ASDEX Upgrade

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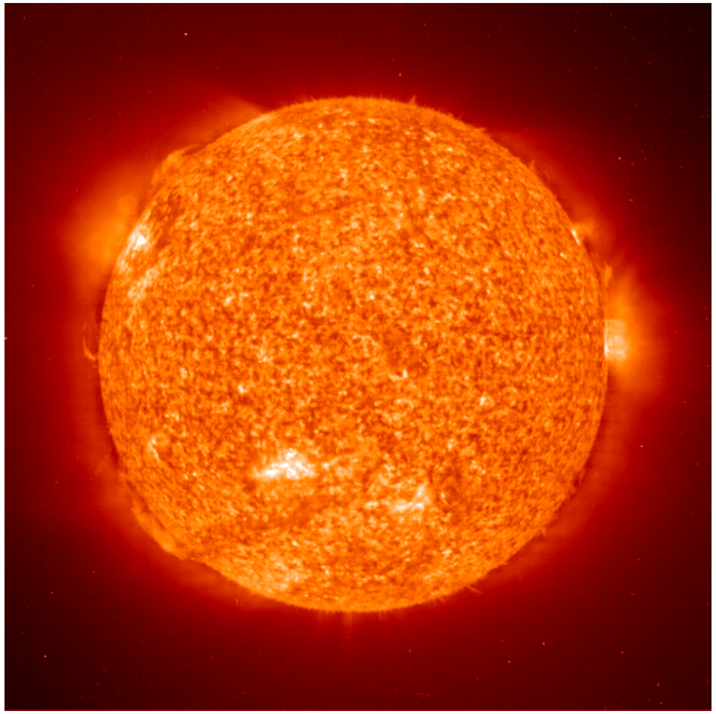
IPP

Tokamak Operational Scenarios



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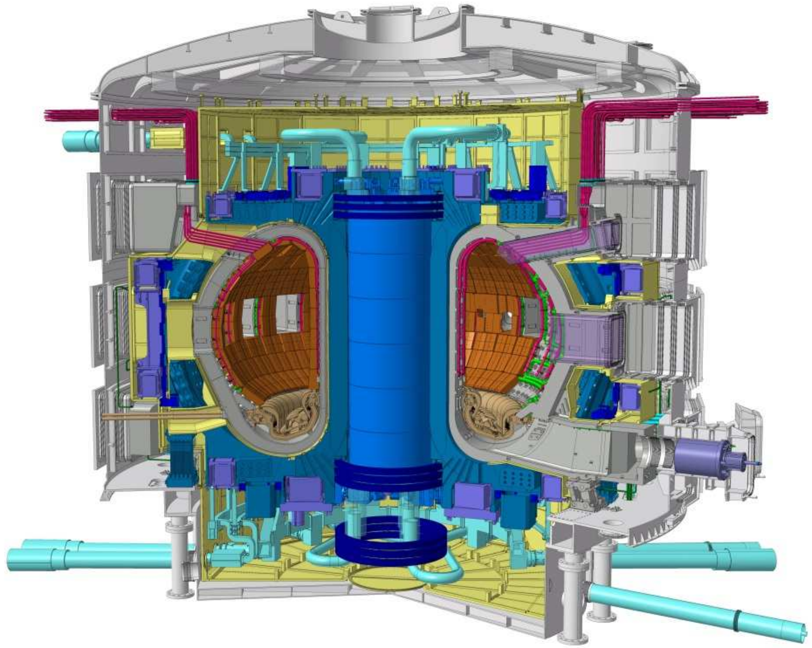


DPG Advanced Physics School
‚The Physics of ITER‘
Bad Honnef, 25.09.2014

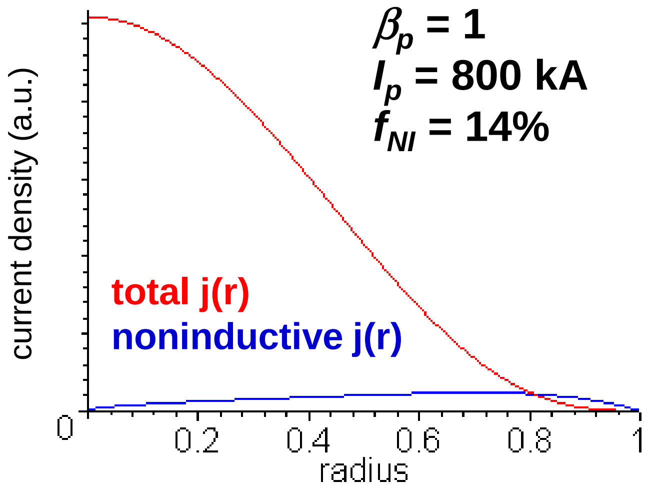
- What is a ‘tokamak (operational) scenario’?
- Optimisation strategies for tokamak scenarios
- Conventional scenarios
- Advanced scenarios
- Summary and conclusions

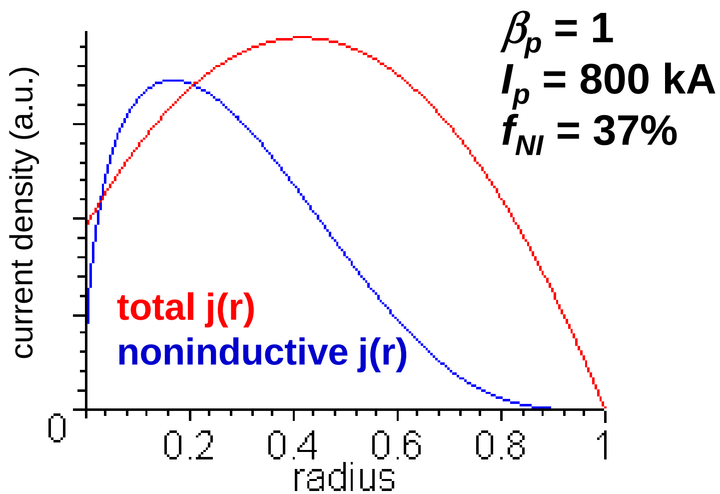
What is a ‚tokamak scenario‘?

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

Plasma discharge characterised by

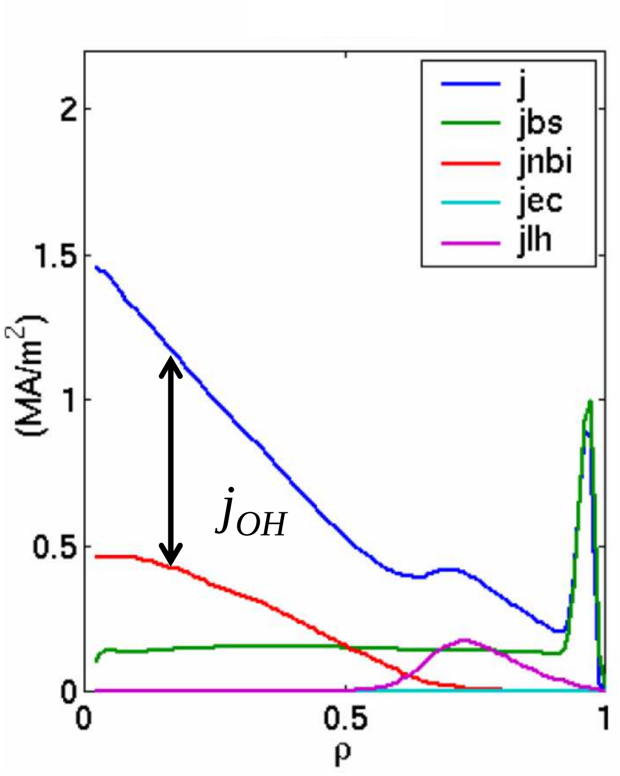
- external control parameters: B_t, R_0, a, κ; δ, P_{heat}, Φ_D…
- integral plasma parameters: $\beta = 2\mu_0 \langle p \rangle / B^2$, $I_p = 2\pi \int j(r)\, r\, dr$…
- plasma profiles: pressure $p(r) = n(r)*T(r)$, current density $j(r)$

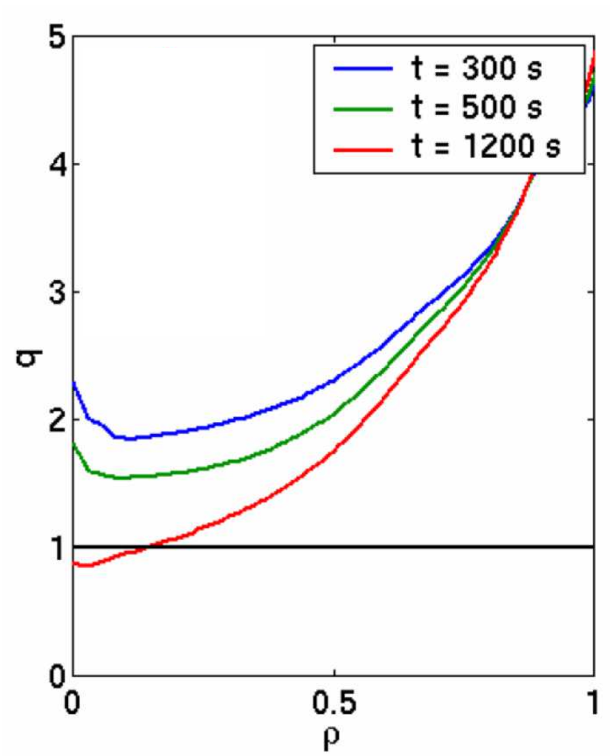




$\rightarrow$ operational scenario best characterised by *shape* of *p(r), j(r)*

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





safety factor:

