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Materials for the plasma-facing components of fusion reactors

H. Bolt [a,*], V. Barabash [b], W. Krauss [c], J. Linke [d], R. Neu [a], S. Suzuki [e], N. Yoshida [f], ASDEX Upgrade Team [a]

[a] *Max-Planck-Institut für Plasmaphysik, EURATOM Association, Boltzmannstrasse 2, D 85748 Garching, Germany*
[b] *ITER-IT, ITER-JWS-Garching, D 85748 Garching, Germany*
[c] *Forschungszentrum Karlsruhe, IMF-3, EURATOM Association, P.O. Box 36 40, D-76021 Karlsruhe, Germany*
[d] *Forschungszentrum Jülich, IWV-2, EURATOM Association, D52425 Jülich, Germany*
[e] *Japan Atomic Energy Research Institute, 801-1 Naka-machi, Ibaraki-ken, 311-0193, Japan*
[f] *Research Institute for Applied Mechanics, Kyushu University, Kasuga, Fukuoka 816-8580, Japan*

Abstract

During reactor operation the plasma-facing materials have to fulfil very complex and sometimes contradicting requirements. At present, tungsten shows the highest promise as plasma-facing material. Experiments in the ASDEX Upgrade tokamak indicate that plasma operation is feasible with walls and divertor surfaces mostly covered with tungsten. Thick tungsten coatings have been deposited by plasma spraying on EUROFER first wall mock-ups and show good adhesion and stability. The performance of tungsten surfaces under intense transient thermal loads is another critical issue, since the formation of a melt layer may favour the generation of highly activated dust particles. Work on 'nanocrystalline' tungsten shall improve the mechanical properties under neutron irradiation which is especially important for designs, where tungsten has also to fulfil structural functions. Alternative divertor heat sink materials with very high thermal conductivity like SiC-fibre reinforced copper composites are presently being developed and should allow operation at reactor relevant coolant temperatures.

1. Introduction

The operation of fusion power reactors will impose very demanding operation conditions on the plasma-facing components (PFCs). The PFCs will be operated in an environment which comprises incident particles and heat flux from the plasma. The surface of the plasma-facing material (PFM) is subjected to erosion by energetic ions and neutral atoms escaping from the plasma [1]. Albeit it is expected that the experience gained by the operation of ITER will allow to operate a fusion reactor under quiescent and non-disruptive plasma conditions, a very limited number of disruptions may still occur. Such off-normal events can cause ablation of material from the heated surface and progressive damage [2,3]. Giving a very stringent limit on the allowable number of disruptions in a reactor the sputtering erosion of the PFM during normal operation will determine the component lifetime. Experiments with high-Z materials, especially tungsten (W), are being carried out in the ASDEX Upgrade tokamak, since the erosion of this material under normal operation conditions is considerably lower than the plasma induced erosion of low Z materials like carbon (C) or beryllium (Be). The disadvantage of high Z materials, however, is the far lower tolerable impurity concentration within the plasma which poses strong demands on plasma control. Further to sputtering, metallic materials might show additional erosion by forming blisters under incidence of ions from the plasma [4]. Tritium from the plasma is absorbed by the PFMs and might enter the coolant of the PFCs. The extend to which tritium is being taken up

* Corresponding author. Tel.: +49-89 3299 2141; fax: +49-89 3299 1212.
E-mail address: harald.bolt@ipp.mpg.de (H. Bolt).

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by the PFCs is highly dependent on the material used, and the state of the internal nano- and microstructure of the material.

Mechanical stress and strain within the materials is caused by the superposition of residual stress stemming from the manufacturing process and the additional thermal stress during operation. Neutron damage can lead to the degradation of the mechanical properties and in some cases to a decrease of the thermal conductivity and dimensional changes of the PFMs. These boundary conditions indicate that the selection of a suitable PFM is difficult, since many of the processes mentioned above can at present only be quantified by making largely simplifying assumptions.

The requirement of steady state operation leads to PFCs which are a compound of the PFM being directly bonded to a heat sink material containing the coolant [5–8].

The anticipated loading conditions for a first power reactor after ITER (DEMO) are listed in Table 1 as derived from [9]. Compared to this first power reactor, commercial reactors will certainly lead to even higher neutron wall loading and neutron fluence to the plasma-facing components. A tentative value for the first wall neutron fluence for a power reactor is 150 dpa.

