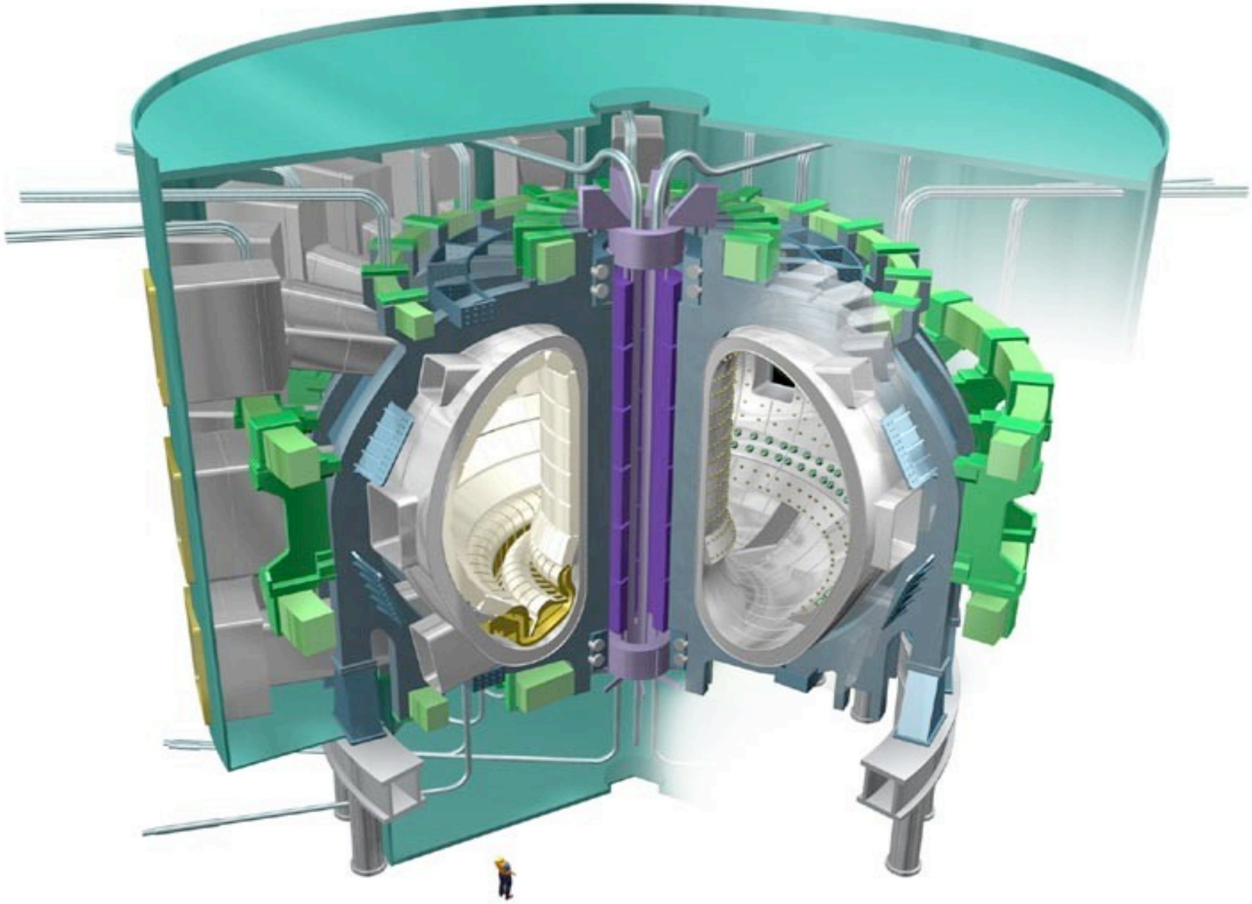
Key plasma-wall interaction physics issues for ITER



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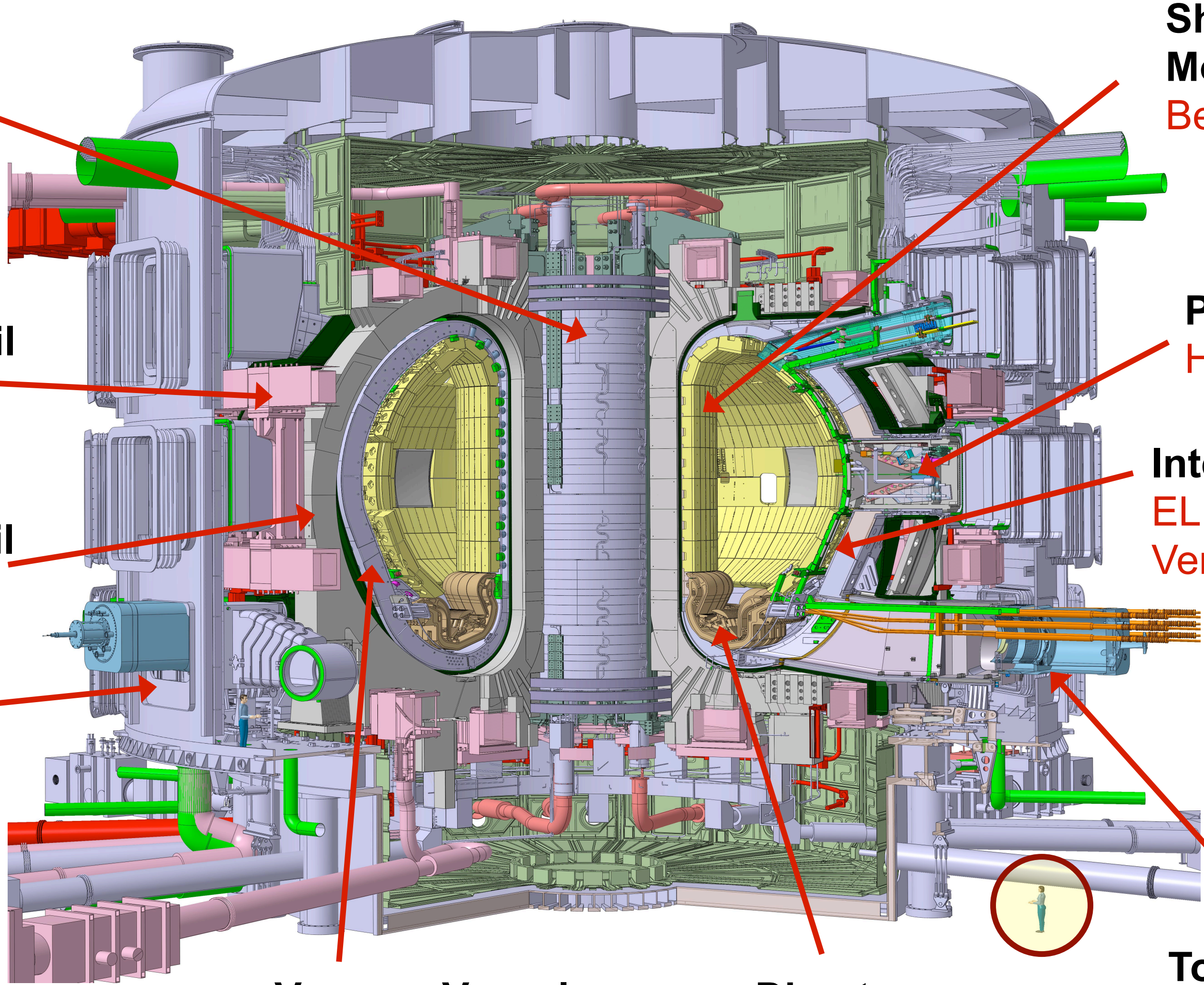
Importance of plasma-surface interaction

"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

Idea here is to introduce the most important physics concepts and describe the challenges → details to be developed in specialist tutorials throughout the week

The ITER Tokamak



Outline

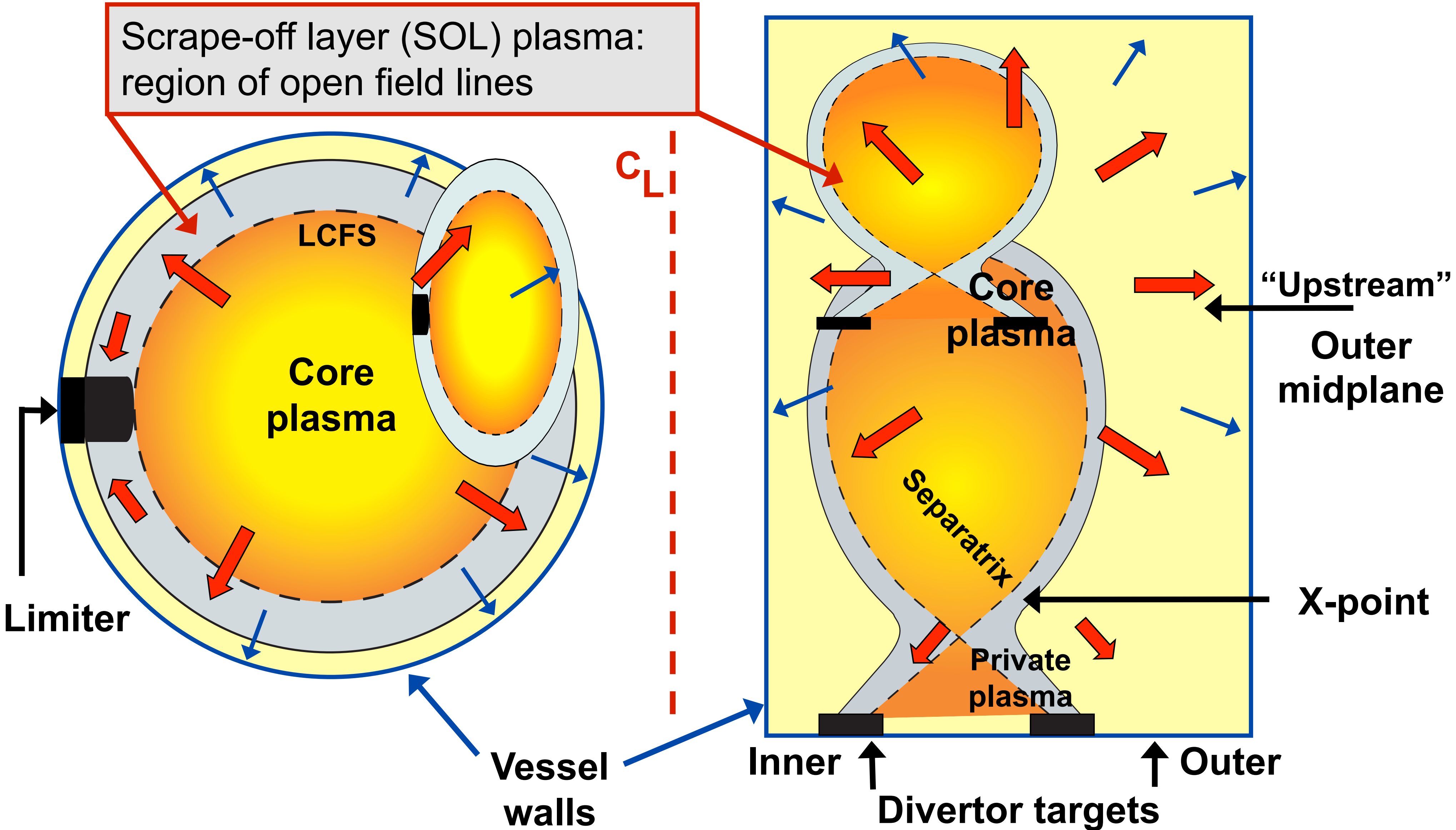
- Part I: The scrape-off layer (SOL) and divertor
  - SOL particle and power widths
  - Divertor detachment
  - Turbulent transport and SOL flows

- Part II: Plasma-surface interactions
  - Material lifetime – erosion and migration
  - Transients (ELMs and disruptions)
  - Tritium retention
  - Dust

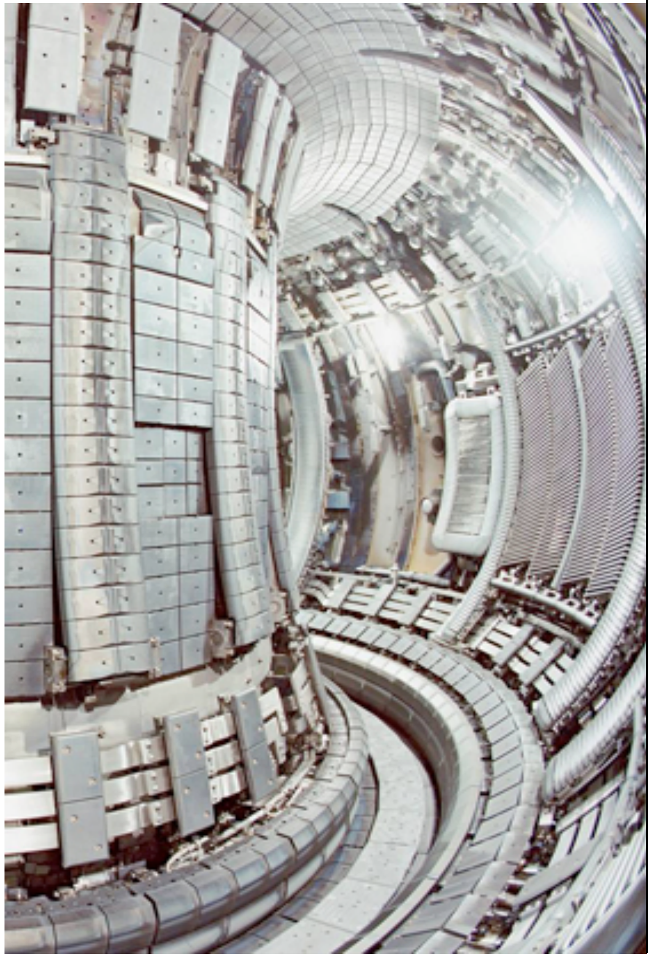
Part I

Divertor and SOL physics

Terminology: limiters and divertors



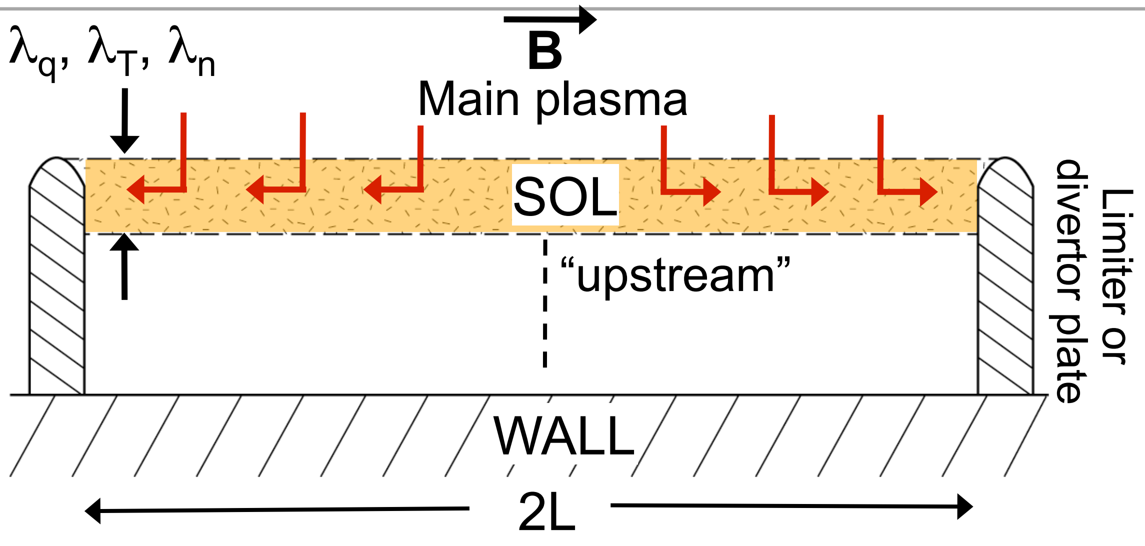
e.g. Limiter and divertor phases in many JET shots



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 → few secs. Full burn divertor phase of ~400 s for the Q_{DT} = 10 inductive scenario

Basics – SOL 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: $\Gamma_\perp \ll \Gamma_{||}$
- 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):

