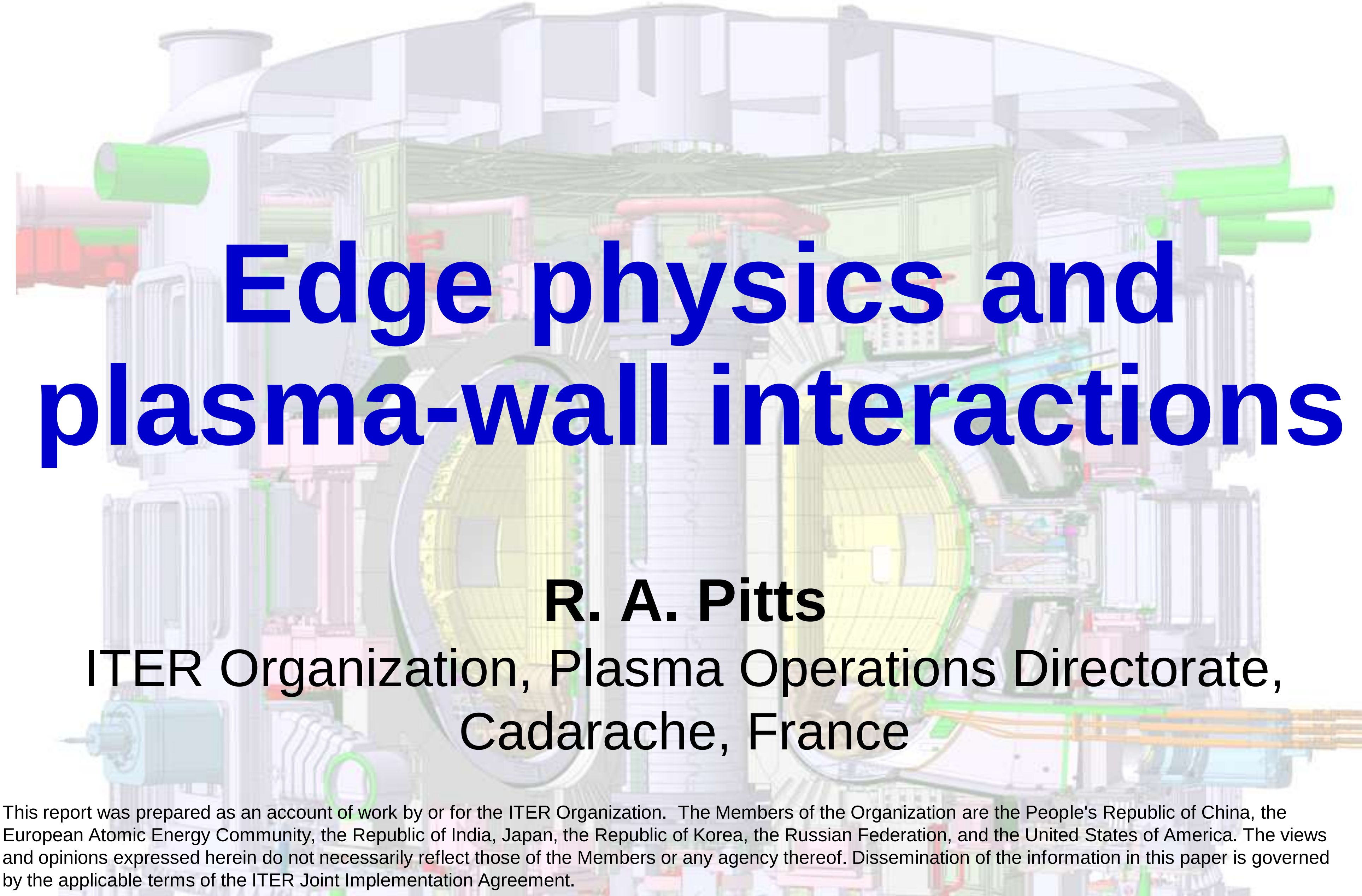

Edge physics and plasma-wall interactions

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Importance of plasma-surface interaction

"The interaction of plasma with first wall surfaces will have a considerable impact on the performance of fusion plasmas, the lifetime of plasma-facing components and the retention of tritium in next step burning plasma experiments"

Progress in the ITER Physics Basis, Chap. 4: "Power and particle control", Nucl. Fusion **47** (2007) S203-S263

CAVEAT: Edge plasma physics and PSI is a vast domain. Can only scratch the surface in a single tutorial. Work referenced throughout the talk is listed at the end.

These slides in this tutorial are rather dense and are to be used more as reference material. More important to listen to the speaker!

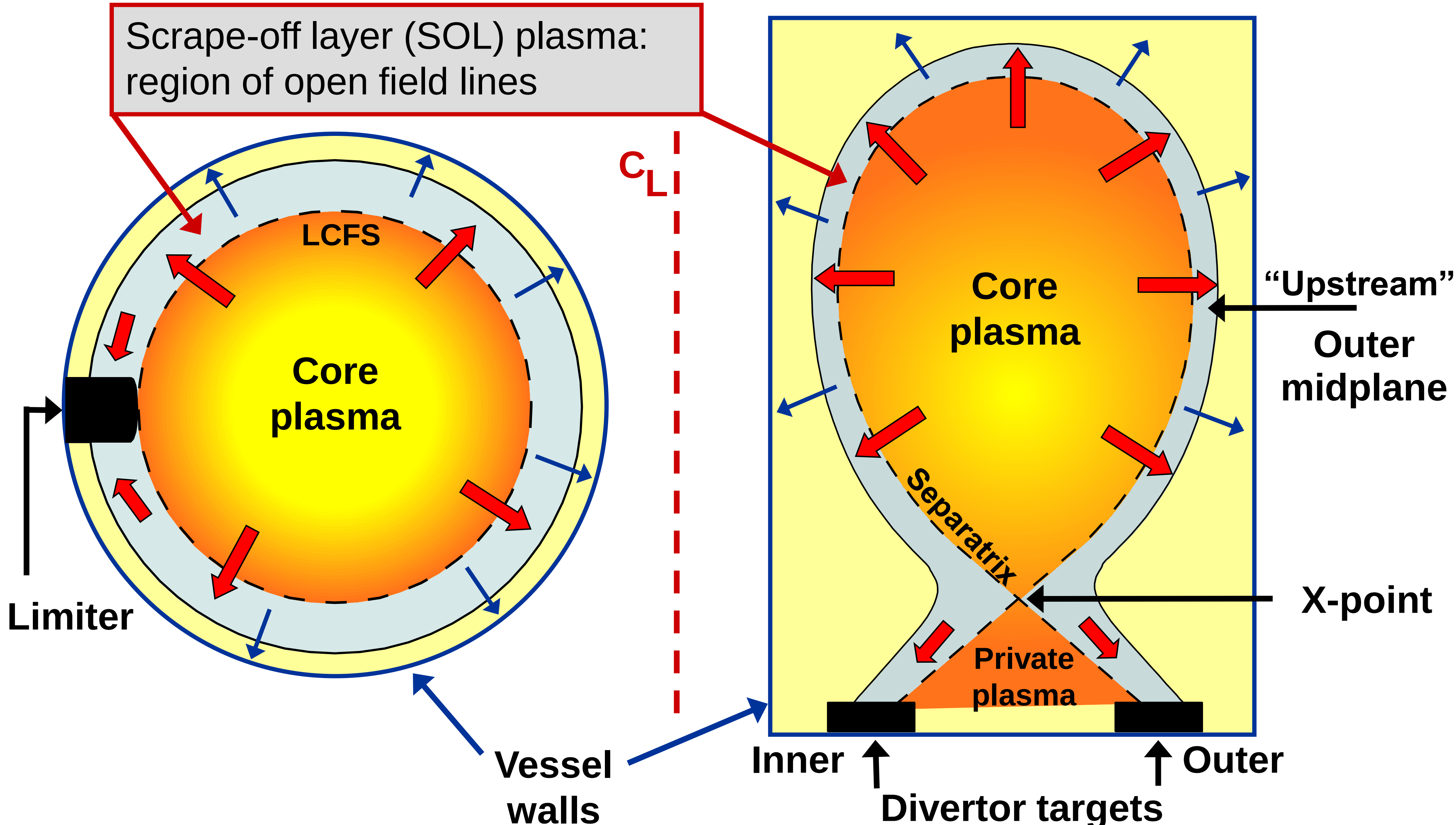
Outline

- Part I: The scrape-off layer (SOL) and divertor
  - SOL particle and power widths
  - Divertor detachment
  - Turbulent transport and SOL flows
- Part II: Plasma-surface interactions
  - Material lifetime – erosion and migration
  - Transients (ELMs and disruptions)
  - Tritium retention
  - Dust

Part I

Divertor and SOL physics

Terminology: limiters and divertors

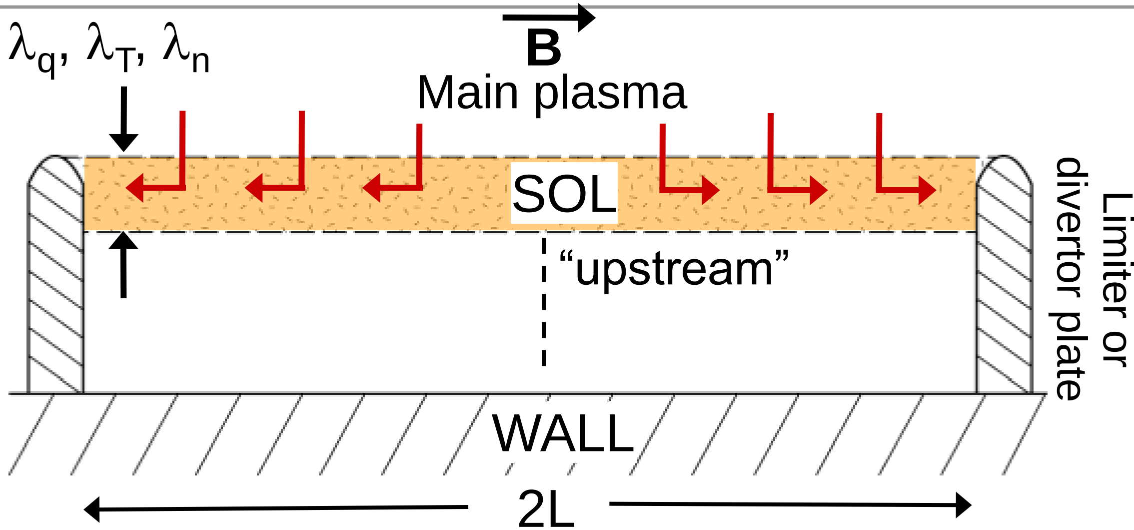


e.g. Limiter and divertor phases in many JET shots

JET #62218, H-mode, I_p = 3.0 MA, B_φ = 3.0 T – notice the “ELM bursts” – more later

Part of the ITER ramp-up and ramp-down will be in limited phase – but quite short → ~10 s. Full burn divertor phase of ~400 s for the Q_{DT} = 10 inductive scenario

Basics – SOL density width, λ_n [1]



Adapted from [1]

- Any solid surface inserted into a plasma constitutes a very strong particle sink
- In the high tokamak B-field: $\Gamma_\perp << \Gamma_{||}$
- Thin Debye sheath (λ_D few 10's μm thick) forms at the surface → controls flow of particles and energy || **B**

Quick and dirty estimate of λ_n with diffusive approx. for cross-field particle transport (all ionisation inside LCFS):

$\Gamma_\perp \equiv nv_\perp = -D_\perp dn/dr \sim D_\perp n/\lambda_n$ →

$v_\perp \approx D_\perp/\lambda_n$, $\lambda_n = \tau_\perp v_\perp$ → $\boxed{\tau_\perp = \lambda_n^2 / D_\perp}$

$v_{||} \approx c_s \sim (kT/m_i)^{1/2}$ → $\boxed{\tau_{||} = L / c_s}$

Then, if $\tau_\perp = \tau_{||}$, → $\boxed{\lambda_n = (D_\perp L / c_s)^{1/2}}$

e.g. L ~ 30 m (typical of JET):
$T_{LCFS} \sim 100$ eV, $c_s \sim 10^5$ ms^{-1},
$D_\perp \sim 1$ m^{2}s^{-1} (near SOL)
→ $\lambda_n \sim 1.7$ cm!!
cf. ITER minor radus = 2.0 m
Could be even worse for energy – see next ……

SOL power width, λ_q

- SOL width for power, λ_q, is also small and is an important parameter of the edge plasma
- As for particle transport, the physics determining λ_q is an extremely active area of research and the experts still do not agree ….
- Scalings for λ_q can be derived from models and experiments, e.g.:
  - “2-point” analytic modelling [1]: $\boxed{\lambda_q \propto P_{SOL}^{-5/9}}$ P_{SOL} = power into SOL
  - Scaling from H-mode experiments on JET [7,8]:

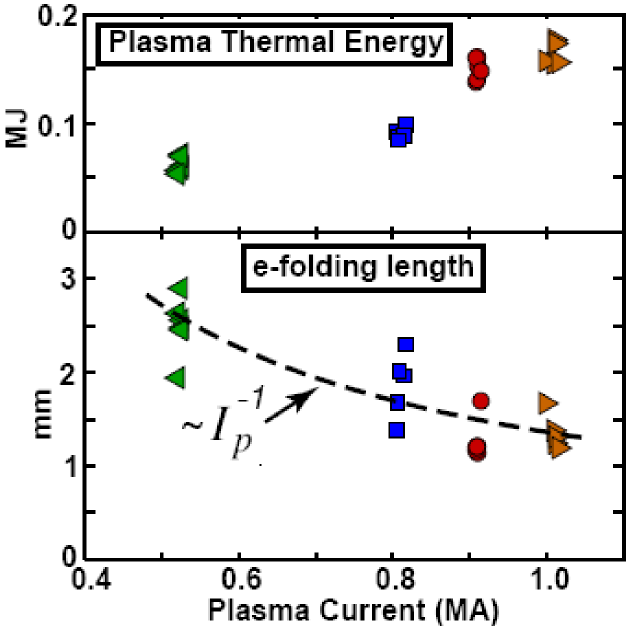
    $\boxed{\lambda_q \propto P_{SOL}^{-0.5} B_{\varphi}^{-1.0} n_u^{0.25} q_{95} R^2}$ if parallel conduction dominates

    $\boxed{\lambda_q \propto P_{SOL}^{-0.5} B_{\varphi}^{-1.0} n_u^{0.25} q_{95}^{0.5} R^{1.5}}$ if parallel convection dominates
  - ITER modelling [9] yields λ_q = 5 mm, JET scaling gives λ_q = 3.5 – 4.5 mm mm (cf. a = 2.0 m for ITER)
  - Fairly recent multi-machine scaling [10] gives λ_q/R ~ constant
  - VERY recent high resolution measurements on several tokamaks are finding $\lambda_q \propto 1/I_p$ (like JET → $\lambda_q \propto q_{95}/B_{\varphi}$)

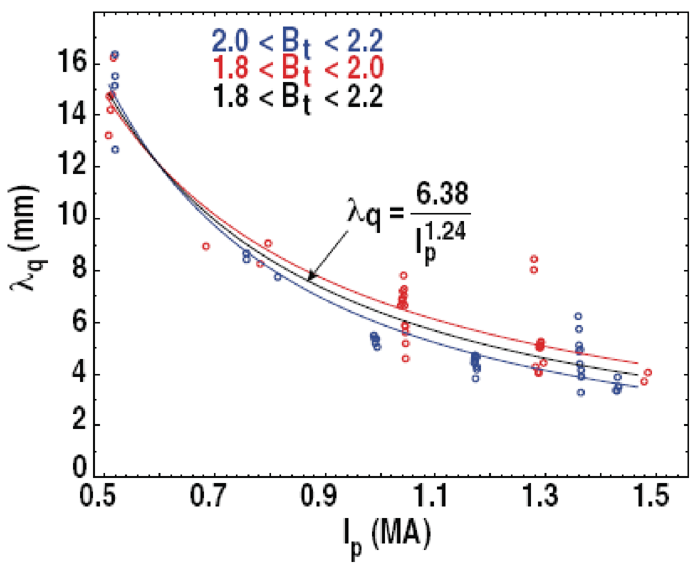
λ_q still the subject of much attention

- Very new high precision near-SOL H-mode (inter-ELM) λ_q measurements reported from NSTX, DIII-D, C-Mod and JET
  - All machines find $\lambda_q \propto I_p^{\alpha}$ in H-mode with α ~ -1.0 → -1.6 and little dependence on P_{SOL}, B_φ, n_e, ….
  - Potentially serious implications for ITER target power handling (I_p = 15 MA!)
  - What sets the value of λ_q? Parallel SOL transport? H-mode pedestal stability?

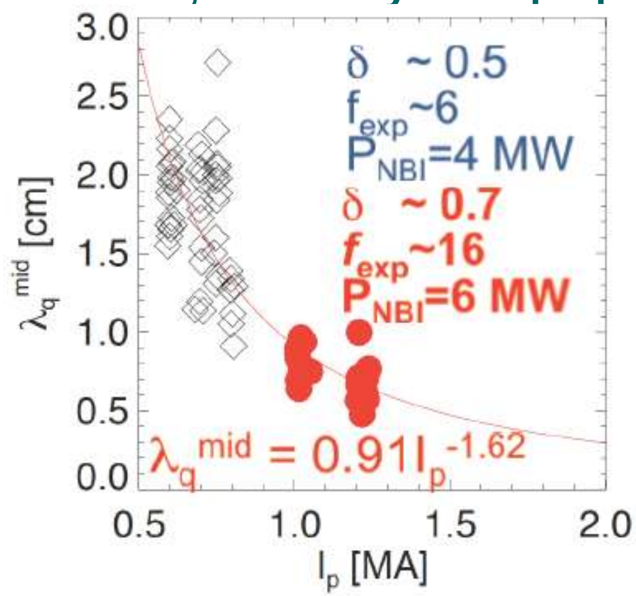
C-Mod, B. LaBombard et al. [11]



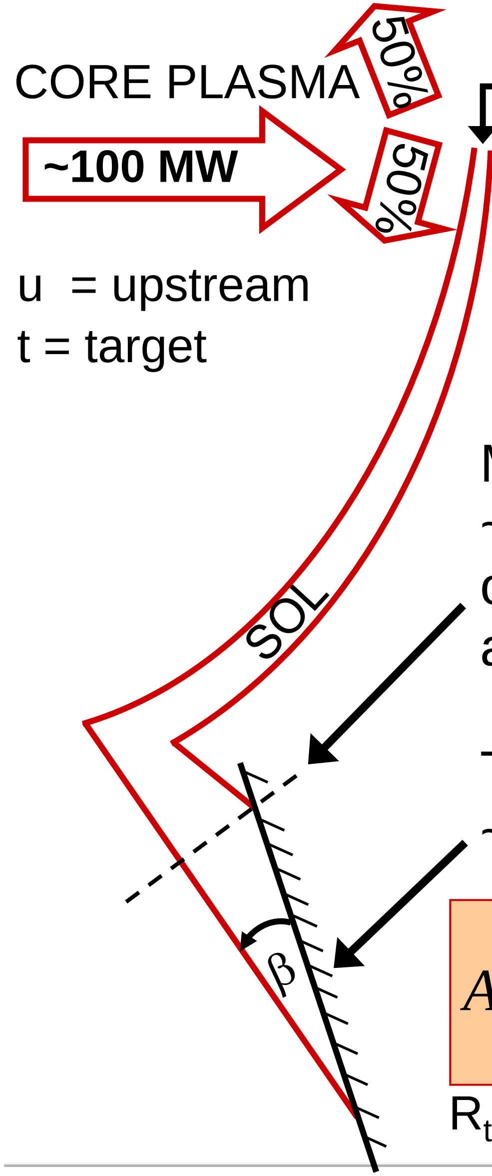
DIII-D, C. J. Lasnier et al. [12]



NSTX, T. K. Gray et al. [13]



Example power handling – ITER case



u = upstream
t = target

$$Area \sim 2\pi R_u \lambda_q (B_\theta / B)_u \sim 0.09 m^2$$

λ_q = 5 mm, R_u = 8.2 m, $(B_\theta/B)_u$ ~0.33

⇒ $q_{||,u}$ ~ 1 GWm^{-2} !!

