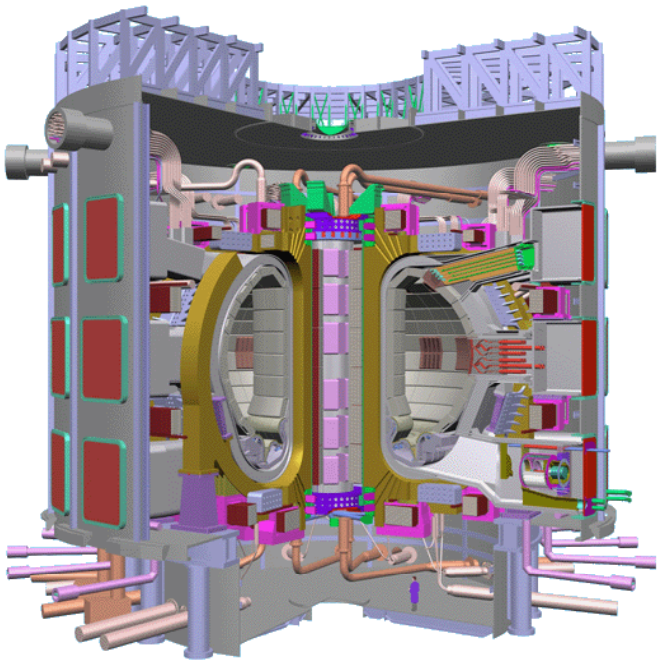
Plasma Wall Interactions in ITER and Implications for Fusion Reactors

G. Federici, *ITER International Team, Garching*



Particular thanks to
A. Loarte (EFDA), G. Matthews (JET).

OUTLINE

- Introduction: Power and particle exhaust in ITER→ Challenge to PFC materials.
- Thermal loads and erosion during transients:
  - ✓ Type I ELMs
  - ✓ Disruptions
- Issues for stationary conditions :
  - ✓ Carbon chemical erosion
  - ✓ C migration to shadowed areas and tritium co deposition
  - ✓ Mixing of materials
- **Implications and strategy for a reactor**
- Conclusions

Power and Particle Exhaust in Next-Step Devices

- Dominant α Heating ($P_\alpha \geq 2\ P_{add}$)

  → $Q_{DT} = P_{fus}/P_{add} \geq 10$
- "Reasonable Fusion Power" ≥ 500 MW

  - $P_\alpha \geq$ 100 MW

  - Power exhaust ≥ 150 MW

    > P_{SOL} ~ 100 MW

    > $q_{||}$ ~ 1 GW/m^2

    > $q_\perp^{PCF} \leq 10$ MW/m^2

    a) Low angle of incidence on target ~3° → $q_\perp^{PCF}$~50 MW/m^2

    b) Large SOL radiation → Divertor → P_{rad} ~0.6 P_{SOL}

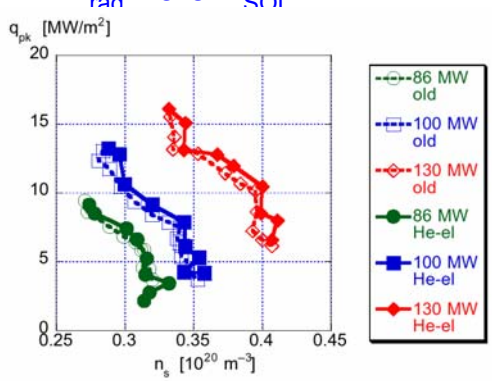
- He exhaust ≥ 1.8 10^{20} He/s
- Tritium throughput :
  - Min. (η_{He} ~ 0.1) ~ 10^{21} s^{-1}
  - Max. limit : Reprocessing and Reasonable inventory (5 10^{22} s^{-1}→0.25 g/s)

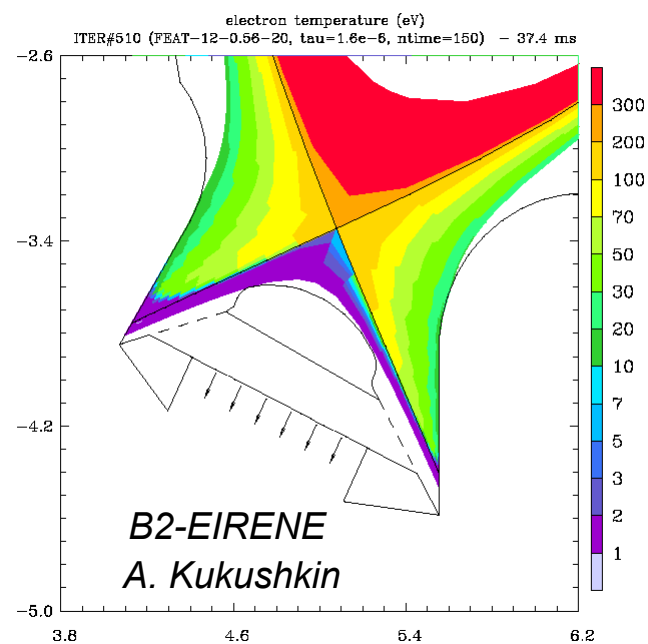
Basic Ingredient

C chemical Erosion ~ 1%

Γ_D ~ 10^{25} s^{-1} → Γ_C ~ 10^{23} s^{-1}

(extrinsic Impurities for fine-tuning)





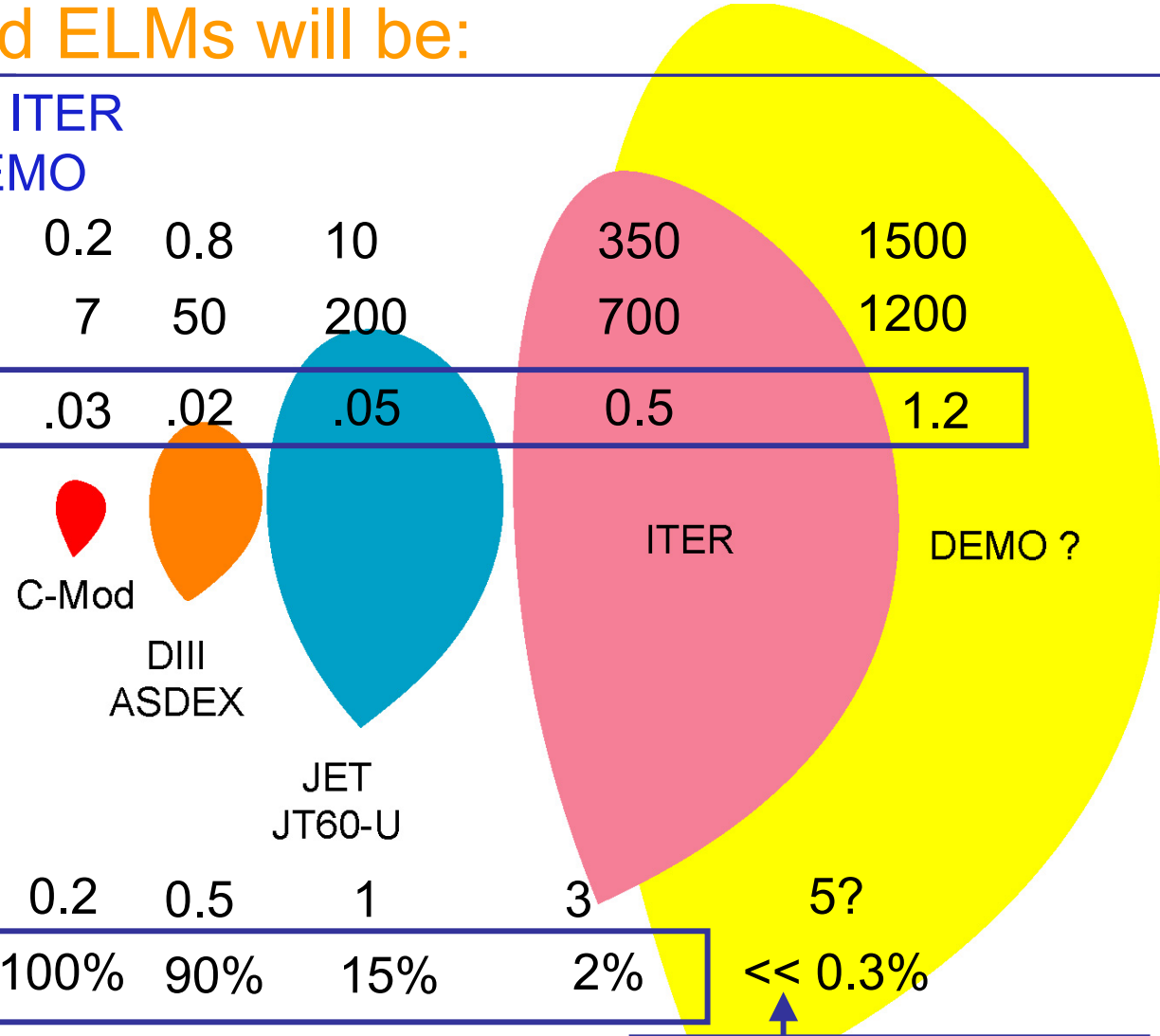
G. Matthews, PSI 2004

Disruptions and ELMs will be:

- strictly regulated in ITER
- not permitted in DEMO

$W_{thermal}$ (MJ) ➔	0.2	0.8	10	350	1500
A_{plasma} (m^2) ➔	7	50	200	700	1200
$W_{thermal}/A_{plasma}$ (MJ/m^2) ➔	.03	.02	.05	0.5	1.2
$A_{divertor}$ (m^2) ➔	0.2	0.5	1	3	5?
$\Delta W_{ELM}/W_{th}$ ➔	>100%	90%	15%	2%	<< 0.3%

Perfect disruption mitigation



Max. ELM size - target ablation

DEMO - limit set by sputtering threshold

Implications on design and machine operation

- **Thermal transients (e.g., ELMs and disruptions) strongly affect operation and drive material choice.**
  - **Type I ELMs may cause unacceptable divertor erosion and damage at the wall, and in the case of C contribute to tritium co-deposition.**
  - **Unmitigated disruptions lead to W melting and probably 'mitigated' disruptions could damage the Be wall (e.g., formation of melt layer whose behaviour and properties are uncertain).**

Bad News!!!

PMIs in ITER will scale up orders of magnitude with increase in stored energy and pulse duration

Parameter	Today's tokamaks	ITER	Factor X times
$E_{therm} \alpha R^5$ (MJ)	1-15	350	25-300
Disruptions (MJ/m^2)	0.03-0.5	~1-10	2-20
Type I ELMs (MJ) (MJ/m^2)	0.1-0.6 0.01-0.3	8-20 0.1-2	10-30
Pulse duration (s)	10-25	400	10-40
Cum. Run time (hr/y)	2-8 hr/y	>200 hr/y	25
C-erosion divertor	7-300 µm/yr	> 1 cm/yr	100
T-injected/ pulse (g)	0.2	~100	>100

- **In-vessel tritium inventory with carbon PFCs could rapidly reach safety limit (>300 g).**
  - **Current estimates ≈ few hundred pulses in ITER.**
  - **Poor validation of models to predict T retention in ITER. They are far from explaining experimental results (underestimates).**
- **No engineering-scale demonstration of fast and efficient tritium removal from tokamaks to date !**

- **Urgent needs**
  - **PWI diagnostics and measurement methods and strategies to learn as we operate.**
  - **We take a risk if we do not test/validate proposed material choices in existing tokamaks.**
  - **Push ahead now to develop zero C scenarios (i.e., high-Z).**

PFC materials selection - 3 materials to start with

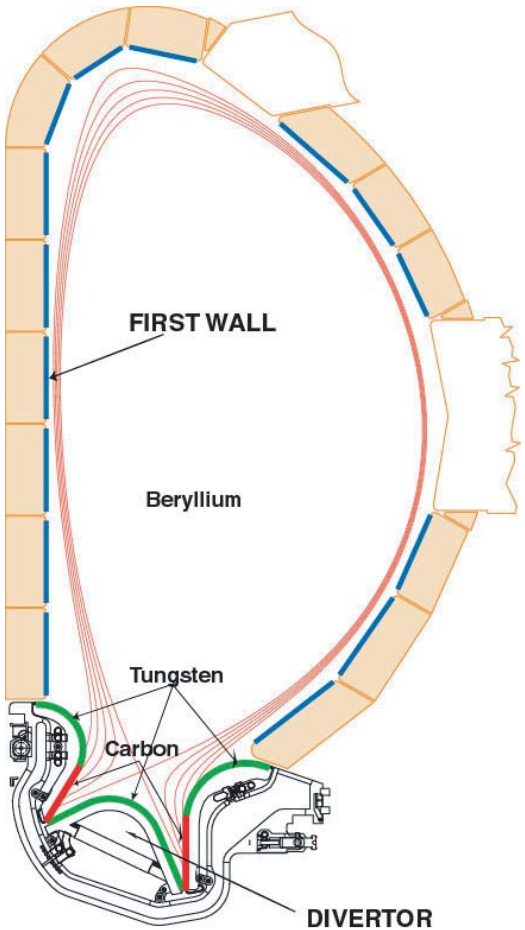
Behaviour under transient and steady-state loads
→ PFC Material Choice for ITER

~ 700m^2 Be first wall : low Z + Oxygen getter

~ 100m^2 W Baffle/Dome : low Erosion, long Lifetime

~ 50 m^2 Graphite CFC Divertor Target (→ W):

a) No Melting under transient Power Loads (ELMs and Disruptions)
b) Compatibility with wide Range of Plasma Regimes ($T_{e,div}$ ~ 1 – 100 eV)
c) C is a very good radiator right where it needs to be (i.e., T_e in the divertor)



- Is this a reasonable choice?
- What are the implications of this choice?
- What remains to be addressed to demonstrate the validity of this choice?
- What are the feasibility issues and operation implications of alternatives?

