Max-Planck-Institut für Plasmaphysik

Operation limits and MHD instabilities in modern tokamaks and in ITER

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Outline of the talk

- Tokamak equilibrium and drives for instabilities
- Operation space of tokamaks
  - Limits and possible extensions
- Performance limiting MHD instabilities
  - In conventional scenario
    - Sawteeth
    - Neoclassical Tearing Mode (NTM)
  - In advanced scenario
    - Resistive wall mode (RWM)
  - Edge Localized Mode (ELM)
  - Fast particle instabilities
  - Disruption
- Summary

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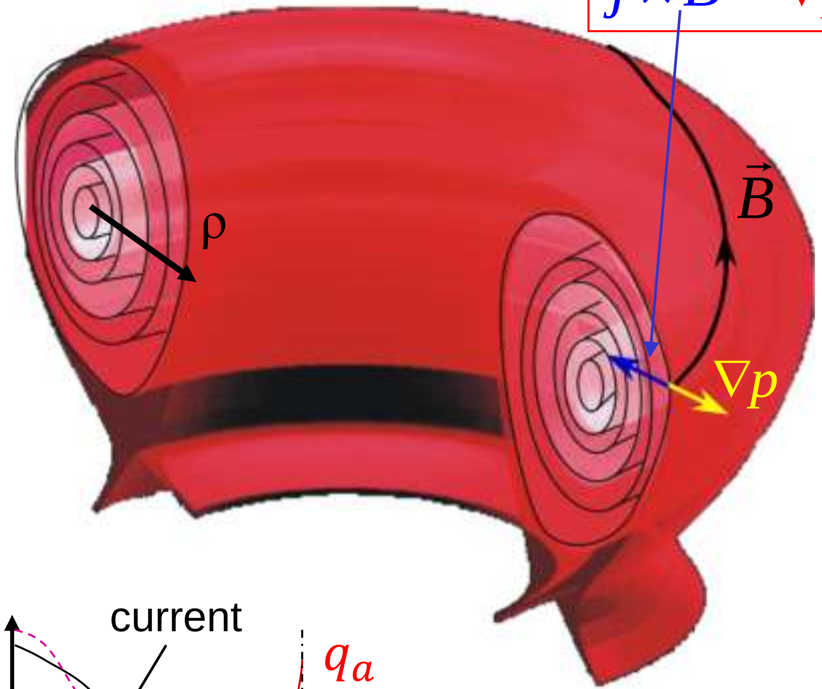
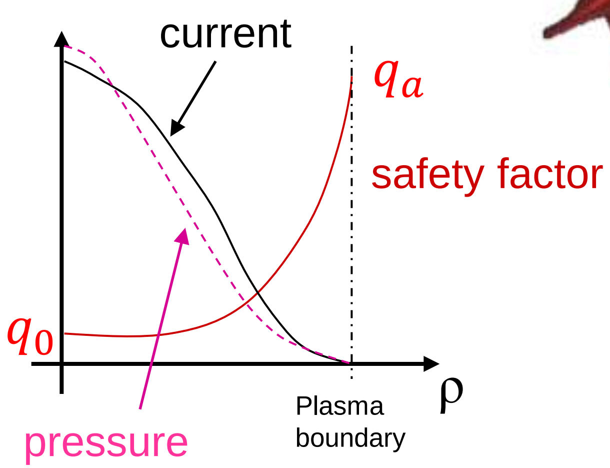
Equilibrium in tokamak

$$\vec{j} \times \vec{B} = \nabla p$$





Equilibrium in tokamak means:

the force balance between pressure gradient and magnetic force in each point inside the plasma.

$$q = \frac{poloidal\_rotation\_angle}{toroidal\_rotation\_angle}$$

$$\beta_t \equiv \frac{\langle p \rangle_{volume}}{B_t^2 / 2\mu_0} = \frac{kinetic\_plasma\_pressure}{magnetic\_field\_pressure}$$

$$\beta_N = \frac{\beta_t a B_t}{I_p}$$

normalized beta

(allows to compare different tokamaks!)

Stability condition in plasma

If the force is unbalanced: $\vec{F} = \vec{j} \times \vec{B} - \nabla p \neq 0$

magnetic force ($\vec{j} \times \vec{B}$)

plasma pressure force (∇p)

One can calculate energy changes for a given displacement $\vec{\xi}$

$$\delta W = \frac{1}{2}\int \vec{\xi} \cdot \vec{F} d\tau$$

Linearization:
$A = A_0 + A_1$
0 – equilibrium
1 - perturbation

$$\delta W = \frac{1}{2}\int_{plasma} \left(\underbrace{\gamma p_0 \left(\nabla \cdot \vec{\xi}\right)^2}_{>0} + \left(\vec{\xi} \cdot \nabla p_0\right) \nabla \cdot \vec{\xi} + \underbrace{\frac{B_1^2}{\mu_0}}_{>0} - \vec{j}_0 \cdot \left(\vec{B}_1 \times \vec{\xi}\right) \right) d\tau + \underbrace{\int_{vacuum} \frac{B_{vac}^2}{2\mu_0} d\tau}_{>0}$$

$$\boxed{\text{Unstable only if} \quad \delta W < 0}$$

Wesson, Tokamaks, 3rd Edition
Freidberg, Ideal MHD

If the force is unbalanced:

$$\vec{F} = \underbrace{\vec{j}\times\vec{B}}_{\text{magnetic force}} - \underbrace{\nabla p}_{\text{plasma pressure force}} \neq 0$$

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Always positive (stable)!

Unstable only if $\delta W < 0$

Wesson, Tokamaks, 3rd Edition
Freidberg, Ideal MHD

Drives for instabilities in MHD are current and pressure profile gradients

Linearization:
$A = A_0 + A_1$
0 – equilibrium
1 - perturbation

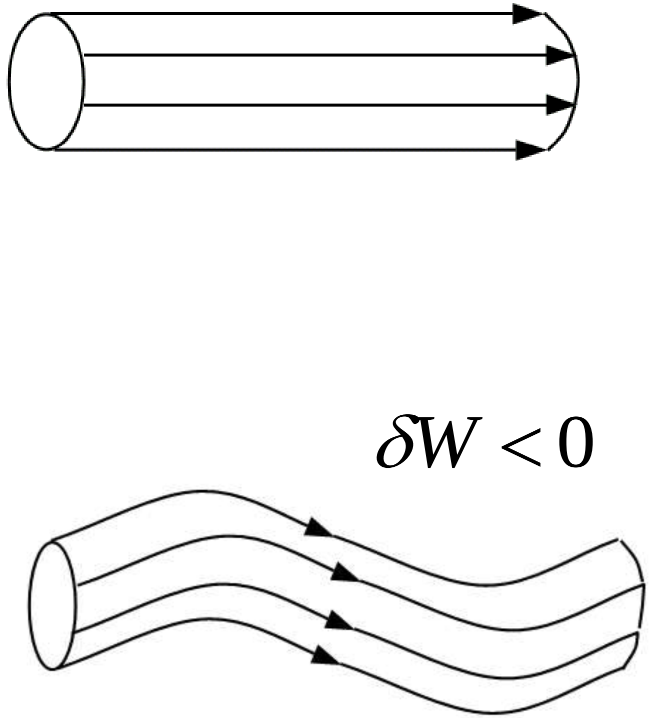
$$\delta W = \frac{1}{2}\int_{plasma}\left(\underbrace{\gamma p_0\left(\nabla\cdot\vec{\xi}\right)^2}_{>0} + \left(\vec{\xi}\cdot\nabla p_0\right)\nabla\cdot\vec{\xi} + \underbrace{\frac{B_1^2}{\mu_0}}_{>0} - \vec{j}_0\cdot\left(\vec{B}_1\times\vec{\xi}\right)\right)d\tau + \underbrace{\int_{vacuum}\frac{B_{vac}^2}{2\mu_0}d\tau}_{>0}$$

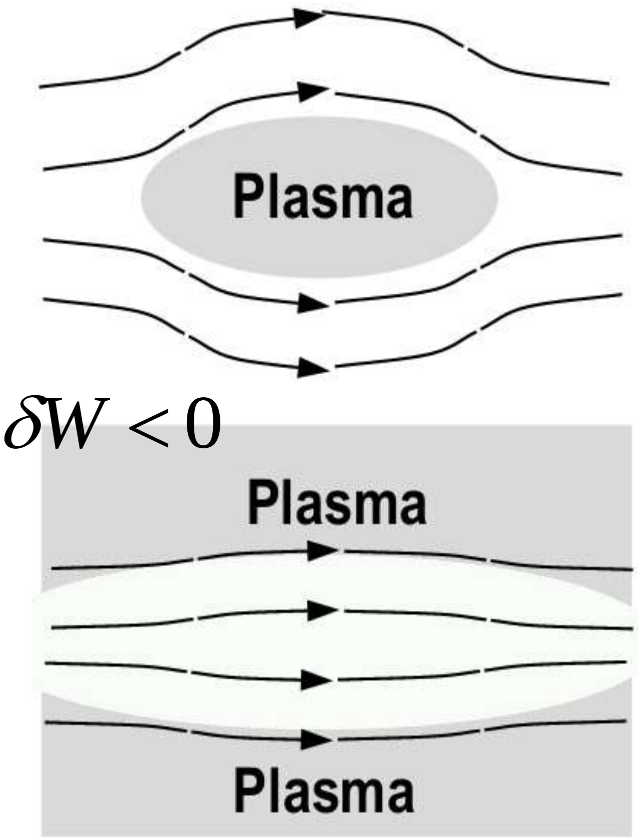
Pressure driven instabilities

Current driven instabilities

$\boxed{\text{Unstable only if } \delta W < 0}$

Wesson, Tokamaks, 3rd Edition
Freidberg, Ideal MHD





current driven instabilities

(kink mode)

pressure driven instabilities

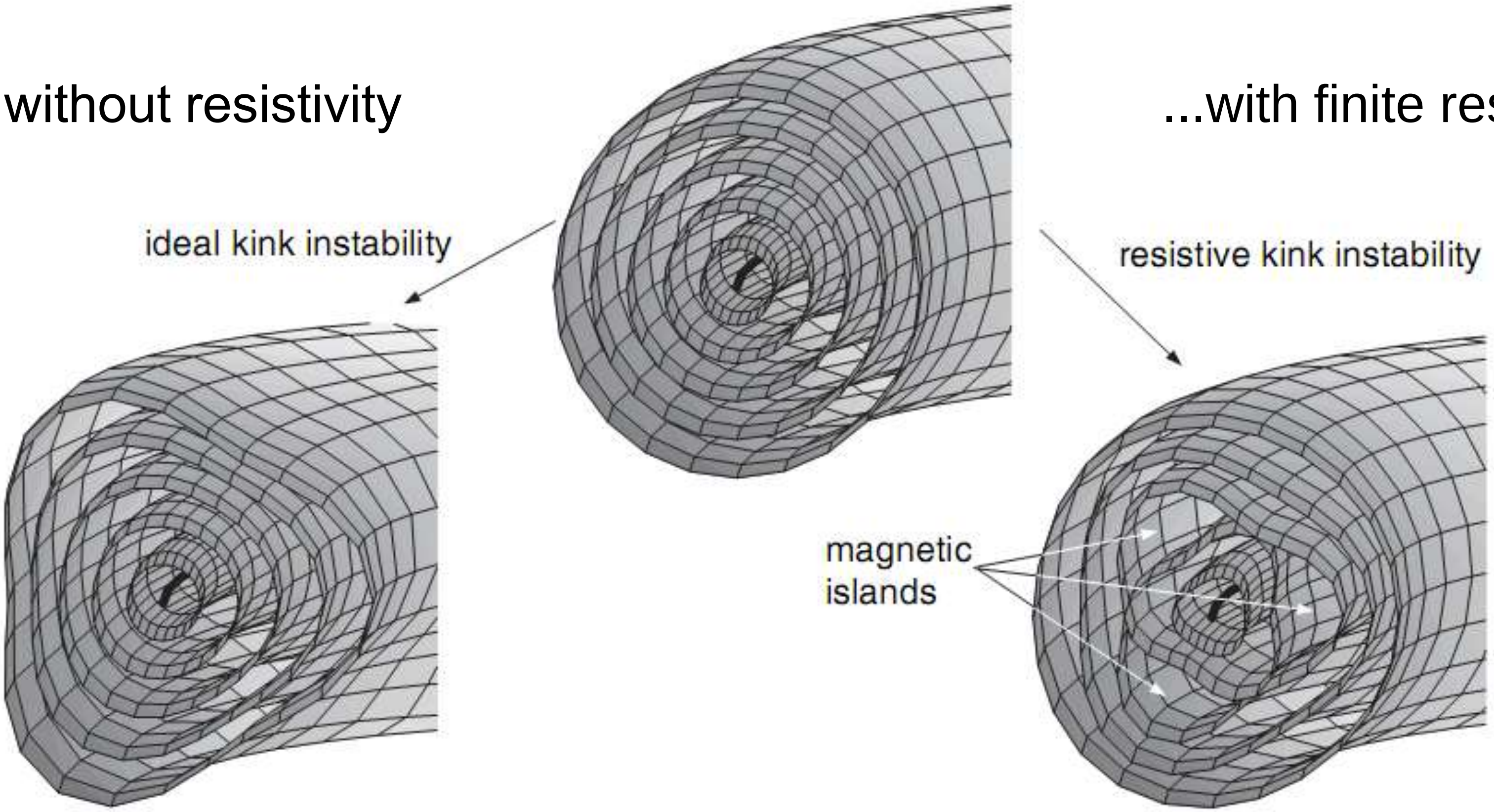
(interchange mode)

Resistivity could be important as well!

MHD instabilities can develop at the rational surfaces

…without resistivity

...with finite resistivity



Zohm,
Scripts

IDEAL

RESISTIVE

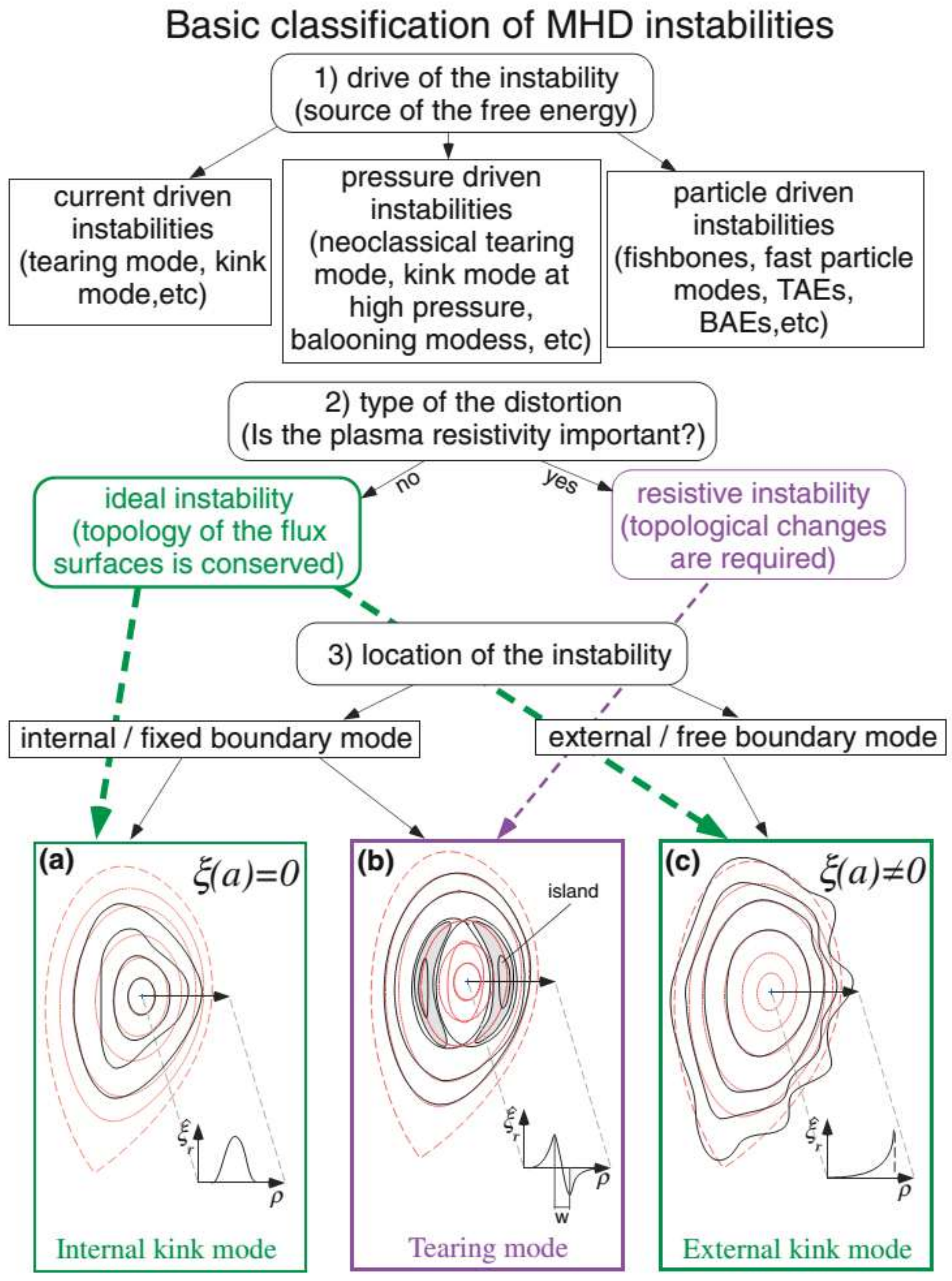
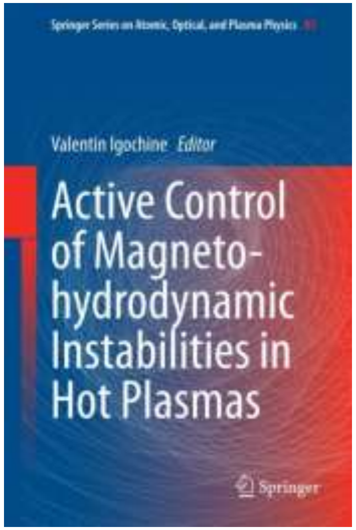


Fig. 2.8 The basic classification of MHD instabilities

from this book ☺



V. Igochine, “Active Control of Magneto-hydrodynamic Instabilities in Hot Plasmas”, Springer Series on Atomic, Optical, and Plasma Physics, Vol. **83**, 2015 (Chapter 2)

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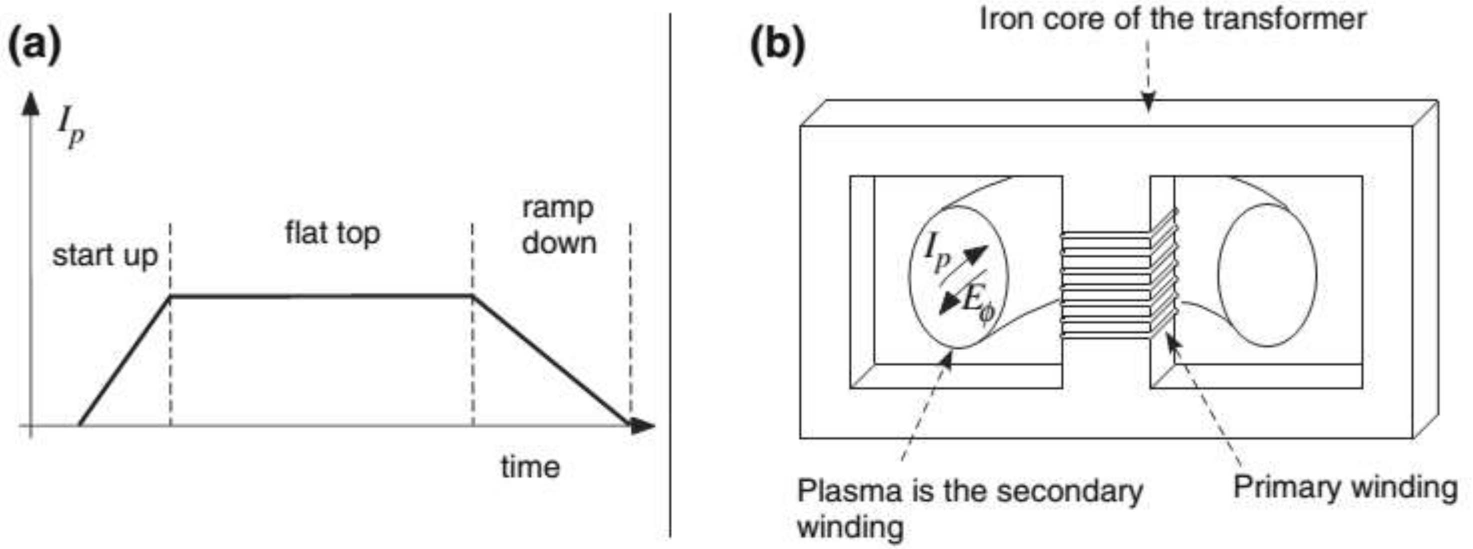
How we operate tokamak?

