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SCIENCE DIRECT®
Journal of Nuclear Materials 329–333 (2004) 20–30
journal of nuclear materials
www.elsevier.com/locate/jnucmat

The European integrated materials and technology programme in fusion

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Abstract

The paper reviews the present EU long-term programme, which is focused on the R&D of the materials and key technologies needed for the DEMO reactor. The mission of the EU power plant conceptual study, some major results and its consequences on future materials development are reported. Thereafter, the EU portfolio is presented including: material R&D from ferritic–martensitic steels and enhanced ODS steels to materials for high temperature application (SiC/SiC and tungsten), theoretical modeling of irradiation effects, the unique effort in building a fusion nuclear database on activation and transport files and codes, design activities for TBMs for ITER and DEMO as well as R&D and design performed for IFMIF.

1. Introduction

The EU long-term strategy in the technology R&D programme is based on the assumptions that ITER operation will start after the year 2014 and that, in the meanwhile, a positive decision is taken to build IFMIF. The present time scale foresees the start of the IFMIF EVEDA (Engineering Validation, Engineering Design Activity) phase late in 2004 and subsequently, construction start in 2009/2010 and the start of operation in 2016/2017 with one accelerator line and in 2019/2020 with two accelerator lines. DEMO relevant test blanket modules (TBMs) need to be installed in ITER at the beginning of the hydrogen operation phase. To be fully relevant, TBMs will be designed and fabricated using DEMO relevant materials and technologies. Towards 2020 sufficient information from ITER and IFMIF could allow to start the DEMO final design and licensing procedure.

Material development and development of breeding blanket are highly interacting and have to be closely related in a common strategy. Assuming that the fast track to a commercially fully competitive and environmentally acceptable fusion power plant (FPP) besides ITER includes only one intermediate step, breeding blankets have to be ready for operation two decades from now. ITER provides the first – and may be the only – facility to test blankets under the most realistic neutron environment.

In addition, the machine after ITER could be equipped during its operational life with more then one 'generation' of breeding blanket. Hence DEMO would be started with breeding blankets that rely on technologically moderate extrapolation, well known techniques and materials, but with the potential to be developed towards more advanced concepts suitable for power plant application.

The European strategy foresees helium-cooled blankets with two different breeding options (liquid metal and ceramic breeder), i.e. a helium-cooled lithium–lead (HCLL) and a helium-cooled ceramic blanket with

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0022-3115/$ - see front matter
doi:10.1016/j.jnucmat.2004.04.339

pebble beds (HCPB). Both use the same basic structural material, a reduced activation 9Cr ferritic martensitic steel, called EUROFER.

Being tested as test blanket modules in ITER from day one of operation it is indispensable to have a qualified material that fulfils all design and operational requirements and is fully code qualified for low and high temperature applications in all mechanical properties. Coatings of structural material that act as anti-permeation barriers and as anti-corrosion layers against liquid metals are required.

Technological development is still needed to enhance pebble beds. Activities oriented to improve the beryllium properties mainly with respect to tritium retention and irradiation swelling, the fabrication of ceramic pebbles, modeling of thermo-mechanical behavior of pebble beds as well as verification of activation and transport codes complete the portfolio of R&D needs for HCPB blankets.

For economic benefit the drive to higher temperature operation is indispensable. Therefore, in parallel to the development of 'generation one' breeding blankets, research on alternative materials has to be conducted and it is indispensable to do this from today and with a sufficient budget.

In the long term Europe therefore sees a logical sequence, by supplementing EUROFER with the next generation of advanced ferritic steels based on oxide dispersion strengthening (ODS) dispersing nanosized particles (yttrium oxide being the first choice), and thereafter, and really alternatively, with the development of fibre-reinforced silicon carbide (SiC_f/SiC).

For economic reasons also, at the same time it is necessary to raise the operational temperature of divertors. Gas cooling is the natural choice as in HCLL and HCPB-type blankets. A coolant temperature of 600–650 °C implies operating temperatures in the structure of 650/700 °C to 1200/1300 °C at the high flux/high temperature heat removal part of a divertor. The candidate that potentially fulfils this requirement is tungsten in combination with a supporting back-bone structure of advanced ferritic (ODS) steel.

Combined material and component tests for proof of principle could be envisaged for helium-cooled divertors at the very end of ITER operation.

Advanced ODS steels and SiC material could be tested under operational conditions in 'second generation' breeding blankets in DEMO. The strategy could be to still stay with an (improved) EUROFER type steel as the workhorse and back-bone for the design and make use of the advanced material in steps of increasing functional and reliability requirements. In other words, prior to using ODS steels and SiC_f/SiC as structural material they could be included as FW plating or for insulating purposes.

