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Invited Paper

The long term development of fusion materials in the EU: achievements and open issues

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The long term development of fusion materials in the European Union for the construction and operation of future fusion power reactors focuses on structural materials such as Reduced Activation Ferritic Martensitic (RAFM) steels, of which the EU reference material is EUROFER, Oxide Dispersion Strengthened (ODS) RAFM and RAF steels, SiC_f/SiC composites, tungsten (W) and W-alloy and functional materials such as Li based ceramics, liquid Pb-15.7at%Li, Be, plasma facing materials and, anticorrosion and antipermeation layers. The structural materials are developed for specific applications and have to satisfy specific requirements with respect to radiation resistance, low activation, high creep strength and good compatibility with all contact media in an as wide as possible operational temperature window. The functional materials should not only have some of these properties, but must fulfill the required functions of: tritium breeding, neutron multiplication, efficient heat transfer, low corrosion and a limited hydrogen permeation rate. An overview of the status of present developments, recent achievements and open issues of these fusion materials is presented here.

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1. Introduction

Life on earth very much depends on the energy from the sun and thus on fusion energy. The regenerative energy sources, wind, water, waves and bio-mass, are all driven by solar power. Coal, gas and oil also come from solar energy collected and stored over millions of years. So it is not too surprising that one of the dreams of mankind is to generate and harvest this energy source on earth. In recent decades a serious effort was made to perform the R&D required for the operation of a future Fusion Power Plant (FPP). Despite considerable progress being made in the last decades (e.g. the triple product - a measure of the quality of plasmas in fusion devices – showed the same steep improvement as the memory density in chips of the semiconductor industry), the realisation of a FPP seems still to be as far away as it was 30 years ago.

In 2006 seven international partners (China, the European Union (EU), India, Japan, the Russian Federation, South Korea and the United States) agreed to embark on the construction and operation of a tokamak type fusion reactor called ITER (International Thermal Experimental Reactor) in Cadarache, France, which is planned to be the final step before the design of DEMO, the first demonstration reactor for fusion power generation, although with low availability. This undertaking is the largest international R&D agreement apart from the International Space Station (ISS) and shows the level of importance attributed by these countries to the development of fusion power. The use of fusion energy should offer considerable advantages: the effective fuel components, deuterium and lithium, are almost inexhaustible on earth and are more or less homogeneously distributed. Any disturbance of the fusion plasma causes its extinction. Emission of green-house gases will not occur. Transuranium nuclides are almost not produced and related proliferation issues play no role. Long-term radioactive waste can be minimized by proper tailoring of the elements used in the materials of a FPP. All the accident scenarios so far studied in FPPs do not require evacuation of the public.

In the next step, ITER, the remaining plasma physics and most of the technology issues shall be addressed. ITER is an experimental test facility and is not foreseen to produce electricity. This shall be done in the following machine, DEMO.

This paper will focus on the special material issues of future FPPs such as the need for high radiation resistance, low residual activation, high creep strength and good compatibility with all contact media in an as wide as possible operational window for the achievement of high thermal efficiency and highlight open issues. The most promising structural materials [1, 2, 3, 4] are: Reduced Activation Ferritic Martensitic (RAFM) steels, Oxide Dispersion Strengthened (ODS) RAFM and RAF steels, SiC_f/SiC composites and tungsten-alloys.

2. The Fusion environment

In the fusion reaction of deuterium and tritium to be used in the first FPPs 3.5 MeV alpha particles and 14.1 MeV neutrons will be produced. The energy of the He nuclides is used to heat the plasma, whereas the 14.1 MeV neutrons transfer their energy into the Breeding Blankets (BBs) of the fusion reactor. The purpose of these BBs is i) to produce tritium by fission of Li by these fusion neutrons to achieve tritium self-sufficiency, ii) to deliver the heat needed for electricity production and iii) to shield the vacuum vessel and magnets. The structural as well as the functional materials (plasma facing materials, tritium breeding materials, neutron multipliers, anticorrosion/ antipermeation layers, etc.) used in these BBs undergo changes generated by these high energy fusion neutrons: activation, transmutation, displacement cascades, subcascades, point defects and more globally the evolution of the microstructure produced by the surviving defects and transmutation products. Displacement rates in the range of 30-40 dpa (displacement per atom) per year will occur in FPPs compared to present European Material Research Reactors (MTRs) with approximately 7 dpa/ fpy. In addition, in FPPs, due to the higher neutron energy the production rates for He and hydrogen are around 12 and 45 appm/dpa, respectively, and thus again far higher than in MTRs. In addition, the plasma facing materials have to withstand a high thermal flux (averaged value: 0.5 MW/m^2) from the plasma requiring a special design with optimised forced cooling.