We plan to use C in the divertor during initial operation

- The rationale for using carbon in the ITER divertor follows directly from:
  - *the projected levels of thermal loads and damage during thermal transients (e.g., type I ELMs/ disruptions). These are still uncertain and need R&D.*
- Two strategies for the ITER divertor:

(1) retain CFC during operation with H- and D. This implies availability of:

- *in-situ* time- and space-resolved, as well as global measurements of D-retention and erosion and validated modelling tools, which would enable to monitor status of D retention in ITER during D-operation and predict conditions during DT operation.
- efficient methods for T-removal, which still need to be developed and tested in tokamaks with the relevant materials mix and temperatures.

(2) start with full-W divertor and develop compatible operating scenarios.

- This means accepting greater restrictions on ITER operating space.
- The issue of operation with an all metal wall (high or low Z) and no carbon still needs to be explored in current machines and scaled with machine size.
- If we are to live without C, we have to find alternative ways to radiate.

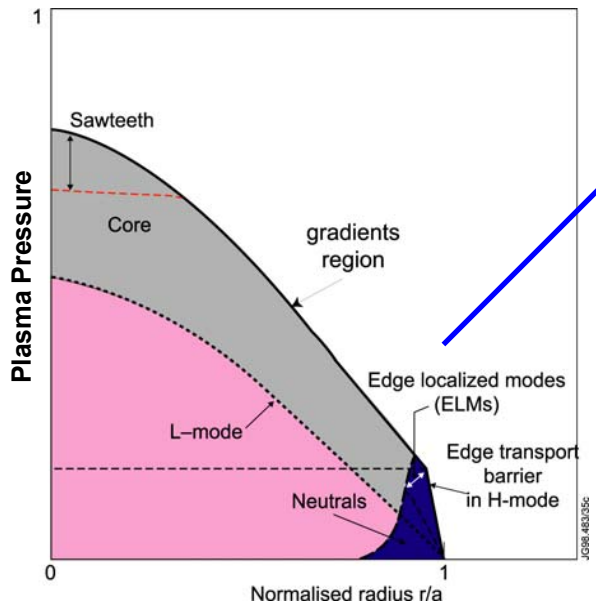
Type I Edge Localised Modes (ELMs)

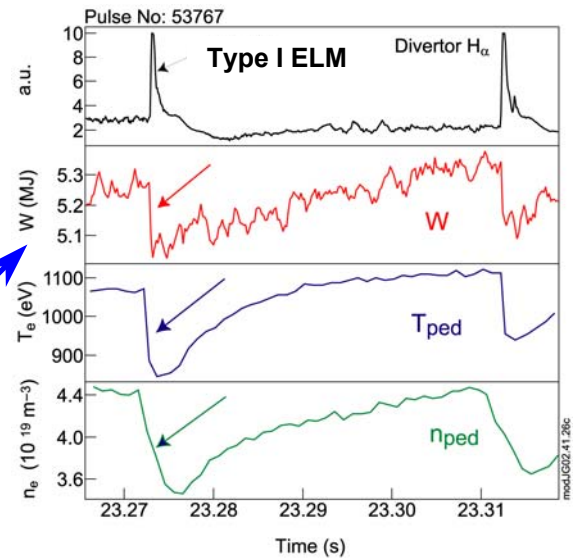
A. Loarte, Plasma Phys. Control. Fusion 45 (2003) 1548

Next step Experiments @ Q_{DT} ~ 10 require high τ_E and n_e Regimes

High (n_e, τ_E) Type I H-mode provides a robust Regime to achieve Q_{DT} ~ 10

H-mode ←→ ETB ←→ Pedestal



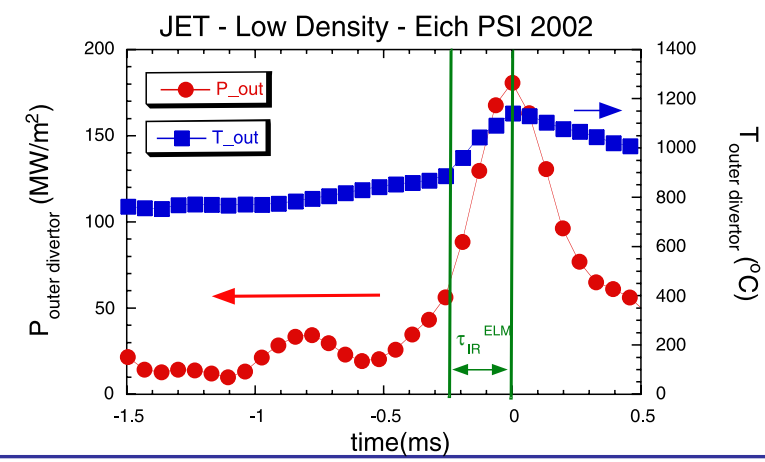


W_{JET} ~ 2 – 12 MJ vs. W_{ITER} ~ 350 MJ

A_{ITER}/A_{JET} ~ 2 – 3

ITER Type I ELM Energy Flux :

- E_{ELM} = 0.5 – 5.0 MJ/m^2
- Duration Energy Pulse= 0.2 – 1 ms
- f_{ELM} = 1 - 10 Hz



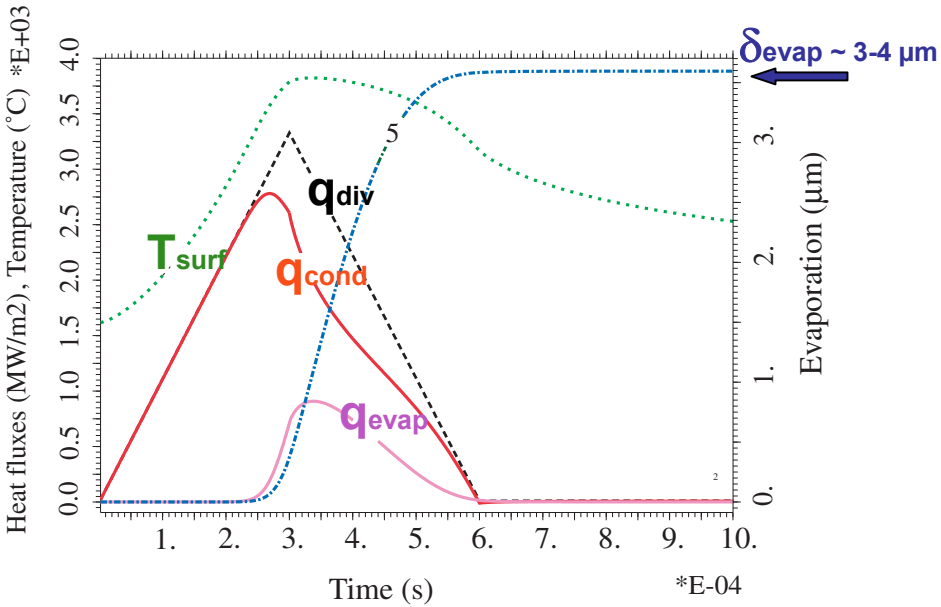
Talk of Janeschitz

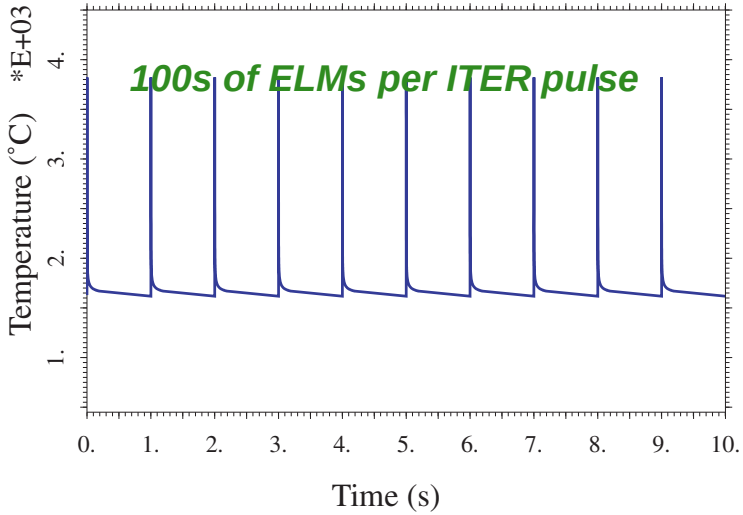
Regimes with high (τ_E, n_e) & small ELMs exist in some devices but have non-negligible drawbacks (q_{95} > 3.5, operation close to DNX, …)

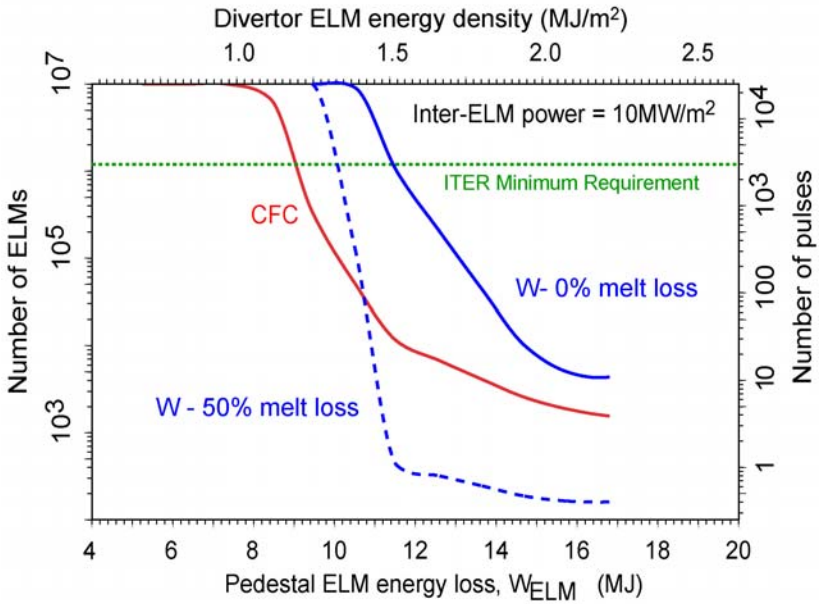
Thermal loads and erosion during ELMs

Main problems arising for ITER:

- *Divertor target erosion lifetime.*
- *Impurity production and plasma contamination.*







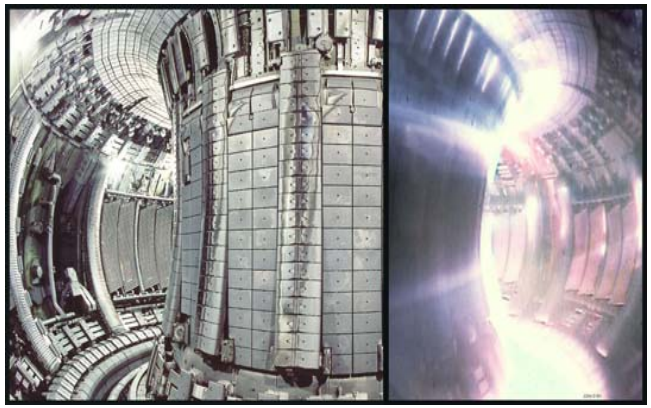
Because of short duration of ELMs steep temperature gradients only near-surface ≤ 300 μm

