Experimental Results from Tokamaks

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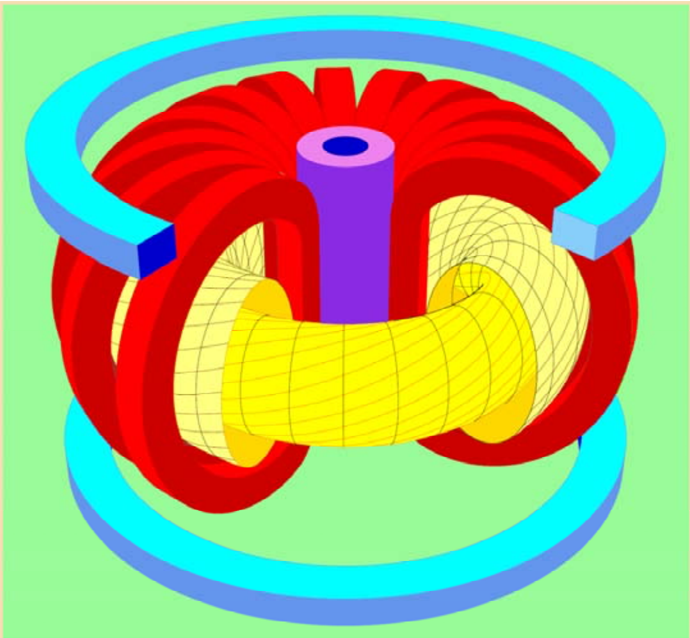
Thanks to R. Dux, R. Neu and W. Suttrop



„Tokamak"

from Russian words:
toroidalnaja ***ka***mera (toroidal chamber)
magnitnaja ***k***atushka (magnetic coil)

Outline

- Introduction: Tokamak Principle – Why tokamaks anyway?
- Why does ITER look like it looks like?
  - maximizing $nT\tau$
  - what is the minimum B_t?
  - wall contact & material
- Additional Issues
- Outlook to Reactor

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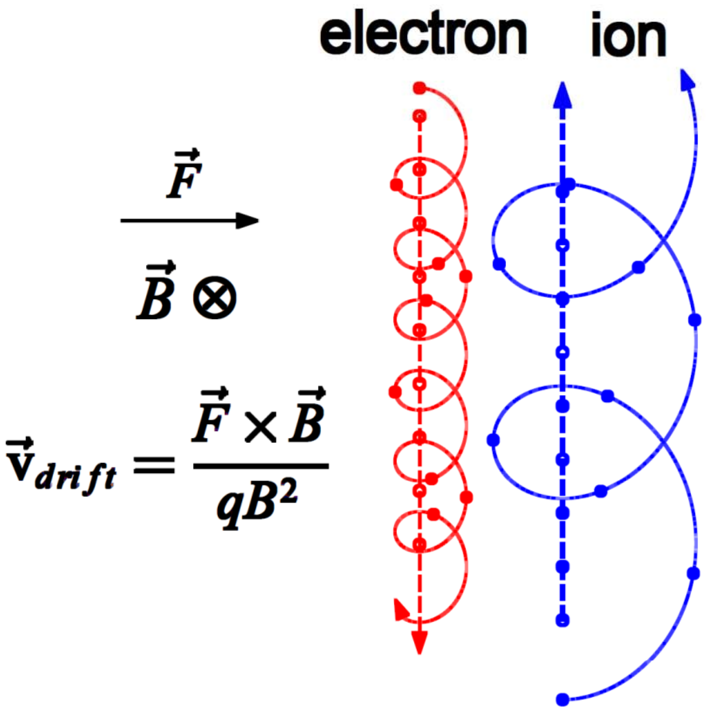
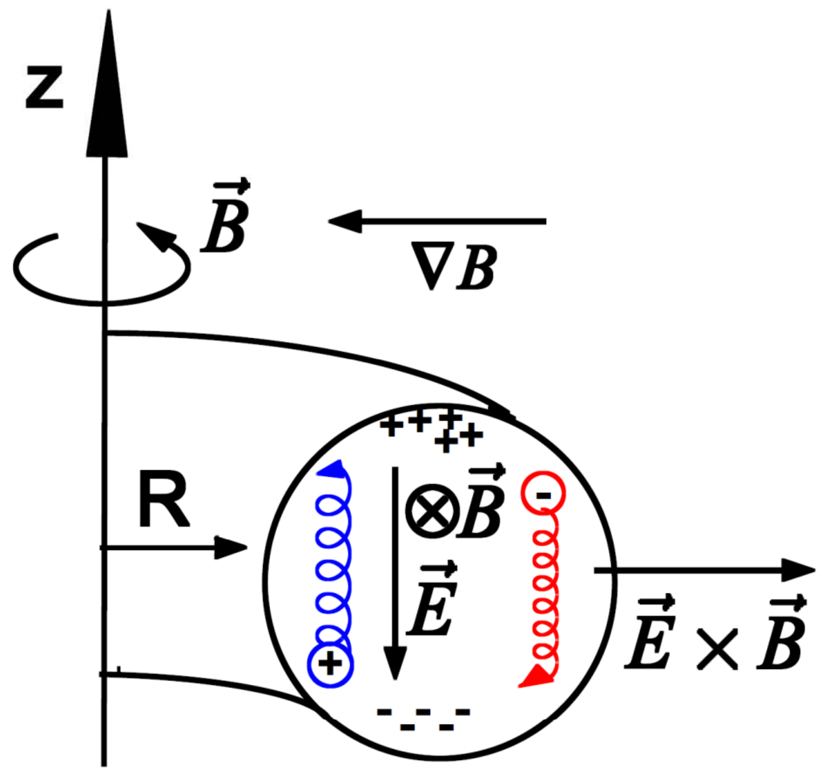
No confinement with pure toroidal mag. field

- Centrifugal force + force on magnetic moment μ in B-gradient
- Corresponding FxB drifts are charge dependent => E-field
- ExB-drift (charge independent) leads to a confinement loss

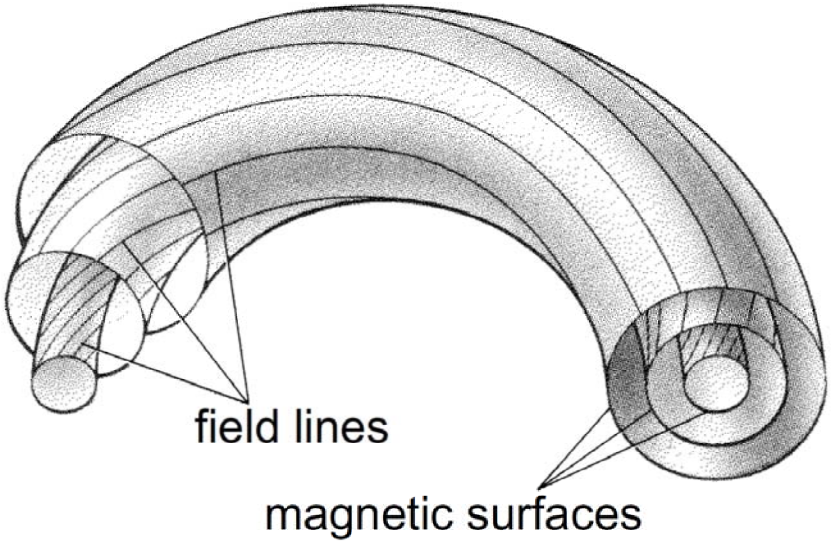


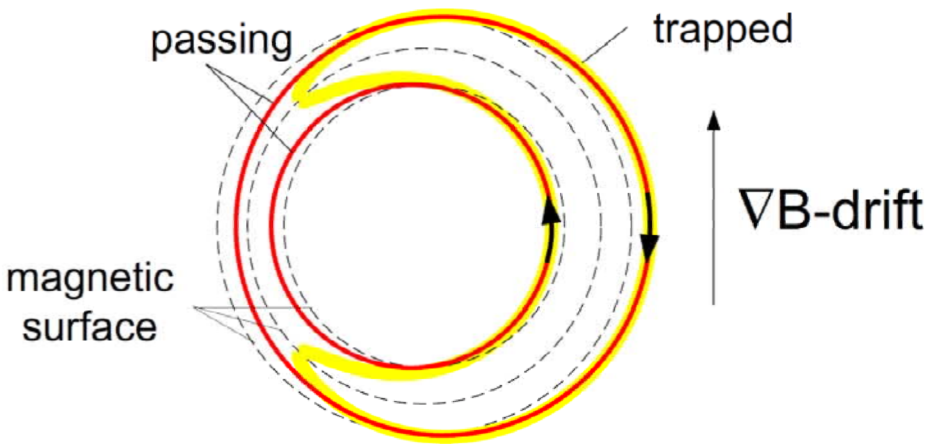


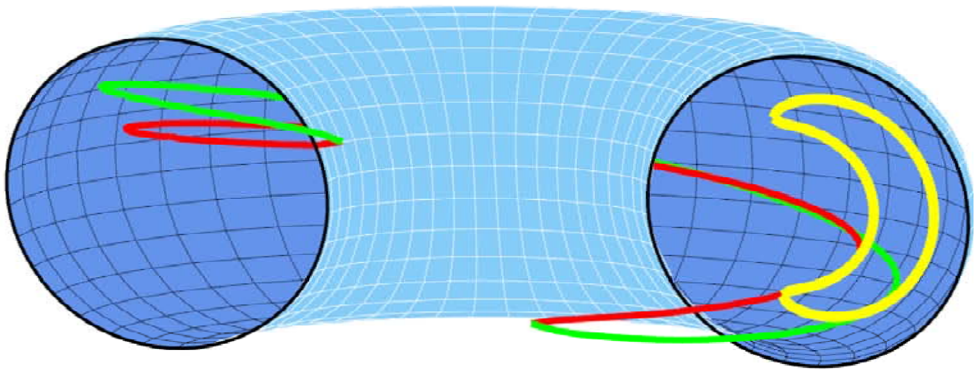
Additional Poloidal Field Compensates Drifts

- Additional poloidal field:
  - ⇨ Helical field lines
  - ⇨ Particles drift away and towards center
  - ⇨ Magnetic mirror configuration for some particles („Trapped Particles“)

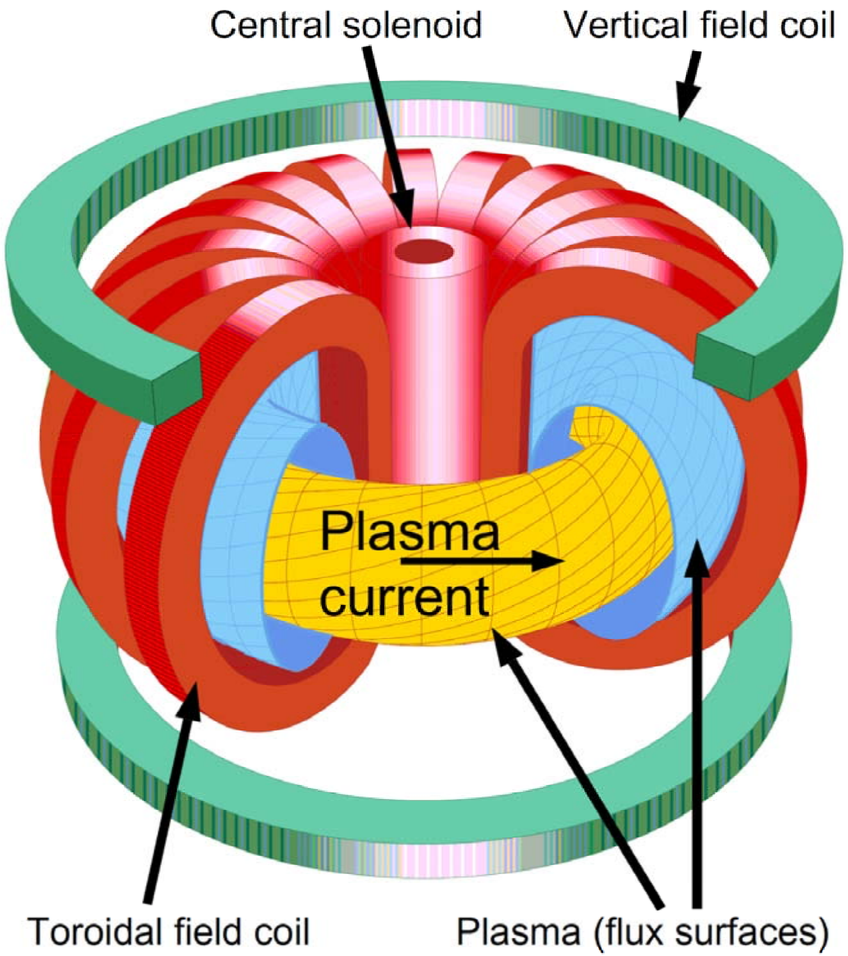
    (μ is a constant of motion…)
  - ⇨ Passing particles => small excursions
  - ⇨ Trapped particles = large excursions







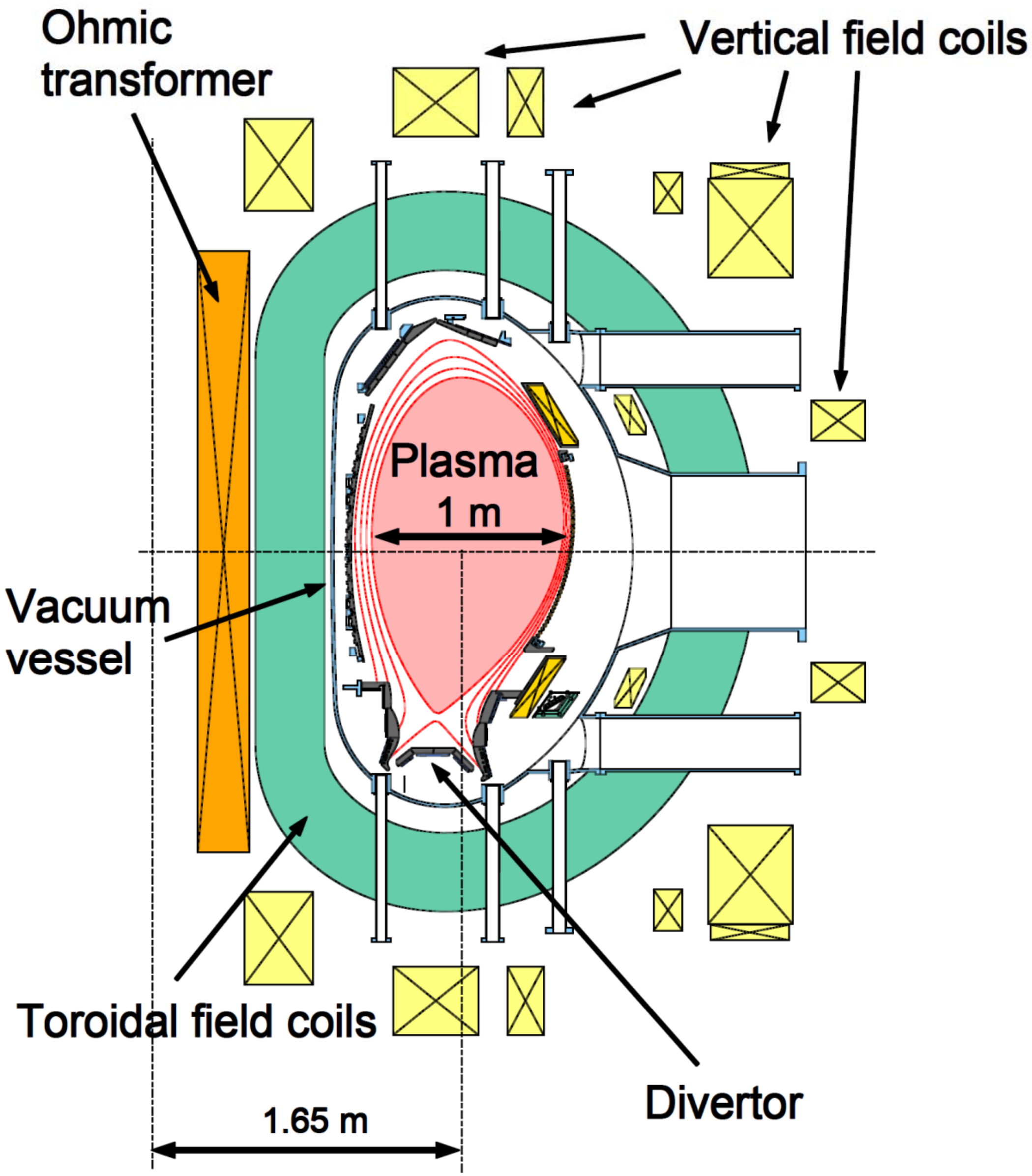
The Tokamak



- Axissymmetric Configuration (2D-problem)
- Toroidal Field+Plasma Current

  => helical magnetic field lines
- Basic Requirements:
  - ⇨Toroidal field coils
  - ⇨Vertical field coils
    - ⇨stability (compensating the hoop force)
    - ⇨ Manipulation of Plasma Shape
  - ⇨Central solenoid for driving plasma current

ASDEX Upgrade
(Axi-Symmetric Divertor Experiment)



Plasma parameters:

Major radius: 1.65 m

Minor radius: 0.5 m

Toroidal field: 3.4 T

Plasma current: 1.4 MA

Auxiliary heating:

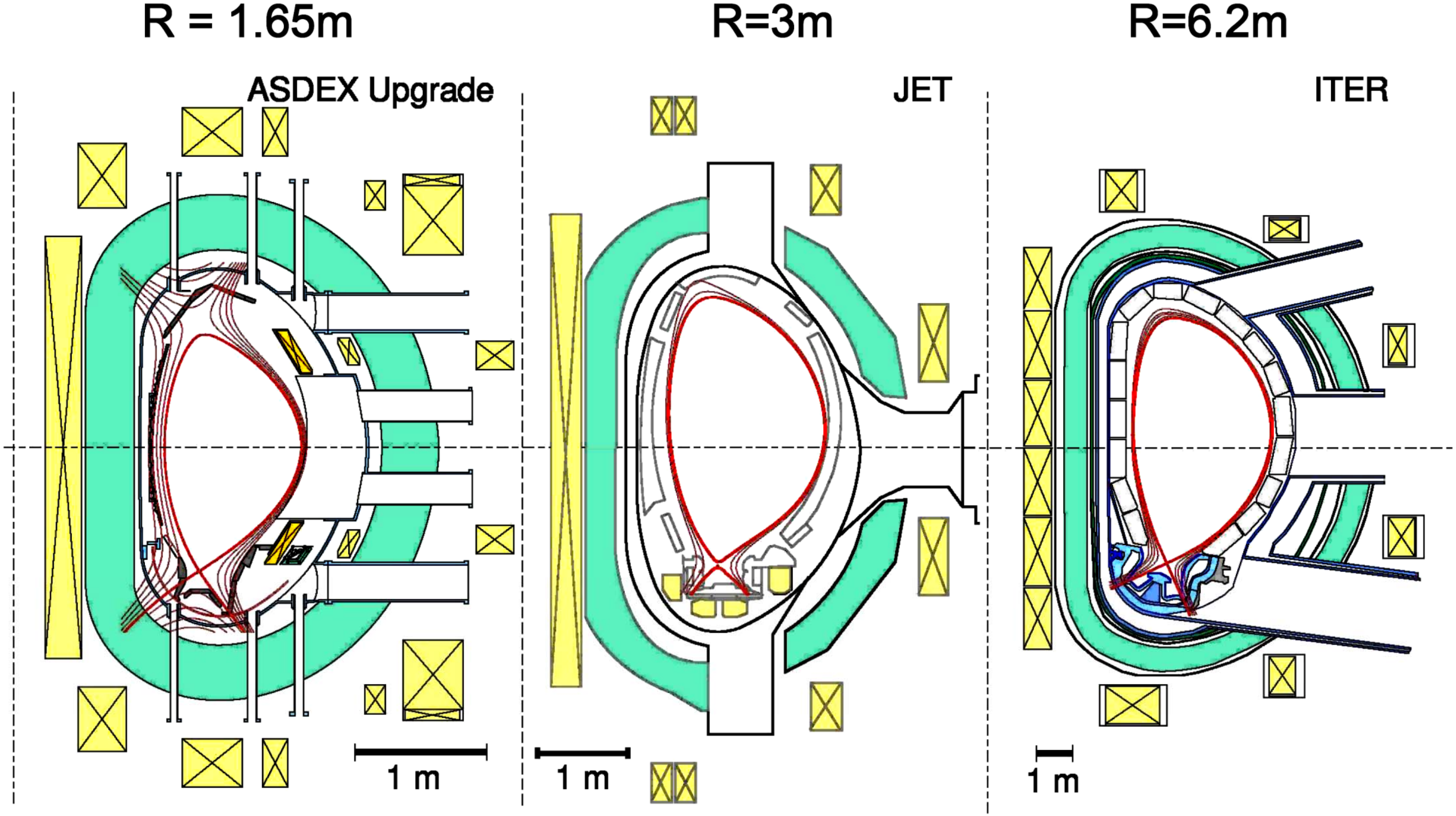
Neutral beam injection:
20 MW (60, 93 keV)

