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STUDIECENTRUM VOOR KERNENERGIE
CENTRE D'ÉTUDE DE L'ÉNERGIE NUCLÉAIRE

ANNUAL REPORT FOR THE STEERING COMMITTEE OF THE ASSOCIATION EURATOM-BELGIAN STATE FOR FUSION 1999

Compiled by Marc Decréton
October 1999
BLG-833
Ref. 1160/99-02

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1. Introduction

This report summarises the work performed in the area of fusion reactor technology by the following three partners:

- the Belgian Nuclear Research Centre SCK•CEN in Mol,
- the S.A. Gradel company in Luxembourg,
- the Department of Metallurgy and Material Engineering of the KUL University in Leuven.

It covers the twelve-month period October 1998 to September 1999, and is prepared for the annual steering committee meeting of the Association Euratom - Belgian State for Fusion.

The work was articulated along two lines: the assessment of first wall, vessel assembly and blanket *materials under radiation and coolant interaction,* and the developments for remote handling in maintenance activities, mainly related to *radiation hardness of components and systems.*

SCK•CEN concentrated its activities on assessing several standard materials as first wall and blanket assembly materials: *beryllium* (integrity tests on annealed irradiated samples, reactivity with air and steam, modelisation of the fracture behaviour); *stainless steel* (fatigue tests on irradiated samples, preparation of double wall tube irradiations); *inconel* (irradiation creep).
Assessment work was also devoted to alternative materials, such as *chromium* (fragilisation under radiation), *reduced activation ferritic-martensitic* steels, with possible *oxide dispersion strengthening* (exploratory work in characterisation of basic materials), *molybdenum* (tensile and fracture toughness after radiation) and *titanium* (analysis of potentialities).
The impact of material choice on *dismantling strategies* and waste *storage reliability* has also been started.

At KUL, after a delayed completion of the experimental facility for testing *erosion-corrosion phenomena in Cu alloys* with incident heat flux, a series of test campaigns is now starting.

SCK•CEN continued its *assessment of components under gamma and neutron flux*, representative of the in-vessel conditions during maintenance and during storage periods. Components needed to drive, sense and connect the remote handling machines have been further identified for all basic functions, especially related to multiplexing capabilities for cables reduction. More attention was put recently on system effects, with analysis of systems used on the Brasimone mock-up for divertor replacement. In this context, the assessment of *optical fibre* communication, sensing and fibroscopy highlighted the wide potentialities of this option.
A new task related to *manipulator control* without cameras has been initiated with the objective to apply this approach on the divertor test bench.

GRADEL worked in closed collaboration with the ENEA Brasimone mock-up site for the development of *divertor movers components.* Reliable locking systems for the plasma facing components attachment blocks have been studied, manufactured and tested.

2. EFDA Related tasks

2.1. Physics Integration: Heating and Current Drive

2.1.1. Neutron Benchmark - Radiation Induced Electrical Degradation (RIED) Experiment (Task T246: Ceramics for Heating and Current Drive and Diagnostics Systems)

PRINCIPAL INVESTIGATOR: M. Decréton

SCIENTIFIC STAFF: S. Coenen, Y. Pouleur

OBJECTIVES

Ceramic insulators will be used in the fusion reactor vessel wall as part of the heating, current drives and diagnostics systems. These insulators will be subject to neutron fluxes and high temperatures. It is known that the electrical characteristics of most insulators present in these circumstances undergo a two step degradation process, called RIED (Radiation Induced Electrical Degradation). After a slow decrease in insulating resistance, a sharp breakdown is observed when a given dpa damage is reached. Such a sudden degradation could have serious consequences on the reactor operation. At present state, only fragmentary results are available, in terms of neutron fluence, energy and flux, as well as temperature and sample material. The objective of the task is to perform a neutron benchmark experiment to obtain more reliable values of the RIED threshold.

PROGRAMME

This task, started in 1996, investigates the RIED effect of sapphire under thermal neutron irradiation, at high temperatures and under vacuum. On-line monitoring of the electrical conductivity of these samples are performed. Different samples are irradiated under different application-relevant conditions during one irradiation experiment, allowing to obtain results for all samples and all applications relevant conditions simultaneously. The task involves:

- the construction of an irradiation rig,
- the adaptation of a measurement system,
- the irradiation test under thermal neutron,
- analysis and reporting of the obtained results.

ACHIEVEMENTS

The design of the experimental rig was initially foreseen for the BR1 reactor. It was changed in favour of a BR2 reactor experiment, allowing a better gamma dose rate to be obtained. The experimental parameters can be summarised as follows:

- Neutron flux: ~ 10^{+12} n/cm².s (E>0.1MeV).
- Gamma heating: 100 Gy/s.
- Temperatures: 350 °C (2 samples), 400°C (1 sample), 450°C (2 samples).
- Vacuum (N_2): 10^{-5} bar.
- Electrical field: resistance measurement under 500 Volt DC.

Technical design of the rig

Considering those multiple requirements, a new rig has been specially designed. It is mainly composed of three modules, centred in a pressure tube to be introduced in the reactor and connected to a control panel for vacuum circuit, heating regulation, measurements and data acquisition. The modules which are thermally independent to achieve the three different required temperatures, are small ovens (figure 1).

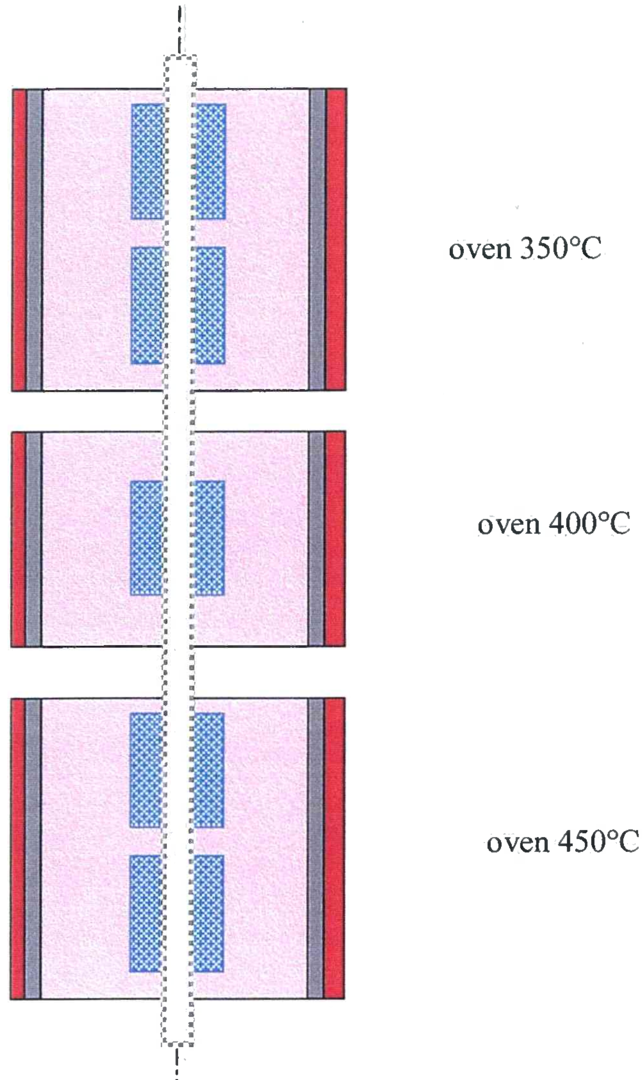


Figure 1: Internal module. Vertical section

Heating wires are mounted on metal support around the samples (figure 2). As the environment is vacuum, thermal transfer is mainly driven by radiation. Temperatures are regulated through the power delivered by the heaters, while basic heating is provided by gamma heating.

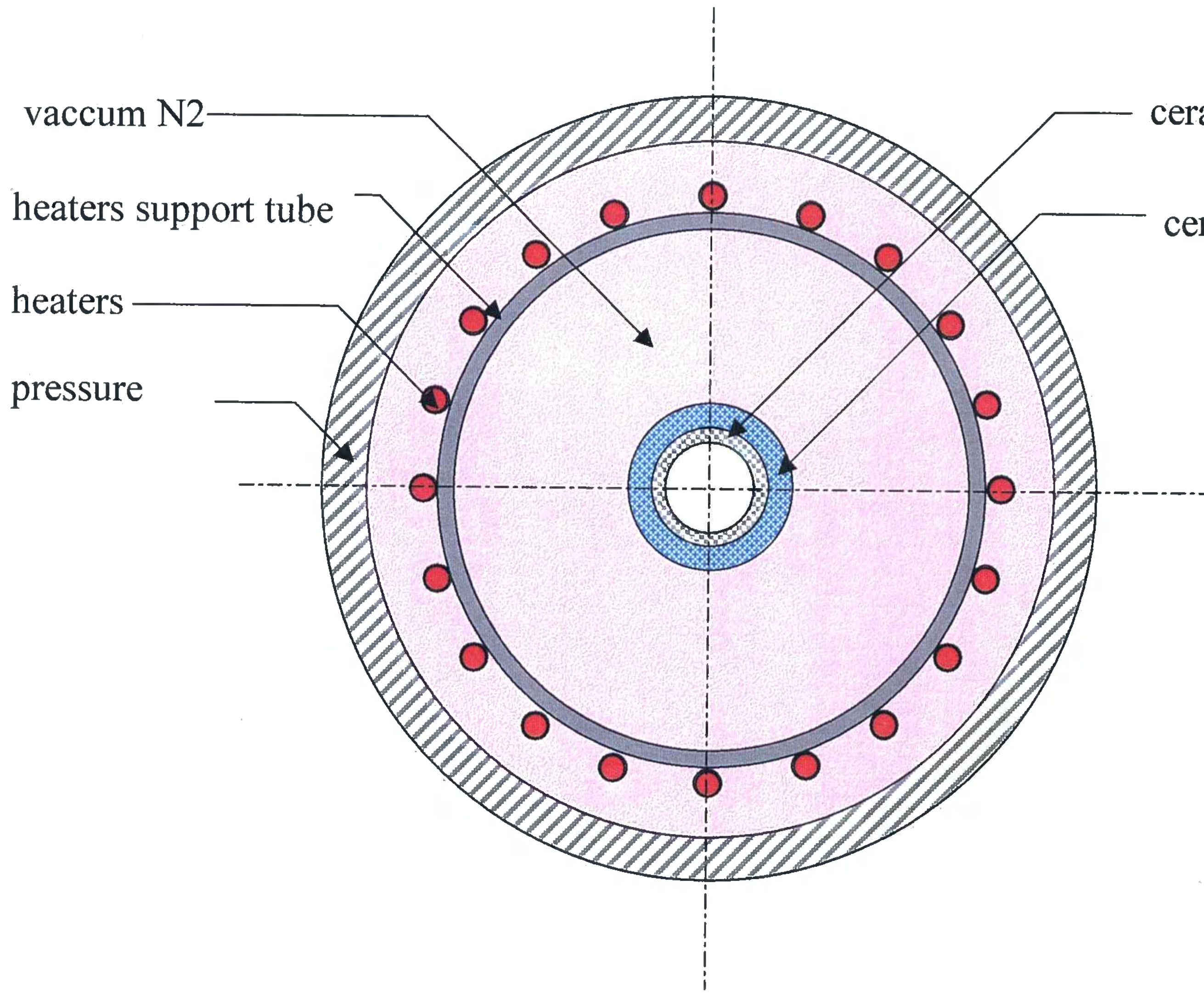


Figure 2: Horizontal section at mid-plane

Electrical resistance measurement is here a delicate task because of the very high resistivity values of the ceramic material and the high voltage involved. It should lay between 10^{+11} and 10^{+7} Ω.m during irradiation. Therefore, the circuit for the electrical measurements, which has to work in an industrial environment, has been designed with a lot of care. All the wires for instance present higher insulation resistances than the ceramics, and small currents (0.00002 – 0.2 mA) are measured by means of an accurate ampere-meter and a voltmeter placed on a reference resistance. Adequate filters are foreseen. Currents are measured on the low voltage side of the samples to minimise the effects of the losses. Moreover, good vacuum is important to avoid gas ions currents. Only materials showing little degassing were thus selected.

Different neutronic calculations have been performed in order to determine the optimum irradiation conditions. Since all irradiation channels of the BR2 reactor at full power show levels of flux and dose rate exceeding the experimental conditions, the experiment must be performed in a special cycle of the reactor, at low power (~ 5 %). The chosen channel is in the reflector and offers a neutron flux of 2 10^{+12} n/cm².s (E>0.1MeV). Activation dosimeters are placed in the rig near the samples and will be analysed afterwards to determine the neutron flux with high accuracy.

Several actions are foreseen to ensure safety and to protect the samples. If a leak or an undesired increase of temperature is detected, the rig will be isolated from the out-of-pile part of the experiment, the heating and the electrical tension source being cut off. The rig will also be withdrawn then, out of the flux region, to prevent any irradiation of the samples under undesired conditions. The same withdrawal of the rig will be used to measure, at regular intervals, the electrical resistances of the samples without neutron flux and with little gamma. This will help to better characterise the RIED effect.

The rig is designed to be reloaded with new modules and samples to offer new opportunities for future irradiations programmes.

FUTURE WORK

The experimental device is completely detailed. Its functionality and safety were examined and approved in September. Construction is now on-going.

The first functional tests should start begin 2000 and irradiation and data analysis should follow in March 2000.

PARTNERS

This task is performed in collaboration with CIEMAT (Madrid, Spain) where the first research work on RIED has been initiated a decade ago.

2.2. Vessel/In-Vessel: Plasma Facing Components

2.2.1. Integrity test and swelling assessment of Irradiated Beryllium after Annealing (Task T216: Shield Blanket Fabrication and Testing)

PRINCIPAL INVESTIGATOR: Leo Sannen

OBJECTIVES

Beryllium, a low atomic number element with excellent mechanical and neutron related properties (high scattering, low capture and high multiplication), is considered suitable for plasma facing and breeding blanket components in future fusion reactors. Potential disadvantages are its low maximum operating temperature and its swelling behaviour, the latter one arising from the helium build up from the nuclear reactions. Russian researchers observed that low temperature irradiated beryllium samples completely disintegrate after heating at threshold values of ~ 1500 appm He and 700 °C. This rose particular concern on these potential disadvantages. Therefore, investigations to check the integrity of low temperature irradiated beryllium exposed to high heat fluxes were performed at SCK•CEN and the assessment of the corresponding swelling was made.

PROGRAMME

The programme involved the following activities.

- Appropriate samples (i.e. appropriate for the remote hot-cell pycnometry density measurements) are cut from broken CT specimens (S65 VHP and S200 VHP type beryllium) and from a cylindrical BR2-archive specimen (BR2 type). The S-type beryllium has been irradiated at 230 °C up to 725 ppm He, and the BR2-type beryllium at 50 °C up to 3800 appm He.
- The samples are examined visually before annealing.
- Annealing is performed at 600 and 750 °C for 5, 27 and 100 hours.
- After annealing the integrity of the samples is checked by visual examination.
- The swelling is evaluated by density measurements.

ACHIEVEMENTS

All samples were annealed in a remotely operated quartz tube vacuum furnace. Figure 3, 4 show a representative sample of the annealing curves out of the totality of annealings, encompassing two annealing temperatures (600 °C and 750 °C) and three annealing periods (5-27-100 h).

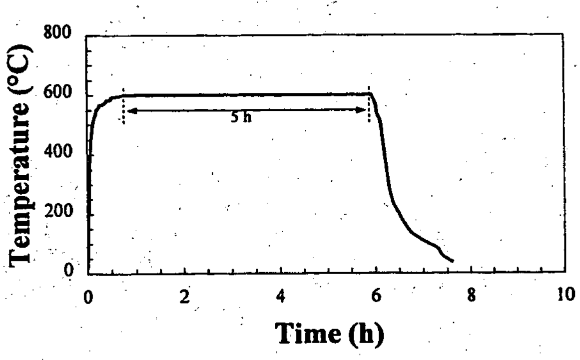


Figure 3: Temperature record of the Annealing at 600 °C for 5 h

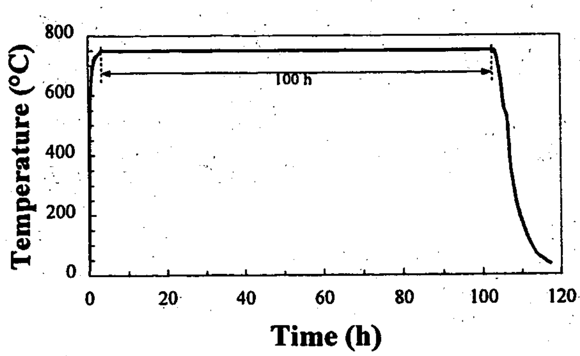


Figure 4: Temperature record of the Annealing at 750 °C for 100 h

All samples still showed the same solid geometry, i.e. no sign of disintegration was observed on any sample (e.g. figure 5).

Before annealing ***after annealing***

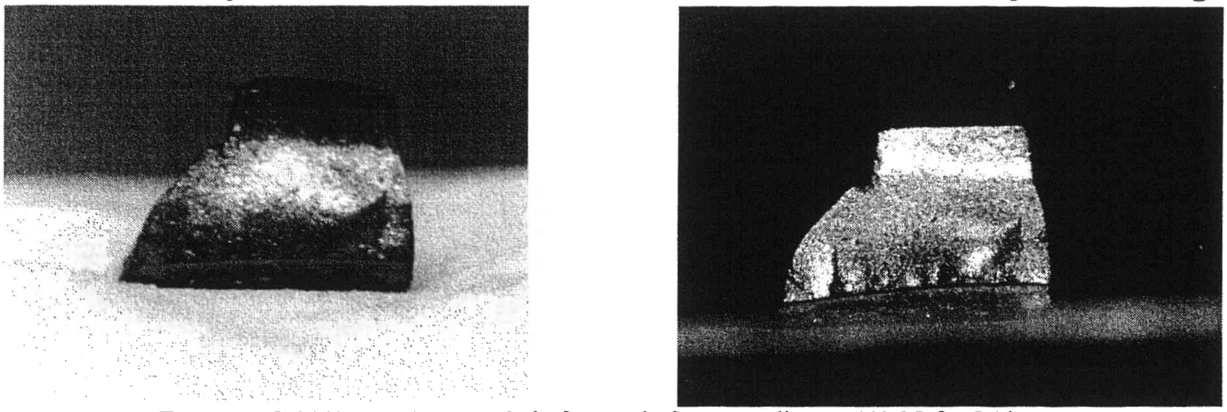


Figure 5: S-200E type Be sample before and after annealing at 750 °C for 27 h

Swelling of the post-irradiation annealed Beryllium was assessed by density measurements using a remotely operated mercury pycnometer. The results are compiled in Table 1.

Be type		S-65			S-200F			S-200E		
Fabrication Data										
BeO content (wt. %)		0.6			1.2			1.9		
Theoretical density (g/cm^3)		1.852			1.856			1.861		
Reported density (g/cm^3) (%TD)		1.850 (99.9)			1.854 (99.9)			1.852 (99.5)		
Measured density (g/cm^3) (%TD)		1.851 (99.9)			1.838 (99.0)			1.852 (99.5)		
Irradiation Data										
Temperature. (°C)		230			230			50		
n-Fluence (10^{25} n/m^2 ; E_n > 1 MeV)		1.5			1.5			9.7		
He-content (appm)		600			600			3900		
Annealing		**Volumetric Swelling ***								
Temperature	Time	vol. %	acc. (2 σ)		vol. %	Acc. (2 σ)		Vol. %	acc. (2 σ)	
°C	H		abs.	Rel.		Abs.	Rel.		abs.	rel.
as-irradiated / non-annealed		+ 0.1	± 0.9	1655%	+ 0.0	± 0.6		+ 0.7	± 0.3	44%
600	5	+ 0.0	± 0.4		- 0.8	± 0.4	51%	+ 0.5	± 0.2	41%
	27	+ 0.4	± 0.4	112%	- 0.8	± 0.4	52%	+ 0.5	± 0.3	57%
	100	+ 0.2	± 0.5	228%	- 0.9	± 0.4	48%	+ 0.9	± 0.4	38%
750	5	+ 1.9	± 0.4	22%	+ 0.5	± 0.4	90%	+ 1.7	± 0.2	13%
	27	+ 2.1	± 0.4	20%	+ 0.4	± 0.4	95%	+ 2.4	± 0.2	10%
	100	+ 2.0	± 0.4	21%	+ 0.5	± 0.4	90%	+ 2.3	± 0.3	12%

* with the measured initial as fabricated density as reference

Table 1: Swelling of post-irradiation annealed beryllium

From these swelling data it is clear that

- significant swelling enhancement is observed at 700 °C with respect to 600 °C;
- the higher BeO content grades swell less;
- the annealing times used have no or only neglegible effect on the swelling.

