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Plasma-wall interaction of advanced materials

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ABSTRACT

DEMO is the name for the first stage prototype fusion reactor considered to be the next step after ITER. For the realization of fusion energy especially materials questions pose a significant challenge already today. Advanced materials solution are under discussion in order to allow operation under reactor conditions [1] and are already under development used in the next step devices. Apart from issues related to material properties such as strength, ductility, resistance against melting and cracking one of the major issues to be tackled is the interaction with the fusion plasma. Advanced tungsten (W) materials as discussed below do not necessarily add additional lifetime issues, they will, however, add concerns related to erosion or surface morphology changes due to preferential sputtering. Retention of fuel and exhaust species are one of the main concerns. Retention of hydrogen will be one of the major issues to be solved in advanced materials as especially composites and alloys will introduce new hydrogen interactions mechanisms. Initial calculations show these mechanisms. Especially for Helium as the main impurity species material issues arise related to surfaces modification and embrittlement. Solutions are proposed to mitigate effects on material properties and introduce new release mechanisms.

1. Introduction

Tungsten (W) is currently the main candidate material for the first wall of a reactor as it is resilient against erosion, has the highest melting point, shows rather benign behavior under neutron irradiation, and low tritium retention. Extensive work has been done to qualify current materials with respect to these issues for ITER, especially for W as first wall and divertor material [2].

For the next step devices, e.g. DEMO, or a future fusion reactor the limits on power exhaust, availability, lifetime and not least on fuel management are quite more stringent. Extensive studies and materials programs [3–8] have already been performed hence it is assumed that the boundary conditions [9] to be fulfilled for the materials are in many cases above the technical feasibility limits as they are understood today. Efforts to establish new advanced plasma-facing material-options are moving forward [1] (and references therein) focussing on crack resilient materials with low activation, minimal tritium uptake, long lifetime and low erosion.

Fig. 1 shows an overview of the mechanisms of plasma-wall interaction typically considered. For the lifetime of the first wall of a fusion reactor the issues of material migration, hence erosion and re-deposition, are crucial considering the function of the material as an armor of the structural components. W is mainly eroded by impinging impurities such as carbon, beryllium and seeding gases, it is however still the best material choice to suppress erosion, due to a high threshold energy for physical sputtering [10–12].

For carbon and beryllium based PFCs the co-deposition of fuel with re-deposited material has been identified as the main retention mechanism (Fig. 1). This retention grows linearly with particle fluence and can reach such large amounts that carbon was eventually excluded in ITER and most likely future devices [13–15]. Tritium retention in PFCs due to plasma-wall interactions is one of the most critical safety issues for ITER and future fusion devices and does remain so for W as implantation and trapping, as well as diffusion into the bulk and permeation into the substructure will always lead to formation of T inventory, even if co-deposition can be avoided altogether.

Ultimately, the benefits of advanced materials have to be demonstrated in conjunction with plasma-wall-interaction (PWI)

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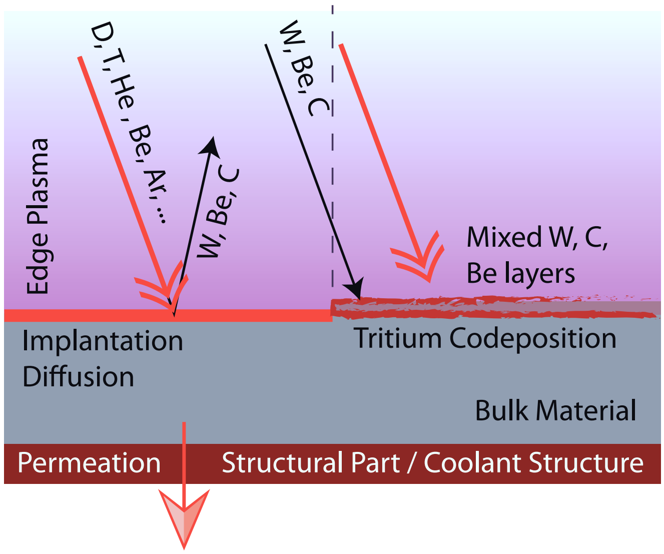


Fig. 1. Typical issues related to plasma-facing materials are ion and neutral impact, retention, erosion and redeposition.

studies from laboratory scale up to full component testing. The goal of this contribution paper is to identify the most critical areas to be tackled and to describe a possible development strategy based on linear plasma devices, modeling, lab-scale experiments and tokamak tests.