This article focuses on the progress of R&D work on plasma-surface interaction with high-*Z* materials, especially W, and the development of new materials and material compounds for future fusion reactors. The progress described here takes as background the status of work which has been described in a previous review paper [1].

2. Plasma surface interactions employing tungsten as plasma-facing material

2.1. Experiments in the ASDEX Upgrade tokamak

A dedicated experimental programme is being performed in ASDEX Upgrade to investigate the plasma operation with W as plasma-facing material in a medium size tokamak.

In ASDEX Upgrade W is being progressively applied as plasma-facing material. At present the W wall coverage is approximately 25 m^2 consisting of a thin W coating of 1–3 μm on graphite and CFC tiles. The development of the coating technology and the characterization of the deposited W films is summarized in [10]. The only major areas which have not yet been covered by W are the lower divertor as well as the protection and antenna limiters on the outboard side. Recent results regarding the plasma operation of ASDEX Upgrade with a W coverage of approximately 15 m^2 are described in [11,12].

The results which are of direct impact on the selection of a high-*Z* plasma-facing material for a fusion reactor are summarized in the following. During the discharges in divertor configuration the high-*Z* atomic impurity fraction could be well controlled to remain below 10^{-5}. The variation from divertor mode to limiter mode and back during a single discharge showed that the wall contact on the inboard side during the limiter phase leads to enhanced sputter erosion and transport of high-*Z* impurities to the plasma centre with resulting strong impurity radiation. The return to divertor operation leads to the disappearance of the impurity source.

Table 1
Anticipated operation parameters and loading conditions for the PFCs of a first fusion reactor after ITER (DEMO), derived from [9]

	First power reactor after ITER (DEMO)			
	First wall		Divertor target	
Component replacements	5 year cycle		5 year cycle	
Av. neutron fluence				
(MWa/m^2)	10		5	
Displ./transmut. production	W	RAFM steel	W	Cu
dpa	30	120	15	60
appm (He);		1200		600
(dpa/%Re for W)	6% Re		3% Re	
Normal operation				
No. of cycles	<1000		<1000	
Peak particle flux (10^{23}/m^2s)	0.02		∼10	
Surface heat flux (MW/m^2)	<1		...10...	
PFM operational temperature (°C)	W: 550–800		W: water coolant: 500–900 He coolant: 650–1450	
ELM energy density (MJ/m^2),	≪1		≪1	
ELM occurrence			'Grassy', controlled?	
Off-normal operation (disruptions)				
Peak energy density (MJ/m^2)	–		30	
Duration (ms)/no. of events	–		1–10, max. 10	

In combination with outward impurity transport from the core plasma this causes a fast reduction of the impurity content and of the impurity radiation.

The operation of a fusion reactor with high-Z PFM would result in a high temperature plasma in front of the divertor target, very high divertor heat loads and strong sputter erosion, if no additional means of cooling of the edge plasma by radiation would be applied. In present devices operating with carbon-based PFMs, the eroded C serves as source for strong radiation in the divertor zone. In a reactor without C-radiation, eventually seeded gaseous impurities will have to act as edge radiation sources. In ASDEX Upgrade Ar gas had been seeded by a feedback controlled system to reduce and adjust the electron temperature of the divertor plasma (see Fig. 1). A flux of 4×10^{19} Ar at./s lead to a reduction of the edge temperature in the divertor from 20 eV to below 10 eV. Sputtering by Ar lead to a slight increase of the W-impurities from 10^{-5} to 2×10^{-5}, which, however, was uncritical to the plasma discharge. Using ELM pacemaking by injection of pellets the ELM activity could be stabilized [13] and it was shown, that in ASDEX Upgrade Ar seeding can be applied to reduce the divertor temperature without causing an uncontrolled increase of the influx of Ar-impurities.

The wall erosion of W by the plasma in ASDEX Upgrade was shown to be mostly driven by the incidence

