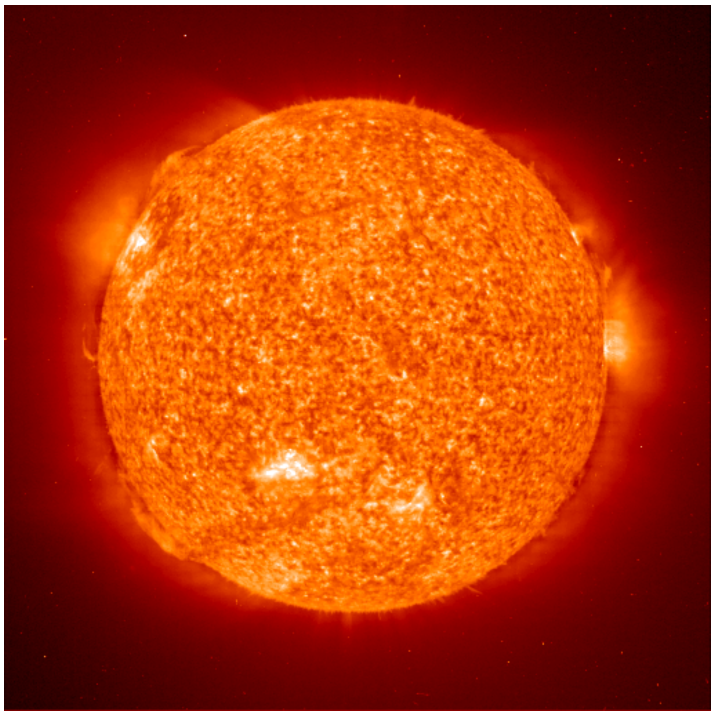
Max-Planck-Institut für Plasmaphysik

Progressing Performance – Tokamak Core Physics



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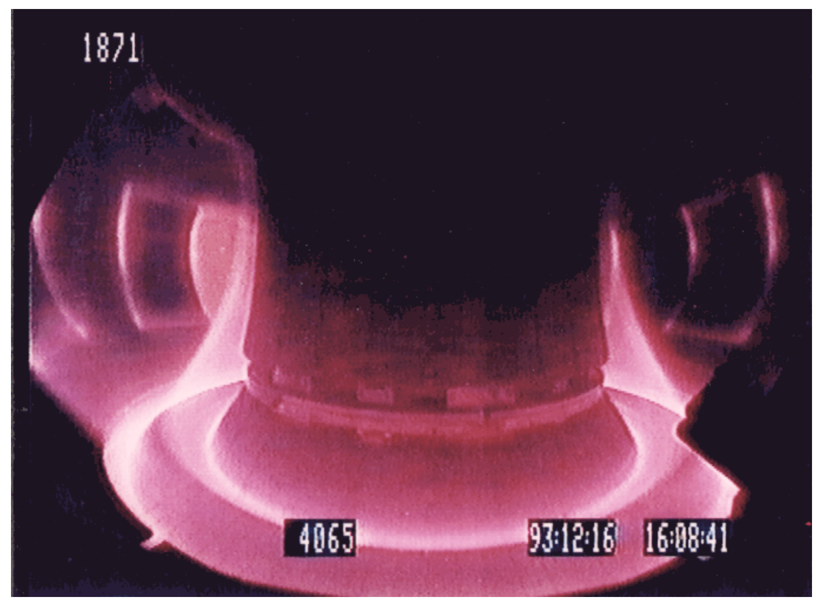


Joint ICTP-IAEA College on Advanced Plasma Physics, Triest, Italy, 2016

- heat insulation (energy transport)
- magnetohydrodynamic (MHD) stability
- tokamak operational scenarios
- exhaust of heat and particles (tomorrow, Wednesday)

- heat insulation (energy transport)
- magnetohydrodynamic (MHD) stability
- tokamak operational scenarios

Reactor energetics: the ‚Lawson' criterion for $n\tau_E$

α-heating compensates losses:

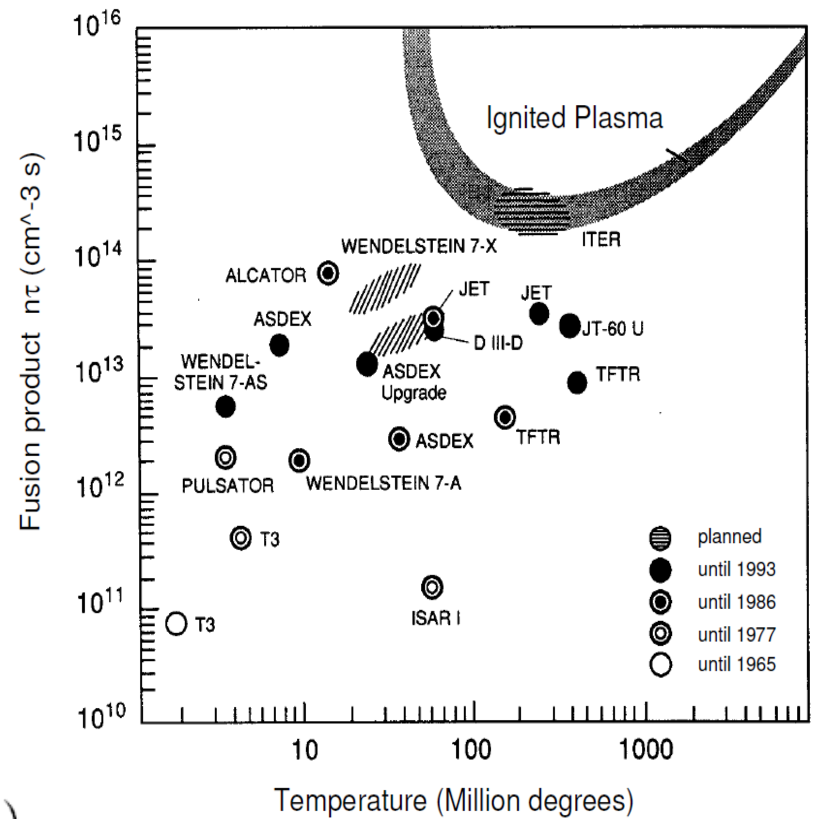
- radiative losses (Bremsstrahlung)
- heat conduction and convection

$$\frac{n_e^2}{4}\langle\sigma u\rangle E_\alpha > c_{Br} n_e^2 Z_{eff}\sqrt{k_B T} + \frac{3 n_e k_B T}{\tau_E}$$

$\tau_E = W_{plasma}/P_{loss}$ **('energy confinement time')**

leads to

$$n_e \tau_E > \frac{3 k_B T}{\langle\sigma u\rangle E_\alpha/4 - c_{Br} Z_{eff}\sqrt{k_B T}} = f(T)$$



which has a minimum for $n\tau_E = 2 \times 10^{20}$ m^{-3} s at T = 20 keV

Figure of merit for fusion performance $nT\tau$

Power P_{loss} needed to sustain plasma

- determined by thermal insulation:
  $\tau_E = W_{plasma}/P_{loss}$ **('energy confinement time')**

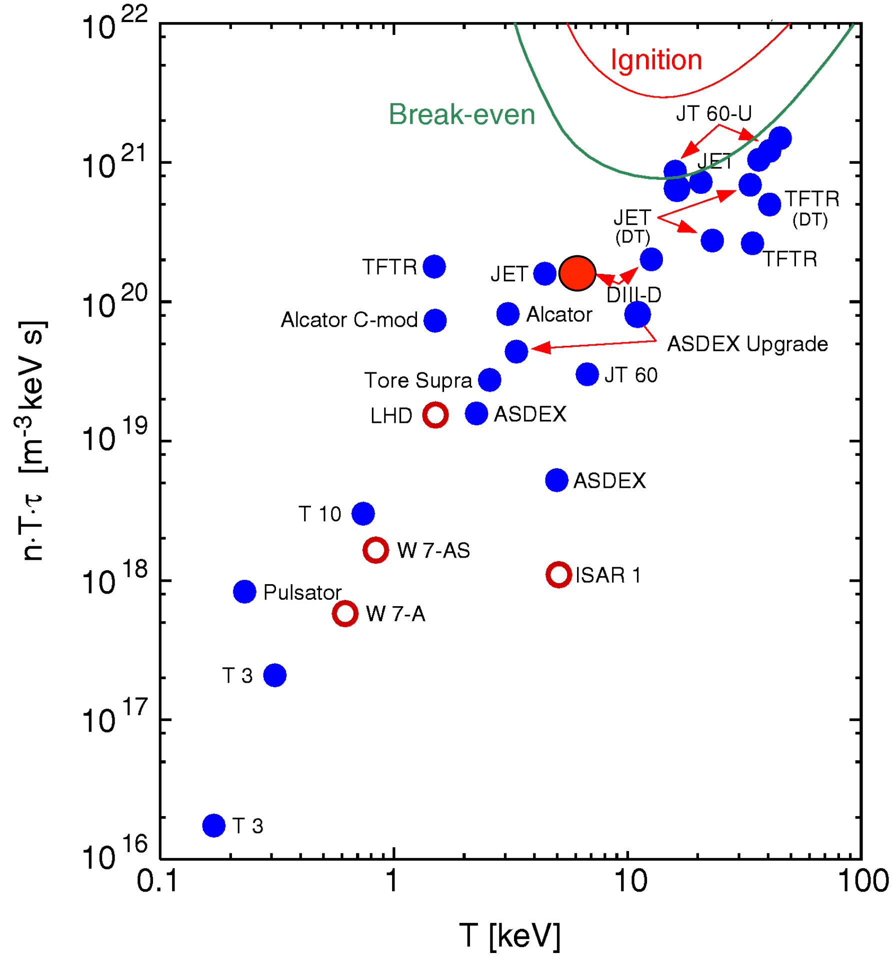
Fusion power increases with W_{plasma}

- $P_{fus} \sim n_D n_T \langle\sigma v\rangle \sim n_e^2 T^2 \sim W_{plasma}^2$

Present day experiments: P_{loss}
compensated by external heating

- $Q = P_{fus}/P_{ext} \approx P_{fus}/P_{loss} \sim nT\tau_E$

Reactor: P_{loss} compensated
by α-(self)heating



- $Q = P_{fus}/P_{ext} = P_{fus}/(P_{loss} - P_{\alpha}) \rightarrow \infty$ (ignited plasma)

How is heat transported across field lines?

Energy confinement time determined by transport

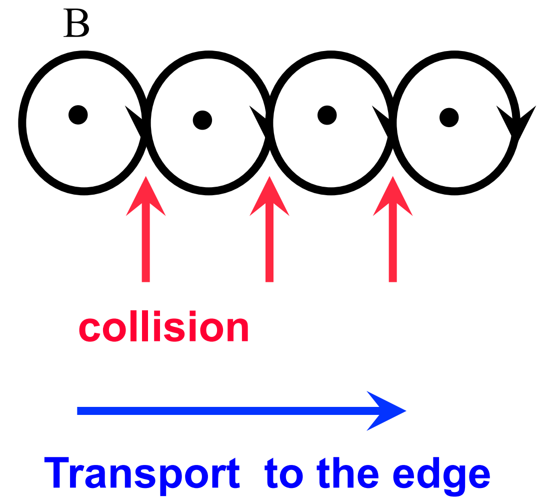
Simplest ansatz for heat transport:

• Diffusion due to binary collisions

$\chi \approx r_L^{\,2} / \tau_c \approx 0.005\ m^2/s$

$\tau_E \approx a^2/(4\,\chi)$

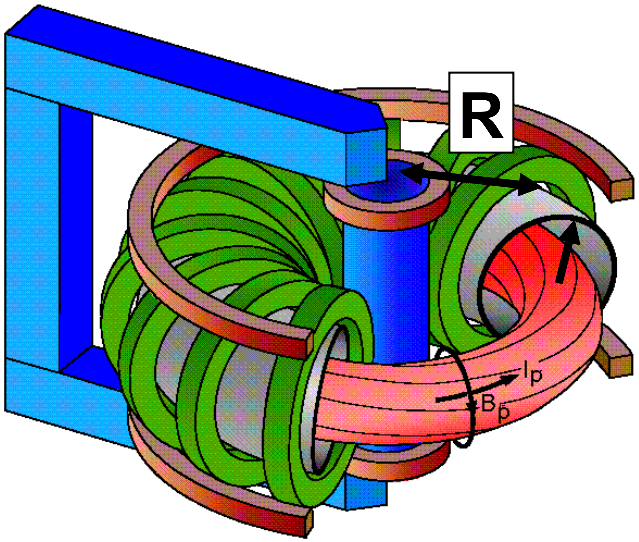
• table top device (R ≈ 0.6 m) should ignite!



