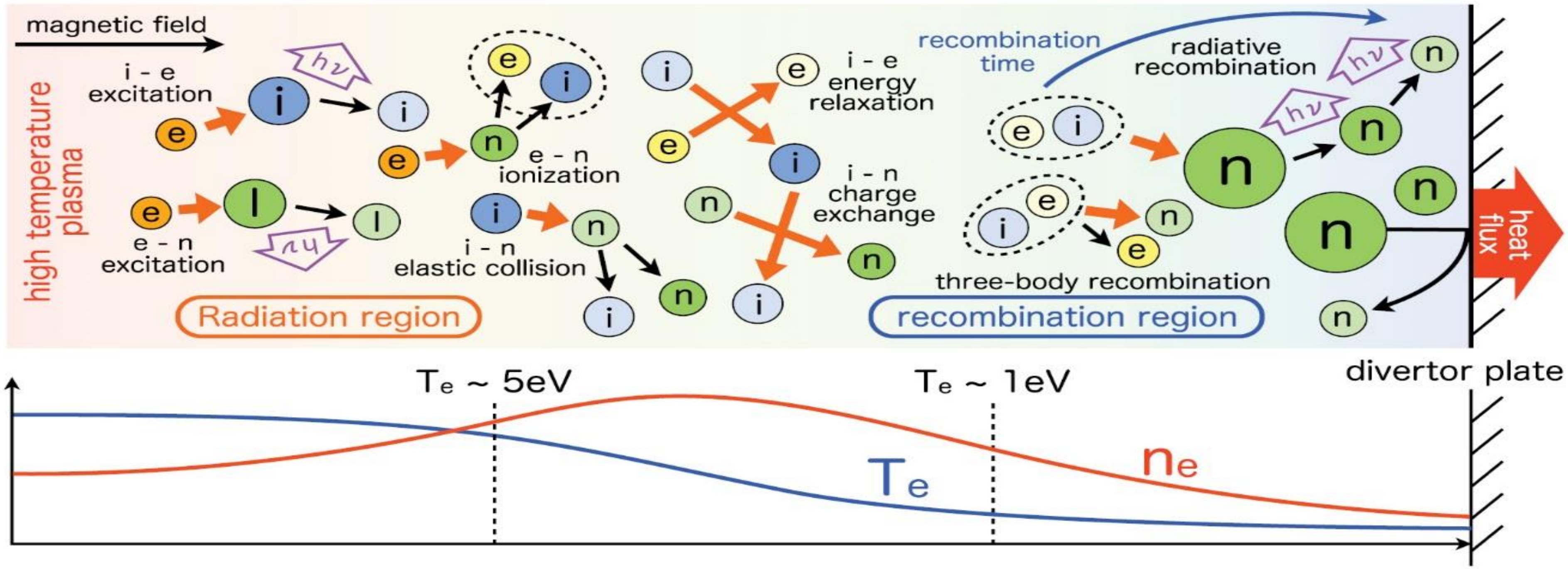

BASIC FUSION BOUNDARY PLASMA PHYSICS

Plasma Surface Interactions

JANUARY 2019 D. REITER

10th ITER international school 2019
The physics and technology of power flux handling in tokamaks. January 21st to 25th KAIST, Daejeon, South Korea

Mitglied der Helmholtz-Gemeinschaft

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Fusion boundary plasma physics:

Like in any applied science: Three questions

I.) WHAT: …happens? Plasma flows, chemistry, wall interaction processes,…

II.) HOW: …can we make the application work? → build ITER

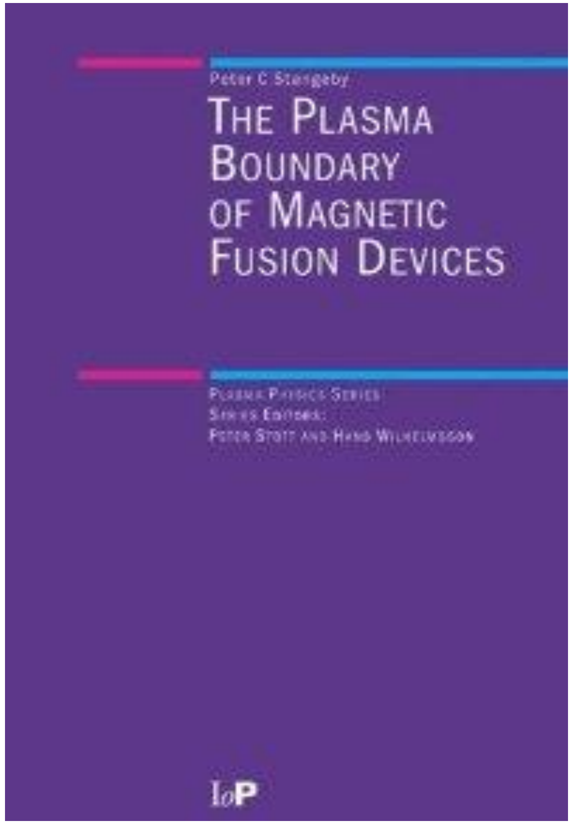
III.) WHY: understanding the "WHAT" (boundary plasma),
based on theory, simple modelling, and
computational bookkeeping (complex codes) of many "basic processes"

Today: we know enough about "What" happens to proceed to "the "How" question (build ITER). Very little still on the "Why" question.
Todays lecture: I and II, in a nutshell

Reading…

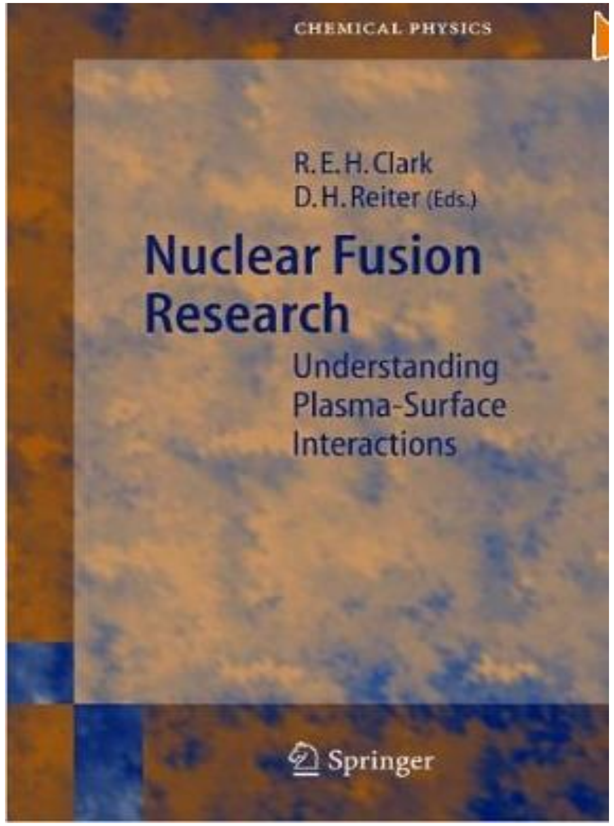
P.C. Stangeby

(IoP, 2000):



Golden standard text book,
boundary plasma science

R. Clark / D. Reiter (eds)
IAEA decennial reviews
(the last one so far: 2014, Daejeon
(Springer series: chemical physics, 2005):



specialized topics,
review articles by experts

Listening…

3.5 hours DETAILED podcast interview with Richard Pitts (ITER-IO): "ITER and Fusion: explained in context"

http://omegataupodcast.net/download-archive/
episode 157-fusion-at-iter/

This lecture is built upon a tutorial by

Bruce Lipschultz, University of York, UK

THE UNIVERSITY of York

\+ some lecturing material, talks, … from

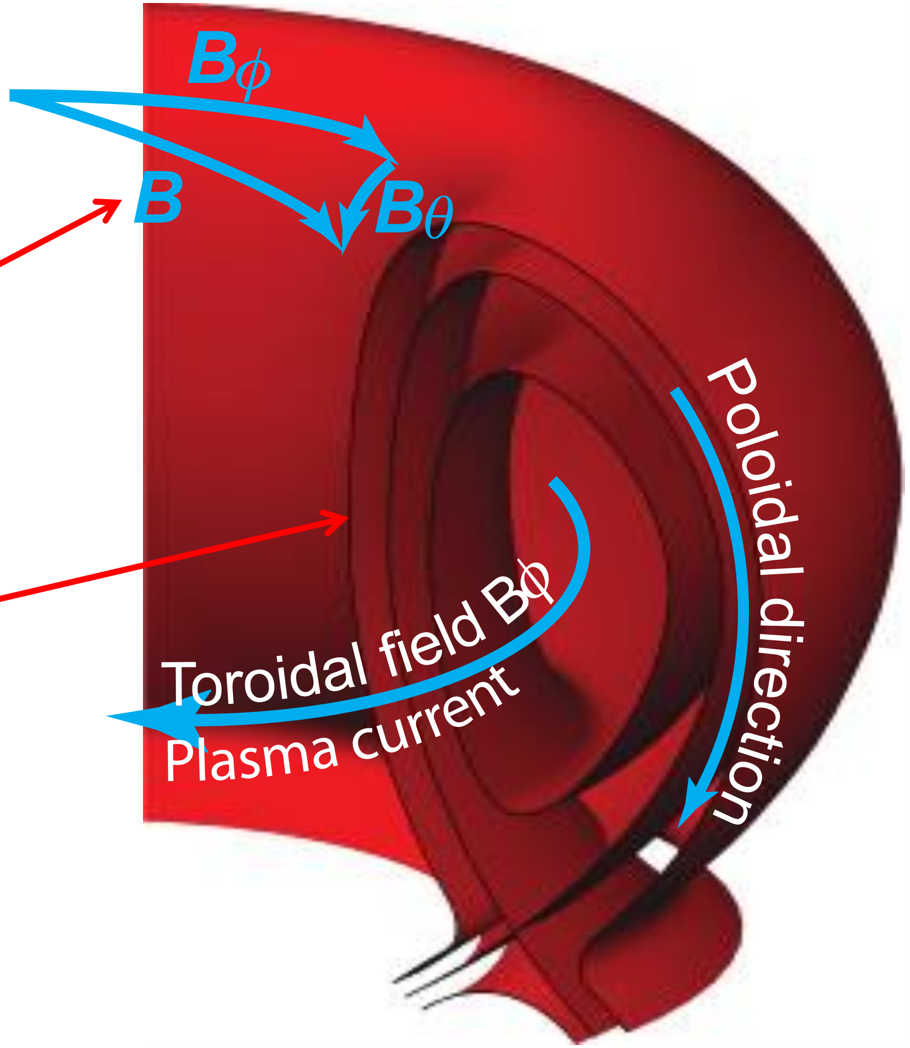
e.g. S. Brezinsek, R. Pitts, W. Fundamensky, and many more

Basic fusion boundary plasma physics and plasma-surface interactions

- Review tokamak geometry
- Heat exhaust challenge in a tokamak: the divertor
  How divertor physics helps reduce/spread the heat flux
- Reactor requirements for erosion rates
  Role of divertor physics in reducing that erosion
- Tritium retention inside the tokamak (…if time permits…)
  Processes and how to control it
- Summary

Tokamak terminology and topology

- Two primary directions
  - Toroidal (ϕ) – also direction of plasma current which makes a poloidal field
  - Poloidal (θ)
- Total field, B, dominated by B_ϕ, loops helically around the plasma (poloidally and toroidally)
- $B_\theta/B \sim 0.1$ in tokamaks
- Surfaces of constant poloidal field flux are formed
- **The boundary plasma is that region, in which the plasma (dynamics, composition, etc..) and the vessel components are directly and strongly mutually affected**
- Today tokamaks are big enough to have a well separated **core plasma region** ⬌ mutual boundary conditions to boundary plasma.



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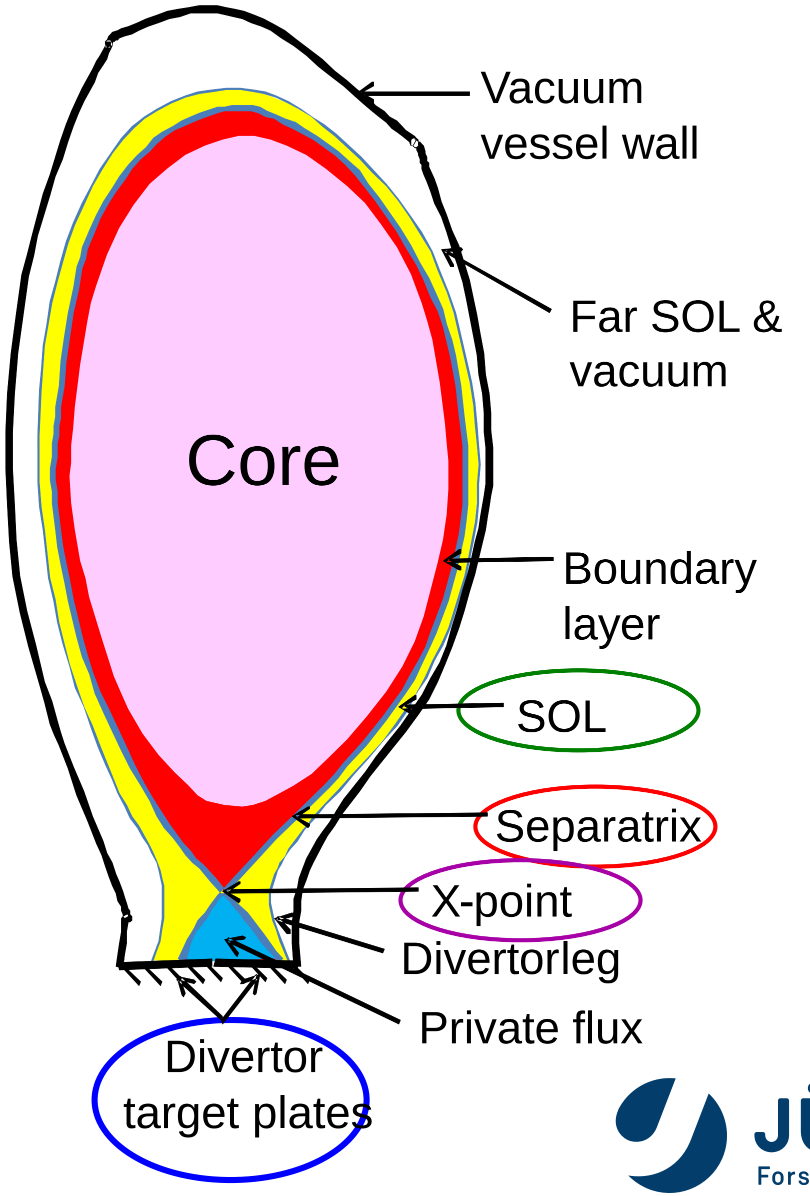
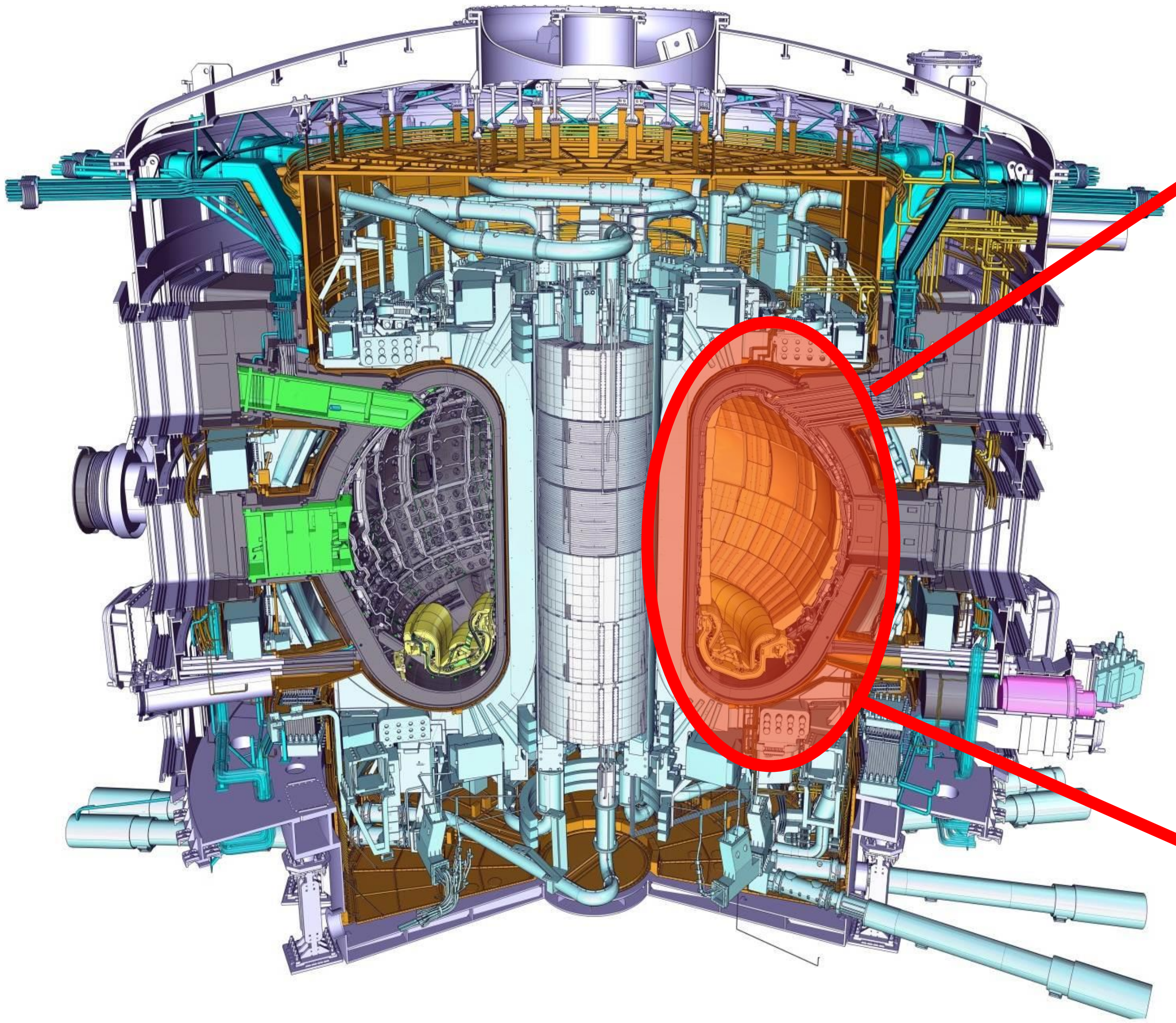
Tokamak terminology and topology, “core”, “boundary”

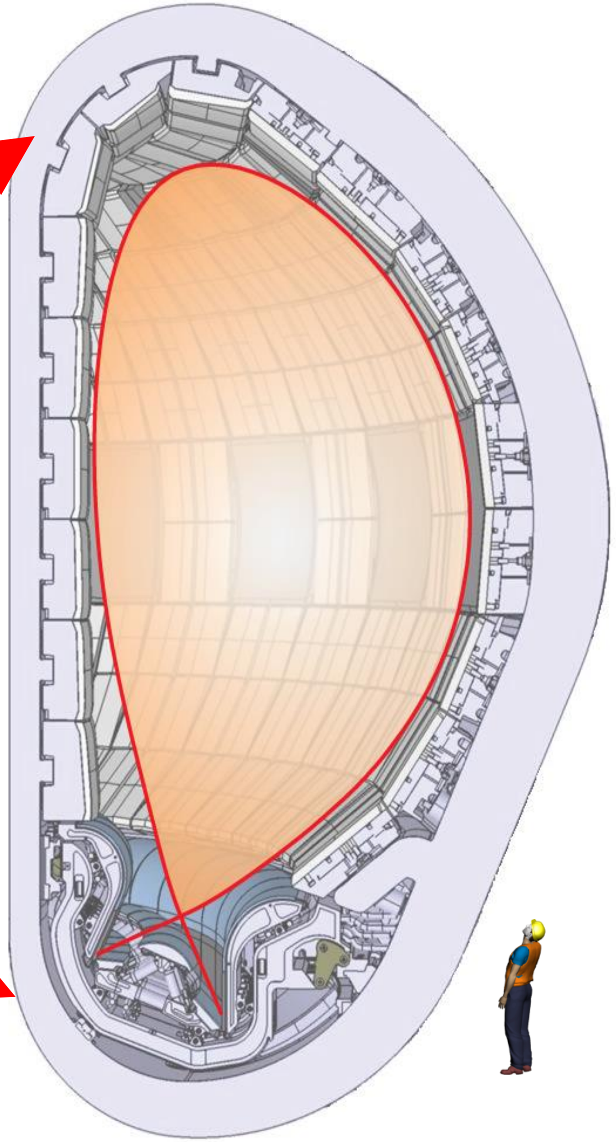
- Two primary directions
  - Toroidal (ϕ) – also direction of plasma current which makes a poloidal field
  - Poloidal (θ)
- Total field, dominated by B_ϕ, loops helically around the plasma (poloidally and toroidally)
- Surfaces of constant poloidal field flux are formed
  - Inside the **separatrix** field lines do not intersect material surfaces – the hot core plasma
  - Outside the separatrix – the Scrape-off layer (**SOL**) where field lines are routed to the **divertor target**
  - The poloidal field goes to zero at the **x-point**
  - (far SOL) plasma chemistry, vacuum region



ITER burning plasma power exhaust



Power balance



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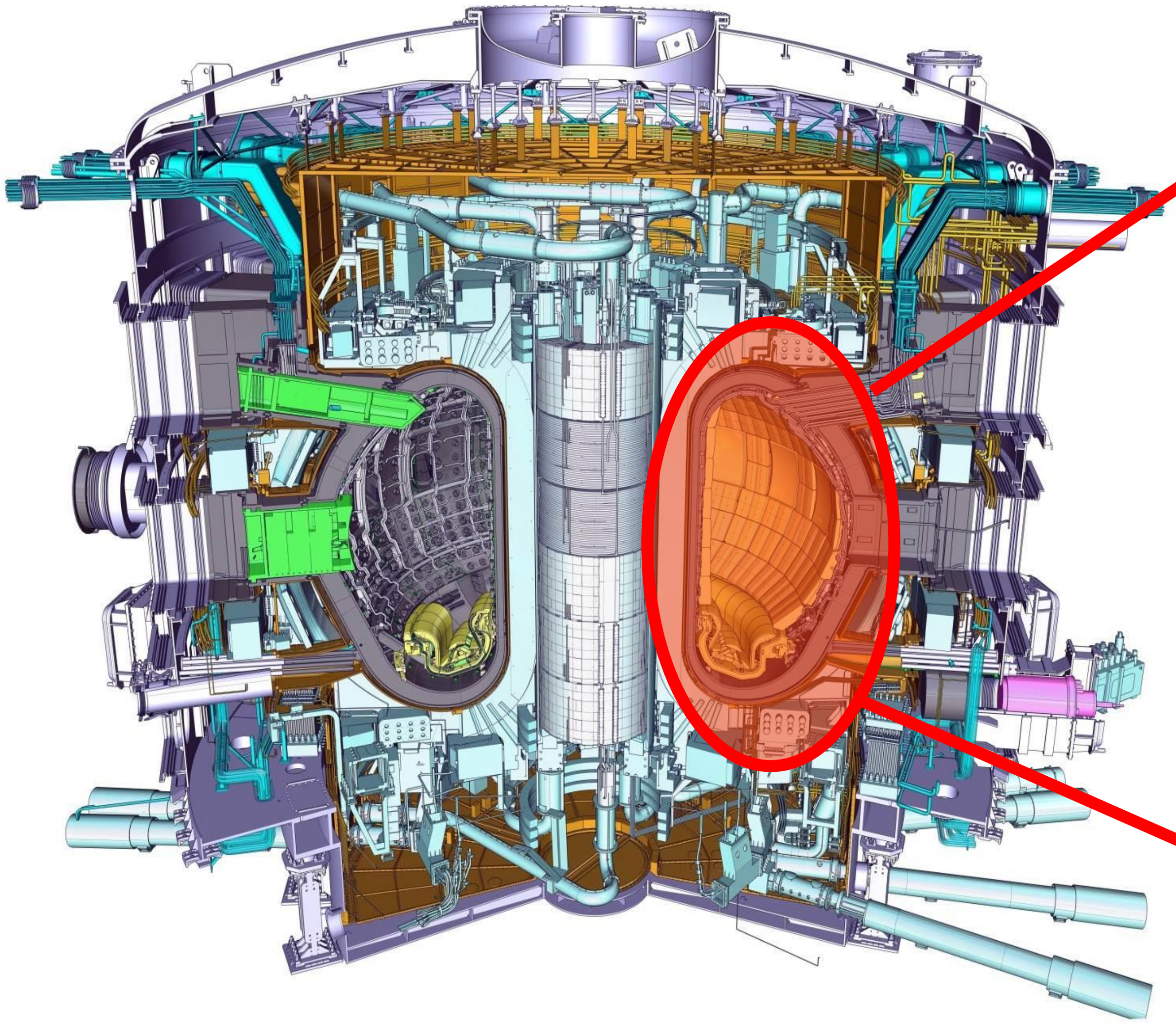
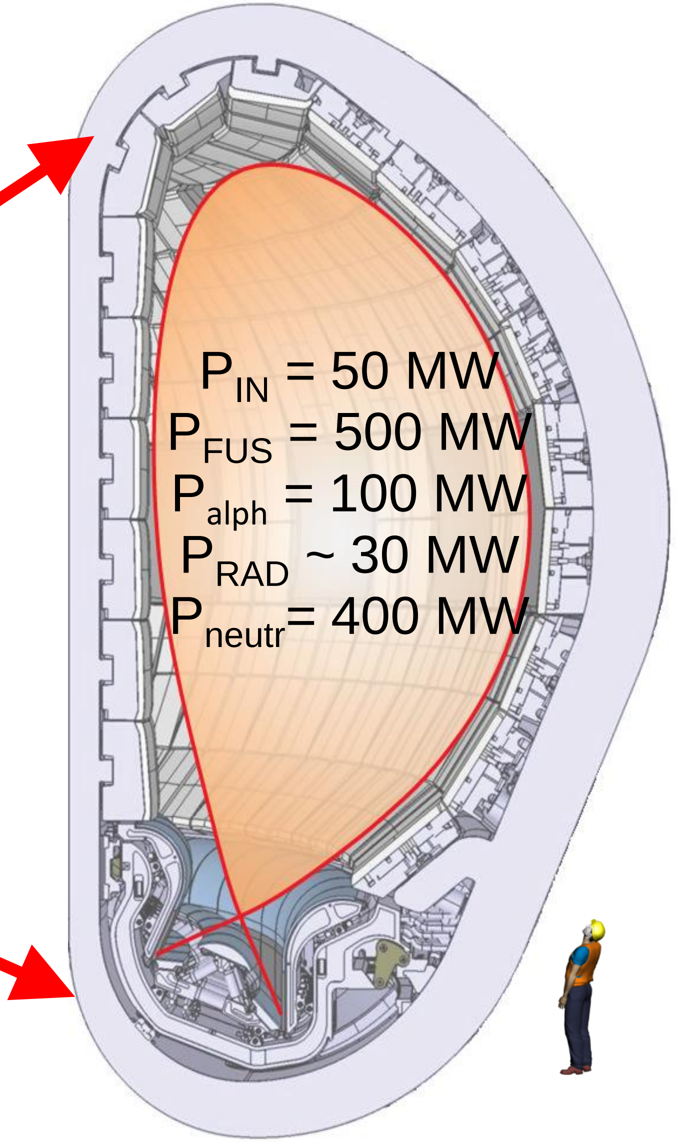
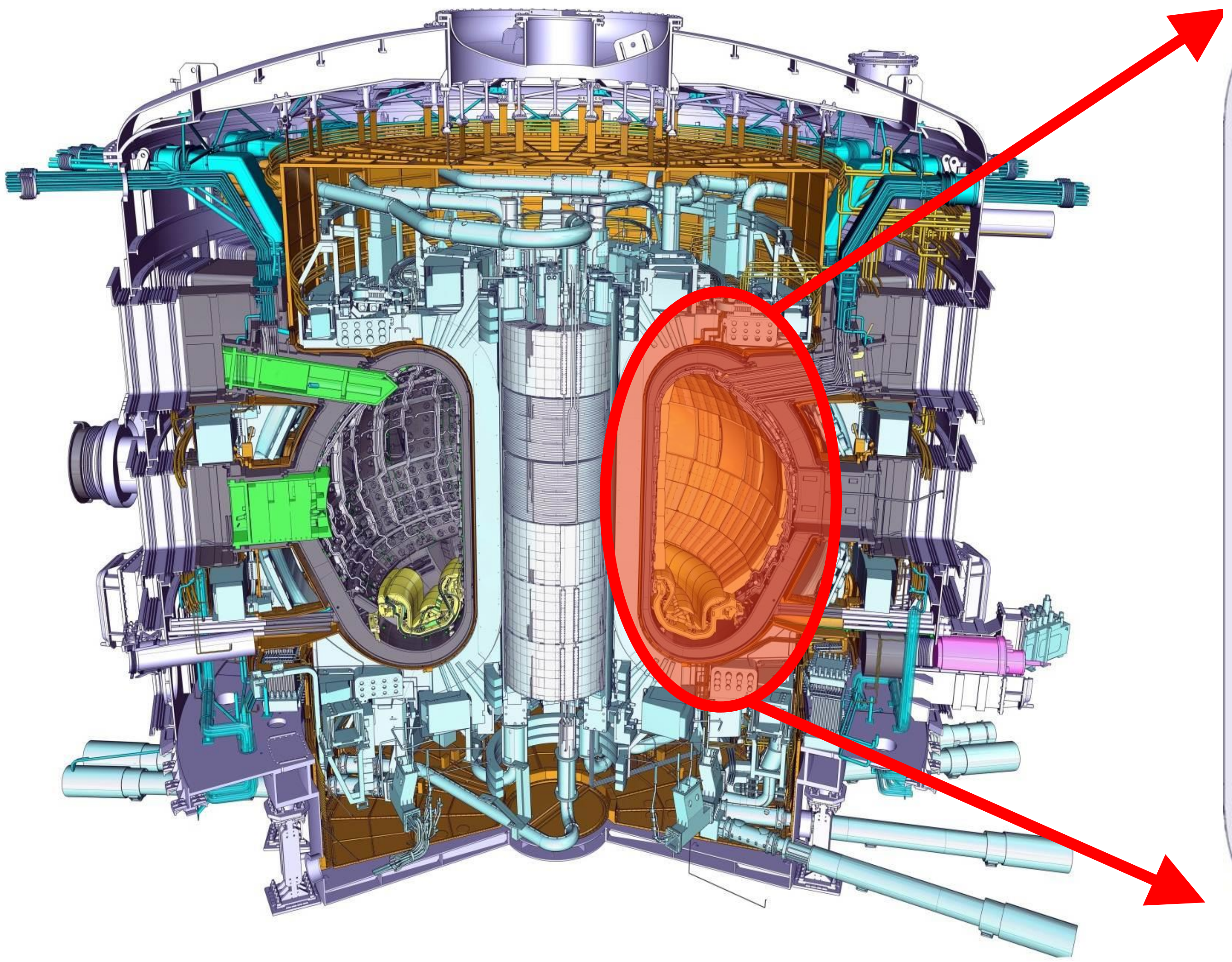
ITER burning plasma power exhaust



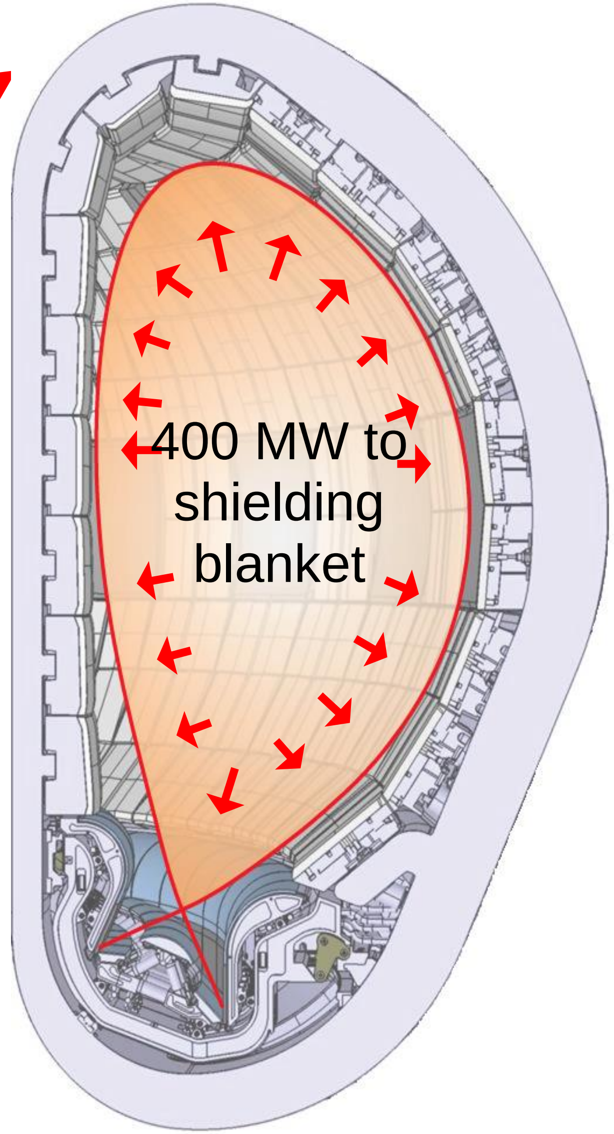
Power balance



ITER burning plasma power exhaust

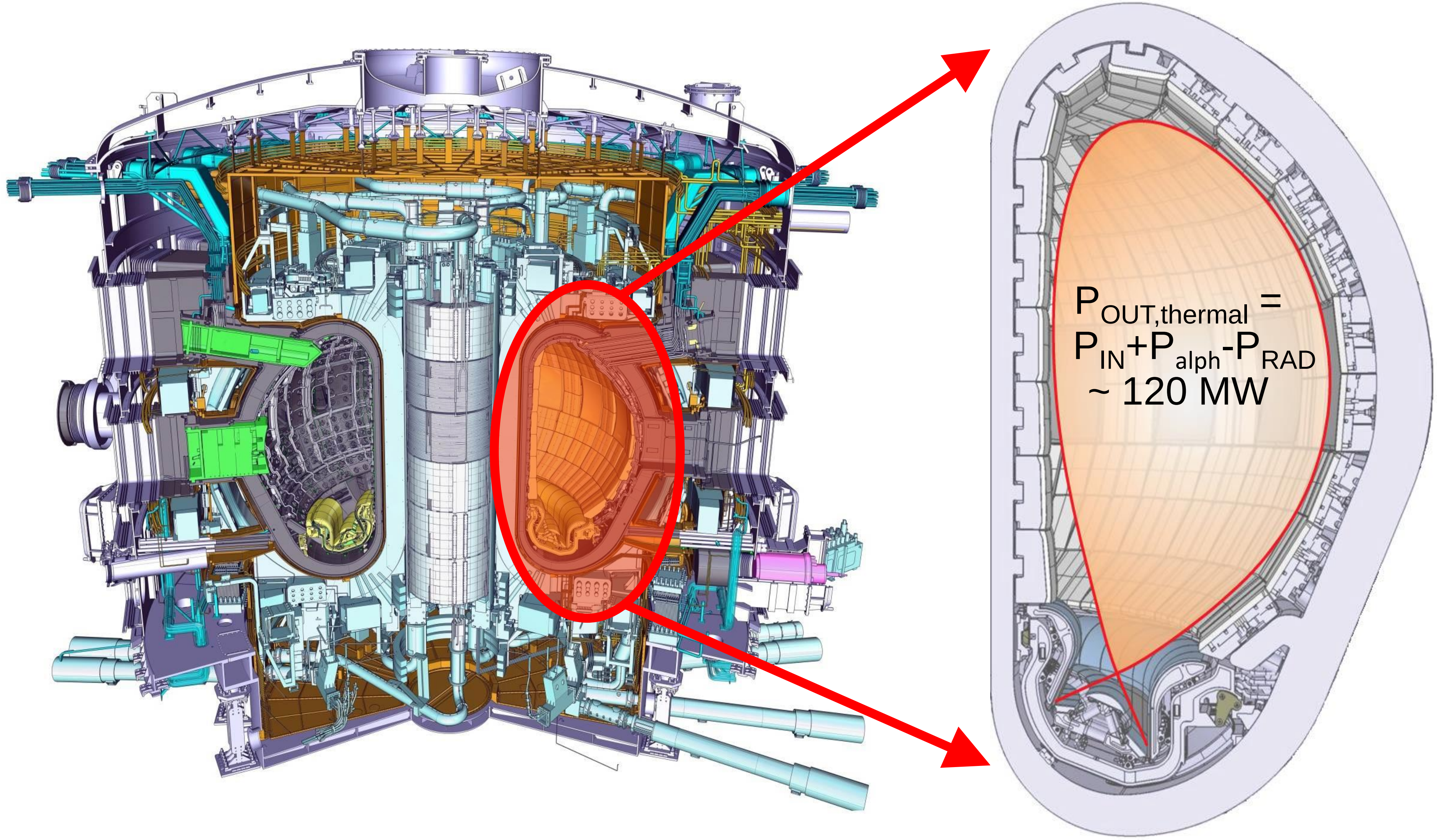


Power balance



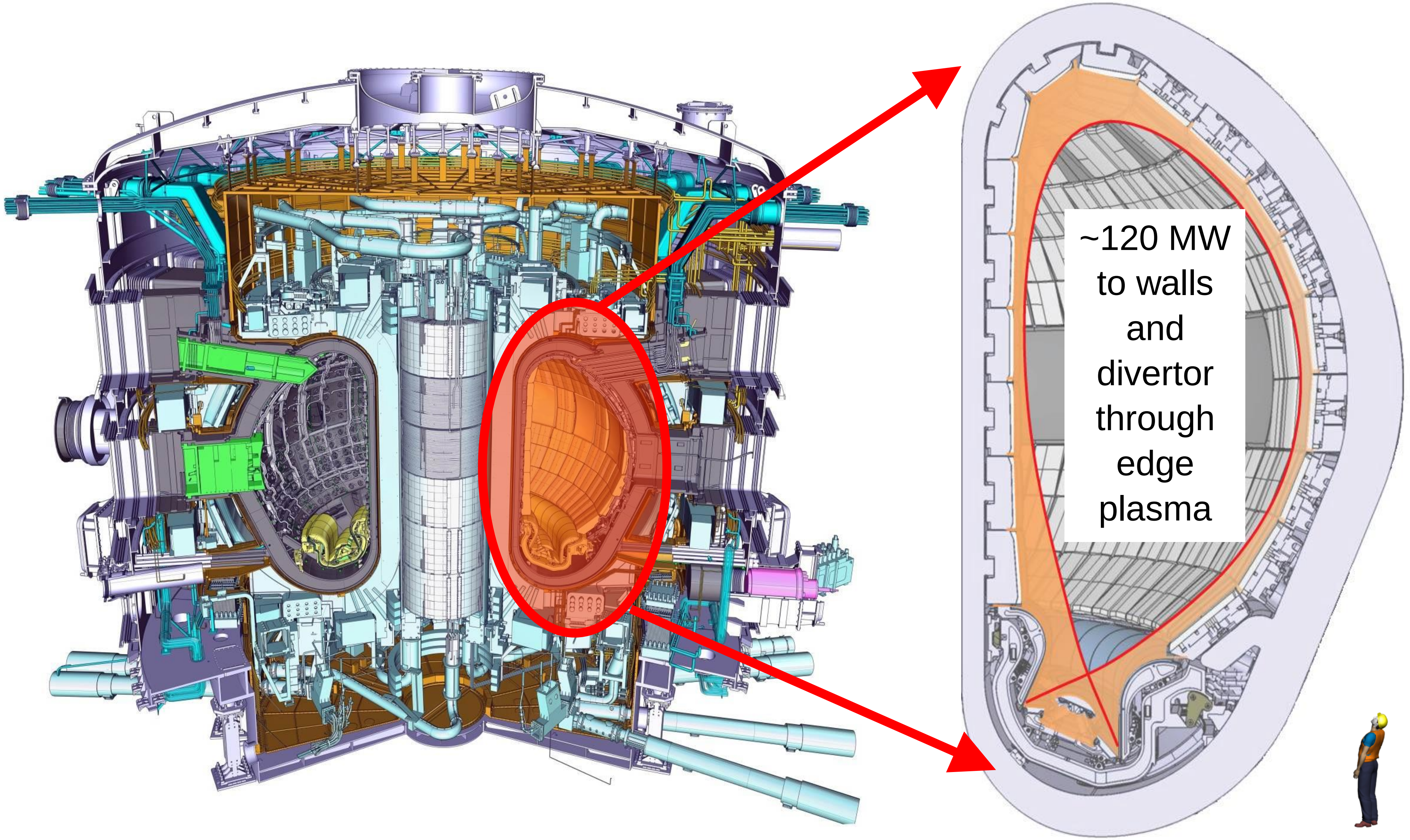
ITER burning plasma power exhaust

Power balance



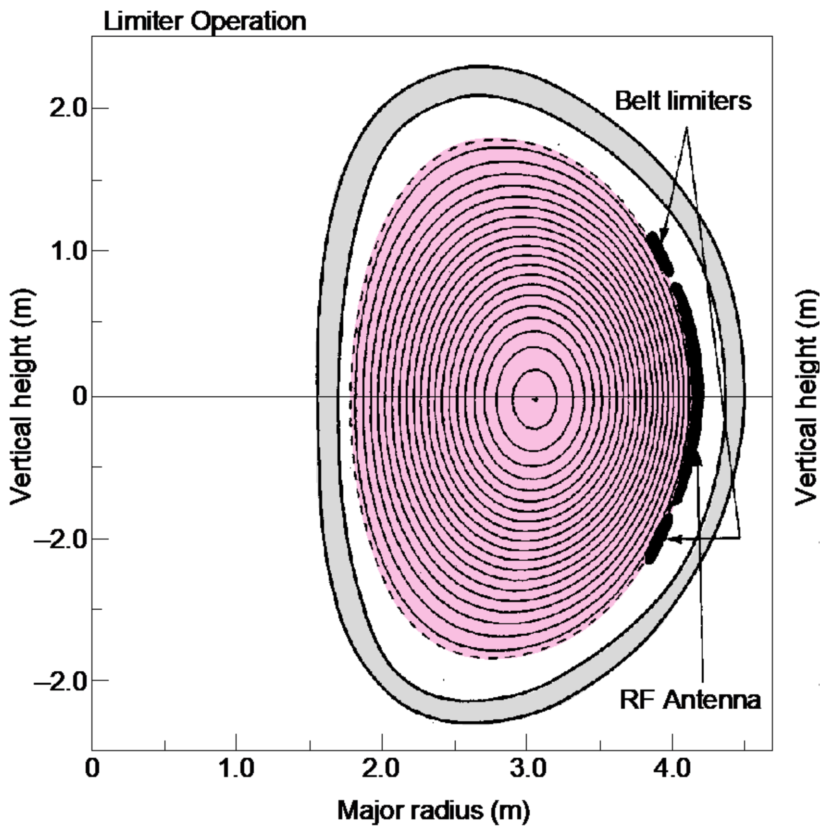
ITER burning plasma power exhaust

Power balance



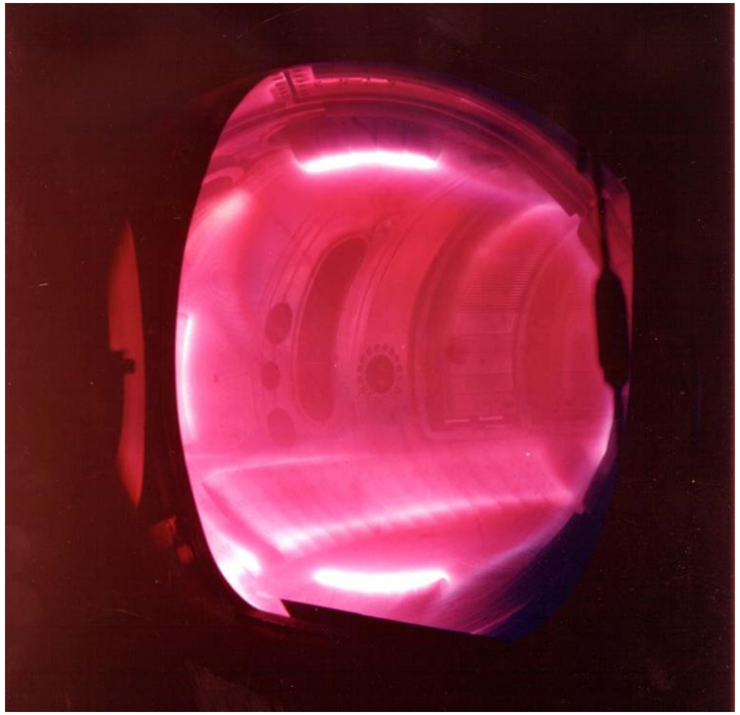
Divertor vs Limiter geometry

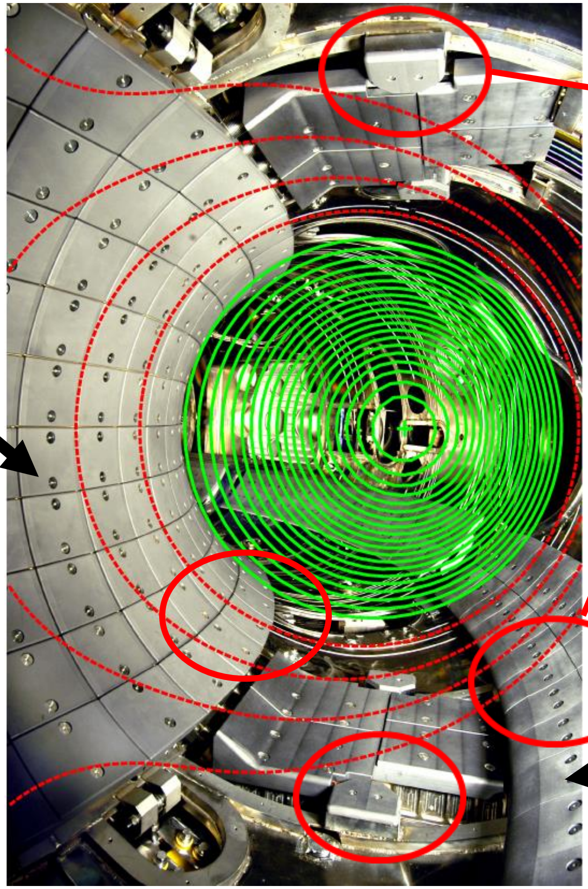
- Until ~ 1980 -1985 the heat flux handling structure was a **limiter** – an aperture that limits plasma size
  - Erosion rate high due to sputtering and evaporation
  - *Impurities immediately go inside the separatrix (confined plasma) – cooling that plasma through excitation radiation (high Z), and diluting the fuel (low Z)*



