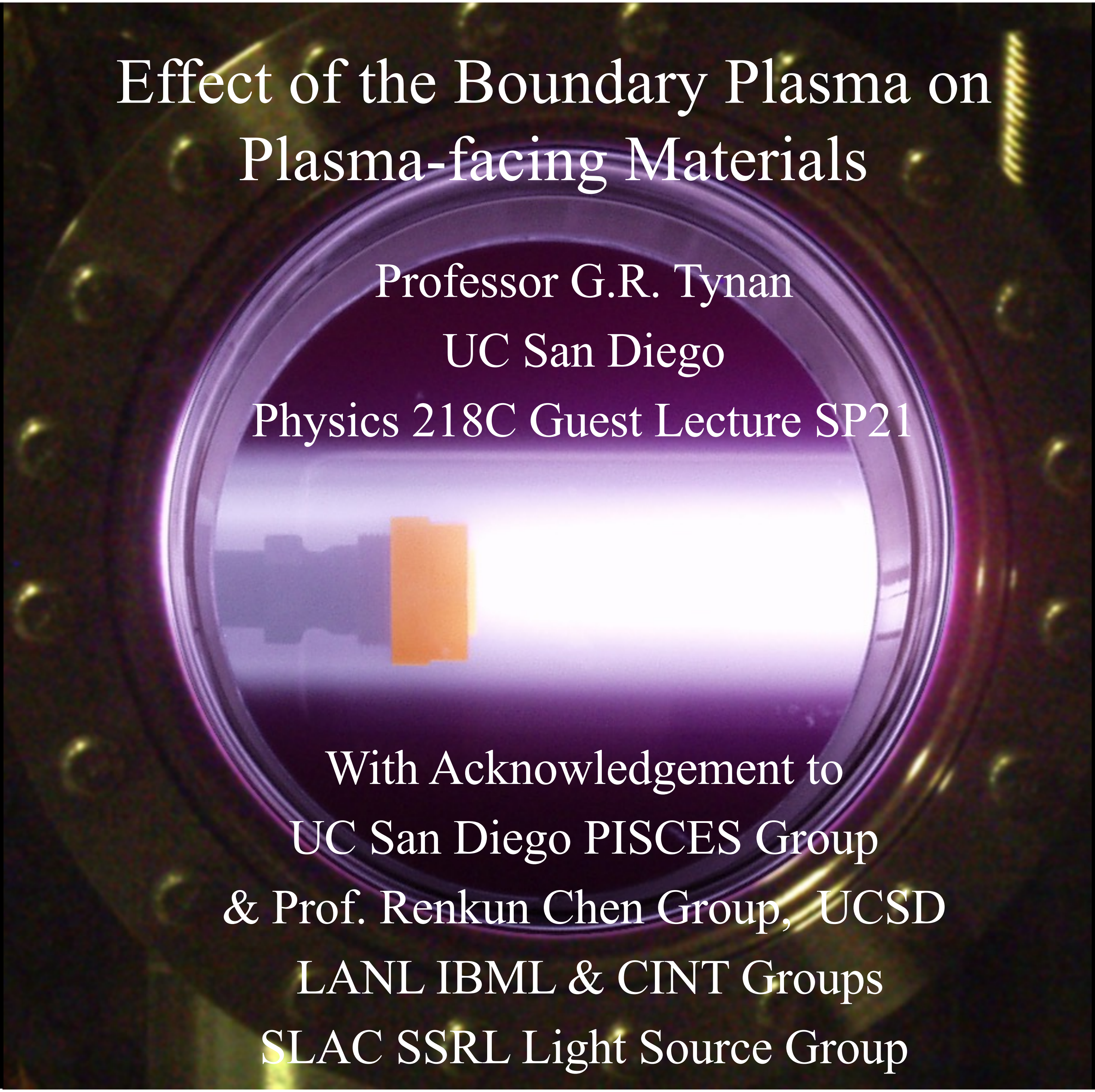
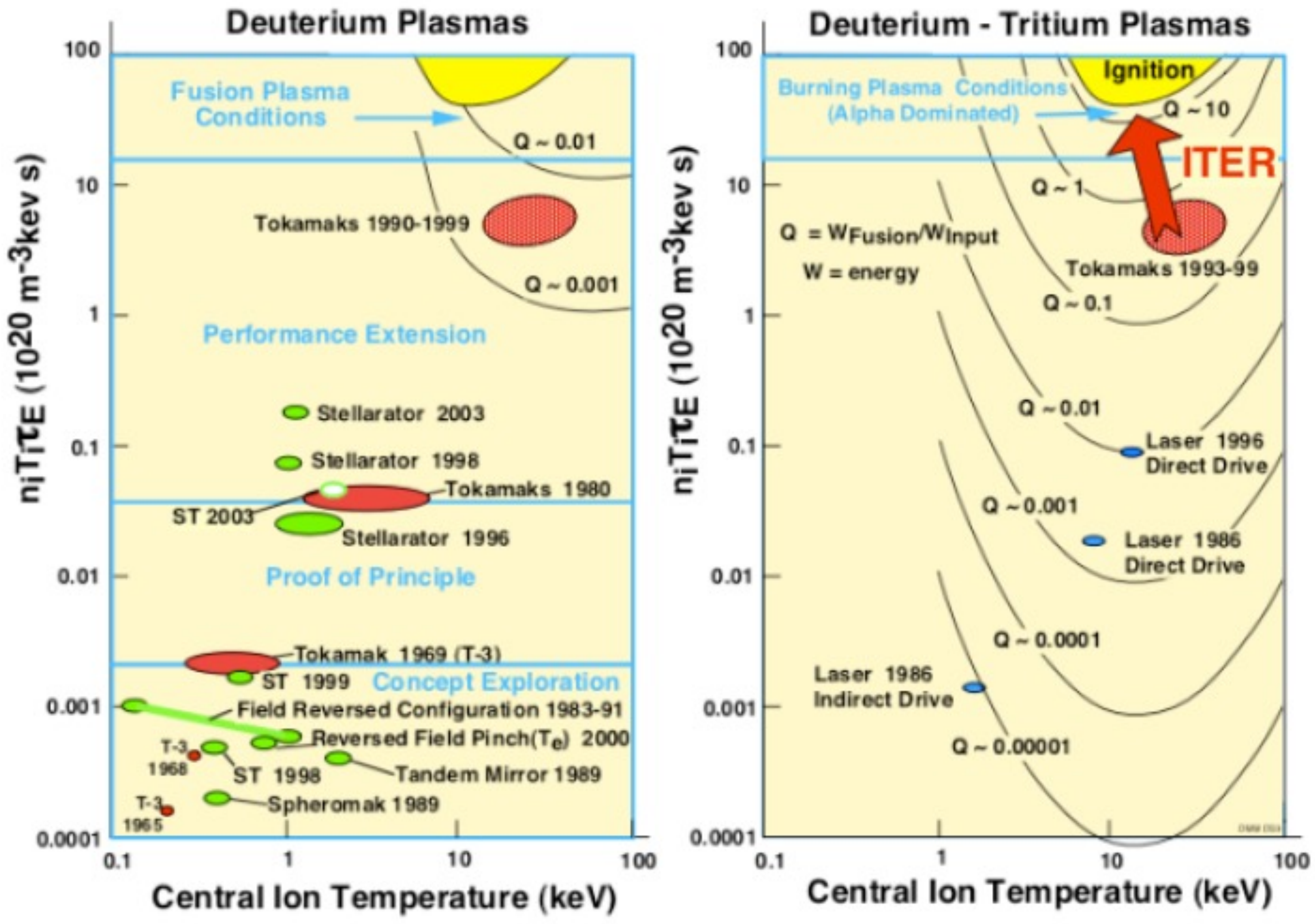


Effect of the Boundary Plasma on Plasma-facing Materials

Professor G.R. Tynan
UC San Diego
Physics 218C Guest Lecture SP21

With Acknowledgement to
UC San Diego PISCES Group
& Prof. Renkun Chen Group, UCSD
LANL IBML & CINT Groups
SLAC SSRL Light Source Group

Progress towards fusion energy production



Ref: Greenwald Report, DOE-SC 2007

Overview of Plasma-Material Interactions

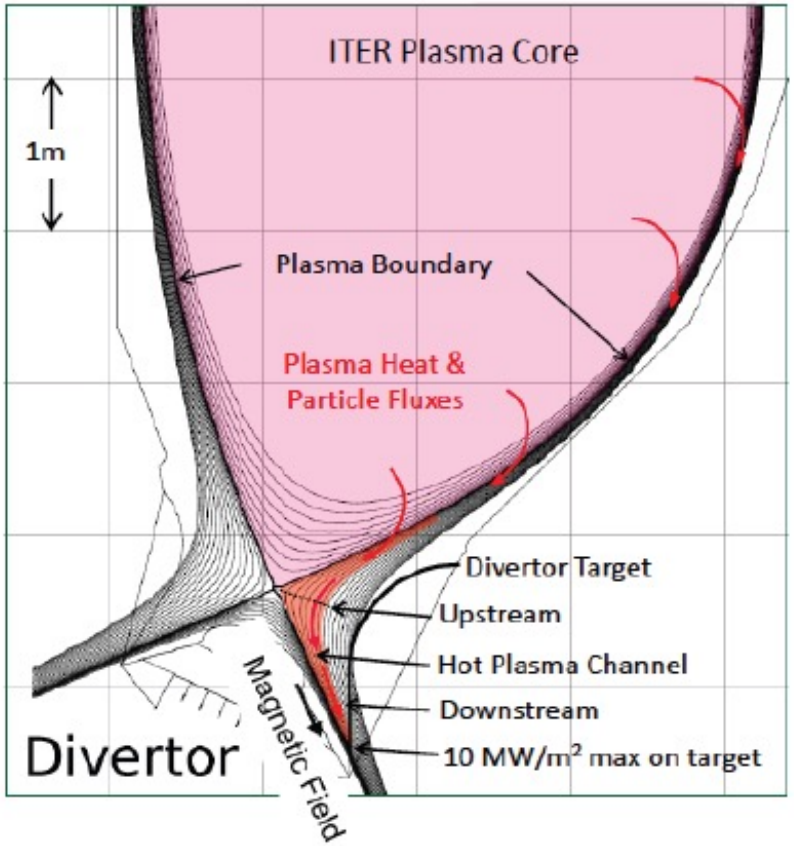
Key Issues
- Erosion lifetime and plasma compatibility
- Tritium inventory
- Thermal transients
- H/He blistering
- Heat removal
- Fabrication technology
- Neutron damage

Leading candidate materials

PFC and Divertor
- Be, W, (C?)

Structural components
- Fe-Cr steels, V-Cr-Ti, SiC

ITER Plasma Material Interface



bulk plasma:

impurity tolerance

$W < 2\ 10^{-5}$, reactor $< 10^{-4}$

Be, C: 10^{-2}

first wall:

modest flux of high energy neutral particles (100s eV),

low energy ions

divertor target:

high heat flux 10 (20) MW/m^2

transient heat loads:

e.g. ELMs, disruptions

Plasma-Material Interface (PMI) includes plasma physics, materials science, atomic physics…

- PMI instantly and continuously remakes the plasma-facing surface, significantly modifying material properties and plasma behavior

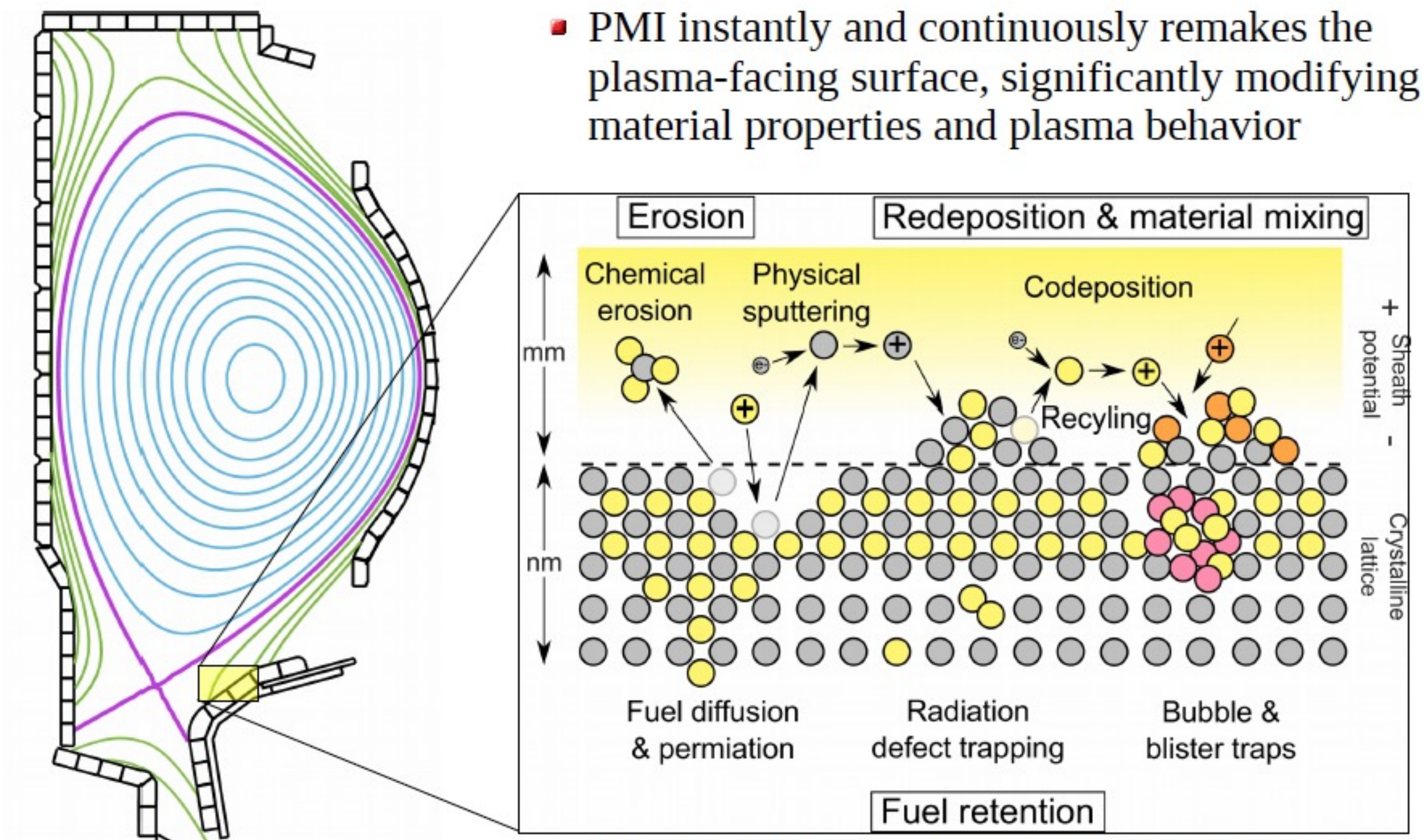
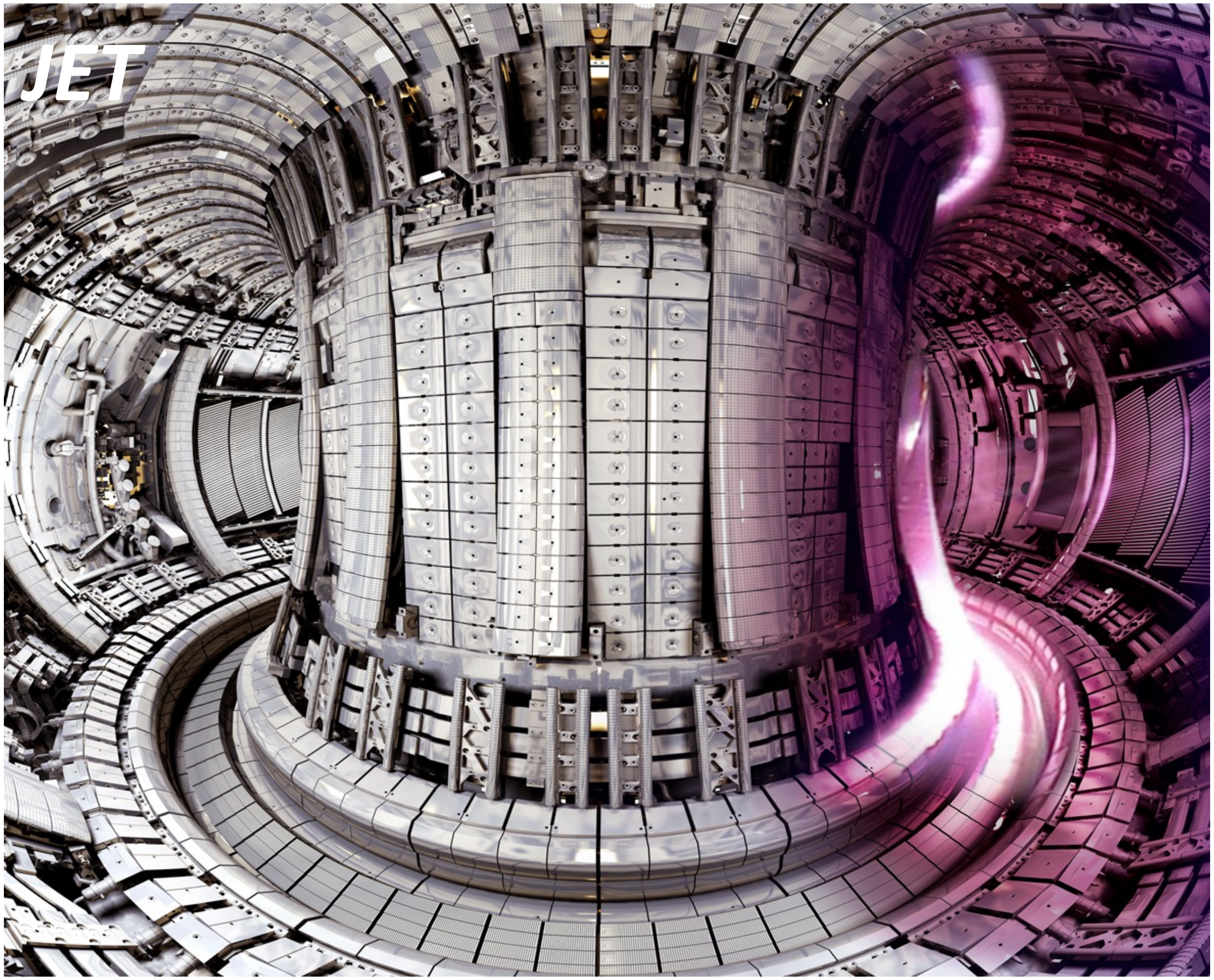


Figure adapted from B. Wirth et al. MRS Bulletin, March 2011

BD Wirth, K Nordlund, DG Whyte, and D Xu, MRS Bulletin, 36, 2011

Plasma-materials interactions are one of the key challenges remaining for fusion energy



https://www.euro-fusion.org/jet/

Outline of Talk

- What is required beyond ITER to get to fusion energy?
- What PMI-related issues emerge from this focus?
- What activities are underway?
- What additional efforts are needed?

