54th Annual Meeting on Hot Laboratories and Remote Handling

HOTLAB 2017

17-22 September, 2017, Mito, Japan

In cooperation with the
Atomic Energy Society of Japan

JAEA

In cooperation with the
International Atomic Energy Agency

IAEA
International Atomic Energy Agency
Atoms for Peace

Venue: Mito Keisei Hotel, 2nd floor
Technical sessions: meeting room “RURI”
Registration desk and exhibitions: RURI Foyer

Conference Program

Sunday, 17 Sep. 2017

15:00–19:00	Conference Registration	RURI Foyer
17:30–19:00	Welcome Reception	RURI

Monday, 18 Sep. 2017

8:30–9:00		*Conference Registration*	RURI Foyer
9:10–10:40		**Opening session** Session chair: K. Minato (JAEA)	
9:00–9:10		Welcome and Opening Remarks	Y. Miura (JAEA)
9:10–10:40		**Post-Fukushima special session - #1** Session chairs: K. Minato (JAEA), A. Leenaers (SCK•CEN)	
9:10–9:40	***F-01***	Decommissioning of 1F – current status and plan	N. Saito (TEPCO)
9:40–10:00	***F-02***	Contributions to the decommissioning of Fukushima Daiichi Nuclear Power Station by JAEA Naraha Remote Technology Development Center	H. Daido (JAEA)
10:00–10:20	***F-03***	The outline of Japan Atomic Energy Agency's Okuma Analysis and Research Center (1) - The total progress of Labolatory-1 and Labolatory-2 -	T. Inoue (JAEA)
10:20–10:40	***F-04***	Hot cell investigation of irradiated fuel debris from the Three Mile Island unit 2 (TMI-2) reactor	D. Bottomley
10:40–10:55		*Coffee Break*	RURI Foyer
10:55–12:35		**Post-Fukushima special session - #2** Session chairs: M. Worley (DOE), M. Osaka (JAEA)	
10:55–11:15	***F-05***	Revisiting the TMI-2 core melt specimens to verify the simulated corium for Fukushima Daiichi NPS	M. Takano (JAEA)
11:15–11:35	***F-06***	Collaborative R&D for advanced remote analysis using pulse laser ablation and related technologies	A. Nishimura (JAEA)
11:35–11:55	***F-07***	Radiochemical analysis of rubble collected from around and inside reactor buildings at Units 1 to 4 in Fukushima Daiichi Nuclear Power Station	Y. Sato (JAEA)
11:55–12:15	***F-08***	Severe accident research activities at the CEA: Methodology and main insights related to source term quantification and fuel behavior	O. Dugne (CEA)
12:15-12:35	***F-09***	International collaborations at JAEA/CLADS toward decommissioning of Fukushima Daiichi NPP	T. Washiya (JAEA)
12:35–13:40		*Lunch*	RURI
13:40–15:40		**HOTLAB operation - #1** Session chairs: M. Jong (UKAEA), T. Washiya (JAEA)	
13:40–14:00	***O-01***	Impact of the Fukushima Daiichi(1F) accident on the nuclear installations in Petten, NL	P. Thijssen (NRG)
14:00–14:20	***O-02***	Damage on the JMTR hot laboratory by the 2011 Great East Japan Earthquake	A.Shibata (JAEA)

14:20–14:40	***O-03***	Current status of the Irradiated Materials Characterization Laboratory at INL with limited PIE microstructural characterization	B. Miller (INL)
14:40–15:00	***O-04***	Post-Irradiation Examination Capabilities of M1 Hotcell in IMEF	Y.J. Kim (KAERI)
15:00-15:20	***O-05***	Overview and current status of the experimental capacity of the LECI hotlab facility	C. Blandin (CEA)
15:20-15:40	***O-06***	Overview and status of the US Nuclear Science User Facilities (NSUF)	J. Kennedy (INL)
15:40–16:00		*Coffee Break*	RURI Foyer
16:00–17:40		**Waste and storage** Session chairs: O. Dugne (CEA), H. Chichester (INL)	
16:00–16:20	***W-01***	Multirecycling plutonium in fast neutron reactor: Advanced design in ASTRID fuel cladding	M. Gennisson (CEA)
16:20–16:40	***W-02***	Research and operational activities on waste management and decommissioning at JRC Karlsruhe and Ispra	V.V. Rondinella (JRC)
16:40–17:00	***W-03***	Experimental setup for hydraulic resistance measurements on spent nuclear fuel	G. Cornelis (SCK-CEN)
17:00–17:20	***W-04***	Safe reconditioning of nuclear fuels stored underwater	S. Milesi (CEA)
17:20–17:40	***W-05***	Lessons learned construction from DIADEM medium-level waste interim storage’s	H. Lagrave (CEA)
17:40–18:00		*Photo Session*	RURI
19:00–20:40		*Conference Dinner*	RURI

Tuesday, 19 Sep. 2017

8:30–		*Registration Desk Open*	RURI Foyer
9:00–10:20		**HOTLAB operation - #2** Session chairs: M. Streit (PSI), K. Kurosaki (Osaka Univ.)	
9:00–9:20	***O-07***	UKAEA Materials Research Facility, into full operation	M. Jong (UKAEA)
9:20–9:40	***O-08***	Operational experience in hot cell transfer systems at Radio Metallurgy Laboratory	T. Ulaganathan (IGCAR)
9:40–10:00	***O-09***	BATAN - IAEA cooperation in the program of Decontamination and Post Irradiation Examination (PIE) in Radiometallurgy Installation Hot Cell - BATAN	B. Briyatmoko (BATAN)
10:00–10:20	***O-10***	Maintenances of the hot laboratory without operation of the system of air supply and exhaust	N. Nakamura (JAEA)
10:20–10:40		*Coffee Break*	RURI Foyer
10:40–12:20		**Facilities and equipment - #1** Session chairs: S. Woodbury (NNL), J. Buckle (Merrick)	
10:40–11:00	***E-01***	Facility development at JRC Karlsruhe for mechanical integrity assessment of fuel rods in view of transport and handling	J. Somers (JRC)
11:00–11:20	***E-02***	The ESS Active Cells Facility construction and design update	M. Gohran (ESS ERIC)
11:20–11:40	***E-03***	Fabrication and installation of VTT’s new hot cells	W. Karlsen (VTT)

11:40–12:00	***E-04***	Materials Analysis and Characterization Building, a new facility for observation of structural and fuel cladding materials of nuclear power plants	T. Sonoda (CRIEPI)
12:00–13:10		*Lunch*	RURI
13:10-14:30		**Facilities and equipment - #2** Session chairs: B. Miller (INL), J. Dexter (MBraun)	
13:10–13:30	***E-05***	The Hot cells ready – first results	O. Srba (CVR)
13:30–13:50	***E-06***	Shielded Focused Ion Beam Field Emission Scanning Electron Microscope (FIB-FE-SEM): Evaluation, technical modification and implementation	R. Brütsch (PSI)
13:50–14:10	***E-07***	Gained experience on the development of equipments used in high-activity cells	S. Martin-Vignerte (CEA)
14:10–14:30	***E-08***	Development of metal corrosion testing method simulating equipment of reprocessing of spent nuclear fuels	M. Matsueda (JAEA)
14:30–14:50		*Coffee Break*	RURI Foyer
14:50–16:10		**Remote handling technology - #1** Session chairs: V.V. Rondinella (JRC), A. Nishimura (JAEA)	
14:50–15:10	***R-01***	ESS Shielded casks' preliminary design and related monolith maintenance operations	L. Astrom (FIAB)
15:10–15:30	***R-02***	Remote target handling and radioactive isotope collection and handling for CERN's MEDICIS Facility	K. Kershaw (CERN)
15:30–15:50	***R-03***	Virtual X-ray vision by 3D scene reconstruction for work in nuclear containment	G.-J. Wei (UNSW)
15:50–16:10	***R-04***	An approach for remote nondestructive testing method for concrete structure using laser-generated ultrasonic	A. Furusawa (JAEA)
16:10–16:30		*Coffee Break*	RURI Foyer
16:30–17:30		**Poster short presentations** Session chair: M. Umeda (JAEA)	
		− All the poster presenters give two minutes short presentation with a slide (no more than 1 slide), which shows outline and highlight of the poster. − Questions and answers should not be made in this session but in the Poster session.	RURI
17:30–19:30		**Poster session**	
		− Posters should be set up on the board indicated with presentation number (P-xx) during lunch break on Tuesday 19. − Poster presenters are due for explanation and discussion on their posters. − Refreshments will be served during the session.	RURI

Wednesday, 20 Sep. 2017

8:30–		*Registration Desk Open*	RURI Foyer
9:00–10:20		**Remote handling technology - #2** Session chairs: L. Astrom (FIAB), T. Ogata (CRIEPI)	
9:00–9:20	***R-05***	Automation testbed for remote basket handling in a hot cell	D. Ryu (KAERI)
9:20-9:40	***R-06***	Automated device for pressure tube samples collect for hydrogen concentration determination in pile	S. Ionescu (ICN)

9:40–10:00	***R-07***	A 21st century remote manipulator: the MT200 TAO programme -Experience feedback and future perspectives	A. Guyot (Getinge La Calhène)
10:00–10:20	***R-08***	Additional payable safety features for remote handling technologies	J. Hedstueck (Wälischmiller)
10:20–10:40		*Coffee Break*	RURI Foyer
10:40-12:20		**Post-irradiation examination - #1** Session chairs: D. Gavillet (PSI), M. Hirai (NFD)	
10:40–11:00	***I-01***	Spallation material preparation using the hotcell facility, a new preparation box and focused ion beam for investigations from mechanical testing up to methods using synchrotron light	M.A. Pouchon (PSI)
11:00–11:20	***I-02***	Cleaning of failed Lead containing Zircaloy-2 Neutron Spallation Target Rods with a Dissolution Process	R. Zubler (PSI)
11:20–11:40	***I-03***	Post-irradiation Examination Using TEM Method for Swelling Evaluation of Baffle Plate in PWR Core Internals	K. Fujimoto (NDC)
11:40–12:00	***I-04***	Shear Punch Testing of Irradiated Cladding Materials from BOR-60 Irradiations	T. A. Saleh (LANL)
12:00-12:20	***I-05***	In situ Raman spectroscopy on nuclear materials in hot cell	S. Miro (CEA)
12:20–13:30		*Lunch*	RURI
13:30–14:50		**Post-irradiation examination - #2** Session chairs: Z.L. Peng (McMaster Univ.), S. Miro (CEA)	
13:30–13:50	***I-06***	ND-PIE on MTR fuel plates at SCK·CEN: a comparison with destructive analysis	Y. Parthoens (SCK/CEN)
13:50–14:10	***I-07***	Application of FE-SEM to the measurement of U, Pu, Am in the irradiated MA-MOX fuel	S. Sasaki (JAEA)
14:10–14:30	***I-08***	Post irradiation examination of fuel bundle from 540 MWe Pressurized Heavy Water Reactor (PHWR) of TAPS-3	J.L. Singh (BARC)
14:30–14:50	***I-09***	Analysis of fission gases released in the void volume of irradiated CANDU type nuclear fuel	M. Mincu (ICN)
14:50–15:10		*Coffee Break*	RURI Foyer
15:10–15:50		**Post-irradiation examination - #3** Session chairs: Z.L. Peng (McMaster Univ.), S. Miro (CEA)	
15:10–15:30	***I-10***	Microstructure Analysis of Irradiated NUE and NU Fuel in NPIC Hot Cells	Z. Fang (NPIC)
15:30–15:50	***I-11***	Spherical Fuel Element Deconsolidation System in INET	T.Wang (Tsinghua Univ.)
15:50–16:20		**Closing session** Session chair: K. Minato (JAEA)	
		Awards for Best Presentation and Best Poster	
		Next HOTLAB Meeting; HOTLAB 2018	
		Closing remarks	K. Minato (JAEA)

List of Poster Presentations

P-1	The outline of Japan Atomic Energy Agency's Okuma Analysis and Research Center (3) - Labolatory-2 -	M. Ito (JAEA)
P-2	Electrochemical corrosion tests for core materials utilized in BWR under conditions containing seawater	Y. Shizukawa (JAEA)
P-3	Development of analytical methods for radioactive waste samples from TEPCO Fukushima Daiichi Nuclear Power Station site at JAEA Okuma Analysis and Research Center	S. Sato (JAEA)
P-4	The outline of Japan Atomic Energy Agency's Okuma Analysis and Research Center (2) - Labolatory-1 -	Y. Sugaya (JAEA)
P-5	Safety aspects of fabrication of Americium-Plutonium Oxide pellets in a glovebox	J.D. Bruin (NRG)
P-6	A simple way for checking sampling representativity in nuclear facilities stacks	F. Minot (CEA)
P-7	Decommissioning program of Research Hot Laboratory in JAEA - Technical review of dismantling works for the Lead Cells Part 2 -	H. Shiina (JAEA)
P-8	Design of pseudo fuel debris fabrication equipment for critical experiment in converted STACY	F. Kobayashi (JAEA)
P-9	SKAPHIA: Shielded electron probe microanalyzer for radioactive samples	M.P. Moret (CAMECA)
P-10	Nuclear material transfer technologies and related operational activities at JRC Karlsruhe Hot Cells	L. Velnom (JRC)
P-11	First window replacement at the Hot Fuel Examination Facility	H.J.M.Chichester (INL)
P-12	Lab-scale continuous vitrification process for high level waste	I. Bisel (CEA)
P-13	Robot-assisted analysis procedure for hydrogen content determination of irradiated fuel cladding	H. Wiese (PSI)
P-14	High power ISOL radioactive target remote handling at TRIUMF	G. Minor (TRIUMF)
P-15	Development of precise manufacturing of irradiated miniaturized testing specimens at UJV Rez Hot Cell Facility	R. Kopriva (UJV)
P-16	Radioactive sample tomography using brand - new SPECT device designed by Research center Rez, Ltd.	P. Švrčula (CVREZ)
P-17	Development of reconstitution technique and testing of reconstituted bending bars in hotcell	H.S. Nolles (NRG)
P-18	Fabrication techniques of the sample supporting jigs for post irradiation examination with 3 dimension printer	H. Miyai (JAEA)
P-19	Radioactive materials post-irradiation examination at CANS	Z.L. Peng (McMaster Univ.)
P-20	Hydrogen content analyze in post-irradiation examination samples	G. Yi-fan (CIAE)
P-21	Irradiation Stability Study on Boron Carbide Reinforced Aluminums Matrix Neutron Absorbing Material	X. Hang (NPIC)
P-22	Wettability of Liquid CsI on Polycrystalline UO_2	H. Ishii (Osaka Univ.)
P-23	Gamma scanning of spent fuel element from nuclear power plant	Z. Xin-xin (CIAE)
P-24	Development of the electrochemical testing techniques in hot-cell	H. Tsuchihashi (NFD)
P-25	High temperature physicochemical properties of irradiated fuels	T. Ishikawa (JAEA)
P-26	ROBATEL Industries - Nuclear solutions provider since 1953.Hot lab Solutions	C. Dane (ROBATEL)
P-27	R&D at ROBATEL Industries - New materials for nuclear safety	T. Garnier (ROBATEL)
P-28	Enhancing MA transmutation by irradiation of (MA, ZR)H_x in FBR blanket - Fabrication of (Ln, Zr)H_x pellets -	M. Hirai (NFD)

Technical Tours (Thursday 21 and Friday 22 Sep., confirmed participants only)

Fukushima area	
8:10	Meet at Robby of Mito Keisei Hotel
8:30	Departure from Mito Keisei Hotel by bus
10:30-11:20	Naraha Remote Technology Development Center (JAEA)
11:40-12:20	Lunch
12:30-16:30	Fukushima Daiichi NPS (TEPCO)
18:00	Arriving at Mito Keisei Hotel

Tokai area		
9:40	Meet at Robby of Mito Keisei Hotel	
10:00	Departure from Mito Keisei Hotel by bus	
10:30-12:00	Nuclear Science Research Institute (JAEA Tokai)	
	Group A	Group B
	RFEF NSRR	J-PARC
12:00-13:10	Lunch	
13:30-15:00	J-PARC	RFEF NSRR
15:30	Arriving at Mito Keisei Hotel	

[Important information for Technical tour]

1. **You will need to bring your passport with you for the identification (original document only; copies cannot be used).**
2. Carrying cameras, camera-equipped cell phones, smart-phones or equipment which has a photography function is **strictly prohibited** in the sites (security measures for nuclear material protection).
3. Please be advised that we may not be able to return your belongings if contaminated.
4. Please wear **"long sleeves" and "long trousers"** to prevent any accidental adhesion of radioactive substances to your body.
5. Please wear shoes which are easy to walk in, and avoid sandals, high-heeled shoes and so on. In addition, for women, please refrain from wearing a skirt.

Abstracts

of

Oral Presentations

Post-Fukushima Special Session #1

Current Status of Fukushima Daiichi Nuclear Power Station after the Great East Japan Earthquake

Noriyuki Saito

Tokyo Electric Power Company Holdings, Uchisaiwai-cho, Chiyoda-ku, Tokyo, 100-8560, Japan,

Abstract

More than six years have passed since the Fukushima accident occurred. TEPCO is undertaking decommissioning of Fukushima Daiichi Nuclear Power Station (NPS) steadily and safely, incorporating domestic and international expertise, to fulfill its responsibility for the March 11, 2011 accident. This presentation shows overview of accident and the main progress of on-site activities.

A series of tsunamis which attacked on site 40minues after the Great East Japan Earthquake devastated Fukushima Daiichi NPS. As the result, reactor cores of Units 1 to 3 which were in operation when the earthquake struck damaged severely. In addition, Unit 1 to 4 were exploded by hydrogen produced by the chemical reaction between fuel claddings exposed from the water and steam.

Since the containment was broken through the accident, contaminated cooling water leaked out of the RPV and PCV and flowed into the basement of the turbine building through the reactor buildings.

To address this problem, a circulating water cooling system was established. In this system, contaminated water is reused for reactor water injection after cesium and salt have been removed using water processing facilities.

A cold shutdown state was accomplished in December 2011 owing to this system. This resulted in keeping the RPV temperatures below 100 degrees Celsius as well as the internal PCV temperatures.

TEPCO have been improving contaminated water management manners using Multi-layered measures. For example, multi-nuclide removal systems were started operation on March 2013 and groundwater bypass system and impermeable wall were also constructed.

About spent fuel management in spent fuel pool, all assemblies at Unit 4 were successfully removed on December 2014. Currently, preparation works for spent fuel removal from spent fuel pools at Unit 1-3 are progressing. For instance, at Unit 3, removal of large rubble from the spent fuel pool was completed in November 2015. Installation of a cover for fuel removal and fuel-handling machine is underway from January 2017.

For fuel debris removal from the damaged core, technologies development is conducted on national R&D project. In parallel, Investigation inside reactor building have been carrying out using by robotics applications.

Solid waste management is becoming notable issue. Solid waste generation have been predicted over the next 10 years based on the planned decommissioning activities. Based on this prediction, TEPCO made the waste management plan which showed how to improve waste management manner.

Improvement of work environment is essential to carry decommissioning work through.

Immediately after the accident, workers had to wear a full-face mask everywhere on site. Based on the progress of measures to reduce environmental dosage on site, workers don’t need to wear a full-face mask without highly contaminated area within Unit 1-4 buildings. Large rest house was started operation on May 2015. Workers can get a hot meal services in this building. On March 2016, the convenience store also opened.

TEPCO is moving forward with decommissioning activities keeping safety as the overriding priority while making every possible effort to facilitate early repatriation of disaster evacuees. TEPCO will also continue to share the information on its decommissioning status globally.

Contributions to the decommissioning of Fukushima Daiichi Nuclear Power Station by JAEA Naraha Remote Technology Development Center

Hiroyuki Daido[1], Shinji Kawatsuma[1], Hisayuki Kojima[1], Masahiro Ishihara[1], Shinichi Nakayama [2]

[1]*Naraha Remote Technology Development Center, Japan Atomic Energy Agency, Naraha-machi, Fukushima-ken 979-0513, Japan*
[2]*Fukushima Research Institute, Japan Atomic Energy Agency, 7-1 Taira-aza-Omachi, Iwaki-shi, Fukushima-ken, 970-8026 Japan*

Abstract

For the decommissioning of the Fukushima Daiichi Nuclear Power Station, the fuel debris and relevant materials should be retrieved under the highly dosed and contaminated environment. The key issues for safe retrieval of the fuel debris is utilization of remote technologies coupled with a mock-up which is a full scale model of a part of a reactor for decommissioning. At Naraha town, in order to promote development of the relevant remote technologies necessary for decommissioning and to provide a facility for remote technology experimentation as well as to support the revitalization of local industries, we have constructed a new facility which is open for all the contributors of the decommissioning of Fukushima Daiichi as well as of relevant activities of revitalization in this area. Actually the Naraha Center started the service on April 2016 as a user's facility where any domestic or foreign users can organize their own work according to their plan.

The facility includes a Mock-up Test Building which is large enough for installing a mock-up of a section of a suppression chamber belongs to Fukushima Daiichi unit 2 reactor. This mock-up is utilized for series of experiments on seeling leakage of water being performed by the International Research Institute for Nuclear Decommissioning (IRID). We have also constructed and organized robot test fields including a studio of motion capture, mock-up stairs and a robot testing pool feature in this Building. Using these, the users can make development of test methods for quantitative evaluation of performance of common robots for decommissioning as well as emergency response robots and so on. Each robot test field clarifies the necessary robot performance levels and the necessary operator skills. In the research management building, we have constructed and operated a cave type virtual reality system which provides us a virtual decommissioning work site for validating of robot designs, planning of decommissioning tasks and training of robot operators. The system also provides a lot of suitable work sites in which heavy preparation and training are necessary before doing actual work. Typical categories of the users are listed as follows; 1st, National project based on the utilization of a mock-up facility supported by the Government, IRID, TEPCO, and NDF etc. From the point of view of our original purpose, the successive mock-up project for continuous contribution to the decommissioning is indispensable. 2nd, decommissioning companies, 3rd, the projects organized or assisted by Fukushima Local Government, 4th, Universities & Colleges such as Fukushima University, Fukushima Technological College, the University of Tokyo and so on, 5th, local industries which directly contribute to revitalization of Fukushima as well as domestic and foreign industries.

At the same time, we are performing our own research and development using our facility. The subjects include the development of robot testing methods for the decommissioning of Fukushima Daiichi Nuclear Power Station and emergency response robots and relevant remote technologies which encourage and contribute to the facility utilization. Experts of remote technology and relevant fields at Naraha Remote Technology Development Center are happy to support and collaborate the facility utilization promoted mainly by users from all over the countries. We also hope that the utilization brings us new subjects of research and development which also push the progress of decommissioning technologies..

