Development of Transient Tolerant Plasma Facing Material

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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.

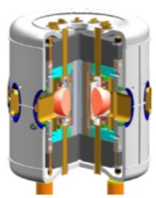
Surface Material is a Key Item for Fusion Development

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

DIII-D

C

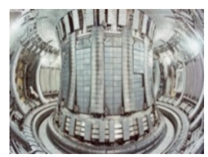


C-Mod

Mo B-coated

AUG

W

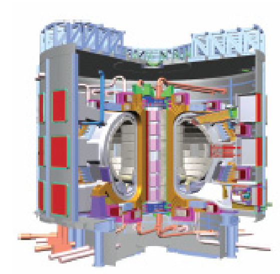


JET-ILW

Be/W/C

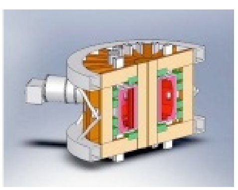
EAST

C/W



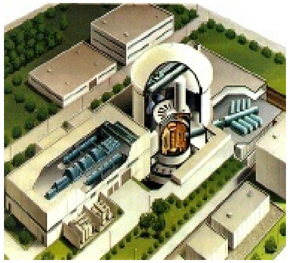
ITER

Be/W/C



CTF/FNSF/FDF

?



DEMO

?

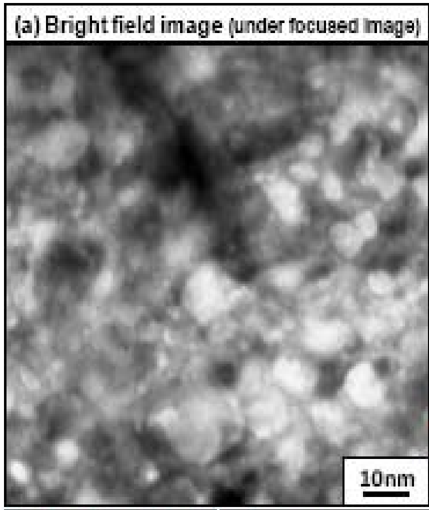
(High neutron and edge alpha fluence)

← **Surface material options** →

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

~ 600 - 700 K | ~ 900 – 1900 K | > 2000 K



PISCES-A: D_2-He plasma

M. Miyamoto et al. NF (2009) 065035

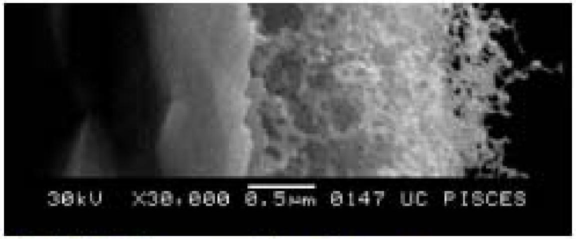
600 K, 1000 s, 2.0×10^{24} He^+/m^2, 55 eV He^+

- Little morphology
- He nanobubbles form
- Occasional blisters

PISCES-B: mixed D-He plasma

M.J. Baldwin et al, NF 48 (2008) 035001

1200 K, 4290 s, 2×10^{26} He^+/m^2, 25 eV He^+



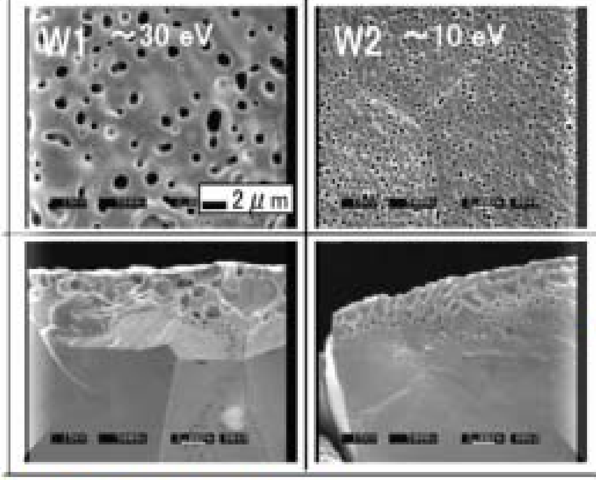
NAGDIS-II: pure He plasma

N. Ohno et al., in IAEA-TM, Vienna, 2006

1250 K, 36000 s, 3.5×10^{27} He^+/m^2, 11 eV He^+

- Surface morphology
- Evolving surface
- Nano-scale 'fuzz'

2.6×10^{27} /m² 3.7×10^{23} /m²s 7200 s 2100 K	0.9×10^{27} /m² 1.2×10^{23} /m²s 7200 s 2600 K



NAGDIS-II: He plasma

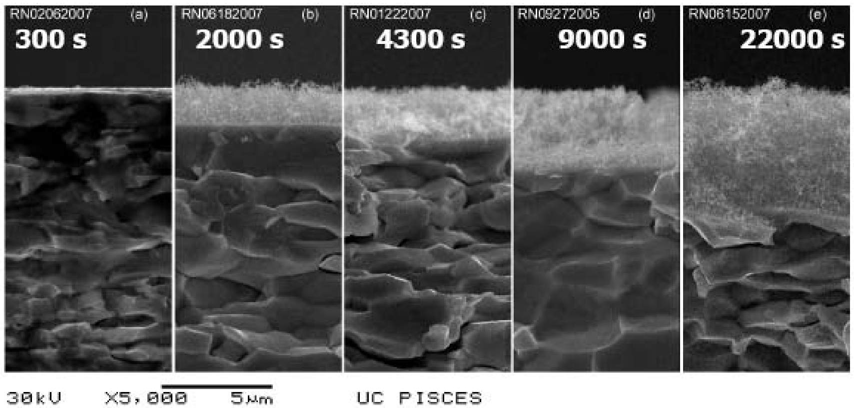
D. Nishijima et al. JNM (2004) 329-333 1029

- Surface morphology
- Shallow depth
- Micro-scale

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

Significant Issues Projected for W-surface Operation

SEM cross-sections of W targets exposed to PISCES-B pure He plasmas.