Limiter tokamak: example: TEXTOR, FZ Jülich 1983 - 2013

inner bumper
limiter/divertor
Hidden behind: DED
RMP coils



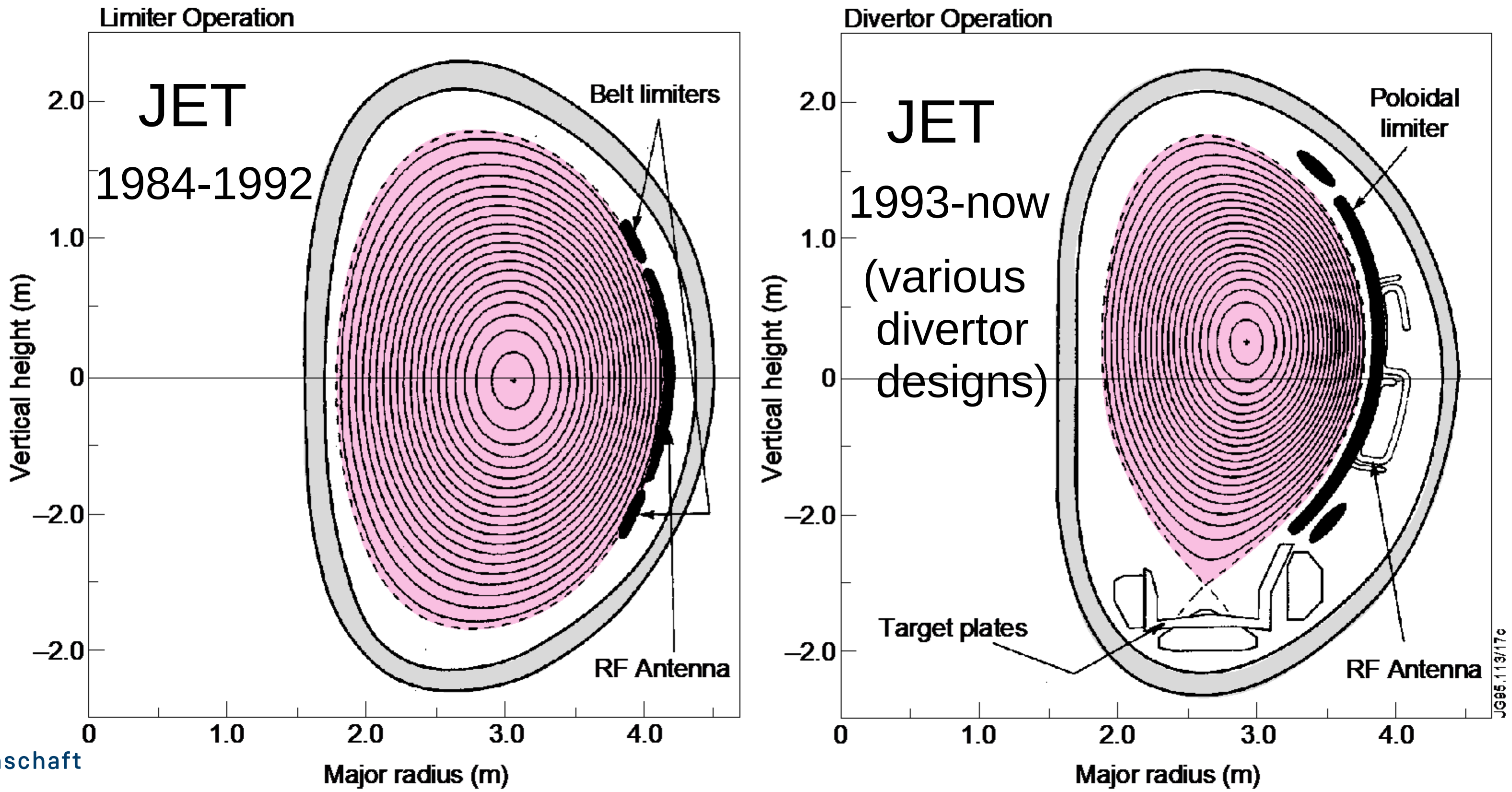


Areas of dominant
“recycling”

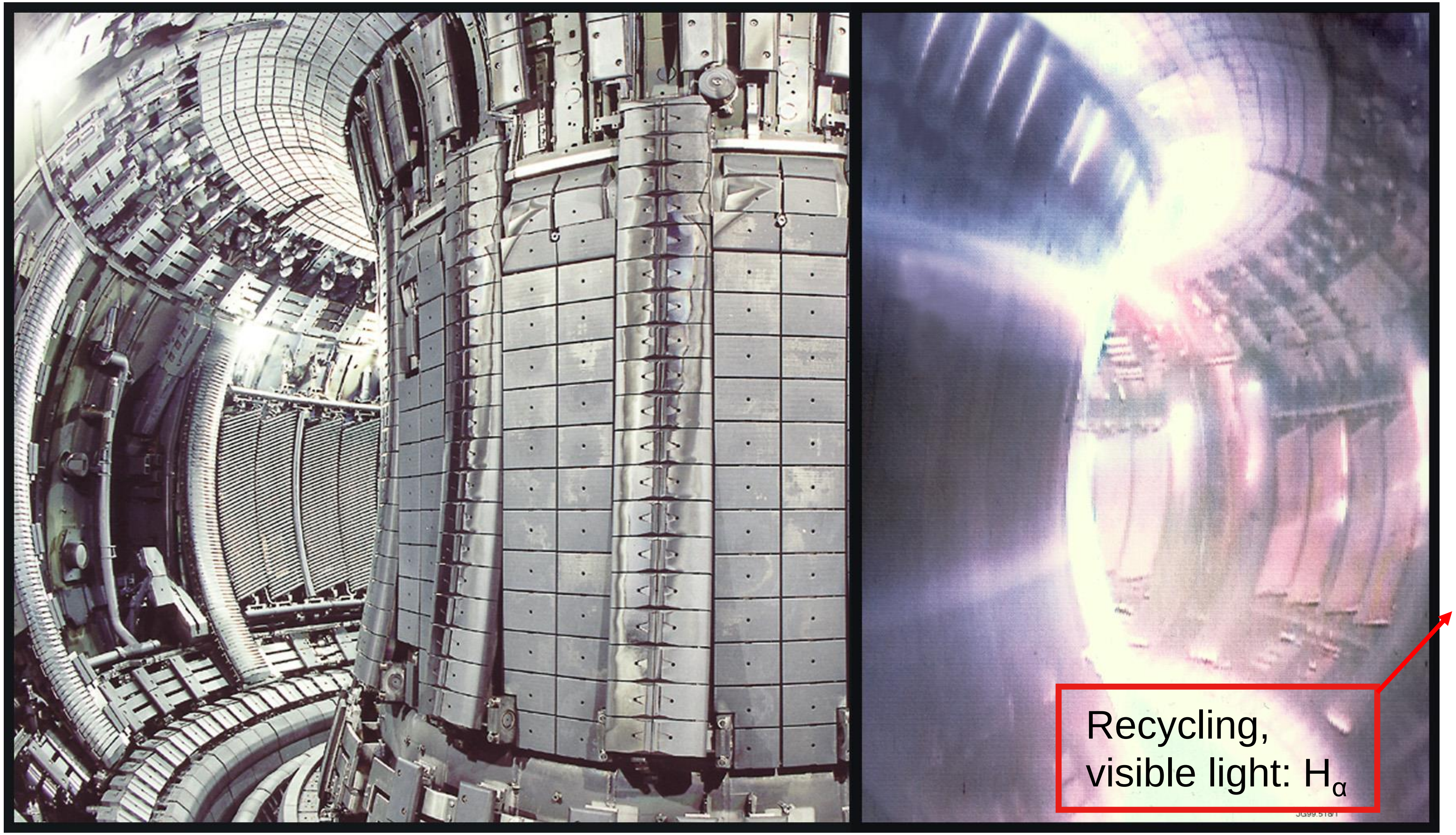
ALT-II
toroid. pump-limiter
Hidden behind:
8 pumping stations

Divertor vs Limiter geometry

- Until ~ 1980 - 1985 the heat flux handling structure was a **limiter** – aperture limits plasma size
  - Erosion rate high due to sputtering and evaporation
  - Impurities immediately go inside the separatrix (confined plasma) – cooling that plasma
- Divertor geometry - plasma-material interaction moved away from the confined plasma
- Still limiters in a divertor-based tokamak to protect the walls and components



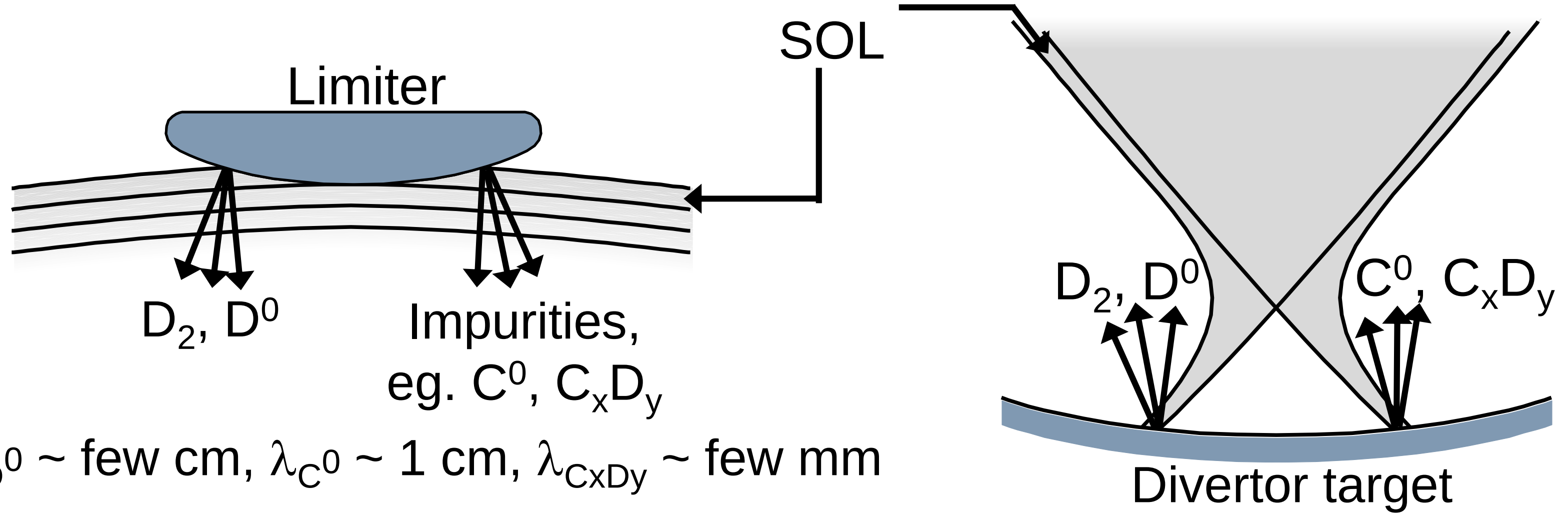
JET (Joint European Torus) : Poloidal Divertor (1993 ---)



Dominant plasma - surface interaction localized in divertor

JET Furnace chamber:
Ø 8.5 m 2.5 m high B = 3.4 T 4.8 MA 1 min

Limiter vs divertor recycling



λ_{D^0} ~ few cm, λ_{C^0} ~ 1 cm, λ_{CxDy} ~ few mm

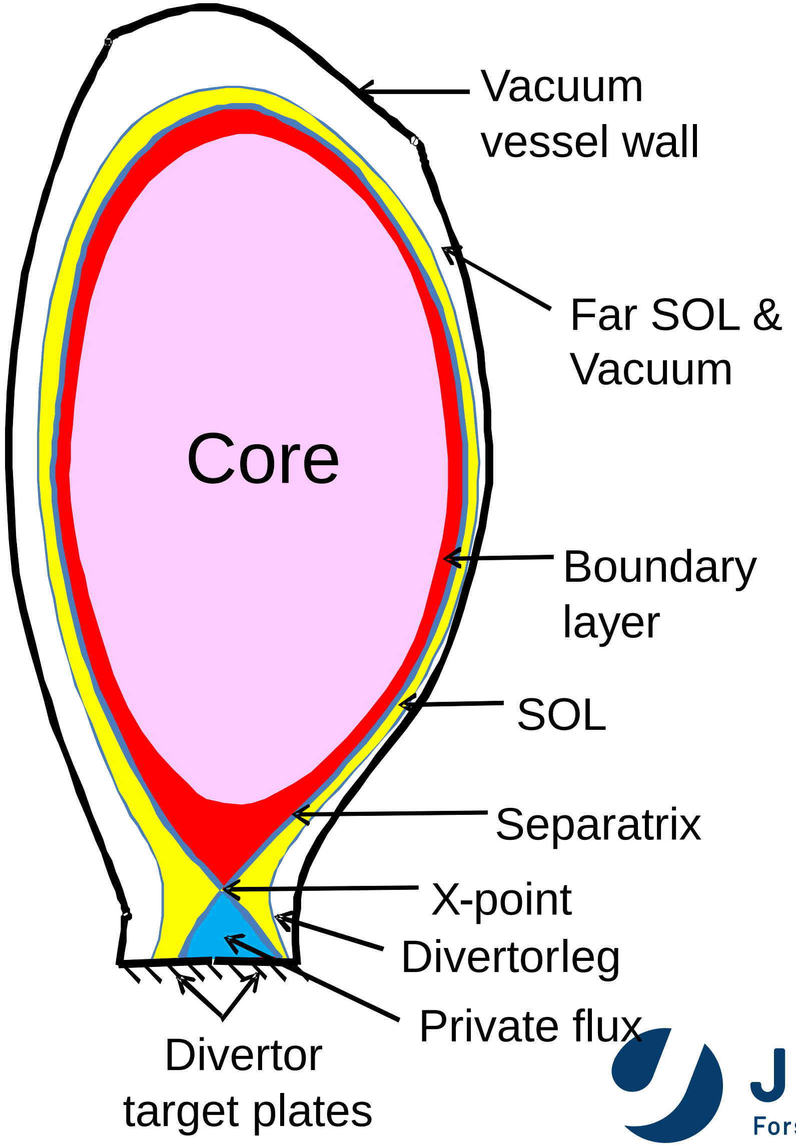
Limiter	Divertor
Intimate contact with edge plasma	PFCs removed from edge plasma
Little recycling/cooling in the SOL results in a hot, tenous SOL plasma	Colder, denser SOL plasma, due to local recycling / cooling
High erosion yields, poor pumping	Lower erosion yields, improved pumping
Strong influx of both fuel and impurity neutrals	Fuel and impurity sources screened
Impure edge & core, i.e. high Z_{eff}	Improved plasma purity, i.e. lower Z_{eff}

Fusion energy production puts severe demands on the surfaces and materials surrounding the plasma

What are the challenges and goals associated with the interaction of the plasma with surrounding material surfaces?

A. Reduce power flow to surfaces to below engineering limits through spreading the power

- Gigawatts/m^2 flowing along the field, but ~10 MW/m^2 engineering limit onto the surface

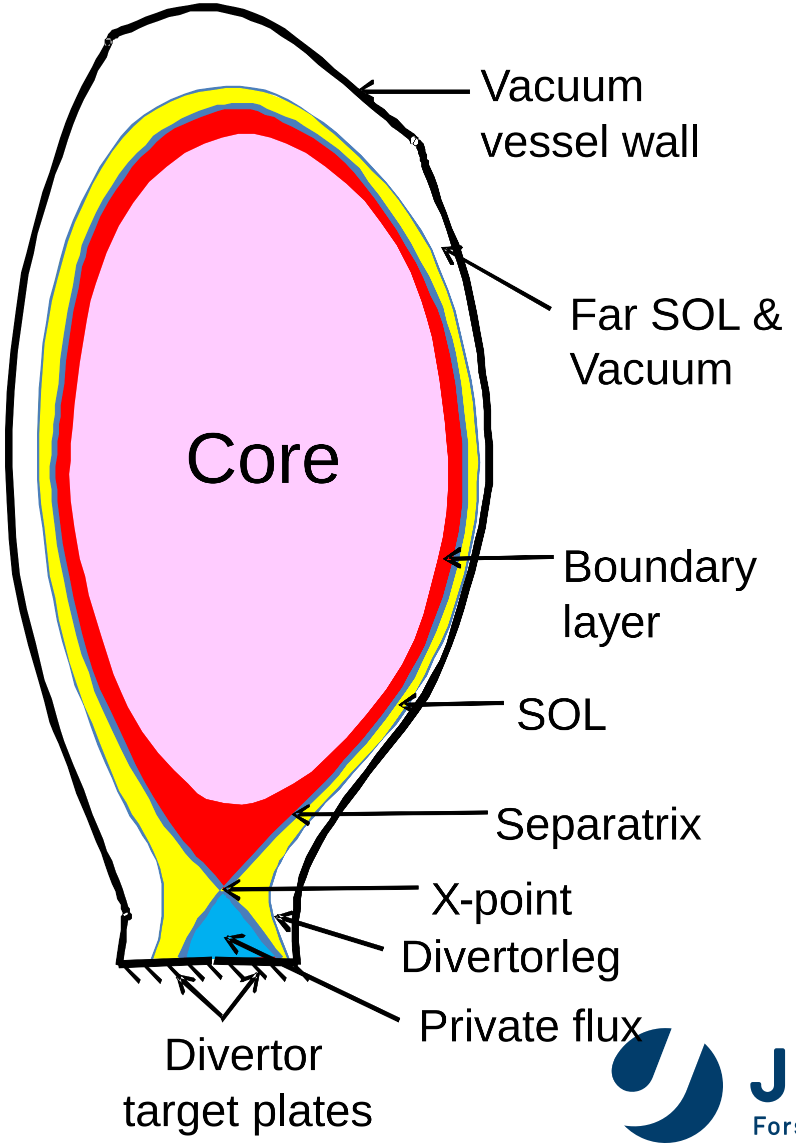


Fusion energy production puts severe demands on the surfaces and materials surrounding the plasma

What are the challenges and goals associated with the interaction of the plasma with surrounding material surfaces?

B. Lifetime of surfaces: Plasma Facing Components (PFCs)

- Erode at a rate consistent with reasonable maintenance and economics
- The material should not fail due to degradation of properties (neutrons, thermal…)

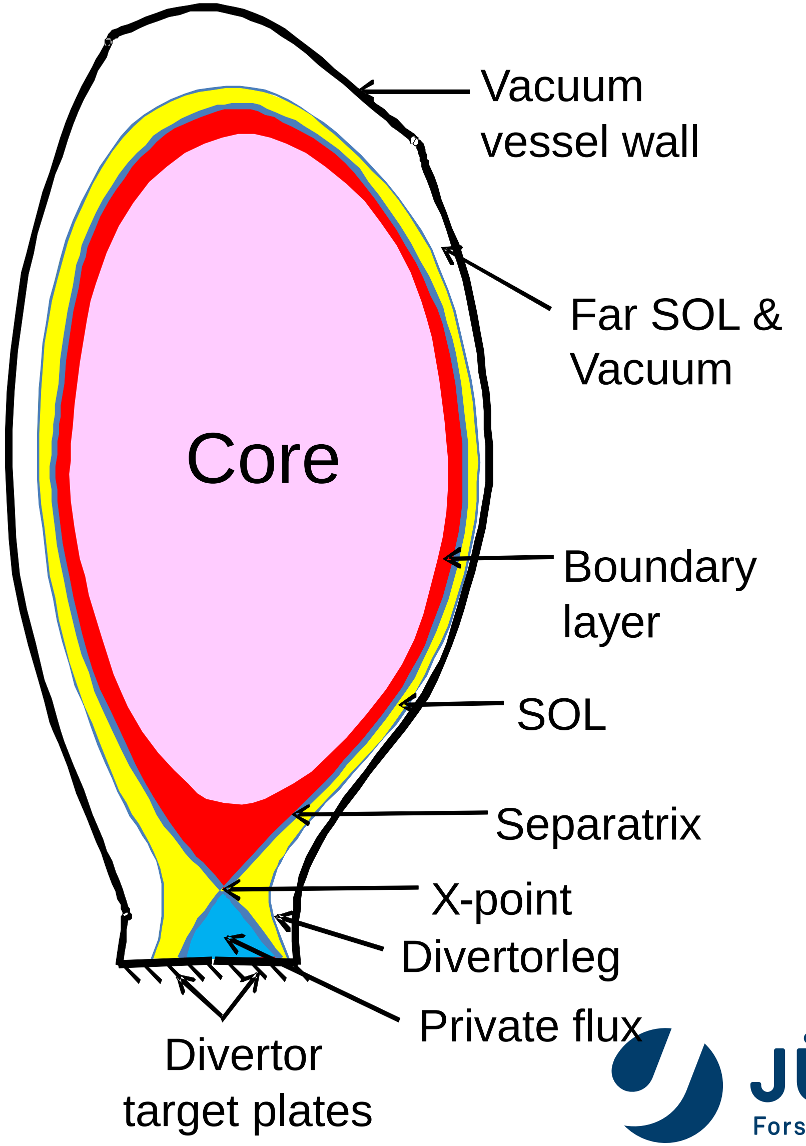


Fusion energy production puts severe demands on the surfaces and materials surrounding the plasma

What are the challenges and goals associated with the interaction of the plasma with surrounding material surfaces?

C. Compatibility – Atoms eroded from the material interface should not lead to degradation of the fusion process occurring in the core plasma

- Dilution due to the material atoms going into the plasma, displacing fusion reactions
- Radiation losses due to surface-eroded impurity atom excitation and ionization – cooling the plasma and lowering the fusion reaction rate

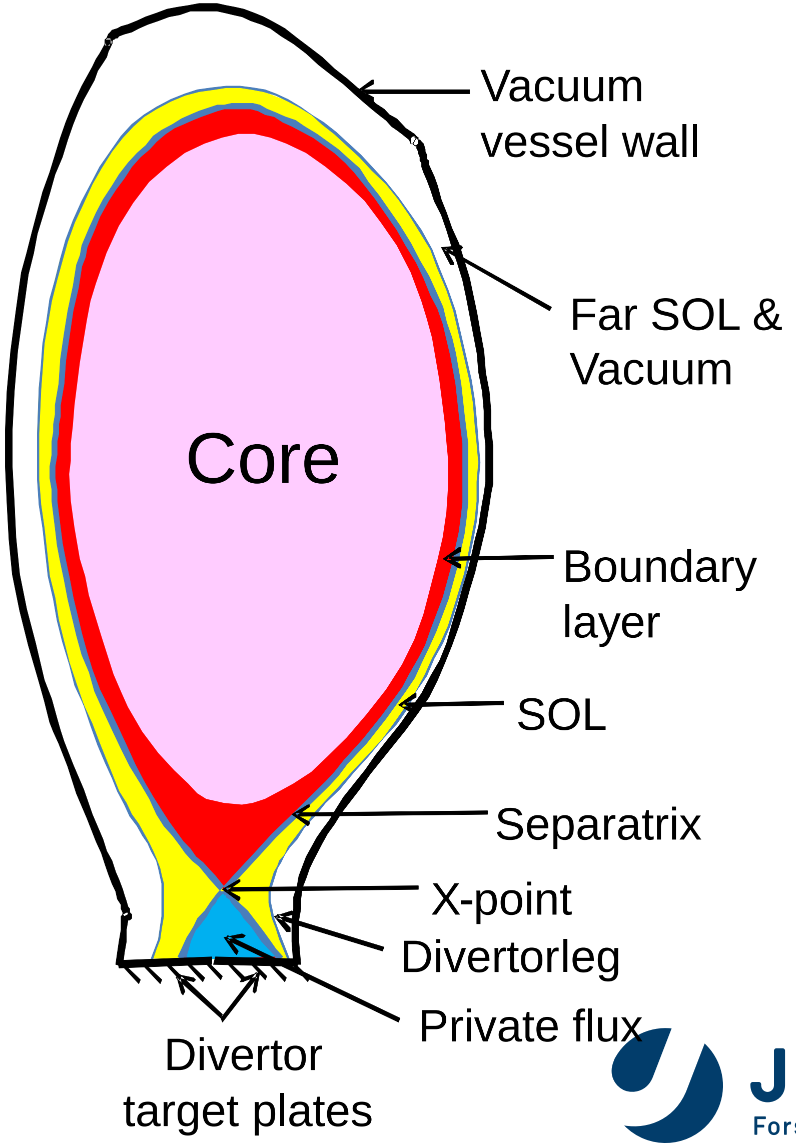


Fusion energy production puts severe demands on the surfaces and materials surrounding the plasma

What are the challenges and goals associated with the interaction of the plasma with surrounding material surfaces?

D. Tritium retention – only a small fraction of the fusion 'fuel' can get stuck in the 'engine'

- T can be implanted or buried in surfaces
- He 'ash' from fusion reaction needs to be removed efficiently

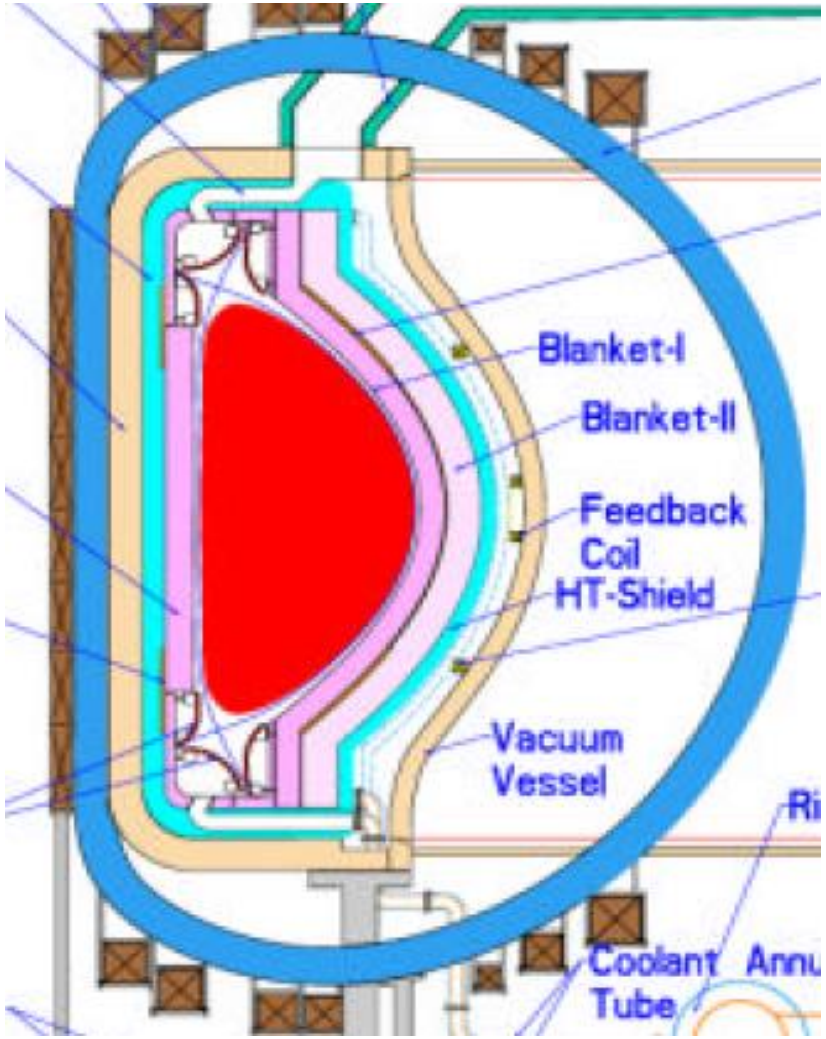


Properly handling the heat exhaust is a primary hurdle in the quest for magnetic fusion energy

Fusion reaction: *D+T -> He^4 (3.5 MeV) + **n** (14.1 MeV)*

- 1/5 of the fusion power (alphas) goes back into heating the plasma – ‘burning plasma’.
- That power has to be exhausted safely
  - “A reliable solution to the problem of heat exhaust is probably the main challenge towards the realization of magnetic confinement fusion”*

*EU -EFDA report ’Fusion Electricity – A roadmap to the realisation of fusion energy’, Nov. 2012



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The 4 challenges (mutually related)

A. Reduce power flow to surfaces to below engineering limits through spreading the power

B. Lifetime of surfaces (Plasma Facing Components, or PFCs)

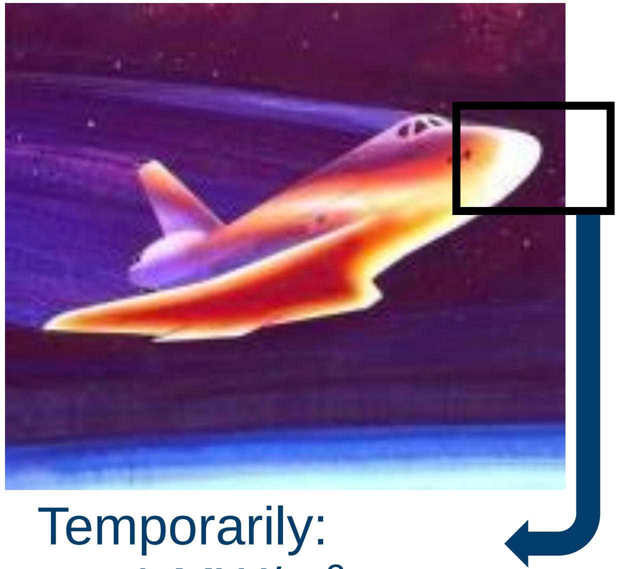
C. Compatibility – Atoms eroded from the material interface should not lead to degradation of the fusion process occurring in the core plasma

D. Tritium retention – only a small fraction of the fusion ‘fuel’ can get stuck in the ‘engine’

A. Power flow to surrounding material surfaces

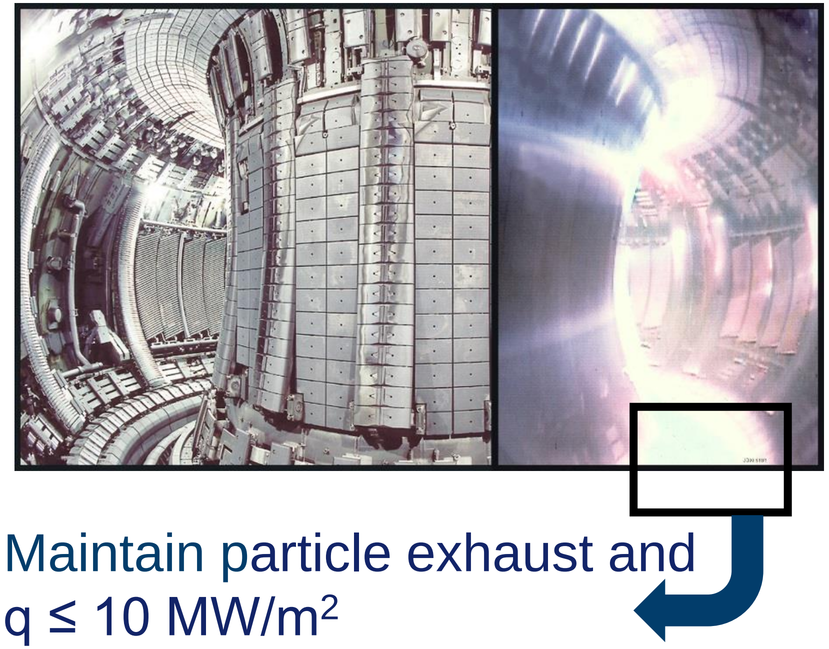
Sunny day
q ~ 0.5 kW/m²

Planetary space re-entry



Temporarily:
q ~ 1 MW/m²
a few minutes

The largest fusion reactor today:
JET (Joint Europ. Torus) :
Ø 8.5 m, 2.5 m high, 3.4 T, 7 MA, 1 min



Maintain particle exhaust and
q ≤ 10 MW/m²
steadily

Arc-welding

[http://www.youtube.com/watch?v=Ng-urnWhBR8&hd=1]

q ~ 40-100 MW/m²

A. How much will the heat flux spread as it flows along the magnetic field?

- The transport of particles and energy *across and along the field* determines where the particles/power go – *the peak heat flux and the 'footprint' on surfaces*
  - Sets the requirements for high heat flux components and where they need to be located
- The conductivity along the ***B*** field is much higher than across the field
  - The good confinement across ***B*** is wanted in the core but not in the SOL

  As for particles, parallel-to-***B*** heat flux width λ_q is determined by the ratio of $\perp$ to $||$ heat 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 theory and experiments.

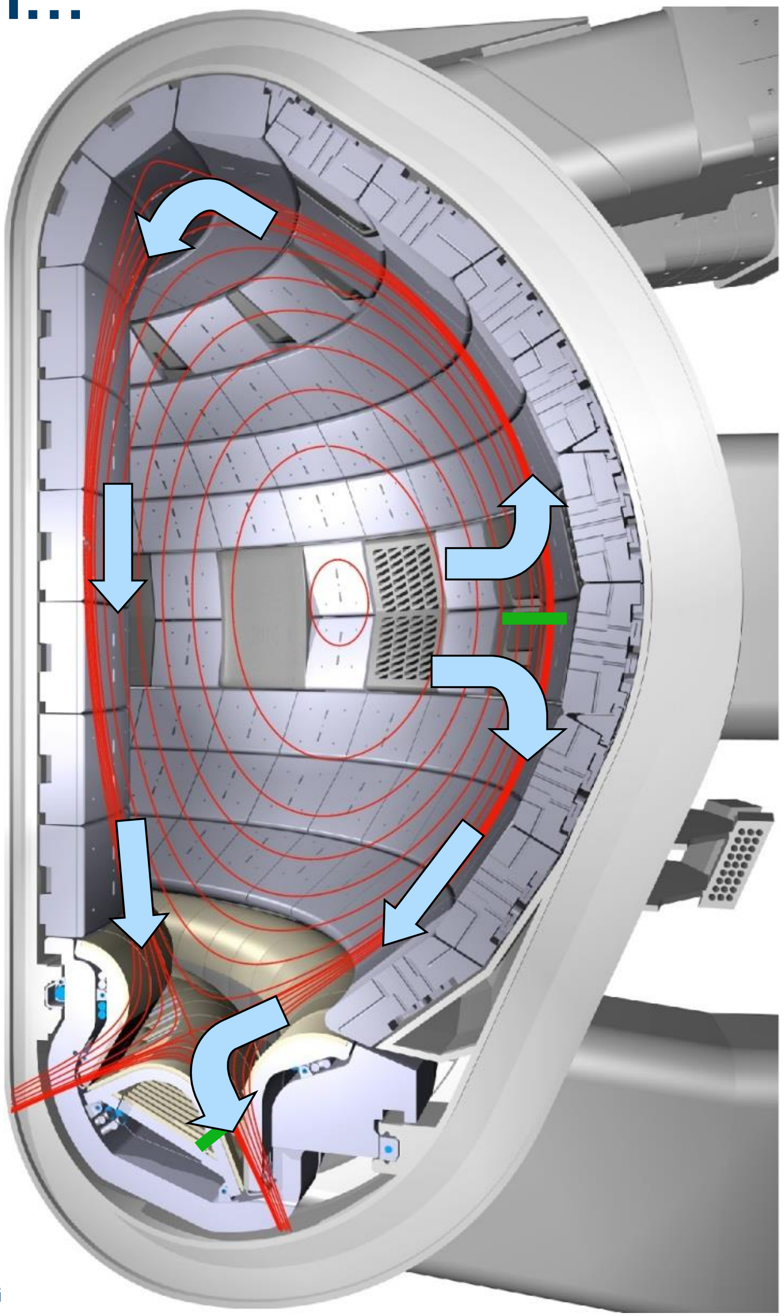
- Analysis of data from existing tokamaks leads to the conclusion that for ITER
  - The parallel-to-***B*** heat flux width*, $\lambda_{q||}$, is ~ 1mm, ~ 0.03% of the minor radius of the ITER plasma!
  - The smaller $\lambda_{q||}$ is, the higher the peak heat flux along the field, $q_{||}$

*M. Makowski, et al, Phys. Plasmas **19**, 056122 (2012);
B. LaBombard et al, Phys. Plasmas **18**, 056104 (2011);
Eich, T. et al., Plasma Phys. & Contr. Fus., 2005, 47, 815;
D. Whyte et al, J. Nucl. Mat. 438 (2013) S435–S439

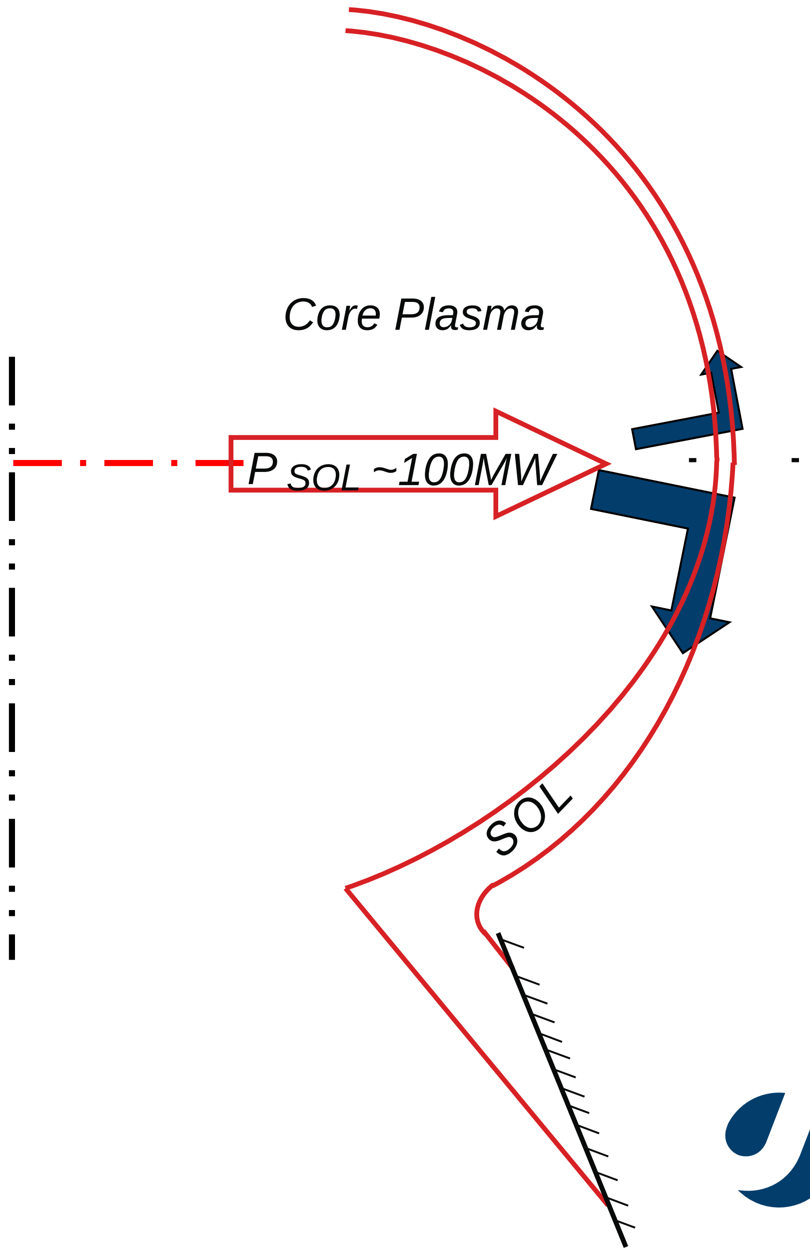
A. How much will the heat flux spread as it flows along the magnetic field?

- multi-machine scaling indicates: $\lambda_q/R \sim$ constant
- Stored energy scales strongly with tokamak major radius, $W \propto R^4$
  But power deposition area in the divertor $A \propto R\lambda_q$ only (~3.0 m^2 in ITER)
- W/A: 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.

From now on…



… a bit more schematic



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A. The power flow across the separatrix in ITER may not seem large but the resultant $q_{||}$ is very large

- For ITER: P_{SOL} ~ 100MW crossing the separatrix
  - The power primarily enters the SOL at the outer edge, or low-field side
  - Roughly $P_{SOL}/2$ flows poloidally toward the outer divertor through a horizontal planar **annulus** of area $2\pi R\lambda_q$
  - λ_q scrape off layer (heat flux) width: a decisive parameter. Best guesses: a few mm (?).

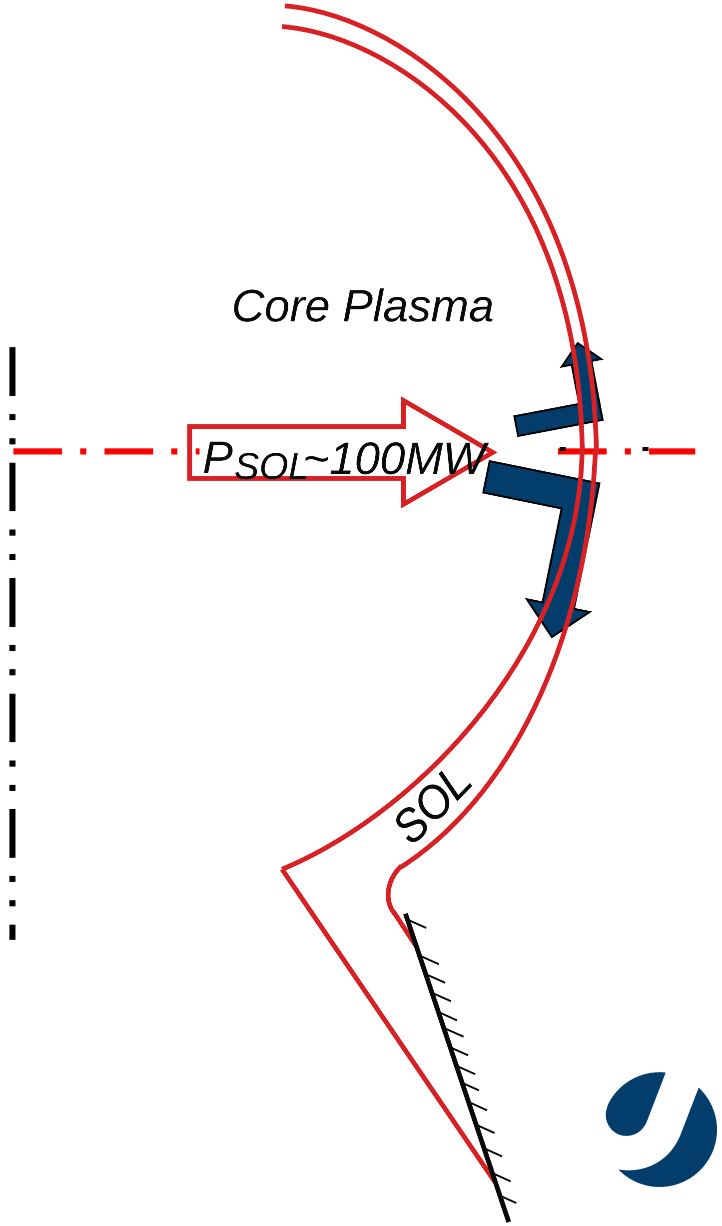
Great battlefield in boundary plasma theory.

$$\lambda_n \sim \lambda_\Gamma \sim \sqrt{\tau_{\parallel n} D_\perp}, \qquad \lambda_T \sim \lambda_q \sim \sqrt{\tau_{\parallel T} \chi_\perp}.$$

$$\tau_{\parallel n} \approx \frac{L_\parallel}{c_s},$$

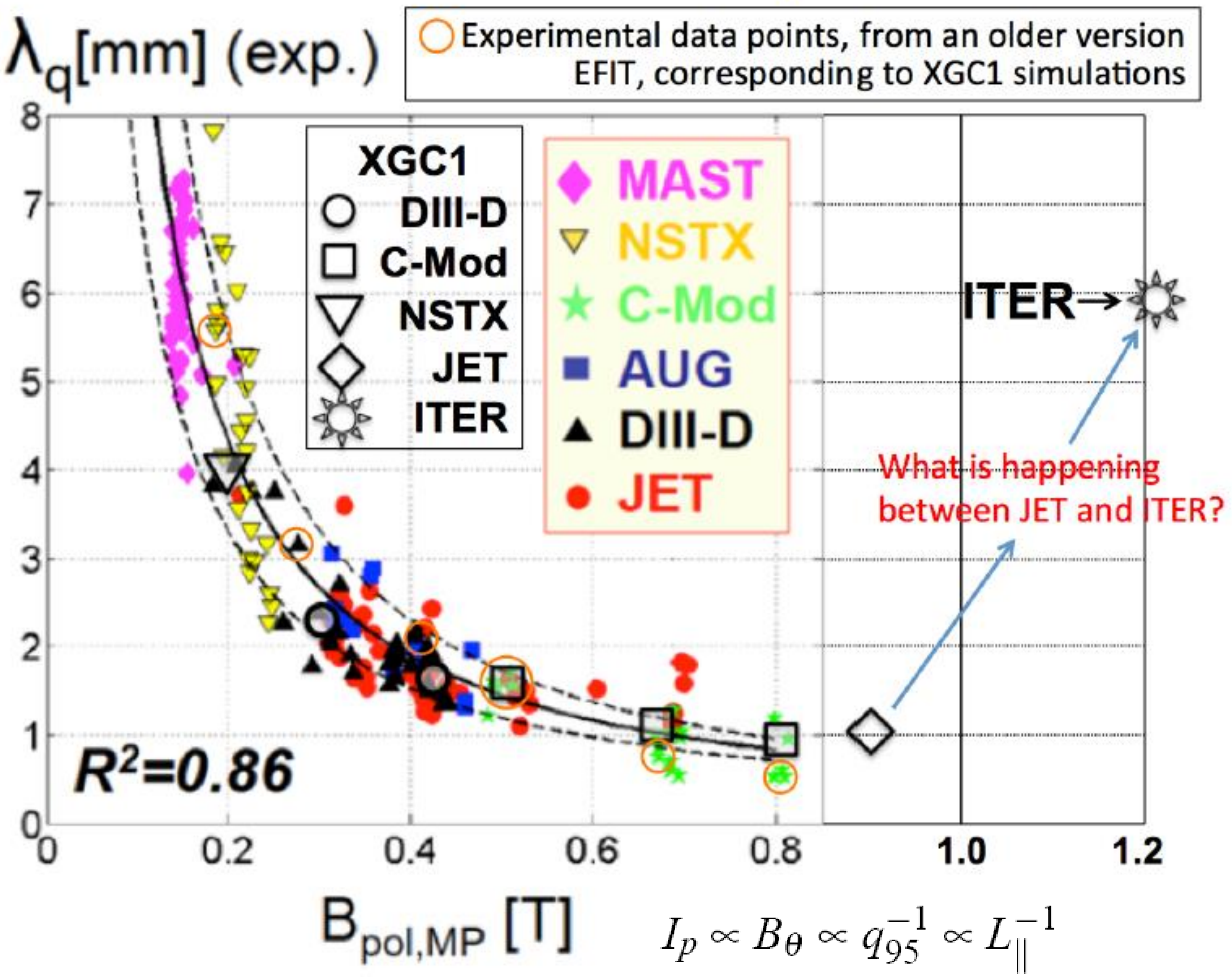
$$\tau_{\parallel T} \approx \frac{L_\parallel^2}{\chi_{\parallel e}}, \rightarrow \text{classical collisional}$$

anomalous



Aside: recent ab initio boundary plasma turbulence simulations indicate: nature might help ?

