IAEA nuclear fusion

Towards reduced activation structural materials data for fusion DEMO reactors

To cite this article: A. Möslang *et al* 2005 *Nucl. Fusion* **45** 649

Nucl. Fusion **45** (2005) 649–655 doi:10.1088/0029-5515/45/7/013

Towards reduced activation structural materials data for fusion DEMO reactors

A. Möslang[1], E. Diegele[2], M. Klimiankou[1], R. Lässer[2], R. Lindau[1], E. Lucon[3], E. Materna-Morris[1], C. Petersen[4], R. Pippan[5], J.W. Rensman[6], M. Rieth[1], B. van der Schaaf[6], H.-C. Schneider[4] and F. Tavassoli[7]

[1] FZK, Institute for Materials Research I, PO Box 3640, 76021 Karlsruhe, Germany
[2] EFDA, CSU, Boltzmannstrasse 2, D-85748 Garching, Germany
[3] SCK-CEN, Boeretang 200, 2400 Mol, Belgium
[4] FZK, Institute for Materials Research II, PO Box 3640, 76021 Karlsruhe, Germany
[5] Institute of Metal Physics, University Leoben, 8700 Leoben, Austria
[6] NRG, MM&I, PO Box 25, 1755 ZG Petten, The Netherlands
[7] CEA/Saclay, DMN/Dir, 91191 Gif sur Yvette Cedex, France

E-mail: anton.moeslang@imf.fzk.de

Received 12 January 2005, accepted for publication 15 April 2005
Published 24 June 2005
Online at stacks.iop.org/NF/45/649

Abstract
The development of first wall, blanket and divertor materials that are capable of withstanding for many years high neutron and heat fluxes, is a critical path to fusion power. Therefore, the timely availability of a sound materials database has become an indispensable element in international fusion road maps. In order to provide materials design data for the short-term needs of ITER test blanket modules (TBMs) and for a DEMOnstration fusion reactor, a wealth of R&D results on the European reduced activation ferritic–martensitic steel EUROFER, and on oxide dispersion strengthened variants are being characterized, mainly in the temperature window 250–650˚C. The characterization includes irradiations up to 15 dpa in the mixed spectrum reactor HFR and up to 75 dpa in the fast breeder reactor BOR60. Industrial EUROFER-batches of 3.5 and 7.5 tons have been produced with a variety of semi-finished, quality-assured product forms. To increase the thermal efficiency of blankets, high temperature resistant SiC_f/SiC channel inserts for liquid metal coolant tubes are also developed. Regarding radiation damage resistance, broad based reactor irradiation programmes deal with several ranges from $\leqslant$5 dpa (ITER TBMs) up to 75 dpa (DEMO). For the European divertor designers, a materials database is currently being set up for pure W and its alloys, and related reactor irradiations are foreseen with temperatures from 650˚C to 1000˚C.

PACS numbers: 28.52.Fa, 61.82.-d, 81.05.Bx, 83.50.Uv

(Some figures in this article are in colour only in the electronic version)

1. Introduction

The development of first wall, blanket and divertor materials that are capable of withstanding high neutron and heat fluxes for sufficiently high neutron fluence of at least 10–15 MW-y m^{-2}, corresponding to a component lifetime of at least five years, is a critical path to fusion power. As the construction of a fusion demonstration reactor, DEMO, for demonstrating competitive electrical power generation will be the next important programme milestone after ITER, the timely availability of a sound materials database has become an indispensable element in international fusion road maps. Even more, fast track scenarios like the Fast Track Experts Meeting convened by the president of the European Union (EU) Research Council and chaired by King [1], or the FESAC Fusion Development Path Panel chaired by Hazeltine [2] confirmed again that the material qualification and, therefore, the required International Fusion Materials Irradiation Facility, IFMIF, are on a critical path. The fabrication technology and the materials database provided for design, construction, licensing and safe operation of a DEMO, must provide highly attractive properties, especially with respect to high thermal efficiency, availability, reliability, irradiation resistance and reduced activation capability.