Important transport regime for tokamaks and stellarators:

• Diffusion of trapped particles on banana orbits due to binary collisions

• neo-classical transport (important for impurities)

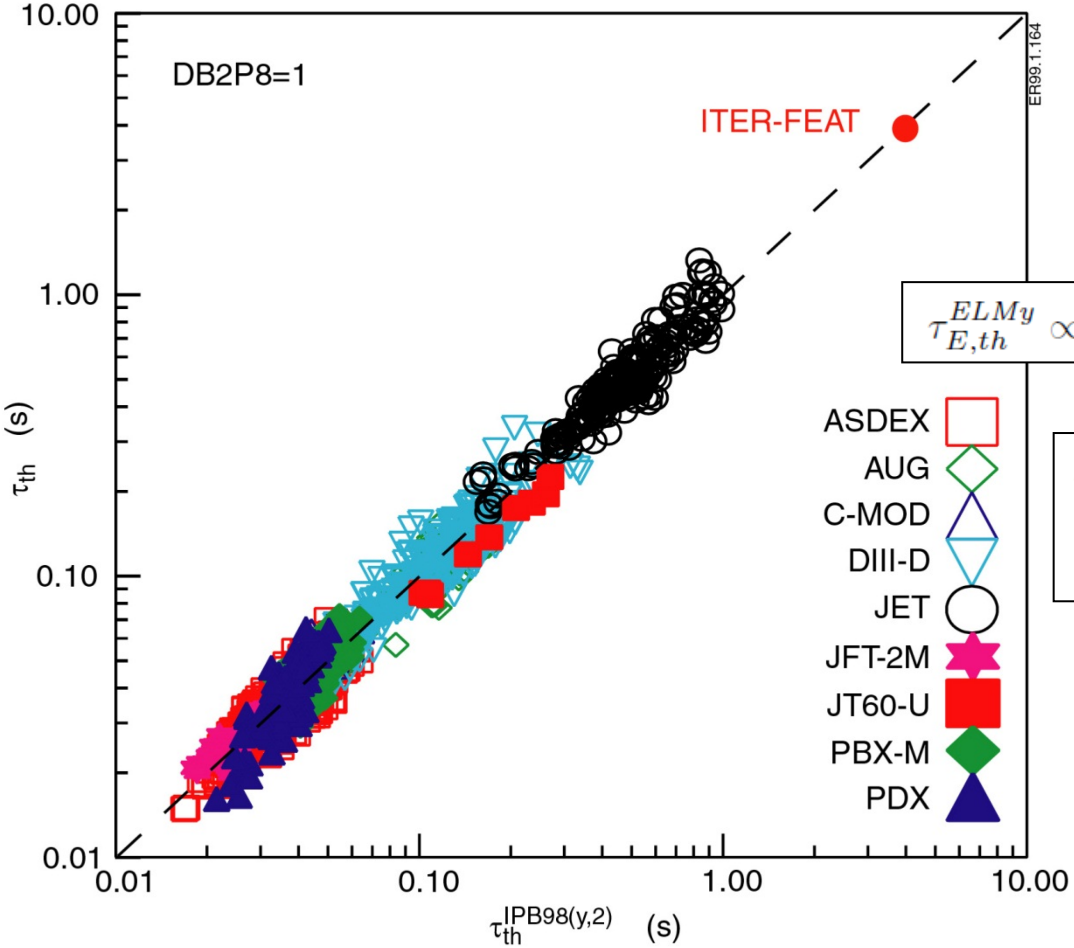


Experimental finding:

• ‚Anomalous' transport, much larger heat losses

• Tokamaks: Ignition expected for R ~ 8 m

Energy confinement: empirical scaling laws



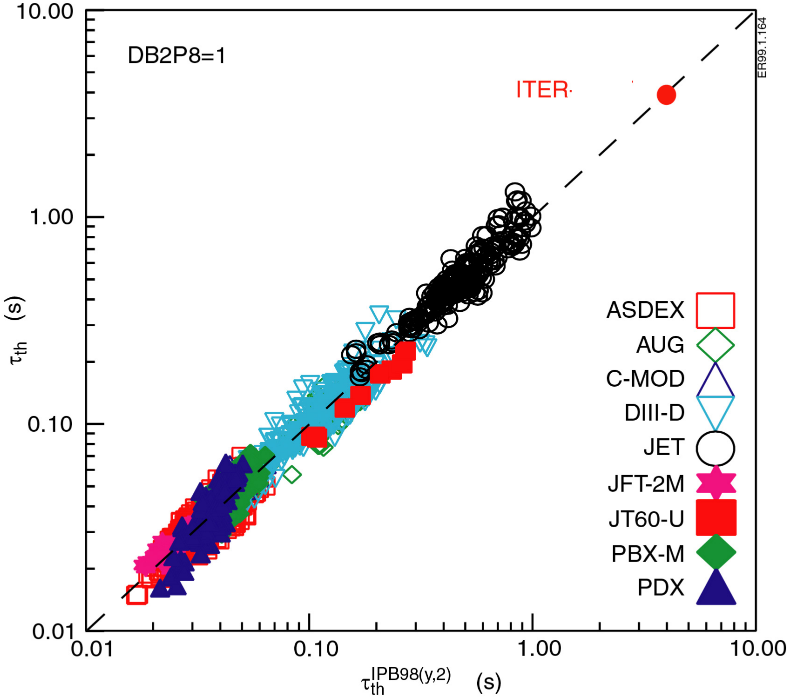
$$\tau_{E,th}^{ELMy} \propto \tau_B \rho_*^{-0.70} \beta^{-0.90} \nu_*^{-0.01} M^{0.96} q^{-3.0} \varepsilon^{0.73} \kappa_a^{2.3}$$

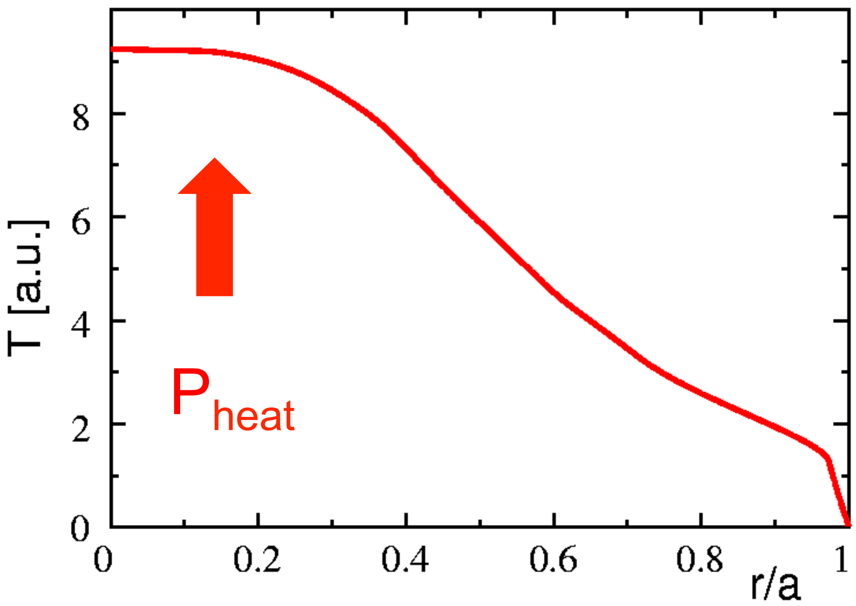
$$\tau_{E,th}^{ELMy} = 0.0562 I^{0.93} B^{0.15} P^{-0.69} n^{0.41} \times M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_a^{0.78}$$

In lack of a first principles physics model, ITER has been designed on the basis of an empirical scaling law

- very limited predictive capability, need first principles model

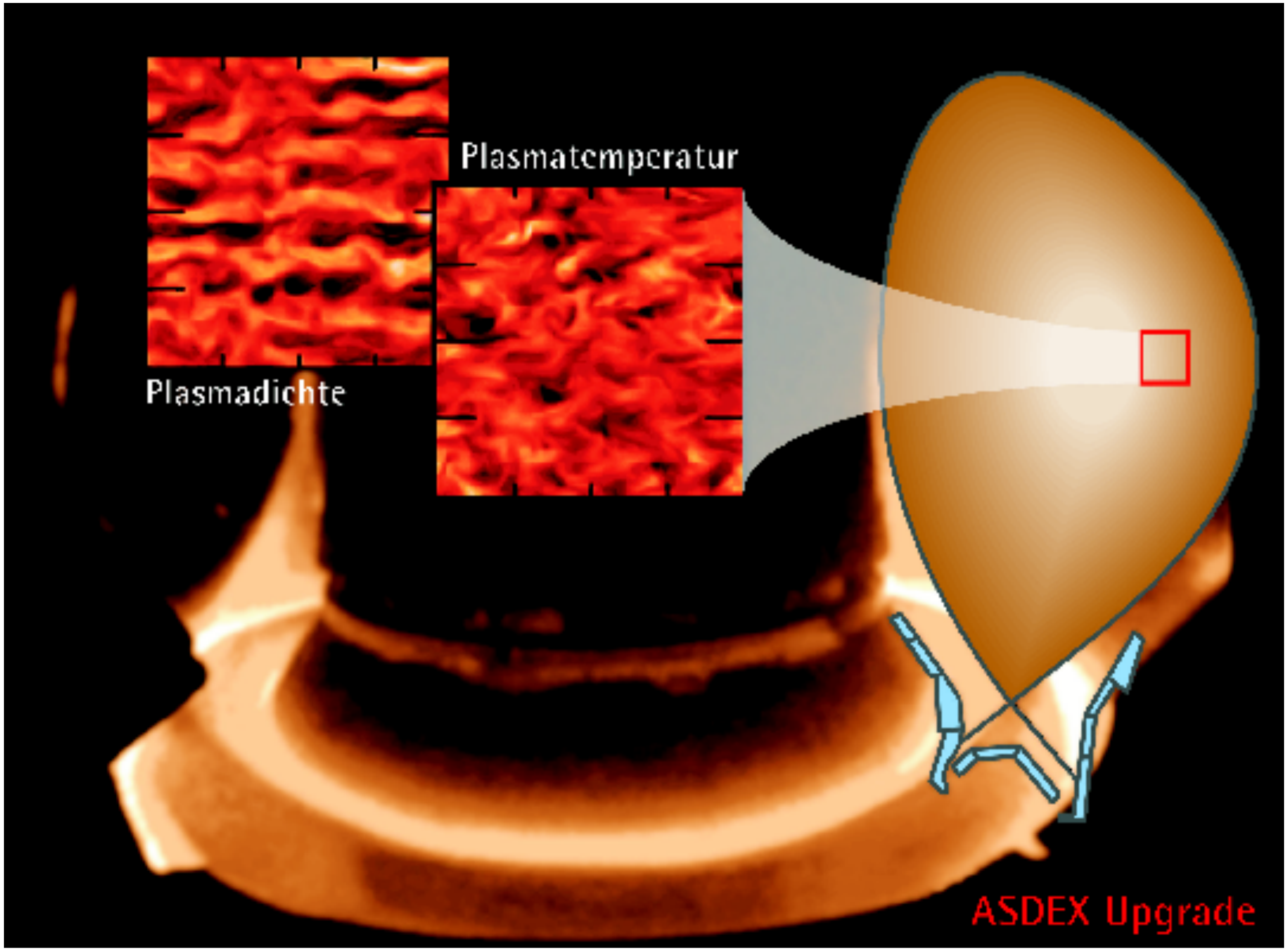
From empirical scaling laws to physics understanding





First principle based understanding of temperature (density, …) profiles

Anomalous transport due to turbulence



Simplest estimation for heat transport due to turbulence:

$D \approx (\Delta r_{eddy})^2/\tau_{tear} \approx 2\ m^2/s$

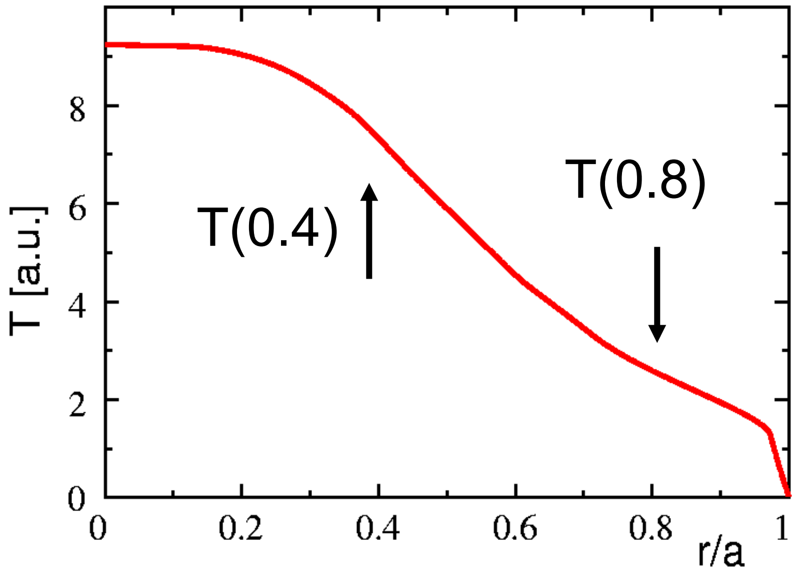
Global Gyrokinetic Simulation of Turbulence in