- XGC1 gyrokinetic simulations consistently show that $\lambda_q \propto 1/I_p$, (up to $B_{pol,MP}$ ~0.8 T), but also that this empirical + ion drift scaling is broken at the ITER scale: $B_{pol,MP}$ ~ 1.2 T
  - Attributed to domination of electron turbulence
- Recent attempt to model JET 4.5 MA discharge gave λ_q as expected from empirical scaling (very indirect "evidence" so far)
  - But no direct experimental λ_q values are available jet for these discharges
- Well diagnosed, high power, highest I_p JET discharges with divertor IR are needed to verify this result

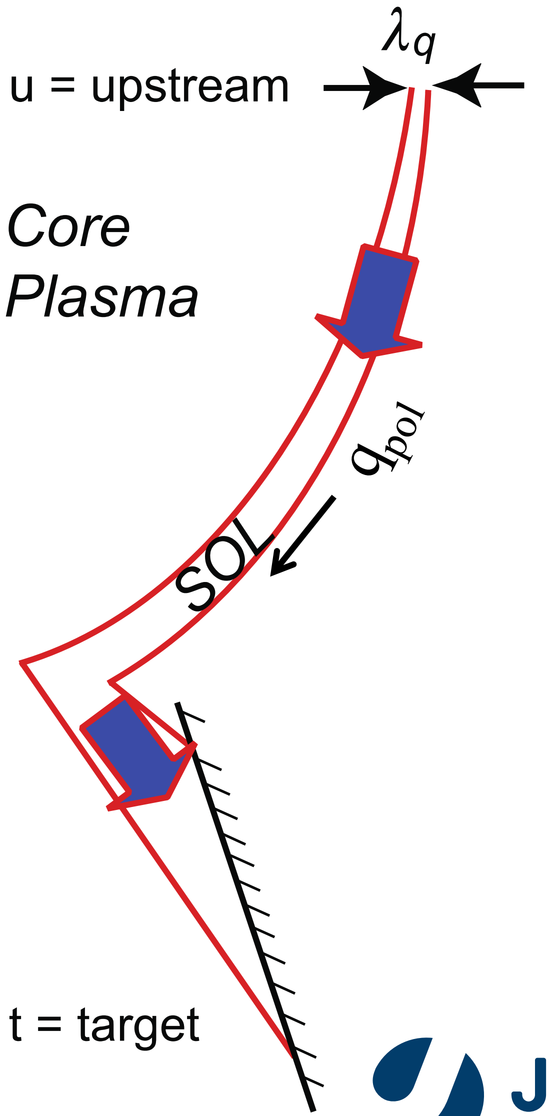


$$I_p \propto B_\theta \propto q_{95}^{-1} \propto L_\parallel^{-1}$$

A. The width of the SOL power flow, λ_q, is very small

- For ITER: P_{SOL} ~ 100MW crossing the separatrix
  - The power primarily enters the SOL at the outer edge, or low-field side
  - Roughly $P_{SOL}/2$ flows poloidally toward the outer divertor through a horizontal planar **annulus** of area $2\pi R\lambda_q$
  - Peak poloidal heat flux can be written:

$$q_{pol} \sim \frac{P_{SOL}/2}{2\pi R\lambda_q}$$



A. The parallel-to-B heat flow is considerably higher

Geometry: field pitch, B/B_θ
flux expansion,
target plate tilting, α wrt. B_θ

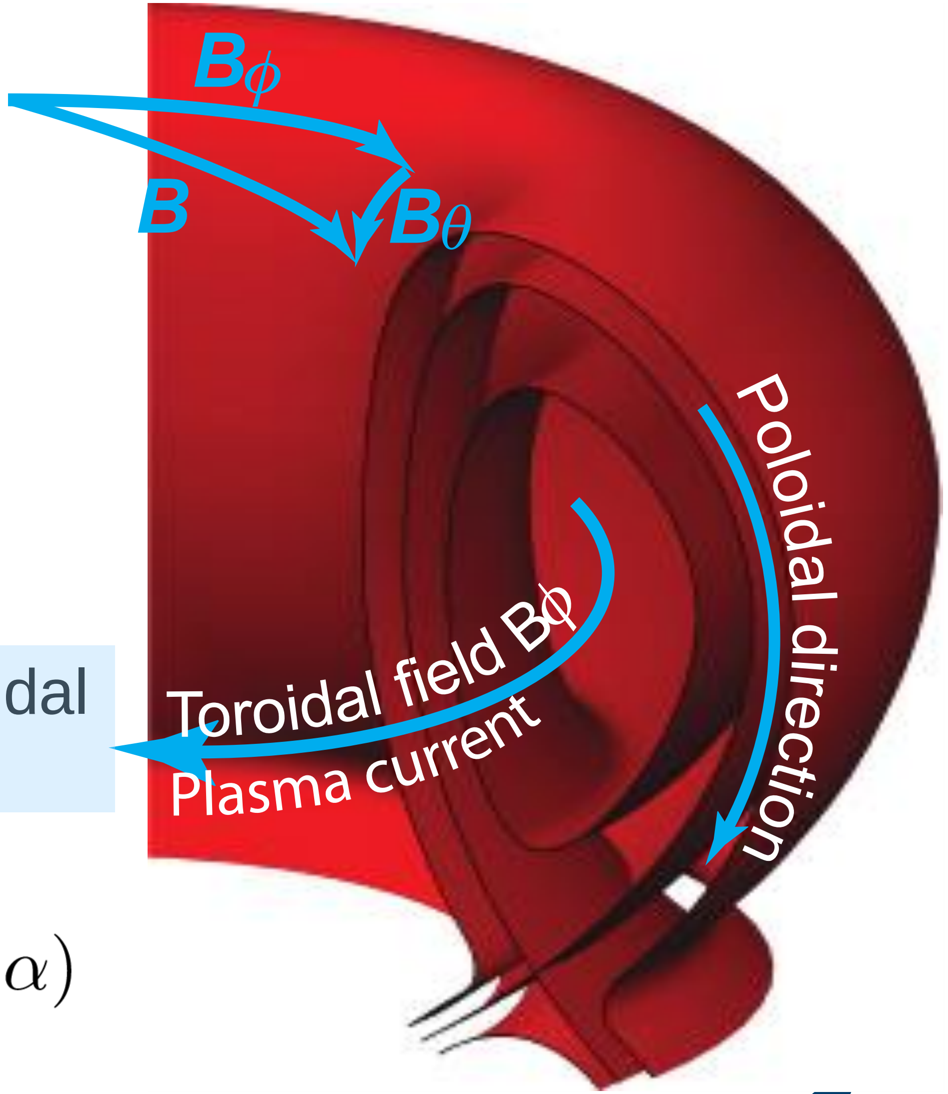
$$q_{pol} \rightarrow q_{||} \rightarrow q_{\perp,T}$$

$$q_{pol} \sim \frac{P_{SOL}/2}{2\pi R\lambda_q}$$

$$q_{||} \sim \frac{P_{SOL}/2}{2\pi R\lambda_q}\frac{B}{B_\Theta}$$

varies along poloidal direction

$$q_{\perp,T} \sim \frac{P_{SOL}/2}{2\pi R\lambda_q}\left[\frac{B}{B_\Theta}\right]_u\left[\frac{B_\Theta}{B}\right]_T \sin(\alpha)$$



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A. The high parallel heat flux leads to high SOL temperatures and plasma conductivity

$$k_{//} = k_0 T^{5/2}$$

The thermal conductivity increases rapidly with temperature

$$q_{//} = k_{//} \nabla T_{//} = k_0 \frac{2}{7} \frac{dT_{sep}^{7/2}}{ds}$$

ITER: $q_{||} \sim 2\ \mathrm{GW/m^2}$

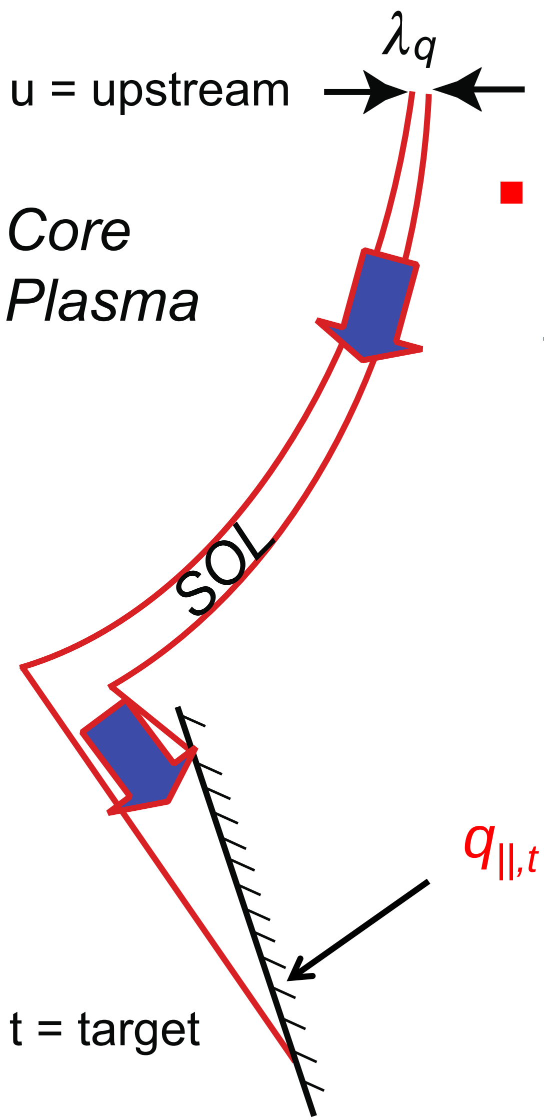
$$T_{sep} \sim \left[\frac{7}{2} \frac{q_{||}}{\kappa_0 L_{||}} \right]^{2/7} \sim 150\ eV \sim 1.5\ x 10^6 K$$

$T_t \ll T_{sep}$ allows us to solve for the separatrix temperature

$$\kappa_{sep} \sim 5x10^4 \frac{W}{m \cdot K} \sim 100\ \ \kappa_{copper}$$

- Even at ~150 eV the plasma just outside the separatrix in the SOL is a very low resistance thermal conductor

A. Ultimately $q_{//}$ must be reduced by at least 100-fold to reduce the surface heat flux below 10MW/m²



$q_{||,u}$ ~1.8 GW/m² (ITER), and ~ 5-10 larger in a Demo

- Physical limits for steady state heat removal by conduction through a surface < 10 MW/m², ~ 200 x less than $q_{||}$

$q_{||,t}$ ~ 2.2 GW/m²

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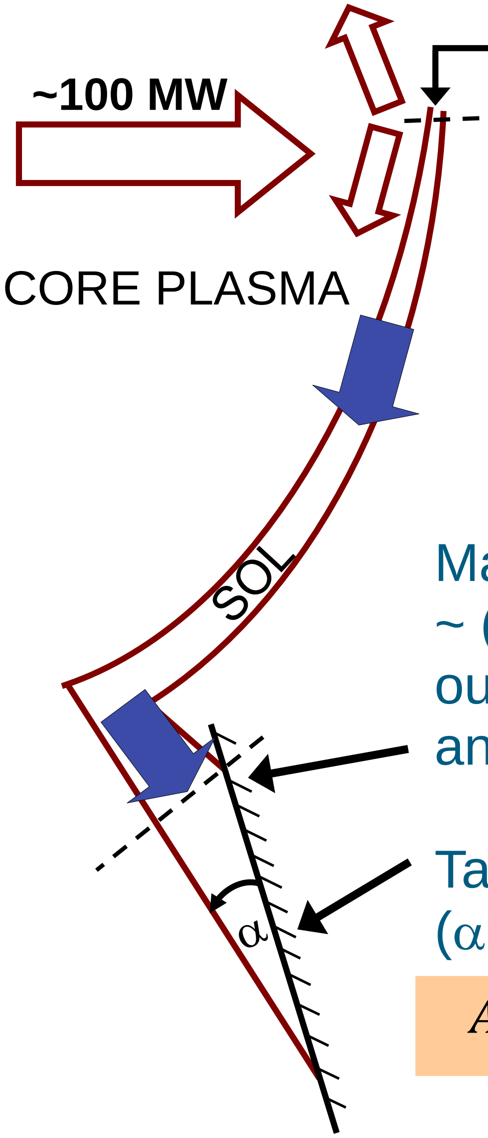
A. Ultimately $q_{//}$ must be reduced by at least 100-fold to reduce the surface heat flux below 10MW/m²



$Area \sim 2\pi R\lambda_q \sim 0.04m^2$ λ_q = 1-2 mm

$q_{||,u}$ ~ 1.8 GW/m² (ITER), and ~ 5 - 10 times larger in a Demo

Magnetic flux expansion
~ $(B_\theta/B)_u$ / $(B_\theta/B)_t$ ~ 4 for ITER outer divertor → low field line angles at strike points (~3°)

+

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

$Area \sim 2\pi R\lambda_q (B_\theta/B)_u / \sin(\alpha)(B_\theta/B)_t \sim 0.5m^2$ per target

$q_{\perp T}$ ~ 100 MWm⁻² per target
if no radiative (or other) dissipation

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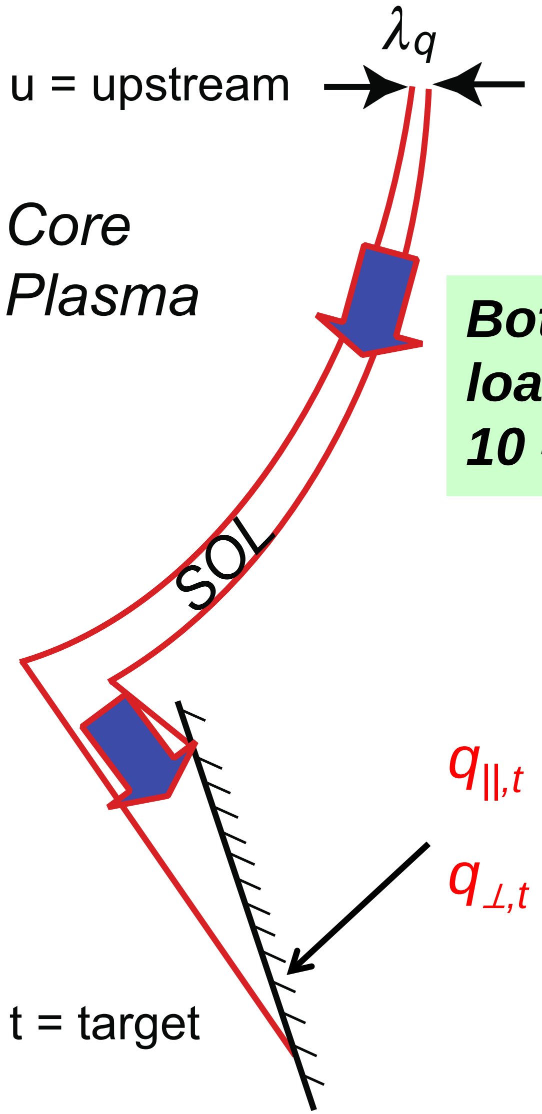
A. Ultimately $q_{//}$ must be reduced by at least 100-fold to reduce the surface heat flux below 10MW/m²



$q_{||,u}$ ~1.8 GW/m² (ITER), and ~ 5 -10 x larger in a Demo

Bottom line – geometry is not enough to reduce the heat loads to below engineering limits; Need another factor of 10 - 50 drop in $q_{||}$

$q_{||,t}$ ~ 2.2 GW/m²

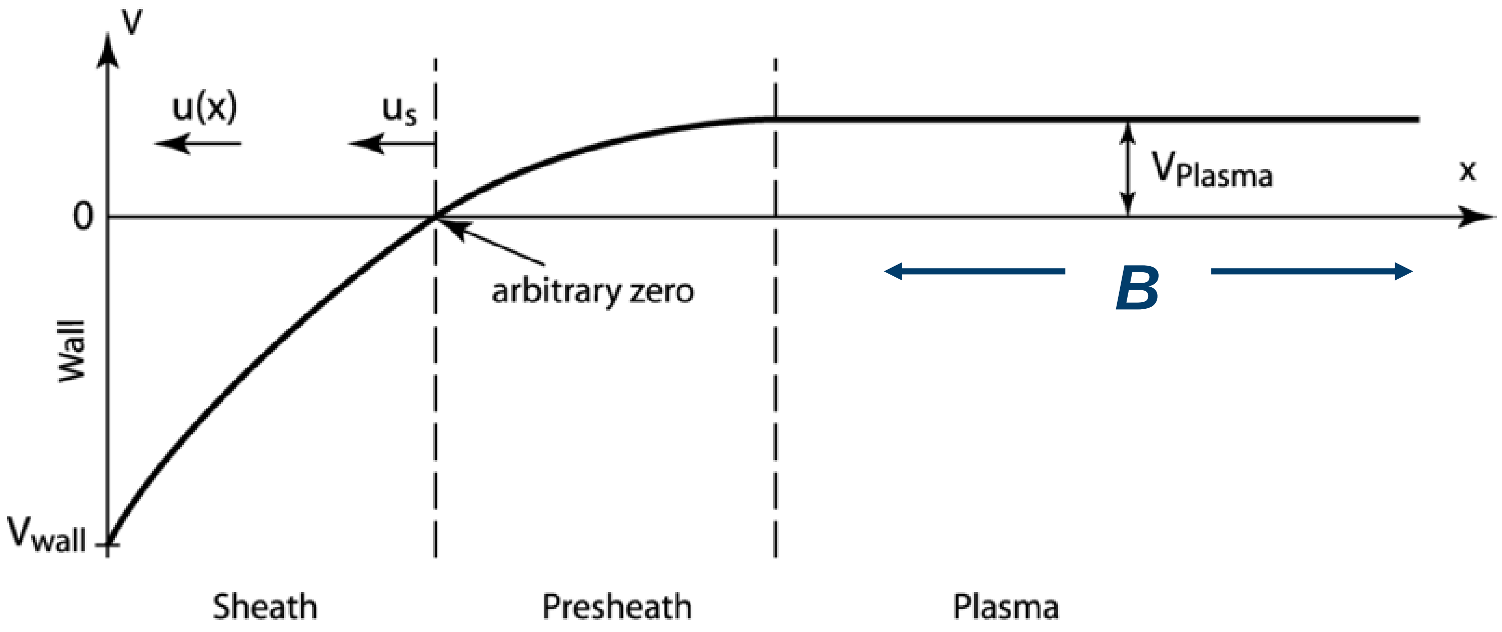
$q_{\perp,t}$ ~ 100 MW/m²

What can be done to bring $q_{||}$ down substantially?

Plasma Physics!

A. Power flow to surrounding material surfaces

- First review electrostatic ‘sheath’ at surfaces – *a concept central to plasmas*
  - At any interface between a plasma and a conducting surface a sheath is formed – a potential barrier to slow the electron flow to the surface to the same level as ions



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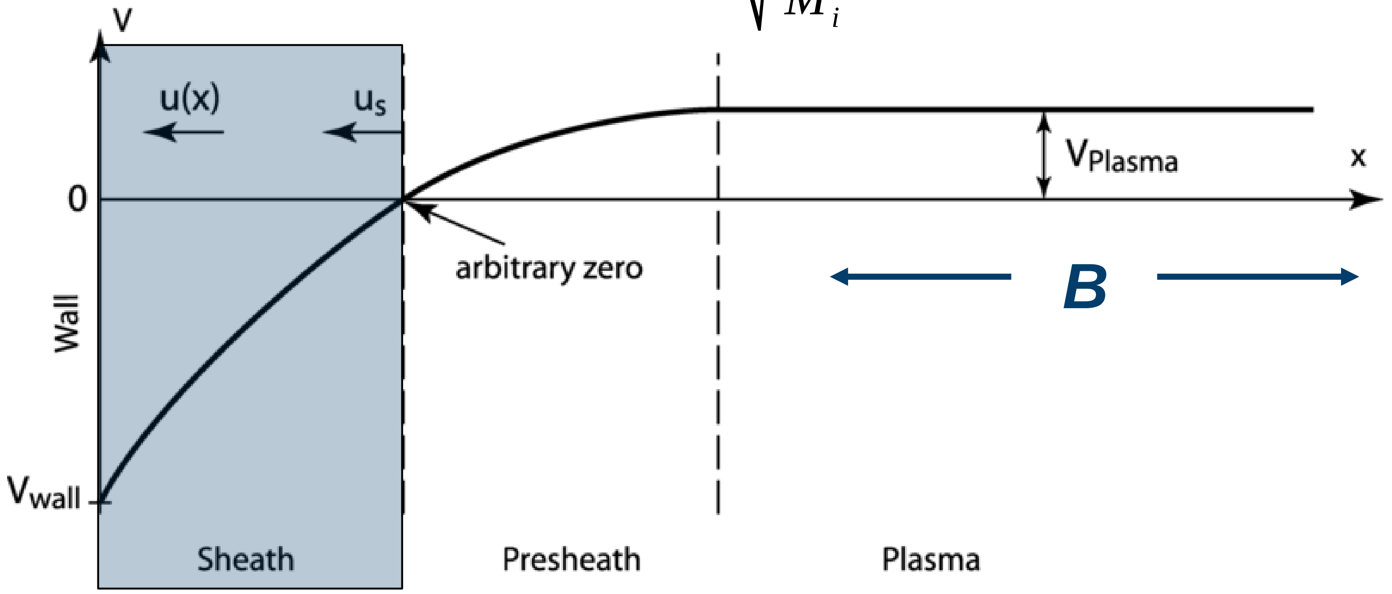
A. Review of the sheath

- Ion thermal velocity, $v_{th,i}$, is $\sqrt{\frac{M_i}{M_e}}$ smaller than electron thermal velocity, $v_{th,e}$.
  - *Electrons flow to surfaces much faster than ions*

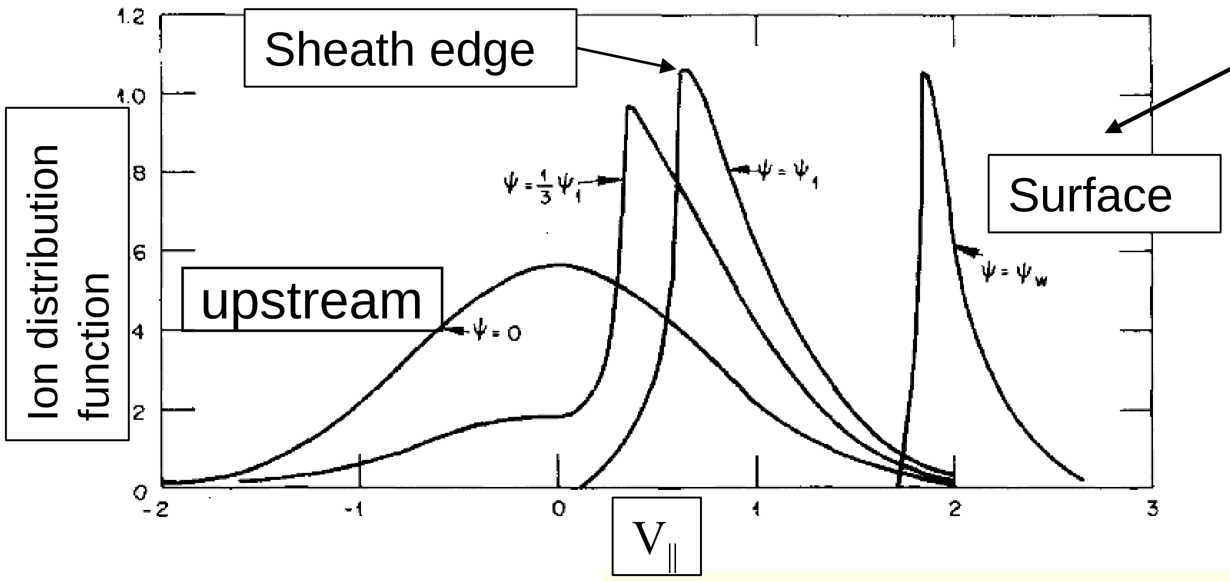
    (unless tilting angle of surface against B field is made too small: "near parallel targets")
- Allow the surface to be floating (not drawing an electr. current)
  - *Wall charges up negative and repels electrons* such that ion (J_i) and electron (J_e) currents are equal ($J_e = J_i$).
  - *Plasma flow at sound speed, $u_s = C_s = \sqrt{\frac{2T_e}{M_i}}$, entering the sheath*



A. Review of the sheath (with ≥ 2º inclination against B field)

- Ion thermal velocity, $v_{th,i}$, is $\sqrt{\frac{M_i}{M_e}}$ smaller than electron thermal velocity, $v_{th,e}$.
  - *Electrons flow to surfaces much faster than ions (are more “mobile”)*
- Allow the surface to be floating (not drawing current)
  - *Wall charges negative and rejects electrons* such that ion (J_i) and electron (J_e) currents are equal ($J_e = J_i$). *Also accelerates ions towards the surface*
  - *Plasma flow at sound speed, $C_s = \sqrt{\frac{2T_e}{M_i}}$, entering the sheath*

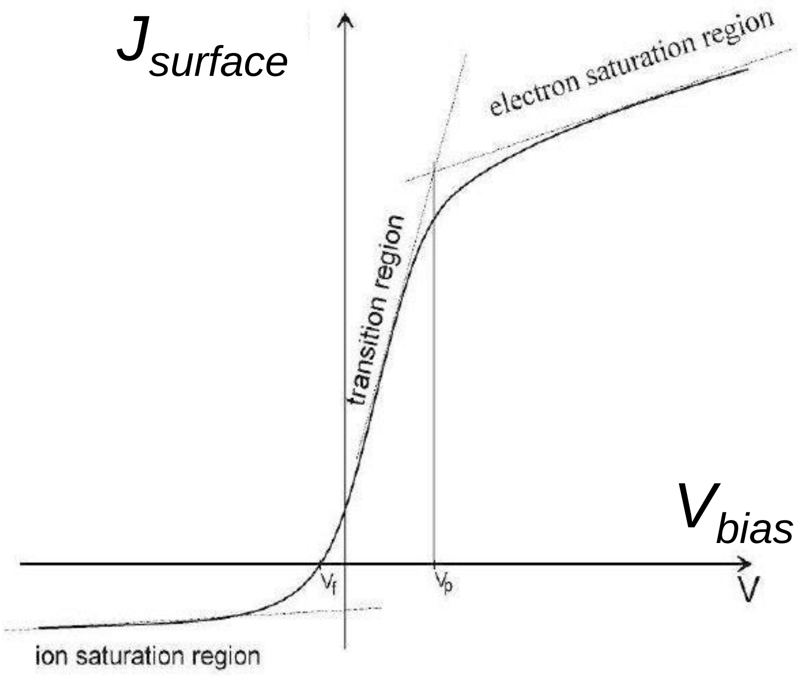
Non-Maxwellian
ion distribution
at sheath entrance



G. Emmert et al., *Phys. Fluids* **23** (1980) 803

Aside: Langmuir probes

- Electrons are in a repelling potential, the (truncated) Maxwellian form at sheath entrance in maintained up to target (at same T_e), just more truncated there.
- As V_{wall} is varied ($J_{surface}$ vs V_{bias}), one obtains a measure of the electron distributions function – that is how Langmuir probes measure T_e.



$$J_{surface} = J_{i,sat} + J_{e,sat} \exp\left[e\left(V_{bias} - V_{plasma}\right) / kT_e \right]$$

A. Review of the sheath: heat flux through sheath

- Heat flux to the surface:

  $q_{||} = \gamma T_e \Gamma_{||} = 0.5\gamma T_e n_e C_s$ @ surface;

  γ~7-8 (the 'sheath transmission coefficient')

  C_s ion acoustic speed ~ $\sqrt{\frac{2T_e}{M_i}}$ (for T_e=T_i)

- Components of the sheath transmission coefficient γ :
  - ~ 2 - 2.5T_e thermal ion energy (for T_i = T_e)
  - ~ 2T_e due to the electron thermal energy through the sheath
  - ~ 2.5 - 3T_e due to the sheath potential (accelerated flow of ions)
  - ~ 1T_e directed flow energy (sound speed: 0.5 T_i+ 0.5T_e, then assume T_i = T_e)

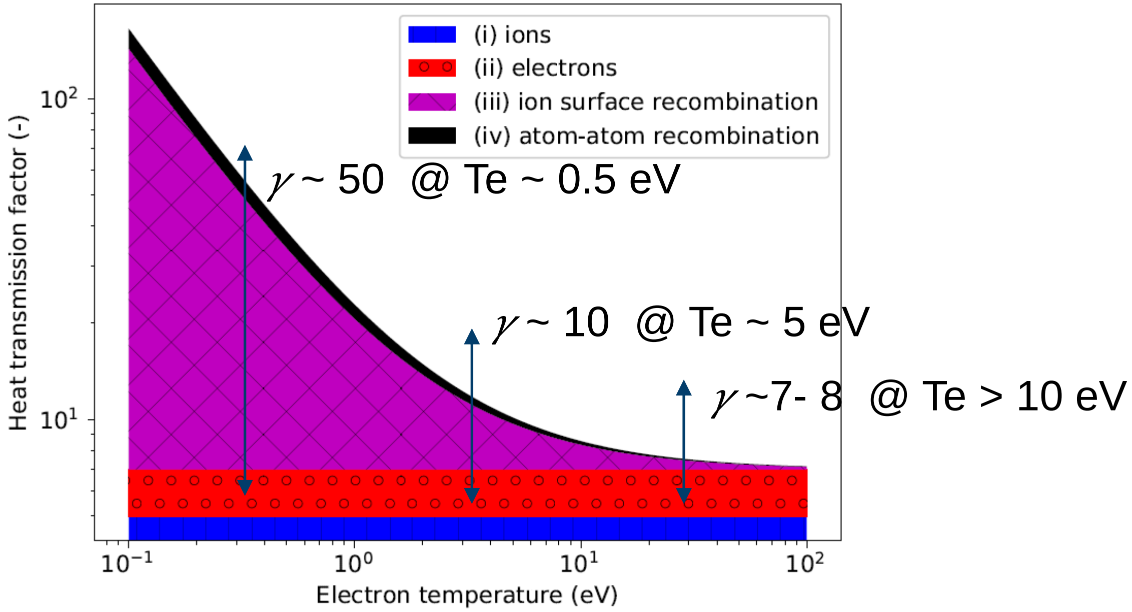
Sheath – target heat transfer factor

$q_{||} = \gamma T_e \Gamma_{||} = 0.5 \gamma T_e n_e C_s$ @ surface;

γ ~7- 8 ('sheath transmission coefficient'), is often taken as constant.

But even for $T_e \rightarrow 0$: Further contributions: $q_{eir} = 13.6$ eV x $\Gamma_{||}$ electron-ion recombination at surface
$q_{aar} = 2.2$ eV x $\Gamma_{||} (1-R_{i,N})$ atom-atom recombination

→ Removing just power may not be enough. Below T_e=5 eV: Must also reduce $\Gamma_{||}$



Even for $\Gamma_{||} \rightarrow 0$:
Further target heating sources:

- cx –neutrals from divertor volume
- Localized divertor radiation

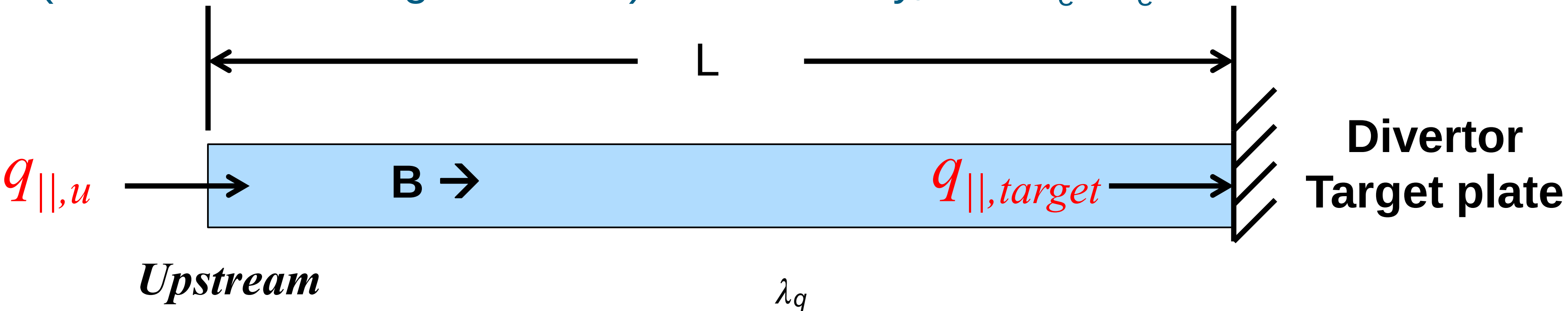
A. Power flow to surrounding material surfaces

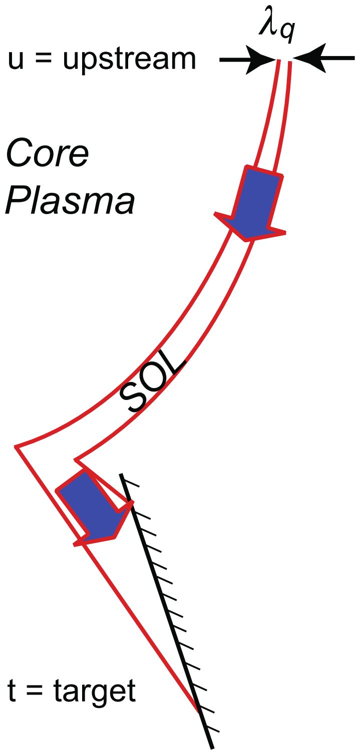
- Reduce the power reaching the divertor surfaces (PFCs) through divertor physics: $q_{||} = \gamma T_e \Gamma_{||} = 0.5\gamma T_e n_e C_s$ *plasma heat flux @* surface;

- Step 1: increase plasma collisionality (density) → parallel gradients
- Step 2: remove power by (impurity) radiation → low T_e , T_i at target
- Step 3: remove momentum (break plasma pressure balance) → low Γ_{target}
- Step 4: turn plasma ions/electrons into neutrals (recombination)

Step 1 – form gradients in ne, Te along flux tube from core edge to divertor (pressure constant)

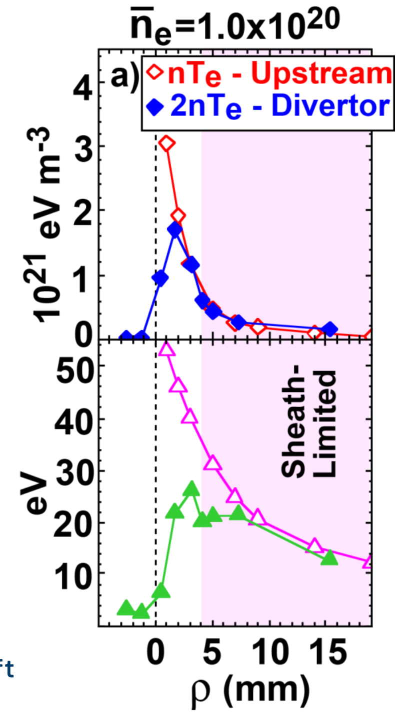
- Raise density (n) such that the mean free path for collisions < flux tube length, L (not free-streaming electrons): collisionality, $\nu^* \sim n_e L / T_e^2$

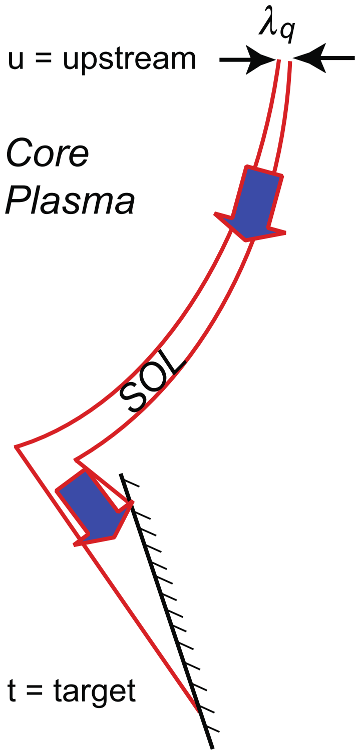




Step 1 – form gradients in ne, Te along flux tube from core edge to divertor (pressure remains constant along such flux tubes)

- Raise density (n) such that the mean free path for collisions < flux tube length, L (not free-streaming electrons): collisionality, $\nu^* \sim n_e L/T_e^2$



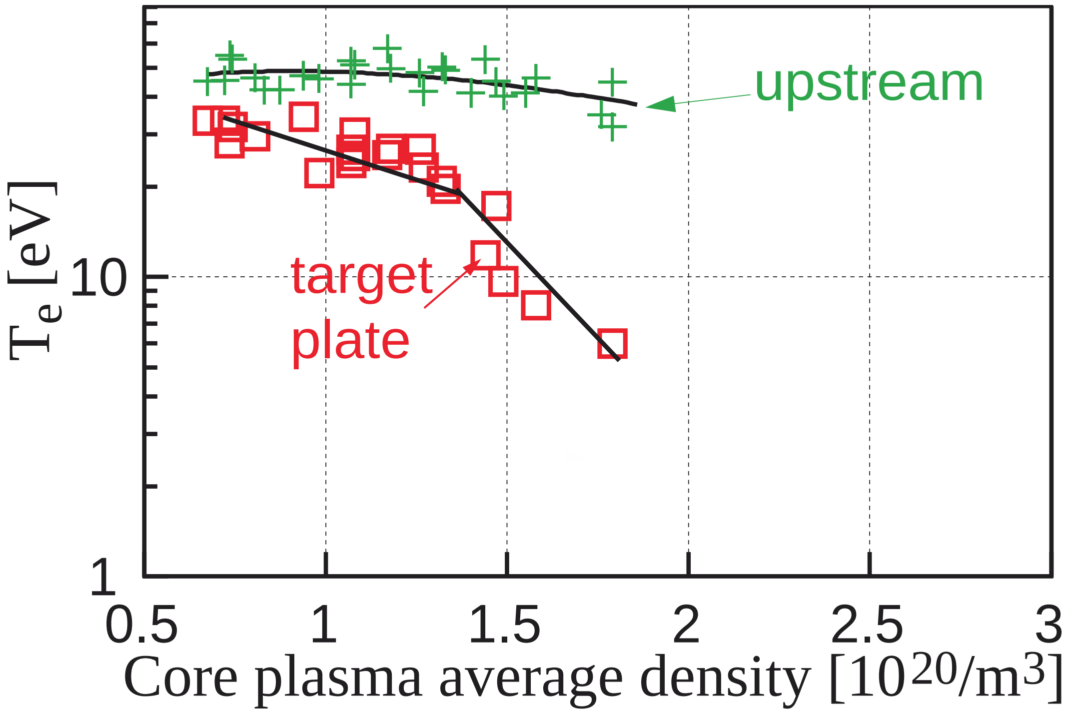


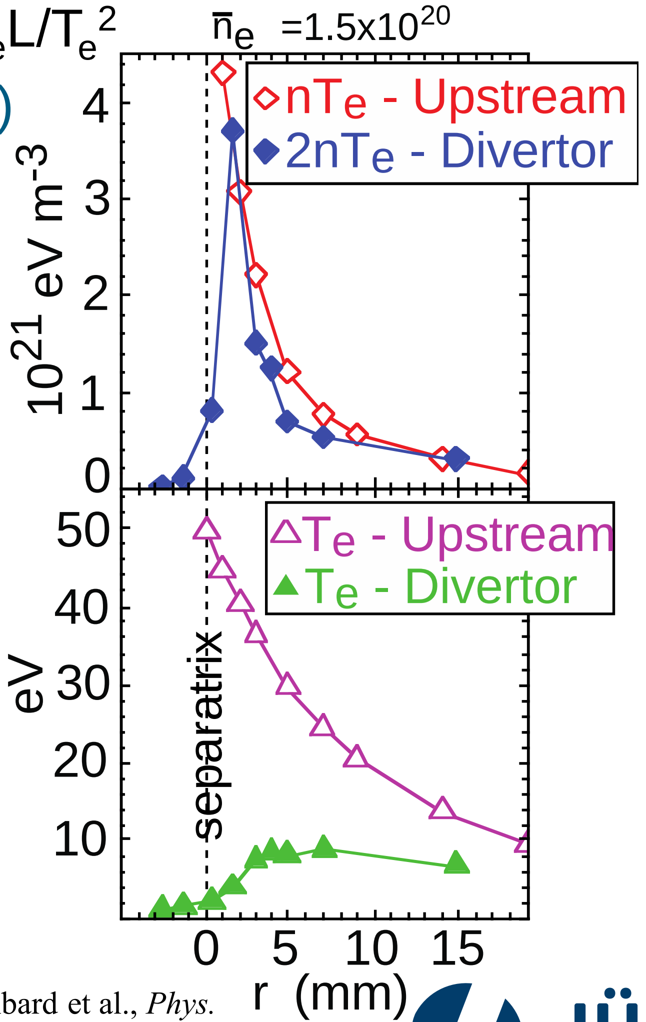
- $q_{||,target} = 0.5\gamma T_e n_e C_s \propto p_{e,tar} T_{tar}^{1/2}$

Step 1 – form gradients in ne, Te along flux tube from core edge to divertor (pressure constant)

- Raise density (n) such that the mean free path for collisions < flux tube length, L (not free-streaming electrons): collisionality, $\nu^* \sim n_e L/T_e^2$
  - Would allow gradients along flux tube & lower T_e (higher n_e) at the divertor target
  - Pressure ($n_eT_e + n_iT_i + 0.5M_iv_i^2$) constant along **B**

B. Lipschultz et al., *Fusion Science Tech.*, **51**, (2007) 369

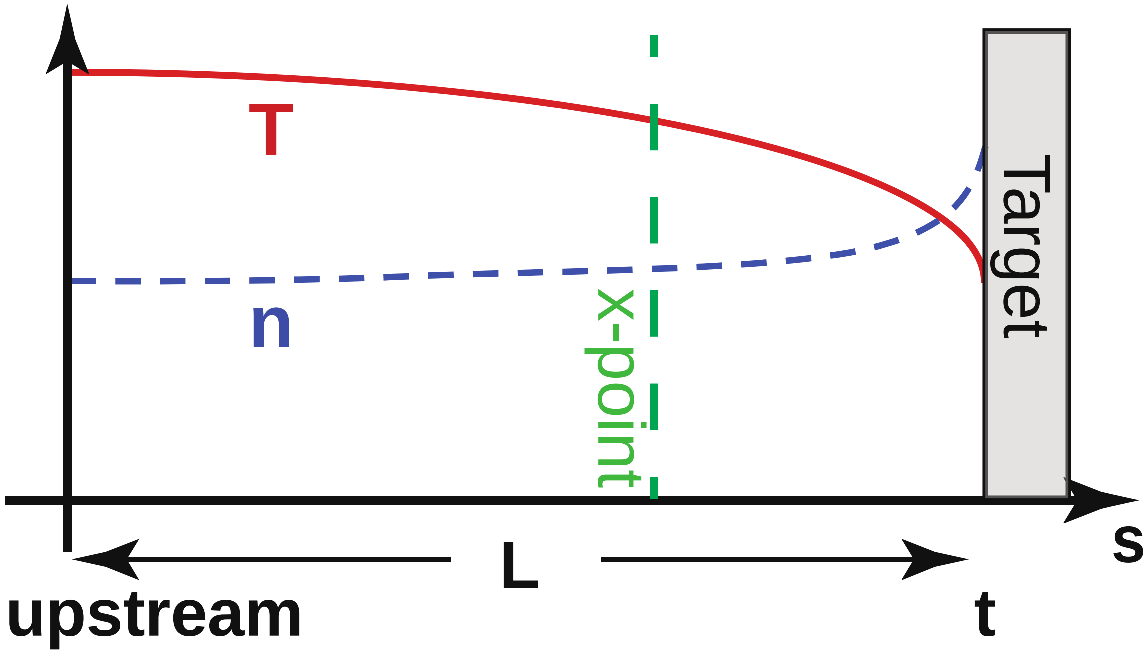




B. LaBombard et al., *Phys. Plasmas*, **2**, (1995) 2242

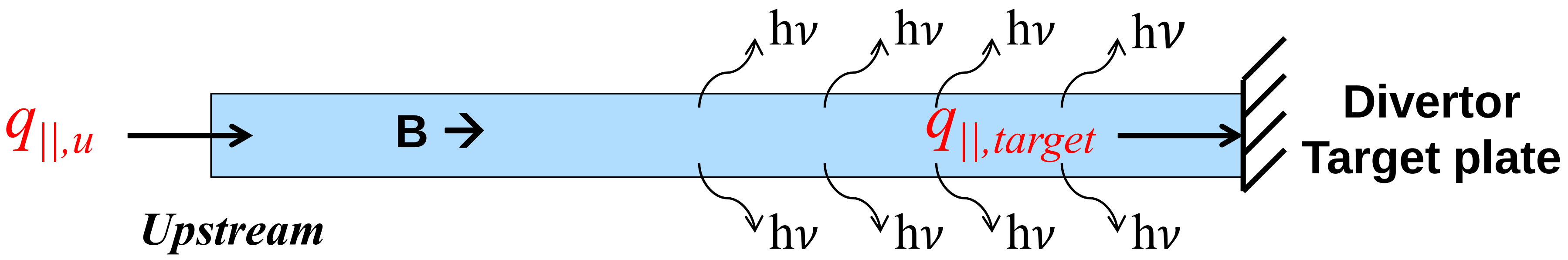
Step 1 – decrease thermal conductivity of flux tube (with pressure constant along B)

