Status of DIII-D Boundary Experiments Including Si-W Surface Material Development

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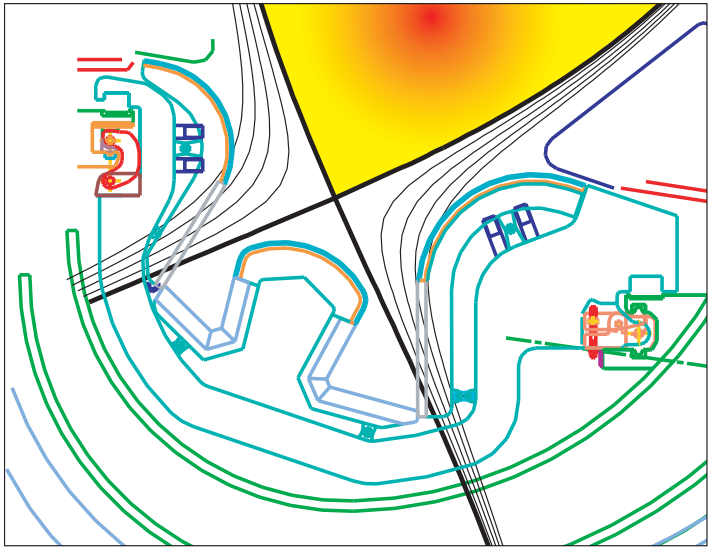
DIII-D NATIONAL FUSION FACILITY

GENERAL ATOMICS

The Boundary Plasma Remains a Challenge for Fusion Research

ITER Divertor



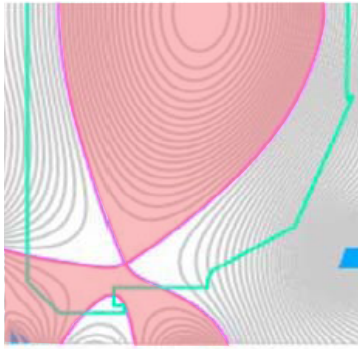
High Power Density and Plasma flux in Future Tokamaks

Device	$q_{\parallel}$ (GW/m^2)	Ion Fluence to PFCs (TC/yr)
ITER	0.9	0.3
FNSF	2.0	4
DEMO	3.0	40

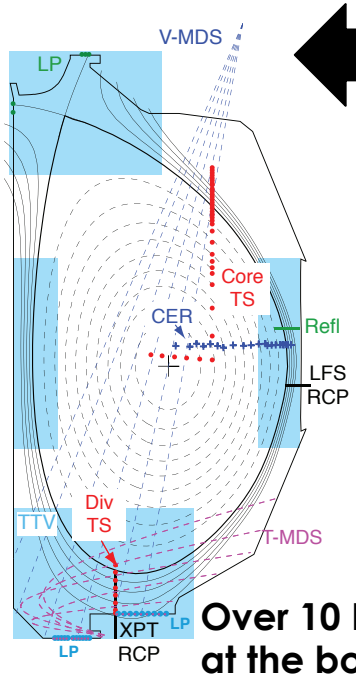
- **A divertor solution should include**
  - High radiation fraction
  - Low plasma temperature to minimize erosion
- **And compatibility with core plasma**
  - Cold divertor stably isolated from core plasma
  - Maintain core density for efficient current drive

TC ~$10^{12}/(1.6\times10^{-19}$ C/e)=6×10^{30} electrons

DIII-D has Exceptional Capabilities for Developing a Boundary Solution for Future Devices