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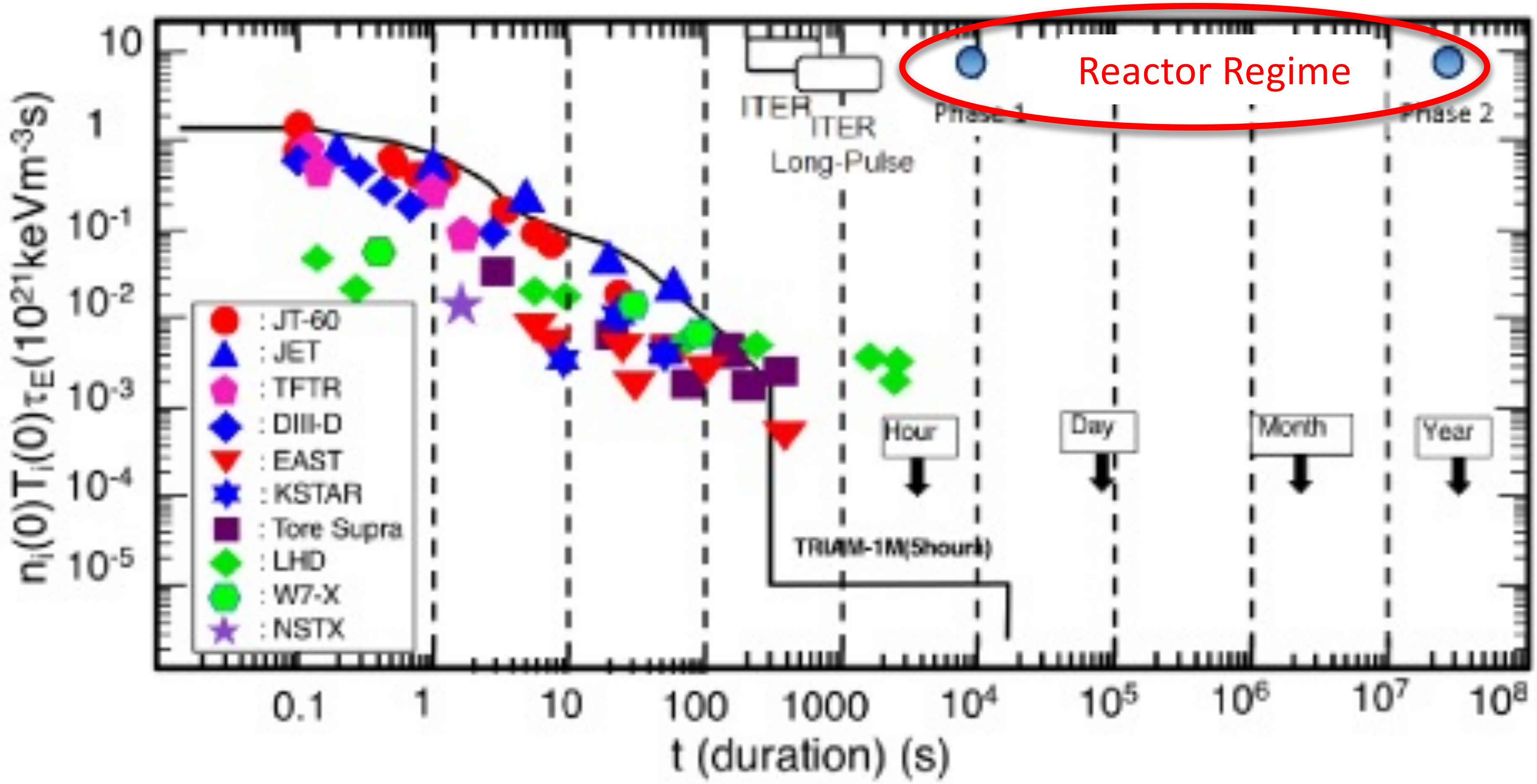
We are far away from long-duration reactor regime



Reactor Regime

Kikuchi, Springer 2015

Roadmap for Chinese MFE Development

PFPP

(~2050)

1GWe, Power Plant Validation

CFETR

(2030 start operation)

I: Q=1-5, steady state, TBR>1, >200MW, 10dpa

II: DEMO validation, Q>10, CW, 1GW, 50dpa

ITER

(~2021)

Phase I : Q=10, 400s, 500MW, Hybrid burning plasma

Phase I I: Q=5, 3000s, 350MW, steady-state burning plasma

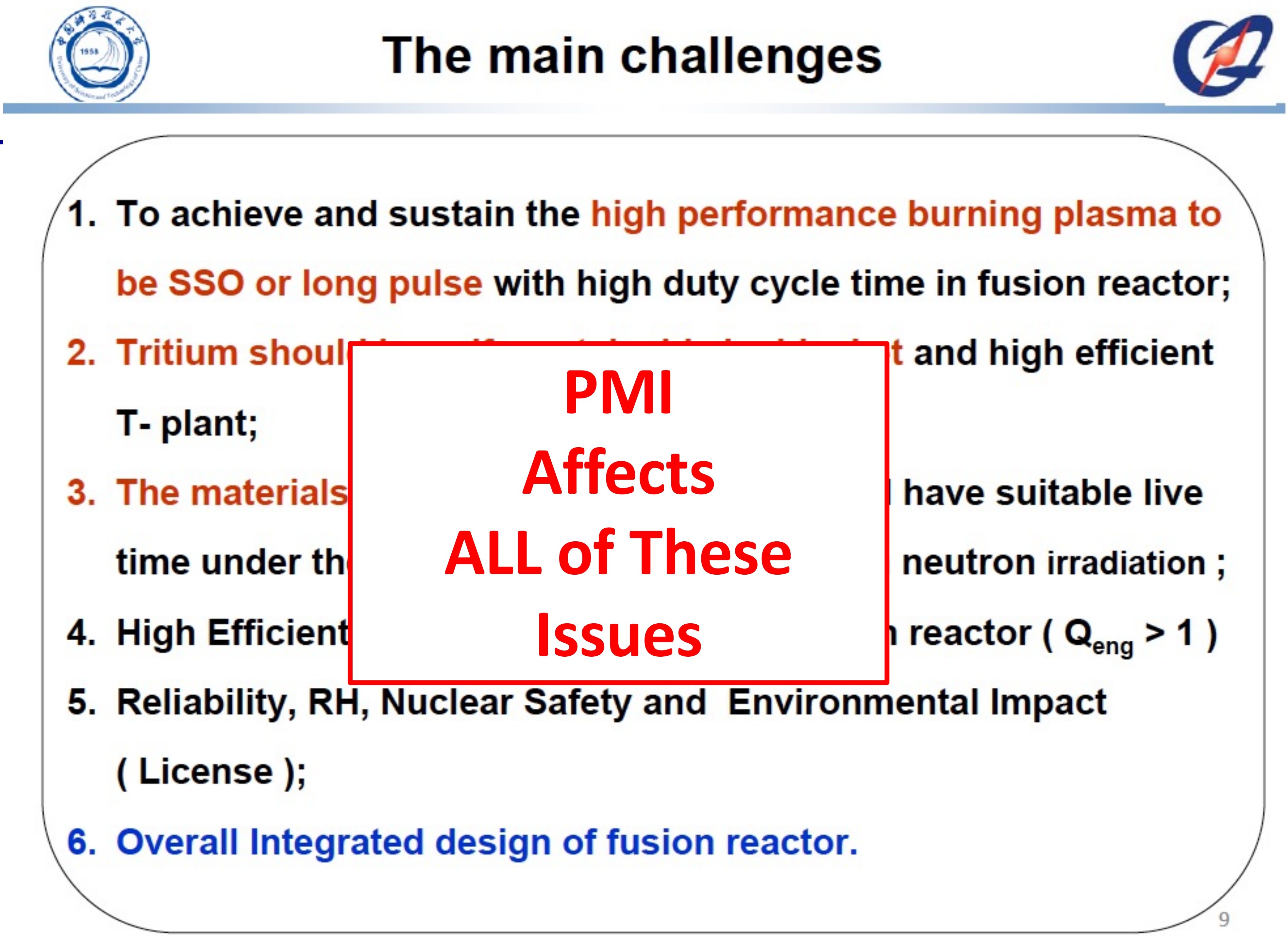
EAST Advance PFC, steady-state advanced operation

HL-2M Advanced divertor, high power H&CD, diagnostics

J-TEXT Disruption mitigation, basic plasma

The main challenges

1. To achieve and sustain the high performance burning plasma to be SSO or long pulse with high duty cycle time in fusion reactor;
2. Tritium should be self–sustainable by blanket and high efficient T- plant;
3. The materials of first wall and blanket should have suitable live time under the high heat load and flux fusion neutron irradiation ;
4. High Efficient electricity generation on fusion reactor ($Q_{eng} > 1$)
5. Reliability, RH, Nuclear Safety and Environmental Impact (License);
6. Overall Integrated design of fusion reactor.



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PMI Affects ALL of These Issues

Plasma-Material Interactions emphasized as a critical area in several community-generated reports

- US – Research Needs for Magnetic Fusion Sciences, Report of the Research Needs Workshop (ReNeW) : (2009)
- US – FESAC Report on Strategic Planning : (2014)
- EU – A roadmap to the realization of fusion energy : (2014)
- Japan - Report by the Joint-Core Team for the Establishment of Technology Bases Required for the Development of a Demonstration Fusion Reactor : (2014)

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PMI Issue	Reactor Impact	Research Need	
Divertor particle & power handling	Dissipate divertor thermal loads ,density control	Edge/SOL transport physics; advanced divertors, transient control	Increasing Timescale
PMI Impact on Confinement	Maintain core plasma performance	Long pulse (1000s seconds) tokamak w/ ***CFETR relevant*** wall conditions	
Surface Morphology Evolution	Loss of performance at high heat flux; dust generation	Understand mechanisms & manage/avoid deleterious conditions	
Helium Accumulation	Effect on D/T Retention, Material performance	In-situ real-time diagnostic for He , D content;	
Fuel Retention Probability ~10^{-6}-10^{-7}	TBR>1	In-situ real-time D, T profiles over <10microns;also need He profiles since He affects retention	
Surface Erosion <~ 1mm/year requires $Y_{net}<10^{-5}$	Wall & Divertor Reliability & Lifetime	In-situ diagnostics Sensitive to ~100's nm over 10micron dynamic range	
Material Migration & Mixed Material Formation	Minimize & Predict evolution of mixed materials	2D SOL Plasma Flows; in-situ mixed material diagnostics	
Rad-damage & Transmutation Effects	New (Degraded?) Materials Properties	Neutron surrogates; neutron irradiation; studies of In-situ retention, material properties	

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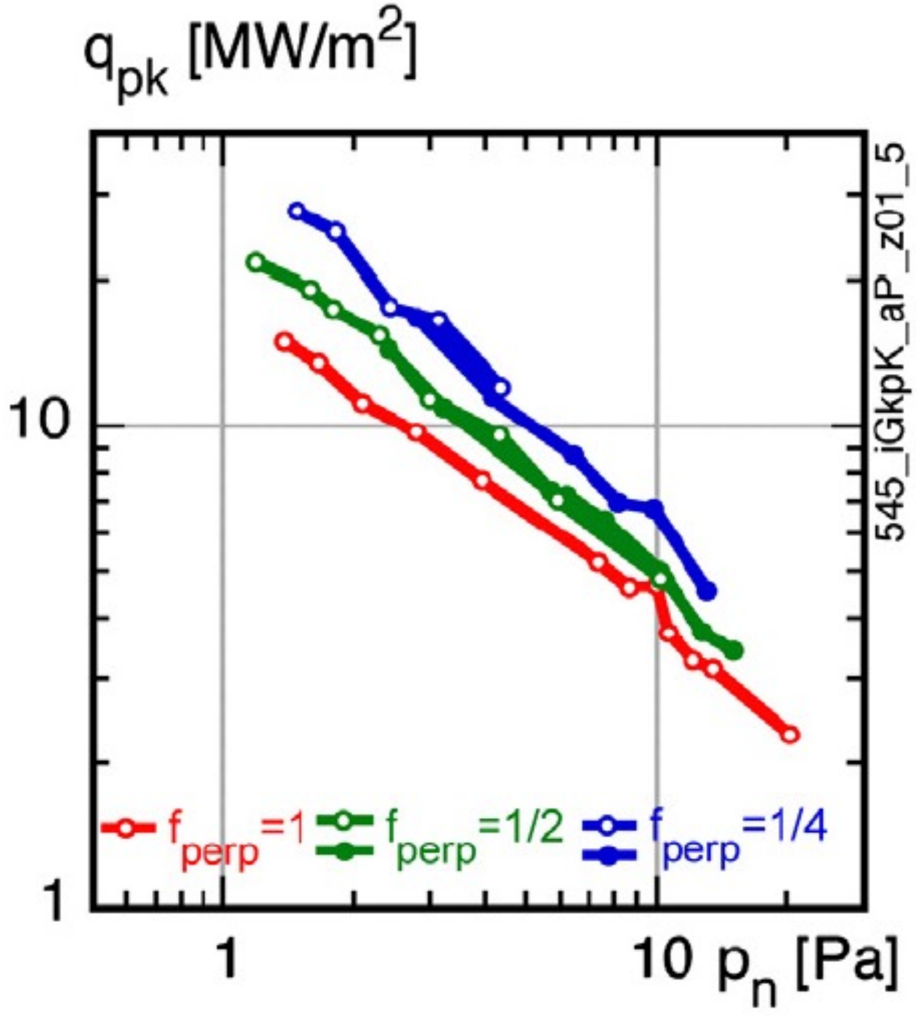
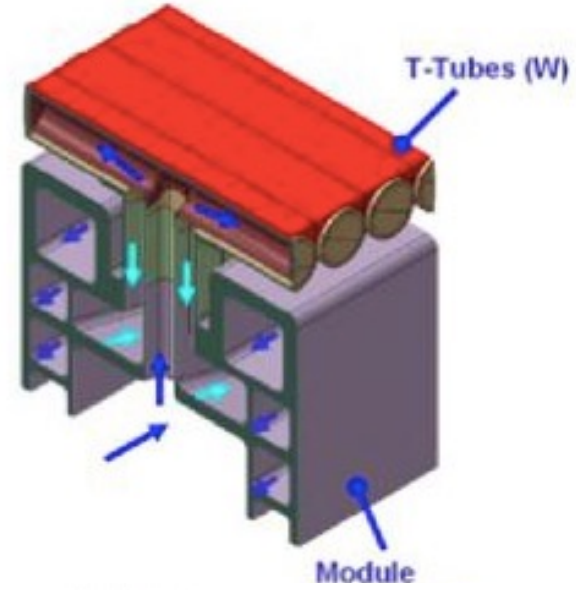
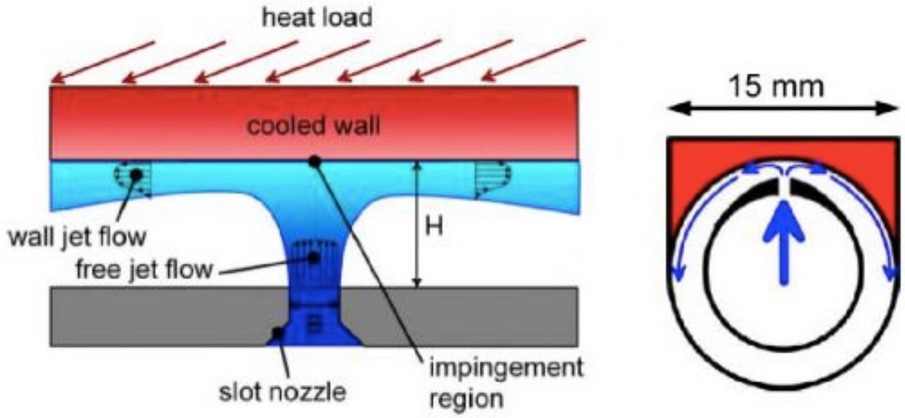
Divertor heat loads force extreme divertor target designs



Kukushkin et al JNM 2013

Ihli et al, Fus. Eng. Design (2005)





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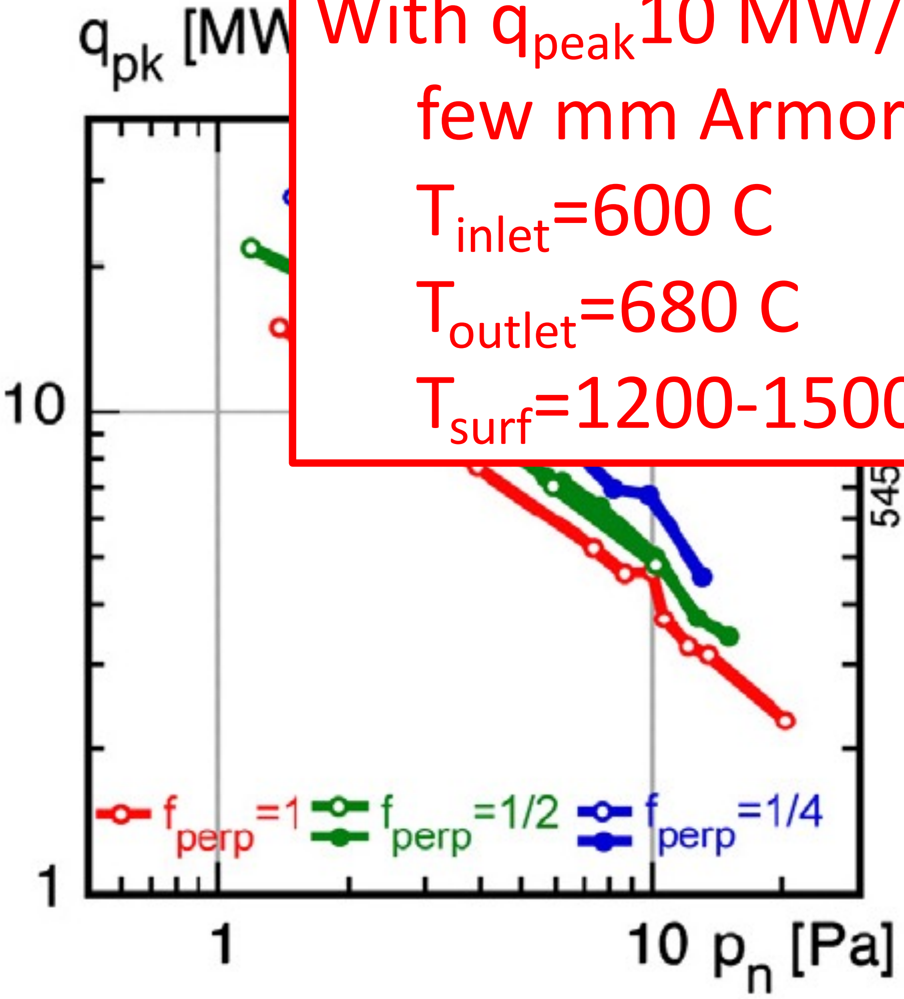


With q_{peak}10 MW/m^2

few mm Armor

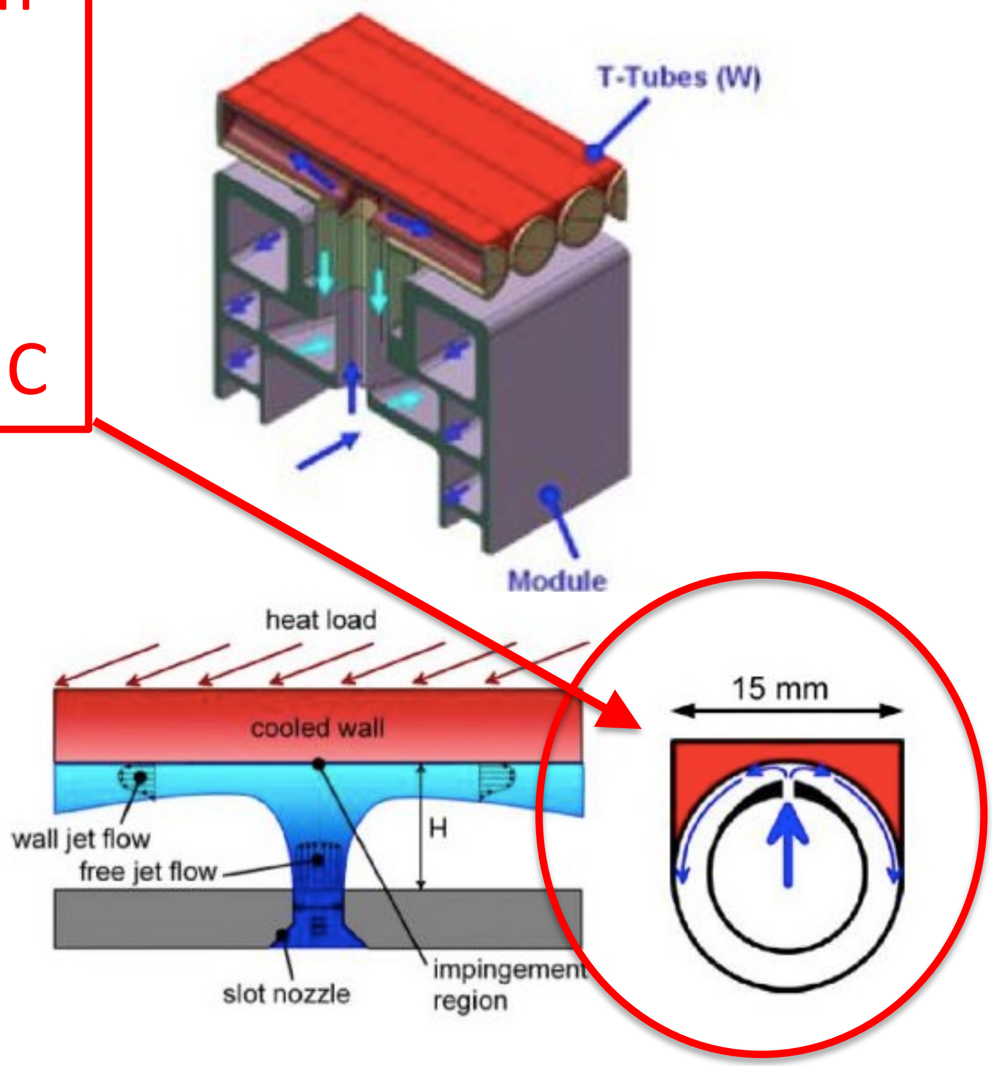
T_{inlet}=600 C

T_{outlet}=680 C

T_{surf}=1200-1500 C

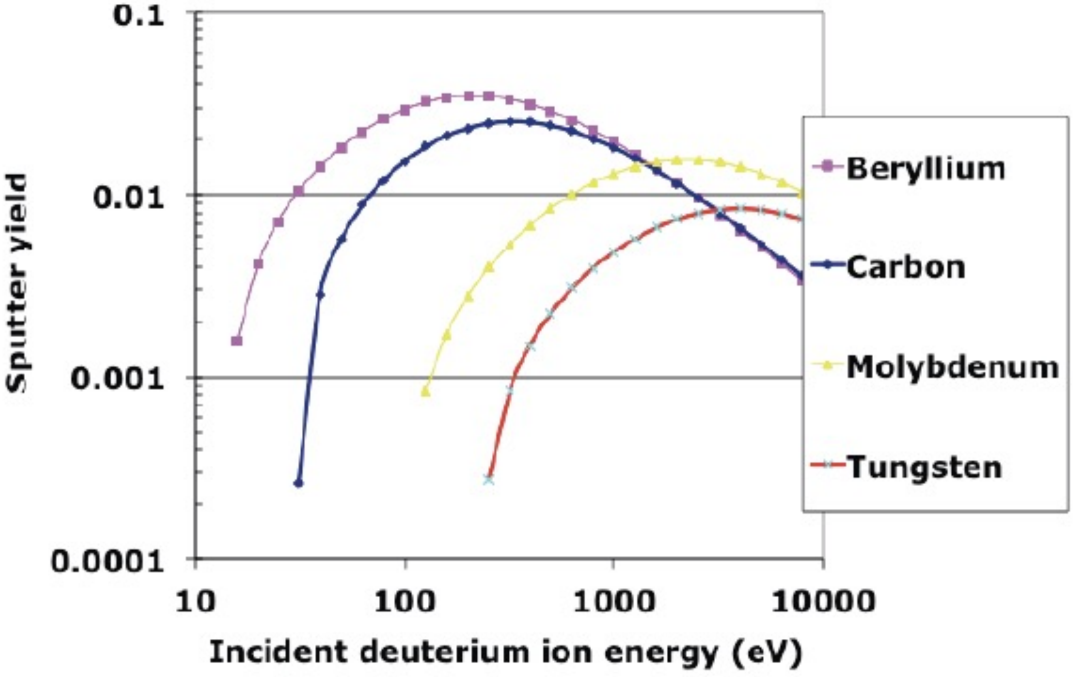
Kukushkin et al JNM 2013

Ihli et al, Fus. Eng. Design (2005)



PMI Challenge: Divertor Particle & Heat Loads

- Divertor target must be thin
  - For q_{div}~10MW/m^2 ΔT~100 deg-K/mm
- Annual divertor target particle fluence ~10^{31}/m^2
- If allow 1mm erosion/yr
  - N_W~6×10^{28} atoms/m^3
  - Area density/mm~6×10^{25}/m^2
- **Allowable net yield Y_{net} ~ 6×10^{-6}**



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Achieved W divertor erosion rate is too high

Parameter	ASDEX-UG[1]	ALCATOR C-Mod[2]	Reactor
Exposure time (sec)	2600	3200	$3\text{x}10^{7}$
Projected or allowable divertor target erosion rate (mm/year)	3.6	0.8	1
Measured or allowable W atom erosion/m^2	$1.5\text{x}10^{22}$	$1.4\text{x}10^{21}$	$6\text{x}10^{25}$
Total ion fluence/m^2	$6\text{x}10^{25}$	$2\text{x}10^{25}$	$3\text{x}10^{31}$
Effective yield	$2.5\text{x}10^{-4}$	$7\text{x}10^{-5}$	**$2\text{x}10^{-6}$**

Need to reduce Y_{eff} by 30-100x➔

Need low T_e and/or lower ion flux at target ➔ Advanced Divertor?

