Tokamak disruptions from an experimental and a theoretical perspective : what we know and what we don't know ; simulations achievements and existing gaps

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Tokamak disruptions represent a serious drawback for fusion magnetic confinement systems and for the development of a fusion reactor concept.

Nuclear fusion power plants require steady state operation of quiescent plasmas and **no disruptions at all are allowed**. In present tokamaks, however, disruptions are almost unavoidable especially for high performances plasmas conditions.

In these lectures I will present an overview of the known, open and critical issues, both from an experimental and a theoretical perspective.

I will mainly concentrate on the magneto-hydro-dynamical (MHD) aspects only briefly mentioning the important issues related to disruptions mitigation using gas injection systems and runaway electrons.

9th ITER International School 20-24 March 2017 Aix en Provence (France)

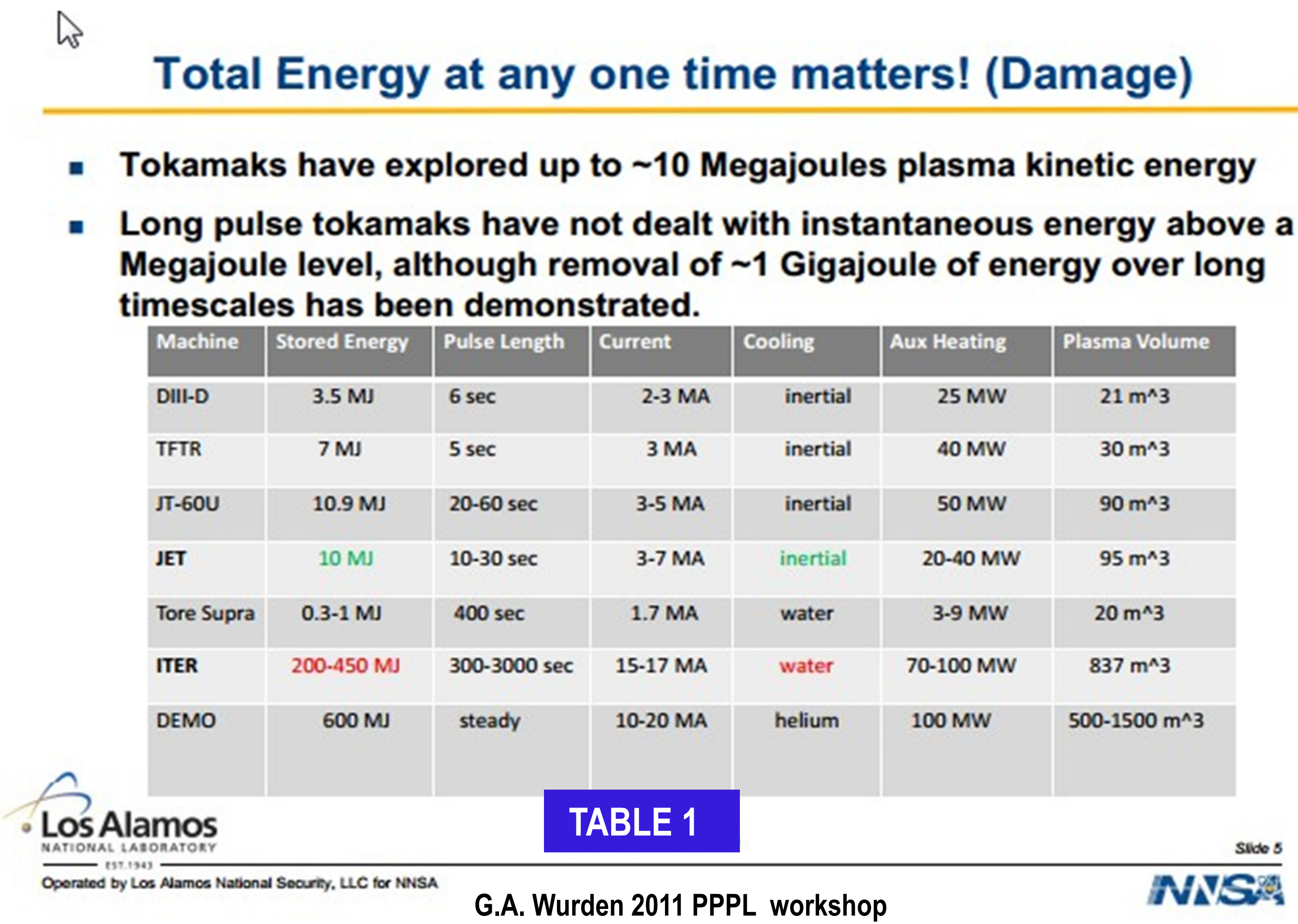
OUTLINE

- Introduction to disruption phenomenology
- Causes and effects of disruptions
- Equilibrium and vertical stability
- Symmetric and non symmetric halo currents
- Boundary conditions
- Hiro and surface currents
- Halo/hiro/eddy currents and flux conservation
- Current asymmetry rotation
- Virial Theorem and angular momentum
- Open Issues for ITER
- M3D simulations various results
- The mistery of the TQ
- FR scenarios and disruptivity
- Plasma rotation mistery
- Disruptions control and RMP
- Radiation and disruption mitigation
- Runaways electrons
- Conclusions

Theory vs experiments

I would like to live in Theory Country.. why?
Because in Theory everything works !

Stored Energy in actual and future devices



Total Energy at any one time matters! (Damage)

- Tokamaks have explored up to ~10 Megajoules plasma kinetic energy
- Long pulse tokamaks have not dealt with instantaneous energy above a Megajoule level, although removal of ~1 Gigajoule of energy over long timescales has been demonstrated.

Machine	Stored Energy	Pulse Length	Current	Cooling	Aux Heating	Plasma Volume
DIII-D	3.5 MJ	6 sec	2-3 MA	inertial	25 MW	21 m^3
TFTR	7 MJ	5 sec	3 MA	inertial	40 MW	30 m^3
JT-60U	10.9 MJ	20-60 sec	3-5 MA	inertial	50 MW	90 m^3
JET	10 MJ	10-30 sec	3-7 MA	inertial	20-40 MW	95 m^3
Tore Supra	0.3-1 MJ	400 sec	1.7 MA	water	3-9 MW	20 m^3
ITER	200-450 MJ	300-3000 sec	15-17 MA	water	70-100 MW	837 m^3
DEMO	600 MJ	steady	10-20 MA	helium	100 MW	500-1500 m^3

TABLE 1

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ITER and DEMO level of stored energy

How much energy are we talking about?

60 MJ of runaways, 400 MJ of thermal quench, 600 MJ of poloidal magnetic field energy

600 MJ will melt ~ one ton of copper

15 MJ is released by 7 sticks of TNT

10 GJoule ≅ A380 flying at 700 km/h

100 MJ: F-14 Tomcat launched by steam catapult

Melting point of copper: 1356 K
Specific heat capacity of copper: 385 Jkg^{-1}K^{-1}
Specific latent heat of fusion (energy required to convert a solid at its melting point into a liquid at the same temperature): 205000 Jkg-1
So to melt 1 kg of copper we need (1056*385 + 205000) J = 611,560 J.

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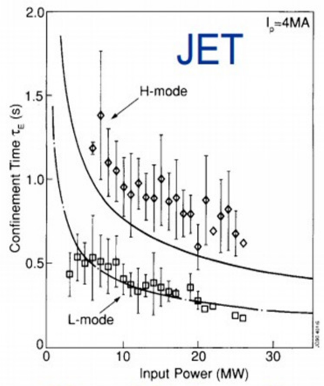
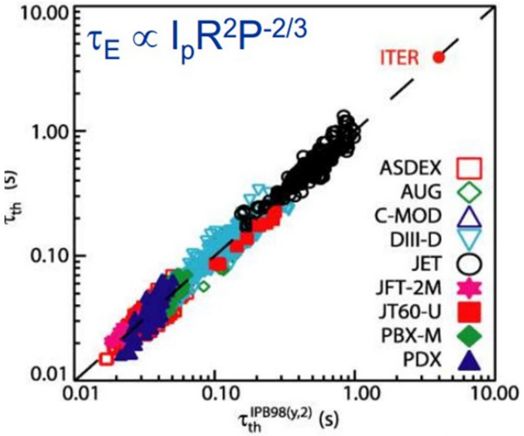
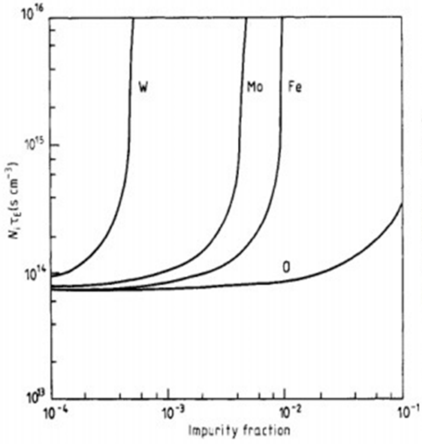
The tokamak plasma energy balance

$$\boxed{\frac{W}{\tau_E} = V_\phi\, I + P_{add} + P_{fus} - P_\Omega - P_{rad}}$$

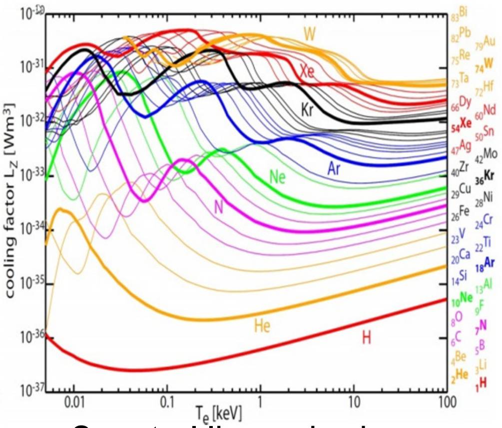


τ_E (H-mode) ~2 x τ_E (L-mode)





Required (n τ) vs ımp. content for break -even @ 10 KeV



Spectral lines+ brehms.

- The global (volumetric) energy balance is at the basis of the plasma confinement

- **A sudden non compensated deficit in this balance can lead to disruptions**

Key elements are (neglecting convection losses):

τ_E : the plasma energy confinement time

V_ϕ I : the transformer ohmic input power (→ 0)

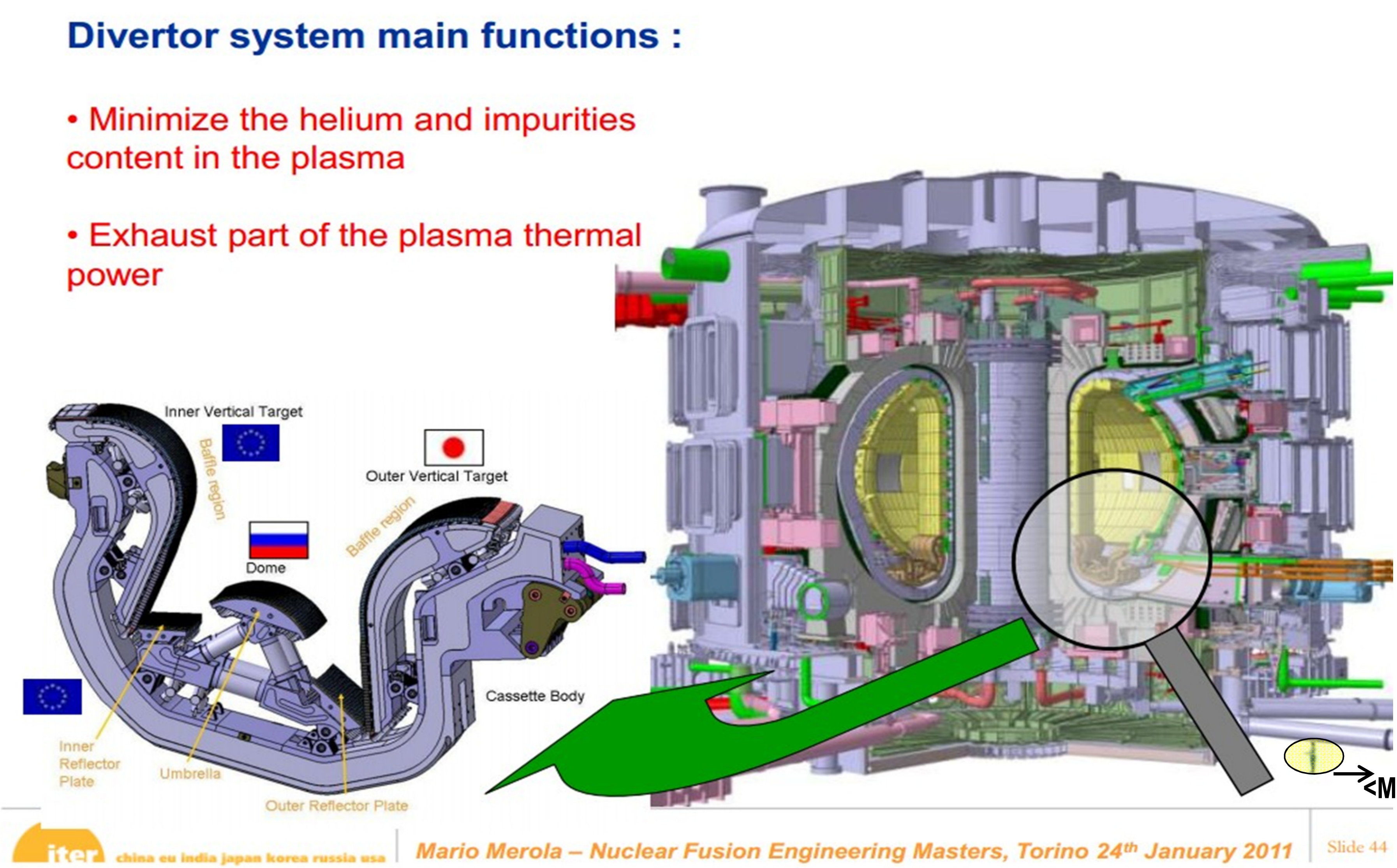
P_{fus}: fusion power ($V\, n_d n_T < \sigma > E_{fus}$)

P_{add} : the additional heating power

P_Ω : the plasma ohmic dissipation ($\approx V\, J^2 Z\, T^{-3/2}$)

P_{rad} : $\mathrm{V}\, \Sigma\, n\, n_z L_z$ (with $\mathrm{Brehms} \approx n^2 T^{1/2}$)

ITER Machine and Divertor System



Divertor system main functions :

- Minimize the helium and impurities content in the plasma
- Exhaust part of the plasma thermal power

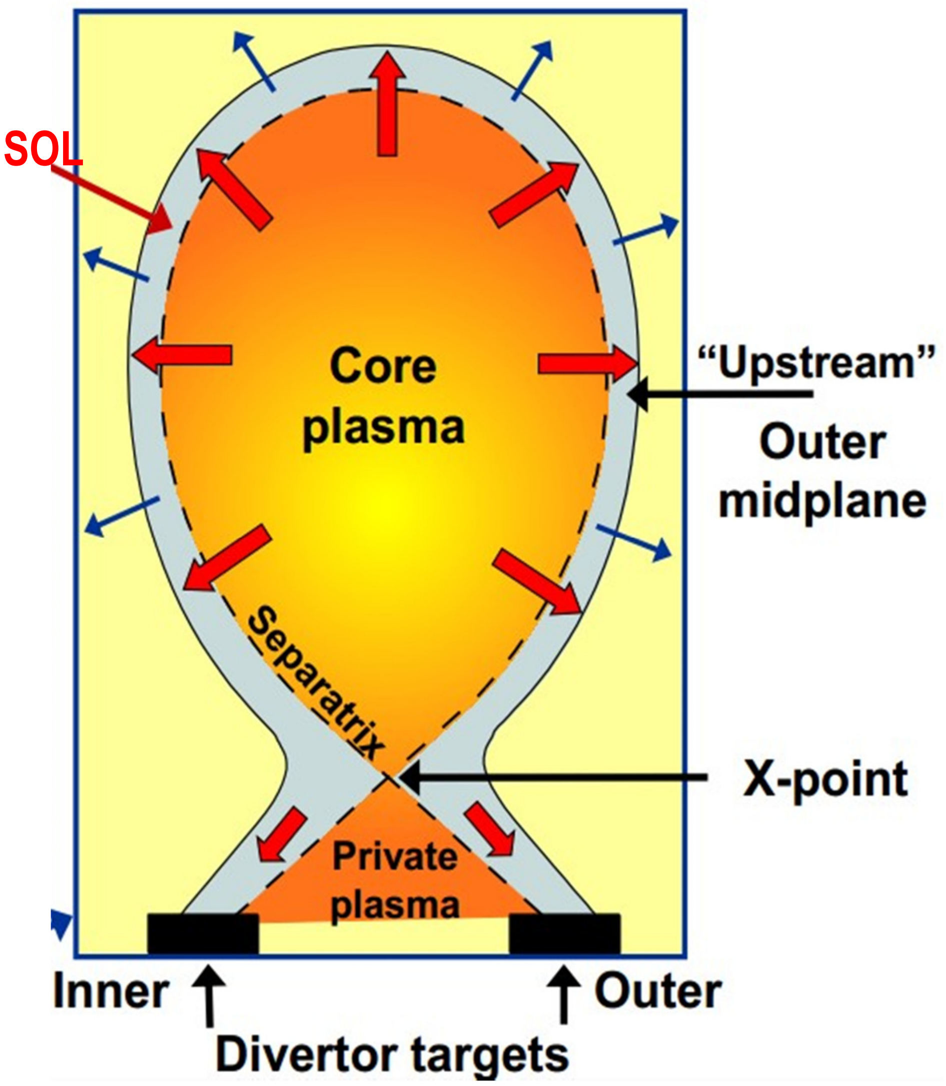
iter china eu india japan korea russia usa