Plasma current time trace and different phases of the discharge

[V. Igochine, “Active Control of Magneto-hydrodynamic Instabilities in Hot Plasmas”, Springer 2015,(Chapter 2)]

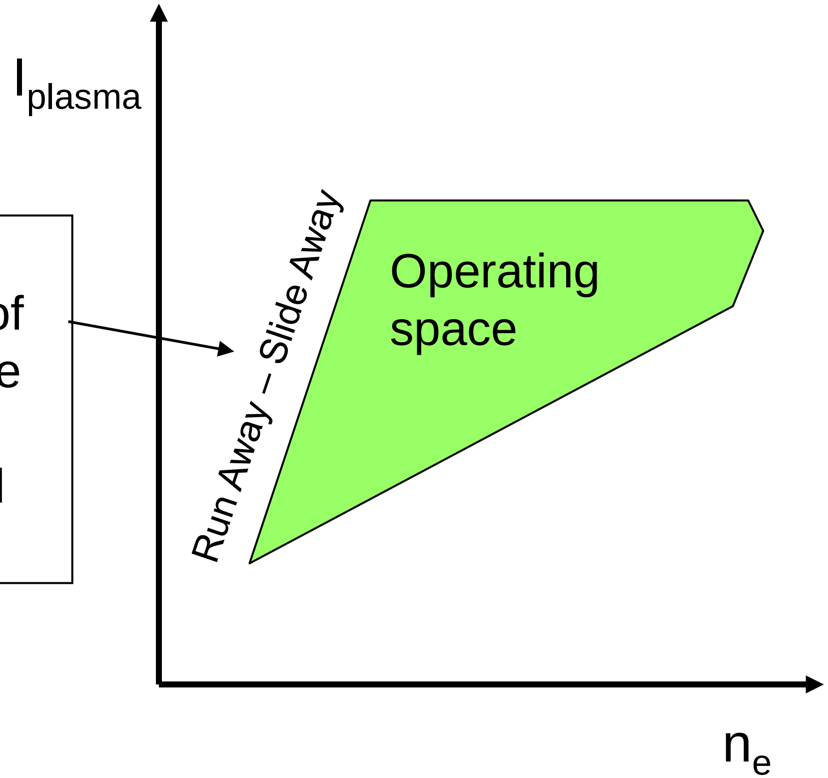


<u>Tokamak startup:</u> Plasma acts as the secondary winding of the transformer. External voltage is applied on the primary winding of the transformer. → toroidal electric field → acceleration of electrons → avalanche process → first toroidal plasma current → ohmic heating → high temperature (ohmic heating is not effective any more) with external heating → current ramp down

Two natural values for control: plasma current and density

Operation space of the tokamak (Hugill Diagram)

Here density is too small. Part of the electrons are continuously accelerated and lost



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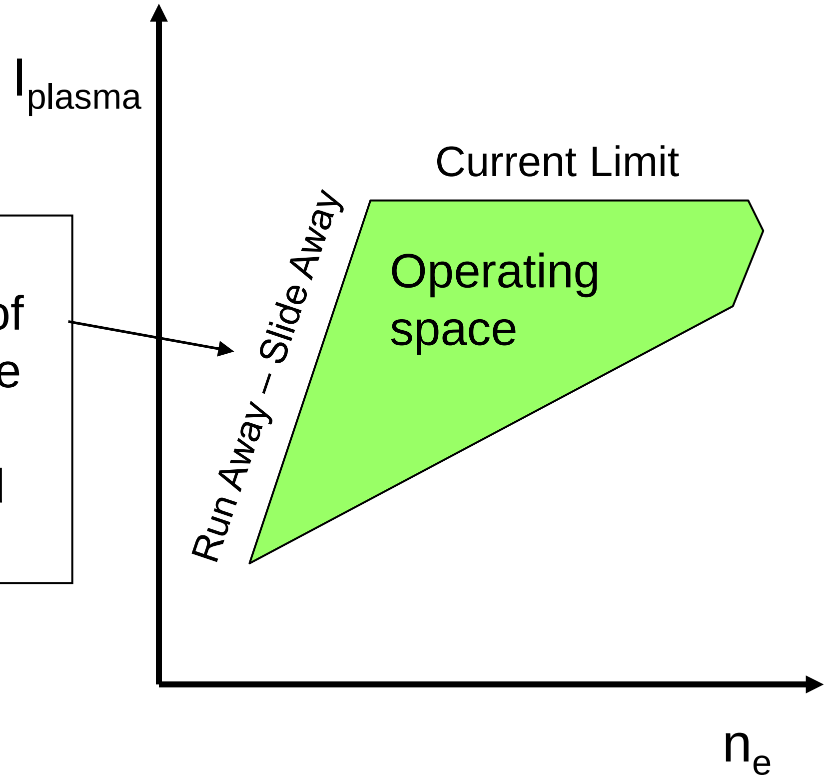


Strong currents lead to unstable situation with respect to kinks

inside plasma, internal kinks

$$\delta W = \frac{2\pi^2 B_z^2}{\mu_0 R_0}\int_0^a [(r\xi')^2 + (m^2-1)\xi^2]\left(\frac{n}{m}-\frac{1}{q}\right)^2 r\,dr$$
$$+\frac{2\pi^2 B_z^2}{\mu_0 R_0}\left(\frac{2}{q_a}\left(\frac{n}{m}-\frac{1}{q_a}\right)+(1+m)\left(\frac{n}{m}-\frac{1}{q_a}\right)^2\right)a^2\xi_a^2$$

Outside, external kinks

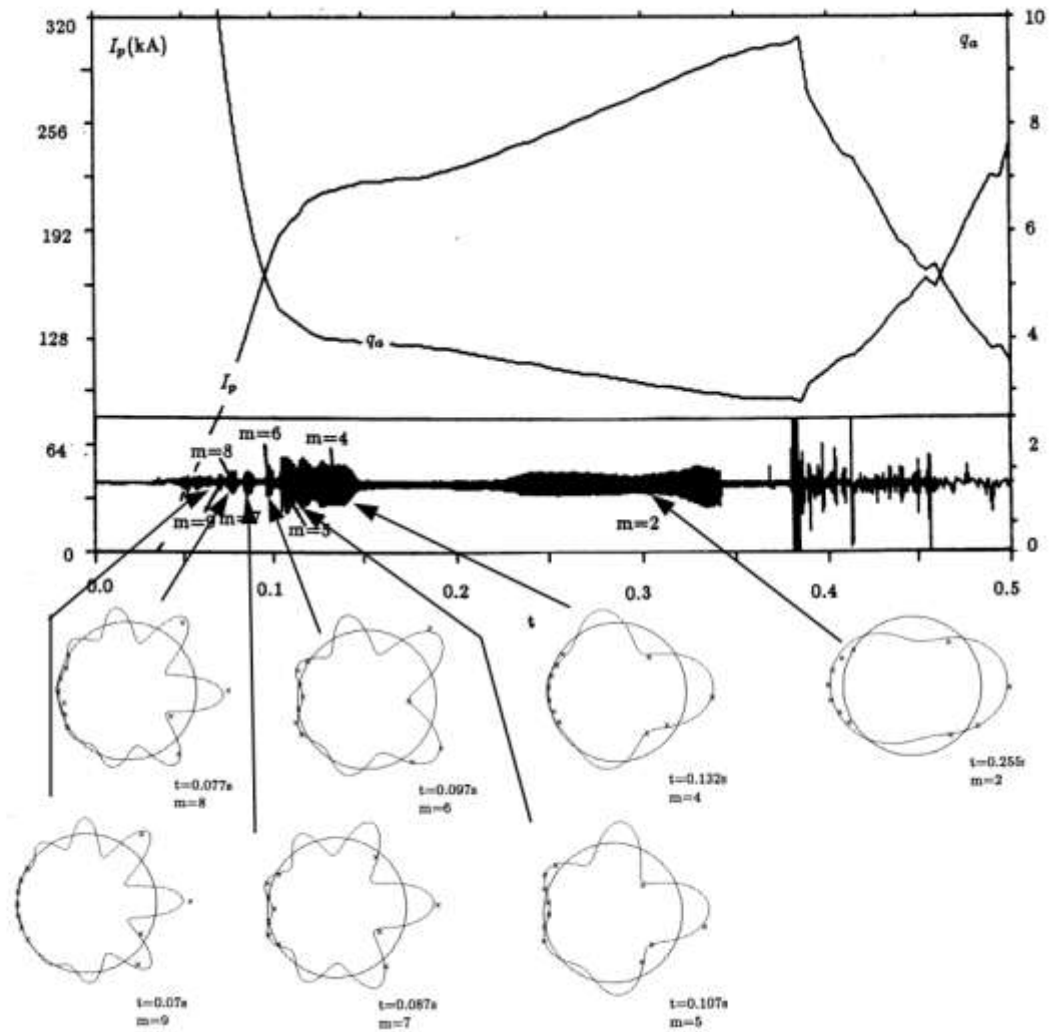
Determine stability

Always positive

Thus, unstable external kinks are possible if $\boxed{q_a > \frac{m}{n}}$

Resonant surface is close to the plasma boundary but is outside the plasma

Strong currents lead to unstable situation with respect to kinks



These kinks can be seen during plasma ramp up (q_a drops down)

But real limits poses q=1 case …

Assume $\xi = const$ and $m = 1$

0

inside plasma, internal kinks

$$\delta W = \cancel{\frac{2\pi^2 B_z^2}{\mu_0 R_0} \int_0^a [(r\xi')^2 + (m^2-1)\xi^2] \left(\frac{n}{m} - \frac{1}{q}\right)^2 r\,dr}$$

no dependence from current profile!

$$+ \frac{2\pi^2 B_z^2}{\mu_0 R_0} \left(\frac{2}{q_a} \left(\frac{n}{m} - \frac{1}{q_a} \right) + (1+m) \left(\frac{n}{m} - \frac{1}{q_a} \right)^2 \right) a^2 \xi_a^2$$

Outside, external kinks

$$\delta W = \frac{4\pi^2 B_z^2}{\mu_0 R_0} n \left(n - \frac{1}{q_a} \right) a^2 \xi_a^2$$

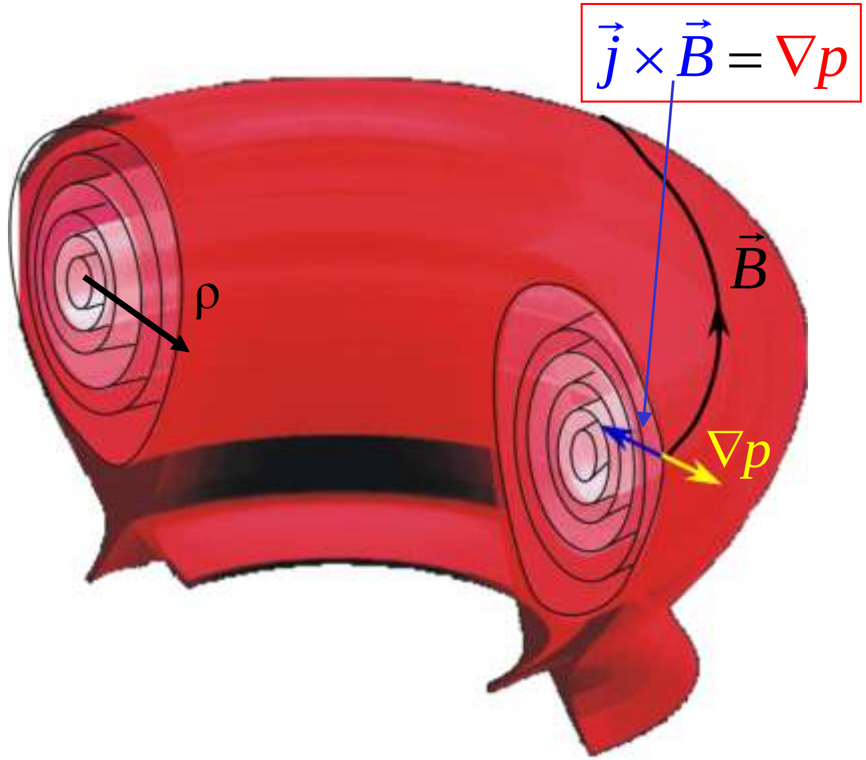
Result requirements for stability

The most restrictive for n=1

$q_a < 1/n$

$\boxed{q_a \geq 1}$

Kruskal-Shafranov limit and pressure limit



We know from experiment that for $q_a<2 \rightarrow$ kink instability!

Normal operations has

$$q_a = \frac{2\pi a^2 B_{tor}}{\mu_0 I_{plasma} R} \geq 3$$

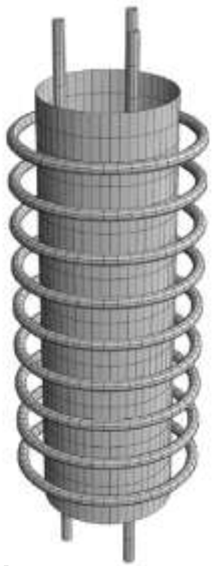
This gives limit for maximal plasma current and thus the force which confine the plasma. Force comes only from toroidal current (poloidal magnetic field).

$$B_{tor} \approx 10 B_{pol}$$

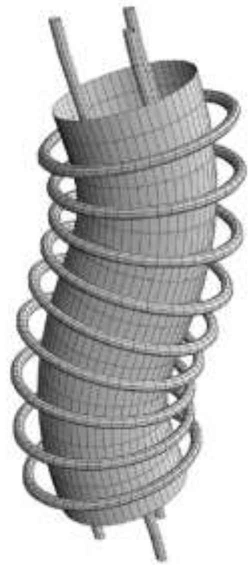
$$\beta_t = \frac{\langle p \rangle}{B_{tor}\hat{}2/2\mu_0} < 10\%$$

Strong currents lead to unstable situation with respect to kinks

unperturbed



perturbed



($B_{tor} >> B_{pol}$)

<u>Suydam´s criterion</u> (cylindrical tokamak) is a test against a localized perturbation around rational surface q=m/n

$$\frac{8\mu_o r p'}{B_z^2} + \left(\frac{r q'}{q}\right)^2 > 0$$

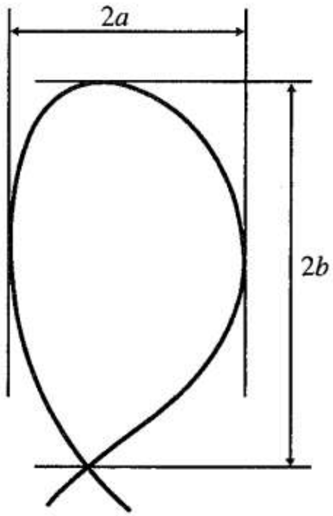
balance between the destabilizing effect of an unfavorable pressure gradient / field-line curvature combination and the stabilizing effect of shear in the magnetic field.

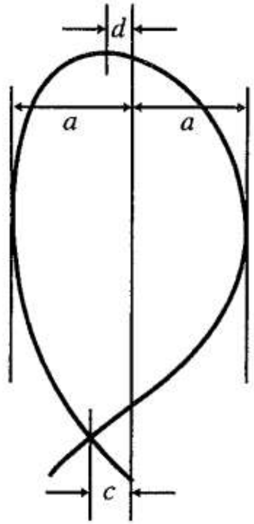
<u>Mercier´s criterion</u> (toroidal geometry, circular cross-section, interchange modes)

$$\frac{8\mu_o r p'}{B_z^2} \boxed{\left(1 - q^2\right)} + \left(\frac{r q'}{q}\right)^2 > 0$$

- changing sign of the pressure gradient for q>1
- around minimum q shear stabilization is ineffective thus, requires q(0)>1 in standard case

Current limit, realistic tokamak configuration





$$\kappa = b/a \qquad \delta = \frac{(c+d)/2}{a}$$

$$q_o \geq 1 \Big/ \sqrt{1 - \frac{4}{1+3\kappa^2}\left(\frac{3}{4}\frac{\kappa^2-1}{\kappa^2+1}\left(\kappa^2 - \frac{2\delta}{r/R_o}\right) + \frac{(\kappa-1)^2 \beta_{pol}(0)}{\kappa(\kappa+1)}\right)}$$

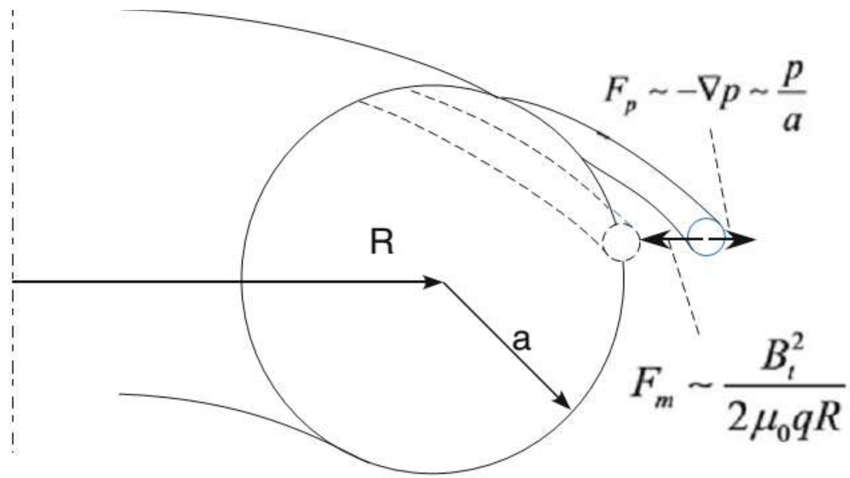
Finite plasma pressure $\beta_{pol}(0)$ and plasma elongation κ are destabilizing,

…but a sufficient triangularity $r\delta/R_o \geq \kappa^2/2$ can turn the effect of elongation into a stabilizing one. Together with the fact that elongation allows a higher plasma current at given aspect ratio, this motivates from the physics side the D-shaped cross-section of all modern tokamak designs. (This allows also divertor configuration!)