The outline of Japan Atomic Energy Agency's Okuma Analysis and Research Center (1)
- The total progress of Labolatory-1 and Labolatory-2 -

Toshihiko Inoue, Miho Ogawa, Yoshinori Sakazume, Hiroshi Yoshimoch,
Soichi. Sato, Shinichi Koyama, Tomozo Koyama, Shinichi Nakayama

Japan Atomic Energy Agency, Taira-Central-Building, 7-1 Omachi, Taira, Fukushima 970-8026, Japan

Abstract

Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station (1F) is in progress according to the Japanese Government's "Mid-and-Long-Term Roadmap" (Roadmap). Radiometric analysis of fuel debris and radioactive wastes such as contaminated rubble and secondary wastes from water processing is needed for the decommissioning. The Roadmap assigned the construction of a hot laboratory and analysis to the Japan Atomic Energy Agency (JAEA). The hot laboratory, "Okuma Analysis and Research Center", will be constructed near the 1F site.

The JAEA's Okuma Analysis and Research Center consists of the three buildings; Administrative building, the Laboratory-1 and Laboratory-2. The Administrative building provides office space and meeting rooms for the researchers, and has apparatus mock-up space. The construction began in October 2016 and plans an operational start in the spring of 2018. The Laboratory-1 and Laboratory-2 are hot laboratories. Laboratory-1 is for radiometric analysis of low and medium level radioactive rubble and secondary wastes. Laboratory-1 will provide the data needed to establish the strategy and methodology for treatment and disposal of the wastes, and mainly equipped with hoods. The license of the Laboratory-1's implementation was approved by The Secretariat of the Nuclear Regulation Authority and the construction started in April 2017 and plans an operational start in 2020. A number of analyst and engineer will be required for these plants operation, then it will be start the plan for the training of personal. Laboratory-2 provides concrete cells, steel cells, glove boxes and hoods for the analysis of the fuel debris and high level radioactive rubble. Samples from the 1F site are transferred into the concrete cells for preparations such as cutting and dissolution, and are measured the mechanical properties. The steel cells are for chemical separation and physical property measurement, and the glove boxes and hoods are for radiometric determination and chemical analysis. The Laboratory-2's major analysis items is reviewed by review meeting organized of cognoscente.

This paper total outlines Laboratory-1 and Laboratory-2, including the basic design, activities and progress. The details about Laboratory-1 and Laboratory-2 are reported the other papers in this conference.

Hot cell investigation of irradiated fuel debris from the Three Mile Island unit 2 (TMI-2) reactor

Paul David W. Bottomley[1],
J. Somers[2], D. Papaioannou[2], P. Pöml[2], S. Bremier[2], D. Serrano[2], S. van Winckel[2], D. Wegen[2] & V. V. Rondinella[2]

[1] Japan Atomic Energy Agency, CLADS Laboratory, 790-1 Motooka Ohtsuka, Tomioka-town, Futaba-county, Fukushima, 979-1151, Japan (formerly JRC-Karlsruhe)
[2] EC JRC-Karlsruhe site, Unit G.III.8, Waste Management, Hermann-von-Helmholtz Pl. 1, 76125 Karlsruhe, Germany

Abstract

The accident at Three Mile island, unit 2 occurred in 28th March 1979. This major civil nuclear power plant accident, was initially investigated by the US DoE, but was followed up with an OECD NEA/CSNI TMI-2 Vessel Investigation Program, managed by Idaho National Laboratories in which they extracted and selected samples to be sent to participating laboratories in Europe and Japan.
JRC-Karlsruhe (ex-ITU) received a range of samples for examination from the fully molten core in the centre to the debris found in the upper cavity. The results of these examinations and similar investigations carried out by other European hot cell laboratories are presented. The central melted core was a mixed $(U,Zr)O_2$ ceramic with ferrous oxide inclusions, while the agglomerate showed a wide variety of fused fuel, cladding and internal structural materials that had incompletely reacted. Debris was an equally wide range of fuel, cladding and other pieces. Rod fragments with little outer oxidation of the cladding demonstrated that the temperatures in the outer regions of the reactor remained low.
For the preparation of the abstracts of oral and poster presentations, the authors are asked to use the present document as a template and follow the recommendations for the style. Your abstract should not exceed 800 words as in the previous HOTLAB meetings.
More detailed examination using different techniques gave further very useful information as the reactions that had occurred during the meltdown as well as conditions (oxygen potential) and temperatures that were attained during the accident. The information and experience obtained here is also of relevance to the current Fukushima decommissioning and remediation.

Post-Fukushima Special Session #2

Revisiting the TMI-2 Core Melt Specimens to Verify the Simulated Corium for Fukushima Daiichi NPS

Masahide Takano[1,2], Atsushi Onozawa[1,2], Miho Suzuki[1,2], Hiroki Obata[1,2]

[1]*Japan Atomic Energy Agency, Tokai-mura, Ibaraki-ken 319-1195, Japan*
[2]*International Research Institute for Nuclear Decommissioning, Minato-ku, Tokyo 105-0003, Japan*

Abstract

For the decommissioning of severely damaged cores of Fukushima Daiichi Nuclear Power Station (1F), the retrieval operation of solidified core melt (corium) and its safe management are essential tasks. To understand chemical and mechanical characteristics of corium specific to the 1F cores, we have prepared and analyzed various types of simulated corium specimens in laboratory scale. The $(U,Zr)O_2$ solid solution with some minor solute elements is the main component of oxide corium. From the numerous experimental data on both as-melted and sintered specimens, we have found that the cooling rate considerably affects the phases, microstructure, and hence the mechanical properties of solidified state. Such effect is derived from the low mutual solubility between UO_2 and ZrO_2 at low temperatures. The rapid-cooled specimens from melt remain as single phase of cubic $(U,Zr)O_2$. In the sintered and annealed specimens simulating slow cooling conditions, on the other hand, the phase segregation into U-rich cubic phase and Zr-rich tetragonal phase occurs in micron scale in accordance with the existing UO_2–ZrO_2 pseud binary phase diagram. The micro hardness, which is one of the important properties to consider machining tools, is higher in the former single-phase structure.

To verify the effect of cooling condition found on the simulated corium, we revisit the actual corium specimens collected from the TMI-2 accident core, which have been stored at the Reactor Fuel Examination Facility (RFEF) in JAEA Tokai since 1991. We select totally six groups of pieces sampled from the molten-pool (2), upper crust (1), upper core debris bed (1), and lower head (2) regions. Polished cross-section of each piece is subjected to optical and SEM observation, elemental analysis by EPMA, X-ray diffraction, and micro Vickers hardness measurement. As a result of elemental analysis, 1 to 4 at% of Fe+Cr originating from stainless steel are dissolved in the $(U,Zr)O_2$ matrix of all the pieces. The O/(U+Zr) ratio is not far from the stoichiometric value judging from lattice parameters, except for the highly oxidized lower head debris.

An interesting contrast is that the remarkable phase segregation is observed in the molten pool debris, where the cooling rate was considered quite low because of the crust layer working as insulator. Figure 1 compares SEM images of the TMI-2 corium (upper crust and molten pool) and the $(U,Zr,Gd,Fe)O_2$ simulated corium specimens. Rapid-cooled specimens (upper row) have dense microstructure and consist of single phase of cubic structure. On the other hand, the slow-cooled specimens (lower row) consist of U-rich cubic and Zr-rich tetragonal phases distributed minutely. Micro-hardness of the rapid-cooled specimens ranges from 12 to 14.5 GPa, and that of the multiple phase microstructure ranges from 8 to 11 GPa depending on the degree of minuteness. From these observations we have confirmed the similar dependence of microstructure and mechanical property on the cooling condition, even though the elemental composition is somewhat different between the TMI-2 corium and the simulated corium for 1F cores.

This work was carried out under the “Project of Decommissioning and Contaminated Water Management under FY2013 and FY2014 Supplementary Budget” subsidized to IRID.

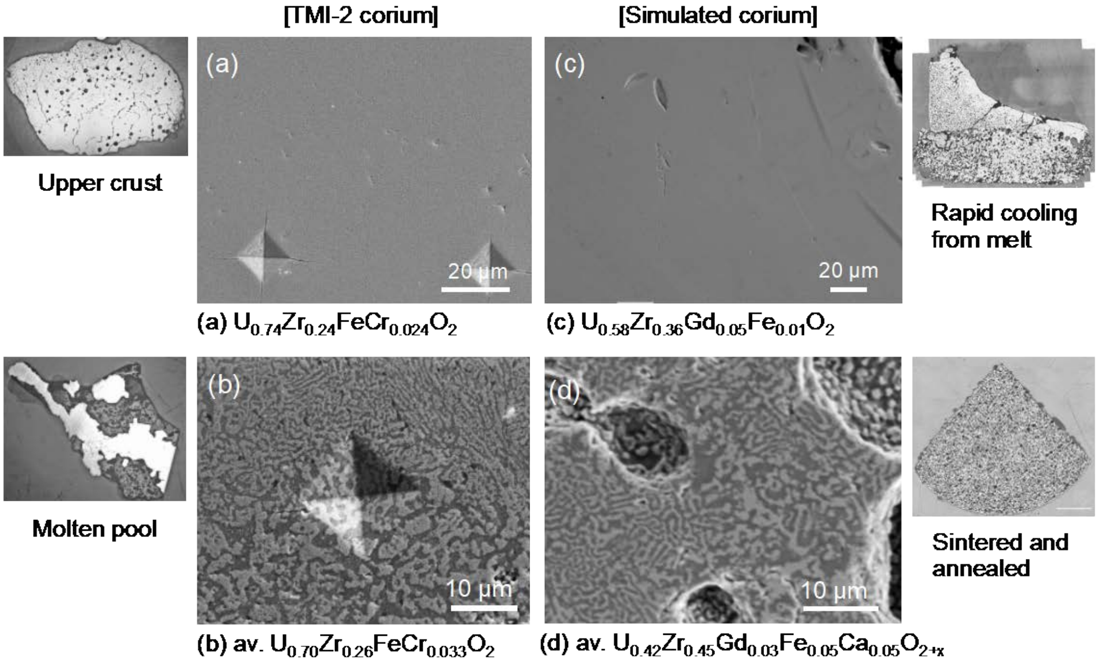

Figure 1 SEM images of polished cross-section of the TMI-2 corium (left) and the simulated corium debris (right). Images (a) and (c) correspond to rapid-cooled microstructure, while (b) and (d) slow-cooled microstructure with multiple phases.

Collaborative R&D for Advanced Remote Analysis using Pulse Laser Ablation and Related Technologies

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Abstract

We have been working over 7 years in Tsuruga site of Fukui to promote laser processing techniques for monitoring and maintenance of NPPs, accompanying with the Human Resource Development. Since the Great East Earthquake of Japan In 2011, decommissioning Fukushima Daiichi NPPs had been one of the most important issues for us [1]. Collaborative activities had started immediately after the accident in each establishment and all sectors of JAEA. Some special teams consisting of different research expertise were organized in the Sector of Fukushima Research and Development of JAEA. One of the teams challenged to develop the remote probing technique and a compact probing device in order to monitor the damaged reactor core of Fukushima Daiichi NPPs, applying pulse laser spectroscopy and telescope observation with radiation resistant optical fiber [2].

For Fukushima reconstruction, Naraha Remote Technology Development Center was established in 2015. The location is 30 km southern direction away from the Fukushima Daiichi NPPs [3]. Summer vacation internship program was arranged in Sep. 2016. Figure 1 shows a bird's eye view of the Center where collaborative activities promoting laser processing research from Tsuruga site to Naraha site were introduced.

Fig.1: Bird's eye view of the Naraha Remote Technology Development Center

R&D for remote sensing technology for radioactive materials is under promotion in the Center. Laser ablation could be a core technology for this purpose. Figure 2 shows the relation of three major categories of R&D around laser ablation. The experimental setup is composed of a nanosecond YAG laser, focusing optics, a multi-channel spectrometer, a laser Doppler interferometer and a test piece of concrete sample. YAG laser pulses are focused on the concrete surface and laser induced ablation plasma is generated.

In Tsuruga site, we experienced that the performance of high-power quasi-CW fiber laser irradiation on concrete was investigated either in the downward or the upward direction. Melting expulsion was induced by recoil pressure evaporation and greatly enhanced by expansion via gaseous bubbles and breakup. The performance of laser induced melting exclusion was significantly enhanced in the upward direction by the assistance of gravity. However, the performance by nanosecond YAG laser irradiation shows completely different aspect in comparison with that by quasi-CW fiber laser irradiation. Higher recoil pressure of ablation plasma, several hundred MPa, is generated with a few nanosecond laser pulse duration. Flashing fluorescence and ejecting decomposed fine particles are accompanied with the ablation plasma. Thus, no melting glassy product is found around a laser spot. Micro particles in the ablation plume rain down on a slide glass for microscope observation. The slide glass is coated with UV curable resin to make the micro particles confined tightly.

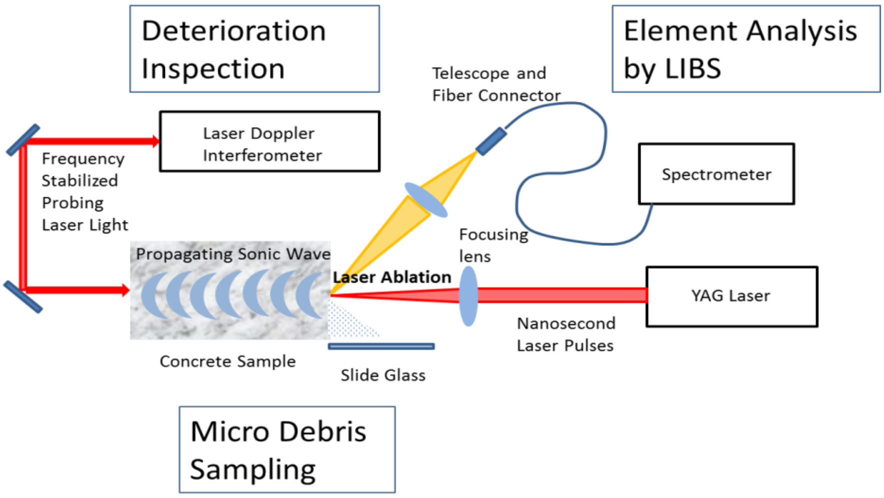

Fig.2: Three major categories of R&D linked with laser ablation

In the abovementioned Fig. 2, the following three major categories of R&D linked with laser ablation are shown schematically. For the inspection for nuclear fuel debris of Fukushima Daiichi NPPs, the proof of principal of **Element Analysis by LIBS** using a radiation resistant optical fiber was finished successfully. To date, the robotic device which can carry a fiber coupled LIBS processing head into the reactor vessel is under planning.

1) **Element Analysis by LIBS** combined with a telescope and/or optical fiber delivery
2) **Deterioration Inspection** by laser Doppler interferometer and numerical simulation
3) **Micro Debris Sampling** for mass spectroscopy and other advanced diagnosis

Deterioration Inspection is now under development, a laser Doppler interferometer being prepared for heating effect of specially fabricated concrete samples. The remote detection of surface vibration can measure the speed of propagating sonic wave.

Micro Debris Sampling is under preparation. Laser ablation on the concrete samples is carried out. Figure 3 shows the fine particles of decomposed concrete which were captured on a slide glass and observed by a phase contrast microscopy. Around 20 μm diameter particles could be useful for LIBS while around 300 nm diameter ones are applicable for efficient ionization of induction coupling plasma.

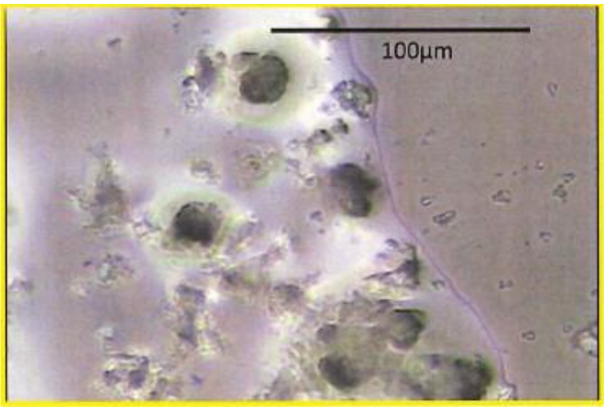

Fig.3: Fine particles of decomposed concrete sample.by laser ablation

In a hot laboratory of Okuma site, nuclear fuel debris and other contaminated objects should be handled and analyzed remotely. Pulse laser ablation and related technologies can contribute to not only reducing radiation exposure but also advancing analytical methods.

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Radiochemical analysis of rubble collected from around and inside reactor buildings at Units 1 to 4 in Fukushima Daiichi Nuclear Power Station

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Abstract

Fukushima Daiichi Nuclear Power Station (F1NPS) was severely damaged by hydrogen explosions resulting from the Great East Japan Earthquake that occurred on 11 March 2011. After the accident, radionuclides, including ^{137}Cs and ^{90}Sr, were released from reactor buildings at Units 1 to 3, and contaminated large amount of rubble. To examine a strategy for treatment and subsequent disposal of the rubble, it was essential to clarify their radionuclide and radioactivity concentration. In order to clarify the contamination of rubble, it is necessary that analysis of alpha and beta nuclides that are difficult to measure as well as analysis of gamma nuclides. In previous research, we developed analytical procedures for radioactive waste generated by research institute. In order to analyze rubble collected from around and inside reactor buildings at Units 1 to 4, we modified analytical procedures for the rubble and applied to it. In this study, from the radioactivity concentrations of the rubble, the characteristics of the contaminated rubble generated by the F1NPS accident are discussed.

Twelve samples were collected from around reactor buildings at Units 1, 2 and 4. Twenty four samples were collected from inside reactor buildings at Units 1 to 3. These samples were analyzed radioactivity concentrations of gamma-ray-emitting nuclides ^{60}Co, ^{94}Nb, ^{137}Cs, and ^{152,154}Eu, beta-particle-emitting nuclides ^{3}H, ^{14}C, ^{79}Se, ^{90}Sr, ^{99}Tc, and ^{129}I, and alpha-particle-emitting nuclides ^{238}Pu, $^{239+240}$Pu, ^{241}Am, and ^{244}Cm. To analyze 16 nuclides, these samples were ground and subdivided into 15 portions whose weight were every same grams. Pretreatments were chosen suitable method according to nuclides characteristics, pretreatments of analyses for ^{3}H, ^{14}C and ^{129}I were chosen combustion and gas trapping method, these of ^{79}Se and ^{99}Tc were chosen alkaline fusion method, and these of the others were chosen acid dissolution with microwave heating method. To quantify the radioactivity concentration of gamma nuclides, samples were measured by gamma-ray spectrometry using a high-purity Ge detector (HPGe, SEIKO EG&G CO., LTD.). Beta-particle-emitting nuclides except for ^{129}I were measured using a liquid scintillation counter (LSC, PerkinElmer, Inc.). Iodine-129 was measured using an inductively coupled plasma mass spectrometer equipped with a dynamic reaction cell (ICP-MS, PerkinElmer, Inc.). And alpha-particle-emitting nuclides were measured by alpha-ray spectrometry using a Si semiconductor detector (SSD, SEIKO EG&G CO., LTD.).

From the rubble collected from around reactor buildings, ^{3}H, ^{14}C, ^{60}Co, ^{79}Se, ^{90}Sr, ^{99}Tc and ^{137}Cs were detected. ^{3}H, ^{14}C, ^{60}Co, ^{79}Se, ^{90}Sr, ^{99}Tc, ^{137}Cs, ^{129}I, ^{154}Eu and alpha nuclides were detected from the rubble collected from inside reactor buildings. The rubble collected from inside reactor buildings were higher radioactivity concentration than that collected from around reactor buildings. Radioactivity concentrations of ^{94}Nb and ^{152}Eu were below than the detection limit, as a result of analyzing all samples. As shown in Figure 1, the radioactivity concentrations of ^{60}Co and ^{90}Sr in the rubble collected from around and inside reactor buildings were correlated with those of ^{137}Cs. Regardless of the sampling position, the ^{90}Sr / ^{137}Cs radioactivity ratios of Units 3 were larger than those of Units 1. These results implied that the radioactivity ratio of ^{90}Sr to ^{137}Cs depended on the accident for each reactor buildings.

The radioactivity ratio of ^{238}Pu to $^{239+240}Pu$ for Units 1, 2, and 3 were 1.8, 2.4 and 2.3 respectively. Those of plutonium ratio calculated by ORIGEN2 for Units 1, 2, and 3 in the reactor core were 2.9, 2.4, and 2.3 respectively. On the other hand the radioactivity ratio of ^{238}Pu to $^{239+240}Pu$ from global fallout were 0.019 - 0.068. Since plutonium ratios of analytical result were approximately equal to those calculated by ORIGEN2, we concluded that plutonium detected from rubble was released from F1NPS.

This work was carried out under the “Project of Decommissioning and Contaminated Water Management under FY2013 and FY2014 Supplementary Budget” subsidized to IRID.

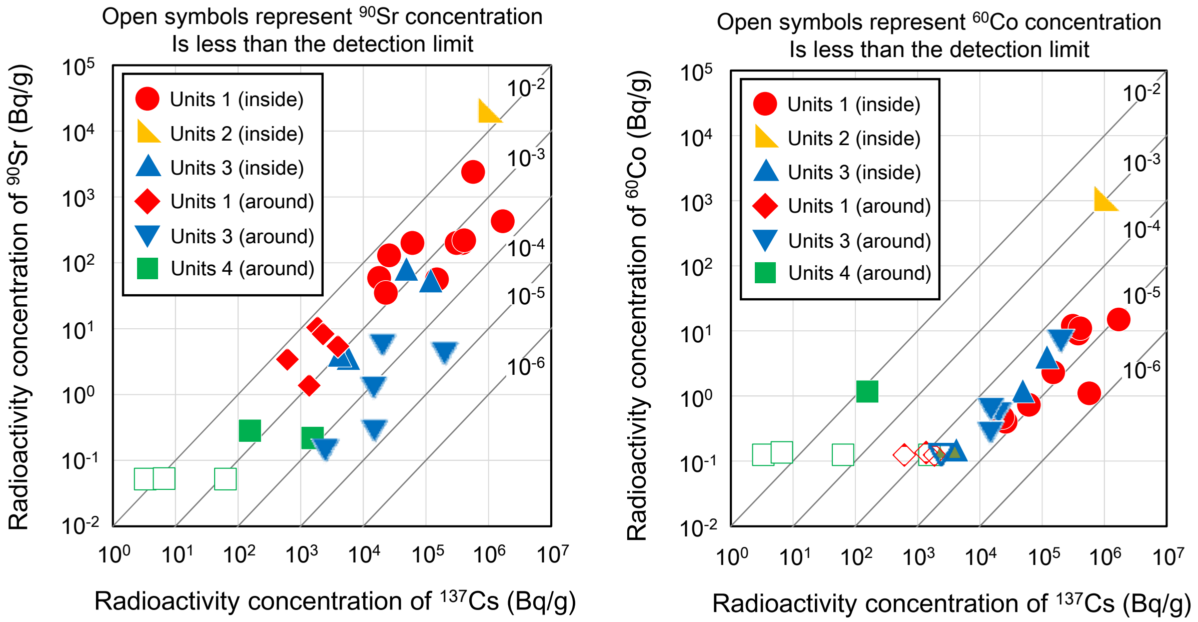

Fig. 1 The Correlation of concentration of detected ^{90}Sr and ^{60}Co as a function of concentration of ^{137}Cs.