Consistent He plasma exposures: T = 1120 K, Γ_{He+}= 4–6 ×10[22] m^{-2}s^{-1}, E_{ion} ~ 60 eV

PISCES

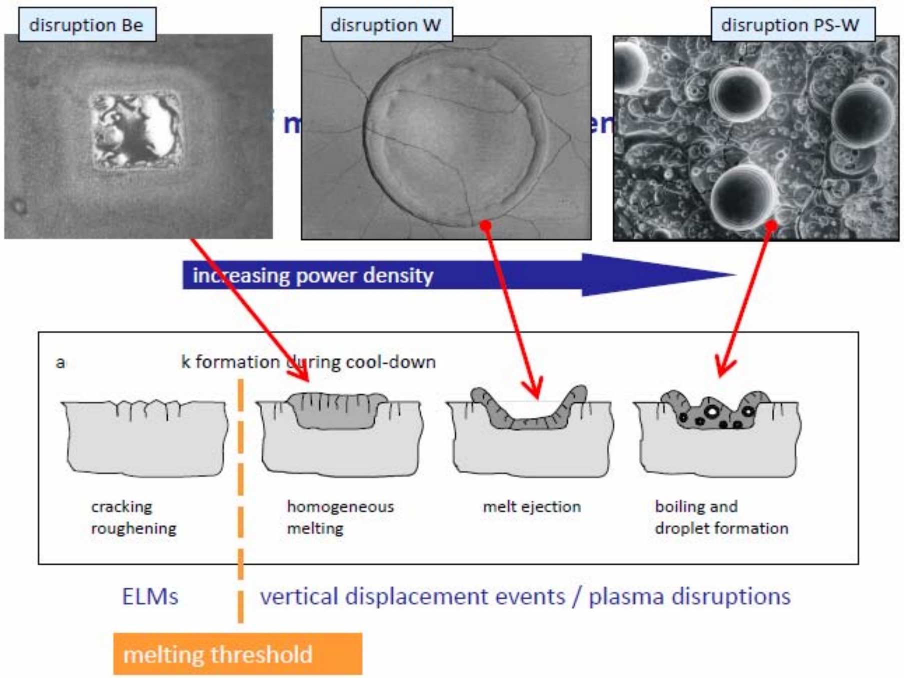
When exposed to He at high temperature, W surface showed growth of W nanostructure from the bottom; the thickness increases with plasma exposure time

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

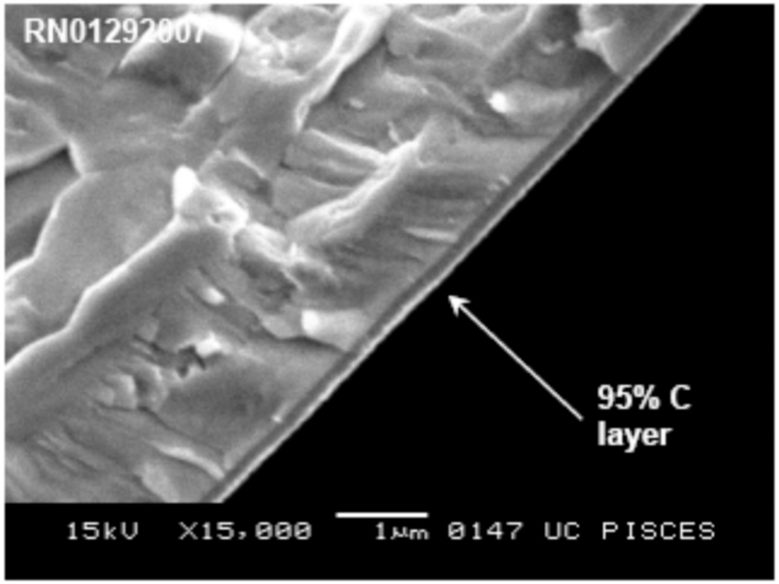
We cannot eliminate unpredicted disruptions even if disruption detection and mitigation work perfectly

Carbon Plasma Impurity Can Inhibit W Morphology Change with D_2-He with Carbon Discharges

PISCES

E_i = 15 eV, T_s = 1100 K, Fluence = 10^{25} He^+/m^2,
$n_{He+}/n_e \sim 10\%$, $n_{C+}/n_e < 0.1\%$ Δt = 3600 s

