EFDA EUROPEAN FUSION DEVELOPMENT AGREEMENT

The Plasma Boundary

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EFDA CSU-Garching

with major contributions from

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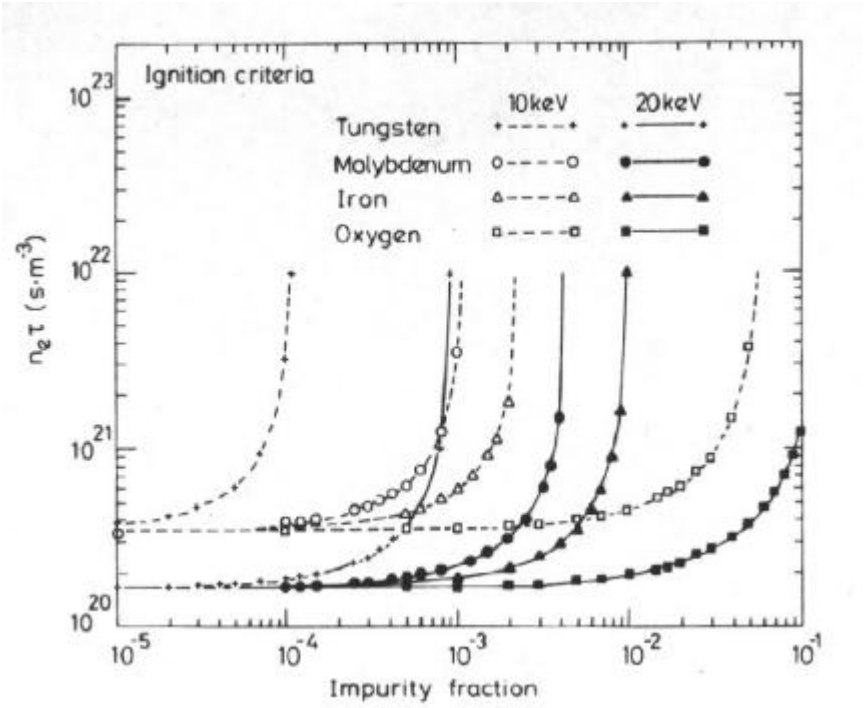
- **the problem and the physics**
- **a strategy for control**
- **edge effect on the core**
- **plasma boundary in ITER and a power plant**

the plasma boundary and impurities

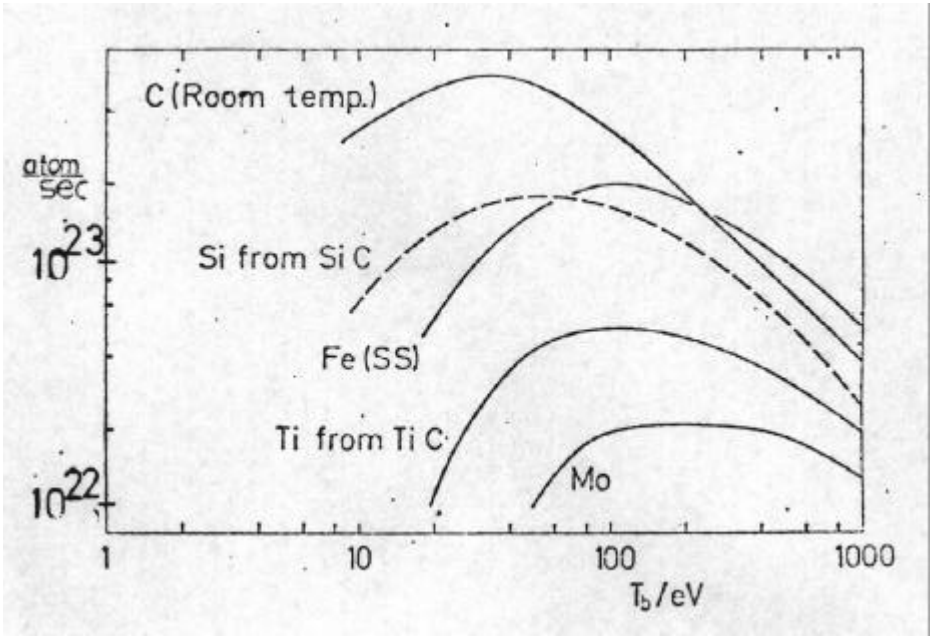
a burning plasma can tolerate only small amount of impurity radiation (particularly high Z)*)

energy and particle flow to wall produces high impurity release rate (particularly low Z)

PPPL



IPP



example for 3 GW thermal reactor,

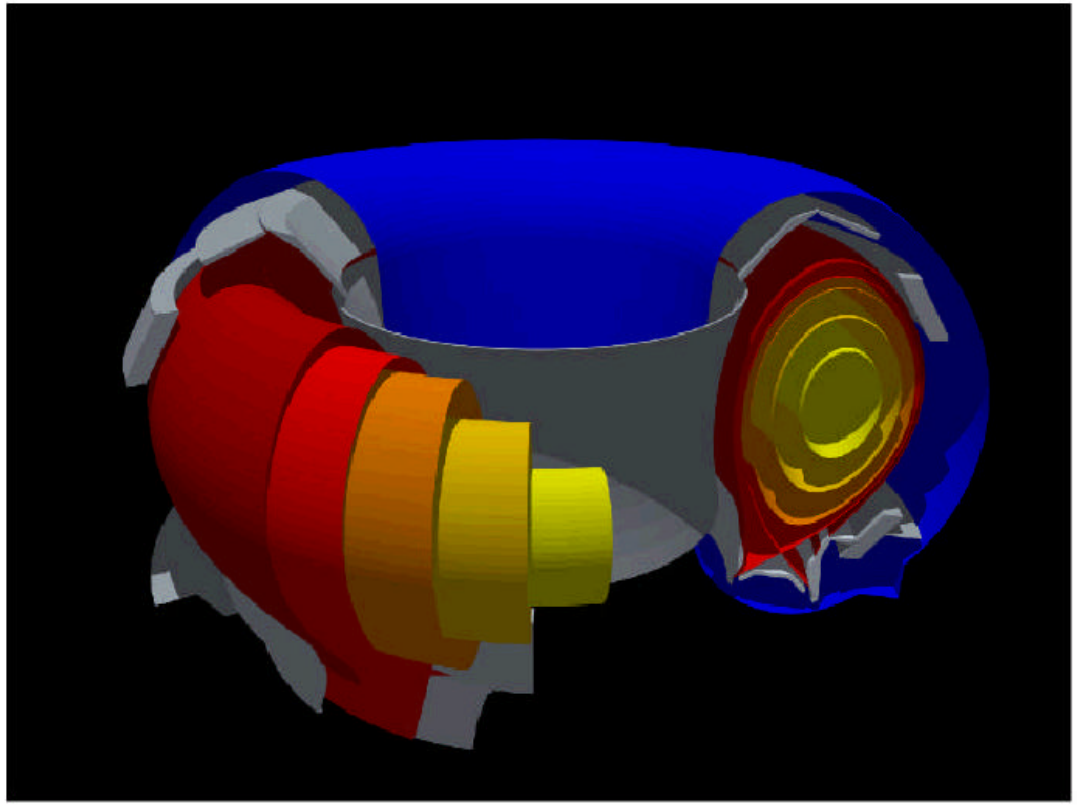
implies $t_E / t_{imp,eff} > 100$

***) low-Z impurities lead to strong fuel dilution**

origin of problem

(1) high single particle energy

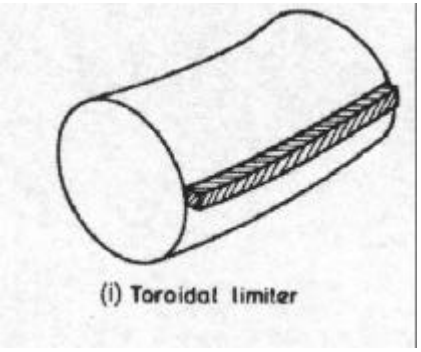
(2) high power flux at contact curve: wall – magnetic surface due to transport anisotropy*)

