# "Plasma Surface Interaction Issues of an All-Metal ITER"[1]

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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 surface, 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, core plasma contamination, ELM response, and He on W irradiation effects. Code package OMEGA computes PFC sputtering erosion/redeposition in an ITER full power D-T plasma edge regime with convective transport. The HEIGHTS package analyzes divertor 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 nm/s) or bare (stainless steel/Fe) wall (0.05 nm/s) for the low duty factor ITER, and are very low (0.002 nm/s) for W. Most wall-sputtered Be is redeposited on the wall itself or baffle region, with about 10% transported to the divertor target. T/Be codeposition in redeposited wall material could be significant (~2 gT per 400 s ITER pulse). Core plasma contamination potential from wall sputtering appears acceptable for Be (~2%), and negligible for W (or Fe) due to near-surface ionization of sputtered W (Fe) atoms and subsequent strong redeposition. A tungsten divertor likewise appears acceptable from the self-sputtering and plasma contamination standpoints, and would have negligible T/W codeposition. Be can grow on/near the strike point region of a W divertor, but for the predicted maximum surface temperature of ~800°C, deleterious Be/W alloy formation may be avoided. ELM's 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 carbon, but critical R&D issues remain, e.g., in areas of transient surface erosion (of all materials), W surface integrity with energetic He etc. bombardment, and in predictive plasma/surface interaction modeling generally. Steps are suggested to ameliorate problems and reduce uncertainties, e.g., via a 300 or 400°C baking capability for T/Be reduction, and using a deposited tungsten first wall test section.

## 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 analyzing 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 coupled studies of plasma edge/scrapeoff layer (SOL) parameters, single and mixed-material sputtering erosion/redeposition, plasma transient response, tritium codeposition, Be/W alloy formation, and He/W effects. We focus 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 ELM and other plasma transient limitations.

## 2. First Wall sputter erosion/redeposition

Fig. 1 shows the ITER PFC system design, with reference Be first wall surface, tungsten "baffle" region and with a tungsten divertor target. The plasma edge/SOL parameters and PFC response was studied via the US OMEGA code collaboration, using models for the

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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 neutrals code, REDEP/WBC code package, TRIM-SP sputter yield code, BPHI-3D sheath code, and other codes.

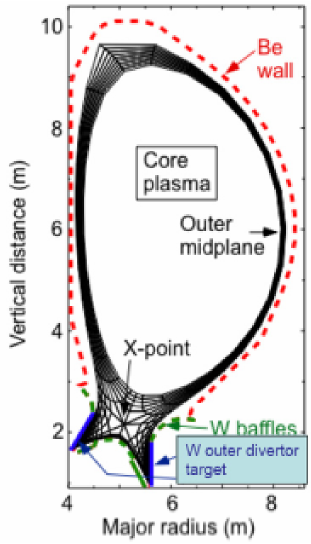


Figure 1  ITER Plasma Facing Component (PFC) system.

Convective transport results in ~x50 increase of both D-T and He ions and charge-exchange (CX) D-T neutrals to the first wall compared to diffusion-only transport [1,2]. This raises obvious concerns about high wall sputter erosion and subsequent transport. Table I shows sputter erosion rates for a Be or W outer first wall, and an alternative bare stainless steel (Fe) wall. Erosion is computed for incident D,T atoms from charge exchange, and D, T, He and trace O ions, using the respective particle sputter yields and energy/angular distributions, and assuming sheath acceleration of ions at the wall. In spite of high particle flux, the results show acceptable Be or Fe erosion rates for the low duty-factor ITER. The W sputter erosion rate is very low and would extrapolate to a high duty factor (DEMO etc) device.

