The Abdus Salam
International Centre for Theoretical Physics

United Nations
Educational, Scientific and
Cultural Organization

IAEA
International Atomic Energy Agency

2028-19

Joint ICTP/IAEA Workshop on Atomic and Molecular Data for Fusion

20 - 30 April 2009

WHAT/HOW - Can we tend the fire?

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Can we tend the fire?

Three lectures course on plasma surface interaction and edge physics

II.) HOW can we make the application work (build ITER) ?

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Thanks to: R. Pitts (ITER), P.C. Stangeby (U. Toronto)

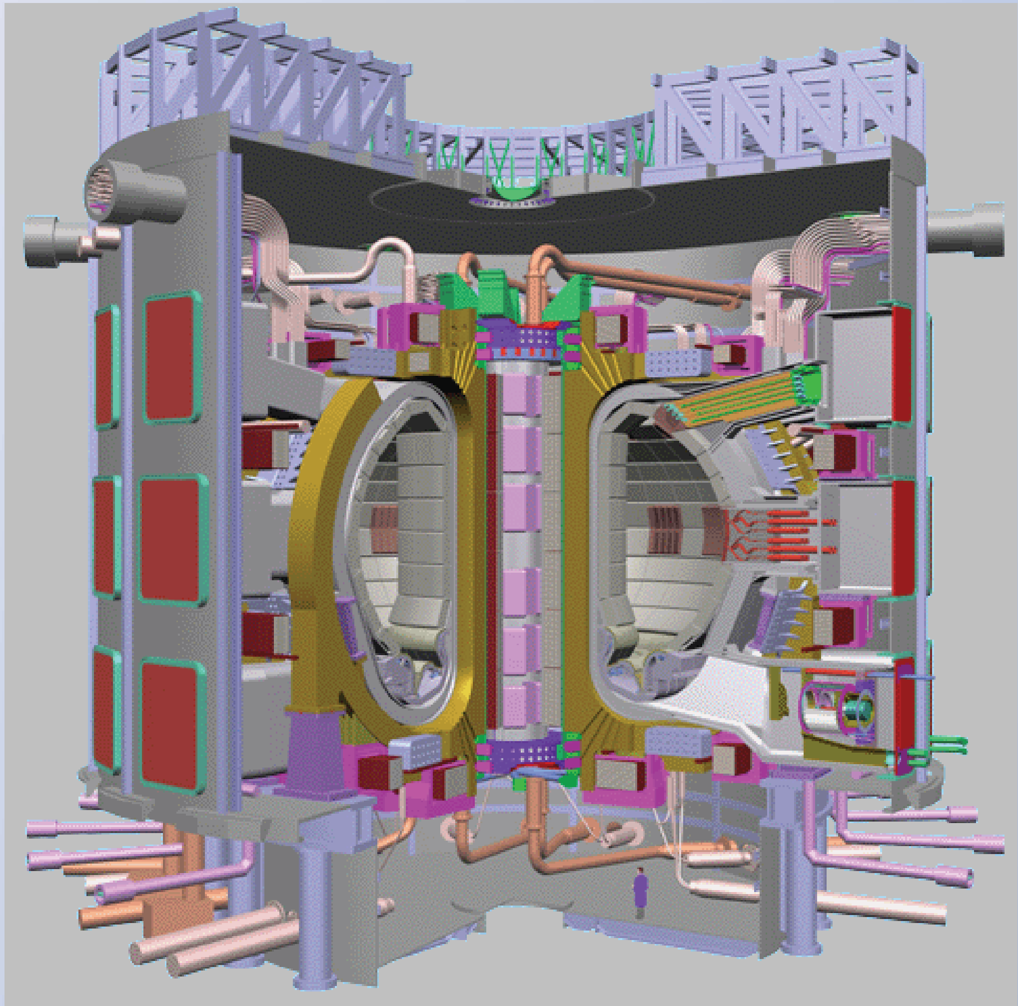
Motivation for Fusion Energy Research

electric power for 1 family, 1 year

Deuterium and Lithium
as an abundant
New Primary Energy Source

75 mg Deuterium
225 mg Lithium
equivalent of
1000 litres
of oil

The ITER Challenge



ITER: Furnace chamber:
Ø 15 m   6.8 m high   5.3 T   15 MA   500 MW   8 min

ITER „the way“

A Joint Project of EU, Japan, USA, Russia, South Korea, China, India

contributions „in kind“

European Legal Entity
(ELE)
Barcelona

Associations Industry

Agreement on site: **Cadarache**

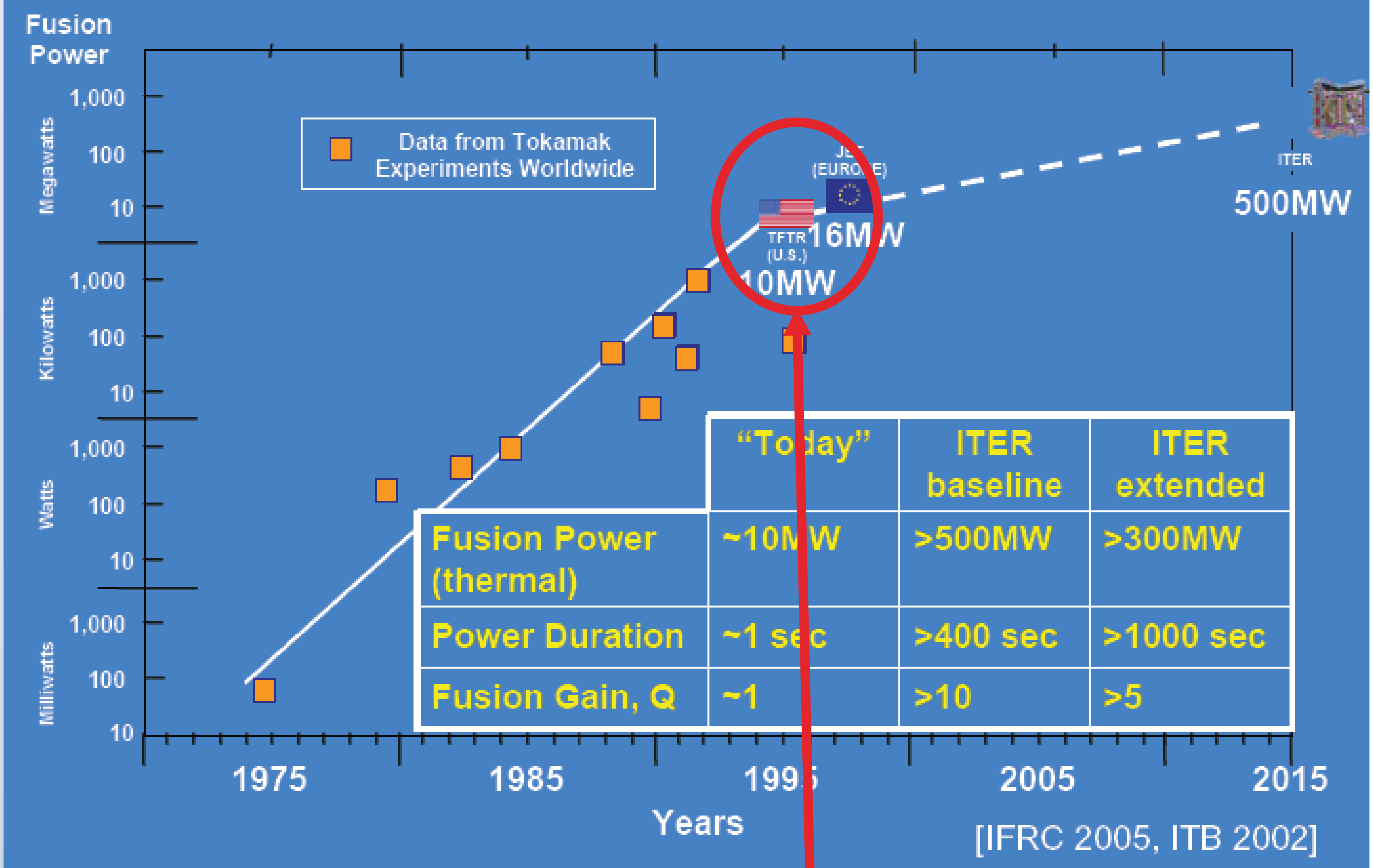
Investment ~ 5 Billion €

24. Mai 2006 Signature of ITER Implementing Agreement

Design Review: 2007

Construction: 2008 – 2016

ITER's Fusion Performance in Context

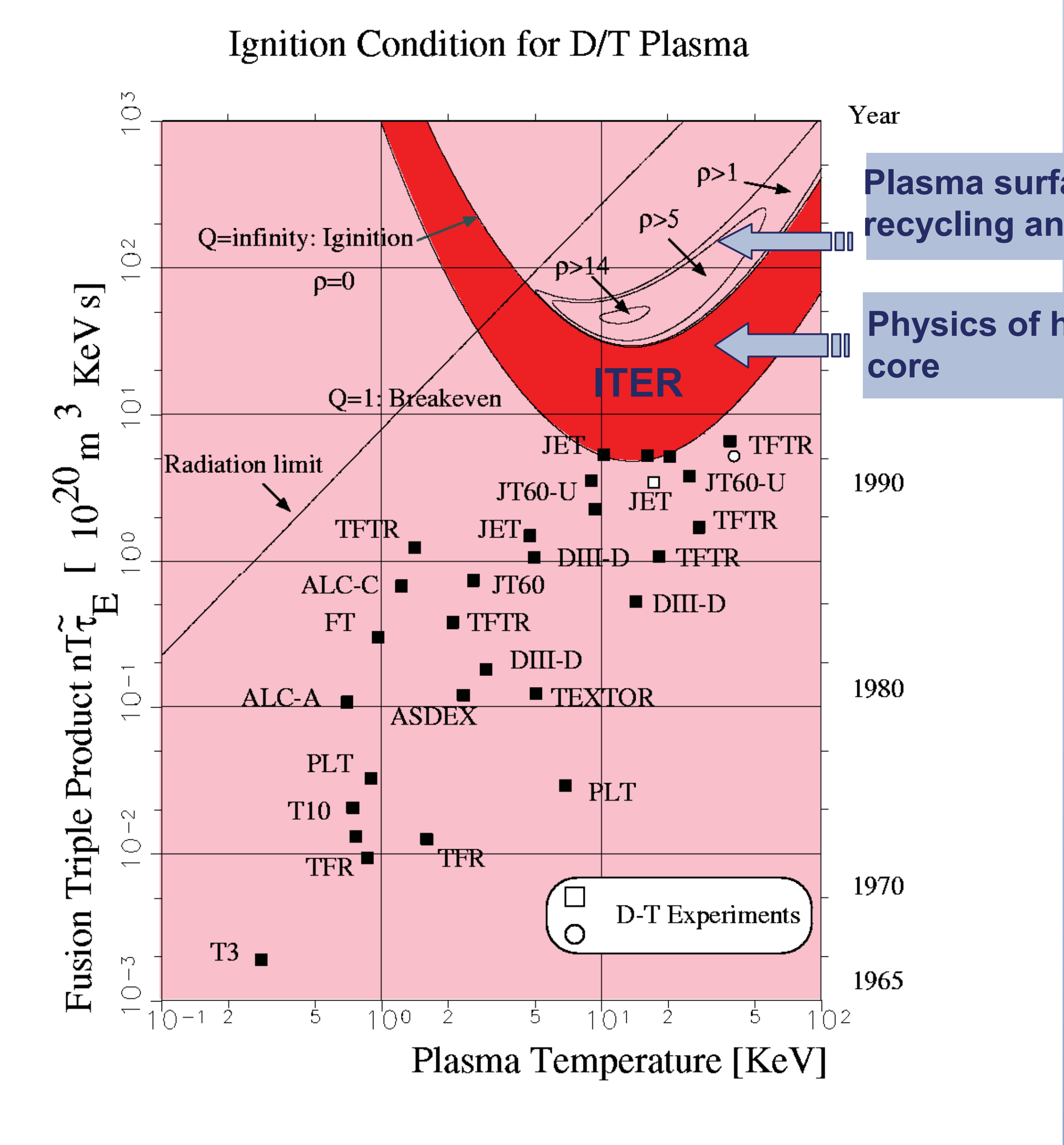


	"Today"	ITER baseline	ITER extended
Fusion Power (thermal)	~10MW	>500MW	>300MW
Power Duration	~1 sec	>400 sec	>1000 sec
Fusion Gain, Q	~1	>10	>5

D. Stotler, PPPL, ICAMDATA 2006

Proof of Principle of Fusion Energy

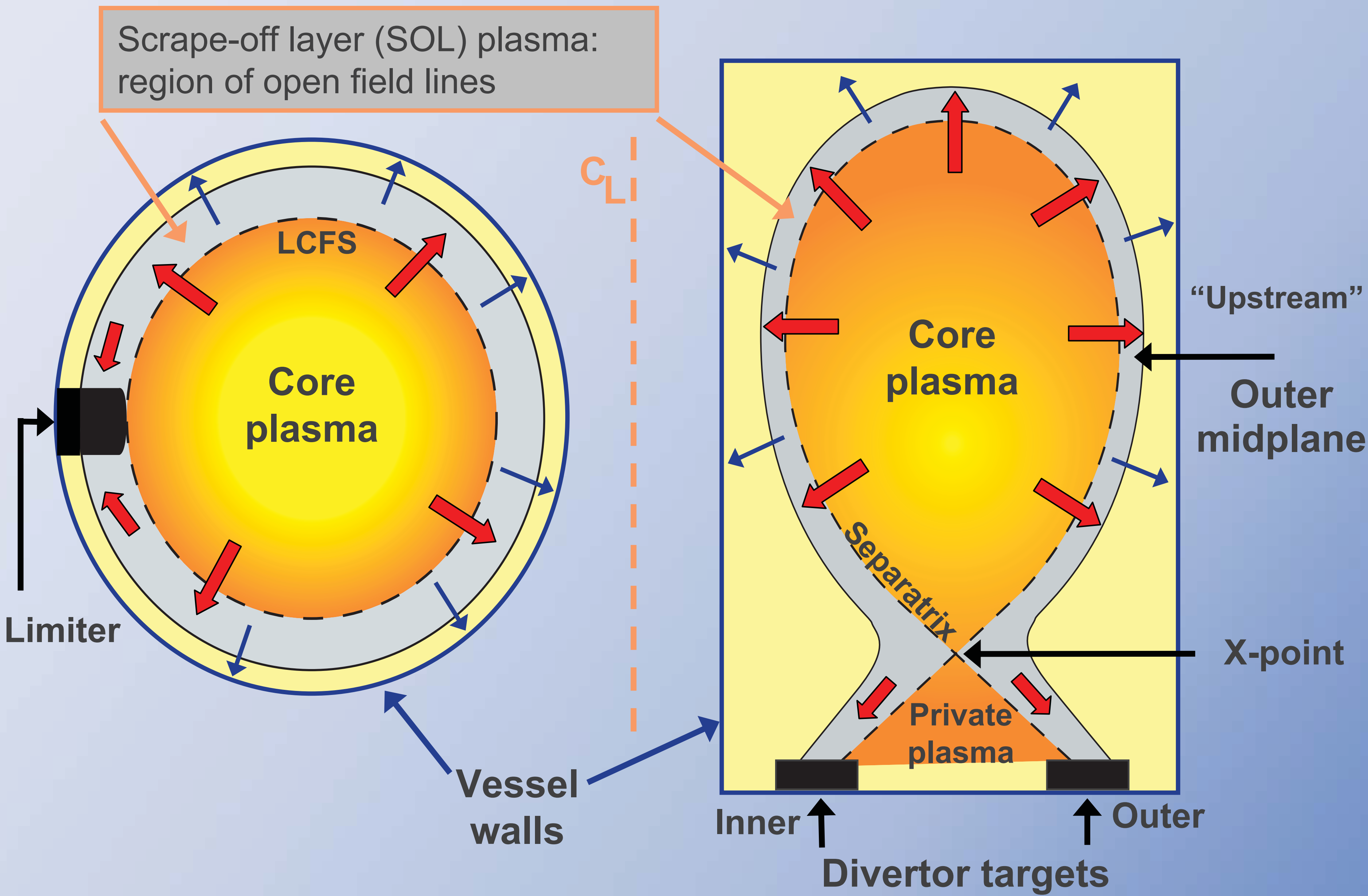
ρ:
heat conduction/part. convection



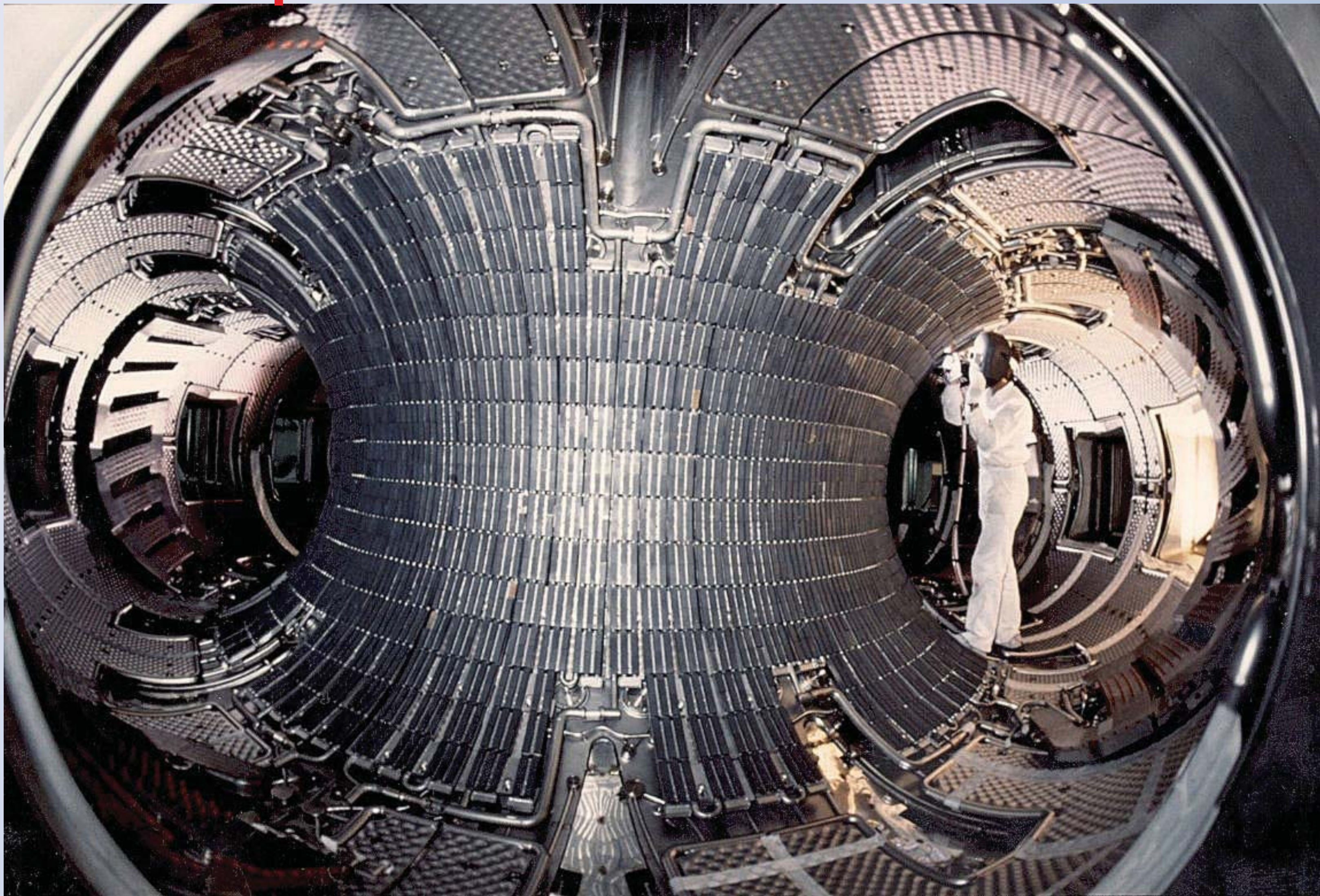
Plasma surface interaction, recycling and edge physics

Physics of hot plasma core

Terminology: limiters and divertors



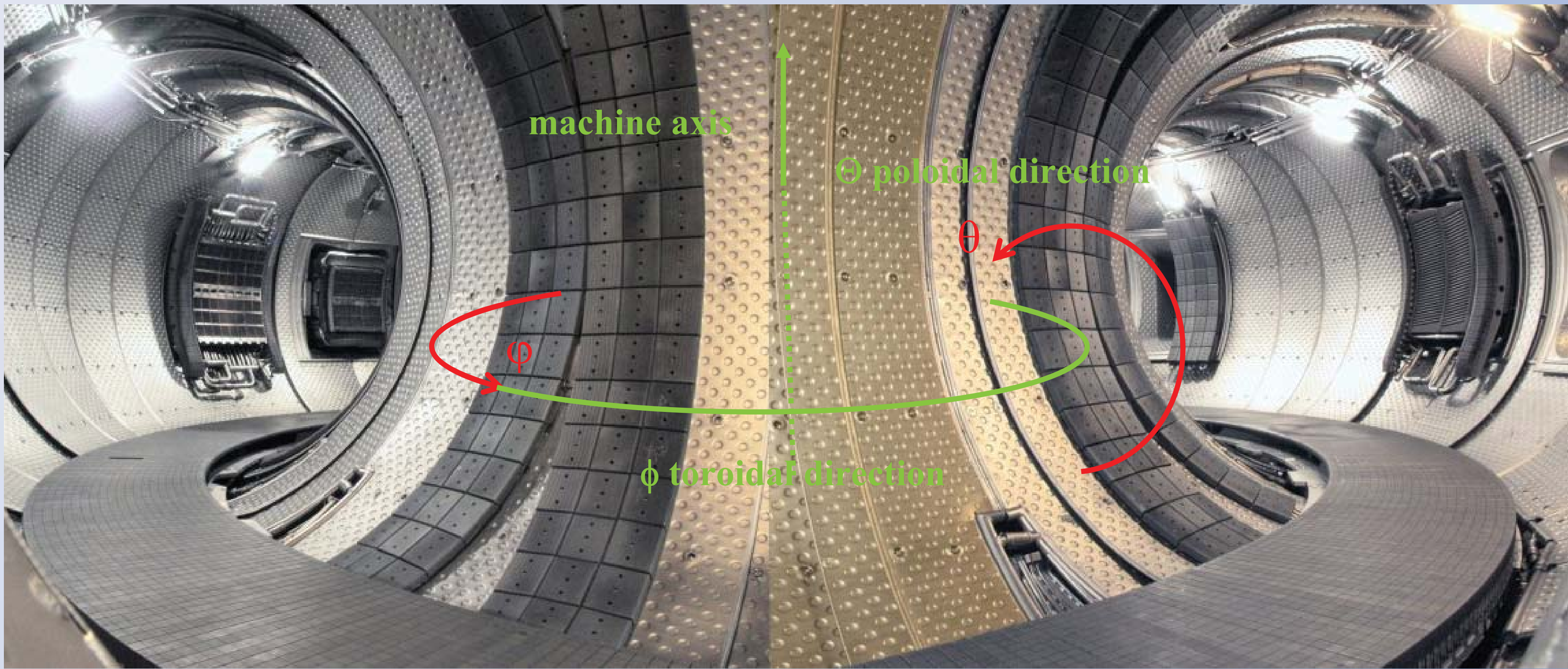
Tore Supra, tor. limiter “CIEL”



Where is the limiter ????

