Plasma Facing Materials Selection: A Critical Issue for Magnetic Fusion Power

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Prof. M. Sawan from the University of Wisconsin; Drs. K. Umstadter and R. Doerner of UCSD and C. Lasnier of LLNL.

Abstract

This paper proposes a possible development approach that may satisfy most of the ten requirements for plasma facing materials (PFM) in a fusion power plant. If the materials selected for ITER were to be used in an advanced DT device, C tiles and Be coatings would suffer radiation damage, and W could also suffer damage from neutrons and helium ions. Li has been proposed as a possible PFM option. However, due to vaporization and subsequent transport into the plasma core, modeling results indicate that Li will have very limited heat flux capability at the chamber wall for power reactor. Another innovative approach based on the use of low-Z loaded W surface is proposed with the loading of Si into an array of small indentations or toroidal grooves in the W to represent over 10 micron equivalent thickness, such that the thin Si-loaded W-surface can withstand an occasional disruption without melting. At the same time it is necessary to control and mitigate rapid transient events like disruptions, type-I ELMs and runaway charged particles in order to reduce the frequency of thermal dump to the chamber wall. Additional efforts will be needed to assure the uniformity of heat and particle flux distributions to the chamber wall and to maintain in real time replenishment of low-Z material onto the chamber surface for steady state operation.

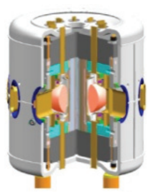
Surface Material is a Key Item for Fusion Development

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

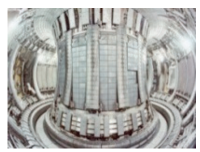


C-Mod

Mo

AUG

W

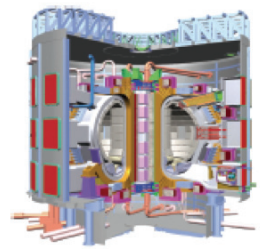


JET-ILW

Be/W/C

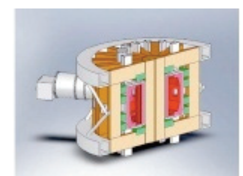
EAST

C/W



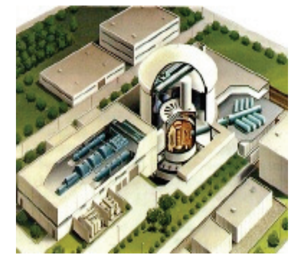
ITER

Be/W/C



CTF/FNSF/FDF

?



DEMO

?

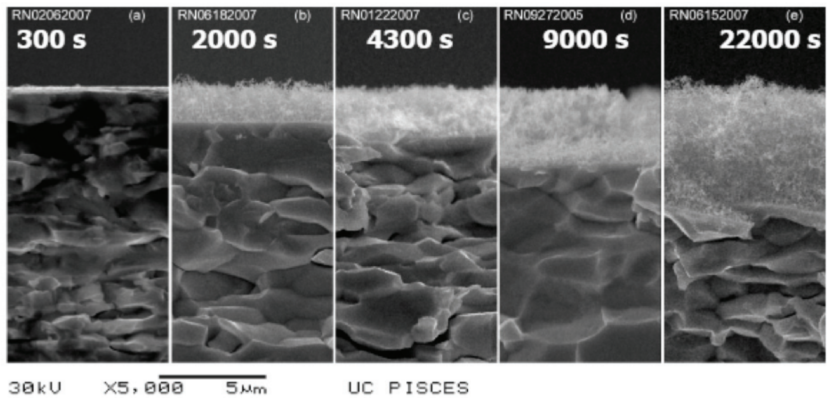
(High neutron and edge alpha fluence)

← **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 feasibility issues have been identified

Significant Issues Projected for W-surface Operation

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



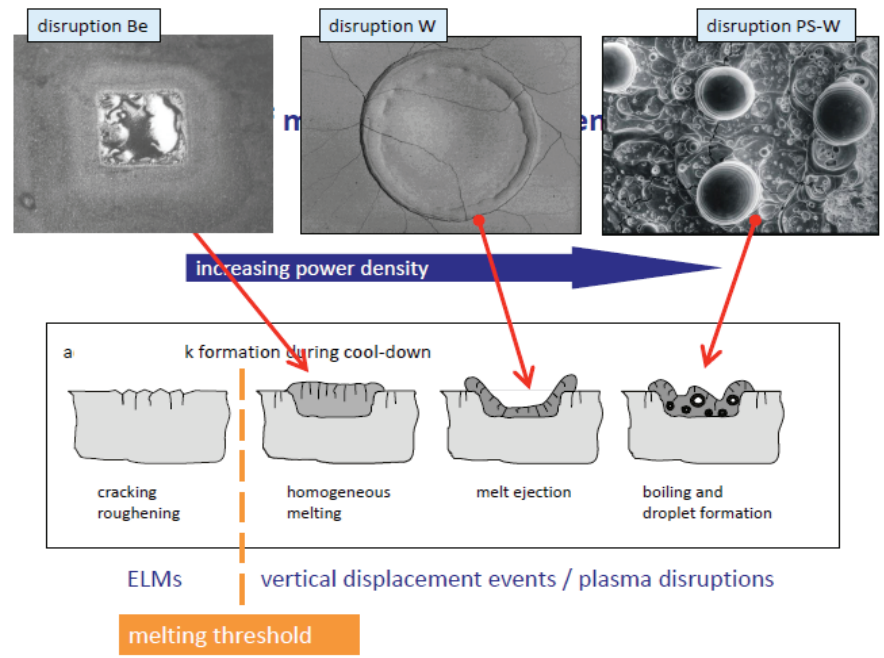
Consistent He plasma exposures: T = 1120 K, Γ_{He+} = 4–6 $\times 10^{22}$ m^{-2}s^{-1}, E_{ion} ~ 60 eV

PISCES

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

Baldwin and Doerner, Nuclear Fusion <u>48</u> (2008) 1-5

ITER disruption loading:
10-30 MJ/m^2 for 0.1 to 3 ms



Irreversible surface material damage

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

We cannot eliminate un-predicted disruptions even if disruption detection and mitigation work perfectly

Damage to W First Wall has also been Projected

Low energy He^+ irradiation in plasma simulator NAGDIS
H bubble and hole formation on W surface at >10 eV

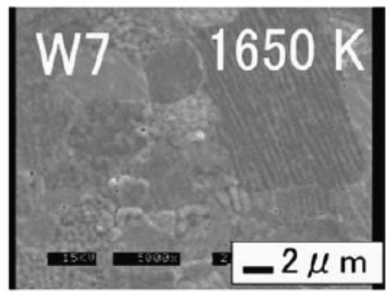
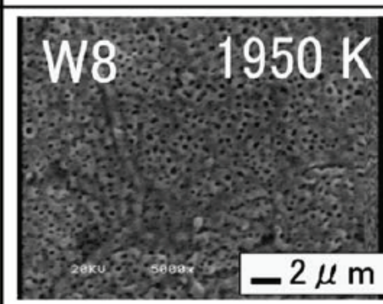
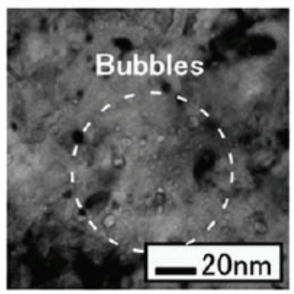
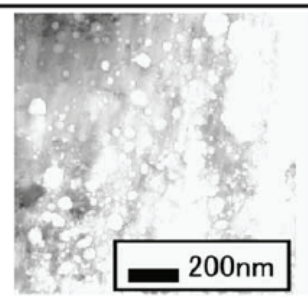
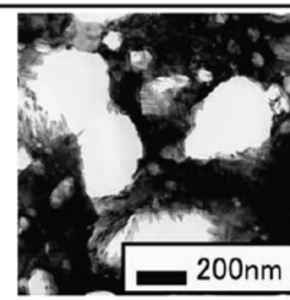
Fluence Time / Ion flux Temp	3.5×10^{26} /m² 3600 s / 1.0×10^{23} /m²s ~20 eV	1.8×10^{26} /m² 1800 s / 1.0×10^{23} /m²s ~25 eV	1.7×10^{26} /m² 660 s / 2.6×10^{23} /m²s ~25 eV
SEM			
TEM			
Bubble size	< 5 nm	< 200 nm	< 500 nm

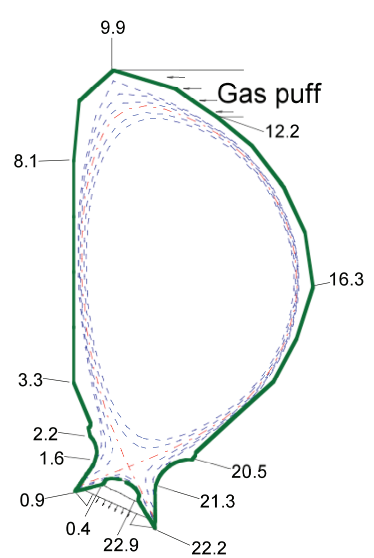
Fig. 3. Temperature dependence of bubble and hole formations on PM-W surfaces at different temperatures; W6: 1300 K, W7: 1650 K, W8: 1950 K. Bubbles or holes clearly appear, and the size of bubble becomes drastically larger with the temperature above the recrystallization temperature of W (1400–1500 K).

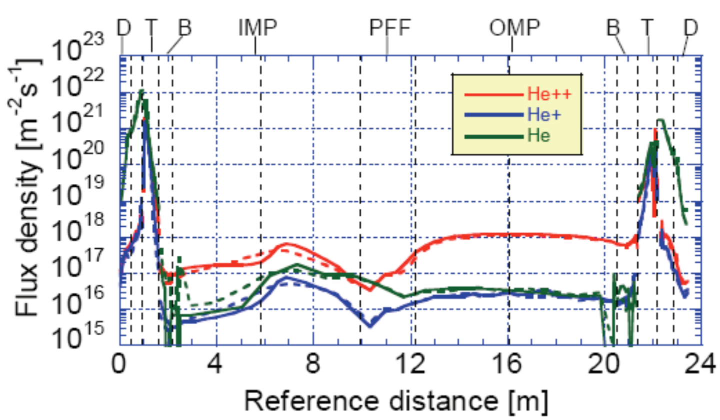
D. Nishijima, Journal of Nuclear Materials 329 (2007) 1029-1033

He^+ Irradiation of W Sample can Simulate DEMO Parameters; DEMO and ITER He^+ Flux are Comparable

	Simulating DEMO first wall	Simulating DEMO Divertor
Flux, #/m^2.s	10^{18}-10^{19}	~5x10^{22}
Fluence, #/m^2	10^{21}-10^{22}	3.5x10^{27}- 2x10^{28}
He^+ energy	200 eV to 8 KeV	10 eV to 80 eV
Temperature	100-1000° C	~1000° C

Results from ion beam, plasma devices, TRIAM-1M and LHD discharges





Kukushkin, ITER D 27TKC6, 14/04/08

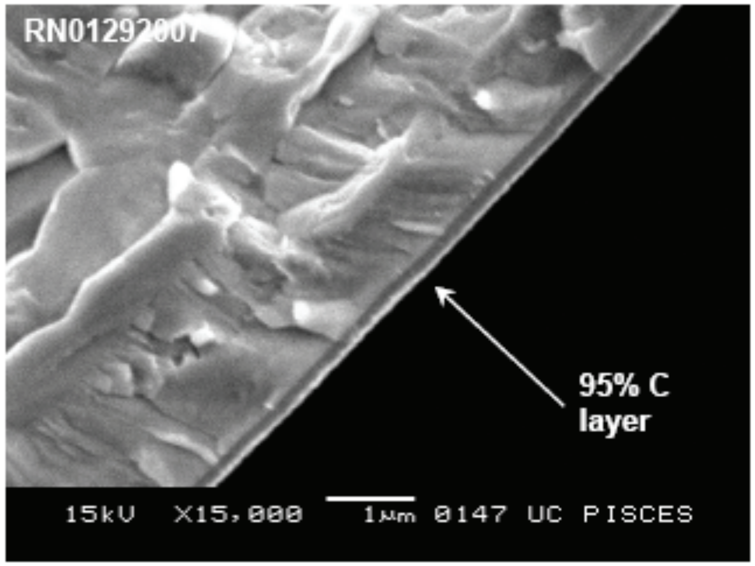
First wall: Flux 10^{17}-10^{18} #/m^2.s

Divertor: Flux ~10^{21} #/m^2.s

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

E_i = 15 eV, T_s = 1100 K, Fluence = 10^{25} He+/m^2,
$n_{He+}/n_e \sim 10$ %, $n_{C+}/n_e < 0.1$ % $\Delta t = 3600$ 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 has not been sputtered away

- At **E_i = 15 eV**, **C deposited on W are not sputtered away**.
  - ⇨ **W-C layers inhibit He induced morphology**.