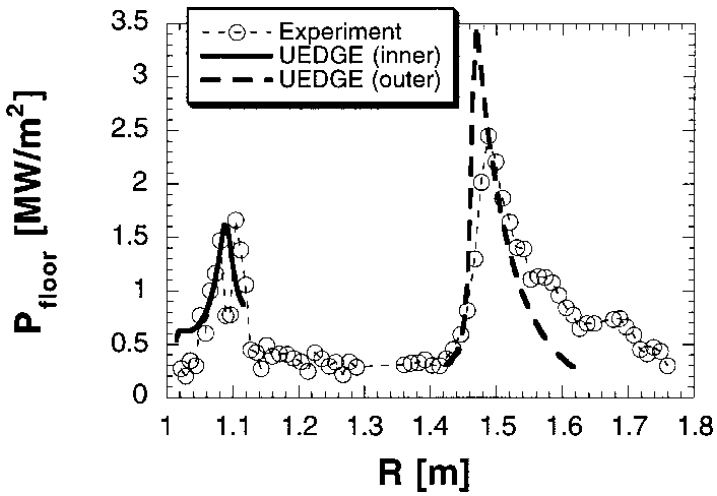


Flexible Configuration & Operational Regimes

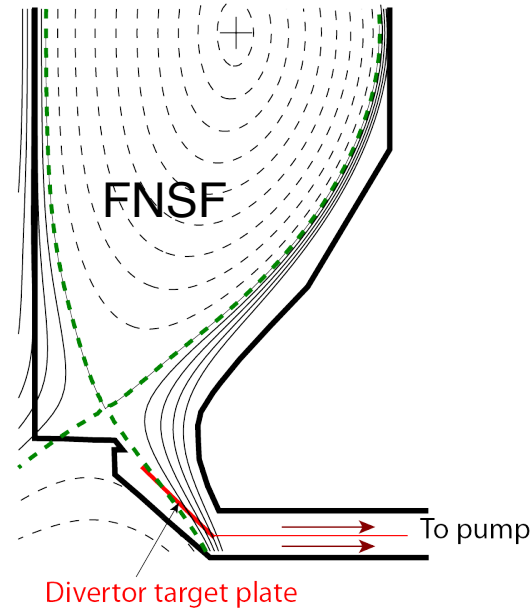


Over 10 Diagnostics at the boundary

Boundary Plasma Models (UEDGE, SOLPS, OEDGE)



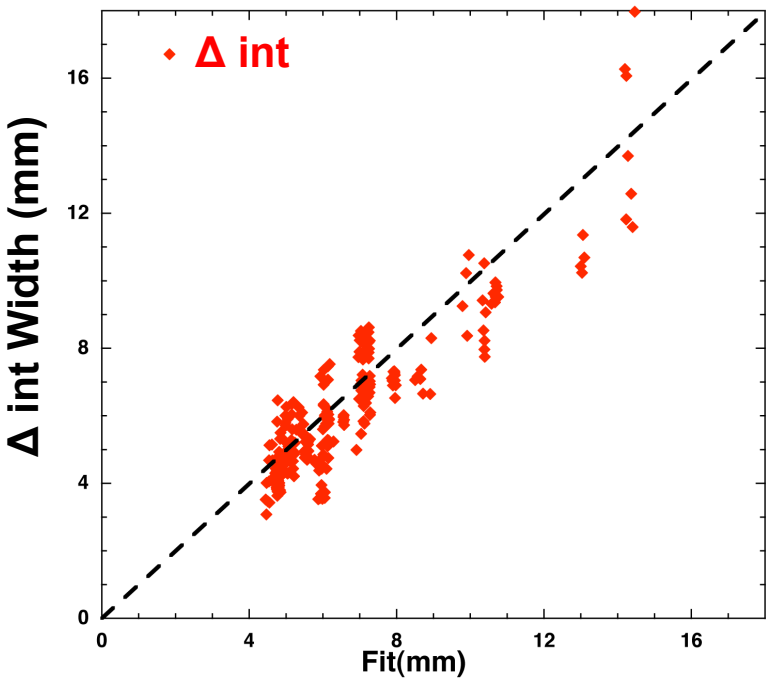
Future Device Divertor



- **A validated model is required to extrapolate DIII-D results to FNSF and ITER**

Significant Progress in Divertor Heat Flux Width Scaling Resulted from 2010 Joint Research Target (JRT)

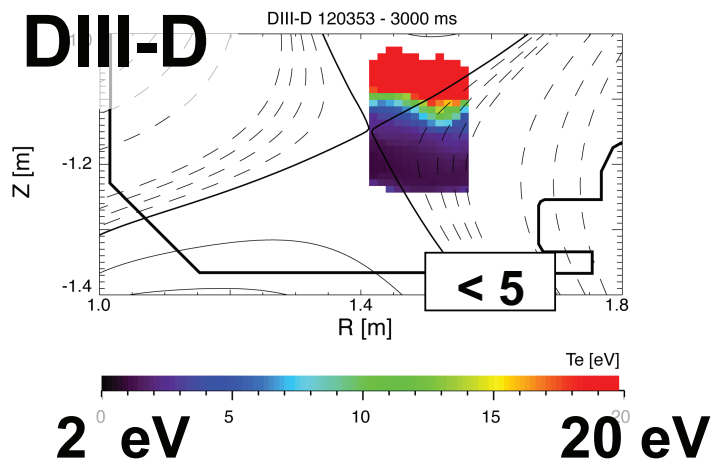
DIII-D Heat Flux Width Primarily Dependenent on I_p