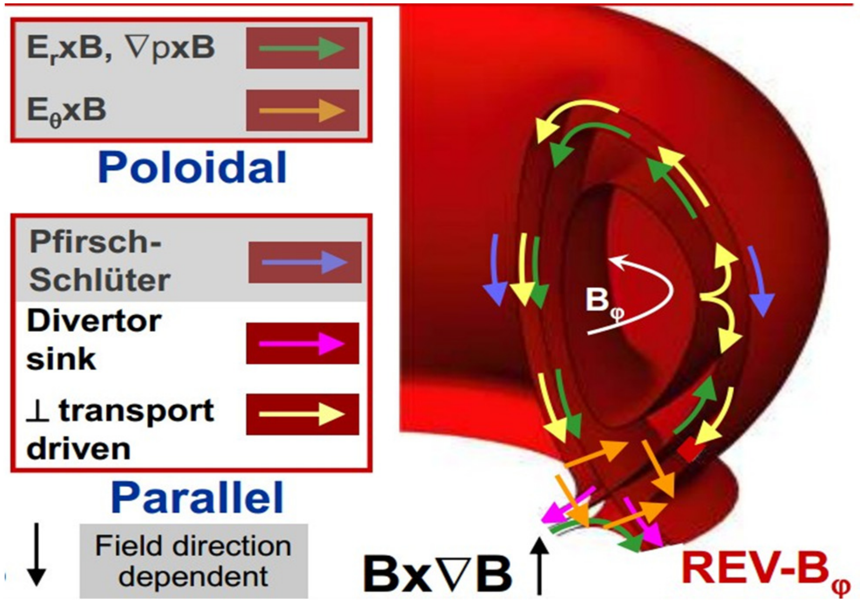
Mario Merola – Nuclear Fusion Engineering Masters, Torino 24th January 2011

Slide 44

<MAN>

Divertor and SOL Layer convective losses

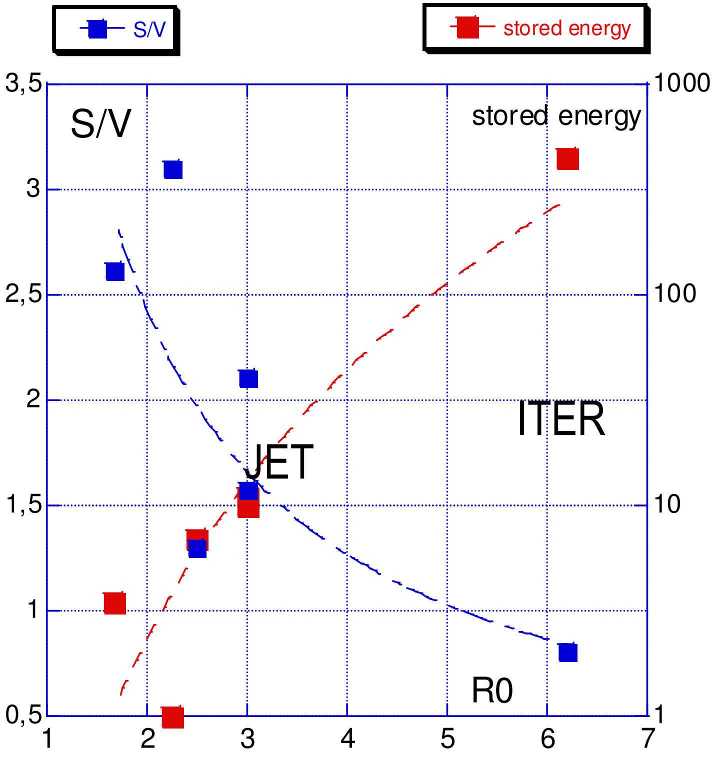




- To mantain a **clean plasma** and to **limit** the **plasma wall interactions** diverted open magnetic field are created in the SOL layer where radiation and convective losses are the main sinks of energy

- main disadvantage is the **limited** divertor **plates surface**

The unfavourable Surface to Volume ratio of the torus

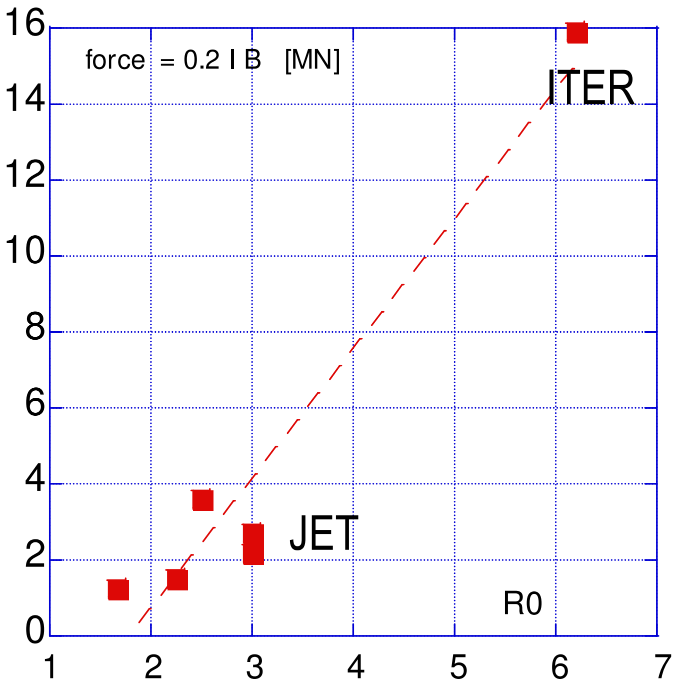


From **TABLE 1** data

One issue which is not well addressed, in my opinion, is that **the S/V ratio scales unfavourably with R**

- the **neutrons per unit area increase with R** (since the number of neutrons is proportional to the plasma volume)
- the **divertor area** is in any case a **fraction of the total surface** also the **thermal load per unit area increases with R** both at the **divertor plates** and also in general **on the entire wall**

The unfavourable scaling of disruption forces



From **TABLE 1** data

- **Assuming that the forces on the structures** (later shown to be reasonable) **scale like** :

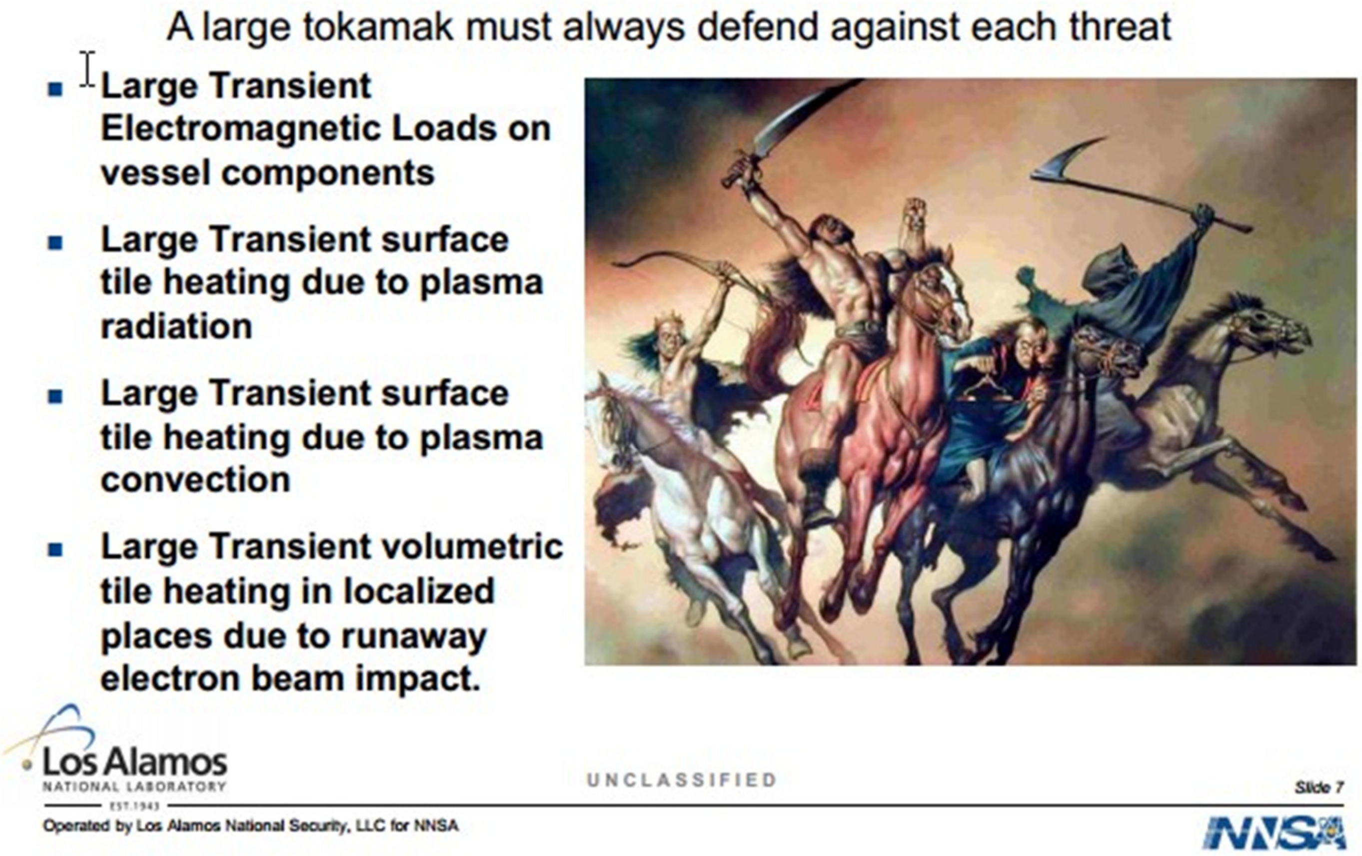
  $F = \alpha \, I_p B_\phi$ with $\alpha = 0.2$

  there is **almost a factor of 10** between actual experiment and ITER

- This simple (but realistic) assumption also show that the **scaling to larger current or magnetic field devices** is quite unfavourable in case of disruptions

What is a tokamak disruption ?

It is a **SUDDEN RELEASE** of this stored internal energy that produces **4 main consequences :**



A large tokamak must always defend against each threat

- **Large Transient Electromagnetic Loads on vessel components**
- **Large Transient surface tile heating due to plasma radiation**
- **Large Transient surface tile heating due to plasma convection**
- **Large Transient volumetric tile heating in localized places due to runaway electron beam impact.**

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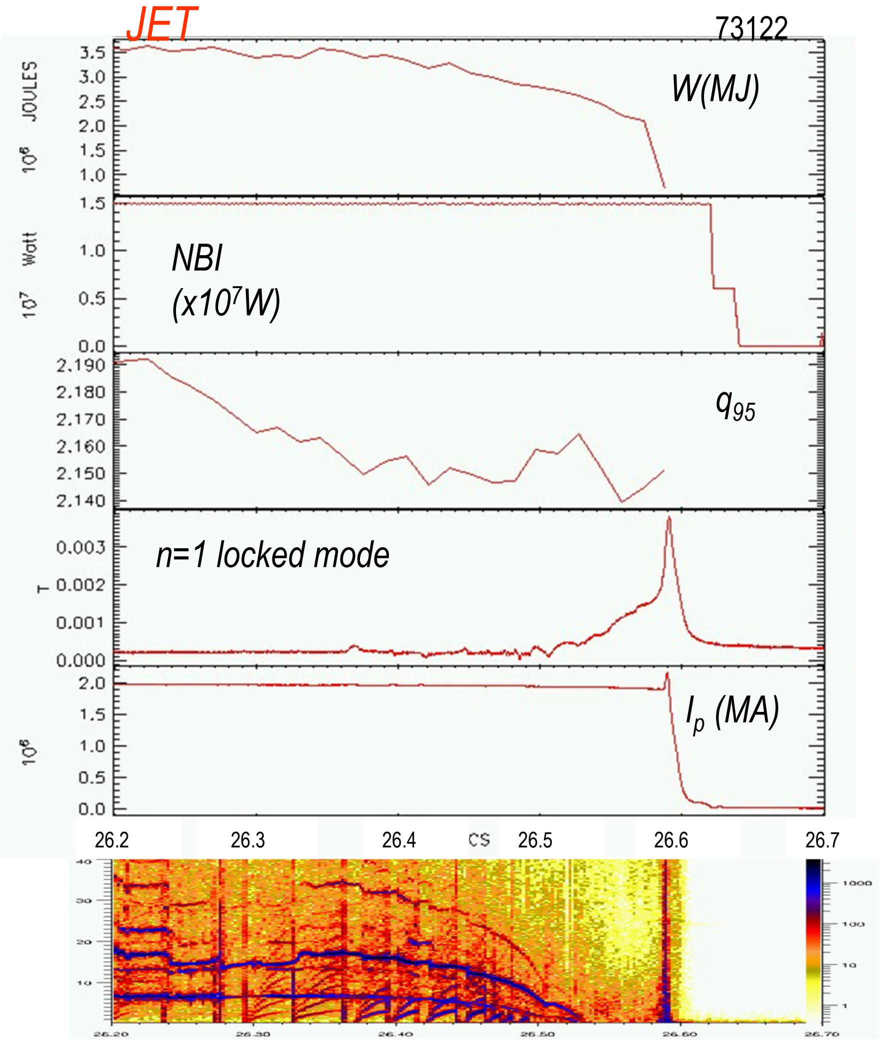
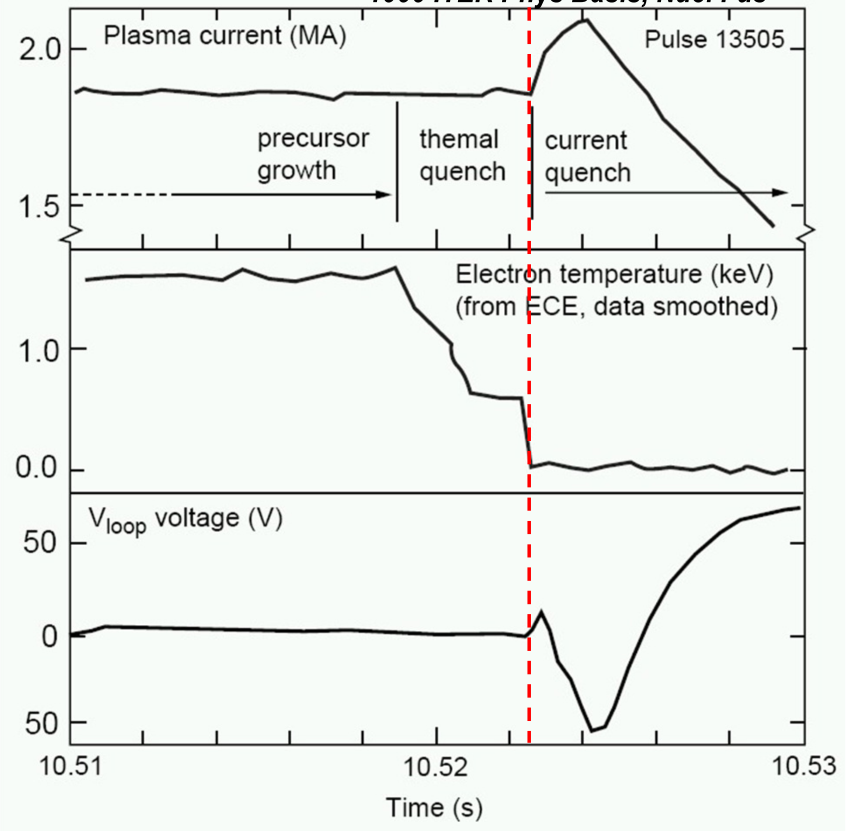
Which is the «typical» disruption phenomenology ?



- *Pre-disruption energy loss, precursors*

T. Hender 2010 CCFE workshop

1999 ITER Phys Basis, Nucl Fus



- ***Thermal quench and current quench***
- ***Consequences heat + EM loads, VDE, halos, runaways***

Which are the important characteristic parameters ?

- ***The Thermal quench (TQ) and current quench (CQ) times : tq and tc***

These times determine the power losses. Generally tq << tc with ***tq*** *of the order of* ***few ms*** *and* ***tc*** *of the order of* ***few tens ms***

In turn ***the shorter these times the stronger the effects*** *of the heat deposition (****melting of plasma facing component****) and of the electromagnetic consequences (****induced eddy currents and stresses on metallic structures****), and available energy to* ***accelerate electrons*** *(****R****unaway* ***E****lectrons).*

*(****too long CQ*** *times can also be an issue -> RE & momentum impulse)*

Moreover the ***TQ*** *can produce the loss of the vertical stability and induces the so called* ***Vertical Displacement Events*** *(****VDEs****) and the generation of large* ***plasma edge halo currents***

- ***Avoidance/Mitigation actions are therefore required***

How to explain the Voltage spike and current behavior ?

