

Materials for DEMO and Reactor Applications - Boundary Conditions and New Concepts

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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. Heat, particle and neutron loads are a significant problem to material lifetime when extrapolating to DEMO. For many of the issues faced advanced materials solution are under discussion or already under development. In particular 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 issues when deciding for new materials. Here W_f/W Composites as well as strengthened CuCrZr components together with oxidation resilient tungsten alloys allow the step towards a fusion reactor. In addition, neutron induced effects such as transmutation, embrittlement and after-heat and activation are essential. Therefore, when designing a component an approach taking into account all aspects is required.

1. Introduction and Boundary Conditions

When considering a future fusion power plant multiple intertwined 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

5 [1, 2]. Already from fig. 1 one can see that at the far branches of the tree multiple times
6 the following issues arise: cooling media, neutron flux and neutron damage, ion impact
7 and sputtering as well as heat loads and transient events.

8 In the following only a subset of those conditions can be evaluated and so far only
9 for the relatively well known conditions of the next step devices such as ITER & DEMO
10 [2].

11 *1.1. DEMO Conditions*

12 DEMO is presently considered to be the nearest-term reactor design that has the
13 capability to produce electricity and is viewed in Europe [3] as a single step between
14 ITER and a commercial power plant [4, 5, 6]. Currently, no conceptual design exists for
15 DEMO apart from early studies. A design has not been formally selected, and detailed
16 operational requirements are only now being developed [7], hence for discussion purposes
17 we simply assume a reactor with the with fusion power of 2GW as given in [2, 8]. From
18 the assumptions presented in [2] an average of typically $10 - 20 \text{ MW/m}^2$ on the divertor
19 is to be expected and with wall loading around $\sim 1 - 1.5 \text{ MW}$. For the neutron loading
20 one can refer to [9] with (40dpa / 5fpy (full power year))

21 This machine is already significantly different in size and performance from the
22 next step device, ITER. Main differences include significant power and hence neutron
23 production , tritium self sufficiency, high availability and duty cycle as well as a pulse
24 length of hours rather than minutes. In addition, safety regulation will be more stringent
25 both for operation and also for maintainability and component exchange [7]. A reactor
26 might even go beyond these requirements, e.g. steady state operation.

27 1.2. First Wall and Divertor

28 Several issues related to materials used in the construction of a future fusion reactor
29 need still to be tackled. Among those are the issues related to the first wall and divertor
30 surfaces, their power handling capabilities and lifetime. For the next generation device,
31 ITER, a solution based on actively cooled tungsten (W) components has been developed
32 for the divertor, while beryllium will be used on the first wall [10, 11]. The cooling
33 medium will be water as is also considered for high heat load components in DEMO
34 [7]. For the first wall of a fusion reactor unique challenges on materials in extreme
35 environments require advanced features in areas ranging from mechanical strength to
36 thermal properties. The main challenges include wall lifetime, erosion, fuel management
37 and overall safety. For the lifetime of the wall material, considerations of thermal fatigue
38 as well as transient heat loading are crucial as typically 10^9 (30Hz) thermal transients
39 (ELMs) during one full power year of operation are to be expected. Tungsten (W)
40 is currently the main candidate material for the first wall of a fusion reactor as it is
41 resilient against erosion, has the highest melting point of any metal and shows rather
42 benign behavior under neutron irradiation, as well as low tritium retention. Erosion of
43 the first wall and the divertor will in addition require a significant armor thickness or
44 short exchange intervals, while high-power transients need strong mitigation efficiency
45 to prevent damage of the PFCs. [11].

46 For the next step devices, e.g. DEMO, or a future fusion reactor the limits on
47 power exhaust, availability and lifetime are quite stringent. As conventional mono-
48 blocks are allowing for $10\text{MW}/\text{m}^2$ [11] and transmutation and radiation damage can
49 quickly diminish the thermal conductivity to 50% [12]. Radiation effects including
50 neutron embrittlement may hence limit actively cooled W components in DEMO to
51 about 3-5 MW/m^2 due to the diminished thermal conductivity or the need to replace

52 CuCrZr with Steels with its low thermal conductivity [13, 8]. Quite extensive studies
53 and materials programs [14, 15, 16, 1] have already been performed hence it is assumed
54 that the boundary conditions [8] be fulfilled for the materials are in many cases above
55 the technical feasibility limits as they are understood today.

- 56 • Extended power handling, i.e., ability to withstand power loads larger than
57 10 MW/m^2 . Here especially the choice of coolant is critical. Water cooling might
58 be required to allow sufficient exhaust at given acceptable pumping power [2, 8].
- 59 • The radiation damage for the divertor is predicted to be close to 3 dpa/fpy. For
60 copper if chosen the value varies between 3 and 5 dpa / fpy
- 61 • It is assumed that despite the radiation damage erosion of the armor is the dominant
62 lifetime determining factor. Here it needs to be considered that maximum thickness is
63 also determined by the required neutron transmission required for tritium breeding.
- 64 • Even when starting up DEMO in phases a final blanket could be required to
65 withstand up to 50 dpa in order to minimize the exchange frequency.

66 In the following we will however try to concentrate on three groups of issues [8, 7]

- 67 • Power exhaust and energy production: The first wall blanket exhausts the neutron
68 power and hence must be operated at elevated temperatures to allow for efficient
69 energy conversion. Here a material must be chosen with a suitable operational
70 temperature window and sufficient exhaust capability. The cooling medium for
71 high temperature operation can be crucial.
- 72 • Mitigate the effect of material degradation due to neutrons and reduce radioactive
73 waste: One can select materials that allow high temperature operation, mitigate
74 effect of operational degradation such as embrittlement and neutron effects linked
75 to transmutation.