Magnetic flux expansion ~$(B_\theta/B)_u/(B_\theta/B)_t$ ~3 for ITER outer divertor → low field line angles at strike points (α ~ 2.5°)

+

Target tilting in poloidal plane (β ~ 25° for ITER outer target)

$$Area \sim \frac{2\pi R_t \lambda_q (B_\theta / B)_u}{\sin\beta (B_\theta / B)_t} \sim 1.2 m^2$$

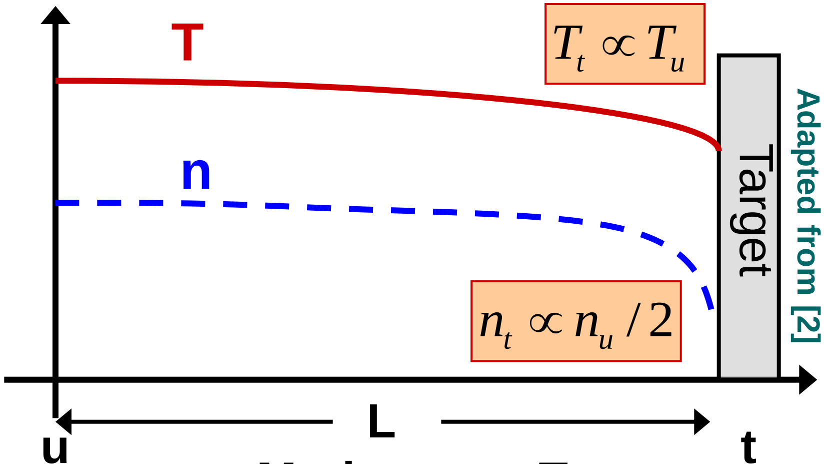
R_t = 5.6 m, $(B_\theta/B)_t$ ~0.11

- Max. steady-state power flux density permitted at ITER divertor targets: $q_\perp \leq 10$ MWm^{-2}
- Magnetic and divertor geometry alone cannot reduce the power to tolerable levels
- Most of the parallel power flux must be prevented from reaching the plates → divertor detachment and high radiative loss

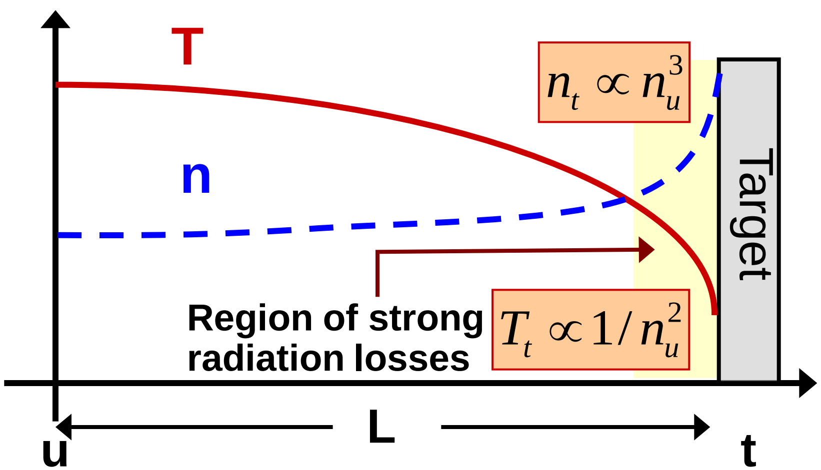
⇒ $q_\perp$ ~ 40 MWm^{-2} per target if no radiative (or other) dissipation → cannot be tolerated for more than ~2-3 s on actively cooled surfaces

The route to detachment (1)

Low n, high T (high P_{SOL})
"Sheath limited"



Moderate n, T
"High recycling"



Mean free paths for particle collisions are long: $\lambda_{coll} \propto T_u^2 / n_u, T_u \sim T_e \sim T_i, \lambda_{ee} \sim \lambda_{ei} \sim \lambda_{ii}$

SOL collisionality: $\nu^* = L / \lambda_{coll}$ low

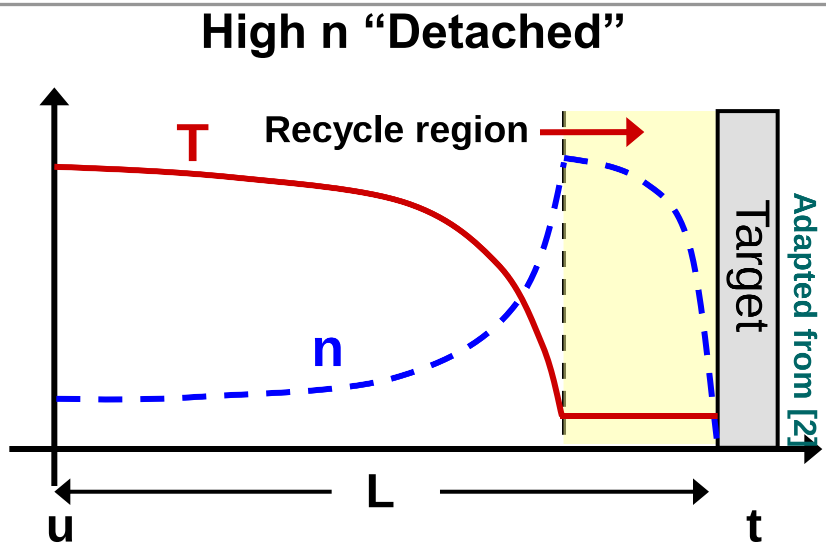
Power flow to surface largely controlled by target sheath: $q_{\parallel,t} = \gamma n_t c_{st} T_t + n_t c_{st} \varepsilon_{pot}$

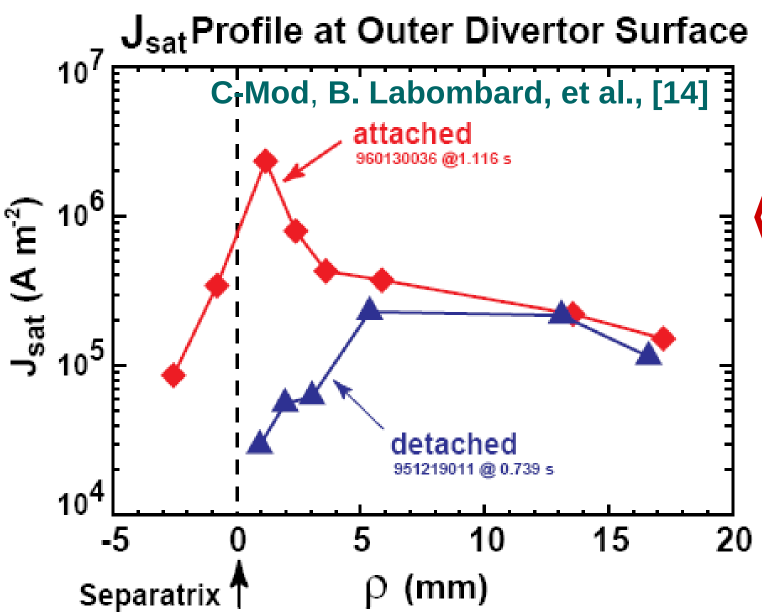
γ = sheath heat transmission coefficient

ε_{pot} = potential energy per incident ion

ν^* rises as n_u rises, finite electron heat conductivity: $q_{\parallel,cond} = -K_\parallel dT / ds_\parallel, K_\parallel = \kappa_0 T^{5/2}$
(note: $\kappa_{0,e} \gg \kappa_{0,i}$)
allows parallel T gradients to develop → T_t decreases, but pressure balance maintained ($\nabla p_{||} \sim 0$) so that n_t rises strongly ($\Gamma_t \propto n_u^2$)
λ_{ion} ($\propto 1/n_t$) decreases so that target recycling increases strongly → flux amplification
As $T_t \downarrow$, radiation loss increases → $T_t \downarrow$ further

The route to detachment (2)



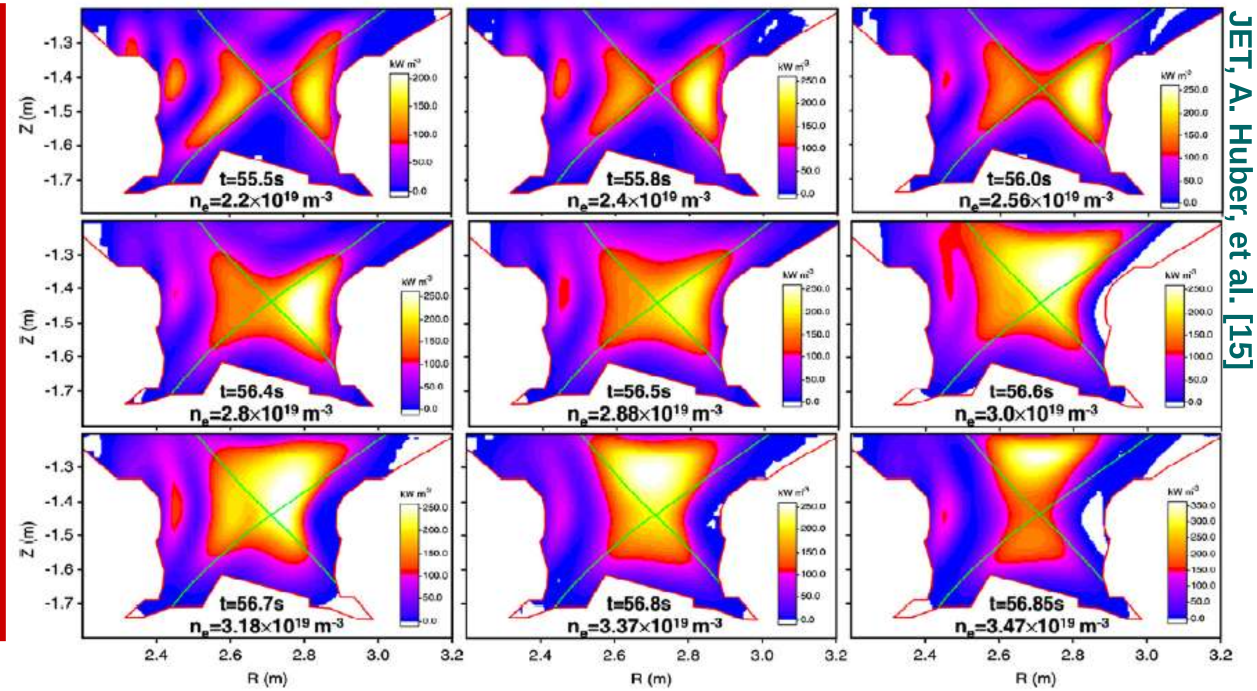


At sufficiently low T_t, (< 5 eV), neutral ionisation rate < ion-neutral friction processes (charge exchange, elastic scattering). Momentum transferred from ions to dense cloud of neutrals in front of the plate (recycle region) → begins to reduce n_t, $\nabla p_{||} \neq 0$ and plasma pressure falls across recycle region. Once T_t ~1-2 eV (and if n_t high enough), volume recombination locally "extinguishes" plasma, reducing target power flux

Detachment seen experimentally in many devices, but complex "volumetric" process – modelling still has problems to reproduce. X-point geometry → long connection lengths → high residence times in low T_e plasma → efficient radiative loss favouring power reductions where $q_{||}$ is highest (i.e. on flux surfaces near separatrix).

Full detachment is a problem

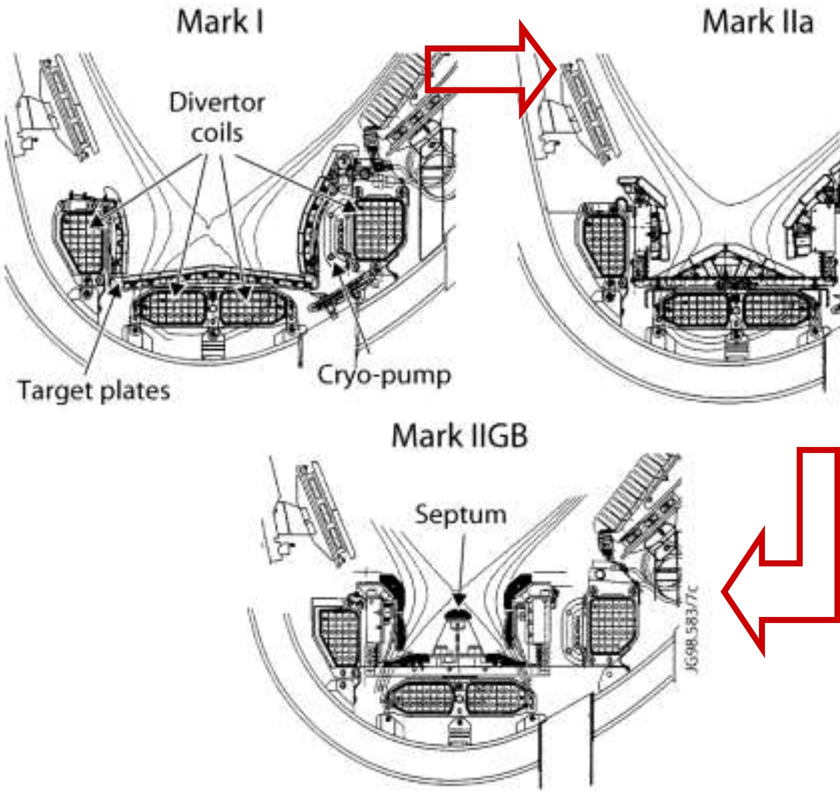
- Detachment which is too “strong” (particle flux reduced across the whole target) is often associated with zones of high radiation in the X-point region and confined plasma (MARFE)
- MARFE formation can drive a transition from H to L-mode (H-mode density limit) or disruption
- MARFE physics still not well modelled

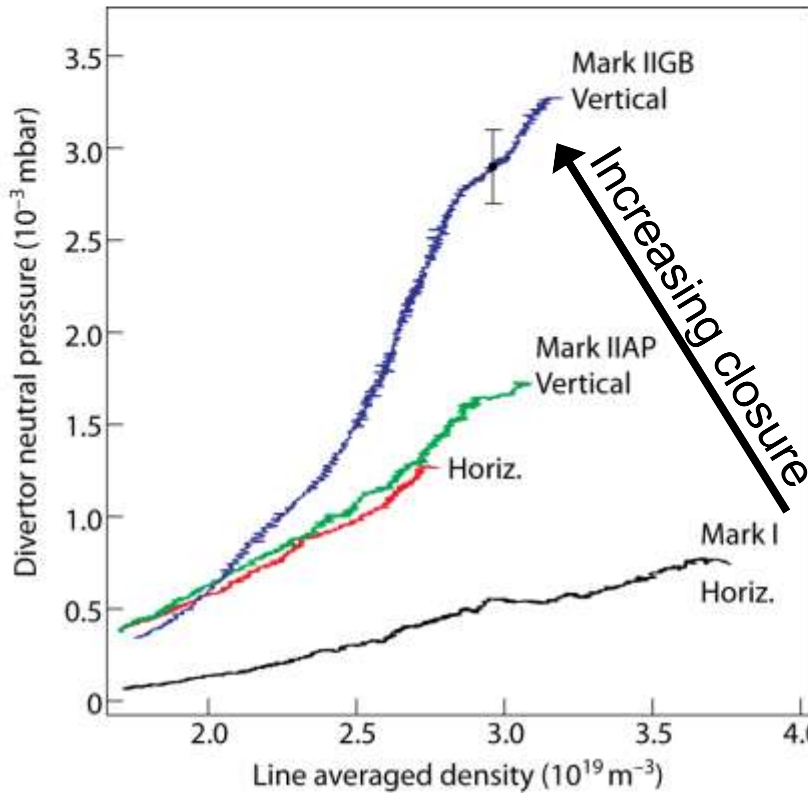


Limit detachment to regions of highest power flux (where it is needed most).
Maintain remainder of SOL in high recycling (attached)
A few ways to arrange that this happens more readily:

Divertor closure | Target orientation | Impurity seeding

Divertor closure





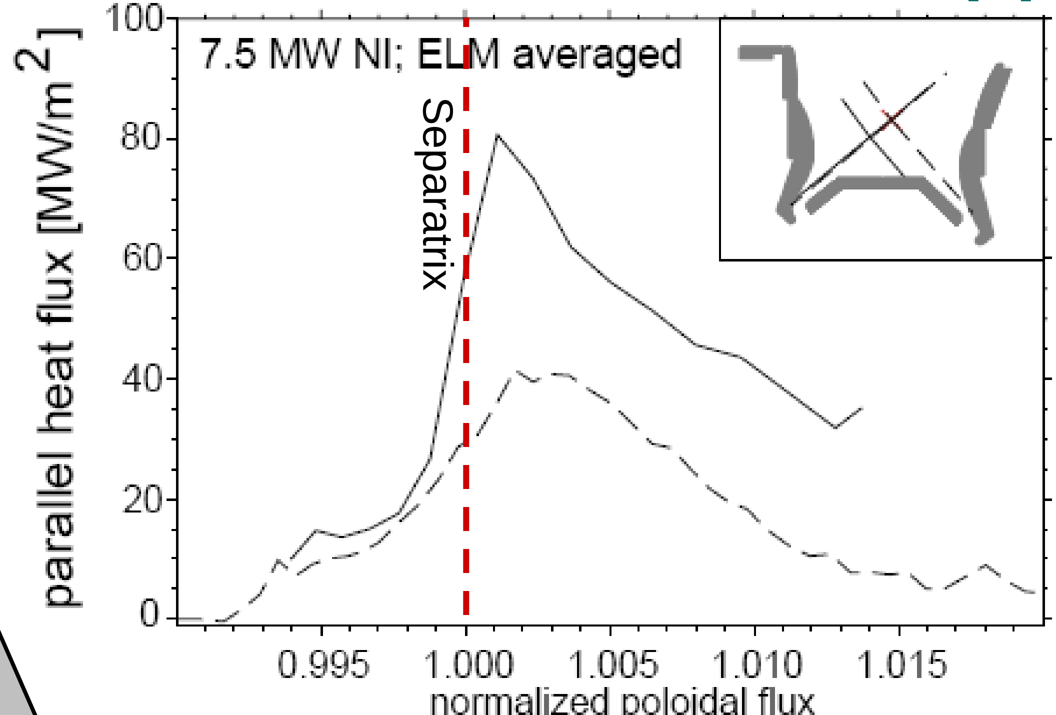
JET, R. D. Monk, et al. [16]

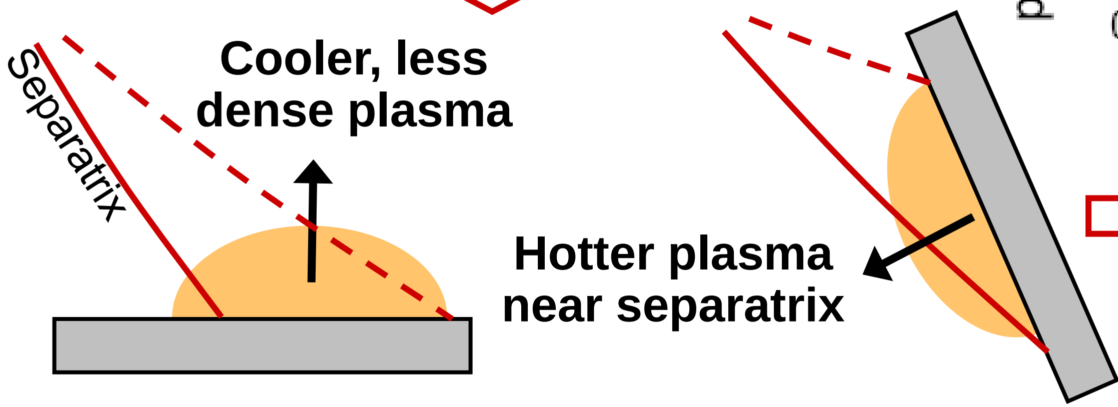
- Increased closure significantly improves divertor neutral pressure → increased neutral density (n_n), promoting earlier detachment
- Closing “bypass” leaks important for increasing n_n
- Divertor closure also promotes helium compression and exhaust – very important for ITER and reactors

Target orientation

- Parallel heat fluxes significantly reduced for vertical cf. horizontal targets
- Underlying effect is preferential reflection of recycled deuterium neutrals towards the separatrix

AUG, A. Kallenbach, et al. [17]





Increased ionisation near sep.