2. Outcome of the EU power plant conceptual study

2.1. The main plant models

The results and outcome of the power plant conceptual study (PPCS) [1–3] guides to a large extent the direction of the EU long-term programme establishing priorities and coherence in the nuclear component development activities. In particular, the mission of the PPCS was to demonstrate the credibility of the power plant designs considered; the safety and environmental advantages of fusion power production as well as its economic viability. Compared to earlier European studies [4–6], the present designs aim to satisfy economic objectives and the plasma physics basis was updated. As a consequence the parameters of the models differ substantially from those of former investigations.

A limited number of four models, called A–D, has been studied as examples of a spectrum of possibilities, ranging from near term to advanced plasma physics, materials and technology [7–12]. In general, increasing economic attractiveness of a concept goes together with larger extrapolation of material properties and technology, including increased risk that the development could finally fail.

The economics of fusion power improves substantially with increase in the net electrical output from the plant. However, large unit size causes problems with grid integration and poses the requirement for very high reliability. As a compromise, the net electrical output was chosen to be 1500 MWe for all the PPCS models. However, their fusion powers vary due to different efficiency.

Major conclusions on the cost of electricity (COE) are: even the near-term models are acceptably competitive. Depending on the model and on the learning effect, internal COE range from 3 to 12 Eurocent/kW h.

All models have very good safety and environmental features. Studies suggest helium-cooled lithium–lead is probably a very promising additional concept, from the safety, environmental and economic viewpoints. The main parameters of the models are summarized in Table 1.

Recently, the helium-cooled lithium–lead concept has been added as preliminary studies showed that it has its own advantages in comparison to the water cooled concept, especially with respect to safety aspects.

2.2. Effects of the PPCS on the strategy for materials development

The EU programme is focused in the short term in developing the reduced activation ferritic martensitic (RAFM) steel EUROFER. This material will be used as structural material in the two EU test blanket modules (TBMs) to be installed in ITER: the helium-cooled lithium–lead (HCLL) and the helium-cooled pebble bed

Table 1
Main parameters of models considered under PPCS

	→ Increasing extrapolation in physics and technology → Increasing operating temperature			
	Model A	Model B	Model C	Model D
Fusion power (GW)	5.0	3.6	3.4	2.5
Net electrical output (GW)	1.5	1.5	1.5	1.5
Physics	ITER-like, High plasma current		'Advanced physics', Low plasma currents, High bootstrap fraction	
Plasma current (MA)	33.5	28.1	20.1	14.1
Net reactor efficiency (%)	27	43	44	61
Blanket (wall load) (MW/m^2)	2.2	2.0	2.2	2.4
	WCLL, Water cooled	HCPB, He cooled	'Dual coolant', Li–Pb breeder, He cooled FW	Self-cooled Li–Pb
Structural material	EUROFER	EUROFER	EUROFER+ODS-layer+SiC/SiC insert	SiC/SiC
Breeder	90%–Li_6 enriched LiPb	Ceramic pebble bed + Be pebble bed	Li–Pb	Li–Pb
Divertor				
Peak load (MW/m^2)	15	>10	>10	5
Cooling	Water	Helium	Helium	He or liquid metal
Structural material	Cu-alloy (CuCrZr) max. 400 °C	Tungsten at high heat removal and ODS as back-bone		

(HCPB). The blanket solution to be chosen for DEMO will be, with high probability, one of those developed in the frame of the DEMO-test blanket programme.

As an intermediate target, further improvements in power plant efficiency/economics and in waste management appear possible by the use of low activation materials that can operate at higher temperature than the RAFM steels. At present in the EU R&D is mainly focused on ODS steels. As a first step a EUROFER type ferritic–martensitic ODS steel will be developed, that, because of its similar thermo-physical properties, can be combined with conventional EUROFER. Ferritic ODS steels are investigated in a second step as the back-bone structural material of a gas (or liquid metal) cooled divertor aiming at an operational window in excess of 700 °C.

In addition, SiC_f/SiC composites are to be developed for a temperature window in the range of 600–1000/1100 °C. A limited but steadily increasing effort during the 6th Framework Programme (FP6) is also devoted to investigate the characterization of tungsten alloys and their improvement. The development of a structural divertor material is extremely challenging. On the one hand, by virtue of its excellent thermal properties (high conductivity and low expansion coefficient), the upper operating temperature could be as high as 1200/1300 °C. On the other hand, a lower operational window in the order of 650–750 °C could be required for a significant overlap in temperature window with a back-bone ODS steel.

Present EU strategy is to test in ITER a series of blanket modules derived from the two EU reference blanket conceptual designs performed for DEMO. These mock-ups should be equipped with all DEMO-relevant technology for which a significant R&D programme is in progress.

Tests are indispensable in order to validate the computational models and codes for the DEMO design (neutronics; tritium production, inventory and recovery; MHD effects; thermo-hydraulics and thermo-mechanics; electromagnetics) and to qualify them in a fusion environment as well as investigating potential knockout factors for DEMO application.