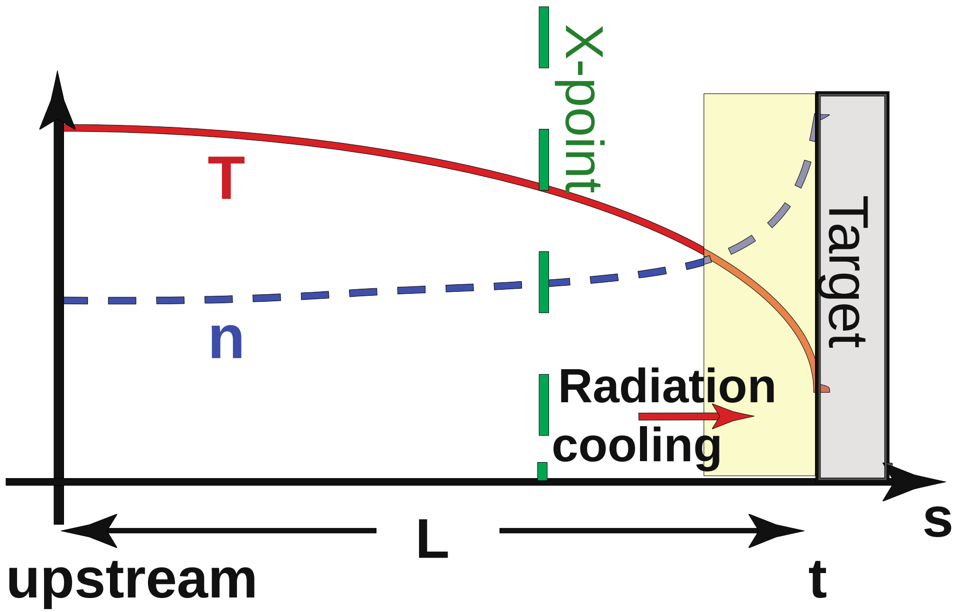
- Raise density (n) such that the mean free path for collisions < flux tube length, L (not free-streaming electrons): collisionality, $\nu^* \sim n_e L/T_e^2$
  - Would allow gradients along flux tube & lower T_e (higher n_e) at the divertor target
  - Pressure ($n_e T_e + n_i T_i + 0.5 M_i v_i^2$) constant along **B**

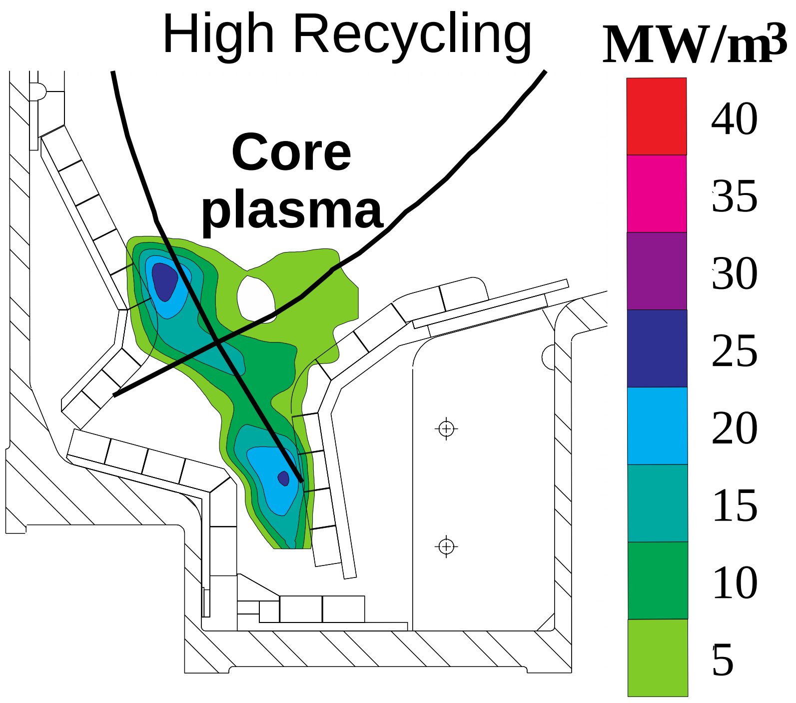


Adapted from C.S. Pitcher & Stangeby, Plasma Phys. Contr. Fusion **39** (1997) 779

Step 2 – remove power from the plasma in the flux tube through impurity radiation

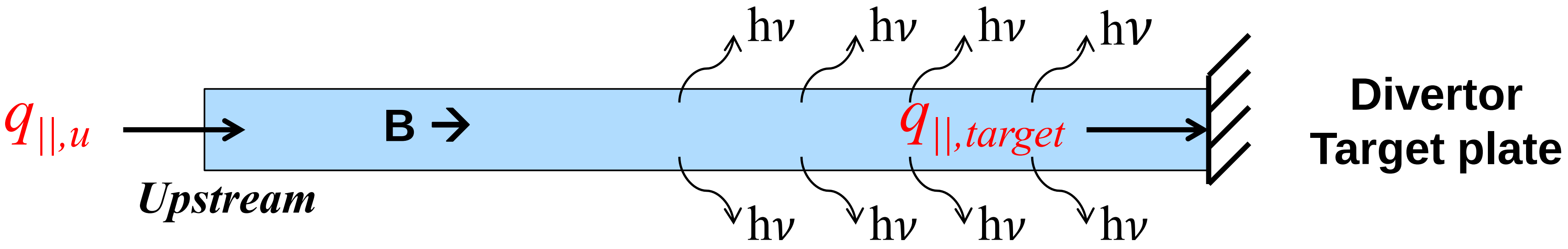




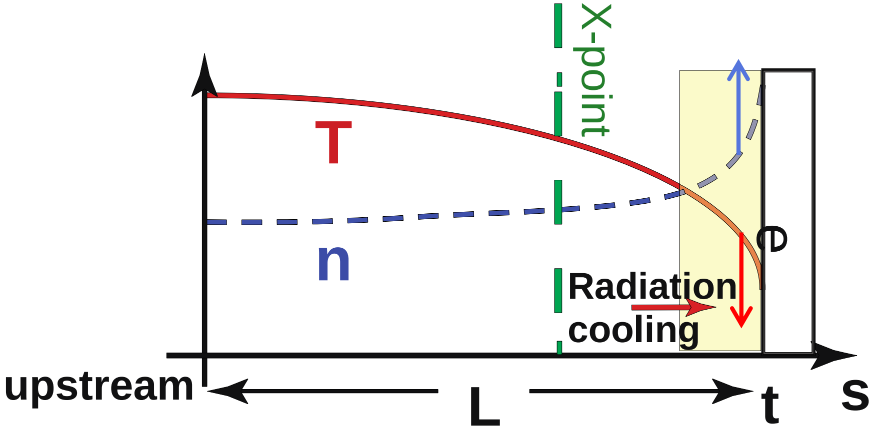


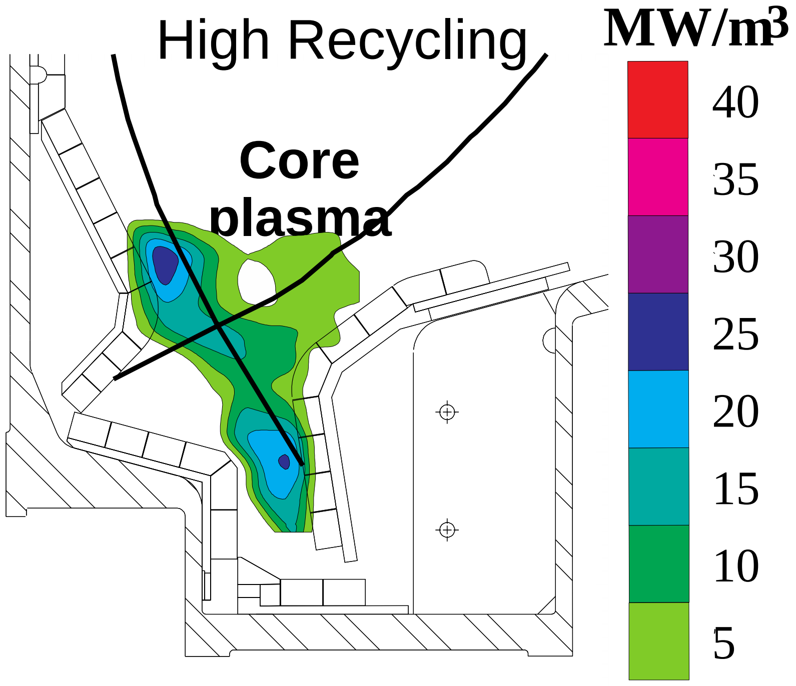
J. A. Goetz, B. Lipschultz et al., *Phys. Plasmas*, **3** (1996) 1908.

Step 2 – remove power from the plasma in the flux tube through impurity radiation



- Spreads the power over larger area, reduces $T_{e,t}$
- Bohm sheath condition at target gives
  - $\Gamma_{||,target} = 0.5 n_{e,t} C_s \propto P_{e,t}/T_{e,t}^{1/2}$
  - $q_{||,target} = \gamma T_{e,t} \Gamma_{||,target} \propto P_{e,t} T_{e,t}^{1/2}$





J. A. Goetz, B. Lipschultz et al., *Phys. Plasmas*, **3** (1996) 1908.

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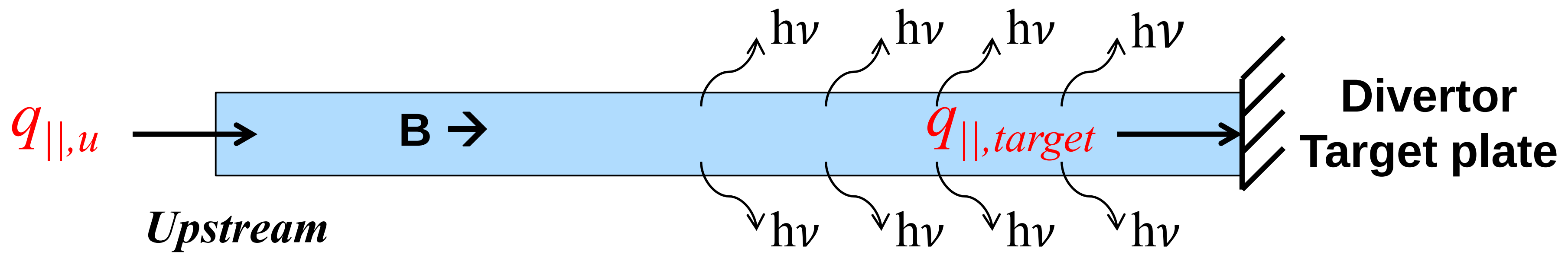
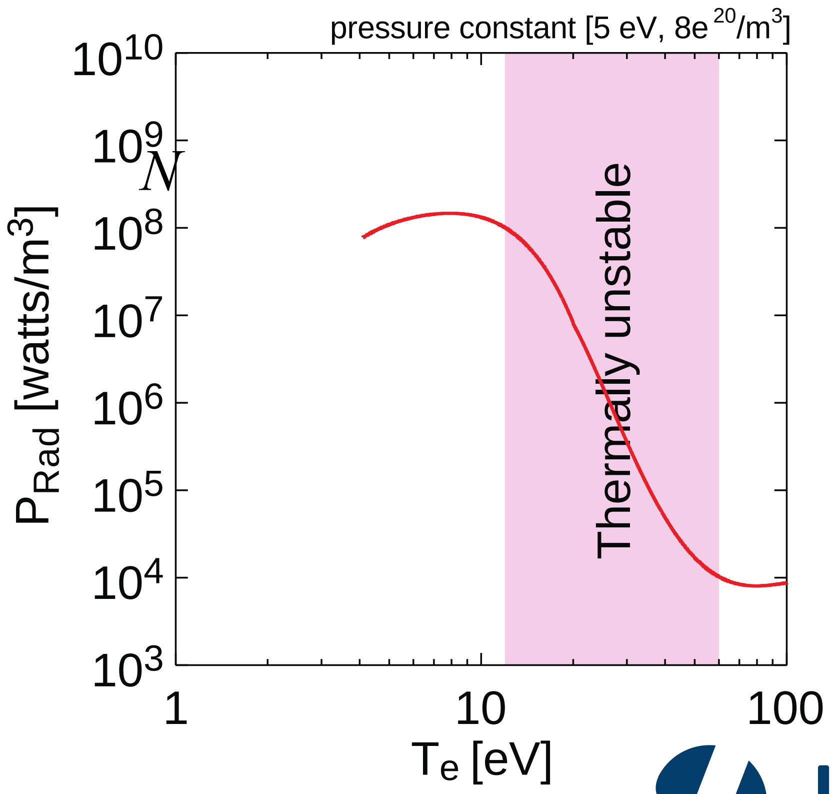
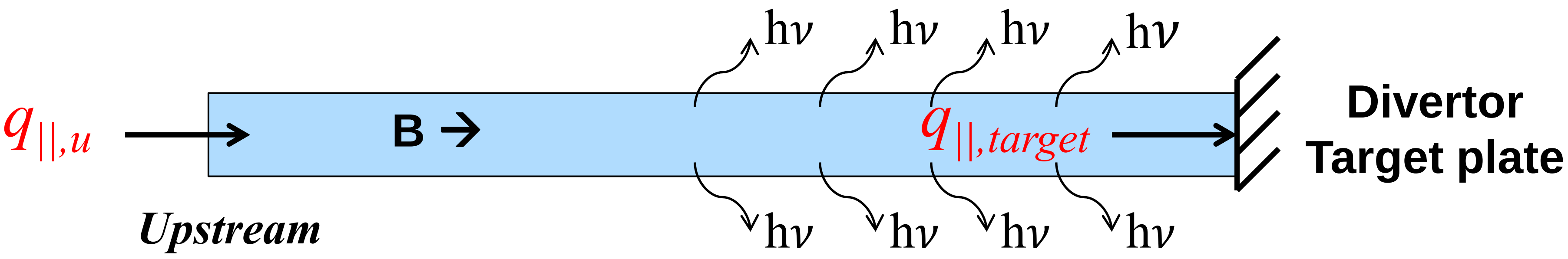


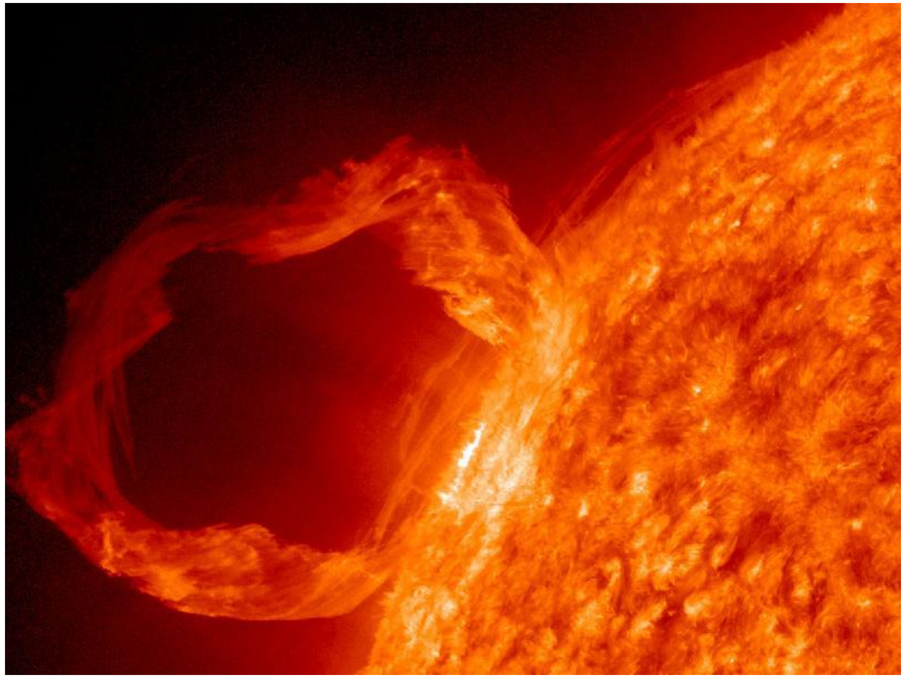
- Radiation power density ∝ $n_e n_Z \langle sv \rangle (T_e)$
  - For constant pressure ($n_e T_e$) and impurity fraction one can solve for the radiation density
  - The slope of the curve gives rise to a **'radiation condensation instability'** whereby as the plasma cools it radiates more and is cooled more (MARFE).
  - Low-Z, 'Seeded' impurities, such as Ne and N_2, are used



Step 2 – remove power from the plasma in the flux tube through impurity radiation

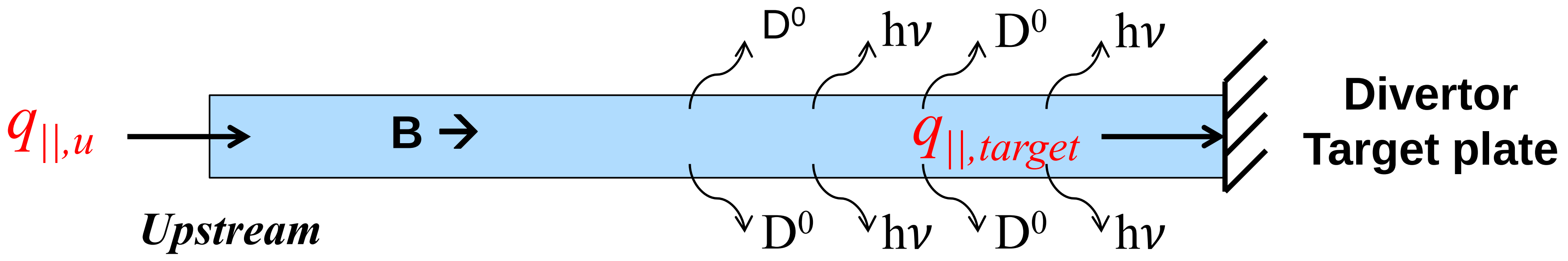


- Radiation power density ∝ $n_e n_Z \langle\sigma v\rangle(T_e)$
  - For constant pressure ($n_e T_e$) and impurity fraction one can solve for the radiation density
  - The slope of the curve gives rise to a 'radiation condensation instability' whereby as the plasma cools it radiates more and is cooled more.
- This 'instability' plays a central role in divertor physics as well as other aspects of tokamaks and solar phenomena (e.g. prominences[1], coronal rain[2])

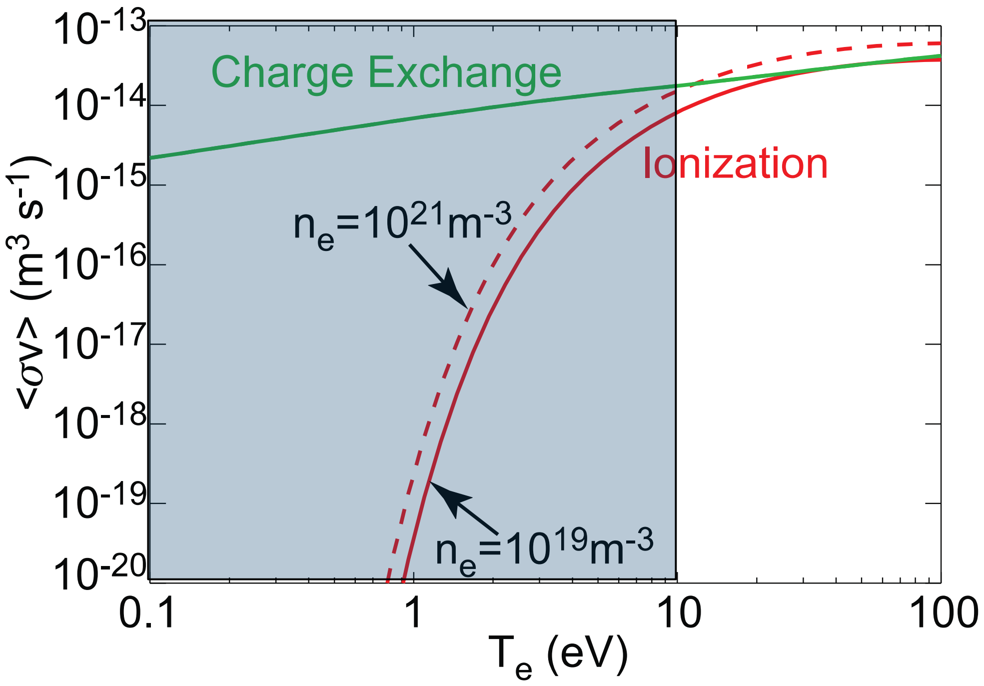


[1]e,g. A. De Groof, et al, A&A 443, 319–328 (2005)
[2]e.g. P. Antolin et al, **280** (2012) 457.

Step 3 – "Detachment"* Remove momentum from the plasma flow in a given flux tube (pressure is not constant)

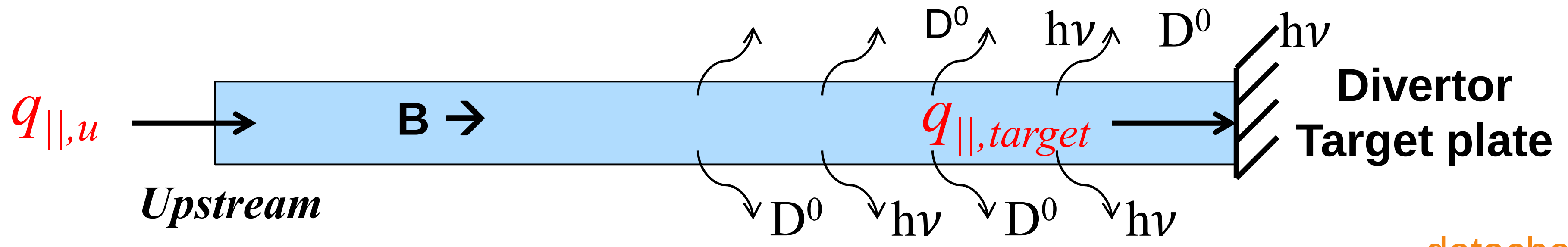


- Charge exchange reactions
  - D^0(cold)+ D^+(hot) -> D^+(cold) + D^0(hot)
  - *neutrals exchange momentum and energy with the plasma* – **pressure can drop**
  - Charge exchange must dominate over ionization ($T_e \leq 5$ eV) so that hot cx-neutrals can escape the plasma and deposit momentum/energy on surfaces

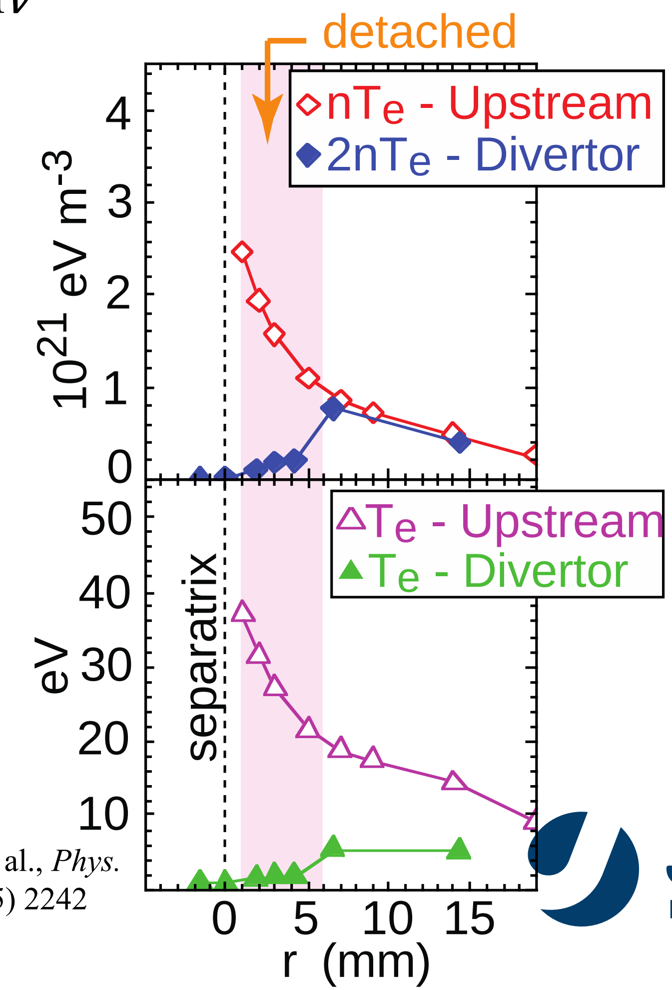


*The plasma can become so cold that the ionization region (around 5 eV) 'detaches' from the divertor plate, moving some distance upstream.

Step 3 – "Detachment": remove Momentum from the plasma flow in a given flux tube (pressure is not constant)



- Charge exchange reactions
  - D^0(cold)+ D^+(hot) -> D^+(cold) + D^0(hot)
  - neutrals exchange momentum and energy with the plasma – **pressure can drop**
  - Charge exchange must dominate over ionization ($T_e \leq 5$ eV) so that hot cx-neutrals can escape the plasma and deposit momentum/energy on surfaces
  - Both $p_{e,t}$ and $T_{e,t}$ drop leading to much larger drops in $q_{||,t}$ ($\propto P_{e,t} T_{e,t}^{1/2}$)
  - Reduces $\Gamma_{||}$ ($\propto p_e/T_e^{1/2}$) as well as $q_{||}$



B. LaBombard et al., *Phys. Plasmas*, **2**, (1995) 2242

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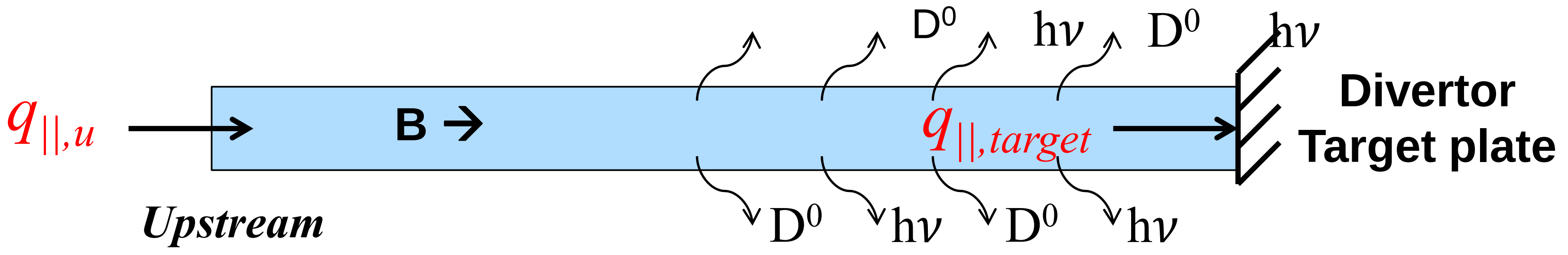
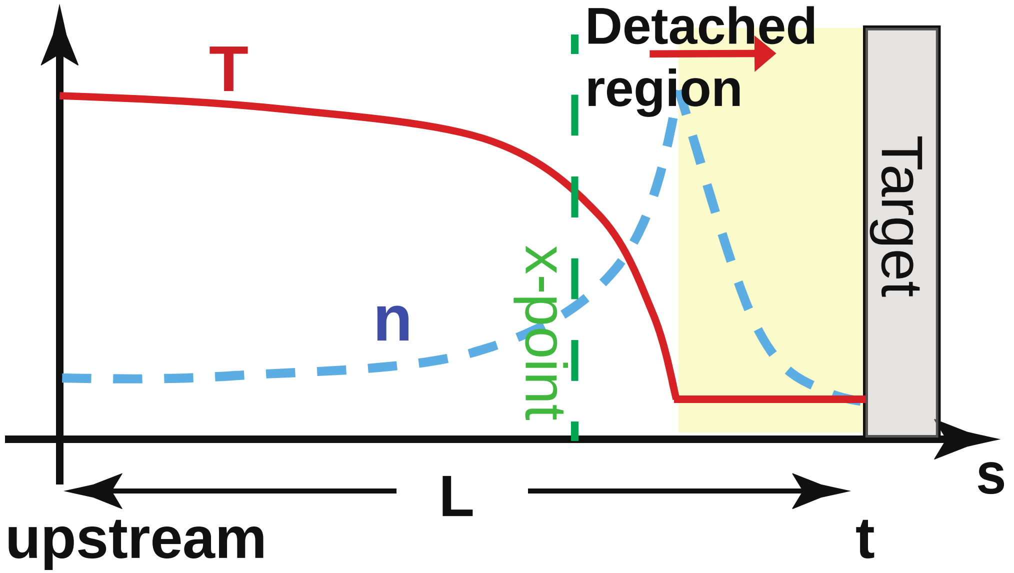
Step 3 – "Detachment": remove Momentum from the plasma flow in a given flux tube (pressure is not constant)

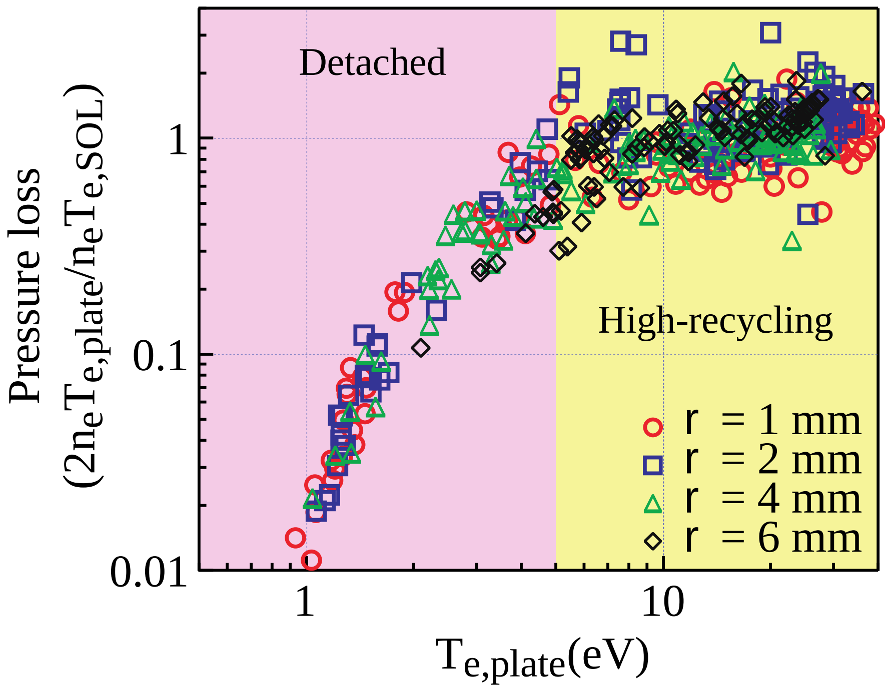


- Charge exchange reactions
  - D^0(cold)+ D^+(hot) -> D^+(cold) + D^0(hot)
  - neutrals remove momentum and energy from the plasma – **pressure can drop**
  - Charge exchange must dominate over ionization ($T_e \leq 5$ eV) so that neutrals can escape the plasma and deposit momentum/energy on surfaces
  - Both $p_{e,t}$ and $T_{e,t}$ drop leading to much larger drops in $q_{||,t}$ ($\propto P_{e,t}T_{e,t}^{1/2}$)
  - Reduces $\Gamma_{||}$ ($\propto p_e/T_e^{1/2}$) as well as $q_{||}$



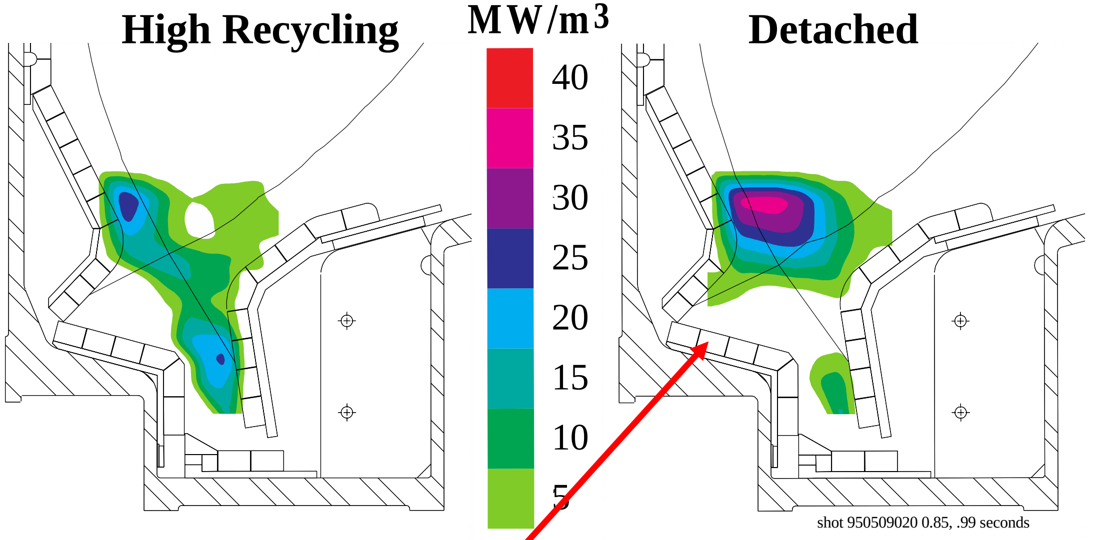
Transition of divertor from high-recycling to detached regimes leads to large changes in divertor conditions

B. Lipschultz et al., *Fusion Science Tech.*, **51**, (2007) 369



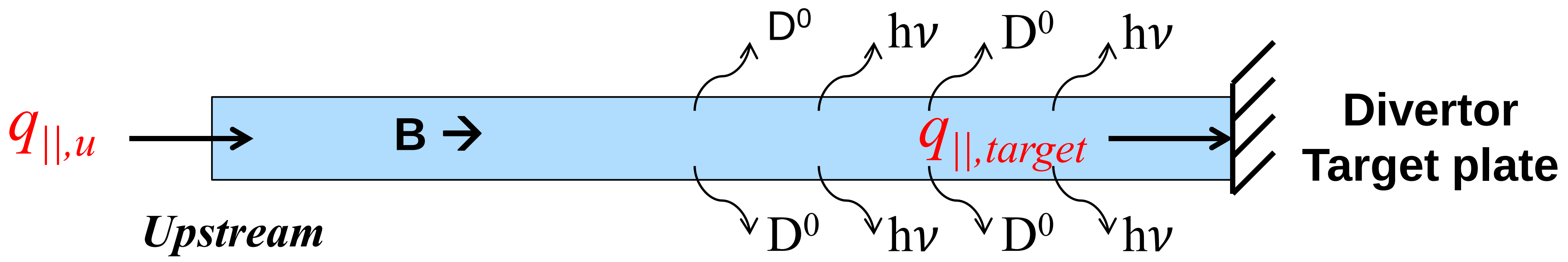
- Loss of pressure on a flux surface can reach x 100
  - Pressure loss averaged over plate ~ x 4 – x 10

- Detachment drawback – Cold region can expand to reach hot core
  - Cools the core plasma and easier for impurities to reach the hot core plasma

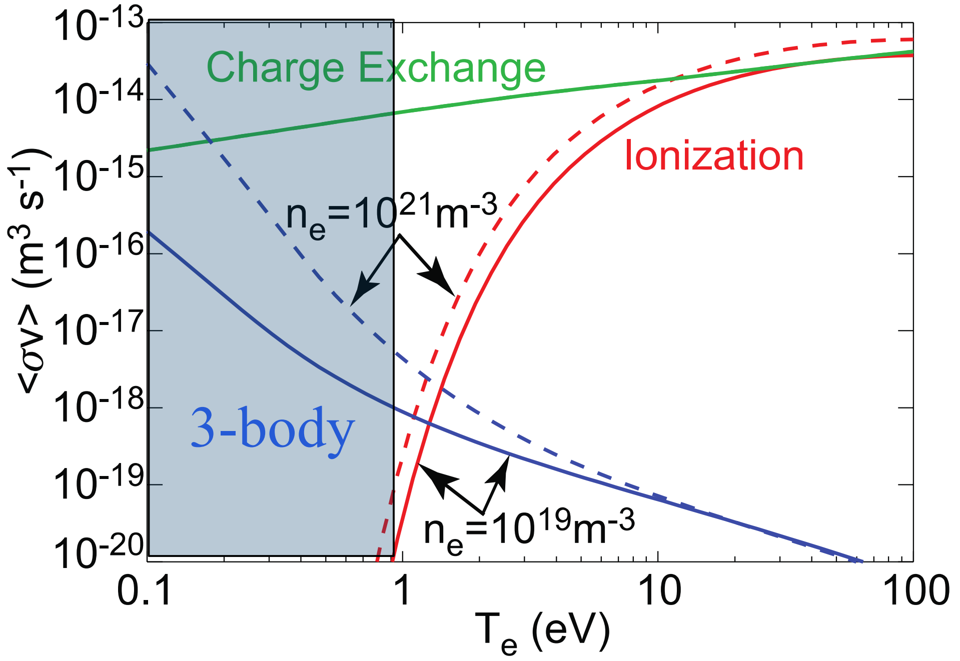


J. A. Goetz, B. Lipschultz et al., *Phys. Plasmas*, **3** (1996) 1908.

Step 4 – Convert plasma to neutrals (occurs with step 3)



- Three body recombination - neutrals remove momentum, energy and particles
  - $e + e + D^+ \rightarrow D^{0*} + e$
  - $q_{||}$ and $\Gamma_{||}$ drop even more
  - Requires temperatures ≤ 1 eV for recombination to dominate over ionization (for $n_H = n_p$)
- Note that 3-body recombination can also be a thermally-unstable process, if its associated radiation cools electrons left behind



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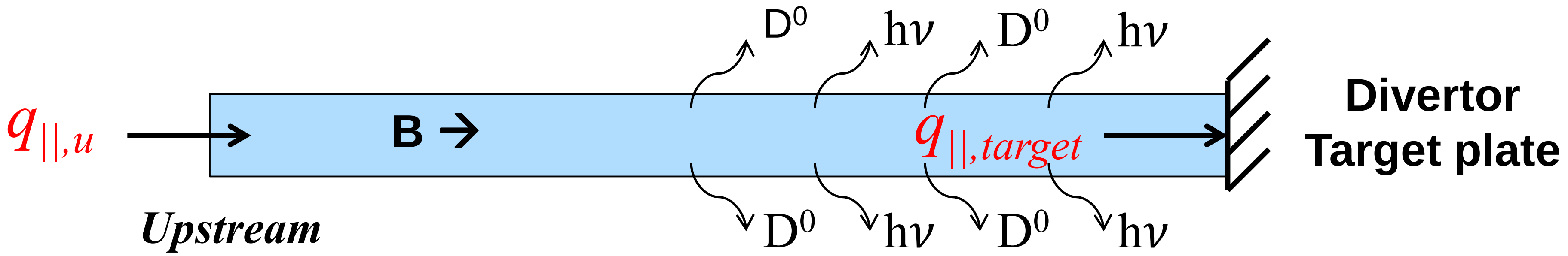
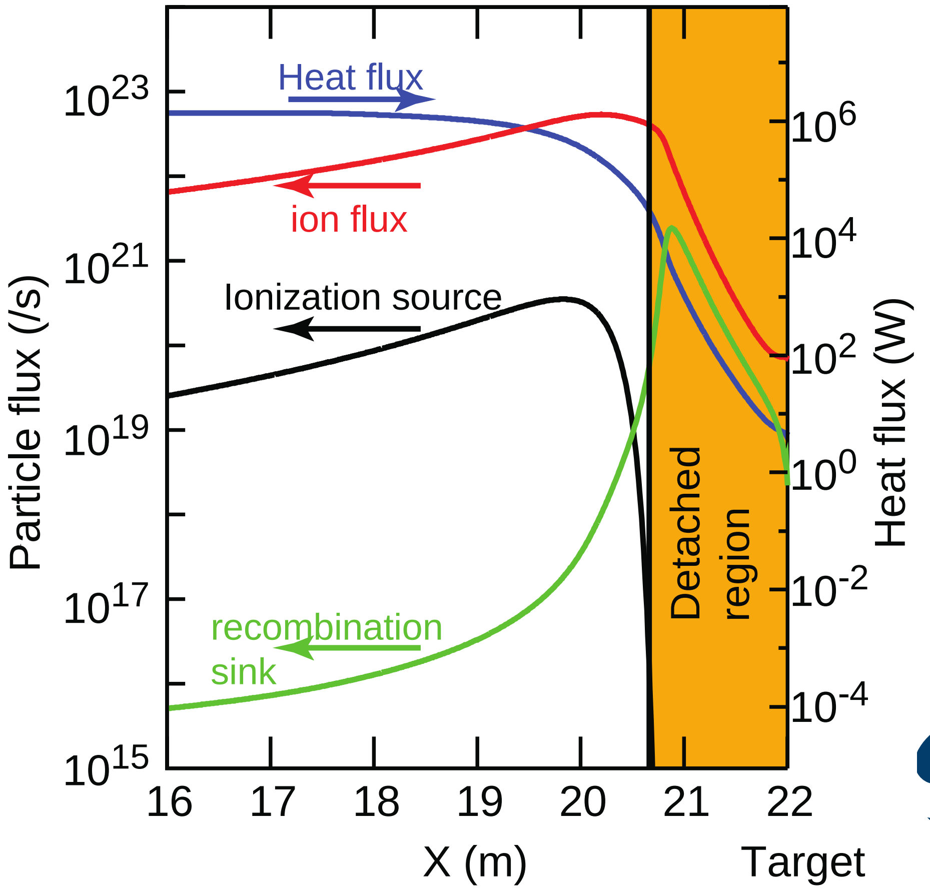
Step 4 – Convert plasma to neutrals (occurs with step 3)



- Mach number for plasma flow goes to ~1 at detachment front ('virtual target')
- Large density gradient driven by recombination.
  - Is a shock forming?
  - Double layer?
- Recombination energy heats electrons, associated radiation cools them. Net effect ?

S. Togo et al.,Plasma Fus. Res. **8** (2013) 2403096 .



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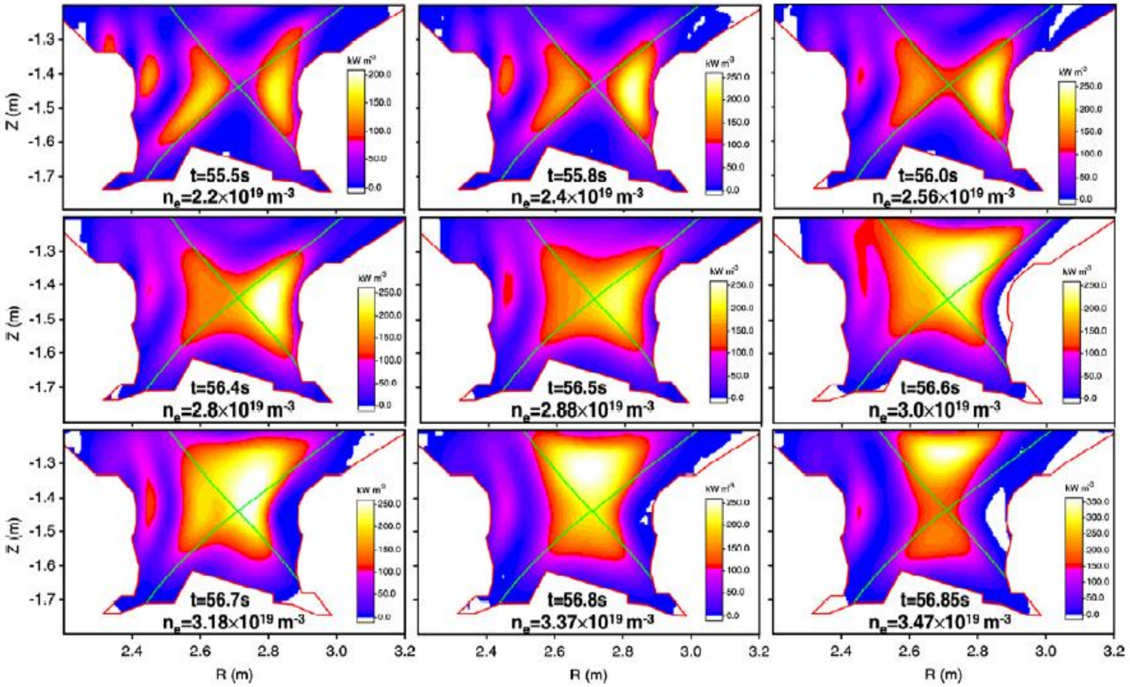
Full detachment is a problem

- 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 understood



JET, A. Huber, et al.

