Fusion Reactor Technology 2
(459.761, 3 Credits)

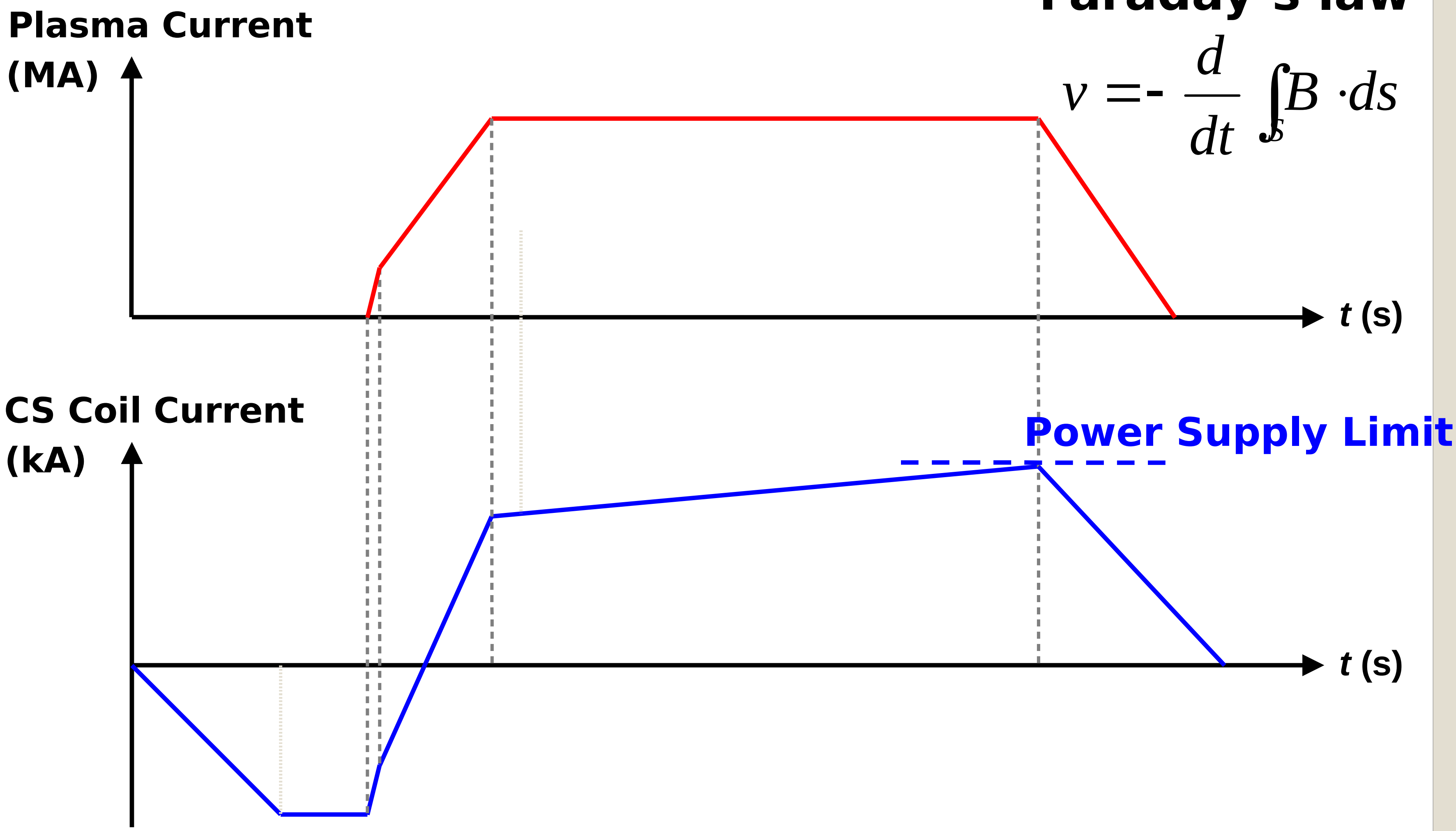
Prof. Dr. Yong-Su Na

(32-206, Tel. 880-7204)

Pulsed Operation

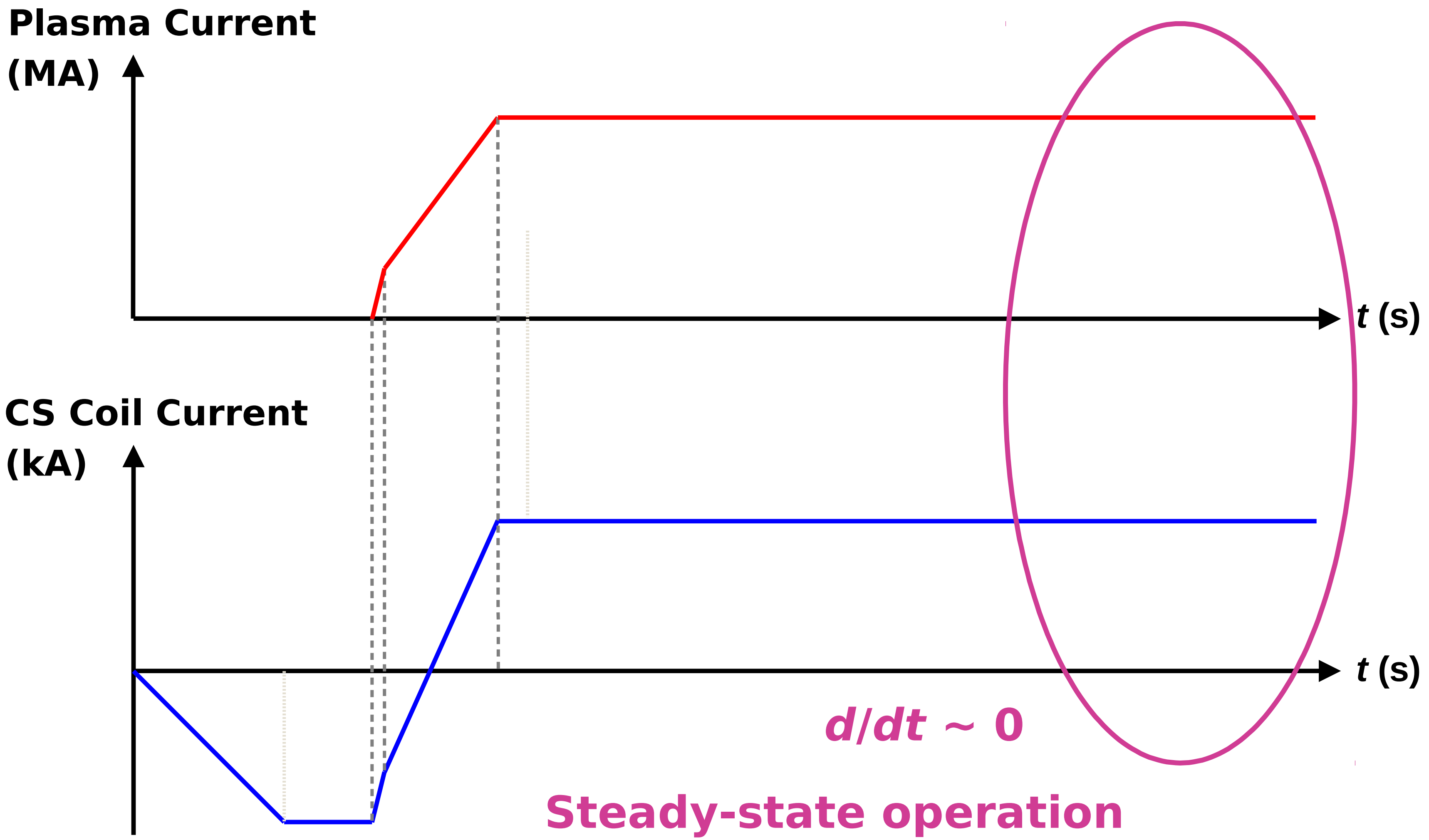
Faraday's law

$$v = -\frac{d}{dt}\int_S B \cdot ds$$



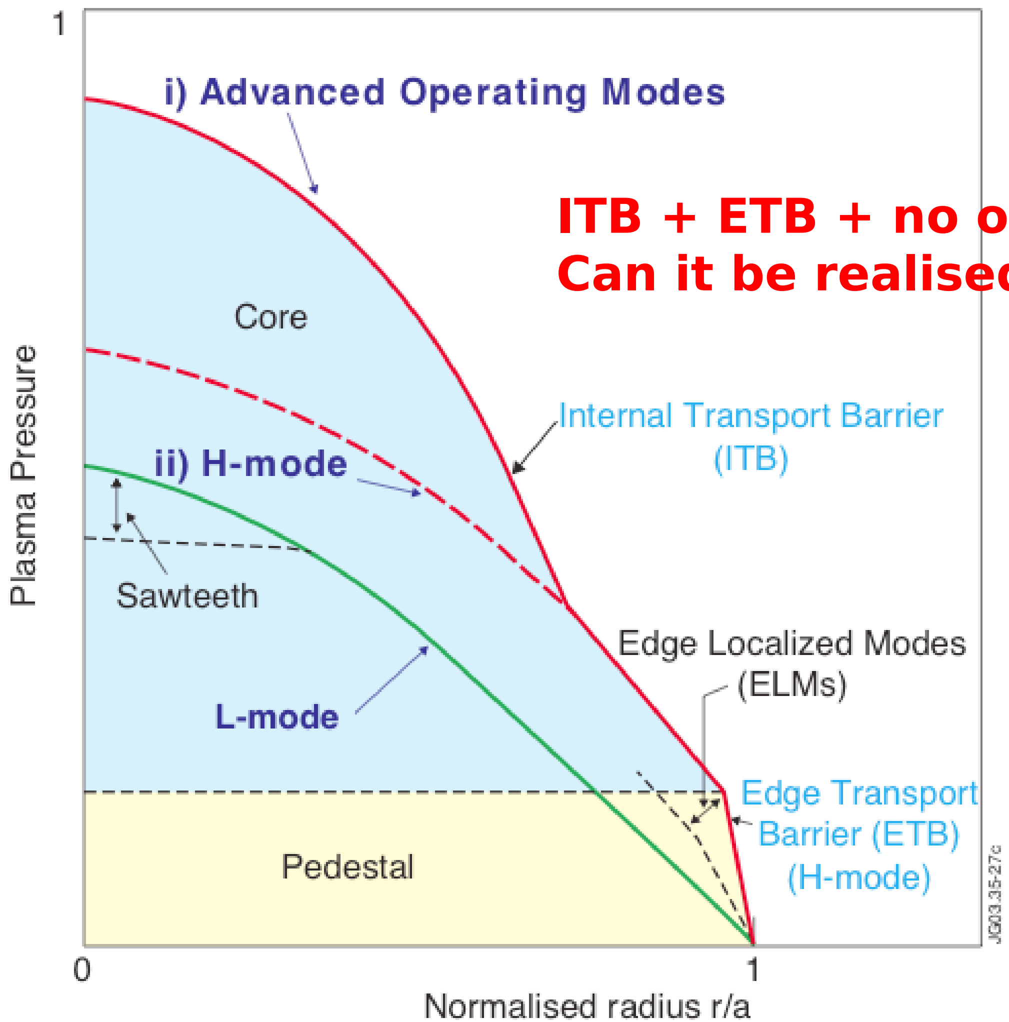
Inherent drawback of Tokamak!

Steady-State Operation



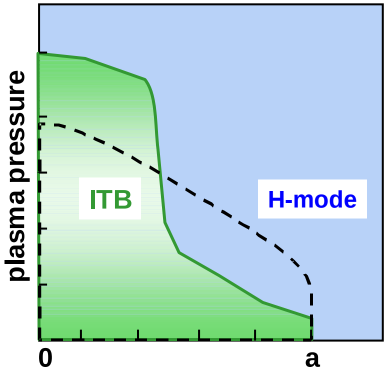
Steady-state operation
by self-generated and externally driven current

Tokamak Operation Modes

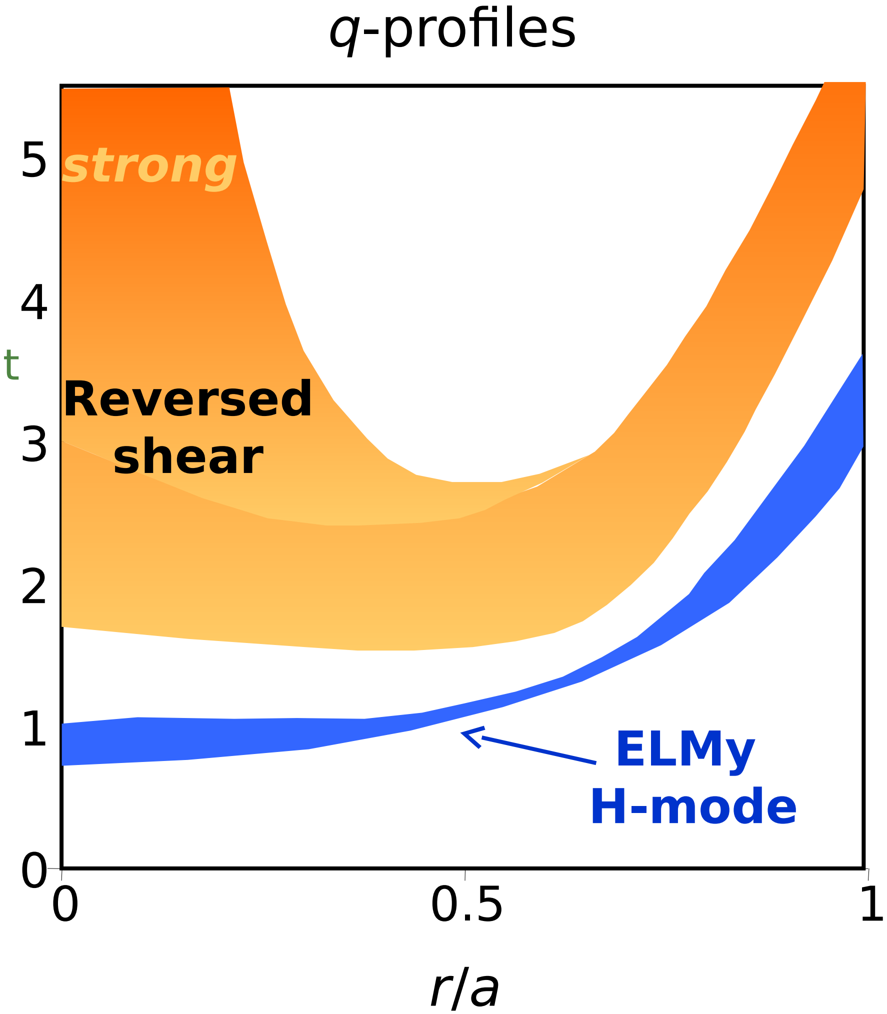


ITB + ETB + no or mild ELM is our dream!
Can it be realised?

