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Section 2. Plasma facing and high heat flux materials

Plasma facing and high heat flux materials – needs for ITER and beyond

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Abstract

Plasma-facing materials (PFMs) have to withstand particle and heat loads from the plasma and neutron loads during reactor operation. For ITER knowledge has been accumulated by operation experience and dedicated tests in present tokamaks as well as by laboratory experiments and modelling. The rationale for the selection of PFMs in ITER (Be, W, carbon fibre reinforced carbon) is described with regard to the critical issues concerning PFMs, esp. erosion during transient heat loads and the T-inventory in connection with the redeposition of carbon. In the fusion reactor generation after ITER the very stringent conditions of increased surface power to be removed from the plasma, a lifetime requirement of several operational years, high neutron fluences and increased operation temperature are exerting even more severe constraints on the selection of possible materials. Comparing these boundary conditions with materials under development and their further potential, only a narrow path is left regarding heat sink and PFMs. In this context the investigations on W as first wall material carried out e.g. in ASDEX Upgrade are being discussed as well as laboratory results on W-based material systems. The implications of these results are the starting point of what should form a consistent programme towards plasma-facing and heat sink materials for a fusion reactor.

1. Introduction

Plasma-facing components (PFCs) of fusion reactors 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. In addition, high transient heat loads during strong edge localized mode (ELM) activity of the plasma or off-normal events like disruptions can cause ablation from the heated surface. Tritium is absorbed by the PFMs or it can be chemically bonded to redeposited material.

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 PFM is difficult, since many of the processes mentioned above can only be quantified by making largely simplifying assumptions.

Tokamaks which have been operated in the past and presently are mostly relying on the concept of passive heat uptake during the plasma pulse and subsequent slow heat removal in between discharges. This allows

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simple design solutions and does not call for solid bonds between PFM and the heat sink. Long pulse machines like Tore Supra and in the near future Wendelstein 7-X require active heat removal during plasma discharges with durations of the order of 100 to several 1000 s to avoid overheating of the materials. This leads to PFCs which are a compound of the PFM being directly bonded to a heat sink material containing the coolant [1,2]. The same holds for ITER which also requires quasi-stationary heat removal from the PFCs during the plasma discharge [3].

Experience during the plasma operation of tokamaks and stellarators showed that the use of low-Z materials greatly reduced the radiation loss from the plasma compared to medium and high-Z PFMs, e.g. [4]. The very short integrated plasma operation time of present devices does not require attention to the erosion lifetime issue as far as materials other than very thin plasma-facing films are concerned. In future devices, including ITER, the issue of the component lifetime due to material erosion becomes important. In ITER the PFM lifetime will be determined mostly by thermally induced erosion during transiently peaked heat flux events. Off-normal events like disruptions have been observed in all tokamaks and are a major cause for thermal erosion. Means are being developed to diagnose early phases of disruption development, to counter their further evolution and to mitigate the consequences of a disruption [5]. In ITER disruptions still play a major role since in an experimental device the boundaries of the operational parameter space will have to be explored. In addition, at present the means to avoid and to soften disruptive events are not yet developed fully.

Regarding a reactor following ITER (e.g. DEMO), advance in plasma control and thus a quiescent plasma operation is presumed. Thus, in a reactor, the sputtering erosion of the PFM during normal operation will determine the component lifetime. Experiments in present fusion devices with high-Z materials, especially tungsten (W), are being carried out, since at least 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 [6].

The aim of this article is to describe the main issues which lead to the selection of PFCs for ITER. Since the criteria for this selection differ from the criteria relevant to a fusion power reactor, a partially different approach towards the selection of PFM and heat sink materials for a reactor has to be taken. To arrive at a consistent materials choice for a reactor, intense research has to be carried out and the corresponding main research needs are outlined in this article.

2. PFM and heat sink materials for ITER

2.1. Operation conditions

Table 1 lists the main operation conditions for the plasma facing components in ITER [7,8] as well as anticipated operation conditions for a first electricity producing reactor (DEMO) after ITER [9].

2.2. Rationale of the material selection for ITER

In the following the main issues are briefly discussed which exert strongest influence on the selection of materials for ITER. Detailed documentation is given in [7,8,10–12]. Present planning for ITER foresees the use of Be as first wall PFM, W and carbon fibre reinforced carbon (CFC) will be applied as divertor PFM.

