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# FUSION REACTOR MATERIALS

## Introduction

Fusion Research in Europe is one of the largest co-ordinated programmes of the European Commission. SCK•CEN focuses its fusion participation on material characterisation under radiation and coolant interaction: *structural material* for the vessel modules, as well as material used for the diagnostics and maintenance *instrumentation*. The latter aspect is presented in the contribution on "Instrumentation" of the present report. The present contribution focuses on radiation induced degradation characteristics of structural materials, but involves also the studies related to safety issues and waste management aspects related to these materials.

## Objective

To contribute to the knowledge on the behaviour, during and after irradiation, of the fusion-reactor materials and components.

## Programme

- Study of the mechanical behaviour of structural materials under neutron irradiation: Reduced Activation Ferritic-Martensitic (RAFM) steels, inconel, chromium, etc. This involves tensile and compact tension tests, microstructural evaluation and corrosion measurement;
- Characteristics of irradiated first wall material such as beryllium, as for instance its reactivity with air and steam.
- Detection of abrupt electrical degradation of insulating ceramics under high temperature and neutron radiation;
- Study of dismantling and waste disposal strategy for fusion reactors.

## Achievement

### *Radiation induced mechanical degradation of RAFM steels*

RAFM (Reduced Activation Ferritic Martensitic) steels represent the most promising option in terms of structural materials, both for fusion reactors and accelerator driven systems (ADS). Presently, their nominal composition envisages Cr contents between 7% and 10%. The most up-to-date RAFM steel nowadays investigated is EUROFER97. Several organisations are studying its mechanical properties both in the unirradiated and irradiated condition.

In 2000, a preliminary irradiation (code name IRFUMA) of EUROFER97 has taken place at 300 °C in BR2, leading to an accumulated dose of about 0.27 dpa. Tensile, Charpy impact and Charpy toughness specimens were loaded in IRFUMA, with the aim of providing a complete characterisation of the strength and toughness properties of the steel via the post irradiation examination (PIE) envisaged in 2001.

A parallel activity is in progress investigating the possible improvements of the EUROFER97 mechanical properties induced by the Oxide Dispersion Strengthening (ODS) technique. Two preliminary batches of EUROFER97 steel, produced via the Hot Isostatic Pressure (HIP) route and with two different contents of $Y_2O_3$ (3% and 5%), have been characterised by means of tensile and sub-size instrumented Charpy tests for two different orientations. We performed the tests in collaboration with the Paul Scherrer Institute (PSI-CRPP, Switzerland); comparison with SCK•CEN data will be performed in 2001.

### *Corrosion behaviour of RAFM steels*

Structural materials for fusion reactors will be subject to corrosion in contact with the coolant, circulating in the blanket. The use of new materials, such as RAFM steels, present a challenge in the particular fusion conditions, as very few experimental data are at present available to evaluate their performance. The influence of irradiation on the corrosion behaviour of these materials represents in particular a major unknown. The study aims at evaluating the electrochemical behaviour of RAFM in high temperature water, with special emphasis on the occurrence of stress corrosion cracking (SCC) on fresh materials and irradiated samples. Also, attention is put on the influence of an intense radiation field on this electrochemical behaviour, involving the influence of radiolysis and direct flux effects, such as the photo-electric effect on corrosion.

*Electrochemical behaviour*: We performed electrochemical tests like potentiodynamic polarisation curve measurements, corrosion potential measurements and Electrochemical Impedance Spectroscopy (EIS) in demineralised water at 100°C and 230°C, with and without oxygen. The experimental results showed that the shape of the polarisation curves changed with temperature, but not with oxygen concentration. With oxygen however the polarisation curves moved slightly to a more anodic region. EIS