After a brief review of strategies for materials development, this paper describes for the major blanket and divertor structural materials, the present status of the fabrication and manufacturing technologies and the

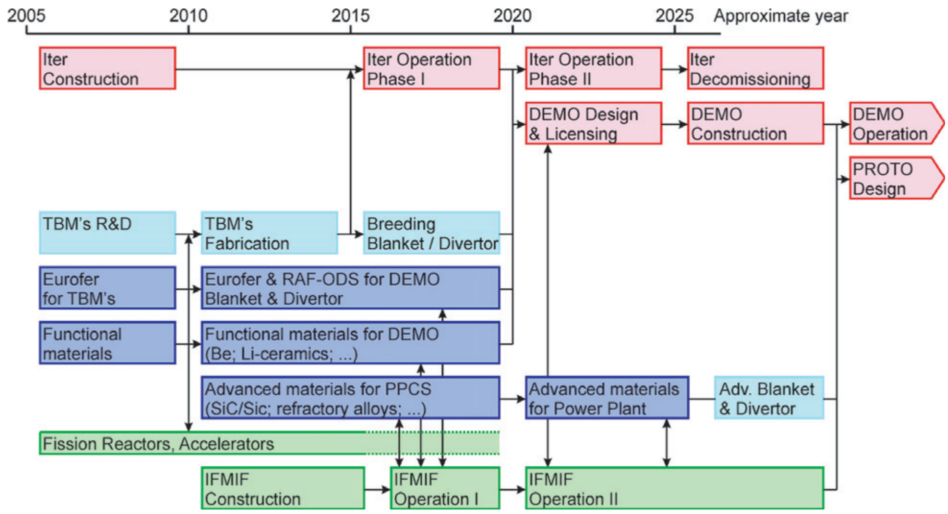

Figure 1. Cross-linking between the materials development and the fusion road map according to [4].

achievements of the materials qualification, highlighting examples of recent progress in Europe.

2. Strategies for materials development

Supported by the 'broader approach' discussions of the ITER negotiations, the major elements of formally developed [3] global fusion strategy scenarios have become very similar for the different parties. Figure 1 shows an example of an updated related road map [4]. The criteria mentioned have led to a worldwide concentration of R&D efforts towards a few materials classes that are currently being developed in a largely coordinated manner under the umbrella of an IEA implementing agreement (Annex-II). For blanket applications, these are the structural materials classes of the reduced activation ferritic martensitic (RAFM) steels (including nanodispersion strengthened variants), the vanadium alloys and the SiC fibre reinforced ceramic SiC composites. Although all three classes already fulfil major requirements of 'low level waste' capability, the different materials are hardly comparable because the level of knowledge differs remarkably. The currently investigated vanadium alloys have an attractive combination of thermo-physical properties, high temperature strength and nuclear inventory decay; however, after irradiation below about 420˚C, they may suffer severe fracture toughness degradation and, because of the pick-up of interstitial impurities (O, C, N), they are subject to pronounced embrittlement above about 550˚C. SiC_f/SiC composites feature attractive short and medium term nuclear inventories and high temperature strength. However, their ability to maintain physical properties and structural integrity under high neutron doses must be fundamentally demonstrated, and pending structural design criteria, the fabrication technology for large SiC/SiC components of complex shapes is not yet available. In view of the extreme heat load of helium gas cooled divertor concepts in some PPCS studies, the favoured materials are at present nanodispersion strengthened refractory alloys (e.g. W–La_2O_3) and W tiles on the plasma side and high temperature resistant oxide dispersion strengthened (ODS) RAF steels on the backbone side.