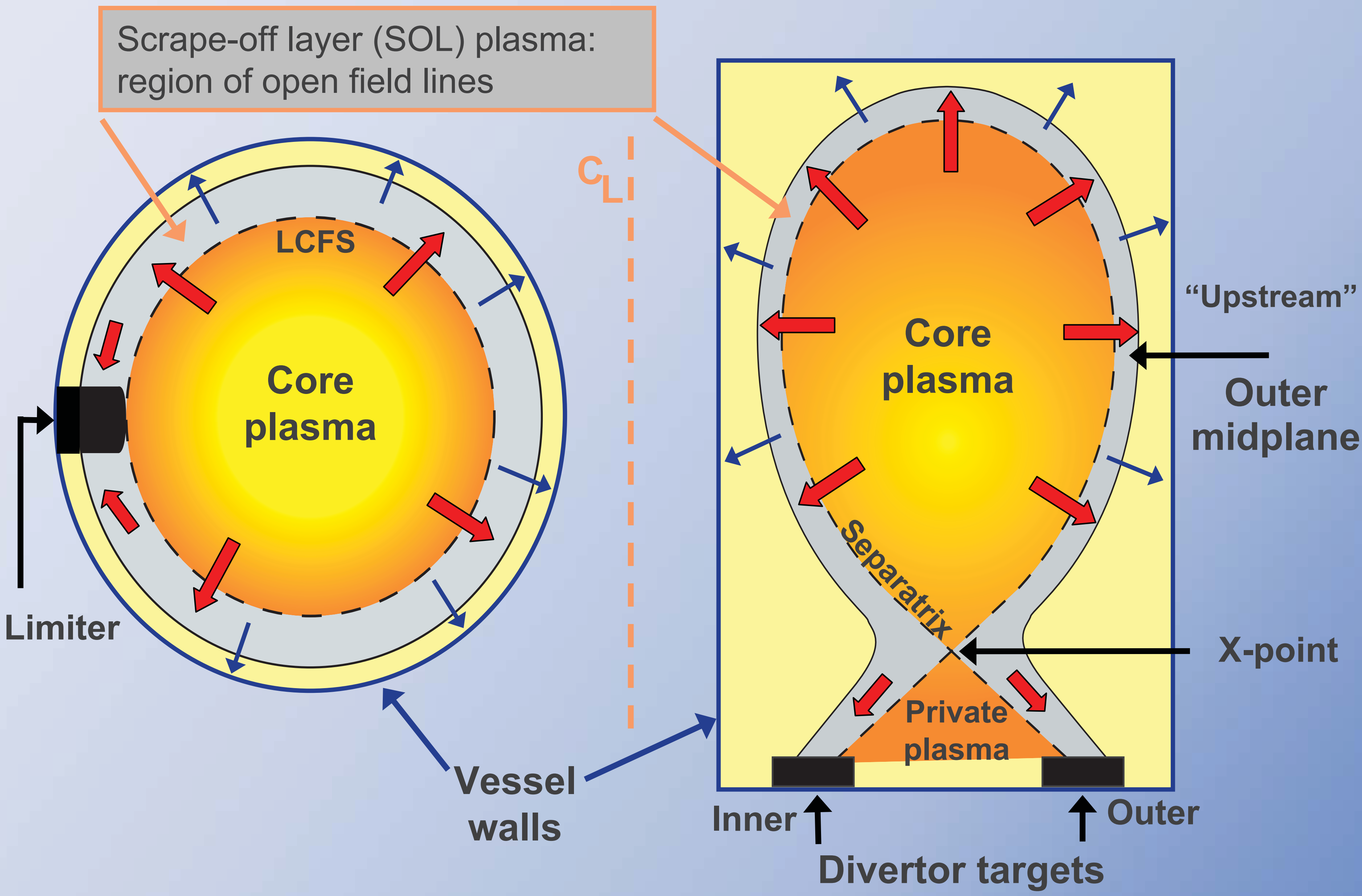
Tore Supra

Interior view of Tore Supra

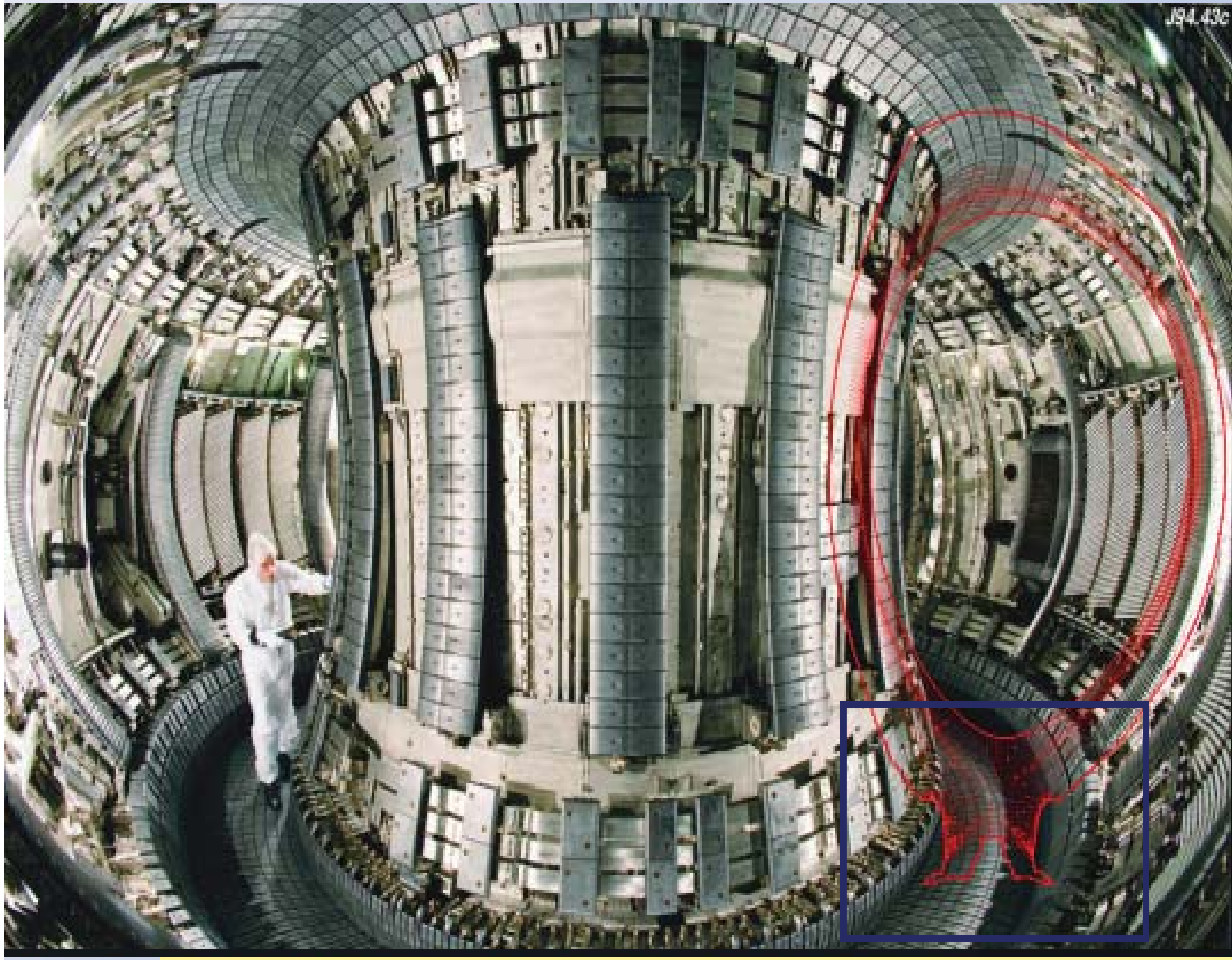


Full toroidal limiter CIEL

Terminology: limiters and divertors



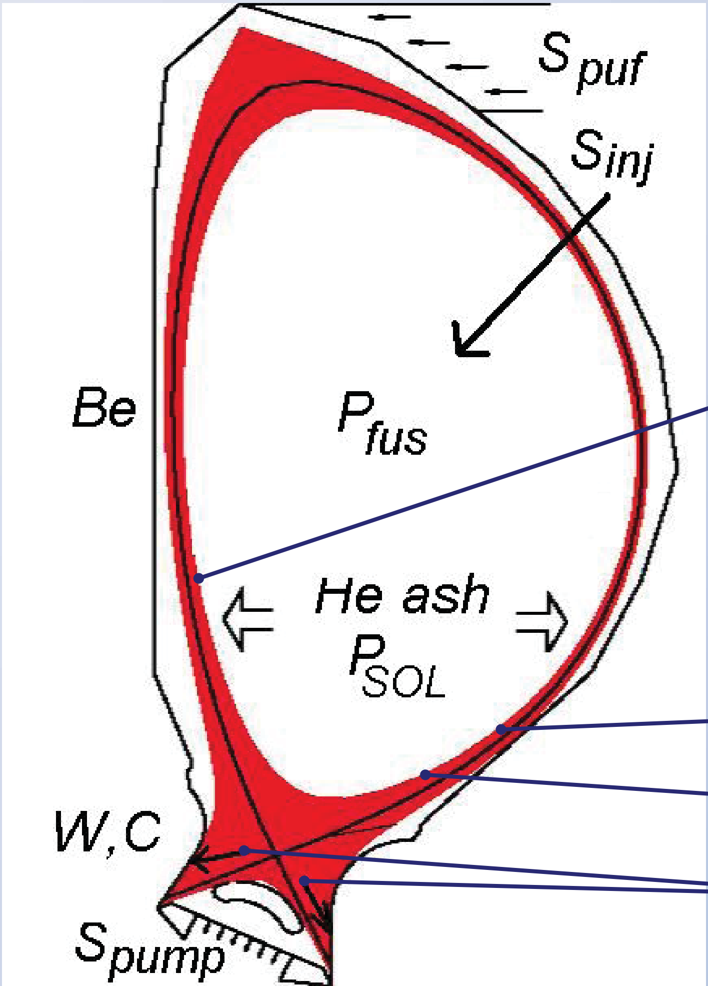
JET: Poloidal Divertor



JET Furnace chamber:

Ø 8.5 m 2.5 m high 3.4 T 7 MA 1 min

Provide sufficient convection without accumulating tritium and with sufficiently long divertor lifetime (availability).



$P_{fus} \approx$ 540-600 MW

$\Rightarrow$ He flux

$\Rightarrow P_{SOL} \approx$ 86-120 MW

$n_s \approx (2\text{-}4) \cdot 10^{19}$ m^{-3}

$S_{inj} \leq 10 \cdot 10^{22}$ s^{-1}

$S_{pump} \leq 200$ Pa·m^{-3}/s

!

$Z_{eff} \leq 1.6$

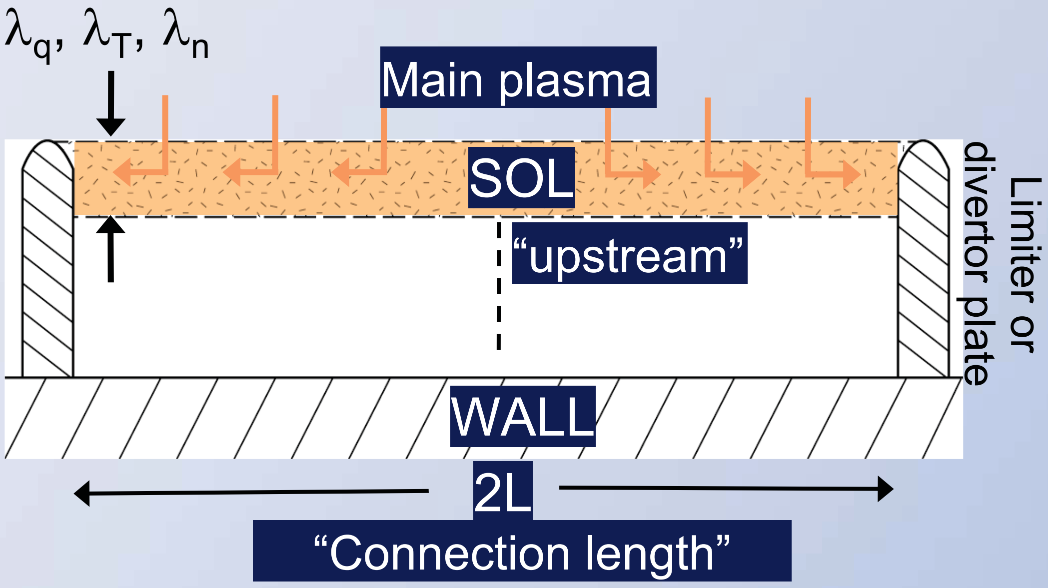
$C_{He} \leq 6\%$

$q_{pk} \leq 10$ MW/m^2

?

Engineering parameter : $S_{puff} \sim (1 \ldots 13) \cdot 10^{22}$ s^{-1}

Basics – SOL width, λ_n



Any solid surface inserted into a plasma constitutes a very strong particle sink

In the high tokamak B-field:
$\Gamma_\perp \ll \Gamma_{||}$

Thin Debye sheath (λ_D few 10's μm thick) forms at the surface → controls flow of particles and energy || B

Quick and dirty estimate of λ_n with diffusive approx. for cross-field particle transport (all ionisation inside LCFS):
$\Gamma_\perp \equiv n v_\perp = -D_\perp dn/dr \sim D_\perp n/\lambda_n$

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

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

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

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

The problem with λ_q

SOL width for power, λ_q, is also small and is an important parameter of the edge plasma

As for particles, λ_q is determined by the ratio of $\perp$ to || transport (e.g. cross-field ion conduction and parallel electron conduction: i.e. $\propto (\chi_\perp/\chi_{||})^{1/2}$), where $\chi_\perp$ is anomalous

Scalings for λ_q can be derived from models and experiments, e.g.:

- "2-point" analytic modelling: $\lambda_q \propto P_{SOL}^{-5/9}$ P_{SOL} = power into SOL
- Scaling from H-mode experiments on JET: $\lambda_q \propto P_{SOL}^{-0.5} B_\varphi^{-0.9} q_{95}^{0.4} n_u^{0.15}$
- ITER modelling assumes λ_q = 5 mm, JET scaling gives λ_q = 3.7 mm (cf. a=2.0 m)
- Very recent multi-machine scaling gives λ_q/R ~ constant

Note also that the parallel power flux, $q_{||} \propto P_{SOL}/\lambda_q$ ~ as much as 1 GWm^{-2} in ITER

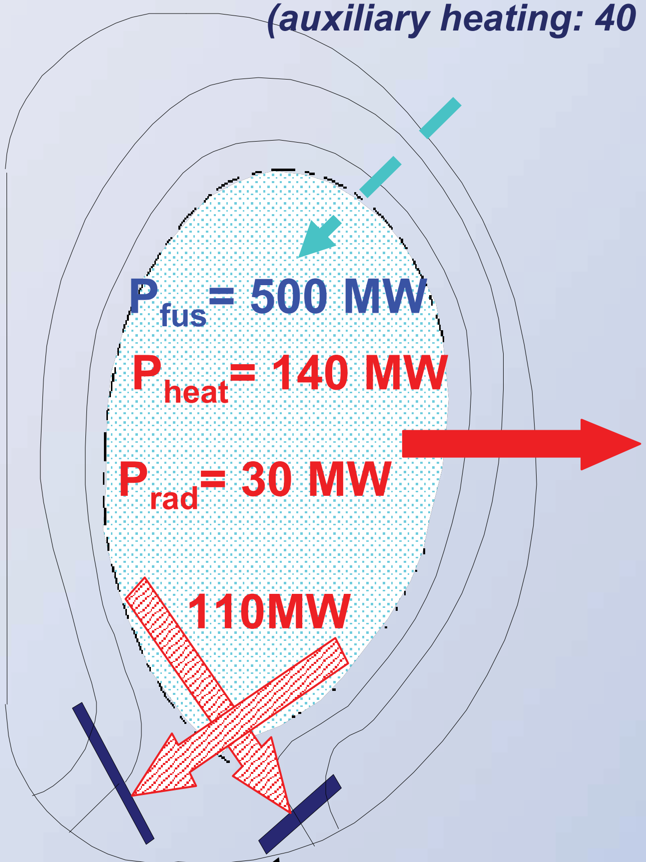
Stored energy scales strongly with tokamak major radius, $W \sim \propto R^4$
But power deposition area in the divertor $\propto R\lambda_q$ only (~3.0 m^2 in ITER)

Bottom line is that despite its increased physical size, ITER will concentrate more power into a narrower channel at the plasma edge than today's devices.

Courtesy: R. Pitts

The power exhaust problem in fusion (ITER as example)

Injected power
(auxiliary heating: 40 MW)



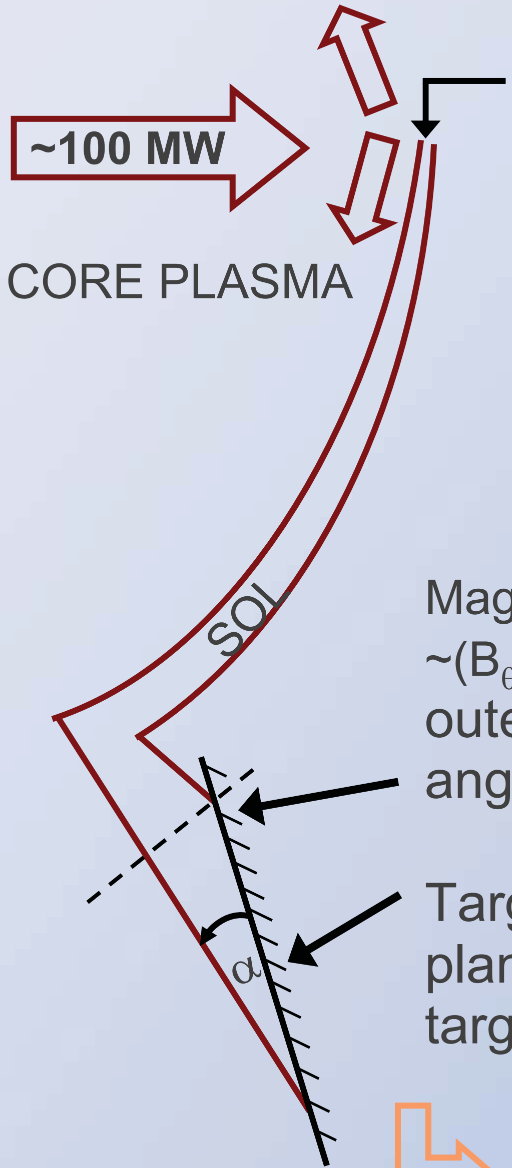
Divertor plates

Fusion power	500 MW
α-heating + auxiliary heating	140MW
Loss: Bremstrahlung+ Synchroton Radiation	30MW
Power load without additional radiation:	110MW
Wetted area: 2*U*width of strike zone (2 *40 * 0.05)	4.0 m²
Power load MW/m²	~ 25

Well above technical limit (10 MW/m²)

The problem results from the very small power SOL width (~ 0.5 cm)

Power handling – ITER case (approx)



$Area \sim 2\pi R\lambda_q \sim 0.2m^2$ λ_q = 5 mm

$q_{||,u}$ ~ 500 MWm^{-2}

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

+

Target tilting in poloidal plane (α ~ 25^o for ITER outer target) $Area \sim 2\pi R\lambda_q (B_\theta/B)_u/\sin(\alpha)(B_\theta/B)_t \sim 2.0m^2$ per target

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

q_T ~ 25 MWm^{-2} per target if no radiative (or other) dissipation

- **Magnetic confinement is now effective enough to contain the main fusion flame, but it is too good for the plasma edge (SOL): very narrow heat-footprints on targets.**

- **Magnetic Confinement Fusion Reactors must operate at reduced target fluxes and temperatures ("detached regime").**

- **n, T upstream (core) fixed by burn criteria, density limit, etc.**

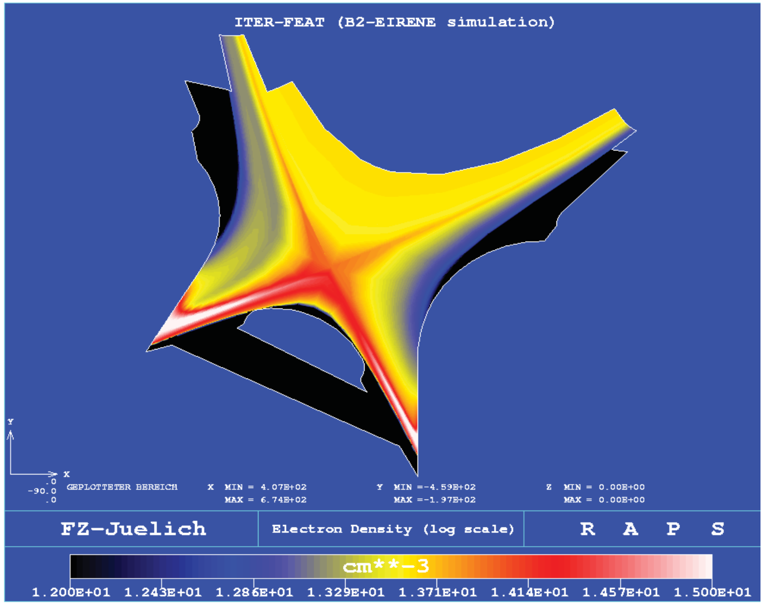
- **For ITER: Detached regime: decrease particle flux to target for given upstream conditions: self sustained neutral cushion (reactive plasma) controlled by PWI and A&M processes.**

- **Divertor detachment physics involves a rich complexity of plasma chemistry not otherwise encountered in fusion devices .**

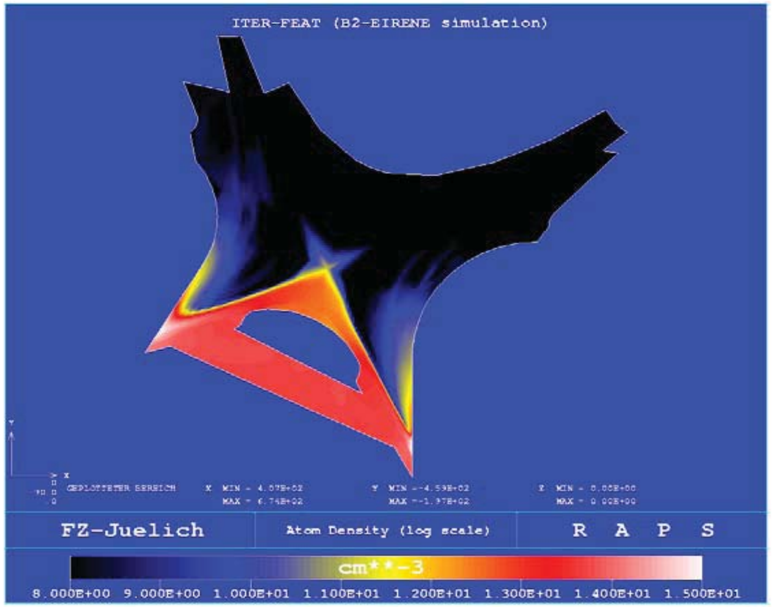
→ RECYCLING

1) Can a reactive plasma protect the chamber from a thermonuclear plasma?
2) Can, simultaneously, sufficient particle throughput be maintained?