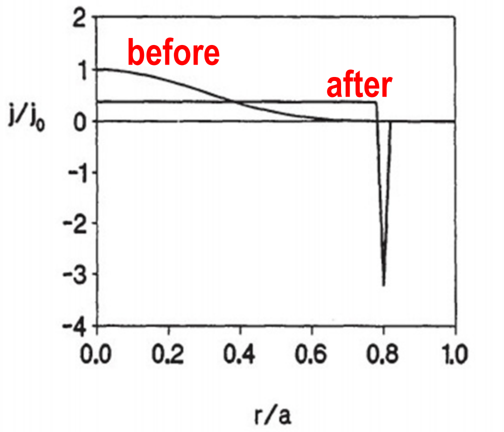
From Wesson et al NF (1990)

$$\boxed{V = -L_v \frac{d}{dt}(I_p + I_v)}$$ with $I_v = \frac{V}{R}$ $\longrightarrow$ $$\boxed{V = -R\exp(-(R/L_v)t)\int_0^t \exp((R/L_v)t')\,\frac{dI_p}{dt}\,dt'}$$

therefore a negative Voltage implies a positive current derivative. On the other hand an internal instability **flatten the current profile** and decreases the internal inductance (next slide).
If we assume that only a **fraction f** of the internal energy goes in increasing the current (and not dissipated), we have:

$$\boxed{\frac{1}{2}L_p\frac{dI_p^2}{dt} = -f\,\frac{1}{2}I_p^2\,\frac{dL_p}{dt}}$$

Therefore a **decrease of the internal inductance** can explain the **plasma current increase**.



To **explain the delay** between the **TQ** and the **current increase** a **negative current spike diffusion process** is also invoked (similar to the **surface current model** to be discussed later)

Inductance vs current peaking

From Stacey:Fusion Plasma Physics (Wiley 2012)

$$p = p_0\left(1 - \frac{r^2}{a^2}\right)$$

$$j = j_0\left(1 - \frac{r^2}{a^2}\right)^{\nu}$$

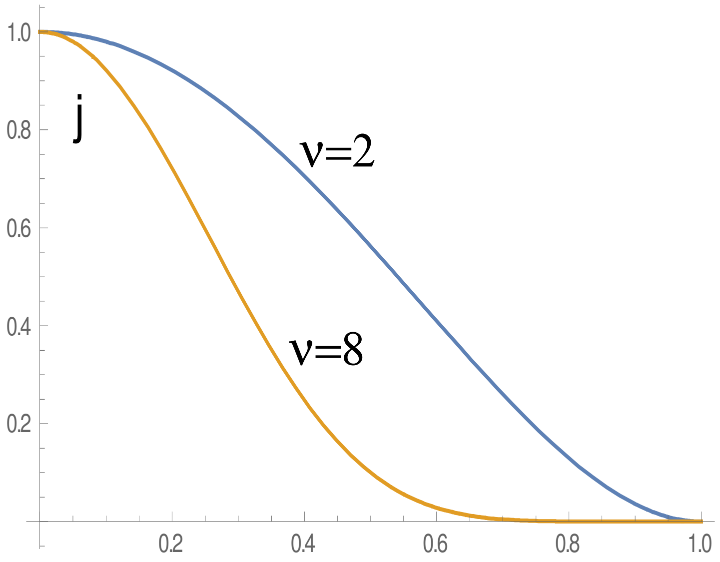
→

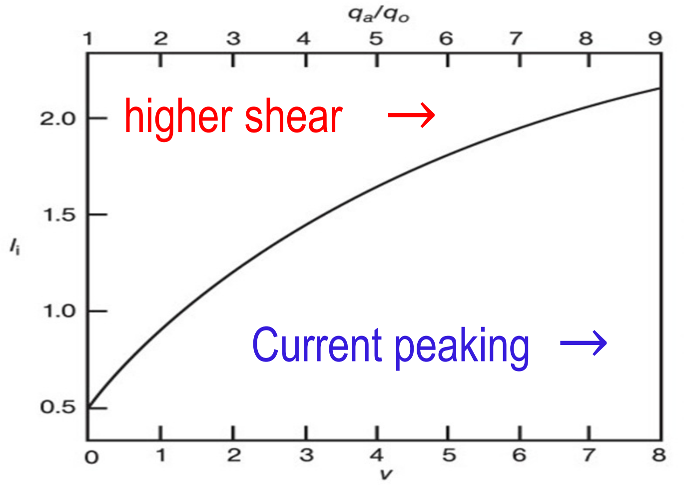
$$l_i = \frac{\bar{B}_\theta}{B_{\theta a}^2} = \frac{2\int_0^a B_\theta^2 r\,\mathrm{d}r}{a^2 B_{\theta a}^2}$$

$$l_i = \ln(1.65 + 0.89\nu)$$

and

$$q_a/q_0 = \nu + 1$$





higher shear →

Current peaking →

What causes disruptions ?

from P. de Vries et.al. NF **51** (2011) 53018

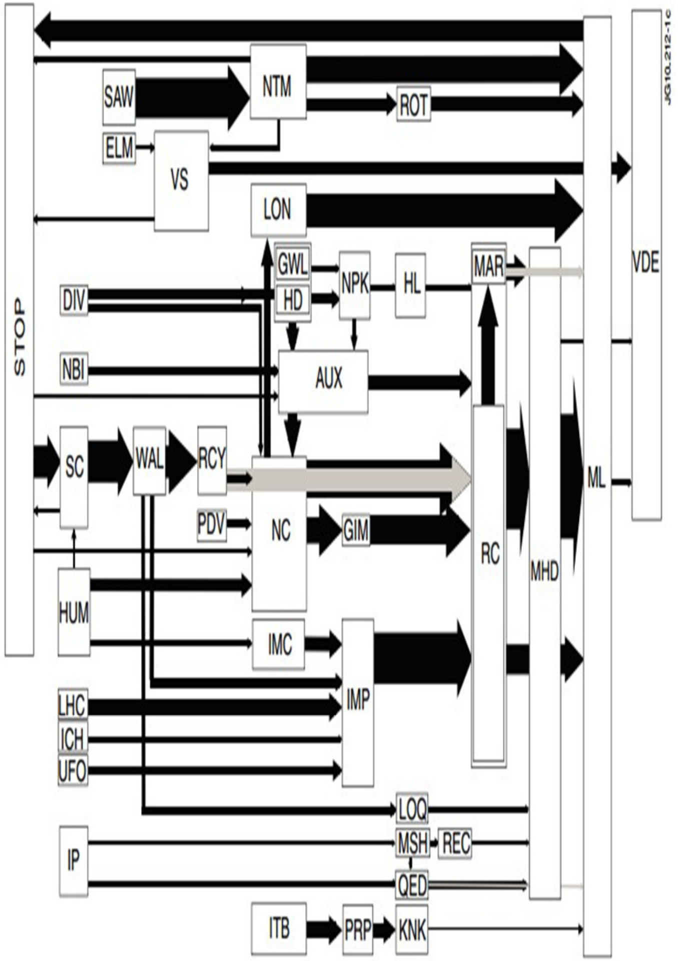


Figure 4. A schematic overview, showing the statistics of the sequence of events for 1654 unintentional disruptions at JET

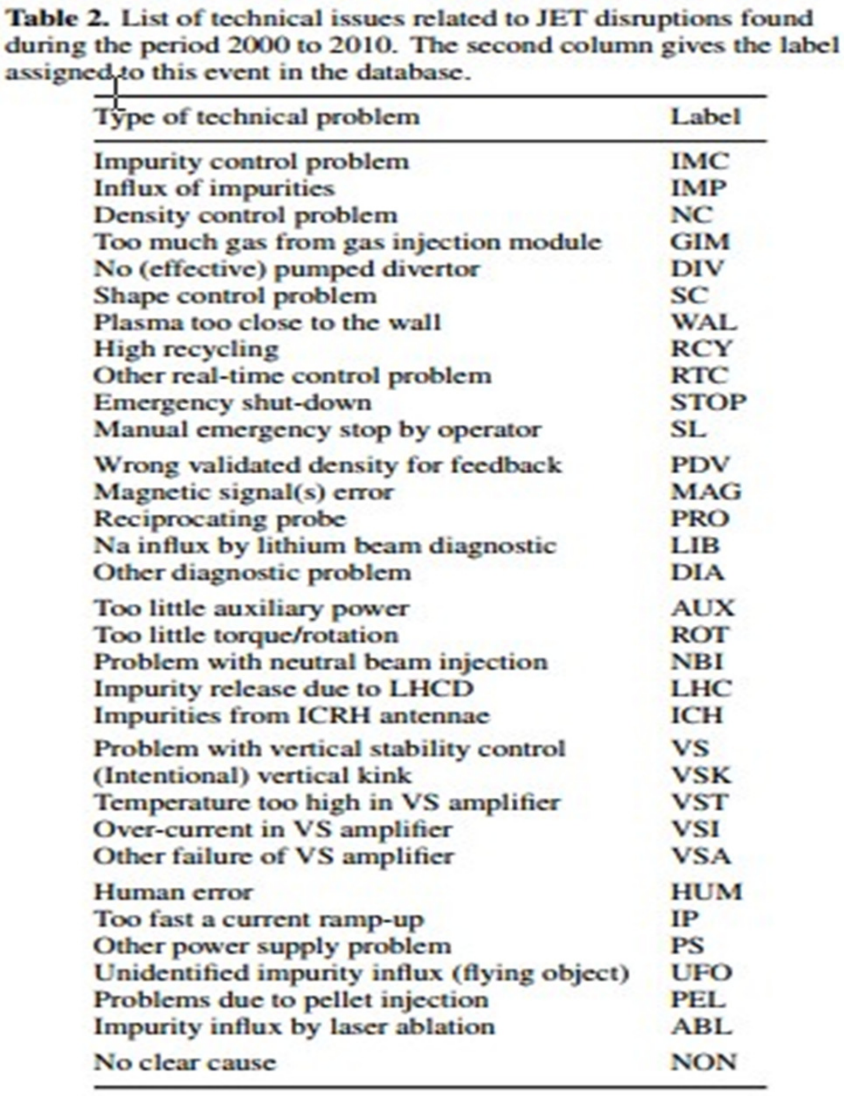


Table 2. List of technical issues related to JET disruptions found during the period 2000 to 2010. The second column gives the label assigned to this event in the database.

Type of technical problem	Label
Impurity control problem	IMC
Influx of impurities	IMP
Density control problem	NC
Too much gas from gas injection module	GIM
No (effective) pumped divertor	DIV
Shape control problem	SC
Plasma too close to the wall	WAL
High recycling	RCY
Other real-time control problem	RTC
Emergency shut-down	STOP
Manual emergency stop by operator	SL
Wrong validated density for feedback	PDV
Magnetic signal(s) error	MAG
Reciprocating probe	PRO
Na influx by lithium beam diagnostic	LIB
Other diagnostic problem	DIA
Too little auxiliary power	AUX
Too little torque/rotation	ROT
Problem with neutral beam injection	NBI
Impurity release due to LHCD	LHC
Impurities from ICRH antennae	ICH
Problem with vertical stability control	VS
(Intentional) vertical kink	VSK
Temperature too high in VS amplifier	VST
Over-current in VS amplifier	VSI
Other failure of VS amplifier	VSA
Human error	HUM
Too fast a current ramp-up	IP
Other power supply problem	PS
Unidentified impurity influx (flying object)	UFO
Problems due to pellet injection	PEL
Impurity influx by laser ablation	ABL
No clear cause	NON

What causes disruptions ? an old issue !

"Disruptions in Tokamaks", F. C. Schueller, Plasma Phys. Control. Fusion 37 A135 (1995). Proceedings of ITER Workshop on Disruptions and VDE's, Garching, March 13-17, 1995.

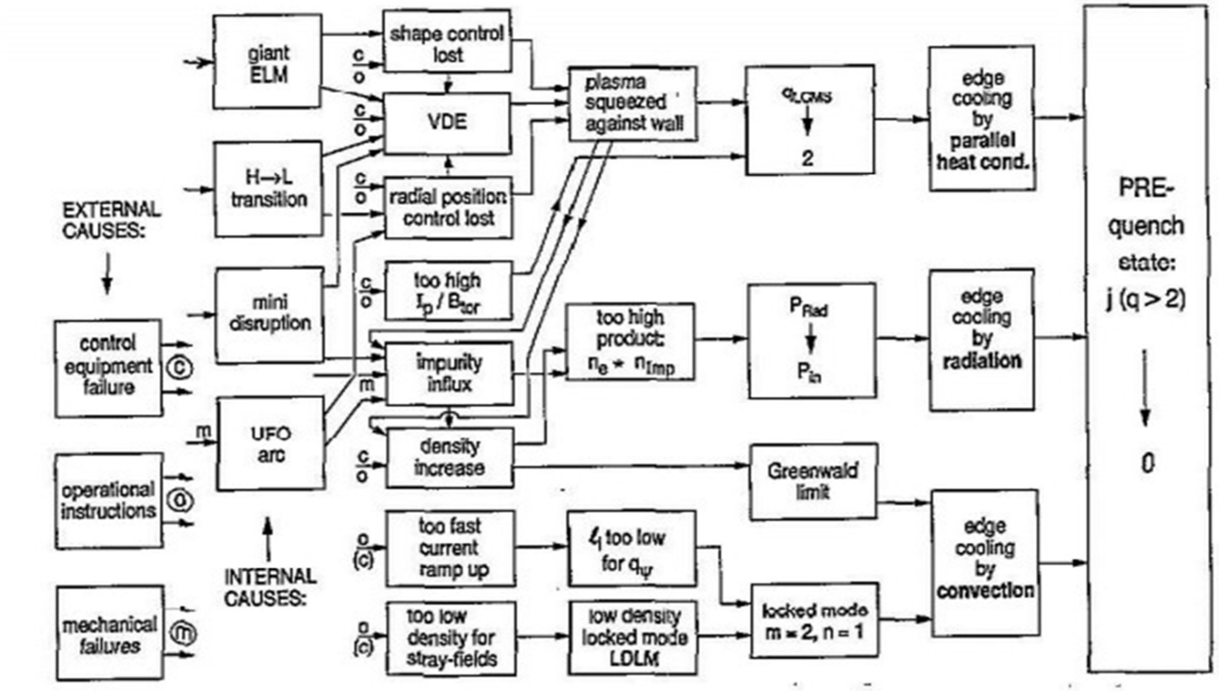


Figure 12. A scheme of possible initiating events and precursor scenarios leading to a prequench state with deficient edge.

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Disruptions effects

June 2008 Alcator C-Mod, in-vessel inspection

localized melt damage most likely due to runaways

Melt damage at upper edges

“Far away” diagnostic harness burned/melted by runaways

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Disruption effects in JET ILW

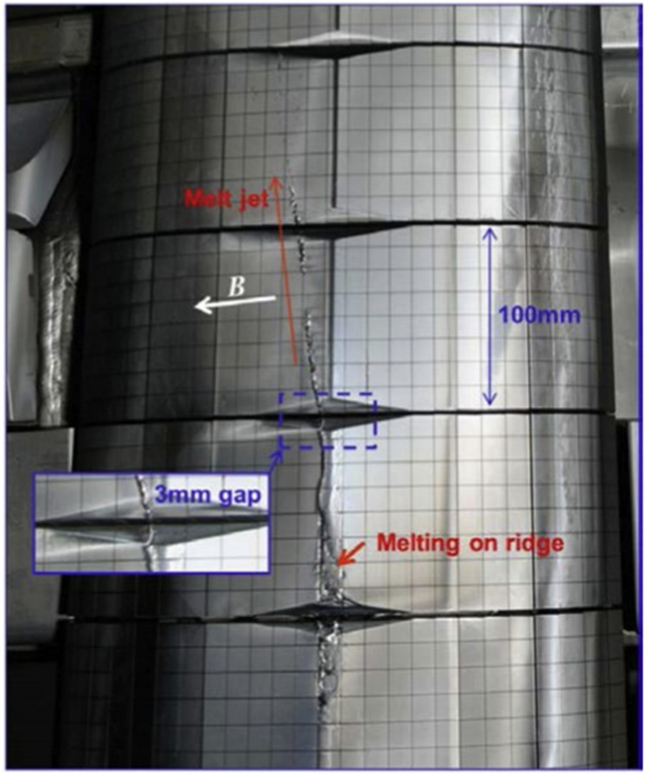


Figure 1. Bulk beryllium melting on the ridge of the JET inner wall limiter (4X).