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

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

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

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

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

Even worse for energy – see next ……

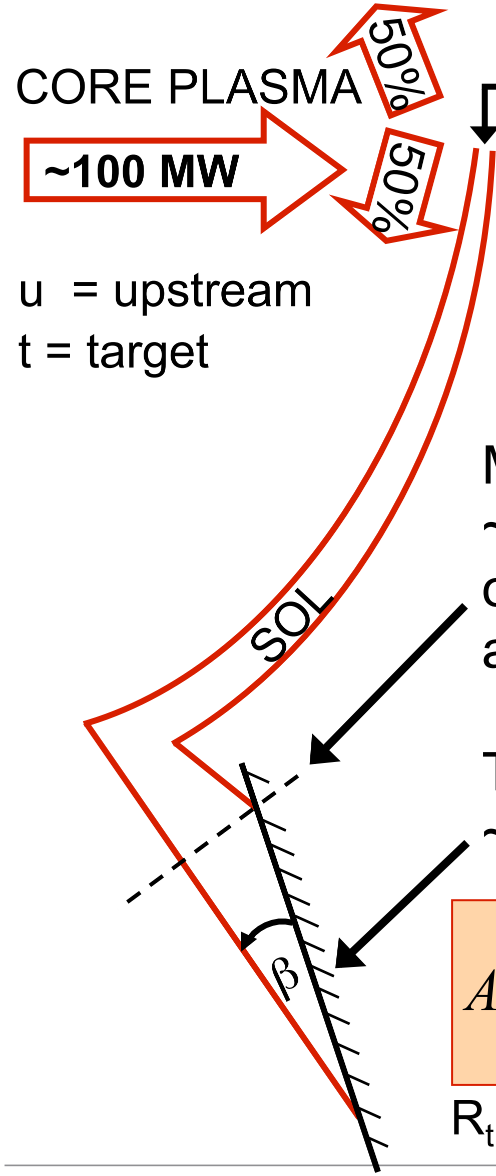
The problem with λ_q

- SOL width for power, λ_q, is also small and is an important parameter of the edge plasma
- As for particles, λ_q is determined by the ratio of $\perp$ to $||$ transport (e.g. cross-field ion conduction and parallel electron conduction: i.e. $\propto (\chi_\perp/\chi_{||})^{1/2}$), where $\chi_\perp$ is anomalous
- 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 [8]: $\boxed{\lambda_q \propto P_{SOL}^{-0.5} B_\varphi^{-0.9} q_{95}^{0.4} n_u^{0.15}}$
  - ITER modelling [9] yields λ_q = 5 mm, JET scaling gives λ_q = 3.7 mm (cf. a=2.0 m)
  - Very recent multi-machine scaling [10] gives λ_q/R ~ constant
- Note also that the parallel power flux, $q_{||} \propto P_{SOL}/\lambda_q$ → as much as **1 GWm^{-2}** in ITER

Stored energy scales strongly with tokamak major radius, $W \sim \propto R^5$ [11]
But power deposition area in the divertor $\propto R\lambda_q$ only (~3.0 m^2 in ITER)

Bottom line is that despite its increased physical size, ITER will concentrate more power into a narrower channel at the plasma edge than today's devices. The use of divertor detachment, radiation and geometry will be used to reduce the surface power flux densities to manageable levels, but careful monitoring will be critical (unlike today's expts., where it is "useful")

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 ~ 2.5º)

\+

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

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

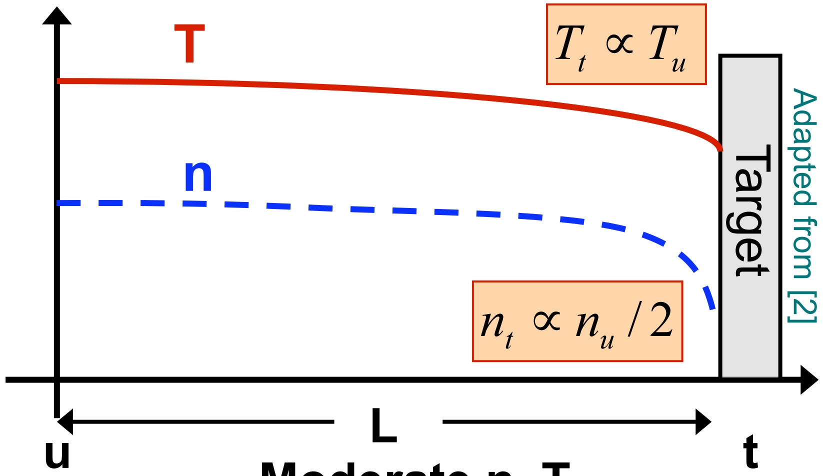
R_t = 5.6 m, $(B_\theta/B)_t$ ~0.11

- 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

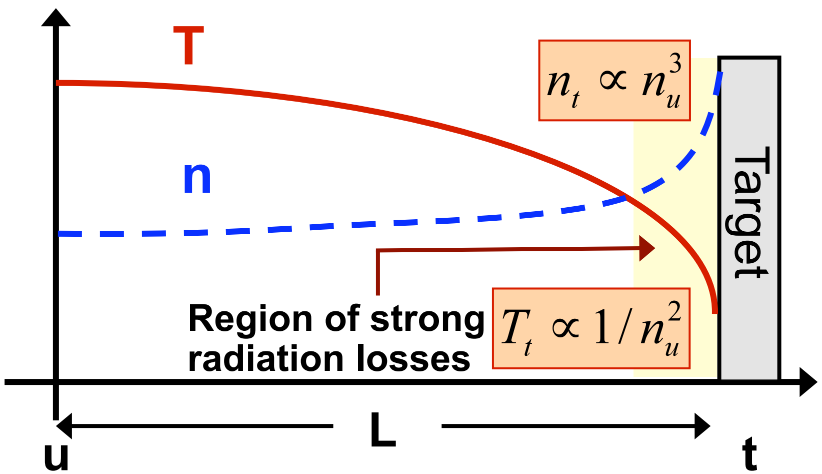
→ $q_\perp$ ~ 40 MWm^{-2} per target if no radiative (or other) dissipation → cannot be tolerated for more than ~2-3 s on actively cooled surfaces

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: $q_{\|,t} = \gamma n_t c_{st} T_t + n_t c_{st} \varepsilon_{pot}$

γ = sheath heat transmission coefficient

ε_{pot} = potential energy per incident ion

ν^* rises as n_u rises, finite electron heat conductivity: $q_{\|,cond} = -K_\| dT / ds_\|, K_\| = \kappa_0 T^{5/2}$

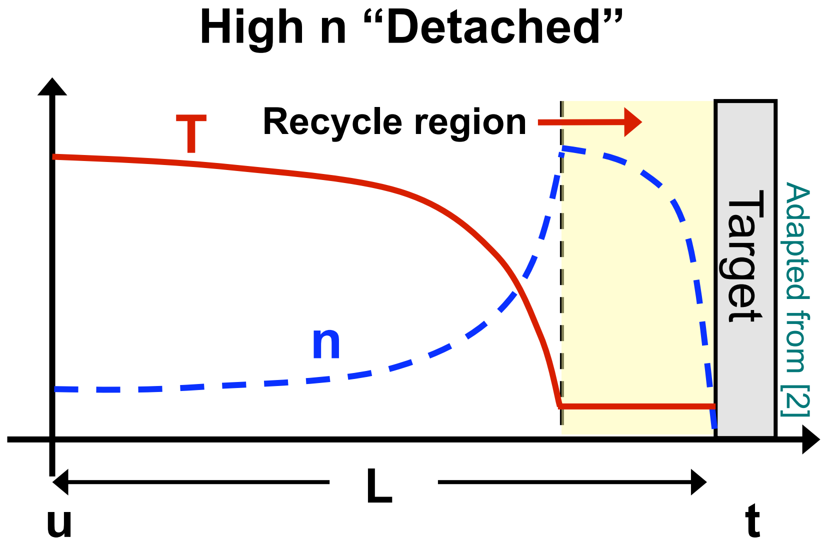
(note: $\kappa_{0,e} \gg \kappa_{0,i}$)

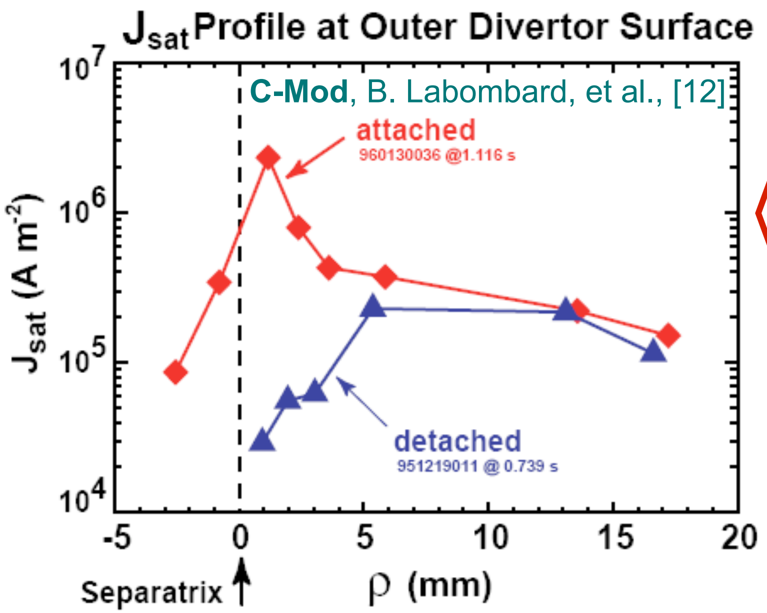
allows parallel T gradients to develop → T_t decreases, but pressure balance maintained ($\nabla p_\| \sim 0$) so that n_t rises strongly ($\Gamma_t \propto n_u^2$)

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)



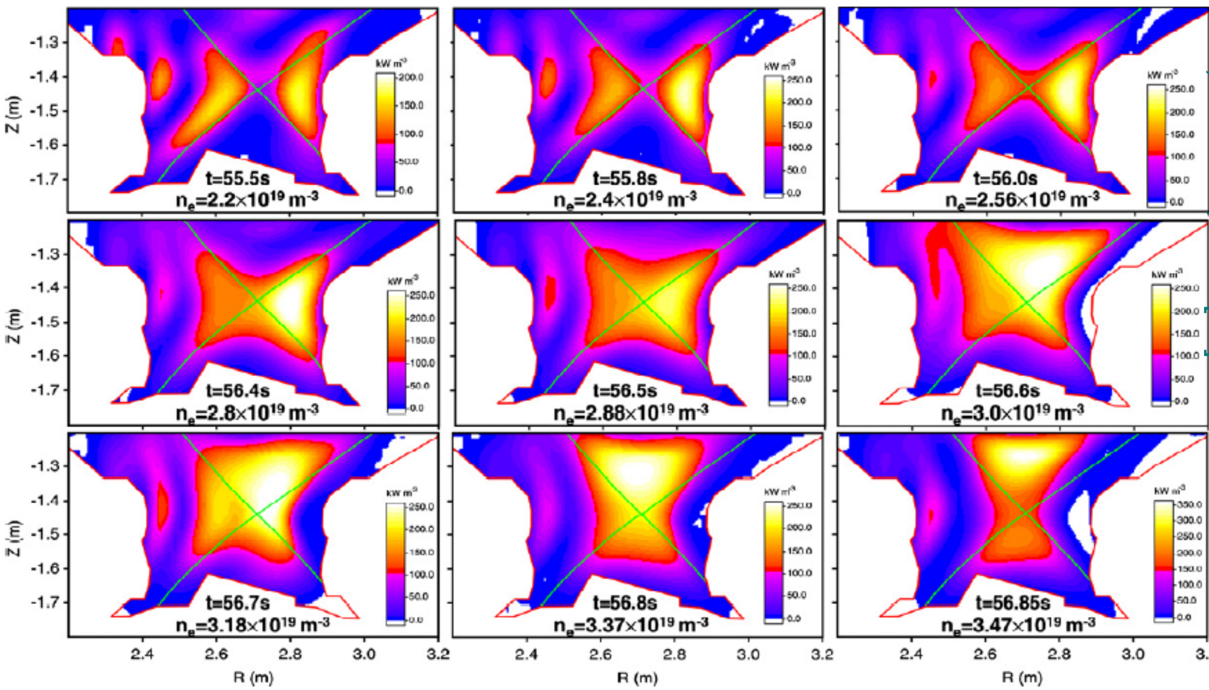


At sufficiently low T_t, (< 5 eV), neutral ionisation rate < ion-neutral friction processes (charge exchange, elastic scattering). Momentum transferred from ions to dense cloud of neutrals in front of the plate (recycle region) → begins to reduce n_t, $\nabla p_{||} \neq 0$ and plasma pressure falls 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



JET, A. Huber, et al. [13]

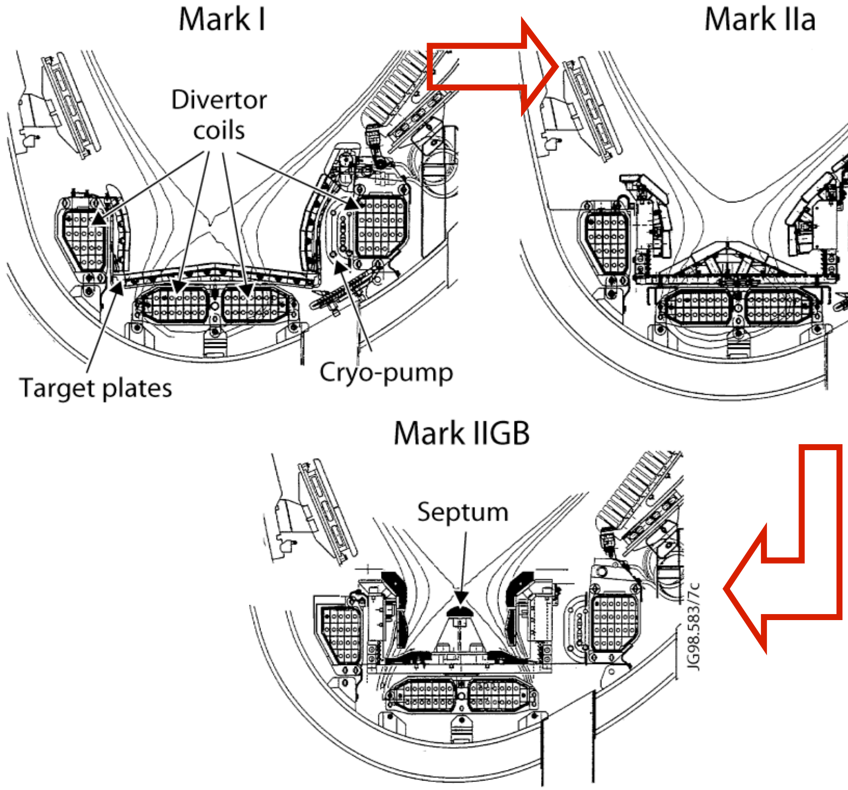
Limit detachment to regions of highest power flux (where it is needed most).
Maintain remainder of SOL in high recycling (attached)
A few ways to arrange that this happens more readily:

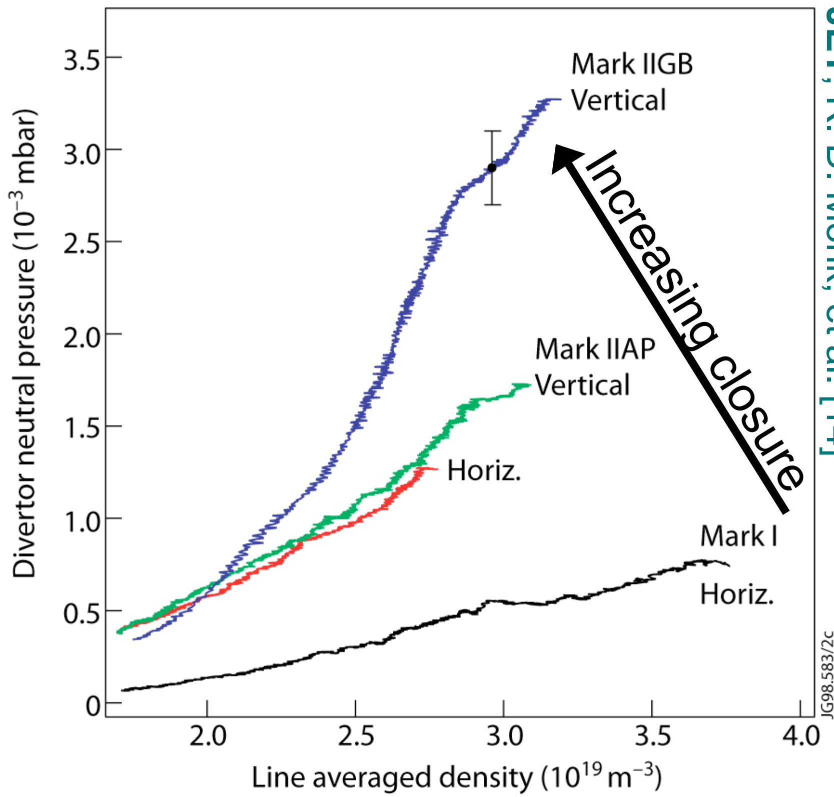
Divertor closure

Target orientation

Impurity seeding

Divertor closure





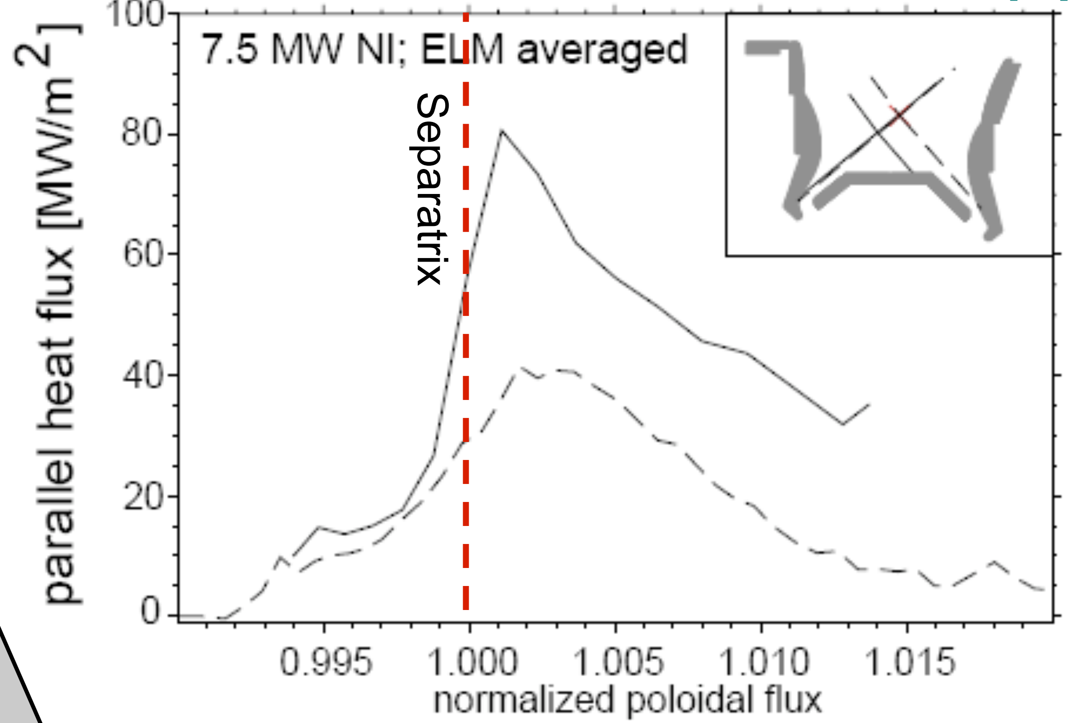
JET, R. D. Monk, et al. [14]