Severe Accident Research activities at the CEA: Methodology and Main Insights Related to Source Term Quantification and Fuel Behavior.

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Abstract

Despite the very high level of control and safety in Nuclear Power Plant, they are not exempt from device malfunction, human errors or even natural disasters, which may lead to nuclear incidents or accidents. When consequences involve a high degradation of the reactor core (melting), associated with the release of Fission Products (FPs) and other radioactive materials from the reactor core, it is classified as a Severe Accident (SA, TMI-2, Chernobyl and Fukushima for instance). These released FPs may be transported by air masses or water and thus cover extensive areas and affect all living beings due to their radiological effect and may as well act as poisons.

The amount and isotopic composition of the radioactive material released from the core is called Source Term, and its assessment has been the main objective of several international research programs for more than thirty years. These research programs are commonly classified into two main groups, according to their approach: (1) Integral programs, such as PHEBUS FP, studying the response of a whole nuclear core during a severe accident, in a reduced scale; (2) On the other hand, analytical programs studying the fuel and FPs when submitted to accidental conditions, by means of Separate-Effect Tests (SET). Examples of the latter are the HI/VI, VEGA, VERCORS and VERDON programs. As result from all the research programs an extensive experimental database has been generated.

However, up to now, predicting correctly the FPs release from UO_2 and/or MOX fuels in SA conditions is still a significant and very important challenge since there are many remaining uncertainties. In order to improve these estimations, the global fuel and FPs behavior during the accidental sequence must be better understood, and specific emphasis has to be put on mechanisms which promote FPs release and fuel relocation. One of the most useful ways to do that is to perform appropriated annealing treatments with representative thermal transients in order to measure the absolute level and kinetics of the released FPs. To understand the promoting mechanisms, theses FPs release measurements have to be coupled with the corresponding fuel micro-structural changes resulting from these thermal transient.

To this end, since the last decade, CEA has set up two complementary research axes, aiming at reproducing conditions representative of nuclear severe accidents, using both high burn-up irradiated fuel samples and model materials. The first axis corresponds to the VERDON program and deals with commercial UO_2 and MOX fuels irradiated in French PWR. Model materials (often called SIMFUELS) consist in natural UO_2 doped with stable isotopes of FP in concentrations that match a targeted burn-up. Therefore, SIMFUELS are representative of irradiated nuclear fuels but without their radioactivity. The importance of such materials lies in the possibility of using powerful characterization techniques, such as X-ray Absorption Spectroscopy, which today are unavailable for large samples of irradiated nuclear fuels.

The present paper, organized in four main parts, presents successively the experimental

facilities available at the CEA Cadarache and Marcoule centers together with the corresponding R&D axes:

- SA experimental VERDON laboratory and associated annealing test device as MERARG,
- Analytical and micro analysis laboratories by which all the pre- and post-test fuels examinations, supported by analytical development on simulated corium samples, are performed
- Use of SIMFUELS methodology.
- The last part of the paper focuses on results obtained with this general approach, with special emphasis on VERDON-1 test.

International collaborations at JAEA/CLADS towerd decommissioning of Fukushima Daiichi NPP

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Abstract

6 years has passed from the severe accident at TEPCO's Fukushima Daiichi Nuclear Power Station (1F). For the decommissioning of damaged reactors, unit 1-3, restoration work is ongoing and inside of reactors state is gradually grasped by inside robot investigation. However, the most challenging task for retrieval of generated fuel debris is still remaining as great concern. The Collaborative Laboratories for Advanced Decommissioning Science (CLADS) inaugurated on April 1, 2015 will be the core of JAEA's research and development (R&D) on the decommissioning of 1F. The 1F accident was different from TMI-2 and Chernobyl accidents, and there are many unexpected challenging tasks ahead. Therefore, as internationally, gathering of the knowledge and expertise on the decommissioning and accident management is indispensable. For the international cooperation, the CLADS has been launched several collaborative activities as worldwide. The collaborative fields are expanding as following; radioactive waste management in the decommissioning (US), characterization of fuel debris and MCCI product (France), damaged fuel handling and treatment and storage of debris (Belgian), investigation on corium solidification process in the core degradation process(Czech), etc.. Also multinational cooperation as IAEA Coordinated Research Project and OECD/NEA projects on thermodynamic data base and fuel debris analysis are promoting within international communities. April 2017, JAEA has built Internal collaborative Research Building at Komioka-machi, near 1F site in Fukushima, where the CLADS is based, now.

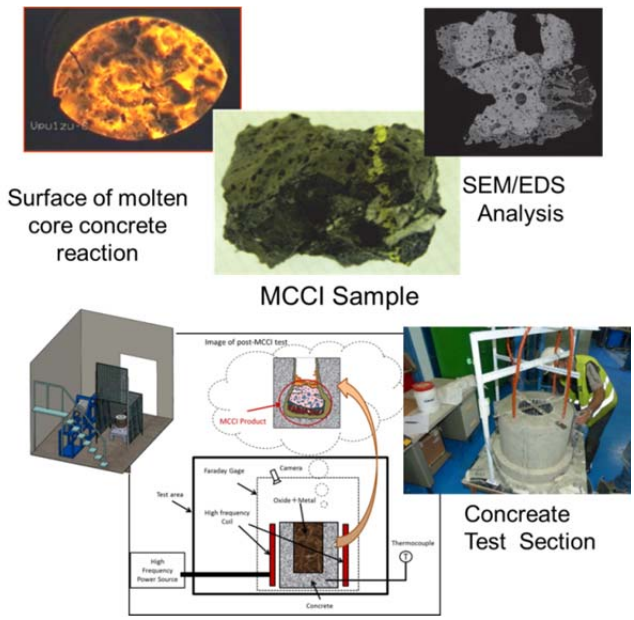

Fig.1 Joint study on engineering scale MCCI phenomenon and characterization at CEA/Cadarache

Fig.2 Internal workshop for decommissioning at JAEA/CLADS

HOTLAB operations #1

Impact of the Fukushima Daiichi(1F) accident on the nuclear installations in Petten, NL

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Abstract

The Fukushima Daiichi(1F) accident has had a worldwide impact. In the Netherlands Complementary Safety Assessments were carried out for all nuclear facilities in order to find opportunities to increase in the robustness of the facilities to extreme situations, beyond their existing safety margins.
NRG has carried out a comprehensive improvement program involving amongst others cliff-edge evaluations of the impact of seismic activity, flooding and extreme weather conditions. These evaluations resulted in proposed measures that largely have been completed. In this presentation examples are given of the measures, showing that in the aftermath of the accident lessons are learned and that our nuclear installations have become safer.

Damage on the JMTR hot laboratory by the 2011 Great East Japan Earthquake

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Abstract

An earthquake with magnitude 9.0 hit eastern Japan on 11th March, 2011. It was the most powerful earthquake since records began in Japan. The earthquake was named “The 2011 earthquake off the Pacific coast of Tohoku” by Japan Meteorological Agency and it also known as “the 2011 Great East Japan Earthquake “. The Japan Material Testing Reactor (JMTR) located about 290 km away from the epicenter. An intensity of the earthquake at the Oarai-town where JMTR located was upper 5 on Japanese seismic scale and the seismometers installed in JMTR recorded 294 cm/s^2.

The JMTR hot laboratory was damaged by the earthquake. This paper reports various damage on the JMTR hot laboratory caused by the earthquake.

The JMTR hot laboratory has three kind of hot cells; steel cells, lead cells and concrete cells. At the concrete cell, an electric lock which closes the shielding door of a hot cell was broken. Cracks were found in the wall of hot laboratory building.

The exhaust stack of JMTR Hot laboratory is a part of gaseous waste treatment system. It was built in 1970 and is 40 meter in height. Thinning was found at some anchor bolts on base of the stack in 2015. When thinning of anchor bolts were investigated, gaps between anchor bolt nuts and a flange plate were found. A cause investigation for the thinning and the gaps were performed. It was concluded that the thinning was caused by water infiltration over a long period of time and the gaps were caused by elongation of thinning part of anchor bolts by series of earthquakes start from the 2011 Great East Japan Earthquake.

Current Status of the Irradiated Materials Characterization Laboratory at INL with Limited PIE Microstructural Characterization Results

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Abstract

The Irradiated Materials Characterization Laboratory at Idaho National Laboratory is a post-irradiation examination (PIE) facility focused on high end microscopy enclosed in shielded enclosures. Planned PIE activities include microscopy, using FIB, EPMA, and TEM, and thermal property measurements. An overview of the facility, its operational status, and its future growth will be provided. A detailed discussion of the shielding/confinement surrounding the PIE equipment will be presented. The shielded sample preparation area will be presented focusing on instruments being incorporated to achieve high quality sample finishes for microscopy and thermal property measurements.

With various instruments of IMCL operational, PIE results of irradiated fuels using the EPMA, FIB, and TEM will be provided. A detailed explanation of the capabilities of the PIE equipment will be presented and how these instruments are being incorporated in preforming high quality microstructural characterization on highly irradiated materials.

Post-Irradiation Examination Capabilities of IMEF M1 Hot Cell

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Abstract

IMEF (Irradiated Material Examinations Facility) in Korea has been conducting post-irradiation examinations (PIE) on materials and fuels irradiated in the HANARO research reactor and commercial reactors.

IMEF has operated regularly since 1993. The hot cells in the ground floor consist of 6 concrete cells (M1 ~ M6) and 1 lead cell (M7). One hot cell (M8) is placed in the basement for ACPF (Advanced spent fuel Conditioning Process Facility). In addition, a pool with a depth of 10m is located in the service area to handle transporting casks. Main functions of each hot cell are as follows:

- M1 ~ M4 line : Non-destructive tests, Capsule dismantling, Specimen preparation, Storage
- M5 line : Mechanical tests (Impact, Tensile, Facture, Fatigue), Dimension measurement
- M6 line : DUPIC (Direct Use of spent PWR fuel In CANDU) experiments
- M7 line : Microscopy, Hardness, Density, SEM
- M8 line : Pyro-process, Demonstration tests

Because R&D fuels irradiated in HANARO such as a coated particle fuel, a metallic fuel, a plate fuel and so on have various shape and characteristic, various equipment and jigs have been developed in IMEF. Especially in case of M1 hot cell in which non-destructive tests has mainly been performed, tests equipment such as a gamma scanning, an X-ray CT system, a fission gas analyzer, a vacuum furnace and so on have been developed and operated.

This paper focuses on post-irradiation examination capability of M1 hot cell and introduces the equipment with research examples.

Overview and current status of the experimental capacity of the LECI hotlab facility

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Abstract

This paper presents the experimental capabilities dedicated to the characterization of irradiated materials behavior in normal, accidental and storage conditions of the CEA Saclay LECI hotlab facility. The main objective of the LECI facility, that operates 45 hotcells, is to provide metallurgical, mechanical and physico-chemical data on irradiated materials (from fuel assembly, core structure, and vessel) in support to the operation of light water reactors as well as the development of future nuclear systems and research reactors. Specifically, a comprehensive review of the available equipment which can be used to characterize a broad range of properties of irradiated materials mainly for Gen2&3 / Gen IV nuclear applications is presented. These data are then used to identify normal, accidental or storage conditions behavior laws and models able to predict the in-service lifetime of the materials.

The mechanical properties are characterized through a complete set of mechanical testing such as tensile, creep and burst tests. Specific experiments have been developed in order to obtain the mechanical properties in axial, circumferential and biaxial conditions, or to simulate LOCA or RIA transients. Fine metrology tools give access to dimensional variations at the micrometric scale after various irradiation experiments performed in Material Testing Reactor (MTR), that address creep and growth under irradiation.

State-of the-art equipment, including optical microscopy with image analysis for hydride content and distribution, FEG-Scanning Electron Microscopy fitted with EDS, WDS and EBSD, Transmission Electron Microscopy and Electron Probe Micro-Analysis allow the fine characterization of key properties of irradiated materials.

A specific focus on on-going projects for implementing high temperature mechanical tensile testing equipment and specific metallurgical equipment (Tomographic Atom Probe, Focused Ion Beam SEM) which will be available in the very short term to characterize the materials at the atomic scale is also presented.

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

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Abstract

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 increase the access of the nuclear energy research community to the unique and specialized nuclear energy capabilities operating across the country and developed over many decades. These capabilities include nuclear test reactors, ion beam accelerators, hot cell post-irradiation-examination (PIE) equipment, synchrotron beam lines, and advanced radiologically qualified materials science PIE instrumentation. For any particular project, the NSUF can support the design and fabrication of an irradiation experiment, the transport of the experiment, the PIE activities, the analysis and interpretation of the data, and final material disposition. The NSUF is a unique user facility in the U. S. in that it is not a single facility but instead a consortium of a number of partner facilities each with its own capabilities and distributed across the country. Four reactors are now available through the NSUF including the Advanced Test Reactor (ATR) at Idaho National Laboratory (INL), the Massachusetts Institute of Technology Reactor (MITR), the Pulstar Reactor at North Carolina State University, and the High Flux Isotope Reactor (HFIR) at Oak Ridge National Laboratory (ORNL). The hot cell facilities accessible for high activity PIE and material handling include those at INL, ORNL, Pacific Northwest National Laboratory (PNNL), University of Michigan (UM), and, as our first industry partner, Westinghouse Materials Center of Excellence. Ion irradiations are now offered through either the Michigan Ion Beam Lab at UM or the Tandem Accelerator Ion Beam at University of Wisconsin – Madison (UW). The application of the Focused Ion Beam (FIB) for small sample preparation first employed in Idaho and used extensively in the Microscopy and Characterization Suite (MaCS) facility in the Center for Advanced Energy Studies (CAES) has opened up a diverse array of advanced materials science instrumentation that can perform studies on materials with low levels of radiation. Additional partners within the NSUF include facilities associated with the University of California – Berkeley, Purdue University, the University of Nevada – Las Vegas, and the Illinois Institute of Technology. The latter partner is an example that allows the NSUF to leverage other national user facilities, namely the Advanced Photon Source (APS) at Argonne National Laboratory (ANL) through the Materials Research Collaborative Access Team (MRCAT) X-ray beamline. All NSUF projects are awarded through competitive processes and are open to university, national laboratory, industry, small business, and international researchers.

Two NSUF initiatives undertaken to assist users in formulating projects and enable DOE-NE to better manage their capabilities and assets are of particular note. First is the Nuclear Energy Infrastructure Database (NEID) that was recently made available to the nuclear energy research community as a web-based searchable tool populated with a variety of linked information on instrumentation, facilities, and institutions comprising the capabilities of interest to the nuclear community. The NEID can be found at nsuf-infrastructure.inl.gov. The second initiative is to strongly enhance the Nuclear Fuels and Materials Library (NFML) that contains specimens and samples from the irradiation tests performed over the decades. The library is intended to reduce costs, avoid redundancy in irradiation tests, and secure irradiated fuels and materials for future studies as new ideas and instrumentation come into being. 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 NFML can be accessed through the NSUF website at nsuf.inl.gov.

Waste and storage

Multirecycling Plutonium in Fast Neutron Reactor: Advanced Design in ASTRID Fuel Cladding

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Abstract

In the frame of generation IV studies, multirecycling plutonium from spent fuel is proposed in sodium fast neutron reactor.
As part of the development of the ASTRID generation IV reactor prototype, specific studies have been carried out on the fuel assembly fabrication.
The work presented here concerns the development of technologies for the fabrication of fuel pins.
Different components must be assembled to prepare a fuel pin: a cladding, a spacer held within the cladding, a fuel pellet column, a spring to maintain the whole, upper and lower plugs, and a wire wound around the pin which ensures suitable spacing between all the pins when they are grouped in an assembly.
One of the first fabrication steps consists in accurately fixing the position of the spacer inside the cladding, this serves as a reference: it constitutes the lowest point for the fuel in the ASTRID core, and is the limit point for the free volume of the lower expansion area for fission gas expansion.
The ASTRID rod designers decided to use crimping to hold the spacer immobile within the cladding. During this operation, the cladding is deformed until it moulds to the shape of the spacer.
A device was specifically designed to achieve this process. The technique chosen consisted in squeezing a polyurethane ring around the cladding holding the spacer. Set up in a fixed, rigid structure, this flexible ring compresses the cladding as it is deformed. The entire crimping setup will be described.
An experimental qualification plan was carried out, in order to determine the optimal functioning parameters which would guarantee that desired rods specifications can be achieved.
During these experiments, we investigated the effect of:

- the ring material (type of polyurethane),
- the ring geometry (thickness and diameter),
- the ring hardness (70 and 90 Shore A),
- the pressure applied on it using a hollow hydraulic jack cylinder.

The spacers crimped in place in their claddings were characterized using:

- Measurements of the cladding exterior profiles, to validate their conformity to the specifications.
- X-ray analyses, to measure the minimum cladding thickness and the gap between the cladding and the spacer in the crimped zone, and to check spacer integrity and the absence of interstitial particles.
- Traction tests, to determine the minimum force leading to the spacer sliding within the cladding, as well as the force necessary for it to detach.
- Visual examinations, to check for the absence of any deposits, of marks or of scratches on the cladding.

All these test results enabled the appropriate ring geometry to be determined, as well as the operating conditions which met the specifications.
In addition, endurance tests were carried out in order to determine the number of crimping operations able to be performed successively with the same ring. The results of these experiments will also be presented.
Thus qualified, the crimping device will be implemented for the fabrication of mock rods which will be used on different test benches during the qualification programme for ASTRID fuel assemblies.

Research and Operational Activities on Waste Management and Decommissioning at JRC Karlsruhe and Ispra

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Abstract

Since July 2016, the Nuclear Decommissioning Department has been established in the Directorate for Nuclear Safety and Security of the Joint Research Centre (JRC). The Department includes a scientific unit located in Karlsruhe, Germany (Nuclear Waste Management) and an operational unit in Ispra, Italy (Nuclear Decommissioning). The scientific scope of the Department encompasses experimental studies on nuclear waste management, covering spent fuel/high level waste in view of storage and geological disposal, and on decommissioning and remediation, including characterization of debris from severe accidents for removal and conditioning. Moreover, the Karlsruhe unit operates the hot cells which constitute the hub for PIE of conventional and advanced fuel in JRC. The Ispra unit is dedicated to the operational decommissioning of a wide array of obsolete nuclear research facilities present on site. The facilities to be decommissioned include research reactors, hot laboratories (including hot cells), waste effluent stations and other installations; moreover, the operational decommissioning programme includes the retrieval and conditioning of old waste packages currently in shallow burial at the site and of other waste forms. This presentation illustrates the main activities in the Department and focuses on the interaction and possible synergies between R&D and operational decommissioning, in the perspective of future co-existence of relatively large scientific research and operational decommissioning activities at the same site.

Experimental Setup for Hydraulic Resistance Measurements on Spent Nuclear Fuel

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Abstract

A laboratory scale hot cell installation was developed for studying challenges in conditioning leaking spent fuel for safe mid-term dry storage. Regardless of the drying approach of spent fuel, axial gas flow through the fuel column is the most important physical fuel property, and it also links directly to various safety aspects, such as fuel behaviour in loss of coolant accident (LOCA). The installation was developed to measure the hydraulic resistance in spent fuel column, for both intact as non-intact, using argon, helium and water vapour. The design took into account several limitations of hot cell environment, such as small dimensions, manipulator operations, degradation of components under radiation, and overall safety issues.

Safe Reconditioning of Nuclear Fuels Stored Underwater

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ABSTRACT

A large majority of the nuclear fuels used for CEA research (from experimental reactors to power reactors) are stored underwater in ponds. This storage solution has the advantage of being able to easily remove the decay heat from the fuel while providing good radiological protection for the staff operating in the facilities.

On the CEA Cadarache site, the storage pond in the PEGASE facility is used for the interim storage of fuel casings. The safety review of this licensed nuclear facility led to the decision to remove these fuels to the CASCAD (CASemate de CADarache) facility specifically built for the dry storage of radioactive material.

Due to the incompatibility of the safety baselines for these two facilities, the PEGASE fuels stored underwater had to be reconditioned before storage in CASCAD

The STAR (*Station de Traitement, d'Assainissement et de Reconditionnement*) facility on the Cadarache site was used for these reconditioning operations, performed in its hot cells according to a suitable safety baseline. The fuels were reconditioned by type (MOX, UO2, graphite-gas, etc.), dried, inerted and placed in leaktight double-jackets casings which meet the CASCAD storage criteria: the containers produced at the end of the process by the STAR facility were transferred to the CASCAD facility in IR500-type nuclear transport casks.

At the beginning, it was necessary to qualify casings for the reconditioning of these fuels so as guarantee their leaktightness, particularly in the case of load drops. To meet the CASCAD facility's criteria a new process was then developed and qualified. This required designing and developing specific equipment and adapting them to the hot cell constraints (remote operation, remote maintenance, resistance to ionising radiation, and operation in neutral gas environments) for the acceptance and treatment of fuel casings having been stored underwater.
Being able to trace each operation of the entire fuel reconditioning process at the STAR facility was key to the success of this project. Traceability means that the progress of the reconditioning process can be known at any time regarding the technical aspects or the management of nuclear materials.

The presentation will explain the specificities of the equipment, as well as describe the different phases of the process and the constraints that have been taken into account. It will provide information on the results obtained and the experience feedback from this 5-year reconditioning campaign (from May 2011 to December 2016) which was carried out round-the-clock.

Lessons learned from DIADEM medium-level waste interim storage's construction

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Abstract

As part of its decommissioning policy, the nuclear energy division of CEA has identified materials whose high enough radioactive features authorize neither sustainable storage in existing facilities nor evacuation to a French operational geological disposal. Pending availability of definitive solutions, it was decided to explore the possibility to store these materials in new facilities, the subject of DIADEM project and the need for a strategic new interim storages construction Program.
As state-own R&D and D&D agency, CEA has developed a recognized expertise both in safety and management of highly active wastes which is an issue of major public concern and in the understanding of the strategic challenges in new nuclear facilities project management including scheduling, funding, resources controlling, contract management and installation performance.

A key point in the safety assessment was the methodology on demonstrating the robustness of the stored waste containers. This qualification process together with LGM and LIGERON companies leads to ensure the containment of the radioactive material in any situation. The facility is planned to receive 3 types of cylindrical stainless steel containers, with identical diameters and increasing heights. The sides' thickness enables the containers to resist handling hazards such as falls, and means higher corrosion resistance. Both drop tests and corrosion measurement's qualification are under development. Non-destructive corrosion detectors will be installed in the hot cell and will monitor the acoustic emission characteristics of the containers stainless steel in the corrosion process.