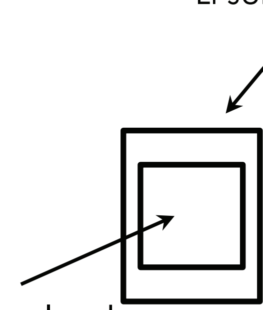
PISCES

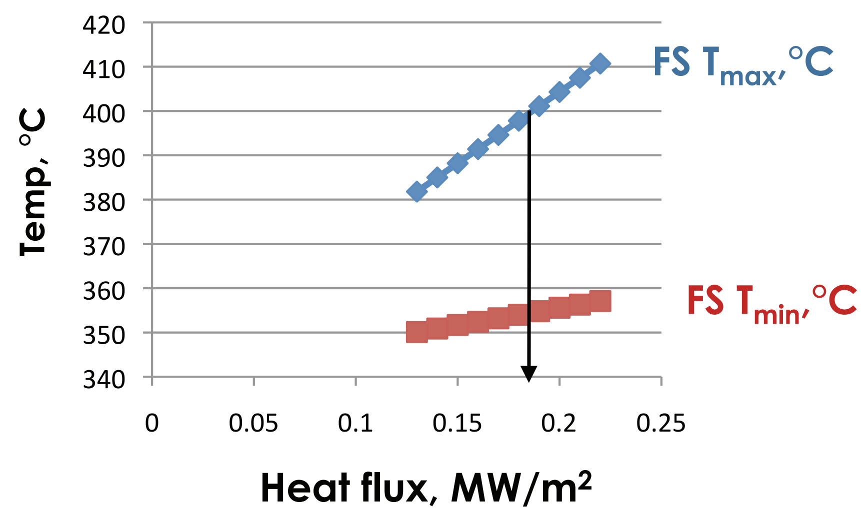
UCSD Center for Energy Research

Baldwin and Doerner, PISCES, UCSD

How about Li-wall? A Very Optimistic Estimate of First Wall Heat Removal Shows the Heat Flux Limit of Li Surface

- **Assumptions**
  - Structural material – ferritic steel, k_{th} at 20 W/m.k
  - Helium coolant at 8 Mpa
  - He Tin at 340°C
  - Coolant velocity at 100 m/s
  - Square channel geometry, width = 2 cm
  - Wall thickness = 4 mm
  - Channel length = 1 m
  - Film drop enhancement = 1.8 via twisted tape
  - Li thickness = 0.0
- **Limitations**
  - Ferritic steel T_{min}> 350°C, limited by radiation hardening
  - Li surface T_{max}< 400°C , limited by vaporization into the plasma core

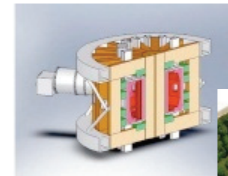


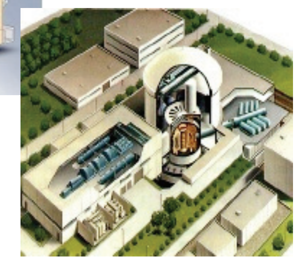


T. Rognlien et al., J. of Nuclear Mater. <u>290-293</u> (2001) 312-316

Result:
- **Li surface at the first wall is limited to <0.2 MW/m², due to the vaporization of Li unless it is demonstrated experimentally that Li will not transport into the plasma core**

Plasma Facing Material Design and Selection Requirements for Next Generation Devices





1. Withstand damage from DT generated He
2. Withstand transient events like ELMs and disruption

Additional critical requirements

Physics performance

3. Material suitable for high performance plasma operation
4. Suitable for edge radiation to reduce maximum heat flux at the divertor
5. Low physical and chemical erosion rate

Engineering performance

6. Transmit high heat flux for high thermal efficiency conversion
7. Minimum tritium inventory
8. Minimum negative effect to tritium breeding performance
9. Low activation materials
10. Replenish damaged surface material suitable for steady state operation and long lifetime
11. Match materials temperature design requirements
12. Withstand high neutron fluence at high temperature

A Possible PFM Concept that could Satisfy all Requirements

W-buttons

W-buttons filled with Si

The concept: Si-filled W-surface (3,4,9)

- Protect the W surface from He damage with the presence of Si (1)
- Exposed W will have a low erosion rate (5)
- 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 (6,8,11)
- 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" (2)
- Should be able to control tritium inventory at temperature ~1000°C (7)
- Suitable real time siliconization can be used to replenish Si when and where needed (10)
  (Satisfying requirements from the last VG, #12 TBD)

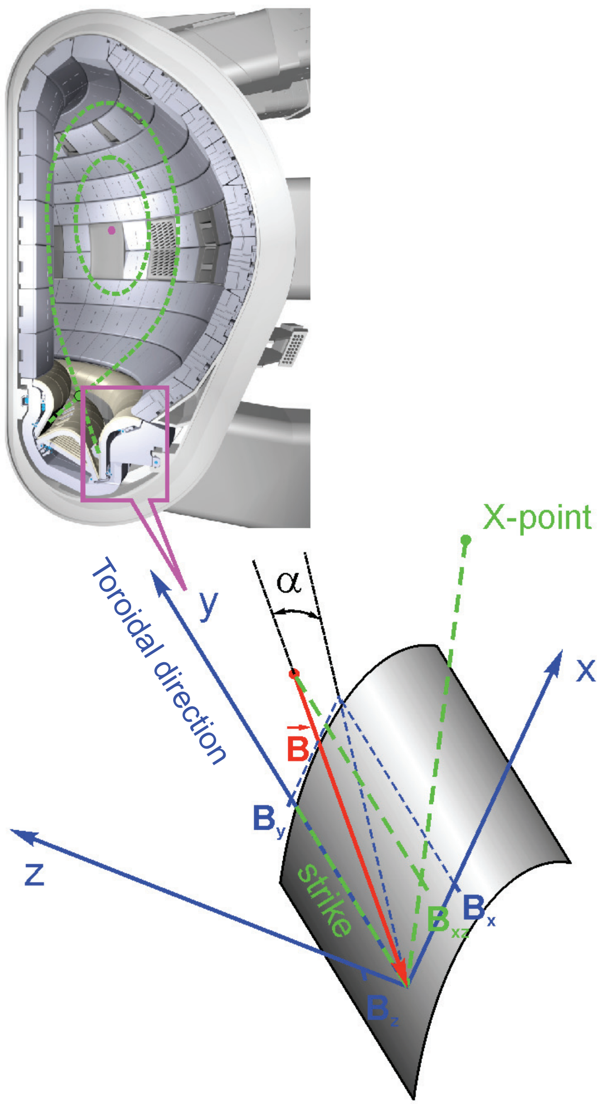
Vapor Shielding Effect

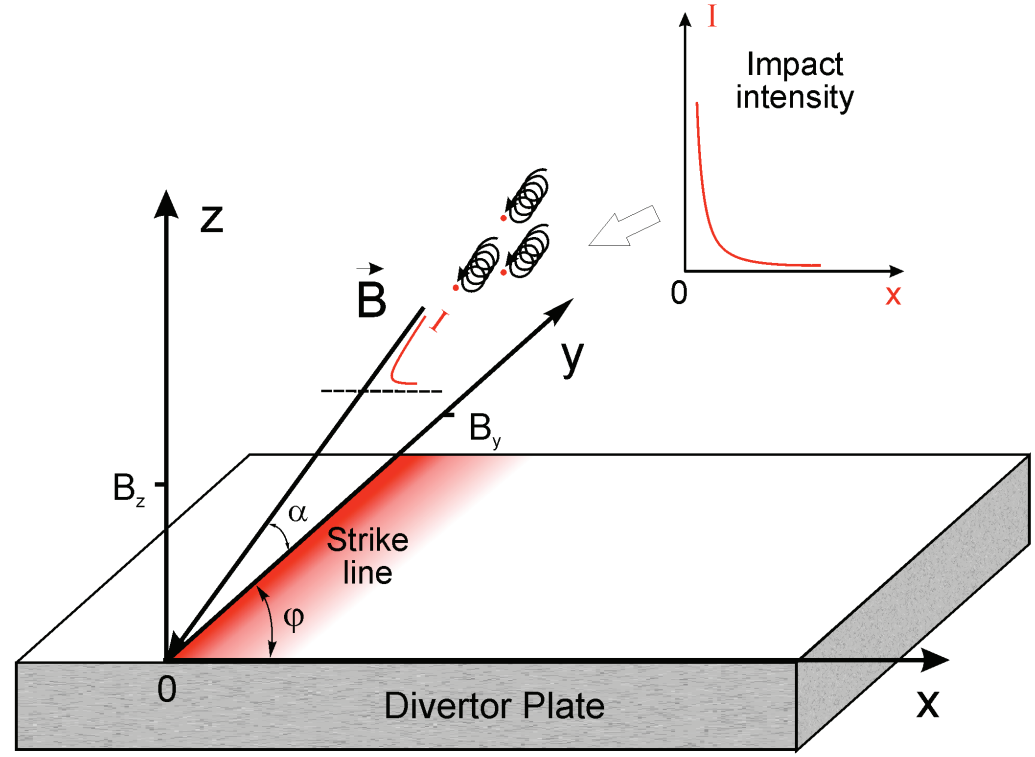
Selected Literature

V. Litunovsky, et al., **Experimental Study of Shielding Layer Plasma Radiation At High Power-Material Interaction**, J. Plasma Fusion Res. Series **2** (1999) 324-327

V. Safronov et al., **Material Erosion and Erosion Products under Plasma Heat Loads Typical for ITER Hard Disruptions**, J. of Nucl. Mater. **290-293** (2001) 1052-1058