As a positive decision for ITER implies its availability by 2014, consequently, materials and fabrication processes must be frozen not later than 2006 to allow the fabrication and tests of the final prototypes of TBMs in due time.

3. The new European HCLL and HCPB DEMO blanket concepts

3.1. General consideration

As a consequence of a blanket selection exercise during the first half of the nineties two concepts have been selected as promising for further R&D towards DEMO, the water cooled lithium lead (WCLL) and the helium cooled pebble bed (HCPB). In order to reduce the financial commitments in Framework Programme 6 (2003–2006) it was decided to concentrate the blanket R&D activities for the TBMs to be tested in ITER on a single coolant. This policy keeps open the possibility of adopting both types of breeder materials under study in EU: the solid (ceramic) and liquid (lithium lead) ones. Therefore, the water cooled lithium lead (WCLL) concept has been replaced by the helium-cooled lithium–lead (HCLL) concept.

In order to reach the objectives to install the first TBMs in ITER at the beginning of the operation phase in hydrogen, a three stage programme has been identified:

- First stage (up to 2006): Finalization of the TBM system and auxiliary systems, design, qualification of the most critical technologies, fabrication and testing of small-scale mock-ups. During 2005 materials and manufacturing technologies for TBMs should be selected.
- Second stage (2006–2009): Fabrication and testing of TBM large-scale mock-ups, detailed engineering design of TBMs components, auxiliary systems and instrumentation to be installed in ITER.
- Third stage (2010–2015): Fabrication of the final TBMs system, pre-assembly, acceptance tests and installation in ITER as well as preparation for licensing.

In addition to the development of structural materials and of fabrication and joining technologies, comprehensive irradiation test programmes on breeder and neutron multiplier materials have been launched with the aim to reach DEMO relevant operating conditions (burn-up, dpa, etc.). Particular emphasis is given to tritium production and retention in beryllium and Pb17 Li. Specific out of pile and in pile tests on TBM mock-ups and components are scheduled before the third stage period.

3.2. Design

3.2.1. Common features of both concepts

Design specification for the HCLL and HCPB blanket have been taken from model B of the PPCS, i.e. a neutron wall load of 2.4 MW/m^2 and a peak surface heat flux of 0.5 MW/m^2 are considered. Modules with dimensions at the FW of 2 m $\times$ 2 m placed at the equatorial outboard region have been taken as reference. The design so far includes the first wall (FW), the breeder units (BU) and the manifold system. The main constraints for the design were that both concepts should share the same type of FW and of blanket box and structure. The module is designed to withstand the full coolant pressure in faulted conditions, in particular the case of an in-box loss of coolant accident. This requirement sets up very severe conditions for the construction and could only be managed by a strong segmentation of the entire box. Both concepts use helium coolant at operational pressure of 8 MPa and inlet/outlet target temperatures of 300–500 °C.

The outer shell of the blanket box is made up from a steel plate with internal cooling channels bent, into a U-shape. Inlets and outlets of all channels are located at the radial back of the box. Welded into the box is a stiffening grid of radial–toroidal and radial–poloidal plates. Each grid plate is cooled by gas flowing through internal channels that are fed from the rear. This grid results in cells open at the back with toroidal–poloidal dimensions of about 20 cm $\times$ 20 cm that accommodate the breeder units (BU). The spacing of the grid is designed in order to allow the box's wall to resist the 8 MPa fault condition. The joints of each group of four stiffening plates form a cross that extends into the radial back and is needed for a strong connection of the grid to the module back plate.

Goals in the design include:

- A tritium breeding ratio (TBR) larger than 1 to assure self-sufficiency of the reactor. A TBR larger than 1.14, computed with a 3D neutronics programme, is required to account for the ports, the uncertainties in the nuclear libraries and the Li burn-up during the life time.
- Optimized coolant inlet temperatures and coolant flow schemes to use the full temperature windows of EUROFER in the range of 300–550 °C is envisaged. The lower limit is dictated by the irradiation embrittlement and the increase in DBBT (ductile–brittle transition temperature), the upper limit basically is given by long-term mechanical properties as creep rupture and fatigue–creep interaction, but also by a reduction in static mechanical properties.

3.2.2. Specific design of breeder units

The breeder unit for the HCPB concept has a base plate (to be inserted from the rear) that holds two breeder canisters, providing space for two shallow ceramic breeder pebble beds. The canister walls contain a dense array of internal, radially oriented, rectangular cooling channels. The space left by the canisters is filled with beryllium pebble beds. Pipes taking purge gas to the front of all pebble beds are welded onto the cooling plates.