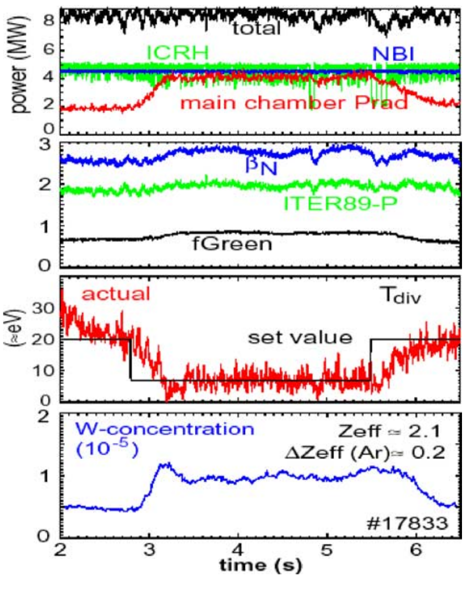

Fig. 1. Time traces of relevant plasma parameters in the ASDEX Upgrade discharge (#17833) with $I_p = 1$ MA, $n_e = 10^{20}/m^3$ and Ar gas puffing for edge cooling from 2.8 to 5.5 s. From top to bottom the parameters are: heating power (ICRH, NBI, total) and total radiated power from the main chamber; normalized plasma pressure (β_N), H-Mode factor according to ITER 89-P L-Mode scaling (ITER 89-P) and density normalized to the Greenwald density limit (f_{Green}); divertor electron temperature (measured and preset value); W-concentration.

of plasma ions rather than by energetic charge exchange neutral atoms. Time resolved spectroscopic measurements during the discharges proved that most of this ion driven erosion occurs during the ramp-up and ramp-down phases of the discharge when the plasma has wall contact. In a fusion reactor operated with proper control of the wall clearance of the plasma during stationary discharges this erosion path by ion erosion could be largely avoided.

In summary, the successful operation of ASDEX Upgrade with W-PFM indicates that this material is a viable candidate for application in a fusion reactor from the viewpoint of erosion and control.

2.2. Laboratory simulation experiments

2.2.1. Electron beam simulation of disruption heat fluxes

Extensive testing of high-Z materials had been carried out using a pulsed high power electron beam to simulate disruption heat loads [14,15]. It was found that pulses of 4.4 ms duration with a deposited energy of 5–7 MJ/m^2 lead to considerable differences regarding the damage to the materials structure, Fig. 2. La_2O_3 doped W shows the formation of deep cracks which follow the orientation of the grain structure whereas W26%Re shows a reduced crack formation in the recrystallized area. Powder metallurgically processed W (PM-W) does not form any objectionable cracks in the heat affected zone after such heat flux exposure. However, cracks have been detected in an area outside direct electron beam impact. In this region the material remained below DBTT and a distinct crack pattern has been developed during the cool down phase after a disruption. Test samples made from PM-W which were preheated above DBTT do not show any disabling crack formation. The cracks in bcc-metals usually propagate along the grain boundaries of the elongated grains of the rolled microstructure. It is thus advisable to apply massive W such that the grain orientation is in direction of the heat flux which is passing the material. Thus delamination of material from the surface caused by crack propagation parallel to the heated surface can be avoided. Plasma sprayed W (PS-W) may, depending on the microstructure, show delamination of particles or layers which have been insufficiently bonded to each other during the spray deposition process.

Transient heat load experiments by electron beam [14] and by pulsed plasma [16] confirmed that a part of the molten material is lost by droplet ablation processes which strongly enhances the overall erosion of the armour material.

2.2.2. He beam irradiation of W

Irradiation by He ion beam had been applied to W in order to investigate the formation of voids and bubbles in the surface region of the material as function of