RAFM steels have by far the most advanced technological base of the three candidates and, therefore, have become the reference structural material for ITER test blanket modules (TBMs) and DEMO blankets for all parties:

ITER: In order to get fully qualified TBMs assembled at the beginning of the operation of phase I (around 2015), a detailed engineering design as well as fabrication and testing of TBMs is required during 2006–09 (figure 1). The ensuing activities (2010–15) include fabrication, assembly and acceptance tests of final TBMs [5].

DEMO: While the displacement damage level in ITER TBMs does not exceed about 3 dpa (displacements per atoms) during the entire lifetime, 25–30 dpa will be achieved in DEMO blankets within one year. That is, the neutron and thermo-mechanical loads of DEMO operating conditions are much more demanding. For DEMO design and licensing (figure 1), all necessary materials should be qualified up to about 70–80 dpa in a fusion relevant neutron environment like IFMIF until around 2022.

Within the EU, the two major breeder blanket concepts currently being developed are the helium cooled pebble bed (HCPB), and the helium cooled lithium lead (HCLL) blankets. For both concepts, sharing the same overall blanket box, different conceptual designs are being discussed for the breeder units with temperature windows in the range 250–550˚C for conservative approaches based on 'reference' RAFM steels. In the future, more aggressive designs taking into account RAFM ODS steels could enlarge the operational window to about 300–650˚C. In addition, high thermal efficiencies beyond ∼42% could be achieved by the use of SiC_f/SiC channel inserts inside LiPb coolant tubes in a dual coolant concept. In order to provide a timely related materials database for the TBMs of ITER and for DEMO, the European strategy for materials development is largely governed by the key issues of (i) high neutron damage resistance, (ii) good mechanical properties at high operating temperature and (iii) low activation properties. Similar approaches with different emphases exist in Japan, the RF and US.

Table 1. Chemical composition of EUROFER-97 compared to F82Hmod; target values in [·]. (*a*) Main alloying elements, (*b*) radiologically undesired elements in mass% and ppm.

		Radiologically desired (mass%)	EUROFER-97 specified (mass%)	EUROFER-97 achieved (mass%)	F82Hmod.Heat 9741 (mass%)
(*a*)	C		0.09–0.12 [0.11]	0.11–0.12	0.09
	Cr		8.5–9.5 [9.0]	8.82–8.96	7.7
	W		1.0–1.2 [1.1]	1.07–1.15	1.94
	Mn		0.20–0.60 [0.40]	0.38–0.49	0.16
	V		0.15–0.25	0.18–0.20	0.16
	Ta		0.10–0.14 [0.12]	0.13–0.15	0.02
	N_2		0.015–0.045 [0.030]	0.018–0.034	0.006
	P		<0.005	0.004–0.005	0.002
	S		<0.005	0.003–0.004	0.002
	B		<0.001	0.0005–0.0009	0.0002
	O_2		<0.01	0.0013–0.0018	(0.01)
(*b*)	Nb	<0.000001	<0.001	0.0002–0.0007	0.0001
	Mo	<0.0001	<0.005	0.001–0.0032	0.003
	Ni	<0.001	<0.005	0.007–0.028	0.02
	Cu	<0.001	<0.005	0.0015–0.022	0.01
	Al	<0.0001	<0.01	0.006–0.009	0.003
	Ti	<0.02	<0.01	0.005–0.009	0.01
	Si	<0.04	<0.05	0.04–0.07	0.11
	Co	<0.001	<0.005	0.003–0.007	0.05