- 76 • Tritium self-sufficiency and safety: 22 kg/year of tritium are required for a 2GW
77 plasma operated at 20% availability, this means $\sim 85\%$ [8] of the in-vessel surface
78 must be covered by a breeding blanket and the loss of tritium without ability to
79 recover needs to be minimized.
- 80 • Accident scenarios need to be considered e.g. loss of coolant and air ingress are to
81 be considered.

82 2. Material Issues

83 As an example the divertor lifetime is considered as the leading parameter. Fig. 2(a)
84 depicts what typically is seen as the main avenues of damage to the material of the
85 divertor. Either high heat-loads cause melting, cracking or recrystallization or neutrons
86 impact the actual microstructure of the material. Surfaces are damaged by impacting
87 ions causing both surface morphology changes and erosion.

88 Fig. 2(b) depicts hence one approach to solve at least some of the problems.
89 Choosing tungsten (W) as the main armor material suppresses sputtering due to the
90 high atomic mass compared to the sputtering ions. Tungsten also has a rather high
91 thermal conductivity (Values at RT; W: $\sim 173\text{W}/(\text{mK})$, Cu: $\sim 390\text{W}/(\text{mK})$, steel:
92 $\sim 17\text{W}/(\text{mK})$) and can hence facilitate higher heat exhaust than e.g. steel. For
93 tungsten also the high melting point is beneficial. Thermal properties, however are
94 intrinsically linked to potential transmutation and irradiation processes (sec. 2.4).
95 In addition, tungsten has a rather low hydrogen solubility and hence facilitates low
96 retention under fusion conditions [17, 18]. Tungsten is, however, inherently brittle and
97 does show catastrophic oxidation behavior at elevated temperatures.

98 *2.1. Operational Window*

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

102 The lower operating temperature limit in metals and alloys is mainly determined
 103 by radiation embrittlement (decrease in fracture toughness), which is generally most
 104 pronounced for irradiation temperatures below $\sim 0.3 T_{melt}$, where T_{melt} is the melting
 105 temperature (tungsten $\sim 3600K$) [19]. The upper operating temperature limit is
 106 determined by one of four factors, all of which become more pronounced with increasing
 107 exposure time such as thermal creep (grain boundary sliding or matrix diffusional creep),
 108 high temperature helium embrittlement of grain boundaries, cavity swelling (particularly
 109 important for Cu alloys), and coolant compatibility such as corrosion issues.

110 This is depicted in fig. 3(b) with fig. 3(a) showing the thus given operational
 111 conditions for a given cooling structure. Fig. 3(b) also indicates with arrow the direction
 112 in which new or advanced materials should extend the operational windows.

113 If the PFC surface is operated around 800 K inside the operational window for W,
 114 and copper is chosen together with water as part of the coolant solution the thickness
 115 (d_i) is automatically determined (with κ_i the heat conductivity). In a simplified one-
 116 dimensional approach for two materials (1,2) one can write:

$$117 \quad q = \frac{T_{surface} - T_{cool}}{d_1/\kappa_1 + d_2/\kappa_2} \quad (1)$$

118 This means that the maximum heat exhaust is determined by the heat conduction,
 119 the potential for recrystallization and the ductile-to-brittle transition behavior of the
 120 material. For a real component this simple approximation will not hold and
 121 temperature gradients along the surface will be present - causing additional thermal
 122 stress and inhomogeneous changes in material properties. Here new material options

123 are required to allow a larger operational window, by overcoming brittleness issues,
 124 keeping in mind that a maximized heat conduction is crucial (e.g. steel).

125 For transient events the limits can even be more stringent when considering the
 126 limited penetration depth of a given heat pulse fig. 3(c) and its maximum surface
 127 temperature rise (eqn. (2)) with κ the heat conductivity, ρ the density and c the heat
 128 capacity).

$$129 \quad \Delta T_{surface}^{\infty}(t) = \frac{q_s}{\sqrt{\kappa\rho c} \cdot \sqrt{\pi}} \sqrt{\Delta t} \quad (2)$$

130 Active cooling for fast transients is not relevant because of the small penetration
 131 depth. From assumptions related to unmitigated ELMs at 1 GW/m² for 1ms [11]
 132 already a temperature rise of 1500K is achieved within only the top 1 mm. Despite
 133 the ELMs being poloidally distributed along the target an unmitigated ELM can still
 134 deposited 1 GW/m² locally. Additions along the poloidal directions will only aggravate
 135 the problem.

136 Cracking or melting is difficult to prevent under such conditions , a loss of control
 137 might push close to the limit as the top surface at 10-20 MW/m² will already operate
 138 close to 2000 °C [2] . Irreparable damage has to be avoided. Fig. 3(d) depicts even
 139 higher thermal wall loads caused by so called disruptions, sudden and uncontrolled loss
 140 of the plasma with deposition of the energy on the wall. Assuming that 50% of the
 141 thermal energy is radiated during the thermal quench of the plasma and with a limited
 142 toroidal and poloidal inhomogeneity of two respectively the thermal disruption loads
 143 are always much above the crack limit of W [20] although still below the melt limit.
 144 Variations of the torus geometry (aspect ratio) provides only a moderate reduction of
 145 the thermal loads [21].