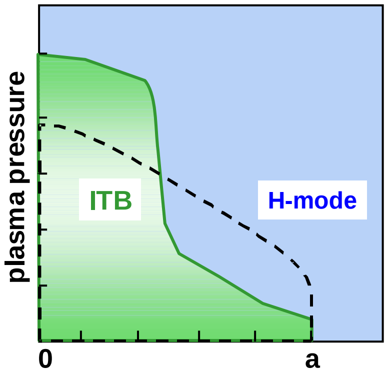
Advanced Operating Modes



- Good confinement
- Poor stability

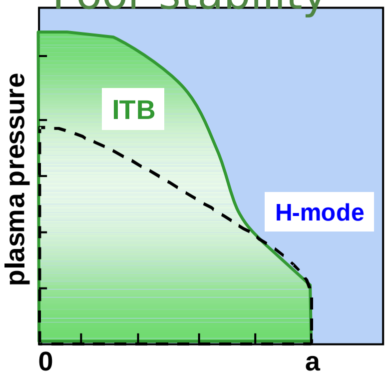


Advanced Operating Modes

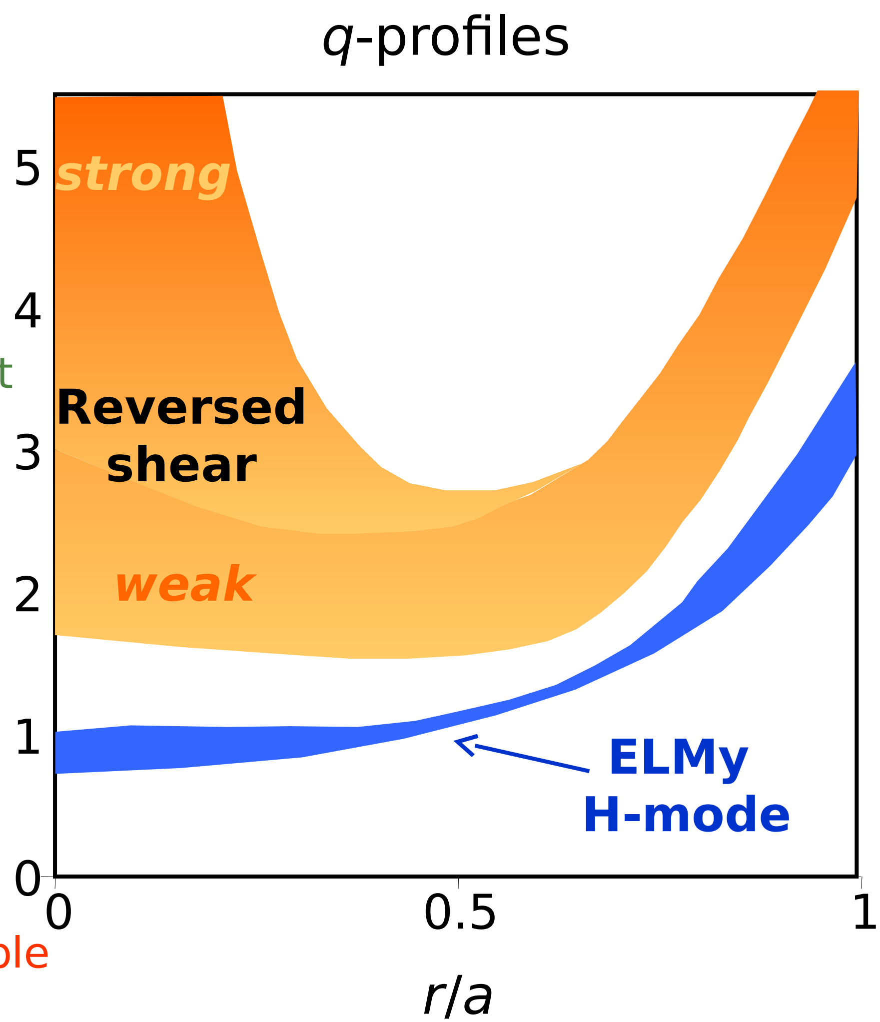


- Good confinement

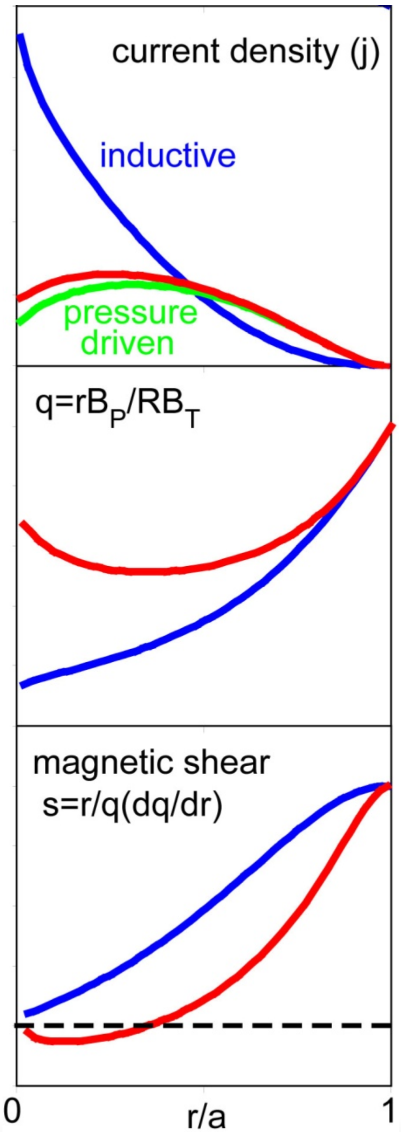
- Poor stability

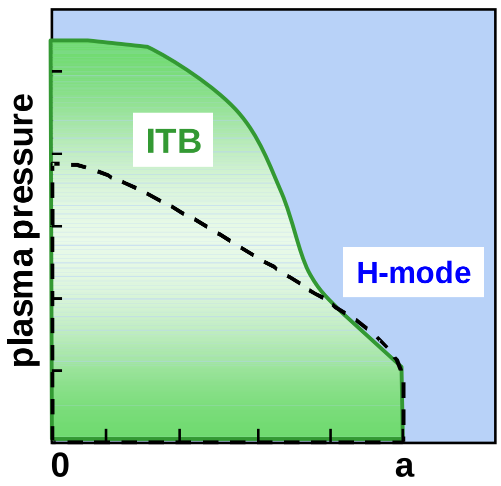


- Only "weak" RS plasmas are stable but they require a delicate active control



Reversed Shear Mode

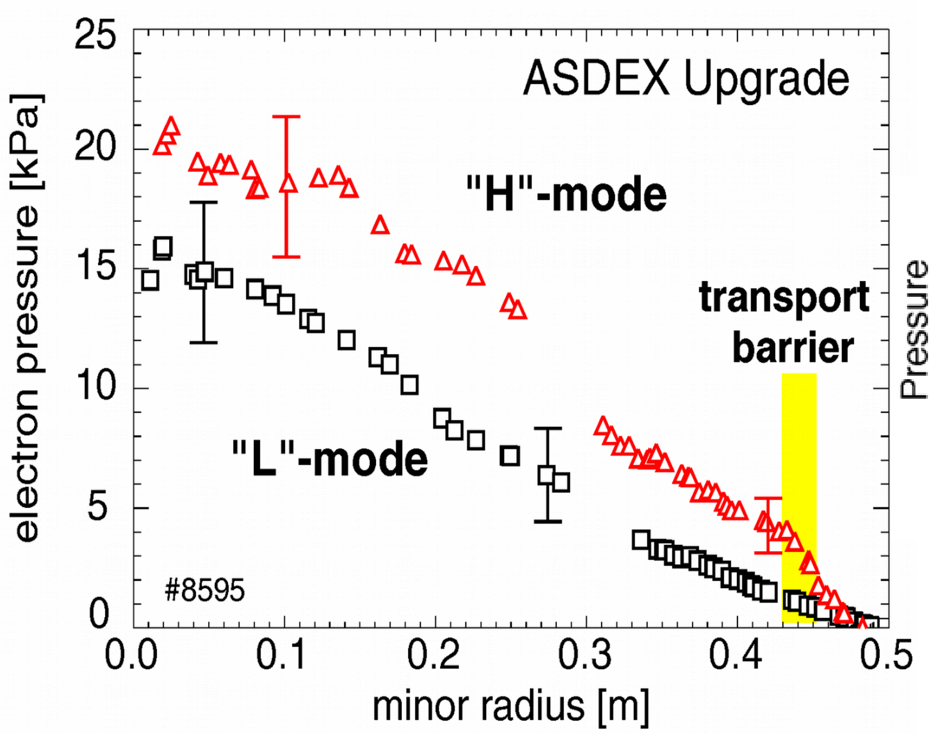




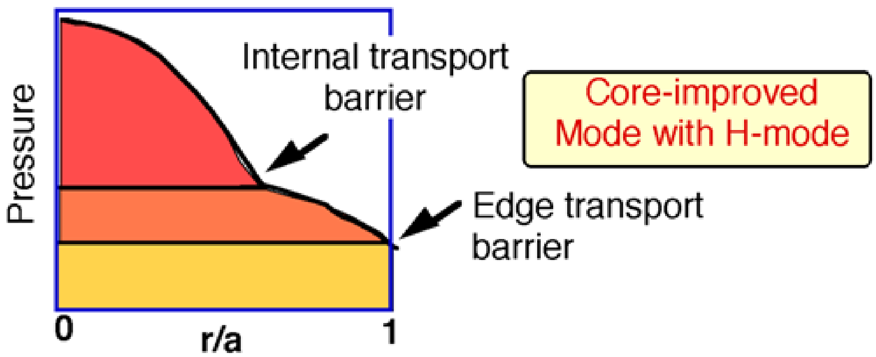
- Higher pressure gradient region in the core with steep edge pedestal
- Hollow current profile
- Reversed *q*-profile
- With negative magnetic shear

Reversed Shear Mode

H-mode

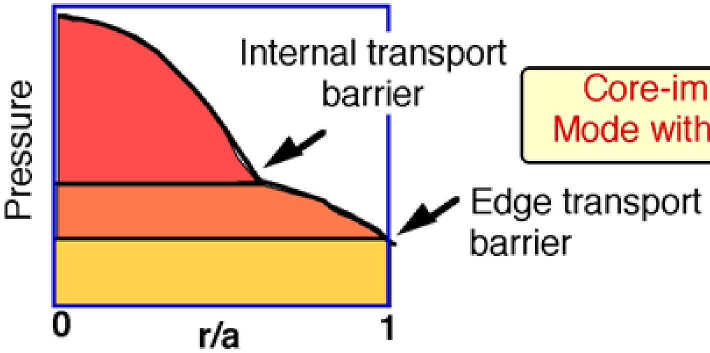


Reversed shear mode



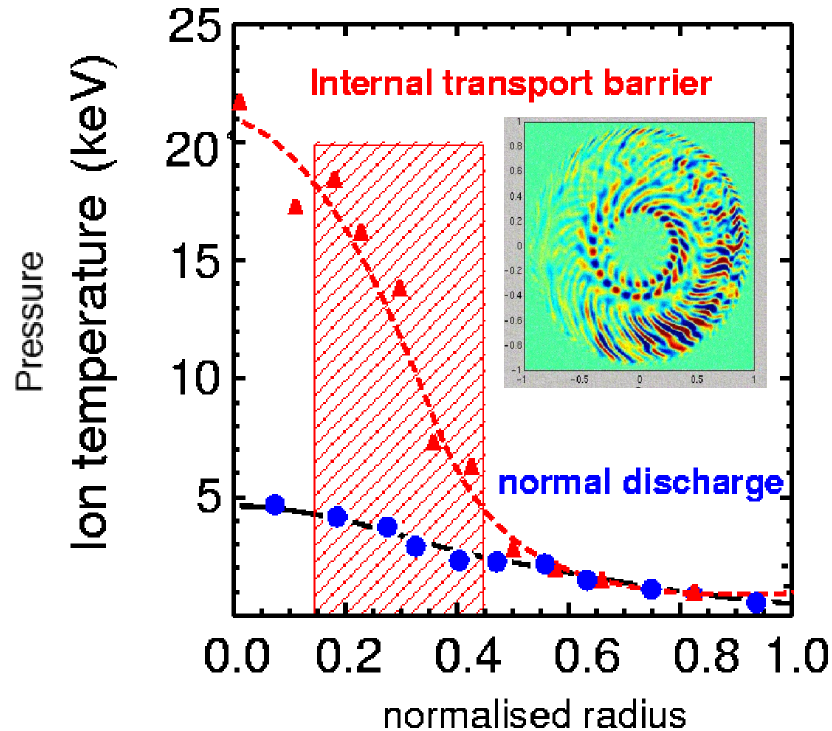
ASDEX Upgrade

Reversed Shear Mode



Core-improved Mode with H-mode

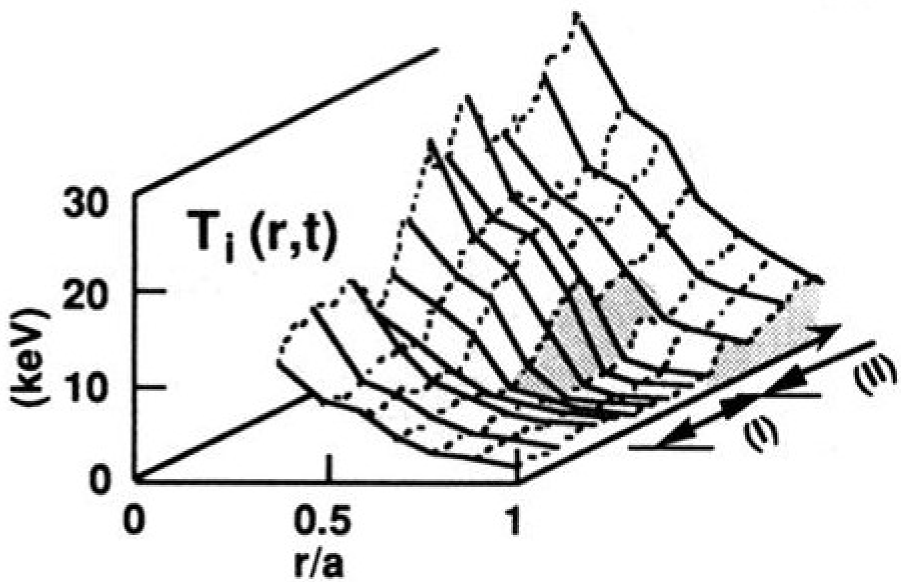
Reversed Shear Mode



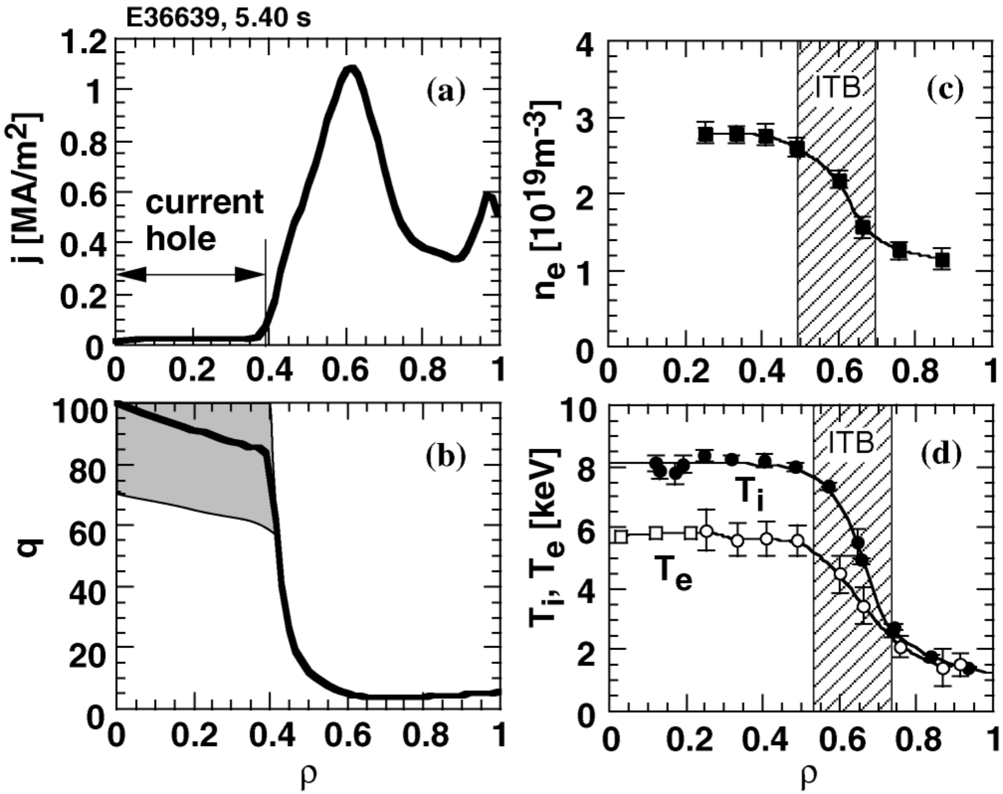
Internal Transport Barrier

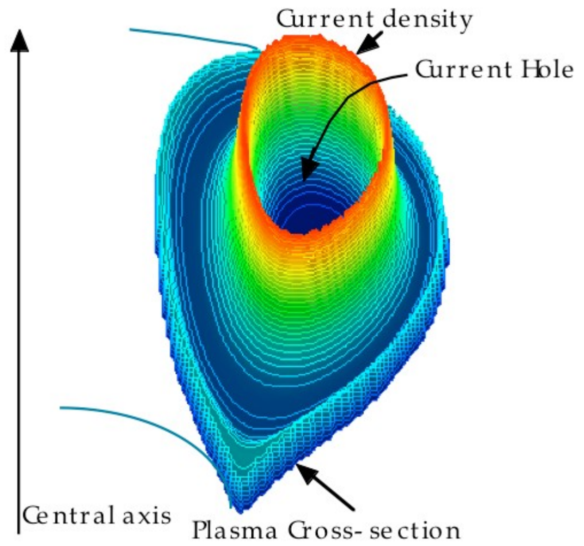
Tokamak operation in the high-β_p regime is a promising concept for a steady-state tokamak reactor [1,2]. Here the poloidal beta is defined as $\beta_p = 2\mu_0\langle p\rangle/B_p^2$, where $\langle p\rangle$ is the volume-averaged plasma pressure and B_p is the averaged poloidal magnetic field on the face. An energy confinement time, τ_E, more that for L mode (for example, ITER89-P [3] in the high-β_p regime to reduce the plasma cu nition and hence to achieve efficient steady-st operation [4]. Improved confinement time w in the high-β_p regime ($\beta_p = 1-2$) in JT- the confinement improvement factor, τ_E/τ creased with $\varepsilon\beta_p$ [5]. In this regime, the "hig a bootstrap-current fraction of up to 58% a ion temperature, $T_i(0)$, of 38 keV were ach taneously. Recently the high-β_p mode regime ed to a lower q regime ($q_{\text{eff}} \sim 4.3$; q_{eff} is the e face safety factor defined in Ref. [6]) by u... profile control to avoid sawteeth. And high fusion performance was attained in this regime [7,8]. This Letter describes two distinctive features of this high-β_p mode: (1) the formation of an "internal" transport barrier near the $q = 3$ rational surface and (2) the appearance of high poloidal plasma rotation velocity of ~ 50 km/s in the plasma interior.