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Divertor challenge is multifaceted

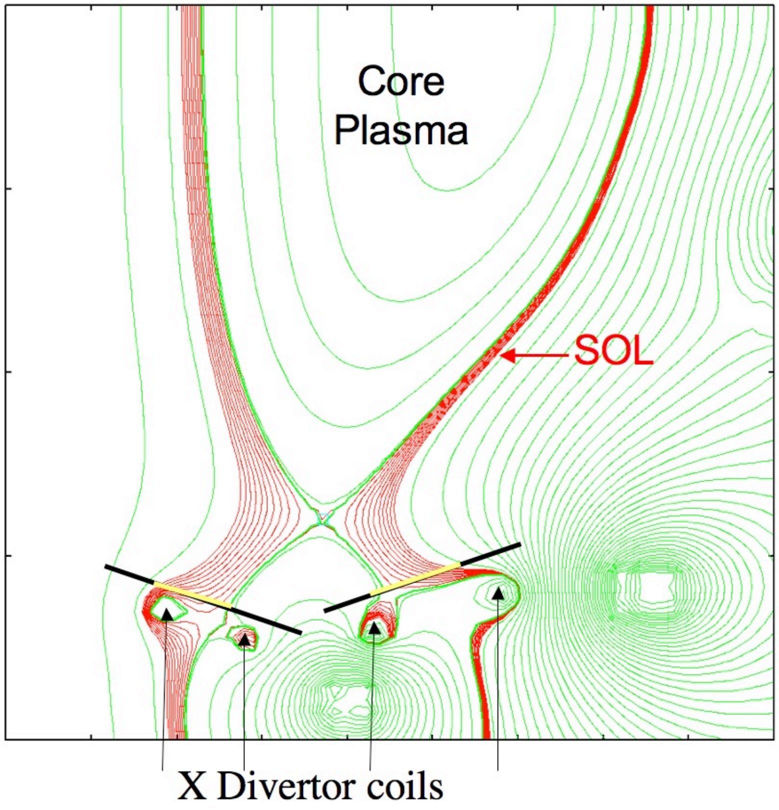
1. **Divertor plate heat flux**
   - Technological limits of ~ 10 MW/m^2 , perhaps less at much higher neutron fluence than ITER
2. **Helium pumping**
   - In simulations, degrades *very rapidly* with power and lower density
3. **Plasma erosion of the plate/plasma impurities**
   - High plasma plate temperature/low density greatly increases sputtering/reduces prompt redeposition
4. **Divertor survival of disruptions/ELMs & other transients**

Need ***integrated*** solution for post-ITER step (CFETR, …)

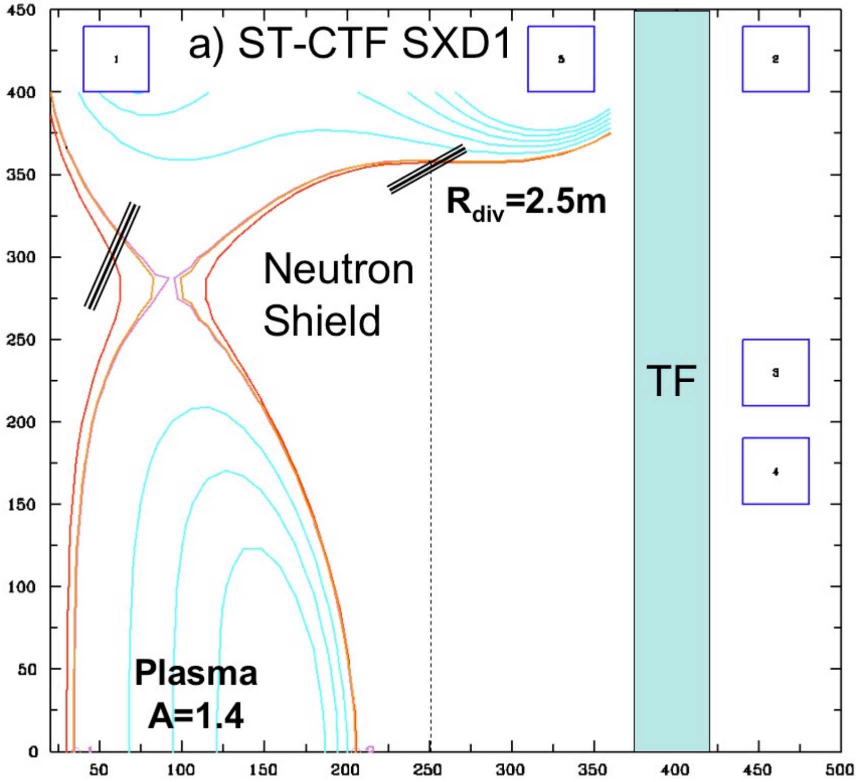
Research Need: Advanced Divertor

Kotschenreuther, ReNeW Presentation

Flux Expansion/Snowflake



Super-X Divertor



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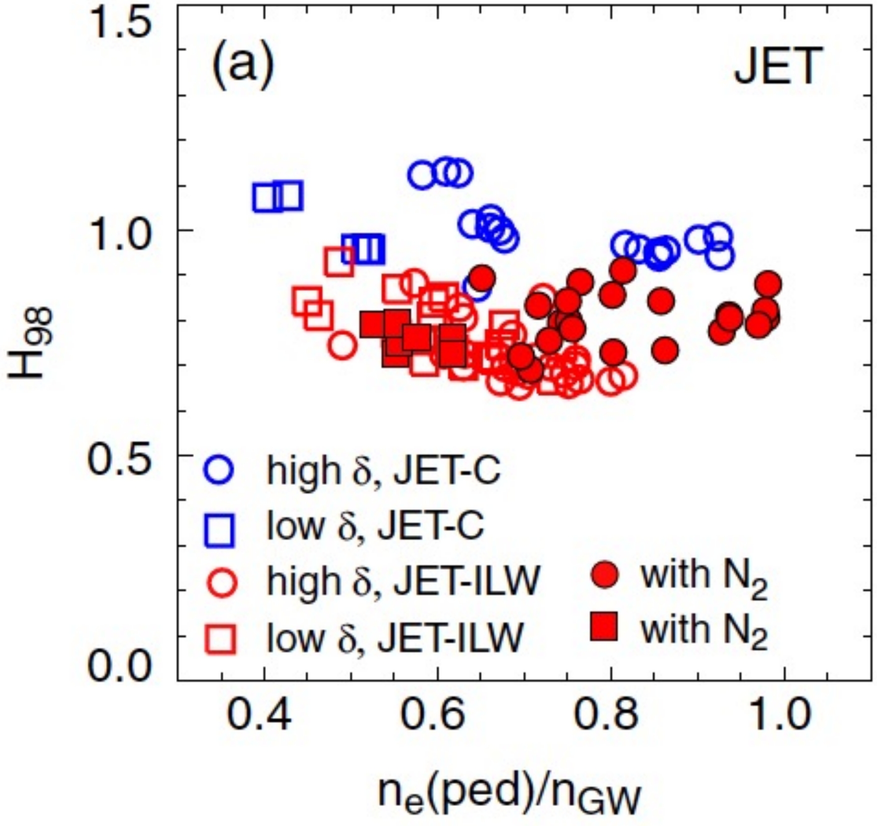
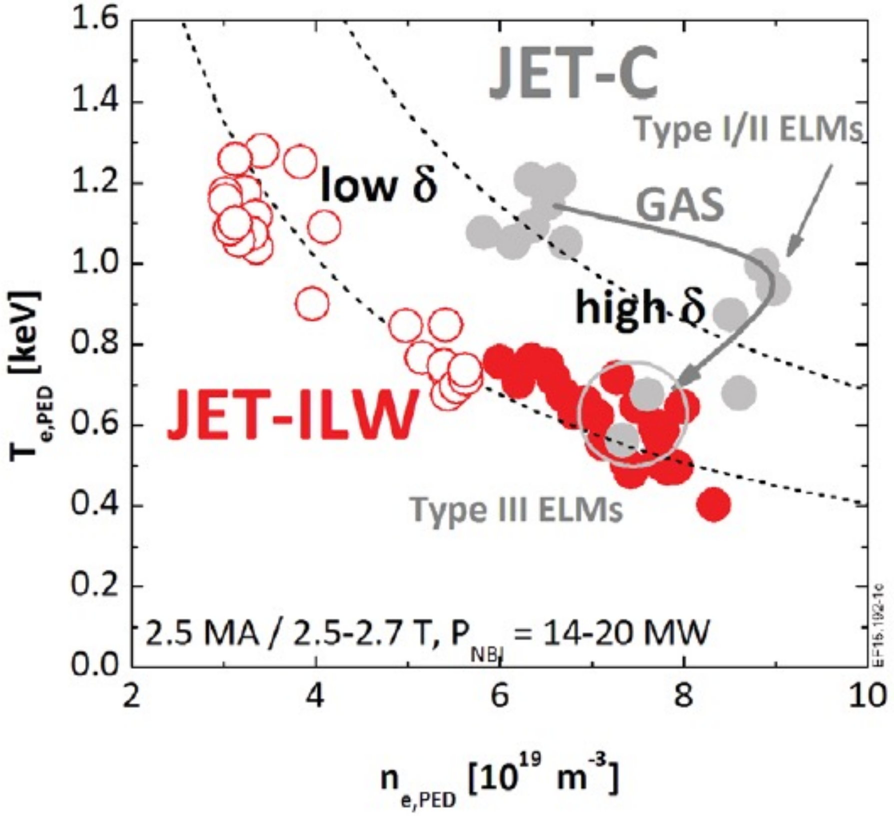
Increasing Timescale

PMI Challenge: Core-wall integration

Buerskins et al, PPCF 2013



Maggi et al, NF 2015



Wall material choice impacts core plasma via poorly understood mechanisms…need to Understand & predict

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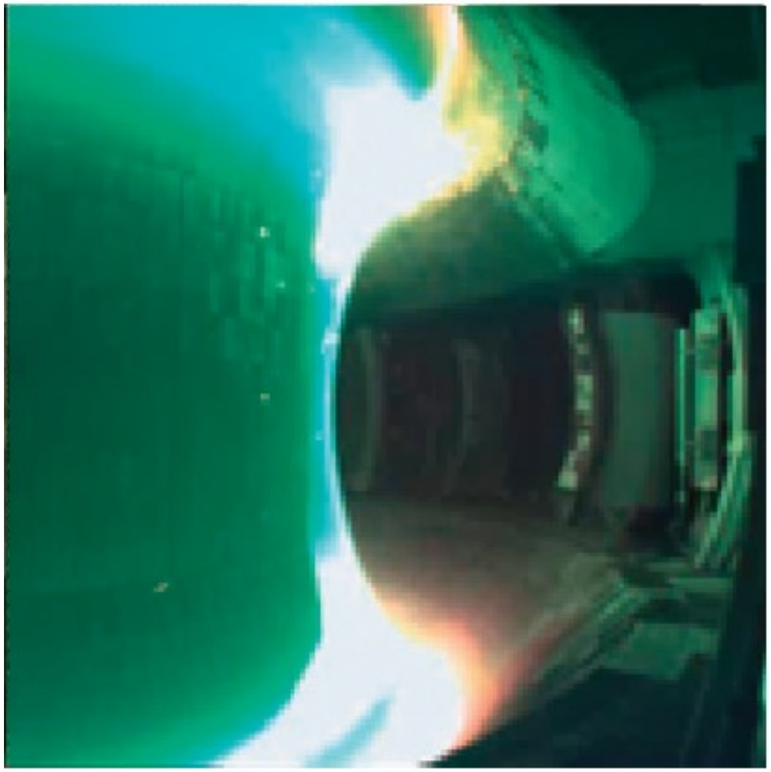
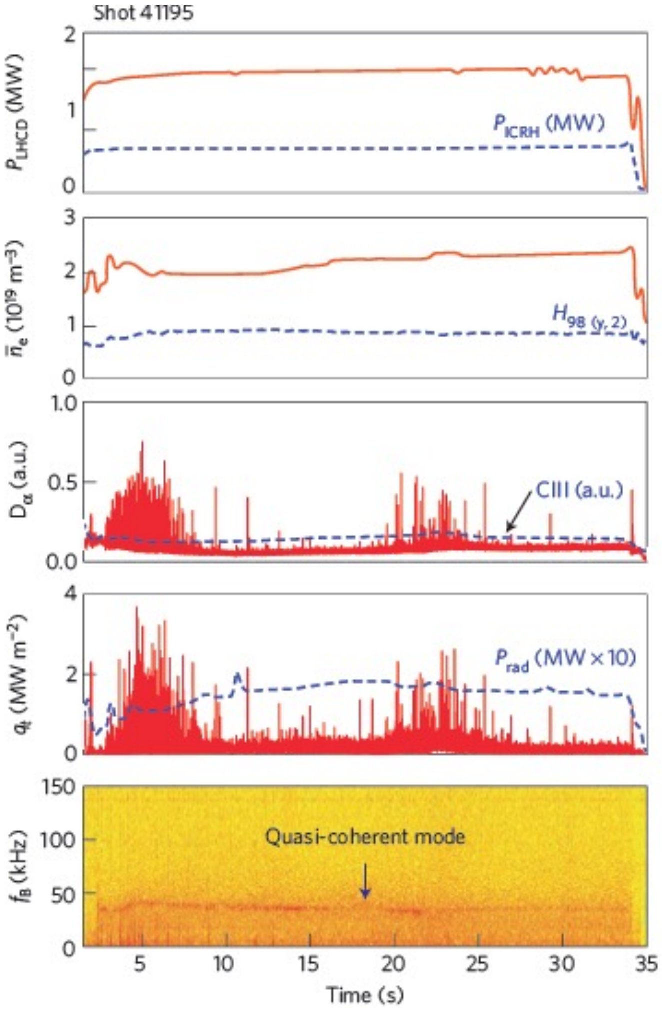
PMI Challenge: Core-wall integration in long pulses

J Li et al, Nature Phys. 2013





Low-Z Coatings (Li, B, …) inevitably used to achieve core Performance; **Questionable utility for steady-state T-breeding device!**

Research Need:

- Why do low-Z coatings have impact on core plasma performance
  - Neutral recyling effect?
  - On Pedestal Fueling?
  - On Flow Shear?
  - Something else?
- How to achieve good core performance w/o low-Z coating?
- If not, can low-Z coating be made compatible with requirements of TBR>1, T-inventory control?

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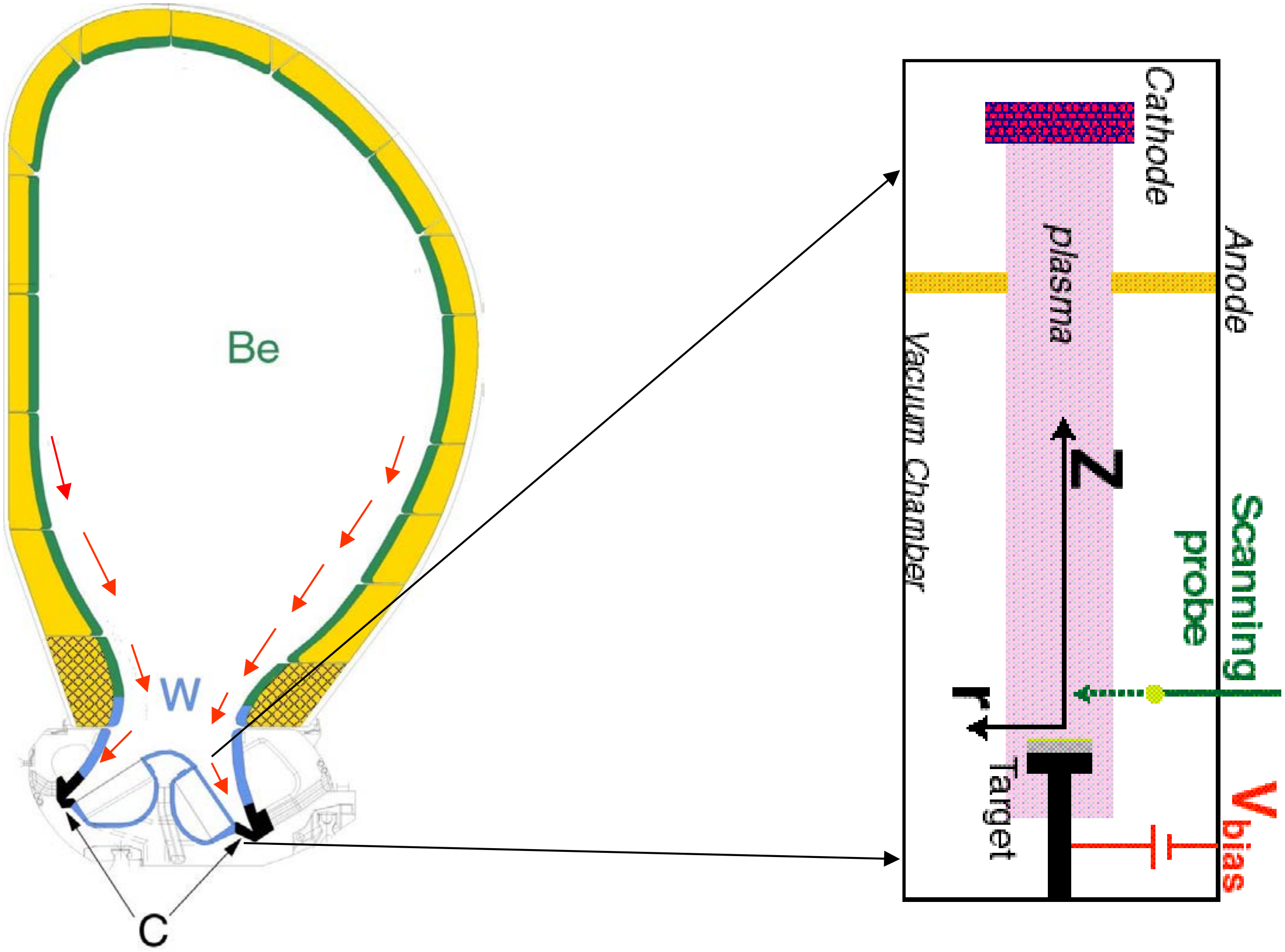
Increasing Timescale

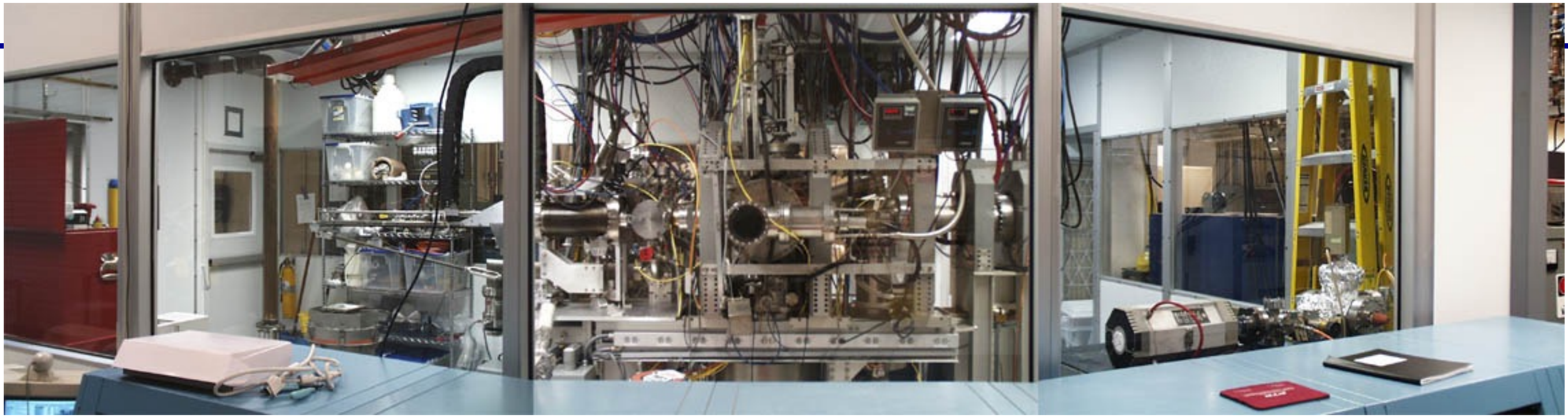
Linear plasma devices simulate many aspects of fusion PMI science