In this contribution an overview is given of new advanced materials within the framework of PWI and their compatibility with the operation of an energy-producing fusion plasma. Their properties are compared to the currently available baseline options for first wall materials. When choosing those materials three main aspects are typically considered for PWI: erosion, plasma compatibility, hydrogen isotope retention and material changes due to helium.

2. Plasma-wall interaction in advanced materials

2.1. Component

For the purpose of this discussion a component based on advanced materials [1,16] is envisioned. As reference a monoblock would be made of tungsten fiber-reinforced tungsten (W_f/W) [17,18,18–20], smart alloy as the matrix material [21–24] with interfaces, between fiber and matrix, based on oxide ceramics [25,26], a copper based cooling tube and integrated permeation barrier layers [27] (Fig. 2). The plasma-facing component can be made up entirely of W_f/W or only some area can be strengthened by including them. Depending on the exposure conditions erosion behavior and retention can hence vary. Based on various methods of building an constructing W_f/W composites either Chemical Vapor Deposition (CVD) [17,28,29] or powder metallurgical processes [19,30] are defining the microstructure of the matrix material.

Although erosion and retention for W are particularly low [2], the impact of plasma exposure, material microstructure, hydrogen diffusion, and the composite character of the component need to be considered. Interactions with helium (He) as exhaust or argon (Ar) as a seeding gas can cause changes in erosion patterns and retention in the upper layers of the material [31]. Considering steel or tungsten alloys such as EUROFER and self-passivating alloys [1] for the first wall, the erosion rate becomes increasingly important, determined by both composition and microstructure. The impact of the preferential erosion of light elements on the plasma performance and material lifetime are addressed in various references [24,32] (e.g. self-passivating alloys) and below under the headline of erosion (Section 2.4). Radiation damage can increase retention in the component by an order of magnitude [33].

The oxide ceramic interfaces introduced in the composite-material, allowing for pseudo-ductility, will also change the hydrogen interaction behavior as these interlayers can act as permeation barriers [27]. Interfaces become increasingly important also for power exhaust. Transmutation can quickly diminish the thermal conductivity to 50% [34]. With a volume fraction for interfaces and fibers, with low thermal conductivity, of ∼ 30% this potentially can become more challenging.

Interaction of helium with W ranges from surface morphology changes [35] to transmutation-induced He embrittlement at high temperatures from neutron irradiation [36]. Here recent work [37] aims at an insight into He in interface bubbles as well as He-induced hardening and how it depends on the interfaces and their surface area in composite materials, potentially also introducing new transport mechanisms.

2.2. Fuel retention and hydrogen interaction

For several reasons fuel retention is crucial when discussing plasma-material interactions in a tokamak. First and foremost it is related to the operational viability of a fusion power plant. In the course of the development of fusion power the breeding of tritium was identified as one of the crucial aspects. For each tritium atom used another needs to be produced with some additional production to cover losses etc. For a DEMO reactor or a future power-plant the tritium breeding ration needs to be of the order of 1.1–1.2 to cover modeling uncertainties and losses and to allow start up of additional power plants [9]. For tritium breeding the material choice can be crucial [38,39].

Fuel retention behavior of W is still subject to present studies especially when considering multi species plasma impacting together with additional heat-loads [31,40]. It was shown that by replacing CFC in JET with W and Be in the Joint European Torus (JET) the retention was significantly reduced [41] as the main mechanisms via co-deposition with carbon was removed. A remaining issue however, is the implantation and diffusion of hydrogen into the material. Especially for composite materials the interaction of hydrogen in the material with all its constituents needs to be clarified and it needs to be shown that for improved properties such as ductility or enhanced strength other aspects like safety and tritium self-sufficiency are not sacrificed. Fig. 3(a) shows the two macroscopic and microscopic issues relevant for W composite materials. Similar two bulk materials issues related to microstructure and material composition can be studied. This depends for example on the grain structure and defects in the material. Here an example for the CVD-W material used in W_f/W is given in Fig. 3(b). Pure CVD W was loaded with 6×10^{24} D/m^2 at 370K after being annealed at 1200K. The retention observed is similar to recrystallized pure W from powder-metallurgy as discussed in [42]. The expectation is hence that the bulk contribution from the matrix and its behavior is similar to bulk W. W_f/W however is a macroscopic 3D structure as depicted in 3(a).

