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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 towards realizing fusion. 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 to be applicable for use 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, surface morphology changes due to preferential sputtering. Retention of fuel and exhaust species are the main concern. Tungsten alloys will most likely not add additional lifetime concerns as they will develop a protecting pure tungsten layer. However retention of hydrogen and helium will be one of the major issues to be solved in advanced materials as especially composites will introduce new hydrogen interactions. Initial calculations show theses mechanisms. Especially for Helium as the main impurity species solutions are proposed to mitigate effects on material properties and introduce new release mechanisms.

Keywords:

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 theses issues for ITER, especially for W as first wall and divertor material [2].

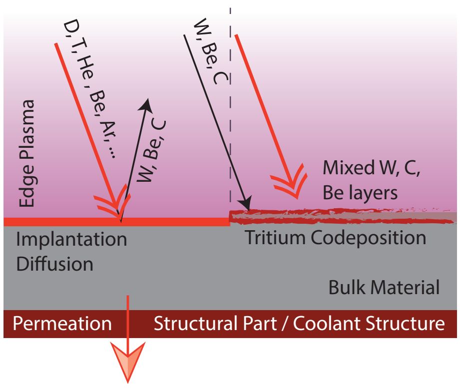

Figure 1: Typicall issues related to plasma facing materials are ion and neutral impact, retention, erosion and redeposition.

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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. Quite extensive studies and materials programs [3, 4, 5, 6, 7, 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.

Figure 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 its low sputter-yield [10, 11, 12].

For carbon-based PFCs the co-deposition of fuel with redeposited carbon 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, 14, 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

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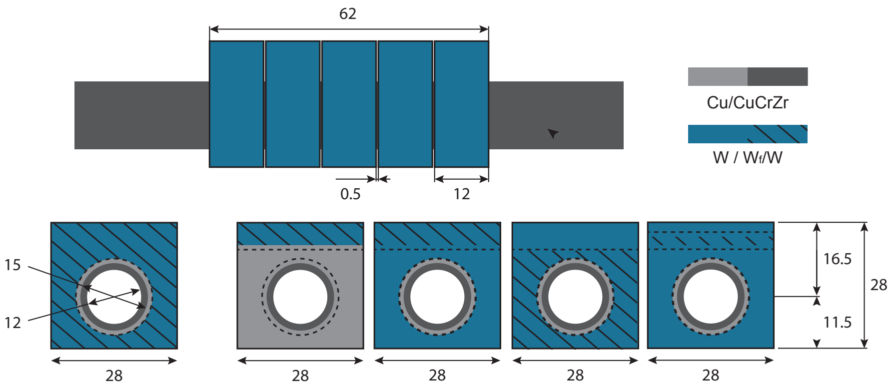


Figure 2: Based on the current designs chosen for ITER and DEMO the mono-block or flat-tile design are favored. Introducing the advanced materials and composites can however be done in various locations.

and trapping is still acting as well as diffusion into the bulk and permeation into the substructure.

Ultimately, the benefits of advanced materials have to be demonstrated in conjunction with PWI 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 in comparison to current understanding of baseline options. When choosing materials for fusion applications three main aspects are typically considered for plasma wall interaction (PWI): retention, erosion and plasma compatibility.

2. Plasma Wall Interaction in Advanced Materials

2.1. Component

For the purpose of the discussion a component based on advanced materials [1, 16] is envisioned. As reference a monoblock would be comprising of tungsten fibre re-enforced tungsten (W_f/W) [17, 18, 19, 18, 20], smart alloy [21, 22, 23, 24] with interfaces based on oxide ceramics [**?** 25] , a copper based cooling tube and integrated permeation barrier layers [26] (Fig. 2). The plasma-facing component can be made up entirely of W_f/W or only some area could 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) [27, 28, 17] or Powder metallurgical processes [19, 29] are driving 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 [30]. Radiation damage can increase retention in the component by an order of magnitude [31].

The in the composite-material introduced oxide ceramic interfaces, allowing for pseudo-ductility, will also change the hydrogen interaction behavior as theses interlayers can act as permeation barriers [26]. Interfaces become increasingly important also for power exhaust. Transmutation can quickly diminish the thermal conductivity to 50% [32]. With a volume fraction for interfaces with low thermal conductivity and fibres of ~30% could potenially become more challenging.

