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55th Annual Meeting on Hot Laboratories and Remote Handling – HOTLAB 2018

Book of abstracts

Wade Karlsen (ed.)

VTT TECHNOLOGY 334

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Preface

We are pleased to welcome you to the 55th Annual Meeting on Hot Laboratories and Remote Handling – HOTLAB 2018, which takes place in the Nordic city of Helsinki from the 16th to the 20th of September, 2018. For the meeting venue we have selected the Helsinki Congress Paasitorni, established already in 1908. Paasitorni is located by the sea in the city center in close vicinity of the Hakaniemi market place and Hakaniemi metro station. The meeting itself takes place in the meeting room Sirkus of Paasitorni, located downstairs, but with its own spacious lobby area and its own restaurant.

Based on the abstract submissions, we have created a program for HOTLAB 2018 that follows the life-cycle of hot laboratory operation in support of nuclear power operations. That means starting from designing new facilities, through their use, and finally a special focus on waste handling and decommissioning, including conditioning high level waste for final repository. We are pleased to offer a particularly extensive collection of companies presenting their offerings as part of a “mini expo” in the lobby are outside Sirkus, as well as a large collection of posters in an area of the lobby.

We also wish to thank the sponsors of some special aspect of this meeting, including IAEA for travel assistance, Nuclear Science User Facilities/Idaho National Laboratory for Best Poster, Wälishmiller HWM for Best Presentation, and Aquila for Graduate Student Travel Grants.

We hope you will enjoy the technical meeting, and take a little extra time to enjoy the remarkable and exceptional beauty of Helsinki!

With best wishes,

On behalf of the whole Organizing Committee

Wade Karlsen

IAEA supported travel for some individuals

NSUF/INL supported the award for best poster.

Wälishmiller HWM supported the award for best presentation

Aquila supported Graduate Student Grants.

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Tour and Social Events

VTT Centre for Nuclear Safety

A visit to the VTT nuclear research facilities will be arranged on September 20th

Morning tour bus departure: **8.00** from Hotel Scandic Paasi, Paasivuorenkatu 5 B, 00530 Helsinki

Afternoon tour bus departure: **13.00** from Hotel Scandic Paasi, Paasivuorenkatu 5 B, 00530 Helsinki. Registration and payment is made when registering for the conference.

Other practical information for the tour:

- Pre-sign up for the tour is required. Guests are **pre-allocated** to morning or afternoon tour times.
- Non-Finnish citizens are required to provide their national ID (passport or EU ID card) prior to arrival. Please submit a PDF or photo (jpg etc) to VTT at least 2 days before the visit. Submission to HOTLAB2018@vtt.fi. You must have the same ID with you when you arrive.
- The facility is a secure facility, and therefore our security protocol requires that each visitor must check-in in the lobby before access to the laboratory is granted.
- Cameras and photographing are not allowed in the laboratory, but free lockers are provided in the lobby; small ones for putting your phones/cameras, and also some bigger ones for bags. Please do not leave any belongings on the bus, as there are multiple buses in use for transport.
- The facility is a radiological facility, and therefore disposable shoe covers and disposable lab jackets will be provided for you. You will be subjected to contamination screening upon exit.
- Although the lab coats are provided, please wear pants, as shorts and skirts are not recommended in the laboratory.
- Please wear flat-bottomed shoes, as e.g. high-heels do not function well with shoe coverings, nor do our contamination monitors function well with heels.
- Food, drinks and chewing gum are not allowed in the laboratory area.

If problems arise regarding your attendance the day of the event, please contact Satu Koskela at +358 40 158 2945 or satu.koskela@vtt.fi.

We look forward to seeing you at VTT.

Wade Karlsen, Laboratory Manager,
VTT Centre for Nuclear Safety

Get together party

Eckert & Ziegler
Isotope Technologies Dresden

A Sunday evening get-together will take place at the Brewery-restaurant Bryggeri Helsinki (Bryggeri.fi/en) that is located on Sofiankatu 2 near Helsinki Senate Square, on Sunday September 16th, 2018 at 19.00. You are warmly welcome to this event, sponsored by Isotope Technologies Dresden GmbH.

Meeting dinner

For those registered for the meeting dinner, the dinner will take place at the opposite side of our conference venue Paasitorni in the beautiful restaurant Meripaviljonki just over the sea (www.ravintolameripaviljonki.fi) on Monday September 17th, 2018 at 20.00.






Useful Information of Helsinki

Here we collected some practical information that maybe of interset during your visit.

Finland is a country of forests and lakes. Inland lakes and rivers make up 10% of the country. The large areas of forest cover almost two thirds of the land mass. Only 6% of Finland is arable. It is one of the Nordic countries and bordering Sweden, Norway and Russia. Area wise the country is the fifth largest in Europe (338,424 km²) with only 5.5 million inhabitants. It is the most sparsely populated country in the European Union, with only 16 inhabitants per km². There are four seasons in Finland winters being cold and summers warm.

Finland is a republic and is an independent country since 1917. It became a member of the European Union in 1995 and part of the European Monetary Union in 2002 and the only Nordic country using the Euro as currency. Finland was the third country in the world and the first one in Europe allowing women to vote. This happened as early as 1906. The electronics, machinery, forestry, high-tech and design industries are Finland's most important revenue sources.

Finnish language is a non-Indo-European language belonging to the Uralic family, along with Estonian and Hungarian. However, language is not a problem. As most Finns take it for granted that you do not speak their language, they are glad to make use of their English or other European languages they master. Finland is in fact a bilingual country, the second official language being Swedish.

Climate and weather

Helsinki's climate combines characteristics of both a maritime and a continental climate. The proximity of the Arctic Ocean and the North Atlantic creates cold weather, while the Gulf Stream brings in warmer air.

Daily maximum temperature in September is typically between +15 and +20°C in Helsinki.

Average temperatures in Helsinki

Entire year: +5.9°C

Warmest month: July +17,8°C

Coldest month: February -4,7°C

For the latest weather forecast, see website (en.ilmatieteenlaitos.fi) by the Finnish Meteorological Institute.

City bikes

The City of Helsinki is attempting to increase levels of cycling in city traffic. The city bike system supports this goal. During 2018 city bike season there will be 1,500 bikes and 150 bike stations in Helsinki.

City sightseeings in Helsinki

Guided tours & Hop On Hop Off in Helsinki. Explore the city on a guided sightseeing tour by bus and see all the top sights of Helsinki! Take a tour with Hop On Hop Off or enjoy a city tour with Helsinki Panorama.

Sightseeing by bus and boat in Helsinki: Enjoy Helsinki by bus and boat with our combination tickets. Two guided sightseeing tours that let you discover the main sights in the capital both from land and sea.

More information: www.stromma.fi/en/helsinki/

Currency and payments

The local currency in Finland is Euro (EUR). All major credit cards are widely accepted in Finland. For an ATM machine, look for the sign “OTTO”. These 24-hour cashpoint machines are widely available and accept the following international ATM and credit cards: Visa, Visa Electron, EuroCard, MasterCard, Maestro, Cirrus and EC.

Banks are usually open on weekdays 10am–4:30 pm.

In addition to banks, there are several foreign currency exchange points available around the city center. See for example: Forex at Stockmann Department Store or Central Railway Station and Tavex on Fabianinkatu 12.

Electricity

Voltage: 220–240 Volts. Electrical sockets (outlets) in Finland are one of the two European standard electrical socket types: ”Type C” Europlug and ”Type E/F” Schuko.

Emergency number

General Emergency number for police, ambulance and fire department is 112.

Hakaniemi Market Hall

While the Hakaniemi Market Hall from 1914 is undergoing a major renovation, the indoor market and its vendors have moved to temporary facilities inside a modern glass hall in Hakaniemi Market Square. The new glass hall houses almost 50 stalls, and its sunny terrace is open throughout the summer. Enjoy the atmosphere and fresh seasonal products, eat a delicious lunch or just stop by for coffees and cinnamon buns. The glass hall is open from 8am to 9pm on weekdays and locates approximately 400 meters from the meeting venue.

Insurance

Please check the validity of your own insurance. The Conference organisers cannot accept liability for personal injuries sustained or loss of, or damage to, property belonging to delegates or the registered accompanying persons during the Conference.

Passport and visa / Letter of invitation

Citizens of the member countries of the European Union and several other countries do not need a visa for entering Finland. Detailed information about the passport and visa requirements can be found at website of the Ministry for Foreign Affairs of Finland.

If you require a letter of invitation from HOTLAB 2018 secretariat, please submit a request for a visa letter invitation to HOTLAB2018@vtt.fi. Your name must be listed exactly as it appears on your passport. Any differences between the name on your passport and the name on your invitation letter or other documentation could lead to a delay and/or denial of your visa.

<u>Note that the visa refusal is no reason for refund of the registration fee.</u>

Postal services

Post offices are open 9 am–6 pm (some offices even until 8 pm) from Monday to Friday. Yellow mailboxes are available for collections on weekdays.

Stamps can be purchased at post offices, bookstores, newsagents, kiosks and hotels. Helsinki General Post Office is located at the center of Helsinki, Elielinaukio 2 F.

posti

Tax free shopping

Citizens of no*N*-European countries are eligible for tax-free returns upon leaving EU territory. Purchases must be made in shops displaying the Tax Free Shopping sign. The minimum total sum of purchased goods must be €40.

Time zone

The time zone in Helsinki is Eastern European Time (EET), 2 hours ahead of Greenwich Mean Time (GMT+2).

Tourist information

Helsinki Tourist Information (www.myhelsinki.fi) offers free information about the city, sights, events and services including a wide range of brochures and maps all year round.

WiFi

You can connect your laptop or mobile phone to a wireless network in many places throughout the city center e.g. in several cafes, restaurants and libraries.

For free WiFi, connect to “Helsinki City Open Wlan”. There are plenty of hotspots available in the city center and at harbors. In addition, most hotels offer their guests a free internet connection. For the international research and education community, it is also possible to use Eduroam service while in Helsinki.

Scientific Programme

Sunday 16.9.2018

19:00 – 22:00 Get together at **Restaurant Bryggeri** (Sofiankatu 2)…

Monday 17.9.2018

Opening

9:00 – 9:20	*Wade Karlsen (VTT, Espoo, Finland)* Welcome and Opening the HOTLAB 2018

Technical Session: Designing and building of new facilities

9:20 – 9:45	*Brandon Miller* *Idaho National Laboratory, Idaho Falls, USA* Conceptual Design of the Sample Preparation Laboratory (SPL) at Idaho National Laboratory
9:45 – 10:10	*Rubén O. González* *Comisión Nacional de Energía Atómica; Buenos Aires, Argentina* New Laboratory for the Study of Irradiated Materials Associated to RA10 Research Reactor in Argentina
10:10 – 10:35	*Kamlesh Pandit,* *Bhabha Atomic Research Centre, Mumbai, India* Engineering Aspects of Hot Cells and In-Cell Equipments
10:40 – 11:10	**Coffee Break in Foyer**
11:10 – 10:35	*Mitchell K. Meyer* *Idaho National Laboratory, Idaho Falls, USA* Commissioning of the Irradiated Materials Characterization Laboratory
11:35 – 12:00	*Dongseok Ryu* *Korea Atomic Energy Research Institute, Daejeon, Korea* Overhead Gantry System and Basket Handling - for the Pyroprocessing Automation Verifying Mock-up
12:00 – 12:25	*Carvyn Jones* *European Spallation Source, Lund, Sweden* ESS Cask Assembly and Systems Engineering Methodology
12:25 – 12:50	*Keith Kershaw* *CERN, Geneva, Switzerland* Remote Handling Design Study Developments for a New Experimental Facility at CERN – The Beam Dump Facility
13:00 – 14:00	**Lunch in Restaurant "Kellari"**

Commercial Session: Designing and building of new facilities, commercial resources

14:00 – 14:20	*Maria Isabel Machado* *Assystem Engineering & Infrastructures, Paris, France* Innovative Hot Lab Concept for Nuclear Industry
14:20 – 14:40	*Hubert Hafen* *Wälischmiller Engineering Gmbh, Markdorf, Germany* Challenges in Designing Hot Cells - How to Avoid Future Difficulties in the Use of Telemanipulators
14:40 – 15:00	*Steve Chunglo* *Central Research Laboratories (CRL), London, UK* Getting a Handle on Improved Telemanipulator Operation HOTLAB 2018

Technical Session: PIE of Fuels

15:00 – 15:25	*Marin Mincu* *Institute for Nuclear Research Pitesti, Arges, Romania* Some Aspects Concerning Post-Irradiation Examination of CANDU Type Nuclear Fuel
15:25 – 15:50	*Wang Xin* *China Institute of Atomic Energy, Beijing, China* Failure Analysis on AFA 3G Gd Rod from Nuclear Power Plant
15:50 – 16:20	**Coffee Break in Foyer**
16:20 – 16:45	*Zhen Wang* *Nuclear Power Institute of China, Sichuan, China* Blistering Test Under the Pressure Condition in Hot Cell
16:45 – 17:10	*Fauzi Helmi Rahmatullah* *Indonesia Center for Nuclear Technology,Tangerang Selatan, Indonesia* Blister Defect Analysis of U3 Si2/Al Nuclear Fuel Cladding by Ultrasonic Test
17:10 – 17:35	*Cad Christensen* *Battelle Energy Alliance, Idaho Falls, USA* Advanced Gas Reactor/TRISO Particle Fuel Re-Irradiation and Safety Testing Experiment Performed in the NRAD Reactor and HFEF Hot Cell
17:35 – 18:00	*Philip Winston* *Idaho National Laboratory, Idaho Falls, USA* Radial Deconsolidation of Irradiated AGR-3/4 Compacts at Idaho National Laboratory
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Tuesday 18.9.2018

Technical Session: PIE of Fuels

9:00 – 9:20	*Shang-Feng Huang* *Institute of Nuclear Energy Research, Taoyuan, Taiwan* Correlation of Pressurized Water Reactor Vessel Material Properties Variation with Neutron Fluence by Surveillance Program
9:20 – 9:40	*Takuji Sugihara* *Nuclear Development Corporation, Ibaraki, Japan* Miniature C(T) Specimen Fabrication for Reutilization of Surveillance Tested Materials
9:40 – 10:00	*Lu Wu* *The First Institute, Nuclear Power Institute of China, Sichuan, China* The Microstructure of Post-Irradiated A508-3 Steel and its Effects on Charpy Impact Energy
10:00 – 10:20	*Venkatasubramanian Karthik* *The First Institute, Nuclear Power Institute of China, Sichuan, China* Mechanical property evaluation of irradiated stainless steels using sub size and miniature specimens
10:20 – 10:40	*Kris Dunn* *Canadian Nuclear Laboratories (CNL), Ontario, Canada* Hotcell Examinations of In-core Inconel X-750 spacers removed from CANDU Reactors for Surveillance
10:40 – 11:10	**Coffee Break in Foyer**

Technical Session: Hot Lab Equipment

11:10 – 11:35	*Mahdi Rezaian* *Atomic Energy Organization of Iran (AEOI), North Kargar, Iran* Criticality Evaluation of a Transport Cask of Irradiated Nuclear Fuel Samples According to the IAEA Regulations for the Safe Transport of Radioactive Material
11:35 – 12:00	*Natalia Zolnikova* *ROBATEL Industries, Genas, France* The R83 Type B(U) Transport Package for Used LEU Fuel: A Versatile Package
12:00 – 12:25	*Ion Man* *Institute for Nuclear Research Pitesti, Arges, Romania* Compact Tension Sample Preparation out of Candu Pressure Tube Using the Numerical Controlled Milling Machine
12:25 – 12:50	*John Stanek* *Idaho National Laboratory, Idaho Falls, USA* Radioactive Materialography Preparation Systems
13:00 – 14:00	**Lunch in Restaurant “Kellari”**

Technical Session: Hot Lab Equipment

14:00 – 14:25	*Gary Butler* *Aquila Nuclear Engineering Ltd., Twyford, UK* Fit for Purpose Design for Remote Operations -Handling the Hot Potatoes
14:25 – 14:50	*Gauthier Jouan* *CEA Marcoule, Bagnols Sur Ceze, France* Implementation of An Innovative Nuclearized SEM in CEA-Atalante Facility
14:50 – 15:10	*Mona P. Moret* *CAMECA, Gennevilliers. France* Shielded Electron Microprobe and Some of Its Main Applications in Hotlabs
15:10 – 15:30	*Robert M. Ulfig* *CAMECA Instruments Ltd., Madison, USA* Three-Dimensional Subnanometer Compositional Analysis of Radiation Damaged Materials with Atom Probe Tomography – Technology and Practical Considerations
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Technical Session: Management of Hot Labs

9:00 – 9:25	*J. Rory Kennedy* *Idaho National Laboratory, Idaho Falls, USA* Overview and Status of the US Nuclear Science User Facilities (NSUF)
9:25 – 9:50	*Ki Seob Sim* *International Atomic Energy Agency (IAEA), Vienna Austria* IAEA Activities on Fuel Irradiation Tests, Post Irradiation Examination (PIE) and PIE Facilities Database
9:50 – 10:15	*Marco Streit* *Paul Scherrer Institute, Villigen, Switzerland* New Integrated Sample Management Software at PSI HOTLAB
10:15 – 10:40	*Frédéric Schmitz* *Bel V, Anderlect, Belgium* Shielded Cells Design and Periodic Safety Review
10:40 – 11:10	**Coffee Break in Foyer**
11:10 – 11:35	*Laurent Loubet* *CEA, St. Paul lez Durance, France* Accumulation of Nuclear Material in Nuclear Facilities: An Iterative Approach in Order to Develop Measuring Stations
11:35 – 12:00	*Jean-Marc Adnet* *CEA,* Bagnols sur Cèze Cedex*, France* TARRA Project: Transfer of MOX R & D Between 2 CEA Sites

Technical Session: Aging Management of Hot Labs

12:00 – 12:25	*Javin DeVreede* *Canadian Nuclear Laboratories (CNL), Ontario, Canada* Rejuvenation of the Canadian Nuclear Laboratories Chalk River Campus
12:25 – 12:50	*Niklas Snis* *Studsvik Nuclear Ab, Nyköping, Sweden* Recent Upgrades of the Studsvik Concrete Hot Cell Facility
13:00 – 14:00	**Lunch in Restaurant “Kellari”**
14:00 – 14:20	*Franck Dominjon* *CEA, Bagnols Sur Ceze, France* Refurbishment of Handling Equipments in A Maintenance Cell of Phenix
14:20 – 14:40	*Basiran Basiran* *Indonesia Center for Nuclear Technology, Tangerang Selatan, Indonesia* Refurbishment of Drum Lifting Device for Radioactive Waste Handling Inside Hot Cell Facility

Technical Session: Decommissioning and Waste Handling

14:40 – 15:00	*Sou Watanabe* *Japan Atomic Energy Agency, Ibaraki, Japan* Treatments of Radioactive Waste Solutions Generated in a Hot Laboratory of Japan Atomic Energy Agency
15:00 – 15:20	*Barbara Charlotte Oberländer* *Institute for Energy Technology (IFE), Kjeller, Norway* Implementation of An Innovative Nuclearized SEM in CEA-Atalante Facility
15:20 – 15:40	*Torje Osen* *Merrick & Company, Greenwood Village, USA* Remote Handling and Waste Containment Approach for Whiteshell Laboratories Standpipe and Bunker Legacy Waste Retrieval
15:40 – 16:10	**Coffee Break in Foyer**
16:10 – 16:30	*Frank Scheuermann* *NUKEM Technologies Engineering Services, Alzenau, Germany* Fuel Inspection Hot Cell at Ignalina B1 ISFSF – Lessons Learned
16:30 – 16:50	*Jae-Han Kim* *Korea Atomic Energy Research Institute, Daejeon, South Korea* Preliminary Study on the Repair and Transportation Methods of Spent Nuclear Fuel Assembly in KAERI
16:50 – 17:10	*Jonas Martinsson* *Studsvik Nuclear Ab, Nyköping, Sweden* High Level Waste (Spent Fuel) from Hotcell to Final Repository – A Standard Procedure in Sweden
17:10 – 17:30	Jani Huttunen Teollisuuden Voima Oyj, Eurajoki, Finland

Closing Session: Awards, Next HOTLAB Location

Thursday 20.9.2018

a.m. Technical Tour to VTT's brand-new hot cell facilities

p.m. Technical Tour to VTT's brand-new hot cell facilities

Friday 21.9.2018

Optional Technical Tour to Posiva’s Final Repository Site in Olkiluoto (All Day)

ABSTRACTS FOR

ORAL

PRESENTATIONS

Technical Session:

Designing and Building of New Facilities

Conceptual Design of the Sample Preparation Laboratory (SPL) at Idaho National Laboratory

Brandon Miller, Mitch Meyer and Bill Landman

Idaho National Laboratory, Idaho Falls, USA
Corresponding author: Brandon Miller <brandon.miller@inl.gov>

The Idaho National Laboratory is designing a new hot cell facility, designated as the Sample Preparation Laboratory (SPL), to support beta-gamma bearing materials such as structural materials and alpha-free claddings. The building is currently under design stage with design completion scheduled for the end of 2018. The main feature of SPL is a hot cell line that consists of bays for experimental receipt and decontamination, a sizing and grinding cell, a sample decontamination cell, and a sample storage cell. Attached to the hot cell line is a shielded mechanical properties testing cell. Instrument cells are present in the laboratory to house advanced characterization equipment, including 4 shielded instrument cells for high dose material characterization. Robotics are being employed in the shielded instrument cells for loading/unloading activities of the instruments. A glovebox/hood line is available for sample preparation activities for lower dose materials. Additional features of the SPL include a manipulator repair area, increased office space for researchers, operators, and users, and future instrument cell growth possibilities.

Purpose for SPL

Sample Preparation Laboratory, SPL, is being constructed at the Materials and Fuels Complex, MFC, at Idaho National Laboratory. This facility supports a need for a facility to accept and manipulate highly radioactive beta-gamma bearing materials that are alpha contamination free. The Advanced Test Reactor, ATR, is used by various programs to irradiate structural and cladding based materials to high doses to study their irradiation behaviour. Typically, these experiments are transferred to the Hot Fuels Examination Facility, HFEF, at MFC for experimental disassembly and other activities. In HFEF, irradiated fuel is characterized and significant alpha, beta, and gamma contamination is present. As a part of the Nuclear Scientific User Facility, low dose structural samples from the irradiated ATR experiments are shipped to other national laboratories and universities for characterization. This requires decontamination of the NSUF samples from alpha bearing contamination from HFEF. This process can be difficult compared to materials with only beta-gamma contamination and often leads to increased dose to personnel to achieve contamination levels acceptable to universities. SPL is being designed with the purpose of being alpha contamination free excluding some use of fixed alpha bearing materials in advanced characterization equipment. Another feature of SPL of interest for INL is a shielded mechanical properties cell. The shielded cell will provide capability to characterize high gamma dose materials that previously could not be analysed at INL. SPL will have other features which will be discussed in detail a later section.

Design Features of SPL

SPL is a three story building that will support sample preparation activities, advanced characterization of irradiated materials, and office space. Figure 1 shows a general layout of the three stories of the SPL building and a conceptual design of the outside of SPL. Office space is designed to provide collaboration between researchers, operators, and outside users of facility.

Figure 1: General layout of the three floors of SPL

Hot Cell Line of SPL

The primary feature of SPL is the hot cell for sample preparation of beta-gamma materials. A source term of 550 Ci of Co-60 (2.035E13 Bq) is assumed for the hot cell line. Figure 2 shows a general layout of the hot cell line. The cell is a partial two story design. Experimental casks enter a cave to mate up to the hot cell. Various casks can be mated up to the cell including the Flying Pig, the BRR cask, the GE-100, and others. Experiments are removed from their cask by use of a crane and lifted into second floor shielded bay. Experiments are decontaminated and checked for any alpha contamination from the shipping cask in this cell. Experiments are dropped into the sizing and grinding cell for disassembly and sample preparation. The sizing and grinding cell contains experimental drop down plugs to store experiments in. Traditional sample preparation equipment including polishers, grinders, and saws will be used for sample preparation activities and disassembly of experiments. An electron discharge machine, EDM, will be used to shape and size larger samples down to sizes appropriate for mechanical property testing and other activities. The EDM will be stationed in a location where a man door is present for the sizing and grinding cell for service ease. A crane is used to move equipment in the sizing and grinding cell with access for the crane to reach the decontamination cell adjacent the sizing cell. A false wall is present between the sizing and grinding cell and the decontamination cell. This false wall can be opened to move larger items between the two cells with an additional basic pass through present below the false wall for transfer of smaller items.

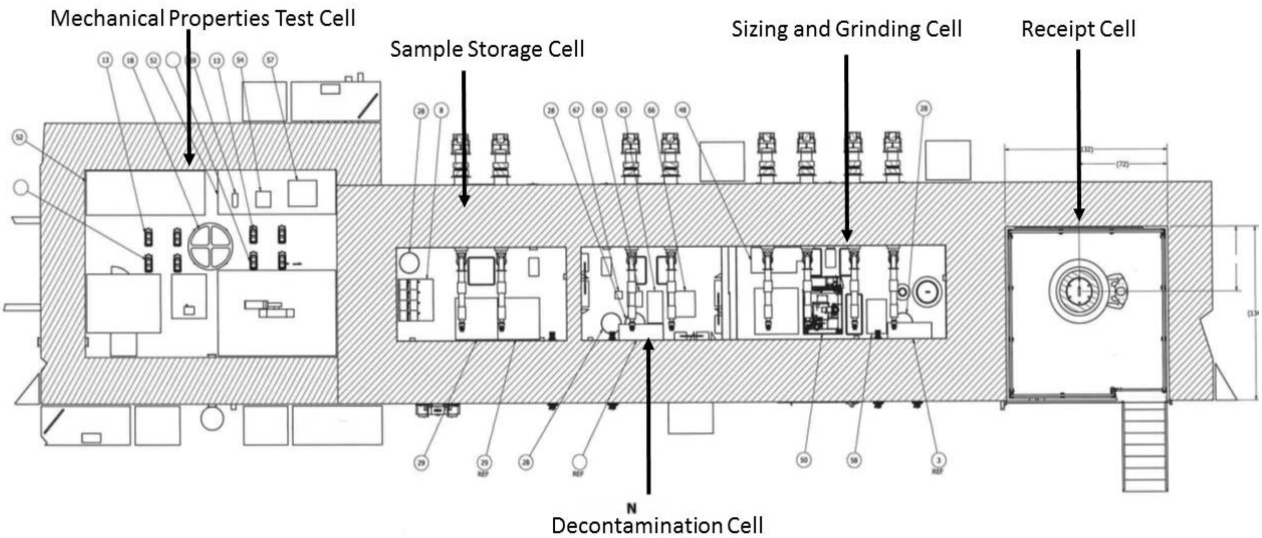


Figure 2: General layout of the hot cell in SPL

The decontamination cell's primary purpose is decontamination of equipment and samples. Radiological waste will be packed in this cell as well. The cell has a plug in the ceiling, allowing access to the crane and to provide another method to allow larger items to be placed into and out of the cells. A small pass through port allows samples to be transferred to the sample storage cell. In the sample storage cell, multiple thousands of samples can be stored in drawers for future use. A ceiling plug is present to have access to the cell when samples are not present. Another feature of the sample storage cell is a pneumatic air transfer system for transferring samples to various locations in the facility such as the shielded instrument cells and glovebox system.

Advanced Characterization Instrument Cells

SPL is designed to have individual instrument cells that will house various advanced characterization equipment. Four of the cells are shielded cells designed to shield 0.25 Ci/9.25E9 Bq of Co-60. Each instrument cell has a separate individual room that is outside of the radiological buffer area that will allow for operation of the instruments. These separate rooms will allow outside users access to run the instruments without having to undergo the radiological training to enter the laboratory. The pneumatic air transfer system is used to transfer high dose materials from the sample storage cell to the shielded cells. A robot system will be used to load/unload the instruments. Vibrating equipment associated with the instruments will be located above the instruments on the second floor of SPL. Some targeted advanced characterization equipment include X-ray diffractometer (XRD), X-ray photoelectron spectroscopy (XPS), and scanning electron/focused ion beam microscopy (SEM/FIB).

Additional Key Features of SPL

Various other key features of SPL are listed below.

- A glovebox/fume hood line is located on the second floor of SPL. Its purpose is sample preparation of materials with low radiation dose. Six fume hoods are associated with the line to support various activities including two non-radiological hoods. All of the radiological hoods and the glovebox are connect using transfer chambers to allow transferring of items easily.
- A manipulator repair area is present on the third floor of SPL. This is needed due to no facility at MFC capable of repair of beta-gamma only contaminated manipulators.
- Additional space on the 1st and 2nd floor of SPL for installation of new equipment and shielded cells.

Timeframe

- End of 2018: Completion of 100% design
- Start construction in 2019 depending on funding from U.S. Government.

New laboratory for the study of irradiated materials associated to RA10 Research Reactor in Argentina

Rubén O. González, Marcela I. Mirandou, Constanza Lugo, Aníbal Rodriguez, Pablo Areas, Evangelina De Las Heras, Lucio Ponzoni and Yasmin Martínez

Comisión Nacional de Energía Atómica, Ciudad Autónoma de Buenos Aires, Argentina
Corresponding author: Ruben O. González <rugonz@cnea.gov.ar>

Currently Argentina has the experience and capacity for the design, construction, commissioning and operation reactors for research, development and production of radioisotopes, such as the RA-10 reactor, which construction is in progress. In the same way, during the last years our country has accomplished the knowledge to design and build the facilities needed to enhance the use of RA10 reactor through its multiple capabilities.

One of the main objectives of *RA10* reactor is the research and development of new nuclear fuels and materials. To fulfill this task, it is mandatory the existence of a facility in which the post irradiation examinations, studies and tests can be done. In this sense, a project has been initiated in CNEA which aim is the construction of a new *Laboratory for the Study of Irradiated Materials*.

This Laboratory will be destined to carry out tests and characterization of irradiated fuels and materials in a versatile and adequate way through specific and complex studies. It will be equipped with high technological level equipment applied to several analysis techniques such as, Optical Microscopy, Scanning Electron Microscopy, Quantitative Dispersive Microanalysis in Energy and Wavelength, X-ray Diffraction, among others.

Technical description

The proposed plant distribution for this laboratory (Figure 3) is composed by a main warehouse divided into four sectors: Office Sector, Equipment-Operation Sector, Intervention Sector and Hot Cells Sector.

In the Office Sector, there will be a main access road for the entry of personnel, while in the Intervention Sector a secondary access road destined to the entry of transport packages.

In the Equipment-Operation Sector, the characterization equipment will be housed. The equipment considered necessary for a proper analysis and characterization of the irradiated samples are a scanning electron microscope, a X-ray diffractometer and an electron microprobe analyser. This sector will have sufficient free space, in case the use of other equipment would be required in specific practices. Each equipment will have associated a workstation for technicians and / or users of it. There will also be support workstations with tools for repairing or making specific devices needed during the development of different experiences.

This sector will also have a Mock up cell for training and testing of procedures and the general control room of the Hot Cell, Equipment-Operation and Intervention sectors (controls, commands and electrical protections of the laboratory).

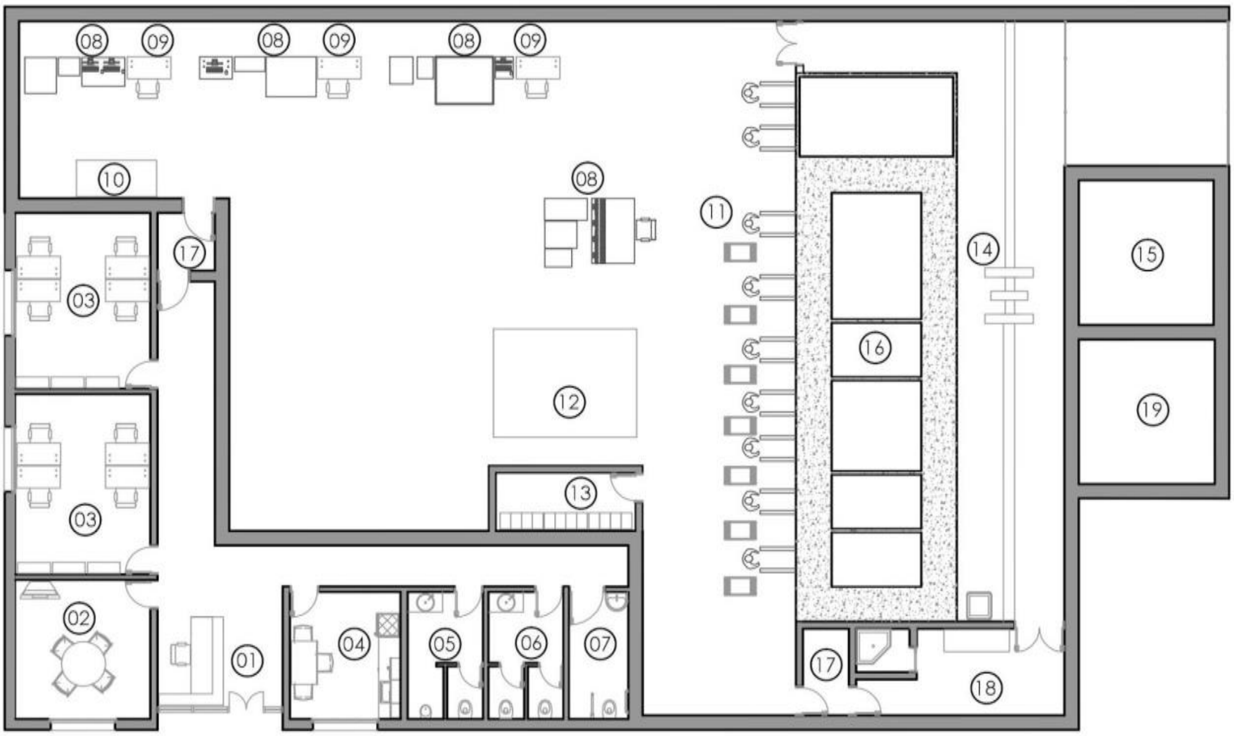
In the Intervention Sector, the samples coming from the reactor in their corresponding containers will be received. It will also have a depot for irradiated samples, a shielded locker for small equipment used inside the cells, which are taken out in order to free the different cells. Also a small room to contain radioactive waste, temporarily, will be placed in this sector.

The Hot Cells Sector will consist of six αβγ shielded compartments, in which highly radioactive materials can be handled. The entire assembly will have adequate perimeter shielding for an equivalent Co_{60} activity of approximately 10^5 Ci.

The work premise is based on the minimization of the samples, their preparation according to the requirements of the different characterization techniques and the conditioning for their extraction from the cell and subsequent analysis. As this laboratory has been thought to carry out both research and development activities, a versatile and comprehensive conceptual design is necessary. Thus, the cells will have different internal sizes with the aim of adapting them to the greater number of activities and required equipment.

In the following paragraphs a brief description of the functionality of each of the six proposed compartments is presented

- *Compartment 1*: Its main function will be the handling of large components. It will have an approximate width of 3 m and double pair of telemanipulators. These two workstations can be used individually or combinated. It will have gates of important dimensions to allow the entry of devices.
- *Compartments 2 and 3*: The main function will be the development of activities that require the use of equipment of reduced dimensions. It will have an approximate width of 1.5 m and a single pair of telemanipulators.
- *Compartment 4*: Its main function will be metallographic preparation of samples. Inside it, cutting, polishing, ultrasonic cleaner machines, among others, will be housed. It will have an approximate width of 2.5 m and double pair of telemanipulators to work independently or in combination.
- *Compartment 5*: It will contain an optical microscope. It will also have free space available to house auxiliary equipment for common use or equipment from other compartments that require a temporary stay inside the cells. Its width will be 1.5 m, approximately, and will have one pair of telemanipulators.
- *Compartment 6*: It will contain the column of an electron microprobe analyser.



1. Central hall,
2. Meeting room,
3. Offices,
4. Kitchenette,
5-7. Rest rooms,
8. Characterization Equipment,
9. Equipment workstations,
10. Tools cabinet,
11. Hot cells workstations,
12. Mock up,
13. Hot cells control room,
14. Crane,
15. Equipment and irradiated samples room,
16. Hot cells,
17. Airlock,
18. Dressing room,
19. Radioactive waste room

Figure 3: Proposed plant distribution for the new *Laboratory for the Study of Irradiated Materials.*

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Engineering aspects of hot cells and in-cell equipments

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Reliable performance of nuclear fuels and critical core components has a large bearing on economics of nuclear power and radiation safety of plant operating personnel. In view of this, PIE is periodically carried out on fuels and components to generate feedback information which is used by the designers, fabricators and the reactor operators to bring about changes for improved performance of the fuel and components. Examination of the fuel bundles has to be carried out inside hot cells due to their high radioactivity. The new hot cell facility (NHF) has been designed, built and commissioned with additional features and capabilities. The NHF consists of two hot cells made of heavy density concrete and a number of lead cells. The new hot cells are designed to handle higher levels of β-γ radioactivity. NHF has fully automated ventilation system with improved safety systems. NHF has a charging port with adequate opening to accommodate irradiated reactor core components. All material transfer system in the cell has electronic and mechanical interlocks which eliminate chance of accidental personal exposure. NHF is provided with cranes in all radioactive areas which facilitate handling of larger irradiated components and heavier shielding cask. Automated data acquisition and monitoring systems are installed in NHF.

Various equipment are installed inside the hot-cells for the material property evaluation and performance assessment. The details regarding the engineering aspects of NHF and in-cell equipment will be discussed. A view of the operating area of the NHF is shown in Figure 4.

Figure 4: Operating area of the hot cells in the NHF showing the viewing windows and master slave Manipulators (MSMs)

Engineering aspects of New Hot Cell Facility (NHF):

The new hot cell facility has two areas for handling radioactive materials. They are:

- ***Hot cells for handling highly radioactive irradiated fuels and structural materials.*** The NHF consists of two hot cells, namely cell-1 & cell-2, and is designed to handle β–γ radiation. The front, rear and side walls of the cells are 1.5 m thick and are made of heavy density concrete (density=3.4 g/cc) (Bhandekar et al., 2015). The bigger cell (cell–1) is around 17 m long and the smaller cell (cell–2) is 5 m long. All regions of cell–1 and cell–2 are provided with lead glass viewing windows, master slave manipulators (MSMs), ports for in-cell camera and service plugs, which are essential for carrying out PIE. The salient features of the NHF are:
    - *Fuel transfer port size*. The new hot cells have transfer port which will facilitate loading of larger components like control blade assembly into the cells for PIE.
    - *Cell dimensions*. The length of the hot cell in the NHF is 17 meters. This provides the advantage of examining longer components such as full length irradiated pressure tube. The NHF provides the facility of continuous scanning of longer fuel elements.
    - *Shielding capacity*. The hot cells at the NHF are capable of handling higher activities up to 2.5 x10^5 Ci of Co^{60} or 2.6x 10^6 Ci of fission products.
    - *Ventilation in NHF*. The ventilation in the NHF is of once-through type and ensures dynamic confinement of radioactive particulates within the radioactive zones of the facility. The ventilation system is based on radioactive area zoning principles and satisfies the regulatory guidelines.
    - *Cranes for material handling.* Both cells have been provided with dedicated in-cell cranes of 2.0-ton capacity. The isolation area is on the rear of the hot cells and acts as a buffer between the cells and the high bay surrounding the cells. The isolation area is provided with a 2T hand operated overhead travel underslung crane. The cell exhaust filters are located in nine separate pits below the isolation area floor. The high bay surrounding the cells and isolation area on three sides is called the warm work area and houses a 40T/5T EOT crane. This area is used for receiving shielded casks containing radioactive materials. An airlock capable of accommodating a 30T trailer truck is provided.

Lead cells and low active laboratories for handling of specimens with a lower radiation field

The low active laboratory is primarily used for carrying out mechanical tests on irradiated test specimens. Towards this an instrumented drop tower, a servo hydraulic & screw driven universal testing machines, creep testing units and static load test setups have been installed in the low active laboratory. The front wall of the lead cells in this laboratory is made of 200mm thick steel cased lead bricks and the rear walls are made of 100mm thick lead bricks. The lead cells are fitted with articulated MSM, viewing windows, hatches/door for personnel entry, transfer ports, and other handling facilities. The radioactivity of the test specimens will be limited to a few mCi of Co^{60} equivalent.

Engineering aspects of the in-cell equipment

The NHF has a comprehensive PIE facility in terms of material characterization and analytical capabilities required for PIE studies on nuclear fuels and materials.

PIE involves dismantling of fuel bundles using mechanical cutting machine having saw blade, visual examination by high definition pan tilt zoom (PTZ) camera having optical zoom of 30X and 4MP resolution, leak testing using liquid nitrogen-alcohol test, laser profilometry, ultrasonic testing of fuel pins immersed in water in horizontal tank is carried out to detect the presence of incipient flaws in its cladding and also for the end-plug defects, gamma spectroscopy and scanning, fission gas release measurement, bow-measurement of the fuel pins, trepanning of 30 mm disc specimens for metallography and mechanical testing from the pressure tubes, drilling and notching for fracture toughness testing, canning and crimping of intact fuel pins after PIE operation for shifting to the storage pool for further processing or storage. All the in-cell equipment are designed so that it can be operated with MSM and maintenance should be easy.

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Commissioning of the Irradiated Materials Characterization Laboratory

Mitchell K. Meyer, Cynthia Adkins, Jeff Bailey, Aaron Balsmeier, Jayson Bush, Miles Cook, Scott Moore, Emil Franklin, Jian Gan, Jeff Giglio, Lingfeng He, Brandon Hernandez, Michael Iervese, Daniel Jadernas, Jesse Kappmeyer, Kevin Keefe, Collin Knight, Brandon Miller, Daniel Murray, Steven Swanson and Kevin Tolman

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The Irradiated Materials Characterization Laboratory (IMCL) is a nuclear research user facility designed for the characterization of the microstructure and properties of high activity nuclear fuels and materials. The 1114 m^2 (12,000 ft^2) laboratory consists of a single bay and a support wing designed to provide a suitable environment for the operation of high resolution scientific instruments. The open laboratory bay allows shielded instrument installations to be reconfigured to adapt to advances in instrumentation over the expected 40+ year life of the facility. Installation of the initial suite of shielded instruments began in 2015 and will be completed in early 2019. IMCL operation began in 2017, balancing scientific exploration with construction activities.

IMCL Instruments and Equipment

Transfers and Sample Preparation. Materials to be examined in IMCL are typically transferred into the Shielded Sample Preparation Area (SSPA) using a transfer container with an internal volume of approximately 1,000 cm^3 shielded with 10 cm of lead, consistent with IMCL's focus on detailed characterization of smaller samples of material. The (SSPA) consists of three lead shielded compartments, an inert atmosphere glovebox, and an airflow hood containing sample preparation equipment and an optical microscope. Radiation dose rates from samples may be measured in the SSPA sample transfer cell. If dose rates can be reduced sufficiently by size reduction, samples may be prepared in the glovebox and transferred to instruments using the air hood and a lightly shielded container.

Instruments

Shielded instruments currently in service consist of a FEI Helios dual beam Plasma FIB (Focused Ion Beam), a FEI Quanta dual beam FIB, and a Cameca SX100 R shielded microprobe. Because IMCL is focused on the examination of irradiated nuclear fuel, sample materials typically contain significant amounts of alpha- and beta-bearing material as well as high energy gamma emitters. Shielding and contamination control (confinement) functions are distinctly separated for the purpose of allowing easy access to the instruments for maintenance and modifications (Figure 5). Shielding for each instrument is provided by a 21.6 cm (8.5 inch) thick steel wall with a height of 2.1 m (7 feet), designed to provide worker protection from samples with dose rates equivalent to 7.4×10^{10} Bq (2 Ci) of ^{60}Co. Confinement of alpha and beta contamination is provided by an inert atmosphere glovebox. The glovebox is mounted inside of the shielded room adjacent to the front shield wall.

Leaded glass windows installed in the front shield wall provide visibility of the inside of the glovebox, instrument loading device, and sample transfer port. Transfers of material into instrument gloveboxes are made through a rapid transfer port without breach of confinement. Samples to be examined are loaded from the glovebox to the instruments using custom sample transfer apparatus and telemanipulators. Each instrument station also contains a sputter coater.

A shielded cell dedicated to the measurement of thermal properties is currently being installed. This cell will include instruments for temperature dependent measurement of thermal diffusivity, heat capacity, and dilation; sample dimensions and mass; and direct measurement of thermal conductivity using surface reflectivity (Hurley et al., 2015). One additional space in IMCL is reserved for a future shielded instrument.

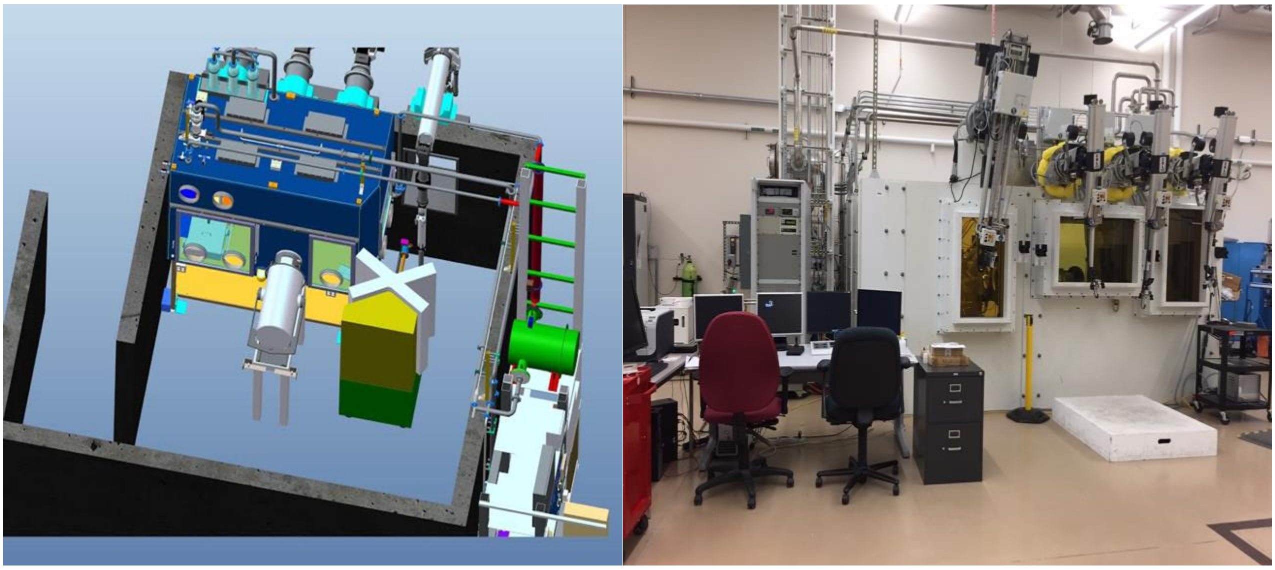

Figure 5: Left - Schematic of shielding and confinement system for IMCL scientific instruments. Gray denotes steel shield walls, blue denotes the confinement glovebox, and yellow/green an electron beam instrument. The IMCL shielded transfer container, depicted in silver, is attached to the center of the glovebox. Right - Photograph of the installed EPMA (Electron Probe MicroAnalysis) shielded cell.

In addition to the shielded instruments described above, IMCL houses an FEI Titan Themis 200 ChemiSTEM, Bruker D-8 micro XRD (X-Ray Diffraction) system, and a PANalytical Empyrean XRD.

Installations in progress include a JEOL 7600 FEG-SEM and a Quantum Design Physical Property Measurement system. A Zeiss Xradia micro computed tomography system will be installed in 2019.

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Overhead Gantry System and Basket Handling for the Pyroprocessing Automation Verifying Mock-up

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The more spent fuel has been stacked in the nuclear power station, the higher apprehension has been growing. Pyroprocessing, one option to recycle the spent fuel, has been studied in Korea Atomic Energy Research Institute. Processing cells were prepared, and various experiments were carried out using surrogate. One salutary lesson from the experiments was the trouble usually comes from the molten salt, rather than radioactivity of the material. Molten salt is highly corrosive, and even worse Pyroprocessing requires to regulate high temperature. Another difficulty from the experiments was the remote handling. For engineering scale experiment, the target objects to handle, such as crucible, material, or electrode, were about 10 ~100 kg, and they usually exceed the payload of the mechanical MSM (master salve manipulator) capability. Simplification of handling operations should be considered in the equipment design, and motorization or automation of equipment were desirable.

As mentioned above, mechanical reliability and remote handling does matter to achieve economic feasibility of Pyroprocessing. To break through the mechanical engineering problem in Pyroprocessing, KAERI tried to resolve the troubled mechanical issues in separate experimental space. The Pyroproccessing Automation Verifying Mock-up (PAVM) was planned to address the mechanical reliability and automation of processing equipment. The figure 1 shows the conceptual drawing of the PAVM.

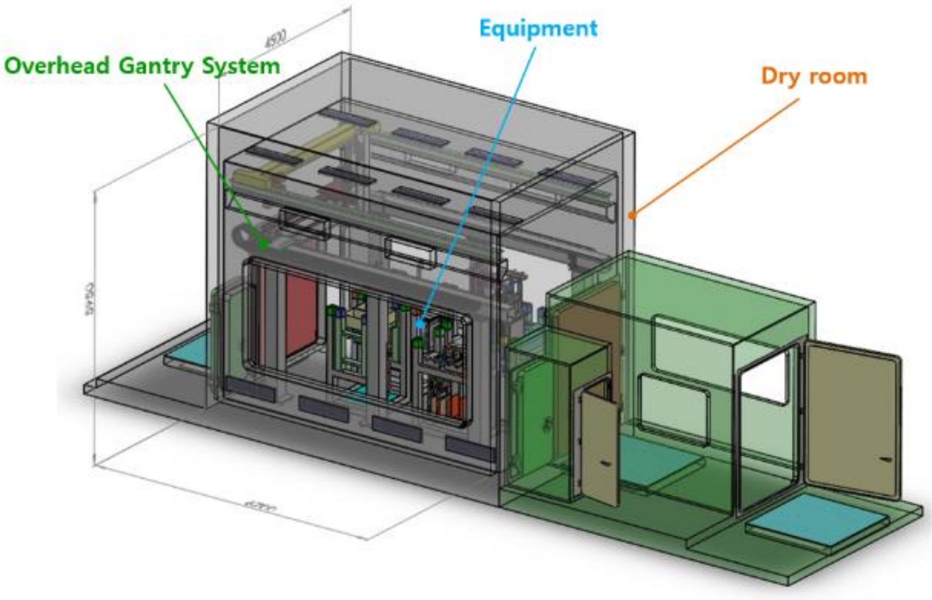

Figure 6: Conceptual drawing of the PVAM.

The main space of the PVAM is a dry room, which would be maintained under the dew point -40 degree Celcious. The size of the dry room was determined as enough to locate two pieces of processing equipment. As the estimated base size of equipment is about 1200 mm by 1200 mm for engineering scale process, then, the dry room size was designed as 4000 mm by 6000 mm. A human worker is allowed to access the dry room through a vestibule, where would be a buffer zone between dry room and atmospheric outside. A crane was installed on ceil of the dry room, and it help to move heavy equipment or material. A precision overhead gantry system was also equipped for automation experiment.