The requirements for the cooling of PbLi allow a simplified design of the BU for the HCLL concept. Cooling plates with meandering channels for the helium are connected to a back plate. The PbLi fills the free space in the cell entirely. The structure of the box and grid is adapted to the flow of the PbLi inside the module. As the PbLi is only for breeding purposes, the flow can be kept at velocities as low as 1 cm/s, which is quasi-stagnant.

The requirement on tritium breeding governs the minimum depth of the modules: neutronic calculations based on PPCS model B demonstrate the achievement of the target TBR with a breeder zone thickness of about 46 cm for the HCPB and 55 cm for the HCLL, where Li_6 enriched breeding material is assumed.

More detailed descriptions of the BU as well as of the complex construction of the back-wall and supplying structure as well as on fabrication technology and joining are provided in this conference [13,14].

3.3. Gas-cooled divertor concepts

The divertor target is recognized to be a very critical component for tokamak-based reactors. Quite successful R&D has been performed for present-day machines and for ITER. Concepts for these machines can withstand very high heat loads and can be manufactured with sufficient reliability.

However, these concepts are not relevant for a fusion power plant (FPP) because of the associated additional requirements such as high thermal efficiency, high reliability and availability, long lifetime associated with high neutron fluence. It has been recognized that the step between ITER and a FPP is too large; it is therefore suggested to take advantage of the expected intermediate step, such as DEMO, in order to develop an evolutionary solution, in a similar way to that foreseen in the blanket development programme.

The proposed strategy is therefore the following.

Instead of proceeding with a moderately evolutionary development of 'near-term' concepts based on water cooling at relatively low temperature, the effort is to make a step forward to more ambitious 'long term' concepts using helium or liquid metals at high coolant temperature coolants. Tungsten and to some extent SiC_f/SiC are candidate structural materials. These concepts are to be designed to cope with maximum heat flux of 10 MW/m^2.

Tungsten is the best potential candidate together with gas cooling. The suggested T-window is 700/800–1200/1300 °C with uncertainties at both ends of the temperature window. Irradiation effects are widely unknown.

SiC_f/SiC can be considered associated with Pb–17Li cooling especially in case this material combination is already used for the blanket. Appropriate design rules for either material need to be developed. Specific emphasis is required for the fabrication of thin walls (1–2 mm) and the development of dissimilar joints with steels.

The choice of a specific divertor concept is only meaningful in relation with the selected blanket system, because of significant integration and safety constraints. Following the same ideas as during the blanket development, R&D priorities should be given in the next decade to: (i) conceptual design of a whole divertor system to derive representative and critical mock-ups; (ii) fabrication of these mock-ups and performing out-of-pile thermo-mechanical tests for screening (especially for joints technology); (iii) in-pile tests, in particular for the 'near/medium term' concepts.

4. EUROFER – the choice for breeding blankets in early DEMO operation

4.1. Development strategy

Two decades ago the discussion was on austenitic vs. martensitic ferritic steels. The advantages of austenitic steels were their already well-developed technical properties at that time, with acceptable creep rupture strength and good compatibility with the fusion helium environment. After some R&D the swelling behavior for cold-worked material under fission reactor irradiation and the DBTT were no longer a concern. However, it turned out that they are very sensitive to He-induced high temperature embrittlement and they are not compatible with liquid Li– and Li–Pb. This strongly reduces the upper operating temperature. Another concern was the high long-term radioactivity. Attempts to replace Ni by Mn were not successful. Austenitic steel is appropriate for devices like ITER with moderate upper temperature and limited neutron fluence but they have limited potential for future breeding blanket.

The EU material programme is focused in developing the 9% CrWVTa RAFM steel EUROFER (97) [15]. The current specification was defined in 1997. In the IEA frame the US, Japan and the EU co-operated to develop FM steels with reduced activation. Development of EUROFER is based on the experience from this IEA collaboration, in particular from the Japanese F82H development and characterization as well as by R&D conducted in several EU labs (FZK, CEA, CRPP, UKAEA and ENEA). The next step in the development is the definition of EUROFER-II which aims at further improvement of mechanical properties after irradiation, in particular the reduction of the DBTT in the irradiated material. A decision on that can be taken by the end of the years 2005/2006 at earliest when significant PIE results from irradiations up to 40 and 80 dpa, respectively, will be available. TBMs likely will be manufactured using EUROFER-II.

It is the impurities that determine the long-term activity. Hence the third step (EUROFER-III) aims towards a real low activation material with re-cycling times within 100 years. From the EUROFER specification this seems to be technically feasible but has to be proven in industrial heats where the impurity control is extremely difficult to manage unless special clean production lines are available.

4.2. Qualification of EUROFER

The development has two different time scales, i.e. day-one of ITER operation and day-one of a subsequent DEMO-operation each with different operating conditions and loading characteristics. The requirements for DEMO are unequally more demanding with respect to irradiation damage. Nevertheless the material has to be fully code qualified for ITER application. TBM are exposed to highly damaging fatigue–creep interaction mechanism due to the high number of short operational pulses in ITER. This altogether constitutes a very comprehensive test matrix.