2.2.1. Preference for low-Z first wall material

Based on the large experience with the operation of low-Z PFM esp. graphite or CFC composites and to a lesser extent Be, it has been established that plasma impurities from these materials can be tolerated also in ITER up to a dilution limit of the D-T plasma of 1%. Further to this, in tokamaks with ITER relevant divertor configuration like JET and ASDEX Upgrade it has been found that C- and to a smaller extend also Be-impurities show strong radiation at low electron temperatures in the divertor [13]. This helps to further reduce the kinetic plasma energy and thus to reduce the localized power deposition at the strike zone on the divertor plates. In contrast, far less operational experience with medium and high-Z materials exists and in earlier fusion experiments, having been mostly limiter devices, sometimes intolerably high impurity concentrations occurred [14,15]. More recent experiments with W as PFM showed that a wide range of plasma operation conditions can be applied also with high-Z divertor and wall materials leading to uncritical W concentrations of $<10^{-5}$ [16,17].

2.2.2. Tritium inventory

In addition to the T-inventory which accumulates in the near surface zones of PFMs, major concern is caused by T which is codeposited with eroded wall material. Especially the codeposition with C on low temperature surfaces ($T < 350$ °C) can lead to hydrogenated carbon films with H-isotope to C-atom ratios of 0.4–1. This material may also be deposited on surfaces distant from the plasma at locations where later removal will be impossible. Estimates for the tritium codeposition in ITER are of the order of a few grams per discharge even with a limited CFC coverage of the divertor strike point modules only [11], Fig. 1. Chemical erosion of CFC is the main source for the redeposition of hydrogenated (tritiated) carbon. This compares to a projected value of

Table 1
Operation conditions for the plasma facing components of ITER and a DEMO-like reactor [7–9]

	ITER		Reactor	
	First wall	Divertor target	First wall	Divertor target
Component replacements	None	Up to 3	5 year cycle	5 year cycle
Av. neutron fluence (MWa/m^2)	0.3	Max. 0.15[a]	10	5
Displacement damage/ transmut. production (dpa/appm (He)) (dpa/%Re for W)	Be 1/1000 Cu 3/30 SS 3/30	CFC 0.7/230 W 0.7/0.15% Re Cu 1.7/16 SS 1.6/16	W 30/6% Re RAFM steel 120/1200	W 15/3% Re Cu 60/600 RAFM steel 60/600
Normal operation				
No. of cycles	30 000	10 000?	<1000	<1000
Peak particle flux ($10^{23}/m^2s$)	0.01	~10	0.02	~10
Surface heat flux (MW/m^2)	<0.5	~10[b]/3	<1	...10...
PFM operational temp. (°C)	Be: 200–300	W: 200–1000 CFC: 200–1500	W: 550–700	W: 350–500
ELM energy density (MJ/m^2)	–	<1	–	Reduced
ELM duration (ms)/{Frequency}	–	0.2/ {few Hz}	–	'Grassy'?
Off-normal operation				
Peak energy density (MJ/m^2)	60 (VDEs)	30 (Disr.)	–	?
Duration (ms)/{Frequency (%)}	300 {1%} (VDEs)	1–10 {<10%} (Disr.)	–	1–10, max. 10 events

[a] Without replacement.
[b] Slow transients 20 MW/m^2 lasting 10 s (10% frequency).

0.2 g T/400 s discharge for a Be first wall which already assumes strong BeO formation that leads to enhanced T codeposition [11].

In fact, the tritium codeposition issue has been the main driver to reduce the C coverage of PFCs in ITER as far as possible. As first wall PFM Be was preferred as low-Z material, while in the divertor regions with moderate off-normal heat loads W has been chosen due to the lower erosion under off-normal heat loads compared to Be. The divertor operation of ASDEX Upgrade with full W coverage of the strike zones showed that W erosion in the divertor area can be controlled and that even eroded W is well retained within the divertor zone [16,18].

2.2.3. Heat removal

The quasi-stationary heat flux to the divertor surface led to the development of high heat flux PFCs, which resulted in qualified developments of Be/Cu, CFC/Cu and W/Cu components. Heat flux tests on model components showed that especially CFC/Cu and W/Cu material combinations are suited to remove heat fluxes of >20 MW/m^2 for at least 1000 thermal cycles, Fig. 2 [10].

2.2.4. Erosion during edge localized mode activity

At present it is still not clear, whether ITER can be operated with H-mode plasma and rather quiescent ELM-activity ('type II ELMs'). H-mode operation at

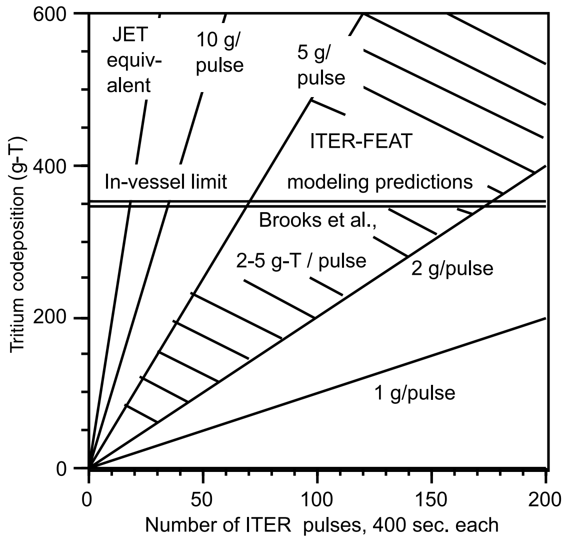

Fig. 1. Tritium retention for ITER, showing modelling predictions and the equivalent rate directly derived from JET D-T experiments. The in-vessel inventory limit is shown by double line [11].