146 2.2. Evolution of Thermal Properties

147 In addition to the above mentioned issues fig. 4(a) shows that the fusion environment
 148 can also drastically change some of initial material parameters. Already a low amount of
 149 transmutation can have a significant influence on the power-exhaust. When calculating
 150 the thermal conductivity based on $\kappa \cdot \rho = L \cdot T$ with κ the thermal conductivity, ρ the
 151 resistivity and L the Lorenz number with a value of $3.2 \times 10^{-8} W \Omega K^{-2}$ for tungsten one
 152 can estimate that κ drops by 60% already at 5 wt% of Re or Os. Already insignificant
 153 transmutation irradiation can change the thermal properties of W. From fig.4(b) one can
 154 determine that especially at lower temperatures κ drops significantly. In any case stable
 155 and predictable material properties are necessary even under radiation - or a detailed
 156 knowledge of the time dependent evolution, to determine lifetime and performance of
 157 components.

158 2.3. Embrittlement

159 Conventional high performance materials offer high strength and stiffness combined with
 160 low density hence weight. However, a fundamental limitation is the inherent brittleness
 161 of tungsten. As seen above cracking hence brittle behavior can be a limiting factor when
 162 operating any tungsten based PFCs in a tokamak [20]. For the fusion environment the
 163 additional problem results from operational embrittlement.

164 Fig. 5(a) shows that already at a moderate neutron fluence corresponding to 1
 165 dpa the DBTT of tungsten moves up to almost 630K. If in addition recrystallisation
 166 occurs (fig. 5) almost no structural load can be given to the tungsten component at
 167 temperatures of a few hundred degrees. For a typical mono-block [11, 22] a tungsten
 168 thickness of 6mm on top of the CuCrZr cooling pipe would mean, based on simple
 169 estimations (eqn. 1) that only the top part of a exposed mono-block would be in the

170 allowed temperature range specified in fig. 3(b). This means for a water-cooled solution
171 tungsten is normally brittle hence only a functional part, suppressing e.g. erosion and
172 allowing for high operational temperatures. Failure is usually sudden and catastrophic,
173 with no significant damage or warning and little residual load-bearing capacity if any.
174 Structures that satisfy a visual inspection may fail suddenly at loads much lower than
175 expected. Cracking must usually be avoided for PFCs and certainly for structural
176 components.

177 *2.4. Activation and Transmutation*

178 An issue that can be quite crucial, especially for complex components with multiple
179 material and alloying components, is the activation and subsequent recyclability under
180 neutron irradiation. As fusion is typically considered a technology with minimal or
181 no long term nuclear waste [23] tungsten and special steel grades [24] have optimized
182 radiation performance with respect to low activation. Molybdenum and aluminium are
183 avoided as they produce long term activation products [9, 23]

184 Fig. 6(a) shows the activation behavior for various elements under a typical
185 fusion neutron exposure with a duration of five years for materials exposed at the
186 first wall. Based on a study provided in [6, 9] with a neutron flux at the first wall
187 of $1.0 \times 10^{15} \text{ncm}^{-2}\text{s}^{-1}$. For materials exposed in the divertor a factor 10 lower neutron
188 rate is expected in the area of the high heat flux exposure due to geometrical reasons
189 [7].

190 Fig. 6(b) shows the values of an assumed component containing W, Cr, Cu and
191 erbium, representing e.g. a mono-block with small interlayers and a copper cooling
192 structure. Already here it is clear that the shielded hands on radiation level can not
193 be achieved after 100 years when using copper in cooling structures at the first wall.
194 Mitigation of these effects need to be considered by utilizing non or low activation

195 materials. e.g. replacing copper for the first wall and removing Er or Al oxides in favor
196 of yttria.

197 *2.5. Retention and Permeation*

198 Tritium retention in PFCs due to plasma-wall interactions is one of the most critical
199 safety issues for ITER and future fusion devices. For carbon-based PFCs the co-
200 deposition of fuel with re-deposited carbon has been identified as the main retention
201 mechanism (fig. 7). This retention grows linearly with particle fluence and can reach
202 such large amounts that carbon is omitted in the activated phase of ITER and therefore
203 basically excluded for future reactors due its large issues related to retention [17] .
204 Instead, tungsten is foreseen as PFC material in the divertor of ITER and tungsten based
205 alloys are the most promising candidates for PFCs in future reactors. Fuel retention
206 behavior of tungsten is subject to present studies. It was shown that by replacing
207 CFC with W in the Joint European Torus (JET) the retention e.g. can be significantly
208 reduced [18]. An issue that however remains is the potential for diffusion of hydrogen
209 into the material. In the breeding blankets especially the interaction of tritium with
210 reduced activation ferritic martensitic (RAFM) steels (e.g. EUROFER-97) can be
211 crucial to minimize fuel retention or loss.

