MAX-PLANCK-INSTITUT
FÜR PLASMAPHYSIK

Introduction to Tokamak Operation Scenarios and Development Considerations

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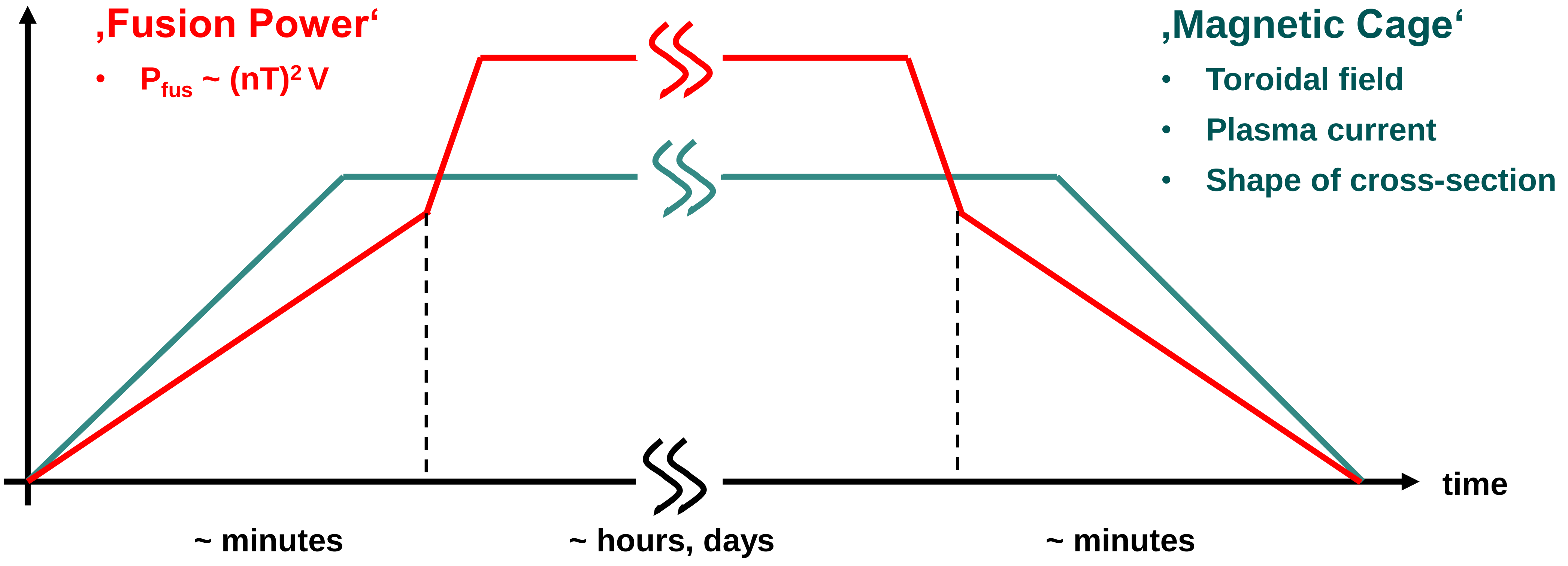
Goal: produce and sustain a burning fusion plasma



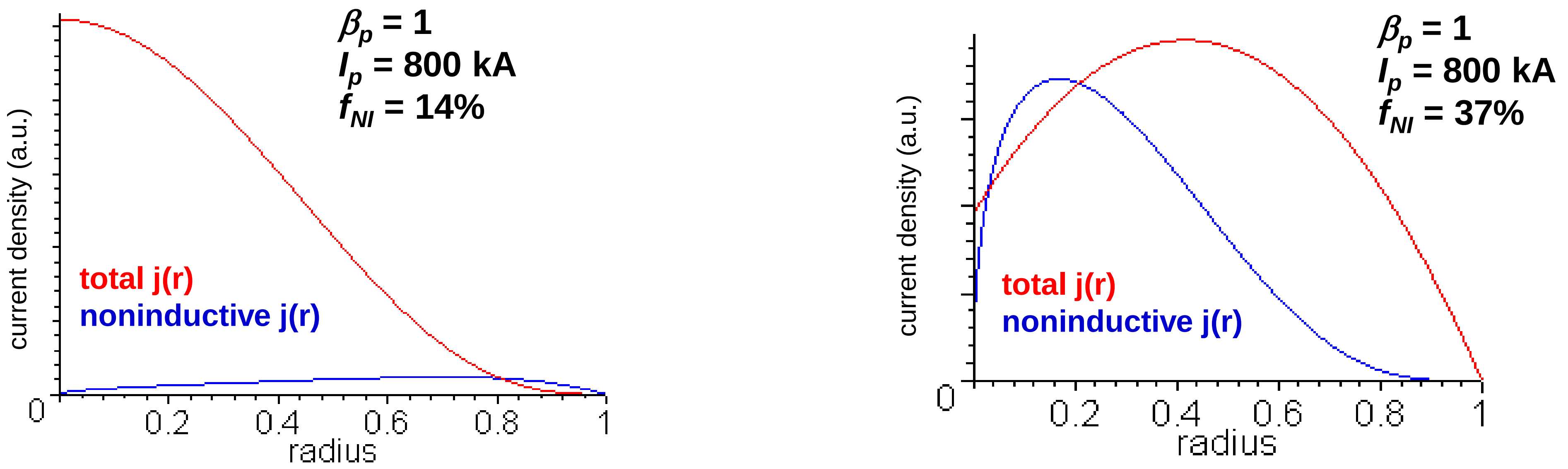
‚Burning' plasma: $Q = P_{fus}/P_{ext} >> 1 \Rightarrow$ fulfill Lawson criterion $nT\tau_E > 5 \times 10^{21}$ @ 15 keV

- achievable n, T τ_E, and pulse length depend on ‚tokamak operation scenario' (= mode of operation)
- note: here, we use the term ‚operation scenario' for the flat-top phase only

What is a ‚tokamak operation scenario'?

A tokamak operation scenario is a tokamak discharge state, characterized by

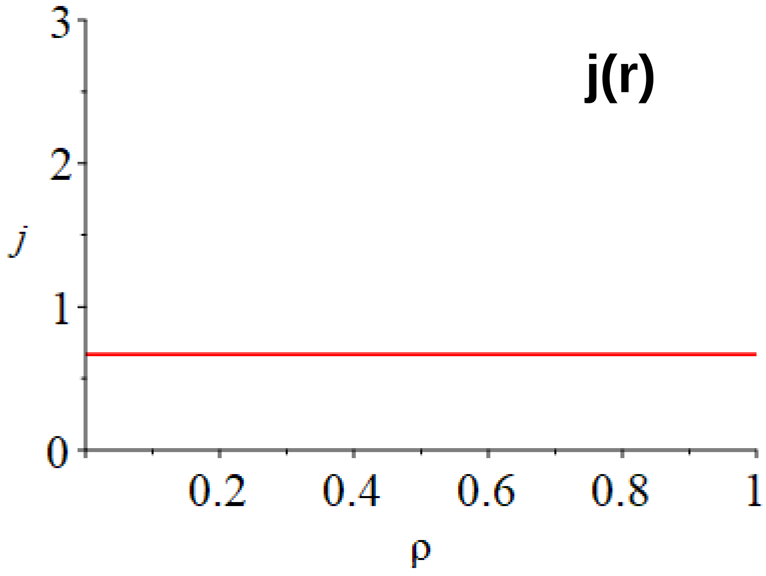
- external control parameters: B_t, R_0, a, κ, δ, P_{heat}, Φ_D…
- integral plasma parameters: $\beta = 2\mu_0\langle p\rangle/B^2$, $I_p = 2\pi \int j(r)\, r\, dr$…
- plasma profiles: pressure $p(r) = n(r)*T(r)$, current density $j(r)$…

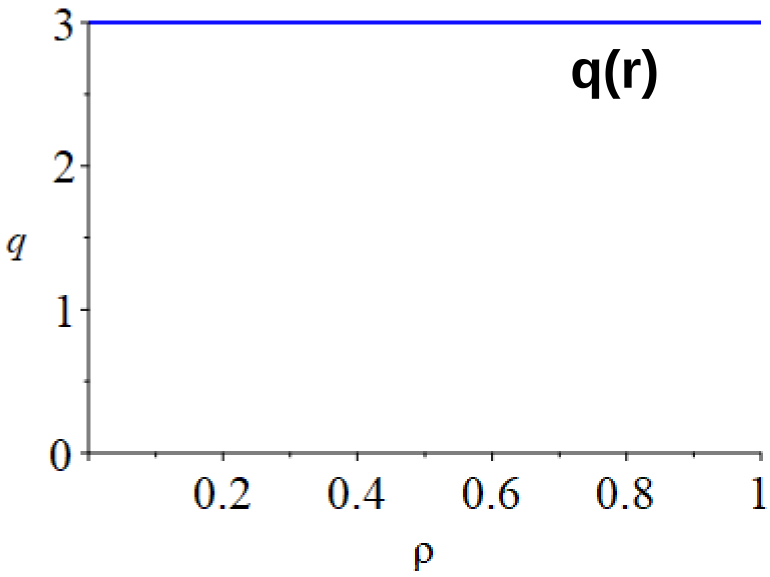


For similar integral plasma parameters, discharge properties can vary significantly

$\rightarrow$ Tokamak operation scenario best characterized by *shape* of $p(r)$, $j(r)$

Link between the current profile j(r) and the safety factor profile q(r)





The radial profile of the toroidal current is directly linked to the radial profile of the safety factor q

$$q \approx \frac{r}{R}\frac{B_{tor}}{B_{pol}} \propto \frac{r^2}{R}\frac{B_{tor}}{I_p(r)}$$

where $I_p(r)$ is the total current inside r :

$$I_p(r) = 2\pi \int_{r'=0}^{r'=r} r'dr'j(r')$$

(formula only holds for a large aspect ratio torus with cylindrical cross-section)

This dimensionless quantity is very important for MHD stability

- too low q leads to kinking of plasma column

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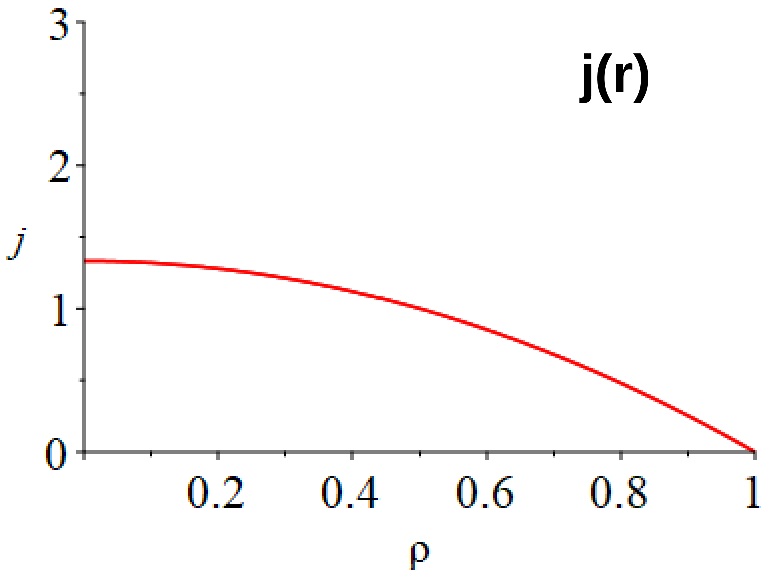

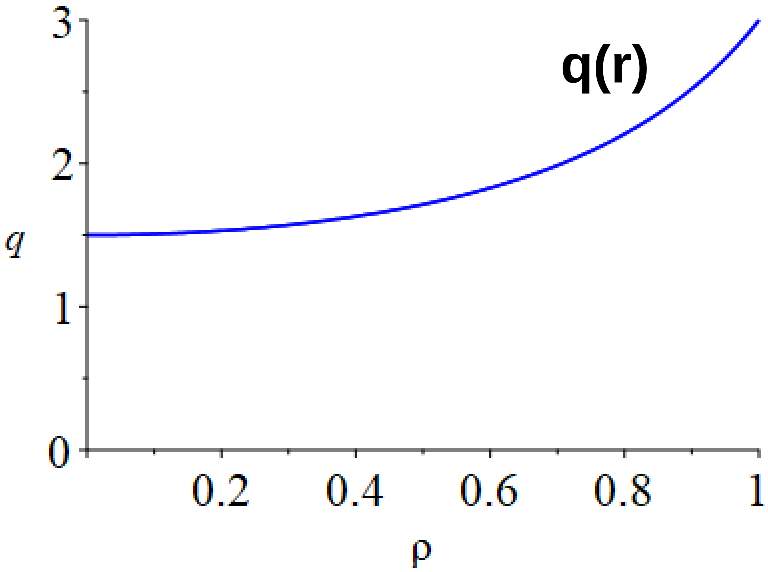

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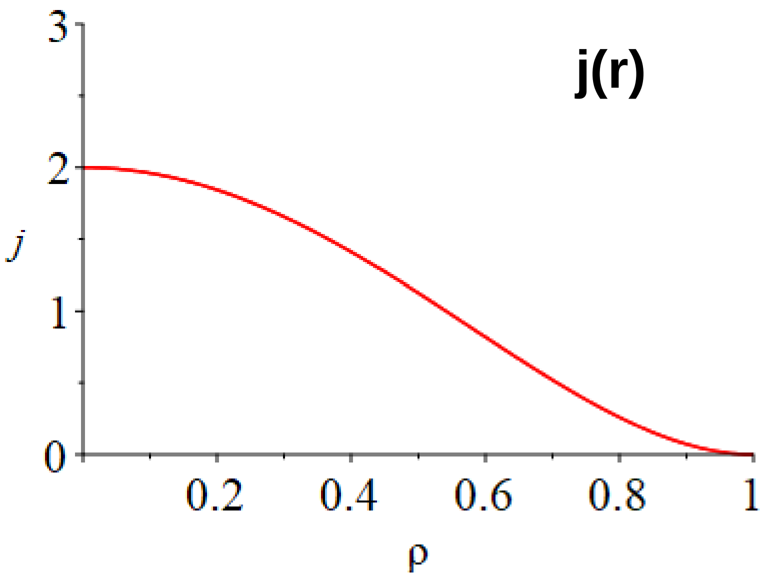

