Marc Decréton

Fusion Reactor Materials

Introduction

The research programme aiming at a prototype fusion reactor is presently the largest co-ordinated R&D action of the European Commission. The development, however, of what could become a major source of electricity production in the future, is still a challenge for a certain number of technological issues. The selection of suitable materials, a reliable instrumentation, adequate safety and environmental procedures, are examples of domains where research and development is on-going, world-wide. Based on its long-term experience and its important available infrastructure, SCK•CEN has established its particular niche in the fusion community, around the study of radiation effects on materials. It covers of course the metal alloys used for the reactor structures: an activity summarised in this actual chapter. However, SCK•CEN's fusion research involves also the assessment under radiation of diagnostics and maintenance instrumentation. This latter aspect appears in the chapter on "Instrumentation" of the present report. Two other fusion studies are also mentioned in the contribution on "Social Sciences". A comprehensive summary of all fusion activities is moreover available in the SCK•CEN fusion annual report, as referenced below.

Objectives

Our activities on fusion materials aim:

- at contributing to the knowledge on the radiation-induced behaviour of fusion-reactor materials and components;
- at helping the international community in building the scientific and technical bases needed for the construction of the future reactor.

Programme

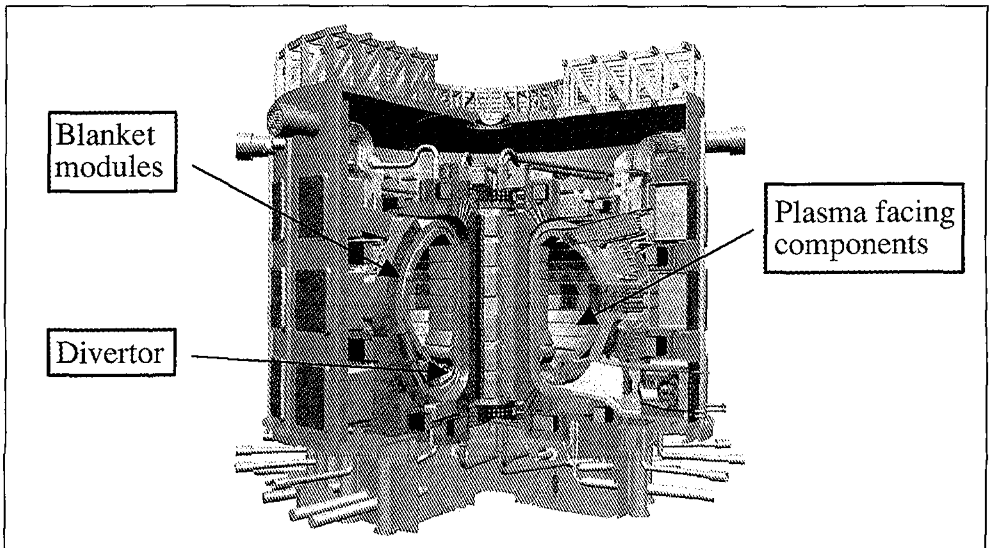
We look for suitable materials capable to withstand the extreme temperature and radiation conditions expected on the first wall inside the torus, on the blanket modules (cooling circuits and tritium breeding systems) and on the plasma diverting structures (see figure). These materials are tested in representative fusion conditions, and their characteristics analysed.

In particular, it involves:

- assessing the mechanical and chemical (corrosion) behaviour of *structural materials* under neutron irradiation and water coolant environment. The main focus is put on the new steel selected for the fusion reactor: Eurofer97, a reduced-activation ferritic-martensitic (RAFM) steel. Other studies concern chromium, copper and titanium;
- characterising irradiated *first wall material*, such as beryllium. We look for surface reactivity as a safety issue, and radiation-induced mechanical behaviour due to helium-induced swelling;
- studying how these materials must be managed during dismantling and disposed as waste.