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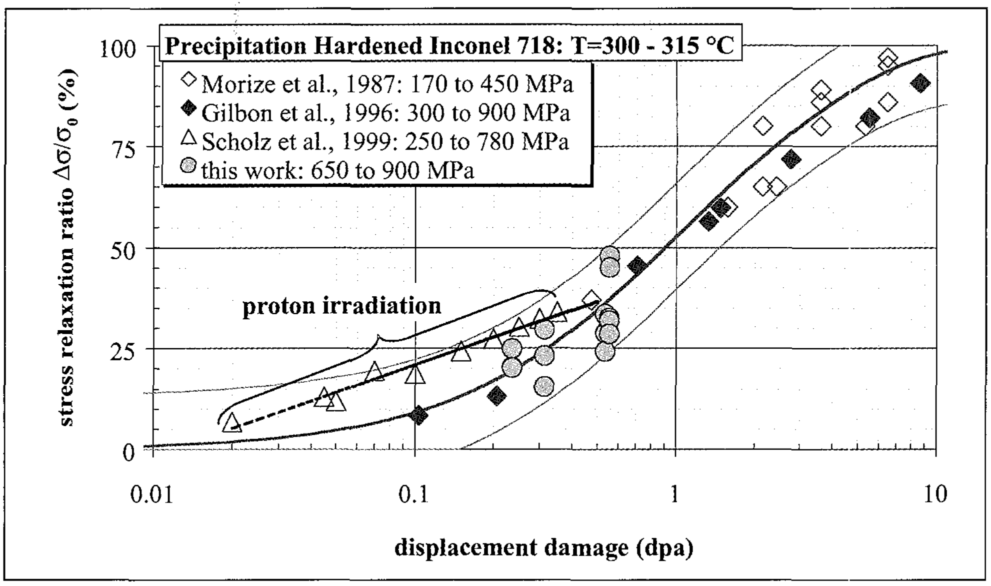
results were used to calculate (uniform) corrosion rate. We measured the highest corrosion rate at a temperature of 230°C with addition of oxygen. The most likely source of oxygen under operational conditions of a thermonuclear fusion reactor is the formation of hydrogen peroxide due to radiolysis of water. We modified therefore the water circulation loop of our corrosion laboratory with a micro pump to inject hydrogen peroxide and an electrochemical sensor to measure the hydrogen peroxide concentration (at the cold end of the loop). First results showed that almost all the injected hydrogen peroxide was decomposed to oxygen in the high temperature part of the loop (autoclave) and that the corrosion potential was increased due to the injection of hydrogen peroxide. Sometimes an initial decrease of the corrosion potential was observed after injection of hydrogen peroxide. After this initial decrease the corrosion potential increased to a more anodic value than before the addition of hydrogen peroxide. We will perform further experiments to investigate the influence of hydrogen injection on the electrochemical corrosion behaviour of RAFM steels using polarisation resistance and corrosion potential measurements.

*Stress Corrosion Cracking*: We selected two ferritic martensitic steels for this study: the EUROFER97, specially developed for the fusion programme, and the BI56 steel, with a higher chromium content (12% instead of 8%) and a higher strength and corrosion resistance. We used samples as delivered: i.e. quenched and tempered. Tests occur in pure, deaireated water, with additions of oxygen, hydrogen or hydrogen peroxide to simulate different possible operation regimes of the cooling systems. Temperatures range from 100°C to 230°C, representative of the expected operating regime of cooling circuit. We evaluated the SCC susceptibility by slow strain rate tensile testing (SSRT) in an autoclave. The strain rate was set to $10^{-6}$/s in all tests. We also performed reference tests in helium to estimate the intrinsic properties of the material at the tests temperature and strain rate. In both alloys, there is a significant reduction in total elongation when the SSRT tests are conducted in oxygenated water. Also, the fracture surface of materials exhibits a significant amount of stress corrosion cracks after testing in oxygenated water. The mechanism of SCC is however different in both alloys: in the EUROFER97, SCC initiates from pits, generated from localised corrosion. The tendency for SCC is higher at 100°C than at 230°C in EUROFER97. In the BI56 steel, there is a transition in SCC mechanism at about 160°C: at low temperature, SCC initiates in a transgranular mode, but propagates in an intergranular way (former austenitic grain boundaries); at high temperature, the intergranular mechanism is suppressed and the SCC propagation is entirely transgranular. The appearance of the fracture surface suggests a hydrogen-embrittlement-assisted SCC mechanism in the transgranular case. The BI56 steel is more sensitive to SCC, which corresponds to its higher yield stress, and this increases the sensitivity of the material to hydrogen-embrittlement-assisted SCC. We consider also ODS variants of EUROFER97 in the future, and extend the study to samples which have been irradiated up to 0.25 dpa in the IRFUMA experiment. As is suggested by the differences in SCC susceptibility in steels with different strength levels, the irradiation hardening may increase the susceptibility of the material to SCC. Further tests will involve samples irradiated up to 2 dpa.