3. Reduced activation steels of EUROFER type

3.1. Alloy selection, heat production, joining and manufacturing technologies

RAFM steels are modified compositions of conventional ferritic–martensitic 8–12% Cr–MoVNb steels, obtained mainly by exchanging Mo, Nb and Ni with W and Ta in order to obtain low activation capability [6]. To meet the ambitious time schedule, an integrated research and test programme has been set up that includes broad based alloy development and qualification activities as well as various joining and manufacturing technologies of test modules and mock-ups. As a result of a systematic screening of many RAFM laboratory heats in Europe [7–10], and taking into account experience from similar programmes in Japan and the US, the 9% CrWVTa alloy EUROFER-97 was specified. Industrial batches of 3.5 and 7.5 tons have been produced by two different EU companies with 10 different heats and 18 different semi-finished, quality-assured product forms (forged bars, plates, tubes wires) in various dimensions. Table 1 shows the chemical compositions of EUROFER-97 and the Japanese reference RAFM steel F82Hmod with upper limits of radiologically undesired elements. The RAFM steels produced already meet the requirement of being low level waste after 80–100 years of DEMO reactor shutdown. The next step is fine tuning the alloy on the basis of substantial results from fission reactor irradiations and the subsequent definition and fabrication of EUROFER-2 after 2006. An additional optimization (EUROFER-III) aims at a substantial reduction of radiological impurities. This is considered to be technically possible at the expense of increasing costs and would further significantly reduce the remaining γ-dose rate beyond 100 years, but would have no measurable impact on mechanical properties.

The main fabrication technologies currently examined for European breeder blanket structures include (i) bending, cutting and machining of different EUROFER plates, (ii) joining of complex shaped semi-finished products with many internal ducts and (iii) heat treatments to ensure long-term ageing resistance and to adjust a well-defined microstructure that guarantees appropriate ductility and fracture toughness properties.

As joining technologies play a key role in the timely qualification of all breeder blankets, a broad based technology programme has been launched in Europe on EUROFER-97 steel products with several associations involved. The joining technologies included until now (i) TIG welds, (ii) electron beam welds, (iii) laser welds, (iv) fusion welds under uni-axial pressure, as well as (v) solid–solid HIP joints and powder consolidated blocks. The welded product forms ranged from laboratory scale tensile and fracture toughness samples for a systematic analysis of post-weld heat treatments (PWHTs) on mechanical and microstructural properties to medium-scale mock-up fabrication with structural integrity tests after PWHT. An overview of EUROFER-specific manufacturing technologies with special emphasis on results of joining techniques is given in [11]. Extensive chipless shaping and joining experience taking into account different welding procedures and powder technology product forms have demonstrated that EUROFER type steel complies with a wide range of established manufacturing processes.

3.2. Materials design limits and status of the materials database

3.2.1. Heat treatment, ageing and physical properties. Broad based characterizations and recommended heat treatments are available for the RAFM steels EUROFER-97 and F82Hmod. For EUROFER-97, the recommended two-stage heat treat treatment varies slightly depending on the product form. Generally, it involves austenitization at 980–1040˚C for 27–30 min, followed by tempering at 750–760˚C for 90–120 min. All produced heats, semi-finished products and joints were post-heat treated and analysed in various EU laboratories with emphasis on traceability and reproducibility. Until now, the in-depth chemical and homogeneity analyses

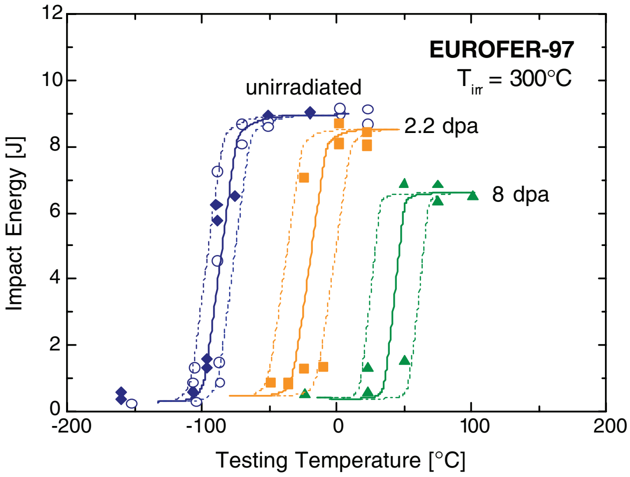

Figure 2. KLST impact transition curves for EUROFER-97 8 mm plate after 300˚C irradiation.

have not revealed any abnormalities. The EUROFER-97 grain size is about 2–3 times smaller than F82Hmod. Continuous CCT diagrams have been analysed in detail together with the transformation, long-term ageing, hardening and thermal expansion (α_m and α_i) behaviour in the temperature range from RT to 1120˚C [8]. Other physical properties include elastic (E) and shear moduli (G), Poisson's ratio (ν), electrical resistivity (Ω), density (ρ), specific heat (C_p), thermal conductivity (k) and diffusivity (a), as well as magnetic properties.