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

| Wall surface | Sputtered current[a] | Erosion rate[b] | Erosion lifetime, 1 mm surface @1% duty factor |
|---|---|---|---|
| | atoms/s | m/s | years |
| Beryllium | $1.9 \times 10^{22}$ | $3.2 \times 10^{-10}$ | ~10 |
| Iron (stainless steel) | $1.0 \times 10^{21}$ | $5.0 \times 10^{-11}$ | ~60 |
| Tungsten | $5.6 \times 10^{19}$ | $1.8 \times 10^{-12}$ | ~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 II summarizes current transport/redeposition results for sputtered/ionized wall material. The analysis method is described in detail in [2]. Briefly, a full kinetic 3-D WBC computation is used, with input D-T full power UEDGE plasma background, with sputtered atoms launched with TRIM-SP derived velocity distributions; resulting ions are then subject to charge-changing and velocity-changing collisions with the plasma, including diffusion and convective force effects.

Table II 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 | .28 | .56 | .75 |
| Fraction to baffle | .62 | .43 | .25 |
| Fraction to divertor | .094 | .008 | $1.4 \times 10^{-4}$ |
| Fraction to edge plasma boundary | .006 | $4.0 \times 10^{-6}$ | 0 |
| | | | |
| 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; for outer wall/divertor components
[b] for sputtered atoms, normal to surface

Most sputtered material is redeposited on the wall and baffle, with from ~0-10% going to the divertor and ~0-1% reaching the edge/core plasma boundary, depending on the material. 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 Ref. [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 Fig. 1). The W ions then diffuse/flow back to the wall and/or flow along poloidal field lines to the other PFC components faster then 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 [3]; however the geometry/plasma is different than ITER. Also, higher than expected divertor Mo erosion/transport was seen in some C-MOD shots [4]; analysis of C-MOD Mo erosion via REDEP etc. codes is currently underway.

### 3. Divertor sputter erosion/redeposition

Fig. 2 shows electron temperature and D-T ion flux at the divertor. Erosion/redeposition analysis of a pure-W divertor, using this plasma solution (with associated density etc. parameters), shows near zero net sputter erosion, due to low gross sputtering to begin with, non-runaway self-sputtering (due to the <30 eV peak plasma temperature at the divertor), very high (~100%) redeposition, and zero core plasma contamination. (This contrasts to a carbon divertor target which has a peak sputter erosion rate of order 10 nm/s).

The effect of wall-sputtered Be transported to an initially W divertor target–not including any surface temperature effects–was assessed in [5] 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 net Be growth over ~2/3 of the target, due to re-sputtering by plasma ions and self-sputtering/reflection, but substantial growth, ~1 nm/s, occurs at/near the strike point.

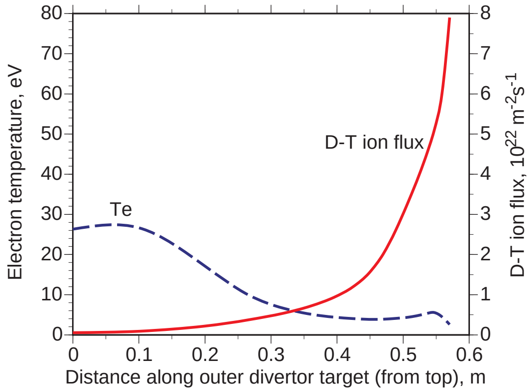


Figure 2  Electron temperature and D-T ion flux along the outer divertor target.  OMEGA computation; ITER edge plasma with convection.  From Ref. [5].  Strike point at ~0.55 m.

**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 ions with a high temperature surface to form low melting temperature tungsten beryllide alloys near the divertor strike point.  PISCES-B experiments show W-Be alloy formation for a W sample surface temperature exceeding 750 °C [6].  The measured diffusion rate for Be in W at 750 °C is $4.3\times10^{-15}$ cm$^2$/s, increasing to 5.8 $\times10^{-13}$ cm$^2$/s at 850 °C.  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.

Fig. 3 shows the heat flux and resulting surface temperature profile we compute for the ITER W monoblock armor divertor.  This is for the above-mentioned ITER base-case D-T plasma with 100 MW input from the core to the edge region.  Since the peak temperature is ~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.