ITER/tokamak geometry

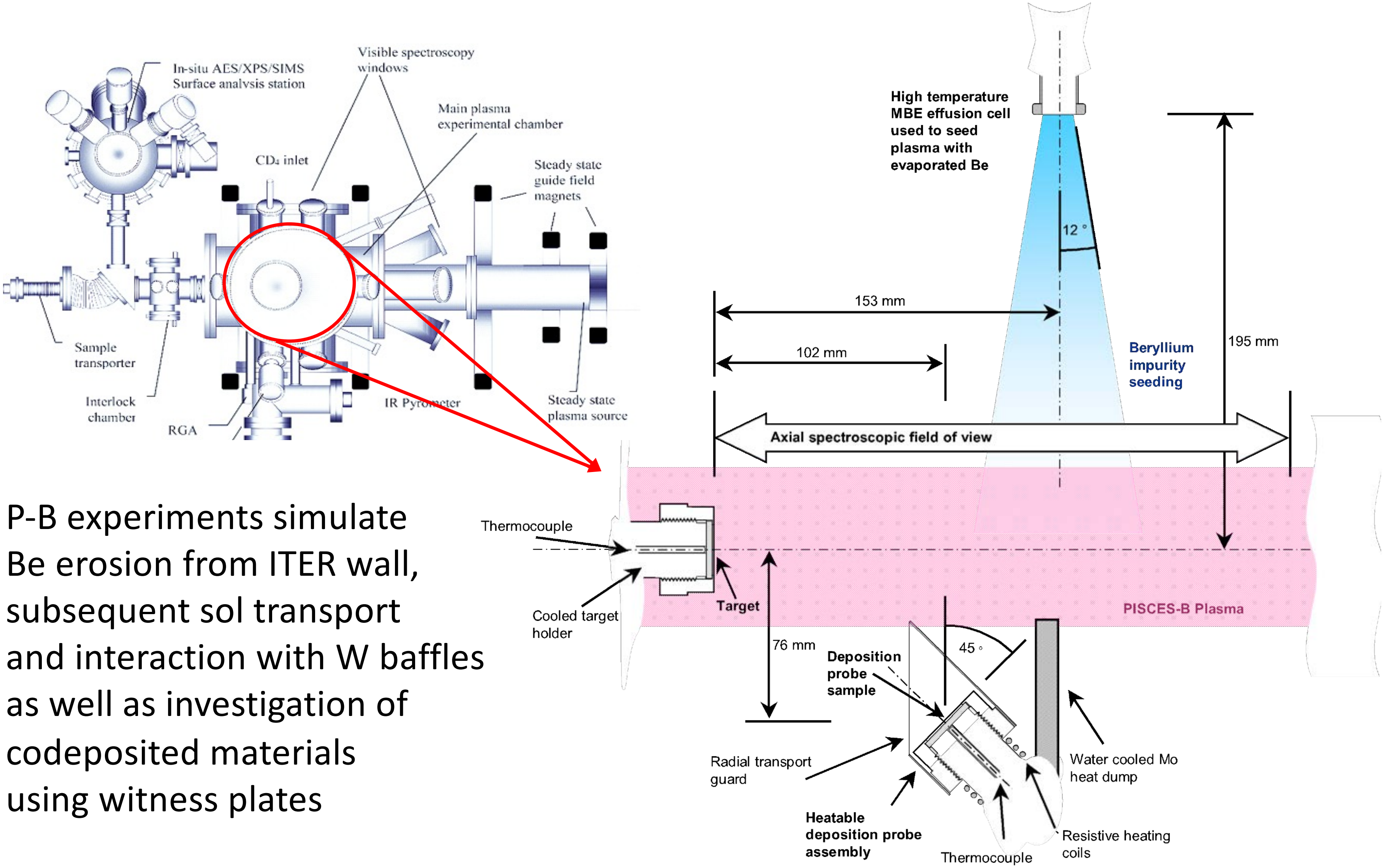
PISCES/linear device geometry

The PISCES-B facility at UCSD



- PISCES-B is located in an air-tight enclosure to allow investigation of Be
- PISCES-A is located outside the Be enclosure to allow easier non-Be investigations and to develop diagnostics for PISCES-B
- The PISCES Program routinely hosts visitors from Japan, EU, China as well as other US Fusion Laboratories

Schematic view

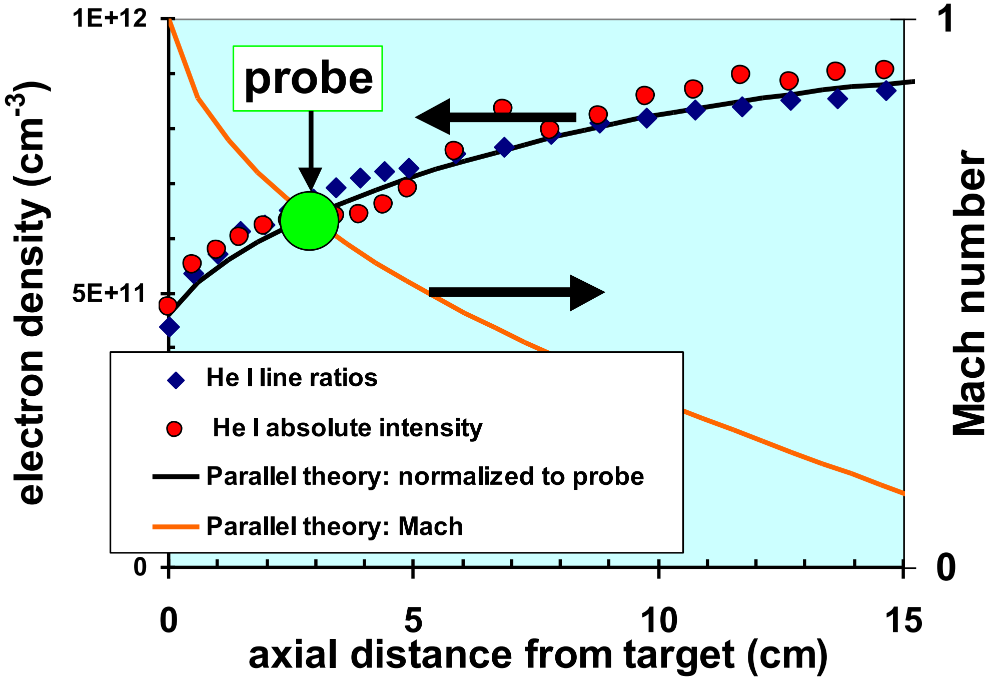


P-B experiments simulate Be erosion from ITER wall, subsequent sol transport and interaction with W baffles as well as investigation of codeposited materials using witness plates

Lab studies give comprehensive plasma, target and impurity diagnostics.

- Plasma impurity concentration
  - calibrated spectroscopy
  - magnetically shielded RGA
  - material surface analysis
  - LIBS surface contamination
- Erosion yield
  - weight loss
  - calibrated spectroscopy
  - full 3-D modeling
- Ion flux by target bias current and probe measurements
- Sample temperature by IR pyrometers and thermocouples

Plasma density is measured by a reciprocating Langmuir probe, He line ratios, absolute He line intensity and compared to parallel sheath theory.



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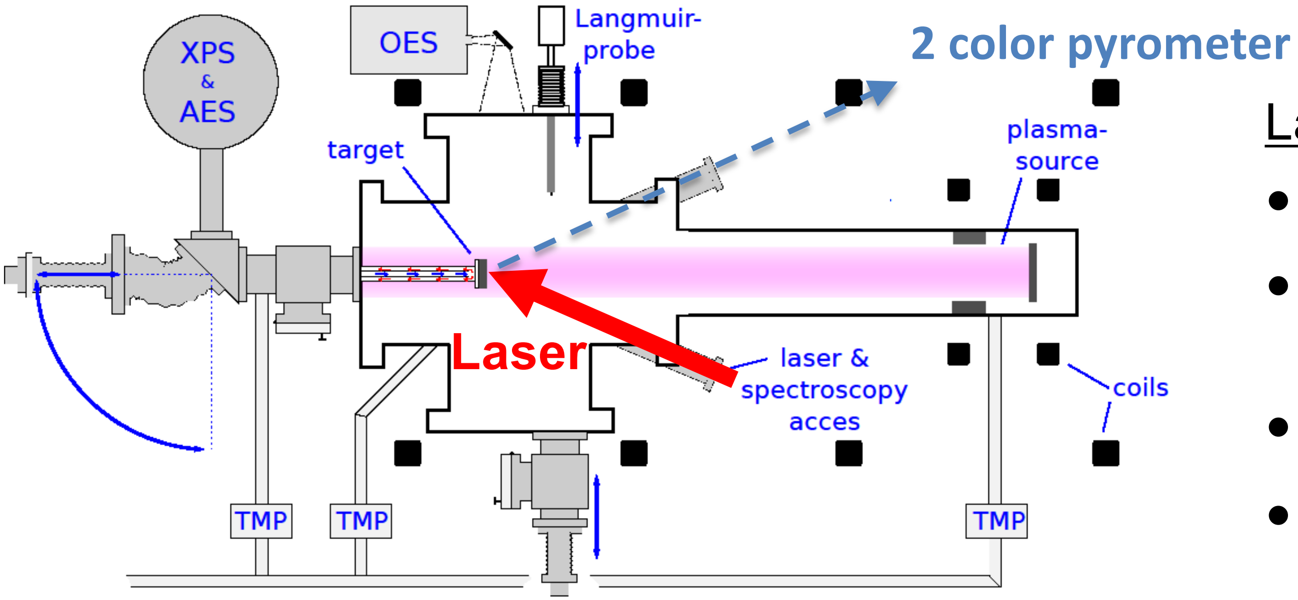
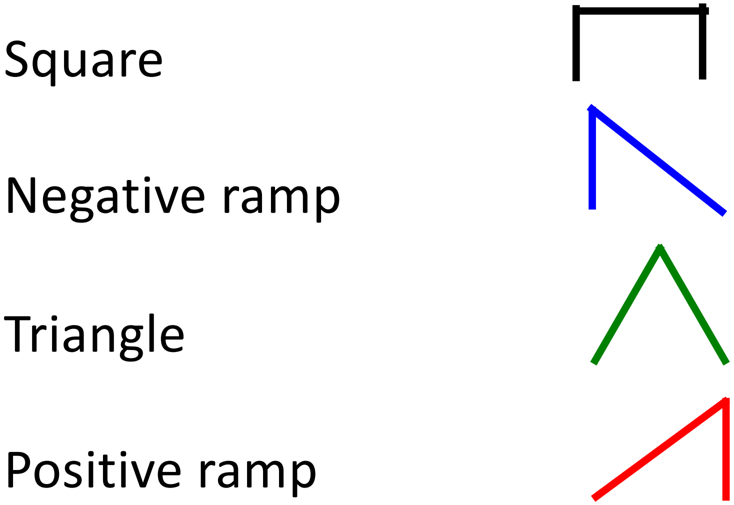
Thermal transient (e.g. ELMs) effects on W surfaces



Laser heating

- Nd:YAG 1064 nm laser
- <2 GW/m^2 of absorbed power density
- Pulse width 1 to 10 ms
- $N_{cycles} = 100$

Laser pulse shape can be controlled. Four shapes were investigated.



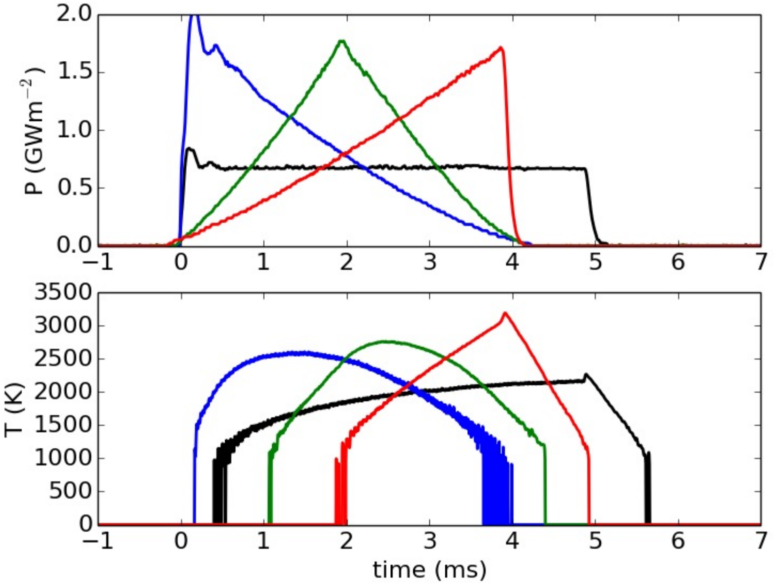
Square

Negative ramp

Triangle

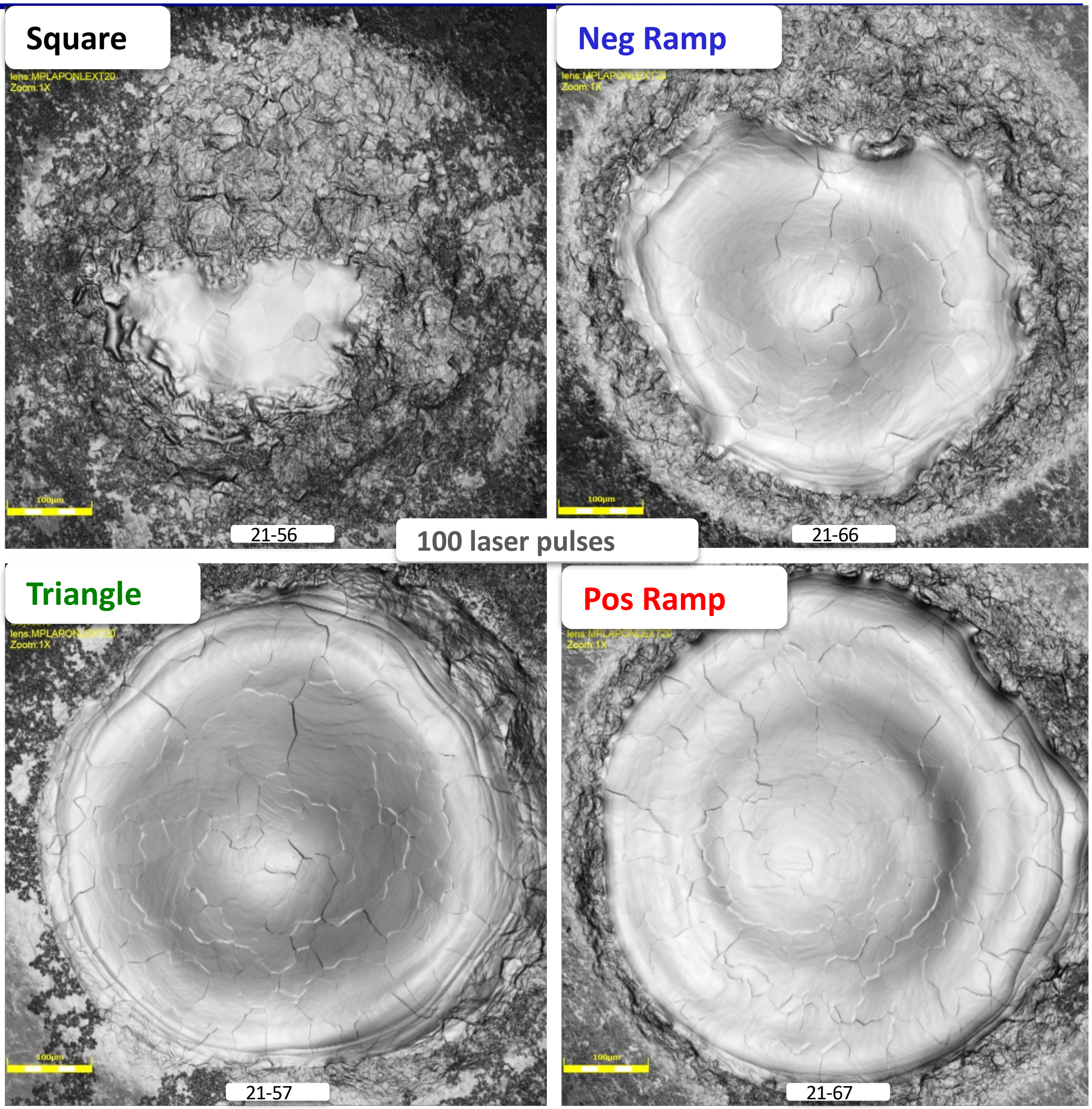
Positive ramp

Damage depends on heat pulse shape



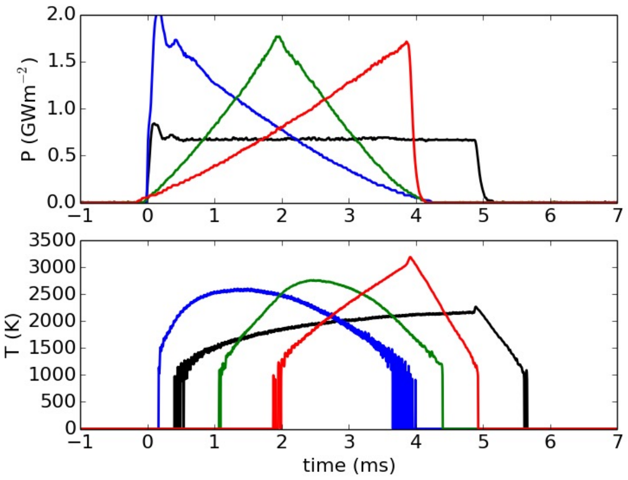
Pulse shape	T_{peak} (K)
Square	2280
Negative Ramp	2600
Triangle	2780
Positive Ramp	3130

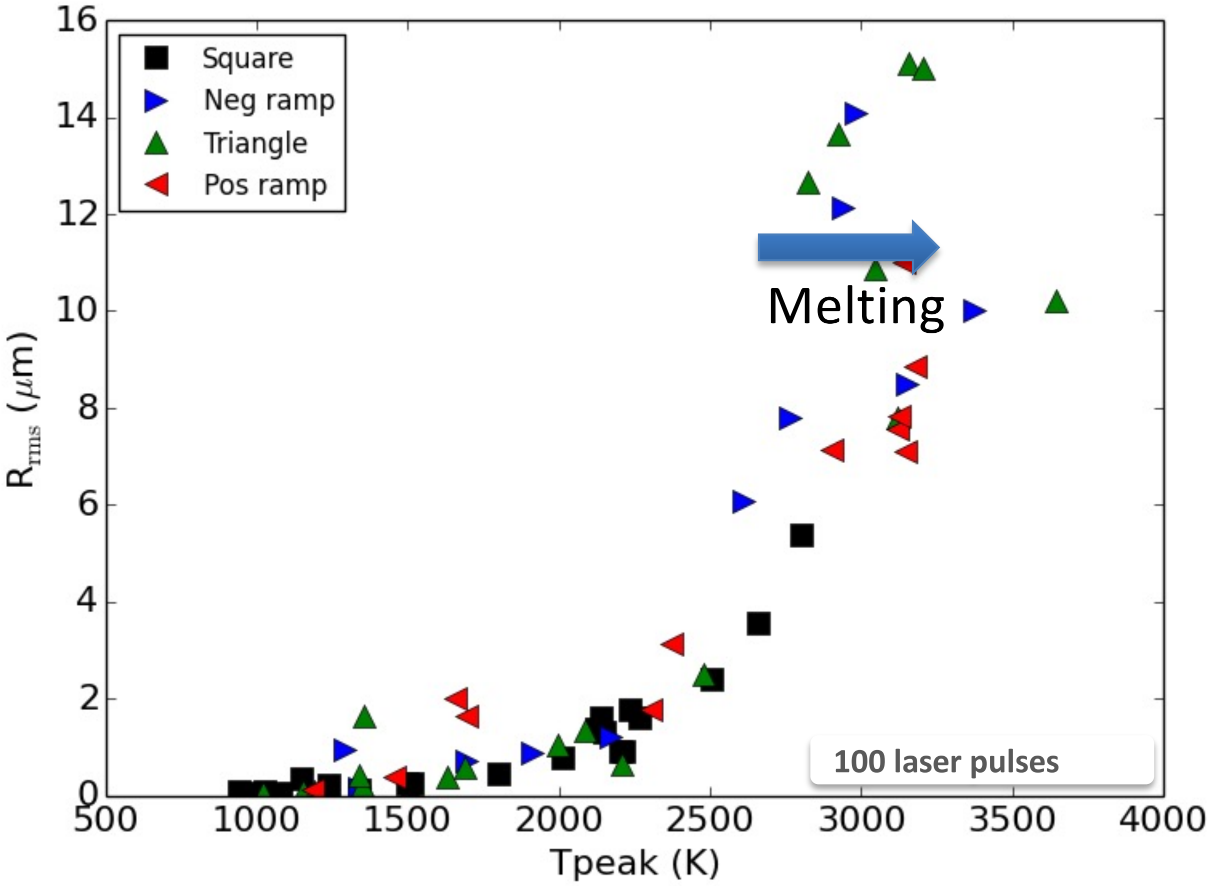
Tungsten T_{melt} = 3695 K,
absolute intensity to pyrometer
is used to compare surface temperature
due to different pulse shapes
(underestimates temperature)



100 laser pulses

Damage correlated w peak surface temperature





Melting

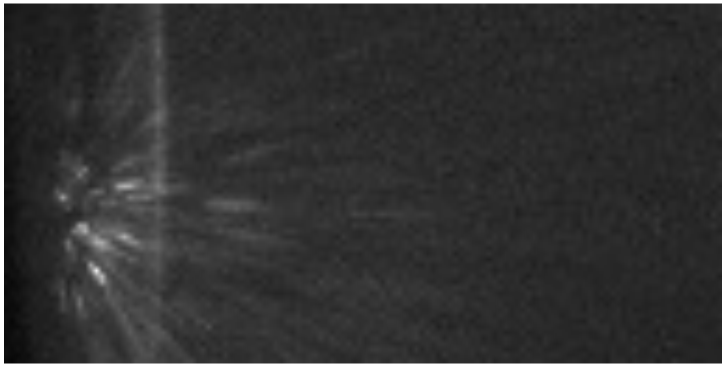
100 laser pulses

It is important to accurately predict and model ELM shapes in ITER to understand the response of the tungsten divertor plates to repetitive thermal cycling.