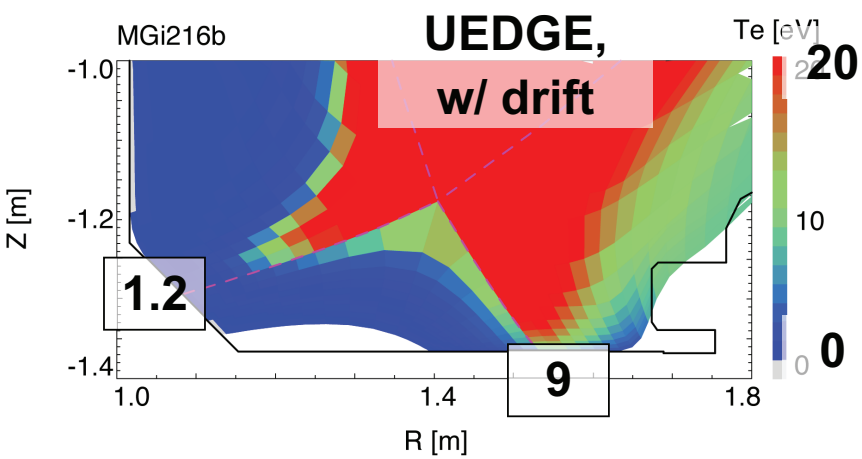


$$\Delta_{Int} = 1.4 \times 10^4 \times I_p^{-1.32} \times B_t^{-0.06} \times P_{SOL}^{0.24} \times n_{GW}^{-0.03}$$

- **Heat flux width, λ_q ($q_{||}$) critical for divertor design; ITER → FNSF → DEMO**
  - Defines required dissipation
  - Sets detachment upstream density, $n_{core} \propto q_{||}$
- **Empirical scaling measured**
  - Strongest dependence, $\lambda_q \propto 1/I_p$
  - Scaling similar in other tokamaks
- **International database beginning**
  - US database: DIII-D, C-Mod and NSTX
  - International effort through ITPA
  - Initial analysis finds little size scaling

Additional Physics is Required to Adequately Describe Detached Radiative Divertor Operation





- **Divertor heat dissipation up to highest available power has highlighted important physical processes**
  - Neutral/ion collisions, recombination, …
- **Additional physics is needed to reconcile**
  - Model prediction of detachment at higher core density
  - Strong flow in SOL
  - Divertor profile instability; rapid onset of core MARFE
  - Large in/out divertor asymmetry

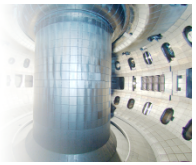
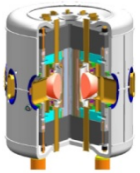
Isolate Additional Physics Required for Codes to Match Experimental Detachment Onset

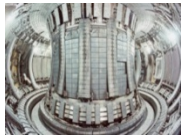
- **Several candidate processes: kinetic parallel transport, sheath formation, neoclassical transport, anomalous pinch,…**
- **First, determine role of plasma flow in the energy, momentum and particle balance of detachment onset**
  - Determine scaling of conditions required for detachment
  - Utilize extensive diagnostics to isolate discrepancies between 1D models, 2D codes and experiment
  - Utilize detached plasmas to address ITER's concern of reattachment timescale
- **Modify detachment onset and stability with divertor configuration**
  - Initial exploration of potential advantages
  - Test model predictions

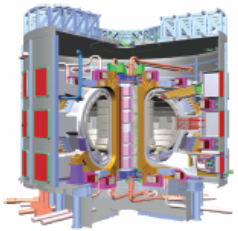
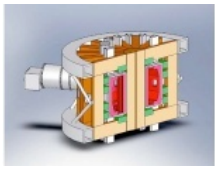
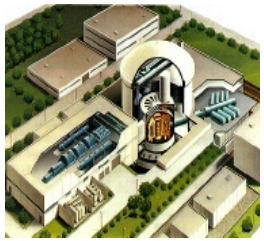
Surface Material is a Key item for Fusion Development

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-Mod	AUG	JET-ILW	EAST	ITER	FNSF	DEMO
C	Mo	W	Be/W/C	C/W	Be/W/C	?	?

