EUROPEAN WORKING GROUP
on
"HOT LABORATORIES AND REMOTE HANDLING"
Villigen, Switzerland

September 27 – 29, 2000

Final Agenda

Wednesday September 27, 2000

11:30	Registration	
12:30	Lunch at PSI	
13:45	Opening address by G. Bart	

Session I: Topical PIE Techniques
<u>Chairman: Barbara C. Oberländer</u>

14:00	1 Visual Inspection and Nuclide Identification System	H.A. Buurveld
14:20	2 A Micro Beam Collimator for High-Resolution XRD Investigations	D. Papaioannou J. Spino
14:40	3 The New Gamma-Scanning and Tomography System in the Hot Cell Laboratory of the Forschungszentrum Jülich	W. Kühnlein
15:00	4 Fuel Bundle Dismantling Machine PLACIDE	J.-P. Léveque F. Berdoula
15:30	**Coffee break**	
16:00	5 Developing a Method for Fractionated Hydrogen Determination in Fuel Cladding Samples	H. Wiese, A. Hermann
16:20	6 Methods for Measuring Hydrogen in Zirconium Alloys Used in Studsvik	L. Nystrand
17:00	Transport to hotel in Zurzach	
19:00	Departure for dinner "Waldhütte Würenlingen" via PSI	
22:30	Transport via PSI to hotel	

Thursday September 28, 2000

Session II: Applied Research Topics
Chairman: Leo Sannen

08:00	Departure by bus from hotel in Zurzach	
08:30	p7 Standard Electron Probe Microanalysis of Irradiated Fuel at PSI	D. Gavillet R. Restani
08:50	p8 On the Analysis of Pu Distribution in MIMAS MOX by EPMA	B. Vos A. Leenaers
09:10	p9 Developments in the use of SEM and SIMS for the study of Irradiated Fuel at Berkeley	W. J. Stephen R.C. Corcoran
09:30	p10 Ion Beam Analysis of Components with High Tritium and Beryllium Content	J.P. Coad D.E. Hole R-D. Penzhorn
09:50	p11 Joint Applications AES and SIMS Methods for Half-Quantitative Measurements of an Element Composition of a Surface Destruction (Interaction) Various Irradiated Materials	Y. Goncharenko L. Evseyev V. Kazakov
10.10	p12 Application of Destructive Analysis Methods at PSI for the Characterisation of Non-Irradiated and Irradiated Nuclear Materials	Z. Kopajtic I. Günther-Leopold B. Wernli F. Gabler
10:30	**Coffee break**	

Session II: continued
Chairman: Leo Sannen

11:00	p13 Post-Irradiation Examination of High Temperature Reactor Fuel Elements	W. Schenk E.H. Toscano
11:20	p14 Recent experience in high burnup fuel rod puncturing	P. Schleuniger A. Hermann
11:40	p15 Device to Test the Long-Term Creep Behaviour of Cladding from High Burn-Up Spent Fuel	E.H. Toscano W. Goll

12:00	16 Review of Methods used at SSC RIAR for Mechanical Testing of Specimens made of VVER Zirconium Fuel Rod Claddings	Y.M. Golovchenko S.G. Yeryomin
12:30	**Lunch**	

Session III: Facilities for the next centrury
Chairman: F. Groeschel

14:00	17 A new Metallography Box in PSI	D. Gavillet M. Gehringer M. Sauder F. Groeschel
14.20	18 "SINQ Irradiation and PIE programs"	Y. Dai
14.40	19 Fusion Materials Development, CRPP-EPFL	P. Spätig
15.00	20 Fabrication of CT specimens with inserts of irradiated materials by milling and EB- welding	B.C. Oberländer
15.20	21 Refurbishment of PSI to comply with requested safety standards after N 40 years of operation	G. Bart L. Wiezel
16:00	Departure for wine tasting in Auenstein	
18:30	Departure from Auenstein	
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Abstracts

Visual Inspection and Nuclide Identification System

H.A. Buurveld, NRG-Petten

The waste storage facility (WSF) is a building, used for the interim storage of high level and medium level radioactive waste. The radioactive waste is stored in drums in special basements, covered by thick steel and concrete leads. After a maximum period of five years, the radioactive waste is transported to the central organisation for radioactive waste (COVRA) for final disposal. At the moment a clean-up programme has started which aim is to reduce and then to move the high level radioactive waste of the Nuclear Research and Consultancy Group (NRG) to COVRA.

To have insight in the costs of the clean up of the WSF it is necessary to know the composition, filling rate and weight of the contents as well as the condition of the drums with high level radioactive waste stored in the WSF. For this purpose the so-called 'measuring and inspection campaign' has been set up. With the obtained information of this campaign, strategies for reduction of the volume of the high level radioactive waste can be made.

For the measuring and inspection campaign a special device, the so called VINISH, was built. VINISH is a Visual Inspection and Nuclide Identification System for High level radioactive waste. It is equipped as follows:

- a CCD-camera and halogen lamps for visual inspection,
- a high purity germanium crystal for gamma spectrometry,
- 4 meters for dose rate measurements,
- a collimator system for reducing the dose rate on the germanium crystal,
- a rotation system which rotates the waste drums during the measurements and visual inspection.

The dose rate measurements, gamma spectrometry and visual inspection are performed at the same time. The rotation of the waste drums is continuous during these operations. A waste drum is examined at 3 or 4 consecutive positions depending on the height of the waste drum.