The lids will be screwed onto the containers in the waste-producer's facilities, and then welded in DIADEM's hot cell before storage. The decontamination process in the hot cell and the lid welding will guarantee a low contamination level for periods of several decades. For waste involving radiolysis phenomena, the lids will also be equipped with metal filters which will allow gases to escape. These special filters are designed to only give gaseous pressure release, while blocking all particles. To avoid corrosion issue on the filters due, a study selected a filtering media according to requirements of nuclear standards and DIADEM's operation. To ensure characteristic of filtering media, a test plan is built, that including accelerated aging studies, hydrogen diffusion rate, filter efficiency and pressure versus flow curves. A specific housing is also designed to adapt the filter to the lid and protect the media. One filter can complete all requirements for one containers, each container is equipped with 4 filters. Finally filters must be removable in order to change them during the storage, therefore remote handle tools come with the filter.

DIADEM's construction phase was entrusted to INGEROP Company who provided design and EPCM services on the design and the construction management for the facility. Nowadays, the civil works progress is about 70 percent, with an expected end of works in 2019. The final safety assessment is also expected next year.
This paper deals with the lessons learned since the start of the construction in 2013.

HOTLAB operations #2

UKAEA Materials Research Facility, into full operation

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Abstract

The Materials Research Facility (MRF) is designed to test irradiated materials for both the Fusion and Fission materials development programs and will be used by Culham Centre for Fusion Energy (also part of UKAEA) as well as industry and academia. In May 2016 UKAEA opened the MRF, a new purpose-built research facility which allows researchers from UKAEA, academia, industry and other organisations to investigate post-irradiation properties of both materials in service in today's nuclear power stations and candidate materials for use in future fission and fusion power stations. The new facility bridges the gap between activity levels that can be handled at university or industrial laboratories and activity levels that require large facilities at nuclear licensed sites.
With the MRF, the UKAEA is working on nuclear readiness for the coming decades. It aims to achieve the following goals:
Serve both Fission and Fusion research with input to future reactor types as well as existing reactor types
Invest in the development of test methods for micro- and macro-sized specimens as well as in international acceptance of those methods
Work actively to create a change in nuclear materials research by focusing on size reduction.

The MRF has the capability to receive and process activated materials with a maximum activity up to 3.75 TBq Co60 (or equivalent). The MRF hot-cell line (a receiving cell and three interconnected hot-cells), provides downsizing, mounting and polish samples. Samples can either be transferred to one of the research rooms for on-site experiments or be transferred to an external partner or customer. Two MRF research room lines have been developed to shield samples with a maximum activity of 3.75 GBq Co60 (or equivalent). Each research room line contains 5 research rooms, all operated and controlled remotely. In the research rooms, microscope techniques (as Scanning Electron Microscopy (SEM), Focussed Ion Beam (FIB), Atomic Force Microscopy (AFM) and Precision Ion-beam Polishing System (PIPS)), mechanical testing (static testing, high frequency fatigue and SEM in-situ testing) and thermo-physical techniques (Laserflash, Dilatometry and Simultaneous Thermo-gravimetric Analysis & Differential Scanning Calorimetry (TGA/DSC)) are available.
Furthermore, gloveboxes lines are installed to use for research on lower activity samples, e.g. for Sample preparation (cutting, polishing, electrolytical polishing), Tritium and Beryllium based research. A setup for Thermal Desorption Spectroscopy (TDS) and rigs for corrosion of tritium loaded materials and plasma behaviour at tritium and deuterium are being installed in the MRF.

In 2018-2020 the MRF is planning to increase its capabilities as follows:

- Increase of the sample preparation capabilities in the hot-cell line (EDM cutting, in-cell welding, dimple grinding, electrolytical polishing, etcetera).
- Doubling the hot-cell capacity by the addition of two more hot-cells for increased flexibility by use off interchangeable inner containments.
- Realisation of an additional research room line for future scientific instruments.
- Additional equipment for microstructural, mechanical and physical characterization.

Operational experience in hot cell transfer systems at Radio Metallurgy Laboratory

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Abstract

The α, β, γ hot cells of Radio Metallurgy Laboratory (RML) at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam, India has been in operation since 1994. This hot cell facility caters to post irradiation examination (PIE) of irradiated fuel and structural materials of Fast Breeder Test Reactor (FBTR). We have established 5 different routes for transfer of materials in and out of the hot cells. They are 1.Vertical Transfer System (VTS), 2. Horizontal Transfer System (HTS), 3.Rear side cell door opening, 4.Top Roof opening system and 5.Quick transfer system.

VTS is employed for transfer of 1661mm long irradiated subassemblies from Irradiated Fuel Storage Area (IFSA) of FBTR to the hot cell #1. Irradiated subassemblies kept inside the argon filled special pot at IFSA of FBTR is transferred to RML hot cell #1 through a 25m long underground trench linking the two buildings. The entire operations of fuel transfer are controlled remotely. The remote alignment of special pot with respect to mating components in the hot cell poses many challenges. The VTS has many safety interlocks to avoid accidental conditions during fuel transfer operation. This system has been successfully used for more than 50 transfer operations over last two decades.

For alpha tight transfer of fuel pins and cut sections with a maximum length of 540mm from hot cells to other facilities a horizontal transfer system is used. This transfer system consists of 5t cask operated on a trolley and transfer takes place in horizontal manner. Challenges faced during the operations of these systems have been solved without any breach of containment of the operating hot cells. The horizontal transfer system has been used for more than 100 transfer operations.

For transfer of high β, γ stainless steel hulls of the subassemblies a special transfer system has been devised based on the rear side cell door opening. It consists of 3 ton cask, mobile compensatory shield, trolley, 1.8m long container, turn table etc. This system and methodology was arrived at as an interim measure for transferring only β, γ waste.

Roof opening system is provided in each hot cell to facilitate the posting of new equipment, taking out of old unserviceable equipment and maintenance of equipment. The openings were adopted for minimally intrusive entry into the hot cell for repair of equipment.

In addition to the above, hot cells have a quick transfer system provided on the operating area side. Transfer of minor tools and consumables that need to be posted into the hot cells in a rapid and leak tight manner is undertaken using this system.

This paper discusses the salient features of various transfer systems established in the hot cell facility of Radio Metallurgy laboratory at Indira Gandhi Centre for Atomic Research (IGCAR), Kalpakkam. Operational experience gained and case studies also will be discussed.

Key Words: Hot cells, remote operations, fast breeder test reactor, fuel subassembly, post irradiation examination, transfer systems.

BATAN - IAEA Cooperation in the Program of Decontamination and Post Irradiation Examination (PIE) in Radiometallurgy Installation Hot Cell - BATAN

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Abstract

BATAN received expert assistance from the IAEA in cooperation of decontamination and post-irradiation examination program in Radiometallurgy Instalation hotcell - BATAN. Some of test equipments in the hotcell has undergone aging and therefore can not function anymore. For the test equipment that has been damaged and can not be repaired must be removed from the hot cells and will be replaced with new equipment. Hot cell decontamination activities has been carried out to remove the test equipment that has been damaged from the hot cells. BATAN staffs have conducted decontamination in concrete hot cells.

Through the cooperation with the IAEA, in 2015 BATAN has held an interregional workshop on various techniques of decontamination in hot cell facilities. Cooperation between BATAN and IAEA continues with the arrival of experts to Indonesia in April 2017 through the IAEA Expert Mission on the decontamination of hot cells and PIE program in Indonesia. The activities carried out in the form of a workshop on decontamination and post-irradiation examination in April 2017. BATAN – IAEA cooperation activities in the decontamination and PIE facility preparation program in nuclear fuel technology center - BATAN will be described in the paper.

Maintenances of the hot laboratory without operation of the system of air supply and exhaust

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Abstract

The JMTR hot laboratory is a facility to perform post irradiation examinations (PIEs) of samples which irradiated in the Japan Materials Testing Reactor (JMTR). The hot laboratory is connected with the JMTR by a water canal. Irradiated samples are transferred through the canal safely without any transportation cask to perform PIEs such as tensile test, impact test and IASCC test in the hot laboratory. However, thinning of anchor bolts of the exhaust stuck of the hot laboratory were found in 2015. The exhaust stuck was removed and all PIEs have been stopped in the hot laboratory. The system of air supply and exhaust have been stopped and negative pressure can’t be kept. To manage the hot laboratory safely under this condition, several maintenances of the hot laboratory have been carried out as follows. Gaps between inside cell and outside cell and between radiation controlled area and non-controlled area in the hot laboratory have been sealed up to prevent spreading of contamination. The sealing is checked every day. Surface concentration of contamination around the sealing is measured every week. These maintenances will be continued until a new exhaust stuck will be rebuild.

Facilities and equipment #1

Facility Development at JRC Karlsruhe for Mechanical Integrity Assessment of Fuel Rods in view of Transport and Handling

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Abstract

The mechanical integrity of irradiated fuel rods is of fundamental importance in view of their packaging, transport and storage. With interim storage times now expected to exceed 100 years, this key safety issue becomes all the more important. The JRC Karlsruhe has developed two new facilities to assess the response of irradiated fuel rods to mechanical solicitations transversal to the fuel pin. A unique capability is the handling of true spent fuel rod segments, with lengths up to 300 mm, and pressurized (Helium) at the nominal end of service levels. The first of these devices is configured in a classical three point bending arrangement, to measure deflection at the contact point as a function of a quasi-static load. The second examines the response of the fuel rod to the impact of an object falling vertically on it. In both devices, the rods are driven to rupture, permitting collection of released fuel for size distribution analysis, ultimately for the assessment of criticality in the event of fuel release (in transport casks). Both devices are extremely compact, having been developed as a set of components that were transferred into the hot cells and assembled therein using telemanipulators. This paper will present the innovative devices, as well as first results attained.

The ESS Active Cells Facility Construction and Design Update

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Abstract

The Active Cells Facility (ACF) at the European Spallation Source is currently being built in Lund, Sweden. The facility is built as a large hot cell with the purpose to protect workers from the hazardous environment resulting from processing of radioactive systems and components. The systems and components are originating from the ESS facility operation where neutrons are produced for science through a spallation process.

The ACF internal dimensions will be around 30 meters long, 12 meters wide and 15 meters in height. Within this volume there will be space for a process cell, a maintenance cell, an interim storage and airlocks for both workers and transports. The ACF static barrier structures is built with high density concrete of 1.3 meter thickness. On top of the inner parts of the facility, there will also be a control room, rack rooms and worker changing rooms. The technical galleries surrounding the ACF provides logistics and systems to be able to perform the tasks within the ACF.

Currently the construction of the ACF is ongoing, hand in hand with the detailed design of re-bars and casted in items like anchor plates etc. The ESS project is responsible for all cast in items in floors, walls and ceilings. All system design, manufacturing and installations are performed as an in-kind contribution from RACE – UKAEA, Culham, United Kingdom within the framework of 17 European countries contributing to the construction and eventually operation of the ESS facility.

This paper will report on the timeline for the project, overall layouts and logistical solutions, novelties in the design, there will for example not be any lead glass windows, current status of the construction, difficulties and lessons learned, as well as current status of systems and components development for the ACF operations.

Fabrication and Installation of VTT's new hot cells

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Abstract

New hot cell facilities have been installed in the VTT Centre for Nuclear Safety as a part of VTT's research and testing infrastructure renewal. The new hot cells are aimed at the mechanical testing and microstructural characterization of beta- and gamma-emitting materials of nuclear power plant structures. The design, construction and installation of the hot cells has been carried out on a turn-key contract by Isotope Technologies Dresden, GmbH. The design includes a suite of six hot cells on the main floor, located above a utility cell in the basement, as well as a light cell with a shielded glovebox. The main floor hot cells are arranged in two rows with a closed access space between the rows, from which the interiors of individual cells can be accessed. Internal containments are a key design feature of the cells. Several special features were incorporated into the design for VTT's particular needs. Several large pieces of equipment that were already existing at VTT needed to be accommodated into the design of some of the cells, and were also moved in as a part of the hot cell installation. These included an electric discharge machine, an electron beam welder, and a hydraulic mechanical testing device. Additionally, two new mechanical testing devices were procured by VTT and integrated into one of the hot cells by ITD as a part of the turn-key contract. A shielded, enclosed conveyor connects the two upstairs cell rows, and features a lift to transport test specimens from the basement utility cell to this shielded conveyor, with the primary purpose of conveniently transporting specimens between the basement and the two upstairs cell rows in a shielded manner. The utility cell features a two-directional transport cask docking system, enabling safe and secure docking of transport casks of different size and orientation (horizontal as well as vertical), with weight as high as 10 tonnes. For directly transporting heavy specimens to the main hot cells, each cell line has a bottom port matched to two different casks used for transports within the facility. This paper describes the construction and installation of the new hot cells and their equipment, with a particular focus on the specific technical challenges presented by the project, and how they were met through close collaboration between VTT, ITD and the subcontractors.

Materials Analysis and Characterization Building, a New Facility for Observation of Structural and fuel cladding materials of Nuclear Power Plants

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Abstract

For the stable operation and safety of nuclear power plants, maintenance of integrity of LWR materials is required. Under neutron, α, β, and γ irradiation conditions, the component materials such as structural materials of RPV, fuel cladding, concrete, and wiring materials of nuclear power plants tend to degrade as increasing fluence of neutron, α, β, and γ. Since these problems may cause the reactor to be shut down for a long period of time, it is necessary to resolve these problems urgently. In order to resolve these problems, the mechanism of several degradations should be clarified by means of several analyses such as microstructure observations, hardness, and hydrogen measurements.

In order to clarify the mechanism of several degradation of materials of NPPs, CRIEPI built a new facility, named “Materials Analysis and Characterization Building” in the CRIEPI’s Yokosuka-district. This building has a license to use radioactive isotopes (RI) and nuclear fuels. Though, because this building does not have full-fledged hot cells and pools, only a limited amount of nuclear fuel materials can be brought in. Main use of this building is to analyze RI and nuclear fuel pollutants materials such as de-fueled cladding, reactor pressure vessel, concrete, and cable insulating materials.

The characteristic of this facility is to install various analyzers necessary for microstructure observations, hardness, and hydrogen measurements. Representative devices to install in this building is as follows, 1) Three dimensional atom probe tomography (3DAP), 2) high resolution Cs-corrected TEM with EDS and EELS, 3) Field-Emission SEM, 4) EPMA, and 5) Focused Ion Beam. Previously, by means of 1) and another Cs-TEM, the dissolution of Fe ions from FeCr and FeNi metal precipitates in neutron-irradiated BWR cladding materials was observed clearly[1].

In this presentation, the features of each analyzers of this building and the outline of the findings that will be obtained from this new building will be reported.

[1] Takashi Sawabe, Takeshi Sonoda and Shoichi Kitajima, Proc. of WRFPM 2014, Paper No. 100139.

Facilities and equipment #2

The Hot cells ready – first results

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Abstract

Paper presents commissioning of the hot cell facility, which is a part of the project SUSEN (Sustainable Energy) at CVR (Research center Řež). This facility contains 10 hot cells and one semi-hot cell. They are divided to 8 gamma hot cells and 2 alpha hot cells. The hot cells and semi hot cell are equipped with experimental devices used for preparation and testing of irradiated materials. The facility contains technologies for a complex samples processing (cutting, welding, machining) and set of equipment for carrying out mechanical tests (sample preparation area, stress testing machine, fatigue machine, creep-testing machine, etc.) as well as to study material microstructure (microhardness and nanohardness tester, scanning electron microscope). Our facility allows work with radioactive samples with activity up to 300 TBq ^{60}Co and with dimension of 2 CT.

Paper is focused on finalization of the airtight boxes (airtight sliding door, sealing, second floor, adjusting of the technologies), tuning and testing of the support technologies (transfer device, active ventilation, liquid waste system, etc.). The transfer device consists from a big 25 tons crane and shielding cask, which is designated for a sample activity up to 300 TBq of ^{60}Co. The test with enclosed ^{60}Co source with activity of 300 TBq was made, as a proof of the shielding efficiency of the transfer device. The operation of this device is fully autonomous.

Early this year was started inactive testing of all technologies in the facility to prove sample preparation and testing abilities, also training of all necessary procedures (entering in the box, remote operations, work in the hermetic suite, etc.). The first testing of irradiated sample will be after fulfilling and proving nominal parameters of the support technologies. This test will be made in June of this year. The final protocol with all tested parameters and result of the test with first irradiated sample confirms that the hot cell facility is fully operational and ready to receive active samples.

Shielded Focused Ion Beam Field Emission Scanning Electron Microscope (FIB-FE-SEM): Evaluation, technical modification and implementation

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Abstract

Scanning electron microscopy (SEM) and energy dispersive x-ray spectrometry (EDS) are important analyses techniques in the PSI Hot-laboratory to characterize nuclear materials. Since 1992 the PSI Hot-laboratory operates the SEM DSM962 on low active irradiated materials, but with the raising operation time the maintenance of the electronic spare parts becomes more and more difficult and therefore the acquirement of a new instrument was indispensable. On the other hand the PSI research division of nuclear materials has the request of materials characterization in the atomic scale, which is possible at synchrotron beam lines. Due to statutory specified limited activity values at these facilities the production of tiny specimens is essential.

To combine the requests the new shielded analytical tool should be upgraded with a focused Ion beam (FIB) gun, a micromanipulator for in-situ sample preparation and the analytical capabilities upgraded with an integrated analytical system containing Energy Dispersive X-ray Spectrometry (EDS), Wavelength Dispersive X-ray Spectrometry (WDS) and Electron Back Scatter Diffraction (EBSD).

In addition the instrument should allow to handle highly radioactive samples like irradiated fuel and cladding. To protect human health and life and the sensitive parts of the instrument some modifications on the conventional purchasable instrument and detectors has to be constructed.

It was decided to purchase a Zeiss Crossbeam 540 fitted with a GEMINI II field emission electron optic and a Capella FIB column with liquid Ga ion source.

Secondary electrons (SE) are detected with an Everhart-Thornley-type chamber detector and with the in-lens SE-detector. For the detection of Back-Scattered Electrons (BSE) a YAG-detector and an in-lens EsB (Energy selective Backscatter) will be available.

The EDAX-Pegasus analysis system contains the “Octane SUPER” silicon drift detector for EDS, the high resolution “DigiView 5” camera for crystallographic analyses with EBSD and the TEXS WDS spectrometer.

To lift out and handle the prepared specimen in the vacuum chamber after ion cutting and polishing a highly precise “Oxford Omniprobe Autoprobe 200.2” micro manipulator will be available.

Since there is a limitation of floor loading in the laboratory, the construction of the hot cell was challenging. The dimensions of the lead shielding had to be large enough for a small glove box and the instrument attached with all the detectors, but in the same time small enough not to exceed the maximum floor loading allowed.

In order to fulfill these requests the instrument is installed in a cabin with a wall thickness of 10 cm lead, detectors are protected with shutters and the remote controlled loading system is realized with manipulators through an attached glove box, which allows safe handling of highly alpha/gamma radioactive and/or contaminated samples.

With this new installation it is planned to prepare tiny specimens for further analyses on various PSI-internal as well on PSI-external beam lines. The complex analysis system of the FIB/SEM itself allows qualitative and semi-quantitative elemental and crystallographic analyses of in-situ prepared surfaces and ion cut and polished open laid structures of any bulk material.

Gained Experience on the development of equipments used in high-activity cells

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Abstract

For many years the LECA-STAR facility in Cadarache has been carrying out high-quality programs in high-activity cells in various technical fields:

- R & D on irradiated fuel: metallography, annealing tests, micro / nanoscale examination
- Manufacturing of experimental rods,
- spent fuels processing and packaging (conditioning, chemical stabilization, …),

To design and build all the necessary equipment for these activities, LECA-STAR has chosen an organization that allows it to centralize the skills necessary for the development and realization of equipment, in order to lead the projects in terms of schedule and quality. These skills are gathered within the LECA-STAR engineering laboratory. This laboratory is composed of project managers responsible for designing, manufacturing and commissioning equipment intended for use in high activity cells. It also brings together various technical skills: mechanical engineering (CAD) and electrical / I&C, calculations (seismic, biological), planning, project monitoring, etc.

The aim of this presentation is to highlight the means and skills of the laboratory, its recent achievements and the experience gained during the past development of benches, on project methodology aspects on the one hand, and technical aspects on the other one.

Development of Metal Corrosion Testing Method Simulating Equipment of Reprocessing of Spent Nuclear Fuels

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Abstract

PUREX (Plutonium Uranium Redox Extraction) process is widely used in the commercial reprocessing plants of spent nuclear fuels. For separating and concentrating Uranium (U), Plutonium (Pu) and fission products (FP) from spent nuclear fuels, nitric acid solution is used. Nitric acid solution is heated up to boiling point to dissolve the oxide spent fuels, then the reprocessing solution contains a lot of oxidizing ions such as Ruthenium, Pu and Neptunium (Np) coming from FP and trans-uranium elements (TRU). Since the boiling nitric acid solutions containing the oxidizing ions are very corrosive, the corrosion attack occurs in the equipment of dissolver and concentrator dealing with heated solutions, which is made of metal materials such as stainless steel. From viewpoints of operation management and lifetime prediction of equipment made in stainless steels, the corrosion progress state and the corrosion rate of metal materials have to be grasped in the process solutions under operation condition of reprocessing plants.

We installed a corrosion test device inside an airtight type concreat cell which has containment capability to avoid internal exposure in WAste Safety TEsting Facility (WASTEF) which is a hot laboratory in Japan Atomic Energy Agency (JAEA). This device enables to use solutions containing alpha-ray emmission nuclides such as Np and nuclear fuel components, and to perform two types of corrosion test conditions at same time; immersion and heat-transfer. In case of the heat-transfer condition, the surface temparature of test pieces can be heated 20°C higher than the boiling point by contacting with the electric heater to simulate the super heating surface of metal over boiling point that accelerate corrosion. Moreover this device can change the boiling point in the test solution under reduced pressure conditions. It is possible to perform continuous corrosion tests more than 500 hours under control of temperature and pressure. The steam from boiled test solution can be condensed and refluxed using a cooling condencer at head of device. The test solution bath is made of Zr which has high corrosion resistant. The maximum capacity of test solution bath is small, that is 150mL, because the amount of Np and U for corrosion test solution becomes reduced. At the bottom of the Zr solution tank, there is a hole to storage ^{60}Co gamma ray source to simulate radiation intensity of high-level radioactive liquid waste. Therefore, resin parts of the device are made of poly ether ether ketone which has high acid resistance and high radiation resistance. The test metal samples can be easily removed from the solution bath to measure the weight or to observe the surface state after the corrosion test. And, this corrosion device enables to perform the electrochemical measurements such as a polarization curve measurement which is important to understand the corrosion mechanism, because the test sample is electrically insulated from the Zr bath. Moreover, the test solutions with oxdizing ions can be prepared by dissolving solid oxides such as NpO_2 into boiling nitric acid useing dissolution implements in glovebox. And surface observation after corrosion tests can be performed by a scanning electron microscope.

In WASTEF, it is possible to operate every examinations related with the corrosion research of materials.