G. Federici, Plasma Phys. Control. Fusion 45 (2003) 1523

Effects of Type I ELMs at the main chamber wall

$\Delta W_{ELM}^{div} \sim 50 - 80$ % of ΔW_{ELM}^{dia}

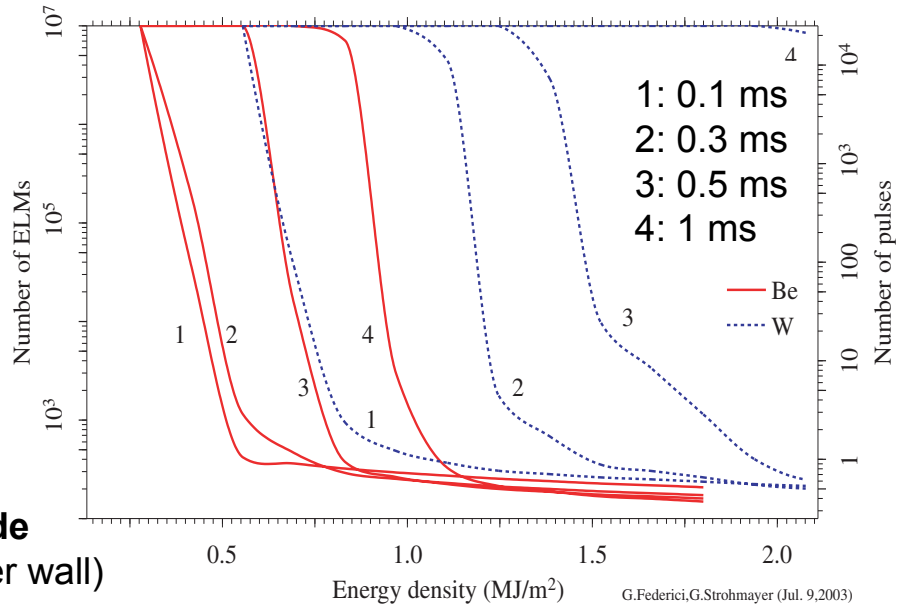
=> **A fraction of the ELM energy reaches the main chamber wall**

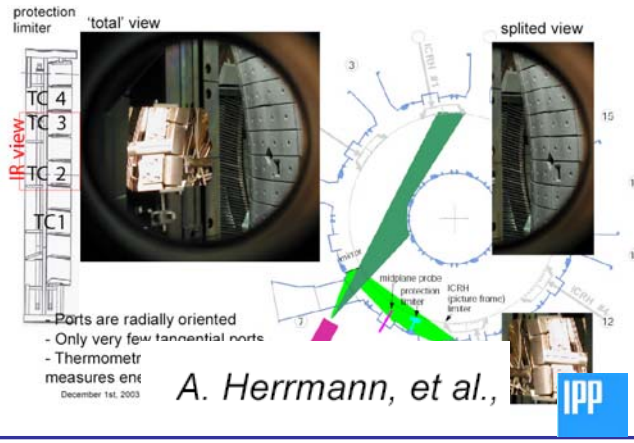


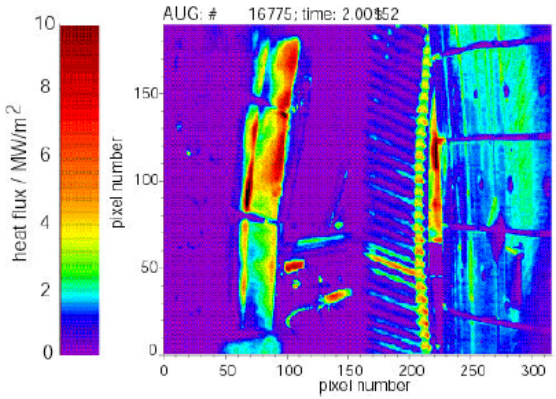
JET – Type I ELM

- **26% of ELM energy mid-plane loss is found outside divertor in ASDEX Upgrade** (14% limiters, 12% inner wall)

- Strong implications for ITER







- Wetted area?
- Protruding surfaces (e.g. start-up limiters (front surface few m²).

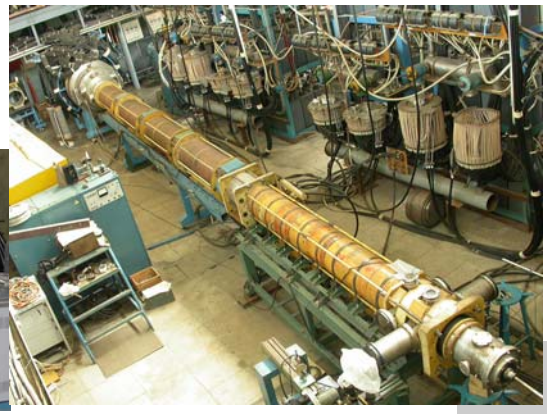
  => tolerable ~0.3-1 MJ/m² depending on material and ELM duration (if ≥ 1 ms W doesn't melt).

Cumulative damage effects during Type I ELMs

ITER

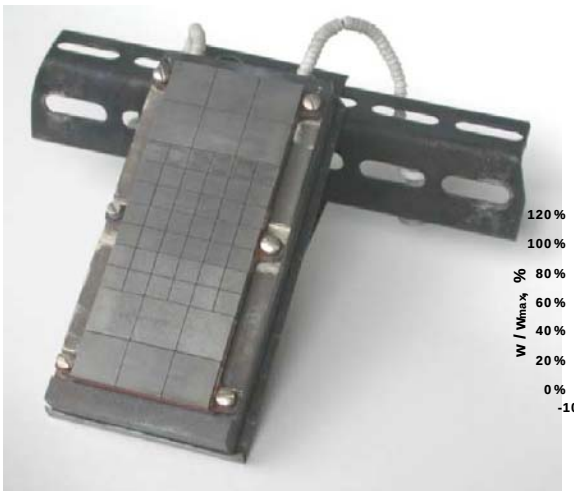
Two Russian plasma guns located in SRC RF TRINITI

QSPA



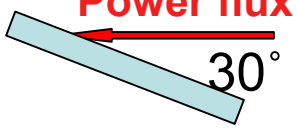
Plasma parameters (ELMs):

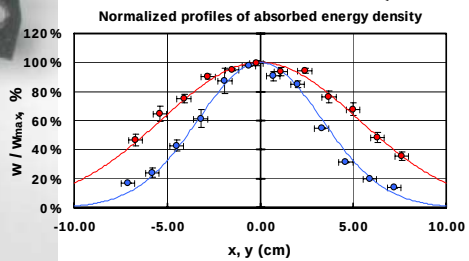
	QSPA	MK200UG
• **Heat load (MJ/m^2)**	**0.5-2**	**0.2–1**
• **Pulse duration (ms)**	**0.1–0.6**	**0.04–0.06**
• Plasma stream ϕ (cm)	5	6-10
• **Magnetic field (T)**	**0**	**0.5-1.2**
• **Ion impact energy (keV)**	**<0.1**	**1.5**
• **Electron temp. (eV)**	**<10**	**100-200**
• Plasma density (m^{-3})	< 10^{22}	(2-5)x10^{21}



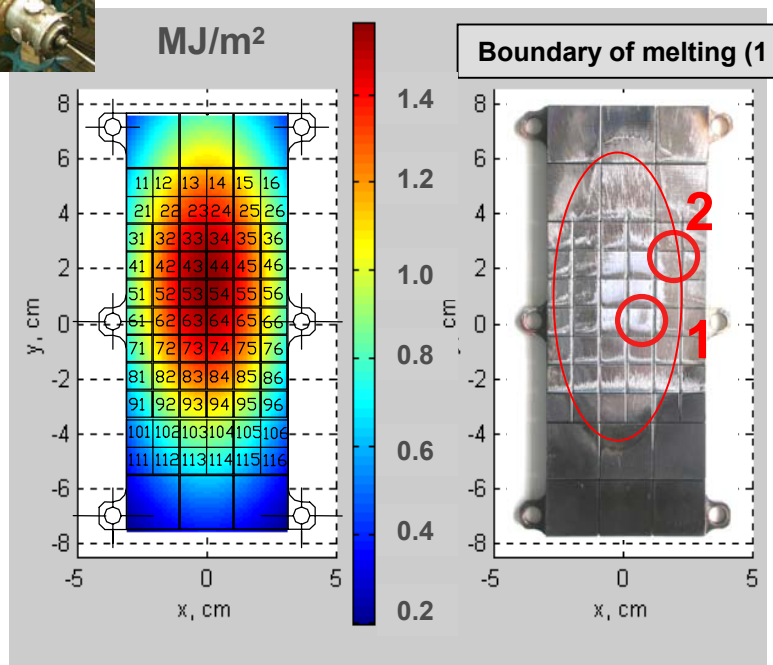
150x60x10 mm³

Mk-200UG

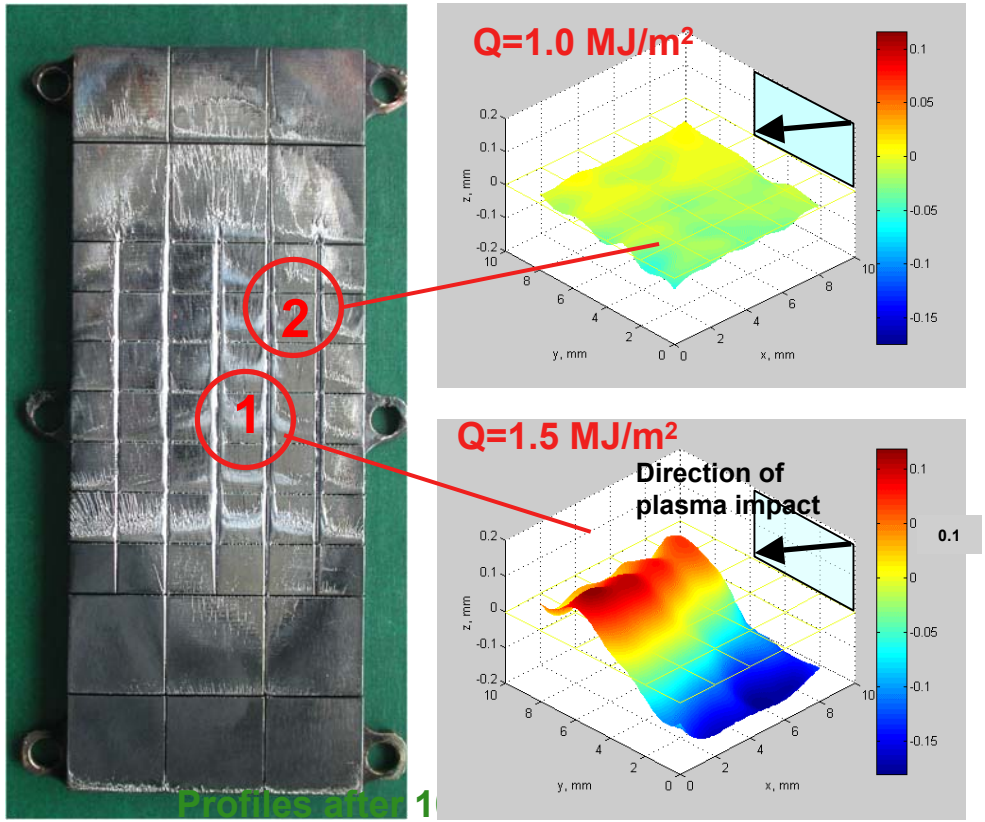


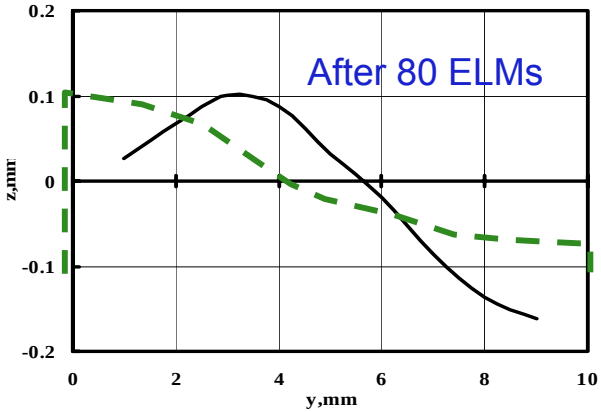


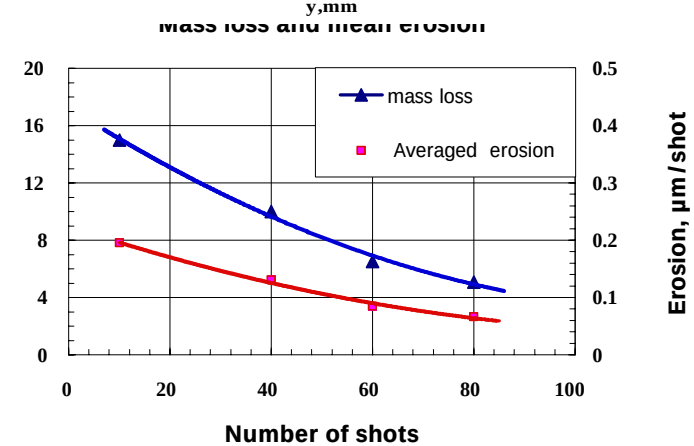
- along the target (y direction)
- across the target (x direction)



W melt layer motion under ELM heat loads at QSPA facility (Δt=0.5 ms)