Abstracts

A micro beam collimator for high-resolution XRD investigations

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Germany

Based on the necessity of obtaining crystallographic information from the some hundredth micrometer thin periphery (rim) zone of irradiated nuclear fuel pellets, a collimating system has been developed for condensing X-rays, providing a very thin but intense and low divergent flat beam with a nominal aperture of 15 µm. Owing to the high brilliance win, the condenser can be operated even with a conventional CuKα-radiation tube, e.g. in a common Bragg-Brentano diffractometer. Acquisition of XRD patterns on very thin layers of materials (20-30 µm) can be thus carried out in the laboratory without need of high-intensity radiation sources (e.g. syncrotron), as demanded by the most nowadays approaches using glass capillaries collimators.

The concentrator has been mounted in a powder diffractometer installed in a lead-shielded glove box used as a hot cell for examinations of toxic and radiactive samples. In the paper presented, the apparatus will be described and some novel micro-XRD observations on UO_2 spent fuels will be discussed, that yielded important information on structural changes occurring along the pellet radius at high burn-ups.

The New Gamma-Scanning and Tomography System in the Hot Cell Laboratory of the Forschungszentrum Jülich

Winfried Kühnlein, Forschungszentrum Jülich

Gamma-scanning and tomography are well-known, proven techniques for non-destructive measurement of activity or density distribution of radioactive materials. A 3-axis scanning device in the Hot Cells of the Forschungszentrum Juelich has been used for the above mentioned purpose, however, due to frequent repairs has now been replaced by a new device.

Based upon the experience with the old device, a universal 4-axis scanning device (3 linear axis and rotation) was designed and installed in an hot cell, allowing linear horizontal (200 mm), vertical (300 mm) and rotational scan (360 Degree), 2D-scan (200 mm x 300 mm), cylinder-scan and tomography. With the third linear axis the distance between scan object and gamma-detector is adjustable. Collimators with varying geometries allow different object activity resolutions.

The components chosen for the scanning apparatus were selected with regard to the highest possible reliability and for easy exchange. Objects with a weight of max. 15 kg can be handled. A new object adapter was designed to hold samples with various geometries.

The PC-based control and data acquisition software was completely redesigned. The 2D-scan (surface scan) was implemented and the entire scan procedure was automated. The scan status is shown by a bar graph. Due to the fact that the scan can take several days, a restart option was added in the case of an interrupt. The scan data can be exported in a format compatible with popular spreadsheet (i.e. Excel).

The new measuring device has proven itself on investigation of irradiated materials for the upcoming European Spallation Source. The activity distribution gives the exact location for further specimen analysis such as hardness, tensile and bending tests. The tomography of objects with a diameter up to 200 mm and 250 mm height is feasible.

Fuel bundle dismantling machine **PLACIDE**

J.-P. Leveque, F. Berdoula
CEA-Cadarache

The (Department d'Etudes des combustibles of the French Commissariat à l'Energie Atomique in Cadarache, France) **DEC/SECI/LIGNE** Engineering Examination has recently developed a remotely controlled system for different inspection and dismantling tasks.
PLACIDE is a tool guiding device, particularly being designed for the dismantling of fuel bundles. After being approved by the licensing organization **DEC/CSI**, hot operation started in June 2000. The 3 axes remotely controlled automatic control device **CN**, consisting of a three translatory axes guiding machine and a removable tool cutting device with 1 rotatory axe, has been developed for the demonstration of cutting of a Zircaloy "**fuel bundle**". This is realised by means of these 3 axes which bend in the same shape as the fuel bundle one.
An other specially designed cutting system can be easily exchanged by means of a guide rail, will be used for different complex cutting tasks.
It is very important to have equipment of non-destructive examination (gamma-scanning and dimensional inspection) in hot-cell to observe the activity profile of irradiated materials. In both cases, translatory axes of PLACIDE are used. Those equipments can be integrated by remote manipulators, so they are designed, manufactured and modified to make exercise easy and no trouble.
Innovative tooling has been required to provide the reliability and dexterity for performing cutting with a good accuracy, gripping, and other such functions. Over the years, a set of components that can be modified and adapted to several applications has been developed.

DEVELOPING A METHOD FOR FRACTIONATED HYDROGEN DETERMINATION IN FUEL CLADDING SAMPLES

Holger Wiese, Armin Hermann, Raimund Bühner, Michael Steinemann and Gerhard Bart

Laboratory for Materials Behavior
Paul Scherrer Institute, CH-5232 Villigen, Switzerland

Hydrogen pickup of the cladding is a life limiting criterion of fuel rods for high burnup. Hot gas extraction of cladding parts including the oxide scale is commonly in use as the current method to measure the hydrogen concentration in cladding samples. Attempts have been made to differentiate between hydrogen picked up by the metal vs. hydrogen within the corrosion layer. One possibility consists in mechanically grinding off the oxide and separately measuring the hydrogen in metal. This method is not precise enough because of the corrugated interface surface between oxide and underlying metal.

PSI has developed an improved hot gas extraction technique which allows (within one fusion step) to differentiate between the hydrogen fraction in the corrosion layer vs. the amount dissolved or precipitated in the base metal.