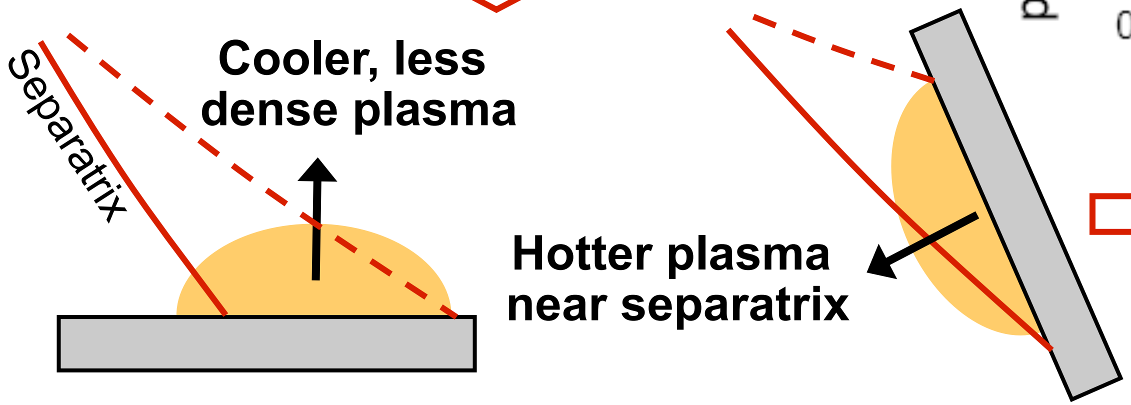
- Increased closure significantly improves divertor neutral pressure → increased neutral density (n_n), promoting earlier detachment
- Closing “bypass” leaks important for increasing n_n
- Divertor closure also promotes helium compression and exhaust – very important for ITER and reactors

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. [15]





Increased ionisation near sep.

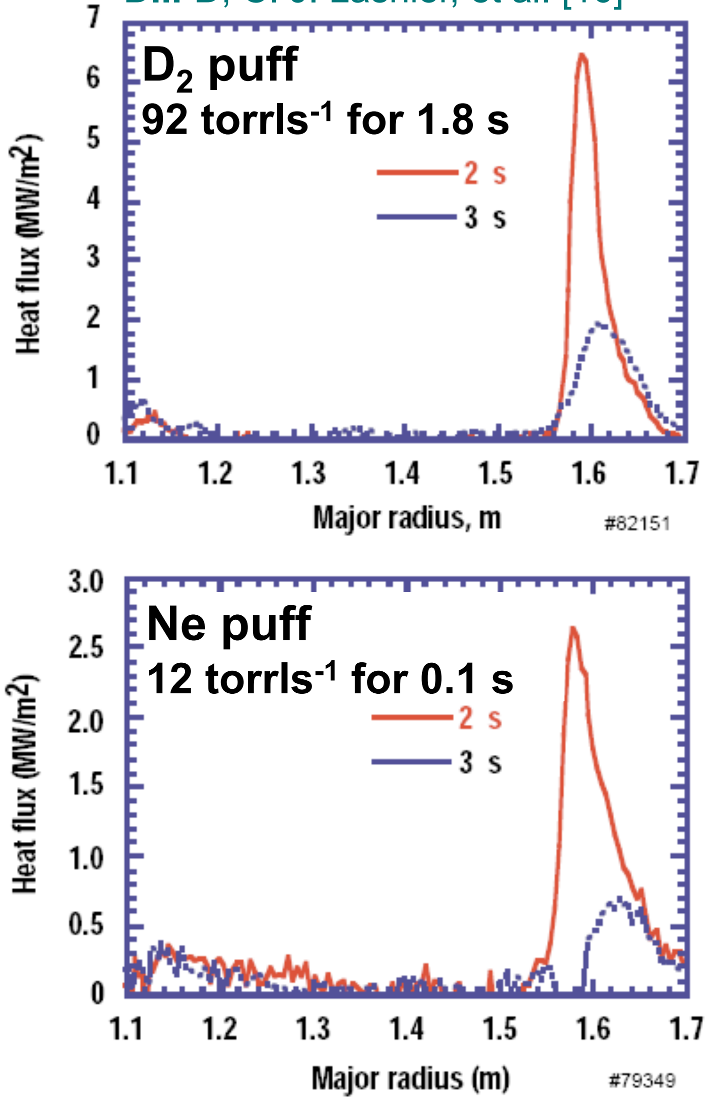
Higher n_t, lower T_t

Higher CX losses

Pressure loss → $q_{||}$ ↓

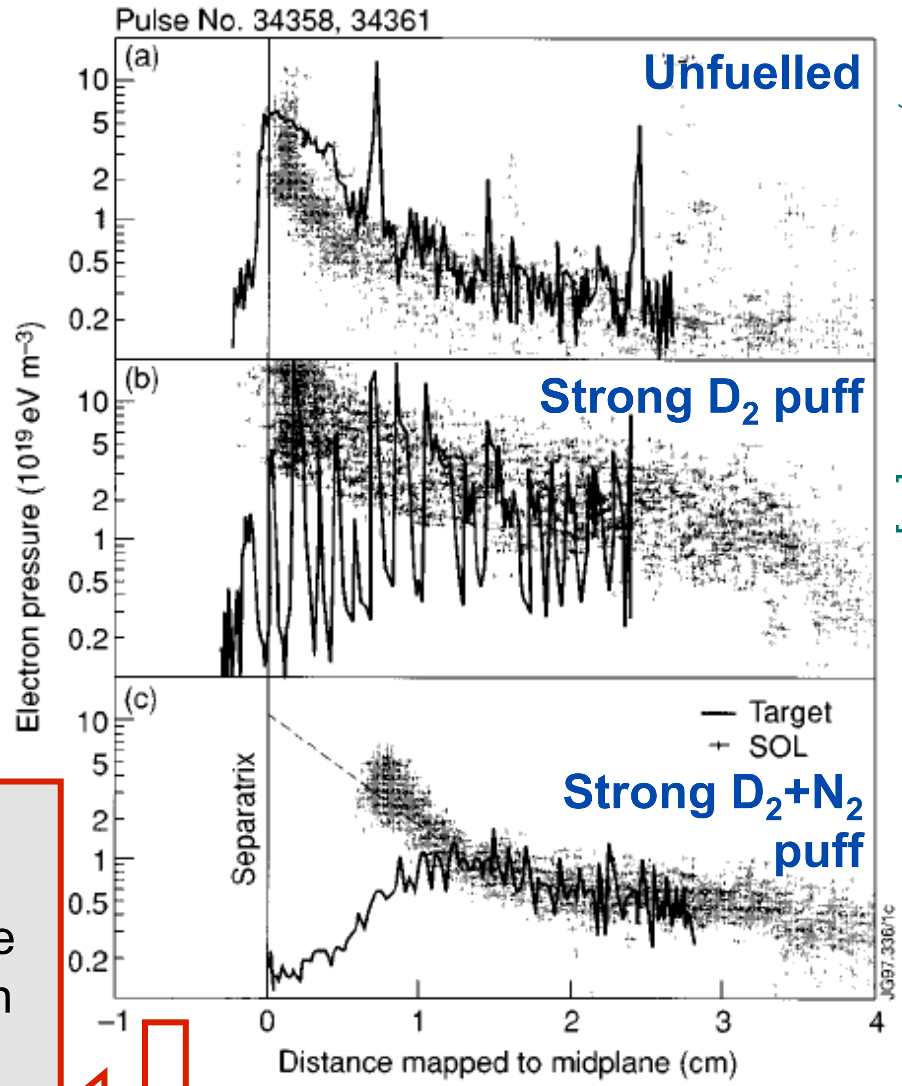
Impurity seeding

DIII-D, C. J. Lasnier, et al. [16]

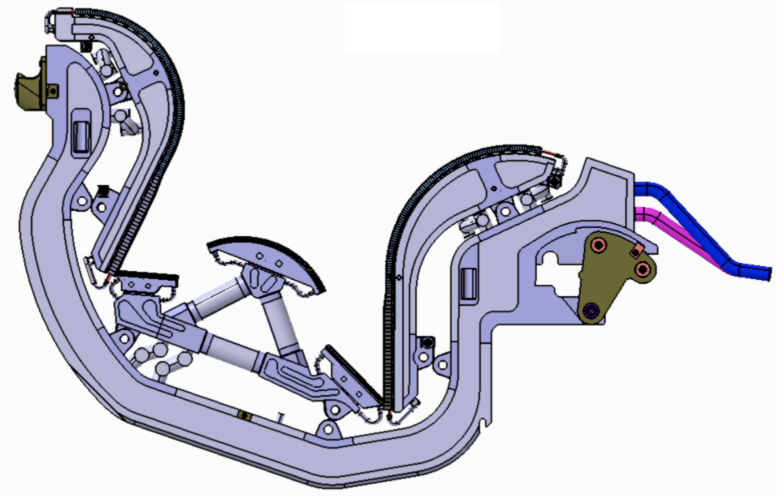


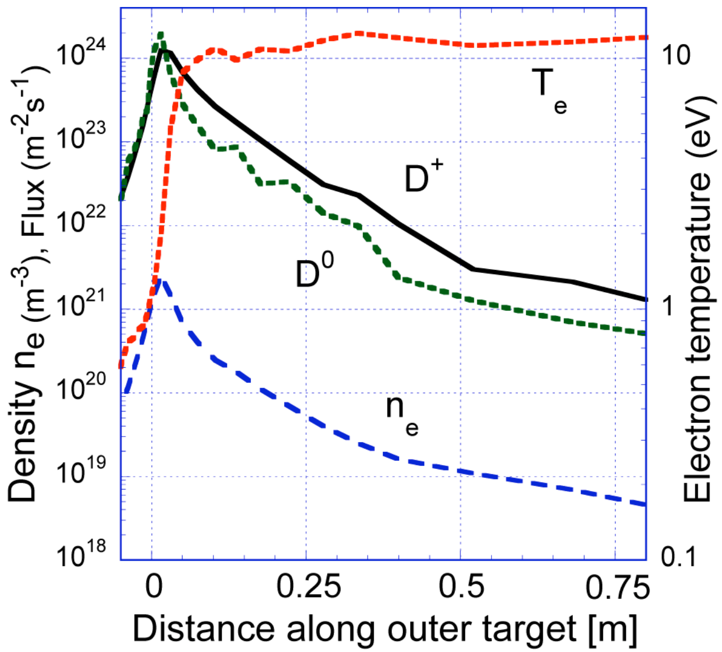
Strong impurity seeding also reduces ELM size but high price can be paid in confinement

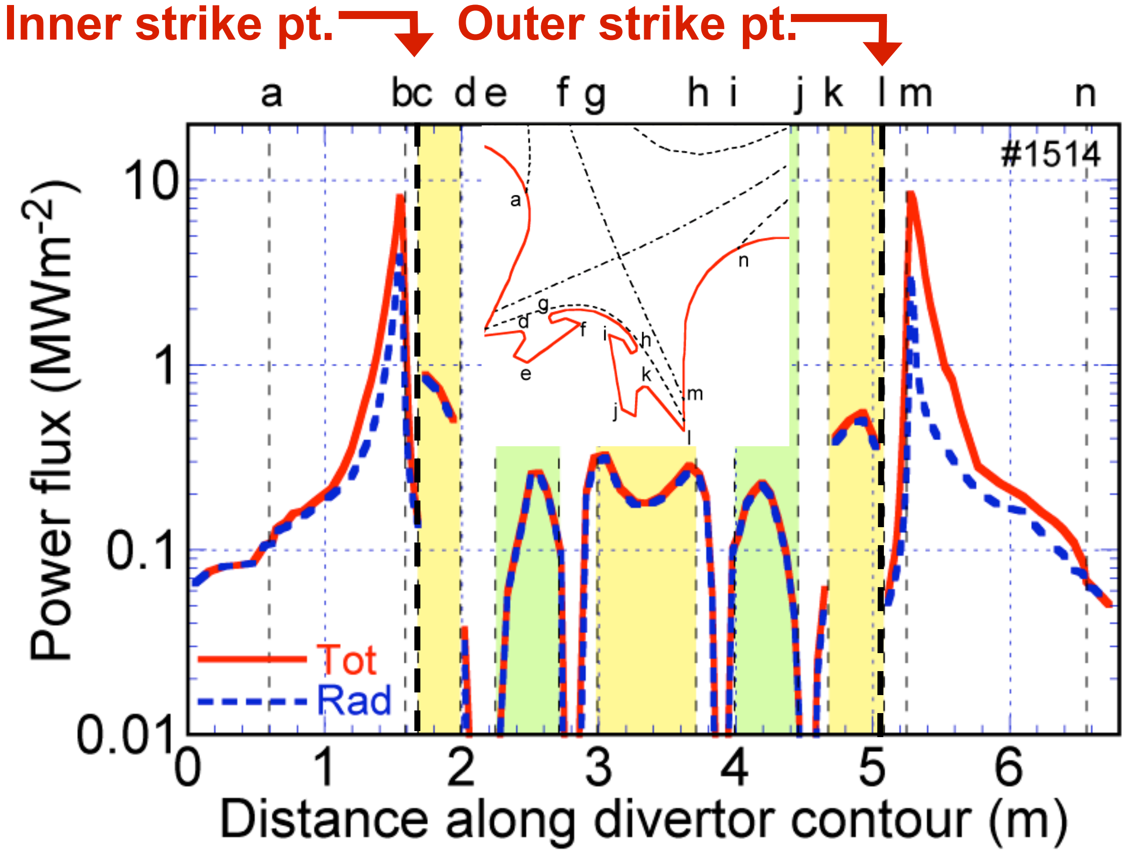
JET, G. F. Matthews et al. [17]



ITER divertor achieves partial detachment





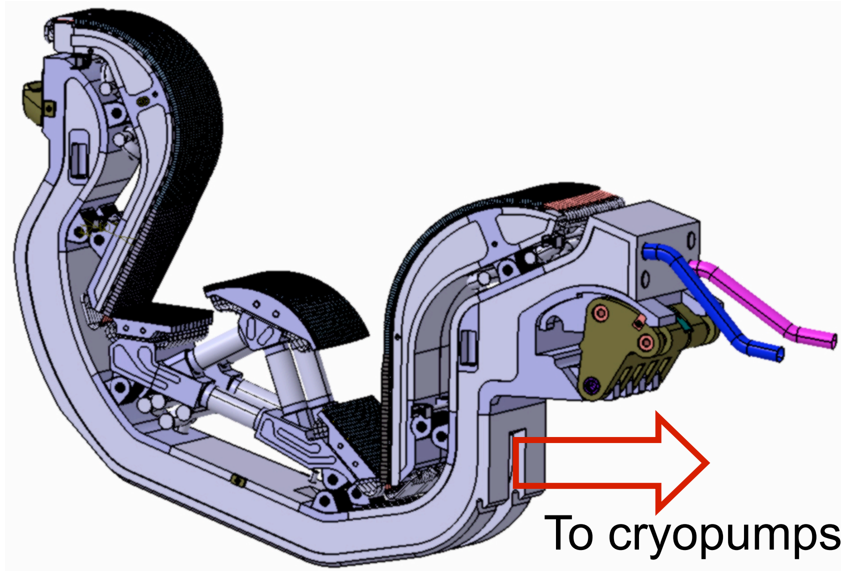


R.A. Pitts, A. S. Kukushkin, submitted to Physica Scripta (2009)

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: $\tau^*_{p,He}/\tau_E \le 5-10$ 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) [18].