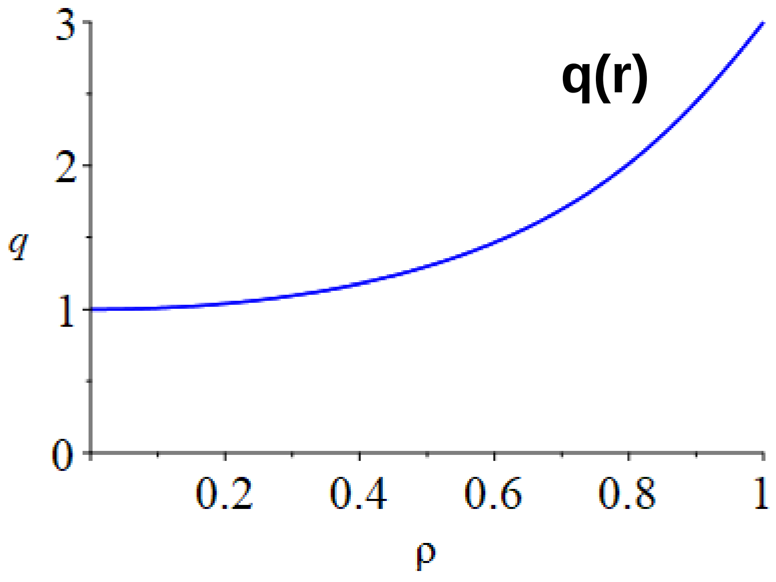

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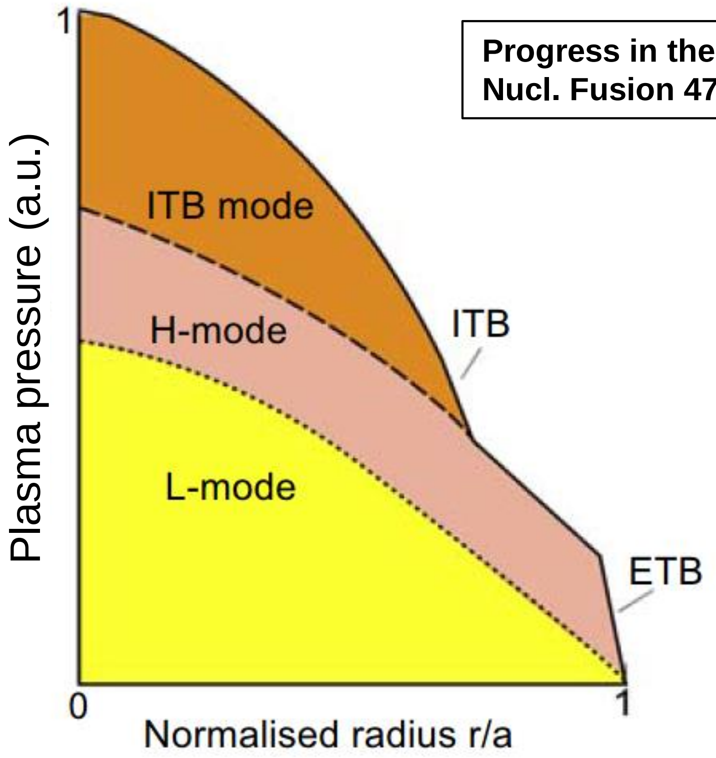
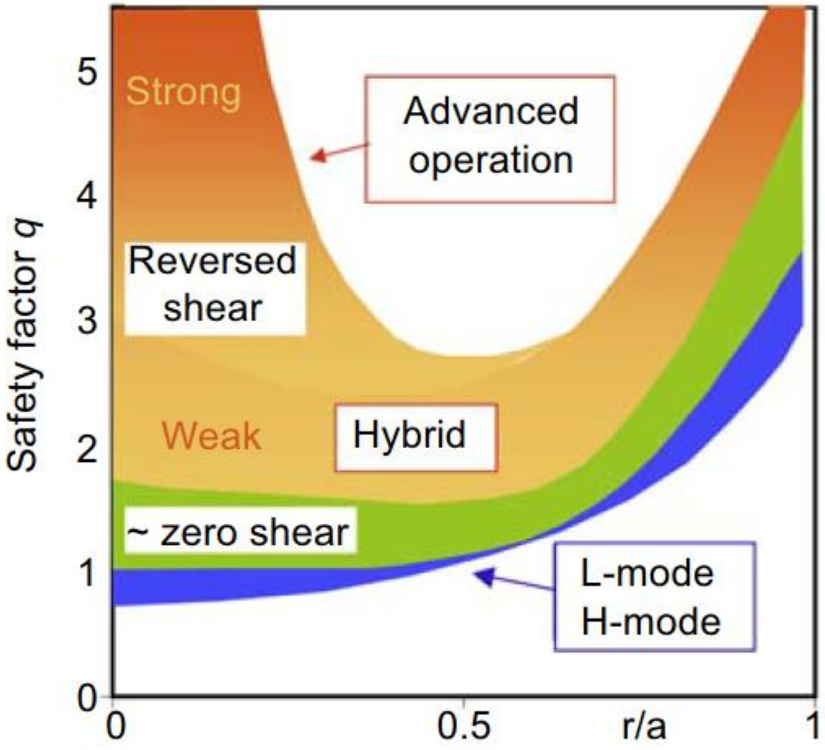
What is a ‚tokamak operation scenario‘?



Progress in the ITER Physics Basis
Nucl. Fusion 47 (2007) S285



Tokamak operation scenario best characterised by *shape* of *p(r), j(r)*

- pressure profile composed of kinetic profiles $p(r) = n_e(r)\ T_e(r) + n_i(r)\ T_i(r)$
- current profile usually parametrised by safety factor $q(r)$

Outline

1) Introduction (just given)

2) Optimisation strategies for tokamak plasmas

3) Scenarios characterised by j(r) and p(r)

4) Scenario access

5) Summary

Outline

Figure of merit for fusion performance $nT\tau_E$

Aim is to generate power, so

- P_{fusion}/P_{ext} should be high

$P_{ext} = P_{heat} - P_\alpha$ determined by thermal insulation:

- $\tau_E = W_{plasma}/P_{heat}$ (energy confinement time)

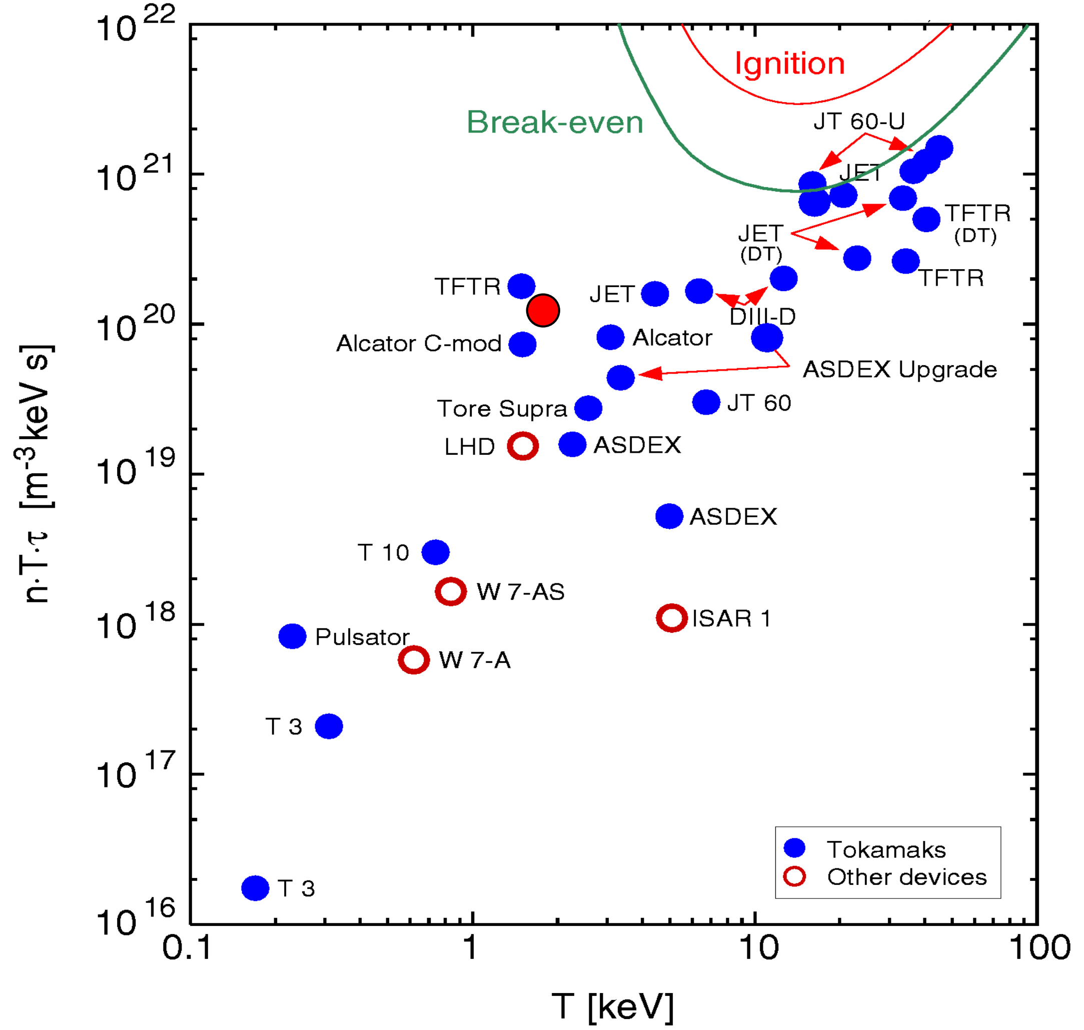
In present day experiments, P_{heat} comes from external heating systems

- $Q = P_{fus}/P_{ext} \approx P_{fus}/P_{heat} \sim (nT)^2/(nT/\tau_E) \sim nT\tau_E$

In a reactor, P_{heat} mainly by α-(self)heating:

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

(ignited plasma, Q no longer $\sim nT\tau_E$)



$\rightarrow$ Optimising fusion performance means optimizing $nT\tau_E$

Optimisation of $nT\tau_E$: ideal pressure limit

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

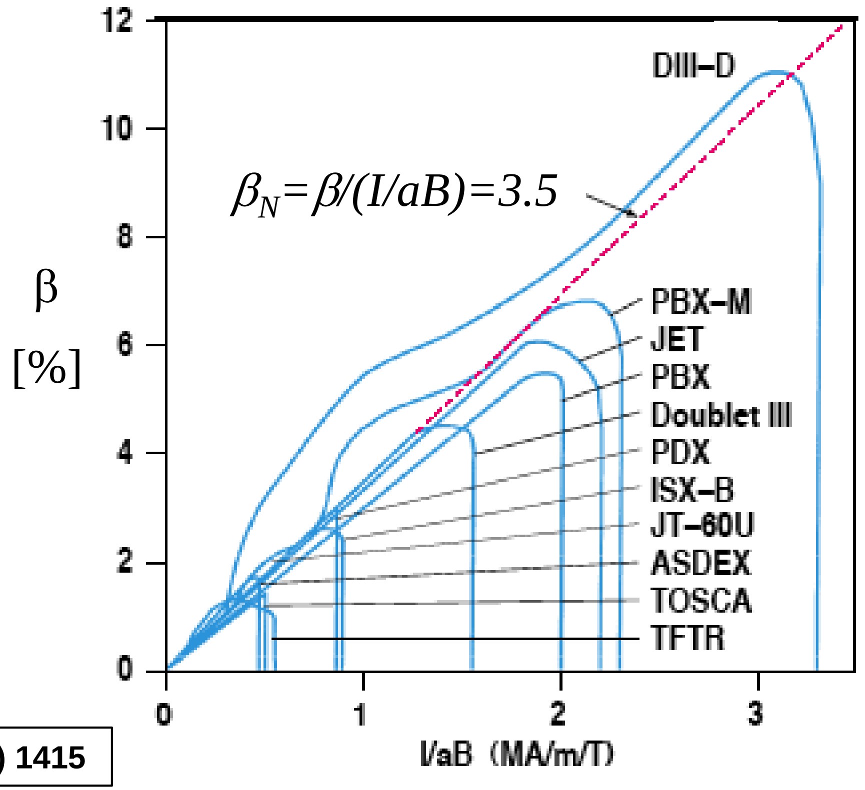
This quantity is limited by magneto-hydrodynamic (MHD) instabilities

'Ideal' MHD limit (ultimate limit, plasma unstable on Alfvén (inertial) time scale ~ 10s of μs)

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

$$\beta_N = {}^{\beta}\!/_{I_p/(aB)} \propto \beta\, A\, q$$

- high plasma current beneficial for achieving high β
- at fixed aB, shaping of plasma cross section allows higher I_p (low q limit, see later)

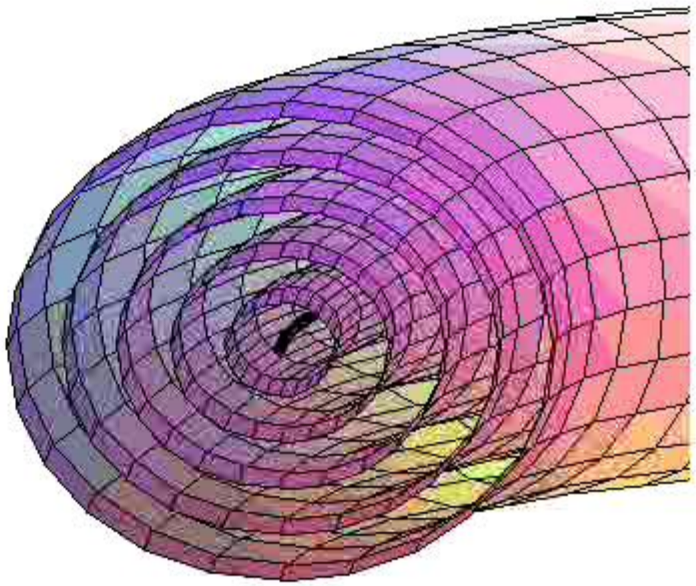


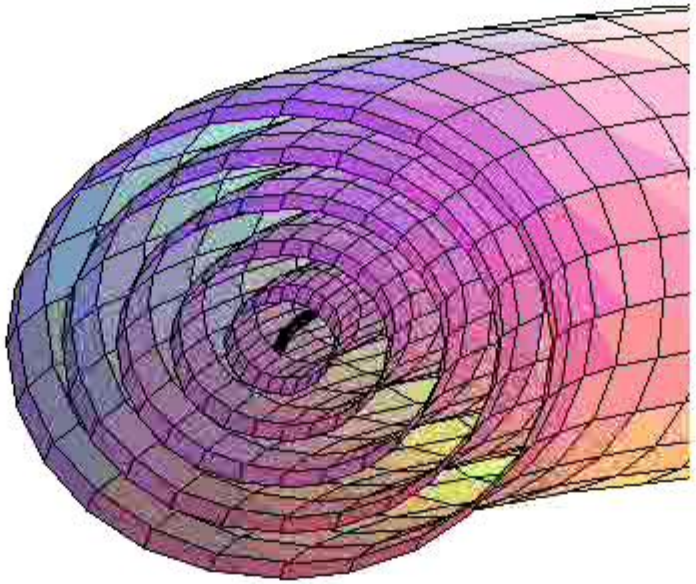
E.J. Strait et al., Phys. Plasmas 1 (1994) 1415

Optimisation of $nT\tau_E$: resistive pressure limit

In an ideal MHD stable plasma, resistive MHD instabilities may occur, leading to magnetic islands

'Resistive' MHD limit (on local current redistribution time scale ~ 10s of ms)



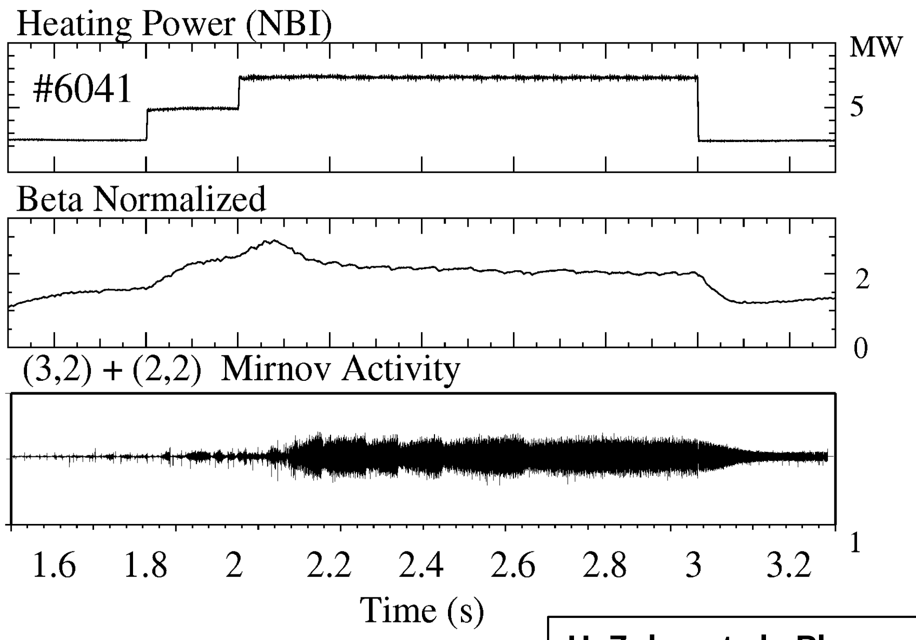


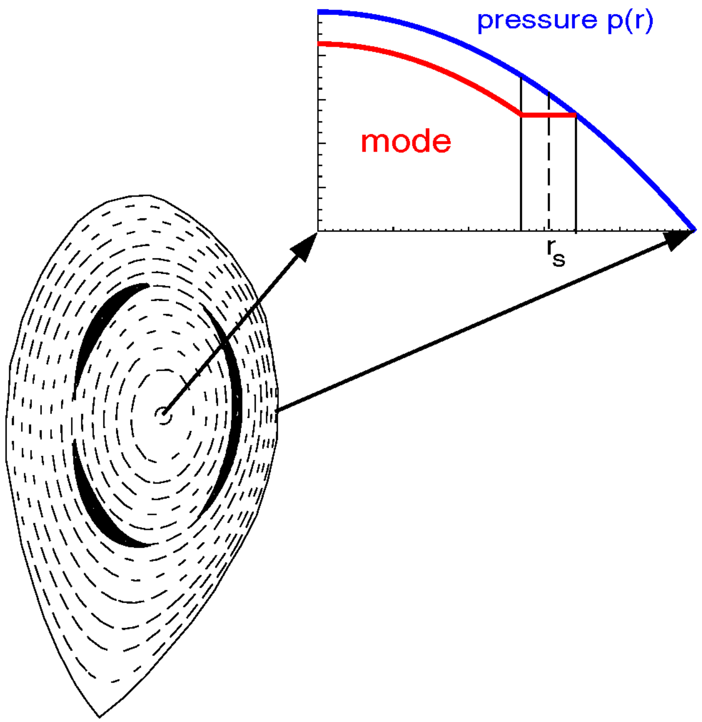
Optimisation of $nT\tau_E$: resistive pressure limit