high densities and temperatures in the separatrix region in present devices leads to intense pulsed power deposition by strong ELM-activity with a frequency of about 10 Hz and pulse durations of several 100 μs (type I ELMs). The scaling of the energy deposition from such

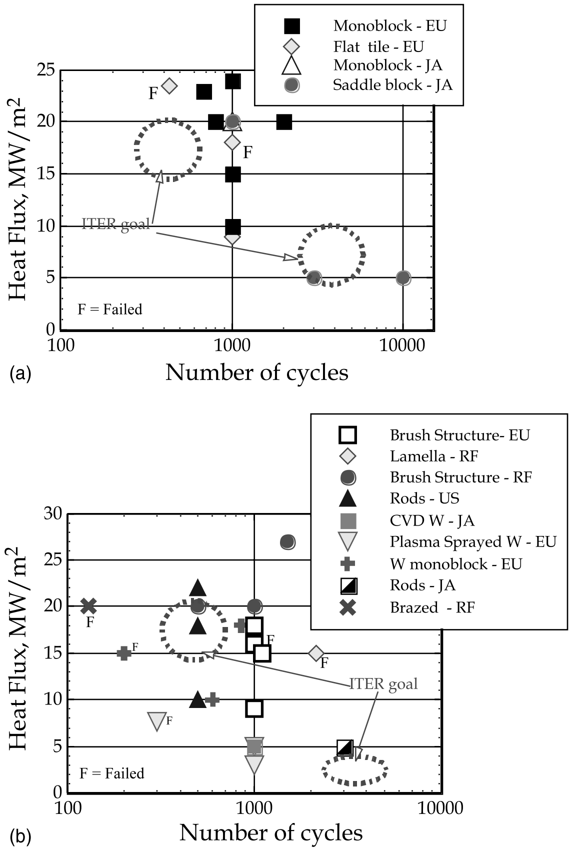

Fig. 2. Summary graphs of heat flux test results on actively cooled divertor test specimens for ITER. Legend indicates specimen design/processing technology and origin. Target values for ITER divertor operation are indicated [10]. (a) CFC armour and CuCrZr heat sink; (b) W armour and CuCrZr heat sink.

ELMs in present devices to ITER parameters [7,48] showed that ablation of divertor PFMs is likely to occur. In sensitivity studies [19] Be was ruled out as divertor material, since melting and evaporation would take place already well below even moderate estimates of possible ELM power deposition. The energy deposition limit (0.3 ms pulses) for Be is $\approx$0.2 MJ/m^2 (melting limit) compared to 0.4 MJ/m^2 for carbon (evaporation limit) and 0.6 MJ/m^2 for tungsten (melting limit) [7,8]. Thus CFC as divertor material would be marginally compatible with ELM power deposition as well as W allowing even slightly higher energy deposition.

2.2.5. Erosion during off-normal events

The thermal quench during disruptions in ITER would lead to energy deposition of up to 30 MJ/m^2 during 1–10 ms on the divertor PFM. The instantaneous ablation of the heated surface causes the formation of a thick and partially opaque vapour shield which leads to the radiative redistribution of the incident energy onto larger areas, mostly within the divertor chamber [19]. The ablation loss on CFC during such events is of the order of 10 μm, whereas the melting of tungsten and the possible partial removal of the melt layer can cause a reduction of armour thickness of the order of 100 μm per event [7,8]. Thus the disruption lifetime of the tungsten armour at the strike point location would be very short. For this reason the use of CFC is envisaged at and near the location of the divertor strike zones.

The quantitative implications of the occurrence of vertical displacement events (VDE) to the first wall are unclear, since the radiative cooling process under strong impurity influx has not been investigated in detail. In any case, active means to mitigate energy deposition of the order of 60 MJ/m^2 during 300 ms should be applied. VDE energy deposition without radiative cooling of the plasma would lead to severe melting and damage both on Be and W as first wall material. Power deposition experiments simulating VDEs by electron beam and calculations indicate melt depths of up to 3 mm thickness on Be and about 1 mm thickness on W during a VDE [20,21]. A significant fraction of the melt layer would be lost from the heated surface.