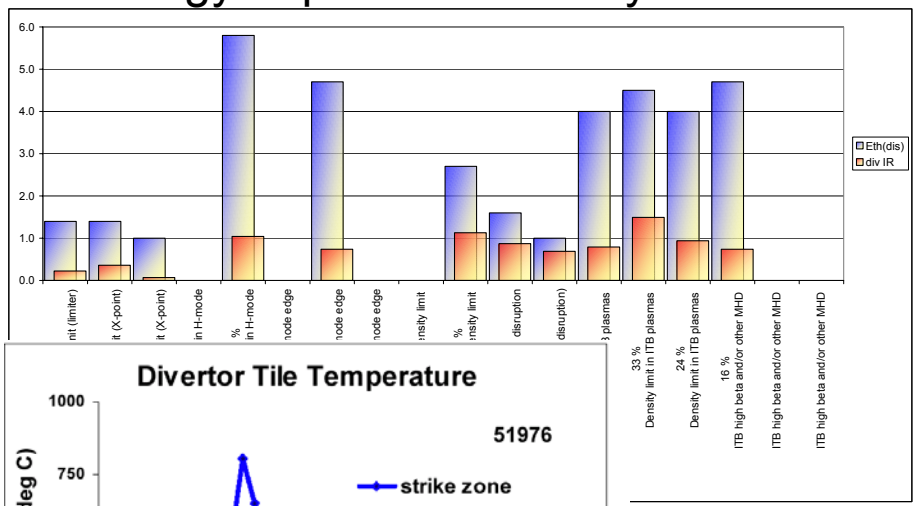




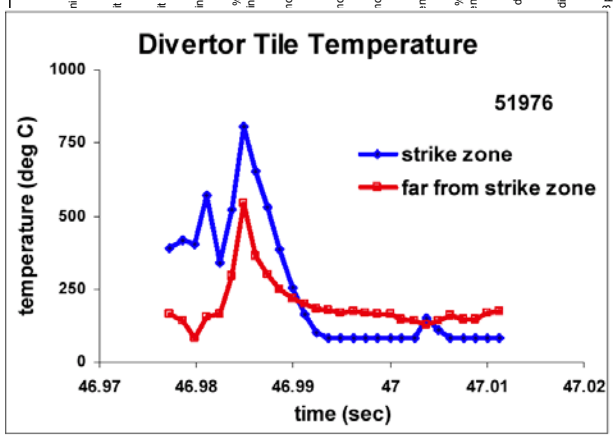
- **W melt layer forms at the edge and shifts within an individual macro-brush element along plasma stream direction but does not close the gaps;**
- **average erosion did not exceed 0.2 µm per shot (evaporation, no melt losses);**
- **maximal roughness of surface reaches 0.3 mm after 100 pulses @ ~1.5 MJ/m^2;**
- **modelling prediction in good agreement with experimental results.**

Thermal quench disruptions

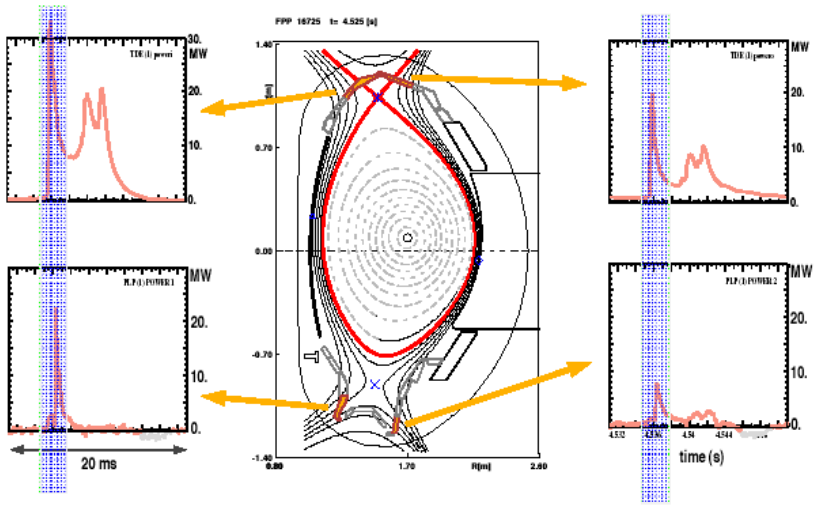
- Energy deposition not only in the divertor ($W_{div} << W_{th}$). Large broadening seen.



V. Riccardo (JET)



P. Andrew (JET)



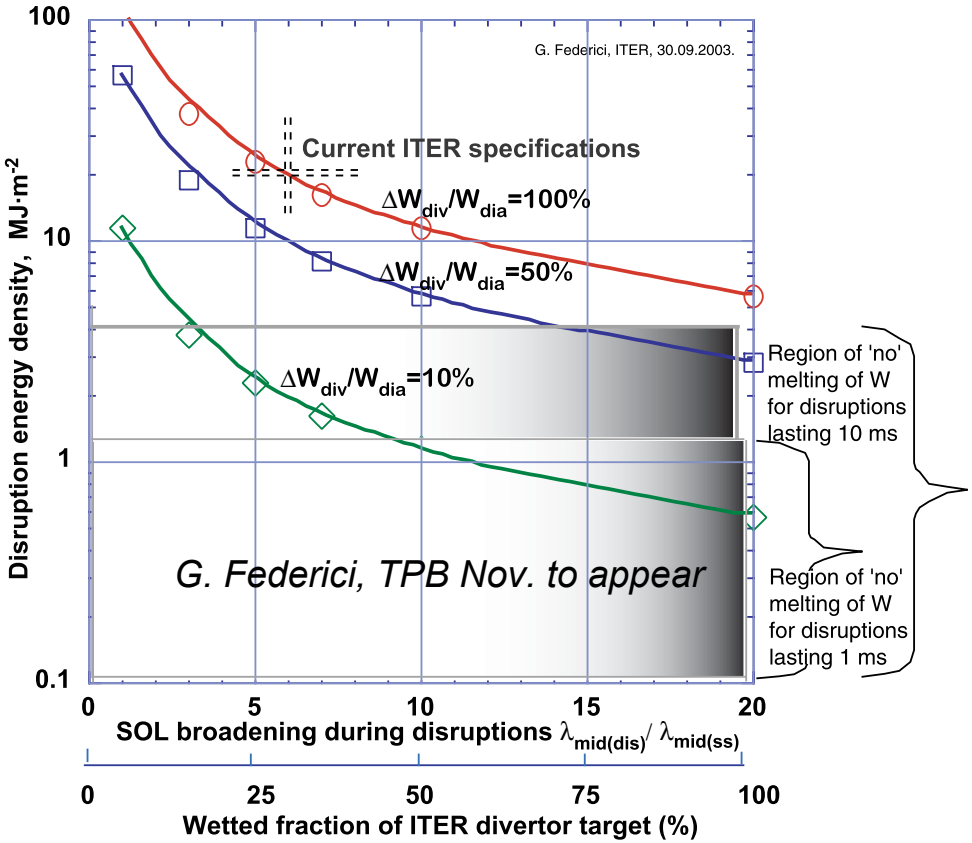
G. Pautasso (ASDEX Upgrade)

- However, it is not known at the moment where and by what processes the missing thermal and magnetic energies are deposited in the main chamber.
- If the JET results extrapolate to ITER, disruptions would not damage a W target.
- If this energy deposition is not sufficiently uniform, then in ITER additional damage to main chamber components might be expected.

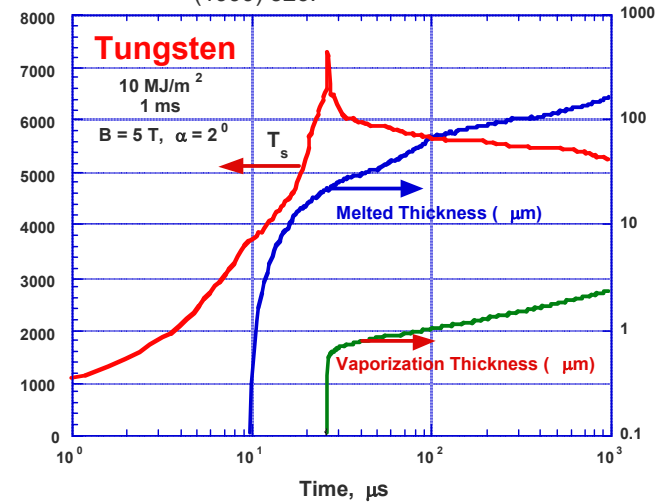
Specifications for thermal quench need revisiting

Current ITER thermal quench specifications (very conservative)

- W_{th} = 350 MJ, duration ~ 1 ms
- $\Delta W_{div}^{disruption}/W_{th}$ ~ 1
- Broadening ~ 3 (very small)



Hassanein, A., J. Nucl. Mater. 273 (1999) 326.

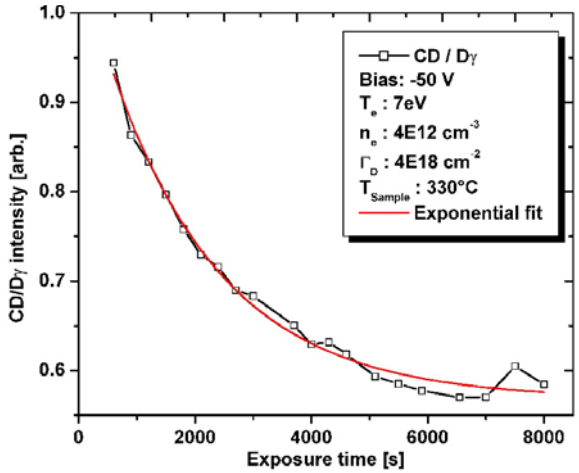


- Important driver for lifetime of W divertor target in ITER
  - ✓ **Concern remains on whether generation during ELMs, disruptions of surface irregularities in tungsten due to melting, and in CFC due to brittle destruction, might form hot spots during normal operation.**

C-erosion during normal operation

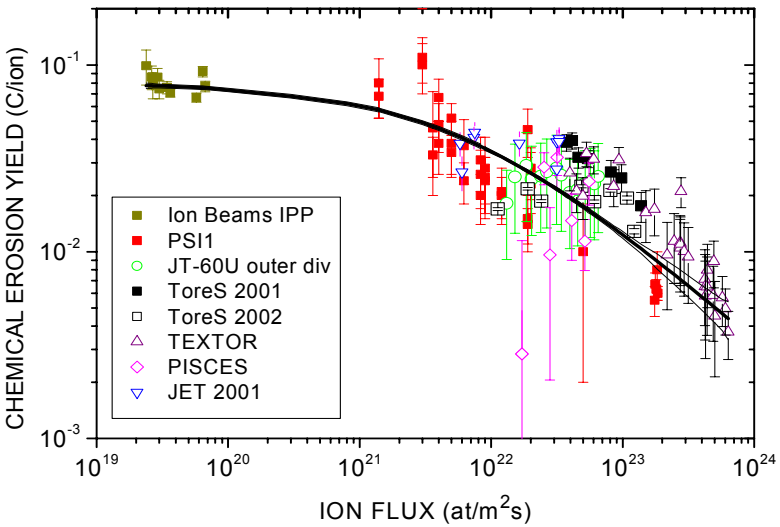
Reduction of source of carbon (chemical sputtering) :

- Flux dependence, high target temp;
- Fluence dependence (DIII-D, PISCES-B);
- Doping.



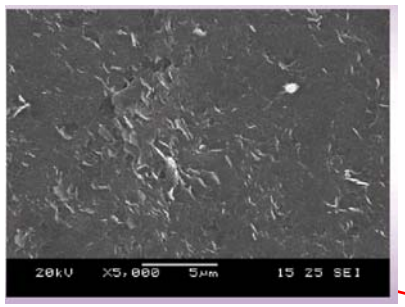
Reduction of Y_C^{chem} at high fluences accompanied by surface modifications

J. Roth et al.
Data after calibration and normalisation to 30 eV

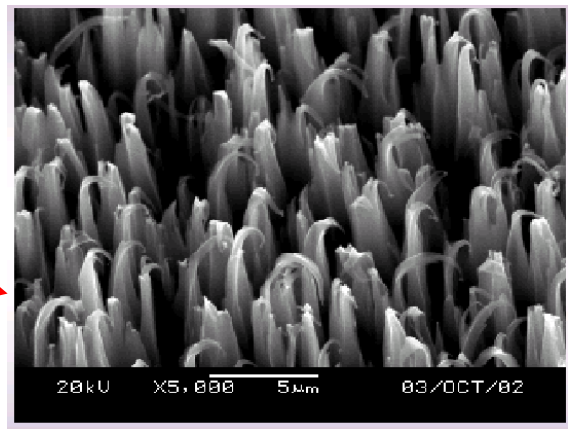


J. Roth (IPP Garching) Nuclear Fusion 2004

PISCES-B
R. Doerner, K. Schmid, et al.