REFERENCES

[1] L. Sannen, *"Integrity and Swelling of Post-Irradiation Annealed Beryllium."*, R-3343, May 1999.

2.2.2. *Analysis of tensile and fracture toughness results on irradiated Molybdenum Alloys, TZM and Mo-5%Re (Task PDS 1.4)*

PRINCIPAL INVESTIGATOR: M. Scibetta

SCIENTIFIC STAFF: R. Chaouadi, J.-L. Puzzolante

OBJECTIVES

The three categories of plasma facing components (PFCs) under consideration in the fusion scientific community are non-metallic materials (graphite, carbon fibre composite), low Z metallic element (beryllium), and high Z metallic element (tungsten, molybdenum).

The PFCs will be submitted to very high heat and radiation fluxes and disruptive loadings. The choice of the PFCs will be dictated by various criteria such as tritium permeation, plasma-PFCs interactions, erosion resistance, fatigue induced by thermal loading, swelling and embrittlement, creep properties, thermal ageing, neutron activation, biological hazards, fabrication capabilities and cost.

The most interesting properties of refractory metals such as molybdenum are the high melting point, the high mechanical resistance at elevated temperatures, the low thermal expansion coefficient and the good thermal conductivity, which results in an excellent dimensional stability and a good resistance to thermal shock. The tensile properties for candidate divertor armour materials Mo-alloys are now well established. However, the assessment of the effect of radiation damage on the tensile, fracture toughness and fatigue properties still requires a large effort. In this context, the tensile and toughness properties of two Mo-alloys (TZM and Mo-5%Re) were investigated at SCK•CEN.

Background

Tensile and Disk-Shaped Compact Tension specimens DC(T) from TZM and Mo-5%Re materials were irradiated previously in the BR2 reactor. Two irradiation experiments were performed:

- MOST1: with a fluence of 3.50 10^{20} n/cm^2 (E>1MeV) and a nominal irradiation temperature of 40°C,
- MOST2: with a fluence of 2.88 10^{20} n/cm^2 (E>1MeV) and a nominal irradiation temperature of 450°C.

ACHIEVEMENTS

Twenty-eight static tensile tests were performed in the baseline and irradiated conditions for temperatures varying from 25 to 450 °C. TZM and Mo5%Re materials show good tensile strength and ductility in the baseline condition in the range 25 °C to 450 °C. However, there is a drastic loss of ductility due to irradiation for both materials. The lower irradiation temperature (MOST1) induces a higher reduction of ductility.

Thirty eight instrumented fracture toughness tests were performed on DC(T) in the baseline and irradiated conditions for temperature varying from 25 to 450 °C. Tests were performed on pre-cracked and notched specimens. In the base line condition, our results were compared to KFA data on the same Mo-5%Re material.

In base line condition, the fracture toughness increases with temperature and the fracture mode is cleavage. As shown in figure 6, a good agreement with KFA results was obtained. These results allow rationalising results obtained on notched specimen (see figure 6). Although a drastic diminution of ductility is observed on tensile test results, the fracture toughness decreases only slightly due to neutron irradiation (see figure 7).

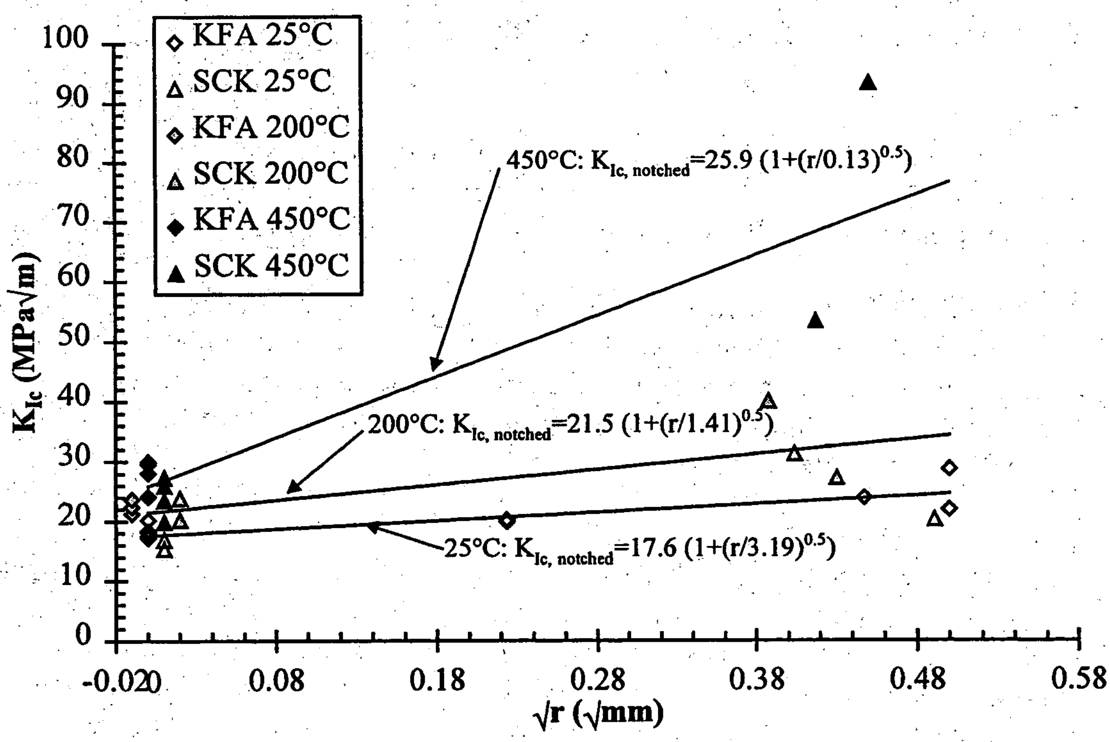


Figure 6: Fracture toughness versus the square root of the notch tip radius. Mo-5%Re material in the base line condition

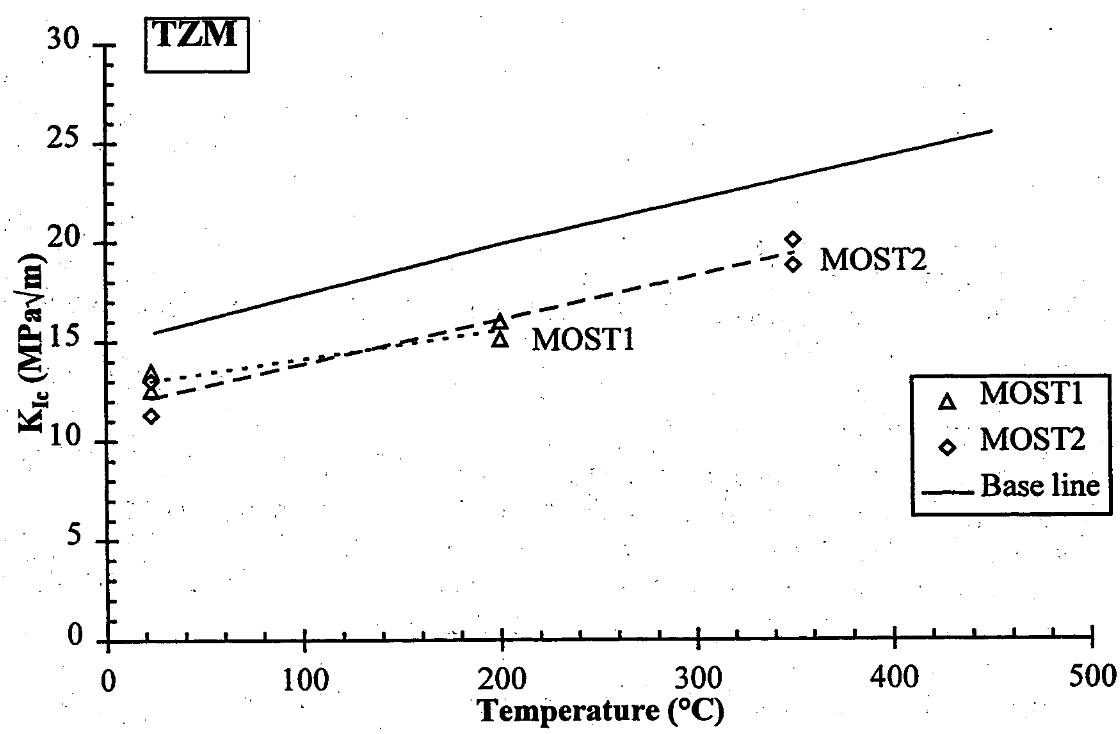


Figure 7: Fracture toughness versus test temperature. Effect of the irradiation on the fracture toughness of the TZM material

Scientific Partners

KFA Jülich GmbH, (Rödig M., Derz H. and Pott G).

References

[1] M. Scibetta, R. Chaouadi and J.-L. Puzzolante, "Analysis of tensile and fracture toughness results on irradiated Molybdenum Alloys, TZM and Mo-5%Re", report BLG-823, SCK•CEN, Mol (Belgium), October 1999.

[2] M. Scibetta, R. Chaouadi and J.-L. Puzzolante, "Analysis of tensile and fracture toughness results on irradiated Molybdenum Alloys, TZM and Mo-5%Re", Ninth International Conference on Fusion Reactor Materials ICFRM-9, Colorado Springs, October 10-15, 1999 (to be published in the Journal of Nuclear Materials).

2.2.3. Erosion-Corrosion Testing of Copper Alloys with Incident Heat Flux (Tasks T217/T222: Aqueous Corrosion of SS and Cu Alloys; Test of Divertor Elements)

PRINCIPAL INVESTIGATOR: W. Bogaerts, KULeuven

SCIENTIFIC STAFF: Jian Hua Zheng

OBJECTIVES

Study of the erosion-corrosion behaviour of Cu-alloys in high velocity water under heat transfer conditions, with the aim of investigating the effect of critical water chemistry parameters.

PROGRAMME

Materials:

- Dispersion Strengthened (DS) copper,
- CuCrZr copper alloy.

Test conditions:

- incident heat flux: about 10 MW/M2,
- mock-up inlet temperature: 140°C,
- pressure: 25 bar,
- water velocity: ca. 12 m/sec,
- water chemistry: controlled H_2 and/or H_2O_2 additions.

Duration:

- max. 6 runs from 100 to max. 1000 hrs.

ACHIEVEMENTS

All of the required experimental facilities are currently in place. A number of technical and/or administrative problems (e.g. increased safety requirements) that delayed progress of the project have been solved. The experimental loop with incident heat flux has been finalized.

On the other hand, the test matrix was redefined during the meeting with NET home team held in October this year. The first test is scheduled to start before the end of 1999.

2.3. Vessel/In-Vessel: Vessel/Mechanical Structure

2.3.1. Irradiation Creep (Stress Relaxation) Testing of Inconel 718 (Task V62)

PRINCIPAL INVESTIGATOR: R. Chaouadi

SCIENTIFIC STAFF: R. Van Nieuwenhove, M. Wéber

OBJECTIVES

The ITER shield modules are attached to the backplate by four radial supports, and connected to it by a pair of electrical straps, all of which are fixed by Inconel 718 bolts. They are exposed to moderate neutron fluence (less than 1 dpa) at elevated temperature (300 to 350 °C). To deal with operational loads, these bolts are pre-stressed during assembly. However, during operation, the bolts lose part of the pre-load due to stress relaxation, but mostly as a result of irradiation creep. Almost no data are found in the literature on the dimensional stability of Inconel 718 under neutron irradiation.

Therefore, the objective of this work is to provide a set of data for loading conditions typical of the actual operation conditions of the bolts in ITER. More specifically, the irradiation creep behaviour of Inconel 718 at 300°C up to 0.5 dpa is investigated. The specimens are pre-stressed to 80 % of the yield strength, with a variation of ± 10 %. The pressurised thin-walled tube geometry is selected, as it is the most adequate geometry for such measurements.

In the previous reporting period, a literature review and the design of the creep test rig were completed.

ACHIEVEMENTS

The material was characterised (microstructure and tensile properties), tubes were fabricated and pre-pressurised and finally loaded into the reactor. Presently, the tube specimens have received a dose of 0.24 dpa (one reactor cycle) and exposure to a second reactor cycle is foreseen before the end of this year, thereby reaching the final dose level in January 2000.

Material characterisation

The tensile properties were determined at two temperatures: ambient and 300°C (see table 2). Heat treatment resulted in a substantial increase of the yield strength (more than twice the initial yield strength). Figure 8 shows the engineering strain – stress curves of the different samples.

Table 2: Tensile test results

Id.	Condition	T (°C)	Yield stress (MPa)	Ultimate stress (MPa)	uniform elongation (%)	total elongation (%)	reduction of area (%)
FINC-1	sol. ann.	20	553	974	36	44	57
FINC-2	sol. ann.	300	464	904	39	44	53
FINC-3	prec. hard.	20	1146	1428	18	24	42
FINC-4	prec. hard.	300	1055	1296	20	24	41

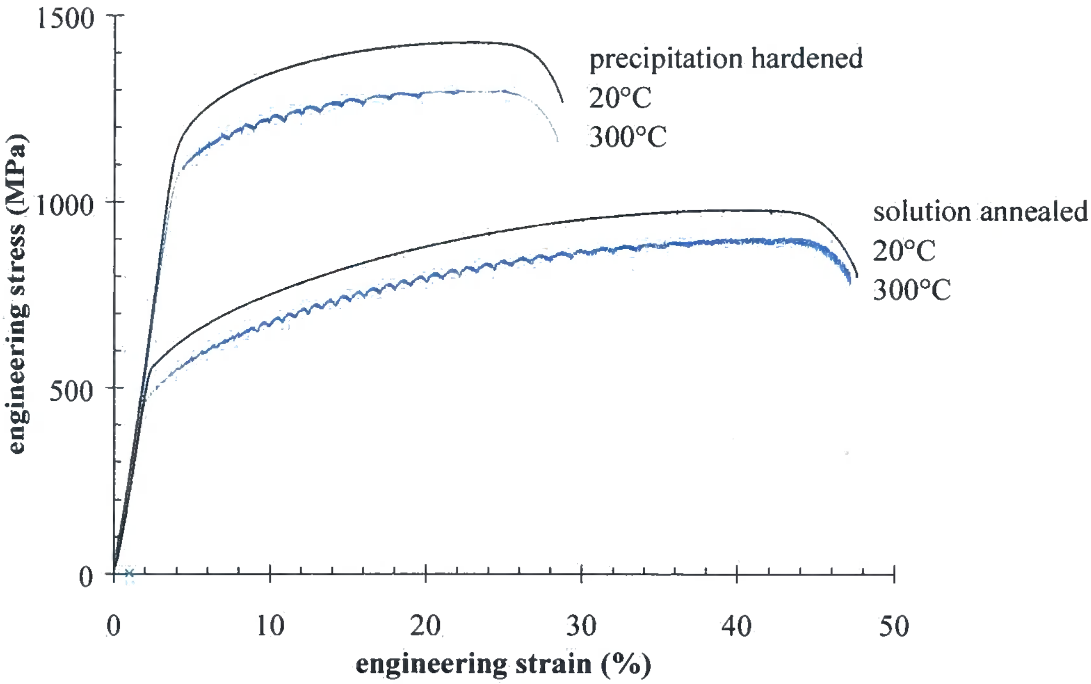


Figure 8: Tensile flow curves of Inconel 718 in the precipitation hardened condition

The heads of broken tensile samples were polished and etched to reveal the microstructure of the Inconel 718 in both solutions annealed (as received) and precipitation hardened conditions. Micrographs at 100× and 1000× magnification for each condition indicated no effect of heat treatment on the grain size. Vickers hardness measurements were performed on two samples in the solution annealed and precipitation hardened condition, and also revealed a significant increase (nearly a factor 2) for the precipitation hardened material. Transmission Electron Microscopy (TEM) specimens are in preparation for examination. Creep tests on pressure tubes of unirradiated material at 300 °C are also in preparation and will serve as a reference for the irradiated samples.

Tube fabrication and pre-pressurisation

The required high stress of 0.9 times the yield stress corresponds to an inner tube pressure of about 1000 bar. Welding the end-plugs to the thin-walled (0.3 mm) pressure tubes, with sufficient resistance to the applied stresses, proved to be very difficult. The laser welding technique was found the most reliable but a large number of iterations was necessary in order to optimise the parameters.

The tubes (see figure 9) were pressurised by introducing a relatively large volume of argon gas at low pressure (around 20 bar) into the small volume of the pressure tube. This is accomplished by lowering the pressure tube in liquid nitrogen, so as to liquefy the argon inside the tube. The resulting pressure has been experimentally verified and crosschecked by the weight increase of the filled tube. The tube diameters have been measured by an accurate laser measuring system before and after the filling. The same system will be used to measure the diameter increase after the irradiation.

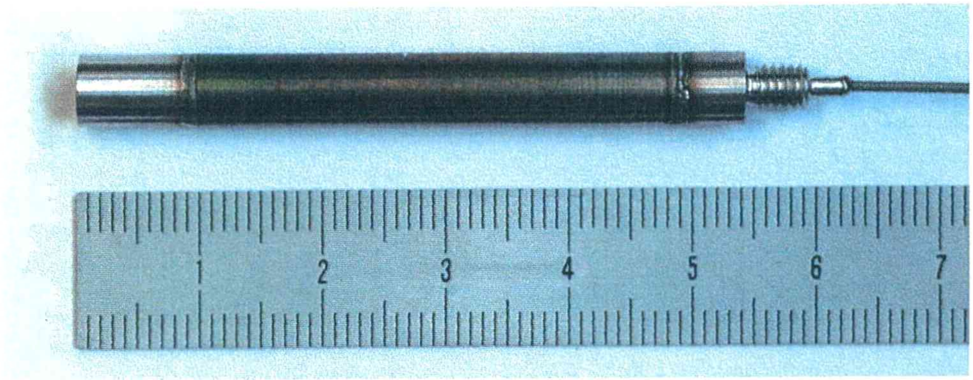


Figure 9: Picture of a pressure tube made of Inconel 718

Irradiation in BR2

The irradiation takes place in the Pressurised Water Reactor loop Callisto at a temperature of 305 °C. Due to the gamma heating of the tubes, however, the wall temperature can increase to 320 °C [2]. Possible rupture of the pressure tubes is monitored continuously by radiation monitors (detection of ^{41}Ar). The testing matrix is given in table 3.

Table 3: Testing matrix for irradiation

Specimens	Pre-stress	neutron fluence 0 %	60 %	100 %
Pressurised tubes	0.7 σ_y	2	2	3
	0.8 σ_y	2	2	3
	0.9 σ_y	2	2	3
Tensile	--	--	3	3

Future work

After the final irradiation cycle in the BR2 reactor (January 2000) the samples will be unloaded and the dimensional measurements of the tubes will be performed in order to determine the creep (relaxation) rate of the Inconel 718. The final report will be available in the middle of 2000.