The programme in the EU is progressing with irradiations of EUROFER97 to a wide range of radiation damage, starting from 0.3 to 1.0, 3, 5,10, 15, 30 and 70–80 dpa. Low-level irradiations support the analytical study of the build-up of radiation damage and its theoretical explanation.

A set of irradiation activities up to 2.5 dpa in the temperature range of 60–550 °C (with main emphasis on irradiations at the 300 °C level) is in progress in order to generate an engineering database requested for TBM design activities and licensing procedures.

In particular in view of DEMO application high dose level irradiation experiments comprising more than 300 specimens have been launched in the fast breeder reactor BOR 60 in Dimitrovgrad, Russia, for neutron damage up to 70–80 dpa at 320–350 °C in an accelerated time frame. This campaign is supplemented by irradiations in HFR, Petten, in the temperature range 250–450 °C up to 15 dpa, where besides EUROFER97 also a few EUROFER-ODS alloys are tested.

Irradiation experiments with the focus on specific manufacturing processes or joining techniques (EB-weld and HIPped) are partly underway and will be complemented if needed to analyze particular TBM design issues.

4.3. Some results of mechanical out-of-pile characterization

The out-of-pile characterization of the base materials (plates, bars, tubes) and joints is almost completed. Mechanical and micro-structural characterization of the ‘as-received’ as well as of material thermally aged up to 10 000 h are compiled [16] in a database and assessed. Moreover, data are evaluated according to the DISDC/ISDC (DEMO Interim Design Criteria/ITER Structural Design Criteria) rules to derive ‘allowable’ stresses and strains for detailed engineering [17].

Some results are highlighted:

(a) Static properties: The mechanical properties, i.e. yield strength, ultimate tensile strength, total elongation measured up to 650 °C are expected to be similar to those of F82H and the OPTIFER-family.
(b) Creep properties: Similar trends as for other RAFM steels have been found at test temperatures in the range of 450–650 °C. Emphasis is given to experiments in the order of 500–550 °C rather than accelerated tests at higher temperatures.
(c) Ductile-to-brittle transition temperature: Depending on the specimen size and evaluation procedure values in the range of −70 °C are found.
(d) Fatigue–creep interaction. Fatigue results (isothermal) are very good. Hold times of 400 s might result in reduction of endurance limit by a factor of 2. This is still in line with safety factors applied in design rules. However the effect of longer hold times and the result that compression hold is more damaging than tensile hold needs to be elaborated in more detail.

4.4. Irradiation performance of low temperature irradiations

Post-irradiation experiments (PIE) of the low to medium dose irradiations at 300 °C up to 3 dpa at BR2 (Mol) and up to 12 dpa at HFR (Petten) [18] show coherent data.

The increase by n-irradiation of the yield stress of 9CrWVTa heats is independent of the exact composition and type of product form if only wrought materials (i.e. with an appropriate microstructure) is considered. The hardening follows consistent a logarithmic trend down to 0.2 dpa (Fig. 1). No evidence of saturation is seen so far. Data from irradiations at 325–350 °C fall much below the curve and are not comparable.

The DBTT shift, determined from some 360 single KLST specimen, with an average of 10 specimens per material/dose combination [19] is summarized in Fig. 2. The reference EUROFER 8 mm plate, 14 mm plate, 100 mm shows better properties than F82H.

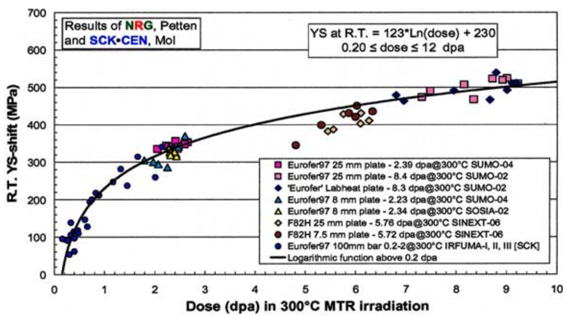

Fig. 1. Increase in yield strength as a function of dose.

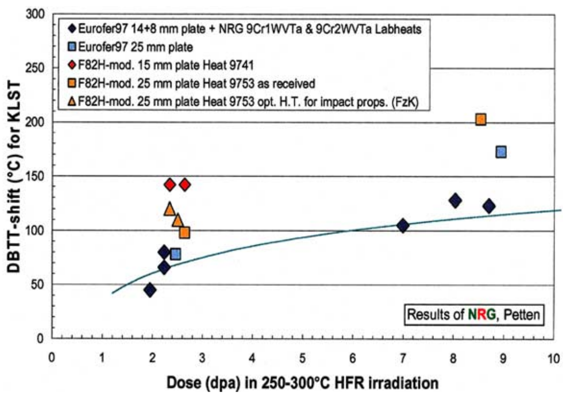

Fig. 2. Shift of the ductile–brittle transition temperature as a function of dose.

