Power exhaust and plasma-wall interaction in tokamaks

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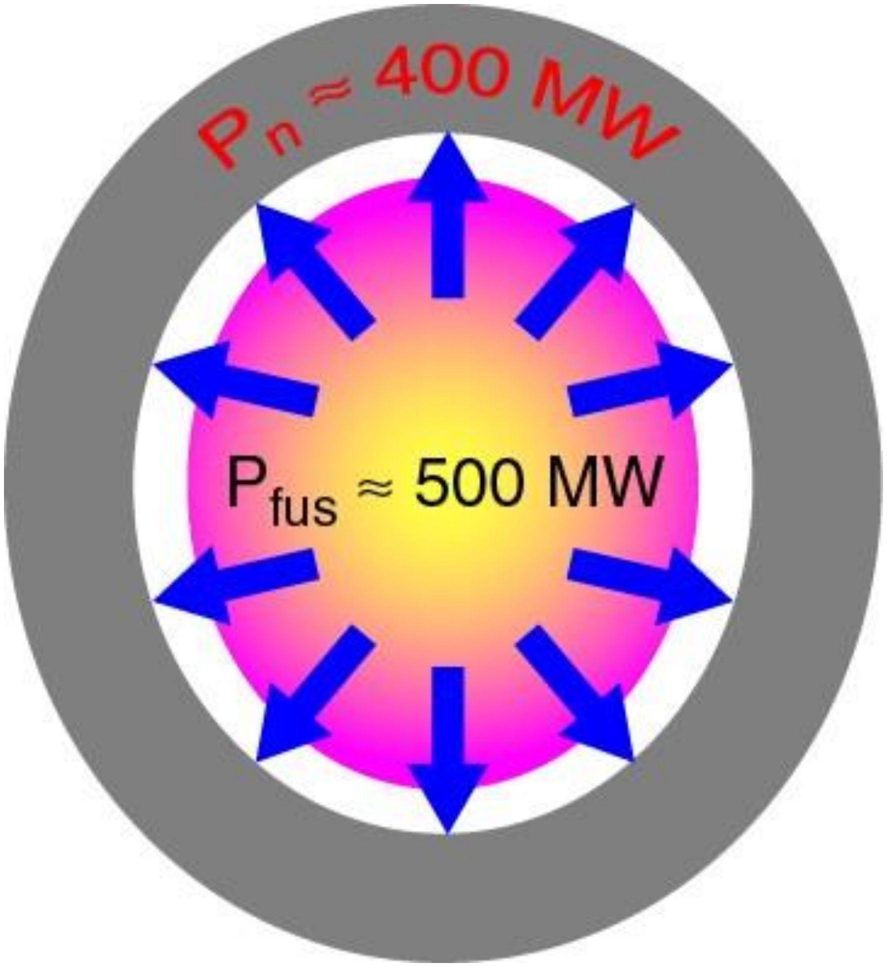
Outline

- **(Neutron and particle) environment for plasma-facing materials:**
  - Need for power exhaust (due to particle bombardment)
  - Material options
  - Impact of neutrons on materials
  - Material migration and tritium retention
- **Physics models to describe scrape-off layer plasma and plasma-material interactions ⇒ required for extrapolation toward future fusion power plants**

The plasma-surrounding (material) walls (vessel) provides a containment and vacuum conditions



- **Fusion requires a base pressure of about 10^{-8} mbar ⇒ pumping system**
- **Power in α-particles and auxiliary must be (eventually) extracted through walls**
  - Power in neutrons converted to heat in blanket wall
  - Tritium breeding
- **Helium removal via in-vessel pumps**

The choice of materials in fusion reactors is driven by plasma/neutron-wall interactions

- **Primary issues are:** material lifetime, tritium inventory, and debris formation

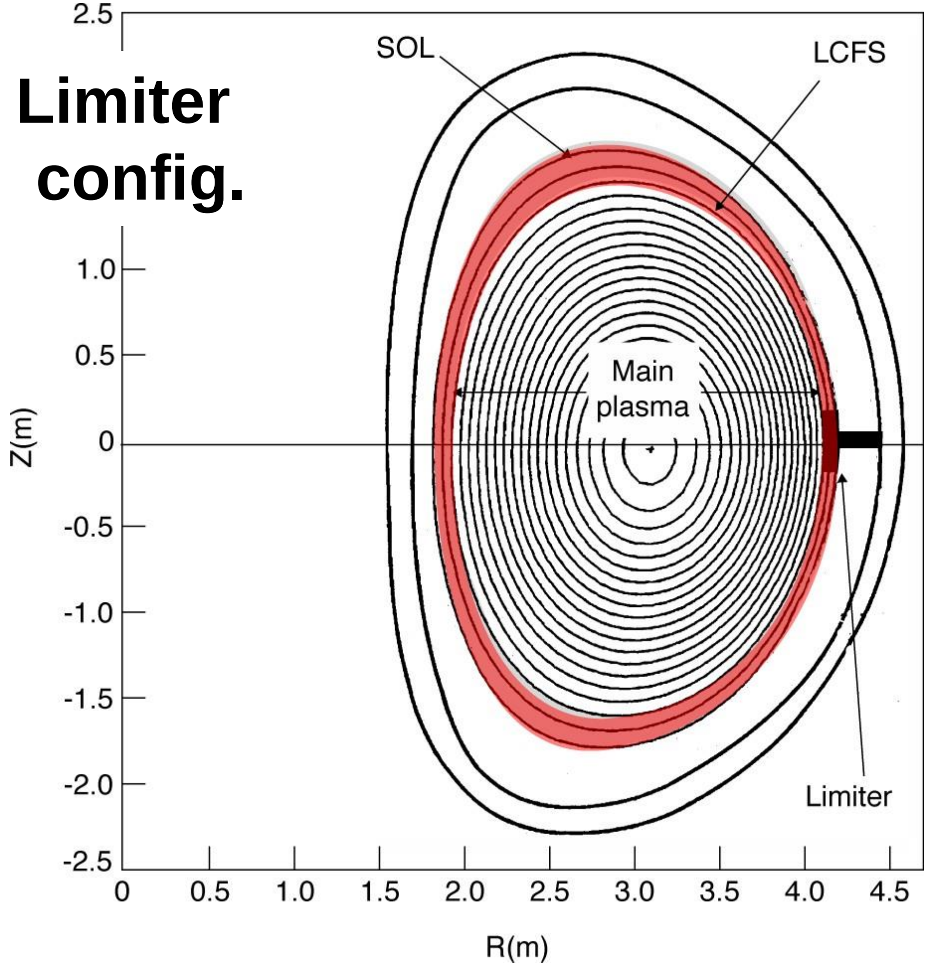
⇒ **Economical/practical aspect, but tritium and debris formation also an additional safety aspect**

- **D-T fusion reaction:** $D + T \rightarrow \alpha$ (3.5 MeV) + n (14.1 MeV)
  - α-particle for plasma self-heating, neutron for blanket heating ⇒ 1 g D-T produces 67.6 GJ in α-particles and 271.8 GJ in neutrons (1kg of coal produces 24 MJ)
  - Following thermalization, α-particles become helium ash ⇒ need to be removed from system

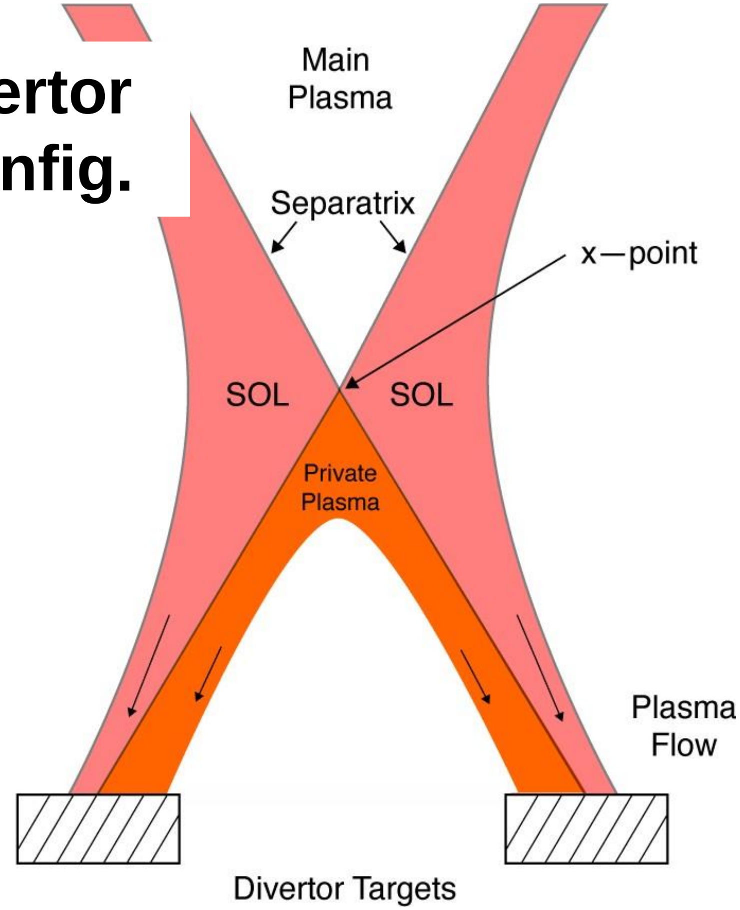
⇒ **In future fusion power plants, both power and particle exhaust and activation of surrounding wall are issues**

Diverting the magnetic field lines to dedicated regions inside the vessel controls plasma-wall interaction

Limiter config.

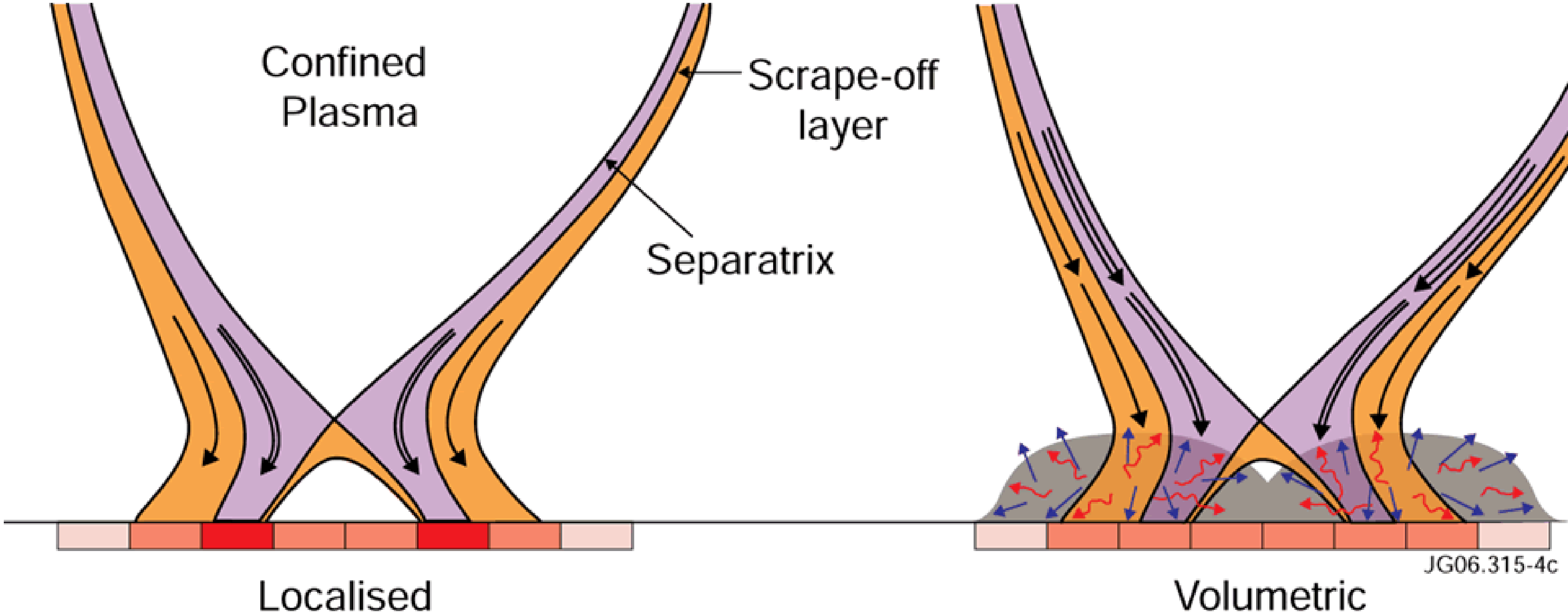


Divertor config.



- **Isolation of divertor from main chamber by adding coils at the bottom of device to produce magnetic null**

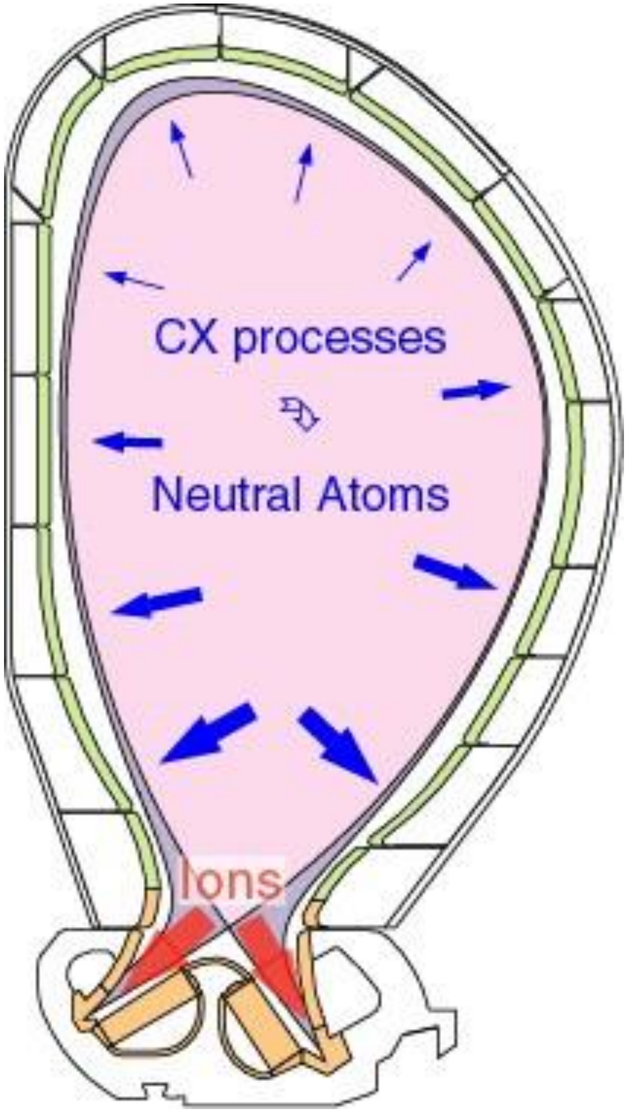
Low divertor temperatures, and thus heat fluxes to the divertor target plates, are achieved in detached conds.



- **Standard picture:** power and particle flow from the confined plasma via the scrape-off layer onto the divertor targets ⇒ transfer of kinetic energy to surface

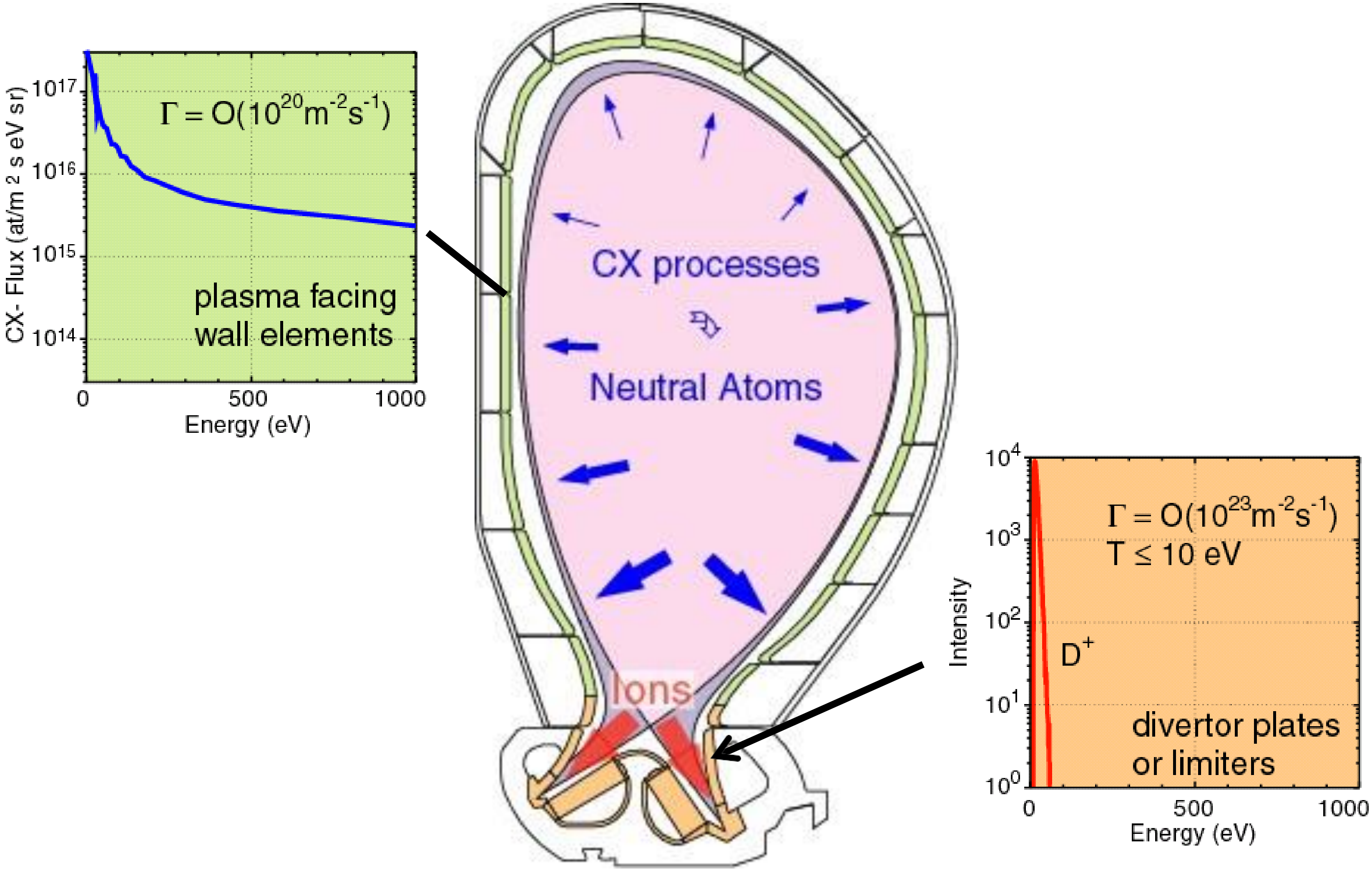
- **Detachment:** recombine plasma to neutrals in front of targets ⇒ power loss in radiation and recombination

