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Plasma–surface interaction issues of an all-metal ITER

J.N. Brooks[1], J.P. Allain[1], R.P. Doerner[2], A. Hassanein[1], R. Nygren[3], T.D. Rognlien[4] and D.G. Whyte[5]

[1] Purdue University, 400 Central Drive, W. Lafayette, IN 47907 USA
[2] Unviversity of California San Diego, 9500 Gilman Drive, CA 92093, USA
[3] Sandia National Laboratories, PO Box 5800, Albuquerque, NM 87185, USA
[4] Lawrence Livermore National Laboratory, 700 E. Avenue, PO Box 808, Livermore, CA 94551, USA
[5] Massachusetts Institute of Technology, 175 Albany Street, Cambridge, MA 02139, USA

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Abstract

We assess key plasma–surface interaction issues of an all-metal plasma facing component (PFC) system for ITER, in particular a tungsten divertor, and a beryllium or tungsten first wall. Such a system eliminates problems with carbon divertor erosion and T/C codeposition, and for an all-tungsten system would better extrapolate to post-ITER devices. The issues studied are sputtering, transport and formation of mixed surface layers, tritium codeposition, plasma contamination, edge-localized mode (ELM) response and He-on-W irradiation effects. Code package OMEGA computes PFC sputtering erosion/redeposition in an ITER full power D–T plasma with convective edge transport. The HEIGHTS package analyses plasma transient response. PISCES and other data are used with code results to assess PFC performance. Predicted outer-wall sputter erosion rates are acceptable for Be ($0.3\,\mathrm{nm\,s^{-1}}$) or bare (stainless steel/Fe) wall ($0.05\,\mathrm{nm\,s^{-1}}$) for the low duty factor ITER, and are very low ($0.002\,\mathrm{nm\,s^{-1}}$) for W. T/Be codeposition in redeposited wall material could be significant ($\sim$2 gT/400 s-ITER pulse). Core plasma contamination from wall sputtering appears acceptable for Be ($\sim$2%) and negligible for W (or Fe). A W divertor has negligible sputter erosion, plasma contamination and T/W codeposition. Be can grow at/near the strike point region of a W divertor, but for the predicted maximum surface temperature of $\sim$800 °C, deleterious Be/W alloy formation as well as major He/W surface degradation will probably be avoided. ELMs are a serious challenge to the divertor, but this is true for all materials. We identify acceptable ELM parameters for W. We conclude that an all-metal PFC system is likely a much better choice for ITER D–T operation than a system using C. We discuss critical R&D needs, testing requirements, and suggest employing a 350–400 °C baking capability for T/Be reduction and using a deposited tungsten first wall test section.

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(Some figures in this article are in colour only in the electronic version)

1. Introduction

The choice of plasma facing component (PFC) surface materials and compatible plasma parameters remains a critical and contentious issue for ITER and beyond. PFC plasma/surface interactions will affect component lifetime, tritium inventory, plasma contamination and plasma operation. The US PFC team has been analysing non-carbon, all-metal ITER PFC performance, in particular for Be and W surfaces, via code analysis coupled to existing data. We report here on studies of plasma edge/scrape-off-layer (SOL) parameters, single and mixed-material sputtering erosion/redeposition, plasma transient response, tritium codeposition, Be/W alloy formation and He/W effects. We assess testing issues. (We also note that the US PFC team is studying other issues, e.g. dust formation, but these are not discussed here.) Our focus is on the high flux outer-wall/divertor region.

The all-metal system is promising, but major R&D is needed. In general, the PFC response will restrict ITER core/edge plasma operations, via edge-localized mode (ELM) and other plasma transient limitations.