Helium enrichment: $$\eta_{He} = \frac{n_{He}^{pump}/2n_{D2}^{pump}}{n_{He}^{plasma}/n_e} = \frac{C_{pump}}{C_{plasma}}$$

is the ratio of He concentration in the divertor compared to the main plasma.

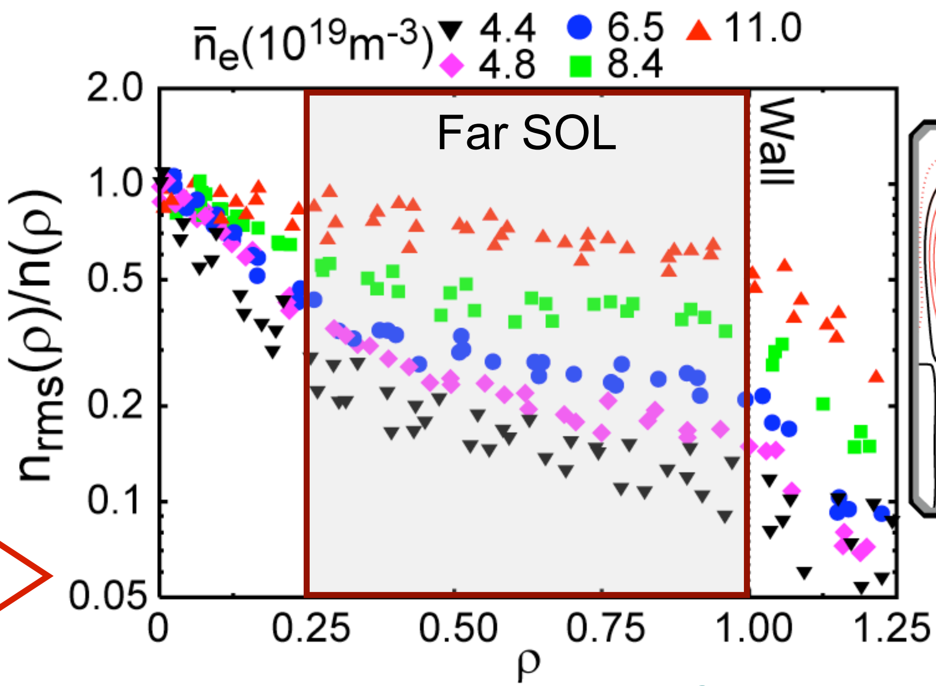
e.g. ITER: He prod. rate ~2×10^{20} s^{-1}
Max. divertor pumping speed ~200 Pa m^{3}s^{-1} ~ 1×10^{23} He atom s^{-1}
→ C_{pump} ~ 2×10^{-3} = 0.2%
Typical acceptable He conc. in the core: ~4% → η_{He} = 0.2/4 = 0.05 is minimum required. The values of η_{He} and $\tau^*_{p,He}$ required for ITER have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

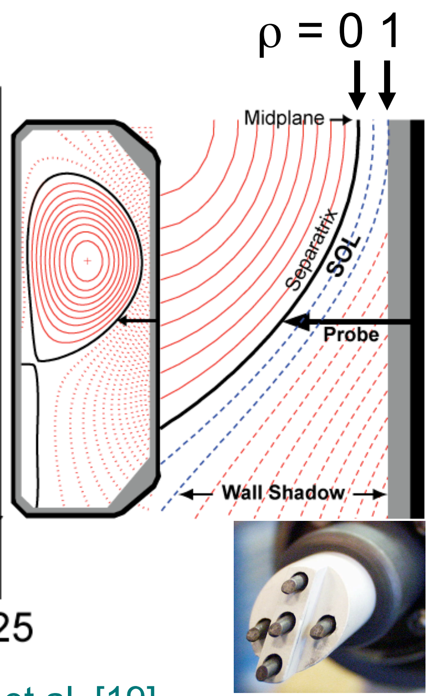
Perpendicular SOL transport

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





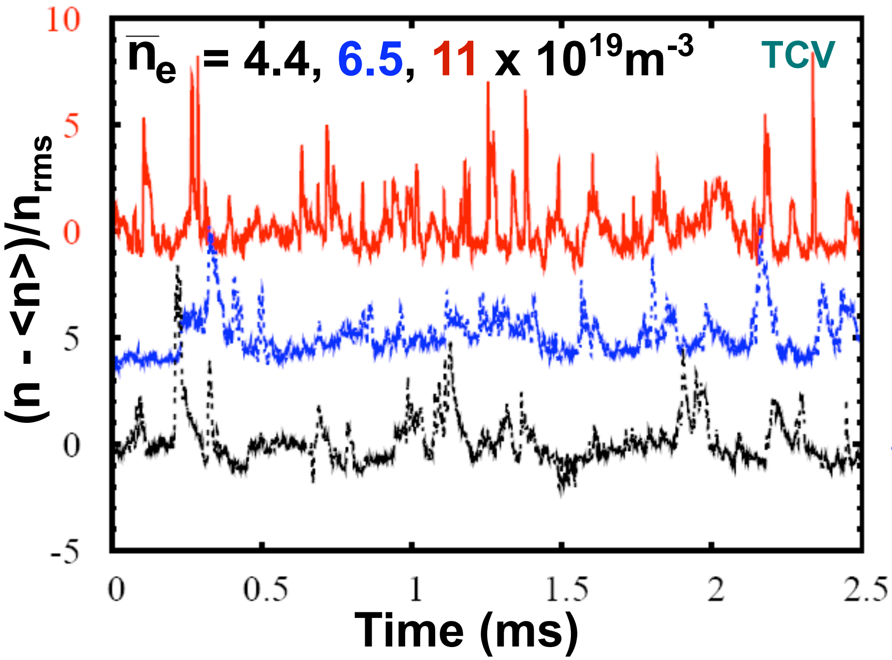
Note how broadening occurs mostly in the "far SOL"

Garcia, Pitts et al. [19]

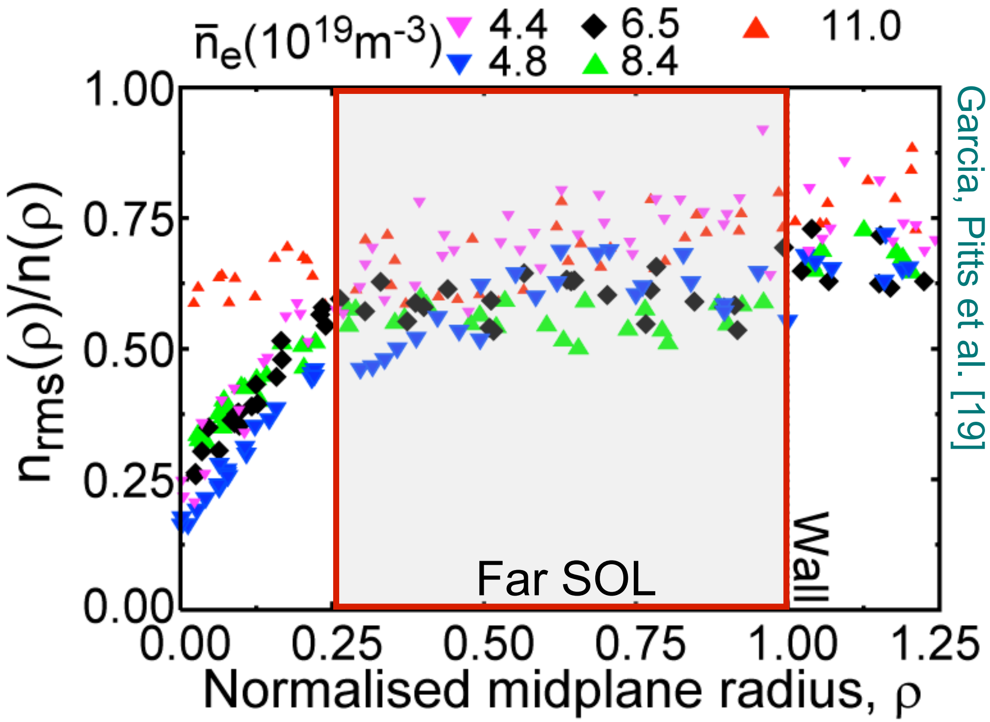
What causes the broad n_e profiles?

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



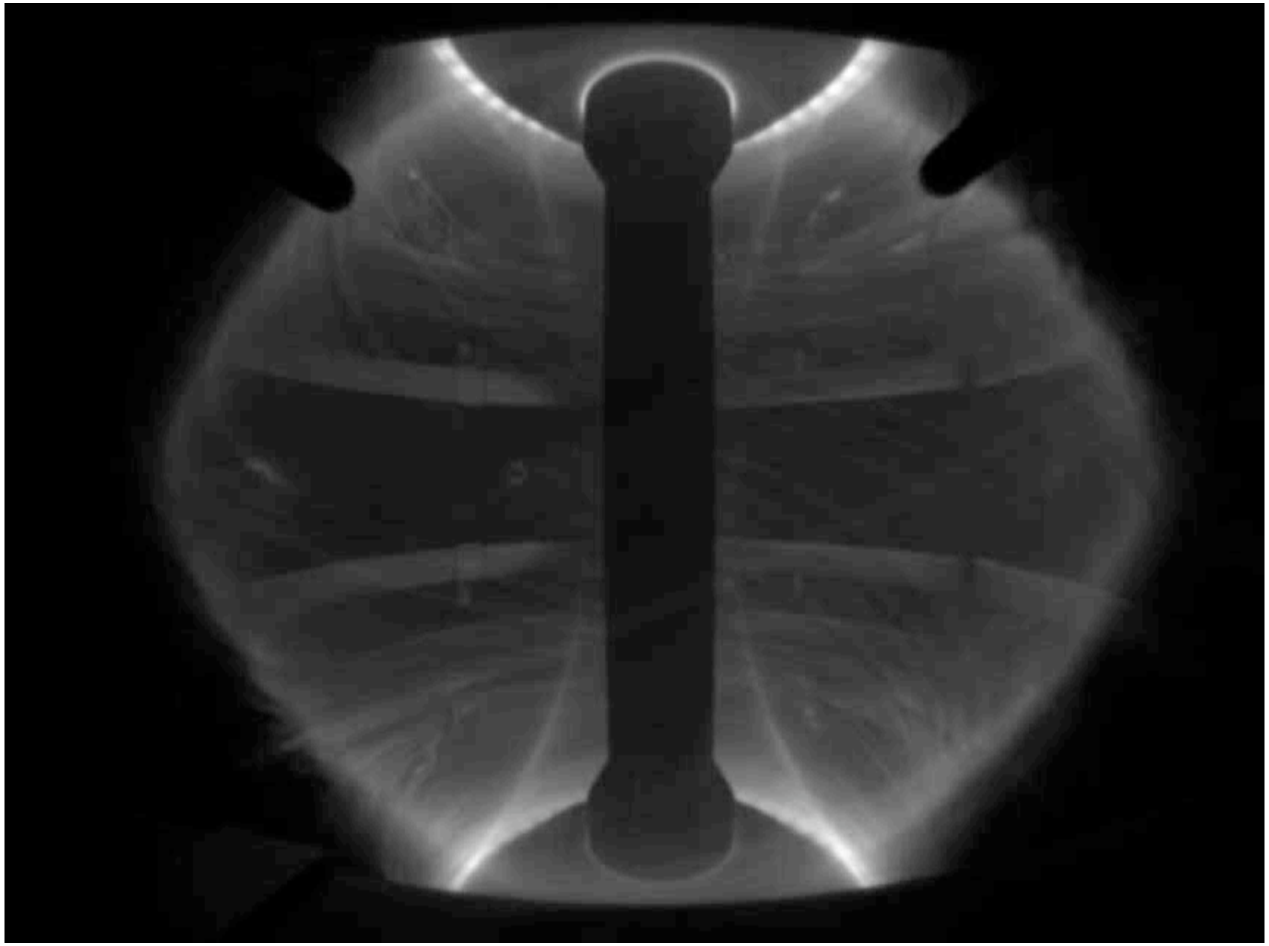
At high densities, bursts more frequent



L-mode filaments on MAST

Courtesy MAST team, UKAEA Culham

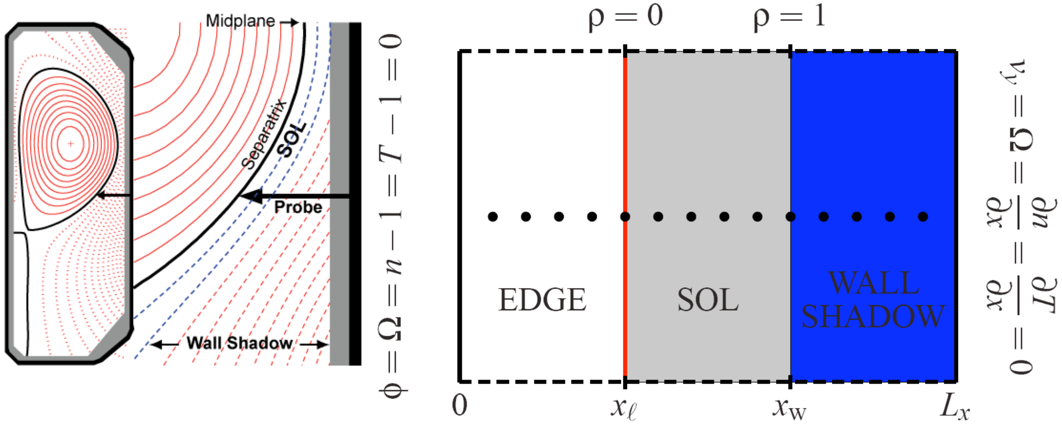
Probes provide quantitative data on the real particle flux, but only at fixed toroidal and poloidal locations. Fast visible imaging allows the 3D picture to be seen but analysis more challenging.



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

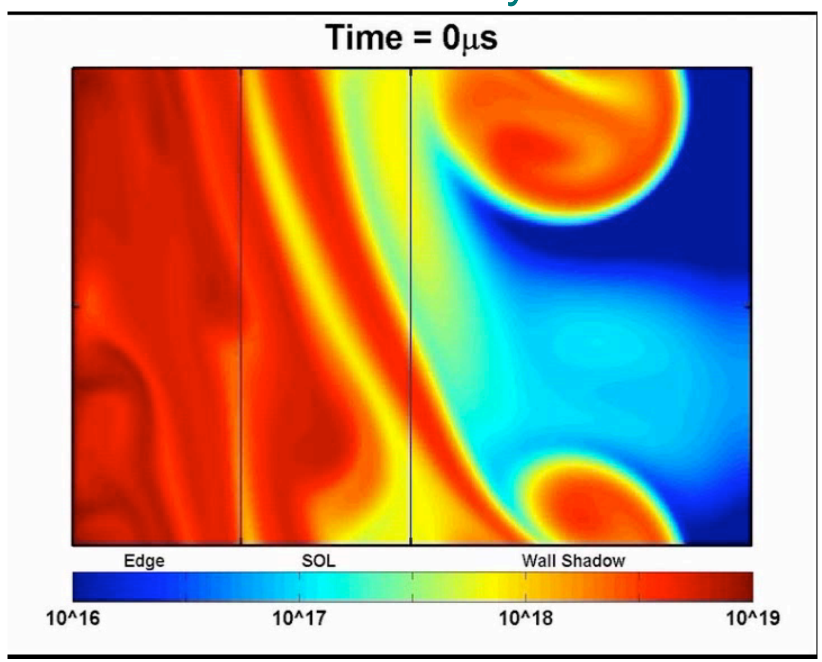
Origin of the bursts?

- 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

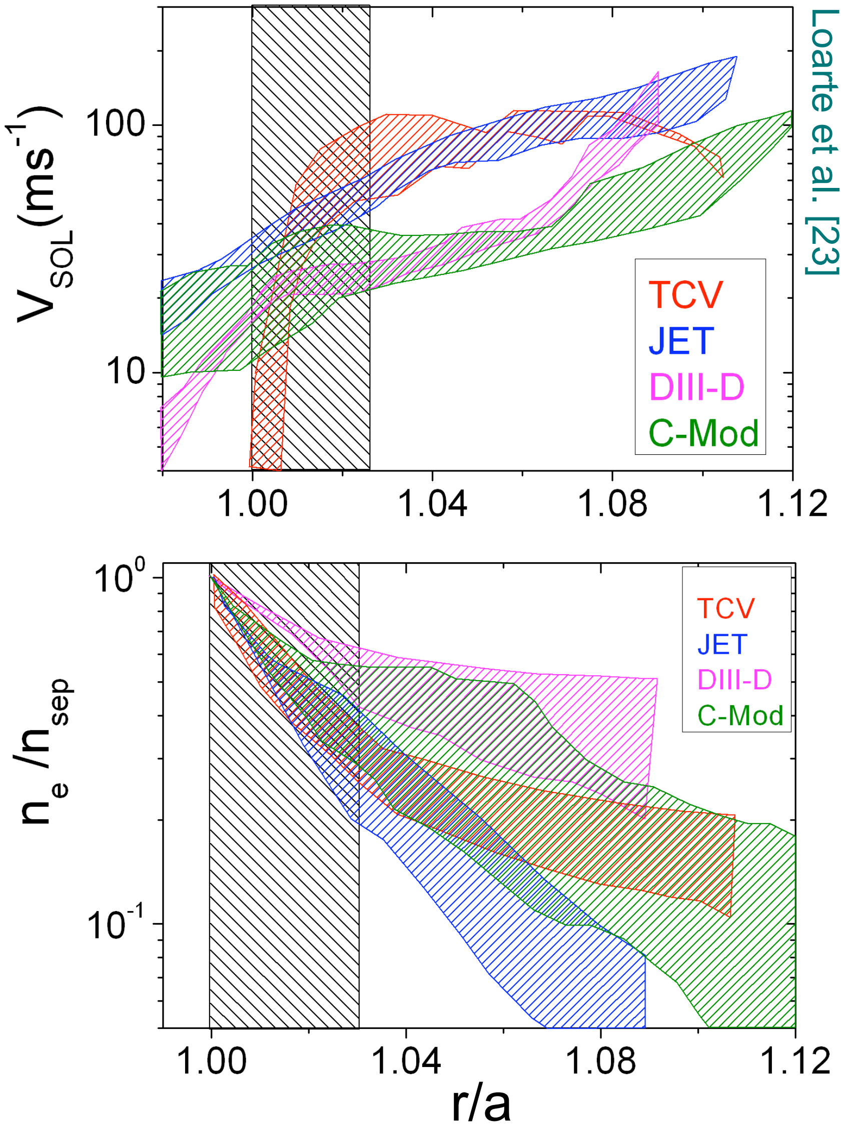


- ESEL code (Risø [20]) simulates 2D region centred on outboard midplane
- Exhaustively tested against TCV high density case [19, 21, 22]

Particle density field



Bursty transport now in ITER baseline!