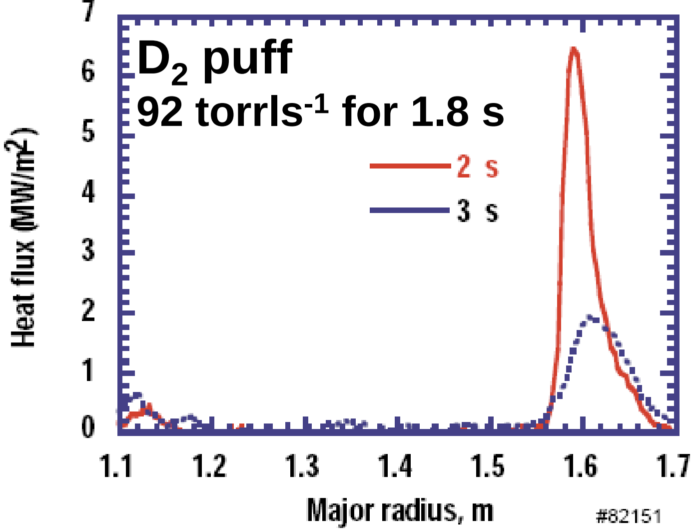
Higher n_t, lower T_t

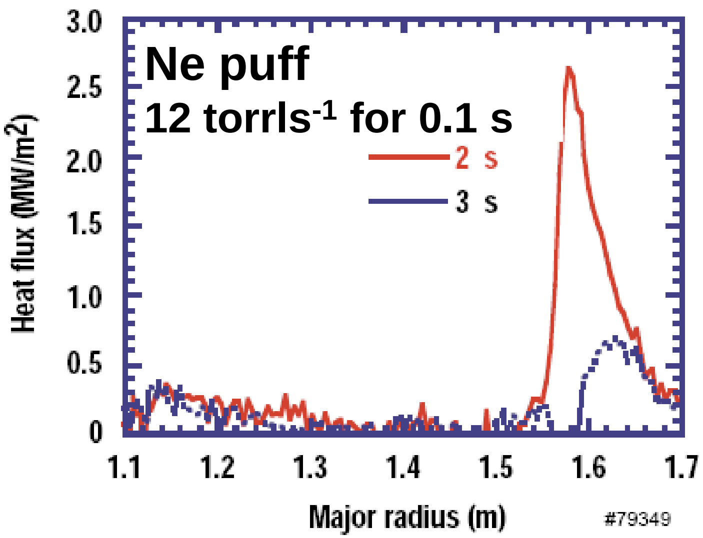
Higher CX losses

Pressure loss → $q_{||}$ ↓

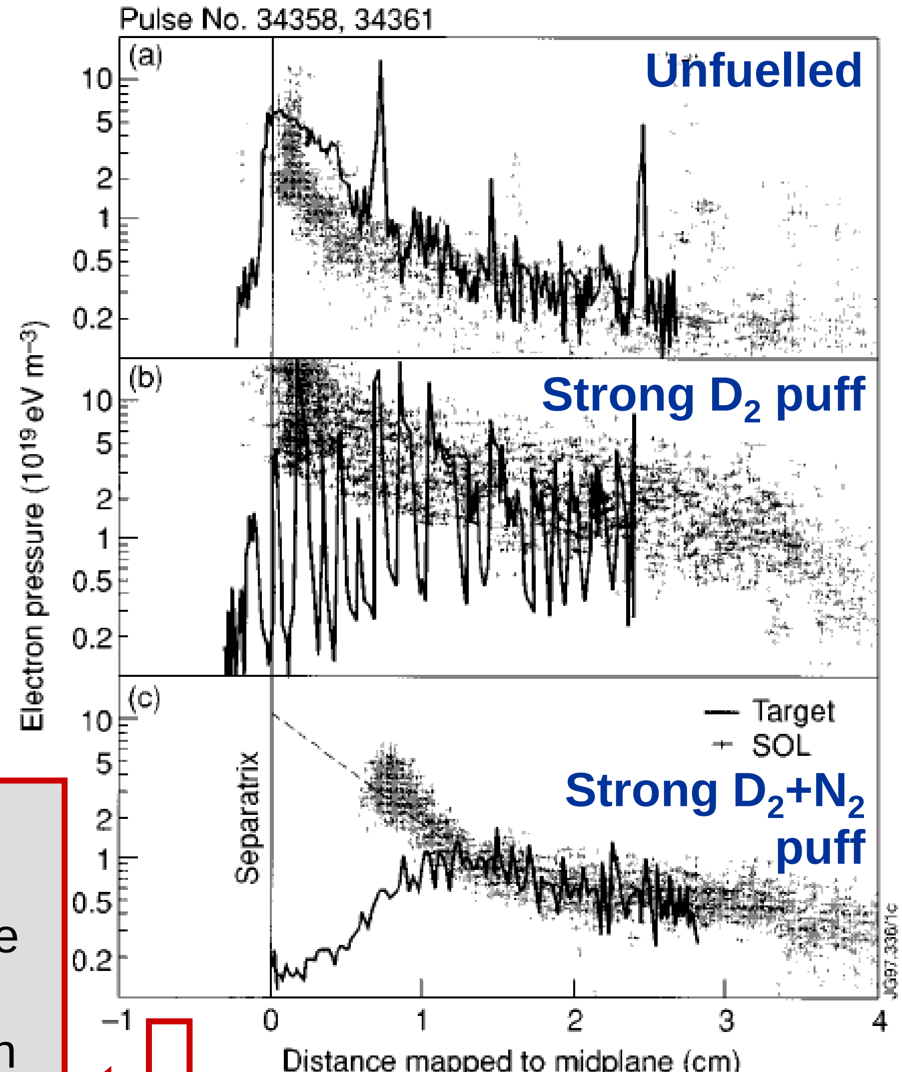
Impurity seeding

DIII-D, C. J. Lasnier, et al. [18]



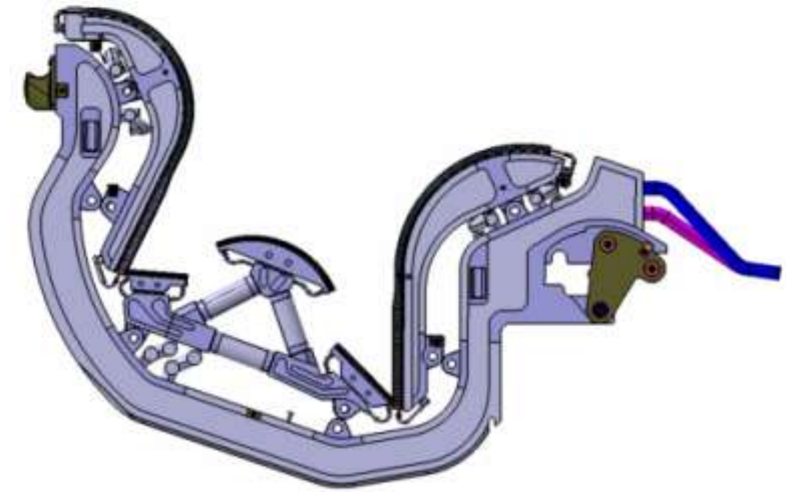


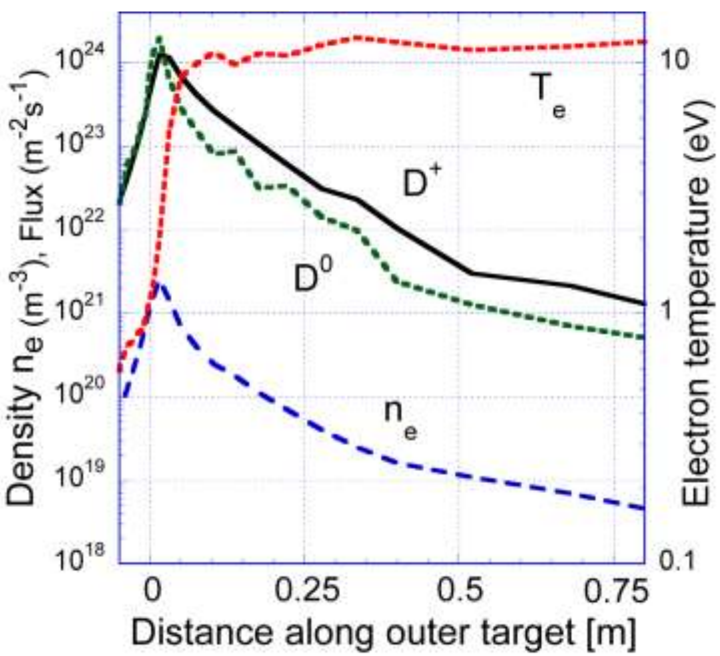
Strong impurity seeding also reduces ELM size but a price may have to be paid in confinement



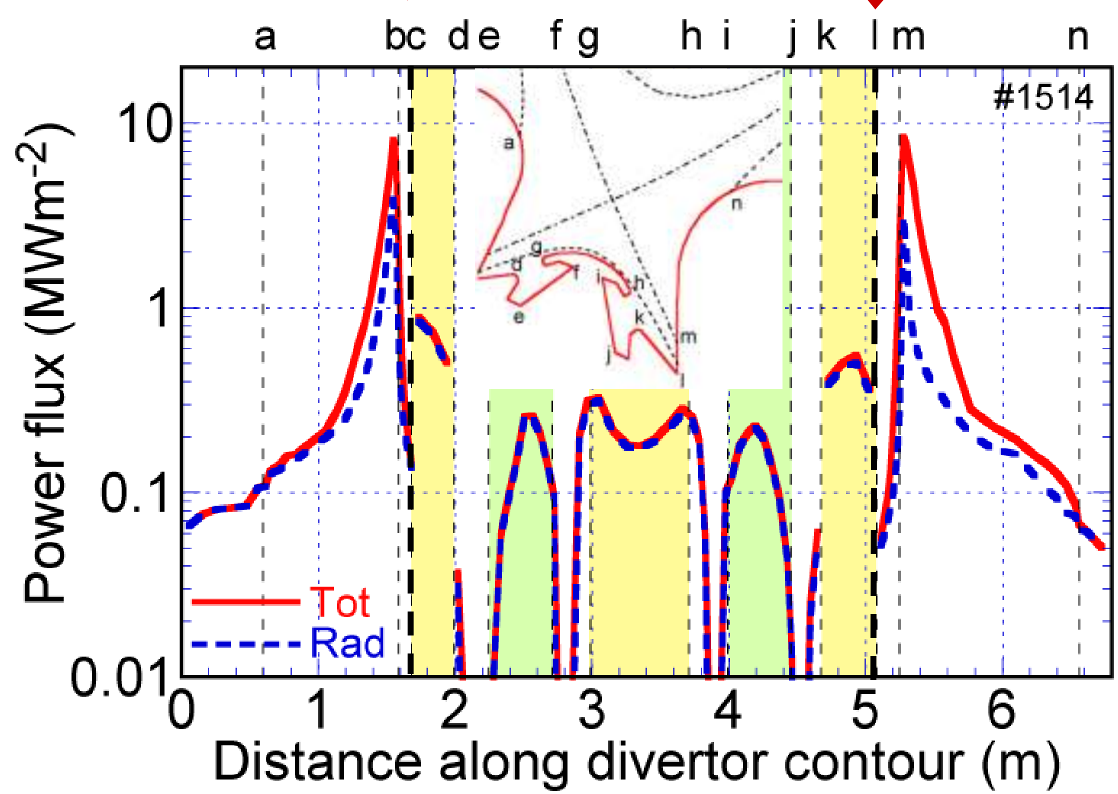
JET, G. F. Matthews et al. [19]

ITER divertor achieves partial detachment





Inner strike pt. Outer strike pt.

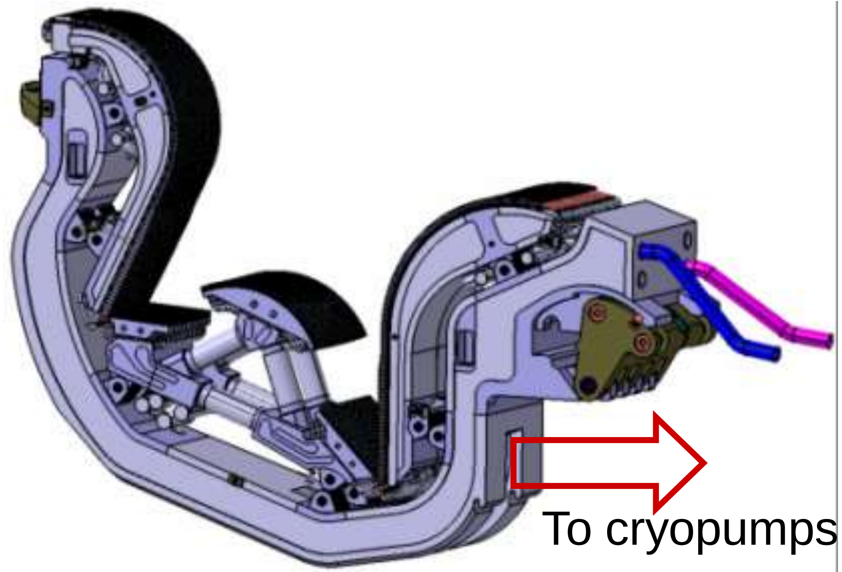


R. A. Pitts, A. S. Kukushkin et al. [20]

Deep V-shaped divertor, vertical, inclined targets
Dome separating inner and outer targets – reduces neutral reflux to the core.
Also helpful for diagnostics, neutron shielding

Divertor helium exhaust

Apart from power handling, primary function of divertor is to deal with He from fusion reactions → compress D, T, and He exhaust as much as possible for efficient pumping (and therefore also good density control).



Critical criterion for an ITER burning plasma is that He is removed fast enough such that: $\tau^*_{p,He} / \tau_E \leq 5-10$ is satisfied.

$\tau^*_{p,He}$ is the global helium particle residence time – a function of τ_p, the He neutral density in the divertor and the pumping speed (conductance) [21].

Helium enrichment: $\eta_{He} = \dfrac{n_{He}^{pump} / 2n_{D2}^{pump}}{n_{He}^{plasma} / n_e} = \dfrac{C_{pump}}{C_{plasma}}$ is the ratio of He concentration in the divertor compared to the main plasma.