In developing this technique a LECO hydrogen analyzer has been programmed in a manner as to allow to estimate the hydrogen evolving from different sources (e.g. oxide layer and metal of a fuel cladding segment) by applying a suitable temperature-time regime. Details of the method as well as the sophisticated calibration procedure will be described. The technique was qualified also by analyzing mechanically separated phases (corrosion layer or metal).

The results show that there exists a significant hydrogen fraction within the corrosion layers of irradiated PWR claddings. Without taking this fraction into account, the hydrogen concentration in the metal phase can be overestimated by up to about 20 % especially with thick oxides and/or thin cladding walls present.

Abstracts

Methods for measuring hydrogen in zirconium alloys used in Studsvik

Anne-Charlotte Nystrand, David Schrire

The characterisation of the hydrogen pick up in fuel cladding is an important factor for understanding the fuel cladding behaviour. Two techniques for measuring hydrogen in non-irradiated and irradiated zirconium alloys are used in Studsvik

→ hydrogen after global info.

One method is based on the hot extraction technique from ASTM standard E146-83 where hydrogen is extracted from the zirconium alloy samples (cladding rings) by heating them in a vacuum system. The hydrogen content in the sample is determined from the volume and pressure of the gas as an average hydrogen content in the sample. The results of the measurements appear to be influenced by the presence of oxide scale on the samples. By heating the samples at different temperatures samples with oxide start to release hydrogen at a lower temperature than samples without oxide.

→ local info

The other method has been developed for preparing scanning electron microscope (SEM) samples suitable for oxide layer imaging and radial hydrogen concentration and morphology determination. The area fraction of the hydride phase is determined by image analysis of backscattered electron images (BEI). The local hydrogen concentration can be determined quantitatively with a spatial resolution of less than 100 μm.

Why mech. behaviour, corrosion behavior, fuel rod. length (modelling)

HVE: ⊖ uncertainty of volume & pressure.
⊖ " residue H in sample chamber.
⊖ abs. of H in crucible.
⊖ uncertainty of cont. H left in sample

SEM: • hydrided mat. has been cooled slowly.
• all hydrides are of δ phase.
• hydride area fraction = hydride volume fraction.

Standard Electron Probe Microanalysis of irradiated Fuel at PSI

R. Restani, D. Gavillet, PSI

Typical electron probe microanalysis (EPMA) consists of the measurement of radial and local elemental concentration profiles of polished, transversal fuel rod specimens.
The elements of main interest are uranium, plutonium, oxygen and among the fission products the mobile Cs and Xe and the burn-up monitor Nd. Complementary fission products may be Ba, I and Zr as well as elements which may form intermetallic precipitates like Ru and Mo.
The EPMA operator faces some specific difficulties when measuring irradiated fuel rods. In the following the guidelines of the standard measurement procedure for fuel analysis are described. In low concentration mesurements the shielding of the specimen and the X-ray detector are important features to reduce the radioactive background disturbance on the measurement especially when the standard material cannot be measured adjacent to the specimen. The radioactive background and interfering characteristic X-rays of other elements for the main fission products and Pu also affords a careful choice in setting and measurement of the backgrounds beside the peak line. The significance of uranium subordinate lines on the measurement of Pu and Cs makes it obligatory to measure and subtract this influence in the evaluation. As this interference is in the range or above the concentration of these elements it causes some uncertainty in the result which proved to be relatively smaller for the Pu measurement.
The calibration standards for the fission products differ in content and composition from the specimen. This lack cannot be compensated by the evaluation procedure ($\phi\rho z$-model) so that the measurement is focussing to give relative concentration differences and approximate contents within a time interval of 5-10 minutes for one full point analysis.
Pu measurements below 5 wt% give the same results calibrated with either Pu and Pu free uranium standard.
Oxygen measurement is used mainly to trace relative oxygen content differences, phase differences and morphological impacts (pores, fissures). It serves also to normalise the element content in the presence of porosity.
The described procedure insures the generation of quality database that can be used for the fuel behaviour modelling.

On the analysis of Pu distribution in MIMAS MOX by EPMA.

B. Vos and A. Leenaers

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One of the main issues in characterising MOX fuel is the quantification of the Pu finely distributed in the matrix. Electron Microprobe Analysis (EPMA) is one of the techniques that can be used to determine elemental distributions. Conventionally, the methods used within EPMA are one dimensional scans (linescans) or two dimensional X-ray maps but the experimenter is confronted with several difficulties when he tries to evaluate this Pu distribution quantitatively. The one dimensional linescans are often not sufficiently representative for the whole sample. The 2D maps on the other hand are difficult to quantify due to the limited signal to noise ratio per pixel which is intimately related to the short acquisition times for each pixel.

In this presentation, we will show that quantification using X-ray maps can be improved. Our approach is based on the acquisition of digital maps, directly calibrated on peak intensities. An advantage of the 2D method compared to linescan acquisitions, is that a 2D representation is more demonstrative and has an improved statistical representativeness. It should be mentioned, however, that a reasonably representative area is at least $1x1mm^2$ wide. If one wants to analyse this area with a spatial resolution of $1\mu m$, 10^6 analysis points have to be acquired. Even with the presently proposed method of quantification, reliable results are obtained only when a sufficient counting time per point (~20 msec/point) is applied. This yields an analysis time of several hours for each zone.

The method is verified using independent linescan acquisitions and applied successfully on un-irradiated MIMAS MOX.