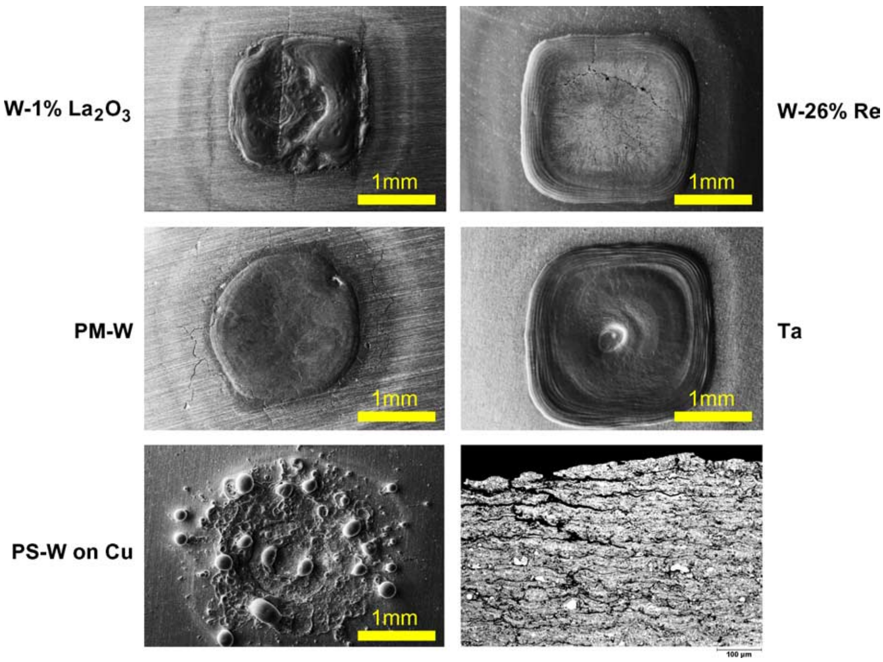

Fig. 2. Surfaces of different high-Z materials after pulsed high heat flux loading by electron beam; parameters: absorbed energy 5–7 MJ/m^2, beam current 160 mA, pulse duration 4.4 ms. The lower figure on the right hand side shows the metallographic cross-section of the PS-W coating shown on the lower left hand side.

substrate temperature, He ion energy and He fluence. At moderate temperatures (1023 K) and 14 keV He ion energy blisters formed and the erosion by sputtering was accelerated by the loss of the blister covers. At higher temperatures of 1273 K migration of the He voids to the grain boundaries took place where larger bubbles formed by void aggregation. This process was observed in beam experiments also at low energies of 250 eV. Only a very low He ion fluence of 4.5 10^{19} He/m^2 was needed to initiate the bubble formation at the grain boundaries [17,18], Fig. 3.

Coimplantation of D and He ions and subsequent thermal desorption experiments showed that the implanted D is quickly desorbed at low temperatures. Most of the implanted D leaves the W material already at temperatures below 800 K [19,20]. This result coincides with thermal D release experiments on W which had been irradiated by high energy protons (800 keV, 0.3 and 8 dpa) and subsequently exposed to a D glow discharge plasma [21]. Also here massive D desorption has been observed already at low temperatures. Thus it is expected that the use of W at elevated temperatures in a fusion reactor will not lead to a major T inventory in the W armour.

2.2.3. Exposure to laboratory plasmas

PM-W was exposed to a He plasma at different ion energies in the NAGDIS-II device [22]. Below 5 eV ion energy no bubble formation was observed. Initial

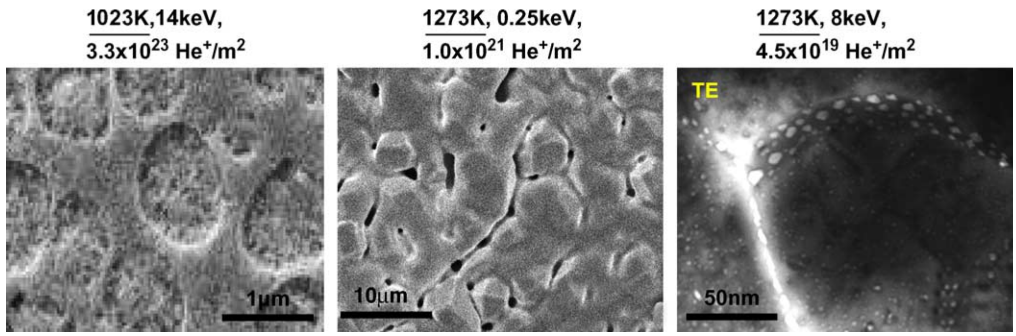

Fig. 3. Formation of bubbles and voids under He ion irradiation.

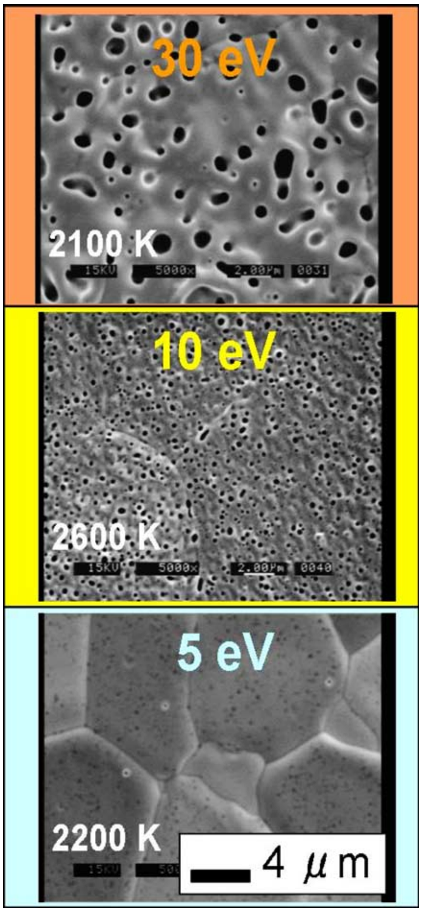

Fig. 4. Formation of He bubbles and voids under He plasma exposure of W [22]. He ion flux 10^{23} m^{-2} s^{-1}, He ion fluence up to 10^{27} m^{-2}.

