Plasma surface interaction issues of an all-metal ITER


IAEA 22nd Fusion Energy Conference
13-18 Oct. 2008 Geneva Switzerland
Surface interaction issues of an all-metal ITER

Jeff Brooks¹
Jean Paul Allain¹
Russ Doerner²
Ahmed Hassanein¹
Richard Nygren³
Tom Rognlien⁴
Dennis Whyte⁵

US Plasma Facing Components (PFC) Group
¹Purdue University, West Lafayette, IN
²University of California, San Diego
³Sandia National Laboratories
⁴Lawrence Livermore National Laboratory
⁵Massachusetts Institute of Technology
Oak Ridge National Laboratory
University of Illinois, Urbana/Champaign
University of California, Los Angeles
Princeton Plasma Physics Laboratory
General Atomics
The choice of Plasma Facing Component (PFC) surface materials remains a critical and contentious issue for ITER and beyond. PFC plasma/surface interactions affect: component lifetime, tritium inventory, plasma contamination, plasma operation.

Why consider all-metal?
--- Use 1 or 2 materials, not 3; reduce/eliminate mixed-material concerns.
--- Eliminate tritium retention in carbon codeposits in D-T phase.
--- Better extrapolation to high-neutron ITER phase and post-ITER devices.

This work examines several critical areas
- Code Package OMEGA analysis of PFC sputter erosion/redeposition
- Divertor surface temperature
- Be/W alloy formation
- Tritium retention/codeposition for Be wall, W wall/divertor
- HEIGHTS analysis of ELM etc. response of W divertor
- Tungsten helium effects assessment
- Testing requirements

Not covered here but under investigation in US
- Consequences of melted surfaces, dust
- Neutron damage on bulk retention
ITER Plasma Facing Components (PFCs)

We examine:

1) Reference Be first wall surface (outer wall region)
2) “Bare first wall” (Fe)
3) Tungsten first wall
4) Tungsten baffle/divertor target
Sputtering erosion/redeposition analysis

Package-OMEGA*

- **UEDGE/DEGAS**: D-T ion and neutral fluxes to divertor, wall; scrape-off layer plasma parameters (fluid/ Monte-Carlo)
- **TRIM-SP, ITMC**: sputter yields (binary collision, single and mixed-material)
- **WBC**: sputtered atom/ion transport in scrape off layer (3-D, full-kinetic, Monte Carlo)
- **REDEP/WBC package**: divertor, wall erosion/redeposition analysis
  (impurity transport, atomic and molecular processes, sheath, sputter yields, tritium codeposition, etc.)
- **W-MIX, MD**: mixed-material surface evolution, molecular-dynamic sputter yields
- **BPHI-3D**: sheath analysis (3-D kinetic)

**Data** (where available)

* Omnibus Modeling of Erosion Generalized Analysis [1-3]

**UEDGE/DEGAS ITER boundary plasma simulation**

Major differences in edge plasma solution with convective flow: Strong first-wall plasma interaction

First wall ~700 m²  
Baffle ~ 100 m²  
Divertor ~ 50 m²

D-T flux to the wall

~X50 wall flux with convection

with convection
higher density at wall

no convection (diffusion only)
ITER first wall sputtering rates; OMEGA-analysis, convective edge plasma regime

<table>
<thead>
<tr>
<th>Wall material</th>
<th>Sputtered current (^a)</th>
<th>Erosion rate (^b)</th>
<th>Erosion lifetime, 1 mm surface (@) 1% duty factor years</th>
</tr>
</thead>
<tbody>
<tr>
<td>Beryllium</td>
<td>(1.9 \times 10^{22}) atoms/s</td>
<td>(3.2 \times 10^{-10}) m/s</td>
<td>(\sim 10)</td>
</tr>
<tr>
<td>Iron (stainless steel)</td>
<td>(1.0 \times 10^{21})</td>
<td>(5.0 \times 10^{-11})</td>
<td>(\sim 60)</td>
</tr>
<tr>
<td>Tungsten</td>
<td>(5.6 \times 10^{19})</td>
<td>(1.8 \times 10^{-12})</td>
<td>(\sim 1700)</td>
</tr>
</tbody>
</table>

\(^a\) outer first wall  
\(^b\) w/o peaking, if any, due to gas puffing charge exchange

- Be sputter erosion acceptable for low duty-factor ITER; but will not extrapolate post-ITER (< 1 year in steady-state device)  
- W erosion very low  
- Bare-wall erosion low
Transport summary of sputtered outer first wall material; WBC code, $10^6$ histories/run. Plasma with convection, reference impurity convection model.