3. PFM and heat sink materials for a reactor

3.1. Operation conditions

As an orientation regarding a first electricity producing DEMO-type reactor following ITER the parameter set of [9] has been adopted. Different to the operation conditions for ITER as an experimental device, the requirements to reactor operation are driven mainly by the need for long operation time between component exchange periods (aim: 5 years), and by a fusion power derived neutron fluence to the PFCs (aim: 60 dpa$_{steel}$ (divertor) to 120 dpa$_{steel}$ (first wall)). Tentative loading conditions for the PFCs are listed in Table 1. The temperature level of the components has to be considerably higher in a reactor compared to ITER to allow for a reasonable thermal efficiency. The upper limits will be determined by the temperature limit of the PFM and heat sink materials. Table 2 lists issues regarding the selection of plasma facing and heat sink materials and qualitatively indicates the relevance of each issue for the materials selection in ITER and in a DEMO-type reactor.

It is assumed that the operation of ITER will yield considerable progress in plasma operational control, such that the issue of transient heat fluxes can be resolved as a question of plasma operation rather than a question of materials response. The rather large material loss rates per disruption and other transient heat flux events, esp. ELMs, which are envisaged for ITER and

Table 2
Relative comparison of the significance of issues for the PFM selection in ITER and DEMO

	ITER	Reactor (DEMO)
Operational flexibility	+	
Plasma erosion	○	++
Transient events	++	
Heat removal	++	++
T codeposition	++	○ (if W applied as PFM)
Neutron damage	+	++
Neutron activation		+

(○): Significant; (+): very significant; (++): crucial.

which in a power reactor would be even more severe due to the larger stored plasma energy lead to the firm conclusion that a reactor can only be operated with tolerable ELMs and a negligible number of disruptions and other transient off-normal heat flux events. In addition, it is presumed, that the pulsed power deposition during ELM activity can be controlled such that erosion and high cycle fatigue processes at the PFM surfaces can be excluded. Thus in the following a quiescent plasma is being adopted as operational scenario for a reactor.

The fusion power of a reactor of 2000 MW implies that the plasma will be heated by 400 MW of α-particle power. Typical ratios of the power share between first wall and divertor in present tokamaks are in the range of 10%/90% to 50%/50%. C-impurities or externally seeded gaseous impurities (e.g. Ne, Ar) allow for strong radiation in the low plasma temperature divertor zone and thus lead to a reduction of the localized convective surface heat load from the incident plasma [13,49]. Since a fusion reactor would have to be operated without C, gaseous radiating impurities like Ne or Ar have to be purposely seeded to increase the radiation of the plasma edge within the main chamber and in the divertor region. In a reactor this method would greatly reduce the power load to be carried by the comparatively small divertor surface area. Also within the divertor chamber the largest fraction of the power incident into the divertor region would also be transformed into radiative heat load. Thus up to 90% of the energy outflow from the plasma can be dissipated by radiation [22] and surface heat loads within the divertor chamber of not more than 10–15 MW/m^2 may be realized under semi-detached operation conditions [23,24]. It is therefore assumed in the following that a power reactor would be operated with Ne- or Ar-gas seeding under detached conditions.

3.2. Indications for reactor PFMs

3.2.1. Erosion, PFM thickness and lifetime

For ITER numerical simulations have been carried out on the wall erosion of Be and of W by neutral atoms originating from charge exchange processes (CX-neutrals) and from impurity ions which also are part of the background plasma existing even distantly away from the separatrix [8,11]. Scaling of these results indicate that low-Z material erosion (Be, C) would be 3–10 mm per burn year, for Fe 1 mm per burn year, and for W between 0.03 and 0.3 mm per burn year and thus considerably lower for W compared to low and medium-Z materials. ASDEX Upgrade results with large wall area W coverage show that wall erosion processes by impurity ions need further detailed investigation [25]. In the reactor case this would correspond to erosion by medium-Z seeded impurity ions. Thus the quantitative assessment of the first wall erosion in a reactor is still open. Although the ITER-based simulations have large inherent uncertainties, and their transfer to reactor operation is not straight forward, the order of magnitude of erosion indicates that on the first wall Be would not provide sufficient lifetime. In addition, the implications of the final redeposition of very large low-Z material quantities during operation within the reactor are unknown. Taking the order of magnitude of W erosion, a first wall with a W coating of the order of 1 mm might provide sufficient PFM lifetime. This would also be marginally compatible with the neutronics requirements of ceramic breeder concepts with Be-multiplier that critically depend on the low neutron absorption in non-breeding components [26]. A bare steel wall might be an attractive solution for recessed wall regions with only negligible impurity ion fluxes and moderate CX-particle fluxes.

During W operation on large wall areas in ASDEX Upgrade no negative effects on plasma performance from W plasma impurities were observed for ITER relevant operation regimes and the W-impurity fraction could be controlled to remain below 10^{-5} [17].