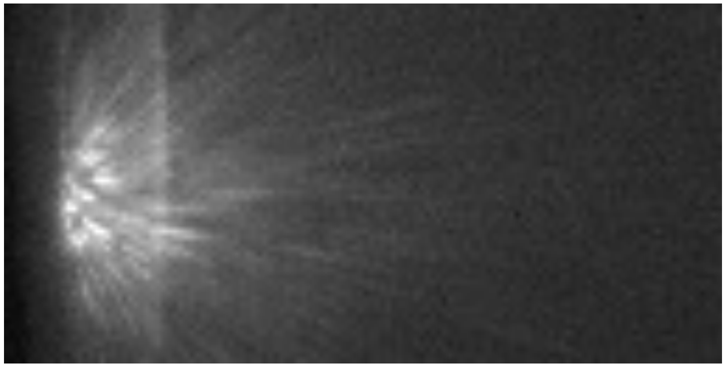
Plasma-implanted D also affects W surface damage

(from K. Umstadter et al., NF 51(2011)053014)

SAMPLE



F = $5^x10^{22}/m^2$



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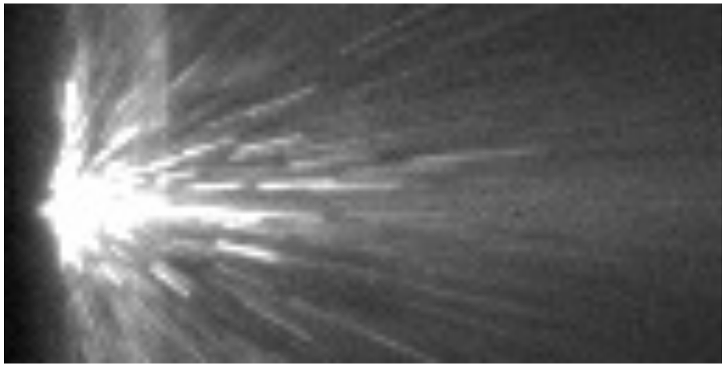
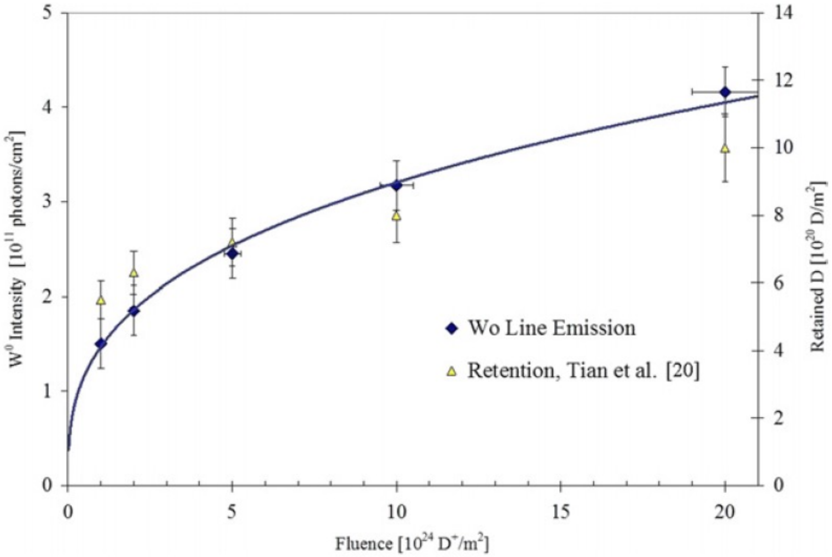


F = $2^x10^{24}/m^2$

Fluence to surface before laser pulse varied

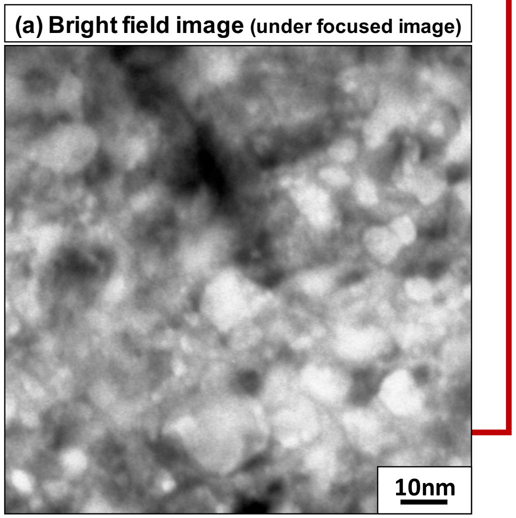
Absorbed
Energy Impact
~45 MJ/m^2 s$^{1/2}$
($\mathscr{R}_{W\ (\lambda=1064nm)}$ ~ 70%)

V_{bias}= -125V
Γ=$2x10^{22}/m^2$-sec
T_e=11eV
n_e=$2x10^{24}/m^3$
T_s ~ 50°C



W Temperature Influences PMI Effects

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



PISCES-A: D_2-He plasma

M. Miyamoto et al. NF (2009) 065035

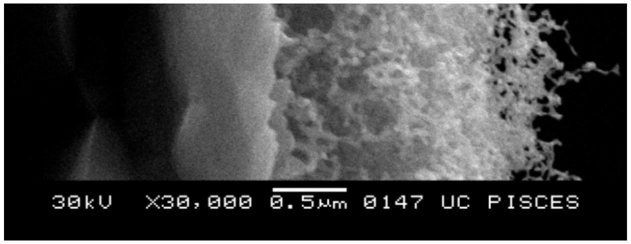
600 K, 1000 s, 2.0x10^{24} He^+/m^2, 55 eV He^+

- Little morphology
- Occasional blisters

PISCES-B: pure He plasma

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

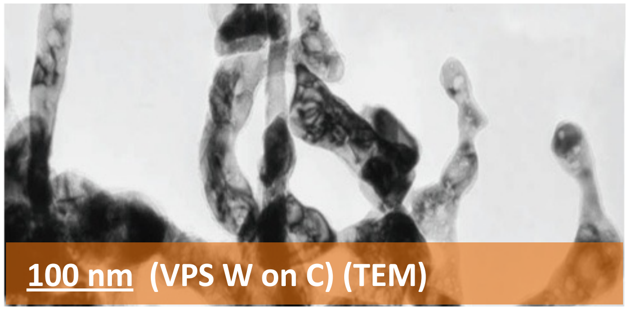
1200 K, 4290 s, 2x10^{26} He^+/m^2, 25 eV He^+



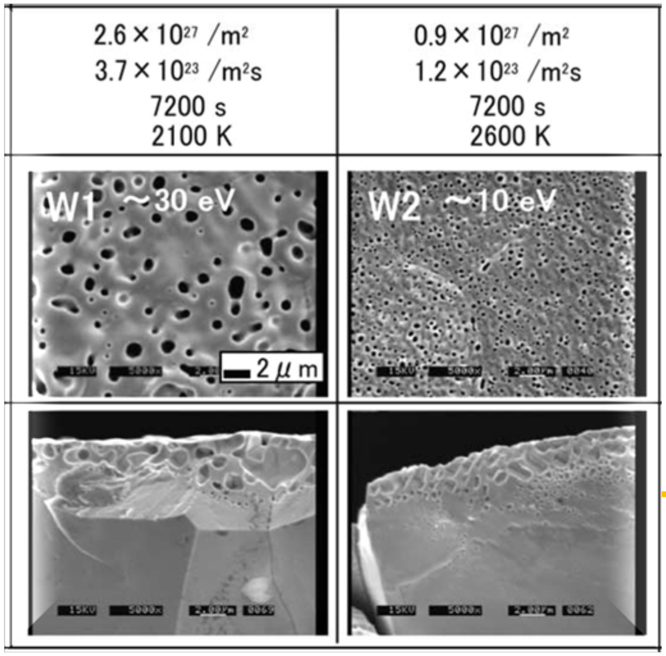
NAGDIS-II: pure He plasma

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

1250 K, 36000 s, 3.5x10^{27} He^+/m^2, 11 eV He^+



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



NAGDIS-II: He plasma

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

- Surface morphology
- Shallow depth
- Micro-scale

Fuzz growth consumes W bulk

EDX reveals indications of plasma interaction only with top-most fuzz structures (A). Interface between fuzz and bulk (B) shows no sign of plasma interaction. Fuzz forms from growth, not redeposition. No mass change to samples.

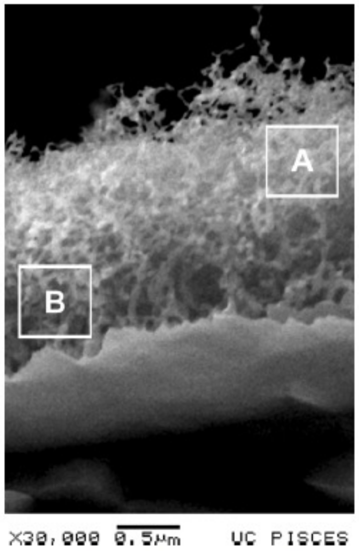
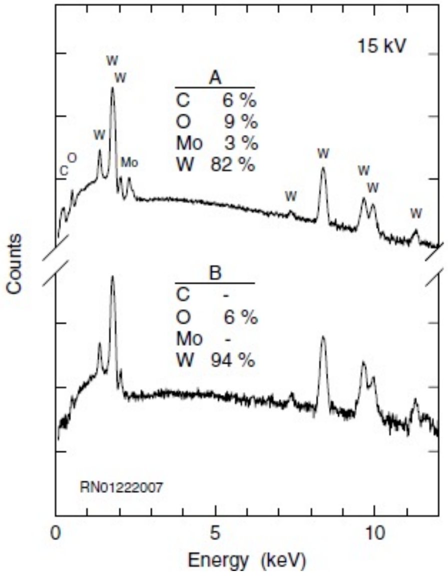
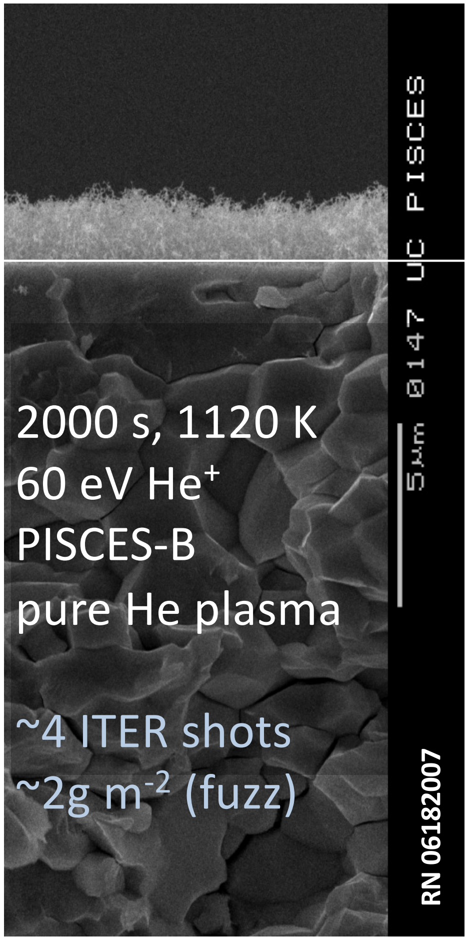


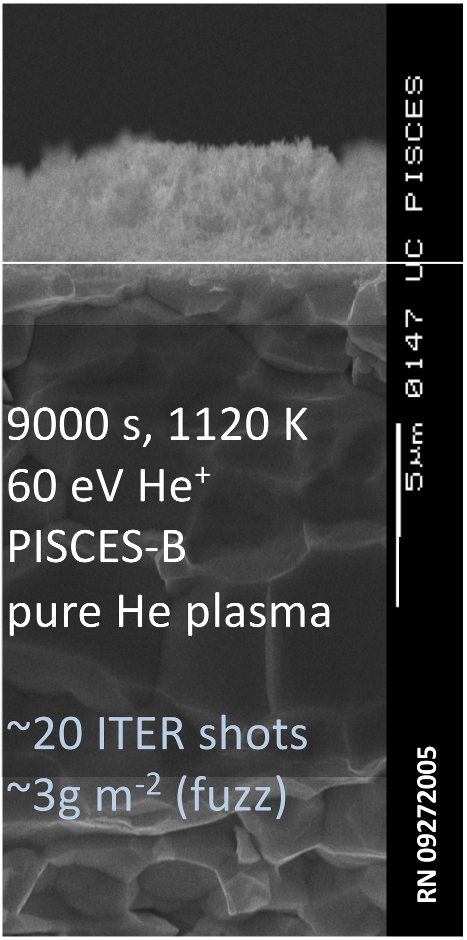
Figure 1. Cross-sectional SEM micrograph of a W target that was exposed at 1120 K to pure He plasma for 4.3 × 10³ s.





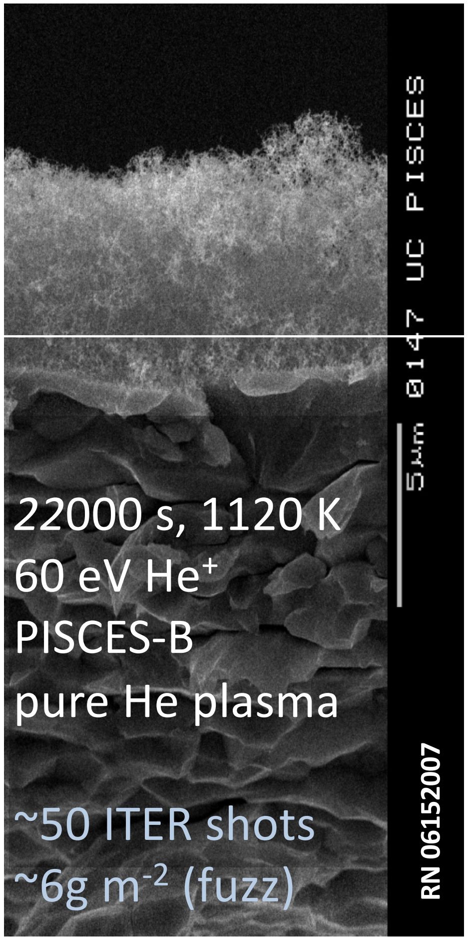
2000 s, 1120 K
60 eV He^+
PISCES-B
pure He plasma

~4 ITER shots
~2g m^{-2} (fuzz)



9000 s, 1120 K
60 eV He^+
PISCES-B
pure He plasma

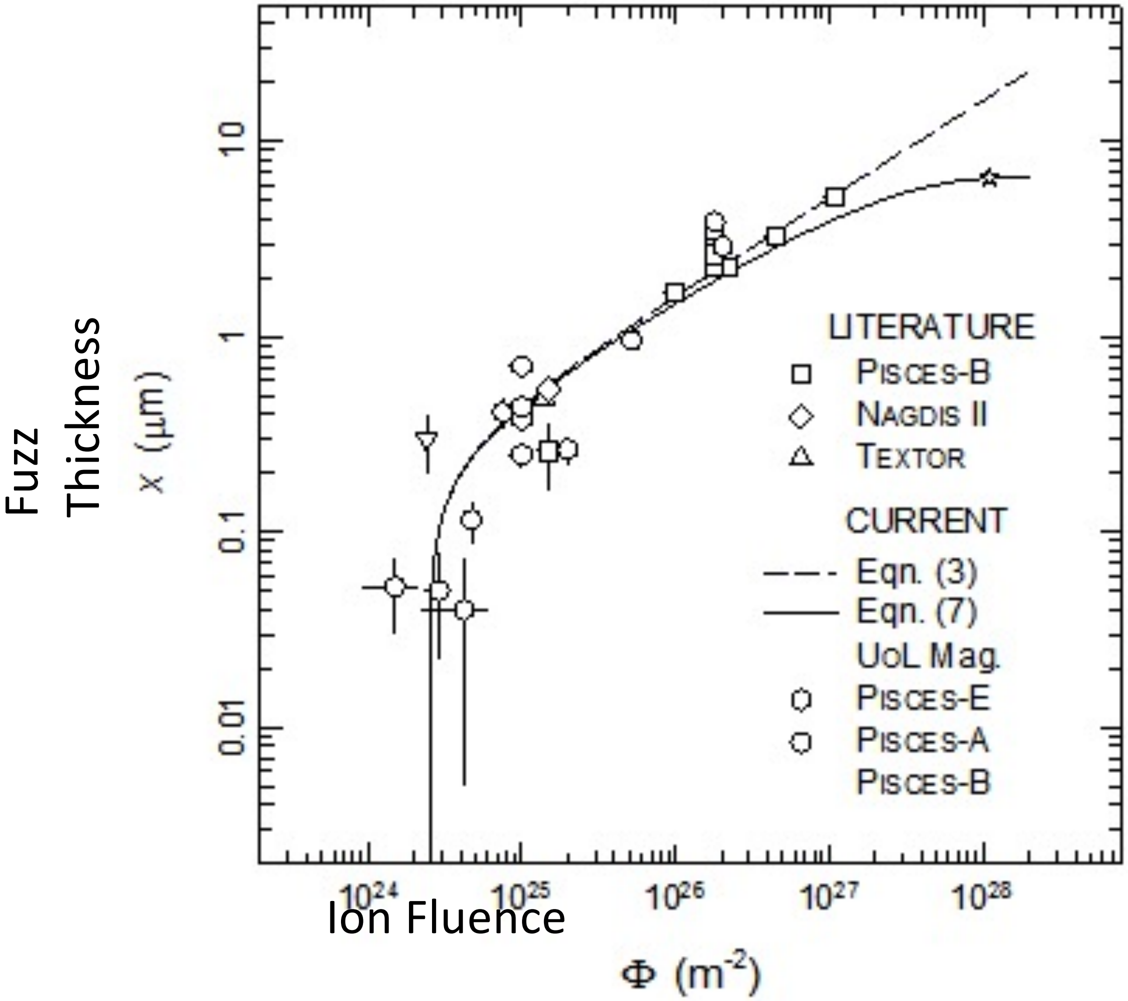
~20 ITER shots
~3g m^{-2} (fuzz)



22000 s, 1120 K
60 eV He^+
PISCES-B
pure He plasma

~50 ITER shots
~6g m^{-2} (fuzz)

Analytic model captures basic physics



PMI Issue	Reactor Impact	Research Need
Divertor particle & power handling	Dissipate divertor thermal loads ,density control	Edge/SOL transport physics; advanced divertors, transient control
PMI Impact on Confinement	Maintain core plasma performance	Long pulse (1000s seconds) tokamak w/ *CFETR relevant* wall conditions
Surface Morphology Evolution	Loss of performance at high heat flux; dust generation	Understand mechanisms & manage/avoid deleterious conditions
Helium Accumulation	**Effect on D/T Retention, Material performance**	**In-situ real-time diagnostic for He , D content;**
Fuel Retention Probability ~10^{-6}-10^{-7}	TBR>1	In-situ real-time D, T profiles over <10microns;also need He profiles since He affects retention
Surface Erosion <~ 1mm/year requires $Y_{net}<10^{-5}$	Wall & Divertor Reliability & Lifetime	In-situ diagnostics Sensitive to ~100's nm over 10micron dynamic range
Material Migration & Mixed Material Formation	Minimize & Predict evolution of mixed materials	2D SOL Plasma Flows; in-situ mixed material diagnostics
Rad-damage & Transmutation Effects	New (Degraded?) Materials Properties	Neutron surrogates; neutron irradiation; studies of In-situ retention, material properties

Increasing Timescale

Near surface He nano-bubbles form in W

300ºC

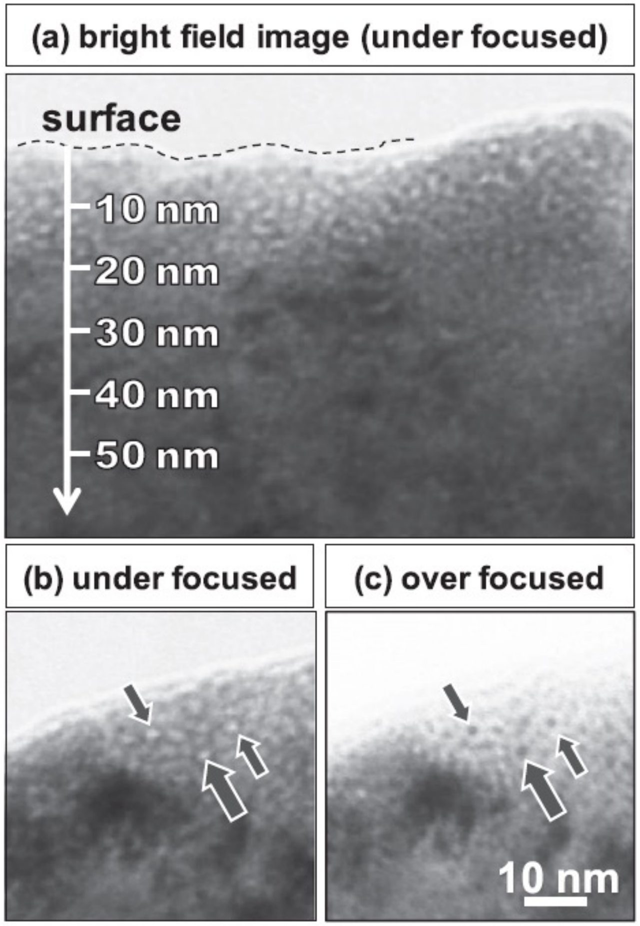


Fig. 2. Cross-sectional microstructure, observed with TEM, in the W sample exposed to D + He mixture plasma at $E_i \sim 60$ eV, $\Phi_D \sim 5 \times 10^{25}$ m^{-2}, $T_s \sim 573$ K, $c_{He} \sim 5\%$. As pointed with arrows, He bubbles have bright and dark contrasts in under (b) and over (c) focused images, respectively.