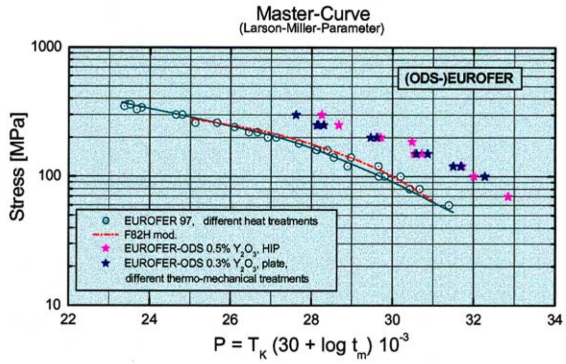

Fig. 3. Creep properties of EUROFER compared to ODS-EUROFER.

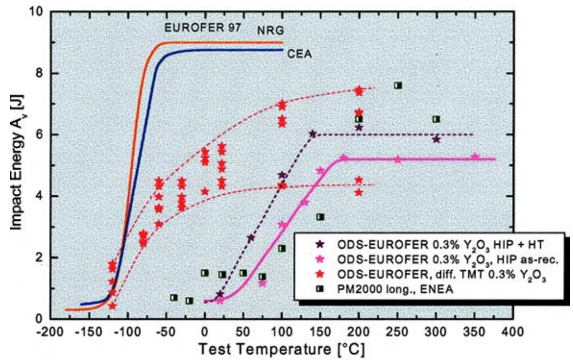

Fig. 4. DBTT properties of EUROFER compared to ODS-EUROFER.

5. ODS steels

5.1. EUROFER ODS

An obvious limitation of EUROFER is the reduction of creep strength above 550 °C. In the EU the general approach so far is to use the EUROFER 97 alloy composition as matrix and to add Y_2O_3 dispersion in the range of 0.3–1 wt%. The activation calculations showed that the Y_2O_3 dispersoids do not worsen the radiological behavior. An improved creep rupture strength has been reported by several groups (this conference) that extends the upper temperature limit of application to 650 or even 700 °C. The present work in European laboratories is concentrated on the optimization of the production route via powder metallurgy mixing of the constituents, mechanical alloying and hot isostatic pressing (hipping). Next year one of the three fabrication routes followed will be selected and a larger heat of 50 kg will be produced aiming at the fabrication of semi-finished products.

Whereas the first results of hipped EUROFER-ODS variants are encouraging with respect to tensile and creep (Fig. 3) the observed impact properties are still not completely satisfactory. For some production route the ductile-to-brittle transition temperature is as low as RT but for an application in highly loaded FW structures of a blanket the upper shelf energy needs still some improvement (Fig. 4) [20].

Future investigations will be concentrated on a refinement of chemical composition in order to retain a fully martensitic structure and to eventually improve the corrosion resistance. A further aspect of development followed by the group at CRPP, Switzerland, is to retain a finer particle distribution by a proper stabilization with Ti precipitates and to reduce grain boundary impurities/precipitates introduced by the powder metallurgy production route. Further work is also directed towards the optimization of the present production routes (i.e. the proper combination of thermal and mechanical treatments).

5.2. Ferritic ODS

Whereas the ODS EUROFER fabrication routes were chosen to obtain similarity in thermal and mechanical properties with EUROFER in order to achieve easy joining, the next step towards higher operating temperatures needs to proceed with high-chromium-containing ferritic steels ODS, where first activities will start in 2004.

In periods of limited budgets and reduced skilled staff as well as lack in irradiation facilities, collaborations on ODS related activities in the frame of the IEA-Implementing Agreement should be reinforced.

6. Structural materials for high temperature application

6.1. SiC_f/SiC

SiC_f/SiC ceramic composites have attractive properties for application in fusion power reactors. They have acceptable mechanical properties for application up to at least 1000 °C and good activation properties if impurities are controlled. Significant progress has been made in the last decade. The progress relies mainly on the availability of almost stoichiometric fibres with higher thermal conductivity, higher thermal stability. A strong interaction with the manufacturer is requested as the properties of the composite can be tailored to the specific application by choosing the appropriate fibres architecture, fibres to matrix interface and densification processing route.

The basic properties that need to be proven (and have potential for optimization) are the radiation resistance of physical and mechanical properties. The inherent brittleness and anisotropy of SiC_f/SiC composite is an inherent property and requires appropriate guidelines for designing components, and rules have to be adapted to these properties. In particular, a basic understanding of micro-structural evolution, change in mechanical properties and critical damage mechanism has to guide the development of optimized composite materials and better designs.