Ion Cyclotron Heating:
6 MW (30-60MHz)

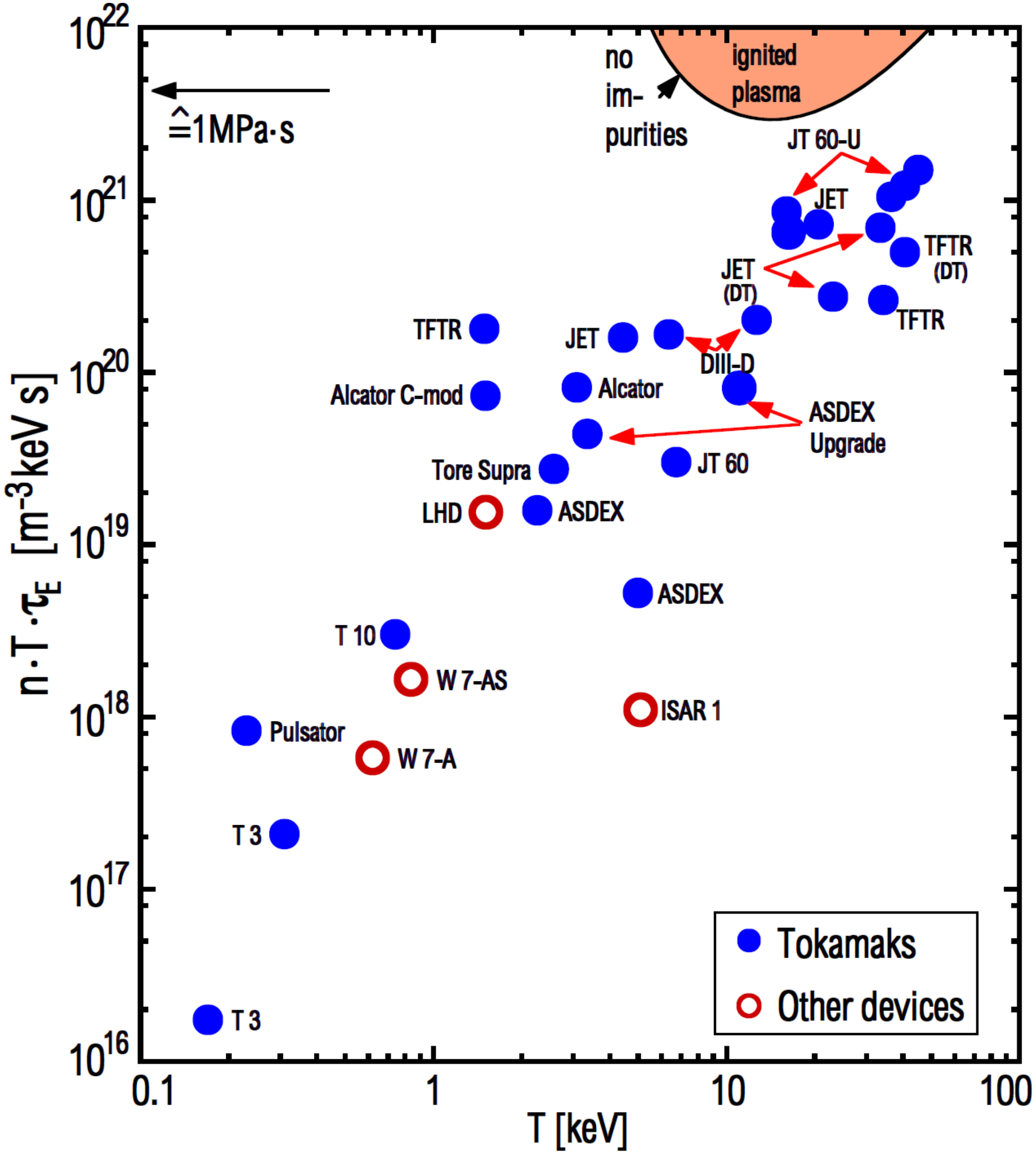
Electron Cyclotron Heating:
4 MW (140/105GHz)

A Few Divertor Tokamaks

Major Radius



Tokamkas:
Reactor Relevant Paramters Achievable



- Ignition parameters are nearly matched by tokamaks
- ITER is a tokamak
- Axi-symmetry facilitates
  - ⇨ Construction
  - ⇨ Operation
  - ⇨ Research

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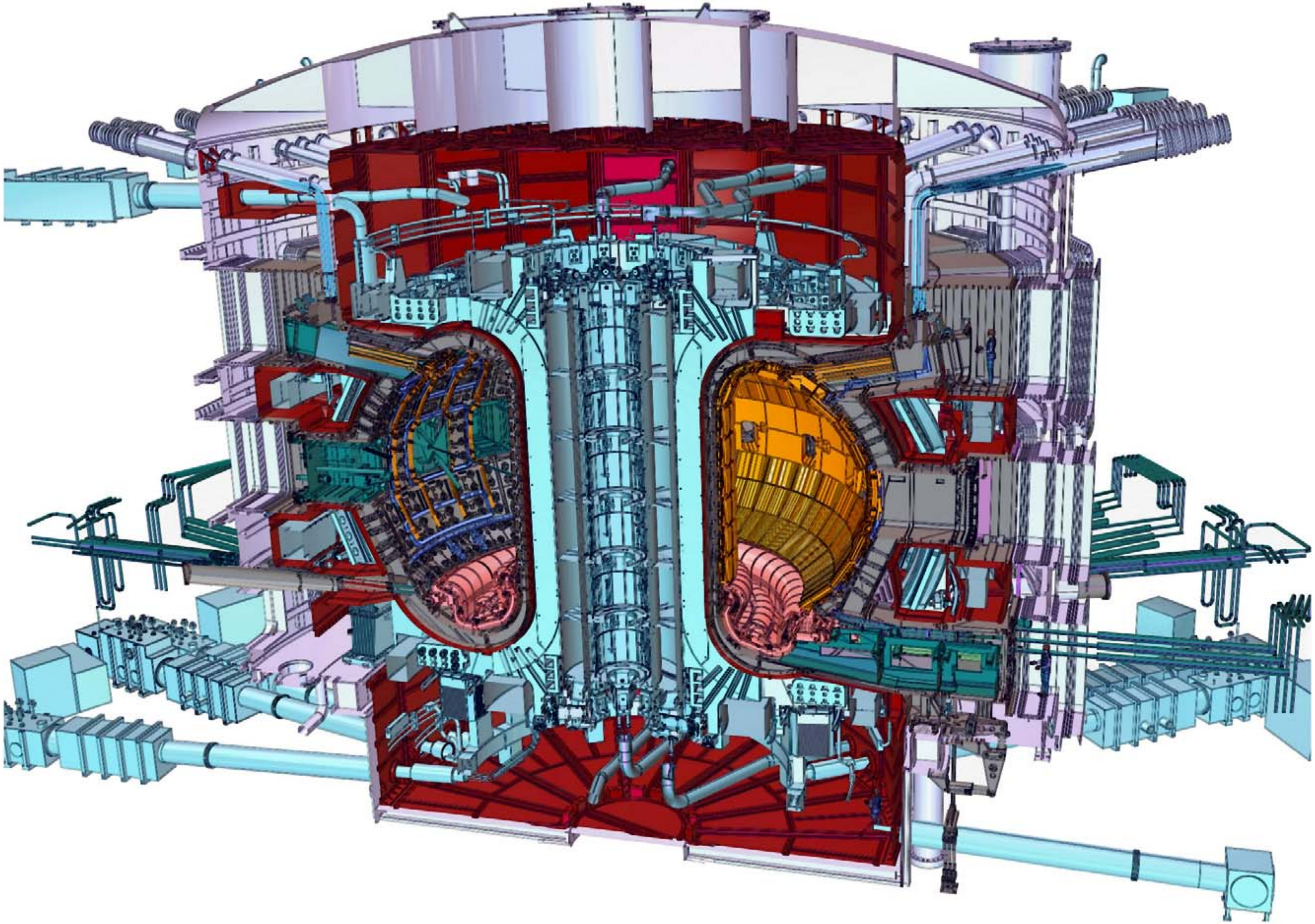
ITER is big, elongated-shaped, large superconducting coils and has a divertor – why?



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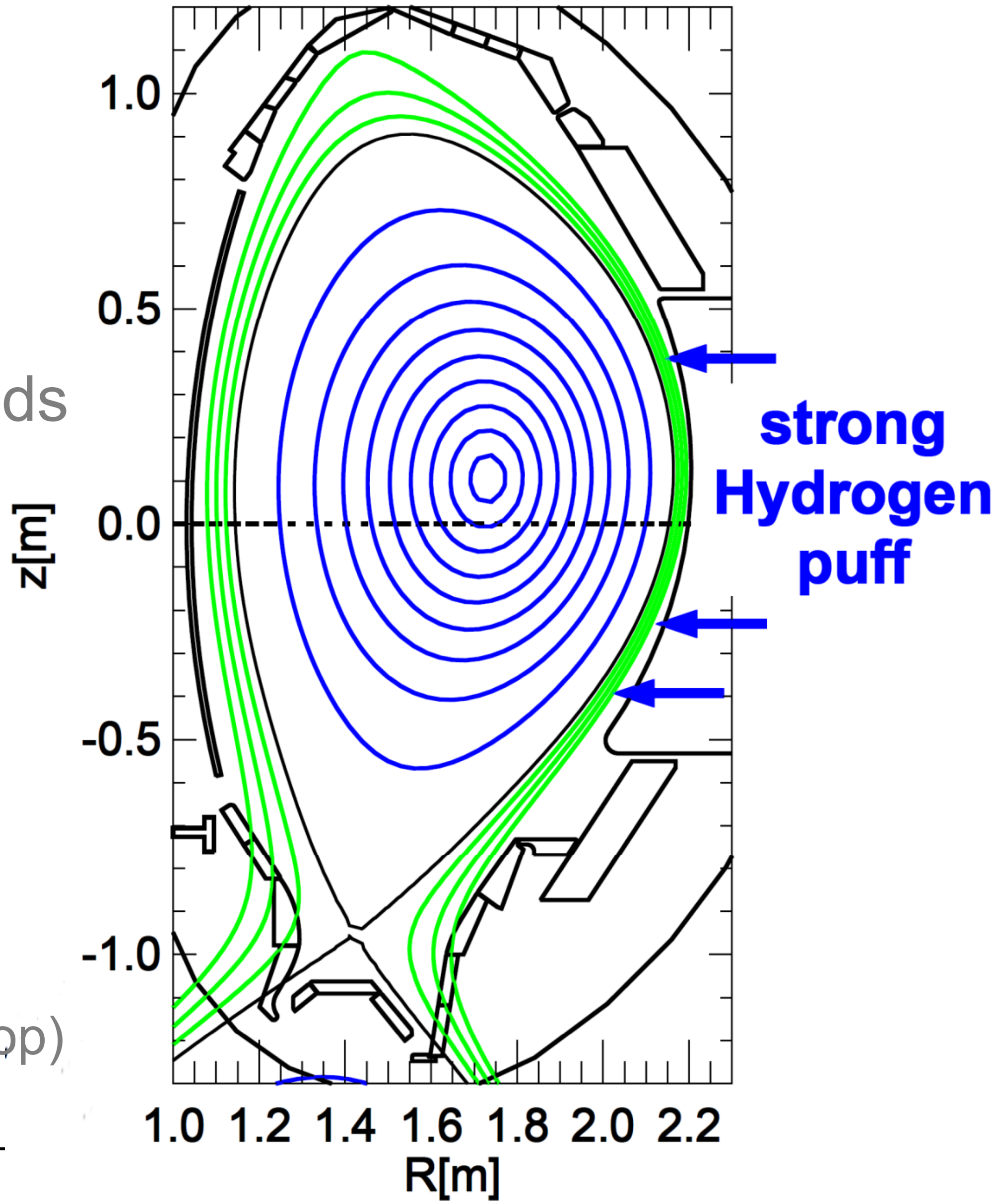
Maximizing Density n – Greenwald Density Limit!

- Strong gas puff, cools down edge and divertor plasma
- Recombination in Divertor leads to ‚Detachment'
- Radiation instability forms:
  - ⇨ At lower Te, more radiation from same amount of impurities
  - ⇨ Cooling of Te by radiation increases radiation (feedback-loop)



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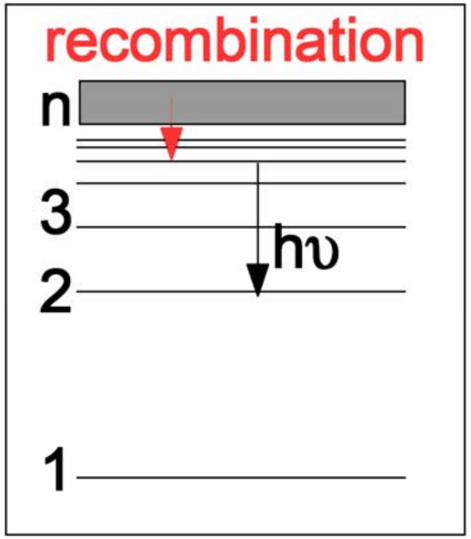

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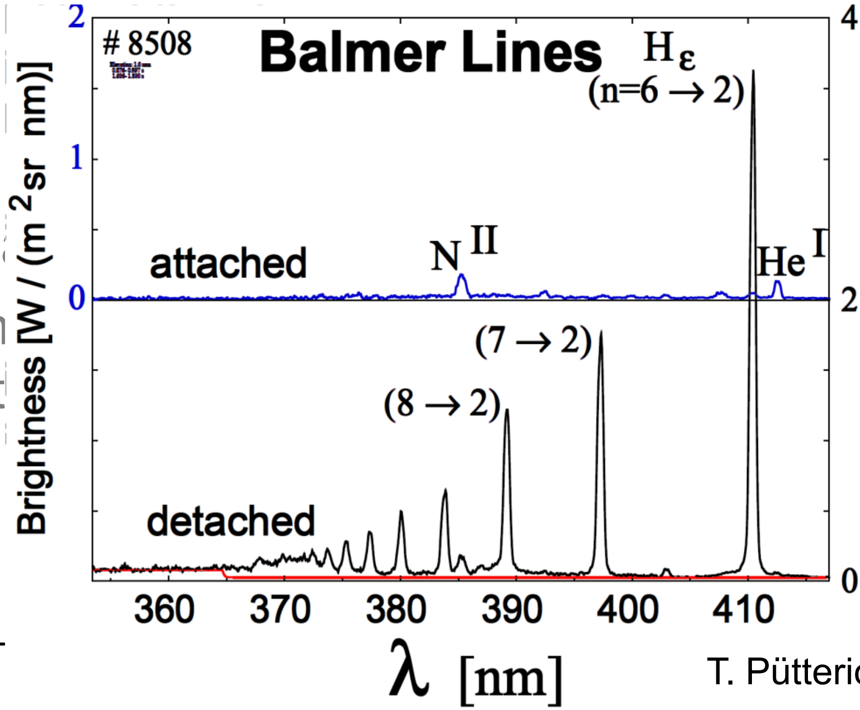
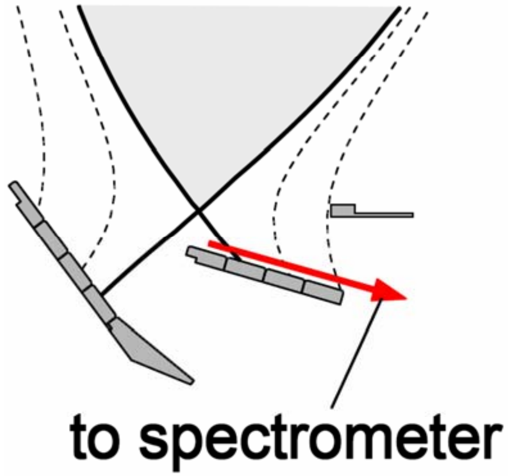


- Radiation inst
  - ⇨ At lower Te, n same amount
  - ⇨ Cooling of Te increases rad



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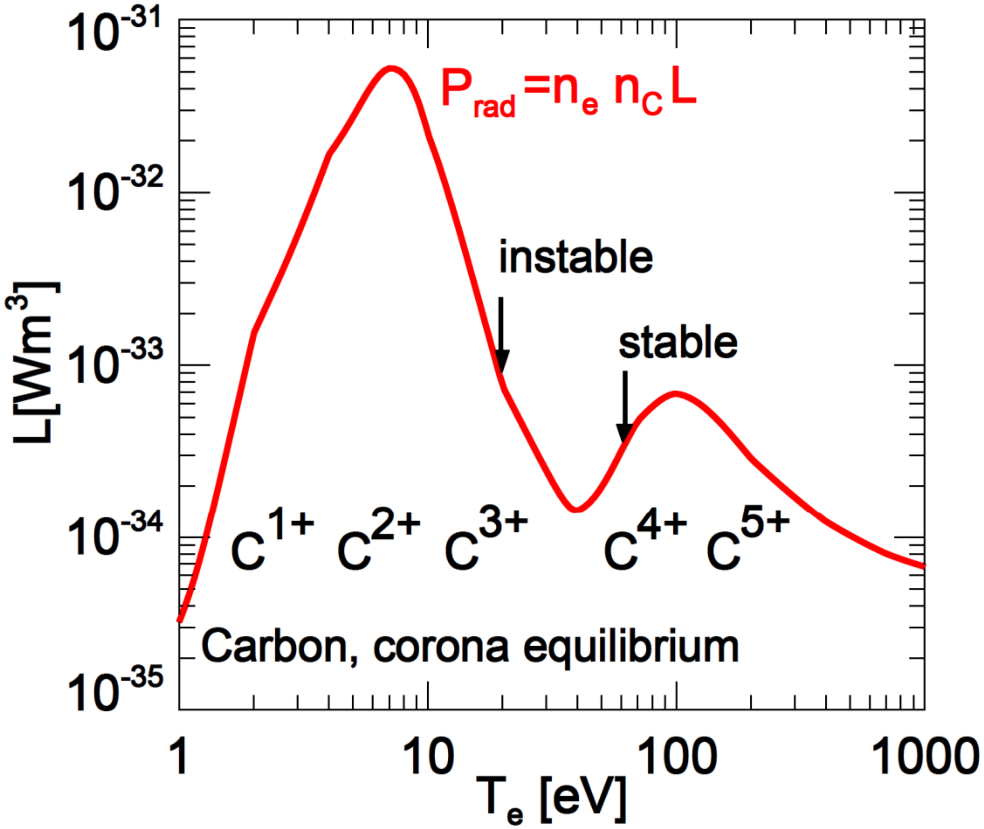
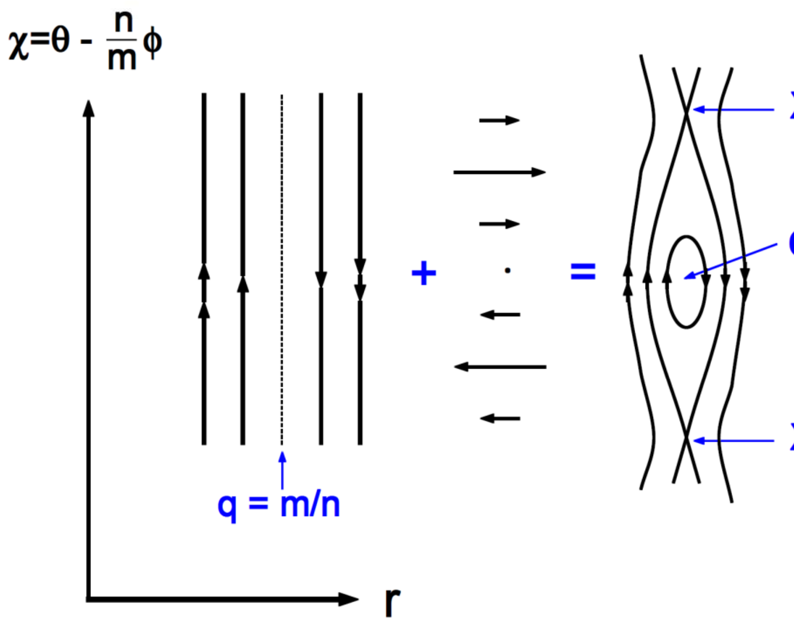


Maximizing Density n – Greenwald Density Limit!

