MINISTERIO DE CIENCIA Y TECNOLOGÍA

Ciemat Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas

XXXIX Plenary Meeting of the European Working Group

Hot Laboratories and Remote Handling

October 22nd to 24th, 2001

Ciemat , Madrid, Spain

ABSTRACTS

CSN CONSEJO DE SEGURIDAD NUCLEAR

enresa Empresa Nacional de Residuos Radiactivos, S. A.

SERIE PONENCIAS

MINISTERIO
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Y TECNOLOGÍA

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XXXIX Plenary Meeting of the European Working Group

Hot Laboratories and Remote Handling

Proceedings of the International Topical Meeting
on Hot Laboratories and Remote Handling

Introduction

1. Objectives

Hot laboratories installations and remote handling facilities are of paramount importance for the nuclear fuel cycle; both when close or open cycles are considered. These facilities have a basic role for drawing new and appropriated strategies for the Nuclear industry in several R&D fields like: life extension of nuclear power plants, increase of the fuel burnup, nuclear materials characterisation, decommissioning and dismantling of nuclear facilities and nuclear waste issues (handling, containers for interim storage, spent fuel characterisation for interim and final storage, etc.),

This meeting has been celebrated regularly from 1963 (see Table) and constitutes of a specific forum for hot laboratories and remote-handling facilities where the different issues in this field can be discussed, both from a technical and scientific point of view.

European Working Group "Hot Laboratories and Remote Handling"

Year	Date	Place	Number of meeting
1963	May 7-8, 1963	Cadarache, France	
1964	May 14-15, 1964 October 29, 1964	Ispra, Italy Petten, Netherlands	Meeting 7 Meeting 8
1965	October 21, 1965	Jülich, Germany	Meeting 9
1966	September 22-23, 1966	Casaccia, Italy	Meeting 10
1967	September 21-22, 1967	Karlsruhe, Germany	Meeting 11
1970	May 25-26, 1970	Cadarache, France	Meeting 12
1971	June 24-25, 1971	Jülich, Germany	Meeting 13
1972	June 08-09, 1972	Petten, Netherlands	Meeting 14
1973	June 14-15, 1973	Geesthacht, Germany	Meeting 15
1974	September 18-19, 1974	Winfrith, England	Meeting 16
1976	May 25-26, 1976	Casaccia, Italy	Meeting 17
1979	June 19-20, 1979	Risø, Denmark	Meeting 19
1981	May 21-22, 1981	Karlsruhe, Germany	Meeting 20
1982	June 10-11, 1982	Mol, Belgium	Meeting 21
1983	June 08-10, 1983	Petten, Netherlands	Meeting 22
1984	June 13-15, 1984	Harwell, England	Meeting 23
1985	June 26-28, 1985	Cadarache, France	Meeting 24
1986	May 21-22, 1986	Brasimone, Italy	Meeting 25
1987	September 23-24, 1987	Ispra, Italy	Meeting 26
1988	September 28-29, 1988	Jülich, Germany	Meeting 27
1989	September 27-28, 1989	Karslruhe, Germany	Meeting 28
1990	June 12-13, 1990	Risø, Denmark	Meeting 29
1991	June 25-26, 1991	Barnwood, England	Meeting 30

Year	Date	Place	Number of meeting
1992	June 24-25, 1992	Suze-la-Rousse (Marcoule), France	Meeting 31
1993	June 15-16, 1993	Chinon, France	Meeting 32
1994	June 14-15, 1994	Mol, Belgium	Meeting 33
1996	May 14-15, 1996	Petten, Netherlands	Meeting 34
1997	June 5-6, 1997	Studsvik, Sweden	Meeting 35
1998	September 21-23, 1998	Windscale, UK	Meeting 36
1999	October 13-15, 1999	Karlsruhe, Germany	Meeting 37
2000	September 27-29, 2000	Villingen, Switzerland	Meeting 38
2001	October 22-24, 2000	Madrid, Spain	Meeting 39

The objective of the last topical meeting was to discuss the actual status of knowledge in the refurbishment of facilities, new implementation and procedures used in hot cell operations as well as the decontamination in these facilities too. The meeting was intended to help to answer questions such as:

- The exchange information and to discuss on the analytical methods, their implementation in the hot cells, the methodology used and its application in nuclear related research.
- The share of experience on common infrastructure matters of exploitation of a hot laboratory, e.g., remote handling techniques, safety features, QA-certification, waste handling procedures, etc.
- To discus updates/renewals, to promote normalisation and to increase co-operation, e.g., by looking at manual complementarities.
- To prospect present and future demands from the nuclear industry towards hot laboratories and to draw strategic conclusions regarding further needs based on the actual situation and perceived future developments.

Finally, the conference aimed to fix the basis for the formation of a formal stable European Network with the support of the EU such as to allow longer European strategic consensus on nuclear research issues and associated infrastructure, i.e., nuclear waste issues as well as nuclear power plant issues.

2. Organisation of the Meeting

The technical programme of the meeting was elaborated by the International Steering Committee consisting of: Dr. L. Sannen (SCK CEN, Belgium), Dr. E. Toscano (JRC-ITU, European Commision), Dr. K. Duyves (NRG The Netherlands), Dr. J.P. Lévêque (CEA, France), Dr. B. Oberländer (IFE, Norway); together with Scientific Committe, constituted by: Mr. J.L. Díaz, Dr. J. Quiñones and Dr. J. Serrano (Ciemat, Spain).

The Meeting was held at the installation of CIEMAT, Madrid, Spain, from October 22nd to 24th, 2001.

3. General Discussions and Conclusions

3.1. Session 1: PIE - Techniques and Fuel Characterisation

Chairmen: ***E.H. Toscano***

Modern approaches to the Post-Irradiation Examination and Characterisation of Fuel were considered at the session. The session included 7 papers, 6 coming from 3 different countries (2 from the Russian Federation, 3 from France and 1 from Belgium) and 1 from an international organisation (European

Community, Joint Research Centre) together with an industrial partner (Framatome ANP) and the University of Helsinki.

1. The first paper came from RIAR-Dimitrovgrad and presented advanced techniques to study different properties of cladding materials and fuels. The characteristics and capabilities of an X-ray diffractometer and of a neutron radiography device (based on a ^{252}Cf-neutron source) were described. Specific engineering features and examples of applications aimed to predict the reliability of reactor core components and of fuel rods cladding materials and hexagonal tubes alloys were presented.

2. The second presentation came from CEA-Cadarache and described the capabilities of an X-ray radiography installation. The device is used mainly for the control of the quality and optimisation of TIG-welding of end-plugs of re-fabricated segments.

3. The third paper detailed the state of advancement of the installation of shielded Secondary Ions Mass Spectrometer at CEA-Cadarache. Some preliminary results n non-irradiated samples were reported.

4. The fourth presentation described the methods available at the Institute for Transuranium Elements concerning the characterisation of hydrides and the quantification of the amount of hydrogen pickup in irradiated cladding materials. The capabilities of a hot extraction device for the quantitative determination of the hydrogen present in the matrix, and the features of a Scanning Electron Microscope with a backscattered electron detector for the morphology and quantification of hydrides were presented. The characterisation of a high burnup cladding, having a dense hydride rim at the periphery was presented as an example of the application of these techniques.