2. First wall sputter erosion/redeposition

Figure 1 shows the ITER PFC system design, with the reference Be first wall surface, tungsten 'baffle' region and with a tungsten divertor target. The plasma edge/SOL parameters and PFC response were studied via the US OMEGA

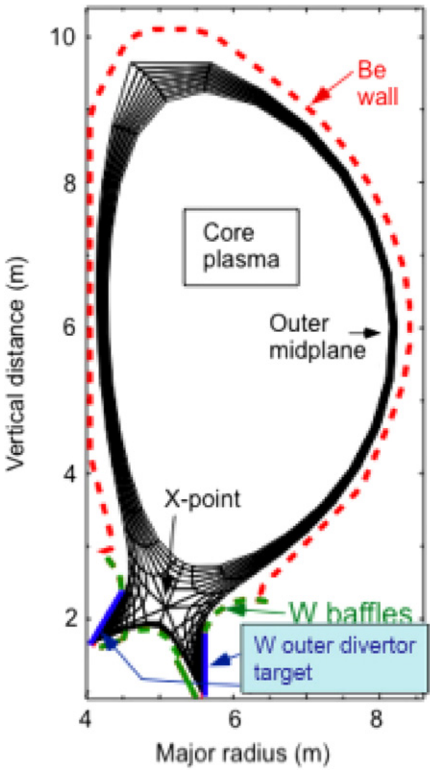

Figure 1. ITER PFC system. Modelling is done for the reference beryllium outer first wall surface, a tungsten wall, a tungsten outer divertor surface (baffle region and target) and a bare wall (stainless steel = iron).

code collaboration, using models for the expected convective ('blob') transport plus diffusive plasma edge transport regime. OMEGA consists of real-time and off-line coupled codes, comprising the UEDGE plasma fluid code, DEGAS charge-exchange (CX) neutrals code, REDEP/WBC code package, TRIM-SP sputter yield code, BPHI-3D sheath code and other codes.

Convective transport results in a $\sim x50$ increase in both D–T and He ions and CX D–T neutrals to the first wall compared with diffusion-only transport [1, 2]. This raises obvious concerns about high wall sputter erosion and subsequent transport. Table 1 shows sputter erosion rates for a Be or W outer first wall and an alternative bare stainless steel wall. Erosion is computed for incident D, T atoms from CX, D^+, T^+, He^{+2} (5%) and trace (0.1%) O^{+3} ions, using the respective particle sputter yields and energy/angular distributions and assuming sheath acceleration of ions at the wall.

A bare wall is modelled here—for the present purposes of assessing the effect of a medium-Z uncoated wall—as pure iron. As for the other materials, iron sputter yields are supplied by TRIM-SP code computations. Electron impact ionization rate coefficients for iron are taken from [3].

In spite of high particle flux, table 1 results show acceptable Be or Fe erosion rates for the low duty factor ITER. The W sputter erosion rate (due ~90% to D–T CX neutrals, and ~10% to He and trace impurity ions) is very low and would extrapolate to a high duty factor (DEMO, etc) device.

Table 2 summarizes current transport/redeposition results for sputtered/ionized wall material. The analytical method is

Table 1. ITER outer first wall sputtering rates; reference case convective edge plasma regime.

Wall surface	Sputtered current[a] (atoms s^{-1})	Erosion rate[b] (m s^{-1})	Erosion lifetime, 3 mm surface at 3% duty factor (yr)
Beryllium	1.9×10^{22}	3.2×10^{-10}	~10
Iron (stainless steel)	1.0×10^{21}	5.0×10^{-11}	~60
Tungsten	5.6×10^{19}	1.8×10^{-12}	~1700

[a] For outer first wall, scaled from lower-half outer-wall results.
[b] Rate approximately spatially uniform; not including local peaking, if any, due to CX from gas puffing.

Table 2. Transport summary of sputtered outer first wall material; WBC code, 10^6 histories/run. Plasma with convection, reference impurity convection model.