Scientific staff
Rik-Wouter Bosch, Aimé Bruggeman, Rachid Chaouadi, Marc Decréton, Frank Druyts, Ludo Eysermans, Jérémie Fays, Patrice Jacquet, Michel Klein, Dario Lodi, Enrico Lucon, Lorenzo Malerba, Jan Marivoet, Vincent Massaut, Léon Put, Leo Sannen, Marc Scibetta, Steven Smets, Bart Smolders, Jef Valenbergh, Steven Van Dyck, Pierre Van Iseghem, Marc Vankeerberghen, Marc Verwerft

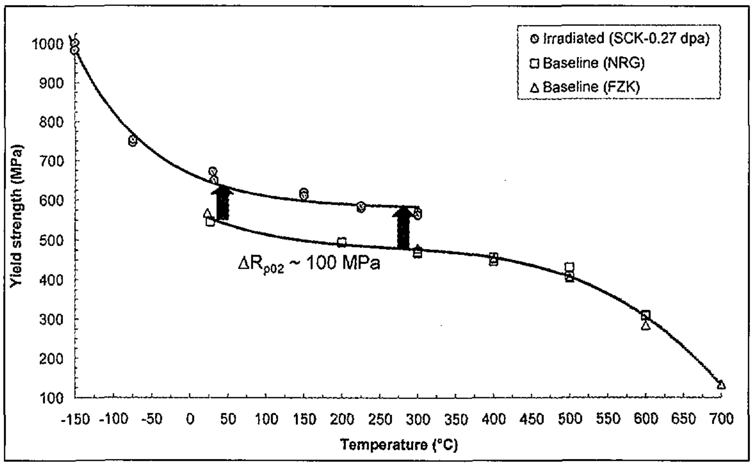
Supporting staff
André Gastmans, Kris Penasse



The ITER-FEAT tokamak. Material studies at SCK•CEN concern the plasma facing components, the blanket module and the divertor

RAFM: mechanical behaviour of irradiated material

RAFM steels represent the most promising structural materials for future fusion reactors. These steels are actually also considered for accelerator-driven reactor applications. Their nominal composition involves chromium as main alloying element. The fusion community has defined one particular RAFM steel, called Eurofer97, with 8% chromium content, as its reference material. In collaboration with several partners, SCK•CEN looks for radiation induced changes in the mechanical characteristics, in comparison with available data obtained on similar materials (as for instance the Japanese F82H steel). Samples of Eurofer97 were irradiated last year up to 0.27 dpa at 290 °C. Tensile tests have now been performed at different temperatures (T = -150 to 300 °C), and results compared with tensile data obtained by NRG and FZK on unirradiated samples. In the next figure, the yield strength values are given and show a yield increase (irradiation hardening) of about 100 MPa in the whole temperature range, when compared to baseline data. The ultimate tensile strength increase is however smaller, ranging from 30 to 50 MPa.



Yield strength values measured on Eurofer97 in the irradiated condition and comparisons with baseline data

Charpy impact tests have been conducted on irradiated samples, with measurements of lateral expansion and shear fracture appearance and precracked Charpy specimens were used for the determination of the fracture toughness. The results show a shift of the ductile-to-brittle transition-temperature (DBTT) by about 26 °C after irradiation, while the upper-shelf energy (USE) level remains unchanged. These results are apparently the first mechanical data obtained world-wide on irradiated Eurofer97 and are therefore of primary importance for the further development of the fusion material developments. Other samples are now further irradiated up to 2.5 dpa.

SCK•CEN investigates also how Oxide Dispersion Strengthening (ODS) can increase the service temperature of Eurofer97. A preliminary characterisation of the mechanical properties of two unirradiated ODS heats (prepared by PLANSEE) with different percentages of Y_2O_3 strengthening (0.3% and 0.5%) was performed. Based on tensile and subsize Charpy tests, the comparison with non-ODS data shows an improvement in tensile strength and elongation, but no effect on ductility. The reduction-of-area values decrease. Heat treatment is crucial for these materials. Untreated ODS shows a spectacular degradation in toughness, with DBTT shifting by more than 150 °C and USE dropping by 35%.