D-shape allows higher current but after plasma shape in tokamak is fixed, no extension of the current limit is possible.

$$q_a = \frac{2\pi a^2 B_Z}{\mu_0 I_p R} \quad (\text{``straight tokamak'' model})$$

$$\beta = \frac{\langle p \rangle}{\langle B^2 \rangle / 2\mu_0}$$



$$\delta W_m = \int \vec{F}_m \cdot \vec{\xi} dV \sim \frac{B_\phi^2}{qR} \xi.$$

$$\delta W_p = \int \vec{F}_p \cdot \vec{\xi} dV \sim \frac{p}{a} \xi$$

$$\delta W_m \sim \delta W_p \rightarrow \frac{B_\phi^2}{qR} \sim \frac{p}{a}$$

Fig. 2.18 A flux tube with length $L \sim qR$ is shifted outwards by plasma pressure. The situation is stable only if the magnetic force is sufficient to counteract the pressure force

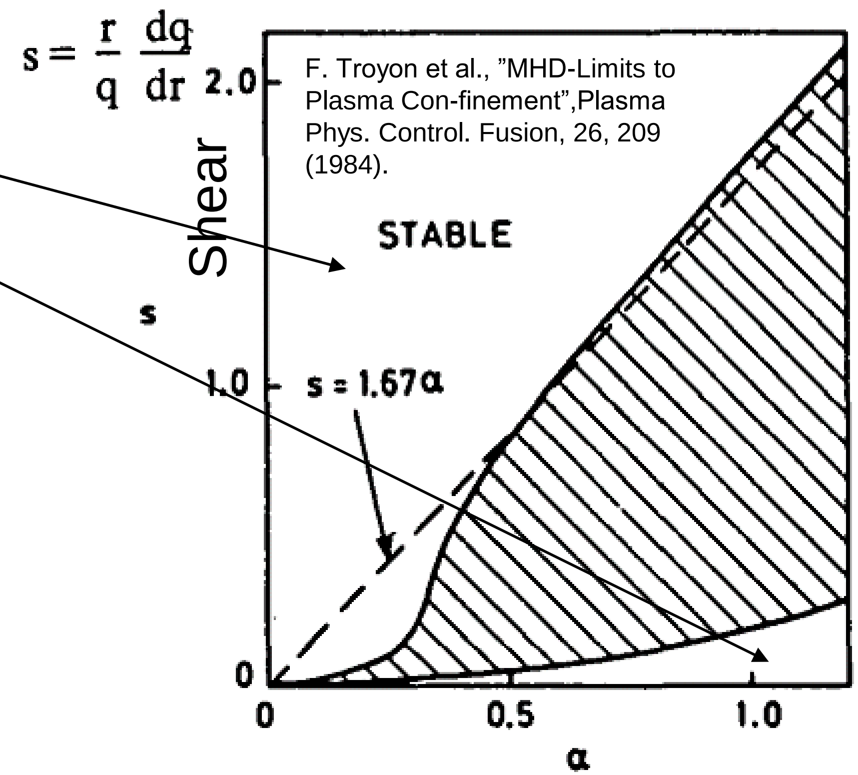
$$\beta_{crit} \sim \frac{I_p}{B_\phi a}$$

This value is different for different tokamaks, but …

This value is a general measure which depends on plasma profiles, plasma shape and other factors and can be compared between tokamaks! (This value is typically around 3%)

$$\beta_{crit} = \beta_N \frac{I_p}{B_\phi a}$$

There are two stability regions which could merge in one



Pressure gradient

$$\alpha = -\frac{2\mu_0 R q^2}{B^2}\frac{dp}{dr}$$

Beta limit is defined by:

- pressure profile (balooning modes)
- current profile (kink modes)
- wall (stabilization of low-n modes)

and should be calculated numerically for particular tokamak taking into account all these ingredients!

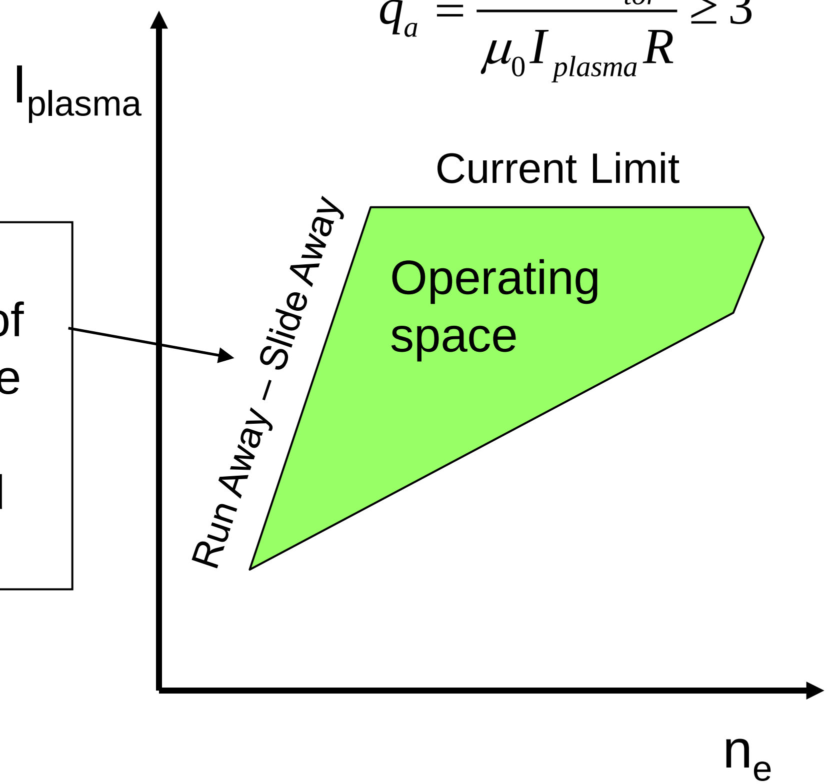
But: experimental beta limit is often determined by NTM or RWM onset at even lower plasma parameters!

Operation space of the tokamak

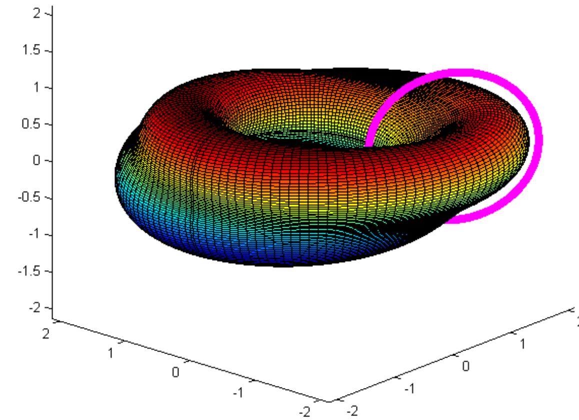
if $q_a<2 \rightarrow$ kink instability.
Normal operations has

$$q_a = \frac{2\pi a^2 B_{tor}}{\mu_0 I_{plasma} R} \geq 3$$

Here density is too small. Part of the electrons are continuously accelerated and lost.



(2,1) kink



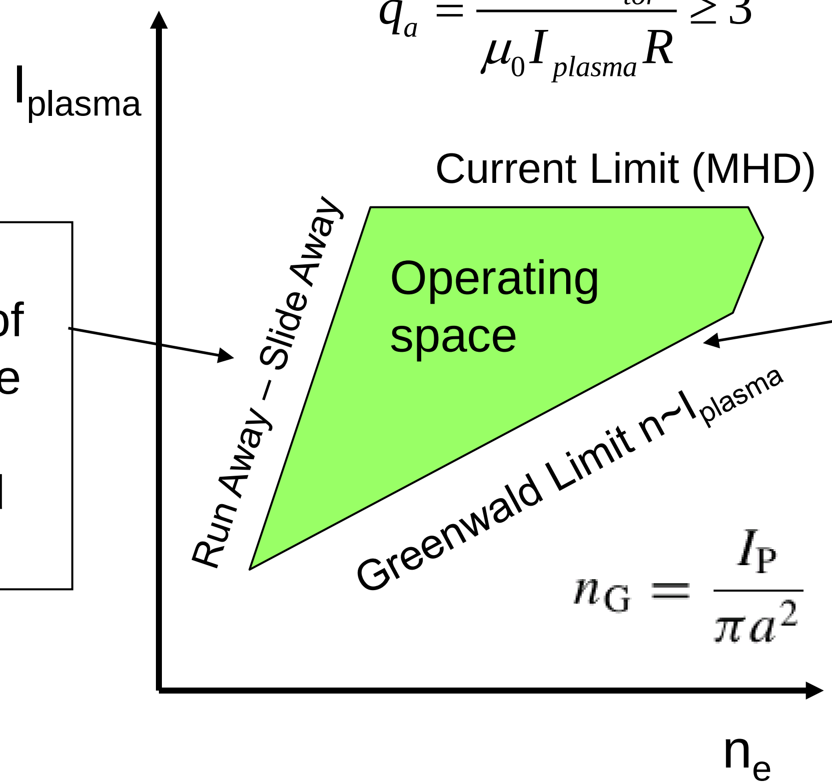
current driven, ideal kink instability

Operation space of the tokamak

if $q_a<2 \rightarrow$ kink instability.
Normal operations has

$$q_a = \frac{2\pi a^2 B_{tor}}{\mu_0 I_{plasma} R} \geq 3$$

Here density is too small. Part of the electrons are continuously accelerated and lost.



This is a phenomenological limit which is not completely understood.

Crossing the limit leads to edge cooling, current profile shrinkage and lost of MHD equilibria. This may lead to disruption.

It involves MHD, transport and atomic processes

Phenomenology of density limit in tokamak:
- lost of global confinement
- H/L transition
- MARFEs (multifaceted asymmetric radiation from the edge)
- divertor detachment

All these events are associated with cooling of the edge plasma

Radiative collapse

n_e increase at constant pressure → Te decreased → strong line radiation from high Zeff

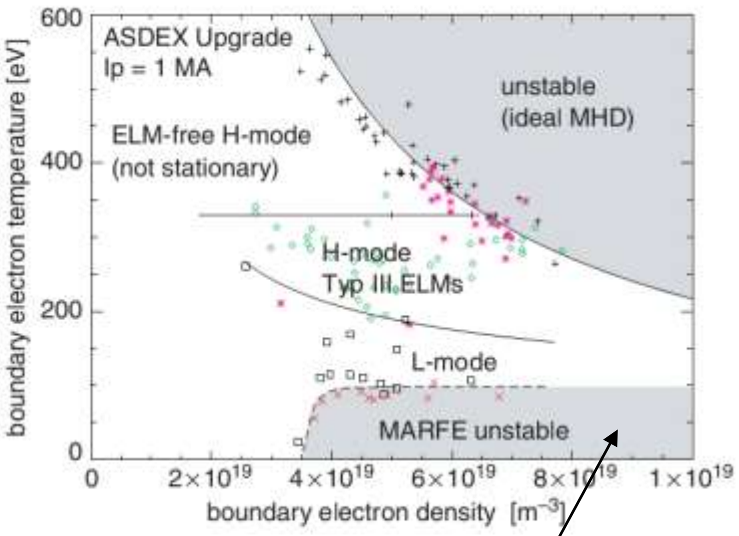
Radiated power = Total heating power

$$n_e^{crit} \propto (P_{heat}/(Z_{eff}-1))^{1/2}$$

F. C. Schuller, "Disruptions in Tokamaks",PlasmaPhys.Control.Fusion, 37, A135 (1995)

MARFEs (multifaceted asymmetric radiation from the edge)

Suttrop W et al1997Plasma Phys. Control. Fusion **39** 2051



Here the density limit is seen to occur when the edge temperature falls below a threshold

- Main energy lost: ionization and charge exchange of incoming neutral particles
- Zone of the high radiation visible on the high field side.

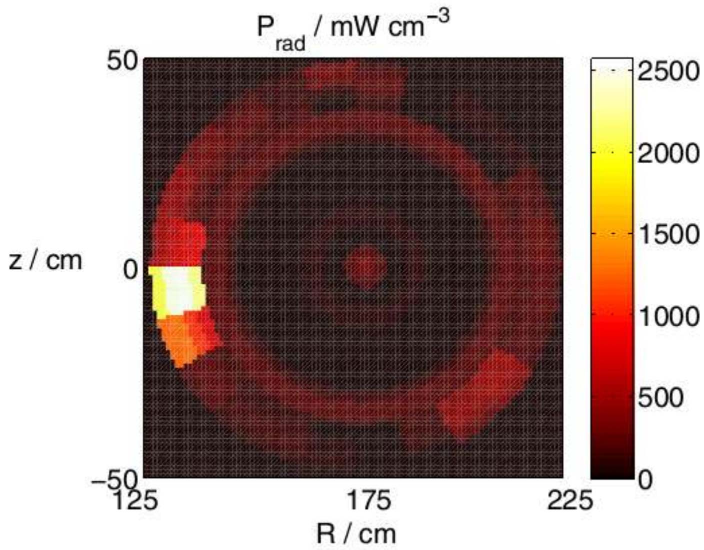
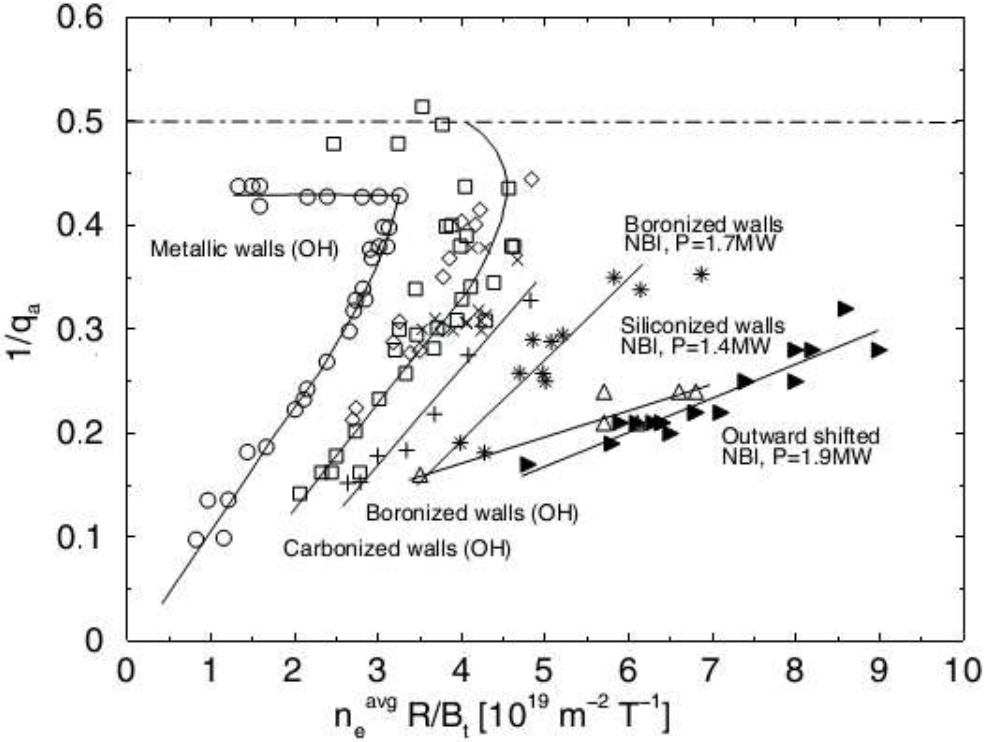


Figure 2: Tomographic reconstruction of the radiated power density during a MARFE in the TEXTOR tokamak.

H. R. Koslowski TRANSACTIONS OF FUSION SCIENCE AND TECHNOLOGY. VOL. 49

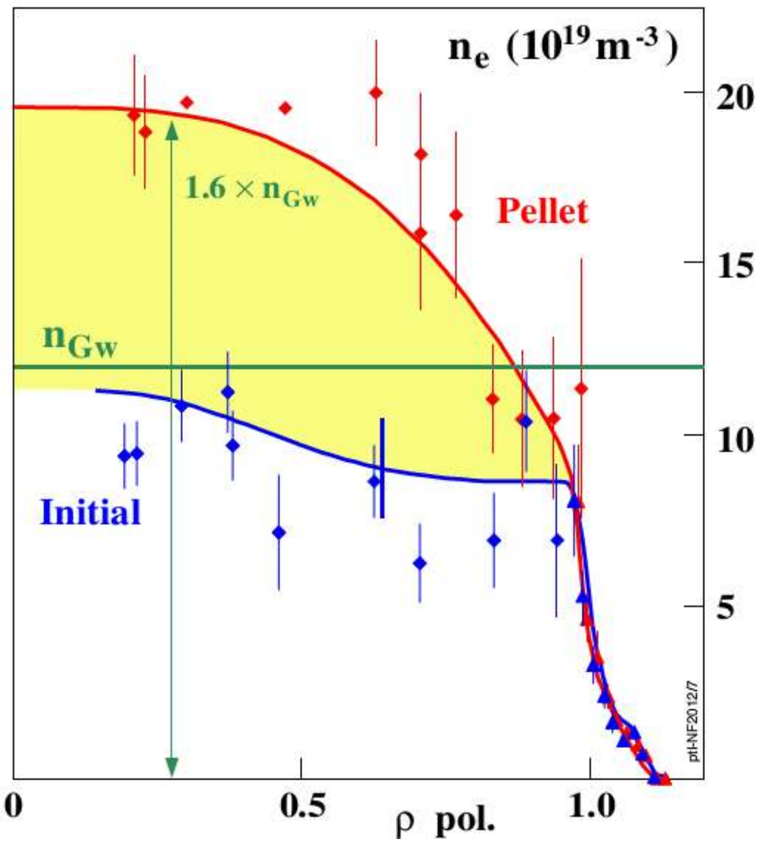
TEXTOR



J. Rapp et al.,. 26th EPS (1999).

Wall conditioning and strong heating shift the limit.