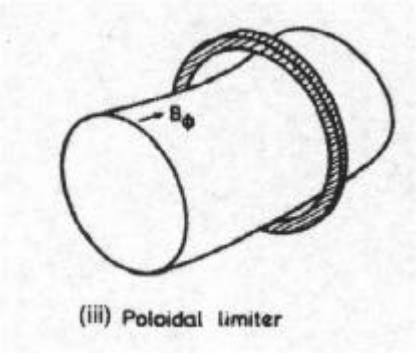


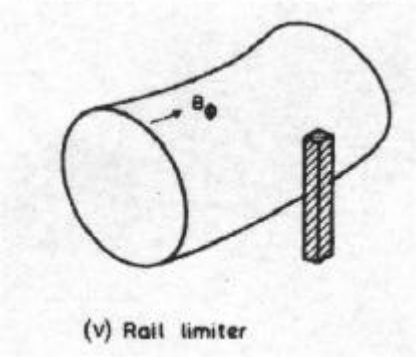
*) in spite of modest energy density

topological alternatives for controlling plasma wall contact

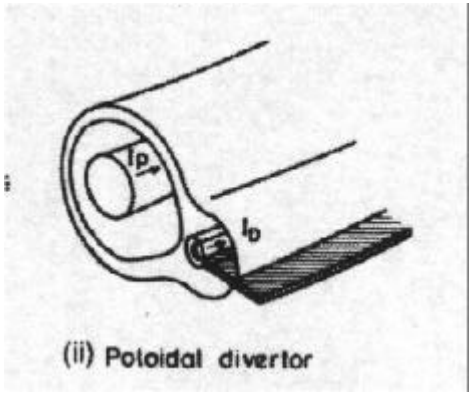
Limiters

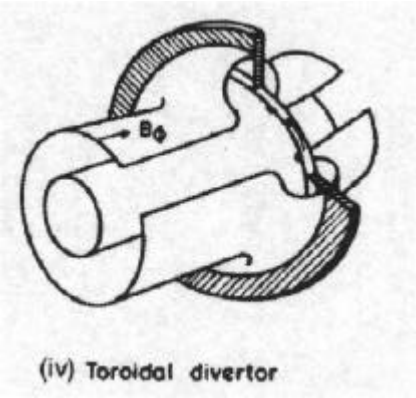


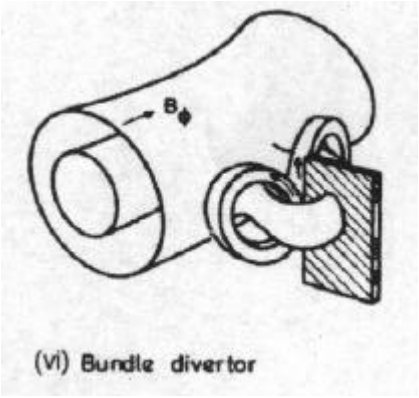




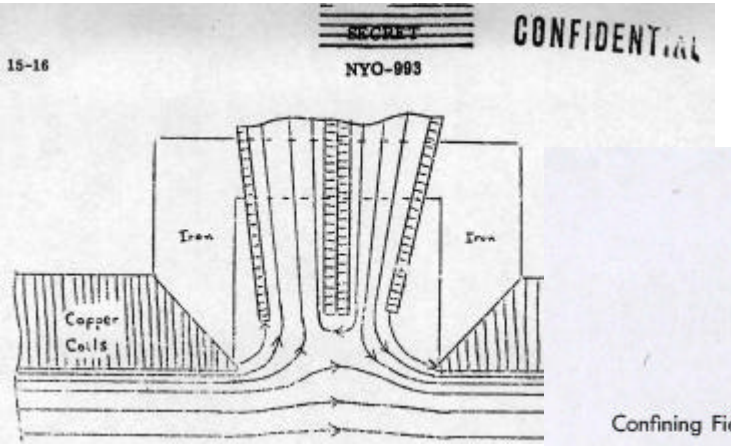
Divertors



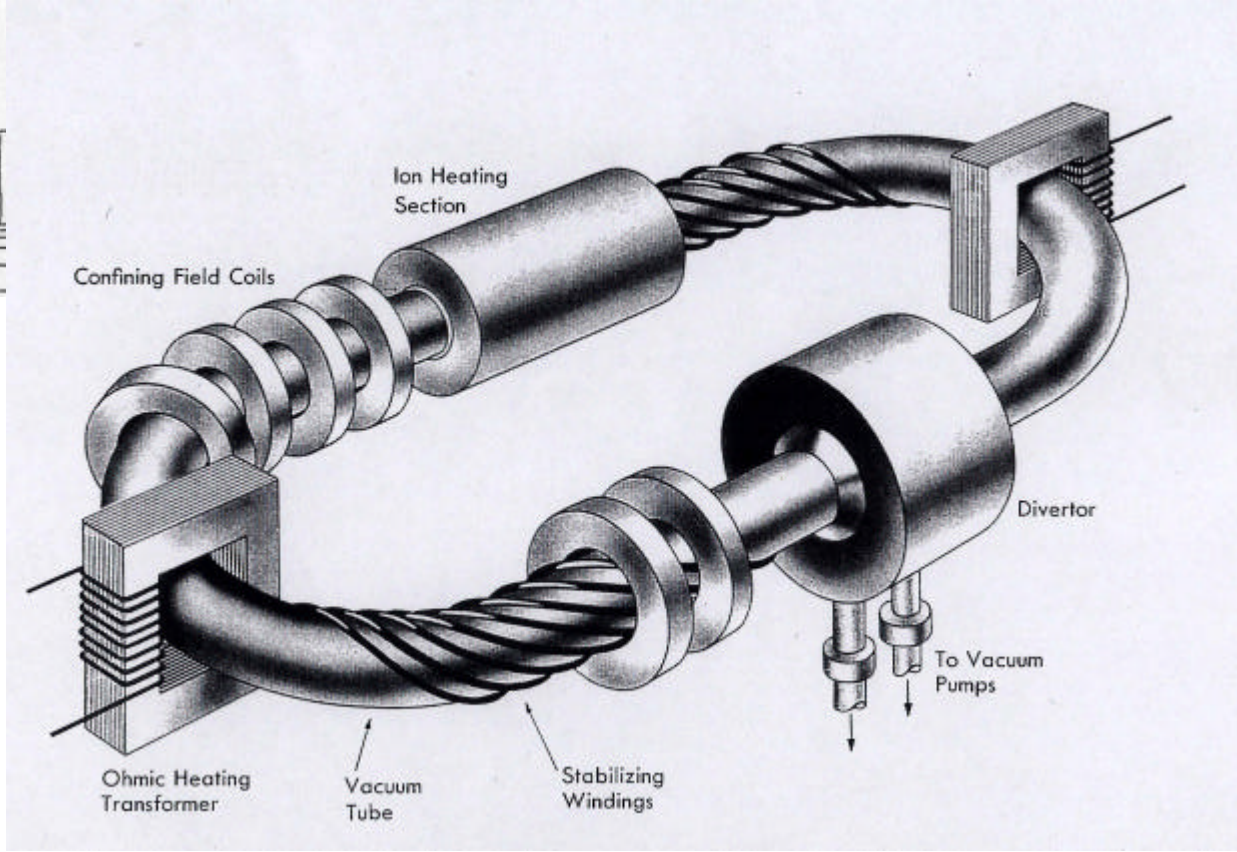




removal of plasma-wall interaction from core plasma vicinity – the divertor idea



L. Spitzer, 1951



Model-C Stellarator

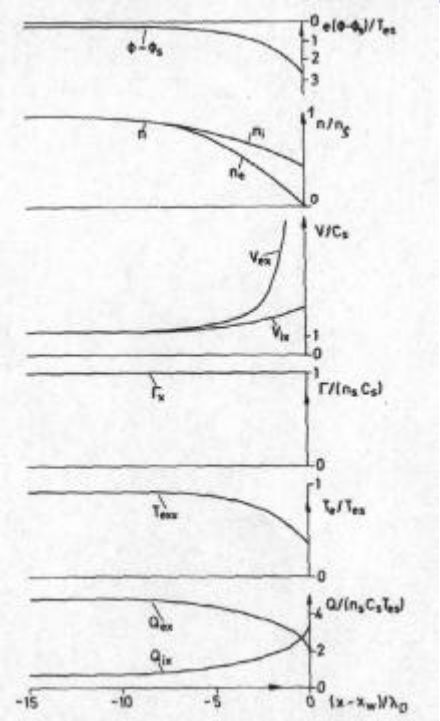
the physics:

the sheath in front of material walls

without (or ⊥) B-field: the Bohm sheath on scale λ_D

entrance condition $\mathbf{v}_{-\infty l} \cdot \mathbf{n} \geq c_s$

sheat potential drop $\Delta\Phi_s = -g_s kT, g_s = 3.18$ (for $D^+, T_i = 0, j = 0$, no secondary emission.)