Overhead gantry system in PVAM

Purpose and requirements. The main purpose of the PVAM is to find reliable solutions using molten salt, and it will also be utilized to explore effective ways to simplify the handling operation and automation method. The operations of equipment could be roughly depicted as like preheating, loading material, installing electrode, processing, unloading material, or replacing electrode. The most frequent operation is loading material, and that is top priority to make automation. To load material, a basket should be handled, inserted on a slot, and immersed into molten salt. The requirements of the overhead gantry system are brought to accomplish the sub task of the material loading operation. The requirement are listed below;

- 3 DOF motion (X-Y-Z traveling)
- Work space 3200 (mm) x 4300 (mm) x 1400 (mm)
- X-Y direction force 1000 N
- X-Y direction max speed 1.8 m/s
- X-Y direction linear motor drive
- Z direction max load 50 kg
- Z direction two stage telescopic mechanism
- Precision of each direction is sub mm.
- Various end-effectors are replaceable on the installed tool changer

By adopting industrial precise gantry solution, the overhead gantry system in PVAM was designed as shown in Figure 7.

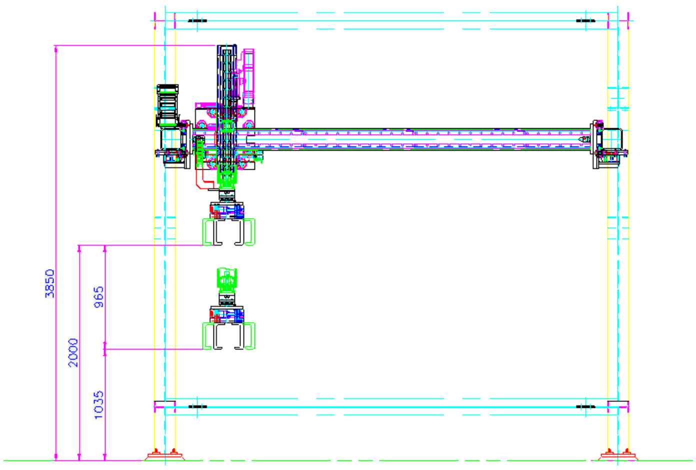

Figure 7. Design of the overhead gantry system in the PVAM.

Installing gantry system and basket handling in a dry room

To minimize the dry volume, two stage telescopic mechanism was installed for z-directional motion. The workspace of the gantry system is shared by the crane system. While the crane is located parking position, the gantry system moves the rail for automation work. On the other hand, the gantry system should be located on the opposite corner to use the crane. Each system is interlocked not to be activated on the same time. A linear motor drive the gantry system, and magnetic scale was utilized for precise position feed-back.

Figure 8. Precise gantry system driving by linear motor.

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ESS Cask Assembly and Systems Engineering Methodology

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The European Spallation Source (ESS) in Lund, Sweden will be a 5 MW long pulsed neutron spallation research facility with planned commissioning 2022. The Cask Assembly, shall ensure the safety and protection of workers, property and the environment from the effects of radiation during Target Monolith maintenance and the internal transport of irradiated Target Monolith components.

The Cask Assembly will transport spent components in the high-bay from the Target Monolith to the Active Cells Facility (ACF), where the irradiated components will be dismantled, separated and prepared for disposal. The Cask Assembly will be lifted by the high-bay crane and, using gamma gates, it will dock to the Target Monolith and ACF to provide extended containment. All the cask internal lifting operations must be remote with no man access to ensure worker safety. The Cask Assembly will also transport new components from the Mock-Up and Test Stand (MUTS) to the Monolith Target for installation. The target station layout is shown in Figure 9.

Target Monolith components that will be handled by the Cask Assembly are: Proton Beam Window, Moderator Reflector Plug, Target Wheel and Shaft, Proton Beam Instrumentation Plug and the Target Monitoring Plug. To access the components, adjacent internal shielding blocks must also be handled.

The internal remote handling devices must provide adequate precision and accuracy to facilitate installation and removal of monolith components.

The Model Based Systems Engineering (MBSE) approach was selected to clarify the project requirements, improve communications, minimise changes during the project, allowing for more realistic time planning, and facilitate simply and appropriate verification and validation.

To apply MBSE effectively, it is essential to consider people, process and tools. One of the key best-practice techniques that have been employed for this project is to provide an Architecture that represents the overall System. The Architecture comprises a number of Views, each of which has its own purpose and delivers some sort of tangible benefit to whichever Stakeholder is using it. Not all Stakeholders will require access to all Views, so it is important to understand who will use each View, what information it should contain and, above all, why they are interested in the View. The Views cover all relevant aspects of the System, such as: requirements, analysis, design, verification and validation, deployment and process. The structure of the Architecture is defined by the Architecture Framework.

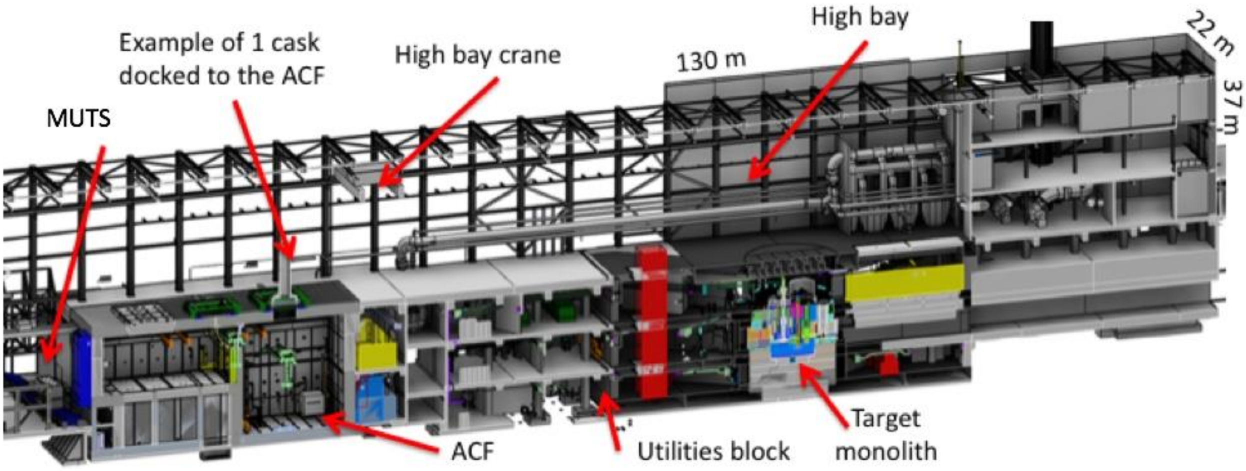

Figure 9. Layout of the target station

This Framework, including the Viewpoints and Ontology, compliant with ISO 15288 and ISO 42010, will be presented as part of this paper. The Framework is being used to provide requirements engineering, context and concept modelling, validation, traceability and procurement tools. The concept modelling includes System Overviews, System Configurations, Interface Definitions and Behaviour Views.

Remote handling design study developments for a new experimental facility at CERN – The Beam Dump Facility

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Introduction to the Beam Dump Facility

CERN has launched a study phase to evaluate the feasibility of a new multi-purpose high-intensity facility with the primary goal of exploring Light Dark Matter – the Beam Dump Facility (BDF). The new facility will require – among other infrastructures – a target complex in which a dense target/dump (referred to in the rest of this document as the "target") will be installed, capable of absorbing the entire energy of the beam extracted from the Super Proton Synchrotron (SPS) accelerator - 355 kW average beam power).

Remote handling and manipulation of the target, the surrounding shielding and adjacent beam line elements will be mandatory due to high residual radiation dose rates. The target complex design is therefore greatly influenced by remote handling considerations.

The target will be used to produce weakly interacting particles, to be investigated by a suite of particle detectors located downstream of the target complex. A new junction cavern and extraction tunnel will be built to house the new beam line taking the proton beam from the SPS to the BDF target area. A large experimental hall is located immediately downstream of the target complex (Figure 10).

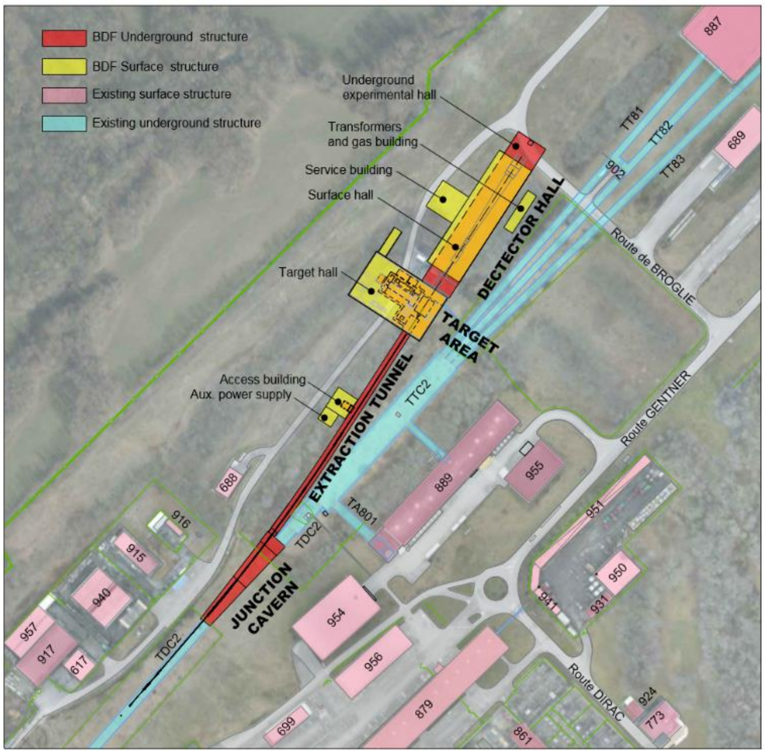

Figure 10: Plan view of the BDF extraction tunnel, target complex and detector hall next to the existing SPS "North Area" (North Area shown in blue).

The target complex. The target will be at the heart of the new facility. High levels of radiation (both prompt and residual) will be produced by the SPS beam hitting the target; a total cumulated dose near the target of around 500 MGy/year is expected. Residual dose rates near the target will be of the order of hundreds of Sv/hr. The target will be located in an underground area (to contain radiation as much as possible) located at about 15 metres below ground level.

The target will be surrounded by approximately 3700 tonnes of cast iron and steel shielding with outer dimensions of around 6.8 m x 9.5 m x 8 m high (the so-called hadron absorber) to reduce the prompt dose rate during operation and the residual dose rate around the target during shutdown. The target and its surrounding shielding will be housed in a vessel containing gaseous helium slightly above atmospheric pressure in order to reduce air activation and reduce the radiation accelerated corrosion of the target and surrounding equipment.

The target and the shielding immediately around it will be water cooled. All the shielding in the helium vessel will be built up of blocks; the layout and geometries of which are designed to avoid direct radiation shine paths and to minimise the number of block movements needed to allow exchange of failed equipment. The target complex design allows for removal and temporary storage of the target and shielding blocks in the cool-down area below ground level and includes dedicated shielded pits for storage of the highest dose rate equipment.

A 40-tonne capacity overhead travelling crane in the target complex building will be used for initial installation and will carry out the handling and remote handling of the shielding blocks and other equipment as needed for assembly and maintenance of the facility.

Target complex design methodology. After the initial work to determine the main requirements and basic layout of the target complex, the target complex design was further developed by going into more detail on the handling and remote handling operations required throughout the life of the facility. This work aimed to demonstrate the feasibility of the construction, operation, maintenance of the BDF target complex along with decommissioning of the key elements.

The study included the conceptual design of lifting, handling and remote handling equipment for the highly activated objects along with the necessary water, helium and electrical connections compatible with the radiation environment and remote handling constraints. These designs were then integrated in to the target complex as a whole.

The two remote handling concepts studied*.* Target complex designs based on two different handling concepts have been developed: the “crane concept” and the “trolley concept”. The crane concept relies on the overhead travelling crane in the target complex building for the movements of the target, shielding etc. during the life of the facility. The trolley concept has the target and its main services installed on a mobile trolley running on rails allowing quicker access to the target.

For both concepts the core elements are essentially the same and are common to the target complex designs produced for both concepts. The main differences between the crane and trolley concepts are in the way the target and water-cooled proximity shielding are supported, installed and removed from the helium vessel and how their services are connected and disconnected.

The presentation will briefly introduce the Beam Dump Facility then describe the two remote handling design concepts and the design study results.

Commercial Session:

Designing and Building of New Facilities, Commercial Resources

Innovative Hot Lab Concept for Nuclear Industry

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Despite the vast use of Hot Labs on the nuclear industry, their current designs and operations do not meet entirely today's needs and requirements of the operations nor tomorrow's. For example, the tools changes/replacements are a factor for delays on the operations that are not completely taken into account today at the design stage. This is due to the difficult access to the interior of the cell. It has to be further emphasized that the maintenance is almost inexistent inside the cells. This complicates the end-life of the Hot Cell with a consequent increase of complexity in the decommissioning operations.

According the aforementioned issues, we propose in this paper to investigate into Hot Cells design and concept in order to integrate new operability modes and functions. The overall idea is to bring modularity, flexibility and adaptability to the Hot Lab nuclear industry.

Figure 11 reports the different phases of the hot cells that will have to be considered during the System engineering approach that will be driven by digital continuity and an efficient information management system that will be discussed later.

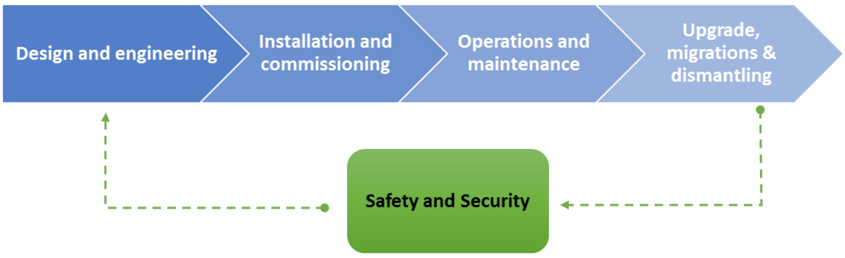

Figure 11: System engineering approach adopted

We initially proposed to focus on the development and implementation of four main aspects: implement the Industry 4.0 concept to the nuclear industry, explore the concept of in service maintenance and predictive maintenance, optimise the decommissioning & dismantling process by taking into account the entire life-cycle process and implement the use of modular and flexible solutions.

This paper proposes to concentrate on a concept of Hot Cell for maintenance operations. This is a case study developed taking several assumptions and could be adapted to any particular project thanks to the global approach taken.

Innovative Hot Lab

The innovations present on the project developed can be divided into three main categories: Modularity, Flexibility and Industry 4.0. The combination of these leads to the complex concept building detailed in Figure 12.

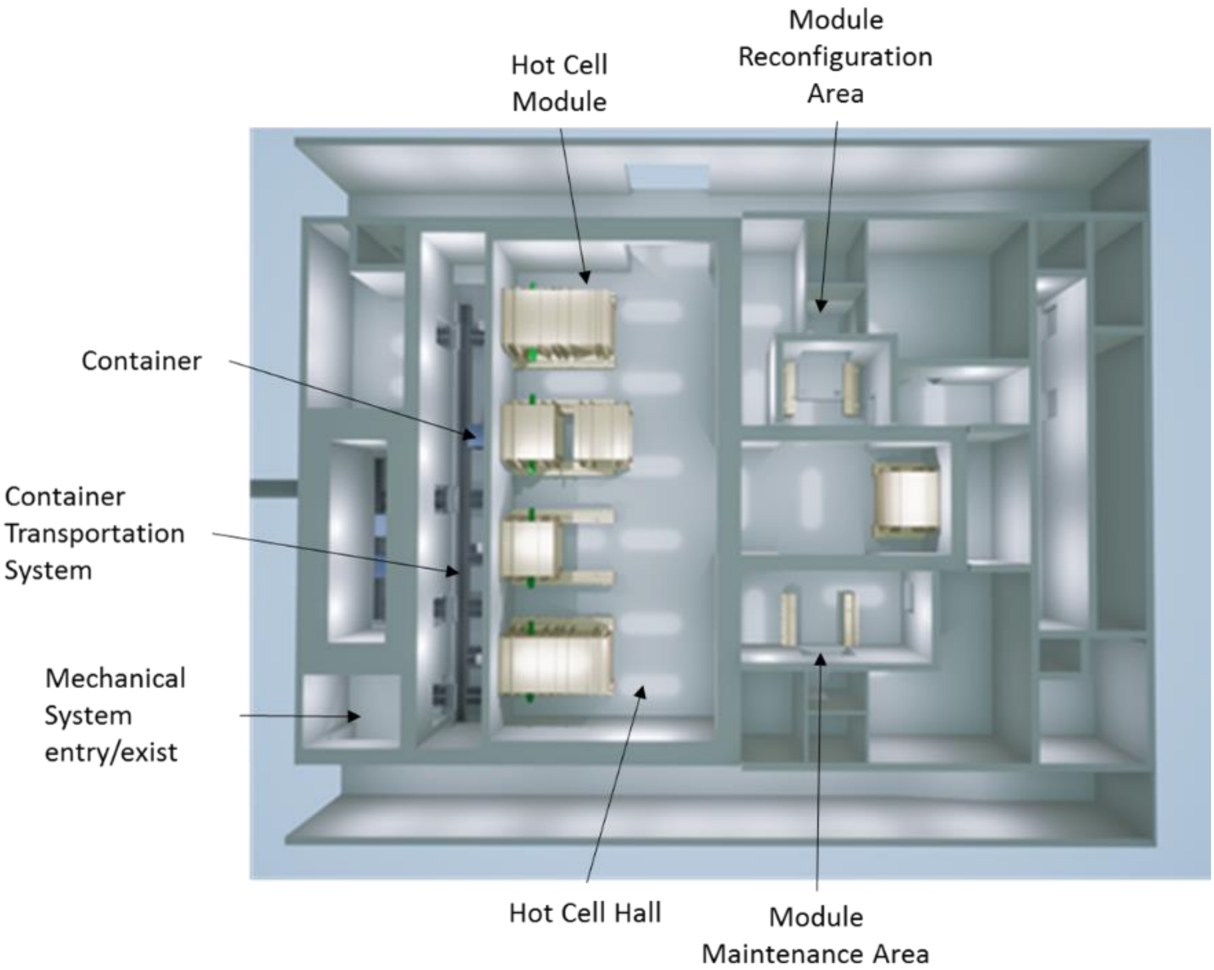

Figure 12: Complex Building Layout

Modularity. The modularity of this concept is achieved by the plug & play characteristic of 9 similar modules present in the Hot Cell Hall. The modules are the operational chambers where the maintenance of the mechanical systems arriving from the reactor will be performed. A Hot Cell assembly can be composed by plugging different type of modules together (see Figure 3).

It is expected that these modules would move to a dedicated area such as maintenance or reconfiguration in the Hot Cell Complex to allow the extension of their operational lifetime.

The modules are mobile and moving to various area and points in the building a standardise docking and alignment interface has been designed.

The modularity has been achieved through the definition of a standard template and simplified interfaces between each sub-systems. This is reflected by the manipulator arms attachment plate to the Hot Cell Module wall. The Container Transportation System (CTS) is also designed to be composed by several modules of rollers that will cover the entire corridor.

Flexibility. The flexibility is mainly achieved by the possibility of fast reconfiguration and re-arrangement of the Hot Cell Modules (HCM). The possibility to easily exchange/replace/modify equipment inside the Hot Cell Module (see Figure 13) according to the operational needs brings a great flexibility to our overall complex.

In the case of maintenance or need for repairs the HCM can also be moved to the maintenance area through the specific innovative transportation mean. Maintenance operations would be achieved to retrieve the Hot Cell Module to a functional state. The reconfiguration and maintenance area are physically separated in order to allow the parallel activities on the Hot Cell Modules.

It has to be noted that while a module is in maintenance or reconfiguration, operations can still be happening in the Hot Cell Hall. This improves the operational efficiency of such a building and reduce downtime of operations.

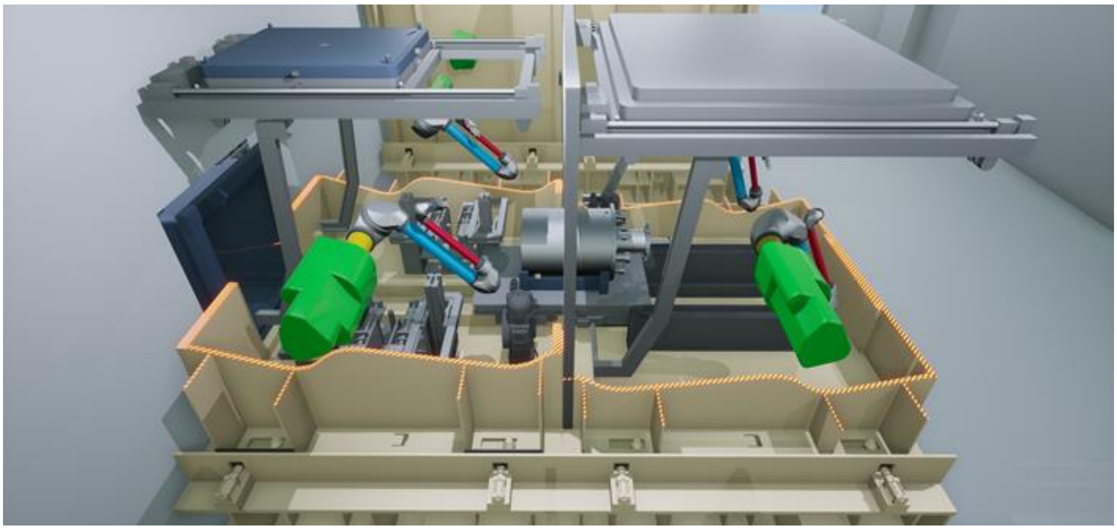

Figure 13: Overview of Hot Cell Modules plugged togeher and their equipment

Industry 4.0. Remotely controlled operations or fully automated operations have been introduced to the operations process. Indeed remote operated manipulators arms, trolleys and cranes have been implemented into the concept design to integrate the functionalities desired.

This intends to improve the activities quality of the operators and the type of work that they might be subjected to. This way it is attempted to automatized the controls and include the newest technologies in the nuclear field.

The activities and supervision of operations will be performed from a control room with remote/virtual access to all the machinery inside the complex. The control room will be also prepared to accommodate the technology upgrades that might occur during the life cycle of the facility. This will lead to use the HCM as laboratory for evaluating new technologies. It has be noticed that this will be made possible with the functionality of flexibility and modularity where some modules are used to evaluate new technologies and others modules are used for nominal operations.

Conclusions & Further Work

In this paper, we proposed a new concept of hot lab that will be enabled by digital continuity and a model based system engineering approach. The architecture proposed is using the most advanced technologies in industry 4.0 to demonstrate the flexibility and modularity requirements. More developments are on-going that will be described in the extended version of the paper and during the conference.

This Hot Lab concept brings the current Hot Lab to the next level as the design has been adapted to meet the current operational needs, implement novel ideas and take into account its entire lifecycle. Developing modular and flexible system is part of the future of the Nuclear Industry and Assystem E&I demonstrated that the Hot Lab had great potential. During this work Assystem has identified topics that would require further development to one day make this concept possible. Therefore collaboration schemes have been set up with research centres, start-ups, and universities.

Through this project Assystem also demonstrated that the use of digital tools is key in the development of a successful project.

Challenges in designing hot cells – How to avoid future difficulties in the use of telemanipulators

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Planning a hot cell implies taking into account different professions working together in a single place: from the building-shell to the in-cell equipment, like lights or media. Also new challenges in the nuclear industry influence the complexity of tasks performed in hot cells, e.g. regulations and technical developments. In the same manner, remote handling technology has been improved in the last decades. All these factors impact the complexity of designing a hot cell. Considering this complexity, small mistakes can have a major impact at the end. From the point of view of a manufacturer and in most cases as subcontractor, this paper will concentrate on giving feedback about experience and technical developments Wälischmiller Engineering made in the last decade in order to highlight some "traps" that can be avoided: By using some examples (part 1) the presentation will highlight how important it is to choose the suppliers at an early stage of the project for special requirements (part 2) and what can be the role of specialised companies in the success of a project (part 3).

Feedback examples

Even if the planning is good, things can be forgotten: e.g. installation or maintenance aspects. In this example, the difficult installation of the manipulators will be highlighted. Cable traces blocked the access of installation equipment. As a result, it has been found that the maintenance according to the maintenance plan cannot be performed. Solutions were found with the customer.

***The importance of a good construction site coordination*.** The second example will highlight the importance of coordination at construction site. The installation of the manipulators has been advanced, despite the fact, that neither the cells nor the building shell were ready. Avoid damage to the manipulators due to welding and concreting works was a real challenge. In addition the work conditions at minus temperatures were more than difficult and brought our staff to their limits.

From these two examples it can be concluded that specialised suppliers should be involved at an early stage of the project to avoid planning mistakes, or at least reduce risk and at the end reduce costs or schedule issues.

Supplier selection at an early stage of the project

Why should the suppliers be selected at the beginning of the project? The selection of the suppliers at the early stage of the project provides a security in the planning through continuity. Both examples highlighted, that the knowledge of Wälischmiller Engineering would have avoided some difficulties during the installation of the remote handling equipment. Involving the experience of suppliers and other stakeholders contributes to the success of projects and reduce the risk of planning mistakes.

To get a better approach of the challenges of each project, it is important to involve specialised companies to identify bottlenecks and to find solutions to solve problems.

Selection criteria. Different criteria have to be taken into consideration to select the supplier: experience, references, solutions oriented and technical capacities.

Role of specialised companies

Experience and expertise. Suppliers with long experience will help to succeed in the project and avoid a lot of problems at the end especially during the installation at site. Companies like Wälischmiller Engineering with many years of experience in the nuclear industry will help to lead to the success of the project. As specialist, Wälischmiller knows the requirements of the nuclear industry and the expectations given to remote handling equipment (narrow spaces, high contamination…).

Since decades we strive to develop customisable solutions for different needs. No project is like the other because each facility is different. Trust and solution oriented relationship is in our opinion the key of the success of many projects. For that reason, we develop our products by keeping in mind the needs of our customers and see it with their eyes.

Reliability in the execution. The planning can be excellent, but sometimes solutions have to be found to solve issues. In this case it is important, that the supplier shows enough flexibility and is solution oriented to overcome difficulties at site during the installation.

A short time frame of the installation procedure is a challenge for companies. The supplier should be able to install equipment within the given time as shut-down of nuclear installation is expensive. Those challenges can be overcome by a continuous, solution oriented and trust based cooperation between customer and supplier.

<u>About Wälischmiller</u>: Wälischmiller Engineering has been providing safe, smart and cost-effective remote handling solutions with the famed German quality and reliability for over 60 years worldwide. Our handling systems offer various mechanical telemanipulators for a wide range of applications. Our models A100 and A200 series were successfully employed in Sellafield, Cadarache and Chernobyl. Other products including remote controlled power manipulators from the A1000 series for handling heavy loads; intervention systems with servo-manipulators for repair and maintenance tasks in hazardous and inaccessible zones as well as remote-controlled and automatic equipment for positioning, transport and sampling tasks.

Getting a Handle on Improved Telemanipulator Operation HOTLAB 2018

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A persuasive argument can be made that one of the most significant days in the evolution of hazardous material handling in industrial applications occurred in 1949. That's when inventor Ray Goertz first publicly demonstrated, at the behest of the U.S. Atomic Energy Commission for use at its Argonne National Laboratory, his invention that would come to be known as a "telemanipulator."

During that same era, three scientists from the Massachusetts Institute of Technology founded Central Research Laboratories (CRL) and began working on developing safer methods for handling hazardous and toxic products. This led to CRL's development of command-remote telemanipulators that could be used for the safe and efficient handling of nuclear materials by eliminating the need for the human operator to have direct contact with what could be extremely harmful and hazardous substances.

Since then, CRL has manufactured and installed more than 7,800 telemanipulators in 22 countries. Every telemanipulator that CRL manufactures is designed for the specific needs of the user and takes into account many varying factors in developing the most customized solution available.

With that commitment in mind, in the ensuing decades the design and operation of telemanipulators have undergone a series of technical enhancements and improvements. Most of these enhancements were to meet application specific needs.

Surprisingly, though, one critical component of the telemanipulator has been relatively immune to change; the handle that the operator grips and manipulates to complete the precise hand movements which are translated to, and mimicked by the telemanipulator tong.

Since the invention of the first telemanipulators, the handle design has stayed roughly the same, though there have been calls for its refinement. This paper will illustrate what the challenges have traditionally been in telemanipulator operation and construction, what changes were required in handle design, and how a new handle design can help improve performance in those areas.

Figure 14: The evolution of the VERSA Handle System.

Technical Session:

PIE of Fuels

Some Aspects Concerning Post-Irradiation Examination of CANDU Type Nuclear Fuel

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During the last years, a few defective CANDU fuel bundles have been transferred from Romanian Unit 2 NPP to Post Irradiation Examination Laboratory (PIEL) and examined in order to establish the cause of the defect. The RATEN ICN Post Irradiation Examination Laboratory was designed as a support for nuclear industry, especially as support for CANDU type power plants and there are available high-performance instruments and methods for irradiated fuel examination using nondestructive and destructive examination techniques.

In the frame of an IAEA program, a system for the transport of the spent nuclear fuel using a B(U) transport container type was designed, manufactured and licensed in order to transfer CANDU type nuclear fuel from power plants to Post Irradiation Examination Laboratory. This paper contains also a brief presentation of the transport container and auxiliary tools used for loading/ downloading the container.

The main methods implemented and used for post-irradiation examination of CANDU type nuclear fuel are: visual examination and photography of entire bundle and separately on fuel rods by a periscope, high precision dimensional measurements and profilometry scanning, axial gamma scanning and tomographic reconstruction of fission product distribution in the cross section of fuel rod, eddy current defect testing and oxide layer thickness measurement, fuel cladding puncture and fission gas analysis by mass spectrometry, metallography and ceramography by optical and scanning electron microscopy and mechanical testing on samples of fuel cladding. Also, radiochemical methods like mass spectrometry (ICP-MS, TIMS), gamma spectrometry, alpha spectrometry and high-pressure liquid chromatography are involved for chemical analysis and burn-up determination.

The paper will present relevant aspects concerning the post-irradiation examination of CANDU fuel bundles and fuel rods that contribute to establishing the cause of the defect.

Post-Irradiation Examination

Fuel transfer from the nuclear power plant to PIE Laboratory. After discharge from CANDU reactor core, bundles suspected of defects are visually inspected underwater and some of them selected for detailed post-irradiation examination in the hot cell laboratory. A system for the transport of the spent nuclear fuel using a B(U) transport container type was designed, manufactured and licensed in order to transfer CANDU type nuclear fuel from power plants to Post Irradiation Examination Laboratory. The transport package consists of a container body, basket with the bundle, two impact dampers, an overpack to tighten it on the truck, auxiliary tools for loading and unloading the bundle in the container and a device for pressuring using helium.

The container is loaded with defective bundle underwater in the spent fuel pool (Figure 15) and unloaded in the hot cell Figure 16) for further post-irradiation examination.

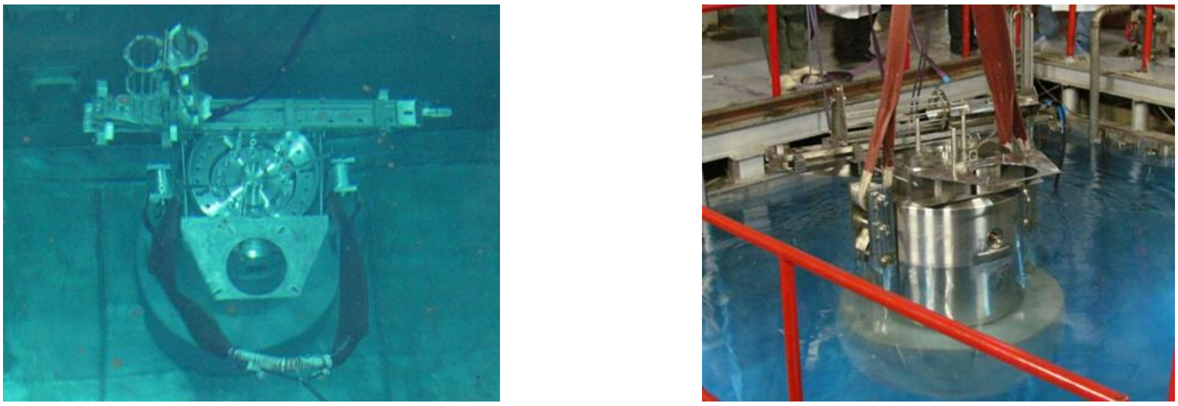

Figure 15: Loading the container in the pool at power plant site

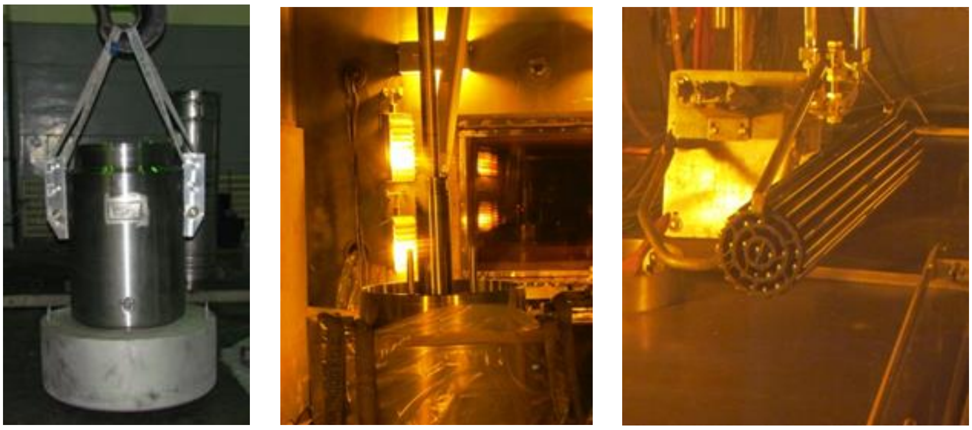

Figure 16: Unloading the container in the hot cell

Visual Examination and Photography. A device for precise positioning of the fuel bundle, a fuel element positioning machine, a periscope with magnification up to x12 and a high-resolution digital camera are used for visual inspection of both the entire bundle end fuel rods. The visual appearance of the surface gives a prime information about the cause of the defect (Figure 17) and also cracks less than 0.1 mm can be located (Figure 18).

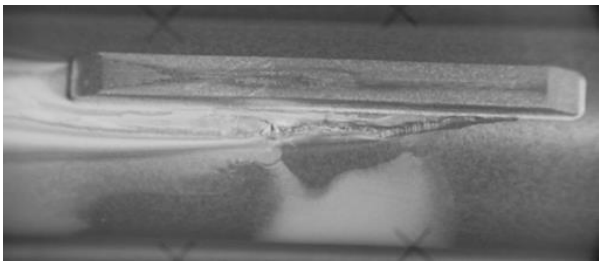

Figure 17: A cladding defect caused by debris

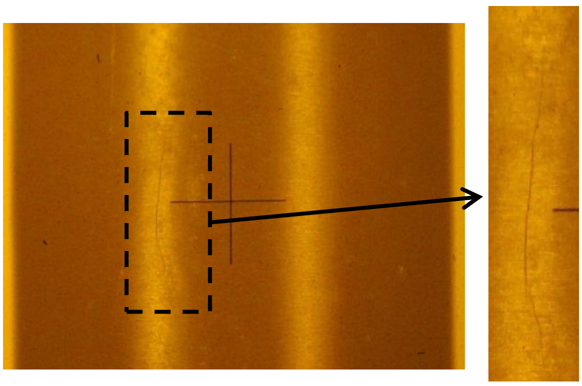

Figure 18: Thin crack highlighted by visual inspection

Macroscopic features of the irradiated nuclear fuel such as: deposits, corrosion, cracks, swelling can be examined by periscope.

Dimensional measurements and Gamma Scanning

A measuring device using two opposed LVDT (Linear Variable Differential Transformer) transducers (measurement range: ±2.5 mm) installed on the fuel element positioning machine is involved for dimensional measurements to obtain the parameters that highlight the dimensional changes of the fuel element during irradiation: profilometry of outside diameter over the entire length (Figure 19), length, bow and ovality with an accuracy of few microns.

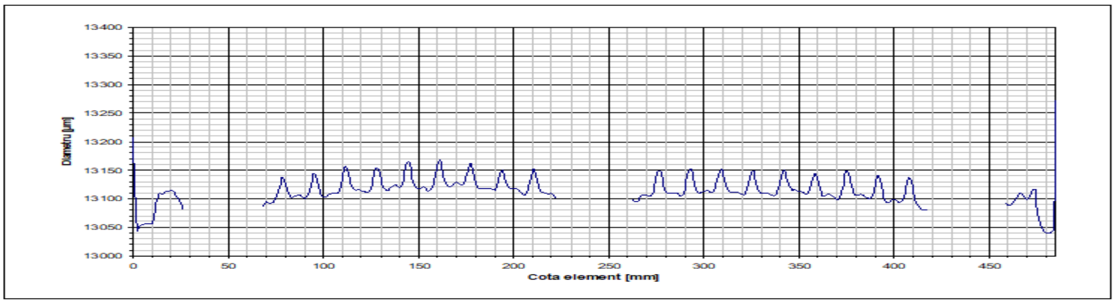

Figure 19: Outside diameter profile

Gamma Scanning

A fuel element positioning machine, a collimator and a high-resolution gamma detector are used for measurement of the axial distribution of fission products (FPs) activity in the fuel column: specific gamma isotopes activity profile such as ^{137}Cs (Figure 6), ^{134}Cs, ^{95}Zr-Nb, ^{103}Ru-Rh, ^{106}Ru-Rh, ^{140}Ba-La, ^{144}Ce-Pr and Burn-up determination.

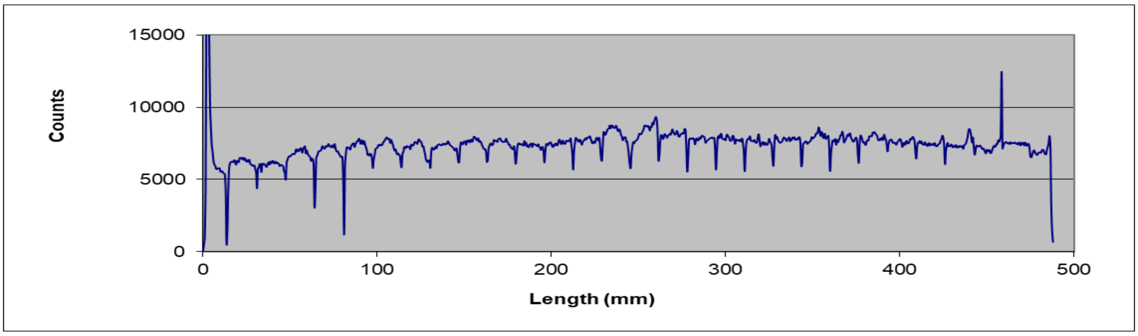

Figure 20: Gamma-scanning axial distribution of ^{137}Cs

Conclusions

During the post-irradiation examination of CANDU spent fuel were found defects caused by debris in the primary coolant, fuel column- end cap interaction, and stress corrosion cracking. Swelling cannot conduct to defects without a corrosion contribution.

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Failure Analysis on AFA 3G Gd Rod from Nuclear Power Plant

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The activity of the reactor coolant can indicate the failure of the fuel rod. The coolant radioactivity increased during reactor operation, and a Gd rod was found to be failed by poolside inspection. The failed rod was then transported to the hotcell for PIE to find out the root cause of the failure. Several reasons can cause the failure of the fuel rod, such as manufacture defects, PCI, grid-rod fretting, debris, handling, and each of these causes has their own feature on the primary failure. Once the failure occurred, the coolant can go inside the cladding and cause secondary failure by hydrogenation. To analysis the root cause of the failure, the primary failure should be identified from others.

The key method of primary failure identification is visual inspection. The high resolution visual inspection confirmed that both ends of the rod were failed, and the hydrogen content analysis showed that the hydrogen content at the position of lower plug welding was 1720 µg/g, while that was only 133 µg/g at the position nearby the upper plug hole welding. That means the failure at lower plug welding was a secondary failure, and the upper failure was primary. The failure root cause of Gd rod was hole welding defect.

PIE method

Body Text Several PIE method was performed on the failed rod for failure analysis, such as visual inspection, dimension measurement, X-ray radiography, γ- scanning, eddy current testing, oxide film thickness measurement, metallography, hydrogen content measurement, SEM and EDS.

PIE result

The high resolution visual inspection confirmed two failure on the rod, one was the hole welding, and the other one was lower plug welding, show as Figures 21 and 22, respectively. The whole hole welding part was missing, and there was a “H-type” cracking at the lower plug welding.

Figure 21: Hole welding failure

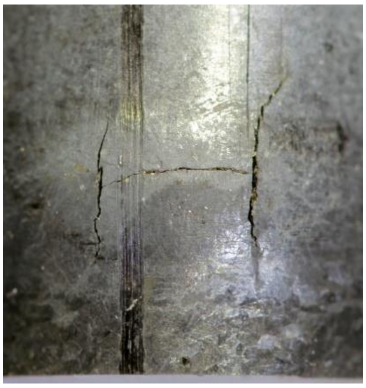

Figure 22: Lower plug welding failure

Hydrogen content analysis showed that the hydrogen content at the position of lower plug welding was 1720 µg/g, while that was only 133 µg/g at the position nearby the upper plug hole welding, the result showed at Table 1.

Table 1: Hydrogen content result

Sample position	Sample index	Mass(g)	Hydrogen content (µg/g)	Average content (µg/g)
5 mm lower than the upper plug	D1-WRU-H-1	0.0575	130	133
	D1-WRU-H-2	0.0582	136	
Lower plug	D1-WRD-H-3	0.1242	1720	1720

Conclusion

Due to the visual inspection result and the hydrogen content measurement, the failure at lower plug welding was a secondary failure, and the upper failure was primary. The failure root cause of Gd rod was hole welding defect.

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ASTM E-1447. C1457-00 (2010)e1 Standard Test Method for the Determination of Total Hydrogen Content of Uranium Oxide Powders and Pellets by Carrier Gas Extraction.

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Blistering test under the pressure condition in hot cell

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The blistering temperature is an important parameter of the plate type fuel assembly after irradiation, which affects its safe operation. How to increase the blistering temperature of fuel assembles is an important aspect of fuel development.

At present, the widely used method of measuring the blistering temperature is to heat to a certain temperature under the condition of atmospheric pressure, and then take out for the observation. This method has a disadvantage that it is impossible to obtain a blistering temperature under pressure condition close to the actual service pressure, which affects the performance evaluation of fuel assembles.

In order to obtain the blistering temperature under a certain pressure condition and study the formation process of the blister, a set of automatic blistering test device with pressure in a hot cell was designed. The equipment has been proved to be functional, and some interesting data are obtained, which can help to promote people's understanding of blistering behavior.

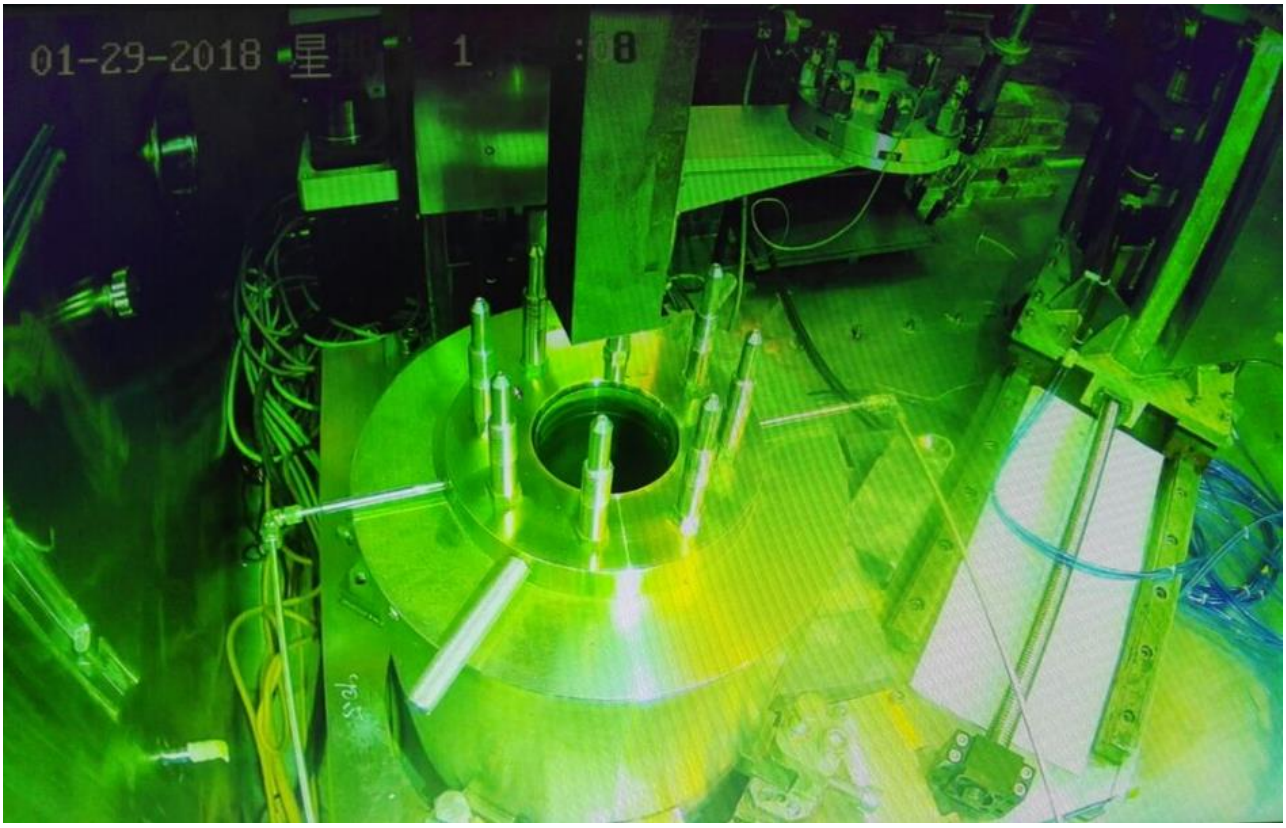

Figure 23: Automatic blistering test device

Blister Defect Analysis of U_3si_2/Al Nuclear Fuel Cladding by Ultrasonic Test

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Fuel integrity is one of safety considerations during nuclear reactor operation. Any defect resulted after reactor operation should be avoided. Thus, in nuclear fuel development, post-irradiation examination is necessary to be performed to ensure fuel integrity. Non-destructive test using ultrasonic method is one of alternative methods to perform post-irradiation examination. The objective of this analysis is to detect blister defect in cladding of plate type nuclear fuel. In post irradiation examination (PIE) using ultrasonic method, the operation parameters are necessary to determine. In this experiment, this analysis method was conducted on U_3Si_2/Al fuel element cladding consisting blister defect. Ultrasonic tests were done using two techniques, i. e. pulse-echo and through transmission techniques. The results were presented in A-scan display. The results show that the longitudinal wave speed in water used in the testing is 1545 m/s. The optimal distance between the probe and the specimen when using pulse-echo technique is 20 mm The optimal distance between the two probes when using through transmission technique is 40 mm. Tests at blister area using pulse-echo technique caused output signal to decerease by 19 %FSH, while through transmission technique caused output signal to decrease by 80 %FSH. According to the results it can be concluded that ultrasonic test is effective to detect a blister defect in plate type nuclear fuel, and through transmission technique exhibited a clearer difference between normal area and blister area in fuel cladding when compared to pulse-echo technique.

Keywords. *ultrasonic test, U_3Si_2/Al fuel cladding, blister, pulse-echo, through transmission.*

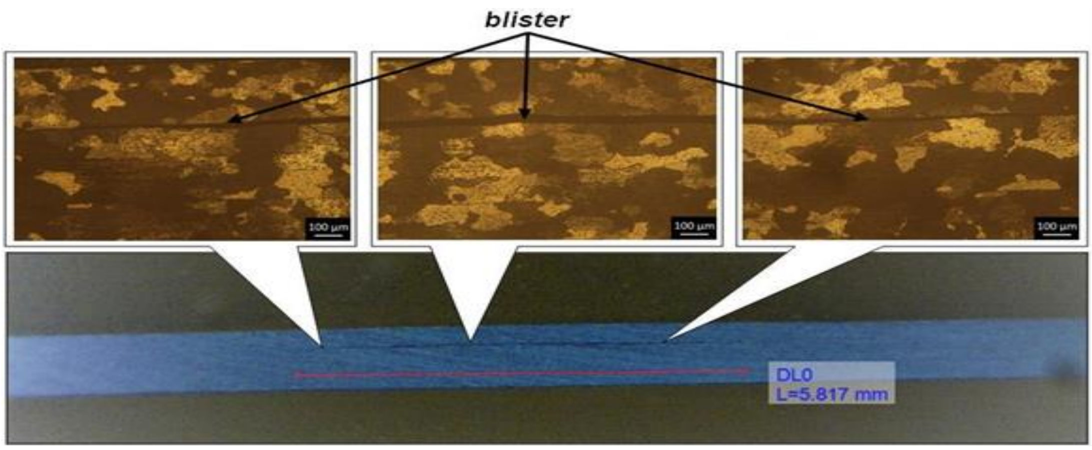

Figure 24: Microstructure blister defect from $U3Si_2$/Al Nuclear Fuel Cladding

Advanced Gas Reactor/TRISO Particle Fuel Re-Irradiation and Safety testing Experiment Performed in the NRAD Reactor and HFEF Hot Cell

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Post-irradiation examination (PIE) and safety testing of tri-structural isotropic (TRISO)-coated fuel particles from the second Advanced Gas Reactor (AGR) irradiation experiment, AGR-2, are in progress at the Neutron Radiography Reactor (NRAD) atthe Hot Fuels Examination Facility (HFEF) at the Idaho National Laboratory (INL). These fuels were irradiated for 559 effective full-power days in the Advanced Test Reactor; however, short-lived fission products with biological significance (such as I-131) decayed away before PIE cold be performed. The purpose of the re-irradiation is to generate short-lived I-131 (half life 8.0 days) and Xe-133 (half life 5.2 days) before heating tests in the Fuel Accident Condition Simulator Furnace (FACS) in the HFEF hot cell. There are three parts to the re-irradiation experiment. The first is to insert particles deconsolidated from irradiated fuel compacts into the NRAD reactor. Second is the heating test of the particles in the FACS. Third is the gamma spectroscopy of the furnace condensation plates, filters, and the fuel particles.

Intact TRISO coatings prevent fission product release. In order to study fission product release from the kernels, defects must be generated in the TRISO layers. Selected particles were cracked in order to damage all three TRISO layers and expose the fuel kernels. Inducing cracks in the particles to expose the kernels allows measurement of fission product releases from exposed kernels.

