TUNGSTEN AS A PLASMA FACING COMPONENT AND DEVELOPMENT OF ADVANCED MATERIALS FOR FUSION

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ABSTRACT

For the realization of fusion energy especially materials questions pose a significant challenge already today. Heat, particle and neutron loads pose a significant problem to material lifetime when extrapolating to DEMO [1, 2] the first stage prototype fusion reactor [3, 4, 5] considered to be the next step after ITER towards realizing fusion [6]. For many of the issues faced tungsten was considered the solution. Recent progress has however shown that new advanced tungsten or material grades maybe required. In particular safety relevant components such as the first wall and the divertor of the reactor can benefit from introducing new approaches such as composites or new alloys into the discussion. Cracking, oxidation as well as fuel management are driving safety issues when deciding for new materials. Considering in all this also the neutron induced effects such as transmutation, embrittlement and after-heat and activation is essential. A component approach taking into account all aspects is required.

I. BOUNDARY CONDITIONS

Figure 1: Materials Issues for fusion - incomplete

When considering a future fusion power-plant multiple interlinked issues need to be evaluated (fig. 1). Some of the main problems a future reactor is faced with are linked to the materials exposed to the fusion environment and their lifetime considerations. Already from fig. 1 one can see that at the far branches of the tree multiple times the following issues arise, cooling media, neutron flux and neutron damage, ion impact and sputtering as well as heat loads and transient events.

In the following a subset of those conditions can be evaluated only and so far only for the relatively well known conditions of the next step devices e.g. DEMO [2].

The devices called DEMO is so far considered to be the nearest-term reactor design that has the capability to produce electricity and is viewed as single step between ITER and a commercial fusion plant. Currently, no conceptual design exists apart from early studies [3, 5]. A design has not been formally selected, and detailed operational requirements are not yet available [7]. For discussion purposes it is simple to assume a reactor with the fusion power of 2GW and a wall area of $1200m^2$.

$$P_{exaust} = P_H + P_\alpha \sim 450MW \quad (1)$$
$$P_n = 1600MW/1200m^2(\sim (40dpa/5fpy[8]) \quad (2)$$
$$P_R = 225MW/1200m^2 \quad (3)$$
$$P_P = 225MW/1200m^2 \quad (4)$$

This means an average of $1.5MW/m^2$ on the first wall with $\sim 1.3MW$ coming from neutrons, typically $10-20MW/m^2$ on the divertor and not yet any transient loads taken into account. This machine is already significantly different in size and performance from the next step device, ITER. Main differences include significant power and hence neutron production ($1dpa \sim 5 \times 10^{25}n/m^2$), Tritium self sufficiency, high availability and duty cycle as well as a pulse length of hours rather than minutes. In addition, safety regulation will be more stringent both for operation and also for maintainability and component exchange [7]. A reactor might even go beyond, e.g. steady state operation.

II. PWI CONSIDERATIONS

Several issues related to materials used in its construction of a future fusion reactor need still to be tackled. Among those are the issues related to the first wall and divertor surfaces, their power handling capabilities and lifetime. For the next generation device, ITER, a solution based on actively cooled tungsten (W) components has been developed for the divertor, while beryllium will be used on the first wall [9]. The cooling medium will be water as is also considered for high heat load components in DEMO [7]. In contrast to a reactor where high wall temperature (> 300° C) facilitate energy production ITER W components are only operated at 70° C and hence in the brittle regime.

For the first wall of a fusion reactor unique challenges on materials in extreme environments require advanced features in areas ranging from mechanical strength to thermal properties. The main challenges include wall lifetime, erosion, fuel management and overall safety. For the lifetime of the wall material, considerations of erosion, thermal fatigue as well as transient heat loading are crucial as typically 10^9 (30Hz) transients, so called ELMs, are to be expected during one full power year of operation.