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

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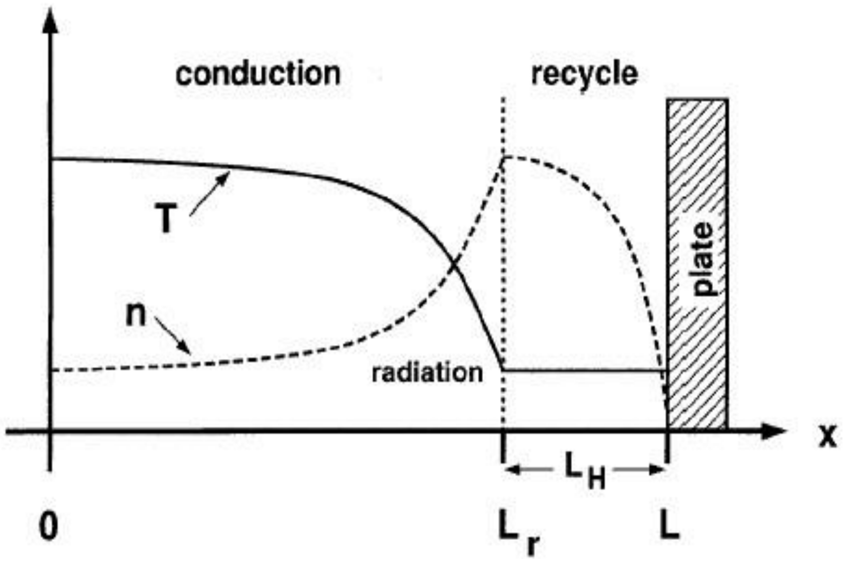
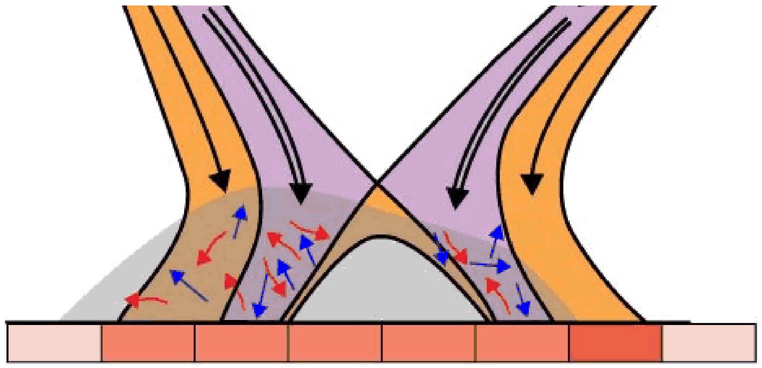
Full detachment vs. partial detachement

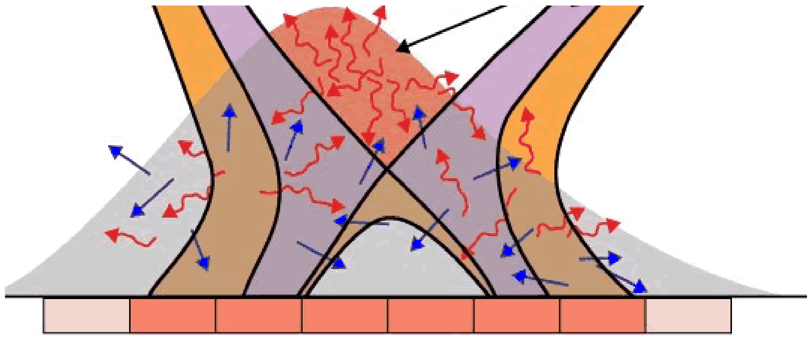


Important now:
CX: resonant charge exchange
ES: elastic scattering
Less important:
EI: electron impact ionization

Partially detached (CX-ES limited):
n_t decreases, $T_t < 5$ eV, $p_t << p_u$

Fully detached (radiation limited):
X-point MARFE





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Aside: Not just charge exchange: "Battle field" of hydrogen molecule: Two-electronic, strongly coupled potential-surfaces of H_3^+

$H^+ + H_2$ is the most fundamental ion-molecule system

We should know all about it P. Krstic, ORNL, US

Proton impact of molecule

$$p + H_2(v) \to p + H_2(v')$$

$$p + H_2(v) \to H + H_2^+(v')$$ Charge transfer

$$p + H_2(v) \to H + H^+ + H$$

$$p + H_2(v) \to H + H^+ + H^+ + e$$ Dissoc Double ion

$$p + H_2(v) \to p + H_2(n, v')$$ Exc. elec. vib.

Processes with molecular ion

$$H + H_2^+(v) \to H + H_2^+(v')$$
$$H + H_2^+(v) \to p + H_2(v')$$
$$H + H_2^+(v) \to p + H + H$$

Numerous other processes with molecules

$$H^- + H_2(v) \to H + H_2(v') + e$$
$$H + H_2(v) \to H + H_2(v')$$
$$H + H_2(v) \to H + H + H$$
$$H_2(v') + H_2(v'') \to H_2(v''') + H_2(v'''')$$
$$H_2 + H_2(v) \to H_2 + H + H$$

Creation of H_3^+

$$H_2^+(v') + H_2(v) \to H_3^+(v'') + H$$

H_3^+ Series of interesting reactions:

DE, DR, branching ratios with electrons

D, DCT with H

- "Interplay" of transport and inelastic processes
- Rotational analysis is missing
- Isotopic constitution: D_2, T_2, HD, HT and DT, sensitive on vib. energy levels

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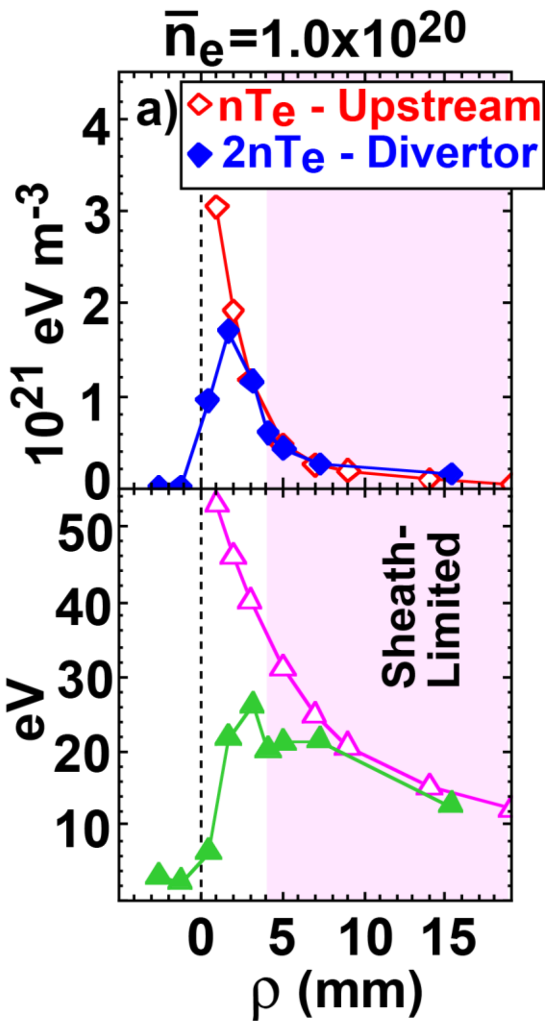
A. Summary: Divertor plasma physics plays an important role in reducing the divertor heat load

$$\Gamma_{||,target} = 0.5 n_{e,t} C_s \quad \propto \quad P_{e,t}/T_{e,t}^{1/2}$$
$$q_{||,target} = \gamma T_{e,t} \Gamma_{||,target} \propto \quad P_{e,t} T_{e,t}^{1/2}$$

- There are **three regimes** for flux tubes that connect the main plasma edge to the divertor target:
  - No gradients along **B** – 'sheath-limited' flow of heat and particles



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A. Summary: Divertor plasma physics plays an important role in reducing the divertor heat load

$$\Gamma_{||,target} = 0.5 n_{e,t} C_s \propto P_{e,t}/T_{e,t}^{1/2}$$
$$q_{||,target} = \gamma T_{e,t}\Gamma_{||,target} \propto P_{e,t} T_{e,t}^{1/2}$$

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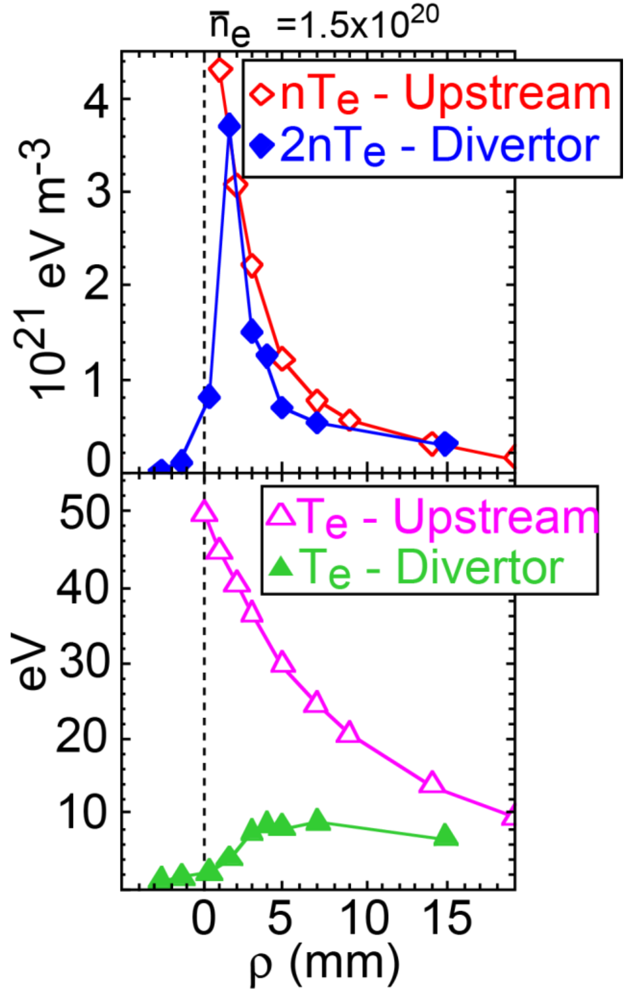
    - Pressure is constant along **B**
    - $\Gamma_{||,target}$ increases



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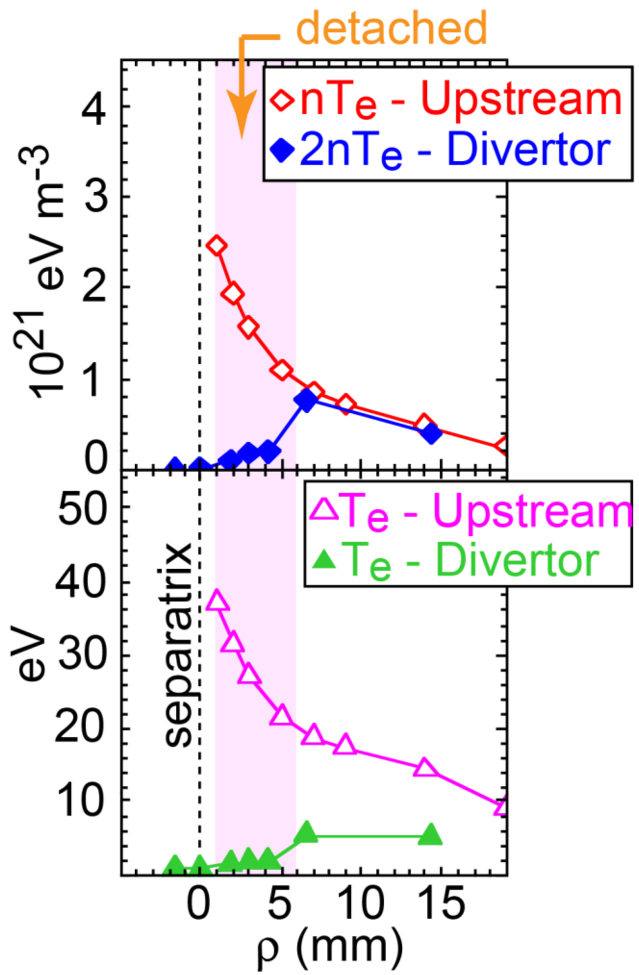
    - Pressure is constant along **B**
    - $\Gamma_{||,target}$ increases
  - Detached regime – pressure loss
    - Low $n_{e,target}$, $T_{e,target}$, $q_{||,target}$, $\Gamma_{||,target}$

Higher density



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A. Summary: Divertor plasma physics plays an important role in reducing the divertor heat load

- At detachment
  - Plasma pressure can drop by up to a factor of 100 within a flux tube.
  - Local T_e drops by a factor of ~10

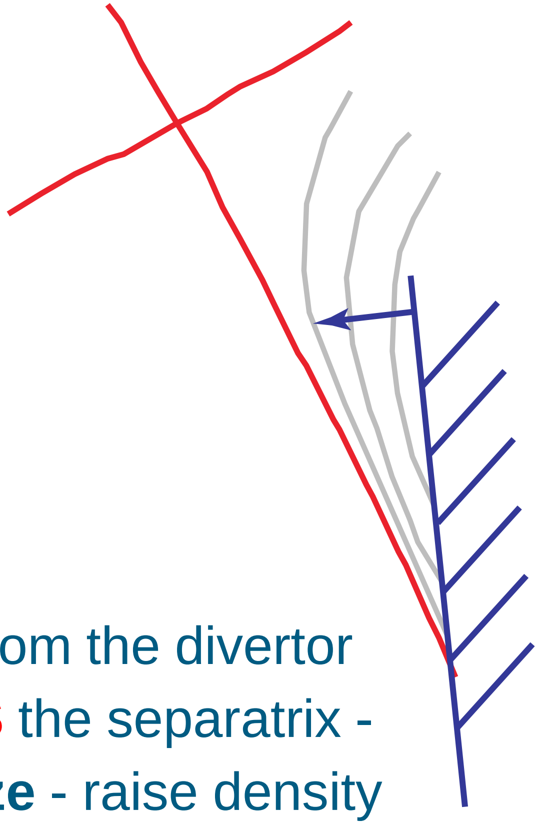
$$q_{\perp}[W/m^2] = \Gamma_{ions}\gamma_{sh}T_e[eV] \propto n_e T_e^{3/2} = P_e T_e^{1/2}$$

  - Total drop in $q_{\perp,target}$ ~ factor of ~100 – in theory, roughly that needed for DEMO after the $q_{||}$ reduction achieved through geometry!

    - New variations of divertor geometry are being studied to determine if power dissipation, He exhaust and core compatibility can be improved further.

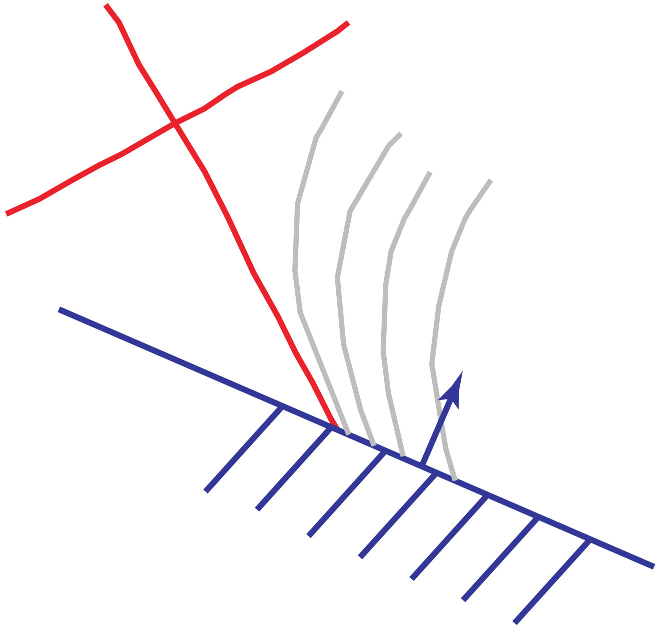
A. Geometry, and how it compresses neutrals, will make it easier or harder to achieve detachment

Vertical plate divertor



- Neutrals recycling from the divertor plate go **TOWARDS** the separatrix - **more likely to ionize** - raise density and lower temperature there - easier to detach

Open, or flat-plate divertor



- Neutrals recycling from the divertor plate go **AWAY** from the separatrix - **less likely to ionize** - don't raise $n_{e,sep}$

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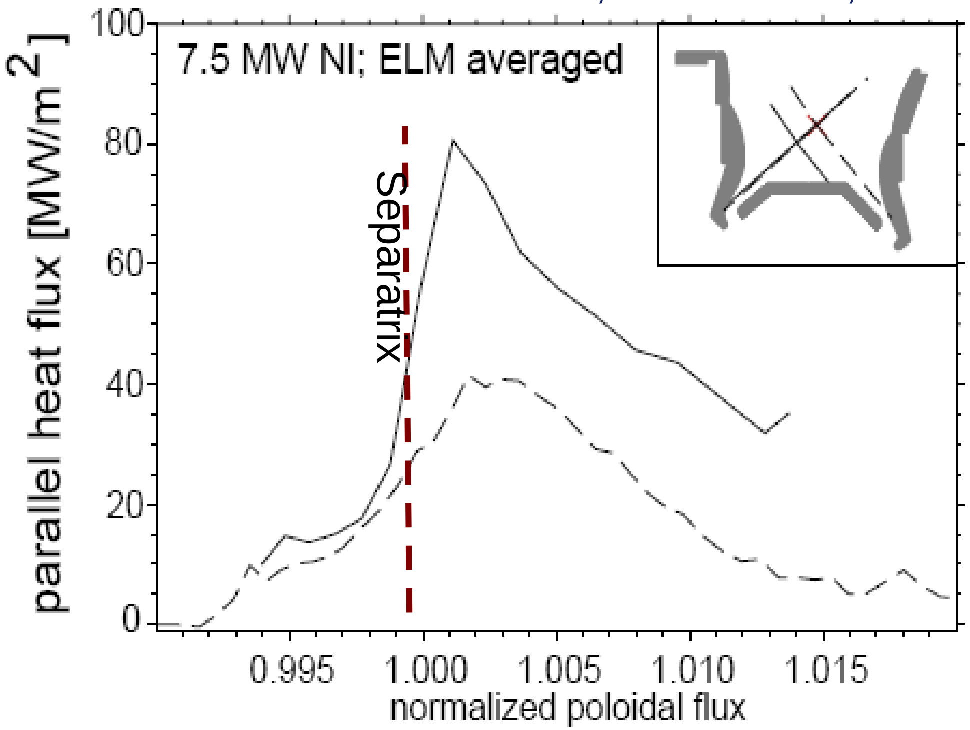
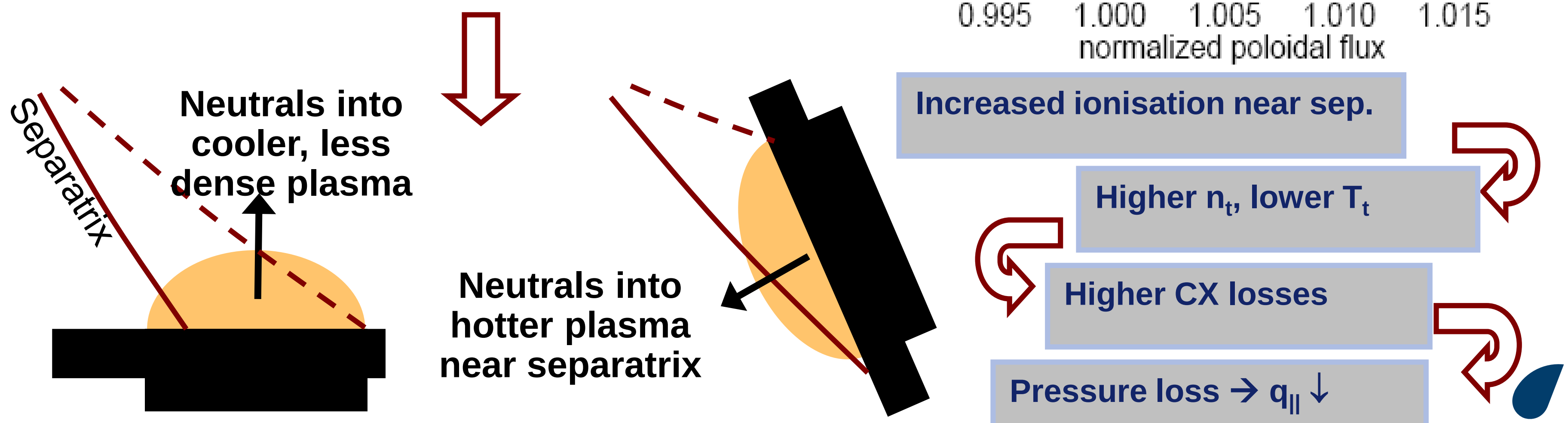
A. Geometry, and how it compresses neutrals, will make it easier or harder to achieve detachment

AUG, A. Kallenbach, et al.

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





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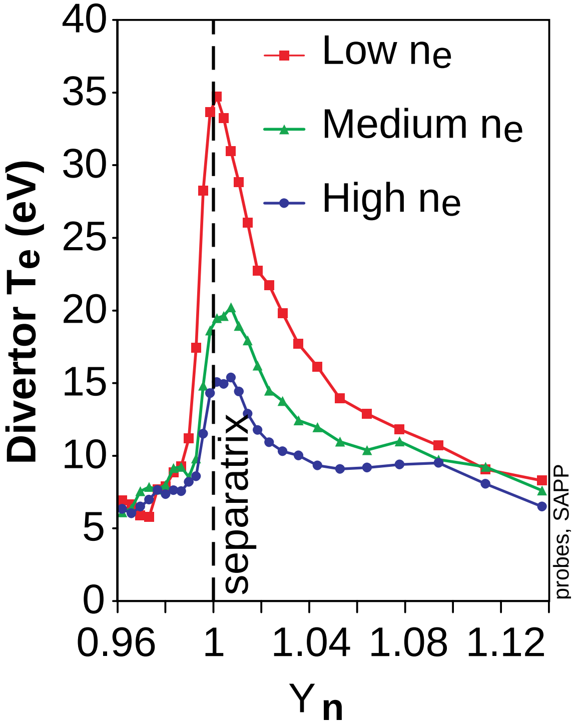
A. Geometry, and how it compresses neutrals, will make it easier or harder to achieve detachment

- **Flat-plate divertor** leads to temperature peaked on the separatrix - higher erosion.

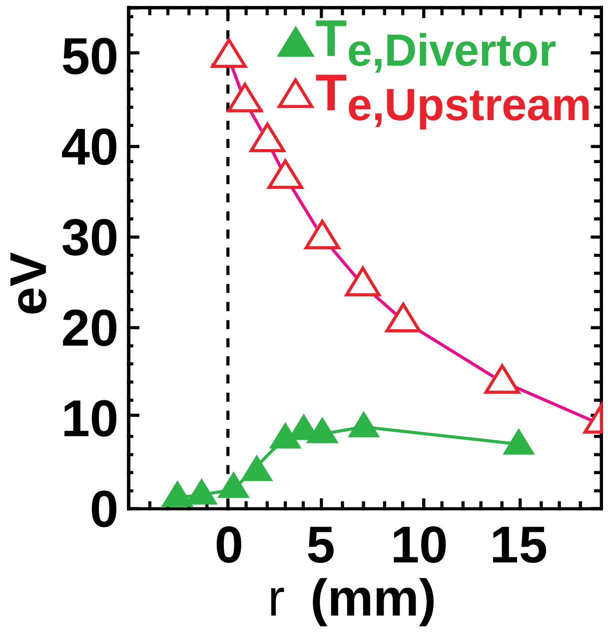


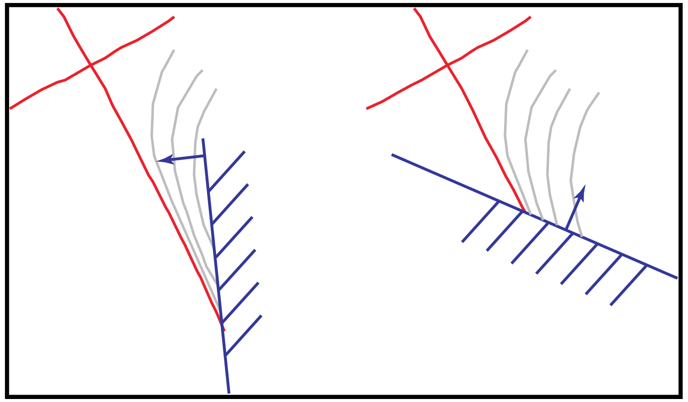
DIII-D:
horizontal target

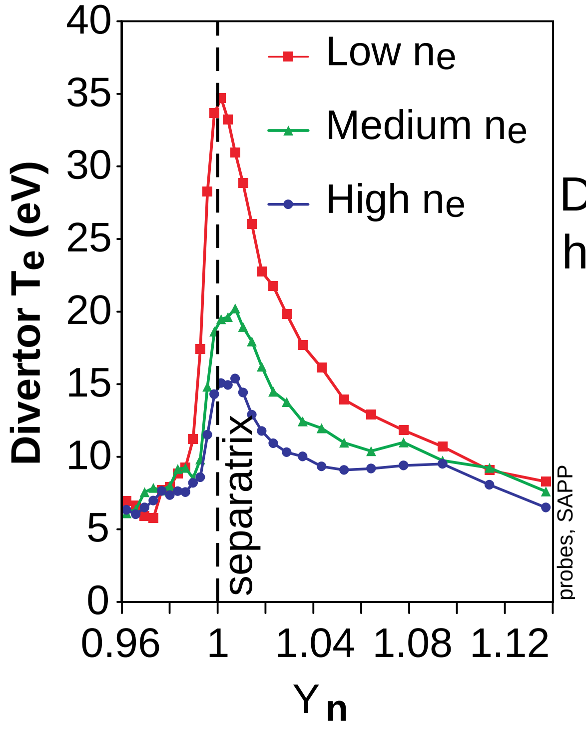
A. Geometry, and how it compresses neutrals, will make it easier or harder to achieve detachment

- **Flat-plate divertor** leads to temperature peaked on the separatrix - higher erosion.

C-Mod: vertical target







DIII-D: horizontal target

- **Vertical-plate divertor:** Enhanced ionization of neutrals on the separatrix leads to higher densities
  - Higher radiation $\propto n_e^2$
  - Lower T_e -> lower $q_{||}$ ($\propto P \cdot T^{0.5}$)
  - => access to $T_e \leq 5$eV and detachment at lower core n_e.

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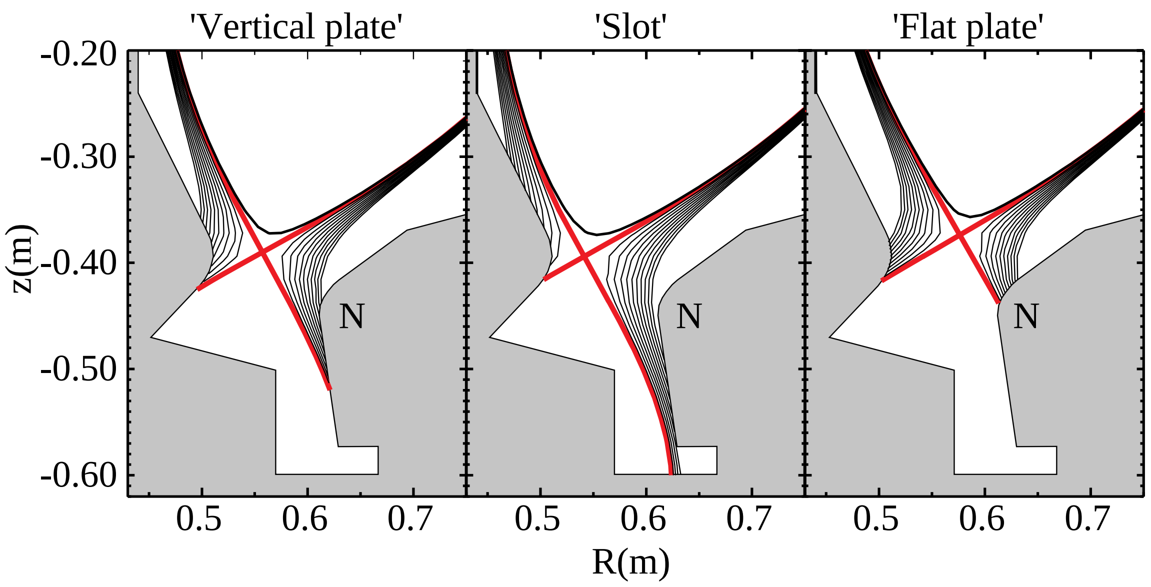
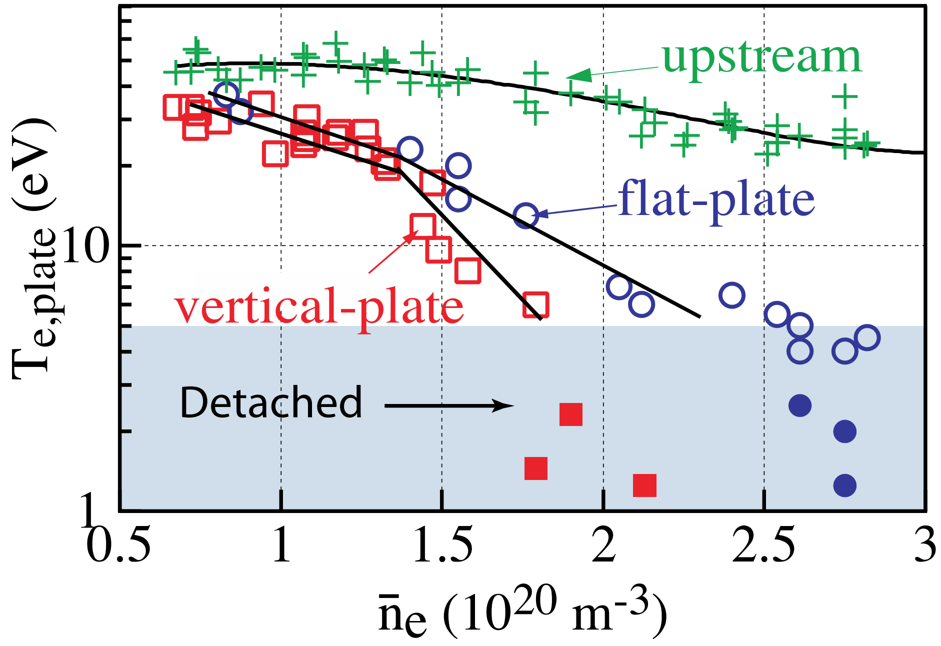
A. Geometry makes it easier to achieve detachment



- Vertical plate geometry, with its effect on neutrals, is the cause of the lower detachment threshold in C-Mod

- Compare different geometries in one tokamak



B. Lipschultz et al., *Fusion Science Tech.*, **51**, (2007) 369

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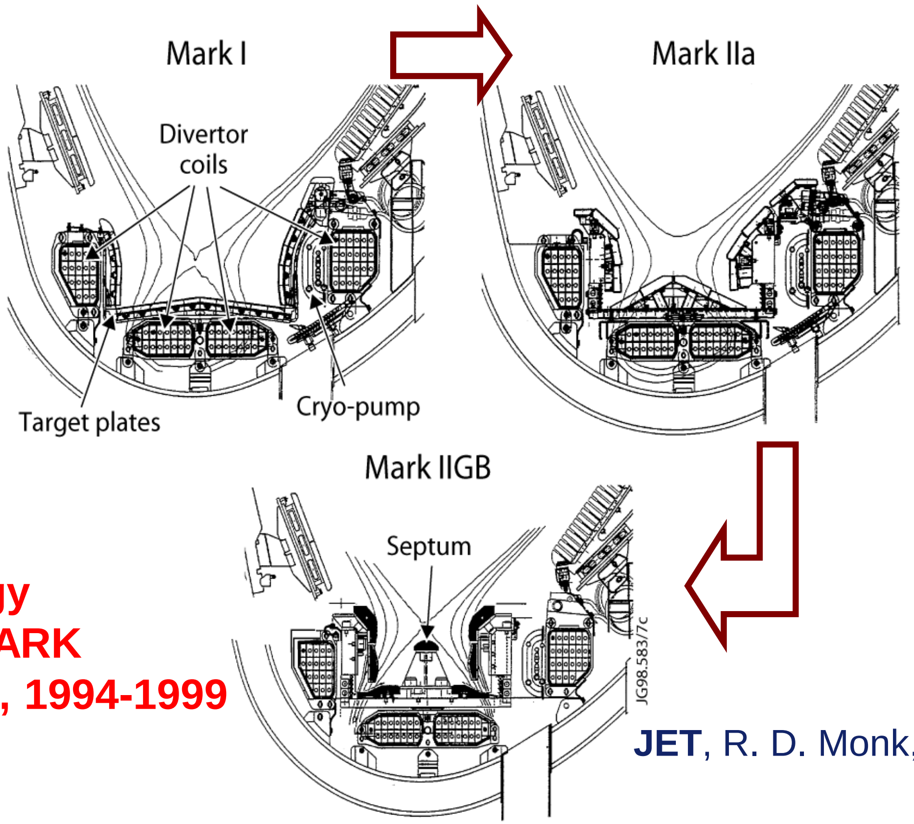
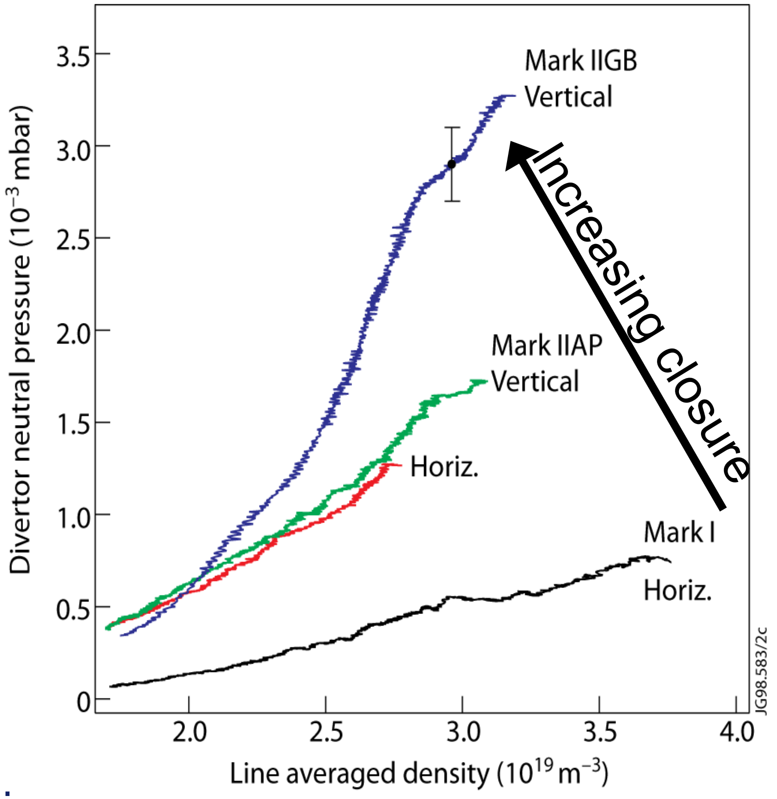
A. Divertore closure: intuition, guided by experiment and modelling, keep flexibility



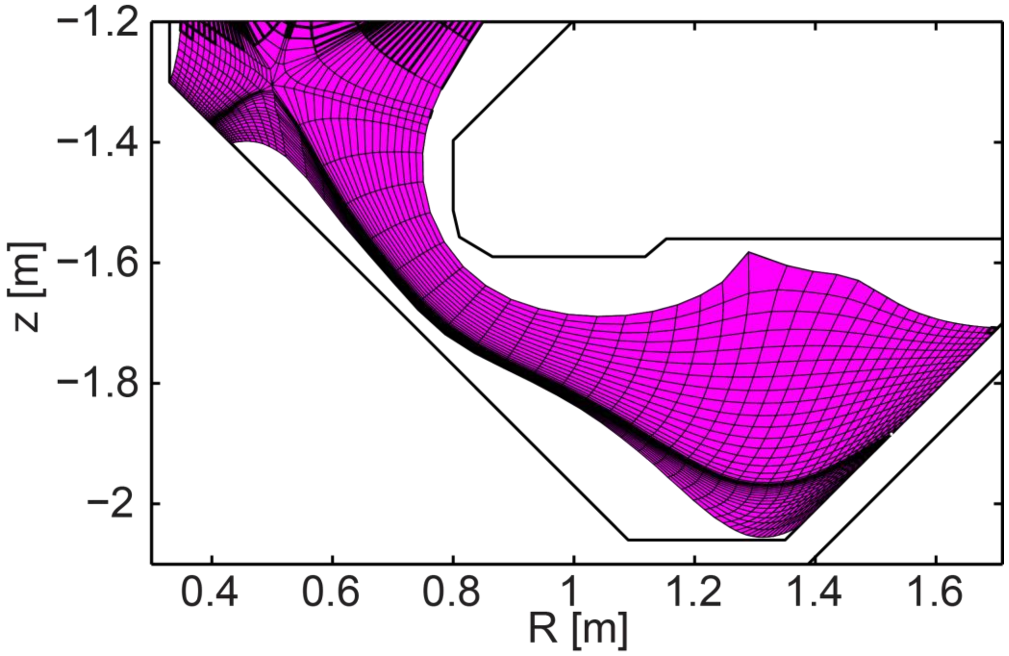
Genealogy of JET MARK Divertors, 1994-1999



JET, R. D. Monk, et al.

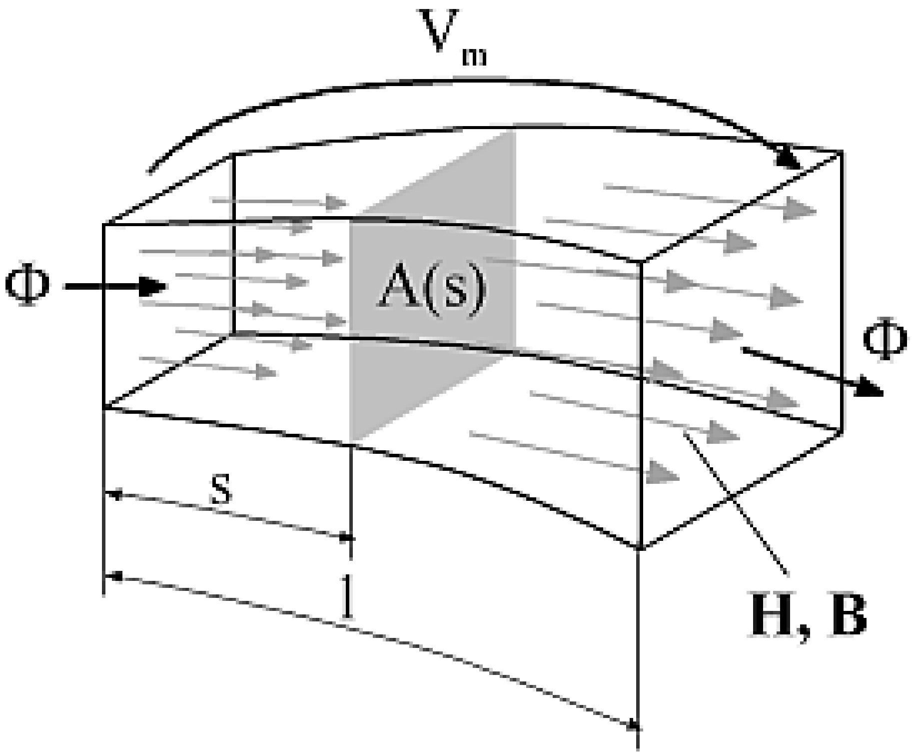
- Increased closure significantly improves divertor neutral pressure → increased neutral density (n_n), promoting earlier detachment
- Closing "bypass" leaks important for increasing n_n
- Divertor closure also promotes helium compression and exhaust – very important for ITER and reactors

A. The MAST-U tokamak provides another study of how geometry can be used to optimize the divertor and detachment



- The outer divertor target is moved to larger major radius, R
  - Maximizes target area lowering peak heat fluxes
  - Difficult for neutrals and impurities to escape the divertor
  - Toroidal (and total) field drops – B ∝ 1/R

A. Variations in the total magnetic field lead to variations in $q_{||}$, the power flow along a flux tube



$$\Phi = B \cdot A_{fluxtube} = const$$

$$B \cdot R \sim const \;\; \text{(tokamak)}$$

$$\Rightarrow \; A_{fluxtube} \propto R$$

Flux tube area increases as total B drops

$$q_{||} \cdot A_{fluxtube} = const$$

$$\Rightarrow \; q_{||} \propto \frac{1}{R}$$

$q_{||}$ also drops as total B drops and R increases
=> Easier to lower target T_e and access detachment

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Summary

- Through divertor geometry and plasma physics the divertor power load can be reduced below 10 MW/m^2, compatible with ITER operation
  - Not yet clear whether the process of detachment is enough to reduce power loads below engineering limits for a reactor

- Our current understanding shows that divertor surface erosion rate can be kept low enough to achieve < 1mm/yr for tungsten in the divertor
  - If one can keep the plasma temperature < 5 eV, compatible with detachment

- Operation of a reactor at high temperature (800-1000K) has positive effects on
  - Reduction of T retention
  - 'mending' of neutron damage
  - Reactor engine efficiency of conversion of heat to electricity
  - Note we don't yet have a demonstrated coolant that works at those temperatures

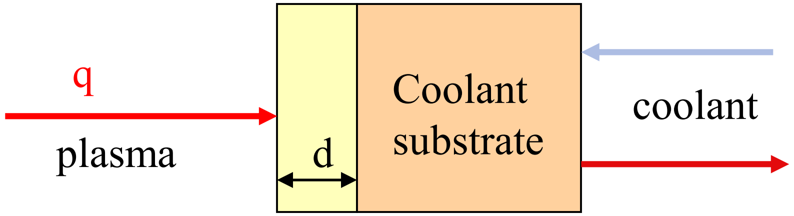
B & C challenges: Net erosion of the plasma facing components must be kept at extremely low levels

Understanding the physics of plasma surface interactions under reactor-like conditions

B. Net erosion of the tile (plasma-facing component: PFC) material must be kept at low values for **PFC lifetime** reasons
   - Only a fraction of the PFC thickness can be eroded per year - ~ 1mm
   - Even at that low rate the amount of material can be large

C. Net erosion of the PFC surface must be minimized for the **effect on the core plasma**
   - Radiation of energy, reducing the core temperature and the fusion reaction rate
   - Dilution of the core fuel

B. Heat exhaust, T_{melt}, material stress and heat conductivity set tile thickness (and erosion rate) (lifetime, availability of reactor)



- Material choices of refractory metals (W, Mo) or graphite.