In the divertor the PFM erosion is difficult to quantify. However, in the case of semi-detached plasma operation within the divertor chamber, it is expected that also the energy of the seeded impurity ion flow will be dissipated radiatively. Thus at least for high-Z materials erosion may be of less concern on divertor surfaces. First calculations of the spatial electron temperature distribution with different Ne-seed impurity concentrations using the EDGE2D-NIMBUS code [50] showed that at the inner divertor a very cold plasma can be established and that also at the outer divertor the electron temperature can be decreased to less than 10 eV [27,51], Fig. 3. In addition to the very low expected erosion of high-Z material in the divertor a part of the material eroded on the first wall surfaces will be transported into the divertor with subsequent redeposition. W-deposition experiments by magnetron sputter coating process showed that for pure W-substrates, the redeposition should lead to the growth of films with good adhesion and thermal conductivity close to bulk material values [28], Fig. 4.

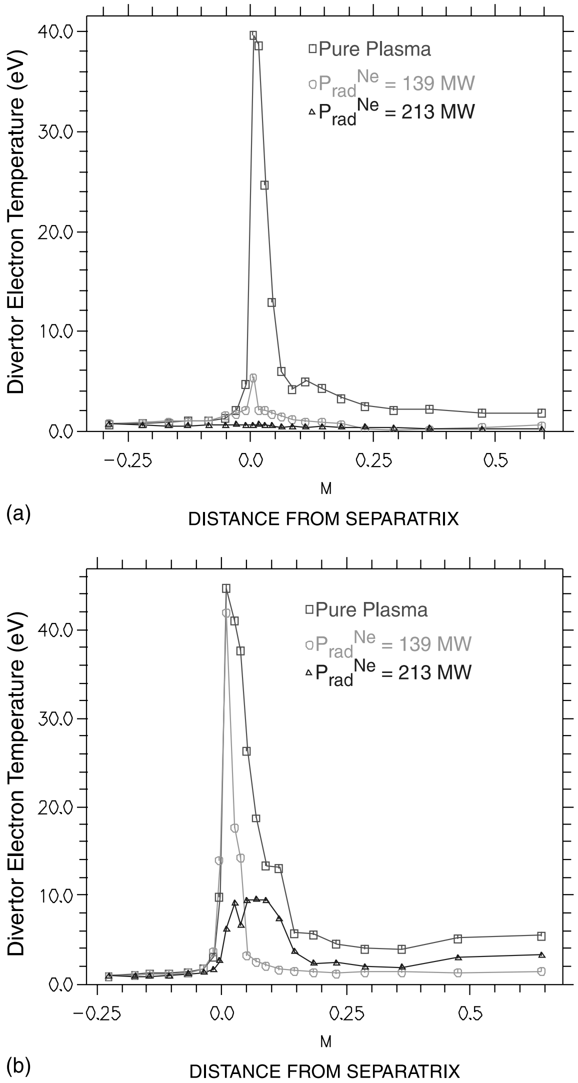

Fig. 3. Spatial evolution of the electron temperature in a reactor divertor as function of the radiative power dissipation by Ne-impurity seeding (EDGE2D-NIMBUS code) [27,50,51]. (a) Inner divertor; (b) outer divertor.

Under presence of a fully oxidized W-surface erosion of W by D-ions has been observed to occur already below the energy threshold for D-ion erosion of pure W [29]. In a reactor a thorough conditioning before the start of operation or after accidental O-ingress is inferred. During normal operation the specified leak rate can lead to a maximum influx of 3×10^{16} O-atoms/s [30]. Assuming O-enhanced W-erosion taking place the upper limit of possible enhanced erosion would be that in subsequent erosion and redeposition steps each O-atom leads to the transport of one W-atom to a remote surface where also final codeposition of the O-atom takes place. Under this assumption the maximum O-enhanced erosion would be 400 g W/burn year for the whole device.

Blistering which has been observed on W-surfaces under irradiation with H-, D- and He-ions could be a channel for enhanced erosion due to the subsequent flaking of the blister caps [31]. It was found that this effect depends on the processing, purity and structure of the W material. The results indicate that blistering occurs preferentially on hot rolled W with low intergranular strength. Low energy implantation of H-, D-ions appears to be the cause of compressive stress in the surface layer with subsequent interlaminar exfoliation taking place along the boundaries of the surface layer grains. No progressive delamination of more than one layer has been observed. Irradiation with non-monoenergetic ions does not lead to blistering, since the implanted H, D can migrate back to the surface through the ion damaged structure. Above 700 °C the implanted hydrogen is thermally released and blisters do not occur [31,32]. Blistering was not observed on plasma sprayed W in laboratory experiments [33]. In addition it was not observed on plasma facing W surfaces exposed to high ion fluxes in tokamaks.