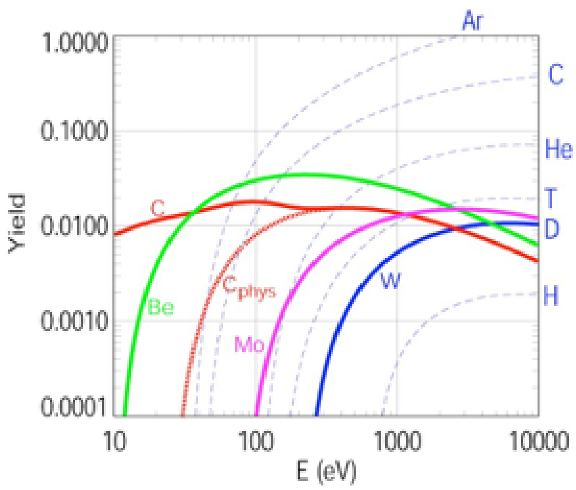


Figure 2: Sputtering yields for C, Mo, Be and W bombarded with D ions [10]. For C, chemical erosion enhances the yield at low energies and yields. For W, impurity sputtering, such as Ar ions, dominates. Based on [11, 10]

Tungsten is the main candidate material for the first wall of a fusion reactor as it is resilient against erosion (Fig. 2), has the highest melting point of any metal and shows rather benign behavior under neutron irradiation as well as low tritium retention. Erosion of the first wall and the divertor will require a significant armor thickness or short exchange intervals, while high-power transients need strong mitigation efficiency to prevent damage to the plasma facing components (PFCs) [12].

One issue that is related to the wall erosion is the fusion performance of the fusion device and hence the amount of tolerable impurities. For tungsten only minute amounts can be tolerated when considering the burn conditions of the plasma and cooling provide by tungsten radiating in the plasma. In [13] the analysis given for only helium as on of the impurities shows that 10^{-4} W atoms per deuterium atom can be enough to extinguish the fusion performance.

For the next step devices, e.g. DEMO, or a future fusion reactor the limits on power-exhaust, availability and lifetime are quite stringent. Radiation effects including neutron embrittlement may limit actively cooled W components in DEMO to about 3-5 MW/m^2 due to the diminished thermal conductivity or the need to replace CuCrZr with Steels [14]. Quite extensive studies and materials programs [15, 16, 17, 1] have already been performed hence it is assumed that the boundary conditions [14] be fulfilled for the materials are in many cases above the technical feasibility limits as they are understood today.

- High divertor power handling, i.e., ability to withstand power loads larger than 10 MW/m^2. here especially the choice of coolant is critical. Water cooling will be required to allow sufficient exhaust efficienvy

- The radiation damage for the divertor is predicted to be close to 3 dpa/fpy. For copper if chosen the value varies between 3 and 5 dpa / fpy (full power year)

- It is assumed that despite the radiation damage erosion is the dominant lifetime determining factor.

- Even when starting up DEMO in phases a final blanket should be capable of lasting up to 50 dpa.

In the following we will however try to concentrate on three groups of issues [14, 7]

- Power exhaust and energy production: The first wall blanket exhausts the power and hence must be operated at elevated temperatures to allow for efficient energy conversion. Here a material must be chosen with a suitable operational window and sufficient exhaust capability. The cooling medium for high temperature operation can be crucial.

- Mitigate material degradation due to neutrons and reduce radioactive waste: One can select materials that allow high temperature operation, mitigate effect of operational degradation such as embrittlement and neutron effects linked to transmutation.

- Tritium self-sufficiency and safety: 22 kg/year of tritium are required for a 2GW plasma operated at 20% availability, this means ~ 85% [14] of the in-vessel surface must be covered by a breeding blanket and the loss of tritium without ability to recover needs to be minimized. Accident scenarios need to be considered e.g. loss of coolant and air ingress are among the possible scenarios.

Tritium retention in plasma-facing components (PFCs) due to plasma wall interactions is one of the most critical safety issues for ITER and future fusion devices. For carbon based PFCs the co-deposition of fuel with re-deposited carbon has been identified as the main retention mechanism (fig. 3).

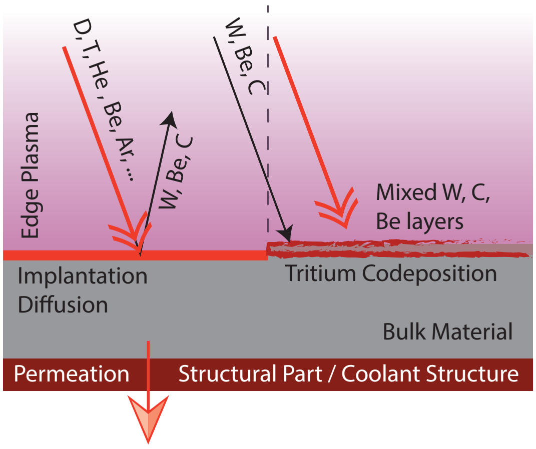


Figure 3: Fuel retetention and permeation issues under plasma exposure conditions