Electrons, Ions



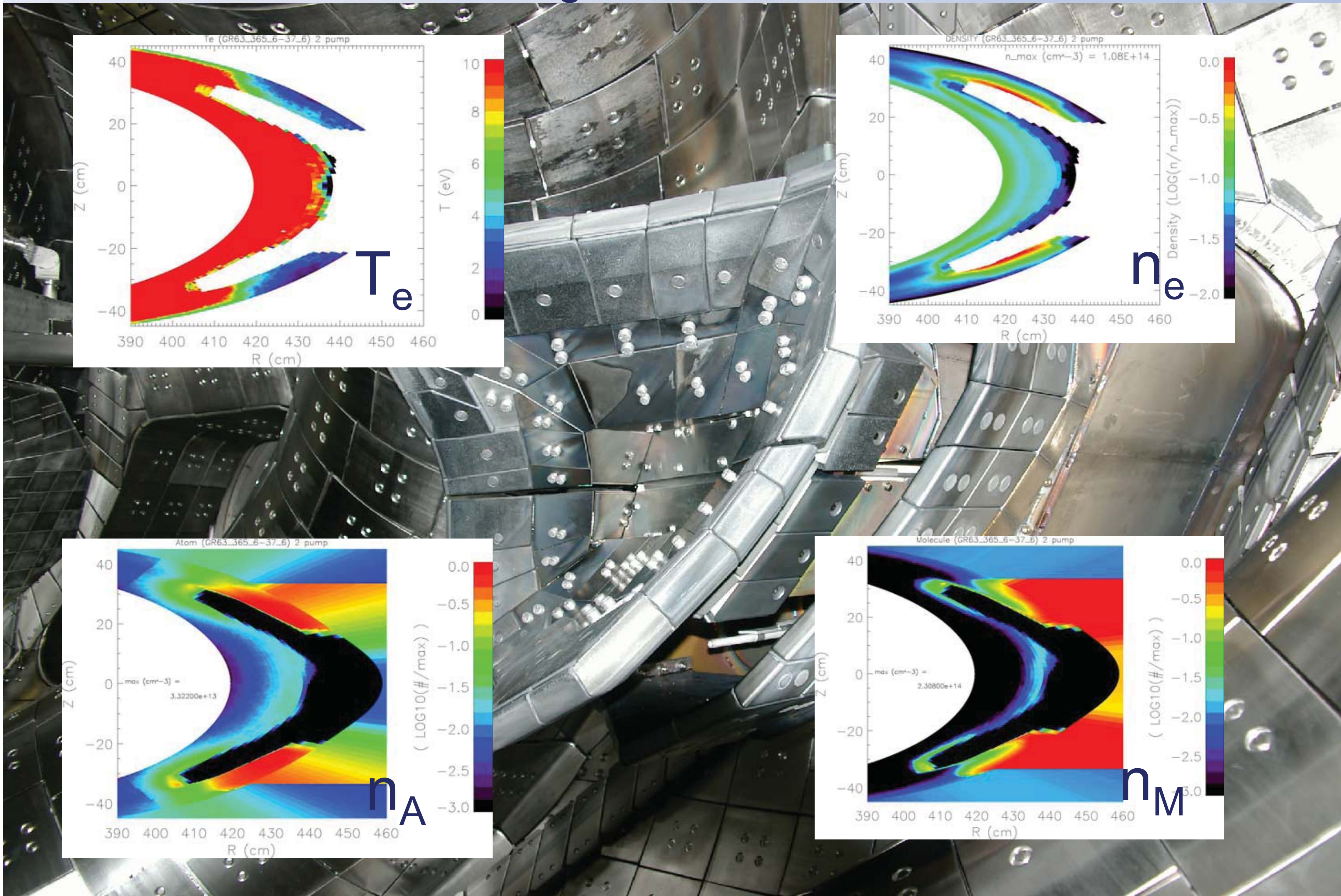
Neutrals



ASIDE

Stellarators will have same problem!
see: Large Helical Device (LHD), Toki, Japan

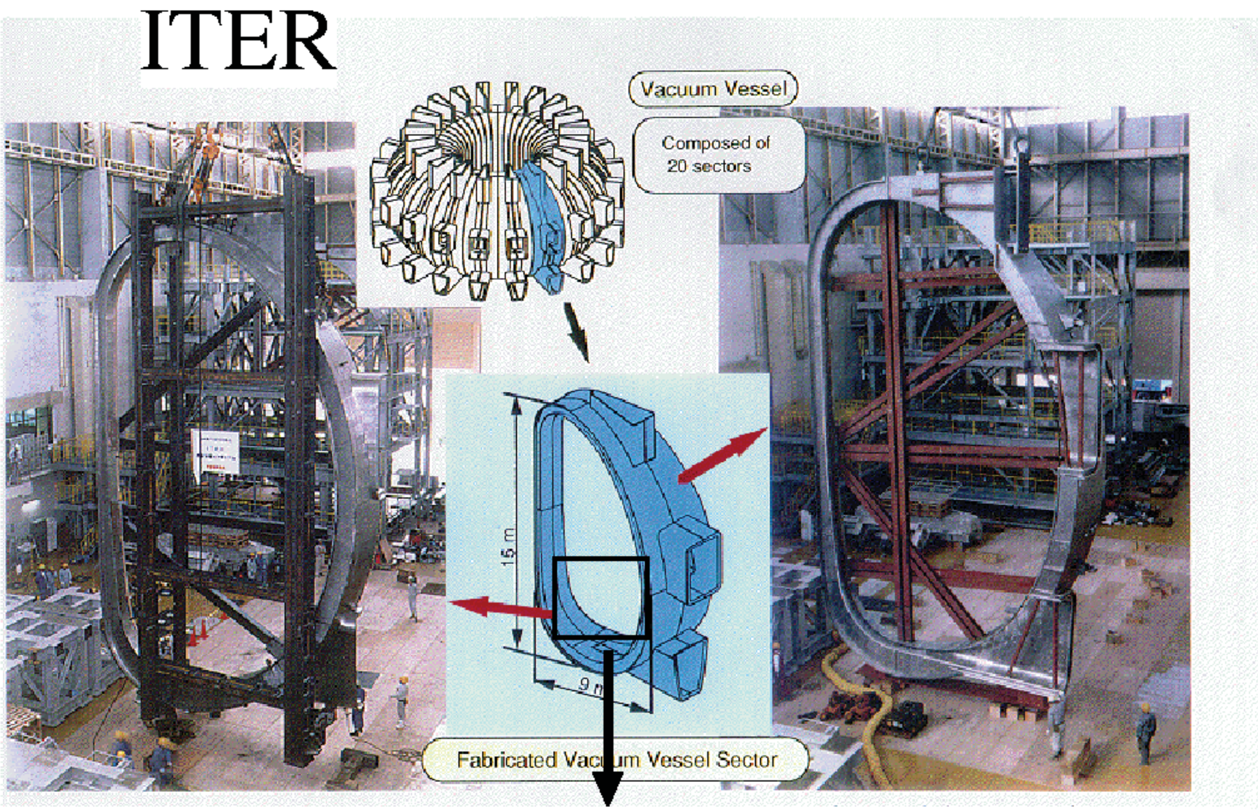
3D LHD Plasma Edge Simulation: EMC3-EIRENE



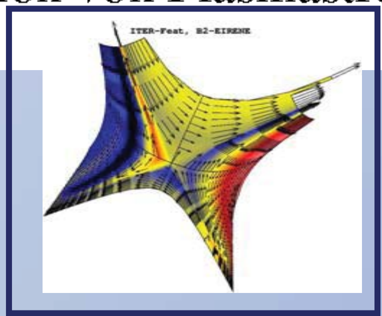
But for rest of this lecture:

Focus on the ITER challenge

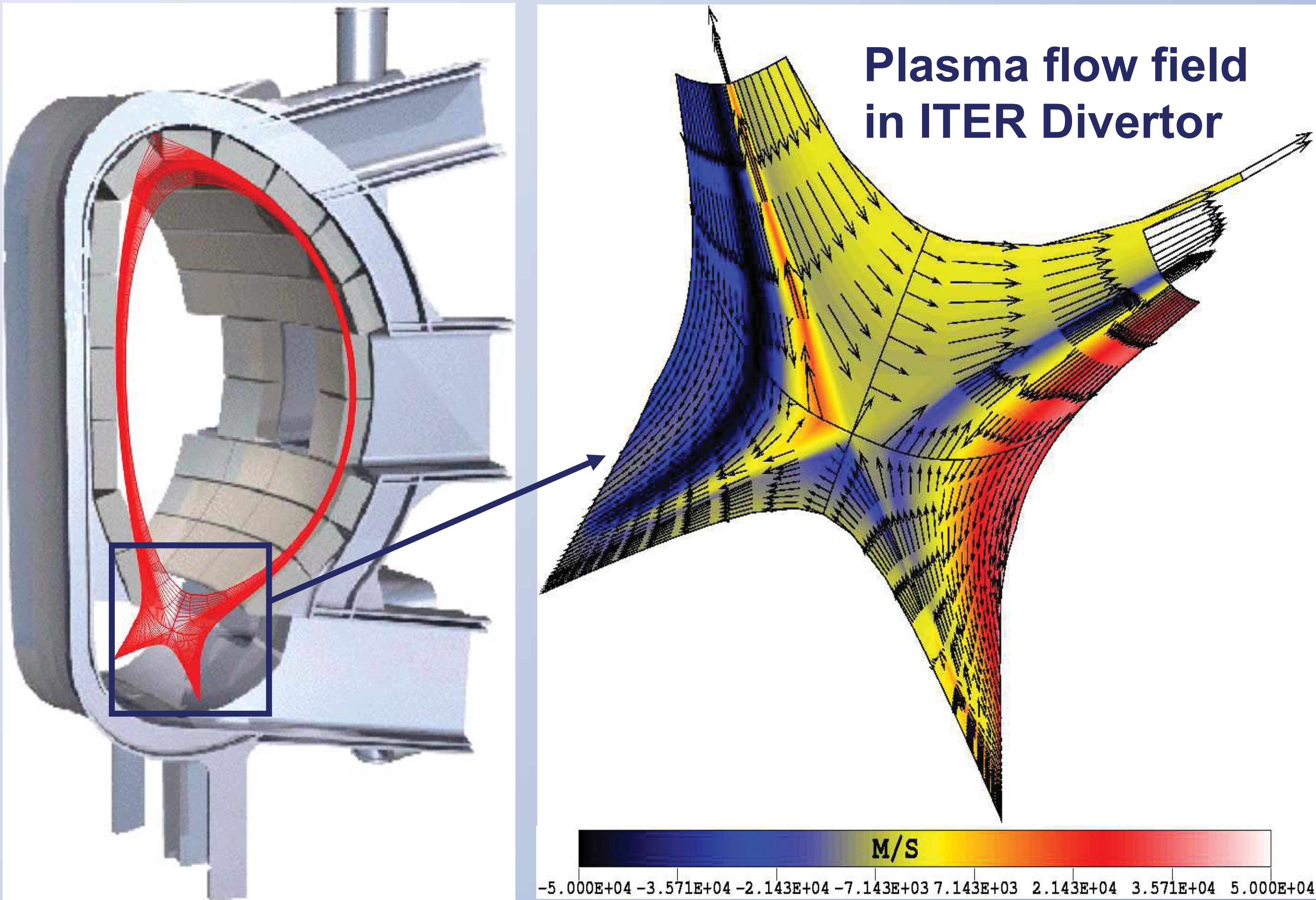
Here: restriction to 2D axi-symmetric plasmas



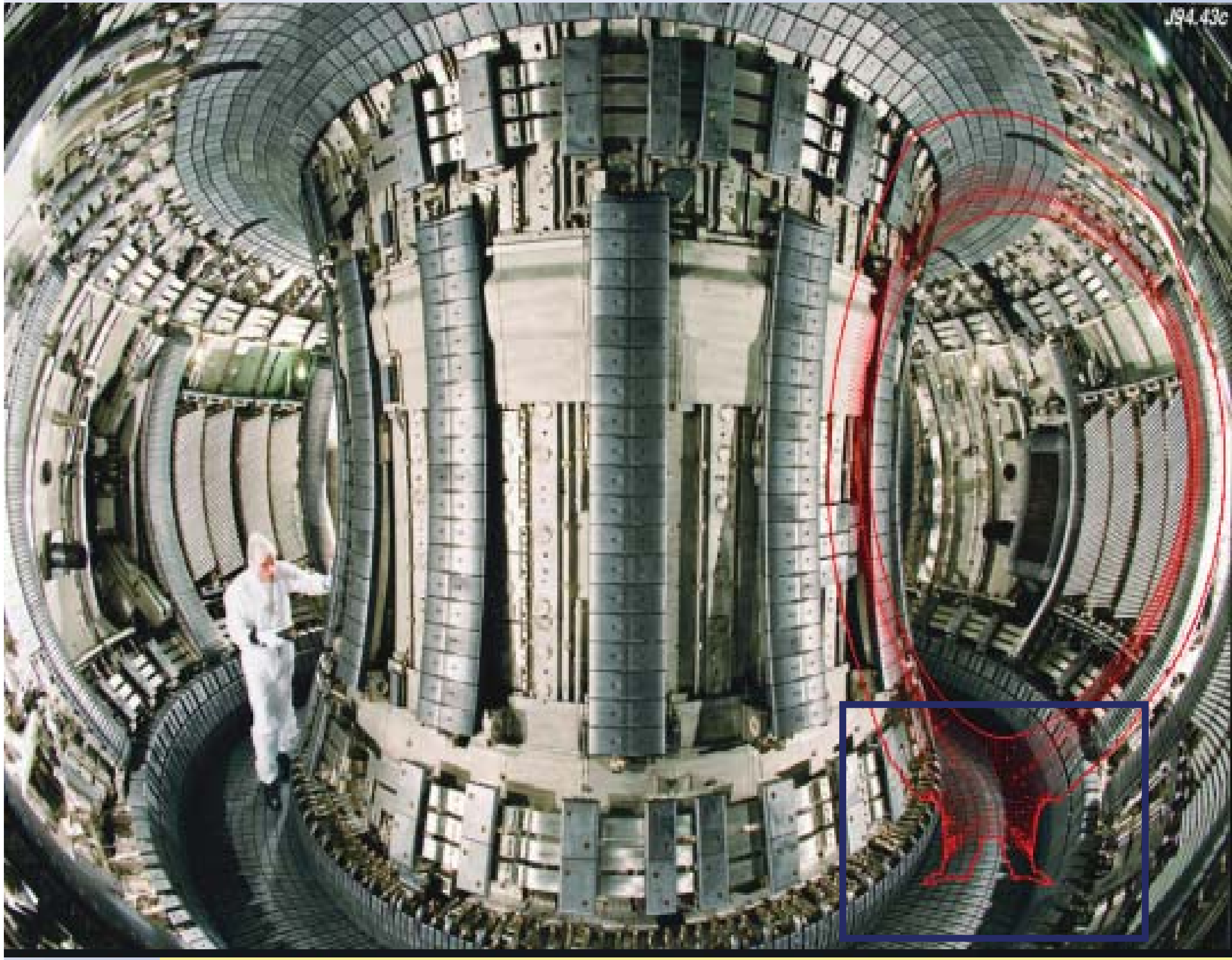
Numerische Simulation von Plasmaströmungen für ITER



The ITER Divertor



Active role of recycling and neutral particle transport
cooling the edge plasma, protecting target surfaces from overexposure



JET Furnace chamber:

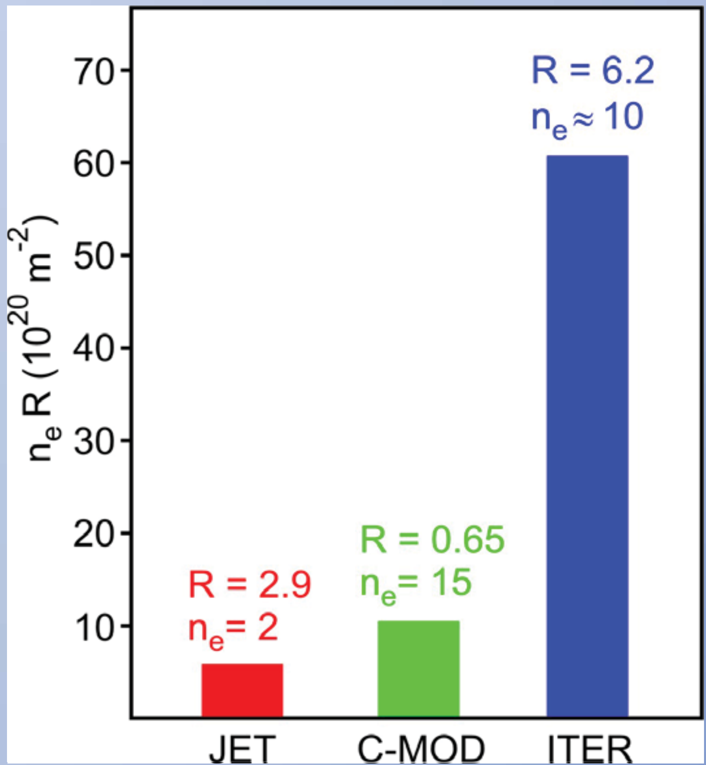
Ø 8.5 m 2.5 m high 3.4 T 7 MA 1 min

Recycling: a) provide convection
b) protect exposed target areas

World-wide effort to understand (and predict?)
Edge Plasma dynamics on the basis of best known
Plasma Surface Interaction and Atomic & Molecular Processes

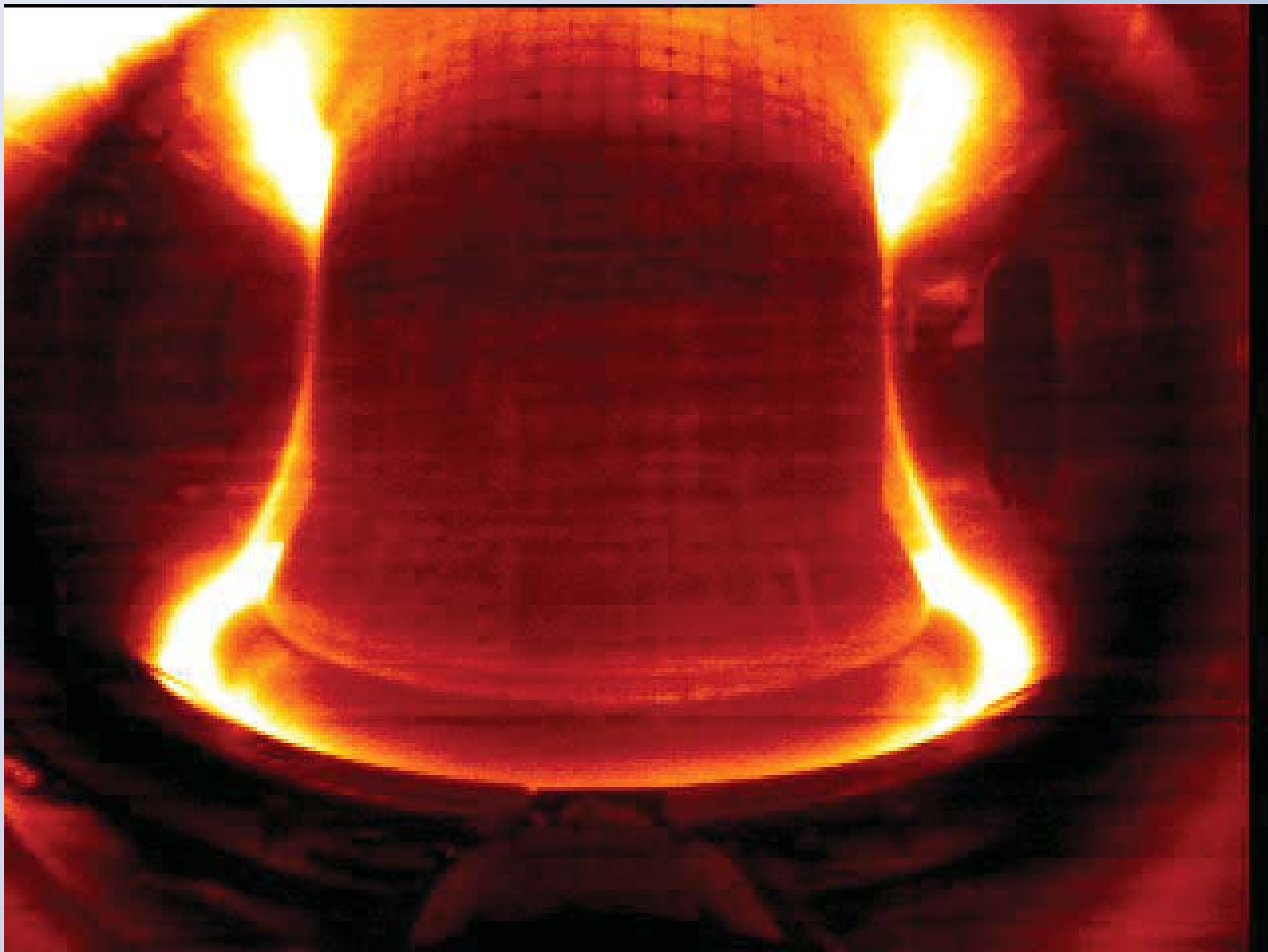
Estimate "Collisionality": n_eR
-n_e-Divertor Plasma density (10^{20} m^{-3})
-R- Major Radius (m)

Alcator C-Mod (MIT)
10 times smaller than ITER
similar shape
higher density



Alcator C-Mod (MIT)

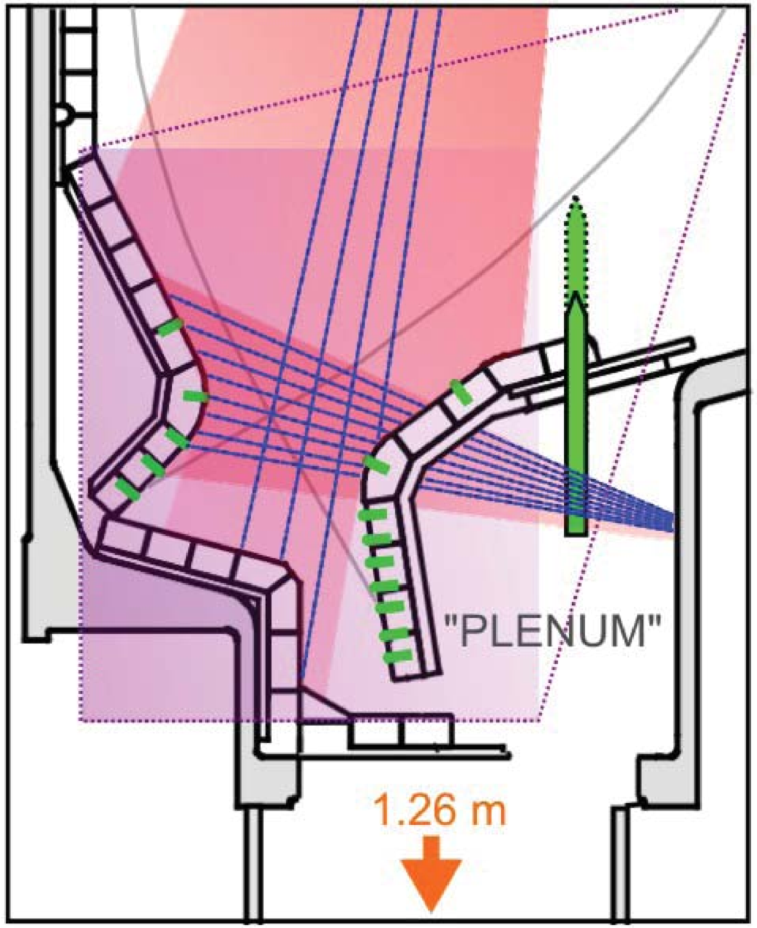
Alcator C-Mod (MIT)



HIGH RESOLUTION DIODE ARRAYS WITH D_α FILTER

TARGET LANGMUIR PROBES AND UPSTREAM RECIPROCATING PROBE FOR n_e AND T_e

DIVERTOR GAS PRESSURE (25±3 mTorr)