At temperatures of 1500 °C and above implanted He can agglomerate to form highly pressurized He-bubbles. This leads to a roughening of the surface, when the near surface bubbles release their gas content and form craters of μm size [34]. Because of the limitations of the maximum heat sink material temperature, such surface temperatures will not be reached on W, see Section 3.3.

Based on these data and using properly processed W it is unlikely that enhanced erosion processes on W which have been observed under specific laboratory conditions will lead to serious lifetime reductions of W when used as PFM.

3.2.2. Heat removal

The heat flux incident on the divertor surface has to be conducted through the PFM to the heat sink material. R + D for ITER has shown that heat flux removal with W or CFC as PFM is possible up to the 25 MW/m^2 level under quasi-stationary conditions [10]. The thermal conductivity of W does not degrade significantly under neutron irradiation compared to CFC and Be [10,35]. In these experiments, rather than the PFM itself, the integrity of the high heat flux component including the interface to the heat sink material and the resulting component thermomechanics appear to be critical.

3.2.3. Neutron damage and radiological implications

Neutron damage at levels above 5–10 dpa lead to the exclusion of graphitic materials (dimensional instability, reduction of thermal conductivity) and Be (gas production and excessive increase of T inventory) [10,35,36]. Refractory metals are subjected to a strong increase of the ductile to brittle transition temperature

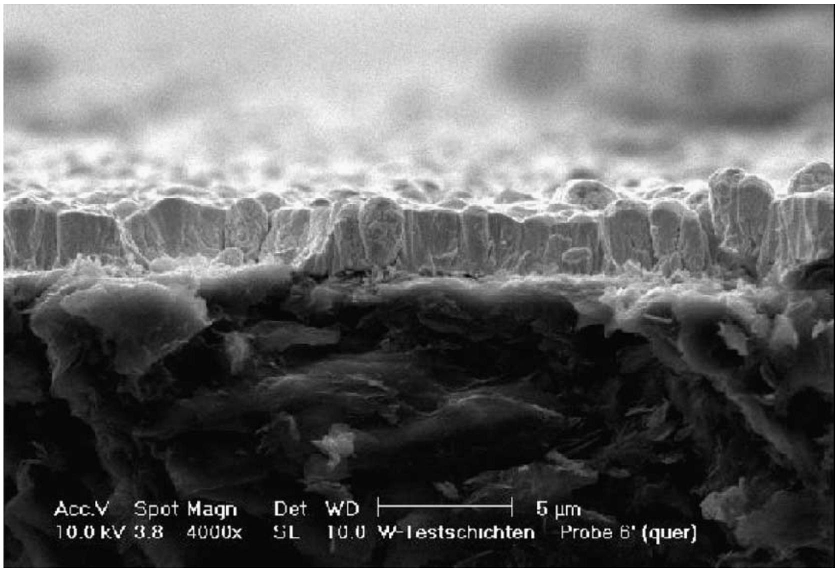

Fig. 4. SEM image of magnetron sputter deposited W coating showing dense columnar structure (substrate graphite).

(DBTT). For W embrittlement is to be expected below 800–1000 °C, indications for embrittlement already under low neutron dose are shown in Fig. 5 [35,37–39]. Embrittlement of Mo occurs below 800 °C [40]. This implies that refractory metals should be used as armour materials without structural function. The optimization of the armour thickness together with small scale segmentation of the armour should allow reactor operation with embrittled armour even on high heat flux components.

Regarding W, the comparatively strong emission of afterheat implies, that the W inventory in the reactor should be kept as limited as possible, Table 3. Depending on a detailed assessment of the erosion rate of W-clad PFCs, the thickness needs to be optimized, e.g. to 1 mm of W. Compared to W, Mo has a higher neutron induced long term activation and surface dose rate [41,42], Table 3. It has to be expected that all surfaces within the plasma chamber will show coverage with redeposited PFM. In the case of Mo having a high surface dose rate, this would mean, that any component surface would need extensive surface cleaning after operation and before any reprocessing procedure. W contamination would be less critical, since W is also being used as major compositional element of structural RAFM steels.