From Matthews et al, Phys. Scripta (2016)

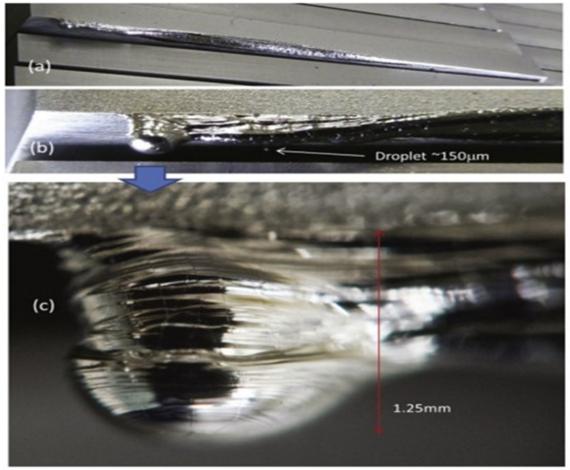
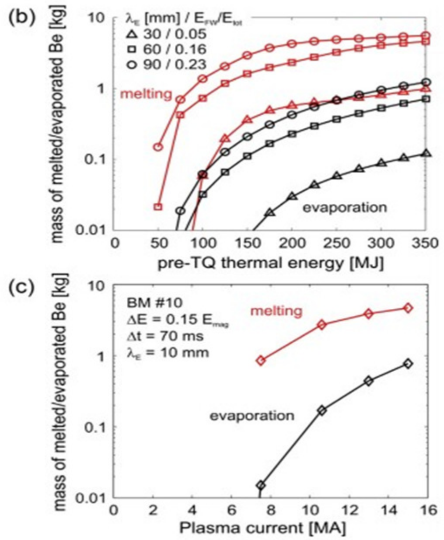
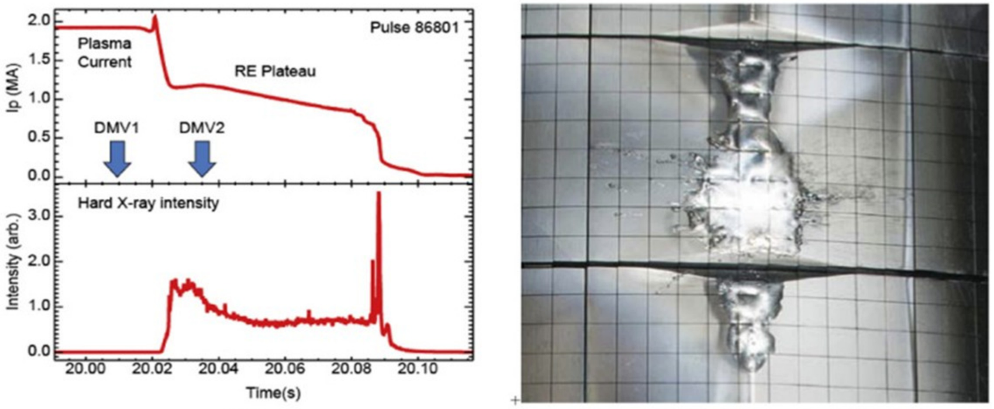


Figure 6. (a) Image of the melted edge of the special tungsten lamella. The lamellas are 5.5 mm wide and 60 mm long. (b) Detail of layering of the migrated material and a small ~150 μm diameter droplet adhered to the side the lamella. (c) Higher resolution image showing layering and cracking of the main droplet.

From Lehnen et al JNM (2015)

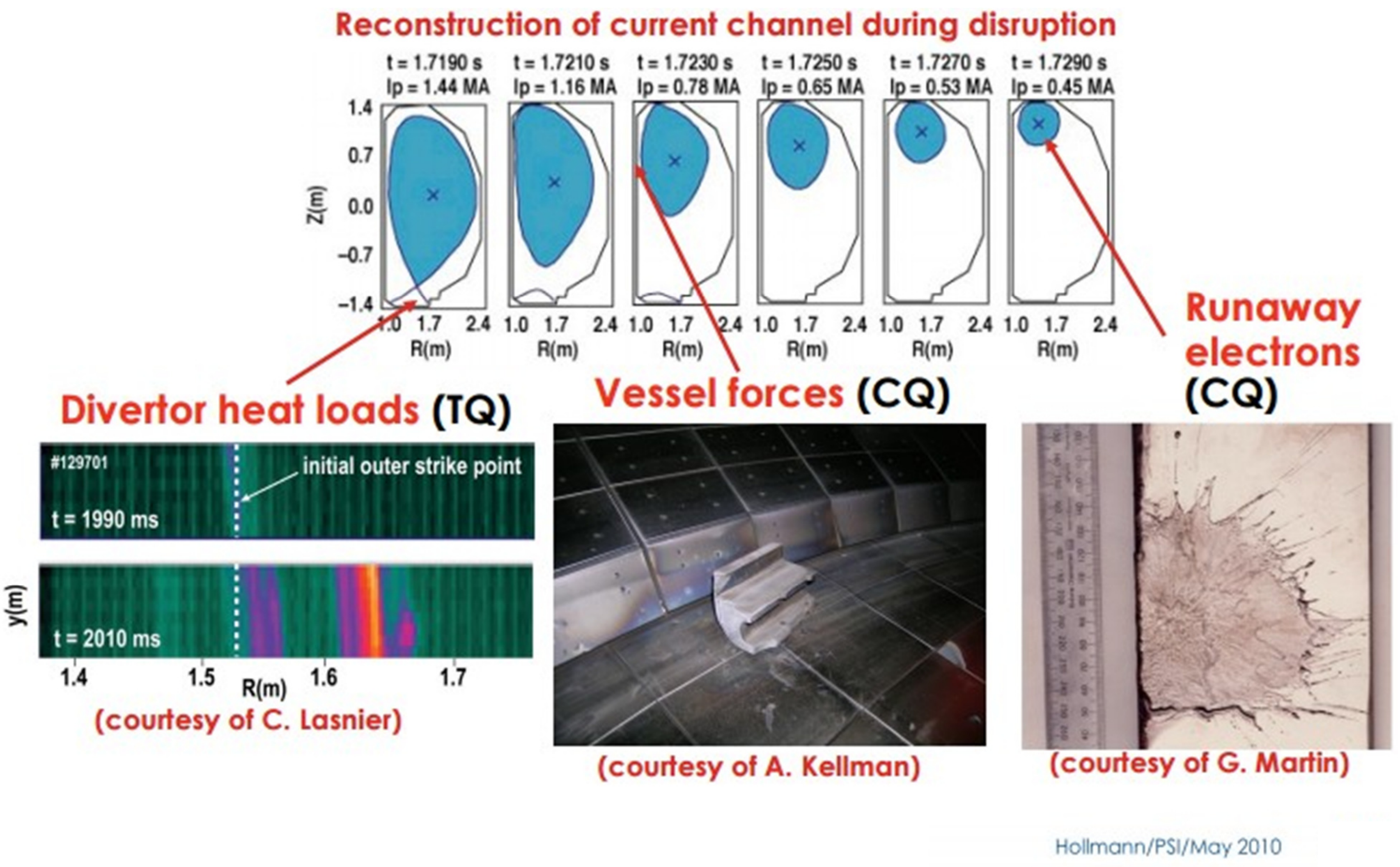




(a) (b)

Figure 4. (a) Plasma current versus time for JET pulse #86801 in which a runaway electron (RE) plateau characterized by hard x-ray emission is produced when argon is injected by DMV1(4.7 bar l). More argon is injected by DMV2(12.7 bar l) in an unsuccessful attempt to mitigate the REs (b) in-vessel image of melt damage due runaway electrons from pulse #86801 in which REs hit the tops of the inner wall limiters about 60 ms after they are created. The castellations are 12 mm^2.

Disruption effects



Reconstruction of current channel during disruption

Divertor heat loads (TQ)

(courtesy of C. Lasnier)

Vessel forces (CQ)

(courtesy of A. Kellman)

Runaway electrons (CQ)

(courtesy of G. Martin)

Hollmann/PSI/May 2010

Example: Tokamak equilibrium and disruptions

To maintain a tokamak plasma in equilibrium the following equation should be satisfied:

$$\boxed{\boldsymbol{J} \times \boldsymbol{B} = \nabla p}$$

where J and B are the current and magnetic fields in the plasma region and p is the plasma Pressure.

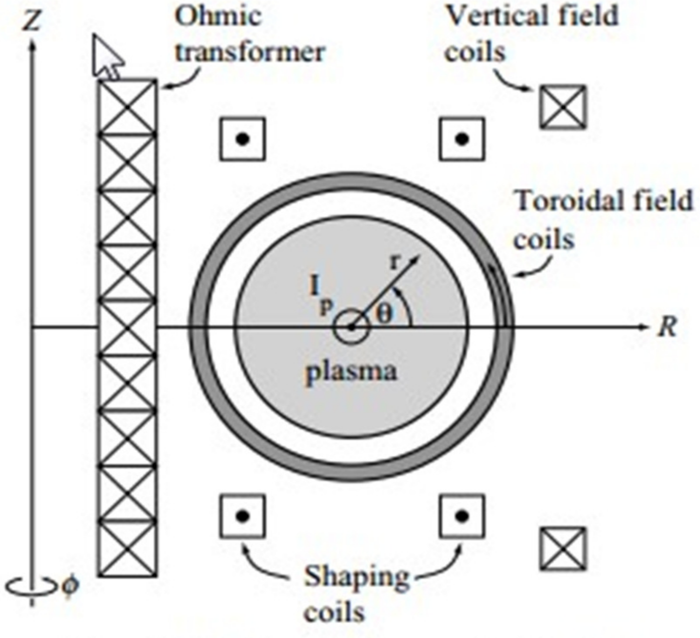


Figure 13.22 Schematic diagram of a tokamak.

However a plasma equilibrium is not possible without external currents !

Theorem :

A magnetofluid cannot stay in MHD equilibrium by forces generated only by its own internal currents

A tokamak equilibrium needs external currents !

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The reason is very fundamental :
it is related to the so called **VIRIAL THEOREM**

Starting from the equation of motion written in conservative form: $\boxed{\frac{\partial \rho \boldsymbol{V}}{\partial t} = -\nabla \boldsymbol{T}}$

where **V** is the magnetofluid velocity and **T** the stress tensor. It can be shown that:

$\boxed{\frac{\mathrm{d}\mathcal{I}}{\mathrm{d}t} = \int_{\mathcal{V}} \rho \mathbf{V} \cdot \mathbf{r} \,\mathrm{d}\mathcal{V}}$ is the moment of inertia and :

$$\begin{aligned} \frac{\mathrm{d}^2\mathcal{I}}{\mathrm{d}t^2} &= \int_{\mathcal{V}} \left(\rho V^2 + 3p + \frac{B^2}{8\pi} - \frac{(\nabla\phi)^2}{8\pi G} \right) \mathrm{d}\mathcal{V} \\ &= 2\mathcal{E}_V + 3(\gamma - 1)\mathcal{E}_p + \mathcal{E}_B + \mathcal{E}_g \end{aligned}$$

where

- Kinetic energy: $\mathcal{E}_V \geq 0$
- Internal energy: $\mathcal{E}_p \geq 0$
- Magnetic energy: $\mathcal{E}_B \geq 0$
- Gravitational energy: $\mathcal{E}_g \leq 0$ (only possible negative term!)

- In an equilibrium, this expression must equal zero:

$$0 = 2\cancelto{0}{\mathcal{E}_V} + 3(\gamma - 1)\mathcal{E}_p + \mathcal{E}_B + \cancelto{0}{\mathcal{E}_g} \quad > \ 0\ !$$

no surface stesses: $0 = -\oint_S \mathrm{d}\mathbf{S} \cdot \mathbf{T} \cdot \mathbf{r}$

Vertical stability of an elongated (why ?) tokamak (1)

- Distance to go around poloidally is larger

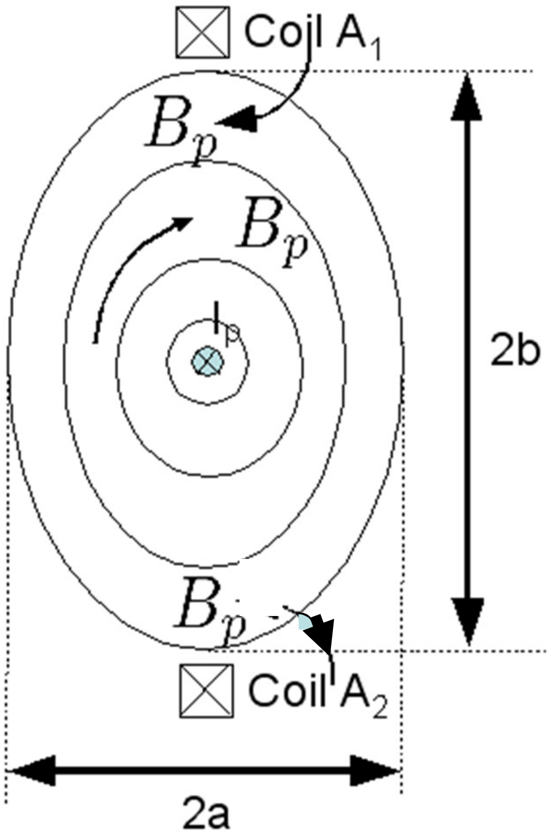
$$q = \frac{2\pi r^2 B_t}{\mu_0 R I} = \frac{2 A B_t}{\mu_0 R I}$$

$$A = \pi ab = \pi a^2 \kappa \quad \kappa = \frac{b}{a}$$

For the same plasma current:

$$\boxed{q_{\mathrm{elip}} = q_{\mathrm{circ}} \kappa}$$

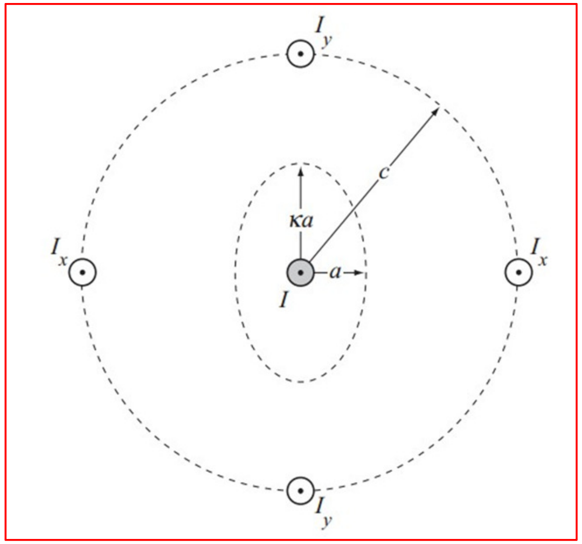
- **If q = 3-4 is the stability limit of operation** one can run a larger current in an elliptically shaped plasma
- ..also easily to be **DIVERTED**



from www2.warwick.ac.uk

Vertical stability of an elongated tokamak (2)

FromJ. Freidberg : Plasma Physics and Fusion Energy , ed. Cambridge (2007)



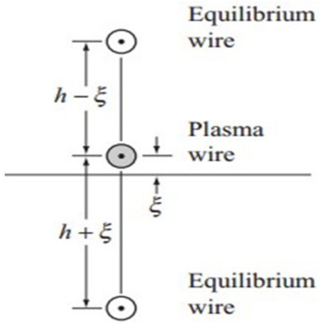
Considering a wire model (as in figure) and an elongated plasma kept in equilibrium by $I_{x,y}$.
Imposing at the plasma boundary: $A_z(0,\kappa a) = A_z(a,0)$

Taking into account that : $A_{z,j} = \frac{\mu_0 I_j}{2\pi}\ln(r_j)$ with r_j the radial distance from each wire to the surface and summing up :

$$I_y \ln\left(\frac{c^2 - \kappa^2 a^2}{c^2 + a^2}\right) + I_x \ln\left(\frac{c^2 + \kappa^2 a^2}{c^2 - a^2}\right) - I \ln\kappa = 0.$$

and for c >>a :

$$\boxed{I_x - I_y = \frac{c^2}{a^2}\frac{\ln\kappa}{1+\kappa^2} I.}$$



Calculating the forces between wires as: $\boxed{\mathbf{F}_{ij} = -(\mu_0 L I_i I_j / 2\pi r_{ij})\mathbf{e}_{ij}}$

where L is the length and $\mathbf{e}_{ij}$ is the radial versor pointing from wire i to j

Therefore the force on the plasma wire is :

$$\boxed{F_y = \frac{\mu_0 L I}{2\pi}\left(-\frac{I_y}{c-\xi} + \frac{I_y}{c+\xi} + 2\frac{\xi I_x}{(c^2+\xi^2)^{1/2}}\right)}$$

Vertical stability of an elongated tokamak (3)

By linearizing: $\boxed{\delta F_y = \frac{\mu_0 L I^2}{\pi}\left(\frac{I_x - I_y}{I}\right)\frac{\xi}{c^2}}$

The condition for stability is (restoring force) : $\delta F_y < 0$

and finally: $\boxed{\frac{\ln \kappa}{1+\kappa^2} < 0}$

(remembering the relation for c>>a for $I_{x,y}$)

Therefore $\kappa > 1$ is always **UNSTABLE VERTICALLY.** (if κ<1 UNSTABLE HORIZONTALLY)

Hence it is clear that an **active control** is needed to maintain the plasma **STABLE**.

Any failure in the control system or any sudden change in plasma shape or internal conditions

can result in a loss of the control and therefore can produce a:

VerticalDisplacementEvent (VDE) and a plasma disruption.