### Inconel: Irradiation Creep Inducing Stress Relaxation

The ITER shield modules are to be 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. It is important to have reliable experimental data on radiation-induced stress relaxation of this material at 300°C to a neutron exposure of 0.5 dpa. We selected a pressurised thin-walled tube geometry as the most adequate sample geometry. The tubes are first laser-welded to end plugs and then filled with argon at liquid nitrogen temperature. We selected the volume of argon in such a way that the desired pressure is reached at the operation temperature of 300°C. The stress levels are 700 to 900 MPa. Specimens exposed to the same conditions, but without neutron exposure, show no creep deformation. On the other hand, the neutron-exposed specimens exhibit an actual creep deformation. The creep data are converted into stress relaxation using well-established equations. The figure below shows the stress relaxation ratio as a function of the displacement damage. Additional neutron irradiation data found in literature are included for comparison. As you can see , there is a very good agreement between the various data sets. As expected, no stress level effect is found. Results from proton irradiation are also found within the 95% confidence bounds of the neutron irradiation data.

According to the design conditions, the bolts for flexible cartridge and primary wall fastening necessitate residual stresses at the end of life of about 670



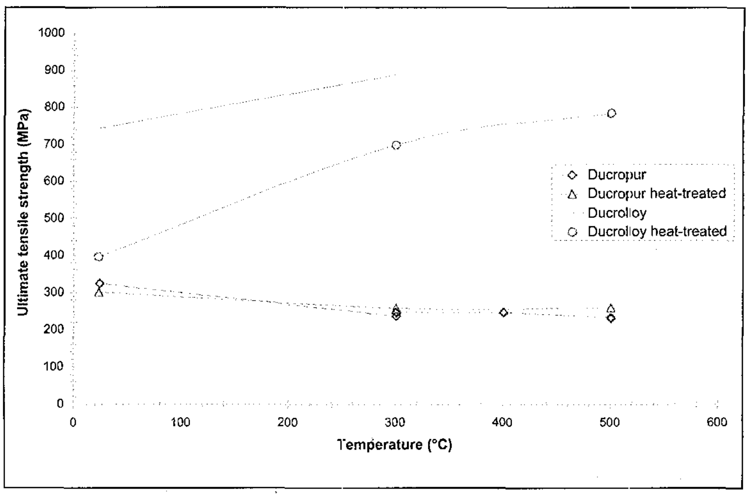

***Comparison between neutron and proton irradiation inducing stress relaxation of Inconel 718, as you can see, there is a very good agreement between the various data sets.***

MPa and 400 MPa, respectively. The estimated neutron dose varies between 0.04 and 0.3 dpa for the flexible cartridge fastening and between 0.65 and 3 dpa for the primary wall fastening. In both cases, the operating/irradiation temperature is about 250°C. Calculation of the maximum doses under these conditions give, respectively, 0.47 and 1.31 dpa for the flexible cartridge and primary wall fastenings. This clearly indicates the importance of an accurately definition of the actual operating conditions, if a reliable evaluation of the material performance is to be achieved.

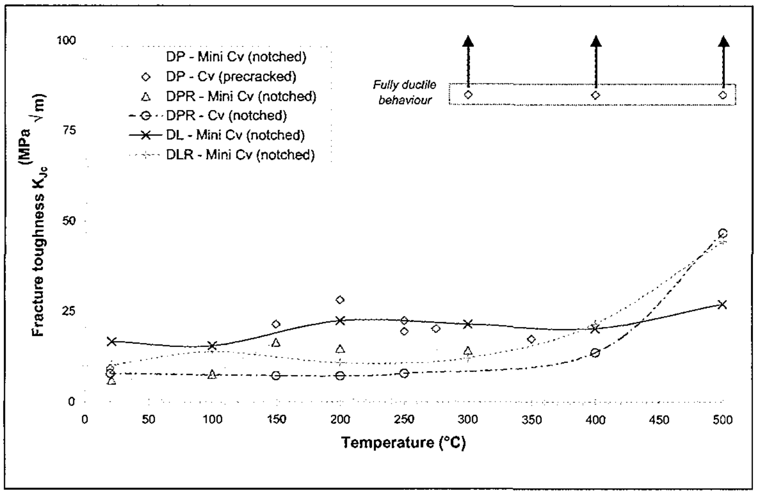
### *Chromium: Characterisation under neutron radiation*