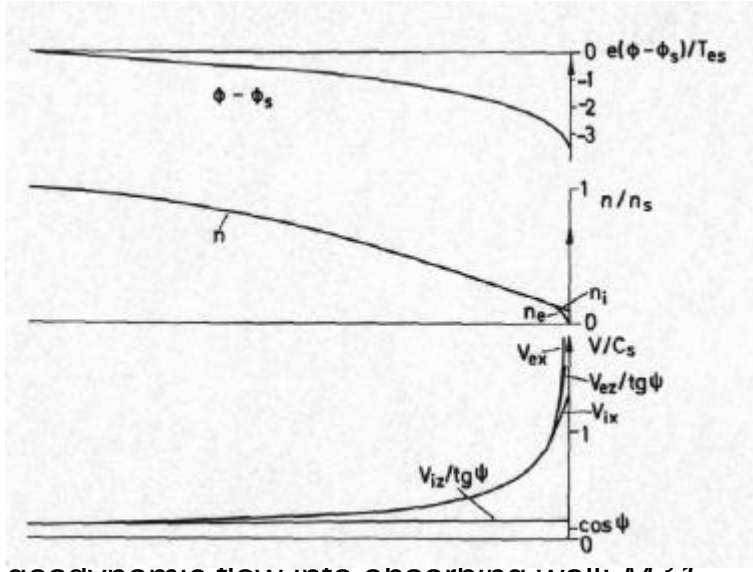


with finite B ‘ : presheath on scale $c_s/\omega_{c,i}$ (Chodura)

$\mathbf{v}_{-\infty r} \cdot \mathbf{b} \geq c_s$

on scale of λ_D, like Bohm $\mathbf{v}_{-\infty l} \cdot \mathbf{n} \geq c_s$

sheath potential drop on two scales:



equivalent to gasdynamic flow into absorbing wall: $M \geq 1$

(complication with impurities: several sound speeds)

the physics:

consequences of the Bohm/Chodura sheath

limitation of energy flux to walls:

$$Q_w = (5.2 + M^2/2) \cdot kT_e \cdot \Gamma_W$$

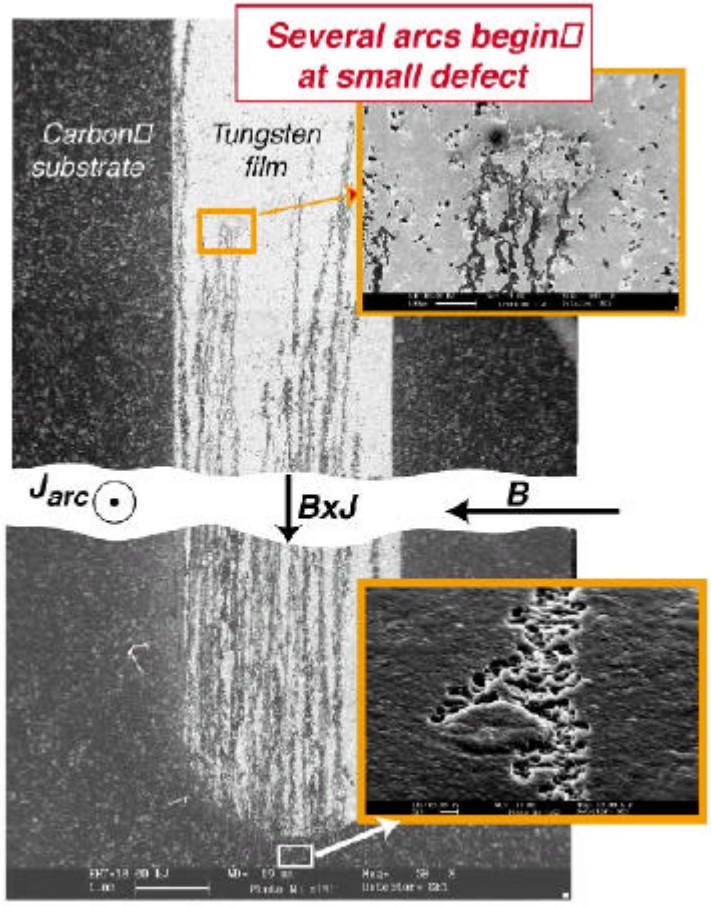
(for conditions on prev. slide)

acceleration of ions (impurities in sheath):

$$\Delta E_{Z,s} = Z \cdot g_s \cdot kT_e$$

possibility of unipolar arc formation

short (< 1 ms), localised (d » 10 µm), high current density (10^{12} A/m^2) discharges across sheath



controlling plasma-wall interaction:

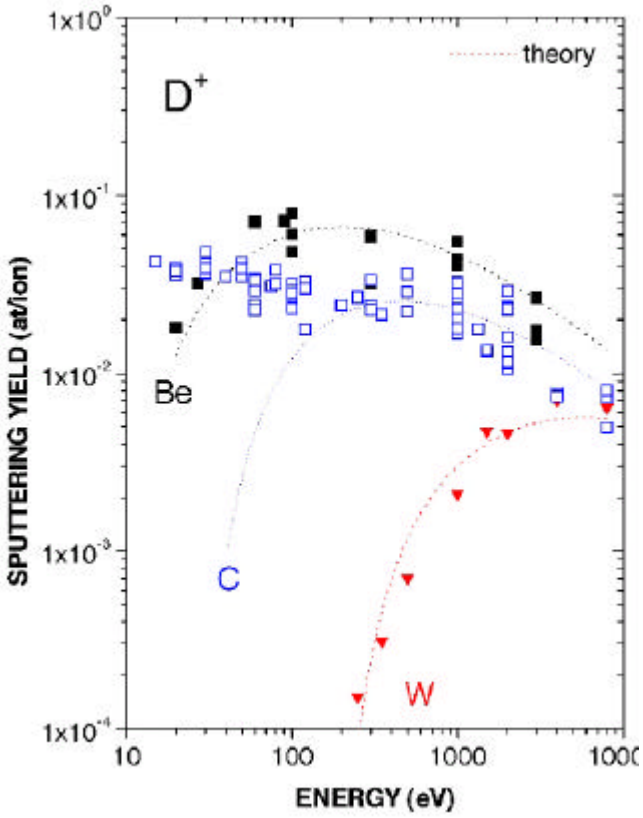
the reaction of the solid: complex and important

physical sputtering:

threshold behaviour

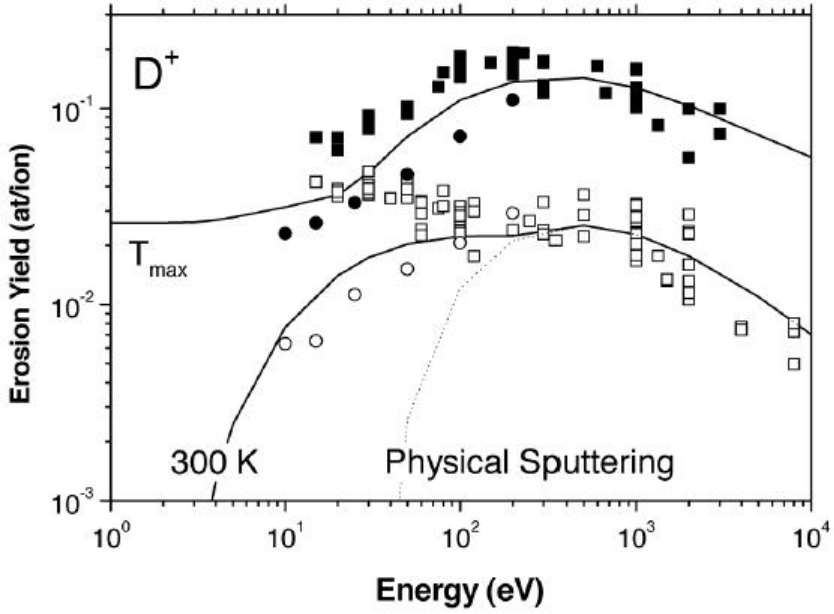
dependence on projectile mass

low for D,T on W



for carbon based materials:

- enhancement due to chemical effects
- chemical sputtering: flux and target temperature dependent



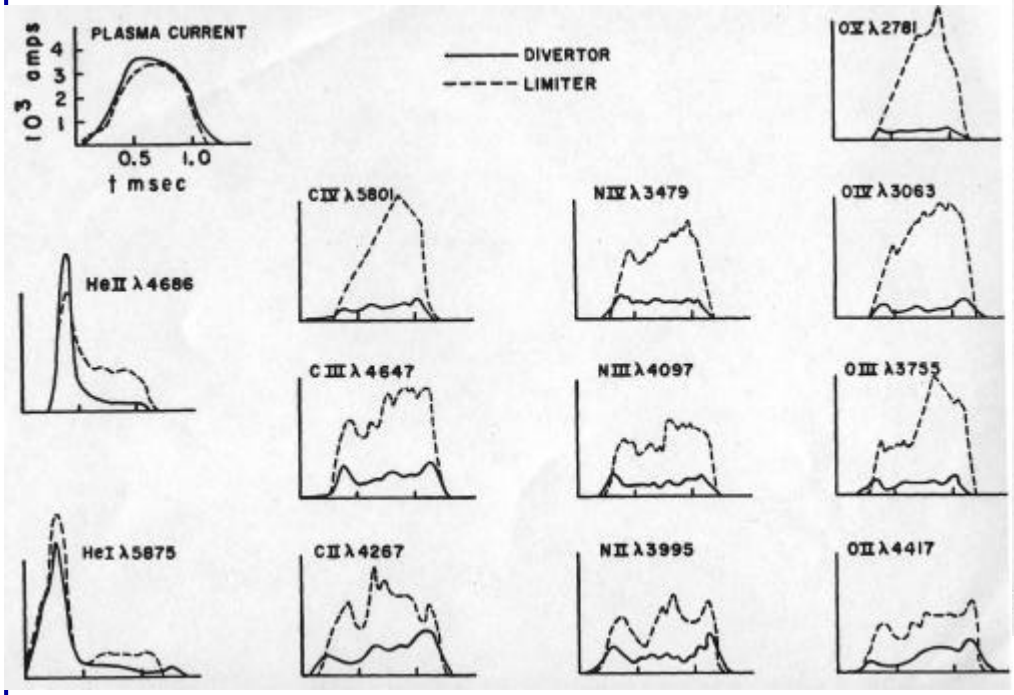
the trade-off: low vs. high Z-materials

material	pro	con
CFC	• low core radiation • good power handling • does not melt	• high chemical erosion, even at low plasma temperatures • plasma dilution • co-deposition with tritium
beryllium	• low core radiation • oxygen gettering	• poor power handling (melting) • high physical sputtering • plasma dilution
tungsten	• low sputtering yield at low plasma temperatures • no tritium co-deposition	• high core radiation • reaction to anormal events

strategy for controlling plasma-wall interaction:

(1) displacing it into Divertor: B-65 and C-Stellarators

oxygen content reduced by factor of 50 in C-stellarator



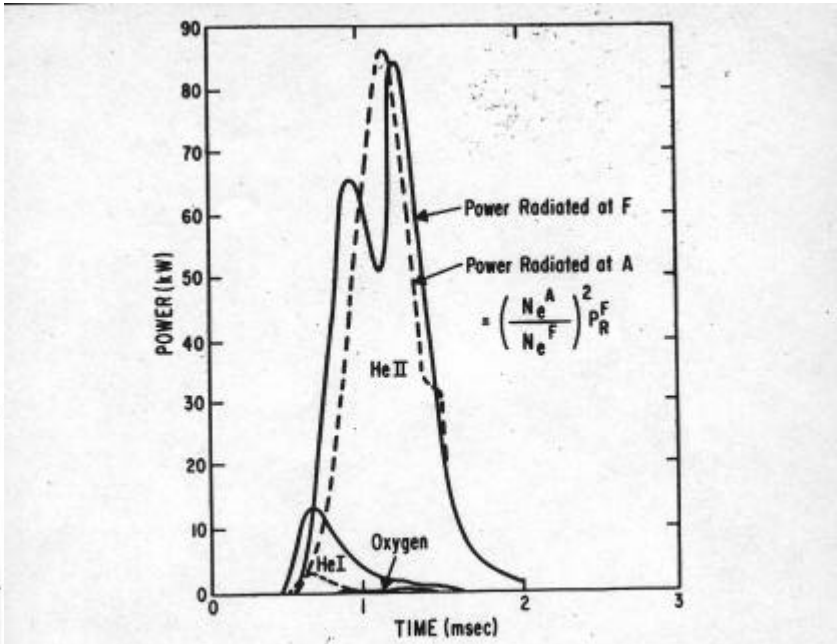


FIG. 6. Power radiated in a 4-kA helium discharge, as measured at port-cross F and computed for port-cross A.

B65: radiation emission with/without divertor coils activated

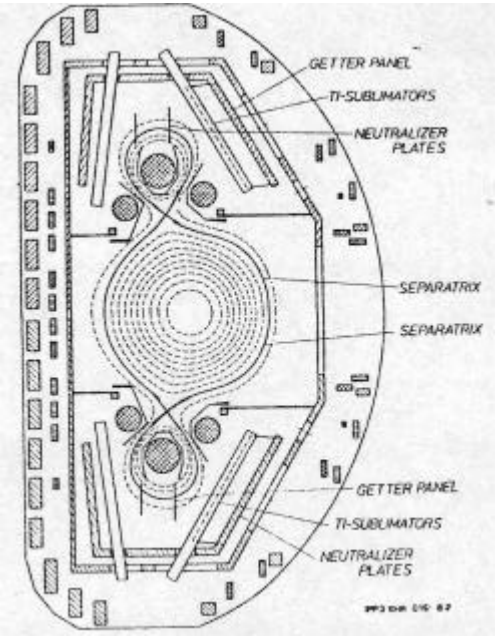
without divertor: O-radiation comparable to He

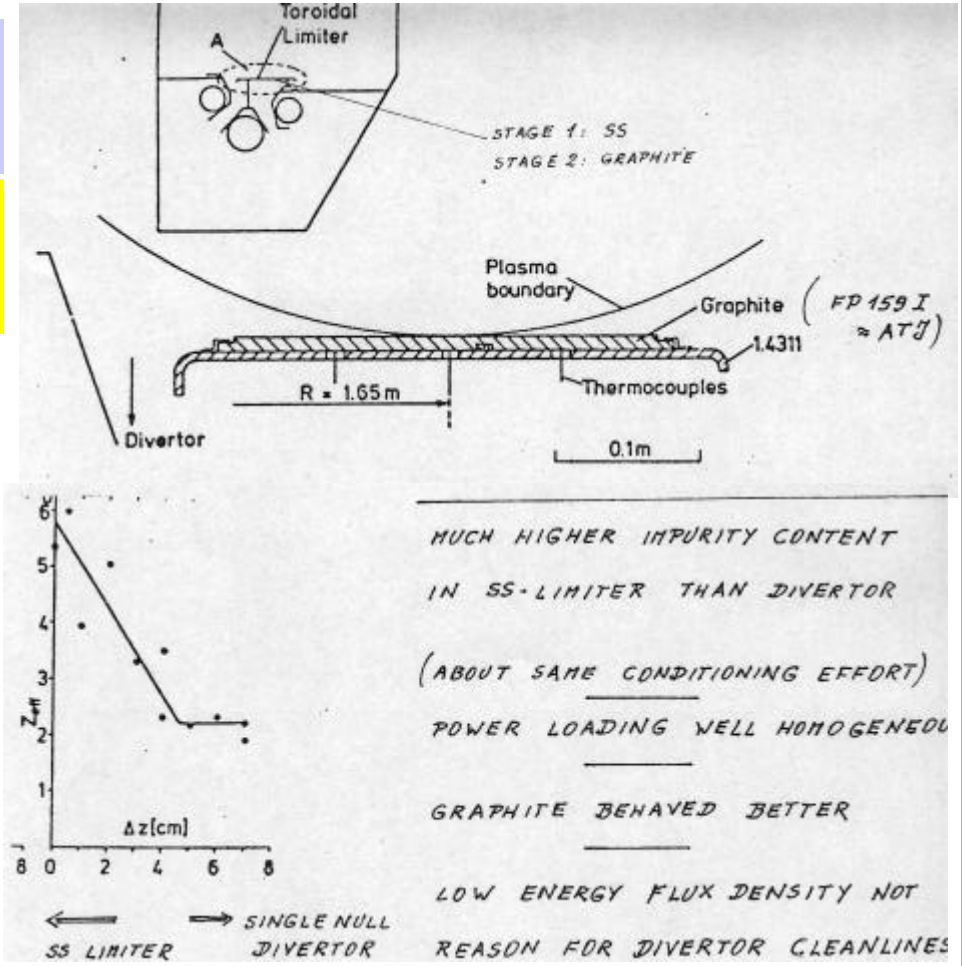
strategy for controlling plasma-wall interaction:

(1) displacing it into Divertor:

good impurity control has been achieved also in limiter tokamaks, with careful controll of geometry, limiter & wall conditioning (TFTR)

divertor: more robust to material (Ti, Cu on ASDEX, W on AUG) and to target plate conditioning





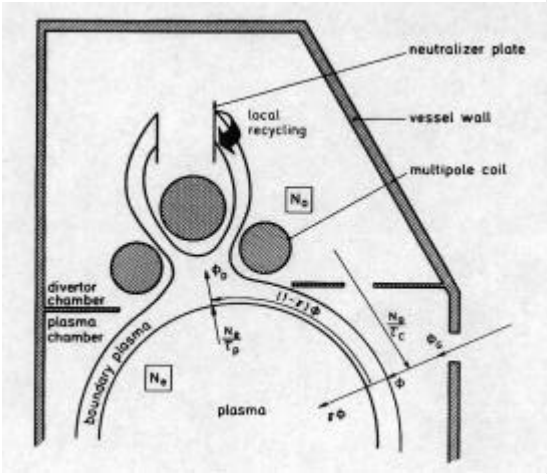
toroidal limiter – poloidal divertor comparison on ASDEX 1982

strategy for controlling plasma-wall interaction:

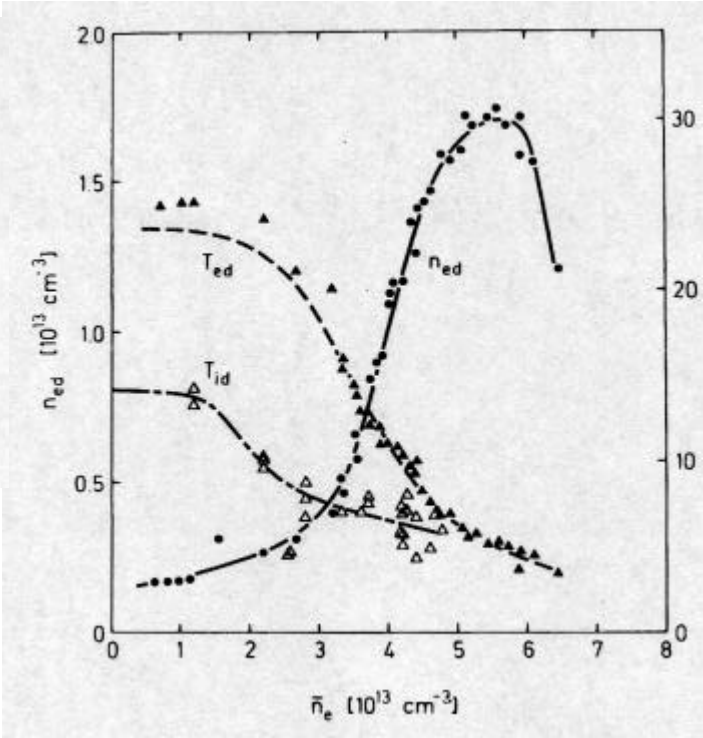
(2) reduce $T_{e,DIV}$ by high divertor recycling

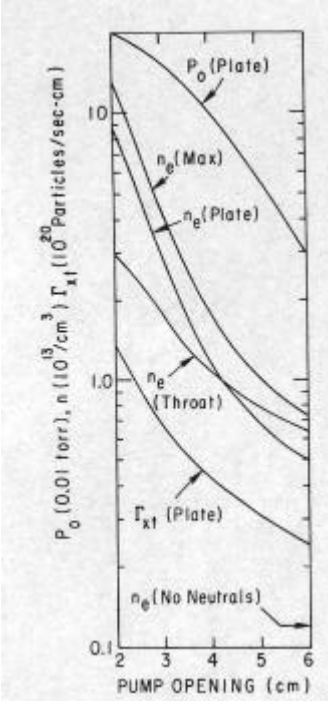
- favoured by closed geometry
- high plasma density

difficult in limiter due to pumping by bulk plasma



ASDEX: divertor geometry and density scan at constant input power (see also - unrecognized - detachment)





PDX: modelled dependence of divertor parameters on throat width

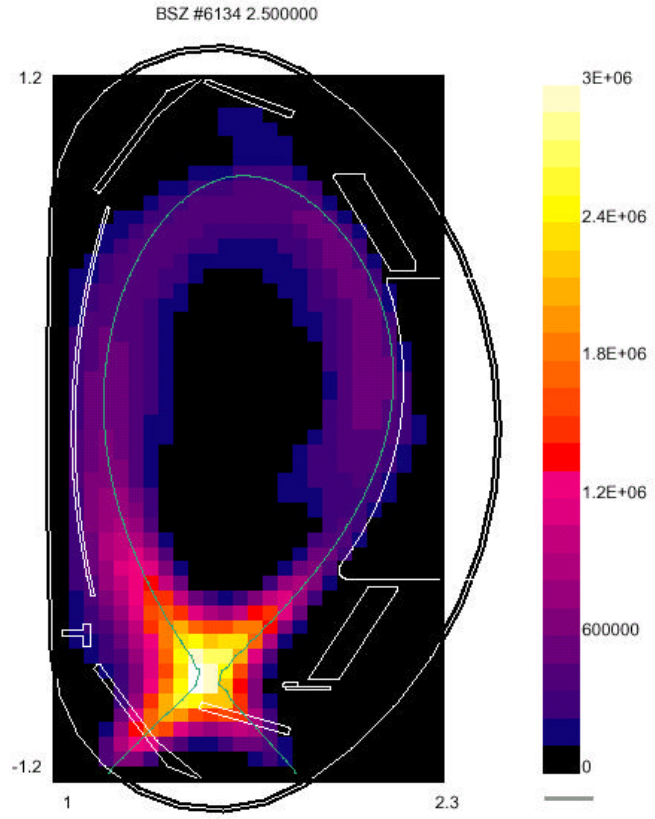
strategy for controlling plasma-wall interaction:

(3) convert conduction/convection power into radiation

for reactor requirements shear power load untolerable (in MW/m^2)

unmitigated	demonstrated	tolerable	desirable
60	20	10	5

- seeding impurities radiating from edge
- relying on self-regulation of „carbon cycle“



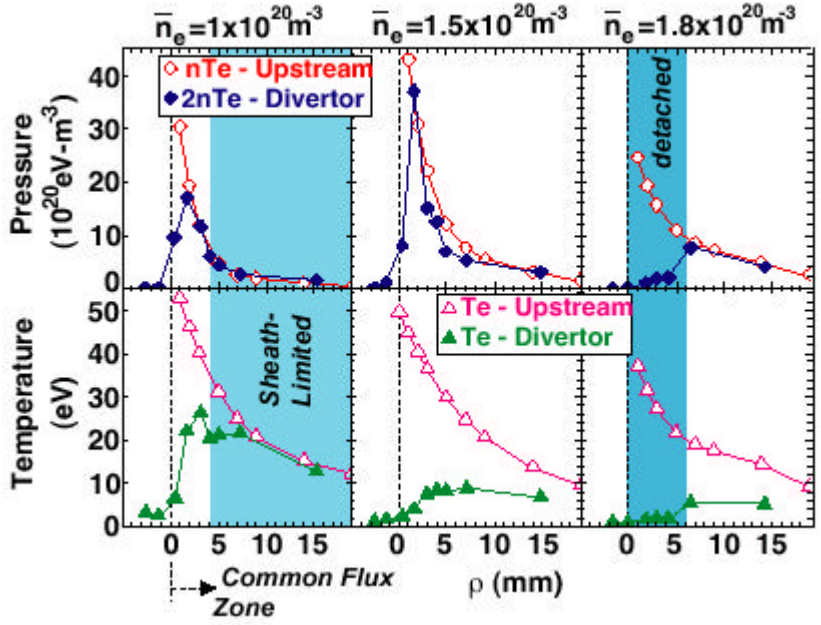
radiation emission from ASDEX-Up

strategy for controlling plasma-wall interaction:

(4) (partial) detachment: when plasma pressure taken up by neutrals

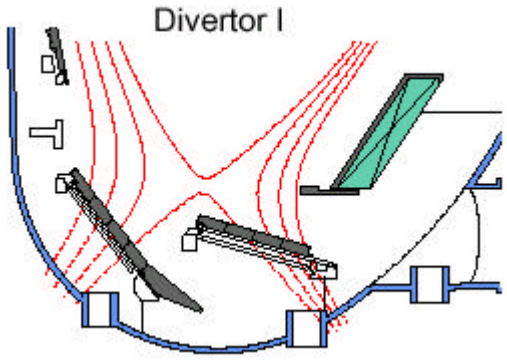
at low residual conducted power and high density,

recombination and CX remove plasma momentum (pressure)

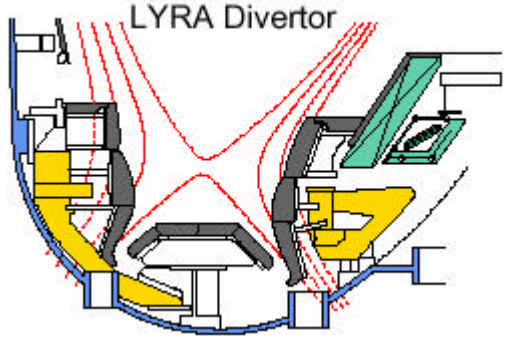


Alcator C-mod: midplane and divertor parameters

favoured by closed geometry or target plate shaping



detached only on inside leg

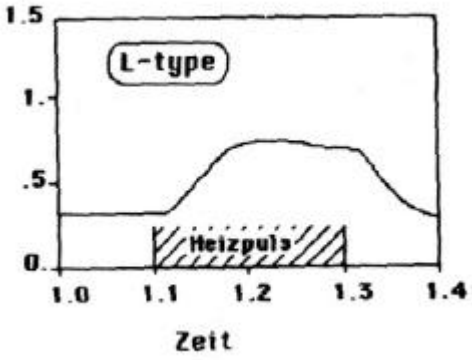


detached readily on both legs

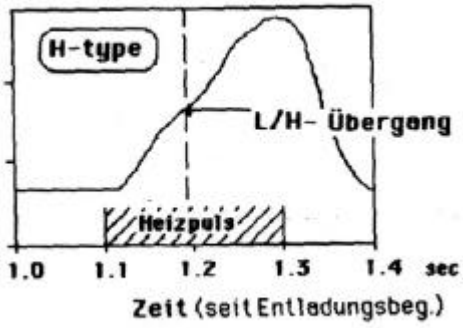
the unexpected side of plasma boundary physics

or: the tail wags the dog!