212 **3. New Material Options**

213 For all the above determined issues or boundary conditions potential solutions need to
214 be developed. We are faced with a multilayer approach for the PFCs including armor,
215 fuel barriers, cooling structures & breeding elements and hence we have to consider a
216 multitude of interfacing materials. From the plasma towards the cooling structure we
217 consider tungsten or tungsten alloys on either copper or steel based structures with

218 functional layers, e.g. permeation barriers or compliance layers. A generally new
219 components concept to circumvent classical definitions of limits is required, applying
220 damage resilient materials such as composites, followed by a much better definition what
221 can be tolerated before a component needs to be exchanged. We need to define lifetime
222 for PFCs with more parameters than erosion and cracking. Composite approaches to
223 enhance material parameters and mitigate damage modes by utilizing mixed properties
224 will be ideal including safety features like passivating alloys etc. Not yet developed ideas
225 on self-healing or damage tolerant materials similar to aerospace applications can be a
226 future field of research, including e.g. liquid metals [25]. Already today smart materials,
227 fibre composites and alloys which adapt to the operational scenario are possible. In some
228 cases detrimental effects such as erosion are actually used to facilitate material functions
229 (sec. 3.2). If W as a first wall material is required to suppress erosion even preferential
230 sputtering can turn the top layer of alloys or steel into a thin layer of erosion suppressing
231 tungsten [26, 27, 28].

232 *3.1. Composites for High Loads*

233 The basic idea is to introduce extrinsic mechanisms which allow energy dissipation.
234 This is the only way to enhance the toughness in brittle materials [29]. A basic strategy
235 to achieve pseudo-ductility is the incorporation of fibres and a weak interface into a
236 matrix, which needs extensive development and validation [30]. To overcome brittleness
237 issues when using W, a W fibre enhanced W composite material (W_f/W), incorporating
238 extrinsic toughening mechanisms can be used. The extrinsic mechanisms enable energy
239 dissipation and thus stress peaks can be released at crack tips and cracks can be stopped.
240 Another option are composite laminates made of commercially available raw materials
241 [31, 16]. The link between W_f/W and laminates is the similarity of fibres and foils. Both
242 show a special microstructure of highly deformed and elongated grains, hence showing

243 high strength and ductility even at room temperature [32, 33, 34]. Accordingly, even
244 in the brittle regime, below the DBTT, these materials allow for a certain tolerance
245 towards cracking and damage in general. In comparison, conventional tungsten would
246 fail immediately. From fig. 8(a) the principle of fibre-composite strengthening behavior
247 can be seen. Even when a crack has been initiated inside the material the energy
248 dissipation mechanisms allow further load to be put towards the component. After
249 reaching the ultimate strength other mechanisms lead to a controlled failure rather
250 than a catastrophic one in the brittle case. First W_f/W samples have been produced,
251 showing extrinsic toughening mechanisms similar to those of ceramic materials [35, 36].
252 These mechanisms will also help to mitigate effects of operational embrittlement due
253 to neutrons and high operational temperatures. A component based on W_f/W can
254 be developed with both chemical vapor deposition (CVD), utilizing a CVD setup, and
255 a powder metallurgical path through hot isostatic pressing [37, 35]. Crucial in both
256 cases is the interface between fibre and matrix. The interface is a thin layer (fig. 8(b))
257 with targeted properties: weak enough to enable the toughening mechanism, as strong
258 as possible to maximize the dissipated energy [38]. This is an idea based on enabling
259 pseudo-ductile fracture in inherently brittle material e.g. SiC ceramics [39].

260 Keeping in mind the above mentioned boundary conditions one can consider that
261 brittleness from either neutron irradiation or elevated temperatures can be mitigated as
262 the pseudo-ductilisation does not rely on any part of the material being ductile, crack
263 resilience can be established [35, 36]. Facilities to produce both CVD as well as powder
264 metallurgical W_f/W are readily available.

265 In order to enable the use of composites in fusion, it needs to be shown that for new
266 materials equally good behavior in terms of thermal conductivity, erosion and retention
267 can be established. As part of the development particularly the choice of the fibre and

268 interface material can be crucial. A sag-stabilized potassium doped fibre can even retain
269 some ductility in addition to strengthening the material [40, 34]. For the fibre-matrix
270 interface a non activating choice is required hence one should move from the so far
271 considered erbia [38, 35] potentially towards the low activating yttria.

272 In addition to conventional composites also fine grain tungsten is an option to
273 strengthen and ductilize tungsten [32] similar to other metals [33]. An option to achieve
274 this for W is powder injection molding (PIM) [41, 42]. PIM as production method
275 enables the mass fabrication of low cost, high performance components with complex
276 geometries. The range in dimensions of the produced parts reach from a micro-gearwheel
277 ($d=3$ mm, 0.050 g) up to a heavy plate ((60x60x20)mm, 1400 g). Furthermore, PIM
278 as special process allows the joining of tungsten and doped tungsten materials without
279 brazing and the development of composite and prototype materials, as described in
280 [41]. Therefore, it is an ideal tool for divertor R&D as well as material science. Figure
281 9(a) show new developed tungsten parts produced via PIM for a study of plasma-
282 wall interaction at ASDEX Upgrade at IPP Garching. Uniaxial grain orientation (see
283 fig. 9(b)), up- & down scaling, good thermal shock resistance, shape complexity and
284 high final density are several typical properties of PIM tungsten materials. Detrimental
285 mechanical properties, like ductility and strength, are tunable in a wide range (example:
286 W-1TiC and W-2Y₂O₃) [42]. Based on these properties the PIM process will enable the
287 further development and assessment of new custom-made tungsten materials as well as
288 allow further scientific investigations on prototype materials.

289 *3.2. Tungsten Smart Alloys*

290 Addressing the safety issue, a loss-of-coolant accident in a fusion reactor could lead to
291 a temperature rise of the first wall components of 1400 K after approximately 30 – 60
292 days due to neutron induced after heat of the in-vessel components [6] as schematically

293 depicted in fig. 10(a).