- As Edge Te drops, resistivity rises
- resistive MHD modes appear and destroy the onion shell structure of the magnetic surfaces
- further loss of temperature ....disruption

Tearing modes (resistive, ∇j driven)



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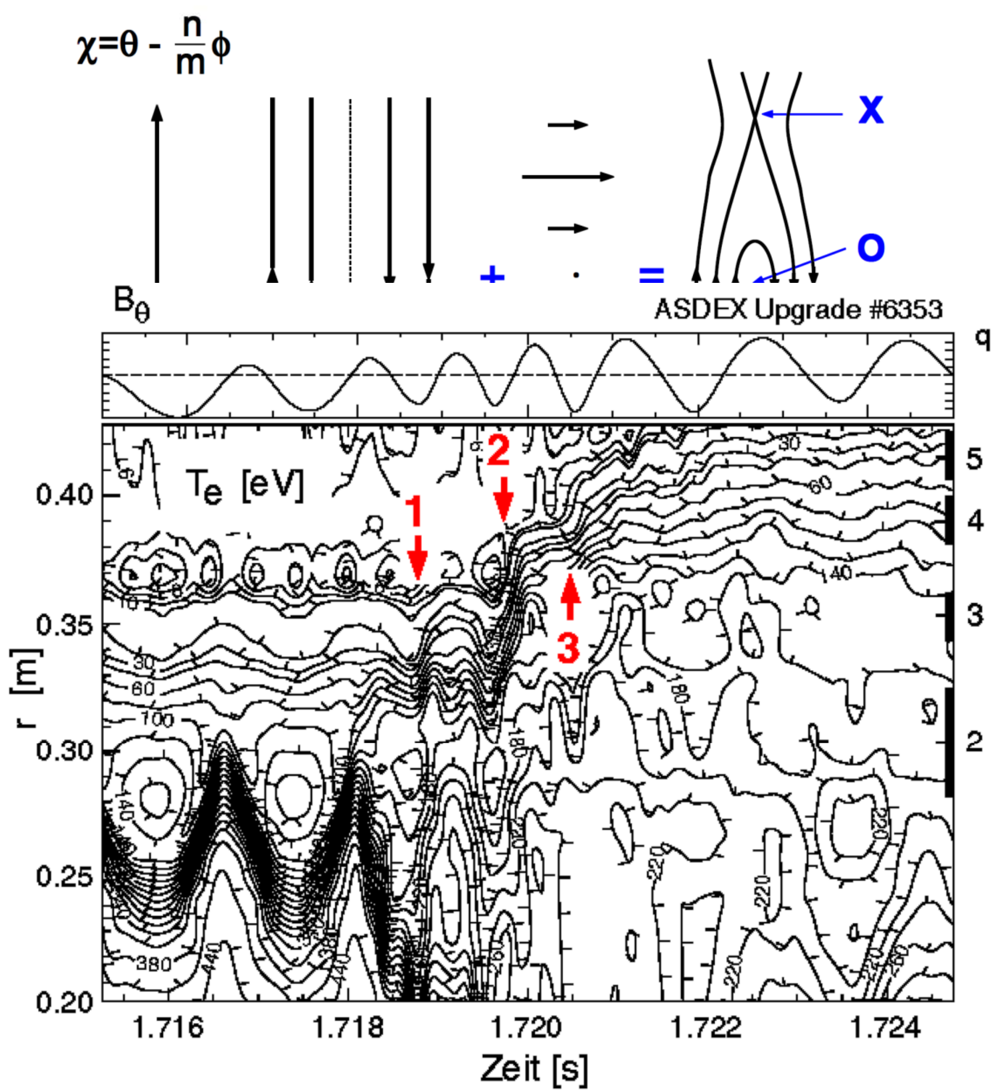


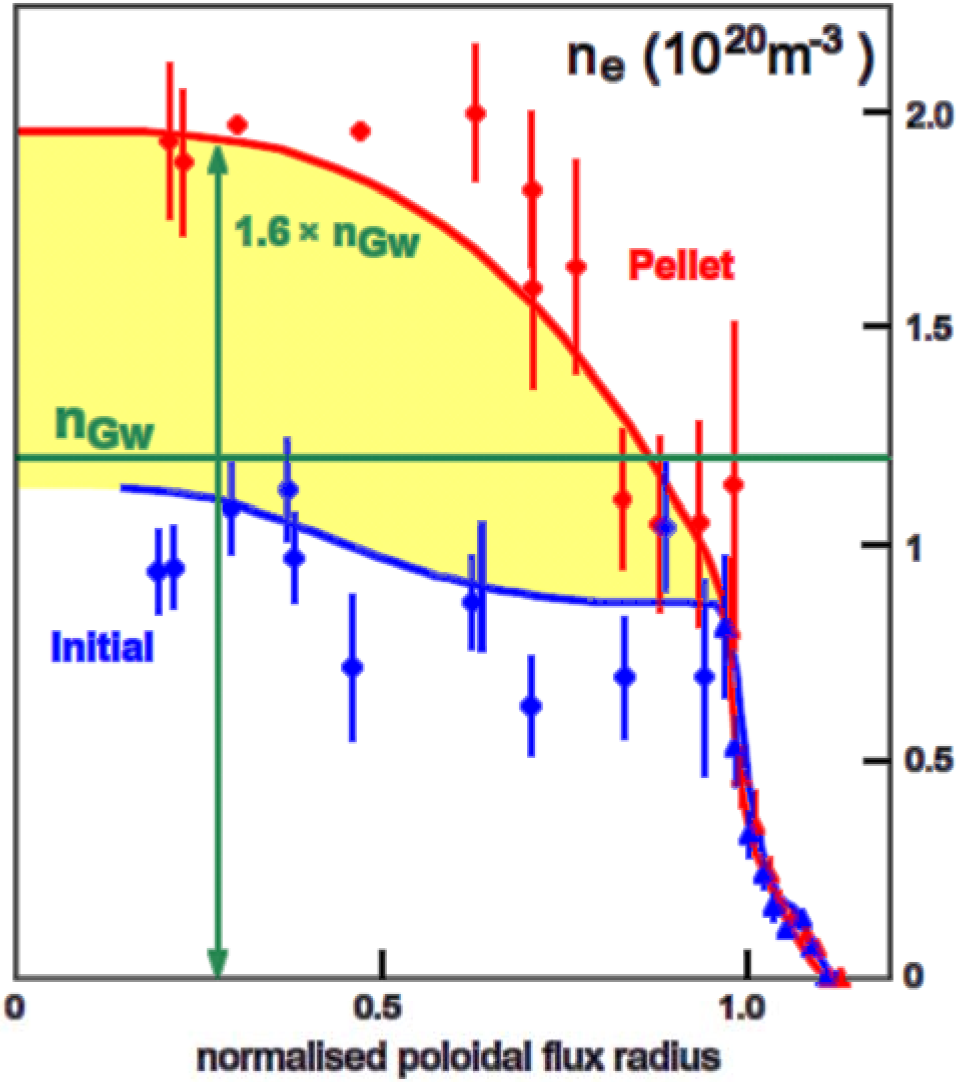
Greenwald Density Limit Applies at Plasma Edge

- Greenwald limit

$$n_{\text{limit}} \propto \frac{I_p}{a^2 \pi}$$

- Applies for gas puffing

- Core fueling by pellets allows to push core density above that limit



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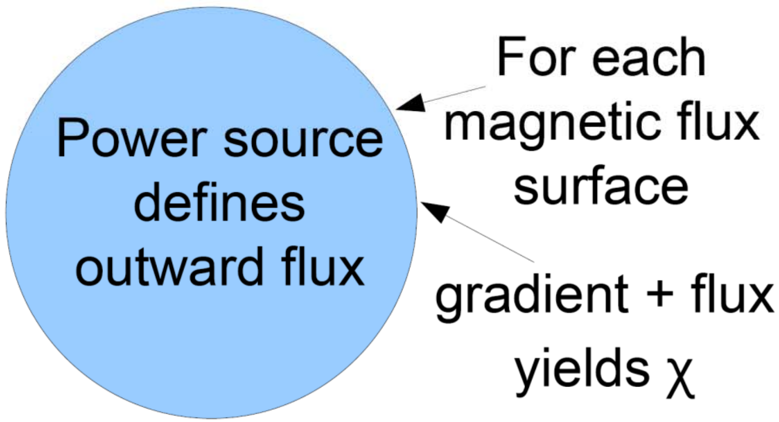
How to Measure Heat Transport – Method A

Heat transport coefficient χ can be obtained from Te-profiles in equilibrium:

Heat flux due to radial transport

$$q = -\chi n_e \frac{\partial T_e}{\partial r}\Big|_{r=r'}$$



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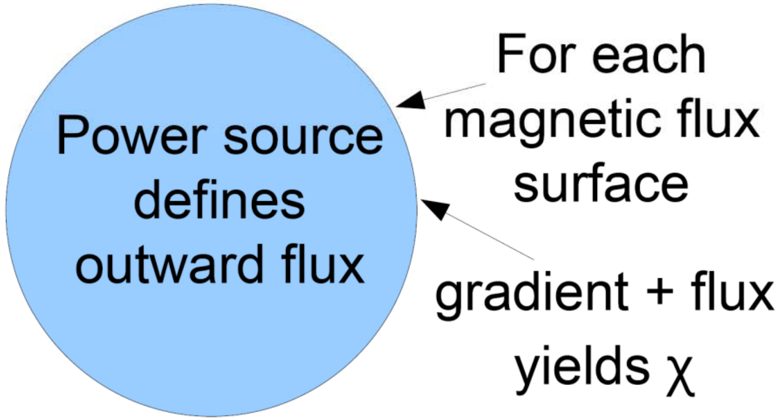

Heat flux due to radial transport

$$q = -\chi\, n_e \frac{\partial T_e}{\partial r}\Big|_{r=r'}$$

equals heating power inside of radius r'

$$q = \frac{1}{r'} \int_0^{r'} P_{heating}\, r\, dr$$

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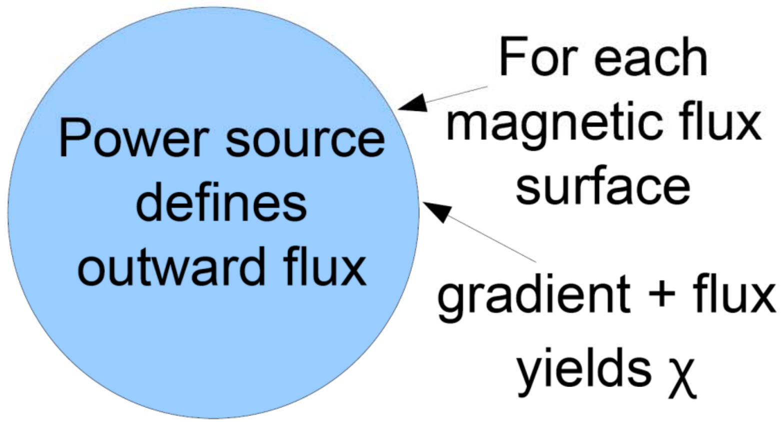

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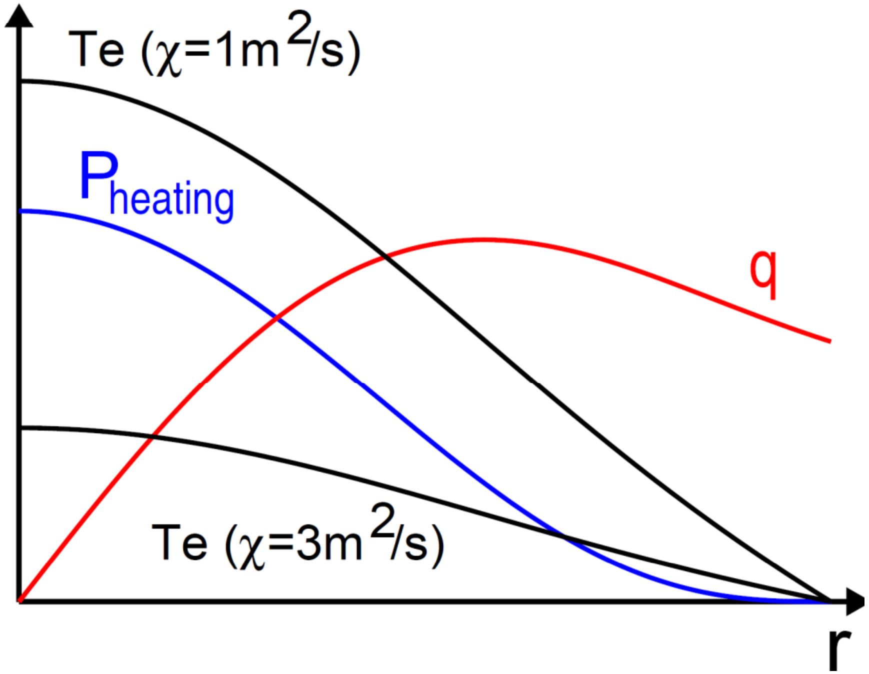
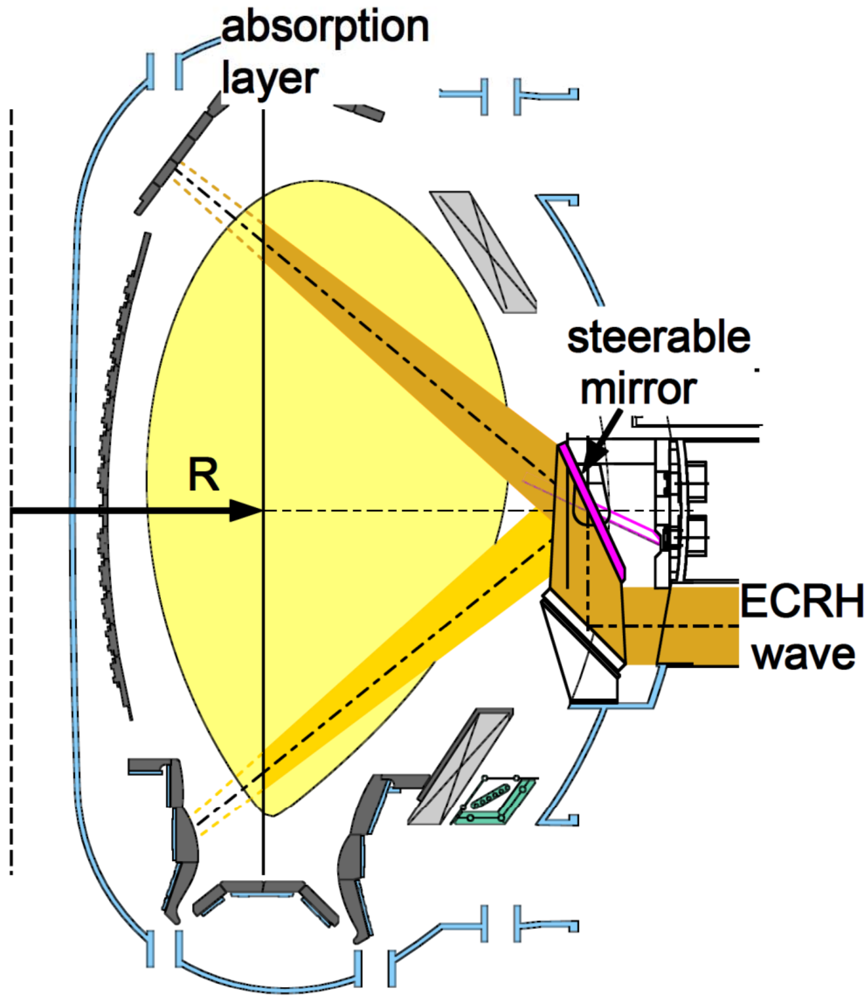


How to Measure Heat Transport – Method B

ECRH heating is ideal to induce local Te perturbation

Fast measurements of Te using electron cyclotron emission allow to track the propagation of the heat pulses