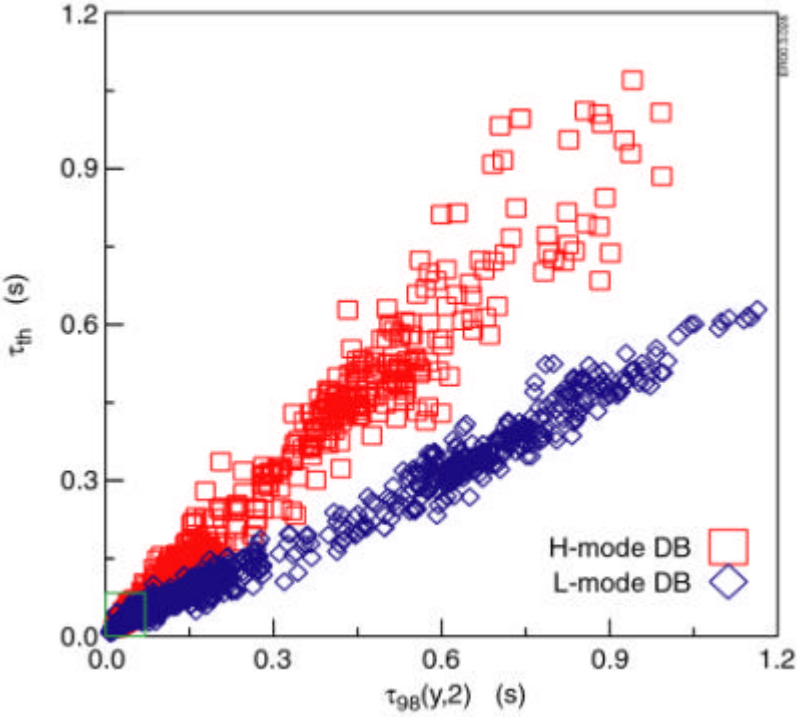
consequences on global confinement beyond those via impurity balance



16a



16b



historic questions:

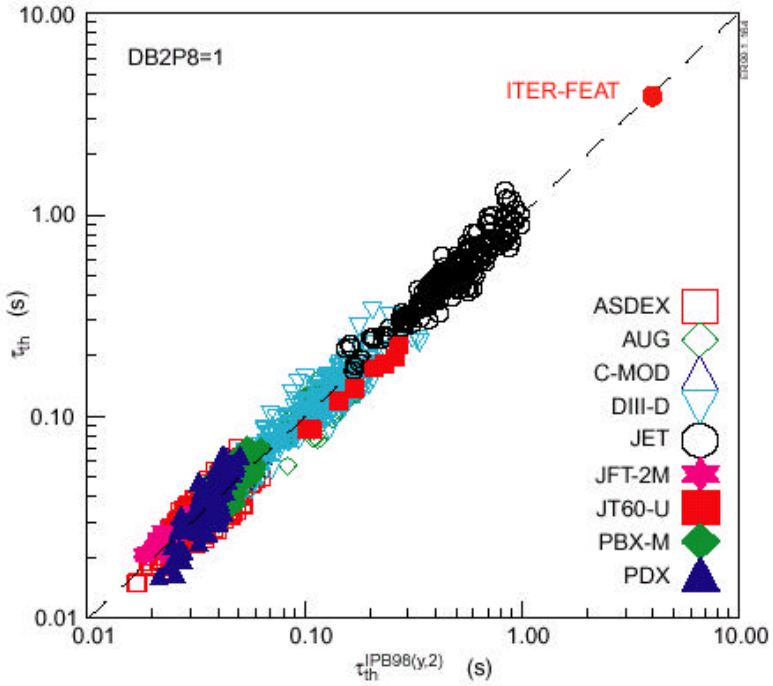
why only in divertors ?

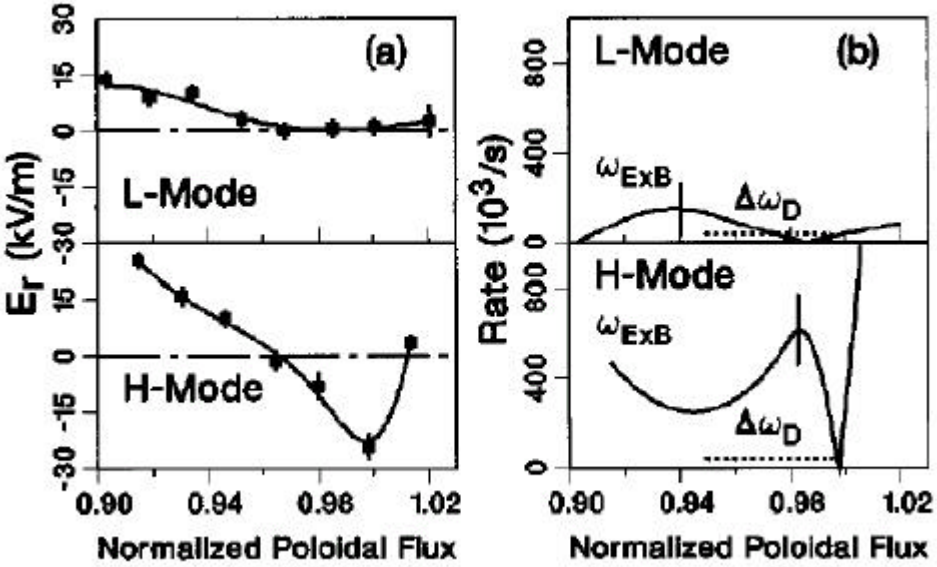
why not internal barriers ?

H-mode:

a reliable tool: ITER-layout basis and reference operating regime

basic paradigm: shear supression of turbulence





several theories (contributing factors) to cause transition:

- self-generation by turbulence
- orbit losses

the physics:

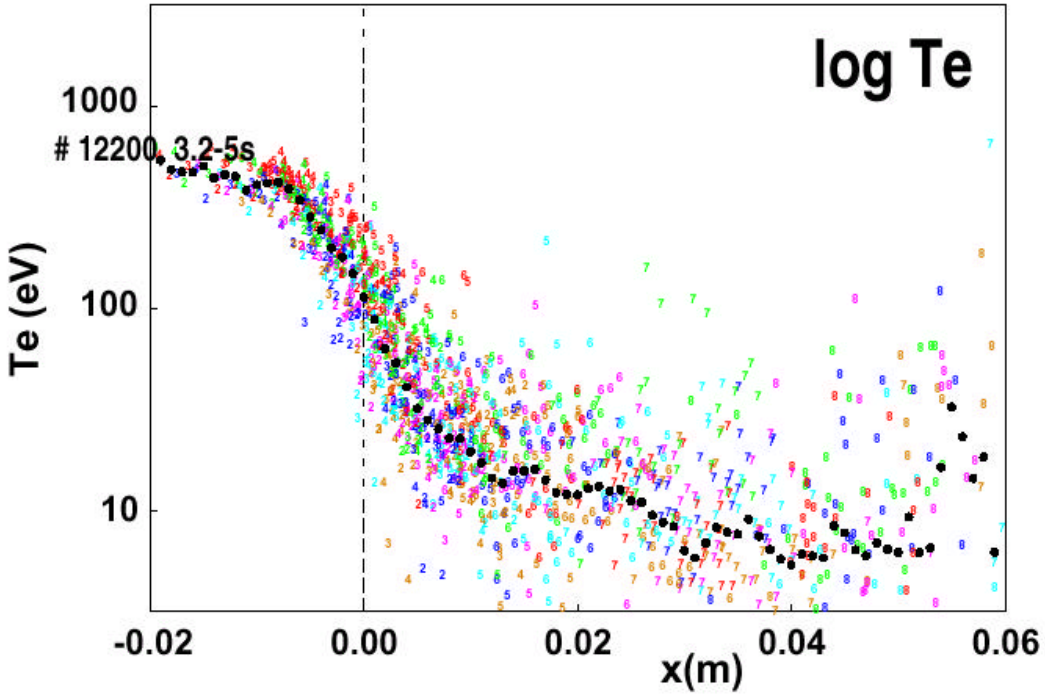
cross-field transport in the SOL

very steep decay lengths
$L_{Te} = (\nabla T/T)^{-1}$ » 0.5 cm in example

continuity of behaviour across separatrix

observed coherence length of turbulence order of L_{Te}

(l_R » 0.2 – 0.8 cm)