5. The paper came from SCC-RIAR Dimitrovgrad and detailed the capabilities of this Institute for the characterisation of the thermal properties of fuel rods: thermal diffusivity, thermal conductivity, heat capacity, thermal linear expansion coefficient and differential thermal analysis. The devices were detailed described and some selected results presented.

6. The sixth paper described the activities at CEA-Saclay, Laboratory for Characterisation of Irradiated Material (LCMI), in the framework of the French safety program on Reactivity Initiated Accident (RIA), in the field of mechanical properties under transient conditions. An induction heating system was described. The system allows the realisation of heating rates to simulate some scenarios of the RIA and LOCA accidents.

7. The last paper, from SCK-CEN Belgium, described the capabilities of an autoclave to study Irradiation Assisted Stress Corrosion Cracking properties of structural materials. The device permits the simulation of the reactor primary circuit environment. The equipment, installed in a hot cell, will allow the handling of pre-irradiated samples.

The panel discussion estimated that the 7 papers reflected very good the present needs concerning the PIE-examinations in hot cells and the new fields opened for research in this field:

- **The need for more specialised techniques:** Increased average discharge burnup caused the appearance of new phenomena in the fuel (e.g., the rim-effect) and in the cladding (oxidation behaviour and hydrogen pickup) calling for specific techniques applied to the detailed research of this new issues. The presentations 1, 3, 4 and 5 reflected this trend. The application of SIMS to both cladding and fuel and SEM techniques to the cladding and the fact that the porous rim in the fuel can modify the heat transfer properties (paper No. 5), are some examples of the present needs in these fields.

- **Spent fuel as a waste form:** Many countries are currently considering the possibility to dispose the spent fuel in intermediate storage. This requires a more detailed investigation of the mechanical properties in such an environment. The conditions in a storage container (high temperature and internal pressure) may cause the appearance of creep phenomena. Since the amount, distribution and orientation of hydrides can affect the mechanical properties, a detailed knowledge of these characteristics in irradiated samples is necessary. Paper No. 5 was a contribution to this goal.

- **Licensing issues:** Reactivity Initiated Accident (RIA) and Ramp Behaviour (LOCA) are one of the important issues raised by licensing authorities all over the world. The proper sample preparation (paper 2) and adequate testing conditions (paper 6) have to be assured. New phenomena due to increased burnup and the utilisation of new cladding materials will keep these issues of interest in the coming years.

- **Life extension issues:** One of the most important issues concerning the economic nuclear energy production is the life extension of the existing reactors. In this context, the safety issues are of paramount importance to obtain a prolongation of the licensing of nuclear reactors. Paper 6 was a contribution to the elucidation of the proper testing conditions to this goal.

3.2. Session 2: Fuel back end waste issues

Chairman: **M. Huntelaar**

In this section 8 presentations where given which can be divided into three subgroups:

A. Reprocessing

B. Storage and behaviour of spent fuels

C. Decommissioning

Ad A Reprocessing

For reprocessing hot cells are mounted to optimise various research items such as: fuel dissolution, and the selective separation of transuranium elements and fission products from dissolved fuels. The goals of these investigations are to reduce the amount of current waste levels and to open future programmes for reducing identified waste streams.

Ad B Storage and behaviour of spent fuels

For long-term storage of spent-fuel a lot of basic research is being performed and will be necessary in the future still, to ensure a well founded safe storage in geological repositories. The stability of storage is found to depend on:

- Storage time,
- Burn-up levels,
- The amount of oxygen available in the system; a relation between oxygen potential, degradation levels and leaching behaviour is found,
- Formation of secondary phases, co-precipitates and fission gas release,
- Type of geological formation and the physical properties of ground water (pH, temperature, amount of dissolved O_2),
- Type of radiation (alpha, beta, gamma) and their contribution in the time frame of storage,
- Type of fuel cycle and waste form, varying from direct storage to closed cycle systems.

All the items mentioned above contribute to the mechanisms of leaching processes. Additional studies are necessary to fill-in the existing blank spots to be able to make better long-term prediction of leaching processes. This knowledge would support a decision on what type of waste form and storage is best.

Ad C Decommissioning

Additional model studies are required to understand and to prevent the formation of waste, as for instance in fusion reactors. Carbon-flakes containing tritium are observed in present tokamaks such as

JET, which lead to a built-up of tritium levels which would inhibit a long-term use in the future. Future experimental modelling studies should concentrate on: methods for removing tritium-debris, use of less carbon plasma facing materials, and the prevention of revolatilisation of hydrocarbons to more remote areas in the fusion reactor.

Traditionally hot cells are used for research on various topics, as for instance mentioned above. Nowadays more and more hot cells are being applied to the back end of the (fuel) cycle: the treatment of waste and the decommissioning of waste sites and the (partial) dismantling of buildings. In order to be able to guarantee the stability of future waste, packaging of new waste should comply with the following:

- Characterisation of waste (chemical state, physical contents, nuclide contents and activity, and weight) and pre treatment may be necessary to avoid problems by degradation of the contents. Specific problems are for instance the gamma-induced degradation of PVC that leads to the formation of chlorine gas and hydrochloric acid. The mechanisms leading to these degradation processes should be known if possible.
- Further minimisation of waste.
- The development and licensing of containers for transport and storage.

Recommendations

- Prevention of waste, not only in activity, weight and volume but also in the occurrence of non-treatable waste which cannot be disposed of.
- Spent fuel modelling. Prediction of long term behaviour of spent fuels stored in geological repositories under various conditions.
- Better co-operation in the use of transport containers. An overview of what containers are available in the various institutes and what types of connection systems are in use would be beneficial to all the institutes.

3.3. Session 3: Hot Labs and Nuclear Facilities

Chairman: **J.Y. Blanc**

The 3rd session was dedicated to the hot laboratories and nuclear facilities. It included eight papers provided by six different countries. They can roughly be classified inside three main topics:

Four papers were dealing with refurbishment of hot laboratories: B. Oberländer presented the new system of metallography preparation at Kjeller in association with Struers Company, O. Rabouille detailed the new microscope area implemented at Saclay, G.L. Tjoa described the new CNC milling machine now available at Petten, whereas J.Y. Blanc gave an overview of the whole refurbishment project going on at Saclay.

Two presentations were focusing on radiological protection, either during normal operation at Jülich, as explained by W. Kühnlein, or during a large refurbishment as presented by R. Brütsch for PSI.

The last two papers were describing shipping casks: a very large one, with the BG18 proposed by W. Claes from Mol and J. Sperlich from Transnubel or a very small one explained by D. Gavillet for PSI.

During discussion, it appeared that:

- Refurbishment of hot laboratories remains a classical topic for these meetings, because most of the European laboratories are old and need renewal in order to be safely operated with state-of-the-art techniques;
- Cask was a good subject for debate, as it appeared that many countries have similar logistics difficulties, and that partaking information can lead to save costs, or resolve shipping

problems: it is always interesting to know that a solution has been developed in a neighbouring country for a very similar problem;

- The increase in burn-up of the latest fuel rods leads to an evolution of cask agreements, it will be interesting to know what will be the best target value for maximum burn-up for a given cask;

- The rules for dealing with very low activity wastes are not similar from one country to another.

It is suggested to propose the cask situation in European countries as one topic for the next plenary meeting.

4. Acknowledgements

The organisers like to acknowledge the financial support by "Consejo de Seguridad Nuclear" the panish segurity council and "Empresa Nacional de Residuos Radiactivos, SA (ENRESA)" (Sapanish agency for radioactive waste management).