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Hot Cells Conference

Developments in the use of SEM and SIMS for the study of Irradiated Fuel at Berkeley

R C Corcoran and W J Stephen
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Abstract

Post Irradiation Examination (PIE) of irradiated UO_2 and Mixed Oxide (MOX) fuel reveal a large number of small intra-granular fission gas bubbles around 1nm diameter. The nucleation and growth of these bubbles, especially under reactor transients can lead to significant fuel swelling. In addition, the migration and coalescence of these bubbles to the grain boundaries leads to the development of populations of circular lenticular or extended fission gas pores on the grain faces. Eventually the intersection of these pores with grain edges leads to diffusion paths out of the fuel, and to fission gas release.

Extensive examination of irradiated Advanced Gas Cooled Reactor (AGR) fuel and Short Binderless Route (SBR) MOX fuel at Berkeley using the fully shielded Scanning Electron Microscope (SEM) along with in-house image analysis techniques, has allowed a large database of inter and intra-granular fission gas bubble populations to be developed. This has allowed models to be formulated on the development of these populations.

Fission gas bubble analysis PIE performed by SEM at Berkeley is described along with the development of a recently installed active Secondary Ion Mass Spectrometer (SIMS) facility.

ION BEAM ANALYSIS OF COMPONENTS WITH HIGH TRITIUM AND BERYLLIUM CONTENT

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An important part of the fusion research at the Joint European Torus (JET) is to study the interaction of the plasma with the surrounding components. The principal techniques used in these investigations are the Ion Beam Analysis (IBA) methods. Since 1989 beryllium has been evaporated regularly over the inner surfaces (to act as a getter for oxygen and to assist with density control) and some solid Be components have also been used. Thus since that time special handling facilities have been developed to prevent exposure to the potentially hazardous oxides of Be. Furthermore, in 1997 tritium was used in addition to the normal deuterium to fuel the plasma, so components from within JET now also contain tritium, and the handling has been modified to cope with the additional radiological hazard. 35g tritium were used in 1997 as fuelling for JET plasmas, and initially 40% was retained in, or as deposits on, in-vessel surfaces. The amount of tritium in the vessel has now reduced to 2g, which is mostly trapped in carbon-based deposits that have flaked off and fallen to the bottom of the vessel; the specific activity of this material is about 1 TBq/g.

Carbon-fibre composite tiles of sizes up to 380x180x50 mm, and samples with a cumulative tritium content of ~5GBq, have been analysed in the IBA chamber. One of the most important uses of IBA has been in understanding the retention of tritium (and deuterium) in fusion devices. At the present time, a critical aspect of design effort for a Next-Step tokamak is to reduce this retention.

JOINT APPLICATION AES AND SIM METHODS FOR HALF-QUANTITATIVE MEASUREMENTS OF AN ELEMENT COMPOSITION OF A SURFACE DESTRUCTION (INTERACTION) VARIOUS IRRADIATED MATERIALS

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The original methodical receptions allowing to carry out investigations of an element composition of between zirconium oxide and metal part of the irradiated fuel cladding from alloys on a zirconium basis are developed.

For research of the zirconium metal/oxide interface of the irradiated fuel cladding from alloys on a zirconium basis by the methods of the surface analysis using ion etching, offer the solution allowing to take into account:

a) It is essential the major velocity of ion etching of metal in comparison with oxide;
b) It is essential the large uniformity of a metal part of fuel cladding in comparison with oxide film.

As a result of realization of these receptions there is a possibility to compensate or essentially to reduce influence of effect of accumulation of an electrical charge on a researched surface.

Offered receptions at application of AES and SIMS methods (the addition them by more traditional methods of the element analysis is possible also, for example electron-probe microanalysis) allow to receive conception about distribution of various elements not only on interface between zirconium oxide and metal part of the irradiated fuel cladding from alloys on a zirconium basis, but can be extend on other similar investigations of study of an element composition of interface between dielectric film (coverings, depositions on the oxide films used fuel elements) and their metal basis.

Application of Destructive Analysis Methods at PSI for the Characterisation of Non-Irradiated and Irradiated Nuclear Materials

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The principles of a sustainable development in the nuclear energy sector include most efficient use of resourses and reduction of waste volumes. These principles guide all projects in the Laboratory for Materials Behaviour at PSI (Advanced Fuel Cycles, Eden, LWR-Contamination), conducted with the financial support of Swiss utilities and plant operators, Swiss Nuclear Safety Inspectorate (HSK), nuclear industry and international research organisations. The R&D activities within these projects can be grouped in the following areas:

- Maximising the use of uranium by investigating the high burn-up behaviour of nuclear fuel and the recycling of plutonium in the existing LWRs.
- Evaluation the feasibility of inert matrix fuels (IMF) for plutonium incineration.
- Exploring the potential for recycling of actinides and long-lived fission products in fast reactors or in accelerator driven systems.
- Studying the corrosion processes and mechanical integrity in high burn-up Zircaloy fuel rod cladding

The main objective of the various international programmes dealing with post-irradiation examinations is to improve the knowledge of the inventories of actinides, fission and spallation products in spent nuclear fuels and targets. Further, the standards of the nuclear industry demand a high purity for the materials used in reactor experiments. Therefore, highly sensitive an accurate analytical methods have to be applied for characterisation of the advanced nuclear fuels (IMF, MOX).