D_2-He with C



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

At E_i=15 eV, C deposited on W is not sputtered away

⇨ **W-C layers inhibit He induced morphology changes**

PISCES

UCSD Center for Energy Research

Baldwin and Doerner, PISCES, UCSD

A Possible PFM Concept that Could Satisfy all Requirements

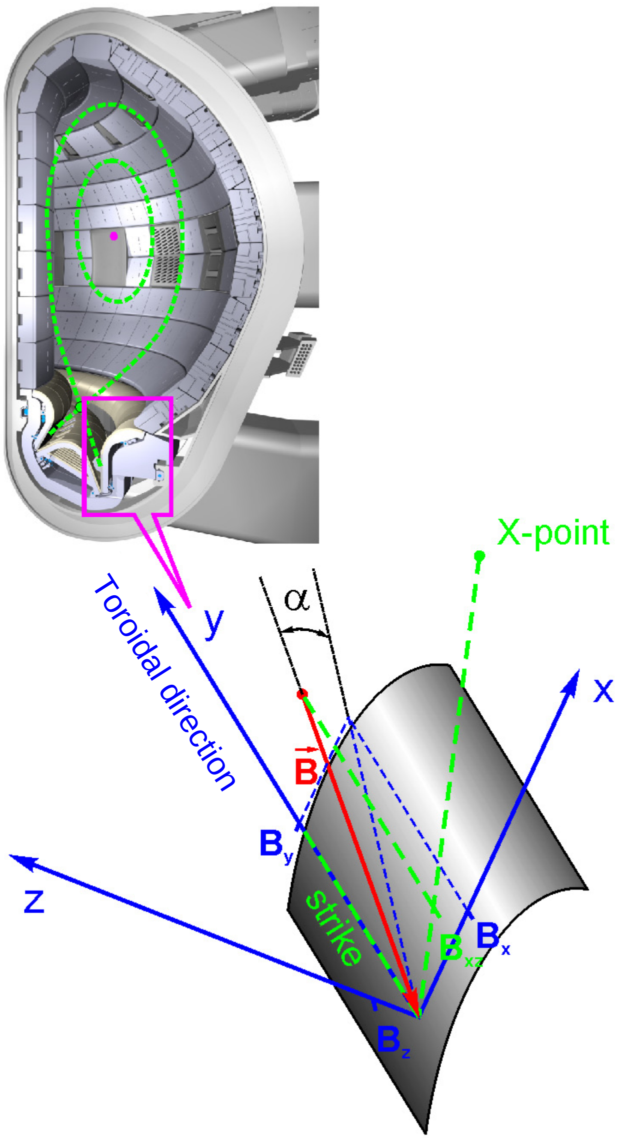
The concept: Si-filled W-surface

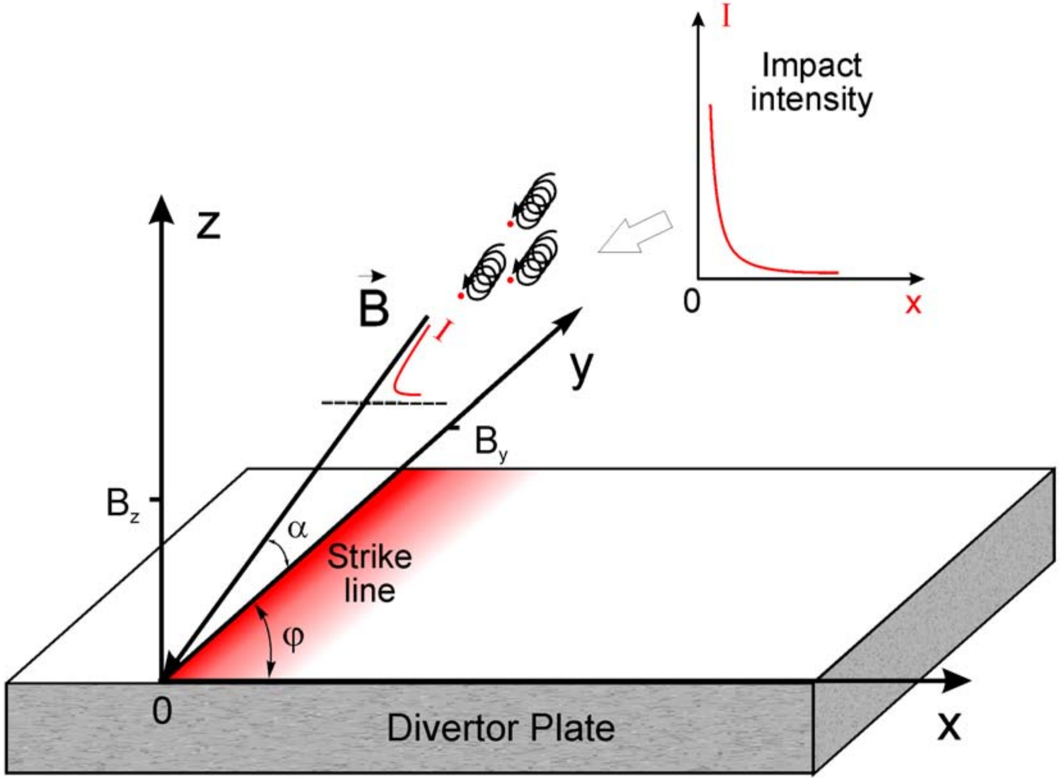
W-buttons

W-buttons filled with Si

- **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 κ_{th} of W layer, necessary for DEMO**
- **Should provide enough Si to withstand ELMs and a few disruptions (modeling showed vaporized Si ~10 µm/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 ~1000°C**
- **Suitable real time siliconization could be used to replenish Si when and where needed**