3.3. Indications for reactor heat sink materials

In contrast to ITER in a fusion reactor the temperature level of the heat removal will be significantly higher due to the need for a reasonable thermal efficiency. Since at present the choice of the coolant for the first wall (water or He) is still open, it has to be assumed that the temperature limitation will be imposed by the temperature dependent strength limit of the first wall structural material. Considering that the PFM will be a rather thin armour having no structural function, the heat sink material would have structural function as well as functions to redistribute the heat flux towards the coolant channels and to provide hermetic coolant confinement. Since on the first wall only a moderate heat flux of about 1 MW/m^2 would need to be removed, a

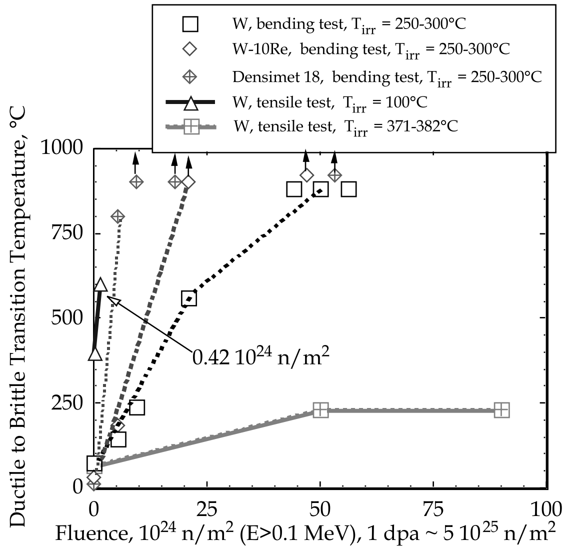

Fig. 5. Influence of neutron irradiation on the DBTT of W; W3.4Ni1.6Fe and W–10%Re alloys [35,37,39].

Table 3
Decay heat and surface dose rate for different potential plasma facing materials

	Decay heat after 1 day (kW/kg) (irrad. 4.15 MW/m^2, 2.5 y)[a]	Decay time to reach surface dose rate of 10^{-2} Sv (years) (irrad. 5 MW/m^2, 2.5 y)[b]
W	2×10^{-1}	150
Mo	3×10^{-2}	2×10^5
Fe	10^{-2}	60
SiC	10^{-3}	10

A surface dose rate of 10^{-2} Sv is regarded as an indicator for the remote recyclability of a material.
[a] Ref. [41].
[b] Ref. [42].

reduced activation ferritic martensitic (RAFM) steel with moderate thermal conductivity appears to be sufficient. Cyclic heat flux tests on a water cooled RAFM first wall mock-up indicates that up to 2 MW/m^2 can be removed by a steel based first wall. Though in the experiment at the heated surface plastic strain induced hardening occurred under 2.7 MW/m^2 heat flux pulses of 15 s duration, the material was not damaged during 5000 cycles, Fig. 6 [43]. The behaviour of W coatings on RAFM steel has not been subject of examination until now.

Applying an oxide dispersion strengthened (ODS) RAFM steel would allow a gain in the upper temperature limit from 550 °C (pure RAFM steel) to 650 °C (ODS steel) [44].

The divertor with heat loads of the order of 10–15 MW/m^2 would require a heat sink material with high thermal conductivity. Thus Cu is a candidate material for this application due to its superior thermal conductivity. A major drawback of Cu is the narrow window of operation temperature. Data up to 5 dpa show that CuCrZr shows irradiation hardening at temperatures below 200 °C. The upper temperature limit due to irradiation induced softening is 350 °C [45], Fig. 7. DS-Cu may result in an upper temperature limit of 400 °C. Preliminary work on SiC fibre reinforced Cu composites is presently under way [46]. Indications are that both strength and temperature limit could be increased with such composites, so that divertor operation at higher heat sink material temperatures would become possible. A further issue with Cu is the rather high activation under neutron irradiation [47]. Since the heat sink material, however, is shielded from the plasma, the contamination of other plasma chamber surfaces with Cu is not to be expected. Thus the activated Cu remains well isolated and would be of small volume of $\approx$2.5 m^3 for the whole divertor. In the long term further innovative material solutions which allow higher divertor operation temperatures could result in a more efficient use of the

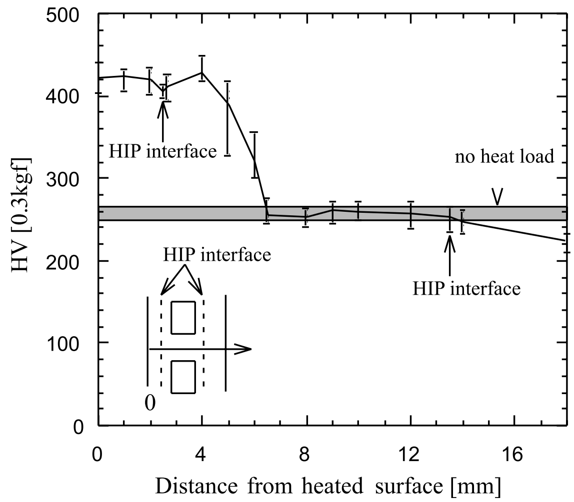

Fig. 6. Hardness distribution in depth direction of an actively cooled F-82H mock-up after surface heat flux testing to 2.7 MW/m^2 for 5000 cycles [43].