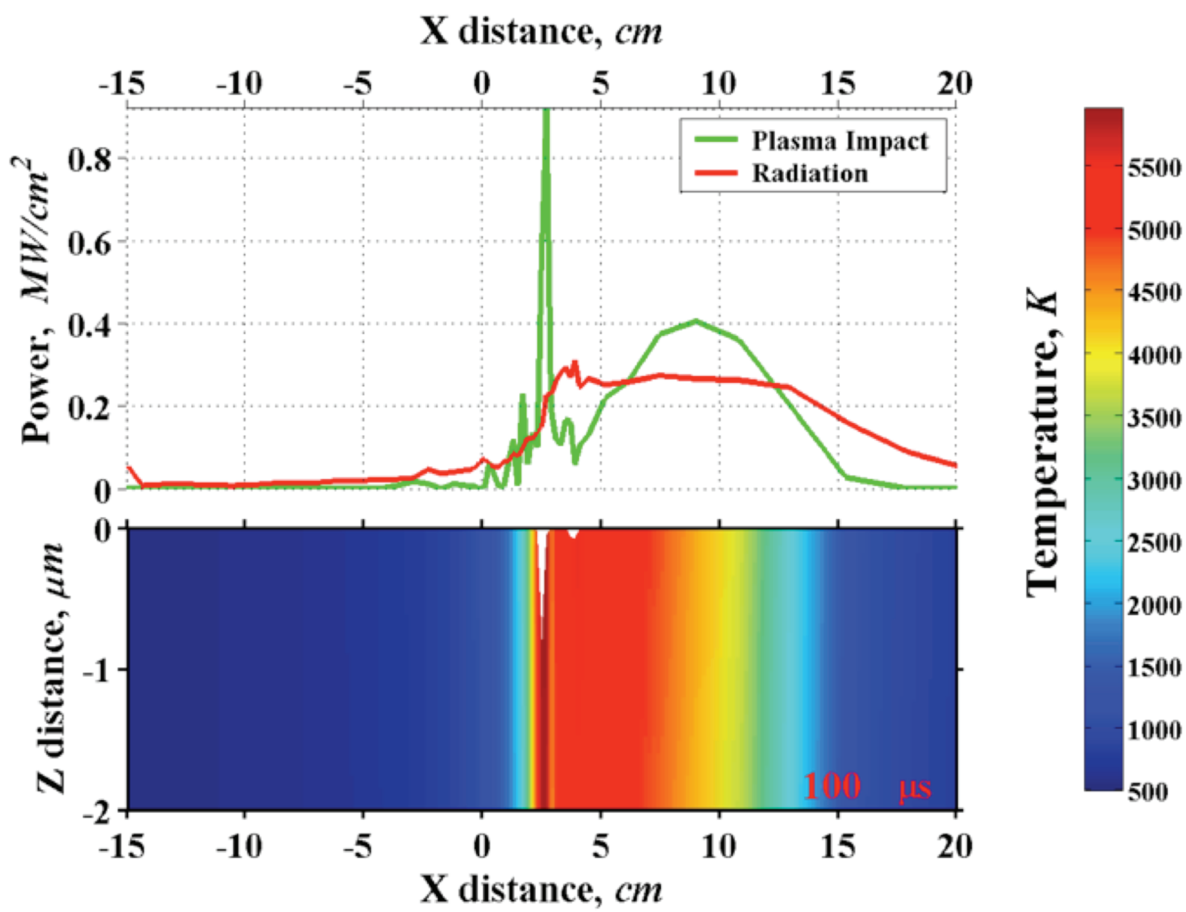
Modeling Geometry



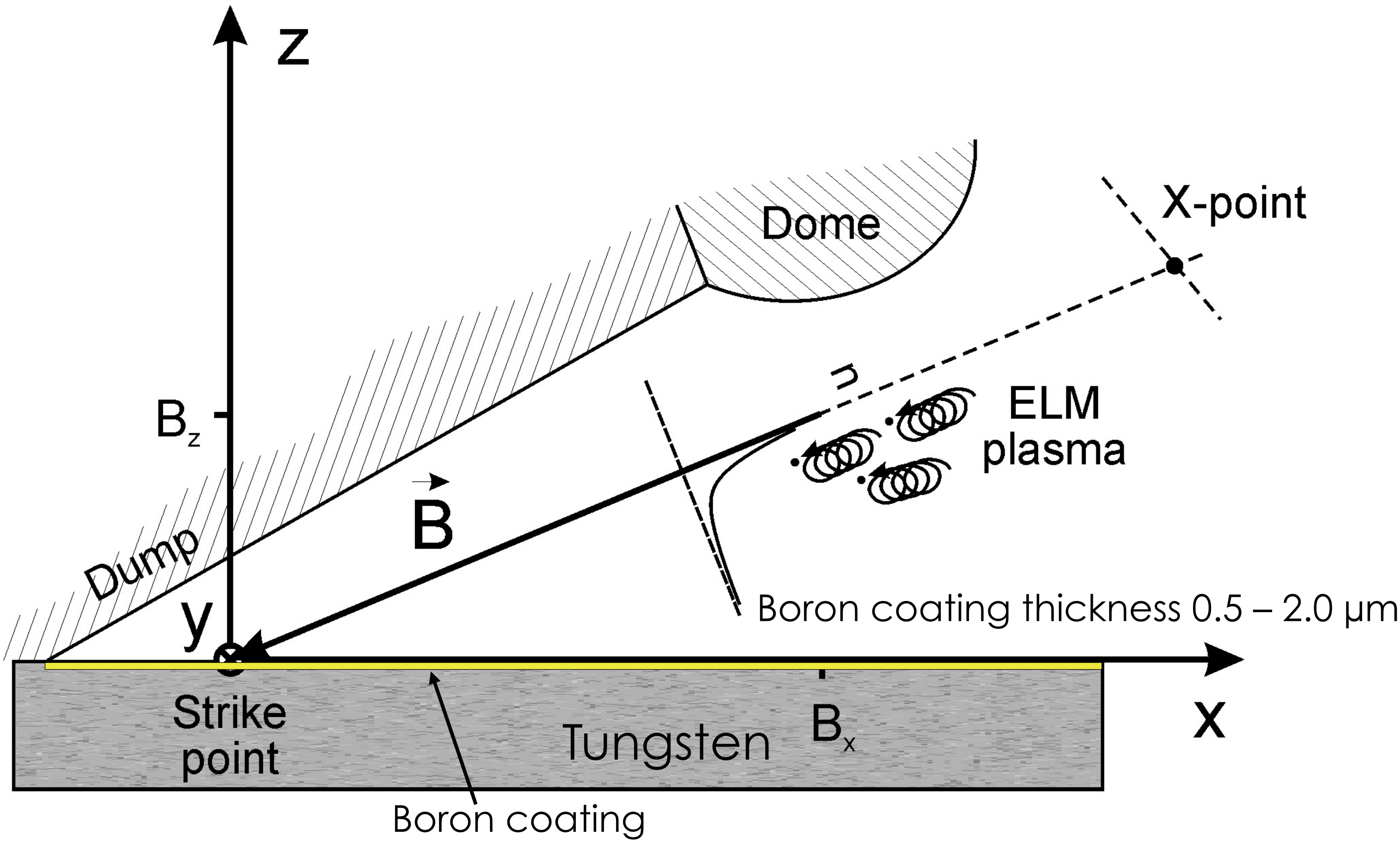


Tungsten Divertor Erosion & Surface Fluxes

Energy density $E = 25\ MJ/m^2$
Impact duration $t = 0.1$ ms
Magnetic field $B = 5.0$ T
Incline angle $\alpha = 5.0$ deg