Direct contact of the plasma with the vessel wall must be limited to certain (controlled) areas

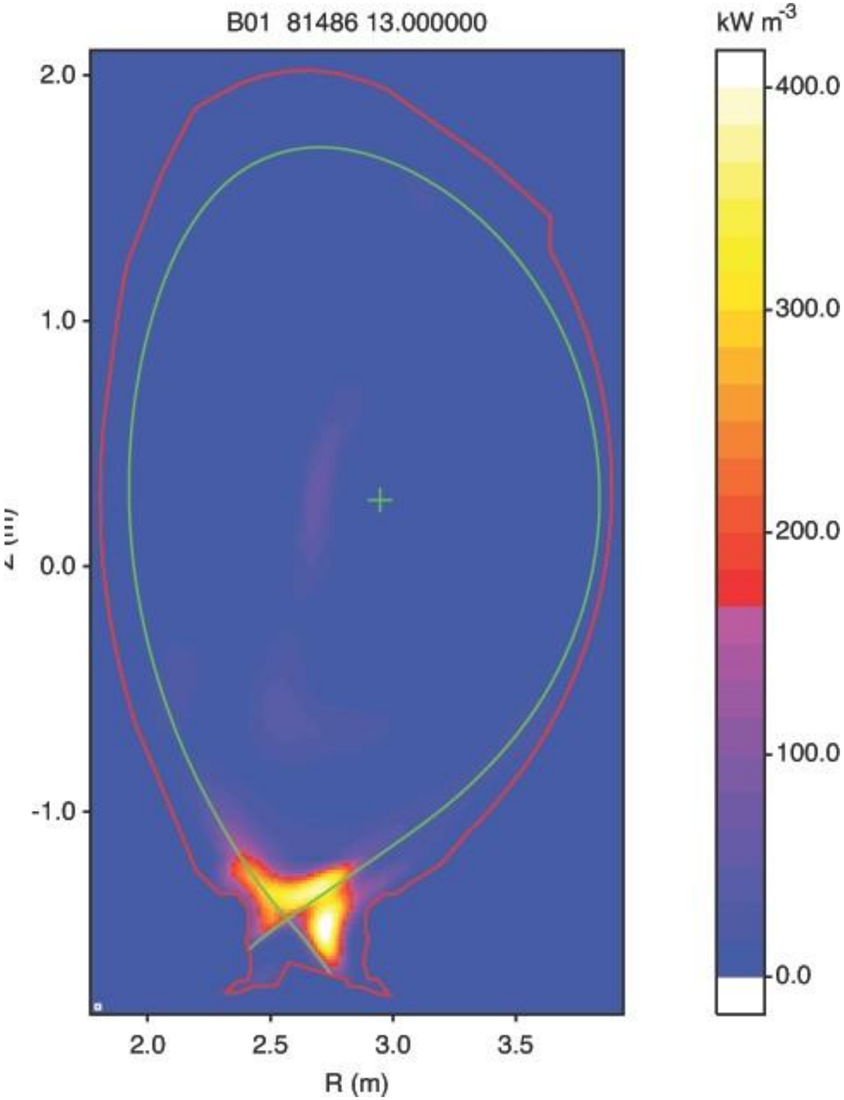


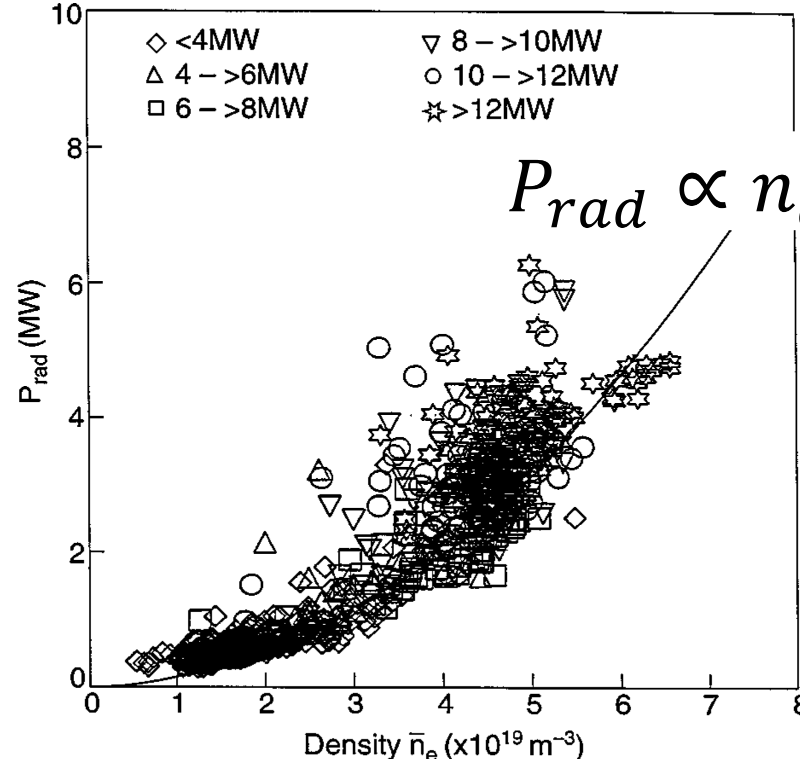
- **Power in the plasma is predominately radiated ⇒ (isotropically) spread over wall**
- **Remaining power in escaping particles**
  - Ions following field lines ⇒ limiter and divertor plates
  - Charge-exchange neutrals ⇒ main chamber and divertor plates

No uniform engineering boundary conditions since escaping particles have a wide range of fluxes and energies



In tokamaks, plasma radiation is concentrated in the divertor region (desired situation!)

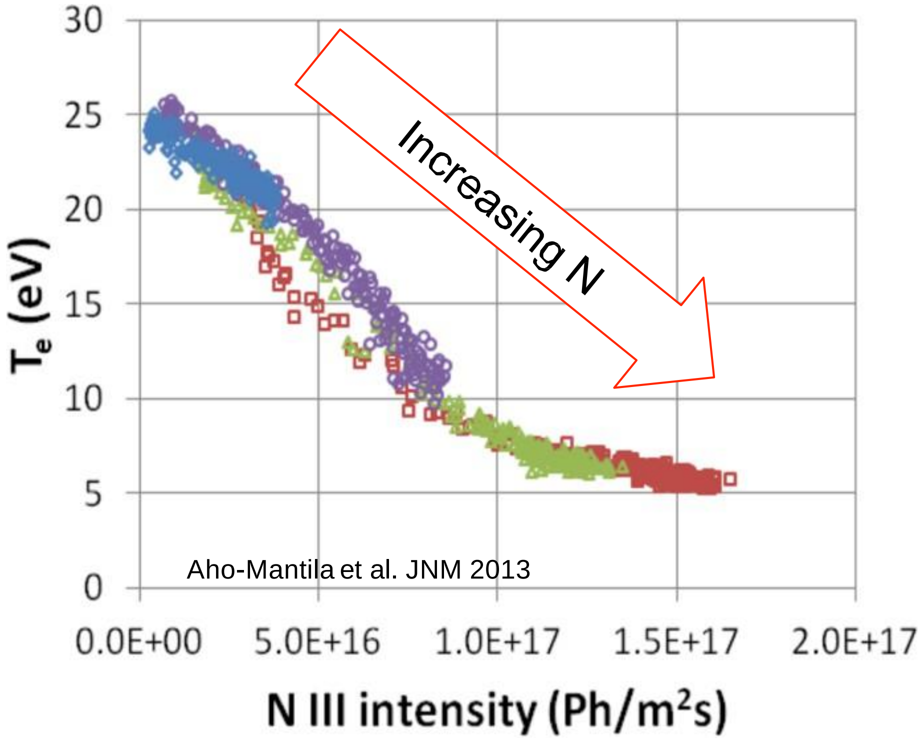


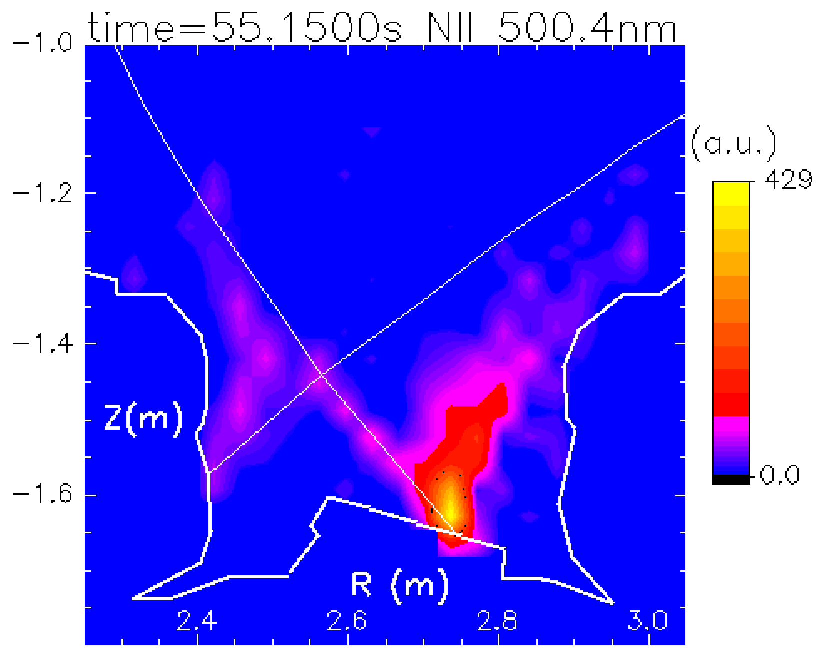


$$P_{rad} \propto n_e^{1.95}$$

- **Main radiation in hydrogen and impurity, and Bremsstrahlung**

Detachment can be achieved by operating at high density and/or by intentionally injecting impurities

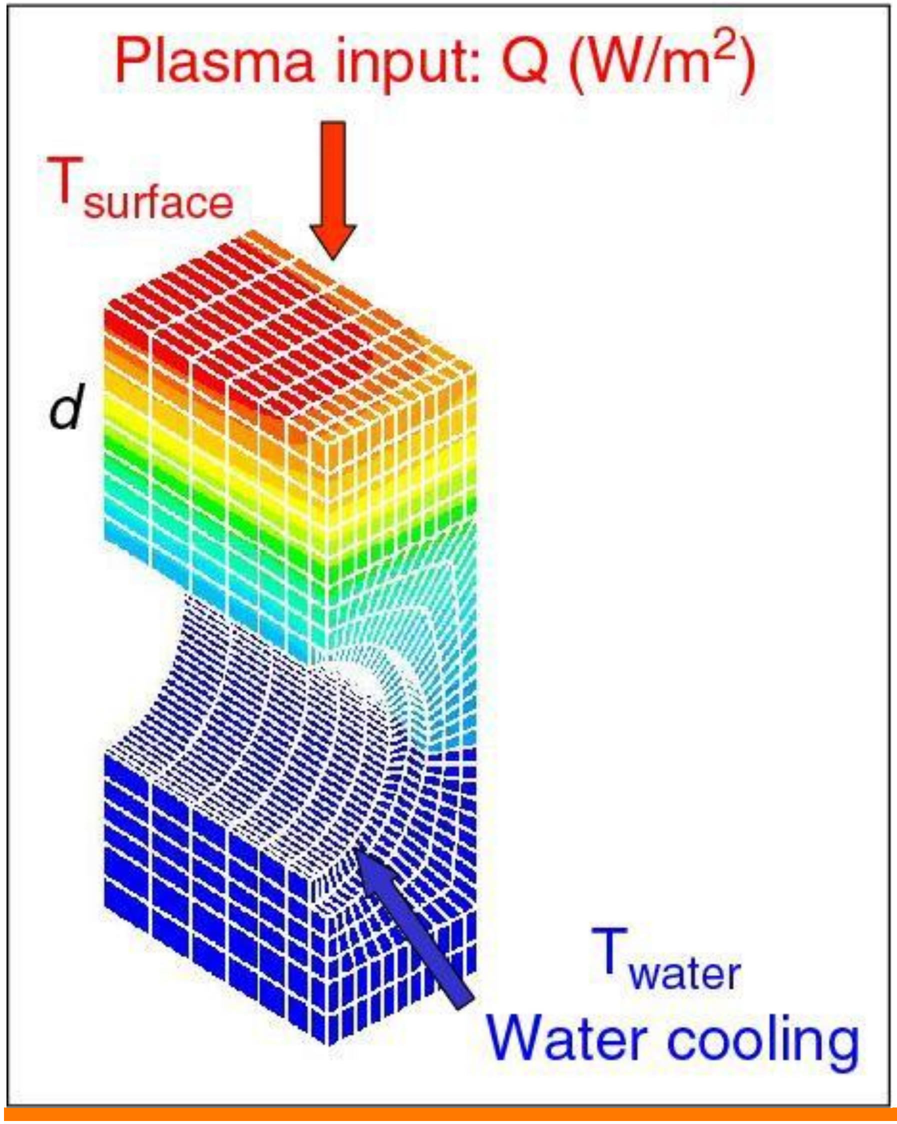




Järvinen et al. PSI 2014

- **Line radiation of nitrogen is strongest at temperatures characteristic of the scrape-off layer**
- **Low-dose "seeding" leads to reduction of T_e to 5 eV ⇒ reduced surface heating**

Steady-state heat removal may be computed using standard (finite element) techniques



- **For given amour material of density (ρ) and thickness (*d*), and with a thermal heat conductivity (λ) and heat capacity (c)**

⇒ Steady state conditions:

$$\Delta T = T_{surf} - T_{water} = Q\, d/\lambda$$

- **$Q \approx 10$ MW/m^2, $\lambda \approx 2000$ W/mK, $d \approx 2$ cm ⇒ $\Delta T \approx 1000$ ℃**

The energy in transient events is too high for any material to absorb ⇒ need to mitigate plasma events

- **Surface temperature follows square-root law with time:**
  $T(t) \propto P \times \sqrt{t}$
- **ITER ELMs ≈ 15 MJ, deposition time ≈ 0.1 – 0.5 ms, deposition area ≈ 6 m^2 ⇒ power density ≈ 10 GW/m^2**

⇒ T_{max} ≈ 6000 ℃, penetration depth ≈ 0.15 mm

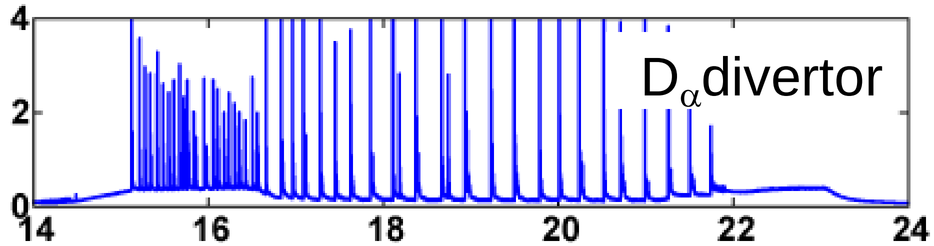
- **Sublimation temperature for graphite ≈ 2200 ℃**
- **Melting and boiling temperature of W = 3410 ℃ / 5560 ℃**

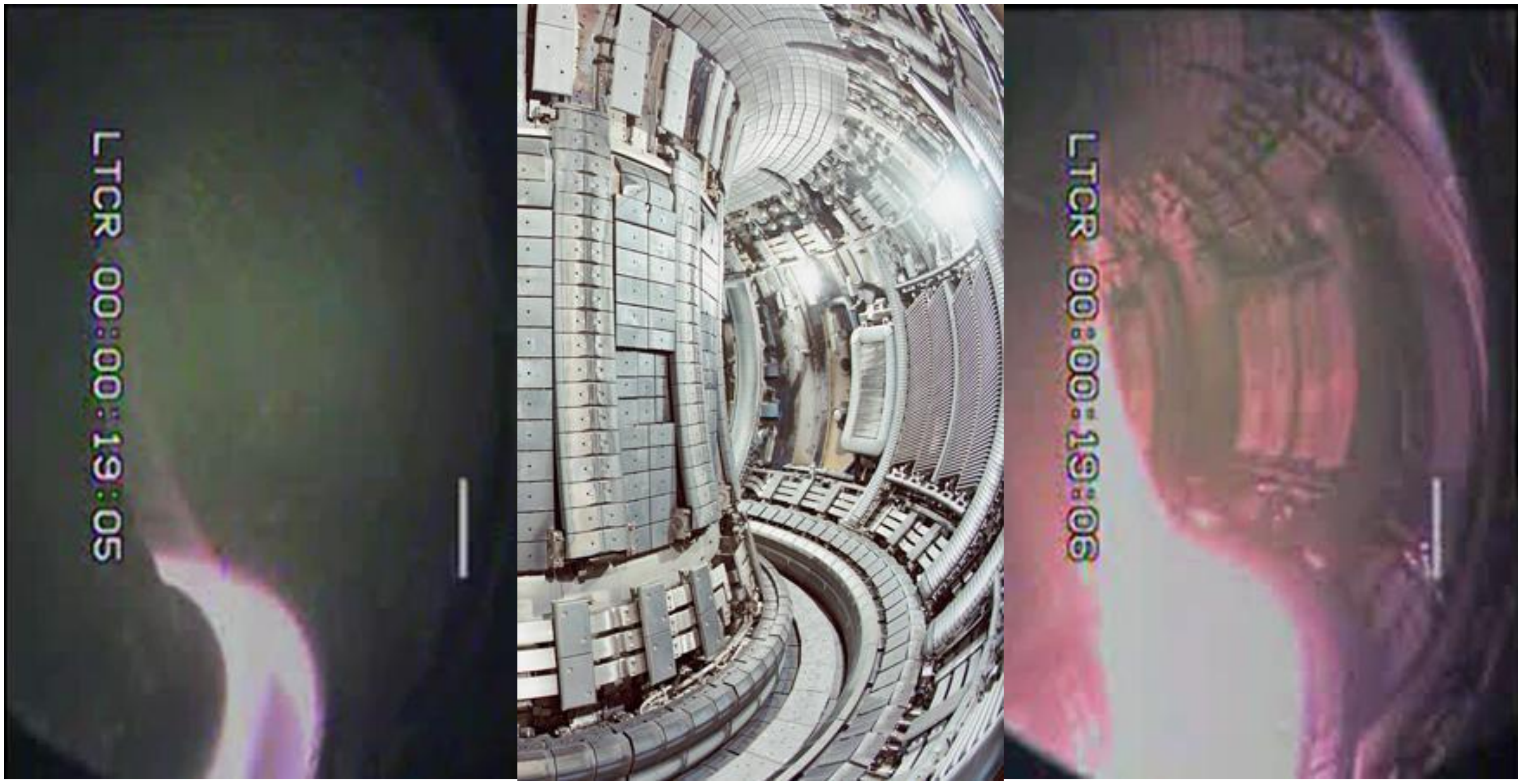
⇒ Graphite will sublimate rapidly, metals will melt

⇒ No immediate material solution, need to mitigate plasma events!