Interaction of helium (He) with W starts with surface morphology changes [33] and ends with transmutation induced He embrittlement at high temperatures and from neutron irradiation [34]. Here recent work [35] aims at an insight into He in interface bubbles as well as He-induced hardening and how it depends on interface area per unit volume in composite materials, potentially also introducing new transport mechanisms. Considering EUROFER or self-passivating alloys [1] for the first wall, the erosion rate becomes increasingly important, determined by both composition and microstructure. The impact of the enhanced erosion of light elements on the plasma performance and material lifetime are addressed.

2.2. Erosion

When discussing lifetime of the first wall of a fusion reactor the issues of material migration, hence erosion and redeposition are crucial considering the function of the material as an armor of the structural components. Currently it is assumed that a W armor is the best way to mitigate lifetime concerns. 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 [36, 37, 38]. As an example of alloys work on EUROfer erosion can be considered [37, 39]. The effect of preferential sputtering will however chenge the surface morphology and potentially introduce additional roughness and micro-structured surfaces [36]. 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.

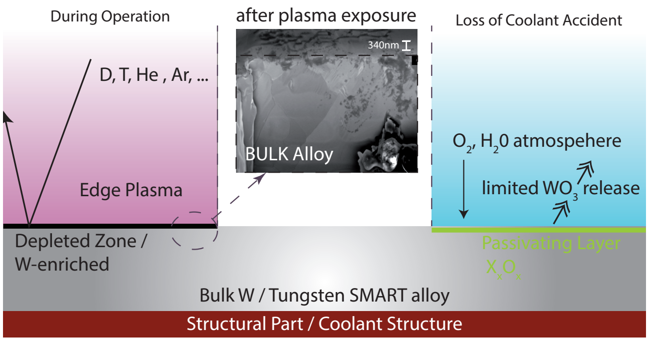


Figure 3: Working principle of a smart alloys based PFC with both the operational and accident mechanisms shown. [21]

One of the issues to be solved with the use of W in a fusion reactor is the formation of radioactive and highly volatile W-oxide (WO_3) compounds during an accident scenario [40, 41, 42]. This is mitigated by the use of so called SMART Alloys [22, 23, 6, 43, 24] which are typically produced as model-systems via Magnetron or on a larger scale via powder-metallurgically.

Enhanced erosion of light elements during normal reactor operation is not expected to be of concern. Fig. 3 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 oxides 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].

As discussed in [21] it is observed that the measured weight loss of sputtered smart alloy sample 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 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 as displayed in fig. 2, introduces additional complications.

As seen in fig. 4 W_f/W consists of multiple interchanging layers of fibres coated by an interface [17, 44, 25] 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, fibre and matrix is eroded. Interfaces currently are typically oxide ceramics[17, 29, 19]. 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 fibre 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 fibre layer would be present and exposed to the plasma. Theses issues are similar to erosion of CFC under fusion conditions discussed in [45, 46]

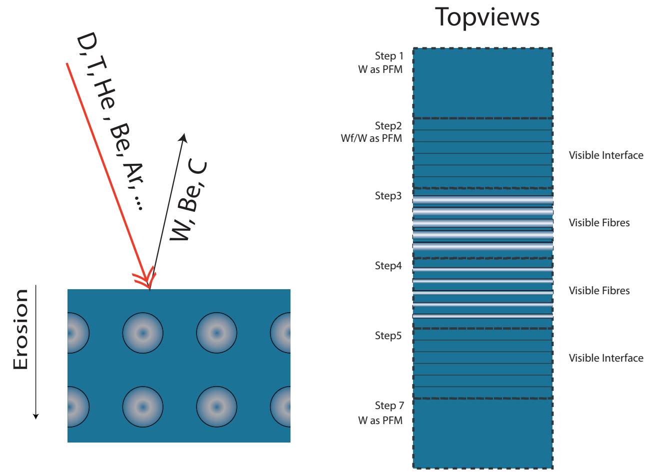
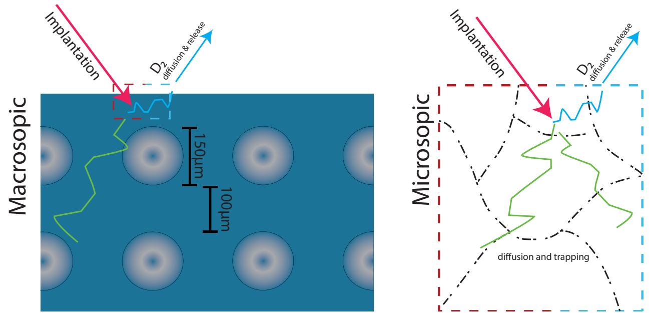