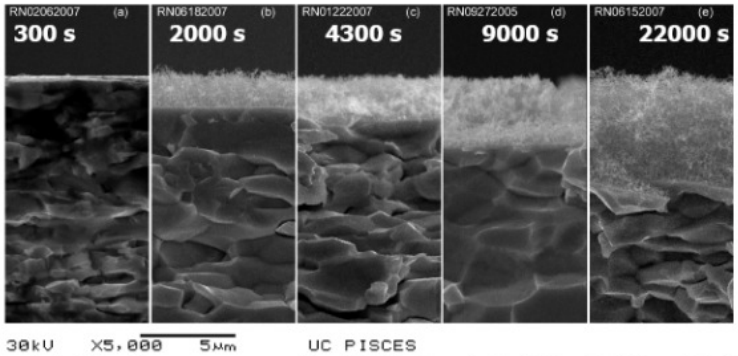
(High neutron and edge alpha fluence)

← **Surface material options** →

C and Be will not be suitable for the next generation DT 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 Independent of Alloy Development

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 nano-structure from the bottom; the thickness increases with plasma exposure time

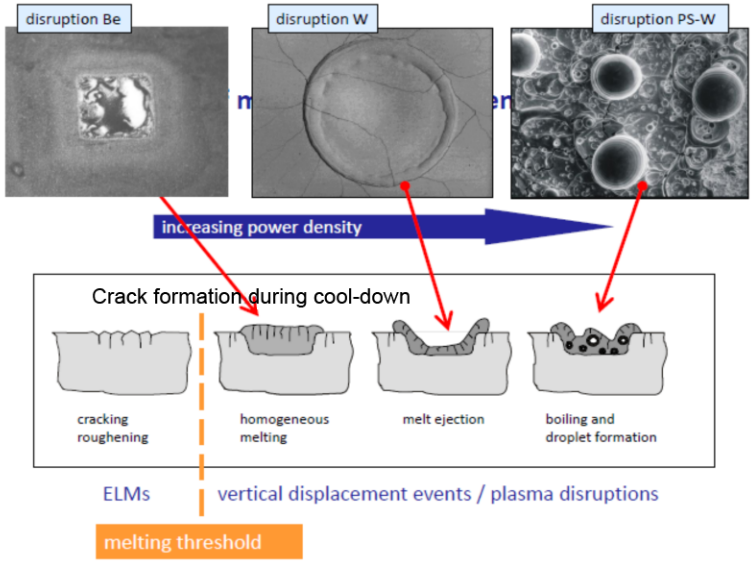
Baldwin and Doerner, Nuclear Fusion 48 (2008) 1-5

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, VLT conf. call Jan. 2011

ITER disruption loading:

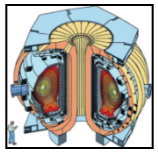
10-30 MJ/m^2 for 0.1 to 3 ms



Irreversible surface material damage

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

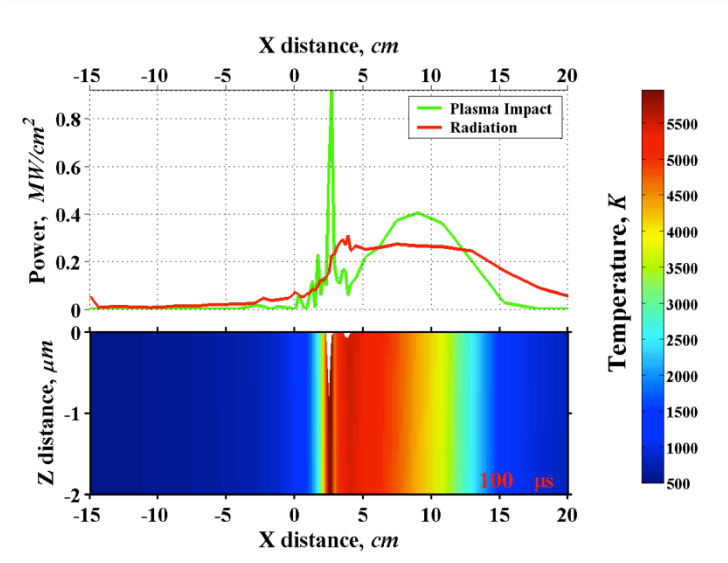
We cannot eliminate un-predicted disruptions even if disruption detection and mitigation work perfectly