In the W_f/W model-system discussed below interfaces consist of oxide ceramics, research on their properties as tritium permeation barriers ranges over a variety of materials [27,43–48], including alumina, Erbia and Yttria. Permeation reduction factors of up to 100 are reported.

In order to asses in a limited 1D model the behavior of such composite structures we are using reaction-diffusion based modeling [49–51] to detect first obvious differences of retention in composites.

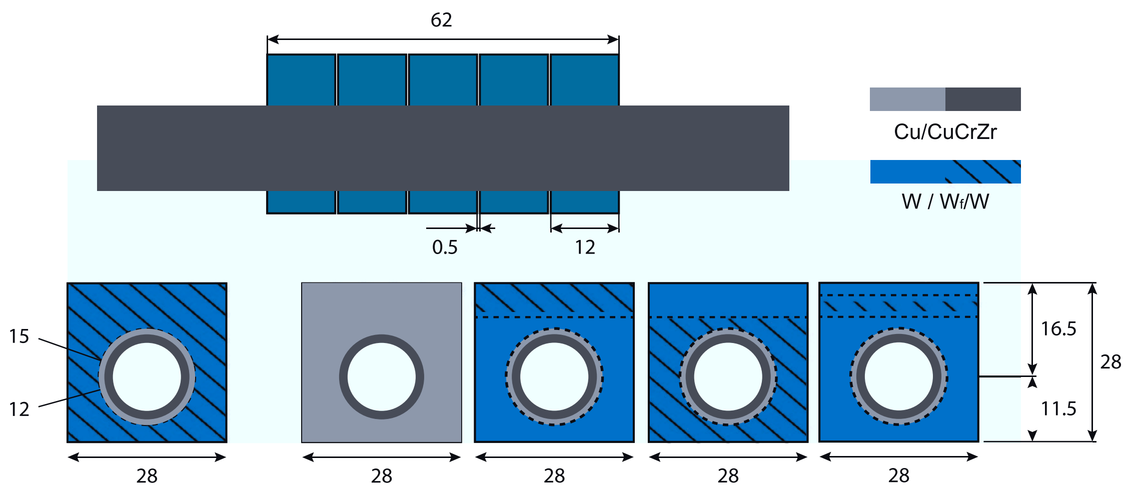
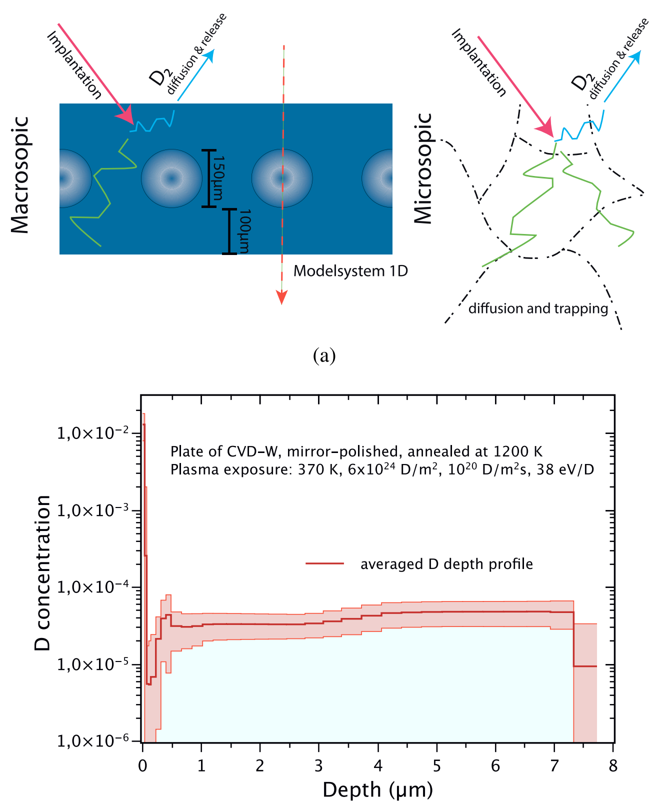