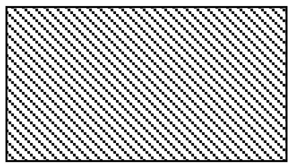


Stationary, inter-ELM power fluxes to ITER main wall now assumed to be dominated by convective, intermittent transport

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

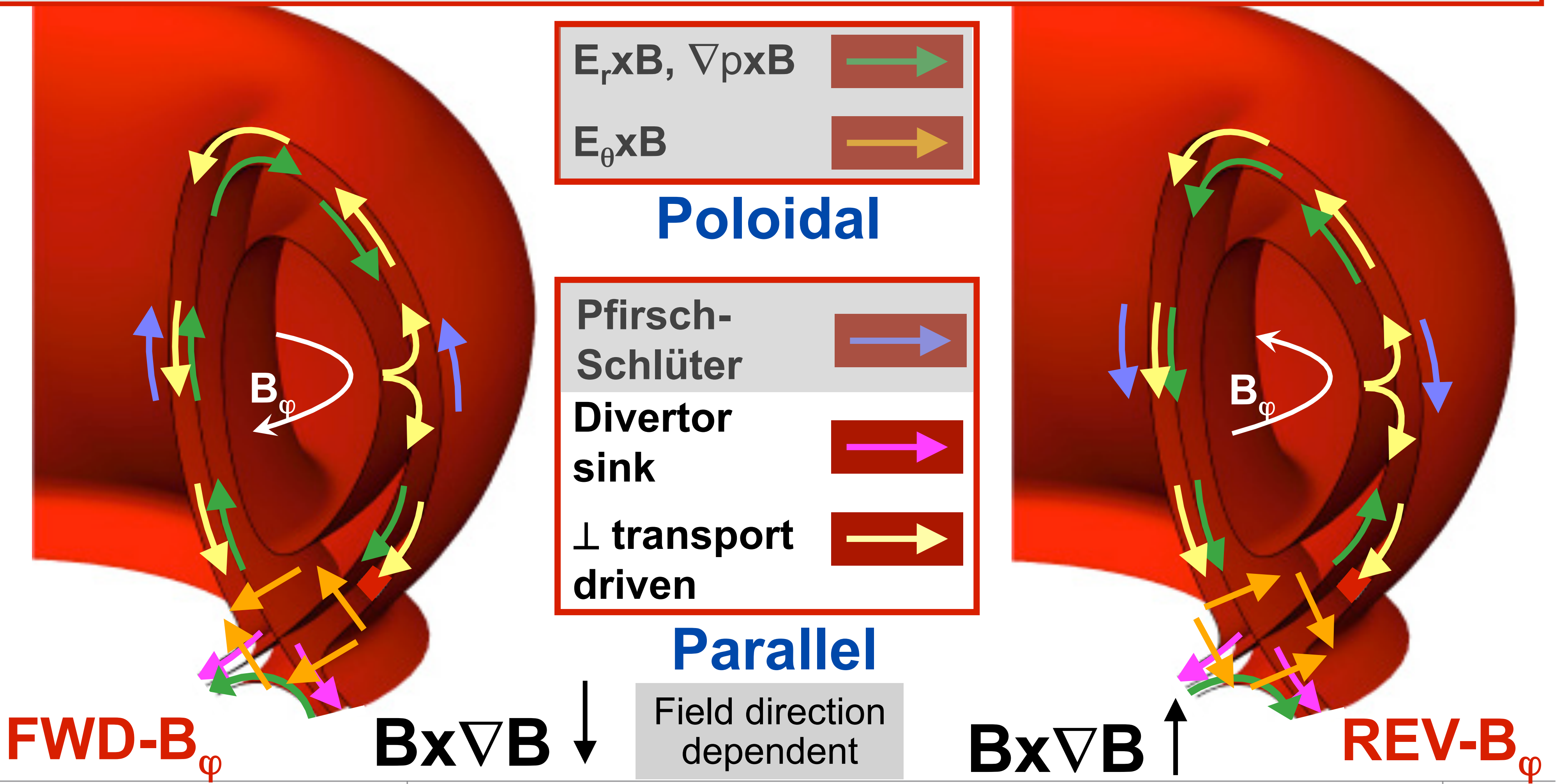
Power to ITER first wall <20 MW (20% P_{SOL})
Part. flux to ITER first wall < 1×10^{24}s^{-1} (10% Γ_{div})



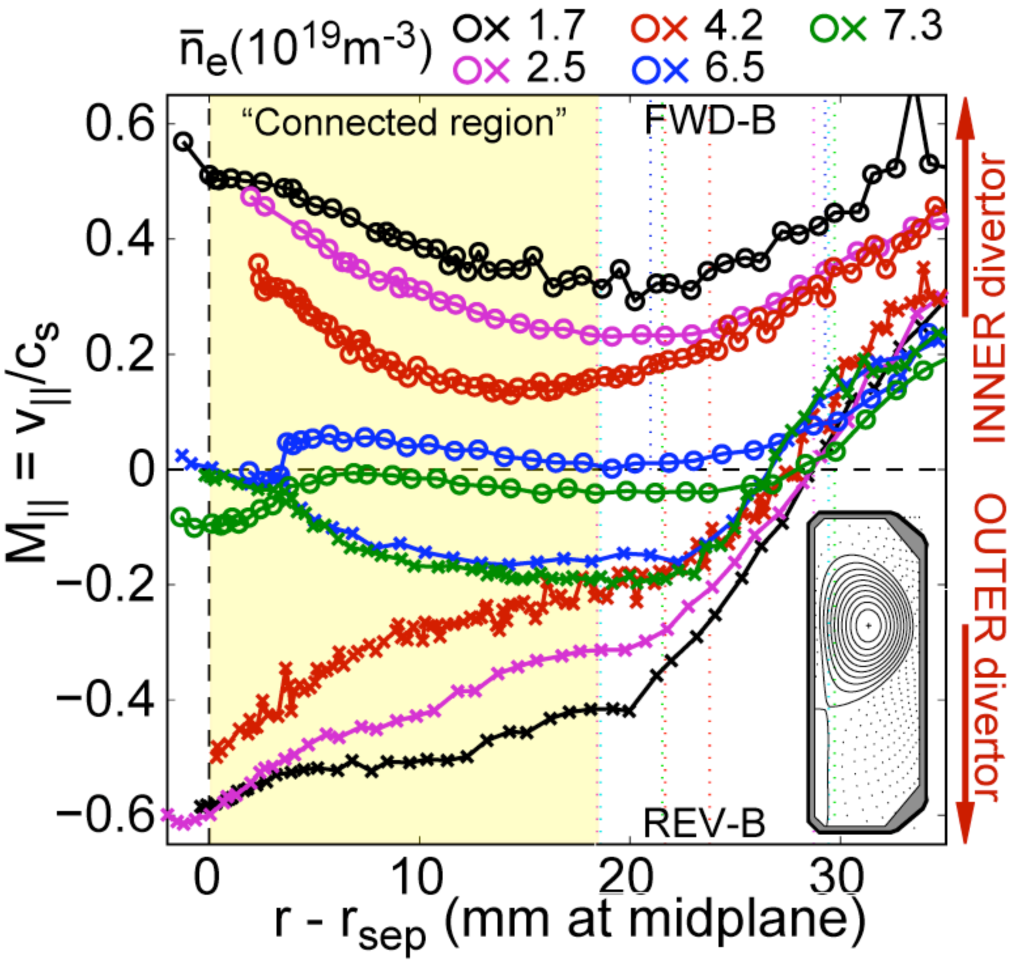
Region of connected SOL from inner to outer strike points

Parallel SOL ion flows

Determine transport of impurities from source to destination in a tokamak – material migration – T-retention (Slides 38-46)



Flows can be very strong

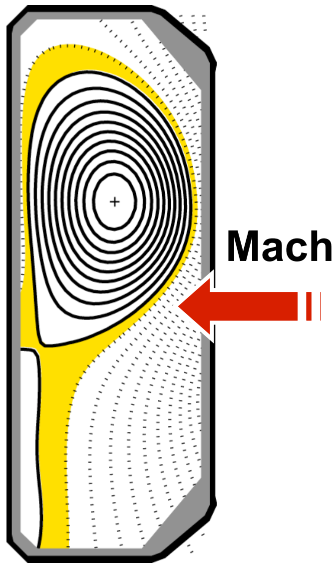


$M_{||} = 0.5 \rightarrow v_{||} \sim 30\ kms^{-1}$!

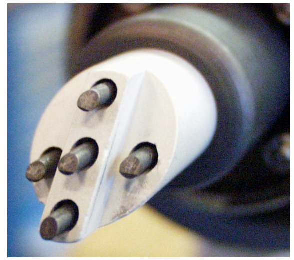
Have been measured on several tokamaks – TCV is a good example [24,25]

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

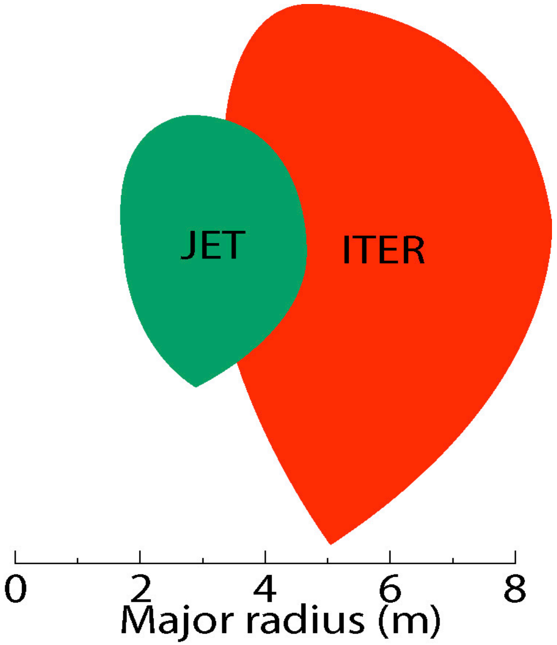


Part II

Plasma-surface interactions

The challenge: upscale to ITER is a 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

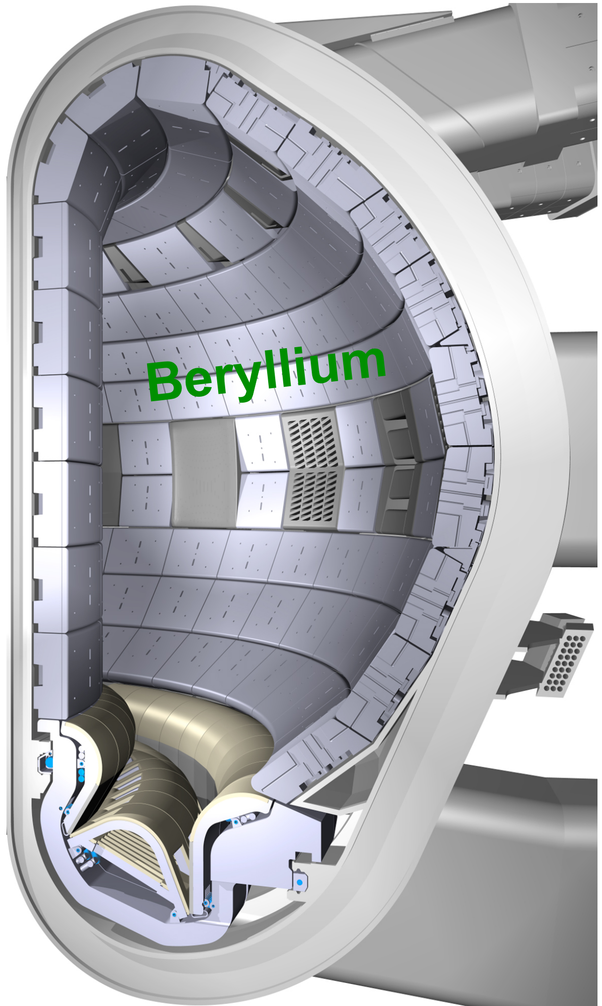
Extracted from Matthews et al. [26]

1 ITER pulse ~ 0.5 JET years energy input

1 ITER pulse ~ 6 JET years divertor fluence

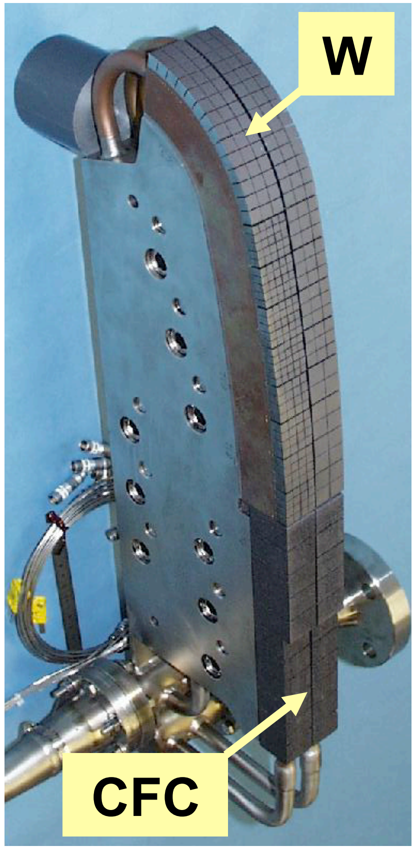
- Stored energy goes ~ $\propto R^5$ → ~35× higher on ITER than JET
- But deposition area for power in the divertor $\propto R\lambda_p$ → $\lambda_{p,ITER} \sim \lambda_{p,JET}$ → ~3.0 m^2 ITER cf. ~1.0 m^2 JET → ITER must project ~35× the energy into only 3 times the area
- High stored energy means that unmitigated disruptions and ELMs far beyond anything tolerable by today's materials.