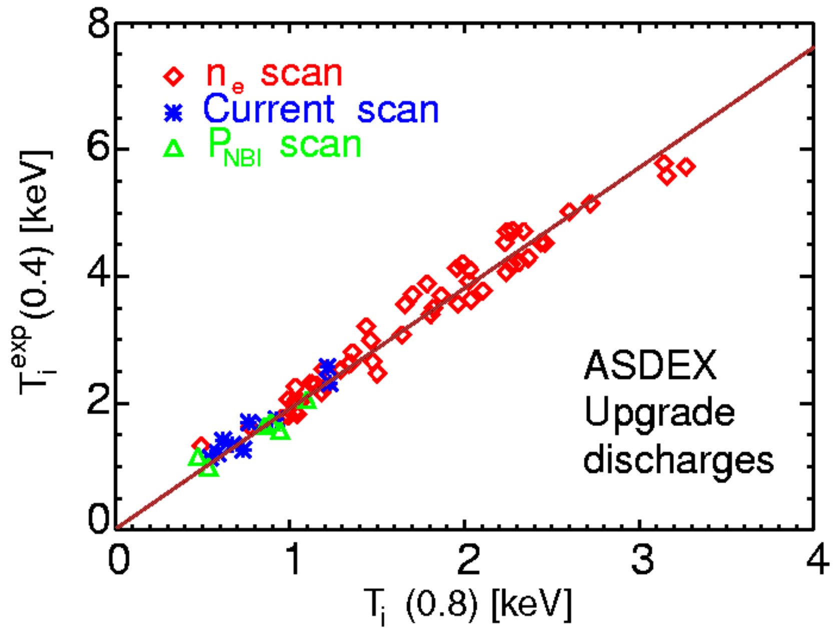
ASDEX Upgrade

GENE

gene.rzg.mpg.de

Energy Transport in Fusion Plasmas





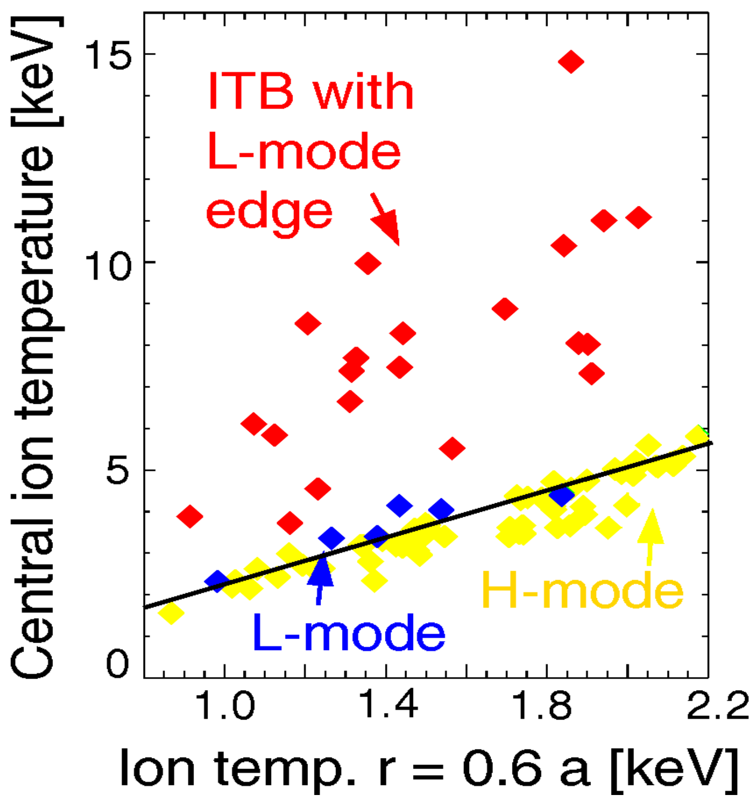
Anomalous transport determined by gradient driven turbulence

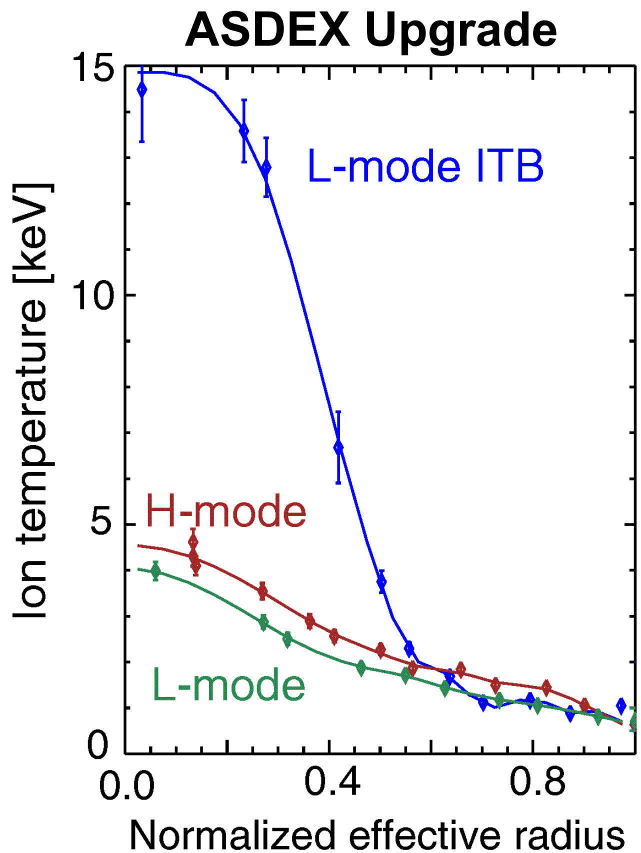
- temperature profiles show a certain 'stiffness'
- 'critical gradient' phenomenon – χ increases with P_{heat} (!)

$\Rightarrow$increasing machine size will increase central *T* as well as τ_E

N.B.: steep gradient region in the edge governed by different physics!

Energy Transport in Fusion Plasmas

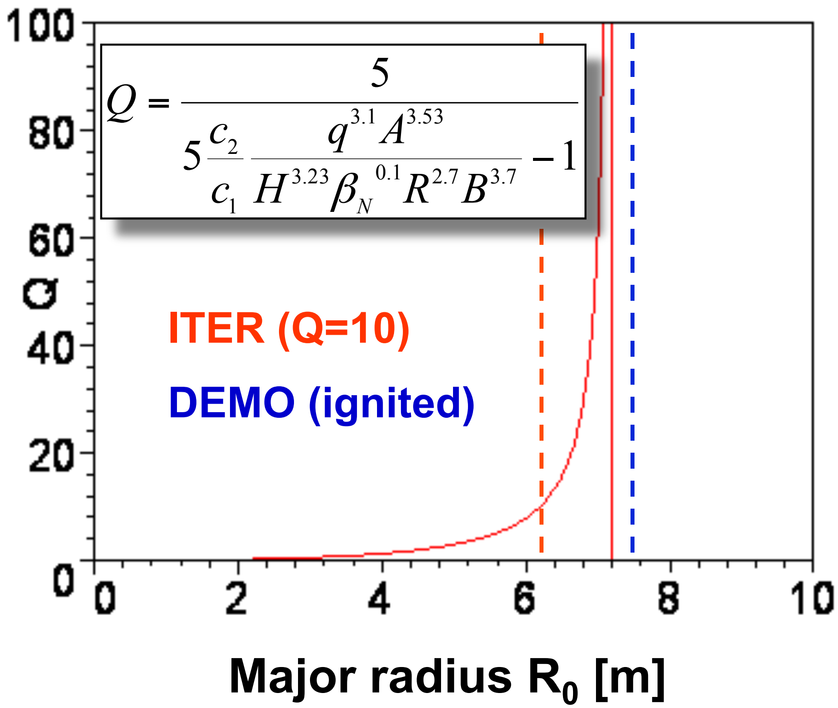


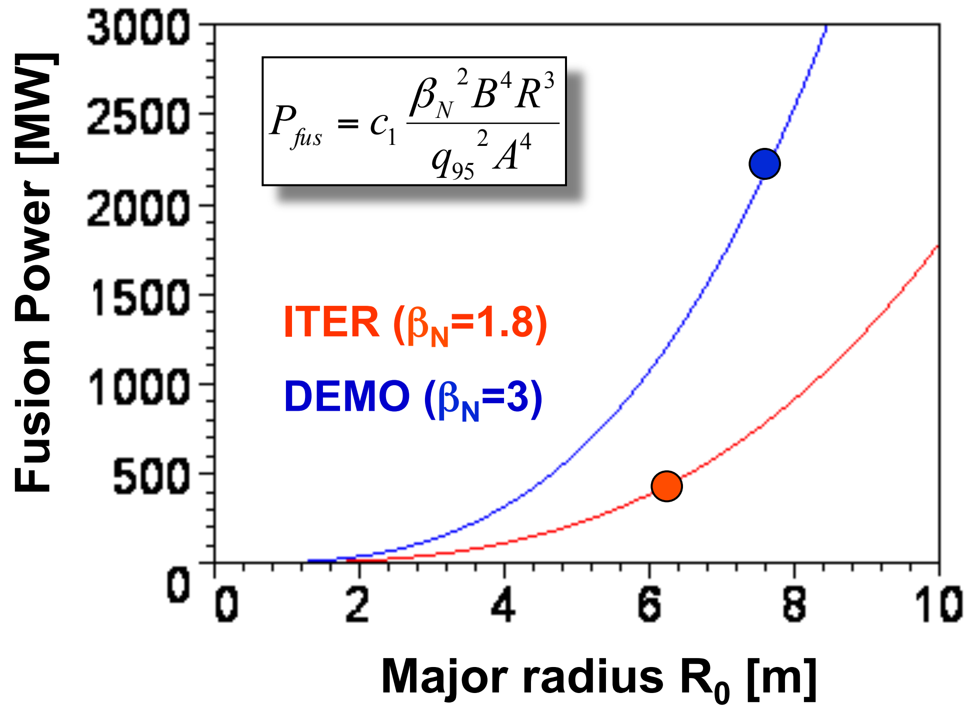


Locally, critical gradients can be exceeded ('Transport Barrier')

- sheared rotation can suppress turbulent eddies
- works at the edge (H-mode, see later) and internally ('ITB')

Anomalous transport determines machine size

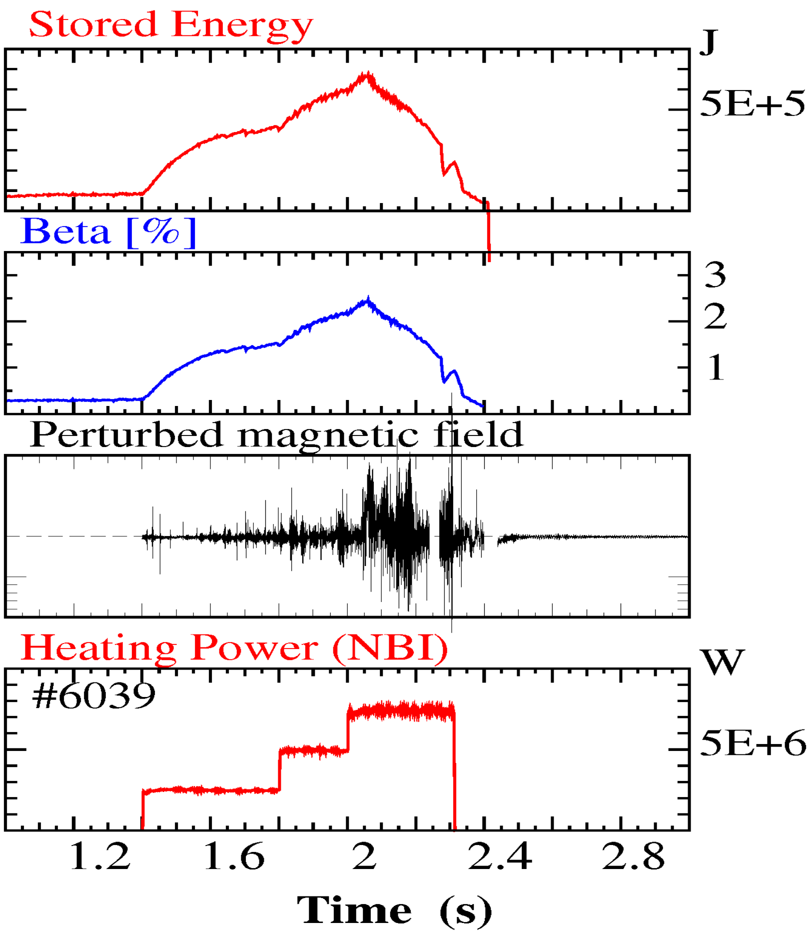


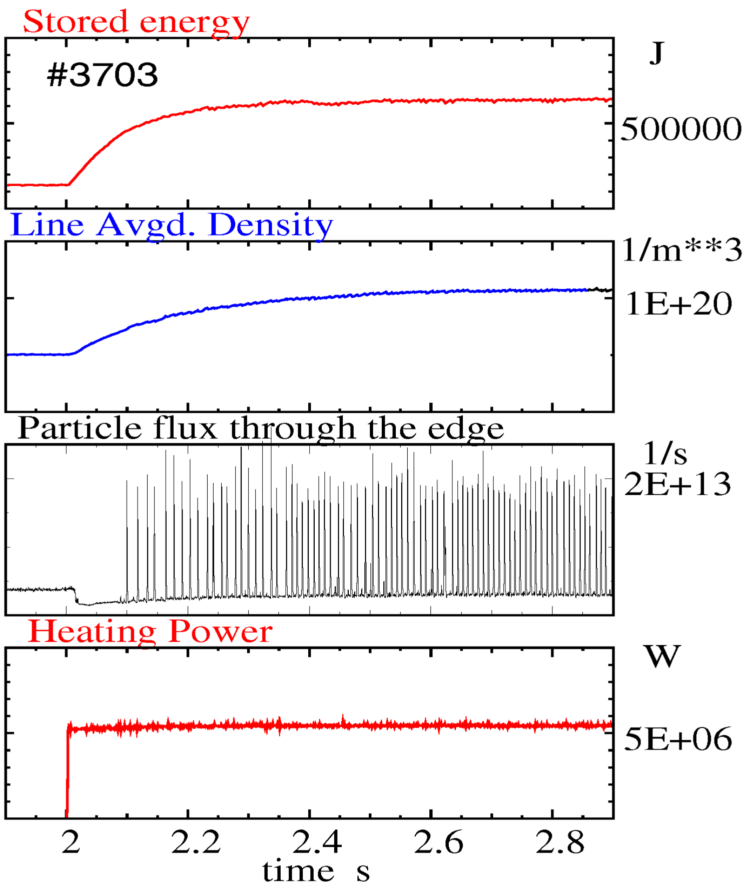


- ignition (self-heated plasma) predicted at R = 7.5 m
- at this machine size, the fusion power will be of the order of 1 GW

- heat insulation (energy transport)
- magnetohydrodynamic (MHD) stability
- tokamak operational scenarios