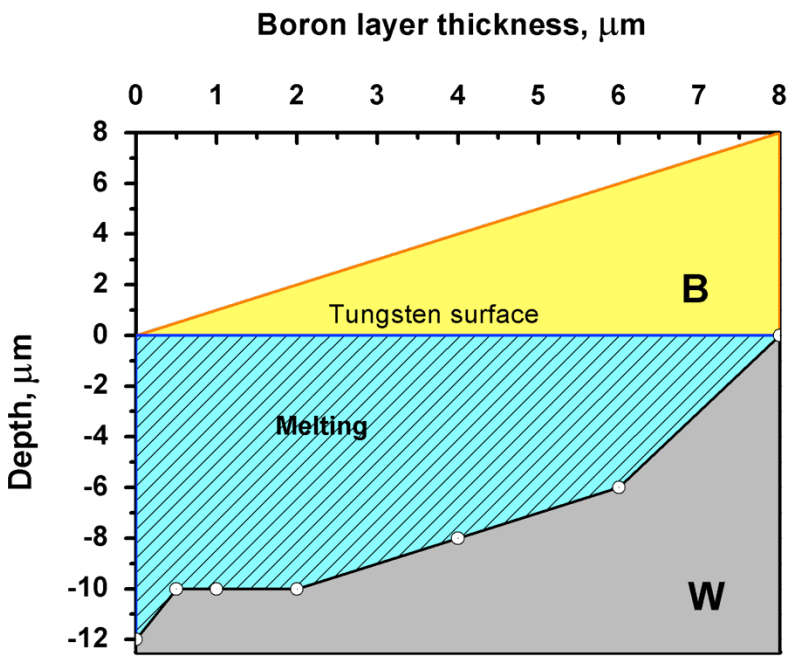


Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->TBM->Power conversion->General

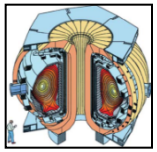
Divertor Surface Erosion and Vapor Shield Protection from Disruptions

Disruption condition:
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



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Disruption Tolerant Surface Development Sequence

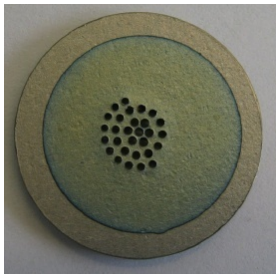
W-mesh

2008

Damaged mesh during high temp formation of B-W compounds

W-disc with holes

2009

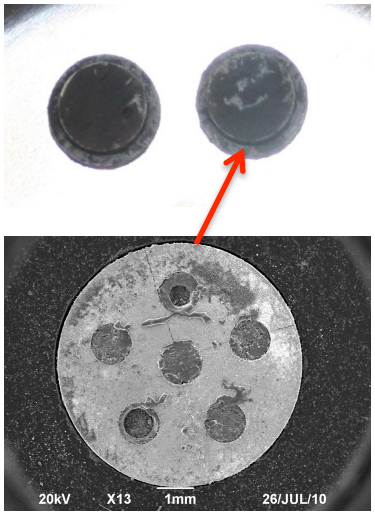


Difficulties of filling holes with B and Si is a better match

Confirmed Boron coating thickness of 0.75 μm < 1 μm

Si filled W indentations

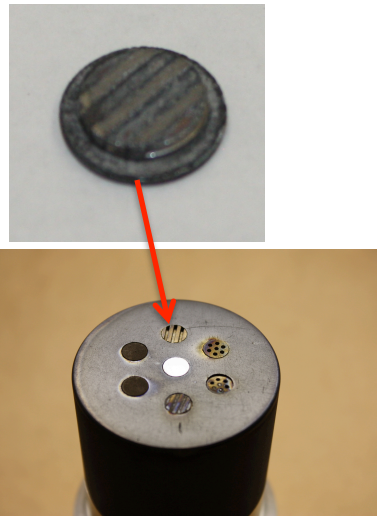
2010



A loose and messy filling; after 7 high power shots found melted Si, but Si retained in indentations