3. Structural materials for FPPs

In the Power Plant Conceptual Studies (PPCS) [5] four different FPP concepts are discussed with their BBs: Model A with a Water Cooled Lithium Lead (WCLL) BB, Model B with a Helium Cooled Pebble Bed (HCPB) BB, Model C with a Dual Coolant (DC) BB and Model D with a Self-Cooled (SC) BB. Recently a fifth concept was studied: Model AB [6] with a Helium Cooled Lithium Lead (HCLL) BB.

3.1. Reduced activation ferritic/martensitic steels

The development of Ferritic and Ferritic/Martensitic steels started approximately 20 years ago based on the experience gained in fission reactors. These steels exhibit greatly superior swelling [7] and embrittlement resistance [8] in comparison with austenitic steels at temperatures between 400° and 550ºC. In Europe the RAFM reference steel for fusion applications (e.g. in Models A, AB, B and C) is EUROFER, a 9% CrWVTa alloy with W, V, Ta content in the wt% range of 1.0-1.2, 0.15-025 and 0.10-0.14, respectively. Reduced activation of these steels is achieved by replacing the elements Ni, Mo and Nb with Cr, W and Ta, respectively. Extensive R&D on EUROFER was performed in the last years. All mechanical and physical properties of virgin EUROFER are collected in a database [9]. Many irradiation campaigns [10, 11, 12] were performed in European MTRs and fast neutron reactors leading to the conclusion that the irradiation induced degradation of mechanical properties (hardening, ductility, embrittlement, fracture toughness) in EUROFER is in general less severe than in 9% Cr-1% Mo steels. No clear limit of the degradation under continued irradiation has been observed until now. Recent annealing experiments of EUROFER at 550 °C after irradiation up to 40 dpa showed that the original properties (tensile strength and Ductile to Brittle Transition Temperature (DBTT)) could be almost fully recovered [13]. An important question is whether this recovery will be achieved in the presence of He concentrations as expected in FPPs.

Other major issues for EUROFER are i) the limited tensile and creep strength above 550 ºC and softening under cyclic loading, ii) the additional influence of the gaseous transmutation products He and hydrogen in FPPs, not accounted for in MTR irradiation campaigns and iii) joining techniques. Recently, irradiation campaigns were started to understand the evolution of the mechanical properties of EUROFER welds and joints during irradiation.

EUROFER will be used for the first time as a construction material for a large nuclear component in the so-called Test Blanket Modules (TBMs) to be installed in ITER from the first day of operation.

3.2. Oxygen dispersion strengthened steels

ODS Ferritic/Martensitic and Ferritic steels are being developed as they offer the prospect of improved strength and creep resistance compared to the original materials at higher temperatures up to 650 and even 750 °C, respectively, to allow higher thermal efficiency whilst aiming for keeping good properties (fracture toughness and DBTT) at the lower end of the temperature window. In the EU 0.3wt% Y_2O_3 powder is added to atomized EUROFER and the ODS EUROFER steel produced by i) mechanical alloying, ii) HIPping and iii) thermal/mechanical treatment (rolling or hot extrusion). Good progress has been achieved for the EUROFER type ODS steels. Further improvement, especially of the production routes, still seems possible. The first irradiation campaigns of ODS EUROFER steel have been completed and post irradiation experiments will be started soon. Laboratory scale heats of ferritic ODS steels have also been produced [14] and are being examined. Irradiation results [15] indicate that the oxide particle may act as trapping centers for He thus mitigating the He embrittlement.

The major issues for further ODS development are i) optimization of the production processes and better characterization of virgin behaviour, ii) the investigation of the irradiation and temperature stability of the micro- and nanostructure in ODS, especially of the oxides and iii) full characterization of all mechanical and physical properties after irradiation. A further open issue is the use of EUROFER ODS as a structural material because conventional welding via melting leads to severe changes of the microstructure. Friction stir [16] or diffusion welding processes may be used in future. Diffusion welds of dissimilar materials (EUROFER and ODS EUROFER) were successfully produced [17].

3.3. SiC_f/SiC composites

SiC_f/SiC composites are proposed as functional and structural material in the DC and SC LiPb BB concepts, respectively. These concepts offer a high technical attractiveness due to high temperatures of up to 700 °C and 1100 °C and net efficiencies of 0.42 and 0.60, respectively. The use of SiC_f/SiC composites [18] in this context is based on their good mechanical strength at these high temperatures, low activation at short and medium terms, expected good compatibility with liquid LiPb, etc.