The mechanical characterisation of two chromium alloys (DUCROPUR and DUCROLLOY), both in the as-received (HIPped) and heat-treated states, has been extensively analysed and reported, as well as presented in two international conferences. The ultimate aim is exploiting their favourable characteristics (low-activation, high corrosion resistance, elevated resistance via the ODS technique) and possibly improving their weakest aspects (scarce ductility and toughness below 300 °C, difficulties in joining and forming, possible incompatibility with Li and Be). A new set of tensile specimens of the two alloys in both states were irradiated in the BR2 reactor at 300 °C up to an accumulated fluence of approximately 0.27 dpa (IRFUMA experiment). We will characterise these samples in order to complete the tensile characterisation in the irradiated state. As far as the unirradiated state is concerned, the characterisation of the mechanical properties (tensile, impact, toughness) of the two alloys in the unirradiated state has been completed. Results have clearly shown that the addition of $Y_2O_3$ to quasi-pure Chromium significantly improves the tensile properties but at the same time dramatically degrades fracture toughness properties (see the two figures below).

Continuation of these activities in the following years will strongly depend on the availability of a new Cr alloy with improved ductility.



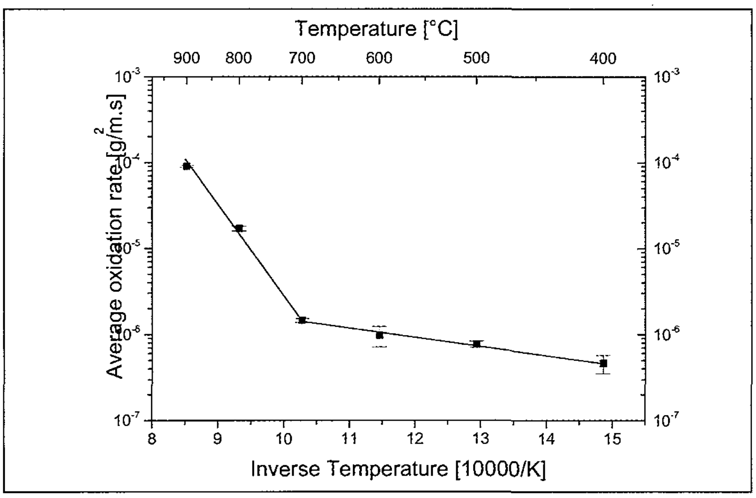

***Tensile properties of the Chromium alloys (Ducropur: 99.96% pure Cr; Ducrolloy: Cr with 5%Fe and 1% $Y_2O_3$), clearly showing that the addition of $Y_2O_3$ to quasi-pure Chromium significantly improves the tensile properties.***




*Fracture toughness properties of the Chromium alloys (DP,DL = Ducropur,Ducrolloy; DPR,DLR = DP, DL heat-treated), clearly showing that the addition of $Y_2O_3$ to quasi-pure Chromium dramatically degrades fracture toughness properties*

## Beryllium: Interaction of pebbles with air and steam

European Helium-Cooled Pebble Bed Blanket (HCPB): The HCPB design foresees beryllium as neutron multiplier in the form of a pebble bed. For the safety assessment of the HCPB design, in the event of accidental contact of pebbles with air and steam, kinetic data are needed to describe the extent of the occurring exothermic reaction. For this purpose, thermogravimetric (TG) experiments with the 1-mm pebbles currently chosen as the reference HCPB material were performed in the Waste & Disposal Department. These pebbles, produced by the rotating electrode method, are nearly spherical in shape and possess low specific surface area and porosity. The figure shows the average oxidation rates as a function of temperature for these pebbles when exposed to air between 400°C and 900°C. One observes a parabolic oxidation kinetics up to 700°C. At 800°C and above, an accelerating behaviour is observed, which is associated with the loss of protective properties of the oxide film. Fitting of the TG curves yielded rate constants of the chemical reaction in air.