In an ideal MHD stable plasma, resistive MHD instabilities may occur, leading to magnetic islands

‘Resistive’ MHD limit (on local current redistribution time scale ~ 10s of ms)

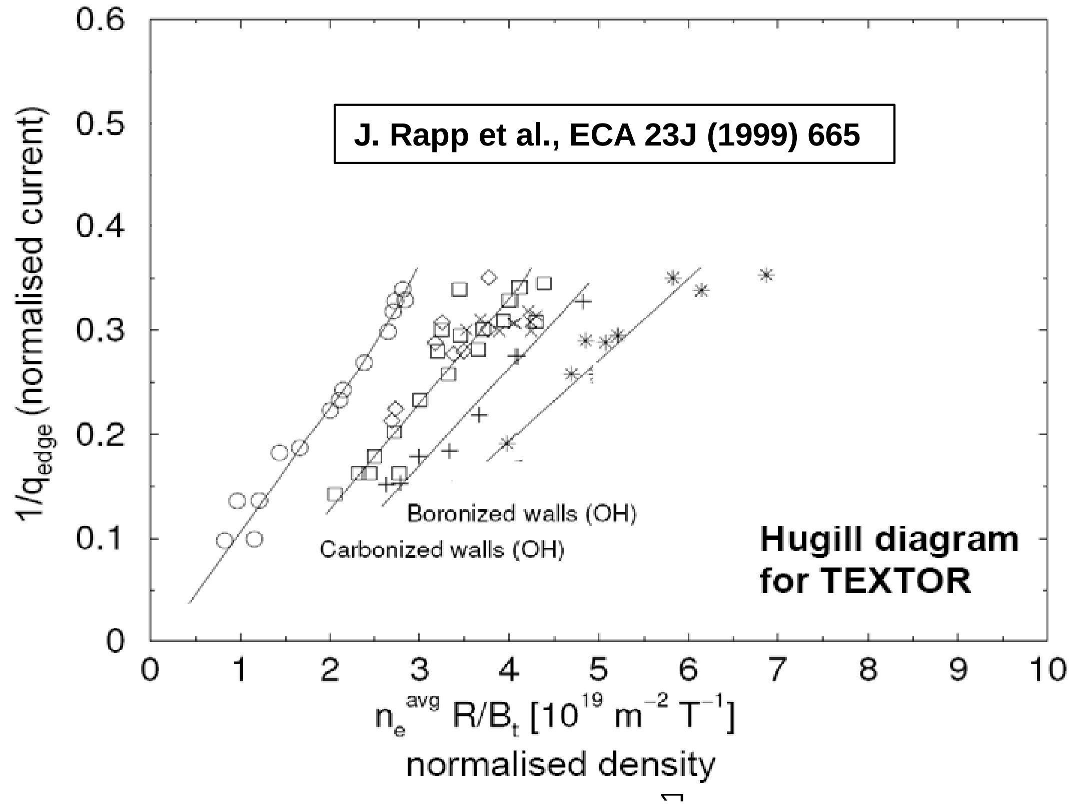
- ‘Neoclassical Tearing Mode’ (NTM) driven by loss of pressure driven ‘bootstrap’ current within magnetic island

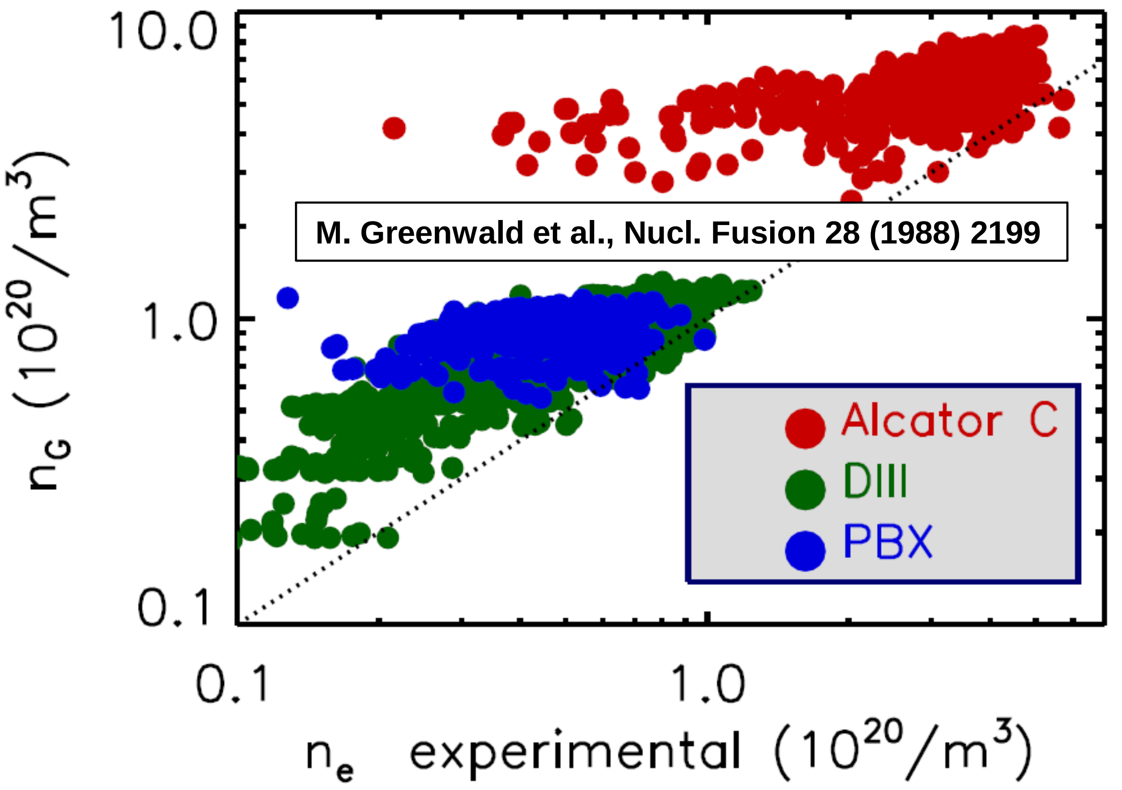




H. Zohm et al., Plasma Phys. Control. Fusion 37 (1995) A313

Optimisation of $nT\tau_E$: density limit



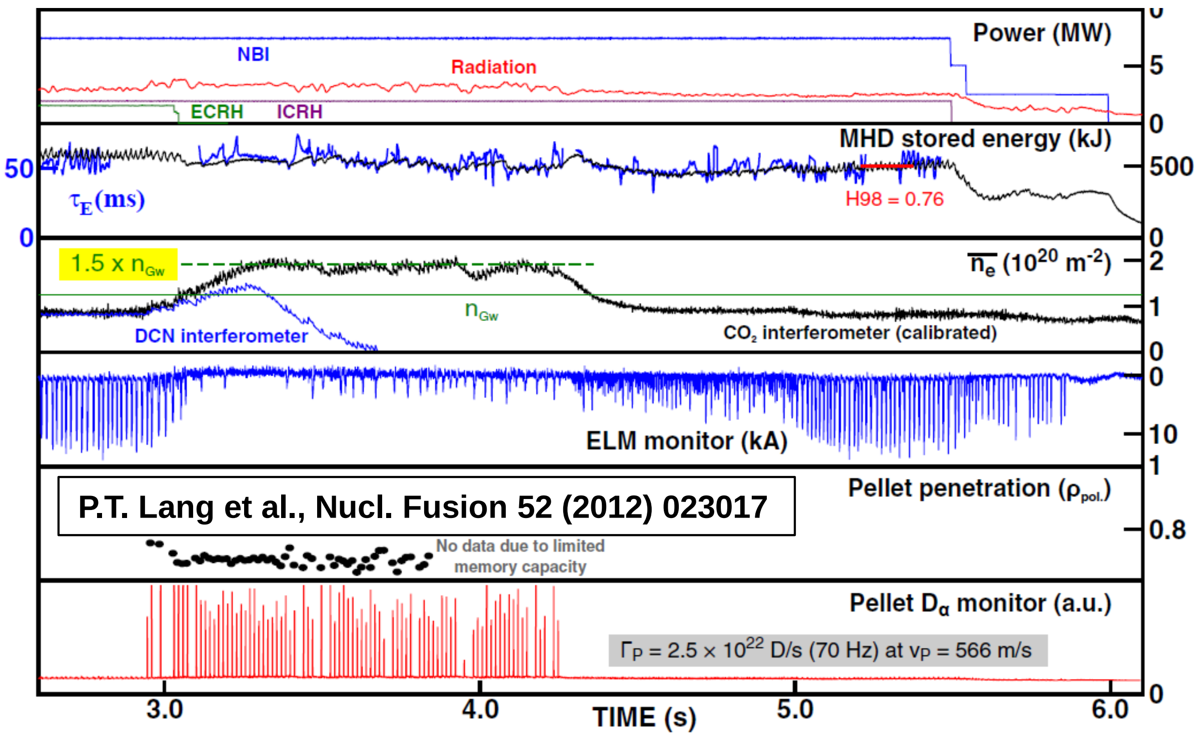


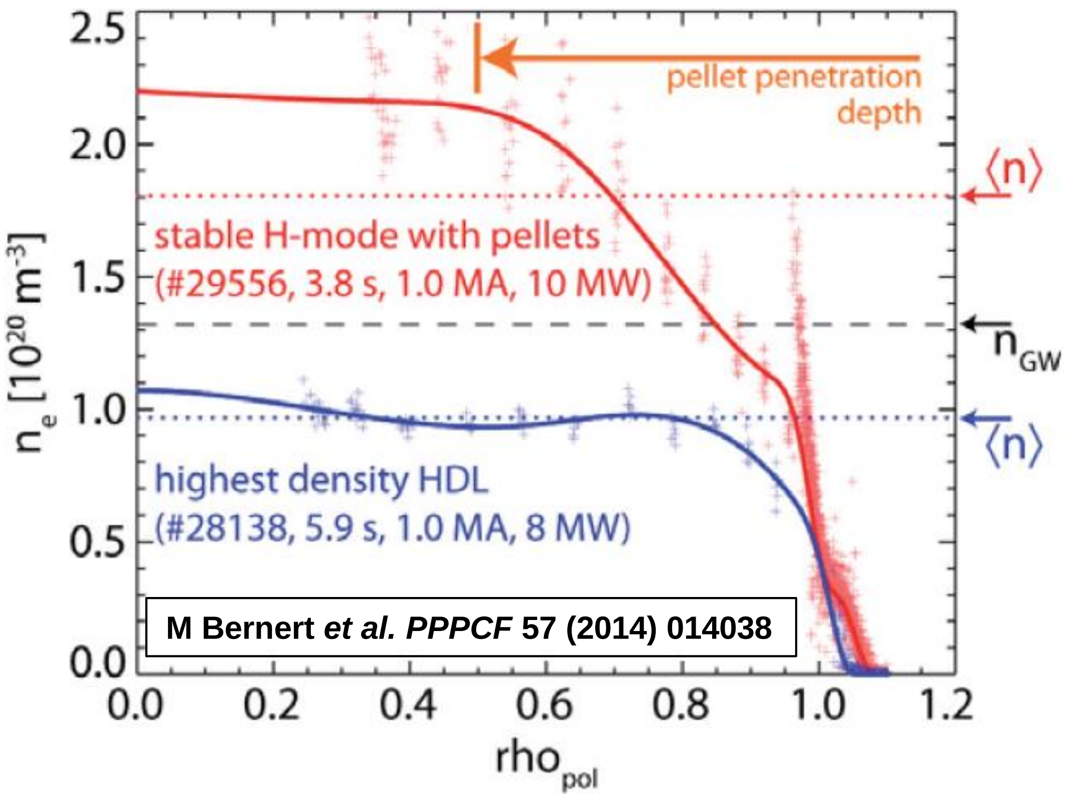
Since *T* has an optimum value at ~ 20 keV, *n* should be as high as possible

- density is limited by disruptions due to excessive edge cooling
- *empirical* 'Greenwald' limit, $n_{GW} \sim I_p/(\pi a^2) \rightarrow$ high I_p helps to obtain high *n*

$\boxed{n_{GW}=I_p/(\pi a^2)}$

N.B.: Operation at $n/n_{GW} > 1$ is possible

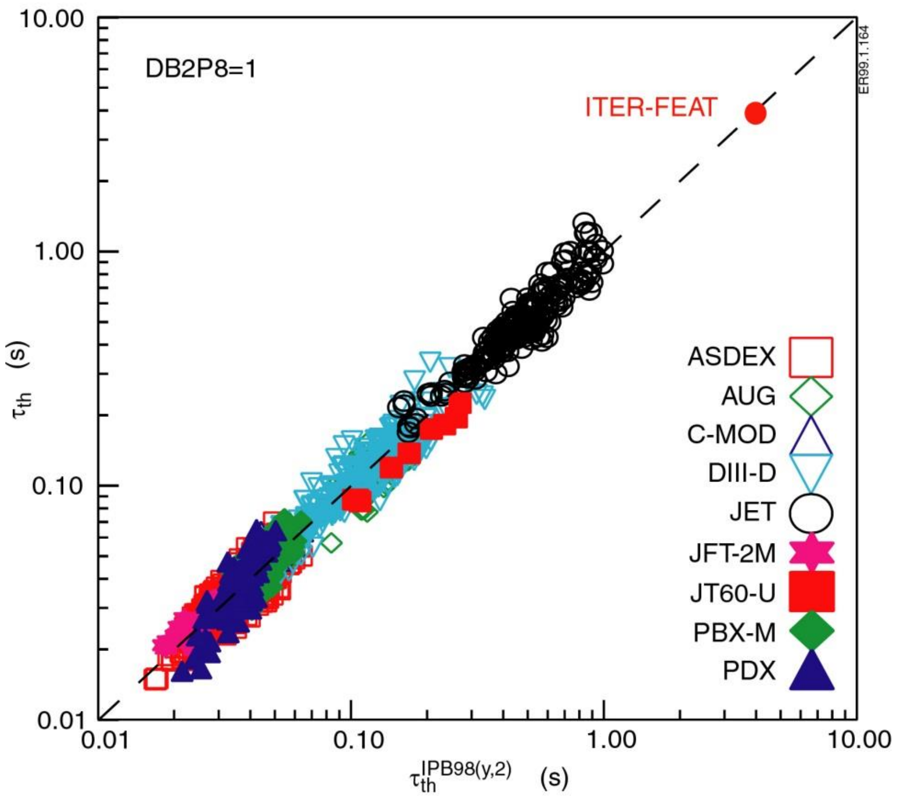




Greenwald limit sets a limitation to the edge density, not necessarily the central density

- can be overcome by central fueling ('pellets') or through inward 'pinch' for particles

Optimisation of $nT\tau_E$: confinement scaling



Empirical ITER 98(p,y) scaling:

$$\tau_E \sim H\, I_p^{0.93}\, P_{heat}^{-0.63}\, B_t^{0.15} \ldots$$

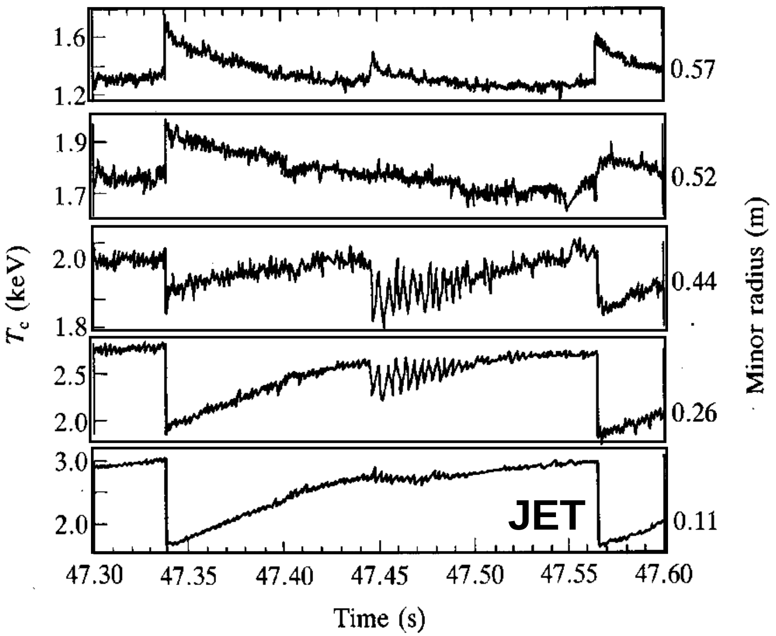
ITER Physics Basis
Nucl. Fusion 39 (1999) 2203