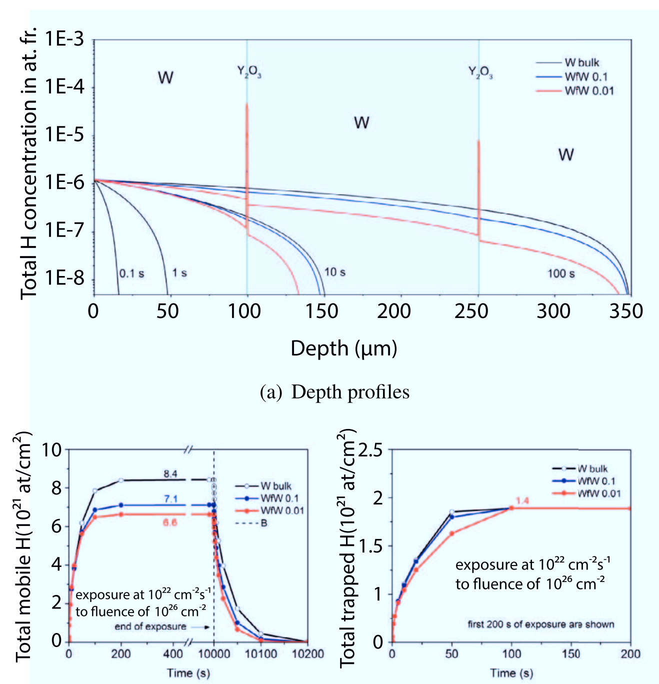
Fig. 2. Based on the current designs chosen for ITER and DEMO the mono-block or flat-tile design are favored. Introducing the advanced materials and composites can however be done in various locations. Dashed lines indicate locations of material interfaces and potential locations of permeation barriers. Dimensions are given in mm.



(a)

(b)

Fig. 3. (a) W_f/W with respect to hydrogen retention - 1D Modelsystem indicated. (b) Measured retention of CVD W similar to the one used in W_f/W [18].



(a) Depth profiles

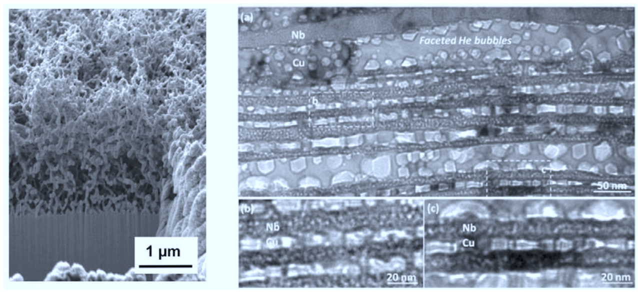
(b) Time evolution

Fig. 4. (a) shows the mobile H depth profiles, after 0.1,1,10,100s of loading, (b)(r) shows the total concentration of mobile and trapped H vs. time, (b)(r) shows the outgassing behavior.

The 1D calculations is based on a 5 layered model-system W(100µm)/Y2O3(1µm)/W(150µm)/Y2O3(1µm)/ W(100µm) similar to what is shown in 3(a).

For the matrix W-bulk properties are assumed, for the interface region similar mechanisms of diffusion are implemented however a reduction in diffusion rate of either 10 or 100 is assumed (cf. Fig. 4 Wf/W/0.1 & Wf/W 0.01). Here more detailed studies regarding the interfaces used and their properties are crucial and should be motivated by this work. The fiber is currently assumed to be behaving identical to the matrix. However the microstructure is significantly different [17–19,30,52] and hence detailed studies also on pure fiber retention properties are warranted. The trap density is set to 1E-7 at.fraction through the entire depth of the model system (incl. matrix, fiber and oxide). This is clearly a value to be adapted by comparison with experiments but allows a simple picture to be compared with expectations. The model system was loaded with 1E22 D/m^2s and a fluence 1E26 D/m^2.