Considerable improvements were made recently as a consequence of the development of i) stoichiometric and crystalline SiC fibres, ii) advanced fiber/matrix interfaces and iii) the chemical vapour infiltration (CVI) process. As CVI leads to porosities in the matrix, “closed” SiC_f/SiC composites are produced by surface coating using chemical vapour deposition (CVD). The EU fusion programme recently focused on the production of large (20 cm x 20 cm x 0.4 cm) 2D and 3D SiC_f/SiC plates using commercially available high quality SiC fibers and the production steps mentioned above. Irradiation campaigns of SiC_f/SiC composites have just started in the temperature ranges of interest. For the future use of SiC_f/SiC material as so-called flow channel insert in the DC concept, a contract for the production of 10 cm x 10 cm x 0.4 cm plates was recently placed with European industry. These SiC_f/SiC flow channel inserts require very low thermal conductivity through the thickness to thermally isolate the fast flowing Pb-15.7at%Li inside the flow channel from the more stagnant one in contact with the structural material EUROFER and very low electrical conductivity to reduce MHD effects. The mechanical requirements are less demanding than for SiC_f/SiC used in structural applications. Thus the flow channel inserts are easier to produce.

The major issues still to be addressed for SiC_f/SiC composites as structural materials in the SC concept are i) structural stability under high thermal loads, ii) joining techniques, iii) leak tightness, iv) corrosion resistance in the presence of Pb-15.7%Li at high temperatures and v) general behaviour under irradiation conditions (swelling, influence of the transmutation products He and hydrogen, reduction of thermal conductivity, etc.).

3.4. Tungsten alloys

Tungsten alloys are discussed as candidate materials for structural applications in high heat flux and high temperature removal units in He cooled divertor concepts because of their high melting point, high thermal conductivity, high strength, high creep resistance and low vapour pressure. They can also potentially be used as plasma facing materials. Their shortcomings are: inherent low fracture toughness which depends on many factors (such as purity, microstructure, production history), low ductility, DBTT well above room temperature and the high cost of fabrication. To gain a better understanding of these alloys more systematic studies were recently started in the EU. They focus on obtaining a better understanding of the basic failure modes in W-alloys and on the production of fine grained materials to improve ductility and fracture toughness by means of severe plastic deformation (SPD). Typical SPD techniques are high speed hot extrusion and high pressure torsion (HPT). The hope is that by alloying with Re, La_2O_3 or K, nano-structured materials can be produced, which show a high resistance to recrystallization at very high temperatures.

Many mechanical properties of these W-alloys are not known under irradiation. Joining of tungsten to other materials, e.g. to RAF steels in the high heat flux components is a challenge as i) the joining with a pressurized component (10 MPa He) is required, ii) the materials are very dissimilar with respect to physical properties (thermal expansion coefficient is ~4-5 10^{-6}/K for tungsten compared to ~12 10^{-6}/K for steels at 870 K, the Young’s moduli are 300-350 GPa and 170-180 GPa, respectively, at 870 K) and iii) both materials are chemically reactive and form very brittle Laves phases at the interface.

In conclusion: the development work on W alloys can be summarized as presently raising more questions than answers.

4. Functional materials

4.1 Tritium breeding materials

Li based materials (liquid Pb-15.7at%Li and Li ceramics: Li_4SiO_4 and Li_2TiO_3) are the breeding materials proposed in the HCLL and HCPB concepts. In the case of the ceramics the materials are used in the form of well defined pebbles in pebble beds. Production processes were developed and the purity, the mechanical and thermal properties, the phase stability and the tritium release determined as a function of temperature and other gases, e.g. H_2 to support isotopic exchange, in the He purge gas. The low solubility of hydrogen in Pb-Li [19] plays a fundamental role as it determines the driving force of tritium for permeation through EUROFER into the He coolant.

4.2 Multiplier materials

Neutron multiplier material is required to help to achieve the tritium production ratio of 1.1. In the case of the HCPB concept the multiplier materials are 0.1 cm diameter Be spheres produced by the rotating electrode method. The volume needed by the Be is about 3 to 4 times larger than for the Li-ceramics. The quality of the Be spheres was assessed with respect to purity, mechanical properties, tritium inventories, tritium releases, oxidation in air and reduction of water steams to hydrogen. Presently the development shifts from pure Be to Beryllites ($Be_{12}Ti$, $Be_{12}V$) which promise a higher tritium release at lower temperatures and show safer behaviour with respect to oxidation and reduction of H_2O to H_2, but are more difficult to produce due to their brittleness.

4.3 Antipermeation and anticorrosion layers

In DEMO approximately 500 g of tritium will be produced in the BBs per day. Due to the high temperatures and thin EUROFER wall thicknesses between the purge gas of the tritium extraction system or the Pb-Li and the He coolant considerable amounts of tritium will permeate into the He coolant as hydrogen permeates easily through EUROFER. Coating of the EUROFER surfaces with materials of extremely low hydrogen permeability (e.g. α-Al_2O_3) can reduce the tritium permeation by orders of magnitude [20]. The capability to produce those layers on all inner surfaces of the BBs and their long-term stability under true operation conditions must be addressed. In Pb-Li corrosion rates of EUROFER [21] of 90 µm/y were measured at 480 °C and for flows of 30 cm/s. Anti-corrosion layers of W produced by chemical and physical vapour deposition techniques are being tested in Pb-Li loops. Further R&D is needed to improve the layer quality. The optimum solution would be to develop anti-permeation and anti-corrosion layers in one coating.