This retention grows linearly with particle fluence and can reach such large amounts that carbon is omitted in the activated phase of ITER and future reactors [11]. Instead, tungsten is foreseen as PFC material in the divertor of ITER and is the most promising candidate for PFCs in future reactors. Fuel retention behavior of tungsten is subject to present studies. It was shown that by replacing CFC with W in the Joint European Torus (JET) the retention e.g. can be significantly reduced [18] as predicted (Fig. 4). An issue that however remains is the potential for diffusion of hydrogen into the material. In the breeding blankets especially the interaction of tritium with Reduced Activation Ferritic Martensitic (RAFM) steels, e.g. EUROFER-97, can be crucial to minimize fuel retention or loss.

III. MATERIAL ISSUES FOR TUNGSTEN

In the following sections several issues are described that arise from the above depicted boundary conditions. As an example the divertor lifetime is considered as the desired parameter. Typically there are three main avenues of damage to the material of the divertor. Either high heat-loads cause melting, cracking or recrystallization or neutrons impact the actual microstructure of the material. Surfaces

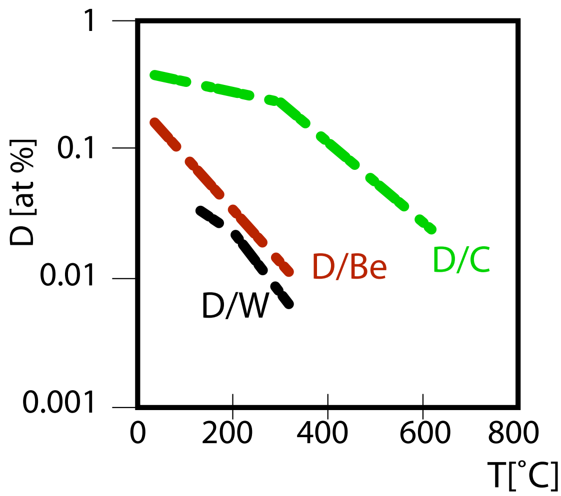


Figure 4: Estimate of retained deuterium concentration in C, Be and W deposits under codeposition conditions. (Sketch based on [11])

are damage by ions impacting and causing both surface morphology changes or erosion. Fig. 5 depicts hence one approach to solving at least some of the problems. Choosing Tungsten (W) as the main wall material suppresses sputtering due to the high atomic mass in contrast to the sputtering ions. Tungsten also has a rather high thermal conductivity (Cu: $\sim$ 390 W/(mK) W: $\sim$ 173 W/(mK) Mo: $\sim$ 138 W/(mK) Steel: $\sim$ 17 W/(mK) and can hence facility higher heat exhaust than e.g. steel, for tungsten also the high melting point is beneficial. Thermal properties however are intrinsically linked to potential transmutation and irradiation processes. In addition it is know that tungsten has a rather low hydrogen solubility and hence facilitates low retention under fusion conditions [18]. Tungsten is however inherently brittle and does show catastrophic oxidation behavior at elevated temperatures.

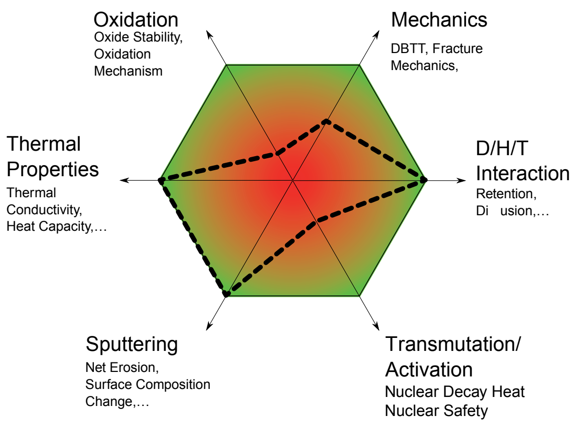


Figure 5: Tungsten as a first wall material

Not always all material properties can be optimized at once. After an optimization step a material might be developed that in its entirety fulfills all criteria by interaction between individual criteria as displayed in fig. 6