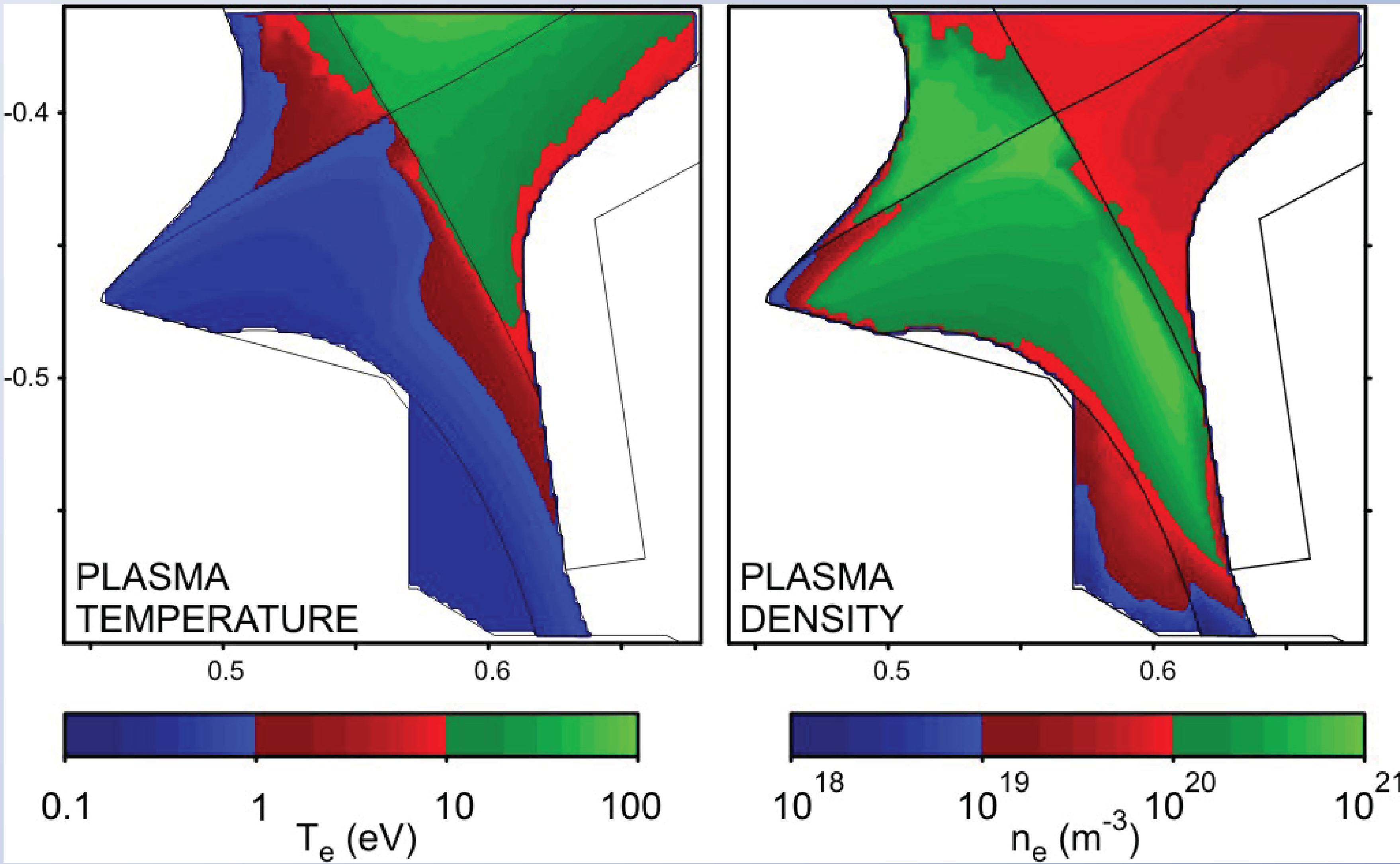


TOROIDALLY VIEWING CCD CAMERA WITH D_γ FILTER

SPECTROMETER FOR VOLUME n_e AND T_e

C. BOSWELL
B. LaBOMBARD
B. LIPSCHULTZ
A. NIEMCZEWSKI
S. PITCHER
J. TERRY

Shot: 990429019, at 950ms,
$\langle n_e \rangle = 1.5\ 10^{20}$, $I_P = 0.8$ MA, $B_{tor} = 5.4$ T
OSM reconstruction (Lisgo et al., 2004)

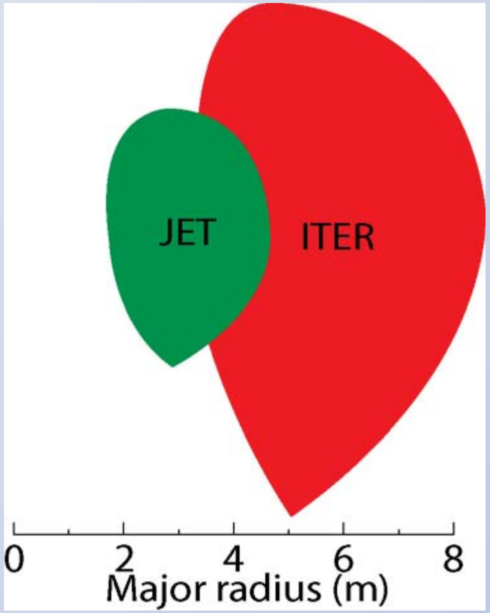


Plasma-surface interactions

Upscale to ITER is a big step

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

*Code calculation

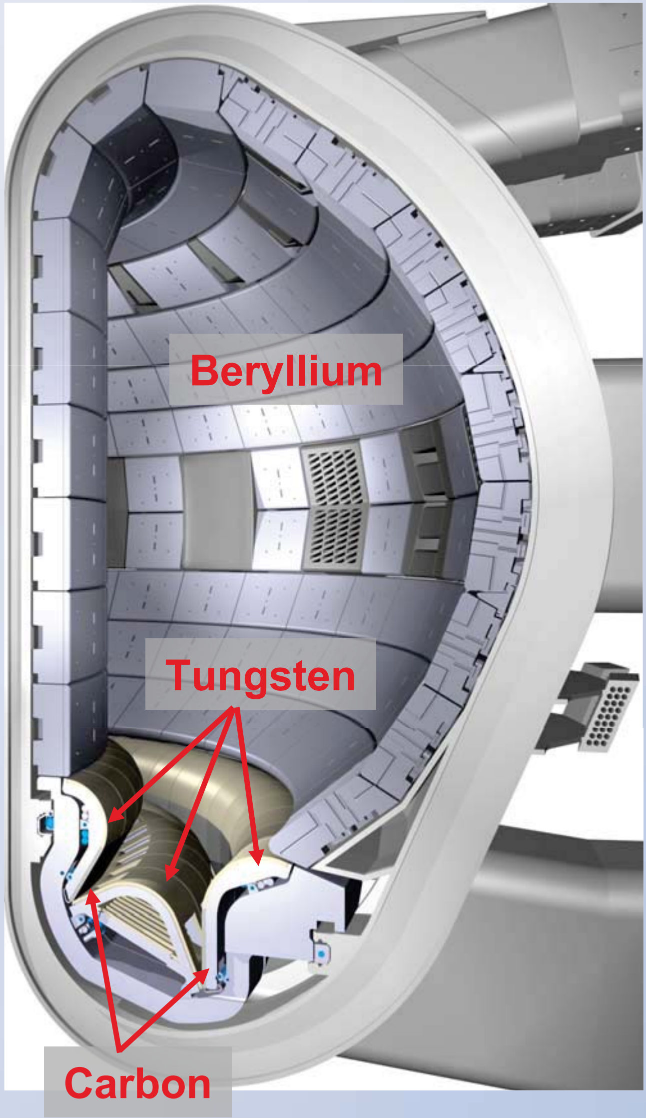


1 ITER pulse ~ 0.5 JET years energy input

1 ITER pulse ~ 6 JET years divertor fluence

Courtesy: G. Matthews

ITER PFC Environment



Courtesy: J. Roth

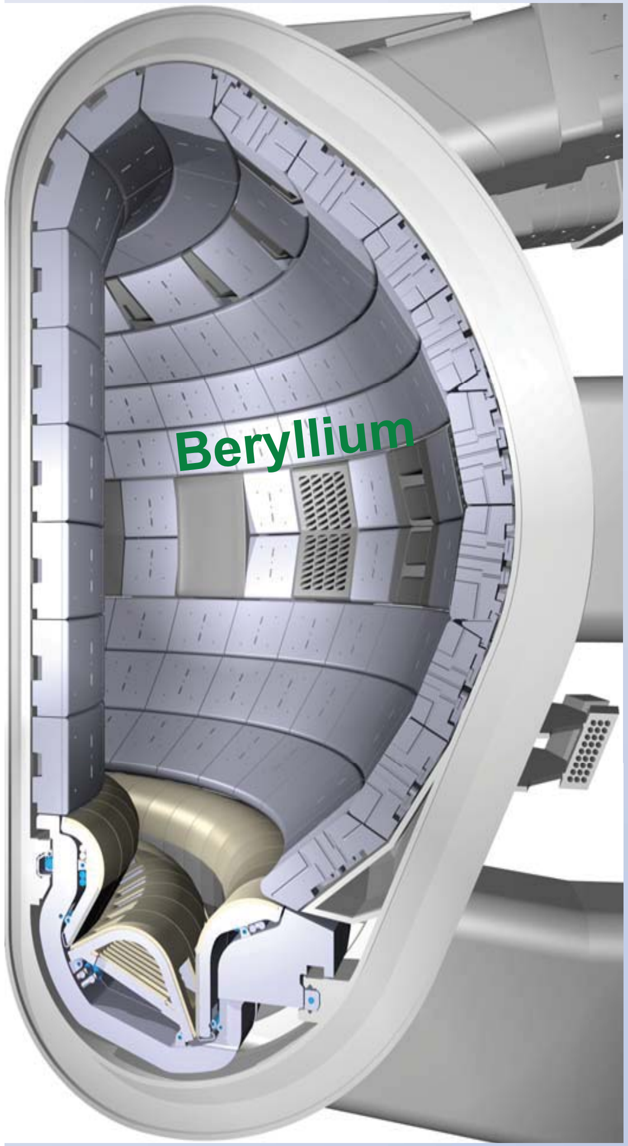
Initial reference material mix (H, D phases):

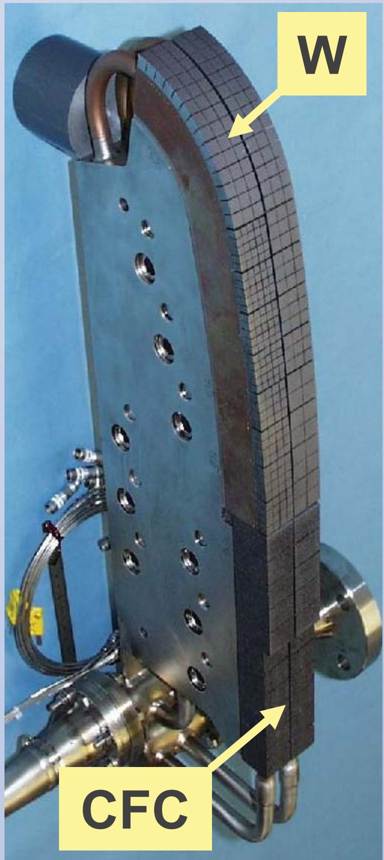
- **700m² Be** first wall and start-up limiter modules
- **100m² W** divertor dome and baffle region
- **50m² Carbon Fibre Composite** (CFC) for the divertor strike point areas

Present strategy for ITER operation

- change to a **full W-divertor** before DT operation
- Decide on specific time for change on the basis of experience on hydrogen retention and dust
- **all-W** as future DEMO relevant choice

ITER materials choices





Courtesy: R. Pitts

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

For H and part of D phase: C for the targets
- Low Z
- Does not melt
- Excellent radiator

W for the dome/baffles
- High Y_{phys} threshold

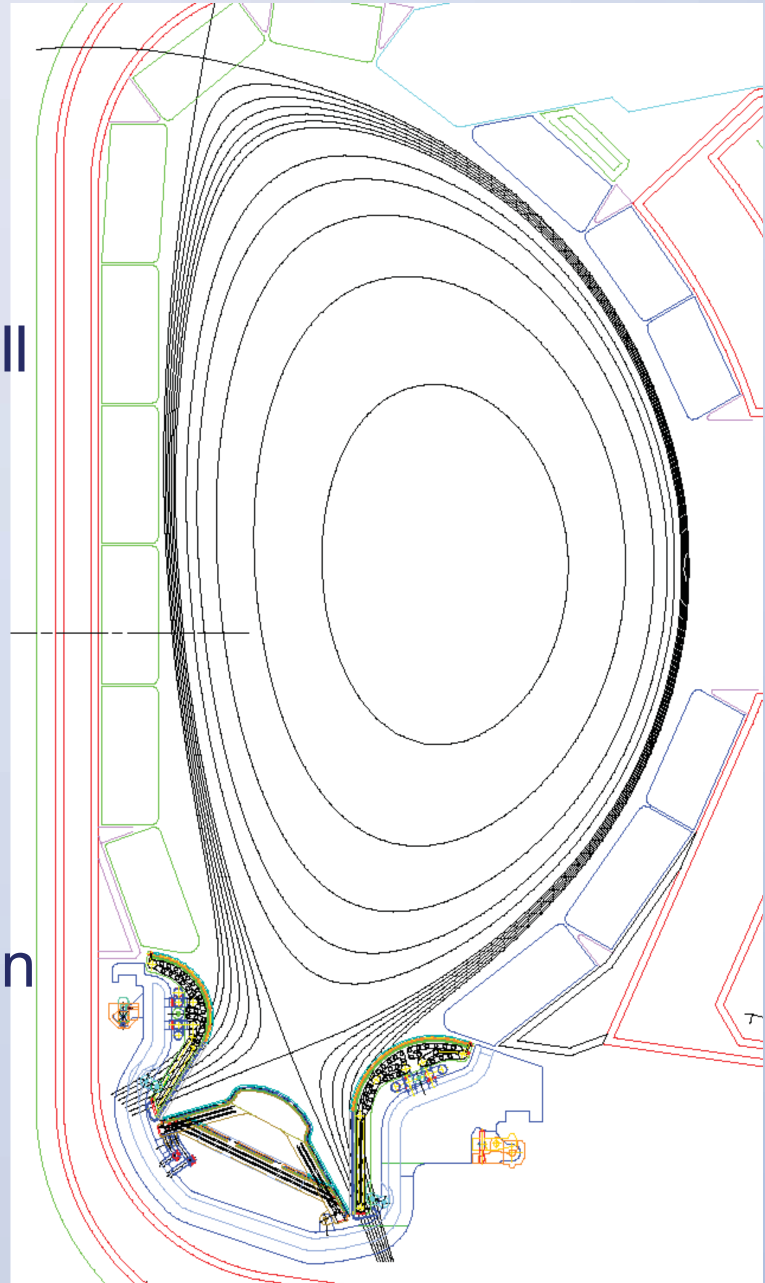
Driven by the need for operational flexibility

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

To avoid problem of T-retention

What are the issues associated with plasma-surface interactions?

ITER



Be-wall

tungsten
baffles

graphite target plates

Erosion

Graphite - a conservative choice
forgiving material, no melting,
3825 ºC sublimation temp

D peak flux 10^{24} m^{-2} s^{-1}
erosion yield about 1%
$\Rightarrow$ 10^{22} C-atoms m^{-2} s^{-1}
for steady state
$\Rightarrow$ 6 000 kg / year or 2.6 m/year

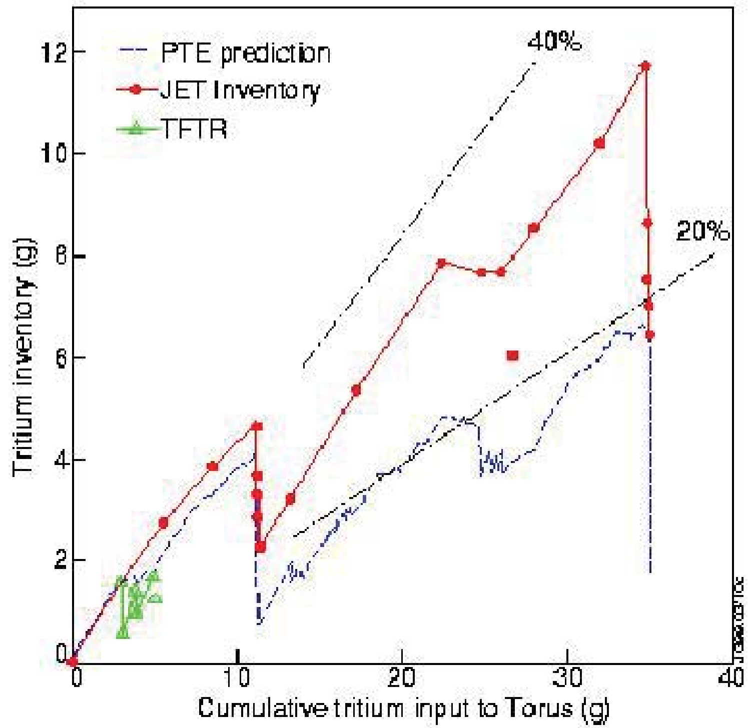
Deposition

the tokamak - a closed system
essentially all eroded particles
are re-deposited

The tritium retention issue:

On JET, operated with tritium, the tritium inventory built up without saturation limit.

JET: Retained Tritium



The rate of T retention in JET during DTE1 was 40% of input, reducible to 17% after cleanup in D, without sign of saturation.
P. Andrew, et al, FED 47 (1999) 233.

Extrapolation to ITER: the permitted in-vessel inventory, 0.5 kg, could be reached in 100 shots

Availability – the main remaining challenge of fusion research

cost of electricity:

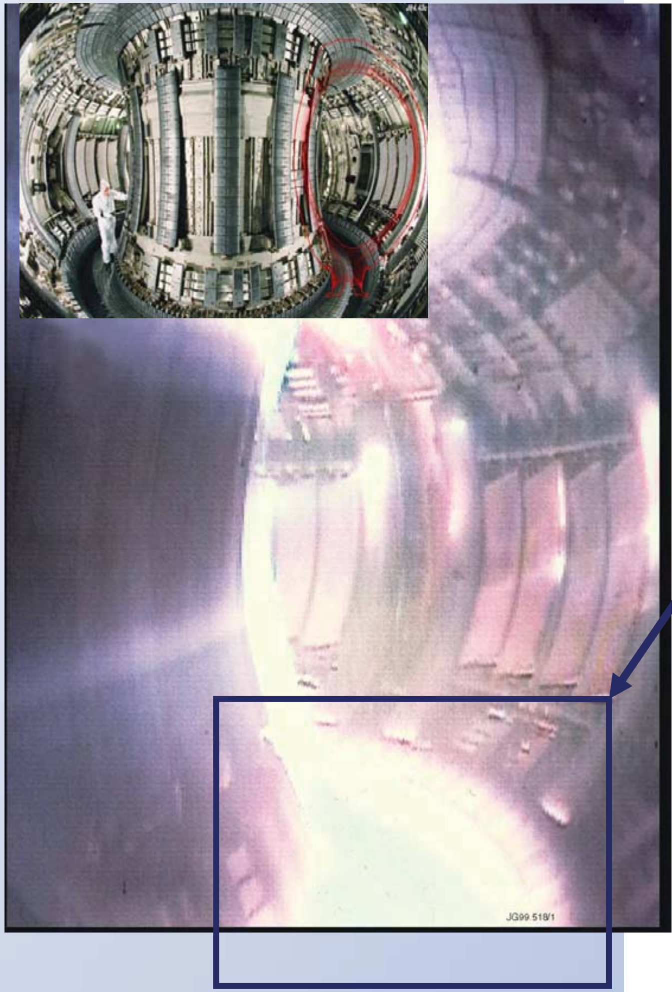
$$\mathrm{COE} \propto A^{-0.6}\,\eta_{th}^{-0.5}\,P_e^{-0.4}\,\beta_N^{-0.4}\,N^{-0.3}$$

availability (A)
thermodynamic efficiency (η_{th})
unit size (net electrical output, P_e)
normalised beta (β_n)
limiting density normalised to the Greenwald density (N)

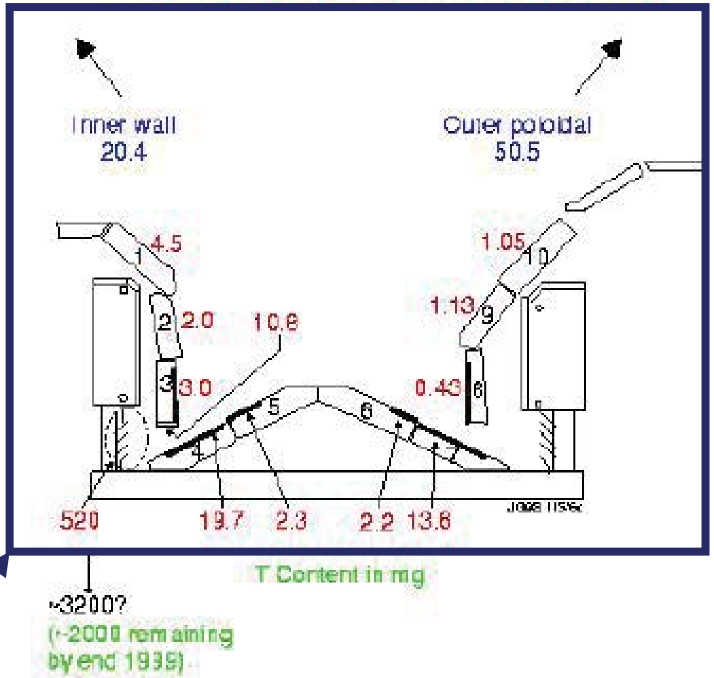
Power Plant Conceptual Study (PPCS) Stage II
D J Ward, I Cook, N P Taylor

the key issues determining the availability:
- life time of wall components
- tritium retention

Carbon re-deposition, Tritium co-deposition



Location of tritium in JET vessel during the post-DTE1 shutdown



The location of the deposition is surprising: only a few mgs were found on typical tiles, but 520 mg were vacuumed up from the cooled, out-of-sight louvers, suggesting up to 3200 mg also that have fallen through to the vessel floor.

J.P. Coad, et al, J Nucl Mater 290-293 *(2001) 224.*

On JET, operated with tritium, the tritium inventory built up without saturation limit.

This problem may be so serious as to rule out the use of carbon in fusion devices.

That, however, would eliminate the leading candidate material, and the one that, by a considerable margin, we know most about.

It would be a setback to be driven to the extreme of not being able to keep the carbon option open.