Components for Re-irradiation Experiment

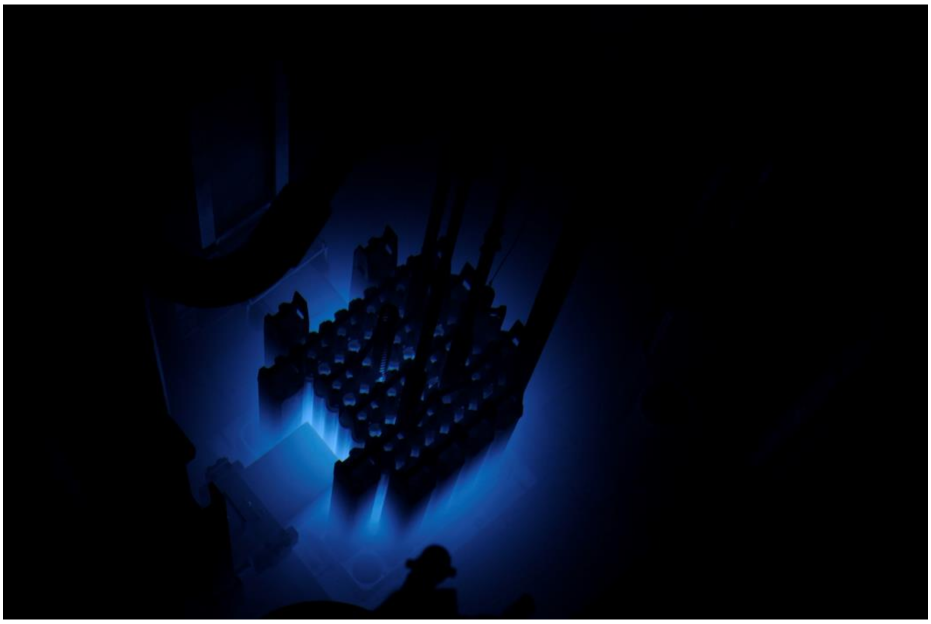

Figure 25: NRAD Reactor core

NRAD Reactor. The particles are placed into a titanium capsule and inserted into the wet tube C-4 SW core position of the NRAD reactor. The capsule is irradiated for up to 4 days of shift operation (8 hours at full power followed by 16 hours at zero power). At the end of the fourth irradiation period, the capsule is left in the reactor for a minimum 15 hours for decay time. The capsule is then transferred to HFEF and placed into the FACS furnace. Some particles were gamma counted in between re-irradiation in NRAD and loading into the furnace.

***Heating test in the FACS furnace*.** The particles are removed from the irradiation capsule and placed on a sample holder in the FACS. As the particles are heated to ~1600ºC, a helium sweep gas flows past the sample. The gas flows past a condensation plate that is attached to a cold finger. Condensable fission products such as I-131, Cs-134, Cs-137,Eu-152, Eu-154, and Ag-110m are collected on the plate. Condensation plates can be exchanged for new plates at predetermined times throughout the heating test. The sweep gas exits the furnace and flows through a silver zeolite filter and into an LN2 cold trap. Fission gases from the particles, such as Kr-85 and Xe-133, are collected in a charcoal canister (in the LN2 cold tarp) and are counted by High Purity Germanium (HPGe) detectors. The zeolite filter is later removed and counted (primarily for I-131) out of cell.

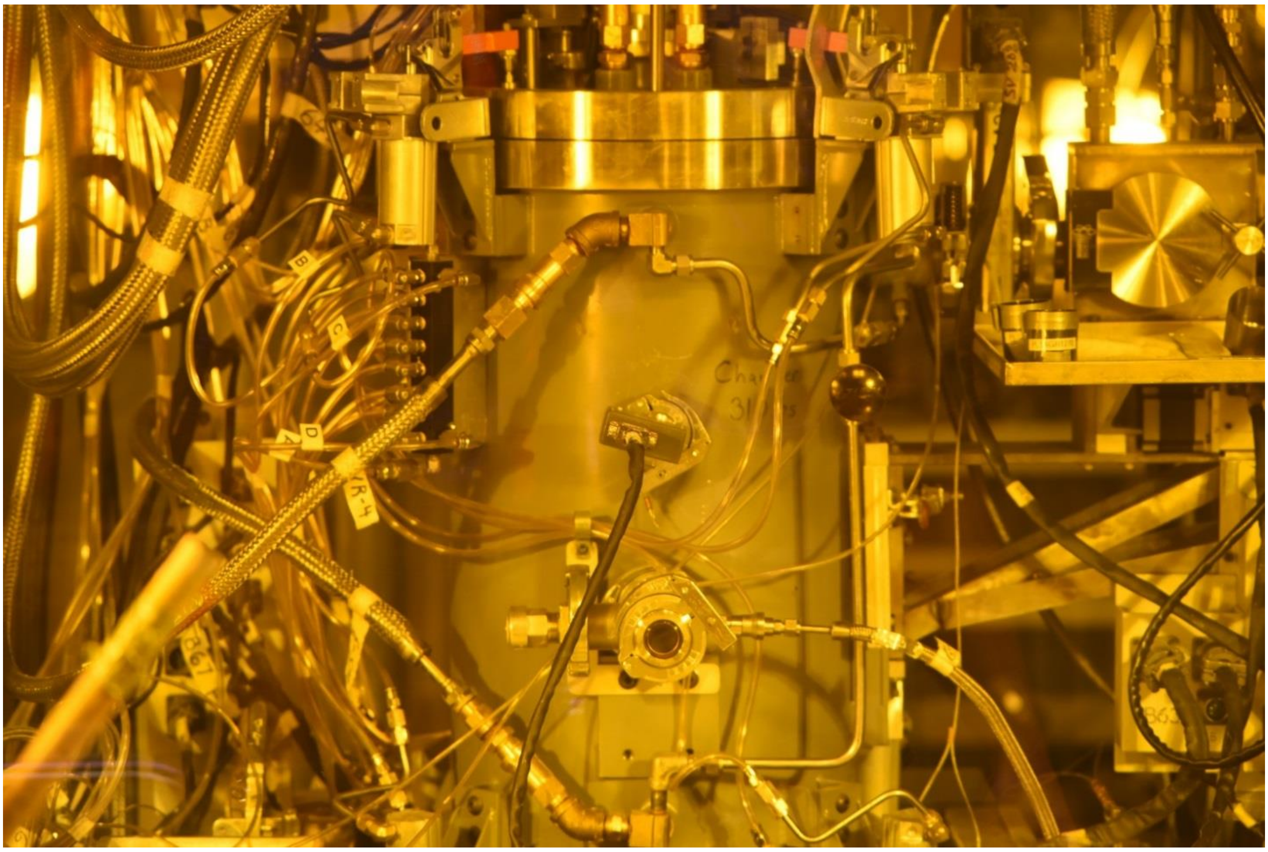

Figure 26: The FACS furnace.

Gamma Spectroscopy in the HFEF Out of cell Gamma Counting Stations (HOGS)

Throughout the run, condensations plates are transferred out of the HFEF hot cell and over to the HOGS. The HOGS consists of 2 shielded HPGe detectors that communicates with an ORTEC DSPEC-502. The DSPEC-502 contains two complete spectrometer electronics, including detector high voltage, amplifier, and an analog-to-digital converter. The information is then sent to a Data Acquisition computer. After the heating test is complete in the FACS, the zeolite filter and particles are counted as soon as possible so that the short-lived fission products (i.e. I-131 and Xe-133) are not lost through decay.

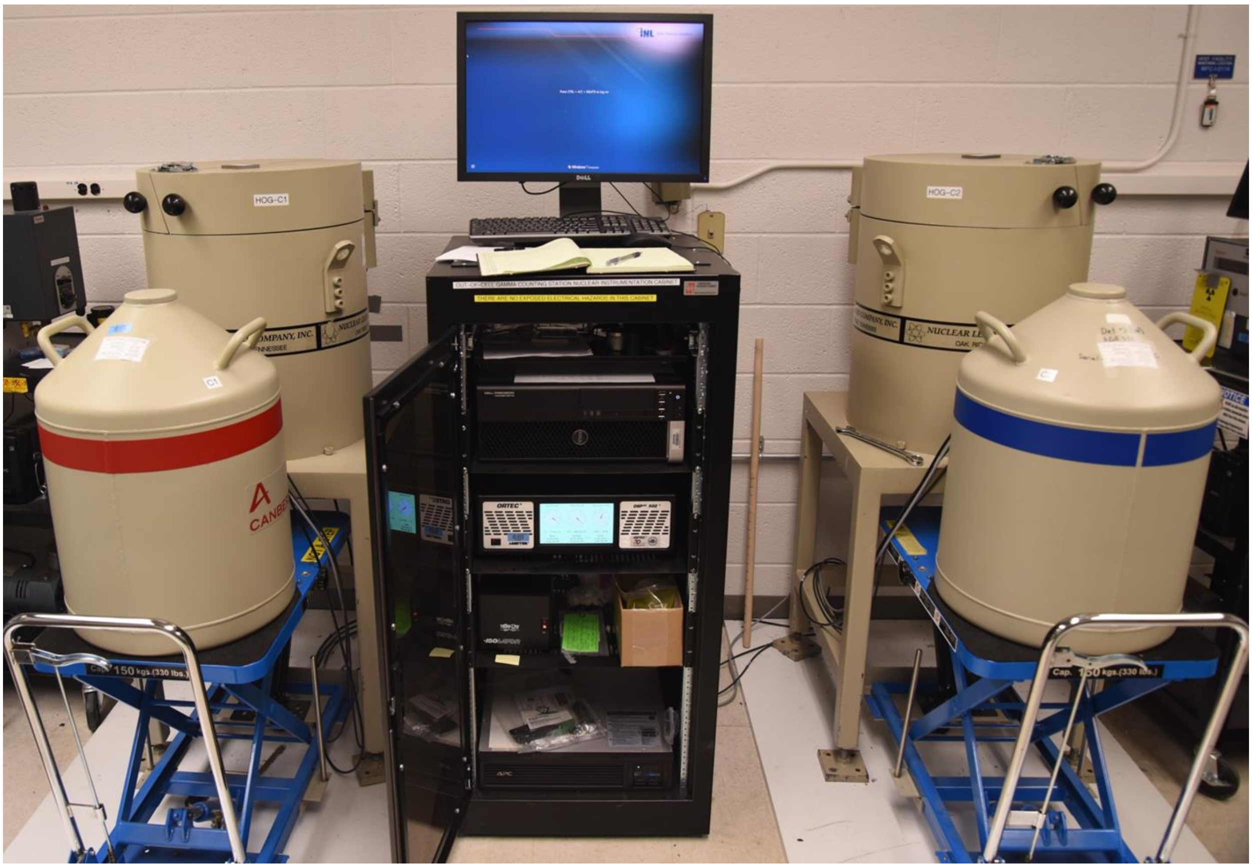

Figure 27: The HOGS

Conclusion

The unique ability the INL has to use the NRAD reactor to generate short-lived isotopes in previously-irradiated fuels, heat the fuel to simulated accident temperatures in the FACS, and analyze the components in the HOGS in a limited time frame gives the AGR program a better understanding of how these isotopes behave in the fuel under accident scenarios.

Radial Deconsolidation of Irradiated AGR-3/4 Compacts at Idaho National Laboratory

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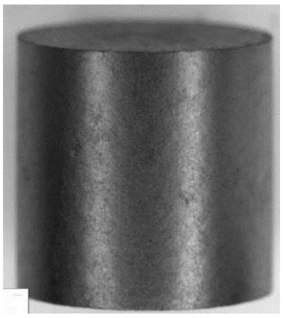
The Idaho National Laboratory (INL) evaluated the concentration gradient of fission products released in the AGR-3/4 experiment by selective removal of annular volumes of the fuel compacts. Purpose-built designed-to-fail particles aligned along the axis of an experimental compact released fission products whose concentration was determined by progressive radial electrochemical dissolution of the binding matrix material. Precise dimensional measurement was achieved by video imaging of the eroded compact at each of several steps. The video measurement of the compact diameter eliminates the need to attempt manual caliper measurements that may lack precision and may damage the sample surface.

AGR-3/4 is part of a series of irradiation experiments of high-temperature gas reactor TRISO particle fuel produced and tested by the Advanced Reactor Technology group of the Idaho National Laboratory. TRISO particles tested in the previous two experiments (AGR-1 and -2) performed at a level in which release of gaseous krypton and xenon fission products was undetectable during irradiation, and during post-irradiation examination, evidence of particle coating failures was no greater than 0.03 %. To investigate fission product transport and deposition in structural graphite, AGR-3/4 was designed to include 20 designed-to-fail (DTF) particles arranged at the axis of 12.3 mm OD x 12.5 mm long compacts, each containing nominally 1872 particles. The DTF particles consist of a standard 350 um diameter TRISO uranium oxycarbide kernel surrounded by 20 um layer of pyrolytic carbon, which is not sufficient to retain fission products. A photo of a typical compact and an x-ray image highlighting the DTF orientation are shown as Figure 28.



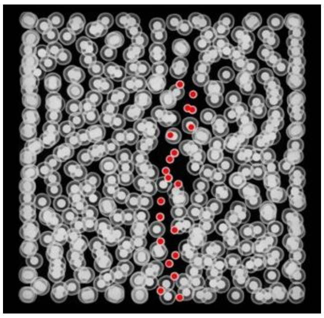


Figure 28: 12.5 mm diameter x 12.3 mm long compact (left), X-ray of compact showing DTF particles in red (right)

To investigate the gradient of fission product transport within the compacts, the normal method of complete electrolytic deconsolidation of the compact is modified and the TRISO particles are separated from the compact by rotating it against a platinum-rhodium screen anode while partially immersed in a nitric acid anode, allowing an approximately 1 mm layer of particles to be removed from the surface of the compact during each sequential step. This work is done in an individual air atmosphere hot cell located at the INL Materials and Fuels Complex MFC-752 Analytical Laboratory. The cell floor area is approximately 3.6 m^2, is viewed through nominal 1 m^2 oil-filled leaded glass window, 60 cm thick and is serviced by 2 CRL Model L-HD manual master-slave manipulators. Because this cell already contains a shielded macroscope, an oxidation furnace and glassware for leaching of particles, the equipment for doing the deconsolidation step was designed to fit into a nominal 30 cm^3 volume.

The image processing software was developed by Grant Helmreich using Matlab® (Mathworks), and analyses individual frames from video recording of the rotating compact to measure the compact diameter at multiple points to a precision approaching 0.01 mm on a 12.3 mm diameter.

Process Description

The process of deconsolidation proceeds by electrolytically breaking down the matrix carbon, which is partially graphitized by the addition of electrical current and nitric acid. This process owes some of its heritage to Heinz Nabielek, who developed a variation of this technique to determine variations within "pebbles" from the German Pebble Bed High Temperature Gas Reactor. The description given here applies to the approach used at INL to selectively remove layers of particles for differential evaluation of the resulting deconsolidation solution. It is the result of development work at Oak Ridge National Laboratory and Idaho National Laboratory.

Mechanically, the compact is glued onto a 6 mm OD stainless-steel shaft that is driven by a 12V DC, 10 rpm gearmotor that rotates the compact against the Pt-Rh anode screen. The shaft and the compact are joined using graphite-filled epoxy (Atom Adhesives AA-CARB-61). The task of accurate joining is done by mounting the geared shaft and its manipulation handle in a machined fixture that aligns the shaft at the center axis of the compact. The 2-part epoxy is mixed outside the hot cell, and transferred into the cell in a 1 ml plastic syringe. The glue is then injected down the hollow shaft, filling the gap between the compact and the end of the shaft. After a 24-hour ambient temperature cure, the geared shaft and compact are mounted in the drive unit and positioned over a beaker of 4M HNO_3 that has a Pt-Ir cathode wire placed in it. The anode is placed in contact with the compact cylindrical surface, and lowered to the point that the compact surface just contacts the surface of the acid. The motor is actuated, and the power supply is turned on to apply a voltage of <10 VDC at a current of < 1 A. As the compact is rotated, the external surface is eroded as the matrix carbon is broken down.

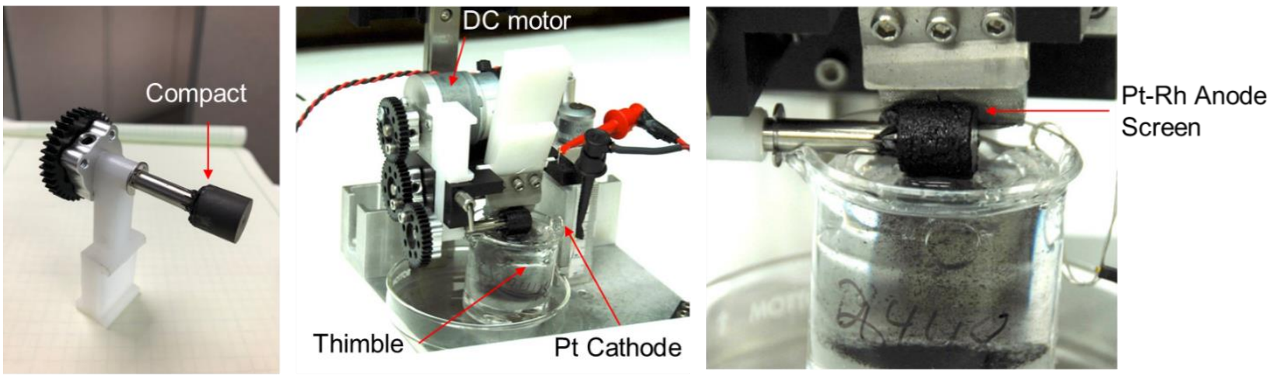


Figure 29: Compact glued on shaft with drive gear and manipulator handle (left), deconsolidation unit (center & right)

The simplified chemical intercalation process is as follows:

- Cathode (reduction):

$$HN(V)O_3 + e^- \rightarrow 2\ N(IV)O_2 + OH^*$$

- Anode (oxidation):

$$C(0) + 4\ OH^* \rightarrow C(IV)O_2 + 2\ H_2O + 4\ e^-$$

The net effect is that the matrix is broken down into carbon dioxide and nitrogen oxide as a result of the imposed current without adverse effect on the TRISO particle coatings. During initial unirradiated development of the axial deconsolidation process, it was determined that electrolytic input in excess of 10 W led to breakdown of TRISO coatings, so the process has been limited to the 10V/1A value.

The particles released from the matrix are collected in a porous-bottomed container called a "thimble". After a prescribed period (typically 15 minutes), approximately 1 mm of the surface has been removed, and the process is stopped. The gear driven shaft and compact are raised from the acid, and a video of the rotating shaft is made through the cell window to produce image data that can be analysed to determine the precise diameter of the compact at each of the completed intervals. The beaker is replaced with one containing fresh acid and an empty thimble, and the process is repeated typically two more times, until the remaining material of the compact is the same diameter as the shaft to which it is glued. The final step is to deconsolidate the shaft diameter material by progressively electrolytically dissolving it in acid with the same cathode in a beaker, but using a Pt-Ir electrode imbedded in the glue to complete the circuit. The acid from each step is analysed by gamma ray spectrometry and strontium separation to determine fission product content. Mass spectrometry is used to detect the presence of uranium and other non-gamma emitting products released from the DTF particles.

Using the volume removed in each deconsolidated layer and data from the fission product solution analysis, it is possible to determine fission product concentrations and evaluate individual product mobility and the gradient within the compact. The compact gamma ray emitting isotope inventories were established prior to deconsolidation using precision gamma scanner located at the Hot Fuel Examination Facility.

Image Analysis

The process of video analysis of the diameter of the progressively eroded compact is used because attempting to measure by physical contact using a mechanical caliper can damage the surface of the compact, as well as break the glue joint, ending the process unless it can be precisely re-glued. The video analysis allows precise prediction of the number of particles removed in each radial deconsolidation step, as well as the volume of the matrix dissolved. Groups of 10 images from a given compact rotation selected from up to 15 minute-long videos taken at 24 fps with 1280x720 resolution were analysed to determine the compact diameter.

The video is done using a Nikon D5000 digital single-lens reflex camera with a Nikkor 80-400mm telephoto zoom lens attached. The camera is positioned at approximately 2 m from the window, with the deconsolidation equipment 30 cm from the inside of the window. The lens is used at the 400mm full zoom setting. The orientation of the camera and the general configuration of the in-cell deconsolidation equipment is shown in Figure 30.

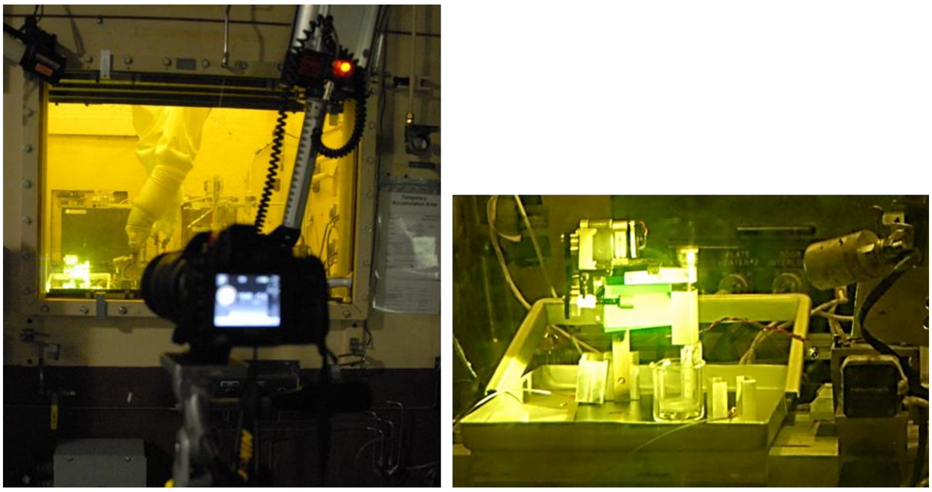


Figure 30: Camera location relative to hot cell window (left), Deconsolidation unit (right)

When the compact and shaft are raised from the acid, a small green-screen background is located behind it, allowing the image-processing software to distinguish the dark, rectangular silhouette of the compact and calculate the area and thus the compact diameter. The image analysis software finds the outline of the compact and the mounting shaft by the use of active contours, then corrects for image tilt based on the observed angle of the mounting rod. The diameter of the compact was then found by taking the average height of the masked compact in the tilt-corrected image.

All of the compacts were manually measured in the HFEF hot cell as a part of dimensional checks done on all components following disassembly of the experiment, so when each compact was mounted on its shaft, it was rotated in the drive system while recording video prior to deconsolidation. This baseline video was used to do initial lighting and alignment adjustments to ensure that the dimensions measured by image analysis were plus or minus 0.01mm.

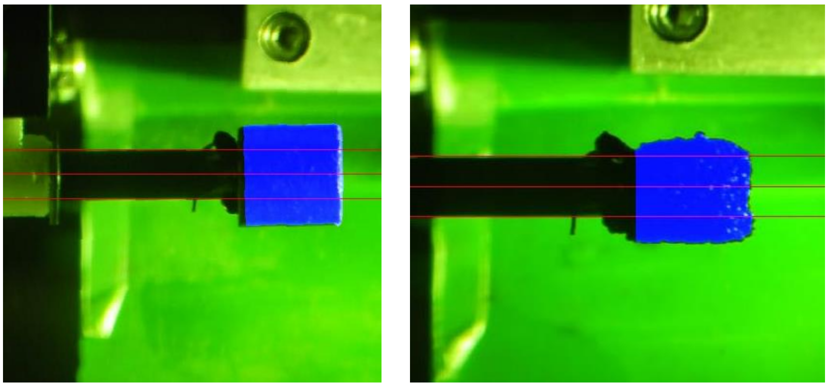


Figure 31: Compact on shaft in false color (blue) and shaft diameter (red) registration lines; Before (left), and after (right) deconsolidation

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Technical Session:

PIE of Structural Materials

Correlation of Pressurized Water Reactor Vessel Material Properties Variation with Neutron Fluence by Surveillance Program

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The surveillance capsules included in the reactor vessel surveillance program of Nuclear Power Plants in Taiwan had been removed from the reactor and shipped to the hot laboratory in the Institute of Nuclear Energy Research (INER) for examination after being irradiated for scheduled periods. The objective of the surveillance program is to correlate the mechanical properties of surveillance specimens with accumulated neutron fluence the specimens experienced during irradiation, and then to evaluate the adequacy of fracture toughness of reactor vessel materials for continued operation. Based on ASTM Standard E185-82, Charpy V-notch impact testing and tensile testing were the main technologies used to measure the mechanical properties of the surveillance test materials. There were also neutron radiometric monitors, including iron wire, nickel wire, copper wire, uranium-238 in U_3O_8 powder and neptunium-237 in NpO_2 powder, encapsulated in the capsule for evaluating the neutron fluence by a combination of activities measurement of neutron radiometric monitors and computer code analysis according to Regulatory Guide 1.190. By means of Regulatory Guide 1.99, Revision 2, the results of mechanical properties measurement and neutron fluence analysis were then employed to evaluate the extent of radiation embrittlement of reactor vessel material specimens. The pressure-temperature limit curves for reactor operation and the PTS (Pressurized Thermal Shock) analysis can also be deduced from the correlation, evidencing that the reactor pressure vessel will be kept being bounded by the requirements of 10 CFR APPENDIX G during its life time.

Hot Laboratory Examinations

***Mechanical Testing*.** The post-irradiation mechanical testing, including Charpy V-notch and tensile tests, was performed in accordance with 10 CFR 50, Appendix H and ASTM Standard E185-82. Three kinds of surveillance test materials were tested: weld, heat affected zone and base materials. As per the specimen orientation and location specified in ASTM E185-82, tension and Charpy specimens were machined from the quarter-thickness (1/4T) locations of the representing surveillance test material plate.

- *Charpy V-notch Impact Test.* The Charpy V-notch impact tests were conducted in accordance with ASTM E23-07a with a Tinius Olsen Impact Tester. The maximum impact capacity of the Tinius Olsen Tester set up in the lead cell of INER is 359 Joules. This tester is operated remotely and the specimens can be loaded in automatically, making it possible to implement an impact test within five seconds upon the specimen being removed from the thermostatic sample room. Gas is used as the media for cooling or heating specimens during the test. Test specimens were kept at the test temperature for 40 minutes before testing in accordance with ASTM E23-07a. In accordance with ASTM E2298, the hammer tip is instrumented with a strain gauge system, which yields dynamic impact energy and additional characteristic values in addition to the force-time curve.

- *Tensile Test***.** According to ASTM E8/E8M-09 and E21-09, the tensile tests were conducted. The tensile tester can exert a maximum axial tensile force (load) of 100kN (22000 lb). The tester is also set up in the lead cell, operated remotely. Besides, the tester is equipped with an electric resistance heater. All of the tensile tests were performed under a constant cross head speed of 0.01cm/min(0.005in/min).

Neutron Radiation and Dosimetry Analysis**.** Neutron fluence rate calculation was performed and interactively compared in two parts: neutron transport analysis and passive radiometric monitor measurement. Computational codes of ANISN and DORT were used to calculate the discrete ordinates transport of neutron (Rhoades & Childs, 1998). Iron, nickel and copper wire together with oxide powder of uranium-238 and neptunium-237 were used as the dosimeter to evaluate the representative neutron flux for neutron energy > 1 MeV. SAND-II (McElroy et al., 1967) offered a method to perform least squares analysis technique for the neutron spectrum unfolding, leading to a neutron fluence rate spectrum closer to reality.

- *ANISN and DORT codes*. In the neutron fluence rate analysis within the reactor geometry, both 1-dimention and 2-dimention discrete ordinates transport computer codes played an important role in the calculations. In the 1-dimention analysis, ANISN supported the calculation of the neutron fluence rate in a cylindrical geometry. On the other hand, DORT was used to construct a 2-dimensional model, including R-Θ geometry and R-Z geometry, calculating the neutron fluence rate distribution for different azimuthal angle and height.
- *SAND-II code*. SAND-II provides an adjustment method by means of combining the results of neutron transport calculations with neutron dosimetry measurements in order to obtain an optimal estimate for neutron spectrum with assigned deviation. All the quantities input into SAND-II are simultaneously and iteratively adjusted to give an output spectrum with the minimum weighted least squares error.

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Miniature C(T) Specimen Fabrication for Reutilization of Surveillance Tested Materials

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Irradiation embrittlement of the Japanese reactor pressure vessel (RPV) is monitored by the tests on the specimen of the RPV material irradiated based on the surveillance test program. Lately, the Master Curve (MC) approach has become the main stream in fracture toughness evaluation.

However, the number of specimens is limited due to capacity of a surveillance capsule. Because some plants have few fracture toughness specimens contained in the surveillance capsules due to capacity and the minimum number of fracture toughness data for the evaluation by the MC method is six, sufficient fracture toughness data could not be obtained.

Therefore, it is expected to obtain additional fracture toughness data using broken specimens of irradiated materials after Charpy test. It is effective in extension of fracture toughness database to utilize broken halves of Charpy specimen because a large number of Charpy specimens are generally contained in the surveillance test capsule.

Specimen reconstitution by welding can be a candidate to solve this problem. However, it requires specific equipment and time-consuming machining operations for welding. Furthermore, only one three-point specimen can be made from a broken half of Charpy specimen to avoid the heat affected zone due to welding.

As an alternative to specimen reconstitution by welding, fracture toughness test using 0.16T-C(T) (Mini-C(T)) specimen, which has dimension of 4×10×10mm, is available without welding, and moreover four Mini-C(T) specimens can be taken from a broken Charpy specimen.

In this study, machining process in hot cell was considered in order to make it possible to fabricate Mini-C(T) specimen for irradiated materials with sufficient accuracy. And, the Mini-C(T) specimen fabrication technique from a broken half of Charpy specimen used in the surveillance test, was established.

Mini-C(T) fabrication from a broken Charpy specimen

***Geometry and dimensions*.** Figure 32 illustrates C(T) specimen design are recommended in the ASTM E1921 (American Standard for Testing and Materials, 2018; Miura & Soneda 2010). Because the ASTM standard do not limit the available specimen size, small size C(T) specimens including Mini-C(T) specimens can be used.

The Mini-C(T) specimen slit orientation is the same as the crack propagation direction of the Charpy specimen, so that 1.25 W defined in Figure 32 is 10mm for the Mini-C(T) specimen as shown in Figure 33. Central Research Institute of Electric Power Industry (CRIEPI) has developed a MC based fracture toughness evaluation method for the Mini-C(T) specimen (Yamamoto et al., 2012; 2013; 2015) and verified applicability of the Mini-C(T) specimen to the MC evaluation has been studied in a series of international round robin test programs coordinated by CRIEPI. In these programs and the related studies, it was demonstrated that the reference temperature (To) can be determined by the Mini-C(T) specimens without any specific difficulties for the unirradiated RPV base metals.

For the above reasons, Mini-C(T) was adopted as specimens taken from broken halves of Charpy specimen.

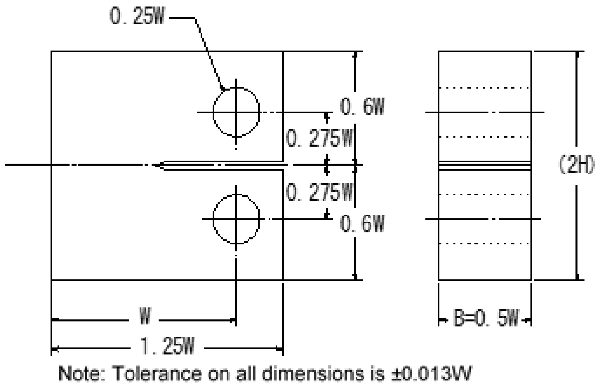

Figure 32: Geometry and dimensions of C(T) specimen in ASTM E1921[1].

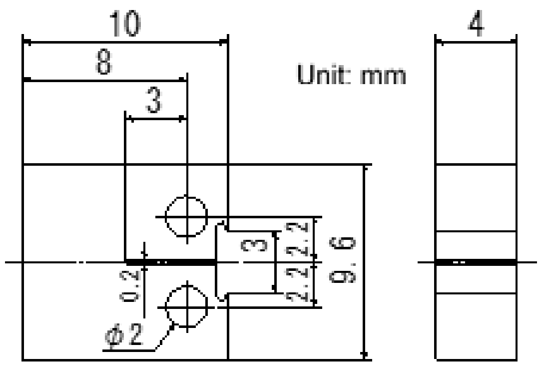

Figure 33: Geometry and dimensions of Mini-C(T) specimen.

Fabrication procedure for Mini-C(T) specimen

Four miniature C(T) specimens were machined from a broken half of Charpy specimen as shown in Figure 34, so that the orientation of the crack coincides with that of the Charpy specimen. According to the ASTM standard, dimensional tolerances are defined by the ratio to the specimen width W as shown in Table 2. This means that the dimensional tolerances for Mini-C(T) specimen are severe compared with larger specimen. For this reason, because fabrication of specimen seems to be difficult in hot cell using the limited existing equipment, it is important to establish procedure for Mini-C(T) specimen fabrication.

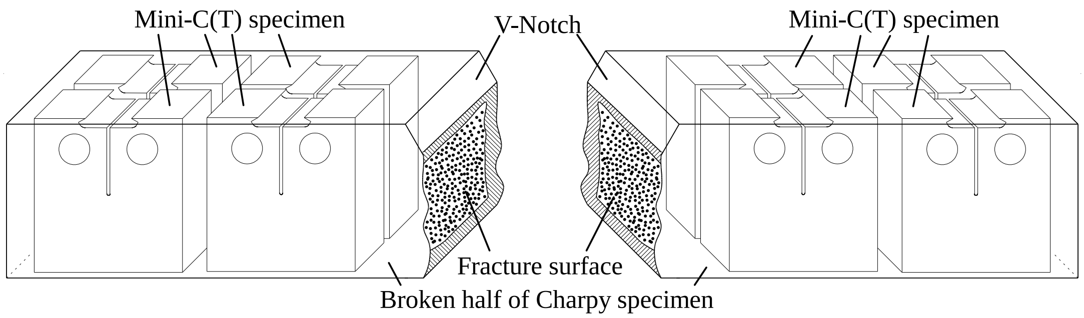

Figure 34: Orientation of Mini-C(T) specimens machined from broken halves of Charpy specimen.

Table 2: Dimensional tolerances of C(T) specimen specified in ASTM E1921.

Specification in ASTM E1921	**Mini-C(T) specimen**	**1/2T-C(T) specimen**
±0.013W	±0.1mm	±0.33mm

In order to fabricate Mini-C(T) specimen in high accuracy, the wire cut EDM (electrical discharge machine) was used mainly. Image of machining process for Mini-C(T) specimens from a broken half of Charpy specimen is shown in Figure 35.

【Machining process】

1. Cutting of fracture surface
2. Machining of prepared holes by drill
3. Machining of outline by wire cut EDM

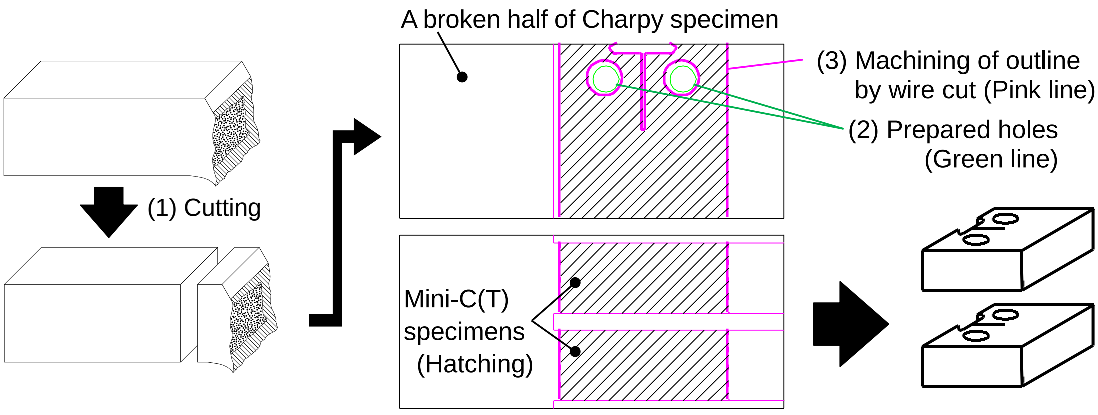

Figure 35: Image of machining process for Mini-C(T) specimens.

Prototypes of Mini-C(T) specimen were machined from a unirradiated material piece, and the prototypes of Mini-C(T) specimen fabricated in radiation controlled area have sufficient accuracy. The procedure established in this study is adopted to Mini-C(T) specimen fabrication from surveillance test materials after Charpy test.

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The microstructure of post-irradiated A508-3 steel and its effects on charpy impact energy

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Irradiation hardening and embrittlement of reactor pressure vessel (RPV) steels, which is a key issue for the safety and lifetime assessment of nuclear reactors, are the consequence of microstructural changes under neutron irradiation (Mazouzi et al., 2011; Azevedo 2011; Little 1986; Terentyev et al., 2015). To apply Chinese A508-3 steel into RPVs, extensive research on its mechanical properties relevant to nuclear power plant environment has been carried out. The previous results of Charpy V-notch impact test reveals that the impact energies of post-irradiated A508-3 steel in the same test environment were unstable.

In order to find the possible reasons causing the instability of Charpy impact energy in A508-3 steel, the microstructure, grain size, defects, and non-metallic inclusions of post-irradiated specimens were observed by using optical microscope (OM), scanning electron microscope (SEM), and energy dispersive spectrometer (EDS) with shield in this paper. The results indicated that no obviously change in the bainite structure and grain size of the domestic A508-3 steel could be observed by OM under this irradiate conditions (fluence of neutron is 2.97×10^{19} n/cm^2, and irradiation temperature is 290℃±15℃), which could not lead to impact energy abnormally. Instead, the direct reason probably is the differences in fraction volume of the defects(cavities) in the matrix, which formed during solidification process. In addition, the defects could be divided into two types, one is filled with layer-like Al_2O_3, MnS, and Al-Mg-O ternary non-metallic inclusions combining together, and the other is voids. Furthermore, the boundary between these non-metallic inclusions and matrix is quite loose, which is easily to decrease the impact toughness. Otherwise, except MnS phase, the morphology and composition of the Al_2O_3 and Al-Mg-O ternary non-metallic inclusions in defects were not modified by neutron irradiation significantly under this irradiate conditions.

Experimental specimens

Detailed information of each sample is shown in Table 3.

Table 3: The results of impact testing

Specimens	Irradiation temperature T（℃）	Fluence（n/cm^2）	Impact temperature（℃）	Impact energy E（J）
1#	290±15	0	24	263
2#	290±15	2.97×10^{19}	0	213
3#	290±15	2.97×10^{19}	0	8

Results and Discussion

In order to analyze the inclusions and cavities in further, and confirm the effect of cavities on impact energy, the unirradiated and post-irradiated samples were analyzed by SEM and EDS. Figure 36 Figure 36 is the SE and BSE images, and X-ray mapping results of the inclusions in 1# unirradiated sample. As shown in Figure 36a, the inclusions exhibit irregular island-like morphology. Combine with the BSE image and X-ray mapping results in Figure 36, it is easy to find that three kinds of inclusions grow together as an ensemble. Based on EDS results, these inclusions (indicated by arrows A in Figure 36a, and B and C in Figure 36 b)have the compositions equivalent to Al_2O_3, $MgAl_2O_4$ (Reshak et al., 2014) and MnS respectively, which are commonly reported in A508-3 steels. In addition, some Ca, Cu, Mo, and Si atoms are dissolved inside the MnS inclusion. Furthermore, although these inclusions combine together very tightly, the boundary between these non-metallic inclusions and matrix is quite loose, which is easily to decrease the impact toughness.

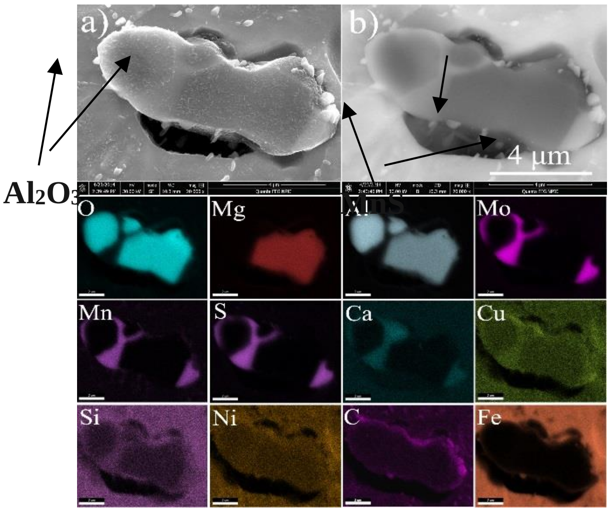

Figure 36: Secondary electron a), backscattered electron b) images and X-ray mapping results for non-metallic inclusions of Sample 1

The SEM images and X-ray mapping results of inclusions in 2# and 3# post-irradiated samples is presented Figure 37 and Figure 38 respectively. The Al2O3 and MgAl2O4 inclusions still could be observed after neutron irradiation. Compare with that, the MnS inclusion is absent Little 1986). In addition, there is no obviously differences in chemical composition between unirradiated and post-irradiated samples in the MgAl2O4 inclusion. But for the Al2O3 inclusion, after neutron irradiation, some Ca atoms(approximately 3.6 to 3.7 at.%) were dissolved inside. At the same time, the concentration of Al element in phase decreased from about 43.5 to 40.0 at.%. Furthermore, compare with Figures 36, 37 and 38, it seems that the Ca atoms which previously dissolved in MnS phase transfer into Al2O3 phase during irradiation process. However, mechanism of the transfer behavior of Ca atoms needs further investigation.

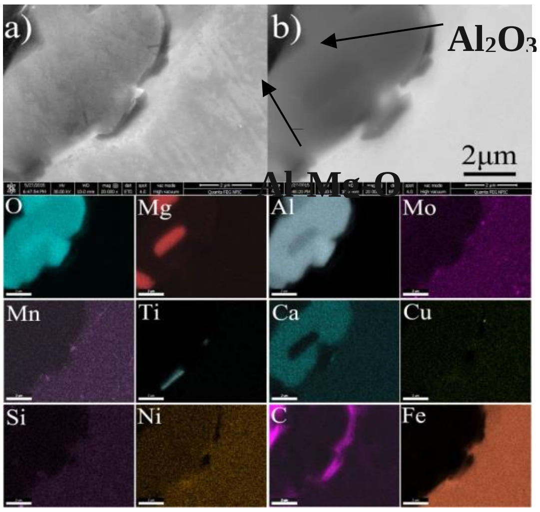

Figure 37: Secondary electron a), backscattered electron b) images and X-ray mapping results for non-metallic inclusions of Sample 2

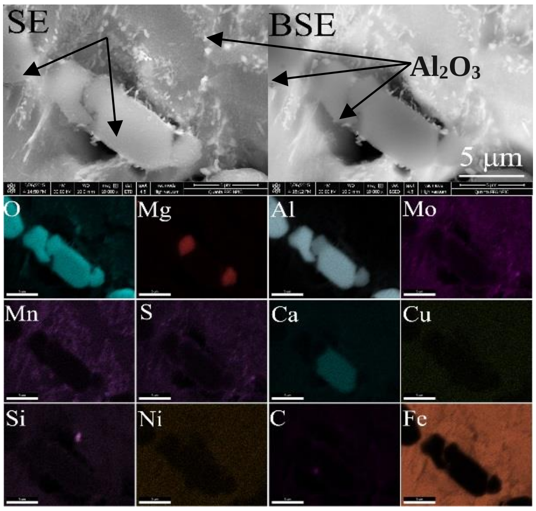

Figure 38: Secondary electron a), backscattered electron b) images and X-ray mapping results for non-metallic inclusions of Sample 3

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Mechanical property evaluation of irradiated stainless steels using sub size and miniature specimens

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Austenitic stainless steel is the major core structural material of sodium cooled fast reactors in India. The various stainless steels chosen for use in the upcoming Prototype Fast Breeder Reactor (PFBR) are (i) alloy D9 (in 20% CW) as material of replaceable core components like fuel cladding and wrapper and (ii) SS316 LN/ SS304 LN as material of permanent structures like main vessel, safety vessel, grid plate etc. The performance of the stainless steels has been studied through irradiation experiments in Fast Breeder Test Reactor (FBTR) and their mechanical properties and microstructure evaluated in hot cell experiments.

The mechanical properties have been evaluated using (a) sub size tensile and impact specimens with dimensions proportionally reduced from standard ASTM specimen and (b) miniature disc specimens of diameter 8.0 mm and 0.5 -1.0 mm thickness. As compared to standard size specimens, sub size and miniature disc specimens (i) permit efficient use of costly reactor space for test irradiations (ii) experience lesser flux and temperature gradients across its volume and (iii) permit easy handling due to reduced activated dose. The test procedures of various mechanical tests on irradiated specimens and the trends in mechanical properties of the stainless steels as function of displacement damage will be presented in this paper.

Experimental - Sampling and test methods

The test samples for mechanical testing were sourced from (i) irradiated clad & wrapper tube components of alloy D9 with displacement damages in the range of 40-60 dpa and (ii) pre-fabricated specimens of SS316 LN and SS304 LN irradiated to displacement damages of about 2-5 dpa in experimental capsules.

For tensile test of alloy D9 claddings, tubular clad section of 60 mm length extracted from various axial locations of the fuel pin were used after removing the fuel by chemical dissolution. Mandrels inserted from the ends of the tube enabled gripping and the free distance inside the clad tube between the mandrels (set as $5.65\sqrt{\text{Area}}$) was taken as the initial gauge length. Sub size flat tensile specimens (total length: 40 mm, gage length: 12.0 mm, gage width: 3.0 mm) were machined from alloy D9 wrapper faces using a 4-axis CNC machine for tensile testing. A customised wedge action gripping system was employed for holding the tubular and flat tensile samples in hot cell UTM fitted with an electric resistance furnace. In addition to the tensile tests, shear punch (ShP) tests were carried out on disc specimens (8.0 mm diameter and 1.0 mm thick) extracted from alloy D9 wrapper. The test fixture, experimental procedures and analysis methodology adopted are detailed elsewhere. (Karthik et al., 2013). The room temperature tensile properties of alloy D9 wrapper were estimated from ShP test data using tensile-shear punch correlations.

For mechanical property evaluation of SS316LN/SS304LN, pre-fabricated Charpy V notch (5 mm x 5mm x 55 mm) specimens of the two steels were irradiated in FBTR in addition to sub size tensile and disc specimens. A 450J capacity pendulum type instrumented impact testing machine calibrated as per ASTM E23 protocols was used for the impact tests. A mobile shielded setup consisting of a steel shielding wall fitted with Master Slave Manipulators (MSMs), viewing glass window and camera, and the test machine with customised specimen aligning tools was erected and used for remote impact testing.

Recent Developments

The use of Digital Image Correlation (DIC) for strain measurement during tensile testing in hot cells is being attempted. The challenges in implementing DIC for hot cell experiments and initial results of remote strain measurements in subsize tensile specimen of 12 mm gage length using DIC will be presented. Further miniaturization of tensile specimen has been carried out with a gage length of 3.0 mm and gage width of 1.5 mm carved out from a disc sample of 10 mm diameter and 0.5 mm thick. (Sham & Natesan, 2017) The ultra subsize (USS) tensile specimen (Figure 39) has been optimised using finite element analysis w.r.t the fillet radius and geometrical tolerances for gage width and thickness. The methodologies developed for implementing the USS specimen for testing irradiated components in hot cells will be presented.

Results

The combined data sets of tensile and shear punch tests of alloy D9 cladding/wrapper shows that for low irradiation temperatures around 673 K-723 K (lower portion of fuel column), there is significant increase in YS and UTS with a decrease in uniform elongation. The hardening effect decreases in both cladding and wrapper with recovery of uniform elongation at axial locations corresponding to upper portions of fuel column where the irradiation temperature increased beyond 723 K.

The trend curves of strength and uniform elongation of SS 316LN and SS 304LN with dpa reveal that SS 304 L(N) exhibited a higher rate of hardening with dpa and correspondingly a lower ductility compared to SS316 L(N) at all dpa (Figure.40). The Charpy-V notch energy (Cv) of SS316 LN showed no significant changes as displacement damage increased to 5.6 dpa, while that of SS 304 LN decreased by ~8% at 4.7 dpa and by ~20% at 5.6 dpa. Both tensile and Charpy test results indicate superior performance of SS 316LN compared to SS304 LN.

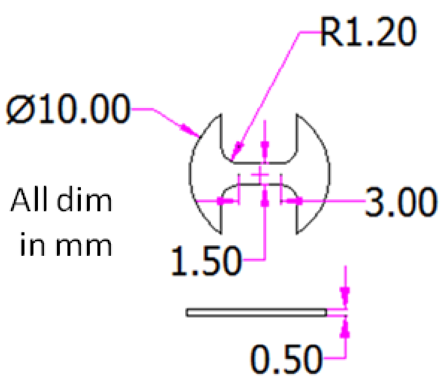

Figure 39: Ultra sub size tensile specimen

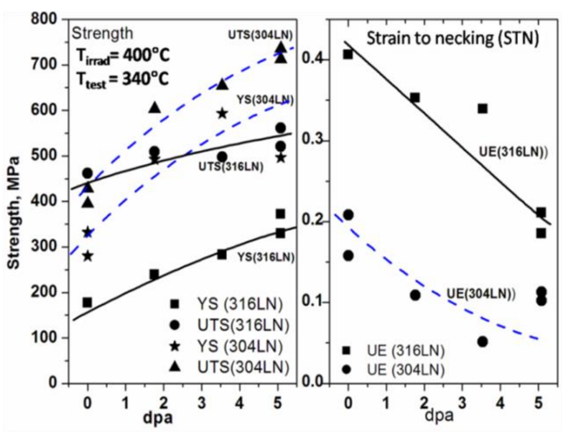

Figure 40: Strength and ductility of SS316/304 LN with dpa.

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Hotcell Examinations of In-core Inconel X-750 spacers removed from CANDU Reactors for Surveillance

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Life cycle management programs require periodic removal of core components from CANDU reactors. One of these components is the Inconel X-750 spacers that serve to prevent contact between the cold calandria tube and the hot pressure tube in the CANDU Heavy Water Reactor design (Figure 41). While serving only a support function, the Inconel X-750 spacer is subject to significant metallurgical transformations in core with the most significant being from the thermal neutron transmutation of nickel leading to high levels of alpha particle production and retention in the form of helium bubbles (up to 3 atomic percent He per the bulk). Surveillance spacers require hotcell examination and testing to ensure that they continue to meet their design requirement. These activities are integral to the overall Inconel X-750 spacer management program (Griffiths, 2013; Judge et al., 2014). The examination and testing activities carried out in the CNL hotcell facilities, equipment used and general findings are discussed.

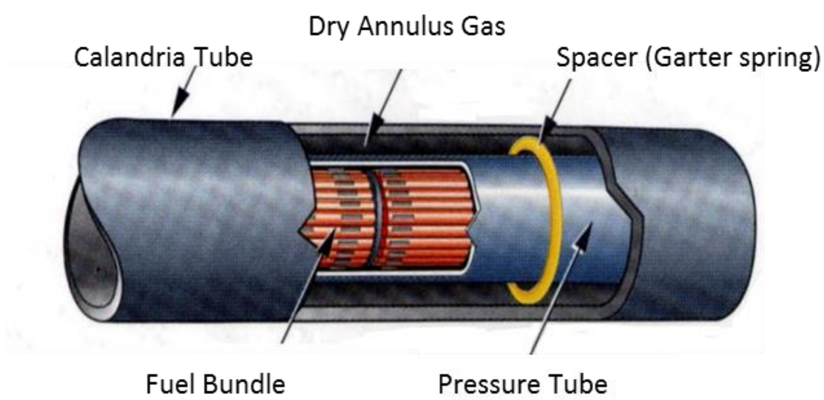

Figure 41: CANDU Fuel Channel schematic cut away showing the key components.

The spacers in a CANDU reactor fuel channel are of a helical spring design (Figure 42). The spacer service life may exceed 25 calendar years in core and hence spacers are highly activated at the time of removal. The most significant activation product is ^{60}Co with levels of up to 3 Ci ^{60}Co per gram material (roughly 30 - 50 curies of ^{60}Co per spacer). The gamma fields of a single spacer at 1 meter would be as high as 65 R/hour. A single coil from that spacer could have a field as high as 4000 R/hr near contact (1 cm). These radiation fields require that all handling takes place within a hotcell until they are sub-sampled to microscopic sizes.