The main instruments for destructive analysis methods (DA) at the hotlaboratory of the PSI include ICP-MS (inductively coupled plasma mass spectrometry, encapsulated in a glove-box) and TIMS (thermal ionisation mass spectrometry) for analysing isotopic ratios and trace element contents, HPLC (high performance liquid chromatography) and IC (ion chromatography) for the analysis of inorganic ions and element contents, LECO TC-436, LECO RH-404 and LECO IR-412 for the determination of H, C, N and O, TGA-851 (thermo-gravimetric analyser) for the determination of the oxygen-to-metal-ratios and TITRINO fully automatic titration for the potentiometric determination of U, Pu, Np and NO_3. Further, for the dissolution of the irradiated and non-irradiated nuclear materials a high pressure digestion apparatus, micro-wave and reflux apparatus encapsulated in a shielded glove-box are used.

Some examples of specific applications of DA for the characterisation of nuclear materials at PSI will be discussed.

POST-IRRADIATION EXAMINATION OF HIGH TEMPERATURE REACTOR FUEL ELEMENTS

W. Schenk* and E.H.Toscano°

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In the framework of the Share Cost Actions (SCA) of the European Commission, a European Project of Development of High Temperature Reactor (HTR) technology has been approved. The project includes developments in the fields of reactor physics, fuel technology, safety, material needs and feasibility of key components and systems.

In the domain of fuel technology a key point is represented by the testing of the irradiation behaviour of new type of fuels and their fabrication methods. In this context, the post-irradiation examination (PIE) of irradiated fuels will be needed to assess the quality of new concepts. Among the PIE-methods the verification of the release behaviour of fission gases (Xe, Kr) and solid fission products (Cs, Sr, Ag, etc.) under accident conditions will be of paramount importance.

In the past, the so-called Cold Finger Apparatus (KÜFA) was developed in the Forschungszentrum Jülich (FzJ) to test HTR-fuel design and fabrication methods. Using this device, the fission product release from defected particles can be tested up to 1800 °C.

In the framework of the SCA/HTR-technology, an up-to-date version of the KÜFA will be installed in the hot cells. In the paper, a description of the apparatus will presented and the experimental programme discussed.

RECENT EXPERIENCE IN HIGH BURNUP FUEL ROD PUNCTURING

Peter Schleuniger and Armin Hermann

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Fission gas release (FGR) is one of the key parameters to qualify the fuel rod behaviour in reactor operation. At PSI the released fission gas is sampled by puncturing the rod in a hot cell and by filling appropriate ampoules for analyses of the gas composition by mass-spectrometry.

Recent experience in puncturing of high burnup fuel rods includes the timely behaviour of such rods in determining the rod free volume as well as insight into the completeness of the sampling procedure.

The experimental setup will be described and examples of measurements will be presented concerning the problems mentioned above.

Abstracts

DEVICE TO TEST THE LONG-TERM CREEP BEHAVIOUR OF CLADDING FROM HIGH BURNUP SPENT FUEL

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In the last years, the discharge burnup of fuel from Light Water Reactors (LWR) has been increased by introducing, e.g., high corrosion resistant materials and advanced licensing methods. For the fuel rod, high burnup means increased neutron fluence and hydrogen content in the cladding and increased rod inner gas pressure due to a higher fractional fission gas release.

Dry cask storage, which is used for the interim storage of spent fuel, has to consider this changing burnup situation including provisions for future burnup perspectives. Unlike earlier experiments, where materials with enveloping creep behaviour were tested under short-time conditions, commercial cladding shall be tested under experimental conditions as realistic as possible.

Therefore, a device for long-term (≥ 1 year) creep testing of irradiated cladding materials from high burnup spent fuel has been developed at the Institute for Transuranium Elements. In the paper the experimental setup as well as the sample preparation steps will be described. Experimental results concerning non-irradiated samples, tested to check the apparatus performance, will be presented.

REVIEW OF METHODS USED AT SSC RIAR FOR MECHANICAL TESTING OF SPECIMENS MADE OF VVER ZIRCONIUM FUEL ROD CLADDINGS

Yu.M.Golovchenko, S.G.Yeryomin

Abstract of the report for Meeting of the European Working Group "Hot Laboratories and Remote Handling"
(Villigen, Switzerland, 27-29 September, 2000)

Methods for testing and investigations of reactor materials mechanical properties used at hot laboratories of the SSC RF RIAR permanently enlarge and improve. In particular it concerns methods for testing and investigations of mechanical properties of the samples made of VVER zirconium fuel pin claddings.

The methods for mechanical testing (axial tension of ring and tube specimens, cladding biaxial load) are briefly described.

Brief descriptions of the methods used for corrosion-thermal treatment of specimens prior to their mechanical testing (overheats in inert medium and water vapors, hardening of the overheated ring specimens in water, controlled flooding of cladding overheated cuts) are given.

The schemes of the methods are shown, and the results of their application are presented.

A new Metallography Box in PSI

D. Gavillet, M. Gehringer, M. Sauder, F. Groeschel

The PSI metallography box has been used for more than 25 years for the investigation of fuel rods and other radioactive materials. The heavy use induced a strong corrosion and a general ageing of the box and infrastructure (specimen preparation machines and microscopes) which impeded the work quality and general performance as well as the safety of the users. Therefore it was decided to built a new box.