500ºC

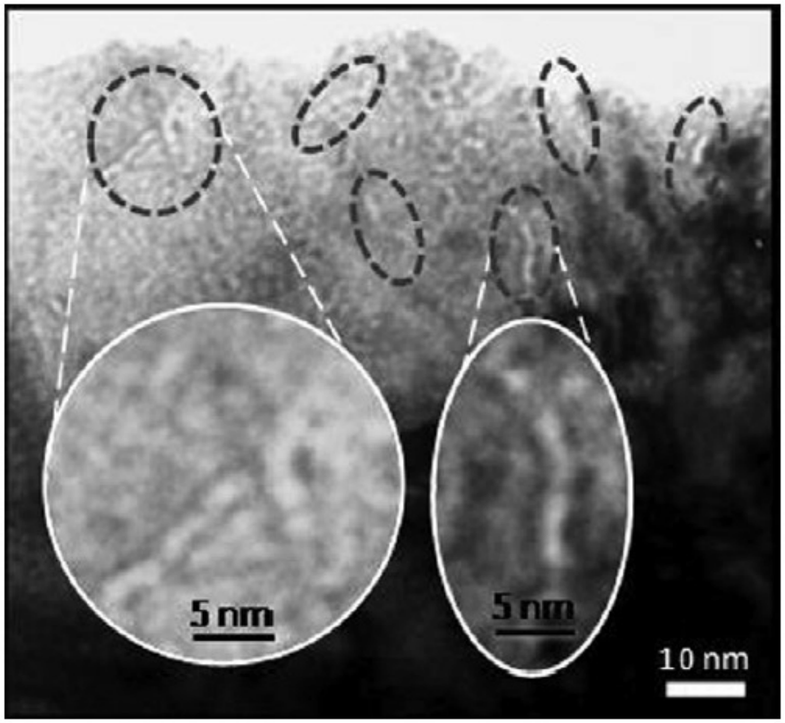


Fig. 4. Cross-sectional microstructure in the W sample exposed to D + He mixture plasma at $E_i \sim 120$ eV, $\Phi_D \sim 5 \times 10^{25}$ m^{-2}, $T_s \sim 773$ K, and $c_{He} \sim 5\%$. As seen in the circles, He bubbles interconnect and make larger clusters.

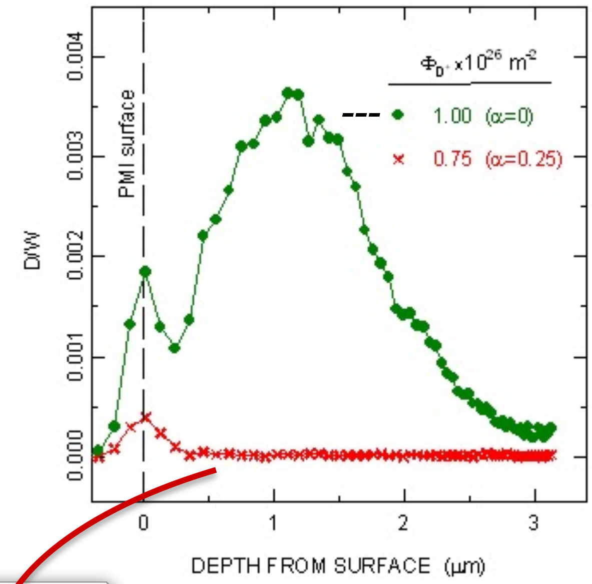
[from M. Miyamoto et al., JNM 415(2011)S657]

These He bubbles act as a diffusion barrier to D

Single step (D, He) exposure

(**D-0.25He**) T_s= 473 K, E_{ion}= 50 eV

Mixed D_2-He compared to **pure D_2**

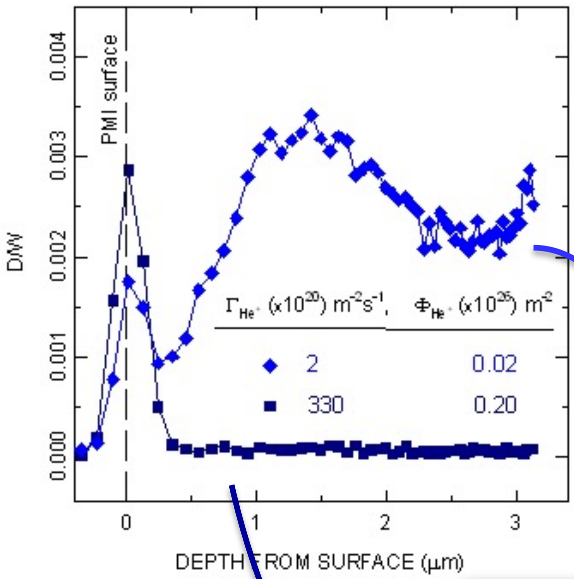


Two step, He pretreat, D plasma exposure

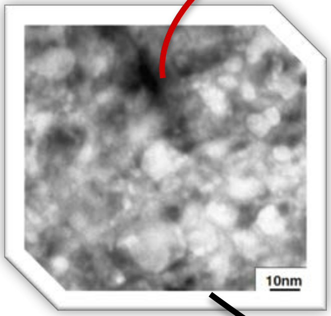
(**He**) T_s= 473 K, E_{ion}= ~30-50 eV

(**D_2**) T_s= 473 K, E_{ion}= 50 eV, Γ_{D+} = 5-8 x10^{26} m^{-2}

Low/**High** flux He prior to D_2, compared

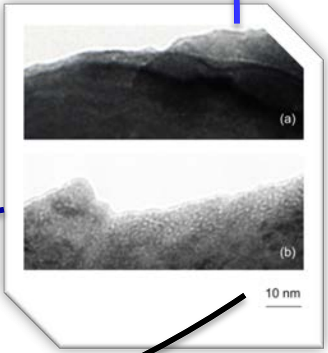


No bubbles



Bubble network provides 'return pathways' to PMI surface interrupting D migration to bulk.

[Miyamoto *et al.*, NF 49 (2009)]



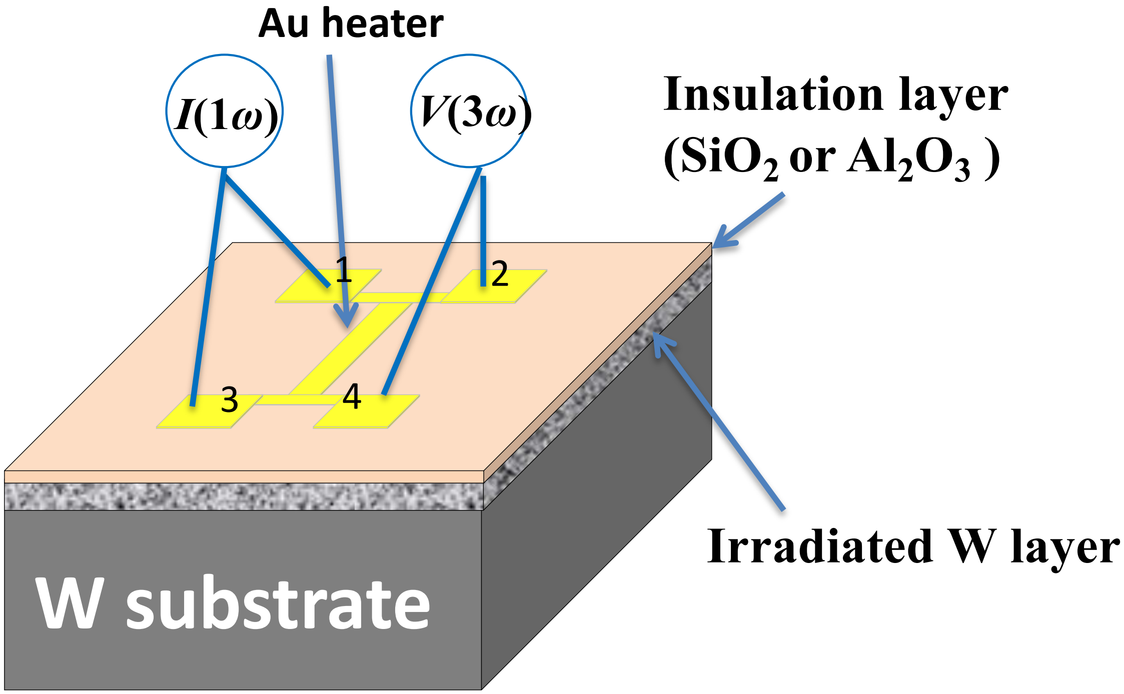
[Baldwin *et al.*, NF (2011)].

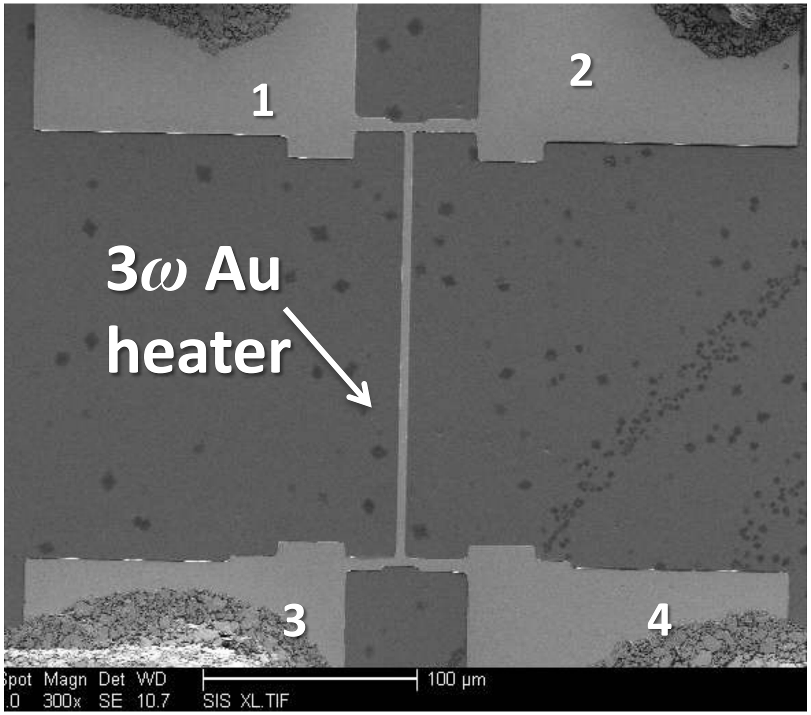
He bubbles

UC San Diego
Jacobs School of Engineering

Thermal Conductivity Measurement of Affected Zone

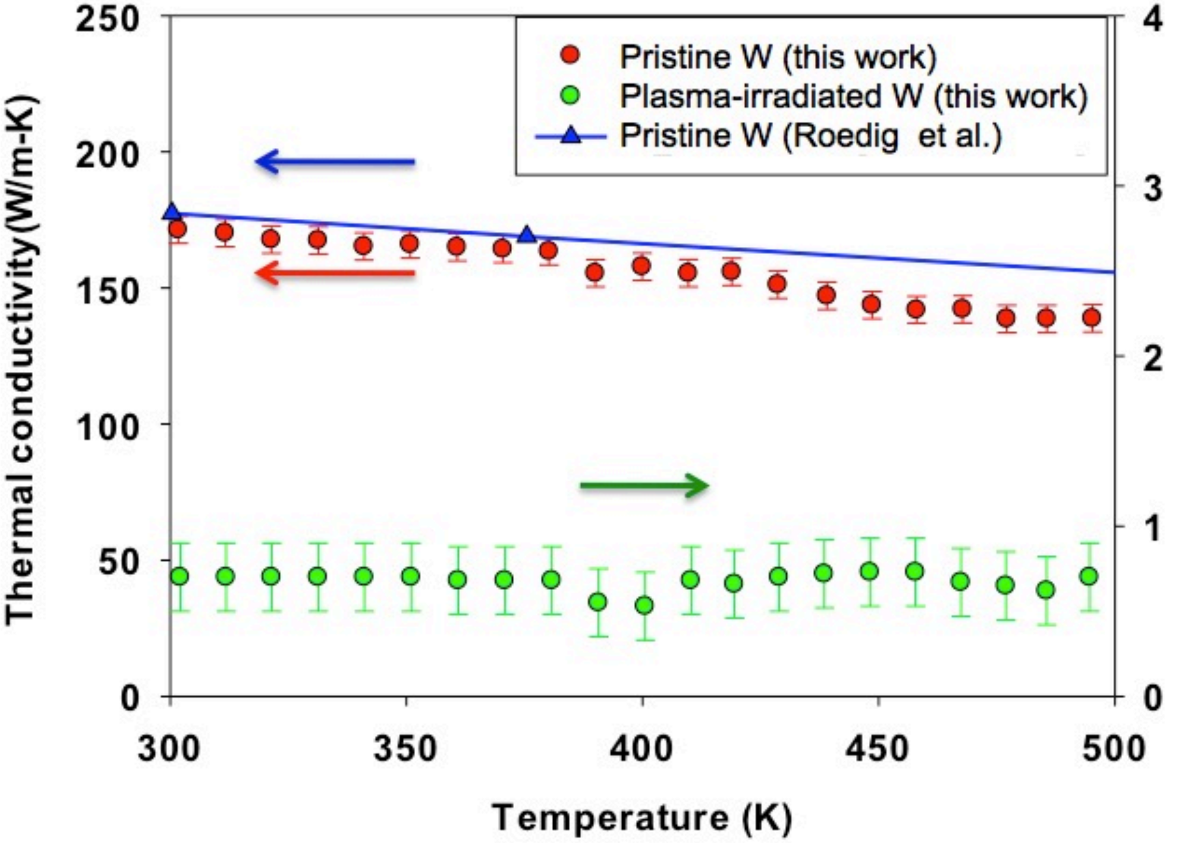
3ω method

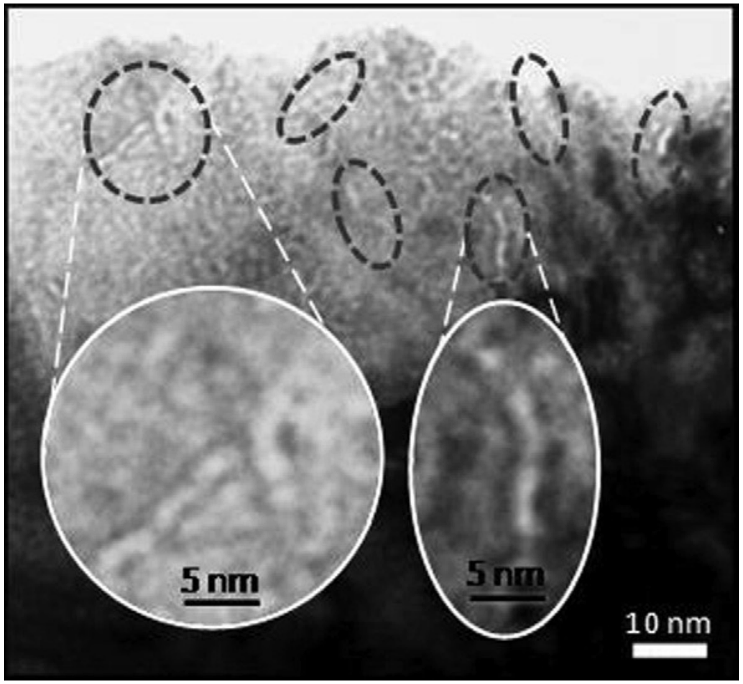




- Apply $I(\omega)$
- T oscillates at 2ω by Joule heating ($Q = I^2R$)
- R oscillates at 2ω ($R=R_o+\alpha T$)
- Can measure T rise from $V(3\omega)$
  - $V_{3\omega} = I(\omega)R(2\omega)$

Reduced Thermal Conductivity of Nanobubble layer in W





M. Miyamoto et al., JNM 415 (2011) S657

- κ of plasma-irradiated W (0.7±0.2 W.m^{-1}K^{-1}) is much lower than that of pristine W, presumably due to the defects formed during the irradiation.
- Between 300 and 500 K, κ of the plasma-irradiated W is independent of the temperature, also indicating that the electron scattering is dominated by the defects rather than phonon.

Cui, Chen, Tynan et al, J Nuc Matl 2017

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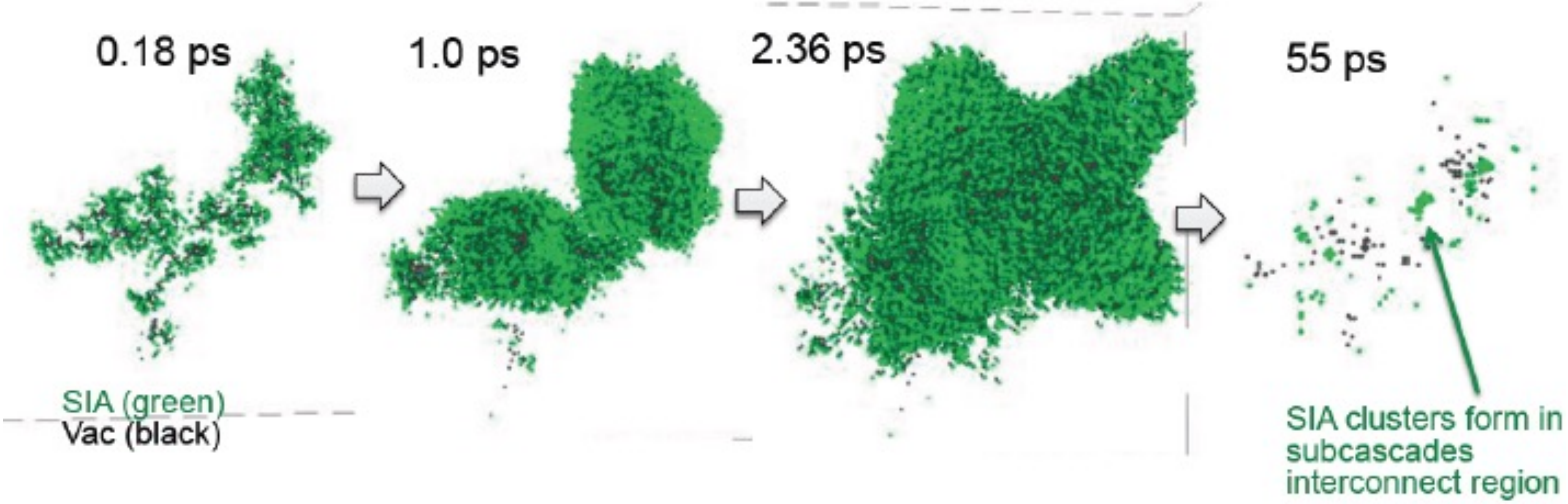
Rad-damage & Transmutation Effects	**New (Degraded?) Materials Properties**	**Neutron surrogates; neutron irradiation; studies of In-situ retention, material properties**

Increasing Timescale

Radiation damage processes impact retention

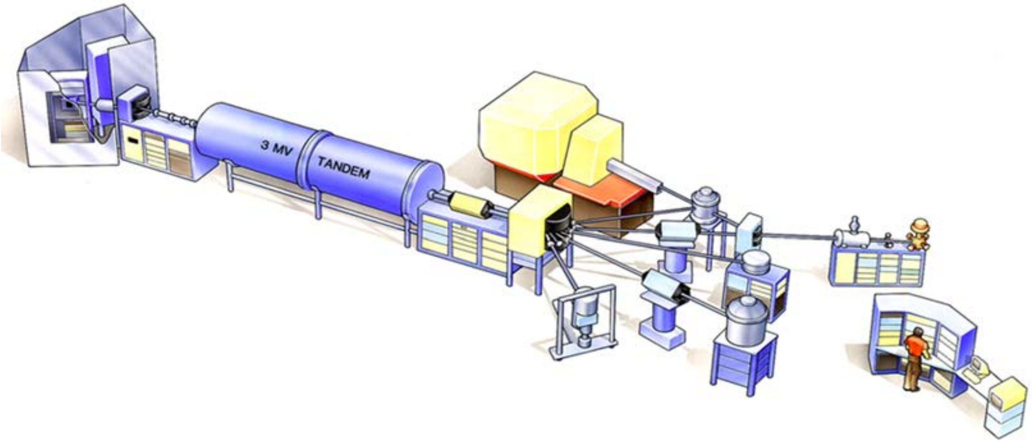
MD Simulation of displacement cascade relaxation (Wirth, private comm.)