Parameter[a]	Beryllium	Iron	Tungsten
Ionization mean free path[b] (cm)	11.5	6.7	3.5
Fraction to wall	0.28	0.56	0.75
Fraction to baffle	0.62	0.43	0.25
Fraction to (outer) divertor target	0.094	0.008	1.4×10^{-4}
Fraction to edge plasma boundary	0.006	4.0×10^{-6}	0.000000
Energy to wall (eV)	61	104	149
Energy to baffle (eV)	118	277	512
Energy to divertor (eV)	273	941	2313

[a] Unless otherwise indicated, average for redeposited ions.
[b] For sputtered atoms, normal to surface.

described in detail in [2]. Briefly, a full kinetic 3D WBC computation is used, with input D–T full power UEDGE plasma background, with sputtered atoms launched with TRIM-SP derived velocity distributions; the resulting ions are then subject to charge-changing and velocity-changing collisions with the plasma, including diffusion and convective force effects.

Most sputtered material is redeposited on the wall and baffle, from ~0–10% going to the divertor and ~0–1% reaching the edge/core plasma boundary, depending on the material. The core plasma contamination potential from wall sputtering—for the reference impurity ion transport model of impurity transport same as D–T ion transport—is acceptable for Be, at ~2% Be/D–T (see [2] for estimate method) and essentially zero for W or Fe. OMEGA analysis of sputtering of the W baffle, both from plasma atoms/ions and from self-sputtering via incident first wall material, likewise shows low erosion rates/contamination.

The reason that wall-sputtered W does not reach the core plasma is fairly simple—sputtered W atoms are ionized close to the wall (~4 cm) and hence far (~18 cm) from the last closed flux surface (see figure 1). The W ions then diffuse/flow back to the wall and/or flow along poloidal field lines to the other PFC components faster than they can diffuse into the core plasma. This effect is enhanced by convective transport but also occurs without it.

This prediction for ITER is in apparent contrast to some present high-Z boundary results, e.g. ASDEX-U, where core W is observed [4], however, the geometry/plasma is different

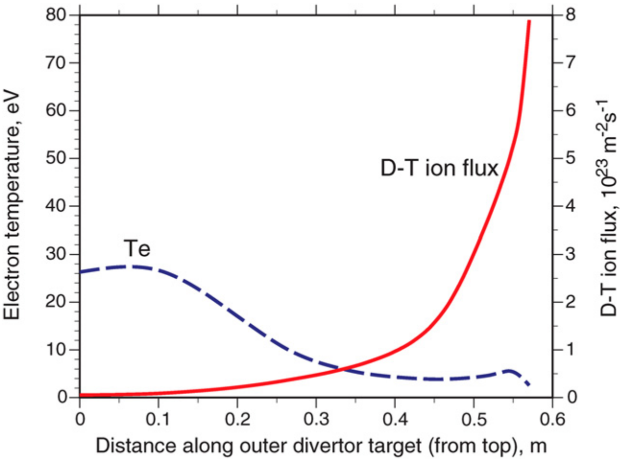

Figure 2. Electron temperature and D–T ion flux along the outer divertor target. OMEGA computation; ITER edge plasma with convection. From [6]. Strike point at ~0.55 m.

from ITER. For instance, ITER will have a longer and more strongly ionizing wall-to-core plasma. Also, ASDEX-U uses a W limiter, where sputtered material easily enters open flux surfaces, and a W antenna screen. Higher than expected divertor Mo erosion/transport was seen in some C-MOD shots [5]; analysis of C-MOD Mo erosion via REDEP, etc codes is underway. There is evidence that enhanced sputtering due to RF heating induced sheaths is important in both ASDEX-U [4] and CMOD [5]. It is not clear whether this is an issue for ITER. Preliminary OMEGA analysis shows that the key concern of a hypothetical ICRF rectified sheath on a tungsten ITER wall, baffle or divertor target would be high redeposited W ion sheath acceleration with possible resulting self-sputtering runaway, and if so, this would impose an antenna/geometry design requirement to avoid this in ITER.