CVD infiltrated Si in W indentations and slots

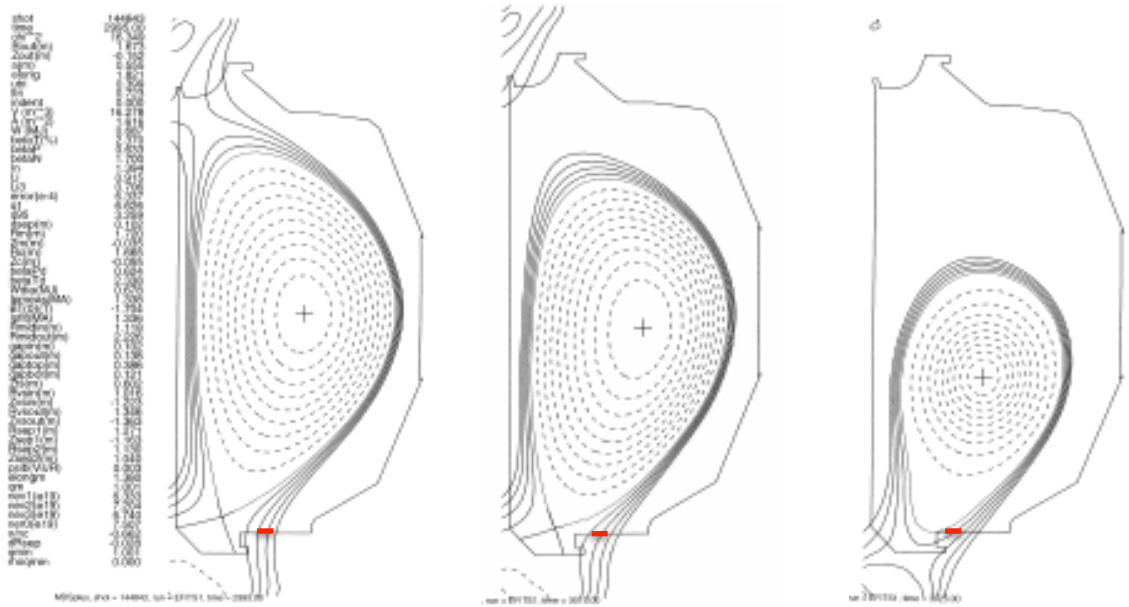
2011



Nicely filled Si in W slots by Ultramet

One Disruption Landed Close to the 2011 Si-W Buttons DiMES

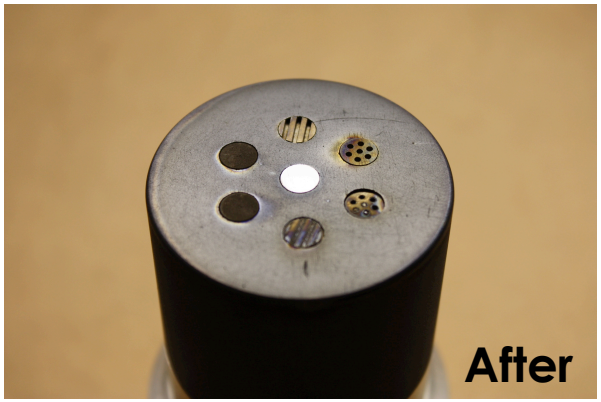
Exposed in DIII-D lower divertor



3005 ms 3015 ms 3025 ms

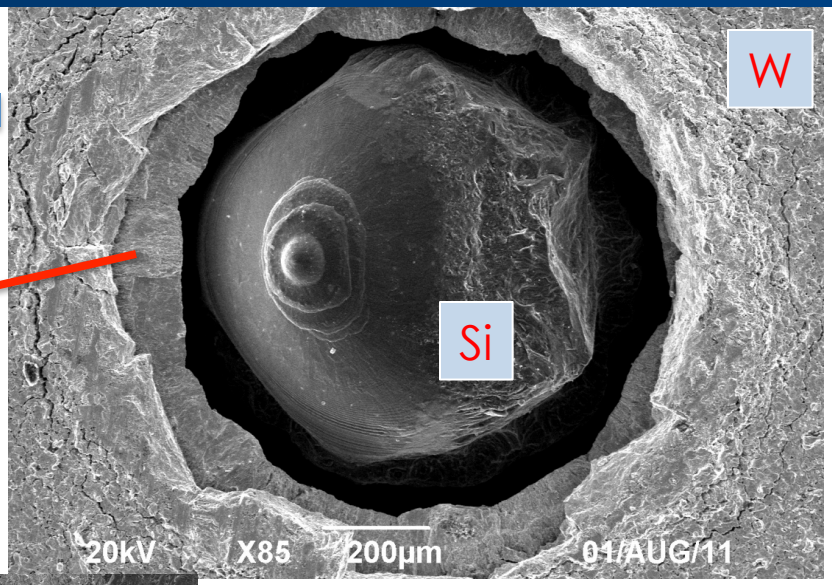
#144843

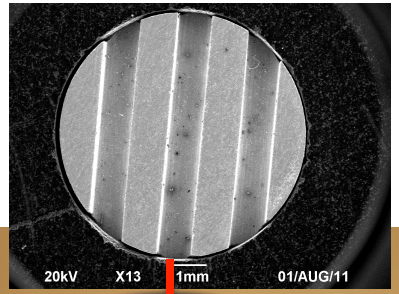