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

ITER Q=10 simulation

- ohmic current coupled to temperature profile via $\sigma \sim T^{3/2}$
  $\rightarrow$ inductive current profiles always peaked, *q*-profiles monotonic
- external heating systems drive current, but with limited efficiency (typically less than 0.1 A per 1 W under relevant conditions)…
- pressure gradient drives toroidal ‘bootstrap’ current: $j_{bs} \sim (r/R)^{1/2}\ \nabla p/B_{pol}$

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
- under reactor-like conditions, dominant α-heating ~ $(nT)^2$
- gas puff is peripheral source of particles, pellets further inside

Shape of profile will strongly depend on (turbulent) heat conductivity and particle diffusivity (i.e. be 'self-organised')

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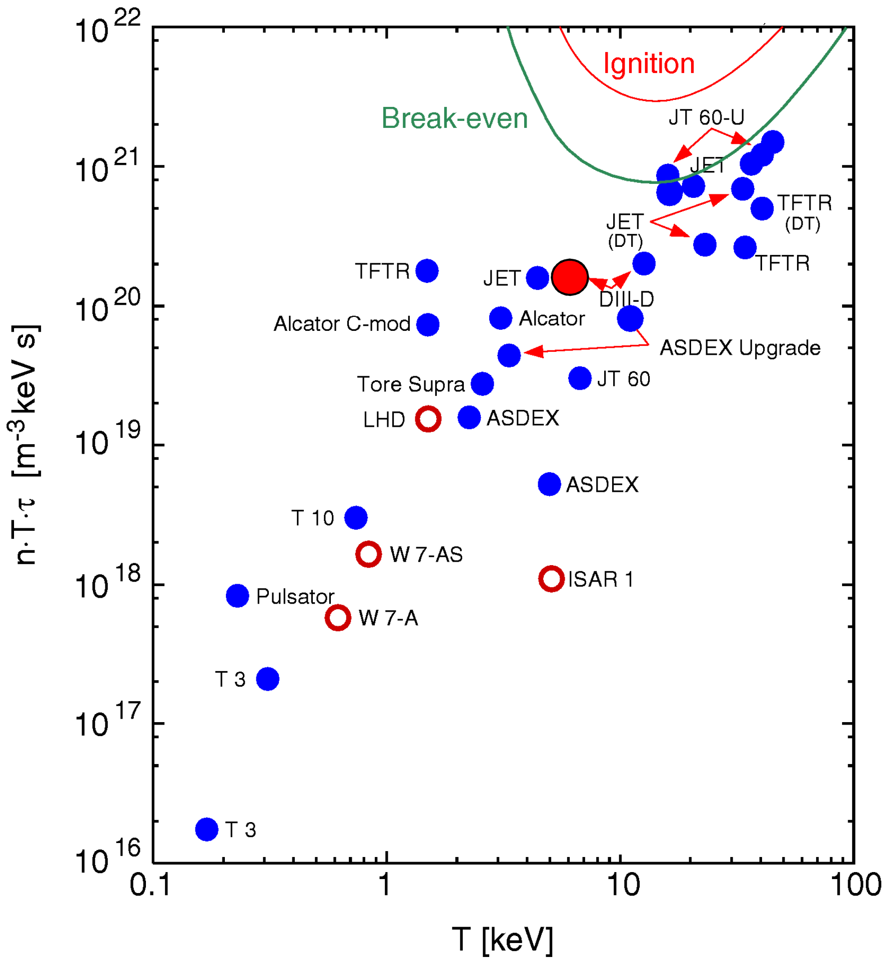
Figure of merit for fusion performance $nT\tau$

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)

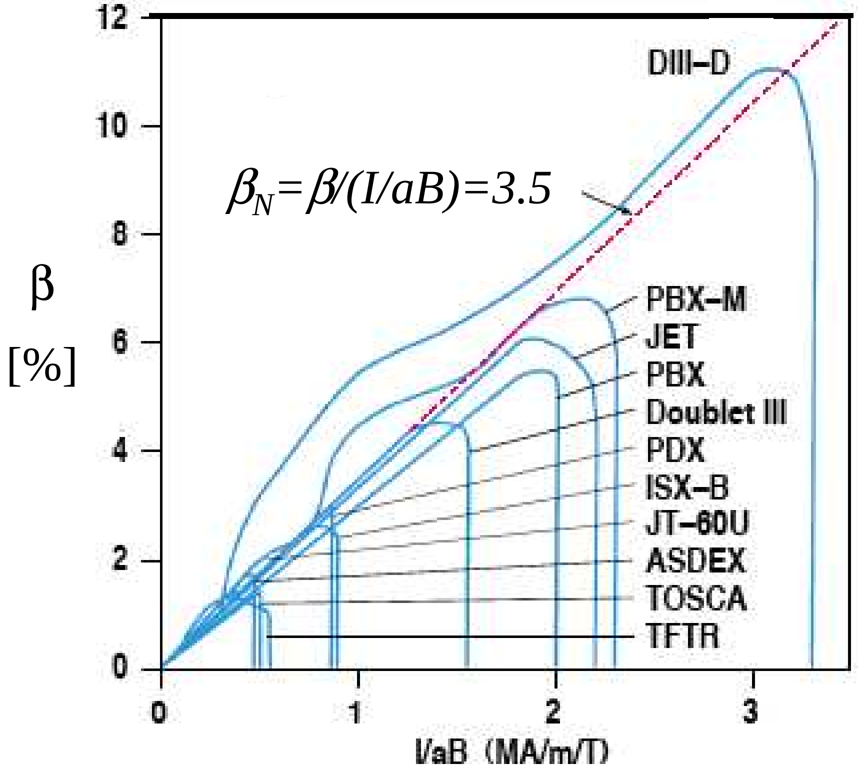
Optimisation of $nT\tau_E$: ideal pressure limit

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 time scale ~ 10 μs, only limited by inertia)

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



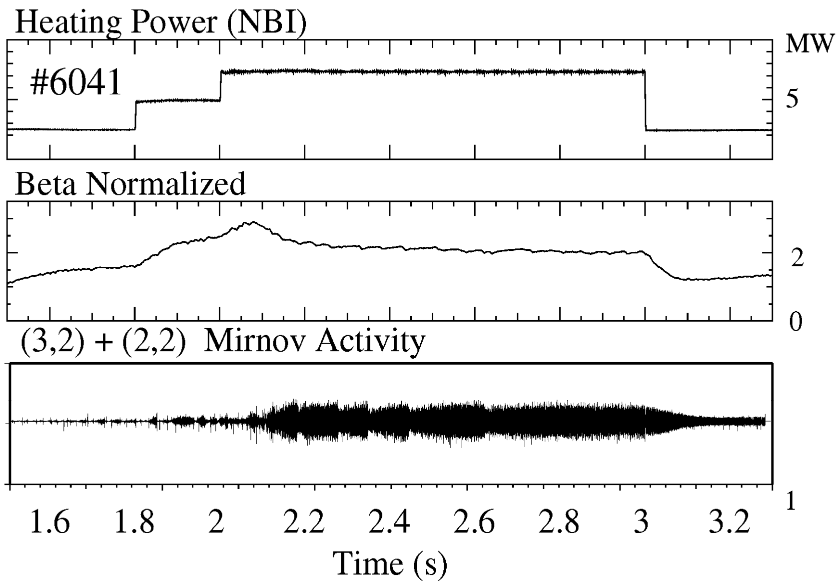
Optimisation of $nT\tau_E$: resistive pressure limit

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

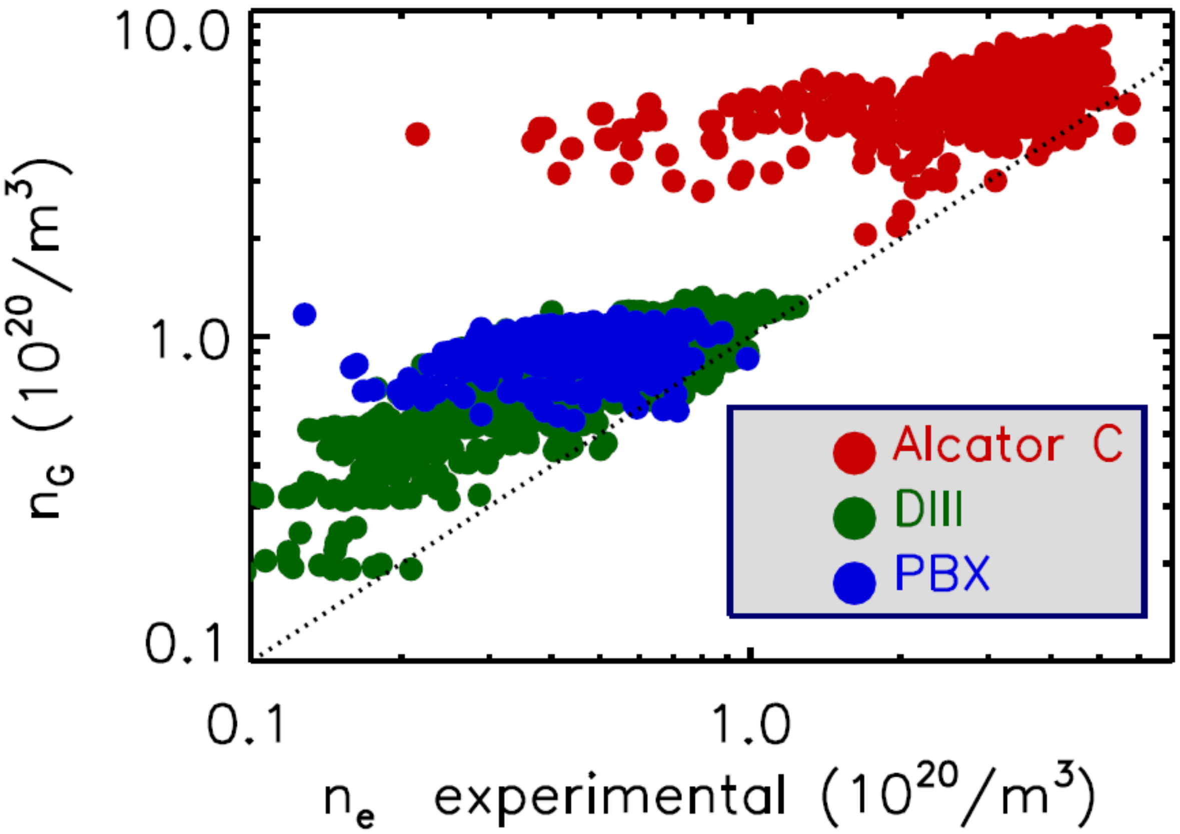
This quantity is limited by magneto-hydrodynamic (MHD) instabilities