3.2.2. Tensile properties. On the basis of work done in a variety of EU laboratories a reliable database exists on different unirradiated EUROFER-97 heats, product forms and orientations in the temperature range RT–750˚C, both on base and welded samples. Tensile strength and ductility are almost unaffected by ageing at 280–600˚C up to 3300 h. In the temperature range around 250–300˚C, which is most relevant for 'low temperature irradiation embrittlement' mixed spectrum reactor irradiations exist up to about 9 dpa [12–14]. The related radiation induced strength increase of $\Delta\sigma_y \cong$ 500 MPa and the corresponding total elongation of a few per cent is within the scatter band in agreement with F82mod data.

3.2.3. Impact toughness and fracture toughness. Impact tests have been carried out with standard and KLST-type specimens on all EUROFER-97 half products in all heat treated, aged and joined conditions. The common KLST impact tests reveal in the as-received condition a ductile to brittle transition temperature (DBTT) of about −90˚C (figure 2). After irradiation at 300˚C to 8 dpa this DBTT is increased to about 40˚C [12]. The DBTT of solid–solid HIP joints is between −60˚C and −90˚C, depending on the PWHT. Besides the DBTT, upper shelf and lower shelf energies and FATT were determined.

Fracture toughness data based on CT, 1/2T CT and pre-cracked Charpy specimens have been measured in the heat treated condition in EU laboratories. According to the master curve approach for CT specimens, T_0 is equal to −130˚C. With respect to the international approach of qualifying miniaturized samples for fracture toughness, an overview is given in [15].

3.2.4. Fatigue and creep fatigue. For isothermal, strain controlled (creep) fatigue experiments complete datasets are available for EUROFER-97 in some EU laboratories from RT to 550˚C up to a total strain range of $\Delta\varepsilon_t = 1.5\%$, and for dwell times in the compression and/or tensile phase from 0 to 600 s. The datasets include a sufficiently high number of stress–strain hysteresis loops per fatigue test and provide a satisfactory correlation between $\Delta\varepsilon_t$ and the number of cycles to failure, N_f, for a given set of machined specimens. However, for reliable lifetime prediction models, the impact of surface roughness on N_f and a correlation between micro-crack initiation and propagation with N_f still needs a lot more R&D work. Thermal creep fatigue with dwell times of 1000 s either in the compression or in the tensile phase, and with temperature windows from 100˚C to (500–550)˚C reveals lifetimes N_f between 400 and 9000 cycles, depending on the experimental conditions. However, N_f of such thermally fatigued EUROFER-97 is reduced significantly during alternating dwell times at high and low temperatures.

3.2.5. Creep and creep rupture. In the design relevant temperature and stress range, broad based creep-rupture experiments between 450˚C and 600˚C up to 30 000 h creep rupture time have been executed on EUROFER-97 heats in two EU laboratories. Creep behaviour, stress exponents, 1% yield limits and rupture times showed no deviations from expected behaviour, confirming long-term stability and predictability [8]. Emphasis is currently placed on the safety relevant low-stress range (100 MPa and below).

