Development of Transient Tolerant Plasma Facing Material

C.P.C. Wong¹, B. Chen¹, D.L. Rudakov², A. Hassanein³, and A.G. McLean⁴

¹General Atomics, PO Box 85608, San Diego, CA 92186-5608, USA
²University of California-San Diego, 9500 Gilman Dr., La Jolla, CA 92093, USA
³Purdue University, West Lafayette, IN 47907, USA
⁴Oak Ridge National Laboratory, PO Box 2008, Oak Ridge, TN 37831, USA

Presented at
13th International Workshop on PFM and components for Fusion Application
1st International Conference on Fusion Energy Materials Science
Rosenheim, Germany

May, 9-13, 2011

*This work was supported in part by the U.S. Department of Energy under DE-FC02-04ER54698, DE-FG02-07ER54917, DE-C52-07NA27344, and DE-AC05-00OR22725.
Abstract

Plasma facing material (PFM) is a critical element of the high performance DT tokamak reactor design. Unfortunately, the commonly proposed material W could suffer radiation damage from charged alpha particle implantation and experience blistering at the first wall and the formation of submicron fine structure at the divertor. Furthermore, it will melt under disruption and runaway electron (RE) events. As a conservative engineering design, the first wall and divertor PFM for steady state power reactor must withstand a few unanticipated disruptions and RE events even when the disruption and RE mitigation techniques are fully engaged. Using a low-Z sacrificial material, like Si, deposited on the W-surface could allow W to withstand a few disruptions and RE events without serious damage while retaining the capability of transmitting high grade heat for power conversion. An equivalent Si thickness of 10 μm is sufficient to form a vapor shielding layer during a disruption that would protect the W substrate from serious damage. Accordingly, transient tolerant PFM surface test buttons have been fabricated and initial results have been obtained with exposure in the DIII-D divertor.
Surface Material is a Key Item for Fusion Development

Surface material is critically important to next generation tokamak devices

- Plasma performance is affected by transport of impurities
- Surface heat removal, tritium co-deposition and inventory will have impacts on material selection for devices beyond ITER
- Radiation effects from neutrons and edge alphas, material design limits and component lifetimes will have to be taken into consideration

C and Be will not be suitable for the next generation devices and DEMO due to surface erosion and radiation damage. Presently W is the preferred choice, but significant issues have been identified.
W Temperature & PMI are Coupled

**PISCES-A: D₂-He plasma**
M. Miyamoto et al., NF 49 (2008) 065035
600 K, 1000 s, 2.0x10²⁴ He⁺/m², 55 eV He⁺
- Little morphology
- He nanobubbles form
- Occasional blisters

**NAGDIS-II: pure He plasma**
N. Ohno et al., in IAEA-TM, Vienna, 2006
1250 K, 36000 s, 3.5x10²⁷ He⁺/m², 11 eV He⁺
- Surface morphology
- Evolving surface
- Nano-scale ‘fuzz’

**PISCES-B: mixed D-He plasma**
M. Baldwin et al., NF 48 (2008) 035001
1200 K, 4290 s, 2x10²⁶ He⁺/m², 25 eV He⁺

**NAGDIS-II: He plasma**
D. Nishijima et al., JNM (2004) 329-333 1029
- Surface morphology
- Shallow depth
- Micro-scale
When exposed to He at high temperature, W surface showed growth of W nanostructure from the bottom; the thickness increases with plasma exposure time.

Baldwin and Doerner, Nuclear Fusion 48 (2008) 1-5

Equilibrium thickness of fuzz is expected to form in the erosion zone of a W-divertor, erosion with lower sputter yield than bulk W

Doerner, UCSD, US VLT conf. call Jan. 2011

**ITER disruption loading:**

10-30 MJ/m² for 0.1 to 3 ms

Irreversible surface material damage

M. Rödig, Int. HHFC workshop, UCSD Dec. 2009

We cannot eliminate unpredicted disruptions even if disruption detection and mitigation work perfectly
Carbon Plasma Impurity Can Inhibit W Morphology Change with D$_2$-He with Carbon Discharges

\[ E_i = 15 \text{ eV}, \quad T_s = 1100 \text{ K}, \quad \text{Fluence} = 10^{25} \text{ He}^+/\text{m}^2, \]
\[ n_{\text{He}^+}/n_e \sim 10\%, \quad n_{\text{C}^+}/n_e < 0.1\% \quad \Delta t = 3600 \text{ s} \]

Similar results were obtained with Be and could be projected for B and Si

At \( E_i = 15 \text{ eV} \), C deposited on W is not sputtered away
\[ \Rightarrow \quad \text{W-C layers inhibit He induced morphology changes} \]

Baldwin and Doemer, PISCES, UCSD
A Possible PFM Concept that Could Satisfy all Requirements