and Poloidal Plasma Spin Up



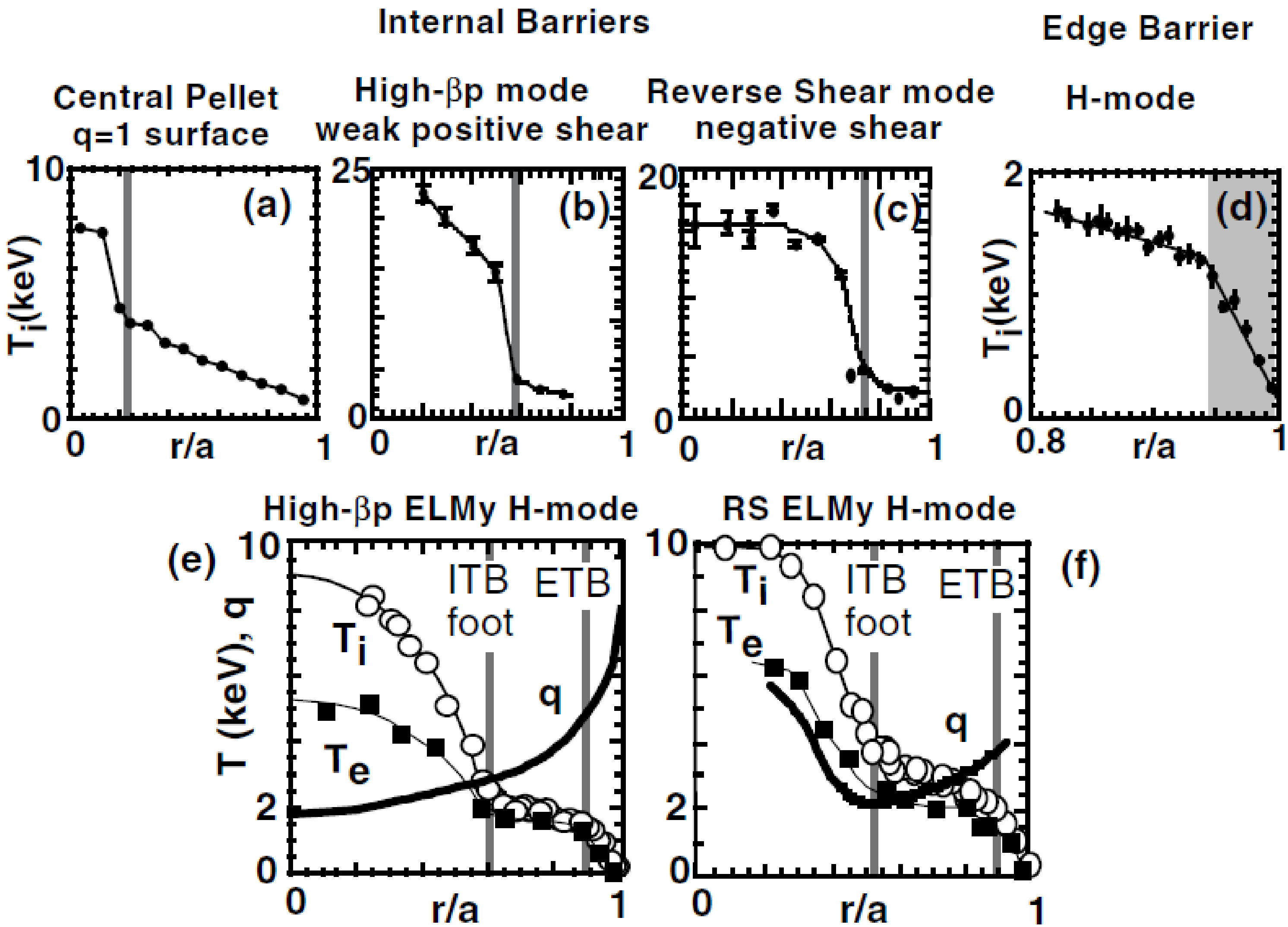
Current Hole Regime





*T. Fujita et al, PRL **87** 245001 (2001)*

Internal+Edge Transport Barrier



ITB without reversed shear

*Y. Kamada, PPCF **42** A65 (2000)*

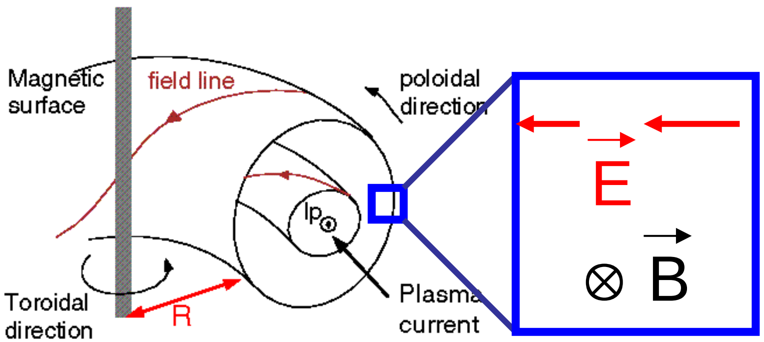
Turbulence Stabilisation

- **Formation of internal transport barriers to improve confinement**

- Reversed magnetic shear

- Rotation shear

} **Stabilises turbulence**



- **One reason:**

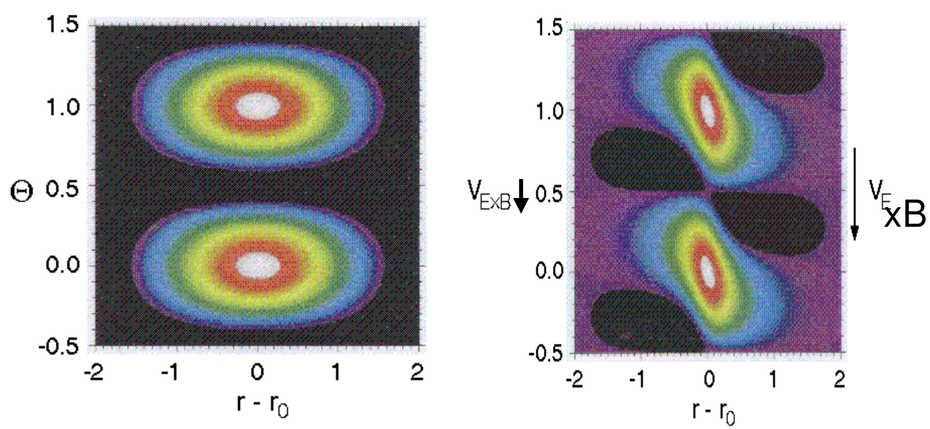
Losses of fast ions at the plasma edge

 sheared radial electric field

 sheared mean ExB rotation

 eddies get tilted and ripped apart

cause turbulence suppression!



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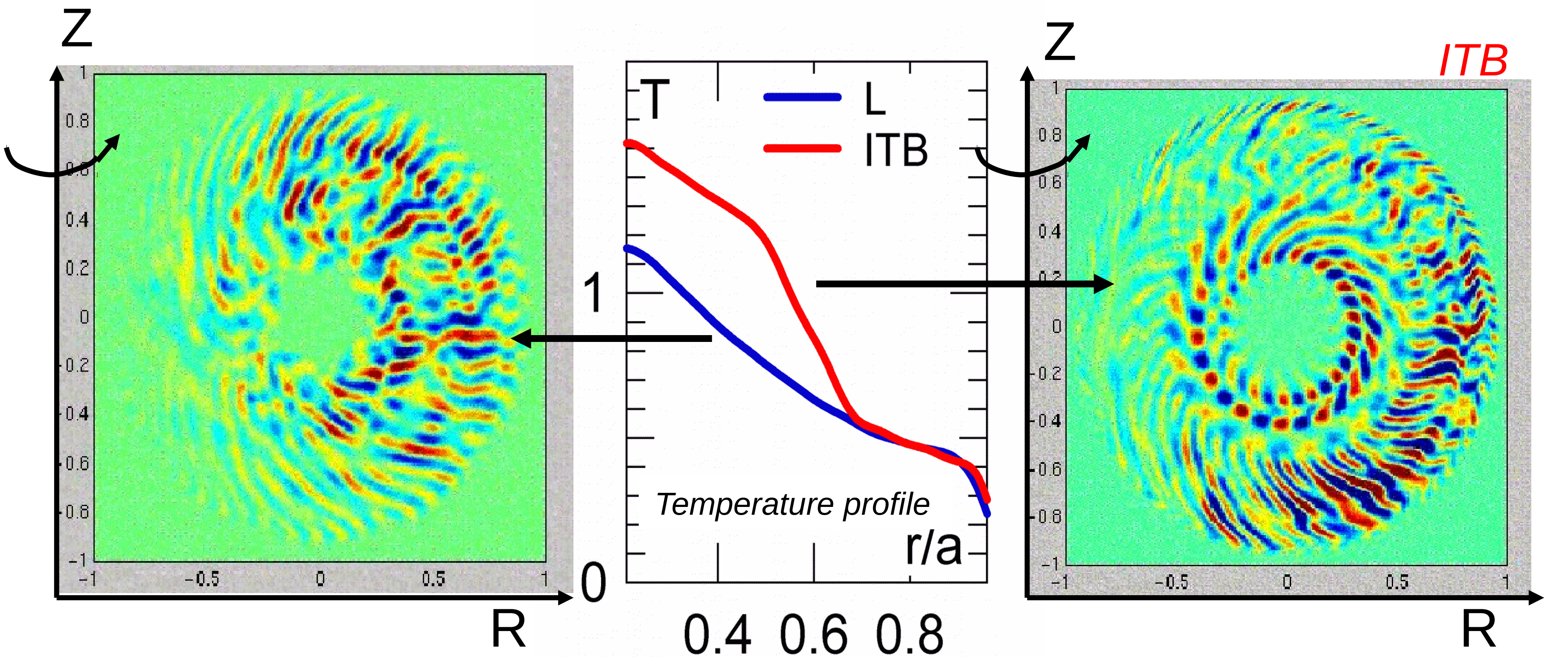
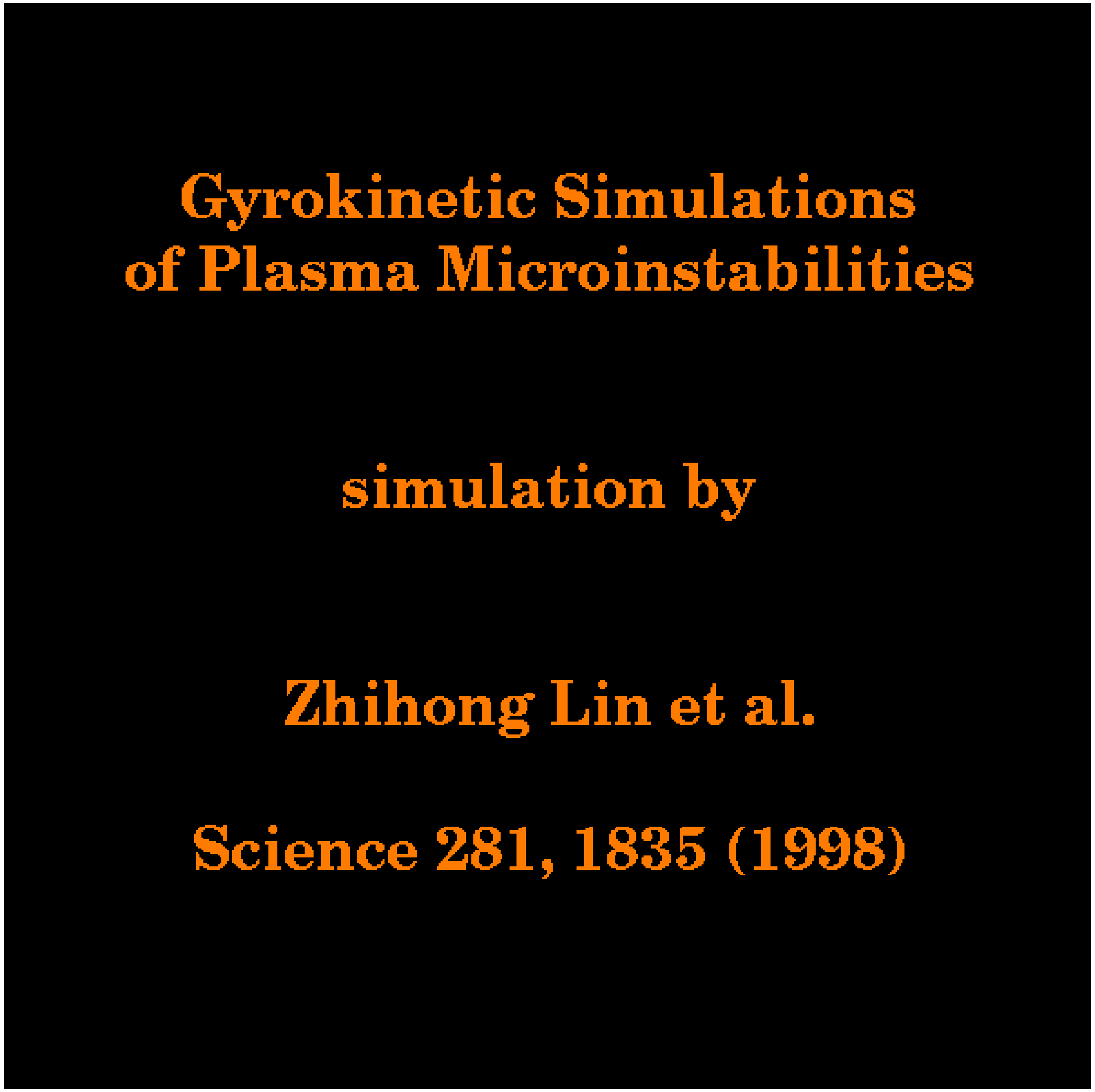


Contour lines of electric potential.

Contour lines of electric potential.