Transient event in the plasmas lead to sudden excursions in heat load and surface temperature

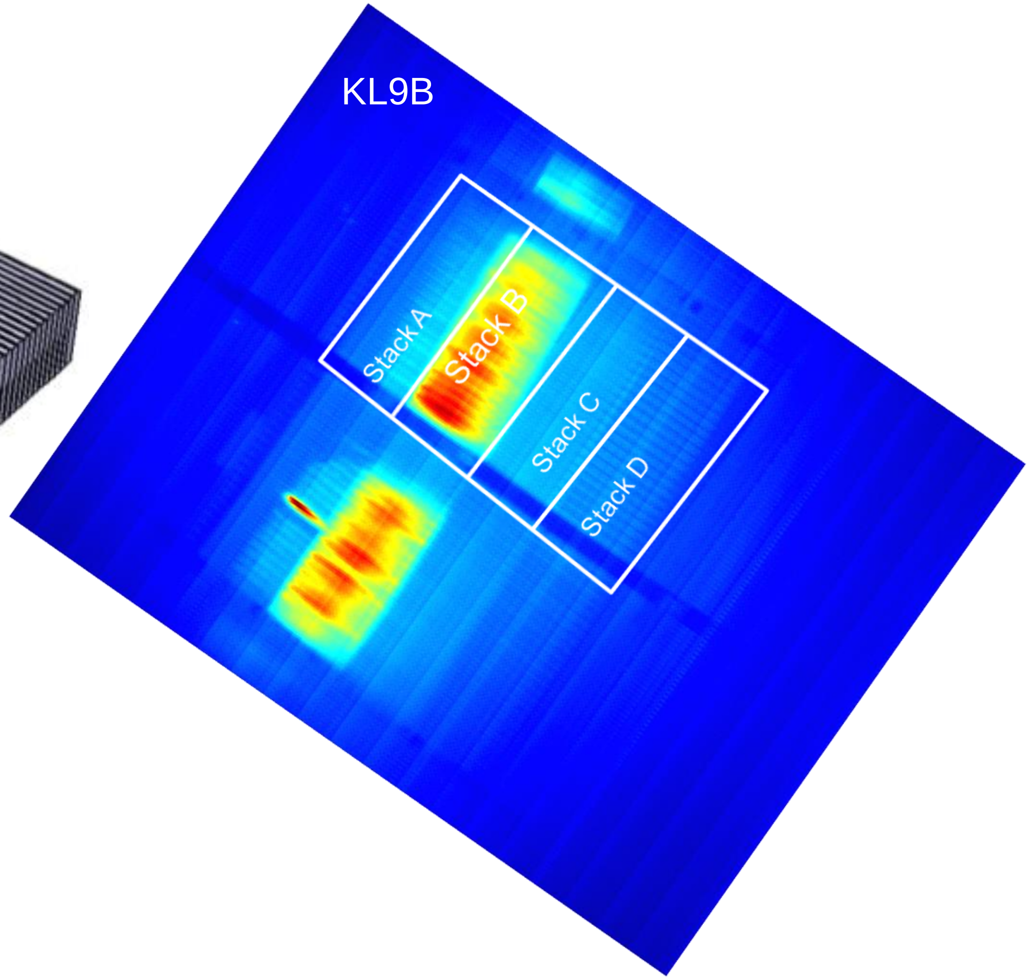
- **ELMs: 2 MJ/m² in 0.5 ms**
- **Disruptions: > 2 MJ/m² in ms**





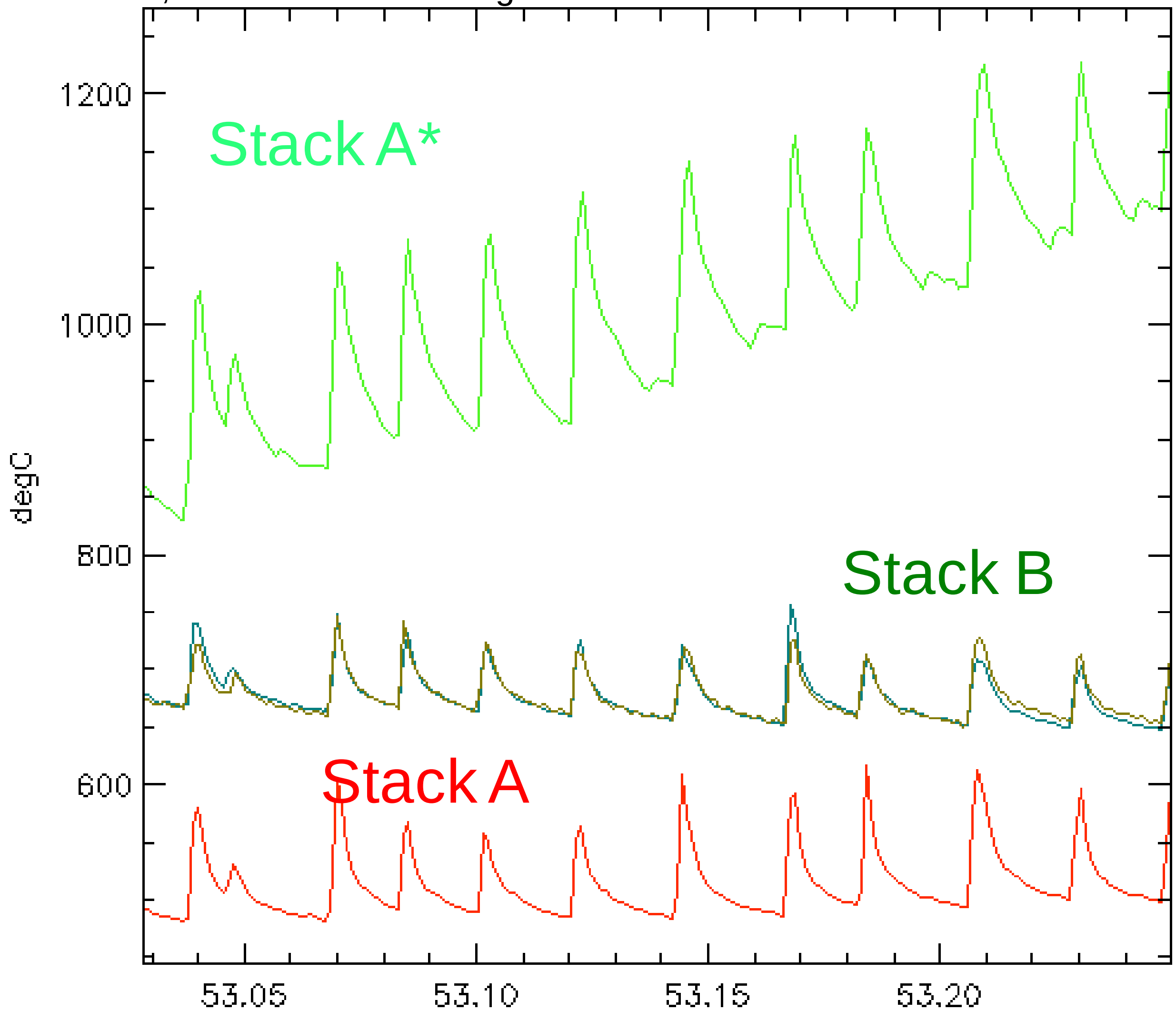
Dedicated JET experiments to intentionally melt and damage the tungsten divertor plates

Coenen, Arnoux et al. JET August 2013



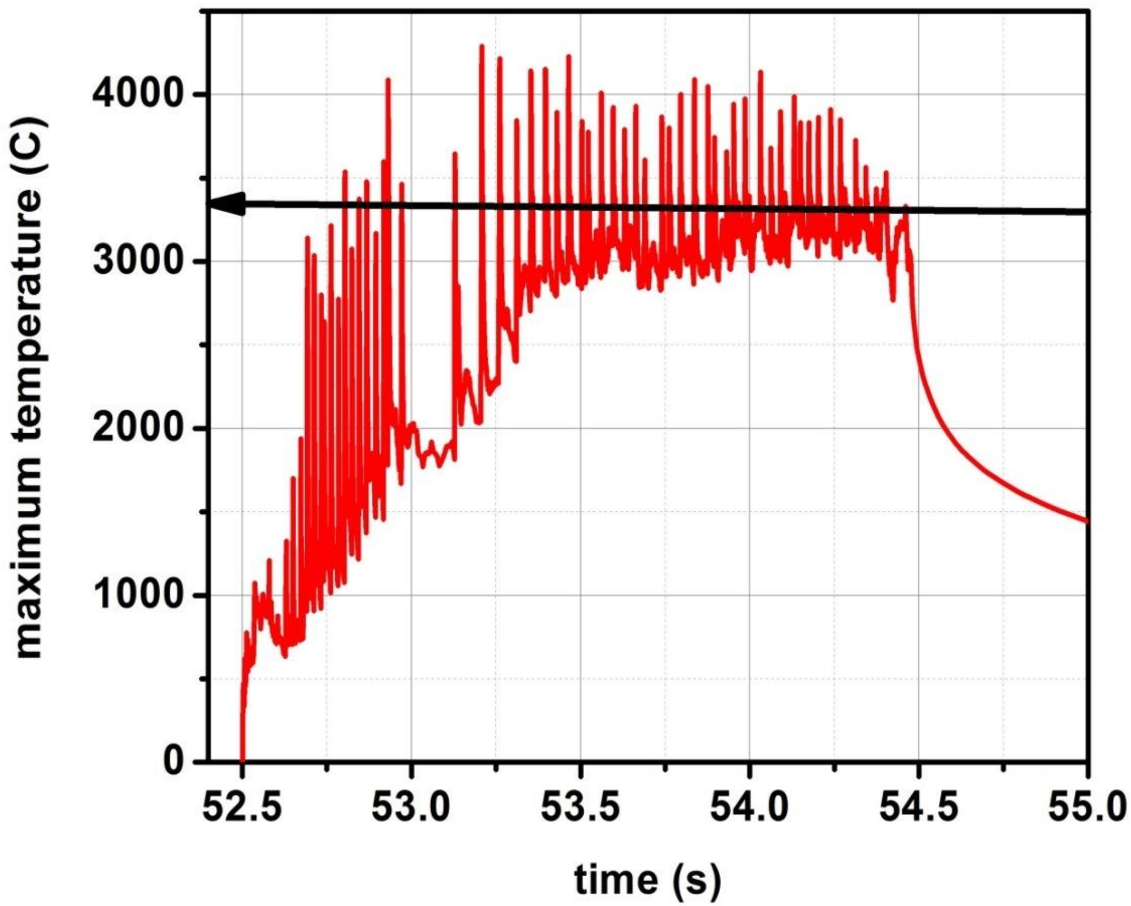
Edge localized modes can raise surface temperature transiently by more than 50%

Coenen, Arnoux et al. JET August 2013



- **Significant melting of W at the location of the elevated lamellae**
- **Careful alignment of (all the) other lamellae prevented W melting**

Controlled flash melting by ELM induced temperature excursions

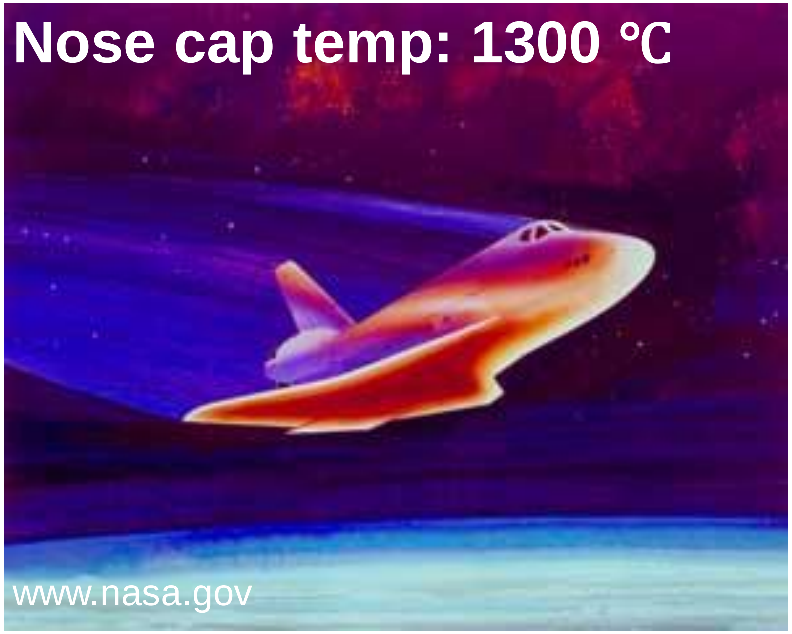


- **Target plate base temperature heated up to ≈ 2800 ℃**
- **ΔT by ELMs ≈ 800–1000 K**

⇒ **ITER decision pro tungsten divertor from start of operation**

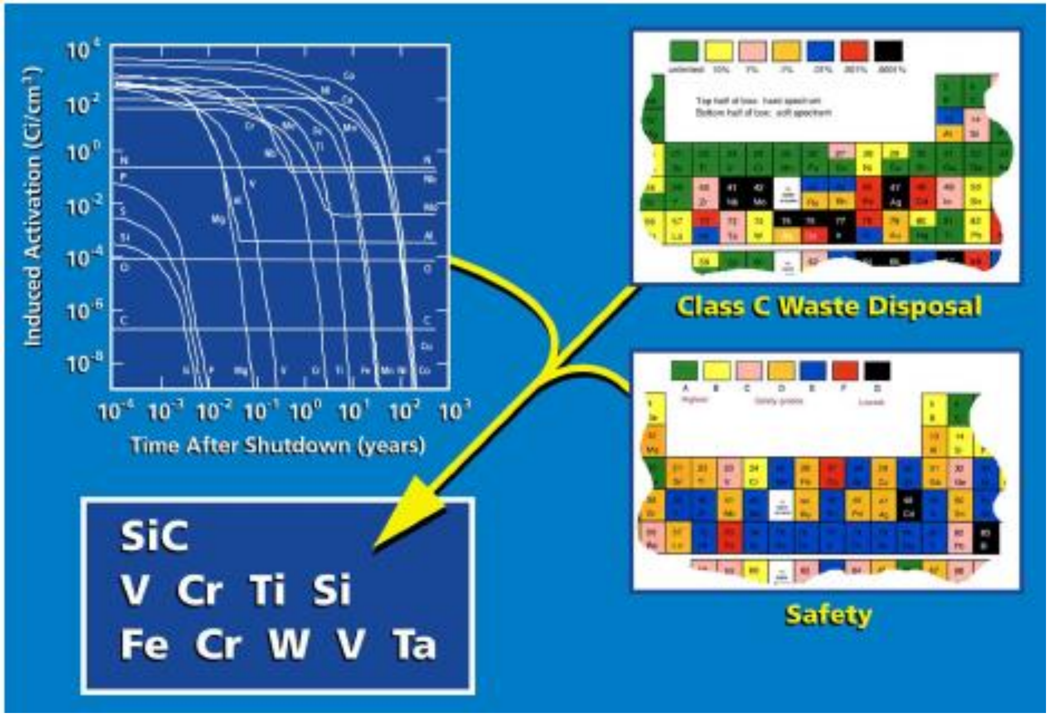
[B. Bazylev et al., 21st PSI 2014, J. Coenen et al., 21st PSI 2014]