ITER materials choices



Surface areas:
Be: ~700 m^2, W: ~80 m^2
CFC: ~40m^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 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

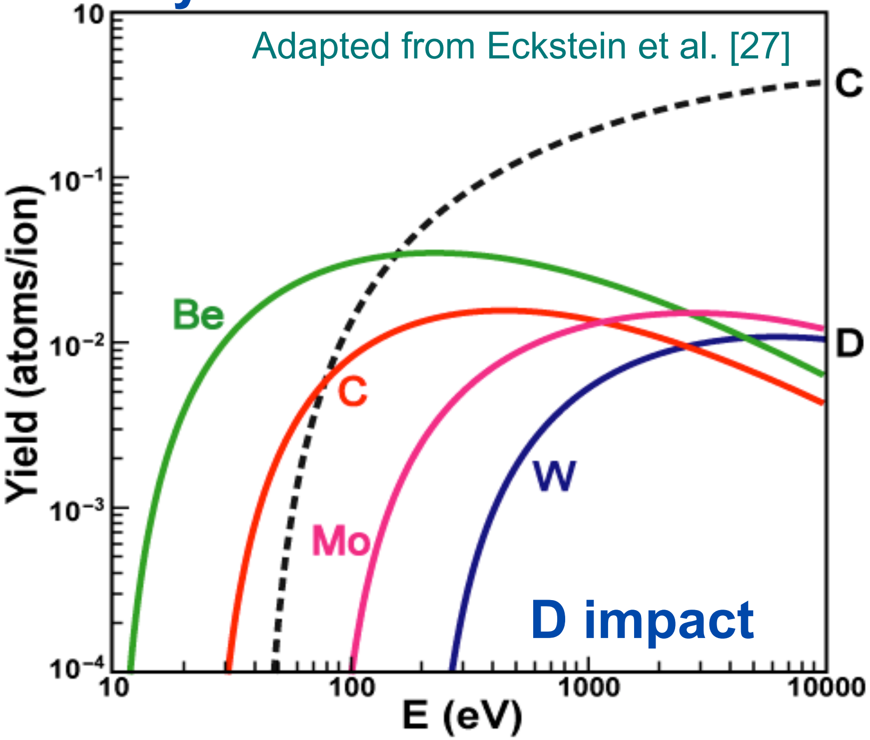
Impurity migration

Migration = Erosion → Transport → Deposition

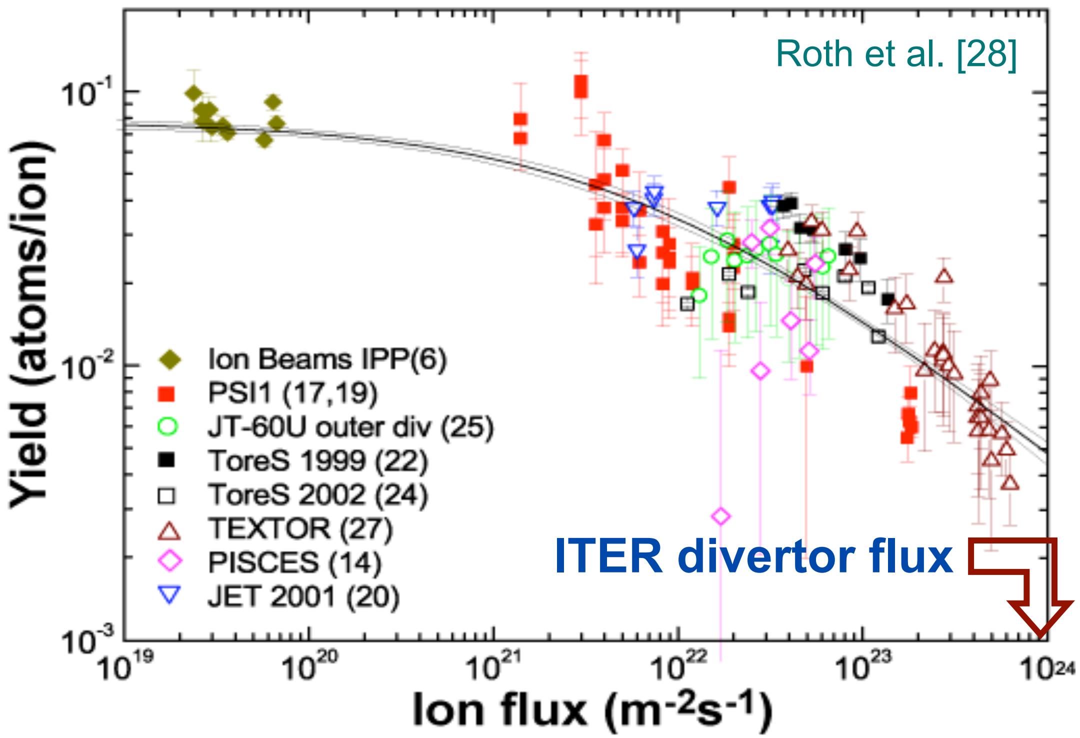
Re-erosion

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 [29]

Transient erosion

Transients are the biggest threat for large scale erosion in ITER. The magnitude of the burning plasma stored energy (~350 MJ) far exceeds that of the largest operating devices (~10 MJ, JET), but surface areas for energy deposition only ~factor 2 larger (slides 9,27)

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

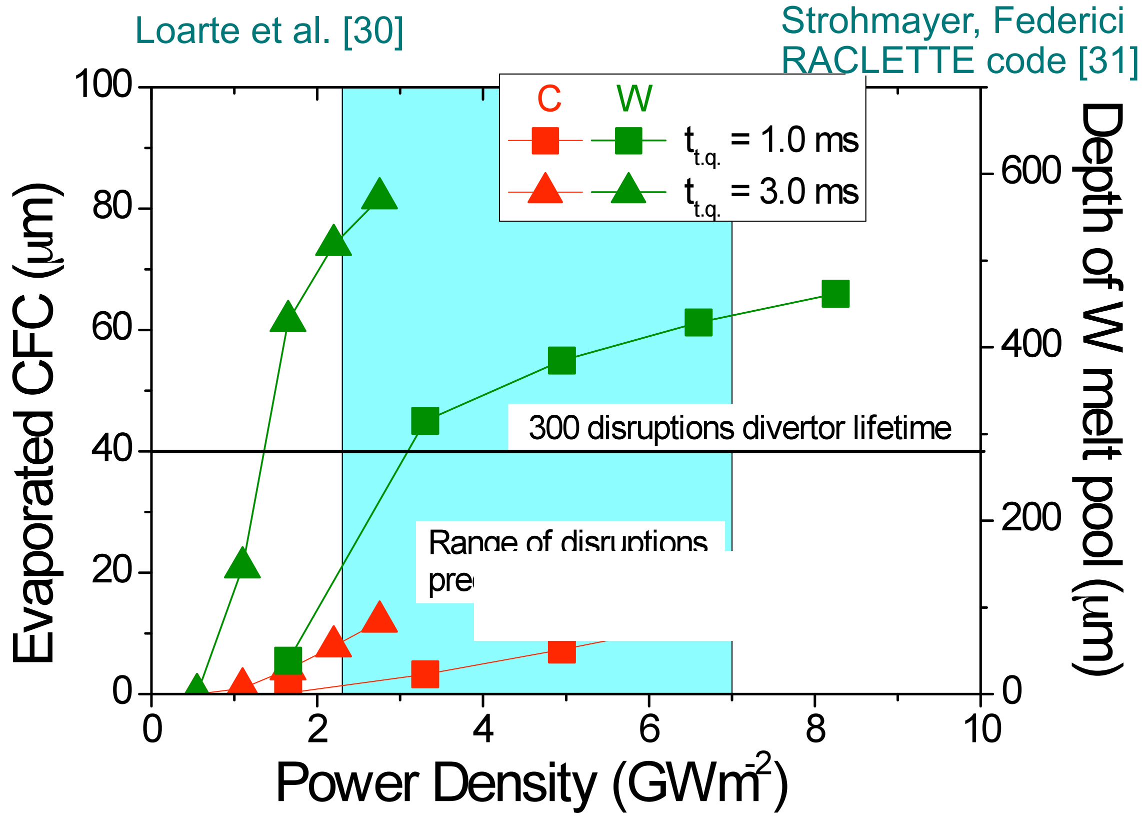
Disruptions ⇩

Worst case full energy disruptions → peak energy densities on the divertor of 5.0-15 MJm^{-2} on timescales of 1.5-3 ms (thermal quench)

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



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

ELM induced erosion

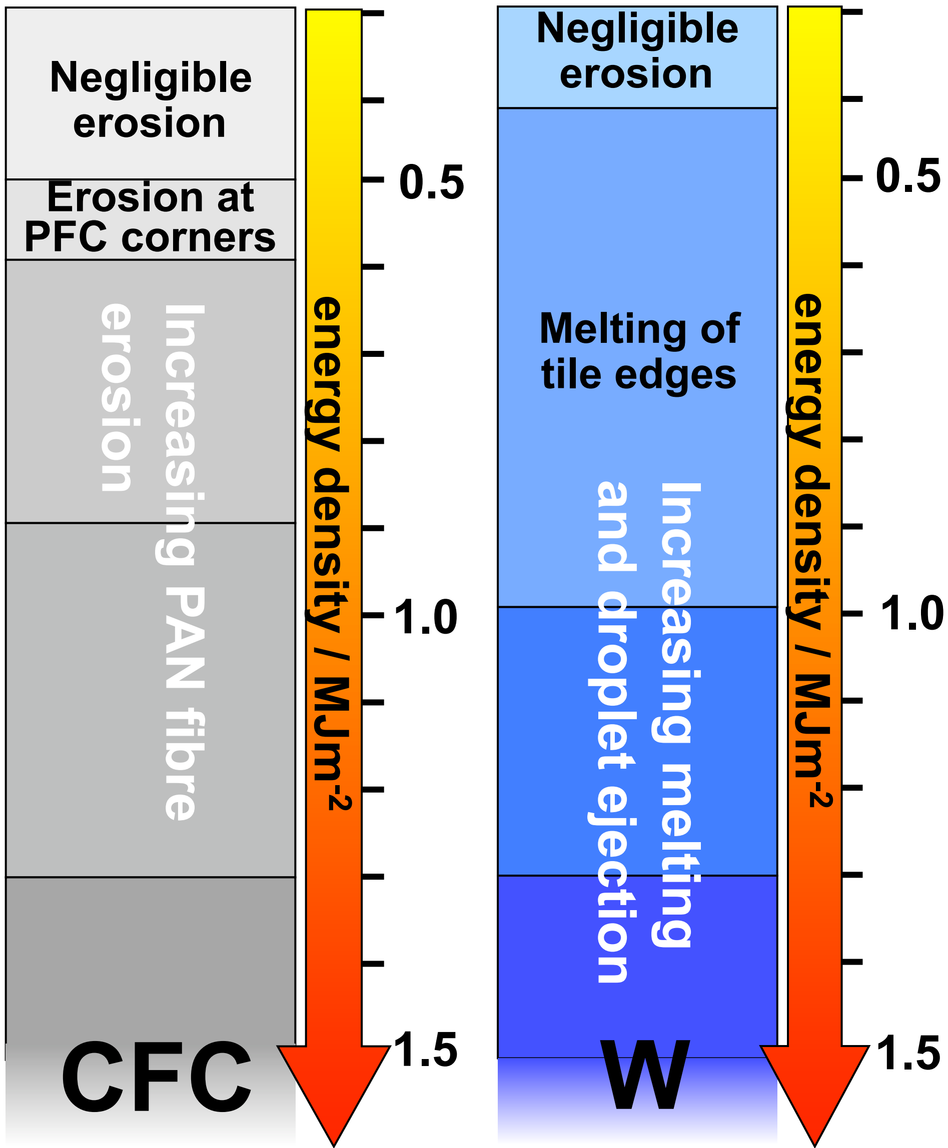
Real material limits are much lower → Results from Russian plasma simulators [32,33]:

Erosion limit for CFC reached due to PAN fibre erosion

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

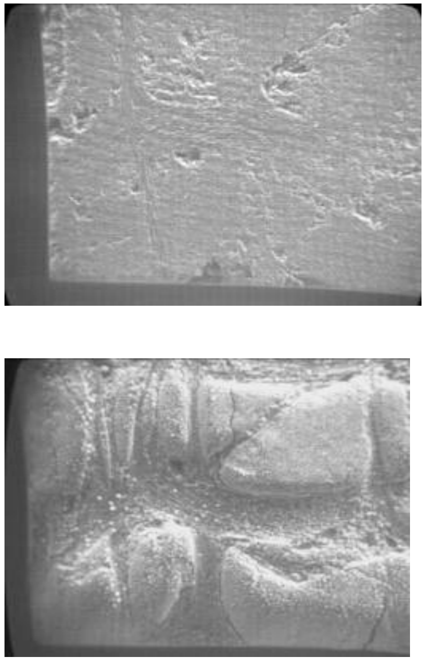


CFC & W erosion under repetitive transients

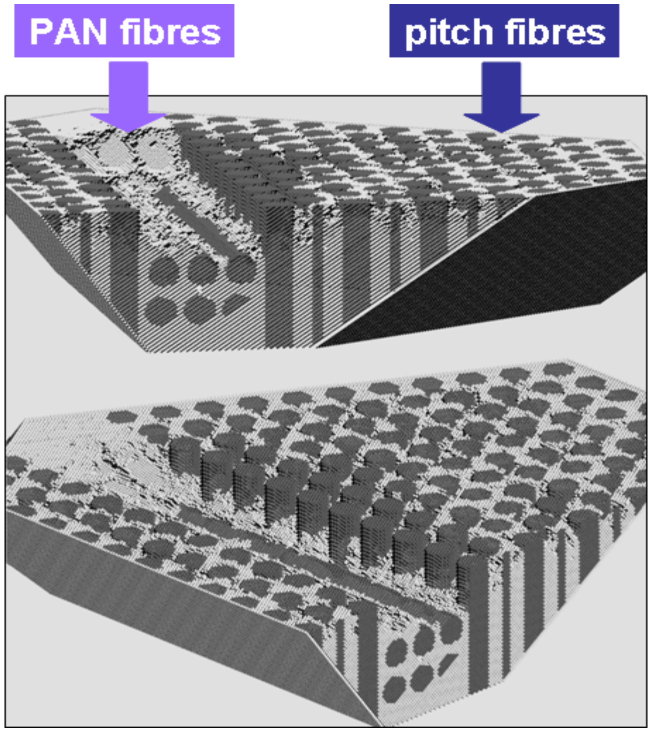
CFC
Enhanced erosion of PAN fibres and brittle destruction

W
Cracking and shallow melting for near threshold loads & droplet ejection for larger loads

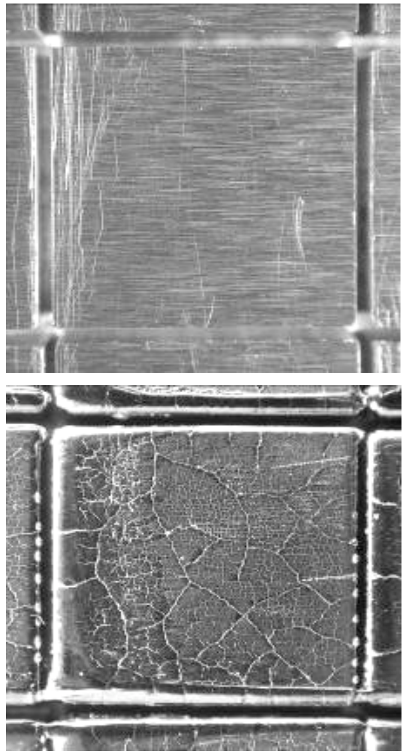
QSPA (experiment)



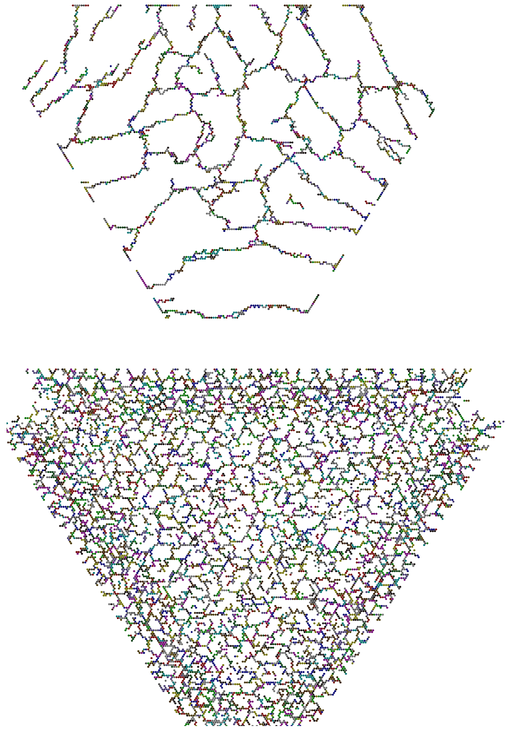
PEGASUS (code)



QSPA



PEGASUS



FACT

Providing reliable methods to mitigate or suppress ELM heat loads and mitigate disruptions on ITER is one of the most challenging issues faced by the project

Impurity migration

Migration = Erosion → Transport → Deposition

Re-erosion

Transport creates and moves impurities

Ions:

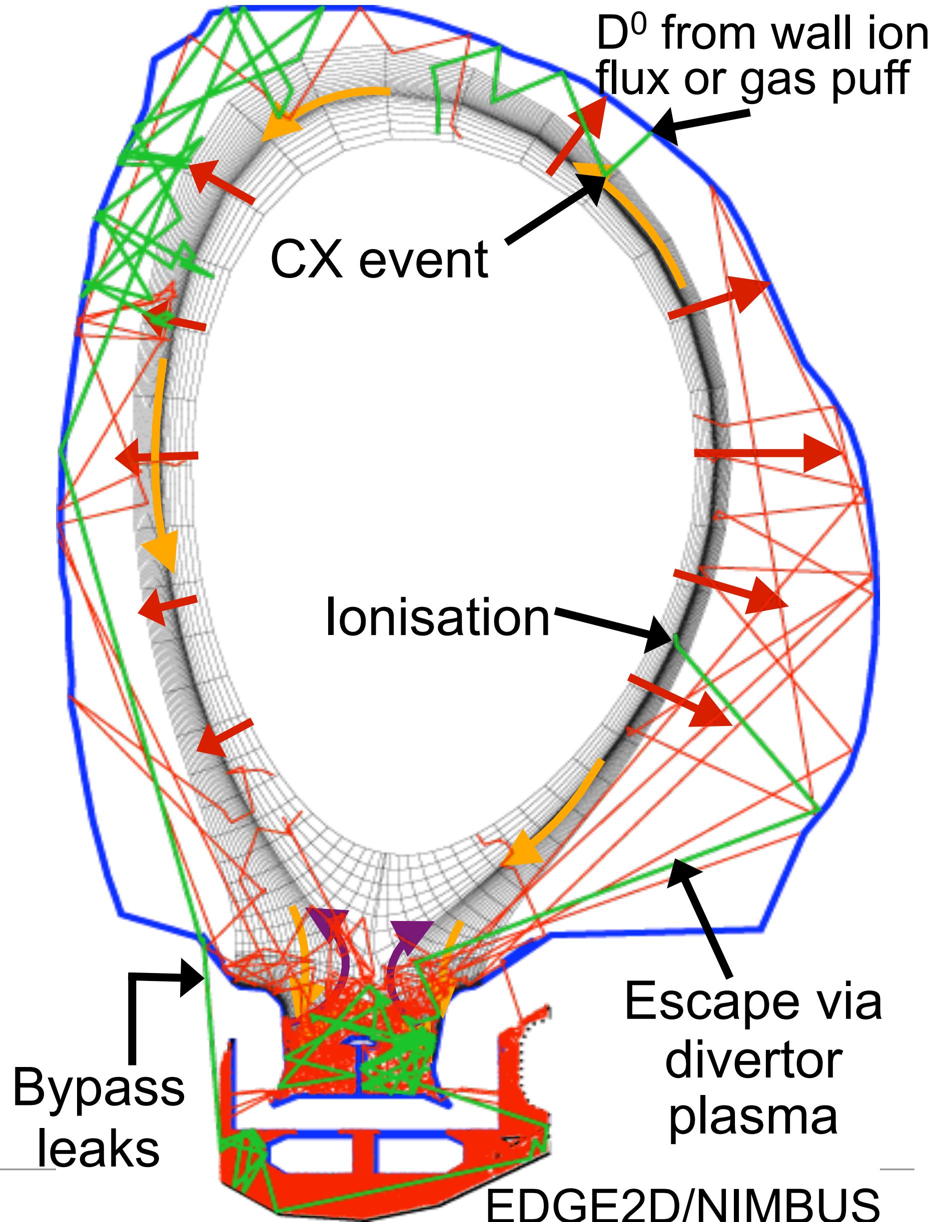
Cross-field transport – turbulent driven far SOL ion fluxes (slides 19-23)
→ recycled neutrals
→ direct impurity release
ELMs can also reach first walls