'Resistive' MHD limit (on local current redistribution time scale ~ 100 ms)

Neoclassical Tearing Mode (NTM)



Optimisation of $nT\tau_E$: density limit

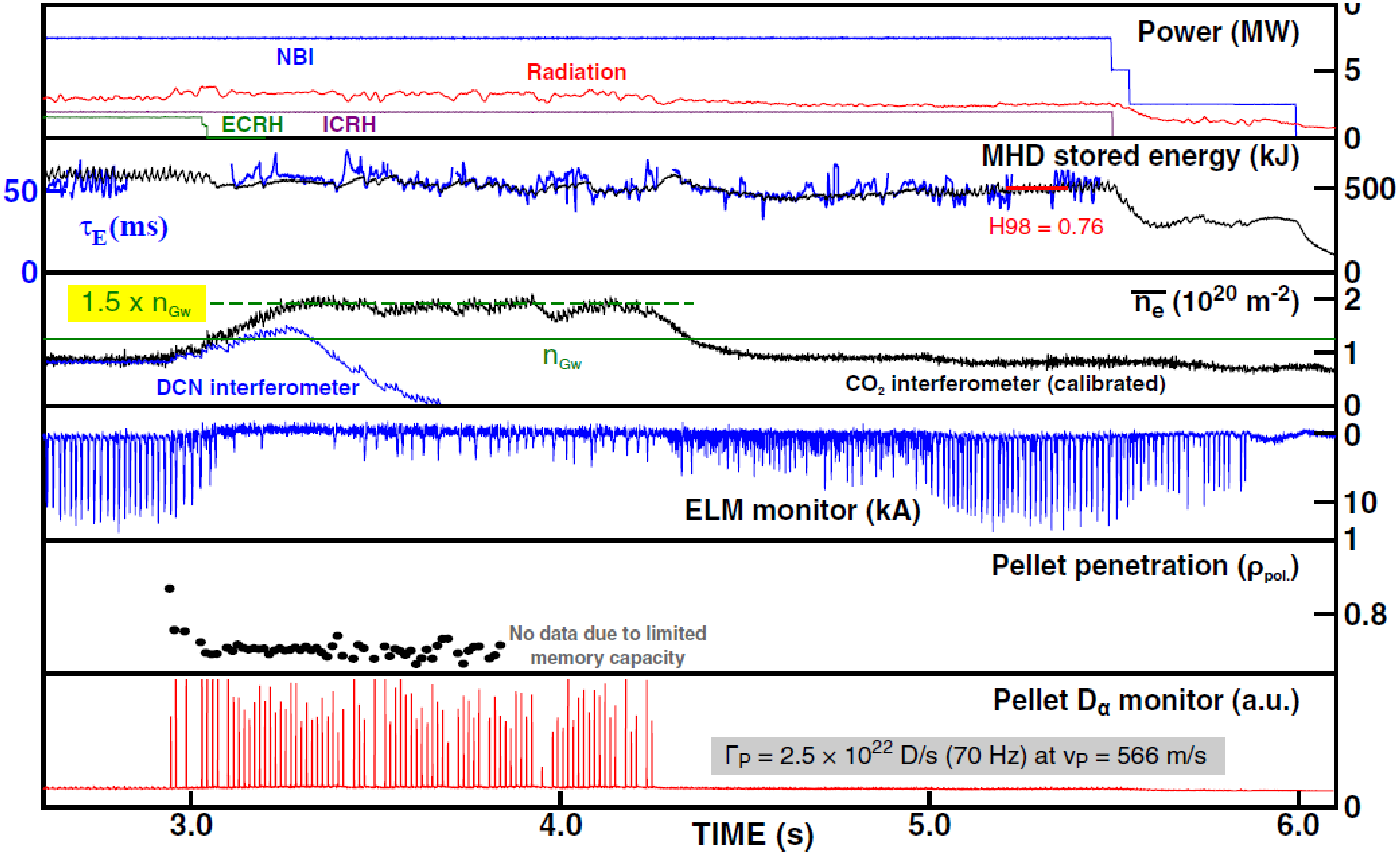


$n_G = I_p/(\pi a^2)$

Empirical ‚Greenwald-limit' describes well maximum density

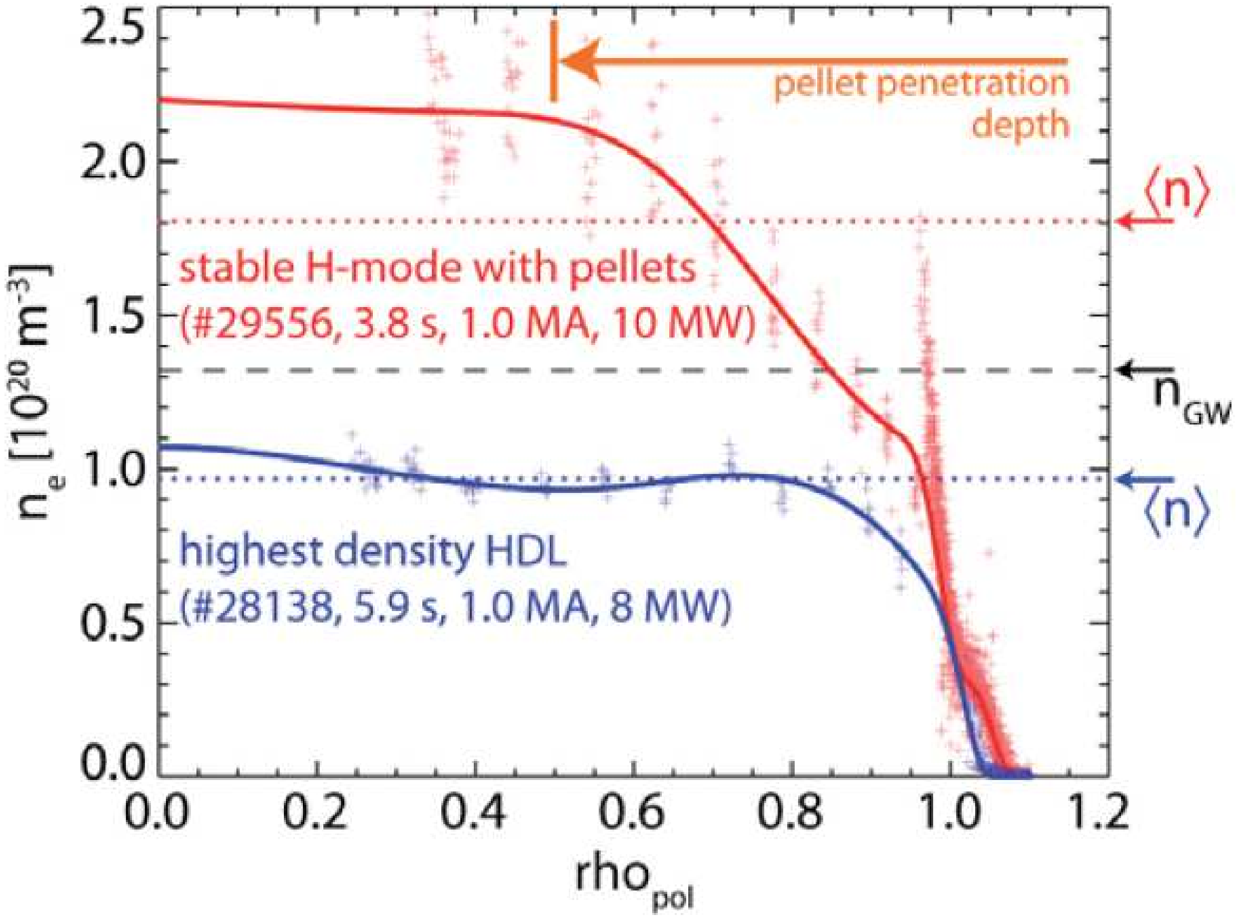
- seems to be linked to a change in edge transport at $n \sim n_G$
- can be overcome if density profile shape is varied (peaked)

Operation at $n/n_{GW} > 1$



Stable operation at n/n_{GW} = 1.5 using pellets

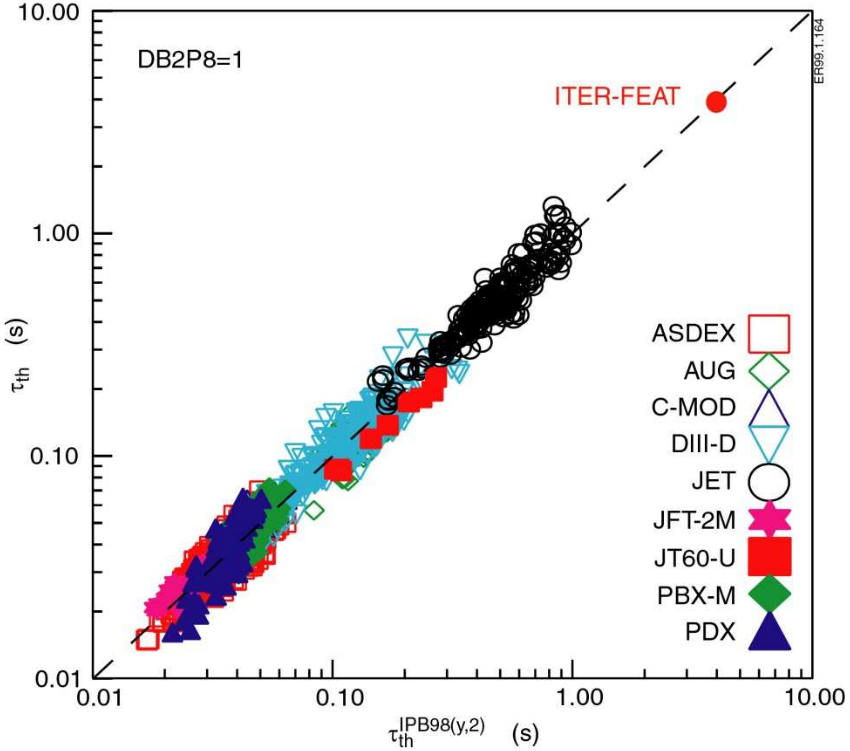
Operation at $n/n_{GW} > 1$



Edge density stays below n_{GW} in all cases (up to $n_{e0} = 4 \times n_{GW}$!)