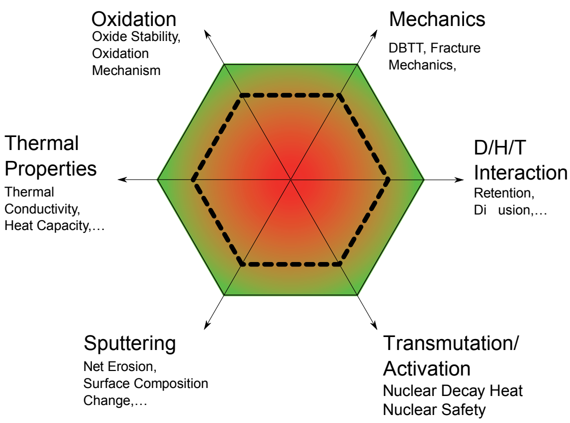


Figure 6: A Compromise 1st Wall Material

A. Operational Window

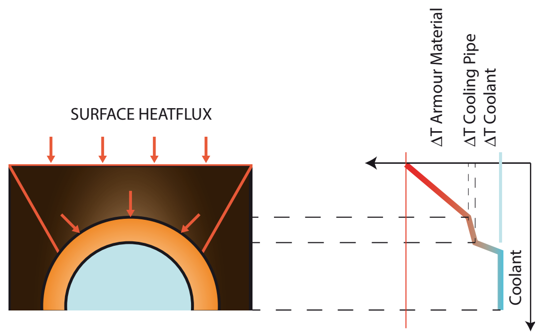
Based on the assumption that W is the option so far to be used as the surface layer the reactor PFCs already quite basic assumptions can be made when picking the operational window and thickness of such components.

The lower operating temperature limit in metal alloys is mainly determined by radiation embrittlement (decrease in fracture toughness), which is generally most pronounced for irradiation temperatures below $\sim 0.3 T_{melt}$, where T_{melt} is the melting temperature (Tungsten $\sim 3300K$) [19]. The upper operating temperature limit is determined by one of four factors, all of which become more pronounced with increasing exposure time such as thermal creep (grain boundary sliding or matrix diffusional creep), high temperature helium embrittlement of grain boundaries, cavity swelling (particularly important for Cu alloys), and coolant compatibility such as corrosion issues.

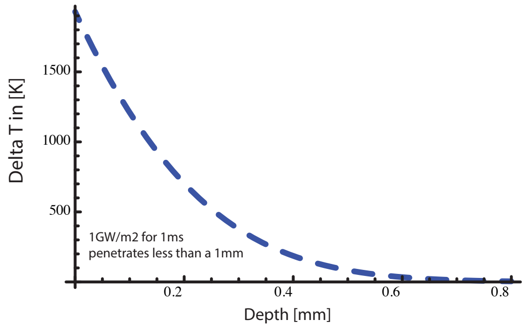
If the PFCs surface is operated at 1100 ° C as optimal for W [20] and copper is chosen together with water as part of the coolant solution the thickness is automatically determined (5) with κ the heat conductivity)

$$q = \frac{T_{surface} - T_{cool}}{d_1/\kappa_1 + d_2/\kappa_2} \qquad (5)$$

This means that the maximum heat-exhaust is determined by the heat conduction, the potential for recrystallization and the ductile to brittle transition behavior of the material. Here new material options are required to allow a larger operational window, by overcoming the limiting factor, keeping in mind that a maximized heat conduction is crucial (e.g. Steel).



(a) Steady state heat flux in a conventional monoblock like structure



(b) Heat flux penetration (in tungsten) from 1GW, 1ms event

Figure 7: Power-exhaust - Issues arising from steady state and transients

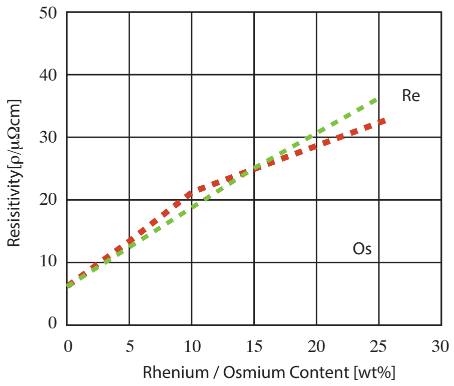
For transient events the limits can even be more stringent when considering the limited penetration depth of a given heat-pulse fig. 7(b) and its maximum surface temperature rise (egn. (6)) with κ the heat conductivity, ρ the density and c the heat capacity). Active cooling for fast transients is meaningless because of the small penetration depth.

$$\Delta T^{\infty}_{surface}(t) = \frac{q_s}{\sqrt{\kappa\rho c} \cdot \sqrt{\pi}} \sqrt{\Delta t} \qquad (6)$$