3. Divertor sputter erosion/redeposition

Figure 2 shows the electron temperature and D–T ion flux at the divertor. These are from the UEDGE edge plasma solution for the above-mentioned reference convective plus diffusive edge plasma case. Erosion/redeposition analysis of a pure-W divertor, using this plasma solution (with associated density, etc parameters), shows near zero net W sputter erosion, due to low gross sputtering to begin with (with no D–T ion sputtering, but only D–T CX neutrals, He^{+2} and trace plasma impurity ions), non-runaway self-sputtering (due to the <30 eV peak plasma temperature at the divertor) and very high (~100%) local redeposition fractions. The high redeposition is due to short (~0.1–1 mm) mean free paths for ionization of sputtered tungsten atoms and subsequent strong W ion flow back to the divertor due to collisions with the incoming plasma, including via electric field forces. (The low tungsten erosion contrasts with a carbon divertor target, which has a peak net sputter erosion rate of order 10 nm s^{-1}). As with a W first wall, the divertor analysis also shows essentially zero core plasma contamination from outer divertor sputtering, due to the high redeposition.

The effect of wall-sputtered Be transported to an initially W outer divertor target—not including any surface temperature effects—was assessed in [7] using OMEGA codes including the W-MIX mixed-material code (and further studied here using differences for the Be source function due to plasma thermal force model variations). In spite of significant Be flux to the divertor target, there is no predicted net Be growth over the upper ~2/3 of the outer target, due to high re-sputtering by plasma ions and self-sputtering/reflection of Be, but substantial growth, of order 1 nm s^{-1}, occurs at/near the strike point, due to Be transport from the upper target region coupled with low sputtering due to low electron temperature.

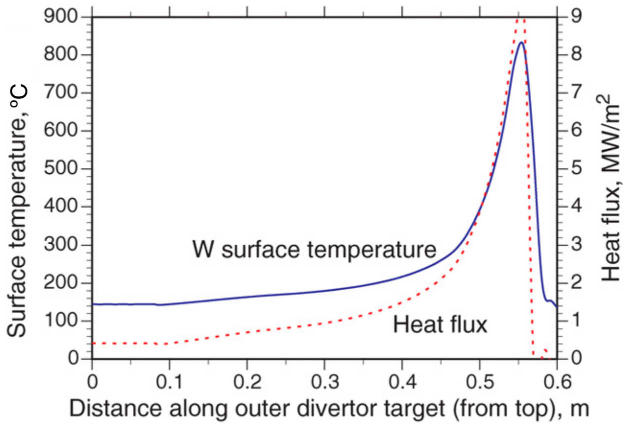

Figure 3. Surface temperature and heat flux for a tungsten monoblock outer divertor.

4. Be/W alloy formation at the divertor

Considering the above results, a potential drawback for a tungsten divertor is the interaction of wall-sputtered Be with a high temperature divertor surface to form low melting temperature tungsten beryllide alloys. PISCES-B experiments show W–Be alloy formation for a W sample surface temperature exceeding 750 °C [8]. The measured diffusion rate for Be in W is quite high, at 750 °C being 4.3×10^{-15} cm^2 s^{-1}, increasing to 5.8×10^{-13} cm^2 s^{-1} at 850 °C. At a similar temperature, the diffusion of Be in W (e.g. 27 nm length scale in 400 s at 750 °C) is about 2 orders of magnitude larger than C in W. However, both sputtering by the incident plasma and sublimation of Be from the hot surface can act to reduce the alloy growth rate by limiting the availability of surface Be.