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Proceedings of the international Plenary Meeting of the European Working Group on Hot Laboratories and remote Handling

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Sponsors:

The organisers appreciate and gratefully acknowledge the financial support of the following organisations and companies:

Centro de Investigaciones Energéticas Medioambientales y Tecnológicas (Ciemat). Ministerio de Ciencia y Tecnología

Consejo de Seguridad Nuclear (CSN)

Empresa Nacional de Residuos Radiactivos, S.A (ENRESA)

Program of Presentation

Monday October, 22nd

9:00	Opening and welcome	Leo Sannen (SCK-CEN) J. L. Díaz (Ciemat)
	PIE – Techniques and Fuel Characterisation Chairs Enrique Toscano and José Antonio Gago	
	Iou. Golovtchenko, V.M. Kosenkov, V.Yu. Shishin	*Equipment and Techniques for Studies on Irradiated Fuel and Fuel Rods Characteristics*
	Francis Berduola	*Implementation of a cabin X-rays in hot cell*
10:00	*Coffee break*	
	B.Pasquet, L.Desgranges, B.Rasser	*Installation of a shielded SIMS in CEA Cadarache*
	E.Heikinheimo, W.Goll and E.H.Toscano	*Evaluation Methods of Hydrogen Pick-up in Irradiated High Burnup Fuel Rod Claddings*
	V.D. Risovaniy, D.N. Souslov, V.D. Shushakov, V.S. Sandakov	*Complex of techniques for investigaton of thermophysical properties of irradiated materials*
	Xavier Averty, Jean-Pierre Pizzanelli, Virginie Basini, C. Duguay	*Induction Heating on dynamic tensile tests*
	Steven Van Dyck	*The installation of an IASCC autoclave test system at the SCK•CEN hot laboratory*
13:00 – 14:00	*Lunch break*	
14:00 – 16:00	**Fuel Back End Waste Issues** Chair Mark Huntelaar and	
	G. Ferlay, N. Reynier Tronche, A. Vaudano, J-P. Dancausse	*Two years of R&D chemistry in C11/C12 Shielded Cells at Atalante Marcoule*
	Andreas Loida, Bernd Grambow, Nikolaus Müller	*Post Irradiation Examinations of High Burnup Spent Fuel Samples: Corrosion Test Procedures, Sampling and Sample Treatment*
	Mark Huntelaar, Herman Buurveld and Peter van der Hulst	*Waste Treatment Hot Cell*
	Sergio Bertelli, Roberto Covini, Carlo Rovei	*Study on containerisation of irradiated fuel and high level liquid wastes for medium/long –term storage at JRC Ispra*
	J P Coad, N Bekris, D Brennan, G Federici, A Perevesentsev et al.	*Tritium retention in JET and Next-Step Fusion Devices*

Tuesday October, 23rd

9:00	A. Martinez-Esparza, J. Esteban, J.L. Gonzalez	*R&D programme on irradiated fuel and new technologies on separation*
	J. Cobos, J.P. Glatz, J. de Pablo, B. Christiansen	*New flow through reactor installed in the ITU Hot Cell laboratory to investigate the dissolution rates of the irradiated fuels*
	Javier Quiñones, Joaquin Serrano	*Studies of the influence of water radiolisis to the spent fuel matrix dissolution process*
10:00	*Coffee break*	
10:30	**Hot Labs and Nuclear Facilities** Chairs J. Y. Blanc and J.L. Díaz	
	Kuehnlein Winfried	*Radiation protection instrumentation at the hot cells of Forschungszentrum Juelich – the new digital data acquisition and visualisation system*
	Olivier Rabouille, Jacky Viard, Martine Ménard, Stéphane Allègre	*The new Isidore microscope*
	B.C. Oberländer, M. Espeland, N.O. Solum	*Experiences from Refurbishment of Metallography Hot Cells and Application of a New Preparation Concept for Materialography Samples*
	G.L. Tjoa, C.M.E. van Thoor and P. Boekhout	*A New Milling Machine in NRG's Hot Cells Facility*
	Jean-Yves Blanc, Christian Chéron, Francis Lefèvre	*Refurbishment of the LECI*
13:00 – 14:00	*Lunch break*	
	W. Claes and J. Sperlich	*The BG 18 container: a B(U)F type packaging for the transport of irradiated fuel rods or materials between nuclear power plants and research laboratories*
	D. Gavillet	*Development of a specimen containment system and measurement procedures for the neutron diffraction analysis on irradiated fuel materials in PSI*
	R. Brütsch	*Overview of the organisation and the measurements procedures used for the radioactive contamination controls during the refurbishment of the PSI hot laboratory*
16:00	*General Discussions and Conclusions*	
21:00	*Official Dinner*	

Wednesday October, 24th

	Travel to Spanish LLW & MLW repository "EL Cabril" (Cordoba)

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2. Fuel Back End and Waste Issues

2.1. TWO YEARS OF R&D CHEMISTRY IN C11/C12 SHIELDED CELLS AT ATALANTE MARCOULE.

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ABSTRACT:

After 4 years work project, a new shielded cells facility is set into service in Atalante Marcoule under the "codename" C11/C12.

It is devoted to the head-end high activity reprocessing studies. Physically, C11/C12 is constituted of eleven working places behind one meter of both concrete and leaded glasses biologic wall. The conception of C11-C12 was decided and organized to permit a large variety of experiments.

These equipments permit among other things studies on the following steps:

- Fuel and target mechanical treatment,
- Thermal treatments under inert gas flow,
- Dissolution using several media like nitric acid or complexing agent,
- Solid-liquid separation by filtration or centrifugation,
- Elements separation using liquid-liquid extraction apparatus like mixer-settlers or centrifugal contactors,
- Elements separation using solid-liquid chromatography.

From June 99 to april 2001, the principal experiments realised were consisted to:

- June-august 99: MOX fuel dissolution
- September 99: Liquid-liquid separation of U, Pu from fission products and minor actinides by the PUREX process,
- October 99: Test of DIAMEX process (selective separation of trivalent elements versus other fission products)
- December 99: Test of SANEX process (selective extraction of trivalent actinides versus lanthanides).
- June 00: Test of SESAME process (selective extraction of americium versus curium)
- September 00: Test of SANEX II process
- November 00: Test of DIAMEX process in presence of complexing reagents to improve selectivity versus Pd,
- April 01: Test of CALIXARENE process for selective extraction of Cs from DIAMEX raffinate
- April 01: dissolution of several pièces of irradiated fuel for metallurgical cladding tests at the CEA / LECI.

The next years will be devoted to increase and diversify experiments in respect with quality and waste management, reducing the size of apparatus and amount of needed radioactive materials.

2.2. POST IRRADIATION EXAMINATIONS OF HIGH BURNUP SPENT FUEL SAMPLES: CORROSION TEST PROCEDURES, SAMPLING AND SAMPLE TREATMENT

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ABSTRACT:

One important objective of Post Irradiation Examinations (PIE) of spent fuel is to study its overall corrosion behavior under conditions, which are as close as possible to the geochemical environment possibly encountered in a repository after an assumed groundwater intrusion. This comprises corrosion tests using real spent fuel samples and near field materials, under the anticipated conditions. The progress of spent fuel dissolution and the associated release/retention of radionuclides is determined by periodical analyses of fission and radiolysis gases, solutions, and after test termination by determination of the amount of radionuclides retained in/on solid phases by various processes as (co)precipitation, formation of new secondary phases, or by sorption.