Remote handling technology #1

ESS Shielded casks' preliminary design and related monolith maintenance operations

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Abstract

The European Spallation Source (ESS) in Lund, Sweden will be a 5 MW long pulsed neutron spallation research facility with planned commissioning 2020. The ESS Remote Handling System within the Target Station will be equipped with several different systems and functions. One of the systems, the shielded casks, shall ensure the safety, protection of workers, property and the environment from the effects of radiation during target monolith maintenance and the internal transport of irradiated monolith components.

Transports will be performed by the shielded casks system in the so called high-bay between foremost the main docking positions on the monolith and the active cells where the irradiated components will be further processed through operations as dismantling, separation and preparation for disposal.

The complete functionality of the shielded casks encompasses many subjects which include among others; radiation shielding, logistics, mechanical design and external and internal lifting equipment. All of which, have to be compliant with regulations in terms of handling of long-lived low and intermediate level waste.

The casks are defined as physical enclosures with their mechanical and electrical interfaces to high bay crane, active cells, mock-up and test stands as well as all internal target components within the monolith.

Target monolith components that will be handled by the shielded casks system are: proton beam window, moderator reflector plug, target wheel and shaft, proton beam instrumentation plug and the target monitoring plug. In order to access the components, adjacent internal shielding blocks must also be handled by the casks system.

Cask's internal remote handling and lifting devices includes a design with adequate precision to facilitate installation and removal of monolith components. Also, the different operations must be carried out in a specific and well planned sequential order.

This paper will mainly describe the preliminary design of the casks system. The paper will also present an overview of the concept for operational sequences for the ESS remote handling, internal transports and handling of irradiated target components.

Remote Target Handling and Radioactive Isotope Collection and Handling for CERN's MEDICIS Facility

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Abstract

A new medical research isotope production facility, "MEDICIS," is currently under construction at CERN. The new facility will use the leftover particle beam of the ISOLDE facility - after it has passed through ISOLDE targets - to irradiate additional targets to produce isotopes for medical research work.

In order to transfer, precisely position and store these additional targets, the isotope production process will use a new remote target handling and storage system. To ensure compatibility with radiation levels, which preclude the presence of electronics in the target handling and storage areas, the remote target handling system integrates modified versions of an industrial robot suspended from a linear axis mounted on the ceiling and industrial monorail transfer system working with custom–designed automated shielding doors, an air lock and remote handling cell.

The isotopes extracted from the irradiated targets also require shielded handling and transfers; for which a new system has been designed.

The MEDICIS facility will be briefly introduced, followed by a description of the target remote handling and storage system and the isotope collection and handling system design.

Virtual X-ray Vision by 3D Scene Reconstruction for Work in Nuclear Containment

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Abstract

The paper presents a low-cost and more interactive implementation of a Real-Time (RT) imaging system for 3-Dimensional (3D) scene reconstruction within a small-scale space in nuclear containment such as a hot cell. The system enables remote inspection and robotic manipulation, or alternatively increases usability and intuitiveness of interaction with objects in a hot cell if the operator is using mechanical master-slave manipulators. 3D scene reconstruction has been highly developed with a vast amount of open-source resources and free platforms associated with computer games development but the technology is not widely applied in the nuclear industry. In our system, we reconstruct a 3D scene on a completely free games developer platform Unreal Engine from depth images captured by low-cost depth cameras. By developing our system on Unreal Engine, we are able to render our reconstructed 3D scene in either 1st/3rd person perspective, or Virtual Reality (VR) forms. Interaction between system user and the reconstructed scene will be largely increased when the user can navigate freely inside the scene. We present a prototype system that is able to reconstruct and render in RT a full color 3D scene in Unreal Engine in high (2K) resolution. Crucially for nuclear containment applications, the depth cameras are in fixed locations, include redundancy, and do not require zoom/pan control. This technology promises to improve operator safety, increase the usability and flexibility of hot-cell facilities, and promises significant cost savings in conventional viewing options such as hot cell windows.

An Approach for Remote Nondestructive Testing Method for Concrete Structure Using Laser-generated Ultrasonic

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Abstract

Decommissioning of Fukushima Daiichi NPPs is one of the most important missions for JAEA since the Great East Japan Earthquake. The decommissioning will be carried out for several decades to come. Testing and monitoring of concrete structures in NPPs is needed in order to guarantee hereafter workability of decommissioning.
Recent work says that Core Concrete Reaction (CCR) advances erosion of the concrete structures of Fukushima Daiichi NPPs and it is difficult to estimate the correct depth of CCR because of undefined shape of the nuclear fuel debris and CCR's complex process [1]. In addition, it is clear that sea water intrusion makes the rebar in the concrete structures corroded gradually. Thus, advanced remote evaluation methods for the deterioration should be considered.
Classical concrete testing methods [2] are unsuitable for remote operation. These methods are fundamentally based on in-situ point-by-point operation by inspectors. Recently, new approach [3] is being reported by some researchers which is rapid, safe and suitable for remote operation, however, it concentrating on the surface cracks of the concrete structures, not on deterioration or corrosion inside.
Deterioration process of the reinforced concrete is shown in Fig.1. Gap or decrease of the adhesiveness between rebar and outer concrete appears in its process. We had a sense of possibility introducing new concrete testing method based on these gap or decrease of the adhesiveness. Concept of the new method is shown in Fig.2, propagated laser-excited ultrasonic gathering the information about the defect or corrosion inside and received at distance with LDV. In this work, we investigate and report how it has effect on propagating ultrasonic along the rebar to decrease adhesiveness between the rebar and outer concrete experimentally.

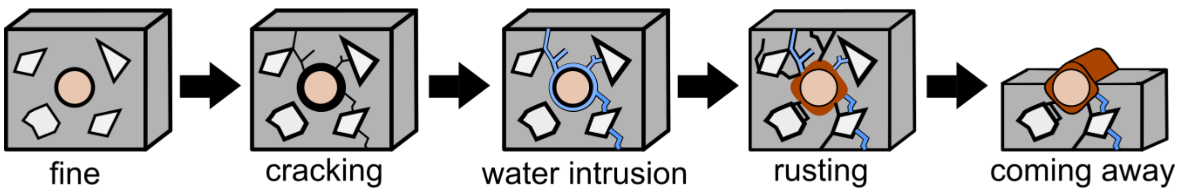

Fig.1: Schematic illustration of the deterioration process of reinforced concrete

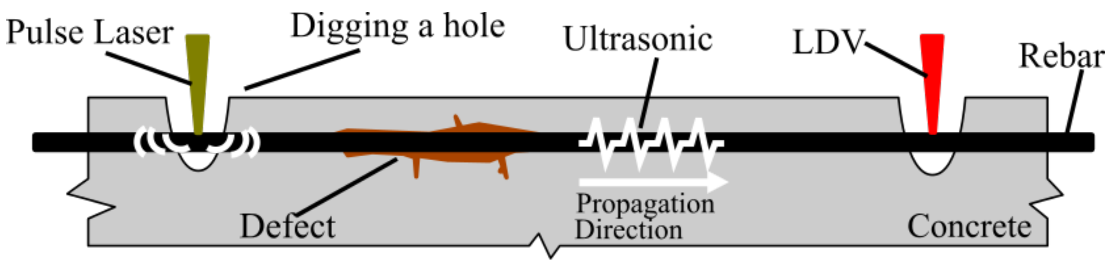

Fig.2: Concept of the remote nondestructive testing method for reinforced concrete structure

Experimental setup is shown in Fig.3. The system is composed of oscilloscope, ultrasonic receiver, nano-sec pulse laser source and test piece. Four kinds of test piece are prepared for this experiment: rebar (bare steel rod), fine test piece, heated and corroded. The bottom of the rebar to excite ultrasonic are irradiated by nano-sec laser pulses. Propagated ultrasonic wave is received at the opposite bottom to be analyzed in time and frequency domain.

Some experimental ultrasonic wave profiles are shown in Fig.4. Fig.4(a) shows the comparison between sound, heated and rebar specimen and Fig.4(b) is the collection of the ultrasonic signal corresponding to the corrosion degree. Remarkable feature seen in the Fig.4 is that wave profile of heated specimen and enough corroded specimen are good agreement with that of rebar. The reason of this behavior is that heat expansion of rebar or rust surface on the rebar decreases the adhesiveness between the rebar and outer concrete, in other words, the adhesiveness decrease gets those specimens close to the rebar. It is found that our approach that propagates ultrasonic on rebar in concrete has enough applicability to test and monitor the reinforced concrete structures via this experiment.

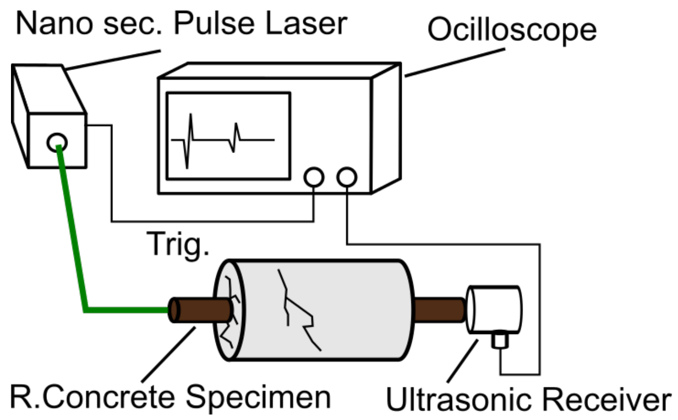

Fig.3 Schematic illustration of experimental method

Future work is to develop a portable in-situ testing device based on this method.

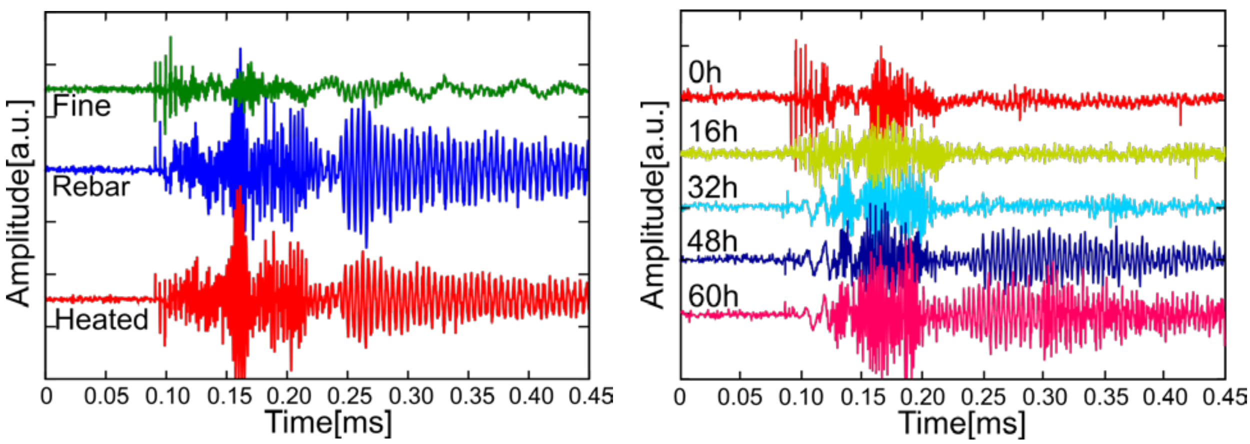

Fig.4 Ultrasonic wave profiles:(a) comparison between sound, heated and rebar specimen (b) collection of the corroded specimen corresponding to the corrosion degree. Numbers in picture mean energization time of electric corrosion.

References

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[2]D. M. McCann, M. C. Forde, “Review of NDT methods in the assessment of concrete and masonry structures”, NDT&E International, vol.34, pp71-84, 2001.

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Remote handling technology #2

Automation Testbed for Remote Basket Handling in a Hot Cell

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Abstract

Nuclear technology contributed to reduce use of fossil fuel and to supply plentiful energy. However, it also brought waste treatment issue. Various ways to reduce the toxic waste in nuclear spent fuel has been explored. Pyro-processing, one option for compressing down the waste by recycling 95 percent of Uranium from the spent fuel, has been extensively studied at the Korea Atomic Energy Research Institute (KAERI). For engineering scale experiments of pyroprocessing, PRIDE (PyRoprocessing Integrated inactive Demonstration facility) was constructed. A large argon processing cell (40.3m x 4.8m x 6.4) was built on the second floor to prevent the high reactivity and corrosiveness of using molten salt. The human operator, therefore, cannot be allowed to access inside the processing cell, and all processes were planned to be conducted by remote operation, as the same as a regular hot cell. Various remote handling systems, such as, mechanical MSMs (master slave manipulator), servo master slave system, crane, small and large transfer lock system, and etc., were installed in the processing cell. For two year operation of the PRIDE processing cell, the manual remote operation strategy caused controversy, and automation in a hot cell was issued for economic feasibility of pyroprocessing.

Preliminary automation concept for pyroprocessing was proposed at KAERI, and a draft evaluation was tried in a virtual environment with 3D CAD models. In the automation concept, the successful transportation of material from equipment to next equipment is most important for its evaluation, but the physical feasibility is questionable only with the virtual simulation. This research propose a physical testbed to verify automation feasibility of basket handling in pyroprocessing.

The automation testbed was proposed to use an overhead telescopic transporter system, conveyors, and a dummy equipment. The target handling objects are dummy baskets and electrode, and there are slots to insert the baskets and electrode on top of the dummy equipment. The given task in the automation testbed is to deliver the basket from conveyor to the specified slot on the equipment and to replace the electrode from one slot to the other. The evaluation scenario is summarized as shown in Fig. 1.

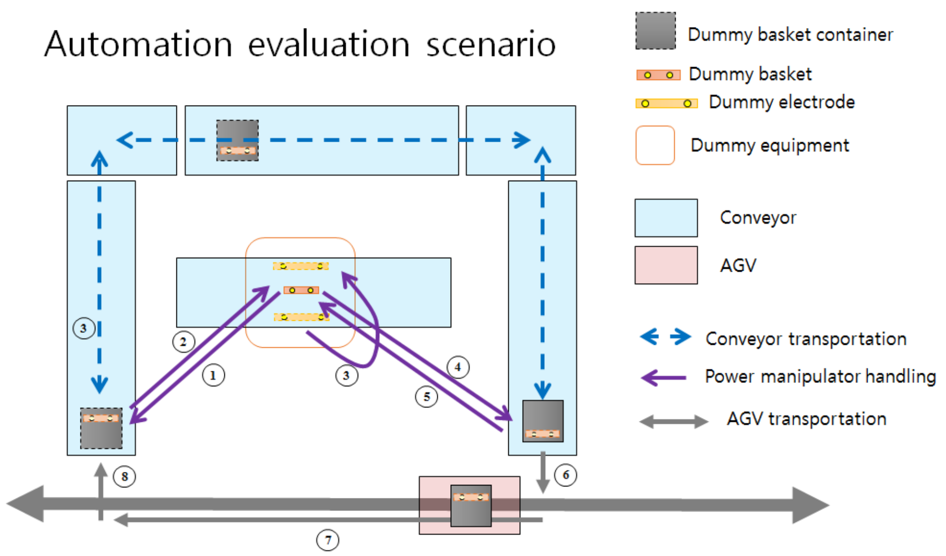

Figure 1. Evaluation scenario for pyroprocessing automation concept.

The overhead telescopic transporter system was integrated with modifying the Power manipulator manufactured by Walischmiller, the well-known German remote equipment manufacturer. The bridge and the telescopic tube of the Power manipulator were position controlled by laser distance sensors, and new end effector was equipped to have a two-point gripping mechanism dedicated to the dummy basket.

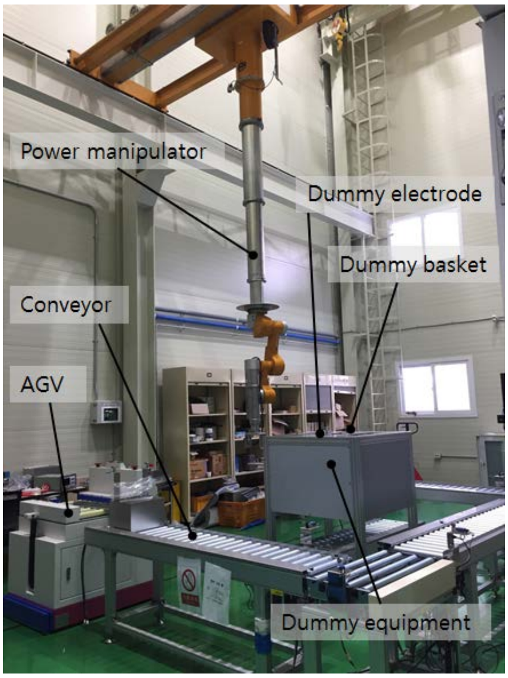

Figure 2. The integrated automation testbed

The automation testbed successfully integrated as shown in Fig. 2, and the given automation tasks were accomplished within the predefined accuracy and repeatability. The evaluation scenario was effectively demonstrated by the integrated automation testbed, and it contributed to substantively evaluate the proposed automation concept.

Automated Device for Pressure Tube Samples Collect for Hydrogen Concentration Determination in PIEL

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Abstract

This paper describes a robotic equipment to be integrated into the technology for the replacement of CANDU 600 reactor fuel channels. This equipment includes a robot part which is remotely operated and moves autonomously inside the pressure tubes and Calandria vessel. The task of the robot part is to perform a visual examination and get video records of the inside part of these tubes and of their rolled joint area.
The main capabilities of this complex robotic equipment are described in our paper. Examples of specific achievements of such devices operating in the nuclear industry in the most advanced countries in the world are also presented.
A detailed presentation of our model is given, both in terms of mechanical and kinematic structure, including the control, remote control and driving systems.
The advantages of using this equipment are revealed and the practical possibilities of manufacturing and testing the experimental model are discussed. The approval tests done in order to eventually homologate the prototype are also presented.

Keywords: pressure tube, CANDU, in-service inspection, robot, visual examination.

A 21st century remote manipulator : the MT200 TAO programme. Experience feedback and future perspectives

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Abstract

The MT 200 TAO programme uses the TAO technology (Téléopération Assistée par Ordinateur, Computer Assisted Remote Manipulation) developed by CEA LIST, which endows remote manipulators with the major benefits of fast calculators and the new generation of electric motors.
After validating the concept from the technical, economic and human angles during a prototype phase, AREVA and GETINGE LA CALHENE managed the entire development and industrialization culminating in the commissioning at La Hague retreatment plant at the end of 2015.
The first Experience feedback confirms that the initial forecasts were correct.
Not only can the TAO product family be perfectly integrated into existing installations, it will also revolutionize the design of future lines of hot cells, in particular thanks to its extensive capacities in terms of work volume, remote steering, versatility and reliability.
Without a doubt this new technology can indeed be heralded as “21st Century TeleOperation”.

Additional payable safety features for remote handling technologies

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Abstract

In active High-Radiation activity settings having clear vision of operations is essential. Equally essential is a precise game plan for work which focuses its attention on the skilled remote handling operators who must produce the results. Project success is based on clean, efficient cuts and take-aways, prepared paths for placement and containment, programmed return paths; and of course on practice, practice, practice on the part of the operators.
According to Joseph Schumpeter, creative destruction describes the "process of industrial mutation that incessantly revolutionizes the economic structure from within, incessantly destroying the old one, incessantly creating a new one". In the same manner remote handling technology has improved in the last decades.
Mechanical design improvements, complex robotic functions and force feedback are examples of progress in that sector. These progresses have also a major impact on the improvement of safety at work. This paper will concentrate on explaining which functions have become cost-effective in the last decades:

1. Virtual Reality
2. Collision avoidance function
3. Tool grapping
4. Linear movement

About Wälischmiller: 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 include 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.

Post-irradiation examination #1

Spallation material preparation using the hotcell facility, a new preparation box and focused ion beam for investigations from mechanical testing up to methods using synchrotron light

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Abstract

The spallation neutron source SINQ at PSI is a continuous source - the first of its kind in the world - with a flux of about $4x10^{14}$ n/cm^2/s. The radiation exposure of the materials due to the neutron flux combined with the incoming proton beam lead to high activation and microstructural degradation of the materials in the target region. In order to address the change in material properties together with the connected scientific questions, materials are extracted from the target and investigated by means of microstructural studies and mechanical testing. Furthermore in the SINQ Target Irradiation Program (STIP) material candidates are explicitly exposed in the target area, reaching damage rates of about 12-15 dpa (in Fe) per year.

As the activation of the materials is very high, such as about 100-200 mSv/h/cm3 for steel samples, the initial separation and conditioning of the samples in the hotcell are inevitable. For the further preparation of the samples, they are transferred to a newly commissioned hot preparation box, which can accept samples up to a cumulative dose rate of 200-300 mSv/h. Here the samples are prepared from e.g. Charpy tested samples for further mechanical testing such as three point bending and small punch testing, including in-situ exposure to lead, or microstructural analyses, e.g. TEM or PAS. All of these tests must also be performed in a shielded environment. For the microstructural studies the TEM samples are either prepared by the classical polishing and electro-polishing methods, or by using the focused ion beam (FIB). Here a newly installed shielded FIB in the PSI hotlab opens new possibilities in the extraction of samples from relevant regions in the material and at the same time to reduce the activity below critical limits allowing studies in lower classified laboratories. The FIB can be used not only for TEM sample preparation, but also to the preparation of special sample geometries for the investigation in a synchrotron facility, the preparation of miniaturized mechanical testing geometries, and the production of fine needles, allowing the investigation using atom probe analysis.

The presentation will illustrate the sample flow from the spallation source thought the different hotcells and shielded facilities to the specific testing devices. It will also show some concrete results from different investigation methods, such as in-situ lead mechanical testing, positron annihilation, atom probe analysis, and microstructural studies.

Cleaning of failed Lead containing Zircaloy-2 Neutron Spallation Target Rods with a Dissolution Process

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Abstract

The Paul Scherrer Institut operates the neutron spallation source facility SINQ. SINQ is designed as a neutron source mainly for research with extracted beams of thermal and cold neutrons, but hosts also facilities for isotope production and neutron activation analysis. The spallation is performed with a proton beam hitting a lead target; the lead is encapsulated in Zircaloy-2 rods. Except for its different process of releasing neutrons from matter by a spallation reaction, SINQ resembles a medium flux research reactor. The Zircaloy-2 target rods have typical cross-sectional dimensions like a fuel rod cladding (the length is just shorter). To avoid too strong neutrons moderation, heavy water is used for cooling.

On June 25th 2016 around one o' clock pm the facility was shut down because of an unexpected target failure (target no. 11). All experiments, unfortunately including one of our own research group had to be stopped for several months until end of October 2016.

The division hot laboratory (AHL) as well as the Laboratory for Nuclear Materials (LNM) are strongly involved in the investigations about the reason for the target failure. The concerned rods are currently still too active for transportation and for the planned tests in the Hotlab. However there are similar older target rods in the hotlab (target no. 9). They are of specific interest because their investigation can help to obtain a preliminary view on the material aging and operational impact onto the material, and shall anticipate an explanation of the later target failure. Foreseen and already partially performed are tests concerning micro-structure and -chemistry (EPMA), hydrides determination (EPMA and SIMS) and radiation embrittlement (micro hardness / metallography). Further, mechanical tests are planned, however, for these the rod sections need to be lead free. Some of the rods filled with lead could not be cleaned by pushing the lead out completely. Thus, chemical cleaning has been envisaged.