Figure 3 shows the heat flux and resulting steady-state surface temperature profile we compute for a W ITER outer lower divertor (fully covered with the type of W monoblocks now used for the baffle design). This is for the above-mentioned ITER base-case D–T plasma with 100 MW input from the core to the edge region. (The peak heat flux is primarily due to convection, but also includes a radiation load.) Since the peak temperature is only ~800 °C, we conclude that evaporation is not a limiting factor in Be buildup. (However, transient heating effects on Be evaporation need assessment). Also, from the point of view of tungsten beryllide formation, the surface temperature is high enough (⩾750 °C) to promote significant alloy growth only in the region extending roughly 2 cm on either side of the strike point. Even though Be may accumulate on other regions of the W divertor, the surface temperature will apparently be too low to promote the formation and growth of the alloys. While this is encouraging, the extrapolation of relevant laboratory results to ITER is highly dependent on assumptions of exposure conditions, and obviously on plasma conditions, and is difficult to reliably predict at this time. Analysis is ongoing to assess the formation and growth rates of the Be_xW alloys under full power ITER operation. Of course, Be/W issues can be completely avoided with an all-tungsten system.

5. Tritium codeposition

Tritium is retained in PFCs by codeposition and bulk trapping; we focus here on the non-saturable codeposition process. For a tungsten divertor, codeposition is negligible due to minimal sputter erosion/redeposition and low T/W trap ratios and likewise for a W wall. This contrasts with estimates of $\sim$3 gT/400 s shot codeposition for a carbon divertor [9, 10].

Be is known to trap T in redeposits at room temperature at rates not that different from carbon but with trapping falling off more steeply with temperature. A scoping estimate of ITER T/Be codeposition was made in [2]. Two still-rough methods are used here to update the T/Be codeposition estimate, both based on laboratory data, and using a nominal 200 °C ITER PFC surface temperature, except at/near (figure 3) the divertor strike point. The first uses a constant value of (D+T)/Be in codeposits of 0.08 at 200 °C [11]. The second uses a scaling law developed [12] for (D+T)/Be in codeposits under varying codeposition conditions. In the DeTemmerman scaling it is noted that energies far exceeding the experimental parameters are not well reproduced by the model. Although the OMEGA-computed CX energy spectrum in ITER is broad ($\sim$1–1000 eV), we approximate it for the trapping estimate with a monoenergetic energy of 100 eV (close to the maximum energy of validity of the model). Also, the value of the Be deposition rate used in these calculations is allowed to exceed the band of experimental values included in the model ($\times$2 at the first wall to $\times$40 at the baffle).

The codeposited tritium calculated using either approach is quite similar, 1.5–1.8 gT/400 s-shot due to outer-wall erosion. However, the codeposit locations are slightly different: predominantly on either the baffle (Causey values) or on the first wall (DeTemmerman scaling), with codeposition being low (<20 mg) on the divertor, or below-divertor (dome region) ($\sim$100 mg) in either case.

It is the tritium release behaviour from codeposits that actually controls the retained tritium inventory. We understand that it is presently envisioned to increase the baking temperature of the ITER divertor and baffle from 240 to 350 °C and to 300 °C for the first wall. During a 240 °C bake out of an ITER PFC component, one could expect to remove only $\sim$20% of the tritium residing in Be codeposits. Increasing bake out to 300 °C would remove $\sim$50% of the tritium, a 350 °C bake out would release $\sim$80% and a 400 °C bakeout would release 85–90% of the tritium. The tritium release behaviour is described in more detail in [13].

Better estimates of ITER T/Be codeposition will require more spatially refined plasma parameter, surface temperature, trapping rate coupling as well as inner wall Be sputtering/transport analysis.

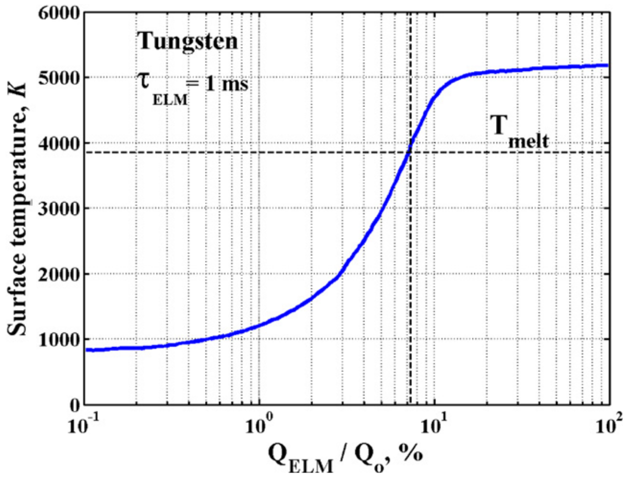

Figure 4. ITER outer tungsten divertor ELM response as a function of ELM energy fraction; Q_0 = 127 MJ released at midplane, ELM duration 1 ms [13].