nanoscale He-bubbles formed near the surface at 10 eV He ion energy and 1300 K substrate temperature. Higher ion energies and/or higher temperatures lead to the formation of larger He bubbles of >100 nm size, Fig. 4. Both, the results from the He ion beam and the He plasma exposure experiments indicate that the creation of interstitial atoms by energetic He is not needed to initiate the formation of voids, but that thermal vacancies act as centers and assist to the formation of voids. At higher temperatures these voids can migrate to grain boundaries and agglomerate to form larger bubbles.

3. High-Z coatings for the first wall of fusion reactors

3.1. Breeding ratio of solid breeder reactors with W first wall protection

Calculations of the breeding ratio of a power reactor with a solid ceramic breeder were carried out to identify the tolerable thickness of a protective W coating on the

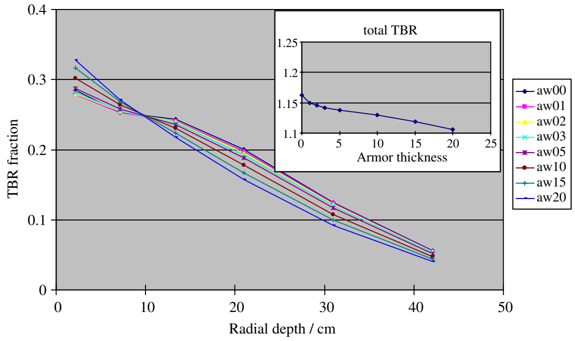

Fig. 5. Tritium breeding ratio through the blanket for different thicknesses of the W armour on the first wall [23]. Insert shows the overall TBR as function of the W armour thickness.

first wall [23]. The high neutron scattering and capture cross-section of W does not lead to significant decrease of the T breeding ratio in a solid breeder blanket if the thickness of the W coating is up to 3 mm, Fig. 5.

3.2. Plasma spray coating of low activation stainless steel

In a fusion reactor large portions of the first wall which are subjected to fluxes of energetic charge exchange neutral atoms from the plasma will have to be coated with W. In order to investigate the technological feasibility of a W coating on a first wall structure of low activation stainless steel, plasma spray coating experiments were carried out on EUROFER and F82H steel substrates. Actively cooled substrates of 60 mm × 190 mm surface area were used for the coating tests. The deposition of a mixed interlayer deposited by simultaneous spraying of EUROFER and W powder lead to excellent adhesion of the subsequently deposited W layer. A thickness of 2 mm of the pure W top layer could be obtained and a further increase of the coating thickness should be possible without major technical problems. Fig. 6 shows a cross-section of a PS-W coating on steel with a mixed interlayer for improved

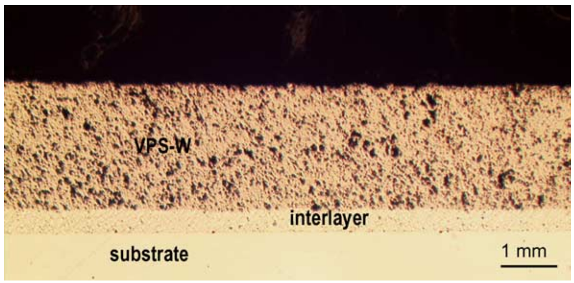

Fig. 6. Metallographical cross-section of plasma sprayed W coating on steel with mixed steel–W interlayer.