- H-mode density limit = edge density limit

Optimisation of $nT\tau_E$: confinement scaling



Empirical ITER 98(p,y) scaling:

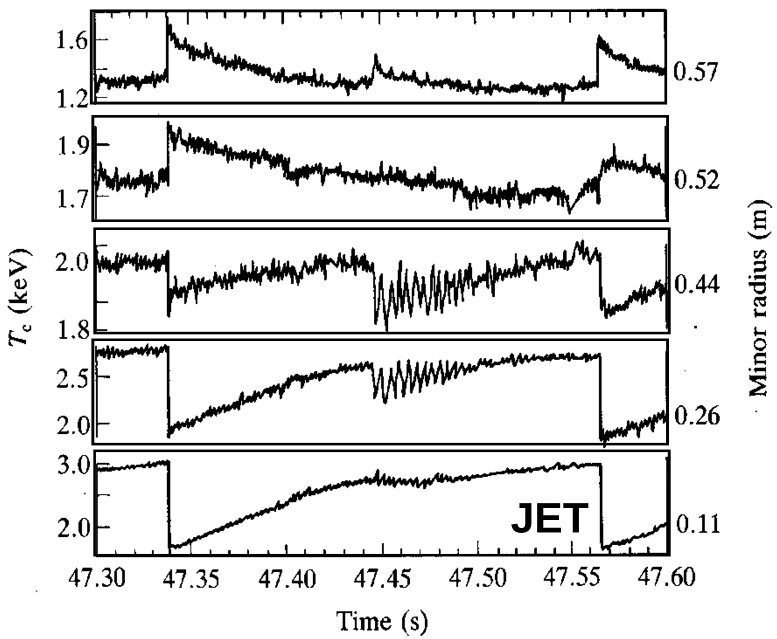
$$\tau_E \sim H\, I_p^{0.93}\, P_{heat}^{-0.63}\, B_t^{0.15} \ldots$$

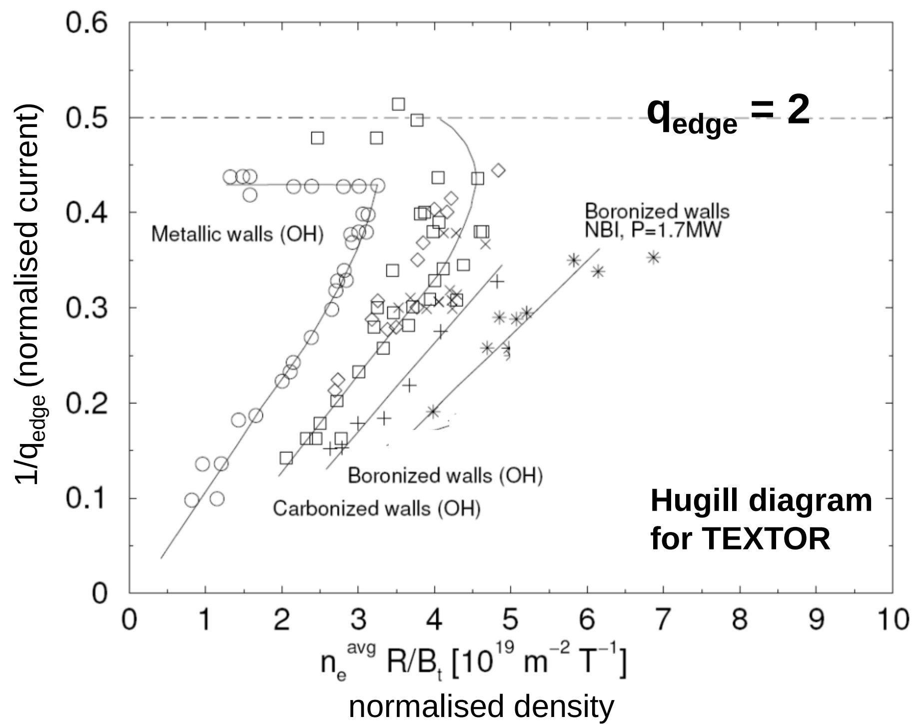
Empirical confinement scalings show linear increase of τ_E with I_p

- 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 is limited by current gradient driven MHD instabilities

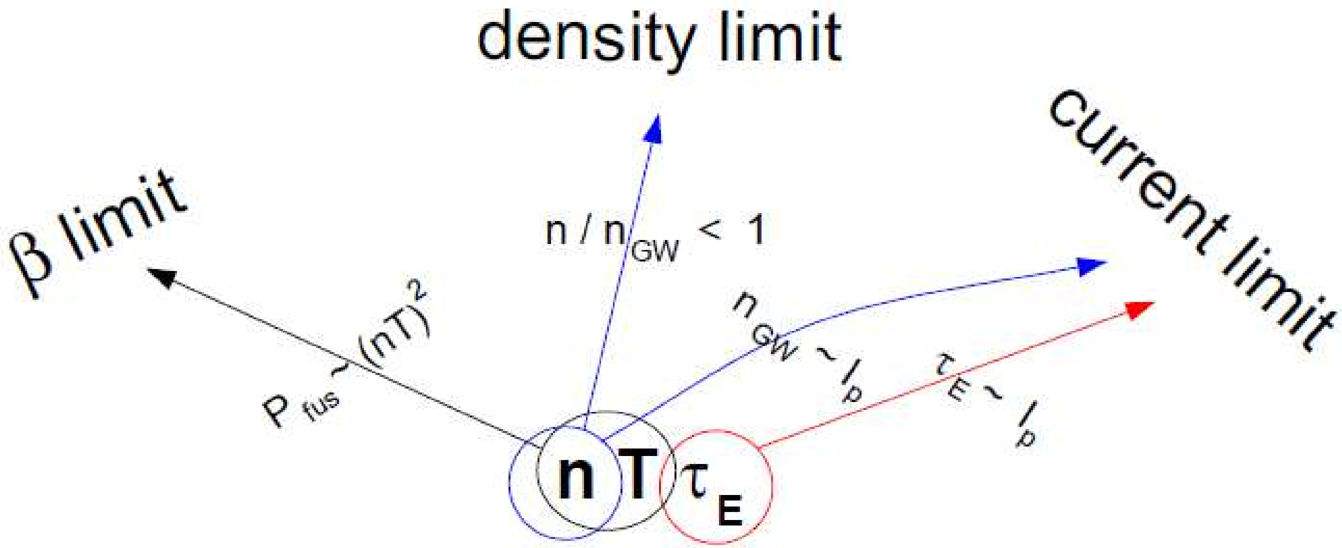




Limit to safety factor $q \sim (r/R)\ (B_{tor}/B_{pol})$

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

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, high I_p is not desirable:

Tokamak operation without transformer: current 100% noninductive

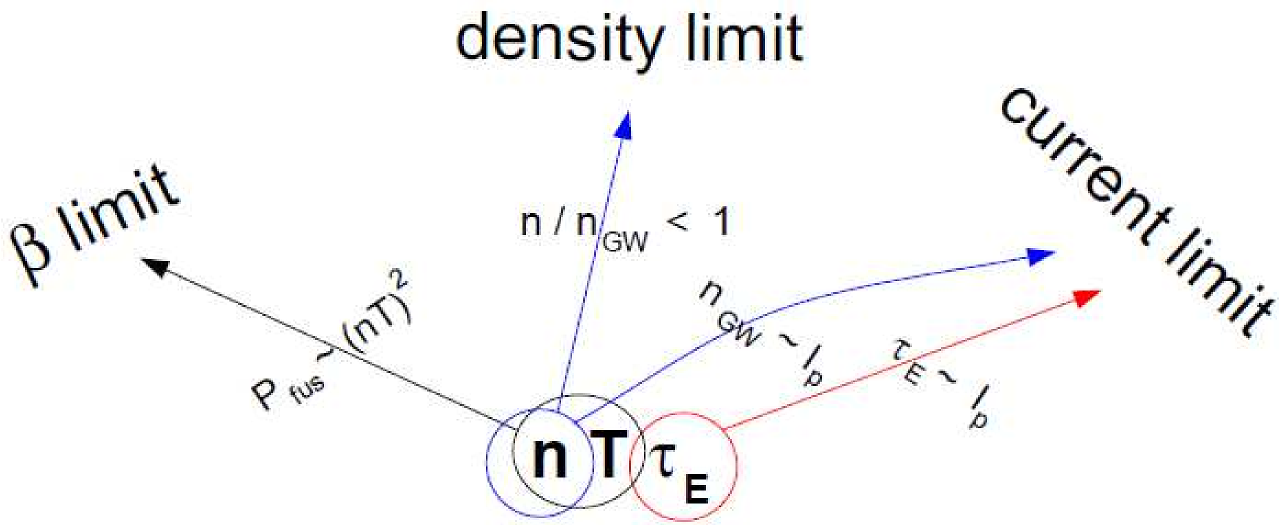
- external CD has low efficiency (remember less than 0.1 A per W)
- internal bootstrap current high for high $j_{bs} \sim (r/R)^{1/2}\ \nabla p/B_{pol}$