6. Divertor ELM response

ELMs are a serious concern during normal H-mode operation of ITER and future tokamaks. During ELMs, an energy, Q_{ELM}, of $\sim$1–10% of total core plasma energy Q_0 is released to the SOL and deposited on the divertor surface in a duration τ_{ELM} of $\sim$0.1–1 ms with a frequency of $\sim$1–10 Hz. The incoming power from the SOL to the divertor plate in ITER-like devices during an ELM can increase from $\sim$10 to $\sim$300–3000 MW m^{-2}. The HEIGHTS simulation package solves detailed plasma transient particle energy deposition, evolution of surface materials, debris formation, vapour MHD, atomic physics, radiation transport and erosion physics [14–16]. Recent enhancements use multidimensional two-fluid hydrodynamic mixing models where the incident D–T plasma is treated separately from the eroded debris cloud of divertor materials.

Figure 4 shows HEIGHTS results for W divertor surface temperature as a function of ELM intensity. Tungsten will start to melt for giant ELMs of energy Q_{ELM} > 7–8% Q_0, released at the midplane. The surface temperature will exceed the melting temperature and a melt layer thickness of 100 μm is developed for giant ELMs (Q_{ELM} $\sim$ 10% Q_0) deposited in 1 ms duration. In this situation, melt layer erosion is a major concern with possible large mass losses due to MHD and splashing effects, as well as the issue of subsequent high heat flux exposure of the damaged surfaces.

A major HEIGHTS result, however, is that carbon has similar ELM and other transient (disruptions, VDE's) concerns as tungsten [14–16]. Carbon may also suffer macroscopic erosion from brittle destruction, particularly at higher power deposition, and both W and C will have significant vaporization losses at shorter giant ELM durations. Also, radiation from the resulting vapour cloud for either material can damage nearby components. We have identified acceptable (no melting) and unacceptable (w/melting) ELM parameter windows for ITER, shown in figure 5 for the W case. For the unacceptable giant ELM of 10% or more deposited core plasma energy in 0.1 ms on the divertor (energy density >3 MJ m^{-2}), the erosion is high for carbon (0.2 μm/ELM). For tungsten there is less

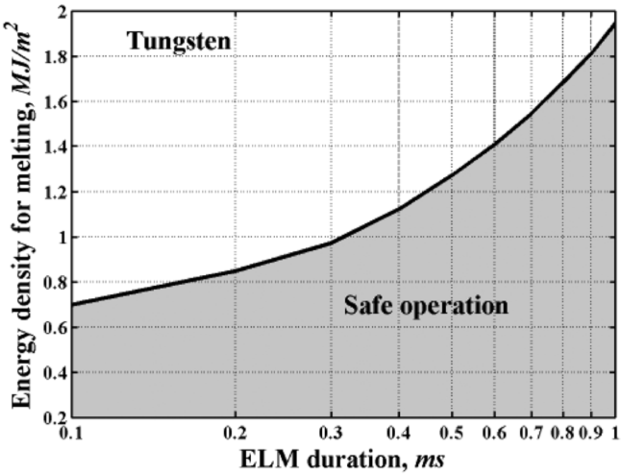

Figure 5. HEIGHTS parameter window for tungsten divertor acceptable (no-melt) ELM response.

vaporization erosion than carbon but, as stated, significant melting occurs. For longer deposition times ($\sim$1 ms) and/or lower ELM energy the surface response for all materials is much better. A further mitigating approach is to inject noble gas to absorb the incoming power and decrease the net power load to the PFC surface below the melting temperature. For example, HEIGHTS analysis shows that for a neon line density of 10^{17} cm^{-2} the resulting W surface temperature is less than 1500 K, which is even lower than the melting temperature of potentially forming Be–W alloys [14].