Modellisation

In collaboration with Risø, a series of low-dpa irradiations are being performed on a selected set of RAFM and pure iron samples, at two different temperatures. The purpose is to contribute to the development of reliable models for radiation-induced degradation. At the same time, an activity was started on the simulation of such an effect, at atomic scale, combining a Monte-Carlo approach with molecular dynamics techniques, for the simulation of the evolution of primary damage state and the calculation of defect-energy parameters. This is conducted in close collaboration with the Université Libre de Bruxelles (ULB) and the University of Madrid, in synergy with the SCK•CEN involvement in the REVE modellisation programme for fission pressure vessel steels.

RAFM: corrosion behaviour of irradiated material

The corrosion of structural materials is an important parameter to take into account in future reactor components, especially within blanket modules. The use of new materials, such as RAFM steels, presents a particular challenge, under the particular fusion conditions. Very few experimental data are actually available to evaluate their performance. In the proposed design of a water-cooled lithium-lead (WCLL) blanket, the material is facing two potentially aggressive media, namely liquid metal and pressurised water, with potential violent reactions if brought into contact. The lifetime prediction of the blanket circuits, separating these two fluids, is a priority concern with regard to safety. SCK•CEN's study addresses two main issues:

- the corrosion behaviour of RAFM steels in hot (300 °C) water: assessing their electrochemical behaviour, with special emphasis on the occurrence of stress corrosion cracking (SCC). Both fresh materials and irradiated materials are considered.
- the radiation-induced changes in electrochemical behaviour of these steels during their actual irradiation in the water environment: looking to the influence of radiolysis, as well as to direct flux effects, such as the photoelectric effect.

Stress corrosion cracking (SCC) of RAFM steels

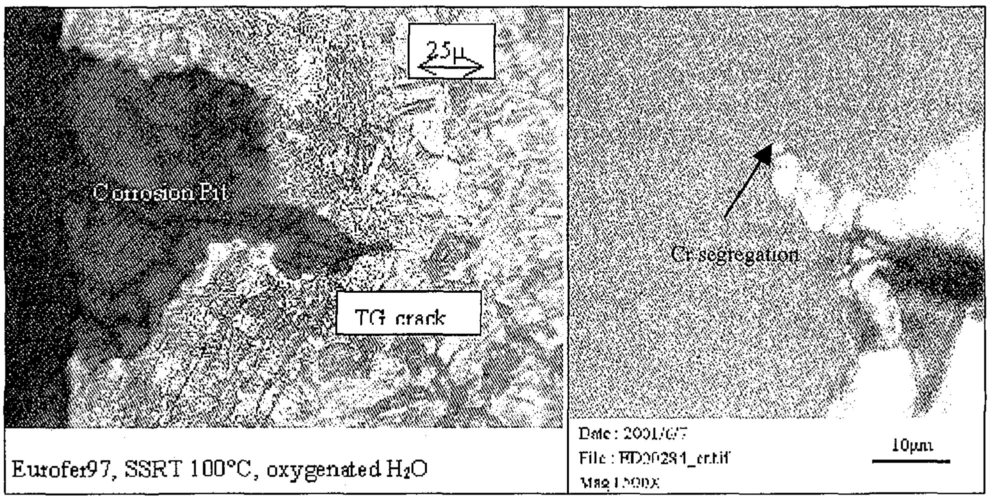
Two ferritic-martensitic steels have been selected for this study: the fusion reference steel Eurofer97 and another steel, the so-called BI56, with a higher chromium content (10%), a higher strength and a better corrosion resistance. The materials were tested in their as-delivered state, i.e. quenched and tempered. The unirradiated samples were first tested in water up to 230°C with different water chemistry. In 2001, the tests were extended to 300 °C, the representative temperature of the WCLL blanket. The susceptibility to SCC was evaluated by slow strain-rate tensile (SSRT) testing in an autoclave. Both alloys show a significant reduction in total elongation under oxygenated water conditions. The fracture surface exhibits even in that case a significant amount of stress-corrosion cracks. The next figure confirms that such stress-corrosion is often initiated by pitting corrosion, or corrosion of chromium-depleted zones on the prior austenite grain boundaries. The experiments will now be further extended to the ODS variant of Eurofer97, as well as to irradiated materials. As is suggested by the variation in SCC susceptibility in steels with different strength levels, the irradiation hardening may increase the susceptibility of the material to SCC. Currently, material irradiated up to 0.25 dpa is available and values up to 2.25 dpa will be provided in 2002.