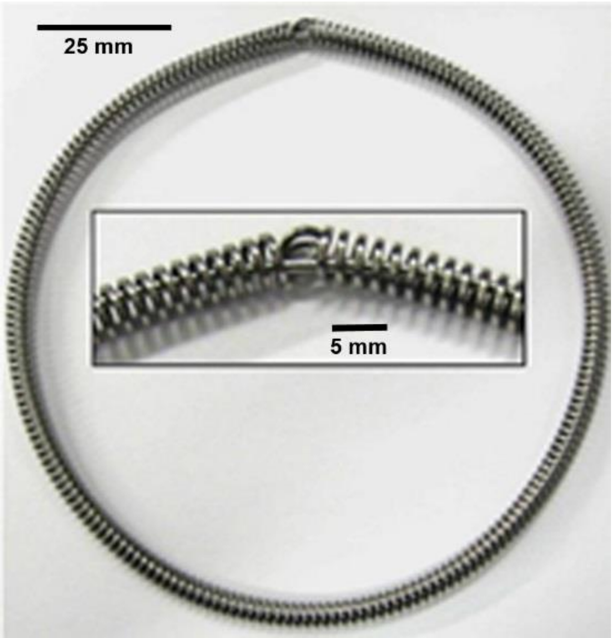

Figure 42: Inconel X-750 spacer and close-up showing attachment hooks and individual coils.

Examination of these spacers begins with an assignment or verification of their in-service location and orientation. The concentric pressure tube and calandria tube lie horizontal in a CANDU reactor so that the bottom portion of the spacer is in contact between the hot pressure tube and the cold calandria tube. As such, the bottom pinched portion experiences much cooler service temperatures than the portion that is only in contact with the hot pressure tube. Since examination and testing requires knowledge of the in-service conditions (temperature and neutron fluence), the orientation and axial location must be determined; information that can only be assumed from the blind removal process. The axial location is determined based on a measurement of the spacer diameter. Since the spacer is conformal to the pressure tube, its inner diameter can be matched to the known outer diameter of the pressure tube usually removed at the same time as the spacers. A calibrated tapered cylinder (Figure 43) enables a simple yet accurate diameter measurement. The pitch of the helical spring also varies around the spacer in ways that reveal the contact region at the bottom of the fuel channel. A rotational indexing stage provides a platform for the spacer and a high resolution video camera provides images for an accurate coil-by-coil measurement of the pitch. These images also provide for a closer examination of the spacer condition after removal from the core.

Once the provenance of the spacer material is established, sectioning for subsequent testing can proceed. Sections are taken for mechanical testing (crush testing, endurance/fatigue testing, micro and nano-indentation (Figure 44), and small-scale micro-mechanical testing), metallographic examinations (optical metallography, post mechanical testing fractography, and high resolution transmission electron microscopy), density determination via immersion techniques, and measurement of the bulk He content with hot vacuum extraction mass spectrometry.

The majority of the mechanical testing requires bulk material, and thus the work is performed exclusively within the hotcells using specially designed tooling. Nano-indentation, small-scale micromechanical testing, and transmission electron microscopy must be performed out-of-cell, and therefore techniques for sectioning and thinning material were developed to reduce the volume of the material down to masses of ~ 2 mg. Samples are then electro-polished to less than 100 microns thick. The fields are then reduced to the levels acceptable for out-of-cell handling. Specimens are first introduced to an active focused ion beam (FIB) for TEM preparation, and small-scale micromechanical testing using an in-situ PI88 Hysitron indenter (Howard et al., 2018). Once prepared in the FIB, the fields have immeasurable activity levels (too low for ordinary radiation protection equipment to measure) to be used for a suite of microscopic analyses. All this surveillance data contributes to a better understanding of the spacer condition and its fitness for continued service.

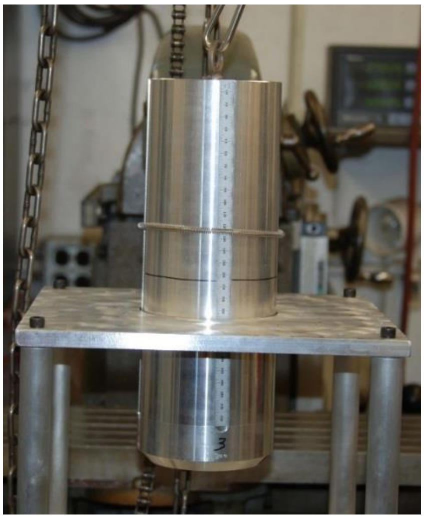

Figure 43: Calibrated tapered cylinder set up for measuring spacer diameter.

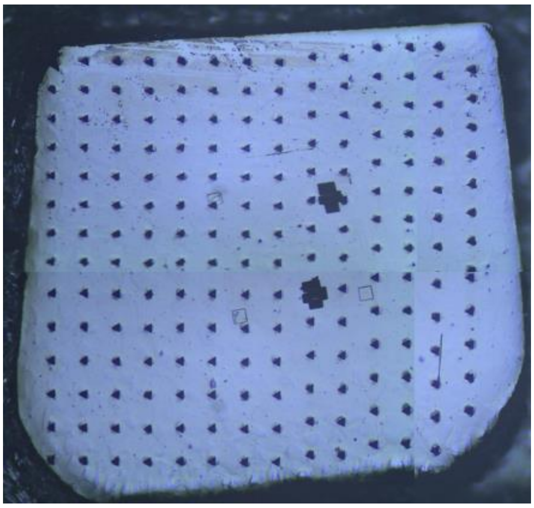

Figure 44: Example of the nano-indentation pattern used on the spacer wire cross section.

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Technical Session:

Hot Lab Equipment

Criticality evaluation of a transport cask of irradiated nuclear fuel samples according to the IAEA Regulations for the Safe Transport of Radioactive Material

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Safe transport of radioactive materials to Post Irradiation Examination (PIE) facilities is one of the most important issues. Criticality safety requirements for packages containing fissile material are provided by the International Atomic Energy Agency (IAEA) Regulations for Safe Transport of Radioactive Material (SSR-6). The general relationship for establishing the acceptance criteria in the criticality safety of a cask is:

$$k_{eff} + n\sigma \leq 1 - \Delta k_m - \Delta k_u \qquad (1)$$

where k_{eff} is the calculated multiplication factor; n is the number of standard deviations considered (2 and 3 are common values); σ is the standard deviation of the k_{eff} value obtained with Monte Carlo analysis; Δk_m is a required margin of sub-criticality; and Δk_u is an allowance for the calculational bias and uncertainty. The parameter of Δk_u is sum of two separate parts, Δk_c and Δk_b; the first is determined by benchmark calculations and the latter is related to the manufacturing tolerances as well as material composition uncertainties. Here, Δk_m=0.05 was used, which is recommended by most authors. The right-hand side of Eq. (1), which is the maximum Upper Subcritical Limit (USL), is given by:

$$USL = 1 - \Delta k_m - \Delta k_u = 0.95 - \Delta k_c - \Delta k_b \qquad (2)$$

In this study, criticality evaluation was carried out for a cask containing irradiated Tehran Research Reactor (TRR) mini plates and fuel rod samples of Bushehr Nuclear Power Plant (BNPP) with shorter active length (about 56 cm height) based on SSR-6 requirements. It was assumed that five mini plates and six fuel rod samples were irradiated in the research reactor. The fuel samples within the capsule are assumed to be irradiated for enough period of time until they reached the maximum burnup. Then, the fuel samples and the capsule are allowed to be cooled inside the adjacent reactor pool before transportation to a PIE facility by the cask.

To determine Δk_c and verify that the computer code accurately predicts the k_{eff}, a set of calculational benchmark problems were employed in this work for bias and uncertainty of the k_{eff}. For Δk_b, a set of calculations were carried out to determine uncertainty of tolerances associated with diameter and thicknesses as well as the material specifications.

The effective multiplication factor of the single cask and the cask array under Normal Condition of Transport (NCT) and Hypothetical Accident Conditions (HAC) were calculated using MCNP code. Considering bias and uncertainties, the USL which determines the minimum effective multiplication factor for the sub-criticality of the system was specified. Regarding the results, the criticality margin of the single and the array under NCT and HAC was far from the USL and therefore the cask can be transported with the assured safe margin.

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The R83 type B(U) transport package for used LEU fuel: a versatile package – HOTLAB 2018

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The International Atomic Energy Agency (IAEA) is very active in international efforts to minimize and eventually eliminate the use of Highly Enriched Uranium (HEU). A significant portion of this support involves the conversion of research reactors (RR) from HEU to low enriched uranium (LEU) fuel and radioisotope production targets[1].

Design of a new package for NRG

To cover the need of Nuclear Research and Consultancy Group (NRG)[1] to transport used LEU fuel from the High Flux Reactor (HFR) in Petten and Hoger Onderwijs Reactor (HOR)[2] in Delft to the intermediate storage facility HABOG at COVRA, Nieuwdorp, as well as waste from the Mo-99 production from its facilities in Petten, ROBATEL Industries is licensing a new type B(U) fissile package, the R83.

One of the challenges to design this package has been to comply with the latest IAEA regulations for the safe transport of radioactive waste while allowing a simple handling at both the HOR and the HFR where the package is loaded underwater, as well as at the HABOG facility in COVRA where the basket is unloaded and placed as a whole in a welded canister for final storage.

Involving of the ROBATELs' innovations in the design process

The final design of this package is based on the multilayer approach stainless steel/lead/neutron absorber that has been the hallmark of ROBATEL Industries' type B(U) packages for the last decades and include the classic ROBATEL Industries PNT7™ concrete as the neutron absorber but also the new FENOSOL™ foam for shock absorbers. To the opposite of wood traditionally used as a shock absorber, this chemical-resistant, moisture-resistant foam is isotropic, avoiding sensitivity of shock absorption with the drop orientation. The drop tests performed with the R83 scale model provide once again the demonstration of the mechanical capabilities of this foam. Moreover, FENOSOL™'s very low heat conductivity and fire resistance make of it an excellent heat and fire protection.

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[1] Nuclear Research and Consultancy Group (NRG) is a Dutch institute that performs nuclear research for the government and private companies. It is the most important producer of radionuclides in Europe and maintains and operates the Petten nuclear reactor. NRG is an internationally operating nuclear service provider.

[2] HOR is a pool-type reseach reactor situated at the Interfaculty Reactor Institute of the Delft University of Technology and has been operated since 1963. It is the one and only university type research facility of its type in the Netherlands.

The main R83 package characteristics and performances are summarized below by focusing on the main safety issues according to regulations.

Table 4: Dimensions and mass of the R83 package model.

Dimensions	Values
Overall Height	≈ 2 140 mm
Outside diameter with shock absorbers	≈ 2 050 mm
Cavity height	950 mm
Cavity diameter	743 mm
Loaded package mass	≈ 16 200 kg

Table 5: Performances of the R83 package model.

Properties	Values
Content maximum mass	1 000 kg
Shielding materials	Refine pure lead Compound PNT7™ Stainless steel Boronated stainless steel
Thermal protection	Compound PNT7™ FENOSOL™ phenolic foam
Containment	EPDM double O-ring system
Closure system	36 x M24 bolts

Progress of the project

In January, 2016 ROBATEL Industries started the project of a new cask R83, Figure 45. The package design safety file was submitted in the end of 2017 to The Authority for Nuclear Safety and Radiation Protection (ANVS) of Netherlands. This year we have started manufacturing of two R83 casks. By the end of 2018 we plan to obtain its Type B(U) fissile license. In 2019 this package needs to be available to guarantee the transportation of LEU material.

Figure 45: Progress of R83 project.

The schematic image of R83 is given on the Figure 46.

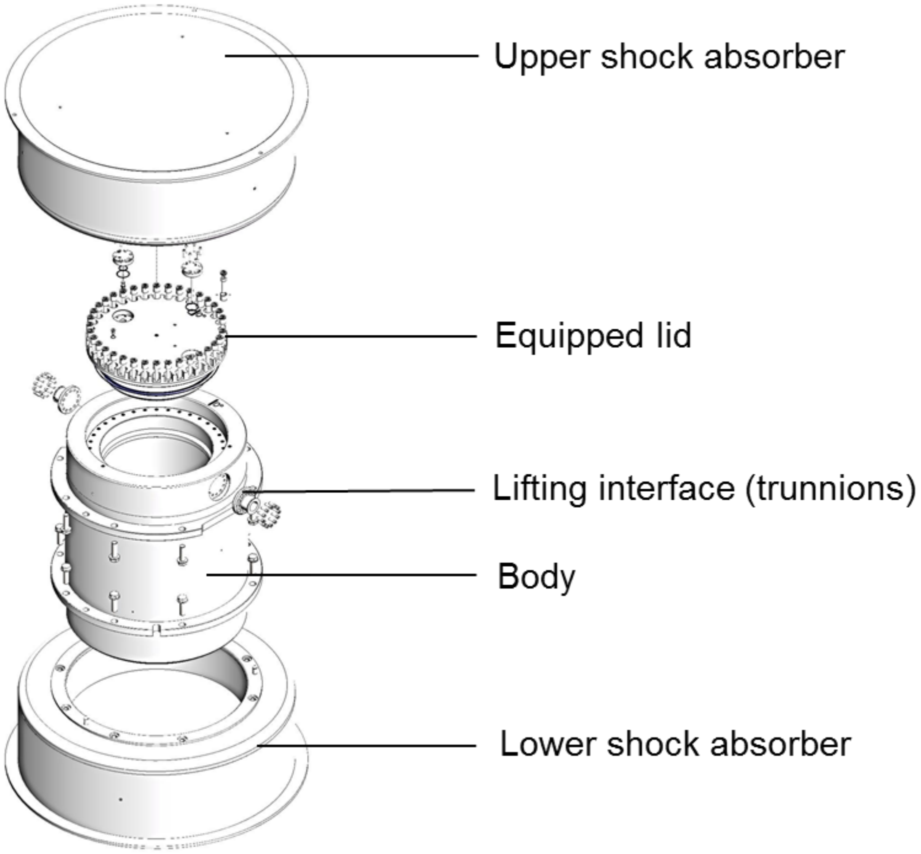

Figure 46: Packaging R83.

Conclusion

ROBATEL has proven that we can design and deliver solutions that fit the best technical and economical requirements for type B packaging. This new cask will replace the current CASTOR MTR-2 cask in view of the conversion of fuel from HEU to LEU. ROBATEL will supply licensing, manufacturing, operation and maintenance documentation packages for this project. Thanks to ROBATELs' expertise and experience in cask licensing, this full cask project will be delivered within 3 years after contract start.

Reference

"IAEA Support of Research Reactor HEU to LEU Fuel Conversion", Research Reactor Section from www.iaea.org

Compact Tension Sample Preparation Out of Candu Pressure Tube Using the Numerical Controlled Milling Machine

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Compact tension (CT) samples are used extensively in the area of fracture mechanics and corrosion testing, in order to establish fracture toughness values for a material. This paper presents the method of preparation of CANDU pressure tube samples for mechanical tests. For samples preparation a DM 1007 Dyna numerically controlled milling machine is used.

The CT sample must have a notch for an initiation of delayed hydride cracking (DHC) along the longitudinal direction. This notch plays a key role in the determination of the threshold stress intensity factor (KIH).

In order to perform the machining of such a CT sample out of a CANDU pressure tube fragment, some devices were manufactured: the device to fasten the pressure tube during machining, the device to fasten the sample during cutting both ends of it and the device to fasten the sample during notch machining. The pressure tube is fastened to the milling machine table, the ensemble of both being allowed to move in the XY plane, whereas the milling tool is moving independently along the Z axis. The programming of sample machining takes into account the size measurement of the pressure tube and the movement coordinates of the milling machine table. A „Point Finder" is used for coordinates measurement of the pressure tube and a similar tool TS27R, fixed on the milling table, is used for dimensional measurements of the milling tools that will be used.

The machining of the CT sample needs six operations for which six different tools are used. All tools are mounted on the tool changer prior to begin the machining and are changed automatically as needed.

This milling machine (Figure 47) will be installed into a heavy concrete hot cell and will be used to prepare CT samples out of irradiated CANDU pressure tube fragments.

Basic parameters of the milling machine:

- X, Y, Z axis travel: 250 x 175 x 250 mm;
- Table size: 450 x 180 mm;
- Maximum table load: 60 kg;
- Maximum X, Y, Z axis travel speed: 5/5/5 m/min.;
- Cutting feed rate: 0 ÷ 5000 mm/min.;
- Spindle rotate speed: 60 ÷ 10000 rpm;
- Spindle nose to table distance: 95 ÷ 350 mm;
- Automatic tool changer, number of tools: 6;
- Tool holder: type BT-30;
- Maximal tool length: 150 mm;
- Simultaneous X, Y, Z axis control;
- Positioning precision: 0.001 mm;
- Feed rate range: 0.001 mm;
- Software SurfCAM 3 Axis SE (DM1007, EM3116(A) & DM2800; Dyna 4M)

Figure 47: Dyna DM 1007

CT sample preparation sequences

The final CT sample should meet the dimensional accuracy and surface quality required by the mechanical tests standards (ASTM E8M; ASTM E399). For all operations needed to obtaining the CT sample (achieving the upper surface, the lower surface and side surfaces, the adjustment of the specimen surfaces, making the holes, the notch preparation) programs edited in „G" codes have been developed in order to control the milling machine (Dyna 4M)

The sequence of operations performed in order to manufacture the CT sample is as follows:

- The lengths and the diameters of all mills which are to be used (Figure 48) and the length of the „Point Finder" were measured using a device TS27R (TS27R Tool setting probe).
- The pressure tube fragment is fastened horizontally on the milling machine table, using the „Fastening device", so that its axis must be parallel to the worktable and with the X direction. One measures the coordinates of the pressure tube fragment (Figure 49), using the „Point Finder", to determine the height and centre of the tube fragment in order to extract the CT sample (Point Finder Touch Sensor. [6]
- Using the cylindrical mills the pressure tube is made, the upper, lower and side surfaces of the CT sample are achieved. The two holes with 4.25 mm diameter of the CT specimen are made using a 3 mm diameter mill (Figure 50).
- After separation of the CT sample from the pressure tube fragment, it is fixed vertically in „The adjustment device" (Figure 51). The CT sample is fixed so that the plane of the holes axes is parallel with the worktable of the milling machine and the axes of these holes are parallel with the Y coordinate. The sample surfaces are adjusted using a 10 mm diameter cylindrical mill.

Figure 48: Tools measurement using the TS27R instrument

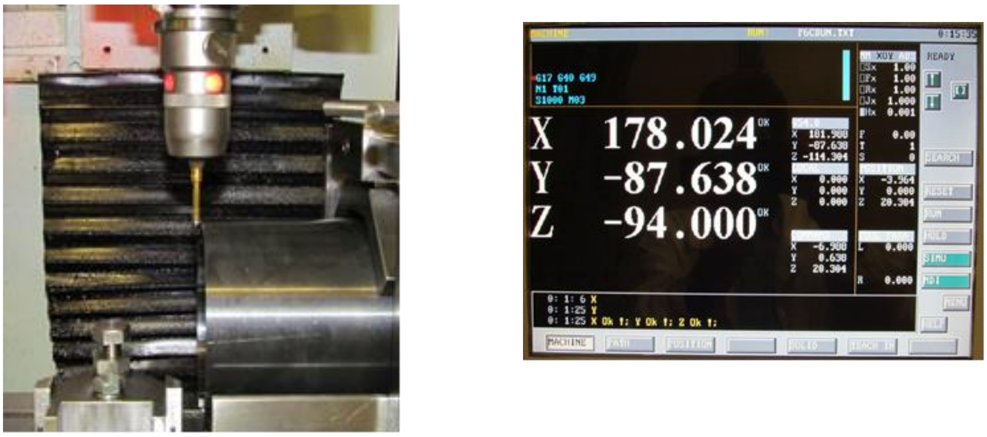

Figure 49: Measurement of the pressure tube coordinates using the „Point Finder”

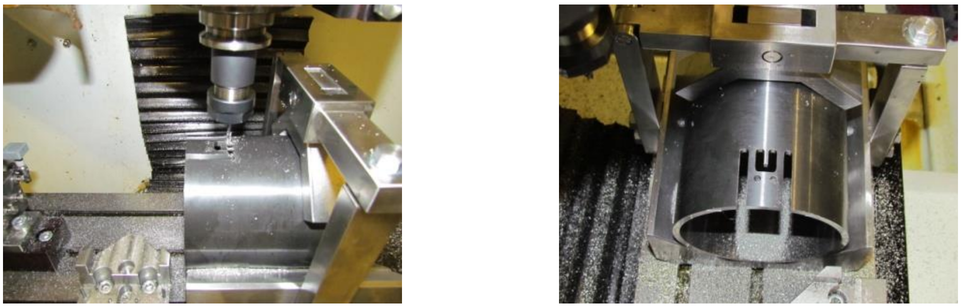

Figure 50: Separation of the CT sample from the pressure tube fragment

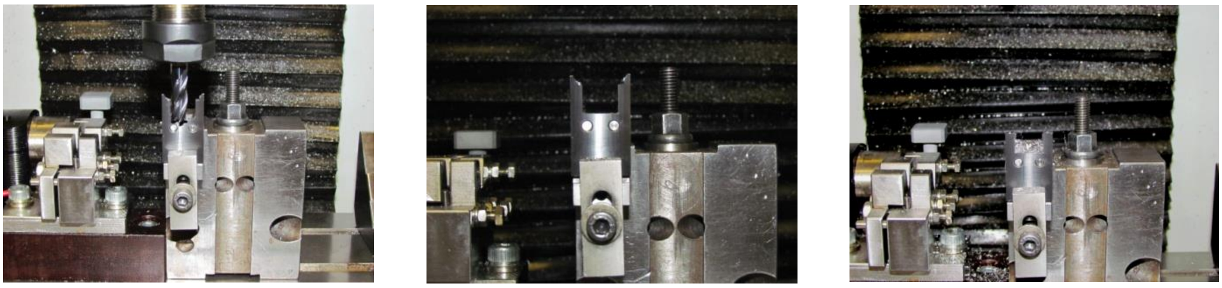

Figure 51: Adjustment of the sample surfaces

- In order to manufacture the notch, the CT sample is placed horizontally in „The notch preparation device”. In the first step, the channel of the central crack is made, using a disk mill with a diameter of 80 mm and the thickness of 1.5 mm. The second step consists in achieving the profile with the angle of 60° at the tip of the crack, using a profiled disk mill (Figure 52).

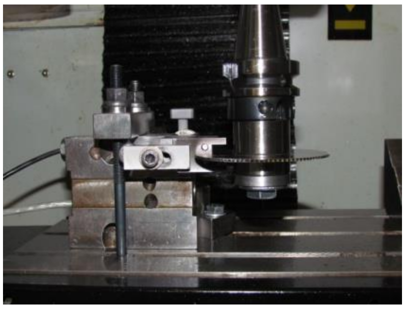

Figure 52: Notch preparation

Results

After all this mechanical procedures, the CT sample out of a CANDU pressure tube fragment is ready (Figure 53).

Taking into account the recommendations of ASTM 399 (Figure 54), the pressure tube dimensions and the specific work conditions, we obtained a specimen with the following specifications: specimen width = 20.4 mm, crack depth = 11 mm, specimen thickness = 4.1 mm, angle between the crack flanks = 60°, total length = 30 mm, diameter of the mounting holes = 4.25 mm, distance between mounting holes = 9.36 mm, the radius at the tip of the crack = 0.08 mm.

Figure 53: CT sample manufactured out of non-irradiated CANDU pressure tube

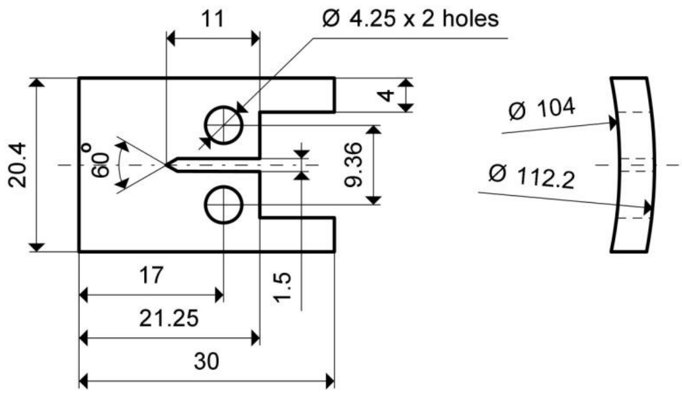

Figure 54: CT sample ASTM 399

The CT samples were obtained out of non-irradiated CANDU pressure tube fragment (made out of Zirconium-Niobium 2.5 %). For this purpose, mills and cutting conditions fitted to the Zirconium-Niobium 2.5 % alloy, precision tools for dimensional measurements („Device for measuring tools", „Point finder"), dedicated devices („The fastening device", „The adjustment device" and „The notch preparation device") were used. The CT samples manufactured in this way correspond to the technical specifications of the standard ASTM E 8M and ASTM E 399.

The milling machine will be installed into a heavy concrete hot cell and will be used to prepare CT samples out of irradiated CANDU pressure tube fragments.

References

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ASTM E 399. Standard Test Method for Linear-Elastic Plane-Strain Fracture Toughness KIc of Metallic Materials. 2012. ASTM E399-12e3.

Dyna 4M – Machine control & Programming Manual

DM1007, EM3116(A) & DM2800 – Operation & Reference manual

Point Finder touch sensor. Nisshin Sangyo CO.Ltd.

TS27R tool setting probe. Renishaw, 2006. H-2000-2171-05-A.

Radioactive Materialography Preparation Systems

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The Idaho National Laboratory (INL) performs materialography on irradiated materials and irradiated nuclear fuel at the Materials and Fuels Complex (MFC) in the Hot Fuel Examination Facility (HFEF)- an argon atmosphere hot cell. Because of the dose rate of these materials, this requires remote systems in a shielded hot cell to prepare samples with the desired surface finish for examination. Remote materialographic sample preparation systems have been developed to replace the existing, aging remote sample preparation systems in HFEF. These systems allow sample preparation of irradiated materials in an inert atmosphere by removing surface deformation with grinding and polishing the material surfaces. This allows researchers to analyze the deformation free microstructure with remote light microscopy, microhardness testing, and other analytical techniques available at MFC. The system consists of two benchtop semi-automatic grinding and polishing machines that are commonly used at materialographic laboratories. The machines have undergone modification by the INL for maximum efficiency for remote use and remote maintenance in HFEF.

Radioactive Materialography

Remote Sample Preparation. Following non-destructive examination techniques in HFEF, materials such as irradiated hardware or irradiated nuclear fuel are destructively examined at microscopic levels for microstructural changes during irradiation. This practice, known as radioactive materialography, requires sample preparation of irradiated materials and is achieved by performing the preparation techniques inside of an inert shielded hot cell. Sample preparation involves the steps of sectioning, mounting, grinding, cleaning, and polishing. Equipment for these steps must be modified for reliability in the hot cell environment and remote use, while maintaining the performance and efficiency of the equipment.

Radioactive Materialographic Grinding and Polishing Systems. INL has procured two Struers Tegra Semi-automatic grinding and polishing machines to replace existing semi-automatic grinding polishing machines that are aging. The Struers Tegra system consists of a base that rotates the platen with the abrasive and a head that provides sample rotation and force. The machines were modified to maximize efficiency for remote operation, prolong system life in a radioactive environment, and for integration with current facility systems in the HFEF hot cell. The system modifications included replacing components that can be degraded by radiation, modifying pneumatics, and adding features to allow operation and maintenance with remote manipulators.

Modification for Remote Operation. Features have been added to the Struers Tegra system to maximize efficiency with remote hot cell use and maintenance. Manipulator grips were added to aid with remote operation. Electrical cables were replaced with radiation resistant cabling with Amphenol® connections that are manipulator friendly. Engineered lifting fixtures to integrate with remote lifting and handling equipment were added to both the base and the head to allow for in cell installation and maintenance of both the head and base. The INL designed a water dosing system for rough grinding that filters and recirculates the water to minimize liquid waste.

The Struers Tegra system utilizes pneumatics on the head to provide a specified force on mounted samples. The pneumatic system was modified to integrate with in-cell compressed argon supply manifolds. This modification included separating the in-cell pneumatics from the control system to mitigate exhausting contaminated argon from the machines into the control system on the operating corridor.

Some components in the Struers Tegra system are susceptible to radiation damage. Components such as electronics and plastic components need to be replaced or relocated in order to ensure system reliability. Electronics were relocated to the control system outside of the hot cell. Plastic components were replaced with compatible metal components.

Figure 54: Remotized Struers Tegra Semi-automatic grinding and polishing machine.

Conclusion

INL has successfully modified two Struers Tegra Semi-automatic grinding and polishing machines for remote operation in HFEF. The system has undergone an out-of-cell qualification process and is now ready for installation in HFEF.

Fit For Purpose Design for Remote Operations - Handling the Hot Potatoes

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Remote handling must be employed in a wide range of situations where more convenient methods of operation are unsuitable due to the need for biological shielding. Working remotely can readily be achieved by various means, however they typically require concessions to productivity, versatility and user operability. Solutions are available which mitigate these drawbacks but they can be costly and complicated to implement, particularly when considering existing facilities or limited/single use campaigns. How do you balance these conflicts to find a fit for purpose solution?

Aquila Nuclear Engineering are an engineering design company providing bespoke solutions for the nuclear industry specialising in remote handling, shielded facilities, containment and radioactive material packaging and transport. In delivering these projects, the engineering team frequently develop remotely operated equipment tailored to the needs of the user and limitations of the facility.

This paper will consider three case studies, looking at what fit for purpose design means for very different applications.

Case Studies

PIE Cave Re-Configuration – National Nuclear Laboratory, Sellafield. The National Nuclear Laboratory (NNL) Windscale Laboratory at Sellafield offers shielded cell facilities to clients for non-destructive and destructive examination of reactor fuel and irradiated materials. The high activity caves are fitted with remote handling equipment and services and may be decontaminated, emptied and refitted as required to suit the requirements of the client.

In 2018, Aquila delivered a number of size reduction machines concerning the refit of one of these caves to support a research project analysing fuel pins. Once commissioned, Master Slave Manipulators (MSM) will be used exclusively to operate and maintain the equipment in a labour intensive campaign expected to last 10 years.

A modular approach was used with standardised components and features where possible minimising installation, operation and maintenance effort. Equipment had to be robust, efficient to use and minimise day to day operator effort with commercial off the shelf (COTS) items used for reliability but adapted for remote operation and maintenance. All interfaces had to be MSM compatible and especially simple for those in daily use. While machine maintenance must also be possible, the difficulty of the procedure could be considered against the likelihood or frequency of occurrence.

Active Waste Vault Retrieval and Export – Magnox, Berkeley. Magnox Berkeley nuclear power plant generated power from 1962 through to 1989 and is leading the rest of the fleet in terms of decommissioning. The reactors have been sealed and are now in long term ‘safestore’ until 2074 but a number of challenges regarding Intermediate Level Waste (ILW) remain on site.

The Active Waste Vaults (AWV) are a series of underground concrete vaults that are currently storing a variety of waste accumulated during the lifetime of the two reactors and the adjoining research laboratory. Emptying one of these required the creation of a temporary hot cell to safely package retrieved waste into shielded flasks for interim storage. Due to the limited quantity of waste and therefore short lifespan (the equipment was ultimately used for 9 months to fill 11 containers), only occasional operator intervention was needed and therefore the purpose designed equipment needed to be simple but safe.

The hotcell is of basic construction in cost effective materials with shielding only provided where required and a fit and finish appropriate to the limited usage. Simple mechanical actions are used to import, open and position the 9,000kg shielded flask in preparation for receiving a waste basket, primarily with only human effort. The system relied on position stops, alignment markers and operator feedback to allow blind operation, although cameras were provided for additional confirmation. Operation under robust procedural controls and a reduction in electrical parts reduced the complexity of the control system resulting in a lightweight Electrical, Controls and Instrumentation (EC&I) package.

Legacy Facility Decontamination and Decommissioning – GE Healthcare, Amersham. The Senior Caves facility in Amersham was constructed in 1957 on what was then a UKAEA site manufacturing radioactive materials for peacetime uses in medicine, scientific research and industry. Originally intended to purify fission products in support of site processes, an accident in 1962 resulted in widespread Cs-137 contamination preventing purification. The plant was used to produce Cs-137 sources until 1964 and in 1966 an attempt to decommission one of the cells resulted in another leak and further spread of contamination.

Returning to the facility in 2010 to begin decontamination and decommissioning, GE Healthcare started with no lights, service or remote handling equipment and solid shielding in front of leaking ZnBr shield windows meaning no internal view. Over the course of the last 6 years, Aquila have supported GE Healthcare with a variety of novel remotely operated equipment, reacting to the unique problems as they have been uncovered.

After initial work to establish a safe route into and out of the complex, work began on decontamination. Tools range from basic components like a 3D printed MSM tool holder allowing operators to quickly make custom tools, to miniaturised grit blasting and vacuuming equipment allowing removal of surface contamination. Due to the single use nature of much of the equipment, trials and close collaboration with the plant operators was vital to minimise delivery time and secondary waste.

Implementation of An Innovative Nuclearized SEM in CEA-Atalante Facility

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The CEA R&D on U-PuO_2 fuels was performed in LEFCA facility in CEA-Cadarache (France) until 2015. LEFCA R&D activities are transferred to Atalante facility in CEA-Marcoule. Within the frame of the TARRA project, the design of new scientific instruments is in progress. Among these equipment, the development of a new nuclearized scanning electron microscope (SEM) dedicated to sintered nuclear samples has been acted. This new SEM is equipped with an energy dispersive spectroscopy (EDS) and electron backscatter diffraction detectors (EBSD).

Choice and study

Technical choices. The fuel characterizations laboratory (LCC) has a large experience on servicing processes of nuclearized SEM connected to a glove box (HOTLAB2016). The final choice of set up is a TESCAN model which is a mix between MIRA3-GM and MIRA3-AMU SEM types. The MIRA3-AMU has a huge and heavy chamber designed for large samples. The entire system is equipped with suspensions based on an active isolation system. The MIRA 3-GM is a classic SEM MIRA3 with the biggest chamber proposed in TESCAN standard. The nuclearization strategy was to adapt the AMU-frame to the GM-chamber connected to a glove box. Active Isolation systems were ensured the connection between the fame and the modified device. By this design (with some optimisations on the GM chamber), it is possible to get rid of SEM disconnection and pneumatic suspensions to maintain it. The absence of cooling system makes its implementation in the ATALANTE-L26 laboratory easier.

This SEM-FEG is equipped with SE Everhart-Thornley, YAG-BSE, SE InBeam and also with EDAX EBSD and EDS detectors. The EDAX technology was chosen for two different reasons. Firstly, the EBSD detector has a window on the SEM chamber that offers the capability to maintain the camera without any containment rupture (no need to nuclearize it). Secondly, ceramic windows (Si_3N_4) used on EDS detector seems to be promising material on radiation protection (it seems stronger than those in polymer). A shielded shutter has also been added to reduce the impact of samples radioactivity.

Study. The Project started in November 2016 to create the TESCAN MIRA3-ATA (ATA for ATAlante). Safety and security rules of Atalante facility have substantially influenced the MIRA3-ATA design (valves, check valves, filters, connectors, earthquake resistance …). A 3D view of the final model is presented in the figure 55.

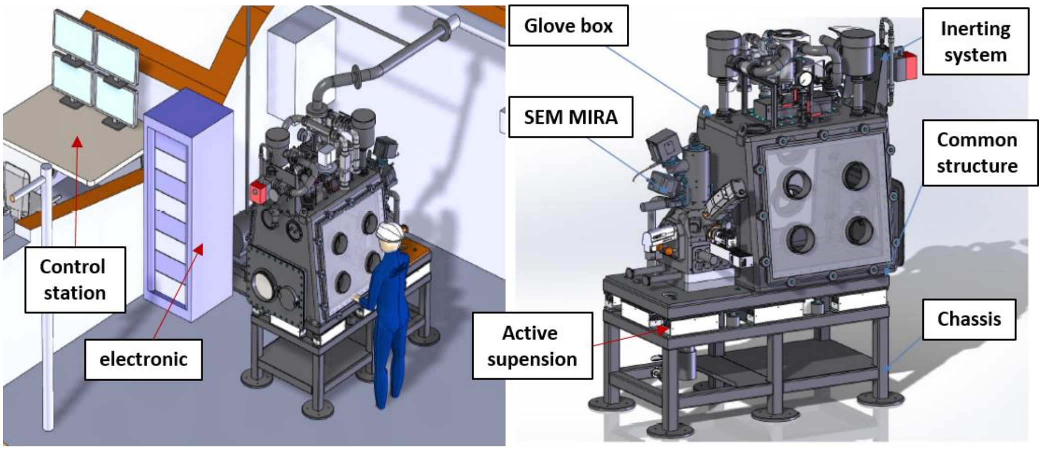

Figure 55: 3D representations of the MIRA3-ATA

Realisation and implementation

Modifications of the SEM chamber were performed by TESCAN. The glove box, ventilation system and the frame have been manufactured by AEMCO Company. NEWTEC SCIENTIFIC has taken over the major modifications of the device.

Some key step of this project will be presented in the conference. To illustrate its implementation, the pictures of the figure 56 show the project progress at the end of May 2018.

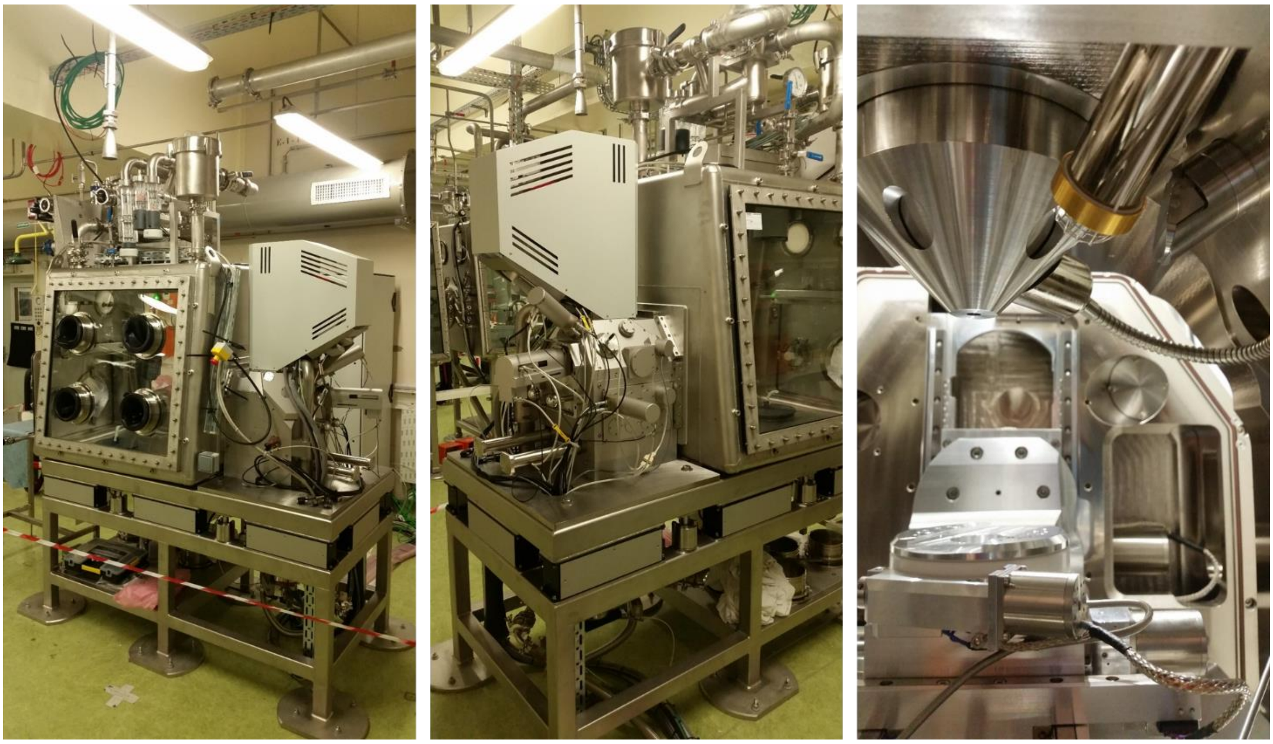

Figure 56: Pictures of the SEM implementation in L26-Atalante in May 2018

Results

A final part of the conference will present some results obtained with this apparatus after its commisionning.

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Shielded Electron Microprobe and some of its main applications in Hotlabs

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Keywords**.** EPMA, shielded electron microprobe, quantitative analysis, nuclear fuel, fission products.

Thanks to its precision, its reproducibility and its stability, Electron Microprobe is a well suited technique for accurately analyzing nearly all chemical elements at concentration levels down to few 10’s ppm with a spatial resolution of about 1 µm, which is relevant to microstructures in a wide variety of materials and mineral specimens. For irradiated samples, EPMA is also one of the technique of choices and can support nuclear fuel development, control, metallurgical or glass analysis. It reveals fine compositional details and the distribution of main and trace elements across the surface of the sample.

CAMECA leader in scientific instruments has been manufacturing Electron Microprobe (EPMA) since 1958 and will present its latest CAMECA shielded EPMA model, SKAPHIA released in 2016 for hotlab facilities. One example of the LECA/STAR shielded EPMA in hotlab handling nuclear fuel will be highlighted. In this example, the microstructure and chemical evolution of fuel pellets submitted to two different nuclear severe accident scenarii are shown. EPMA helped to highlight the impact of the atmosphere (i.e the oxygen potential) on the behavior of the fuel and fission products during these tests. These studies are of particular importance to evaluate and better predict the consequences of such an accident in term of contamination.

Three-Dimensional Subnanometer Compositional Analysis of Radiation Damaged Materials with Atom Probe Tomography – Technology and Practical Considerations

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Over the past 15 years, the number of peer reviewed publications referencing the use of atom probe tomography (APT) has grown by nearly a factor of five (Larson et al., 2013). The performance of the typical atom probe today is orders of magnitude better than the early systems in terms of data collection rate, field of view, mass resolving power, reliability and software. The availability of easy to use laser pulsed systems coupled with the changes in performance and availability of FIB-SEM sample preparation has dramatically opened the array of applications that can be analyzed.

Modern local electrode atom probe systems (LEAP®), can identify subnanometer, spatially resolved, 3D information of any element or isotope with up to 80% detection efficiency at millions of atoms per minute with a field of view (FOV) that can exceed 250 nm and an achievable sensitivity limit in the low parts per million range. The imaging, time-of-flight mass spectrometer, of the LEAP 5000™ from CAMECA® includes improvements in FOV and uniformity, multi-hit sensitivity, faster data acquisition, especially in voltage-pulsed mode, and a new control software platform that improves yield and the speed at which a user can optimize run conditions. Additionally, a new atom probe platform, the EIKOS™ (Larson et al., 2018), offers dramatic improvements in the simplicity of design and operation, including an over 50% reduction in the cost of ownership while still providing high-performance voltage/laser pulsed APT data.

In addition to these instruments performance developments, it is now possible to complete specimen transfer at ultra-high vacuum (UHV) levels with the ability to keep the specimen at cryogenic temperatures to enable new applications such as water-based systems, fast oxidizers like lithium, and the ability to analyze the distribution of hydrogen/deuterium in materials. Vacuum transfer systems can also be used to safely move radioactive specimens. Also, new sample preparation techniques and software developed in cooperation with EDAX® to combine transmission electron backscatter diffraction (t-EBSD) and APT to provide synergistic information (see Figure 57.) and improve APT data reconstruction (Rice et al., 2016).

Although the most common APT applications remain metallurgical in nature, these hardware and software improvements continue to open new applications by enabling sufficiently high yield and data quality to provide novel information. Recent new applications include, failure analysis of FinFET devices, 3D printed alloys, high entropy materials, rapid oxidizers, zeolites, cryogenically preserved biomolecules, H/D distribution in materials as well as traditional analysis of metals, metal oxides and protective coatings. Examples of APT in the analysis of nuclear structural materials and fuel will be reviewed.

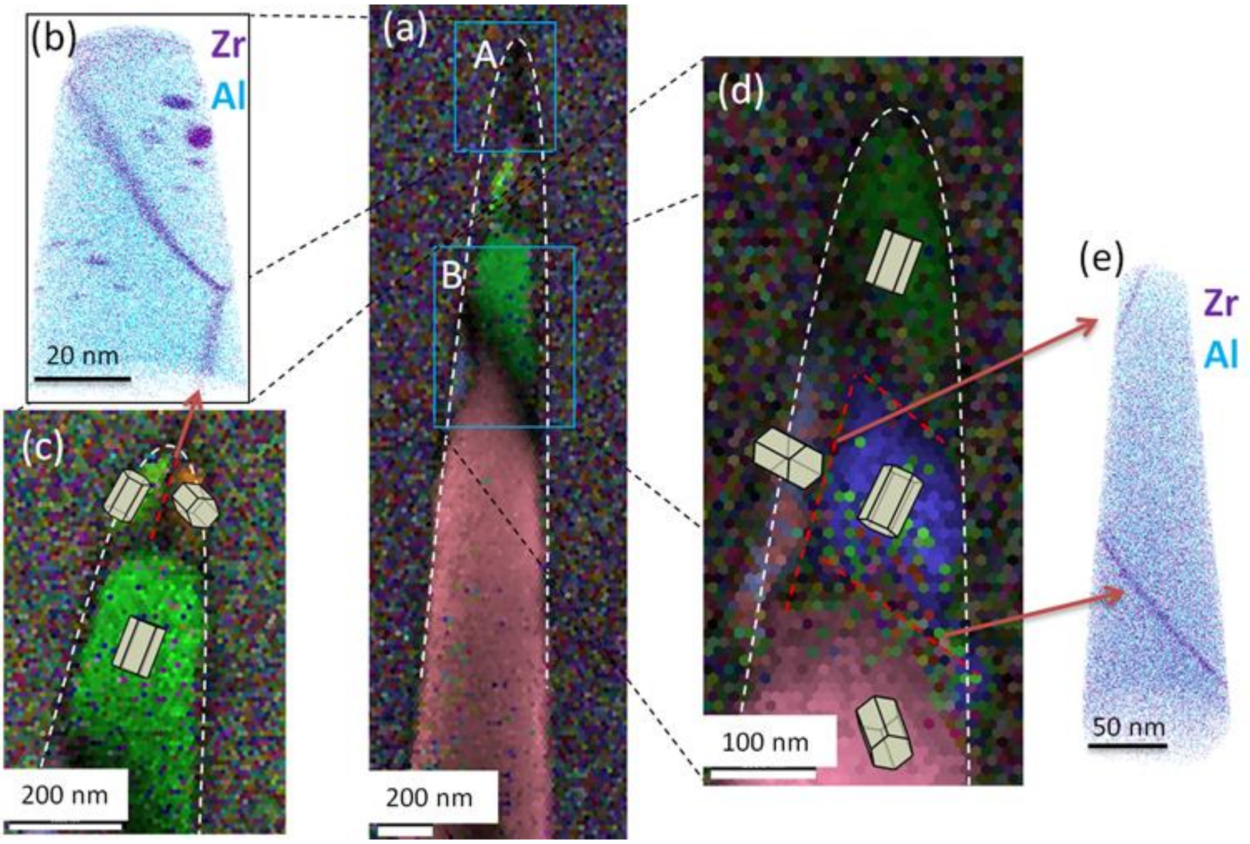

Figure 57: t-EBSD maps of alumina grains and correlative APT results from an alumina scale contraining zirconium.

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Technical Session:

Management of Hot Labs

Overview and Status of the US Nuclear Science User Facilities (NSUF)

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The Nuclear Science User Facilities (NSUF) was established in 2007 under the US Department of Energy Office of Nuclear Energy (DOE-NE) in order to better enable access of the nuclear energy research community to the unique and specialized nuclear energy capabilities operating across the US and developed over many decades. Although initially intended as a single institution government funded user facility at the Idaho National Laboratory (INL), the NSUF has grown and expanded to include facilities at an additional seven National Laboratories, eleven universities (plus four universities at the Center for Advanced Energy Studies in Idaho), one industry site, and one international affiliate. Access to the NSUF capabilities is gained through competitively reviewed proposal processes. For larger projects that can include the complete range of activities from full reactor irradiations through post irradiation examination (PIE) studies and can last many years, the annual Consolidated Innovative Nuclear Research (CINR) solicitation is appropriate. For more “standard” user facility access, i.e. small projects with limited instrumentation time, the thrice yearly Rapid Turnaround Experiments solicitation is appropriate. Information on all aspects of the NSUF can be found at the website (nsuf.inl.gov).

Capabilities of the NSUF

Figure 58 illustrates the breath of capabilities offered through the NSUF and the institutions with which these capabilities are associated. The NSUF offers a full suite of neutron reactor irradiation capabilities covering a range of powers and fluxes. Gamma and ion irradiations are also possible, providing also coupled *in-situ* TEM observation capabilities with the latter. Critically important hot cell and shielded cell capabilities enable not only “classic” PIE studies but also advanced materials science studies on highly radioactive nuclear fuels and materials using state-of-the-art instrumentation. As to be expected, most of these are located at National Laboratories. Driven in large part by the application of focused ion beam (FIB) sample preparation techniques to radioactive materials, whereby very small samples can be prepared with concomitant reduction in radioactivity, even more advanced instrumentation has become available in laboratories able to handle materials with only low activity. Of particular note here are micro- and nano-mechanical property measurements, advanced microscopy, and high resolution chemical analysis (atom probe tomography). Specialized and intense beamlines available at other US national or institutional user facilities can be accessed through the NSUF and include X-ray, neutron, and positron spectroscopic and scattering analyses. Finally, in addition to all the experimental capabilities available for the study of nuclear fuels and materials, the NSUF offers high performance computing access at INL and promotes the coupling of experiment with modelling and simulation.

The NSUF continues to maintain and expand the Nuclear Energy Infrastructure Database (NEID) and the Nuclear Fuels and Materials Library (NFML), two initiatives undertaken to more effectively and efficiently utilize the nuclear energy assets of the US. Both databases are web-based searchable tools with the former populated with a variety of linked information on instrumentation, facilities, and institutions and the latter populated with specimens and samples from irradiation tests performed over the decades. The NEID can assist users in formulating projects and enables DOE-NE to better manage their capabilities and future investments. The NFML is intended to reduce costs, avoid redundancy in irradiation tests, and secure irradiated fuels and materials for future studies as new ideas and instrumentation become available. Significant effort is being exerted to identify caches of materials, check and document the provenance of each, and position the materials in easy to access locations. The NSUF has recently begun work on the CoMET, the Combined Materials Experiment Toolkit, that will offer users an integrated platform incorporating the NEID, NFML, a subject matter expert database, and a projects database. CoMET will likewise be offered as a web-based searchable utility.