3.2.6. Fission reactor irradiations. In addition to existing results, mainly from the mixed spectrum reactors BR2 and HFR in the lower dose regime, broad based HFR irradiations on 11 different heats have been performed between 60˚C and 450˚C with displacement damage levels up to 15 dpa, and in the fast breeder reactor BOR60 at 330˚C up to 30 dpa. The different heats include some reference RAFM steels, various heats of EUROFER-97, ODS-EUROFER, as well as ^{10}B and ^{11}B doped EUROFER to simulate helium effects. Significant post-irradiation examination results will likely be available by the end of 2005. 75 dpa data after 330˚C irradiation will become available in 2006–08. Although irradiated EUROFER specimens have been examined at present at only 10 dpa and below, it can be stated that irradiation induced hardening, ductility reduction and fracture toughness degradation are substantially smaller than irradiated conventional ferritic–martensitic steels like EM10, HT9, MANET and T91. After irradiation, the diffusion welded material performs worse than plate in the impact and fracture tests, but HIP welded material gives good impact results after irradiation.

3.2.7. Materials design limit data, modelling and constitutive equations. The materials community has to meet requests coming from 'General Design Requirements Documents' or 'ITER Interim Structural Design Criteria' in due time. Therefore, not only has a sound database on materials

Figure 3. Example of data sheets from two European materials databases.

properties been set up for EUROFER-97, but also data files containing traceable information on (i) procurement specification, (ii) grades of material used in the ITER TBM relevant specifications and (iii) section designations corresponding to material properties groups. These properties groups contain the analysed materials behaviour, especially the 'Material Design Limit Data'. For EUROFER-97 such data are drafted in [16]. Besides the materials data, a wealth of modelling activities is needed, not only for a scientific understanding of properties but also for safety and availability assessments, as well as for lifetime prediction of components. Finally, within the materials design interface activities, constitutive equations need to be further established in strong interaction with fusion specific code developments.

3.2.8. Materials database. The data from the resulting R&D activities are compiled in relational databases in Japan and Europe. Examples of the two main databases currently used in the EU by several associations contributing to the fusion materials programme are given in figure 3. Both databases contain broad based, design relevant data on RAFM steels, especially EUROFER-97. One database is available on a CD

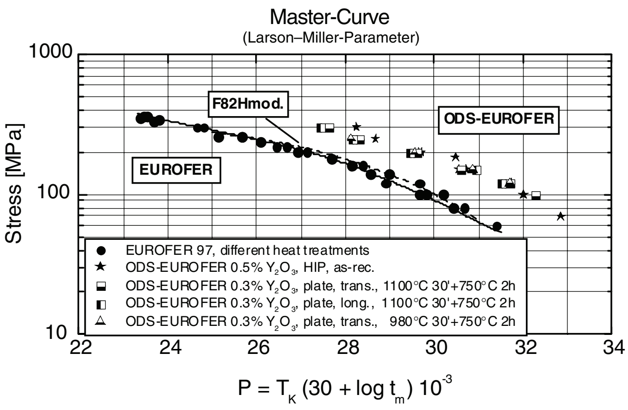

Figure 4. Larson–Miller plot for EUROFER-97 with different heat treatments in comparison with oxide dispersion strengthened ODS-EUROFER and F82Hmod (broken line).

as a Runtime Solution for different operating systems [16, 17]. It also includes (i) an appendix A with design acceptances for various material properties, and (ii) a 'summary database' with an overview of major properties for a given material or plate (figure 3, bottom).