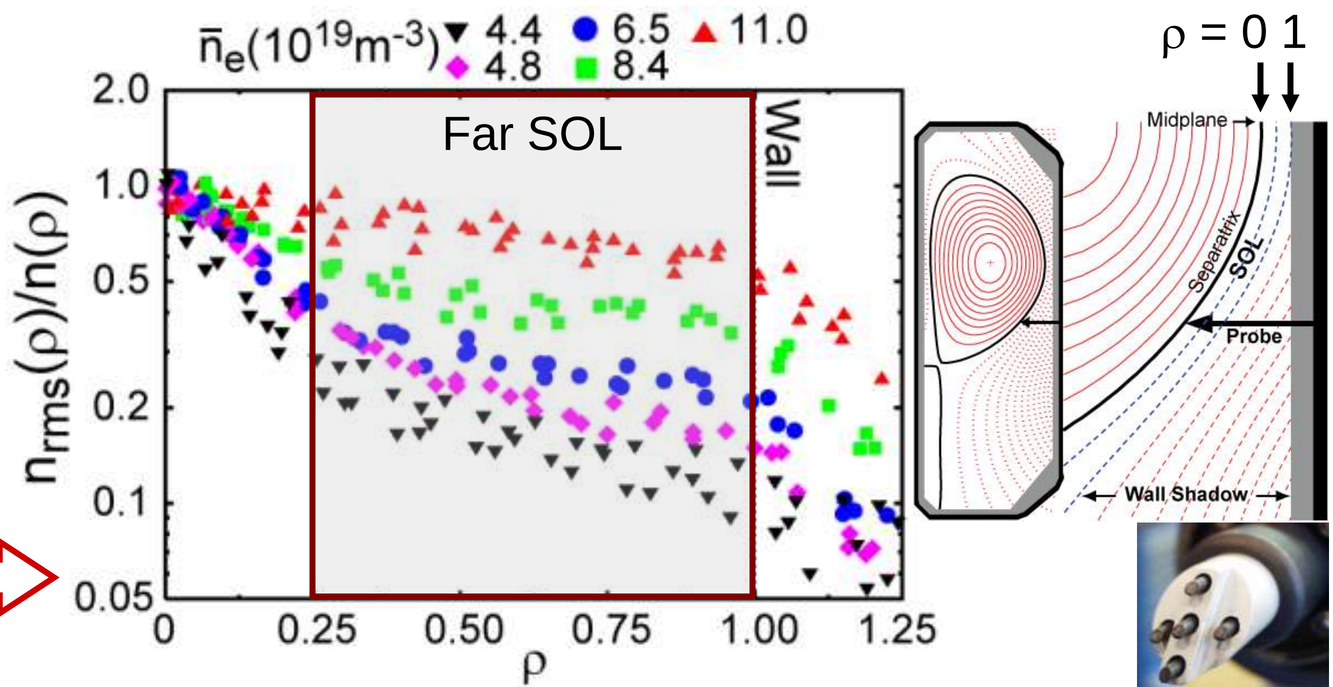
e.g. ITER: He prod. rate ~2×10^{20} s^{-1}
Max. divertor pumping speed ~200 Pa m^{3}s^{-1} ~ 1×10^{23} He atom s^{-1}
→ C_{pump} ~ 2×10^{-3} = 0.2%
Typical acceptable He conc. in the core: ~4% → η_{He} = 0.2/4 = 0.05 is minimum required. The values of η_{He} and $\tau^*_{p,He}$ required for ITER have been achieved experimentally

Perpendicular SOL transport

For many years in early tokamak research, measured density profiles in the SOL plasma often seem to obey an exponential fall off, implying that a Fick's Law type diffusive ansatz is an appropriate description, e.g.: $\Gamma_\perp = -D_\perp dn/dr \sim D_\perp n/\lambda_n$ (see slide 7)

But in fact, the SOL density profile, when looked at more closely, often has more structure, itself dependent on discharge density/ SOL collisionality

Example from the TCV tokamak



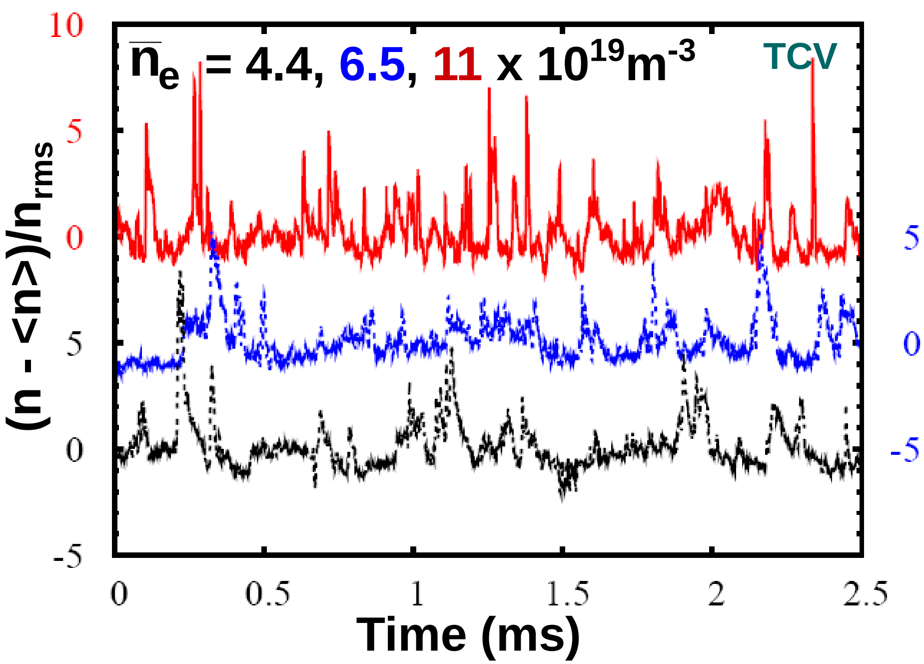
Note how broadening occurs mostly in the "far SOL"

O. E. Garcia, R. A. Pitts et al. [22]

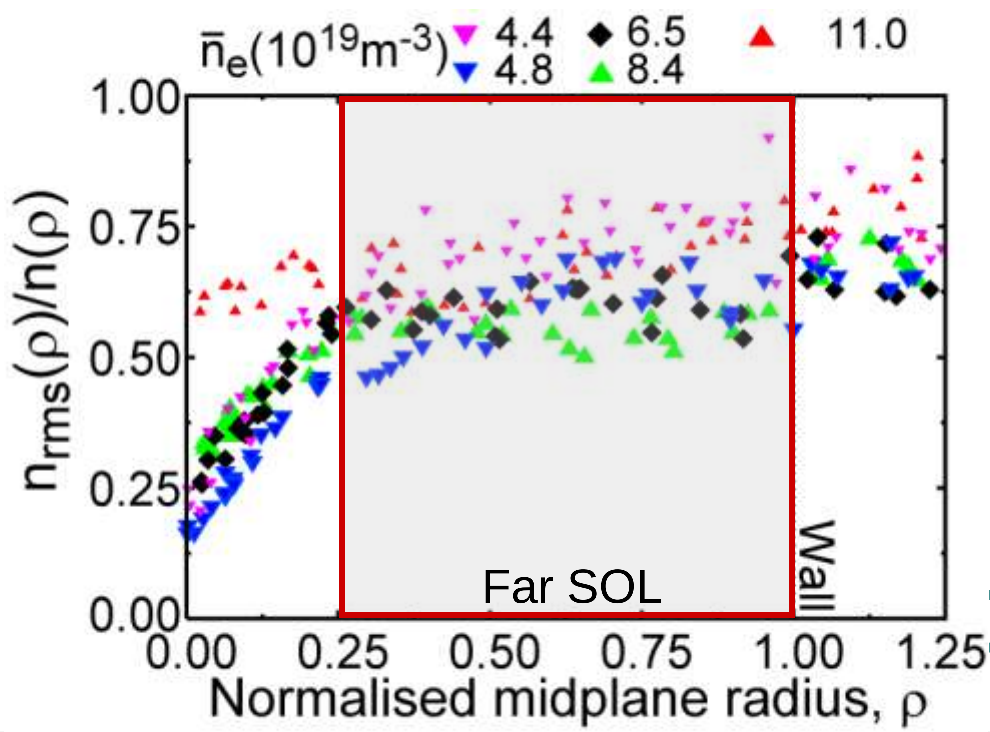
What causes the broad n_e profiles?

The particle transport is intermittent, mostly convective, not diffusive. Particle flux time series are bursty. Most of the transport occurs during the bursts

- Relative amplitude of the bursts very high: $n_{rms}/\langle n\rangle \rightarrow 1$ at high density and in the far SOL at all densities
- These bursts take the form of magnetic field aligned "filaments" as they propagate through the SOL



At high densities, bursts more frequent

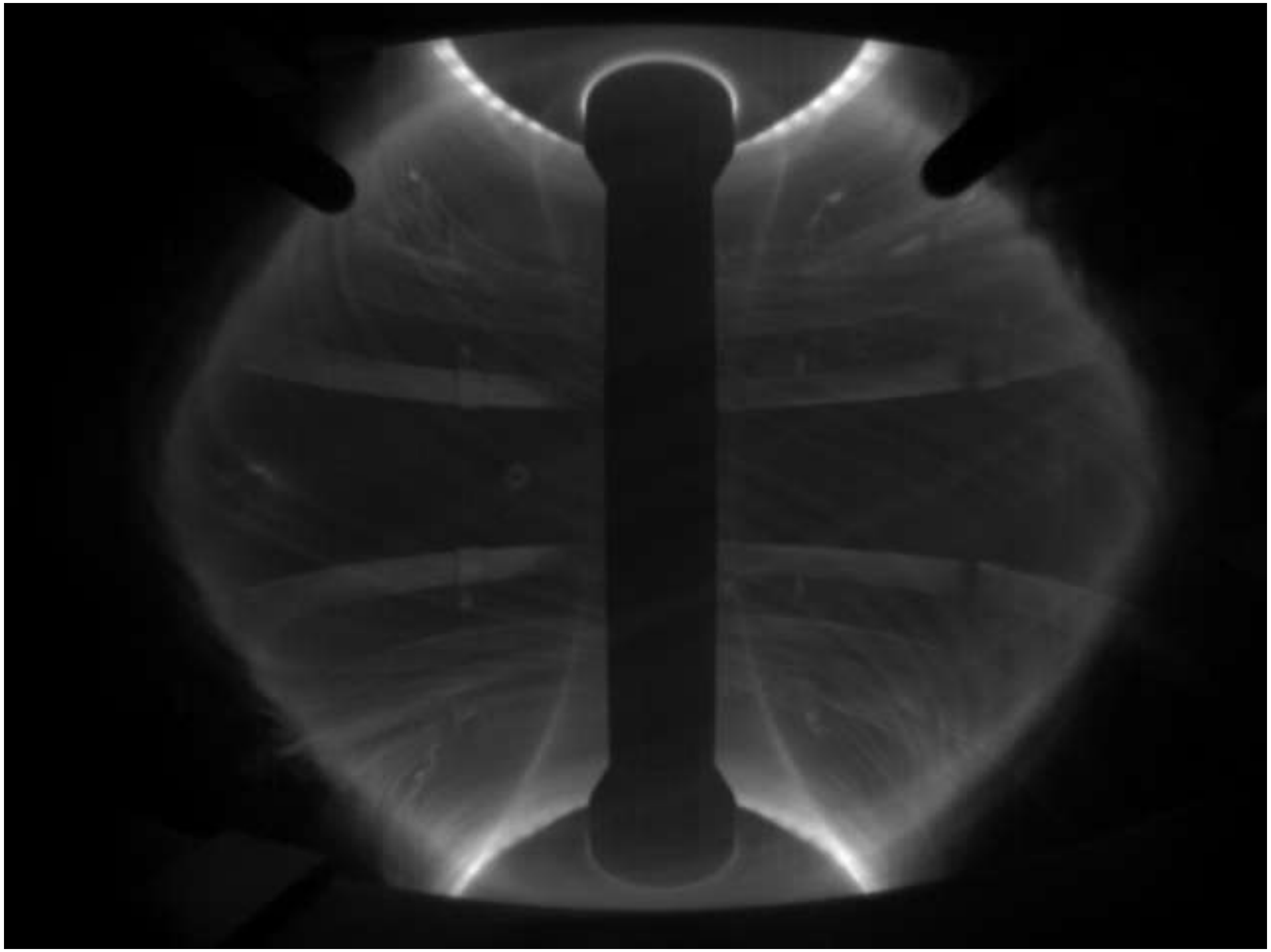


O. E. Garcia, R. A. Pitts et al. [22]

L-mode filaments on MAST

Courtesy MAST team, CCFE Culham

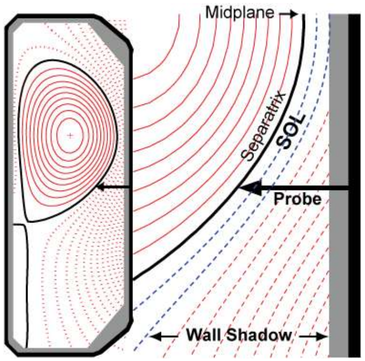
Probes provide quantitative data on the real particle flux, but only at fixed toroidal and poloidal locations. Fast visible imaging allows the 3D picture to be seen but analysis more challenging.

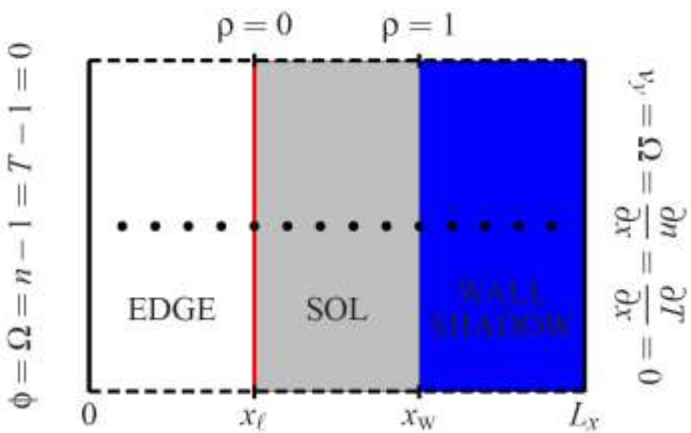


These L-mode filaments are nothing more than small amplitude versions those seen during Edge Localised Mode (ELM) events

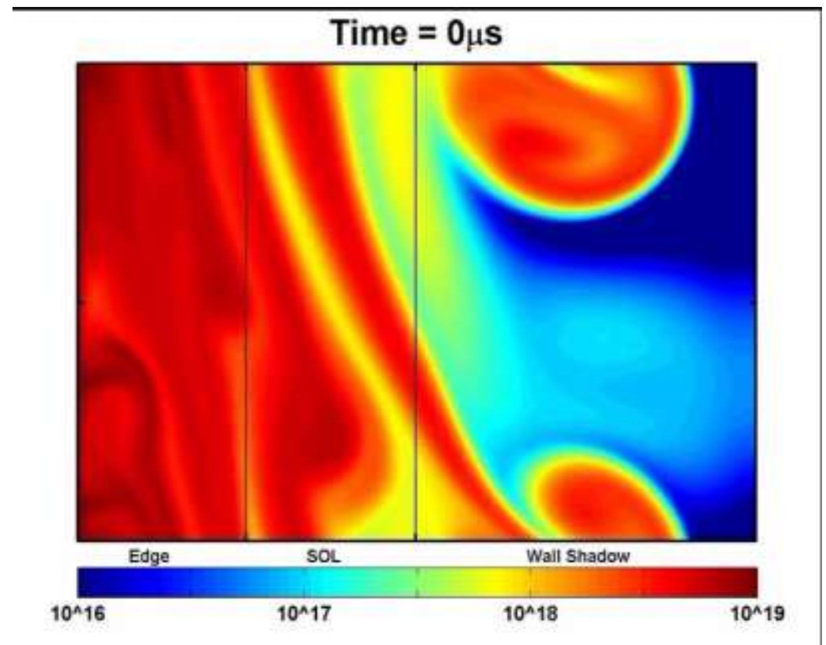
Origin of the bursts?

- Thought now to be due to electrostatic interchange turbulence produced in the near SOL region
- Local relaxations in the edge pressure profile → ejection of bursts of excess particles and heat into SOL → radial motion due to electric drift (B×∇B charge separation → E×B drift), damping by parallel losses on open field lines
- All the basic physics captured by recent 2D interchange turbulence simulations



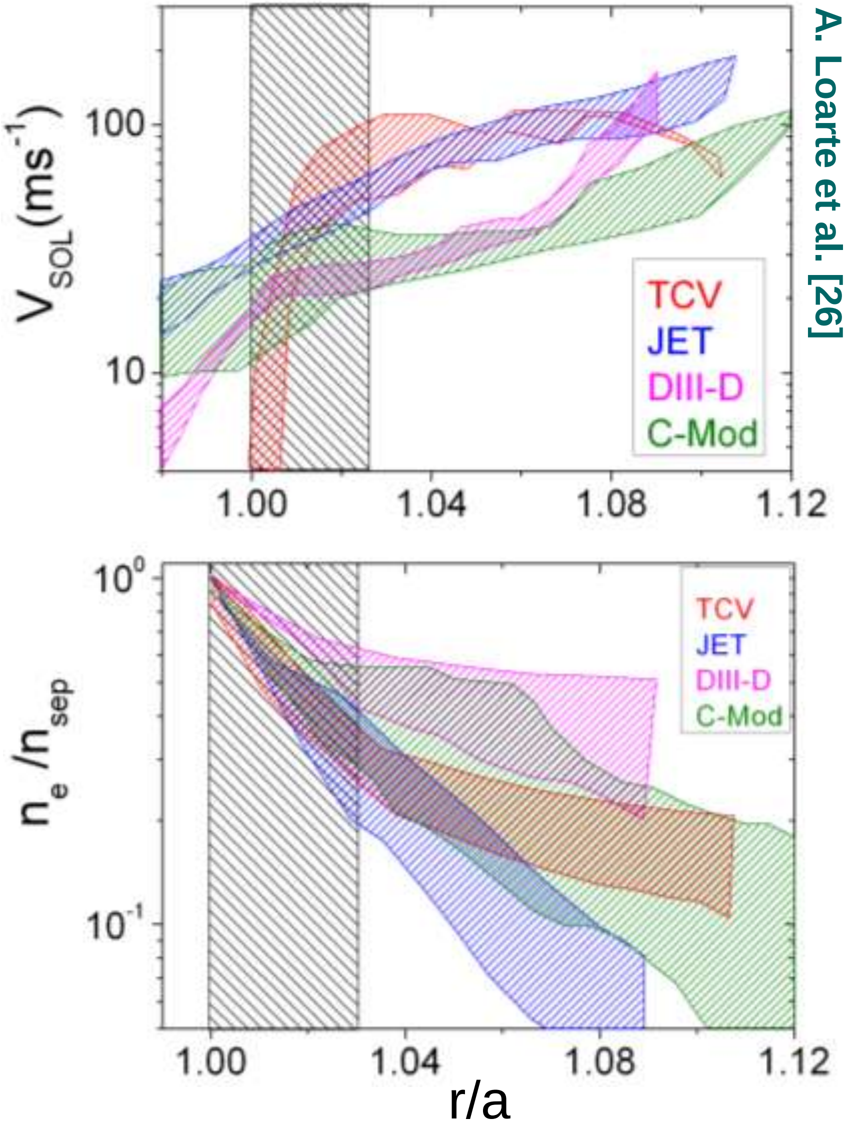


Particle density field



- ESEL code (Risø [23]) simulates 2D region centred on outboard midplane
- Exhaustively tested against TCV high density case [22, 24-25]

Bursty transport now in ITER baseline!