  $\rightarrow f_{NI} \sim I_{bs}/I_p \sim p/B_{pol}^2 \sim \beta_{pol}$

'Advanced' scenarios, which aim at steady state, need high β, low I_p, have to make up for loss in τ_E (e.g. through transport barriers)

Without the 'steady state' boundary condition, a tokamak scenario is called 'conventional'

Optimisation of $nT\tau_E$



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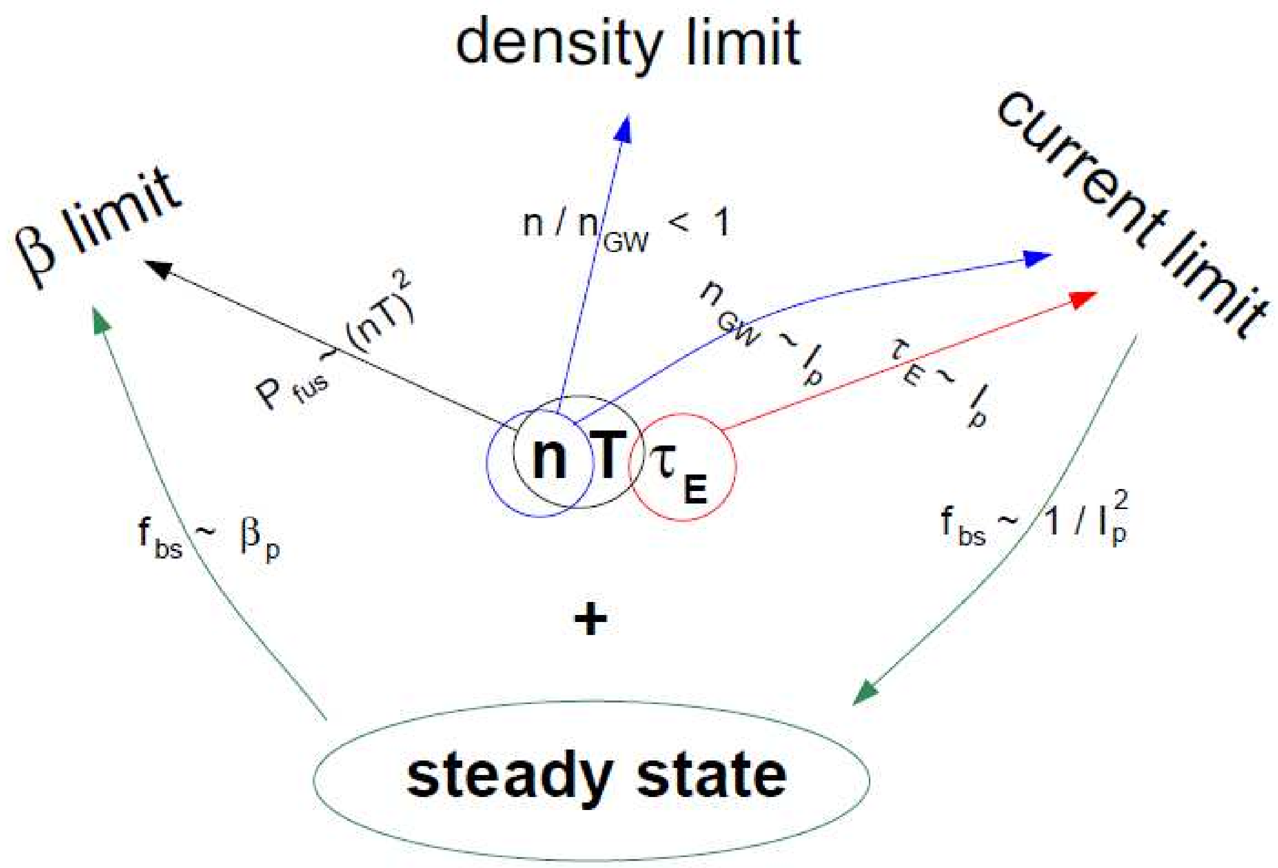
Optimisation of $nT\tau_E$



Including the steady state constraint emphasizes the β-limit

(and de-emphasizes current limit)

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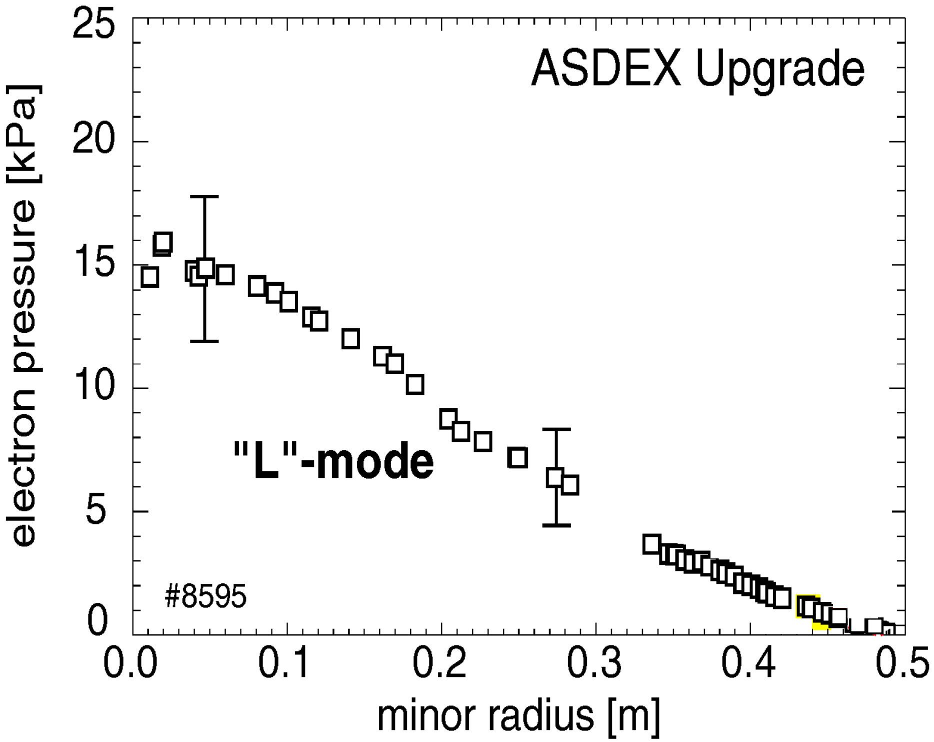
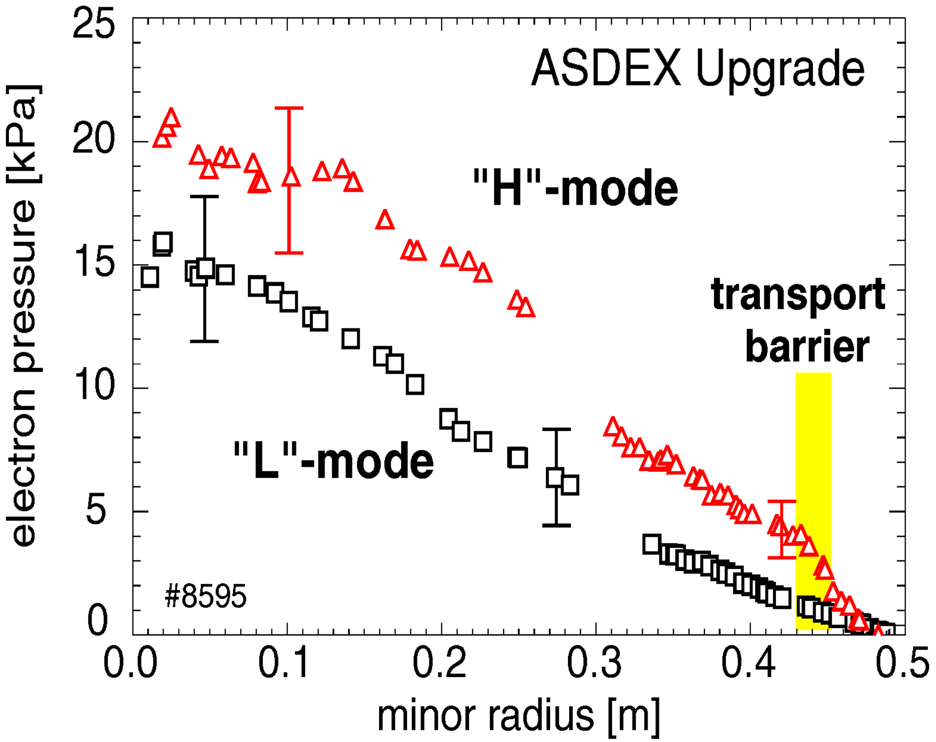
The (low confinement) L-mode scenario