Turbulence Stabilisation

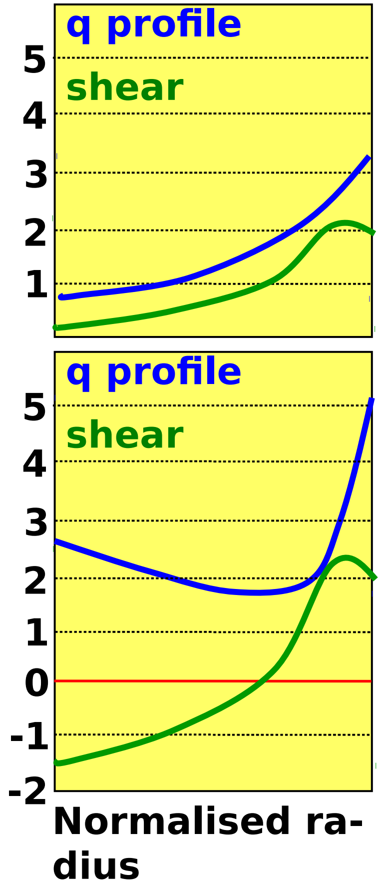


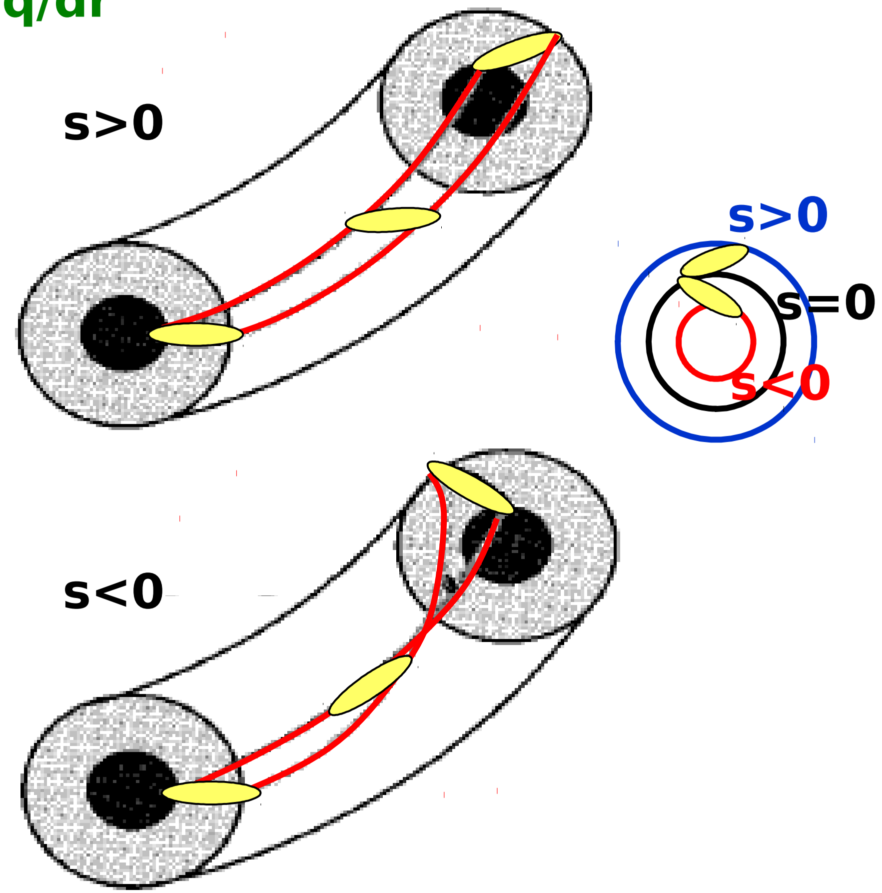
Turbulence Stabilisation

q & magnetic shear: s = r/q dq/dr

Standard scenario

Advanced tokamak scenario





Magnetic shear can twist plasma disturbances

Turbulence Stabilisation

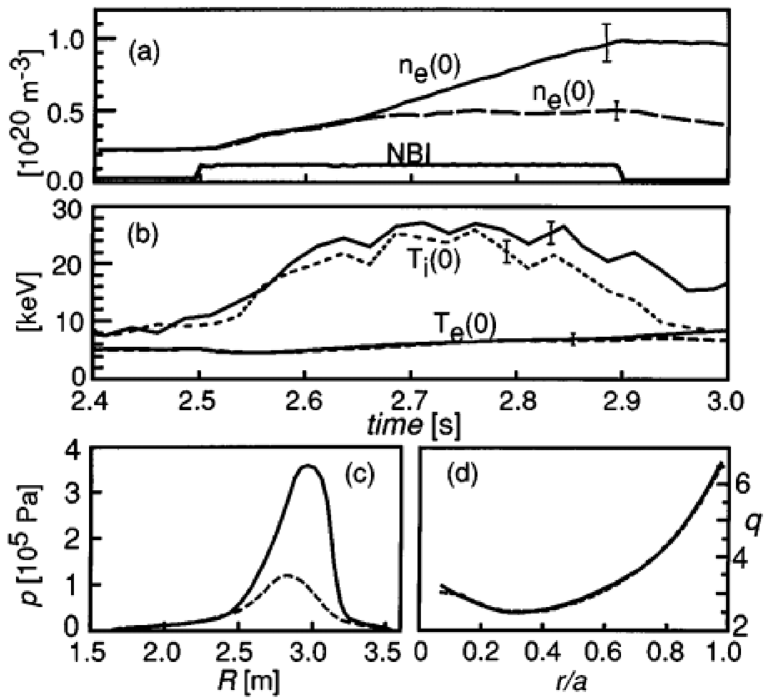


FIG. 1. Time evolution of central electron density (a) and temperatures (b) in ERS (solid line) and RS (dashed line) plasmas with 29 and 27 MW of balanced NBI, respectively. The bottom graphs show the radial profiles of the plasma pressure (c) at $t = 2.9$ s, and the safety factor (d) at the bifurcation time ($t \approx 2.65$ s).

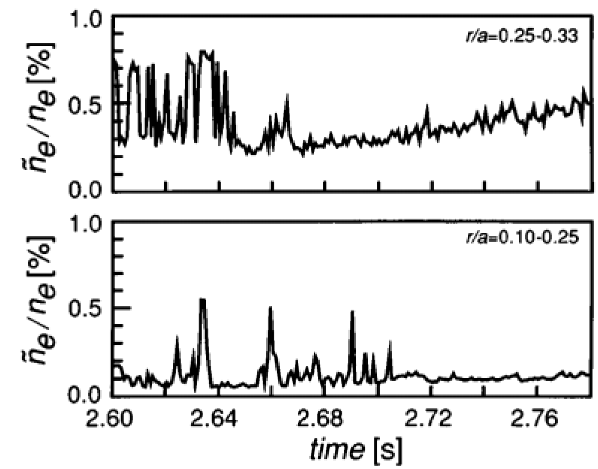


FIG. 3. Time evolution of density fluctuations in the ERS mode.

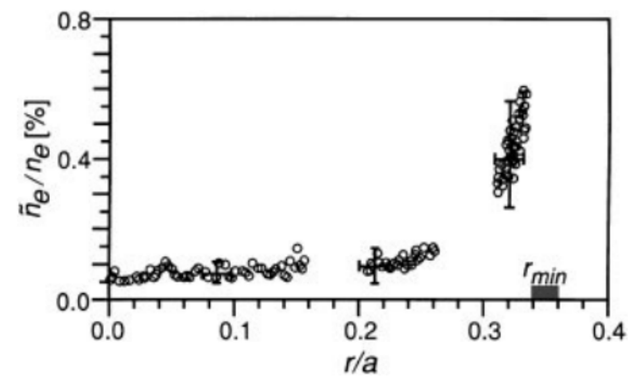
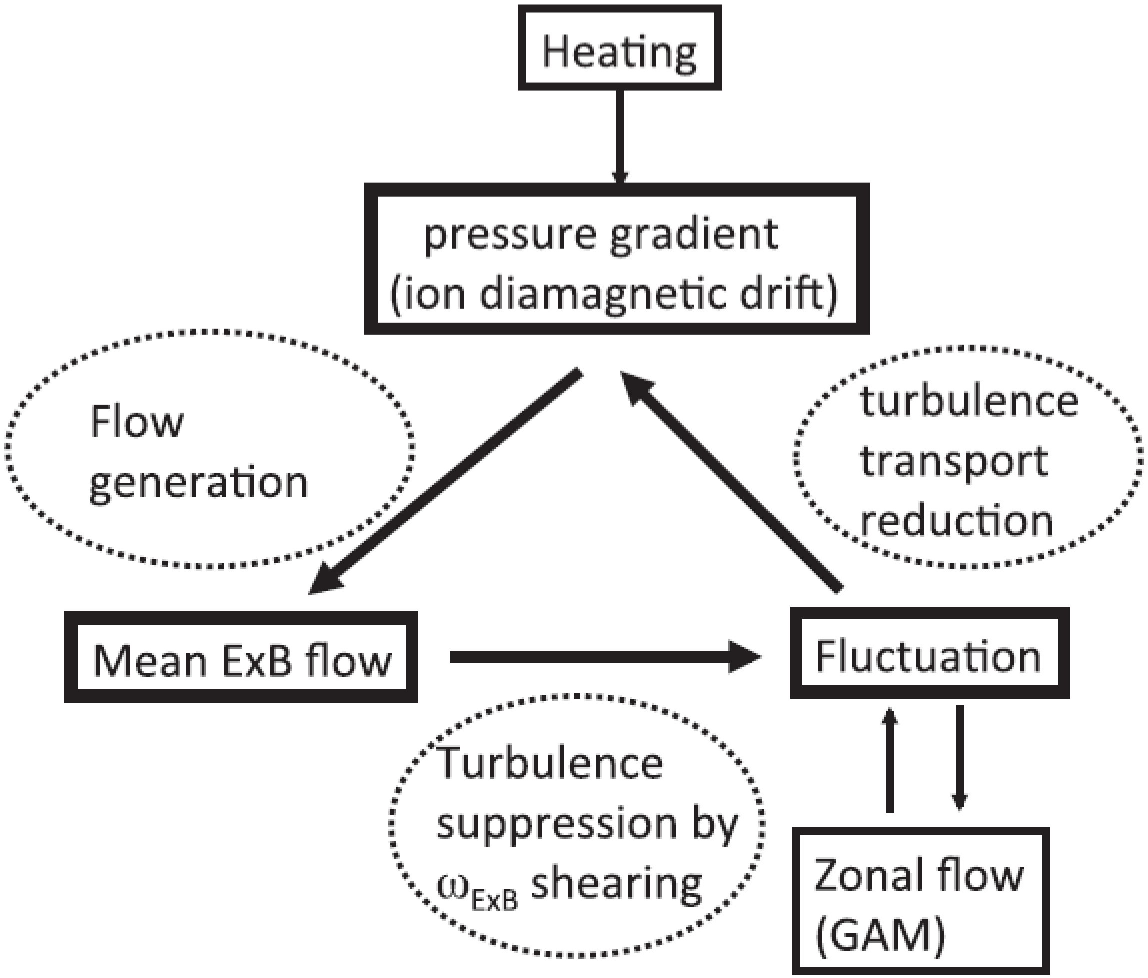


FIG. 4. Amplitude of density fluctuations in the ERS mode at $t = 2.72-2.78$ s; $r_{\min}$ is the radial position with minimum q.

*E. Mazzucato et al, PRL **77** 3145 (1996)*

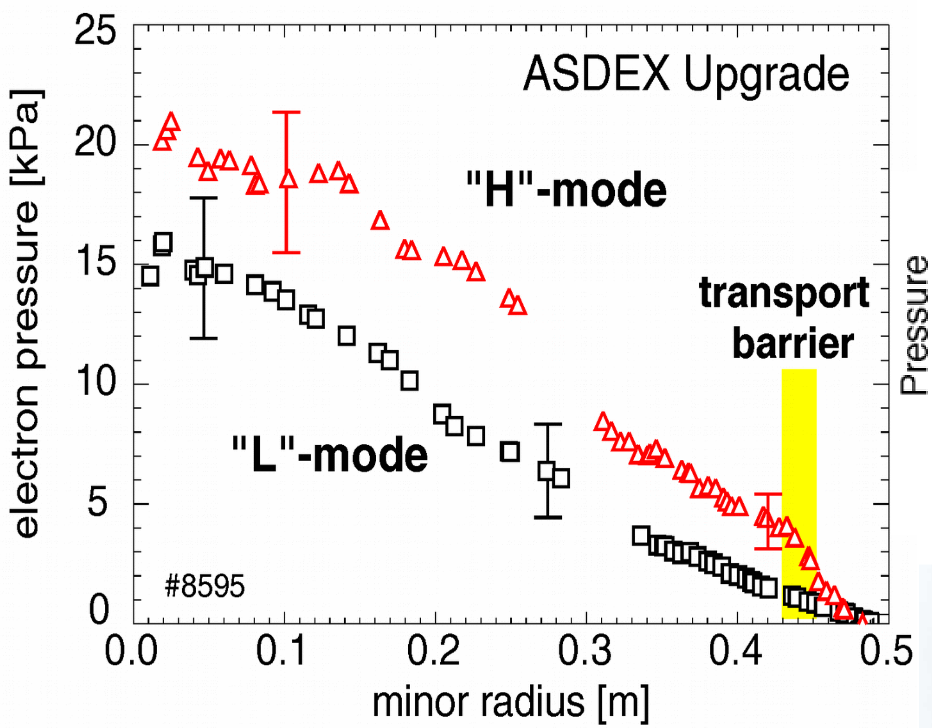
Turbulence Stabilisation



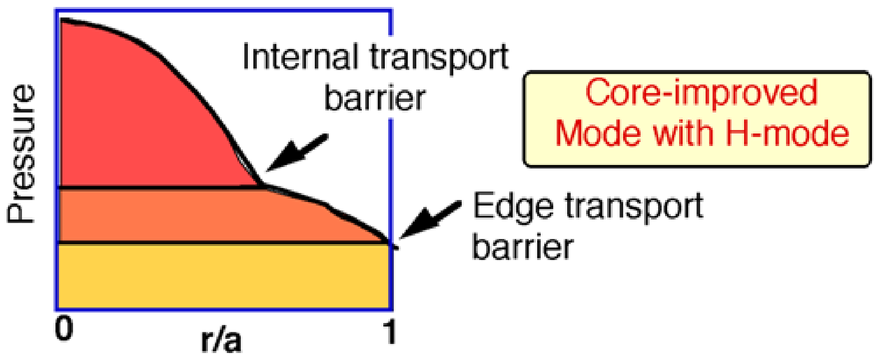
*K. Ida and T. Fujita, PPCF **60** 033001 (2018)*

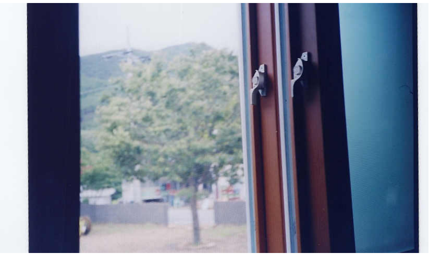
Reversed Shear Mode

H-mode



Reversed shear mode





Reversed Shear Mode

HW: How can one identify the ITBs (electron or ion or heat or particle or rotation) in the experiment?

Explain any diagnostics or theories which can be used to identify them.

Find scalings of threshold power for ITB formation.