ASDEX Upgrade

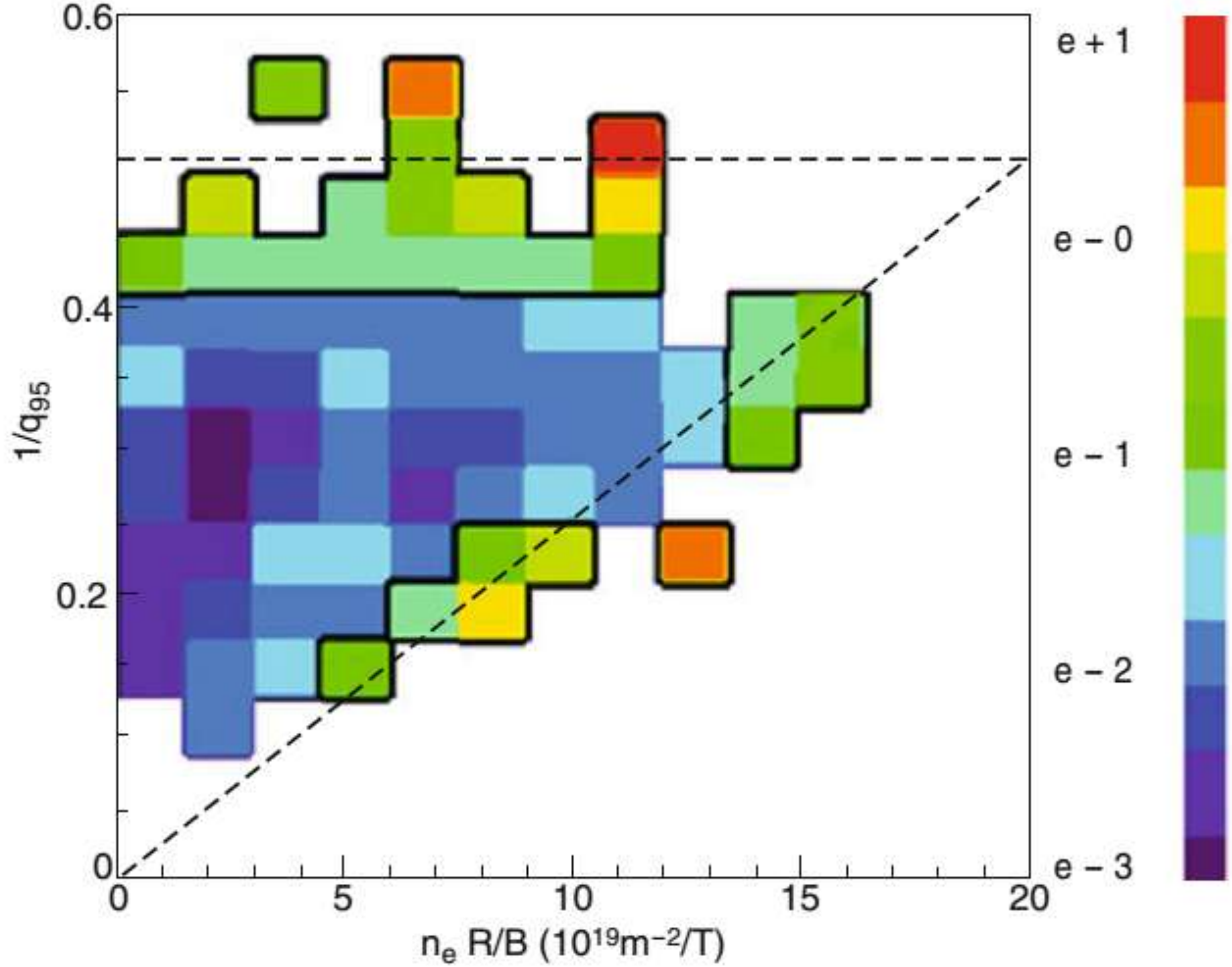


External coils + pellets

Lang Nucl. Fusion 52(2012) 023017

Plasma disruptivity shows Hugill Diagram

P.C. de Vries et al., Statistical analysis of disruptions in JET. Nucl. Fusion49, 055011 (2009)



Operation space of the tokamak

V. Igochine "Active Control of Magneto-hydrodynamic Instabilities in Hot Plasmas", Springer Series on Atomic, Optical, and Plasma Physics , Volume **83**, 2015 Chapter 2.

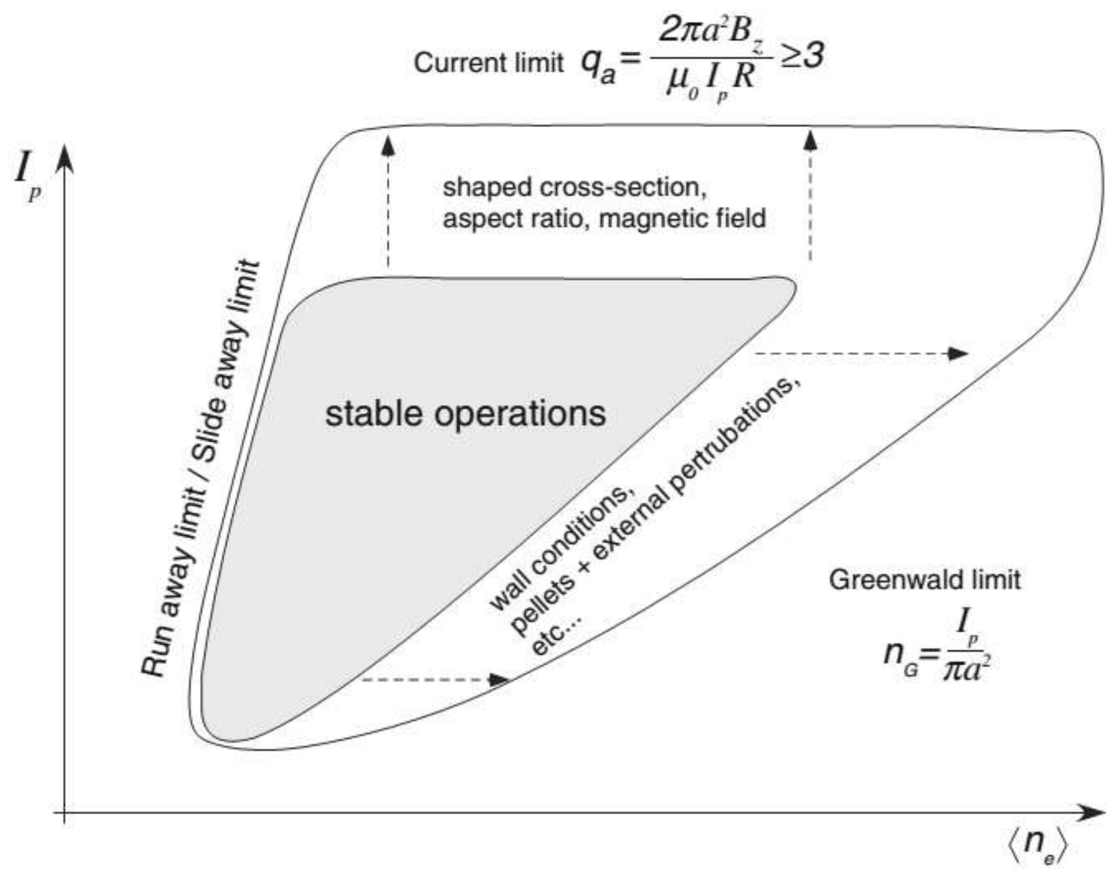


Fig. 2.10 The Hugill diagram and the main limits for plasma operations

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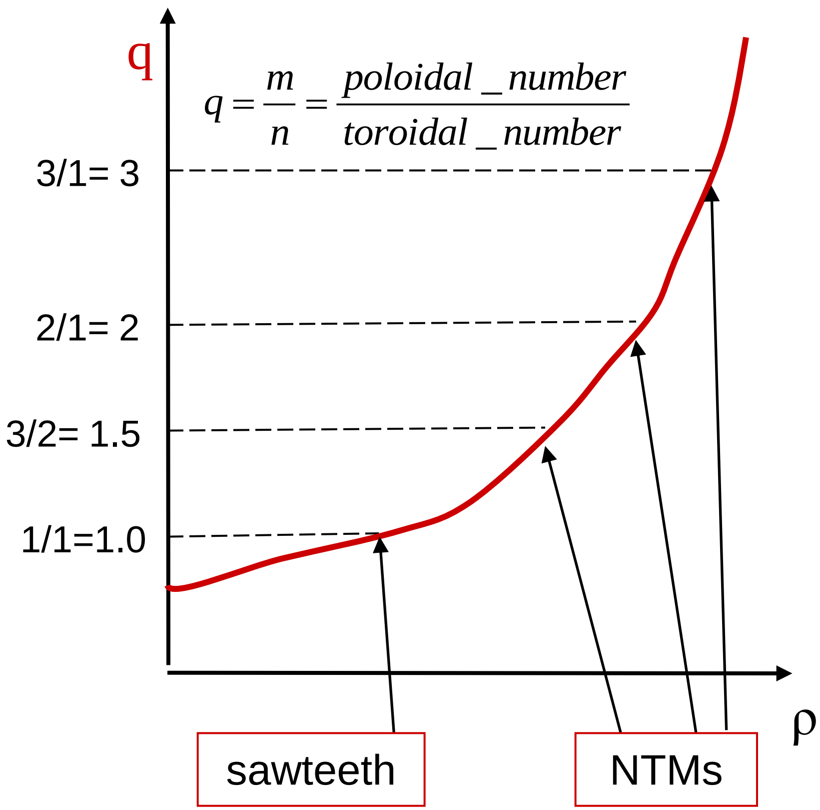
Conventional Tokamak Scenario

Typical safety factor profile for conventional tokamak scenario



Robust, well established, the main scenario for ITER, but … only pulsed operations.

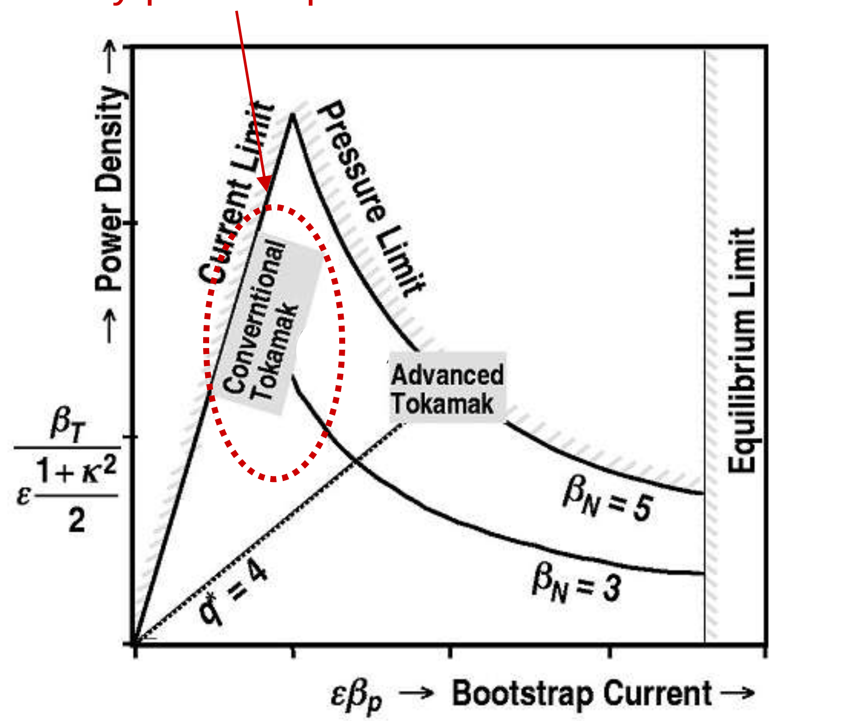
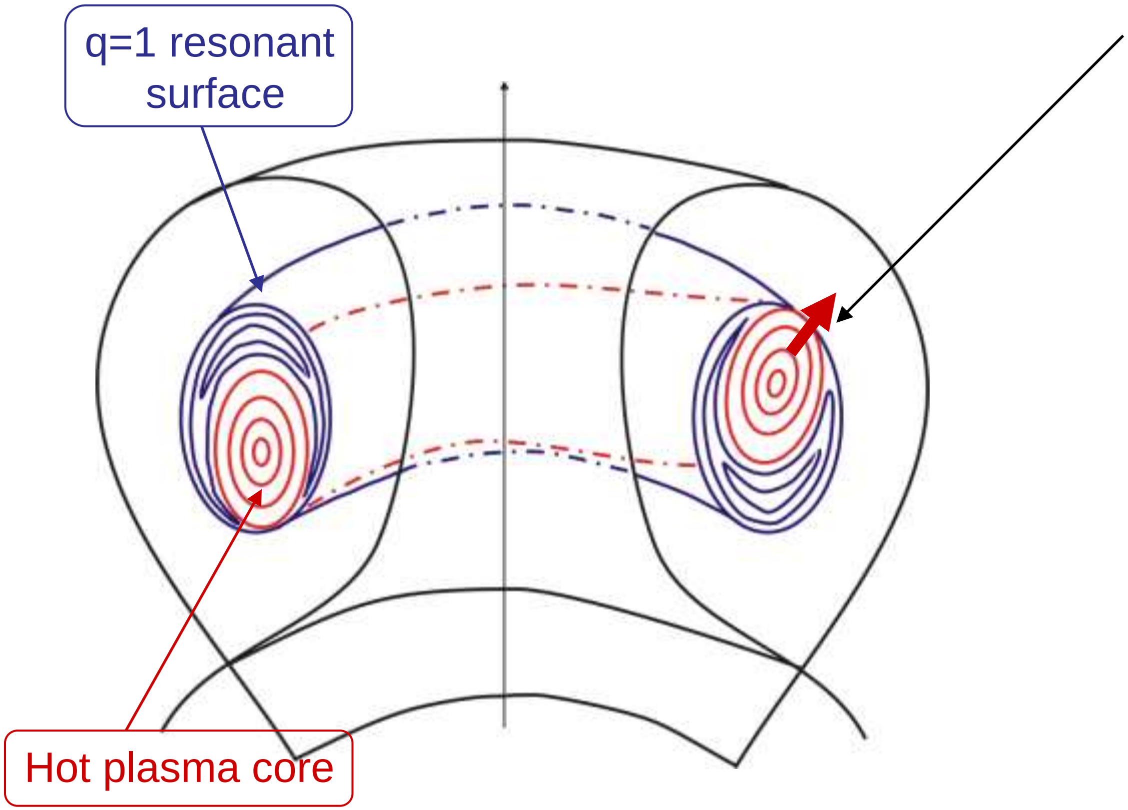


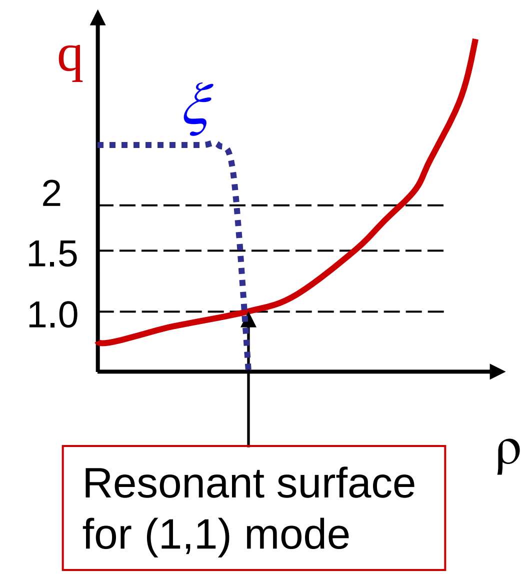
FIG. 1. Conventional tokamaks operate near the current limit, but advanced tokamaks operate near the pressure limit. The tradeoff between high bootstrap current and high power density makes it essential that the AT operate at high β_N.

[C. M. Greenfield et. al. PoP 2004]

Sawteeth: internal (1,1) kink mode.

Heat flow out of q=1 during the crash





A Tilt and Shift of the Core Plasma.

Why do we need to control sawteeth?

- Long Sawteeth **have been shown to** trigger Neo-classical Tearing Modes
  - Long Sawteeth → NTMs
  - Short Sawteeth → Avoid NTMs
- **NTMs degrade plasma confinement**
- Even bigger problem in ITER

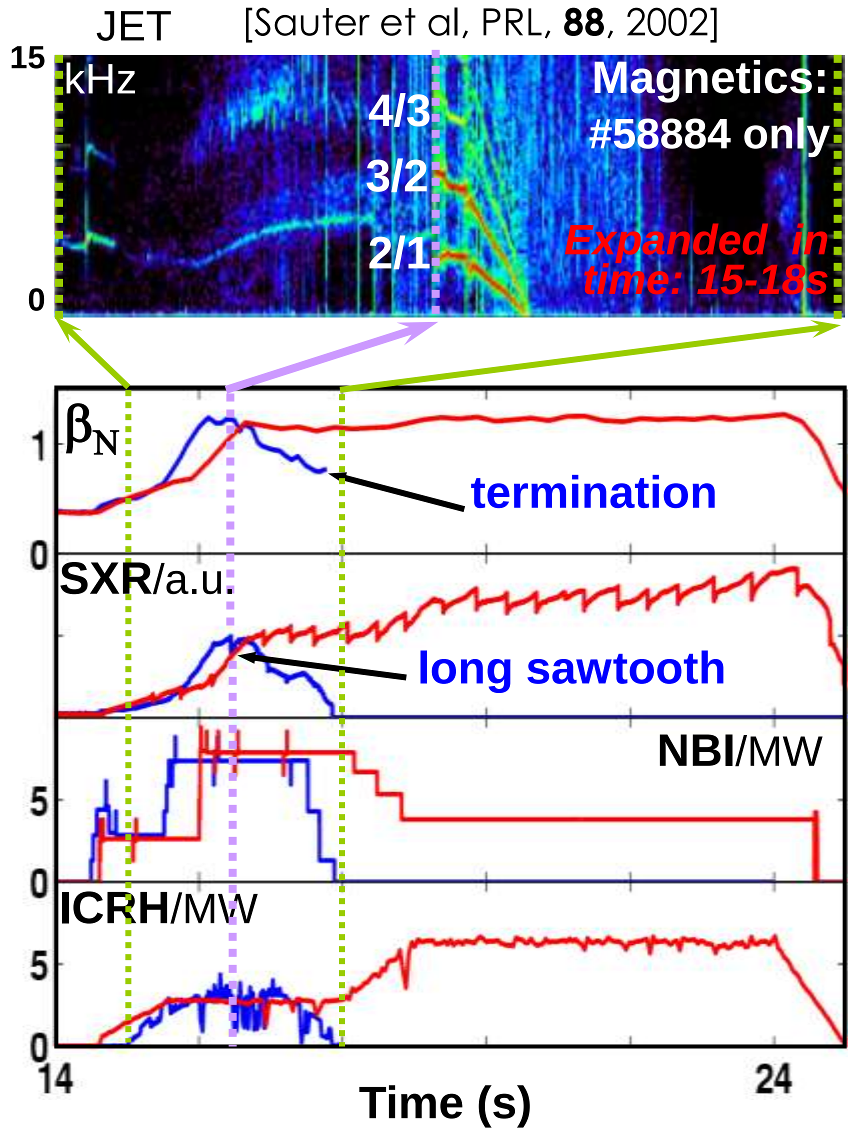
Fusion born α's

⇓

Long sawtooth periods

⇓

More likely to trigger NTMs



Influence of fast particles on sawteeth is now a subject of very intense investigations

On axis NBI - stabilised sawteeth
(longer and bigger)

Off axis NBI - destabilized sawteeth
(sorter and smaller)

Similar experiments were made in
ASDEX Upgrade,
MAST,
TEXTOR,
ToreSupra

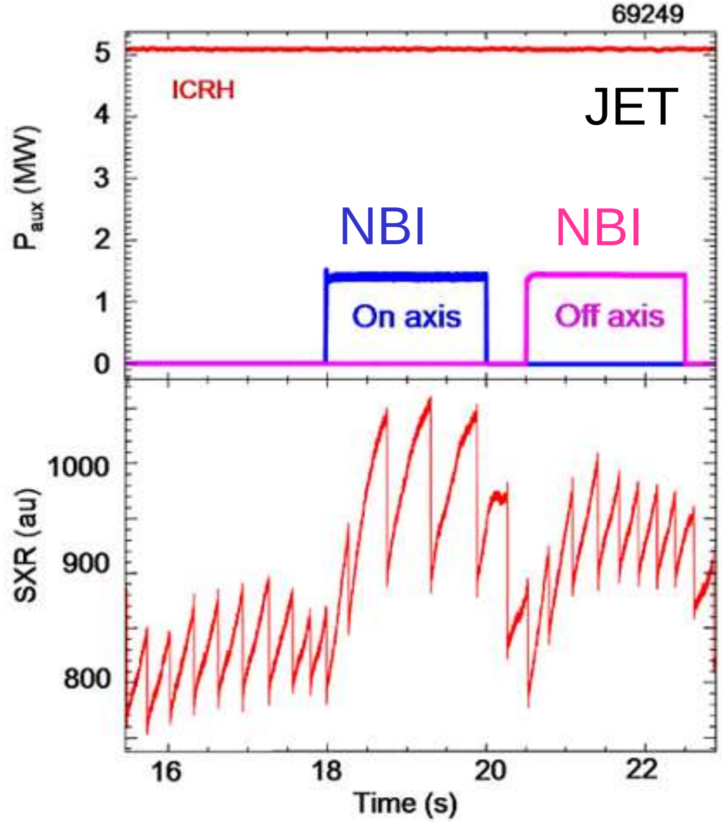


Figure 2. The SXR emission trace for JET discharge 69249 together with the auxiliary heating waveforms. The sawteeth are significantly shorter during the off-axis NBI heated phase, than when on-axis NBI is applied.