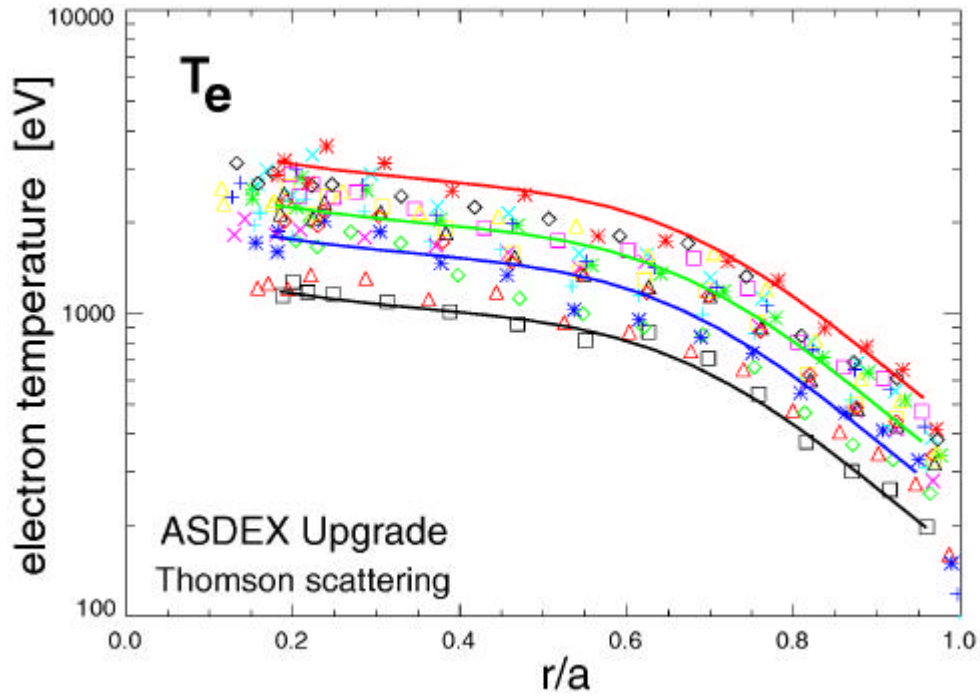
the physics:

profile resilience – the quality of the edge (H-) barrier determines core profile

the ITG mode paradigm:

(unless additional internal barriers develop)

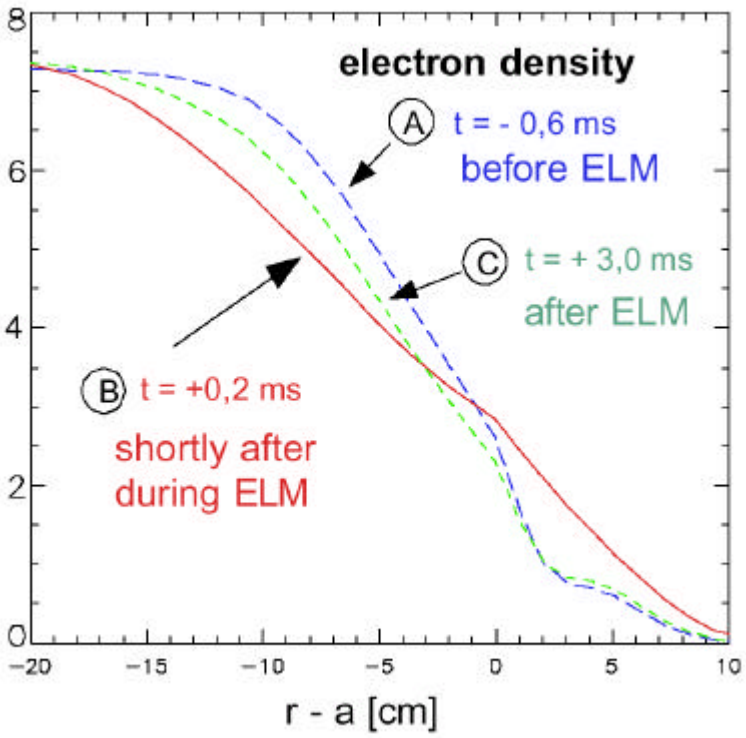
instabilities with ∇T/T – threshold propagate edge characteristics across whole discharge

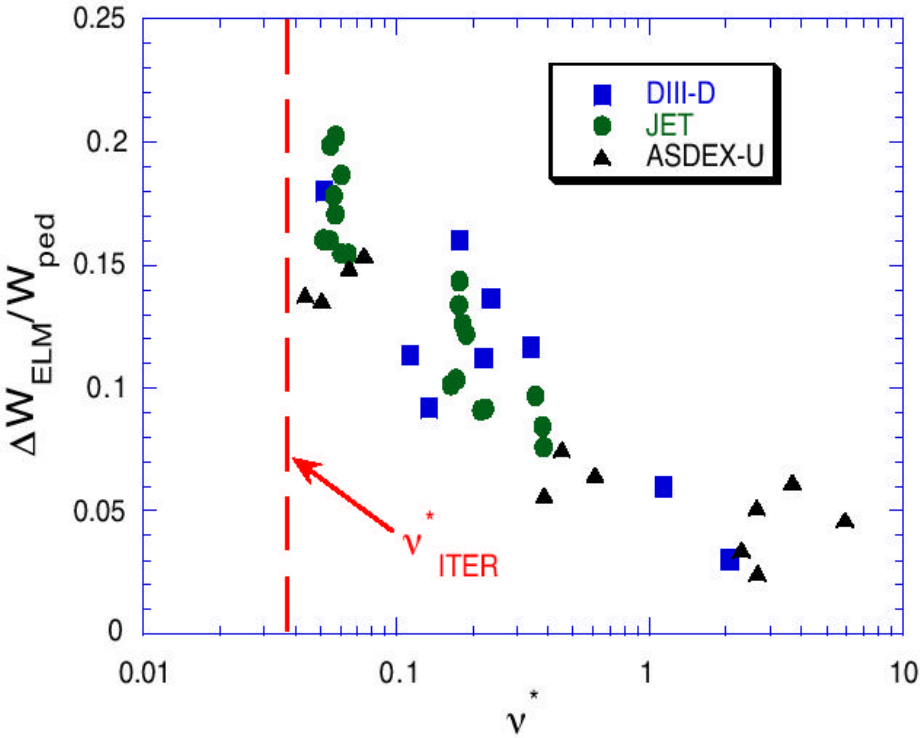


$$c_i = c_i * (T_e / T_i, \hat{s}, q, Z_{eff} ..) \cdot \frac{r_i^2 v_{ti}}{L_n} \cdot G\left(\frac{R|\nabla T_i|}{T_i} - s \right)$$

limits to the H-mode
the ELMs

pressure an/or current density gradients limited by instability: the ELM cycle





associated impulsive heat loads one major problem of ITER/reactor operation

solution: type 2,3 ELMs?

plasma wall interaction for ITER and a Power Plant

paradigma shift:

- from damage of wall to plasma
- to damage of plasma to wall

in any plasma experiment:

radiation losses

(confinement devices have low volumetric power densities)

for a burning plasma

plasma (fuel) dilution

for ITER and a power plant:

target plate damage

tritium inventory

increase in pulse length and duty cycle

divertor heat load tends to increase with size

$$l_{sol} \propto R^{g}, \quad g<1 \quad \Rightarrow q_{div} \propto P_{heat}/R^{1+g}$$

as heating power can go like

$$P_{heat} \propto R^{2} \quad \text{for externally applied heating}$$

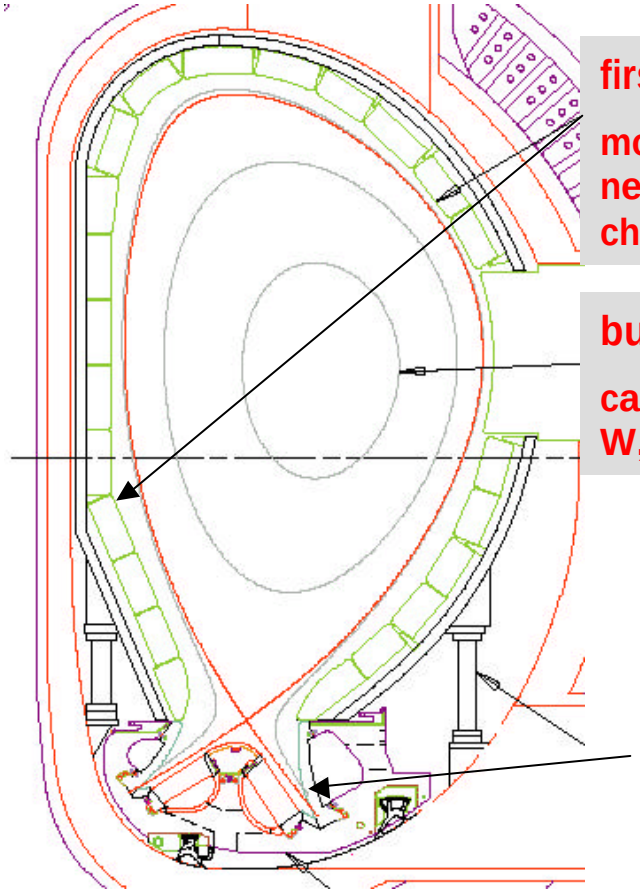
and

$$P_{heat} \propto R^{3} \quad \text{for intrinsic (fusion) heating.}$$

	P_{fus} [GW]	R_o [m]	P_{rad}/P_{heat} 1)	$q_{div,nom}$ 2) [MW/m^2]
ASDEX-Upgrade	0.02 (P_{heat})	1.6	0.8	5
ITER-ref.	0.5	6.2	0.8	5
ITER-SS	0.36	6.2	0.8	5
PPCS	> 4	> 7.5	0.9	11
ARIES-RS	2.2	5.5	0.9	11

1) $P_{heat} = (P_{fus}(1/5+1/Q))$
2) $q_{div,nom} = (P_{heat}-P_{rad})/(4pR_o l F)$ with geometry factor F = 10
and mid plane heat flux width scaling like $l = 0.003 R_o^{0.5}$