A. Loarte et al. [26]

Stationary, inter-ELM power fluxes to ITER main wall now assumed to be dominated by convective, intermittent transport

Multi-machine study shows far SOL cross-field convective velocity weakly dependent on device size

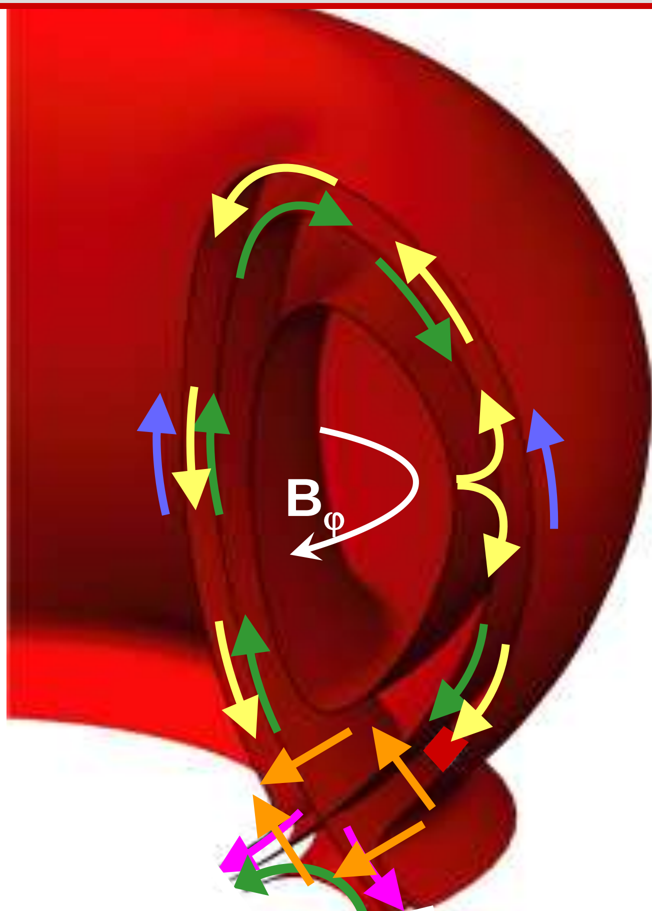
$\lambda_{SOL} = Lc_s/v_{SOL}$ = 4 – 17 cm, L~60 m for ITER

Power to ITER first wall <20 MW (20% P_{SOL})
Part. flux to ITER first wall < $1\times10^{24}s^{-1}$ (10% Γ_{div})

Region of connected SOL from inner to outer strike points

Parallel SOL ion flows

Determine transport of impurities from source to destination in a tokamak – material migration – T-retention (Slides 41-46)



$E_r xB$, ∇pxB

$E_\theta xB$

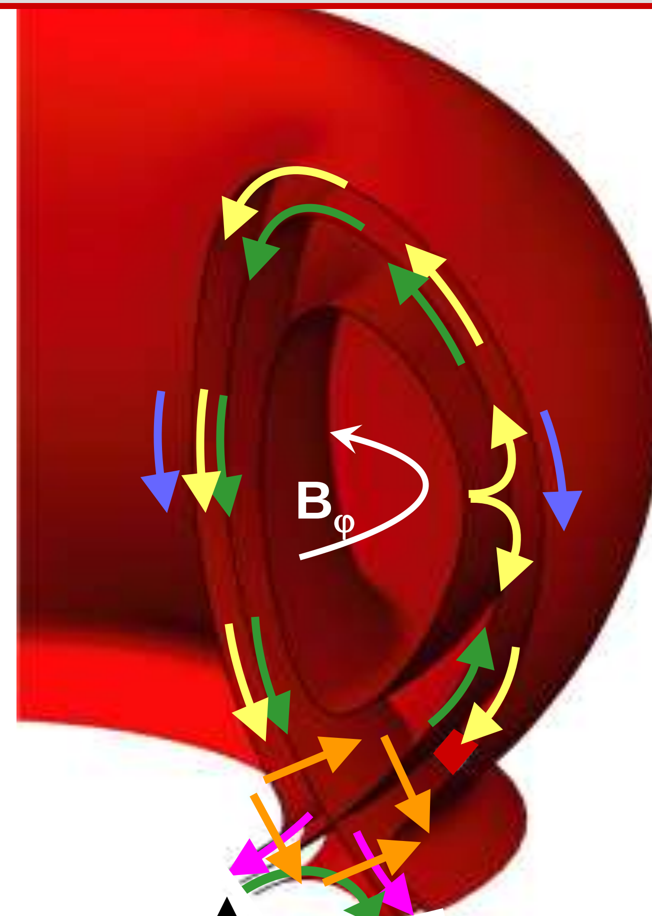
Poloidal

Pfirsch-Schlüter

Divertor sink

$\perp$ transport driven

Parallel

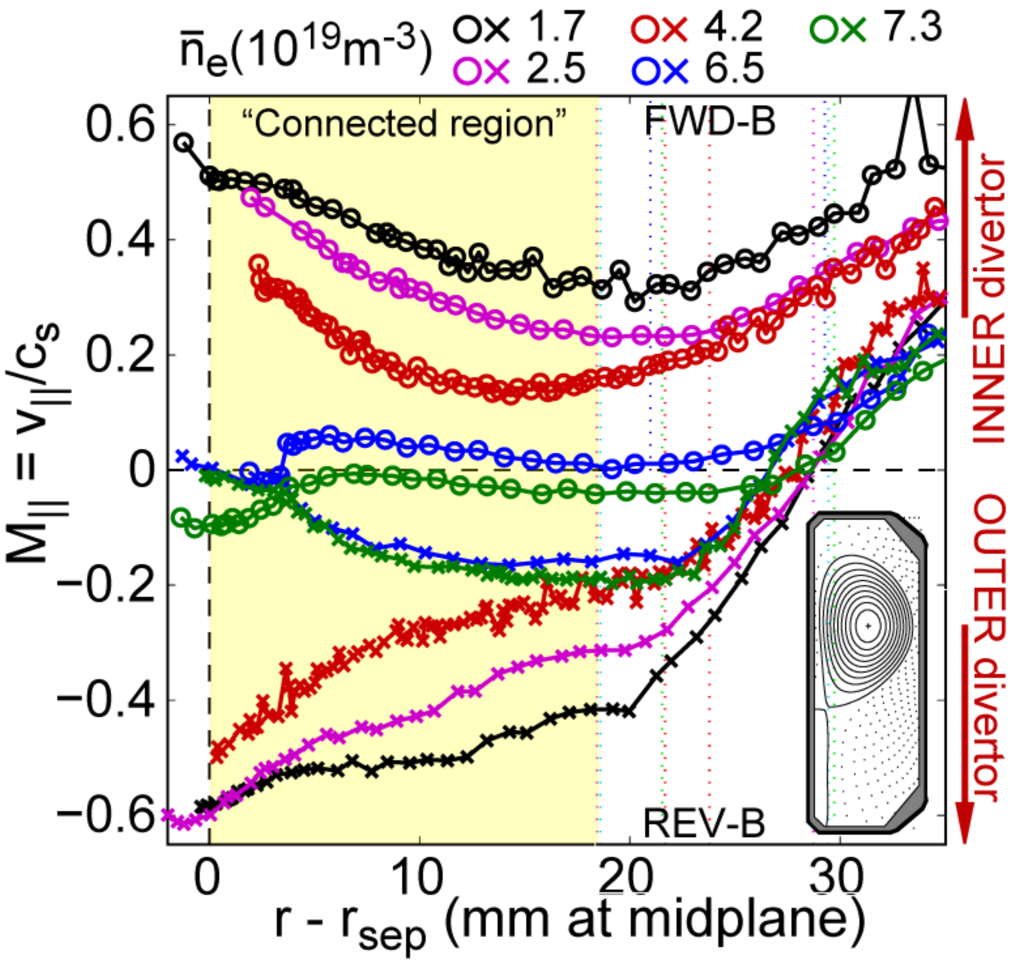


FWD-B_φ **$Bx\nabla B \downarrow$**

Field direction dependent

$Bx\nabla B \uparrow$ **REV-B_φ**

Flows can be very strong

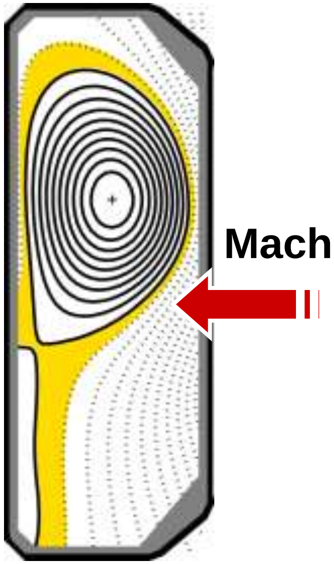


$M_{||} = 0.5 \rightarrow v_{||} \sim 30\ kms^{-1}$!

Have been measured on several tokamaks – TCV is a good example [27,28]

Main parallel flows are field direction dependent, density dependent and in the same direction as the plasma current

→ Consistent with Pfirsch-Schlüter (neoclassical) flow



Mach probe

Part II

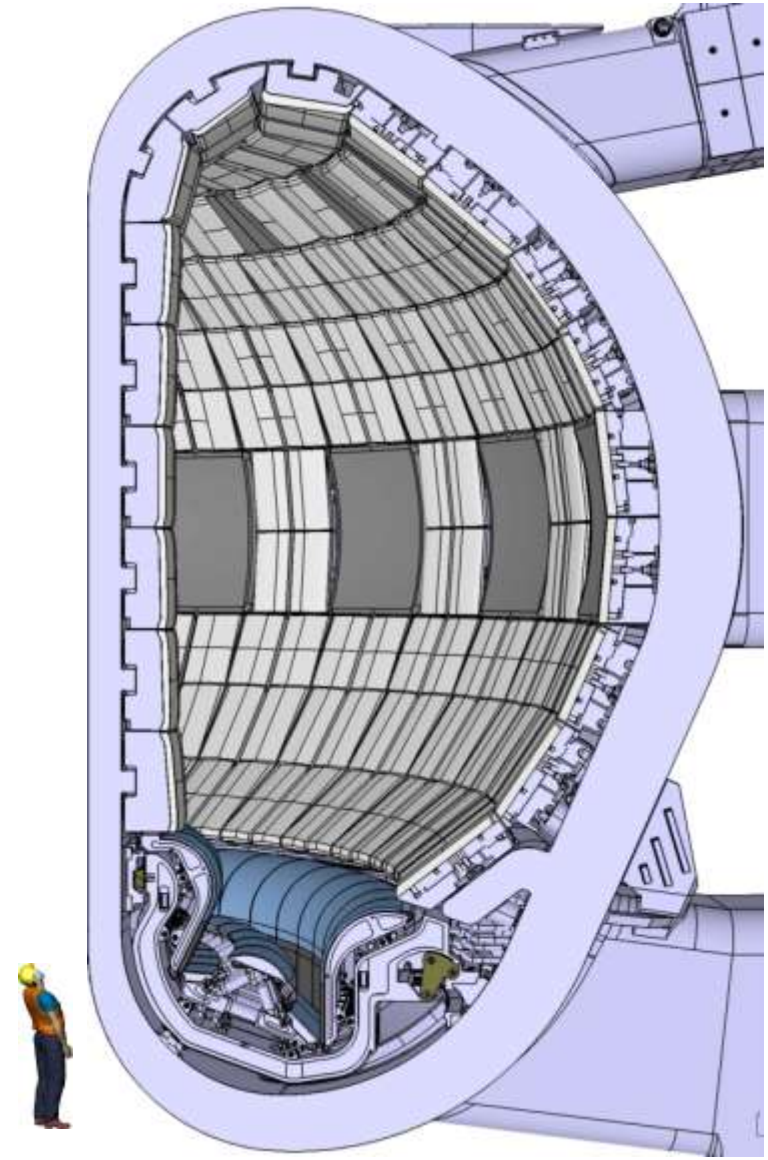
Plasma-surface interactions

The plasma-wall interaction challenge

Perhaps the most critical area for ITER and for the long term feasibility of fusion energy production

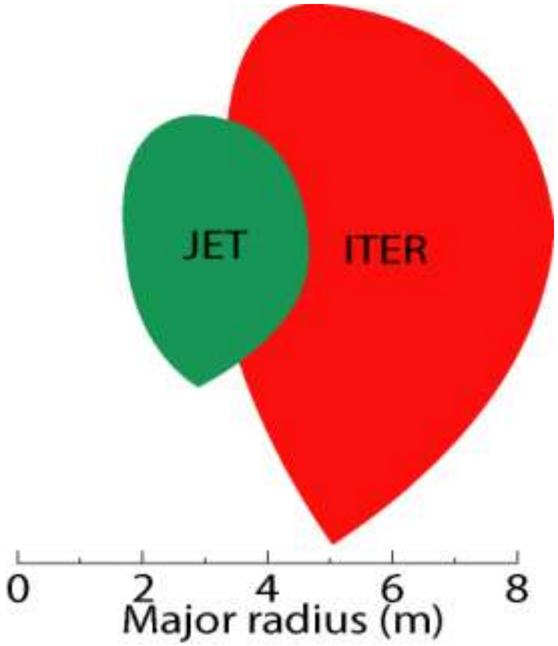
Must guarantee:

- Management of stationary heat flux densities at the limit of cooling technology
- High throughput fuel cycle (low burn-up fractions) with low Tritium retention
- Material erosion rates compatible with adequate lifetime and burning plasma purity



Upscale to ITER is a very big step

Comparison with JET (World's largest operating tokamak) for illustration



Parameter	JET MkIIGB (1999-2001)	ITER
Integral time in diverted phase	14 hours	0.1 hours
Number of pulses	5748	1
Energy Input	220 GJ	60 GJ
Average power	4.5 MW	150 MW
Divertor ion fluence	1.8×10^{27}	$*6 \times 10^{27}$

*Code calculation

Extracted from Matthews et al. [29]

1 ITER pulse ~ 0.5 JET years energy input

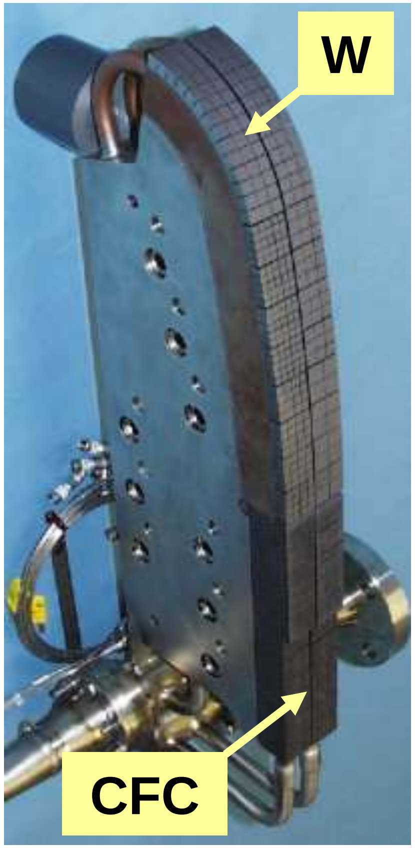
1 ITER pulse ~ 6 JET years divertor fluence

- Stored energy ~ $\propto R^5$ → ~35× higher on ITER than JET [30]
- But deposition area for power in the divertor $\propto R\lambda_p$ → $\lambda_{p,ITER} \sim \lambda_{p,JET}$ → ~2.5 m^2 ITER cf. ~1.0 m^2 JET → ITER will project ~35x the energy into only ~3x the area

ITER materials choices

Surface areas:
Be: ~700 m^2, W: ~120 m^2
CFC: ~35m^2

- Be for the first wall
  - Low T-retention
  - Low Z
  - Good oxygen getter



- For low-active phase: C for the targets
  - Low Z
  - Does not melt
  - Excellent radiator
- W for the dome/baffles
  - High Y_{phys} threshold

Driven by the need for operational flexibility

- For D and DT phases:
  - Be wall, all-W divertor

To avoid problem of T-retention

Critical PWI issues

Long term tritium retention

Material lifetime

Dust production

Short and long range material migration

Material mixing

Steady state erosion

Transient erosion (ELMs, disruptions)

Redeposition

All strongly interlinked

T-retention and dust production are safety critical items and form part of the ITER Nuclear Licensing process

Impurity migration

Migration = Erosion ⇒ Transport ⇒ Deposition

Re-erosion

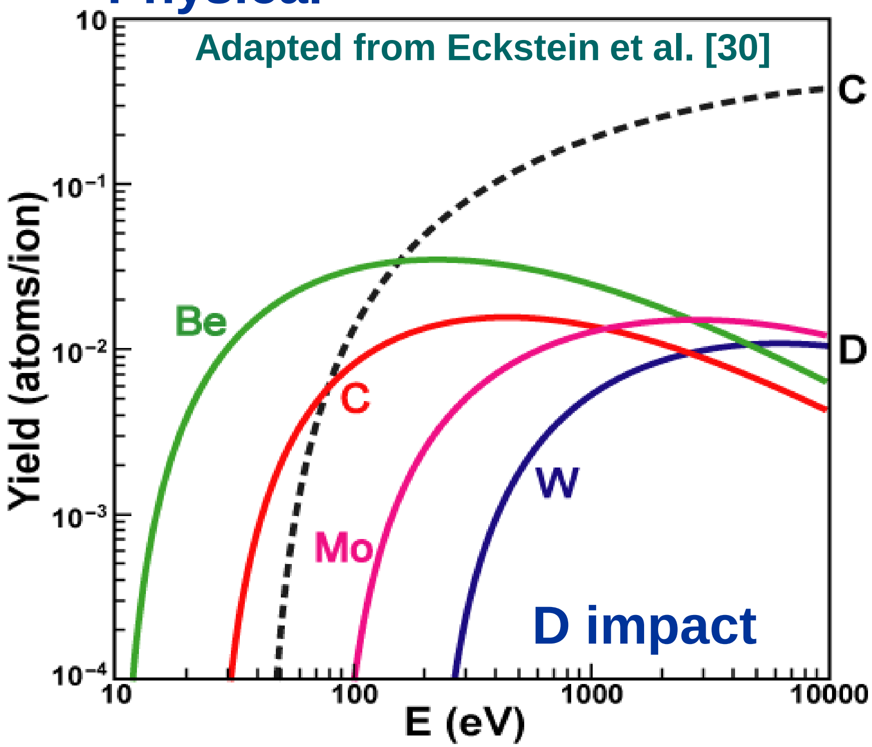
Impurity migration

Migration = Erosion ⇨ Transport ⇨ Deposition

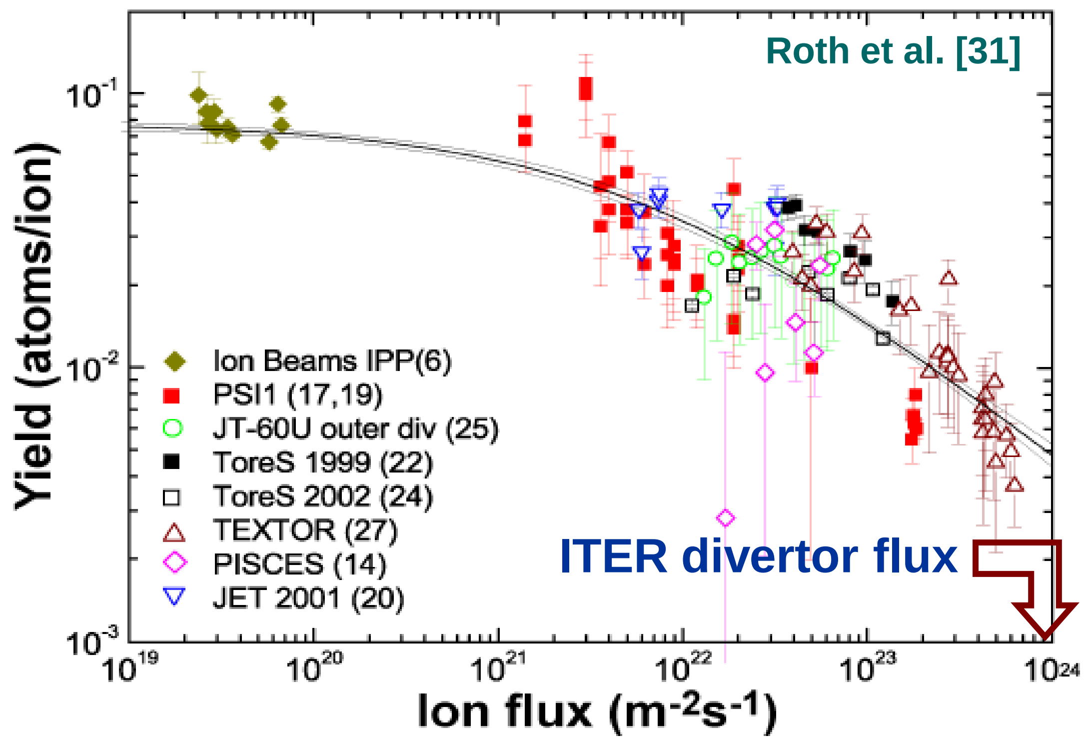
Re-erosion

Steady state erosion: sputtering

Physical



Chemical (carbon)