***Average oxidation rates for REM manufactured 1-mm beryllium pebbles exposed to air at temperatures between 400°C and 900°C. Rates are based on the specific surface area and the total mass increase after 340 minutes of exposure to air.***

*Doping and coating pebbles in ITER:* The foreseen ITER ceramic breeding blanket uses a beryllium pebble bed as neutron multiplier and water as coolant, leading to the same safety concern as for HCPB. There exist experimental indications that the reactivity of the pebbles in steam might be reduced by doping or coating the beryllium material. The temperature range concerned during normal operation of ITER and in the case of an in-box loss-of-coolant accident resulting in contact between the beryllium pebbles and steam is below 500°C. Literature was surveyed, focusing on mitigation strategies for the chemical reactivity of beryllium in a water/steam environment. Three approaches can be distinguished: adding alloying elements, coating the surface, and modifying directly the oxide film by surface engineering. The most promising techniques, not taking into account the effects on activation and mechanical properties, should be calcium-doping and pre-oxidising the beryllium pebbles in order to thicken the oxide layer.

*Chemical reactivity of dust and flakes and deuterium mobilisation experiments from co-deposited layers:* During the operation of a fusion reactor, plasma disruptions produce dust that accumulates in the vacuum vessel. The reaction of this dust with steam raises also safety concerns. It is therefore imperative to get sufficient knowledge on the chemical reactivity of such dust. We prepare now coupled thermogravimetric analysis and mass spectrometry to investigate the chemical reactivity of dust and flakes.

## Insulation ceramics: radiation induced electrical degradation effect (RIED)

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. Electrical characteristics of most insulators undergo a two-step degradation process. After a slow decrease in insulating resistance, we observed a sharp breakdown (RIED) when a given dpa damage is reached. Such a sudden degradation must be avoided during 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 work, carried out by the Reactor Experiments and Instrumentation Departments of SCK•CEN, is to perform a neutron benchmark experiment to obtain more reliable values of the RIED threshold, under representative fusion conditions. A rig has been specially designed to work under vacuum, and the BR2 reactor will be operated at reduced power (5%). Irradiation conditions will be as follow: high vacuum, a neutron flux of $10^{+12}$ n/cm$^2$.s (E>0.1MeV), a gamma heating of 100 Gy/s, temperatures between 350°C and 450°C. The rig is mainly composed of modules, centred in a pressure tube to be introduced in the reactor and connected to a control panel for vacuum circuit, temperature regulation, measurements and data acquisition. Electrical resistance measurement is particularly delicate, due to the very high resistivity of the ceramic samples and the high voltages involved. The rig is designed to be reloadable, using new modules and new samples, for future similar tests. During 2000, the irradiation rig and control panels were made operational. Functional tests showed a high contamination of the ceramic samples by out-of-specification elements, becoming volatile under vacuum and high temperature. We have this problem now under control and we foresee the irradiation during the first half of 2001.

### Beryllium: Waste conditioning strategies

Future fusion reactors are expected to generate large quantities of irradiated beryllium. If not recycled, this beryllium will need to be conditioned and disposed of. For that purpose, we initiated a study in the Waste & Disposal Department to identify suitable conditioning strategies. First, we examined the chemical and radiochemical properties of existing beryllium waste emanating from a fission reactor, such as BR2. We carried out sampling, in order to obtain representative beryllium samples for low, average, and high neutron fluence. This allows modelling of the neutron activation of the existing beryllium and extrapolating the data to beryllium irradiated under fusion reactor conditions. On the basis of these results, and based on an on-going survey of possible conditioning processes for irradiated beryllium, we will analyse the technical feasibility of a suitable conditioning process.

### Waste strategy: dismantling options

It is important at the design phase of a new nuclear installation, to take into account the waste and dismantling issues. This is also true for future fusion reactors. SCK•CEN has a thorough know-how in this domain, through the dismantling of its PWR fission reactor BR3, carried out by the Site Restoration Department. The dismantling of a PWR reactor contains of course specific aspects not directly relevant to fusion installations, but sufficient common generic features are present, so that the lessons learned are of primary use for the designers of the future fusion machines. One aspect for instance is the debate between immediate and deferred dismantling. For fission reactors, we showed that deferred dismantling by, let us say, 65 years, was not sufficient to give a significant radiological gain, especially for handling the hot parts. The same techniques are anyway to be applied. Another example is the comparison between underwater remote dismantling and dry hot-cell environment. This showed that the underwater alternative brings important advantages. Other useful lessons are related to the choice of cutting techniques, water and air purification techniques, the need for specific auxiliary waste management infrastructures, the important issue of recycling highly activated materials, the impact of an accurate waste characterisation approach, even for low level waste, on the waste management cost, the need for a good modelisation of the dose exposure to insure an efficient ALARA approach, etc. Material choice (purity level has a critical impact on final activation levels) should already reflect dismantaling constrains and the remote maintenance handling unit should be designed also for final dismantling