Transport creates and moves impurities

Ions:

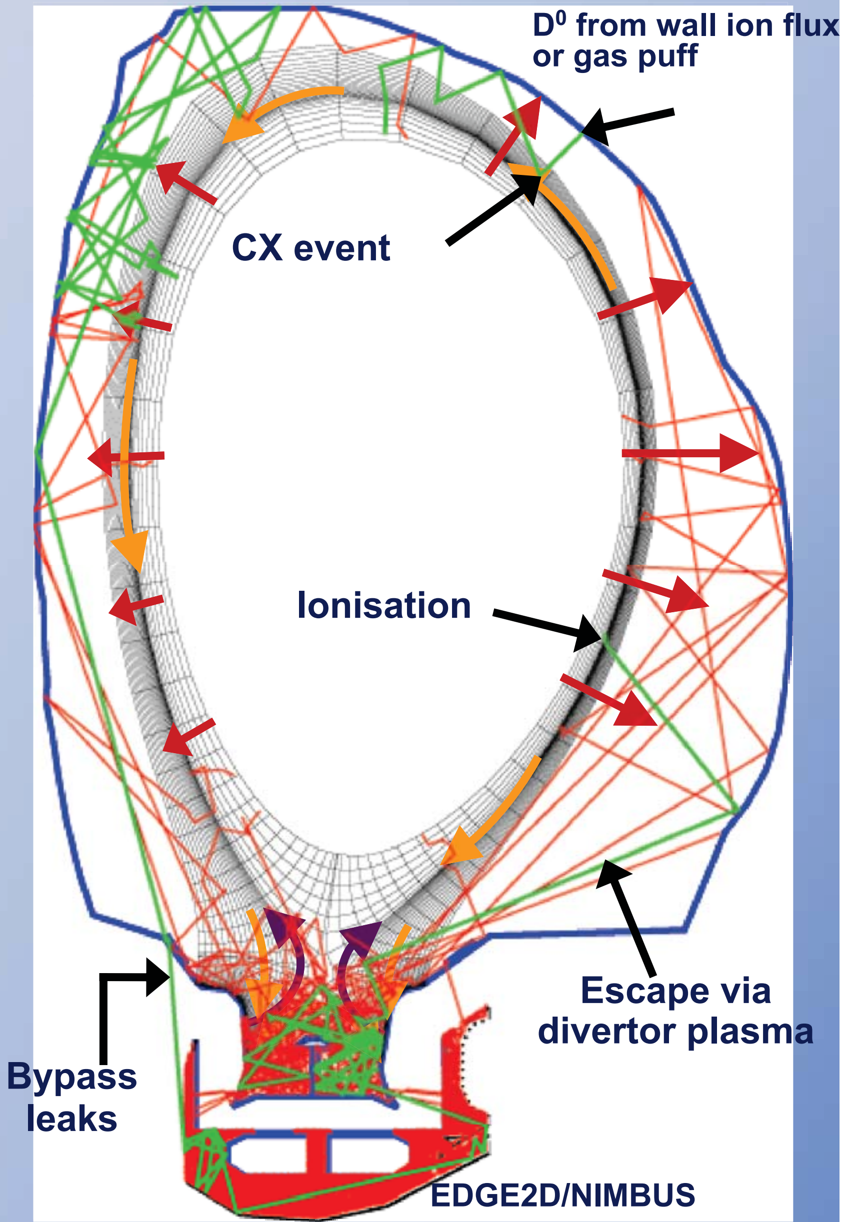
Cross-field transport – turbulent driven ion fluxes can extend into far SOL
→ recycled neutrals
→ direct impurity release
ELMs can also reach first walls

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

SOL and divertor ion fluid flows can entrain impurities

Neutrals:

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



Courtesy: R. Pitts

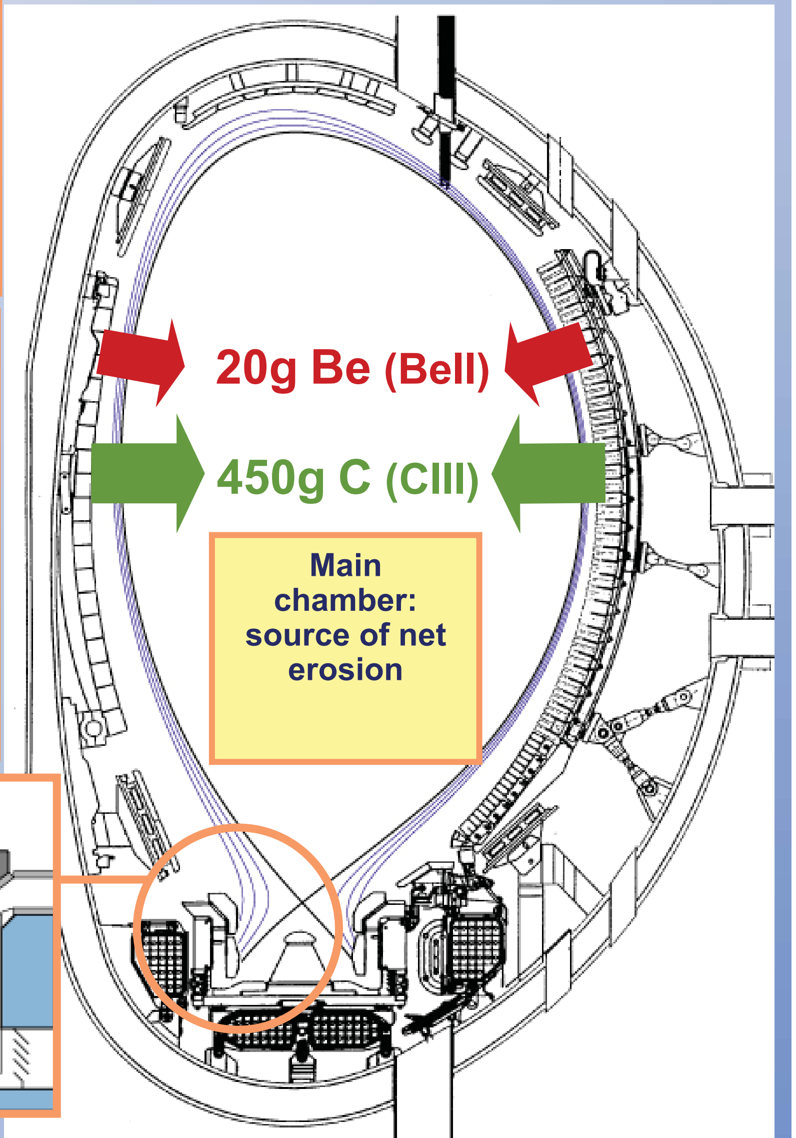
Migration balance – example from JET

Make balance for period 1999-2001 with MarkIIGB divertor:
14 hours plasma in diverted phase (50400 s, 5748 shots)

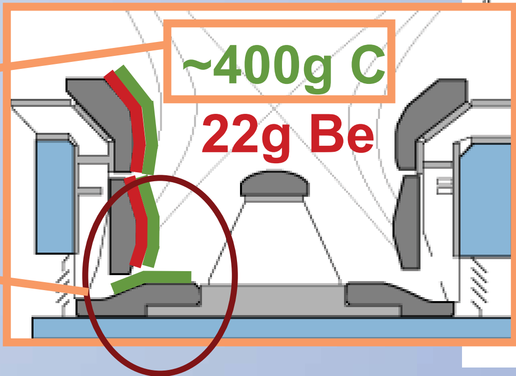
Use spectroscopy and modelling to estimate main chamber sources

Post mortem surface analysis

- Deposition almost all at inner divertor
- Surface layers are Be rich → C chemically eroded and migrates, Be stays put
- Outer divertor – region of net erosion or balanced erosion/redeposition – BUT mostly attached conditions (not like ITER)



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



Courtesy: R. Pitts

Tritium retention (1)

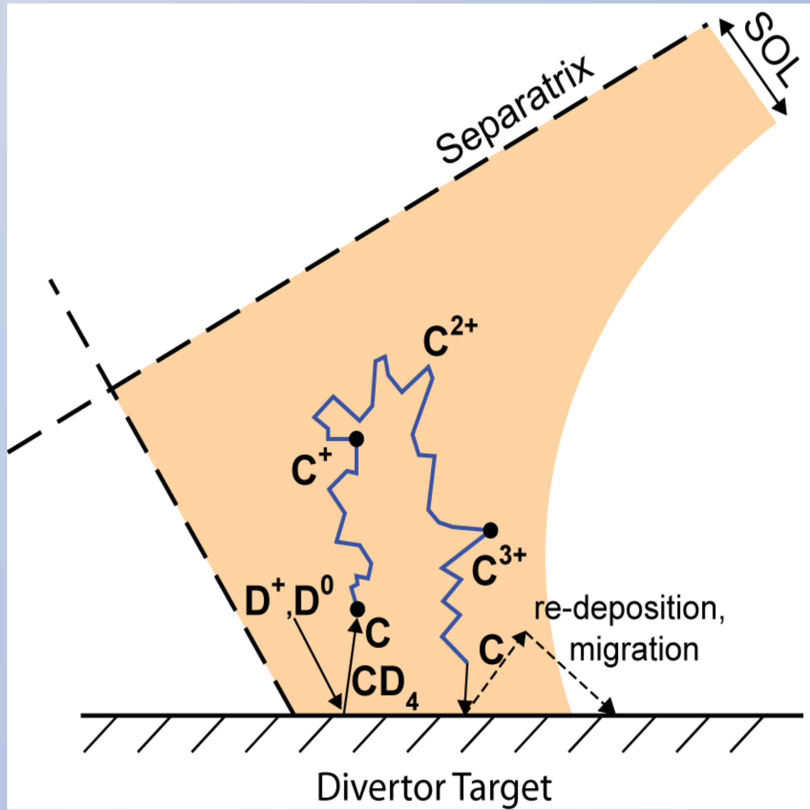
One of the most challenging operational issues for burning plasmas

If carbon present, complex interplay between erosion → hydrocarbons → dissociation/ionisation → transport → re-deposition → migration to remote areas with high sticking coefficients and retention in co-deposits

- Carbon traps D, T very efficiently
- D/C ratio can be in the range ~0.4 → > 1 depending on the type of re-deposited layer

Retention very hard to characterise in today's mostly carbon dominated devices

Dependent on materials, T_{surf}, geometry (limiter/divertor), operating scenarios (H-mode, L-mode, low/high dens.)



Reported measurements range from **3-50% retention**

e.g. on JET, ~3% obtained from long term, post mortem surface analysis, ~10-20% from gas balance.

Courtesy: R. Pitts

Tritium retention (2)

A 400 s Q_{DT} = 10 ITER discharge will require ~50 g of T fuelling (cf. 0.01-0.2 g in today's tokamaks)

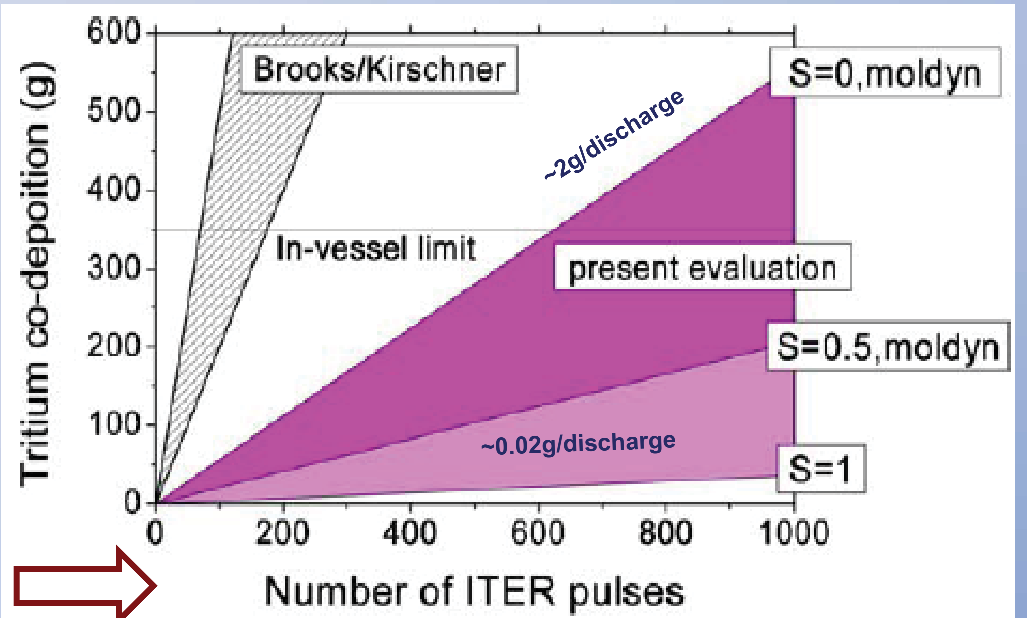
Working guideline for max. in-vess. mobilisable T in ITER ~1kg

World supply of T is also limited

Must avoid build-up in inaccessible locations

Predicting the expected retention in ITER is notoriously difficult

C targets, T_{surf} = 800°C, chemical+physical sputtering



ITER target is a retention level of ~0.05 g/discharge → ~7000 shots before major shutdown for T-removal

Accurate measurement of T-retention and the development of efficient T-removal methods will be critical for the success of ITER

The consequences of tritium retention for ITER

Extrapolations of tritium retention results to ITER

after how many ITER pulses do we reach the limits for tritium retention ?

Extrapolation from experiments	D,T flux (#/s)	T-retention rate (T/ion)	ITER retention gT/s extrapolation (flux: $1.8\ 10^{24}$/sc)	shots /T-limit (400 sec)
TEXTOR	$5\ 10^{20}$/s	$6.4\ 10^{-4}$	0.0064	136
JET T experience	$1.2\ 10^{22}$/s (inner only)	$1.75\ 10^{-2}$ (only louver)	0.10g	9
JET GB on tiles	$2\ 10^{22}$/s	$2.7\ 10^{-3}$	0.024	36
JET C5 on louver from QMB	$1.9\ 10^{22}$/s	2.9 10-4	0.0026	340
Modelling				
ERO-code (2% CxHy er.)			0.006	145
WBC code			0.007	125

large uncertainties, but in any case critical

Mixed Materials

No fusion device operating today contains the material mix currently planned for the ITER first wall and divertor: Be, W, C. Cross contamination of the material surfaces will be unavoidable. This is likely to have several consequences:

Material property changes due to mixing

- Formation of metallic carbides → diffusion of C into bulk material at high temperatures
- Formation of Be-W alloys → melting point can be reduced by as much as ~2000ºC

Effect on H-isotope retention

- Retention of H in BeO can be as high as in C
- Retention in W can be increased by C or oxide layers but is very low in pure W or Be
- Very complex – difficult to predict yet for ITER

Effect on material erosion

- Can both increase and decrease erosion!
- Heavy ions (e.g. Be^{Z+}, C^{Z+}) on C, W → increased phys. sputt. but surface coverage (e.g. Be on C) reduces chemical sputtering.

Preparations underway at JET to test a Be/W & Be/W/C wall mix from ~2011

Courtesy: R. Pitts

Divertor and SOL physics

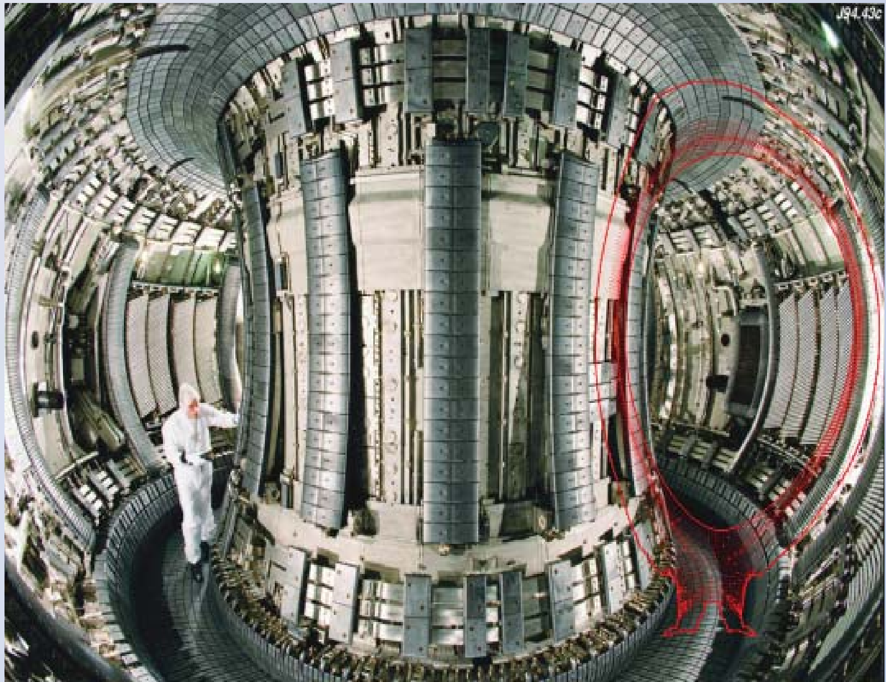
- Experimental finding:
  Sheath limited flow $\Rightarrow$ high recycling $\Rightarrow$ detachment

- Theoretical hypothesis:
  This is brought about by power- and flux dissipation
  due to a chemically rich self sustained
  plasma formed near exposed target surfaces,
  by the recycling process.

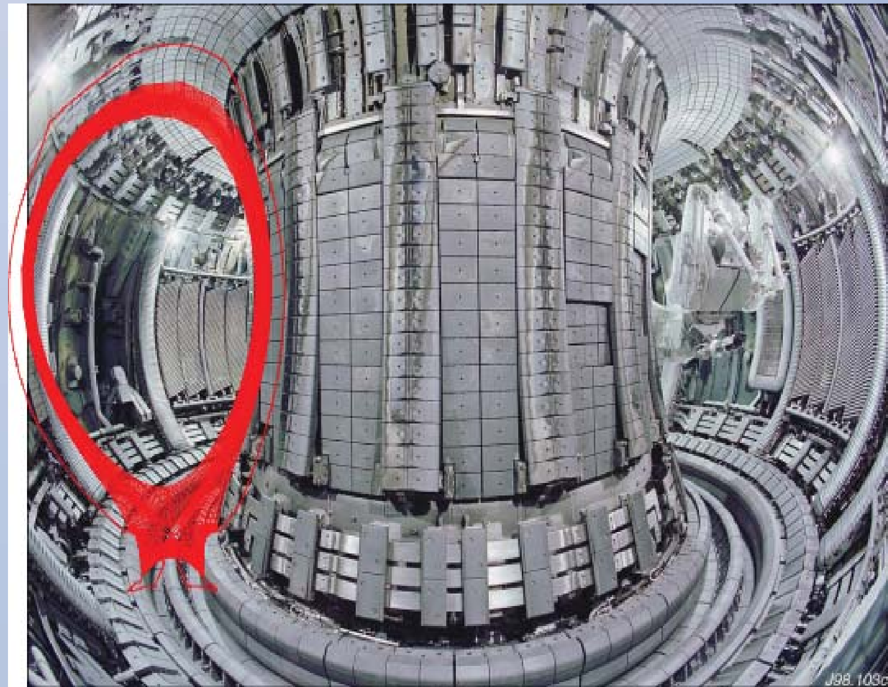
- Experimental tests:
  numerical experiments with
  integrated computational plasma edge models

Experimental findings

JET, 1994, MARK-I Divertor



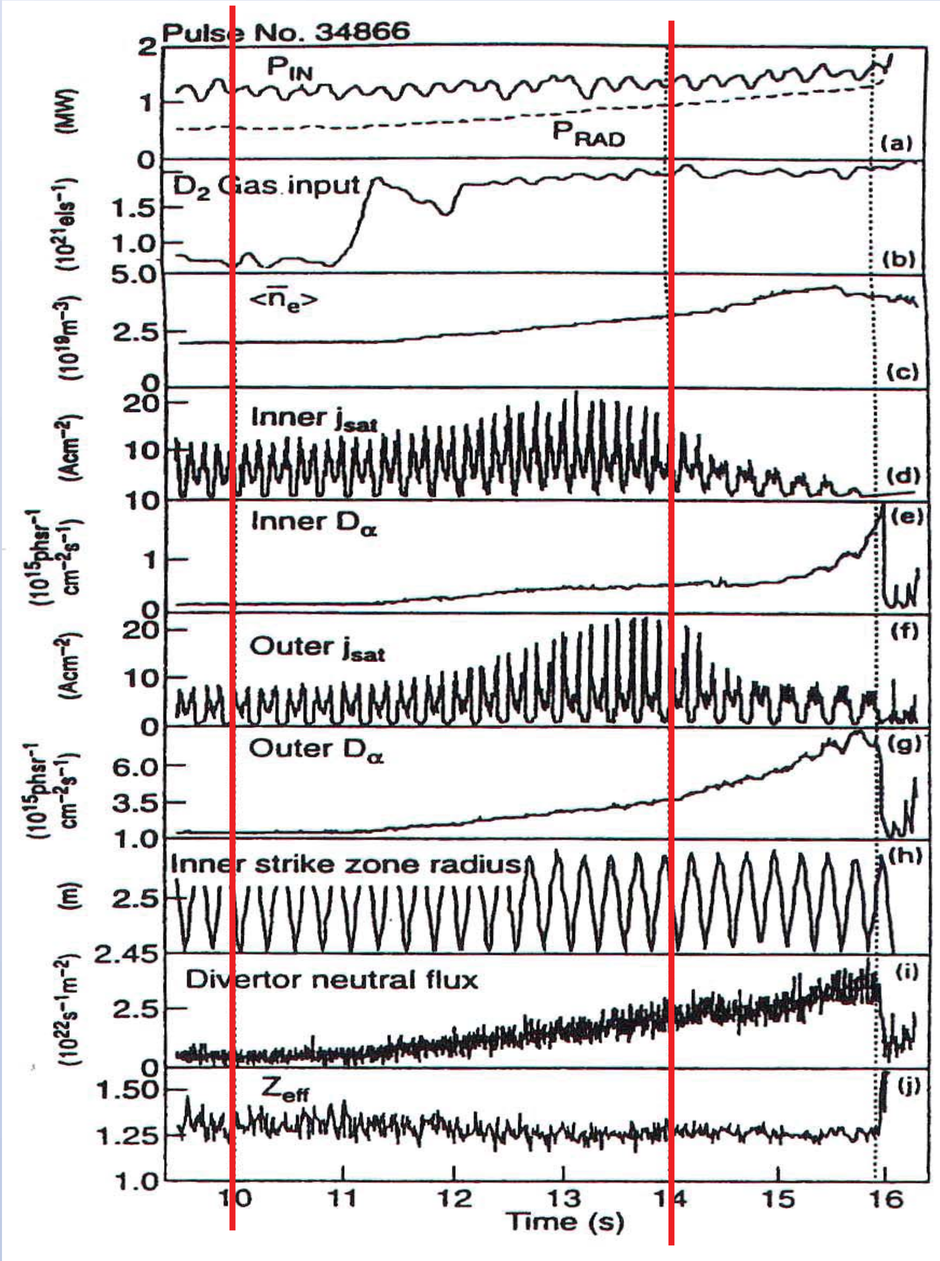
JET, 1998, MARK-II Divertor



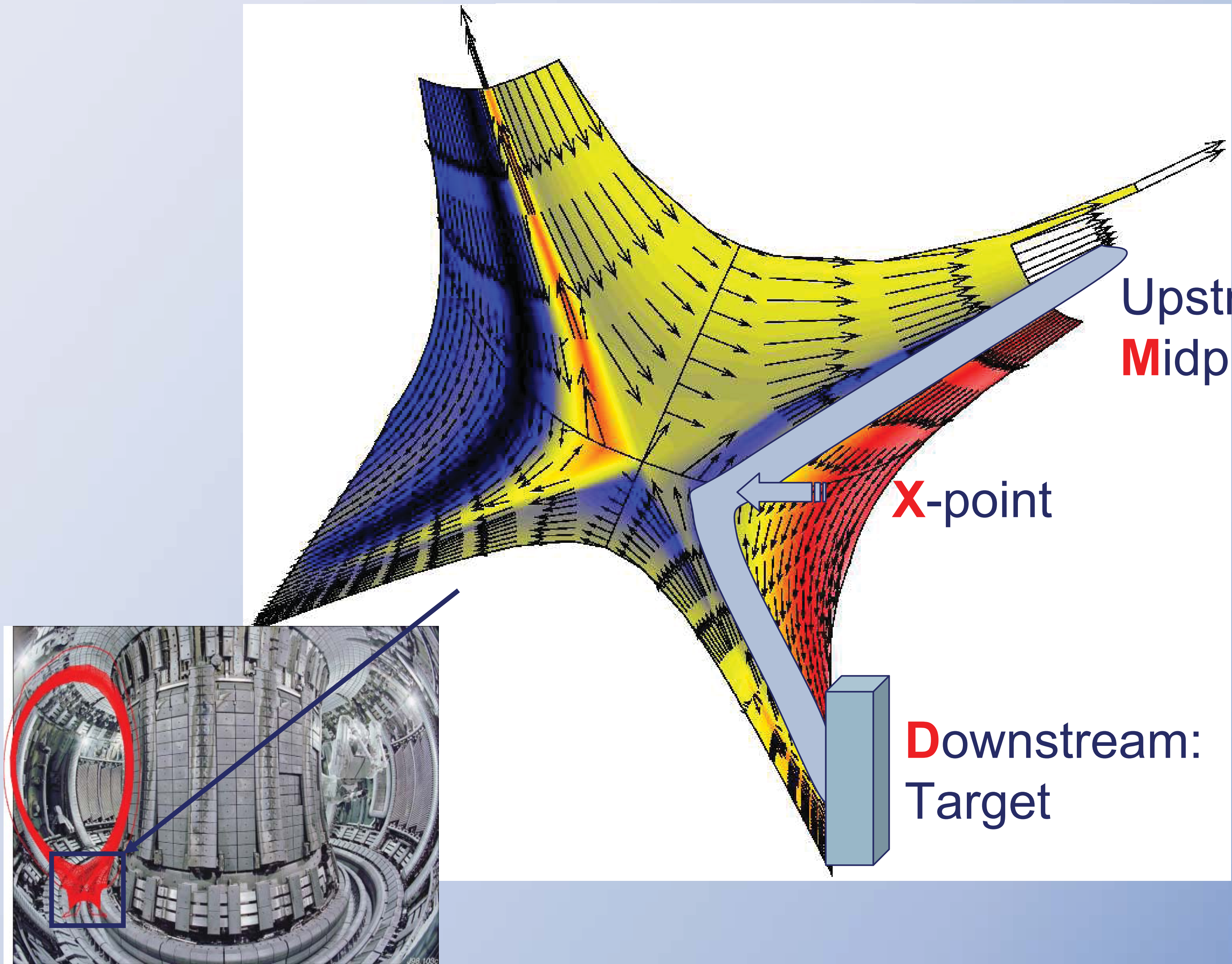
Also indicated: B2-EIRENE-computational grids for JET simulations