7. Tungsten helium irradiation effects

Understanding the role of high-intensity and large dose exposure of He and D–T particles on a tungsten PFC surface is an important challenge for ITER and beyond. Experiments show that tungsten may suffer from He-induced embrittlement and mechanical degradation. Of concern are bubble and 'fuzz' formation under He ion irradiation, blistering, neutron-created T trapping sites, dust generation and overall surface integrity. A key point is that helium damage mechanisms are extremely sensitive to the material temperature, since this largely sets He diffusion and trapping, and of course to He fluence.

W fuzz is a commonly seen micrometres-scale tendril formation of plasma exposed surfaces due to the action of >10 eV He at elevated surface temperature, e.g. [17–19]. Data suggest that the tendrils are set by He mobility/diffusion in the W, and therefore grow with a square-root time dependence. It is suspected that when the tendrils grow to Debye sheath length, i.e. greater than about a micrometre, they will delaminate from the surface due to poor thermal contact. Experiments indicate a lower limit of about 800 °C for this damage [18]. This would imply that an ITER first wall, baffle and most of the divertor—per figure 3 temperature results—would avoid fuzz problems.

At the divertor strikepoint, noting tendrils grow to $\sim 2\,\mu$m depth at 10^{26} He m^{-2} [19], and for a He flux of 5% of the D–T flux (and assuming a pure-W system) implies an effective fuzz erosion rate of $\sim$0.2 nm s^{-1}, i.e. an acceptable $\sim$1 mm net erosion in 25 000 ITER shots. (Be net deposition, if any, and/or ELM effects could reduce this erosion even further). Post-ITER, tendril growth will be of greater concern since the average temperature may be elevated and He fluence will be larger due to steady-state operation.

At lower temperatures, i.e. in the divertor region outside the strikepoint, and for the first wall, the concern is blistering and development of nano-structures. It is presently difficult to project their effects on tungsten erosion. First, it is unclear if the surface damage is detrimental, through loss of associated micro-particulates, or beneficial due to the opening of the surface to release trapped H/He gases. Secondly, the exact flux and fluence dependences are unknown and W surface damage is not universally seen in tokamaks. The most pessimistic projection comes from laboratory observation at RT, reviewed in [18], of $\sim$20 nm structures formed at $\sim 5 \times 10^{21}$ He m^{-2}, which projects a 4–40 nm s^{-1} erosion rate if the said structures are prone to immediate removal. Such a rate could both limit W lifetime through erosion and dust production, and more importantly, the removed particulates may escape local redeposition. However, present experience with high-Z tokamak divertors [5, 20], which typically have some low fraction of He from wall conditioning, is a peak net erosion rate $\sim$0.1 nm s^{-1} campaign-integrated. Recent W exposure in the LHD divertor with pure He plasmas [21] showed a net erosion $\sim$6 nm s^{-1} from blistering, which if scaled with ITER He fluence, also projects $\sim$0.1 nm s^{-1} net erosion. Therefore, the preponderance of experience points towards acceptable net erosion rates, although the full impact of such 'non-atomistic' W erosion for ITER and beyond is still to be determined.

8. Testing

Verifying the robustness of ITER's PFCs armoured with W or Be tiles is crucial to reduce risk and involves three tiers of test activities. The first is verifying that erosion of the plasma facing material during long pulse operation with ELMs, and with disruptions, is acceptable. The test conditions should include surface modifications from ion and neutron damage and from mixing of materials that may exacerbate erosion and the retention of tritium and formation of metallic dust. The second is verifying the performance of metal-armoured PFCs in steady state, cyclic and transient high heat flux tests. The third tier of activity is the deployment of test targets in a tokamak environment.