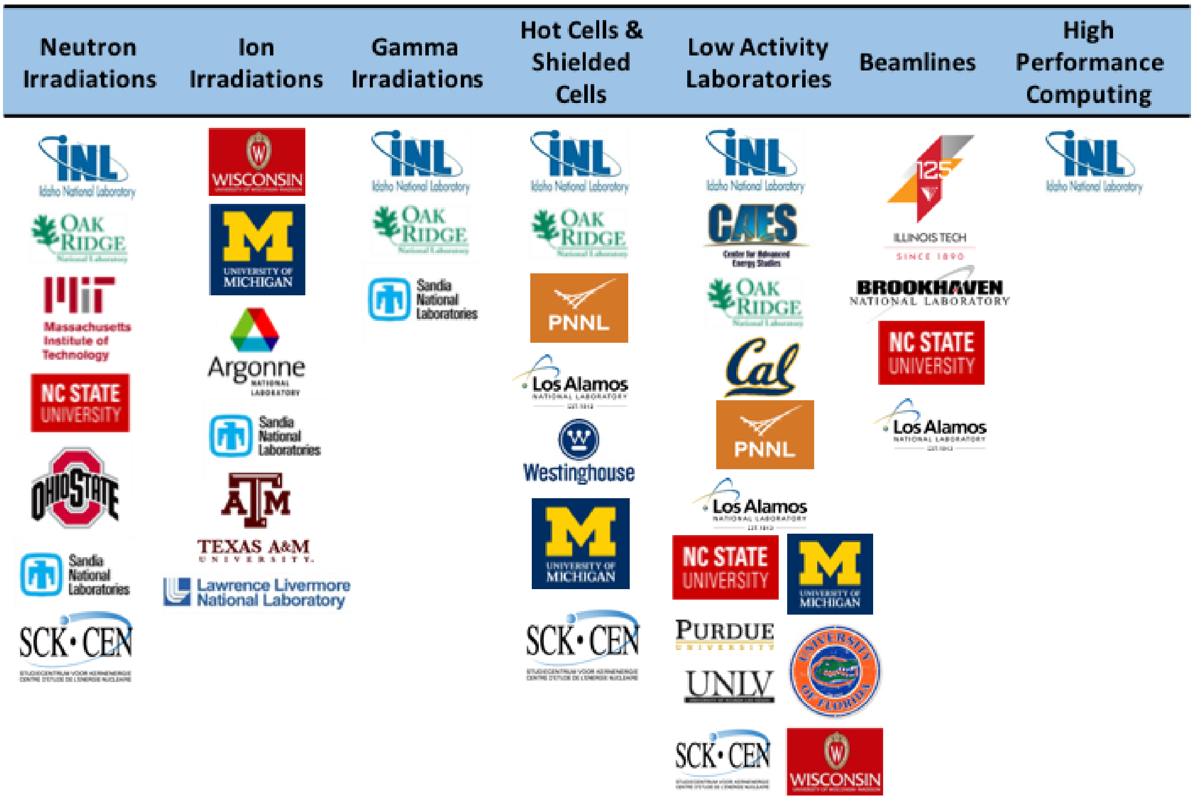

Figure 58: The capabilities of the NSUF according to the institutions that house them. The institutions are represented with their logos. (Figure courtesy of Brenden Heidrich and Alison Hahn).

Investigative Areas of the NSUF

The focus of the NSUF is irradiation effects in nuclear fuels and materials. Within this scope, NSUF projects investigate specific scientific questions that remain open to complete our understanding of the irradiation behavior of fuels and materials, regardless of the Technical Readiness Level of the material. Thus, NSUF projects study materials that are new and innovative all the way to materials that have been long implemented in the commercial realm in order to deepen our scientific understanding. Topical areas of NSUF projects include advanced fuels, advanced cladding and structural materials, radiation resistant materials, sensors, materials from advanced manufacturing techniques, fundamental understanding of reactor materials, welding and joining behavior of materials, and high-performance computing associated with modeling and simulation including model validation. Under these general topic areas, the NSUF currently manages and conducts approximately 130 projects both large and small.

IAEA Activities on Fuel Irradiation Tests, Post Irradiation Examination (PIE) and PIE Facilities Database

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Fuel irradiation tests and related post irradiation examinations (PIEs) are necessary to demonstrate the acceptable performance of the power reactor fuels under various plant states and to justify the specified maximum burnup limit for a new design. Under sub-programme 1.2.2, the IAEA has traditionally led international cooperation in these areas, aiming at discussion about status, problems and perspectives of fuel irradiation testing and PIE facilities; development and application of new techniques for PIE; and characterization of conventional and innovative power reactor fuels. The IAEA also supports maintaining a database for PIE facilities under this sub-programme.

The cost of fuel irradiation tests in material test reactors and PIE continues to increase, while the availability of facilities for such activities has steadily decreased in the last decades. Under these circumstances, the IAEA has recognized that there is a need to promote efficient use of such facilities. Maintaining the PIE facilities database could be the first step moving forward in this direction. As the next step, an advanced model could be considered to facilitate international collaboration, by which Member States can gain timely access to relevant infrastructure based on existing PIE facilities for training or research purpose. Indeed, the IAEA has developed a set of services to support building of nuclear competence using existing nuclear facilities. The International Centre based on Research Reactors (ICERR) is an example to illustrate such IAEA's efforts.

In this presentation, the authors are intended to provide an overview of IAEA activities related to fuel irradiation tests and PIE as well as status of PIE facilities database that is maintained by the IAEA. The overall concept of ICERR is also introduced as an example for further consideration of international collaboration.

New integrated sample management software at PSI HOTLAB

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The department Hotlab (AHL) of Paul Scherrer Institute (PSI) accounts, according to internal as well as external regulations, nuclear materials and moderators. The capacity of the Hotlab is limited by various dispositions by the Swiss regulator ENSI (Eidgenössisches Nuklearsicherheitsinspektorat) and the Swiss Federal Office of Energy (SFOE).

For the detection and control of nuclear fuel samples and monitoring with respect to criticality safety, AHL used since 1998 KBuch.

In autumn 2011, a logic error was detected in the software during a routine booking with extraordinary nuclear fuel. This bug was reported to the national regulator ENSI (2012) and appropriate administrative measures have been taken to prevent the recurrence of the logic error.

ENSI additional requirement: “Given the fact that the logic fault is inherent and can’ t be resolved in the short term, effective measures to prevent future accounting errors must be taken to medium term. The software should be developed to state of the art as soon as possible”.

Subsequently, a project (Streit et al., 2014) was launched to update the used software as required. The project was set up to integrate the new sample and order management (IPV) software (Figure 1) into the Quality Management Software IQSoft used at PSI Hotlab (Zubler et al., 2014).

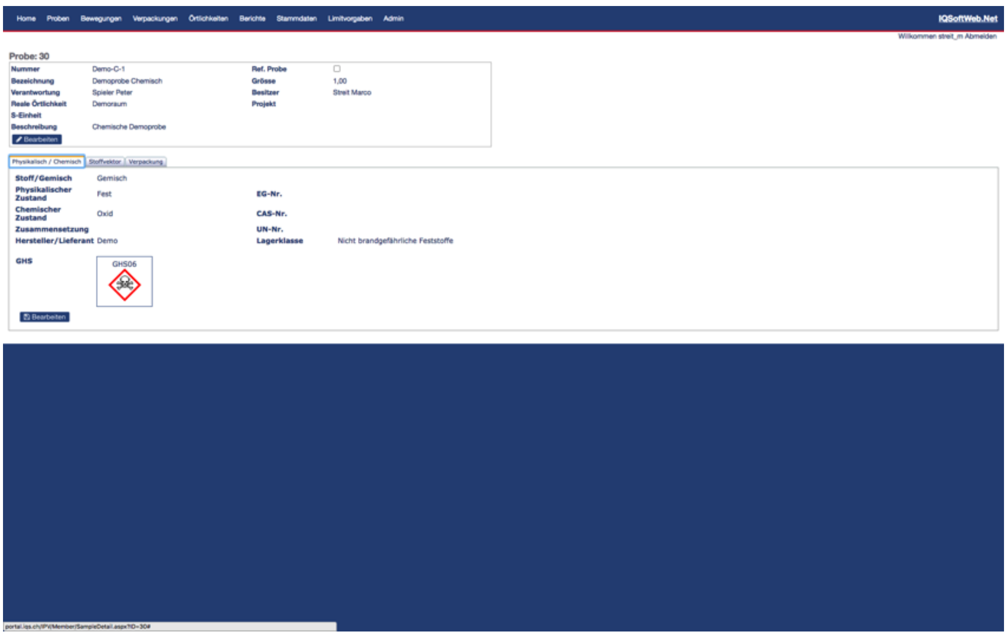

Figure 59: Print screen of the Integrated Sample and Order Management Software

A strong interaction with IQS Ltd. was created to ensure information exchange at the right time, during development as well as the extensive testing phase (Jany, 2015; Streit et al., 2015) of the new software up to today were the software is in use and further user-friendliness is under development.

In Spring 2017 the software was approved by the Swiss regulator (ENSI, 2018) and with the 1st of September 2017 the old KBuch was shut down and IPV was taken over its functionality.

AHL and IQS Ltd are further developing the functionality of the Software together. Due the possibilities of the Database setup special reports can be generated by AHL itself.

The actual presentation is a follow up of the presentations given in 2014/2015 and will shortly summarize the concept, the testing phase and lessons learned during this period. The today's status after approval of the regulator and the further planned development are presented.

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Shielded cells design and periodic safety review

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The periodic safety review consists in a systematic reassessment of the safety of a facility carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience, technical developments, new regulation and guidelines comparison and siting aspects, and is aimed at ensuring a high level of safety throughout the operating lifetime of the plant. In Belgium, the periodic safety review is mandatory for waste facilities, interim storage facilities, radiopharmaceutical producers, etc. In such a facility the number of shielded cells that have to be reassessed is consequent. It could also be highlighted that some facilities have been built in the 70's under legislation, regulation and guidance very different from the present one. The comparison of the design of the shielded cell, with a series of more recent guidelines and practices, makes it possible to highlight design defects, poorly controlled modifications, wear and tear of certain parts, etc. The corrective actions proposed by the operators make it possible to reduce the possible gap between up to date references and an old design.

Regulation

In Belgium, the safety authority is the Federal Agency for Nuclear Control (FANC). Some of its tasks have been delegated to Bel V (www.belv.be), its subsidiary. Bel V inspects class I facilities (2001) and a subcategory of class II facilities (called class IIa) mainly consisting of cyclotrons and irradiators. Every ten years, class I facilities are asked to perform a periodic safety review (IAEA SSG25). In Belgium this exercise is not only mandatory for NPP but also for radioactive waste facilities, radioactive interim storage facilities, fuel fabrication facilities, radiopharmaceutical producers… In such a facility the number of shielded cells that has to be reassessed is consequent. It could be also highlighted that some facilities have been built in the 70's under legislation, regulation and guidance very different from the present ones. According to IAEA definition, the periodic safety review consists in a systematic reassessment of the safety of a nuclear power plant carried out at regular intervals to deal with the cumulative effects of ageing, modifications, operating experience, technical developments and siting aspects, and is aimed at ensuring a high level of safety throughout the operating lifetime of the plant.

Safety function

Two safety functions are mainly reviewed and reassessed during the periodic safety review of a shielded cell: the integrity of the static confinement and the dynamic confinement, i.e. mainly the underpressure cascade of the shielded cells in their environment. A third safety function of the shielded cell is the shielding in itself. Usually no problem is highlighted on this third safety function. A reassessment concerns also the processes hosted by the hot cell and the equipment annexed like the fire extinguisher system.

Static confinement

To review the static confinement, the methodology followed by most of the licensees consists usually in retrieving the plans “as built” of the shielded cell (sometimes from the 70’s), identifying the modifications that have been set up by the licensees on the installation (especially on the alpha box) often to ease the process inside the hot cells. An identification of the weak points, subsystem by subsystem compared to guidelines (ISO 11933 - 1, 2, 3 & 5) is done by the licensees and a proposal to correct them is submitted to the regulatory body that will also perform an independent review and possibly add extra remarks and corrective actions. This may lead, in the worst case to a foreseen and programmed stop of the activity, a full decontamination and a full refurbishment of the shielded cells. Most of the time after this heavy work, a leak test following a well-known methodology described in a guideline (ISO 10648 - 1 & 2) agreed with the licensee is asked before the re-start of the activity. This test is necessary to demonstrate the efficiency of the shielded cell’s refurbishment and improvement.

Dynamic confinement

The dynamic containment of the cell in relation to the immediate environment is also subject to periodic safety reassessments. The methodology used essentially consists of first reassessing the physico-chemical characteristics of the source term in order to determine and classify the shielded cells in relation to a risk level. In a second step, the depression cascade, the redundancies of some systems and the renewal rates implemented in the alpha box of the shielded cells are compared to guide documents (ISO 17873). A re-evaluation of the filtration means (ISO 11933: 4; ISO 2889; ISO 16170; NFEN 779; NFEN 1822) in direct contact with the cell’s atmosphere is also carried out by taking into account the type of gaseous effluents that can be generated by the process located in the shielded cells.

Process

Shielded cells usually house a very wide variety of processes. During the periodic safety review, the operator determines the physico-chemical constraints generated by its process. In this way temperature, pressure, dose rate ranges as well as compatibility with the chemicals used are determined. The suitability of the materials used in the process (tubing, joints, materials, etc.) is then estimated in relation to the constraints and compared with guide documents (NFEN 10204; ASME - IX; NBNEN 287 – 1; ISO 15614 – 1; ISO 9712; ISO 3585; ISO 7619; ISO 10380; EN 1779). The impact of the process on the filters is also evaluated. This may concern, for example, the means of liquid retention or the fire detection and suppression system.

Discussion

Typically a series of weaknesses are identified during periodic safety reviews. The list of the weaknesses is discussed between the regulator and the licensee. These affect both the safety functions assigned to the shielded cells and also the process. Concerning the static confinement, weaknesses are usually identified at the interfaces between the alpha box and ancillary equipment such as remote manipulators, doors, sample transfer connections, rotating parts passing through the alpha box, connections for liquid transfers, etc. One of the points of attention concerns in particular inflatable joints whose poor condition, wear over time or inadequate design can have an impact on both static and dynamic confinement via uncontrolled introduction of compressed air into the shielded cell. Feedback on the operation of fire suppression systems using inert gas and studies to re-evaluate their effectiveness may lead to their replacement or to the adoption of new extinguishing techniques. The absence of redundant equipment, for example at the level of the filtration of the atmosphere of the cells, can lead to important modifications of the shielded cells.

During this reassessment work, the modifications made to the shielded cells are traced and documented if this had not been the case. After the weaknesses identification, corrective actions are proposed by the licensee. These corrective actions have also been discussed between the regulator and the licensee, implemented and at the end of the process, approved by the regulator.

Conclusion

The periodic safety review of the shielded cells design verifies that the condition of the shielded cells reaches at least the same level of safety as when they were installed. The comparison of the design of the shielded cells, sometimes installed more than 40 years ago with a series of more recent guidelines, makes it possible to highlight design defects, poorly controlled modifications, wear and tear of certain parts, etc. The list of the weaknesses is discussed between the regulator and the licensee. Corrective actions are discussed between the regulator and the licensee, implemented and at the end of the process, approved by the regulator. The corrective actions proposed by the operators make it possible to reduce the gap between current references and guidelines and an old design.

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ISO 11933-2 : Components for containment enclosures-Part 2: Gloves, welded bags, gaiters for remote handling tongs and for manipulators.

ISO 11933-3 : Components for containment enclosures-Part 3: Transfer systems such as plain doors, airlock chambers, double door transfer systems, leaktight connections for waste drums.

ISO 11933-5 : Components for containment enclosures-Part 5: Penetrations for electrical and fluid circuits.

ISO 10648-1 : Containment enclosures-Design Principles.

ISO 10648-2 : Containment enclosures-Classification according to leak tightness and associated checking methods.

ISO 17873 : Nuclear facilities - Criteria for the design and operation of ventilation systems for nuclear installations other than nuclear reactors.

ISO 11933-4 : Components for containment enclosures-Part 4: Ventilation and gas-cleaning systems such as filters, traps, safety and regulation valves, control and protection devices.

ISO 2889 : Sampling airborne radioactive materials from the stacks and ducts of nuclear facilities.

ISO 16170 : In situ test methods for high efficiency filter systems in industrial facilities.

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Accumulation of Nuclear Material in Nuclear Facilities: An Iterative Approach In Order To Develop Measuring Stations

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Measuring the deposits of nuclear material accumulated during processes in nuclear facilities is a major challenge in terms of safety and criticality. The characterization of the nuclear material, "holdup", has to be taken into account since the design of new facilities, but also during their operation, and finally for the dismantling of historic equipment and facilities. Considering the diversity of encountered configurations, the holdup measurement is specific to each case. In this context, the Nuclear Measurement Laboratory of CEA Cadarache is specialized in developing and implementing gamma and neutron measuring stations, based on preliminary design and performance assessment by numerical simulation, then on iterative calculations taking into account the feedback of field measurements. In this paper, we illustrate this approach on different case studies, such as glove boxes in hot labs, covering the design, exploitation and dismantling phases of nuclear equipment and facilities.

Context and need

French and international safety authorities require that facilities using sensitive nuclear material (U, Pu, Th) guard against the risks of loss, theft and diversion of these nuclear materials (US Nuclear Regulatory Commission).

Many solutions such as weighing, measurement, physical monitoring can meet this absolute need in operation. However, over time, the deposits of low amounts of nuclear materials that accumulate during processes in nuclear facilities, hot labs, hot cells, and glove boxes can lead to the retention of significant quantities of nuclear materials. This "holdup" has to be taken into account both in old installations, in order to dismantle them properly, in installations currently in operation, and preventively in installations under construction.

Therefore, measuring low amount of deposits of nuclear material is a major challenge. Today, the possibilities offered by non-destructive nuclear measurement provide solutions for many installations (Los Alamos National Laboratory).

Contribution of the Nuclear Measurement Laboratory of CEA Cadarache on hold-up measurements

Analysis context and methodology. The Nuclear Measurements Laboratory (LMN) of CEA Cadarache is specialized in developing and implementing gamma and neutron measuring stations, based on preliminary design and performance assessment by numerical simulation, then on iterative calculations taking into account the feedback of field measurements.

In our methodology, the first step in designing such measuring stations, in a glove box or in a hot cell for example, is to establish the list of radiations emitted by the studied nuclear material in order to select the more suitable nuclear measurement.

This makes then possible to choose the sensor and the detection chain whose performances are the most adapted for a given configuration. For this, the laboratory relies on its long experience in nuclear gamma and neutron measurements, and photon imaging as well.

Finally, numerical modelling of the selected detector in its operating environment is carried out in order to determine, after several iterations, the optimal measurement configuration (type, size and position of the detector) and to define the sensitivity of the measurement (to different parameters such as the type, mass and distribution of nuclear materials).

Recent examples of measuring stations design by the LMN. Our lab is solicited for different studies ranging from the simplest case of new installations, where the design of the measurement system is open, to the most difficult case of historic installations, where it is necessary to adapt to existing implementation constraints, and to more or less known history leading in some instance to poor knowledge on nuclear materials.

The two examples below illustrate both cases:

- The first situation is concerning a hot cell under construction where nuclear material will be reconditioned. The nuclear material is well known, the geometry of the equipment is still modifiable. The choice and the position of the detector can be optimized by successive numerical modelling comforted by experimental campaigns (Figure 60).

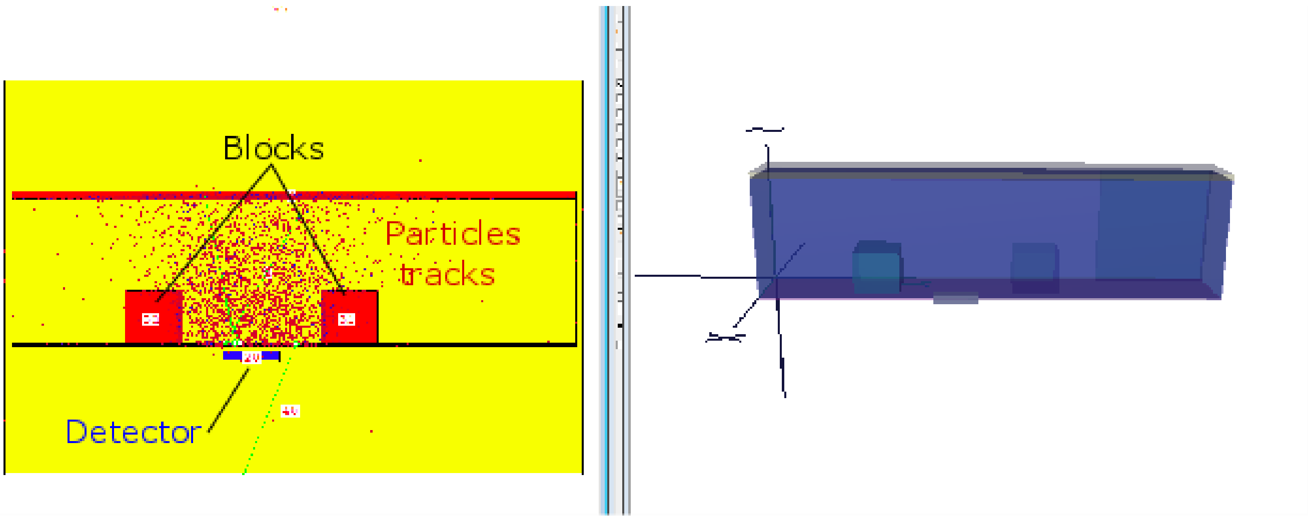

Figure 60: Numerical modelling of a holdup measuring station for a reconditioning glove box.

- The second case concerns an old equipment used to characterize nuclear materials. This equipment had to be moved in a safe way. In this case, numerical simulation has been used in order to select the best measurement configurations. Then experimental results have been interpreted using numerical simulation to obtain quantitative results on the nuclear material hold-up (Figure 61).

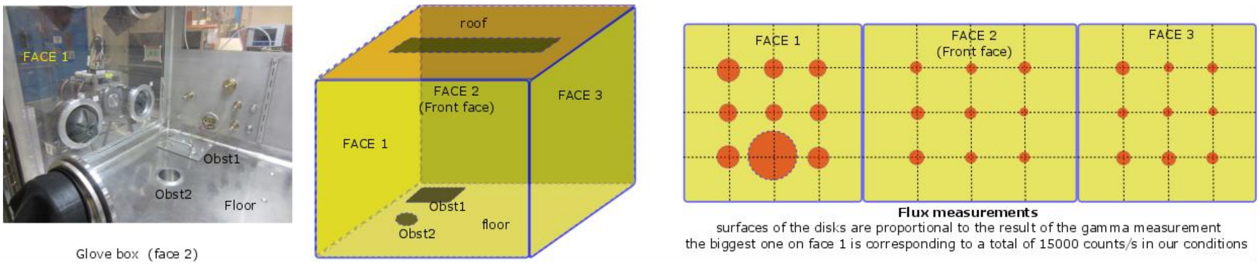

Figure 61: Measurements and quantitative results for holdup in an equipment in view of its transport.

The methodology used by our lab for those cases can easily been applied to other situations.

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TARRA Project: transfer of MOX R & D between 2 CEA sites

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The CEA R&D on U-PuO2 fuels was performed in LEFCA facility in CEA-Cadarache (France) until 2015. The TARRA project is in charge of the transfer of LEFCA activities to Atalante facility in CEA-Marcoule (France) which will be achieved at the end of 2018. This paper describes the installation of 25 glove boxes spread into 2 laboratories of fabrication (Figure 62) and characterization (Figure 63. It focuses on project management, safety and security aspects, new nuclear equipment design.

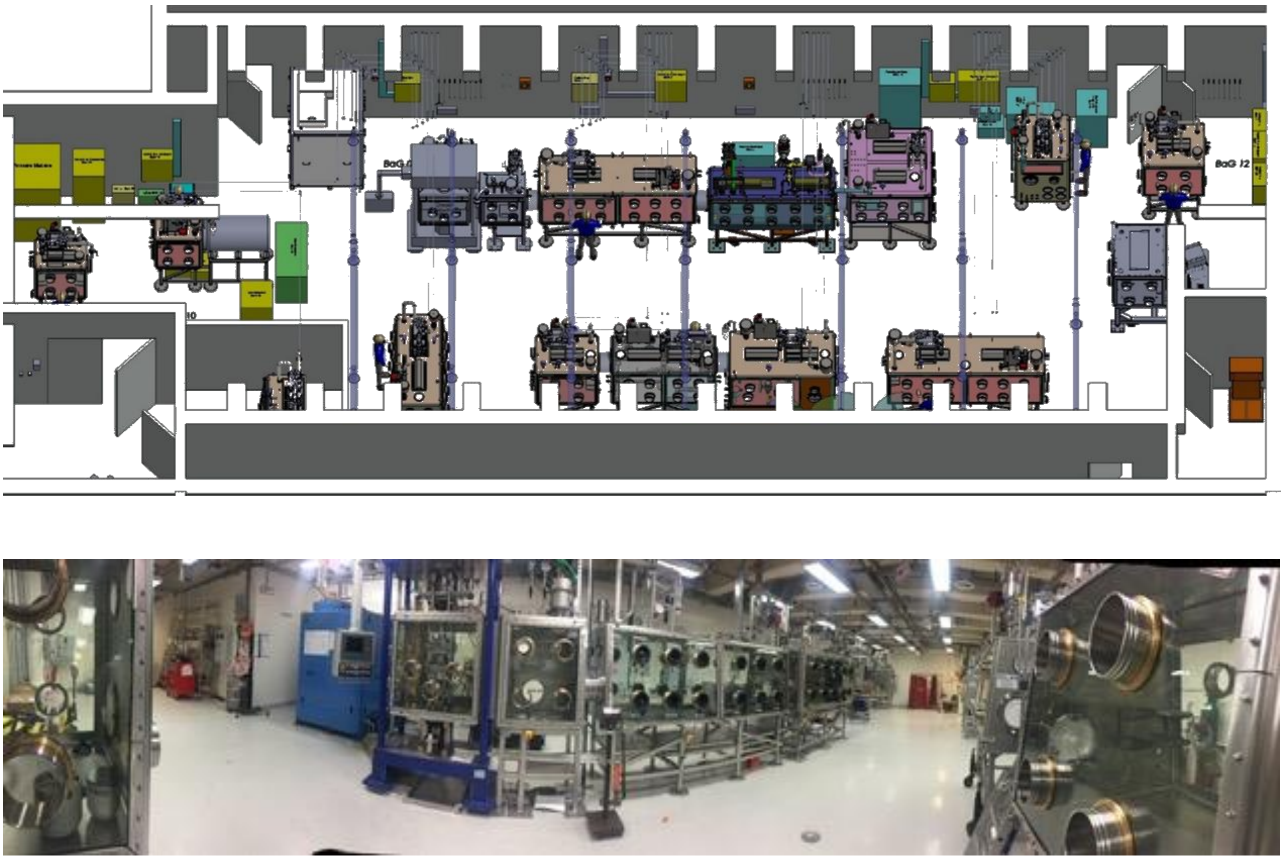

Figure 62. Fabrication laboratory

Figure 63: Characterization Laboratory

Technical Session:

Aging Management of Hot Labs

Rejuvenation of the Canadian Nuclear Laboratories Chalk River Campus

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The Canadian Nuclear Laboratories (CNL), formerly known as Atomic Energy of Canada Limited (AECL) prior to restructuring in 2015, is in the midst of major site rejuvenation activities. Much of the site's buildings and infrastructure was constructed or installed from the 1940s to the 1960s. As a result, many of the buildings and facilities as well as much of the infrastructure is near or at the end of its service life. The Government of Canada has made a significant investment into the Canadian Nuclear Laboratories to bring the site to modern standards and to expand on our existing research and commercial capabilities over a 10-year span. As a result, a number of capital initiatives are underway to make good on this investment.

These major activities include the decommissioning and demolition of a number of outdated or obsolete buildings and facilities, the refurbishment of some existing active laboratory facilities, and the construction of new, modern hot cells, active and inactive laboratories, administrative buildings, and infrastructure to allow CNL to continue with its existing commercial and research missions and to allow CNL to expand into new business and research opportunities that are currently limited by the physical constraints of the existing facilities.

In parallel with these large decommissioning, construction, and refurbishment projects are efforts to ensure that existing facilities remain safely operational until a smooth transition to the newer facilities is possible. There are also ongoing projects to pursue new opportunities outside of the standard CANDU-type work that makes up the majority of the work performed at the Canadian Nuclear Laboratories. This presentation will describe a number of these major activities which are underway, their current states of development, and their paths forward.

Site Rejuvenation Initiatives

Site Construction Projects. Currently there are a number of major capital projects underway at the site not directly related to new or refurbished laboratories and hot cells including, but not limited to, the following:

1) Construction of a new logistics building at the outer gate of the site to ease deliveries and reduce vehicular traffic on-site;
2) Construction of a new office building to consolidate office workers who are currently scattered across site with some in satellite offices located in the nearby town of Deep River;
3) Construction of a new conference centre to better allow for large presentations and conferences to be held on-site;
4) Construction of a new switch-yard and power house for bringing reliable services to the new and remaining facilities and buildings; and,
5) Separation of pedestrian and vehicular traffic pathways to improve industrial safety on-site.

Advanced Nuclear Materials Research Centre

One of the major facilities that will allow for this increased capability is the Advanced Nuclear Materials Research Centre. Currently, the CNL has multiple active laboratories and hot cell facilities that are in different buildings across the Controlled Area of the Chalk River Laboratories site. The ANMRC will consolidate these labs and hot cells into one common building which will increase efficiencies and reduce the number of shielded out-of-building transfers required.

Through the use of strategic functional adjacencies and in-building transfer systems, such as through-wall transfers between adjacent hot cells and transfers to other areas via pneumatic tube systems, active material transfers will require fewer external, shielded transfers which translates to better ALARA practices.

The larger size and layout of this new facility will allow a wider variety of shipping flasks to be handled and unloaded and a wider variety of fuels, such as full length LWR bundles, to be handled. The additional size, layout, and planned workflow of the new hot cells is also expected to significantly reduce the number of conflicts in the cells which will allow for more fluid scheduling and better schedule flexibility.

Hot Cells Bridging Project

With the ANMRC facility still in its conceptual design phase, it is not expected to be realized until the mid-2020s. As such, a bridging project was established to sustain the existing shielded facilities in the meantime. Extensive condition assessments were strategically conducted on the facility infrastructure, safety related systems, and critical processes. In total, 170 facility upgrades were identified in the two facilities. A risk-based methodology was used to assess prioritize the upgrades, taking into consideration the condition, the consequence of failure, and the expected remaining life balanced with the time it will require to conduct the upgrades. A project plan was created to address the most critical jobs in a phased approach that can be adjusted as the conditions in the facility change. The first phase is well underway, and the second phase will be initiated in 2019.

Light Water Reactor Fuel PIE

In parallel to the aforementioned activities, a significant engineering effort is currently underway to allow both non-destructive and destructive PIE of full length LWR fuel elements to be performed in the existing hot cells at the Chalk River Laboratories as opposed to the normal ~500mm CANDU elements that these facilities were designed to handle. This work involves a number of custom designs, including a custom carriage containing a suite of NDE tooling which performs a number of operations simultaneously, modifications to our existing fission gas collection rig, and design of a new fuel cutting tool. Note that, since the hot cell being used for this activity is not dedicated to this work, all of the equipment is being designed for modularity and ease of installation, decontamination, and removal from the hot cell.

An existing flask that is to be modified is mated to the face of the cell to allow for the fuel to be translated through the NDE suite given the limited length of the hot cells that are available for performing this work.

Recent upgrades at the Studsvik concrete hot cell laboratory and cell waste management

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Upgrades of the Studsvik hot cell laboratory over the past two years include installation of an industrial remote control of the in-cell power manipulator, a new fission gas sampling system, a 120 kN in-cell tensile testing machine, new redundant breathable air compressors supporting up to 4 simultaneous frog suits, operator ergonomics, new out-of-cell ICP-MS, ICP-OES and powder x-ray diffractometer and new Parr fuel leaching autoclaves for final repository studies. Here we report some details for the remote-control installation, the custom built tensile tester and how to get rid of bulky waste from the concrete cells.

Remote control of power manipulator

The concrete hot cell line in Studsvik was commissioned in 1960. It consists of five DxLxH 2x2x4 m and two 2.5x4x4 m cells with 1 m wall thickness. The two big cells, each with 2 windows and 2 sets of HWM-A100 master slave manipulators, are divided by a retractable wall. Removing the wall creates a cell of 8 m length. Laboratories for chemical analysis, final repository studies, electron microscopy, low and medium level waste treatment and a mechanical shop for master-slave manipulator maintenance and an automated storage room for in-cell equipment surrounds the cell line.

To move larger equipment such as fuel assemblies, transport cast insets, full-length fuel rods or bulky machinery through the big cells a PAR M3000 power manipulator (PM) can be used. Originally it was fitted with a joystick control box on wheels. The control box connects with wires to the control cabinet of the PM. The operator must move from cell window to cell window to observe the motion of the PM and rolling along with the joystick control was not easy with all cabling and the rather bulky size of the joystick control. To facilitate the surveillance the joystick control was exchanged to an industrial remote control (Åkerströms M300J), see Figure 64. Operator safety was also improved as the emergency stop function via the remote control complies with the latest standards and the risk of stumbling over wires is excluded.

In-cell 120 kN tensile tester

The Studsvik fabricated in-cell tensile testing machine see Figure 64**,** is fully mechanical and loads the specimens by means of a motor and a gear. The specimen under test is held in place by hydraulic grips. Although the load cell has a capacity of 150 kN the maximum load capacity is constraint by the grips and therefore only up to 120 kN. An Instron 8800 controller is used for load control while a Eurotherm controller is used for split-tube furnace surrounding the specimen under test. The furnace can reach operational temperatures of up to 1000 °C. The machine makes it possible to do tensile tests on reactor internal materials under typical in-reactor temperatures.

Cell waste management

Bulky cell waste, like broken fuel cutting machines or discarded PWR fuel assembly spacers, that are impossible to decontaminate needs to be handled remotely all the way to the final repository. Such wastes are packed in standard boxes dedicated for the final storage for intermediate level waste for long lived nuclides (SFL). The outer dimensions of the standard box are 1.2x1.2x1.2 m^3. The internal dimension varies depending on the shielding necessary. For hot cell waste e.g. fuel cutting machines, the box is made of concrete with a wall thickness of 1 dm and weighing about 2 tons empty. Inside the concrete box is a steel box with wall thickness of about 5 mm.

Upon getting waste out of the cells the concrete box is placed in the service area adjacent to one of the big cells. The wall between the big cell and the service area is removed (it moves on rails). The steel box is then lifted from the concrete box by the power manipulator, transferred into the big cell and the wall is closed. The cell waste is now stowed into the steel box by master slaves. When finished a lid is fastened on the box. The process is then reversed i.e. the wall is removed, the power manipulator transfers the steel box and places it inside the concrete box. A concrete lid is put on the concrete box. Depending on the dose rate on the concrete box it can be put in a lead box before transport to the interim underground storage at the Studsvik site.

Figure 64: In cell 120 kN tensile tester (left) and industrial remote control of the power manipulator.

Refurbishment of Handling Equipments in A Maintenance Cell of Phenix

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PHENIX is a French fast breeder nuclear power plant which went critical in 1973 and turn off power operation in 2009. The decommissioning project started in 2005, in order to start immediately after the final shutdown and achieve the dismantling operations as soon as possible in accordance to the French Safety Authorities principals. The critical path of the plant dismantling is the evacuation of the fuel elements and after the evacuation of the Lateral Neutron Shielding Assemblies.

The "Commissariat à l'Energie Atomique et aux Energies Alternatives" (CEA) is the PHENIX operator.

The decommissioning project is headed by the CEA.

Statement

The way of evacuation of the fuel elements from the plant called handling route, needs the availability of several equipments, mainly handling equipments.

This handling route is only at half of its life because the evacuation of the fuel elements and of the lateral neutron shielding assemblies uses the same route to get out of the plant. The analysis of the handling route and the roadmap to increase the reliability of each equipment has been presented in HOTLAB 2015.

The refurbishment of handling equipments in the maintenance cell located on the upper part of the Irradiated Fuel Cell is the first step to increase the reliability of the handling route.

Roadmap

The two main handling equipements in the maintenance cell are the overhead crane and the biological shield plug gantry crane. The aim is to ensure at all times to close the biological shield plug. When opened, direct intervention on equipments would be forbidden, due to the irradiation level.

The actions that condition the closure of the biological shield plug are:

- The handling back of a load or the hook of the crane to the maintenance cell during a handling in the fuel cell,
- The movement of translation and put back into place of the plug with the gantry crane.

Results

The equipments replaced to achieve the objectives are:

- The trolley and the translation motors of the crane,
- The lifting system and the translation's motorization of the gantry crane,
- The entire electrical system.

The added features are:

- For the crane
  - A troubleshooting system of the crane translation movement installed through the wall,
  - A trolley running on the crane frame with two winding drums of hoisting cables fully redundant, an irreversible reducer to hold on the load, two redundant motors for direction,
  - Setting up camera to improve vision
- For the gantry
  - A troubleshooting system of the translation movement of the gantry,
  - Two hoisting cables fully redundant to hold on the load in case of rupture of one cable and to be able to put down the biological shield plug with one cable.

The complementary design requirements are:

- The new crane trolley has to be lighter than the existing one (no change for safety load calculations),
- Use of usual spare parts.

All the new equipments were fully tested before assembly in the maintenance cell in order to limit the cell unavailability. The work in the Irradiated Fuel Cell kept ongoing during the assembly in the maintenance cell.

Conclusion

The refurbishment of the crane and of the gantry in the maintenance cell was carried out successfully between October 2016 and March 2017. The new equipments were commissioned in March 2017.

The second step to increase the reliability of the handling route is the refurbishment of the handling equipments in the Irradiated Fuel Cell. It starts in June 2018 with the replacement of the transfer lifting unit.

Refurbishment of Drum Lifting Device for Radioactive Waste Handling inside Hot Cell Facility

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Radiometallurgy Installation – BATAN is a post-irradiation examination (PIE) facility for nuclear fuel element and structural components of nuclear reactor. Due to the high radiation exposure of the spent fuel so many test equipment and tools that has been installed inside the hotcell was broken, including Drum Lifting Device (DLD). DLD is a tool to move non-nuclear waste out from inside the hotcell. Therefore it was necessary to be carried out refurbishment of the DLD. Remote decontamination, personel intervention and repairing DLD components were steps to refurbishment of DLD. After that refurbishment, operators can move non-nuclear wastes out easily without contact those directly anymore.

***Keywords**: refurbishment, DLD (Drum Lifting Device), spent fuel, Hotcell.*

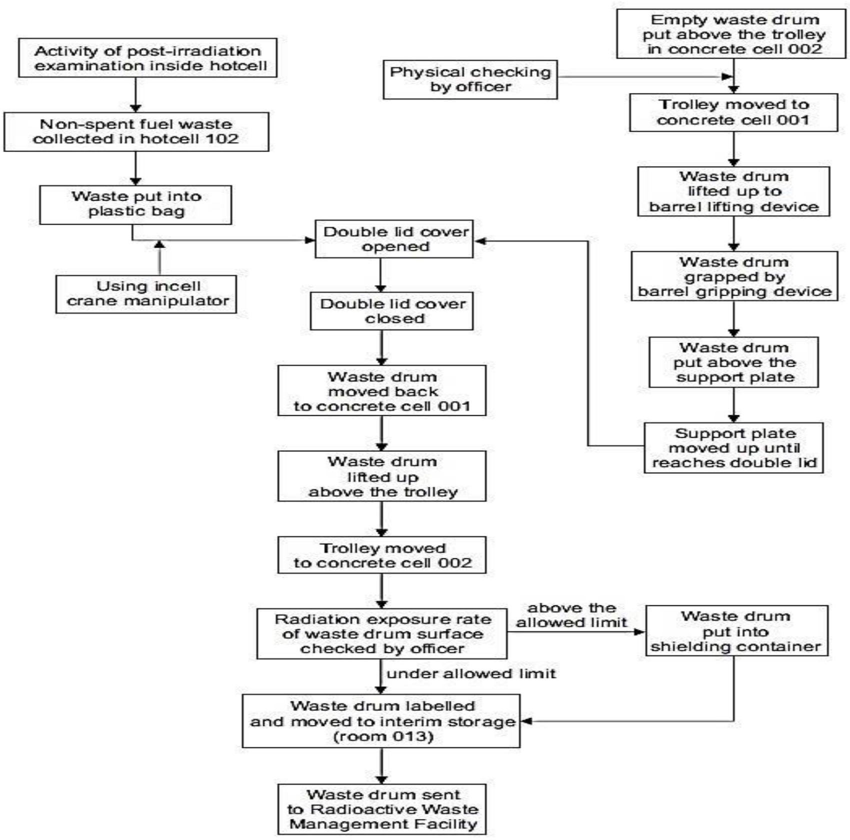

Figure 65: Mechanism of movement of non-nuclear wastes in RMI (Radiometalurgy Installation).

Technical Session:

Decommissioning and Waste Handling

Treatments of Radioactive Waste Solutions Generated In A Hot Laboratory of Japan Atomic Energy Agency

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History and current status of CPF

Chemical Processing Facility (CPF) was constructed in 1980 in Nuclear Fuel Engineering Laboratories of Japan Atomic Energy Agency (JAEA) for researches on reprocessing of spent fast reactor (FR) fuels and on vitrification process of high level radioactive liquid wastes. The facility equips hot cells, glove boxes and hoods, and irradiated and non-irradiated MOX fuels can be handled. The first experiment on radioactive samples was performed in 1982, then applicability of PUREX process on FR MOX fuels with high burn-up and high Pu content has been examined through 21 times active experiments on fast reactor (Joyo, Phoenix and DFR) irradiated fuels from 1982 to 2005. Few fuel pins were treated in a series of the aqueous reprocessing experiments i.e. shearing, dissolving and solvent extraction processes. During this periods, fundamental data of vitrification on high level active waste solution generated from Tokai Reprocessing Plant was systematically collected, and that information contributed to design of Tokai Vitrification Facility (TVF).

In 1996-2002, renovation of the facility was carried out to start new research projects of advanced aqueous reprocessing and of pyrochemical reprocessing (Aose et al., 2007). In the new aqueous reprocessing, separation procedure in the PUREX was replaced by crystallization of U, U/Pu/Np co-recovery by modified PUREX flow-sheet and trivalent minor actinides (Am and Cm) recovery by solvent extraction or extraction chromatography (Funasaka & Itoh, 2007). About 10 times experiments for the new process have been carried out in the hot cell so far. Special glove boxes with Ar atmosphere which equip electric furnace were installed in a laboratory for pyroprocess under collaboration with Central Research Institute of Electric Power Industry (CRIEPI), and electrorefining experiments on few grams of U and Pu has been carried out (Kitawaki et al., 2011). Experimental studies about those new reprocessing technologies are still undergoing.

After the accident in Fukushima Daiichi Nuclear Power Stations, not only R&D on the reprocessing but also various analyses on samples taken in the Fukushima site are one of the most important tasks of CPF. Currently future operation and maintenance plans of CPF are under discussions in JAEA.

Waste solution stored in CPF

More than 30 years experimental and analytical activities inside the hot cells and glove boxes have produced plenty amount of radioactive liquid wastes. The facility has waste solution tanks for storage of active solutions however it does not equip functions of processing the solutions for disposal. Nuclear fuel materials in the waste solutions were recovered using conventional reprocessing procedures and then they were converted to oxide form for storage. The waste solutions with relatively simple compositions such as nitic acid solution or spent PUREX solvent have been transferred into the tanks after the U and Pu recovery. The solutions containing reactive chemical compounds have not been mixed with each other in the tanks in order to avoid unexpected hazardous chemical reactions during the storage, and those have been temporarily stored inside the hot cells or glove boxes separately in small bottles. Those solutions will obviously be one of the most troublesome wastes at the time of decommissioning of this facility which is already under discussion,

Those chemical compounds are necessary to be removed or decomposed before mixing with other solutions inside the shielded environment with specific restrictions. Appropriate treatment procedures of those solutions have to be individually developed after careful risk assessment. JAEA has started systematically investigation on safety treatments of the radioactive waste solution accumulated in the CPF from 2015, and several kinds of the solutions generated by the experiments have been successfully processed inside the hot cell and residual solutions were transferred into the tanks. However, treatment procedures for large part of solutions generated by the analyses and organic solutions are still uncertain due to their complicated compositions. In order to develop appropriate processes for those liquids, collaborative research programs with several universities and national research organizations were started from 2017.

Treatments of waste solutions in CPF

Treatments on most of the aqueous experimental wastes and on a part of the aqueous analytical wastes have already been finished at the end of 2017. Examples of the treatment procedures are followings;

- Phosphoric acid solution
  The solution was used as a electrolyte of electrolytic decontamination experiments inside the hot cells, and it contained 3 mol/L of phosphoric acid, 7×10^5 Bq/mL of ^{137}Cs, 4×10^5 Bq/mL of ^{239}Pu + ^{240}Pu and 8×10^5 Bq/mL of ^{239}Pu + ^{241}Am. Those solution was solidified by adding Al^{3+}, and then disposed as high level solid wastes.
- Lactic acid
  This solution was used as reductant of Pu in a modified PUREX process experiment conducted in a hot cell, and it contained 2 mol/L lactic acid, 1×10^3 Bq/mL of ^{137}Cs, 2×10^4 Bq/mL of ^{239}Pu + ^{240}Pu and 1×10^5 Bq/mL of ^{239}Pu + ^{241}Am. All lactic acid was oxidatively decomposed using Fenton reaction to be CO_2, acetic acid and formic acid without releasing hydrogen gas. The residual solution was sent to one of the liquid waste tank.
- Solution containing Chloride ions (Tada et al., 2017)
  The solution was generated by series of experiments on pyrochemical process and by analysis on contaminated water containing sea water samples at the Fukushima site. So far, chloride ions have not been aggressively treated except in the glove boxes for pyroprocess due to their corrosiveness. The waste solution of the pyrochemical experiments contained totally 70 g of U and 12 g of Pu. Those elements were recovered through solvent extraction with PUREX solvent after exchanging anion from chloride to nitrate by adding $Ag(NO_3)$ for AgCl precipitation formation. The precipitation was disposed as the solid wastes, and U and Pu are stored as oxide form. Chloride ions in the analytical samples of the Fukushima were treated with the same manner with the previous one.

Appropriate treatment procedures of those solutions have been experimentally examined by inactive experiments in advance with operations on the genuine waste solutions. The residual solutions will be treated based on small scale inactive and active demonstrations whose procedures will be developed through the collaborative researches. In our current schedule, treatments of all those solutions in CPF will be completely finished by the end of FY2020.

Current activities of the collaborative studies; STRAD project

The collaborative studies were combined to be a one project at the beginning of 2018 in order to share experiences and knowledges between collaborators. The purpose of this project is contributions not only to the developments for waste solution treatment in CPF but also to waste management or decommissioning of other nuclear facilities. The project was named as Systematical Treatment of Radioactive waste solution for Decommissioning (STRAD), and targets of study and collaborators are currently increasing to address forthcoming decommissioning of various facilities treating radioactive nuclides. Current representative studies in this project are decomposition of reactive chemicals in analytical waste liquids, recovery of nuclear materials from organic solvents and solidification of waste liquids to confine reactive compounds inside the solid. Some achievements of these studies will be shown in the presentation. The project is expected to produce beneficial waste management database which can be referred world widely.

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Management and surveillance of aging underground storage site and aging Al-clad metallic uranium legacy fuel

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Norway is in the process of decommissioning storage sites with chemically unstable Al-clad metallic uranium legacy fuel and is doing preparatory steps for final deposition of the fuel.

Since the mid-60s IFE manages an underground storage site for canisters with spent, legacy fuel elements. The fuel was used from 1951 to 1967 in the first Norwegian research reactor JEEP I. JEEP I is decommissioned many years ago and has been replaced by other Norwegian research reactor. Documentation and knowledge of the legacy fuel are original design drawings with material specifications as well as in- and out-of-pile visual inspection records. Each JEEP I legacy fuel element consists of 2 Al-clad, metallic uranium fuel rods. Records give written information on each fuel element, namely on time in-pile, burn-up (<600 MWd/tU) and fuel element status as observed by visually during biannual interim inspections and after discharge. In the 50s and early 60s discharged JEEP I fuel elements were placed in a cooling pond. In the mid-60s the elements were transferred to dry canisters and placed in steel clad vertical pits in an underground storage site, where they are kept until today. One canister in each pit. In the early 80s, some pits and open Al-canisters were subjected to water ingress. Subsequently, the pits were dried and painted. Further, after >20 years storage all elements were cold and were repacked into new, dry, gas-tight stainless steel canisters with screw lid. The pits were kept sealed by IAEA until about 2008.

Figure 66: Spent fuel storage pits and example canister.

The main risks associated with storage of spent, chemically unstable Al-clad natural U metal fuel with graphite end plugs are:

- Radiation, if canister is lifted up from underground storage position.
- Contamination of surrounding environment by leakage from canisters, if water ingresses into defect canisters and fuel elements.
- Fire, if a (defect) fuel element is exposed to high temperatures, friction, mechanical shock, and free access of oxygen (to the U metal or UH_3).
- Explosion, if a critical amount of hydrogen gas is released instantly from a canister.

After the seal was taken off, a storage site inspection programme was performed including a full non-destructive and destructive examination of a stored fuel element. The fuel element selected had documented small defects caused by uranium metal corrosion. The fuel rod sample showed as expected in the uranium metal some small corrosion spot with oxidised uranium hydride UH_3.

Recognised issues to be worked on in the underground storage site after long time storage were:

- Unknown integrity of black-steel clad pits – unknown number of intact pits.
- Unknown integrity of stainless steel storage canisters, corrosion marks on their exterior, their interior atmosphere and leak tightness for hydrogen and uranium.
- Unknown number of canisters containing defect fuel elements.
- Unknown dose-rate profile alongside the canister axis.
- Unknown integrity of defect fuel elements and unknown fraction/percentage of corrosion product UH_3 present in defect Al-clad fuel elements. UH_3 formed at higher temperatures e.g. in-pile or in the early stages of fuel storage.
- Unknown amount of hydrogen in fuel canister from ongoing anaerobic corrosion/oxidation of metallic Uranium fuel, Aluminium cladding and from ongoing anaerobic oxidation of UH_3. (e.g. at lower temperatures during storage). During the long time “dry” storage a thin protective oxide layer formed, namely underneath the Al cladding on the reactive uranium and on the UH_3 corrosion products.
- Unknown extended of Wigner effect / fire in the irradiated graphite end plug.

The actual management programme for the storage site and the stored fuel includes:

- Maintenance and improvements to the infrastructure of storage site itself
  - maintenance of cracked concrete floor, crane, installation of air conditioning (temperature and humidity), ventilation, outlet filters, fire alarms, fire extinguisher,
- Surveillance and radiological monitoring of the air and ground in and around the storage site
  - radiation monitors, water sampling and radioisotope analyses,
  - environment analyses

- Characterisation and maintenance of storage pits
  - Check movability of canister in pits
  - Make canister movable in pits
  - Control water ingress
  - Wipe outer surface of canister clean and dry
  - Cleaning of pits for rust and removal of natural condensed water in the pits
- Characterisation and monitoring the dry storage itself, namely the canisters
  - Control all canisters with regard to leak tightness (pressure test/leak test) and corrosion (visual inspection)
  - Analyze environment inside the canister with respect to humidity and hydrogen (gas-sampling)
- Characterisation and monitoring of the fuel elements
  - Dose rate measurements and X-ray examination of the entire length of all fuel canisters to document the entire fuel element on integrity, corrosion attacks of the U-metal, and dose rate profile.

What are next treatment steps of the stored fuel? As known, the U metal fuel is chemically unstable. Uranium metal reacts with water and air and therefore cannot be placed in a repository without stabilization measures. The goal of all U-metal-fuel treatment and stabilization measures is to make the U-metal-fuel fit for final depository by transforming/oxidising chemically unstable U-metal and UH_3 into stable U-oxide. Stabilization can be achieved in a slow gas treatment (oxidation) or by wet chemical processing. Necessary preparatory steps prior to a transport abroad for stabilisation or reprocessing are

- Qualitative and quantitative documentation of the fuel inventory and defects in fuel and cladding, their location, volumes and sizes.
- Atmosphere control in canisters (dry storage). When ongoing corrosion with hydrogen production is observed the atmosphere in the canisters and fuel element has to be dried (vacuum dried) and changed into a slightly oxidizing atmosphere.
- Dismantling elements and perform size reduction measures of fuel rods in a controlled atmosphere (glove box or hot-cell).
- Repacking of fuel segments for transport.

The presentation shares experiences from characterisation techniques e.g. pressure testing, hydrogen measurements and x-ray examination of fuel elements.