All kind of experimental work focused on spent fuel is associated with a high extent of technical expenditure, because in any case it has to be carried out under remote operation in hot cells due to the high activity of the fuel samples. Additionally, high efforts are necessary to protect the samples permanently against O_2 uptake from air. Because of the deficit of the spent fuel matrix in oxygen extreme small quantities of O_2 are already sufficient to cause its very rapid oxidation upon the surface. Thus, in the case of pre oxidation the UO_2 matrix dissolution rate will be enhanced orders of magnitude, which will in no case reflect the reality. Moreover, during the entire corrosion test, usually performed under anoxic conditions, the effective protection of the system "spent fuel – solution - gas phase" against air oxygen is required.

Spent fuel corrosion tests are performed in gas tight vessels, either in Ti/Pd lined stainless steel autoclaves, or in quartz glass vessels. Both types of vessels are equipped with fittings, which allow (1) the introduction of the desired atmosphere (mostly Ar), (2) to take samples of the gas phase, (3) to take samples of the solution under streaming Ar, and (4) to remove samples of solid particles during the test. The tightness of the system against air intrusion is verified by the results of the gas analysis, in particular by the amount of N_2, which belongs neither to the fission nor to the radiolysis gases.

The evolution of the gas composition in particular the release of fission gases is considered to be an additional independent indicator for the progress of matrix degradation, because they are not involved in any retention processes as sorption, precipitation or secondary phase formation. Sampling of gases is carried out by connecting the reaction vessel and a pre evacuated gas collection cylinder, with a vacuum pump and an Ar gas supply. To remove the air the connecting tubes are flushed with Ar and afterwards evacuated several times prior to open both the reaction vessel and the evacuated gas collection cylinder, which is plugged in outside the hot cell via filter gasket. Hence, no decontamination procedures are necessary before establishing the connection between the gas collection cylinder and the quadrupole mass spectrometer to analyze the gas composition.

Sampling of solution is performed by means of a syringe, which is connected by flexible Tygon tubes via a three port valve with the solution inside the vessel and an Ar gas supply. The connecting tube system is flushed with Ar prior to take the solution sample. To avoid the uptake of fuel particles the

solution sample will be filtered by using a 0.45 µm filter. Afterwards an aliquot is filtered through a 1.8 nm filter to identify the upper limit of possible colloid formation by means of radiochemical analyze of the filtered and the non filtered aliquot.

To handle solid material present in a reaction vessel during spent fuel corrosion an optical light microscop allowing steric view in association with micromanipulator technique will be applied for first inspections and for selection of fragments for further investigations. Solid materials of interest are e.g. particles of spent fuel, corroded Fe-powder, Zircaloy cladding and near field material. The micromanipulator technique enables the selection and the separation of very small particles, by fixing them on a sample holder. These specimens are characterized by a dose rate, which allows their transfer from the interior of the hot cell to the analytical equipment for further investigations (e.g. SEM, XPS, XRD). None of the investigated samples, inclusive spent fuel particles exceeded a dose rate of 18 µSv/h.

Besides detailed descriptions of the applied experimental procedures and achieved experiences related to our work on spent fuel some recent results will be presented.

2.3. WASTE TREATMENT HOT CELL

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ABSTRACT:

After completion of the HABOG facility (High Active Above Ground Storage Building) in Borsele (The Netherlands) in the year 2003, the stock of high-active nuclear waste at ECN / NRG will have to be prepared for storage in this new facility. To this purpose a dedicated waste treatment Hot Cell will have to be built in the Hot Cell Laboratories.

To minimise the amount of waste and to guarantee the long-term stability the waste will have to be conditioned in a number of steps. After opening of the present waste drums a manual selection of high-active (> 20 $mSv{\cdot}h^{-1}$) and intermediate-active waste (< 20 $mSv{\cdot}h^{-1}$) will be performed remotely. To this purpose the Hot Cell will be equipped with a gamma camera and a cutting device as additional tools. The intermediate level waste will be transported as usual.

Historically also PVC containing materials were disposed of in the same high-level environment. Due to the high gamma field of for instance Co-60 this PVC has disintegrated into a tar-like product and chlorine gas. In time and in combination with water this provides an aggressive chemical environment unsuited for long term intermediate storage. Therefore a special facility will be developed and integrated into the new Hot Cell to separate this PVC contamination from the high-active waste.

To avoid unnecessary contamination at the HABOG-facility the exterior of the new waste drums is not allowed to be contaminated. This inhibits the loading of the waste drums inside the Hot Cell. In stead only the interior will come in contact with the environment of the Hot Cell by means of an opening and an appropriate seal. Before the waste drum is loaded into the transport container the dose rate, the nuclide inventory, and the weight will have to be measured.

2.4. STUDY ON CONTAINERISATION OF IRRADIATED FUEL AND HIGH LEVEL LIQUID WASTES FOR MEDIUM/LONG –TERM STORAGE AT JRC ISPRA

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ABSTRACT:

During the last 40 years huge amounts of wastes arising from past experiments have been generated at JRC Ispra. These wastes are now stored on site in unconditioned form and must be characterised and re-conditioned to ensure their acceptance by future repositories.

Among the several types of wastes processed, spent fuel has a great importance in JRC waste management. In fact, there are more than ten tons of irradiated material, varying widely from commercial to experimental fuel elements or pins, in form of oxides and metal fuel, with very different geometry, dry and wet stored.

In addition, a few hundreds of HLLW have been produced in post-irradiation experiments, namely chemical dissolution, separation and partitioning processes, etc..

This study is an investigation on the most suitable solutions for medium/long term storage for both spent fuel and HLLW assuming that medium/long-term storage is sensitive to the volumes involved and the categorisation of wastes generated.

Preliminary studies on spent fuel have led to two main options: 1) containerisation of the material, as it is, in suitable casks or 2) reprocessing.

Concerning the HLLW, two options are under scrutiny: 1) vitrification in an external facility or 2) in-house conditioning by means of cementation.

The most relevant technical aspects of both projects are here discussed.

Moreover, the lack of historical data linked to the material under discussion and the more stringent rules imposed by regulatory authorities for transportation and storage, have induced JRC to perform a characterisation campaign on such material. The preliminary results of this campaign are also briefly presented.

2.5. TRITIUM RETENTION IN JET AND NEXT-STEP FUSION DEVICES

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ABSTRACT:

A major issue for future fusion power-generating plants is the retention of tritium within the machine, since this reduces the efficiency of the recycling process (leading to costs for extra tritium) and introduces additional safety considerations. The retention seen in existing machines is primarily associated with the re-deposition of carbon eroded from contact points between the plasma and the surrounding walls. In JET after operation with deuterium-tritium mixtures, the great majority of the tritium trapped in the vessel was in carbon-based deposits that flaked off and fell to the bottom of the vessel, where access is difficult. Future machines will also have an allowable limit for the amount of dust, and in JET the break-up of these flakes represents a possible source.