5. Irradiation facilities

The above discussion shows that the presently available MTRs and fast reactors can only give a non-conservative limit for the real degradation of the mechanical and physical properties in a fusion environment because the higher transmutation and activation rates caused by the more energetic fusion neutrons are absent in fission reactors. On the other hand, no difference in the annealing stages of point defects produced in fast fission reactors or by 14.1 MeV neutrons was observed [22]. The high energy displacement cascade produced by a high energy fusion neutron splits into many subcascades produced by knock-on atoms of approximately 20 keV [23]. A direct and important conclusion of these two finding is that the degradation of the mechanical and physical properties only by the ballistic effects of the 14.1 MeV neutron can be well simulated by irradiation in fast fission reactors as long as the new activation and transmutation channels in connection with the high neutron energies are not taken into account.

For the validation of the materials used in future FPPs a facility with a fusion similar neutron spectrum is required. Such a facility is the proposed International Fusion Materials Irradiation Facility (IFMIF) [24] which has presently entered the so-called EVEDA (engineering validation and engineering design activities) phase under the Broader Approach, a bilateral agreement between Japan and EU to support commonly agreed important fusion projects in addition to ITER. In IFMIF neutrons will be produced by injection of 40 MeV deuterons into a fast flowing Li stream. Neutrons with energies up to 60 MeV will be produced. Neutronics calculations show that this facility will have a very similar He to dpa ratio as in fusion devices. In a limited volume of half a liter 20 to 55 dpa per full power year can be produced which allows accelerated testing. Another facility to determine the evolution of the microstructure in the presence of He and hydrogen is JANNUS [25], a multiple beam facility allowing the simultaneous implantation of e.g. He and/or hydrogen and of damage (dpa's). In this way thin samples with different He, hydrogen and dpa levels can be produced quickly and their microstructure observed by transmission electron microscopy (TEM), in situ with implantations and without any activation.

6. Modelling of radiation defects

Modelling the radiation induced changes in the microstructure and the accompanied changes of macroscopic properties is nowadays a very important tool – and of growing importance. Considerable progress was recently achieved in modelling thanks to the density functional theory (DFT) approximation which allows the determination of the energetics of defects and defect clusters in irradiated materials with high accuracy. These values are then used as the basis for various rate theory calculations or kinetic Monte Carlo and lead to the understanding of the evolution of the microstructure. They are also the basis for fitting realistic empirical interatomic potentials for molecular dynamic simulation of extended defects such as displacements cascades and dislocations. A future final goal is to assess possible swelling under fusion relevant conditions and calculate the macroscopic mechanical properties from the evolution of the microstructure.

Modelling has contributed considerably to the understanding of the irradiation effects in solids and will be used to interpolate the numerous irradiation measurements performed in various irradiation facilities in the past. It can be used to optimize the number and type of samples to be irradiated in the limited volume of IFMIF and to extrapolate to the conditions of FPPs. It can be argued that the modeling activities will soon pay back the effort and budget presently spent for them by reducing the number of expensive irradiation campaigns and post-irradiation examinations (PIE). The above mentioned facility JANNUS will be extensively used as validation tool for the modeling. One of the present and future issues of modeling will be to design model oriented/engineered experiments which allow perfect validation of modeling tools by the use of so-called model alloys. In the European fusion community a dedicated effort is made to explain the behavior of EUROFER under irradiation. New important results were achieved concerning phase stability and He diffusion and clustering in model systems such as Fe-Cr and Fe-C [26].

7. Conclusions

The largest part of the current and future materials development program in the EU is and will be dedicated to characterization of EUROFER after irradiation, especially of joints, the development of improved welding and HIPping techniques for the assembly of the TBMs and the production of design rules. The ODS activities have recently been expanded to produce and characterize first ferritic laboratory heats. Irradiated ODS EUROFER samples are cooling down for PIEs. First attempts were recently made to fabricate SiC_f/SiC plates for the SC BB concept. In the case of W a basic R&D program is performed to improve ductility and fracture toughness.

The irradiation campaigns in MTRs and fast reactors suffer from a low He/dpa ratio in comparison to fusion. Via experiments in JANNUS, which permits the simultaneous implantation of He, H and dpa's, the evolution of the microstructure under fusion revelant conditions can be studied, but IFMIF is urgently needed and mandatory to determine the change of the macroscopic material properties under irradiation. The effort in modeling radiation effects in EUROFER should be increased as remarkable results, not thought possible a few years ago, were recently achieved.

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