Fiber-carbon composites are some of the most heat-resilient materials known to-date



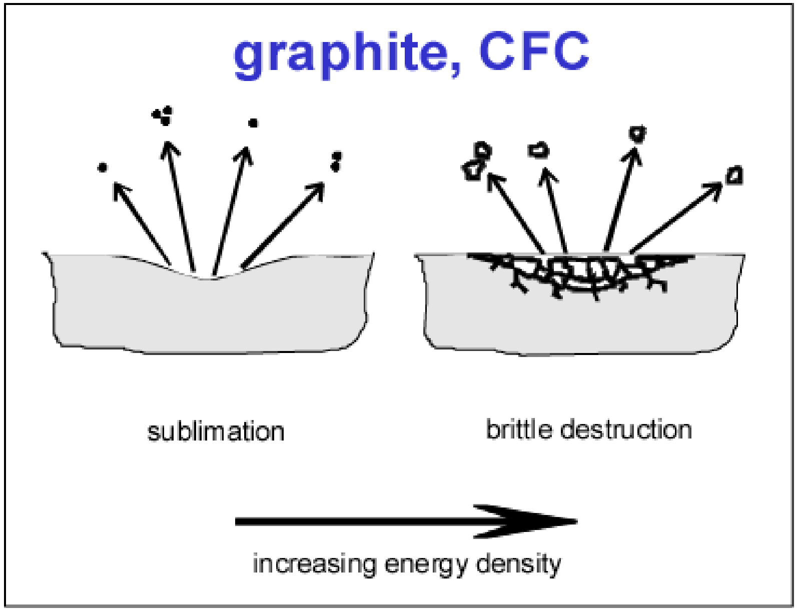
- **Principle factors that ensure spacecraft reentry are shape of the vehicle, angle of reentry and usage of range of materials**
- **Compared to continuous fusion device operation, replacement of space shuttle components is routine …**

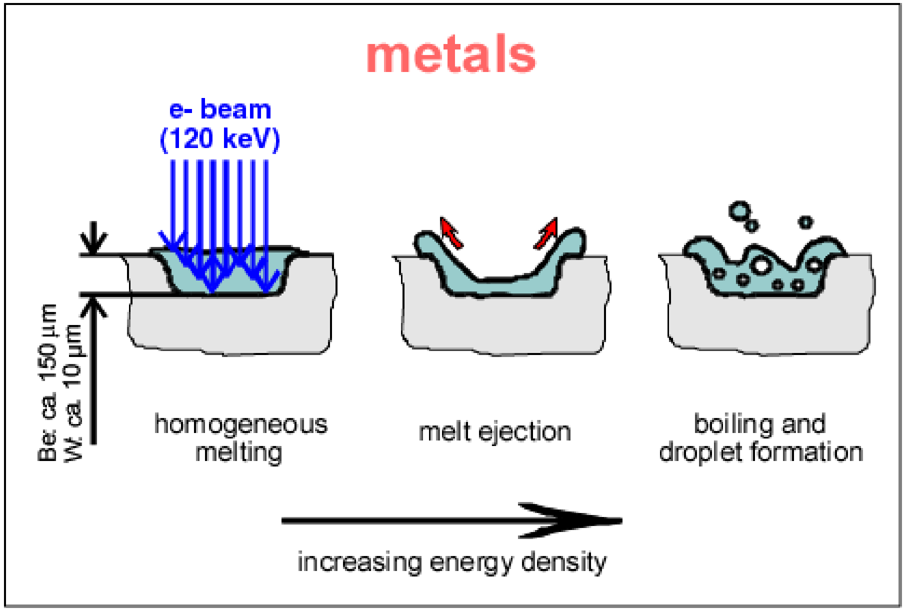
Choice of materials in fusion devices depends on the stage and readiness of building a fusion power plant



- **Heat flux and erosion capability, high duty cycle operation**
- **Compatibility of materials with plasma performance**
- **Change of thermo-mechanical properties in neutron environment, transmutation**
- **Tritium retention, safety**
- **Post-operation waste management**

While carbon-based plasma facing components sublimate, metals melt above a certain temperature

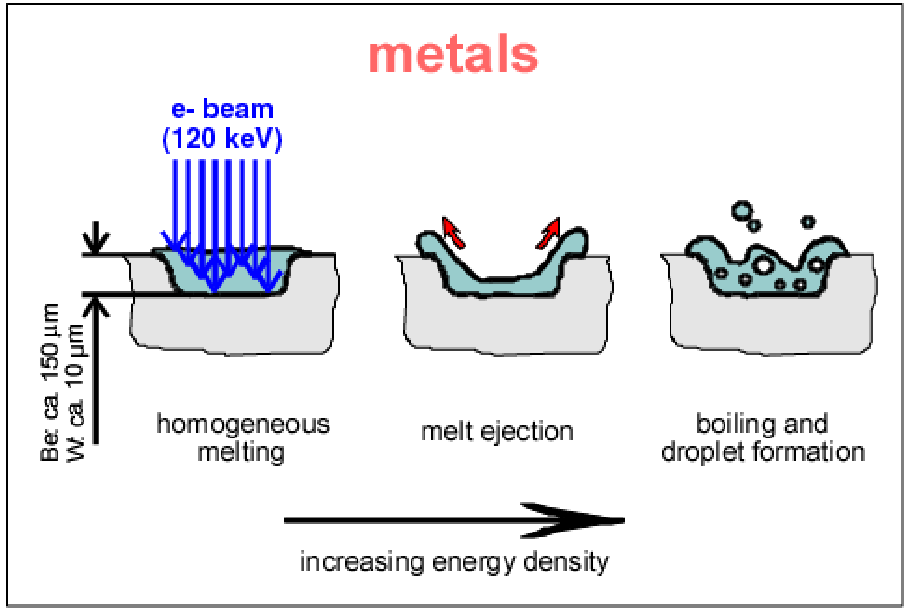


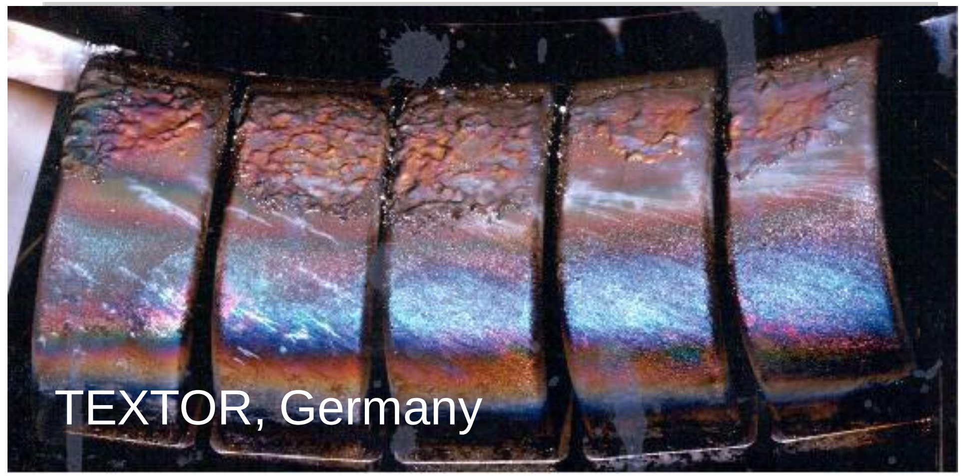


FOR CARBON:
Above a certain power load (threshold) emission of debris
⇨ ***BRITTLE DESTRUCTION***

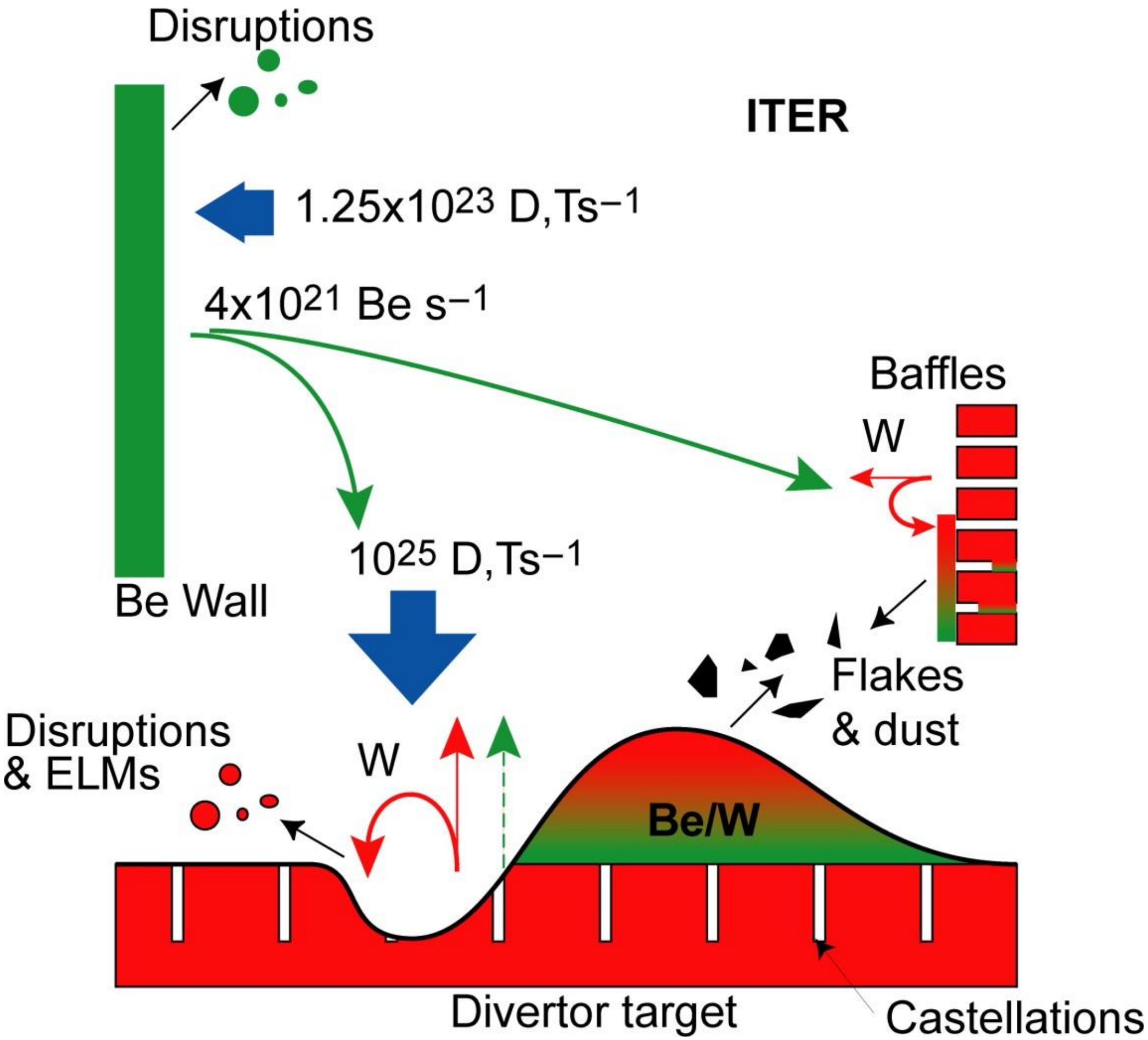
FOR METALS:
Splashing
Formation of droplets
Formation of dust

While carbon-based plasma facing components sublimate, metals melt



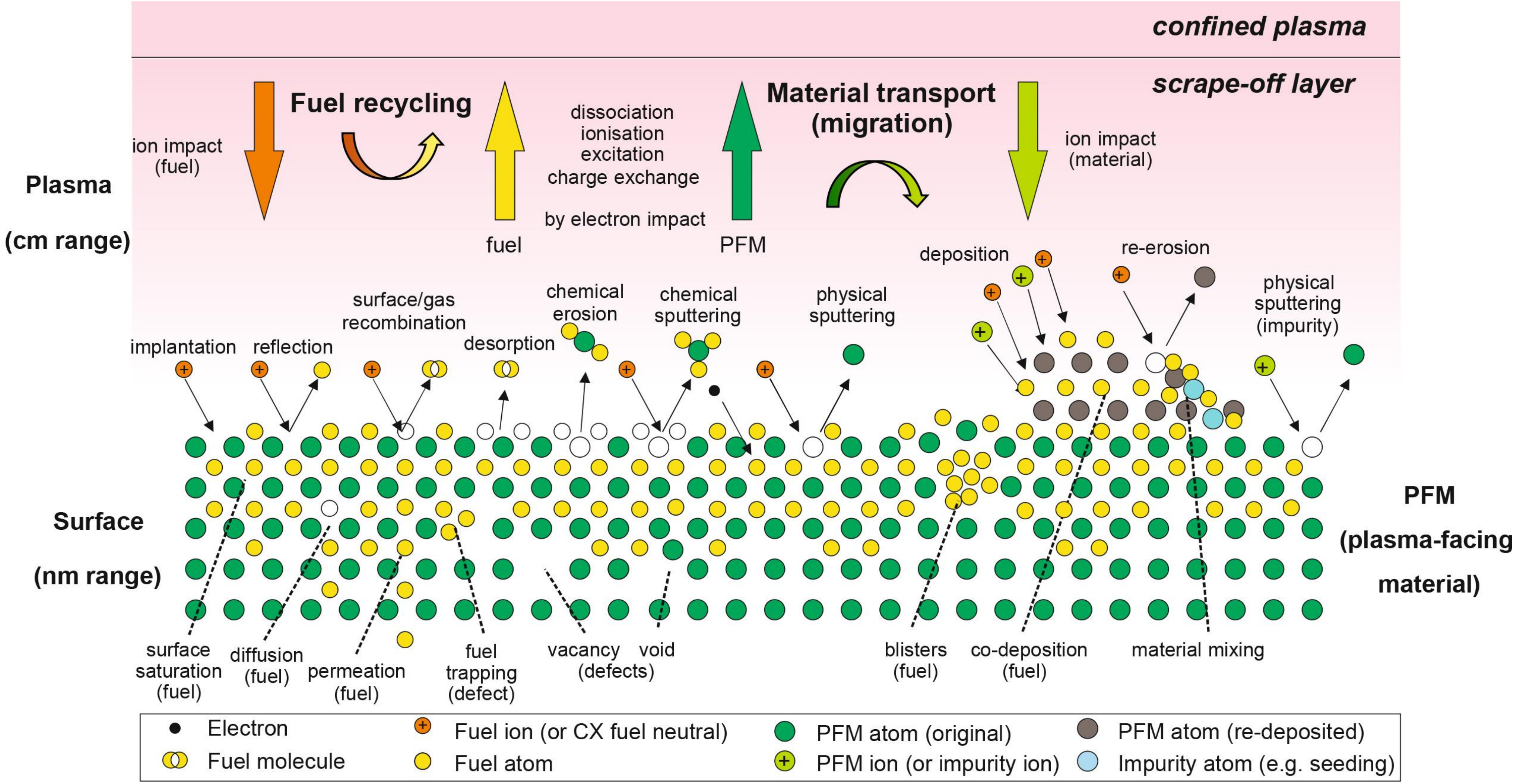


Both globally and locally the various wall elements are in dynamic equilibrium



- **Be erosion ⇒ layer formation ⇒ T retention**
- **Be transport into remote areas ⇒ T retention in plasma-shadowed areas**
- **W erosion, prompt re-deposition**
- **Be/W dust formation**

A wide range of processes take place at the plasma-material interface, including sputtering and implantation



Material migration leads to long-term modification of plasma-facing components and alloy formation