Figure 4: Different scenarios of placing W_f/W and their impact on erosion

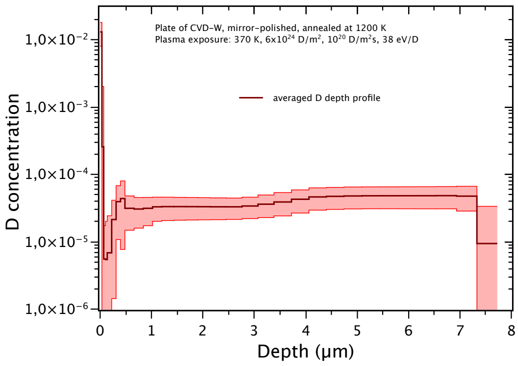
In addition to conventional composites also fine grain W is an option to strengthen and ductilize W [47] similar to other metals [48] . An option to achieve this for W is powder injection molding (PIM) [49, 50]. PIM as production method enables the mass fabrication of low cost, high performance components 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 ((60x60x20)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 [49]. Therefore, it is an ideal tool for divertor R&D as well as material science. Detrimental mechanical properties, like ductility and strength, are tunable in a wide range (example: W-1TiC and W-2Y2O3) [50]. 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.

2.3. *Fuel Retention and Hydrogen Interaction*

For several reasons fuel retention is crucial when discussion plasma material interaction 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 on 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 [51, 52].



(a)

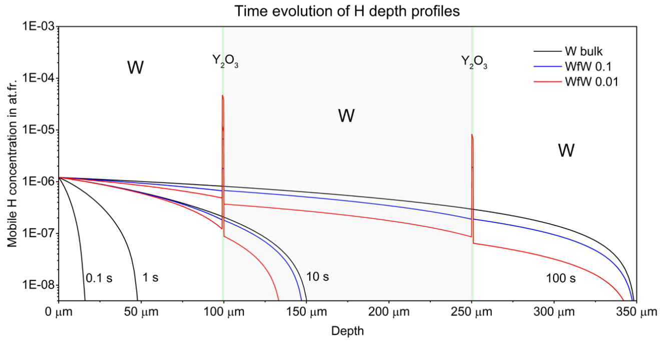


(b)

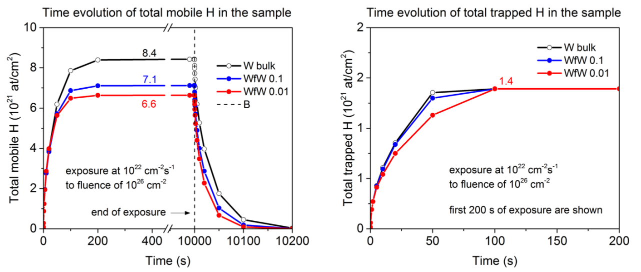
Figure 5: (a) W_f/W with respect to hydrogen retention (b) Measured retention of CVD W similar to the one used in W_f/W [18]

Fuel retention behavior of W is still subject to present studies especially when considering multi species plasma impacting together with additional heat-loads [30**?**]. It was shown that by replacing CFC with W in the Joint European Torus (JET) the retention e.g. can be significantly reduced [53] compared to e.g. Carbon. An issue that however remains 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 advanced properties such as ductility or enhanced strength one does not sacrifice aspects of safety and tritium self-sufficiency. Figure 5(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 on the grain structure, etc. Here an example for the CVD-W material used in W_f/W is given in fig. 5(b). Here 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 [54]. 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 5(a).

In the W_f/W model-system discussed below interfaces would typically made from oxide ceramics, research on their properties as tritium permeation barriers ranges over a variety of materials [55, 26, 56, 57, 58, 59, 60], including alumina, Erbia and Yttria. Permeation mitigation factors of up to 100 are reported.



(a) Depthprofiles



(b) Time Evolution

Figure 6: (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.