294 Thereby, a potential problem with the use of W in a fusion reactor is the formation
295 of radioactive and highly volatile tungsten oxide (WO_3) compounds. In order to
296 suppress the release of W oxides tungsten-based alloys containing vitrifying components
297 seem feasible, as they can be processed to thick protective coatings with reasonable
298 thermal conductivity, e.g. by plasma spraying with subsequent densification as already
299 demonstrated for titanium and tantalum coatings [43]. Enhanced erosion of light
300 elements during normal reactor operation is not expected to of concern. Preferential
301 sputtering of alloying elements leads to rapid depletion of the first atomic layers and
302 leaves a pure W surface facing the plasma as per the given different sputtering yields.
303 [44, 45]. This mechanism is similar to the above mentioned EUROFER-97 surface
304 enrichment. Fig. 10(b) displays the basic mechanism. During operation plasma ions
305 erode the light constituents of the alloy, leaving behind a thin depleted zone with only
306 tungsten remaining. Subsequently, the tungsten layer suppresses further erosion, hence
307 utilizing the beneficial properties of tungsten. In case of a loss-of-coolant and air or
308 water ingress the tungsten layer oxides releasing a minimum amount of WO_3 and
309 then passivating the alloy due to the chromium content. W-Cr-Y with a tungsten
310 fraction of up to 70 at% shows a 10^4 -fold suppression of tungsten oxidation due to
311 self-passivation [46]. Test systems are being produced via magnetron sputtering and
312 evaluated with respect to their oxidation behavior. Production of bulk samples is
313 ongoing. Rigorous testing of oxidation behavior, high heat flux testing and plasma
314 loads as well as mass production for candidate materials are under preparation. The
315 material can be considered for both first wall and divertor applications especially when
316 combined with the strengthening properties of the W_f/W composite approach. The
317 PWI behavior and potential neutron or temperature embrittlement still needs to be

318 assessed.

319 *3.3. Functionally Graded Materials*

320 Having discussed tungsten as the main candidate for the PFMs of a fusion reactor
321 the joint to the underlying cooling structure or wall structure in general is crucial.
322 From the differing thermal expansion coefficients for the different materials (copper
323 $\sim 16.5\mu\text{m}/(\text{mK})$, tungsten: $\sim 4.5\mu\text{m}/(\text{mK})$, stainless steel: $\sim 12\mu\text{m}/(\text{mK})$) it is clear
324 that a mature solution of joining them needs to be established.

325 As example systems the development of functionally graded materials (FGMs)
326 between W as the PFM with the structural material EUROFER-97 can be considered,
327 exhibiting complementary volumetric gradients of W and EUROFER-97. As depicted in
328 [47] FGMs are promising candidates for interlayers between components of two different
329 materials especially when considering applications such as the blanket modules of a
330 DEMO [7, 48] or even a helium cooled tungsten divertor with low to medium heat-
331 flux ($1\text{-}5\text{MW}/\text{m}^2$) for which the heat conductivity of EUROFER-97 may be sufficient.
332 Fig. 11 shows a potential development cycle, which comprises further optimization of
333 FGM manufacturing techniques, joining, mechanical testing and thermal cycling. This
334 will determine the viability of the FGM concept and also allow the comparison with
335 conventional joints. Similar ideas are developed for the transition between copper and
336 W [49, 50] potentially being used as solution for a water-cooled high heat-flux divertor
337 [7, 8]

338 *3.4. Permeation Barriers*

339 Moving from the plasma facing material towards the structural part of the reactor
340 tritium management is an issue in particular for the breeding blankets. In order to
341 prevent fuel loss and radiological hazards it is important to suppress permeation of

342 tritium towards the cooling channels. Research on tritium permeation barriers ranges
343 over a variety of materials [51, 52, 53, 54, 55, 56, 57], including alumina and erbia.
344 Permeation barriers require high permeation reduction factors, high thermal stability
345 and corrosion resistance as well as similar thermal expansion coefficients compared
346 to those of the substrate. Investigation of the permeation reduction factor requires
347 controlled experiments. A new gas-driven permeation setup is established at FZ
348 Jülich to investigate deuterium permeation e.g. through different ceramic coatings
349 on EUROFER-97, which significantly reduce the deuterium permeation. Several
350 deposition techniques can be considered, e.g. filtered arc deposition, chemical routes,
351 and magnetron sputter deposition. A reduction factor of 50-100 is essential to allow
352 a safe operation and a reasonable tritium breeding ratio. In addition to permeation
353 reduction and mechanical feasibility, compatibility with neutron irradiation needs to
354 be considered. Here especially the promising barrier candidates alumina and erbia
355 do have issues. Yttria has a better activation behavior as those candidates, see fig.
356 6. First permeation measurements of yttria coatings on EUROFER-97 show a similar
357 permeation reduction factor as erbia [58]. Studies on yttria are ongoing.

358 **4. Summary and Outlook**

359 Considering all the above mentioned issues when using materials in a fusion reactor
360 environment a highly integrated approach is required. The lifetime of PFCs and joints
361 due to erosion, creep, thermal cycling, and embrittlement needs to be compatible
362 with steady state operation and short maintenance intervals. Thermal properties of
363 composites and components have to be at least similar to bulk materials when enhanced
364 properties in terms of strength are not to hinder the maximization of operational
365 performance. Damage resilient materials can here facilitate small, thin components

366 and hence higher exhaust capabilities. The components need to be compatible with the
367 aim of tritium breeding and self-sufficiency and hence mitigate tritium retention and
368 loss.