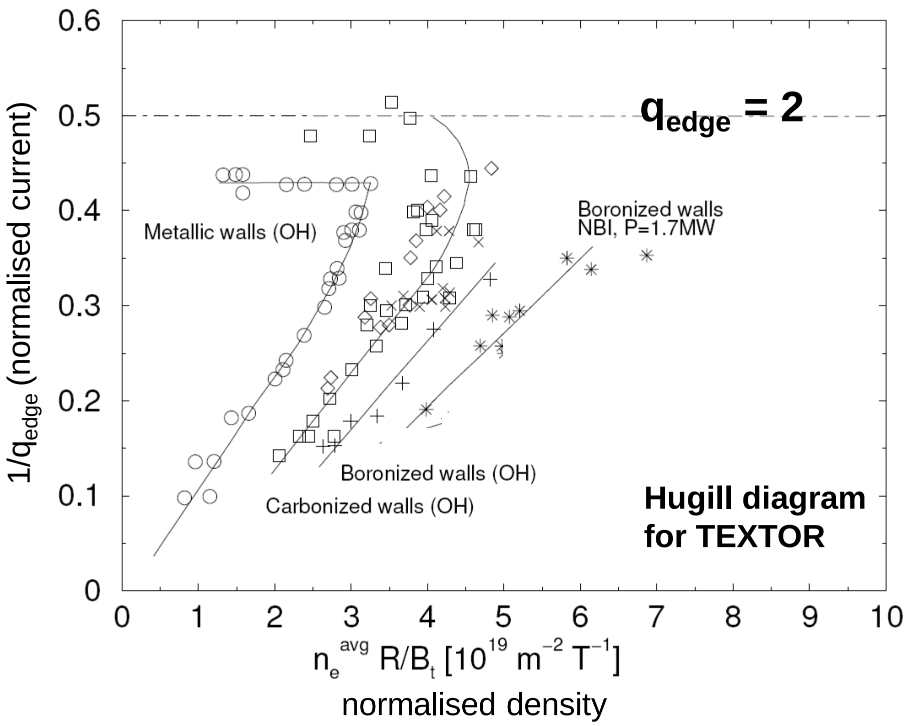
Empirical confinement scalings show linear increase of τ_E with I_p

- note the power degradation (τ_E decreases with P_{heat}!)
- 'H-factor' H measures the quality of confinement relative to the scaling

Optimisation of $nT\tau_E$: current limit

BUT: for given B_t, I_p and j(0) are limited by current gradient driven MHD instabilities





Limit to safety factor $q \sim (r/R)\ (B_{tor}/B_{pol})$

- for q < 1, tokamak unconditionally unstable → central ‘sawtooth’ instability
- for $q_{edge} \rightarrow 2$, plasma tends to disrupt (external kink) – limits value of I_p (usually want $q_{edge} \geq 3$)

Additional constraints from power exhaust

Tolerable heat flux on components limits power flux across the separatrix

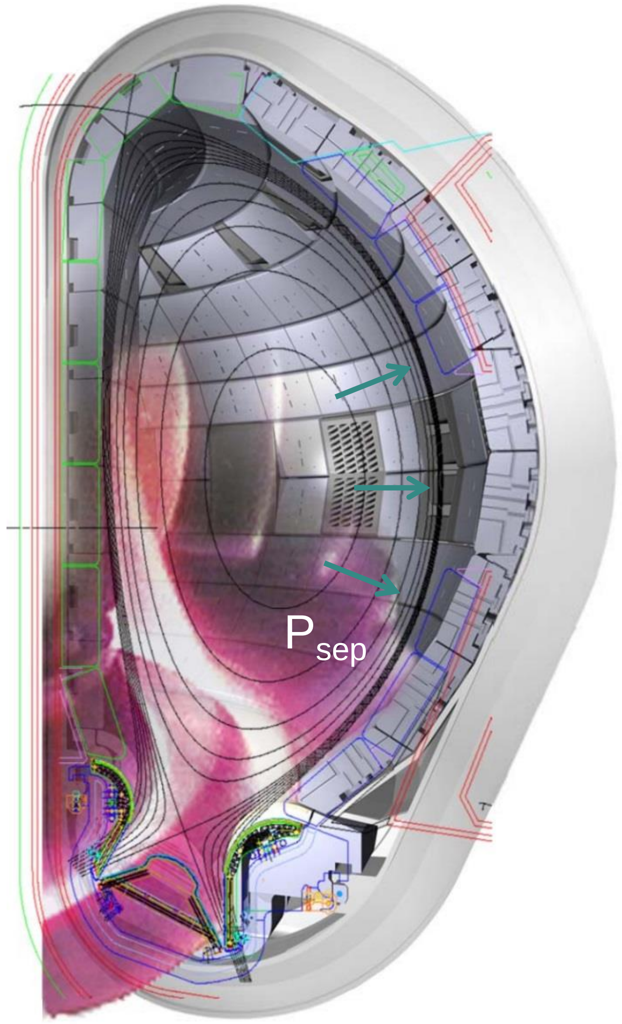
- heat comes down to the divertor in narrow layer of width λ_q ~ cm
- experimentally, λ_q does not increase with R ☹

Need to dissipate power (additional radiation, charge exchange) before it reaches the target plate

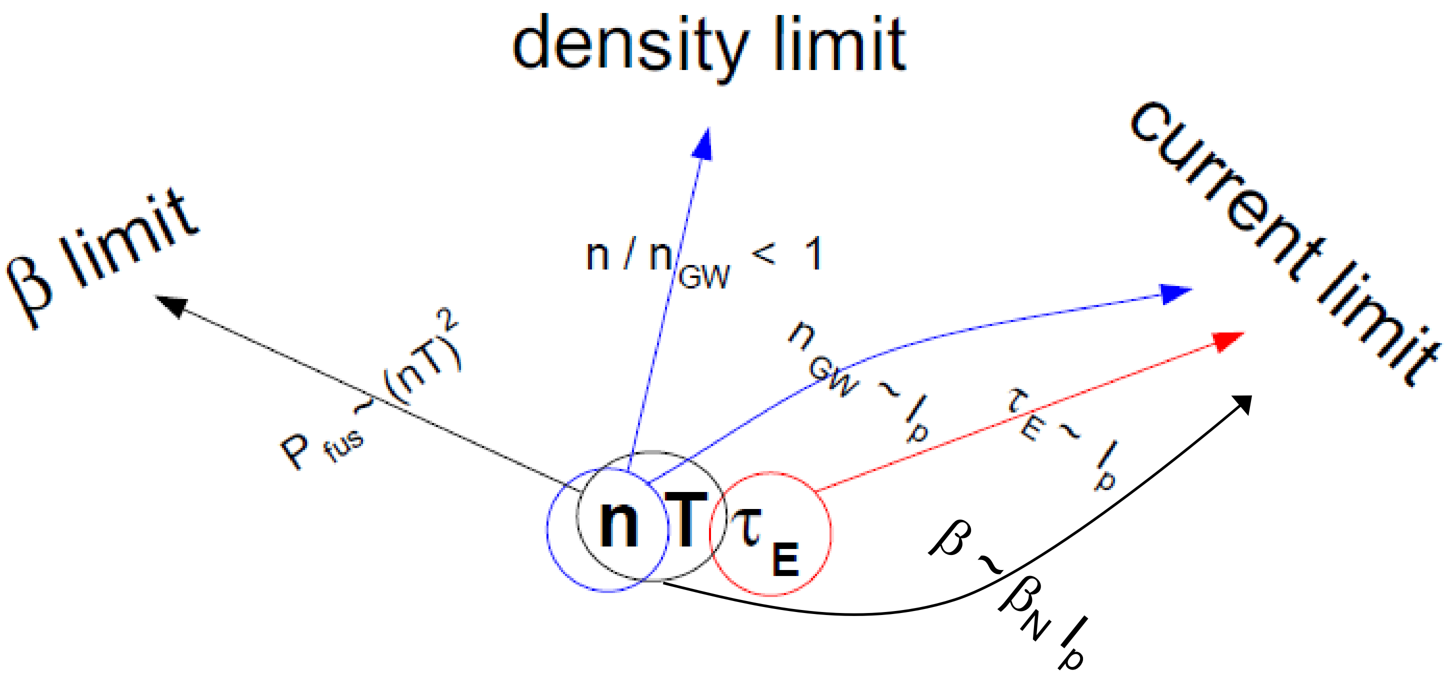
- additional constraint: excessive radiation decreases central

(not addressed in the remainder of the talk, but serious, $\rightarrow$ 2019 IIS)

N.B.: difficult to study in present day devices since exhaust needs high density, current drive needs low density…



Optimisation of $nT\tau_E$



Optimising for $Q=P_{fus}/P_{ext}$ drives operational point close to operational limits

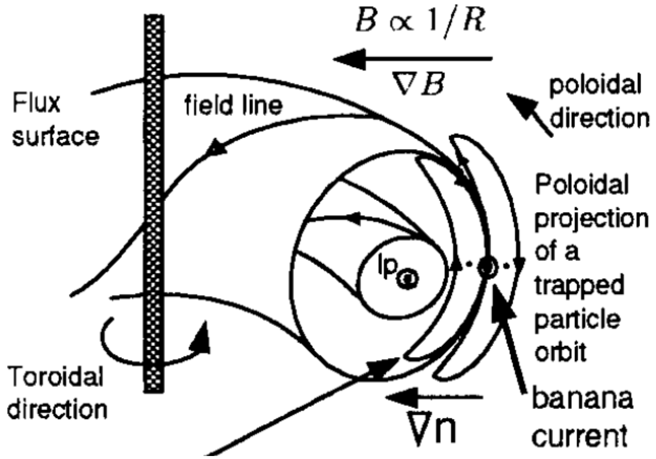
Tokamak optimisation: steady state operation

For steady state tokamak operation (100% noninductive), high I_p is not desirable

- external CD has low efficiency (usually less than 0.1 A per W)
- internal bootstrap current high for high $j_{bs} \sim (r/R)^{1/2}\ \nabla p/B_{pol} \ \rightarrow f_{bs} = I_{bs}/I_p = c_{bs} A^{-1/2}\beta_{pol}$

Optimisation strategies for pulsed and steady state scenarios differ

- ideal stability limits ~ β_N, fusion power ~ β^2, bootstrap fraction ~ β_p



A.G. Peeters, Plasma Phys. Control. Fusion 42 (2000) B231.

Optimise for fusion power and Q (pulsed tokamak, ITER Q=10):

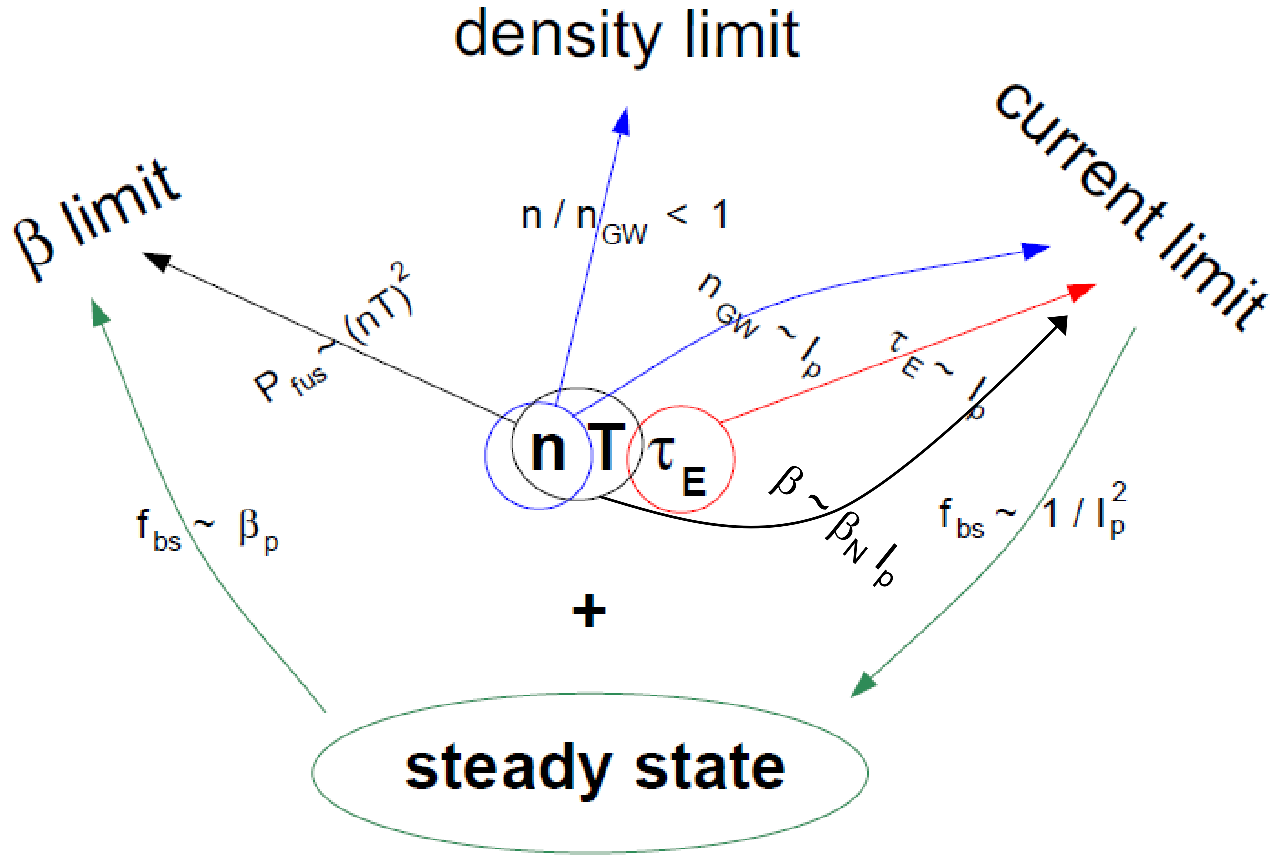
$$\beta \sim \frac{\beta_N}{Aq} \qquad \Rightarrow \text{low q operation}$$

Optimise for bootstrap fraction (steady state tokamak, ITER Q=5)

$$\beta_p \sim \beta_N\, Aq \qquad \Rightarrow \text{high q operation (in conflict with ignition)}$$

$\Rightarrow$ Steady state tokamak at high q strongly profits from higher H and β_N:

Optimisation of $nT\tau_E$

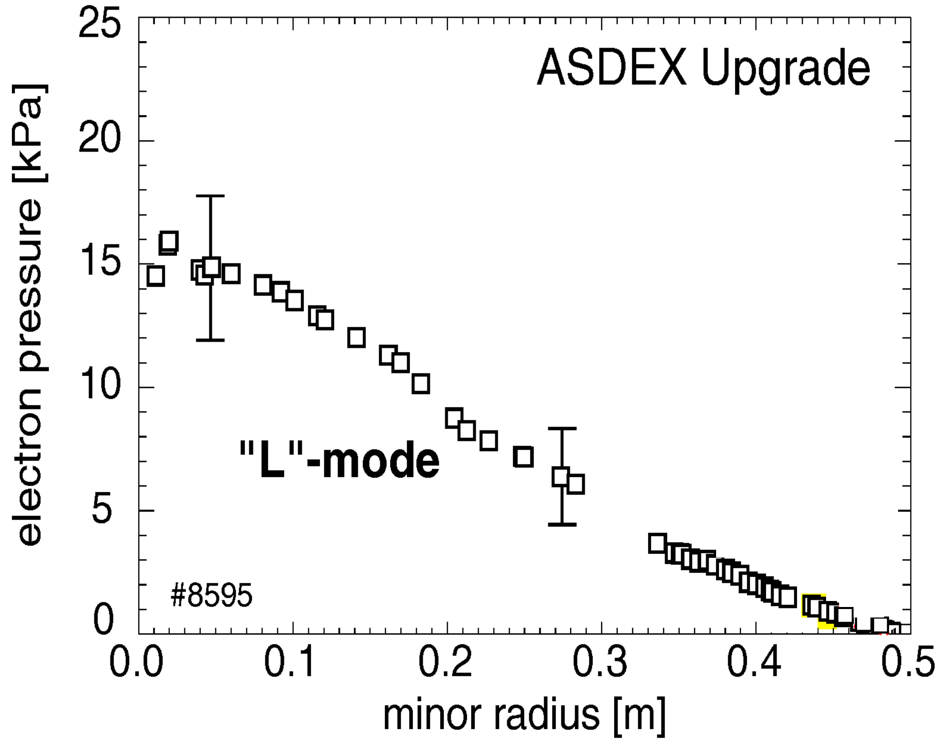


Including the steady state constraint emphasizes the β-limit

(and de-emphasizes current limit)