The concept: Si-filled W-surface

- Protect the W surface from He damage with the presence of Si
- Exposed W will have a low erosion rate
- Transmit high heat flux, e.g. the W-disc can be about 2 mm thick and with indentations, thus retaining high effective $\kappa_{th}$ of W layer, necessary for DEMO
- Should provide enough Si to withstand ELMs and a few disruptions (modeling showed vaporized Si $\sim$10 $\mu$m/disruption including vapor shielding effect) “W-$T_{melt}$ at 3410°C, Si-$T_{melt}$ at 1412°C, Si-$T_{boil}$ at 2480°C”
- Should be able to control tritium inventory at temperature $\sim$1000°C
- Suitable real time siliconization could be used to replenish Si when and where needed
Divertor Surface Erosion and Vapor Shield Protection from Disruptions

Disruption condition, ITER parameters:
Energy density $E = 25 \text{ MJ/m}^2$
Impact duration $t = 0.1 \text{ ms}$
Magnetic field $B = 5.0 \text{ T}$
Incline angle $\alpha = 5.0 \text{ deg}$

Results from Prof. A. Hassanein, Purdue U.
Projected DEMO PFC FW and Divertor Design Approaches

**First Wall**

- DEMO module
- Plasma side
- One-sided roughened channel
- He

**Divertor**

- He impingement heat transfer
- ARIES CS design

**Temperatures & limits**

- Tokamak coating?
- W-layer >700°C
- Joint (TBD)
- ODFS <700°C
- Joint (TBD)
- RAFM >350°C
- <550°C
- 8 MPa at ~350°C

**Deimos module**

- He

**Plasma side**

- One-sided roughened channel

**Concept 2**

- Slot jet

**General Atomics**
Layered First Wall Design Could Handle up to 1 MW/m² with 2-D, 3-D One-sided Roughening of He Coolant Channels

Neutron wall loading at 3 MW/m²

Heat flux, MW/m²

$T_{\text{max}}$, °C

$T_{\text{max}}$ - W, $K_{\text{th}}$ at 25 W/m.K (A conservative value)

$T_{\text{max}}$ - ODFS, $K_{\text{th}}$ at 20 W/m.k

$T_{\text{max}}$ - FS, $K_{\text{th}}$ at 20 W/m.k

$T_{\text{min}}$ - FS

ΔX = 2 mm

ΔX = 4 mm Assumed thicknesses

ΔX = 3 mm

He heat transf. coeff. enhanced with 2-D, 3-D roughening
Si-W Surface Development

- 2008: started with BW-mesh, but the presence of C formed B₄C, WB, W₂B, W₂B₅, WC, and W₂C, thus breaking up the mesh

- 2009 changed from mesh to plate, but B fill fell out of the holes

- Switched to Si due to much better match in the coeff. of thermal expansion between Si and W

- High melting temperature of Si can form low melting point W-Si compounds

- DIII-D boronization confirmed B coating thickness of < 1 μm

- 2010: Drilled indentations on W-button and they were filled with Si in powder form with binder and sintered

- Si filled W buttons exposed in DIII-D
Initial Results of Transient Tolerant Si-filled W-buttons

Si filled W-buttons

Loaded DiMES sample
2 Si-W, 3 graphite, 2 W buttons

W-buttons with 1 mm dia. indentations

Sample exposed To 4 LSN discharges

Exposed in DIII-D lower divertor

After one additional disruption shot without thermal dump on DiMES

Shot 14261-14264

Shot 142706

2011 buttons
Plasma Shot #142706, with Relative Stable Plasma Shape

Added neutron beam injected at 2000 ms, plasma ended with disruption ~ 3100 ms but migrated upward, no thermal dump on DiMES.

Langmuir probe 23
Data on electron density, #/cm³

Langmuir probes

55 ms 105 ms 505 ms 1005 ms 2005 ms

Plasma evolution

DiMES
Mostly CII/CI\textsubscript{III} Emission Measured During Discharge and Disruption (387, 392, 407 nm), Additional CI (375 nm) in Disruption

Ocean Optics USB2000, DiMES-viewing chord, DIII-D shot 142706
Vertical lines for nearby CI, CII, CIII, and CIV lines from Atomic Line List
Details Show Melted Si but Minimal Transport
Si-W Buttons Summary

- As expected, surface Si on the W button readily got removed during discharges, at least from the first 4 shots ($B_T = 1.88$ T and $I_p = 1.08$ MA); Si melting could have occurred during these shots.

- Favorable result was that much of the Si is retained in the indentations even under additional exposure (142706) ($B_T = 1.7$ T and $I_p = 1.2$ MA); the radiation is mainly from carbon.

- Retained Si could demonstrate the vapor shielding effect to protect the W-button surface from melting under disruption and RE events, but this needs to be confirmed.

- W-buttons were not damaged, observed cracks could be due to drilling of the indentations.

- New samples have been fabricated and will be exposed to disruption and RE events during the 2011 DIII-D operation campaign.