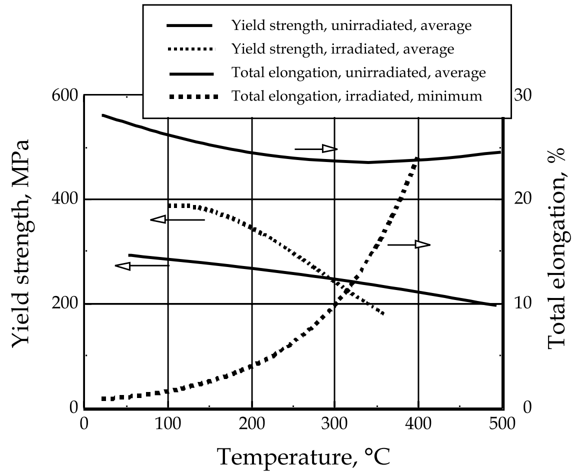

Fig. 7. Yield strength as function of temperature of CuCrZr unirradiated and after irradiation up to 5 dpa [45].

extracted heat from the divertor for electricity production (e.g. development of W-based materials with improved ductility and thus also structural applicability).

4. Main research needs towards viable PFM/heat sink materials solutions

A consistent research programme which aims to arrive at a viable technical solution for the PFCs of a fusion reactor will have to employ experimentation in fusion devices, activities in materials development and qualification as well as investigations concerning neutron damage on PFMs, heat sink materials and their compounds under reactor relevant neutron loads.

4.1. ITER and satellite experiments

The needs to be addressed by ITER are twofold.

Firstly with ITER reactor relevant plasma operation regimes have to be established which result in loading conditions of PFCs that assure the lifetime requirements of a reactor. These are

- virtually disruption free operation,
- quiescent plasma operation with minimal power transients from ELMs,
- balance of the power flow to first wall and divertor and semi-detached divertor operation to limit peaked divertor heat loads.

Secondly, detailed operation experience with W as PFM is needed. This implies operation with a fully W-covered divertor to assess the divertor erosion and re-deposition under reactor relevant operation conditions. Operation of a low-Z material free device would provide firm data on the compatibility of high-Z material with plasma operation applying seeded impurities as well as giving detailed data on the local wall erosion. This work could be shared among ITER and satellite experiments.

These data together with detailed modelling would allow the extrapolation to reactor conditions and allow materials selection and a detailed assessment of the locally required PFM thickness.

4.2. Materials development

Main aspects of future materials development would cover the development of highly reliable bonding techniques of thin and brittle refractory armour material to heat sink materials and the thermomechanical stability of such compounds with embrittled armour. Regarding heat sink materials, an increase of the upper operation temperature would greatly enhance the economic attractiveness of a fusion reactor. Thus strengthening and reinforcement mechanisms of metals should be subject of investigations which could eventually turn out a new class of metal-matrix composite materials.

4.3. Assessment of neutron damage

Directly linked to the development of PFM and heat sink materials, irradiations of materials and bonded compounds in an intense neutron source have to be carried out. Data for higher neutron fluxes (>10 dpa) are very scarce at present and originate from fission reactor irradiations. Thus the combined effects of transmutation induced gas production and displacement damage cannot be investigated to a definite level at present, since a sufficient transmutation rate only occurs at neutron energies higher than those of typical fission neutrons.

5. Conclusions

The selection of plasma facing materials in ITER is to a large extent driven by

- the wish for operational flexibility, leading to a preference for low-Z materials;
- the need to control the tritium inventory, leading to a minimization of the use of carbon, and thus the application of Be on the first wall;
- the maximization of the materials resistance to transient peaked heat loads, which leads to the application of CFC on highly loaded divertor target areas and of W on less loaded divertor surfaces.

Before a first electricity producing fusion reactor after ITER (DEMO) will be built, the control of transiently peaked heat fluxes has to be established, such that surface erosion during ELMs and disruptions will not drive the selection of the reactor-PFM. Also a broad database on the use of high-Z materials as PFM, esp. W, needs to be established to allow a detailed assessment of erosion and thus the definition of the armour thickness on the respective components.

Based on present knowledge, a thin plasma facing W armour on RAFM steel structure should be examined for reactor first wall application. For the divertor with incident heat flux of the 10–15 MW/m^2 level a thin W layer on a Cu-based heat sink may be feasible and is based on an extrapolation of ITER technology. Other advanced materials concepts allowing higher operation temperature with higher attractivity should be pursued in parallel.

In order to reach the goal of a timely solution for the plasma facing and heat sink materials in DEMO a consistent research programme involving ITER, a fusion relevant neutron source, and dedicated materials R + D are necessary.

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