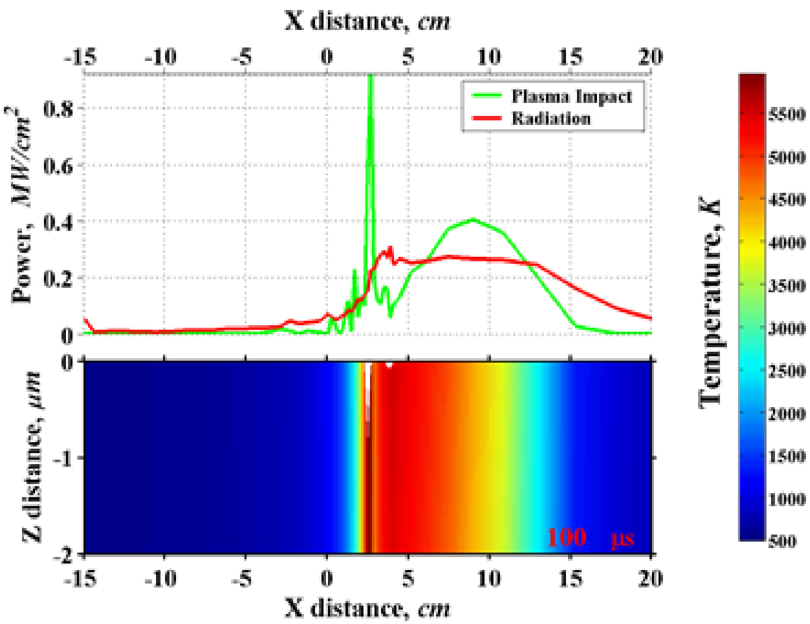
Vapor Shielding Modeling Geometry

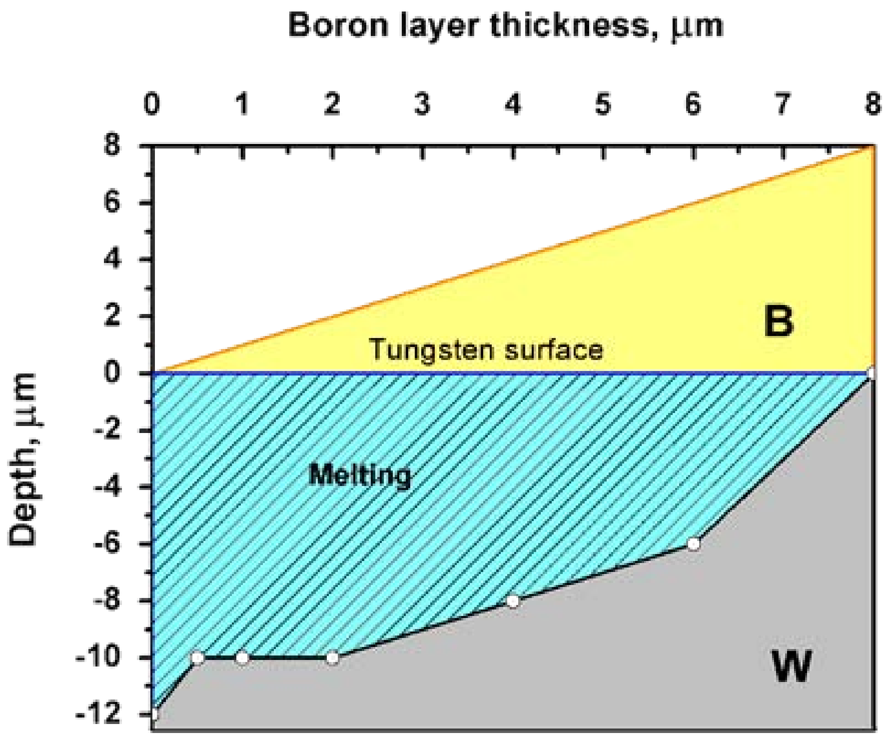




Divertor Surface Erosion and Vapor Shield Protection from Disruptions

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



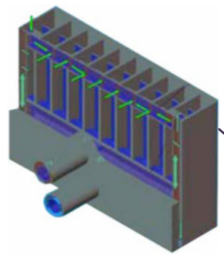


Results from Prof. A. Hassanein, Purdue U.

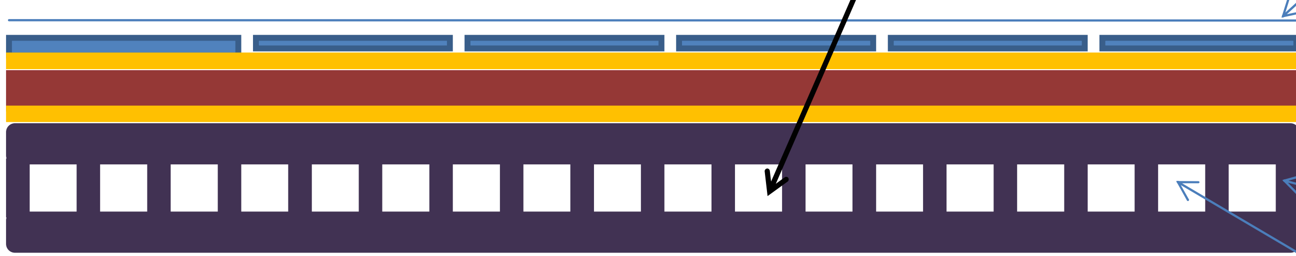
Projected DEMO PFC FW and Divertor Design Approaches

First Wall

DEMO module



Plasma side



One-sided roughened channel

Temperatures & limits

Tokamak coating?

W-layer >700°C

Joint (TBD)