Outline

The (low confinement) L-mode scenario

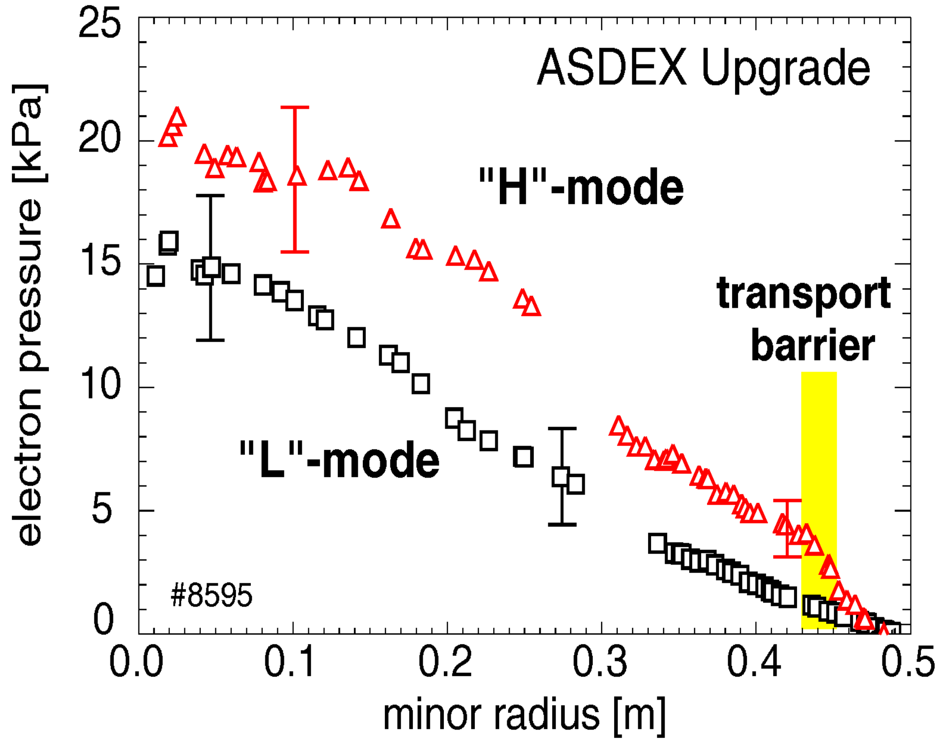


W. Suttrop et al., Plasma Phys. Control. Fusion 39 (1997) 2051

Standard scenario without special tailoring of geometry or profiles

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

The (high confinement) H-mode scenario

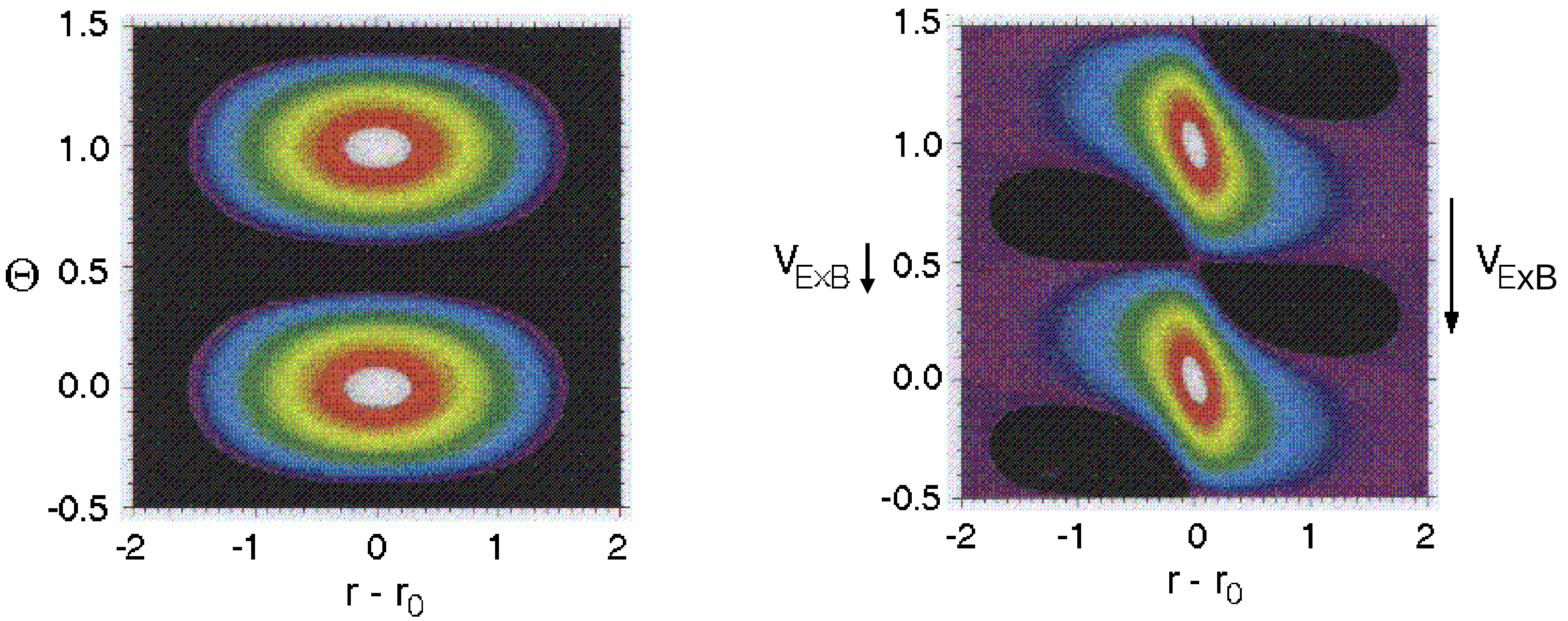


W. Suttrop et al., Plasma Phys. Control. Fusion 39 (1997) 2051

With hot (low collisionality) conditions, edge transport barrier develops when $P_{heat} > P_{threshold}$

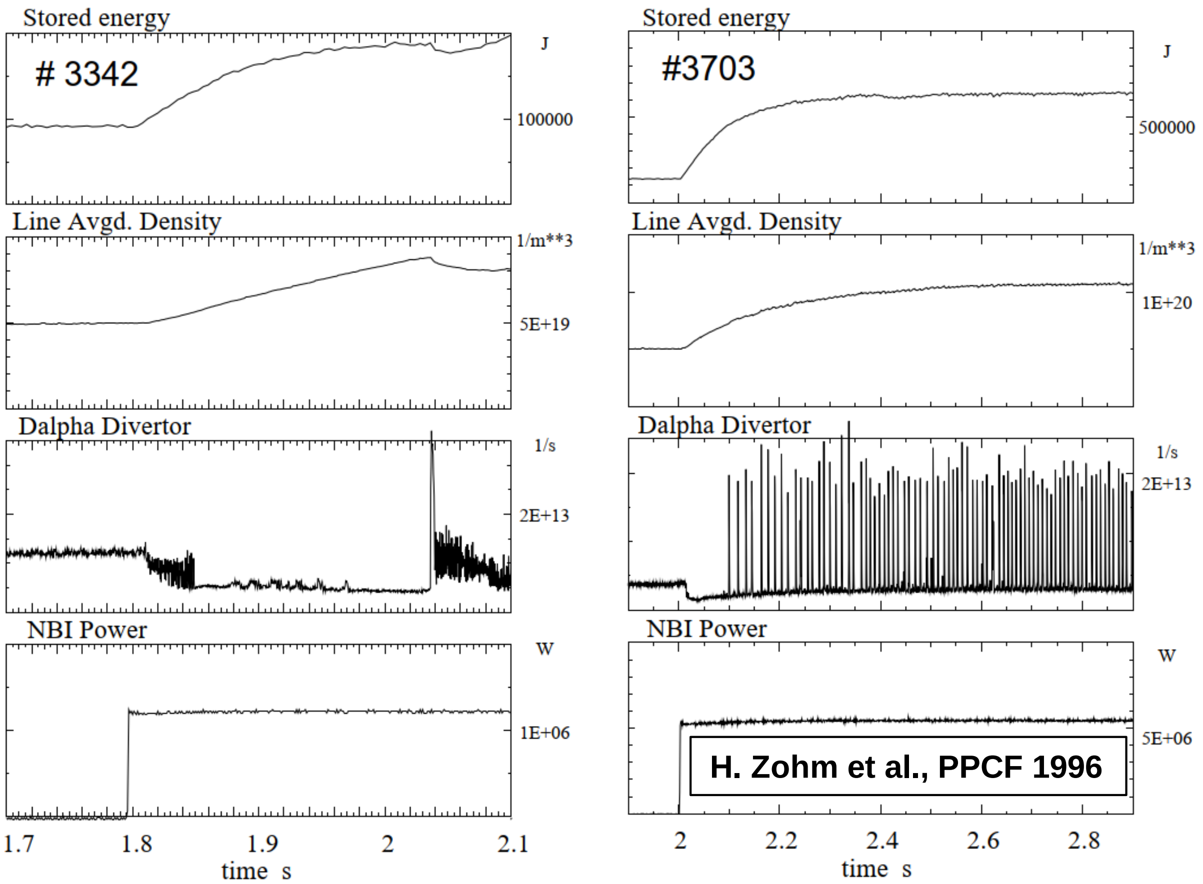
- gives higher boundary condition for ‘stiff’ core temperature profiles
- global confinement τ_E roughly factor 2 better than L-mode
- extrapolates to more attractive (R ~ 8 m, I_p ~ 20 MA) pulsed reactor

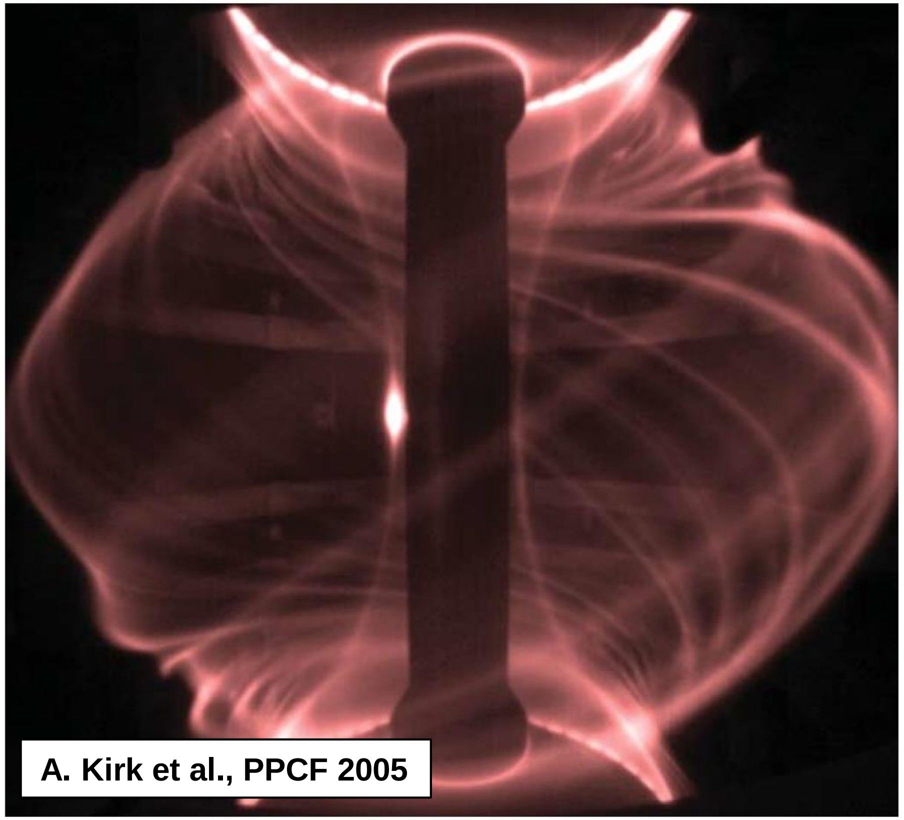
Mechanism for edge transport barrier formation



- in a very narrow (~1 cm) layer at the edge very high plasma rotation develops ($E = v \times B$ several 10s of kV/m)
- sheared edge rotation tears turbulent eddies apart
- smaller eddy size leads to lower radial transport ($D \sim \delta r^2/\tau_{decor}$)

Standard H-mode discharge stationary through ELMs

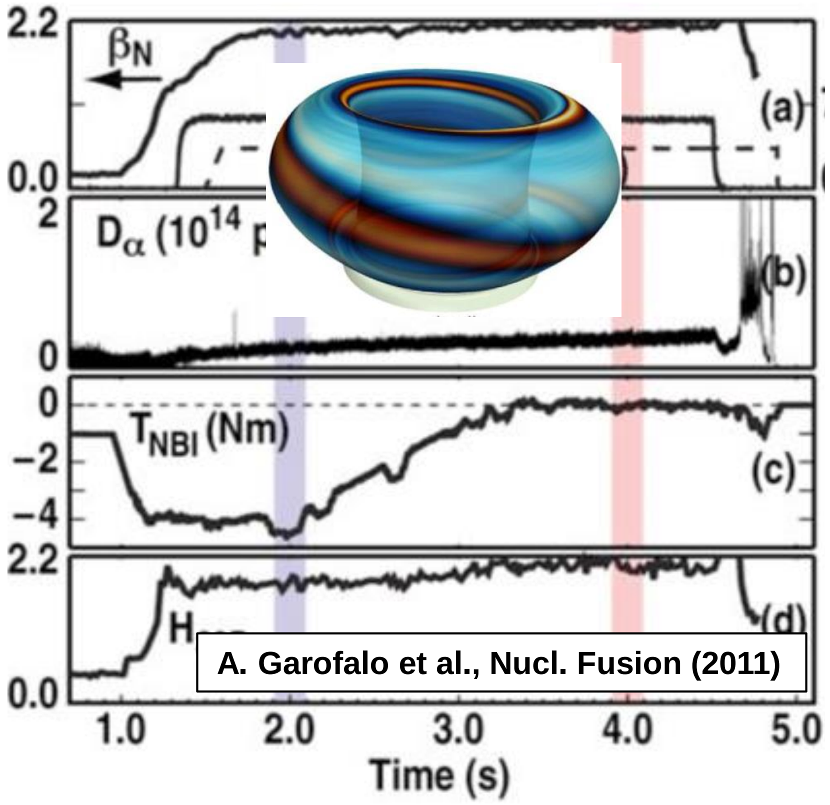




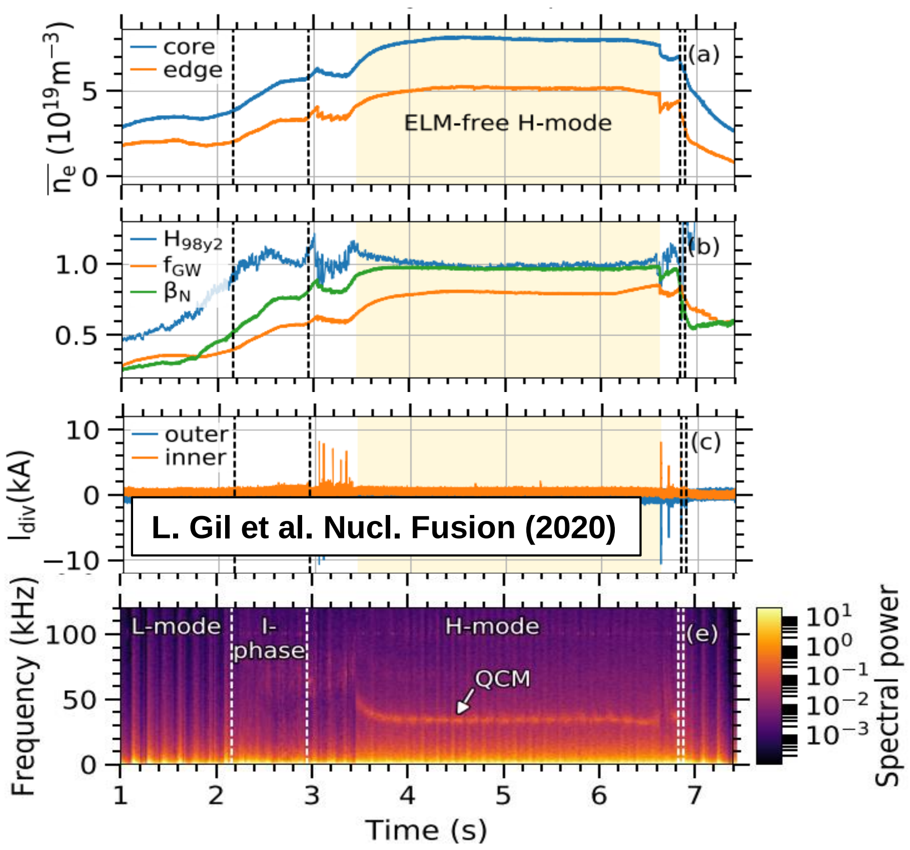
Steep edge gradients drive periodic relaxation instability: Edge Localised Modes (ELMs)