[I.Chapman, V. Igochine et.al. NF 2009]

Influence of ICRH on the sawteeth in JET

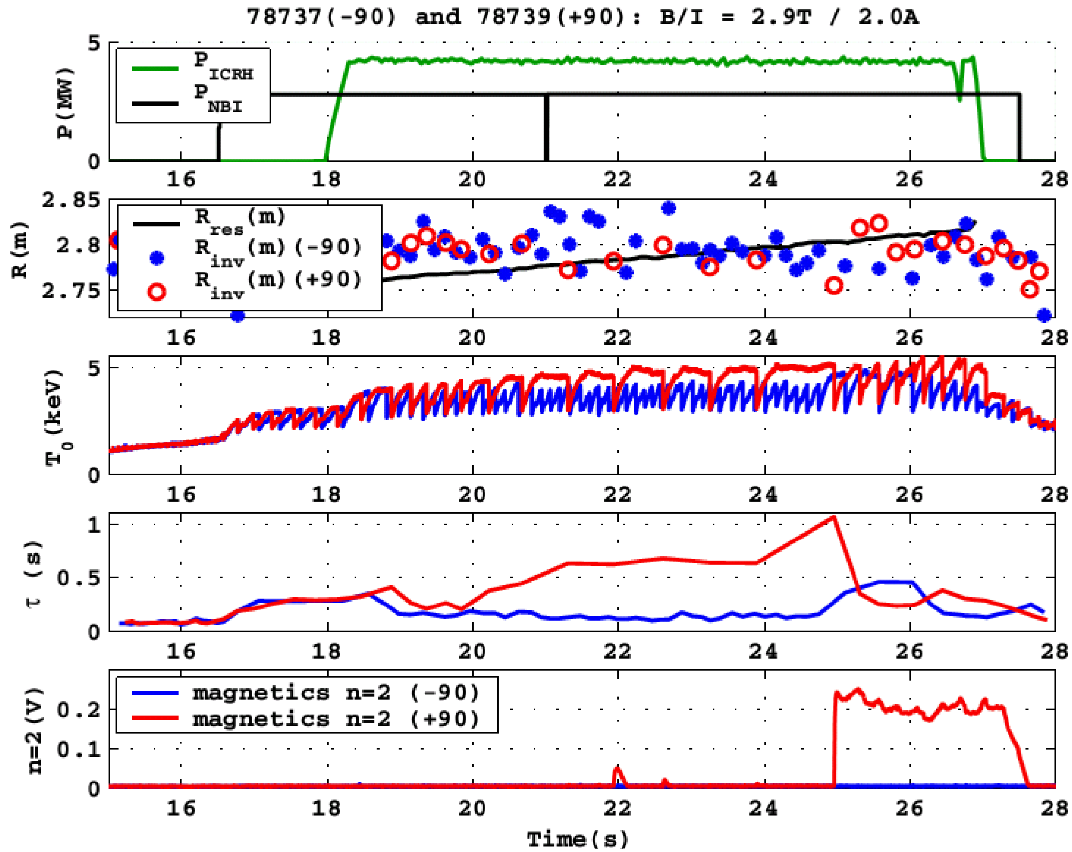


Figure 20. JET pulses 78737 and 78739 with −90° and +90° ICRH off-axis, respectively. The NBI fast ions in the core lead to long sawtooth periods. When the He^3 minority heating is deposited off-axis, the fast ions destabilize the sawteeth with −90° phasing, but stabilize them with +90° phasing, to the extent that a 1 s long sawtooth triggers an NTM. Reproduced with permission from Graves J P *et al* 2010 *Nucl. Fusion.* **50** 052002

ICRH influence stability of the (1,1) mode by:

1) acting on the minority

2) heating of the plasma

Destabilization of fast particle stabilized sawteeth by current drive ECCD

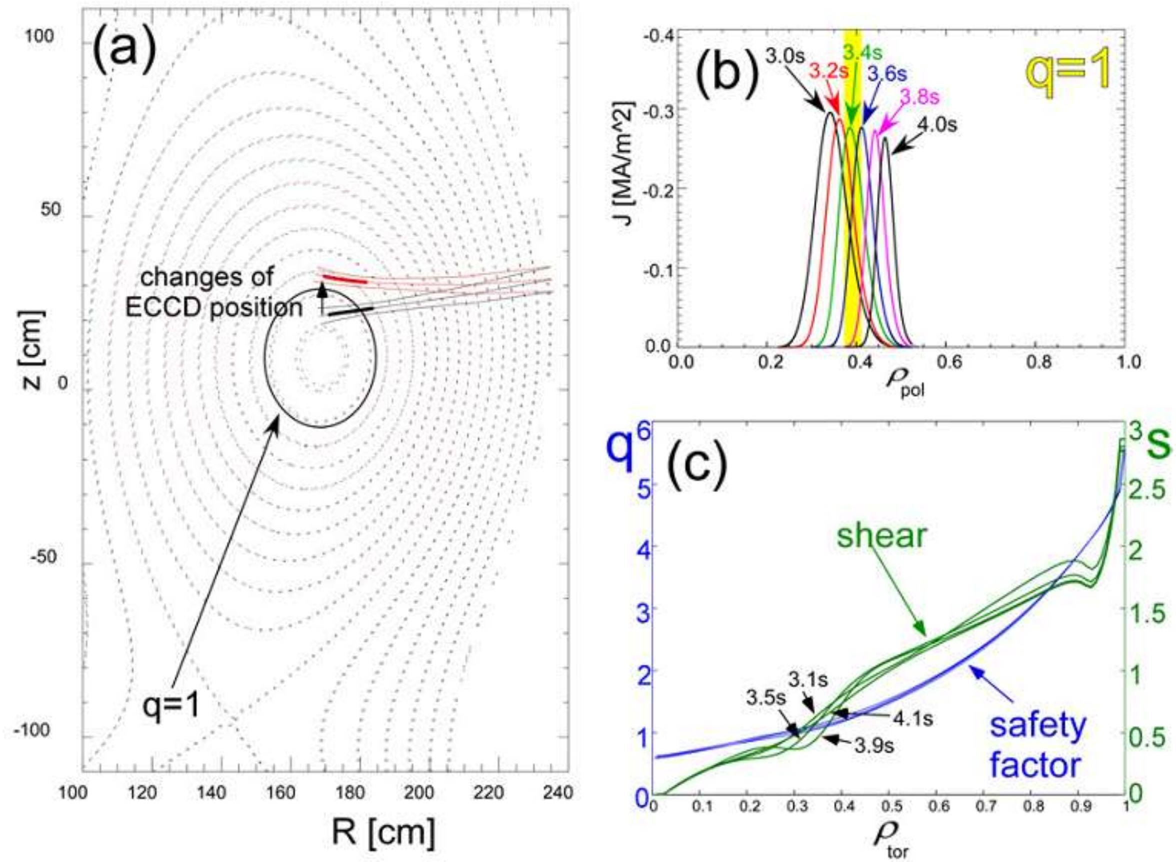
Main aim now is to construct and investigate **ITER relevant situations** in present tokamaks:

• Long sawteeth production with NBI and central ICRH (mock up α-particles in ITER)

• Stabilization and destabilization with ECCD

<u>Results:</u>

First successful experiments in ToreSupra and ASDEX Upgrade

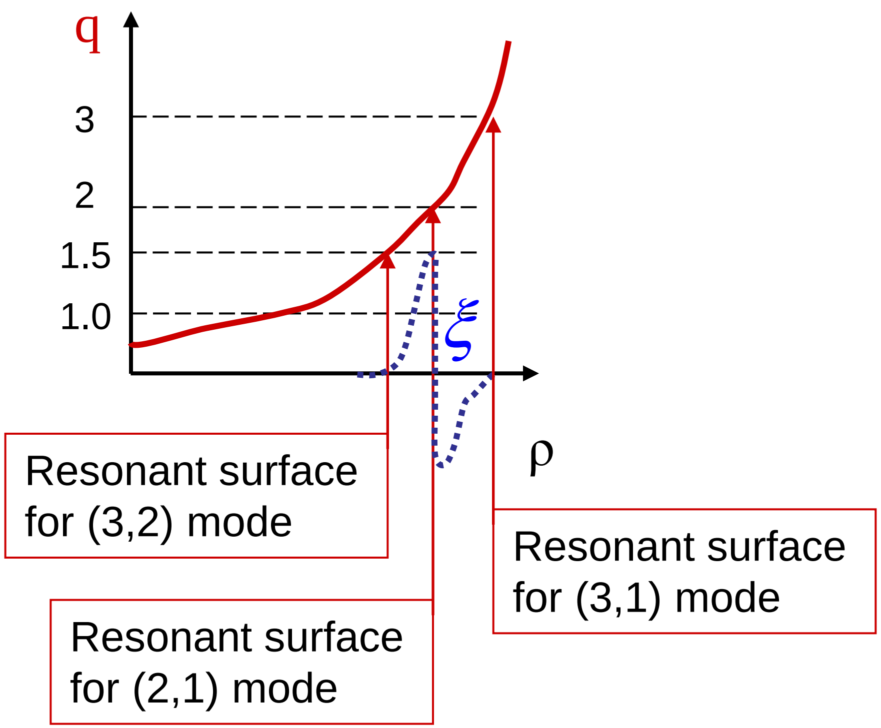


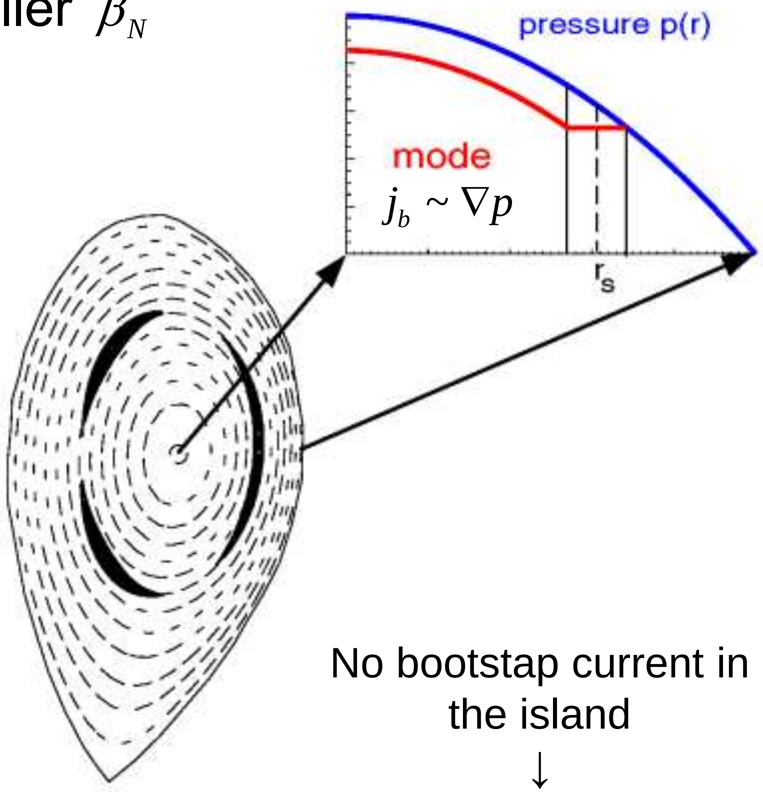
About 40% reduction of the sawtooth period is achieved

[Igochine et.al. Plasma Phys. Control. Fusion, 2011]

Neoclassical Tearing Mode (NTM)

Main problem: Neoclassical Tearing Mode flattens pressure and temperature profile → smaller β_N (Fusion power ~ β_N^2)





No bootstap current in the island

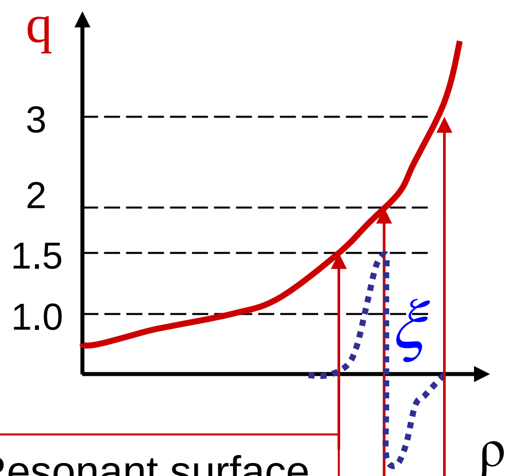
↓

Current hole in the island

↓

Mode grows

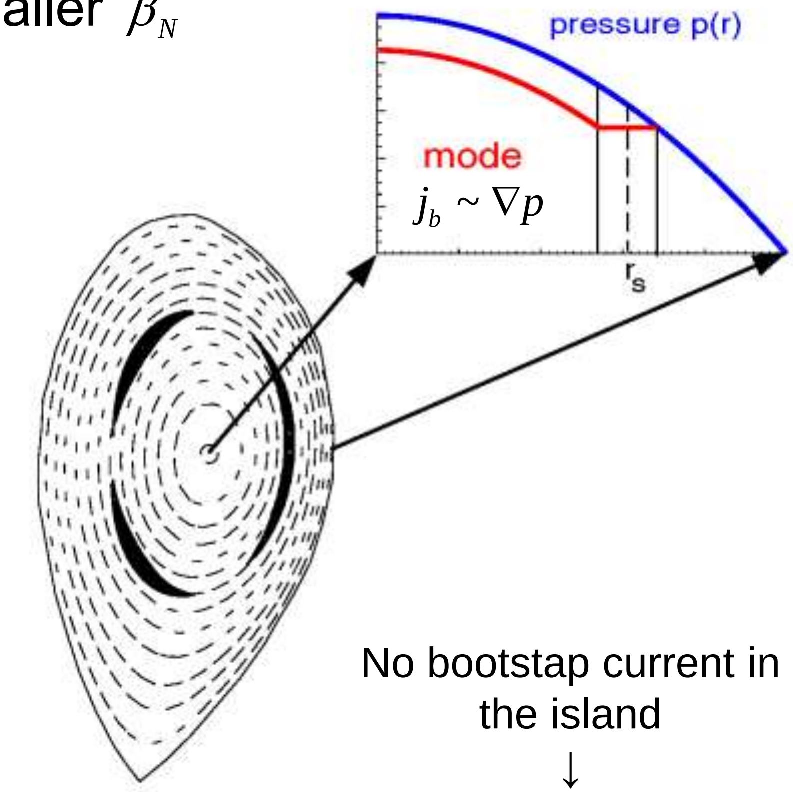
<u>Main problem:</u>Neoclassical Tearing Mode flattens pressure and temperature profile → smaller β_N
(Fusion power ~ β_N^2)



Resonant surface for (3,2) mode

Resonant surface for (2,1) mode

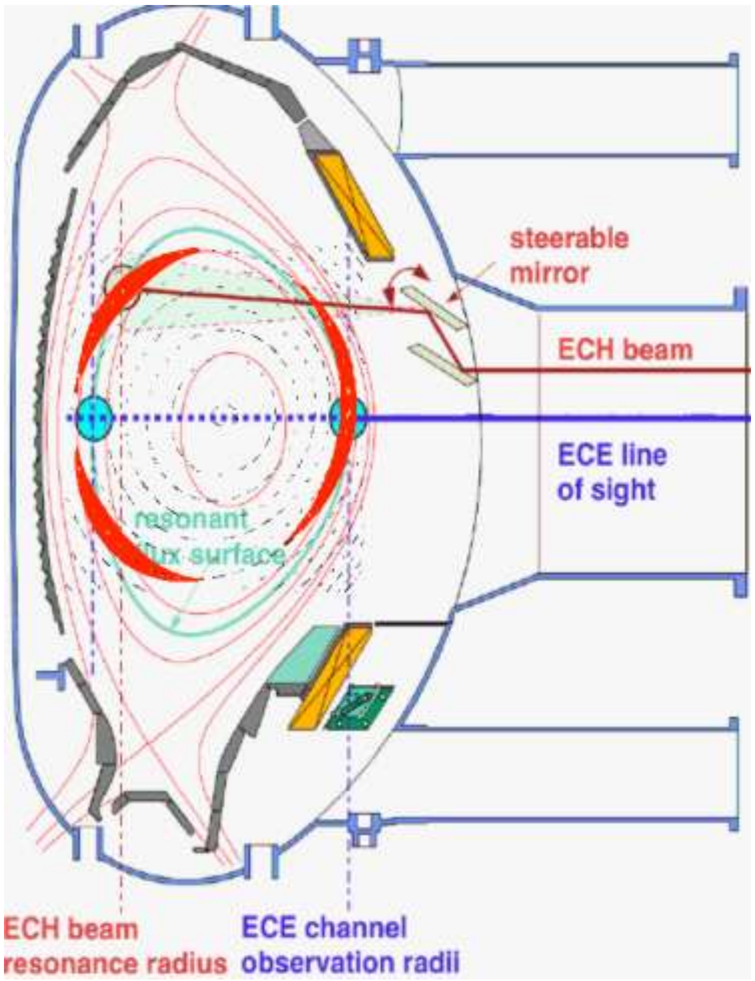
Resonant surface for (3,1) mode



No bootstap current in the island
↓
Current hole in the island
↓
Mode grows

Solution: fill the current hole

Control of NTM with Electron Cyclotron Current Drive (ECCD)



NTM was stabilized by local Electron Cyclotron Current Drive (ECCD) in ASDEX Upgrade [Zohm,NF, 1999].

Since that time the method was confirmed to be robust on other tokamaks and is foreseen for ITER.

Current activities:

• more efficient suppression
  (modulated current drive, only in O-point of the island, [Maraschek, PRL, 2007])

• online control and feedback actions on the mode [DIII-D, JT60U, ASDEX Uprgarde, etc]

Frequently interrupted regime of neoclassical tearing mode

A new regime was discovered in ASDEX Upgrade in 2001. The confinement degradation is strongly reduced in this regime. [A. Gude et. al., NF, 2001, S.Günter et. al. PRL, 2001]

Neoclassical tearing mode never reach its saturated size in this regime. Fast drops of NTM amplitudes appear periodically.