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

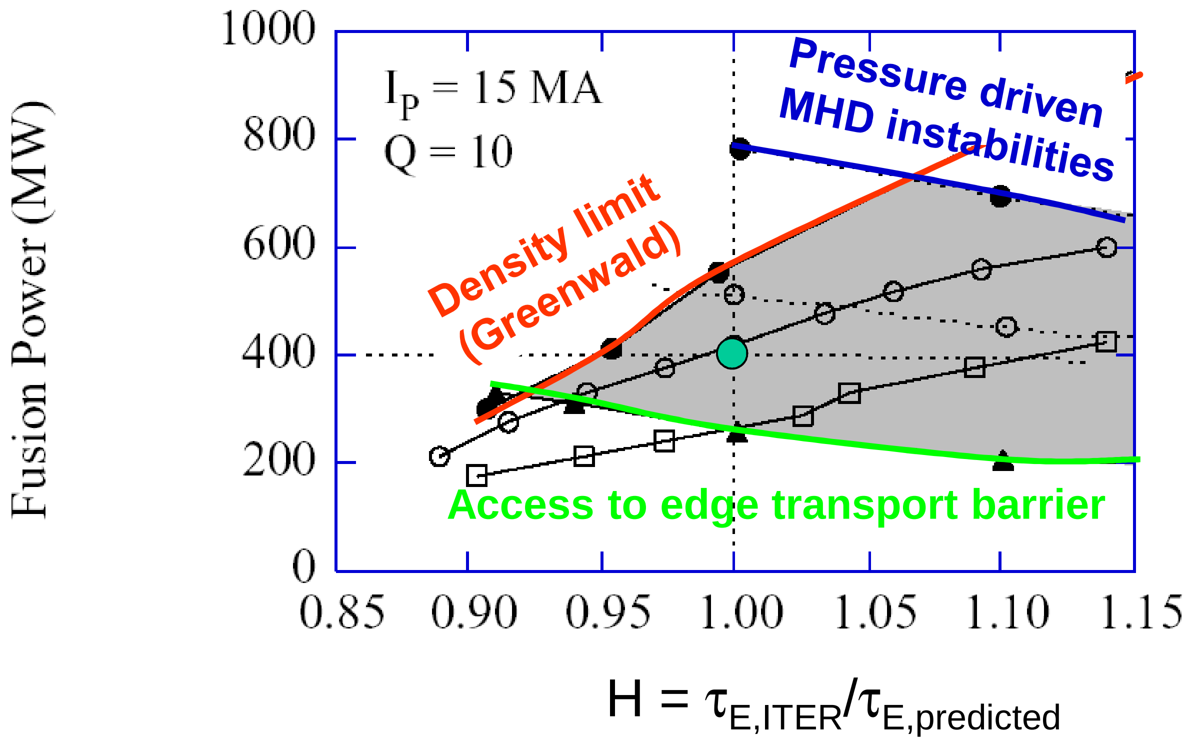
The (high confinement) H-mode scenario



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

H-mode is ITER standard scenario for Q=10…



The design point allows for…

- …achieving Q=10 with conservative assumptions
- …incorporation of ‚moderate surprises‘
- …achieving ignition ($Q \rightarrow \infty$) if surprises are positive

…but some open issues remain…

Need to minimise ELM impact on divertor

- reduce power flow to divertor by radiative edge cooling
- special variants of the scenario (Quiescent H-mode, type II ELMs)
- ELM mitigation – pellet pacing or Resonant Magnetic Perturbations

Need to tackle NTM problem

- NTM suppression by Electron Cyclotron Current Drive demonstrated, but have to demonstrate that this can be used as reliable tool

- What is a ‘tokamak (operational) scenario’?
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Advanced tokamak – the problem of steady state

Advanced scenarios aim at stationary (transformerless) operation

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

  $\rightarrow f_{NI} \sim I_{bs}/I_p \sim p/B_{pol}^2 \sim \beta_{pol}$

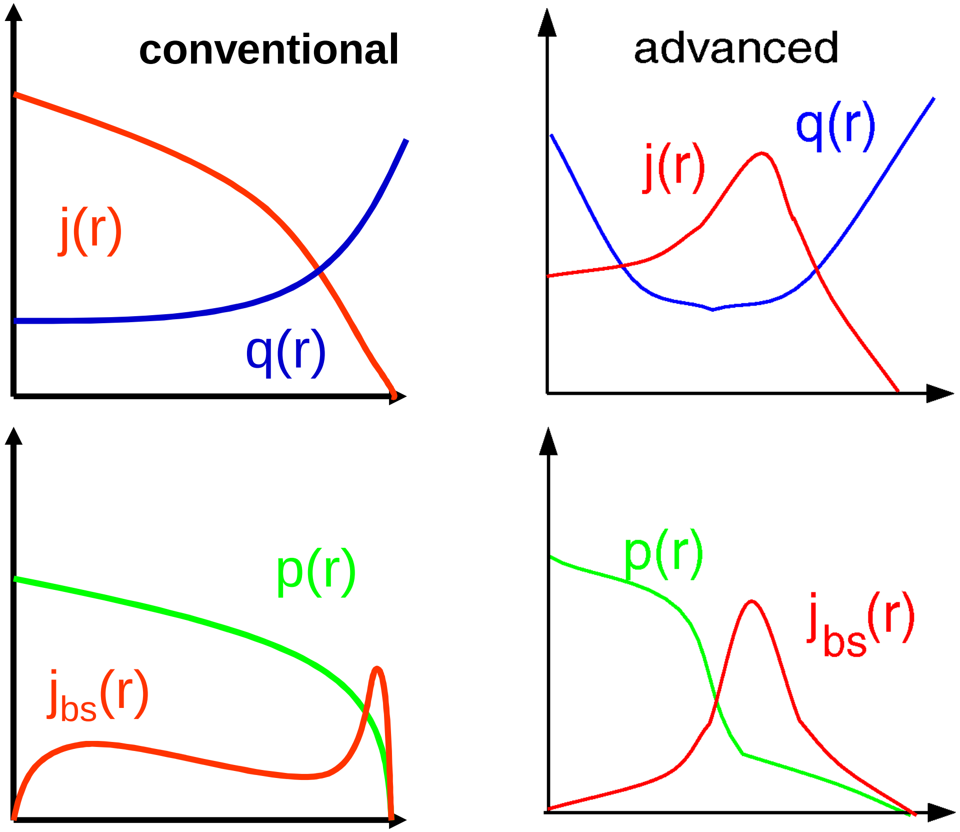
Recipe to obtain high bootstrap fraction:

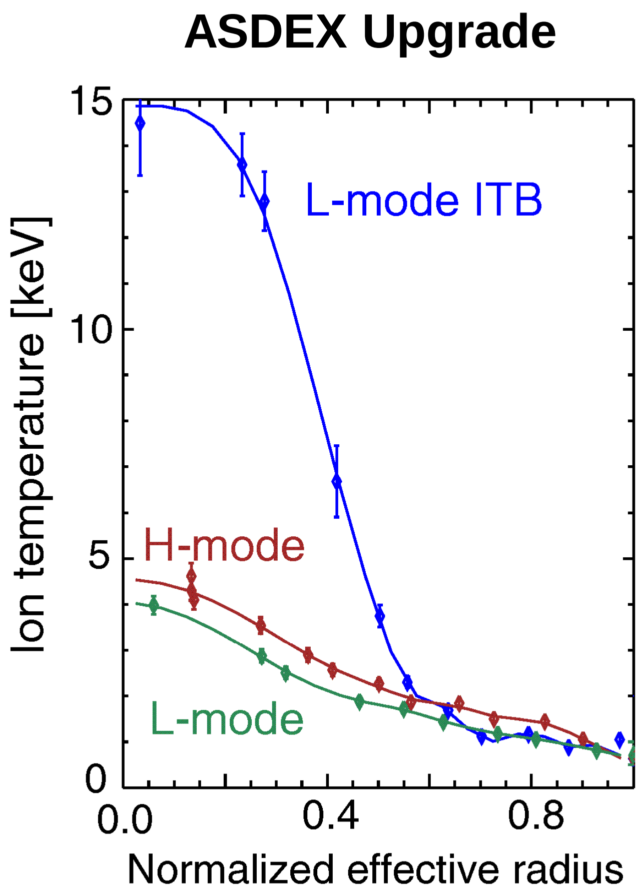
- low B_{pol}, i.e. high q – elevate or reverse q-profile ($q=(r/R)(B_{tor}/B_{pol})$)
- eliminates NTMs (reversed shear, no low resonant q-surfaces)
- high pressure where B_{pol} is low, i.e. peaked $p(r)$

Both recipes tend to make discharge ideal MHD (kink) unstable!

In addition, j_{bs} profile should be consistent with q-profile

Advanced tokamak – the problem of steady state





A self-consistent solution is theoretically possible

- reversing *q*-profile suppresses turbulence – internal transport barrier (ITB)
- large bootstrap current at mid-radius supports reversed *q*-profile

Problems of the Advanced tokamak scenario

Broad current profile leads to low kink stability (low β-limit):

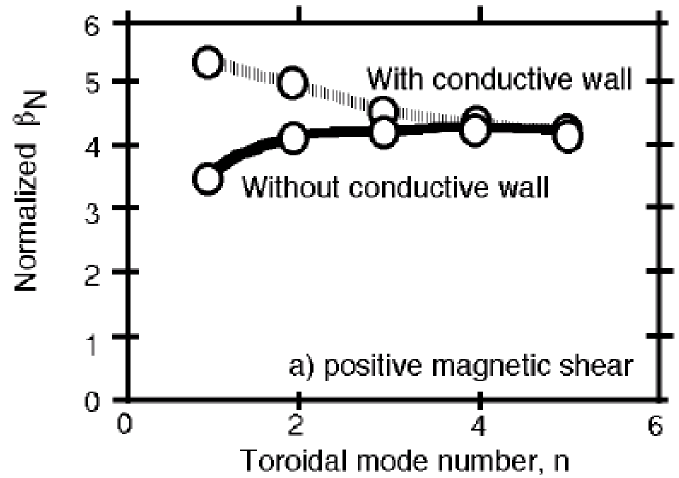
- can partly be cured by close conducting shell, but kink instability then grows on resistive time scale of wall (Resistive Wall Mode RWM)
- can be counteracted by helical coils, but this needs sophisticated feedback