$$2\omega_e(R)=2\frac{e\,B(R)}{m_e}=\omega_{ECRH}$$

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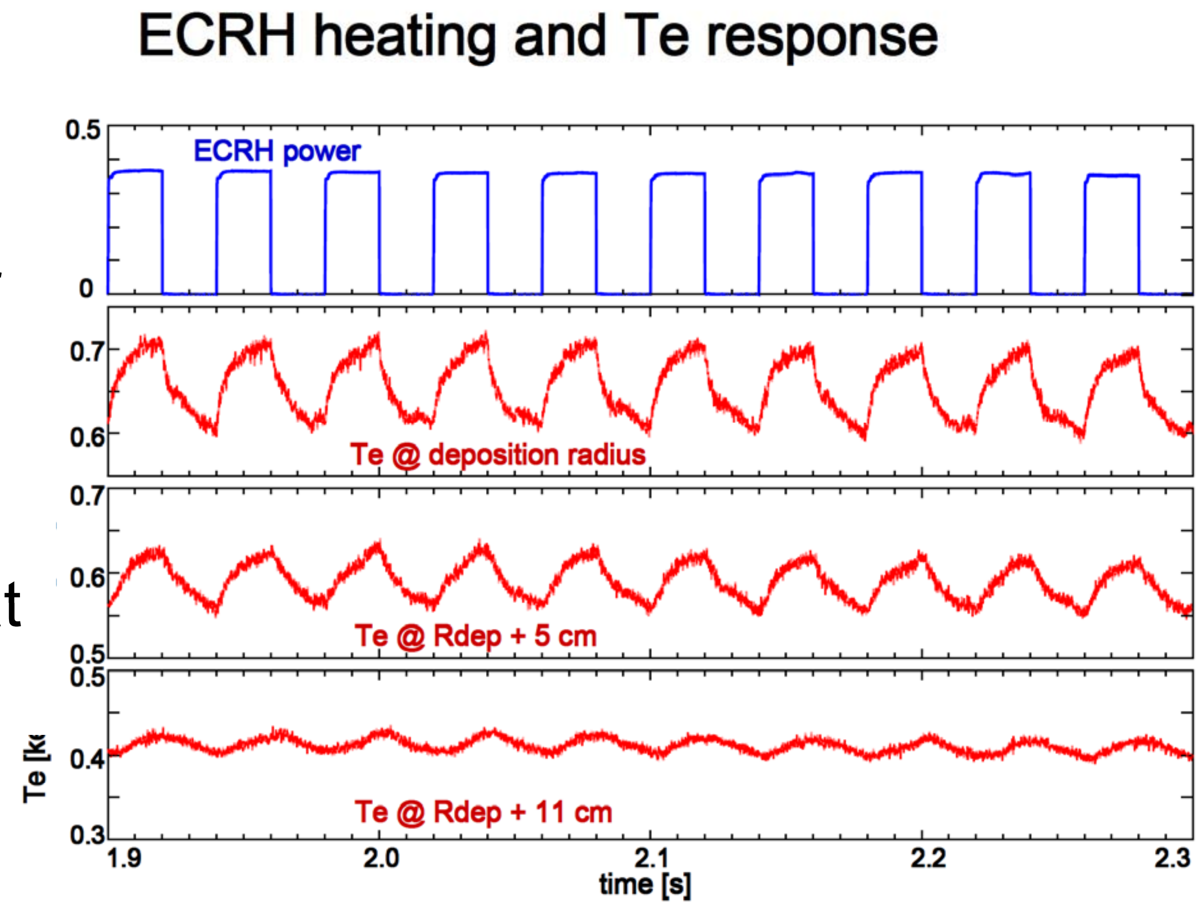
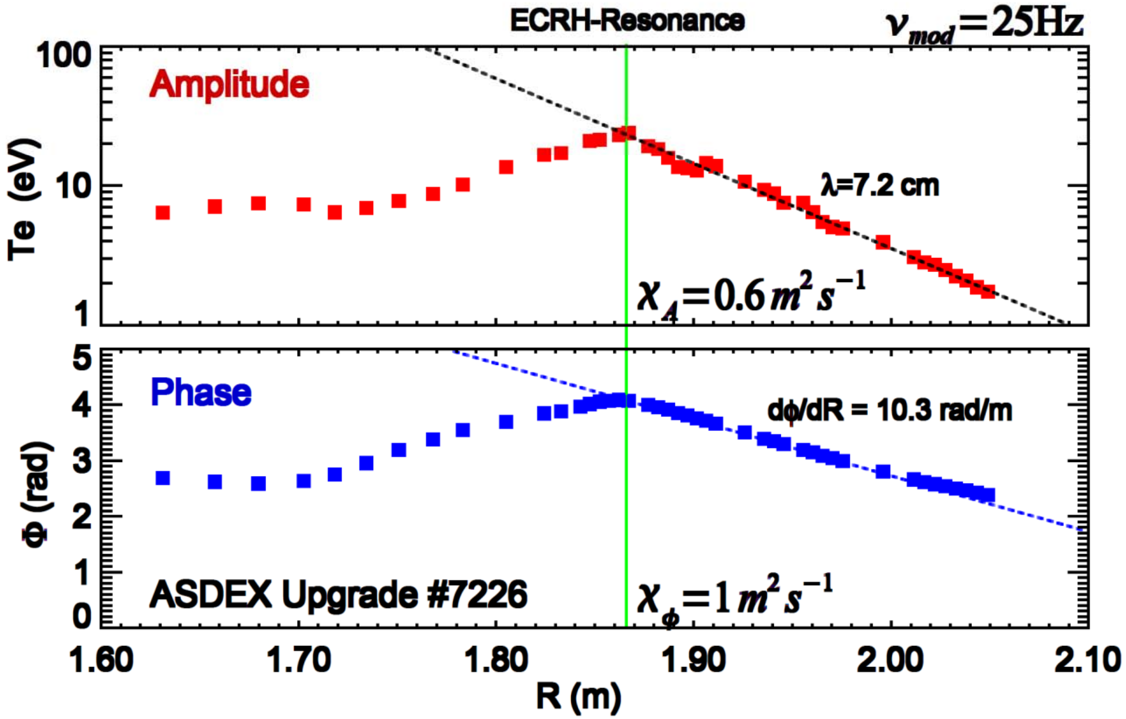


How to Measure Heat Transport – Method B

Fourier transform: Take A, Φ profiles at modulation frequency $\omega_{mod}/2\pi$



Slab geometry:

Decay of amplitude:

$$\tilde{T}=\tilde{T}_0 e^{-\Delta R/\lambda}$$

$$\chi_A=\frac{3}{4}\omega_{mod}\left(\frac{1}{\tilde{T}}\frac{\partial \tilde{T}}{\partial R}\right)^{-2}=\frac{3}{4}\omega_{mod}\lambda^2$$

Change of phase

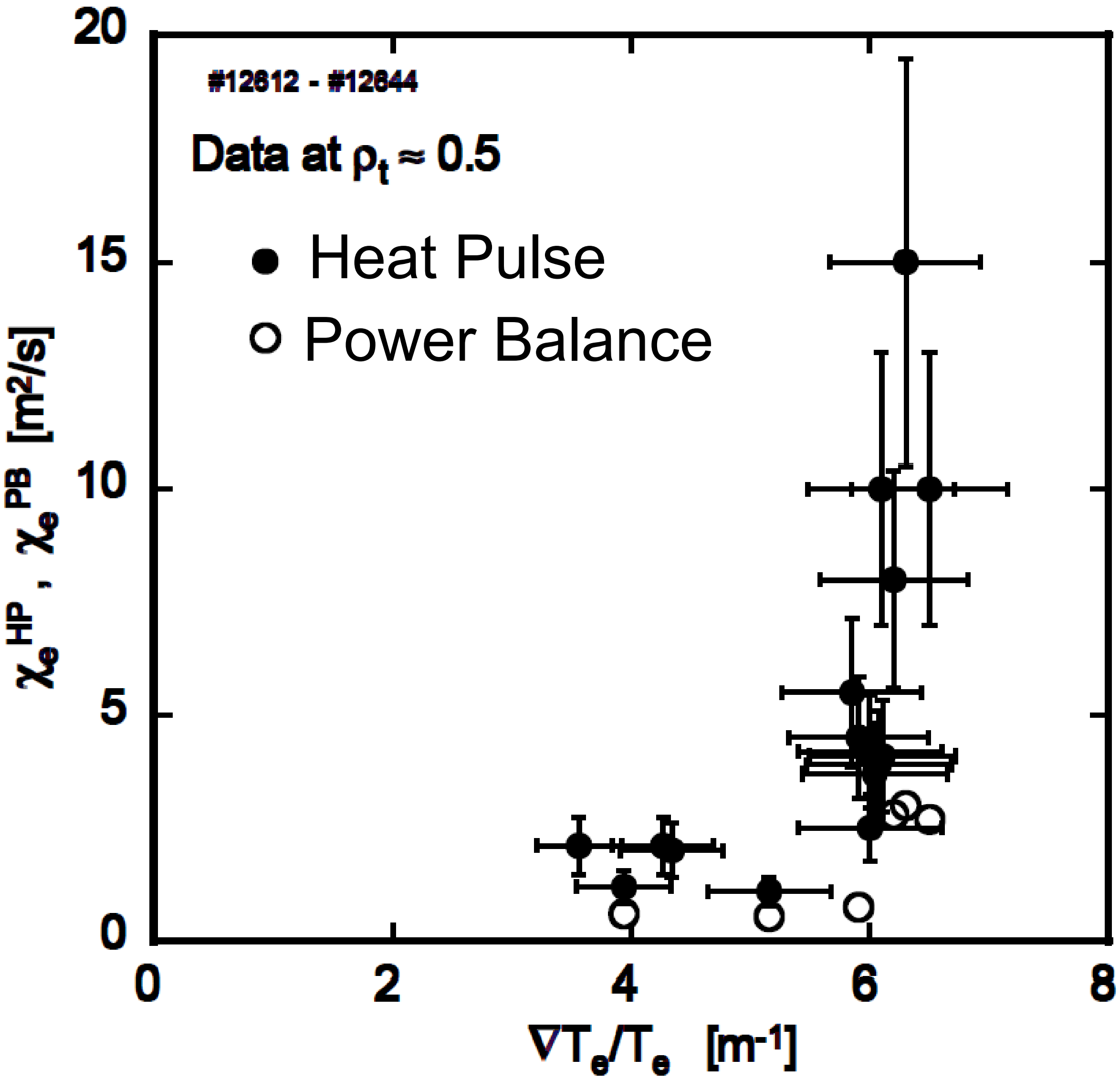
$$\chi_\Phi=\frac{3}{4}\omega_{mod}\left(\frac{\partial \Phi}{\partial R}\right)^{-2}$$

Difference of χ_A and χ_φ due to damping (e.g. e-i collisions)

Heat pulse transport coefficient: $\chi_{HP}=\sqrt{\chi_A\chi_\Phi}$ Example: $\chi_{HP}=0.8\,m^2s^{-1}$

Turbulent Heat Transport has Threshold => stiff T-profiles

- Threshold behaviour
- Strong transport as soon as threshold is reached
- Threshold characteristic for turbulence
- Sufficient heating leads to T-profile that features the critical gradient everywhere



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How to Break These Basic Rules for n and T => Get More Confinement

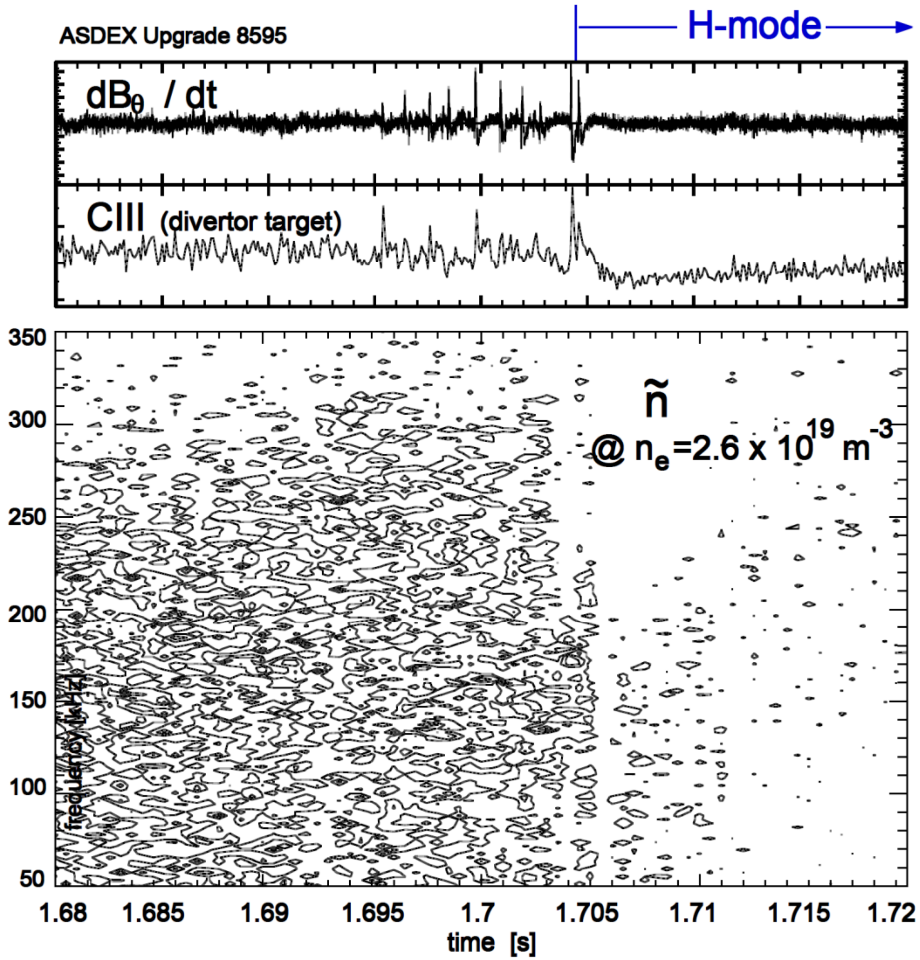
- Suppress turbulence: ‚transport barriers‘

- Optimize parameters that influence turbulence

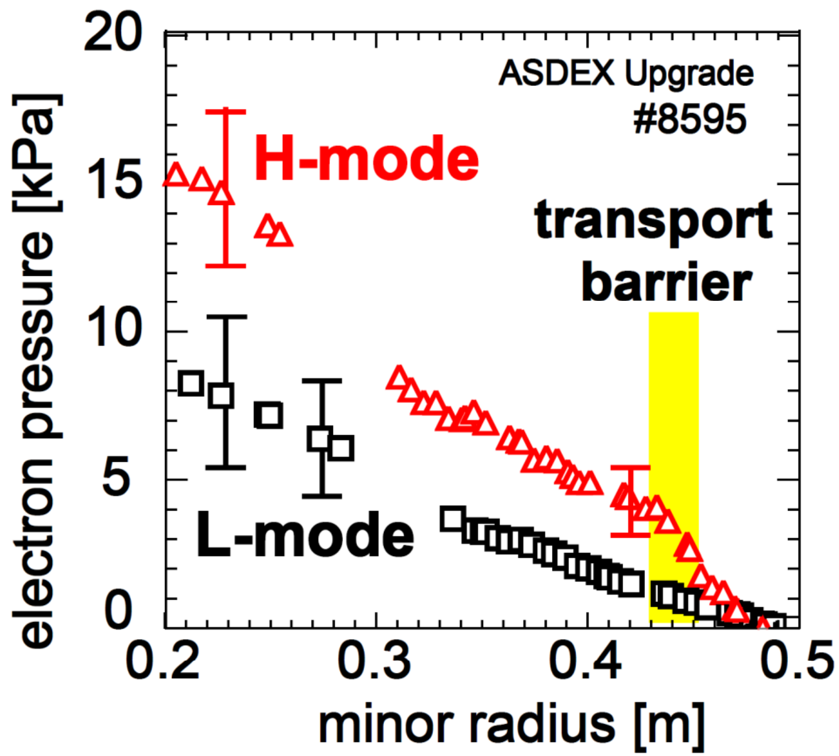
Edge Transport Barrier = H-mode

1984 ASDEX:

Transition to H-mode = state with reduced turbulence at the plasma edge



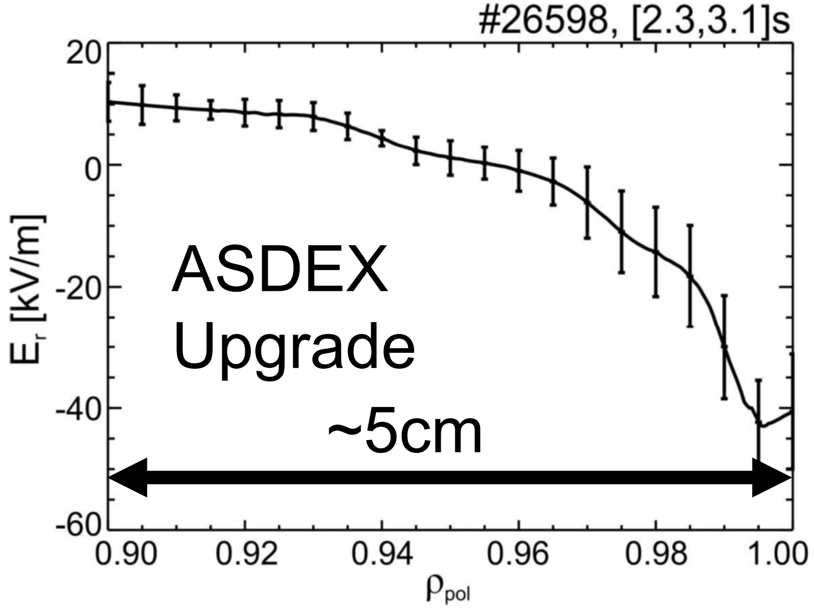
Formation of an edge "transport barrier" = steep pressure gradient at the edge

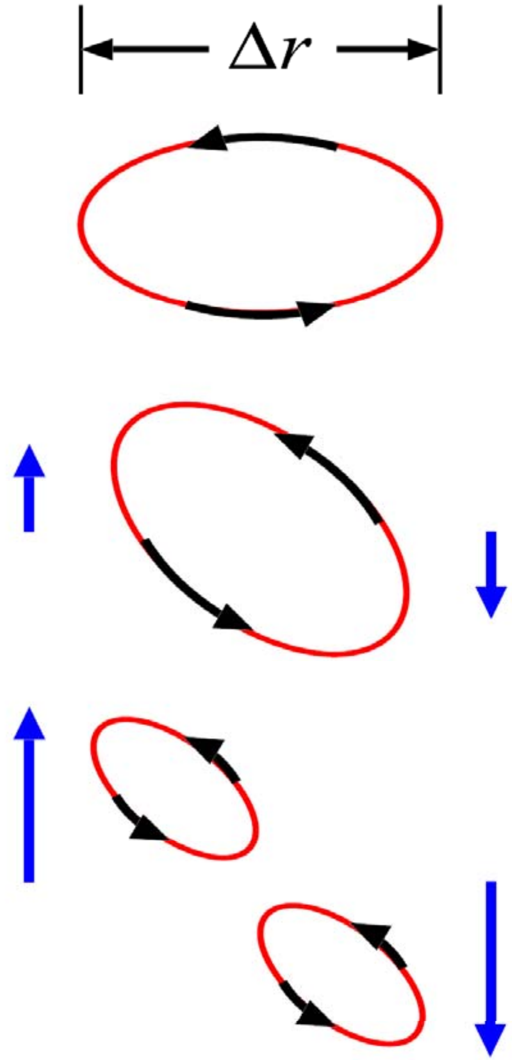


Reduction of transport coefficients to "neoclassical" level often found

Edge pressure limited by stability

Model: Turbulent Eddies are Decorrelated/Strained





- Radial electric field means perpendicular flow
- Gradients in Er means shear in flow
- Highest shearing rates correlate with steepest pressure gradients
- Internal transport barrier also possible

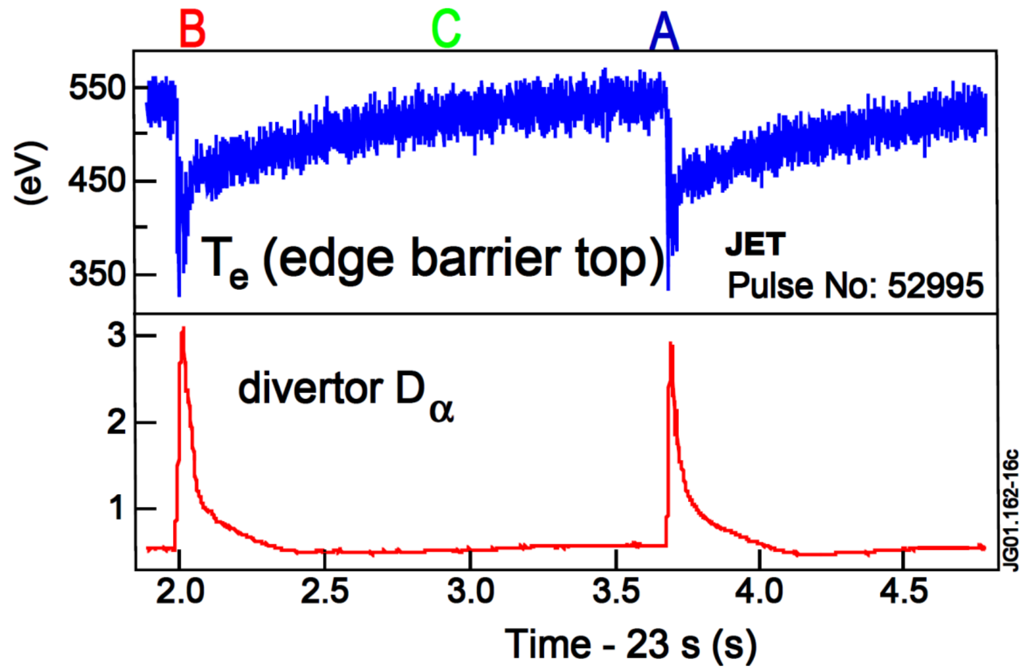
H-mode comes with ELMs

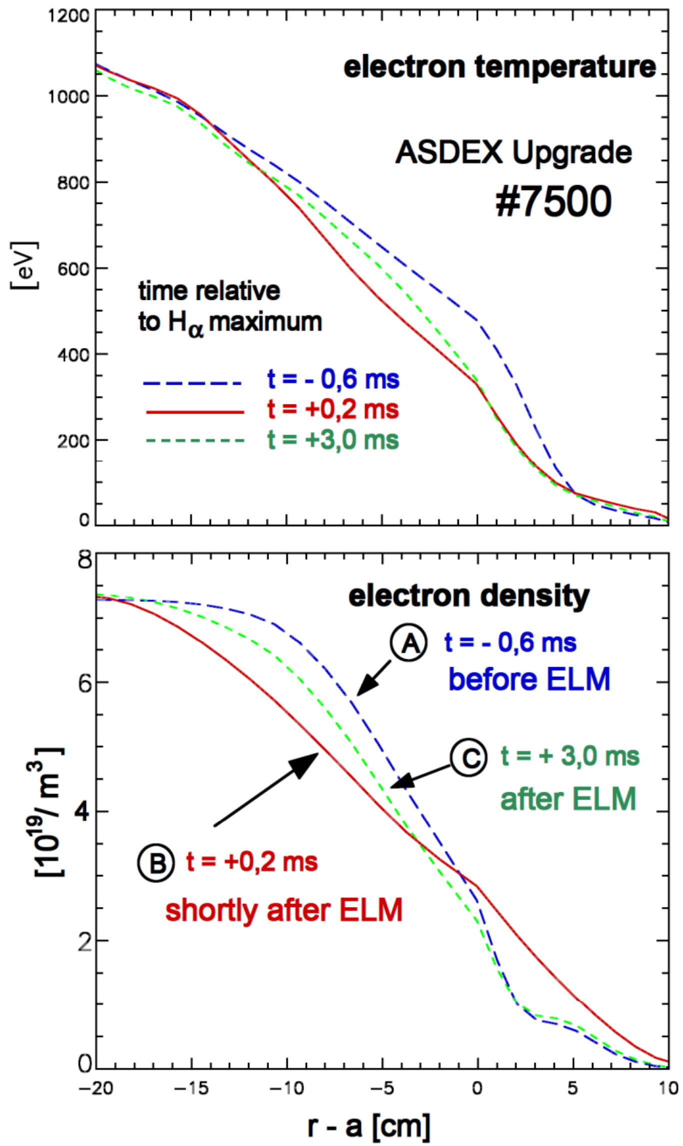
ELM oscillations:

A. Critical ∇p in H-mode barrier region reached
$\rightarrow$ short unstable phase (ELM event)

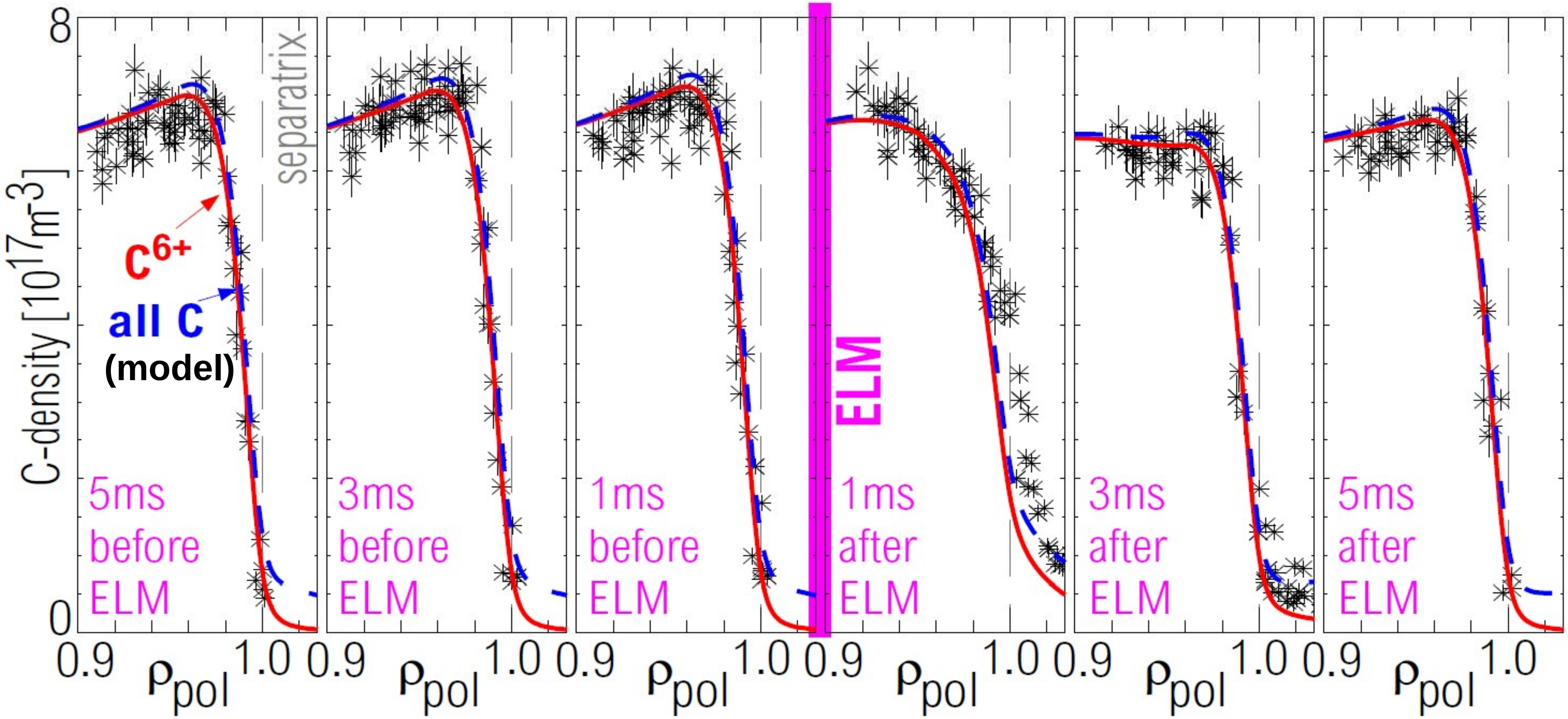
B. Energy and particle loss reduces gradients

C. Gradients build up during reheat/refuelling phase





Transport Barrier require Impurity Control



- Transport Barrier makes neoclassical transport visible
- Inward pinch ~Z => strong gradients
- Need Flushing of Impurities – In this case ELMs are good!

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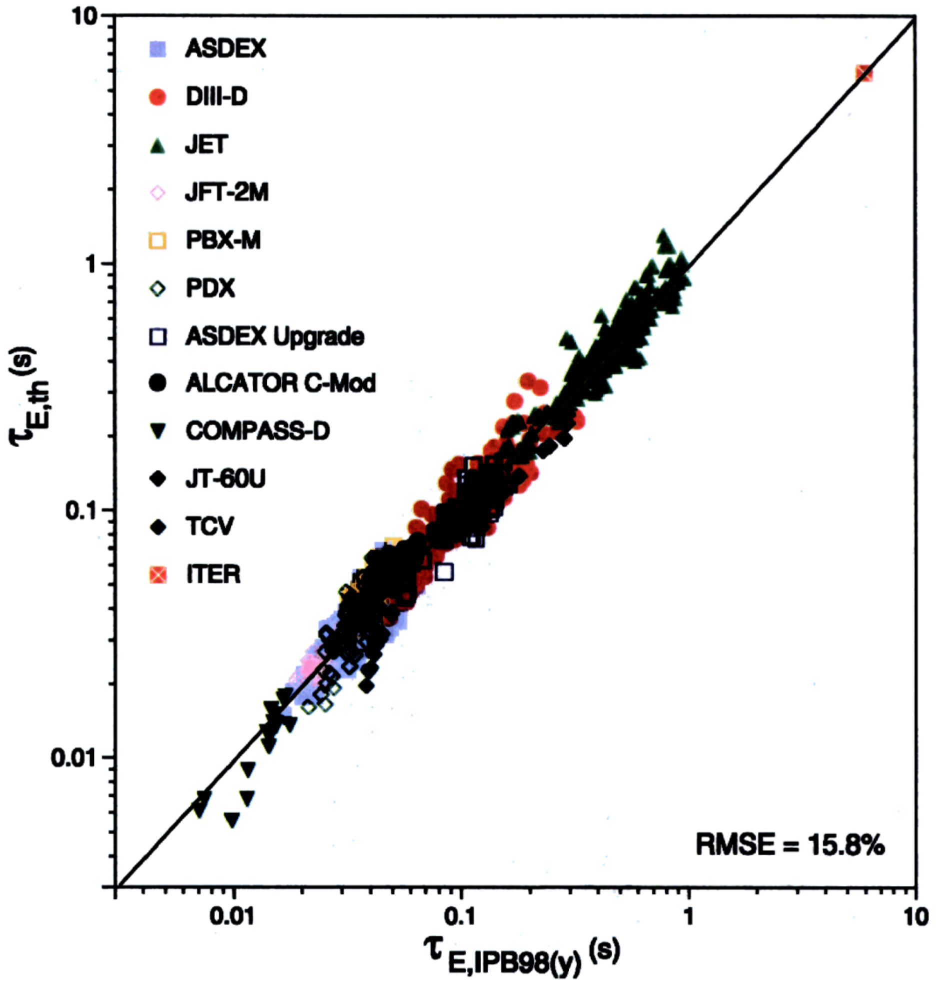
Multi Machine Database for Predictions (no change in phyiscs assumed)

$$\tau_{E,th}^{ELMy} = .0365\, I_p^{0.97}\, B_{tor}^{0.08}\, P_{heat}^{-0.63}\, n^{0.41}\, M^{0.2}\, R^{1.93} (a/R)^{0.23}\, \kappa^{0.67}$$

- Scaling formula derived from multi-machine database
- ITER performance predicted for H-mode with divertor
- I_p, R and κ elongation important!
- B_t not important, but expensive!



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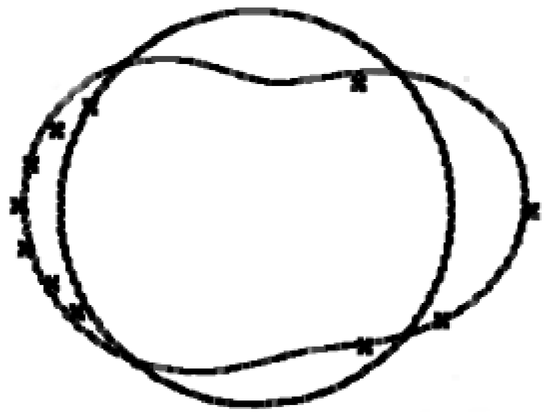
Minimizing B_t for Fixed I_p Means Minimizing q

- Rational Surfaces are adding up error fields
- rational q-Number q=m/n more instable for small m and n
- q=2 at the plasma boundary is locking with wall structure
  - ⇨ ‘external kink‘
  - ⇨ Lock of mode rotation = fast mode growth
  - ⇨ disruption

Poloidal view of an m=2 and n=1 external kink



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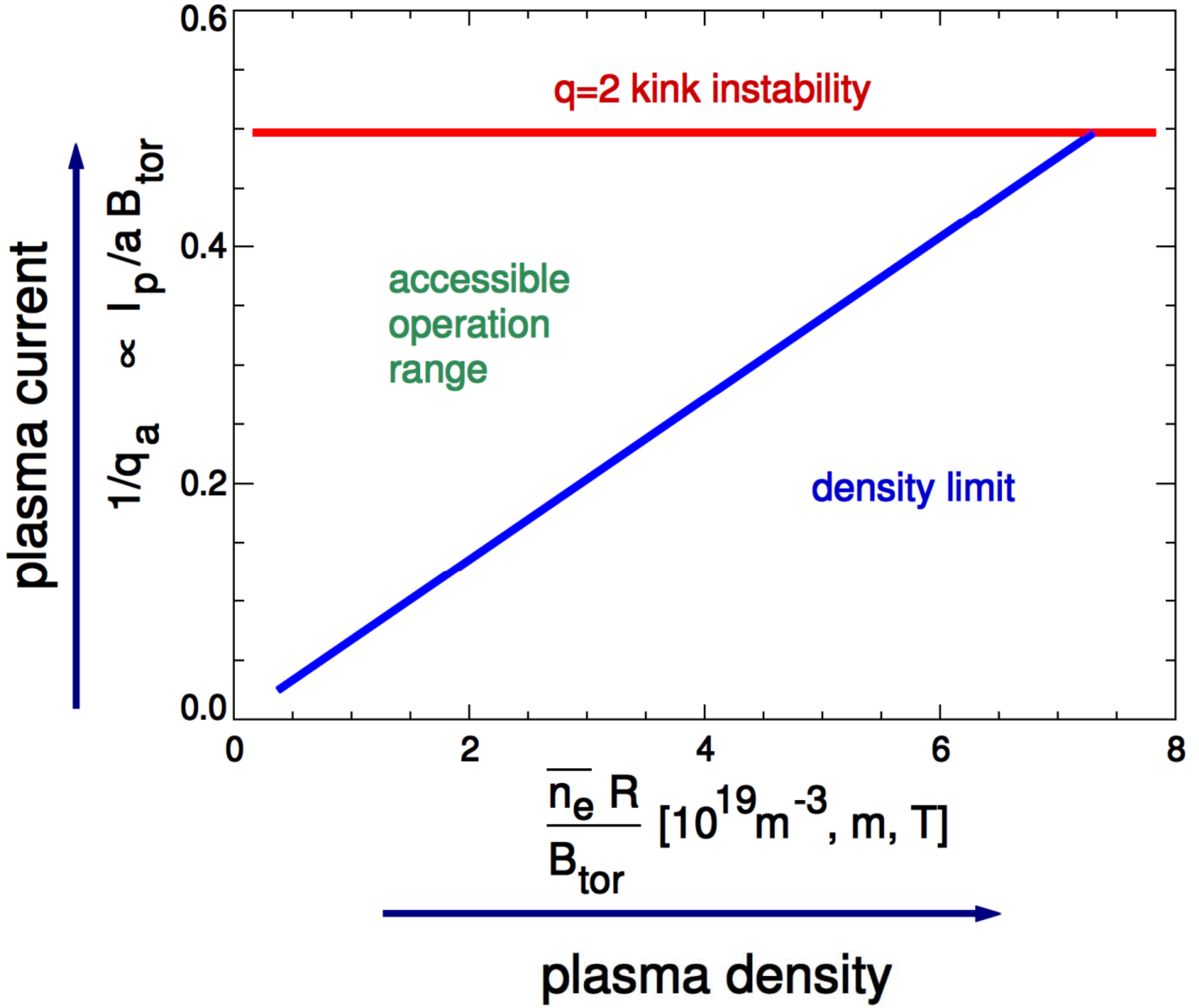

- Hugill diagram
- Combining density limit and external kink q=2

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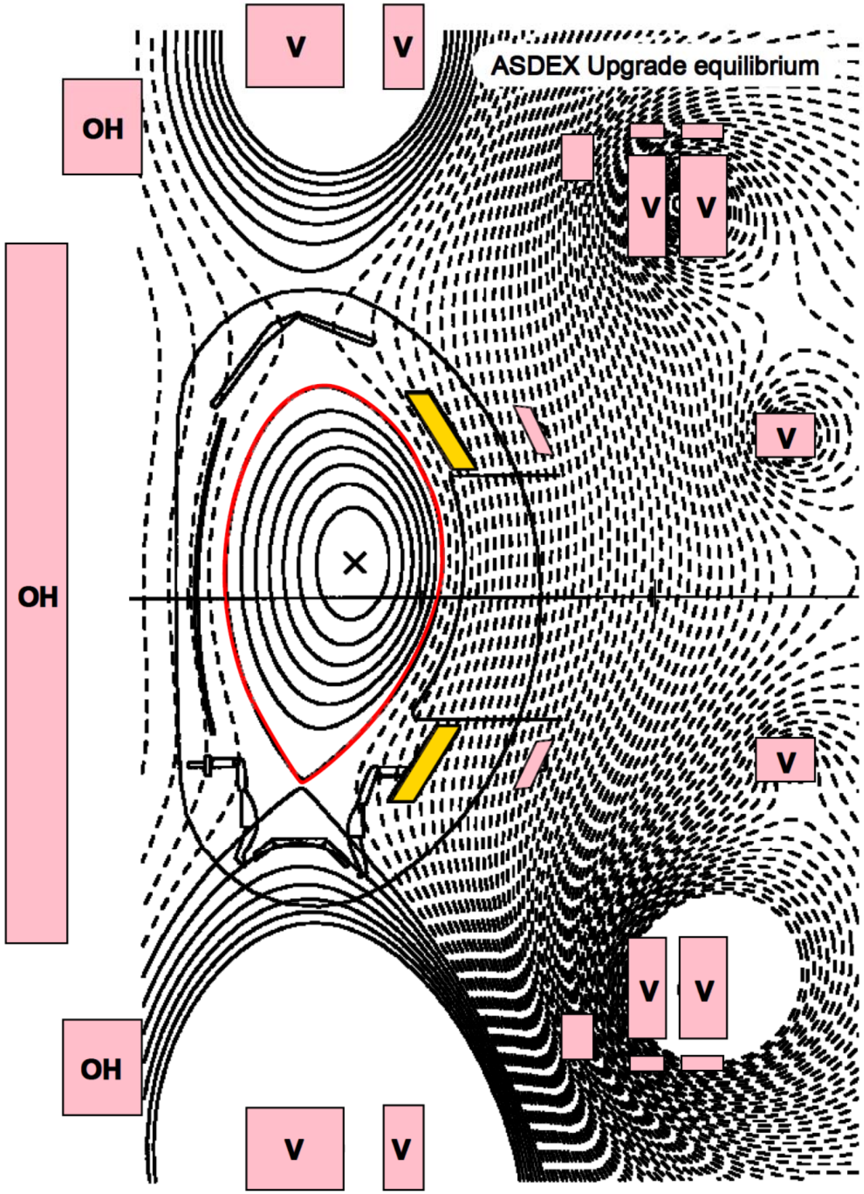
Divertor Results From External Magnetic Fields



- B-field of external V-coils cancel B-field due to plasma current
- Field null, X-points
- Other V-coils determine shape and position of plasma

Grad-Shafranov equation:

$$-\Delta^{*}\Psi = \mu_0 (2\pi R^2) p' + \mu_0^2 I_{pol} I_{pol}'$$

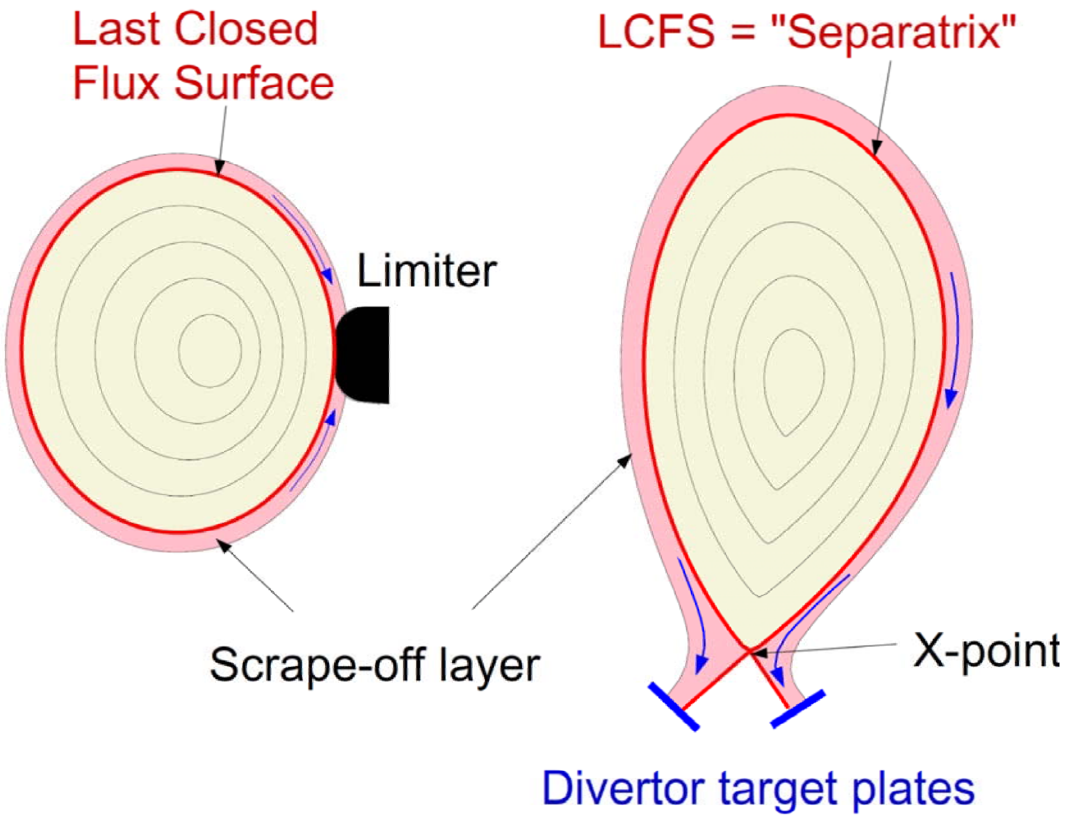
Limiter vs Divertor Configuration

Limiter:

Material structure defines plasma…

Divertor:

Magnetic X-point introduced by quadrupole field coils

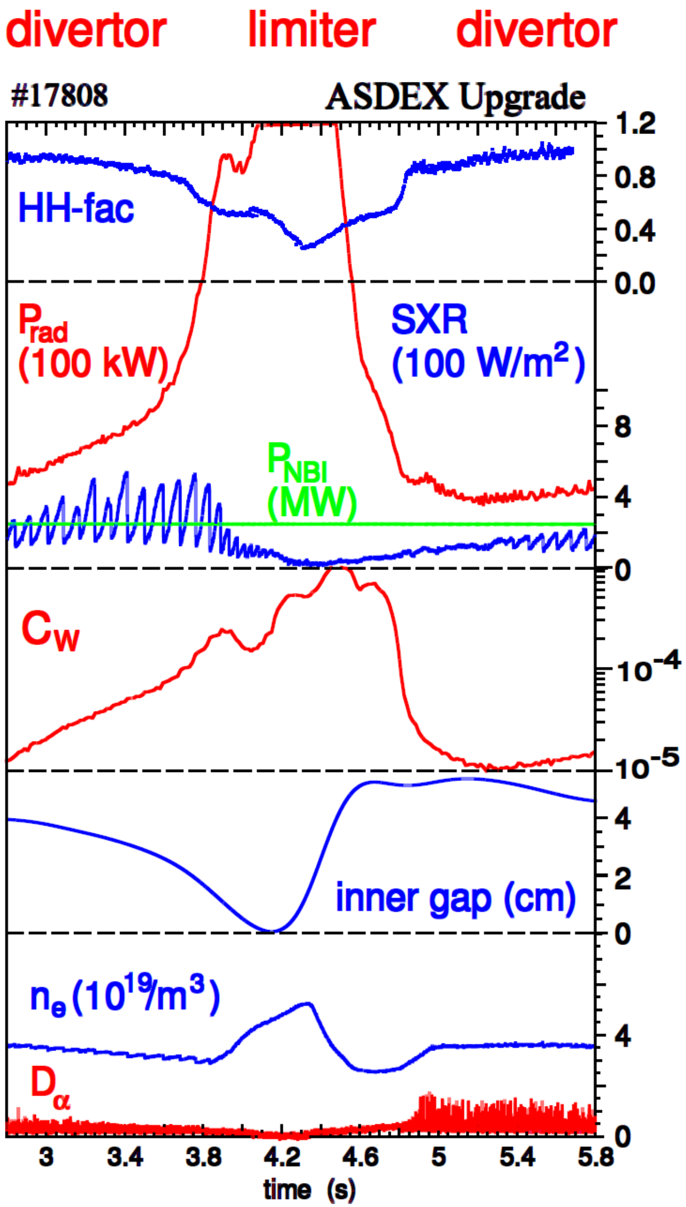


<u>Advantages of Divertor:</u>

- Plasma-Wall interaction further away from main plasma
- High density (recycling) leads to low temp., as pressure constant on field lines
- Access to H-mode

Cold Divertor Visible in Video Camera

Tungsten (W) Wall only Possible with Divertor



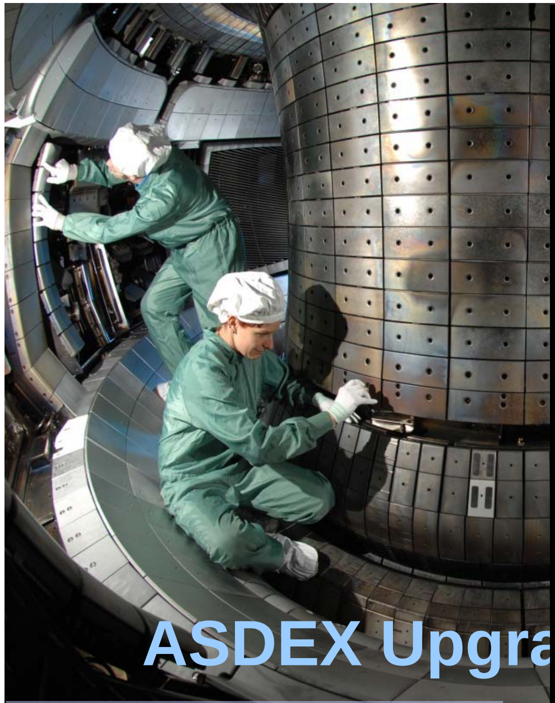
- Divertor configuration cleans up plasma
  - ⇨ Beneficial for low-Z, but crucial for high-Z
- All major Tokmakas studying reactor relevant plasmas have a divertor (H-mode access)
- ITER has a divertor

Tungsten in AUG and JET-ILW

Since 2007:
All W-wall

Since 2011:
W-divertor

Tungsten in AUG and JET-ILW



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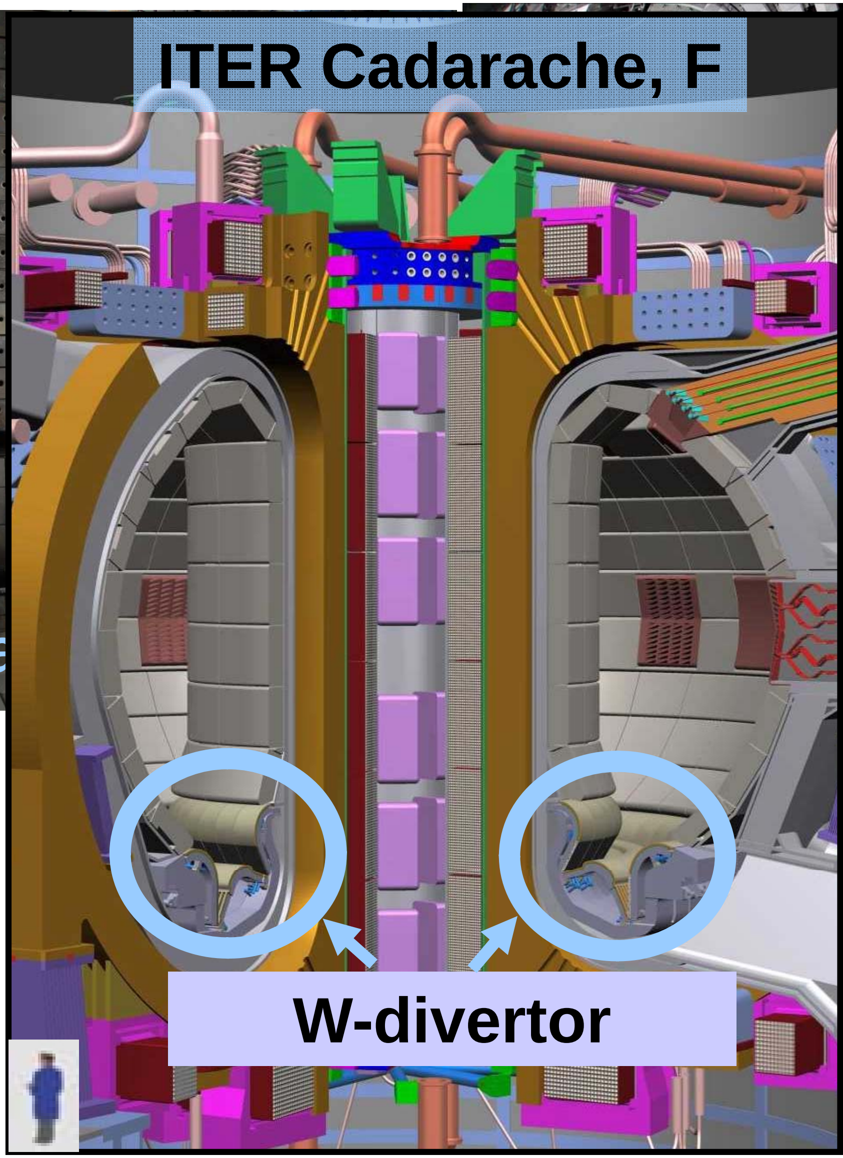
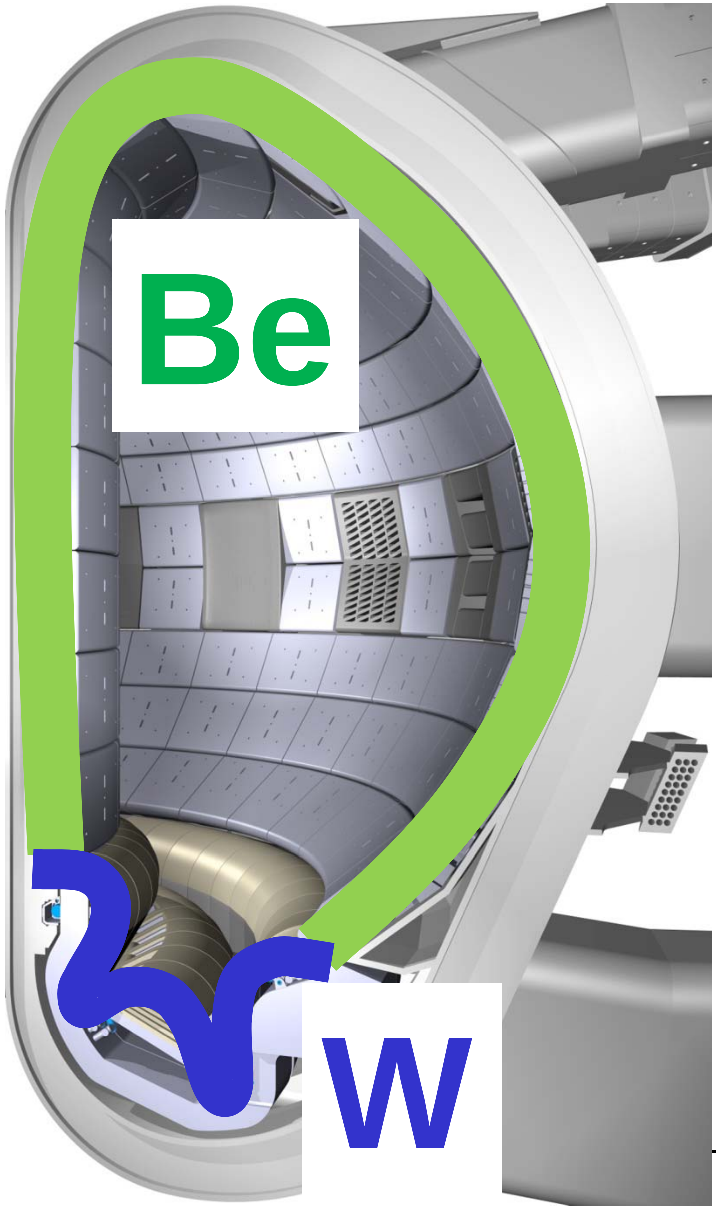


Since 2011:
W-divertor

ITER-wall mix (now investigated at JET)



- W limits erosion
- W can handle the power
- T storage tolerable

- Be getters oxygen
- Not compatible in reactor
- Main chamber impurity penetrates well into plasma
  - ⇨ Be safe choice for ITER

- Carbon not possible: erosion, T storage by codeposition

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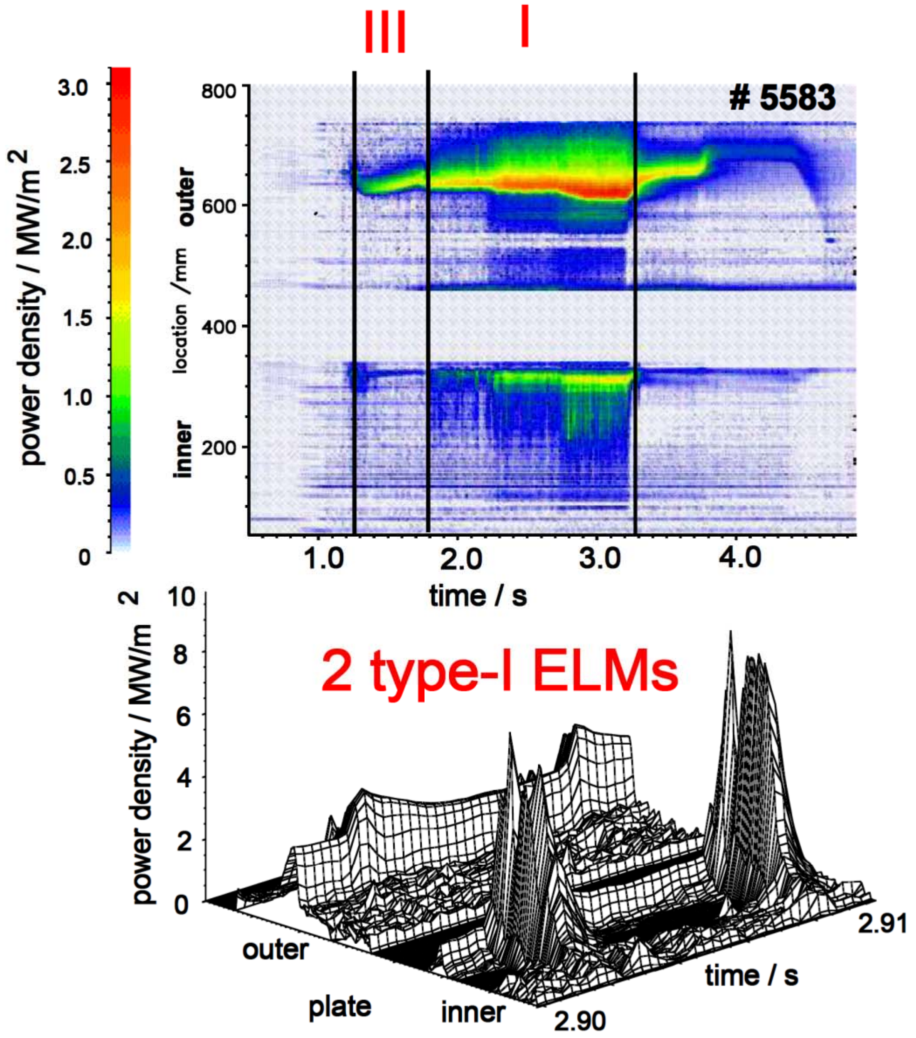
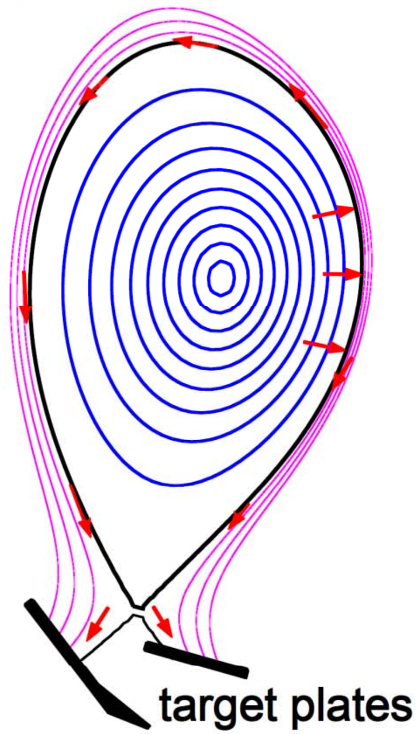
ELMs must be Mitigated to Protect Wall

- significant power deposited on divertor plates
- important to find ways to mitigate ELMs



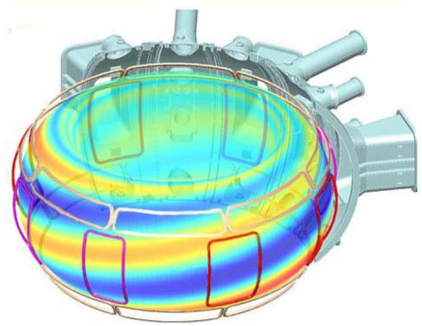


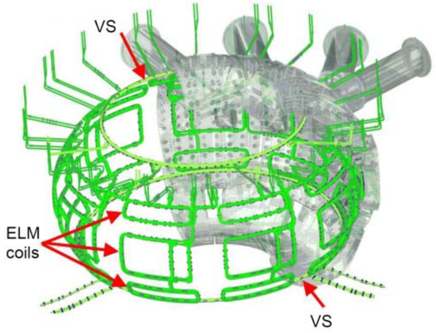
Different ELM types exist:
"Type I" (large)
"Type II, III" (small)

Combine highest pedestal pressure with small ELM losses

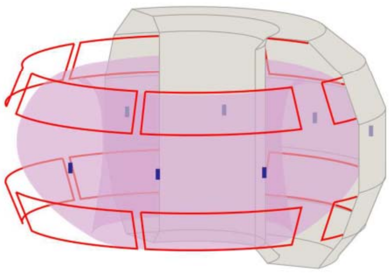
An example of ELM mitigation using magnetic perturbations (10^{-3})

- Perturbation coils have been implemented in many devices
- Effect on ELMs not straightforward