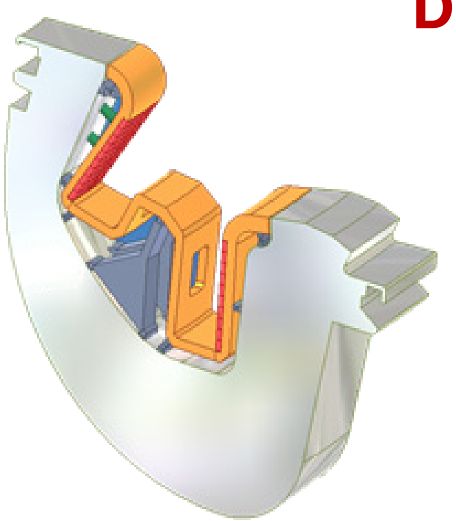
ODFS <700°C

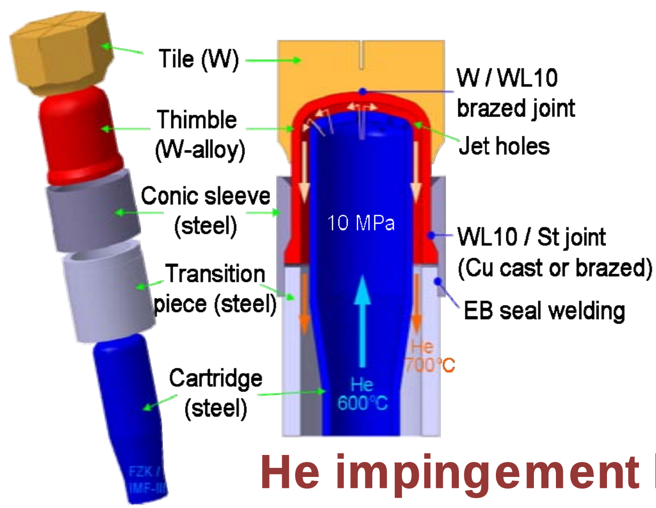
Joint (TBD)