369 Despite using various alloying components, interlayers or coatings maintainability
370 and recycling of used materials is required to make fusion viable and publicly acceptable.
371 Last but not least, large scale production of advanced materials is crucial. We hence
372 propose to utilize the composite approach together with alloying concepts to maximize
373 the potential of the tungsten part of a potential PFC. Together with W/Cu composites
374 at the coolant level and W/EUROFER-97 joints high-performance components can be
375 developed. Rigorous testing with respect to PWI and high heat-flux performance are
376 planned for all concepts to have prototype components available within 5 years for
377 application in existing fusion devices.

378 **Acknowledgements**

379 This work has been carried out within the framework of the EUROfusion Consortium
380 and has received funding from the Euratom research and training programme 2014-2018
381 under grant agreement No 633053. The views and opinions expressed herein do not nec-
382 essarily reflect those of the European Commission

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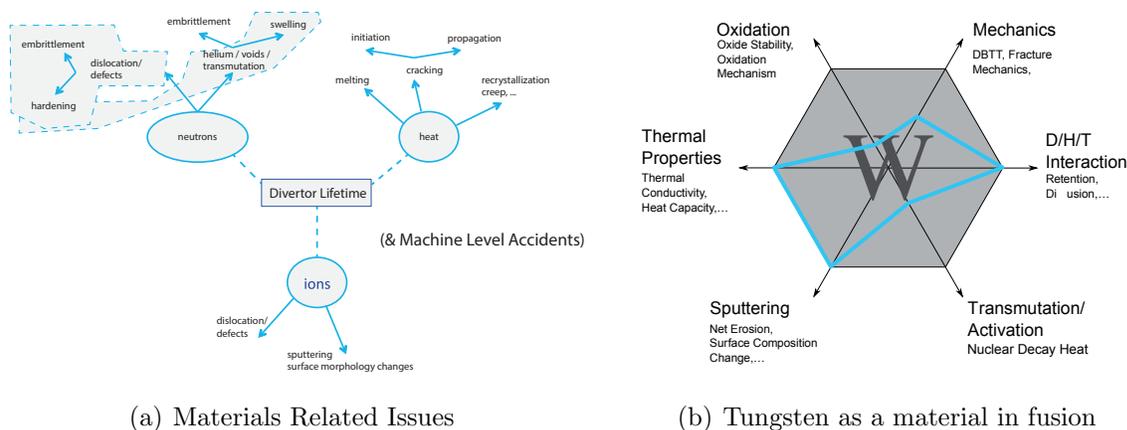
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463 Figures



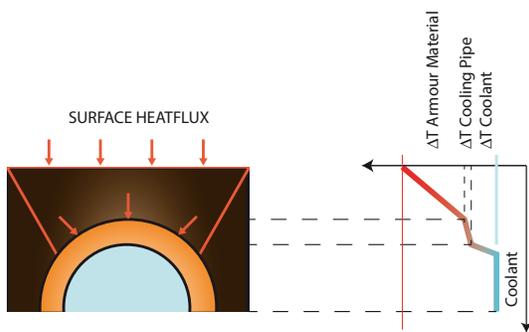
Figure 1: Materials in Fusion face not a single but a multitude of interlinked challenges.



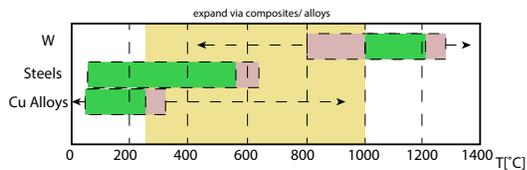
(a) Materials Related Issues

(b) Tungsten as a material in fusion

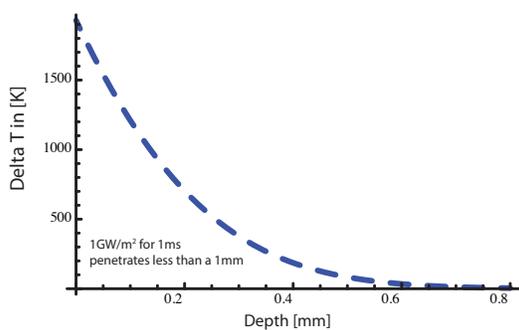
Figure 2: Material Issues for PFCs.



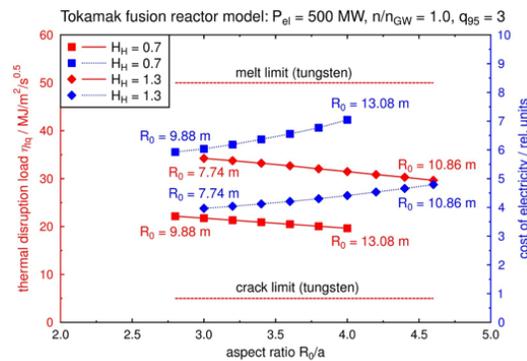
(a) Steady state heat flux in a conventional monoblock like structure



(b) Operational Windows for Structural Materials in Fusion (based on [59])