6.1.1. Status and future activities

At present, 2D and 3D EU reference SiC_f/SiC with improved properties are produced. The choice of the manufacturing route, CVI (chemical vapour infiltration) vs. PIP (polymer impregnation and pyrolysis), will be done on the basis of the best results obtained in terms of thermal conductivity and mechanical properties [21]. The metallurgical and mechanical characterization of the new materials will be systematically organized in a EU database. The irradiation behavior at low (point-defect swelling regime) and high (void swelling regime) temperature and their compatibility with Pb–17Li up to very high temperature will be investigated up to 5 dpa starting in spring 2004.

Technological issues are: Compatibility tests with Pb17Li at temperatures above 800 °C and under flowing conditions are still requested. Reliable brazed joints have been achieved in EU under laboratory condition. Infiltration of the braze within the composite is now well controlled. Joining of large-scale plates will be considered at a later stage.

R&D activities on SiC_f–SiC are performed in EU, Japan and US. Exchange of information is held through an IEA agreement. As an example the EU is performing irradiation at the OSIRIS reactor at CEA, France, at two temperature levels, 600 and 1000 °C. Japan contributes by developing and providing advanced SiC_f/SiC composites and in particular those based on the LPS/NITE (liquid phase sint/nano-infiltration and transient eutectic-phase) process. JA also characterizes the samples provided by the EU and produced by EU industry. The EU is in charge of characterizing the SiC_f/SiC composites after neutron irradiation at 600–1000 °C, and also to test their compatibility against Pb17Li and to assess their joinability.

Still, a stronger integration of this R&D remains necessary to address the effort to the fundamental issues of SiC_f–SiC composites.

6.2. Tungsten

Tungsten alloys have been selected as primary materials candidates in the EU for divertor structural and armour application. The main issues include: (i) low fracture toughness at low temperature. (ii) The ductility (facture toughness, deformability, impact toughness) is very sensitive to production history, alloying elements, temperature, loading direction, irradiation and irradiation temperature. There exists no systematic study with respect for structural application other than to formability. (iii) Generally speaking, W alloys are not used as structural materials so far in large components at industrial scale and with the reliability requirements set-up in FPP. So to understand their behavior under combined thermal and mechanical loadings is essential. This will give ideas how to deal with a material of reduced ductility. Guidance and rules on damage-tolerant design as well as damage-tolerant concepts are not really available. (iv) Finally, knowledge on neutron irradiation effects (especially at higher temperatures) and swelling behavior is very limited.

At present, investigation of the possible improvement in fracture behavior of tungsten based materials by forming a submicron or nano-grained microstructure by means of severe plastic deformation (SPD) are performed at OEAW, Austria. The task includes the production of small batches of forged W, W–5Re, W–1% La_2O_3 with different grain size and low impurity content and, subsequently, the determination of their re-crystallization temperature as well as the selection and production of the most promising samples for further characterization in irradiation experiments. First results show that W with small additions of potassium and W–1% La_2O_3 are the more promising candidates considering their thermal stability.

In the next future the following activities are considered to have priority:

- Further investigations to improve, produce, characterize different W alloys. Emphasis will be put on nano-structured materials, with low interstitial content, to improve the radiation resistance. Metallurgical, physical and mechanical characterization are planned in 2004.
- Medium and high temperature irradiation (600/1000 °C) will be performed up to 5 dpa in the OSIRIS reactor, CEA, France, to investigate potential range of application for structural application.

- Set up of an initial fusion relevant database to support DEMO helium-cooled divertor design. In reviewing existing data, is recognized that additional mechanical characterization (e.g. creep, fatigue) is badly needed to perform even basic engineering analyses of gas cooled divertor designs.

7. Accompanying programmes

7.1. Nuclear data

The nuclear design of fusion devices such as ITER, Demo and IFMIF rely on the results of neutronics calculations. A well-qualified nuclear database and validated computational tools provide the basis for reliable neutronics and activation calculations and the assessments of the associated uncertainties. Supporting experiments are required for fusion design calculations. Analyses include neutron and photon transport calculations to provide the neutron/photon flux spectra which then form the basis for the calculation of nuclear responses of interest when convoluted with related nuclear data. Special emphasis is put on high-quality data around 14 MeV. Dedicated computational tools and data are required for neutronics calculations of the IFMIF neutron source. These tools must be capable of simulating the transport of neutrons generated by Li(d,xn) reactions and of photons produced both in the lithium target and the material test assembly. Cross section data must be provided over the whole neutron energy range of IFMIF, which extends up to 55 MeV.