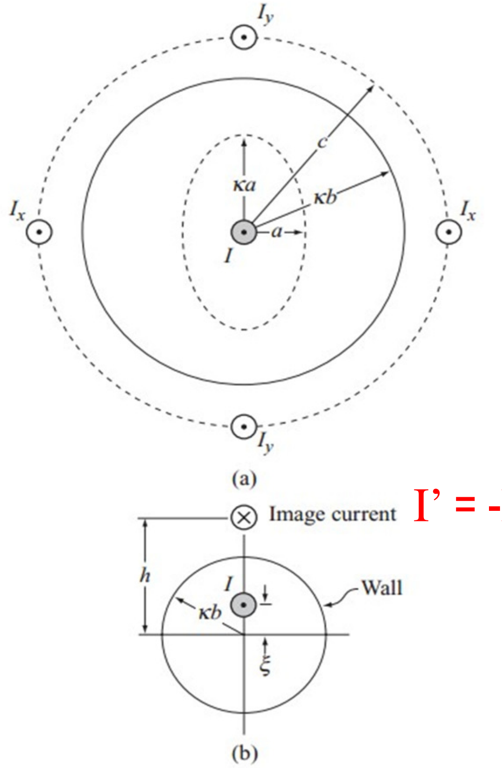
Vertical stability of an elongated tokamak with a perfectly conducting wall (4)

(Assuming that the field of the wires has penetrated the wall before it becomes Ideal !)

Assuming : $h = \frac{\kappa^2 b^2}{\xi}$ (in this way the wall is a flux surface)

In presence of a wall eddy current the force becomes:

$$\boxed{\delta F_y = \frac{\mu_0 L I^2}{2\pi}\left[2\left(\frac{I_x - I_y}{I}\right)\frac{\xi}{c^2} + \left(\frac{I'}{I}\right)\frac{1}{h}\right]}$$



I’ = -I

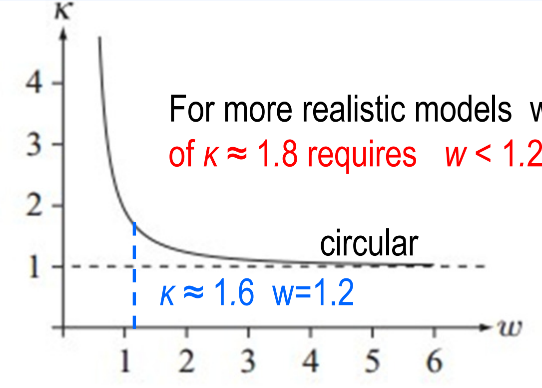
For stability:

$$\boxed{2\left(\frac{I_x - I_y}{I}\right)\frac{\xi}{c^2} + \left(\frac{I'}{I}\right)\frac{1}{h} < 0}$$

After substitution of $I_{x,y}$ and I’=-I, it follows:

$$\boxed{\frac{2\kappa^2}{1+\kappa^2}\ln\kappa \le \frac{1}{w^2}} \longrightarrow$$

with w=b/a



For more realistic models with *R0/a* ≈ 3, an elongation of *κ* ≈ 1.8 requires *w* < 1.2−1.4

Vertical stability of an elongated tokamak with a real conducting wall (5)

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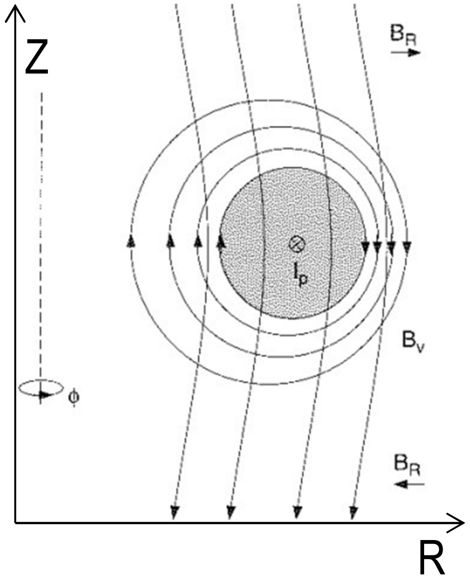
From H. Zohm : Magnetohydrodynamic stability of tokamaks, ed. Wiley (2014)



Vertical stability can be discussed introducing a radial field (see fig.)

Introducing the stability index: $n = -\frac{R\,\partial B_z}{B_z \partial R}$

Since in vacuum $\frac{\partial B_z}{\partial R} = -\frac{\partial B_R}{\partial Z}$ and stability requires $\frac{\partial B_R}{\partial Z} > 0$ (see fig.) and B_v is in negative z direction (for positive plasma current) i.e.

n > 0 for vertical stability.

To elongate the plasma n<0 (equal currents up and down as seen above)

with $B_z = \alpha_S \frac{\mu_o I_p}{4\pi R_o}$ (α_S depends on the plasma conditions: β_p, l_i)

Since:

$$F_{\text{destab}} = 2\pi R_0 I_p \frac{\partial B_R}{\partial z}\Big|_{(R_0,z_0)} (z - z_0) = -2\pi I_p B_z n (z - z_0)$$

Assuming further $(z - z_o) \propto \exp(\gamma t)$ it follows:

$$m_p \frac{d^2 z}{dt^2} = F_{\text{destab}} = -n \alpha_S \frac{\mu_0 I_p^2}{2R_0}(z - z_0) \rightarrow \gamma^2 = -\frac{v_{\text{A,pol}}^2}{R_0^2}\alpha_S n$$

with $V_{A,pol} = \left(\frac{\mu_o I_p}{2\pi a}\right)\frac{1}{\sqrt{\mu_o \varrho}}$

Vertical stability of an elongated tokamak with a real conducting wall (6)

From the last equation it is clear that for unstable cases (n<0) the growth rate is of the order of the Alfvèn velocity → **too fast** → non accessible for feedback systems !

Some sort of passive stabilizing wall is therefore needed !

Flux balance for the conductor reacting to the plasma current, I_p :

$$\boxed{\psi_c = M_{cp}I_p + L_cI_c} \qquad (1)$$

Due to a change of the plasma vertical position :

$$\boxed{\frac{d\Psi_c}{dt} = I_p\frac{\partial M_{cp}}{\partial z}\frac{dz}{dt} + L_c\frac{dI_c}{dt} = -R_cI_c}$$

Therefore the conductor current changes as:

with $\tau_R = \frac{L_c}{R_c}$

$$\boxed{I_c = -I_p\frac{\partial M_{cp}}{\partial z}\frac{z}{L_c}\frac{\gamma\tau_R}{\gamma\tau_R + 1}}$$

Considering an eq. like (1) for the plasma (with $M_{cp} = M_{pc}$) the induced (by I_c) radial field is:

$$\boxed{B_R = -\frac{1}{2\pi R_0}\frac{\partial\Psi_p}{\partial z} = -\frac{1}{2\pi R_0}\frac{\partial M_{pc}}{\partial z}I_c = \alpha_c\frac{\gamma\tau_R}{\gamma\tau_R + 1}\frac{z}{R_0}\frac{\mu_0 I_p}{4\pi R_0}}$$

where:

$$\boxed{\alpha_c = \frac{2R_0}{\mu_0 L_c}\left(\frac{\partial M_{cp}}{\partial z}\right)^2}$$

Vertical stability of an elongated tokamak with a real conducting wall (7)

The stabilizing force due to this current can be calculated as:
($F_{stab} = -2\pi R_o B_R I_p$)

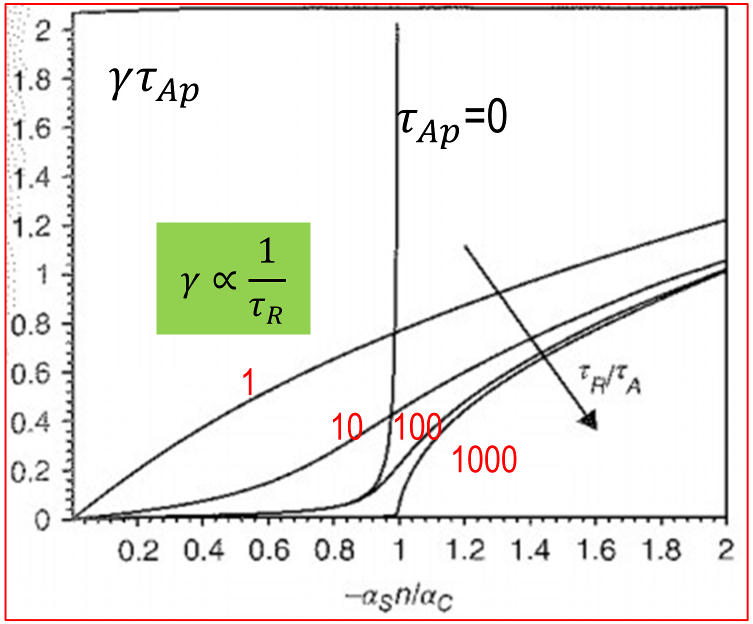
$$F_{\rm stab} = -\alpha_c \frac{\gamma\tau_R}{\gamma\tau_R + 1} \frac{\mu_0 I_p^2}{2} \frac{z}{R_0}$$

and finally the dispersion relation becomes:

$$\gamma^2 \tau^2_{\rm A,pol} + \alpha_S n + \alpha_c \frac{\gamma\tau_R}{\gamma\tau_R + 1} = 0$$

- Natural elongation in toroidal geometry (with a pure vertical field) doesn't need feedback if:
  $\kappa < \kappa_{nat} = 1 + \frac{1}{2(A-1)}$ with A= R/a

- this calculations **assume** no change in shape **i.e plasma rigidity**
  → not completely true !



Symmetric VDEs and halo currents (1)

From Nakamura et al 37th EPS Conf. (2010)

- The standard model of **halo currents** consists in a layer (pink area) of **poloidal currents** that circulate in the open field line region at the boundary of the vertically moving plasma.

- 2D codes (like DINA , TSC) contain specific models to describe the halo current layer evolution, in terms of width and temperature of the halo

- **the halo free parameters** are **adjusted** to match the experimental data

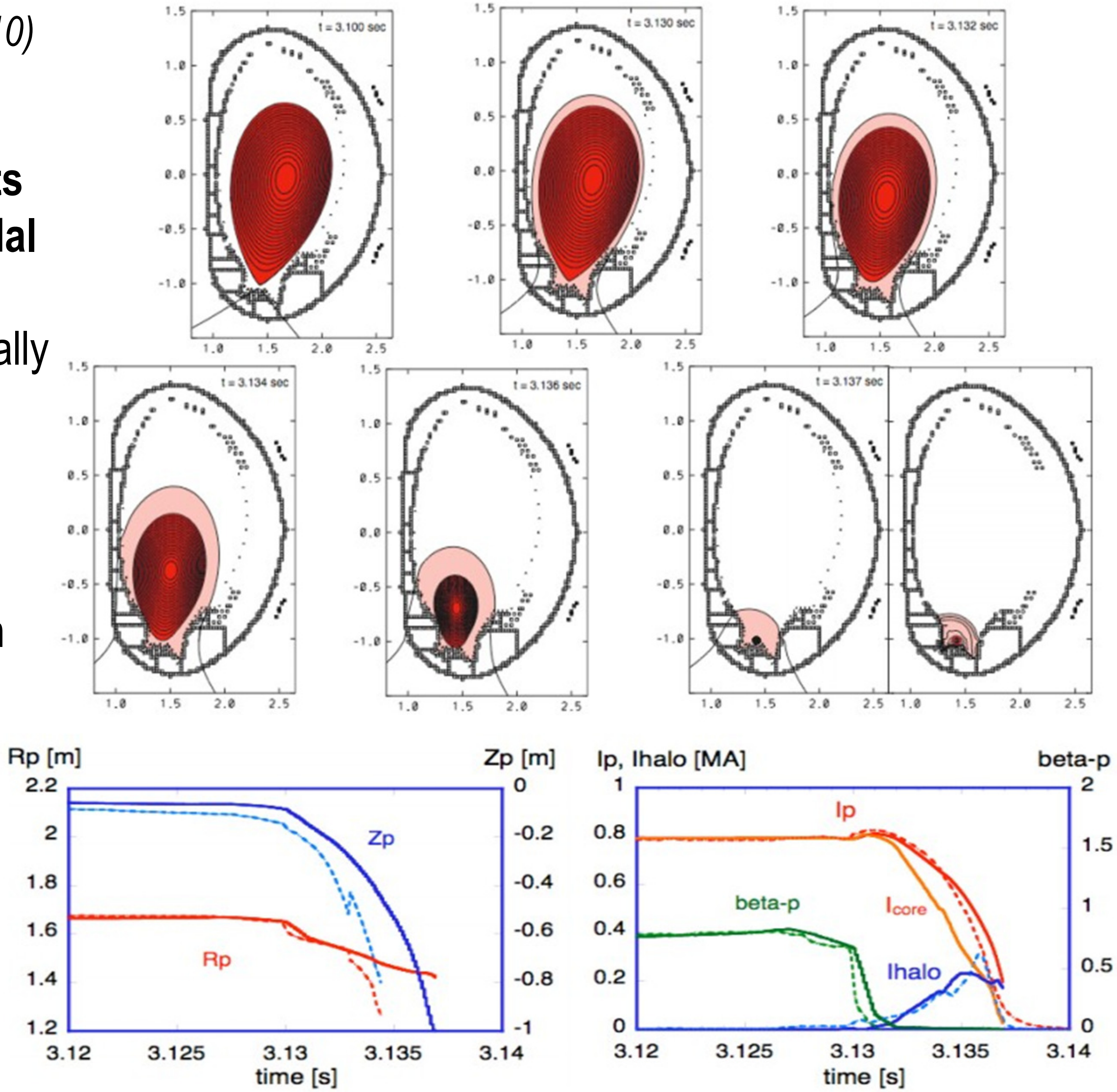


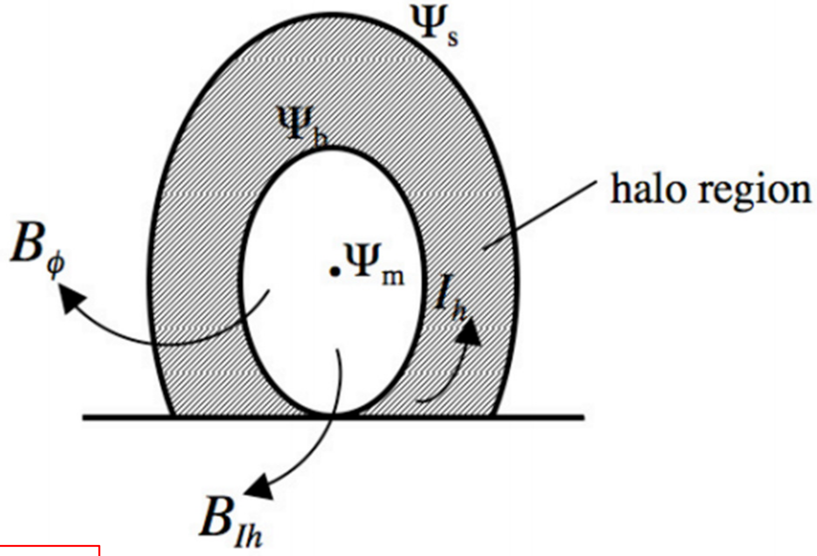
Fig. 2.1 TSC simulation (solid line) and experimental observation (broken line) of ASDEX Upgrade

Symmetric VDEs and halo currents (2)

from M. Windridge phd Thesis (2009)

In DINA code the **halo flux** is defined as a fraction (**w**) of the flux inside the plasma:

$$\begin{aligned}\Delta\Psi_{halo} &= \Psi_b - \Psi_s \\ &= w(\Psi_m - \Psi_b)\end{aligned}$$



w is calculated setting: γ=1 with :

$$\gamma(t,w) = \frac{S_0}{S(t,w)}\left[C + \left(\frac{I_p(t,w)}{I_{p0}}\right)\right]\frac{1}{C+1}$$

It is just an **empirical relation**

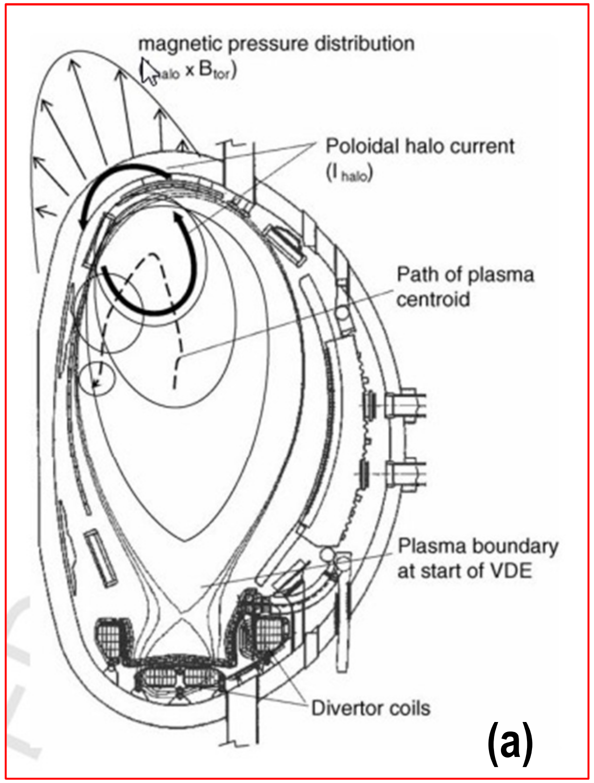
where ‘o’ means before the thermal quench time, S is the total area (plasma+halo) and C is a free constant: **for large C the total S is conserved** and S_p shrinks S_h grows. For C=1 : $\frac{S}{S_o}=\frac{I_p}{I_{po}}$