RAFM >350°C <550°C

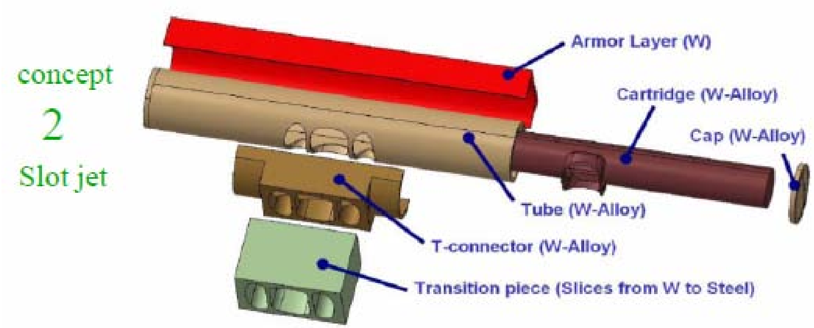
8 MPa at ~350°C

Divertor



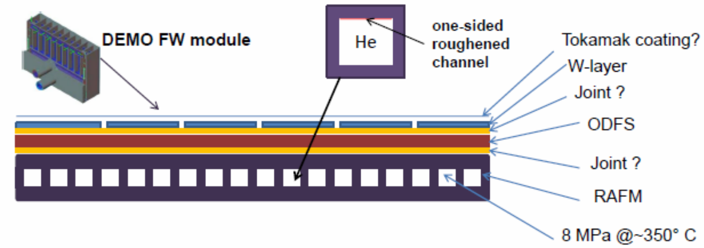


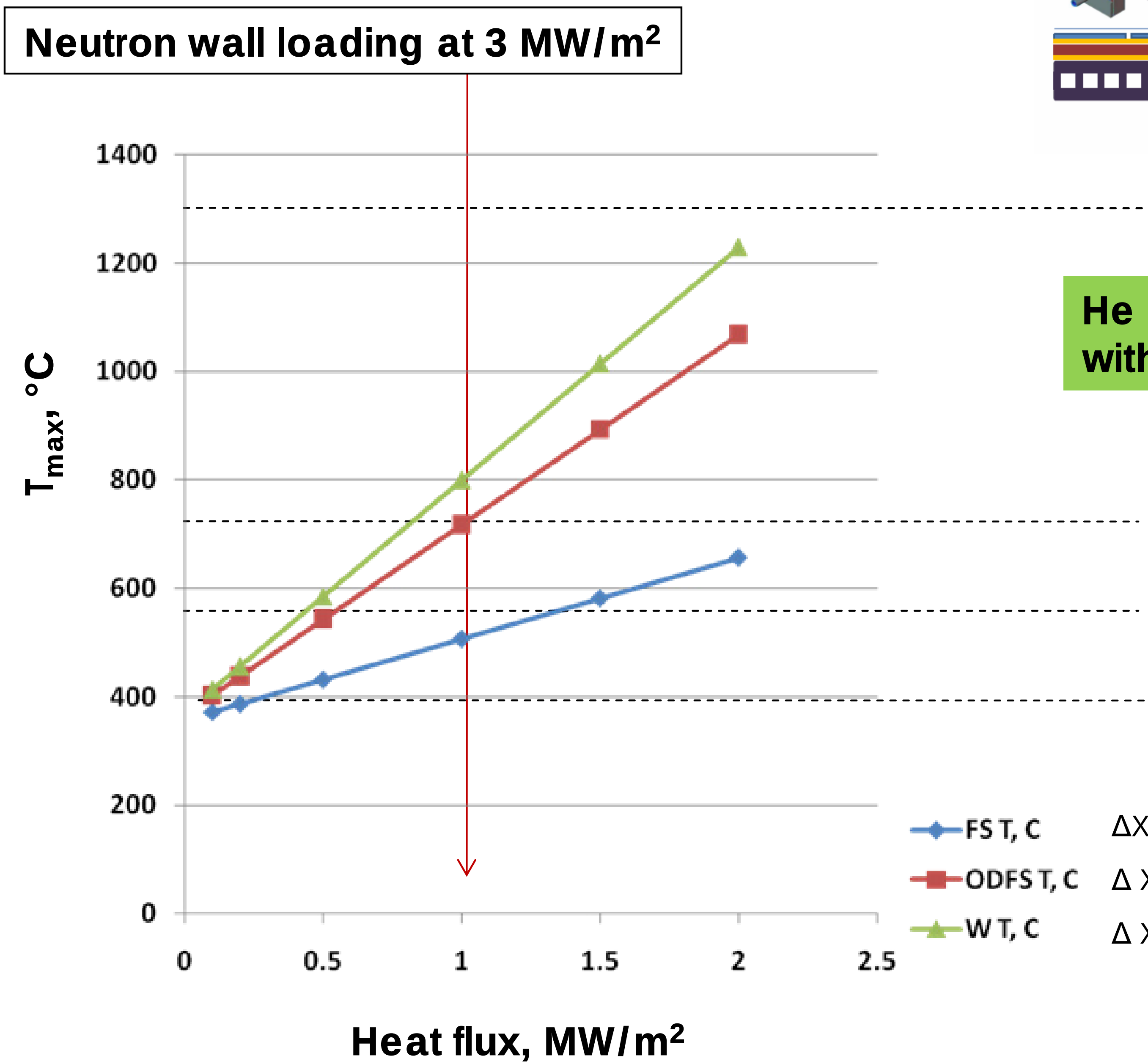
• ARIES CS design



He impingement heat transfer

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

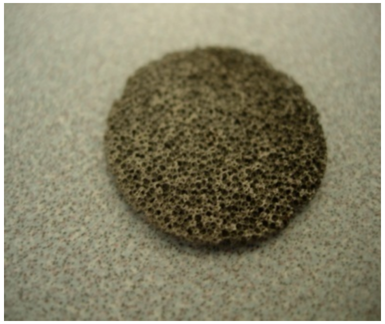




He heat transf. coeff. enhanced with 2-D, 3-D roughening

Si-W Surface Development

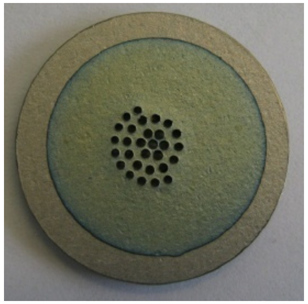
- 2008: started with BW-mesh , but the presence of C formed B_4C, WB, W_2B, W_2B_5, WC, and W_2C, thus braking up the mesh
- 2009 changed from mesh to plate, but B fill fell out of the holes
- Switched to Si due to much better match in the coeff. of thermal expansion between Si and W
- High melting temperature 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



W-mesh

Damaged W-mesh

W-disc



B-coating

W-buttons

W-buttons with Si

Initial Results of Transient Tolerant Si-filled W-buttons

Si filled W-buttons

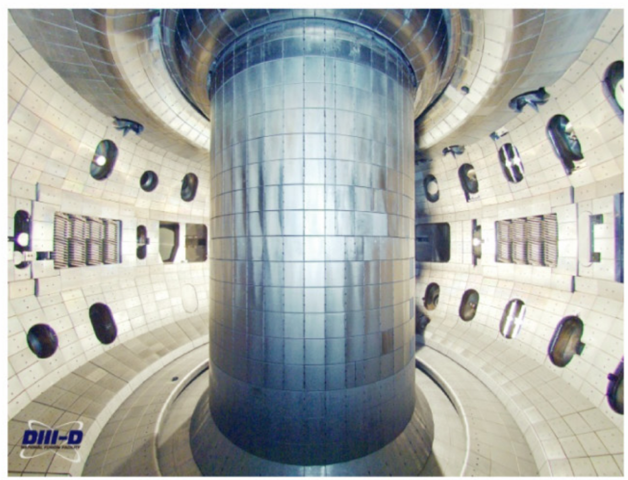
Loaded DiMES sample
2 Si-W, 3 graphite, 2 W buttons