- Energy threshold → higher for higher Z substrate
- Much higher yields for high Z projectiles – important if using impurity seed gases

- No threshold → dependent on bombarding energy, flux and surface temperature

Steady state divertor target erosion rates (ERO modelling) due to Y_{phys} and Y_{chem} estimated at ~0.4 - 2 nms^{-1} for ITER [32]

Transient erosion

Transients are the biggest threat for large scale erosion in ITER. Burning plasma stored energy (~350 MJ) >> in the largest operating devices but surface areas for energy deposition only factor ~3x larger (slide 29)

ELMs

"Natural" ELMs expected to expel ~6% of W_{plasma} at 1-2 Hz [18] → peak energy densities on ITER divertor of 5-10 MJm^{-2} on timescales of 250-500 μs

Disruptions/VDEs

Worst case full energy disruptions → peak energy densities on the divertor of 5.0-20 MJm^{-2} on timescales of 1.5-3 ms (thermal quench)

Current quench: runaway electron currents up to 12 MA with 10-20 MeV in localised areas on timescales of a few tens or hundreds of ms → energy densities of 35-70 MJm^{-2}

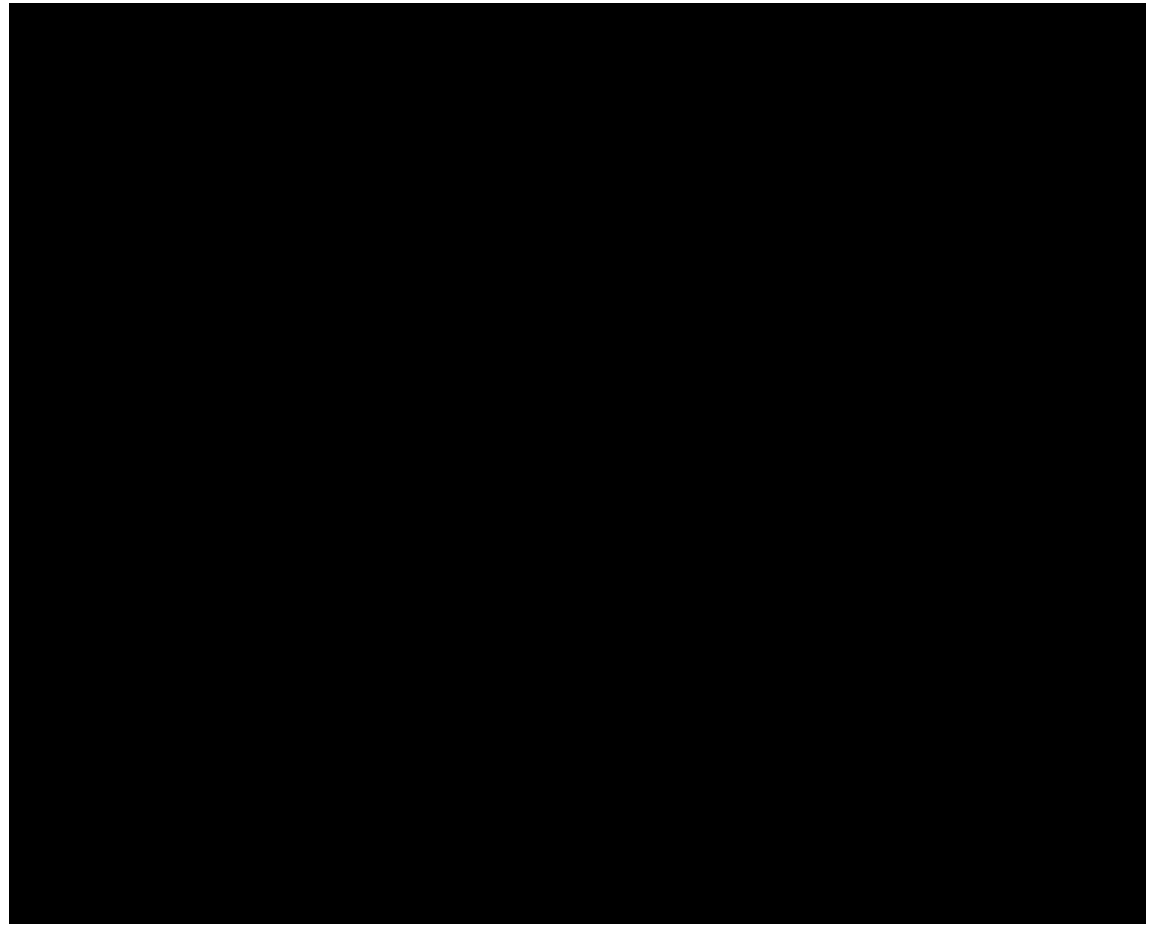
Runaway electrons on Tore Supra

Courtesy J. Bucalossi, CEA Cadarache

Deliberate creation of runaway electron plateau by injection of Neon impurity during I_p ramp up.

Thermal quench occurs between -9.552 < t < 0.948 ms in the video

Subsequent attempts to control the runaway beam using tokamak PF control circuits → one possible solution being sought for ITER



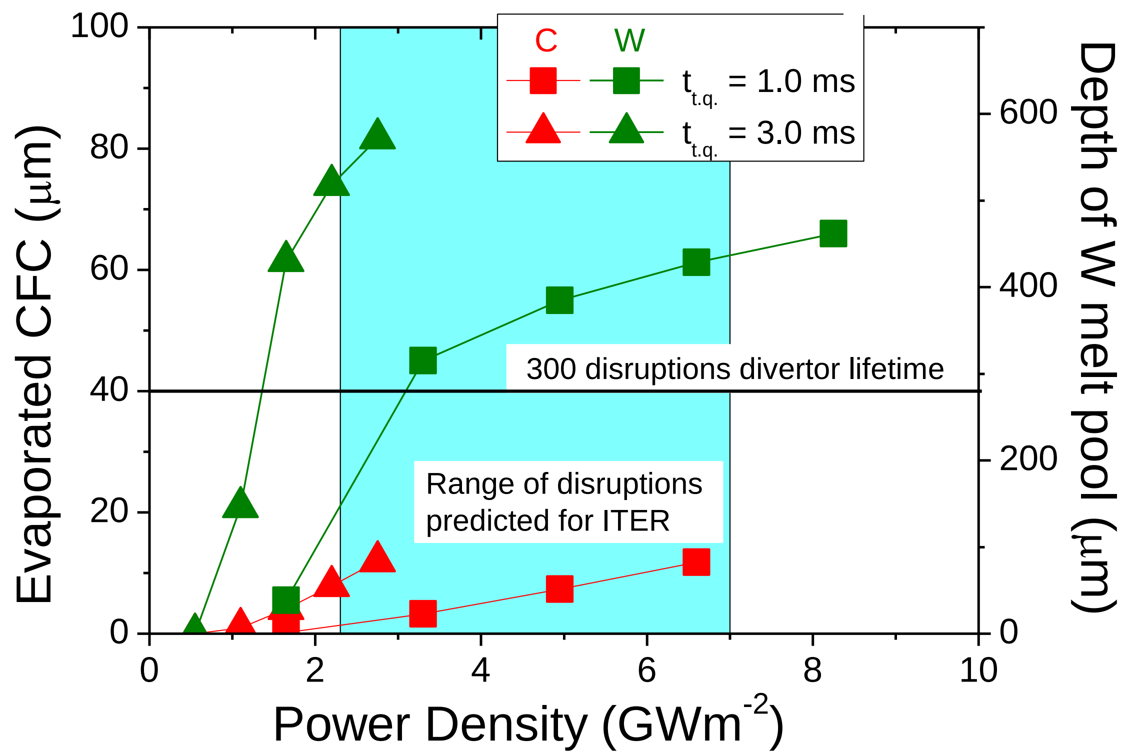
Disruption induced erosion

Vapour shielding reduces CFC erosion

Loss of melt layer on W occurs if layer deep enough and force (evaporated layer plasma pressure, eddy currents) sufficient to trigger liquid instabilities (Kelvin-Helmholtz, Rayleigh-Taylor) → droplet ejection

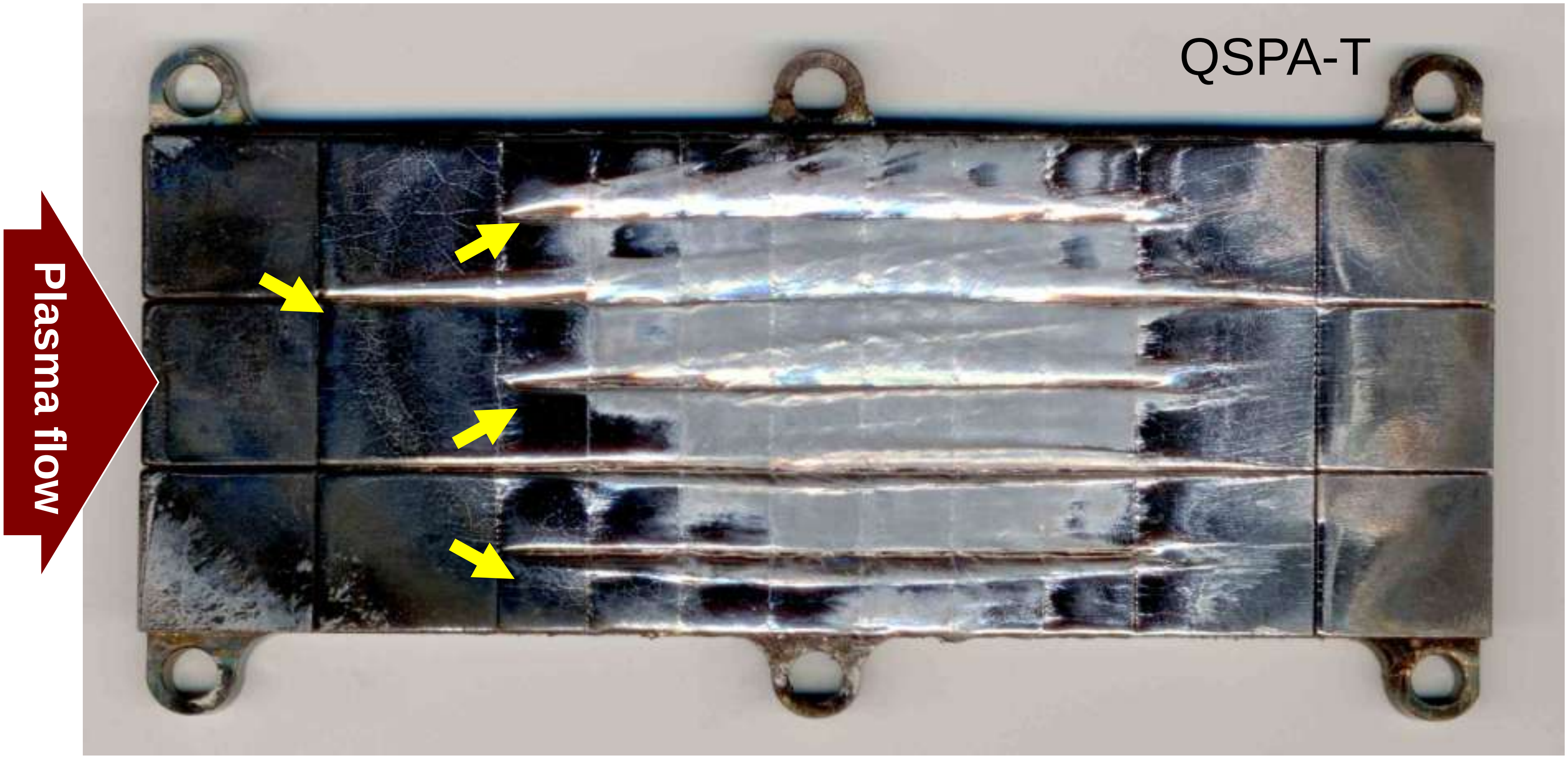
A. Loarte et al. [33]

G. Federici, RACLETTE code [34]



Assuming 10% melt-layer loss, W divertor lifetime (0.3 cm PFC end of life thickness) exceeded in ~300 disruptions → efficient disruption avoidance or mitigation techniques required in ITER → major research priority for the ITER team

e.g. W exposed to 100 pulses of 1.5 MJm^{-2}



A. Zhitlukhin et al., [35]

- Significant melting, bridging of castellation gaps, lifetime reduction
- Issues of operability on damaged targets, dust production

ELM induced erosion

Real material limits are much lower
→ Results from Russian plasma simulators [36]:

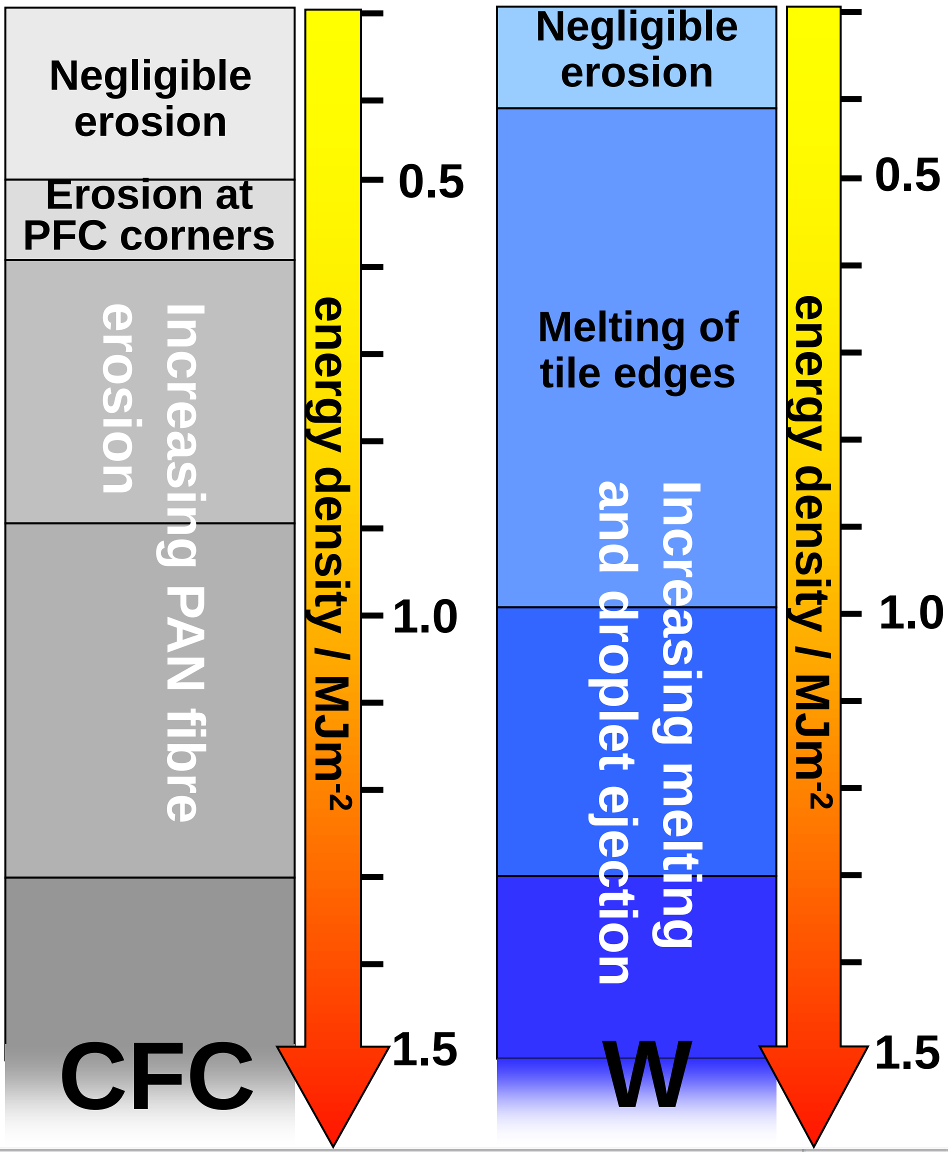
Erosion limit for CFC reached due to PAN fibre erosion (fibres parallel to surface)

Erosion limit for W reached due to melting of tile edges

Crack formation on W observed at energy densities ≥ 0.7 MJm^{-2}

⇩

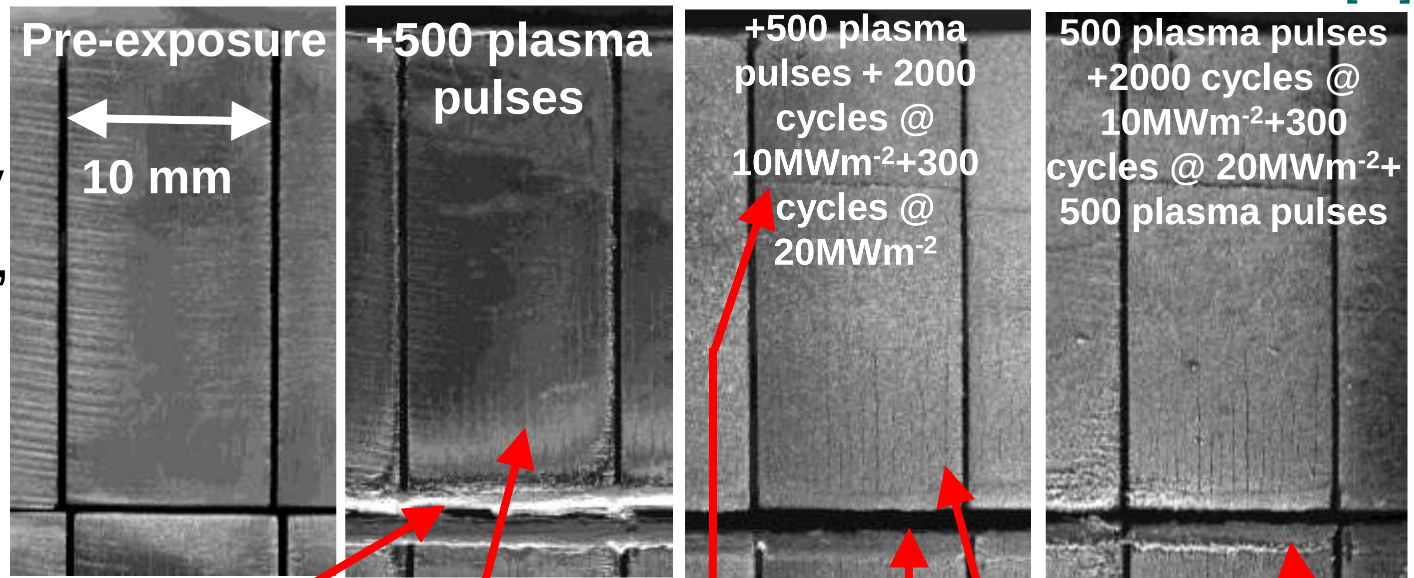
Recommended damage threshold ~0.5 MJm^{-2} now adopted by ITER
→ Will require ELM mitigation strategies to keep E_{ELM} < 1 MJ



Could even be worse for W ….