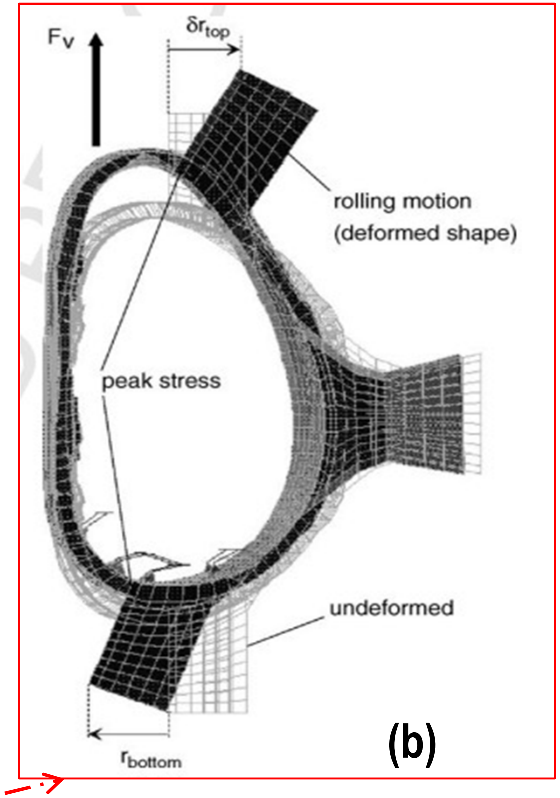
Symmetric VDEs and halo currents (3)

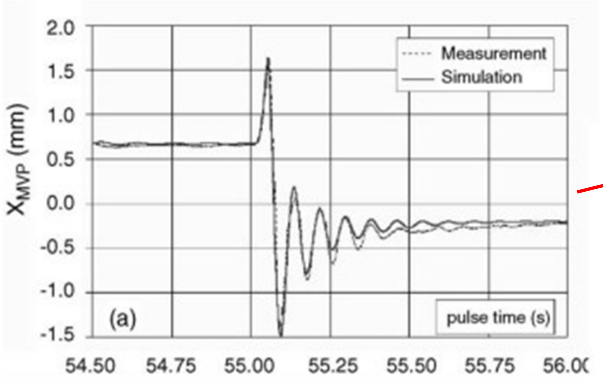
From Buzio et al Fus. Eng. Des. (2006)

Vertical forces **(a)** during the VDE and vessel rolling motion **(b)** in JET due to dampers (MVP)

In JET vertical forces up to **3-4 MN** lasting for several (10-50) ms have been measured







The force scales with the **poloidal halo current** crossed with the **toroidal magnetic field:**

$$\boxed{F_v \approx \; I_{halo} \, B_\phi}$$

with $I_{halo} = f \; I_p$ and $f \approx 0.1 - 0.2$

VDEs symmetric and non symmetric events (1)

From Bertolini, Fus. En. Des. (1996)

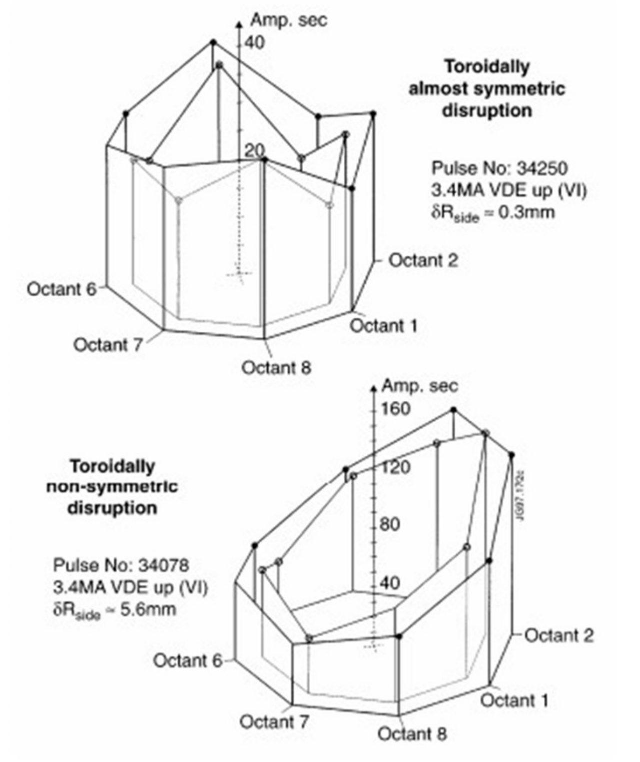
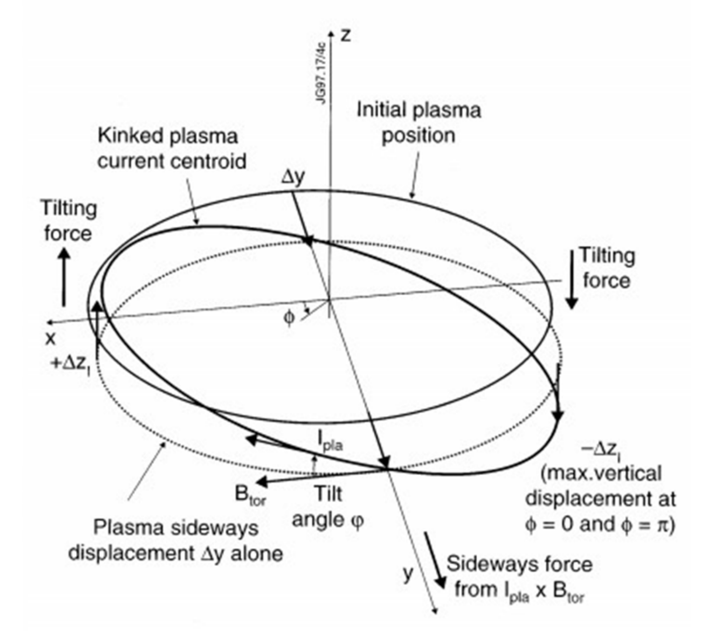


Fig. 15. Time integrated mushroom tile halo currents in VDEs with and without torus sideways displacement: the plasma configuration before the VDE was almost the same in both cases.

- **Tilted/shifted** (m=1,n=1) **wire model** and sideway forces

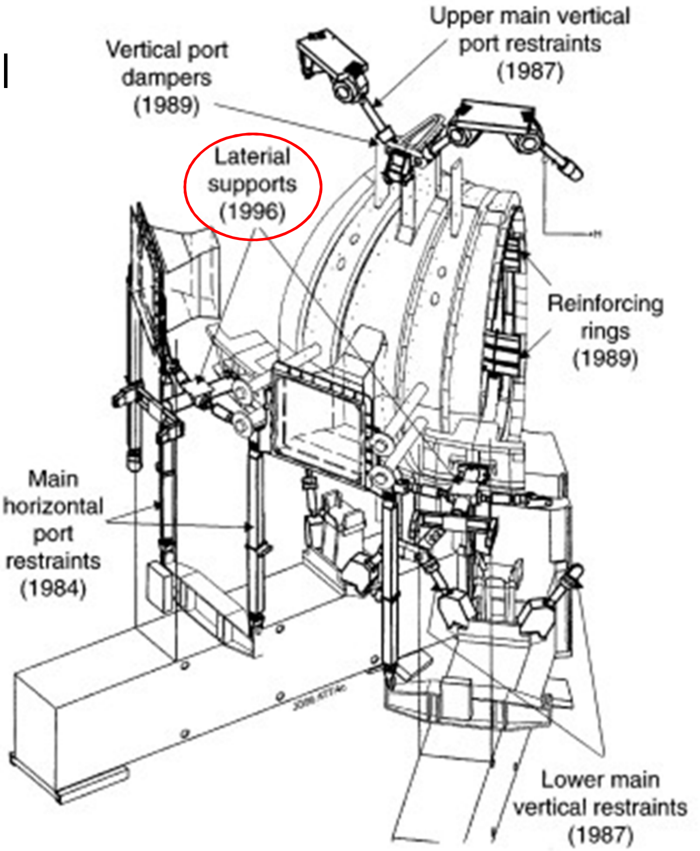


VDEs symmetric and non symmetric events (2)

In JET lateral support to withstand to sideways forces have been installed **in 1996** after a vessel serious damage **in 1994**

After this event it has become clear that sideways forces due to toroidal non axi-symmetric halo currents distribution **are extremely dangerous** and should be avoided

The fact that **an n=1 mode** could explain the observations led to hypothesize that the responsible agent could be **an MHD mode** grown at relatively high amplitude



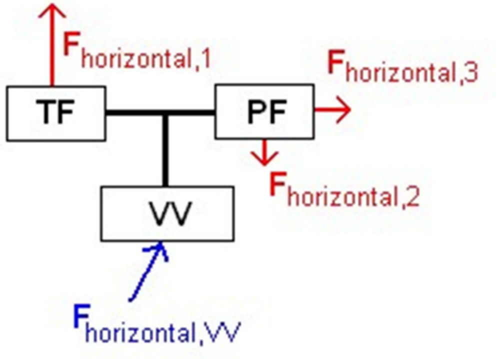
Horizontal force components (tilted/shifted wire)

From Bachmann, ITER report (2007)

- $F_1 \approx I_{pla} \sin(\phi)\, B_{tor}$ (due to the vertical comp. of the current)
- $F_2 \approx I_{pla}\, (B_{pol}(y+\Delta y) - (B_{pol}(y-\Delta y))$ (due to the poloidal field variation with y : shift)
- $F_3 \approx I_{pla}(B_{pol}(z+\Delta z) - (B_{pol}(z-\Delta z))$ (due to the poloidal field variation with z : tilt)

F_1>> $F_{2,3}$ (30-40 MN >> 2-3 MN) in ITER

$$F_{horizontal,VV} = \sqrt{(F_{horizontal,1} - F_{horizontal,2})^2 + F_{horizontal,3}{}^2}$$

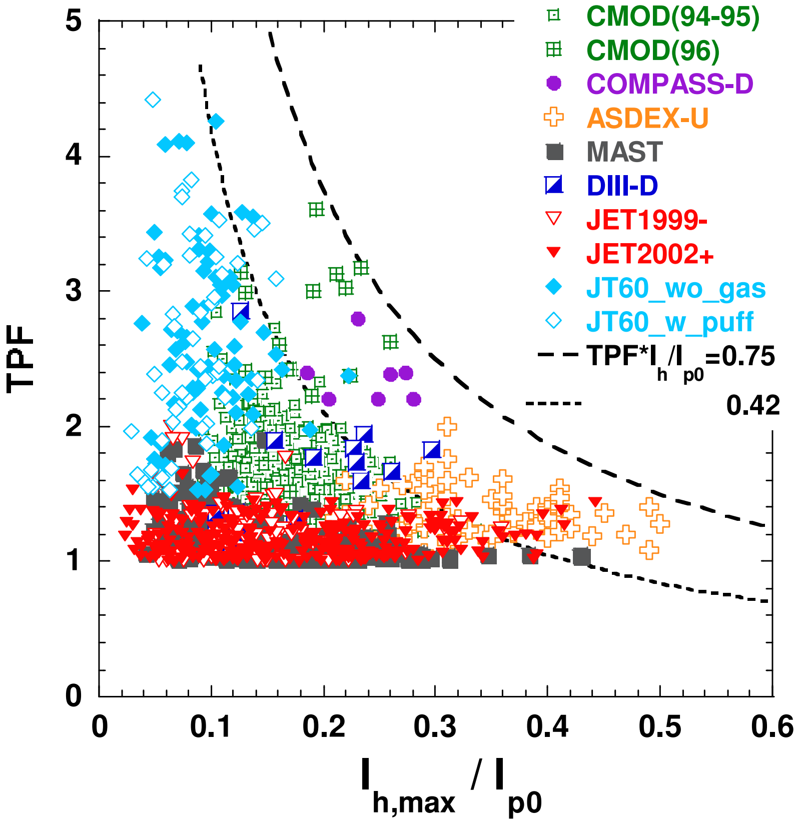


Experimental characterization of non symmetric events

To characterize using a simple parameter
the occurrence of non symmetric disruptions the
Toroidal Peaking Factor (TPF) was introduced:

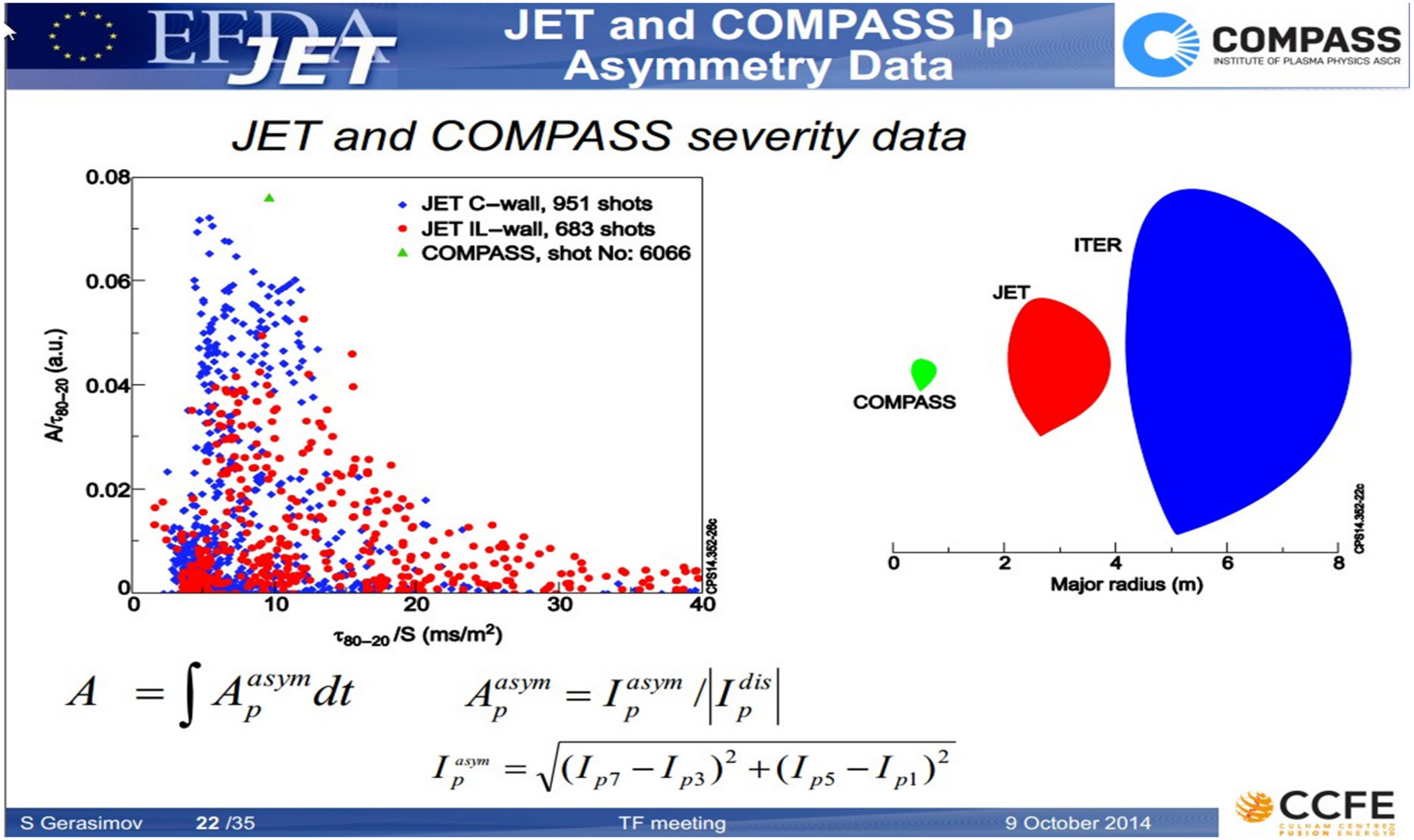
$$\boxed{TPF = \frac{Max(I_h(\phi))}{< I_h(\phi) >}}$$

while hf = $\frac{I_{h,max}}{I_{po}}$ is said the **halo fraction**



In ITER the product (TPF*hf) should remain below 0.75 (see Fig.)