T_{surf} ~ 440 °C

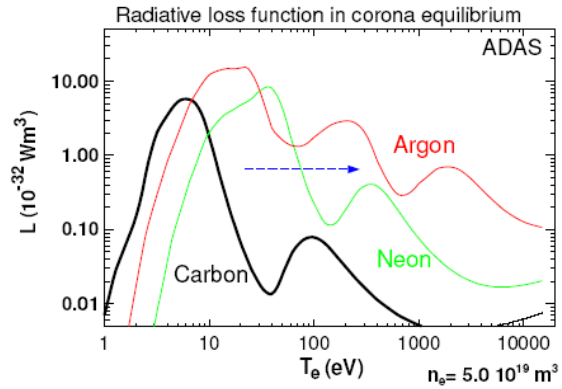


Carbon erosion during normal operation

C necessary to radiate power in the divertor –

- $Y_C^{chem} \sim 1\% \rightarrow P_{RAD}/P_{SOL} \sim 0.6$
- If $Y_C^{chem} << 1\% \rightarrow$ strong (Ne, Ar) Seeding

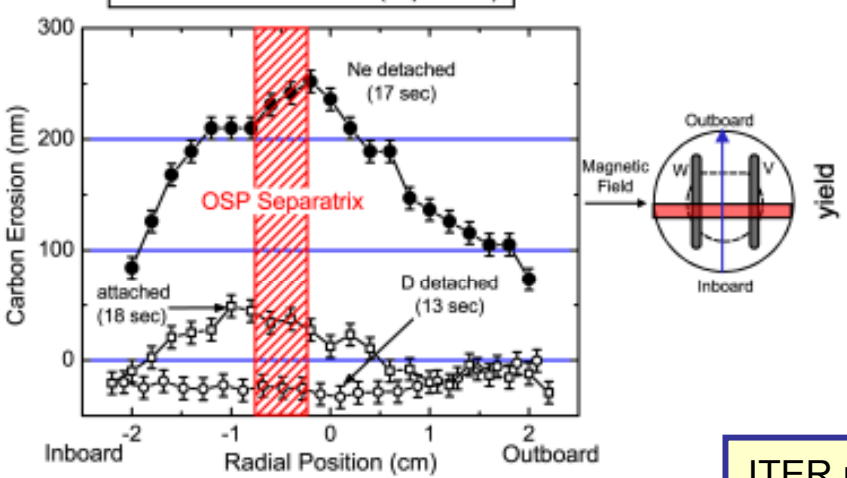
Ne, Ar seem to enhance erosion in laboratory experiments and tokamaks !!



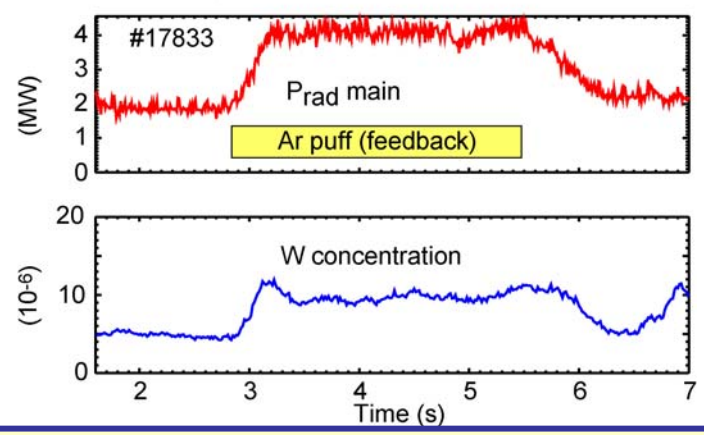
noble gases not effective at low Te
→ very high divertor compression needed

DIII-D, W. Wampler

Peak net erosion rate for carbon:
neon detached 15 nm/s erosion,
attached 3 nm/s erosion,
D detached -2 nm/s (deposition)



ASDEX Upgrade 9 MW combined NBI + ICRH (central)



Impurity seeding will impact high Z erosion

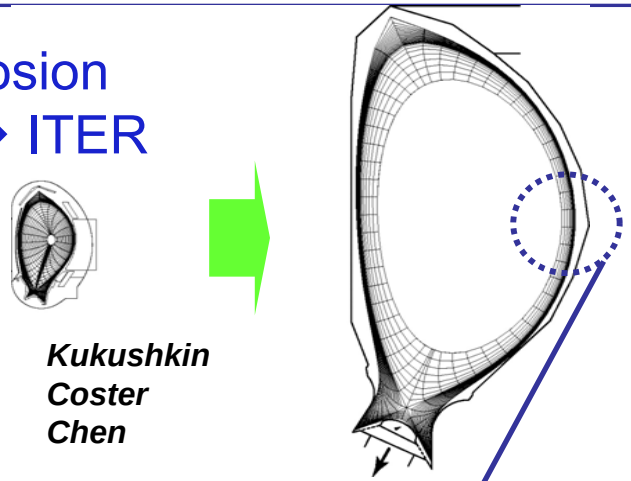
ITER requires $P_{rad}/P_{heat} \geq 60\%$ (for 10MW/m^2 at target)

DEMO requires $P_{rad}/P_{heat} \geq 85\%$ (for 15MW/m^2 at target)

Main chamber erosion→$D^{0,+}$, Z^{n+} fluxes sputter walls

- impurity source
- divertor deposition

Scaling neutral erosion of W walls AUG ➔ ITER



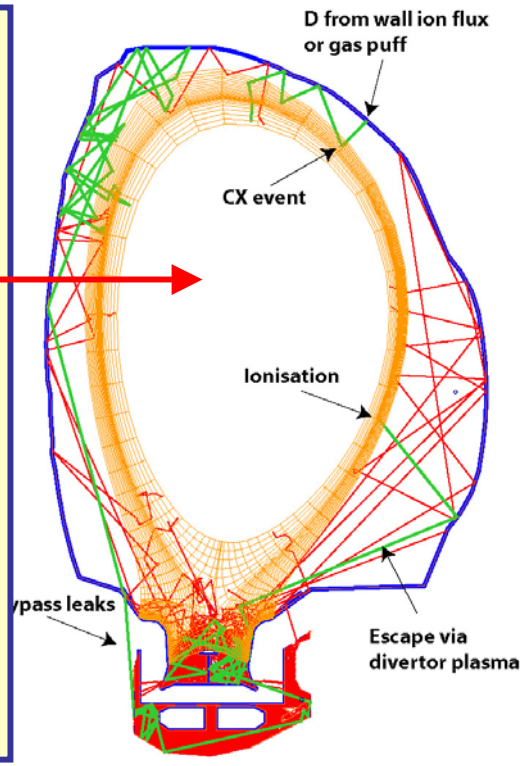
Kukushkin
Coster
Chen

What determines $D^{0,+}$ fluxes?

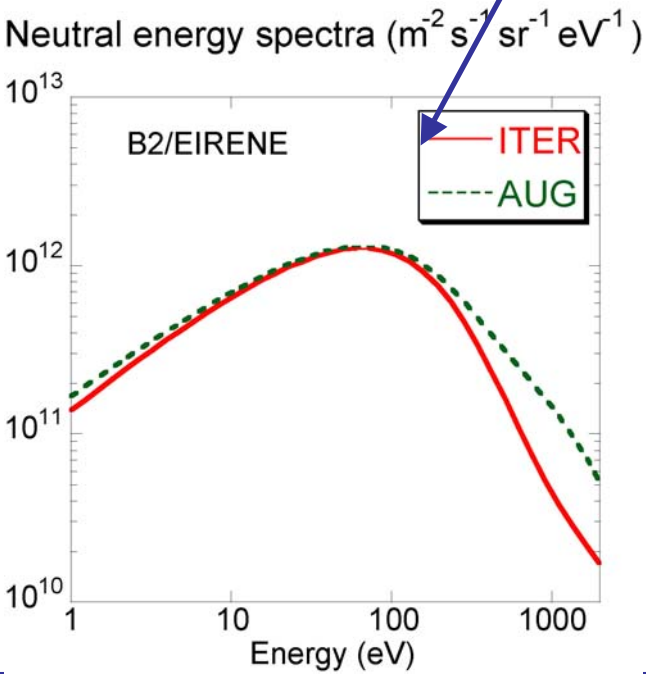
- **D^0 escape via divertor plasma**
- **D_2 gas puff**
- **D_2 bypass leaks**

→ Predictable
+ Controllable?

- **Ion flux to main walls due to:**
  - Far SOL transport (C-Mod)
  - Flux due to ELMs

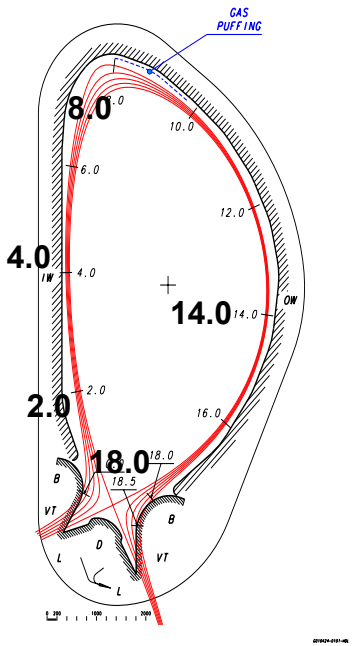


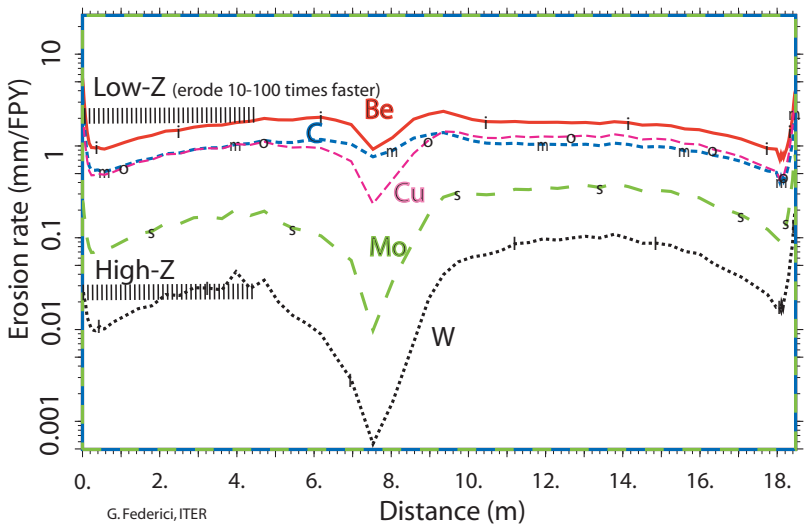
Matthews, PSI 2004



Main chamber erosion→$D^{0,+}$, Z^{n+} fluxes sputter walls

→Eroded Be will deposit in the divertor and then …….



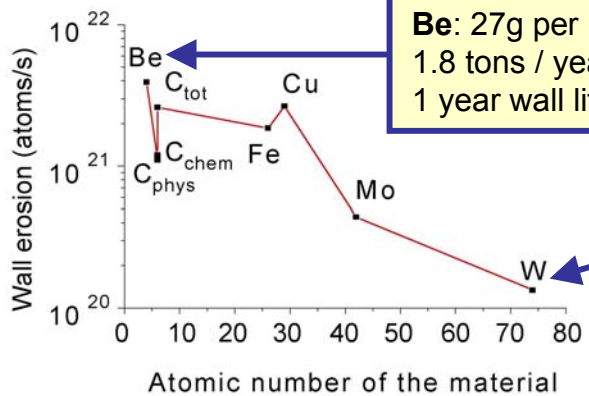


G. Federici, et al. JNM 290-293 (2001) 260.

- **Be peak erosion rate of ~0.1 nm/s (~0.3 cm/burn-yr) is acceptable for the low duty-factor operation of ITER, but not for a reactor.**

Comparison of reactor main wall materials

	Erosion (g/ITER pulse)	Co-deposition (g-T//ITER pulse)
Be	27	<<1?
C	21	~1
W	9	0



Be: 27g per ITER pulse
1.8 tons / year (1m^3)
1 year wall lifetime

W: 9 g per ITER pulse
0.7 tons / year (0.03m^3)
20 year wall lifetime

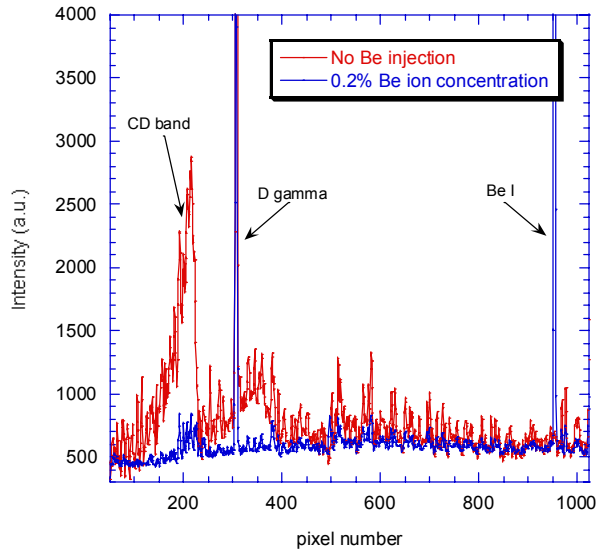
Behrisch PSI02

Effects of Be deposition at the carbon target

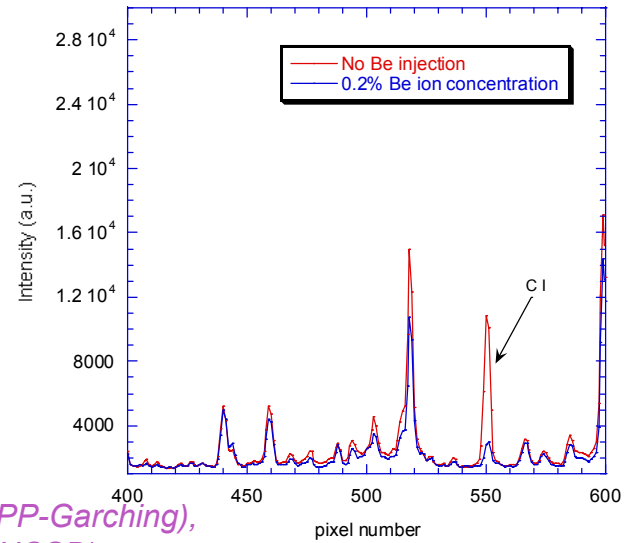
A small Be impurity concentration (~0.2%) in the plasma causes Be surface layers to form, suppressing both chemical and physical erosion of carbon