W-buttons with 1 mm dia. indentations

2011 buttons

Shot 14261-14264

Sample exposed
To 4 LSN discharges

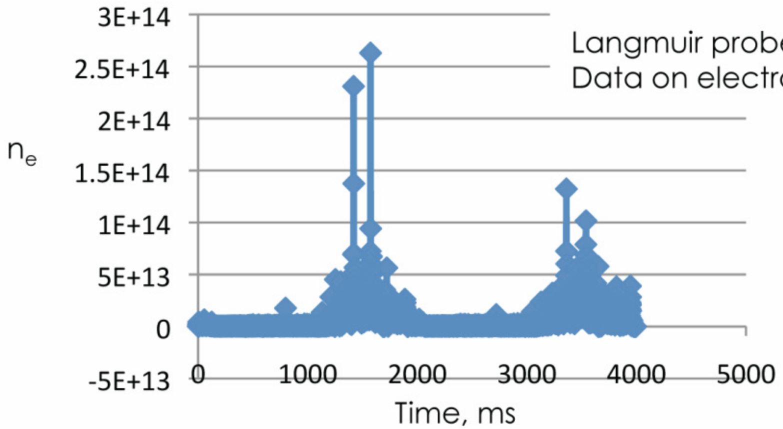


Exposed in
DIII-D lower divertor

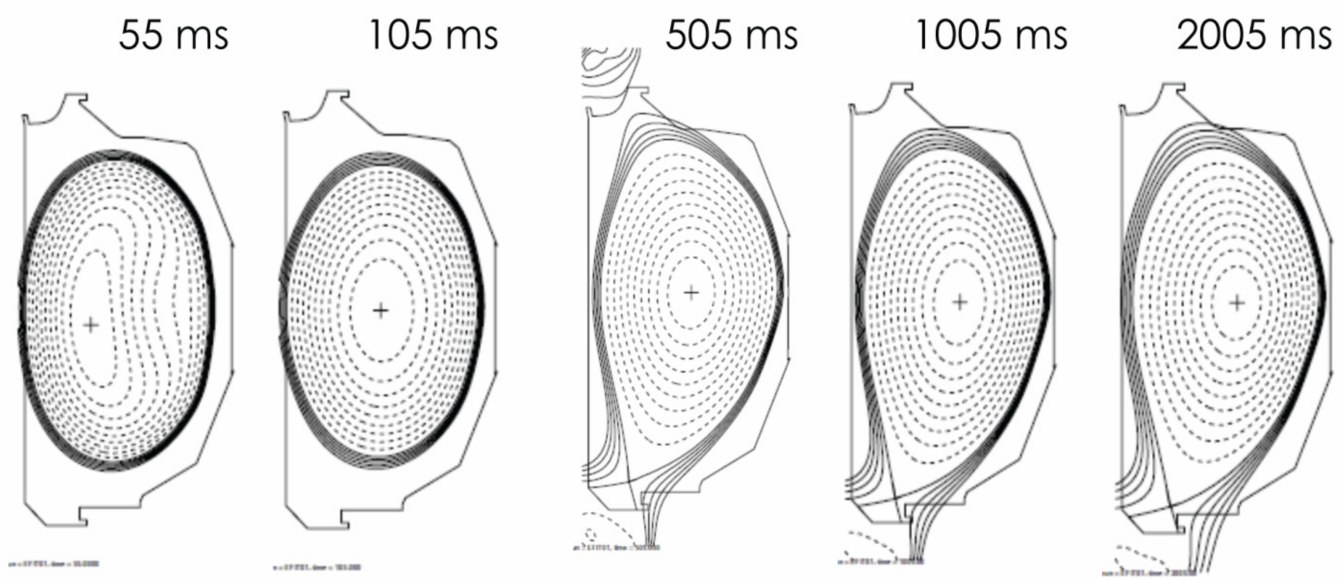
Shot 142706

After one additional disruption
shot without thermal dump on DiMES

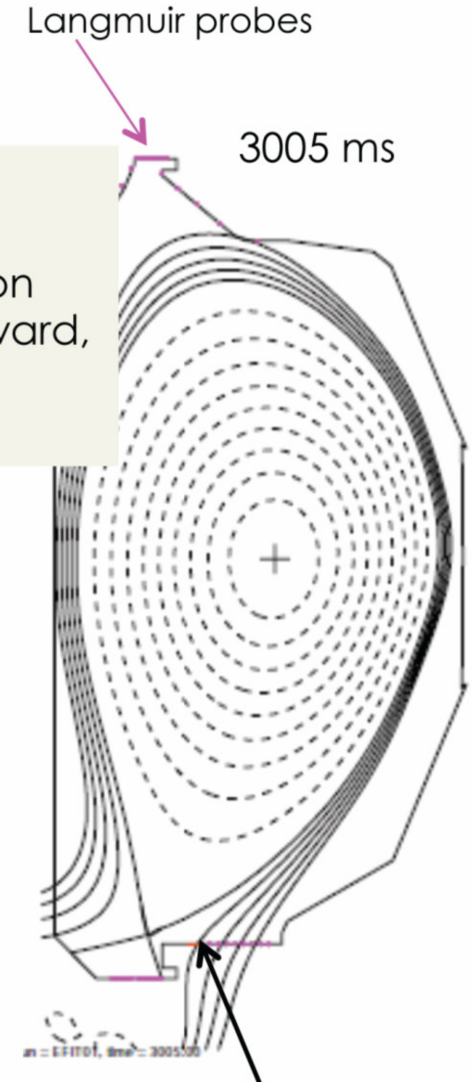
Plasma Shot #142706, with Relative Stable Plasma Shape



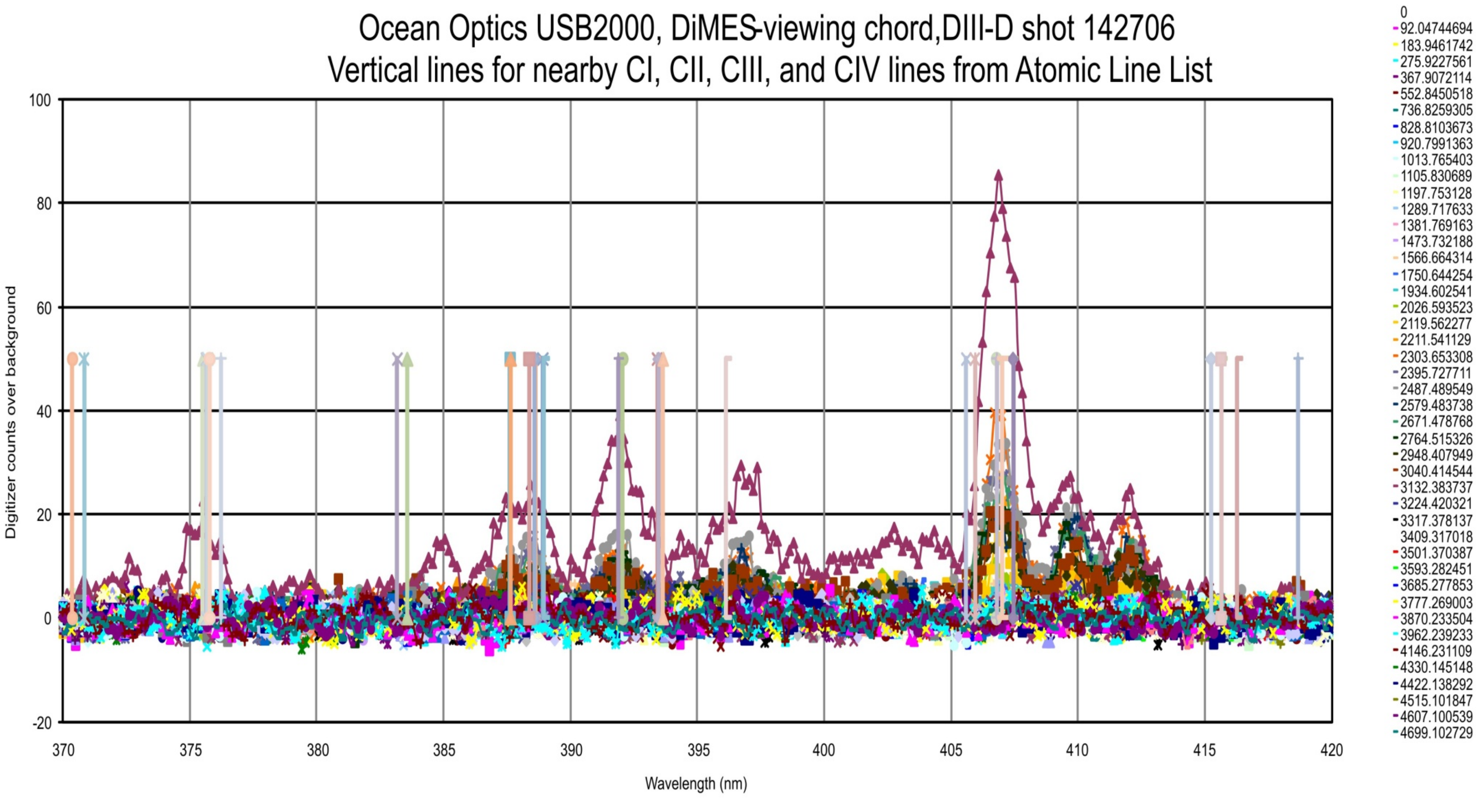
Added neutron beam injected at 2000 ms, plasma ended with disruption ~ 3100 ms but migrated upward, no thermal dump on DiMES