REFERENCES

[1] R. Chaouadi, R. Van Nieuwenhove, M. Scibetta, B. Smolders and M. Wéber, Irradiation Creep Testing of Inconel 718, Preparatory Phase, Progress Report, May, 1999, SCK•CEN, R-3346.

[2] P. Benoit, M. Wéber, Boucle CALLISTO, Rapport du cycle 03/99 A, SCK•CEN, MI.57/D088010/43/PB/MW.rw.

2.3.2. Low cycle fatigue of AISI 316L stainless steel plate and weld (Task PSM 3-3)

PRINCIPAL INVESTIGATOR: J.-L. Puzzolante

SCIENTIFIC STAFF: R. Chaouadi, Ch. De Raedt, M. Scibetta
W. Vandermeulen (VITO)

OBJECTIVES

The austenitic stainless steel type AISI 316L was selected as the main structural material of the next-step ITER fusion device, in particular for the first wall, blanket modules and vacuum vessel components. Although this steel was extensively investigated under different aspects, most results concern irradiation temperatures above 300°C. The purpose of this task is to determine the low cycle fatigue properties of the reference 316L steel in both solution annealed and TIG metal deposit conditions before and after neutron irradiation up to 5 dpa at 42°C. The tests are performed at room temperature.

Background

The plate specimens (named F2) and two orientation of TIG metal deposit weld specimens (named MD1 and MD2) were neutron irradiated in the BR2 materials testing reactor up to 5.4 dpa, corresponding to a neutron fluence of 7.96 10^{21} n/cm² (E>0.1 MeV) [3] in an irradiation rig named 'LOTION 1' (LOw Temperature IrradiatiON) and cooled by the BR2 primary water (42°C) [4].

ACHIEVEMENTS

The fatigue tests on non-irradiated material F2 and MD1 have been performed at the Flemish Research Institute (VITO) [5] using a closed loop servo hydraulic INSTRON test frame type 1340. The same type of machine was used to test non-irradiated material MD2 and all irradiated materials in hot-cell at SCK•CEN. All tests were performed according to the ASTM E-606 standard at room temperature.

The analysis of the test results shows that neutron irradiation induces hardening in both plate and weld materials without significant reduction of the fatigue life. The results are shown in Figures 10 and 11 for the plate and the weld materials, respectively. Our test results on both materials were compared to the fatigue data of the ITER database. As shown in figure 10, a very good agreement is found in spite of different test temperatures (75, 250, 400 and 450°C for the weld; 450, 527 and 550°C for the plate material) and different neutron doses (0.3, 15.6 and 31 for the weld; 0.3, 3.2, 3.3, 15.3, 23.4, 31 and 35 dpa for the plate material).

For strain ranges above 5% (ε>0.5%), the irradiated materials (plate and welds) exhibit continuous softening. The two TIG-metal deposit welds are nearly equivalent in terms of fatigue life and stress in both non-irradiated and irradiated conditions which means that no orientation effect is observed.

For both materials (plate and weld), fatigue life is not affected by irradiation although a significant hardening is induced by irradiation.

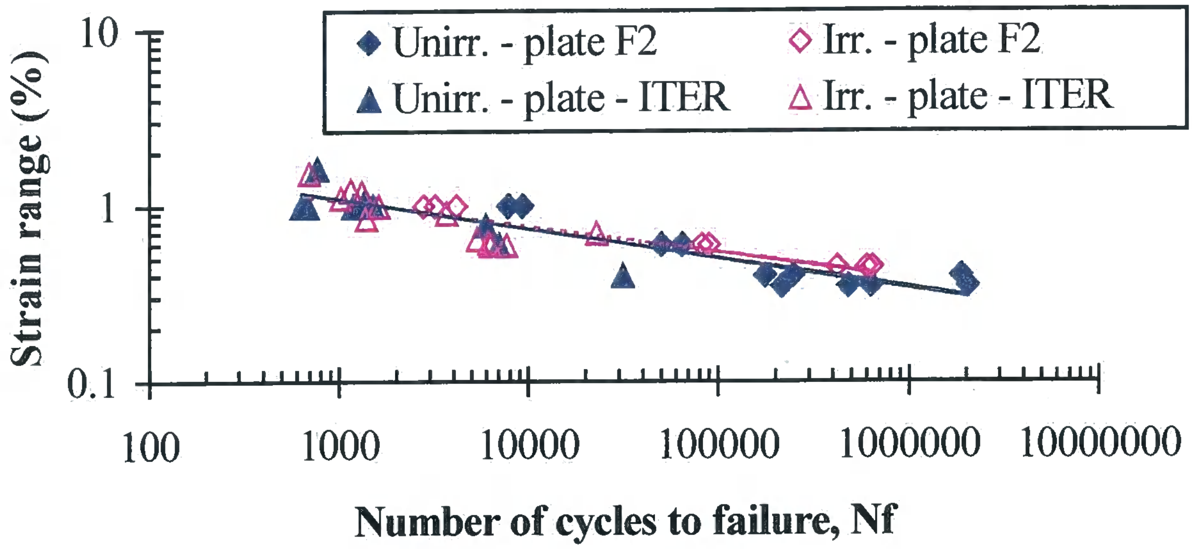


Figure 10: (Δε - Nf) curve of the plate material compared with the ITER data.

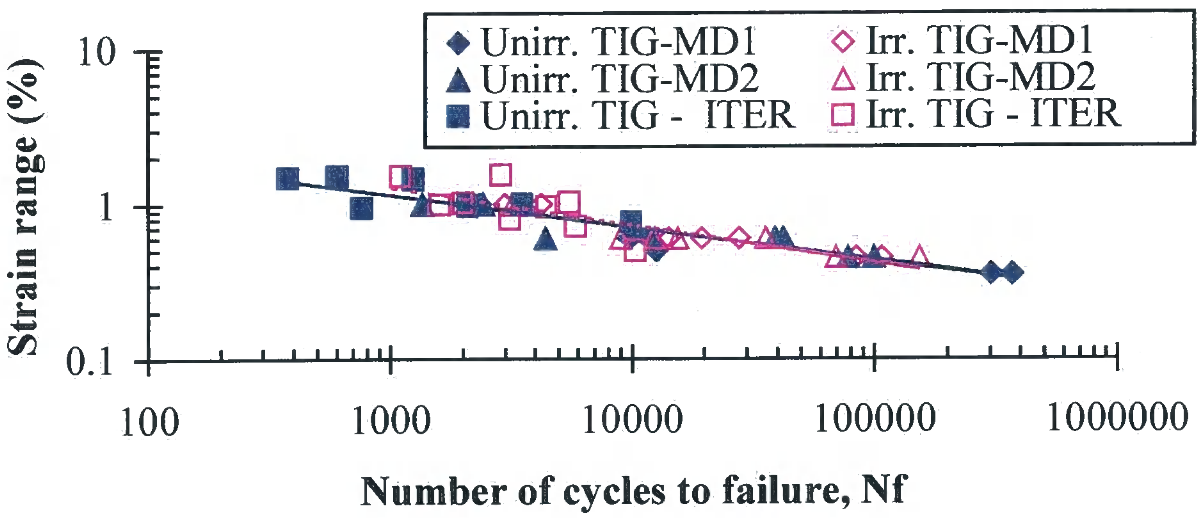


Figure 11: (Δε - Nf) curve of the TIG-metal deposit weld material compared with the ITER data.

SCIENTIFIC PARTNERS

Vlaamse Instelling voor Technologische Onderzoek (VITO) – Mol, Belgium

REFERENCES

[1] J.-L. Puzzolante, M. Scibetta, R. Chaouadi, W. Vandermeulen, "Tensile and low cycle fatigue properties of solution annealed and TIG weld AISI 316L stainless steel irradiated at low temperature (42°C)", BLG-821, September 1999.

[2] J.-L. Puzzolante, M. Scibetta, R. Chaouadi, W. Vandermeulen, "Tensile and low cycle fatigue properties of solution annealed and TIG weld AISI 316L stainless steel irradiated at low temperature (42°C)", Ninth International Conference on Fusion Reactor Materials ICFRM-9, Colorado Springs, October 10-15, 1999 (to be published in the Journal of Nuclear Materials).

[3] C. De Raedt, P. Van Mechelen, Corrected thermal and epithermal fluence values and helium formation in the LOTION 1 and LOTION 2 experiments in BR2 (tasks PSM 3.3 and PSM 1.3), SCK•CEN, Fuel research, ChDR/rw/411/94-02/FT/MOL/94-04, February 1994.

[4] F. Moons, LOTION 1 – Irradiation at 315K of AISI 316L and TIG welds, low cycle fatigue specimens, progress report PSM 3.3, SCK•CEN, FT/MOL/93-01, January 1993.

[5] W. Hendrix, 'Fusieprogramma: LOTION', technical note, VITO, MOL, MAT/N/WH/me/93-050, October 1993.

2.4. Vessel/In-Vessel: Assembly and Maintenance

2.4.1. Remote handling for In-Vessel maintenance - Radiation Tolerance Assessment of Remote Handling Components (Task T252)

PRINCIPAL INVESTIGATOR: S. Coenen

SCIENTIFIC STAFF: M. Decréton, C. Van Ierschot

OBJECTIVES

The ITER Fusion reactor concept involves important maintenance work to be performed in and around the vacuum vessel during reactor shutdown. This maintenance includes:

- inspection of the first wall,
- inspection of the plasma heating and purification components,
- repair of all in-vessel structures,
- replacement of large components, such as divertor plates, blanket segments, ...

Moreover, the remotely operated machines will have to cope with large deployment distances and heavy payloads, and direct vision will often not be possible. This will make manual teleoperation tricky and a certain level of computer aid to the operator will be essential to provide a safe and reliable operation. Since the in-vessel environmental conditions will be quite hostile, with respect to both temperature and radiation, the choice of components to be used on such manipulators becomes dramatically limited. Gamma dose rates of up to 30 kGy/h with total dose levels for a complete mission up to 100 MGy are expected. Diagnostics tasks just after shut-down are expected to be performed at temperature as high as 200 °C. All sensitive components such as motors, sensors, cabling, viewing systems, etc. have to be chosen and/or designed to withstand these extreme conditions. Besides the problems of remote maintenance, instrumentation is also placed near and around the vessel to monitor the reactor during operation. This instrumentation also has to cope with severe environmental conditions, involving gamma and neutron radiation, both high and low (cryogenic) temperatures, vacuum atmospheres and electromagnetic interference. The *main objectives* of this task are:

- to provide the reactor engineers *guidelines on the components*,
- to provide *reliability and expected lifetime data* under these severe conditions of various selected components,
- to provide *adapted specifications* for industrial procurement and prototyping.

PROGRAMME

The *objectives* of these tasks are *achieved* through:

- general *studies* around the *degradation phenomena* of specific components under high radiation and high temperatures,
- the *design of improved prototypes* in collaboration with industry and research institutes,
- *testing under representative conditions*, both gamma and neutrons,
- *setting up* a general *database* for radiation tolerance of remote handling components.

ACHIEVEMENTS

Co-ordination work
The SCK•CEN has co-ordinated the execution of task T252 among the European partners. This work involved insuring proper communication between the partners and between the task and the NET/ITER management. An annual progress report of the T252 Task was issued in June 1999 [1]. A visit to the real size ITER mock-up at ENEA-Brasimone has been organised to get a better understanding of the actual needs of the remote handling tasks.

Motors
Two new motors, manufactured by MACCON (Germany) have been delivered for irradiation testing up to 100 MGy. During the pre-irradiation characterisation, problems occurred with the brakes and the bearings. During the tests at a temperature of 100 °C, the brake blocked. After cooling down to room temperature, the brakes could only be unlocked manually. The front bearings had become noisier during the test at 100 °C as well. Both the motors were shipped back to MACCON for further investigation and to fix the problems. The problem with the brakes was caused by a misalignment due to thermal expansion of the parts, and has been taken care of. The problem with the front bearings is still under investigation.

Sensors: Several irradiation campaigns have been performed to collect data on the radiation tolerance of specific sensors. Tests have been conducted at dose rates up to 33 kGy/h, up to total dose levels of more than 100 MGy. The tested components are ultrasonic sensors, accelerometers and strain gauges. Detailed results of the tests have been reported in [2], [3], and [4].

Communication systems between the sensors and the control system must also be radiation tolerant. This involves the *cables* to be used as well as the *multiplexing* capabilities to ease the umbilical management.

Cables: Collaboration has been initiated with a cable manufacturer (KABELWERKE Eupen, Belgium) to identify suitable cable types for the remote handling operations. The collaboration is planned in two phases. The first phase (1999) consists in identifying suitable candidate materials for cable insulating material. The second phase (2000) consists in the production of a prototype hybrid cable. This hybrid cable should contain instrumentation wires, power lines and optical fibre lines. Irradiation tests on various materials have been started. The irradiation is carried out on different dose rates: 0.1, 1, and 10 kGy/h. Although the lower dose rates are not of real interest for the fusion conditions, those dose rate levels have been chosen as well to get a better understanding of the degradation processes of the materials. All the irradiation tests have been finished. The samples are shipped back to KABELWERKE Eupen for post irradiation testing. A report of all the results will be issued by the end of 1999.

Multiplexing systems have always been considered as a critical element for the use of robotics in remote handling equipment in a nuclear environment. Since a high degree of reliability and a certain degree of intelligence is required for such machines, a large number of sensors and hence a large number of cables is needed. To avoid umbilical management, multiplexing the signals is the most obvious solution.

Various possibilities are open to solve the multiplexing problem: the use of normal commercially available components, space graded components, new radiation hardened technologies, etc. For high total dose applications (1 MGy or more) however, none of the above techniques provides satisfying results. A solution for such high total dose levels can be found using discrete components (such as bipolar transistors, relays, etc.) which can be used up to high total dose levels. Each individual component degrades, but still remains functional with less performance. Incorporating these intrinsically rad-hard components in a circuit design, which takes into account the degradation of each component, can lead to rad-hard solutions up to 10 MGy.
The basic elements of a multiplexing circuit for nuclear environment are:

- switching circuits: to collect analog and digital signals into one single line,
- analog to digital converter: digitising analog signals avoids problems due to noise and reduces losses due to long transmission lines.

To build a digital to analog converter, various possibilities are open. Two elements however are often encountered in the basic block diagrams of A/D converters: a comparator and a counter (based on flip-flop circuits in cascade). A comparator circuit and a JK flip-flop have been designed and built. The design is done in such a way that even with heavily degraded transistors (e.g. h_{fe} degraded from 100 to 15) the circuit still remains functional. The degradation of the transistors has been modelled in SPICE, based on irradiation tests performed in the past. The circuits have been simulated with SPICE and irradiation campaigns up to a total dose of 22 MGy to validate the simulated results have been done. Complete results can be found in [5].

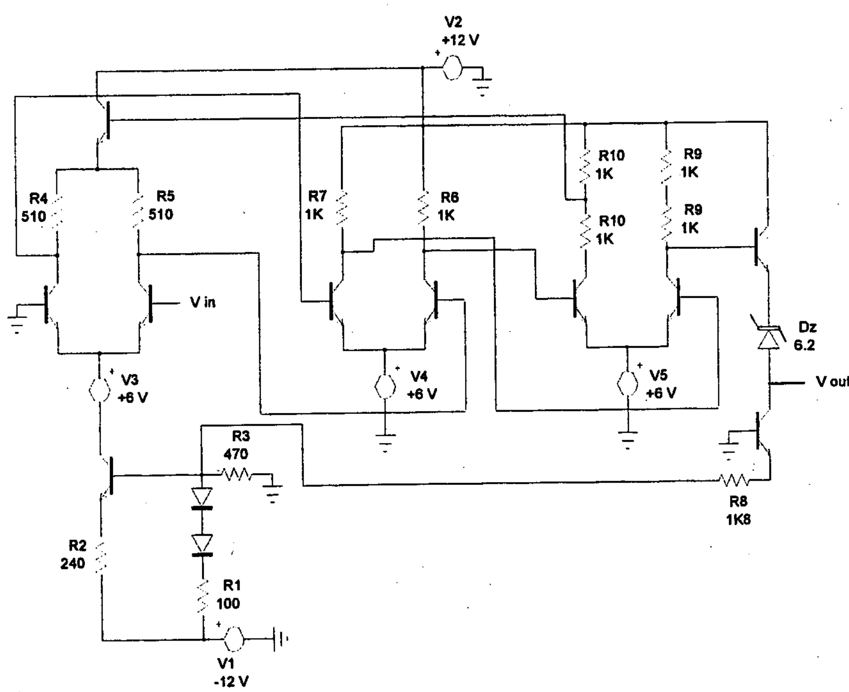
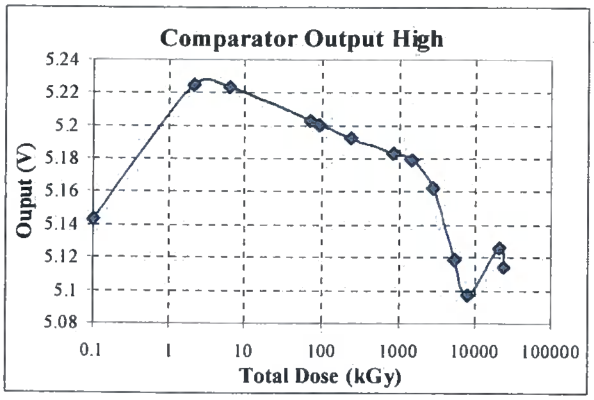


Figure 12: Comparator circuit

The final *comparator* circuit is shown in figure 12. The basic building blocks of a comparator are an ideal current source and a differential amplifier. To compensate for the degradation of the forward current gain h_{fe} of the bipolar transistors, we added an additional amplifier stage and a positive feedback into the circuit.

The circuit is in fact designed such that even with h_{fe} degraded from 200 down to 15, the circuit should still be functional. Spice simulations show that the output values are -0.7 V for the low value and +5.1 V for the high value.
Figure 13 shows the results from the irradiation campaigns. The figure clearly shows that the level of the output voltage High and Low does not change significantly and that these levels remain well within the standard values for most logic circuits. The pre-irradiation value for the low state output of the comparator is slightly different on the actual circuit compared to the SPICE simulations (-0.3 V instead of -0.7V). This has to be accorded to the differences between the SPICE parameters of each component and the actual components mounted on the PCB-board.



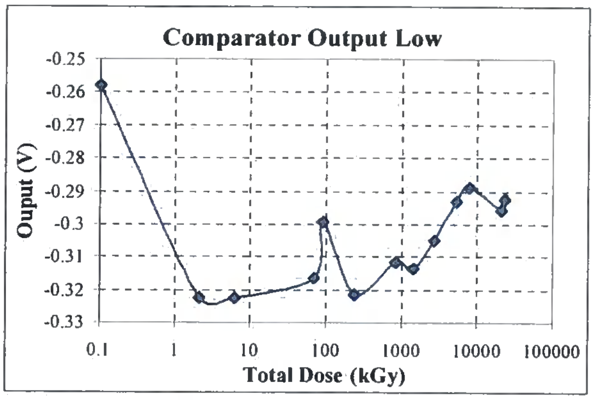


Figure 13: Comparator output High and Low Vs total dose

The basic layout of the *flip-flop* we designed is that of a JK flip-flop. The block diagram is given in figure 14. All logic ports (AND, NAND, and NOT) are built around the 2N2369A transistors in a way similar to TTL technology.
The JK flip-flop is a basic building block. Placing eight of such flip-flops in cascade allows building an 8-bit counter for use in an 8-bit A/D converter.