The conceptual design of the box started in 1998 and was completed at the end of 1999. It consists to a brand new steel shielded box with enlarged stainless steel alpha-containment for the specimen preparation box and an independent shielded containment for a new fully remotely controlled microscope with up to date features and performances, both including the modern safety requirements.

The project concept is described including the work statement for the main components. The specific requirements for the instrumentation (microscope and polishing machines) are described and justified. The project flow including a description of the offer evaluation processes is also given.

An intrinsic part of the project is the dismantling and removal of the old metallography box. The work status and experience gained during this process will be offered.

Finally the construction status of the new box including a detailed description of the main features is given.

European Working Group on Hot Laboratories and Remote Handling

Abstracts

SINQ Target Irradiation and PIE Programs

Yong Dai
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The SINQ Target Irradiation Program (STIP) has been going on since 1998 under an international collaboration. In the first experiment (STIP-I) 1646 small size samples were irradiated in SINQ target MARK-II which received a total of 6.8 Ah proton charge after two year operation. The maximum proton fluence obtained by specimens is about 3.2×10^{25} m^{-2} which produces about 8 dpa in steels. Together with the contribution of spallation neutrons the maximum dose is about 11 dpa. The second experiment (STIP-II) has been started since March 2000. 2087 small size specimens are under irradiation which will be run through the years 2000 and 2001. A maximum dose of 20 to 25 dpa is expected to be achieved.

n flux 10^{14}/cm^2

The samples from STIP-I have been transferred to the hot-lab of PSI. They will be unpacked, sorted and distributed. PIE of these samples will be performed jointly by PSI, CEA, FZJ, JAERI and ORNL. Some results will be available in the first half year of 2001.

Abstracts

Fusion Materials Development, CRPP-EPFL

PIREX irradiation facility

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Fusion Technology CRPP
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5232 Villigen PSI

For about two decades, the Fusion Technology-Materials Group of the Centre de Recherches en Physique des Plasmas at the EPFL has been studying the irradiation effects on materials, developing new alloys and carrying our more fundamental research in the fields. In order to achieve our goals, we make use of a secondary 590 MeV proton beam line of the Paul Scherrer Institute, on which our irradiation facility PIREX is built. The 590 MeV protons are actually used to simulate the radiation damage of the 14 MeV fusion neutrons. Up to date, tensile specimens have been irradiated at doses up to about 2 dpa over a temperature range of 40C to 450C. PIREX allows also to perform in-situ experiments of low cycle fatigue on tubular specimens.

The talk will present an overview of the PIREX irradiation facility and it will be outlined how the irradiated specimens are handled and tested in the hotlab. Furthermore, a description of our testing facilities in the hotlab will be done. Those include tensile and low cycle fatigue testing machines, wire saw, confocal microscopy, Charpy machine, 3 points bending etc…Finally, a description on the ongoing research and future of the program of the group will be presented.

Fabrication of CT specimens with inserts of irradiated materials by milling and EB- welding

Barbara C. Oberländer

Metallurgy Department, Nuclear Fuel, Institutt for Energiteknikk (ife), N-2007 Kjeller, Norway

The age of nuclear power plants is increasing steadily. Nuclear power plant life extension studies demand machining of irradiated materials for further testing. Therefor the Hot Lab at Kjeller has enlarged its remote handled machining capabilities with remote milling. A desription of the milling facility in the hot cell is given. Using the newly installed remote handled milling machine and EB-welding, compact tension specimens (CTs) with irradiated steel inserts in non-irradiated steel CT bodies were successfully fabricated. The irradiated inserts are located in the crack propagation region. The fabrication steps for CT specimens is decribed and a report given on crack propagation in the irradiated inserts and on quality control of the CT samples.

Refurbishment of PSI to comply with requested safety standards after N 40 years of operation

Conference on remote techniques applied

in Hotlabs, PSI, Herbst 2000

G. Bart, L. Wiezel

The PSI hotlaboratory has started its operation in 1962. From the very beginning the hotcell wing served for handling and gross PIE analysis of reactor core internals and highly active material from accelerator target stations. (figure 1). In the radiochemistry wing microstructural and chemical analysis of small highly active samples was subsequently performed with equipment installed in individual lead shielded cells. The radiochemistry wing also served for radiopharmaceutical nuclide dispatching and actinide ceramics preparation. Several constructions and building enlargements have been added since 1962 and naturally the safety infrastructure was improved case wise.

In the course of reevaluating the principal safety documentation for the hotlaboratory during 1994-95 it was realized that the building concept with its class A radioactive areas did not comply any more with modern safely standards, in particular with fire protection regulations and operator safety. In accident scenario analysis it was further demonstrated that radionuclide release to the environment could cause intolerable health risks to the surrounding population. It was therefore decided to principally refurbish the building infrastructure particularly with respect to fire protection, media, and laboratory instrumentation and control. The chosen concept consists in adding a so called media installation corridor on top and along the radiochemistry wing (figure 2) from which the individual labs on two floors are reached by vertical media access channels. Since a) the reconstruction outage time had to be minimised and b) there was not enough storage capacity to remove the whole Hotlaboratory equipment at once, a step wise reconstruction was planned with separation of specified blocks of labs which are freed from activity and which are accessed from the building outside, while the rest of the labs are still (or again) in use or being prepared for the construction work. The presentation describes the principal of the refurbishment campaign including costs, time table and logistics. Since the conference takes place at PSI, a tour would be available for interested visitors to discuss the logistical approach on site.