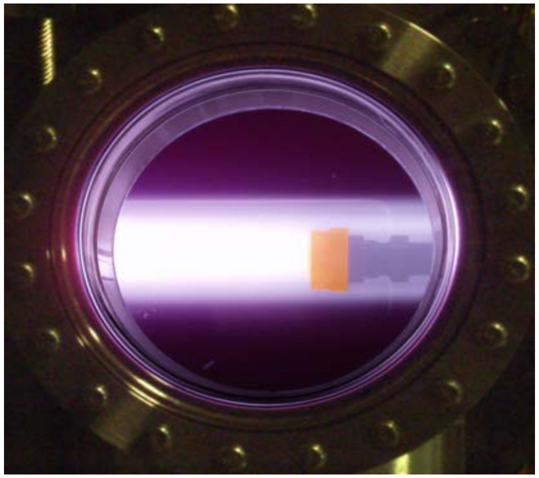
- Multiple displacement events create interstitial/vacancy pairs (Frenkel pairs)
- These have deep (>1eV) trap energies and provide sites for trapping D, T, He
- Can use MeV ion beams to replicate some aspects of this physics

D retention in displacement-damaged W

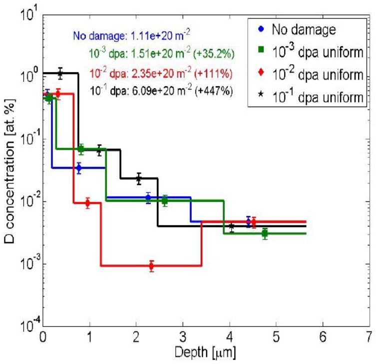
1) Induce Damage w/ Heavy Ion Beams

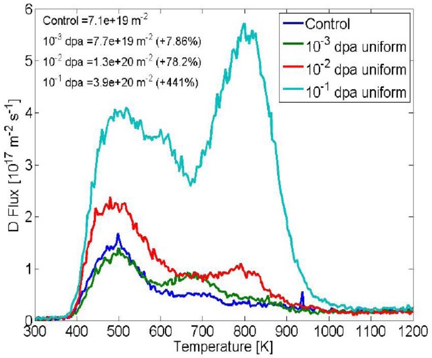


2) Implant D in Linear Plasma Device



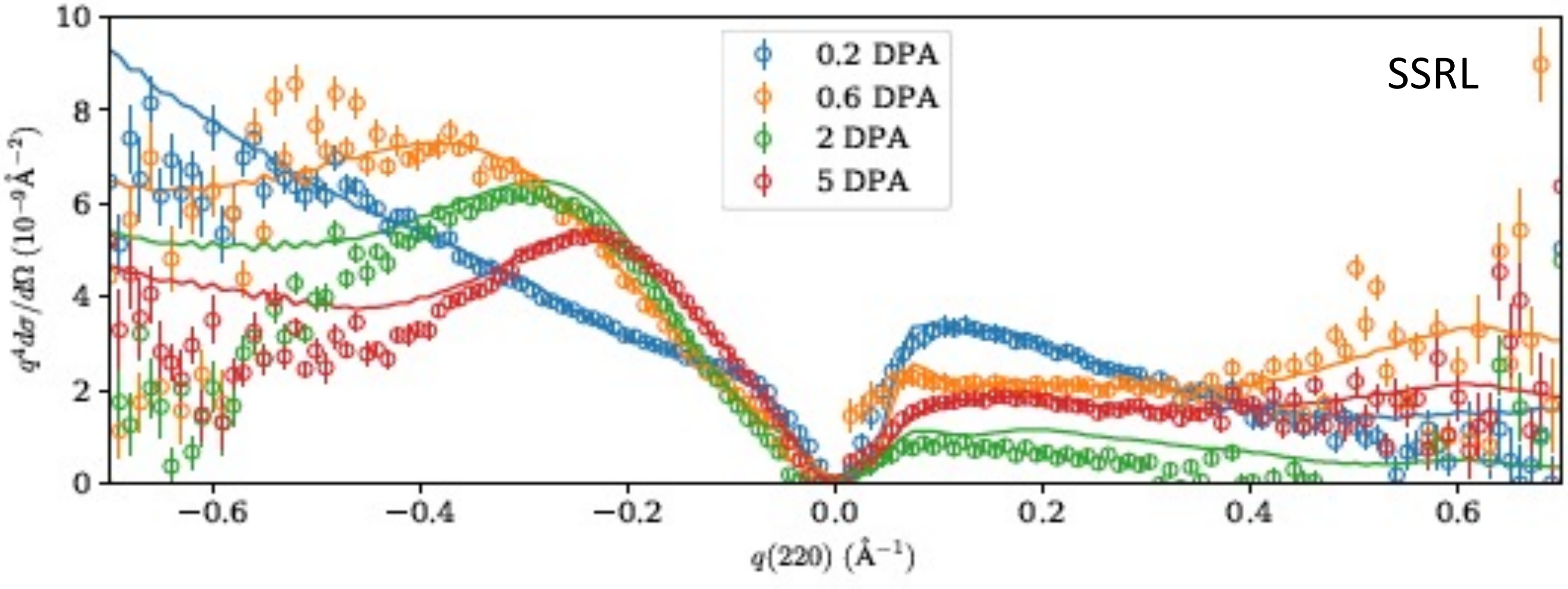
3) Measure D profile, content





Wavenumber spectrum of defect sizes in radiation-damaged W

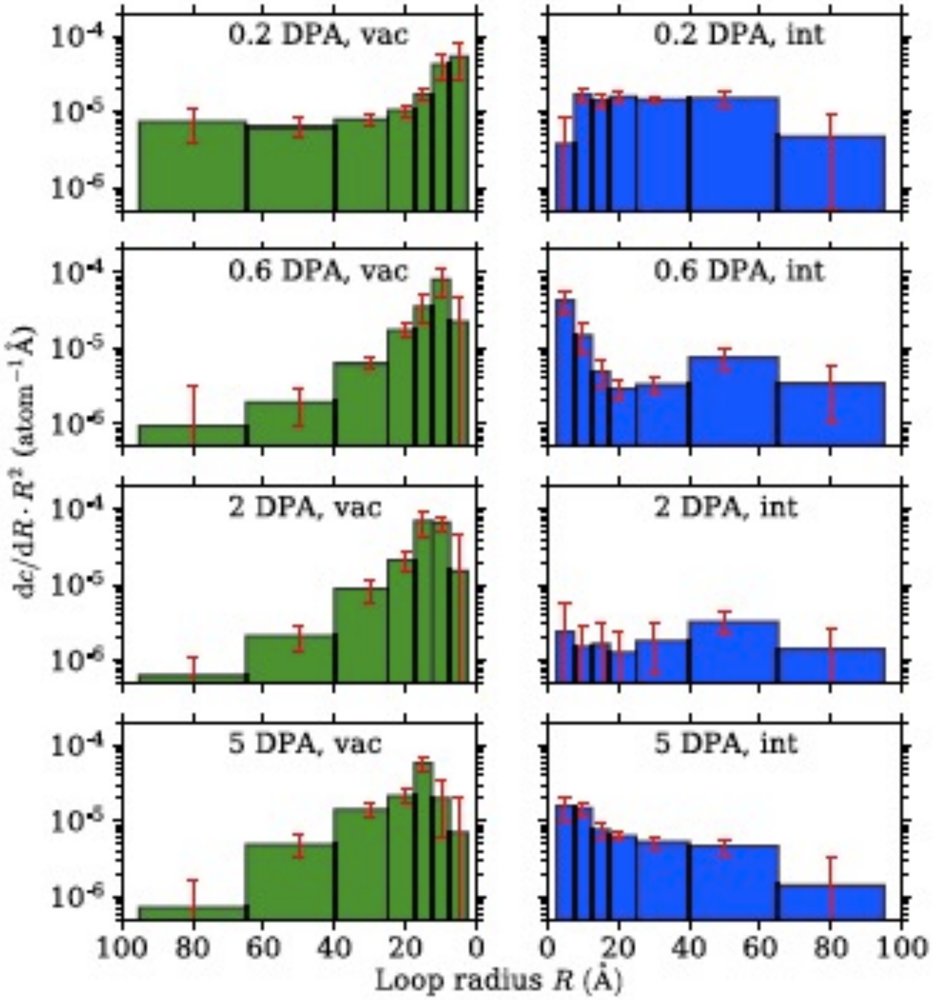
XRD Differential scattering cross-section vs defect wavenumber



P. Sun, P. Heimann, Tynan et al, J Nuc Matl's 2018

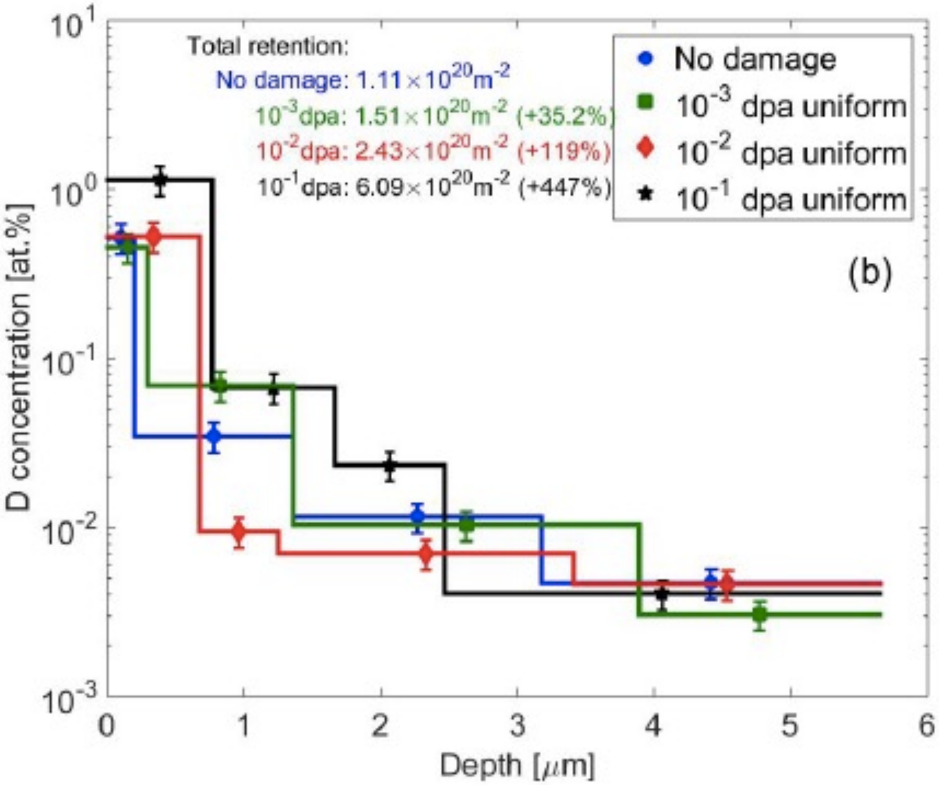
UCSD-LANL-SLAC Collaboration

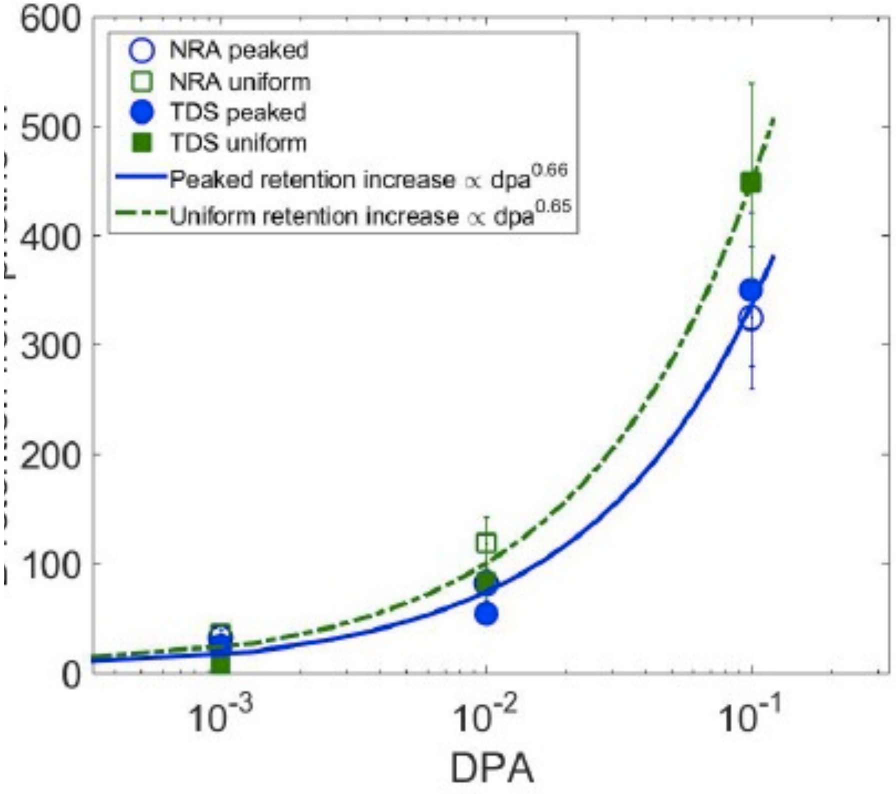
10-50 Angstrom dislocation loops dominant in ion-beam damaged W



Sun, JNM 2018

These defects trap plasma-implanted D



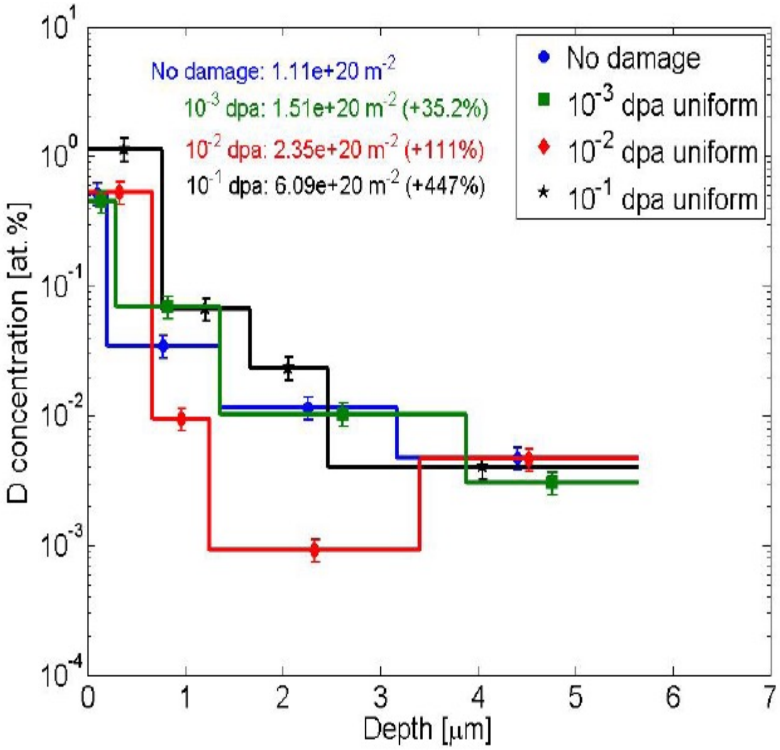


Barton, NF 2016

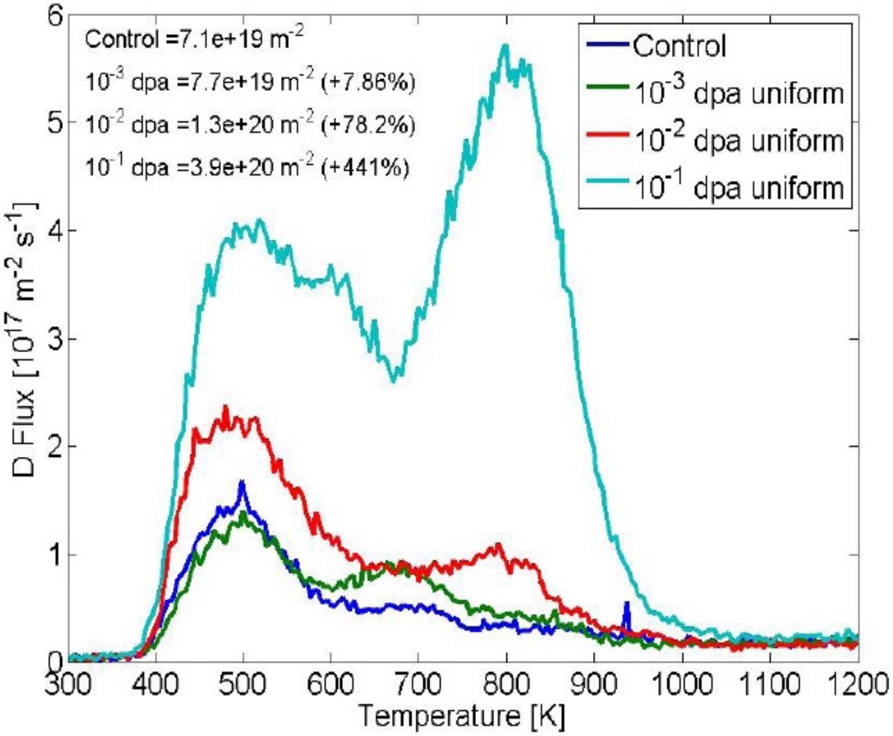
D Retention Increases with Displacement Damage

J. Barton, 2015

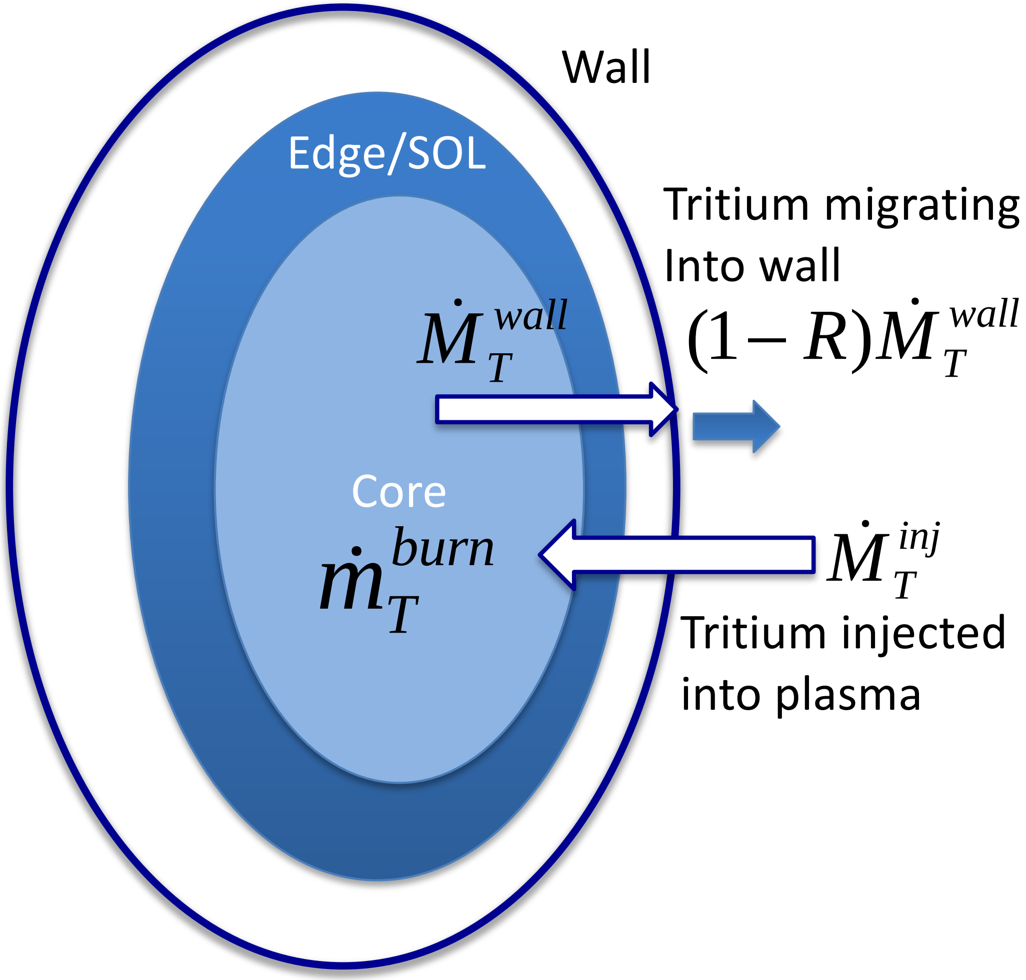
NRA Profiles



TDS Desorption



These defects can impact T self-sufficiency…



Wall

Edge/SOL

Core

$\dot{M}_T^{wall}$

$\dot{m}_T^{burn}$

Tritium migrating
Into wall

$(1-R)\dot{M}_T^{wall}$

$\dot{M}_T^{inj}$

Tritium injected
into plasma

T burnup probability, p_{burn}~0.05

Fueling efficiency, η_{fuel}~20-30%

Mass balance at wall:

$$\dot{M}_T^{inj} = \dot{m}_T^{burn} + (1-R)\dot{M}_T^{wall}$$

Mass balance –core plasma:

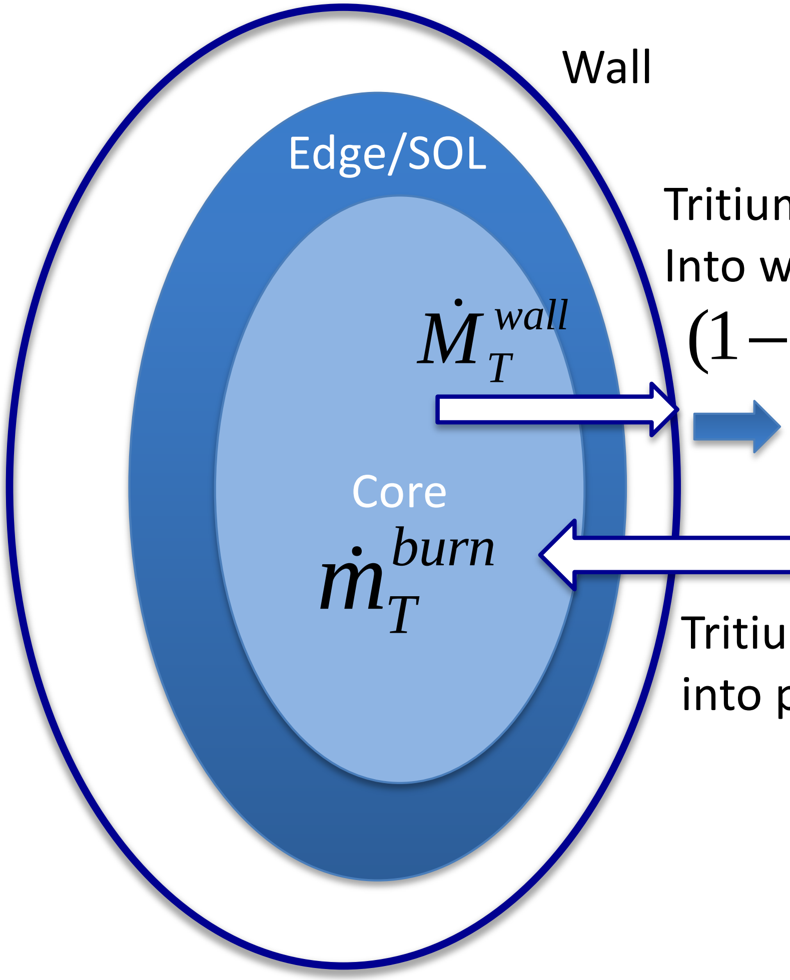
$$p_{burn}\eta_{fuel}\dot{M}_T^{inj} = \dot{m}_T^{burn}$$