-50 V bias, 200ºC, T_e = 8 eV, n_e = 3 e 12 cm^{-3}

Chemical erosion



Physical sputtering



K. Schmid (IPP-Garching), R. Doerner (UCSD)

- **ITER is expected to have ~1-10% Be concentration in the divertor plasma, but several open questions remain:**
  - ✓ **Be-deposition/Be-diffusion (high T_{surf})**
  - ✓ **Behaviour for (Ne, Ar) seeding**
  - ✓ **Power handling/transient power loads**
  - ✓ **In/out coverage asymmetries ?**

⇨ ***A tokamak demonstration under ITER relevant conditions* is urgently needed**

Predictions of Tritium Co-deposition in ITER

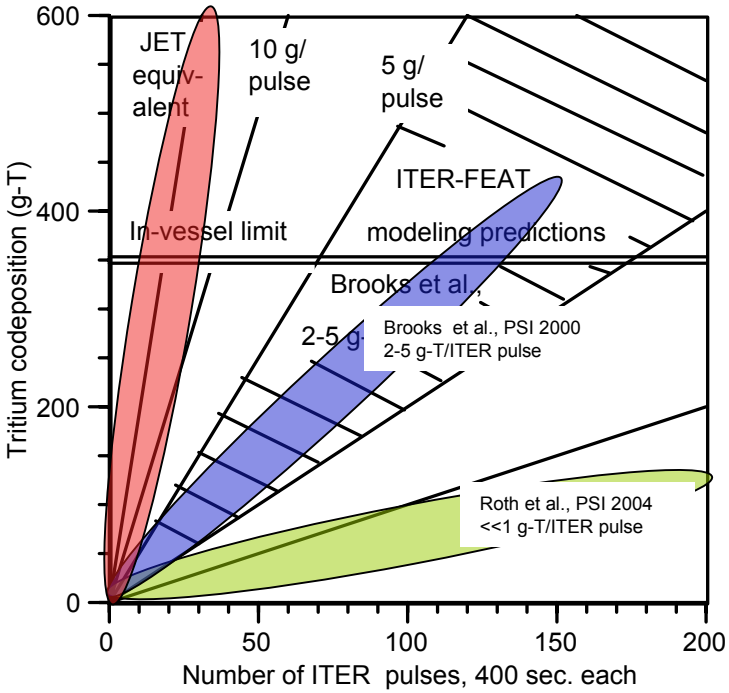
Extrapolation from existing C-machines is very difficult

ITER

- Predictions still uncertain due to
  - *chemical erosion yields at high temp. and fluxes,*
  - *contribution of type I ELMs,*
  - *effects of gaps,*
  - *effects of mixed materials,*
  - *lack of code validation in detached plasma.*

Note:

- *ITER predicted rate is 10x less than that in JET*
- *Model underestimates JET retention by factor x40.*

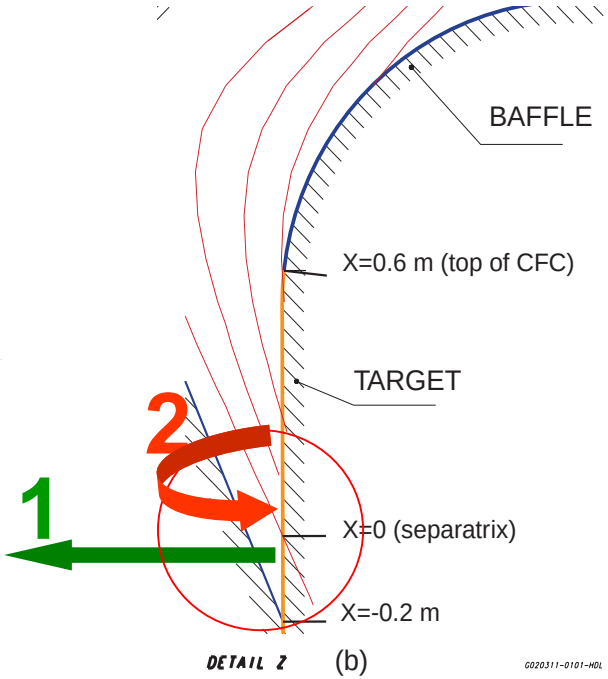


- T issues will be heavily scrutinised by authorities.
- Scale-up of removal rate required is 10^4.
- Potential options for T removal techniques for ITER.

1) Remove whole co-deposit by:
   - ***oxidation (maybe aided by RF)***
   - ***ablation with pulsed energy (laser or flashlamp).***
2) Release T by breaking C:T chemical bond:
   - ***Isotope exchange***
   - ***Heating to high temperatures e.g. by laser, or ...***

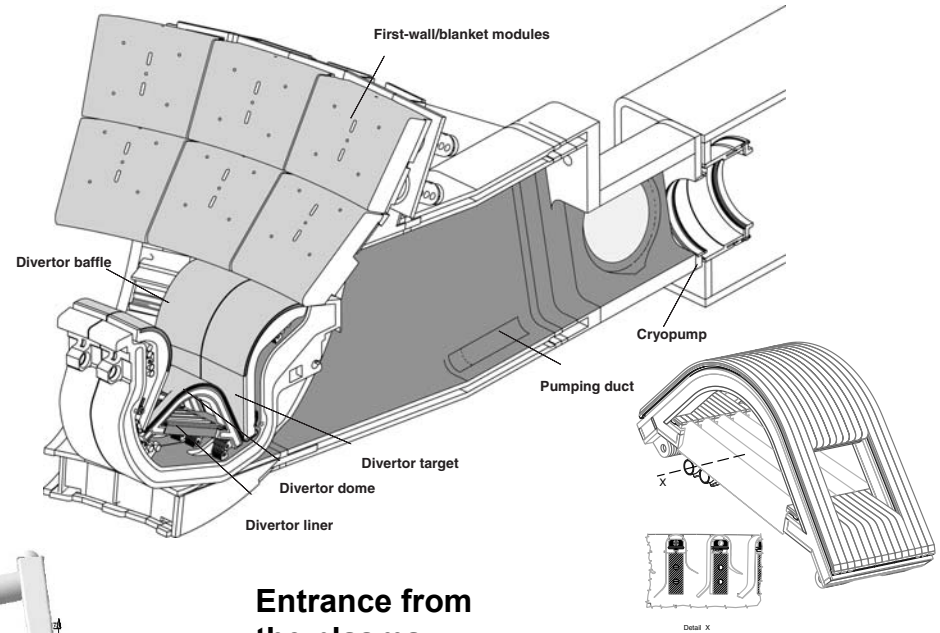
- Constraints:
  - 6.1 Tesla field at inner divertor
  - 10,000 Gy/hr gamma field from activation, 3 h after shutdown.
  - Access difficult, especially to hidden areas

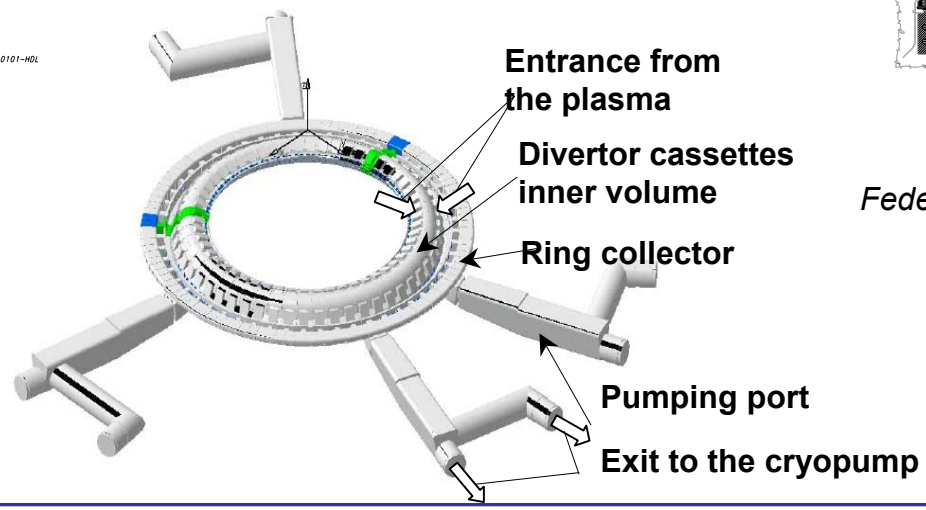
C -deposition in remote areas



to private region

back on plate

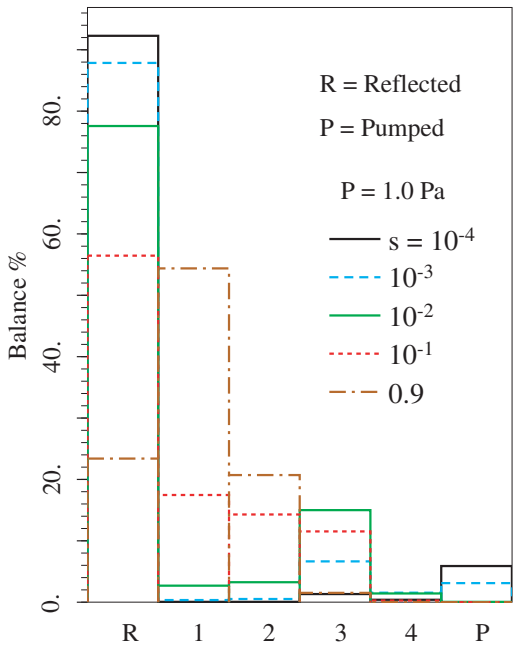


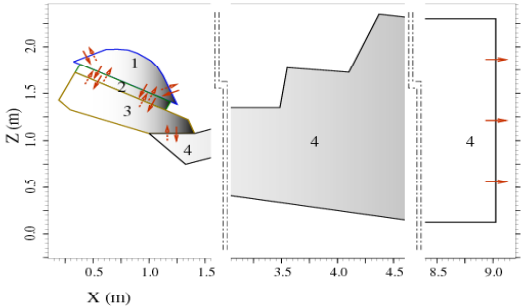


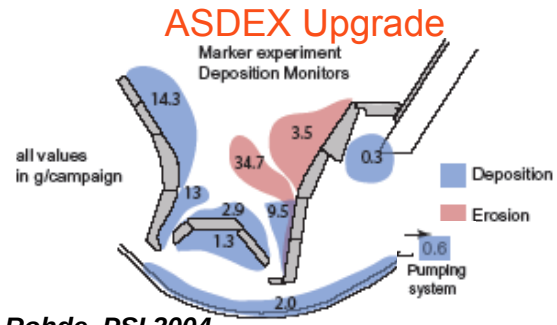
Federici, PSI 2004

C -deposition in remote areas

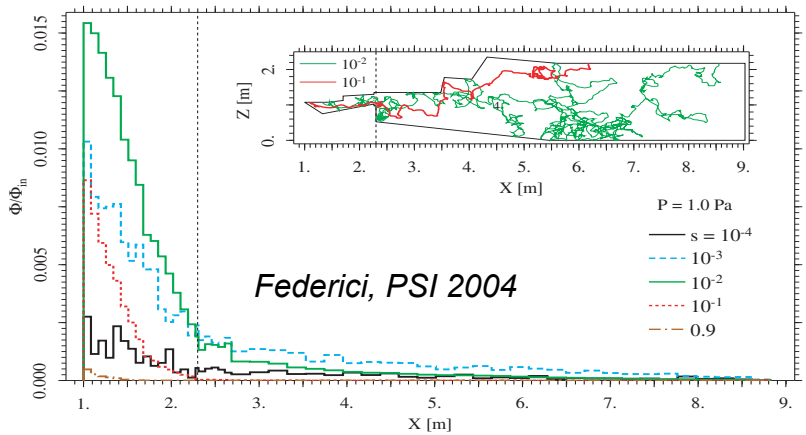
Deposition in the various regions of the model for different values of sticking - 1 Pa.