- beneficial for particle (impurity) exhaust, but heat pulses challenge divertor components

H-mode scenarios with small or no ELMs



,QH' mode: ELMs replaced by saturated kink



,EDA' mode: ELMs replaced by quasicoherent turbulence

In recent years, several small/no ELM scenarios have been developed

- key: extra pedestal transport that flushes impurities, but keeps pressure below MHD limit

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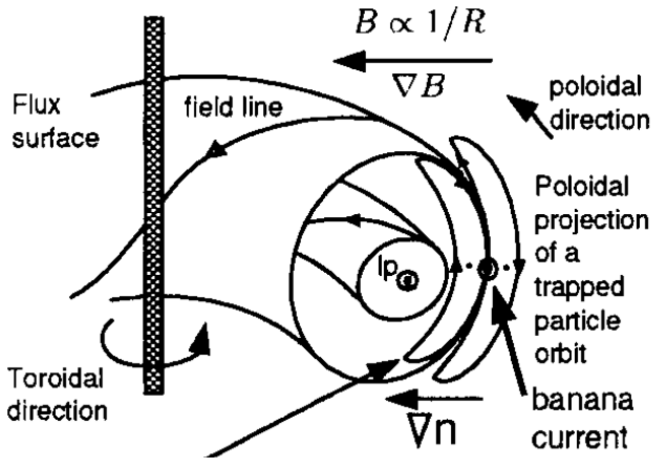

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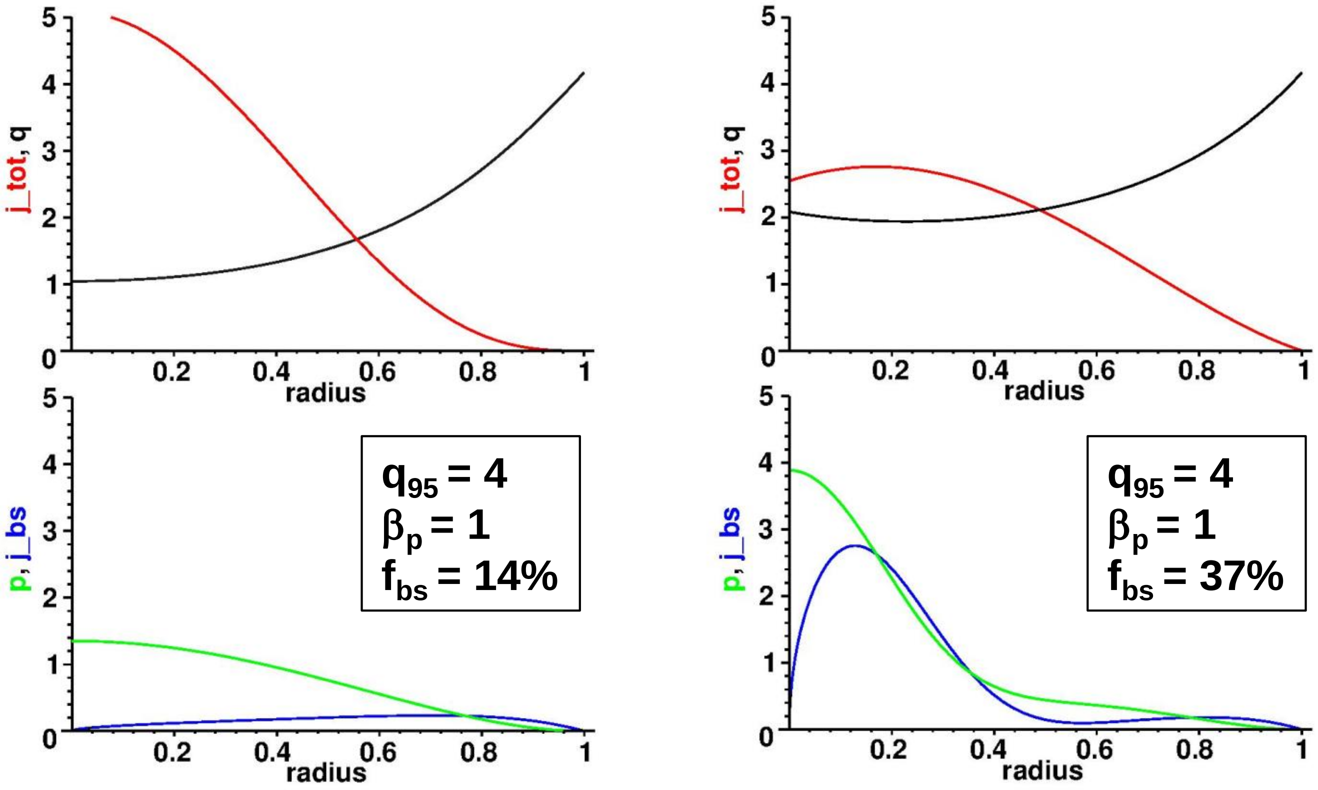
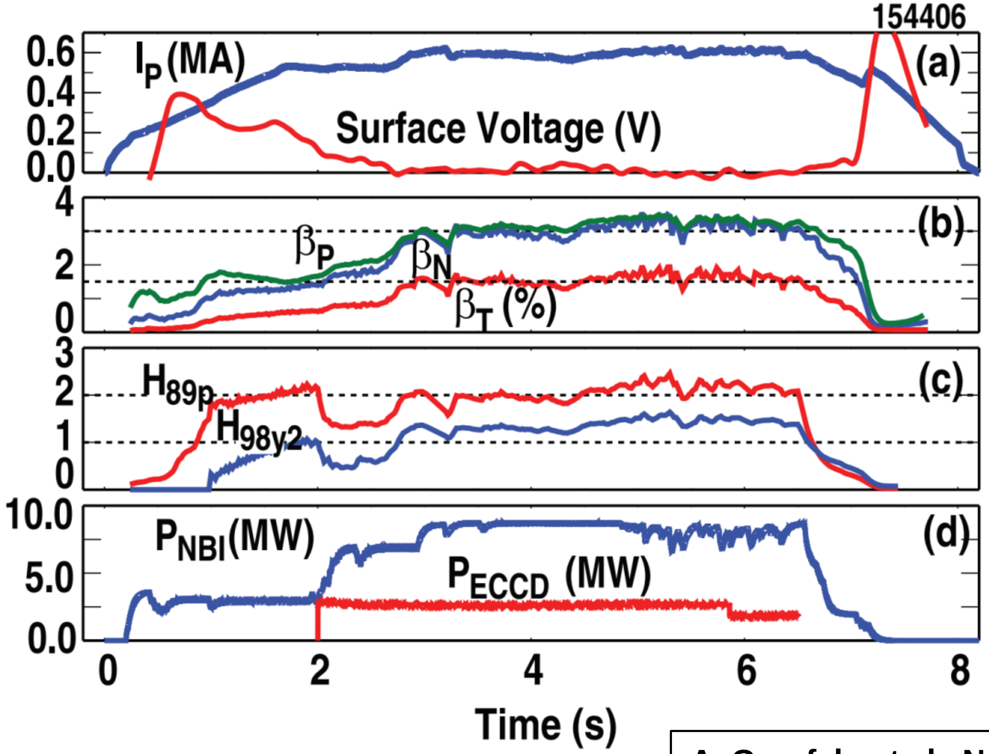
How to maximise the bootstrap fraction – 1-D

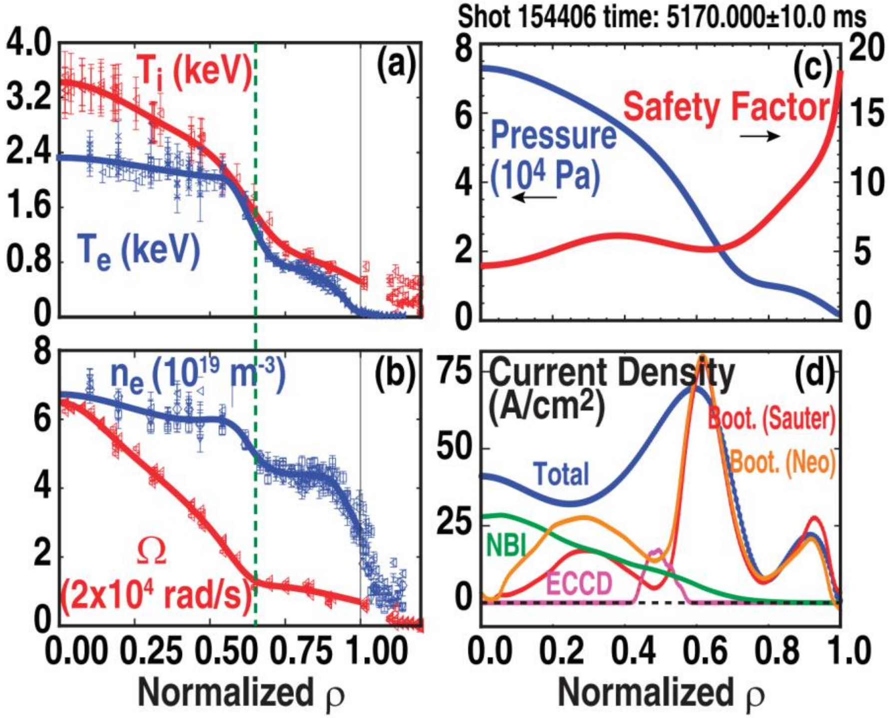


Profile shapes *p(r)* and *j(r)* affect c_{bs}: $j_{bs} \propto \sqrt{\frac{r}{R}}\frac{\nabla p}{B_{pol}} \propto q\,\nabla p$ $\Rightarrow$ elevate q where p drops

- leads to *j(r)* being peaked off-axis and *q(r)* being flat or reversed

Reversed shear – ITB scenario





A. Garofalo et al., Nucl. Fusion 55 (2015) 123025

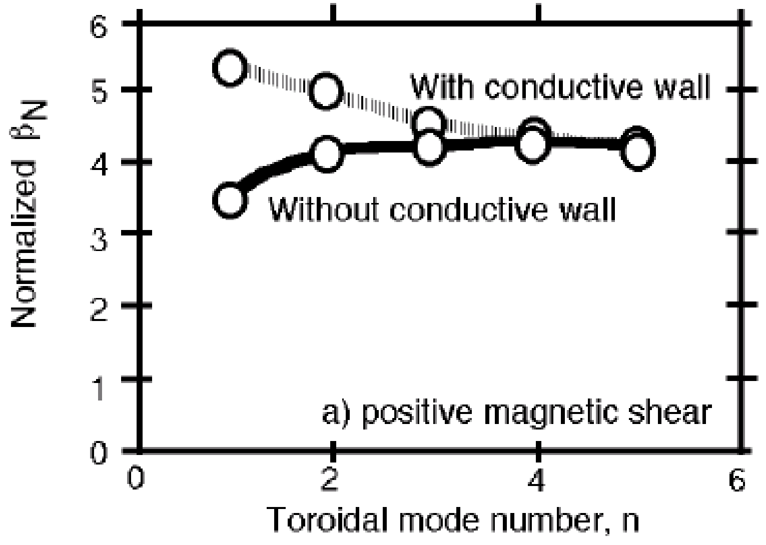
High bootstrap fraction, but convincing demonstration so far only at high q_{95}

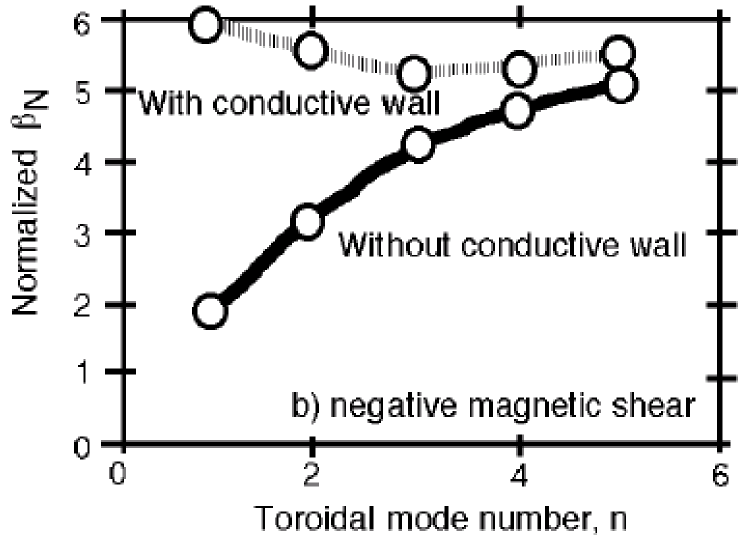
- delicate MHD stability at lower q_{95}

This route is presently followed at high q_{95} (remember $\beta_{pol} \sim \beta_N q_{95}$)

Broad current profiles have a low ideal β-limit

for peaked current profiles (standard H-mode): modest need and gain





For broad current profiles (advanced scenarios): high need and significant gain

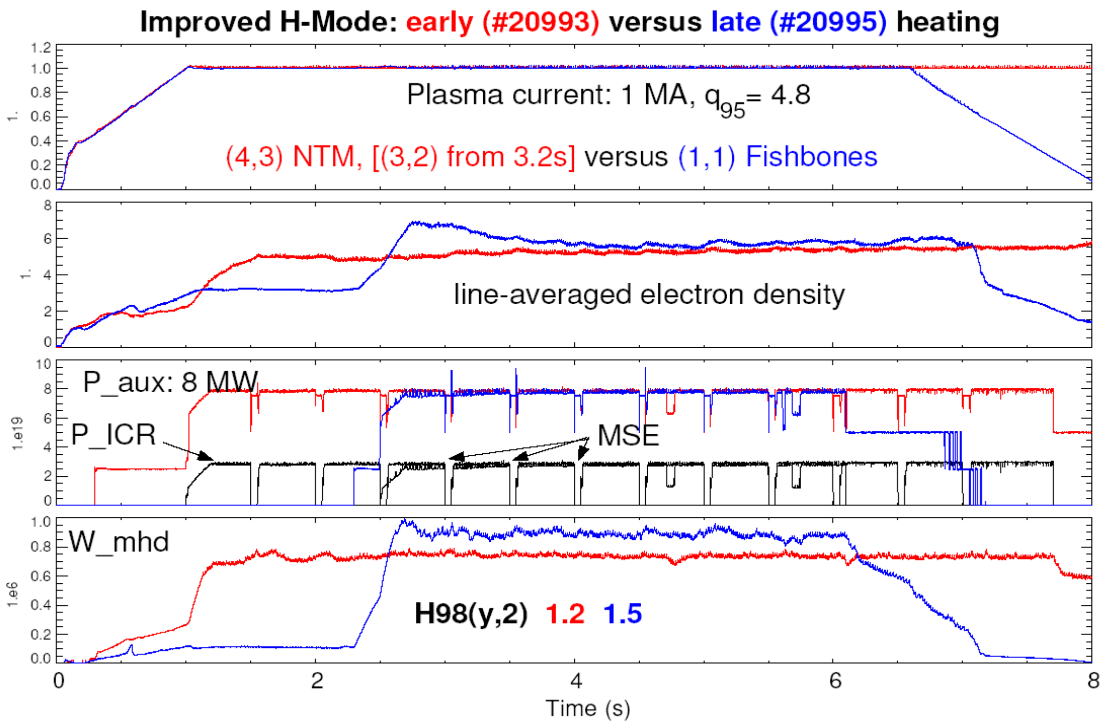
J. Manickam et al., Phys. Plasmas 1 (1994) 1601

…but close-by conducting shell transforms ideal external kink into Resistive Wall Mode (RWM)