Plasma discharges can be subject to instabilities





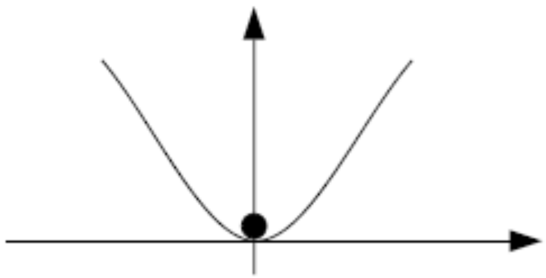
Desaster

β-limit, disruption

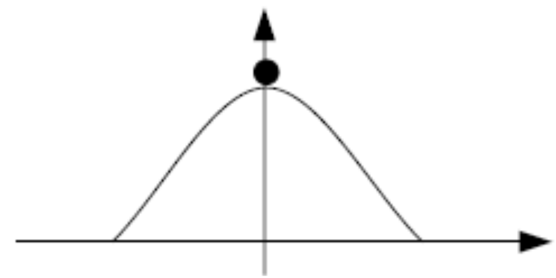
Self-organisation

sationarity of profiles *j(r), p(r)*

Plasma discharges can be subject to instabilities



linearly stable



linearly unstable

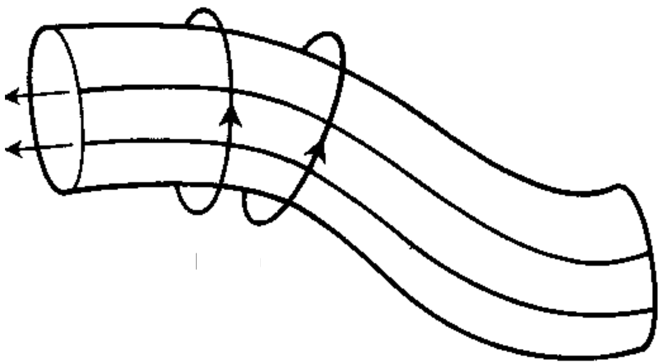
Equilibrium $\nabla p = j \times B$ means force balance, but not necessarily stability

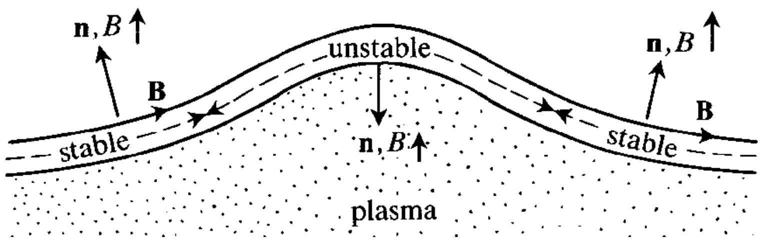
Stability against perturbation has to be evaluated by stability analysis

Mathematically: solve time dependent MHD equations

- linear stability: small perturbation, equilibrium unperturbed, exponentially growing eigenmodes
- nonlinear stability: finite peturbation, back reaction on equilibrium, final state can also be saturated instability

Free energies to drive MHD modes





current driven instabilities

Ex.: kink mode

(only tokamaks)

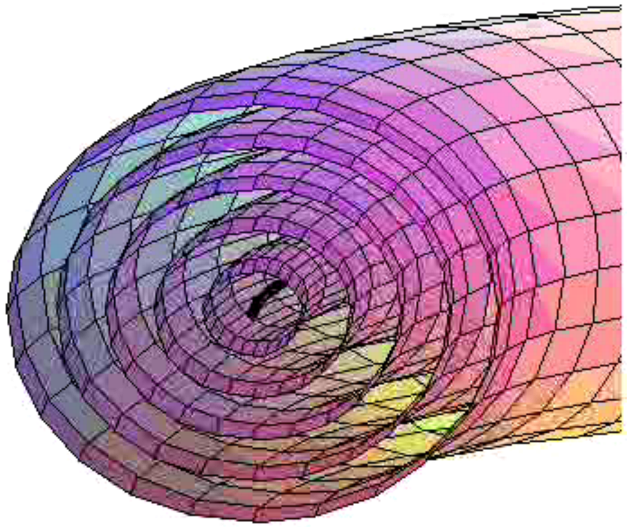
pressure driven instabilities

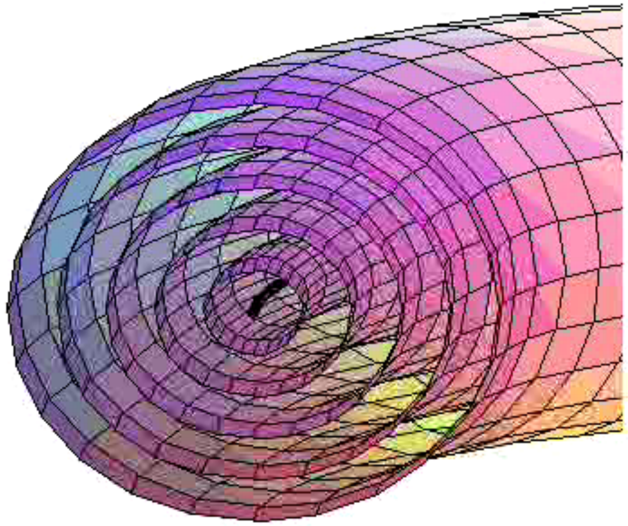
Ex.: interchange mode

(tokamak and stellarator)

N.B.: also fast particle pressure (usually kinetic effects)!

Ideal and resistive MHD instabilities





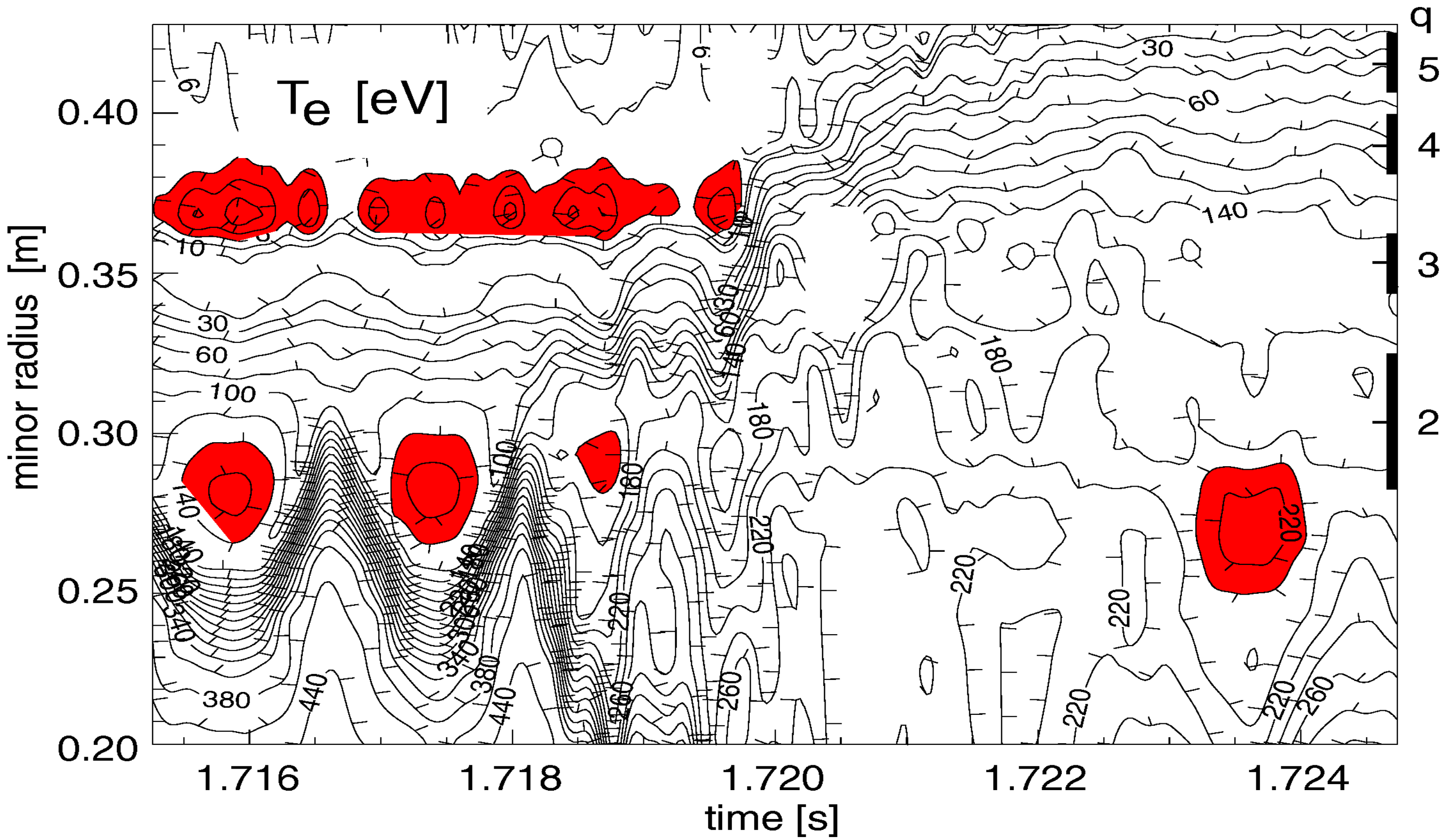
Ideal MHD: $\eta = 0$

- flux conservation
- topology unchanged

Resistive MHD: $\eta \neq 0$

- reconnection of field lines
- topology changes

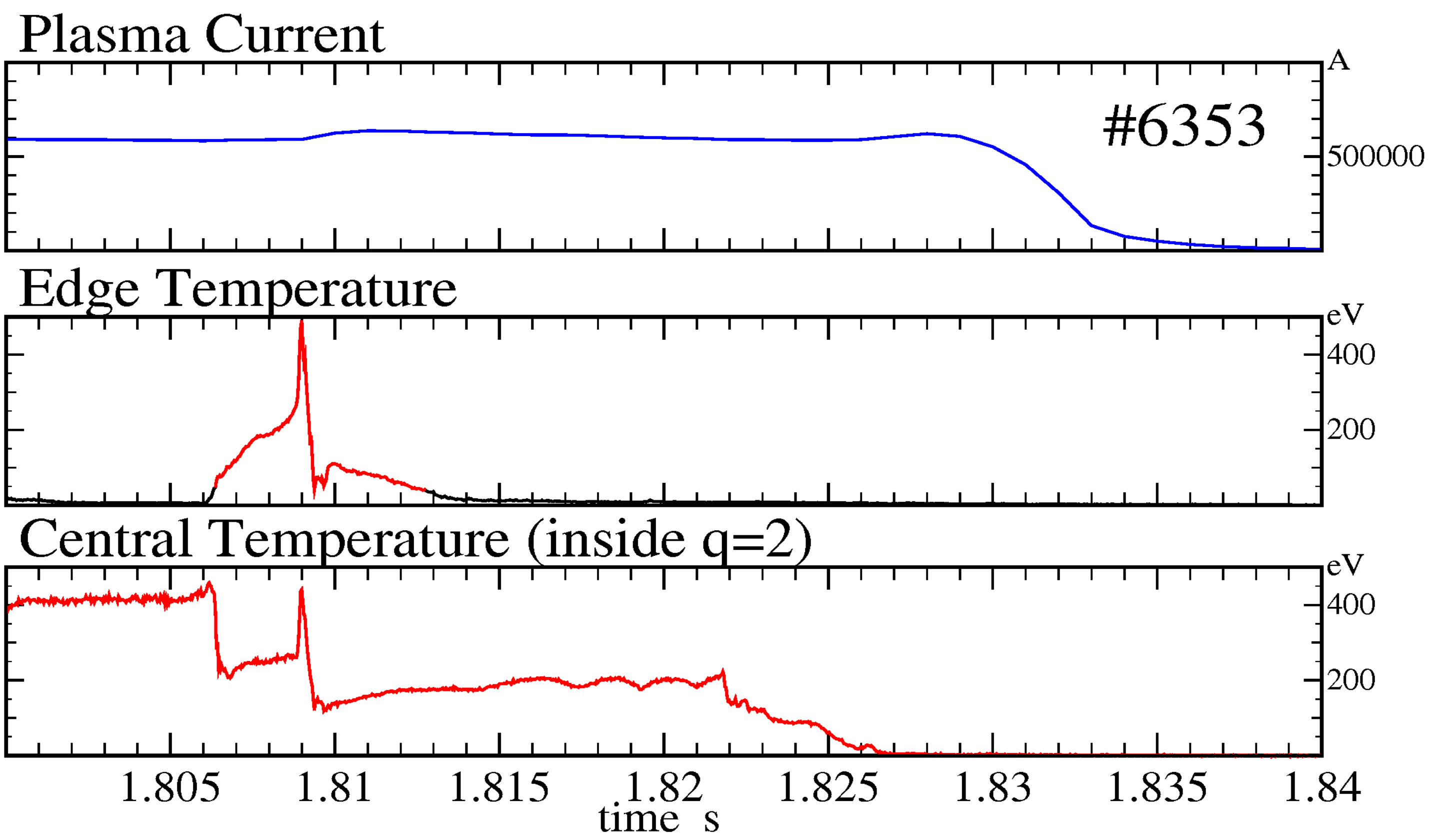
Magnetic islands impact tokamak discharges



coupling between island chains (possibly stochastic regions)

⇒ sudden loss of heat insulation ('disruptive instability')