Rohde, PSI 2004

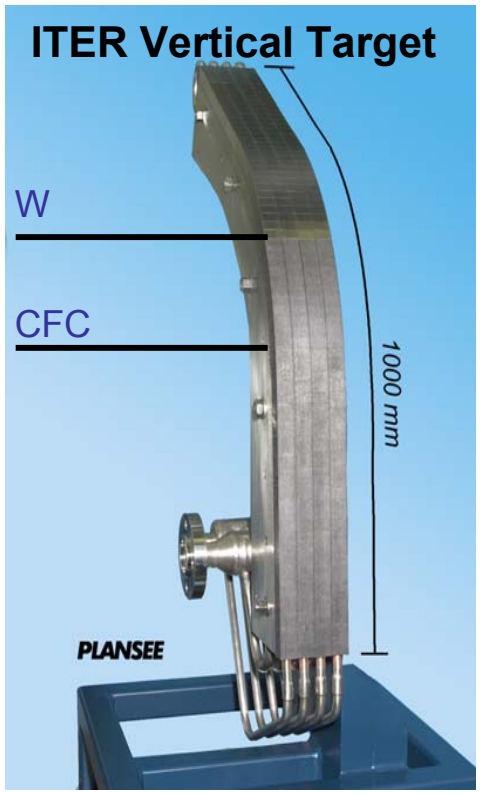


Federici, PSI 2004

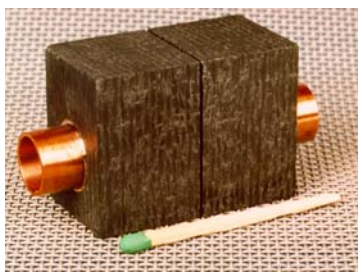
Deposition pattern along the duct for different values of sticking at 1 Pa.

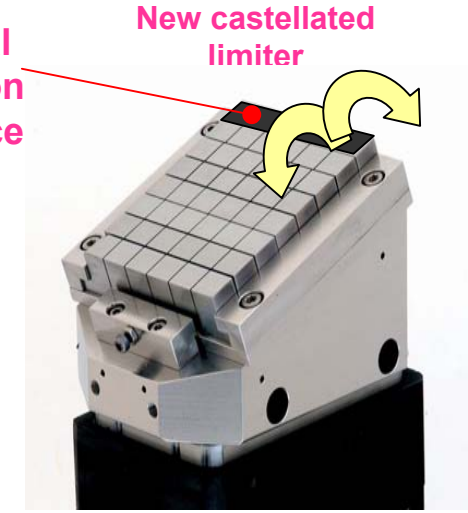
- In agreement with experimental findings in existing tokamaks we find that:
  - ✓ *majority of radical species entering the divertor private regions will stick on the nearby surfaces (i.e., underneath the dome or in the sub-divertor region);*
  - ✓ *only a small fraction of low sticking probability species can enter the pumping duct;*
  - ✓ *only species with very low sticking probabilities ($\leq 10^{-3}$) can reach the pump.*

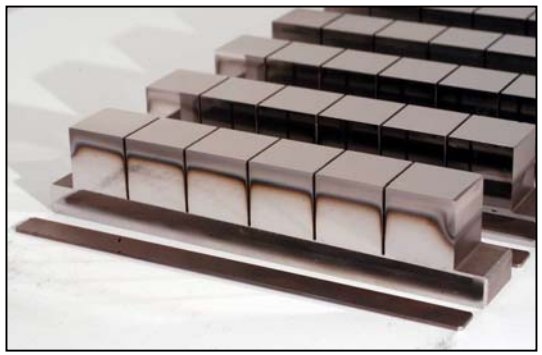
C -deposition in PFC gaps



CFC Monoblock







General parameters:

*N_e=(3.5=>4.3)*10 ^{13} cm^{-3};*

I_{plasma}=350 kA; B_t=2.25 T;

NBI1=1.2 MW; 39 effective shots;

Plasma duration 219 seconds;

Radial position: R=47.5 cm T_{lim}=200-260^{0}C

Experiments on TEXTOR

The castellated limiter with ITER-like geometry of cells was exposed to plasma in TEXTOR to assess the fuel accumulation in the gaps.

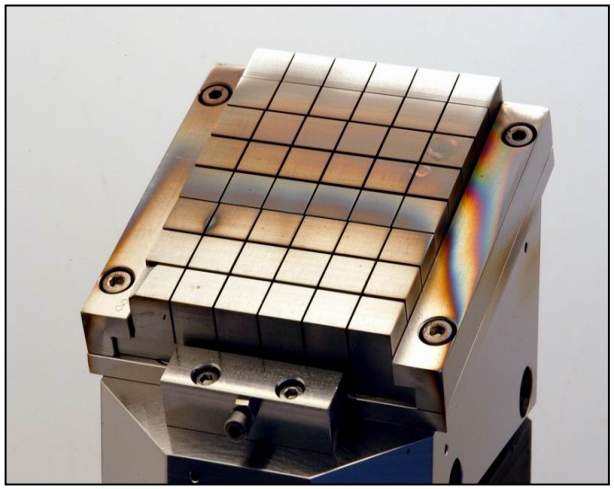


Photo: castellated limiter after exposure in TEXTOR. Dimensions of castellation: 10x10x10 mm, gaps width 0.5 mm.

Gaps contain at least 30% of fuel deposited on plasma-facing surfaces

V. Philipps, A Litnovsky (FZJ)

What Materials Strategy towards a Reactor? (I)

ITER should be an essential element in validating this strategy

ITER

- Low-Z materials are "marginally acceptable" for ITER but clearly inadequate for a DEMO and future fusion power reactors.
  - CFC chemical erosion, sublimation and tritium co-deposition
    - **Short erosion lifetime**
    - **Needs efficient technique to minimise/recover tritium in the co-deposited layers**
  - Be low melting points limit its application to the main chamber wall
    - **Erosion during mitigated disruptions/VDE**
    - **Moderate erosion and material mixing in the divertor.**
- W is seen by many as the only credible candidate. None the less, much work is still needed. Main problems with W are:
  - Power exhaust without C-radiation in the divertor.
  - HHFCs technology is ready to 20 MW/m^2 (water actively cooled).
    - **The absence of carbon as low T_e divertor radiator requires and active control of the radiated power with seeded impurities**
  - Melting and surface irregularity as a result of thermal transient events.
  - Erosion and control of W impurity content to prevent plasma contamination.
    - **$C_W \geq 10^{-4}$ in the central plasma would lead to unduly high radiation losses that would prevent ignition.**
- ITER should be an essential element in validating this strategy.

What Materials Strategy towards a Reactor? (II)

Risks and benefits!

ITER

- As preparation for DEMO, ITER would probably have to test a high Z first wall.
- In principle there is no technical problem with increasing the W coverage of the ITER first wall but that the physics implications are not yet clear.
- If we start as we plan with a different option we need to replace the 1st wall.

Risks

Option 1: Start with Be

- ✓ Change-out of the whole first-wall is a very challenging task.
  - o*Technical feasibility issues, consequences on machine downtime*
  - o*Remote handling optimisation*
  - o*Cost and machine down-time implications*
- ✓ Uncertainties of T retention and mixed-materials during early operation. Show-stoppers?

Option 2: Start with W (everywhere) *only in combination with better control of ELMs and disruptions* *(Promising results from ASDEX Upgrade and other tokamaks)*

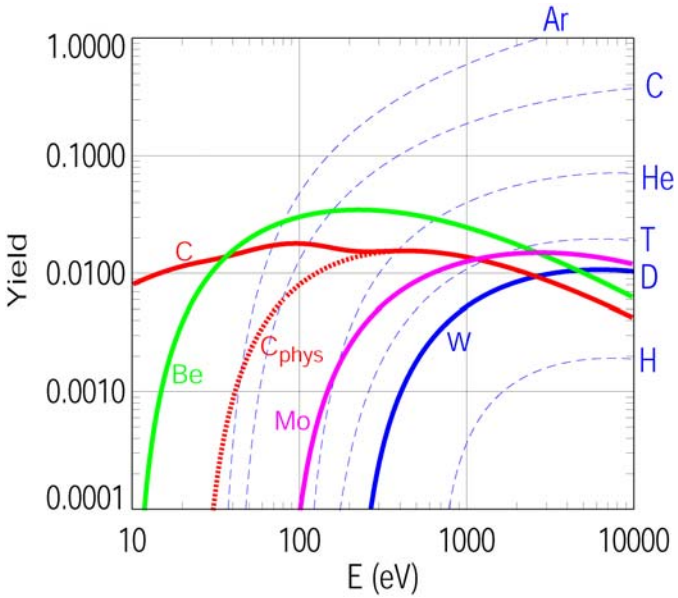
- ✓ Need to develop compatible operating scenarios.
- ✓ Accept a possible reduction of ITER operating space.

What are the Main Problems with High-Z? (I)

Plasma contamination and melting

ITER

High sputter threshold for high Z ➔ low erosion



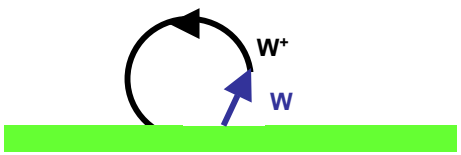
Control of steady-state and transient power loads essential in ITER

G. Matthews, PSI 2004

Ion impact energy

$$E_i = 3ZT_e + 2T_i$$

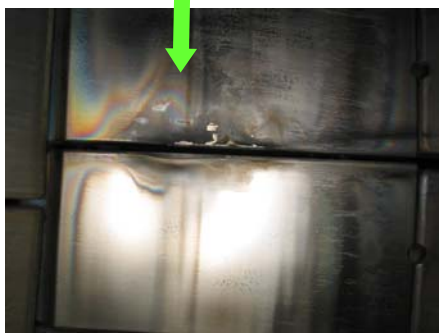
Prompt re-deposition also helps high Z at divertor



- Very low target erosion in C-Mod (Mo) ➔ 4.5mm per year (continuous) at outer strike point
- In ASDEX and C-Mod erosion dominated by impurity sputtering

ASDEX, C-Mod

Melting



ASDEX Upgrade W Upper divertor tile

C-Mod Mo divertor baffles

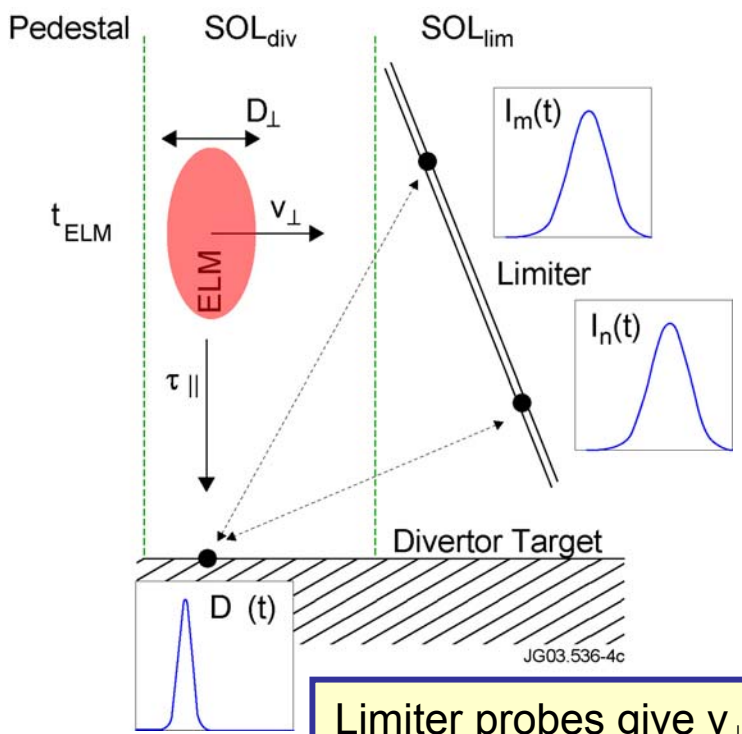
What are the Main Problems with High-Z? (III)

What is the ion impact energy during ELMs at the wall?