Electrochemical behaviour of RAFM steels

The influence of radiation on the electrochemical behaviour of RAFM is studied by numerical and experimental simulation (injection of hydrogen peroxide as simulated radiolysis product). In parallel, electrochemical measurements were carried out under actual reactor irradiation (COFUMA experiment in BR2), in order to verify the influence of the irradiation flux (both neutron and gamma) on the corrosion behaviour, due to radiolysis and direct photoelectric effects. The material characterisation is measured on-line (electrochemical impedance, polarisation resistance) at various reactor power levels and positions in the core. The results allows to simulate more accurately the in-pile conditions during post-irradiation corrosion tests in hot cell environment.

Chromium, copper and titanium: radiation induced mechanical behaviour

Besides steels, other materials have to be used for the first wall and the blanket structures, in order to cope



SCC, initiated at the bottom of a corrosion pit. The crack initiates in the pit at a chromium segregation band, associated with the prior austenite grain boundary in the material

with the extreme thermal loads to be expected, during operation of the reactor. In 2001, SCK•CEN has studied three of these other materials:

- chromium, as a potential refractory metal for high temperature components with very low radiation induced activation;
- copper, for its excellent thermal conductivity in blanket modules cooling circuits;
- titanium: a low Young modulus material for flexible attachment parts on the first wall shielding cartridges.

Chromium

We completed in 2001, in collaboration with ESI in Austria, the mechanical characterisation of two chromium alloys (Ducropur: high-purity 99.7% chromium; and Ducralloy: chromium alloyed with 5% Fe and 1% Y_2O_3). Tensile tests at variable strain rates have been performed in the temperature range 300-500 °C. The influence of strain rate was found to be modest. The characterisation of the tensile properties of unirradiated material up to 500 °C was completed, as well as the study of fracture toughness, in both its original hot-isostatic pressing (HIP) condition and after heat treatment.

Copper

The use of standard CuAl25 alloy has shown increasing problems with low fracture toughness and more recently a reduction in fatigue lifetime, when exposed to combined creep loads, as expected during ITER operation. At the same time, new fabrication routes have been developed for the manufacturing of the blanket modules, which allow an alternative alloy, the CuCrZr, to be used in optimum conditions. It is known indeed that the mechanical and physical properties of this alloy are more sensitive to its thermal history. In collaboration with Risø, radiation-induced creep-fatigue effects. SCK•CEN provides a set of irradiated samples at low-dpa values, irradiated in BR2 at two different temperatures (50 °C and 300 °C) representative of the ITER blanket operating condition.

Titanium

The ITER first wall/shield modules are attached to the back plate by a set of four radial flexible supports, for which a given titanium alloy has been proposed (Ti-6Al-4V) due to its excellent mechanical properties and low Young's modulus. This alloy was extensively studied in previous years, except at the intermediate temperature of 200 °C. In collaboration with Risoe, a low-dpa irradiation of tensile, fatigue and fracture toughness specimens is now in preparation, in order to complete the data set related to this alloy.