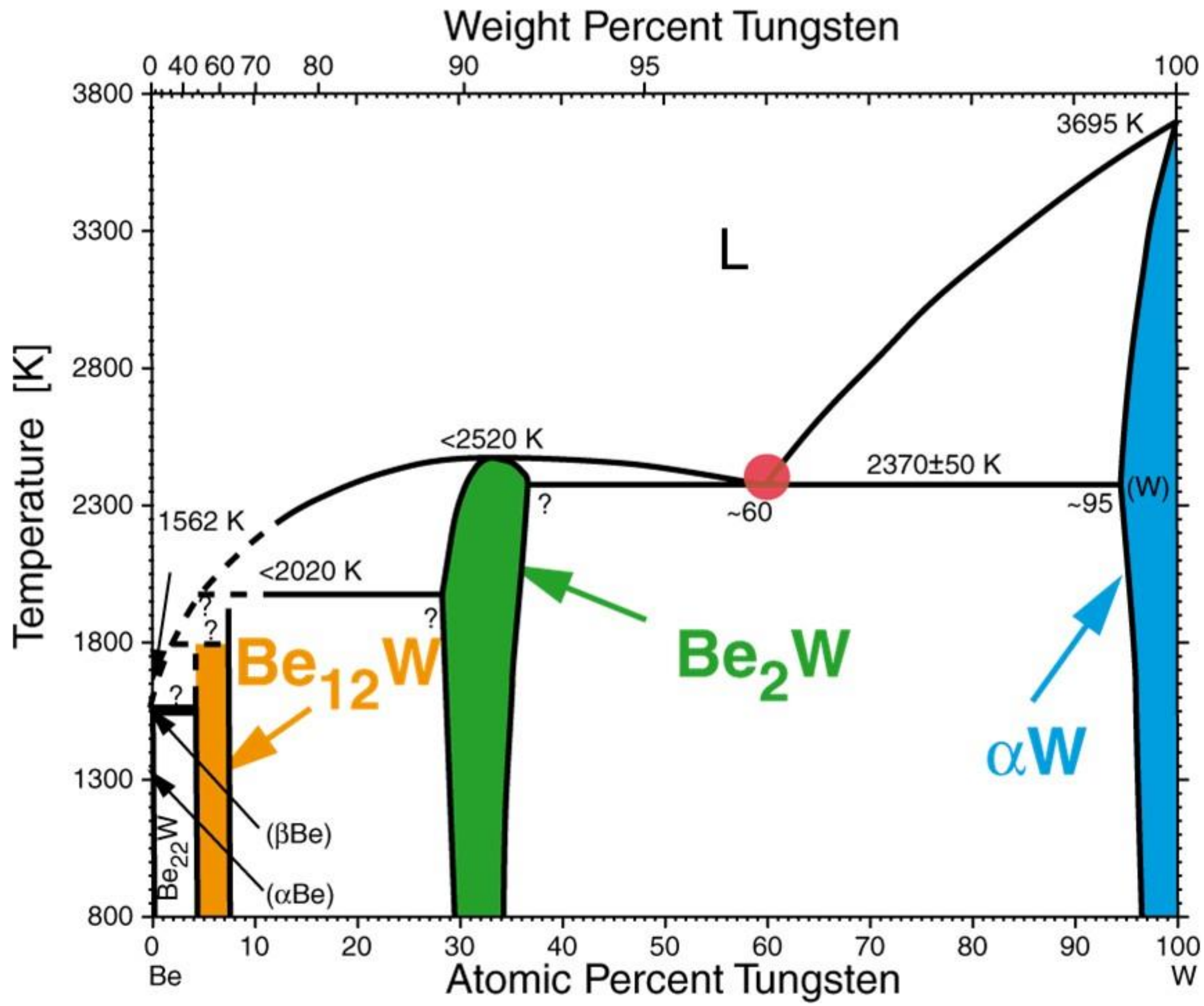
Plasma

Fuel ions + atoms (charge exchange) + impurity ions bombard 1st wall

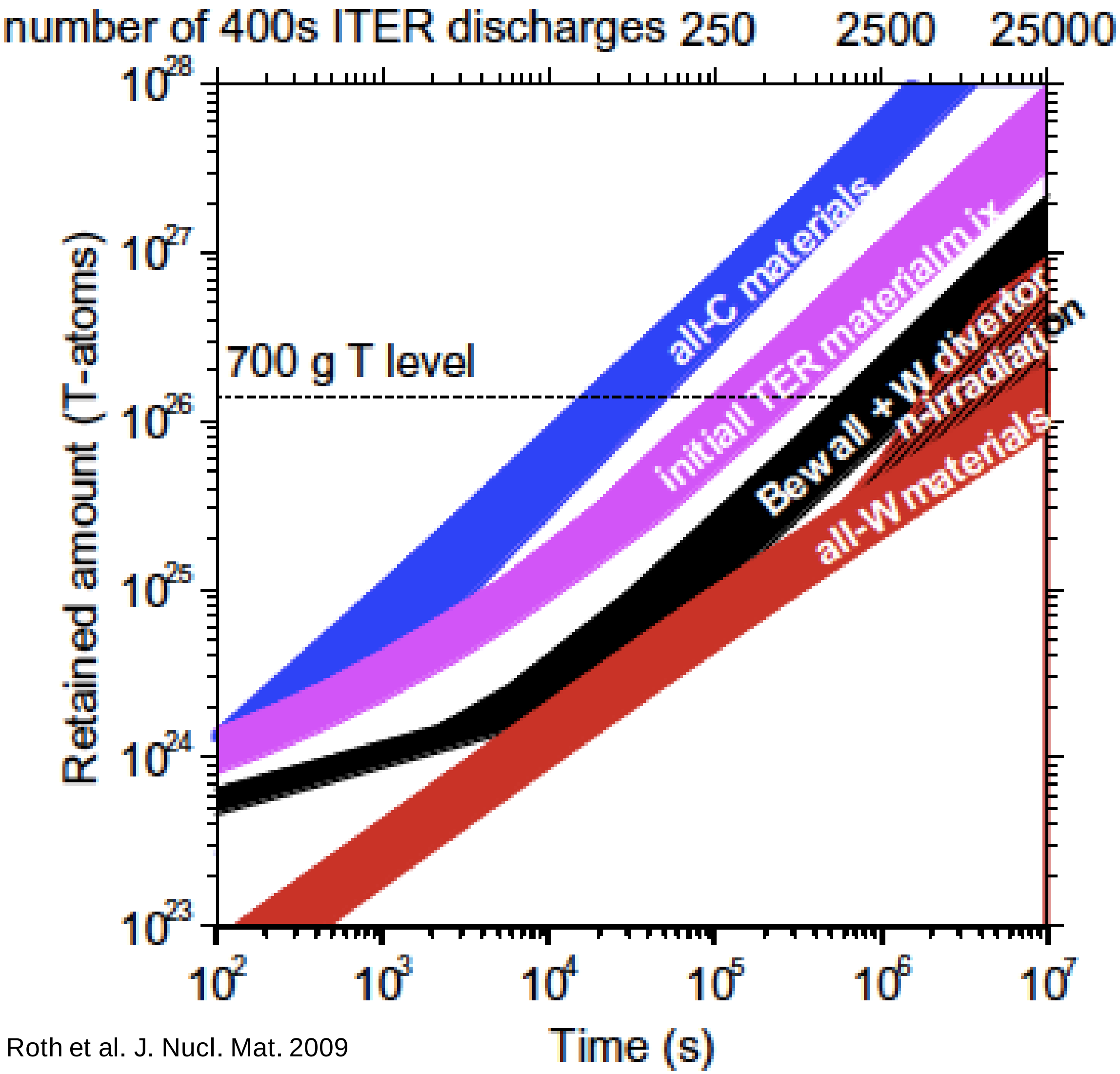
Wall materials

Erosion → Transport → Deposition → Re-erosion → Transport

Formation of Be-W alloys on tungsten surfaces reduce the melting temperature from 3695 to ~ 1570 K



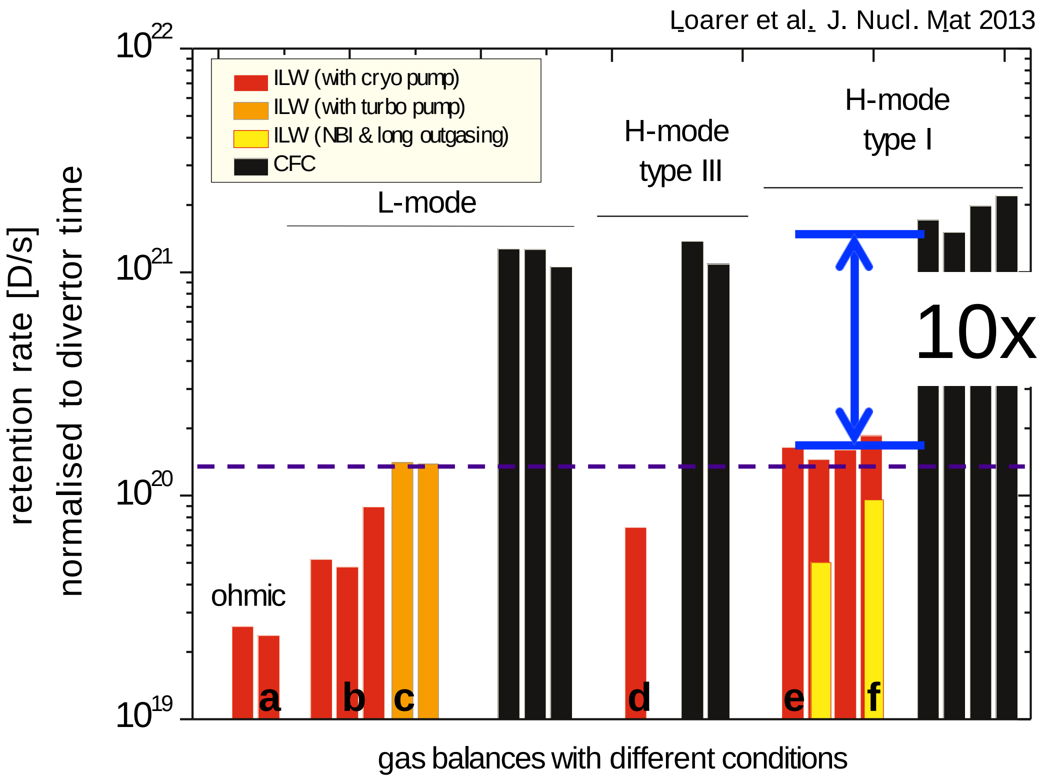
Implantation and co-deposition of tritium on plasma-facing surfaces administratively limits ITER operation

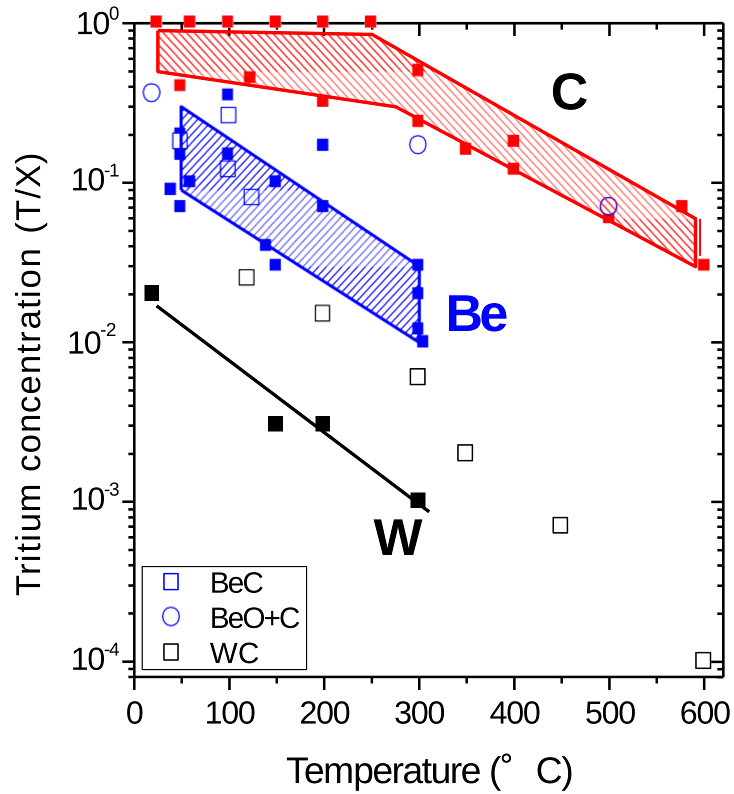


Roth et al. J. Nucl. Mat. 2009

- **Tritium is radioactive, most hazardous to the public in T_2O ⇒ tritium management**
- **Metals are significantly less susceptible of absorbing tritium than carbon ⇒ preferred (and decided!) for ITER**

Changing the JET wall from carbon to beryllium and tungsten reduced the hydrogen retention by 10x

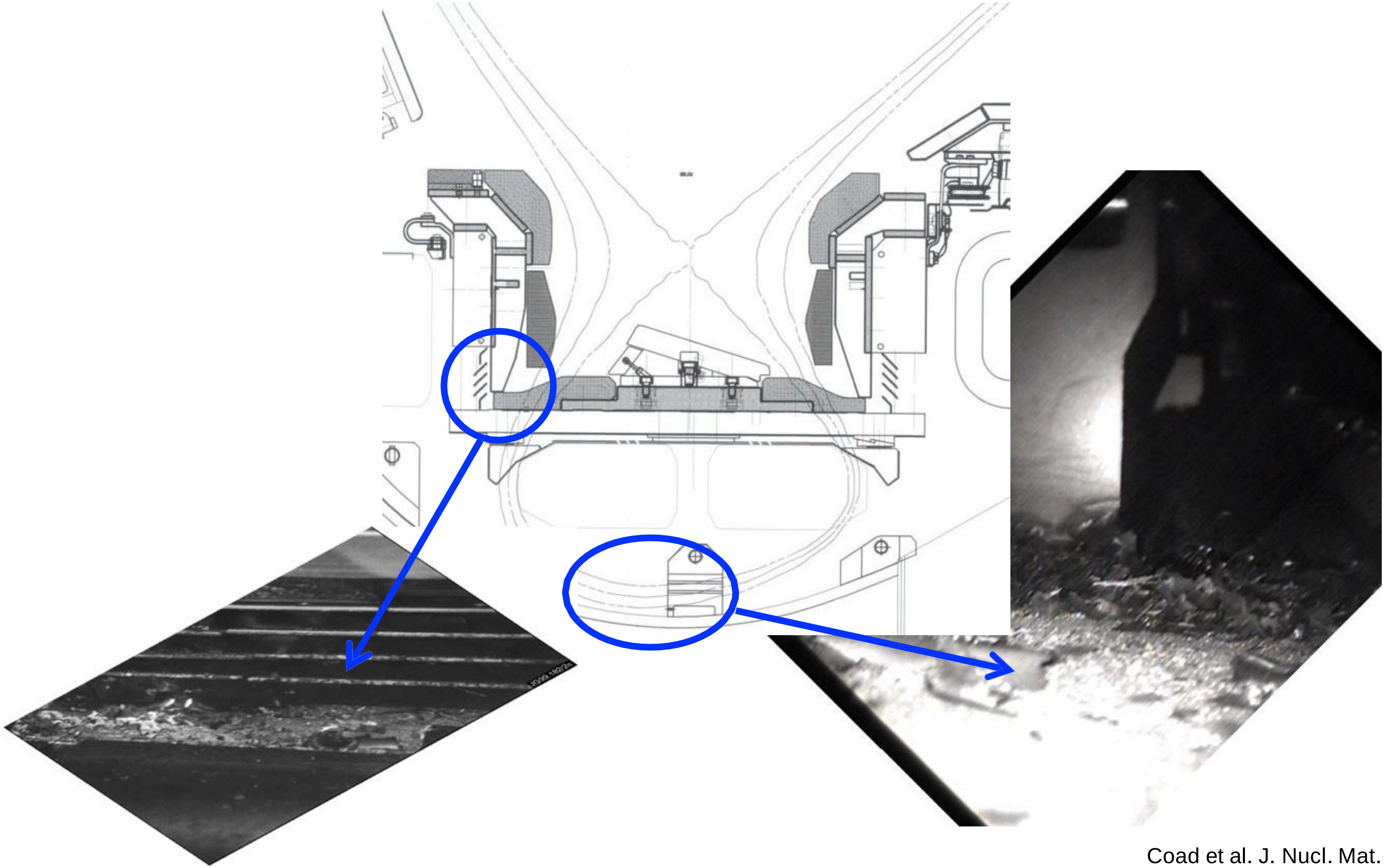




- **Hydrogen retention likely due to co-deposition with beryllium**

⇒ **Another 10x expected for going to full tungsten device**

Thick deposition layer can also delaminate and thereby forming radioactive and chemically reactive dust



Coad et al. J. Nucl. Mat. 2001

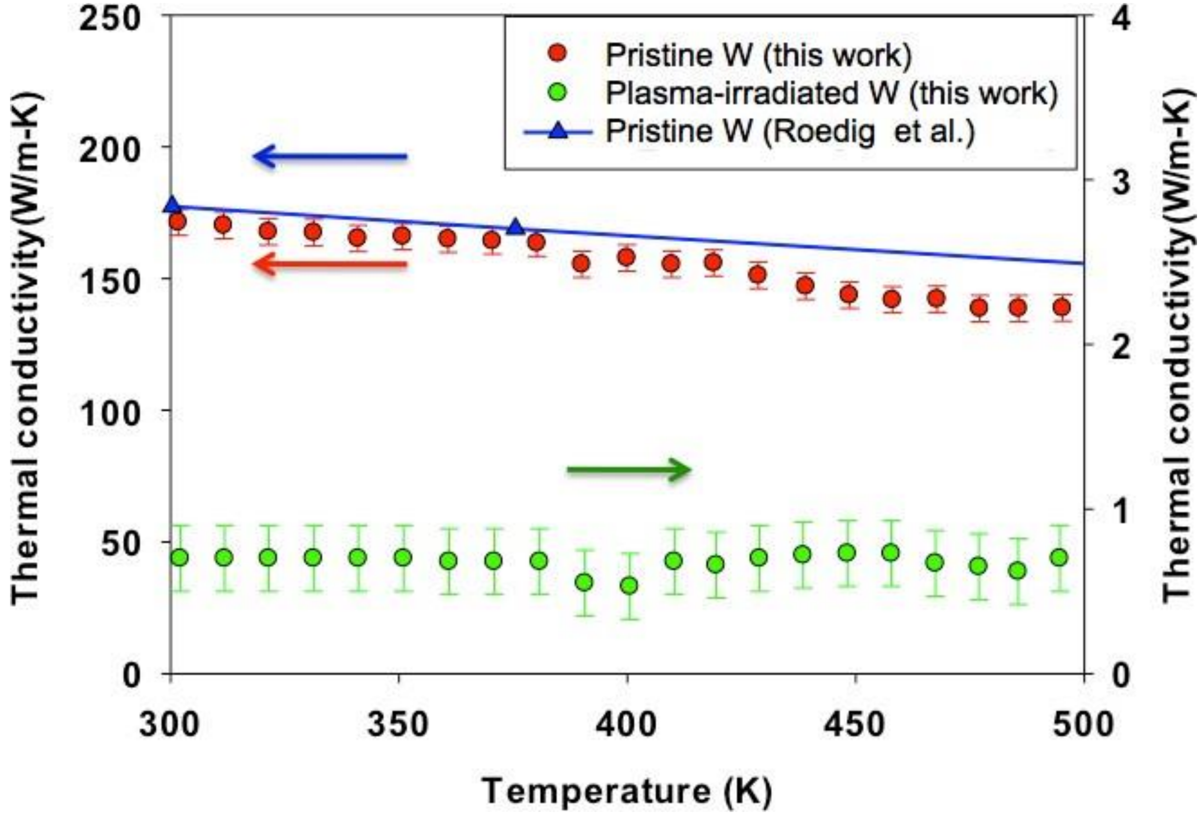
Neutrons significantly change the thermo-mechanical properties of materials