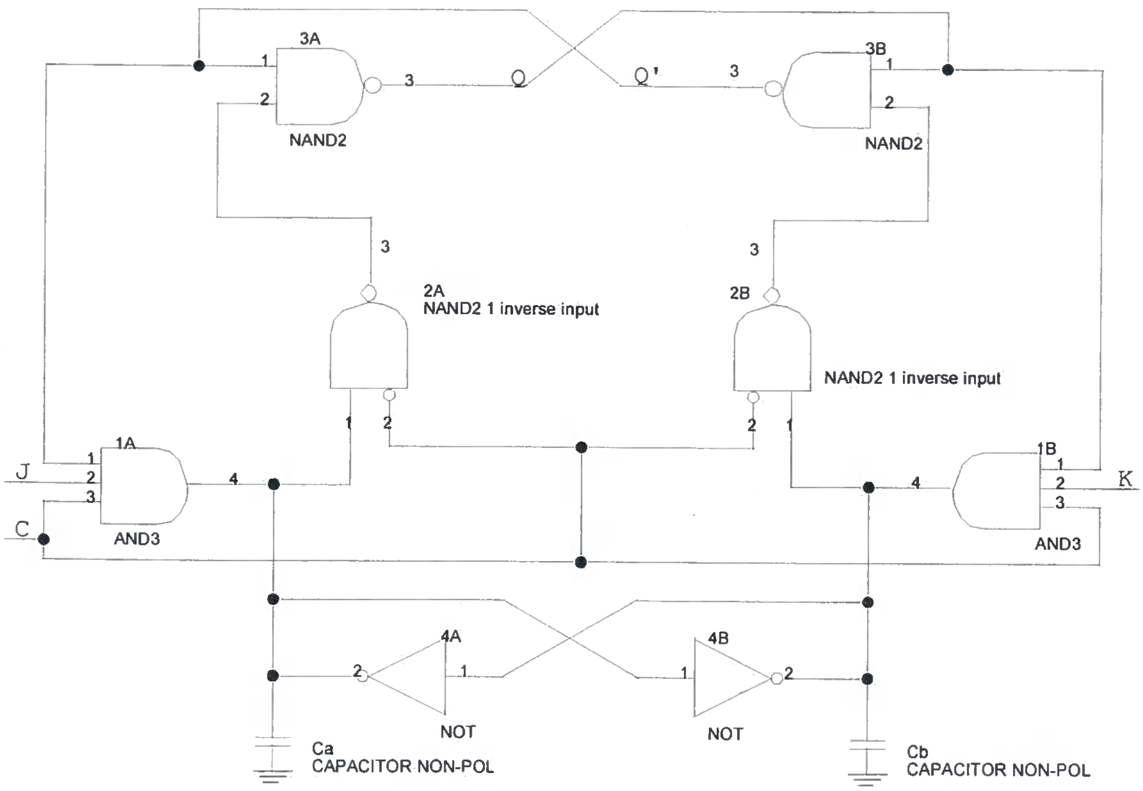


Figure 14: Block diagram of the flip-flop circuit

The values for Q high and low for the circuit with degraded transistors ($h_{fe} = 15$) still remain extremely well within the limits for most logic circuits: 125 mV for Q low, and 6.3 V for Q high. The output values for the flip-flop output as a function of total dose (see figure 15) clearly shows no degradation of any significance. Here also there are minor differences between the pre-irradiation values of the actual circuit and the values given by SPICE simulations. The low value (logic 0) slightly increases from 57 mV to 64 mV, but remains far below the limit value of standard logic circuits.

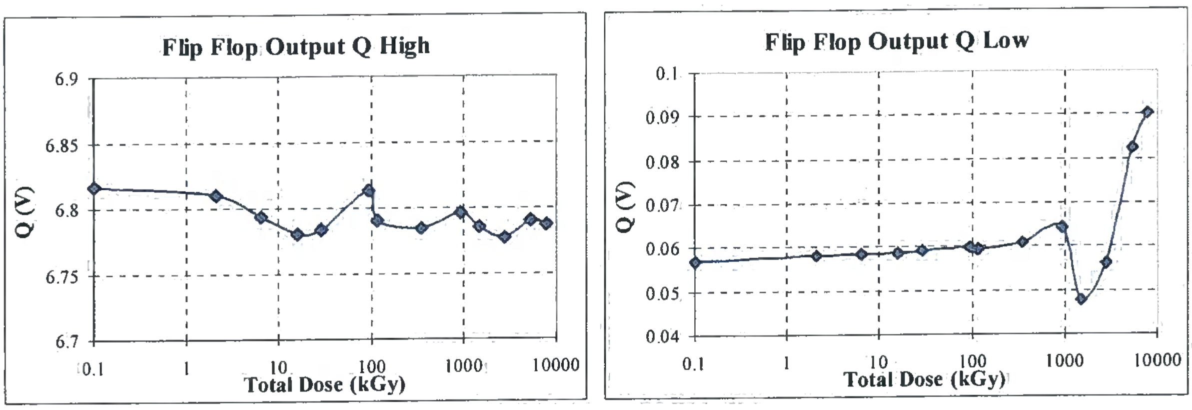


Figure 15: Flip-flop output high and low versus total dose

FUTURE WORK

As soon as the problems of the bearings for the motors are solved, irradiation will start up to a total dose of 100 MGy. The collaboration with CEA/LETI on irradiation testing of opto-electronic components will continue during the next year. Further work will be directed towards developing and testing of systems or subsystems designed for specific tasks.

SCIENTIFIC PARTNERS

Task T252 involves collaboration with CIEMAT (Madrid, Spain), CEA/LETI (Saclay, France), as well as KABELWERKE (Eupen, Belgium).

REFERENCES

[1] S. Coenen, ITER Task T252 Radiation Assessment of Standard Components (S 72 TT 13)Annual Progress Report April 1998 – March 1999, Ref 504/99-04.

[2] S. Coenen, *ITER T252 - Radiation Assessment of Standard Components - Final Report on Ultrasonic Sensors*, SCK•CEN Report R-3354, June 1999.

[3] S. Coenen, *ITER T252 - Radiation Assessment of Standard Components - Final Report on Strain Gauges*, SCK•CEN Report R-3355, June 1999.

[4] S. Coenen, *ITER T252 - Radiation Assessment of Standard Components - Final Report on Accelerometers*, SCK•CEN Report R-3356, June 1999.

[5] S. Coenen, *ITER T252 - Radiation Assessment of Standard Components - Final Report on Multiplexers*, SCK•CEN Report R-3357, June 1999.

2.4.2. Neutron Testing of Common Instrumentation - Assessment of Sensors and Communication Equipment under Fast Neutrons (Task T338)

PRINCIPAL INVESTIGATOR: S. Coenen

SCIENTIFIC STAFF: S. Coenen, M. Decréton, C. Van Ierschot

OBJECTIVES

During the reactor run of the ITER fusion reactor, parts of the remote handling equipment and inspection tools will be stored near the reactor vessel, or, for some particular cases, in the vessel wall itself. These components will require specific attention with respect to dose build-up, including neutrons. In particular, it concerns lens optics of cameras, distance, force and vibration sensors, lubricants and the communication systems. These components will see an environment, which will be different from the standard fission reactors: higher energy neutrons (close to the fast spectrum around the reactor due to the thermalisation of the 14 MeV fusion neutrons), extreme temperatures (high in the blanket, cryogenic in the magnets), strong electromagnetic interferences, vacuum, etc...

Existing experience from gamma irradiation tests and long-time experience from the nuclear industry therefore cannot be extrapolated as such. It has to be noted also that the complex geometry of the fusion reactor and the hazardous conditions around it renders any equipment replacement difficult and costly. Remote handling and inspection systems must therefore have a high degree of reliability, especially with respect to radiation induced degradation, and be able to operate even after a long storage time.

The main goal of the T338 task is to assess, under fast neutron flux, common instrumentation used on the fusion reactor, in particular sensors and their related communication equipment. These components are basically related to remote handling equipment stored near the reactor during operation.

The **main objectives** of this task are:

- to provide the reactor engineers **guidelines on the components**, and **the procurement specifications,**
- to provide **reliability and expected lifetime data** under the conditions around the reactor vessel.

PROGRAMME

The **objectives** of these tasks are **achieved** through:

- **testing under representative conditions**, of components previously qualified under gamma radiation,
- the **design of improved prototypes** in collaboration with industry and research institutes.

ACHIEVEMENTS

The main effort was put on the assessment under neutron radiation of various components, which were qualified before under gamma radiation. Although the materials of each component were selected to show almost no activation, some of the components were too activated to be measured directly after irradiation. A long waiting time was needed for some of the components for post-irradiation examination.

The accelerometers could only be measured after 6 months. The fibre optic connectors still show a too high radiation background to be measured.

actions in case of emerging malfunctions of the machines such as stalling or jerking of the motors. A series of accelerometers, manufactured by Bruel & Kjaer, have previously been assessed under gamma radiation up to 100 MGy. The same type of sensors have been irradiated under fast neutron flux as well up to a total fluence of 1.15 $*10^{17}$ n/cm^2. The results show a decrease of about 20 % of the power spectrum of the accelerometers, as shown in figure 16.

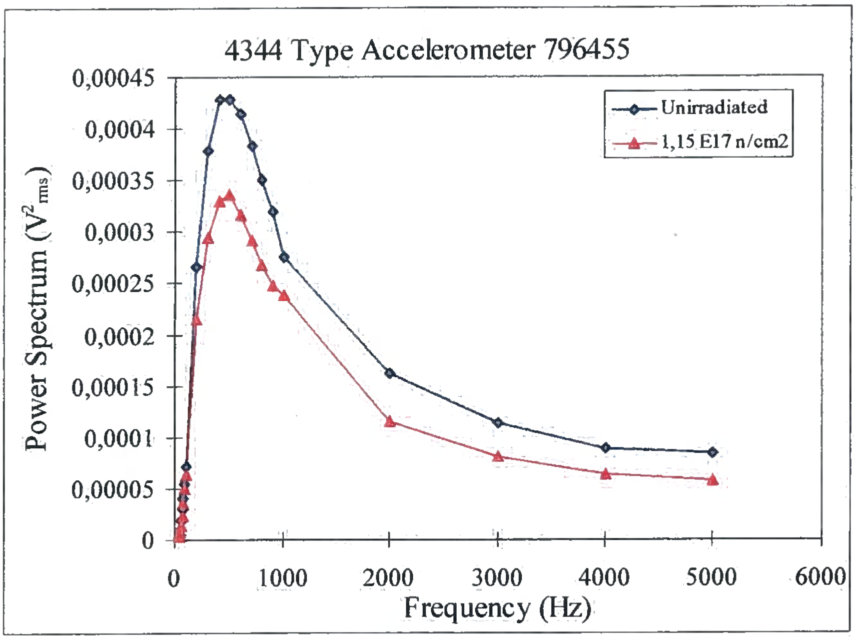


Figure 16: Power spectrum of the 4344 type accelerometer 796455

Future Work

For the follow up in further programmes, other components to be investigated have to be defined in collaboration with industrial partners and in synergy with the actual design of the fusion reactor and the actual maintenance and repair strategies. Negotiations have started with industry for instance for specific bearing types and lubricants.

Not only individual components should be tested, but also complete systems or subsystems. During the next phase, emphasis therefore should be put not only on individual components, but also on complete systems. Failure of one of the critical components of a system could lead to a breakdown of other components, even when these elements have been qualified as excellent solutions.

2.4.3. Divertor maintenance, preliminary qualification test (Task 308/4)

PRINCIPAL INVESTIGATOR: E. Biver, Gradel S.A. Luxembourg

SCIENTIFIC STAFF: J-C. Haux

OBJECTIVES

- Manufacture items for preliminary qualification tests related to divertor maintenance operations:
  - Test the locking system between a divertor cassette and the toroïdal rail (Deliverable 1 - 3).
  - Test the behaviour of the rack and pinion under rescue scenario conditions and the behaviour of the wheels and rails during different mover operations (Deliverable 4 & 5).
  - Test the handling possibilities for removal installation of the divertor cassette keys during the refurbishment of the cassette (Deliverable 6).
  - Technological tests (Heat exchange, load transfer) to validate the key concept (Deliverable 7a).
  - Manufacture of new **P**lasma **F**acing **C**omponent (PFC) mounting arrangement to test multi-link attachment concept in the divertor mock-up facility (Deliverable 7b).

Background
Divertor remote handling has been identified by ITER as one of seven key engineering issues. To be demonstrated are:

- the replacement of a divertor in the ITER vessel by remote handling techniques within an acceptable time and to required tolerances;
- the remote refurbishment of divertor cassettes in a hot cell.

The project includes the fabrication of remote handling tools and test facilities, their use to perform all of the individual operations such as locking, cutting, welding, testing and repositioning, and the demonstration of a complete divertor exchange within 6 months.

PROGRAMME

Continuation of Task T308/4

- Purchase raw material and standard components according to ENEA Brasimone specifications and to Gradel's quality assurance organisation.
- Manufacture (machining, assembling, functional test) of locking system:
  - deliverable 1 to 6 achieved;
  - deliverable 7a: Keys set-up for technological tests;
  - deliverable 7b Multi-link PFC attachment blocks.

ACHIEVEMENTS

- deliverable 7a: sent to ENEA Brasimone on 14/05/98;
- deliverable 7b: to be delivered beginning October 1999.

Future activities
During the next phase of this work, the objective is to further develop the facilities at Brasimone to establish a hot cell environment which is closely representative of that required for an ITER class machine. This will involve the design and manufacture of new cassette mock-ups and test pieces.

2.4.4. *BLINE: The blind man's approach to the local geometric modelling of a nuclear environment* *(Task T329/5: In-Vessel Remote-Handling Dextrous Operations)*

PRINCIPAL INVESTIGATOR: J. De Geeter

SCIENTIFIC STAFF: M. Decréton, C. Van Ierschot

OBJECTIVES

The removal/installation of vacuum-vessel components, such as the divertors, has to be done remotely. Due to the activation induced by neutron irradiation, dose rates of up to 10^4 Gy/h are expected during this task. Under these conditions, cameras cannot be used for on-line close-range imaging. The most radiation-resistant sensors, such as touch probes or ultrasonic sensors, return however only very sparse, local data, unlike cameras. These data are difficult to interpret by a human operator. The situation is comparable to the blind, who explores the environment with the white cane. They use a model of the environment, stored in memory, which they occasionally verify/update with a few well-chosen measurements of the cane.

Therefore, the challenge of this task is to complement the main divertor replacement equipment that does the heavy work in well-defined circumstances, with a dextrous robot arm mounted on an in-vessel mover that does the tasks that require more dexterity in less predictable circumstances. What if something goes wrong? What if something is dropped, or a mover gets jammed? Then the dexterity of the robot arm is essential to go and explore the environment, very much like the blind as discussed above. The objective of the BLINE module in this task is to verify/update a geometric environment model locally in the area of interest (to find a lost object, or relocate a jammed mover), using the measurements taken by the dextrous robot arm. Then, the operator of the remote-handling equipment can rely on this model to do the actual intervention.

PROGRAMME

Task 329/5 has three partners: CEA-FAR (Fontenay-aux-roses), SCK•CEN and TEKES.

CEA-FAR (deliverables 1-5) develops, delivers, and demonstrates the Remote Intervention System (RIS). This system consists of a hydraulic MAESTRO dextrous robot arm, its controller and a Graphical Supervisory system (RIS-GS). The robot arm is to be installed on a mover in the Divertor Test Platform (DTP) at ENEA, Brasimone. This RIS will be used to demonstrate the following tasks that require the dexterity of the MAESTRO arm: (i) locking/unlocking of the cassette locking system; (ii) cassette cooling pipe installation/removal in the ducts; and (iii) a rescue operation on a partially failing Cassette Toriodal Mover (CTM). For these tasks, the MAESTRO arm will be equipped with a ball probe tool to measure points in the environment, similar to a co-ordinate measurement machine.

TEKES (deliverables 8-9) designs, manufactures and tests water hydraulic joints to upgrade the MAESTRO arm.

SCK•CEN (deliverables 6-7, subject of this report) develops, delivers and tests the BLINE software module. BLINE has to use the measurements of the ball probe on the MAESTRO arm to update/verify a geometric model of (part of) the DTP. The first part of the work is to specify the interfaces with RIS-GS, and to define the required functionality of the BLINE module.

The second part is the actual implementation of the BLINE algorithms. The third part consists of the testing of the BLINE module together with RIS-GS at CEA-FAR. The fourth part is the installation and testing of BLINE together with the complete RIS at ENEA in Brasimone.

ACHIEVEMENTS

Specification of the interfaces and required functionality of BLINE

Interfaces

The interface between RIS-GS Supervisor and BLINE has been specified as described in the meeting report by C. Leroux (CEA minutes of the meeting "Interface between RIS_GS supervisor and BLINE software", ref DPSA/STR/LTO/99.331, 10 September 1999). Agreement has been made on the communication medium (ethernet sockets using PONY software provided by CEA) and the format and content of the messages to be exchanged. For simplicity, it has been decided to transmit all measurements (probe positions) in a single file to reduce the number of interactions of RIS-GS and BLINE.

CEA has to calibrate the Maestro robot, so as to compensate for all non-random positioning errors, and to provide a covariance matrix of the remaining random errors on the probe position.

Functionality

The task will concentrate on a rescue operation, whereby both position of the Cassette Toroïdal Mover (CTM) and the divertor cassette, relative to one another and relative to the Divertor Test Platform (DTP), have to be determined before the problem can be solved. Therefore, BLINE should be able to estimate the position of the CTM relative to the cassette, and relative to the DTP, using the measurements of a spherical probe mounted on the MAESTRO arm. The first problem amounts to the estimation of the position of the front plane of the cassette. The second problem is much more complicated as the geometry of the DTP is rather complex. In addition, the latter problem is badly conditioned since only a small part of the DTP can be reached by the MAESTRO arm. A set of DTP features (cylinders and planes) have been identified that are suitable for estimation, and can most likely be reached by the MAESTRO arm. The definitive selection of features has not yet been made.

As discussed sub A., BLINE has to process all measured probe positions at once, and return a transformation describing actual position of the cassette or the CTM.

Development of BLINE algorithms.

The BLINE algorithms basically consist of:

- functions that define the geometrical relationship between the spherical probe position and the geometric feature (measurement equations);
- functions that define the geometrical relation between the different features of the DTP (constraint relations);
- an optimisation algorithm that uses these relationships and all the probe measurements to determine best estimate of the position of features of interest.

As of September 1999, two measurement relations have been derived and tested for measurements with a spherical probe on planes and on cylinders. Due to the bad condition of the estimation problem, special attention has been paid to the parameterisation of the features. These parameterisations have been selected based on their numerical stability for estimation.

As an optimisation algorithm, the Kalman filter has been selected. The correctness of the Kalman filter equations and the measurement equations have been tested using simulated measurements in Monte-Carlo simulations.

Future work

Interface and functionality specification: The definitive interface specification should be finalised and implemented by the end of October '99. The functionality specification should be finished as soon as possible (selection of the features of the DTP that will be measured by the probe, which is to be discussed with ENEA and CEA).

Development of BLINE algorithms: a measurement equation for a cylindrical patch in addition to the one for a complete cylinder. The measurement equation for the complete cylinder performs well in Monte Carlo simulations for measurements that are randomly chosen on the complete cylinder. However, this equation might not perform properly in practice when the measurements are coming from a small patch. Therefore, this problem will be studied more in depth in the next few months. Also the constraint equations will be developed during the next few months.

We will also look for alternative algorithms instead of the Kalman filter. Since it has been agreed that the measurements will be available all at once, there is no added value in having a recursive estimator. Although a recursive and a batch least-squares estimator should theoretically yield the same result, it is expected that a batch estimator will perform better in practice.

The integration with RIS-GS and tests will take place around the end of 1999.

The installation and tests at ENEA are scheduled around mid 2000.