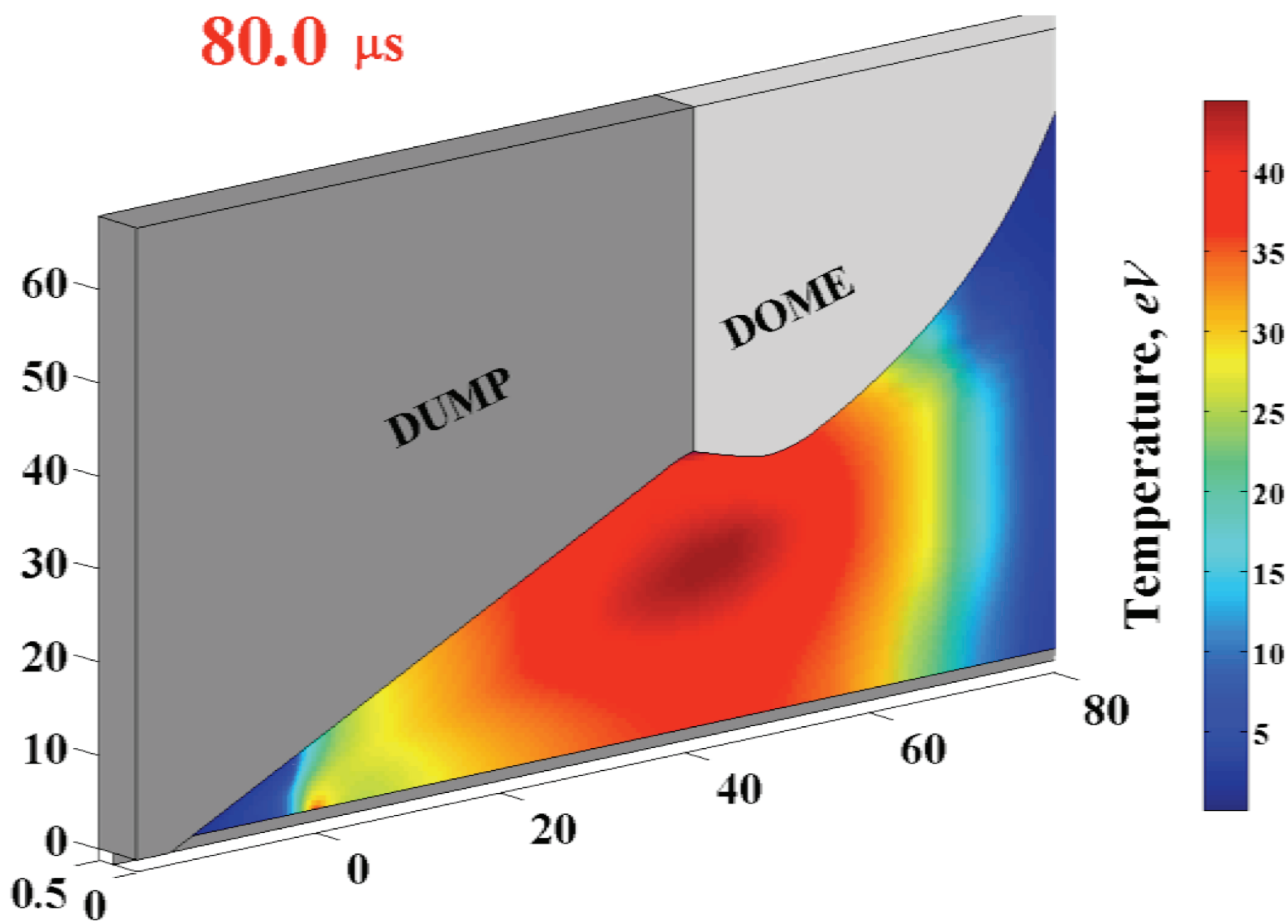
Boron Coating on Tungsten Surface



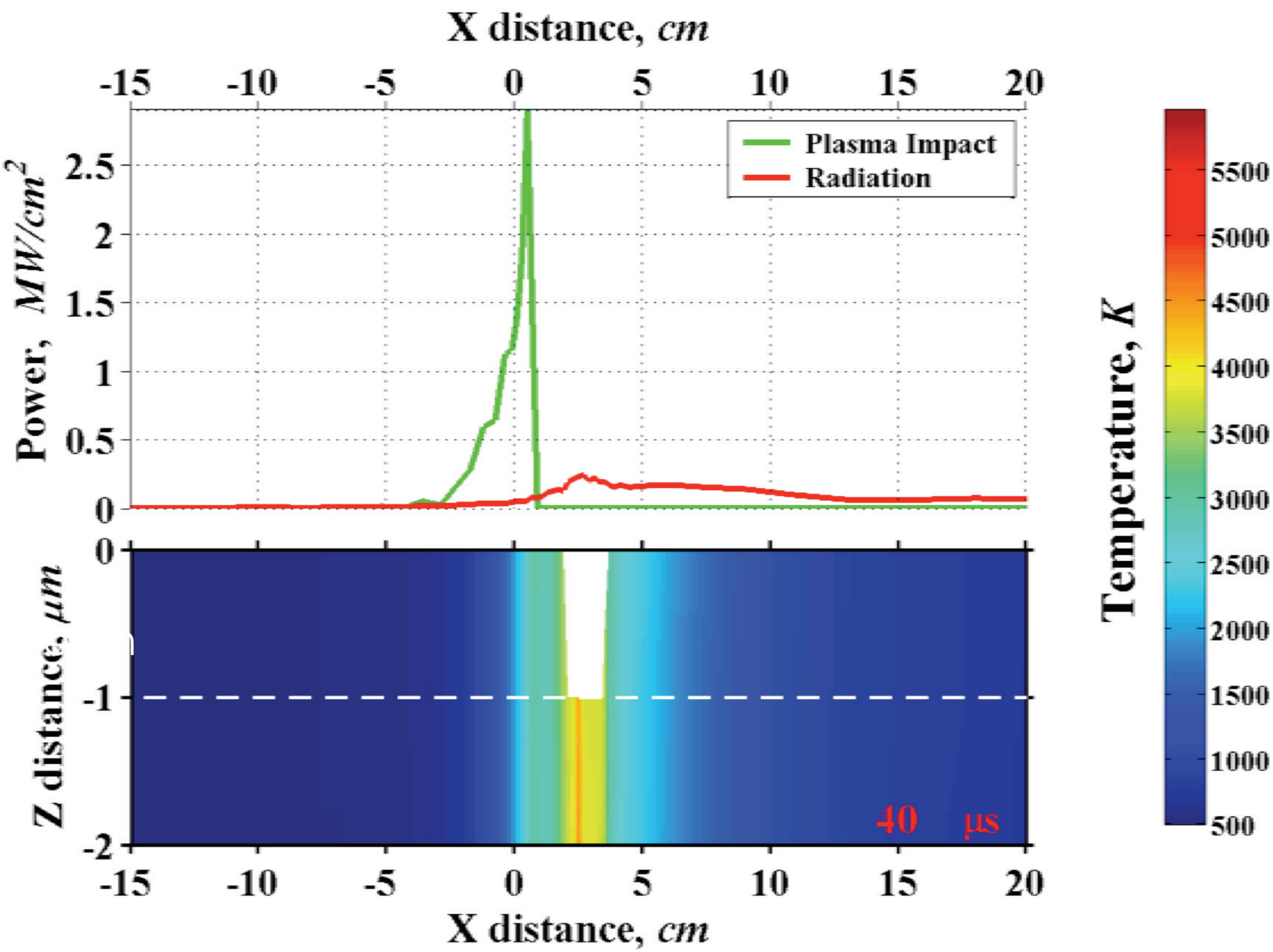
Boron Atomic Data and Plasma Properties

- Self-consistent Hartree-Fock-Slater (HFS) method is used to calculate atomic data
- Collisional-radiative equilibrium (CRE) model is used to calculate the populations of atomic levels and ion and electron plasma concentrations
- Total absorption and emission coefficients are the sum of contributions from bound-bound, bound-free and free-free transitions
- Calculated absorption and emission spectra represent the detailed spectra with 99,998 spectral points in the range of vapor cloud temperatures
- Detailed spectra including all 99,998 spectral points as well as spectral group averaging (~4,000 groups) were used in the Monte Carlo calculations of the radiation transport
- Absorption and emission coefficients of the boron plasma were calculated over a wide density and temperature range: from 10^{14} up to 10^{21} cm^{-3} and from 0.05 up to 250 eV