From assumptions related to unmitigated ELMs at 1 GW/m^2 for 1ms [12] already a temperature rise of 1500K is achieved in only the top 1 mm. Cracking or melting is difficult to prevent here. Irreparable damage has to be avoided at any cost. Even higher thermal wall loads caused by so called disruptions, sudden and uncontrolled loss of the plasma with deposition of the energy on the wall. Assuming that 50% of the thermal energy are radiated during thermal quench of the plasma and with a limited inhomogeneity in toroidal and poloidal direction respectively the thermal disruption loads are always much above the crack limit [21] even-though below the melt limit. Variation of the torus geometry (aspect ratio) provides only moderate reduction of loads.

B. Evolution of Thermal Properties

In addition to the above mentioned issues fig. 8 shows that the fusion environment can also drastically



(a) Electrical resistivity of W containing various amounts of Re or Os. The red line and green line stand for W-xRe and W-xOs respectively [22]

Figure 8: Change of electrical and thermal properties of tungsten under neutron irradiation and transmutation

change some of the set assumptions. Already a small amount of transmutation can have a significant influence on the power-exhaust. When calculating the thermal conductivity based on $\kappa \cdot \rho = L \cdot T$ with κ the thermal conductivity, ρ the resistivity and L the Lorentz number with a value of $3.2 \times 10^{-8} W\Omega K^{-2}$ for tungsten one can estimated that κ drops 60% already at 5wt% or Re or Os. From previous work [23] one can determine that especially at lower temperatures κ drops significantly (30%). In any case one does depend on stable and predictable material properties even under radiation - or a detailed knowledge of the time dependent evolution to determine lifetime and performance of components.

C. Embrittlement

Conventional high performance materials offer high strength and stiffness combined with low density hence weight. However, a fundamental limitation of the current approach is the inherent brittleness of tungsten. As seen above cracking hence brittle behavior can be a limiting factor when operating any PFC in a tokamak [21]. For the fusion environment the additional problem becomes operational embrittlement. An issue related to embrittlement is certainly the recrystallization of tungsten. at temperatures of 1400K only mere hours are required to complete recrystallize the material [24].

Fig. 9 shows that already at moderate neutron fluence corresponding to 1 dpa the DBTT of tungsten moves up to almost 900° C. If in addition recyrstallisation takes place (fig. 9) almost no structural load can be given to the tungsten component at temperatures of a few hundred degrees. For a typical mono-block [12, 23] a tungsten thickness of 6mm on top of the CuCrZr cooling pipe would mean, based on simple estimations (egn. 5) that only the top part of a exposed mono-block would be in the allowed temperature range [20]. This means for a water-cooled solution tungsten is normally a brittle hence only a functional part, suppressing e.g. erosion and allowing for high operational temperatures. Failure is usually sudden and catastrophic, with no significant damage or warning and little residual load-carrying capacity if any. Structures that satisfy a visual inspection may fail suddenly at loads much lower than expected. Cracking is usually avoided for PFCs and certainly for structural components.

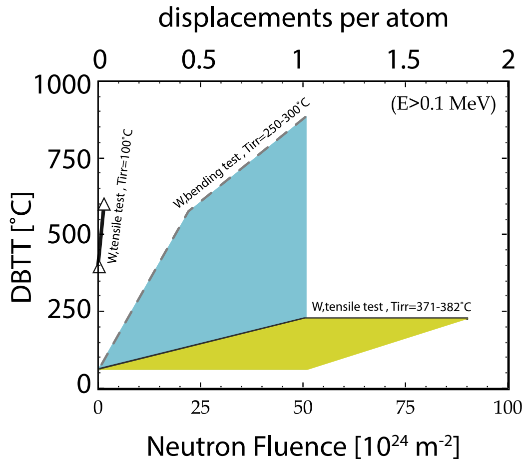


Figure 9: DBTT dependence after neutron irradiation based on [25]

D. Activation & Transmutation

An issue that especially for complex components with multiple material and alloying components can be quite crucial is the recyclability and activation under neutron irradiation. As fusion is typically considered a technology with minimal or now longterm nuclear waste [25] tungsten and e.g. special steel grades [26] have optimized radiation performance with respect to low activation, e.g. molybdenum and aluminium are avoided as they produce long term activation products [8, 25]

Based on a study provided in [5, 8] with a neutron flux at the first wall of $\sim 1.0^{15} ncm^{-2}s^{-1}$ one can estimate the activation of materials after a 5 year period. For materials exposed in the divertor a factor 10 lower neutron rate is expected in the area of the high heat flux exposure due to geometrical reasons [7].