The issues of tritium retention and associated dust in a Next-Step device are being addressed in several ways. Firstly, the use of carbon is being minimised. Carbon is the only material that is tolerant of high surface power loads resulting from off-normal events, so it is not yet clear if the material of the critical contact area may be substituted. Secondly, the deposition mechanisms are being studied, with the aim of minimising the quantity of deposits and, within them, the fraction of tritium retained. Thirdly, ways are being developed of removing any deposits that do form within the device, ideally in such a way that the tritium can be returned (as tritium gas) to the storage beds. This is coupled with work on the more general issues of de-tritiation, such as methods of reducing the tritium content of carbon tiles, and reducing medium-level waste to low-level waste for subsequent disposal.

This work has been conducted under the European Fusion Development Agreement and is partly funded by the UK Department of Trade and EURATOM.

2.6. R&D PROGRAMME ON IRRADIATED FUEL AND NEW TECHNOLOGIES ON SEPARATION

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ABSTRACT:

Enresa, the Spanish Radioactive Waste Management Agency was set up in Spain in 1984 as a state-owned company, in order to be responsible for management of this type of wastes in the country.

The company is supervised by the Government, to whom is obliged to submit an Annual Report of Activities and a Proposal for general Radioactive Waste plan on a yearly basis

After creation of Enresa (1985) there was a change of strategy for HLW, LWR spent fuel is considered a waste. In different radioactive waste plans approved by the Government and in R&D programmes geological disposal is the only option.

In 98/1/29 the government takes decision of delay up to 2010 the approval by the Parliament of the option more convenient for final disposal according with the state of the technology.

The Research & Developments programmes of Enresa linked to the general Radioactive waste management plans are based in: Financing and management by Enresa. Development and creation of experimental research external groups (Universities, companies, institutions). Closed linked to EU and international RDT programmes.

From 1986 up to 1999 ENRESA has implemented 3 R&D programmes. The principal objective of those plans was the development of technologies for the final waste management based in a deep geological disposal (AGP). Those plans were closely linked to the 3rd and 4th EU RDT programmes.

From 1999 - 2002 ENRESA 4th R&D plan (99-2002) in force, linked to the 5th EU RDT programme, will have as principal objective to deepening in knowledge and applications and verifications of the technological infrastructure created in previous plans based in irradiated fuel as waste form for disposal. The principal management option will be the storage of spent fuel in a repository. Partitioning and transmutation as way of decreasing volume and radiotoxicity of waste form will be studied, so that, any decision than might be taken around year 2010, will be based on any feasible strategy.

A disposal concept for spent nuclear fuel and high level waste based on a deep geological repository was defined. In the last years, a large range of activities has been devoted to the performance assessment of a generic repository, called Enresa 2000.

Spent fuel performance assessment requires evaluation of its long-term ability to isolate and immobilise individual radionuclides after groundwater contact and it is necessary to know the waste behaviour under repository conditions.

The evolution of irradiated fuel under interim storage or deep geological conditions and the oxidation (O/M) rate before the water access to the fuel surface, is a factor of great influence on enhancing dissolution.

Those studies have been undertaken by means of chemical analogues, real spent fuel and artificially

aged irradiated fuel.

This work presents results on ENRESA's previous Research &Development plans, and the activities in progress carried out specially on irradiated fuel, in accordance with the 5th General Radioactive Waste Plan (GRWP)in force, focused specially in tests performed with real fuel in hot-cells of CIEMAT or in international collaboration with Institute of Transuranium elements of Karlsruhe, although the main part of the program is based on use of chemical analogues of spent fuel, such as SIMFUEL an UO_2., due principally to its lower costs and than presently there aren't hot cells facilities in operation for PIE fuel studies availables in Spain. Besides, there is a collaboration agreement on spent fuel research with ITU, following the guidelines of the GRWP to promote international cooperation.

The 5th General Radioactive waste Plan means a change in the orientation of the R&D that Enresa has been performing up to date. On the performance of spent fuel, this plan will focus in clarifying questions remaining open from earlier programmes, in order to increase understanding and improving the performance of irradiated fuel as a waste form. Basic research and infrastructure of other spent fuel management strategies will be implemented. The principal management option will be the storage of spent fuel in a repository. But partitioning and transmutation will be studied so that any decision that might be taken around year 2010, will be based on having enough information on any possible strategy.

A general overview and significant conclusions of this programme are presented, focused on projects in the field of spent fuel lixiviation and the research groups and infrastructure created and developed.

Some resources will be dedicated to exploratory research on partitioning and transmutation as high-level waste management strategies, aiming at show that a given elementary process could be efficient in the separation of nuclear materials.

In this direction has been created two new groups in Ciemat for studies on hydrometallurgical and pyrometalllurgical separation, joining to the European projects in this area.

2.7. NEW FLOW THROUGH REACTOR INSTALLED IN THE ITU HOT CELL LABORATORY TO INVESTIGATE THE DISSOLUTION RATES OF THE IRRADIATED FUELS

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ABSTRACT:

In order to study the dissolution rates for the different radionuclides, the effect of water radiolysis and to elucidate the dissolution *mechanisms of the different radionuclides contained in* the spent fuel matrix, irradiated spent fuel and UO_2 was used. This study is performed as a part of the collaboration programme [ENRESA - CIEMAT - ITU (EC DG/JRC)] to provide a source term for use in a performance assessment calculation.

For the determination of the dissolution rates a continuos flow through reactor specially designed for hot cell handling was built. This reactor allows the control in situ of different important parameters for leaching experiments such as, redox potential, pH and temperature.

These leaching experiments reported the effects of four important parameters (redox potential, pH, carbonate concentration and temperature) on the dissolution kinetics of the spent fuel matrix phase. The kinetic of dissolution of irradiated UO_2 fuel has been studied in synthetic granite groundwater under oxidizing conditions at room temperature. Preliminary results indicate that for spent fuel, dissolution rate depends on the burnup, being the dissolution rate calculated for the UO_2 LWR fuel with a burnup of 53 MWd/kg U of 2.66 10^{-10} mol m^{-2} s^{-1} and of 6.77 10^{-11} mol m^{-2} s^{-1} g^{-1} for the spent fuel of 29 MWd/kg U.

2.8. STUDIES OF THE INFLUENCE OF WATER RADIOLISIS TO THE SPENT FUEL MATRIX DISSOLUTION PROCESS

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ABSTRACT:

The disposal of high level radioactive waste in geological deep repositories relies on the long term stability of spent fuel matrix, which must be assured for thousands of years. One of these factors considered within the studies of performance assessment on spent fuel under final repository conditions is the effect of the radiation on its leaching behaviour. Due to the radiation from spent fuel can modify some properties of both solid phase and leachant and therefore it would alter the chemical behaviour of the near field.

Particularising in the effect of the radiation on the leachant, it will cause generation of radiolytic species that could change the redox potential of the environment and therefore may bring on variations in the leaching process.

In this work, we compiled the leaching experiments performed in a irradiation facility (Nayade), in order to emulate γ radiation field of a spent fuel at different cooling times. Initial dose rate used was 0.014 Gy/s using source of ^{60}Co. The spent fuel chemical analogue utilised was SIMFUEL (natural UO_2 doped with non-radioactive elements simulating fission products) and the leachants selected were saline and granite bentonite waters both under initial anoxic conditions.

Preliminary results indicate that radiation produces an increase of the uranium dissolution rate, being the concentrations measured close to those obtained in oxic atmosphere without radiation field. In addition the solubility solid phases from experimental conditions were calculated, for both granite *bentonite water and 5 m NaCl media.*

On the other hand, a tentative approach to model the role of γ radiolysis in these SIMFUEL tests has been carried out as well.