JET, MARK-I, density ramp-up

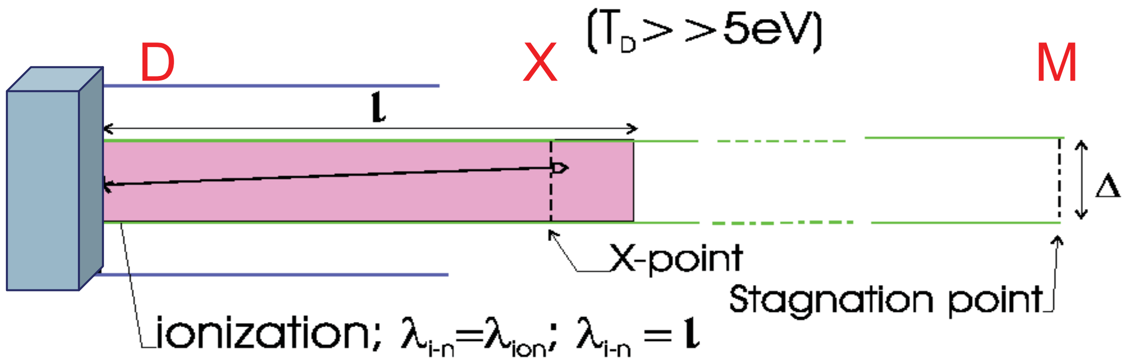
-ohmic
-no imp. injection
-simply: D2-puff



Stretching out a flux-tube: M ⇒ X ⇒ D



Linear, sheath limited regime, convection (.... 1985)



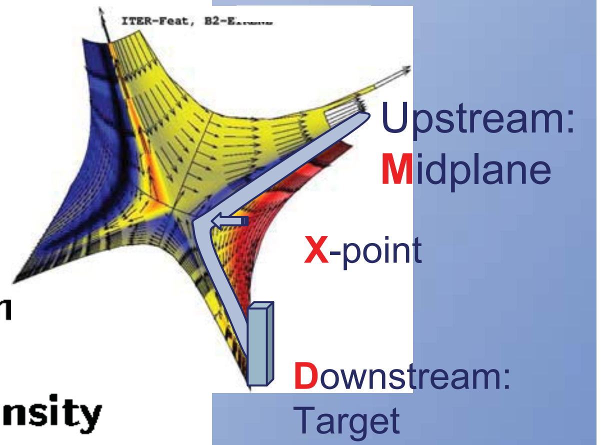
- **low density: high temperature, plasma profiles along fieldlines nearly constant, low radiation losses**

- **energy balance: sheath dominates**

$$q_{\parallel,M} \approx q_{\parallel,D} \approx \delta_e^* T_D \Gamma_D$$

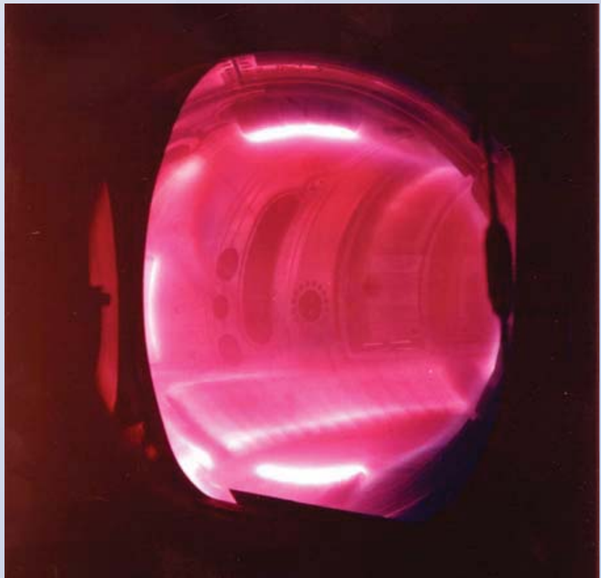
- **small particle flux to the plate: neutral mean free path >> divertor dimension**

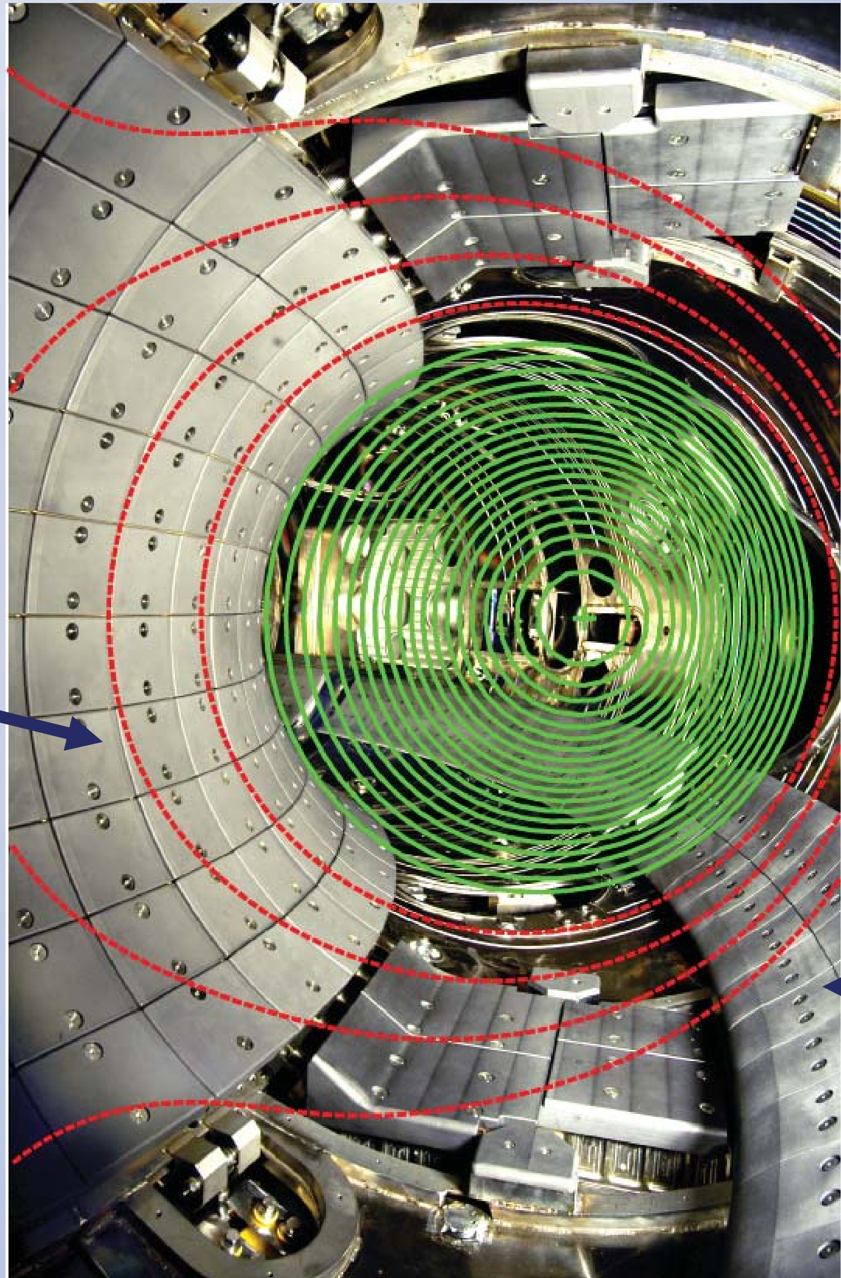
- **divertor density linearly follows midplane density**



TEXTOR
FZJ

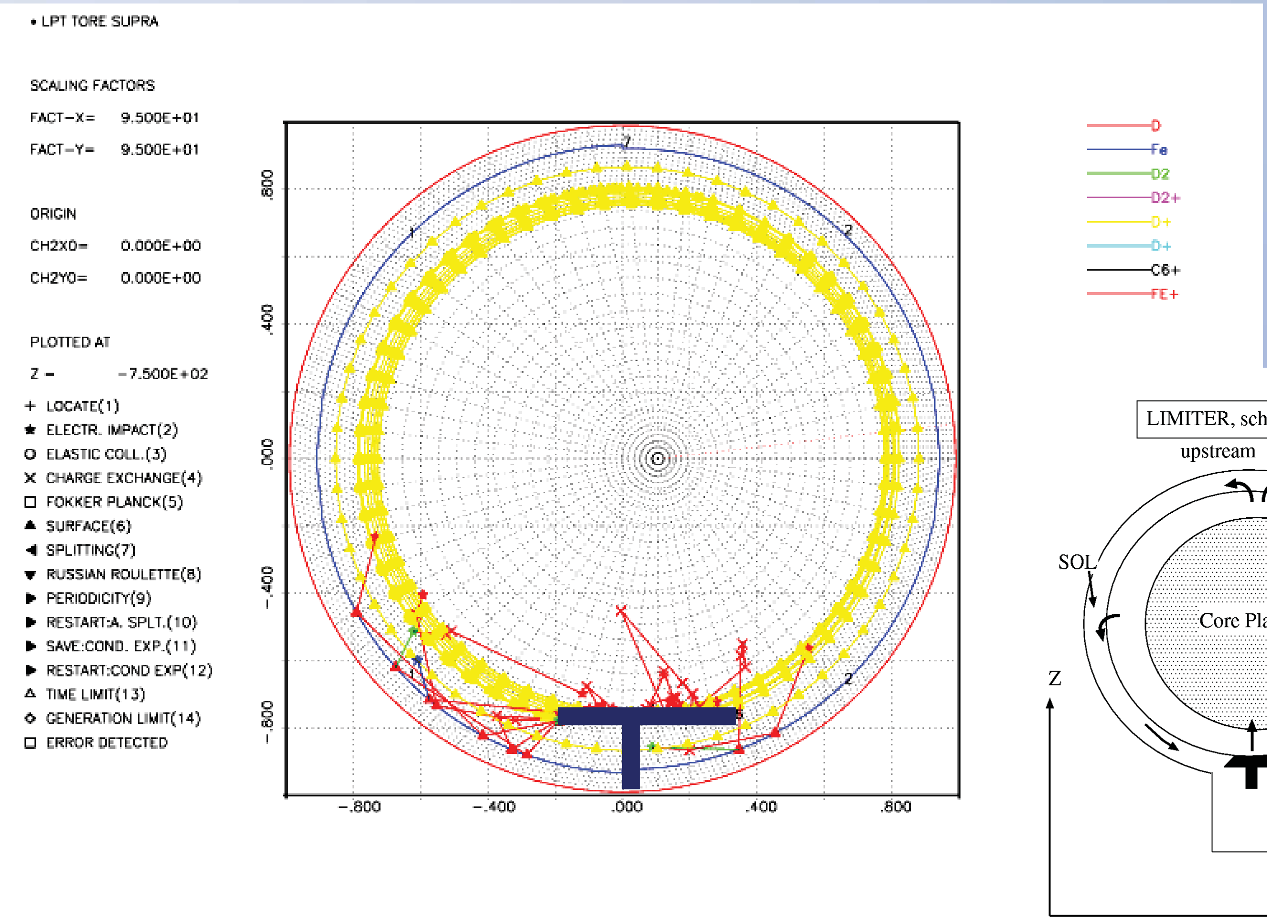
inner bumper
limiter/divertor
Hidden: DED

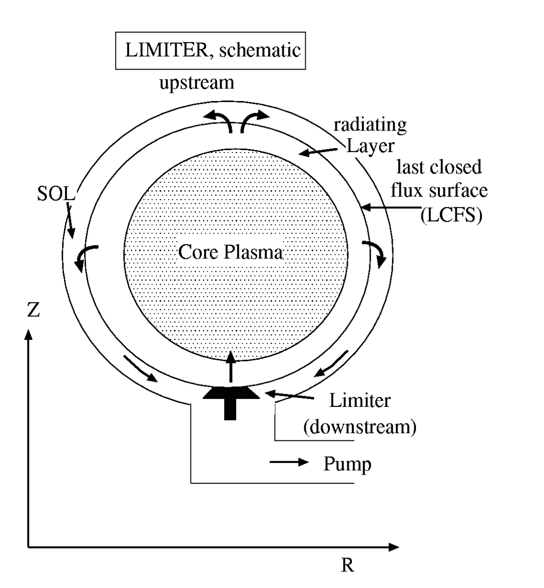




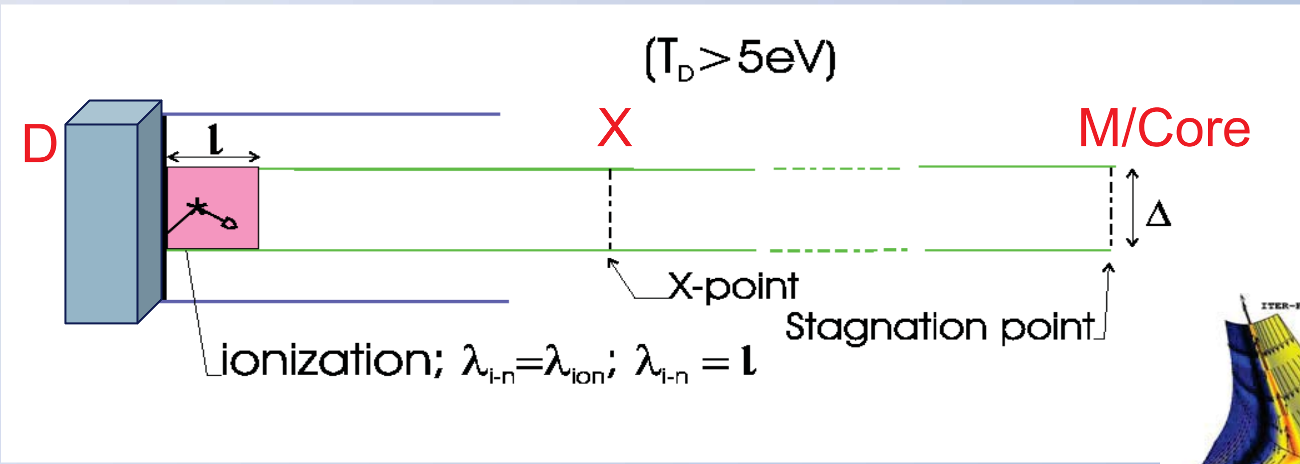
ALT-II
limiter

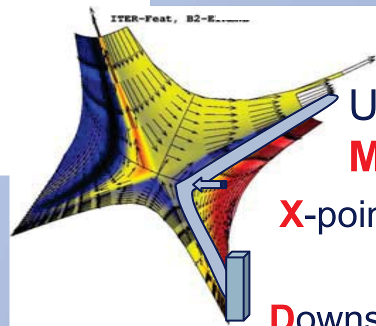
Linear, sheath limited regime: Tore-Supra, TEXTOR





Conduction limited (high recycling): dilution by multiple recycling (1985-1995, ITER CDA)

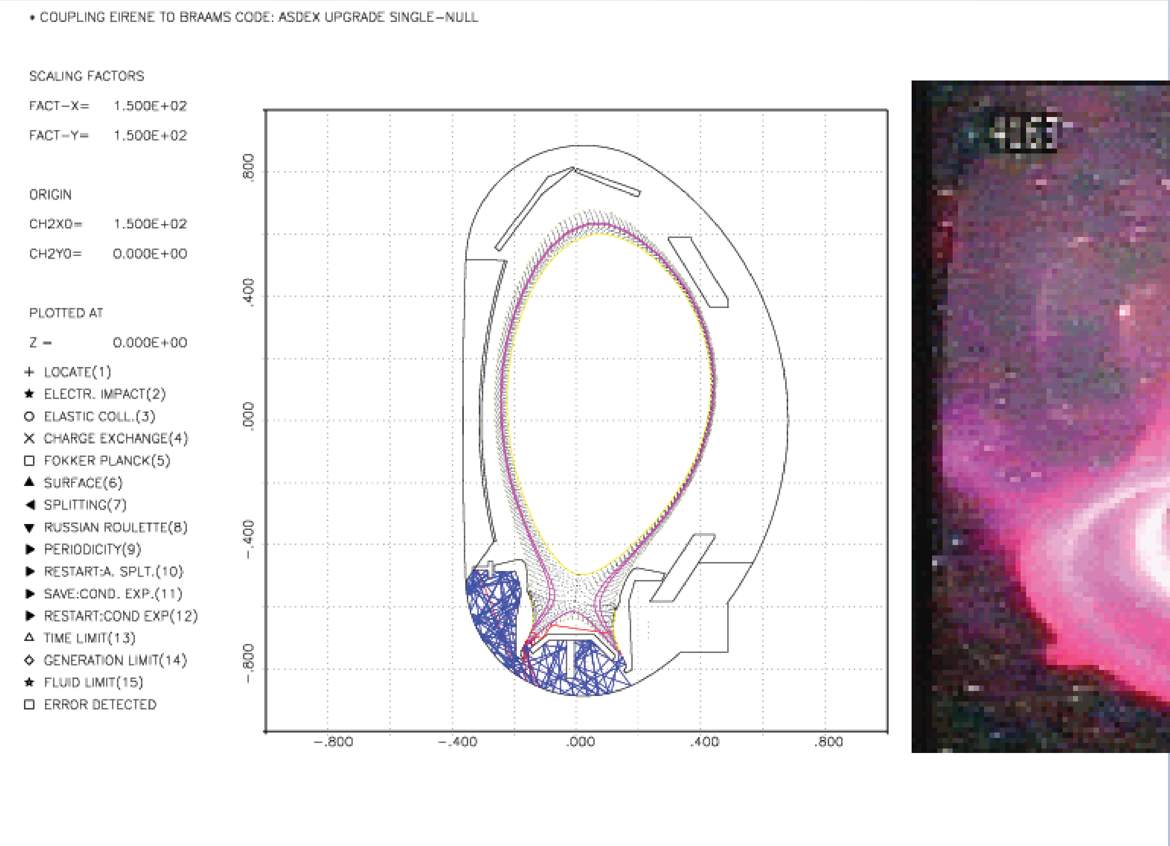


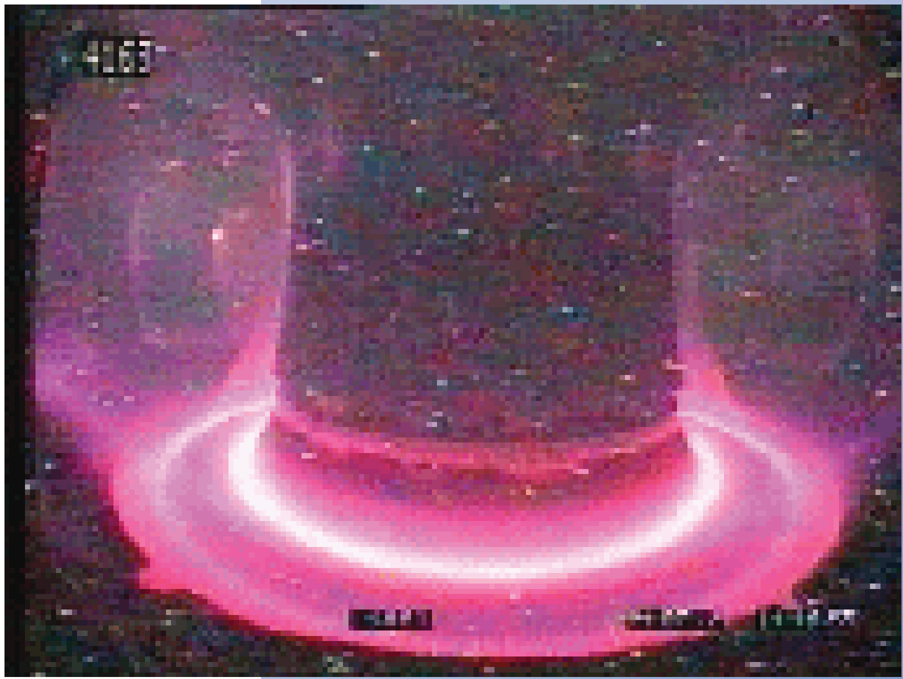


- Lower Midplane temperature (higher density), reduced convection (near target re-ionisation)
  $\Rightarrow$ parallel temperature gradients: low T_e, high n_e near the target D

- Non-linear regime: $T_D \sim n_M^{-2}$, $n_D \sim n_M^{3}$ and $\boxed{\text{flux } \Gamma_D \sim n_M^{2}}$

Trapping of neutral particles in the divertor: high recycling and detachment regime

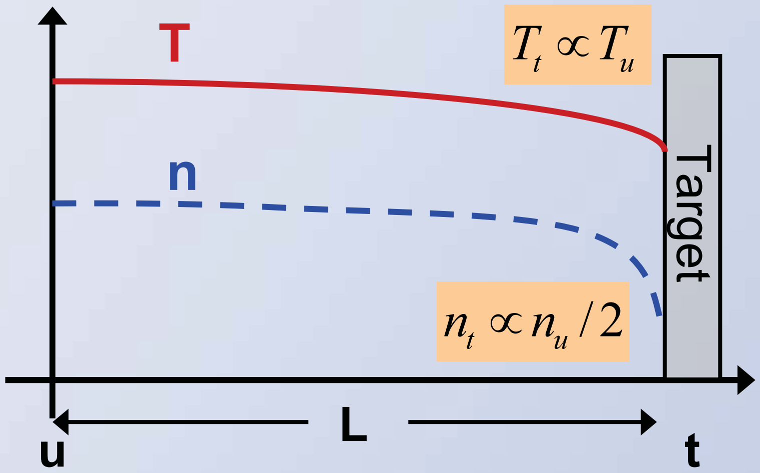




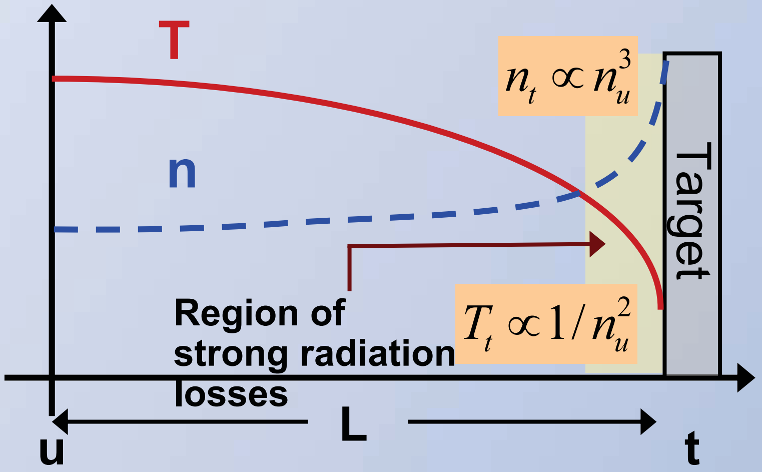
Particle simulation: PWI, A&M **Visible light from ASDEX-U divertor**

The route to detachment (1)

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



Moderate n, T
"High recycling"



Mean free paths for particle collisions are long: $\lambda_{coll} \propto T_u^2/n_u, T_u \sim T_e \sim T_i, \lambda_{ee} \sim \lambda_{ei} \sim \lambda_{ii}$
SOL collisionality: $\nu^* = L/\lambda_{coll}$ **is low**