Within the EFF (European Fusion File) and EAF (European Activation File) projects [22–24], the EU is conducting a unique effort to collect and analyse nuclear data for fusion technology applications. This effort has led to the development of nuclear data libraries such as EFF-series and EAF-series tailored to the ongoing and varying needs of the EU fusion programme. The evaluation includes the needs for nuclear analyses of materials applied in ITER-TBM, shield modules, vacuum vessel, plasma facing components and super-conducting magnets. For the TBM as an example the materials to be considered include Be, Li_4SiO_4, Li_2TiO_3, Pb–Li, and EUROFER (Fe, Cr, W, Ta, V, Mn, C, ...). The main elements have been already evaluated and for the major elements considered in the fusion device internal components the nuclear data uncertainties have been significantly reduced and are typically no longer critical. Impurities play a major role on the dose rate at long times (N and Ni for EUROFER): some additional measurements are needed to investigate the effects of impurities with higher accuracy. The to-do-list in the next three years includes Ta, W, Pb at high priority and next Be, Li, Si, O, C are on the list.

Benchmarking of the data for complex geometries is needed: use of existing steels will be possible for further benchmark measurements with 14 MeV neutrons. Assessment of HCPB as the most critical concept (high fraction of Be) was started with the aim to reduce uncertainties in TBR. Mock-up design is under way.

7.2. Multi-dimensional modeling

In general there is the need for increasing theory support in the fusion materials development:

- It could be most helpful in understanding the mechanisms.
- It would help to settle the validity of extrapolating the ranges of experimental data and allow to limit the costly experimental work in the irradiation devices to the essentials.

The neutron damages in materials is strictly dependent on the neutron energy spectrum; in fusion conditions high He production in relation with damage production is a major issue. IFMIF will be the facility designed to complete the validation process of the materials to be used in DEMO. On the other hand, considering the relatively small irradiation volume, IFMIF will not be sufficient to fully satisfy all irradiation requirements for DEMO design. It is essential therefore to develop models, tools and database in order to correlate irradiation results obtained in different devices (fission reactors, spallation sources, IFMIF, ITER).

A co-ordinated long-term programme focused on the study of the radiation effects in the EUROFER steel under fusion relevant condition has been started in 2002 and continues with increasing effort and budget.

The main activities include: ab-initio inter-atomic binding energy and PKA evaluations; molecular dynamics calculations (displacement cascades, defect accumulation, interaction with impurities and precipitates, transmutations, grain boundaries and dislocations); kinetic Monte-Carlo calculation (time and temperature dependent evolution of micro-structure and defect accumulation).

To qualify these models, specific ion irradiation experiments leading to high He and H production at clearly defined temperature and damage levels are foreseen.

Emphasis is given to guide the programme towards problems inherent in alloys. In particular the development of Fe–Cr potentials has priority. Tests on He-implantation samples are the base for comparison of model prediction to experiments and to intercom pare analytical against numerical predictions (e.g. rate-theory vs. object and event Monte-Carlo simulations). At the atomic level vacancy–dislocation interaction with and without helium are in the focus as well as the effect of grain boundaries and in particular the effect of grain size.

8. IFMIF

It is well recognized and widely agreed that an appropriate high energy, high intensity neutron source is required to test and verify material performance when subjected to extensive neutron irradiation of the type encountered in a fusion reactor. Exploratory work is very important since it can guide the final selection of the most promising materials for commercial fusion reactors. Even more important, the calibration of results from simulation irradiations in presently used fission reactors and accelerators as well as the generation of engineering data for licensing of FPP are additional tasks for this facility. The D–Li neutron source IFMIF (International Fusion Materials Irradiation Facility) is considered the best choice to test within a realistic time scale all the materials needed for the DEMO fusion reactor design and licensing. The history of the IFMIF project, the current status and the conceptual design report as conclusion of the Key Element Technology Phase (KEP) was widely discussed during this conference [25].

The IFMIF parties (EU, JA and US) agreed that the IFMIF project, after KEP, is ready to enter in a next phase called EVEDA (Engineering Validation Engineering Design Activities). The EVEDA must be considered as a distinct phase differing from the activities implemented up to now under Annex II as it is a project-oriented activity and it requires a centralized co-ordination by a joint team of limited size. Therefore a new Legal Framework in the frame of the IEA implementing agreement is now considered for its implementation.

9. Conclusions and outlook

The EU programme is on a good track to reach the goal to test DEMO relevant breeding blanket design in ITER. It includes a comprehensive portfolio of theoretical modeling and development of materials, from FM steels to advanced materials, design activities for TBMs for ITER and future fusion power plants with particular attention to interface and interaction problems, IFMIF R&D and design. Activities are organized in projects in view of obtaining a well co-ordinated effort. The financing of the EU programme through four years Framework Programmes allows assuring continuity to the activities by long-term programme.

In view of international discussion on a fast track to fusion, which would in our opinion, require an increased effort to be achieved, there is a need for increased international collaboration aside of ITER and IFMIF in technology and in particular in materials development. Scientific and technological cooperation, as already promoted through IEA collaborations, should be enhanced.

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