Experimental characterization of non symmetric events (1)



EFDA JET

JET and COMPASS Ip Asymmetry Data

COMPASS
INSTITUTE OF PLASMA PHYSICS ASCR

JET and COMPASS severity data

$$A = \int A_p^{asym} dt \qquad A_p^{asym} = I_p^{asym} / \left| I_p^{dis} \right|$$

$$I_p^{asym} = \sqrt{(I_{p7} - I_{p3})^2 + (I_{p5} - I_{p1})^2}$$

S Gerasimov 22 /35 TF meeting 9 October 2014

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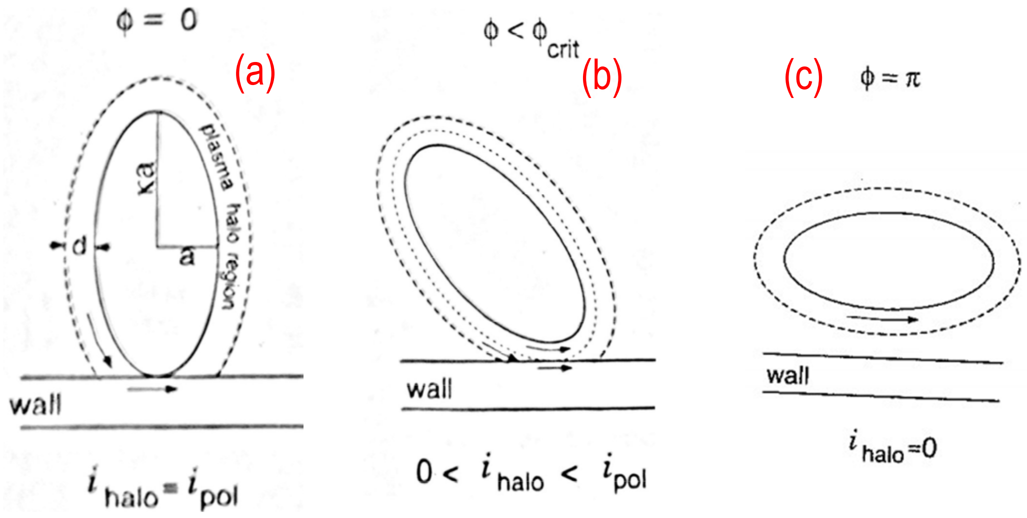
Not only is important the **amount of non axi-symmetry** but also **how long it lasts**:
a parameter **A** is defined to this purpose and it measures the severity of the impulse

A simple model for TPF vs Halo

From Pomphrey et al NF (1998)

2/1 mode plasma -wall Interaction along the torus



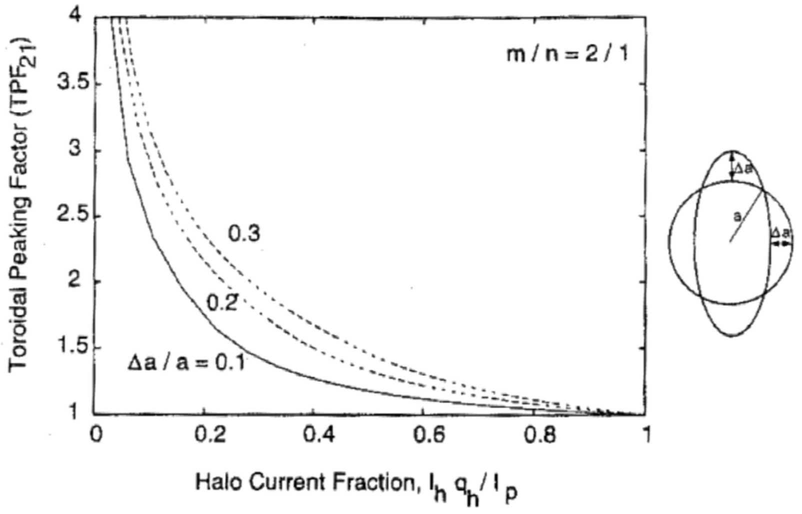
Assuming a 2/1 mode and a force free plasma at the boundary :

$$X = R + r\cos(\theta) \qquad Z = r\sin(\theta)$$

$$r^2 = a^2\frac{\rho}{2}\left((\kappa^2+1) - (\kappa^2-1)\cos(2\theta - \phi)\right)$$

$$\boldsymbol{J} = \lambda\,\boldsymbol{B}$$

$$I_{pol} = \int_0^{2\pi} i_{pol}\,d\phi \qquad TPF^{2,1} = \frac{i^{h,max}}{< i^h >}$$



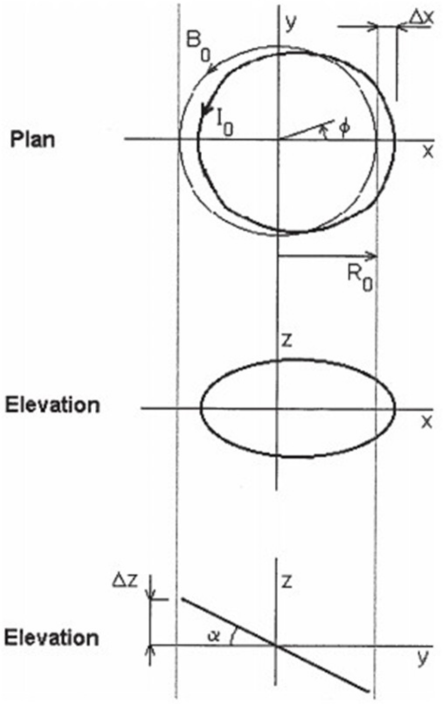
Tilted wire model and non symmetric forces (1)

From Riccardo et.al. Fus. Eng. Des. (2000)

Defining the elevation $\alpha = \frac{\Delta z}{R_o}$ (y goes in toroidal direction) of the current ring and assuming also a shift in x direction, Δx .
For **small tilt and shift**, the magnetic field **at R** can be expressed as:

$$\boxed{\begin{aligned} B_R &\approx B_0 R_0 (\Delta x \sin\varphi - \alpha z \cos\varphi)/R^2 \\ B_\varphi &\approx B_0 R_0 (R - \Delta x \cos\varphi - \alpha z \sin\varphi)/R^2 \end{aligned}}$$

$$\boxed{B_z \approx B_0 R_0 \alpha \cos\varphi / R}$$



The element force is:

$$\delta F_x = \oint \delta I_0 B_z(R,\varphi) \cos\varphi R d\varphi \approx \pi \delta I_0 B_0 R_0 \alpha$$

and the total:

$$\boxed{F_x \approx \pi I_0 B_0 R_0 \alpha = \pi I_0 B_0 \Delta z}$$

can be expressed as:

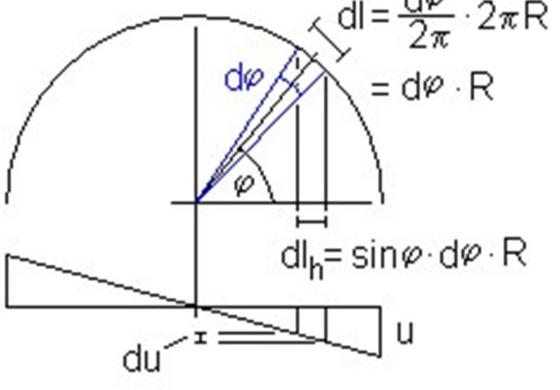
$$\boxed{F_x \approx \frac{\pi}{2} \Delta M_z B_0}$$

(Noll formula)

where ΔM_z is the difference between the current moment at $\phi = -\frac{\pi}{2}$ $and\ \phi = \frac{\pi}{2}$

Tilted wire model and non symmetric forces (2)

From Bachmann, ITER report (2007)



$$du = \frac{dl_h}{R} \cdot u = \sin\varphi \cdot d\varphi \cdot u$$

$$F_{rad} = \int_0^{2\pi} B_{tor} \cdot I_{pl} \cdot u \cdot \sin\varphi \, d\varphi, \qquad F_{horizontal,1} = \sin\varphi \cdot F_{rad}$$

$$F_{horizontal,1} = B_{tor} \cdot I_{pl} \cdot u \cdot \int_0^{2\pi} \sin^2\varphi \, d\varphi = B_{tor} \cdot I_{pl} \cdot u \cdot \pi$$

An equivalent (more clear) way of calculating the **dominant force** of the **tilted-shifted wire**, considering the Lorentz force between the vertical (z-component)of the current and the toroidal magnetic field.

Therefore (similarly to the symmetric case) :

$$\boxed{F_{hor} \approx f' I_p B_\phi}$$

with $f' \approx 0.1 - 0.3$

Approximate Magnetic Field of a tilted coil

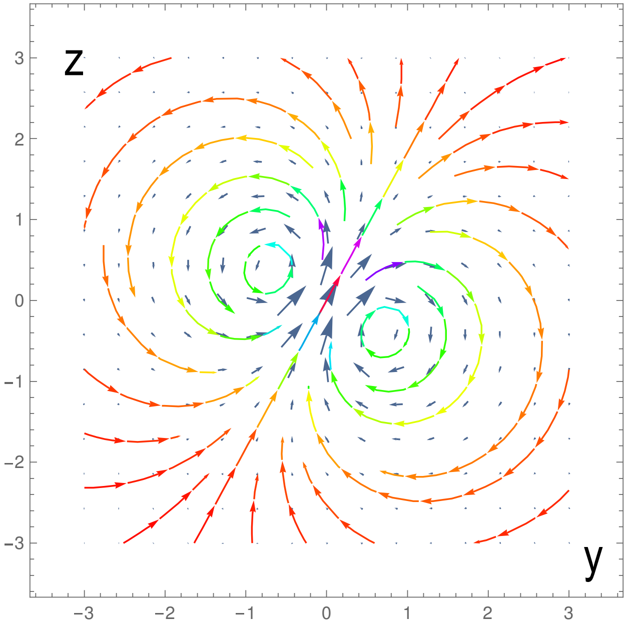
A good approximation^ for the vector potential of a non tilted current loop is given by :

(demonstrations.wolfram.com/MagneticFieldOfACurrentLoop/)

$$A_\phi\ (y,z) = \left(\frac{\mu_0}{4\ \pi}\right) \frac{(\pi\ a^2\ I_c\ y)}{(a^2 + y^2 + z^2)^{3/2}}\ \left(1 + \frac{15\ a^2 y^2}{8\ (a^2+y^2+z^2)^2}\right)$$

With the transformation below, the z axis is rotated by the angle α :

$$y' = y\cos(\alpha) - z\sin(\alpha)$$
$$z' = y\sin(\alpha) + z\ cos(\alpha)$$



$A_\phi(y', z')$ contours are plotted in the figure with the magnetic field vector obtained from:

$$B_y = -\frac{\partial A_\phi}{\partial z} \quad \text{and} \quad B_z = \frac{\partial A_\phi}{\partial y}$$

^ without using elliptic integrals

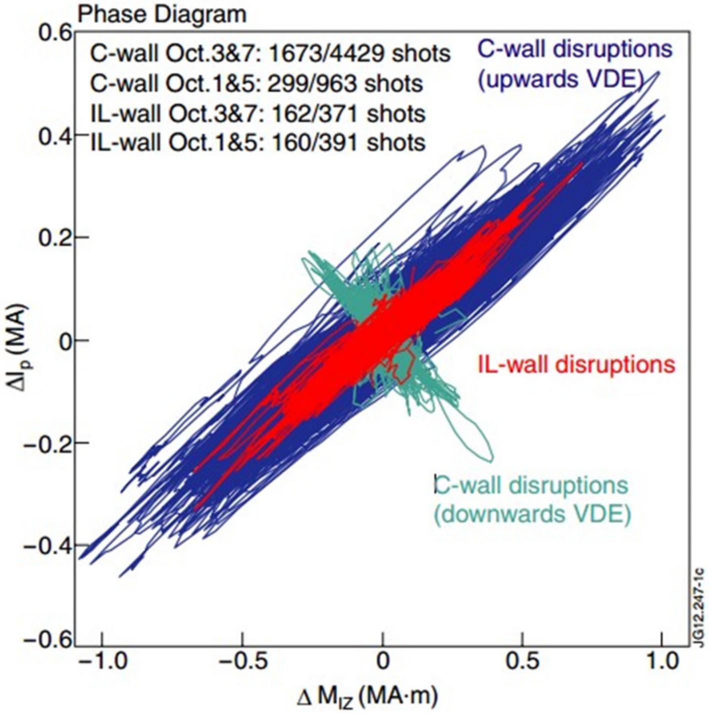
Experimental characterization of non symmetric events (2)

In **Jet** it has been observed that the horizontal force is well approximated by the Noll formula

Also it is observed that there is a linear corrrelation between the current moment and the current asymmetry (see figure)

This correlation is **not completely obvious** since it could be expected that the current becomes lower when the plasma touches the wall , instead **exactly the opposite** is observed

This interesting and simple observation has led to different attempts of interpretation:



From Gerasimov et al NF (2014)

- **a surface current model** (remainding Wesson's paper)
- **a nonlinear MHD model**
- **a passive stuctures model**

→ **All these models claim to be able to explain the JET observations**

Boundary conditions at ideal wall

- Plasma is surrounded by a **perfectly conducting wall** ($\boldsymbol{n}$ is outward-pointing normal vector)
  - $\boxed{(\boldsymbol{n} \times \boldsymbol{E})|_{r_{wall}} = 0}$
  - $\boxed{(\boldsymbol{n} \cdot \boldsymbol{B})|_{r_{wall}} = 0}$
  - $\boldsymbol{n} \times (\boldsymbol{E} + c^{-1}\boldsymbol{V} \times \boldsymbol{B})|_{r_{wall}} = 0 \Rightarrow (\boldsymbol{n} \cdot \boldsymbol{V})|_{r_{wall}} = 0 \Rightarrow \boxed{(\boldsymbol{n} \cdot \tilde{\boldsymbol{\xi}})|_{r_{wall}} = 0}$

Not appropriate to study VDEs

Boundary conditions for plasma – vacuum interface :

- Plasma is surrounded by a **vacuum region** (which is described by $\boldsymbol{\nabla} \times \hat{\boldsymbol{B}} = \boldsymbol{\nabla} \cdot \hat{\boldsymbol{B}} = 0$, with $\hat{\boldsymbol{B}}$ the vacuum magnetic field)
  - $\boxed{(\boldsymbol{n} \cdot \hat{\boldsymbol{B}})|_{r_{wall}} = 0}$
  - plasma surface is free to move $\Rightarrow$ $(\boldsymbol{n} \cdot \tilde{\boldsymbol{\xi}})|_{r_{plasma}} =$ arbitrary
  - $\boxed{[[\boldsymbol{n} \cdot \boldsymbol{B}]]|_{r_{plasma}} = 0}$ (with $[[\cdots]]$ denoting a jump across the plasma surface)
  - $\boxed{[[\boldsymbol{n} \times \boldsymbol{B}]]|_{r_{plasma}} = (4\pi/c)\boldsymbol{K}}$, with $\boldsymbol{K}$ the surface current density
  - $\boxed{[[p + B^2/8\pi]]|_{r_{plasma}} = 0}$

BC appropriate for KTM / surface current

Boundary conditions at a thin resistive wall

$$[\boldsymbol{n}\cdot\boldsymbol{B}]_{rwall}=0 \qquad\qquad [\boldsymbol{n}\times\boldsymbol{B}]_{rwall}=\mu_o\delta\,\mathrm{J}=\mu_o\frac{\delta\,\boldsymbol{E}}{\eta_{wall}}$$

Can be rewritten more explicitly as:

$$\hat{n}\cdot\mathbf{B}_V=\hat{n}\cdot\mathbf{B}_P$$

$$\mathbf{B}_V=\nabla\phi_V \qquad \nabla^2\phi_V=0$$

$$\frac{\partial}{\partial t}\hat{n}\cdot\mathbf{B}=\frac{\partial}{\partial l}\frac{\eta_W}{\delta}\left[\mathbf{B}_P\cdot\hat{l}-\mathbf{B}_V\cdot\hat{l}\right]+\frac{1}{R}\frac{\partial}{\partial\varphi}\frac{\eta_W}{\delta}\left[\mathbf{B}_P\cdot\hat{\varphi}-\mathbf{B}_V\cdot\hat{\varphi}\right]$$

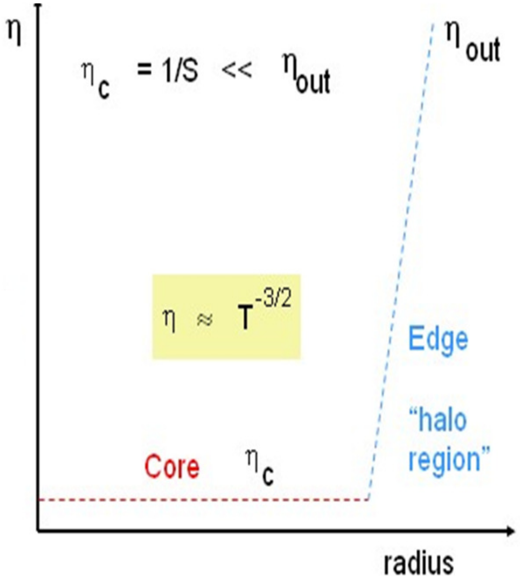
$\mathbf{B}_V$ = magnetic field on vacuum side of wall
$\mathbf{B}_P$ = magnetic field on plasma side of wall
ϕ_V = magnetic scalar potential in wall
η_W = resistivity of wall
δ = thickness of wall

Generally also **assuming** for **v**: $(\boldsymbol{n}\cdot\boldsymbol{V})|_{r_{wall}}=0$

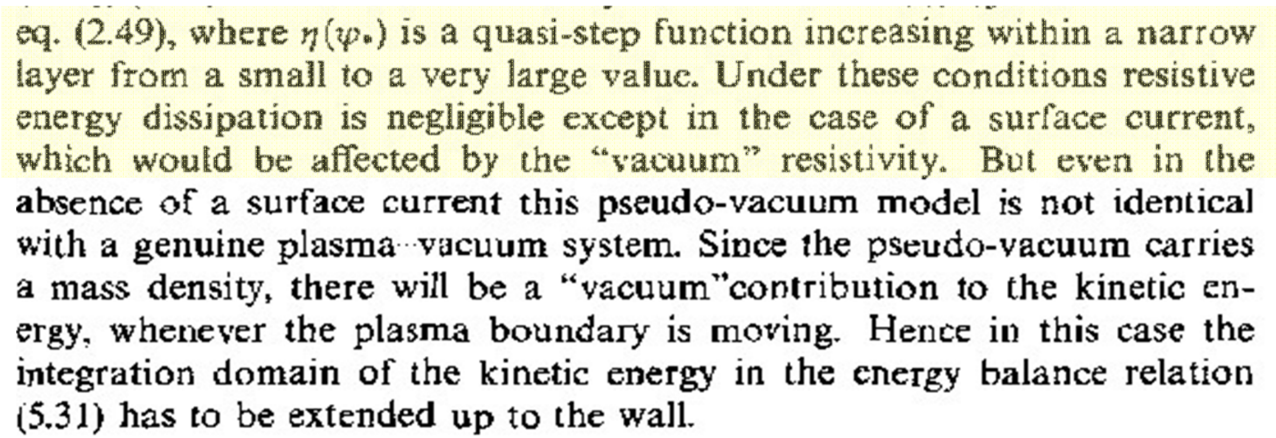
good for VDEs studies without surface currents
(only wall currents are allowed)