REFERENCES

[1] J. De Geeter , H. Van Brussel, J. De Schutter, M. Decréton, *In-vessel maintenance of a fusion reactor: the blind man's approach* , Proc. of the 20th Int. Symposium on Fusion Technology, pp. 1075-1078, Marseilles, France, Sept. 1998.

[2] J. De Geeter, and M. Decréton and E. Colon, *The challenges of telerobotics in a nuclear environment* , Robotics and Autonomous Systems 28, pp.5-17, Elsevier, 1999.

[3] J. De Geeter, J. De Schutter, H. Van Brussel, M. Decréton, *Task-directed sensing with robust sensors for the local geometric modelling of a nuclear installation* , Proc. BNES International Conference on Remote Techniques for Hazardous Environments, London, April 1999, British Nuclear Energy Society.

[4] J. De Geeter, J. De Schutter, H. Bruyninckx, H. Van Brussel, M. Decréton, *Geometric modelling of a nuclear environment* , Proceedings of Uncertainty 99: 7th Int. Workshop on Artificial Intelligence and Statistics , Fort Lauderdale FL, Jan. 1999, pp. 41-50, D. Heckerman and J. Whittaker eds., Morgan Kaufman.

2.5. Tritium Breeding and Materials: Breeding Blanket

2.5.1. Design of Tritium Permeation Barrier – In-pile tests of small submodules (Task A.4.3.3: EBP, WCLL Blanket Concept)

PRINCIPAL INVESTIGATOR: Ph. Benoit

SCIENTIFIC STAFF: Ph. Benoit, Ch. De Raedt, P. Jacquet

OBJECTIVES

In the WCLL blanket concept (for Water Cooled Lithium-Lead), tritium is generated in a 17Li-Pb liquid metal bed, cooled by pressurised water circulating in Double-Wall Tubes (DWT). The permeation of tritium to water is limited by Tritium Permeation Barriers (TPB), which are a coating applied at the outer surface of the DWT. The DWT and the TPB constitute critical elements of the WCLL concept. Therefore, two experiments are proposed to test the TPB performance and the DWT behaviour in conditions as realistic as possible. The experiment aims at confirming the general behaviour of the components and in particular the tritium permeation rate in an environment as close as possible to the future operation conditions.

PROGRAMME

For both experiments, the project involves the study of an irradiation rig, for which are to be issued:

- a general description,
- principle drawings,
- preliminary thermohydraulic, strength, tritium and neutronic calculations,
- planning and estimation of the budget and manpower requested to build and irradiate the module, to examine it and to dispose it after irradiation.

ACHIEVEMENTS

WCLL project

The first experiment is aimed towards the TPB. Two DWT segments are plunged in a lithium-lead pool. The inside of the DWT is flushed by helium and collects the tritium permeating through the wall. One of the DWT has received its TPB coating, the other not, so that comparative measurements are possible. The irradiation position chosen for this experiment in BR2 allows a prototypical tritium production rate, so that a realistic tritium partial pressure will exist in the lithium-lead.

Special precautions are taken to measure the partial pressure of tritium in the lithium-lead, which is the driving force of the permeation process. In particular, a detailed balance of the tritium losses through the different walls has to be performed, which implies a dedicated tritium measurement system.

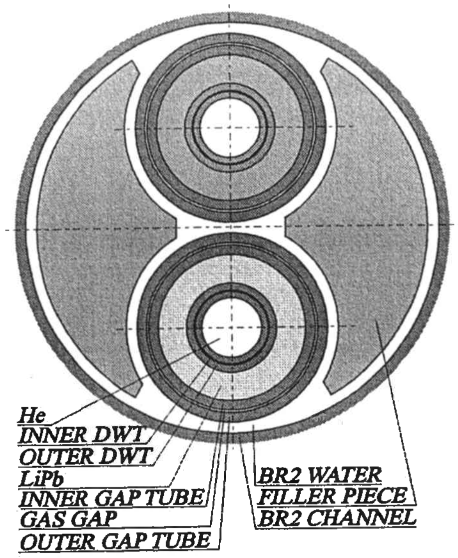


Figure 17: Testing of TPB
Experimental device cross section

For the second experiment, a DWT segment is heated by the gamma flux of BR2 and cooled at the inside by pressurised water. Hence, the temperature gradient tends to separate the two tubes. The purpose is to verify how the brazing agent performs at doses up to 3 dpa. The irradiation device is installed inside the central hole of a BR2 fuel element, in order to maximise the fast neutron component of the flux received by the target.
As an alternative for this experiment, it has also been proposed to perform the irradiation in a special basket in the Callisto loop. Costs are strongly reduced by the use of existing installations, with adequate, if not ideal, irradiation conditions.

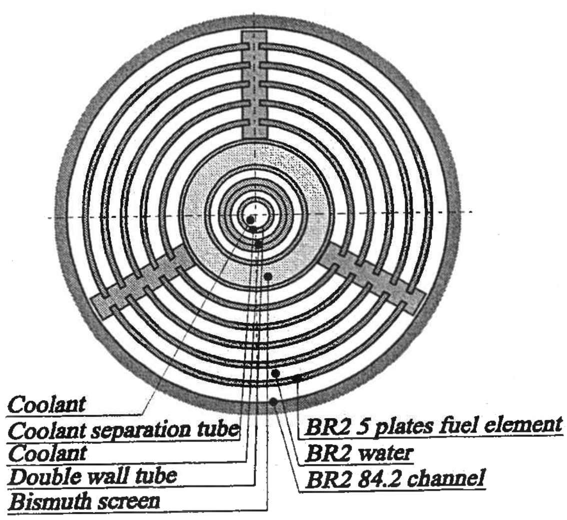


Figure 18: Testing of DWT
Experimental device cross section

A proposal was submitted for the experiments. From discussions with the task management, it became apparent that the budgets available would not allow to perform in the near future any irradiation experiment linked directly to the WCLL blanket development. Hence, we proposed to re-orient the project towards a flexible device for the irradiation of blanket material samples.

MISTRAL Project

The purpose of the MISTRAL project (for Multipurpose Irradiation System for Testing of Reactor Alloys) is to irradiate blanket material samples (such as tensile and Charpy specimens) in the temperature range of 200 – 350 °C. The requirements for the experiment are:

- A multi-purpose irradiation system composed of one small device that can be installed either inside the central hole of a BR2 fuel element or in a standard ϕ 84 mm channel. The central hole of a fuel element allows the highest fast neutron dose and the fastest dpa accumulation rate. Three identical devices could also be installed inside a standard BR2 84 mm channel, which will allow irradiating a larger number of samples with a smaller fast neutron dose.

- The control of the irradiation temperature must be performed with an accuracy of ± 5 °C.

Among the possible solutions which fit those requirements, the best candidates are:

- A double wall capsule filled with NaK, a He/Ne mixing panel to control the temperature.
- A single wall boiling water capsule, the pressure regulation controls the temperature.

SCIENTIFIC PARTNERS

Commissariat à l'Energie Atomique (CEA) - Saclay, France

REFERENCES

[1] Ph. Benoit, Ch DE Raedt; M. Decreton; In-Pile Testing of WCLL Double Wall Tubes, 5th International Symposium on Fusion Nuclear Technology (ISFNT-5), Roma, September 19-24, 1999 (to be published in "Fusion Engineering and Design").

2.6. Tritium Breeding and Materials: Materials Development

2.6.1. Assessment of the Potential of Titanium Alloys (Task SDS 2.2: Scoping Investigations on Titanium Alloys)

PRINCIPAL INVESTIGATOR: L. Coheur

OBJECTIVES

The interest of using titanium alloys in the future fusion reactors has already been largely recognised and discussed. However its lower temperature capability compared to other materials has made it less attractive and the development works on this material were soon abandoned. At present however, activation potential has received an increased attention among the selection requirements of fusion materials. Titanium is therefore worth being considered again. Compared to other low activation material like SiC or vanadium, titanium has specific advantages: lower cost, well known fabrication, rich data base in the unirradiated condition, large mineral resources. This makes it worth restarting the development and qualification works necessary for its use as fusion material.

ACHIEVEMENTS

A report finalising the review task has been issued in April 1999 [1]. In this report, one reviews briefly the main points concerning the use of titanium in a fusion device: activation potential, temperature capability, irradiation behaviour, compatibility with coolants and hydrogen, fabricability, having in mind that all these points are correlated.

The development of titanium metallurgy is primarily a consequence of the need of the aeronautical industry for high specific strength alloys. A lot of compositions has emerged for these extensive studies and is now available in large amounts. A large database also exists covering the physical, mechanical and corrosion properties of these alloys.

Titanium alloys can be very attractive for use in a fusion reactor because they have good mechanical strength at temperatures up to 500°C, good thermal properties, low activation potential, good weldability, good corrosion resistance.

The titanium resources are also very satisfactory. Contrarily to ferritic steels or vanadium alloys, titanium alloys have never been tested for fission reactor core applications and the knowledge of their behaviour under or after neutron irradiation is rather poor.

The limited results from the irradiation experiments carried out in the US before the abandon of the research on titanium alloys show that in terms of swelling and phase stability the two alpha alloys Ti-6242S and Ti-5621S have the greatest promise. They have also the greatest irradiation creep strength at least at low fluence.

Apart from these US experiments, very few references have been found on the behaviour of titanium alloys under irradiation. The Japanese experiments are limited to very low neutron fluences or are related to ion irradiations with only an examination of the irradiated microstructure. No spectacular modifications are observed, so that large swelling values or big changes in the mechanical properties are not to be expected. A Russian communication at the 1995 ICFRM conference in Obninsk gives results of swelling and mechanical properties after neutron irradiation up to 80 dpa. These scientists conclude that titanium alloys may compete with austenitic steels.

Compatibility with hydrogen and hydrogen isotopes is probably the most difficult specification with which titanium alloys must comply. In general, the alpha alloys have a lower solubility for hydrogen than the beta alloys. However, hydrogen permeation through 5621S titanium has been found three thousand times greater than through 300 series stainless steels. Nitride coating or anodising reduce the permeation at best by one order of magnitude. Considering the very large scatter in these measurements, further investigation in this field is needed. Concerning the low activation potential of the titanium alloys, the most recent calculation of Kelzenberg and Ehrlich (J. Nucl. Mat. 236, 1995, p 319) shows that, with the exception of molybdenum, aluminium is the worst of the alloying elements presently used in commercial titanium alloys. One can consider to limit the aluminium concentration to 1-2 wt% or to replace aluminium by higher tin concentration or to develop completely new alloys.

Compatibility with the various coolants envisaged for a fusion reactor is poorly known. In general, titanium has a very good corrosion resistance in aqueous media. Its behaviour in contact with liquid lithium seems also to be satisfactory. This is again a subject which needs further investigation.

In short, we can conclude that:

- alpha alloys offer the best possibilities;
- some commercial alloys, if their good irradiation behaviour is confirmed, could be proposed for some parts of the fusion reactor (vacuum vessel, shield);
- for some other parts (first wall, blanket) new compositions must probably be developed.

REFERENCES

[1] L. Coheur, Assessment of the Potential of Titanium Alloys for Fusion Reactors, SCK•CEN Report No, BR3-1247/99-01, April 1999.

2.6.2. Assessment about water chemistry and corrosion (Task SM 3.5.2)

PRINCIPAL INVESTIGATOR: W. Bogaerts, KULeuven

SCIENTIFIC STAFF: J. H. Zheng

OBJECTIVES

- Review international literature with special emphasis on ferritic/martensitic steels, but also including experiences from fission (LWR) as far as possible.
- Review on corrosion modelling.
- Review of the status of water corrosion/compatibility experiments from previous programmes with special emphasis on ferritic/martensitic steels.
- Analysis and assessment of all corrosion data in the programme.

ACHIEVEMENTS

Based on the literature information collected from various international journals, computerised literature searching programs, and internal reports, the following objectives have already been achieved.

Materials aspects: literature study about properties associated with metallurgical variables of ferritic/martensitic steels which could be related to their corrosion behaviour.
Water chemistry: review on water chemistry applied to light water reactors, particularly PWR and on radiolysis data both in fission and in fusion (computation).
Experimental data and Corrosion modelling: review on the corrosion data already existed in the literature, and on mechanism modelling of chromium steel in high temperature water environments.

However, a review of the status of water corrosion/compatibility experiments from previous programs, and the analysis and assessment of the corrosion data in the programme were delayed, as some reports on corrosion data were hard to obtain.
The completion of the final report is on-going.

2.6.3. Improvement of RAFM (Reduced Activation Ferritic-Martensitic) steels for DBTT (Ductile-to-Brittle Transition Temperature) & Irradiation Hardening (Task SM.1 - European Blanket Programme – Materials)

PRINCIPAL INVESTIGATOR: E. Lucon

OBJECTIVES

- Evaluation and assessment of results from 95-98 period.
- Production of laboratory heats with promising compositions and qualification in reference condition.
- Irradiation of the steel to "saturation" radiation levels (1.5 ÷ 2.5 dpa) and screening of toughness, strength and ductility afterwards.
- Improvement of specification for RAFM steel composition.

ACHIEVEMENTS

Several heats of RAFM steels have been produced, irradiated and tested in the previous years by different partners. The new reference industrial RAFM steel, EUROFER'97, is now being produced. Evaluation and further refining in terms of chemical composition, towards the final steel (called EUROFER'2000) to be used for building ITM's and blankets were planned.
The availability of steel samples was however delayed and SCK•CEN work was therefore limited in 1999 to general bibliographical data gathering, as well as contacts with the task partners. In March 1999, a meeting was held in NRG Petten [1], with the aim of revising the work of the previous years. A Draft Report [2], prepared by B. van der Schaaf and A.A. Tavassoli, was circulated at the end of August 99. SCK.CEN provided comments and remarks.

Conclusion and foreseen future work

A meeting will be held before the end of the year. In this meeting, the activity (irradiation and material characterisation) for the following years will be discussed.

REFERENCES

[1] R. Chaouadi and E. Lucon, SCK•CEN Concise Mission Report, 3 March 1999.
[2] B. van der Schaaf and A.A. Tavassoli, "Hardening and Toughness from Radiation, and Reference Characterisation of F82H and Screening Steels", *EPB Structural Materials Milestone 4*, Draft, 20027/99.25387/P, Petten, 17 August 1999.

2.6.4. *ODS RAFM (Oxide-Dispersion Strengthened) Behaviour Assessment (Task SM6 - European Blanket Programme – Materials)*

PRINCIPAL INVESTIGATOR: E. Lucon

OBJECTIVES

This task is concerned with the production and the assessment of properties of bulk RAFM steel plates. In close co-operation with industry, different manufacturing routes and ODS versions will be explored. The main objectives are as follows:

- Manufacturing procedure for bulk production of ODS RAFM steel.
- Assuring the required increase in mechanical properties at 900 K.

ACHIEVEMENTS

A kick-off meeting was held in FZK Karlsruhe (Germany) on June 23, 1999 [1]. In addition to more precisely defining the requirements for the best candidate ODS alloy, the following actions were agreed upon:

- continuation of the EUROFER fabrication line (FZK);
- gathering of information about the purchase of commercially available ODS alloys (FZK);
- literature review on past SCK•CEN activity on ODS alloys production (SCK•CEN);
- literature review on the effect of Chromium content on the mechanical properties of ODS alloys (all partners).

Actions will be reviewed in a second meeting to be held in PSI (Switzerland) on Oct 5, 1999.

Conclusion and foreseen future work

Once the agreement will be reached on the best material(s) to be characterised, a planning of the activity (preliminary testing/main testing) will be established between the different partners involved in the work package. Only preliminary tests are foreseen in 2000.

REFERENCES

[1] E. Lucon, Concise Mission Report, SCK•CEN/EL-MR5-99, 24 June 1999.

2.6.5. *Chromium Alloys Characterisation (Task ADV2: Advanced Materials Characterisation)*

PRINCIPAL INVESTIGATOR: E. Lucon

OBJECTIVES

In terms of characterisation of Cr alloys properties, the main objectives of the task are as follows:

- obtain experimental toughness, tensile and creep data;
- characterise the microstructure, also after neutron irradiation;
- provide feedback to the material supplying industry;
- provide data for evaluation by designers for blanket application.

The outcome of the present activity will support decisions on eventual continuation and intensification of the Cr alloy development, helping designers to assess the potential of such alloys for blanket construction.

ACHIEVEMENTS

A kick-off meeting was held in SCK•CEN Mol on March 19, 1999 [1]. An agreement was reached with PLANSEE (Austria) for the supply of two commercial alloys: DUCROPUR (Cr 99,96%) and DUCROLLOY ($Cr5Fe1Y_2O_3$), in as-received and recrystallised condition (4 material to be characterised). The test matrix, including testing of materials in baseline (non-irradiated) condition, irradiation (in BR2 and HFR-Petten) and post irradiation examination (PIE) testing, was also basically agreed upon.
As far as SCK•CEN is concerned, the following steps have been carried out.

- 12 tensile specimens, for baseline and PIE testing, were delivered at the end of June:
  - 4 specimens (2 per as-received material) have been tested in baseline condition, 2 at room temperature and 2 at 300 °C; only the DUCROPUR specimen tested at 300 °C has shown appreciable ductility;
  - 8 specimens (2 per material) have been loaded in the BR2 reactor (CALLISTO rig, [2]) and irradiated for two cycles (first cycle terminated in August [3] and second in October).
- 60 sub-size Charpy-V specimens (15 per material) for baseline testing were delivered at the end of August:
  - instrumented impact tests have started at the beginning of September; first results, still to be fully evaluated, indicate an extreme fragility of all materials up to temperatures as high as 500 °C (maximum temperature that SCK•CEN sub-size specimen test facility can reach).
- 40 standard Charpy-V specimen (10 per material) were delivered in September: testing is on-going.

Conclusion and foreseen future work

Preliminary test results in the baseline condition, although yet not fully analysed, obtained on the different materials clearly indicate an extreme fragility; the recrystallization treatment does not seem to improve the toughness of the two selected alloys.

As far as the sub-size Charpy-V specimens are concerned, in view of the results obtained so far, static fracture toughness testing will be considered instead of impact tests; the feasibility of fatigue pre-cracking will be investigated. If this is inapplicable (due to the fragility of the materials), 3-point-bend tests on notched specimens will be carried out and toughness data equivalent to pre-cracked samples will be evaluated using literature correlations.
For the standard Charpy samples, impact testing is foreseen up to temperatures around 1000 °C (using a different furnace); depending on the significance of results, some static 3-point-bend testing could also be performed. Details as in the previous paragraph.
All the activity defined in the kick-off meeting will be terminated and reported by early 2000; the activity for the following years, is being presently discussed and will depend on the outcome of the first series of tests.

REFERENCES

[1] E. Lucon and R. Chaouadi, Concise Mission Report, EL-MR1-99, 22/3/99.
[2] E. Lucon and M. Wéber, "IRMAS: Summary Report - Revision 1, Interne Memo MI.57/B043004/39/EL/MW.nvdb, 29/6/99.
[3] P. Benoit and M. Weber, "Boucle CALLISTO: Rapport du cycle 03/99 A - Irradiation du 21.07.99 au 11.08.99", Memo Interne MI.57/D088010/43/PB/MW.rw, 7/9/99.

2.7. Safety and Environment: Safety Analysis and Environmental Impact

2.7.1. Reaction of Beryllium with air and steam (Task SEAL-1.1)

PRINCIPAL INVESTIGATOR: F. Druyts

SCIENTIFIC STAFF: P. Van Iseghem

OBJECTIVES

When considering beryllium as a primary choice for the plasma facing components in future fusion reactors, the following accident scenarios have to be taken into consideration: the loss of vacuum accident (LOVA) and the loss of coolant accident (LOCA). The safety analysis of these hazards requires knowledge on the kinetics of the beryllium/air and beryllium/steam reactions respectively. In particular, additional data on high temperature/pressure conditions and on the reactivity of irradiated beryllium are needed. The main goal of this research programme therefore, is to determine the beryllium/air and beryllium/steam reactivities with thermal analysis. At the beginning of the reporting period, the analysis of the beryllium/air reaction was almost complete and the analysis of the beryllium/steam reaction had begun.