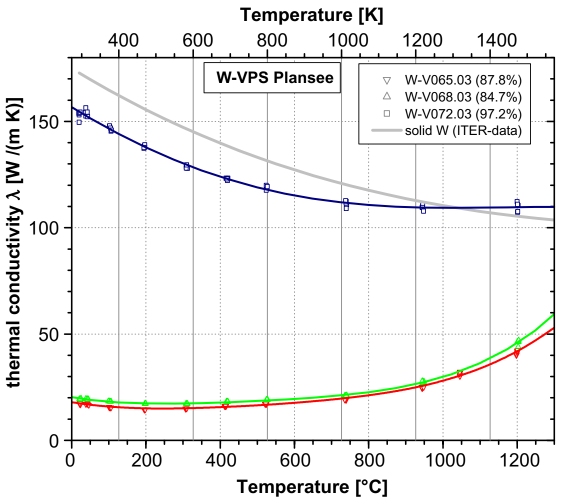

Fig. 7. Thermal conductivity of plasma sprayed W coatings with low density (84.7% and 87.8%), and high density (97.2%). Also indicated is the thermal conductivity of massive powder metallurgy W (solid W).

adhesion. The thermal conductivity of two W coatings with porosity in the range of 12–16% was measured, Fig. 7, and found to be sufficient for FW applications. For comparison, the thermal conductivity of a highly dense PS coating deposited at high temperature and the values for monolithic PM-W are shown in the figure as well. It is expected that the thermal conductivity of the PS-W coatings on steel could still be improved, if necessary.

4. Materials and compounds for fusion reactor divertors

4.1. He gas cooled divertor

The use of a He gas cooled divertor in a fusion reactor will pose several advantages compared to water cooled concepts. The plant safety is increased, since this would allow to operate a reactor with He gas coolant only and thus possible dissociation reactions of water steam under off-normal conditions are avoided. In addition, the heat convected to the divertor which is of the order of 10% of the fusion power could be used for highly efficient energy conversion processes. An initial design exercise is being carried out to identify the most important R&D needs for the realization of a gas cooled divertor [24,25]. As a result, materials and processing activities are being launched to investigate the most pressing issues. As protection material pure PM-W has to be joined to a La_2O_3 doped W-structure containing the He gas at 10 MPa pressure. This W-structure is to be joined to a ODS-EUROFER structure which connects the heat removal elements to the coolant distribution collectors and pipes.

Bonding tests of PM-W which acts as protection material to the La_2O_3–W structure by using a Ti-interlayer have been carried out successfully and evaluation of the mechanical properties of the bonds is under way. Also initial joining trials have been carried out for the bond of La_2O_3–W to ODS-EUROFER by using OFHC-Cu as compliant material. These bonds withstood thermal cycling in the operation temperature range between 600 and 750 °C.

Further activities are being carried out to identify and develop processing or machining methods for the sub-mm structured heat transfer enhancing W inserts. A relevant shaping technology would have to provide high throughput at reasonable cost for the approximately 500 000 inserts required for a fusion reactor divertor [26].

The most critical issue is the development of a W grade (possibly La_2O_3 doped) material with suitable microstructure and sufficient mechanical properties after neutron irradiation. This material has to show ductility in the temperature range of approximately 600 °C (bonding to ODS-EUROFER) to approximately 1450 °C without grain growth or recrystallization. Initial activities on the development of nanostructured W-based materials are presently being carried out or being initiated by several institutions [27].

4.2. Water cooled divertor employing low activation steel as structural material

The design of water-cooled divertors follows a more evolutionary approach which is based on a technology extrapolation of the knowledge gained during ITER divertor development. First activities were performed by producing a F82H based actively cooled divertor specimen from solid material without further joining. The specimen was subjected to a thorough evaluation of the fatigue properties, both numerical and experimental by electron beam loading. The cyclic loading of the specimen with 5 MW/m^2 lead to fatigue fracture after 650 cycles, which is roughly twice the predicted value from a theoretical estimate. Cyclic loading to 3 MW/m^2 for 10 000 pulses resulted in plastic deformation, but no failure.

As the heat flux which can be removed by massive F82H divertor elements is insufficient for reactor divertor operation due to the limited thermal conductivity of the material, further activities will be directed towards a W-based monoblock design applying F82H as ductile liner material with a wall thickness of 0.8 mm.

4.3. Cu-based metal matrix composites as heat sink material

A method to realize a high thermal conductivity material for the divertor is Cu-based metal–matrix composites (MMC). Conventional Cu materials like CuCrZr or DS-Cu are prone to embrittlement under neutron irradiation below 200 °C and to irradiation induced creep above 350 °C [8]. Regarding a fusion reactor divertor, the remaining temperature window is too narrow to apply existing Cu materials. SiC-fibre reinforced Cu may provide a highly creep resistant alternative for operation at higher temperatures. The SiC-fibres of the reinforcement provide the necessary high temperature strength, whereas the Cu matrix mainly provides high thermal conductivity and hermeticity. Initial developments of SiC_f–Cu show promising results in terms of micromechanical properties like interfacial strength between fiber and matrix as well as an ultimate tensile strength scaling which follows the rule of mixture between SiC-fibres and Cu matrix [28]. In parallel to this experimental activity the implications for the use of SiC_f–Cu are investigated in the frame micromechanical analyses and scale-bridging numerical modelling of the mechanics of divertor components [29].