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Remote Handling and Waste Containment Approach for Whiteshell Laboratories Standpipe and Bunker Legacy Waste Retrieval

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Veolia Nuclear Solutions (VNS) has been contracted by Canadian Nuclear Laboratories (CNL) to design and deliver a system to retrieve and repackage legacy waste. The waste has a variety of forms and conditions as described below and requires a flexible and robust set of remote handling devices housed within 3 major shielded containments to complete the tasks. The focus of the presentation is to highlight the remote handling, material handling and containment philosophies being used on the project for processing retrieved materials that are analogous to hot-cell design and operation. This approach is based on past experience and uses a range of remote handling devices and techniques matched to the variety of tasks in the project. The waste material fully contained remote processing design approach from retrieved waste receipt, processing and packaging will be discussed and presented.

CNL's Whiteshell facility is under decommissioning and it is necessary to remotely retrieve, characterize, condition, and repackage some legacy waste stored underground on-site using a variety of systems. The waste is stored in underground standpipes and bunkers. Standpipes are concrete pipes mounted vertically in the ground, which were capped when full. The bunkers are underground rectangular concrete bunkers with above ground roofs. The waste includes high dose rate materials, fissile materials, potentially pyrophoric materials, chemicals, and combustible gases. It must all be handled safely in a remote and contained environment. The waste being removed is in a variety of forms coming from reactor fuel experiments. It is packaged in boxes, bags, barrels and paint cans. The condition of the packages varies from intact to degrees of rupture/disintegration, all of which must be handled by the system in their current condition.

Remote Handling

Overall Approach. The Whiteshell project is challenging requiring a variety of remote handling approaches. This project is a good reference to highlight an overall remote handling approach to complex decommissioning jobs.

For the Whiteshell project the remote handling equipment must able to handle a range of waste (Solids, Sludge, and Liquids) in a difficult environment (Cold, Rain/Snow, Wet). Operation vary from remote concrete demolition to delicate waste segregation activities.

This range of activities means that one piece of equipment is not sufficient, and a variety of devices must be used to accomplish the total scope of the project. The remote handling presentation discussion will further focus on the following topics: A. Basic Requirements for all Remote Handling. B. Manual Work. C. Semi-Remote/Manual Assisted Work. D. Lightly Modified Commercial Equipment for Heavy Work. E. Significantly Modified Commercial Equipment for Heavy Work. F. Custom Equipment for Fine Work.

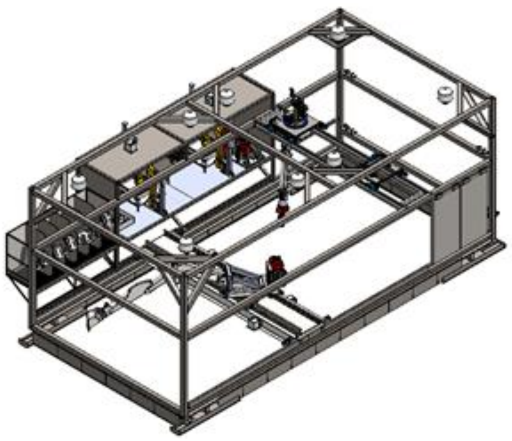

Figure 67: Bunker Remote Excavator System

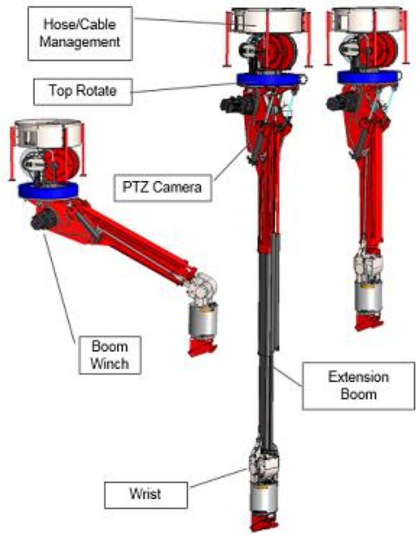

Figure 68: Long Reach Waste Retrieval Tool

Material Handling and Containment

Overall Approach. The Whiteshell project process the waste materials within three major shielded, fully remote operated, and portable, containment structures. The units are designed in accordance with and uses proven hot cell concepts and guidelines. In addition, each unit is designed and equipped for broad operational flexibility for known and also unknown and changing condition waste. The Sorting and Conditioning Unit (SCU) is used for sorting, segregating, characterizing and packaging the waste and is shown in Figure 69 below.

The material handling and containment strategy for the waste is movement of discrete waste packages using a tray-based material handling system. The trays are moved using robust and reliable conveyors that can be remotely maintained in the unlikely event of failure. The waste whether it is being moved from unit to unit, or while being processed in the final sorting unit, remain within containment and behind shielding. The Material Handling and Containment presentation discussion will further focus on the following topics: A. Contamination Containment Design. B. Waste Tray and Material Handling Equipment. C. Waste Transfer Containers. D. Waste Transfer Ports. E. Processing Equipment.

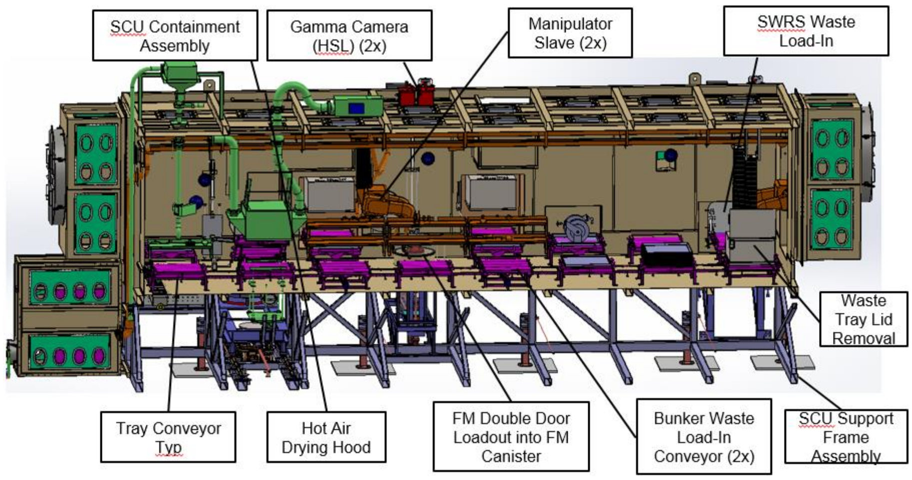

Figure 69: Sorting/Segregating/Characterizing/Packing Unit (shielding not shown).

Fuel Inspection Hot Cell at Ignalina B1 ISFSF – Lessons Learned

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FIHC in Ignalina B1 ISFSF

For storage of the spent nuclear fuel from Ignalina NPP (Lithuania) the Consortium GNS-NUKEM built the dry type Interim Spent Fuel Storage Facility (ISFSF) (B1 Project). The containers with the spent nuclear fuel will be safely stored in the ISFSF. The storage period is 50 years with the possibility, if necessary, to prolong the period. The total capacity of the Interim Spent Fuel Storage Facility is 202 places for the CONSTOR® RBMK1500/M2 casks; the currently planned number of casks with the spent nuclear fuel is 190.

For the unlikely case of any defect of one of the casks (expected frequency 0 to 1 during the storage period) a Fuel Inspection Hot Cell (FIHC) with a dry storage well is implemented into the B1 facility.

The FIHC allows complete removal of a cask content (182 fuel bundles) into a dry storage well inside the FIHC in a manner, that the presumed defective cask can be fully emptied. After removal of the defective cask, the fuel bundles are transferred inside the FIHC from the storage well into a new cask.

The presentation will explain lessons learned during planning, erection, construction and testing of the FIHC and will consider aspects, like

- Optimisation of location of wall penetrations, i.e. additional shielding vs. easy access for maintenance
- Optimisation of safe access
- Calculated dose rates vs. measured dose rates
- Necessity of close survey during erection / installation to avoid formation of gaps inside of the concrete walls
- Necessity of survey of shielding efficiency

taking into account the purpose and the designed frequency of use of the Hot Cell.

Preliminary Study on the Repair and Transportation Methods of Spent Nuclear Fuel Assembly in KAERI

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The Post Irradiation Examination Facility (PIEF) at the Korea Atomic Energy Research Institute (KAERI) has been performing extensive research on investigating the mechanical integrity and chemical characteristics of spent fuel since 1987. The spent fuels in PIEF have been received from three different nuclear power plant sites, and are stored in PIEF pools and hot cells. The spent fuels include an intact fuel assembly (FA), damaged FAs without a top nozzle in a storage rack, intact and damaged fuel rods in a rod basket, and rod segments/specimens.

Recently, the social and political environment around KAERI has significantly changed because of the increased concerns of radiation in the local community. This is mainly due to the storage of spent fuel and radioactive waste in KAERI. Accordingly, KAERI has decided to tranport all spent fuel back to the nuclear power plants. In this paper, the storage condition of intact and damaged fuel assemblies in PIEF is introduced, and the preliminary study on the repair and transportaion methods of fuel assembly is discussed.

Spent fuel assembly in KAERI-PIEF

Storage condition of spent fuel assembly

- *Intact Fuel Assembly.* The intact FA from KORI unit 1, WH STD 14 x 14 (Figure 70) standard fuel assembly type, has been stored in KAERI-PIEF. The safe handing of intact FA can be achieved without an additional task because of the top nozzle. Therefore, the transport of intact FA using an open can will be adopted if there are no failed or defective fuel rods in the FA. In the case of the presence of failed or defective fuel rods in the FA, the dismantling and repair of the top nozzle is required to extract the failed or defective fuel rods. Dummy rods will be inserted instead of the extracted fuel rods to maintain the mechanical integrity of the FA. The transportation solution for failed or defective fuel rods is out of the scope of this paper.

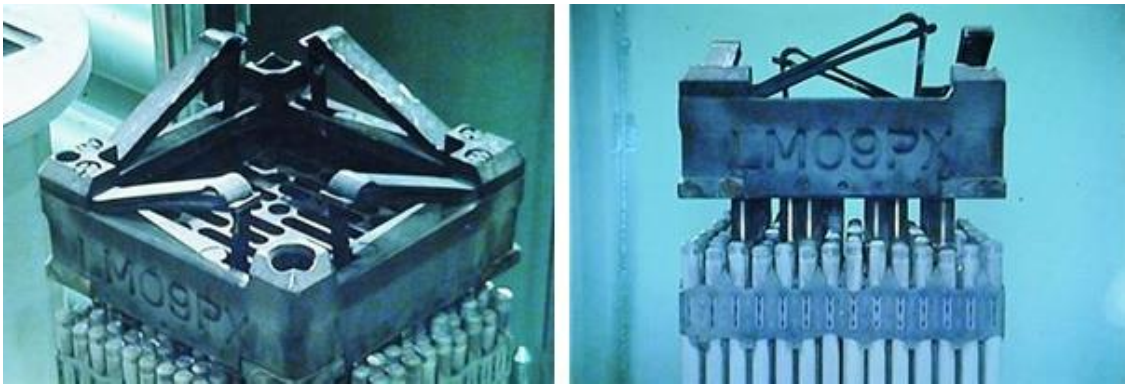

Figure 70: Intact fuel assembly with top nozzle received from KORI unit #1 (Westinghouse, WH STD 14 x 14).

- *Damaged Fuel Assembly without Top Nozzle.* Several damaged FAs without the top nozzle are stored in the PIEF pool. (Figure 71) The top nozzle of the FAs was cut to extract the fuel rods for the purpose of research in PIEF. Therefore, the top nozzle repair for the FA is the first step to achieve the handling of FAs in the pool. When the FA is repaired, the FA is moved from a storage rack to a visual inspection stand to investigate the condition of the fuel rods. At this step, the dismantling of the top nozzle is required to extract the failed or defective fuel rods from the FA. Then, the dummy rods are inserted, and the second top nozzle repair proceeds. To determine an optimum solution for the repair and transportation of damaged FAs, KAERI-PIEF has been performed the preliminary study with Korea Hydro & Nuclear Power (KHNP) and nuclear energy companies.

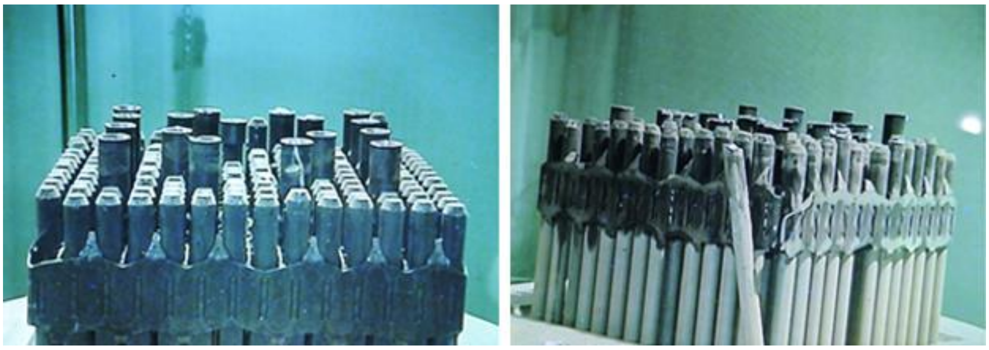

Figure 71: Examples of damaged fuel assemblies without the top nozzle received from KORI unit #1 (Westinghouse, WH STD 14 x 14).

Handling of failed fuel: from reactor to final repository by reconditioning in Studsvik concrete cells

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Failed fuel has been stored in fuel pools at the nuclear power stations in Sweden since start-up of the reactors. Typically, after detection of failure the failed rod is extracted from the bundle and placed in a failed fuel cartridge. The failed fuel cartridge has the size of a fuel bundle and may contain several more or less severe failed rods. Each reactor fuel pool normally has one or several failed fuel cartridges. There are also examples of fuel failures where the failed rod or rods are stuck in the bundle and thus the whole bundle needs to be stored as failed in the fuel pool.

The failed fuel leak radionuclides to the pool water thus causing increased doses but also cause larger amounts of ion exchanger resin and related costs for handling and final disposal. Per Swedish rules, failed fuel shall be removed from the reactor pools without delay.

In the Swedish system for final disposal of spent nuclear fuel, KBS-3 (Birgersson, 2016), the fuel bundles are placed in a dry copper canister, 12 BWR bundles or 4 PWR bundle per copper canister. The copper canisters are placed in tunnels 500 meters down in the solid rock. The filling material around the copper canisters is bentonite clay. KBS-3 will be built near Forsmark nuclear station in mid-east Sweden.

The failed rods contain water and need to be dried before encapsulation for intermediate and final storage in KBS-3. Remaining water in the failed rods may cause further oxidation of the UO2 matrix to unstable oxides such as UO3 and U3O8. Radiolysis of the water may cause leaching of the fuel matrix in the copper canister for final disposal. Radiolysis of water also produce oxygen and oxidants and in presence of air in the copper canister this may produce nitric acid which may damage the copper canister. (About 10% air in argon are expected in the copper canisters after closure). Therefore, SKB has decided that all failed rods shall be dried before encapsulation for final disposal.

Transport from NPP

The failed fuel is transported from the reactor pool to Studsvik's Hot Cell Laboratory (HCL) in mid Sweden. Studsvik arrange the transports using either its NCS-45 cask with BWR bundle dimensions or its 29 ton cask with PWR bundle dimensions. The preferred option is to transport whole fuel cartridges with failed rods but for difficult cases whole bundle skeletons with failed rods stuck in the skeletons are transported. It is also possible to transport failed single rods using a suitable inner basket for the transport cask. In cases when the transport cask certificate does not cover the actual content to be transported, special arrangement for those transports must be applied for to the Swedish Radiation Safety Authority, SSM. The transports are normally by truck on road but for transports under special arrangement an INF 3 vessel is used directly from the NPP harbour to the Studsvik harbour. The sea transport is a compensatory measure in the special arrangement.

Cutting and drying

After unloading of the failed rods in hot cell, Figure72, the end plugs are cut off and the rods are cut into approximately 1 m segments. The 1 m segments then have open ends on both sides which is beneficial to ensure a proper vacuum drying and no enclosed water. After cutting, the rod segments are placed in a capsule for vacuum drying. The vacuum drying of one capsule typically requires 24 – 48 hours.

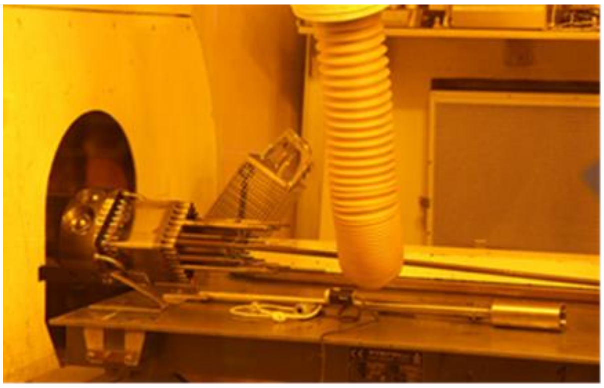

Figure 72: Unloading of failed fuel into hot cell

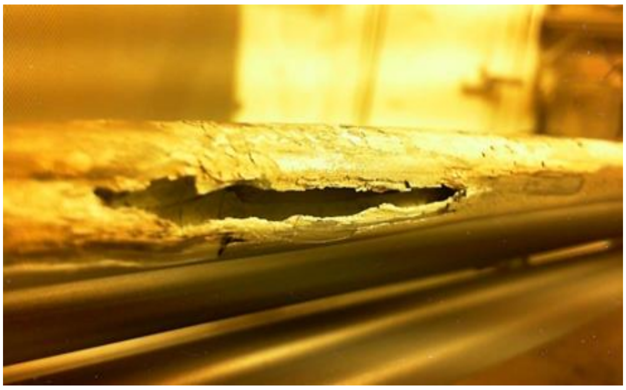

Figure 73: Failed fuel rod

To verify that the capsules are dry, Studsvik performs a pressure rebound test (PRT) according to Standard Guide for Drying Behavior of Spent Nuclear Fuel, ASTM C1553 (2012). At this test the system is closed after vacuum drying and the pressure is monitored for 30 minutes. If the pressure is lower than 4 mbar after 30 minutes the rod segments are considered to be dry.

Encapsulation and final storage

After vacuum drying the inner canister with failed rod segments is placed in a stainless steel primary canister. A lid with a center hole is welded. A final vacuum pumping and PRT is performed and finally the primary canister is filled with He and tight welded. The primary canister is approved by SKB, according to Swedish regulations, for final storage of fuel residuals in KBS-3. The stainless steel primary canister and welding of lid is shown in Figures 74 and 75.

Figure 74: Stainless steel primary canister

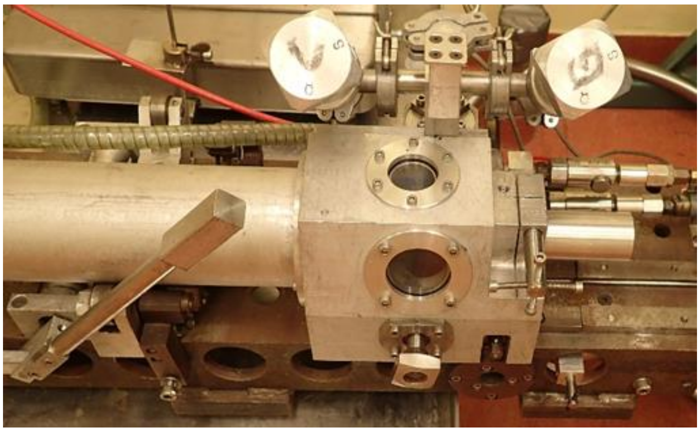

Figure 75: Hot cell welding equipment

After encapsulation 12 primary canisters are assembled in a transport box to the size of one PWR assembly. Alternatively, 3 primary canisters with wider dimensions are assembled to the size of one BWR bundle. Up to 100 failed rods can be packed in one PWR transport box and up to 40 failed rods in one BWR transport box.

The transport box with failed fuel are then transported to CLAB in Oskarshamn. CLAB is the intermediate storage for spent nuclear fuel in Sweden, operated by SKB. Transport boxes with encapsulated fuel residuals has regularly been sent from Studsvik to CLAB the past 30 years and recently also shipments of transport boxes with encapsulated failed fuel have started. Finally, all spent nuclear fuel in Sweden, including fuel residuals and failed fuel will be sent to the KBS-3 final storage.

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Posiva's disposal of spent nuclear fuel - the concept and the encapsulation facility

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Posiva will be the first organization to dispose spent nuclear fuel in its facility in Eurajoki, Finland. The planned schedule will be rather soon, starting in 2020's lasting roughly 100 years. The spent fuel is from Loviisa and Eurajoki, where the former has two VVER-type reactors and the latter two BWR-type reactors and soon also an EPR (PWR)-type reactor. Thus, the current goal is to dispose the spent fuel from five reactors, major part being located already in Eurajoki, whereas the fuel from Loviisa has to be transported to the encapsulation facility.

Here we shortly introduce the concept of the nuclear waste management including the encapsulation process and final disposal. Moreover, we highlight how fuel experiments are related with the concept's phases.

Posiva

Posiva Oy has been established in 1995 and employs roughly 80. Posiva is owned by the Finnish power companies Teollisuuden Voima Oyj (60%) and Fortum Power & Heat Oy (40%). Teollisuuden Voima Oyj has its reactors in Eurajoki (Olkiluoto) and Fortum Power & Heat Oy in Loviisa (Hästholmen). The disposal facility is located in Olkiluoto, Eurajoki, where all the spent nuclear fuel will be disposed. (Please see more from www.Posiva.fi/en. Moreover, all the figures are from the pages.)

The concept - multiple barriers

The concept is based on the KBS-3 method developed in cooperation with Swedish Nuclear Fuel and Waste Management Company, which carries on nuclear waste management in Sweden. More precisely, the design is KBS-3V, meaning that the canisters are placed vertically in individual deposition holes. The disposal concept relies on multiple barriers. The barriers are:

1. Fuel pellet,
2. Fuel rod and fuel assembly,
3. Canister steel insert,
4. Copper canister,
5. Bentonite buffer and tunnel backfill
6. More than 400 meters of bedrock

The idea of the barriers is to create a series of obstacles for radioactive substances penetrating from disposal.

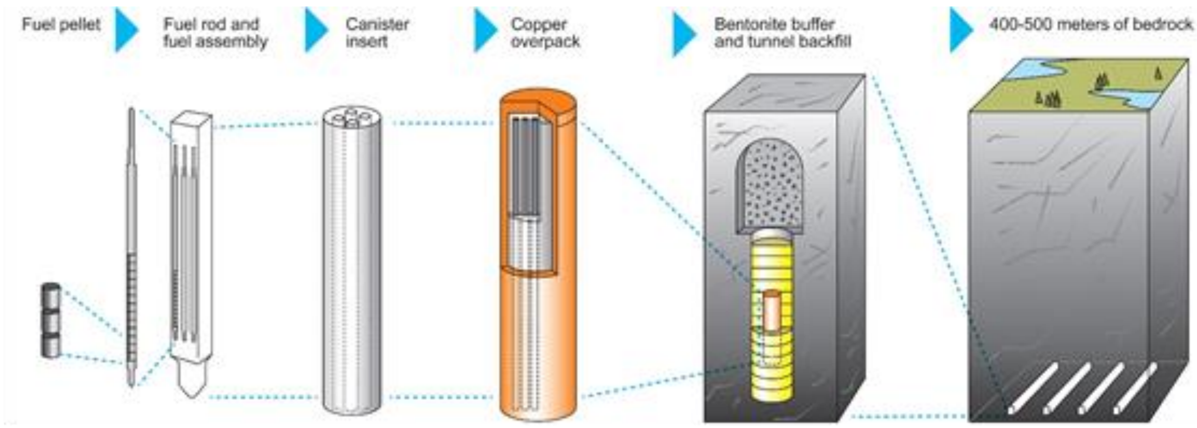

Figure 76: The barriers of the KBS-3 concept implemented by Posiva. With multiple barriers radioactive substances are not easily drifted from the disposal and being harmful in the biosphere.

Long-term storage

The disposal has designed in such a manner that the barriers last over 100 000 years including even ice ages. Posiva has done and is doing a great effort to ensure that the long-term safety is secured for this very long time period.

The encapsulation facility for the encapsulation and the repository

The encapsulation of spent fuel into canisters will be established in the encapsulation plant. Then the canisters will transferred by a lift to the underground repository. The repository's deposition tunnels are located about 400-450 m depth and the final disposal canisters will be placed in those tunnels. The repository can be divided into three parts: 1) deposition tunnels (each canister placed in an individual deposition hole), 2) central tunnels (connecting the deposition tunnels), 3) technical auxiliary facilities.

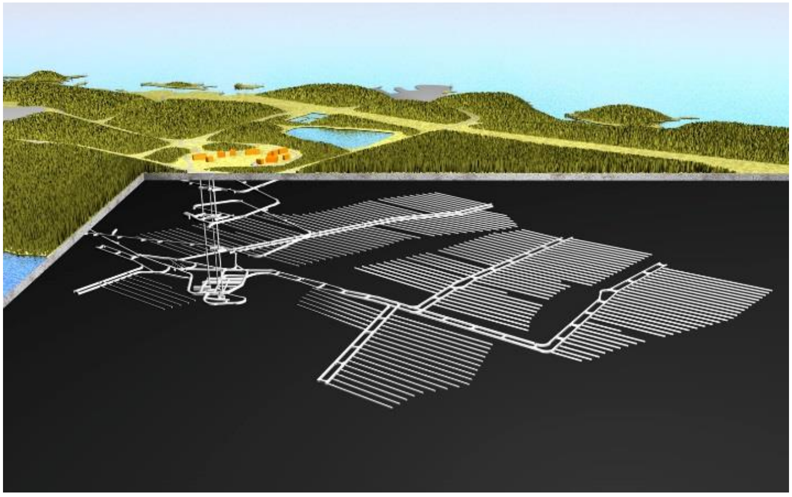

Figure 77: The facility contains deposition tunnels, central tunnels and technical auxiliary facilities.

The spent fuel and the canister

The total amount of the spent fuel is estimated to be 5500 tU which means approximately 2800 final disposal canisters. Moreover, the spent fuel can be classified into three different groups: 1) the fuel from the power plants 1 and 2 in Olkiluoto, 2) the fuel from the power plants 1 and 2 in Loviisa, 3) the fuel from the power plant 3 from Olkiluoto, which will soon be in electric generation. All these types require an individual canister containing its own canister design.

Figure 78: The spent nuclear fuel is packed into the copper-steel canister, where the steel insert has positions for the fuel assemblies and the copper shell protects the fuel from the corrosion.

The spent fuel transport

The spent nuclear fuel is stored currently in the nuclear power plant sites, Eurajoki and Loviisa. The fuel will be transported from the power plants to the encapsulation plant in transport casks.

Schedule

Posiva's plan is to start the disposal sometimes in 2020's and continue for roughly one hundred years. The plan is to dispose all the spent nuclear fuel from Olkiluoto1-3 and Loviisa 1-2 reactors.

Spent fuel studies

Posiva is conducting spent fuel studies to qualify the codes used for nuclear and operational safety, decay heat calculations and to reduce the uncertainties in radionuclide release and transport modelling part of the safety case. Hot cells studies involve radioisotopical measurements of samples from BWR and VVER spent fuel from the Olkiluoto and Loviisa nuclear power plants. Posiva also participates to the EU project DISCO studying the radionuclide releases of radionuclides in disposal conditions

ABSTRACTS FOR

POSTER

PRESENTATIONS

The European Spallation Source Active Cells Facility – Challenges in Construction – HOTLAB 2018

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The Active Cells Facility (ACF) at the European Spallation Source ERIC (ESS) is currently being built in Lund, Sweden. The facility will be a process facility with the main function of processing activated and contaminated metallic waste originating from the neutron spallation process enabling research at ESS. Sub-functions of the ACF are segregation and size reduction of metallic waste, interim storage, receiving and shipping of waste as well as protecting workers from the hazardous environment.

The ACF internal dimensions will be around 30 meters long, 12 meters wide and 15 meters in height and a special feature of the facility is the windowless design, forcing the in-cell viewing system and control system to be highly integrated with the operations of the facility. The baseline throughput of waste is calculated to be around 170 tons/5 year cycle and the most activated components reaching levels up to 3e+09 μSv/hour The ACF static barrier structures is built with high density concrete and the wall thickness is 1.3 meter. The ACF is currently being built as part of the construction of the ESS target station building and at time of writing, the concrete works for the ACF is about half way done.

This paper will concentrate on the challenges of the interfaces between the cast in items enabling the installation of the internal stainless steel liner and other mechanical waste process systems and the civil construction of the ACF superstructure.

Civil construction interface challenges

The civil construction of the ACF (and ESS as a whole), is performed under a contract with a large construction company as well as a range of civil engineering companies providing the detailed design and solutions for the superstructures of the buildings. Based on best available techniques as well as cost and schedule limitations, the construction methods are not always suited for high tolerances and other specific requirements necessary for the ACF cast in items.

In total, there are about 3500 items that will be casted into the concrete, for example; liner beams, confinement penetrations, anchor plates, media penetrations and electrical conduits. All of which have their own requirements on tolerances and unique features. The subparagraphs below are highlighting some of the main challenges that the machine/construction project interfaces has experienced so far.

Mechanical versus civil construction tolerances

Casting in items manufactured in accordance with mechanical tolerances with the expectation that alignment, fitting and interfaces to the later installed machinery in the cell will be in accordance with these mechanical tolerance is not expected to work. Regardless of how well these structures are aligned pre-casting of concrete, the formworks as well as the casting process including a lot of vibration and hydraulic pressure will have an impact on the positions of the cast in items.

ESS has currently deployed a number of mitigations to handle this, one being the use of an internal steel frame in the walls (see Figure P1, middle picture) that, for most of the components, will separate the interface between re-bars and the cast in items. The components are structurally fastened to the steel frame and very fine pre-cast tolerances can be achieved, however, the post-cast tolerances are in some areas observed to be close to the maximum allowable.

Other ways of dealing with tolerances is for example to cast the floor slabs in more than one sequence, the first casting is to nominal level -200 mm allowing floor embedment items to be installed by drilling anchors into the concrete, then pour the final concrete layer up to nominal level of the floor.

The ACF will be cladded with stainless steel plates, to accommodate these an over dimension of the casted in liner beams, which the plates will be welded to, was chosen. This allows for deviations in construction and the stainless steel plate dimensions.

Re-bar design and casting sequences

The amount of re-bars used for the ACF is about 500 kg/m^3, which is about the double amount of re-bars used when building a concrete bridge for example. This as well as how the concrete construction is segmented in the walls (height and length of one single casting sequence) are limiting the way the cast in items can be designed as well as placed. Nominally, anchor plates for example have to be designed with a centre-to-centre distance between the anchor bolts of multiples of 150 mm to accommodate the positions of the re-bars.

No cast in item can be placed longitudinal along a casting joint, which also have forced the design to change in concurrence with progress of the detailed design of the civil works (see Figure P1 right picture).

Uneven concrete surfaces

For both walls and floors, the ideal result from the concrete surfaces, to be able to install the stainless steel liner plates, would be for the surfaces to be completely flush with the liner beams, confinement penetrations and anchor plates. This has, despite the use of spring loaded rods to establish a high pressure between liner beams and concrete form works, been proven difficult. The main reason for this is that the formwork is somewhat flexible (some millimetres) as well as that in between the formworks sections, there are normal wood that when wet, swells differently than the formworks (see Figure P1 right picture).

This has led to that the walls so far have had to be grinded down (only on the hot side of the wall), to be able to fasten the stainless steel liner plates to the liner beams. If the gap is too large, it would not allow the 2 mm thick liner plates to get in contact with the liner beams and welding them together would not have been possible (the way it is designed).

Installation of cast in item in tight spaces

In many positions, the cast in items location and size are either difficult for the civil works to manage (see Figure P2 – left picture), or the installation of the cast in items are very difficult due to tight spaces (often due to that re-bar installation has already been done). This requires either good interaction between mechanical and civil construction to minimize surprises on site, or that as large items as possible are pre-manufactured prior to installation (this can also save time on site). One example is the large pre-manufactured storage pits, see Figure P2 – right picture, that were transported to site and lifted directly into the ACF prior to casting.

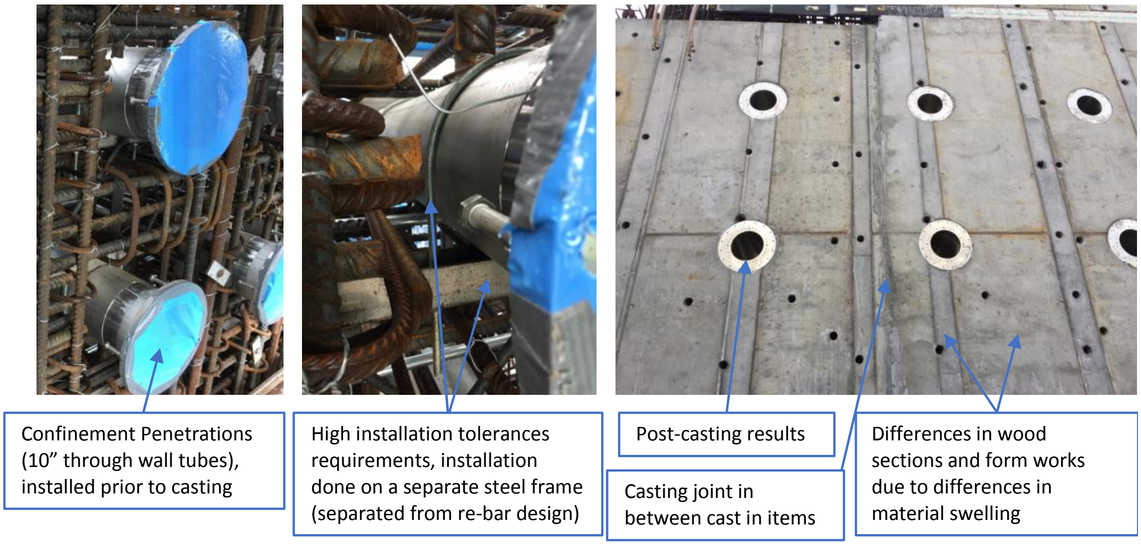

Figure P1: Confinement penetrations, pre- and post-concrete casting[1]

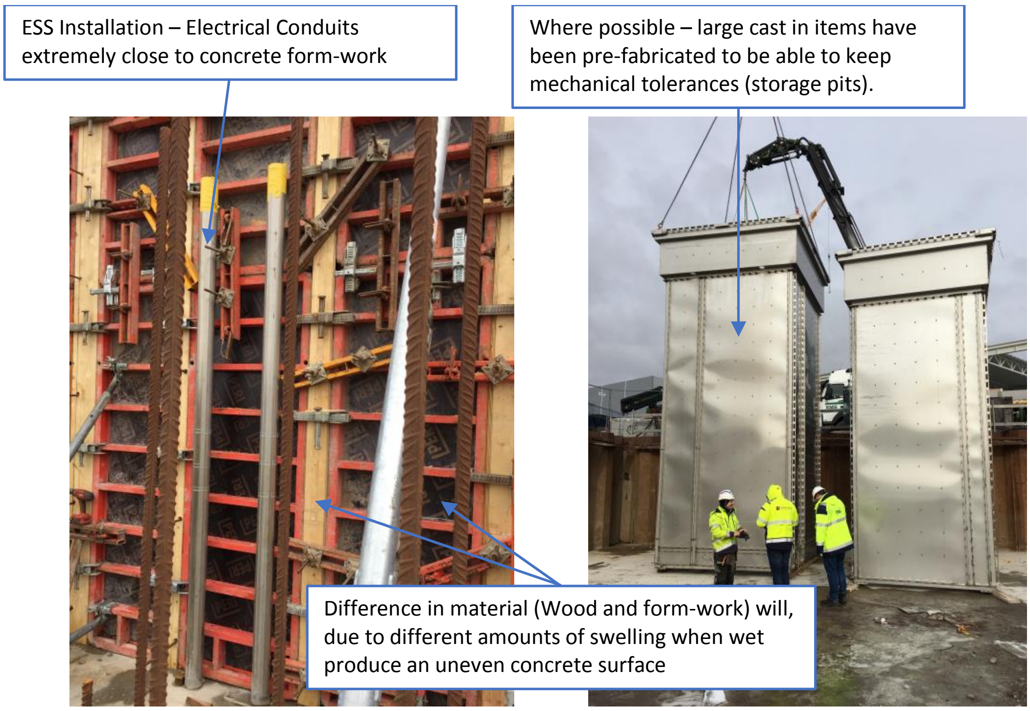

Figure P2: Challenges and solutions of some of the interfaces with the civil works of the ACF[1].

[1] Pictures from site taken by Magnus Göhran, ESS.

Size Reduction Equipment in the ESS Active Cells Facility Abstract for Poster Presentation at HOTLAB 2018

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The European Spallation Source (ESS) is a multi-disciplinary research facility located in Lund, Sweden. Currently under construction, this facility will become the world's most powerful neutron source.

Because of the vast amounts of high-energy neutrons produced, many of the components within the ESS Target Station will become highly radioactive. These components weigh up to 14.5 tonnes, and are 6m in length, removing the possibility of placing them straight into a storage facility. A novel approach to remote size reduction and handling operations is needed to safely cut up, package, and store these activated components, maintaining maximum operation time of the ESS facility.

The ESS Active Cells Facility (ACF) is a windowless hot cell facility designed to reduce the size of the irradiated material produced by ESS, so it can be transported from the facility and stored. As part of the UK's in-kind contribution to the project, RACE have been contracted to deliver the ACF to the ESS project in collaboration with industry.

Size Reduction Equipment Selection

ESS and UKAEA followed a systems engineering approach to define requirements for the facility, including a full list of components to be disposed of; radiation dose rates from these components; defined waste container sizes; and activity and mass limits within the containers. Sub-systems were derived to fulfil these requirements; such as Size Reduction, Handling, Confinement and Shielding, Control, Power and Signalling, and more.

Within the Size Reduction work package, a cutting plan was developed to define component cut locations. One of the driving requirements was to minimise the production of contaminated dust and swarf, so the number of cuts required was minimised. The cutting plan was developed using the following methodology:

A. Identify all the components that will meet the shielded container limits with no size reduction.
B. Those that will not fit within a shielded container, specify cut locations so that they would.
C. Group cuts into similar cut types.
D. Iterate between b & c to reduce the number of cut types required.

This process refined the Size Reduction equipment into five categories; shaft cutting, precision cutting, pipe cutting, lip-weld cutting and bolt removal. With the locations of the cuts defined, packing allocations were presented, and surface dose rates were calculated. This ensured that all components fit within the shielded containers and did not exceed the mass and surface dose limits.

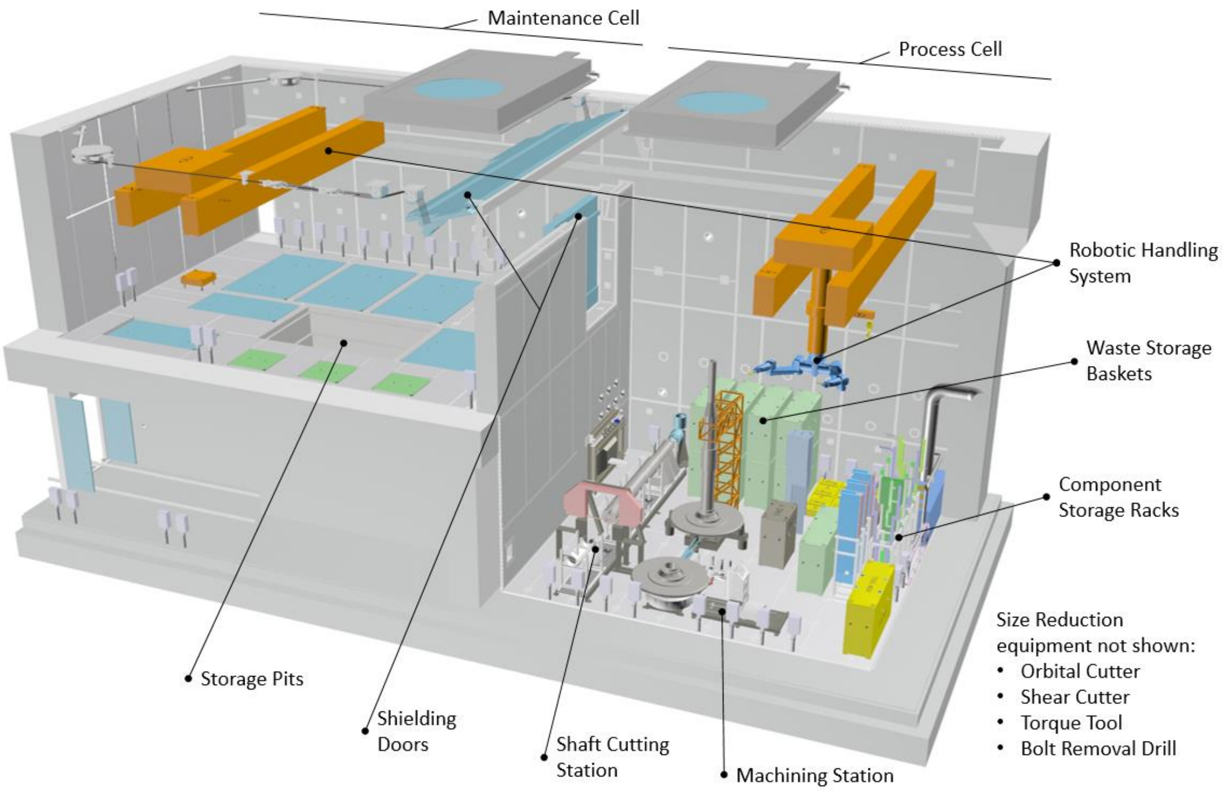

Figure P3: Cross-section of the ESS Active Cells Facility

Size Reduction Equipment Verification

UKAEA identified potential technologies for each of the cut types and concepts were selected on a set of criteria including contamination control, remote maintainability, swarf production and more. Concept studies were carried out on the selected technologies to gain confidence that the system requirements could be met, and cutting trials were performed on representative cut samples to address three main concerns: 1. dry-cutting, as fluid coolants are prohibited in the cell, 2. the most challenging cuts were possible with available technology, 3. dust levels produced were not excessive. Other physical trials were also performed, such as the testing of a remote bandsaw blade changing rig. These were successfully completed for the major Size Reduction equipment categories, shaft cutting and precision cutting, providing high confidence in the candidate concepts.

The equipment will now be developed, manufactured, installed and verified in collaboration with industry.

Size Reduction Equipment Summary

Large activated components within the ESS facility shall be produced on a regular basis, needing size reduction and safe storage. The ESS Active Cells Facility shall be the hot cell facility to remotely handle, size reduce, package, and store these components.

The size reduction cutting equipment can be grouped into five categories: shaft cutting, precision cutting, pipe cutting, lip-weld cutting and bolt removal.

The two major categories being the shaft and precision cutting stations were validated through physical cutting trials and theoretical analysis to provide confidence in the system design.

UKAEA are seeking industrial collaborators to deliver the systems to ESS.

PLATOM’s Expertise and Capabilities to Support Construction and Operation of Hot Cell Facilities

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Platom is a consulting company with more than 30 years’ experience in providing nuclear industry with services covering, e.g., radioactive waste treatment management, equipment and process system deliveries, modelling and simulation of NPP processes and control systems as well as design and delivery of I&C systems.

Platom has designed, supplied and commissioned evaporation and waste treatment box to be used in Hot Cell applications. In addition, we have practical experience of Hot Cell laboratory work with several techniques, e.g., crack growth rate testing of irradiated stainless steels, mechanical testing of irradiated reactor pressure vessel materials and Electro-Discharge Machining (EDM) of irradiated materials.

Design, Manufacturing and Commissioning of Evaporation and Waste Treatment Box for Hot Cell Laboratory

Design Requirements. The safety requirements for the equipment are set in Regulatory Guide ST 6.1, “*Radiation Safety when using unsealed sources*”. In addition, the requirements in the Government Decree 400/2008 on the Safety of Machines were considered in the design process.

Manufacturing. The above-mentioned Regulatory Guide 6.1 and Government Decree 400/2008 were also the main guiding documents during the manufacturing of the evaporation and waste treatment box. All the surfaces which are subjected to radioactivity, are made of materials which are easy to decontaminate. All the sub-suppliers were approved by Platom before starting the manufacturing of the equipment.

The structure of the evaporation and waste treatment box. The main parts of the evaporation and waste treatment box are:

- radiation-shielded drum
- evaporation and waste treatment box
- drying equipment

The picture of the box during assembly work is presented in Figure P4.

Figure P4: The evaporation and waste treatment box during assembly work.

The main components of the evaporation and waste treatment box are presented in Figure P5. The operating principle is shortly described in Ch. 3.

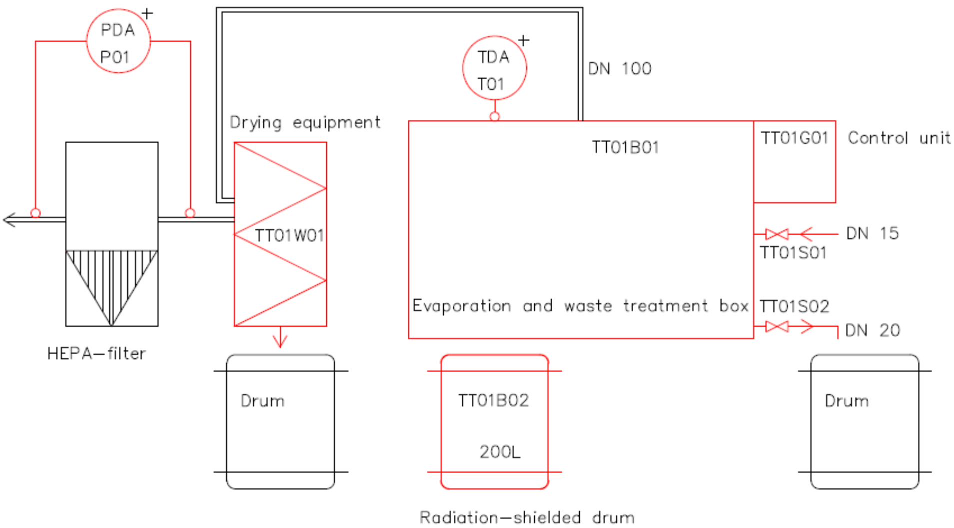

Figure P5: The main components of the evaporation and waste treatment box.

Operating principles of the evaporation and waste treatment box

Liquid radioactive waste will be transported in 200 l drums, with or without radiation shield, depending on the dose rate. The drum is positioned underneath the box, and external heaters are connected around the drum. The drum is lifted so, that it can be tightly connected to the box. The progress of the evaporation process can be followed with mirrors. When the waste is dry, the drum will be detached, covered and transferred to the storage or to the customer.

If the dried waste needs to be solidified, this can be done in the mixer placed in the box. The recipes are determined case-by-case, and different types of sorbents can be utilised.

Different types of wastes, including ion exchange resins, solvents and oils, can be solidified with this equipment.

The preliminary study for safety design of JAEA's Radioactive Material Analysis and Research Facility "Laboratory-2" dedicated to fuel debris analysis at TEPCO's Fukushima Daiichi Nuclear Power Station site

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According to "Mid-and-Long-Term Roadmap towards the Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station" (Roadmap) established by the Inter-Ministerial Council for Contaminated Water and Decommissioning Issues, prospects of a processing/disposal method and technology related to the safety of radioactive material should be made clear by around FY2021. Regarding the fuel debris, it is noted that a processing and disposal method of fuel debris will be decided after the start of retrieving. The analysis and research facility to receive vision for processing and disposal method of fuel debris is now being designed.

The Japan Atomic Energy Agency (JAEA) is now constructing Radioactive Material Analysis and Research Facility. It is built near the Tokyo Electric Power Company Holdings, Incorporated (TEPCO) Fukushima Daiichi Nuclear Power Station (1F) site in order to perform and collaborate with TEPCO's activities in 1F. The facility consists of three buildings: the administration building, Laboratory-1 and Laboratory-2 shown in Figure P6.

The administration building started operation in March 2018 provides office space, meeting rooms for researchers, and apparatus mock-up. Laboratory-1 is now under construction for radioactive analysis of low and medium level radioactive rubbles and secondary wastes. Laboratory-2 is planned as dedicated hot laboratory for radioactive analysis, and mechanical and chemical characterization of fuel debris. Therefore, some specific issues, such as shielding and criticality safety should be investigated as preliminary studies.

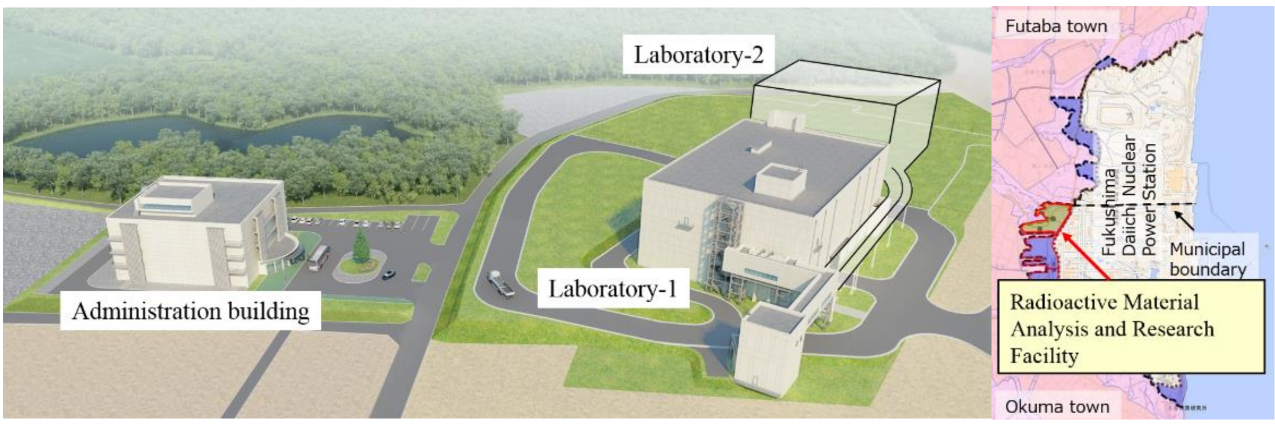

Figure P6: Completion image (left) and location (right) of Radioactive Material Analysis and Research Facility

Description of Laboratory-2

Analytical items and Fuel debris treatment process. In order to characterize chemical and mechanical property of fuel debris in Laboratory-2, candidates of analytical items are shown in Table P1. The amount of fuel debris transported using conventional fuel cask from the 1F site is scheduled to be 12 times/year and set less than 60 kg/year (less than 5 kg per each transportation). The analysis flow of fuel debris is shown in Figure P7. After pre-treatment such as cutting, polishing and dissolution performed in concrete cells, samples (processed fuel debris) are transferred to glove boxes after chemical separation in steel cells. The samples are analyzed in glove boxes using several kinds of equipment. After analysis, samples including not analysed (residual) fuel debris are stored in the concrete cell pit, and then will be returned to the 1F facility.