Beryllium: surface reactivity with air and steam

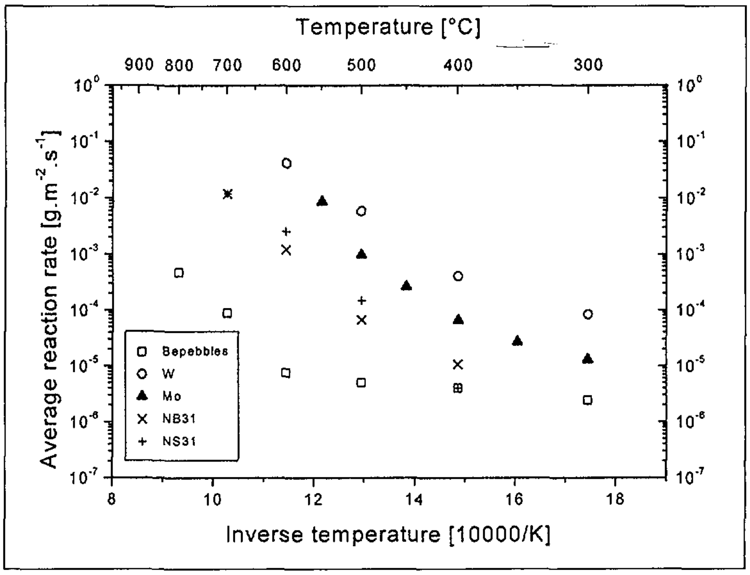
ITER foresees to use beryllium, tungsten and molybdenum, as well as carbon fibre-reinforced carbon composites (CFCs), either as neutron multiplier in the HCPB (Helium-Cooled Pebble-Bed) blanket module, or as armour material on the divertor, respectively. In case of accidental conditions (e.g. loss of vacuum or coolant accidents – LOVA or LOCA), these materials may come in contact with air and steam and their reactivity (production of hydrogen) is a safety concern. A dedicated experimental facility has been installed, in order to provide kinetic data on the chemical surface reactions, for use in safety evaluations. The installation consists of a generator for oxidising atmospheres, a thermo-gravimetry and differential thermal analysis (TG/DTA) equipment, two cryogenic condensers and a quadrupole mass spectrometer. The chemical reactivity in air of beryllium pebbles, as specified for the HCPB blanket module, has been measured. The same tests have been performed on molybdenum, tungsten and two grades of CFC (undoped and doped with 8-10 wt.% silicon). The next figure gives an overview of the results. The three metals show a transition temperature where the trend changes from parabolic to linear kinetics. Parabolic kinetics is associated with the formation of a protective oxide layer, while linear kinetics is associated with non-protective oxidation, leading eventually to the depletion of the base metal. This transition temperature is around 600°C for beryllium, but lower for tungsten and molybdenum (500-550 °C). Although safety studies indicate that the total amount of hydrogen produced during a LOCA would not impair the containment safety of ITER, it is desirable, from a licensing point of view, to minimise this amount. Reactivity reduction approaches were therefore investigated. Below 500 °C, the reaction kinetics is parabolic and the reaction rate is determined by diffusion of Be^{++} cations through the oxide film. This temperature corresponds actually to the maximum operating temperature of ITER, as well as to the expected temperature during an in-box LOCA. Three mitigation strategies can be considered at that temperature: adding alloying elements, coating the surface, or modifying directly the oxide film by surface engineering. Collaboration was started with ITN in Portugal to implant ions (calcium, aluminium and chromium) directly into the oxide layer, using their 3.1 MV Van de Graaff accelerator. The implanted samples will be studied by thermogravimetry and mass spectrometry techniques in Mol and by Rutherford Back-Scattering in Lisbon.

Beryllium: swelling, helium release and creep of irradiated material after annealing

Using highly irradiated beryllium coming from the BR2 reactor matrix, a study was launched on swelling, helium release and creep at conditions relevant to a fusion reactor end-of-life status, i.e. helium concentrations up to 30 000 appm and temperature peaks up to 800 °C. The objective is to provide data for the development of the ANFIBE code, developed at FZK in the framework of the HCPB blanket design. This code describes the beryllium pebble bed behaviour during the whole reactor life. Irradiated material samples (~ 20 000 appm He) have been extracted from the BR2 second matrix and are now prepared to undergo long-term annealing campaigns at 500 °C, 750 °C and 900 °C (during three, six and nine months). Helium content, density and porosity will be measured to characterise the gas behaviour, the radiation-induced porosity and the concurrent swelling. A creep test is being designed in order to allow for creep properties to be evaluated as well.