Rate of T inventory build-up:

$$\Delta\dot{m}_T = (TBR-1)\dot{m}_T^{burn}$$

Tynan, Nuc Matls & Energy 2017

PMI Challenge: TBR>1 Imposes severe retention constraint



Wall

Edge/SOL

Core

$\dot{M}_T^{wall}$

$\dot{m}_T^{burn}$

Tritium migrating Into wall

$(1-R)\dot{M}_T^{wall}$

$\dot{M}_T^{inj}$

Tritium injected into plasma

To avoid impact on TBR, probability of trapping T in wall,

$$p_{trap} = \dot{M}_T^{trap} / \dot{M}_T^{wall}$$

$$p_{trap} << (TBR-1)(1-R)\frac{p_{burn}\eta_{fuel}}{1-p_{burn}\eta_{fuel}}$$

$$with \quad TBR \sim 1.05 \quad R \sim 0.99 - 0.999$$

$$\boxed{p_{trap} << 10^{-6} - 10^{-7}}$$

Tynan, Nuc Matls & Energy 2017

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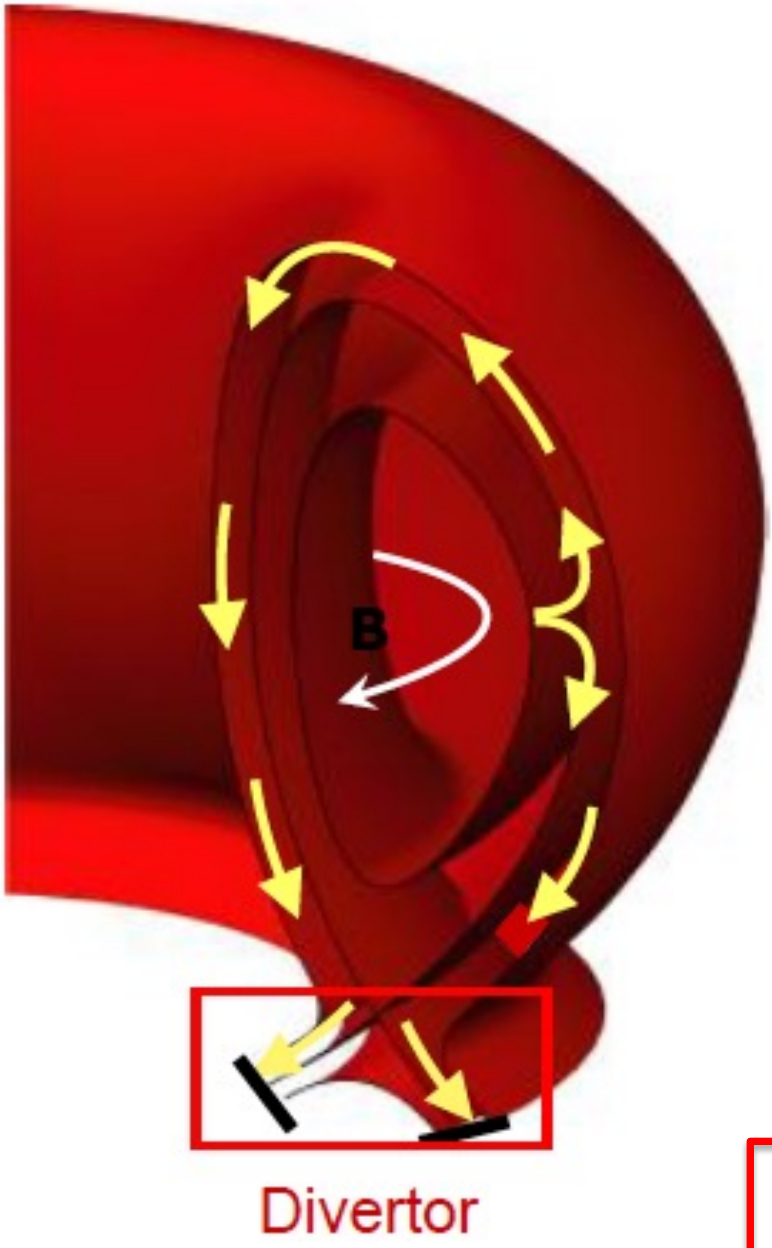
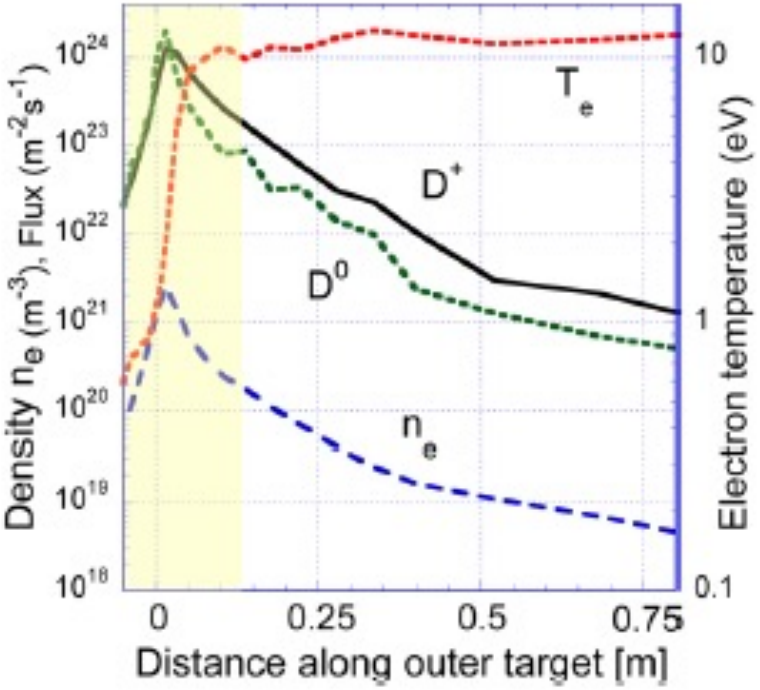
PMI Challenge: In-vessel T Inventory Control



B2-Eirene Simulations, Kikushkin



Particle Flux Into Divertor: ~10^{24}/m^2-sec

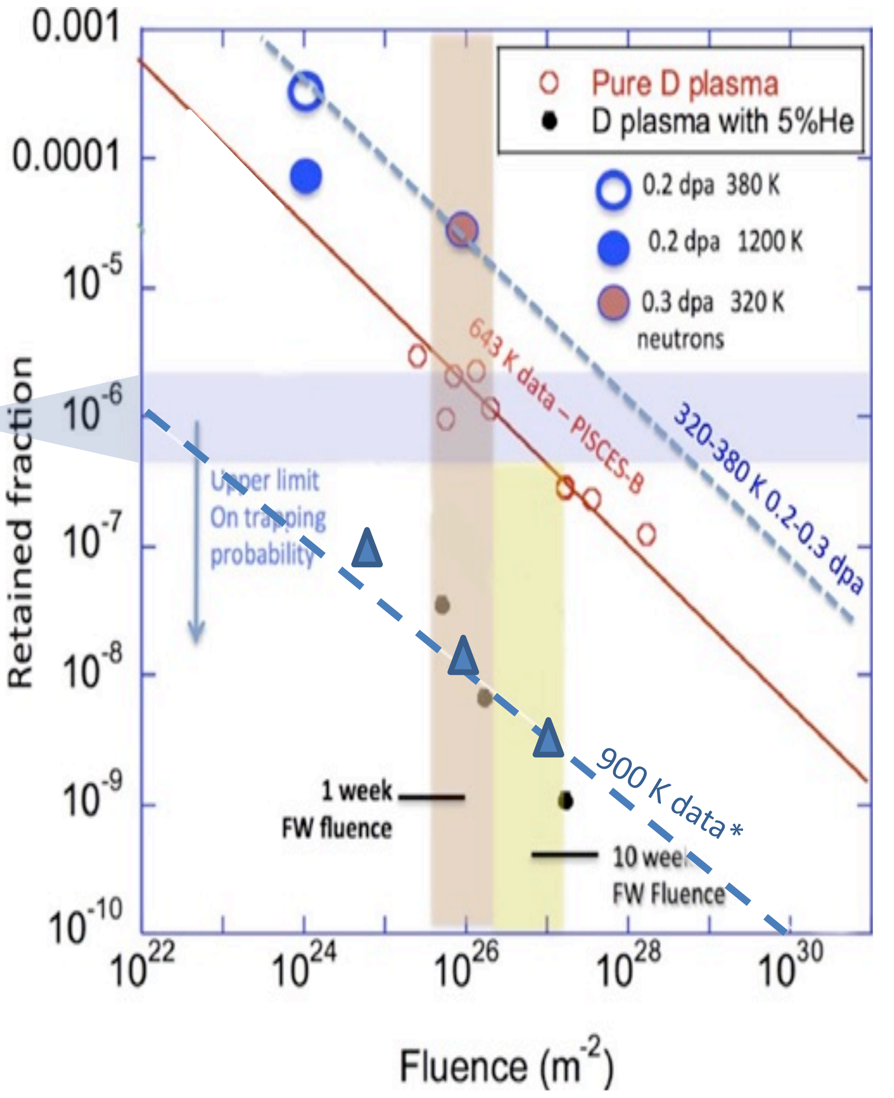
Annual T fluence into divertor: 300 Tonnes-T/year

Maximum allowable mobilizable
in-vessel T inventory: O(1kG)

Maximum allowable T retention probability: $3x10^{-6}$

PMI w/ Radiation damage could limit T self-sufficiency

- DT fusion energy fundamentally relies on T self-sufficiency, TBR > 1.
- **TBR > 1 requires retention probability in FW materials below 10^{-6}.**
- In PISCES, D in W retained frac. at 643 K reaches ~10^{-6} in ~1 week FW fluence.
- At 900 K, <1 h.



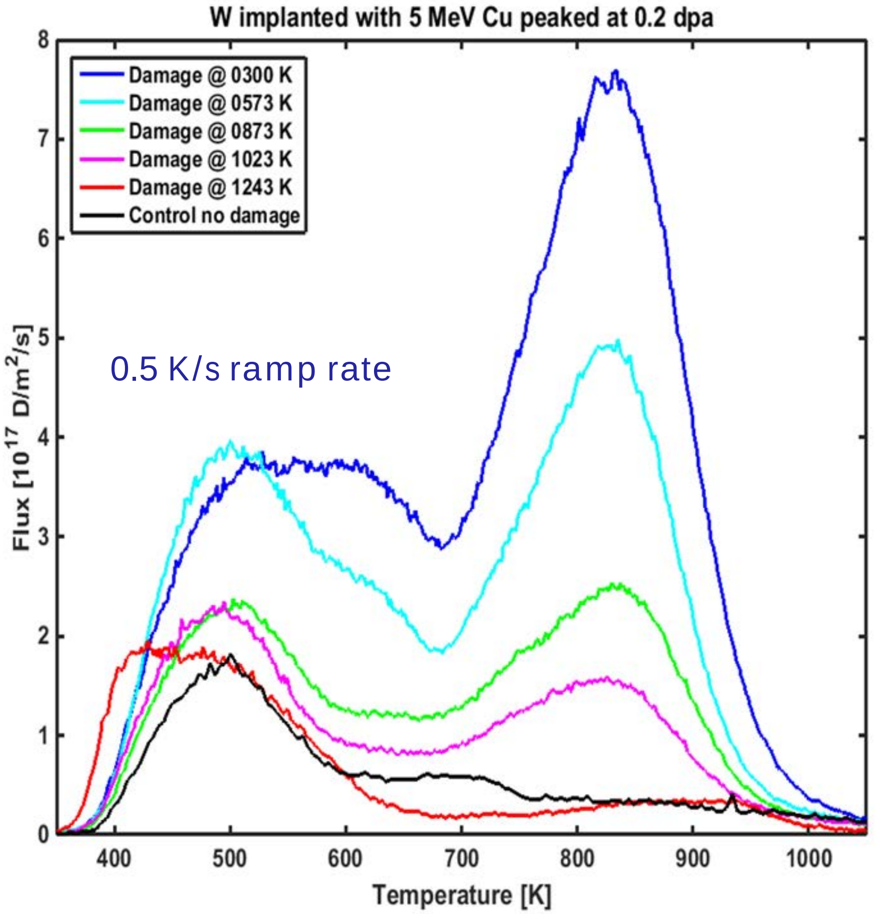
- He can play a beneficial role, but effect is known to be less efficient in damaged W.
- Sequential damage / PMI on W requires larger fluence 'cost'.
- ***Simultaneous plasma-displacement damage effects unknown***

Tynan Nuc Matls & Energy 2017, Doerner et al JNM 2019

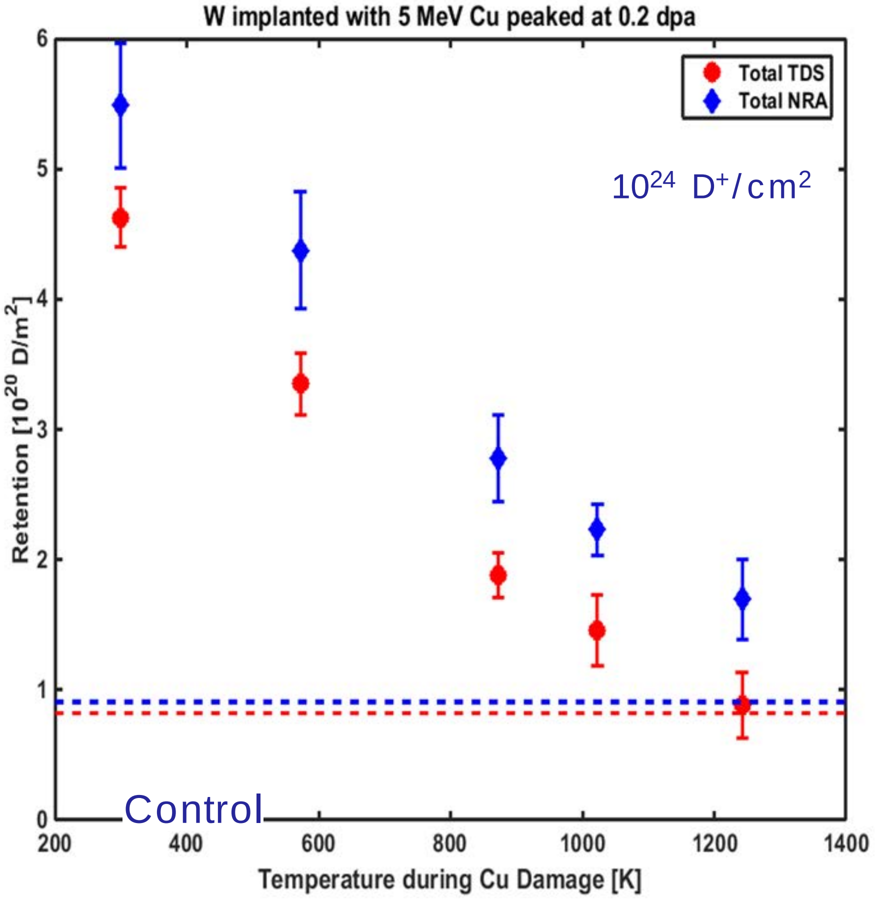
Annealing at high temperature partially heals displacement damage effects in plasma-facing armor materials

RETENTION CAN RECOVER W/ HIGH TEMPERATURE EXPSOSURES

M. Simmonds, 2015



0.5 K/s ramp rate

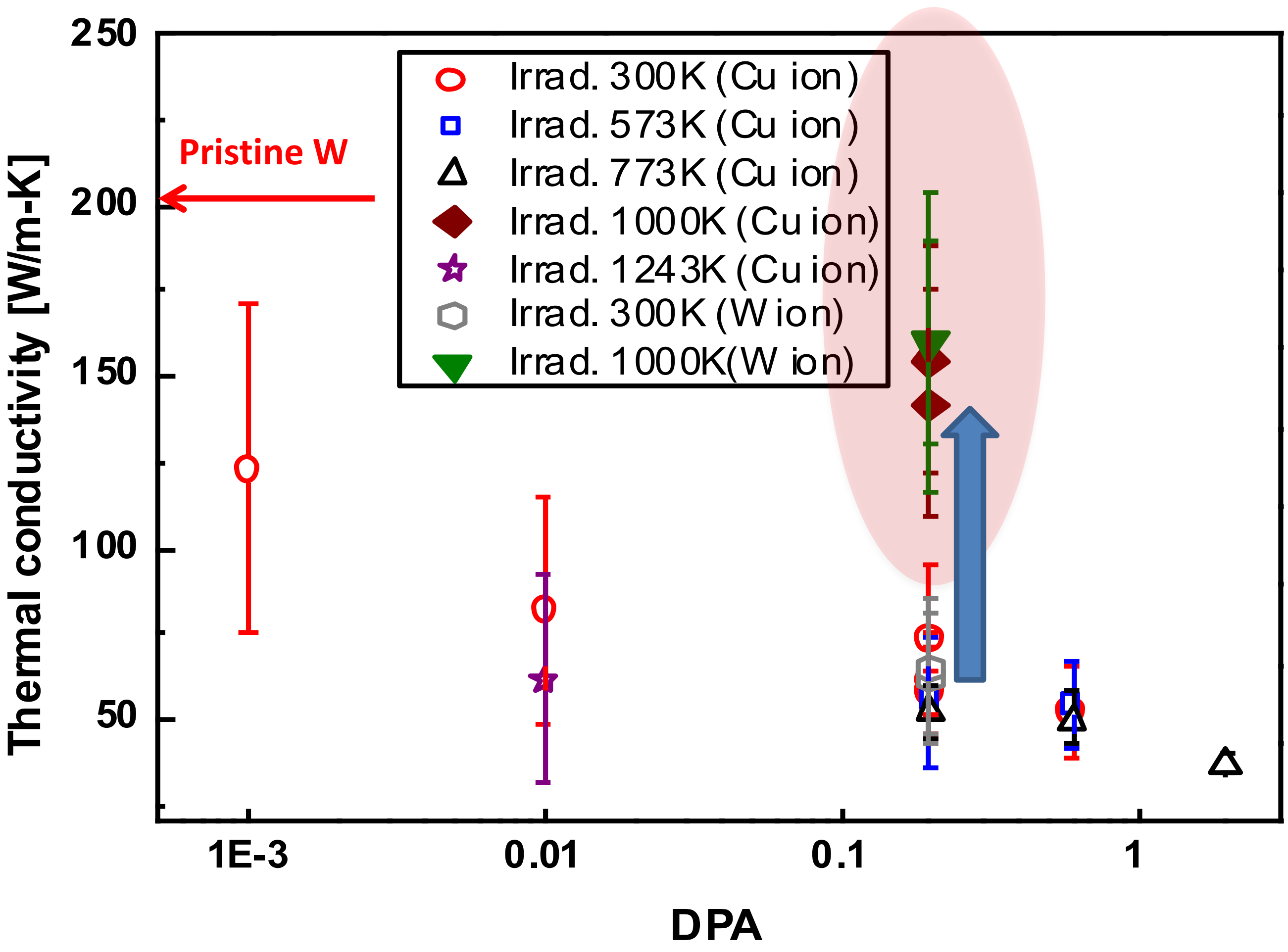


10^{24} D^+/cm^2

Control

Annealing at high temperature partially heals displacement damage effects in plasma-facing armor materials

THERMAL CONDUCTIVITY CAN RECOVER W/ HIGH TEMPERATURE EXPSOSURES



S. Cui, R. Chen et al, J Nuc Matl's 2018.

Outline of Talk

- What is required beyond ITER to get to fusion energy?
- What PMI-related issues emerge from this focus?
- What activities are underway?
- **What additional efforts are needed?**

Many PMI issues to study in lab-scale experiments

PMI Issue	Science Question	Possible Approach
Material Erosion	How does high particle flux affect erosion rate?	Implanted depth markers & Ion-beam NRA; Plasma spectroscopy
Material Redeposition	How quickly is material being redeposited, and what type of mixed materials are formed?	Ion-beam NRA, LIBS, Plasma Spectroscopy & 2D imaging
Fuel retention in D, D-T, and D-T/He Plasmas	Is D/T retetion low enough for TBR>1?	Ion-beam Rad-damage, NRA, LIBS, Ex-situ TDS
Rad-damage effects on PMI	Are there synergistic PMI/Rad-damage effects? Effects on retention? He effects?	Combined plasma/ion beam studies using He & Heavy Ions; GIXRD
Managing divertor heat flux	How do injected divertor impurities affect material surfaces?	Divertor simulator w/ PMI capabilities

Some critical PMI issues require confinement expts

- Adequate divertor & FW component lifetime
  - Control plasma erosion rate via divertor plasma physics (radiative divertor, Super-X, Snowflake, etc…)
- Redeposition, material migration & Fuel retention
  - Tritium inventory & Closing Fuel Cycle, Safety
- High Performance (Q>>1) Long-pulse (days to weeks) Plasma
  - Integrate divertor solution w/ core plasma regime
- Demo Adequate reliability, maintainability