Fig. 10 shows the values of an assumed component containing W, Cr, Cu and Er, representing e.g. a typical mono-block with small interlayers and a copper cooling structure. Already here it is clear that the shielded hands on radiation level can not be achieved after 100 years when using copper cooling at the first wall. Mitigation of these effects need to be consid-

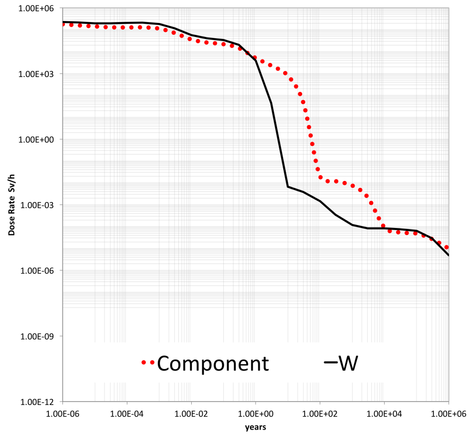


Figure 10: The activation of tungsten (first wall) is shown in comparison to a component (W 79.7wt%, Er 0.6wt%, Cr 12.1wt%, Cu 7.5%) for the first wall can be estimated as an upper bound (based on [8]). Divertor components in general are less prone to activation. Shielded hands-on level: $2\,mSv/h$, Hands-On Level: $10\,\mu Sv/h$

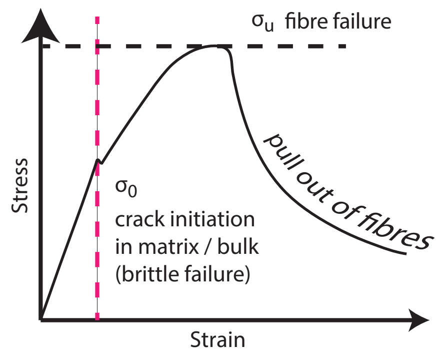
ered by utilizing non or low activation materials. e.g. replacing copper for the first wall and removing Er or Al oxides in favor of Ytrria.

IV. NEW MATERIAL OPTIONS

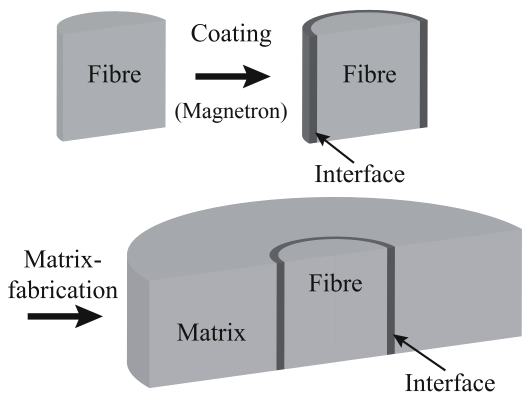
For all the above described issues or boundary condition potential solutions need to be developed. We are faced with a multilayer approach for the Plasma-Facing-Components (PFCs) including armor, fuel barriers, cooling structures & breeding elements and hence we have to consider a multitude of interacting materials. From the plasma toward the cooling structure we consider tungsten or tungsten alloys on either a copper or steel structure with functional layers e.g. permeation barriers or compliance layers. A generally new components concepts to circumvent classical definitions of limits is required with damage resilient materials such as composites followed by a much better definition what can be tolerated before a component needs to be exchanged. We need to define lifetime with more parameters than erosion and cracking for PFCs. Composite approaches to enhance material parameters and mitigate damage modes by utilizing mixed properties will be ideal inclunding safety features like passivating alloys etc. Not yet developed ideas on self-healing or damage tolerant materials similar to aerospace applications might be a future field of research including e.g. liquid metals [27]. Already today smart materials, fiber composites and alloys which adapt to the operational scenario are possible. In some cases detrimental effects such as erosion are actually used to facilitate material functions (sec.). If W as a 1st wall material is required to suppress erosion even preferential sputtering can turn the top layer of alloys or steel into a thin layer of erosion suppressing tungsten [28, 29, 30].