Affected global parameter	Microscopic change
Heat conductivity	Lattice defects
Swelling	Void formation, gas bubbles (e.g., n $\rightarrow$ Be)
Ductility (i.e., ability to stretch material into a wire)	Neutron and helium induced hardening and embrittlement
Composition	Transmutation products
Trap sites for tritium (retention)	Blister formation

- **Investigations into neutron damage of materials requires dedicated facilities (e.g., IFMIF for fusion neutrons, heavy ions) ⇒ need for up to 100 dpa**

Irradiation of tungsten with heavy ions and He reduces the thermal conductivity by factors of 200

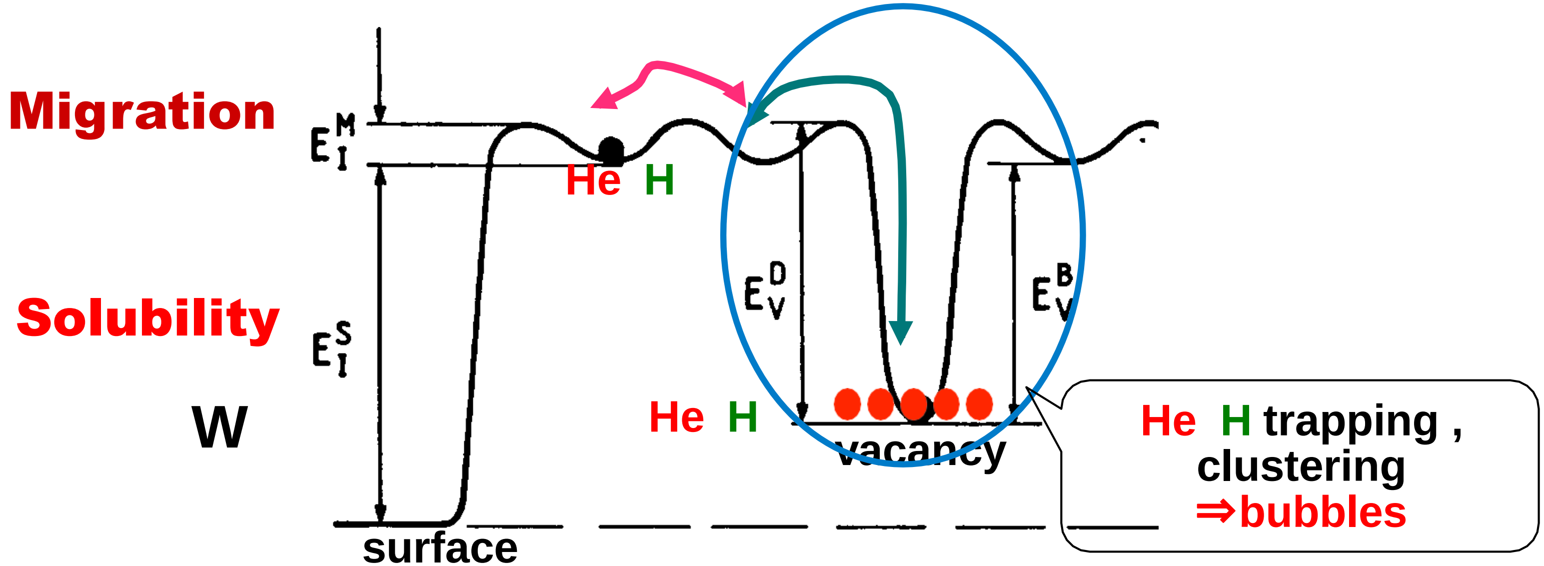
Doerner et al. PMIF-PFC 2013



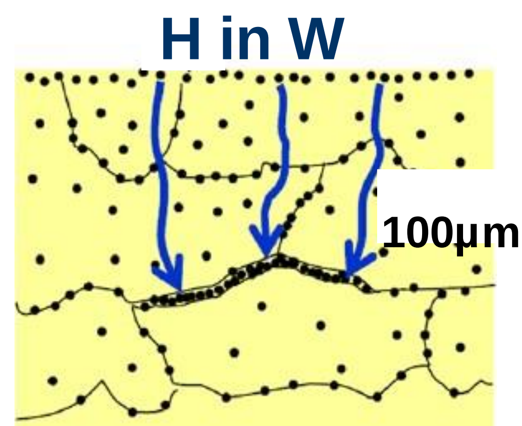
- **Irradiation by heavy ion beam source (up to 18 MeV)**
- **Simultaneous He implantation**

- **Other effects include increase of Ductile Brittle Transition Temperature (DBTT), void swelling, increase of tritium retention**

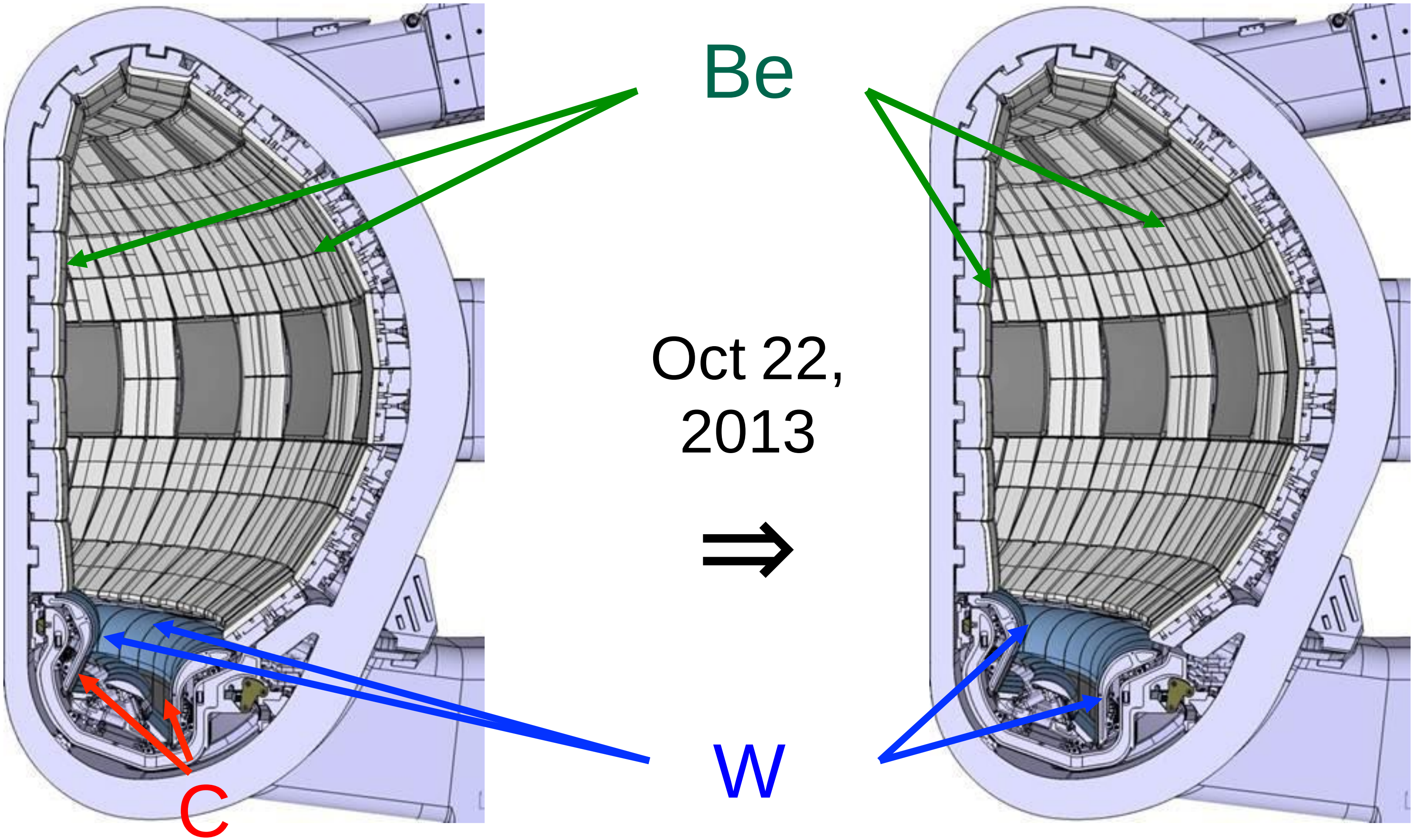
Both hydrogen and helium can be trapped deeply in tungsten leading to bubbles and blisters



- **Low solubility of H and He (E_I^S: 3.5 eV 5.5 eV)**
- **Fast interstitial migration into grain boundaries (E_I^M : 0.35 eV 0.24 eV)**
- **Deep trapping in vacancies (1.4 eV 4.7 eV)**



Because of tritium retention issue with carbon, and the good experience with the JET-ILW, ITER opted for a full-W divertor from day-one material



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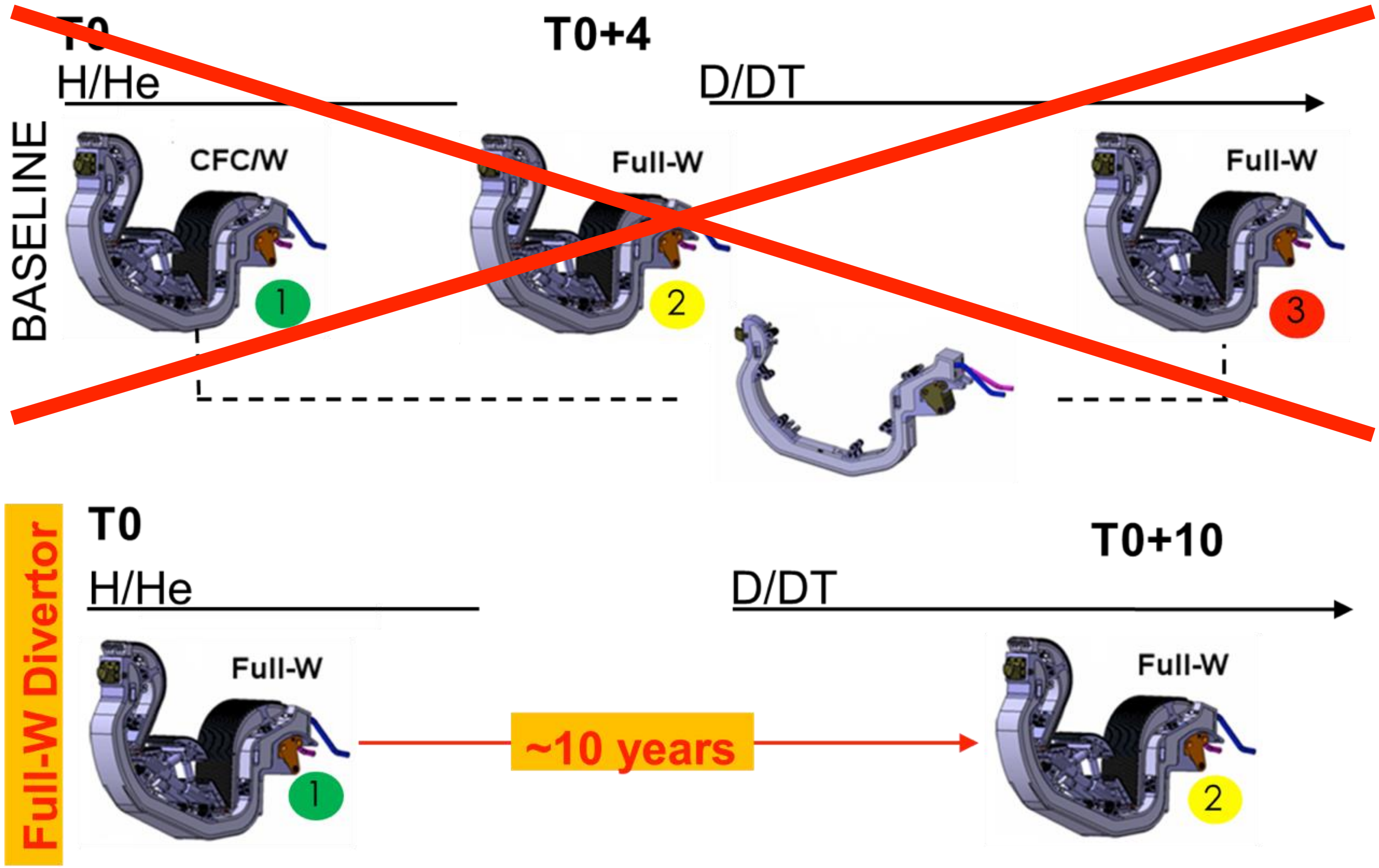


Presemo quiz #1

https://presemo.aalto.fi/fet/

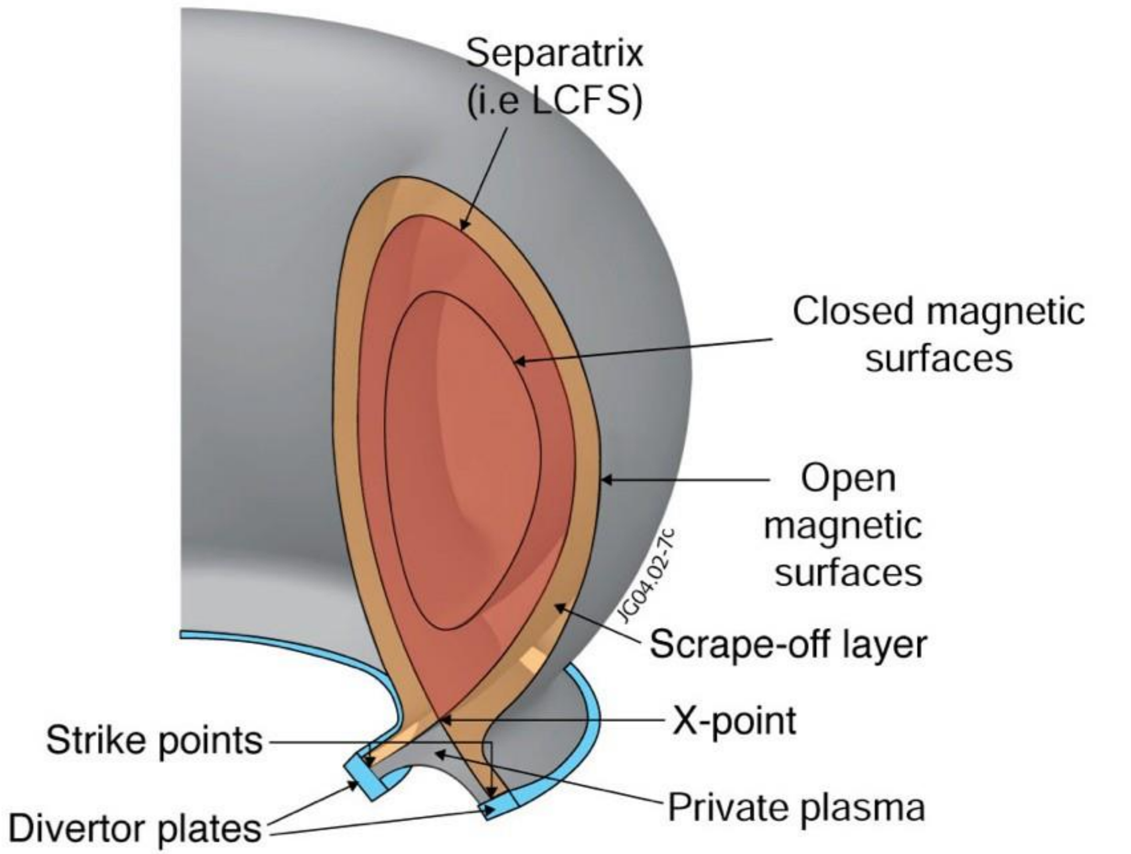
Physics models

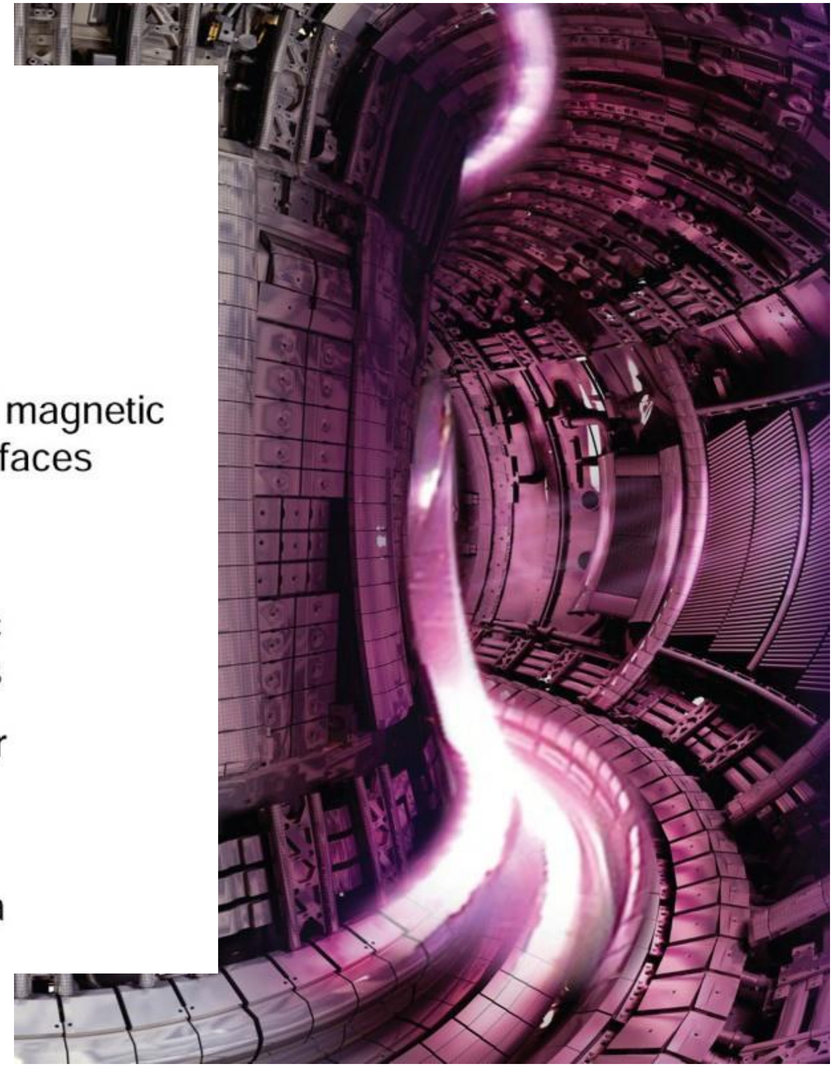
(to predict power exhaust and plasma-material interaction in future reactors)