Plasma evolution

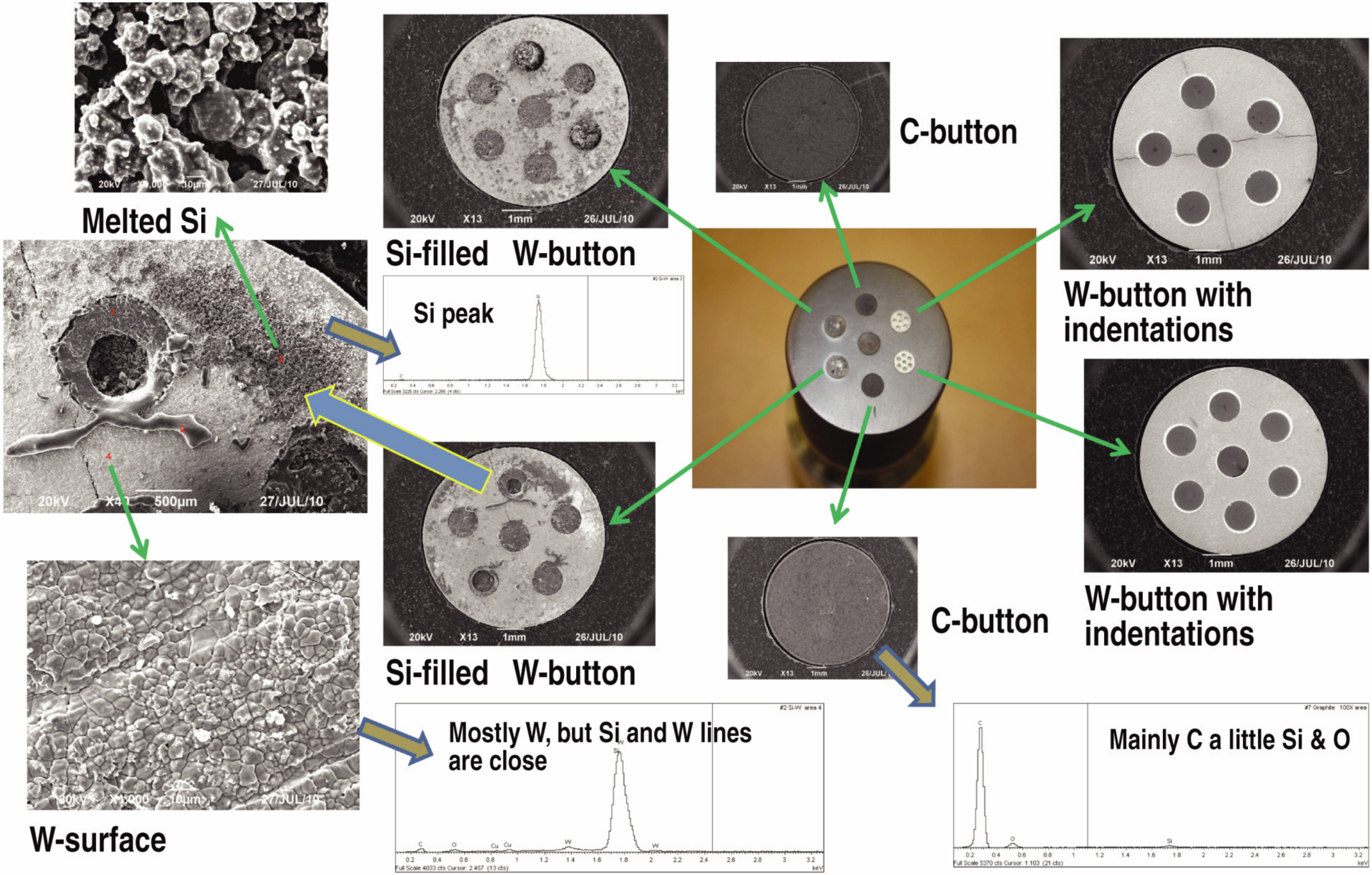


Mostly CII/CIII Emission Measured During Discharge and Disruption (387, 392, 407 nm), Additional CI (375 nm) in Disruption



Emission lines from the 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