Fig. 4 is showing the results of the modeling. In Fig. 4(a) it is observed that the mobile H concentration in the oxide layers increases due to slower diffusion as expected. In the 1D modelsystem this also means a drop in mobile hydrogen in the fiber and subsequent layers. Based on these assumptions the hydrogen traps are completely filled after 100 s. In principle they fill somewhat



(a) (b)

Fig. 5. (a) shows a FIB-cross section of the W-fuzz at the surface and the material below produced in PSI-2 [53], (b) He precipitate networks in Cu-Nb multilayer nano-composites [54]. (b)(b) and (b)(c) are magnified views of the corresponding boxed areas in (b)(a). They show incipient self-assembly of He clusters into interconnected networks.

slower but are inevitably filled despite slower diffusion in fiber-interfaces.

Fig. 4 (b)(l) shows that there is less mobile H in the bulk for W_f/W for this simplified assumptions. The diffusion barrier facilitates outgassing via the plasma exposed area rather then deep diffusion. Potentially this means that introducing a mechanisms that stops deep penetration of hydrogen in the component one can mitigate retention in composites. Here the fiber volume fraction plays a major role and more complex calculations need to shows this also for 2D and 3D structures. Here the ratio of volume to surface area of fiber, matrix and interface will play a crucial role. Assuming e.g. 30–50% volume fraction of fibers one can imagine quite a change in transport behavior.

Outgassing as shown in 4(b)(r) is not slower in the studied test system as a major part of the mobile H leaves the modeled structure through the plasma exposed side, a real 2D case or even 1D case with multiple fibers can be quite different.

For the model compared to the actual CVD material the trap density given defines the maximum retention hence.e.g. solute. The issues here are related the lack of input data for the materials used.This fact which needs to be mitigated as part of the PWI qualification of new advanced materials. It needs to be added that T retention in W is dominated by traps made by neutron irradiation during steady state operation of a fusion reactor. Permeation barriers will change the confinement of T within the fibers or the matrix regions in between. Whether permeation barriers can hence effectively reduce the fuel retention hence needs to be studied carefully.

2.3. Helium interaction

Similar to hydrogen also the impact of helium needs to be considered for any viable PFC concept, as helium is the exhaust product of the fusion reaction and hence is present as part of the impinging plasma impurities. One issue raised from linear plasma devices is the production of so called W-fuzz, surface nano-structures growing on W exposed at elevated temperature to helium plasma. W-fuzz 5(a) has been studied in various configurations [55–57].

A series of measurements coupling plasma exposures in PISCES and DIII-D [58,59] have been performed on W samples, with various surface morphologies. During these experiments a mitigated erosion behavior has been found as well as no additional roughening of the surface during ELMs. Here W-fuzz actually improves the PWI behavior. In addition the high heat-flux performance is changed [53] as grown nano-structures 5(a) are modified. The combination of He plasma with transient thermal shock events results in a severe modification such as reduced height or agglomeration of the sub-surface He-bubbles and of the created nano-structures, i.e. W-fuzz. In addition helium will cause high temperature embrittlement [36] and swelling if present in large enough quantities. In addition to the helium stemming from the fusion reaction transmutation of materials needs also to be considered [8,60,61]. Transmutation into helium is however a minor problem for W [62].

One of the promising new developments regarding the management of helium is the controlled outgassing of He through self-organized precipitate networks in metal composites. Helium (He) implanted into a metal rapidly precipitates out into gas-filled bubbles [63]. In single-phase metals, these bubbles tend to decorate defects, such as grain boundaries [64,65] or dislocations [54,66]. Aside from this tendency, however, their spatial distribution is typically uniform, on average. However, He precipitate morphologies may be markedly non-uniform in multi-phase composites of many metal phases. Non-uniform He precipitate distributions have been observed in studies on He-implanted layered composites of copper (Cu) and niobium (Nb) [67,68]. For example, Fig. 5(b) shows a Cu-Nb nano-composite synthesized by accumulated roll bonding after He implantation at a temperature of 480C. The figure shows markedly different bubble sizes in Nb and Cu layers. The former contains bubbles with diameters predominantly in the 1–2 nm range while the latter contains much larger, faceted H-filled cavities. Indeed, the size of He precipitates in Cu appears to be limited by the thickness of the Cu layers: precipitates may grow to span an entire Cu layer, but do not penetrate into the neighboring Nb layers.