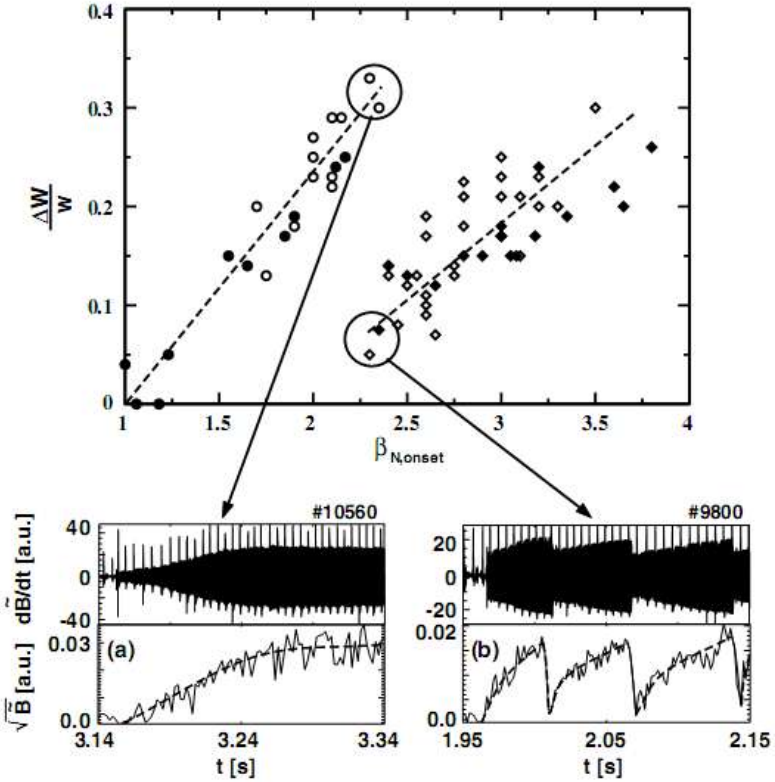


Figure 9. Comparison of reduction in energy confinement ($\Delta W/W$) due to (3,2) NTMs on ASDEX Upgrade (open symbols) and JET (full symbols). Very good agreement is seen, both in the relative confinement degradation as well as in the β_N value above which FIR-NTMs cause less energy losses. The lower figure shows the NTM behaviour for two ASDEX Upgrade discharges at about $\beta_N = 2.3$. The time-averaged amplitude for the FIR-NTM is significantly smaller (b) than the saturated amplitude of the smoothly growing mode (a). [T. Hender et. al. NF, 2007]

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Transition to this regime may be an option for ITER.

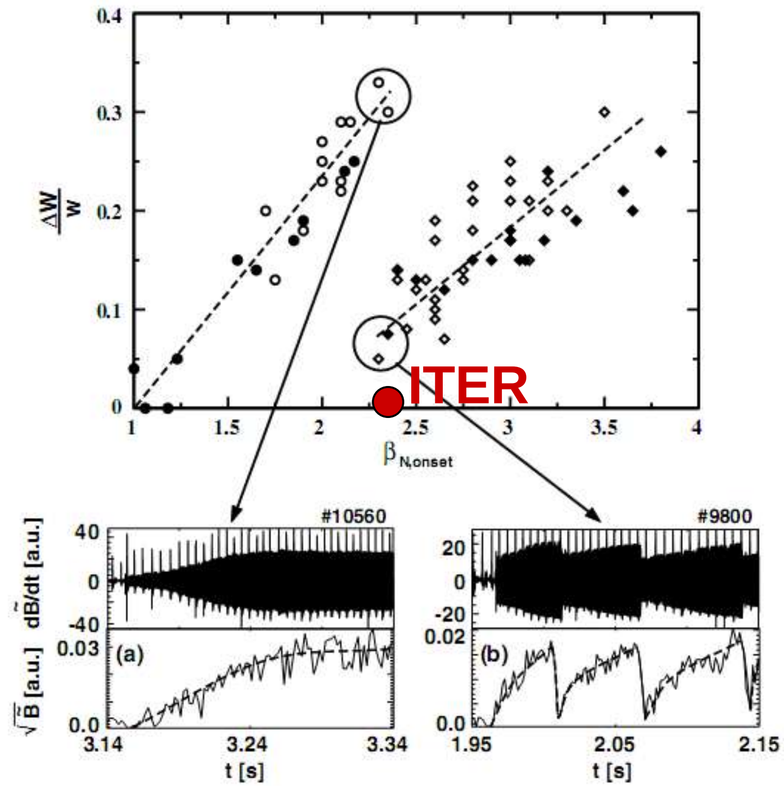

Figure 9. Comparison of reduction in energy confinement ($\Delta W/W$) due to (3,2) NTMs on ASDEX Upgrade (open symbols) and JET (full symbols). Very good agreement is seen, both in the relative confinement degradation as well as in the β_N value above which FIR-NTMs cause less energy losses. The lower figure shows the NTM behaviour for two ASDEX Upgrade discharges at about $\beta_N = 2.3$. The time-averaged amplitude for the FIR-NTM is significantly smaller (b) than the saturated amplitude of the smoothly growing mode (a). [T. Hender et. al. NF, 2007]

It was found that the reason for this fast periodic drop is interaction of the (3,2) neoclassical tearing mode with (1,1) and (4,3) ideal modes. Such interaction leads to stochastization of the outer island region and reduces its size. (The field lines are stochastic only during the drop phase.)

(3,2) only

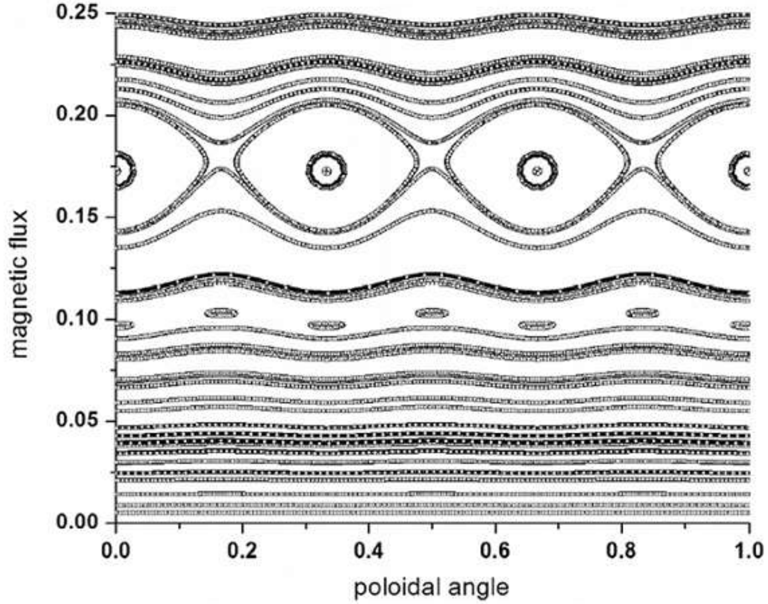

Figure 3. The (3, 2) mode is used as a perturbation. Shape of the perturbation is shown in figure 2. The ASDEX Upgrade discharge No #11681, $t = 2.98$ s.

(3,2) + (1,1) + (4,3)

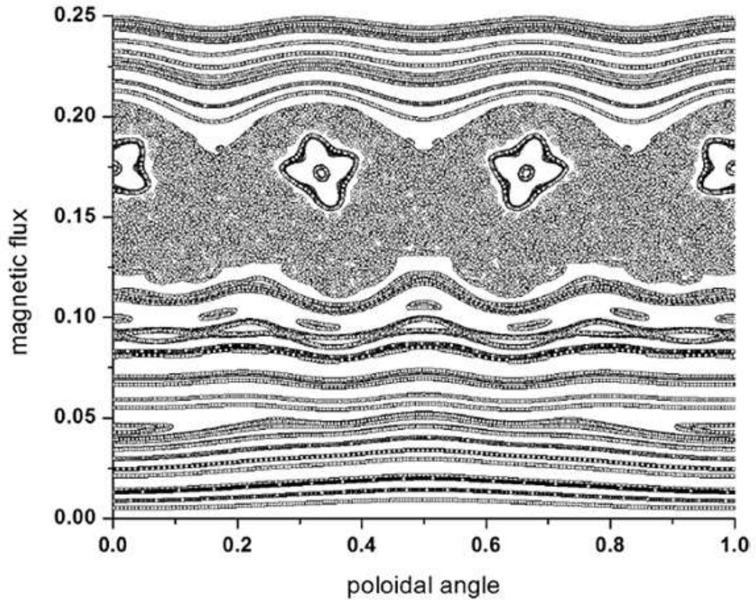

Figure 5. The (1, 1), (3, 2) and (4, 3) modes are used as perturbations. Shapes of the perturbations are shown in figure 2. The ASDEX Upgrade discharge No 11681, $t = 2.98$ s.

[V. Igochine et. al. NF, 2006]

Outline of the talk

- Tokamak equilibrium and drives for instabilities
- Operation space of tokamaks
  - Limits and possible extensions
- Performance limiting MHD instabilities
  - In conventional scenario
    - Sawteeth
    - Neoclassical Tearing Mode (NTM)
  - In advanced scenario
    - Resistive wall mode (RWM)
  - Edge Localized Mode (ELM)
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  - Disruption
- Summary

Disrupt more often, only transient up to now, but … steady state operations are possible.

This scenario has to have higher β_N compared to conventional scenario to be attractive.

$\beta_N \approx 5$

$\beta_N \approx 3$

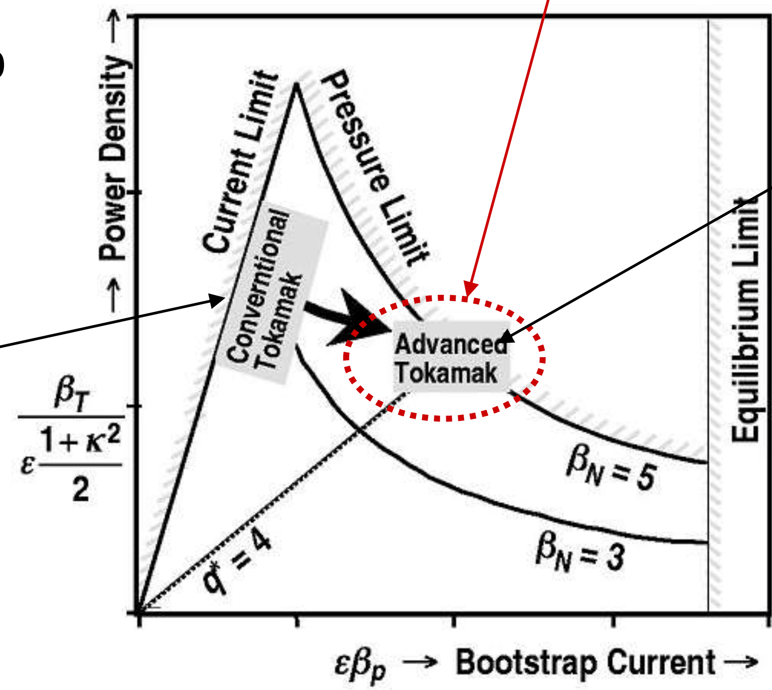
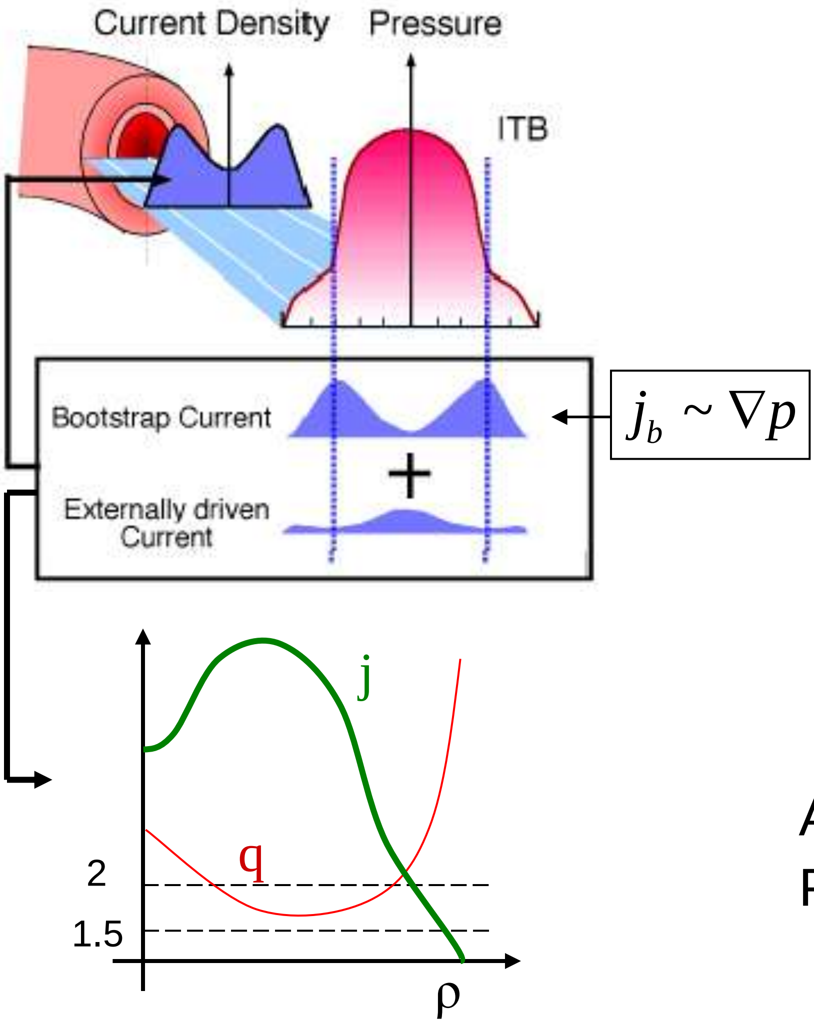


FIG. 1. Conventional tokamaks operate near the current limit, but advanced tokamaks operate near the pressure limit. The tradeoff between high bootstrap current and high power density makes it essential that the AT operate at high β_N.

[C. M. Greenfield et. al. PoP 2004]



Flat or hollow current profiles

↓

Suppression of the turbulence

↓

Internal Transport Barrier (ITB)

↓

Reduced energy transport

↓

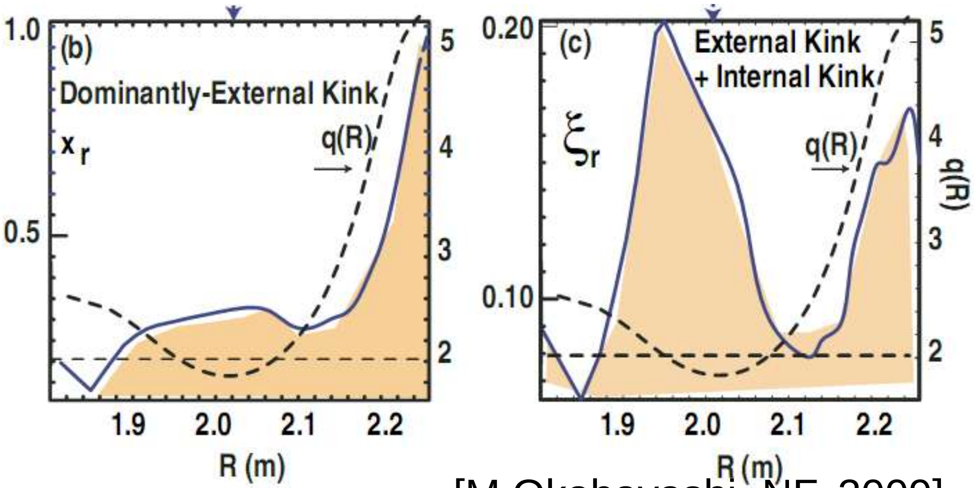
Promises steady state operations

Aim: Steady state operations

Problem: Up to now this is only a transient scenario

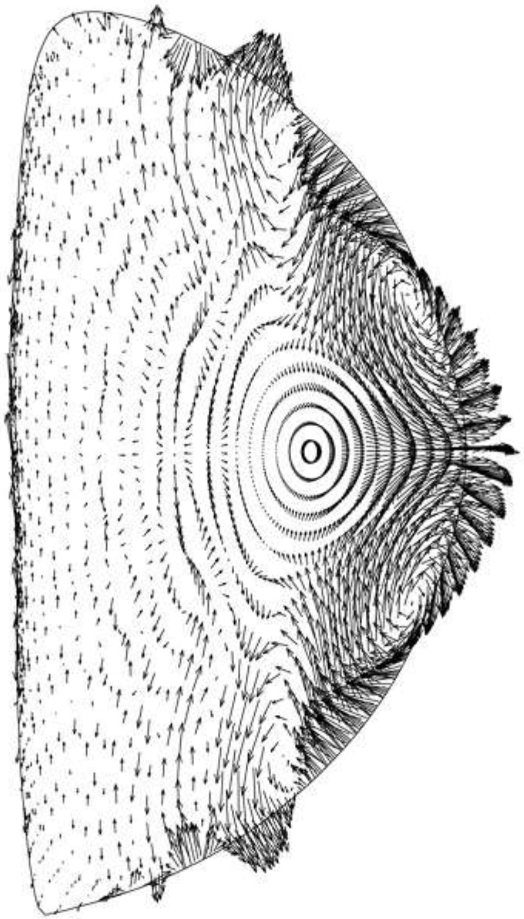
Resistive wall mode is an external kink mode which interacts with the vacuum wall.

The mode would be stable in case of an ideally conducting wall. Finite resistivity of the wall leads to mode growth.