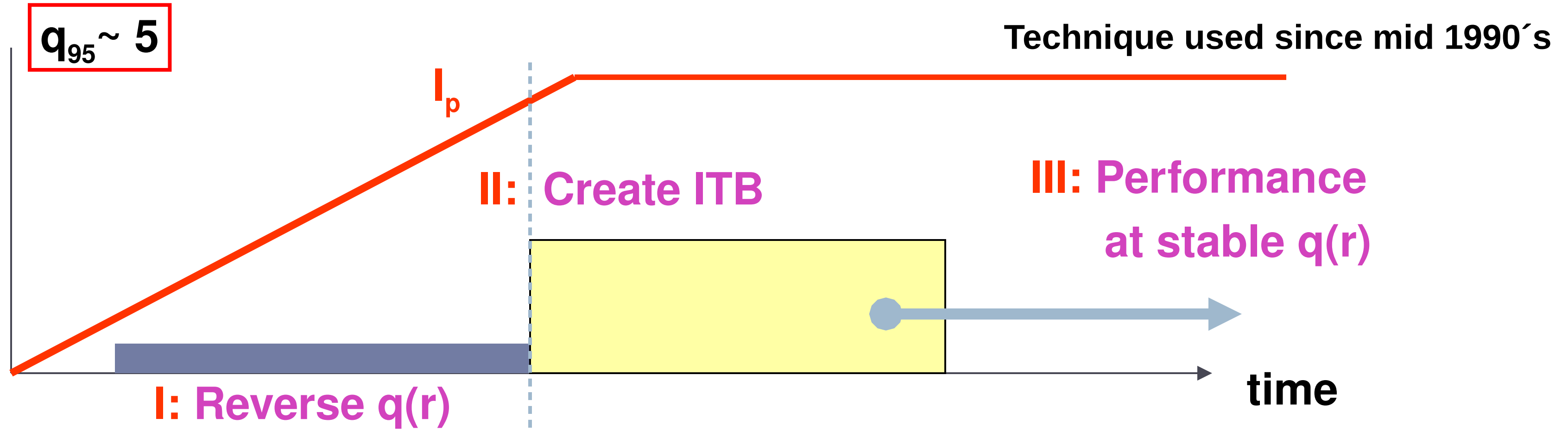
Reversed Shear Mode

- **Operation at lower plasma current:** $f_{BS} \sim \beta_p \sim I_p^{-2}$

→ **Confinement degradation:** $\tau_E \sim H_{98}(y,2)\ I_p^{0.93}$

→ **To get enough fusion power:** $H_{98}(y,2) > 1$ **(advanced)**

Reversed Shear Scenario



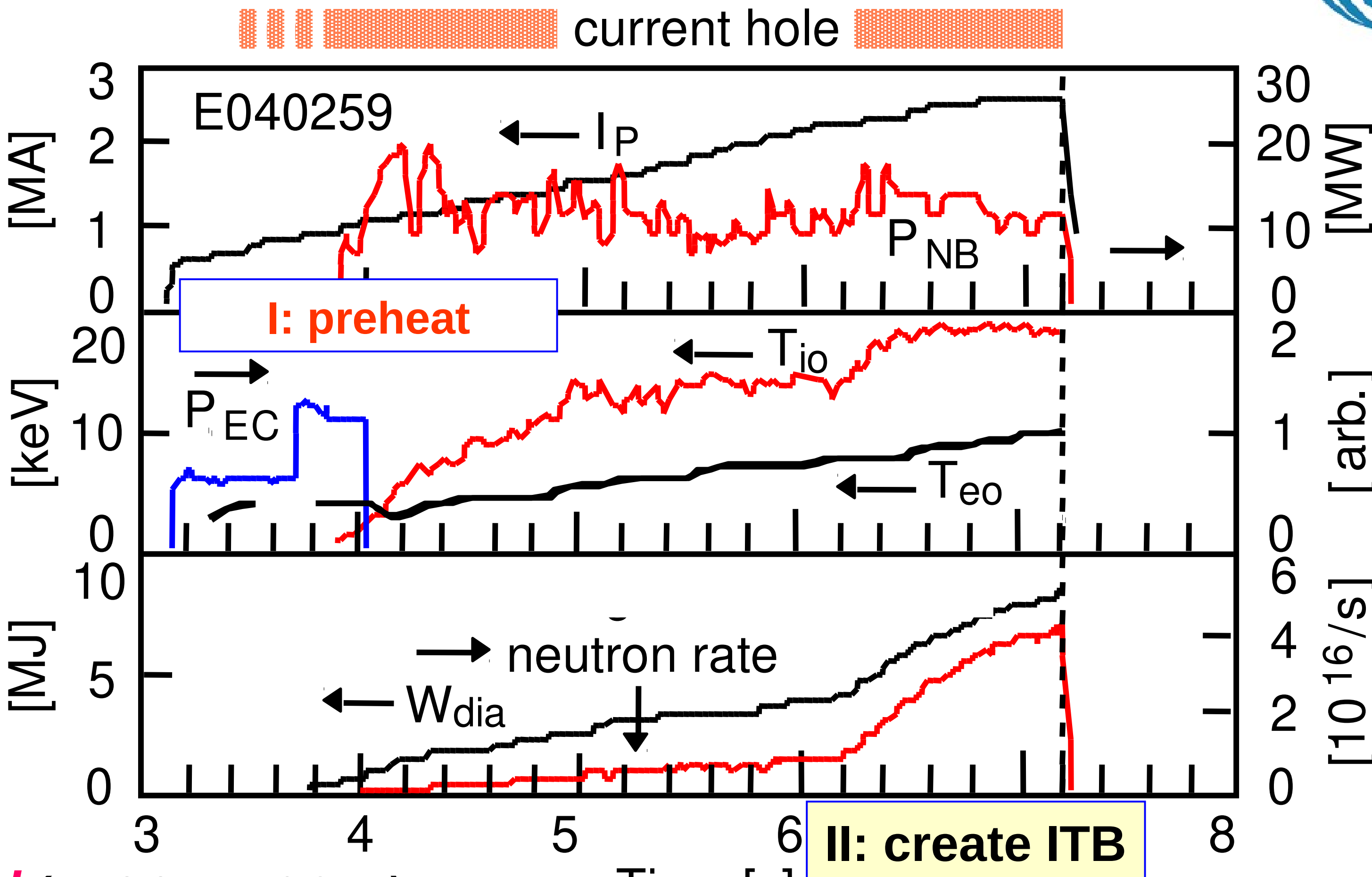
I: Heat during current rise, external current drive (reversed q).

II: Increase heating power to produce ITB (turbulence stabilised) and improve plasma confinement, try to increase pressure (β_N)

III: Keep going
(e.g. ITER steady state scenario at 9 MA : H_{98} ~ 1.6, β_N ~ 3.0, 50% external current drive (73MW), 50% bootstrap fraction)

T Fukuda and the JT-60U team, PPCF ***44*** *B39 (2002)*

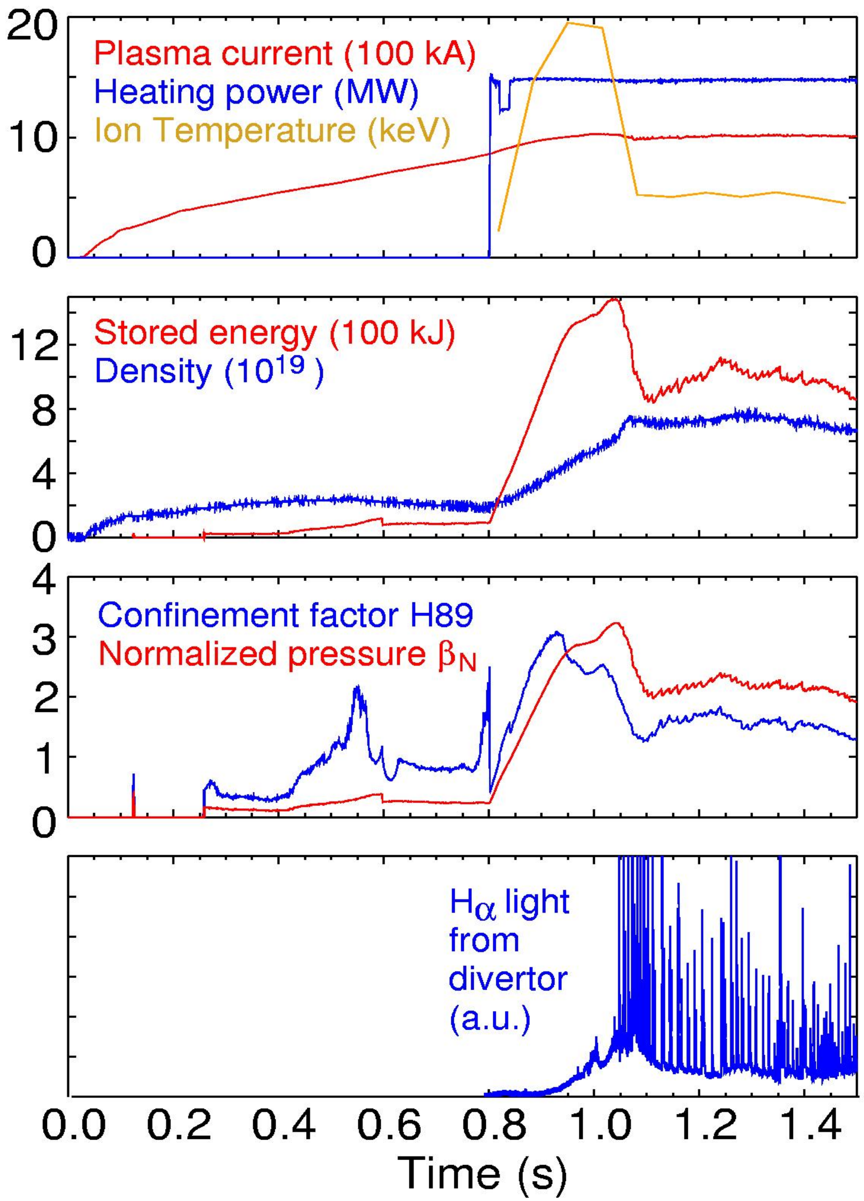
Reversed Shear Scenario



JT-60U ***(R_0=3.3m. a=0.8m.)***

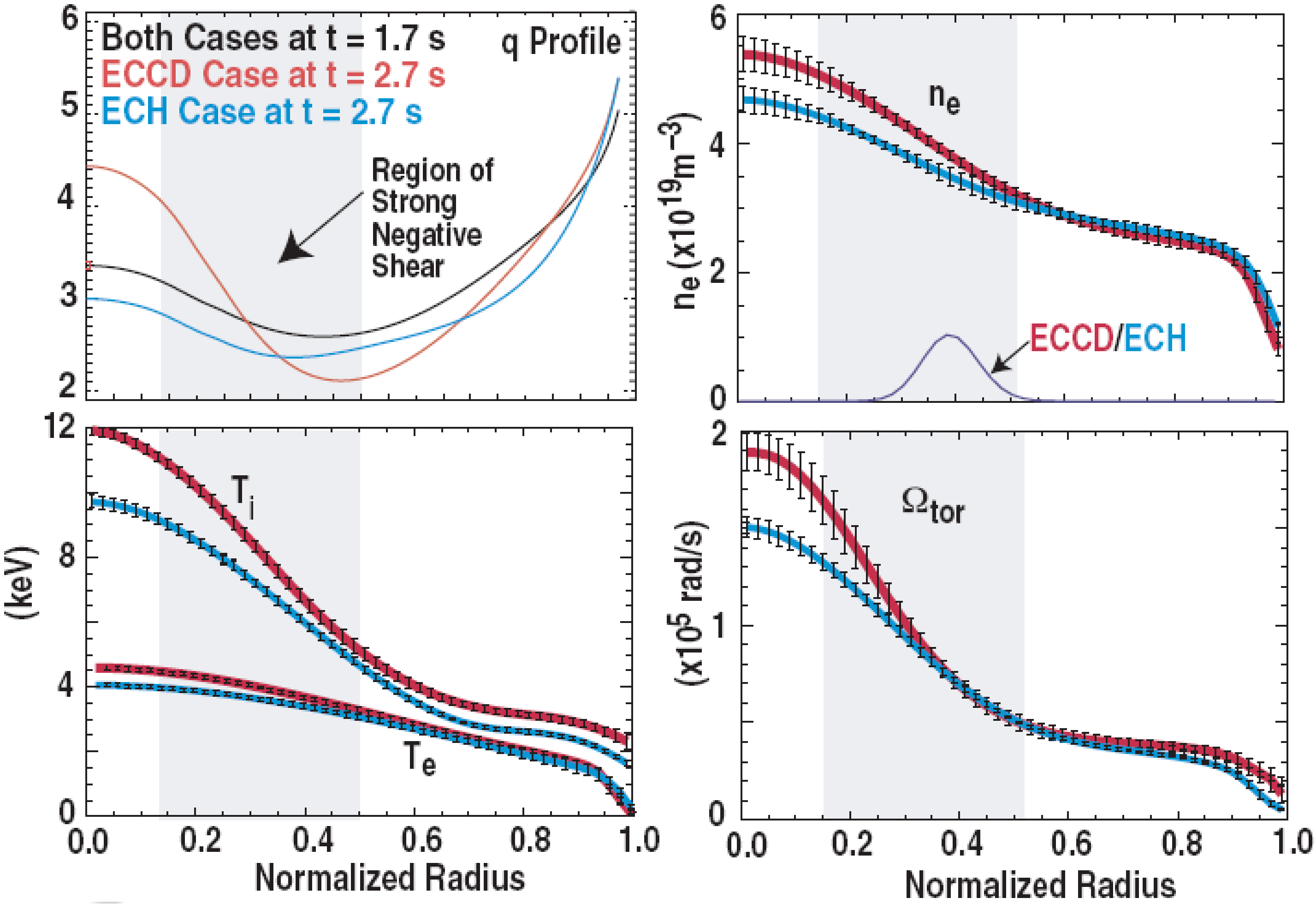
I: Form q(r), II: create ITB, III: But discharge terminates (unstable)

Reversed Shear Scenario



- Formation of an ITB at low n_e, with 15 MW NBI power $T_i > T_e$, high rotation shear
- ITBs are relatively short lived, only few τ_E
- Good, transient performance: $H_{89} \sim 3$, $\beta_N \sim 3$
- ITB not compatible with H-mode edge barrier and large ELMs

Reversed Shear Scenario



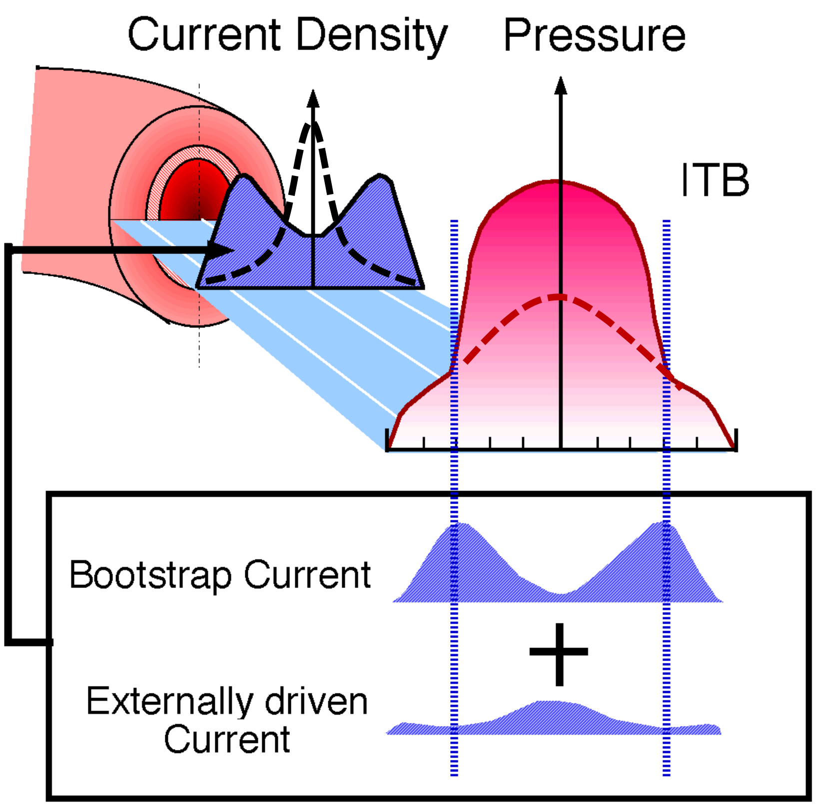
DIII-D NATIONAL FUSION FACILITY SAN DIEGO

(R_0 = 1.7 m, a = 0.6 m)