<table>
<thead>
<tr>
<th>Parameter (redeposited ions)</th>
<th>Beryllium</th>
<th>Iron</th>
<th>Tungsten</th>
</tr>
</thead>
<tbody>
<tr>
<td>Ionization mean free path *, cm</td>
<td>11.5</td>
<td>6.7</td>
<td>3.5</td>
</tr>
<tr>
<td>Fraction to wall</td>
<td>.28</td>
<td>.56</td>
<td>.75</td>
</tr>
<tr>
<td>Fraction to baffle</td>
<td>.62</td>
<td>.43</td>
<td>.25</td>
</tr>
<tr>
<td>Fraction to divertor</td>
<td>.094</td>
<td>.008</td>
<td>1.4 x10^{-4}</td>
</tr>
<tr>
<td>Fraction to edge plasma boundary</td>
<td>.006</td>
<td>4.0 x10^{-6}</td>
<td>0.000000</td>
</tr>
</tbody>
</table>

* sputtered atoms, perp. to wall

- **Be core plasma contamination**: Acceptable (~2%)
- **W core plasma contamination**: Negligible
- Most material redeposits on wall or baffle

{Note: These W contamination results for ITER in possible contradiction to some high-Z tokamak results, e.g., Mo/C-MOD data-analysis in progress.}
Tritium codeposition

- For a tungsten divertor and first-wall;
  - Codeposition is negligible due to minimal sputter erosion/redeposition and low T/W trap ratios. (*Implantation / permeation not considered here*).
  - This contrasts to estimates of ~3 gT/400-s shot codeposition for a carbon divertor.

- For a beryllium wall:
  - Two 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 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.

Tritium codeposition in Beryllium

- The codeposited tritium calculated using either approach is quite similar, 1.5–1.8 gT/400s-shot due to beryllium outboard first-wall erosion.

- However, the T codeposit locations are slightly different:
  -- Predominantly on 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 long-term retained tritium inventory. It is presently envisioned to increase the baking temperature of ITER from 240 °C to 300-350 °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
-- A 400 °C bake out would release 85-90% of the tritium.
Beryllium / Tungsten interaction

Be-W interaction can lead to extreme material failure

e.g. PISCES crucibles for molten Be

<table>
<thead>
<tr>
<th>Contact W wall</th>
<th>Inner wall coating</th>
<th>Crucible failure zone</th>
</tr>
</thead>
<tbody>
<tr>
<td>7% W, 3% O</td>
<td>4% W, 95% Be, 1% O</td>
<td>9% W, 70% Be, 14% C, 7% O</td>
</tr>
<tr>
<td>( \text{Be}_2\text{W?} )</td>
<td>( \text{Be}_{12}\text{W?} )</td>
<td></td>
</tr>
</tbody>
</table>
Computed heat flux and surface temperature at the ITER tungsten monoblock outer divertor (strike point at ~0.55m)

• Maximum temperature ~ 800 °C in very small region ~ 2 m²
OMEGA/W-MIX results: Net beryllium growth rate on initial tungsten divertor (and gross rate from wall-to-divertor transport)

• No beryllium growth over most of the surface: remains underlying tungsten due to intense Be re-sputtering and ionization reflection “down” target.
• 50% of Be is “lost” from target regions (e.g ends up on domes)
• Be growth occurs at/near strike point (region of low $T_e$)
• Thermal effects (evaporation) do not affect results (but transient heating needs evaluation)
Be/W Alloy Formation-Assessment

• The surface temperature is high enough (≥750 °C threshold for Be/W formations) to promote significant alloy growth, if at all, only in the region extending roughly 2 cm on either side of the strike point.

• 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.

• Of course, Be/W issues can be completely avoided with an all-tungsten system.
ELM Analysis

Plasma-Target Interaction Physics Models in code package HEIGHTS
HEIGHTS results: ITER tungsten divertor ELM response as a function of ELM energy fraction; $Q_0 = 127$ MJ released at midplane, ELM duration 1 ms

- Melting occurs for high energy fractions (with considerable material loss)
• A safe-operation window exists for tungsten.
• Note: Carbon does not melt, but ELM material losses not fundamentally different than tungsten.
• Effects of repetitive damage remains to be examined.
Helium effects on Tungsten

- Tungsten “fuzz” is a widely seen nano-scale tendril formation of plasma exposed surfaces due to the action of >10 eV He, at elevated surface temperatures (>1100 K).

- At lower temperatures the concern is blistering and nano-structure development.

- It is unclear if the surface damage is detrimental (through loss of micro-particles), or beneficial (due to release of trapped H/He).

- We assessed present high-Z tokamak results, and LHD pure-He plasma experience. Estimate effective erosion rate ~ 0.1 nm / s.

- For the moderate ITER surface temperatures, the preponderance of experience points toward acceptable tungsten “non-atomistic” erosion rates. Yet this effect may dominate atomistic sputtering.
We suggest a tungsten first wall test section for ITER
- Starting with H phase?
- Fe test section?

Example: 1 \( \mu \)m thick, \(~50m^2\) area W deposited on the reference Be wall surface (e.g., as per Ch. Linsmeier et al., J. Nucl. Mater. 363-365(2007)1129.)

Test sputter erosion, transport, core plasma contamination
We assessed several 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:

- These results support eventual elimination of carbon divertor material in the D-T phase—tungsten has same energy density limit as carbon in ELM/transient erosion, and eliminates sputtering erosion and tritium/carbon codeposition.
- A beryllium first wall has acceptable sputter erosion for ITER (~0.3 nm/s), but would not extrapolate past ITER. T/Be codeposition is a concern (~2g/400s). PFC baking capability of 400 °C – even if only possible in the baffle and selected wall regions—is recommended to minimize T/Be codeposition.
- 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 effects on W may be acceptable in ITER due to moderate temperatures.
- A ~50 m² 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 for tungsten (or any material) although a reasonable operating window seems possible.
- Key research needs include He, Be, N impingement effects on tungsten, ELM effects/mitigation, and continuing plasma edge/plasma-material interaction analysis and code/data validation.