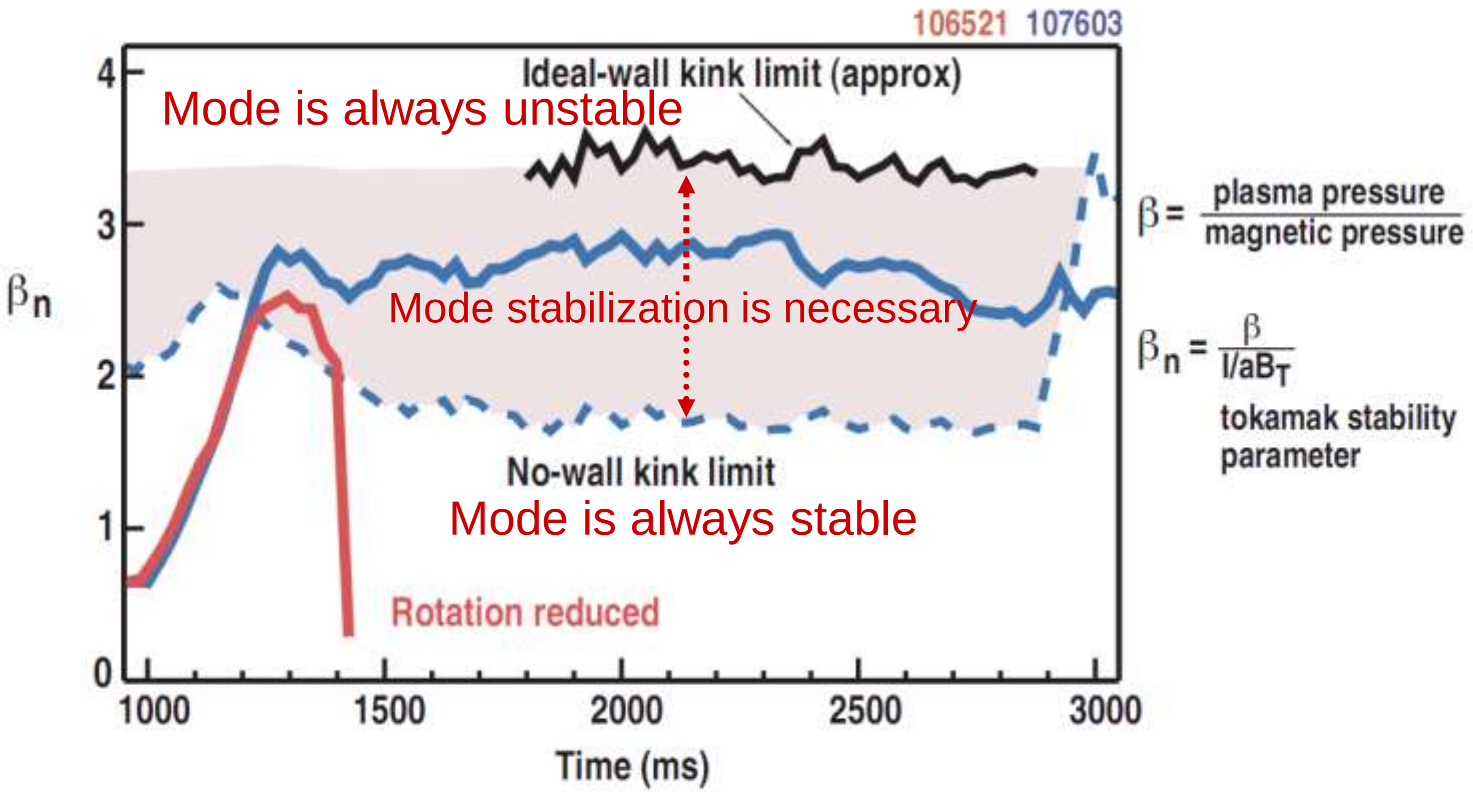
[M.Okabayashi, NF, 2009]

[T. Luce, PoP, 2011]



RWM has global structure. This is important for “RWM ↔ plasma” interaction.

Resistive Wall Mode (RWM)



[DIII-D review 2002]

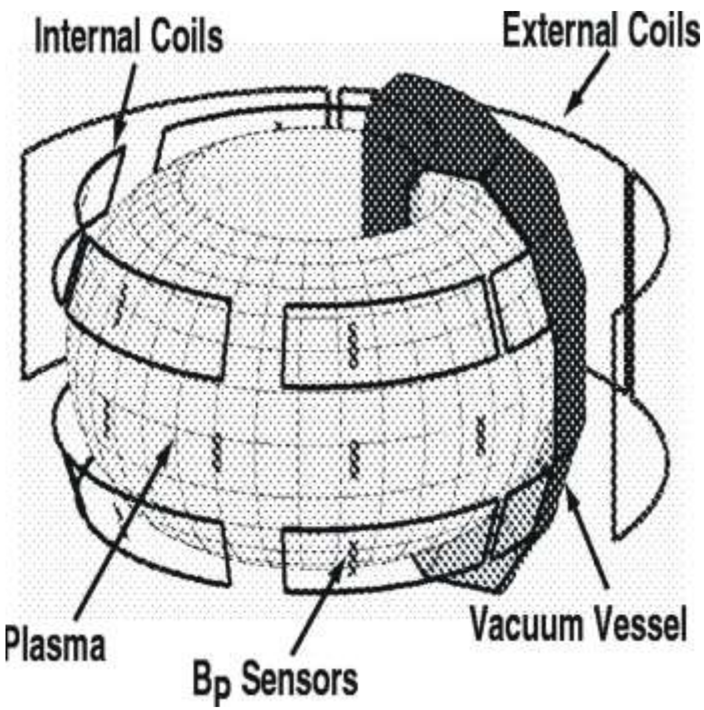
Stabilization gives approximately factor 2 in β_N , which is about factor 3-4 for fusion power.

Stabilizing effect gives:

• plasma rotation (expected to be small in ITER)

• feedback control with external coils (foreseen in ITER design)

Error fields gives destabilizing effect on the mode.



Strait et al.,PoP **11**,(2004)2505

Resent results:

• Stabilization at low plasma rotation [H.Reimerdes, et. al., PRL, 2007]

• Other MHD instabilities (ELMs, fishbones) could trigger this mode [M.Okabayashi, NF, 2009]

[Liu, NF, 2010]

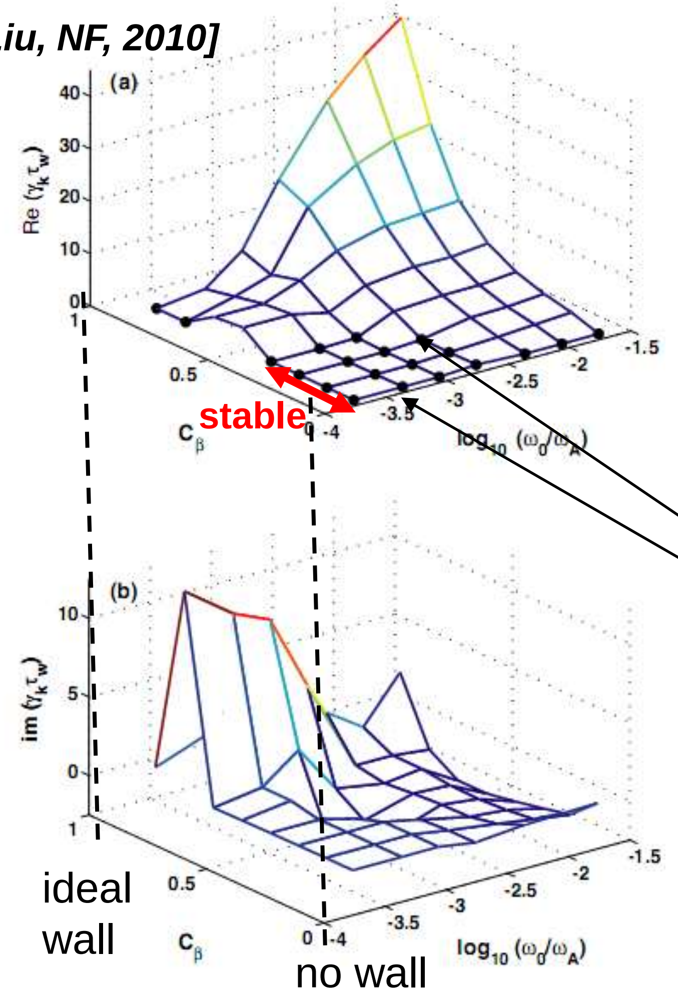


Figure 13. 2D plots of (*a*) real and (*b*) imaginary parts of the RWM eigenvalue for ITER advanced tokamak plasmas, predicted by the self-consistent kinetic calculations. Only precessional resonance damping is included. The black dots indicate stable RWMs with practically vanishing growth rates.

Particle effects are very important!

Stabilizing effects comes from fast particles and from thermal particles (from wave particle interaction)

Black dots are zero growth rates

RWM is stable at low plasma rotation up to $C_\beta \leq 0.4$ (40 % above no wall limit)

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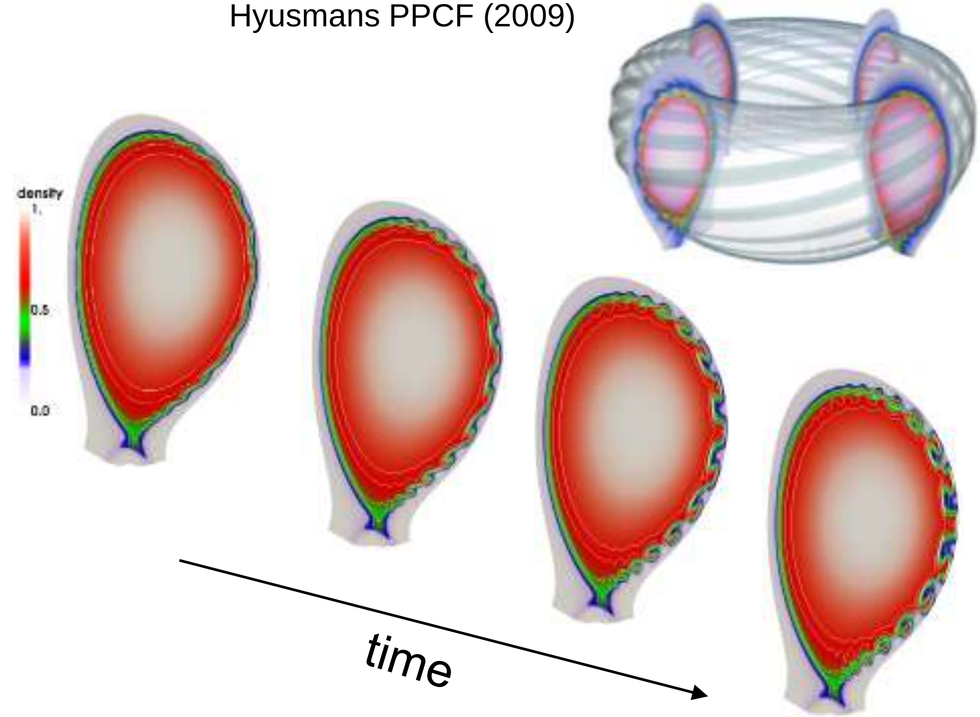
Edge Localized Mode (ELM)

Hyusmans PPCF (2009)



<u>Main problem:</u>

Large heat loads on the plasma facing components.

The maximal heat loads should be reduced.

(Tungsten melting, droplets, surface cracks if W_{ELM}>1MJ.
... but predicted for large ELMs: $W_{ELM,ITER}$~30MJ!
ITER divertor life-time = only few shots with ELMs!)

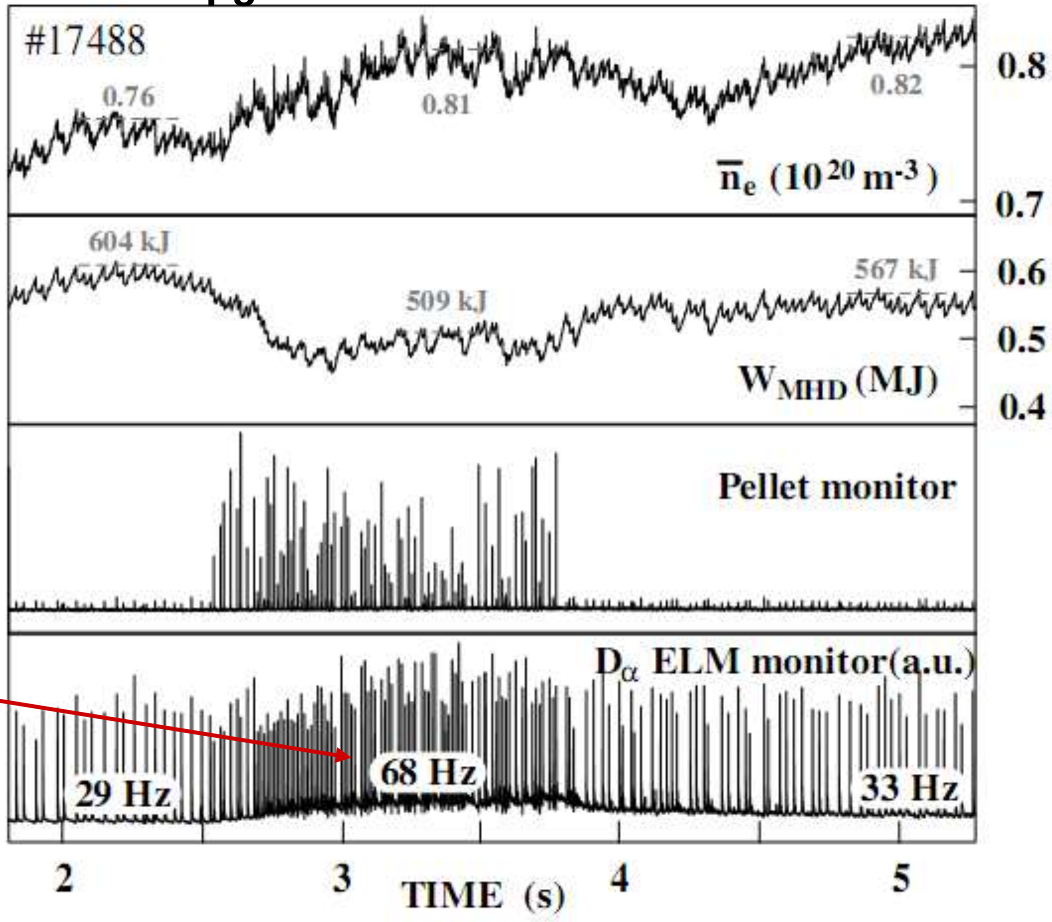
This mode consists of many harmonics and is localized at the plasma edge

Control of Edge Localized Mode (ELM) by pellet injections

One of the ways to solve the problem is injecting a small cold piece of Hydrogen or Deuterium (so called: PELLET) which triggers an ELM.

Increase of the ELM frequency during the pellet phase

ASDEX Upgrade



P. Lang, et. al., 30th EPS Conference

Current status:

- DIII-D: ELM suppression
- MAST: no suppression
- JET : ELM mitigation
- NSTX: ELM “triggering”
- ASDEX Upgrade: mitigation
- KSTAR: mitigation

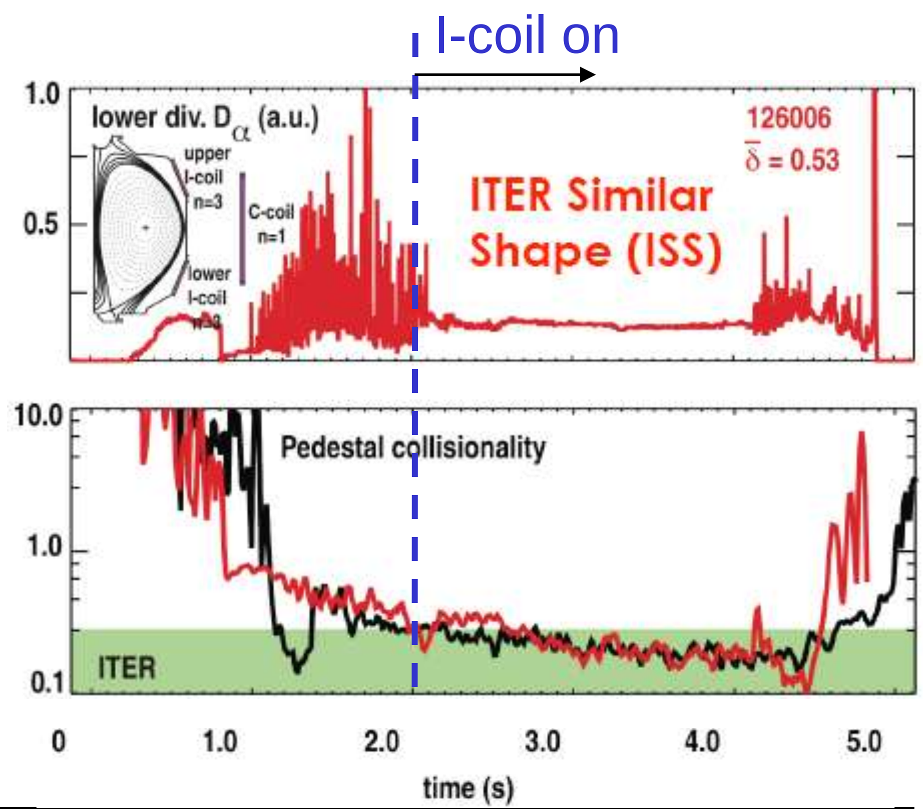
There are no definitive answers up to now. Intensive modeling and experimental efforts are focused on this issue.

Possible explanation for ELM triggering is stochastization of the plasma edge (DIII-D). But this is not always valid (ASDEX-Upgrade). **Non resonant fields also give mitigation effect!**

DIII-D results:

I-coils, n=3 : total ELM suppression at ITER-like collisionality

[Evans, MHD ITPA ,Naka,2008]



ELM control in ITER

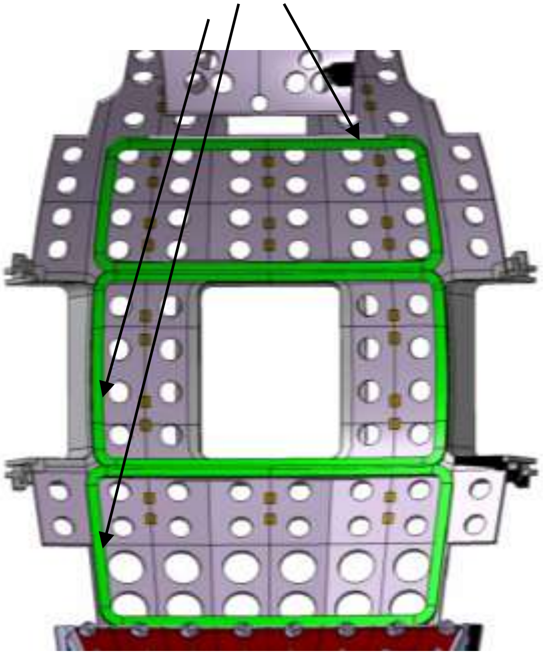
Both ELM control possibilities are currently being explored for ITER.

The other way is scenario developments:

• Scenarios with much smaller and more frequent ELMs (''type II'', ''type III'')

• Scenarios without ELMs (''Quiescent H-mode'')

ELM coils in ITER, low field side



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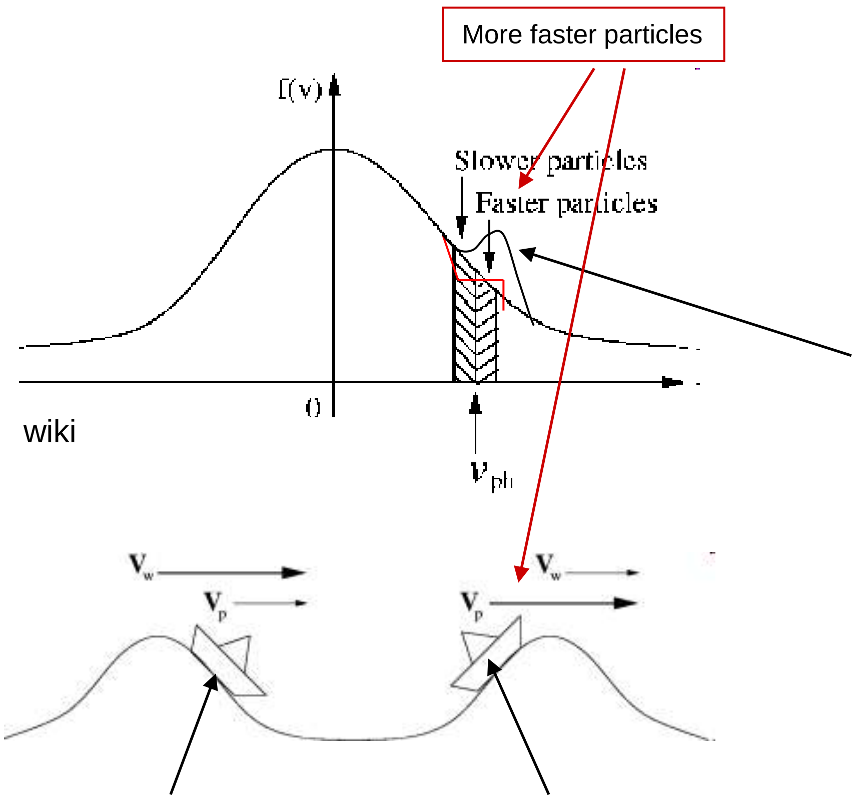
Landau damping and fast particle modes

Energy exchange between a wave with phase velocity v_{ph} and particles in the plasma with velocity approximately equal to v_{ph}, which can interact strongly with the wave.



accelerated decelerated

During this process particle gains energy from the wave without collisions.