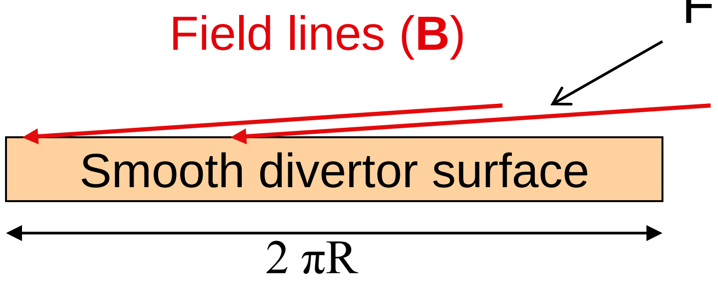
$$d_{tile} = k_{plate} \frac{(T_{\max, surface} - T_{coolant})}{q_{target}}$$

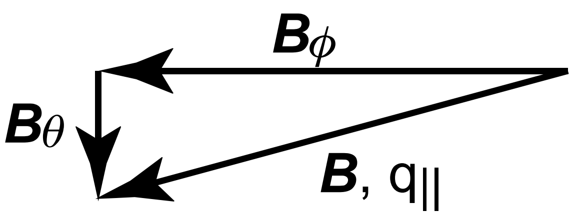
- Plasma facing surfaces must be thin to get the heat out without high front surface temperatures
- But also allowing 2-3 mm erosion before replacement. This forces the erosion rate to be very low ~ 1mm/year

Q_{target}=2MW/m^2 $\Delta T \sim 1000$K	Tungsten	CFC carbon
κ (W/m/K)	150	300
$T_{melt,sublime}$ (K)	3700	3900
d_{tile}	$\leq$ 1 cm	$\leq$ 2 cm

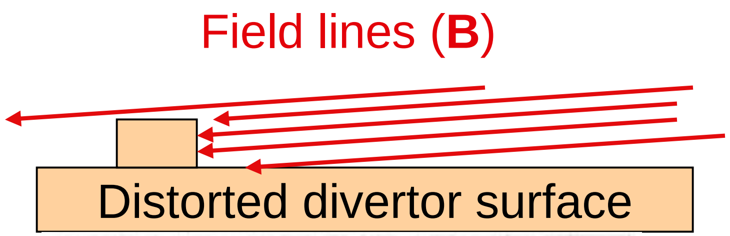
B & C challenges: The surface must be kept uniformly smooth and no ‘edges’ as one source of melting

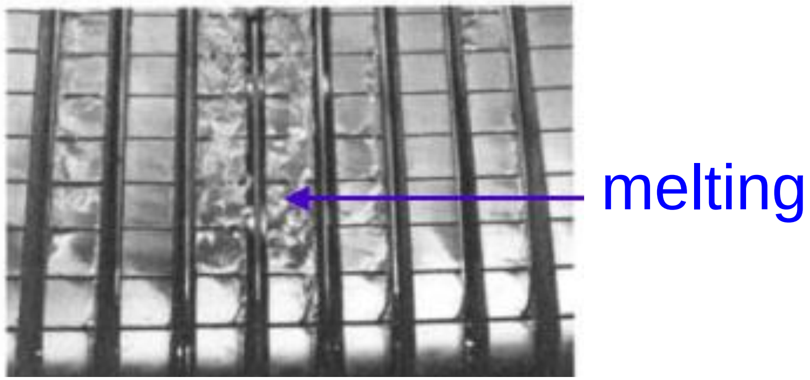
Field line angle to surface ~ 2.5 degrees





$$q_{\perp,\mathrm{target}} = q_{//}\left(\frac{B_p}{B_\phi}\right)_{divertor} \quad @q_{//}\,0.05 \approx 10\,\frac{MW}{m^2}$$



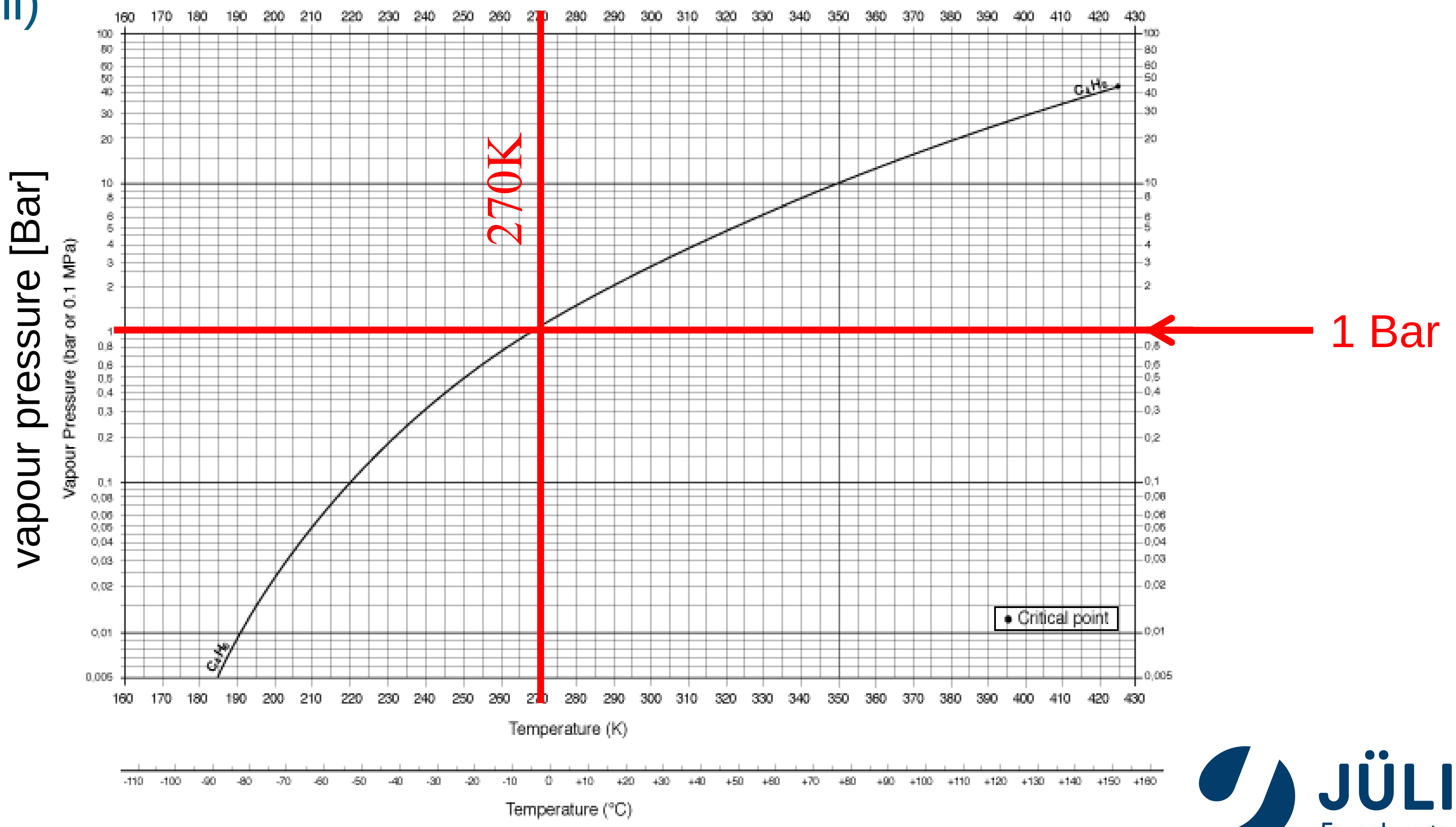


JET tiles, prior to first divertor installation

- BUT – surface distortion leads to perpendicular edge. This can receive $q_{//}$ ~ 100 MW/m² → melting
  - Effect on PFC lifetime
  - Also can lead to boiling coolant
- Cannot allow for such problems: **just: DO NOT MELT**

B & C challenges: Evaporation is the next concern for eroding the surface

- Common example rubber (butadene)
  - 270 K leads to atmospheric (Bar) vapour pressure – large erosion rate of atoms (causes the smell)



B & C challenges: Refractory materials have very low vapour pressure

- Metals such as tungsten (W) have much lower vapour pressure – not reaching atmospheric pressure till > 4000K => lower erosion rate

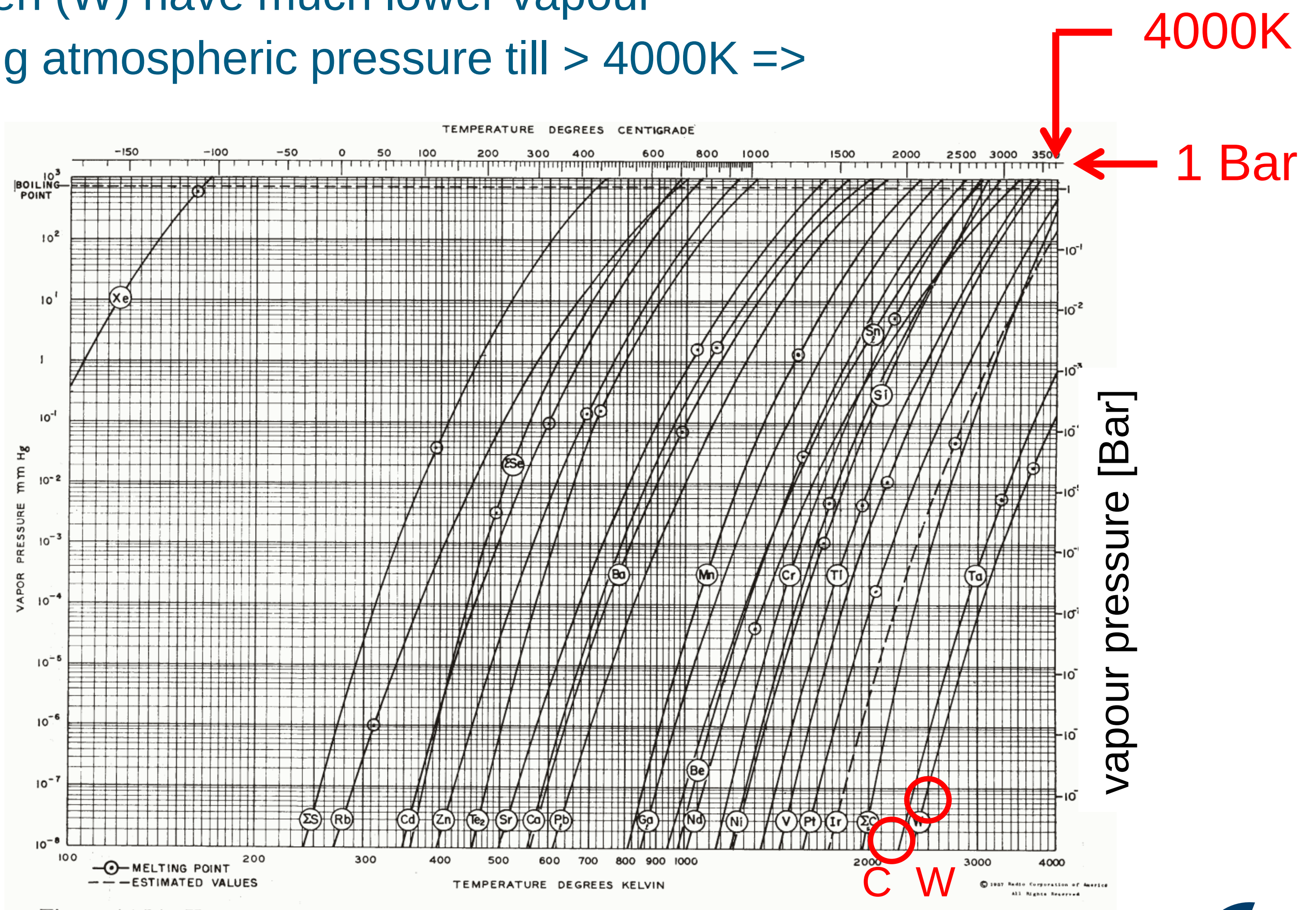


Figure A1(b). Vapor pressure curves for the more common elements (cont.). After Honig (Ref. 5:14). (Courtesy RCA Laboratories.)

Temperature (K)

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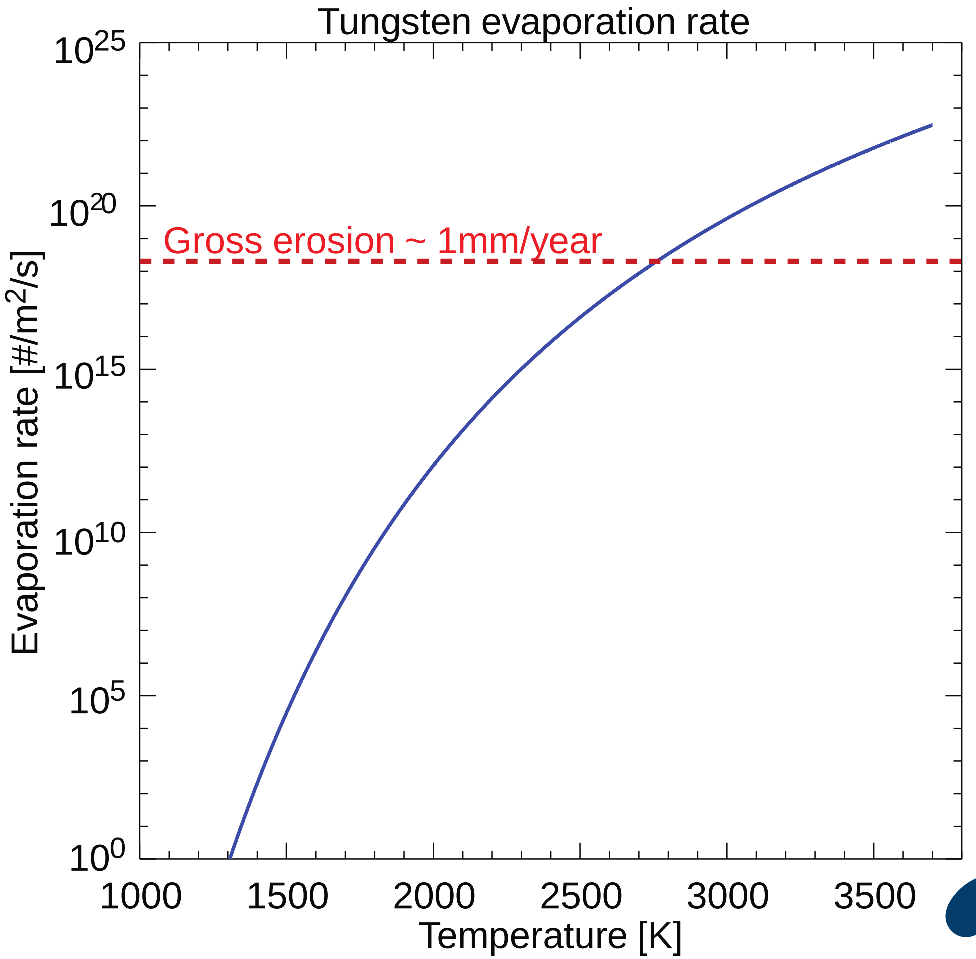
B & C challenges: Evaporation is not a concern at reasonable surface temperatures

- The vapor pressure of tungsten, P[Torr] ~ $5.6 \times 10^{10} \times 10^{-\{45385/T[K]\}}$
- The resultant rate of tungsten atom loss is $dN_W/dt \sim 1 \times 10^{24} \times P[Torr]$

  - Staying well below ~ 2700K (as planned) will keep the gross tungsten loss rate insignificant
  - Given active cooling of all components this should not be an issue
    - From the materials point of view, operated W above the DBTT (700C) and below the recrystallisation temperature (1200-1400C) → good to minimize evaporation



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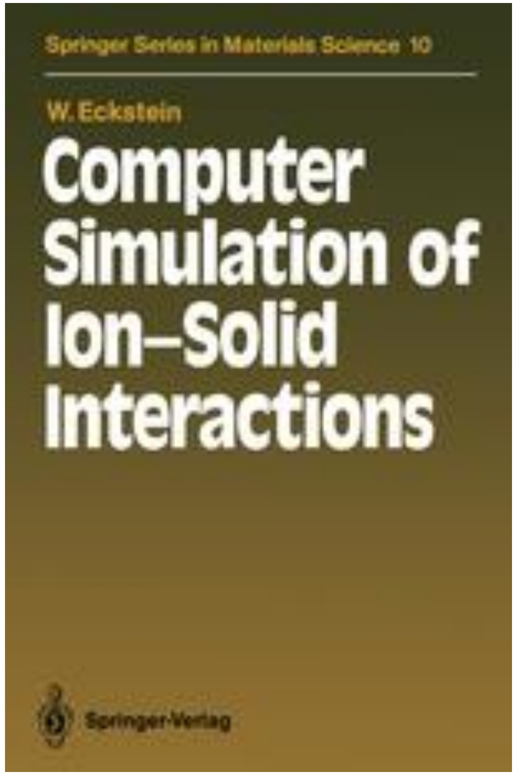
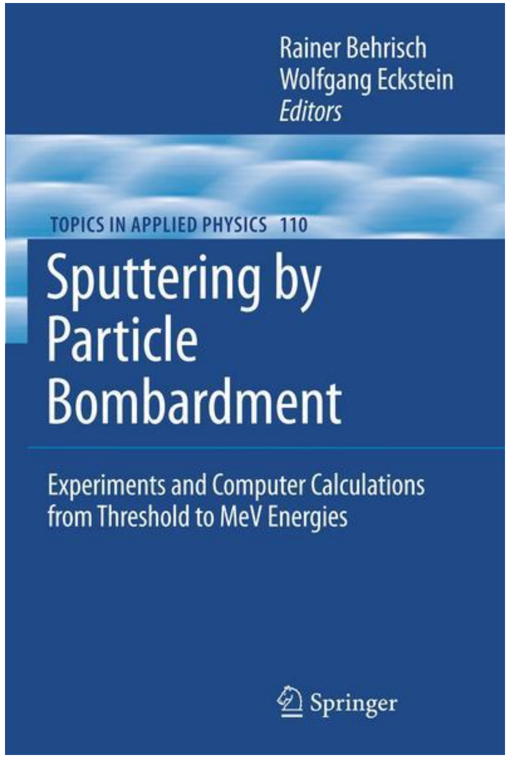
B & C challenges: 'Physical sputtering' is the main concern for steady state erosion

- An ion, incident on the surface with enough energy and mass, will lead to ejection of a surface atom





by W. Eckstein.
The classical monograph on physical sputtering, Springer 1991
Softcover reprint of the original 1st ed. 1991 edition
(November 22, 2011)

State of the art, advanced topics, Springer 2007

‘Physical sputtering’ is the main concern for steady state erosion

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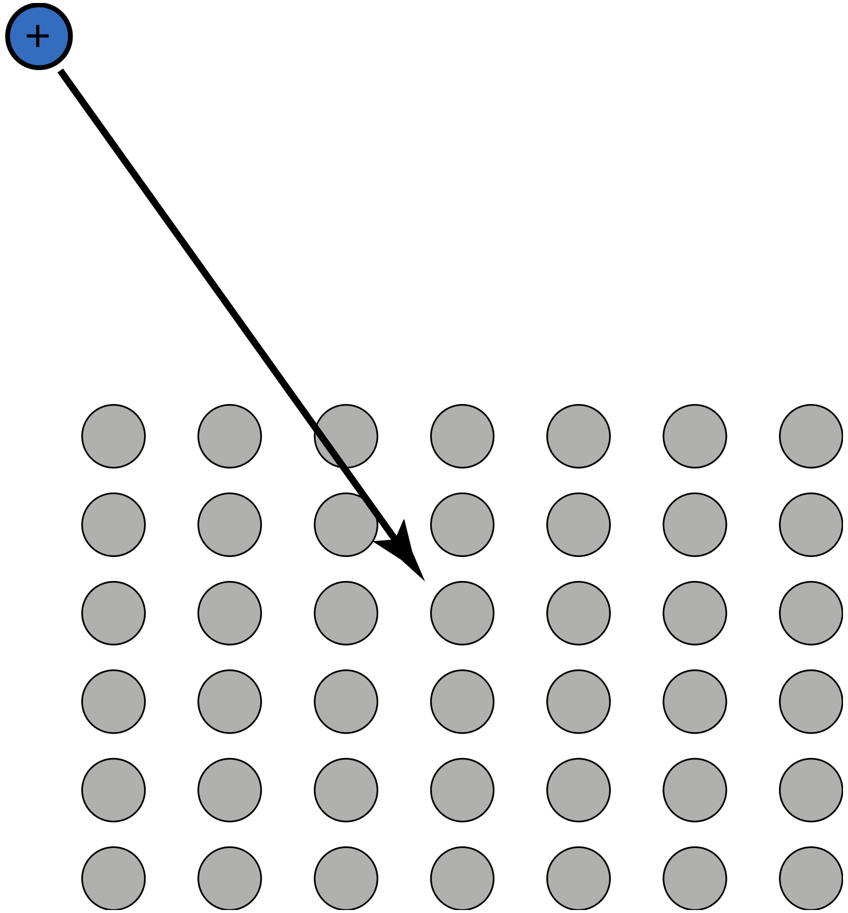

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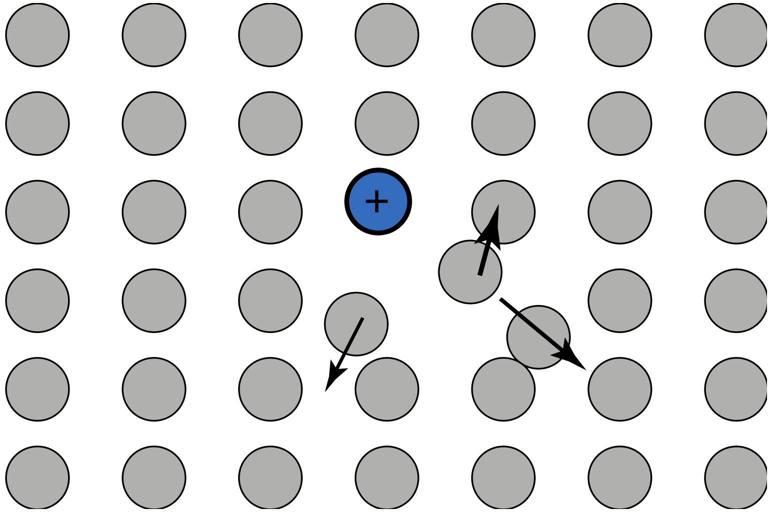

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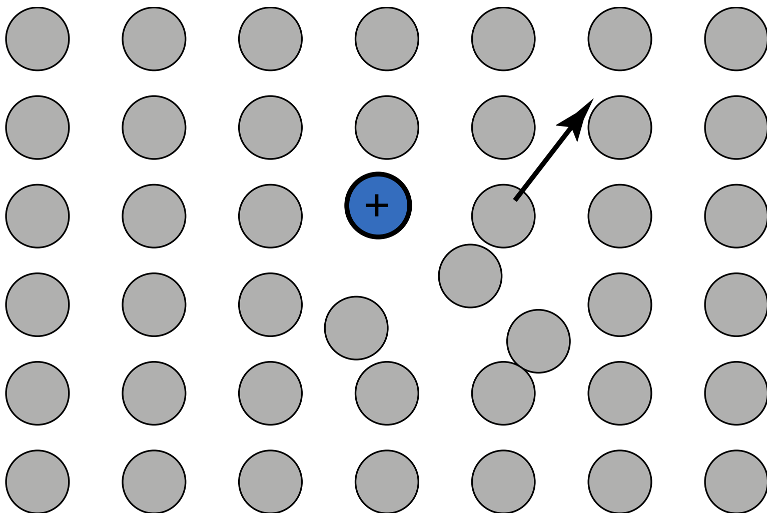

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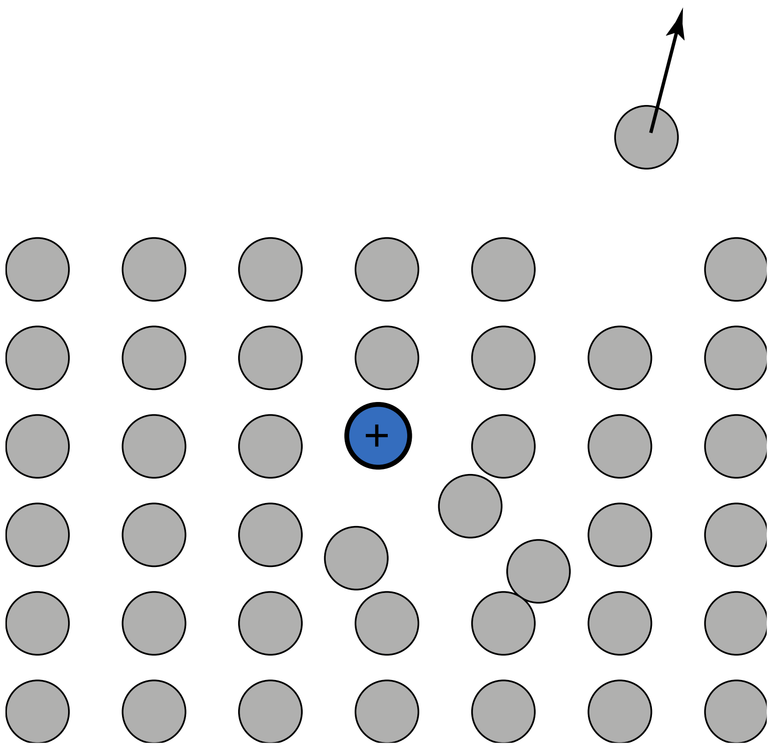
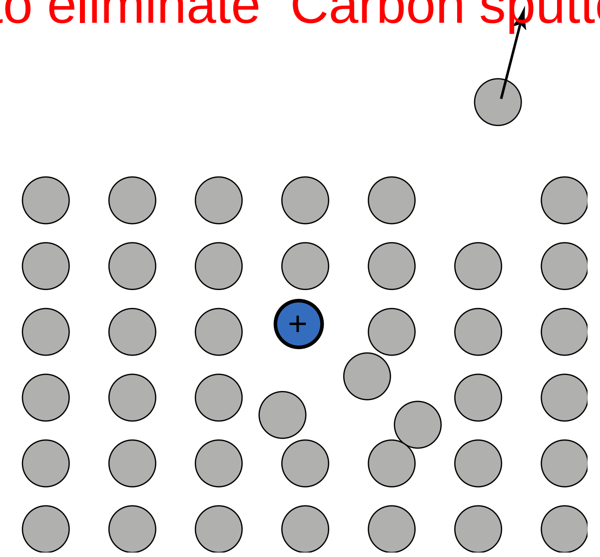
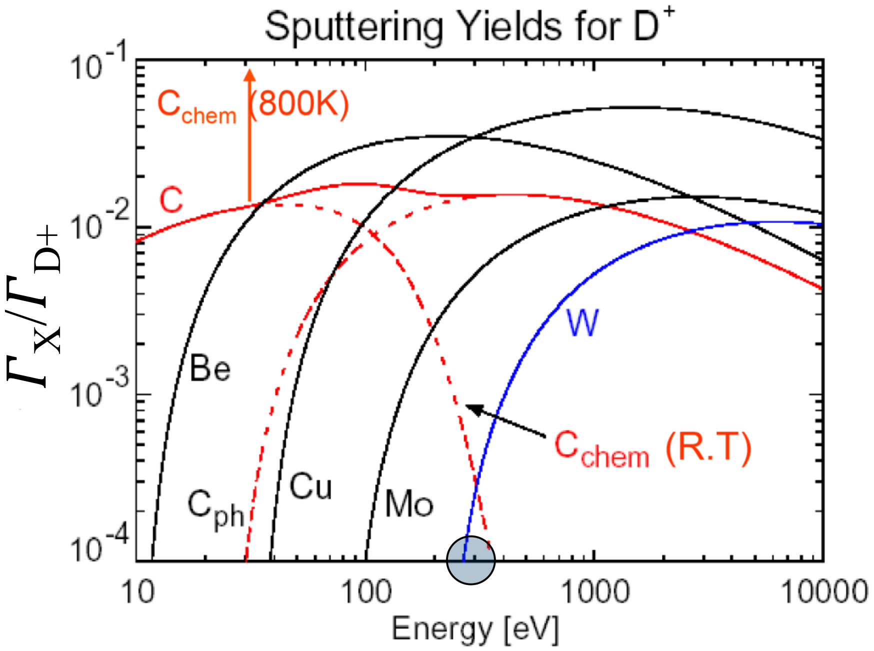


'Physical sputtering' is the main concern for steady state erosion

- An ion, incident on the surface with enough energy and mass, will lead to ejection of a surface atom
  - A lattice atom can leave the surface if it has more energy than the surface binding energy
  - Operation of a detached divertor, discussed earlier, lowers the incident ion energy, lowering the ion (D^+, T^+, Z^+) energy below the sputtering threshold -> no erosion!
    - That increases lifetime of the surfaces and reduces 'pollution' of the core plasma
  - Carbon has an additional "chemical sputtering" channel => it is essentially impossible to eliminate Carbon sputtering





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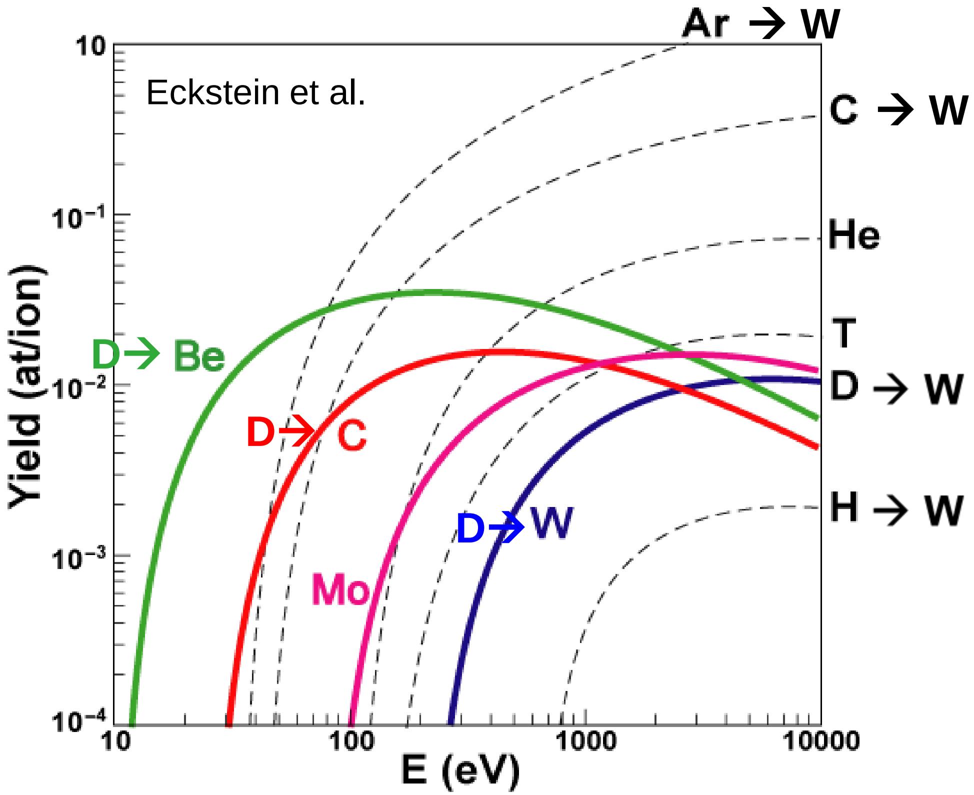
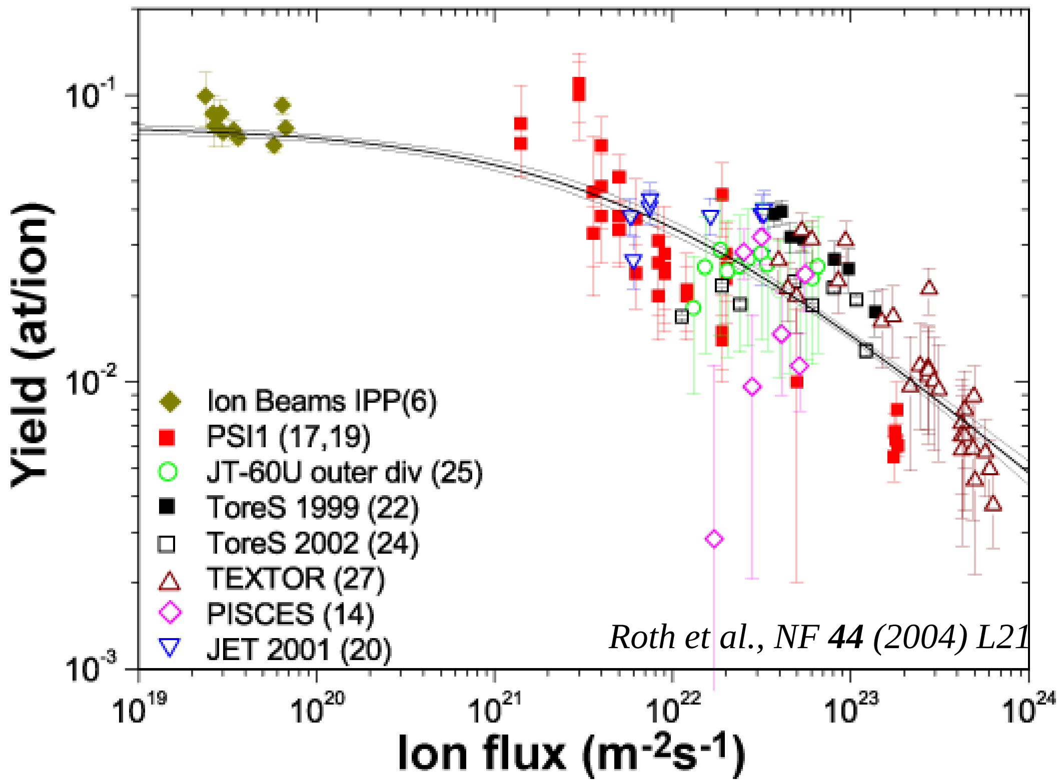
Physical vs chemical erosion

Physical sputtering yield…



increases with projectile energy and mass, while decreasing with target (PFC) material atomic mass

Chemical erosion yield (D on C)…



decreases with D ion flux and is sensitive to C target temperature

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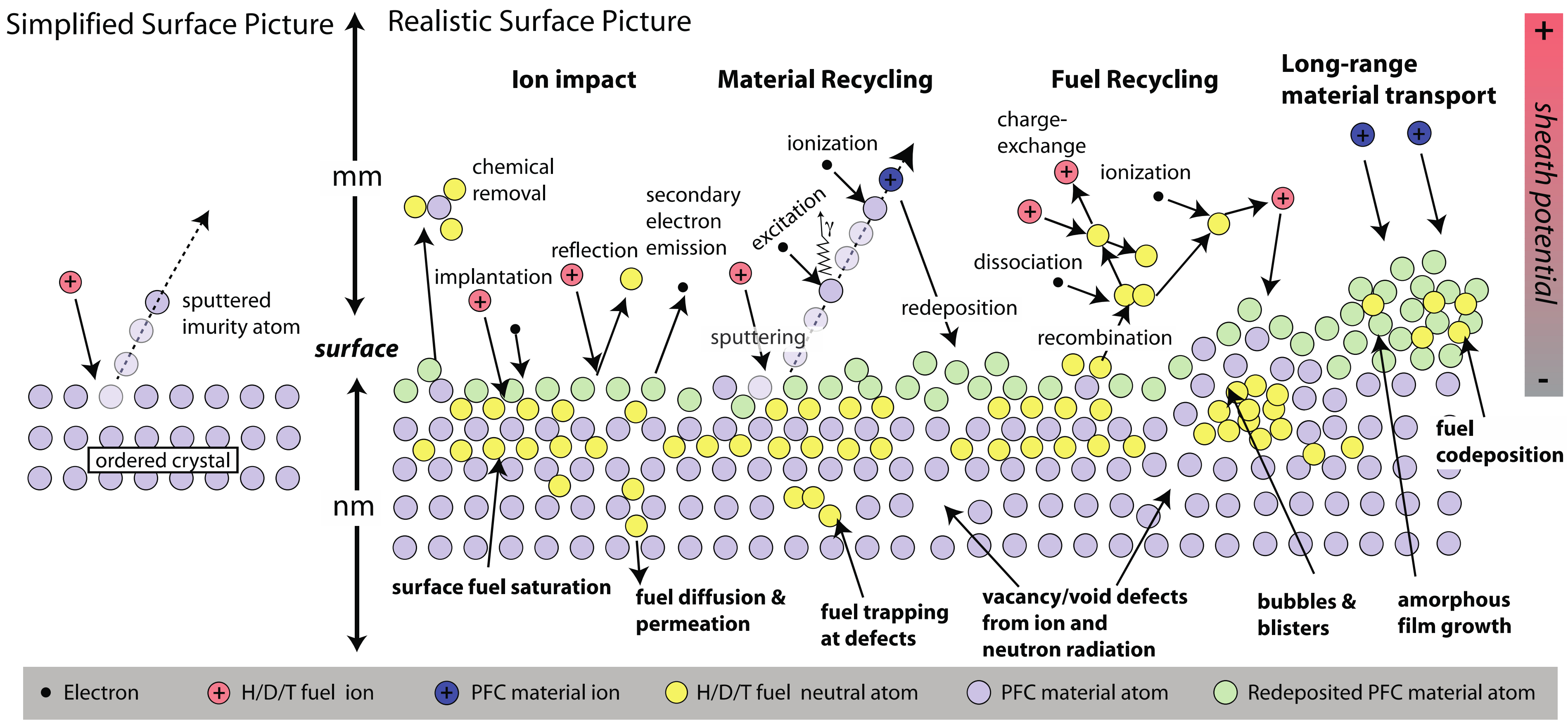
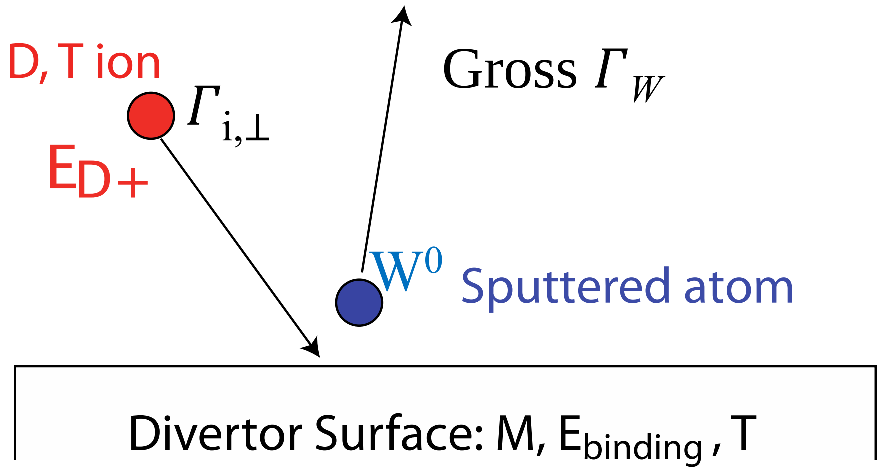
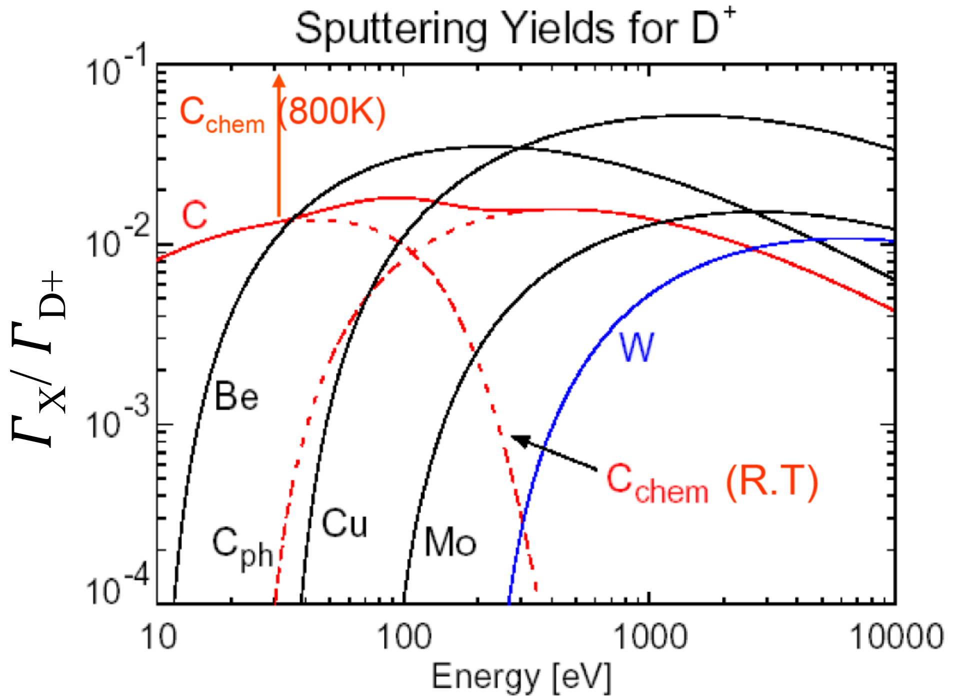
A more realistic picture:
→ complex bookkeeping tools, aka: “edge codes” (SOLPS-ITER)



B. Also of importance is to understand and take into account '**prompt redeposition**' of the sputtered surface atoms

- We have been talking about the gross erosion rate through physical sputtering





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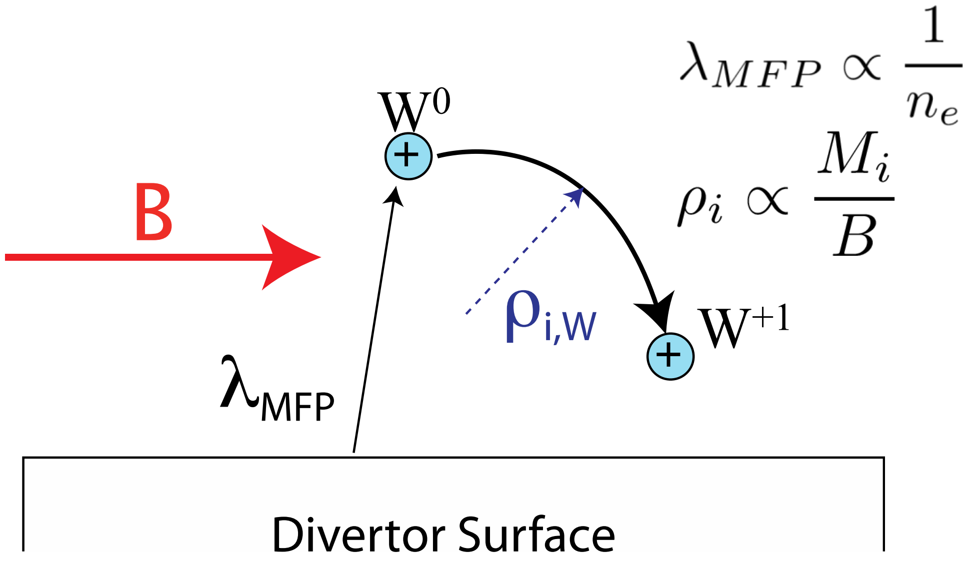
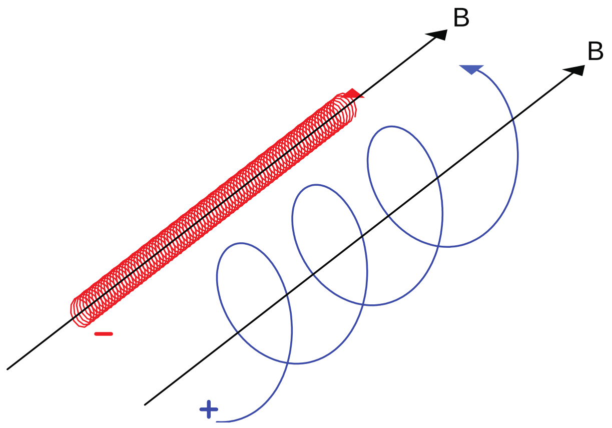
- Some fraction of those eroded atoms are ionized at a distance within a Larmor radius, $\rho_{i,W}$, of the surface – they then rotate around **B**.
  - They can then return to the surface (and stick) while orbiting the magnetic field line – 'Prompt redeposition'
  - At typical magnetic fields and densities in a tokamak fusion reactor divertor the prompt redeposition rate for W should be in the range of 90-99% - **lowering net erosion**.





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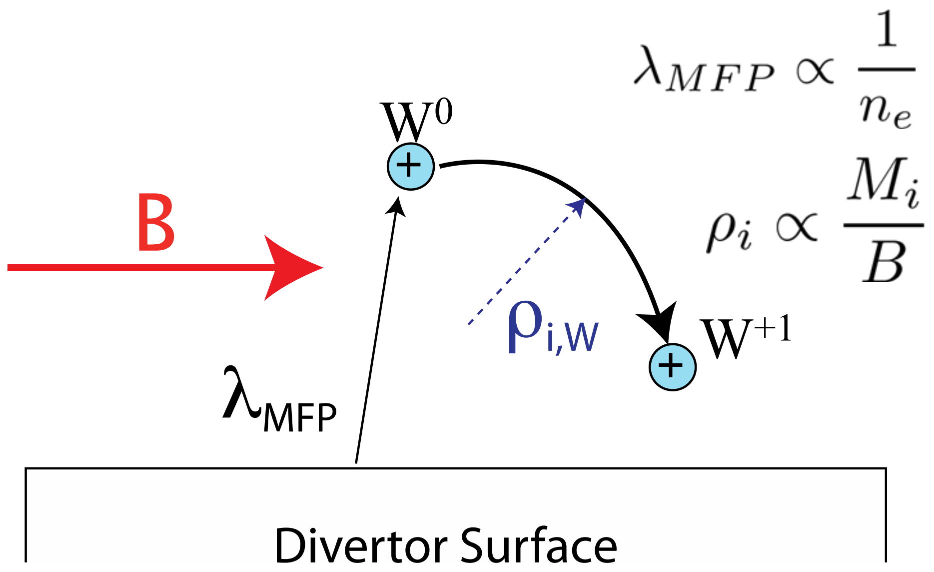
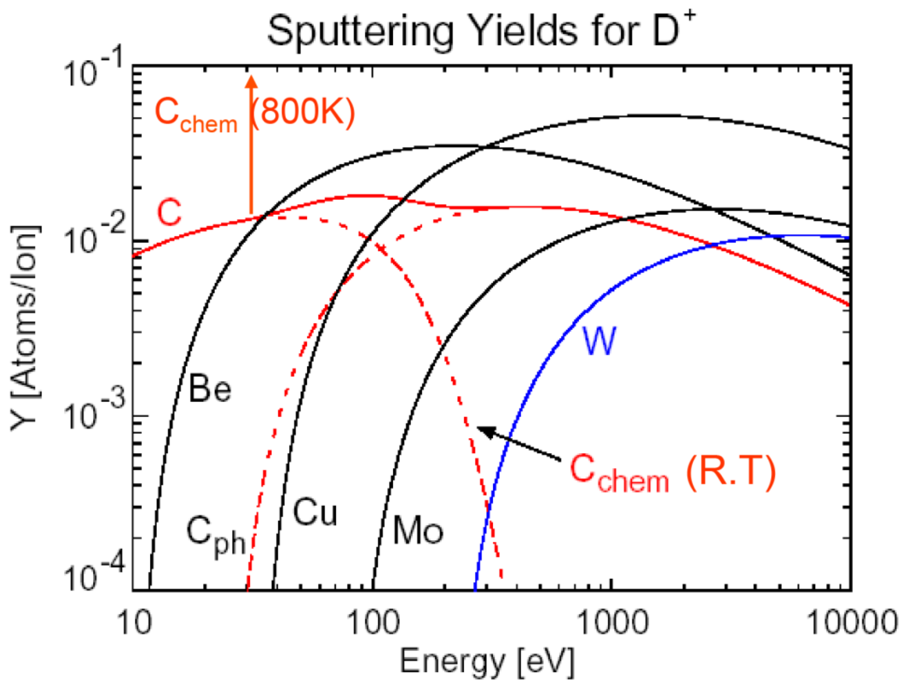


Note - prompt redeposition moves PFC atoms away from the incident – potentially *making valleys and hills – not good*

What are the thermal/mechanical properties of the re-deposited material?

B. Even with knowledge of the sputtering process and re-deposition it is complicated to estimate the net erosion rate

- The net effect of low gross erosion rate, together with prompt redeposition, could potentially bring the net erosion, $\Gamma_W/\Gamma_{i,\perp}$, below the required 10^{-6} (1mm/yr erosion).
  - Hydrogenic ion sputtering is easy to eliminate as the sputtering threshold is high
  - Impurity ions with charge 2-3 will likely be the limiting factor
    - High energy due to their Z (sheath!) and mass
    - Such impurities used to enhance radiative losses in the divertor
  - Rather complex multi-physics models are required for an accurate divertor solution consistent for D+, T+ and impurities, → see: SOLPS-ITER suite of codes.