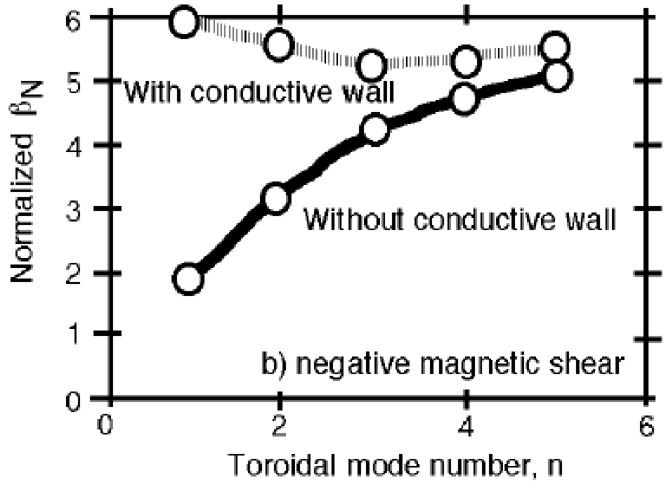
Position of ITB and minimum of *q*-profile must be well aligned

- needs active control of both *p(r)* and *j(r)* profiles – difficult with limited actuator set (and cross-coupling between the profiles)

Effect of a conducting shell on stability…

for conventional H-mode:
modest need and gain





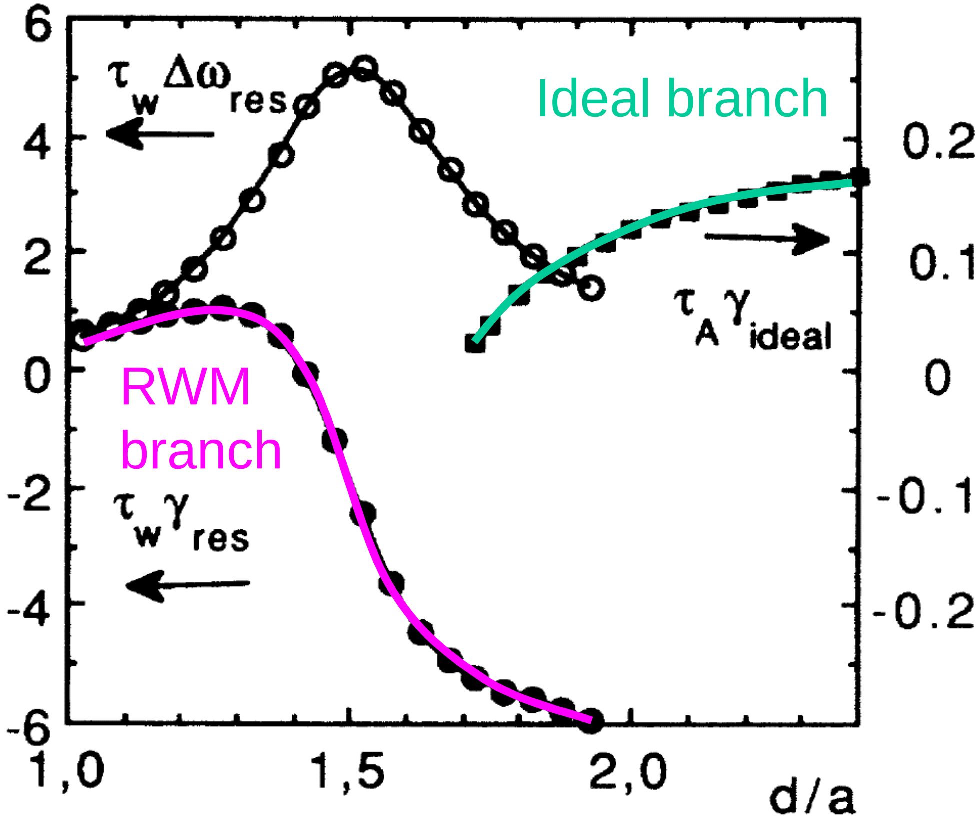
for advanced ITB scenarios:
strong need and significant gain

Close-by conducting wall can stabilise external kink instability

- usually not an issue fo conventional scenario, but advanced scenarios prone to external kink due to broad current/peaked pressure profile

…but no wall is ideally conducting!

A. Bondeson et al., PRL 94



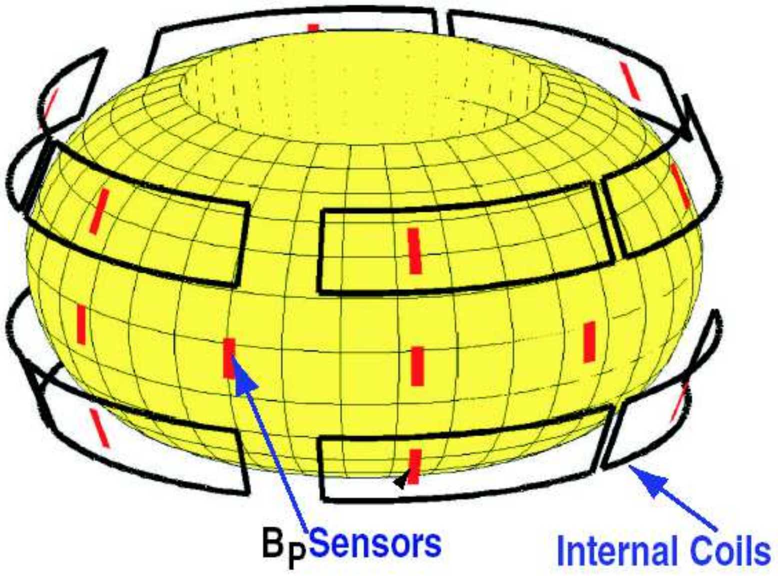
Ideal regime: Ideal kink stable if wall is close enough

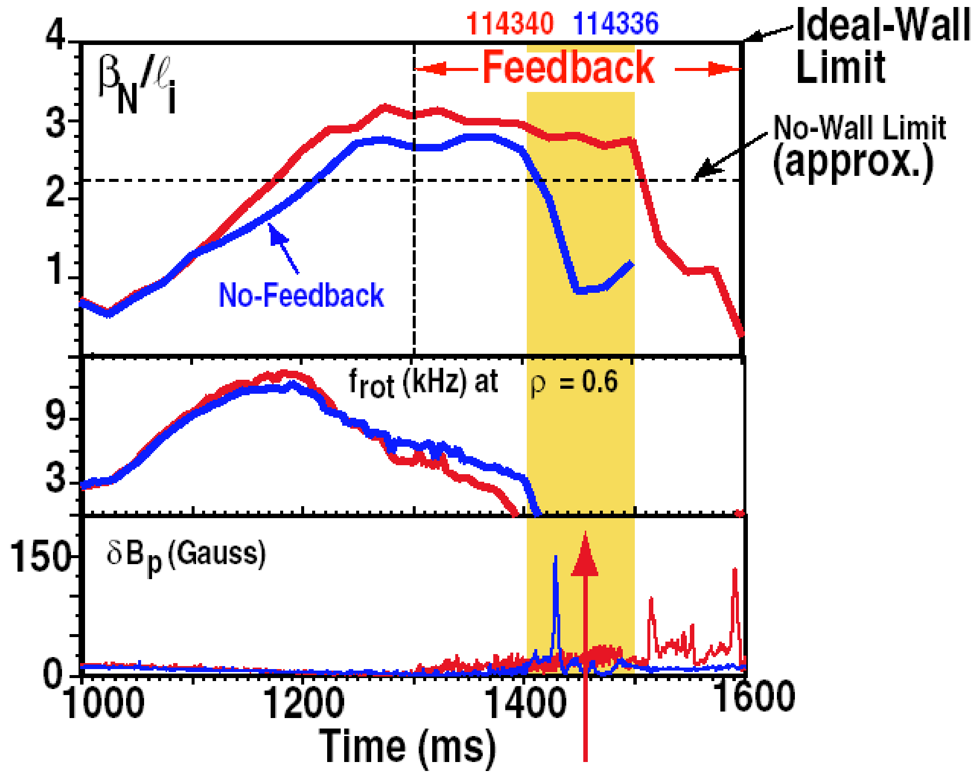
RWM regime: RWM is stable when slipping between mode and wall is large enough

When ideal kink is wall stabilised, RWM can grow on wall time scale

- rotation w.r.t. wall can stabilise the RWM if $\omega_{rot} >> 1/\tau_W$
- balance between wall drag and (rotating) plasma drag on mode