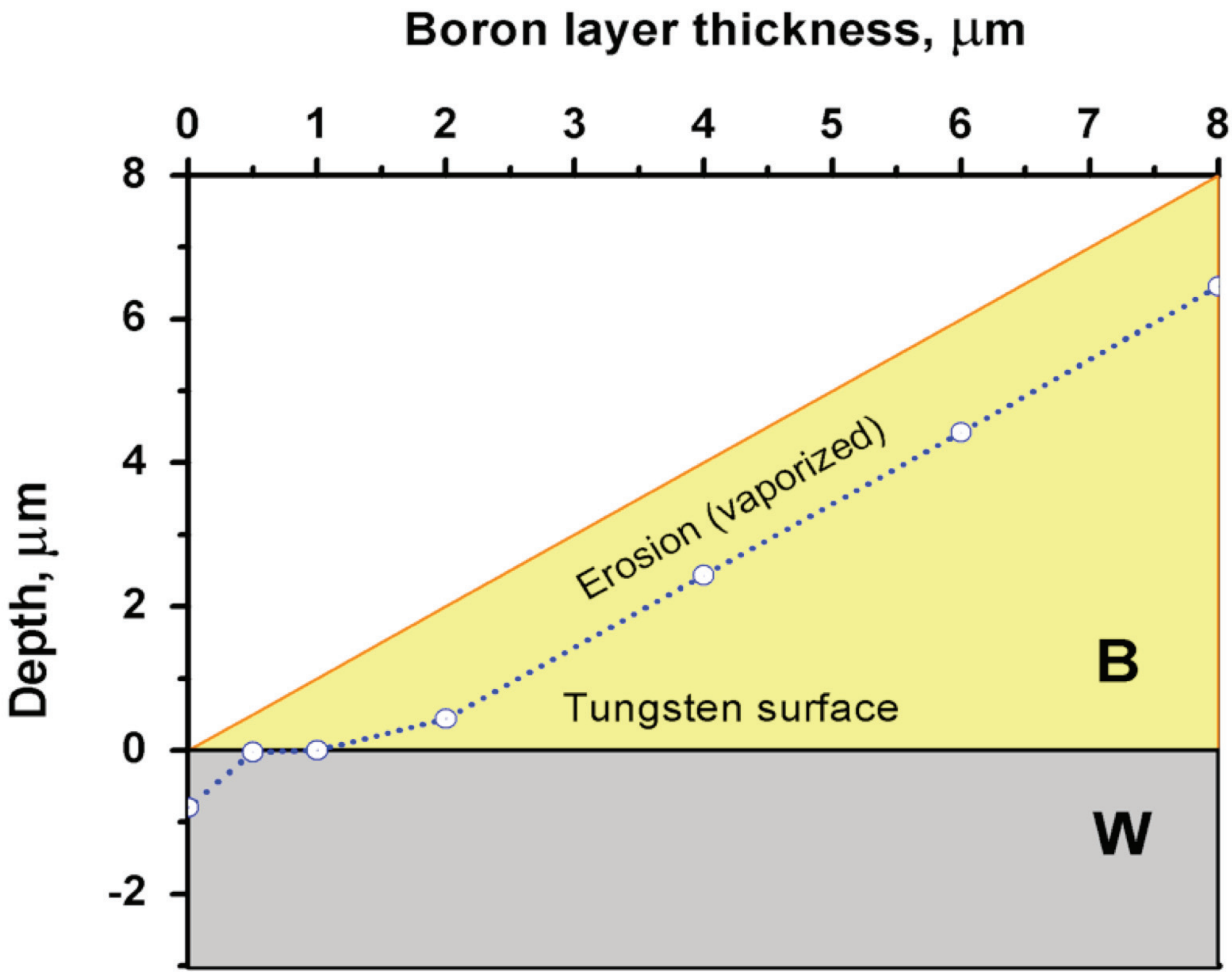
Plasma Temperature Evolution



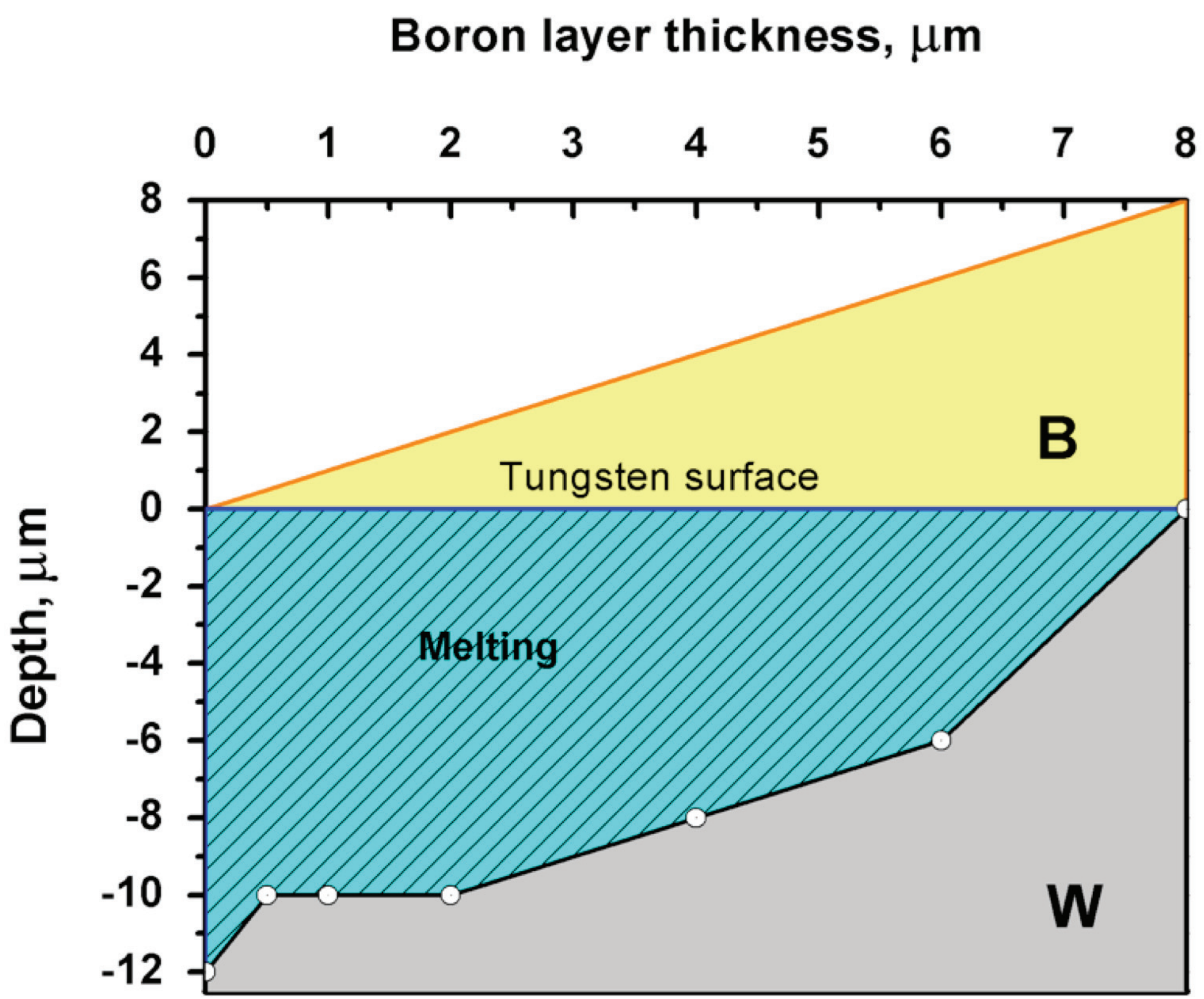
Boron Plasma Shielding



Divertor Surface Erosion



Depth of Tungsten Melting under Boron Layer



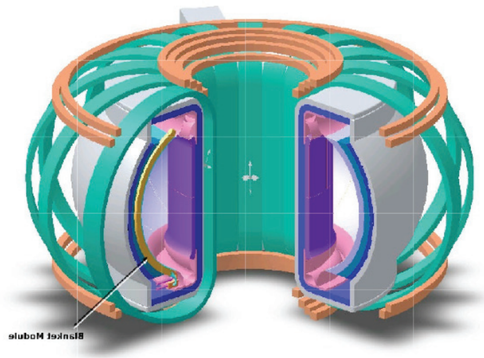
Summary

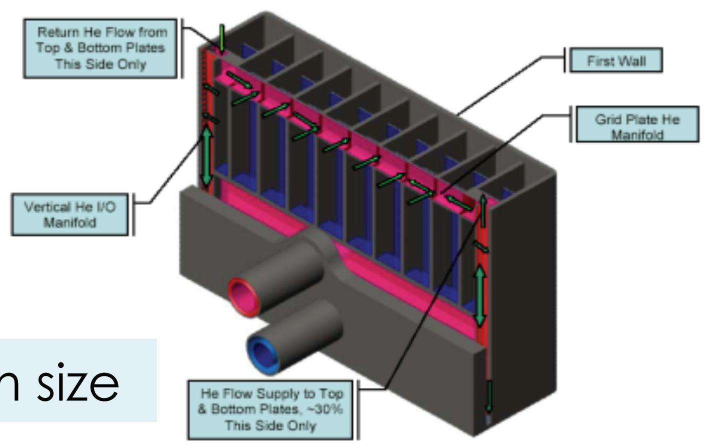
- Low-Z coatings, if replenishable, may protect tungsten divertor plate from disruption and ELMs
- Coating thicknesses in the range of several microns are adequate to mitigate the damage from disruption effects
- Need to optimize coating thickness, coating material (e.g. boron vs silicon, etc), ways to recoat, etc before this concept becomes realistic
- Similar analysis could be performed for ELMing and radial transport heat flux (blobs) to the first wall
- Si could be more effective than B in protecting the W surface, but melt layer losses and splashing from hydrodynamic instabilities and overheating could result in higher erosion of the coated layer

Actively helium cooled first wall and divertor design of next generation tokamak devices

Conceptual He-Cooled Blanket Design has been Assessed for DEMO

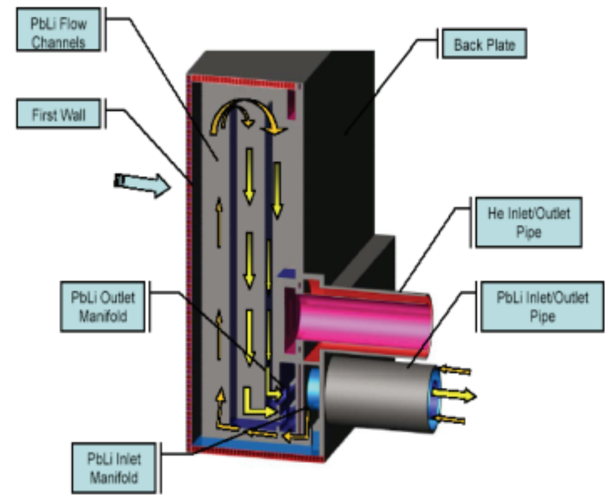
Neutron wall loading: 3 MW/m^2,
FW Surface Loading: 0.5-1 MW/m^2

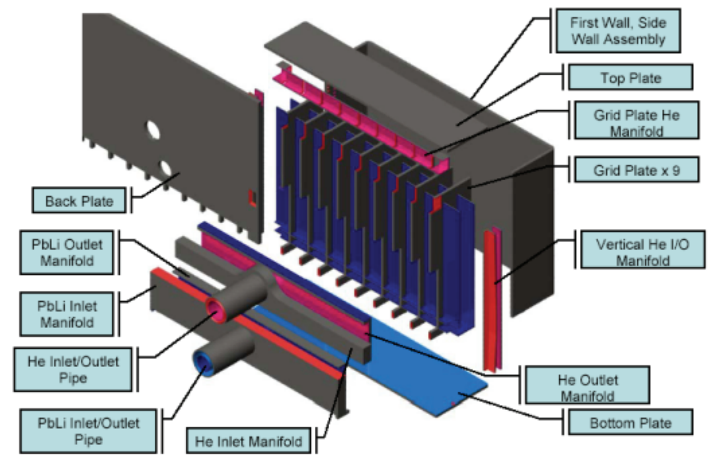




Module size in m size

DCLL concept



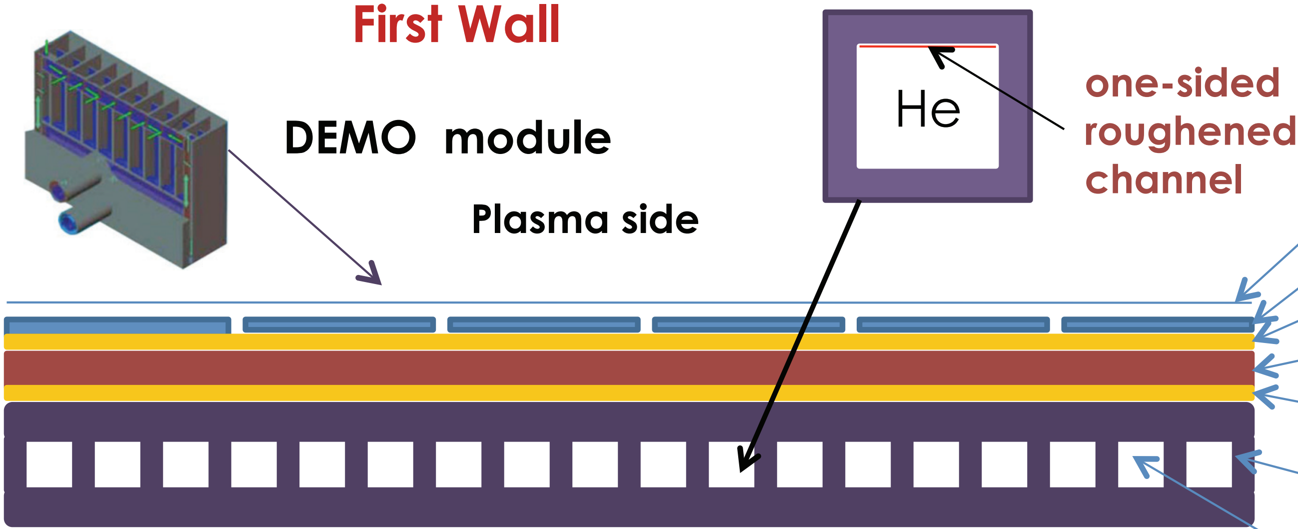


Ferritic steel is the structural material:

Nuclear performance, FW helium cooling, waste disposal, structural design, safety impacts including LOCA, power conversion with CCGT have been assessed.

Projected DEMO PFC FW and Divertor Design Approaches

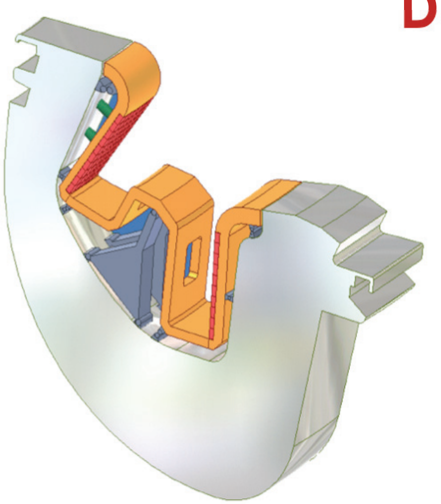
First Wall

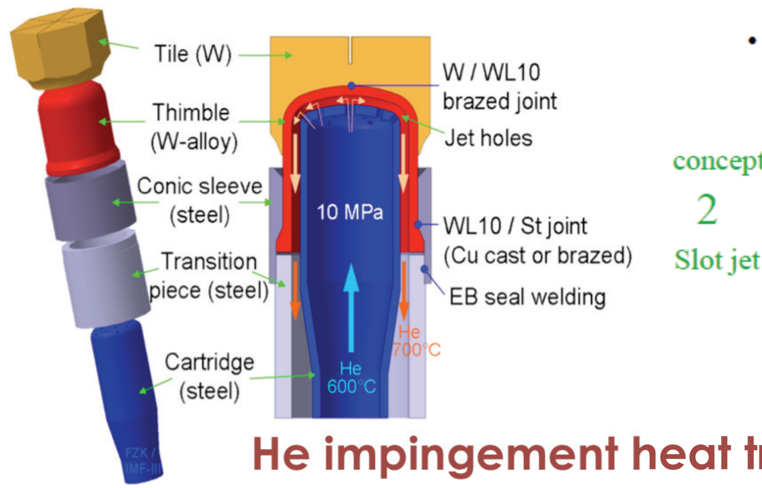


Temperatures & limits

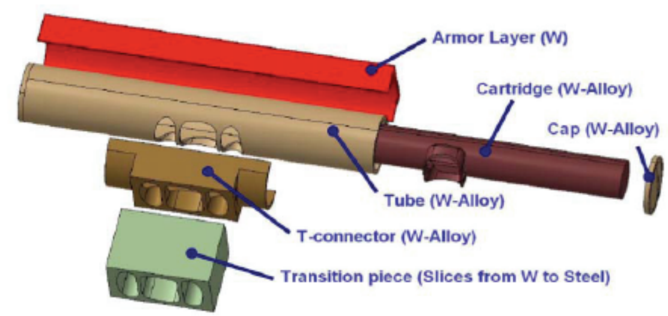
Tokamak coating?
W-layer >700°C
Joint (TBD)
ODFS <700°C
Joint (TBD)
RAFM >350°C
<550°C
8 MPa at ~350°C