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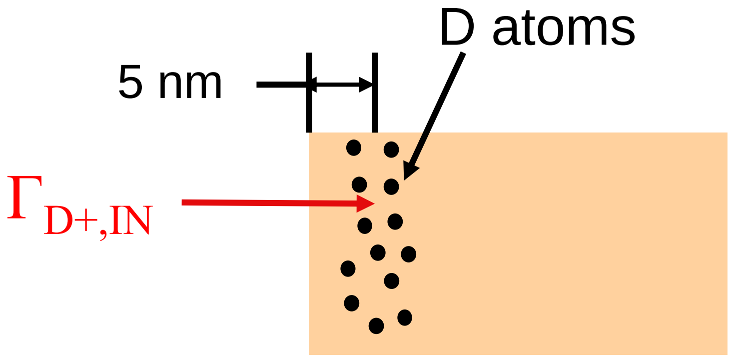
B. The tokamak environment can lead to changes in the PFC surface material characteristics

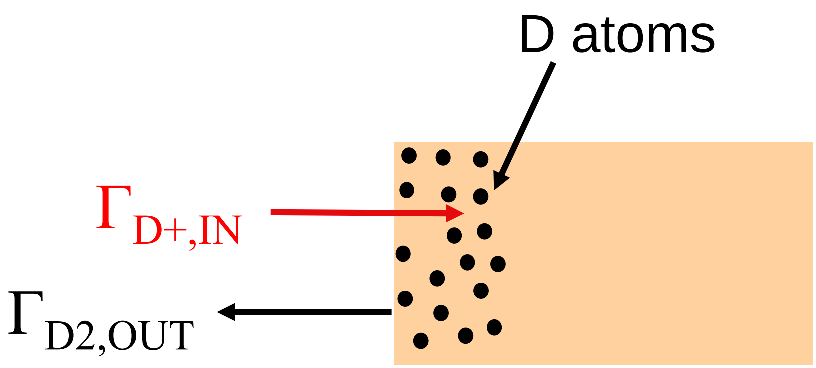
High ion fluxes disturb/damage the material lattice

- Ions travel into the surface (range < 5 nm) picking up an electron - D^0 atoms



B. The equilibrium between incoming ion flux and recombining molecules determines the near surface hydrogen density

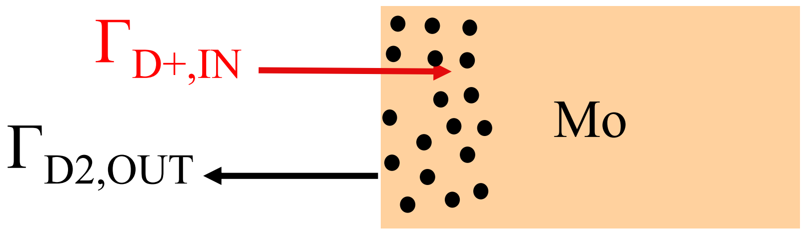
- Ions travel into the surface (range < 5 nm) picking up an electron - D^0 atoms
- Atoms diffuse to the surface where they must combine with other atoms into a molecule to leave the surface (otherwise: energetically unfavorable)
  - The rate of D_2 leaving the surface is determined by the local density, n_D, and the atom-atom recombination coefficient, Rec
- n_D rises such that $0.5 \times \Gamma_{D2,OUT} \sim \Gamma_{D+,IN}$
  - n_D/n_W can reach 10% >> normal D solute level
  - n_D/n_W (solute) ~ $10^{-7} – 10^{-9}$



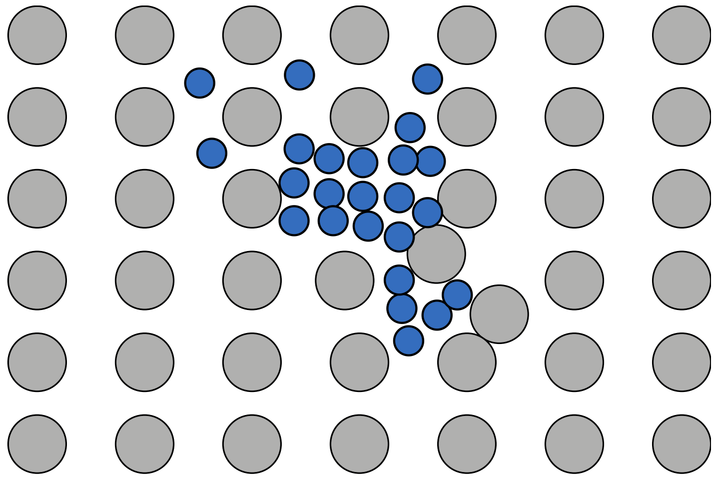
$$\frac{1}{2}\Gamma_{D2,OUT} = \frac{1}{2} n_D^2 \cdot Rec$$

B. Implanted hydrogen leads to high pressures within the lattice and leads to lattice distortions

- High n_D leads to stresses in the lattice
- Stresses are relieved through deformation of the lattice and the creation of vacancies, interstitials or voids -> 'traps' [1-7] with deep potential wells



$$\Gamma_{D+,IN} \approx \frac{1}{2}\Gamma_{D2,OUT} = \frac{1}{2}n_D^2 \cdot Rec$$



Such deformations/traps are a rich area of study
- They can store T which is hard to remove
- The traps diffuse, and can be annihilated

[1] M. Poon et al., J. Nucl. Mater. 307-311 (2002) 723
[2] O. Ogorodnikova et al., 313-316 (2003) 469
[3] G. Wright, PhD thesis, U. Wisc. 2006
[4] V. Alimov, J. Roth, Phys. Scripta T128 (2007) 6.
[5] O. Ogorodnikova et al. J. Appl. Phys. 103 (2008) 034902, P2-61
[6] G. Wright et al., J. Nucl. Mater., **363–365** (2007) 977
[7] G.M. Shu et al., J. Nucl. Mater., **390–391** (2009) 1017–1021

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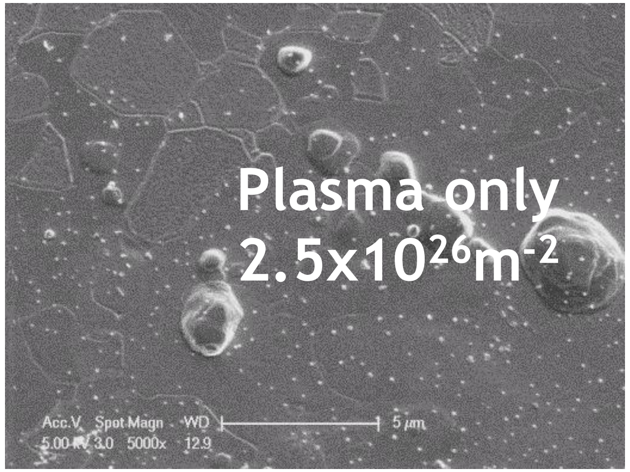
B. Combined particle and heat fluxes lead to synergistic effects on the material

Both plasma fluxes and surface heating modify material surfaces

- Combination of the two brings additional changes to the surfaces
- Probably related to changes in atom mobility at higher temperatures



[1] T. W. Morgan et al J. Nucl. Mater 438 S784–S787 (2013)

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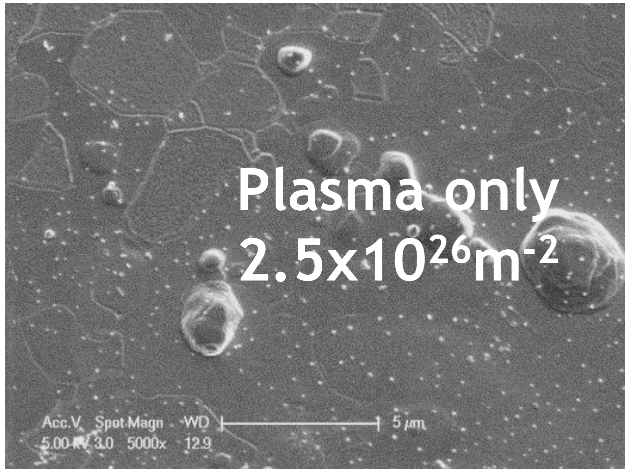

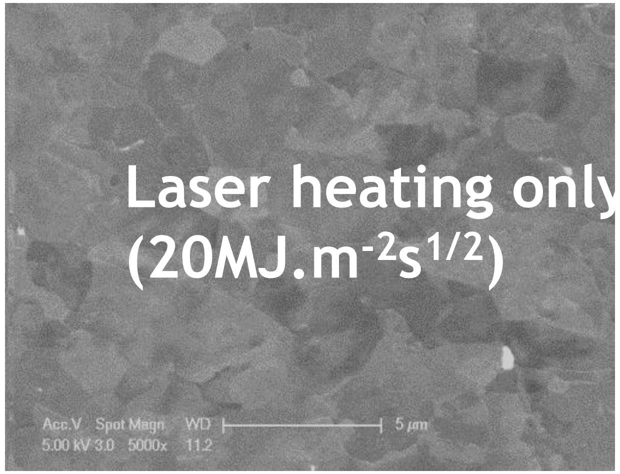

[1] T. W. Morgan et al J. Nucl. Mater 438 S784–S787 (2013)

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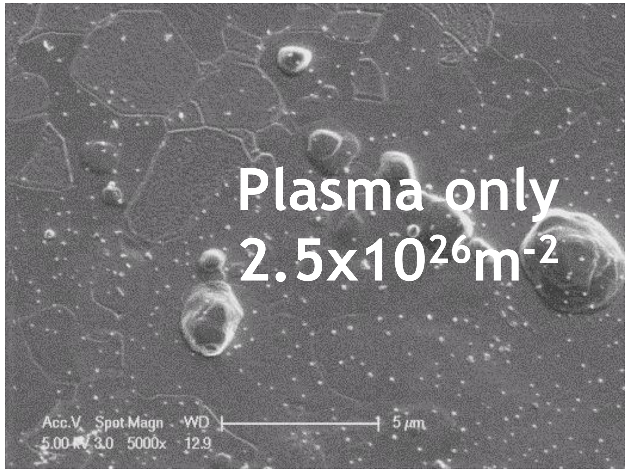
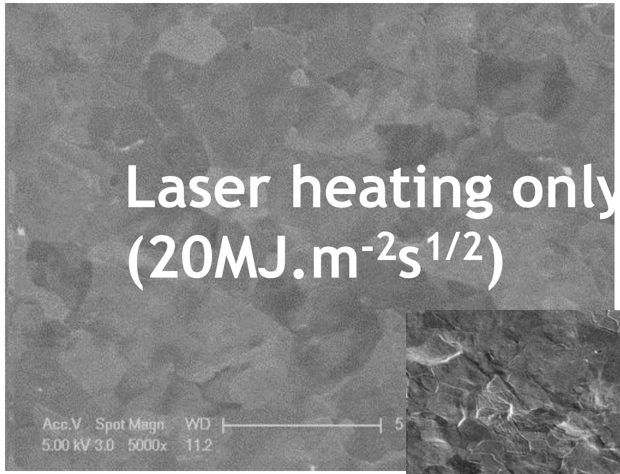
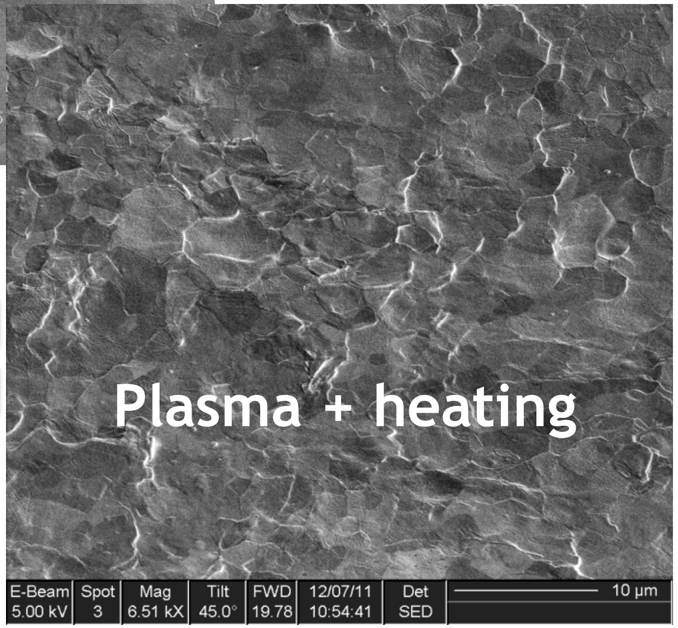


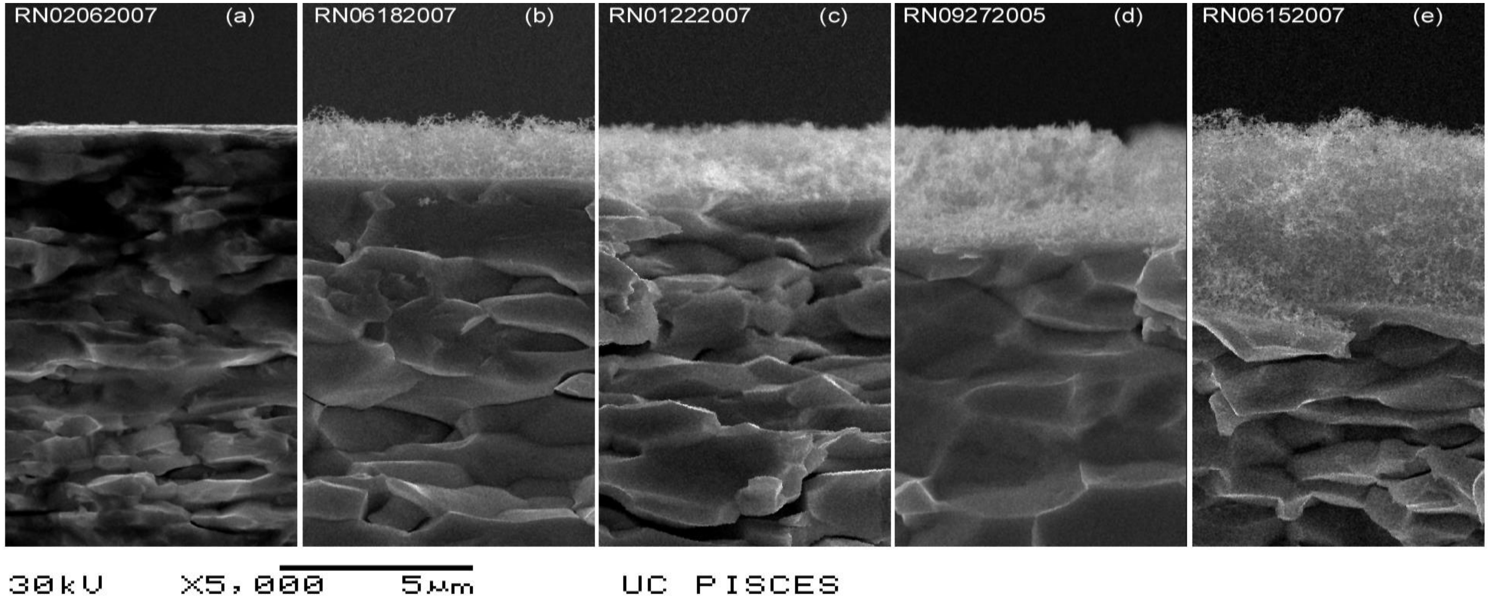


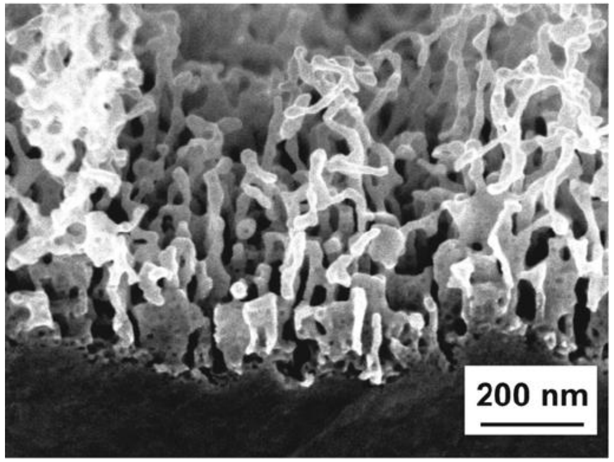


[1] T. W. Morgan et al J. Nucl. Mater 438 S784–S787 (2013)

B. He⁺ fluxes to surfaces at the right surface temperature lead to another synergistic effect – the growth of tungsten ‘fuzz’

- Adding a few % of He ions (He ash from fusion reactions) to the ion fluxes incident on the right temperature W surfaces (~800-1000C) leads to accretion of the W at spots & the growth of tungsten ‘fuzz’ – tendrils ~ 100nm in diameter – and up to microns in length

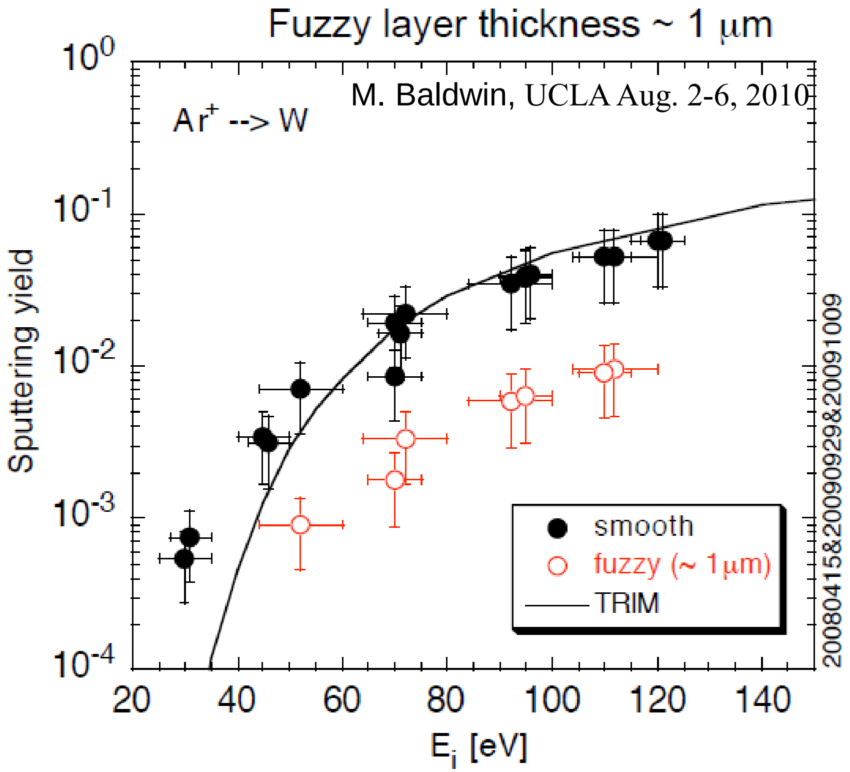




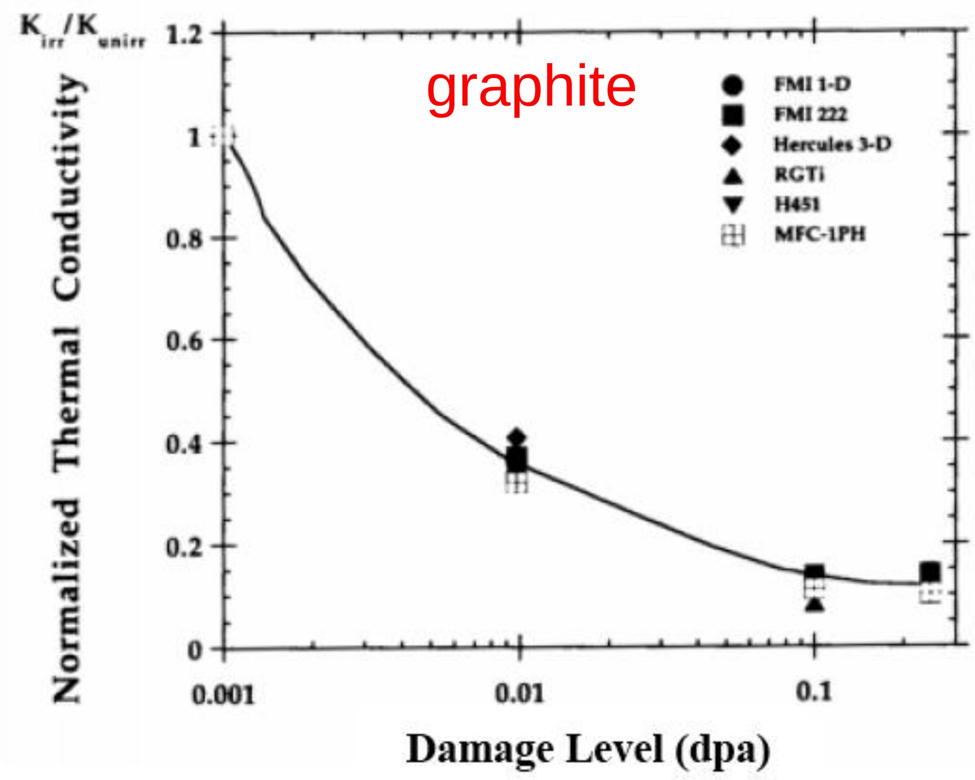
[1] S. Kajita et al. Nucl. Fusion **49** (2009) 095005

B. Tungsten nano-tendril "fuzz" clearly demonstrates how the reactor plasma + thermal environment "re-makes" materials

- Adding a few % of He ions (He ash from fusion reactions) to the ion fluxes incident on the right temperature W surfaces (~800-1000C) leads to accretion of the W at spots & the growth of tungsten 'fuzz' – tendrils ~ 100nm in diameter – and up to microns in length
- The question of whether W fuzz growth is good or bad may depend on how one looks at the issue
  - Reduced sputtering yield,
  - increased resistance to thermal cycling,
  - possible increase in dust production,
  - higher likelihood of arc generation & melting at fuzz tips



B. Developing neutron tolerant materials will probably be last problem solved for fusion

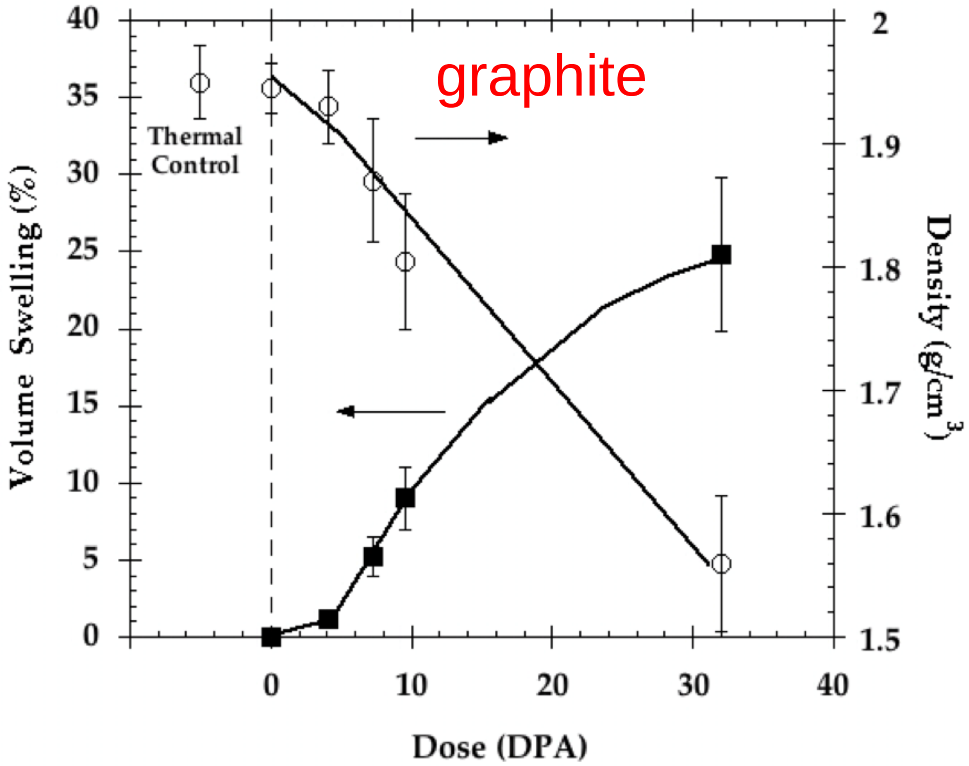


L. Snead, J of Nucl. Materials **224** (1995) 222-229

Example of graphite

- Uniform material bombardment by 14 MeV neutrons
  - ~ 1m thick blanket to thermalize, shield neutrons & breed Tritium
- Displacements per atom in wall ~10 - 20 per year for 1 GW
  - Leads to serious thermal degradation of materials.
  - Internal p, He production by nuclear reactions is also an issue

B. Developing neutron tolerant materials will probably be last problem solved for fusion



L. Snead, J of Nucl. Materials **417** (2011) 629–632

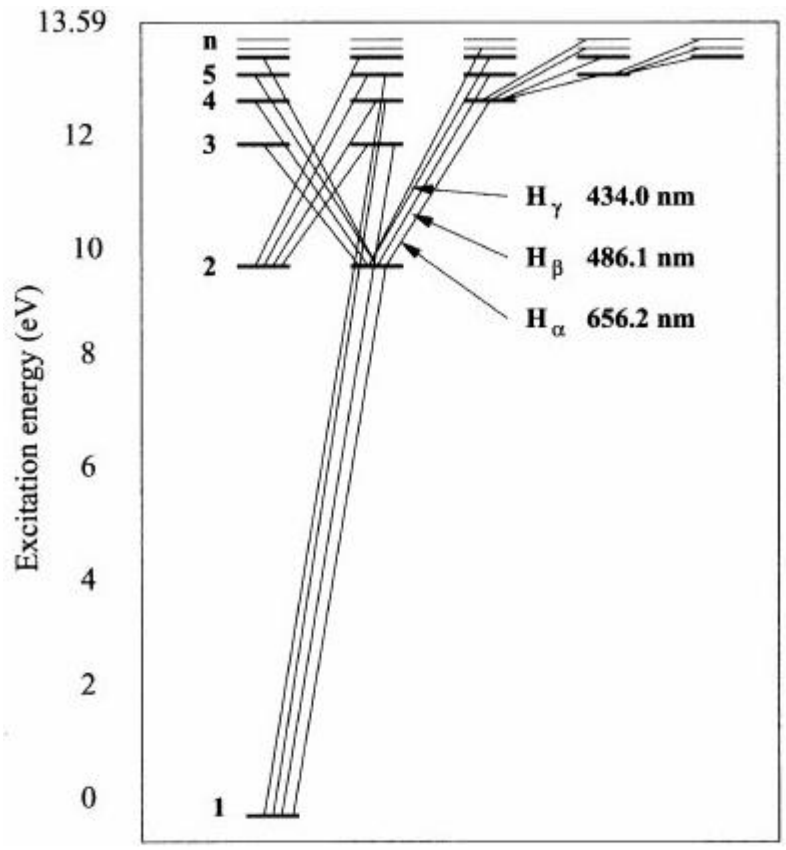
- Tungsten and some 3D graphites have better nuclear damage properties

Example of graphite

- Uniform material bombardment by 14 MeV neutrons
  - ~ 1m thick blanket to thermalize, shield neutrons & breed Tritium
- Displacements per atom in wall ~10-20 per year for 1 GW
  - Leads to serious thermal degradation of materials.
  - Internal p, He production by nuclear reactions is also an issue
- Solution will require dedicated experimental & modeling, probably exploiting self-annealing at high material temperatures

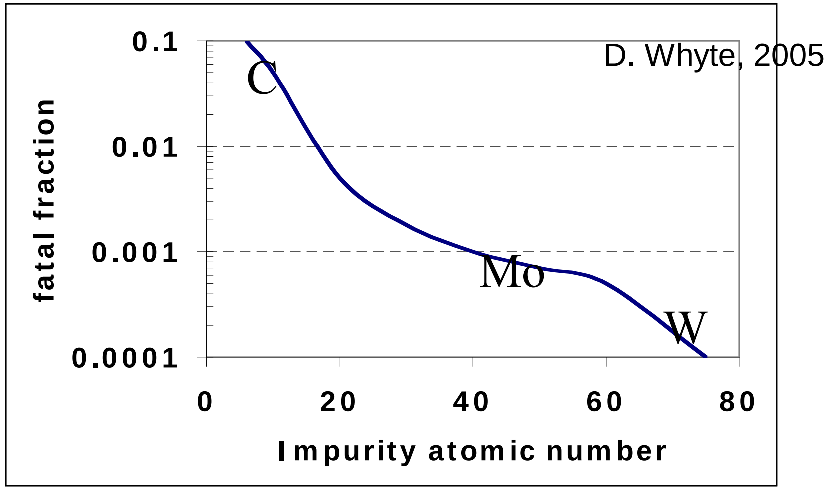
C. Impurity radiation in the core plasma is a constant power loss

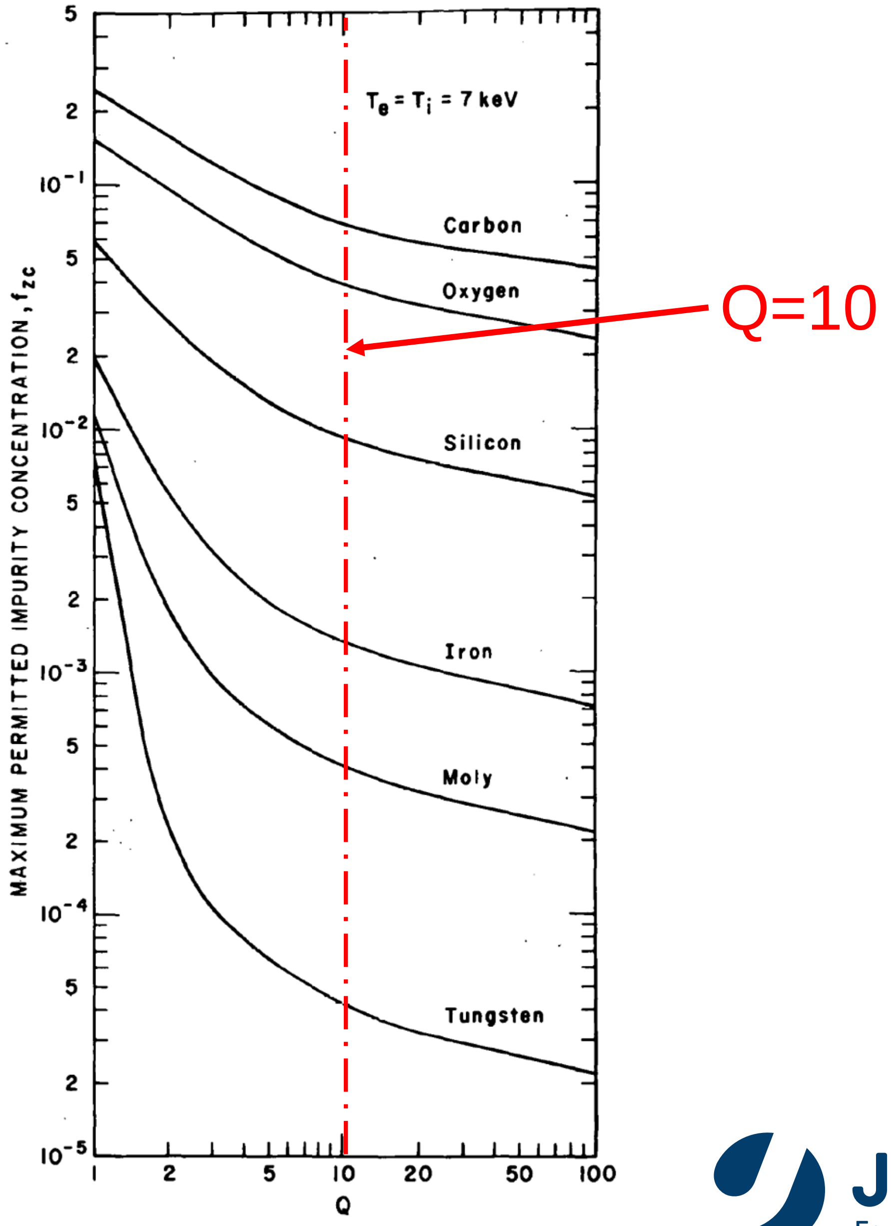
- As the impurity travels through the core plasma it loses more and more electrons.
  - The excitation of orbital electrons before each electron is lost (ionization) leads to radiative losses
- Most light impurities are ‘fully stripped’ of electrons -> they no longer radiate in an ITER plasma
- On the other hand, heavy impurities such as tungsten are never fully stripped → continuously radiating, loss of energy → very little tungsten allowed in the plasma



C. Impurity radiation in the core plasma is a constant power loss (reactor plasma operation)

- Q=(fusion power)/(power in); goal for ITER: Q > 10
  - Allowed concentration (n_W/n_e) of ~ $5x10^{-5}$
  - An injection of a mm diameter droplet of W would lead to radiative collapse in ITER
  - **Melting must be avoided ☺**





Jensen PPPL report #1350, 1977

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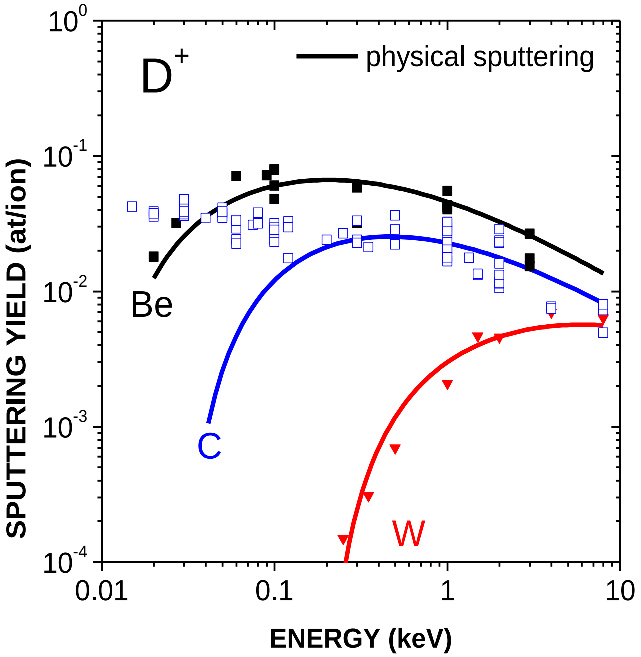
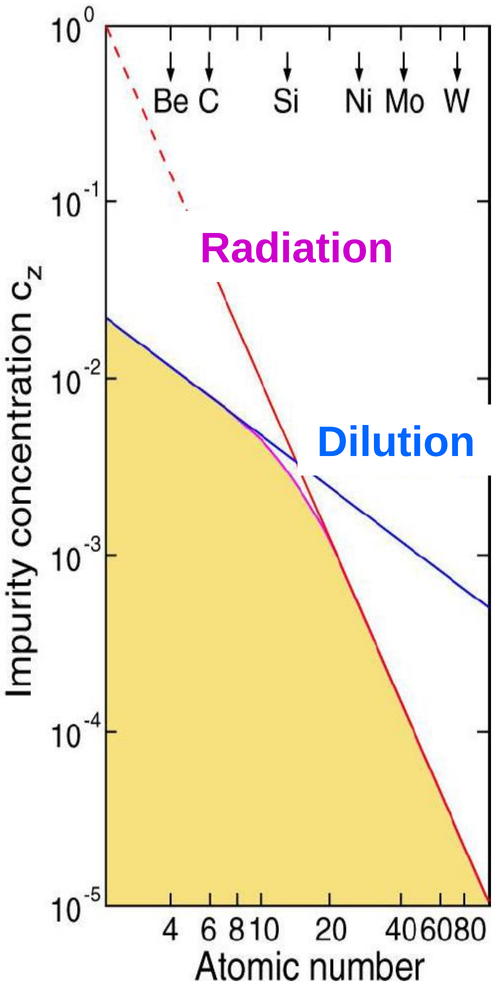
C & D: Criteria for material choice

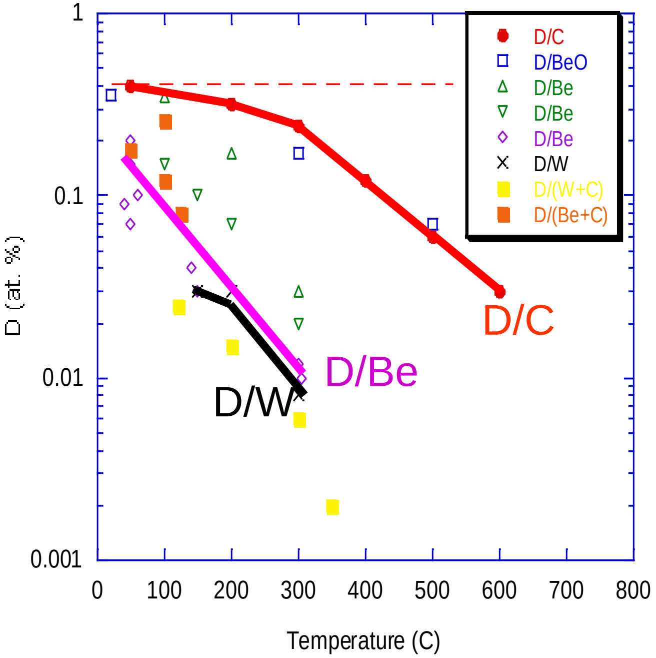
- Wall erosion, tritium retention: → use high Z wall material
- Impact on plasma (dilution, radiation): → use low Z wall material



Wall erosion best with high Z
(but no melting of carbon!)



minimum impact on fusion
performance with low Z



Tritium retention -
an issue for carbon!

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B&C: Summary

Net erosion of the plasma facing components must be kept at extremely low levels, both for reactor lifetime and plasma operation

- Must avoid melting, **by all means**

- Evaporation can probably be controlled (active cooling)

- Sputtering: rather well understood:
  - properties of redeposited layers?
  - complex transport pathways

B&C: Summary

- Through divertor geometry and plasma physics the divertor power load can be reduced below 10 MW/m^2, compatible with ITER operation
  - Not yet clear whether the process of detachment is enough to reduce power loads below engineering limits for a reactor

- Our current understanding shows that divertor surface erosion rate can be kept low enough to achieve < 1mm/yr for tungsten in the divertor
  - If one can keep the plasma temperature < 5 eV, compatible with detachment

- Operation of a reactor at high temperature (800-1000K) has positive effects on
  - Reduction of T retention
  - 'mending' of neutron damage
  - Reactor engine efficiency of conversion of heat to electricity
  - Note we don't yet have a demonstrated coolant that works at those temperatures

D. Tritium retention – Control of Tritium & He ash within the vessel plays a crucial role in optimizing fusion reactors

- The He ash cannot build up in the plasma, or else the fusion reactivity drops
  - Constraint on the allowed fraction of He in the plasma

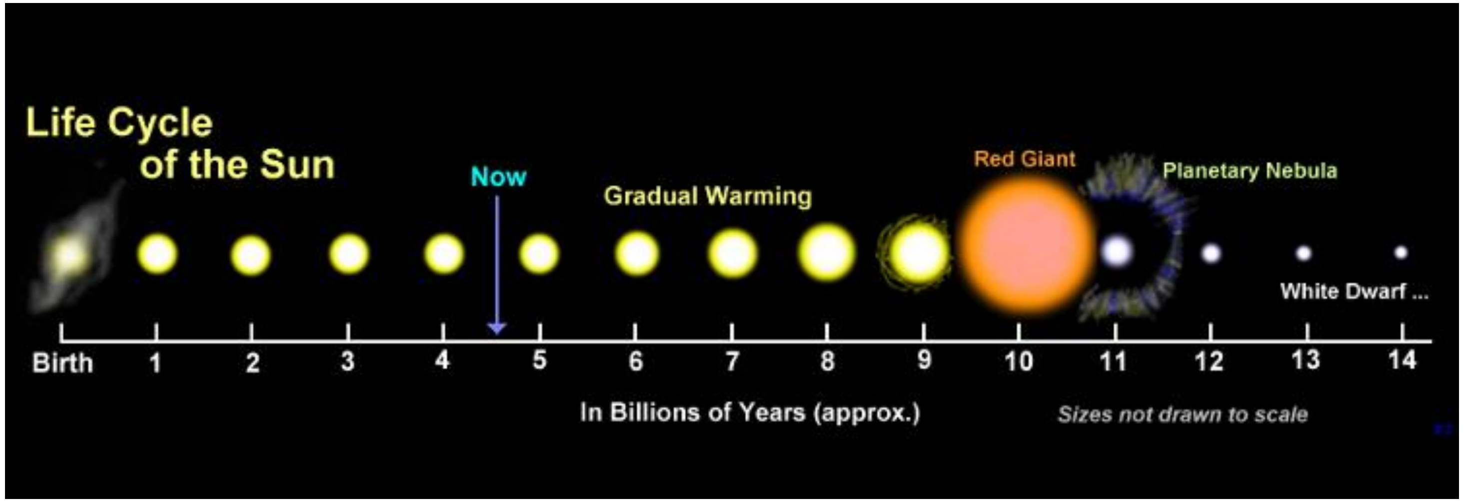
    poses lower limit on pumping and hence on plasma surfaces fluxes
- The T retained in vessel surfaces has to be very low
  - Constraint on the fraction of T ions incident on the surface that stay in the surface (retained) **poses upper limit** on plasma surfaces fluxes.