I_P=1.34 MA
B_T=-1.7 T
W=0.69 MJ
κ=1.38

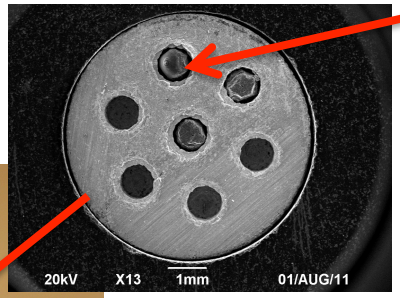


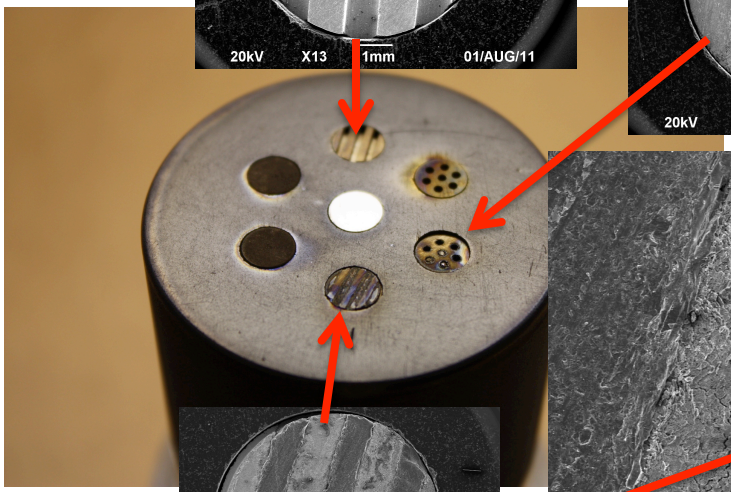
After Disruption Exposure: Si in W Slots Maintain Good Contact, Si in Indentation Partially Melted Possibly because of Poor Contact

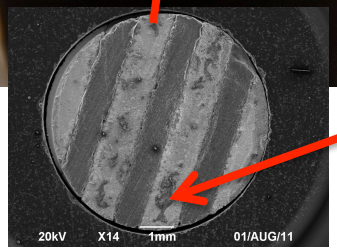
- Si was not in good contact with W
- Some Si was melted
- What about the cracks on the surrounding W