RWM control by Resonant Magnetic Perturbations

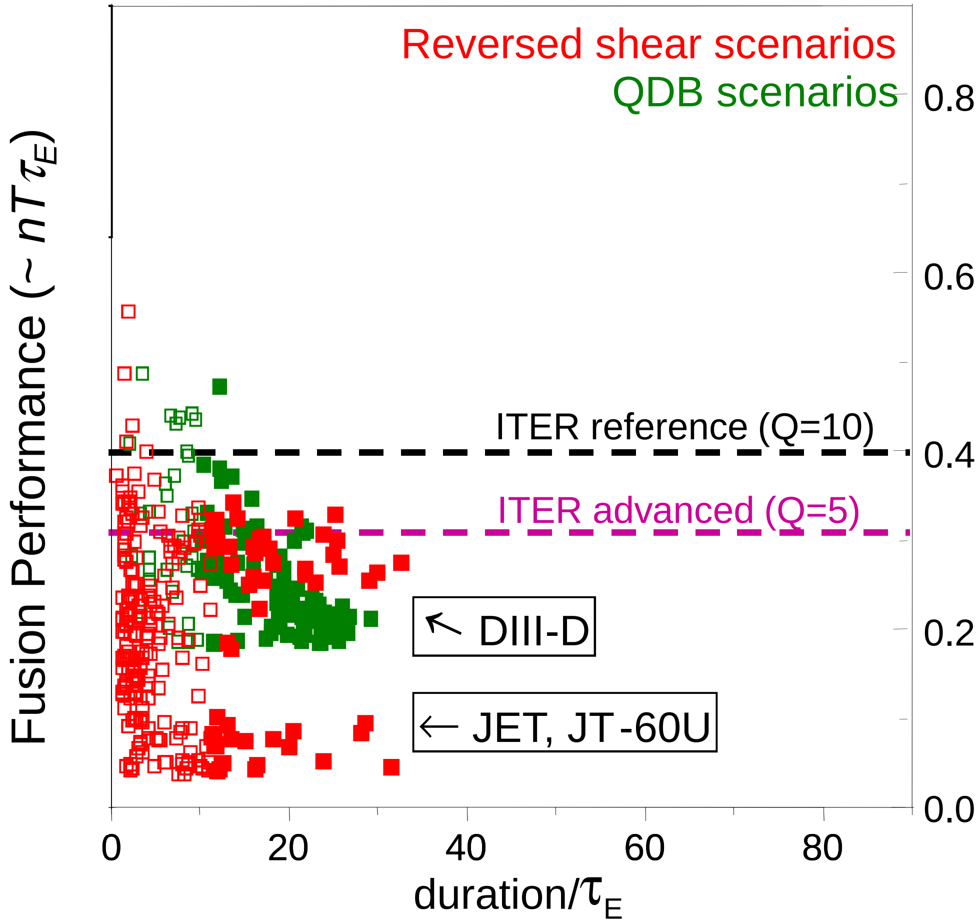




Feedback control using RMPs shows possibility to exceed no-wall β–limit

- rotation plays a strong role in this process and has to be understood better (ITER is predicted to have very low rotation)
- fast particle stabilisation can help substantially even at low rotation

Advanced Tokamak Stability is a tough Problem



Good performance can only be kept for several confinement times, not stationary on the current diffusion time (10 – 50 τ_E in these devices)

A compromise: the ‚hybrid‘ scenario

Reversed shear, ITB discharges

+ very large bootstrap fraction

+ steady state should be possible

- low β-limit (kink, infernal, RWM)

- delicate to operate

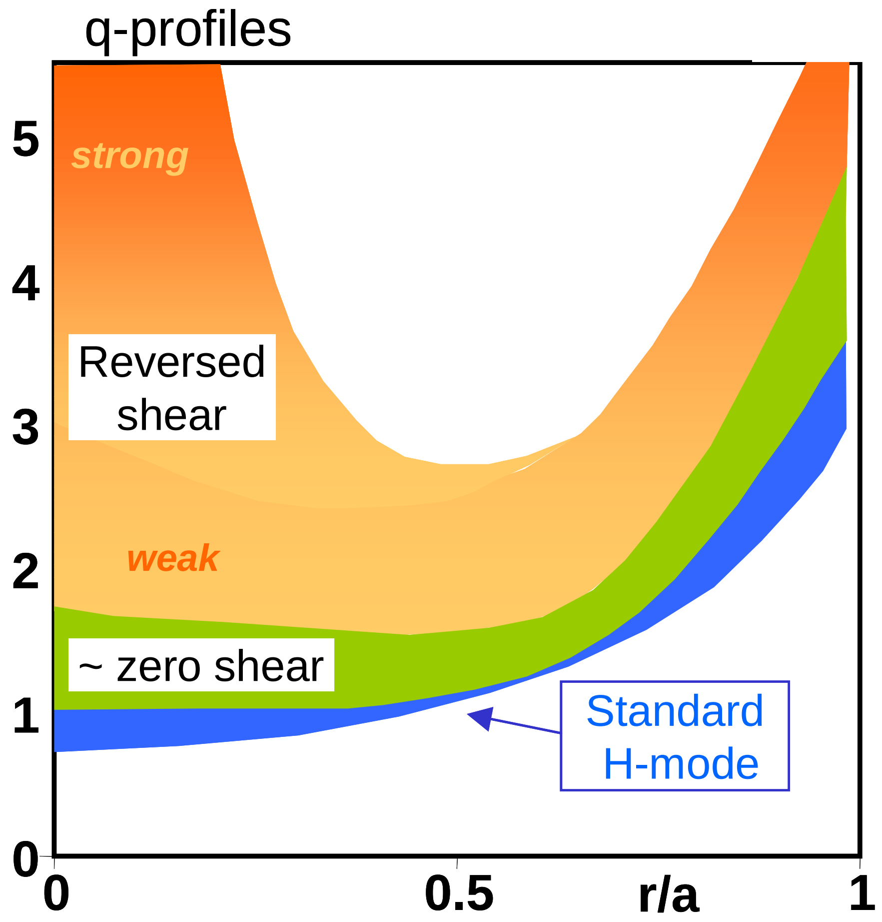
Zero shear, ‘hybrid’ discharges

+ higher β-limit (NTMs)

+ ‘easy’ to operate

- smaller bootstrap fraction

- have to elevate *q(0)*
  (also avoids sawteeth, NTMs)



Hybrid operation aims at flat, elevated q-profile discharges with high q(0)

Not clear if this projects to steady state, but it will be very long pulse…

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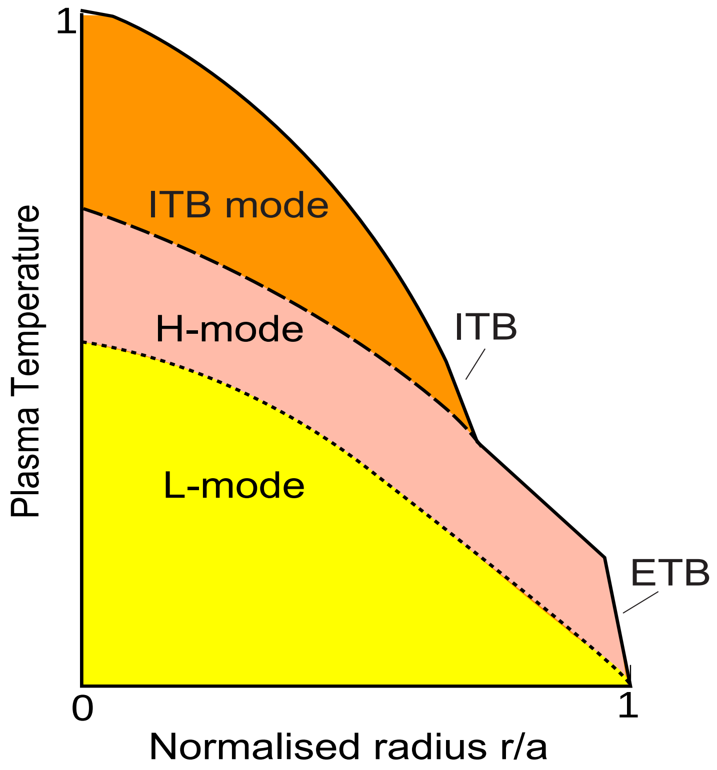

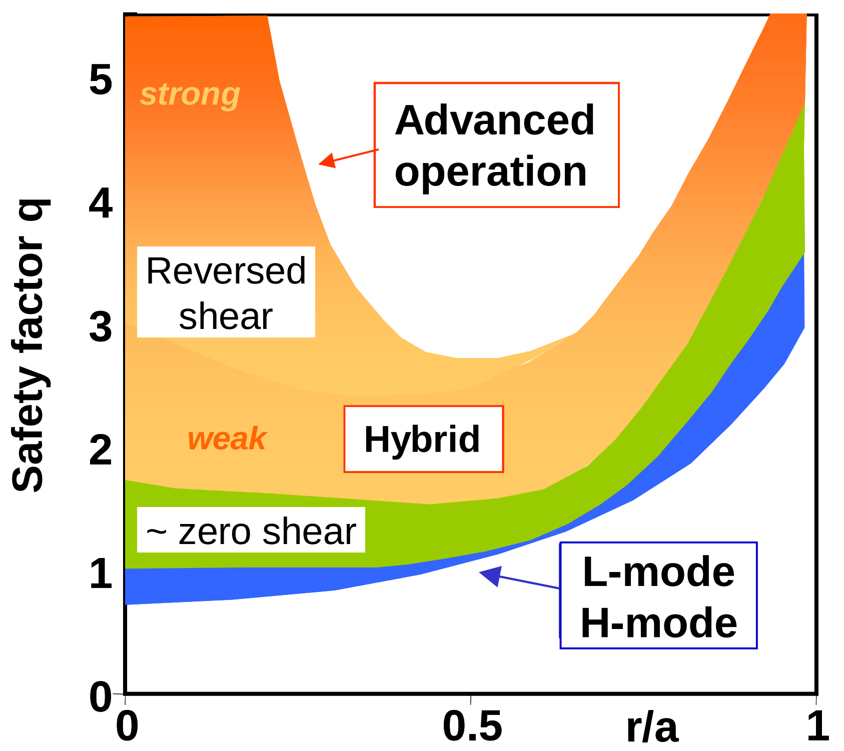

A variety of tokamak operational scenarios exists

- L-mode: low performance, pulsed operation, no need for profile control
- H-mode: higher performance, pulsed operation, MHD control needed
- Advanced modes: higher performance, steady state, needs profile control

Summary and Conclusions

ITER aims at operation in conventional *and* advanced scenarios

- demonstrating Q=10 in conventional (conservative) operation scenarios
- demonstrating long pulse (steady state) operation in ‚advanced‘ scenarios

Scenario:	***Standard***	***Low q***	***Hybrid***	***Advanced***
Ip [MA]	15	17	13.8	9
Bt [T]	5.3	5.3	5.3	5.18
βN	1.8	2.2	1.9	3
Pfus [MW]	400	700	400	356
Q	10	20	5.4	6
tpulse [s]	400	100	1000	3000

One mission of ITER and the accompanying programme is to develop and verify an operational scenario for DEMO

- DEMO scenario must be a point design (no longer an experiment)
- actuators even more limited (e.g. maximum of 2 H&CD methods)