- weak RS
- $H_{98} = 1.37$
- $\beta_N = 2.62$
- $q_{95} = 5$
- „Weak“ ITBs

*M. Wade et al., Nucl. Fusion **43** 634 (2003)*

Current drive and current profile control



Non-monotonic current profile

↓

Turbulence suppression

↓

High pressure gradients

↓

Large bootstrap current

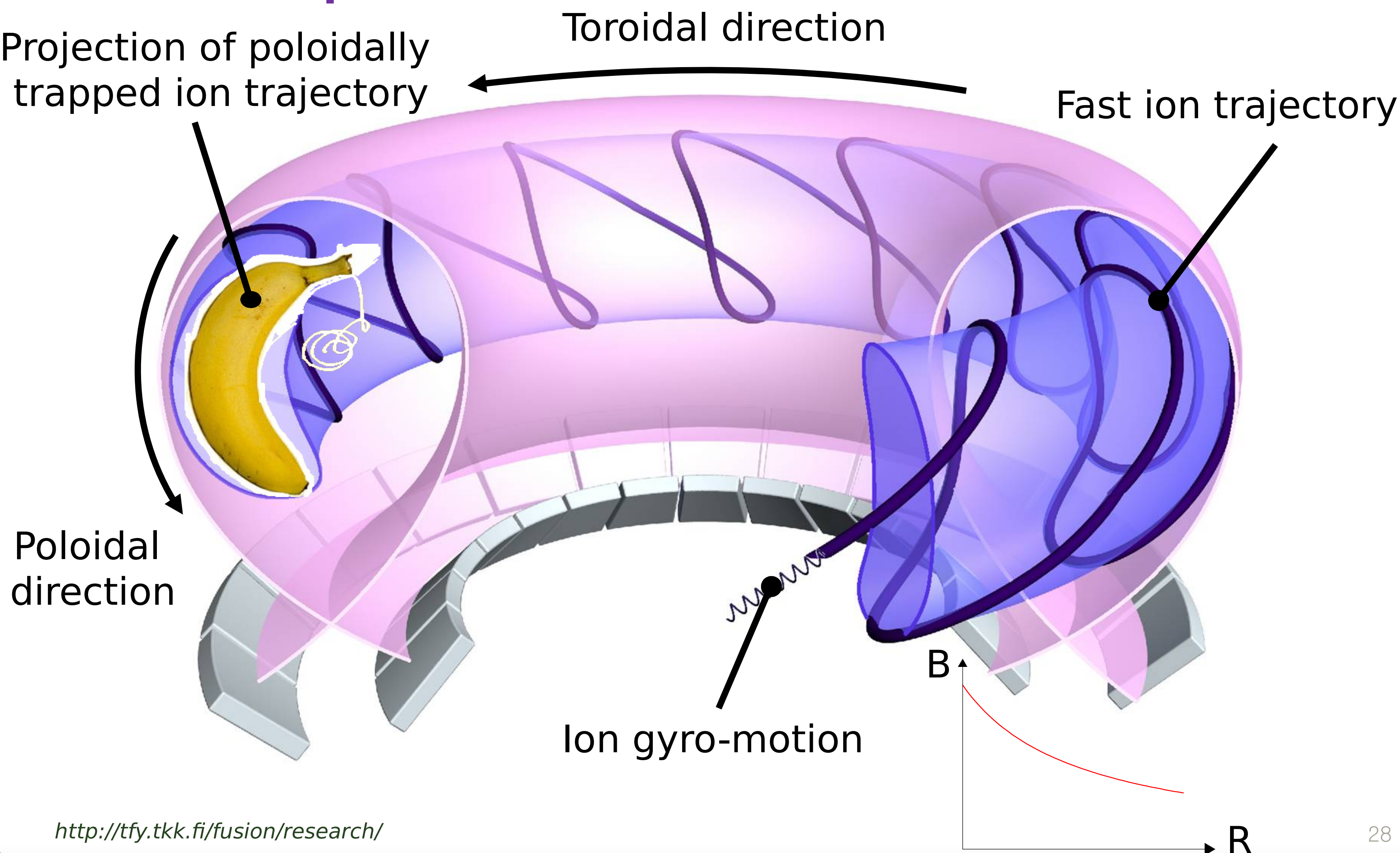
↓

Non-inductive current drive

Does it make a closed loop? **Unfortunately, no!**

Current drive and current profile control

- **Bootstrap current**

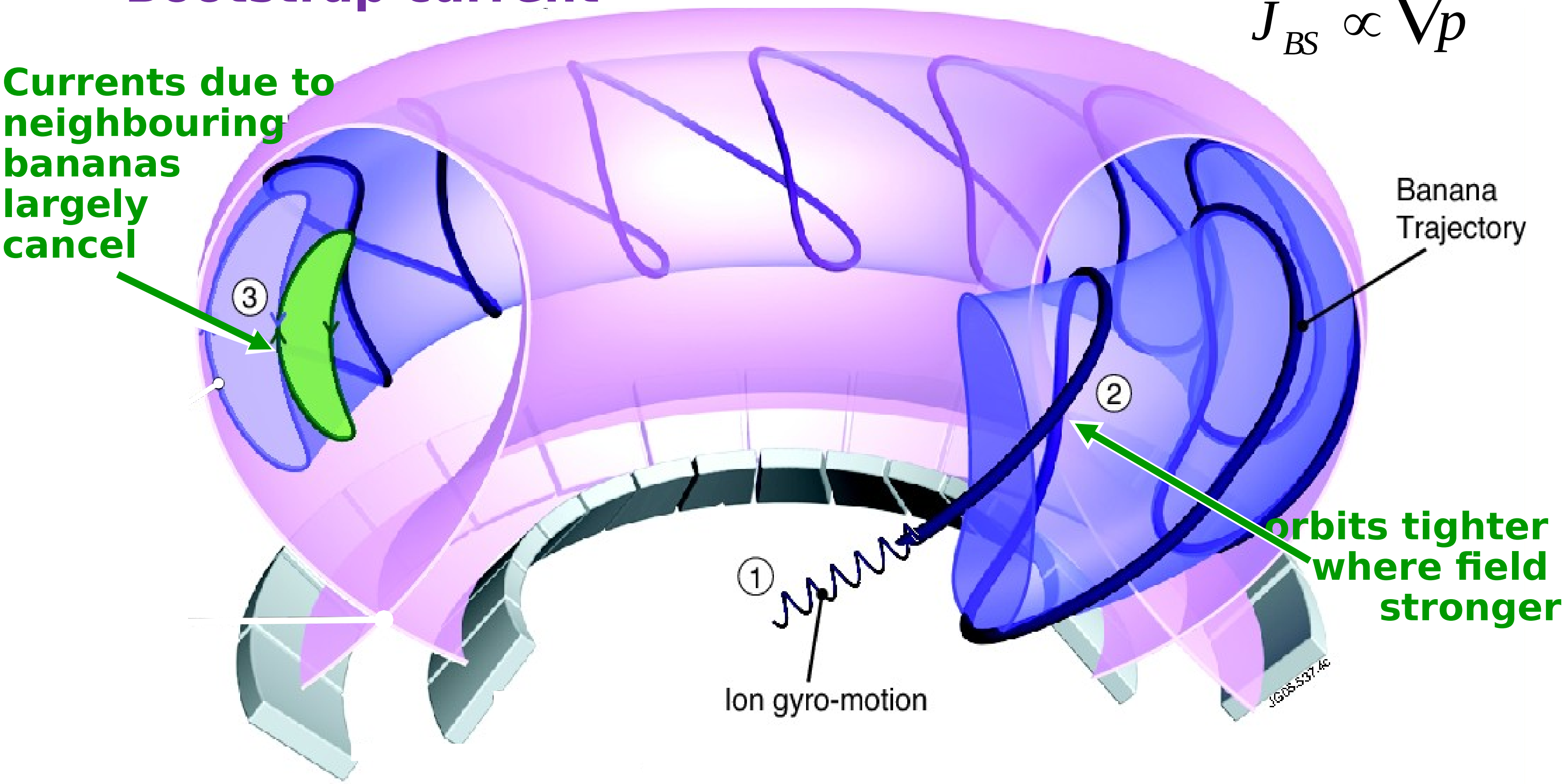


http://tfy.tkk.fi/fusion/research/

Current drive and current profile control

- **Bootstrap current**

$$J_{BS} \propto \nabla p$$



- More & faster particles on orbits nearer the core (green .VS. blue) lead to a net "banana current".
- This is transferred to a helical bootstrap current via collisions.

Current drive and current profile control

- **Bootstrap current**

110 NATURE PHYSICAL SCIENCE VOL. 229 JANUARY 25 1971

HW. How can one measure bootstrap current?
Give examples of 100% bootstrap current drive.

Diffusion Driven Plasma Currents and Bootstrap Tokamak

by

R. J. BICKERTON, J. W. CONNOR & J. B. TAYLOR

UKAEA Research Group, Culham Laboratory, Abingdon, Berkshire

In toroidal systems of plasma confinement the intrinsic diffusion driven toroidal current modifies estimates of the maximum ratio of plasma pressure to magnetic field pressure. This intrinsic current may also make possible a type of Tokamak machine which operates in a steady state, unlike present pulsed designs.

in what we call a "bootstrap" Tokamak. Such a machine could operate in a steady state, unlike present pulsed designs, because refuelling and thermonuclear reactions provide a continuous source of plasma to diffuse across the lines of force.

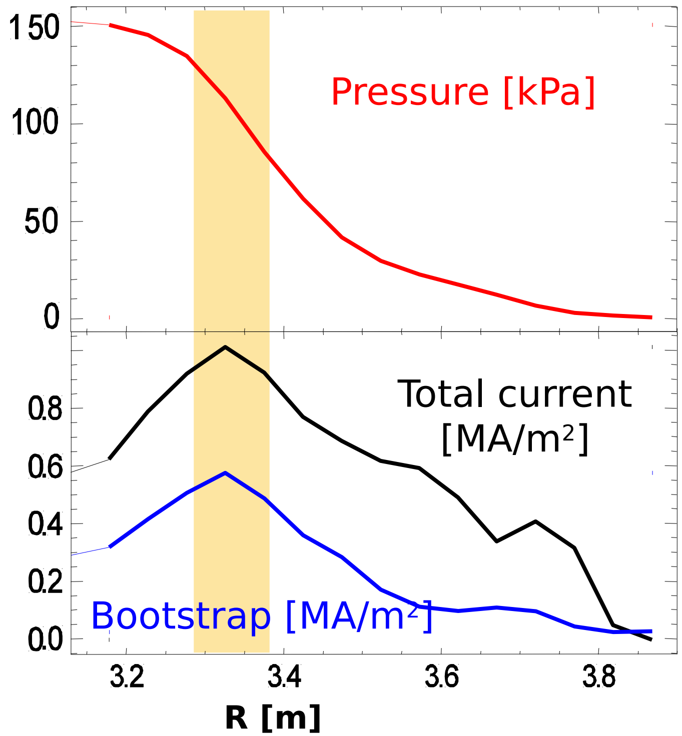
The existence of the intrinsic toroidal current is implicit in all calculations of toroidal diffusion and its value may be obtained easily from such calculations, so that only the result need be quoted here. For simplicity we consider the usual axisymmetric system with concentric magnetic surfaces $B\varphi = B_0/h$, $B_\theta = \Theta B\varphi$ with

$$\Theta = \varepsilon \frac{\iota}{2\pi} \ll 1,\ h = 1 + (r/R)\cos\theta, \text{ and } r,\ \theta,\ \varphi$$

Current drive and current profile control

- **Bootstrap current**

Pulse 53521 (I_p = 2.0 MA)



Bootstrap current is not well aligned with the ITB foot.

→ q_{min} moves to the centre.

→ ITB lost finally.

Bickerton 1971 & Galeev 1970

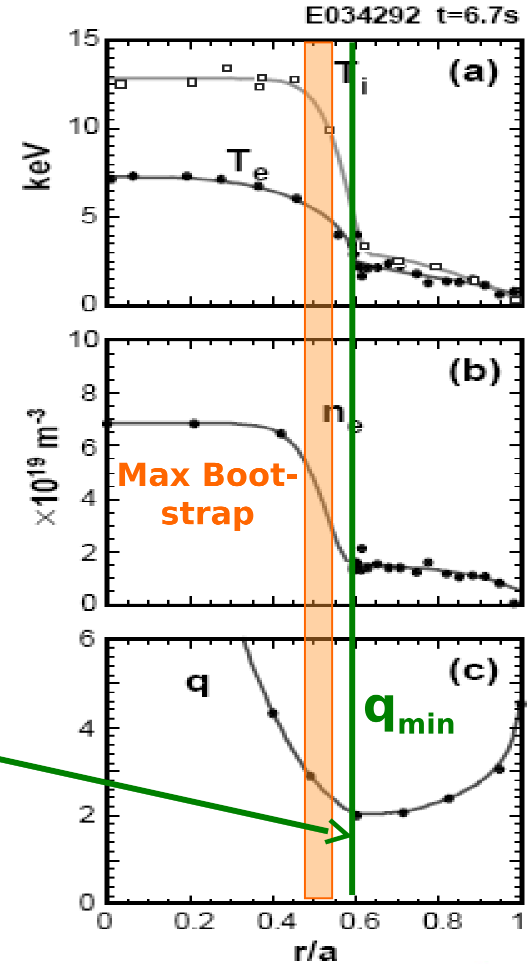
Current drive and current profile control

ITB foot location often correlated with q_{min} radius

∦

Pressure gradient and bootstrap current location

Current drive required to maintain the ITB in steady state (JT-60U)



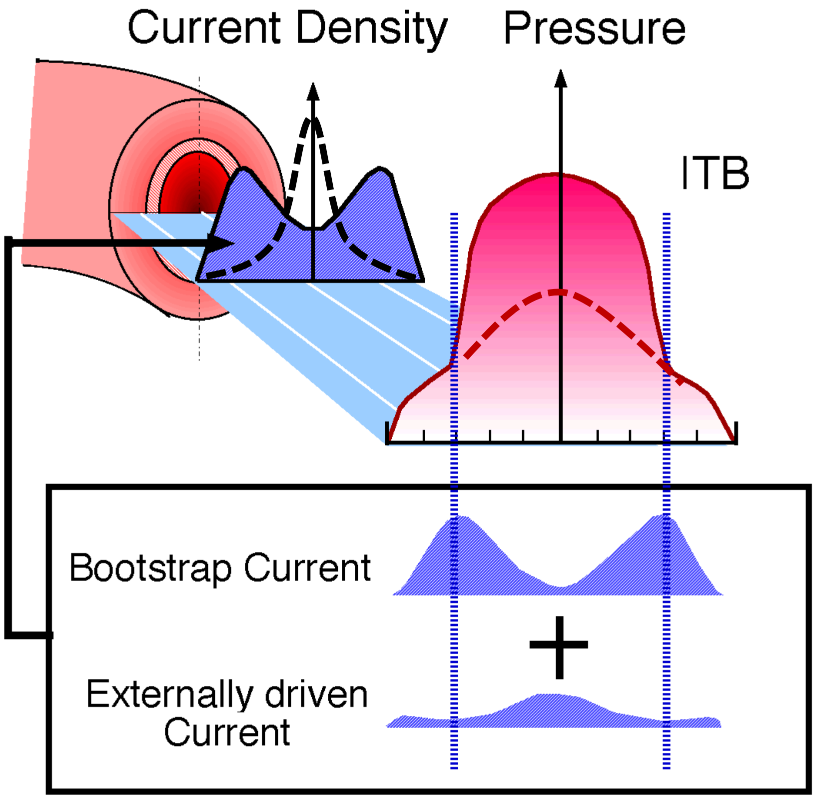
- 100% bootstrap not likely:
  need current drive at r/a~0.7 for broad ITB

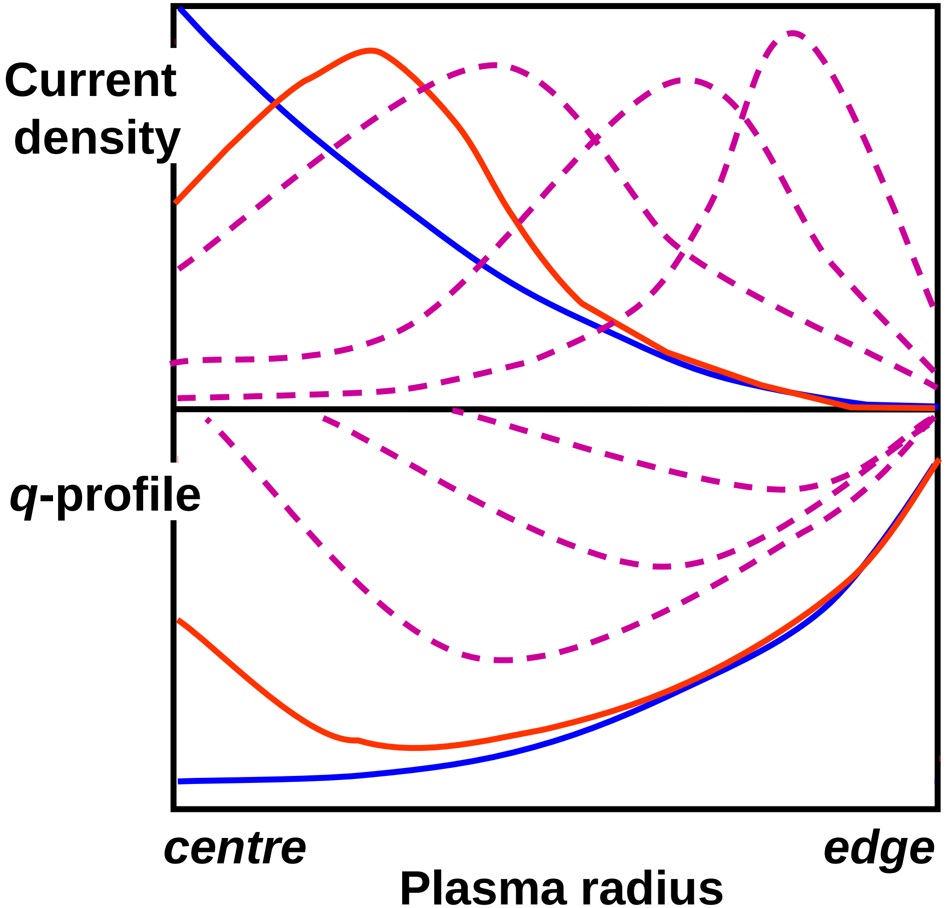
Oikawa Nuc Fus 2000, Kikuchi PPCF 2001, Fujita PPCF 2002, Coda EPS 2007

JT-60U

Sustainment of Non-monotonic Current Profile

- Plasma current diffusion into the core from the edge

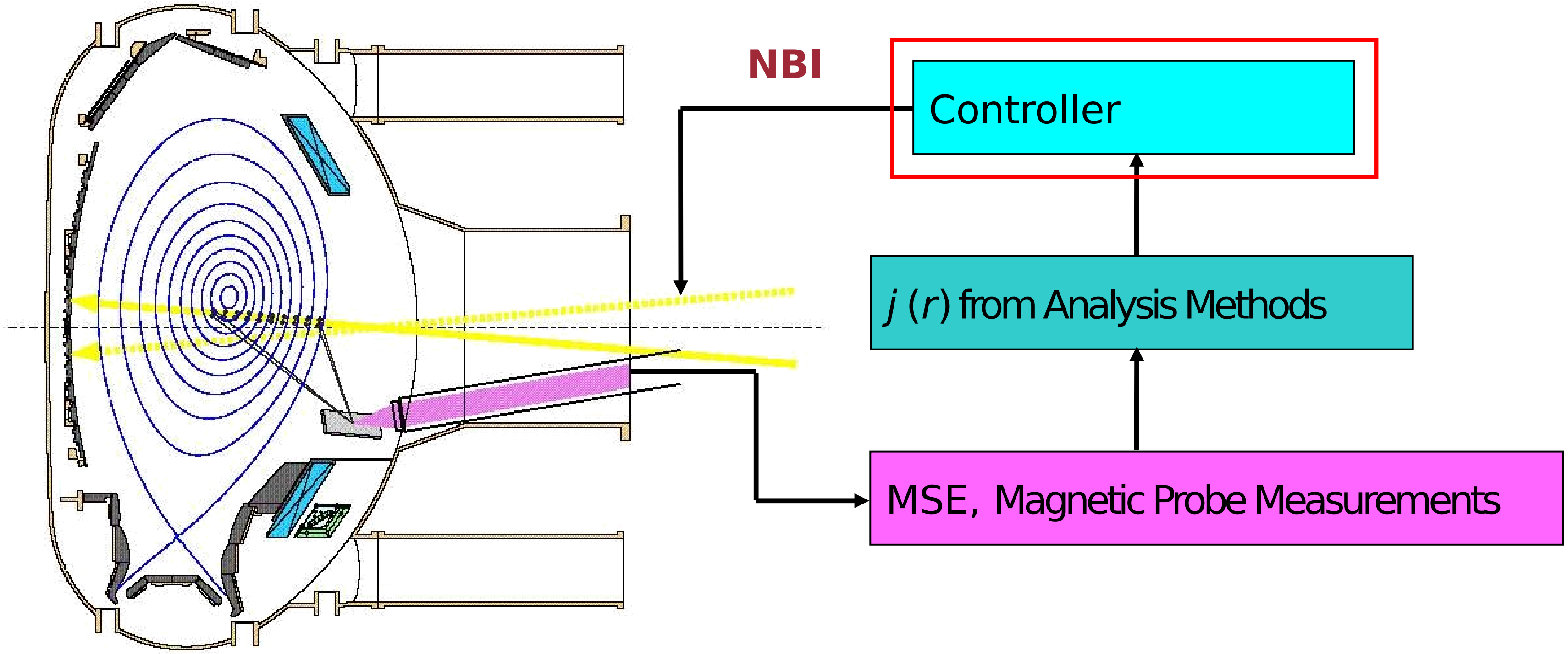




Current and pressure profile control !