Observations such as those in Fig. 5(b) point to intriguing opportunities for designing metal composites that outgas He in a controllable fashion. Yuryev and Demkowicz have proposed [69] that it may be possible to synthesize layered nano-composite materials where He precipitates interact, coalesce, and ultimately self-assemble into an interconnected network of clusters. Any additional He introduced into such a material would diffuse through this network and eventually outgas to the environment, preventing damage. One study suggests that He may indeed outgas along interfaces between phases in metal composites without causing morphological instabilities on the sample surface [70]. Stable outgassing of He along interconnected He precipitate networks is a plausible explanation for these findings. This idea is currently under investigation at Los Alamos National Laboratory.

As composite structures are considered to be used in fusion such mechanisms might be included in W_f/W or other composites to manage to helium or hydrogen content and hence its detrimental effects. Whether or not this will in fact mitigate embittelement is to be tested , as he is in fact localized more strongly in these materials. However , one can imagine using the interfaces regions composites (e.g. Wf/W) to introduce new channeling abilities.

2.4. Erosion

As seen above retention is the crucial element when considering new materials, nevertheless lifetime concerns need to be addressed. When discussing lifetime of the first wall of a fusion reactor the issues of material migration, hence erosion and re-deposition, are crucial considering the function of the material as an armor of the structural components. Currently it is assumed that only W armor provides a suitable life time. If W is hence required as armor material all new concepts need to make sure that W is the main element visible to the plasma at all times.

Preferential sputtering can be used as a mechanism to turn the top layer of alloys or steels into a thin layer of erosion suppressing W [71–73]. As an example erosion of EUROfer can be considered [72,74]. The effect of preferential sputtering will however change the surface morphology and potentially introduce additional roughness and micro-structured surfaces [71]. An issue

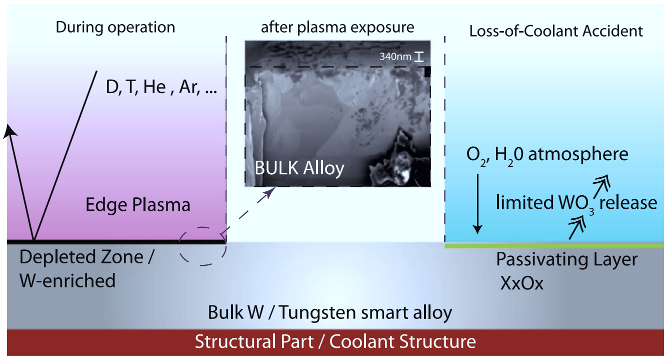


Fig. 6. Working principle of a smart alloys based PFC with both the operational and accident mechanisms shown [21].

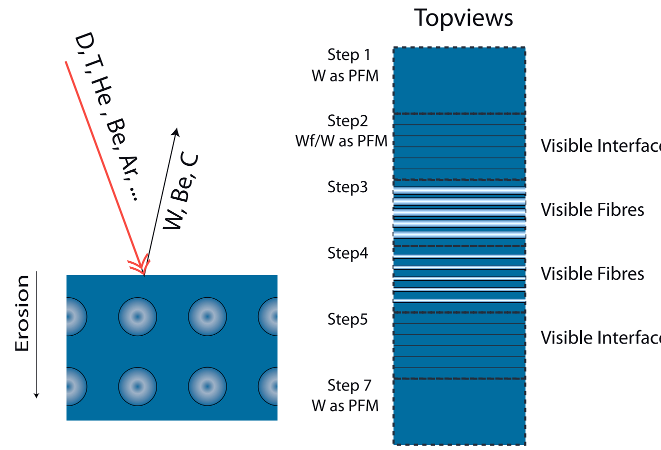


Fig. 7. Different scenarios of placing W_f/W and their impact on erosion.

raised in experiments [73] is that this effect depends on the temperature of the exposed material. This means that diffusion processes might very well counteract the surface enrichment at high temperatures.

In terms of plasma compatibility major concerns are only raised if the erosion of alloying elements is not fully suppressed- in such a case additional plasma impurities need to be considered.