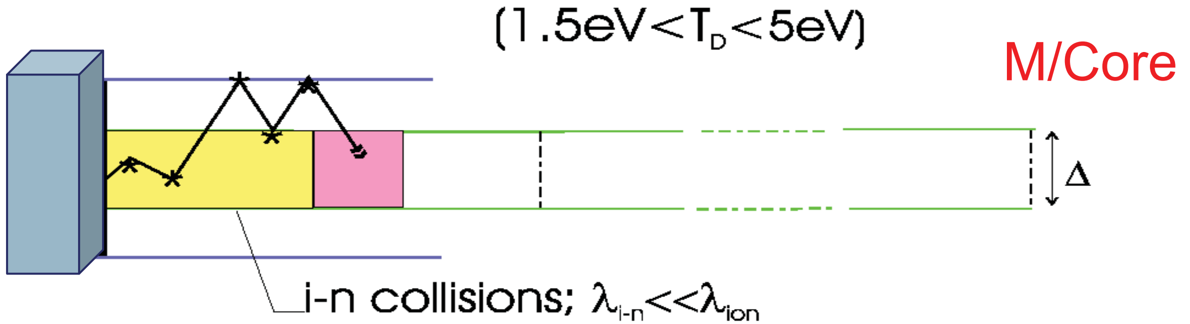
Power flow to surface largely controlled by target sheath: $q_{\parallel,t} = \gamma n_t c_{st} T_t + n_t c_{st} \varepsilon_{pot}$
γ = sheath heat transmission coefficient
ε_{pot} = potential energy per incident ion

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

Courtesy: R. Pitts

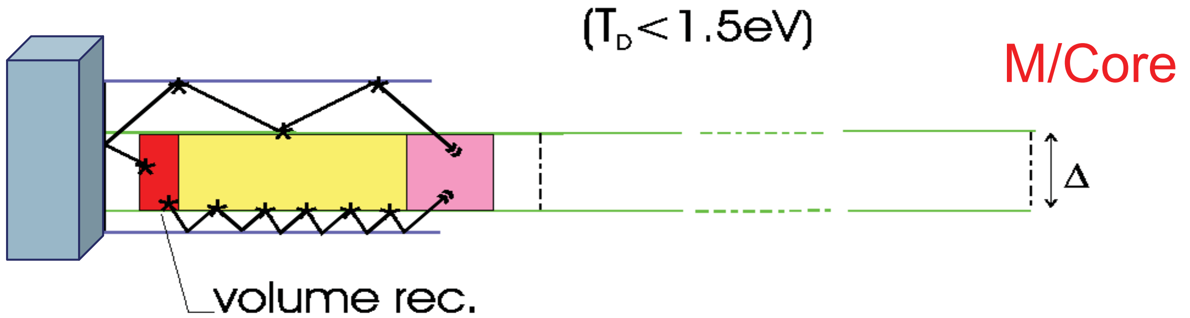
Weak (partial) detachment

(1995 …., ITER EDA)



- **below 5 eV strong momentum dissipation through self sustained A&M processes (CX)**

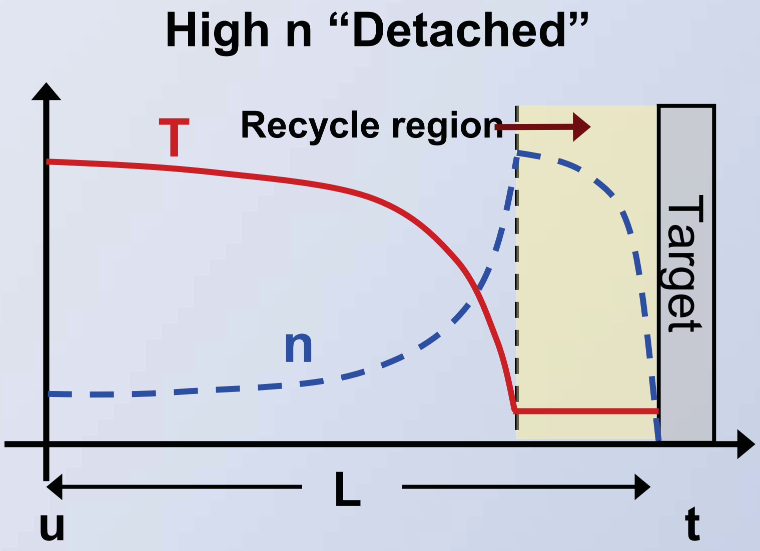
Strong (complete) detachment

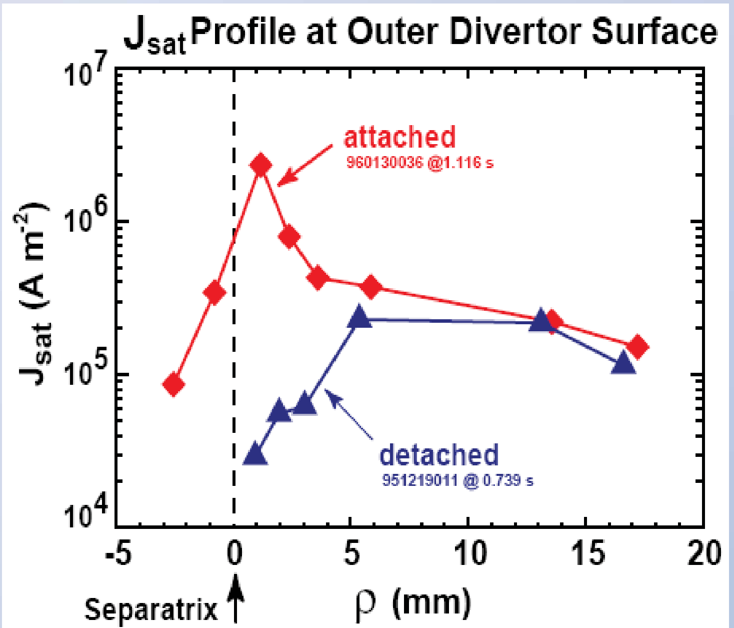


- **Below 1.5 eV additional reduction of plasma flux by volume recombination (virtual target, neutral cushion). Escape of neutrals to the sides followed by ionisation in hotter plasma (6-7 eV) further upstream**

$\Gamma_D \sim n_M^2$ dependence is broken

The route to detachment (2)





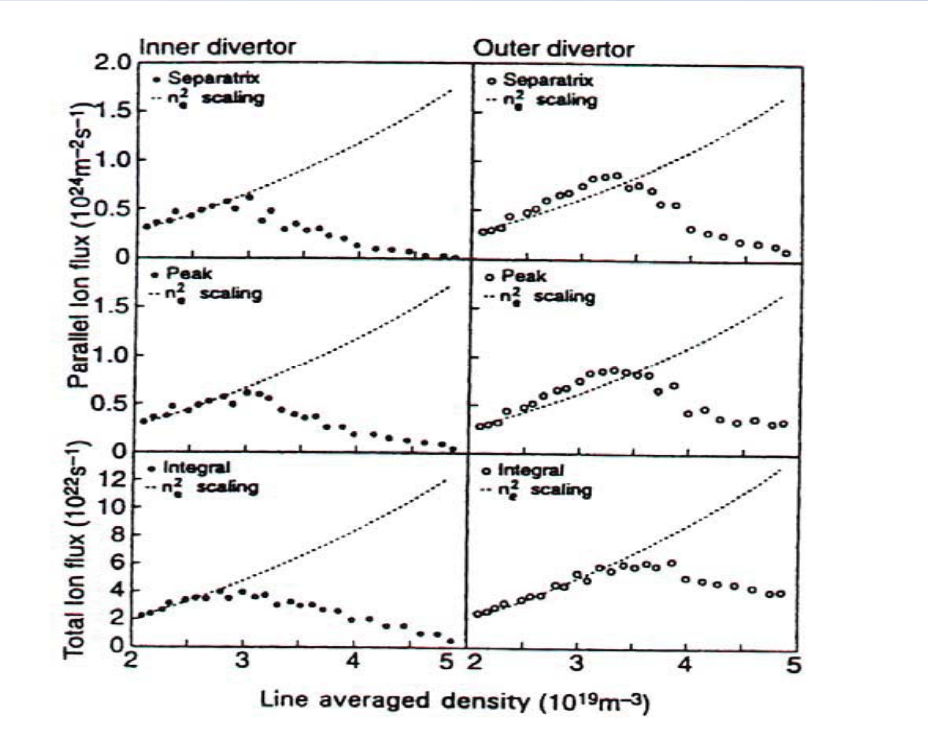
C-Mod, B. Labombard, et al.,

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

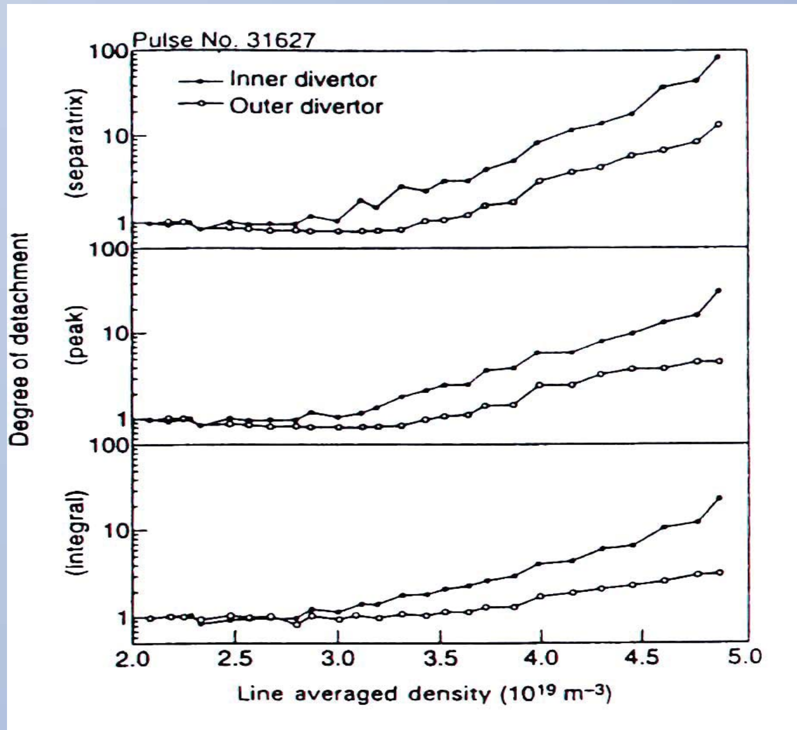
Detachment seen experimentally in many devices, but complex “volumetric” process and relative importance of ion-momentum friction vs. recombination still unclear.
X-point geometry → long connection lengths → high residence times in low T_e plasma → efficient radiative loss favouring power reductions where $q_{||}$ is highest (i.e. on flux surfaces near separatrix).

JET, (ohmic), DETACHMENT

Measured and extrapolated ion fluxes to inner and outer divertors, density ramp

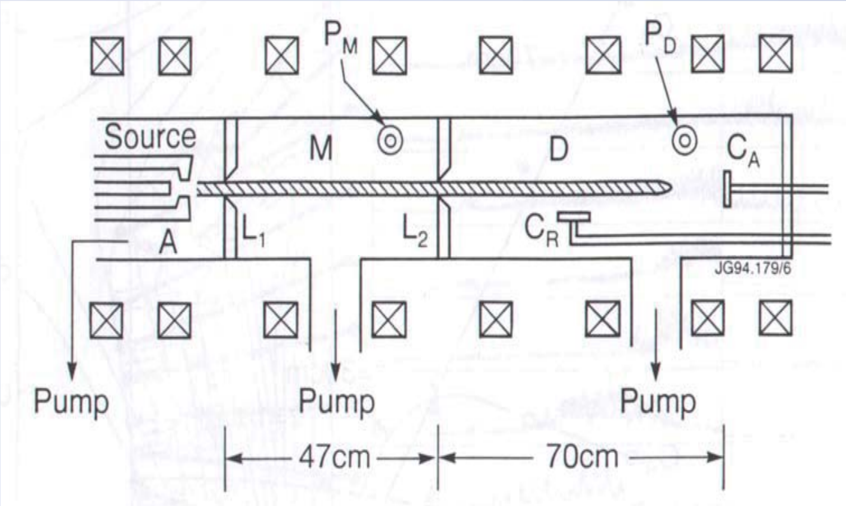


Degree of detachment (DOD)

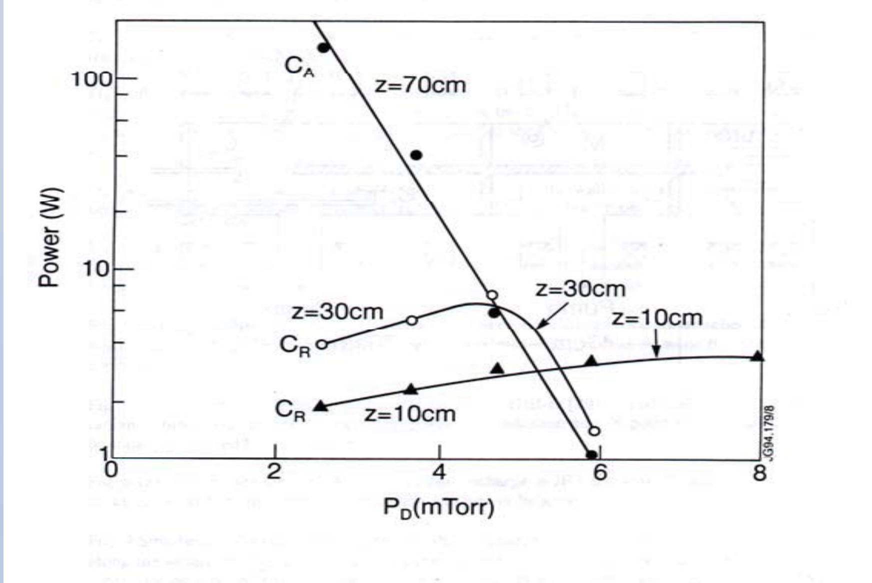


Princeton QED device
(gaseous Divertor concept simulator)

Schematic



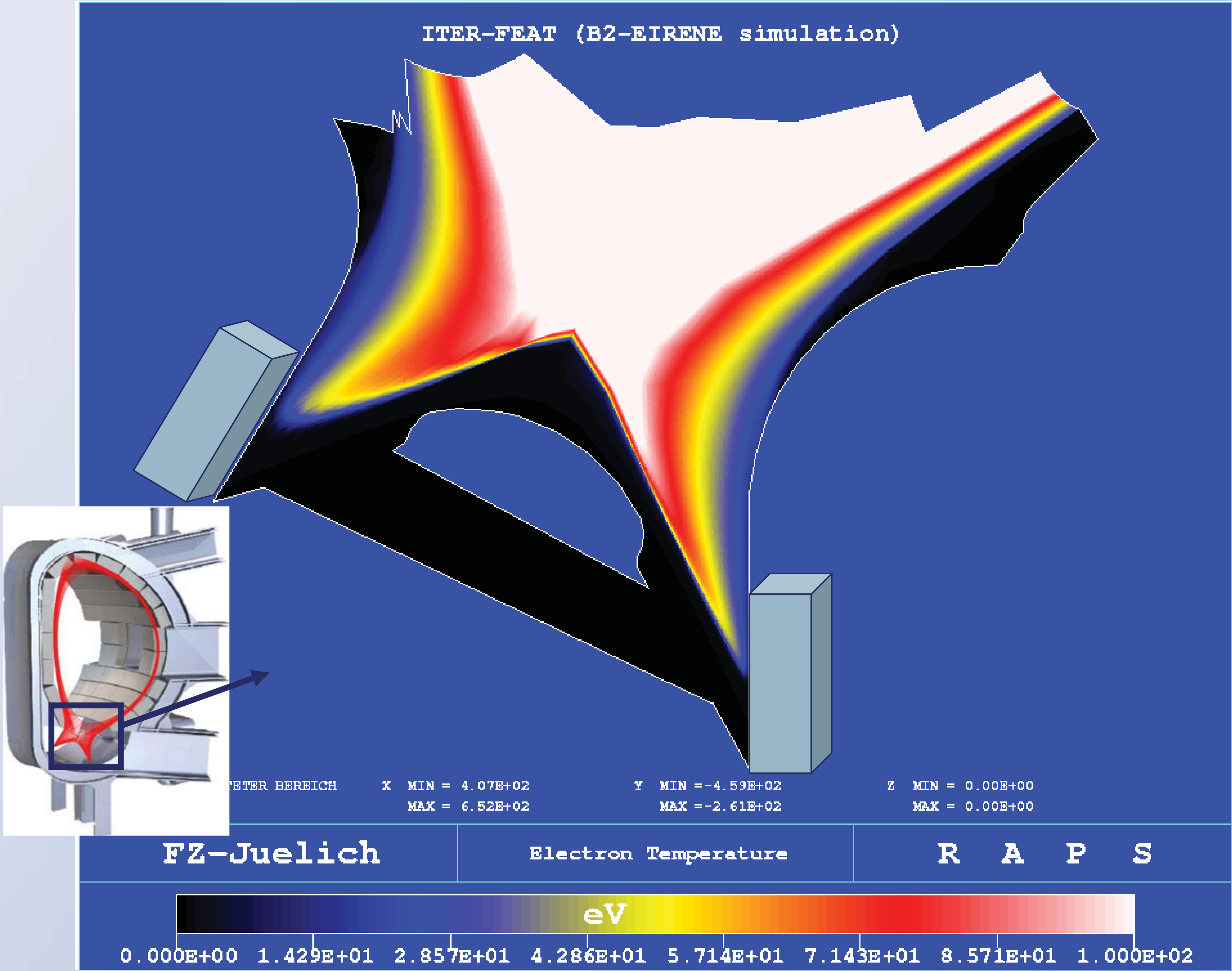
Scaling of calorimeter signals with gas pressure



Hsu et al., PRL 49, 1001 (1982): QED
Schmitz et al., J.Nucl.Mat. 196-198, (1992): PISCES
Ohno et al., PRL 81, 818 (1998): NAGDIS

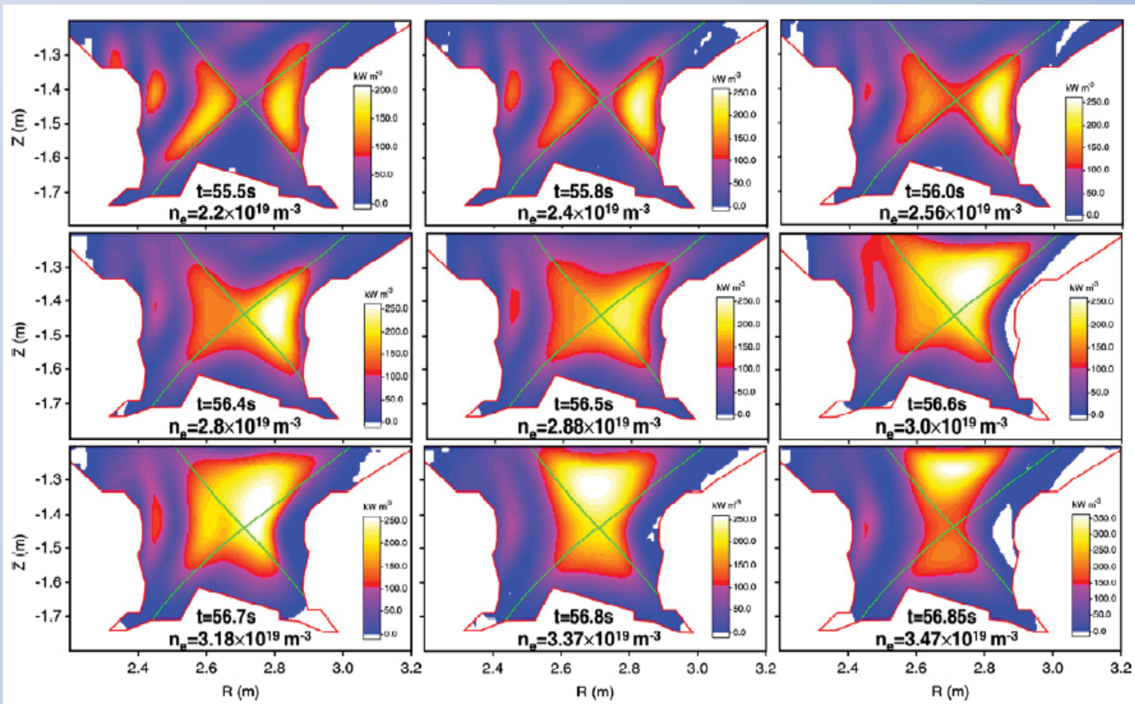
Key difference: here: P_{gas} given. In a fusion device the neutral cushion must be self sustained by recycling process. This issue will be addressed in linear MAGNUM device (FOM)

ITER, B2-EIRENE simulation, fully detached, T_e field



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



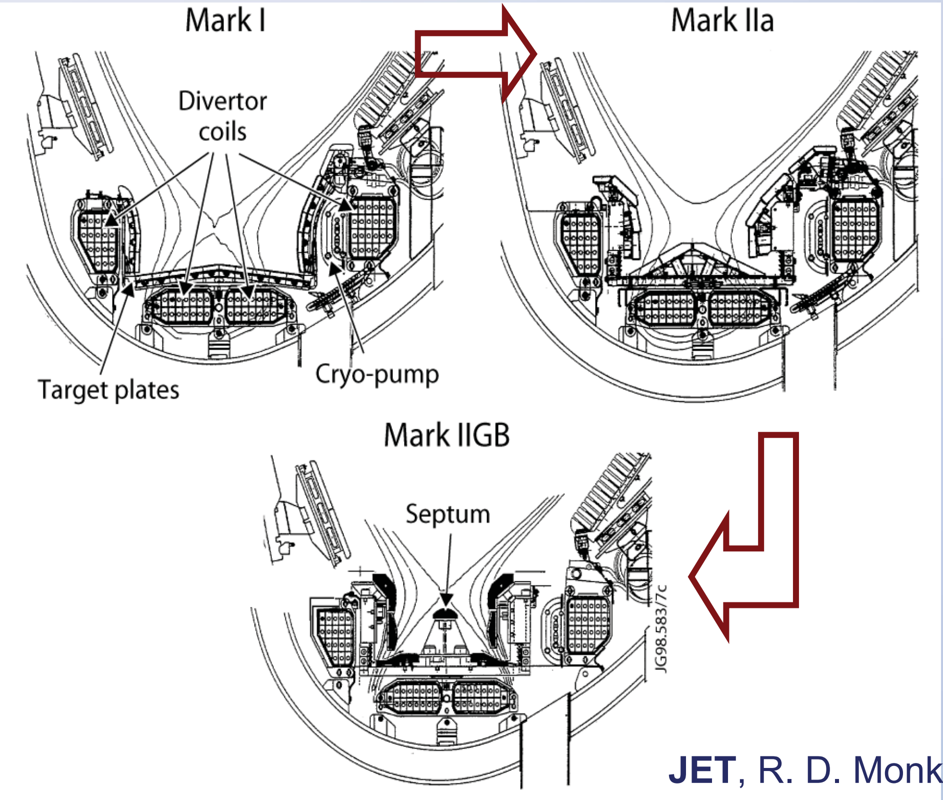
JET, A. Huber, et al.