Beryllium waste: Selecting the optimum conditioning technique

Operating fusion reactors will involve large quantities of beryllium, for which a suitable waste management approach must be defined. In many fission research reactors, the beryllium waste is still stored on site, as no standard route exists yet for its processing. Recycling in nuclear applications could offer an interesting solution, but would probably not cover the whole beryllium inventory anyway. The final disposal route is an alternative option to consider, and acceptable conditioning methods must be found. The study considered five possible options:



Summary of chemical reactivity tests results for beryllium (REM produced pebbles), tungsten (W), molybdenum (Mo), undoped CFC (NB31), and silicon doped CFC (NS31). The reported average reaction rates are based on the total mass gain/loss after six hours of exposure to air and on the initial geometric surface area of the samples

encapsulation in cement, encapsulation in bitumen, incorporation in a glass matrix, incorporation in phosphate ceramics and direct metal disposal, where the waste is simply put in drums, with sand filling the voids. These options were compared on the basis of existing experience, known advantages/drawbacks and cost. Beryllium is toxic and reacts with water to produce hydrogen. Conditioning options involving the waste remaining in its metallic form (cement, bitumen and direct disposal) are therefore inappropriate. The preferred options are incorporation in silicate glass or in phosphate ceramics. Vitrification can rely on a long experience in the nuclear waste conditioning, and this approach was therefore chosen for an experimental demonstration on real activated beryllium. In parallel, the chemical impact of beryllium on the biosphere was estimated using the Belgian reference scenario for deep geological disposal in a clay layer. Even with very conservative estimations of the migration parameters, it appears that the beryllium concentration in the surrounding aquifers remains several orders of magnitude lower than the official drinking water limit. On the other hand, beryllium, in comparison with typical fission waste, has too little an activity and too short a half-life to influence significantly the radiological impact of the repository on the biosphere.

Comparing dismantling strategies and assessing waste clearance and recycling options

SCK•CEN is gaining considerable experience in decommissioning strategies by dismantling its BR3 PWR reactor. This asset is now used to analyse the particular fusion case. It is still possible to orient the fusion dismantling strategy, in order to take full advantage of the existing fission experience. This includes cost assessment methodology. The study highlighted specific suggestions. Fission reactor dismantling is using extensively underwater cutting techniques. This option should be considered in parallel to the use of available maintenance tools to dismantle in dry hot cells. Decontamination of cooling loops prior to dismantling is a valuable option to consider. It greatly reduces the dose rate, allowing even hands-on dismantling to be performed in some cases, and minimising contamination spread when opening the loops. In both types of reactors, a large volume of low-level radioactive metal and concrete is generated and must be evacuated. Large efforts are put on recycling options in the fission industry, either unconditionally (clearance), or for re-use in the nuclear industry. The development of thorough decontamination processes and the availability of several radioactive melting facilities, as well as the progressive acceptation of clearance and recycling limits, make now a recycling option feasible, even for large amounts of materials. At present, such an approach is however limited to low level waste (75 to 200 Bq/g) and to the fabrication of low commercial value components. Fusion aims on the contrary at recycling higher value materials (beryllium, tungsten, etc.) with higher activation levels. Such an extrapolation needs further developments, as no existing facility exists yet for this type of waste. The study was also extended to contaminated and activated concrete. Here, recycling after clearance is common practice. Low-activated concrete becomes for instance grout used for radioactive waste conditioning, an approach that could be applied to fusion reactors too.