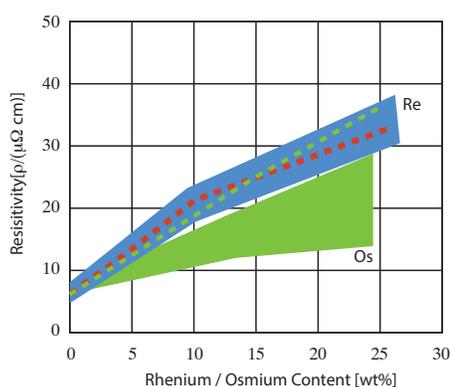


(c) [Heat flux penetration (in tungsten) from $1GW/m^2$, after 1ms

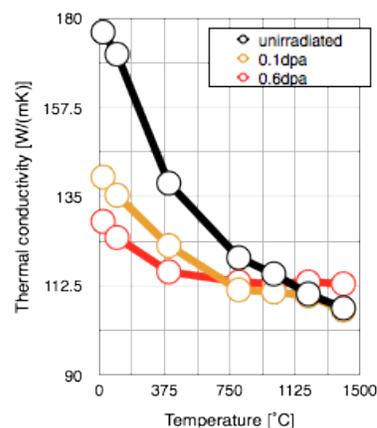


(d) Disruption heat loads - material limits [21, 11, 20]

Figure 3: Power-exhaust - Issues arising from steady state and transients

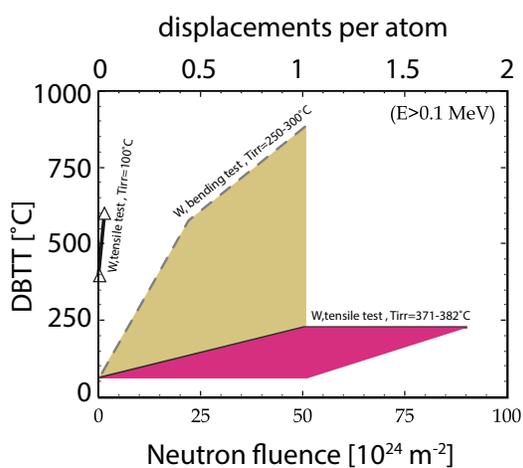


(a) Electrical resistivity of W containing various amounts of Re or Os. The blue band shows WxRe, the green band depicts WxOs for various cases irradiated up to 1.5 dpa. The red line and green line stand for W-xRe and W-xOs unirradiated respectively [12], $T_{\text{irr}} = 300 - 750^{\circ}\text{C}$

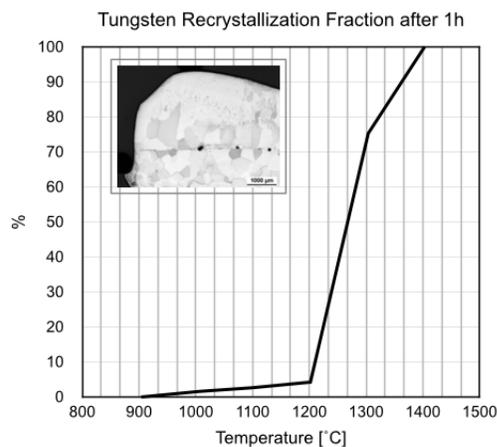


(b) Thermal conductivity of W before and after neutron irradiation (0.1 and 0.6 dpa, $T_{\text{irr}} = 200^{\circ}\text{C}$) [22]

Figure 4: Change of electrical and thermal properties of tungsten under neutron irradiation and transmutation

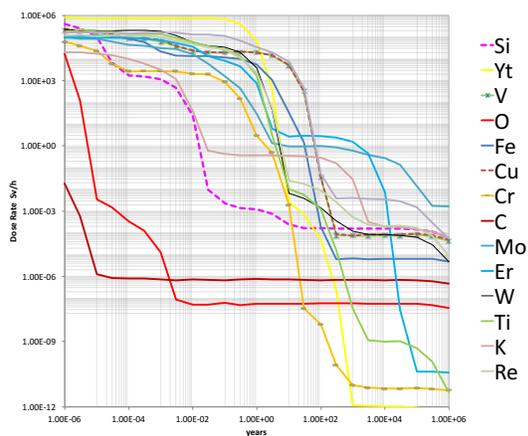


(a) DBTT dependence after neutron irradiation based on [23]

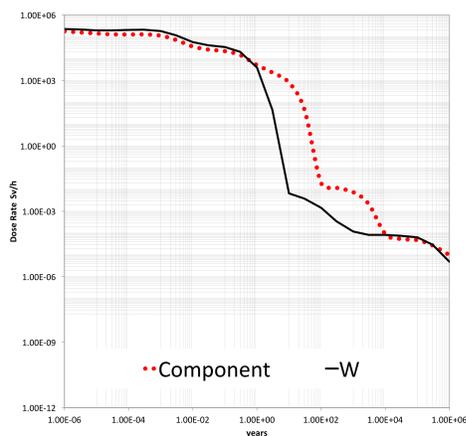


(b) Recrystallization of tungsten after 1h at given T, based on [?]

Figure 5: Factors determining operational embrittlement are neutron irradiation and recrystallization



(a) Residual activation of various elements



(b) W 79.7wt%, Er 0.6wt%, Cr 12.1wt%, Cu 7.5%

Figure 6: The activation of materials for the first wall can be estimated as an upper bound (based on [9]). Divertor components in general are less prone to activation. Shielded hands-on level: 2 mSv/h , Hands-On Level: $10\text{ }\mu\text{Sv/h}$