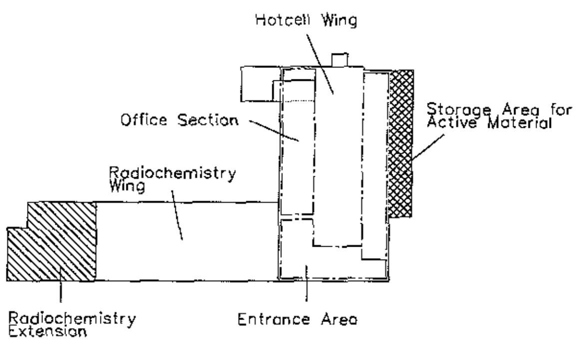

Figure 1: Layout of PSI Hotlaboratory

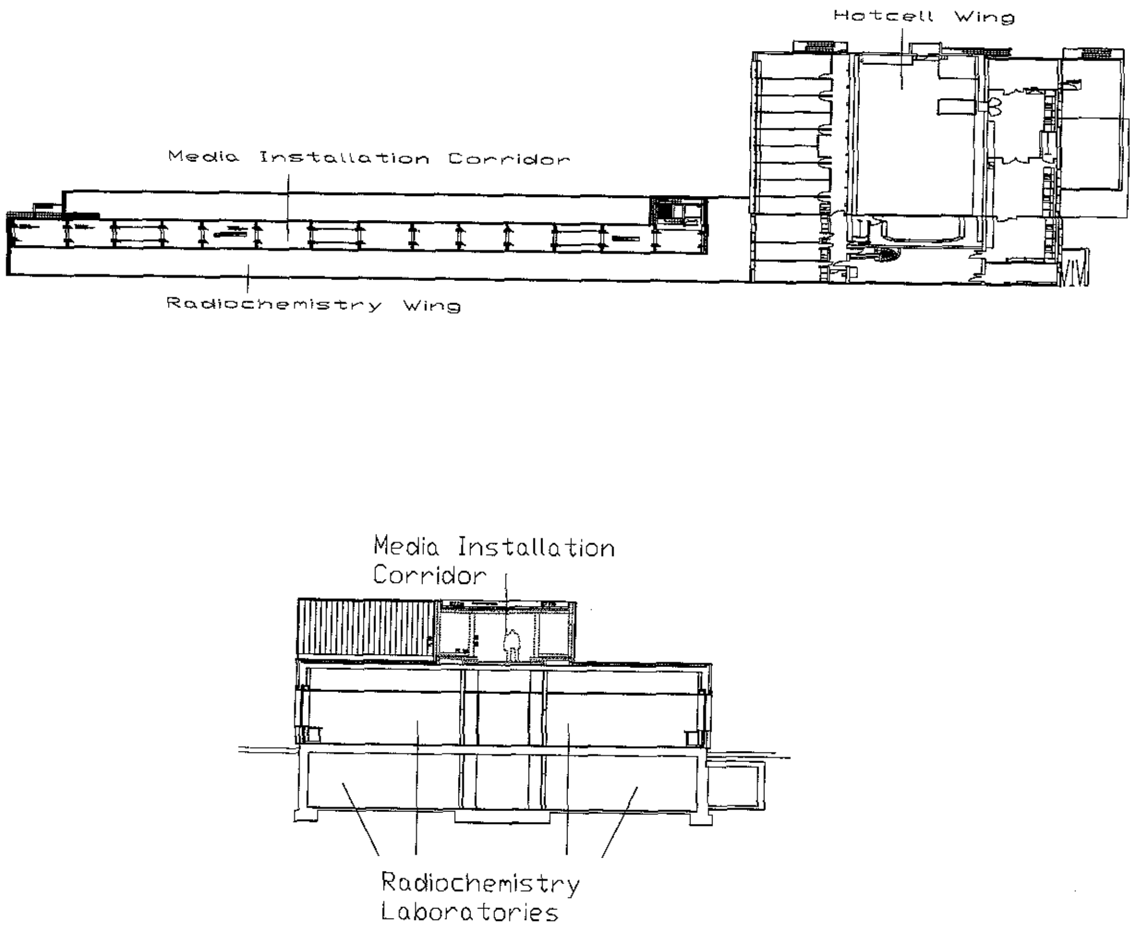

Figure 2: Concept of media installation corridor on the top of the radiochemistry wing

1.	S.	Abolhassani	PSI
2.	A.	Al Mazouzi	PSI
3.	G.	Bart	PSI
4.	G.	Berntsson	Studsvik Nuclear
5.	J.-Y.	Blanc	CEA Saclay
6.	R.	Brütsch	PSI
7.	H.A.	Buurveld	NRG Petten
8.	J.P.	Coad	Jet Project
9.	G.	Crawford	PSI
10	Y.	Dai	PSI
11	J.L.	Diaz	CIEMAT
12	K.A.	Duijves	NRG Petten
13	N.	Fogaca	LABMAT/CTMSP
14	D.	Gavillet	PSI
15	W.	Goll	Siemens Nuclear Power
16	I.	Golovtchenko	SCC RIAR
17	Y.	Gontcharenko	SCC RIAR
18	F.	Groeschel	PSI
19	A.	Hermann	PSI
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27	A.	Leenaers	SCK-CEN
28	J.-P.	Lévêque	CEA Cadarache
29	R.	Manzel	Siemens Nuclear Power
30	G.	Marangio	ENEA
31	M.	Martin	PSI
32	R.	Muzi	CEA Cadarache
33	P.	Novosad	Nuclear Research Inst. REZ
34	A.-Ch.	Nystrand	Studsvik Nuclear
35	B.C.	Oberländer	Institutt for Energiteknikk
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37	L.	Sannen	SCK-CEN
38	P.	Schleuniger	PSI
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40	Ph.	Spätig	CRPP-EPFL
41	W.J.	Stephen	BNFL Magnox Electric
42	J.	Suttcliffe	PSI
43	E.H.	Toscano	CCE CCR TUI
44	B.	Vos	SCK-CEN
45	B.	Wernli	PSI
46	H.	Wiese	PSI