ACHIEVEMENTS

The experimental work was finished and the obtained data from were analysed. The reactivity of three types of beryllium in air and steam was determined with combined thermogravimetric and differential thermal analysis (TG/DTA):

- unirradiated dense S-200F Be ("UD");
- irradiated dense S-200F Be, irradiated up to a fast neutron fluence (E>1MeV) of $1.6\ 10^{21}$ n/cm^2, resulting in a helium content of 300 appm and a theoretical density of 99.9 %;
- irradiated porous S-200E Be, irradiated up to a fast neutron fluence of $4\ 10^{22}$ n/cm^2 (21000 appm He, 97.2 % dense).

Figure 19 summarises the results of the beryllium/air experiments at temperatures between 600°C and 1000°C, while figure 20 shows the measured beryllium/steam reactivities. The reaction rates were determined from TG curves and are based on the geometrical surface area of the beryllium samples. At low temperatures (600°C and 700°C for the dense material, 600°C for the porous beryllium) kinetics tend to be parabolic, indicating protective oxidation during the length of the experiments. At higher temperatures kinetics are accelerating linear. This behaviour is associated with diffusion of air/water molecules through the grain boundaries and subsequent detachment of grains, leading to an increased effective surface area.

The reactivity of irradiated beryllium is slightly higher than that of unirradiated beryllium. This effect is bigger in air than in steam.

In our tests, we did not observe any pronounced influence of porosity, but from the literature it is clear that the 3 % porosity of the investigated irradiated porous material might too low to have any practical effects on the reaction rate.

Comparison of the measured hydrogen generation rates for beryllium in steam with data from INEEL in general yielded a good agreement. Only for 86 % dense beryllium from INEEL, the reaction rate is one order of magnitude higher than the reaction for dense material or with theoretical densities above 92 %.

Additional measurements were made on polished samples to elucidate the causes of the large scatter in results at low temperatures. Surface analysis with Scanning Electron Microscopy of the reacted non-polished samples revealed that oxidation initialised preferentially at ridges produced by the production process of the beryllium discs. From the TG/DTA results for polished samples at 600°C and 700°C the following conclusions can be drawn:

- the scatter in results decreases after polishing, and
- the polished samples have a lower reactivity than the non-polished samples.

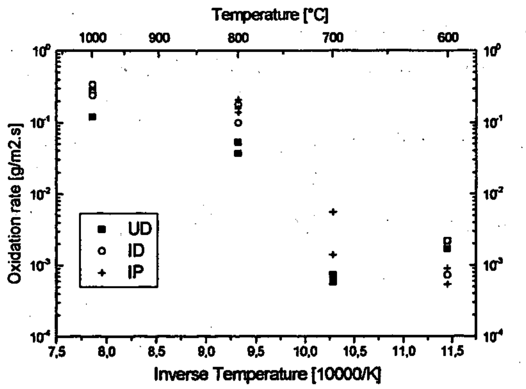


Figure 19: Oxidation rates of beryllium samples exposed to air at different temperatures, derived from TG data and based on the geometric surface area of the samples: unirradiated dense (UD), irradiated dense (ID) and irradiated porous (IP) beryllium.

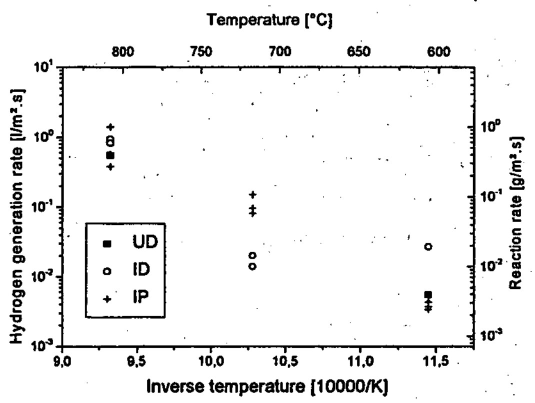


Figure 20: Oxidation rates of beryllium samples exposed to steam at different temperatures, derived from TG data and based on the geometric surface area of the samples: unirradiated dense (UD), irradiated dense (ID) and irradiated porous (IP) beryllium.

The main goals of the research programme were reached. All final reports dealing with experimental results were delayed for an average of six months. This was due to the tight experimental schedule and the associated delay in 1998.

Future Activities

Additional activities were continued after the delivery of the final report for SEAL1 (thermal analysis and surface analysis of polished samples, advanced validation of the results including parabolic fitting of the low temperature curves). These were not reported in [2] A revised report, including the latest results and integrating results from thermal analysis and surface analysis, will be submitted in December 1999.

REFERENCES

[1] F. Druyts, P. Van Iseghem, "Interaction of Beryllium with Air/Steam: Determination of Chemical Reactivities with Thermal Analysis. Final Report," (1999). SCK•CEN, BLG-806.

[2] F. Druyts, P. Van Iseghem, "Interaction of Beryllium with Air/Steam: Determination of Chemical Reactivities with Thermal Analysis," (1999). SCK•CEN, BLG-806.

[3] A. Leenaers, M. Verwerft, "Be/Air and Be/Steam Reactions: Microstructure of the Beryllium Samples. Final Report," (1999). SCK•CEN, R-3329.

[4] F. Druyts, R.A.H. Edwards, F. Scaffidi-Argentina, V. Van Alsenoy, F. Joppen, E. Colino, P. Van Iseghem, "Properties of Irradiated Beryllium. SEAL1 Summary Report," (1999). SCK•CEN, BLG-817.

[5] F. Druyts, P. Van Iseghem, "Thermal analysis of the beryllium/steam reaction," paper accepted for 5th International Symposium on Fusion Nuclear Technology, Rome, Italy, September 19-24, 1999.

[6] F. Druyts, P. Van Iseghem, "Determination of Beryllium Reactivities with Thermal Analysis," paper accepted for the 4th IEA International Workshop on Beryllium Technology for Fusion, Karlsruhe, Germany, September 15-17, 1999.

2.7.2. Chemical Energy Inventories (Task SEAFP2-1.3, Energy Inventories and Hazards)

PRINCIPAL INVESTIGATOR: F. Druyts

SCIENTIFIC STAFF: P. Van Iseghem, M. Aertsens

OBJECTIVES

In the framework of SEAFP-2, three candidate Reactor Models are considered for DEMO. The purpose of this microtask was to assess the chemical energy inventories The purpose of SEAFP microtask 1.3 is to assess the chemical energy inventories for these three reactor models. The focus was on the beryllium reactions, as these are the most exothermic of all possible reactions in the Reactor Models in case of an accident. Also, new data on the beryllium/air and beryllium/steam reactivities from the SEAL subtask 1.1 ('Beryllium reactivities') could be used for the calculations. The work package of microtask 1.3 consisted of three parts:

- establishing an inventory of the materials in the main components of the three reactor models;
- reassessing the chemical energy inventories established in SEAFP-1, and;
- kinetic modelling of SEAL data.

ACHIEVEMENTS

The beryllium present in the divertor, first wall, and blankets constitutes the main chemical hazard for the three reactor models. In SEAFP-1, it was assumed that all beryllium in the plasma facing components (7.17 tons) reacts with steam, yielding a chemical energy of 285 GJ. If we assume that the beryllium reacts completely with steam, a reassessment of the chemical energy inventory yields 295 GJ for each reactor model. Reactor Models 1 and 3, however, are cooled with helium, which makes the assumption of a beryllium/steam reaction unrealistic. Therefore, a second approach was used, first defining a loss of coolant accident (LOCA) and a loss of vacuum accident (LOVA), and then calculating the inventories for these two accident scenarios for each of the Reactor Models. In case of a LOCA, only Reactor Model 2 releases a chemical energy of 295 GJ (in case of complete reaction), while the chemical energy inventory is equal to zero for Reactor Models 1 and 3. In case of a LOVA, all three Reactor Models have the same chemical energy inventory of 495 GJ for the beryllium/air reaction.
Kinetic modelling was performed on the data from thermogravimetric analysis (TG) obtained in the SEAL framework. At low temperatures (600°C and 700°C) a diffusion model was used, and at higher temperatures (800°C, and 1000°C) a Monte Carlo based model was used to fit the linear part of the TG curves. The results were rather poor at 600°C and 700°C, because of the large scatter on experimental TG data. At 800°C and 1000°C, fit results were consistent. Also, a second, more direct approach was used to obtain kinetic data from the TG curves. This approach consisted in considering the maximum reaction rate. Results for both the beryllium/air and the beryllium/steam reaction are reported in [1].

REFERENCES

[1] F. Druyts, M. Aertsens, P. Van Iseghem, "Chemical Energy Inventories: Final Report for SEAFP-2 Microtask 1.3," (1999). SCK-CEN, R-3338.

2.8. Safety and Environment: Waste Management

2.8.1. Study of Dismantling Strategies (Task 5-2a: Management of Activated Materials)

PRINCIPAL INVESTIGATOR: L. Denissen

SCIENTIFIC STAFF: V. Massaut, M. Klein

OBJECTIVES

Present-day design of nuclear installations includes planning for decommissioning. This covers not only technical capabilities, but also a financial analysis. In order to cover the nuclear liabilities proceeding from the use of nuclear facilities, a cost estimate for the decontamination/dismantling phase must be calculated, allowing for suitable provisions to be made during the lifetime of the facility. This applies also to fusion reactors, as they will similarly generate radioactive material. Therefore it is necessary to look at the impact of different decommissioning strategies on the management of activated fusion materials.

For this cost estimate, a detailed analysis of decontamination/dismantling strategies and technical alternatives has to be carried out.

ACHIEVEMENTS

The past ten years SCK•CEN has gained a lot of know-how and expertise during the decommissioning of a small fission reactor. The BR3 reactor is a 10 MW_e pressurised water reactor (Westinghouse) used as fuel test and training facility for NPP-operators. The reactor was shutdown in 1987 after 25 years of operation. It was selected as one of the four pilot dismantling projects by the EU for its R&D programme on decommissioning of nuclear installations.

During the past ten years, different decommissioning activities have been carried out. The activities which are similar or could be of interest for the dismantling of a fusion reactor have been analysed and studied and make part of this progress report.

In the framework of the progress report on fusion reactor decommissioning the first chapter handles the know-how gained during the decommissioning of a fission reactor.

A comparison between immediate and deferred dismantling

During operation, the original, so-called Westinghouse, reactor internals were replaced by another set. Those WH internals were than stored in the reactor pool for more than 30 years. This gave us the opportunity to compare an immediate dismantling with a deferred dismantling (WH internals 30 years decay). The conclusion is that there is no significant radiological, technical or economical profit gained by deferred dismantling, at least for the deferred period considered.

Comparison of different cutting techniques for cutting highly activated components

For the dismantling of the highly activated reactor internals a study has been made for three different cutting techniques: the plasma arc torch, EDM and mechanical sawing. This allowed us to compare the amount of secondary produced waste for these three cutting techniques. As a result the mechanical cutting machine was chosen because the production of secondary waste is low and the type of waste (chips and swarfs) can easily be collected in filters.

For cutting the activated parts, remotely operated cutting tools have been used. Those tools should be simple and easy to control. This can only be an advantage for the operators. The difference with the dismantling of a future fusion reactor lay is the fact that during normal operation of the fission reactors no remote equipment is used except for the fuel manipulations. This is a big advantage for the decommissioning of a fusion reactor where remote handling is already available during the operation time. This equipment can also be used during the dismantling phase of the highly activated parts inside the cryostat.

Decontamination of primary, highly contaminated loops

The first activity was a full system decontamination of the primary loops. It is important to mention that a decontamination and cleaning of highly contaminated circuits should be done as soon as possible after shutdown of the plant. A lot of money can be saved when you can make use of the plant equipment without any investments in maintenance and repair of the equipment that can become unusable after 20 - 30 years after the plant is shutdown. Also the know-how of the plant operators is then still available. It is also important in the scope of ALARA, while a dose reduction for the operators during dismantling activities can be obtained. This is relevant for the blanket modules and their cooling circuits.

In the decommissioning of nuclear installations the waste cost is of strong importance in the overall decommissioning cost, therefore a lot of experience is gained trough tests for different decontamination techniques for concrete and metallic parts to minimise the radio active waste during dismantling operations. These techniques include manual decontamination (simple washing), decontamination with abrasives as well as chemical decontamination.

To control the waste stream a waste management system is set up with a computer database to identify every part that is dismantled. This allows following each dismantled part during each step of the whole material route.

Future work

After issuing the foreseen report end of 1999, the work will be continued with the following items:

An in depth comparison between the decommissioning of a fission reactor and a prototype fusion reactor (e.g. ITER) in the field of cutting techniques, decontamination and waste treatment.

The extensive use of remote handling systems with viewing systems that still can work under high radiation doses.

Due to the much higher doses to be expected inside the cryostat the radiological aspects have to be taken into account for the workers as well as for the public.

The condition of several components after shutdown under neutron irradiation during several years has to be analysed.

Depending on the type of material proposed/used for the future fusion reactors the waste stream management should be analysed.

Also the presence of tritium has to be studied and the influence of the tritiated waste on the waste stream.

2.8.2. Analysis of Waste Site Intrusion Scenarios (Task 5-2b Management of Activated Materials)

Principal Investigator: J. Marivoet

Scientific Staff: X. Sillen, S. Labat

Objectives

Analyses of the performance of possible repositories for the disposal of radioactive waste that will arise from the operation of fusion reactors have already been carried out for the case of the normal evolution scenario [1]. In 1998 SCK•CEN has been contacted by the co-ordinator of the Safety and Environmental Studies Programme to carry out a complementary assessment of the performance of a fusion waste repository system in the case of human intrusions.

Classical approach to the analysis of human intrusion scenarios

Until 1995 the analyses of the radiological consequences of human intrusion scenarios were strongly focusing on drastic "disruptive" scenarios in which the technical and natural barriers of the repository system are short-circuited. Those scenarios will yield, of course, the most severe consequences. In the case of deep disposal, mainly a core examination scenario and a "residence" scenario are considered.

The core examination scenario has already been defined by NRPB [4] in 1987 and it has since then been used in various performance assessments of deep disposal systems for radioactive waste arising from fission reactors as a stylised scenario for human intrusion. The scenario considers the drilling of a geological reconnaissance borehole from which cores containing fragments of the disposed waste are taken. It is assumed that because of the discovery of an "unusual formation" the cores will be extensively examined by a geological worker, who is not aware of the presence of radioactive materials in the core. This scenario can lead to doses of several tens of Sievert in the case of disposal of high-level waste and spent fuel arising from fission reactors.

The "residence" scenario considers the drilling of a borehole through one of the disposal galleries at a repository site. It is assumed that the cuttings, among which would occur remnants of the disposed waste, were left in the neighbourhood of the drilling site. This causes a contamination of the drilling site. Afterwards a house might be constructed on the site and the residents would receive doses via external irradiation from the soil, inhalation of dust and the ingestion of vegetables cultivated on a contaminated field.

The above scenarios are not certain to occur and it was considered that their probability of occurrence is significantly small. In the classical approach to human intrusion scenarios, the probability of occurrence of the scenarios was estimated on the basis of extrapolation of drilling frequencies during the last decades. The product of the estimated probability of occurrence with the calculated dose gave the risk corresponding to the considered scenario.

New developments

In 1995 the U.S. National Academy of Sciences (NAS) published a report on technical bases that can be applied to define public health and safety standards to be applied for the proposed repository at Yucca Mountain (NAS, 1995).

The NAS concluded that it is not technically feasible to assess the probability of human intrusion into a repository over the long term, because future economic conditions and exploration technology cannot be predicted. The drastic intrusion scenarios are not considered as being relevant for a given site and repository concept, because they lead to the same consequences independently of the site or concept. On the other hand the NAS report recommends to carry out calculations of the consequences for particular types of intrusion events, for example drilling one or more boreholes into and through the repository.

The NAS report identifies three categories of hazards resulting from intrusion:

- hazards to the intruders (drillers, handlers of material taken from the repository);
- hazards to the public from material brought to the surface by the intrusive activity;
- hazards that arise because the integrity of the repository's engineered or geologic barriers have been compromised by the intrusion.

For the first two categories the NAS report concludes "that analysing the risks to the intrusion crew and the risks from any material brought directly to the surface as a consequence of intrusion is unlikely to provide useful information about a specific repository site or design", but it recommends "that the compliance analysis should concentrate on the third category of hazard posed by human intrusion, the one resulting from modification of the repository's barriers". The NAS report describes how this last category can be analysed: a stylised intrusion scenario consisting of one borehole of a specified diameter drilled from the surface through a canister of waste to the underlying aquifer and by considering current drilling technology, but assuming sloppy practices, such as not plugging the hole carefully when abandoning it.

Begin 1999, the NAS recommendations have been largely taken over by the Nuclear Regulation Commission (NRC) in a Proposed Rule concerning the proposed repository at Yucca Mountain (NRC, 1999).

Adapted approach to the analysis of human intrusion scenarios
The drastic intrusion scenarios have no longer to be considered in performance assessments of deep disposal systems, because they are no longer considered as relevant for the selected site or the proposed repository design. However, to illustrate the impact of the relatively fast decrease of the radioactivity in fusion waste, we will make calculations for the core examination scenario and compare the radiological consequences calculated for fission waste with those calculated for fusion waste.

The analysis of human intrusion scenarios is now reoriented on the analysis of the impact of a borehole drilling through a repository on the functioning of the main barriers of the repository. We consider that a borehole with a diameter of 30 cm is drilled through the repository and that this borehole is abandoned without sealing it. Ground water will flow through the unsealed borehole and will come in contact with radionuclides from the disposed waste.

However in the case of disposal into a plastic clay formation, such as the Boom Clay layer, the host formation has a strongly self-healing characteristic: convergence of the clay will result in the sealing of the borehole. The time during which ground water will flow through the borehole will be estimated. In a next step the radiological consequences resulting from the temporary flow of ground water through the unsealed borehole will be calculated.

REFERENCES

[1] Broden K., M. Lindberg, G. Olsson (1999), Repository analysis for fusion reactor waste. Studsvik RadWaste, Nyköping, Technical note RW-98/46.

[2] NAS (1995), Technical bases for Yucca Mountain Standards. National Academy Press, Washington.

[3] NRC (1999), Proposed Rule: Disposal of High-level Radioactive Wastes in a Proposed Geological Repository at Yucca Mountain, Nevada. Federal Register, February 22, 1999, Vol. 64, Number 34, pages 8639-8679.

[4] Smith G.M., H.S. Fearn, C.E. Delow, G. Lawson and J.P. Davis (1987), Calculations of the Radiological Impact of Disposal of Unit Activity of Selected Radionuclides. NRPB, Chilton, report NRPB-R205.

3. Underlying Technology

3.1. Vessel/In-Vessel: Assembly and Maintenance and Physics Integration/Diagnostics

3.1.1. Environmental tolerance assessment and testing of optical fibres

PRINCIPAL INVESTIGATOR: F. Berghmans

SCIENTIFIC STAFF: B. Brichard, M. Van Uffelen

OBJECTIVES

The objective of this activity is to investigate whether fibres can be used for communication, sensing and imaging in monitoring and remote-handling applications around future thermonuclear reactors. The task is focused on the study of the environmental tolerance of optical fibres. The work involves theoretical studies on the physical mechanisms responsible for radiation induced degradation in fibres and has been extended to other essential fibre-optic link components, such as optical sources, optical detectors, Bragg-gratings and fibre connectors. Dedicated optical fibres for distributed fibre-optic dosimeters have been investigated as well.