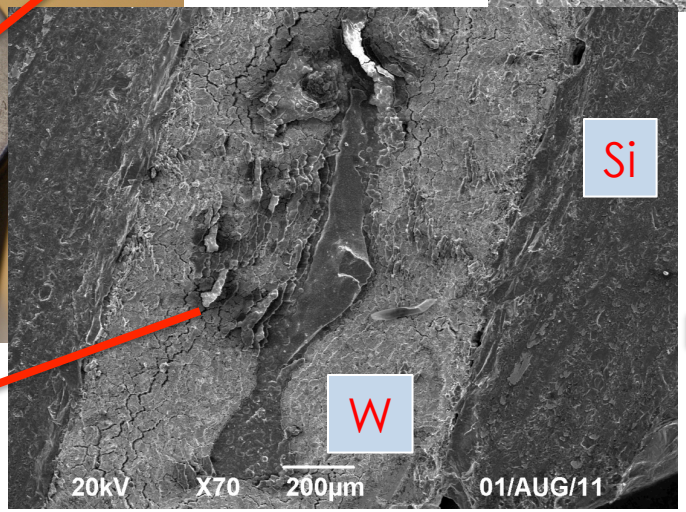








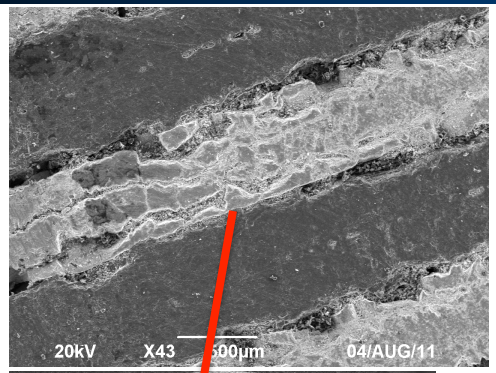


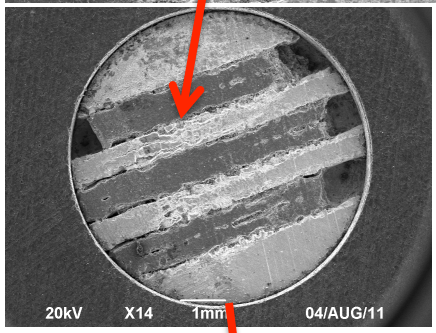


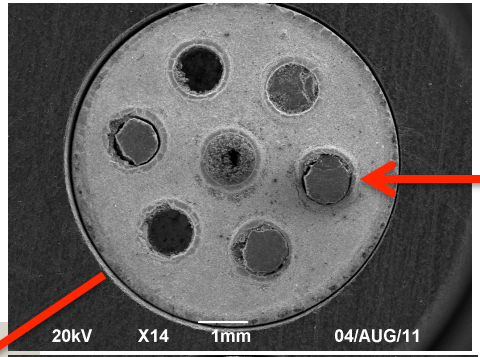
- Si was in good contact with W
- No melting noted
- Why was the W surface damaged ?

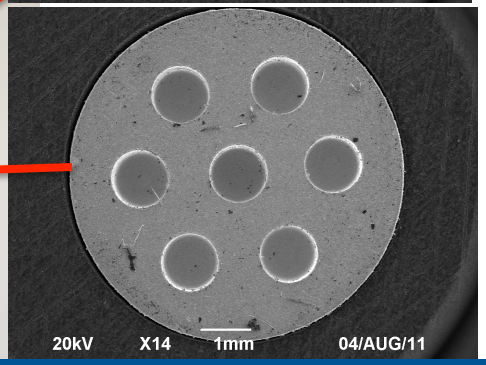
Un-irradiated Buttons, CVD Filling of Si is Much better than with Si Powder. But, the W Surface got Damaged During Filling of Si, Why?

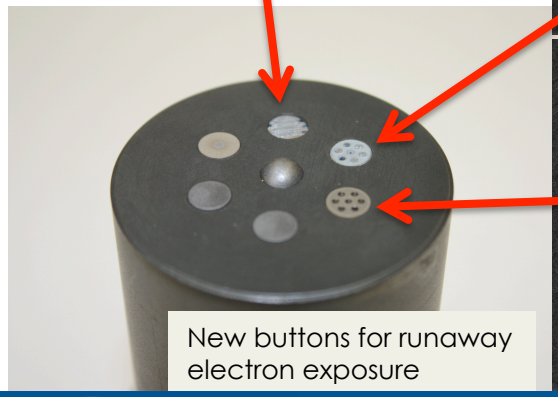
- **Assessment on the damage of W surface during the filling with Si will need to be made before deciding on the next step**

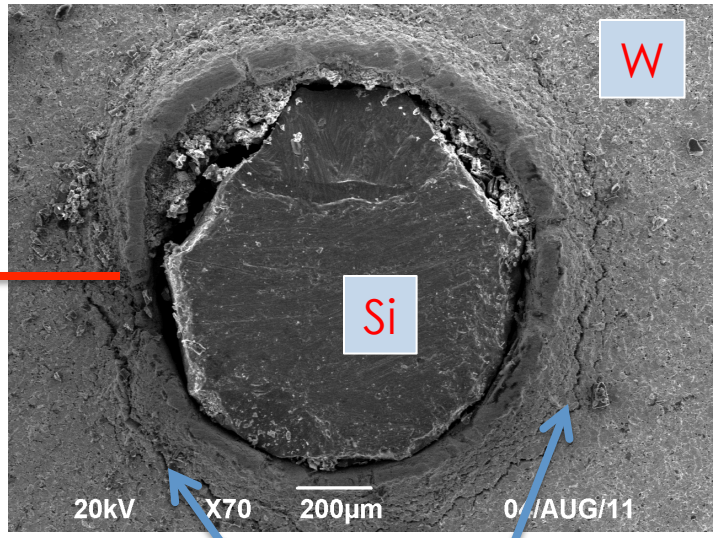












Cracks

Surface Material Development Summary

- We continued the development of Si-W surface material concept with material preparation and exposure to disruptions and runaway electrons
- Using CVD, Si was implanted into W-indentations and W-slots (2011)
- We found gaps between Si and W on indented samples, no gaps were found on slotted buttons, which is a significant improvement from before
- Cracks were found around W-indentations and the surface of W-slots was also damaged. This could be formed during the process of CVD implantation
- Buttons were exposed to a near disruption and a spot of the Si in one of the indentation was melted
- We will improve the process of Si-implantation before requesting system tests with linear machine