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



EDGE2D/NIMBUS

Impurity migration

Migration = Erosion $\Rightarrow$ Transport $\Rightarrow$ Deposition

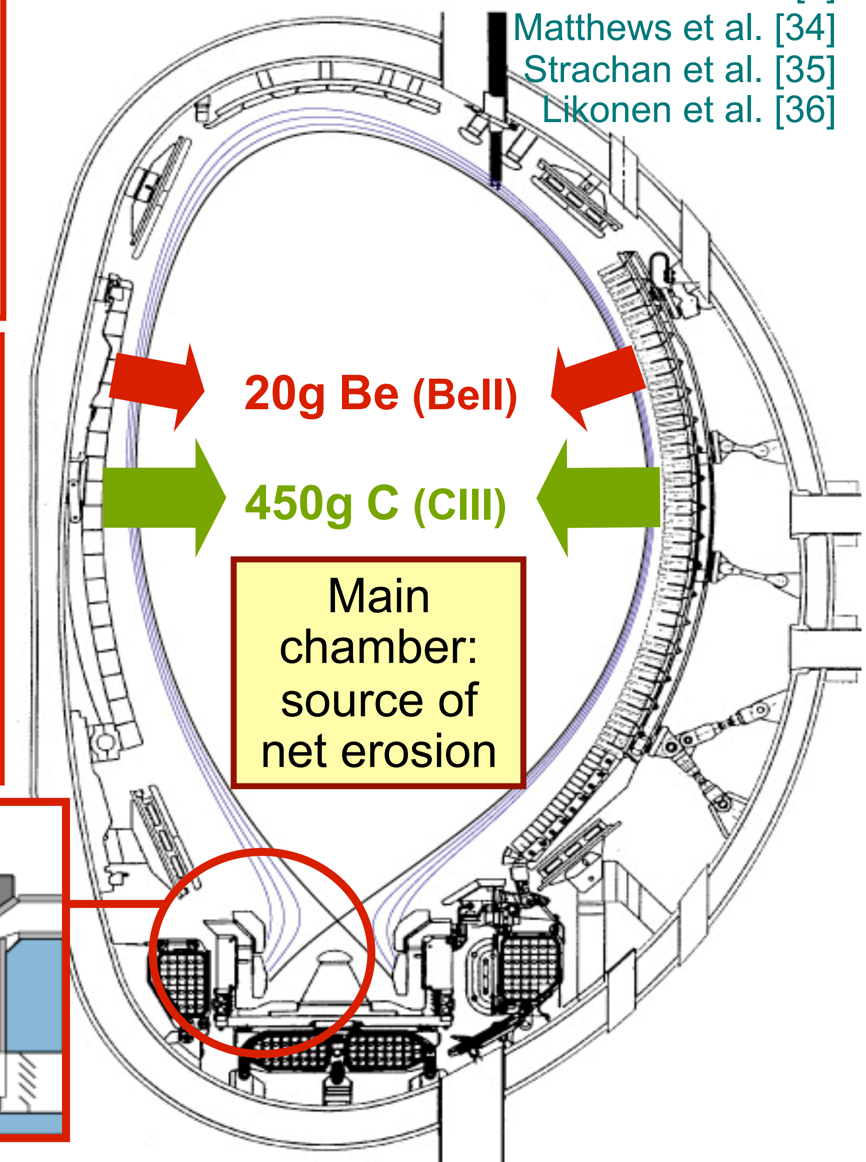
Re-erosion

Migration balance – example from JET

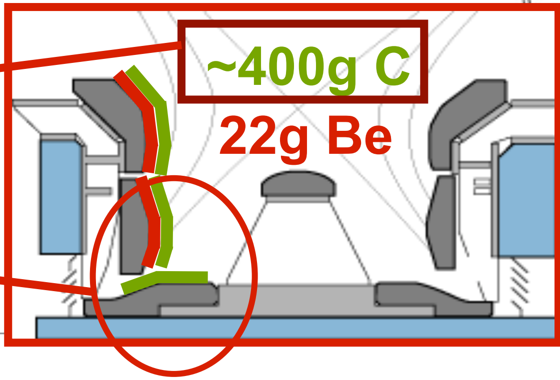
- Make balance for period 1999-2001 with MarkIIGB divertor: 14 hours plasma in diverted phase (50400 s, 5748 shots)
- Use spectroscopy and modelling to estimate main chamber sources

- 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)

Pitts et al. [4]
Matthews et al. [34]
Strachan et al. [35]
Likonen et al. [36]



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

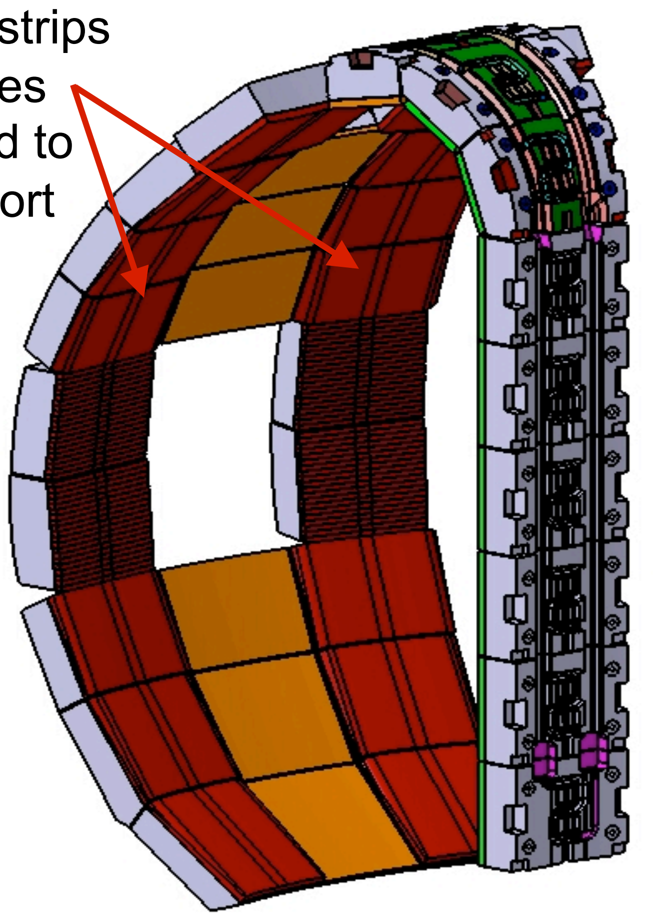


The problem of net deposition

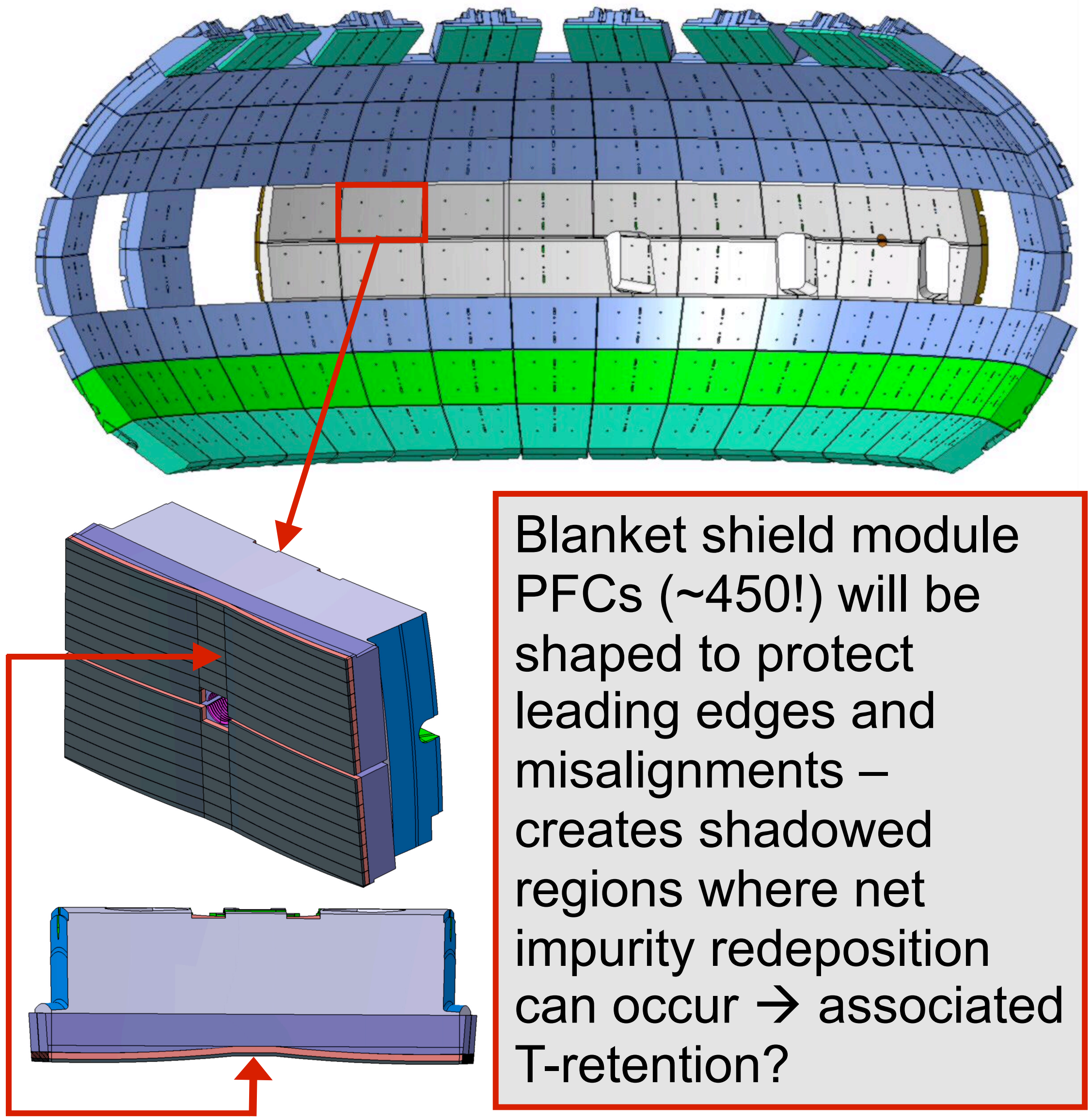
- Material deposition is always accompanied to some extent by retention of tritium → a critical problem for ITER since inventories are limited by nuclear licensing (see slide 44)
- Material deposition creates layers which can become unstable or be destroyed by transient power loads (ELMs, disruptions) → creation of dust → also a safety issue (see slide 47)
- General picture from today's divertor tokamaks is of erosion from the main walls (CX fluxes, long tails in the steady state ion fluxes, ELMs) → entrainment by flows in the SOL → deposition in the divertor region (slides 39, 41)
- But will it really be the case in ITER?

Wall deposition?

Poloidal strips of modules advanced to protect port regions



Recessed area for remote handling – protected against plasma flowing along field lines at low attach angles



Blanket shield module PFCs (~450!) will be shaped to protect leading edges and misalignments – creates shadowed regions where net impurity redeposition can occur → associated T-retention?

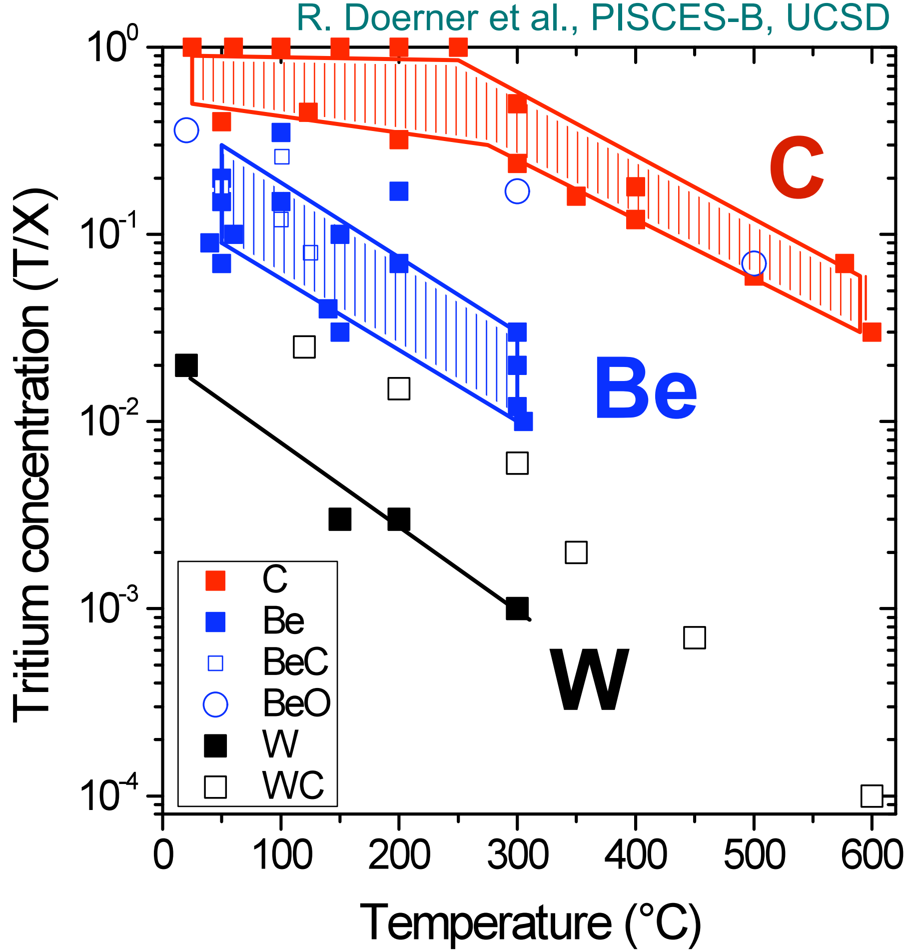
Tritium retention (1)

- 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 [37]
  - 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 – still not completely clear how important it will be