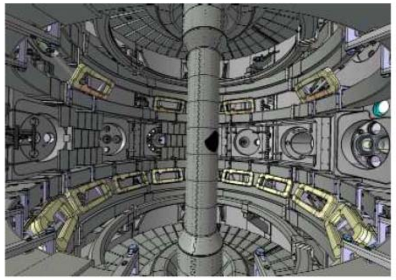




ITER

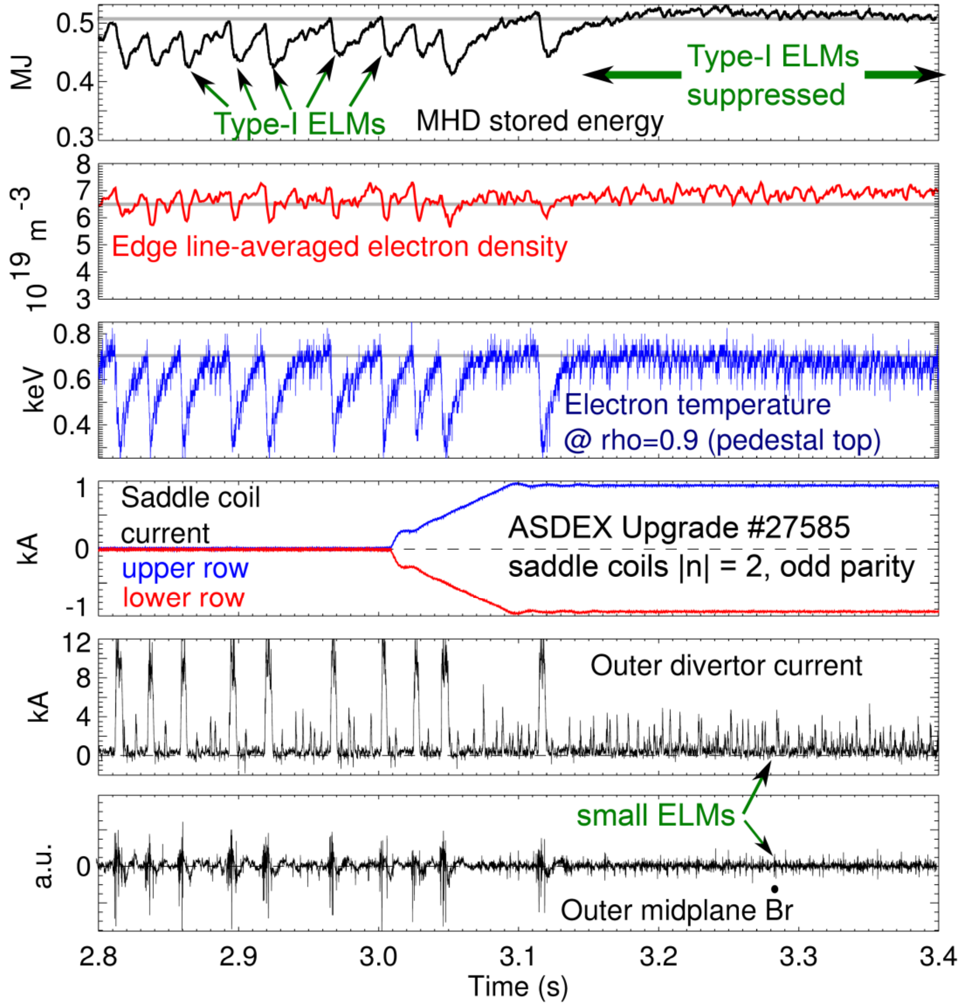


DIII-D

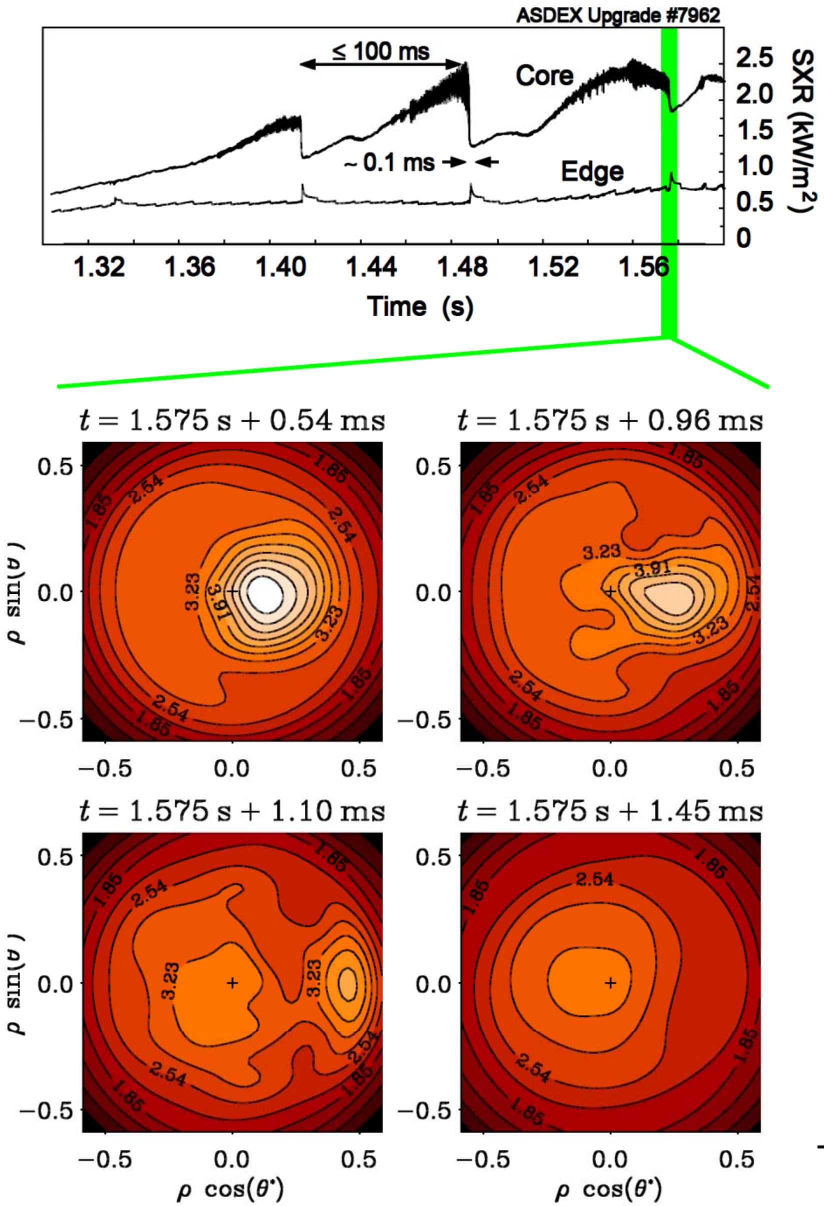


MAST

+ EAST, JET, KSTAR, NSTX



Central (1,1)-Mode ‚Sawteeth'

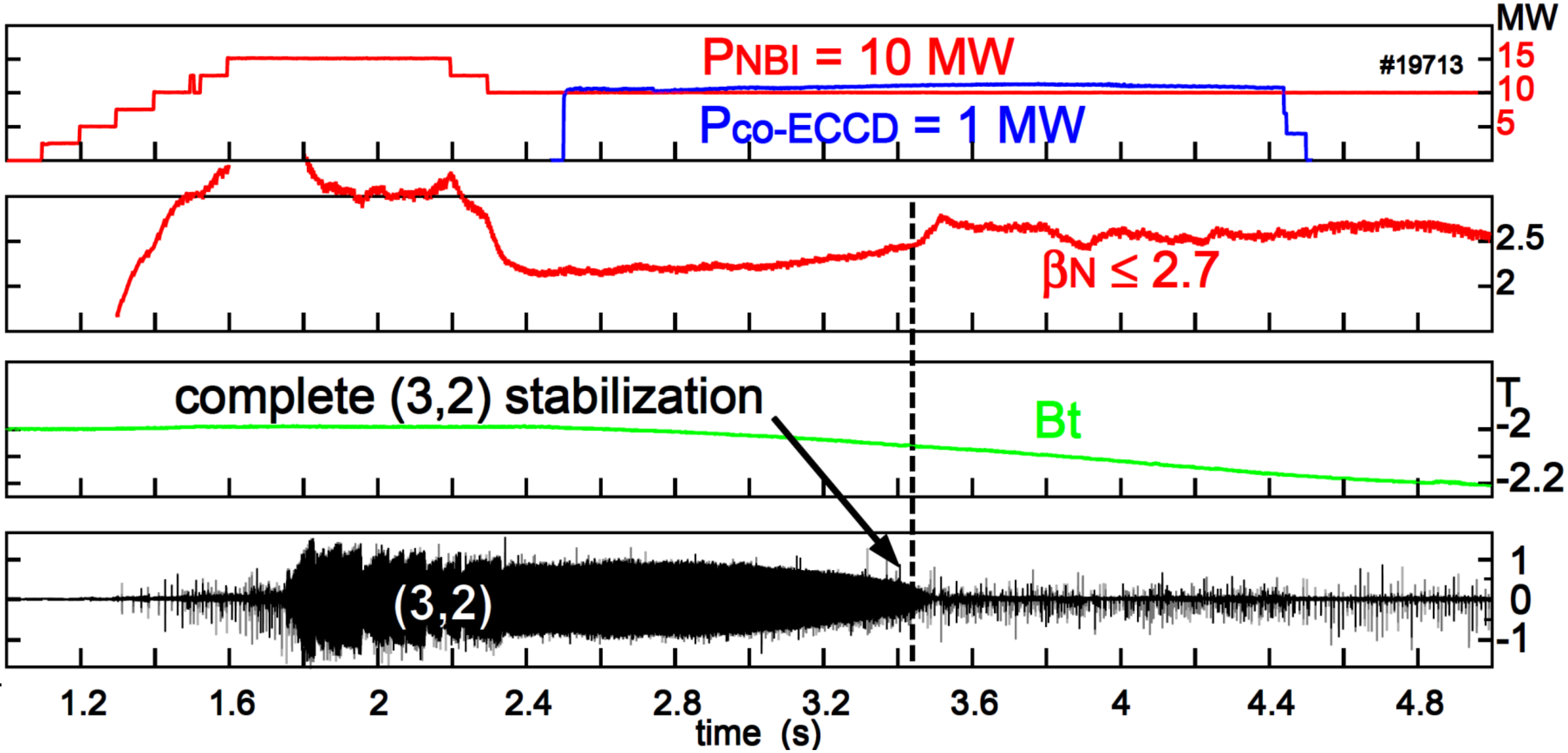


- Stirs up core plasma
- Considerable T-drop + current redistribution
- Provides Seed-Islands for NTMs
- In ITER/reactor:

Possibly removes alphas from core

Neoclassical Tearing Modes

- NTMs triggered where much bootstrap current exists
- Seed-island (sawtooth!) means lack of bootstrap-current
- Lack of current increases island size
- Can be repaired by current drive within Island



Outline

- Introduction: Tokamak Principle – Why tokamaks anyway?
- Why does ITER look like it looks like?
  - maximizing $nT\tau$
  - what is the minimum B_t?
  - wall contact & material
- Additional Issues
- Outlook to Reactor

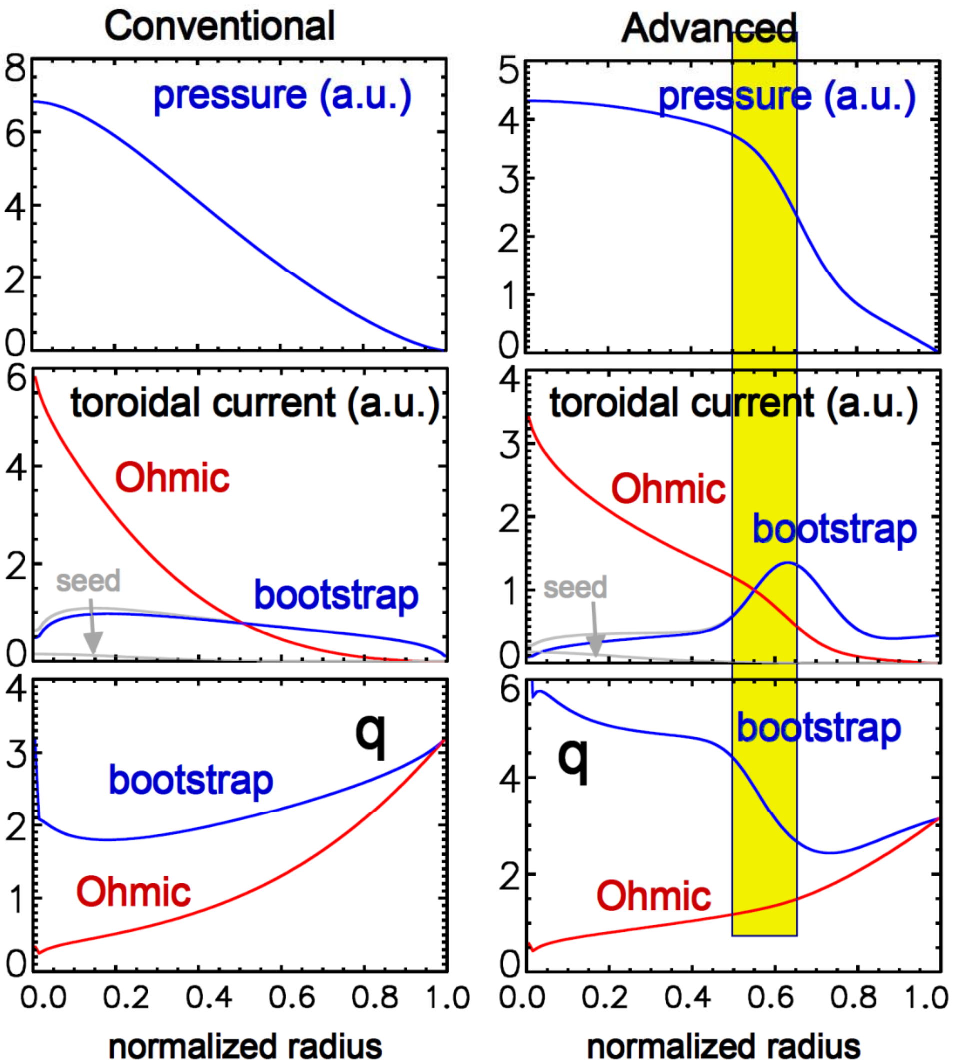
Outlook to Reactor

- Reactor must Run 24/7
- Thus requires a robust wall – probably high-Z
- Thus power exhaust must be solved
- Thus pulse length must be considerable, if tokamak:
  - Bootstrap current important (advanced tokamak)
  - Efficient Current drive (high T)

- Requires highest possible fusion yields
  => density above Greenwald desireable

- Structural Materials must be stable + recycable

Advanced Tokamak



High pressure gradient in transport barrier

↓

Off-axis "bootstrap" current

↓

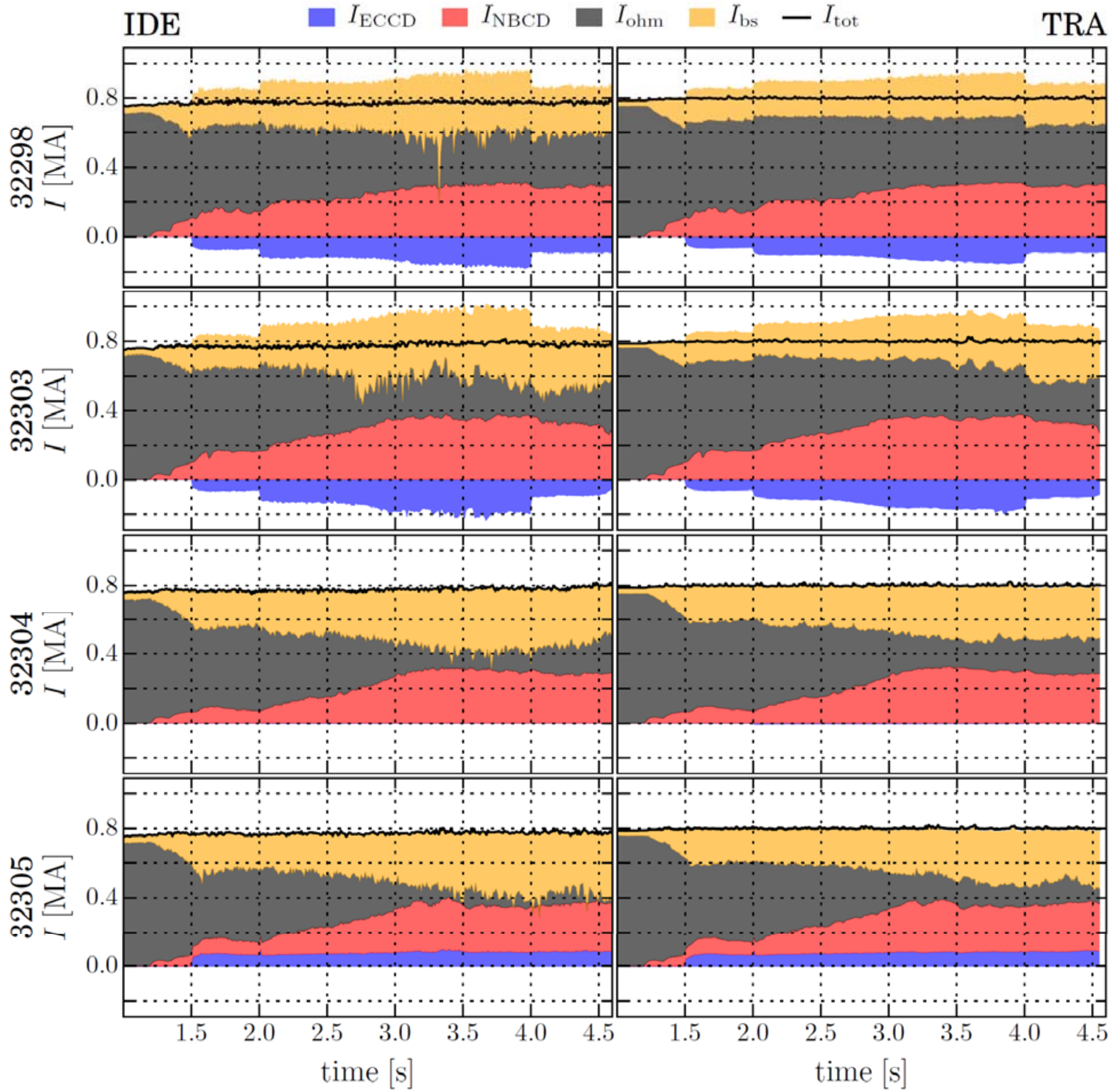
q profile associated with BS current creates negative central shear

↓

Negative shear stabilises turbulence → reduced transport

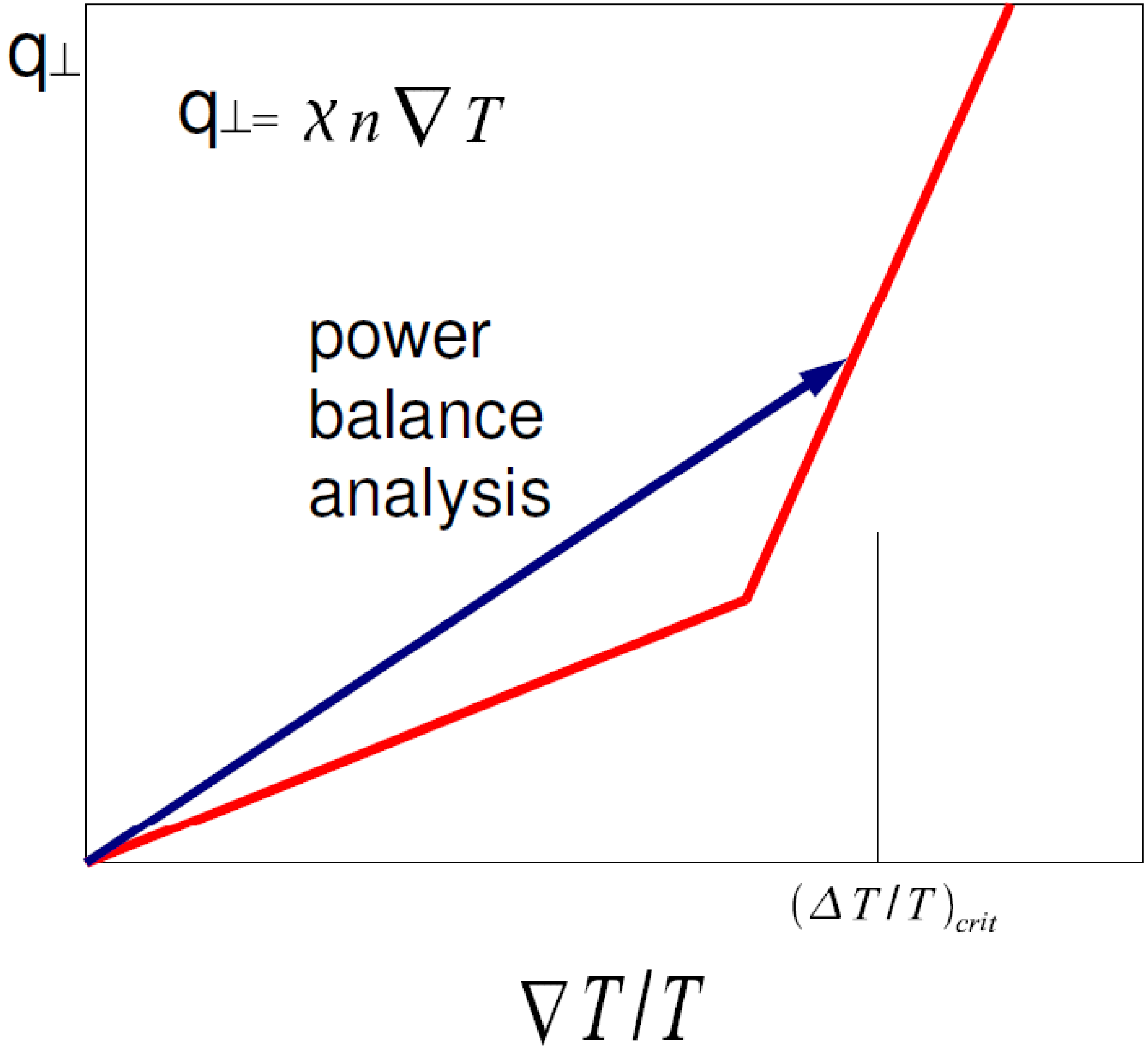
"Steady-state" tokamak without inductively driven current ?