A. Composites for High Loads

A basic strategy to achieve pseudo-ductility is the incorporation of new ductile matrices and fibres, which needs extensive development and validation [31]. To overcome brittleness issues when using W, a W-fiber enhanced W-composite material (Wf/W) incorporating extrinsic toughening mechanisms can be used.The composite approach enables energy dissipation and thus stress peaks can be released at crack tips and cracks can be stopped. Another option is a composite laminates made of commercially available raw materials [32, 17].



(a) Strain-Stress-Curve for a typical Composite Material



(b) fibre enhanced composite and interface layer

Figure 11: Composite approaches based on pseudo-ductilisation.

Accordingly, even in the brittle regime this material allows for a certain tolerance towards cracking and damage in general . In comparison conventional tungsten would fail immediately. From fig. 11(a) the principle of composite strengthening behavior can be seen. Even when a crack has been initiated inside the material the energy dissipation mechanisms allow further load to be put towards the component until at a later stage also the fiber and hence the overall material fails.

First W_f/W samples have been produced, showing extrinsic toughening mechanisms similar to those of ceramic materials [33, 34]. These mechanisms will also help to mitigate effects of operational embrittlement due to neutrons and high operational temperatures. A component based on W_f/W can be developed with both chemical infiltration (CVI), utilizing a newly installed CVI-setup and a powder metallurgical path through hot-isostatic-pressing [35, 36]. Crucial in both cases is the interface between fiber and matrix. The interface is a thin layer which provides a relatively weak bond between the fiber [37] and the matrix for enabling pseudo-ductile fracture in the inherently brittle material, similar to e.g. SiC ceramics [38].

Keeping in mind the above mentioned boundary conditions one can consider that brittleness from either neutron irradiation or elevated temperatures can be mitigated as the pseudo-ductilisation does not rely on any part of the material being ductile, crack resilience can be established [33, 34]. Facilities to produce both CVI as well as powder metallurgical W_f/W are now available. It now needs to be shown that for those components equally good behavior in terms of thermal conductivity, erosion and retention can be established. As part of the development especially the choice of the fiber and interface material can be crucial. A sag-stabilized potassium doped fibre can even retain some ductility in addition strengthening the material. For the interface a non activating choice is necessary hence one can move from the so far considered erbia [37, 33] potentially towards yttria.

In addition to conventional composites also fine grain tungsten is an option to strengthen and ductilize tungsten [39] similar to other metals [40] an option to achieve this for W & DEMO applications is Powder Injection Molding (PIM) [41, 42]. Powder Injection Molding (PIM) as production method enables the mass fabrication of low cost, high performance components with complex geometries. The range in dimension of the produced parts reach from a micro-gearwheel ($d = 3mm, 0.050g$) up to a heavy plate ((60x60x20)mm, 1400 g). Furthermore, PIM as special process allows the joining of tungsten and doped tungsten materials without brazing and the development of composite and prototype materials. Therefore, it is an ideal tool for divertor R&D as well as material science.

B. Tungsten smart alloys

Addressing the safety issue, a loss-of-coolant accident in a fusion reactor could lead to a temperature rise of 1400 K after $\sim 30-60$ days due to neutron induced afterheat of the in-vessel components [5].

Thereby, a potential problem with the use of W in a fusion reactor is the formation of radioactive and highly volatile WO_3 compounds. In order to suppress the release of W-oxides tungsten-based alloys containing vitrifying components seem feasible, as they can

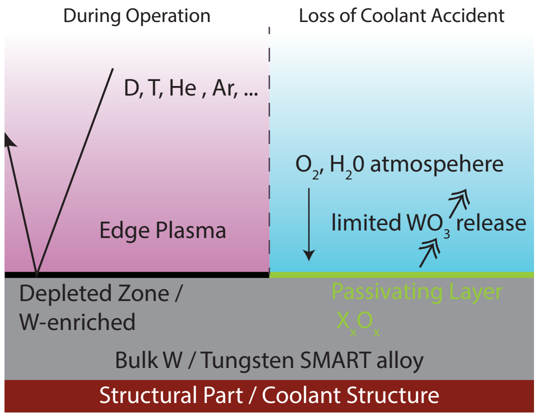


Figure 12: Working principle of a smart alloys based PFC with both the operational and accident mechanisms shown.