Divertor



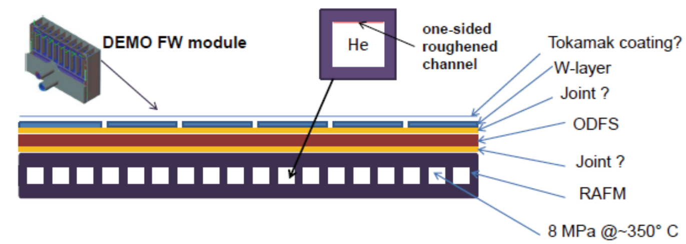


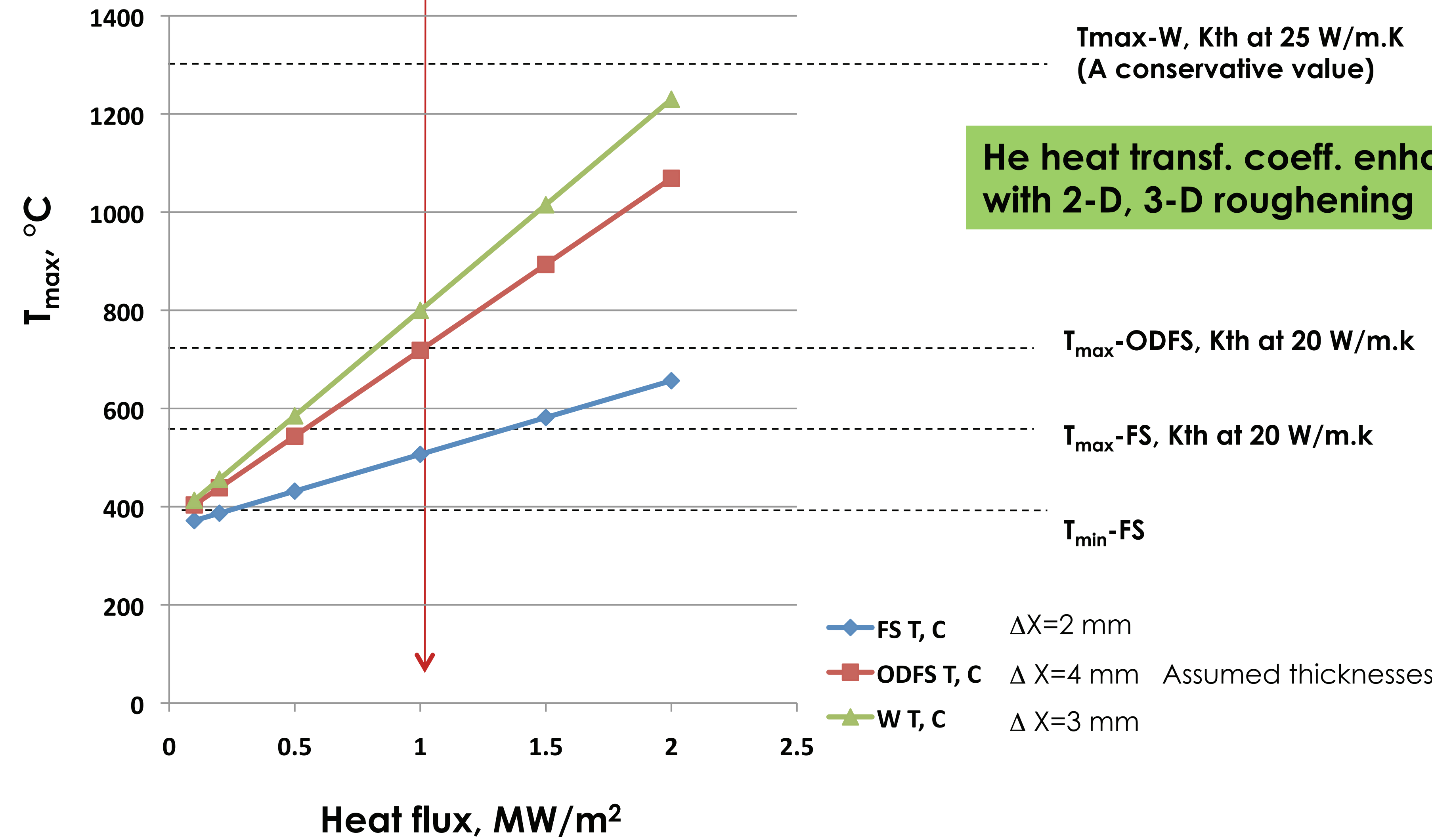
• ARIES CS design



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

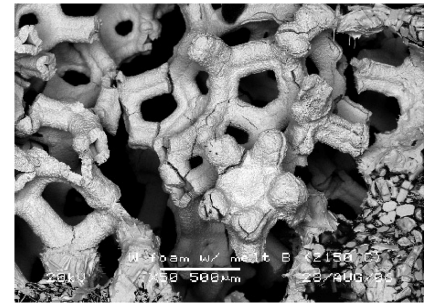
Development of the Si-W surface, potentially a transient tolerant design

We started with W-mesh, then indentations and could become toroidal grooves

Si W Surface Progress

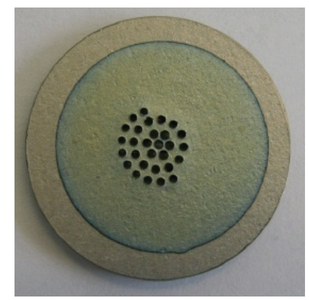
- 2008: started with BW-mesh , but the presence of C formed B_4C, WB, W_2B, W_2B_5, WC, and W_2C, thus broke up the mesh
- 2009 changed from mesh to plate, but B fill fell off the holes
- Switched to Si due to much better match in the coeff. of thermal expansion between Si and W
- High melting temperature of Si can form low melting point W-Si chemicals
- DIII-D boronization confirmed B coating thickness of 0.75 μm < 1 μm coated
- 2010: Drilled indentations on W-button and the Si was filled 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

Si-W Buttons and Sample before Exposure

W-buttons with 1mm diameter and
1 mm deep indentation

W-buttons filled with Si

DiMES button sample module
front face 4.78 cm in diameter

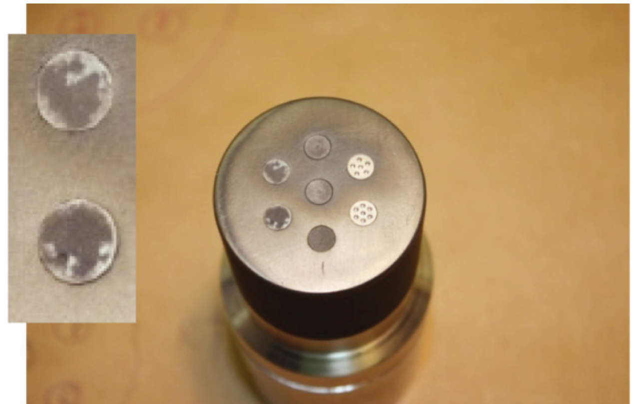
From left to right:

2 Si-filled W-buttons
3 graphite buttons
2 W-buttons with indentations

W-buttons from UCSD
Via K. Kumstadter

Initial Results of Transient Tolerant Si-filled W-buttons

Si filled W-buttons

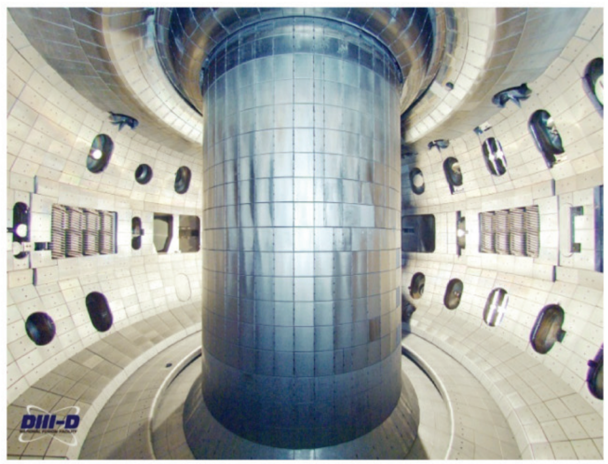


Loaded DiMES sample
2 Si-W, 3 graphite, 2 W buttons

W-buttons with
1 mm dia. indents

Shot 14261-14264

Sample exposed
To 4 LSN discharges

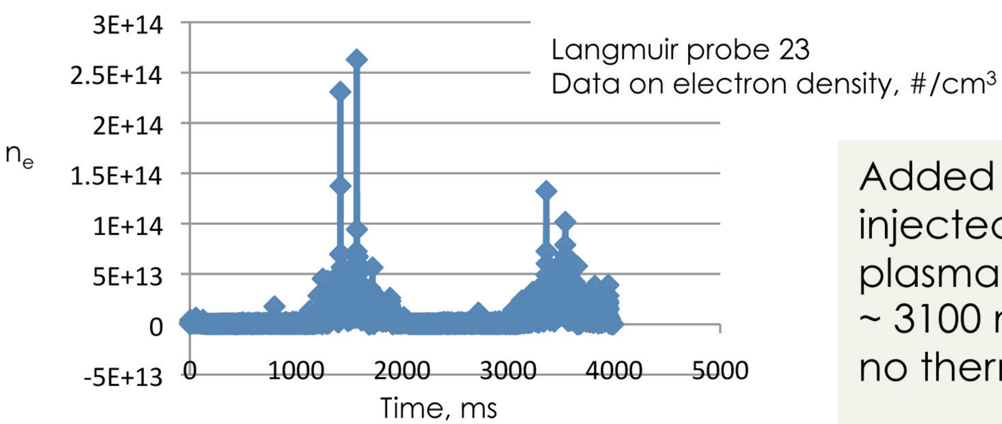


Exposed in
DIII-D lower divertor

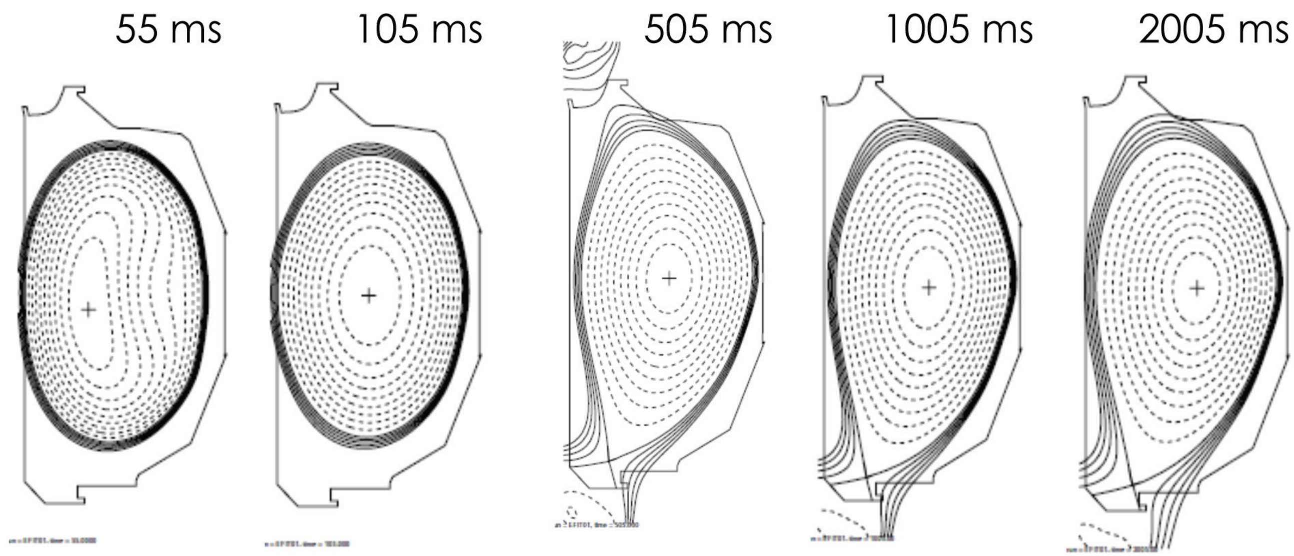
Shot 142706

After one additional disruption
shot without thermal dump on DiMES

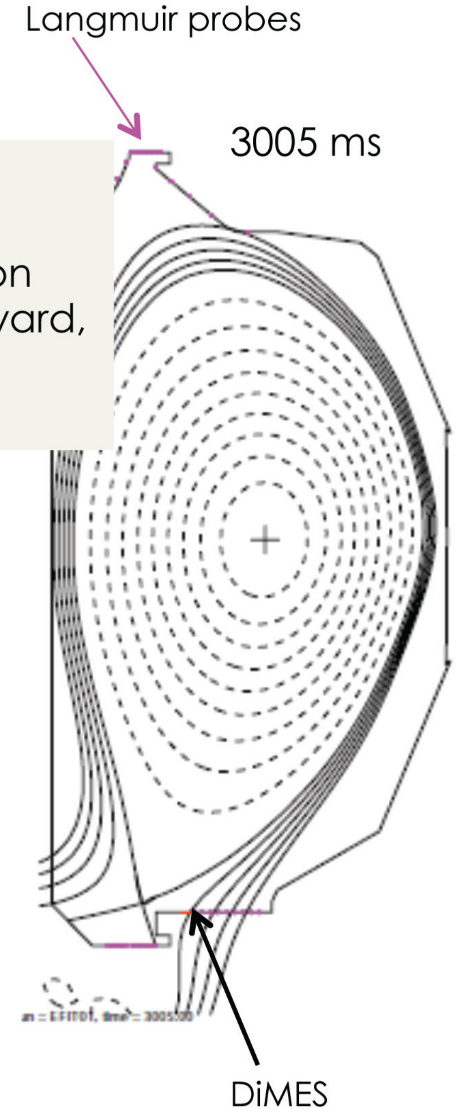
Plasma Shot #142706, with Relative Stable Plasma Shape