ACHIEVEMENTS

Fibre-optics for communication

Optical fibre technology is seriously considered for communication and monitoring applications during the operation and maintenance of future thermonuclear fusion reactors. Their environment is characterised, in particular, by possibly high gamma dose-rates and total doses up to 100 MGy. The feasibility of applying photonic technology in such intense radiation fields therefore needs to be assessed. Whereas many reports deal with the radiation behaviour of a variety of fibre-optic devices, only little information is available on the radiation tolerance at high total dose (e.g. > 1 MGy). We have conducted high total dose (up to 15 MGy) irradiation experiments on a variety of COTS fibre-optic devices, including edge-emitting laser diodes, vertical-cavity surface-emitting lasers, PIN photodiodes and single-mode optical fibres. In a first experiment, we submitted these devices to a gamma dose-rate around 3 kGy/h up to a total dose exceeding 3 MGy. A pure silica (Oxford Electronics Ltd.) and two Ge doped single mode optical fibres (Philips and Corning) for telecommunication purposes have been gamma irradiated up to relatively high total doses. A remarkably low induced loss of about 30 dB/km at a total dose of 3 MGy was obtained with the pure silica core fibre (see figure 21). This is significantly lower than in the conventional Ge-doped fibres, making the Oxford Electronics Ltd. fibre an excellent candidate for application in the ITER environment.

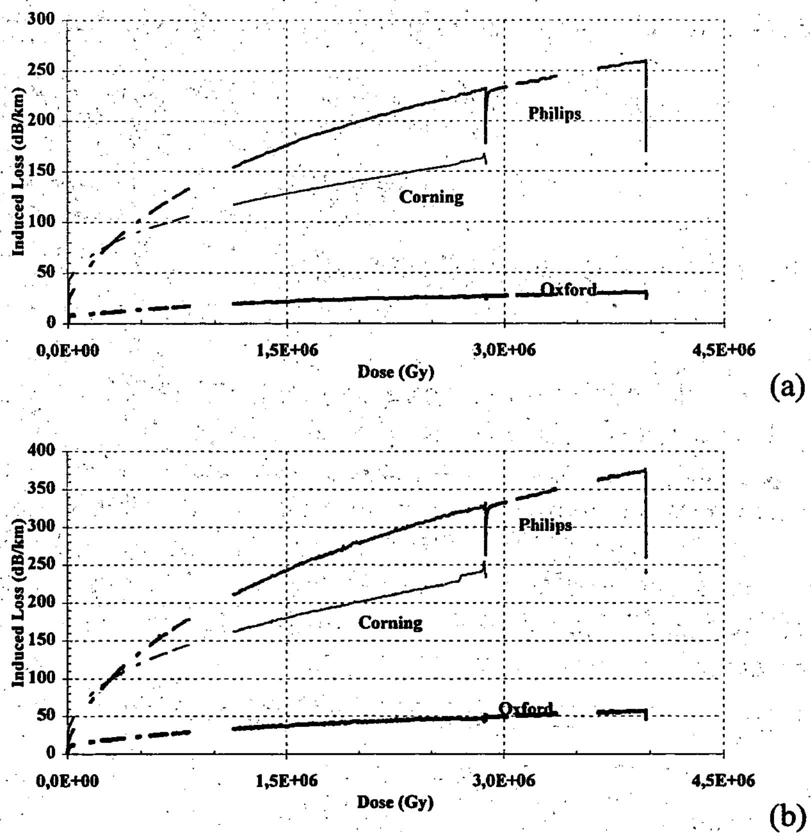


Figure 21: Radiation induced loss in three single mode fibres at 3.3 kGy/h.
(a) 1310 nm and (b) 1550 nm

The different types of emitters showed maximum losses of 8 dB at doses on the order of 1 MGy. This should allow to construct emitter-fibre assemblies for transmission over lengths of 100 m with radiation induced losses on the order of 11 dB for dose-rates of a few kGy/h and temperatures around room temperature.

This, however, is no longer the case when higher dose-rates of 25 kGy/h and higher temperatures are considered. Higher losses, from 15 dB to 20 dB, were observed in both LEDs and VCSELs (see figure 22), whereas the edge-emitting laser diodes kept a loss around 7 dB at MGy doses but revealed more sensitive to temperature changes. These higher losses are probably due to the influence of the packaging and lensing of the devices. Photodiodes still remain the weakest elements in the fibre-optic link. Operation up to 1 MGy (at 2.7 kGy/h) is possible for InGaAs PIN photodiodes, but unfeasible for Si photodiodes. More complete and statistically relevant research on the **combined effect of high dose-rates and temperature variations** on photonic devices is therefore necessary to assess their possible use in enhanced radiation environments like those present around ITER. The mechanical degradation of connectors and fibres also need to be the subject of such studies. Eventually, these investigations should be completed with the evaluation and qualification of fibre-optic systems, including the emitter, the fibre and its connections as well as the receiver, if applicable.

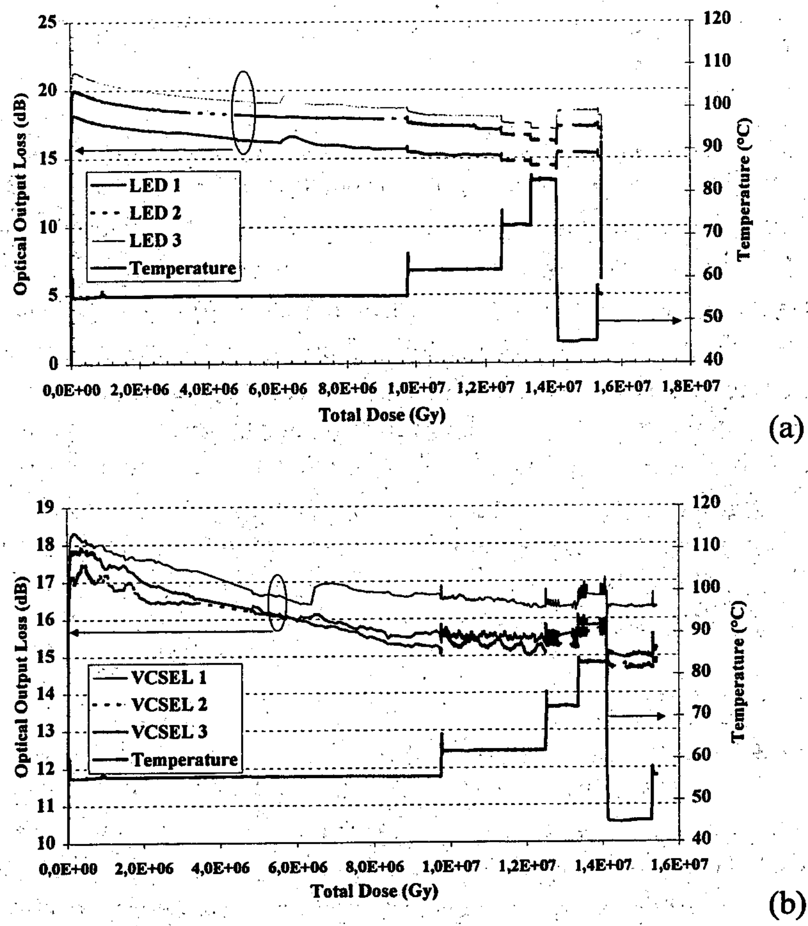


Figure 22: Radiation induced optical output loss of
(a) Honeywell LEDs at a 76 mA forward current and
(b) Mitel VCSELs at a 12 mA forward current at 29.4 kGy/h

Fibre Bragg-gratings

In fibre Bragg-grating (FBG) sensors can be employed to measure different physicals parameters such as temperature and strain. Their main advantage in radiation environments is that the absolute measurement is wavelength-encoded. Measurements should therefore be insensitive to the radiation-induced attenuation in the fibre. Such optical fibre sensors (OFS) have other numerous advantages, which include immunity to electromagnetic interference, intrinsic safety, small size, a possibly high sensitivity, multiplexing capabilities, and remote interrogation. The feasibility of using OFS in a radiation environment still needs to be assessed. Moreover, we recently demonstrated that commercial-off-the-shelf (COTS) optical fibre sensors have to be redesigned to withstand radiation. Here, we report on irradiations performed to provide a sound basis for the design of an optical fibre sensor capable to operate in fusion relevant environments.

The parameters of FBGs were monitored under radiation with the total accumulated dose in excess of 1.5 MGy. The gratings were written using the phase mask technique in 1) one photosensitive Ge-doped fibre; 2) two hydrogen-loaded telecommunication Ge-doped fibres; 3) one N-doped fibre. To write a FBG in the N-doped fibre, 193-nm excimer laser was used, while inscription in Ge-doped fibres was done with UV-light from KrF (248 nm) and Ar (300 nm) lasers.

The reflectivity of the FBGs ranged from 1 to 7 dB. The gratings written in the hydrogen-loaded Ge-doped fibres were annealed at 80°C during 24 h to remove the remaining hydrogen.

The operation of a grating sensor is based on the dependence of the Bragg wavelength λ_B on the temperature T. For a restricted temperature interval (0-100 °C), this dependence is linearly approximated. The temperature can be calculated from the equation: $T = T_0 + (\lambda - \lambda_0)/\alpha_B$, where α_B is the temperature sensitivity coefficient and λ_0 is the Bragg wavelength at T_0. A possible application of a fibre grating sensor in radiation environments therefore depends on to what extent α_B and λ_B are changed by radiation. Radiation influences the refractive index of the fibre, which results in a shift of the Bragg wavelength. Such a shift corresponds to a fictive change of the temperature. Therefore, it is required to decrease the sensitivity of the Bragg peak to radiation. Hydrogen loading is often used to enhance the fibre photosensitivity. However, we found that it increases simultaneously the sensitivity to γ-radiation (see figure 23). Therefore, it seems preferable to use unloaded fibres. For the FBGs under study the value of α_B remained unchanged under radiation, although it can be influenced by radiation for heavily Ge-doped fibres.

The choice of a fibre for writing radiation-resistant FBGs depends also on the expected accumulated dose. N-doped fibres showed the lowest sensitivity to γ-radiation for doses below 100 kGy, although at MGy dose level a γ-radiation induced Bragg peak shift is several times higher than that observed in Ge-doped fibre without hydrogen loading and is comparable to the shift observed in the hydrogen-loaded fibres. We found that the sensitivity to γ-radiation also depends on fibre grating's parameters. Our experimental data indicate how the parameters should be optimised.

In conclusion, our results show that a dedicated design of a FBG sensor can significantly increase its radiation hardness. The behaviour of such fibre Bragg gratings under radiation has been studied. The experimental results show that the temperature sensitivity is not affected by radiation within the accuracy of 3 %. The amplitude and the width of the Bragg resonance also remain unchanged under γ-radiation. The change of the Bragg wavelength as a result of irradiation is not higher than 25 pm and saturates at doses of 0.1 MGy. Recent experiments showed the dependence of the Bragg wavelength behaviour on the fibre type, the irradiation history and confirmed that the radiation sensitivity of fibre Bragg gratings can be optimised by a proper choice of the fibre type.

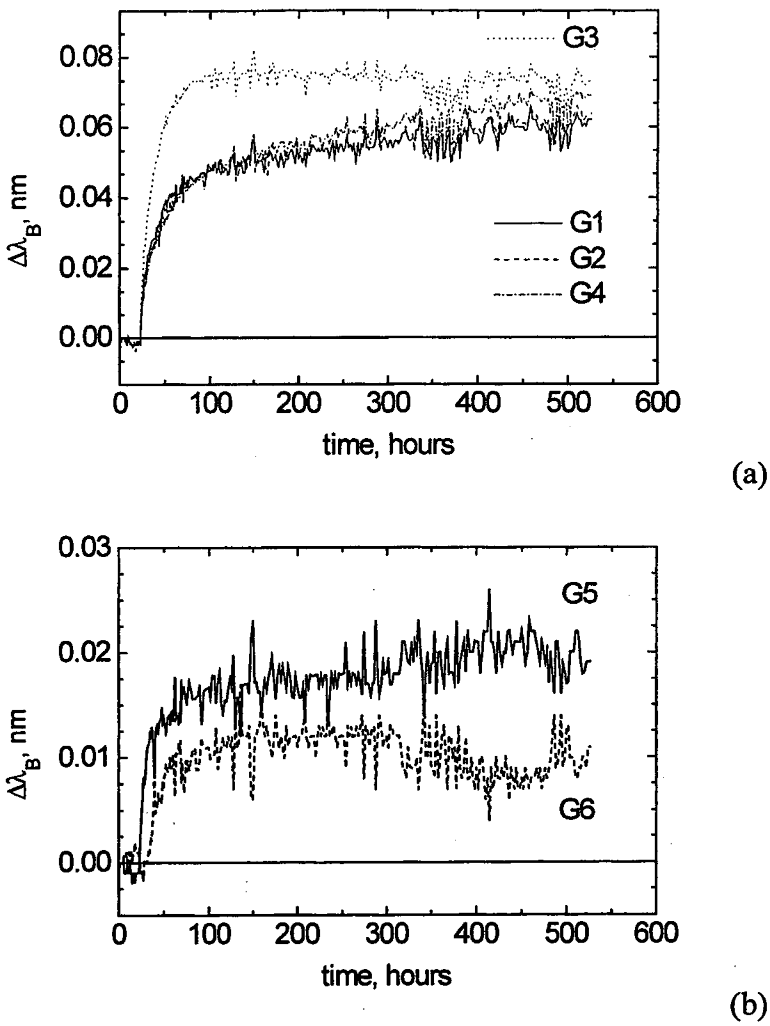


Figure 23: Shift of the Bragg peak under gamma-radiation for FBGs written in gratings written in (a) hydrogen-loaded fibres (G1 – G4) and (b) highly Ge-doped photosensitive fibres (G5, G6)

Fibres for dosimetry

The radiation sensitivity of optical fibres originates from the formation of colour centres. In practice, the main effect of radiation is an additional loss (with respect to the intrinsic attenuation) in the irradiated fibre. This is called Radiation-Induced Attenuation (RIA) which can be a sensing principle for developing an optical fibre dosimeter. An ideal fibre dosimeter should show a high RIA response but also a negligible recovery effect (also called fading effect). In general, these two characteristics can not be achieved with commercially available optical fibres. Therefore, a key issue is to develop dedicated optical fibres that show weak recovery effects. This can be realised by optimising the doping concentration and the fibre geometry as well. Another complementary line of research is to implement a suitable data processing technique, based on a multiple wavelength analysis, which automatically reconstructs the applied dose and compensates for the recovery effect. Gamma irradiations have been performed on phosphorous doped SPPF fibre and Ge co-doped SPGePF, manufactured by FORC (Moscow), as a candidate for developing an optical fibre dosimeter. The dosimetry principle is based on the measurement of the radiation-induced attenuation in the optical fibre. This research indicated that a particular wavelength could be identified for a particular fibre composition at which the induced absorption grows linearly with dose and remains free of thermal recovery.

Experiments to compile a database including the spectra of radiation induced attenuation at various temperatures and dose rates for particular fibre types (including rare-earth doped fibres) are currently on-going.

In our recent work, we located an optimum RIA response region in the range of 1100 and 1400 nm depending on the temperature (see figure 24). We calculated the RIA sensitivity of the P-doped fibre and its variation along the dose.

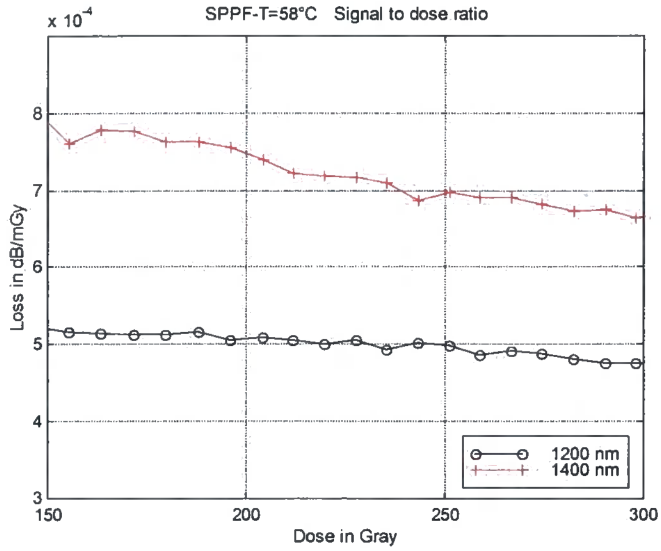


Figure 24: Evolution of the sensitivity ($dB \cdot m^{-1} \cdot Gy^{-1}$) as a function of the dose (2° irradiation) at 1200 nm and 1400 nm and T=58°C

We evidenced a hardening effect when the dose increases. This hardening effect is slightly more pronounced if the fibre is placed in an elevated temperature environment. Due to this hardening, specific cares should be taken with the linear model at high temperatures in terms of extrapolation of the dose. From a practical viewpoint, a standard 1310 nm wavelength may be used to reconstruct the dose at one wavelength. Within a 20 % precision, we may expect to reconstruct the dose by means of a second order polynomial (see figure 25). A linear polynomial could also be used, but only at a temperature lower than 30°C.

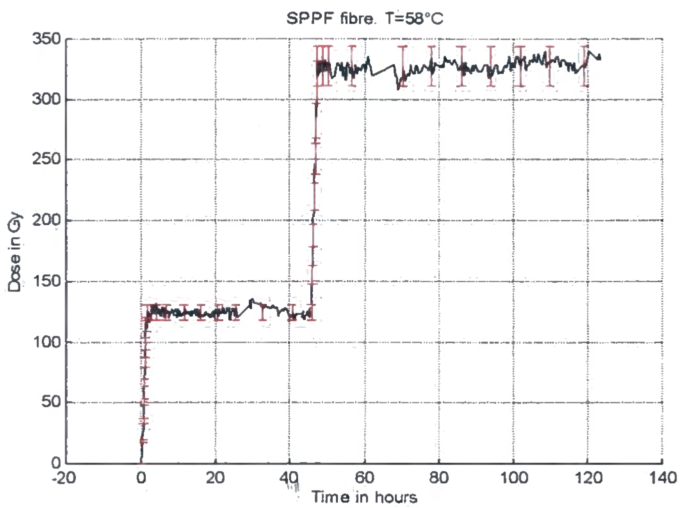


Figure 25: Dose reconstruction at 1300 nm with a second order model at 58°C. 10 % error bars on the estimated dose are also displayed

Conclusion and future work

Our recent work in the October 98- September 99 period has shown the potential for application of optical fibres around thermonuclear reactors. With specific care, they can be applied for a variety of purposes, e.g. data-communication with acceptable radiation induced loss and sensing with radiation hardened fibre-optic sensor designs. Our research has specific spin-offs to ITER relevant applications, including remote-handling and plasma diagnostics. Before fibre-optics can be fully qualified for operation in fusion reactor environments, additional studies are still necessary.

Future work will essentially be a continuation of the former work, with a more pronounced emphasis on fibre related components, which are critical in the overall degradation of the systems : sources and detectors, connectors, front-end optics. To obtain statistically relevant information, larger numbers of samples should be investigated. Former work on fibroscopy and on dedicated fibres for plasma diagnostics will be continued as well.

REFERENCES

[1] A.I. Gusarov, D.B. Doyle, F. Berghmans, "Analysis of photoinduced stress distribution in fiber Bragg gratings", accepted for publication in *Optics Letters*, 1999.

[2] A.I. Gusarov, F. Berghmans, O. Deparis, A. Fernandez-Fernandez, Y. Defosse, P. Mégret, M. Decréton and M. Blondel, "High total dose radiation effects on temperature sensing fibre Bragg gratings", *IEEE Photonics Technology Letters* **9**, pp. 1159-1161, 1999.