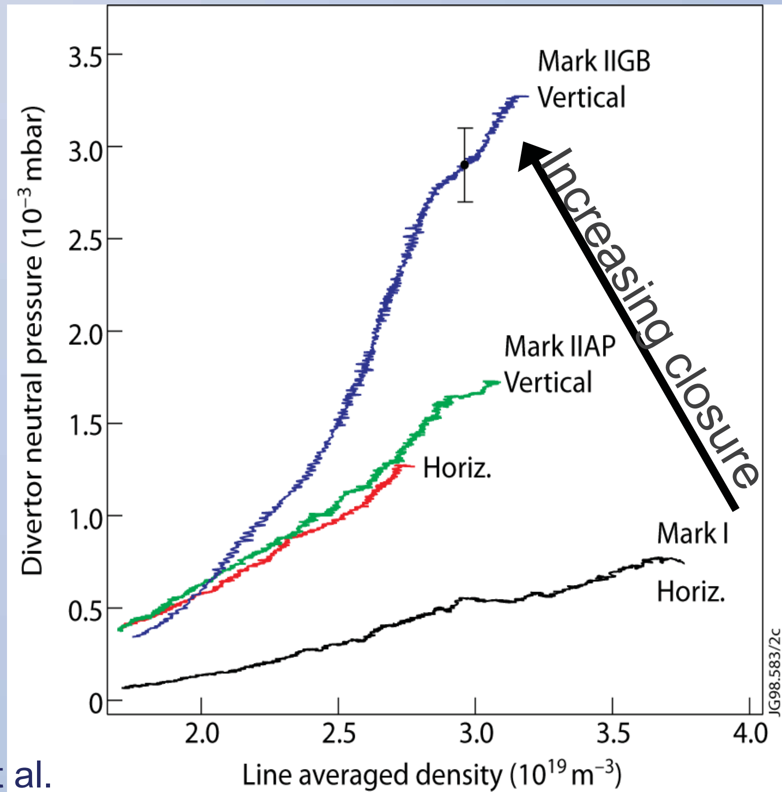
Limit detachment to regions of highest power flux (where it is needed most).
Maintain remainder of SOL in high recycling (attached)
A few ways to arrange that this happens more readily:

Divertor closure | **Target orientation** | **Impurity seeding**

Courtesy: R. Pitts

Divertor closure





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

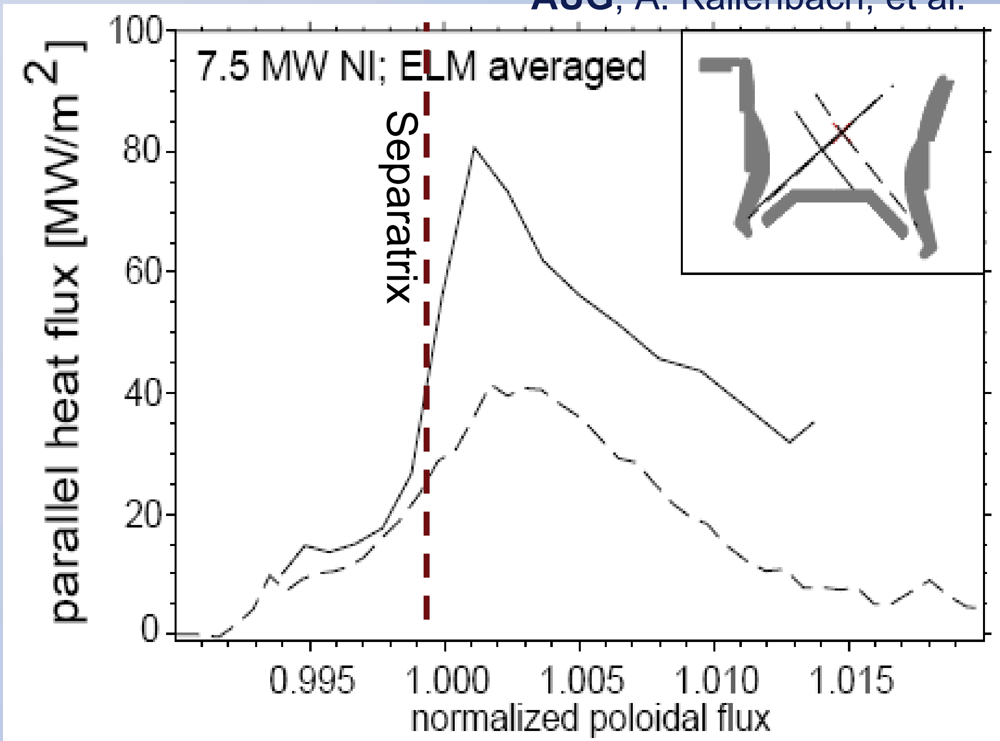
Divertor closure also promotes helium compression and exhaust – very important for ITER and reactors

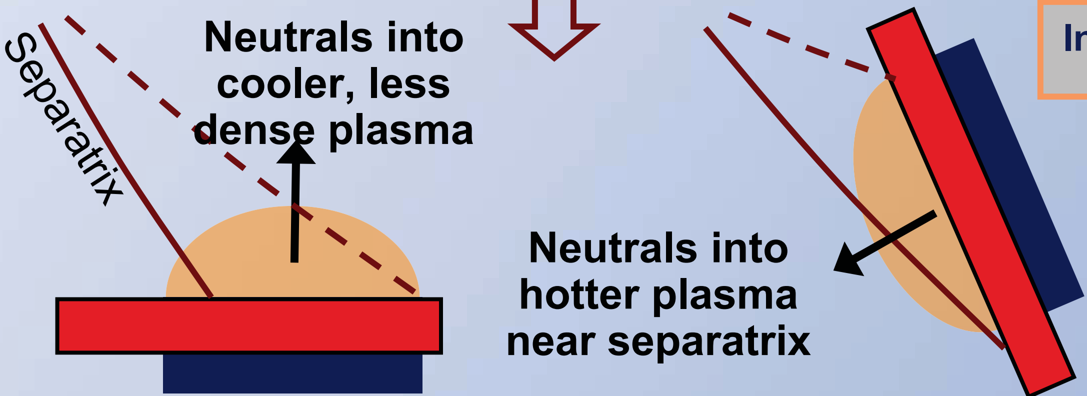
Courtesy: R. Pitts

Target orientation

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

AUG, A. Kallenbach, et al.





Increased ionisation near sep.

Higher n_t, lower T_t

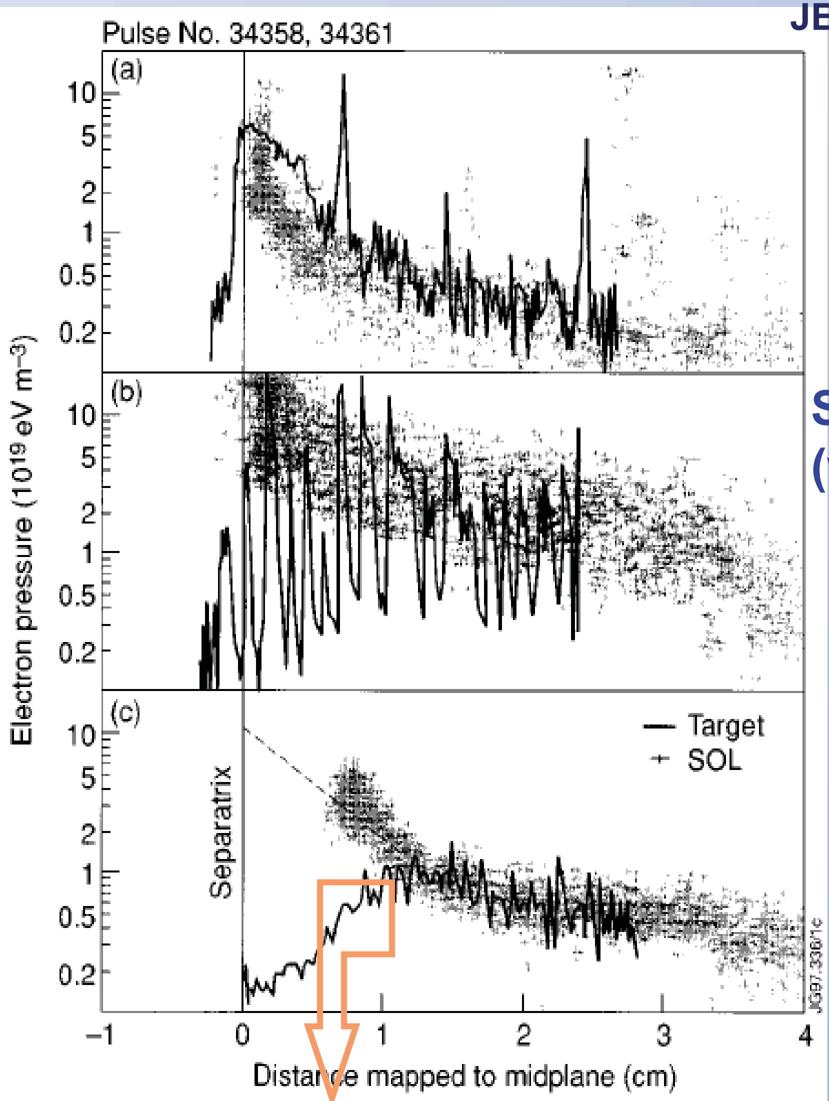
Higher CX losses

Pressure loss → $q_{||}$ ↓

Courtesy: R. Pitts

Impurity seeding

JET, G. F. Matthews et al.



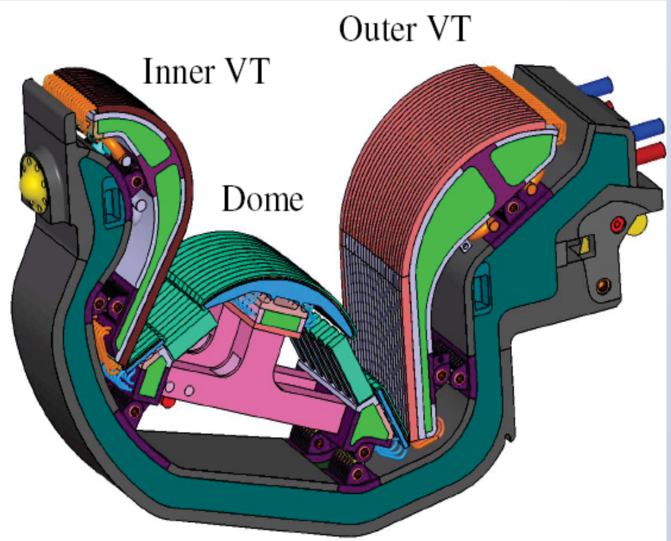
Unfuelled (attached)

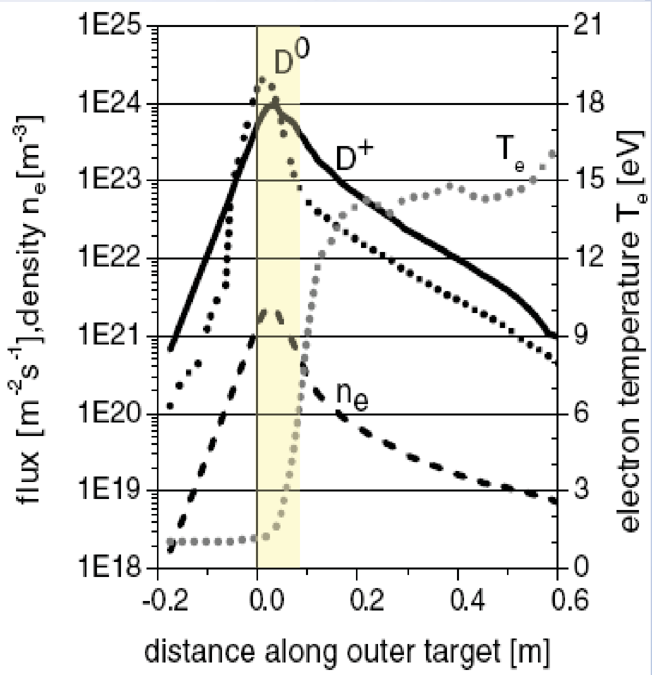
Strong D_2 puff (weak detachment)

Strong D_2 +N_2 puff (stronger detachm.)

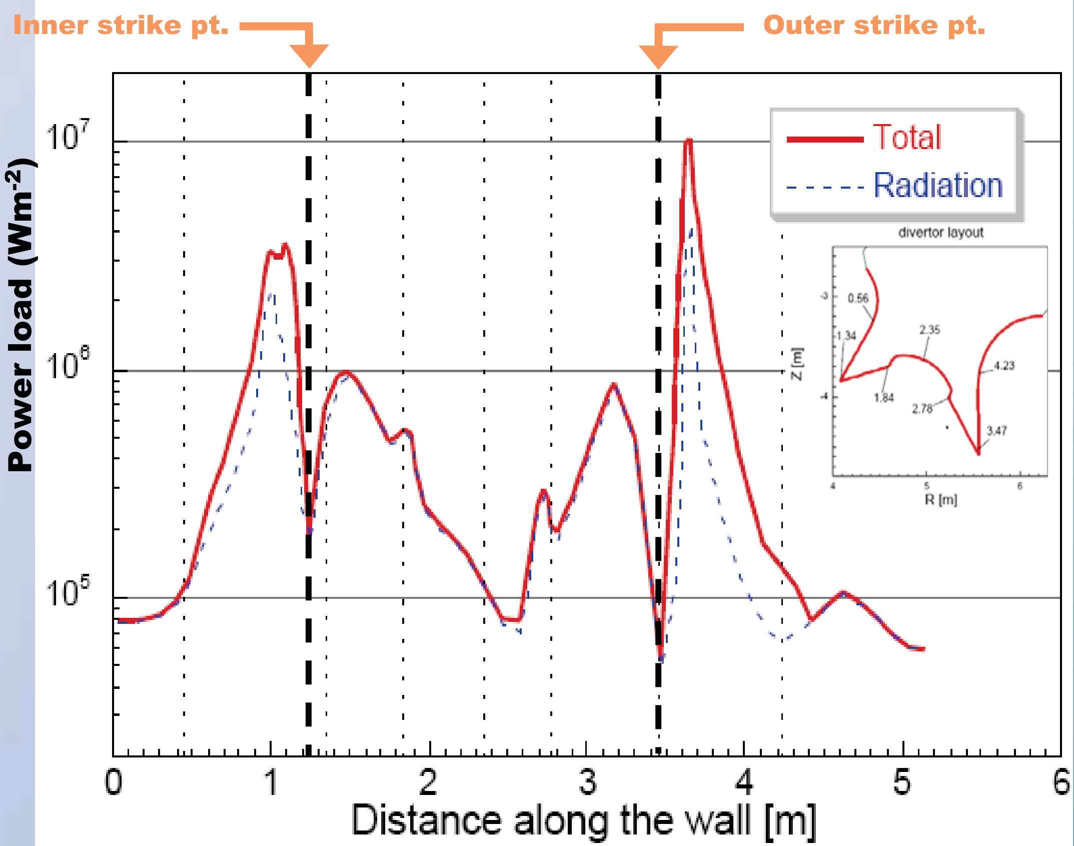
Strong impurity seeding also reduces ELM size but high price can be paid in confinement

ITER divertor designed to achieve partial detachment





Courtesy: R. Pitts

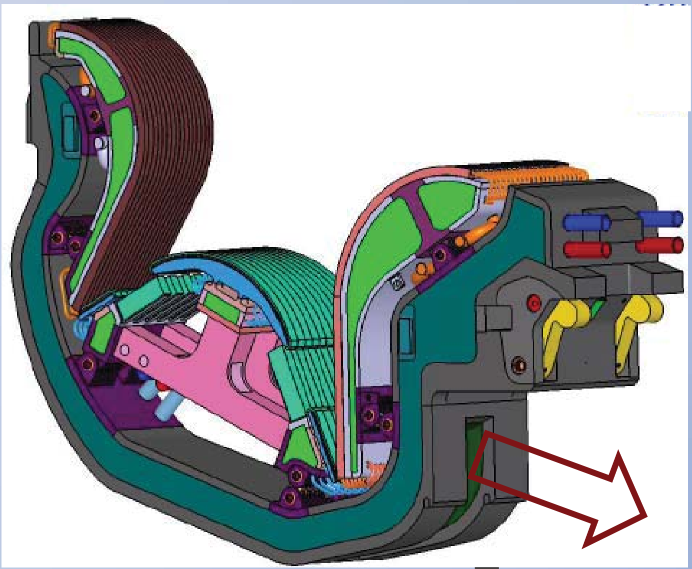


ITER Divertor DDD 17, Case 489 (SOLPS4.2 runs by A. Kukushkin)

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

Divertor exhaust

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



To cryopumps

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

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

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

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

e.g. ITER: He prod. rate ~$2\times10^{20}s^{-1}$
Max. divertor pumping speed ~200 Pa m^3s^{-1} ~ 1×10^{23} He atom s^{-1}
→ C_{pump} ~ 2×10^{-3} = 0.2%
Typical acceptable He conc. in the core: ~4% → η_{He} = 0.2/4 = 0.05 is minimum required. The values of $\tau^*_{p,He}$ and η_{He} required for ITER have been achieved experimentally

Courtesy: R. Pitts

The JET divertor design philosophy

Michael Pick has used to describe the design of the JET divertor:

"The only way to do research is to tell the complete truth. And the truth is that research is often based partially on intuition, which is a perfectly acceptable basis for research in the face of a lack of evidence and verified predictive models.

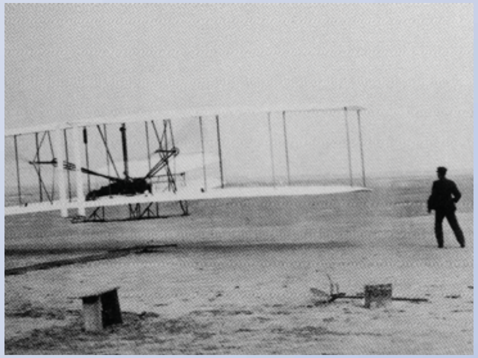
We built the divertor based on what we thought would be a reasonable solution, based on simple extrapolation, models and intuition, leaving open the possiblities to change."

Still true for ITER, despite significant progress in edge plasma science and predictive quality of models
See lecture III

- One and a half decade ago we lacked a credible solution to the divertor problem.

- With the discovery of the cold, detached, radiating divertor in the 1990s, we now have (the makings of) a divertor solution for high power magnetic confinement devices.

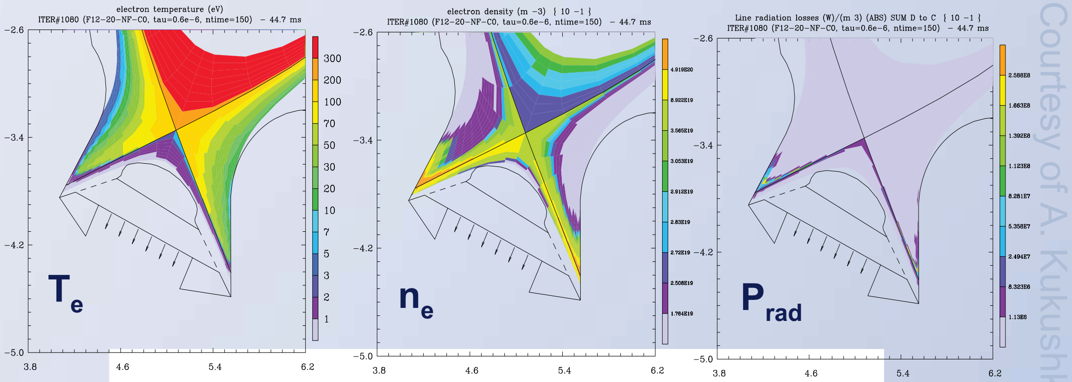
We now have enough understanding of „WHAT“ (JET, Tore-Supra, D-IIID, ASDEX, LHD, W7AS,…..) to proceed with the „HOW“ (to build ITER,…)
Very little on the „WHY“ question still, see lecture III
But we are ready to go: Bring on ITER!

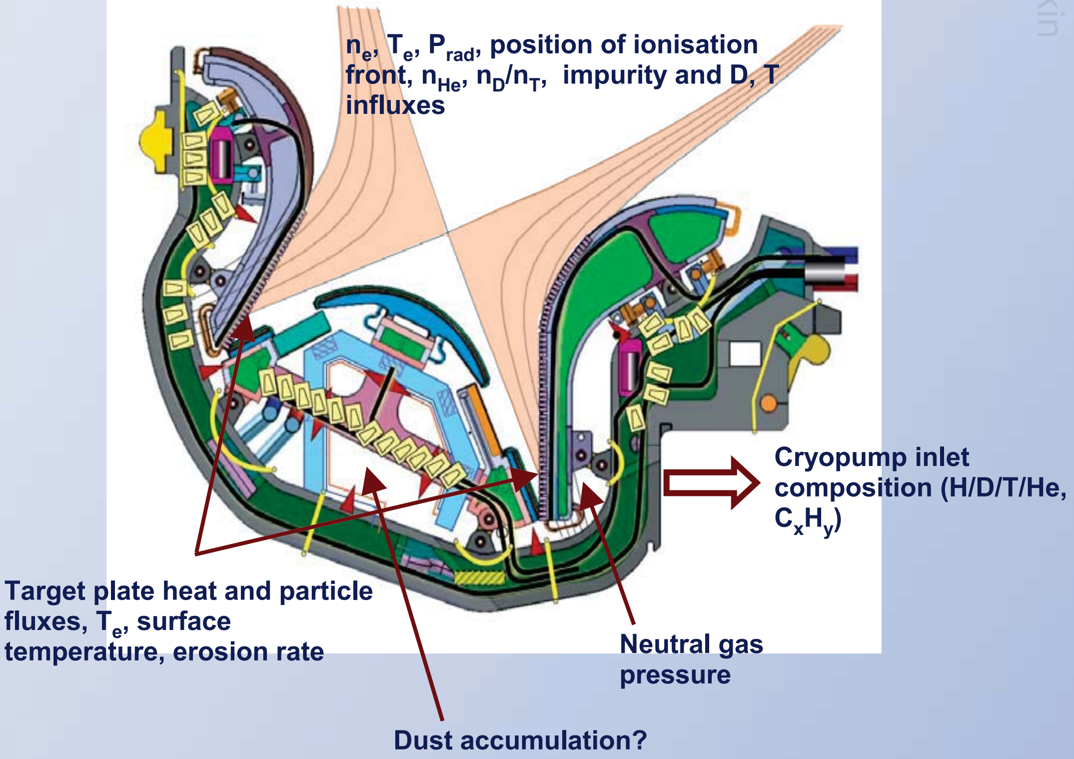


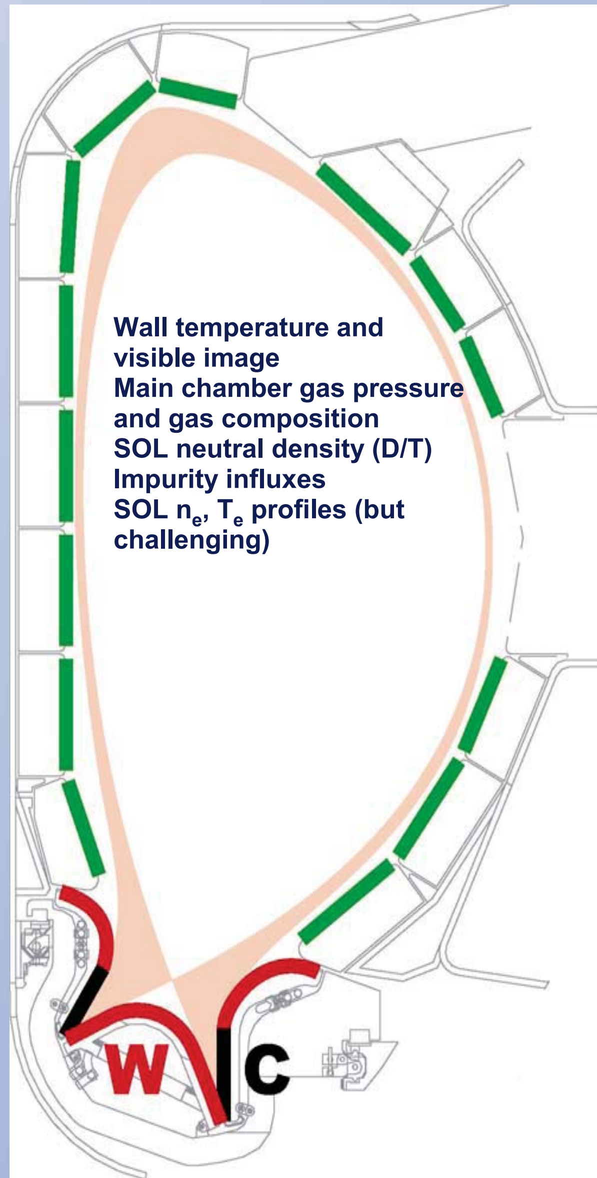
Compare to similar situation after first flight of Wright brothers

Reserve slides

Edge Diagnostics on ITER will be critical







“Mission statement” for this talk ...

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

Progress in the ITER Physics Basis, Chap. 4: “Power and particle control”,

Nucl. Fusion 47 (2007) S203-S263