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

The radiological consequences of human intrusion scenarios are considered to be important indicators for the long-term safety of fusion waste disposal sites. Two intrusion scenarios are usually considered: a core examination scenario (extraction of disposed waste by geological workers, not aware of their presence) and a destruction of engineered barriers by drilling boreholes. To illustrate the short period of concern of fusion waste, comparison was made with similar scenarios applied to fission waste sites. The first scenario was analysed previously. In 2001, the impact of borehole drilling on a multi-barrier repository system was studied. It appeared necessary to distinguish several variants of this scenario, depending on the use, or not, of a casing tube or filling material. With open borehole, the convergence of the plastic clay quickly closes the borehole. The resulting short contact time (typically a few months) between waste and groundwater considerably limits the consequences. A worse variant appears when the borehole is accidentally filled with a coarse material, such as gravel. This hinders the convergence of the clay and allows water flowing through the borehole. The results are now being processed, and will be compared with equivalent data for fission.

Better water detritiation systems using an improved catalyst

Fusion reactors produce tritiated waste, mostly under the form of tritiated water. An adequate water detritiation system is therefore a crucial part of the reactor plant. The process usually considered is the so-called liquid phase catalytic exchange (LPCE). It relies on isotope separation appearing during the exchange of tritium between hydrogen gas and liquid water, flowing in counter-current way, at relatively low temperature. This allows for a good multiplication of the separation factor along a simple packed bed column. The chemical exchange requires however a catalyst, capable to cope with the low solubility of hydrogen in water and the resulting low diffusion rate. SCK•CEN developed in the past such a hydrophobic catalyst, capable to avoid its "poisoning" by liquid water. The project was linked to reprocessing activities, which were discontinued in Belgium. Fusion, however, and in particular the JET facility, has recently shown a renewed interest in these developments, with the request of further improving the process and reducing manufacturing costs. The activity is performed in collaboration with FZK. Together with manufacturers, the catalyst particles were successfully reproduced. Overall gas transfer heights between 0.2 and 0.25 m at 40 °C, with a hydrogen flow rate of 10 mol.s.m^{-2} were achieved and the results confirmed by tests performed at the D. Mendelev University in Moscow. Further evaluation is planned at JET. To situate the catalyst performance, one should remind that the JET application requires the detritiation of 28 000 kg H_2O (HTO) per year, corresponding to a LPCE treatment of 0.074 $mol.s^{-1}$ H_2(HT). Compared to alternative solutions, the use of the SCK•CEN catalyst allows a smaller LPCE column to be sufficient (diameter < 0.1 m; height < 6 m), while still keeping a decontamination factor higher than 10 000.

Looking for synergy between fusion and accelerator-driven systems material research

We organised a Topical Day (May 15, 2001) on the material studies for new reactor concepts, with a particular focus on the existing synergy between the fusion programme and the development of accelerator driven systems (ADS). Material research is indeed identified as a key issue in both developments. The high neutron fluences and energies, the high heat fluxes, the interaction with liquid metal and the aim of reducing induced activation are challenges to be met. The topical day identified the similarities and differences between both approaches and requirements. By gathering nearly one hundred experts from ten different countries, it gave an excellent opportunity to trigger further collaborations between these two communities.

Perspectives

Material studies form an essential part of the development of a high-performance and safe fusion reactor. SCK•CEN will keep its major involvement in fusion in this particular niche. The studies related to first wall and structural materials will be continued with a particular accent put during the following years to the radiation-induced mechanical and corrosion behaviour of RAFM steels, as used in the water cooled blanket module. For beryllium, radiation induced creep and swelling on very highly irradiated material will be the main item. A new approach will be developed for the copper tests for ITER: on-line dynamic loading of the samples will be made possible during the irradiation. The waste strategy will focus on two specific issues: the capability to recycle the produced waste (metals and concrete), and the feasibility of limiting waste disposal to shallow-land buried sites. It is also the aim to see the developed catalyst integrated next year in the JET detritiation installation, and an optimum beryllium waste conditioning approach to be experimentally validated.