3.3. *Improvement of high temperature strength—the RAF(M) ODS steels*

For specific blanket and divertor applications, a replacement of currently considered conventionally produced RAFM steels by suitable ODS alloys allows a substantial increase in the operating temperature from ∼550˚C to about 650˚C or even more. This has been shown by long-term creep rupture tests (⩽10 000 h between 600˚C and 700˚C) on specimens made of ODS-EUROFER (figure 4). A large material characterization programme, including reactor irradiation to 15, 30 and 70 dpa, going on. A breakthrough has been achieved at FZK in overcoming the poor high temperature ductility and the DBTT of first generation RAFM-ODS and commercial ferritic ODS alloys: the production route of the mechanically alloyed EUROFER–0.3 wt% Y_2O_3 included rolling in the so-called cross-roll technique with industrial partners, followed by appropriate thermal treatment. As a result, yield and ultimate tensile strength of the ODS-EUROFER below 450˚C are raised by 50% and more, compared to the non-ODS RAFM steels like EUROFER-97 and F82Hmod. The total and uniform elongation of the ODS steel over the whole temperature range from RT to 700˚C are similar to common RAFM steels or even better. Impact tests on sub-size KLST specimens show greatly improved DBTT, which could be shifted from +120˚C for hipped ODS-EUROFER of the first generation to values well below 0˚C. To establish reliable joining between ODS and RAFM steels, a diversified programme on diffusion weldments was performed, showing that between RT and 700˚C all joints reached the strength and ductility of the base EUROFER steel. Analytical high resolution TEM has been used to explain the encouraging behaviour of the European RAFM-ODS batches.

3.4. *SiC_f/SiC composites and refractory alloys*

SiC_f/SiC composites are considered in the world as a most promising structural material option for a very high temperature operating window for blankets. Currently, the results of neutron irradiation effects on SiC_f/SiC composites up to 6 dpa are collected in Europe with irradiation temperatures of ∼600˚C and 900˚C. Reductions in the bending modulus and dynamic modulus of some types of SiC_f/SiC composites have been observed, whereas others remained unaffected. Although neutron radiation strongly reduces the thermal conductivity, the results indicate that with proper selection of manufacturing processes improvements can be realized. In the EU both two- and three-dimensional SiC_f/SiC composites have been manufactured on an industrial scale in close co-operation with Japan and the US to optimize the three-dimensional fibre architecture. Strength values of 200 MPa and a thermal conductivity of over 20 W mK^{-1} have been achieved for unirradiated material.

In addition to the blanket activities, a modular He gas cooled divertor is being developed within the framework of the EU power plant conceptual study. As ∼10 MW m^{-2} and more needs to be removed, the design is currently based on tiles made of W (∼2000˚C), as well as on structural materials like W-alloy (∼700–1300˚C) and RAF(M)-ODS steel (∼650˚C). Also, for such divertor designs, a suitable materials database must be generated.

Tungsten and its alloys offer high melting points, low vapour pressures, excellent thermal conductivity, high erosion resistance and good thermo-shock properties. Like other bcc materials, tungsten shows a DBTT that is very dependent on its processing and impurities. This complicates the machining (casting, turning, etc) of complex parts. Thus, a low brittleness is essential and desired. In contrast to the usual, widely used deformation processing mechanisms (rolling, drawing, etc), severe plastic deformation (SPD) has shown a significant improvement of the ductility, but it does not prevent re-crystallization at 1200˚C even after short exposure times of a few hours.

Tensile, ageing and long-term creep rupture tests (up to 5000 h) on commercially available W and W–1% La_2O_3 are currently being done in the temperature window of interest, in order to get a preliminary database for divertor designers. To evaluate the fracture toughness of the three different alloys (i) pure W, (ii) W–1% La_2O_3 (WL10) and (iii) potassium doped tungsten alloy (WVM) in the as-sintered condition, CT-specimens with $W = 15$ mm were machined according to the ASTM E399 standard. To pre-crack the specimen a notch with a length of 8 mm was cut with a wire, followed by a cutting step using a razor blade to sharpen the crack. The final fatigue crack was induced by cyclic compression of the samples, 15 000 cycles at an R-value of 0.5. This led to mean fatigue crack lengths of 100 μm for pure tungsten, 130 μm for WL10 and 75 μm for WVM, respectively. Crack formation occurred only in the direction of the pre-notch, crack-branching and propagation perpendicular to the pre-notch was not found. Table 2 shows the fracture toughness values of the alloys that were determined. All values shown in table 2 were valid K_{Ic} values. No R-curve behaviour was found.