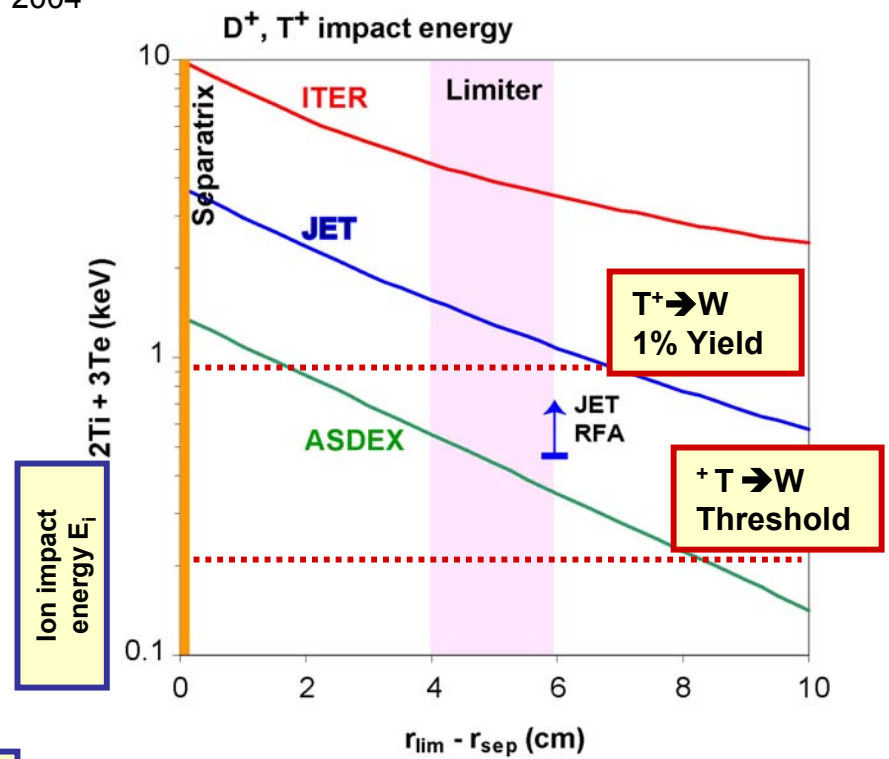
➔ *Increased erosion of high Z walls in large tokamaks*

Plasmoid model has been fitted to JET data

Fundamenski, PPCF2004



Limiter probes give $v_\perp \approx$ 1km/s (similar value predicted for ITER)



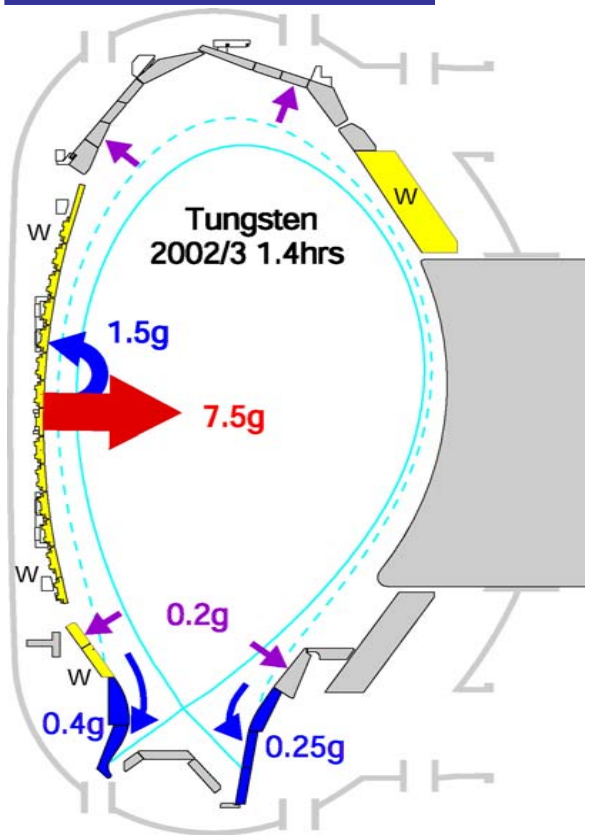
G. Matthews, PSI 2004

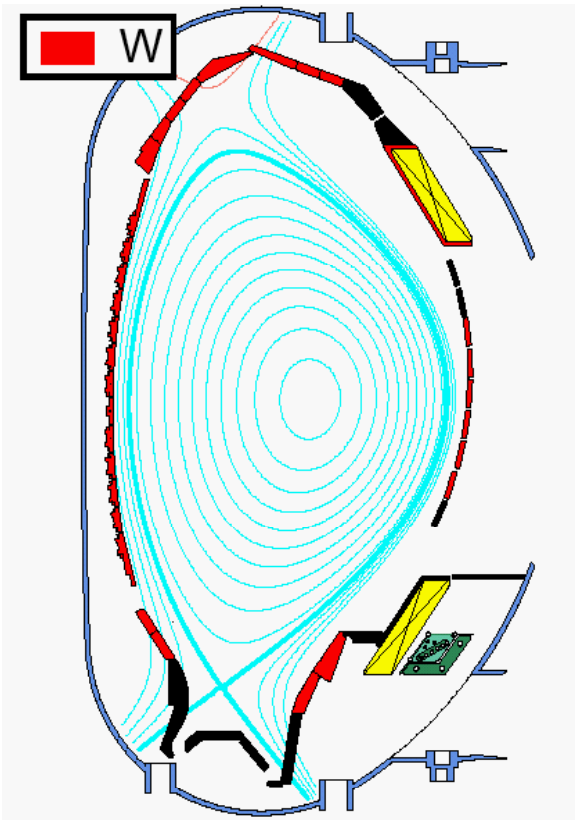
What are the Main Problems with High-Z? (II)

Control of W concentration in ASDEX Upgrade

ITER

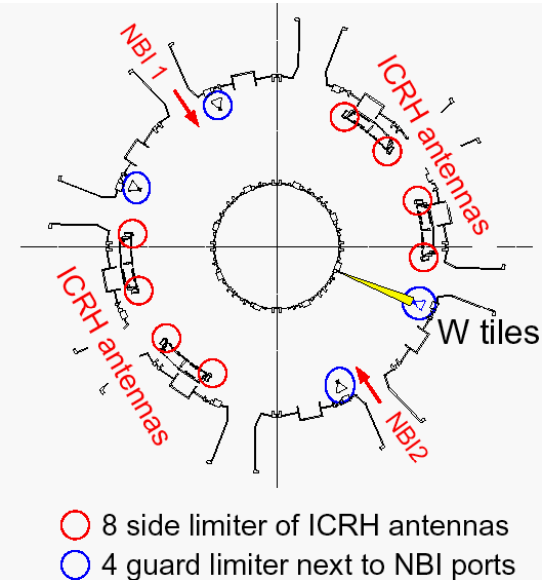
W eroded by impurities
➔ mainly limiter phase





- **Step-by-step increase of W-coated plasma facing components towards a Carbon free ASDEX-Upgrade.**

At present, 65% of total area of PFCs is made of W.



W migration in ASDEX - Upgrade

Krieger, PSI 2004

Total C Deposition
220Kg / year➔ same as JET

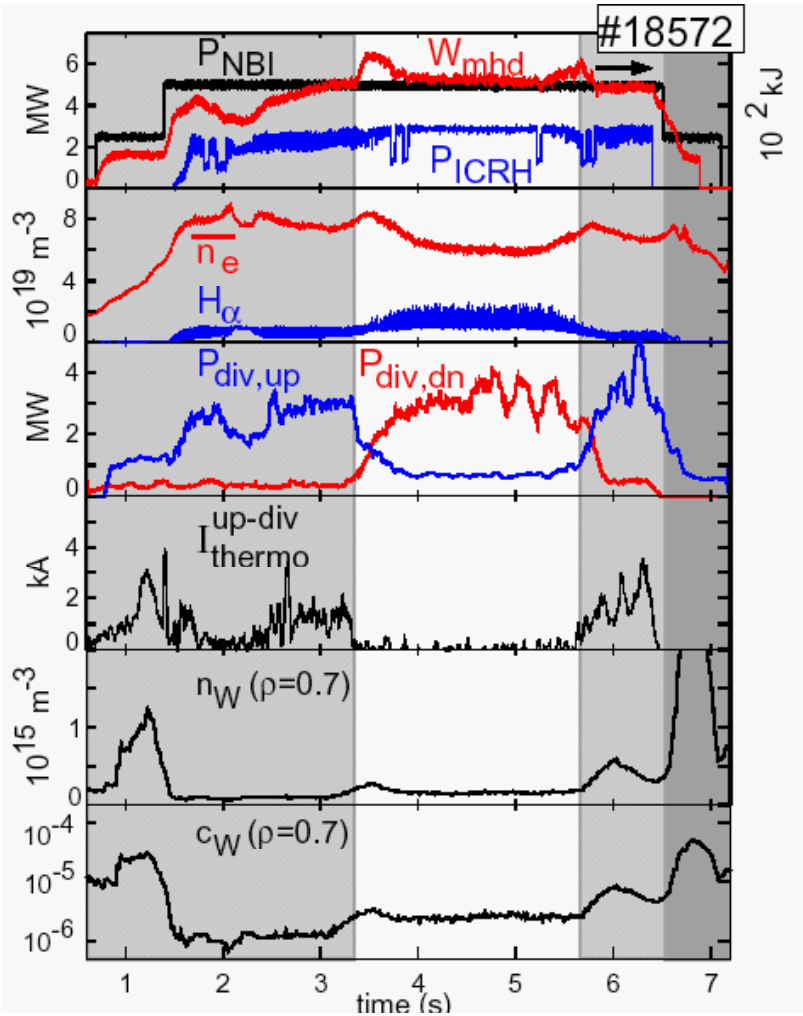
Ralph Dux, PSI 2004

What are the Main Problems with High-Z? (II)

Control of W concentration in ASDEX Upgrade

ITER

- C is still main impurity: typically 1% level
- W concentrations (critical value of 10^{-4} in reactor): typ. $<10^{-5}$
- Conditions for low W-concentration
  - Divertor configuration
  - Control impurity transport in H-mode edge transport barrier - avoid long ELM free H-phases
    - Stay away from H-L threshold: sufficient heating power.
    - Near H-L thershold: ELM pace making.
  - Control impurity transport in plasma centre.
  - Avoid peaked density profiles and low anomalous transport (=>neoclassical accumulation of W)
    - Use central heating by ECRH/ICRH.



Ralph Dux, A. Kallenbach, PSI 2004

Open Questions and Future Work

Still few years before starting PFC material procurement for ITER

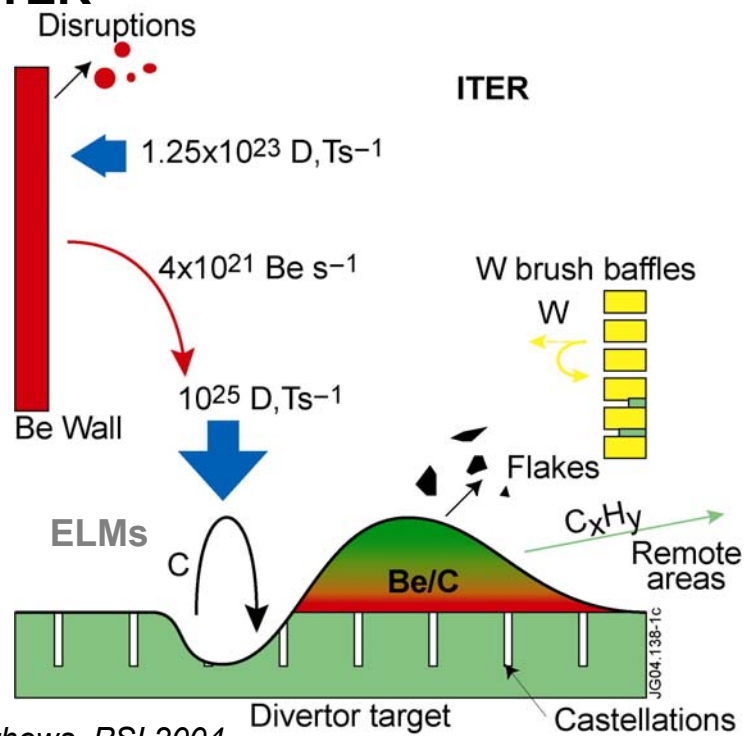
ITER

- Type I ELM and disruption heat loads - energy loss, energy to divertor and wall, duration, broadening, impurities, etc.
- Control/mitigation of ELMs and disruptions.
- Damage effects due to disruptions (and mitigated disruptions) and ELMs.
- Mixed material effects: erosion, T-codeposition, T-removal.
- The issue of operation with an all metal wall (high or low Z) and no carbon still needs to be explored in current machines and scaled with machine size.
- Modelling of W as option for ITER first wall and divertor:
  - investigate issues of plasma compatibility, performance during various phases of operation arising from use of W in ITER on: start-up limiter/ full W target/ main wall.
- Validation/improvement of C erosion/deposition codes.
- Mitigation of co-deposition (temperature tailoring, reactive species (N_2)).
- Carbon removal from hidden areas - e.g. baking with oxygen.
- Plasma interaction with irregular and/or molten W surfaces.
- PMI measurements/diagnostics.

Concluding remarks

Opportunities and challenges

ITER



Matthews, PSI 2004

- Some risk to construct ITER without first testing proposed material mix in existing tokamaks (Be wall in JET and full-W ASDEX-U). **Now is the right time!!!**
  - *According to ITER construction plans, **still few years** remain for further R&D and physics input to divertor/first-wall design and PFM choices.*

DEMO

- Disruptions and ELMs will be strictly regulated in ITER and not permitted in DEMO.
  - Materials protection schemes ➔ ITER operating instructions
- Low-Z materials are “marginally acceptable” for ITER but clearly inadequate for DEMO.
- W is seen by many as the only credible candidate. Nonetheless, much work still needed.
- Erosion and control of W impurity content to prevent plasma contamination is one of the toughest problems.
  - Transient free cold edge for DEMO
- Also brittleness in a neutron environment needs improvement.
- ITER should be an essential element in validating strategy towards DEMO, and probably will need to test a W first-wall.