One of the issues to be solved with the use of W in a fusion reactor is the formation of radioactive and volatile W-oxide (WO_3) compounds during an accident scenario [75–77]. This is mitigated by the use of so called smart alloys [6,22–24,32] which are typically produced as model-systems via Magnetron sputtering or on a larger scale via powder-metallurgically [16].

Preferential erosion of light elements during normal reactor operation is not expected to be of concern. Fig. 6 displays the basic mechanism. During operation plasma ions erode the light constituents of the alloy, leaving behind a thin depleted zone with only W remaining. Subsequently, the W layer suppresses further erosion, hence utilizing the beneficial properties of W. In case of a loss-of-coolant and air or water ingress the W layer oxidizes releasing a minimum amount of WO_3 and then passivating the alloy due to the chromium content. W-Cr-Y with a W-fraction of up to 70 at% shows a 10^4-fold suppression of W oxidation due to self-passivation [23]. Currently it can be demonstrated that this effect works up to 1200 °C

As discussed in [21] it is observed that the measured weight loss of sputtered smart alloy samples corresponds very well to that of pure W providing experimental evidence of good resistance of smart alloys to plasma sputtering. The exposure in plasma was followed by the controlled oxidation of smart alloys to test their behavior after exposition. The detailed results of this investigation are given in [21].

Going one step further however by introducing W_f/W, as a strengthening component into the mono block design as displayed in Fig. 2, introduces additional complications.

As seen in Fig. 7 W_f/W consists of multiple interchanging layers of fibers coated by an interface [17,25,26] and layers of pure W - based on CVD or powder-metallurgy. Depending on the details of the armor layer or mono-block either pure W or a mix or interface, fiber and matrix is eroded. Interfaces currently are typically oxide ceramics [17,19,30]. This will change the erosion characteristics and needs to be studied in detail in linear plasma devices, or tokamak experiments. Similar to preferential erosion of smart alloys one can assume that layers containing fibers will show inhomogenous erosion behavior. It needs to be established if e.g. always an armor layer of pure W needs to be positioned on top of the W_f/W enhanced layers. After eroding such an armor again a fiber layer would be present and exposed to the plasma. These issues are similar to erosion of CFC under fusion conditions discussed in [78,79].

In addition to conventional composites also fine grain W is an option to strengthen and ductilize W [80] similar to other metals [81] . An option to achieve this for W is powder injection molding (PIM) [82,83]. PIM as production method enables the mass fabrication of low cost, high performance components with complex geometries. The range in dimensions of the produced parts reach from a micro-gearwheel (d = 3 mm, 0.050 g) up to a heavy plate ((60 × 60 × 20) mm, 1400 g). Furthermore, PIM as special process allows the joining of W and doped W materials without brazing and the development of composite and prototype materials, as described in [82]. Therefore, it is an ideal tool for divertor R&D as well as material science. Mechanical properties, like ductility and strength, are tunable in a wide range (example: W-1TiC and W-2Y2O3) [83]. Based on these properties the PIM process will enable the further development and assessment of new custom-made W materials as well as allow further scientific investigations on prototype materials. Here initial plasma exposures shows no obvious enhanced erosion as to be expected from pure W a full qualification is ongoing.

3. Conclusion and outlook

By introducing either alloys or composite structures one does change significantly the behavior of the components with respect to plasma-wall interaction. First and foremost the changes are linked to erosion behavior and lifetime concerns and the retention and interaction with plasma species like hydrogen and helium. A typical model component is consisting of a tungsten fiber re-enforced tungsten (W_f/W) [17], smart alloy [6,22–24,32] with interfaces based on oxide ceramics, a copper based cooling tube and integrated permeation barrier layers [27] (Fig. 2). For the matrix material it seems erosion is similar to the pure W-bulk candidates discussed for current machines. Introducing composite structures however changes this and might cause inhomogeneous erosion. This needs to be studied in detail. Retention and permeation of hydrogen is a particularly crucial point and needs to be studied on model system and all the elements comprising the composite to allow model validation and extrapolation. The effects of helium in fusion materials are well known hence a mechanism related to composite materials and model-systems has been proposed [70] and is described above.

In general it can be said that composite materials offer benefits with respect to material properties and even their PWI behavior. Theses benefits will be further studied based on the development of model-systems and dedicated qualification under fusion relevant conditions.

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