The first task was to find an optimal dissolution process and an appropriate acid, respectively. We also had to make sure, that there was no change of the hydrogen content in the samples during the boiling with the acid. Concerning the solubility of lead we found some hints in a work description of a former project and in the literature. Several inactive dissolution tests were made and a process was defined. We reached good results with an acetic acid mixture. It was clearly shown, that neither the hydrogen content nor the sample surface change during the treatment.

For the active test we have chosen the heavily lead shielded dissolution box which is equipped with manipulators. Before starting the procedure for the active material, a safety consideration has to be formulated. Despite the fact that a box made for fuel and fuel contaminated specimens is used, some of the present spallation products are very exotic form a nuclear fuels point of view and pose a challenge: e.g. polonium, mercury (Hg-194) or gold (Au-194). The amount of a possible gaseous release as well as aerosols of these nuclides into the exhausting system had to be clarified. Emission limits have to be met by using additional activated carbon filters. Together with a Swiss company, specialized in nuclear industry products, the development of appropriate filter holders for existing standard filters has been started. Besides this work, a safety report is established; in particular, personal safety equipment and a safe procedure need to be defined.

Post-irradiation Examination Using TEM Method for Swelling Evaluation of Baffle Plate in PWR Core Internals

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Abstract

Irradiation Assisted Stress Corrosion Cracking (IASCC) occurs in core internal structures that receive high neutron irradiation in a pressurized water reactor (PWR) environment. The swelling data of 304 stainless steel which is necessary for reliability evaluation of baffle structure in core internals, it is considered to be important to obtain data using baffle plate material sampled from the actual plant.

However, it is difficult to sample the baffle plate directly from the actual plant under service in the effort of the post-irradiation examination (PIE). It was common to evaluate using materials sampled from baffle former bolts or flux thimble tubes. It is relatively close to the core, and the same kind of austenitic stainless steel (Cold worked 316 stainless steel) as baffle plate. Therefore, in order to improve measurement accuracy and reliability in the swelling evaluation of the baffle plate, it is the same material type, and it is desired to develop systematic data on the PWR irradiation condition (irradiation temperature / dose) and swelling characteristic.

In this study, directly evaluation was made possible by conducting the PIE using the sampling of the irradiation material from the decommissioning plant. We tried to obtain systematic swelling data against irradiation conditions by sampling from a baffle plate of Spain's decommissioning ZORITA plant which was operated for about 40 years (26 EFPY).

Swelling data of the baffle plate was obtained using transmission electron microscope (TEM) method with sampling technique from micro region by PIE in hot laboratory as follows:

- ✓ Based on the simulated analysis results of the dose and the irradiation temperature, the sampling position was selected using the current swelling evaluation formula. A size of 12 mm × 36 mm × 29 mm (thickness) from the actual baffle plate was obtained.
- ✓ Swelling evaluation by TEM method, using dose (33 ~ 47 dpa) and irradiation temperature (299 ~ 327 °C) as parameters with PIE in Nuclear Development Cooperation (NDC) hot laboratory.
- ✓ In order to reduce radiation exposure to the human body due to radioactivity by TEM observation in the hot laboratory, processing was punched out in a micro region at φ 1 mm from irradiation material, embedded in an un-irradiated stainless steel material of φ 3 mm, and the thin film was processed by electrically polishing methods for high magnification cavity observation.
- ✓ As a result, a swelling rate of 0.02 to 0.08% was obtained based on irradiation conditions. It was possible to show the validity of the current swelling evaluation formula.

Shear Punch Testing of Irradiated Cladding Materials from BOR-60 Irradiations

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LA-UR-16-27519

Abstract

Multiple engineering materials were irradiated in the BOR-60 reactor to doses of 16 dpa at temperatures of roughly 375 and 400C, as part of the Ion Beam Simulation of High Dose Neutron Irradiation project, an Integrated Research Program (IRP) Award from the U.S. Department of Energy, Nuclear Energy University Programs (NEUP). The purpose of this project is to compare fast reactor generated neutron irradiation damage to that generated from ion irradiation. The samples were in the form of 3mm diameter by 0.25mm thick TEM specimens. Mechanical properties were measured by shear punch test method, yielding effective shear stress vs displacement curves. Special attention will be given to the fixturing for this test and similar tests. The data collected can be correlated to and compared to yield and UTS data generated from tensile tests. The alloys HT9, T91, 14YWT, and 800H were tested, along with a number of model alloys. This data will be compared to control specimens, as well as samples from other irradiations. Hardness measurements and corresponding microscopy will be presented as well.

In situ Raman spectroscopy on nuclear materials in hot cell

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Abstract

Micro-Raman spectroscopy is a powerful technique for analyzing nuclear materials (fuels, claddings and simulated fuel debris). For this purpose, the confocal Raman spectrometer of the Atalante facility (CEA/Marcoule), coupled to an optical microscope in a hot cell, is used to study the aging of nuclear materials in service conditions (reactor, interim storage, waste disposal) or in accidental conditions, subjected to complex scenarios of irradiation, temperature and interaction with the surrounding environment. This contribution aims to illustrate, through examples, the interest of Raman spectroscopy in the characterization of nuclear materials in a wide range of chemical composition and solicitation.

Firstly, analyses of UOx and MOx fuels allow highlight the effect of irradiation damage with the presence of a triplet of defect bands as observed during ion beam irradiation [1]. Moreover, the position of Raman-active band for the fluorite structure inform on the Pu content, FP concentration and oxidation state of the matrix [2]. Moreover, when subjected to water interaction our studies have shown the precipitation of secondary phases like uranium peroxide whose nature depends on the local redox conditions.

On zirconium alloys, used to encapsulate nuclear fuel in pressurized water reactors, two oxide layers are formed, at the water/cladding and the fuel/cladding interfaces. Micro-Raman spectroscopy allows analyze these oxide layers, highlight the presence of two crystallographic phases of zirconia (monoclinic and tetragonal) and the presence of bands of defects also observed after ion beam irradiation and probably generated by ballistic collisions [3, 4].

During a severe accident, the nuclear fuel can interact with the Zircaloy cladding, and the surrounding metallic structure forming a partially or completely molten mixture. In this context, simulated fuel debris partially representative of those of Fukushima has been characterized before and after leaching experiment under oxidizing conditions. Indeed, Fukushima's fuel debris will be in contact with deionized water during transportation and interim wet storage after defueling. These analyzes revealed the presence of (U, Zr)O_2 phases in the raw material and, after water interaction, the precipitation of secondary phases on these fuel debris.

These analyzes show the interest of using micro-Raman spectroscopy to observe at a the micron scale the heterogeneities of highly radioactive materials and improve the understanding of their behavior during usage conditions, but also the materials formed during severe accident as Fukushima's fuel debris and the secondary phases formed after interaction with the surrounding environment.

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Post-irradiation examination #2

ND-PIE on MTR fuel plates at SCK•CEN: a comparison with destructive analysis

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Abstract

ND-PIE (Non-Destructive Post Irradiation Examination) on full size MTR (Material Test Reactor) fuel plates have been conducted for some years at SCK•CEN. Within fuel qualification programs, ND-PIE such as outer oxide layer measurements, fuel plate thickness measurements, and gamma spectrometry burn-up analyses are performed.

To introduce the examinations in a hot cell environment, existing measurement devices have been adapted and special equipment were built, such as the BONAPARTE bench (Bench fOr Non-destructive Analyses of Plate And Rod Type fuel Elements). In order to calibrate these measurement systems, methods were developed using appropriate traceable standards.

To confirm the quality of the measurement techniques, a comparison study with destructive analyses was performed. This paper will present the results achieved within this comparison study and evaluate the predicted uncertainty on the ND-PIE measurements.

Application of FE-SEM to the measurement of U, Pu, Am in the irradiated MA-MOX fuel

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Abstract

It is important to make observations and elemental analyses of irradiated minor actinide-containing mixed oxide fuel (MA-MOX) for the irradiation behavior investigation of MA-MOX. However, irradiated MA-MOX specimens have high radioactivity and emit alpha-particles. In order to make detailed observations of microstructure and elemental analyses of irradiated MA-MOX, a field emission scanning electron microscope (FE-SEM) equipped with a wavelength-dispersive X-ray spectrometer (WDX) was modified as follows.

1) To prevent leakage of radioactive materials, the instrument is attached to a remote control air-tight sample transfer unit between a shielded hot cell and the instrument.

2) To protect operators and the instruments from radiation, the instrument is installed in a lead shield box and the control unit is separately located outside the box.

By using the modified FE-SEM/WDX, MA-MOX specimens irradiated in Joyo were made observations and elemental analyses. As a result of observation by FE-SEM, microstructure changes was observed in irradiated MA-MOX specimen. As a result of elemental analysis, the characteristic X-rays peaks (U, Pu, Am) were detected by WDX successfully. By measuring the intensities of characteristic X-rays, it was tried quantitative analysis of U, Pu, Am along radial direction of irradiated MA-MOX specimens. Thereby, it was able to grasp the change of microstructure and the change of U, Pu, Am radial distribution of irradiated MA-MOX.

The technique has the great advantage of being able to evaluate the changes of microstructures and the changes of element distributions in MA-MOX due to irradiation.

Post Irradiation Examination of Fuel Bundle from 540 MWe Pressurized Heavy Water Reactor (PHWR) of TAPS-3

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Abstract

A key requirement in any life cycle management plan is the material surveillance programme of in-service components to assess whether materials degrade from known or unknown mechanisms; to estimate the remaining life; and to determine appropriate mitigation, when required. 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.

The 37-pin fuel bundle was loaded in the 540 MWe Pressurized Heavy Water Reactor (PHWR) in Tarapur Atomic Power Station-3 (TAPS-3). It was finally discharged from the reactor due to fuel failure in the year 2014, after a residence of 147 days in the reactor and accumulating a burnup of 3576 MWd/tU. The fuel bundle was received at hot cells of Post Irradiation Examination Division for post irradiation examination (PIE), after a cooling period of 1.5 Years.

After receipt of the failed bundle, yellow deposit was observed in the end plate region of the fuel bundle during visual examination. Elemental analysis of the deposits showed they are rich in Fe, with the presence of some amount of Cr, Ni, Co, Zn and alkali metals. The fuel bundle was dismantled using a pneumatic hack saw machine.

The fuel pins were cleaned thoroughly with alcohol and observed under camera in the hot cell to examine the condition of the appendages, end plug welds and the integrity of the cladding. Examination of the fuel pins of the bundle revealed presence of cracks with perforation in a few fuel pins. Leak testing by liquid nitrogen-alcohol test also confirmed the presence of through the wall cracks. A laser based profilometer measurement showed that the diameter in the fuel pins was relatively uniform along the length of the fuel pin except the failure location. Ultrasonic testing (UT) indicated crack in the clad near the failure locations of all the failed fuel pins. The axial crack length was measured by the probe movement from the start and up to the end of the signal. The UT defect signals showed presence of lack of fusion and root cracks in the weld, which were marked for metallography.

Gamma scanning of failed pin showed variation of Cs^{137} counts near the failure location as compared to uniform counts in an un-failed fuel pin, whereas, Ru^{106} counts are constant. Variation in Cs^{137} near the defect location indicates the possibility of a higher fuel temperature, causing migration of volatile Cesium (Cs) from hotter to cooler locations.

The pressure of the released fission gas in fuel pin from the peripheral ring and the inner ring was measured to be in the range of 1.49 atm and 1.95 atm, respectively which is normal for this level of burn-up.

Detailed metallographic observations and β-γ autoradiograph were carried-out in the fuel pins which gave an estimate about the center line temperature, clad oxide layer, restructuring details, multiple cracking of the clad layer. Hydride blister was observed towards the outer surface of the cladding. The sun burst type of hydride blister on the inner surface was also found. Ultrasonic testing of the end plug to clad tube weld region showed lack of fusion with

the defect covering about 140% of the wall thickness and also severe hydriding in the end plug with cracks in the inner ring.

Internal hydriding was the root cause of failure of the outer fuel pin based on the observations of clad diameter increase, presence of blister, higher fuel temperature, thicker oxide layer on clad inner surface and higher level of hydrogen in the clad near the failure location. Moisture in the pellet or insufficient baking of the graphite layer could be the possible source of hydrogen inside the fuel pin. It was recommended that strict control of moisture/ hydrogen in the fuel pin is required to avoid such fuel failures by controlling fuel pellet drying parameters like temperature, duration and vacuum.

Dark ring observed on the clad surface near the end plug weld in a few pins of the fuel bundle which may be due to higher heat input. Root crack defects were observed in two such fuel pins which are generated either due to higher heat input or higher tensile stress. The lack of fusion defect and root crack observed in the fuel pin suggests that fuel might have failed due to weld related defects allowing coolant ingress oxidizing the pellets and hydriding the clad. It was recommended that better control on the welding parameters during fuel pin fabrication will eliminate weld related defects. The grossly defective welds may be eliminated by proper ultrasonic testing of end cap welds on 100% basis. The presentation will cover all the important findings of the examinations carried out in hot cells.

Analysis of Fission Gases Released in the Void Volume of Irradiated CANDU type Nuclear Fuel

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Abstract

The gaseous fission products tend, due to their nature and their insolubility, to produce changes in the fuel pellet volume and increase pressure in the void volume of irradiated fuel elements. Considering the impact that these changes can have in terms of nuclear safety, it is necessary to study the behavior of fission gas during irradiation. In order to study the production and release of fission gas during irradiation and post-irradiation experiments, various experiments can be designed. This paper presents the installation for cladding puncture and analysis of fission gases released into the void volume of irradiated fuel elements and recent experimental results obtained on CANDU type fuel elements. The concordance with result provided by the computation codes will be also discussed.

The installation for cladding puncture and fission gas analysis was designed and manufactured at RATEN ICN Pitesti. It is used for:

- Measurement of the pressure and volume of gases in the void volume of the fuel rod;
- Measurement of the fuel rod internal void volume;
- Determination of the chemical composition of fission gases, including isotopic composition of the fission gases where applicable.

The paper contains also a description of the method used for the analysis of fission gases. A special attention is paid to the calibration method used for gas analysis by quadrupole mass spectrometry. A dedicated device was designed in order to mix pure gases in different concentrations for the calibration of the mass spectrometer.

Post-irradiation examination #3

Microstructure Analysis of Irradiated NUE and NU Fuel in NPIC Hot Cells

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Abstract

The natural uranium equivalent (NUE) fuel development program is proposed by Third Qinshan Nuclear Power Co., Limited (TQNPC) and Candu Energy, to improve the uranium resource utiliztion rate and reduce the storage pressure of spent fuel. In order to evaluate fuel irradiation performance and pellet cladding interaction of NUE fuel elements, irradiation tests were performed in Qinshan CANDU 6 reactor Unit1 and then post-irradiation examination were carried out. The irradiation swelling along the axial and radial, the crack distribution, and the shape, size and distribution of grain and pore in UO_2 pellet of NUE and NU fuel were analyzed by OM and SEM in NPIC hot cells. The results show that there is no significant difference on microstructure between NUE and NU fuel.

Spherical Fuel Element Deconsolidation System in INET

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Abstract

For High-Temperature Gas-cooled Reactor in China, coated fuel particles are bonded into spherical fuel elements by a graphite matrix. In conjunction with the development of a system to determine fuel failure fractions through γ-analysis of individual and undamaged fuel particles, an automated spherical fuel element Deconsolidation (SFED) system was developed in this study to separate the particles from the graphite matrix by disintegrating the matrix into fine graphite powder, based on the anodic oxidation of graphite in an electrolyte containing nitric acid.

The apparatus in this work contains electrolyzing cell, rotator, anode, Pt cathode and HNO_3 electrolyte, DC electrolytic power supply and control system based on the anodic oxidation. During the deconsolidation, the spherical fuel elements are treated by DC power in electrolyte, and the graphite matrix is intercalated, oxidized and integrated to powder. Afterwards, the mixture of electrolyte and graphite powder is treated by vacuum filtration, and coated fuel particles are sampled from outside to inside in sequence. This system is compact, easy to operate and remote monitoring. Also, the position information can be maintained during the deconsolidation treatment, in favor of coated fuel particle breakage research and analysis of fission products in PIE, avoiding cross contamination among different samples.

Abstracts

of

Poster Presentations

P-01 Poster

The outline of Japan Atomic Energy Agency's Okuma Analysis and Research Center (3) - Labolatory-2 -

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Abstract

Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station (1F) is in progress according to the Japanese Government's "Mid-and-Long-Term Roadmap" (Roadmap). Radiometric analysis of fuel debris and radioactive wastes such as contaminated rubble and secondary wastes from water processing is needed for the decommissioning. The Roadmap assigned the construction of a hot laboratory and analysis to the Japan Atomic Energy Agency (JAEA). The hot laboratory, "Okuma Analysis and Research Center", will be constructed near the 1F site.

The JAEA's Okuma Analysis and Research Center consists of the three buildings; Administrative building, the Laboratory-1 and Laboratory-2. The Laboratory-2 be used for develop technologies related treatment and disposal methods of fuel debris and high level radioactive rubbles and secondary wastes.

About analysis contents and their priority level in the Laboratory-2, meeting has been starting FY 2016 by specialist, As a results, the Laboratory-2 will be installed concrete cells, steel cells, grove boxes, fume foods, and various analysis apparatuses to estimate radioactive analysis and mechanical and chemical property of fuel debris and high level radioactive rubbles and secondary wastes.

Detail design of the Laboratory-2 will be started since FY 2017. By the design at the moment, the Laboratory-2 will be installed fourteen concrete cells, twelve steel cells, nineteen grove boxes and eight fume foods. In the concrete cells, sample processing, X-ray CT and gamma spectrometer will be performed. In the steel cells, SEM, EPMA, XRD, XRF, ICP, etc. will be installed and performed metal organization observation and chemical analysis. In the grove boxes and the fume foods, ICP, gas chromatography, alpha-spectrometer, etc. will be installed and performed chemical analysis and radioactivity measurement. Also, utilities such as manipulator, in-cell crane, emergency power unit, disposal facility, etc. of radioactive waste has been designing.

Above facilities are designed to meet standards and requirement-spec for handling of fuel debris and high level radioactive rubbles and secondary wastes. In additional, analysis and observation apparatus are remodeled to remote type and/or shielding type to handle high level radioactive sample.

This paper outlines the Laboratory-2, including analysis content and concept of building, equipment and analysis apparatus. The Okuma Analysis and Research Center's outline and the Laboratory-1's detail are reported the other papers in this conference.

P-02 Poster

Electrochemical corrosion tests for core materials utilized in BWR under conditions containing seawater

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Abstract

As one of the decommissioning works of Fukushima Daiichi Nuclear Power Plant (1F-NPP), the new and spent fuel assemblies, which had experienced seawater exposure by emergent cooling of the spent fuel pool (SFP) of Unit 4 and had been stored there, have been already transported to another common pool. These fuel assemblies were temporally exposed to solution with high chloride ion concentration (about 10000 ppm) and high temperature (around 80 °C) in the early 1F-NPP accident. Then the concentration and temperature kept low level over a long time by conducting some countermeasures such as switching to fresh water injection, decease of dissolved oxygen concentration by an addition of boric-acid solution and recovery of the coolant circulation system with removal function of chloride ions. Thus, there was no reports on significant corrosion occurrence for these assemblies under simulated SFP water conditions because it was considered that they were less likely to be attacked under these conditions.

Recently, it has been reported to exhibit seawater incursion to the nut surface of the upper end plug due to the gap structure between the plug and tie plate, when detail inspections were carried out for the new fuel assemblies which had been stored in the SFP of Unit 4. Therefore, at the site having the gap structure in the fuel assemblies, the crevice corrosion might have occured during the storage in the common pool.

In this study, the electron corrosion tests, which were repassivation potential measurements for crevice corrosion, were carried out using artificial seawater to investigate the effects of temperature and chloride ion concentration on crevice corrosion.

304L SS used to a nut part of upper end plug was investigated. The artificial seawater conditions were systematically determined as parameters of the temperature and chloride ion concentration based on the water quality environment change in the SFP of Unit 4 of the 1F accident; the temperature and chloride ion concentration ranges were from room temperature to 80 °C and from 10 to 10000 ppm, respectively.

As a result, it was found that the repassivation potential was lower than a spontaneous potential (E_{sp}) when the temperature was above 50 °C and the chloride ion concentration was over 100 ppm. This indicates that there is a high possibility that crevice corrosion will occur in 304L SS. Similarly, in a case whose temperature were 50 °C and the chloride ion concentration was 10 ppm, a repassivation potential was lower than the E_{sp}. It was clarified that a part of basic conditions concerning crevice corrosion of 304L SS in chloride solution. These results suggested that the 304L SS stored in chloride solution had large possibility of crevice corrosion. Then, further accumulation of experimental data will be needed to clarify detail behavior on the crevice corrosion of fuel assemblies of 1F-NPP.

P-03 Poster

Development of analytical methods for radioactive waste samples from TEPCO Fukushima Daiichi Nuclear Power Station site at JAEA Okuma Analysis and Research Center

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Abstract

The Tohoku Earthquake and the tsunami that followed occurred in March 2011. The surge from the tsunami caused a loss of cooling system in Tokyo Electric Power Company Holdings Fukushima Daiichi Nuclear Power Station (1F), Units 1-4. The loss of cooling system allowed the fuel in reactors 1-3 at least partially meltdown and caused hydrogen explosions. 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” in the 1F site, which provide analytical data to characterize the radioactive waste samples for decommissioning of the 1F site. The Okuma Center campus consists of three buildings; administrative building, Laboratory-1 and Laboratory -2. Laboratory -1 is to obtain analytical data of radioactive nuclide contents in rubbles collected from 1F site and wastes from liquid waste treatment facility.

Radioactive waste samples from the 1F site are planned to be analyzed by mainly conventional radiometric analysis at the Laboratory-1: Gamma spectrometry of HPGe detector is used counting gamma rays to identify gamma emitters. Beta decay isotopes with negligible or no gamma emission are planned to be determined by beta activity, which is measured by a liquid scintillation counter and a low background gas flow counter. These instruments require chemical separation before the measurement, because beta counter has poor energy resolution. Each nuclide needs complicated purification treatment process so that well-trained analysts should be required in order to obtain reliable data.

For long half-life nuclides, Coupled Plasma-Quadrupole Mass Spectrometry, ICP-QMS, has an advantage of sensitivity over radioactivity analysis. This alternative method could eliminate the need for isolation of radio nuclides.

For short half-life nuclides, such as Sr-90 and Ni-63, conventional radiometric analysis requires complicated pre-treatment purification process typically including dissolution, filtration and extraction. If this time-consuming process could be automated and/or standardized, efficiency of pre-treatment process would be improved. We studied to optimize the automatization and/or standardization for short half-life nuclides.