EUROPEAN WORKING GROUP
on
HOT LABORATORIES AND REMOTE HANDLING

Villigen – Switzerland
PSI
September 27 – 29, 2000

Session II : Applied Research Topics
Chairman: Leo Sannen

The **first part** of Session II, encompassing 6 papers, was mainly dealing with **instrumental analyses**.

The first two papers dealt with the well-known ***EPMA*** technique exploring more in depth its application on both irradiated and unirradiated fuel.

The ***PSI*** paper focused on improved methodology in the standard application of EPMA on the quantitative determination of the radial distribution of U, Pu and the Fission Products. It was shown that good quantitative results can be obtained, despite radioactive background and the overlap of different lines, when applying appropriate measurement and analysis procedures.

The ***SCK•CEN*** paper showed the application of an advanced 2D measurement methodology enabling a well qualified quantitative determination of the Pu distribution in unirradiated MOX fuel. A representative area of 1×1 mm² can be analysed with high spatial resolution, high accuracy of global Pu content and medium accuracy of local individual measurements by applying an appropriate measurement methodology.

The third paper showed the application by ***BNFL***, Berkeley, of the ***SEM*** to quantify the morphology as well as the number density of Fission Gas bubbles at the grain boundaries of irradiated nuclear fuel. ***TEM*** is applied to reveal the smaller intragranular FG bubbles. Both these measurements contribute to a better understanding of the FG behaviour. Furthermore the development of a recently installed active ***SIMS*** facility has been described.

The forth paper, from the ***JET*** project, showed the ***ion beam analysis*** being an appropriate highly sensitive technique for the determination of the buildup of (hazardous) tritium and beryllium in fusion reactor plasma facing components as well as in the deposits resulting from the erosion of the plasma facing wall. As such it is a a very important tool to study the interaction of the plasma with the surrounding components and the therefrom resulting retention of tritium (and deuterium) in fusion devices.

The fifth paper, of ***RIAR***, reported that the combination of two surface analyses techniques, ***AES*** and ***SIMS***, and the use in both techniques of appropriate ion etching procedures, allows to determine the element distribution at the ZrO_2 – Zry interface. It is reported that this measurement methodology can be extended to study the element composition a any interface between a dielectric film and its metal basis.

The sixth paper compiled the ***sample preparation techniques*** and the wide variety of ***analytical methods*** that are available at the ***PSI*** hot lab for the characterisation of nuclear fuel. Regarding sample preparation, the dissolution of nuclear materials in a high pressure digestion apparatus, micro-wave and reflux apparatus has been described. Some examples of instrumental methods are the thermogravimetric method tuned for the O/M ratio measurement in MOX fuel and the HPLC-ICP-MS analysis of U, Pu and FP's omitting the need for elaborate and time-consuming separations to be performed.

The **second part** of the session, encompassing four papers, dealt with **specific applications**.

The first paper, of ***TUI***, described a ***cold finger apparatus*** to determine the FP retention capabilities of HTR fuel (coated particles type fuel). It enables the study of both the FG release characteristics as well as the solid FP release characteristics at temperatures of up to 1800°C, i.e. under accident conditions.

The second paper, of ***PSI***, dealt with the problem of ***gas communication*** within a high burnup fuel rod at the occasion of ***puncturing*** for fission gas release and free rod volume determination. Observation of the gas pressure evolution in time and puncturing both the top and bottom side of the rod, resulted in appropriate knowledge on the gas communication in high-burnup rods and in the implementation of an appropriate puncturing procedure which ensures proper fission gas release and free rod volume determination.

The third paper, by ***SIEMENS***, described a device to test the ***long-term creep behaviour of cladding***. The integrity of fuel rods during their long term intermediate storage has to be ensured. This is especially important for high burnup fuel characterized by increased n-fluence, H_2-content and increased rod inner gas pressure as resulting from the higher fractional FGR. Samples of irradiated fuel rods cladding are leaktight loaded with high He gas pressure and loaded in a furnace where they are heated up to 400 °C. They stay here for about one year and are unloaded at regular intervals to evaluate the creep.

The forth and last paper reviewed the ***mechanical test methods*** applied at ***RIAR*** on irradiated VVER fuel rod cladding. Appropriate samples are machined from the cladding to perform uniaxial tensile tests. Gas loading of tubular clad specimens is applied for biaxial testing. Both internal and external pressure is applied in the biaxial tests in order to perform full representative tests. Iodine attack from within the tube is applied as well, as well as fatigue testing including external hydraulic pressure cyclic changes. Finally a brief discription of the methods used for corrosion-thermal treatment of the specimens prior to their mechanical testing is given.

Leo Sannen