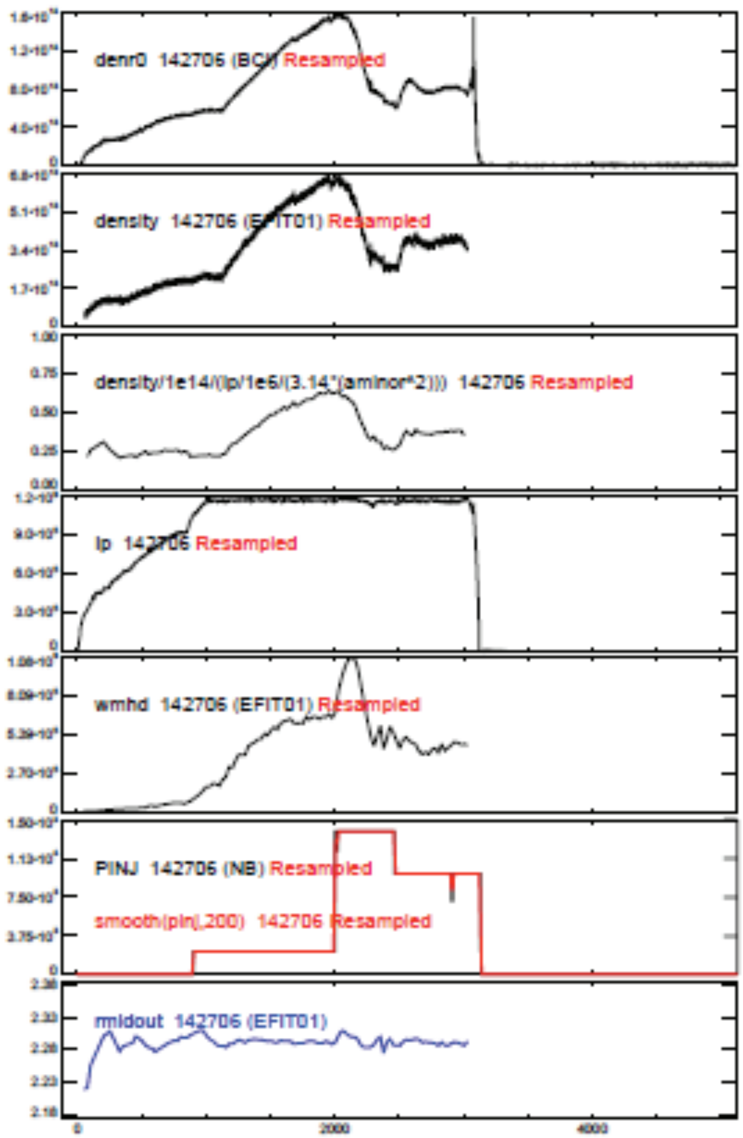
Added neutron beam injected at 2000 ms, plasma ended with disruption ~ 3100 ms but migrated upward, no thermal dump on DiMES



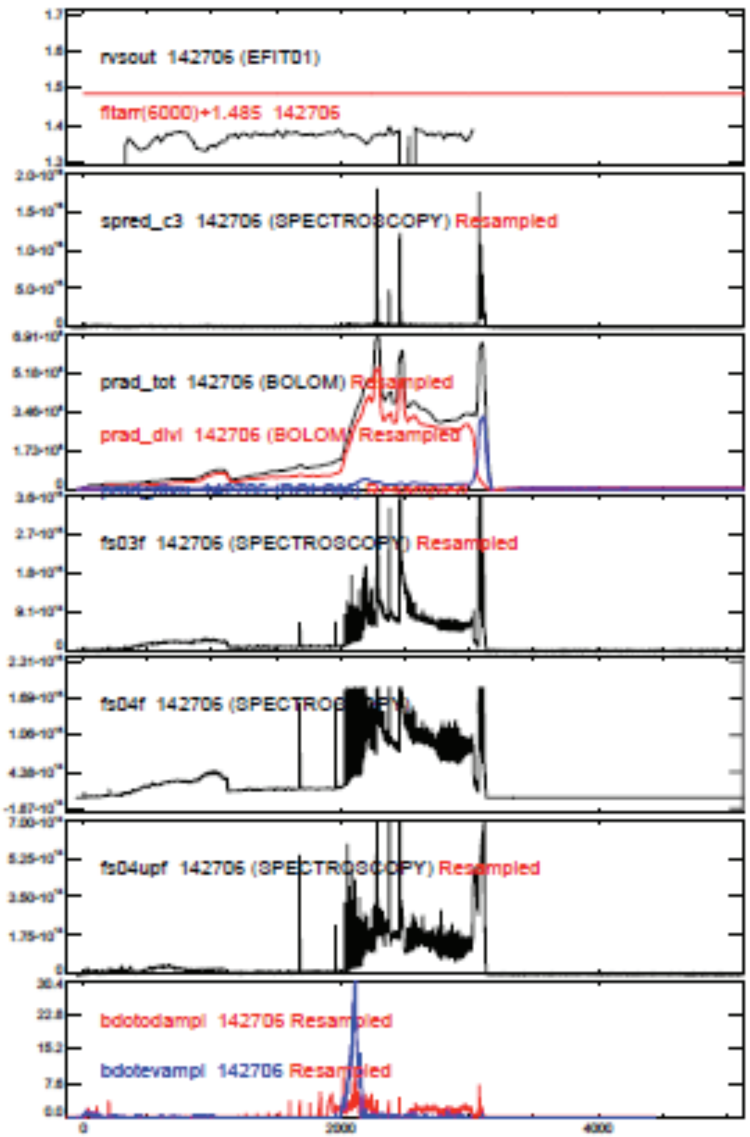
Plasma evolution



Discharge # 142706



NBI



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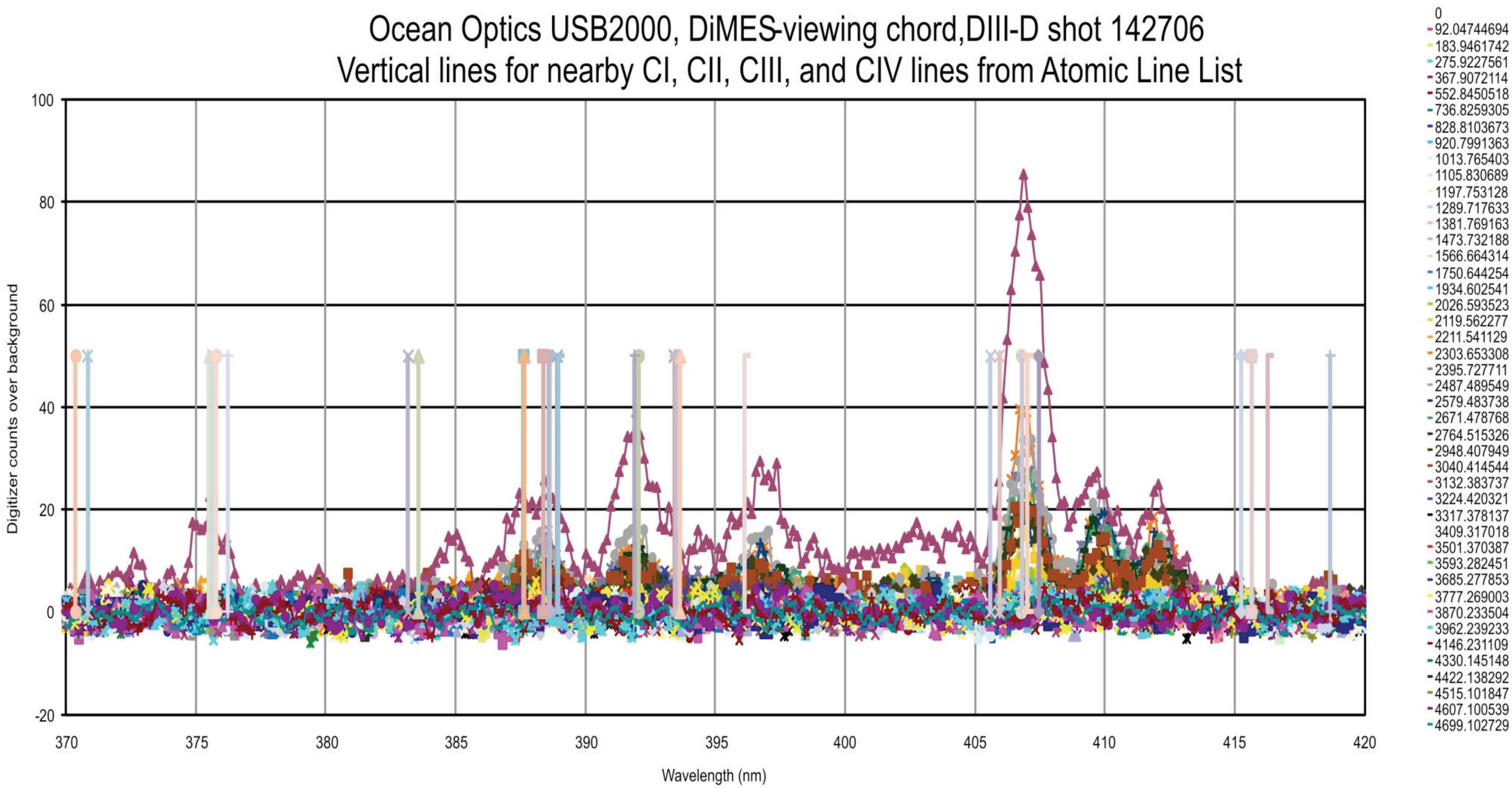
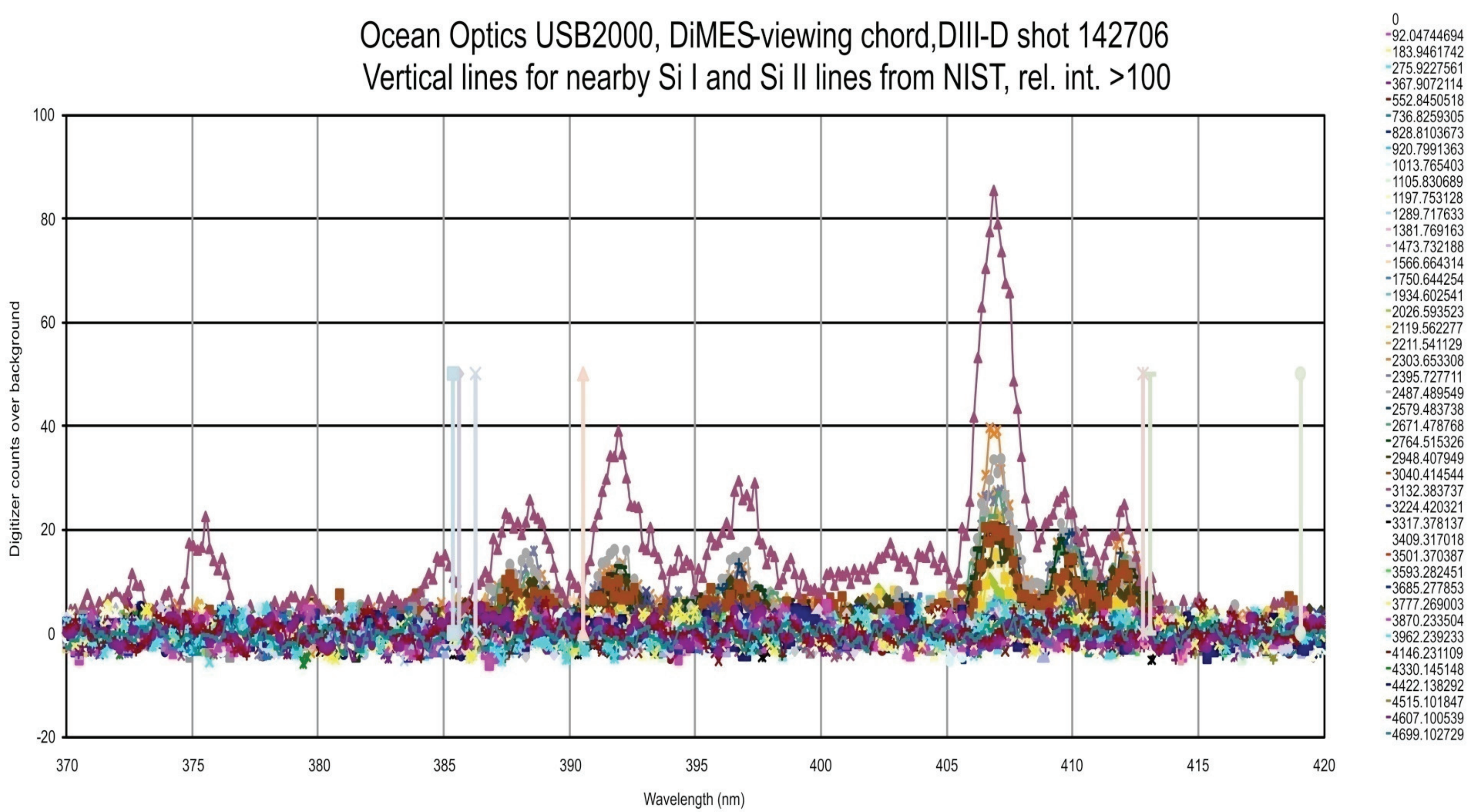
271-10/CPCW/jy