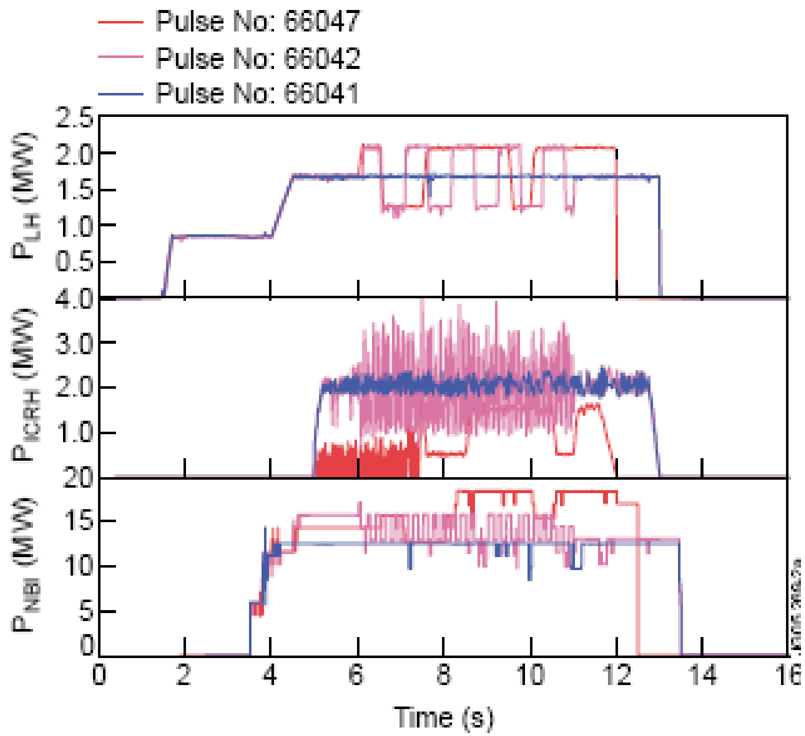
Reversed Shear Scenario

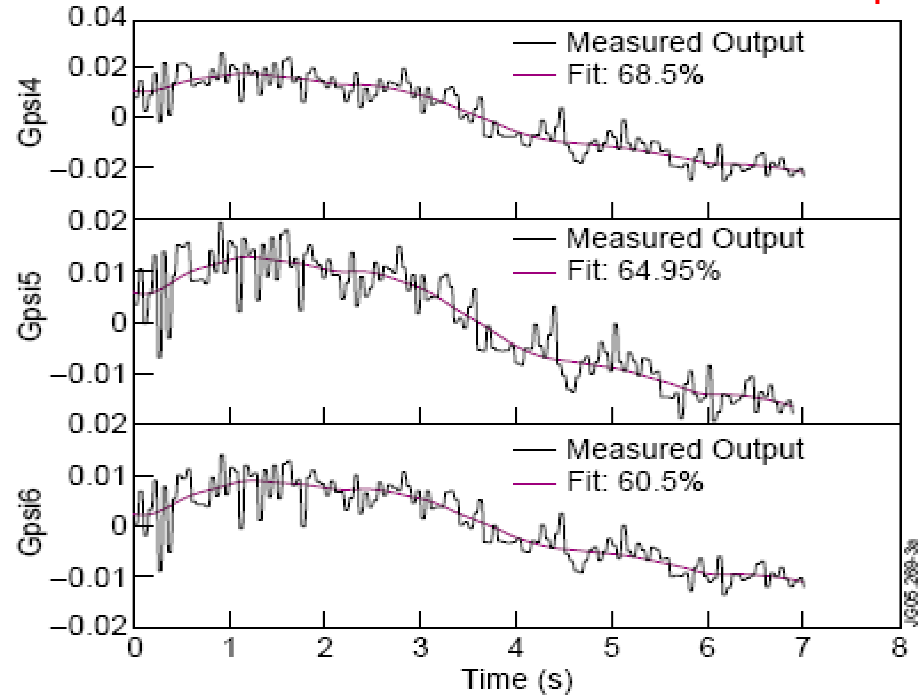
- Current density profile control at ASDEX Upgrade



RT Current and Pressure Profile Control

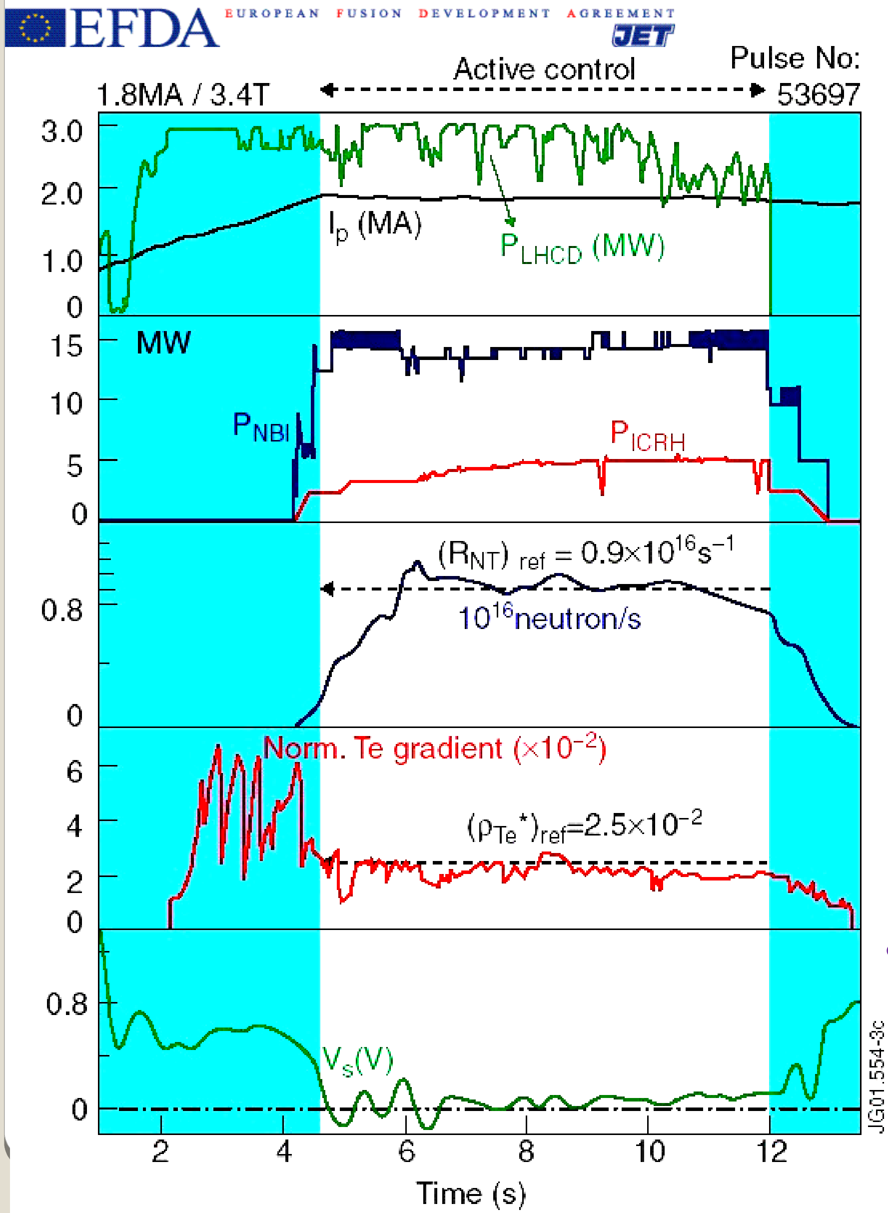
- Simultaneous control of distributed magnetic and kinetic paramters
- Dedicated experiments to identify controller coefficients

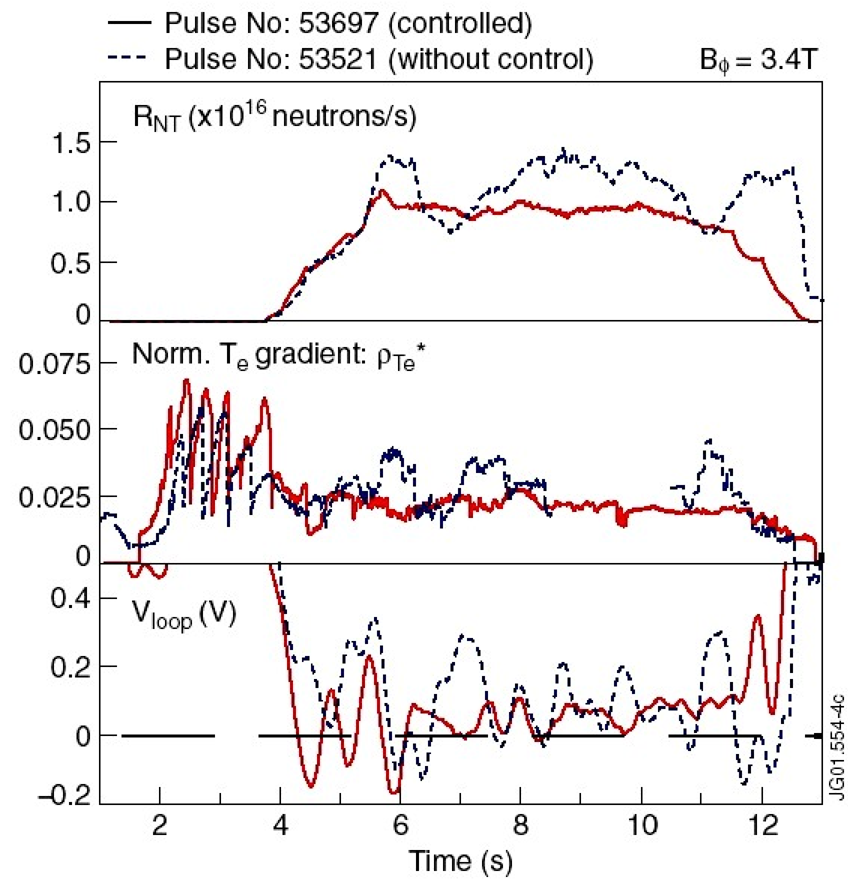




- Modulation combinations of actuators (NBI, LH, ICRH) to infer the coefficients of the state space model of the slow loop.
- Two control loops, 4 actuators (NBI, LH, ICRH, PF)

RT Current and Pressure Profile Control



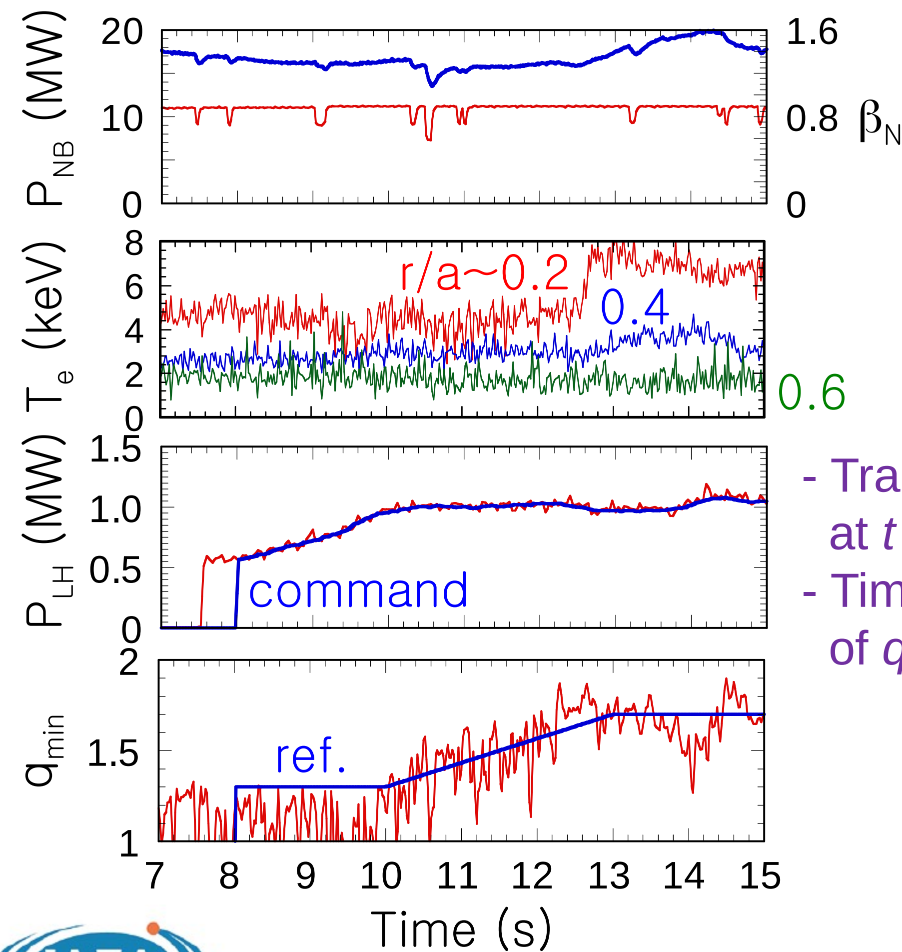


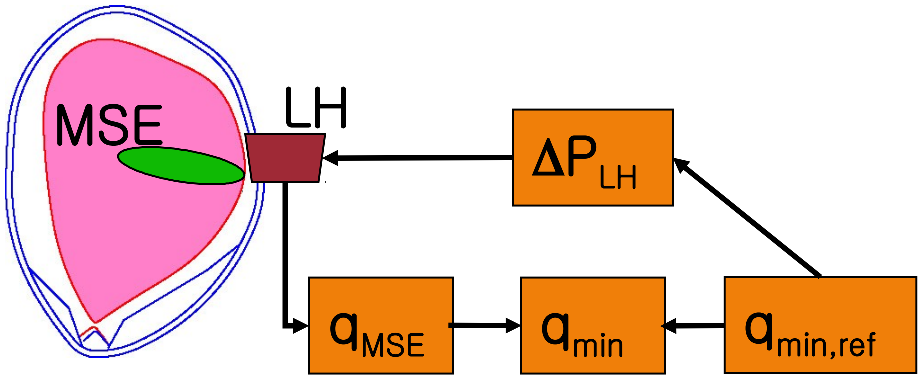
- Real time pressure profile and *q*-profile control to keep ITB steady

Mazon et al, PPCF (2002)