B. Riccardi et al. [37]

Crack formation and melting under ITER-like pulsed and steady loads on water cooled, W monoblock targets
Pulsed loads
≤0.5 MJm^{-2} → controlled ITER ELMs



Edge melting, bridge formation

Transversal cracks

Longitudinal cracks

Transversal cracks widening

Melted areas peeled off

Edge melting, restarts

Previous observations of surface crack saturation might be optimistic

Repetitive edge melting and crack propagation a worry for power handling and dust formation

Monoblock design issues → insufficient confidence yet to build all-W divertor for ITER

Impurity migration

Migration = Erosion → Transport → Deposition

Re-erosion

Transport creates and moves impurities

Ions:

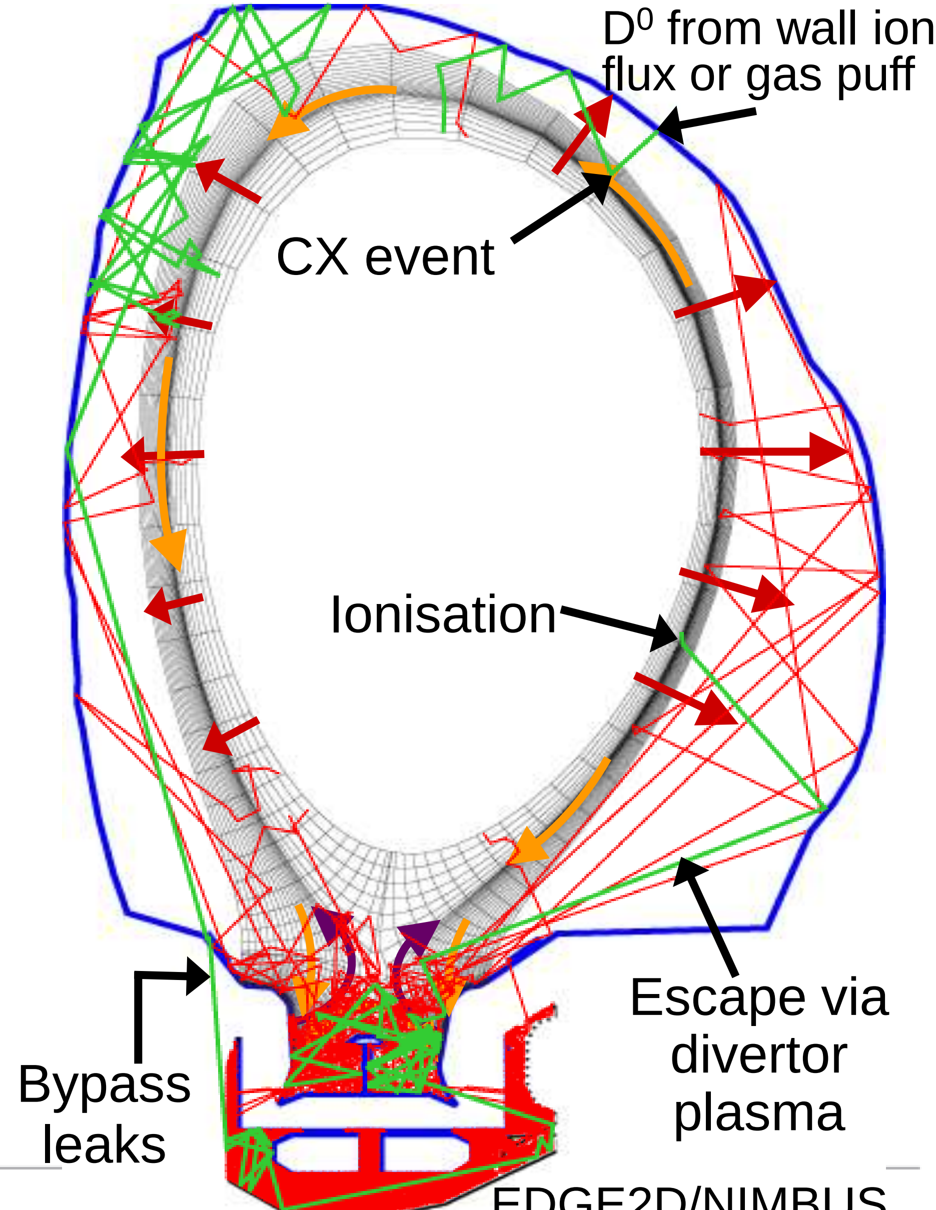
Cross-field transport – turbulent driven far SOL ion fluxes (slides 19-23)
→ recycled neutrals
→ direct impurity release
ELMs can also reach first walls

Eroded Impurity ions "leak" out of the divertor (∇T_i forces)

SOL and divertor ion fluid flows can entrain impurities (slides 24-25)

Neutrals:

- From divertor plasma leakage, gas puffs, bypass leaks → low energy CX fluxes → wall sputtering
- Lower fluxes of energetic D^0 from deeper in the core plasma



Impurity migration

Migration = Erosion ⇨ Transport ⇨ Deposition

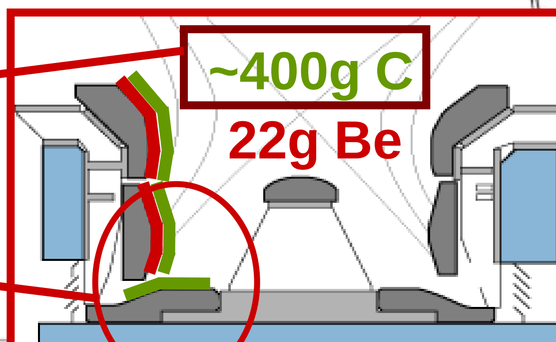
Re-erosion

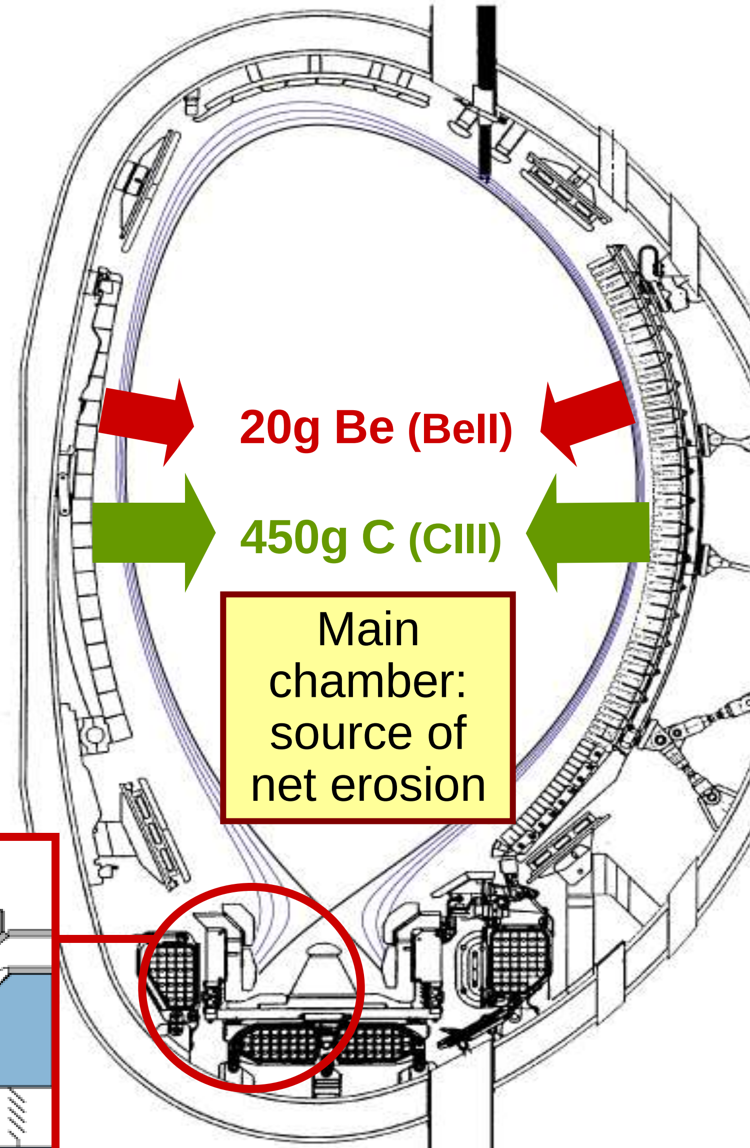
Migration balance – example from JET

- Make balance for period 1999-2001 with MarkIIGB divertor: 14 hours plasma in diverted phase (50400 s, 5748 shots)
- Use spectroscopy and modelling to estimate main chamber sources

- Post mortem surface analysis
  - Deposition almost all at inner divertor
  - Surface layers are Be rich → C chemically eroded and migrates, Be doesn't move
  - Outer divertor – region of net erosion or balanced erosion/redeposition – BUT mostly attached conditions (not like ITER)

~250 kg/year if JET operated full time!
C migrates to remote areas forming D-rich soft layers (high T-retention)





R. A.Pitts et al. [4]
G. F. Matthews et al. [38]
J. D. Strachan et al. [39]
J. Likonen et al. [40]

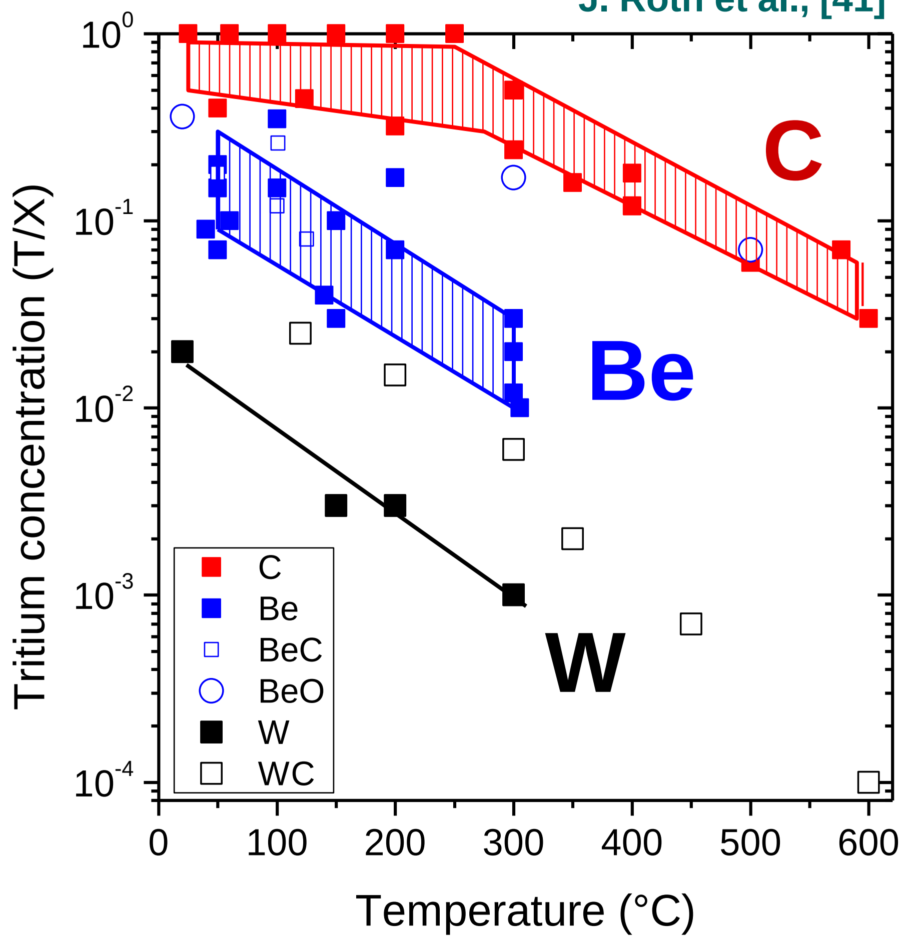
Tritium retention (1)

- A 400 s Q_{DT} = 10 ITER discharge will require ~50 g of T fuelling (cf. 0.01-0.2 g in today's tokamaks)
- Maximum in-vessel mobilisable T in ITER limited to 1kg [41]
  - This is a safety issue
- In practice, administrative limit of ~700 g
  - 120 g in cryopumps
  - 180 g uncertainty
- Predicting the expected retention in ITER is fraught with uncertainty but progress is being made

Tritium retention (2)

- For C, complex interplay between erosion → hydrocarbons →dissociation/ionisation → transport → re-deposition → migration to remote areas with high sticking coefficients and retention in co-deposits
  - Carbon traps D, T very efficiently
  - D/C ratio can be in the range ~0.4 → > 1
- For Be, co-deposition of T also possible - large potential source of Be from first wall
- For W, most of retention will be from implantation → not thought to constitute a large reservoir
- BUT effects of increased trapping due to neutron irradiation of metals – does not look like an issue from recent results

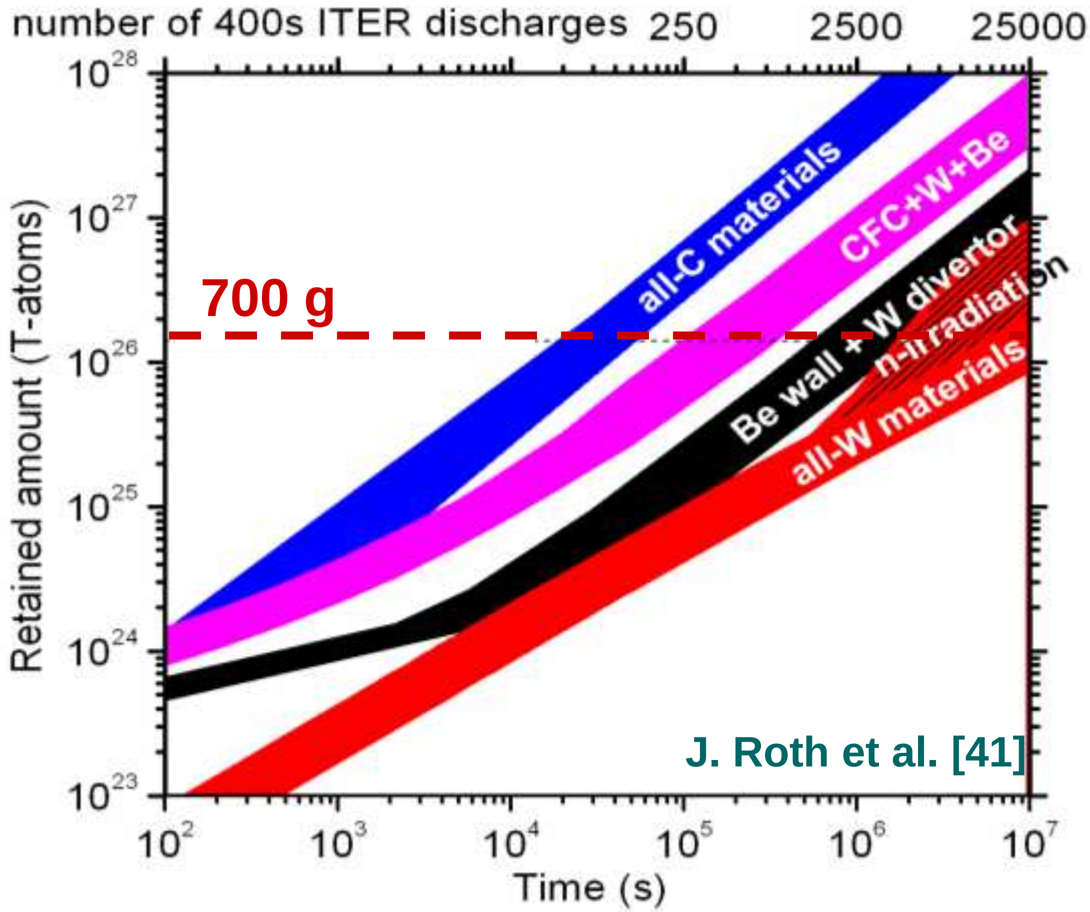
J. Roth et al., [41]



Co-deposition with C and Be depends sensitively on deposition rate, incoming particle energy, surface temperature [42]

Tritium retention (3)

- EU-PWI Task Force and ITPA DIVSOL group have recently tried to estimate ITER T-retention
- Assume erosion determines co-deposition:
  - T-retention = erosion rate x total co-deposition concentration
- Add T implantation in W
- Compare materials options

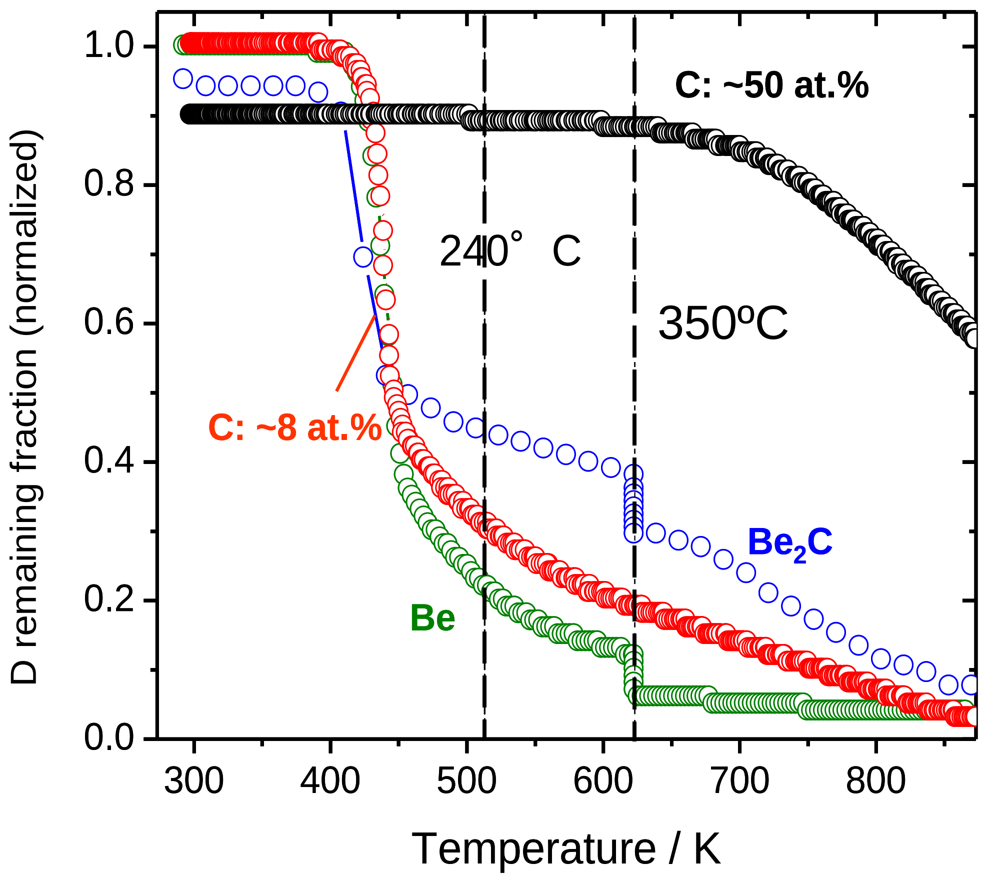


Main driver of current ITER baseline strategy to begin D-T operations with full W divertor – only ~few 100 full performance DT shots predicted before T-inventory limit exceeded if CFC divertor used in tritium phase