Table P1. The expected analytical items in Laboratory-2

	Laboratory -2 (TBD)
Main Analysis items	Radioactivity Elemental analysis Organic matter Surface analysis Chemical analysis Hydrogen gas Mechanical characteristics Specific surface area / Particle size distribution Density Thermal property Calorific/Heating value Others (High-temp. properties, etc.)
Main Preparation	Cutting machine Polishing machine Electric discharge machine
Main Pre-treatment	Alkali dissolution units Acid dissolution units
Others	X-ray computed tomography scanner Pneumatic carrier system

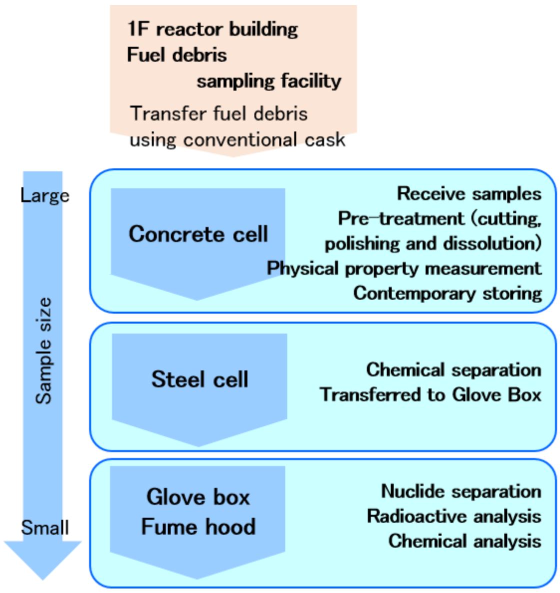

Figure P7: The analysis flow of fuel debris

Preliminary evaluation of safety design condition

The difference between Laboratory-2 and other hot laboratories is the fact that there is no reliable information on fuel debris to use as safety design conditions. For Three Mile Island (TMI) accident and Chernobyl accident, there have been many reports on the characterization of fuel debris. On the other hand, different factors, such as fuel composition and fuel type (MOX fuel) should be considered in the 1F reactors. Thus, it is necessary to consider the existence of fuel debris as various compositions depending on the location as is shown in Figure P8. Therefore, practically conservative assumption is requested for the safety design condition of Laboratory-2, especially concerning the radiation shielding and criticality safety parameters.

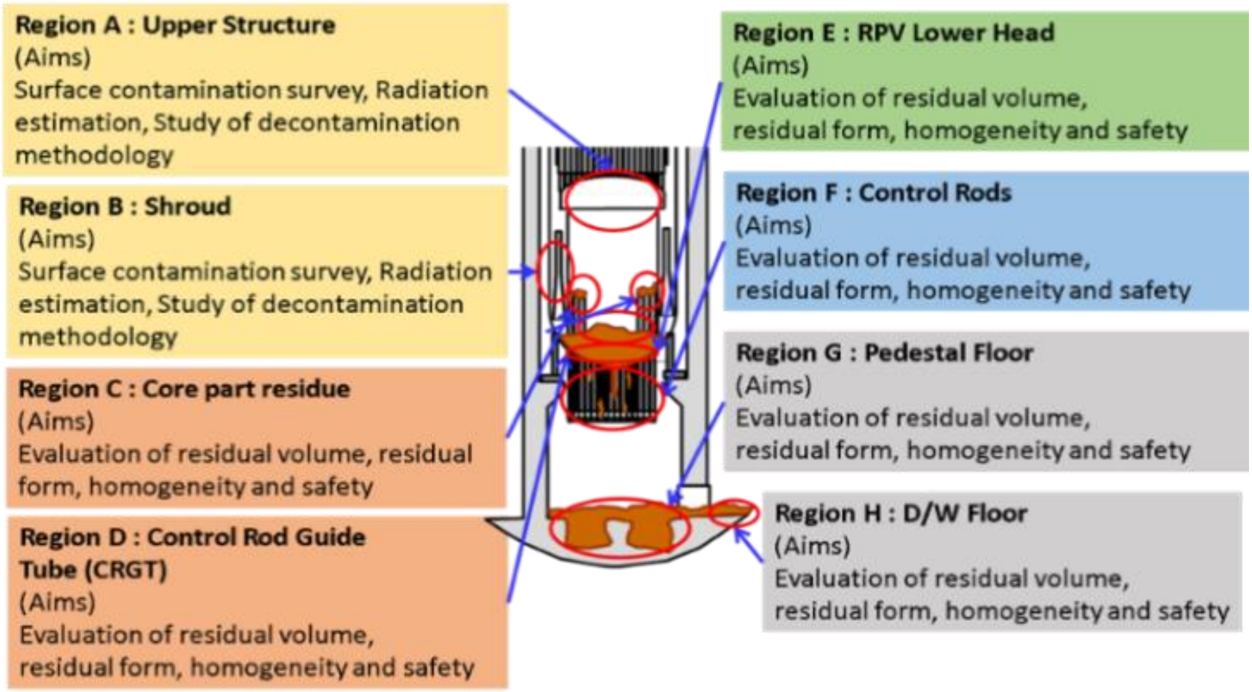

Figure P8: Image of 1F core

For evaluations of radiation shielding to the public at site boundary and workers in Laboratory-2, dose rates have to be evaluated. As a first step, it is necessary to estimate the source intensity for shielding evaluation by calculation. The calculation of source intensity was performed using estimated conservative parameters based on nominal data of reactor core before the 1F accident. Burn-up, uranium enrichment and nuclide composition of MOX fuel were adopted as parameters. The effect of activation of structural material was also included. As shielding calculation system under conservative condition, the radiation source was set as void to exclude self-shielding, because there are no data on parameters of fuel debris that affect the self-shielding such as density and composition.

As a preliminary evaluation of criticality safety, the critical mass limit was evaluated under conservative conditions such as fresh fuel and 30 cm water reflection, considering both uranium fuel and MOX fuel.

Knowledge from results / Summary

Through preliminary calculations, it is found that shielding design of Laboratory-2 is achievable even if calculations are performed under conservative conditions. Through preliminary calculation of critical mass limit, handling of 5 kg fuel debris is concluded to be feasible, considering both uranium fuel and MOX fuel.

As a result of the preliminary evaluation by using conservative condition and nominal 1F fuel data, it is implied that the condition as applied to post irradiation examination facility of irradiated fuel can be adopted to start safety design for our Laboratory-2.

Preparation of experiments at CVR Hot-cell

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Paper presents the preparation for new experiments and first measurements done in hot-cell facility at CVR (Research centre Rez).

The hot cell facility were built within the project SUSEN (Sustainable Energy) at CVR (Research centre Rez). The project uses existing building converted for the purpose of placement of new hot cells. Within this project a new complex of 10 hot cells and one semi-hot cell. Our facility allows work with radioactive samples with activity up to 300 TBq ^{60}Co and with dimension of 2 CT.

Presentation is focused on preparation of new method of swelling measurement (dimension change) of small specimens, cladding testing and first results on active materials.

Using the replicas of surface of highly irradiated specimen with modern 3D scanning device and computer 3D modeling we can replicated the specimen shape. With measuring before and after irradiation even very small changes in dimension (volume or surface) can be described. Also with some 3D modeling surface damage can be excluded from the measuring.

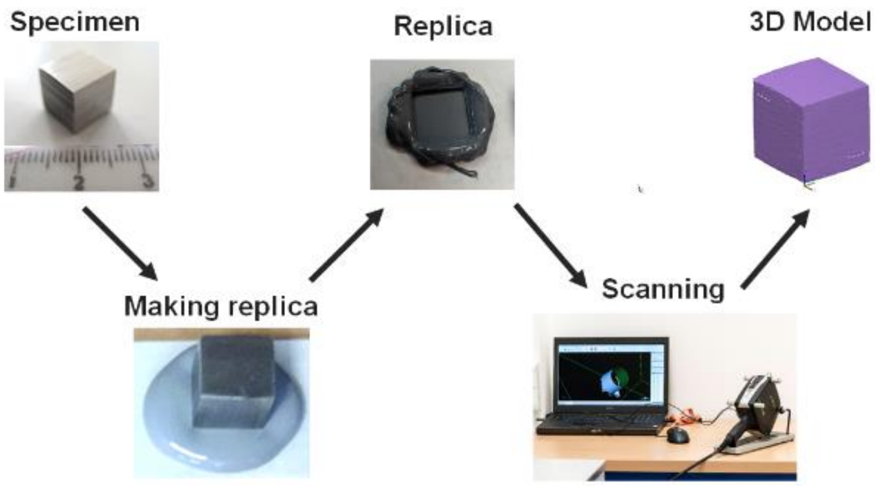

Figure P9: Schematic procedure.

Two type of specimen holder were design for two shapes of cladding specimen for tensile and creep test. During preparation work 3D printer was used for quick production of prototype holder during the evolution of construction design. This process allowed to shorten the time for preparation of infrastructure for new test and specimen shapes.

First SEM and nanoindentation results will be presented. The SEM is equipped with EBSD and WDS sensors and nanoindentation device is equipped with heating table and multi-indent head.

An overview of the Remote Handling solutions and equipment at JRC Karlsruhe's Hot Cells facilities

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At the Joint Research Centre’s (JRC) Karlsruhe site, the Waste Management unit has under its mandate the operation of the entire hot cell suite, which constitutes the hub at JRC for (i) PIE of conventional and advanced fuel and (ii) the R&D on the back end of the fuel cycle. Therein, the 'Alpha-Gamma' intervention team within the unit develops, adapts and operates various Remote Handling technologies with the highest care with respect to contamination monitoring and radioprotection regulations. In total we operate and maintain about 90 Master Slave Manipulators (MSM) of various sizes and types together with several Robot arms for heavy duty or complex operations. These tasks range from operation of scientific instruments to the maintenance of large process equipment, with all operations requiring dexterous manipulations. Most of them are unstructured and require real-time human intervention.

This presentation shall give an overview of our range of articulated and telescopic manipulators with various load capacities and size, both mechanical and electrically assisted models, for work in the different types of hot cells, together with some developments for innovative solutions for manipulators, booting and grippers

Self-threading electrical discharge machine – hot cell modifications and the first year of active machining

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Electrical discharge machining (EDM) is very useful when fabricating metallic samples that requires tight dimensional tolerances, e.g. samples for mechanical tests such as tensile testing, ring compression testing and crack propagation testing. If the samples are radioactive with high dose rates, like samples from reactor vessels or fuel cladding, the operator of the EDM must be shielded from the machine i.e. the EDM must be put in a shielding cell.

The old EDM at Studsvik (Sodick A280L) had reach its end-of-life after more than 20 years of active machining. At the end the cutting wire was broken repeatedly and on this machine the wire had to be threaded manually requiring man entry into the cell. The repeated man entry yielded unnecessary high radiation doses to the personal. So, Studsvik decided to replace the machine with a new self-threading EDM.

Requirements of the new EDM

The new EDM had to fulfil the following requirements:

- It needed to have the electronic control system separate from the machining system, so the control system can be placed outside of the hot cell.
- It needed to fit the existing hot cell without too much modification of the hot cell.
- It needed to be able to automatically thread new wire to avoid unnecessary man entries in the hot cell.
- The EDM must be fitted with an in-cell filter system to remove solid debris from the working fluid.

The choice of fell on FANUC ROBOCUT α-C400iA a modern self-threading EDM with separate control, see Figure P9.

Hot cell modifications of the EDM

A new filtered EDM fluid filter system was designed and fabricated. It compromises a coarse and a fine filter, see Figure P10. The filtering system is located within the hot cell and the filters can be exchanged by use of manipulators. The fluid is stored in an in-cell tank. Before the fluid is released to the on-site active sewage system the fluid is filtered by the in-cell system.

All electrical wires connecting the control system to the EDM system was extended and some integrated electronics were moved out of the EDM-system, see Figure P10. Also, since the EDM is placed inside a hot cell some unnecessary operator protections was removed. These changes to the EDM spurred several warnings and errors in the control system but it was all sorted out by close co-operation with FANUC technicians.

The hot cell was cleaned out and repainted. A new 7-inch-thick lead door was installed so that the new and higher EDM could be transferred into the cell, see Figure P11. The cell is divided into a sample mounting section and an EDM section, each operated by two simple manipulators. On the roof of the hot cell a long manipulator is situated. It is used to pick up object that is unreachable with the other manipulators. To help the operator to see all corners of the cell it has been equipped with a IP-camera and a portable waterproof web-camera.

Figure P9: The FANUC ROBOCUT α-C400iA has controls (right) separated from the EDM system (left).

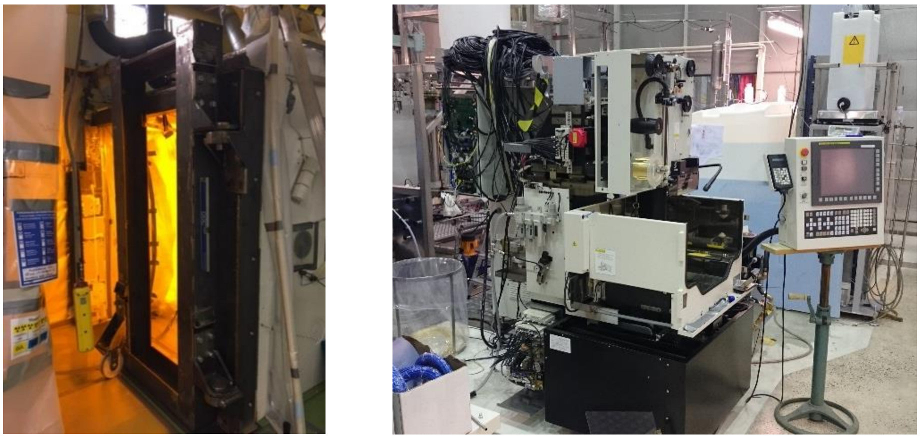

Figure P10: In-cell filters for EDM-fluid (left) and EDM under modification (right)

Figure P11: A new 7-inch door was fitted to the hot cell (left). Operator control panel of the installed EDM (center). A steerable in-cell IP-camera helps the operator to see all corners of the cell.

First year of active machining

The new EDM has been producing active samples for mechanical testing for about a year now. The self-threading feature of the machine has significantly reduced the dose to the operators since man-entry for wire threading is not needed any more. The quality of the machined samples is as good or even better than with the old machine. One lesson learned is that modern technology (especially software) can frighten old operators. Over the first year no radiation related malfunction of the EDM have occurred.

A remote technique for a preparation of tension test specimens from the irradiated round bars

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In Ukraine the fifteen WWER nuclear power plant units have being operated, two of which are WWER-440 units located at Rivne NPP site. In 2010 a thermal annealing of the RPV weld 4 for the Rivne NPP unit 1 was performed. The thermal annealing was a key condition in the procedure of a license renewal for long term operation of this reactor pressure vessel. A new surveillance program for a material science support of RPV safe operation after the thermal annealing was developed according to the requirements of a national regulatory body.

The round tension specimens have been provided for an estimation of the changes in mechanical properties of RPV metal due to a re-irradiation. However, the round bars have been put in the surveillance capsules instead of standard round tension specimens. So, there was necessity to prepare the tension specimens from the round bars. The paper presents the technical details about an application of the new technique for a preparation of tension specimens from the irradiated round bars.

Brief description of a remote equipment

For machining the round specimens from irradiated bars, a new remote lathe with a computer control LEON-01 has been developed (Figure P12). The new equipment is placed in the special zone of a hot cell laboratory designed for handling the irradiated material. The CNC control electronics runs the step motors of the machine. The CNC control unit is the interface between a PC software Mach 3 with a setup under Windows 10 and the lathe mechanics. The software provides the machine with the geometry information for a travel of the cutter considering a small deflection of the bar during a machining process.

Two step motors provide precise travel of the tool in two mutual perpendicular directions. There is a special holder to quickly put the round bar in the collet of the lathe. The three-end cutter is applied as the machining tool. A temperature of the specimen must be less than 100°C during the machining process, so a cooling system has been provided in the lathe design.

The lathe is operated with a power supply of 380 V. The rotation speed of the spindle was selected to be 3000 rpm and the travel speed of the cutter was 30 mm / min to ensure the necessary quality of the specimen surface. A material layer of 0.05 mm thick is removed for each pass of the tool.

Specimen preparation

The reference specimens have been prepared to check an effectiveness of the new equipment. An archive metal of WWER-1000 RPV support shell has been used in this study. Firstly, round bars with a length of 26 mm and a diameter of 6 mm were cut from halves of the tested Charpy specimens using an electric discharge machine (figure P13a). After that the tension specimens were machined from the round bars using the lathe with remote control. The measurements have shown that the dimensions of a reduced section meet the DSTU EN 10002-1:2006 standard requirements.

A view of the machined specimen with a diameter of 3 mm and the gage length of five times the nominal diameter is shown in Figure P13b.

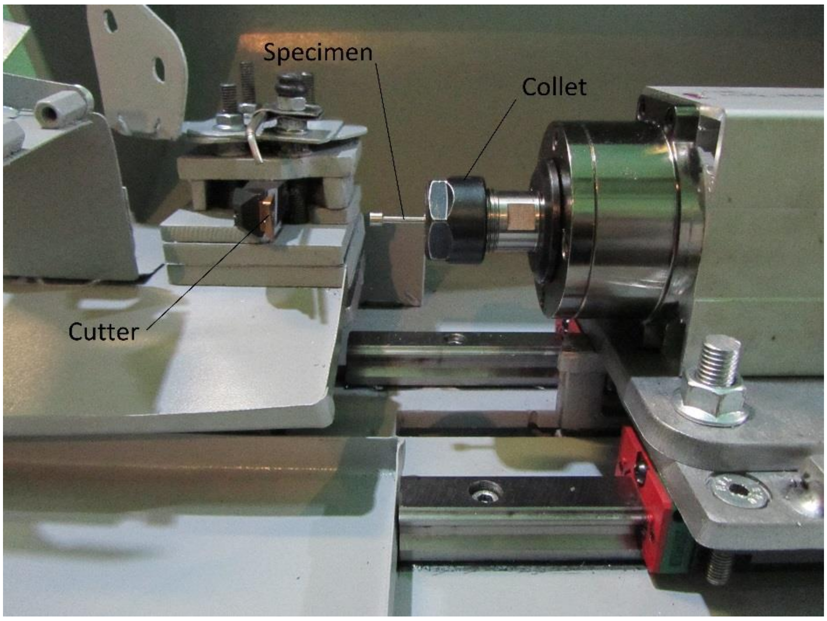

Figure P12: Working area of the lathe with remote control


a)


b)

Figure P13**:** A round bar cut from a half of the tested Charpy specimen (a), a tension specimen machined from the round bar (b)

Finally, the four remotely machined specimens have been tested at room temperature. An analysis of tension test data has shown that a scatter of strength and plasticity parameters which is characterized by standard deviation (3 MPa and 0,5 % respectively) is small enough, therefore, the new equipment allow us manufacturing the round specimens and getting acceptable test results. A location of fracture within the specimen gauge length also indicates the validity of procedure for the specimen preparation.

Conclusion

The report provides information on the development of remote technology for manufacturing tensile specimens from irradiated round bars. The operation principle and components of the lathe with remote control are given. A brief description of machining process is presented. The tension test data has shown that new equipment can be successfully applied for manufacturing the tensile specimens.

Using Novel Small Scale Mechanical Testing to Link the Mechanical Properties and Deformation Mechanisms of High-Dose Activated Inconel X-750

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As fuel-assembly hold-down springs, control guide-tube support pins, jet pump beams, and core internal bolts within light water reactors and fuel channel annulus spacers in modern CANDU reactors. Radiation effects on Ni-based alloys are highly dependent on the neutron energy spectrum and temperature in the reactor core with which they operate. The high-dose material presented in this study ($T_{irr\ avg}$ = 180 ºC and $T_{irr\ avg}$ = 300 ºC) functioned in a high thermal-neutron flux spectrum where (n,α), (n,p), and (n,γ) transmutation reactions accounted for the bulk of the total displacement damage. This damage, plus contributions from fast-neutron direct collision cascades, causes the material to incur dose rates between 2 and 4 dpa/yr in service. In addition, the transmutation reactions generate high internal helium and hydrogen gas contents, > 2.6 at% He and > 0.5 at% H (Judge et al., 2013), in the highest dose Inconel X-750. The high helium content has been linked to helium grain-boundary embrittlement in the bulk component that leads to intergranular failure as shown in Figure P14 (Judge et al., 2015).

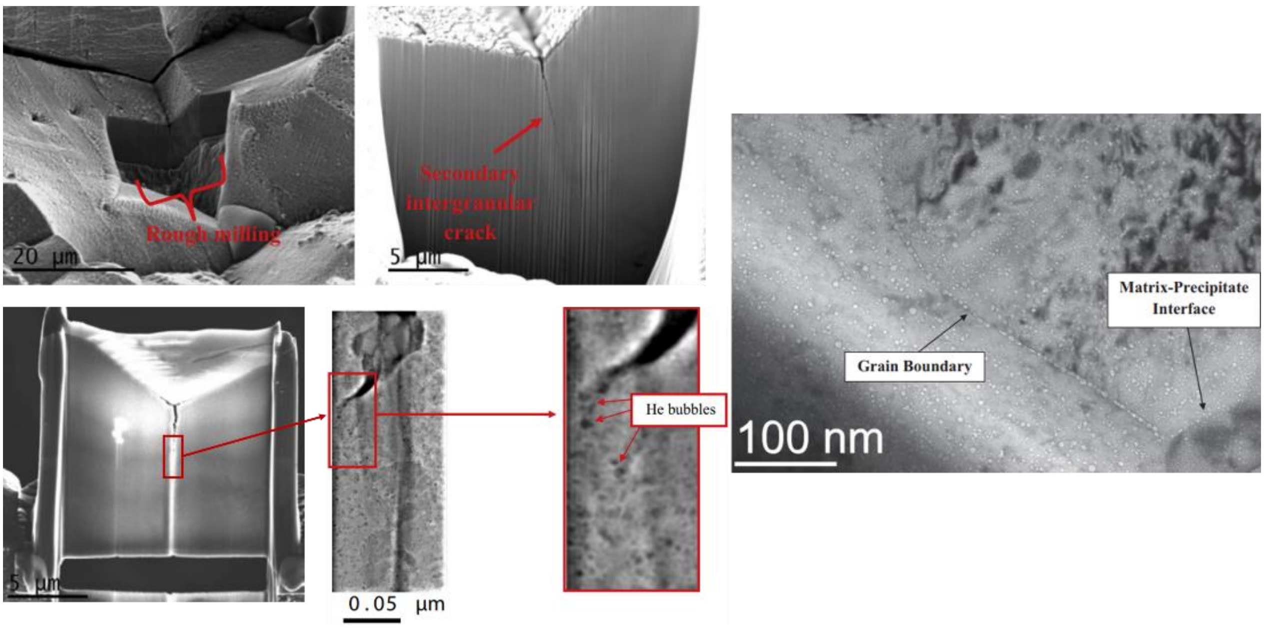

Figure P14: Direct evidence of helium grain boundary embrittlement linked to intergranular failure in high dose (53 dpa, 1.8 at% He) Inconel X-750 (Judge et al., 2015).

Previous post-irradiation experiments utilizing novel, micro-three-point bend testing quantified the flexural yield strengths of material irradiated to 53 dpa and 67 dpa at both 180 °C and 300 °C as shown in Figure P15, taken from (Howard et al., 2015; 2018). Differences in the yielding behavior of the high dose material at the two irradiation temperatures can be attributed to differing helium bubble hardening mechanisms. High temperature material with average bubble sizes > 2 nm forces dislocations to change course more than in low temperature material with average bubble sizes ~1 nm. This is reflected by large yield-strength increases in high-temperature material, whereas there is little difference in the yield strengths of non-irradiated, 53 dpa, and 67 dpa material irradiated at 180 °C.

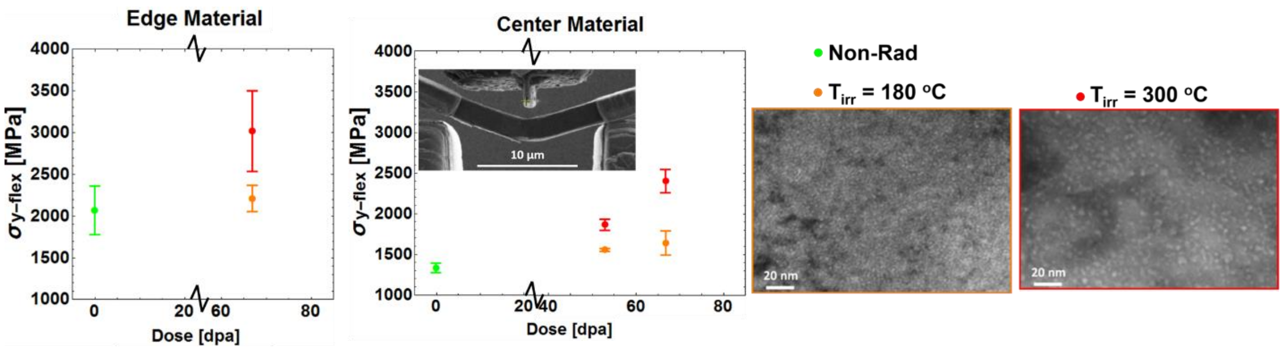

Figure P15: Flexural yield strengths of high dose, Inconel X-750, micro-three-point bend specimens (Howard et al., 2015; 2018 and their associated matrix helium bubble microstructures (Judge et al., 2015).

Assessing the mechanical properties beyond yielding via this testing technique was rather difficult because the specimens could not be tested to ultimate failure. There were a few indications of cracks along grain boundaries in specimens irradiated to 67 dpa after they yielded via deformation slip bands seen in SEM test videos, but more precise, direct evidence of failure strengths, degradation mechanisms, and ductility parameterization was still desired.

New processes developed at CNL utilizing the Fuel and Materials Cells (FMC) in combination with an active FEI Versa Focused Ion Beam (FIB) workstation and post-test TEM analysis led to the development and refinement of a second, novel, push-to-pull, micro-tensile, post-irradiation examination small scale mechanical test. Sample preparation techniques target specific boundaries of interest by coupling EBSD analysis with nanomanipulator lift-outs of active volumes of material (~1.25 μm x ~1 μm x ~2.5 μm). Micro-tensile testing of high-dose material irradiated to 80-85 dpa provides strong evidence for a mixed-mode failure mechanism involving both dislocation slip bands/channels which develop in the grain interior and grain boundary failure. Grain interior deformation shears and elongates helium bubbles such that they coalesce, leading to channel fracture (Figure P16a). Grain-boundary inclusions have also been shown to initiate cracks within the grain boundary, fracturing the boundary (Figure P16b). Micro-tensile testing also quantifies the mechanical properties of high-dose Inconel X-750 in terms of its critical resolved shear stress, yield strength, failure strength, and total elongation.

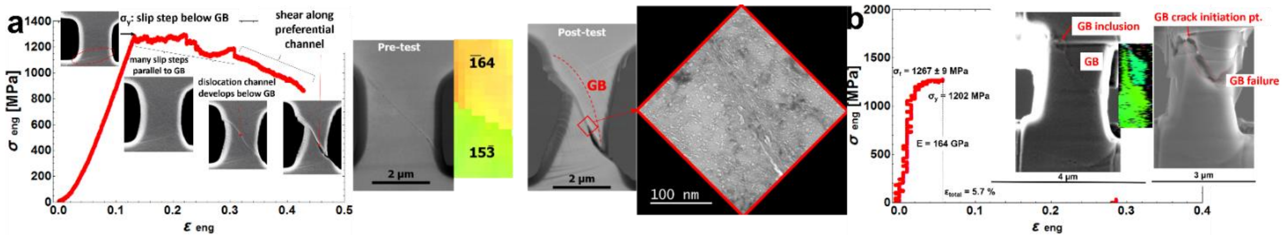

Figure P16: Micro-tensile specimens exhibiting (a) channel fracture and (b) intergranular fracture

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High Energy X-ray Study on Nondestructive Detection of Fuel Assemblies

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Nuclear fuel assembly is the core component in reactor, the characteristics of assembly is directly related to the safe operation of nuclear reactors and the effective use of nuclear fuel components, During the running of reactor, fuel pellets will be swelling and cracking under the condition of solid and gas fission products, thermal stress, neutron irradiation etc, and will be breaking in severe cases, which will directly threaten the safety of nuclear facilities. Therefore, periodic testing of nuclear fuel assembly is particularly important, irradiated nuclear fuel components have strong radioactivity, and there are some technical difficulties for the detection of nuclear fuel assembly, according to investigation and related research ,nondestructive detection technology research of High energy X-ray is a kind of efficient means of detection.

China institute of atomic energy has performed the technique research for several years,we have designed detector system, precision machinery system,motion control system,imaging system, simulation of nuclear fuel assembly etc., a platform of nuclear fuel components detection system has been set up,and has realized tomography imaging acquisition of simulated fuel assembly.now we are performing further research of detection system optimization ,data interference reduction algorithm under strong radiation environment,effective method study of improving the quality of imaging. All of these will lay a solid foundation for the detection of nuclear fuel assembly in hot cell.

System composition and structure

The detection system of nuclear fuel component consists of multiple subsystems, each sub detection system working normally and coordinating orderly is the premise of effective detection , the subsystem is mainly composed of high energy X-ray source, detection system, high precision mechanical movement and the control system, imaging detection system, etc.The system structure diagram is shown in Figure P17.

Results

As we know, irradicated nuclear fuel components have strong radioactive ,In order to reduce strong radiation interference and realize effective detection, high energy linear accelerator ray source is very necessary. we use 9 Mev hign energy X- ray source in this system, high-energy accelerator mainly include X-ray head, water cooling system, modulator, control system, etc.The research group conducted exploration and research on simulated nuclear fuel assemblies. The detection results are shown in Figure P18.

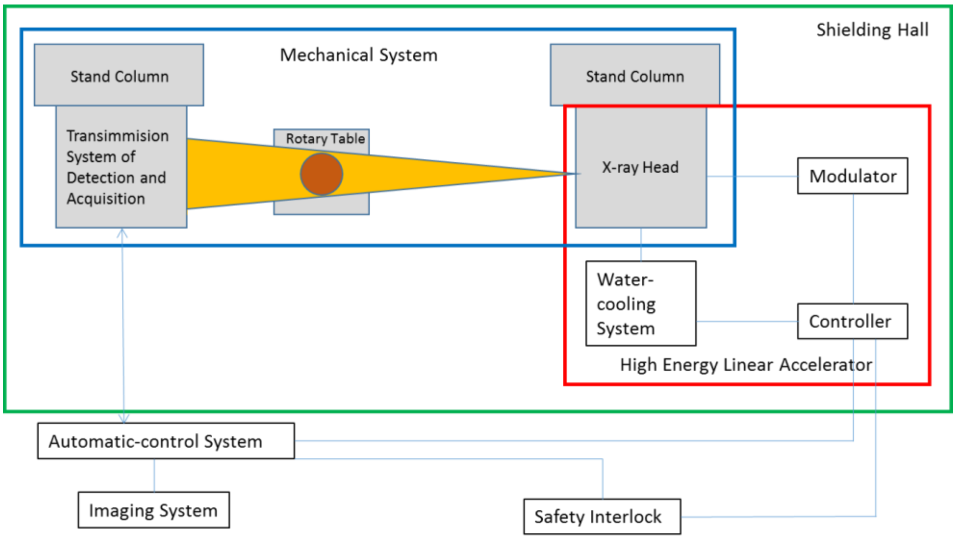

Figure P17: Schematic diagram of system structure.

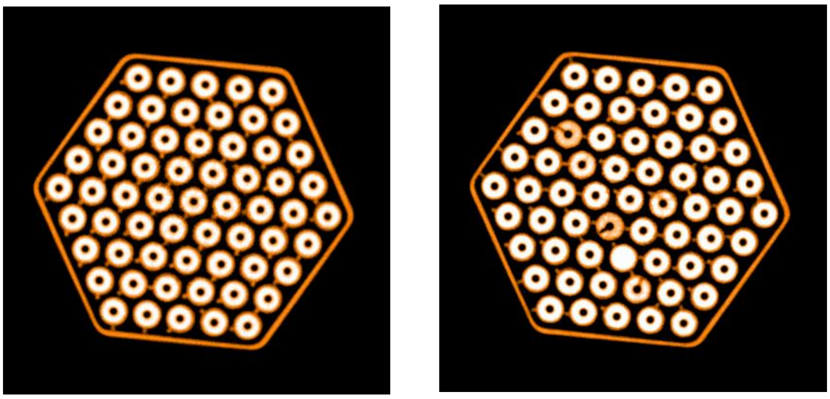

Figure P18: Computered tomography image of simulated nuclear fuel assembly

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Research on Closed-end Burst Testing of Irradiated Fuel Cladding Tube

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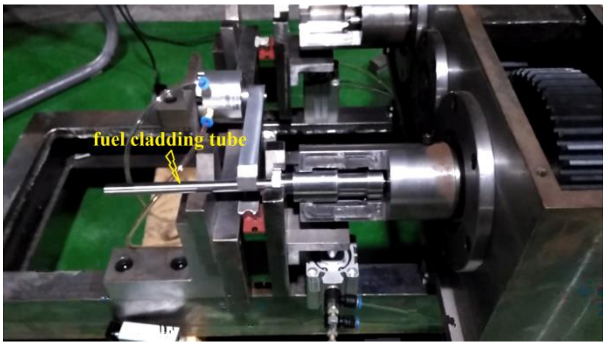
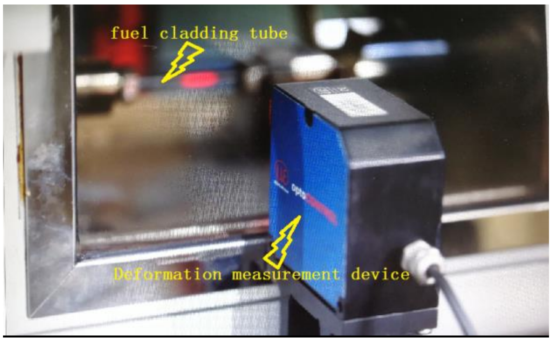
Closed-end burst testing of irradiation fuel cladding tube is one of the important detection for irradiation fuel assembly. The fuel cladding tube is endured internal pressure formed fuel core swelling and fission gas during reactor operation. Pressure resistance directly affects the integrity of cladding tube that is essential for safe operation of reactor. Closed-end burst testing can simulate stress state during condition of high temperature and high pressure. Therefore, closed-end burst testing is one of the important inspection items for fuel assembly. Due to the radioactivity of the cladding tube after irradiation, the sealing performance and deformation measurement of cladding tube are one of the key technologies for the closed-end burst testing. Therefore, this paper designed an automatic locking sealing device and deformation measurement continuously device based on the principle of closed-end burst testing, See Figures P19 and P20.



FigureP19: Automatic locking sealing device



FigureP20: Deformation measurement device

Closed-end burst testing was carried using cladding tube of Zr-4 alloy, and obtained the burst strength and circumferential elongation of Zr-4 alloy before and after irradiation, Burst crevasse analysis of fuel tube before and after irradiation was carried out. The test results showed that the automatic sealing and locking device of the cladding tube can achieve the blasting pressure of over 200MPa, and the continuous deformation measurement device can realize non-contact and real-time measurement of the diameter variation in the process of closed-end burst testing of cladding tube, and the measurement accuracy can reach 5μm, which the circumferential deformation of the cladding tube in the process of closed-end testing was obtained with the internal pressure change. The two devices designed were well applied to the burst testing of the irradiated cladding tube, and avoided the irradiation dose of the experimenter.

Key Words: Closed-end burst testing, Fuel cladding tube, Automatic locking sealing device, Deformation measurement device

The research on oxide dispersion strengthened (ODS) ferritic steel by chemical method

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Iron-based alloys, strengthened with a high number density of nanosized yttrium oxide particles, are typically referred to as oxide dispersion strengthened (ODS) steels, which exhibit better high temperature strength and creep properties than conventional ferritic/martensitic steels while maintaining the attractive properties of this class of alloy such as a low thermal expansion coefficient, high thermal conductivity, and excellent void swelling resistance. ODS ferritic or ferritic/martensitic alloys are being developed for fission applications, in particular as candidate materials for fuel cladding of fast reactor and Gen IV reactors. They are also considered as blanket material for future fusion reactors. However, a challenge in the development of these alloys is maintaining adequate fracture toughness in combination with high strength. According to the literatures (Verhiest et al., 2009; Degueldre et al., 2005; Klueh et al., 2005; Klimenkov et al., 2009; Ukai et al., 2009), ODS ferritic or ferritic/martensitic steels are materials with high temperature strength but inferior impact properties with respect to conventional ferritic or ferritic/martensitic steels.

In our present work, we use a novel chemical processing to generate a homogenous distribution of Y2O3 particles deposited on an argon atomized steel powder, then the composite powder has been consolidated by hot isostatic pressing (HIP), followed by forging and heat treatment. The ODS ferritic steel with nominal composition Fe-12.5Cr-2.5W-0.25Ti-0.2V-0.4Y2O3 (designated 12Cr-ODS) has been fabricated by this manufacturing route, and the microstructure, mechanical properties, hot deformation behavior and (H+/e–) and (He+/e–) irradiation of the ODS steel have been preliminarily investigated.

Experimental procedures

The microstructure of the specimens was investigated using transmission electron microscopy (TEM). TEM was performed at 200 kV on a JEOL 2010 equipped with energy dispersive spectrum (EDS) device.

Tensile specimens were machined according to ASTM standard E-8, tensile tests were carried out using an MTS 810 machine equipped with a heating chamber. Round tension test samples of 5 mm diameter and 25 mm gauge length were subjected to tension with a strain rate of 10^{-3} s^{-1}.

V-notch KLST specimens were machined with the dimensions of 10^{T} mm×10^{W} mm×55^{L} mm, 2 mm notch depth, 0.25 mm notch root radius, and 45° notch angle.

Fracture surfaces of the tensile and Charpy impact samples were examined by scanning electron microscopy (SEM) equipped with an energy dispersive spectrometry system.

The hot compression tests were performed using the Gleeble-1500 thermal stimulation machine at constant true strain rates of 0.002, 0.02 and 0.2 s^{-1} and at temperatures of 1070, 1120 and 1150°C.

The effects of (H+/e–) and (He+/e–) dual-beam irradiation on the microstructure damage behavior and oxide particles stable were studied using a combined irradiation device connected by a high energy ion accelerator with a hyperbaric electron microscope.

Results

Microstructure. Hot rolling results in a heterogeneous microstructure, the typical duplex (martensite plus a-ferrite) microstructure of the 12Cr material is observed. Transmission electron microscopy reveals uniformly distributed grains and a non-uniform dispersion of oxide particles, as shown in Figure P21a. A refinement of the microstructure consisting of equiaxed grains is observed for severely hot forged alloy. In the as-forged ODS material, a fine dispersion of Y_2O_3 particles was found with sizes ranging from 10-100 nm. The TEM image in Figure P21b shows nanoparticles that seem to be quite homogeneously distributed in the inner region of the grains, although they were occasionally found on grain boundaries. After forging, the distribution of the oxide particles changes from cluster to dispersed, therefore, forging is an effective way to eliminate the agglomeration and non-uniform distribution of Y_2O_3 particles throughout the matrix to obtain dense and uniform alloy, which would contribute to the improvement in the mechanical properties.

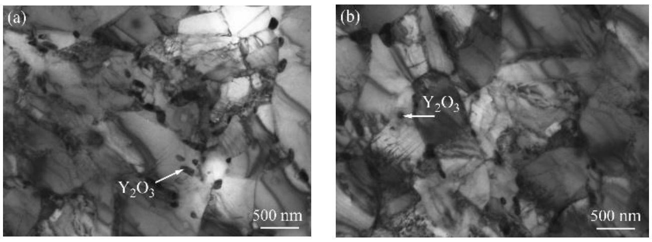

Figure P21: Bright field TEM images showing matrix grains and yttria dispersoids of (a) as-rolled alloy and (b) as-forged alloy.

Tensile properties and Impact properties. The results of tensile testing obtained for the 12Cr-ODS alloys in hipped, rolled, and forged conditions are presented in Table P2. Forging or hot rolling significantly improves the mechanical properties of the as hipped ODS ferritic steel. The as rolled ODS ferritic steel has highest yield strength, ultimate tensile strength, hardness, and bending strength. At room temperature the as rolled alloy exhibits ultimate tensile strength and the uniform elongation of 1147 MPa and 7.0%, respectively; The as forged alloy has excellent mechanical properties, at room temperature the ultimate tensile strength and the uniform elongation were 1104 MPa and 10.9%, respectively. At 550°C, The ultimate tensile strength and the yield strength have a slight decrease as the temperature increases to 550°C, and this decrease in strength is accompanied by an increase in uniform elongation. The as-forged 12Cr-ODS steel exhibits very attractive Charpy impact properties with upper shelf energy of 22 J and a low ductile-to-brittle transition temperature of about −15°C.

Table P2: Tensile properties of the 12Cr-ODS alloys tested in different conditions.

Temperature (°C)	As-hipped alloy			As-rolled alloy			As-forged alloy		
	YS (MPa)	UTS (MPa)	UE (%)	YS (MPa)	UTS (MPa)	UE (%)	YS (MPa)	UTS (MPa)	UE (%)
RT	453	623	2.9	763	1147	7.0	738	1104	10.9
550	406	589	3.1	713	1109	9.5	685	1052	11.4

Hot deformation behaviour. The deformation behavior of the as forged ODS ferritic steel has been investigated by using isothermal hot compression tests. The true stress-true strain curves reveal character of steady state type, which can be represented by a Z parameter in the hyperbolic-sine constitutive equation. The activation energy for hot deformation of the ODS ferritic steel is about 384.487 kJ/mol. The processing maps of the material are obtained according to the dynamic materials model, as shown in Figure P22. The map exhibits two domains: the stability domain in the temperature range of 1070-1135°C and strain rate of 0.002-0.02 s^{-1}, and the instability domain in the temperature range of 1075-1120°C and strain rate of 0.02-0.2 s^{-1}.

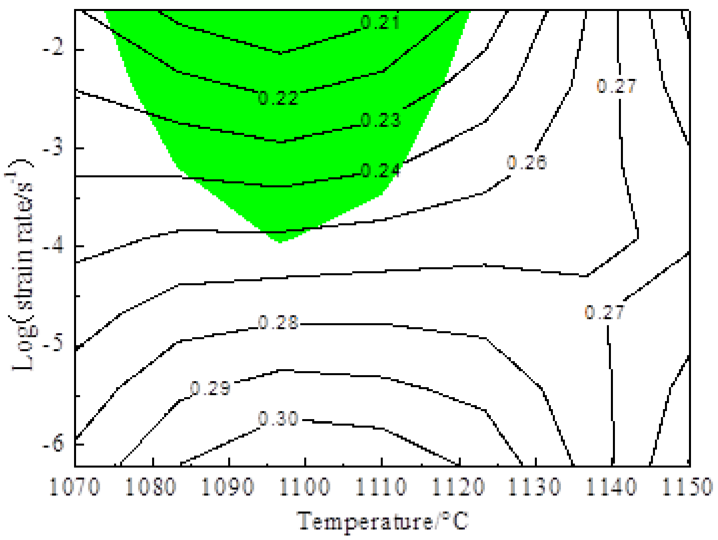

Figure P22: Processing maps for 12Cr-ODS ferritic steel at strain of 0.3, contour numbers represent percent efficiency of dissipation and shaded region corresponds to instability domain.

(H^+/e^-) and (He^+/e^-) irradiation. The as forged ODS ferritic steel shows excellent swelling resistance under (H^+/e^-) and (He^+/e^-) irradiation at temperature range of 350-550°C, the total void swelling was all less than 0.15% at dose of 15 dpa. The size of the dispersed oxide is stable, no solubility process was observed. Chemical compositions of the oxide particle, the interface, and the matrix have no obvious change before and after irradiation.

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Remote Metal Fuel Slug Fabrication System Based on Injection Casting

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The KAERI (Korea Atomic Energy Research Institute) has been developing a technology for fabricating TRU (Transuranium) metal fuel using TRU ingot produced from Pyroprocessing. This TRU metal fuel will be used as a fuel material of SFR (Sodium-Cooled Fast Reactor). The TRU metal fuel fabrication processes should be conducted in a fully remote manner at a hot-cell because of a nature of a high radioactivity of TRU ingot.

This paper presents an engineering-scale fabrication system for remotely manufacturing metal fuel slugs developed at KAERI. This remote fabrication system mainly consists of two sub-systems: an engineering-scale fabrication equipment and a remote fabrication mock-up. The engineering-scale equipment includes an injection casting furnace, a vacuum pump, a generator, a pressure tank, and a control panel. An injection casting furnace consists of an upper chamber, a lower chamber, a mold assembly, a mold assembly holder, a mold assembly transporter, a locking device, and a base platform. A high frequency non-cooled induction coil system was engaged inside the lower chamber of the injection casting furnace. A mold assembly can contain a maximum of 78 molds. Each mold made of quartz configures a length of 450mm, and an outer diameter of 5.70mm and an inner diameter of 8.70mm. This casting furnace was designed and constructed in modules to facilitate a remote operation and maintenance. The injection casting furnace can produce a maximum of 78 metal slugs at one operation. A remote fabrication mock-up is a facility to demonstrate an engineering-scale fabrication equipment in a remote manner. This mock-up was designed and constructed to evaluate the remote operability and maintainability of the developed fabrication equipment and test its performance fabricating metal slugs using cooper or depleted uranium before TRU ingot is used. The mock-up includes tools to remotely handle the casting furnace. These tools are a pair of telemanipulators, overhead crane, and a flexible viewing window, etc. The telemanipulator (HWM, A110) has an effective handling capacity of 15 kg in any position within its workspace. A 2-ton overhead crane is mounted on tracks installed above the mock-up and provides remote handling capabilities over entire mock-up space. A single viewing window is designed to be connected with a pair of telemanipulators by a holder and can be moved in a left or right direction with respect to the front wall of the mock-up. The position of the viewing window can also be varied with the movement of the operating position of telemanipulators depending on the tasks required. Such design provides operator with a more efficient means for testing equipment located within a limited space of the mock-up.

Currently, an engineering-scale fabrication system has been established at the Fuel and Material Test Facility of KAERI. The injection casing furnace will be tested and evaluated in the remote fabrication mock-up from the remote operability and maintainability viewpoint. And then, metal fuel slug fabrication will be conducted using cooper or surrogates in the middle of this year.

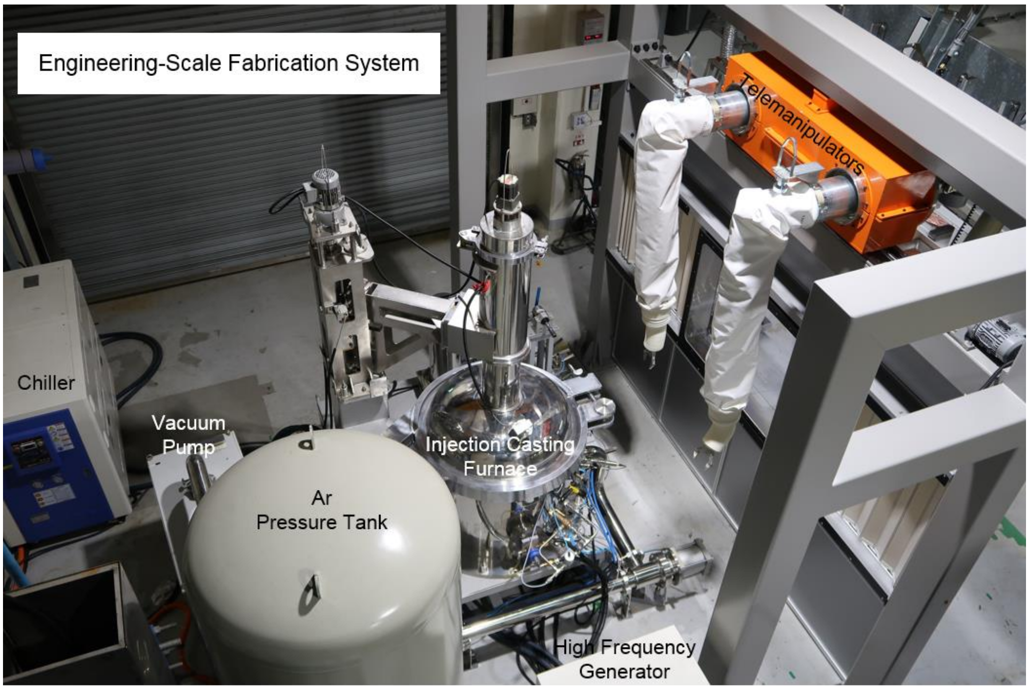

Figure P23. Engineering-Scale Fabrication System.

This work was supported by the National Research Foundation of Korea funded by the Ministry of Science and ICT (2017M2A8A5014888).

Sample Preparation Techniques for Post Irradiation Examinations in the Reactor Fuel Examination Facility

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Introduction

Reactor Fuel Examination Facility (RFEF) is one of the largest hot laboratories in Japan. In RFEF, several kinds of PIEs are performed to evaluate the safety and reliability of spent nuclear fuels, and the data from PIEs is provided to the customers such as nuclear researchers and fuel vendors. In these years, the requirements from our customers become more complicated and more accurate, so that the sample preparations such as cutting, holding or defueling prior to PIE itself must be improved to meet their requirements. In this report, several sample preparation techniques for PIEs are described.

Sample preparation techniques for PIEs

Cutting of various shaped samples. Most of the sample preparation devices installed in RFEF's hot cells are designed to apply a rod- or a pellet-shaped samples which has cylindrical forms. On the other hand, one of the PIE requests was to perform the composition analysis for the TMI-2 debris which has a non-cylindrical shape formed from the molten reactor core elements during the accident. To cut the TMI-2 debris for the SEM observations, the previous type of sample holder is not applicable because of its complicated shape. Therefore, the improved chucking device was necessary to hold the non-cylindrical shaped sample for its cutting. Figure P24 shows the improved chucking device for TMI-2 debris which has the multiple clamp bolts to hold the debris sample by several different angles. The device can hold the non-cylindrical shaped samples completely, and samples can be cut at the aimed cutting line precisely.

Figure P24: Normal sample holder and improved chucking device

Defueling device for hydrogen analysis of cladding tube. It is very important for the precise hydrogen analysis of cladding tube to prevent the influence of the hydrogen from the fuel pellets. Therefore, the defuleing from the cladding is one of the most important procedures for the sample preparation of the hydrogen analysis to ensure its precision. Figure 2 shows the appearance of the defueling device developed in RFEF. This device mainly consists of the precise hand grinder, rotary scroll chuck and XYZ stage. The fuel cladding sliced with the fuel pellet as approx. 1 mm thickness is set on the rotary scroll chuck with its dedicated attachment holder and the pellet is grinded with hand grinder to the utmost. Attachment holder protects cladding tube from scratches and makes handling of 1 mm thick samples easy. With this defueling device, the influence of the fuel pellet on the precision of hydrogen analysis can be reduced as possible, and at the same time this device is also effective to save the gamma-ray exposure of the operators and to reduce the contamination of the glove box in which the hydrogen measurement apparatus is installed.