Disruptive instability limits achievable density

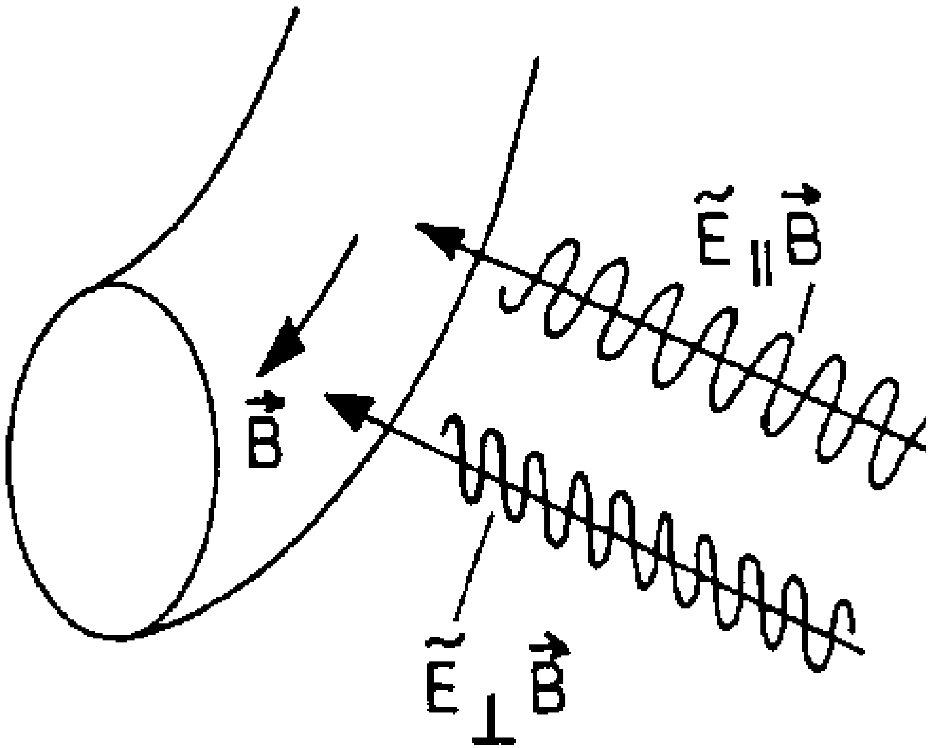


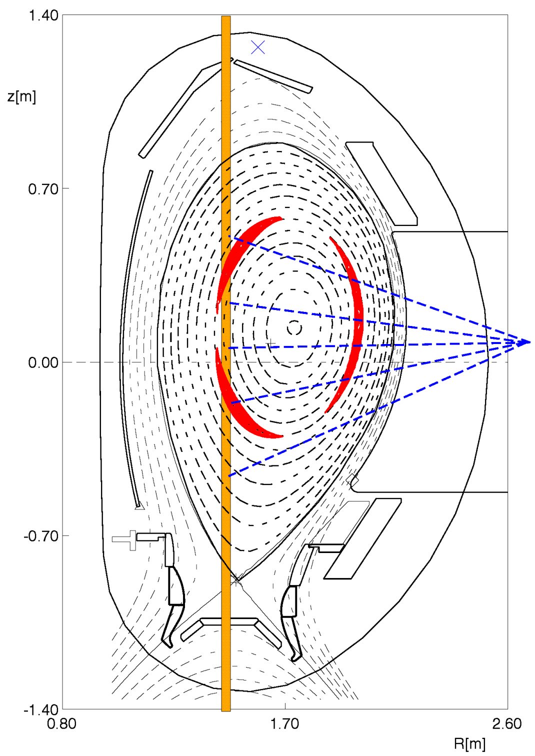
High density clamps current profile and leads to island chains

excessive cooling, current can no longer be sustained

disruptions lead to high thermal and mechanical loads!

Removal of magnetic islands by microwaves





$$n\ \nu_{ECR} = \nu_{wave} - k_{||} v_{||}$$

Electron Cyclotron Resonance at $\nu = n\ 28\ GHz\ B$ [T]

Plasma is optically thick at ECR frequency

Deposition controlled by local B-field ⇒ very good localisation

Ideal MHD instabilities limit achievable pressure

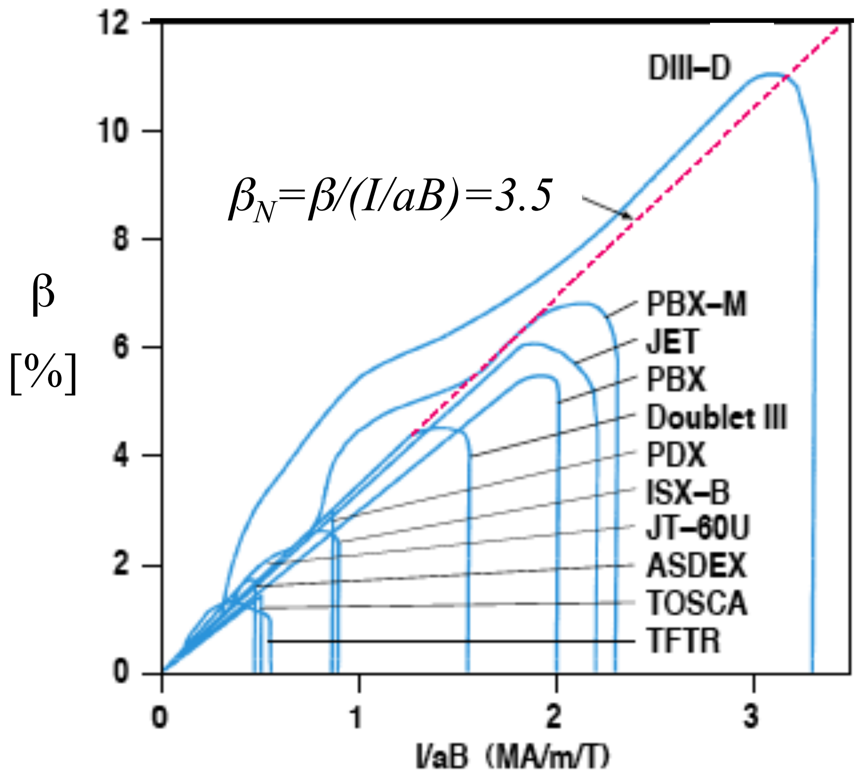
Optimising nT means high pressure and, for given magnetic field, high dimensionless pressure $\beta = 2\mu_0 <p> / B^2$

This quantity is ultimately limited by ideal instabilities

'Ideal' MHD limit (ultimate limit, plasma unstable on Alfvén time scale ~ 10 μs, only limited by inertia)

- 'Troyon' limit $\beta_{max} \sim I_p/(aB)$, leads to definition of $\beta_N = \beta/(I_p/(aB))$

- at fixed aB, shaping of plasma cross-section allows higher $I_p \rightarrow$ higher β



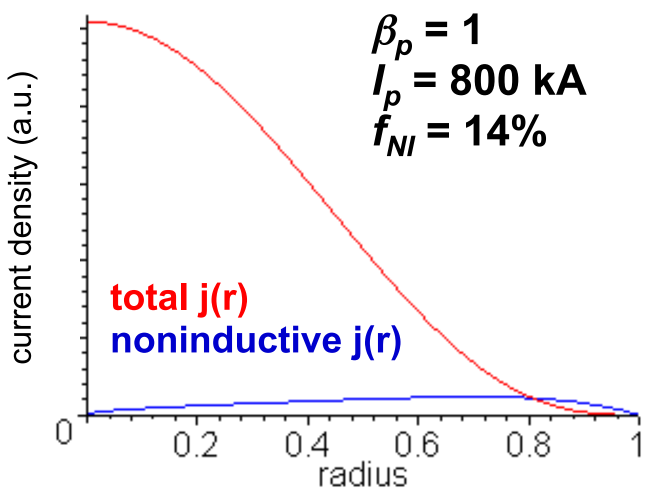
- heat insulation (energy transport)
- magnetohydrodynamic (MHD) stability
- tokamak operational scenarios

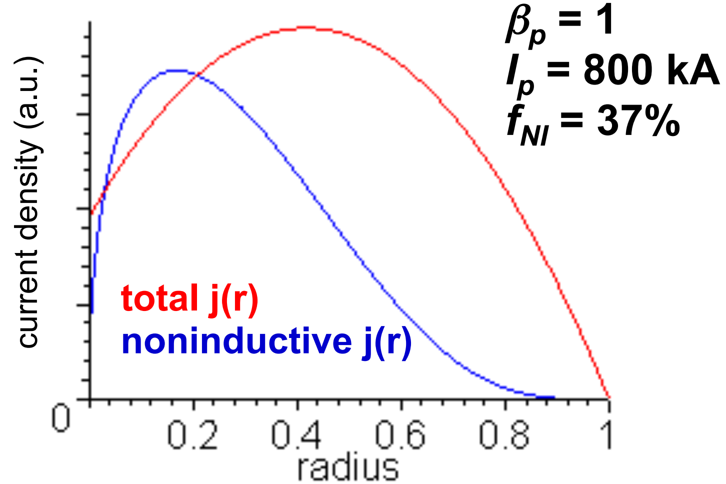
What is a ‚tokamak scenario‘?

A tokamak (operational) scenario is a recipe to run a tokamak discharge

Plasma discharge characterised by

- external control parameters: B_t, R_0, a, κ, δ, P_{heat}, Φ_D…
- integral plasma parameters: $\beta = 2\mu_0\langle p\rangle/B^2$, $I_p = 2\pi \int j(r)\, r\, dr$…
- plasma profiles: pressure $p(r) = n(r)*T(r)$, current density $j(r)$





→ operational scenario best characterised by *shape* of $p(r)$, $j(r)$

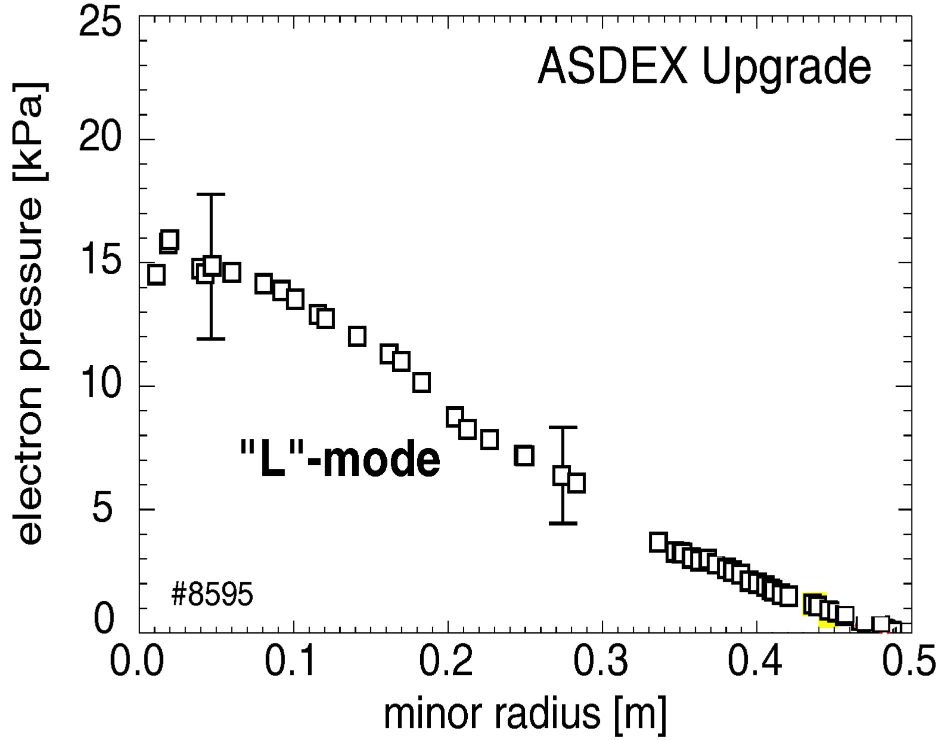
Control of the profiles j(r)and p(r) is limited

Pressure profile determined by combination of heating / fuelling profile and radial transport coefficients

- ohmic heating coupled to temperature profile via $\sigma \sim T^{3/2}$
- external heating methods allow for some variation – ICRH/ECRH deposition determined by *B*-field, NBI has usually broad profile
- gas puff is peripheral source of particles, pellets further inside

but: under reactor-like conditions, dominant α-heating ~ $(nT)^2$

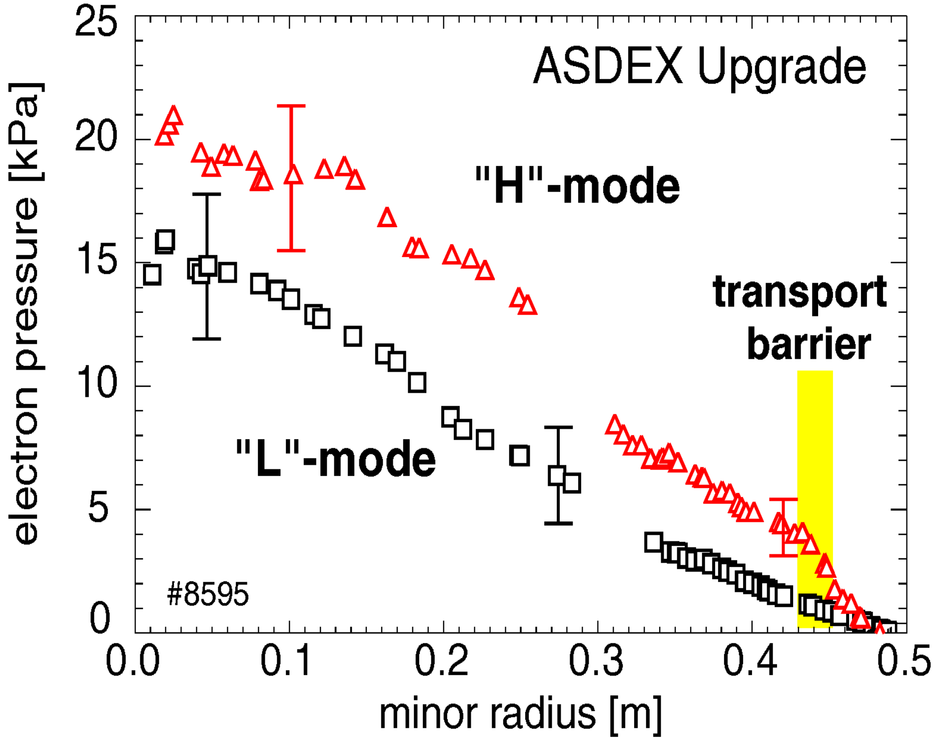
The (low confinement) L-mode scenario



Standard scenario without special tailoring of geometry or profiles

- central current density usually limited by sawteeth
- temperature gradient sits at critical value over most of profile
- extrapolates to very large (R > 10 m, I_p > 30 MA) pulsed reactor