This paper describes development of analytical methods, ICP-MS for long half-life nuclides and automated/standardized analytical system for short half-life nuclides, to be applied in Laboratory-1 of Okuma Analysis and Research Center.

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

The outline of Japan Atomic Energy Agency's Okuma Analysis and Research Center (2) - Labolatory-1 -

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Abstract

Decommissioning of TEPCO's Fukushima Daiichi Nuclear Power Station (1F) is in progress according to the Japanese Government's "Mid-and-Long-Term Roadmap" (Roadmap). Radiometric analysis of fuel debris and radioactive wastes such as contaminated rubble and secondary wastes from water processing is needed for the decommissioning. The Roadmap assigned the construction of a hot laboratory and analysis to the Japan Atomic Energy Agency (JAEA). The hot laboratory, "Okuma Analysis and Research Center", will be constructed near the 1F site.

The JAEA's Okuma Analysis and Research Center consists of the three buildings; Administrative building, the Laboratory-1 and Laboratory-2. Among them, Laboratory-1 will provide the analytical data needed to establish the strategy and methodology for treatment and disposal of low and medium level radioactive wastes from 1F site. The subjects of analysis in Laboratory-1 are mainly contaminated concretes and metals from buildings, ashes which are generated by incineration of radioactive wastes, and secondary wastes from water processing. About these subjects, radionuclides which should be assessed for safe disposal of 1F wastes and appropriate analytical methods were determined.

Analytical methods in Laboratory-1 are radiochemical analysis, chemical composition analysis, mechanical and physical property measurement. Radiochemical analysis provides data of radionuclide concentrations in wastes needed to treat and dispose these, and information of chemical state that affect the transmission coefficient in soils. Chemical composition analysis provides information of co-existing substances for the solidification of radioactive wastes and hazardous substances that may affect the environment. Mechanical and physical property measurement provides information for strength evaluation of building structures in 1F, and physical property data for treatment and disposal wastes. Laboratory-1 which was designed to perform these analysis certainly and smoothly have four iron cells, ten glove boxes and fifty fume foods.

Laboratory-1 is currently in the construction phase, and we have been attempted to further improve efficiency and labor saving of analysis procedure.

This presentation outlines Laboratory-1 in Okuma Analysis and Research Center, including the basic design, activities and progress. The details about Laboratory-2 are reported the other presentation in this conference.

Safety aspects of fabrication of Americium-Plutonium Oxide pellets in a glovebox

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Abstract

Oxide fuels for Generation IV systems and IMF (Inert Matrix Fuel) may contain high concentrations (up to 50%) of plutonium and minor actinides. High concentrations of plutonium in oxide fuels strongly limit the solubility in nitric acid. However, the effect of the Minor Actinide-content (specifically Am) on the dissolution capability is still unknown. Yet this effect forms a crucial aspect in the design of a reprocessing flow-sheet for oxide fuels for Generation IV systems, as well as for IMF. The objective of the research is to study the basic dissolution properties of oxide fuels with a high concentration of minor actinides. This enables establishing a relationship between Am-content and dissolution capability, addressing open issues concerning the maximum achievable Minor Actinide content in fuels and targets for transmutation purposes.

Within this research, fresh Am-Pu oxide fuel pellets had to be prepared by powder metallurgy method. NRG operates an actinide laboratory equipped with gloveboxes and analytical apparatus for actinide (U, Pu) research. In case of Am-241, the pellet fabrication by powder metallurgy (done by hand) is quite challenging in terms of radiation protection, as Am-241 is an alpha and gamma emitter. The standard glove box does not give sufficient protection against gamma radiation. A HAZID (Hazard Identification) has been performed where mainly radiological aspects have been evaluated with conclusion that extra shielding is necessary. This additional protection to limit the radiation influence on the workers could be provided by a glove box shielded by lead glass plates. As the Actinide laboratory is not equipped with shielded glove boxes, a new glove box with suitable shielding had to be designed and installed. Other aspect regarding radiation protection concerns the finder dose; special shielded gloves were necessary to limit the finger dose. The steps of powder metallurgy method were analysed as well and where possible, additional measures like distance tools and/or automatization were taken to further limit the finger dose.

The fuel pellets with 30wt% en 50 %wt AmO_2 in PuO_2 have been fabricated and characterized by density measurements and the chemical composition was checked by XRD. Fuel pellets of good quality have been prepared and will be used for further research on dissolution behaviour.

The measures taken for radiation protection purposes (shielded glovebox and gloves) worked sufficiently, as the collective finger dose obtained during the fabrication campaign stayed below few mSv with the collective dose lower than 50 µSv. It can be concluded that small fabrication campaigns for research can be performed safely in this way.

A SIMPLE WAY FOR CHECKING SAMPLING REPRESENTATIVITY IN NUCLEAR FACILITIES STACKS

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Abstract

Because of obvious safety and health reasons, and consequently of regulatory constraints, the monitoring of gaseous releases from nuclear facilities in the environment and the air-cleaning efficiency measurement are an important challenge for nuclear operators and radiation protection teams. They are based on regular measurements of contaminants' concentrations in stacks and ventilation systems. The pollutants' concentration distribution may be heterogeneous at the measuring point if the distance setting of the mixing is not sufficient. To obtain a reliable measurement of the gaseous effluents released in the atmosphere, the homogeneity of gazes has to be characterized in the sampling area of the stack.

A simple way of characterization is presented in the poster, based on He-4 detection in the stack by a mass-spectrometer. A controlled quantity of He is injected in the HVMC, and then detected in several points of the stack sampling area. A rapid statistic study of the measured He rates gives reliable information about the homogeneity of the mixing in the stack. Helium as tracer gas, can simulate radioactive gases and up to 3 µm-diameter airborne radioactive particles which are typical pollutants in stack downstream HEPA filters.
Acceptance criteria for homogeneity are given by ISO 2889 norm.
Another important element is the piping between the sampling point and the measuring system of radioactive pollutant. Leaks, plugs and trapping of particles can occur giving a wrong measure of the released radioactivity in the stack. In a similar way as previously, comparison between He rates at the sampling point and in air-monitoring device enables the operator to qualify the sampling point (type, nozzles number and their position in the air flow…) and the sample transportation to the measuring system.

Decommissioning Program of Research Hot Laboratory in JAEA -Technical Review of Dismantling Works for the Lead Cells Part2-

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Abstract

The Research Hot Laboratory (RHL) in Japan Atomic Energy Agency (JAEA) is the first facility in Japan for the post irradiation examination (PIE) on reactor fuels and structural materials, which had contributed to advancement of the fuels and materials since 1961. The building of RHL consists of two stories above ground and a basement, in which 10 heavy concrete and 38 lead cells were installed. In RHL, all operations for PIE had been completed in 2003. Then the decommissioning program has been implemented in order to promote the rationalization of research facilities in JAEA.As the first step of the program, PIE apparatuses and irradiated samples were removed from the cells, which have been managed as radioactive wastes. The dismantling of lead cells was initiated in 2005. At present 26 lead cells are successfully dismantled. This paper shows technical review of dismantling operations for the lead cells.

P-08 Poster

Design of Pseudo Fuel Debris Fabrication Equipment for Critical Experiment in converted STACY

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Abstract

Towards the decommissioning of the Fukushima Daiichi Nuclear Power Stations (1F), Japan Atomic Energy Agency (JAEA) has designed fabrication equipment of a pseudo fuel debris for the evaluation of the criticality characteristics of 1F fuel debris.

In the 1F decommissioning project including defueling, the fuel debris needs to be treated with great care from the standpoint of criticality safety, due to uncertainty of its chemical composition and physical state. In the accident at 1F, it is speculated that uranium, zirconium and iron solid solution is one of the major materials of fuel debris, according to the review and analysis of information released from Tokyo Electric Power Company and government authorities. In addition, porous structure of the fuel debris may form with concrete composition by molten-core-concrete-interaction (MCCI).

For development of criticality control of these fuel debris, JAEA has been planning to perform critical experiments using a pseudo fuel debris, at the Static Experiment Critical Facility (STACY). Also the core of STACY is being converted from a uranyl-nitrate-solution-fuel type to a UO2-fuel-rod-and-water-moderator type. The critical experiments using the converted STACY require the pseudo fuel debris with simulating the criticality characteristics of the 1F fuel debris. And high dimensional accuracy is required for the pseudo fuel debris to evaluate the criticality characteristics of the fuel debris with high accuracy. JAEA designed new equipment to fabricate these pseudo fuel debris. In designing, in order to confirm the feasibility of the fabrication-method, some fuel pellets mixed with uranium oxide and structural materials (iron, silicon, zirconium, etc.) were manufactured. The properties such as pressing and sintering condition were obtained by the prototyped fuel debris. The pseudo fuel debris fabricating equipment reflecting these properties is designed in 2017 and now constructed. The equipment will be installed to start the fabrication in 2018.

In this work, the pseudo fuel debris manufacturing results and the design of the new fabricating equipment will be described.

This study includes the results of contract work funded by the Secretariat of the Nuclear Regulation Authority (NRA) of Japan.

SKAPHIA: Shielded Electron Probe Microanalyzer for radioactive samples

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Abstract

Capitalizing on more than fifty years of success in developing and servicing shielded microanalytical instruments for nuclear fuel characterization, irradiated materials behavior investigation and radioactive waste management [1-3], CAMECA recently launched the SKAPHIA Shielded Electron Probe Initiative, bringing together key players in nuclear research & industry towards the development of the next generation Shielded EPMA (Fig. 1).

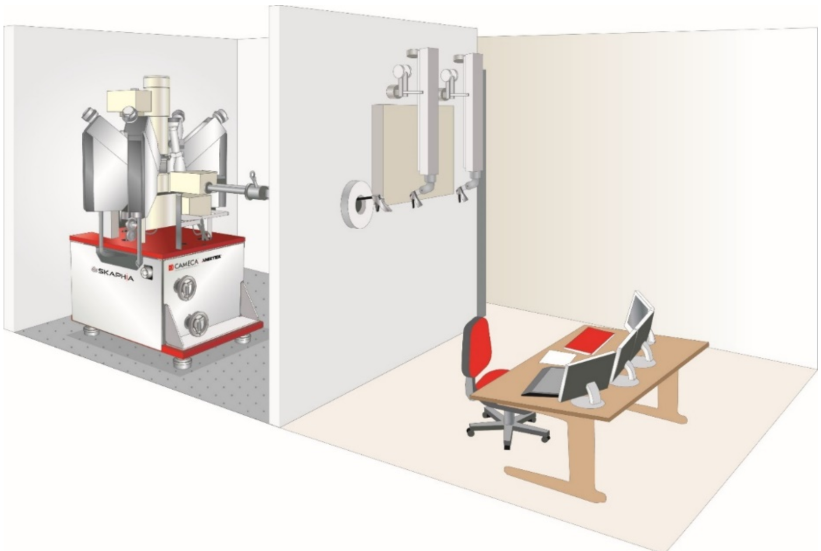

Fig. 1: Representation of SKAPHIA instrument in the lab.

While providing a safe environment for manipulating and analyzing nuclear samples (column, spectrometers and sample stage installed in a hot cell, remote-manipulators used to insert and mount the radioactive samples, instrument fully remote-controlled, shielded WDS and detectors to prevent the background caused by the γ radiations, secondary electron detector with a special orientation to avoid γ ray perturbation, stage made of Denal material and dedicated software providing all necessary features for quantitative analysis, X ray mapping, line profile acquisition and data processing) SKAPHIA will provide benchmark analytical performance. Derived from CAMECA state-of-the-art EPMA instruments, our next generation Shielded EPMA will analyze almost all elements of the periodic table, revealing compositional information for both major and trace elements of radioactive sample. This information will be obtained from sub-micron areas with ultimate precision and accuracy. The system will be designed to accept samples emitting β and γ radiations with a maximum acceptable γ radiation level of 111 GBq at an energy of 0.75 MeV (Fig. 2).

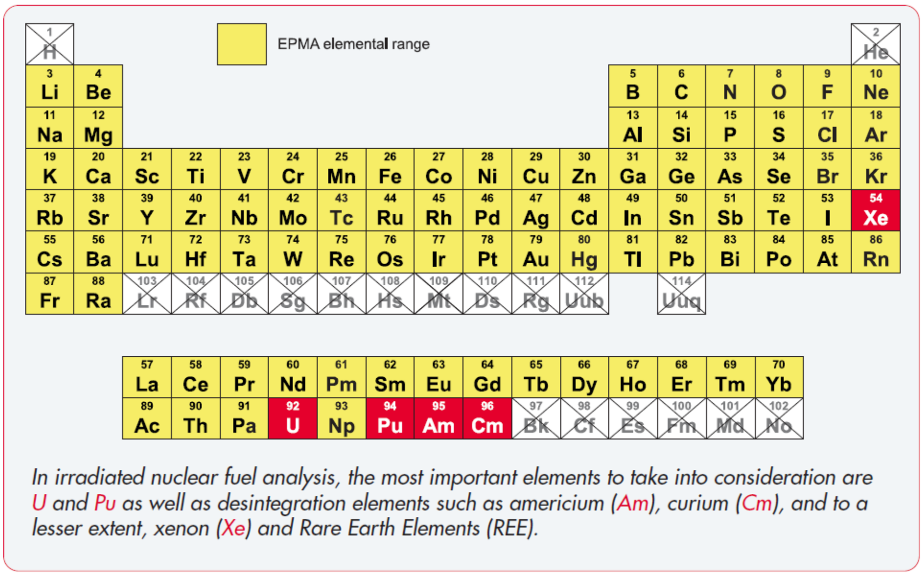

Fig. 2: Periodic table indicating the EPMA elemental range (yellow) and the most important elements to take into consideration in Irradiated nuclear fuel analysis (red).

SKAPHIA will allow scientists to gain a deeper understanding of fuel performance, to explore irradiated material behavior and radiation damage processes, to develop innovative alloys and structural materials, to optimize the nuclear fuel cycle and to achieve better nuclear waste management, thus contributing to a safer world.

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Nuclear Material Transfer Technologies and related Operational Activities at JRC Karlsruhe Hot Cells

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Abstract

At the Joint Research Centre's (JRC) Karlsruhe site, the knowledge generation Waste Management unit has under its mandate the operation of the entire hot cell suite, which constitutes the hub for PIE of conventional and advanced fuel in JRC. To run the various experimental studies on irradiated and/or nuclear material the 'Alpha-Gamma' intervention team within the unit develops, adapts and operates various transfer technologies with the highest care with respect to contamination monitoring and radioprotection regulations. Our nuclear transfer operations cover the introduction of irradiated material into Hot Cells, the loading of samples to be shipped to external users, and all internal sample distribution within the facility. Samples can be transferred to the different facilities of our Hot Cells using a pneumatic transport system connecting the cell lines or with appropriate shielded (50-100 mm Pb) casks according to sample size and dose rate. Docking is achieved by a variety of double lid coupling systems. The poster gives an overview of the technologies implemented and operated in the JRC – Karlsruhe Hot Cells along with an assessment of the experience gained in view of personal safety and waste reduction in the laboratory.

First Window Replacement at the Hot Fuel Examination Facility

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Abstract

Operations at the Hot Fuel Examination Facility (HFEF) at Idaho National Laboratory (INL) began in 1975. Windows at HFEF provide essential viewing capability and shielding for operations inside the hot cell. These windows consist of six panes of glass. The protective A-slab is attached from within the hot cell, with the thicker B-slab next, secured to the outside of a large window tank unit. Inside the tank unit, there are three more slabs (C, D, and E) of leaded glass in a bath of mineral oil to facilitate clearer viewing. The F-slab is attached to the outer surface of the window tank unit in the operating corridor.

An oil leak from the 1M window tank unit into the hot cell was detected in 2015. Project teams convened to design and implement replacement plans. This work called for diverse teams including engineering, facility operations, machine shop, radiological control and engineering, nuclear safety, manipulator repair group, heavy equipment operators, mockup shop, vendor team from Hot Cell Services Corporation, and construction subcontractors.

Before the tank unit could be replaced, the A-slab inside the hot cell needed to be replaced to provide a confinement seal. Design, fabrication, mockup testing, and installation were completed in 2015. The mock-up group provided the tooling needed to remove and maneuver the A-slab via remote manipulation, including a rotating transfer stand and a pneumatic wrench with cameras. A mock-up system allowed engineering, the manipulator repair group, and mock-up personnel to practice and refine the operational steps before actual removal and installation with the in-cell crane and manipulators.

Meanwhile, the engineering team developed the design for the new window tank unit. Engineers from several disciplines provided calculations, designs, and recommendations for challenges such as tooling and window design, suitable viewing angles, and shielding properties. Structural analyses personnel ensured safe floor loading in the truck lock and operating corridor for the 6,350-kg (14,000-lb) window tank on a 2,130-kg (4,700-lb) custom-built cart. Hot Cell Services Corporation, a subcontractor, fabricated the cart and replacement tank unit.

Before tank removal work began, radiological engineers determined any doses received would be ALARA, or as low as reasonably achievable. To satisfactorily reduce radiation, engineers designed a steel shield to block the window space while the tank unit was pulled out. Next, operators transferred the 4,200-kg (9,250-lb) steel shield assembly in-cell and hung it over the window with the crane. They also relocated items inside the cell away from the window to further reduce radiation levels. In addition, carpenters set up a containment enclosure tent around the 1M window work area. Industrial hygiene provided and tested the tent design and welders built components of the tent's air filtration system, which controlled airborne debris.

After the meticulous preparation, construction work took place over a month period, starting at the end of September, to coincide with a scheduled maintenance outage at HFEF. The actual window tank unit swap was completed within a five-day span. Details of the window design and installation will be presented.

Lab-scale continuous vitrification process for high level waste

Christophe Jégou, Thierry Marcillat, Frank Doreau, Jean-Luc Dussossoy, Sylvain Peuget, Maxime Fargard, Isabelle Bisel

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Abstract

As borosilicate glass is the containment matrix for fission product solutions from nuclear fuel reprocessing R&D activities, a continuous vitrification process is being set up in a hot cell of the Atalante facility.

To produce the glass, the waste is dried and heated to convert the nitrates into oxides in a rotary kiln. This is mixed with glass-forming chemicals and heated by induction to very high temperatures (approximately 1000 °C) in a metallic furnace to produce the melt, which is poured into a containment vessel where it cools to form a glass. The containment vessel will be placed in a storage shaft in the hot cell itself before final disposal or further R&D use.

The process is also equipped with tanks and supply lines, a rotating glass frit dispenser and a complete off-gas cleaning line.

When commissioned, the process will be able to treat fission product solutions at the rate of about 1L/H and will produce 1kg glass blocks.

P-13 Poster

Robot-assisted Analysis Procedure for hydrogen content determination of irradiated fuel cladding

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Abstract

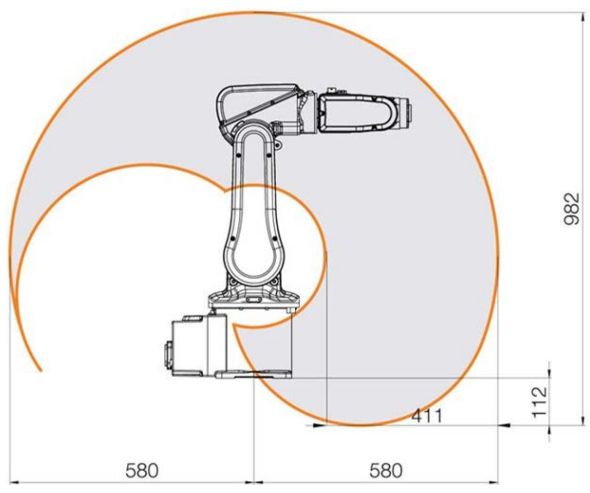
To improve the economy of nuclear fuel assemblies, further development of cladding alloys and assembly designs is aimed for by utilities, suppliers and vendors. To confirm the effectivity of the developments as well as the compliance with safety margins, Paul Scherrer Institut (PSI) Hotlab conducts a Post Irradiation Examination (PIE) program, where fuel pins irradiated at swiss commercial plants are characterized on a regular basis since 30 years. The fuel pins to be investigated are chosen based on fuel assembly design changes and heat generation or burn-up criteria and PSI offers the complete range of non-destructive and destructive analyses. One of the destructive tests is the measurement of hydrogen content in samples of the irradiated cladding. Until now, it is carried out using a LECO RH-402 analyzer installed in a shielded cell fitted with standard remote handlers. After outgassing the crucibles, they are loaded with the weighed samples. By RF heating of about 4 MHz, the hydrogen release is following a function depending on time and temperature. The hydrogen content is calculated by comparing the heat conductivity of a nitrogen gas flowing in a reference channel to a nitrogen gas enriched by the released hydrogen at the same flow rate.

The requirements for the procedure are:

- Cutting of segments from a pin and removing the fuel
- Cutting of each ring into four quarters and measuring one by one
- Keeping the azimuthal orientation of the samples
- Distinguishing between hydrogen in the oxide layer and in the metal

The requirements for the handling in the shielded cell are:

- Unload a distinct cladding sample from the transport tray
- Weigh the sample and the crucible
- Move the sample into the oven loader
- Weigh the empty crucible and place it between the oven electrodes
- After outgassing, remove the crucible including the molten sample from the oven
- Move the crucible back to the transport tray or to the waste bin



A new type of shielded cell was designed at PSI Hotlab fitted with a LECO ROH-600 and an ABB IRB120 six-axis robot arm to reduce cell size and weight. All handling requirements are fulfilled by the robot, saving time and space for the analysis technique. The robot's base and one of the articulated joints have been irradiated before the decision for the IRB120 and the cell was designed around the robot's workspace. The system is already in service for inactive samples and the first irradiated cladding samples will be handled in 2018. The poster presents aspects of the new cell, explains the programming in RAPID with Robot Studio 5.6 and shows the movement path of the TCP along the required worksteps.

High Power ISOL Radioactive Target Remote Handling at TRIUMF

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Abstract

TRIUMF, Canada's national laboratory for particle and nuclear physics, currently maintains a high power ISOL (Isotope Separation On-Line) target station facility called ISAC (Isotope Separator and ACcelerator), commissioned with first beam in 2001. At ISAC, refractory metal and actinide compound targets are bombarded with 50-100 kW of 500 MeV protons from TRIUMF's cyclotron accelerator to produce rare isotope species for a variety of experimental applications, including nuclear astrophysics and fundamental nuclear structure. The ISAC proton beam targets expire, and must be exchanged on an operating cycle that produces about 10 waste target units per year. About 1 week after removal from service, these waste targets have a residual gamma radiation field producing a dose rate of about 1 Sv/h at 1 m, and must be stored on site for decay for several years, prior to shipment in flasks to a national disposal facility. TRIUMF has remote handling infrastructure in place for servicing the ISAC target facility, which includes hot cells and a remotely operated bridge crane. TRIUMF is currently designing a next-generation ISOL facility called ARIEL (Advanced Rare IsotopE Laboratory), with improved remote handling infrastructure. ARIEL will apply modern technology and lessons learned from ISAC.