First Wall and Divertor: materials in a hostile requirement



first wall:

modest flux of high energy neutral particles (~ 100s eV), chemical erosion

bulk plasma:

cannot tolerate impurities (= 10^{-5} W, 10^{-2} Be, 5 x 10^{-2} He)

tritium inventory issue:

- tritium in plasma vessel must be kept low for safety
- sputtered, desorbed, evaporated material co-redeposits with high hydrogen fraction

divertor target plates:

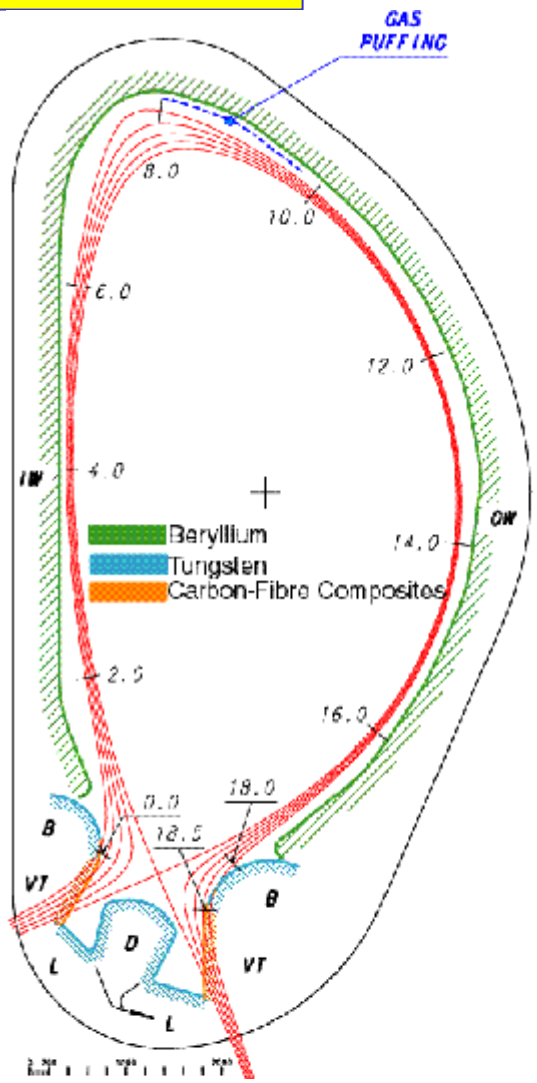
high heat flux (without radiation protection: ? 60 MW/m2, with r.p. ? 10 MW/m2)

impulsive heat loads: ELMs, disruptions

Main technological Problem for ITER: plasma facing materials

- Issues for the CFC divertor targets are
  - Erosion lifetime (chemical sputtering), T-co-deposition (450g T limit inside the vessel) and C dust production (~200kg limit)
- Issues for the Be FW are
  - Be dust production (100kg limit) in particular on hot surfaces (6kg limit) -> Hydrogen production in off normal events - explosion ?
- Issues for W clad divertor targets are
  - Disruption erosion - melt-layer loss
  - W dust production (100 - 300kg limit) causing a radiological hazard in case of a by-pass event
  - Plasma compatibility

PFCs	Area (m^2)
IW	300
OW	380
L	10
B	50
D	30
VT	55
Liner	60



Present preference: 3 different materials

Replacement of CFC by W for divertor strike zones considered

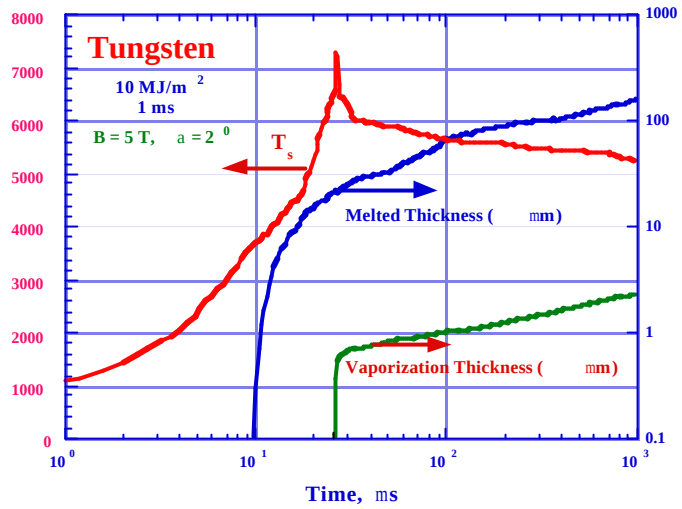
W in the divertor is practically only eroded by disruptions

- The issue is how much of the melt layer (~ 100µm) will be lost
- What kind of longer term distortions of the target surface are to expect

Plasma compatibility, i.e. the W impurity migration into the main plasma is a concern

- Promising results from ASDEX-UP suggest that W should be more seriously considered

From Engineering point of view W clad targets have reached almost the same development level as CFC



A sacrificial thickness of 10 mm could last only for about 50-100 disruptions

Situation for carbon might not be better, if brittle destruction plays a role.

Longer disruption times can further reduce disruption lifetime.

Experimental data regarding formation of melt layers (and their properties)are very meagre.

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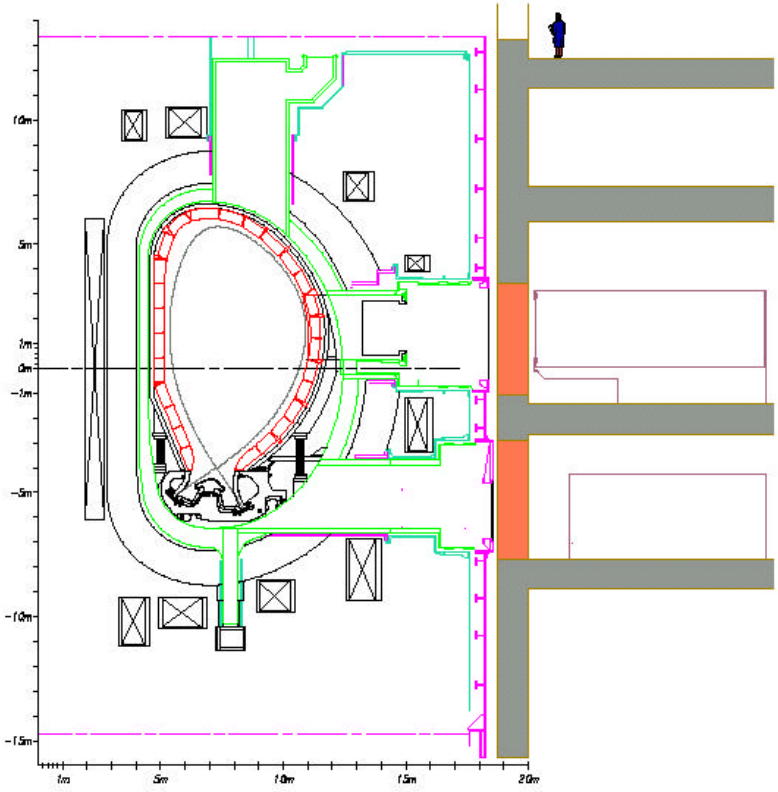
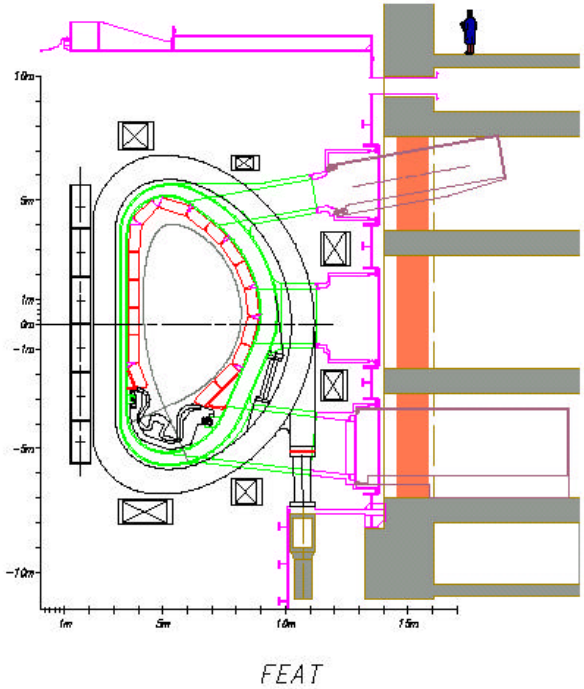
at the occasion of its

anniversary

PPPL is happy to announce that all electric energy requirements of New Jersey will hitherto be met by clean fusion power

Princeton 13.9.01

ITER-FDR vs. ITER-FEAT

Major radius	8.14 m	6.2 m
Minor radius	2.8 m	2.0 m
Plasma current	21 MA	15 MA
Toroidal magnetic field	5.7 T	5.3 T
Plasma elongation	1.7	1.85
Fusion power	1500 MW	400
Q	> 100 (“ignited”)	> 10
Burn duration	> 1000 s	> 300 s