3. Hot Labs and Nuclear Facilities

3.1. RADIATION PROTECTION INSTRUMENTATION AT THE HOT CELLS OF FORSCHUNGSZENTRUM JUELICH – THE NEW DIGITAL DATA ACQUISITION AND VISUALISATION SYSTEM

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ABSTRACT:

In nuclear facilities and radioactive laboratories radiation protection is an important task. An extensive instrumentation is necessary to monitor the radiation levels and limits at the different locations and to record the acquired data for archiving. In the past different systems were used starting with analogue measuring instruments and plotting devices and ending with digital instruments and printers.

Since the beginning of the '80s, at the control room of the Hot Cells of the Forschungszentrum Juelich analogue display instruments (panelmeters), plotting devices and a digital process computer system were used for the acquisition and documentation of the signals from the different radiation measuring devices. In the middle of the '90s, the government demanded, that the data have to be recorded and stored for 30 years. Since the hardware became older and older, and support was no longer available for hard- and software of the process computer system, it was decided to find a new data acquisition and documentation system, that would meet the demand of the supervising authorities and fulfil additional wishes for an advanced system.

First of all the new system should use standard components. Together with SIEMENS company a combination of SIEMATIC S7 data acquisition system and the MEVIS data processing and visualisation software was chosen. This software is operated on a standard PC under the operation system Windows NT4. Analogue signals (0 – 20mA) and digital levels from the radiation measuring instruments are acquired at time steps of 1 ms and compressed to 15 s-data. These data are checked against the threshold values. The mean values of the time step of one minute and the min- and max-values of this time step are stored in the data acquisition system for two hours and transferred to MEVIS data processing system for monitoring and storage on the computer hard disk. The history of interesting data can be displayed graphically on the monitor with different time scales. A printout of the average values during the last 8 hours of all measuring locations is triggered automatically. In case of alarms and exceeding of limits the location number and the actual value are always printed out in addition to the signalisation on the display board of the control room. Limits are not only defined by absolute values but also by the signals slope. Exceeding of limits can be predicted at a very early state. All data are stored for about two month on the hard disk of the computer and than archived on a CD-ROM. Since the computer is connected to the local area network, registered users have access to the stored data for displaying and analysing locations of interest.

The new system has been in operation since the beginning of this year and shows good results of performance and availability.

3.2. THE NEW ISIDORE MICROSCOPE

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ABSTRACT:

In the frame of the refurbishment of LECI hot laboratory in Saclay, it was decided to renew one of the two metallography lines of the building. This line is situated at one end of the "Isidore" line of lead-shielded hot cells.

The work started by the cleaning of 5 out of 9 cells in Isidore line. Two were 2 m x 1.5 m cells, whereas the 3 others were smaller. Decontamination was difficult in both larger cells, because a lot of metallographic preparation had been performed there and because the cleaning of the lower parts of the cell, below the working area, was uneasy by remote manipulators.

The refurbishment of the cells included :

- changing the windows, because old windows were made of glass panels separated by oil, which is now prohibited by safety requirements,
- putting of a new pair of manipulators on one large cell, and adding bootings on manipulators on both large cells,
- changing all the ventilation systems in these cells (new types of filters, new air-ducts),
- modifying and changing metallic pieces constituting the working area inside the cell,
- increasing the height of the small cells in order to add a manipulator for charging the sample on microscope or on hardness machine,
- simplifying the electrical wiring in order to decrease the fire risk in the hot cell line,
- add a better fire protection between the working area and the transfer area, i.e. between the front and the rear part of the cells.

The scientific equipments for these cells are :

- an Olympus microscope, modified by Optique Peter (company based in Lyon), equipped with a motorised sample holder (100 x 200 mm), maximum size of sample : O.D. = 100 mm, 6 magnifications : x12.5, x50, x100, x200, x500 & x1000, two microhardness positions : Vickers et Knoop, Polaroid image and digital camera with SIS image analysis system
- a new periscope manufactured by Optique Peter, magnification x3 and x9, digital image and SIS system, and an old periscope,
- a Testwell hardness machine : Vickers, Knoop and Rockwell from 29.4 to 981 N,
- preparation system for TEM thin foils,
- an embedding system for EPMA sample designed for Sn-Bi metallic embedding,
- several polishing equipments and epoxy embedding system for metallography.

After completing installation of the equipments, tests have been performed to check the radioactive shielding of the cells, as well as a control of air tightness. The work is now nearly finished and the cells will soon be in operation. The main difficulties have been encountered when cleaning the larger cells, and also to perform the refurbishment with other works going on in neighbouring cells (electrical wire modifications, changing windows).

Next step will be to perform the same type of work on the other metallography line, which includes 3 cells. If funding is available, this can start in 2002 or 2003.

3.3. EXPERIENCES FROM REFURBISHMENT OF METALLOGRAPHY HOT CELLS AND APPLICATION OF A NEW PREPARATION CONCEPT FOR MATERIALOGRAPHY SAMPLES

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ABSTRACT:

After more than 30 years of operation the lead shielded metallography hot cells needed a basic renewal and modernisation not at least of the specimen preparation equipment.

Preparation of radioactive samples for metallography and ceramography in hot cells can be challenging and time consuming since it demands a special design and quality of all in-cell equipment and often a high degree of skill and patience from the operator. Essentials in the preparation process of samples for metallography and ceramography are simplicity and reliability of the machines, and a good quality, reproducibility and efficiency in performance. Desirable is process automation, flexibility and alara. amount of radioactive waste produced per sample prepared.

State of the art preparation equipment for materialography seems to meet most of the demands, however, it cannot be used without modifications in hot cells.. Therefore, IFE and Struers modified a standard model of a Struers precision cutting machine and a microprocessor controlled grinding & polishing machine for Hot Cell application. The later machine supports a state of the art grinding and polishing system using magnetic discs. Design modifications necessary to the machine included adapting of parts to allow remote handling and modification of parts to resist the radioactive environment. The system is equipped with a memory and control unit to control every single parameter in each step. Parameters such as preparation time, force, rotation (rpm), type of abrasive suspension, lubricant, dosing level and preparation disc used can be saved, recalled and run again at any time.

Hot cell utilisation of the microcomputer controlled grinding & polishing machine and the existing automatic dosing equipment made the task of preparing radioactive samples more attractive. The new grinding & polishing system for hot cells provides good sample preparation quality and reproducibility at reduced preparation time and reduced amount of contaminated waste produced per sample prepared. The sample materials examined were irradiated cladding materials and fuels.

3.4. A NEW MILLING MACHINE IN NRG'S HOT CELLS FACILITY

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ABSTRACT:

Preparations are in progress to install a new milling machine in the NRG's hot cells facility in the course of this year.

The milling machine is CNC controlled and adapted for use in a hot cell environment. Special arrangements are made to comply for this purpose. Since a number of components are not fully resistant to radiation, the machine is wheeled allowing it to be removed from the hot cell if not needed.

Some technical specifications from the various manufacturers will be discussed which finally lead to the selected supplier. Primarily, this only concerns the flexibility of the total equipment to adapt for remote control and not the general technical aspects of the equipment.