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



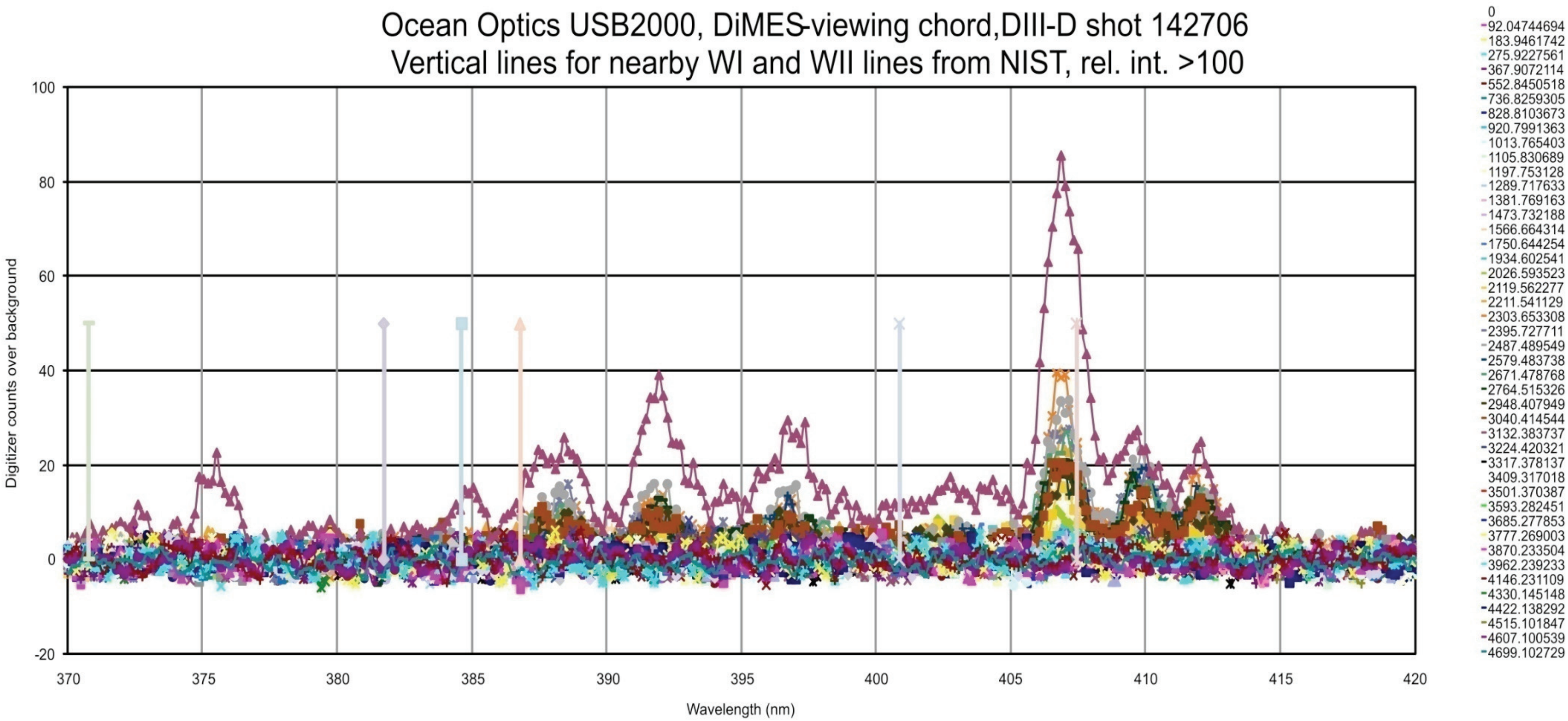
Emission lines from the Atomic Line List

No Significant Si Lines Measured Except Possibly at 385 nm (Seen in the Disruption Only)



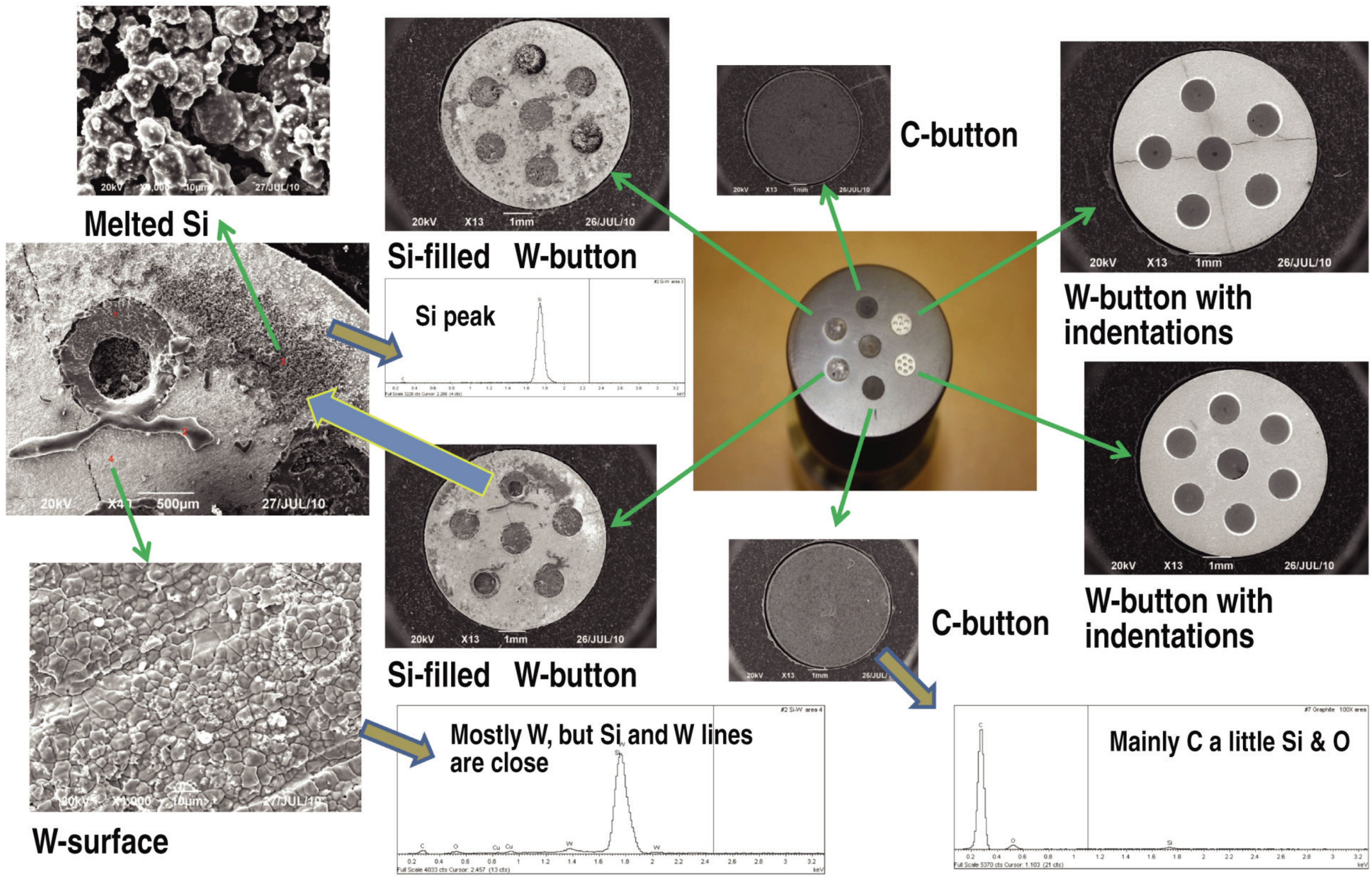
Emission lines from the NIST database

Emission of W Unlikely Due to Lack of Strong 400.8 nm Line Emission (rel. int. 1000)



Emission lines from the NIST database

Details Show Melted Si but Minimum Transport



Si-W Buttons Exposure Observations

- As expected surface Si on the W button got removed during discharges easily 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 shows 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 thermal dump, which will need to be confirmed
- W-buttons were not damaged, observed cracks could be due to drilling of the indentations
- Additional analysis to be performed on other 4 shots (no Ocean Optics was available) and look at the heat flux variation for the melting of Si
- Disruption tolerance to be demonstrated along with variations in W-material, and optimization on indentation geometry and Si-fill method
- New samples will be fabricated and exposed to thermal dump during disruption

Conclusions

- The first Si-filled W-surface buttons with indentations have been fabricated and exposed to high power discharges in piggyback mode in DIII-D
- Impacts from the high power thermal dump from disruptions still have to be demonstrated, but initial results could support the retention of Si in the indentations of the W-surface, thus allowing the possible demonstration of vapor shielding effects during the thermal dump from disruptions
- This experiment demonstrates the possibility of developing a robust PFM design
- Efforts will continue on the selection of suitable W material, indentations or toroidal groove geometry, and the fabrication of the Si-filled W-surface
- Key critical issues have been identified and the development of the Si-filled W-surface concept will continue
- At the same time it is necessary to control and mitigate rapid transient events like disruptions, type-I ELMs and runaway charged particles in order to reduce the frequency of thermal dump to the chamber wall and divertor surfaces
- Additional efforts will also be needed to achieve uniformity of heat and particle flux distributions at the chamber wall, to minimize peak heat flux at the divertor and to learn how to maintain in real time the replenishment of Si in the indentations or toroidal grooves for steady state operation