Physics models are needed to extrapolate and mitigate plasma-material issues

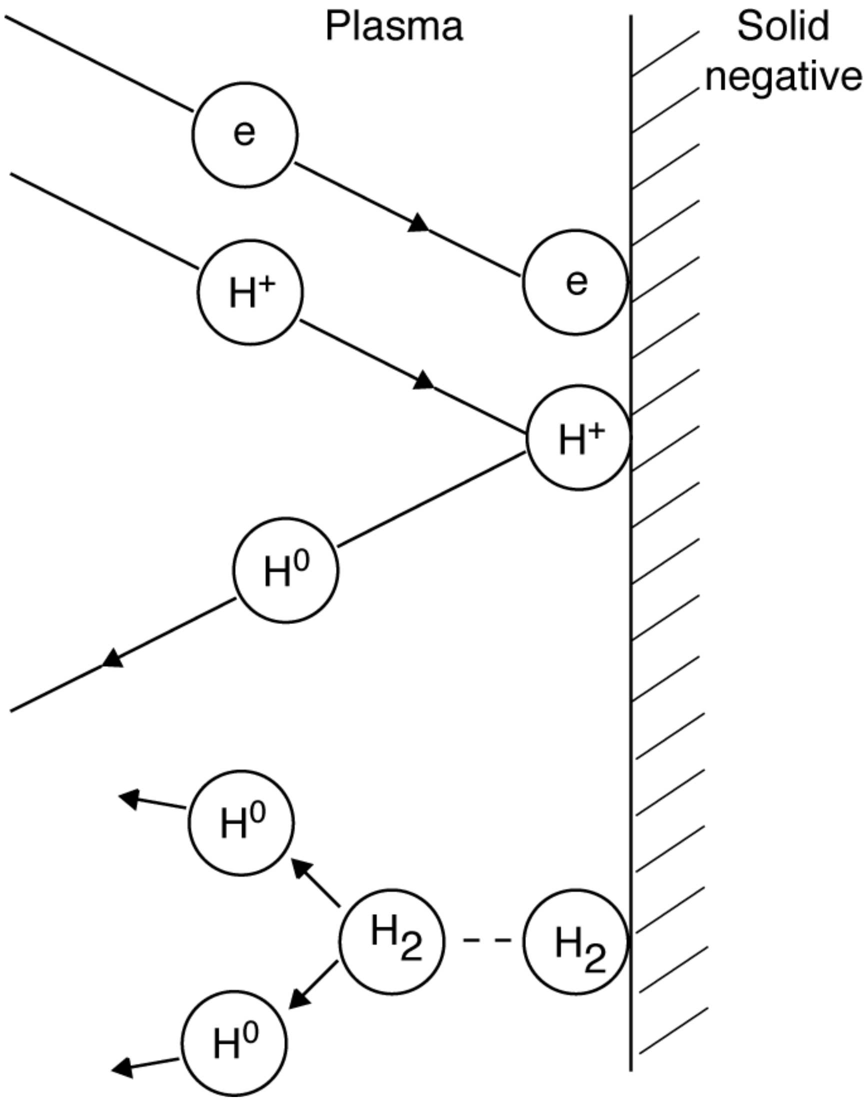
Parameter/s	Issue/s
Plasma radiation, power flux and total energy to surface	Power exhaust
Particle flux and fluence	Erosion and impurity influxes ⇒ plasma impurity content
Plasma temperature	Power exhaust, sputtering yield, total erosion
Plasma density (impurity seeding)	Detachment/power exhaust (fuel dilution, density limit)
Helium	Fuel dilution
Dust	Fuel dilution, explosion hazard

In diverted configurations, the separatrix divides the core and the SOL, and defines a private plasma region



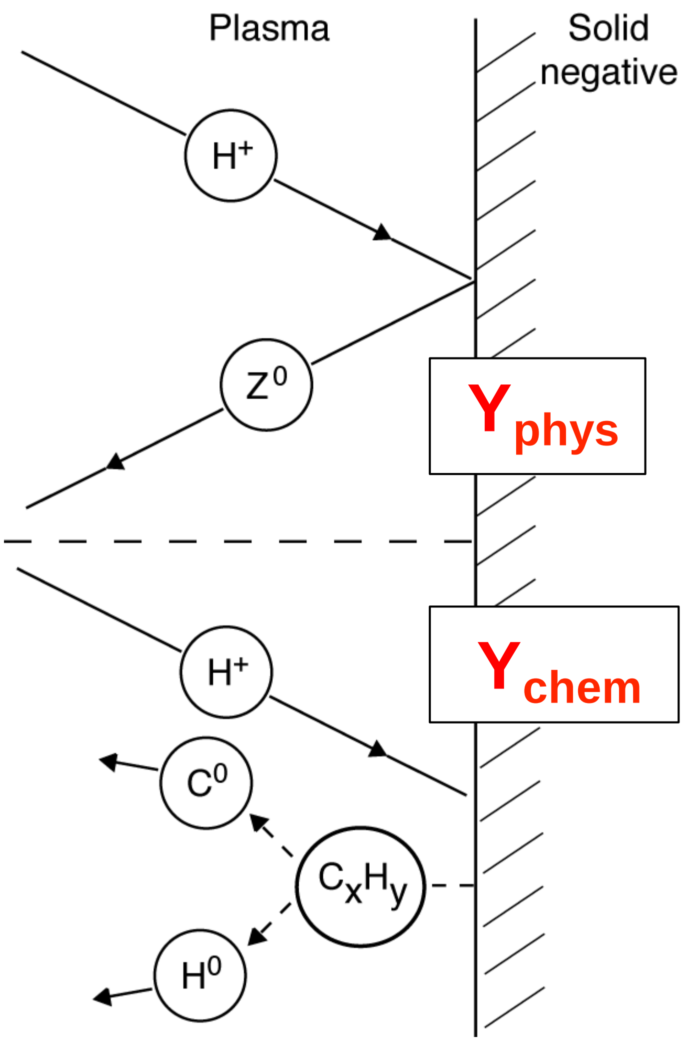


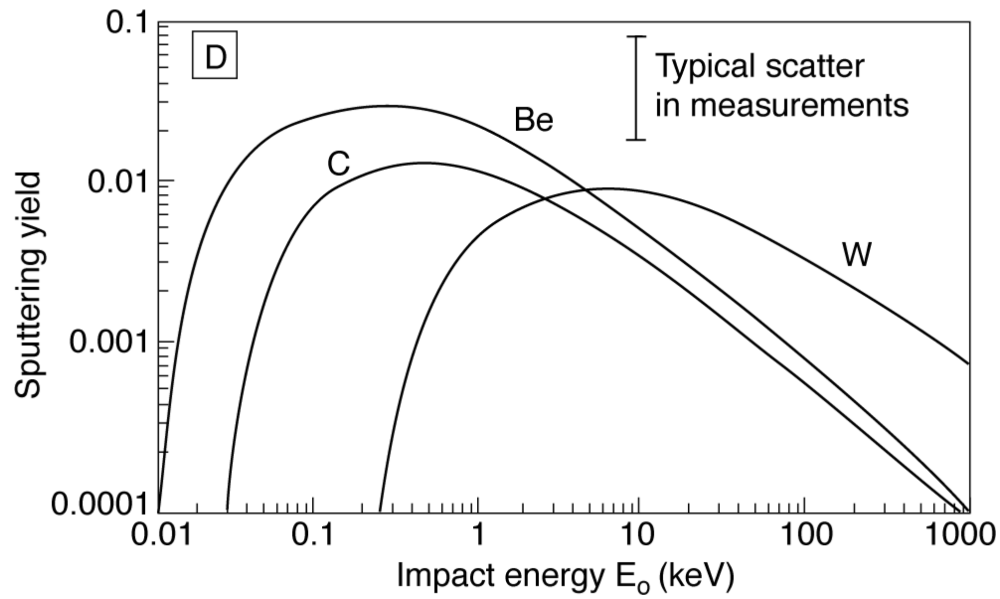
Plasma electrons and ions can stick to material surface, and recycle as neutral ('natural' fueling)



- **Impinging ions recombine at surface**
- **Particles can remain at surface, diffuse into material, or are released back into plasmas as atoms (backscattering) or molecules (thermal release)**
- **Walls acts both as particle sink and source**: strongest fueling process in tokamaks

Plasma-wall interaction leads to sputtering and macroscopic erosion of material



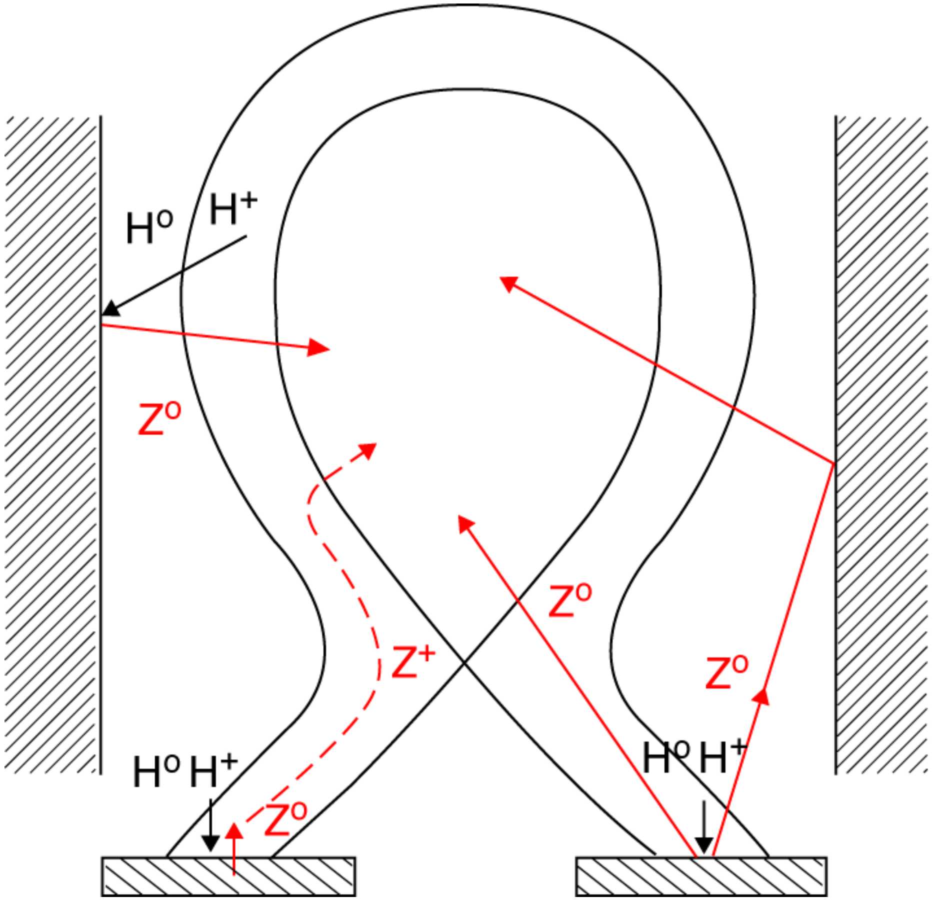


- **Phys. and chem. sputtering processes due to hydrogen ions and neutrals, and impurities (self-sputtering!)**
  ⇒ effective yields

Sputtering due to ion and neutral impact on the material surface leads to release of impurities

- **Physical sputtering:** momentum transfer of incoming particle to lattice
  - Threshold energy: $Y_{phys} \rightarrow 0$ for $E_0 \rightarrow E_{thresh}$
  - Peak yield correlates with maximum ion/neutral-substrate momentum transfer
  - Yields are strong function of material ⇒ future reactors favor high-Z materials
  - Self-sputtering of same-mass impurities can lead to $Y_{phys,eff} > 1$ ⇒ run-away process

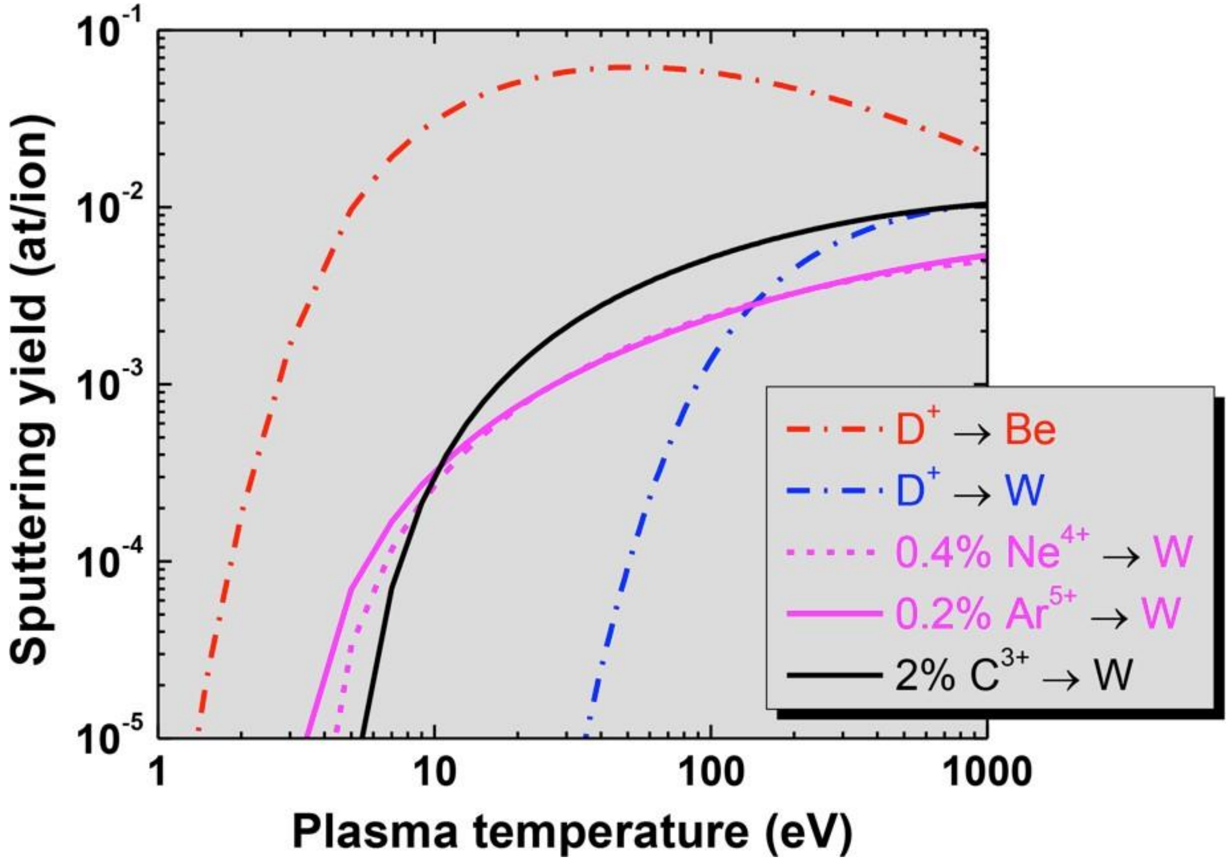
Impurities are generated at both the main chamber walls and divertor plates



- **Impurities can enter the main plasma as neutrals or ions**
- **Principal pathways include**
  - Source (distribution)
  - Edge transport
  - Core transport
- **Impurity migration**

“If we understand the impurity source distribution, we can mitigate the impurity issue almost entirely!” Quoting P.C. Stangeby