### Waste storage: Analyses of human intrusion scenarios into a fusion waste disposal site

To illustrate the short period of concern of fusion waste in comparison with fission waste, we made comparisons of the results of assessments of the radiological consequences of human intrusion into a fusion waste repository with results of similar analyses carried out for fission waste. This allows also identifying the relative contributions of materials impurities on the intrusion doses. The Waste & Disposal Department carries out this work. Drastic geological examinations of a core digged up at the repository site are often taken as reference intrusion scenario. However, recent performance assessments of fission waste repositories focus now more the attention on other ones, such as a borehole drilling through the repository. We analysed such a scenario, and compared the results with fission relevant equivalent data. In the case of the core examination scenario the highest dose to a geological worker is calculated for the Be coating of the first wall of the fusion reactor. A somewhat surprising result is that, also for the fusion waste, actinides, such as Am-241 and Pu-239, are the main contributors to the intrusion doses. The U impurity in the Be is only about 32 ppm, but the very hard fusion neutron spectrum appears to convert a considerable fraction of this uranium into higher actinides. A comparison with intrusion doses calculated for a fission spent fuel repository shows dose values are at least 5000 times lower for fusion, even when the most active fusion waste is taken into account. The dose resulting from a close inspection of a core containing spent fission fuel drops under 0.5 Sv after 1 million years, whereas this dose level, under which no deterministic health effects are expected, is never reached in the case of fusion waste. We eleborated now a preliminary description of the borehole-drilling scenario and we discussed values of the related parameters with borehole-drilling experts. In the case of a plastic clay layer, such as the Boom Clay, which is the reference host layer for geological disposal of high-level and medium-level fission waste in Belgium, one can expect that a borehole without lining will get closed after some time by the convergence of the clay. Geomechanical analysis shows that such a sealing of the borehole takes only a few months. This short contact time between disposed waste and groundwater considerably limits the consequences of such borehole scenario. The results will now be compared with the data coming from a similar on-going analysis carried out by for fission waste repository in Belgium. Interactions with the fusion reactor development and the materials research programmes are foreseen to evaluate the impact of design modifications and further purification of candidate materials on the possible long-term consequences of the resulting radioactive waste.

## Partners

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| - | PLANSEE (Plansee, Austria). |
| CEA | Commissariat à l'Energie Nucléaire (Saclay, France) |
| CIEMAT | Centro de Investigaciones Energéticas, Medioambientales y Tecnológicas (Madrid, Spain) |
| CRPP | Centre de Recherche sur la Physique des Plasmas (Lausanne, Switzerland) |
| EFDA-CSU | European Fusion Devopment Agreement - Close Support Unit (Garching, Germany) |
| ENEA | Ente per le Nuove Tecnologie, Energia e l'Ambiente (Brasimone, Italy) |
| ESI | Erich Schmid Institute of Material Science (Leoben, Austria) |
| FZK | Forschungszentrum Karlsruhe (Karlsruhe, Germany) |
| NFR | Nuclear Fusion Research (Stockholm, Sweden) |
| NRG | Nuclear Research Group (Petten, the Netherlands) |
| OEAW | Austrian Academy of Science (Vienna, Austria) |
| RISOE | Risoe National Laboratory (Risoe, Denmark) |
| ULB | Université Libre de Bruxelles - Physique Statistique et Plasmas (Brussels, Belgium) |
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## Sponsor

| | |
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| EC | European Commission through its European Fusion Development Agreement (EFDA) and its Underlying Technology Programme (Brussels, Belgium) |

## Scientific output

### Publications

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X. Sillen, J. Marivoet, M. Zucchetti, "Analysis of human intrusion scenarios for the deep disposal of fusion wastes", IAEA Technical Committee Meeting on Fusion Safety, Cannes, June 13–16, 2000.

E. Lucon, E. van Walle, M. Decréton, "Mechanical Properties of Two Chromium Alloys in As-Received and Heat-Treated Conditions", 21st SOFT (Symposium on Fusion Technology), Madrid, September 11-15, 2000 (to be published in Fusion Technology).

F. Druyts, J. Fays, P. Van Iseghem, "Chemical reactivity of beryllium pebbles in air", 21st SOFT (Symposium on Fusion Technology), Madrid, September 11-15, 2000 (to be published in Fusion Technology).

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