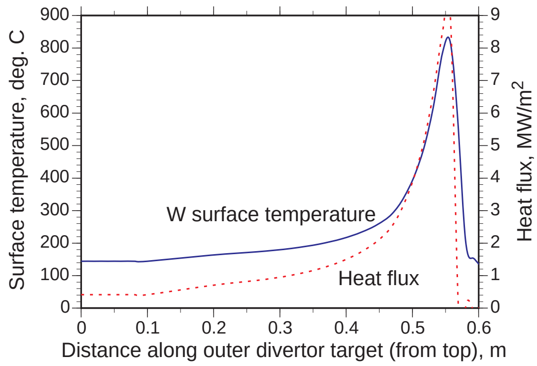


Figure 3.  Surface temperature and heat flux at the tungsten monoblock outer divertor.

## 4. 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 to estimates of ~3gT/400-s shot codeposition for a carbon divertor [7-8].

Be is known to trap T in redeposits at room temperature rates not that different than 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 (Fig. 3) the divertor strike point. The first uses a constant value of (D+T)/Be in codeposits of 0.08 at 200 °C [9]. The second uses a scaling law developed [10] 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 (~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 (x2 at the first wall to x40 at the baffle).

The codeposited tritium calculated using either approach is quite similar, 1.5–1.8 gT/400s-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) (~100 mg) in either case.

It is the tritium *release* behavior from codeposits that actually controls the retained tritium inventory. It is presently envisioned to increase the baking temperature of ITER from 240 °C to 300 °C. During a 240 °C bake out of the entire ITER vessel, one could expect to remove only ~20% of the tritium residing in Be codeposits. Increasing bake out to 300 °C would remove ~50% of the tritium and a 400 °C bake out would release 85-90% of the tritium. The tritium release behavior is described in more detail in [11]. Better estimates of ITER T/Be codeposition will require more spatially-refined plasma parameter, surface temperature, and trapping rate coupling, as well as inner wall Be sputtering/transport analysis.

## 5. Divertor ELM response

Edge-localized modes (ELMs) are a serious concern during normal H-mode operation of ITER and future tokamaks. During ELMs, an energy, $Q_{ELM}$ of ~1-10% of total core plasma energy $Q_o$, is released to the SOL and deposited on the divertor surface in a duration $\tau_{ELM}$ of ~0.1-1 ms with a frequency of ~1-10 Hz. The incoming power from SOL to divertor plate in ITER-like devices during an ELM can increase from ~10 MW/m$^2$ to ~300-3000 MW/m$^2$. The HEIGHTS simulation package solves detailed plasma-transient particle energy deposition, evolution of surface materials, debris formation, vapor MHD, atomic physics and radiation transport, and erosion physics [12-14]. 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. Fig. 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 melting temperature and a melt layer thickness of 100 µm is developed for giant

ELMs ($Q_{ELM}$ ~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.

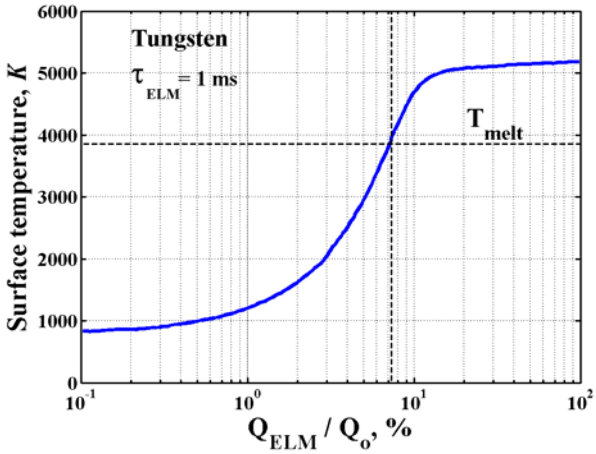


Figure 4  ITER outer tungsten divertor ELM response as a function of ELM energy fraction; $Q_0$ = 127 MJ released at midplane, ELM duration 1 ms [12].