Advanced Scenario in Experiment



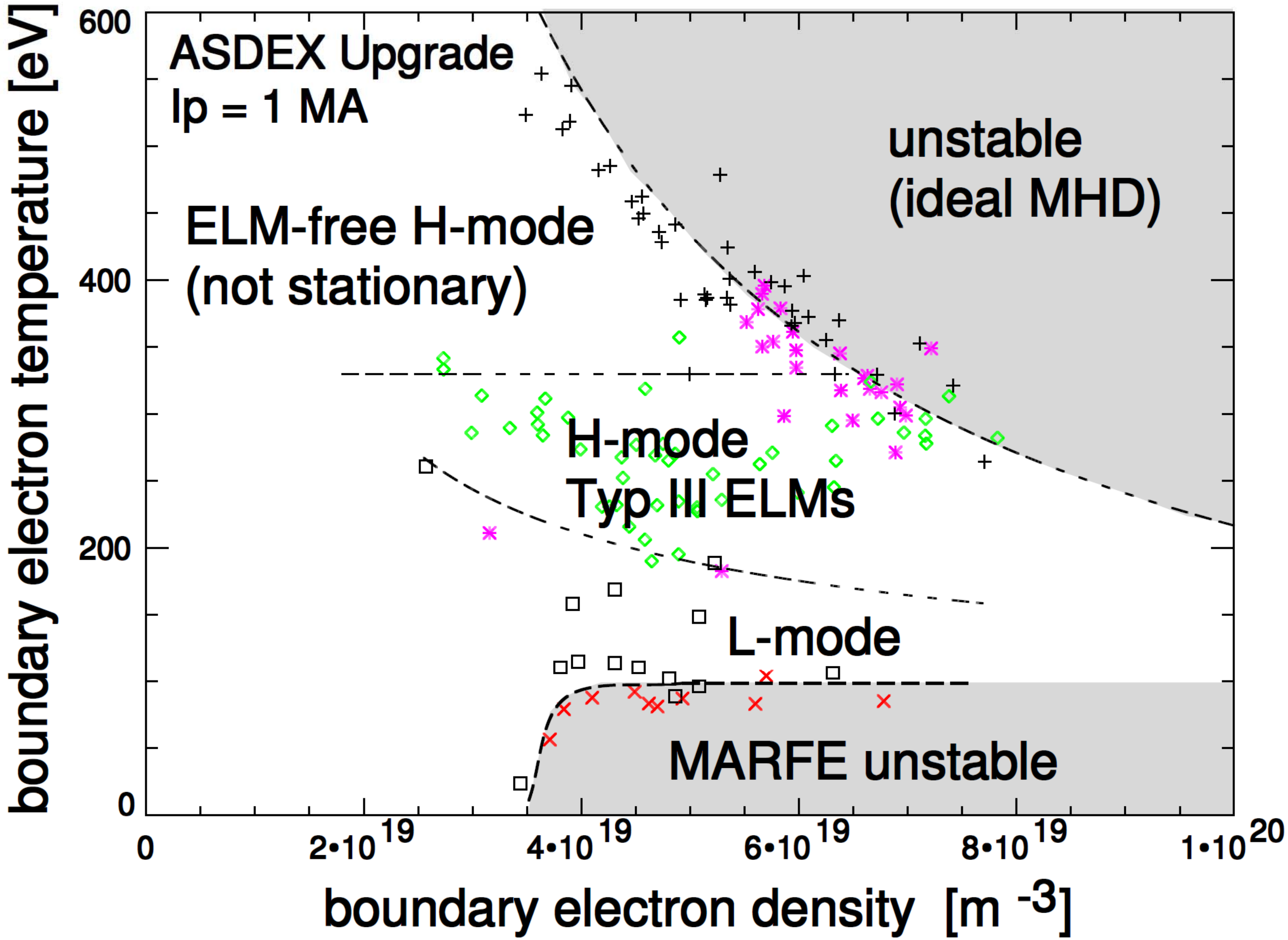
- IDE & TRA are two different analyses
- Various discharges
- Current drive (ECCD, NBCD)
- Bootstrap current BS
- Last: Close to non-inductive

heat transport measurements from power balance ≠ from heat pulse analysis



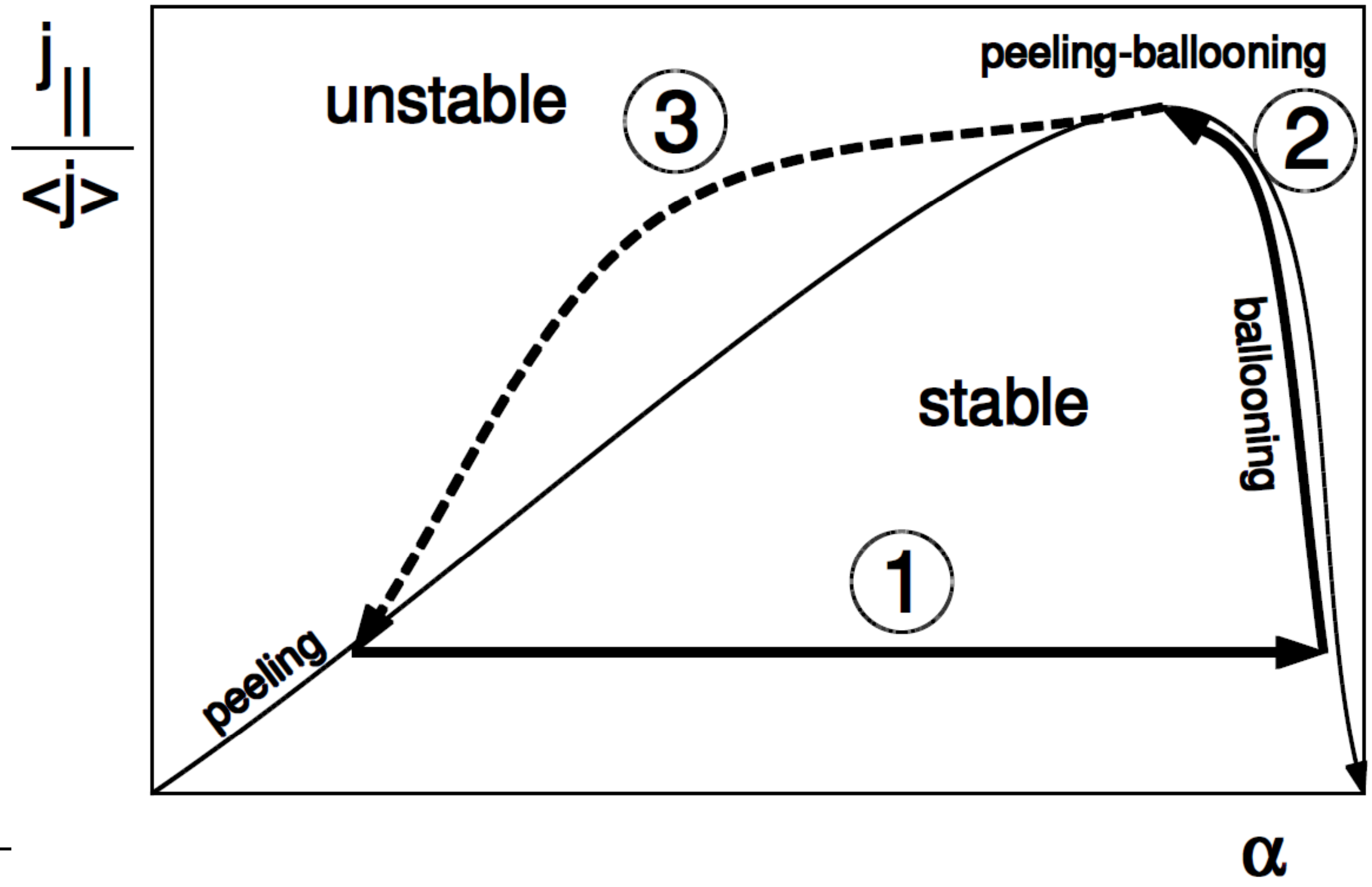
- PB: probing the average chi
- Heat pulse probing the local slope

ELM-types in the Operational Space in AUG



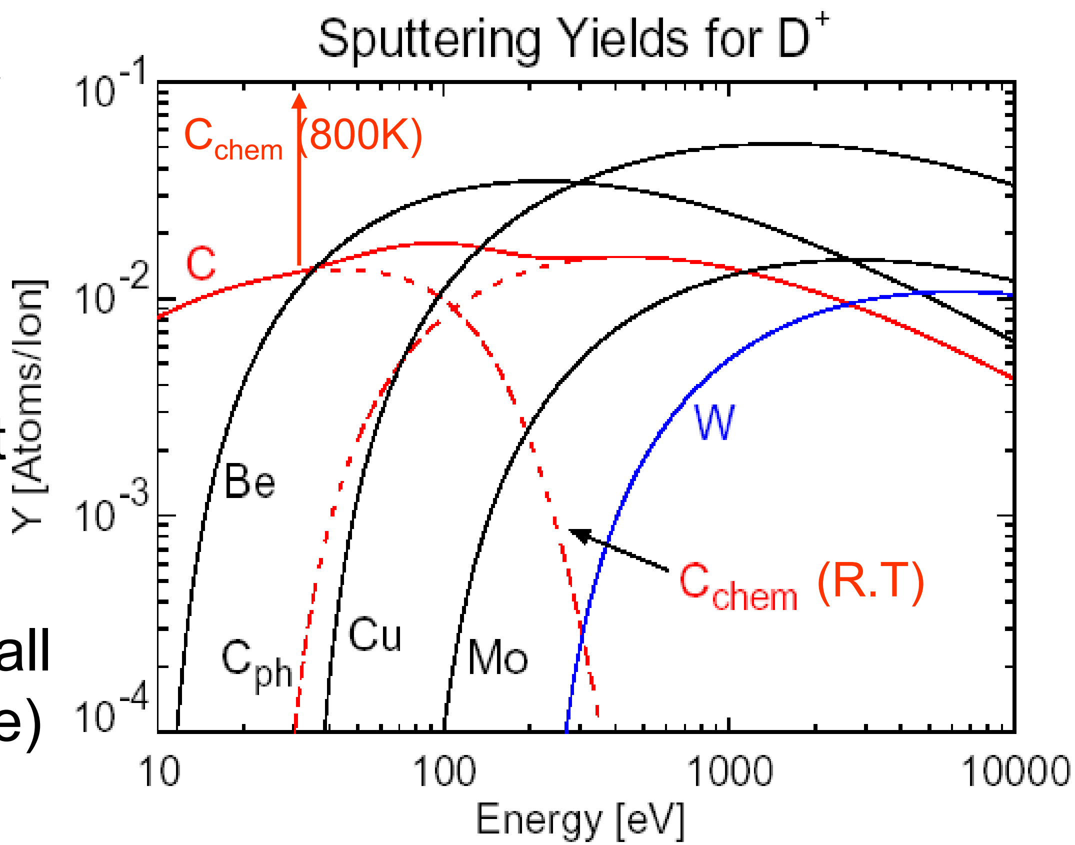
Peeling-Balooning Model for ELM-cycle

- 1: Pressure gradient builts up quickly
- 2: pressure gradient limited by balloing inst., edge-j increases
- 3: ELM crash (kink, peeling inst.) resets conditions



Tungsten is Worthwhile the Effort

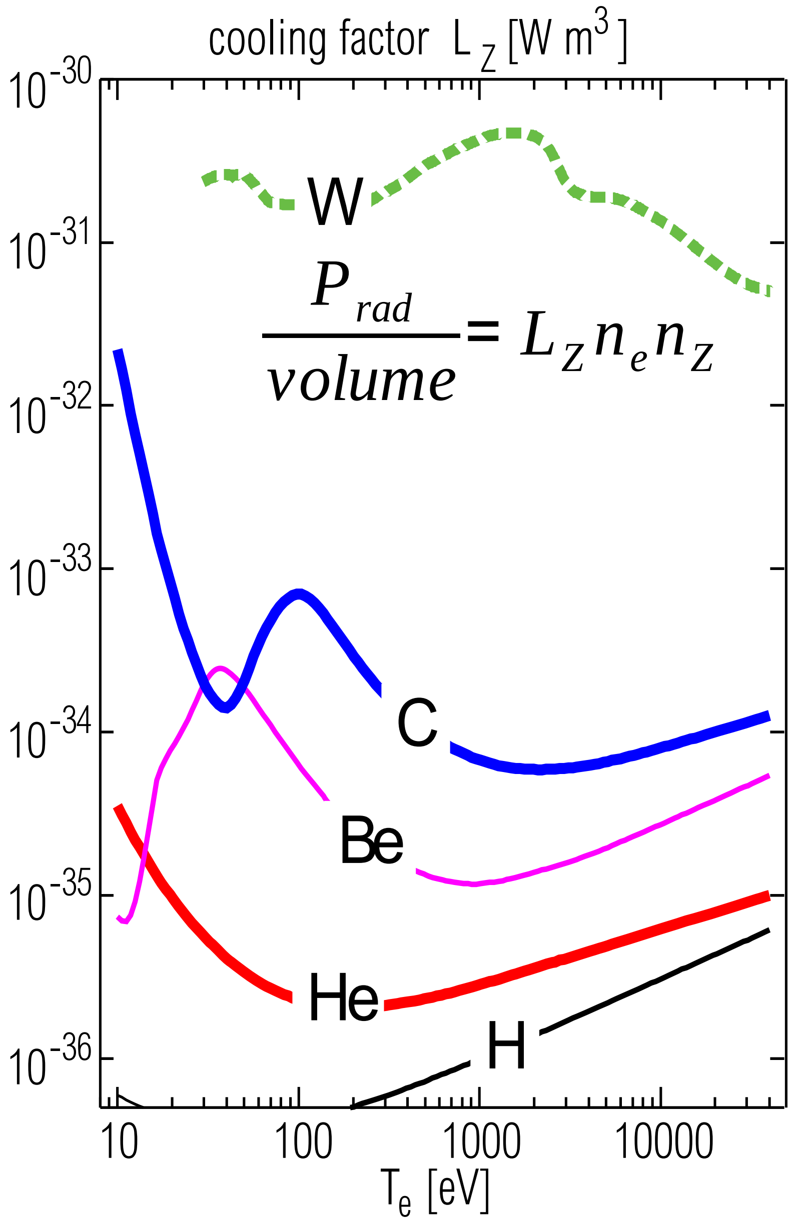
- First wall erosion needs to be very small (Reactor operation 24/7 requires high-Z wall)
- W can handle high heat fluxes (metal with highest melting point)
- Fuel retention in W is small (Co-deposition is no issue)



⇒ **W is a plasma impurity**

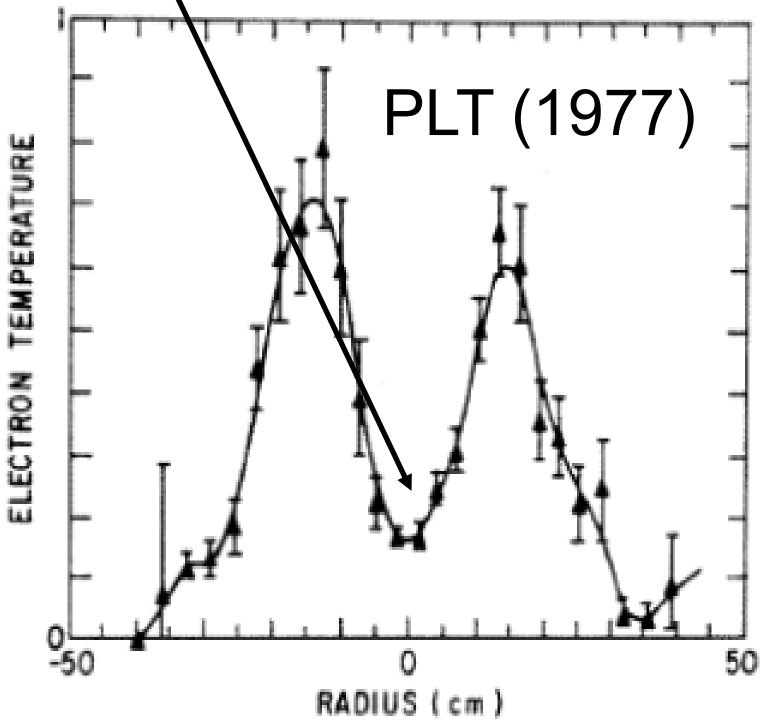
⇒ **Radiative Cooling due to W demands caution**

Caution: Tungsten May Radiate in the Plasma Core



- Transport non-favourable
- High cooling factor at high T_e
- Radiative cooling also in central plasma

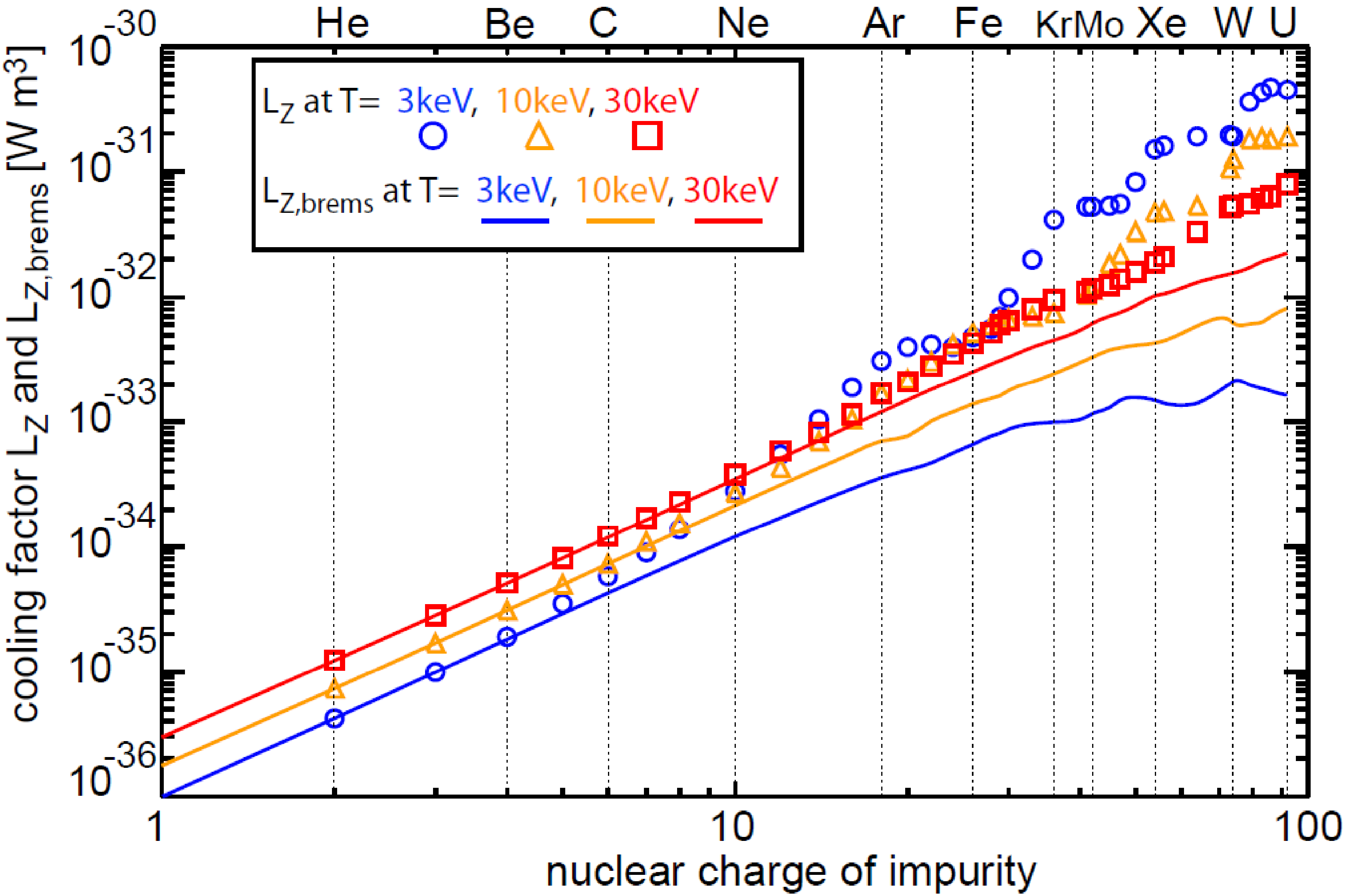
issu



V. Arunasalam et al., EPS 1978 Prague

Interior of ASDEX Upgrade – Since 2007 full-W wall

What Type of Impurity Radiation is Important?



What Type of Impurity Radiation is Important?

Cooling Factor for W (best quality is blue dashed)