[3] B. Brichard, F. Berghmans, P. Borgermans, M Decréton, A.L.Tomashuk, I.V.Nikolin, R.R.Khrapko, K.M.Golant, " Dedicated optical fibres for dosimetry based on radiation-induced attenuation: experimental results and application ", Conference on Photonics for Space and Enhanced Radiation Environments, EOS/SPIE Symposium on Remote Sensing, 20-24 September 1999, Florence, Italy.

[4] F. Berghmans, M. Van Uffelen, A. Nowodzinski, B. Brichard, F. Vos, P. Jucker and M. Decréton, " High total dose irradiation experiments on fibre-optic components for fusion reactor environments", Conference on Photonics for Space and Enhanced Radiation Environments, EOS/SPIE Symposium on Remote Sensing, 20-24 September 1999, Florence, Italy.

[5] A.I. Gusarov, D.S.Starodubov, F. Berghmans, O. Deparis, Y.Defosse, A. Fernandez-Fernandez, M. Decréton, P. Mégret and M. Blondel, "Design of a radiation-hard optical fibre Bragg grating temperature sensor", Conference on Photonics for Space and Enhanced Radiation Environments, EOS/SPIE Symposium on Remote Sensing, 20-24 September 1999, Florence, Italy.

[6] A.I. Gusarov, D.S.Starodubov, F. Berghmans, O. Deparis, Y. Defosse, A. Fernandez-Fernandez, M. Decréton, P. Mégret and M. Blondel, " Behaviour of fibre Bragg gratings under high total dose gamma radiation", RADECS 99, 13-17 September 1999, Fontevraud, France.

[7] P. Borgermans, B.Brichard, F.Berghmans, M.Decréton, K.M.Golant, A.L. Thomashuk and I.V. Nikolin, "On-line gamma dosimetry with phosphorous and germanium co-doped optical fibres", RADECS 99, 13-17 September 1999, Fontevraud, France.

[8] M. Van Uffelen, A. Nowodzinski, Ph. Jucker, F. Berghmans, B. Brichard, F. Vos and M. Decréton, " Fibre-optic link components for maintenance tasks in thermonuclear fusion environments ", RADECS 99, 13-17 September 1999, Fontevraud, France.

[9] F. Berghmans, A. Gusarov, B. Brichard, M. Decréton, O. Deparis, P. Mégret, M. Blondel, I. Veretennicoff and H. Thienpont, " Photonics for nuclear industry : issues, problems and potential solutions ", *18th triennial Congress of the International Commission for Optics ICO XVIII*, San Francisco, USA, 2-6 August 1999, *SPIE Proceedings* **3749**, pp. 536-537, 1999.

[10] A.I. Gusarov, D.S.Starodubov, F. Berghmans, O. Deparis, Y. Defosse, A. Fernandez-Fernandez, M. Decréton, P. Mégret and M. Blondel, "Comparative study of MGy level γ-radiation effect on FBGs written in different fibres", *11th International Conference on Optical Fiber Sensors*, Kyongju, Korea, 12-16 April 1999, *SPIE Proceedings* **3746**, pp. 608-611, 1999.

[11]F. Berghmans, G. Verschaffelt, B. Brichard, S. Coenen, F. Vos, M. Decréton, H. Thienpont and I. Veretennicoff, "High total dose gamma radiation response of vertical-cavity surface-emitting lasers", OPTORAD 98, 10 December 1998, INSTN, Gif-sur-Yvette, France.

[12] B. Brichard, F. Vos, F. Berghmans and M. Decréton, " Comparison of the gamma radiation sensitivity of conventional and radiation hardened telecommunication fibres ", OPTORAD 98, 10 December 1998, INSTN, Gif-sur-Yvette, France.

[13] A.I. Gusarov, A. Fernandez-Fernandez, F. Berghmans, O. Deparis, Y. Defosse, P. Mégret, M. Décreton, M. Blondel, " Temperature monitoring with fibre Bragg grating sensors for MGy dose level gamma-radiation environments ", OPTORAD 98, 10 December 1998, INSTN, Gif-sur-Yvette, France.

3.2. Vessel/In-Vessel: Plasma Facing Components and Blanket

3.2.1. Fracture Behaviour of Beryllium

PRINCIPAL INVESTIGATOR: R. Chaouadi

SCIENTIFIC STAFF: A. Leenaerts, J.-L. Puzzolante and M. Scibetta

OBJECTIVES

Experimental data on fracture toughness behaviour of beryllium are quite limited, especially in the irradiated condition. This is not only due to the high costs of such tests but also to the inherent toxicity of beryllium, limiting therefore the laboratories qualified to perform such tests. On the other hand, tensile tests are easy to perform and widely used for investigating the radiation effects on materials. The main objective of this work is to investigate the correlation between tensile properties and fracture toughness.

Background

Within the European Long Term Fusion Technology Programme related to the ITER blanket design, a task T23 was performed at SCK•CEN to generate tensile and fracture toughness data on four ITER grade Beryllium alloys. Specimens were irradiated at the BR2 Materials Testing Reactor in various temperature and neutron fluence conditions. Tests were performed in a large temperature range covering brittle as well as ductile fracture. This data set offers thus a good opportunity to investigate the correlation between tensile properties and fracture toughness.

ACHIEVEMENTS

An extensive effort was put into re-evaluating the fracture toughness test results. Indeed, the data of the previous evaluation were underestimating the actual fracture toughness. Finite element calculations support the updated data. As no major difference was found between the various grades, the modelling work was concentrated on the S65 VHP grade material. Indeed, this grade was selected as the reference material in ITER for plasma facing component and fist wall applications.

The S65 VHP compact tension specimens were systematically investigated with scanning electron microscopy (SEM). In the baseline condition, two fracture mechanisms were identified: cleavage for test temperatures below 300°C and ductile above this temperature. In the irradiated condition, all samples fail in a brittle manner, with tendency to intergranular fracture for specimens irradiated at 600°C.

Two modelling approaches are investigated:

- semi-empirical modelling: here, analytical relations allow to express fracture toughness as a function of parameters derived from the tensile test:

  - model of Conrad and co-workers (1973) for cleavage fracture,
  - model of Hahn and Rosenfield (1968) for ductile fracture.

- micromechanical modelling of fracture: here, the micromechanisms of fracture are taken into account in association with the stress and strain history during the whole loading process. Finite element calculations are then required to determine the local stress and strain history.

  - Ritchie-Knott-Rice (1968) RKR model for cleavage fracture: fracture occurs when the maximum principal stress reaches a critical microcleavage stress value over a critical distance ahead of the crack tip.
  - Rice and Tracey (1969) model for ductile fracture: fracture initiation occurs when the void growth rate reaches a critical void growth rate value over a critical distance ahead of the crack tip.

In all these models the characteristic distance plays an important role.

The application of the semi-empirical models to our data are summarised in figure 26 which compares predicted fracture toughness values to predicted ones. For cleavage fracture, the characteristic distance was taken equal to 7µm and 70µm for HIP and VHP grades, respectively. The Conrad et al. model considers the characteristic distance as a fitting parameter. Within the experimental uncertainties, the agreement is fairly good. For ductile fracture, the characteristic distance according to Hahn and Rosenfield is proportional to the strain-hardening exponent. Figure 26 shows that this model underestimates the actual fracture toughness although a quasi-linear correlation between predicted and actual fracture toughness is found.

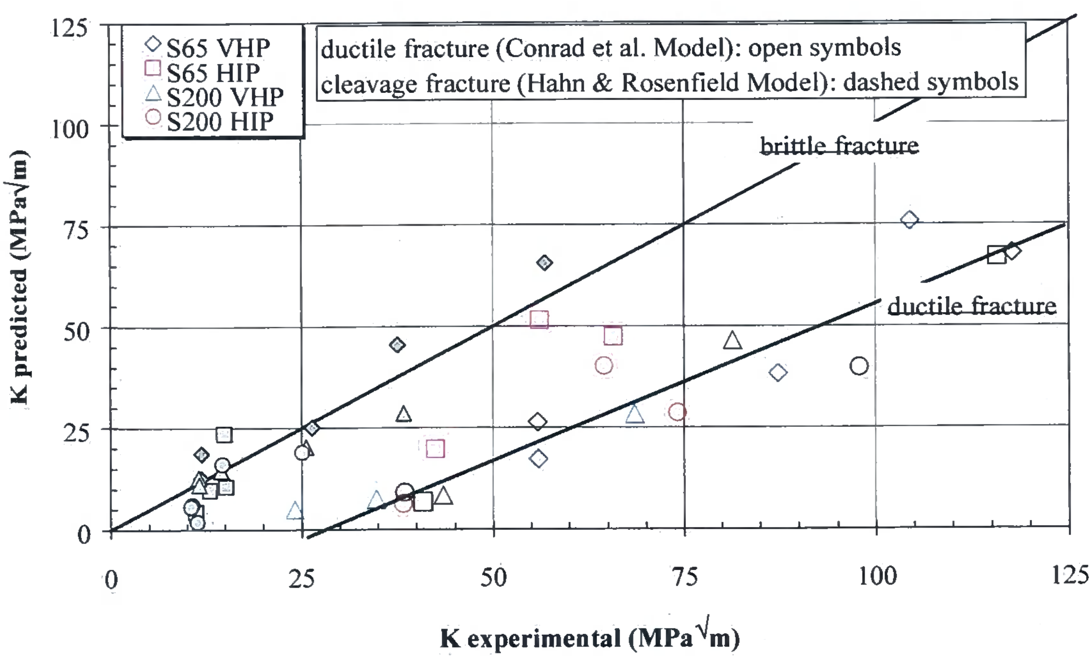


Figure 26: Prediction of fracture toughness from tensile test parameters using semi-empirical models

Finite element calculation were performed on 6 specimens of S65 VHP material tested at 20, 230, 310, 455, 540 and 605°C. Specimens tested at 20, 230 and 310°C exhibited a brittle fracture behaviour, typically cleavage while the three specimens tested at higher temperatures show a ductile fracture behaviour.

In order to apply the RKR cleavage model, two parameters are required: the characteristic distance of course and the microcleavage fracture stress. Unfortunately, none of these parameters is known. In particular, the fracture stress determined from the tensile tests cannot be used as the stress and strain state are not representative of the one present ahead of a crack tip. The fracture initiation site on the compact tension specimens should be determined in order to estimate through finite element calculations the microcleavage stress. This stress should be characteristic of the material and independent from test temperature. This task is in progress.

For ductile fracture, two parameters are also required: the critical void growth rate and again the characteristic distance. The critical void growth was approximately evaluated from the tensile tests. Preliminary analysis on the damage distribution at fracture initiation is shown in figure 27. The characteristic distance lies between 120 and 220 µm. An in-depth evaluation is in progress to better evaluate this model, in particular the sensitivity study of the various parameters.

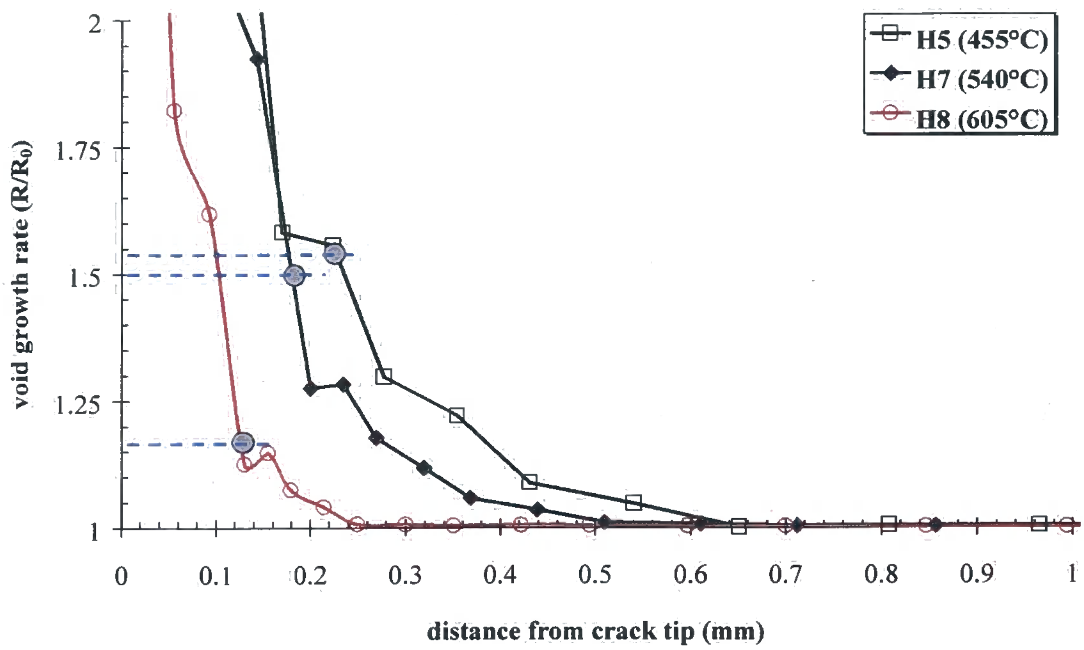


Figure 27: Damage distribution ahead of the crack tip.
The characteristic distance lies in the range of 120 to 200 µm

REFERENCES

[1] R. Chaouadi, M. Scibetta and A. Leenaerts, "Micromechanical Modeling of Fracture Initiation of Beryllium", presented at the 4th International Workshop on Beryllium Technology for Fusion, Kalsruhe, 15-17 September, 1999.

[2] R. Chaouadi, A. Leenaerts, J.L. Puzzolante and M. Scibetta, "Radiation Effects on the Mechanical Properties of Beryllium", presented at the 4th International Workshop on Beryllium Technology for Fusion, Kalsruhe, 15-17 September, 1999.

3.2.2. Crack-initiation and crack-growth detection on-line methods

PRINCIPAL INVESTIGATOR: R. Van Nieuwenhove

OBJECTIVES

Irradiation Assisted Stress Corrosion Cracking (IASCC) continues to cause failures in fission reactors and a world wide effort is presently going on to better characterise this type of failure in Boiling and Pressurised water reactors. Since the first wall if ITER will be cooled by water at high temperature (between 100 °C and 200 °C) at high pressure, IASCC has also been considered as the most important issue for cooling channels made out of austenitic stainless steel (such as AISI 316 LN-IG), because of the poor understanding of IASCC and of the difficulty to validate the corrosion resistance for actual components. In order to assess IASCC under fusion relevant parameters, special instruments have to be developed for the online monitoring of crack-initiation and crack-propagation.
In the previous reporting period (1998), initial qualification tests of electrochemical noise measurements have been performed. Possible indications of crack initiation processes on AISI 316 steel have been observed but no conclusive demonstration could be obtained. From these preliminary tests, a number of possible improvements were however identified and are the subject of this report.

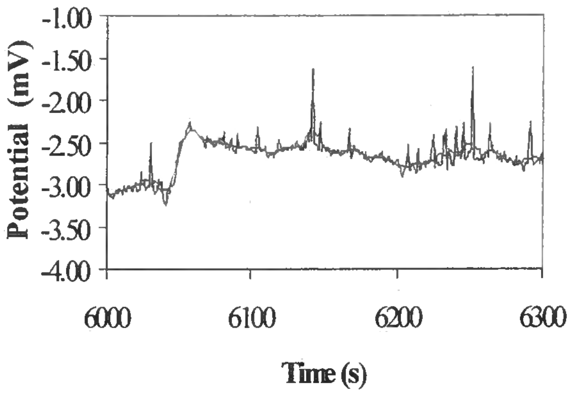
ACHIEVEMENTS

Electrochemical Noise Measurements
By measuring the spontaneous current or voltage fluctuations (noise) between a reference electrode and a test sample, information can be gained about processes at the surface of the electrode. In the first design (reporting period '98), the reference electrode or noise sensor consisted of a thin Pt perforated foil surrounding the pressure tube (test sample). Despite the fact that a lot of measurements have been made with this type of sensor, transient fast peaks (possible indication of crack initiation) on the electrochemical potential noise were only rarely observed and difficult to discriminate from the general noise level. Because of relatively poor results of the foil-type sensor, measurements with a wire type sensor were made. This sensor consisted of two Pt wires at two diametrically opposite positions of the pressure tube, connected to each other and insulated from the pressure tube.

The motivation for this design was based on the following considerations: i) If the surface of the sampled tube is too large potential spikes originating from different regions of the tube surface could overlap and thereby become indiscernible and ii) The high capacitance of the double layer (for a large surface) could possibly filter out fast transient potential spike. In the experiment, the pressure tube and sensor (from sensitised AISI 304 stainless steel) were subjected to a Pressurised Water Reactor environment, as simulated in a recirculating high pressure water loop.

The pressure inside the tube was gradually increased above the yield strength of the material, thereby creating small microcracks on the surface of the tube wall.

These crack-initiation events could now clearly be observed as sharp spikes on the measured potential (see figure 28) and after the experiment they were also observed on pictures taken with an electron microscope (see figure 29).



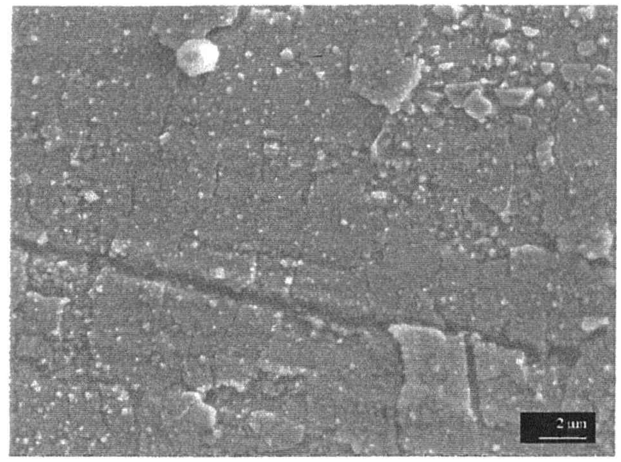


Figure 28: The creation of small micro-cracks shows up as sharp potential spikes on the electrochemical noise sensor

Figure 29: Electron microscope picture revealing surface cracks, with some side branches

Besides these autoclave tests, an irradiation experiment called CORONA (Corrosion Online Assembly) has been designed in which two pressure tubes with electrochemical noise sensors (improved design) are located. This experiment is now in the fabrication stage, after having obtained official approval (second phase) by a reactor safety committee. The out-of-pile pressurisation system and the data-acquisition system have already been completed.

Acoustic emission measurements

Whenever a material cracks, mechanical elastic energy is released in the form of a transient high frequency acoustic pulse (acoustic emission). Acoustic Emission (AE) detection is, besides electrochemical noise, a powerful tool for the on-line detection of cracks. In the previous reporting period ('98) an extensive literature study has been performed to identify sensors (crystals) which could be employed in the harsh conditions of a reactor environment. The most promising candidate was found to be strontium niobate ($SrNb_2O_7$). Although such a material was fabricated, it turned out to be very difficult to electrically 'pole' this material, to make it suited as acoustic sensor. Therefore, a different approach has been taken : The acoustic signal is guided inside a metal bar (waveguide) towards a region outside the harsh, high temperature environment, where commercially available sensors can perform the detection. First tests in a static autoclave, equipped with a slow strain rate test unit, look very promising but more work is needed to reduce the impact of electromagnetic interference.

Conclusions and foreseen future work

Both electrochemical noise (EN) and acoustic emission can be used for the online detection of crack initiation under fusion relevant conditions. In the near future, a further validation of the EN technique is foreseen while increasing its accuracy and time resolution. For the acoustic emission technique, more work is needed to reduce the impact of external electrical noise. Once this technique is working satisfactory, it will be combined (in one experiment) with the EN technique.

References

[1] R. Van Nieuwenhove, *Electrochemical Noise Measurements under PWR conditions,* Third IAEA WACOL meeting (High Temperature On-line Monitoring of Water Chemistry and Corrosion), 6-9 June 1999.

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