A major HEIGHTS result, however, is that carbon has similar ELM and other transient (disruptions, VDE's) concerns as tungsten [12-14].  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 vapor cloud for either material can damage nearby components.  We have identified acceptable and unacceptable ELM parameter windows for ITER, shown in Fig. 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 vaporization erosion than carbon but, as stated, significant melting occurs.  For longer deposition times (~1ms) 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 1500K which is even lower than the melting temperature of potentially forming Be-W alloys [12].

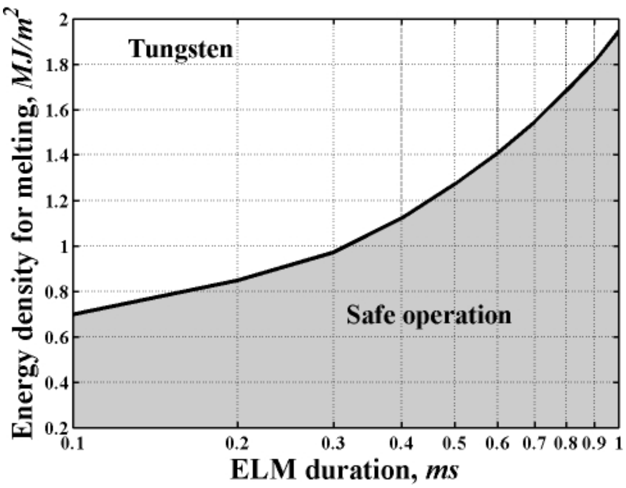


Figure 5  HEIGHTS parameter window for W divertor acceptable (no-melt) ELM response.

## 6. 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.

W fuzz is a widely seen nano-scale tendril formation of plasma exposed surfaces due to the action of >10 eV He at elevated surface temperature, e.g., [15]. Experiments indicate a temperature threshold of about 700 °C for fuzz formation. This would imply that an ITER first wall, baffle, and most of the divertor–per above temperature calculation–would avoid fuzz problems.

At lower temperatures, i.e. in the divertor region outside the strikepoint, the concern is blistering and development of nano-structures. It is presently difficult to project their effects on W 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 dependencies are unknown and the W surface damage is not universally seen in tokamaks. The most pessimistic projection comes from laboratory observation at RT, reviewed in [16], of ~20 nm structures formed at ~$5x10^{21}$ He/$m^2$, which projects a 4-40 nm/s erosion rate if 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 [4,17], which typically have some low fraction of He from wall conditioning, is a peak net erosion rate ~0.1 nm/s campaign integrated. Recent W exposure in the LHD divertor with pure He plasmas [18] showed a net erosion ~ 6 nm/s from blistering, which if scaled with He fluence at ~2%, also projects ~ 0.1 nm/s net erosion. Therefore, the preponderance of experience points toward acceptable net erosion rates, although the full impact of such "non-atomistic" W erosion for ITER and beyond is still to be determined.

## 7. Testing

Verifying the robustness of PFCs armored with W or Be tiles is crucial for ITER. We have a history of experiments on water-cooled PFCs with metal armor as well as ongoing tests, e.g., First Wall Quality Mockups for ITER. 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 tungsten test section on the ITER outer first wall is suggested, e.g., by means of a W coating on ITER Be tiles. Deposition of tungsten metal on Be substrate has several challenges including proper growth and synthesis conditions, film stability and appropriate coating strategies. W coatings on tokamak floor tiles has been achieved in the past by plasma arc deposition of about 1-4 um [19]. 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 [20]. One primary issue involves thin layers of W (~200 nm or more) deposited on Be and at temperatures above approximately 1000K where segregation of Be to the W surface is measured and a defined inter-mixed W-Be region is formed at the near surface.

## 8. 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, we conclude:

- Our results support 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.
- 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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