In order to asses in a limited 1D case the behavior of such composite structures we are using B reaction-diffusion based modeling [61, 62, 63] to detect first obvious differences of retention in composites.

The 1D calculations is based on a 5 layered model-system system W(100μm)/Y2O3(1μm)/W(150μm)/Y2O3(1μm)/ W(100μm) similar to what is shown in 5(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. Here more detailed studies regarding the interfaces used and their properties are crucial and should be motivated by this work. The fibre is currently assumed to be behaving identical to the matrix, however the microstructure is significantly different [17, 18, 29, 19] and hence detailed studies also on pure fibre 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 will allow a simple picture to be compared with expectations. The model-system was loaded with 1E22 D/m^{2}s and a fluence 1E26 D/m^2.

Figure 6 is showing the results of the modeling. In figure 6(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 fibre and subsequent layers. Based on theses assumptions the hydrogen traps are totally all filled after after 100s. In principle they fill somewhat slower but are inevitably filled despite slower diffusion in fiber-interfaces.

Figure 6(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 ratio between W-bulk, fibre etc 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 fibre , matrix and interface will play a crucial role. Assuming e.g. 30 to 50% volume fraction of fibers one can imagine quite a change in transport behavior.

Outgassing as shown in 6(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.g. solute. The issues here are related the lack of knowledge which needs to be mitigated as part of the PWI qualification of new advanced materials.

2.4. Helium Interaction

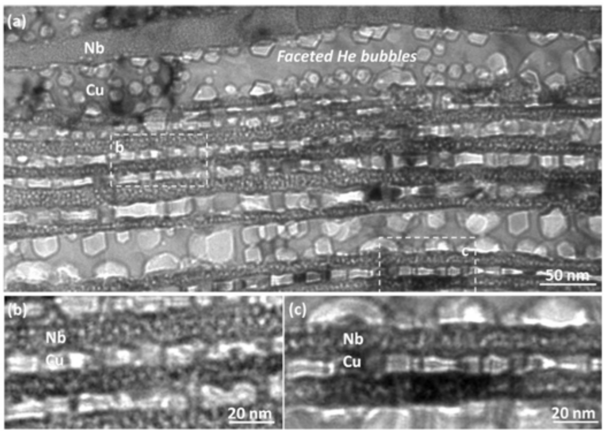


Figure 7: He precipitate networks in Cu-Nb multilayer nano-composites [64]. (b) and (c) are magnified views of the corresponding boxed areas in (a). They show incipient self-assembly of He clusters into interconnected networks

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 has been studied in various configurations.[65, 66, 67]. A series of measurements coupling plasma exposures in PISCES and DIII-D [68, 69] have been exposing W samples, with various surface morphologies, during theses experiments a mitigated erosion behavior has been found as well as no additional roughening of the surface during ELMs. Meaning that fuzz under theses condition actually improves the PWI behavior. In addition helium will cause high temperature embrittlement [34] 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 [70, 8, 71]. Transmutation into helium is however a minor problem for W [72]

On 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 [73]. In single-phase metals, these bubbles tend to decorate defects, such as grain boundaries [74, 75] or dislocations [64, 76]. 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 demonstrated in studies on He-implanted layered composites of copper (Cu) and niobium (Nb) [77, 78]. For example, Fig. 7 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 1-2nm range while the latter contains much larger, faceted bubbles. 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. 7 point to intriguing opportunities for designing metal composites that outgas He in a controllable fashion. Yuryev and Demkowicz have proposed 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 [79]. One study suggests that He may indeed outgas along interfaces between phases in metal composites without causing morphological instabilities on the sample surface [80]. 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 a proposed mechanisms might be included in W_f/W or other composites to manage to helium content and hence its detrimental effects.

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. Here a model component is comprising of a tungsten fibre re-enforced tungsten (W_f/W) [17], smart alloy [22, 23, 6, 43, 24] with interfaces based on oxide ceramics, a copper based cooling tube and integrated permeation barrier layers [26] (Fig. 2). For the matrix material it seems erosion is as critical as for 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 of hydrogen is a particularly crucial point and needs to be studied on model system and all the elements comprising the composite to allow modeling 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 in the past 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, but development of model-systems is ongoing similar to the fusion qualification.

Acknowledgement

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