The (high confinement) H-mode scenario



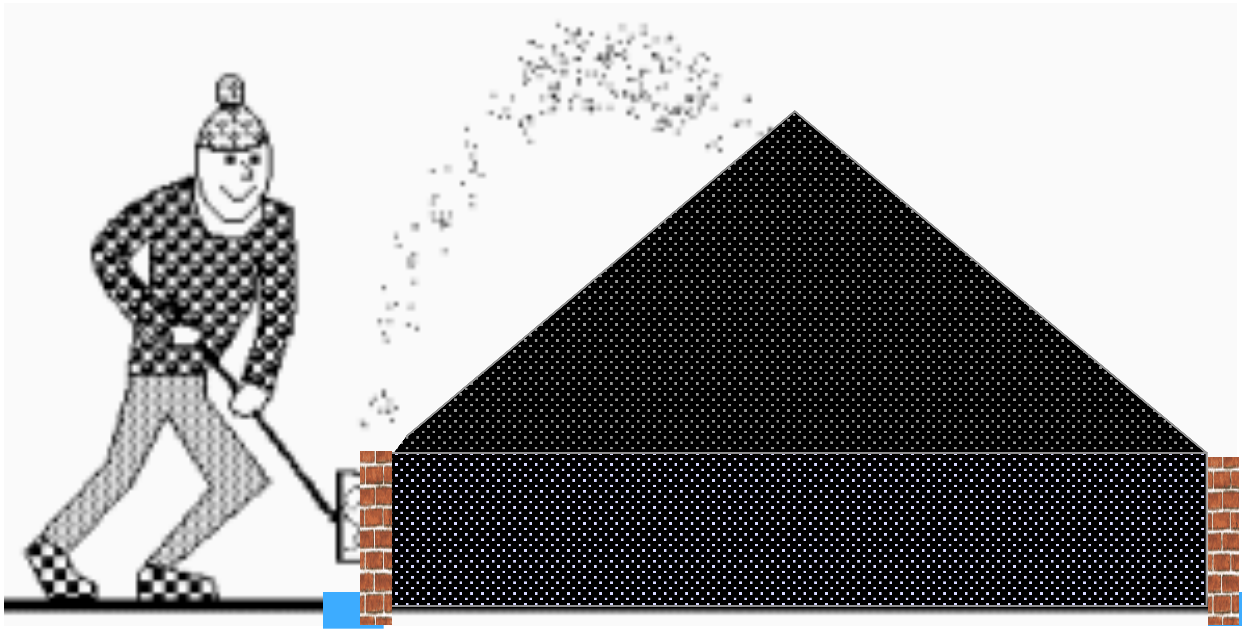
With hot (low collisionality) conditions, edge transport barrier develops

- gives higher boundary condition for ‘stiff’ temperature profiles
- global confinement τ_E roughly factor 2 better than L-mode
- extrapolates to more attractive (R ~ 8 m, I_p ~ 20 MA) pulsed reactor

Quality of heat insulation

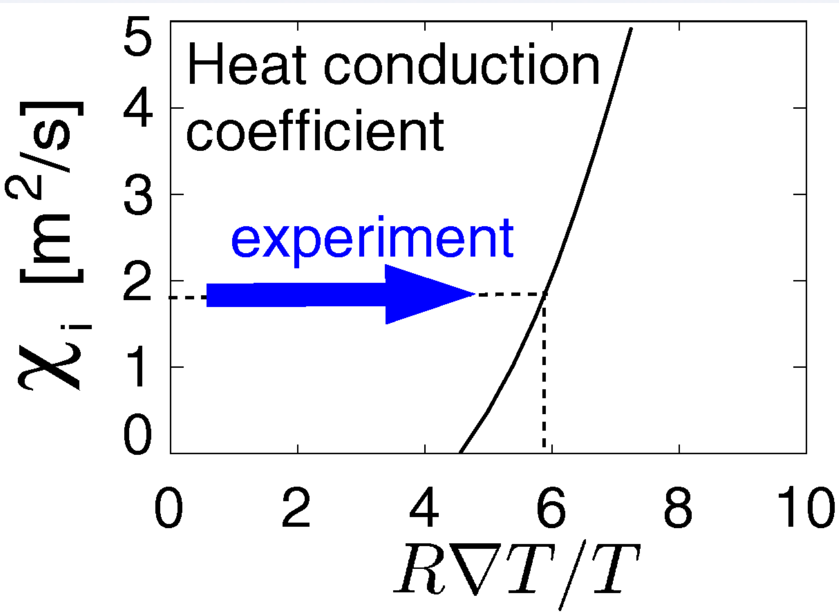
Turbulent transport limits (on a logarithmic scale) the gradient of the temperature profile

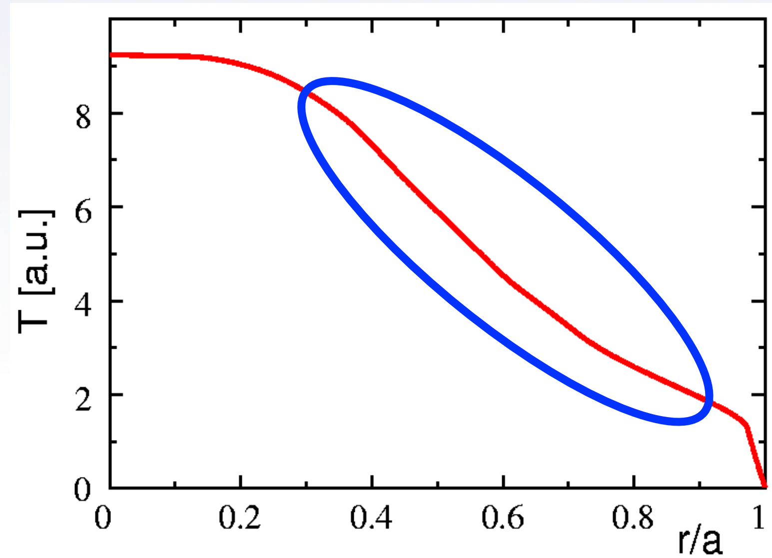
Analogy of a sand pile: limited gradient



But total height is variable by barriers

Turbulent transport strongly increases with logarithmic temperature gradient





Existence of a critical logarithmic temperature gradient (nearly independent on heating power)

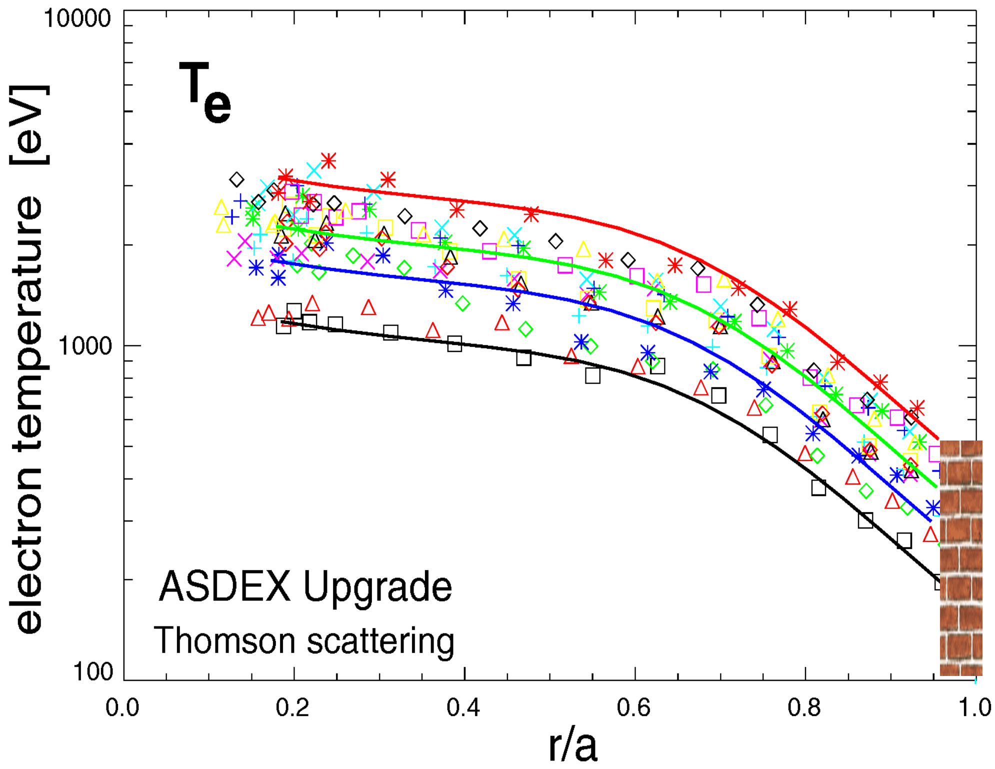
“stiff” temperature profiles

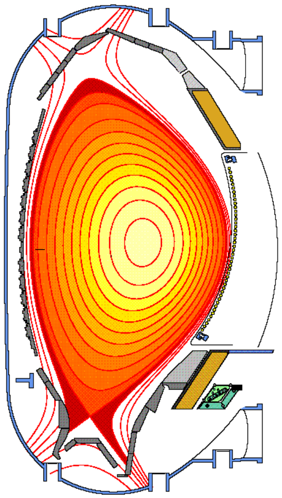
$$\frac{d\,\mathit{ln}\,T}{dr} = \frac{\nabla T}{T} = -\frac{1}{L_{T,cr}}$$