be processed to thick protective coatings with reasonable thermal conductivity, e.g. by plasma spraying with subsequent densification. Enhanced erosion of light elements during normal reactor operation is not expected to be a concern. Preferential sputtering of alloying elements leads to rapid depletion of the first atomic layers of light alloying elements and leaves a pure W-surface facing the plasma [43]. This mechanism is similar to the above mention EUROFER-97 surface enrichment. Fig. 12 displays the basic mechanism. During operation plasma ions erode the light constituents of the alloy leaving behind a thin depleted zone with only tungsten remaining. Subsequently the tungsten layer suppresses further erosion hence utilizing the beneficial properties of tungsten. In case of a loss of coolant and air or water ingress the tungsten layer oxides releasing a minima amount of WO3 and then passivating the alloy due to the chromium content. W-Cr-Y with up to780 at% of W content already shows 10^4-fold suppression of tungsten oxidation due to self-passivation [44]. Test systems are being produced via magnetron sputtering and evaluated with respect to their oxidation behavior. Production of bulk samples is ongoing. Rigorous testing of oxidation behavior, high heat flux testing and plasma loads as well as mass production for candidate materials is under preparation. The material can be considered for both first wall and divertor applications especially when combined with the strengthening properties of the W_f/W composite approach. The PWI behavior and potential neutron or temperature embrittlement need to be quantified.

C. Functionally Graded Materials

Having discussed tungsten as the main candidate for the PFMs of a fusion reactor the joint to the cooling structure or wall structure in general is crucial. From the values of thermal expansion for the different materials (copper $\sim 16.5\mu m/(mK)$, tungsten: $\sim 4.5\mu m/(mK)$ molybdenum: $\sim 4.8\mu m/(mK)$, stainless steel: $\sim 12\mu m/(mK)$ it is clear that a mature solution of joining them needs to be established.

As one of the example systems the development of Functionally Graded Materials (FGMs) between W as the PFM with the structural material EUROFER-97 can be considered. As depicted in [45] FGMs are a candidate especially when considering applications such as the blanket modules of a DEMO [7] or even a helium cooled tungsten divertor with low to medium heat-flux ($1 - 5MW/m^2$) for which the heat conductivity of EUROFER-97 maybe sufficient.

Similar ideas are developed for the transition between copper and W [46, 47] potentially being used as solution for a water-cooled high heat-flux divertor [7, 14]

D. Tritium Management

Moving towards the actual structural part of the reactor tritium management is an issue especially for the breeding blankets. In order to prevent tritium loss and radiological hazards it is important to suppress permeation through the reactor walls. Research on permeation barriers ranges over a variety of materials [48, 49, 50, 51] including erbia and alumina. Permeation barriers require high permeation reduction factors, high thermal stability and corrosion resistance as well as similar thermal expansion coefficients compared to those of the substrate. Establishing the permeation mitigation requires controlled experiment. A new gas-driven permeation setup is established at FZJ to investigate deuterium permeation through different ceramic coatings on EUROFER-97, which significantly reduce the deuterium permeation. Several techniques to apply the coatings can be considered e.g. Arc Deposition, Chemical Routes, Magnetron Sputtering. A mitigation factor of 50-100 is essential to allow safe operation and allow a reasonable tritium breeding ratio.

In addition to permeation mitigation and mechanical feasibility, compatibility with neutron irradiation needs to be enforced. Here especially erbia and alumina but also zirconia [52] do have issues. Permeation barriers from Ytrria [53] may be a potential low activation element element (fig. 10) and in addition is quite similar in terms of thermal expansion when considering EUROFER-97 as the substrate.

V. SUMMARY AND OUTLOOK

Considering all the above mentioned issues when using materials in a fusion reactor environment a highly integrated approach is required. The lifetime of PFCs and joints due to erosion, creep, thermal cycling, embrittlement needs to be compatible with steady state operation and short maintenance intervals.Thermal properties of composites and components have to be at least similar to bulk materials when enhanced properties in terms of strength are not to hinder the maximization of operational performance. Damage resilient materials can here facilitate small, thin components and hence higher exhaust capabilities. The components need to be compatible with the aim of tritium breeding and self-sufficiency and hence mitigate tritium retention and loss.

Despite using various alloying components, interlayers or coatings maintainability and recycling of used materials is required to make fusion viable and publicly acceptable. Last but not least, large scale production of advanced materials is crucial. We hence propose to utilize the composite approach together with alloying concepts to maximize the potential of the tungsten part of a potential PFC. Together with W/Cu composites at the coolant level and W/EUROFER joints high-performance components can be developed. Rigorous testing with respect to PWI and high heat-flux performance are planned for all concepts to have prototype components available within 5 years for application in existing fusion devices.

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