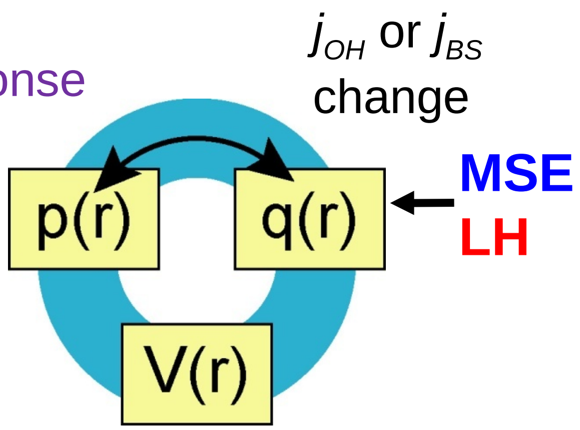
RT Current and Pressure Profile Control

- JT60-U: Real time q_{min} control with MSE diagnostics and LHCD

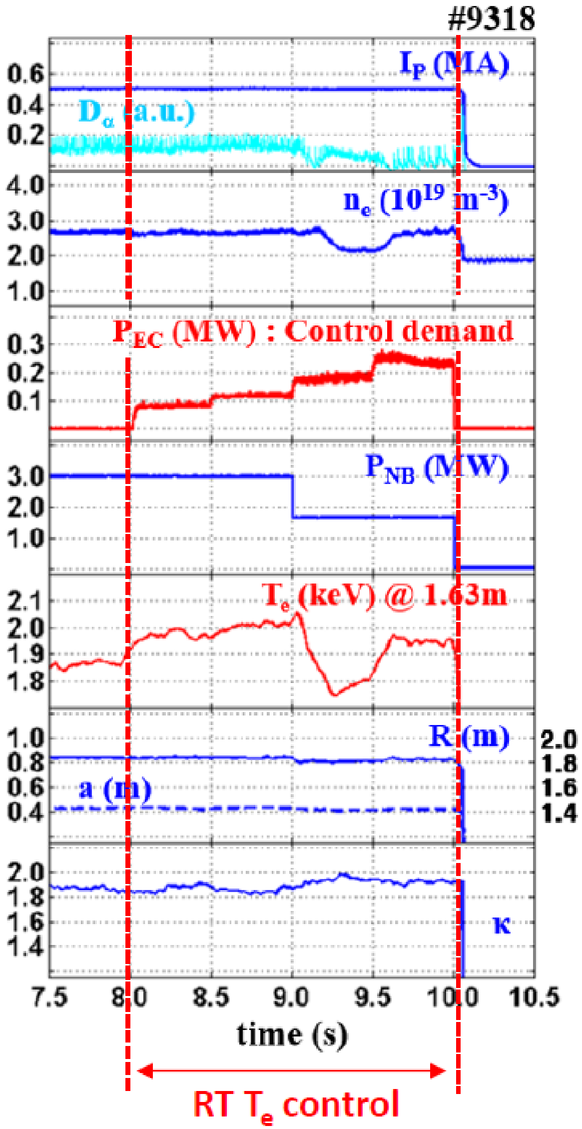


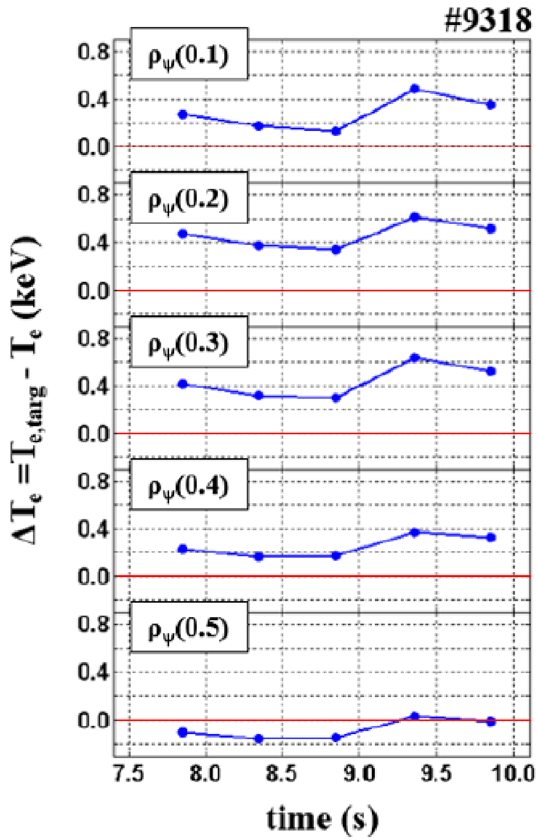


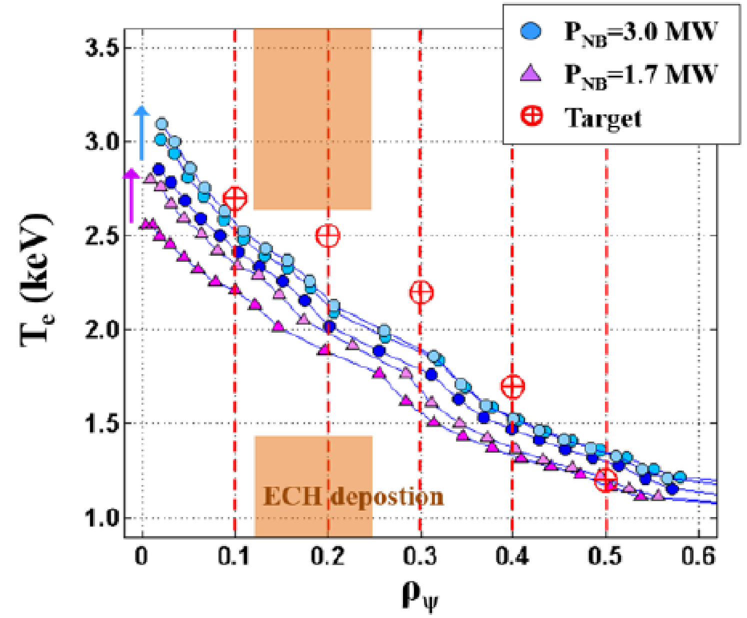
- Transport reduction at t = 12.4 s
- Time delay in response of q_{min}



Experimental result of RT T_e control by ECH (H-mode)

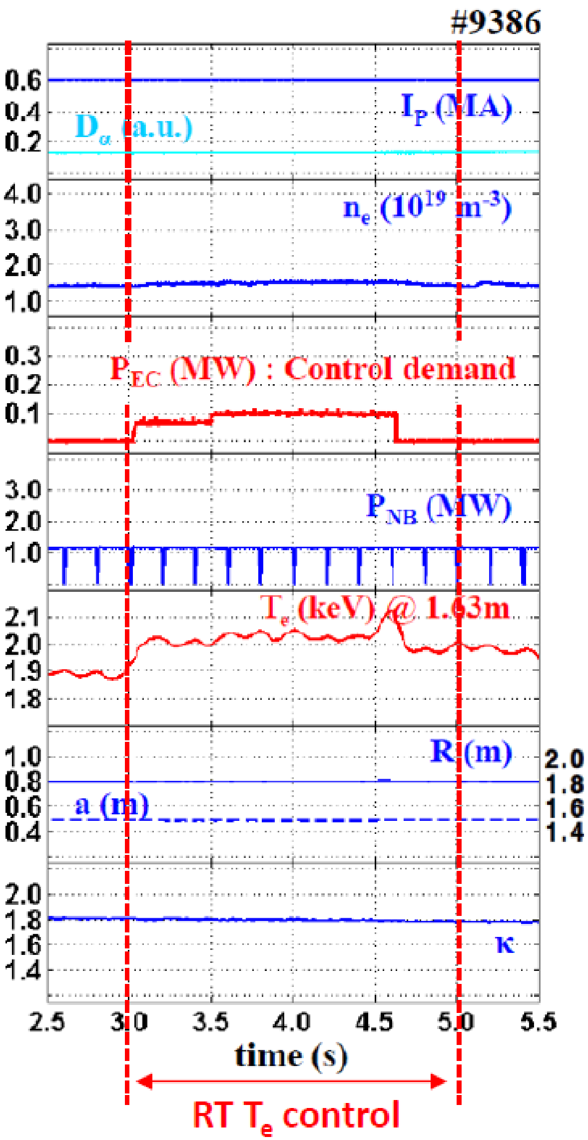


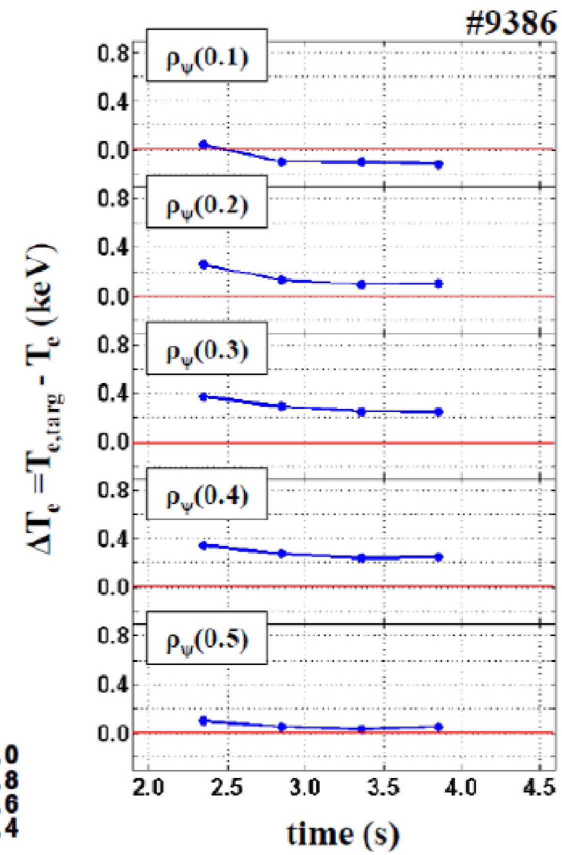


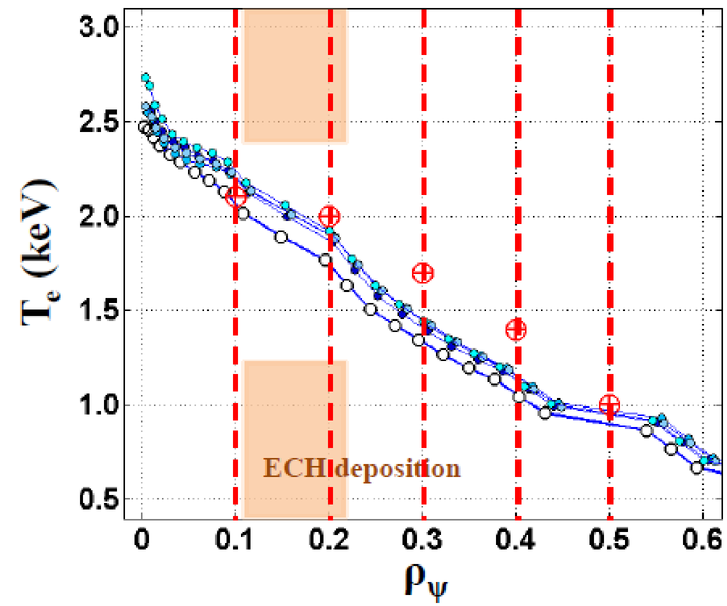


✓ In H-mode discharge, T_e profile was controllable even in the presence of transient injection of P_{NB} with the model estimation, however T_e profile could not reach to the target due to limitation of EC source.

Experimental result of RT T_e control by ECH (L-mode)



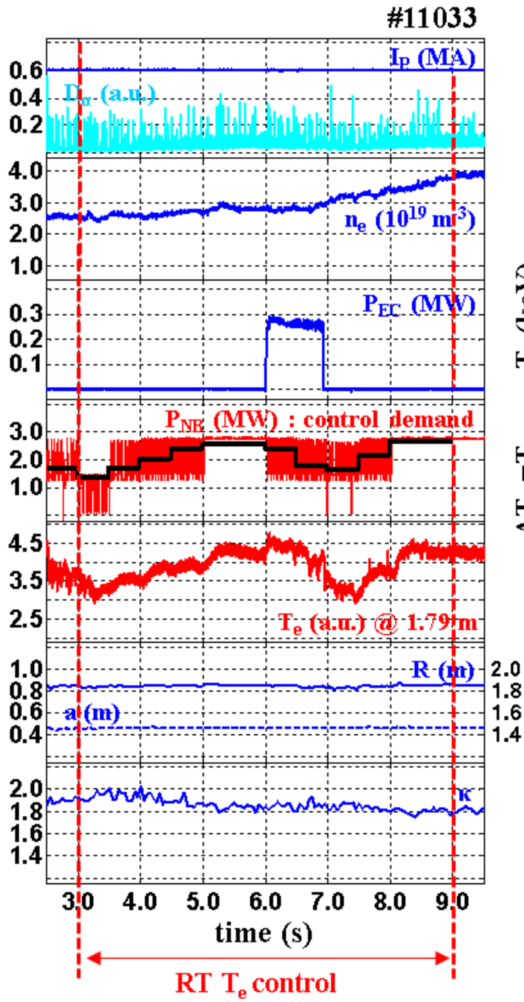


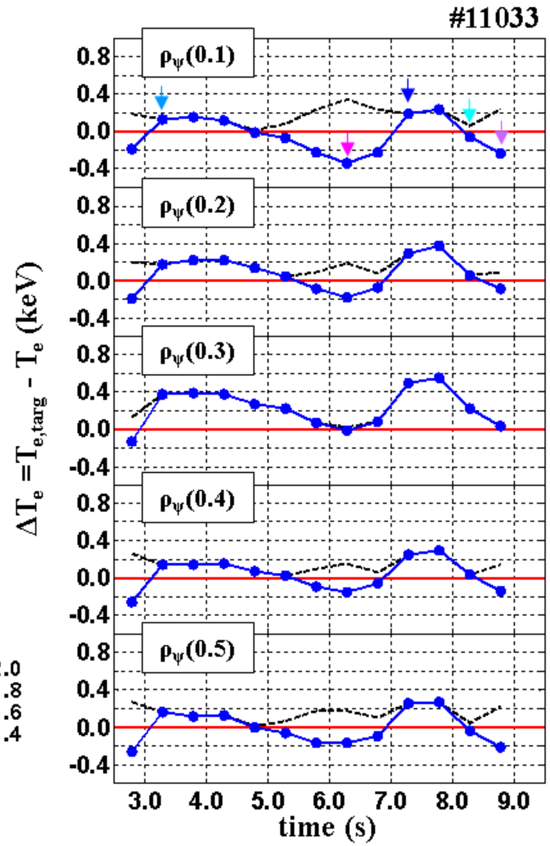


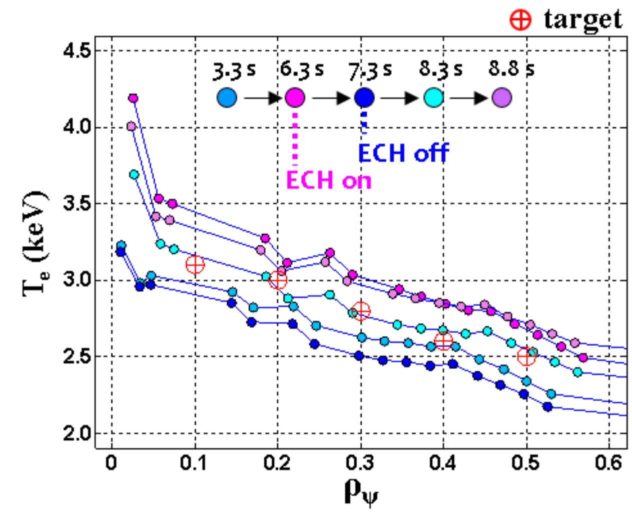
✓ In L-mode discharge, T_e profile was also controllable, however the control demand was stable right away to the 0.1 MW of P_{EC} based on the evaluation of SVD technique.

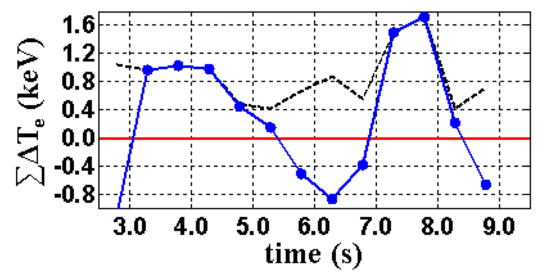
Experimental result of RT T_e control by NBI (H-mode)

➢ **Experimental result of real time T_e profile control (actuator: NBI)**









➢ **In H-mode discharge, T_e profile was controllable by modulation of NBI power even in the presence of sudden injection of P_{EC} with the static system matrix estimation.**

Hyun-Seok Kim et al, IAEA FEC 2016

References

- *A. C. C. Sips*, Seminars