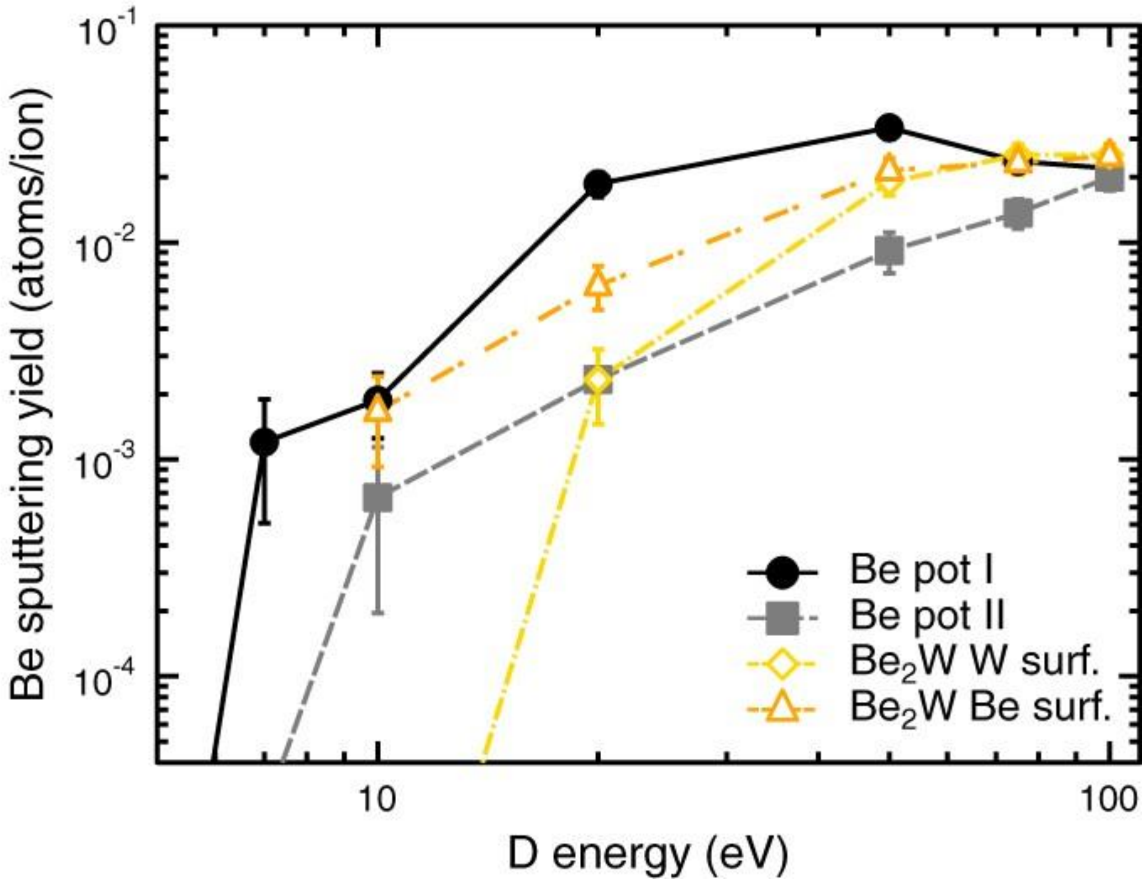
Trace amounts of impurities in the plasmas can significantly diminish benefits of high-Z materials



- **For neon and argon impinging on tungsten, already less than 0.5% is sufficient to drop E_{thresh} from 35 eV to 5 eV**

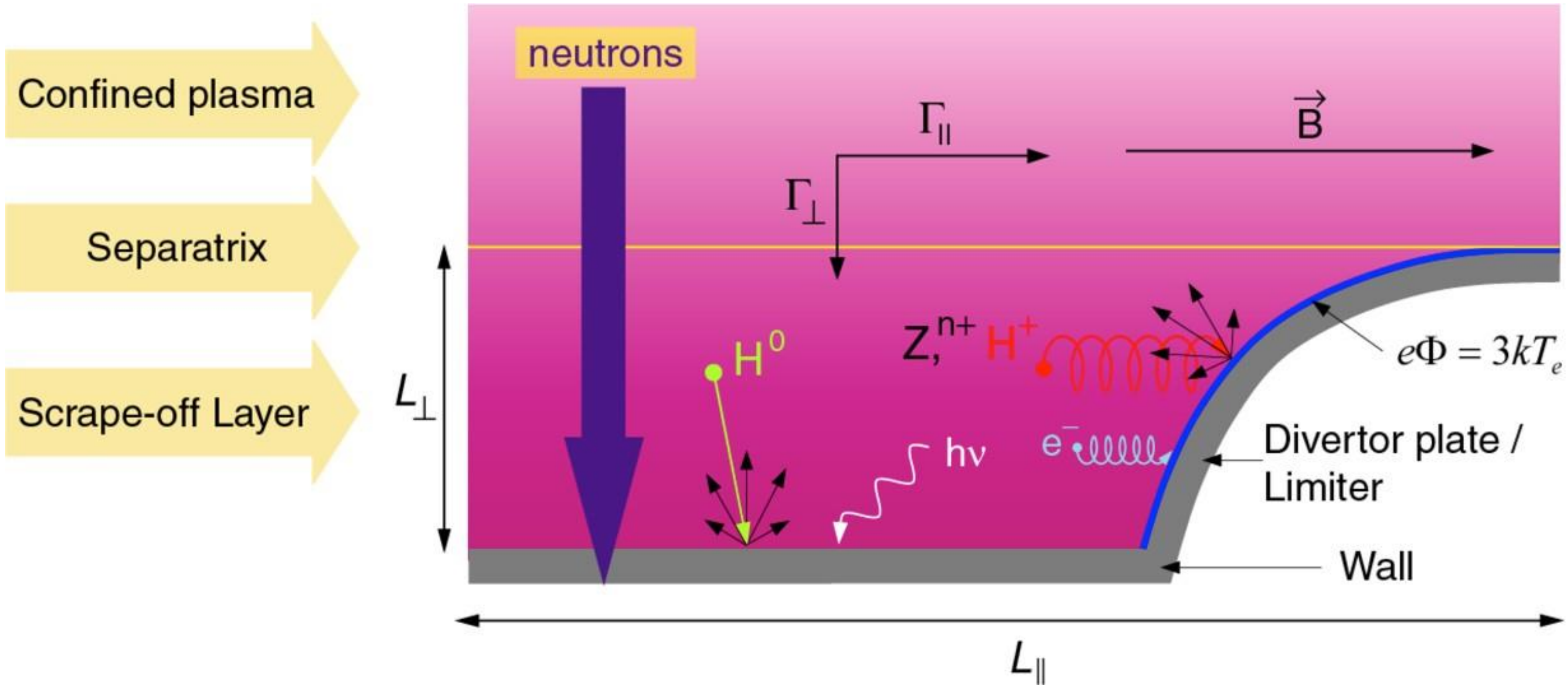
Chemical sputtering also occurs on metals (more common feature for carbon)

Figure 2 from C Björkas et al 2013 Plasma Phys. Control. Fusion 55 074004



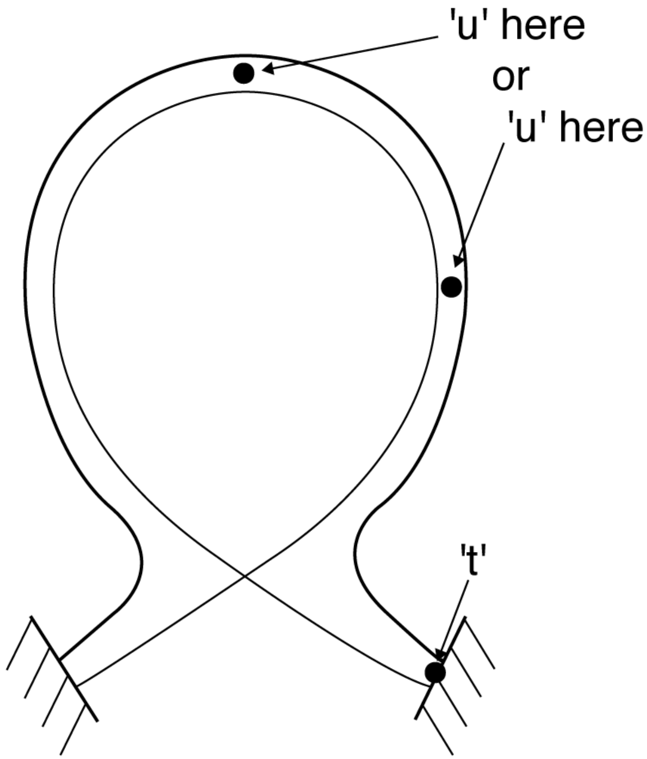
- **Yields are strong functions of substrate temperature, alloy composition, and magnitude of fluxes**

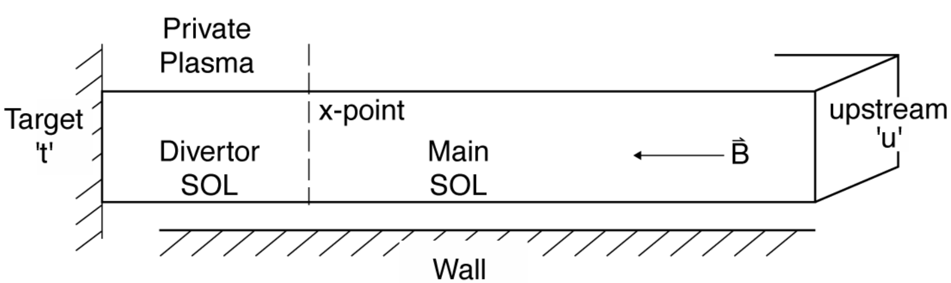
Plasma ions crossing the separatrix into SOL experience ‘attractive’ force of limiter and divertor plate



- **Upon plasma initiation, negative charged sheath forms in front of limiter/plate (while SOL remains neutral!)**
- **SOL width is determined by competition between parallel-B and perpendicular-B transport ⇒ order of cms**

The divertor target conditions are given by the upstream conditions for power and density





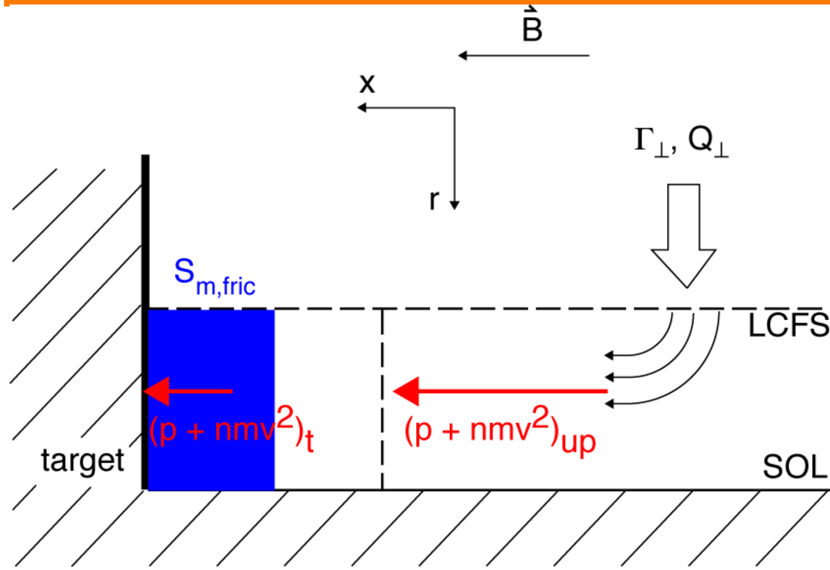
$$2n_tT_t = n_uT_u$$

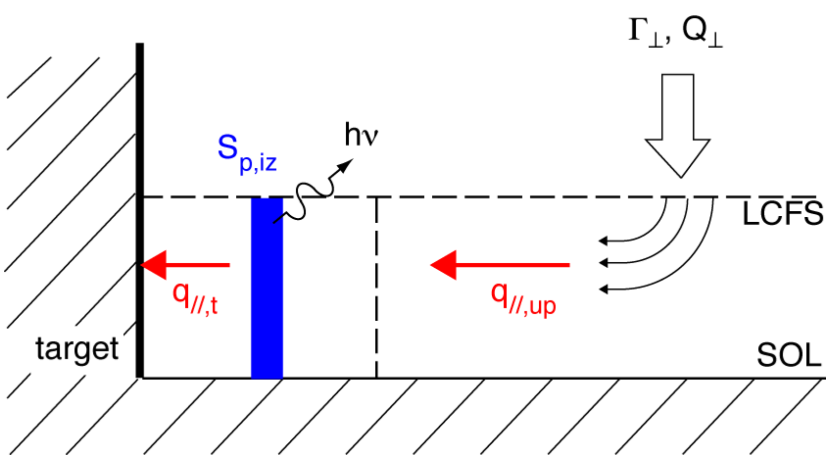
$$T_u^{7/2} = T_t^{7/2} + \frac{7}{2}q_{\parallel}\frac{L}{\kappa_{0e}}$$

$$q_{\parallel} = \gamma n_t k T_t c_{st}$$

- **Conservation of particles, momentum, and energy ⇒ SOL 2-point model (1-D)**
- **Eqs. can be manipulated to obtain n_t, T_t, and T_u for given $q_{||}$ and n_u**

In detached conditions, momentum and power losses occur in the SOL in front of target plate





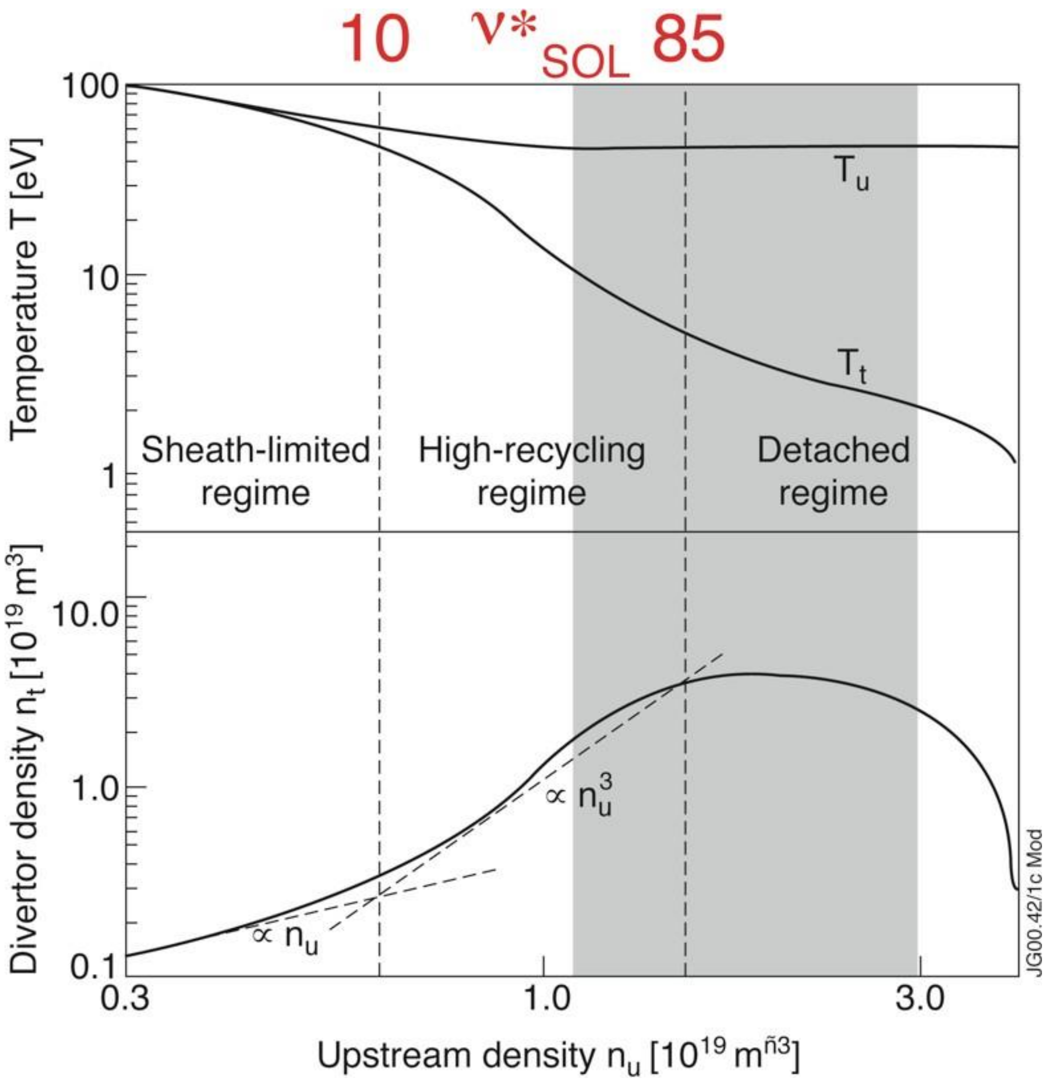
$$2n_tT_t = \boldsymbol{f_{mom}}n_uT_u$$

$$T_u^{7/2} = T_t^{7/2} + \frac{7}{2}\boldsymbol{f_{cond}}q_\parallel \frac{L}{\kappa_{0e}}$$

$$q_\parallel = \frac{\mathbf{1}}{\mathbf{1} - \boldsymbol{f_{power}}}\gamma n_t k T_t c_{st}$$

- **Momentum losses due to (CX) friction of plasma with neutrals (recycling and volumetric)**
- **Surface heat load is dispersed by line radiation (line radiation, recombination)**

The most attractive regime for fusion reactors is the detached regime at high upstream density



- **High upstream densities required for high core plasma density ('natural by-product')**
- **Plasma temperature in front of plate** **1 eV, or below ⇒ low sputtering**
- **Plasma ionization moves off plate**

Summary

- **A container (vessel) is required to provide the vacuum conditions for fusion**

- **Materials are exposed to extreme neutron and particle fluxes ⇒ currently, limited solution to materials issue**

- **Carbon and metals (beryllium, molybdenum, tungsten) have been tested in tokamaks and linear devices**

⇒ Deterioration of thermo-mechanical properties under neutron irradiation and tritium retention swayed ITER to opt for metals (Be and W) only

⇒ Plasma physics (e.g., achieving low plasma temperatures at material surfaces and mitigation of transient events) needs to solve materials issue

Presemo quiz #2

https://presemo.aalto.fi/fet/

Backup material