The machine will be used mainly to manufacture mechanical testing samples from irradiated materials, obtained from both welding experiments and other irradiated components. Special auxiliary tools are made for this purpose to facilitate the machining of the samples. Next, the convenience to program the system for machining mechanical testing samples to meet the specified requirements is also important.

Before installing the equipment in the hot cell a try out is performed to evaluate the system.

3.5. REFURBISHMENT OF THE LECI

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ABSTRACT:

The LECI is a hot laboratory built in Saclay in the early sixties for examinations on fuel rods, with 25 hot cells. Around 1995, a refurbishment programme up to 2004 was decided and started. It includes the renovation of about half of the cells of the existing building and the construction of a new building with about twenty lead-shielded hot cells for mechanical testing. At mid 2001, this paper presents the status of the project and the perspectives for the next years.

These modifications aims :

- to increase sample preparation and examination capacities on nuclear metals : mainly zirconium, steel and aluminium alloys,
- to keep existing P.I.E. facilities on short P.W.R. fuel rod as support for ramp testing programmes in the nearby Osiris reactor and as support for new cladding development programmes,
- to gather in LECI mechanical testing facilities which are up to now located in another facility to be shut down at the end of 2003.

Concerning the existing building, most of the planned refurbishment has been performed and 10 cells have been cleaned and 8 of them will be reequipped at the end of 2001 : a metallography line with new microscope, hardness testing, periscopes, TEM thin foil and EPMA preparation, two cells for tooling mechanical samples (milling machine, lathe, spark erosion), one cell for clad creep testing on long term storage conditions and a cell with a 25-tons tensile machine.

The new building is built, the lead cells will be installed in 2002 and most of the scientific equipments have been ordered. They include: wire erosion machining, 3 tensile machines with extensometry, 2 Charpy, different creep and internal pressure machines, autoclaves, EPMA and Raman analyses. The schedule is to open this building to irradiated materials (no fuel except on EPMA) at the end of 2003.

Some difficulties such as the public enquiries have been successfully overcome, some financial constraints have delayed the project of about one year, and technical difficulties have slightly modified the scientific equipments, but on the whole the objectives have been completed up to now.

In the next years, the challenge will be to successfully achieve in time the completion of the new building including a completely renewed set of safety reports, to refurbish a few other hot cells in the old building to comply with always more stringent safety requirements, altogether with going on with the examinations for our customers.

3.6. THE BG 18 CONTAINER: A B(U)F TYPE PACKAGING FOR THE TRANSPORT OF IRRADIATED FUEL RODS OR MATERIALS BETWEEN NUCLEAR POWER PLANTS AND RESEARCH LABORATORIES

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SCK•CEN
**TNB

ABSTRACT:

The BG 18 package was constructed in 1984 and has been approved according to the applicable IAEA safety regulations until 1995.

In 1998 the SCK•CEN and TNB (Transnubel) started with the upgrading of the container in order to make its use easier. The approval certificate in accordance to the IAEA safety regulations 1985 Edition (As amended 1990) to be delivered by the BfS in Germany is expected by mid 2001.

The allowable contents cover most of the industrial or experimental irradiated fuel types: UO_2 or MOX, PWR or BWR, including high enrichment and burnup.

The design of the BG 18 container and the available auxiliary equipment make possible the handling and vertical loading/unloading under water as well as horizontal loading/unloading against a hot cell.

The capacity of the inner containment system allows the transport of up to 30 full-length fuel rods.

The SCK•CEN as owner of hot cell facilities and of the container is in a position to offer a complete service for examination of irradiated materials.

Transnubel as applicant of the BG 18 container can offer a complete transport service including all the transport formalities and the technical assistance for loading and transporting.

A first application of the modified package for full size irradiated fuel rods from a German power plant to the SCK•CEN hot laboratories is already planned for 2001.

3.7. DEVELOPMENT OF A SPECIMEN CONTAINMENT SYSTEM AND MEASUREMENT PROCEDURES FOR THE NEUTRON DIFFRACTION ANALYSIS ON IRRADIATED FUEL MATERIALS IN PSI

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ABSTRACT:

PSI is actually operating a large neutron diffraction facility based on a neutron spallation source (SINQ). All spectrometer are installed in a large open hall without the required safety devices for the manipulation of irradiated nuclear fuel.

In order to make a phase analyses of a corium specimen, we have developed a safe container system that can be loaded in the PSI hot laboratory and transported in the neutron diffraction hall without further manipulation of the specimen. A set of safety procedures have also been defined in order to allow the measurement of irradiated fuel material.

The containment system will be fully described and the procedure developed for the contamination control and transport will be explained. Results obtained on a corium specimen will be presented to demonstrate the interest of the method for fuel analysis.

3.8. OVERVIEW OF THE ORGANISATION AND THE MEASUREMENT PROCEDURES USED FOR THE RADIOACTIVE CONTAMINATION CONTROLS DURING THE REFURBISHMENT OF THE PSI HOT LABORATORY

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ABSTRACT:

The PSI Hot-Laboratory (HL) is undergoing a large refurbishment of its ventilation system in order to bring it to the modern fire-safety requirement and earthquake safety requirement.

During the realization of the work, a large amount of materials had to be removed from the laboratories. In order to reduce the amount of radioactive waste, a large part has been thorough fully checked for possible radioactive contamination and if necessary cleaned. This procedure has allowed the disposing of a large part of the materials as normal waste, reducing greatly the costs.

In addition, the mounting of the new installation had to be realised by workers who are not declared as radiation-exposed personnel and therefore not checked for radiation absorption or incorporation. Therefore, the laboratories had to be out-zoned. It means that all free surfaces in the laboratory had to be cleaned, checked and declared as radioactive contamination free. Also, in order to insure a total safety, all material and tools used in the refurbishment is checked for possible radioactive contamination.

The PSI health physic team has clearly not the manpower capacity to realise such a work. A team of well skilled and experienced HL-staff was organised for the practical realisation of this crucial task.

The presentation will give an overview of the amount of work it involved, the organisation plan, the measurement procedures as well as the criteria's used for the realisation of the task in the tight schedule defined by the project manager. A summary of the experience gained during the HL refurbishment process will also be shared.

1. PIE - Techniques and Fuel Characterisation

1.1. EQUIPMENT AND TECHNIQUES FOR STUDIES ON IRRADIATED FUEL AND FUEL RODS CHARACTERISTICS.

Iou. Golovtchenko, V.M. Kosenkov, V.Yu. Shishin

ABSTRACT:

To justify and predict the reliability of the reactor core elements it is necessary to know properties of these elements as well as their components. Different equipment, devices and techniques are used at SSC RF RIAR for this purpose.

The following items are described in the report:

1. Specific engineering features and technical potentials of the new X-ray diffractometer. Experience in using such diffractometers to study macrostresses on the surface of the cylindrical claddings of the fuel rods and hexagon tubes, rim-layer amorphization in the fuel rods with oxide fuel and heterogeneous fuel.

2. Specific engineering features of neutron autoradiography and examples of its using in material science studies. ^{252}Cf neutron sources fabricated at SSC RF RIAR are used in this technique.

3. Specific engineering features and application examples of "fuel swelling stresses" measurements procedure.

1.2. IMPLEMENTATION OF A CABIN X-RAYS IN HOT CELL

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ABSTRACT:

The Fabrice process for the reconstituted short length irradiated rods in a hot cell was developed by the CEA, especially for ramp testing. This technique requires intricate operations in a hot cell with *specially adapted equipment and great skill people* :

An end plug is inserted under pressure and fitted to the opening end of a cladding tube. The meeting surfaces of the end plug and the opening end are welded by a TIG (tungsten inert gas) process.