Finally, R&D has started to improve the high temperature stability of W-alloys, and irradiation campaigns are being

Table 2. Fracture toughness values for pure tungsten, an ODS strengened W alloy (WL10) and a potassium doped W alloy (WVM).

	K (MPa m$^{0.5}$)		
T (°C)	W	WL10	WVM
25	5.1	4.7	4.7
200	6.1	4.3	4.6
400	10.7	12.8	16.7

launched in Europe to get an initial database at two temperature levels between 650°C and 1000°C.

Acknowledgments

We appreciate all researchers of the European fusion materials development programme, and besides EFDA and the EU commission, also other institutions and associations involved for their ongoing support.

References

[1] King D. 2001 *Conclusions of the Fast Track Experts Meeting* pp 1–5, http://www.fusion.org.uk/culham/fasttrack.pdf
[2] Hazeltine R. 2003 *A Plan for the Development of Fusion Energy* pp 1–82, http://fire.pppl.gov/fesac_dev_path_final_rpt.pdf
[3] Ehrlich K., Bloom E.E. and Kondo T. 2000 *J. Nucl. Mater.* **283–287** 79–88
[4] Gasparott M. *et al* 2002 European Material Assessment Meeting *Report* EFDA-T-RE-2.0 (Garching) pp 1–66
[5] Andreani R., Diegele E. Laesser R. and van der Schaaf B. 2004 *J. Nucl. Mater.* **329–333** 20–30
[6] Ehrlich K., Cierjacks W., Kelzenberg S. and Möslang A. 1996 The development of structural materials for reduced long-term activation *17th Int. Symp. on Effects of Radiation on Materials* ed D. Gelles *et al* (ASTM STP 1270) pp 1109–22
[7] Ehrlich K., Harries D.R. and Möslang A. 1997 Characterization and assessment of ferritic–martensitic steels *FZK Report* 5626
[8] Rieth M., Schirra M., Falkenstein A., Graf P., Heger S., Kempe H., Lindau R. and Zimmermann H. 2003 EUROFER 97—tensile, charpy, creep and structural tests *FZK Report* 6911
[9] Van der Schaaf B., Tavassoli F., Fazio C., Rigal E., Diegele E., Lindau R. and Le Marois G. 2003 *Fusion Eng. Des.* **69** 197–203
[10] Lindau R., Möslang A. and Schirra M. 2002 *Fusion Eng. Des.* **61&62** 659–64
[11] Cardella A. *et al* 2004 *J. Nucl. Mater.* **329–333** 133–40
[12] Rensman J., Lucon E., Boskeljon J, van Hoepen J., den Boef R. and ten Pierick P. 2004 *J. Nucl. Mater.* **329–333** 113–116
[13] Lucon E., Chaouadi R. and Decreton M. 2004 *J. Nucl. Mater.* **329–333** 1078–82
[14] Materna-Morris E., Schneider H.-C., Dafferner B., Rolli R., Romer O. and Möslang A. 2003 Mechanical properties and structural analysis of martensitic low-activation alloys after neutron irradiation *20th Symp. on Fusion Engineering (San Diego, 14–17 October 2003)* (Nuclear Plasma Society) ISBN: 0-7803-7909-8, p 246
[15] Spätig P., Campitelli E.N., Bonade R. and Baluc N. 2004 *Fusion Energy 2004: Proc. 20th Int. Conf. (Vilamoura, 2004)* (Vienna: IAEA) CD-ROM file FT/1-5 and http://www-naweb.iaea.org/napc/physics/fec/fec2004/datasets/index.html
[16] Tavassoli A.-A.F. 2004 EUROFER steel—Demo Interim Structural Design Criteria (DISDC), version 3, private communication
[17] Tavassoli A.A.F. *et al* 2004 Materials design data for reduced activation martensitic steel type EUROFER *J. Nucl. Mater.* **329–333** S.257–62