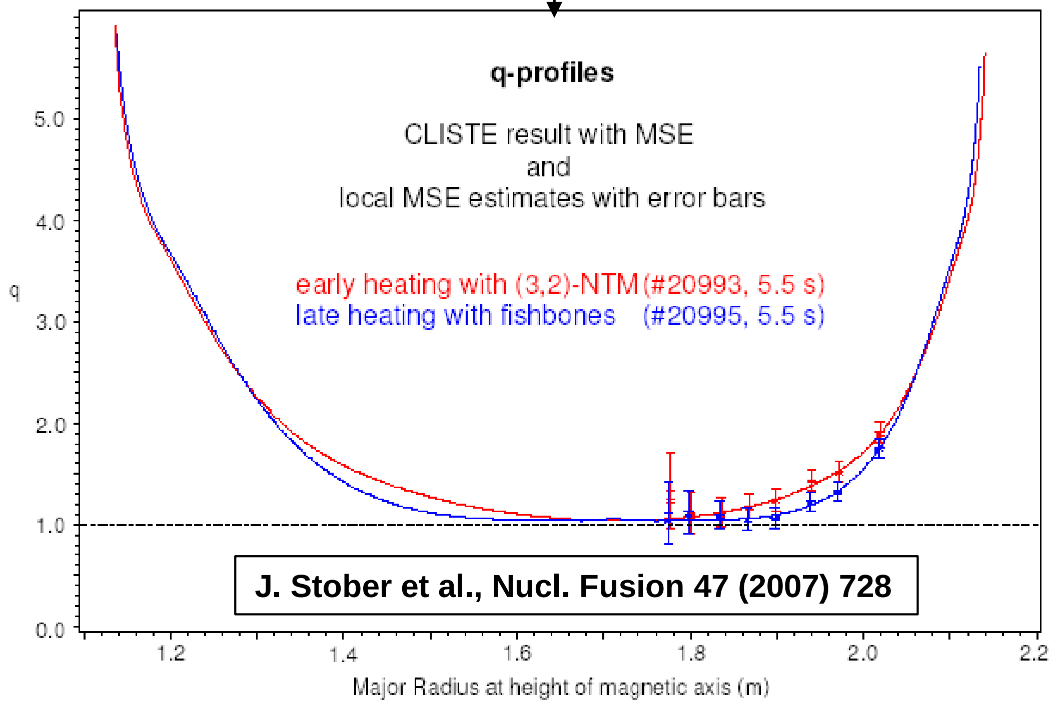
- RWM stability determined by non-ideal effects (rotation, fast particle resonances)

$\Rightarrow$ still an active area of tokamak physics research

A ‘compromise’: the hybrid scenario

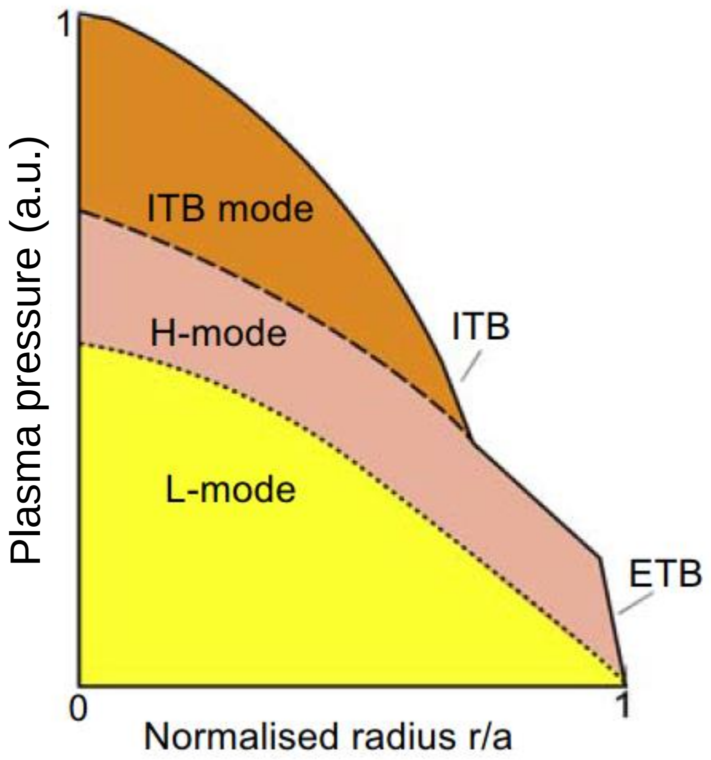


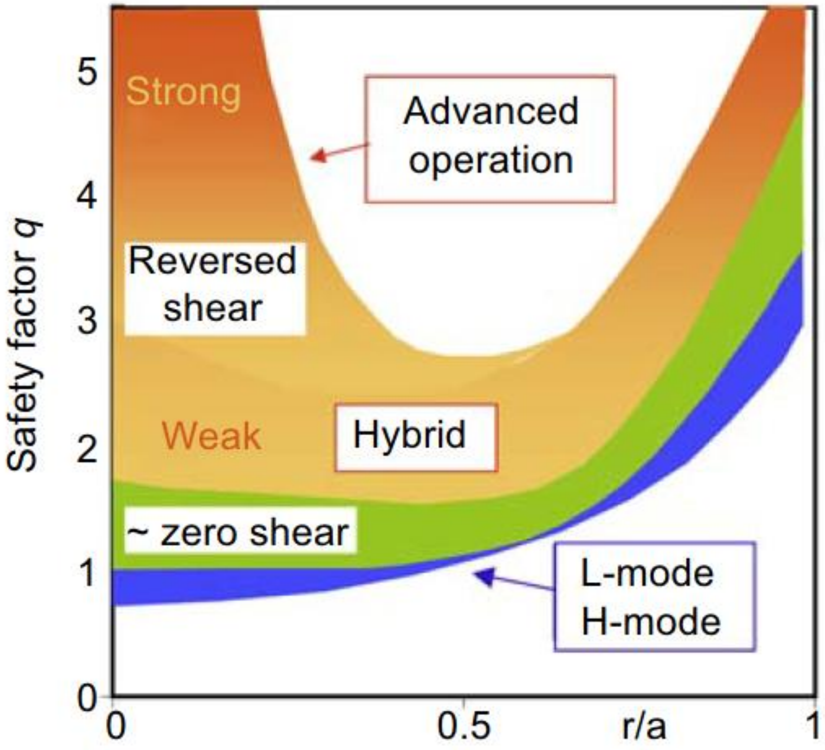
Clue: subtle changes in the current profile have large impact on performance



- robust operation at 4 < q < 5
- confinement (H_{98}) and stability (β_N) improved w.r.t standard H-mode
- q(0) kept above 1 by MHD dynamo (i.e. no sawteeth)

What is a ‚tokamak operation scenario'?





Tokamak operation scenario best characterised by *shape* of *p(r), j(r)*

- pressure profile composed of kinetic profiles $p(r) = n_e(r)\, T_e(r) + n_i(r)\, T_i(r)$
- current profile usually parametrised by safety factor q(r)

Outline

Access to a tokamak operational scenario is a non-trivial task

Actuators (H&CD systems, fueling systems, PF coil system), have limited efficiency:

- external H&CD power should be low in a reactor ($Q = P_{fus}/P_{ext} > 30$), $P_\alpha \sim (nT)^2$ dominates
- external CD has low efficiency (usually less than 0.1 A per W)
- PF coils for shaping and position control need to be far from plasma in a reactor

Equally important, the plasma is a nonlinear dynamic system in which *p(r)* and *j(r)* interact

- depending on the time history, plasma state can be different for the same actuator values
- aim for a plasma scenario that is in a robust (nonlinear) equilibrium state
- the trajectory towards the equilibrium matters and is subject to optimization!

Access to a tokamak operational scenario is a non-trivial task

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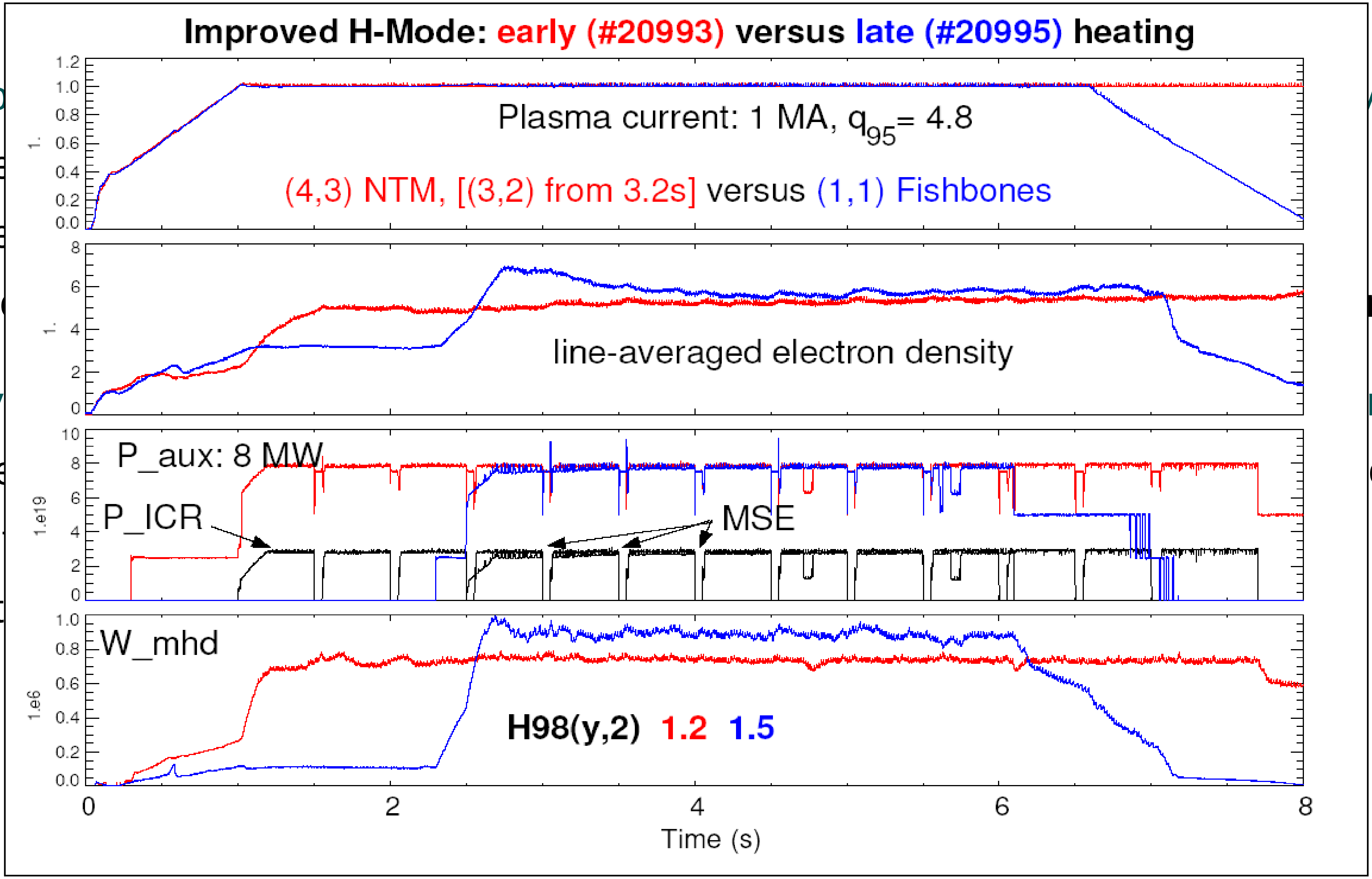
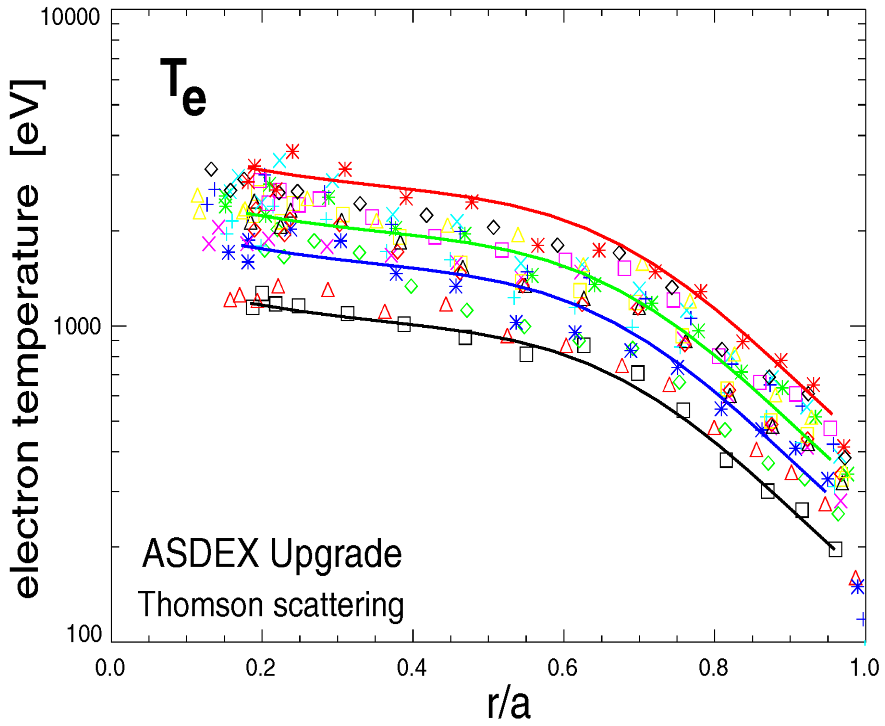
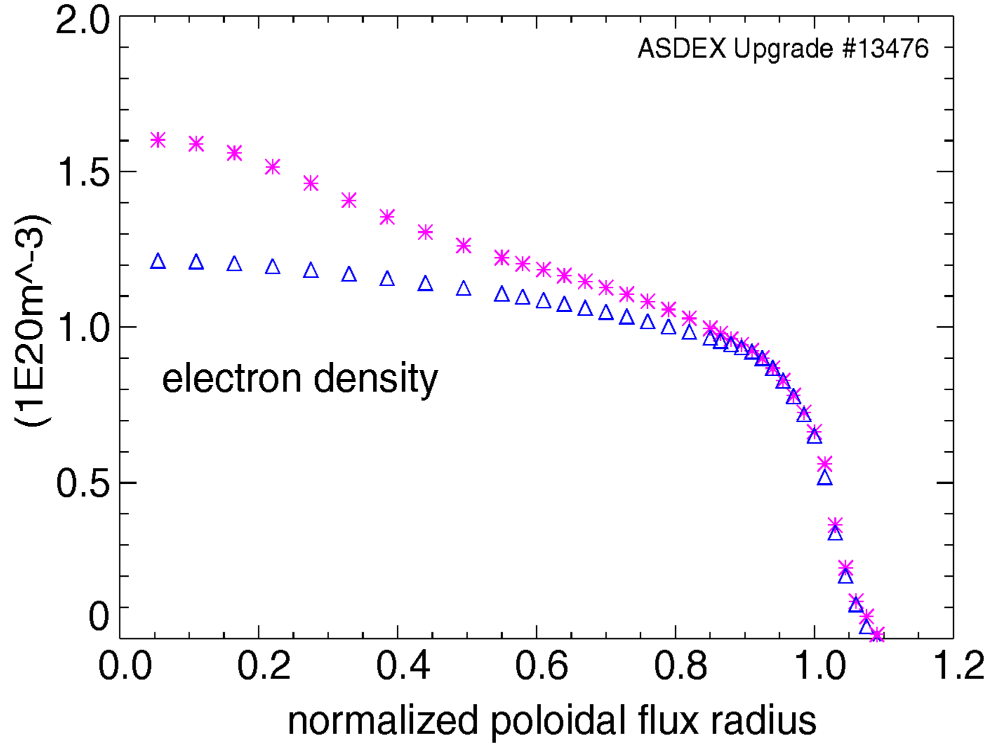