5. Further research needs for developing and validating reactor PFMs and PFCs

In order to provide a credible basis for the further development of fusion power, the research efforts in this field are presently being enhanced.

The lack of knowledge regarding the application of W as protection material for fusion reactor PFCs is a major obstacle. At present major effort is being carried out in ASDEX Upgrade to establish operation experience with W in a divertor tokamak. A consistent activity leading to the validation of W as fusion reactor PFC, however, would require further study of plasma–wall interaction processes in an existing large-sized tokamak in order to lead to a solid physics base for the extended use of W as PFM in ITER as the most reactor relevant device. The operation of ITER with full W coverage would be the final validation of W as PFM for a fusion reactor.

To support the physics and PWI base resulting from studies of tokamak operation with W based PFMs, laboratory investigations are necessary to yield quantitative data on surface processes and especially on the erosion of W under incidence of hydrogen and He particles as well as plasma impurity species. Erosion, transport and redeposition processes of W eroded at the wall have to be quantified by a combined effort of modelling, laboratory and tokamak experiments.

A very serious lack exists regarding the knowledge of the behaviour of W under neutron irradiation. This concerns the mechanical properties of bulk W when being applied in structural function. It also concerns the integrity of W-based coatings which do not have a structural function, but are subjected to severe stress under thermal load and from the thermal expansion mismatch with the substrate material to which they are bonded.

For divertor applications where W is being foreseen with structural function over a wide temperature range, W-based materials with high temperature stability against grain growth and recrystallization as well as a low DBTT after neutron irradiation would be required. This demands a consistent development programme on nanostructured W with stabilization of the nanoscaled grains by suitable dopants. On these trial materials again neutron irradiations need to be carried out to investigate the effect on mechanical properties.

Design studies of fusion reactor divertors will yield the detailed requirements for further materials development, e.g. of suitable heat sink materials and of bonding, processing and machining technologies. Also for these materials and their compounds the behaviour under neutron irradiation needs to be assessed. The heat removal capability of compounds and the operational integrity after neutron irradiation requires verification.

6. Summary

Results from large scale application of W as PFM in the ASDEX Upgrade tokamak prove that plasma operation with W as PFM is possible in a divertor tokamak without major constraints on the operational flexibility. In a reactor operating with very long pulses and a sufficient clearance between wall and plasma the wall erosion should not be governed by plasma ions, but by the comparatively low flux of energetic charge exchange atoms which would result in very low wall erosion values.

Even in a fusion reactor being operated under well controlled plasma conditions a very limited number of disruptions may occur. Electron beam simulation experiments on different high-*Z* materials show that some material grades may suffer damage such as crack formation and ablative loss of particles besides intense melting of the surface.

Laboratory experiments applying He ion beams and He plasmas indicate the formation of surface blisters and submicron sized He-bubbles in the temperature range around 1000 °C. This might result in a slightly enhanced erosion of W during operation. The inventory build-up of D or T in plasma-facing W under both ion and high energy proton irradiation simulating neutron damage was found not to be an issue as the reactor wall would be operated at temperatures sufficiently high to allow for the thermal release of implanted D and T.

Calculations of the breeding ratio of a fusion reactor with ceramic breeder blanket showed that even for this design which is sensitive to W as PFM a W protection thickness of up to 3 mm is acceptable.

The basic technological feasibility of thick plasma sprayed W coatings on low activation steel is being established by an experimental programme on coating deposition and characterization.

The most critical component needing attention and intense development is the divertor. Design studies of a He gas cooled divertor are being carried out along with initial development activities to identify and resolve critical materials related issues. Regarding the design of a water-cooled divertor which is a technological extrapolation from ITER knowledge, development activities focus on a W monoblock design with an integrated liner of low activation steel. Further work is being directed to the development of SiC-fibre reinforced Cu(SiC_f–CU) metal matrix composites which would provide high thermal conductivity together with the thermal stability required for the high temperature operation as heat sink material.

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