Tritium retention (4)

- Even though Be can readily trap tritium, fuel is released from Be co-deposits at much lower temperature than for C
  - The main ITER fuel recovery strategy → divertor bakeout capability to 350ºC is part of the Baseline design → most of the Be co-deposits expected in the divertor
  - NB: main wall bakeable only to 240ºC

K. Sugiyama et al., [43]



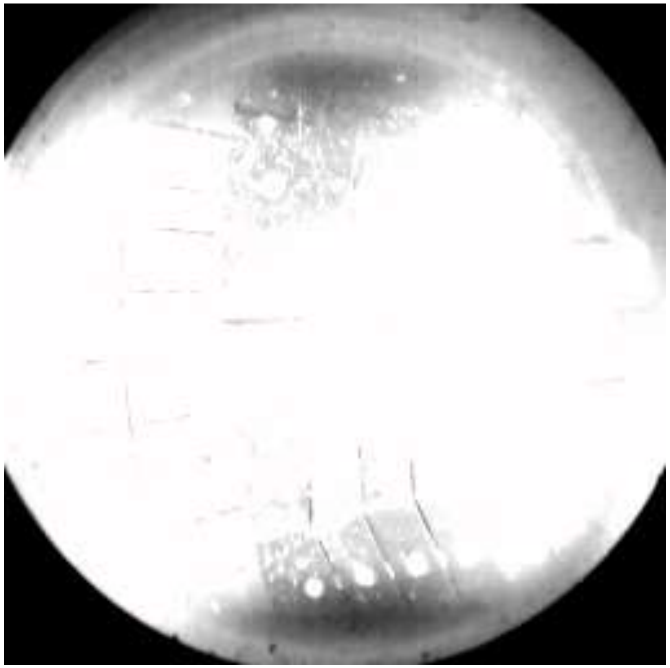
Dust – why worry?

- Expectation is that increase in duty cycle and erosion in ITER will lead to large scale-up in quantity of dust particles produced
- Like T-retention, dust is a safety issue [44,45]
  - dust particles radioactive (tritium + activated metals)
  - potentially toxic (Be)
  - potentially responsible for a large fraction of in-VV mobilisable tritium
  - chemically reactive with steam or air
- Radiological or toxic hazard depends on how well dust is contained in accident scenarios and whether it is small enough to remain airborne and be respirable
  - size needs to be <~ 100 μm
  - depends on how dust is produced, e.g. crumbling of co-deposited layers or destruction (thermal overload) of tritiated layers during off-normal events
  - tritiated dust can levitate in electric fields as a result of self-charging due to emission of beta electrons

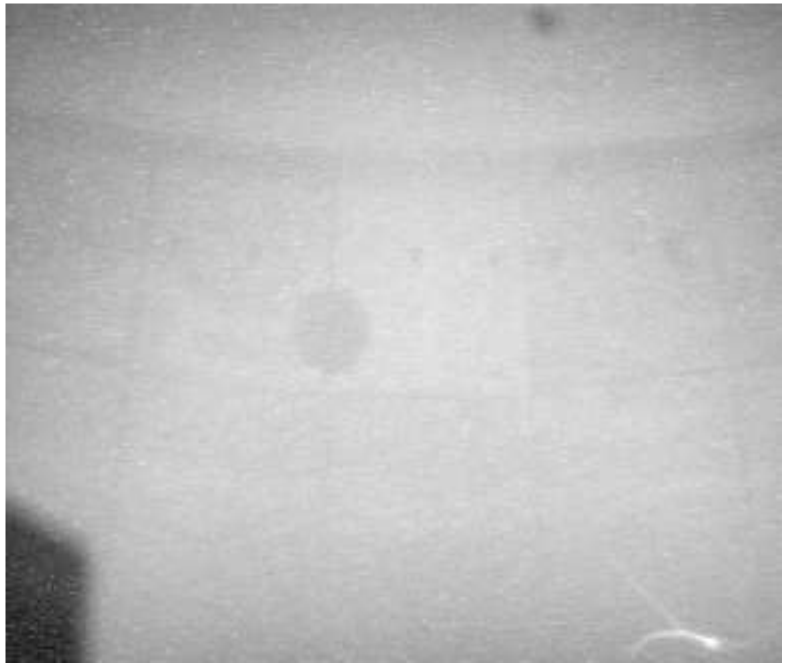
Dust – seen in all tokamaks

- Dust is seen in all tokamaks, especially with C walls, but most often in first plasmas after long vent, or after disruptions when plasma touches surfaces not normally in contact with high heat/particle flux – not usually an operational issue
- First papers to recognize the potential importance more than 10 years ago [46]

TCV: floor viewing IR camera during disruption, #33448 (J. Marki, R. A. Pitts)

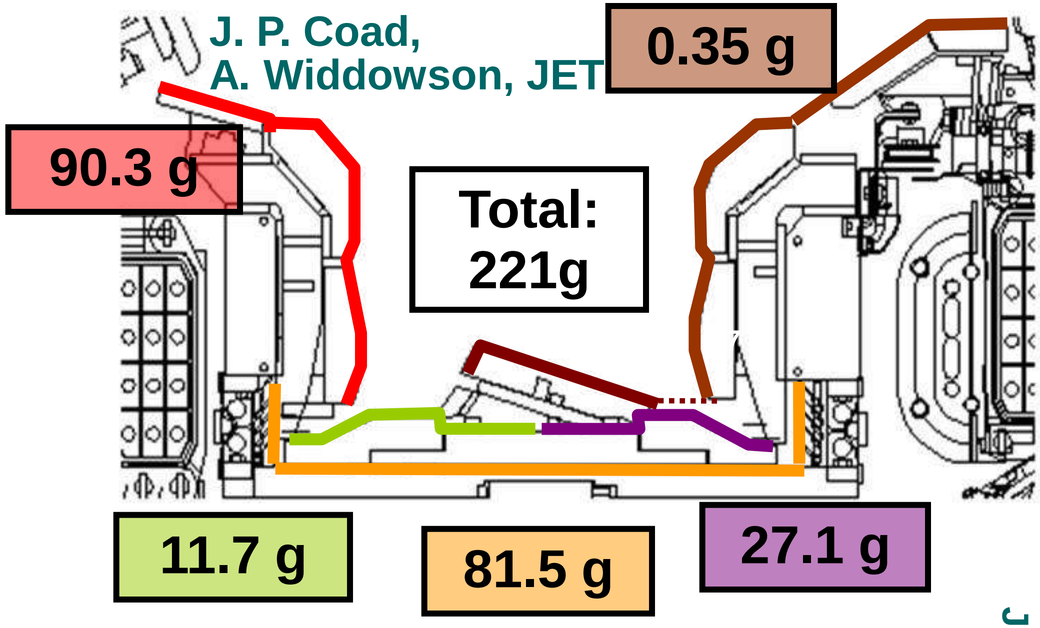


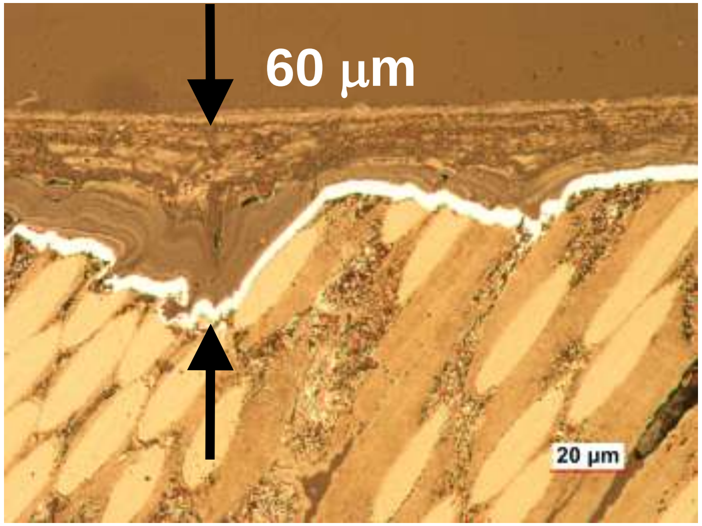
DIII-D: floor viewing DiMES TV with near IR filter. 2nd shot in 2007 after “dirty vent”, #127331. Courtesy of D. L. Rudakov [47]



Significant quantities collected in JET

- Very recent dust collection from JET after ~6 years → dominated by C but Be rich due to Be wall evaporation
  - NB: 1 ITER pulse ~ 6 years JET operation in terms of divertor fluence (based on 1999-2001 campaigns)
  - We need to better understand dust formation, transport, fuel retention and decide how to measure it
  - In a Be/W environment, is dust formation all due to disintegration of deposited layers?
  - When does a layer of given thickness detach? What drives this process?





J. P. Coad, et al. [48]

ITER Dust – safety inventory limits

- Global quantity in the vacuum vessel (VV) – 1 tonne during D-D and D-T operation
- On hot surfaces (corresponds to amount of dust that could produce up to 2.5 kg of H_2 during accident in case of full reaction with steam – requires air ingress):
  - $Be + H_2O \rightarrow BeO + H_2$, $C + H_2O \rightarrow CO + H_2$, $W + 3H_2O \rightarrow WO_3 + 3H_2$
  - Complete reaction: $T_{surf} > 400°C$
  - 6 kg of C, 6 kg of W and 6 kg of Be (for CFC/W/Be mix)
  - up to 11 kg of Be (for Be alone) and up to 77 kg W (for W alone)
- When this dust inventory limit is reached (or if T-inventory reaches 1 kg) ITER operation must be stopped and dust removed
- The real dust inventory will be reduced by measurement uncertainties (estimated to be about 30%)

References (1)

A 1.5 hour lecture can only hope to scratch the surface of such a vast field. Some good reference sources covering aspects of the material shown in this presentation are the following:

[1] "The plasma boundary of magnetic fusion devices", P. C. Stangeby, IoP Publishing Ltd, Bristol, 2000

[2] "Experimental divertor physics", C. S. Pitcher and P. C. Stangeby, Plasma Phys. Control. Fusion **39** (1997) 779

[3] "Plasma-material interactions in current tokamaks and their implications for next step fusion reactors", G. Federici et al., Nucl. Fusion **39** (1997) 79

[4] "Material erosion and migration in tokamaks", R. A. Pitts et al., Plasma Phys. Control. Fusion **47** (205) B303

[5] "The plasma-wall interaction region: a key low temperature plasma for controlled fusion", G. F. Counsell, Plasma Sources Sci. Technol. **11** (2002) A80

[6] "Recent advances on hydrogen retention in ITER's plasma-facing materials: beryllium, carbon and tungsten", C. H. Skinner et al., Fus. Sci. Tech. **54** (2008) 891

A number of additional papers have been used to prepare the slides in this presentation. They are listed below in order of appearance in the talk.

[7] "Boundary plasma energy transport in JET ELMy H-modes", W. Fundamenski and W. Sailer, Nucl. Fusion **44** (2003) 20

[8] "Modelling of ELM-averaged power exhaust on JET using the EDGE2D code with variable transport coefficients", G. Kirnev et al., Plasma Phys. Control. Fusion **49** (2007) 689

[9] "Scaling laws for edge plasma parameters in ITER from two-dimensional edge modelling", A. Kukushkin et al., Nucl. Fusion **43** (2003) 716

[10] "Plasma-surface interaction, scrape-off layer and divertor physics: implications for ITER", B. Lipschultz et al., Nucl. Fusion **47** (2007) 1189

[11] "Scaling of the power exhaust channel in Alcator C-Mod", B. LaBombard et al., 52nd APS Conference, Chicago, USA, November 8-12 2010: to be published in Phys. Plasmas (2011)

[12] "Scaling of divertor heat flux profile widths in DIII-D", C. Lasnier et al., 19th PSI Conference, San Diego, USA, May 24-28, 2010: to be published in J. Nucl. Mater (2011)

References (2)

[13] "Dependences of the divertor and midplane heat flux widths in NSTX", T. K. Gray et al., 23rd IAEA Fusion Energy Conference, Daejeon,South Korea, October 11-16 2010, Paper EXD/P3-13

[14] "Experimental investigation of transport phenomena in the scrape-off layer and divertor", B. LaBombard et al., J. Nucl. Mater. **241-243** (1997) 149

[15] "Improved radiation measurements on JET – first results from an upgraded bolometer system", A. Huber et al., J. Nucl. Mater. **363-365** (2007) 365

[16] "Recent results from divertor and scrape-off layer studies on JET", R. D. Monk et al., Nucl. Fusion **39** (1999) 1751

[17] "Scrape-off layer radiation and heat load to the ASDEX Upgrade LYRA divertor", A. Kallenbach et al., Nucl. Fusion **39** (1999) 901

[18] "Study of target plate heat load in diverted DIII-D tokamak discharges", C. Lasnier et al., Nucl. Fusion **38** (1998) 1225

[19] "Studies in JET divertors of varied geometry: II Impurity seeded plasmas", G. F. Matthews et al., Nucl. Fusion **39** (1999) 19

[20] "Status and physics basis of the ITER divertor", R. A. Pitts et al., Phys. Scr. **T138** (2009) 014001

[21] "ITER Physics basis: Chapter 4, power and particle control", Nucl. Fusion **39** (1999) 2391

[22] "Fluctuations and transport in the TCV scrape-off layer", O. E. Garcia et al., Nucl. Fusion **47** (2007) 667

[23] "Computations of intermittent transport in SOL plasmas", O. E. Garcia et al, Phys. Rev. Lett **92** (2004) 165003

[24] "Interchange turbulence in the TCV SOL", O. E. Garcia et al., Plasma Phys Control. Fusion **48** (2006) L1

[25] "Collisionality dependent transport in TCV SOL plasmas", O. E. Garcia et al., Plasma Phys Control. Fusion **49** (2007) B47

[26] "Power and particle fluxes at the plasma edge of ITER", A. Loarte et al., 22nd IAEA Fusion Energy Conference, Geneva (2008) paper IT/P6-13

[27] "Parallel SOL flow in TCV", R. A. Pitts et al., J. Nucl. Mater. **363-365** (2007) 738

[28] "Neoclassical and transport driven parallel flows on TCV", R. A. Pitts et al., 34th EPS Conference on Plasma Phys. Warsaw, 2 - 6 July 2007 ECA Vol.**31F**, O-4.007 (2007)

References (3)

[29] "Material migration in JET", G. F. Matthews et al., Proc. 30th EPS Conf. on Control. Fusion and Plasma Physics (St. Petersburg, 2003) **27A** (ECA) P-3.198

[30] "Steady state and transient power handling in JET", G. F. Matthews et al., Nucl. Fusion **43** (2003) 999

[30] "Sputtering data", W. Eckstein et al., IPP Garching report number 9/82 (1993)

[31] "Flux dependence of carbon chemical erosion by deuterium ions", J. Roth et al., Nucl. Fusion **44** (2004) L21

[32] "Modelling of tritium retention and target lifetime of the ITER divertor using the ERO code", A. Kirschner et al., J. Nucl. Mater. **363-365** (2007) 91

[33] "Transient energy fluxes in tokamaks : Physical processes and consequences for next step devices", A. Loarte et al., 34th EPS Conf. on Control. Fusion and Plasma Physics (Warsaw, 2007)

[34] "Assessment of the erosion of the ITER divertor targets during Type I ELMs", G. Federici et al., Plasma Phys. Control. Fusion **45** (2003) 1523

[35] "Effect of ELMs on ITER divertor armour materials", A. Zhitlukhin et al., J. Nucl. Mater. **363-365** (2007) 301

[36] "Experimental study of PFC erosion under ITER-like transient loads at plasma gun facility QSPA", N. Klimov et al., J. Nucl. Mater. **390-391** (2009) 721; "Experimental study of PFC erosion and eroded material deposition under ITER-like transient loads at plasma gun facility QSPA", N. Klimov et al., 19th PSI Conference, San Diego, USA, May 24-28, 2010: to be published in J. Nucl. Mater (2011)

[37] "Preliminary results of the experimental study of PFCs exposure to ELMs-like transient loads followed by high heat flux thermal fatigue", B. Riccardi et al.,26th Soft Conference, Porto, Portugal, 27 Sept. – 1 Oct., 2010: to be published in FUs. Engin. Design (2011)

[38] "Material migration in divertor tokamaks", G. F. Matthews et al., J. Nucl. Mater. **337-339** (2005) 1

[39] "JET carbon screening experiments using methane gas puffing and its relation to intrinsic carbon impurities", J. D. Strachan et al., Nucl. Fusion **43** (2003) 922

[40] "Beryllium accumulation at the inner divertor of JET", J. Likonen et al., J. Nucl. Mater. **337-339** (2005) 60

[41] "Recent analysis of key plasma-wall interaction issues for ITER", J. Roth et al., J. Nucl. Mater. **390-391** (2009) 1

References (4)

[42] "An empirical scaling for deuterium retention in co-deposited beryllium layers", G. De Temmerman et al., Nucl. Fusion **48** (2008) 075008

[43] "Consequences of deuterium retention and release from be-containing mixed materials for ITER tritium inventory control", K. Sugiyama et al., 19th PSI Conference, San Diego, USA, May 24-28, 2010: to be published in J. Nucl. Mater (2011)

[44] "Tritium inventory in ITER plasma-facing materials and tritium removal procedures", J. Roth et al., Plasma Phys. Control Fusion **50** (2008) 103001

[45] "The safety implications of tokamak dust size and surface area", K. A. McCarthy et al., Fus. Eng. Design **42** (1998) 45

[46] "Dust in fusion devices – experimental evidence, possible sources and consequences", J. Winter et al., Plasma Phys. Control. Fusion **40** (1998) 1201

[47] "Observations of Dust in DIII-D Divertor and SOL", D. L. Rudakov et al., 1st Workshop on "Dust in Fusion Plasmas", 8-10 July 2007, Warsaw, Poland

[48] "Erosion and deposition in the JET MkII-SRP divertor", J. P. Coad et al., J. Nucl. Mater. **363-365** (2007) 287