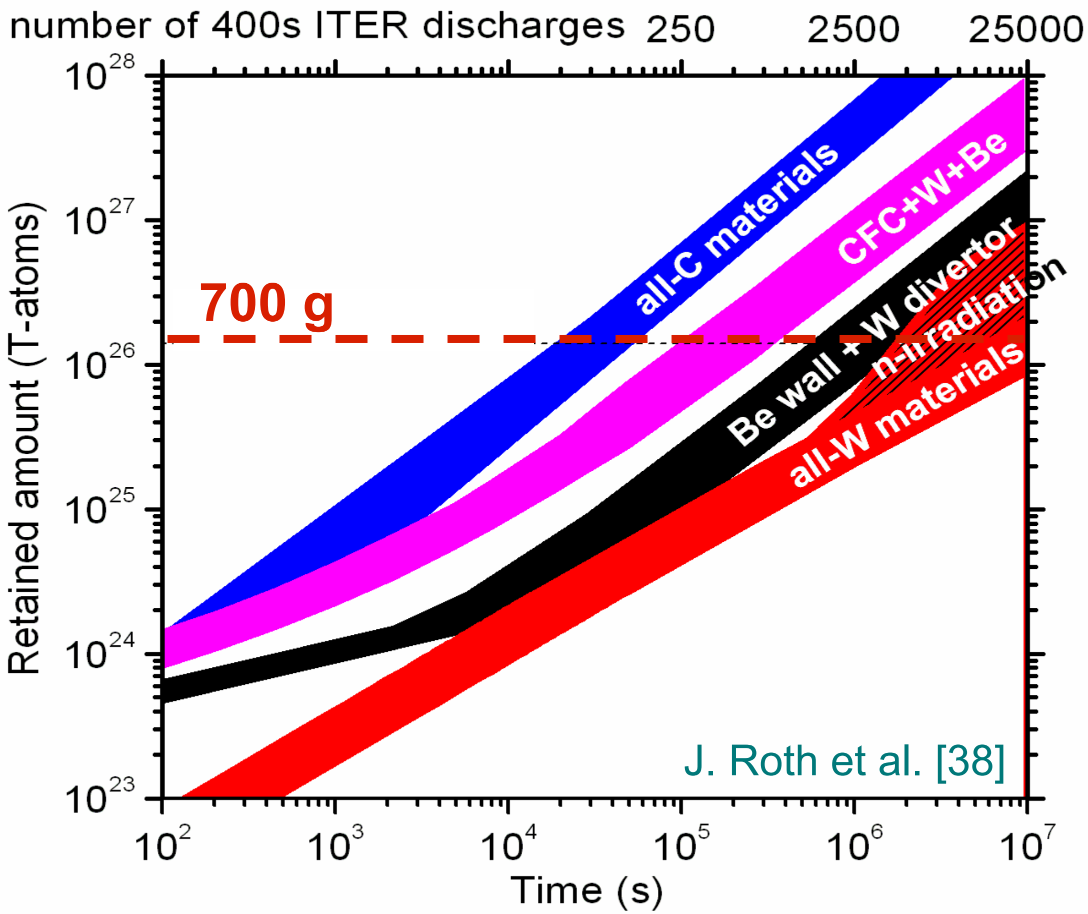
R. Doerner et al., PISCES-B, UCSD



Co-deposition with C and Be depends sensitively on deposition rate, incoming particle energy, surface temperature [38]

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

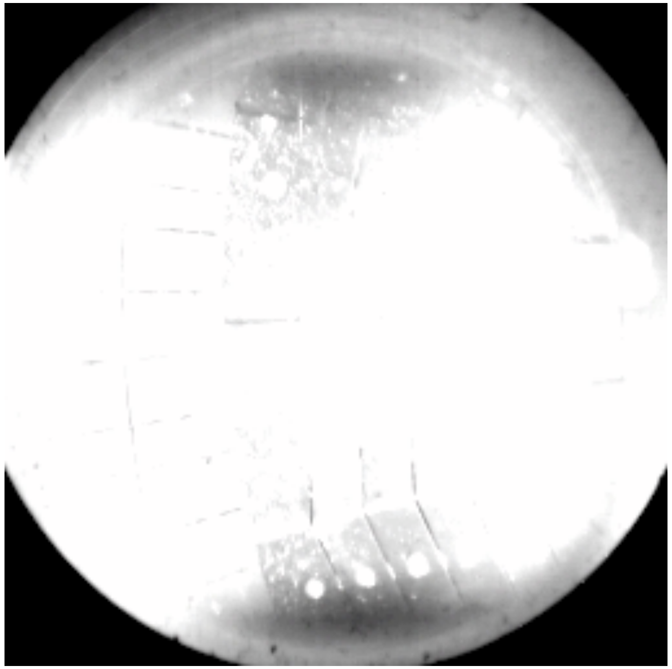
Dust – why worry?

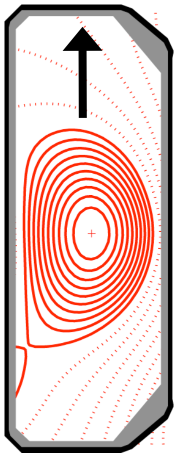
- Expectation is that increase in duty cycle and erosion in ITER will lead to large scale-up in quantity of dust particles produced
- Like T-retention, dust is a safety issue [37,40]
  - dust particles radioactive (tritium + activated metals)
  - potentially toxic (Be)
  - potentially responsible for 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

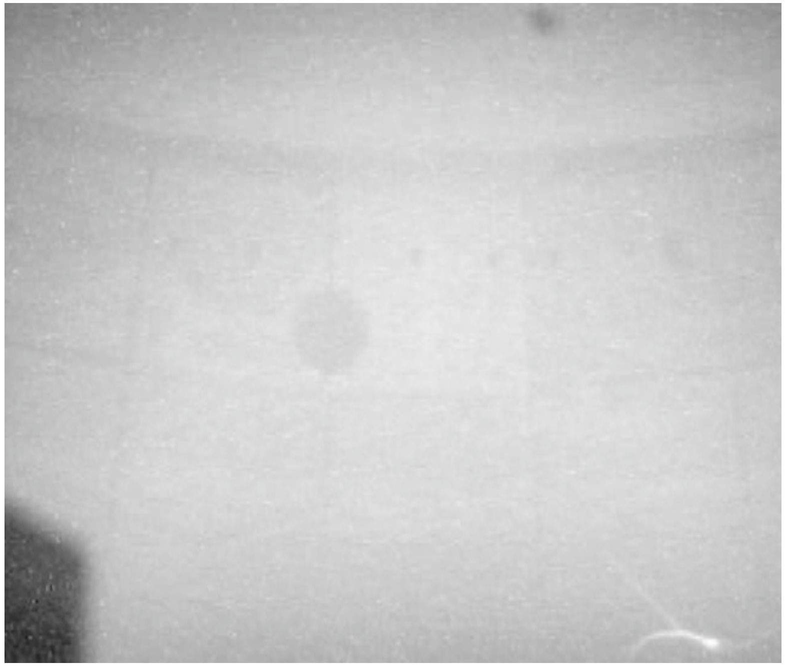
- Dust is seen in all tokamaks, especially with C walls, but most often in first plasmas after long vent, or after disruptions when plasma touches surfaces not normally in contact with high heat/particle flux – not usually an operational issue
- First papers to recognize the potential importance more than 10 years ago [41]

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 & W. P. West [42]



ITER Dust – safety inventory limits

- Global quantity in the vacuum vessel (VV) – 1 tonne durring D-D and D-T operation [37]
- 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: $T_{surf} > 400°C$ (Be or W), $T_{surf} > 400°C$ (Carbon)
  - 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
- The real dust inventory will be reduced by measurement uncertainties (estimated to be about 30%)

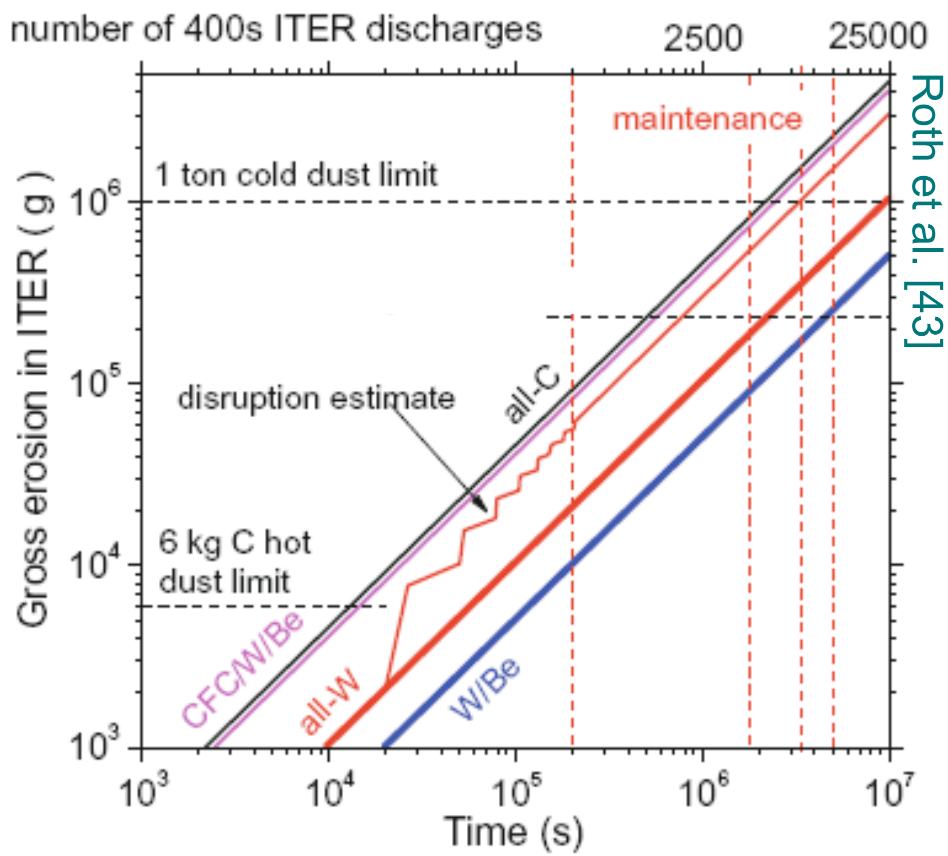
ITER dust: how much?

Assume

Dust generation dominated by erosion → migration/deposition → layer disintegration

Conversion factor $f_{dust} = 1$ from erosion to dust → extremely conservative, for safety reasons

Today's experiments typically have $f_{dust} \sim 0.1$ but this is a challenging measurement



On the basis of these estimates, dust safety levels will not be exceeded before currently scheduled maintenance or divertor cassette exchanges
Still many uncertainties though, e.g. the fraction of dust residing in hot areas (castellation gaps)

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] "Plasma-material interactions in current tokamaks and their implications for next step fusion reactors", G. Federici et al., Nucl. Fusion **41** (2001) 1967

[7] "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.

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

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

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

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

References (2)

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

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

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

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

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

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

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

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

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

[23] "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

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

[25] "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)

[26] "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

[27] W. Eckstein et al., IPP Garching report number 9/82 (1993)

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

[29] "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

[30] "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)

References (3)

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

[32] "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

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

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

[35] "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

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

[37] "An integrated approach to in-vacuum vessel dust and tritium inventory control in ITER", S. Ciattaglia et al., 25th SOFT Conference, Rostock, Germany, Sept., 2008

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

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

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

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

[42] "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

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

Reserve slides

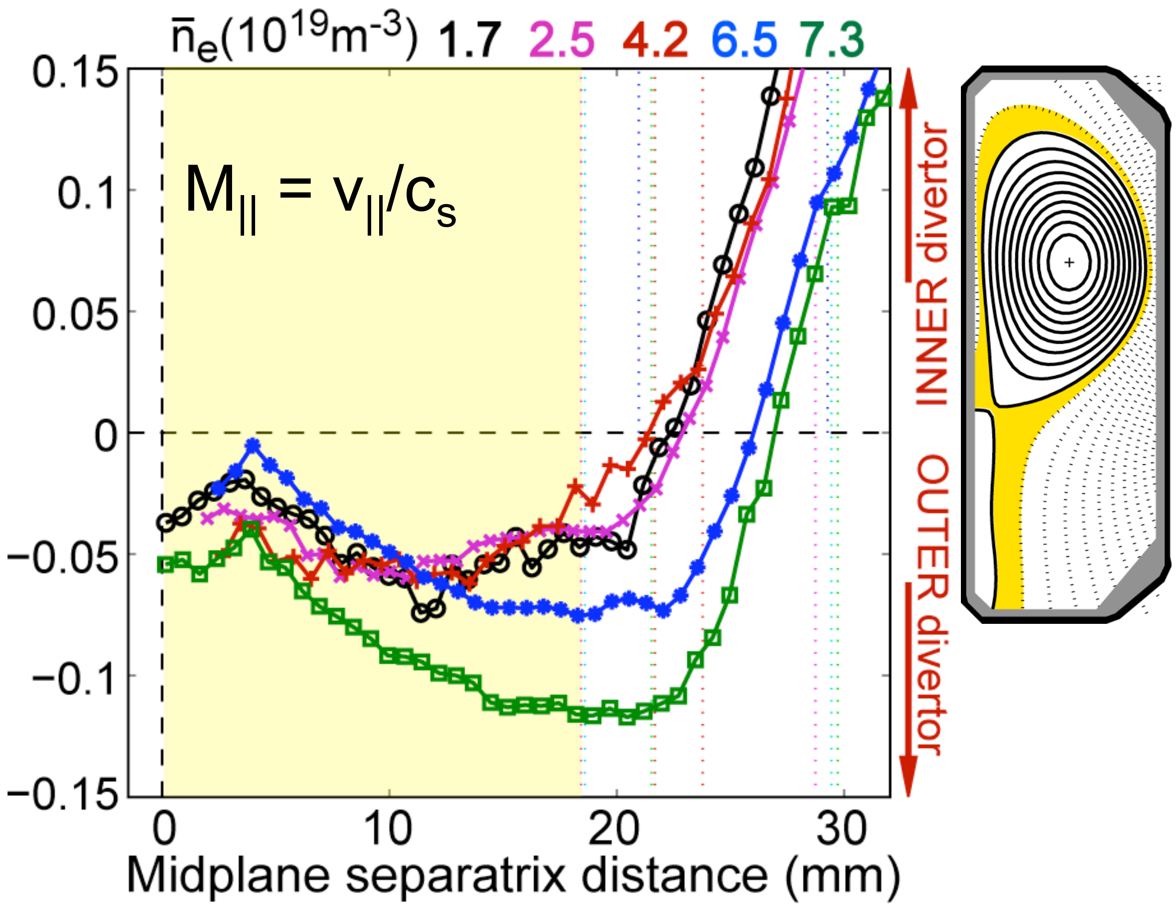
Parallel flow offset

Take mean of flows for 2 field directions → reveal any field independent components

Find $M_{||}$ ~0.05 – 0.1 (up to 10 kms^{-1})

It is this component which can drive the largest impurity migration (the Pfirsch-Schlüter flows generally close on themselves)

Have shown on TCV that interchange turbulence can account for this transport driven component [19,24]



ITER dust strategy

- To satisfy nuclear safety regulators, a strategy for dust management has recently been adopted into the ITER Baseline
- Set of measurement and removal techniques for dust (and tritium) integrated into operation and maintenance programme to maximise machine availability with minimum impact on design
  - Measure dust inventory from time to time and remove when limits approached
  - To meet ALARA safety criterion (As Low As Reasonably Achievable), removal to be performed as completely as possible – reduce radioactive inventories to minimum possible.
- Dust measurement based on global erosion measurements, dust monitors and sampling
- Increased divertor bakeout temperature (350°C) to remove T from Be codeposits
- Partial dust removal during VV vents and almost complete removal during divertor cassette changeout
- Use R&D during construction, plasma operations in H and D phases to validate techniques, refine physics model (reduce uncertainties), optimize techniques

ITER dust – how much?

- DT phase (i.e. Be wall, W divertor):
  - Assume ~65 g "steady-state" erosion per pulse
    - Due to ELMs and continuous background ion flux
    - Main chamber erosion → Be deposition in divertor → erosion → ~130 kg/2000 pulses
  - Assume ~5 kg "steady-state" erosion per major disruption
    - W from the lower divertor (the majority) + Be from the machine top (~5%) – assuming 10% melt layer loss and ~40 m^2 affected area in both locations
    - ~25 major disruptions per 2000 shot campaign → ~130 kg
  - Assume dust conversion factor = 1
    - All erosion converted to mobilisable dust – very conservative
  - Assume VV opening every ~2000 pulses (~15-16 months)
    - Use vacuum cleaning introduced with baseline Remote Handling tools to access as much of divertor as possible (facilitated by new, more open divertor design) – hope to recover at most ~15% dust
  - Assume divertor replacement every ~6000 pulses
    - Complete recovery of dust – assume ~30-50 kg non-recoverable
  - Assume total uncertainty of ~1/3 on 1 tonne limit
    - Mobilisable dust inventory maintained below ~670 kg – uncertainty based on assumed uncertainty on erosion measurement.

Erosion: ELM size must be mitigated in ITER

Materials tests (QSPA and MK-200UG plasma gun facilities in RF) show that CFC PAN fibre erosion and W surface cracking do not occur for energy densities $\leq$ 0.5 MJm^{-2} for ~500 μs, triangular pulse lengths (250 μs up and down). How does this convert to an ITER tolerable ELM size [25]?

What do we know about ELMs at the divertor target?

1) Rise time to peak $T_{surf} \sim \tau_{||}$ [26] ($\tau_{||} = c_s/L \sim 250\ \mu s$ for ITER)
2) Toroidal peaking factor ~ 1 [27]
3) $\lambda_{q,ELM} \sim \lambda_{q,inter\text{-}ELM} \sim 5$ mm (ITER)
4) $A_{\perp,IN} = 1.4\ m^2$, $A_{\perp,OUT} = 1.9\ m^2$
5) No ELM energy to main walls
6) Strong in-out asymmetry in ELM power loading (not understood yet)

$$\Delta W_{ELM} = q_{ELM} \times A_{\perp,in} \times (1 + E_{out}/E_{in})$$

$$= 0.5\ MJ/m^2 \times 1.4\ m^2 \times 1.5 \sim \mathbf{1\ MJ}$$

This is only ~0.3% of plasma stored energy for ITER Q_{DT} = 10 baseline scenario!

Natural Type I ELMs this small do not exist

And note that ELMs have a size distribution!

→ Mitigation techniques will be required in ITER (e.g. pellet pacing (see talk by P.Lang), active ergodisation coils [30, 31])