Development of Precise Manufacturing of Irradiated Miniaturized Testing Specimens at UJV Rez Hot Cell Facility

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Abstract

Present system of nuclear power plant life management requires precise and accurate information about the structural materials degradation and evolution of its mechanical properties. Knowledge of the actual material performance and its prediction is essential for the assurance of safe and long term operation of its components, mainly of reactor pressure vessel and its internals. However, current approved system is mainly based on the programs of surveillance specimens and availability of original archive surveillance materials very limited.

Recently UJV Rez, a. s., Integrity and Technical Engineering Division, has started to pay considerable attention to other perspective methods as a supplementary source of experimental data to foster the process of component degradation evaluation. One of the ways how to enlarge the database of testing results is to machine new samples from the already tested ones. Besides the long term experience with the machining of two new samples from tested Charpy type specimen (using electric discharge machining and electron beam welding), lot of effort was made in the area of machining sub-sized compact tension specimens for static fracture toughness testing (dimensions 10 x 10 x 4 mm). This method enables machining up to the eight new specimens from one single broken Charpy sample.

Paper present all the necessary steps in the process of precise irradiated miniaturized samples manufacturing implementation into the portfolio of manufacturing and testing methods at Hot Cell facility of UJV Rez, a. s.

P-16 Poster

Radioactive sample tomography using brand – new SPECT device designed by Research center Rez, Ltd.

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Abstract

The paper focus on unique SPECT device, which was developed and constructed in the Hot cells facility in the Research center Řež. This facility was built within the project SUSEN and operates with highly irradiated samples.
The SPECT device allows a 3D screening of distributed activity in small radioactive samples with a diameter from one to tens millimeters. The device is capable detecting areas with increased activity within the sample or centers with significantly reduced activity, indicating the presence of cavities, cracks or areas with different chemical composition. The parameters of the collimator allow a resolution of better than 1 mm^3.
The device consists from an accurate 3D robotic scanner and an ionizing radiation detector equipped with two sensitive GM tubes and a massive lead collimator manufactured by Research center Řež. 3D robotic scanner was constructed in accordance with the design from a technical designer from dosimetry of the hot cells facility. Dosimetry data are exported to a computer via analog signal and further processed into tomograms in a specialized software developed by the researchers in department of material and mechanical properties.
The development of this device lasted approximately one year. In March 2017 was the device optimally tuned and first 3D screening was made. The SPECT scanner was successfully tested on several samples consisting of a 3mm diameter radionuclide source ^{137}Cs with an activity of 10 MBq eccentrically placed inside steel capsule of 10 mm thickness.
The scanner is designed for 3D screening of the samples activated in the nuclear reactor with activity from 1 MBq to 1 TBq. It is placed within one of the boxes in the hot cells facility.
Future steps aim to use scintillation detector instead of GM tubes and switching to digital data transfer for more accurate screening and much better resolution.

Development of reconstitution technique and testing of reconstituted bending bars in hotcell

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Abstract

Introduction

NRG is doing research in the field of long term operation (LTO) of current light water reactor pressure vessels (RPV). The shift in transition temperature data obtained by fracture mechanics tests on surveillance specimens is important to make predictions about the RPV lifetime. The available surveillance specimens are often very limited and re-use of the broken specimens is warranted to generate the required data for LTO.

This abstract describes the development of a three point bending setup which can be used to perform fracture mechanics tests according to the ASTM norms E1820 and E1921.

Additionally, a reconstitution technique, according to ASTM E1253, has been developed at NRG to be able to re-use the material of a tested specimen again to fabricate one (or more) new specimens from the limited amount of available specimens.

The procedure is as follows:

1. Perform three point bending fracture toughness test on ISO/DIS 14556-V notch Charpy specimens (SENB)
2. Make inserts from the broken specimen halves by use of a milling machine
3. Attach cylindrical studs to the ends of the inserts by use of studweld equipment
4. Make a new ISO-V notch Charpy specimen by milling to the required specimen tolerances

Three point bending setup

A three point bending test setup was developed for use on an Instron test machine, to perform K_{IC} and J_{IC} tests according to ASTM standards. Testing can be performed in a temperature range from -170°C to 300°C. A clip gauge is used to measure the crack mouth opening displacement (CMOD) and a Direct Current Potential Drop (DCPD) system is used to measure the crack extension (in case of J_{IC} testing). Testing can be performed on surveillance specimens for reactor pressure vessel (RPV) lifetime assessment based on T_0 shift measurements. Special tools were made to improve the ease of mounting knife edges and wires in hotcell for the DCPD measurement.

Milling inserts and welded studs

Inserts from broken test specimen are produced using an in-cell milling machine. The inserts are eventually welded with dummy cylindrical studs on both sides. After stud-welding, new bending specimens (including notch) are fabricated using the same milling machine. Special clamping- and measuring tools have been developed to ease the clamping, alignment, milling and measuring steps in the hot cell.

Studwelding studs on inserts

Welding of unirradiated studs on the irradiated inserts is done using a semi-automatic studweld system, which was specially developed for this purpose. The power source allows us to set the required current- and welding time to get optimal welding results.

Current status

The challenge is to get good welds with a small heat affected zone (HAZ). Currently, several

trial reconstitution welds are produced outside hot cell with the goal optimizing the weld parameters. Hardness and microstructural investigations are being performed to characterize the weld quality and size of HAZ. Temperature measurements on the inserts are planned to measure the temperature profile. Finally, the specimen reconstitution technique will be assessed by verification tests on a selection of reconstituted specimens.
With regard to the status of fracture mechanics setup, several tests on unirradiated material have been performed in the lower shelf and transition regime to test the functionality of the setup. Reference J_{IC} tests using DCPD for crack growth measurement are in the planning for upcoming period.

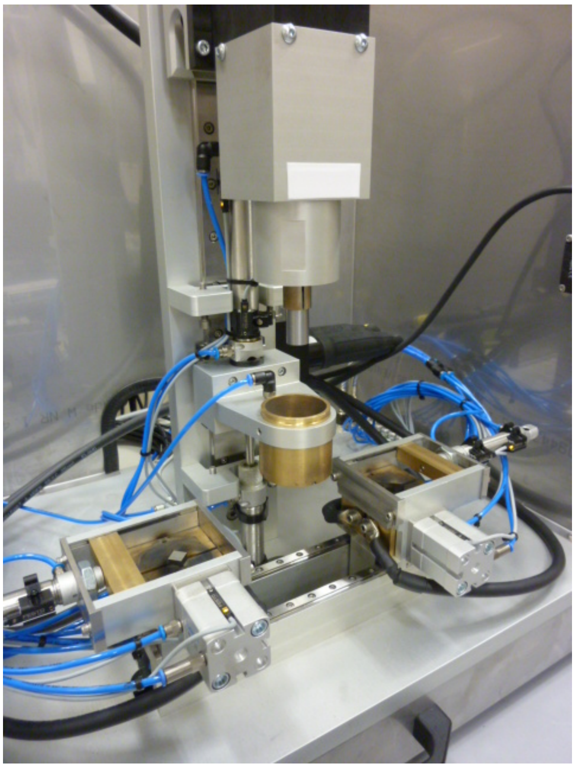

Figure 1: Studweld equipment

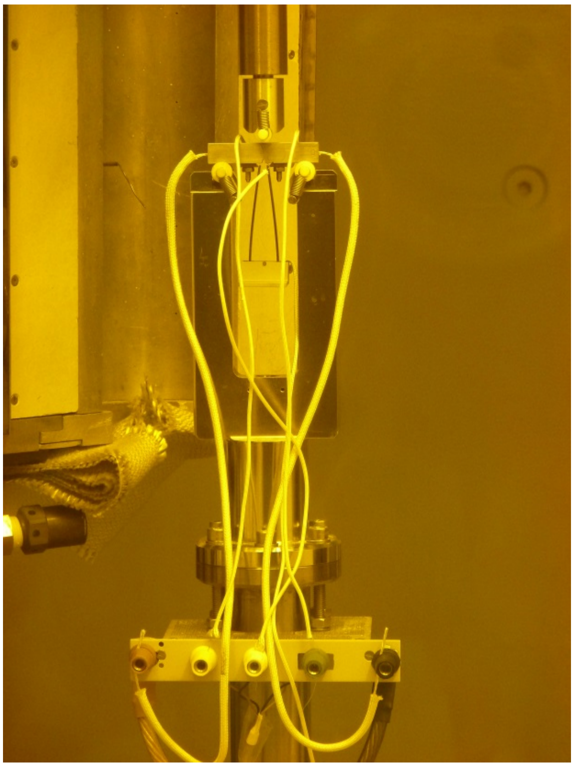

Figure 2: Three point bending setup

Fabrication Techniques of the Sample Supporting Jigs for Post Irradiation Examination with 3 Dimension Printer

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Abstract

In the Reactor Fuel Examination Facility (RFEF) of Japan Atomic Energy Agency (JAEA), Post Irradiation Examinations (PIEs) have been carried out for a long time in order to verify the reliability and the safety of spent fuels. Nowadays, samples handled at the RFEF are not only the spent fuels but also irradiated materials which are smaller than spent fuels and these samples have various shapes. In order to facilitate the handling of the samples using a manipulator, the several kinds of jigs have been used for PIEs at RFEF.

Those jigs are usually manufactured by machining process. We tried to make the jigs, which is PLA resin, with 3D printer instead of machining process for the reduction of the manufacturing time and the improvement of the dimensional accuracy of the jig this time.

It became clear that the actual dimensions of the jigs manufactured with 3D printer were roughly smaller at the concave section and larger at the convex section compared with the dimensions of the plan. So it is necessary to make a plan for the jigs after consideration of the characteristic of the 3D printer.

The jigs can be applied to SEM observation, because the deposition of carbon film onto the jigs was well. And the jigs can be used for the metallography, because the jigs were applicable without any harmful effects on polishing and etching processes.

Radioactive materials post-irradiation examination at CANS

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Abstract

The McMaster CANS facility, a hot cell laboratory designed for radioactive materials characterization research at the atomistic level, is scheduled to be fully commissioned in 2017. This paper is to present the experimental techniques available for post-irradiation examination.

The facility includes five cells: receiving, CNC machining, sample preparation, mechanical testing, and optical testing. Each cell consists of a thick lead-infused shielded glass window, in-cell lifting equipment with crane, and a pair of remote manipulators for handling material such as zirconium alloy reactor fuel tubes. The materials received can be processed through a sophisticated assembly line that logs samples, machines coupons for metallurgical testing with MTS equipment, polishes and etches, and views microstructure evolution on each machining step with an Olympus DSX 500 Opto-digital microscope. The polished sample with low activation is then transferred to FEI Versa 3D FIB/SEM dual beam system for characterization.

The FEI Versa dual-beam system combines a Ga^+ FIB column and a field-emission SEM column in one tool. The SEM with energy and wavelength dispersive x-ray spectroscopy (EDS, WDS) and electro backscatter diffraction (EBSD) is used for topographic features, elemental analysis, grain microstructures and their crystallographic orientations. The Ga^+ beam can perform a variety of tasks such as site-specific etching and polishing for the preparation of electron-transparent specimens and transmission electron microscopy (TEM) or sharp tips for atom-probe tomography (APT), flat and free of critical residual surface for EDS, WDS and EBSD analysis. The first research conducted at CANS was the preparation of radioactive steel TEM samples using FIB. The details of sample preparation and preparation-damage using FIB will be discussed in this work.

Hydrogen Content Analyze in Post-irradiation Examination Samples

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Abstract

To analyze the hydrogen content of fuel rods cladding we found an analyze system. The gas, which releases from sample melting by pulse heating furnace, get into the infrared detector and the results can be calculated by the calibration curve. Due to the high radioactivity of post-irradiate sample, the system have been optimal improved and experimental procedure designed reasonable to satisfy the standard of radioactive concentration and exhausting. Several samples have been analyzed, pieces are normal position and the others are failure position. The results show that the hydrogen contents are about 100ppm in normal position samples, whereas the hydrogen contents of failure position samples are higher than that of normal position samples as expected.

Irradiation Stability Study on Boron Carbide Reinforced Aluminums Matrix Neutron Absorbing Material

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Abstract

Boron carbide reinforced aluminums matrix composites are widely applied in spent fuel storage and transportation field. ^{10}B of boron carbide can capture thermal neutron, and occurring ^{10}B (n, α) ^{7}Li nuclear reaction. These nuclear reactions and elastic collisions between neutrons and atoms will lead to radiation damages, which will limit its lifetime in nuclear material. In order to study the structure evolution and performance change, the full dense 26wt.%B_4C-Al composites fabricated by powder metallurgic method, and irradiation tests were carried out in neutron irradiation environment. The results demonstrated B_4C has good self-healed performance and irradiation stability. 26wt.%B_4C-Al composites were put in nuclear reactor and underwent the neutron and γ ray (neutron average dosage is 5.8E+19n/cm^2 and γ average dosage is 4.021E+11 rad（E≥1MeV）), due to Al irradiation swelling, the dimension and weight of composites were decrease slightly while the density increase a little. After neutron bombardment, irradiation hardening happened to Al, inducing the microhardness and yield strength increase slightly; and the areal density had barely changes after irradiation. These results imply that neutron and γ irradiation has only a little influence on properties of 26wt.%B_4C-Al composites, part of properties are even enhanced. In addition, neutron attenuation measurement were adopted to study thermal neutron shielding performance with different thickness of samples, the experimental results revealed that the sample with 3mm thickness can shield 99% thermal neutron, it is agree with the theoretical value. These research results revealed that 26wt.%B_4C-Al composites prepared by hot powder metallurgic possess both excellent thermal neutron absorption property and irradiation stability.

Keywords: Boron carbide reinforced aluminums matrix composites; neutron and γ irradiation; irradiation stability

Wettability of Liquid CsI on Polycrystalline UO_2

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Abstract

A nuclear severe accident occurred at the Fukushima Daiichi Nuclear Power Plant in 2011. Through this accident, huge amounts of volatile fission products (FPs) such as caesium (Cs) and iodine (I) were released and caused radioactive exposure and environmental contamination. Very recently, our group has reported the ultra-high wettability of liquid caesium Iodine (CsI) on the polycrystalline UO_2 solid surface[1]. Additionally, this study[1] has revealed that liquid CsI penetrates into the polycrystalline UO_2 deeply through the grain boundaries. However, the mechanism of such high wettability of liquid CsI against solid UO_2 has not been clarified. Here, the sessile drop tests were performed for liquid CsI on various solids including UO_2 to understand the behavior of CsI in UO_2. On the solid surface of polycrystalline UO_2, yttria-stabilized zirconia (YSZ) (100) plane and TiO_2 (100) plane, CsI melted immediately and spread to the solid surface after reaching the melting temperature. The contact angle between these solid surfaces and liquid CsI was virtually 0°. However, on the solid surface of MgO (100) plane, liquid CsI did not spread immediately and the contact angle was measured to be 27°. One of the reasons of this ultra-high wettability of liquid CsI against UO_2 is the effect of common properties of solid surfaces of UO_2, YSZ, and TiO_2. It has been reported that the wettability is very sensitive to oxygen-defects at oxide solid surface[2]. In this work, the relationship between the oxygen to metal ratio of solid oxides and wettability of liquid CsI was studied.

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Gamma Scanning of Spent Fuel Element from Nuclear Power Plant

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Abstract

As one of non-destructive testing method in hot cells, gamma scanning is used to provide data relevant to fuel behaviour. 4 complete elements and 4 damaged elements from PWR were measured with burnup range of 9600-45000MWd/tU. Axial burnup distribution and the leakage of Cs were obtained. According to the axial distribution of Cs-137, total loss of Cs into the primary coolant is 20.9% and 27.2% respectively, for two serious damaged elements. The fuel pellet stack length growth rate was less than 0.6%. The low volatile fission product Eu-154 was used to determine the leakage of fuel.

Development of the electrochemical testing techniques in hot-cell

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Abstract

In order to evaluate intergranular corrosion properties of irradiated austenitic stainless steels, the electrochemical measurement technique including sample preparation, which is able to perform in hot-cell using manipulator, was developed. Consistency of electrochemical test results obtained with the developed measurement technique inside and outside hot-cell was confirmed using unirradiated austenitic stainless steel samples.

High Temperature Physicochemical Properties of Irradiated Fuels

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Abstract

Nuclear fuels are generally used under high temperature conditions in nuclear power plants (NPPs). High temperature behavior of the fuels, therefore, must be made consideration in the fuel design for normal operational conditions. As the maximum temperature of the fuels during irradiation is limited within the design criterion to prevent fuel melting, melting temperature is one of the important physical properties to evaluate the thermal behavior of the fuels. Melting temperature for non-irradiated fuels is indispensable to understand irradiation behavior of the fuels at the beginning of life. On the other hand, melting temperature of highly burnt fuel is also of crucial importance for evaluating the integrity and soundness of the fuels at the end of life.

In addition to the knowledge of the high temperature behavior under normal operational condition, the understanding of the fuel behavior in a severe accident of NPPs is also important. In particular, the radionuclide release behavior from fuels at high temperature region under transient and accident conditions has been a main safety concern.

Post irradiation examination (PIE) data including the melting temperatures of the fuels and the radionuclide release behavior from the fuels are needed for better understanding of the high temperature behavior of the irradiated fuels. Available information for evaluating the high temperature behavior of the fuels based on the PIE results, however, is limited due to experimental difficulties encountered in the handling of irradiated fuels.

Research and development (R&D) of high temperature behavior of irradiated fuels have been conducted for many years in the Alpha-Gamma Facility (AGF) of Japan Atomic Energy Agency (JAEA). An apparatus for measuring melting temperatures for irradiated fuels and an apparatus for the evaluation of radionuclides (fission products; FPs and actinides) release behavior from the irradiated fuels were installed in hot cells in AGF.

The apparatus for measuring the melting temperature is equipped with a high-frequency induction furnace and the furnace is capable of heating fuel specimens up to a maximum temperature of 3273 K. The melting temperatures are determined by using thermal arrest method during the heating tests. Burn-up dependence of the melting temperature of uranium-plutonium mixed oxide (MOX) fuels for fast reactors have been evaluated and the decrease rates of the melting temperature have been determined.

The FP release behavior test apparatus is equipped with a high-frequency induction furnace, solid FP sampling systems, a fission gas sampling system, gas analysis equipment and a gamma-ray spectrometer. As the radionuclides release behavior, the radionuclides released from high temperature fuels are quantified and the release rates of radionuclides are obtained. Many fundamental findings concerning the radionuclides release behavior from MOX fuels for fast reactor have been accumulated in heating tests.

In this paper, the activities of the R&D of the high temperature behavior of irradiated fuels are reviewed and detailed descriptions of the apparatuses are provided.

ROBATEL Industries - Nuclear Solutions Provider since 1953. Hot lab Solutions

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Abstract

ROBATEL is a major actor in the French Nuclear Industry since the 1950s, with a clear focus on international development as a driver of new growth for the group. It is present in France with four ROBATEL Industries sites (Genas, La Hague, Cadarache, Marcoule) and in the USA with Robatel Technologies (Roanoke, VA).

ROBATEL develops, licenses, manufactures and maintains transportation and storage casks for radioactive wastes/ sources, spent fuels; LLW, ILW and HLW.
We also provide custom design and fabrication of equipped Glove Boxes, Hot Cells, Shield Doors and Hatches for the nuclear industry and nuclear medicine.
Adding to these products a long experience in the back end, we thus supply waste management lines, including sorting, cementation, incineration, repackaging. This activity makes of ROBATEL a major supplier for the MOX and MELOX plants (France).
These activities have lead ROBATEL to develop neutron and thermal shield compounds and shock absorbing foam that are now used around the world:
PNT3™, PNT7™, Compound 9™, Compound 10™, Compound 21™, Compound 22™, FENOSOL™.

The following article will focus on presenting our hot cells and type B casks for Research Reactor Spent Fuels capabilities and recent solutions that have been implemented.

R&D at ROBATEL Industries – New materials for nuclear safety

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Abstract

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We also provide custom design and fabrication of equipped Glove Boxes, Hot Cells, Shield Doors and Hatches for the nuclear industry and nuclear medicine.
To provide the most reliable products, ROBATEL develop specific materials to ensure nuclear safety.
Neutron shielding materials have been a long tradition at ROBATEL with trademark like PNT7™ for neutron shielding concrete or Compound 22™ for polymer based shielding. Thanks to its internal R&D program and collaboration with the European Center for Ceramic, new generation shielding materials are being developed with improved neutron shielding and temperature range of use. ROBATEL Industries R&D program is also active in the mechanical and fire protection, with the FENOSOL foam, a fire-proof, shock-absorbing foam for nuclear applications.

The following article will present the R&D program currently held at ROBATEL Industry, with a focus on the safety materials.

Enhancing MA Transmutation by Irradiation of (MA, ZR)Hx in FBR Blanket
- Fabrication of (Ln, Zr)Hx Pellets -

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Abstract

MA transmutation cycle by using MA-Zr hydride has been investigated as one of the most effective candidates. It is one of the key items to this cycle to establish the fabrication technique of MA-Zr hydride pellets with homogeneous MA composition and without loss of MA during fabrication process. Especially, the loss of Am during alloying is one of important items to be solved due to its high vapor pressure, difference in melting temperature between Am and Zr, and high chemical activity of reaction between crucible and specimen . In order to solve these items, the following two processes were investigated by using the Sm or Nd as substitute material, 1) the alloying process by using Zr alloy with low melting temperature (82wt%Zr-12wt%Fe-6wt%Al), and 2) the mechanical alloying process. Although the hydride technique for Zr base alloys is considerably established as Sieverts' method, the small equipment was designed as to be installed in a glove box for hot tests by using Am due to the high radio activity of Am.

The alloying tests were performed for mixture of 82wt% Zr alloy and 18wt% Sm metal up to 1523 K for 1 hour. Sm evaporated obviously even at 1320 K and the sample weight decreased about 30% during heating. The lid prevented evaporation of Sm from the system. Sm dispersed homogeneously as small precipitates of ca. 20 μm in Sm-Zr after 1523 K for 1 hour and quench.

Zr powder and Nd powder were mixed at 300rpm for 180 min by attritor mill (Powder Lab; Nippon Coke & Eng. Co., Ltd) as the trial of mechanical alloying. The agglomerations of Sm dispersed homogeneously as small precipitates in spite of short mixing. From this result, the effectiveness of this method was prospected.

The small hydrogenation equipment was designed and the performance was confirmed by hydrogenating Zr of 3 mm to 8 mm in diameter and 3mm to 8 mm in length.

The present study includes the result of “Development of MA-Zr hydride for early realization of transmutation of nuclear wastes” entrusted to Tohoku University by the Ministry of Education, Culture, Sports, Science, and Technology of Japan (MEXT).

The HOTLAB working group and JAEA wish to express their deep appreciation to their sponsors/exhibitors.

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