$$T(a) = T(b)\,\mathit{exp}\left(\frac{b - a}{L_{T,cr}}\right)$$

Core temperature determined by temperature at the edge…

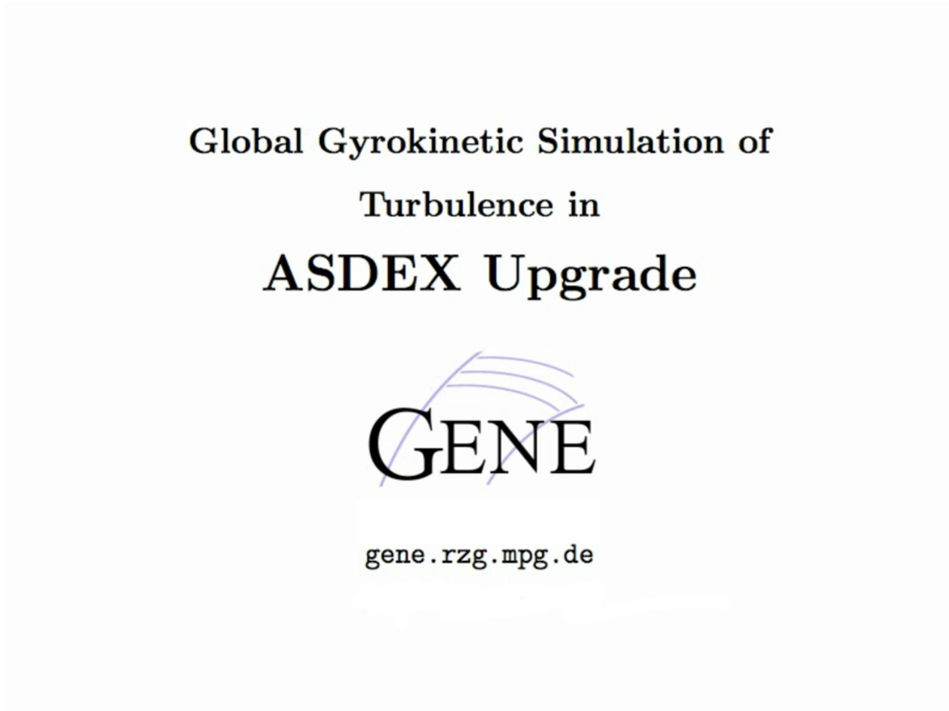
… nearly independent of heating power





Transport barrier at the edge (“high” confinement mode) in divertor geometry

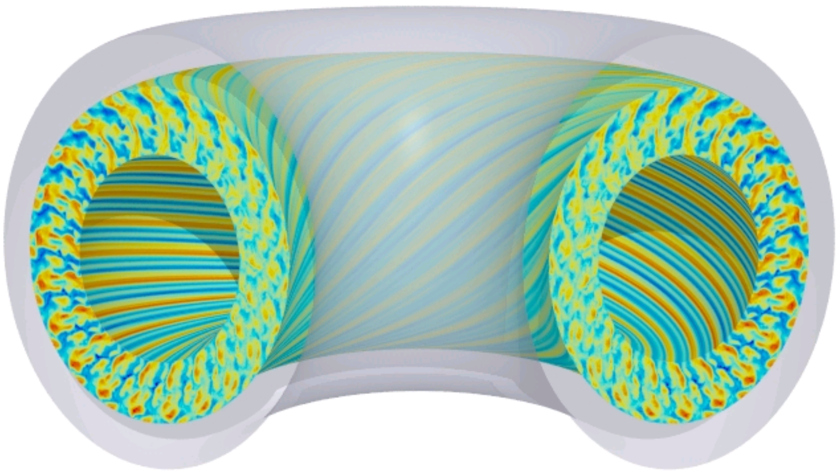
Energy Transport in Fusion Plasmas

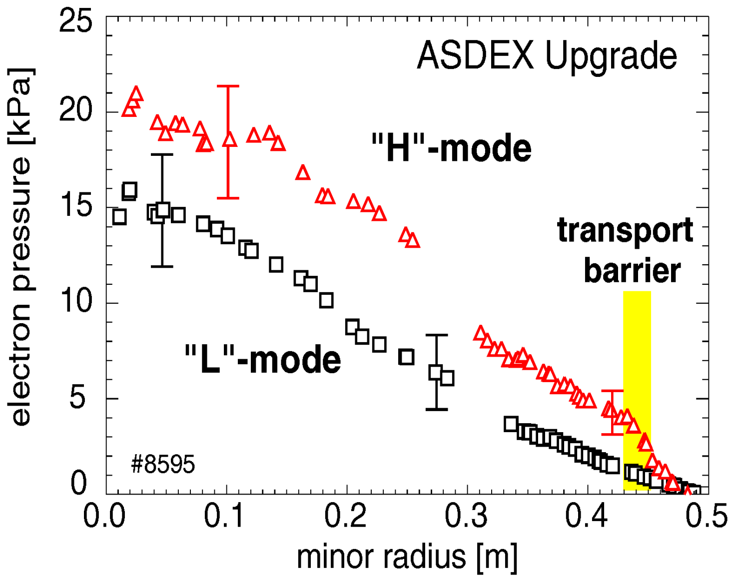


Anomalous transport determined by gradient driven turbulence

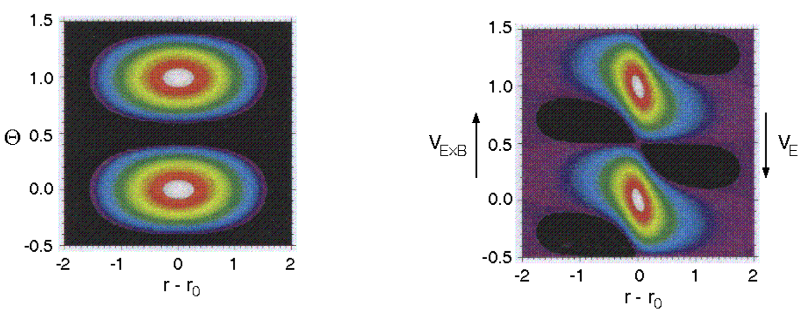
- linear: main microinstabilities giving rise to turbulence identified
- nonlinear: turbulence generates ‘zonal flow’ acting back on eddy size
- (eddy size)^2 / (eddy lifetime) is of the order of experimental χ–values

Sheared flows – the most important saturation mechanism

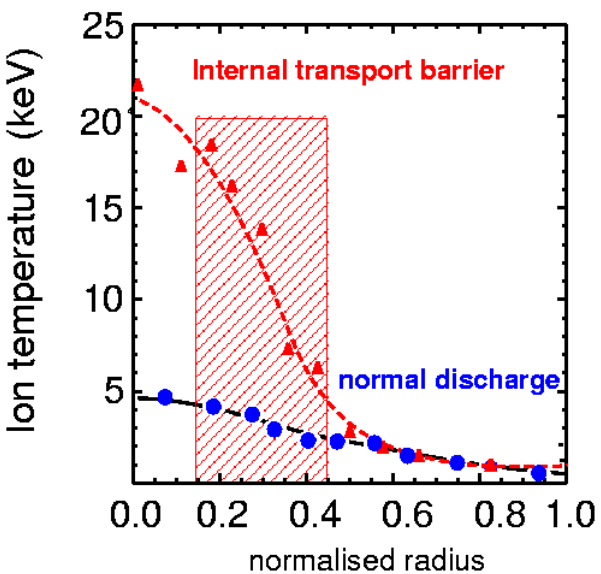




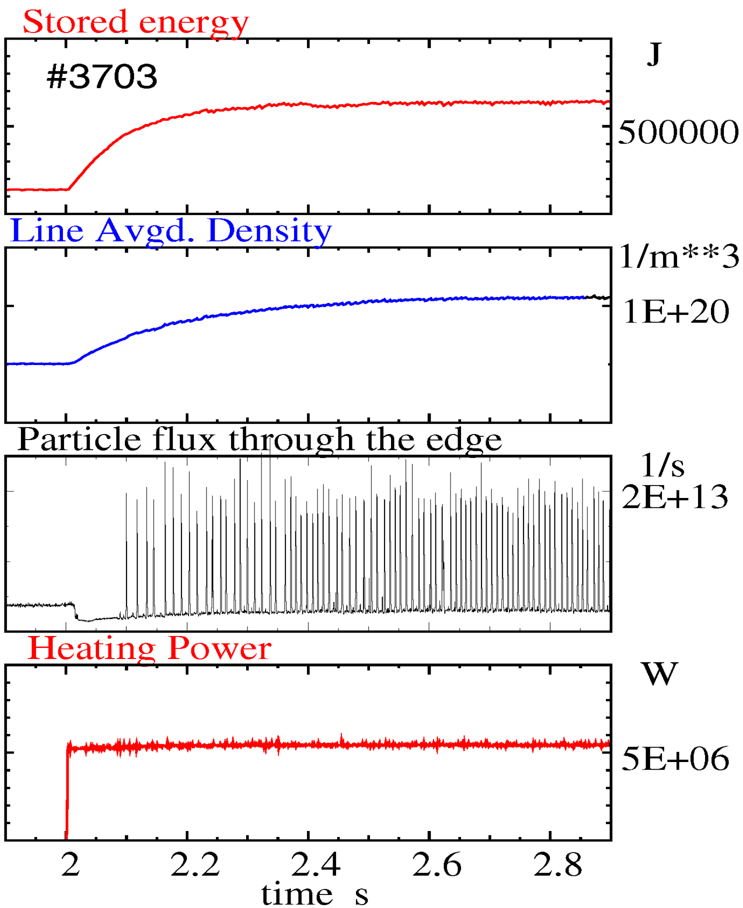
Macroscopic sheared rotation deforms eddies and tears them

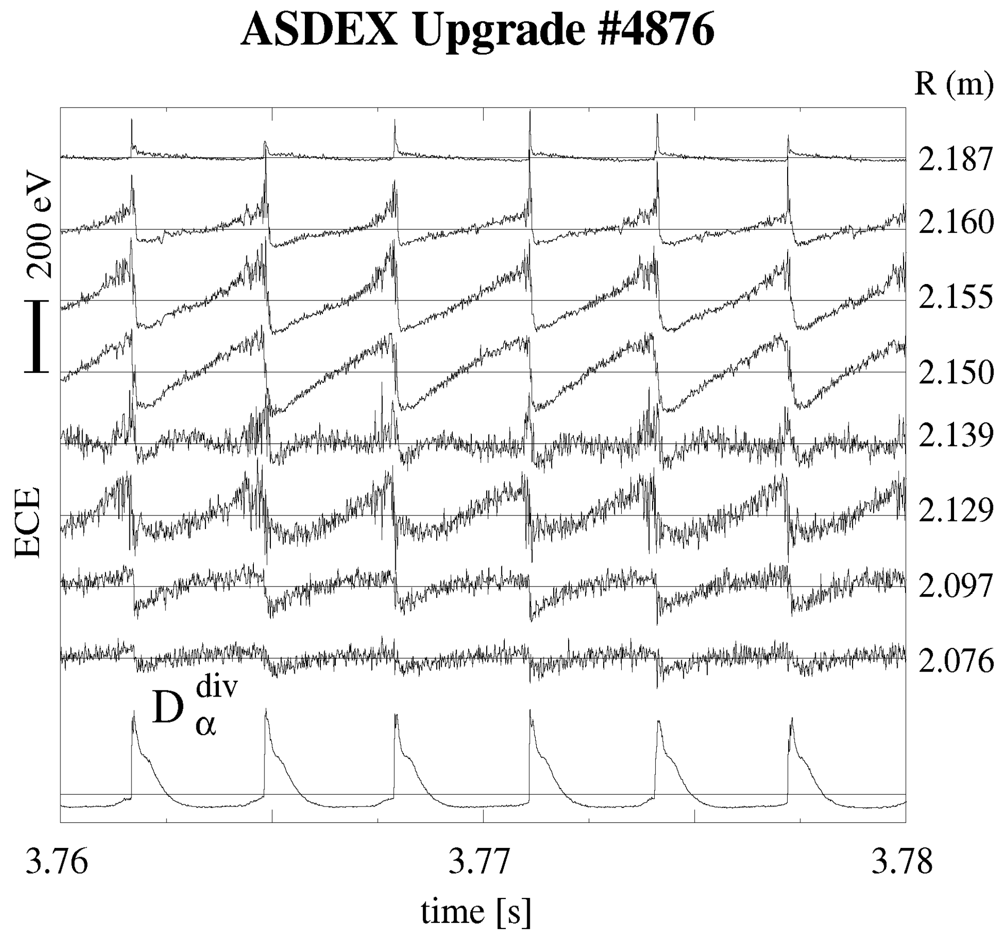


Radial transport increases with eddy size



Stationary H-modes usually accompanied by ELMs

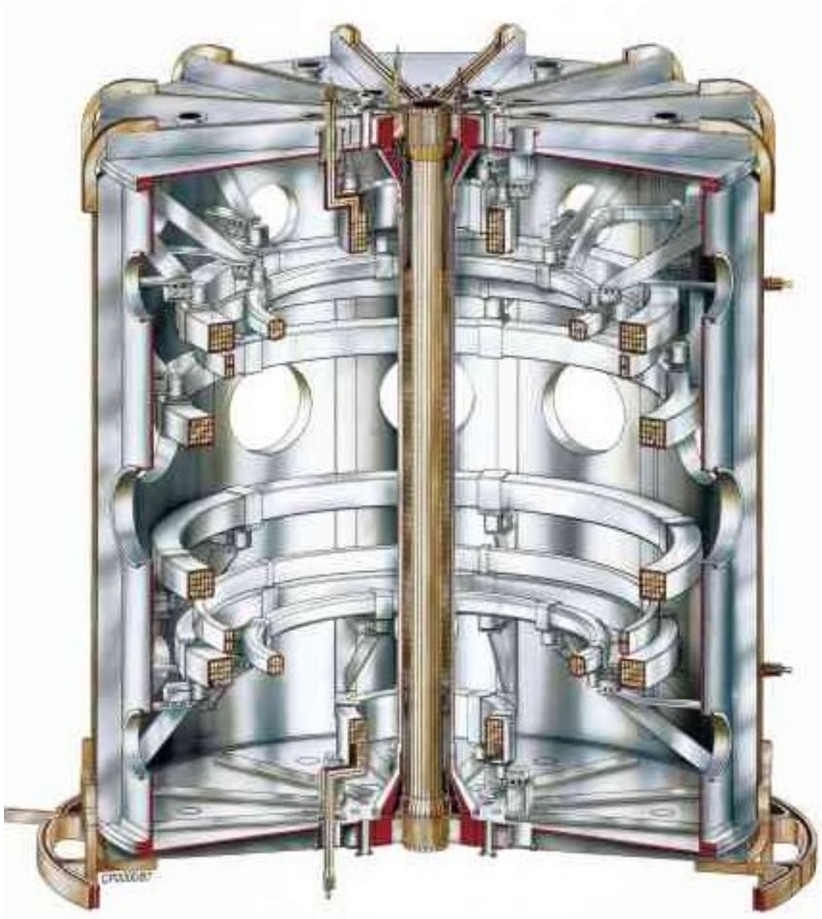




Edge Localised Modes (ELMs) regulate edge plasma pressure

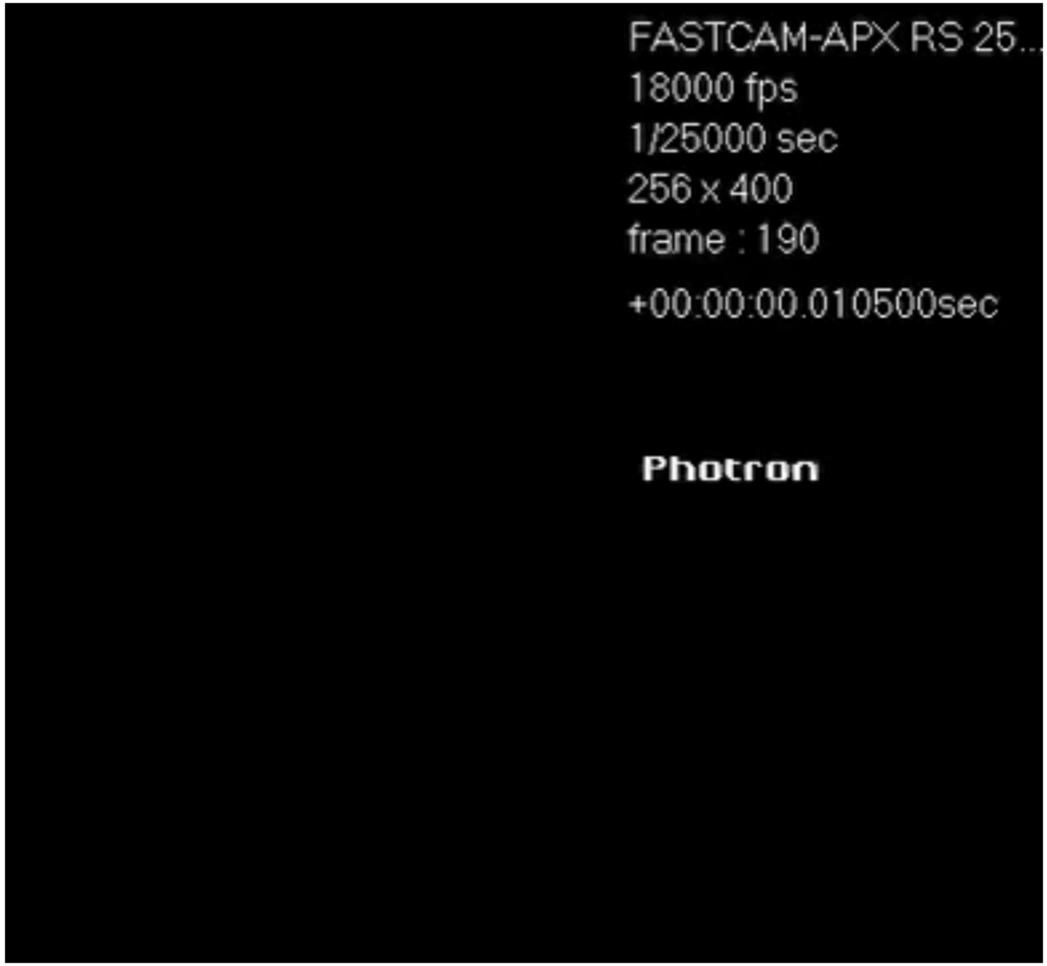
- without ELMs, particle confinement ‚too good‘ – impurity accumulation