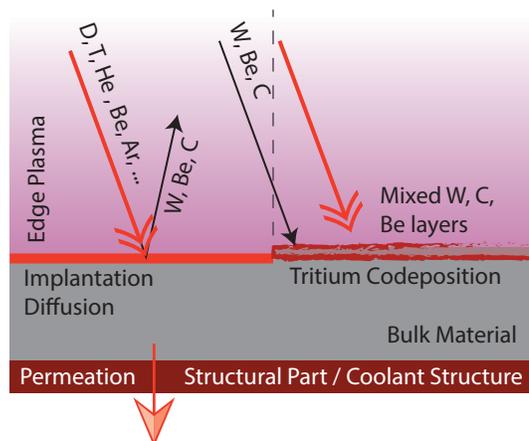


Figure 7: Fuel retention and permeation issues under plasma exposure conditions

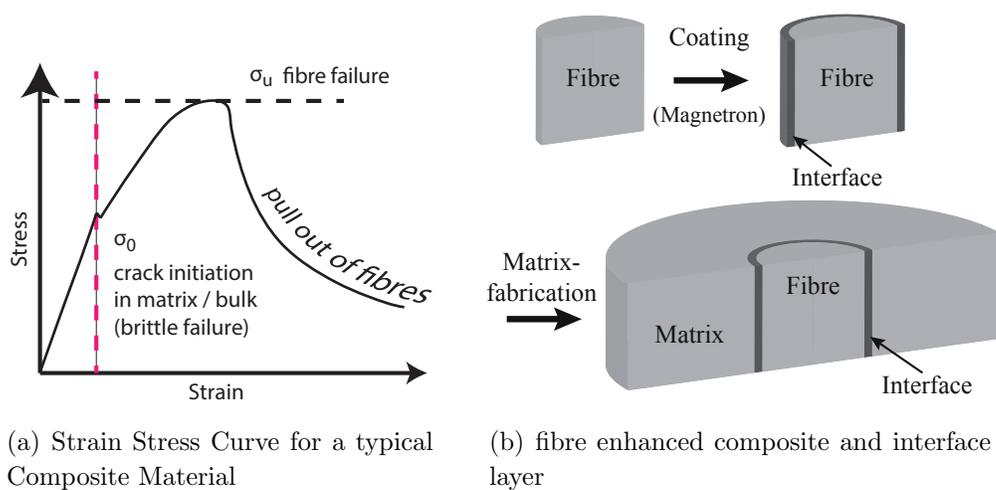
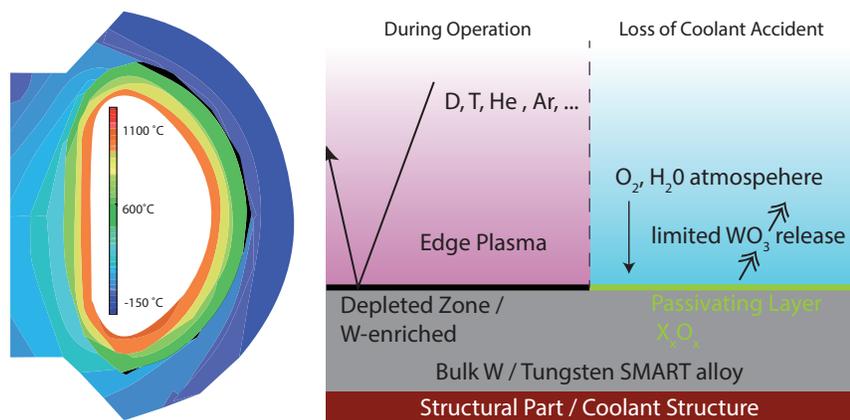


Figure 8: Composite approaches based on pseudo ductilisation.



Figure 9: W-PIM parts produced for a study of plasma-wall interaction in ASDEX Upgrade (a), EBSD image of the uniaxial grain orientation (b).



(a) Loss of coolant accident scenarios - Wall temperature estimate based on [6], 10 days after incident

(b) Working principle of a smart alloys based PFC with both the operational and accident mechanisms shown.

Figure 10

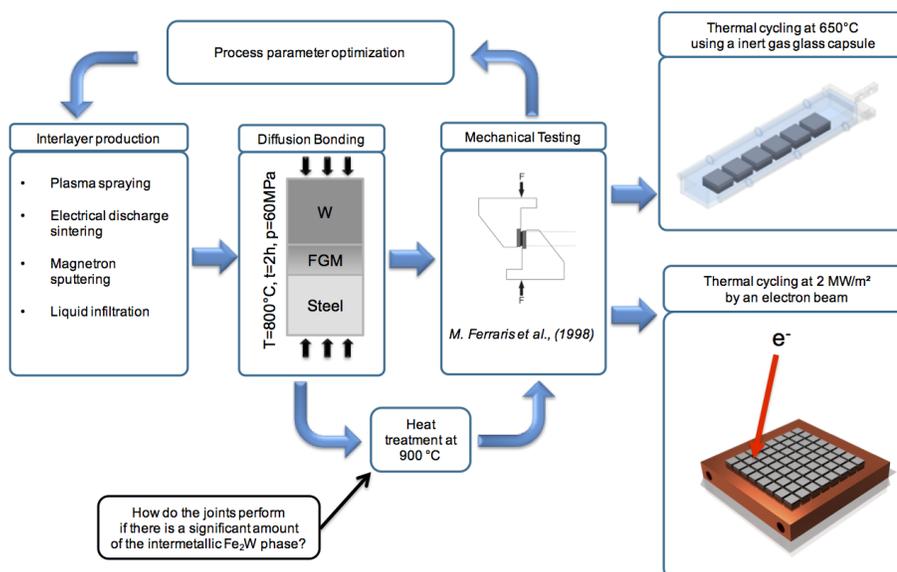


Figure 11: Functionally Graded Materials and a potential optimization and testing procedure.