Experimental work to optimize the welding parameters of TIG method was achieved.

Nevertheless, some predominate defects may occur in the end plug weld joints, such as lack of penetration and cavity. So, particular attention must be paid to non-destructive examination in particular X-ray control of welding areas.

A radioscopy technique has been applied to the control of TIG welds of the end plugs to rod assemblies in a high radioactive environment. This X-rays method enables immediate monitoring of any welding defaults on a TV screen.

A remote positioning system for the Fabrice rod was developed to position fuel rods below a X-ray source. Radioscopy pictures are recorded during remote positioning of the rod movement.

This document presents the modifications achieved by the constructor in cooperation with our laboratory staff, concerning the nuclearization of the apparatus as well as its implementation in the shielded hot cell n°2 of the CEA-DEC/SLS/LECA Laboratory at Cadarache.

1.3. INSTALLATION OF A SHIELDED SIMS IN CEA CADARACHE

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ABSTRACT:

A shielded IMS 6f has been installed in the LECA in the same room where EPMA and SEM were already settled. The nuclearisation was performed by the CAMECA company which sells the standard IMS 6f because the CEA requirements need in-depth modifications of the apparatus itself. Despite these modifications the shielded SIMS has the same level of performance as the standard apparatus. The design of the apparatus and of its modifications will be presented, the safety aspects will be emphasised.

Some preliminary results will be presented on non irradiated samples.

The shielded SIMS should be allowed to handle irradiated samples in October 2001. The location of the SIMS makes its possible to perform SEM, EPMA and SIMS at the same position on the same sample without any transfer by cask.

1.4. EVALUATION METHODS OF HYDROGEN PICK-UP IN IRRADIATED HIGH BURNUP FUEL ROD CLADDINGS*

E.Heikinheimo[1], W.Goll[2] and E.H.Toscano[3]

1 - Helsinki University of Technology, Laboratory of Metallurgy
2 - Framatome ANP GmbH
3 - European Commission, JRC-Karlsruhe, Institute for Transuranium Institute.

CONTENT

- General description of the hydrogen pick-up process
- High burnup phenomena
- Hydride distribution
- Quantitative hydrogen determination. Hot extraction method
- Metallographic evaluation
- SEM-quantitative determination
- Correlation with the corrosion phenomena

1.5. COMPLEX OF TECHNIQUES FOR INVESTIGATON OF THERMOPHYSICAL PROPERTIES OF IRRADIATED MATERIALS

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State Science Center of the Russian Federation **"Research Institute of Atomic Reactors"**

ABSTRACT:

The report covers the complex of the techniques for investigations of thermal properties of irradiated reactor materials, which are necessary for calculations of temperature, thermal stresses and phase stability of the reactor core elements used at SSC RIAR.

In particular, the report describes the principal diagrams, operation principles, accuracy of the techniques, and the examples of the results obtained.

They are as follows:

- The technique for measurement of thermal diffusivity of irradiated powders by radial and linear-in-time heating of a cylindrical sample;
- The technique for determination of thermal conductivity and capacity of compact materials in the form of disks by monotonic heating of a sample;
- The technique for determination of thermal conductivity and capacity of structural materials by the "flash" method;
- The technique for measurement of thermal linear expansion coefficient of irradiated materials with by the quartz dilatometer DKV-5AM-01;
- The technique for high temperature differential thermal analysis (VDTA-8), allowing for investigation of the stability of the irradiated materials within a wide temperature range.

1.6. INDUCTION HEATING ON DYNAMIC TENSILE TESTS

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ABSTRACT:

The LCMI (Laboratory for characterization of irradiated materials), located in Saclay, is in charge of the mechanical tests on irradiated materials. The dynamical tensile machine, in a hot cell equipped with two remote telemanipulators, has been first improved in 1995, to fulfil the French safety programs on Reactivity Initiated Accident (RIA). One objective of this machine is to obtain mechanical property data on current Zircaloy cladding types needed to quantify the cladding's response under RIA or LOCA transient loading and thermal conditions. For the RIA, this means testing at strain rates up to 5 s^{-1} and heating rates up to 200°C.s^{-1}, while for Loss Of Coolant Accidents (LOCA) testing at strain rates of 10-3 s^{-1} and heating rates of 20°C.s^{-1} would be appropriate.

The tensile samples are machined with a spark erosion machine, directly from pieces of cladding previously defueled. Two kinds of samples could be machined in the cladding:

- axial sample in order to test axial mechanical characteristics
- ring sample in order to test transverse mechanical characteristics, more representative of RIA conditions

On one hand, the axial tensile tests were performed using the Joule effect, and heating rates up to about 500°C.s^{-1} were obtained. This enabled us to perform the axial tests in a satisfactory manner.

On the other hand, the tensile ring tests were first performed in a vertical furnace with a heating rate about 0.5°C/s and a thermal stability about 1°C. For temperatures above 480°C, the mechanical characteristics showed a sharp drop which could be attributed to irradiation defect annealing. Therefore we have recently developed an induction heating system to reach heating rates high enough (200°C.s^{-1}) to prevent any significant annealing before performing the ring tensile tests. To apply a uniaxial tangential tension, two matching half-cylinder are inserted inside the ring and are pulled apart.

The main objective of this paper is to summarize the difficulties encountered in setting up a system that could be telemanipulated and would achieve heating rates up to 200°C.s^{-1} while taking into account the requirement for air-cooled coils in the hot cell. The same interface is used for induction heating equipment, in order to control the specimen temperature synchronously with the load/extension, collecting data from all transducers connected including temperature.

1.7. THE INSTALLATION OF AN IASCC AUTOCLAVE TEST SYSTEM AT THE SCK•CEN HOT LABORATORY

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ABSTRACT:

Irradiation assisted stress corrosion cracking (IASCC) is becoming a major concern for ageing power reactors world-wide. In order to assess the influence of irradiation damage on the stress corrosion cracking resistance of materials, it is imperative that one has the facility to subject irradiated material to a relevant environment in a well-defined and controlled stress state. In order to simulate the primary reactor environment in an LWR, an autoclave system with recirculation loop is used, able to operate up to 350°C and 25MPa. The loop is equipped with a parallel system of water purification and on-line chemistry monitoring system, so the sample is exposed to a constant and controlled environmental condition. For the stress corrosion testing, an active loading system is available, with a pull rod penetrating the autoclave cover. According to the test type, different types of samples (tensile, precracked fracture mechanics specimens such as compact tension or notched round bars) can be subject to various loading schemes (constant load, slow cycling load or slow strain rate tensile loading).

Concepts, which are known from testing non-radioactive materials, have been adapted to fit the practice of hot-cell testing of irradiated samples. The autoclave systems have been adapted to facilitate remote handling and reduce the risk of failure and the consequences thereof. The system has been designed in such way that the activity is confined to well defined parts of the system, so that hands-on manipulation of the loop part of the system is possible, while the autoclaves are placed inside a hot-cell. The cell itself has been refurbished for accepting the autoclave system inside and the loop part in the basement room below. In addition, special manipulation devices and storage capacity for the samples have been provided in the cell.