Control of the kinetic profiles *T(r)* and *n(r)*



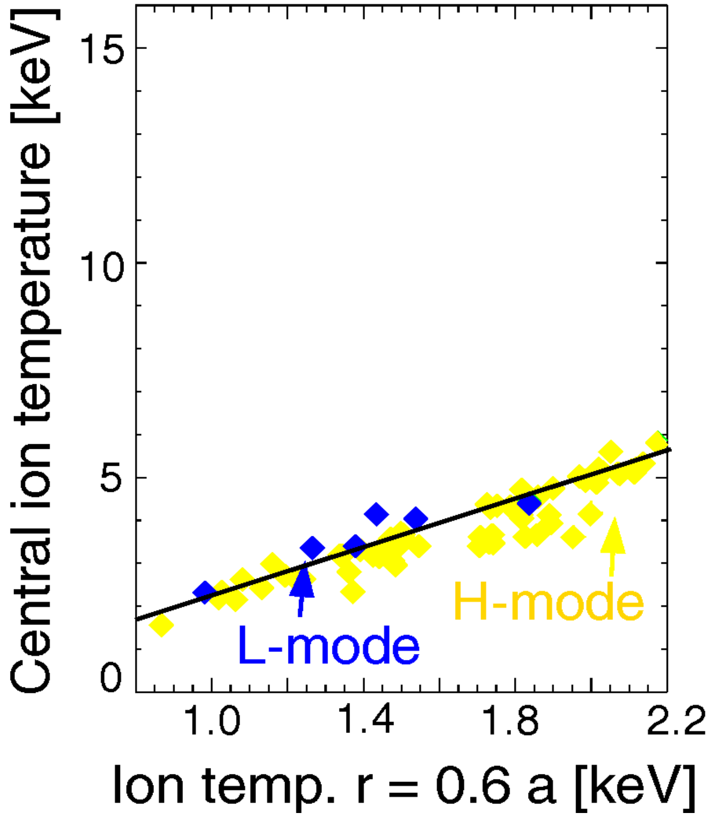


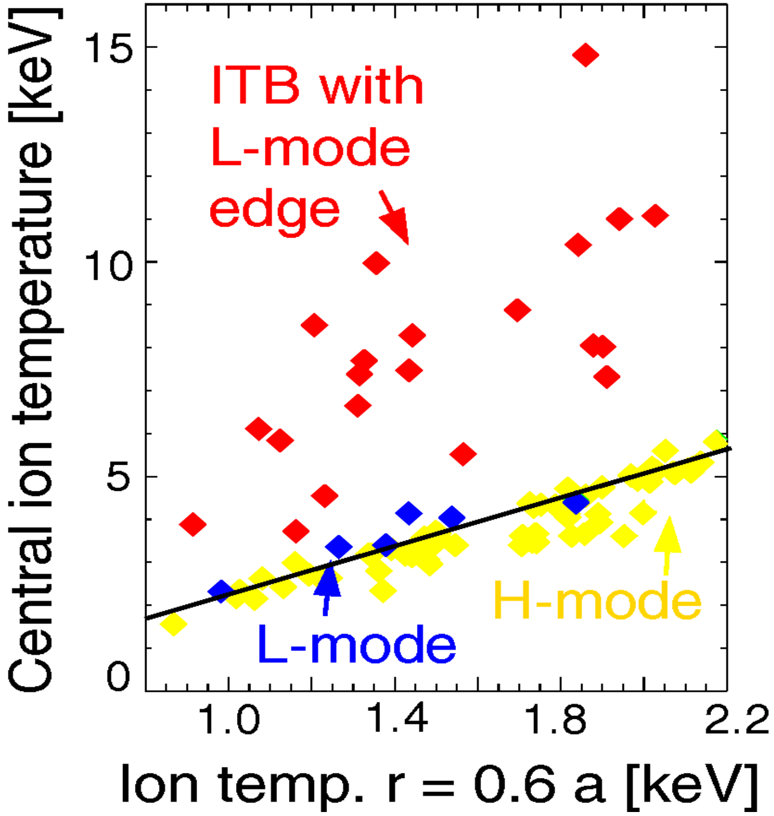
Pressure profile determined by heating / fueling profile and radial transport coefficients

- turbulent heat transport leads to ‘stiff’ temperature profiles (critical gradient length $\nabla T/T$)
- density profiles not stiff due to existence of a (collisionality dependent) ‘inward pinch’

⇒ Controlling transport is crucial to achieving the desired scenario

Control of the kinetic profiles *T(r)* and *n(r)*

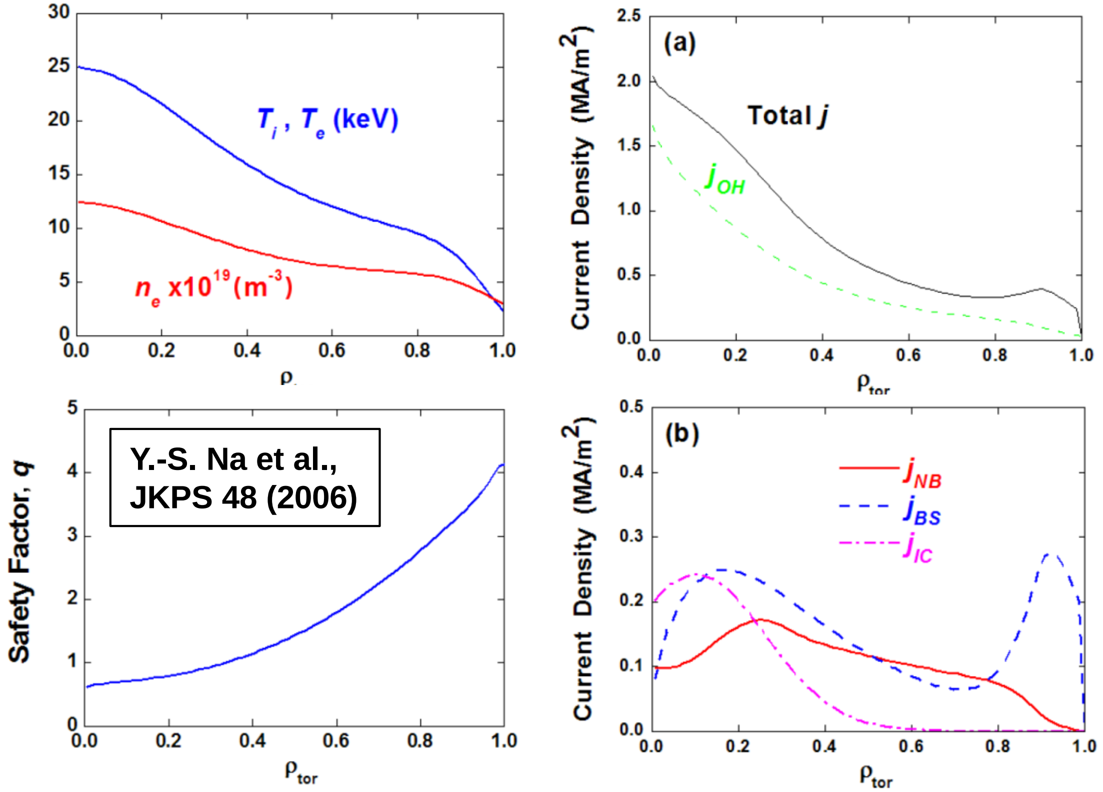


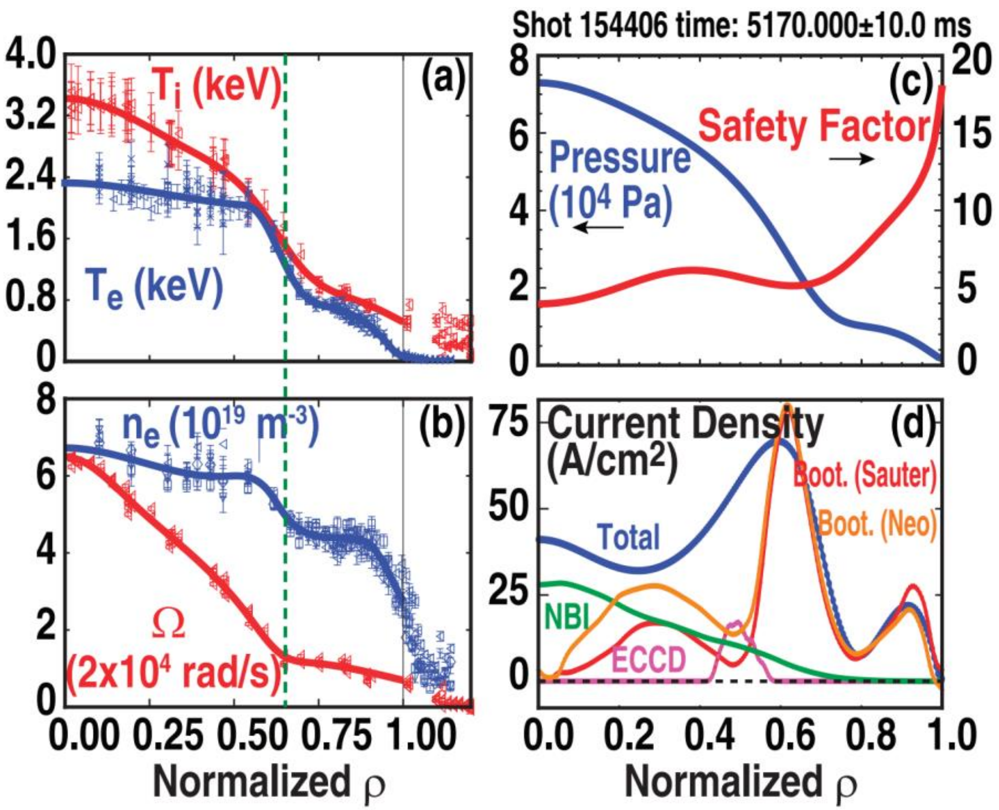


Stiffness can be overcome locally by sheared rotation

- suppressed turbulence allows edge transport barrier (H-mode) / internal transport barrier (ITB)

Control of the current profile linked to kinetic profile(s)





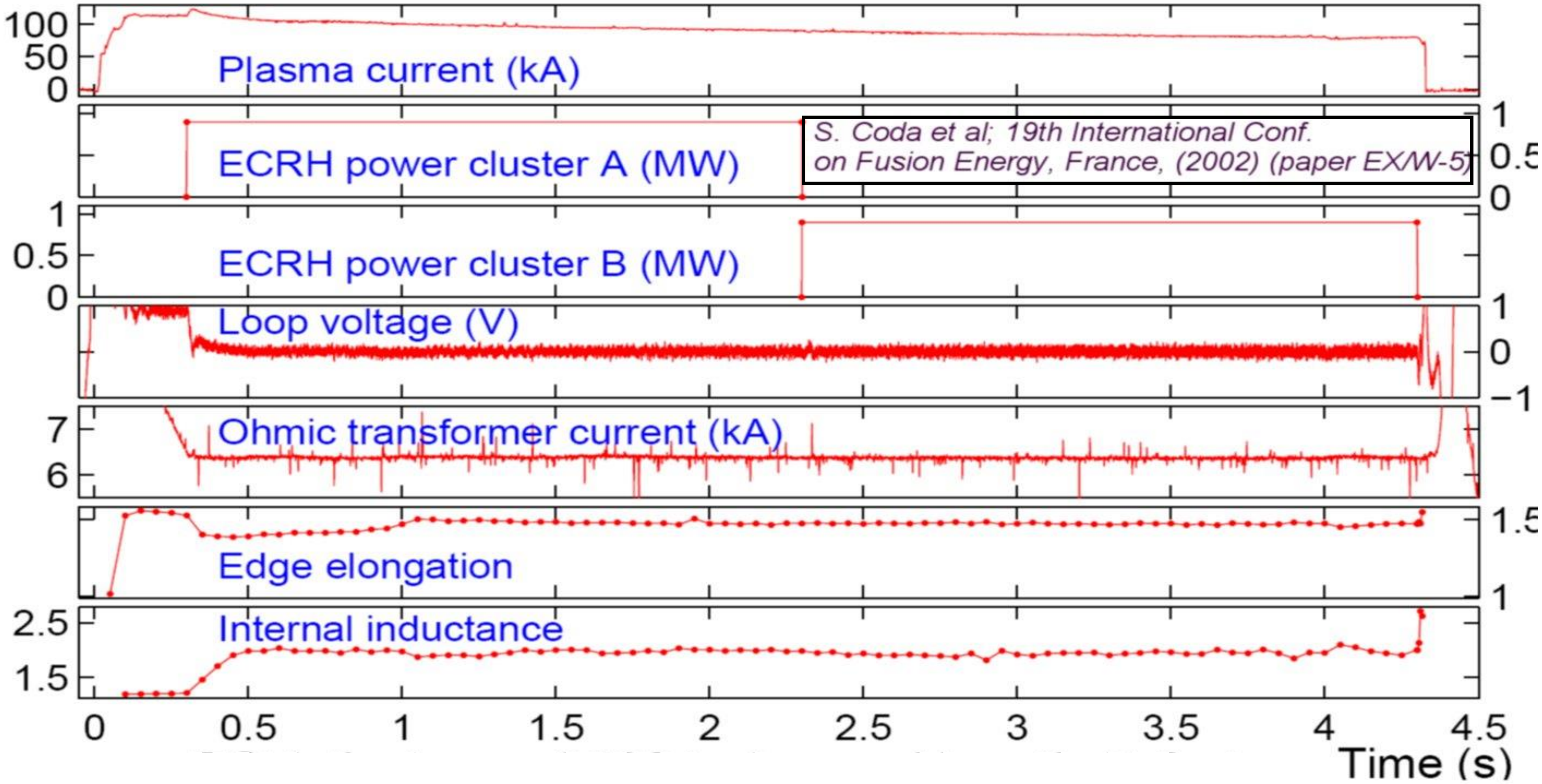
conventional / hybrid

advanced / elevated q

ohmic current coupled to temperature profile j(r) ~ $T(r)^{3/2} \rightarrow$ inductive component always peaked

bootstrap current linked to pressure and current profiles: $j_{bs} \sim (r/R)^{1/2}\ \nabla p/B_{pol}$

There is hope that stable stationary non-inductive solutions exist…



…but it is a challenging task to produce them at plasma performance sufficient for a reactor

Outline

1) Introduction (just given)

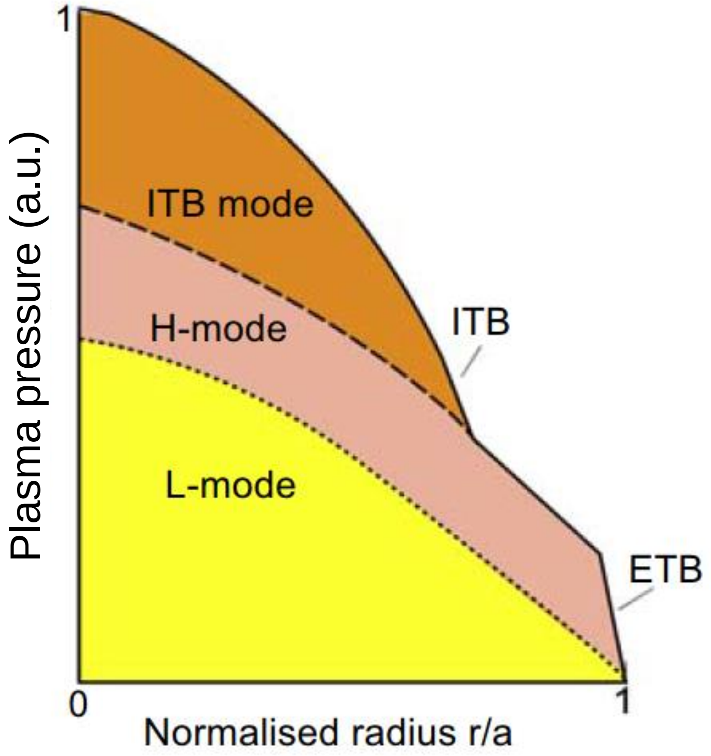
2) Optimisation strategies for tokamak plasmas

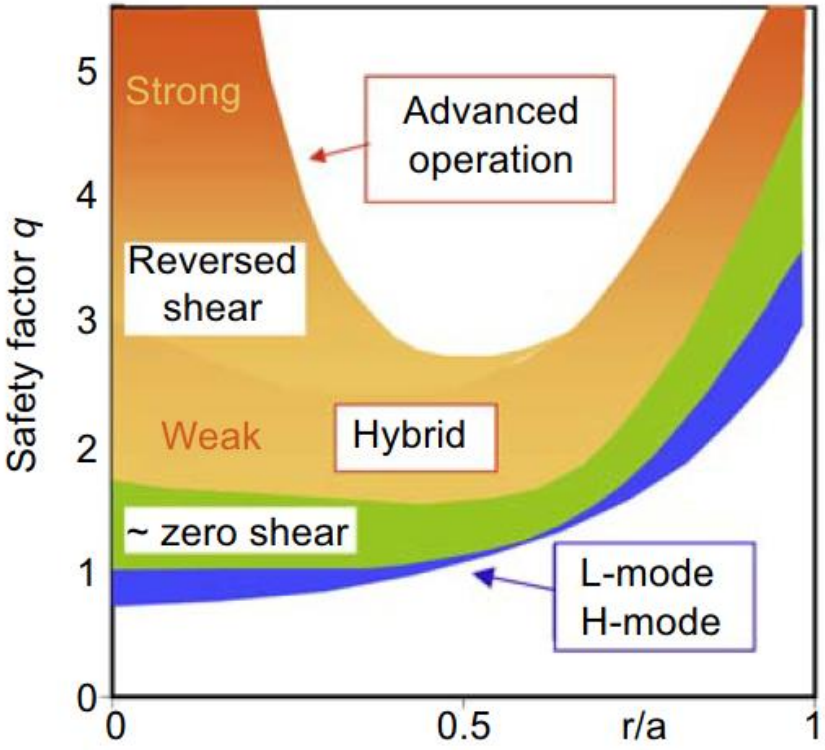
3) Scenarios characterised by j(r) and p(r)

4) Scenario access

5) Summary

Summary





Tokamak scenarios are characterized by a combination of pressure and current profile

- accessing and maintaining scenarios can be challenging due to the nonlinear plasma physics
- control strategies have to be developed to robustly operate the desired scenarrio