But if the distribution function different the result could be opposite! <u>Waves (instabilities) will gain energy</u> from the fast particles. This produces fast particle driven mode.

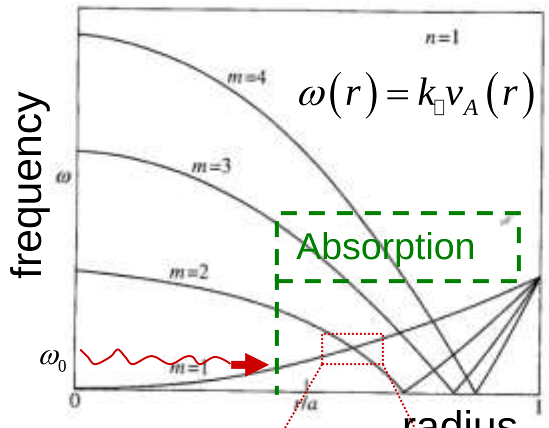
Particles could drive MHD mode unstable as well!

Burning fusion plasma is a source of fast α-particles.

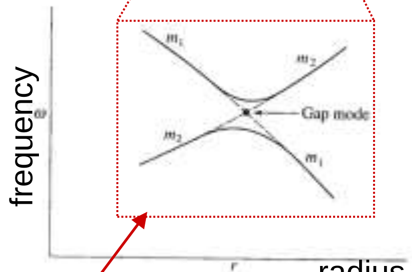
<u>Main problems which could appear in ITER:</u>

- these particles have to be confined long enough in order to transfer their energy to the background plasma
- MHD modes interact with fast ions and redistribute them
- fast ions can excite MHD modes
- fast particle flux could damage the wall

Alfven resonances in cylindrical case



Toroidal case



Wesson, Tokamaks, 3rd Edition

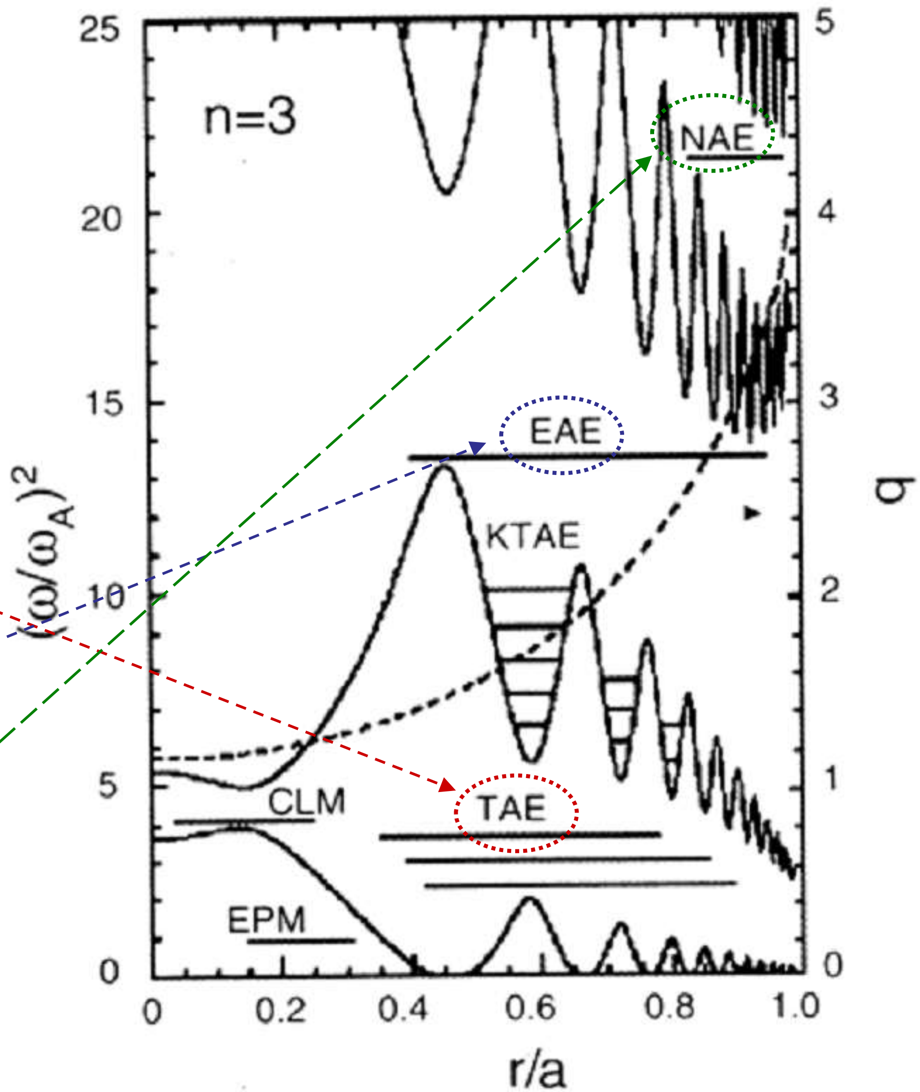
Toroidicity introduces weakly damped gap modes (TAEs). These modes can be destabilized by interaction with fast particles.

List of main fast particle modes (1)

Fast particles in modern tokamaks comes from Neutral Beam Injection (NBI) and Ion Cyclotron Resonance Heating (ICRH).

Observed modes are:

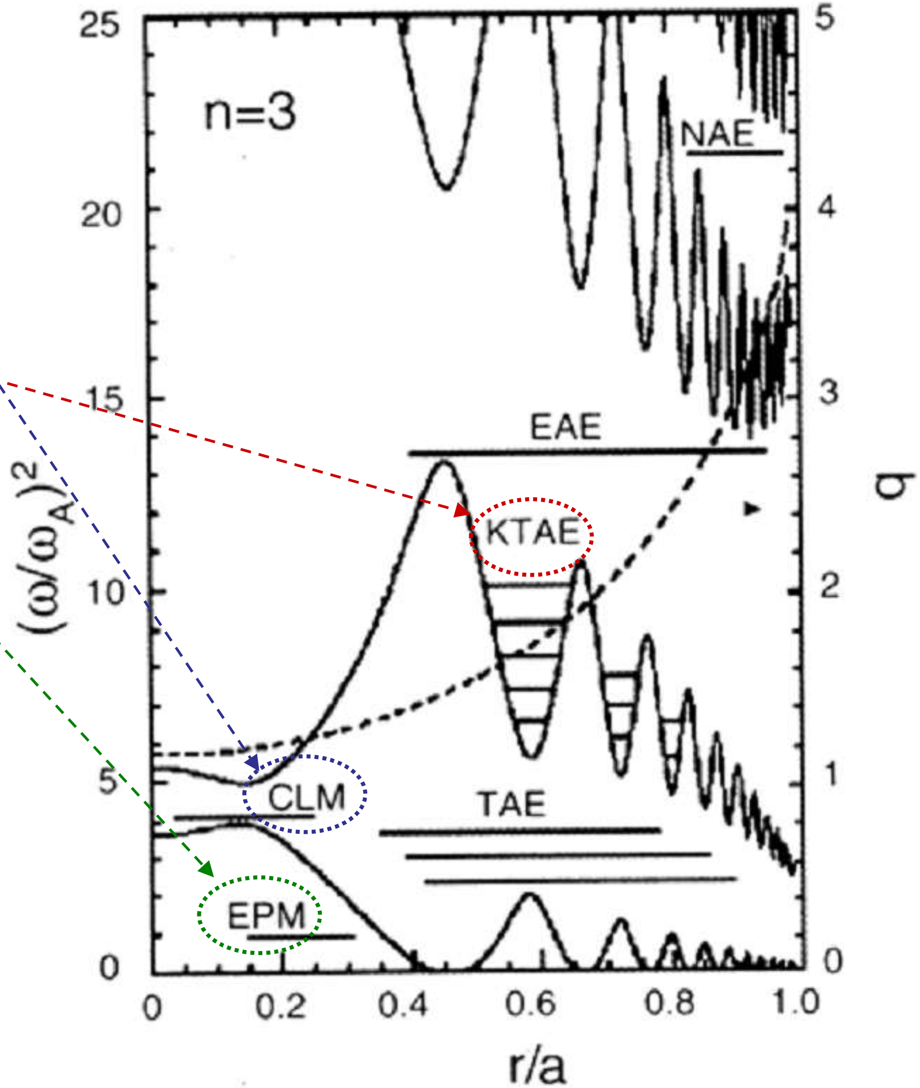
- TAE (Toroidal Alfvén Eigenmodes) ← toroidal effect
- EAE (Ellipticity Alfvén Eigenmodes) ← ellipticity effect
- NAE (Noncircular triangularity Alfvén Eigenmodes) ← triangularity



Miyamoto K. Plasma physics and controlled nuclear fusion

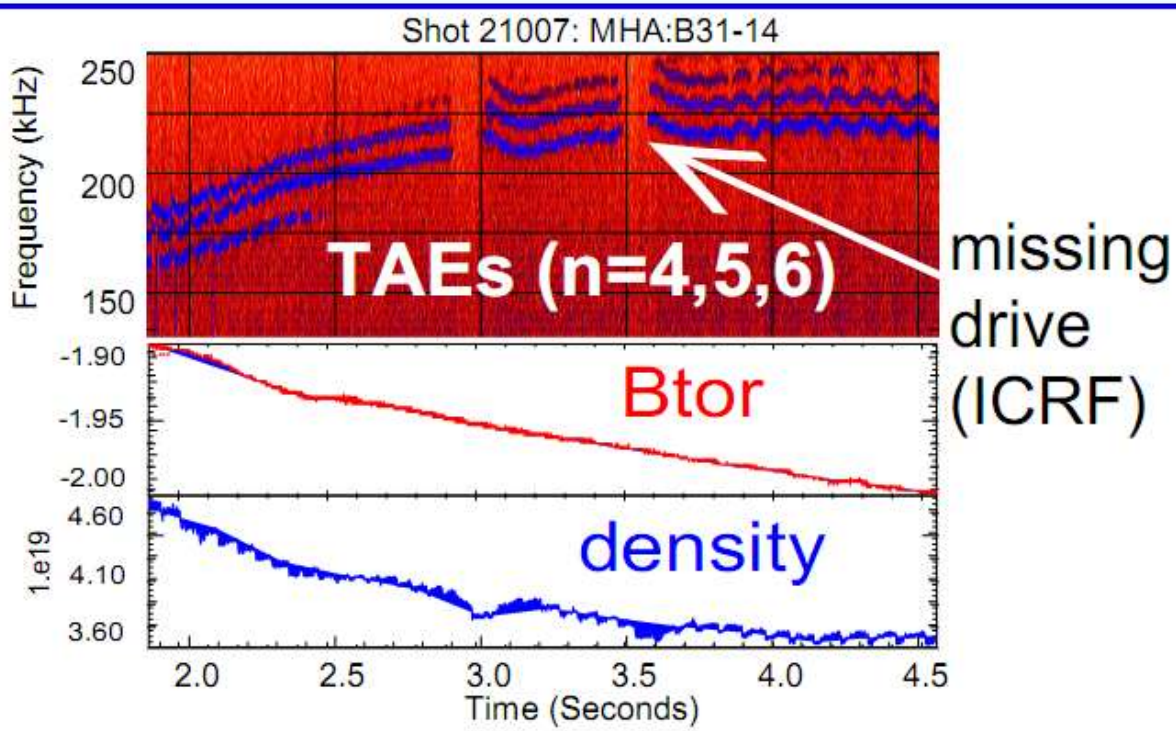
List of main fast particle modes (2)

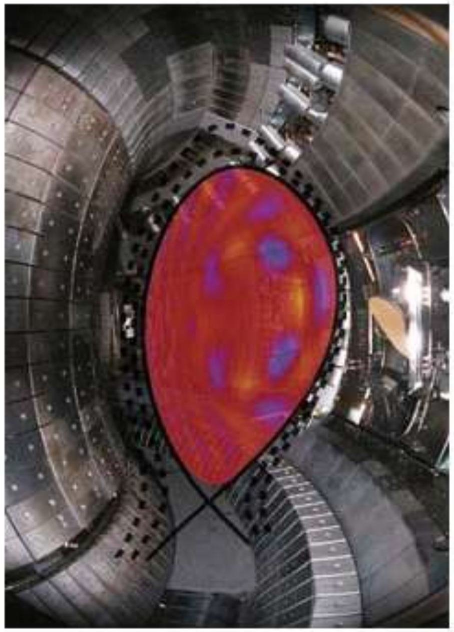
- CLM (Core Localized Modes) ← low shear version of TAEs
- KTAE (Kinetic TAE) ← kinetic effects (finite Larmor radius, etc.)
- EPM (Energetic Particle mode) ← population of energetic particles
- BAE (Beta induced Alfvén Eigenmodes) ← finite compressibility (coupling to sound waves)
- RSAE (Reversed Shear Alfvén Eigenmodes) ← reversed safety factor profile (advanced tokamak scenario)
- ……



Miyamoto K. Plasma physics and controlled nuclear fusion

TAEs at ASDEX-Upgrade (#21007, Mirnov coils)





[P.Lauber, IPP Colloquium, 2009]

Fast particles → collisionless excitation of the weakly damped modes
No fast particles → no drive → no fast particle modes

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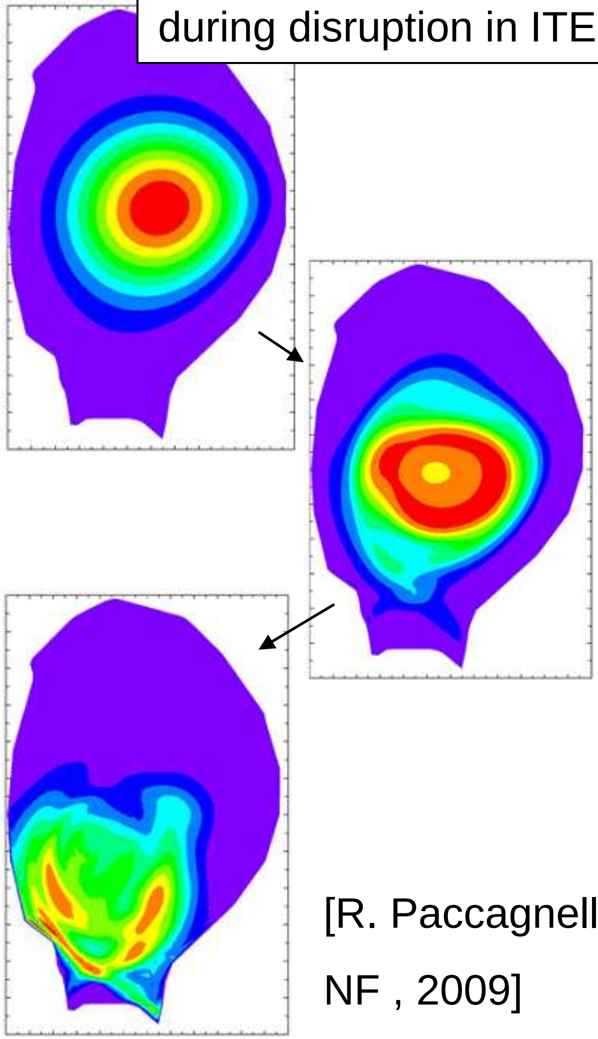
Disruption

Disruption is a sudden lost of plasma confinement.

Key issues to be resolved for disruptions:

- Forces
- Heat Loads
- Runaways
- Mitigation
- Prediction and avoidance

Temperature evolution during disruption in ITER



[R. Paccagnella, NF , 2009]

Disruption is a sudden lost of plasma confinement.

Key issues to be resolved for disruptions:

- Forces
- Heat Loads

Detailed modeling of the machines

- Runaways

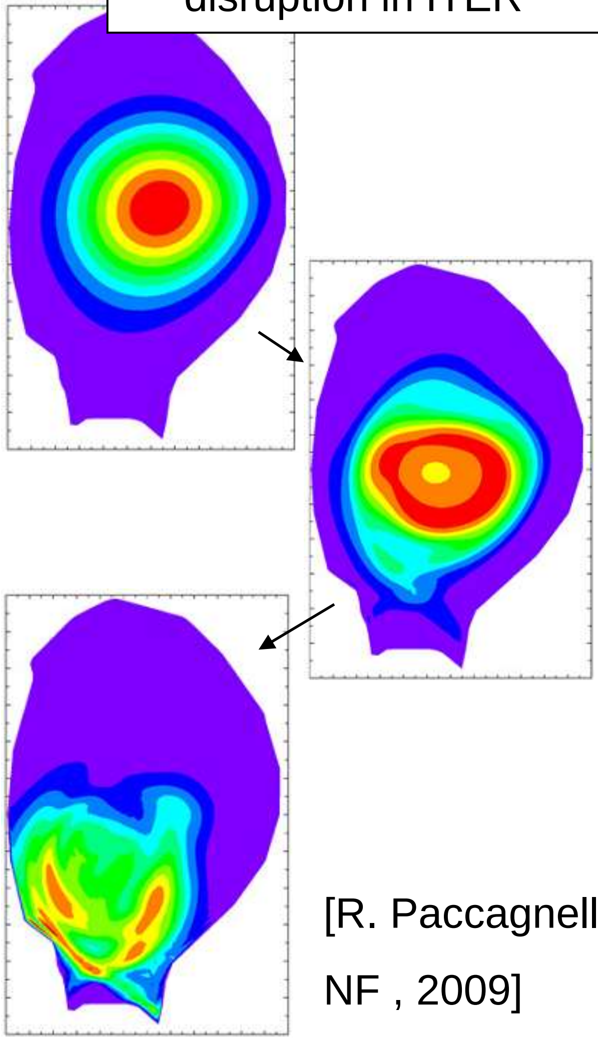
beam of high energetic electrons (massive gas injection, …)

- Mitigation

with killing pellets, with Electron Cyclotron Resonant Heating (ECRH),

- Prediction and avoidance

Temperature evolution, disruption in ITER



[R. Paccagnella, NF , 2009]

Disruption can be predicted rather accurate in modern tokamaks by using:

• different sensors (special magnetic coils for locking modes,…)

• neural network (here a set of disruptions is needed for NN training)

Problem: we can not make a set of disrupted pulses in ITER!

Possible solutions:

• transfer the data base for neural network from smaller tokamaks if possible

• improve modeling and predict disruption from physics side

Summary

A lot of activities in fusion labs are focused on solving MHD problems for ITER needs (experiments and modeling).

Several different approaches for the same problem are investigated simultaneously to have at least one working in ITER.
(for example: ELMs triggering with pellets, with RMPs and development of new scenarios)

Expertise from other types of fusion devices is used for ITER needs.
(for example: RWM controls from reversed filed pinch)

I think that at the start of ITER we would have better new scenarios and new control techniques for MHD instabilities.