Partners

-	D. Mendelev University, Moscow (Russia)
EFDA	European Fusion Development Activities (Garching, Germany)
ESI	Erich Schmid Institute of Material Science (Leoben, Austria)
FZK	Forschungszentrum Karlsruhe (Karlsruhe, Germany)
ITN	Instituto Tecnológico e Nuclear (Secavém, Portugal)
JET	*Joint European Torus* (Abingdon, United Kingdom)
NRG	Nuclear Research Group (Petten, The Netherlands)
PLANSEE	(Plannsee, Austria).
Risø	Risøe National Laboratories (Risø, Denmark)
ULB	Université Libre de Bruxelles (Physique Statistique et Plasmas) (Brussels, Belgium)
Univ. Madrid	Polytechnic University of Madrid, Institute of Nuclear Fusion (DENIM), Madrid, Spain

Sponsors and customers

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

Scientific output

Publications

F. Druyts, J. Fays, P. Van Iseghem, F. Scaffidi-Argentina, "Chemical reactivity of beryllium pebbles in air", to be published in Fusion Engineering and Design, 2001.

E. Lucon, E. van Walle and M. Decréton, "Mechanical properties of two Chromium alloys in as-received and heat-treated conditions", 21st SOFT (*Symposium on Fusion Technology*), September 11-15, 2000 Madrid (to be published in Fusion Technology, 2001).

E. Lucon, E. van Walle and M. Decréton, "Mechanical properties of two Chromium alloys in as-received and heat-treated conditions", Fusion Engineering and Design, Vol. 58-59, pp 767-773, Dec 2001.

Presentations

E. Lucon, "Mechanical tests on two batches of oxide dispersion strengthened RAFM steel (Eurofer97)", to be presented at the 6th International Symposium on Fusion Nuclear Technology, ISFNT-6, San Diego, California, April 7-12, 2002.

E. Lucon, "Results obtained at SCK•CEN on Chromium and on ODS Eurofer97", EFDA Materials Development *Programme Meeting, Petten (The Netherlands), February* 26-28, 2001.

R. Chaouadi, "Review of some materials issues investigated at SCK•CEN within the ITER short term R&D program", Topical Day on Materials for New reactor Concepts – A Synergy between Fusion and ADS Research Programmes, Mol, May 15, 2001, Book of Abstracts BLG-878, p. 11.

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R. W. Bosch, S. Van Dyck, "The influence of hydrogen peroxide on the corrosion potential of Ferritic-Martensitic stainless steels for fusion applications in high-temperature water", Eurocor 2001, Riva di Garda, October 1-4 2001.

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M. Decréton, "Annual report 2001 – SCK•CEN Fusion Programme", BLG-899, November 2001.

E. Lucon and M. Wéber, "Irradiation of Mechanical Specimens of Eurofer97 and Chromium Alloys in the BR2 Reactor": the IRFUMA Experiment, BLG-872, February 2001.

E. Lucon, "Mechanical Tests on Two Oxide-Dispersion Strengthened (ODS) Batches of EUROFER97", BLG-874, March 2001.

M. Klein, V. Massaut, L. Denissen, "European Fusion Long Term Programme – Safety and Environment, Task 55.2 Management of activated materials – Waste strategy: Dismantling options", Intermediate Report, SCK•CEN, R-3490, 20 February, 2001.

F. Druyts, "Chemical reactivity of beryllium pebbles in air and steam, SCK•CEN report", R-3493, January 2001.

E. Lucon and M. Wéber, "Irradiation of Mechanical Specimens of Eurofer97 and Chromium Alloys in the BR2 Reactor: the IRFUMA Experiment", BLG-872, Feb. 2001.

D. Lodi, J. Perlado, L. Malerba, "Multiscale Modelling of Pulsed versus Continuous Irradiation Damage Production in alpha-Iron in Inertial Fusion Conditions", SCK•CEN report to be published, December 2001.

M. Decréton, Book of Abstracts, "Topical day on Materials for New Reactor Concepts: A Synergy between Fusion and ADS Research Programmes", BLG-878, May 2001.

J. Fays, F. Druyts, C. De Raedt, "Conditioning methods for irradiated Be waste", progress report to be issued in December 2001.

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