- In a fusion power plant a compromise between the two must be found

The vision of nuclear fusion research:
A miniature star in a solid container. He ash control

The sun as a role model: no He ash control there

Hydrogen burn → Helium accumulation → Collaps



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Sun ≈ 10 Billion years

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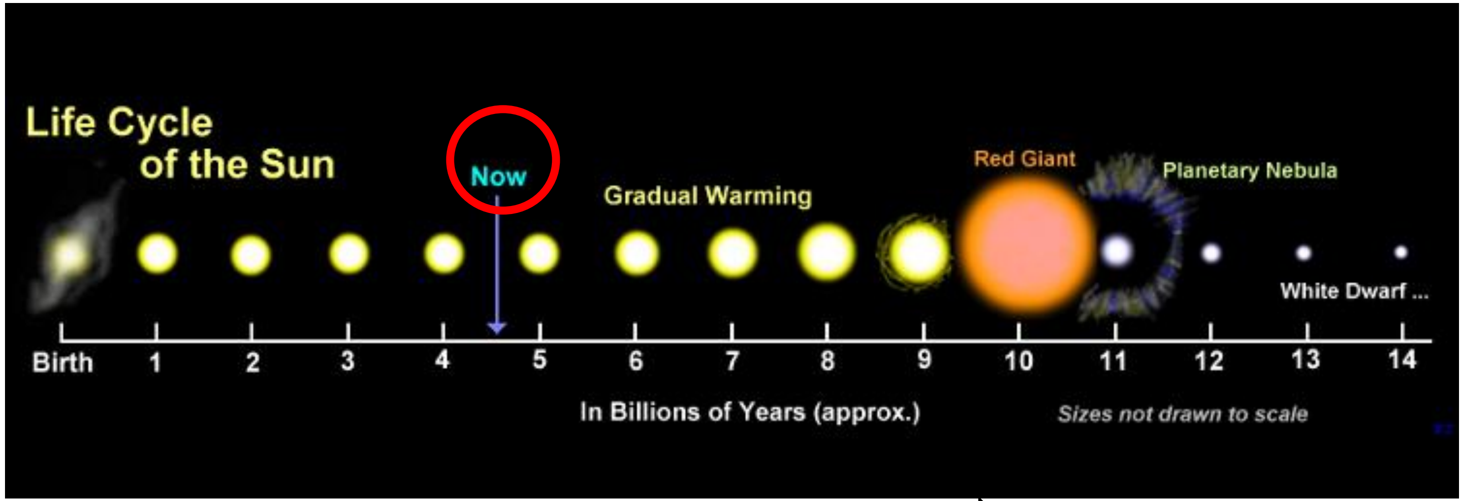

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Sun ≈ 10 Billion years

Fusion flame on earth ≈ 100-200s

well short of ITER goals

D. He 'ash' reduces the number of D/T ions in the plasma and thus the fusion reactivity

- Plasma local quasi-neutrality (#Z of positive charged ions = # of electrons) means that any impurities in the plasma dilutes the number of fusion reactions between D^+ & T^+
  - For every impurity atom entering the plasma with N electrons – N fewer D,T ions are allowed in the plasma – dilution of the fuel

$$n_e = n_D + n_T + Z_{He} n_{He}$$

Note: n_e is limited by B (plasma β)

$$1 - Z_{He} \frac{n_{He}}{n_e} = \frac{n_D + n_T}{n_e}$$

$Z_{He} = 2$, almost everywhere in plasma

$$1 - Z_{He} f_{He} = \frac{n_D + n_T}{n_e}$$

- Want to keep f_{He} low to not lower the fusion rate too much ($\propto n_D n_T$).
- Other low- to mid-Z impurities can also dilute the fuel before their radiation becomes too large

D. He 'ash' reduces the number of D/T ions in the plasma and thus the fusion reactivity

Power balance of a steadily burning fusion plasma, → burn condition (aka: "Lawson criterion")

$$\boxed{n\, T\, \tau_E = g(T)}$$

Core plasma physics, energy confinement τ_E

But the same process that releases fusion energy also produces He particles

Additional: Particle balance for helium ash:

$$\frac{1}{4} n^2 \langle \sigma v \rangle = \frac{n_{He}}{\tau_P} = f_{He} \cdot \frac{n}{\tau_P} = f_{He} \cdot \frac{n}{\rho \cdot \tau_E}$$

Boundary plasma physics, particle confinement τ_P

τ_E : Energy confinement time, „given by nature….",

τ_P : Particle confinement time: recycling, pumping, plasma surface interaction, A&M processes

$$\rho = \tau_P / \tau_E$$

$$\boxed{n\, T\, \tau_E = g(T,\rho)}$$

A boundary plasma parameter ρ explicitly appears in the burn condition

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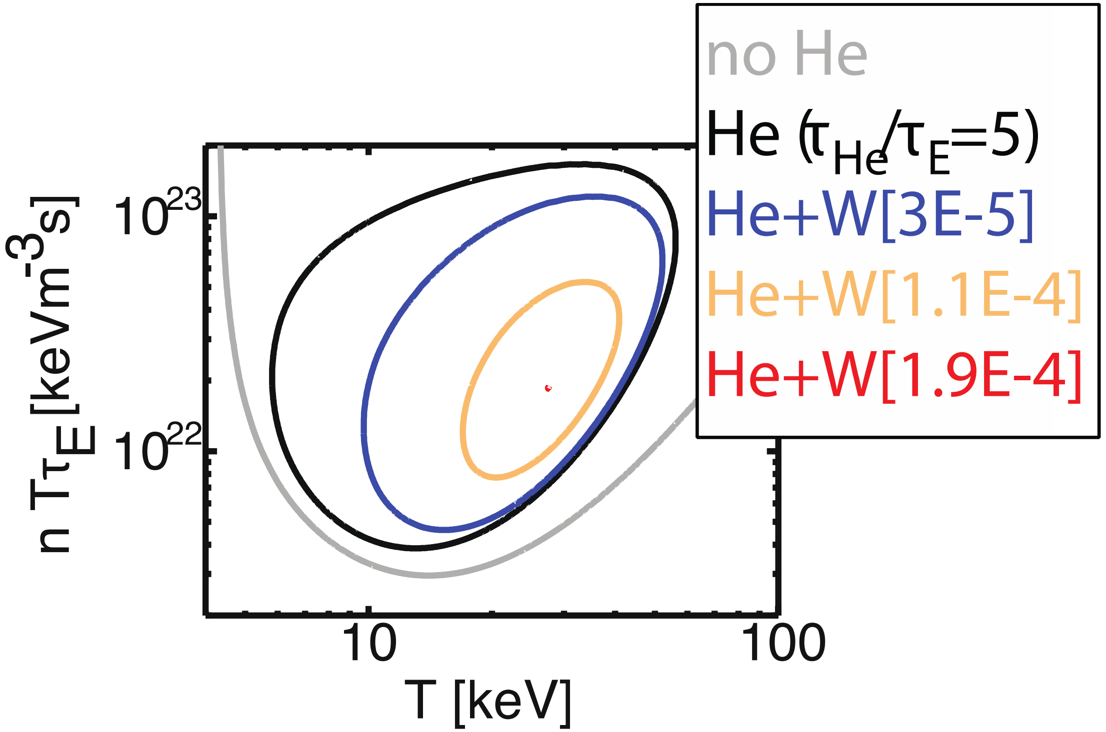
D. Both dilution and radiation play a role in the 'operational space' for fusion

- Both dilution and radiation can be taken into account in determining the size of operational space for fusion burn through the **"Lawson criterion" (burn criterion)**

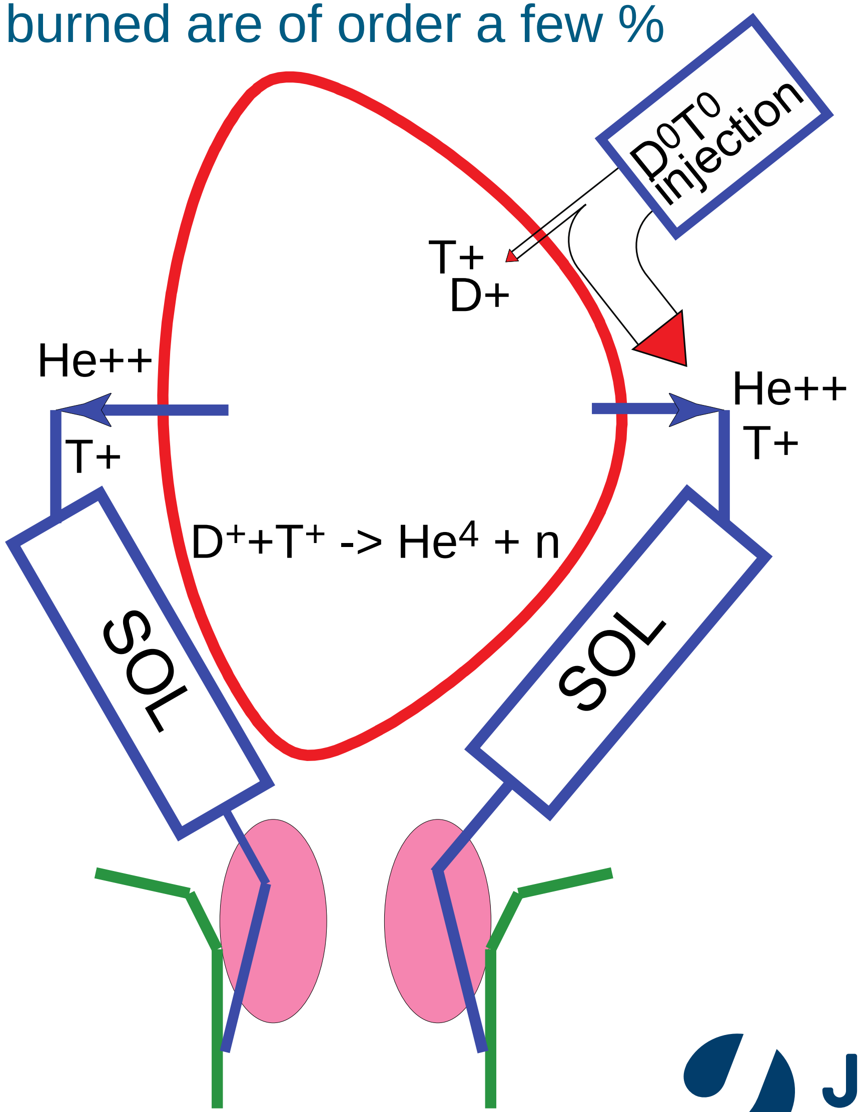


- Higher $nT\tau_E$ and higher T are needed as the He and other impurity concentrations increase.

D. Reiter et al., Nucl. Fus. 30 (1990) 2141;
T. Pütterich, EFPW Split, Dec 2014 p112

D. Tritium burn fraction in a fusion reactor will likely be low

- Most predictions for the fraction of injected T that is burned are of order a few %
- The burn fraction is lower than one would like – there is more T in the exhaust gas that needs to be extracted
- **However,** the fusion power density is still impressive
  - ITER fusion power density ~ 10MW/m^3
  - Sun fusion power density ~ 100W/m^3
  - Human body power density ~ 100W or ~ 1KW/m^3.
- The fusion power density can be increased through higher plasma pressure (β, **B**)



D. Two processes lead to T retention in plasma facing components (PFCs)

- Two processes lead to the fusion fuel getting ‘stuck’ in the fusion engine
  - **Co-deposition of the fuel** with those PFC atoms that return to the surface – ‘burying’ the tritium
  - **Implantation of the fuel** into the surface material lattice

- Both are dependent on the incident tritium flux. The divertor is where the flux and fluence are the largest.

D. Retention of T in surfaces is related to ion fluxes to those surfaces which we relate back to heat flux

- Ion flux Γ_{ions}, leads to both surface and bulk retention of tritium in the plasma facing component
- We can relate back to the maximum allowed heat flux - 10MW/m^2

$$q_\perp[W/m^2] = \Gamma_{ions}[1/(m^2 s)]\gamma_{sh}T_e[eV] \times 1.6 \times 10^{-19}[J/eV] = \text{ 10MW/m}^2$$

- Where γ_{sh} is the sheath transmission coefficient which specifies the ion and electron contributions to the heat flux at the target - γ_{sh} ~ 7- 8
- Also assume $T_{e,\text{target}} \leq$ 5 eV and: 50% reactor operation/availability:

$$\Gamma_{ions}[1/(m^2 s)] = \frac{q_\perp[J/(m^2 s)]}{\gamma_{sh}T_e[eV] \times 1.6 \times 10^{-19}[J/eV]}$$

$$= \frac{10^7[J/(m^2 s)]}{7 \times 5[eV] \times 1.6 \times 10^{-19}[J/eV]}$$

$$= 1.8 \times 10^{24} D/T ions/m^2/s$$

→ Yearly ion fluence of ~ 2.7x10^{31}/m^2

D. Use the *in-vessel T safety limit* to determine the maximum tolerabe T retention rate R_{T^+}

Retention in PFC: $R_{T^+} = \Gamma_{retained\ T^+} / \Gamma_{incident\ T^+}$

Assuming fusion operation half the year,

$$\Gamma_{incident,D^+ + T^+} \approx 2.7 \times 10^{31}/m^2/yr$$
$$\Gamma_{incident,T^+} \approx 1.3 \times 10^{31}/m^2/yr$$

- Assuming an area, $S_{div} \sim 10m^2$, the incident mass of T^+ is

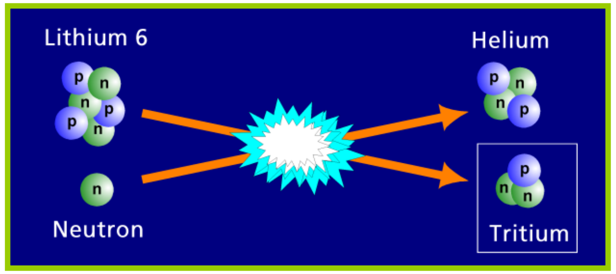
$$\Gamma_{incident,T^+} S_{div} \approx 2 \times 10^5 [kg, T/yr] = 200 \textbf{ tons } T/yr$$

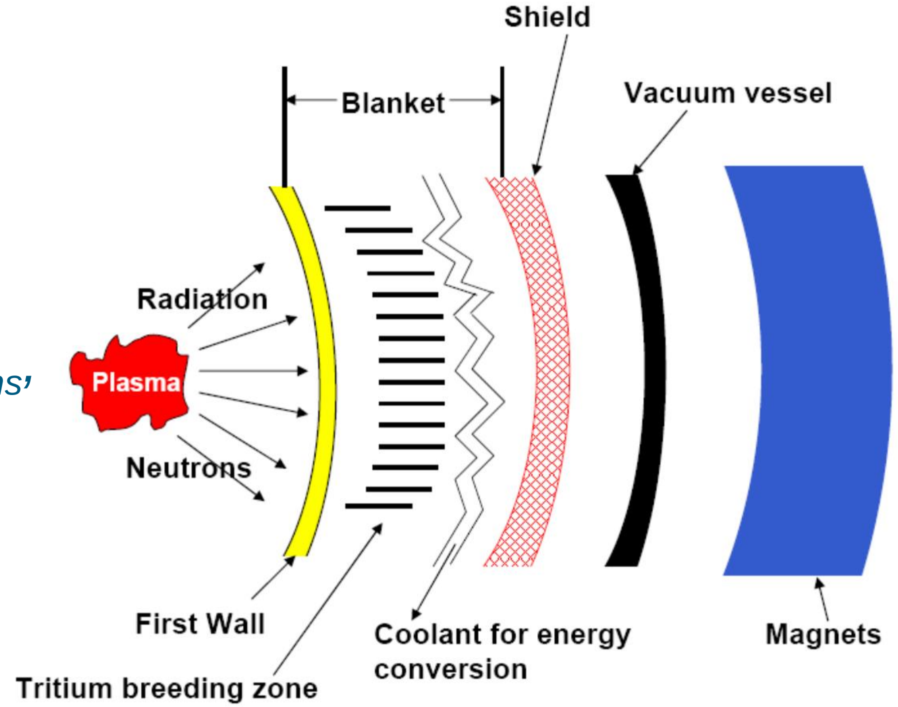
- In-vessel tritium safety limit $\Gamma_{retained\ T^+} = 640g$, forcing $R_{T^+} \sim 10^{-6}$ to reach limit after 3 years of operation

D. Use the *T breeding ratio and fusion rate* to determine the maximum T retention rate R_{T^+}

T breeding ratio B_T is the ratio of the T produced in blanket S_T to the fusion rate: $B_T = S_T / \Gamma_{neutrons}$

- Assume 2.5GW reactor and B_T = 1.02 (2% additional T breeding over the fusion rate)





- Normally we think of this surplus T produced, $(B_T - 1) \times \Gamma_{neutrons}$, as being used to start the next reactor. But here we assume it replaces the T retained in tiles, $R_T \times \Gamma_{incident\ T+}$

$$(B_T - 1) \times \Gamma_{neutrons} = R_T \times \Gamma_{incident,T^+}$$

$$R_T = \frac{(B_T - 1) \times \Gamma_{neutrons}}{\Gamma_{incident,T^+}} \approx \frac{0.02 \times 0.4\ kg/\mathrm{day}}{1260\ kg/\mathrm{day}} = 8 \times 10^{-6}$$

- Again, we need the retention R_{T^+} to be very low

D. Dependence of Tritium co-deposition on tile material

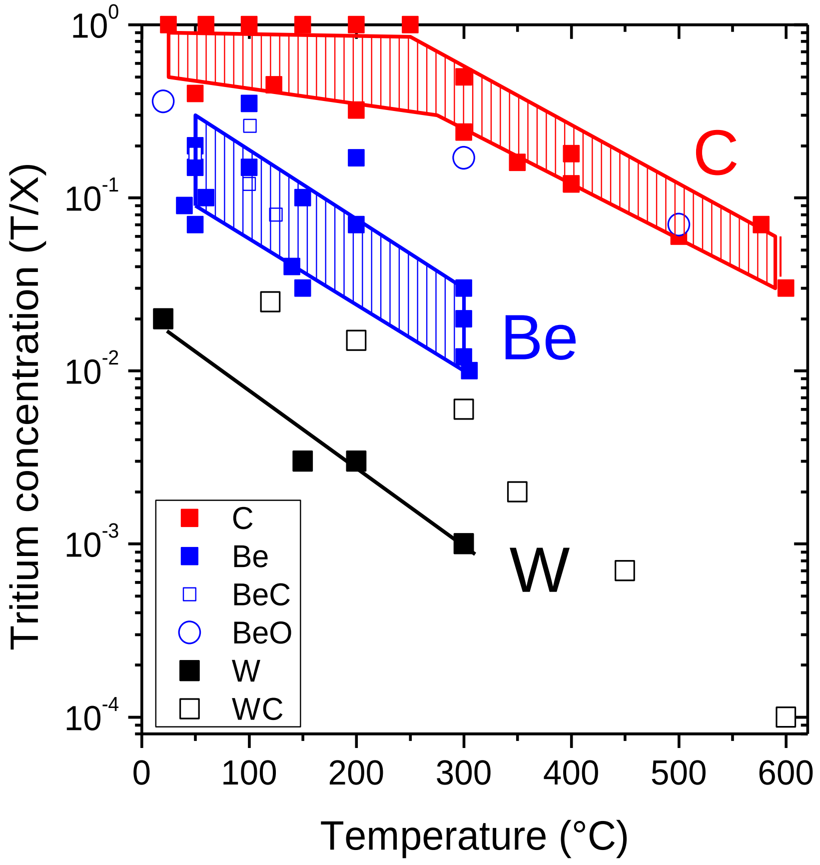
Co-deposition:

- Rough estimate: total net erosion rate x co-deposition concentration
- Detailed evaluation: impurity transport including re-erosion,
  co-deposition concentration depending on final deposition conditions

Tritium inventory due to co-deposition

- much smaller for high-Z
- The erosion rate varies as well
- Example – compare Be to W:
  40 (erosion) x 20 (co-deposition)
  = 800 more co-deposition

Even larger difference for C to W:



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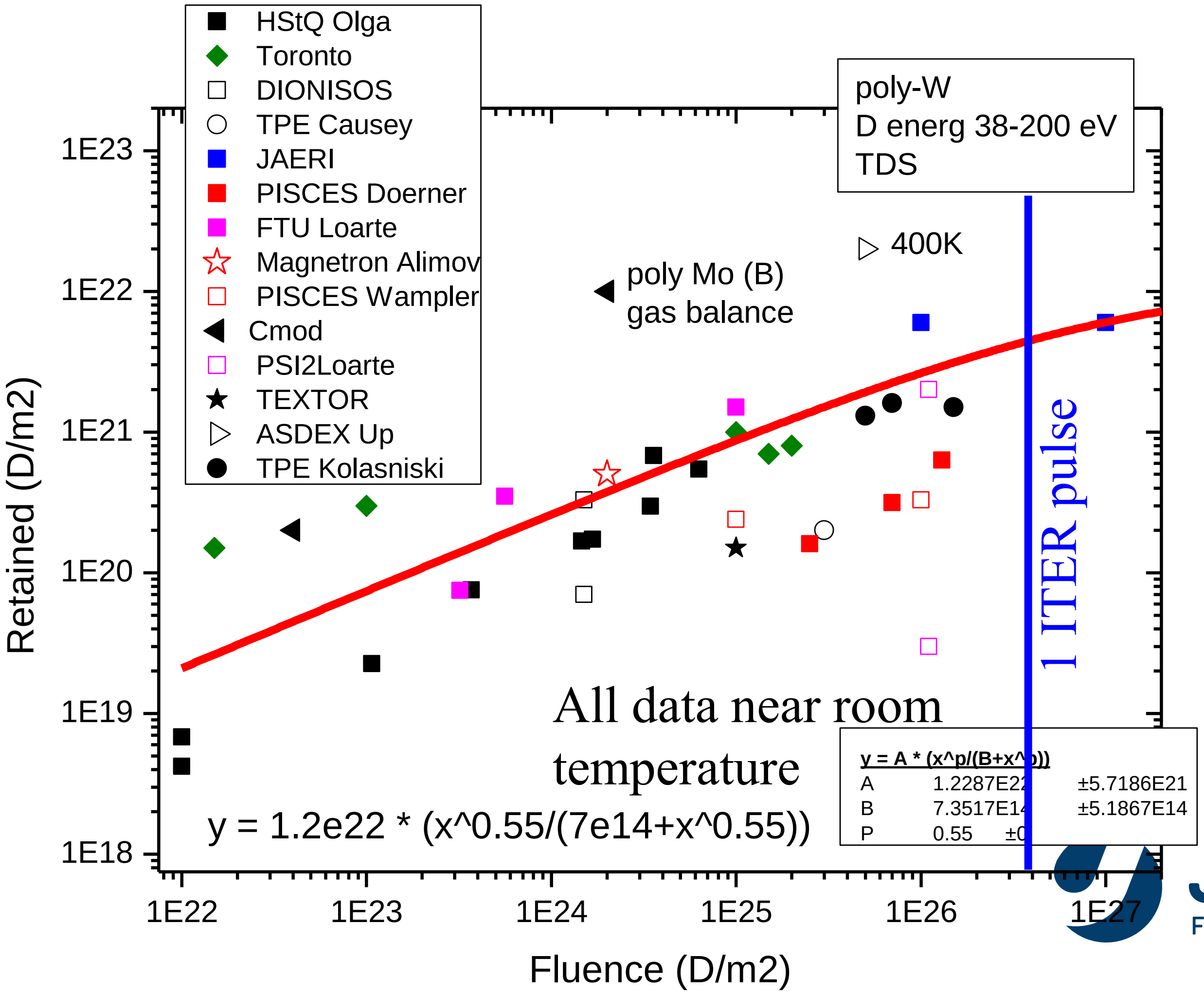
D. Implantation is the dominant retention process for refractory metals such as W

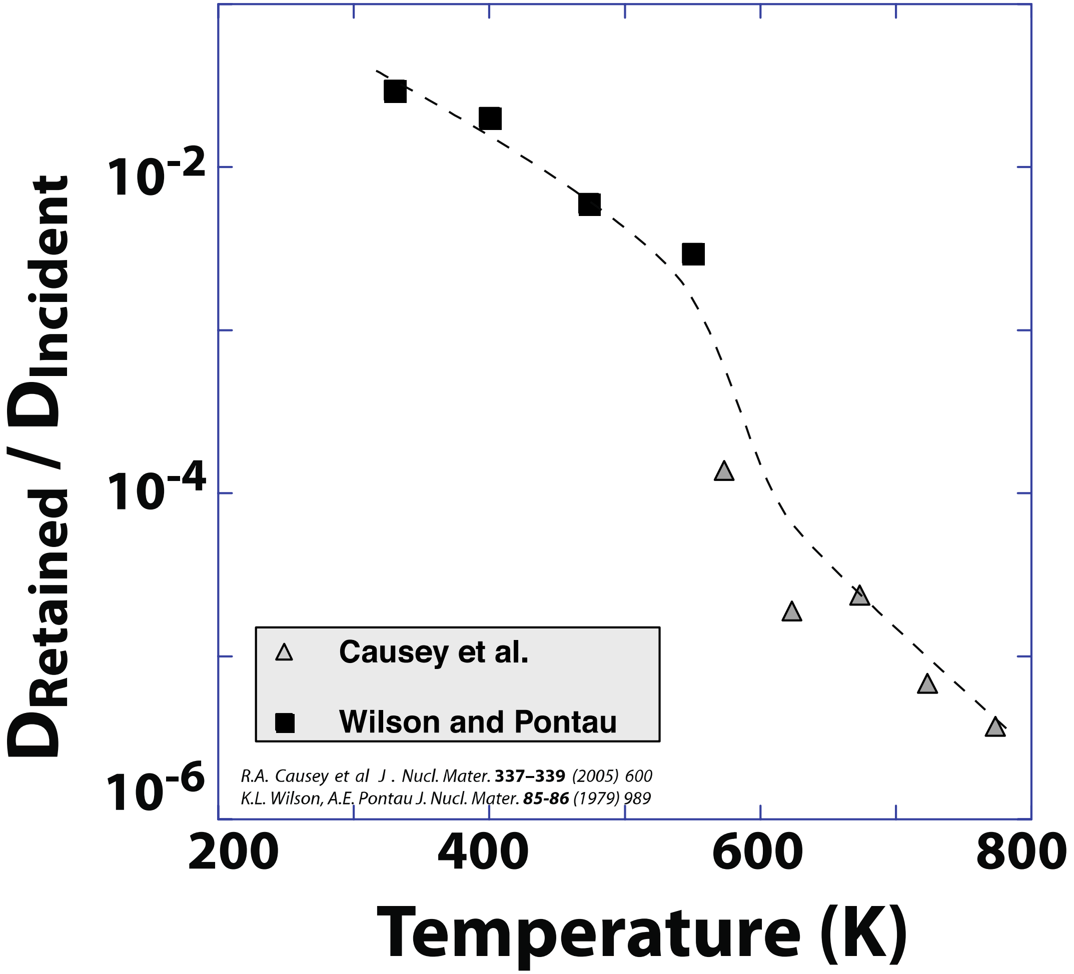
Laboratory studies show that the retention is very low in tungsten when there is no nuclear damage

- Retention high at low fluence, decreasing with increasing fluence
- Retention scaling roughly as $(\text{fluence})^{0.55}$ => dominated by diffusion
- Retention in surfaces < 1 in 10^5 at high fluence and near room temperature



D. Operating at higher temperatures lowers the retention even further

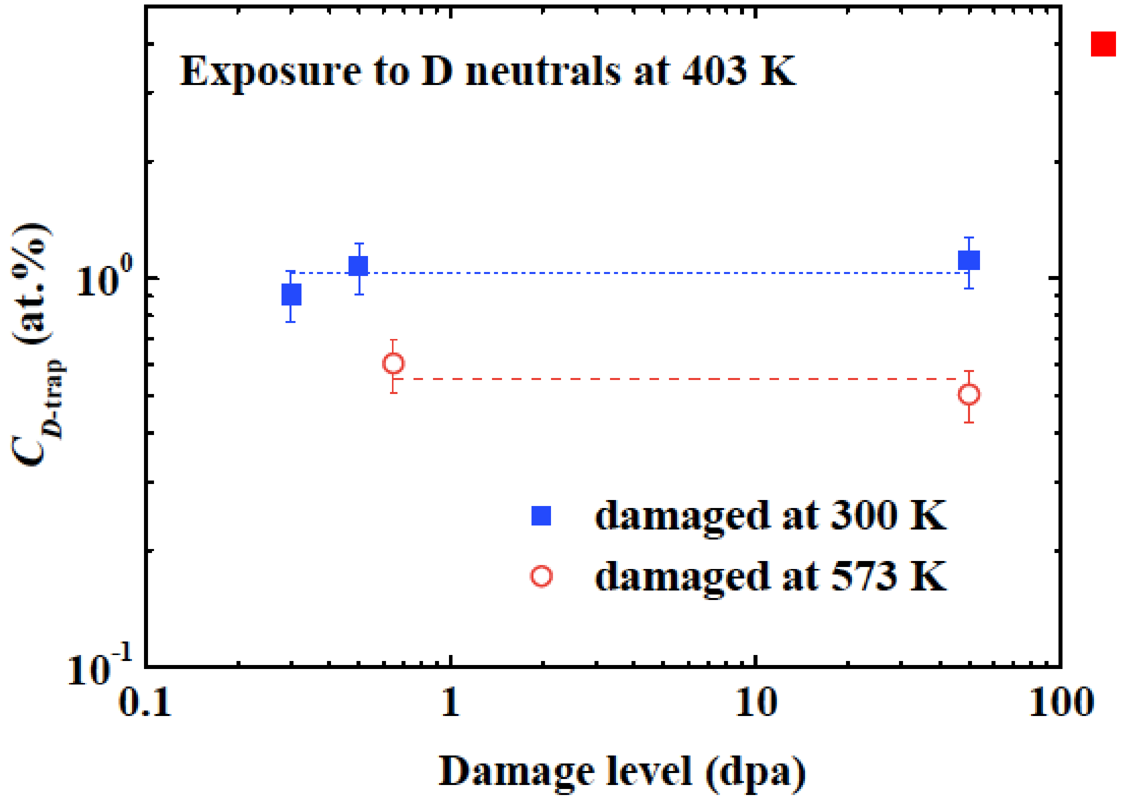
- Drop in retention with increasing material temperature
  - At higher PFC temperature the T atoms have more thermal energy and can escape normal potential wells in the lattice



Ideally operate PFCs at high T

D. Retention increased by neutron damage to material; some results show a saturation of trapping sites

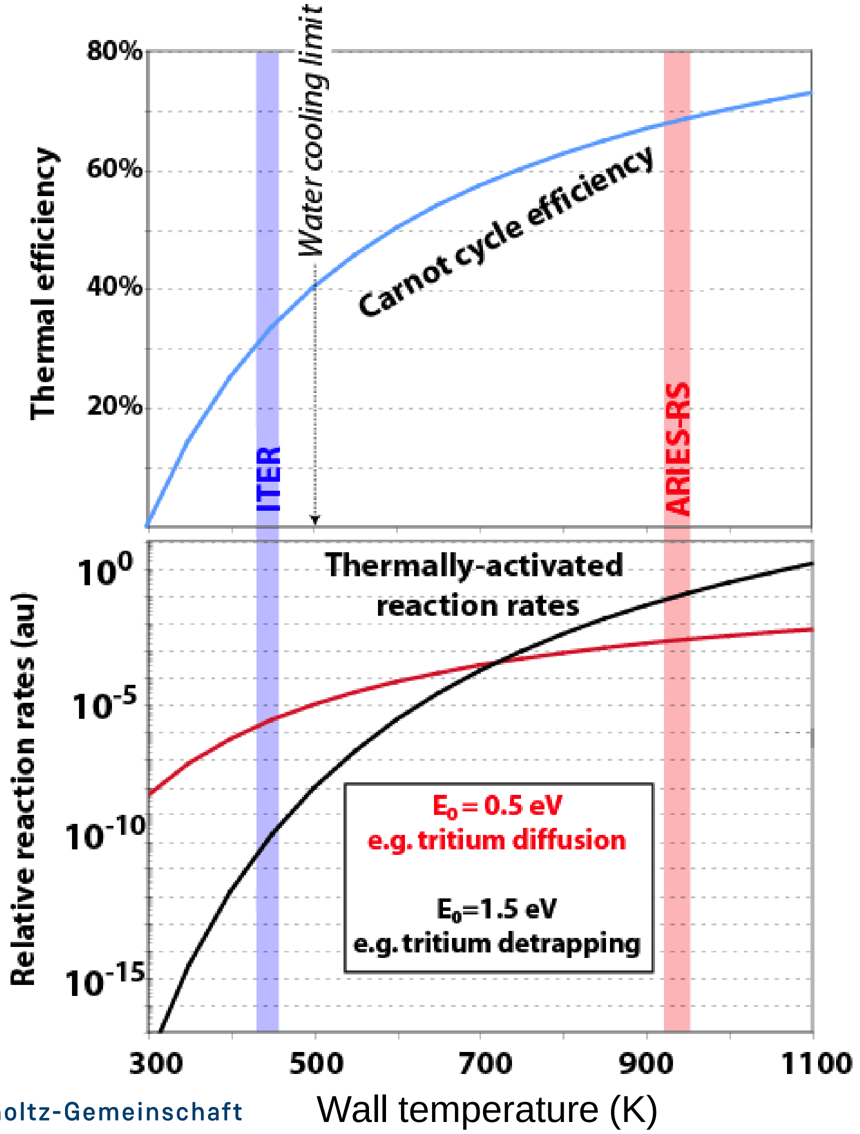
Add neutron damage → 1% tritium storage capacity at defects

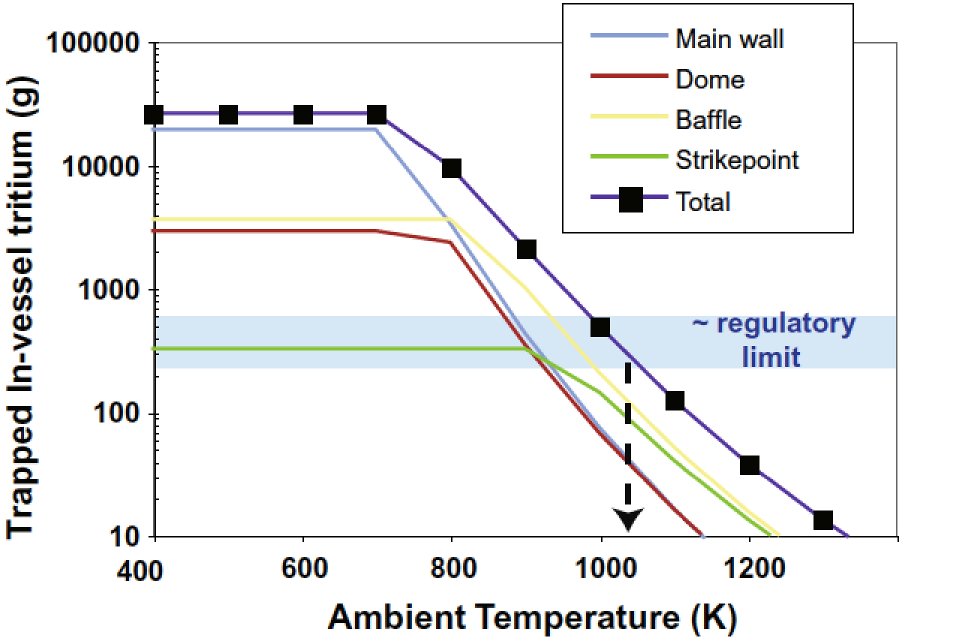


Y. Hatano PSI 2012

- Neutron damage leads to deeper potential wells in the lattice
  - Harder to get the T out of potential wells and diffuse out of the PFC
  - Still – higher temperature lowers the retention

D. Tritium control, thermal efficiency and annealing of neutron damage addressed through high material temperature





D. Whyte PSI 2008

$$Rates \propto \exp\left(-\frac{E_o}{T_{material}}\right)$$

- 'Mending' of neutron damage: neutron-induced potential wells diffuse and can 'recombine' with displaced atoms

Summary:
Tungsten appears to be the reactor PFC material choice

- Graphite has disadvantages with respect to tungsten
  - Erosion rate is much higher at optimal divertor conditions of detachment and low Te
    - Lifetime and dust (safety) issues
  - Carbon has a high rate of T being retained in the surface due to co-deposition
    - T hard to remove without removing more C
  - Structural strength of graphite reduces more with nuclear damage
  - Other materials (e.g. liquid metals) are being considered as well

- Tungsten still has questions
  - Want high operating temperature to reduce T retention and repair DPA damage
  - Too high a temperature, above the ductile to brittle transition (DBTT) reduces ductility and makes it more likely to fail structurally.

Summary:
Power plant costs drive fusion reactor design towards more difficult PFC challenges

ITER example

$P_{fusion} = 500$ MW $S_{area} = 700$ m^2 Cost ~ 20 G$ $T_{wall} \sim 450$ K $f_{on} \sim 0.1$ (duty factor)

Assume: cost of electricity: 0.1 $/kWh

Assume: conversion efficiency: $\eta_{th} = 0.25$

Summary:
Power plant costs drive fusion reactor design towards more difficult PFC challenges

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Assume: cost of electricity: 0.1 $/kWh

Assume: conversion efficiency: $\eta_{th} = 0.25$

$$\frac{\$cost}{\$_{electric}/yr} = \frac{20 \times 10^9 \$}{P_f \eta_{th} (f_{on})(8760\ hr/year) 0.1\$/kWh} \rightarrow \boxed{1800\ yr}$$

After 1800 yr operation ITER becomes a profitable power plant

Summary
Power plant costs drive fusion reactor design towards more difficult PFC challenges

ITER example

$P_{fusion} = 500$ MW $S_{area} = 700$ m^2 Cost ~ 20 G\$ T_{wall} ~ 450 K f_{on} ~ 0.1 (duty factor)

Assume: cost of electricity: 0.1 \$/kWh

$$\frac{\$cost}{\$_{electric}/yr} = \frac{20 \times 10^9\$}{P_f \eta_{th} (f_{on})(8760\ hr/year)0.1\$/kWh} \rightarrow \boxed{360\ yr}$$

P_f x 5 → 2500 MW reduces operation time until pay off to 360 years

JÜLICH Forschungszentrum

Summary
Power plant costs drive fusion reactor design towards more difficult PFC challenges

ITER example

$P_{fusion} = 500$ MW $S_{area} = 700$ m^2 Cost ~ 20 G\$ T_{wall} ~ 450 K f_{on} ~ 0.1 (duty factor)

$$\frac{\$cost}{\$_{electric}/yr} = \frac{20\times 10^9\$}{P_f\eta_{th}(f_{on})(8760\ hr/year)0.1\$/kWh} \rightarrow \boxed{180\ yr}$$

P_f x 5 → 2500 MW

η_{th} x 2 → 0.5

$$h_{th} \sim \frac{T - T_{sink}}{T}$$

JÜLICH Forschungszentrum

Summary

Power plant costs drive fusion reactor design towards more difficult PFC challenges

ITER example

P_{fusion} = 500 MW S_{area} = 700 m^2 Cost ~ 20 G$ T_{wall} ~ 450 K f_{on} ~ 0.1 (duty factor)

$$\frac{\$cost}{\$_{electric}/yr} = \frac{20 \times 10^9 \$}{P_f \eta_{th} (f_{on})(8760\ hr/year) 0.1\$/kWh} \rightarrow \boxed{18\ yr}$$

P_f x 5 → 2500 MW

η_{th} x 2 → 0.5

f_{on} x 10 → 1

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Forschungszentrum

Summary

Power plant costs drive fusion reactor design towards both: more difficult plasma boundary physics and PFC challenges

ITER example

P_{fusion} = 500 MW S_{area} = 700 m^2 Cost ~ 20 G\$ T_{wall} ~ 450 K f_{on}~0.1 (duty factor)

$$\frac{\$cost}{\$_{electric}/yr} = \frac{20 \times 10^9 \$}{P_f \eta_{th} (f_{on})(8760\ hr/year) 0.1\$/kWh} \rightarrow \boxed{18\ yr}$$

P_f x 5 → 2500 MW | η_{th} x 2 → 0.5 | f_{on} x 10 → 1

P_f / S ~ 4 MW m^{-2} | T_{wall} > 1000 K | > yr-long fluence

All these differences to ITER make the plasma boundary physics more demanding and the PFCs more risky in a power plant

JÜLICH Forschungszentrum

Summary

- Through divertor geometry and plasma physics the divertor power load can be reduced below 10 MW/m^2, compatible with ITER reactor operation
  - Not yet clear whether the process of detachment is enough to reduce power loads below engineering limits for a power plant

- Our current understanding shows that divertor surface erosion rate can be kept low enough to achieve < 1mm/yr for tungsten in the divertor
  - If one can keep the plasma temperature < 5 eV, this is compatible with detachment

- Operation of a reactor at high temperature (800-1000K) has positive effects on
  - Reduction of T retention
  - ‘mending’ of neutron damage
  - Reactor engine efficiency of conversion of heat to electricity
  - Note we don’t yet have a demonstrated coolant that works at those temperatures

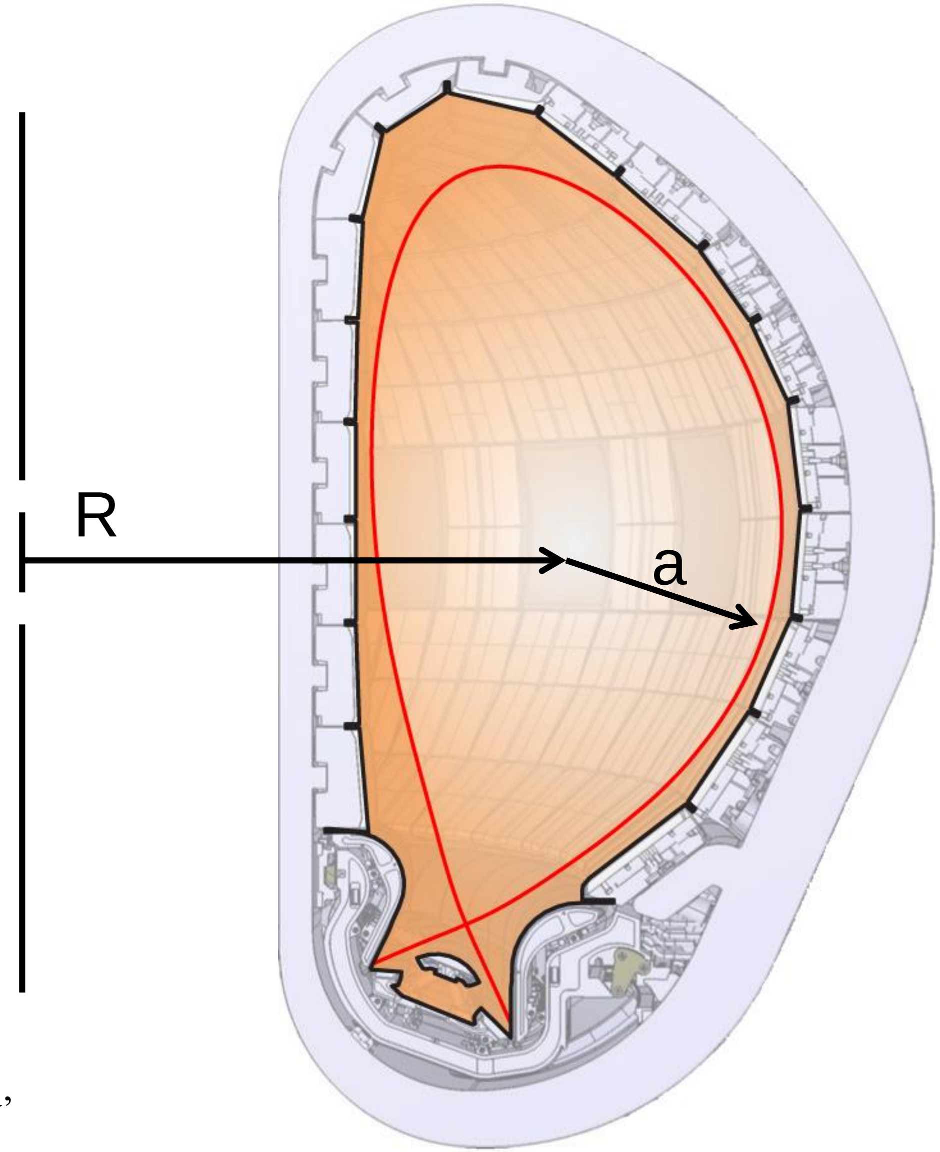
THANK YOU FOR YOUR ATTENTION

JANUARY 2019 | DETLEV REITER

A. ITER: Next step towards a magnetic fusion energy reactor

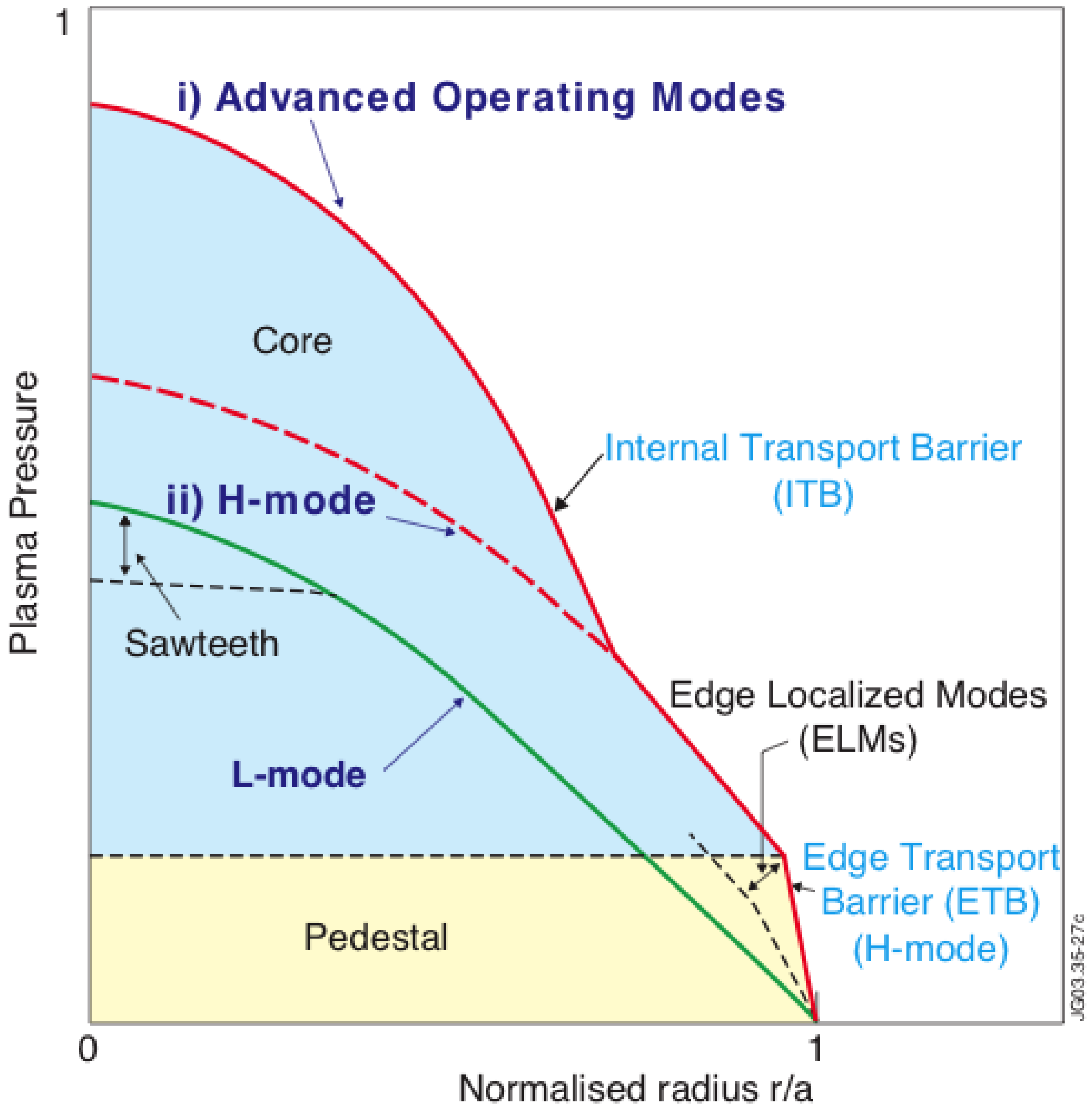
- **International Thermonuclear Experimental Reactor***
- ITER is being built in Cadarache France by 7 international partners – Goals are to
  - Produce 500 MW of fusion power
  - Achieve Q=(fusion power)/(power in) > 10
  - Study alpha particle physics
  - Test exhaust physics/technology
- > 8 years till first plasma

	ITER	Reactor (2.5 GW)
B toroidal field [T]	5.3	~5 (ITER value)
R [m]	6	6
a [m]	2	2
P_{SOL} [MW] = $P_{heat} + P_{\alpha} - P_{rad}$	100	500



*EU -EFDA report 'Fusion Electricity – A roadmap to the realisation of fusion energy'

Summary: Impact of PFCs on fusion gain



Summary: Impact of PFCs on fusion gain

$Q_{DT} = P_{fus} / P_{heat} \sim p_{DT}\, \tau_E\, f(Z_{eff})$

Profile stiffness: T(0) / T(a) ~ const

Critical temperature gradient

Core plasma DT pressure

Impurity accumulation

DT fuel dilution with Z_{eff}

Turbulent transport

MHD equilibrium & stability

Collisional transport

need cool, dense edge plasma

or active ELM size control: pellets, RMPs

Edge plasma DT pressure

radiative impurity seeding

need cool, dense edge plasma

ELM size decreases with edge collisionality

Edge localised modes (ELMs)

Edge & SOL transport: collisional and turbulent

partially detached divertor operation

Transient loads on PFCs

Steady loads on PFCs

must not exceed: ~ 0.5 MJ/m2 in 250 us

Erosion, ablation, melting, cracking

must not exceed: ~10 MW/m2; ~10 eV

Achieving fusion burn vs. exhaust criteria: a complex system

achieve ← $Q_{DT} = P_{fus} / P_{heat} \sim 10$ → maintain

- **Fuelling**
  - gas
  - pellets
  - beams
- **Heating**
  - RF
  - NBI
  - alpha
- **Confinement**
  - equilibrium
  - stability
  - transport

- **Particle exhaust**
  - He ash
  - Intrinsic Z
  - Extrinsic Z
- **Power exhaust**
  - neutron
  - photon
  - plasma
- **First wall design**
  - mechanical
  - thermal
  - nuclear

+ current drive,

disruptions, tritium, dust,…