Cross section of the spherical tokamak MAST



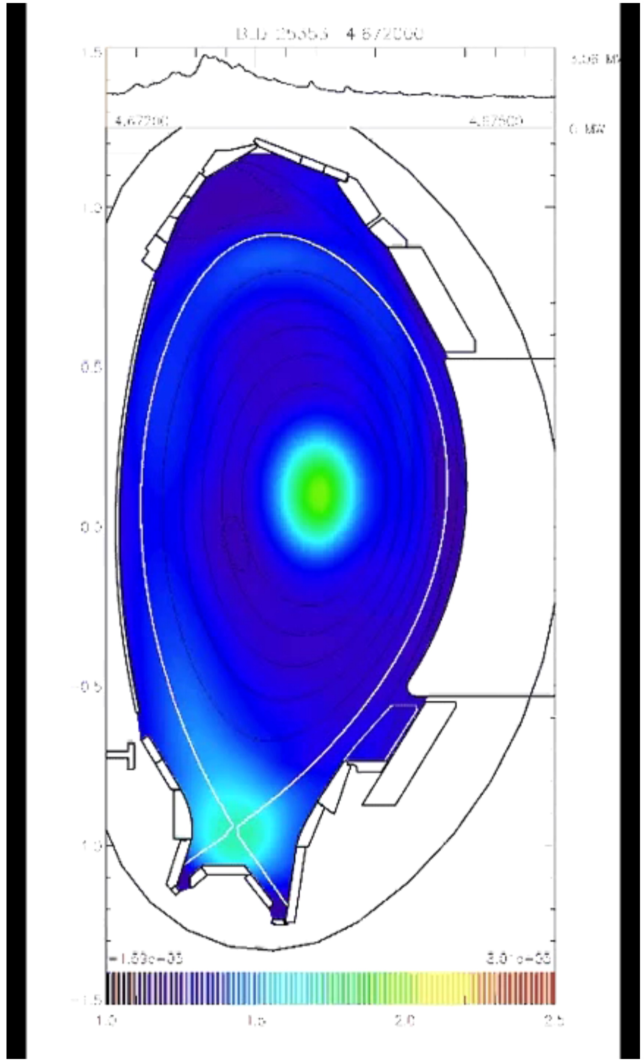
MAST, CCFE, UK

Plasma discharges can be subject to instabilities



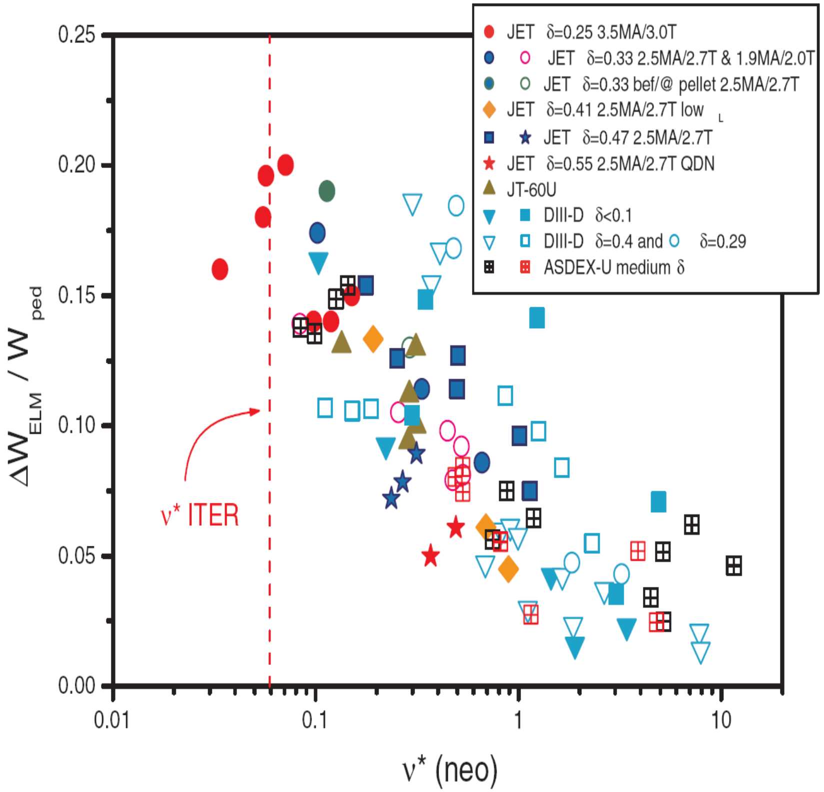
MAST, CCFE, UK

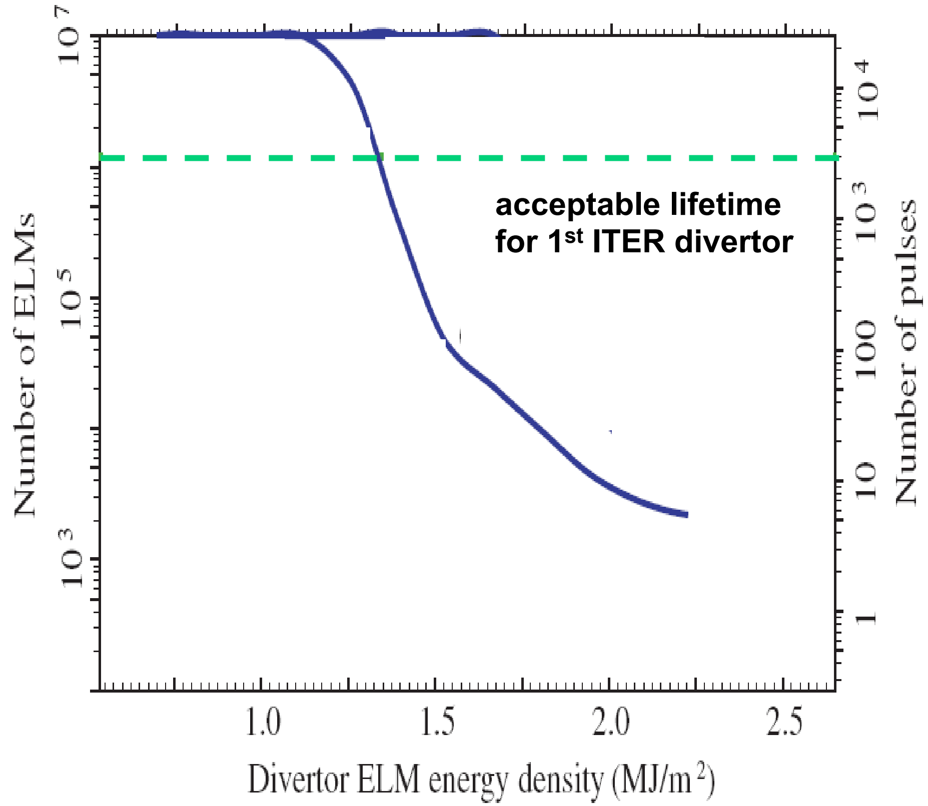
Instability also measured in total radiation



ASDEX Upgrade – tomographic reconstruction of AXUV diods
By M. Bernert on youtube

Stationary H-modes usually accompanied by ELMs



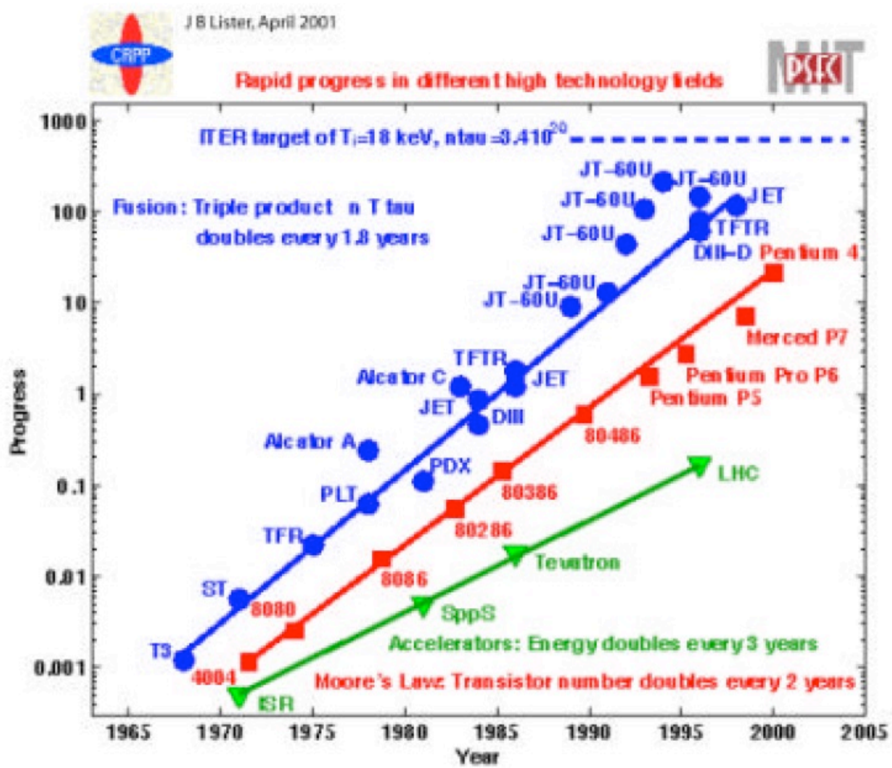


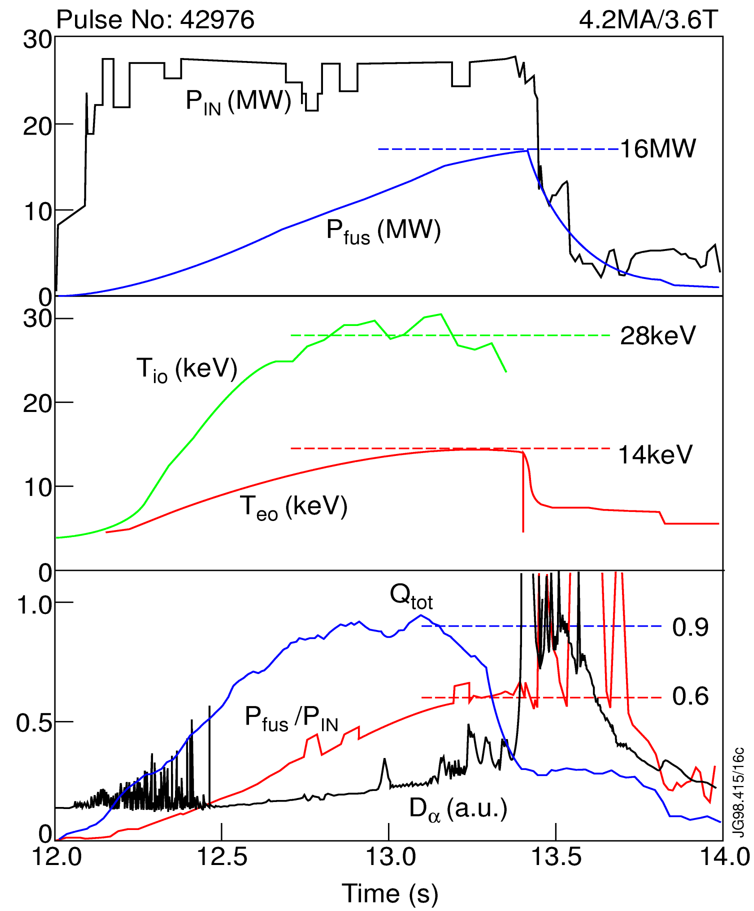
But: ELMs may pose a serious threat to the ITER divertor

- large 'type I ELMs' may lead to too high divertor erosion

Progress…

Tokamaks have made Tremendous Progress





- figure of merit $nT\tau_E$ doubles every 1.8 years
- JET tokamak in Culham (UK) has produced 16 MW of fusion power
- present knowledge has allowed to design a next step tokamak to demonstrate large scale fusion power production: ITER