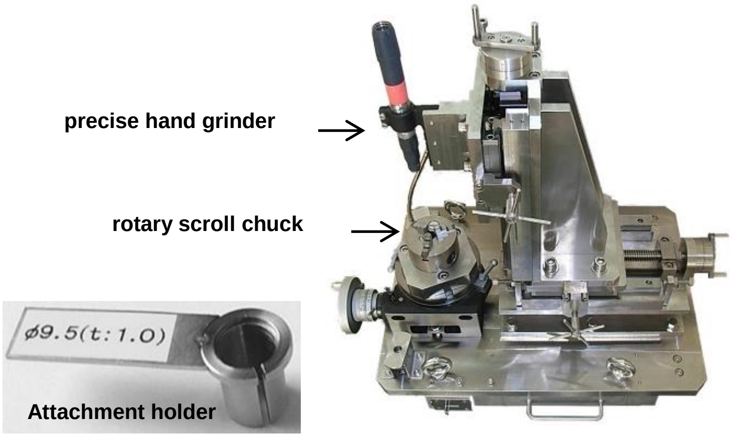
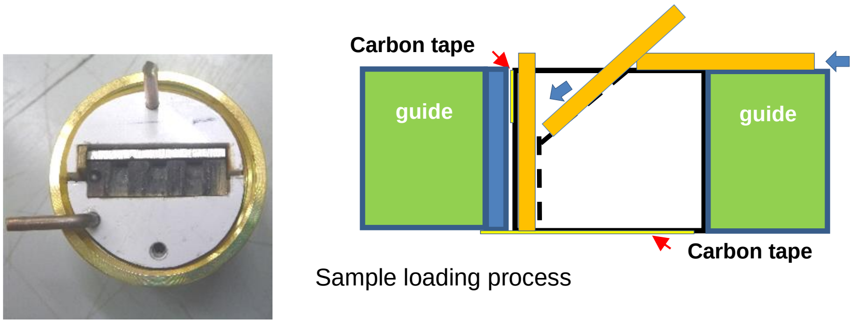


Figure P25: De-fueling device and attachment holder

Sample holder for minimum sample piece. As the PIE requests become more precise, the sample size gets smaller and smaller. RFEF is the hot laboratory designed to perform PIEs with scale ranging from the fuel assemblies to the fuel pellets/claddings, so that the manipulator claws are not fit to handle the millimetre scale samples. Regarding one of the PIE requests, < 1mm thick sample should be inserted into the 1 mm gap slit to observe the fracture surface by SEM. To perform this preparation easily, the dedicated sample holder was designed as shown in Figure P26. Sample is placed on the top side of the holder horizontally and moved to the holding slit along the inclination using the manipulator. By this procedure, the sample is set in the slit vertically and kept the position by carbon tapes.



Sample loading process

Figure P26: Sample holder

Development of Laser Ablation Inductively Coupled Plasma – Mass and Optical Emission Spectrometry Methodologies for Elemental Analysis in a Medium Active Cell Environment

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As Post Operational Clean Out and Decommissioning become more prevalent on the Sellafield Site, there is a requirement for a new analytical facility to be developed. Sellafield Ltd, National Nuclear Laboratory (NNL) and Axiom are developing existing High Active (HA) facilities and constructing a new Medium Active (MA) cell suite within NNL's Central Laboratory. A key part of this project is the development of technologies that Sellafield Ltd and NNL have not previously implemented in a HA/MA cell environment. Examples of new technologies include Laser Ablation Inductively Coupled Plasma Mass Spectrometry (LA-ICP-MS) and Laser Ablation Inductively Coupled Plasma Optical Emission Spectroscopy (LA-ICP-OES).

LA-ICP-MS/OES is a well-developed technique for the elemental analysis of solid materials. The capability to directly analyse solids would greatly eliminate the need to perform dissolutions on samples, reducing the amount of liquid waste generated through analysis. The requirement for sample preparation within the MA cell can cause difficulties as it will need to be undertaken using a form of remote handling. Uncertainties surrounding analytical capability of the techniques also require reviewing to ensure that customer expectations and quality requirements can be met.

In order to determine the validity of LA-ICP-MS/OES, British Geological Survey (BGS) were commissioned to perform a series of investigations into potential sample preparation techniques, including their applicability within a MA cell. Sample preparation techniques which showed promise were then used to analyse certified reference materials and the quality of analysis achievable reviewed for both LA-ICP-MS and LA-ICP-OES.

Nuclearization projects of a tomographic atom probe and of an electropolishing machine for researches on neutron irradiated materials at the atomic scale

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The Tomographic Atomic Probe (TAP), located in LECI, in the CEA/Saclay LECI hotlab facilities, has become an essential technique for the fine characterization of nuclear materials. Indeed, this high-resolution three-dimensional microscope is a unique instrument because it allows an understanding of the phenomena at very fine scales namely the atomic scale with unparalleled chemical and spatial sensitivity, hence it presents a major interest for the characterization of defects created by irradiation that are mostly visible at these scales. In the aim of studying neutron-irradiated materials with this instrument, LECI undertook the nuclearization project of the TAP. This project, which is a major stake for LECI, includes several milestones, namely the manufacturing of a glove box, the instrumental nuclearization, and then the nuclearization of the rooms (engineering studies and building works to accommodate irradiated samples) associated with the nuclear safety and radiation protection studies of this project. As TAP needs specific samples (with a needle shape) to be studied, this project is associated to the nuclearization project of a sample preparation machine.

Nuclearization project of a tomographic atom probe

This project is part of the "Equipement d'Excellence" called GENESIS[2],[3] that is a characterisation platform dedicated to the analysis of irradiated materials at nanometric scale where LECI is associated to GPM (Rouen, France) and CIMAP (Caen, France). Within this project, LECI is currently installing a TAP (and a FIB/SEM which will not be presented here). Scheduled to be operational at the end of 2018, this instrument will allow observing irradiated nanometric samples in order to improve safety studies associated to the aging of actual and new material used in nuclear power plants, and to study the degradation mechanisms produced by neutron irradiation at atomic scale.

To complete this project, several actions were carried out. The first one consisted in the instrumental nuclearization that is it to say the modification of the TAP itself and the manufacturing of a glove box.

The modification of the TAP including the connection of the TAP to a device for vacuum transfer of samples and its docking port, the shielding of the detector to minimize background noise, the remote handling of the vertical rod and a system to recover fallen irradiated samples in the analysis chamber. – CAMECA supplied these changes (see Figure P27).

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2 This work has benefited from a state aid managed by the National Research Agency under the program "Investments for the future" with the reference ANR-11-EQPX-0020

3 Groupe d'Etudes et de Nanoanalyses des Effets d'Irradiations

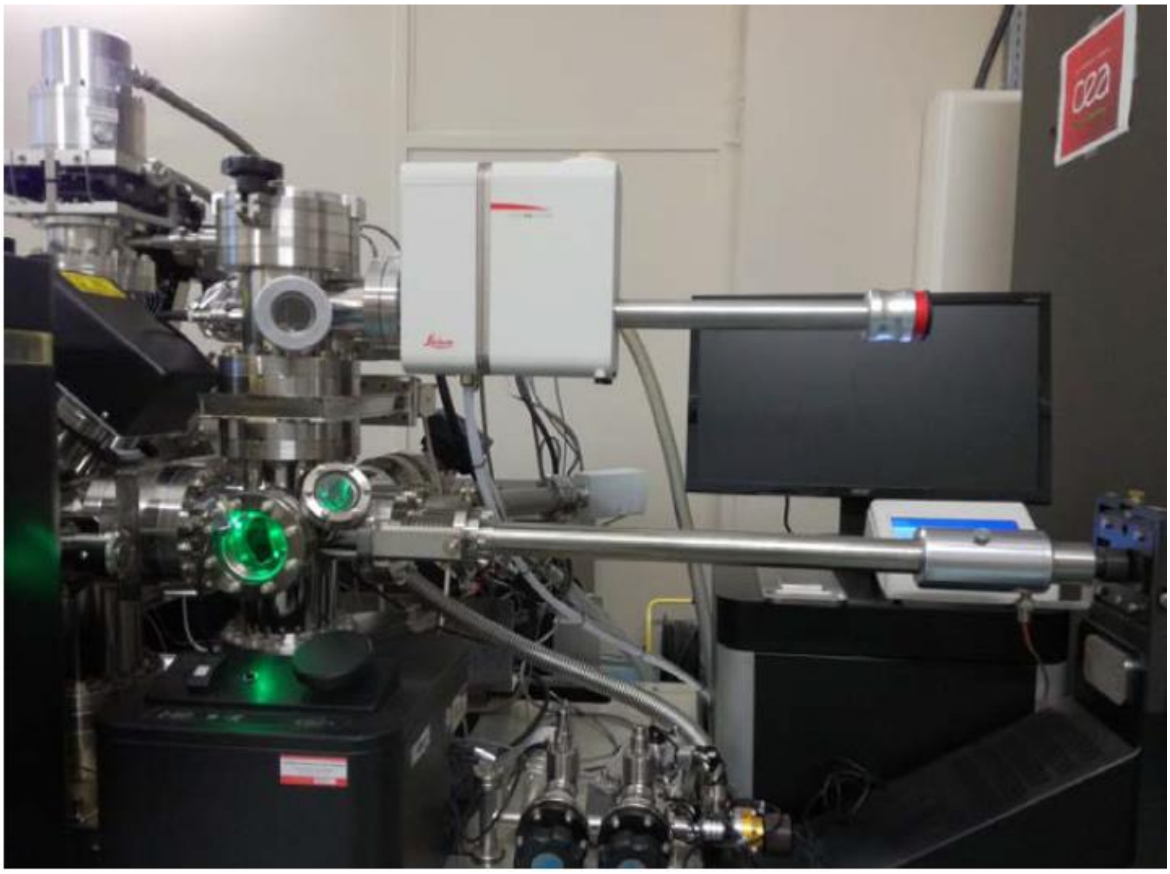

Figure P27: The nuclearized TAP

A glove box was also supplied (by DEFI SYSTEMS) ensuring many functions as (see Figure P28):

- the reception of the neutron irradiated samples from the sample preparation machines (electopolishing or MEB/FIB),
- the storage of the irradiated samples in a vacuum shielded cabinet,
- the preparation of the samples on the holders appropriated to the TAP (or the TEM),
- the possibility to clean samples and holders with a plasma cleaner.

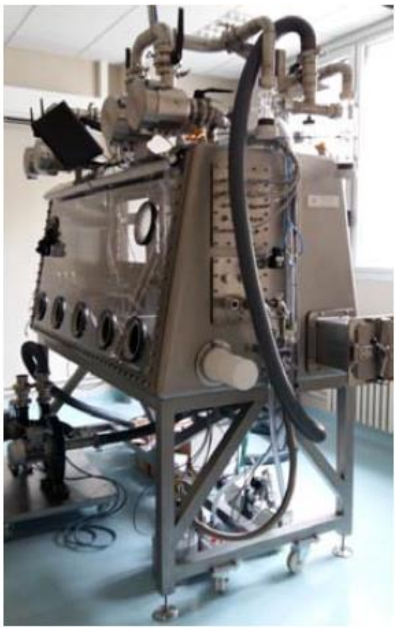

Figure P29: The TAP glove box

Another important point of the project was to nuclearize the rooms that will host the nuclearized TAP and the glove box. Many engineering studies and building works were conducted with the collaboration LECI operating team. Moreover, nuclear safety and radiation protection studies were realized to obtain the authorization from the regulator to analyse neutron-irradiated materials with the TAP. The final decision of the regulator is expected by the end of 2018.

Nuclearization project of a sample preparation machine

A TAP study requires a special preparation of the samples, because they must be fabricated in the form of a needle with an end radius between 10 and 100 nm. Electropolishing is one of the methods used for this particular sample preparation. Its principle is to chemically smooth a sample of 1 cm long and approximately 0.25 mm^2 or circular in cross section to obtain a needle (**Figure P30**).

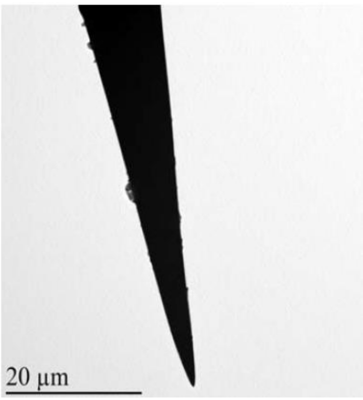

Figure P30: A needle for a TAP study

To characterize irradiated samples with the TAP, the laboratory of microscopy of the LECI hot lab studied and designed an electropolishing machine to be integrated in a glove box dedicated to the preparation of samples with chemical products (see Figure P31).

Figure P31**:** The glove box for the electopolishing machine

The electropolishing machine is composed of many elements that is to say the mechanical part and electronic modules (power generator and motion controller). Moreover, the laboratory developed a software for the control of the machine with LabVIEW. A picture of the software interface is visible in Figure P32.

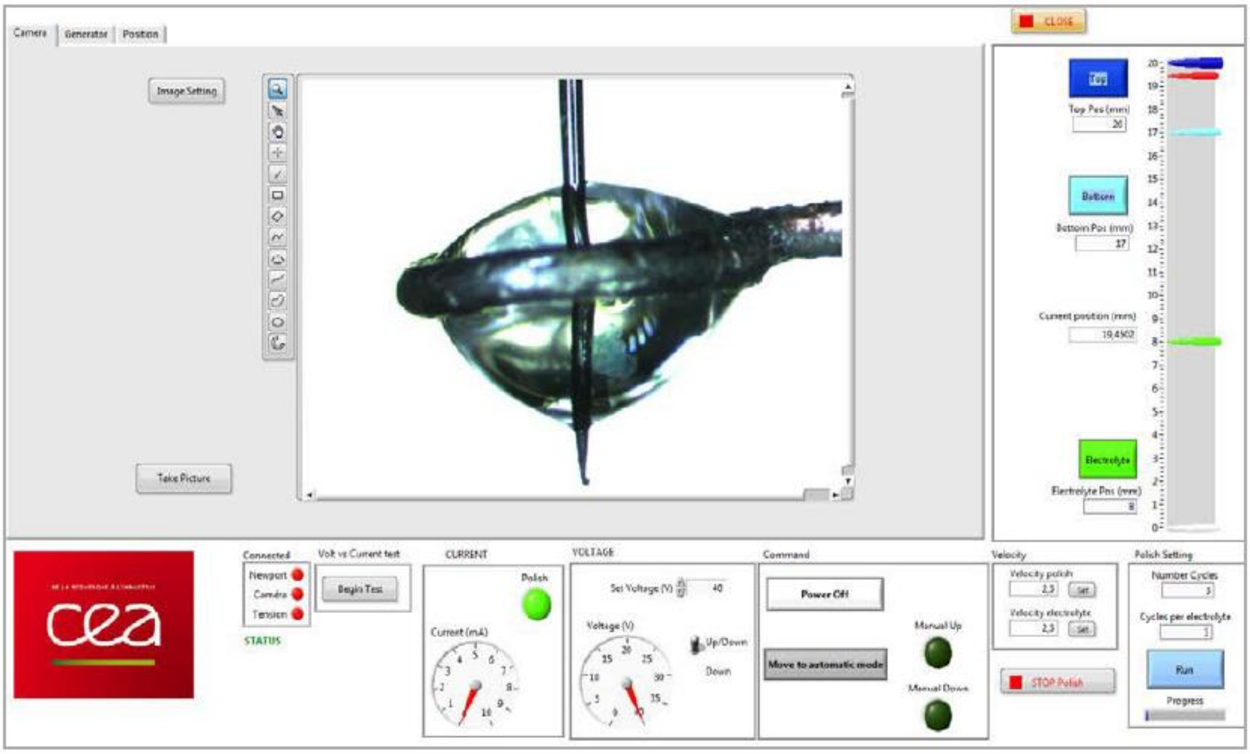

Figure P32**:** Print screen of the interface of the electopolishing machine software developed by CEA

The next step for this project is to integrate the electropolishing machine in the dedicated glove box with suitable connectors. The hardware and relevant electronic modules will be implemented in the room out of the glove box. It will be done by September 2018.

Nuclearized Raman microscope coupled with a hot-stage: new tool to study (U, Pu)O_{2-x} fuel microstructure

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In the frame of the development of uranium-plutonium mixed oxide fuels for Sodium-cooled Fast Reactors (SFRs), characterizing nuclear materials by various techniques is paramount. These fast neutron reactors imply the use of a (U,Pu)O_{2-x} ceramic fuel with a Pu/(U+Pu) content between 19 and 30 mol.%. Furthermore, the physico-chemical and microstructural properties of such fuels, such as chemical homogeneity, oxygen stoichiometry (O/(U+Pu) ratio) and crystallographic structure, have to meet precise criteria for being introduced in the reactor core. As shown in recent studies by Talip *et al.* (2018) and Elorrietta *et al.* (2017), Raman microscopy is a promising tool for characterizing the physico-chemical properties such as, among many others, the cation distribution homogeneity, the grain size, the crystal defects that are of main interest for the production of nuclear fuels.

The development of a new *in situ* Raman device dedicated to handling transplutonium-bearing materials is currently in progress in the new hot-laboratory L26 located at the ATALANTE facility (CEA Marcoule, France) in the framework of the TARRA project[4]. This new experimental set-up installed in a glovebox is complementary to the Raman set-up existing at the C19 hotcell also at the ATALANTE facility and mainly dedicated to study irradiated fuels (Jérou et al., 2010).

This new experimental set-up designed by Optique H. Peter company consists in a confocal optical microscope coupled with an iHR-320 Raman spectrometer supplied by HORIBA company. Moreover, a new nuclearized Raman micro-furnace will allow us performing *in situ* Raman measurements up to 1800°C under controlled atmosphere.

Experimental set-up

Confocal microscope. The confocal optical microscope designed by the Optique H. Peter company and shown in Figure P33a, is installed inside a nitrogen-filled glovebox (Cf figure P34). The glovebox stage has been reinforced in order to increase its rigidity to minimize vibrations induced by the laboratory/facility ventilation. The microscope is fixed on an optical table which is fixed on the glovebox floor. The microscope is equipped with two turrets which can be manually exchanged: the first one equipped with Olympus© objectives (x5 to x100 magnification) and dedicated to optical

[4] More details about the TARRA project and the L26 laboratory can be found in O. Dugne *et al.* abstract entitled « TARRA Project: transfer of MOX R & D between 2 CEA sites » presented during the same conference.

observation and Raman mapping on polished samples; the second one dedicated to *in situ* Raman experiment Mitutoyo© objectives (x5 up to x50 magnification) with long working distance in order to accommodate the micro-furnace geometry. The turret dedicated to optical observation is also equipped with a micro-indenter device in order to perform Vickers hardening tests. On each turret, the change of objective is motorized. The sample displacements (x, y, z and rotation) are performed using independent motorized stages.

Raman spectrometer

The spectrometer is an iHR-320 supplied by the HORIBA© company located outside the glovebox. The spectrometer is equipped with 2 lasers (532 nm and 660 nm) with adjustable power by the labspec software developed by HORIBA©. Each laser is connected to a HORIBA Raman superhead mounted on the microscope (Cf Figure P33a) via an optimized optical fibber (10 µm diameter). The Raman signal is transmitted to the spectrometer using a 100 µm diameter optical fibber. Thanks to the microscope motorized stages, Raman mapping/imaging at micrometre scale of polished sintered samples can be easily performed.

Raman micro-furnace

The nuclearized version of Raman micro-furnace initially developed by Montagnac *et al.* (2013) is shown in Figure P33b. The furnace is a confined 1 mm diameter heating-wire (Pt with 20% Ir alloy) flattened in the middle where the sample will be placed. The wire is connected to a power supply and the temperature is monitored/measured using an optical pyrometer from the Lumasense© company. The latter is installed on a dedicated x,y,z stage in order to compensate sample displacements due to the wire thermal expansion. This heating system has low thermal inertia and it is possible to change the temperature between room temperature up to temperatures >1700 °C in few seconds (Neuville et al., 2014). The micro-furnace is water-cooled using a close circuit coupled to a heat exchanger. Thanks to gas inlet and outlet, a continuous flushing of a controlled atmosphere around the sample during a heat treatment can be performed. Furthermore, a set-up controlling, imposing and measuring the variations in the oxygen partial pressure is also available.

As illustrated by the 3 figures, the whole experimental set-up is already installed at L26 laboratory. As describe here, the whole system is optimized to minimize as much as possible sample and equipment handling: once the sample is installed (except a turret change), the experiments/observations are performed/conducted remotely in an office located near the hot laboratory The first radioactive experiments on $(U,Pu)O_{2-x}$ samples are scheduled in November 2018.

a. b.

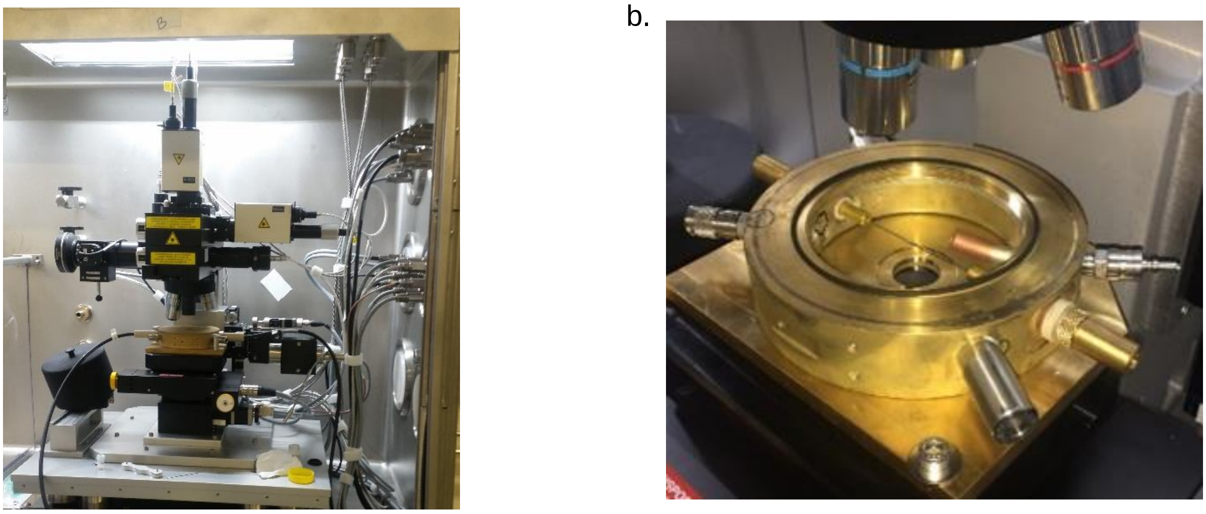

Figure P33: Detailed views of (a) the Raman microscope installed inside the glovebox; (b) the nuclearized Raman micro furnace installed on microscope motorized stage.

Figure P34**:** View of the new glovebox dedicated to Raman microscope installed in the L26 laboratory (ATALANTE Facility, CEA Marcoule, France).

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Application of ICP-MS to analysis of nuclear fuel debris and radioactive wastes

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The Tohoku Earthquake and the tsunami that followed occurred in March 2011. The surge from the tsunami caused loss of cooling system in Tokyo Electric Power Company Holdings Fukushima Daiichi Nuclear Power Station (1F). The loss of cooling system allowed the fuel in reactor units 1-3 at least partially meltdown and caused hydrogen explosions in unites1 and 3. As the results of the explosions, a large amount of radioactive materials were released into the environment and contaminated a vast area containing the 1F site. Japan Atomic Energy Agency, JAEA, currently sets up an “Okuma Analysis and Research Center” next to the 1F site, which aims at characterizing radioactive wastes and fuel debris generated in the 1F site.

Radioactive waste samples taken in the 1F site are planned to be analysed by mainly conventional radiometric methods. Gamma spectrometry of HPGe detector is used to identify gamma emitters. Beta emitters with negligible or no gamma-ray emission are planned to be determined from measurement of beta activity, which is measured by a liquid scintillation counter or a low background gas flow counter. These measurements require complicated and time-consuming process including chemical separation and purification in advance. To avoid the chemical separation, Inductively Coupled Plasma-Quadrupole Mass Spectrometry, ICP-QMS, which has an advantage of sensitivity over radiometric analysis for long-lived radionuclides, was applied. Isobaric interference is the major task to be solved for the mass spectrometric analysis.

The procedure of the mass-based analytical method for radionuclides and the detailed results for an example case for Zr-93 are described in this presentation.

Analytical methods

Conventional radioactivity measurement. Radiation counting method has been used many years. Gamma spectrometry of HPGe detector is used measuring gamma rays to determine the gamma emission radionuclides after just simple preparation. Beta decay isotopes with very low, or no gamma emission, were planned to be measured its activities by beta counting methods, such as a liquid scintillation counter and a low background gas flow counter. It requires chemical isolation of element before the measurement, because beta counter has poor energy resolution. Alpha decay radionuclides emit close energy alpha particles. Chemical purification is also required, even if alpha spectrometry has some energy resolution. Electron capture radionuclides would be measured low energy X-rays (eg. Low Energy Ge Detectors). Purification process is necessary to avoid increasing background signals by Compton scatter of gamma rays. From the results, almost radionuclides should be purified before counting its radio activities.

Analytical instruments are planned to employ radiometric analysis, namely liquid scintillation counter, low background gas flow counter, alpha spectrometer and gamma spectrometer. All these instruments based on radiometric analysis are very reliable, as they have been used for many years.

ICP-QMS spectrometry. ICP-QMS has been a powerful analytical instrument to measure very low concentration isotopes. Ohtsuka compared detection limit of radiation counting method with ICP-QMS (Ohtsuka, 2006). ICP-QMS has an advantage in long half-life, approximately more than 1,000 years, radionuclides measurements.

As long as using ICP-QMS of single quadrupole mass spectrometer, isobaric interference could not be ignored. Zr-93 is one of the most difficult radionuclides to be analysed by mass spectrometer. When Zr-93(half -life 1.5×10^5Y) is measured by ICP-MS, Nb-93(stable) and Mo-93(half-life 4.0×10^3Y) will interfere, because the same mass number. Triple quadrupole mass ICP-MS (QQQ-ICP-MS: Agilent 8900) is selected to analyse Zr-93. Two quadrupole mass separators (QMS) are tandemly arranged, and octupole-based collision reaction cell is set between them, as shown in figure 1. Sample solutions are heated and ionized in the Ar plasma. These ions are led to the first QMS, that is used as a mass filter to pass through only ions of M/Z= 93, such as $[Zr\text{-}93]^+$, $[Nb\text{-}93]^+$ and $[Mo\text{-}93]^+$. Then these ions are reacted with reaction gas to make different molecule ions, whose mass number will also be different. The second QMS separated ions to be measured and collected by electron multiplier detector.

We studied possibility of Zr separation from Nb and Mo with QQQ-ICP -MS using reaction cell, introducing reaction gas, oxygen or ammonium gas. In the measurement, elemental standards were used pure elemental solutions and concentration was adjusted to 1ppb.The first Q-mass was used as a mass filter, M/z was 93, and the second Q-mass was varied its M/z from 93 to 275.

Figure P36 shows ion production rate detected through the second Q-mass of using two reaction gases. The left figure is for the ion production rate without any gases, “no gas” mode. Right upper figure is for using oxygen and right lower figure is for ammonium gas as reaction gas. Mass shift, difference of M/z, was plotted on the abscissa and ionic intensity on the ordinate. Nb and Mo isobaric interference to Zr were not removed with “no gas” mode, because mass shift has not been observed Zr, Nb and Mo measurement. Small amount of NbO^+ and MoO^+ were found with O_2 mode. Almost Nb and Mo were formed di-oxide ion, but Zr was mono-oxide, this estimate reducing isobaric interference, when the Zr concentration is much higher than those of Nb and Mo.

When using NH_3 as reaction gas, Nb, Mo and Zr, forms different ions, $Nb(NH_4)(NH_2)_4^+$, Mo^+ and $Zr(NH_3)_6^+$, respectively. These elements were separated by QQQ-ICP-MS just adding NH_3 gas as reaction cell. From the result, QQQ-ICP-MS can make possible to measure several nuclides, simultaneously, without any chemical pre-treatment before delivering sample solution to the instrument. Therefore, separation process can be simplified just introducing mixed solutions into the instrument.

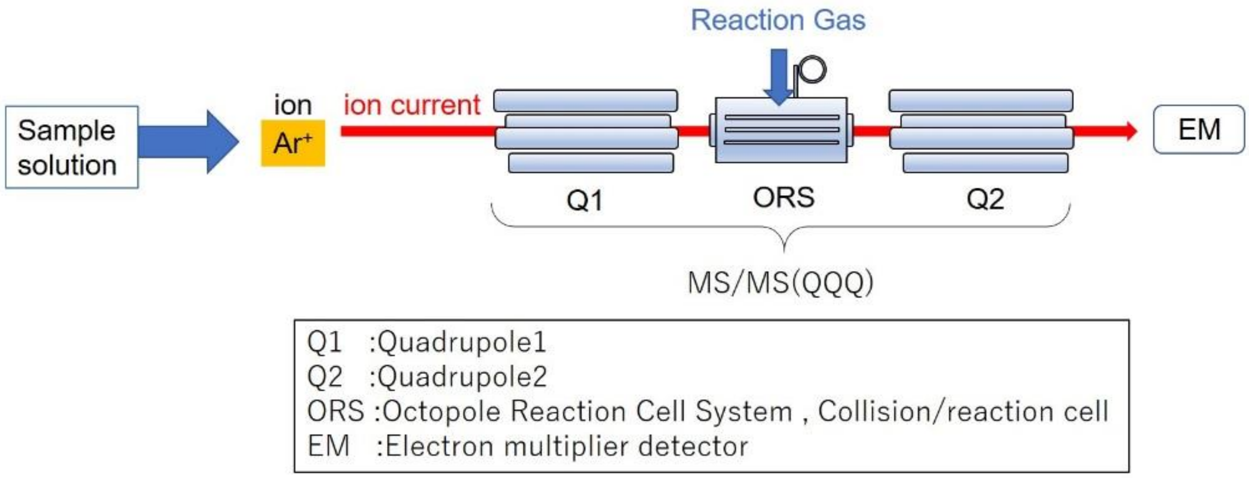

Figure P35: Schematic diagram of ICP-QQQ-MS

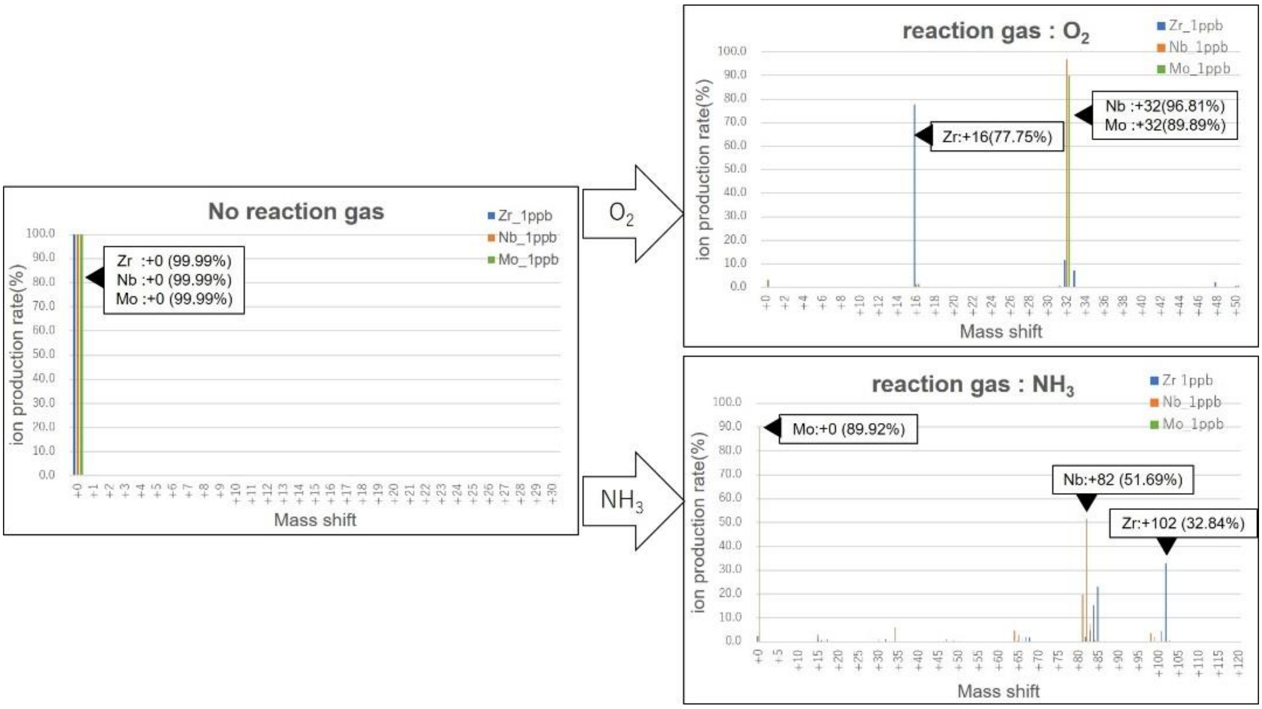

Figure 36: Ion production rate (Mass spectra) of QQQ- ICP -MS

Conclusions

In the QQQ-ICP-MS using O2 and NH3 as a reaction gas, separation condition of Zr from Nb and Mo, was found without chemical purification. QQQ-ICP-MS could measure several nuclides at one time. In this manner, pre-treatment process could be simplified and separation time could be shortened. This can make possible to increase the number of analysis samples.

Acknowledgement

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Design, development, and installation of hot cell instrumentation for Spent Fuel Autoclave Leaching Experiments (SF-ALE)

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Accurate determination of radionuclide release from spent nuclear fuel under geological repository conditions is a critical aspect with respect to various safety scenarios and with a significant impact on the societal acceptance of long-term fuel disposal. Understanding the kinetics of this release - in short (instant) and long-term time domains - allows us to support more accurate models and solutions for safe long term spent nuclear fuel strategies. The work on fuel dissolution kinetics was initiates at SCK•CEN in the FIRSTnuclides project (www.firstnuclides.com). While continuing on this basis, the current efforts are intended to take a step further by improving the fuel dissolution and sampling instrumentation and developing experimental in-cell capabilities, such as Raman spectroscopy.

The project SF-ALE (Spent Fuel Autoclave Leaching Experiments) was developed to prepare a complete hot cell infrastructure for performing spent fuel dissolution experiments with high precision. The project has currently two stakeholders, NIRAS-ONDRAF (Belgium) and Forschungszentrum Jülich (Germany), who both operate three independent autoclaves under specific dissolution conditions.

The poster will describe the equipment needed to perform such a dissolution experiment in a hot cell. The autoclaves were purchased and then adapted for use in hot cell. A support frame was developed to move it from a standby position towards sampling position. During the 18-months experiment several samples need to be taken. A method was developed for sampling liquid and also for the gas phase without the opening of the autoclaves to avoid as much as possible the air contamination. The hot cell was adapted for this purpose with extra feedthroughs. Liquid samples will be taken inside the hot cell 04 and they will be diluted in an adjacent hot cell before analysis. The gas phase will be sampled in a container outside the hot cell using an in-house developed valve system. These valves system is placed in a closed cabinet with in and outlet filters which are connected to the main ventilation system of the hot cells.

A new gas mass spectrometer was taken into service. The gas sampling procedure was developed in order to assure that an optimal pressure of 25 mbar is obtained for the mass spectrometry. Radiochemical analyses for the solution samples include: Alpha spectroscopy, Gamma spectroscopy, LSC + separation, TIMS+separation and HR-ICP-MS analysis. Further development is currently ongoing to commission in-cell Raman and harness hot-FIB to further investigate the link between the physical state of fuel and the nuclide release.

Figure P37: Valves system for gas samplingUse the Caption style and label Figure

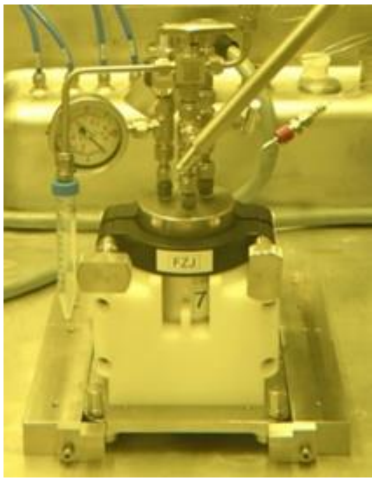

Figure P38: Autoclave in his parkingspot in hot cell 04

The Decommissioning and Waste Management programme of the European Commission Joint Research Centre

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The European Commission Joint Research Centre (JRC) Nuclear Decommissioning and Waste Management Programme (D&WM) is tasked to progressively dismantle all EURATOM nuclear installations at 4 JRC sites (Ispra in Italy; Petten in The Nederlands, Karlsruhe in Germany and Geel in Belgium), including those already shutdown and, at the end their operational life, those currently in use for research activities.

The scope of the programme includes a variety of installations, ranging from research reactors to hot cells, laboratories and other facilities where radioactive substances were and are still handled. It also intends to treat “historical” waste and waste arising from the dismantling operations.

All D&WM activities are implemented safely and according to relevant Member State regulations. The decommissioning activities (including pre-decommissioning activities, dismantling, waste clearance or safe conditioning and removal) are to be pursued as soon as possible after the end of the R&D operation to avoid unnecessary ageing of the infrastructure, and to limit costs escalation arising from extended safe conservation and loss of know-how. The final objective is to release the buildings and areas free of radiological constraint.

The programme is assumed to last for several decades until the decommissioning of the last JRC nuclear installation and the related waste disposal are complete. This contribution illustrates the present status of the JRC D&WM programme with examples of completed decommissioning and dismantling of facilities and descriptions of ongoing and future activities.

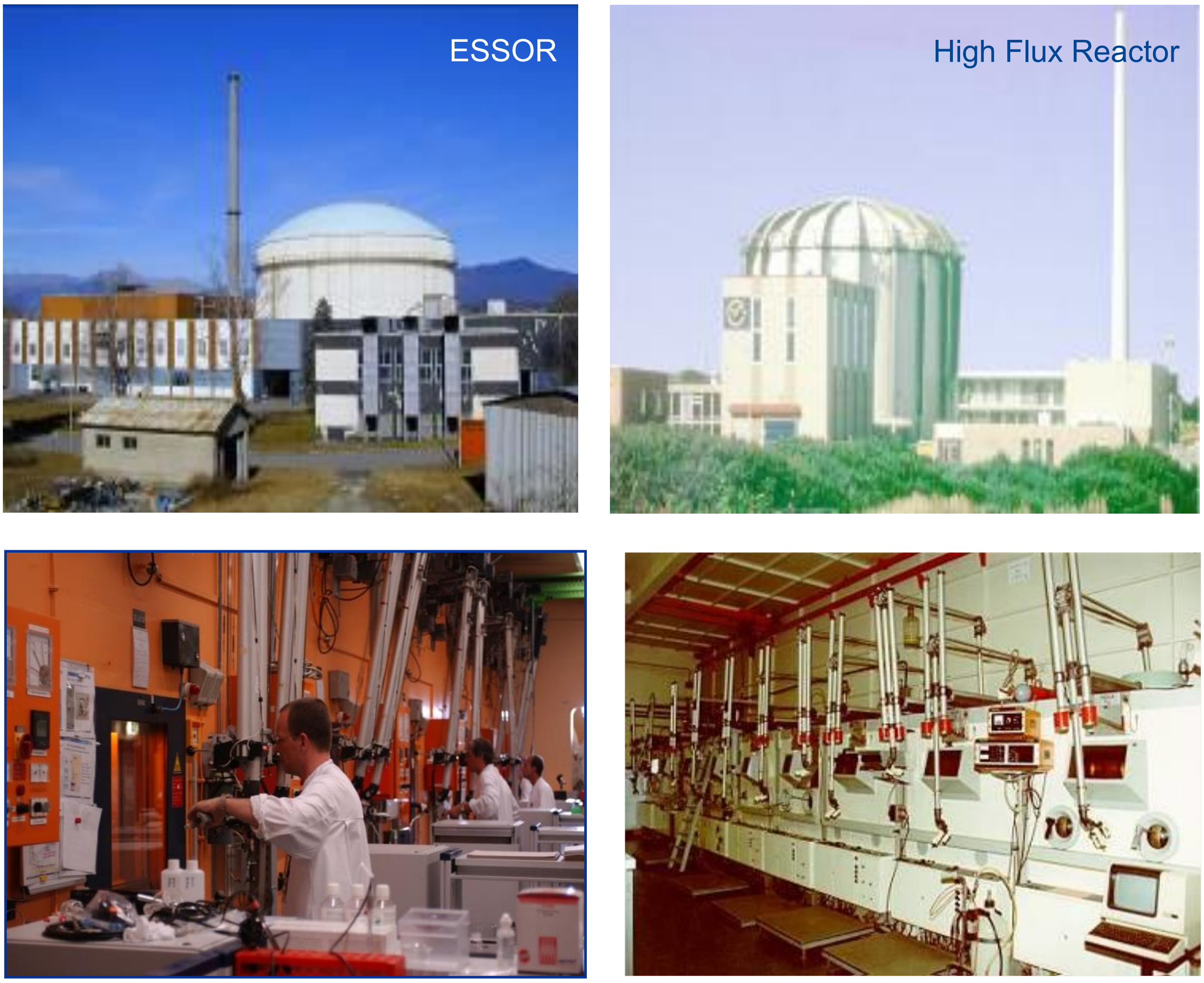

Figure P39**:** The JRC D&WM programme: Examples of facilities located in the different sites of the JRC. Top: ESSOR and HFR reactors located in JRC-Ispra and JRC-Petten. Bottom: View of Hot Cells laboratories in JRC-Karlsruhe and JRC-Ispra.

Transportation capabilities of hot cell facility

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The purpose of this paper is to present new transportation cask TERA 300. The paper focuses on designing, testing and approving of this cask. This cask was designed as a part of the new hot cell facility. Design of this cask was made by ŠKODA Nuclear Engineering in cooperation with Research center Rez and company ÚJP. The main goal was to create cask with reasonable dimensions, good shielding abilities, sufficient inner space and great variability of use.

TERA 300 is designed to be used for transport irradiated samples from variable materials (e.g. steel, concrete, ceramic) with activity up to 300 TBq for 60Co. The cask allows vertical and horizontal receiving with a great range of travel length. Due to the great variability of use, it is a complex machine with an electric driven travel of inner nest. Shielding material is made of lead, tungsten, and uranium.

The cask itself isn't hermetic, a hermetic function is ensured by inner hermetic capsule. The hermetic capsule is made of stainless steel. The capsule is designed in two dimensions, smaller and bigger one. The cask can accommodate 4 pcs of smaller capsules or 2 pcs of bigger capsules. Inner space of smaller version is sufficient to accommodate 1 CT sample. Capsules are designed for manipulation in the hot cell by manipulators.

The cask was produced in 2017. We made operational tests at the beginning of this year. First test was vertical receiving on the ceiling of the cells. We used non active samples for the purpose of this test. Second test was done on the manipulating stand and verify horizontal receiving. The purpose of the last test was verify shielding ability with high active source. This tests showed us some critical points and possible improvement of some parts on the cask.

The cask will be approved as a type B(U) from The State Office for Nuclear Safety at the end of the year 2018.

First transport campaign of new type B(U) packaging for hotlabs

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Research labs worldwide investigate radioactive materials that come from and go to nuclear power plants or research reactors. This can be nuclear fuel, irradiated material samples, instrumentation … all materials with different composition, dimension, weight, residual heat, fissile material. The transport of these radioactive materials often requires the use of a type B transport packaging according to the IAEA SSR-6 recommendations. To be able to fulfil this specific transport services, Transnubel has, amongst others, a new package design at its disposal: TNB 170.

Since the beginning of 2018, the new-build packaging TNB 170 has been in operation. This packaging was developed based on the experience gained during over 40 years of nuclear transports. TNB 170, a type B(U) packaging, is designed for the transport of fresh or irradiated UOX/MOX fuel, sealed or unsealed radioactive sources and neutron sources of type Xx-Be. The loading and unloading can be done vertically in or horizontally against a hotcell.

During the first transport of irradiated fuel samples, the loading of the TNB 170 was executed horizontally against a hotcell and the unloading was performed by placing the packaging vertically inside a hotcell. Specific tools and liners were designed and fabricated for executing the loading/unloading and transport operations in a safe way.

There have been a number of challenges that had to be solved regarding the design of the liner, interfaces between loading/unloading facilities, tools, transport frames … to guarantee a safe transport cycle. Meeting these challenges, especially in the environment of hotlabs with limited access and tools, is important for all future transports with the TNB 170.

Mechanical test of spent fuel at KAERI-PIEF

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Introduction

The PIEF in KAERI has been facilitating a wide range of research on the nuclear fuels irradiated at commercial nuclear power plants to evaluate the fuel integrity through hot cells and pool examination for more than twenty years. The PIEF consists of six hot cells to examine the spent fuel rods and to prepare mechanical test specimens, as well as hot laboratories to investigate the mechanical properties of spent fuel rod cladding and a spacer grid. In this paper, the mechanical tests for spent fuel conducted in the PIEF are introduced.

Mechanical test of irradiated cladding

To predict the behaviour of the fuel rod cladding during storage, particularly in interim dry storage, various types of tests, such as creep, hydride reorientation, fatigue, and compression, have been conducted. Prior to the cladding tests, the spent fuel pellet in the segment of a fuel rod is removed by the mechanical defueling and chemical dissolution. Furthermore, oxide layers on both ends of the cladding, to which the high-pressure fittings are connected, are removed completely to prevent a leakage of the internal pressure (Figure P40).

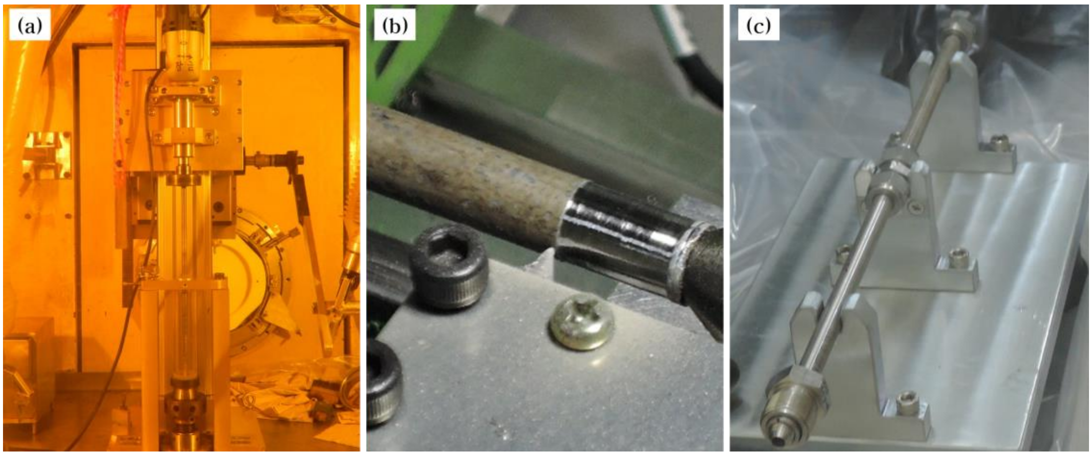

Figure P40: Test specimen preparation: (a) fuel removal, (b) oxide layer removal, (c) test specimen

The creep and hydride reorientation tests are performed with 150-mm-long specimens under internal gas pressurization. In each test, the temperature and hoop stress are set considering the conditions in dry storage. In particular, in the hydride reorientation test, the cooling rate and thermal cycling condition are also considered as major variables. The test equipment and one of the hydride reorientation test results are shown in Figures P41 (a) and P41 (b).

To evaluate the effect of creep and hydride reorientation on the mechanical property of irradiated cladding, a ring compression test is conducted. The test specimens are prepared to 10 mm in length from creep and hydride reorientation test specimens, and the ring specimens are heated to the target temperature and compressed in the radial direction at a rate of 1 mm/min. Figure P41 (c) shows the specimen after the ring compression test.

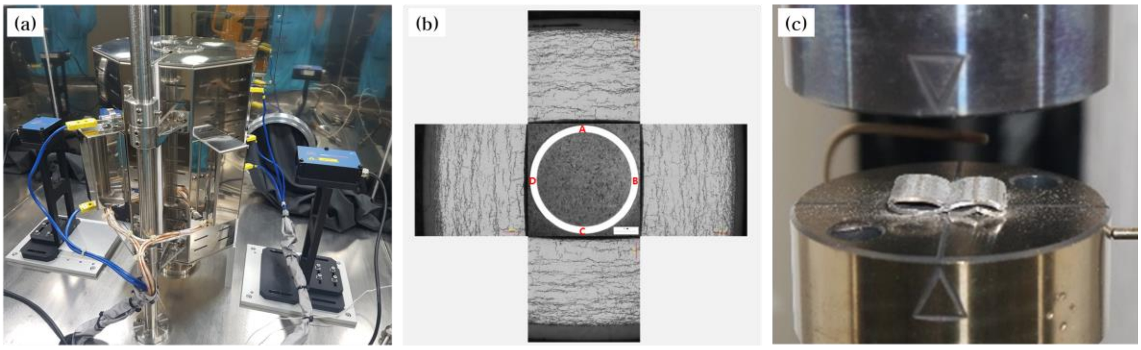

Figure P41: (a) Creep and hydride reorientation system, (b) metallography of irradiated cladding after hydride reorientation, and (c) ring specimen after ring compression test

Mechanical test of irradiated spacer grid

A spacer grid has a critical role in protecting the fuel assembly from the external impact load under severe accident conditions. Thus, it is essential to investigate the structural integrity of an irradiated spacer grid. A crush test, which is accomplished through the pendulum movement of the hammer to impact on the side of the spacer grid, has been performed to produce various buckling characteristic data, such as the impact strength and buckling mode. Figure P42 shows the crush test system and one of the test results.

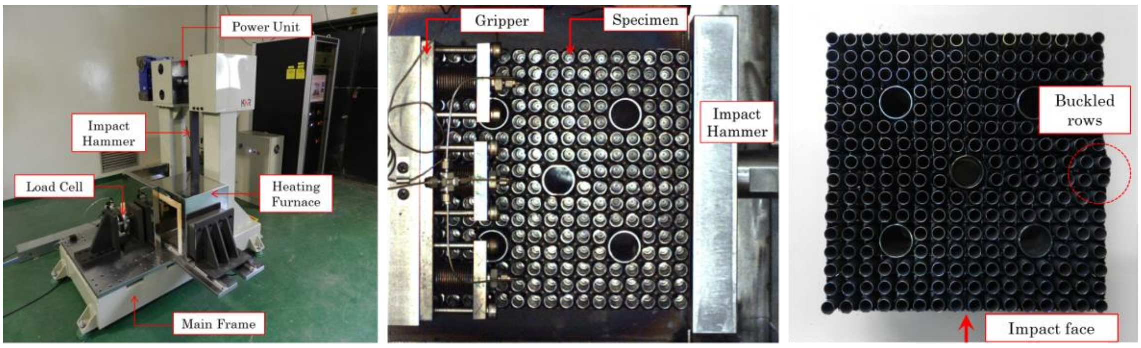

Figure P42. Crush test system for spacer grid and buckling mode shape of mid-spacer grid

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Abstract	The 55th Annual Meeting on Hot Laboratories and Remote Handling – HOTLAB 2018, took place in Helsinki, Finland from the 16th to the 20th of September, 2018, hosted by the Technical Research Centre of Finland Ltd. Reflecting on the contemporary nuclear scene in Finland, the program was arranged to follow the life-cycle of hot laboratory operation in support of the peaceful use of nuclear power. The first sessions described designing and building of new hot cell facilities. These were followed by sessions featuring examples of hot cell utilization, covering PIE of nuclear fuel and nuclear power plant structural materials, and by sessions that highlighted special equipment for use in hot laboratories. Some aspects related to the management of hot cell facilities were highlighted in dedicated sessions, including special sessions on the management of aging facilities. Finally, the extension of aging management to decommissioning and handling of radioactive waste were highlighted in the last sessions of the meeting, concluding with conditioning of high-level waste like spent nuclear fuel for final disposal. In all the HOTLAB 2018 included almost 50 oral presentations and over 20 poster presentations.
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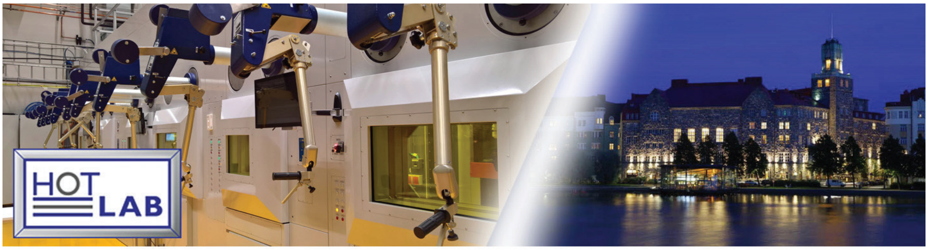

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