Among the provocative issues that face ITER and fusion in general is the balance between (a) our testing of PFCs using relatively few specimens under limited conditions that represent the anticipated operational environment, (b) provisions for the consequences of failures such as missing or damaged tiles and (c) the engineering diagnostics that can determine the status of operating PFCs. A recent paper [22] provides an overview of the development, characterization and testing of plasma facing materials and components for future fusion devices.

Ideally, testing and modelling should go forward together such that test conditions and results are measured with the accuracy and completeness that are useful for benchmarking models. Also, individual experiments should contribute to a useful set of complementary data world-wide. This consideration is particularly important in the choice of materials. Tungsten is available in various wrought and powder metallurgy forms and there are tungsten alloys as well as ultra-fine grain tungsten being considered as alternatives to pure (brittle) tungsten. Additives that act to form grain boundary precipitates, improve mechanical properties and retard recrystallization are considered beneficial. But minor constituents can also affect the formation of damage sites and trapping of helium and hydrogen isotopes and alter the plasma–surface interactions noted previously. There are exciting possibilities to improve materials, such as the 'nano-dispersoids', e.g. [23] but we will face pressure to pare down the possibilities and focus on a relatively few choices to follow through more intensive stages of development. One obvious 'wild card' is the influence of neutron damage on the behaviour of materials, and we will expect to utilize a fusion materials irradiation facility. A sensible case is also being made for one or more test facilities for in-vessel components. Reference [24] discusses many of the challenging issues noted here and proposes one such facility.

We suggest a tungsten test section on the ITER first wall, at an early stage of operation, e.g. by means of a W coating on ITER Be tiles. Deposition of tungsten metal on a Be substrate has several challenges including proper growth and synthesis conditions, film stability and appropriate coating strategies. A W coating on tokamak floor tiles has been achieved in the past by plasma arc deposition of about 1–4 μm [25]. These coatings were deposited mostly on CFC surfaces. The application of tungsten coatings on Be surfaces raises questions on the stability of W films on Be. Be–W alloys are believed stable and can form after deposition of tungsten on a Be substrate [26]. One primary issue involves thin layers of W ($\sim$200 nm or more) deposited on Be and at temperatures above approximately 1000 K where segregation of Be to the W surface is measured and a defined inter-mixed W–Be region is formed at the near surface.

9. Conclusions

We assessed key plasma/surface interaction issues for an all-metal ITER PFC system for full power D–T operation. Subject to numerous model/data uncertainties (including edge plasma solutions, convective forces/impurity transport, tritium and helium effects on surfaces, Be/W alloy formation) we draw the following conclusions.

- Our results support the ITER research plan of eliminating carbon divertor material in the D–T phase—tungsten is essentially no worse than carbon in ELM/transient erosion—and eliminates the major issues of sputtering erosion and tritium/carbon codeposition. (The material of choice for *initial* ITER operation is largely outside the scope of this work.)
- A beryllium first wall has acceptable sputter erosion for ITER but would not extrapolate past ITER. Tritium/beryllium codeposition is a concern. PFC baking capability of 400 °C—even if only possible in the baffle and selected wall regions—is recommended to minimize T/Be inventory.
- A bare stainless steel wall works well from the sputter erosion standpoint.
- An all-tungsten PFC system offers very low sputter erosion, apparently negligible plasma contamination and eliminates T/Be codeposition and Be/W alloy formation concerns. He/W irradiation effects are probably tolerable in ITER.
- A tungsten wall test section is suggested, to study erosion/plasma contamination, and a simple W on Be coating implementation may be feasible.
- Divertor response to ELMs and other transients will restrict the acceptable plasma operating regime, although a reasonable operating window may be possible.
- Key research needs include He, Be, etc impingement effects on tungsten, ELM effects/mitigation and continuing plasma edge/plasma–surface interaction analysis and code validation.

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