Replacing plasma-vacuum BC with cold plasma layer

Plasma-Vacuum BC are **often replaced in nonlinear codes** (as in M3D) by a **two region model** assuming a thin and cold plasma layer at the edge between the **hot core** and the **wall**

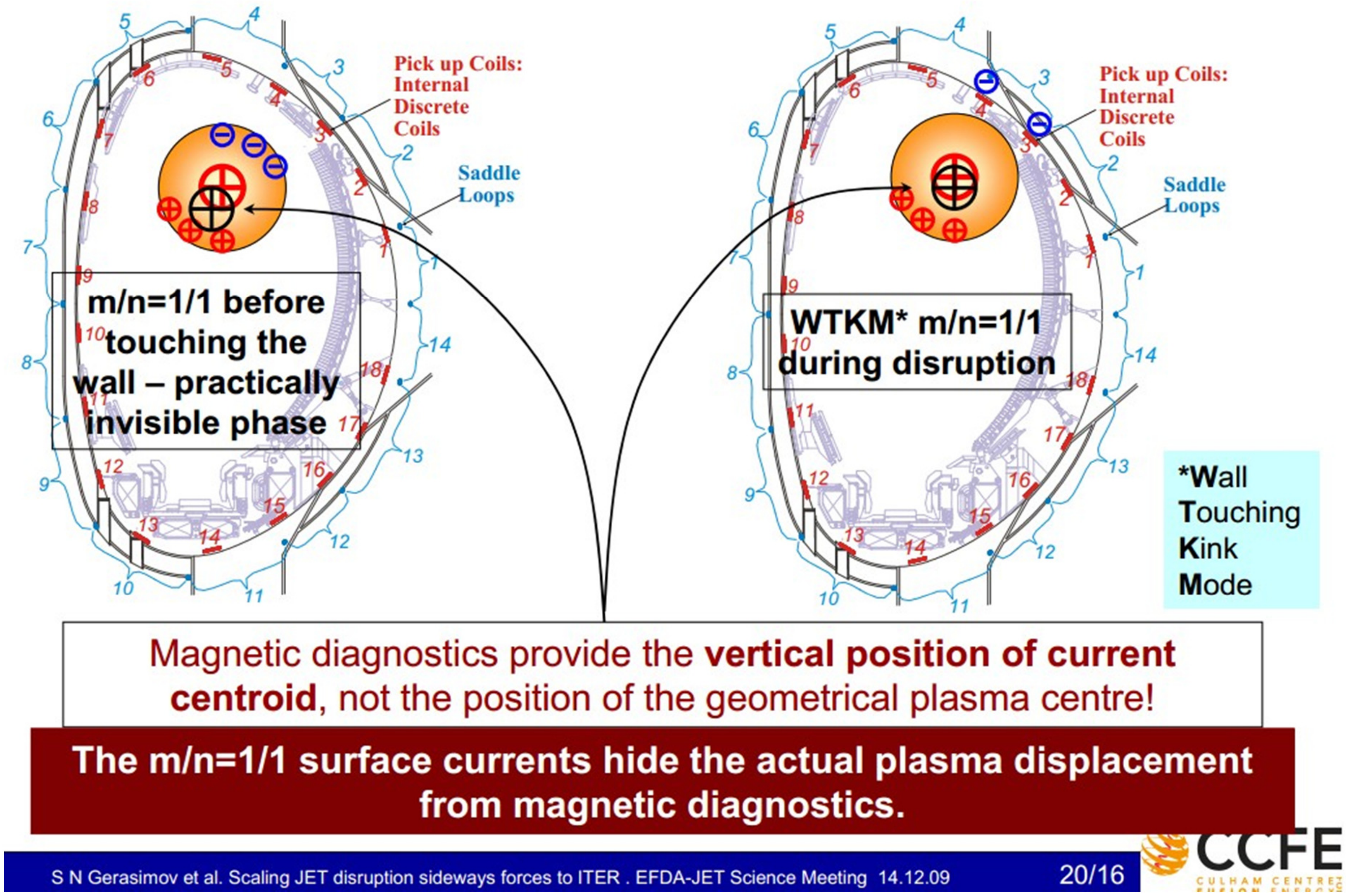


From Biskamp «Nonlinear MHD»


eq. (2.49), where $\eta(\psi_*)$ is a quasi-step function increasing within a narrow layer from a small to a very large value. Under these conditions resistive energy dissipation is negligible except in the case of a surface current, which would be affected by the "vacuum" resistivity. But even in the absence of a surface current this pseudo-vacuum model is not identical with a genuine plasma-vacuum system. Since the pseudo-vacuum carries a mass density, there will be a "vacuum" contribution to the kinetic energy, whenever the plasma boundary is moving. Hence in this case the integration domain of the kinetic energy in the energy balance relation (5.31) has to be extended up to the wall.

Hence effectively in cases in which **a surface current plays a** fundamental **role on the dynamics**, a **cold plasma layer model could not be considered equivalent** to a **vacuum-plasma** model.

The surface current model (**K**ink **T**ouching **M**ode) (1)



m/n=1/1 before touching the wall – practically invisible phase

WTKM* m/n=1/1 during disruption

*Wall Touching Kink Mode

Magnetic diagnostics provide the **vertical position of current centroid**, not the position of the geometrical plasma centre!

The m/n=1/1 surface currents hide the actual plasma displacement from magnetic diagnostics.

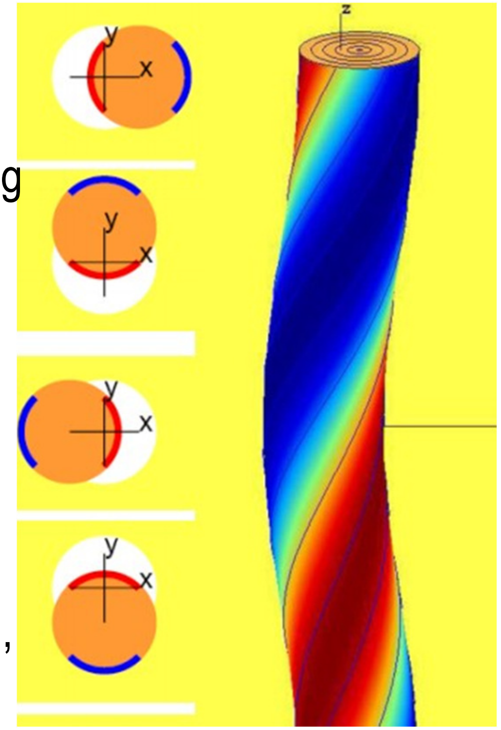
S N Gerasimov et al. Scaling JET disruption sideways forces to ITER . EFDA-JET Science Meeting 14.12.09 20/16

The surface current model (Kink Touching Mode) (2)

From L. Zakharov PoP (2008) and (2012)

The idea is that the plasma reacts to the **1/1 kink** deformation by a **surface current** which tend to slow down the kink achieving a **quasi-equilibrium state**.

$$\mu_0 i_{11}^{surf} = -2\frac{B_\phi \xi_{11}}{R} + \frac{1-q_a}{q_a}\frac{2\lambda}{1-\lambda}\frac{B_\phi \xi_{11}}{R}$$
$$= -2\frac{B_\phi \xi_{11}}{R} - \mu_0 i_{11}^{eddy}.$$



«Hiro» currents

Where **the first term** contributes to the kinked MHD equilibrium, while **the second** shields the eddy currents from the wall at the position of the surface current layer

The deduced force is consistent with Noll's formula:

eddy currents term

$$F_x^{theory} = \pi B_\phi I_{pl}\left(1-\frac{\lambda}{q_a}\right)\delta z = \left(1-\frac{\lambda}{q_a}\right)F_x^{Noll}$$

3D nonlinear MHD simulations results

From H. Strauss PoP (2014)

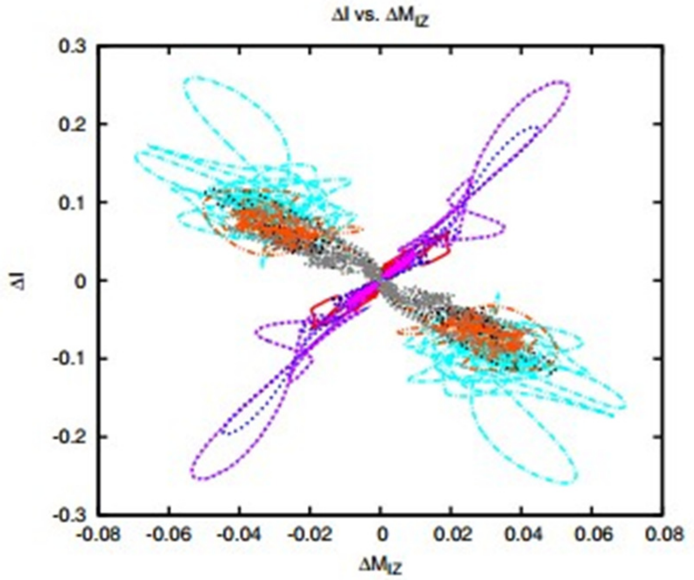
Using the nonlinear MHD code M3D (described later) and defining :

$$\Delta I = \frac{1}{V}\left(\oint d\phi \tilde{I}^2\right)^{1/2},$$

$$\Delta M_{IZ} = \frac{1}{V}\left(\oint d\phi \tilde{M}_{IZ}^2\right)^{1/2},$$

$$V = (2\pi)^{1/2}\int dRdZ.$$

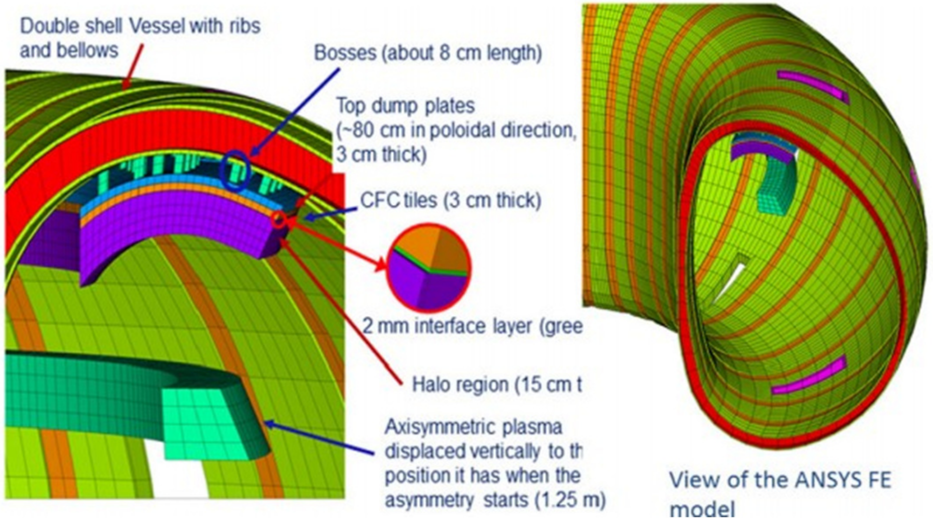
The results plotted in the figure, that seem **consistent with the JET data** are obtained
(**sin and cos** components of **ΔI** and **ΔM** are shown):



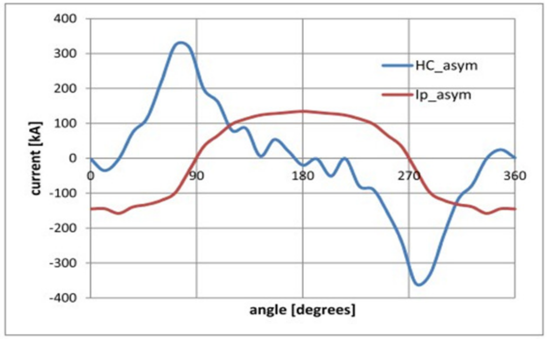
Eddy currents model in the JET wall

From Roccella et al NF (2016)

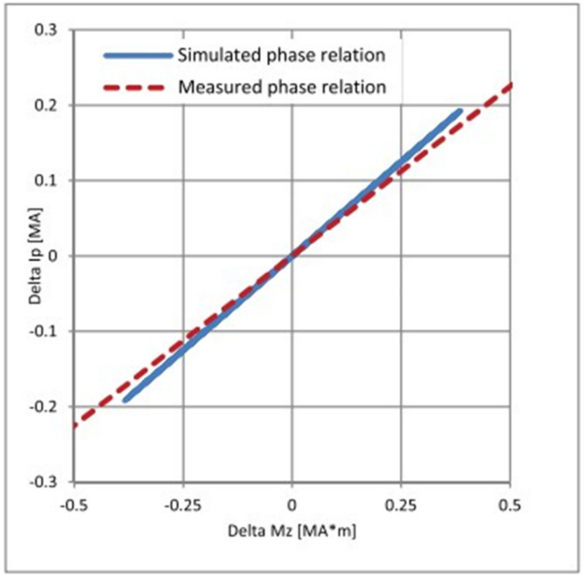
Providing **a detailed description of the JET wall** a model is developed that can explain the toroidal current asymmetry correlation with the magnetic moment as a result of eddy currents flowing from the wall to the plasma where the plasma touches (and short circuit) some wall elements



View of the ANSYS FE model



Halo current and toroidal current asymmetry are **90 degrees phase** shifted as in JET measurements



The emphasis of the model is on the **necessity** of a **detailed description of the passive structures** surrounding the plasma

Flux conservation: Halo, Hiro and Eddy currents (1)

An interesting observation is that **Halo, Hiro or even Eddy** currents are originated by the attempt of the plasma (for halo and hiros) and of the external conductors (for eddies) to **oppose** the **flux variations** associated with the plasma movements/rearrangements due to the MHD phenomena.

Halo, Hiro and Eddy are all **stabilizing currents** that tend to slow down and **counteract** (to some extent at least) **the plasma Instabilities**.

Halo and Hiro rise to preserve the magnetic flux in the plasma region, while **eddy currents screen the plasma region flux variation to the outside world.**

The amount of these currents depends **critically** on the **plasma edge electrical conductivity (for halo and hiro)** and on the **wall conductivity** **(for eddy)**

Flux conservation: Halos, Hiros and Eddies currents (2)

The flux conservation in an **ideally conducting plasma** can be written as:

$$\frac{\partial \boldsymbol{B}}{\partial t} = -\nabla \times \boldsymbol{E} = \nabla \times (\boldsymbol{v} \times \boldsymbol{B}) \rightarrow \frac{\partial}{\partial t} \iint \boldsymbol{B} \cdot d\boldsymbol{S} - \oint (\boldsymbol{v} \times \boldsymbol{B}) \cdot d\boldsymbol{S} = 0$$

or $\frac{\partial}{\partial t} \iint \boldsymbol{B} \cdot d\boldsymbol{S} + \oint (\boldsymbol{v} \times d\boldsymbol{S}) \cdot \boldsymbol{B} = 0$ i.e the FROZEN IN CONDITION

The physical meaning of the expression is that:

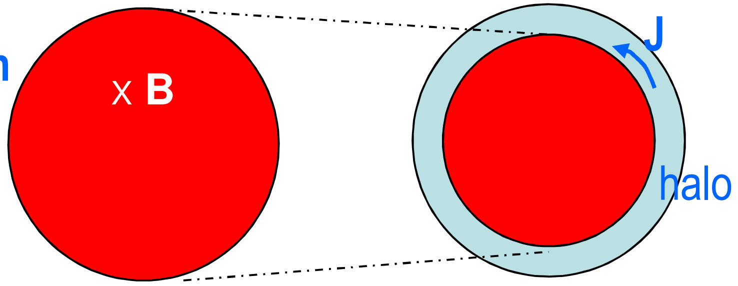
- the magnetic field is comoving with the fluid in ideal MHD
- i.e. the flux through every flux tube is constant as the tube moves around in space
- i.e. the field lines are attached to the fluids
- i.e. the magnetic field cannot change its topology,
- i.e the fluid cannot move across the magnetic field (it is only free to slide along B)

All the above **is not true in a resistive plasma: but the higher is the plasma temperature the better the ideal condition is satisfied**

Flux conservation: Halos, Hiros and Eddies currents (3)

- **The case of halos:**

i.e non ideal compression



Assume that the plasma shrinks:
and the toroidal flux **decreases (!)**
in the plasma region a poloidal current will rise in the halo region to oppose the flux variation

- **The case of hiros:**

If a similar shrinkage happens but no halo is formed outside the plasma, i.e. **a true vacuum** region surrounds the plasma the only way to preserve the flux is that a **surface current** rises at the boundary (so the **J current** is now flowing in a **narrow layer** on the plasma-vacuum «moving» interface)

- **The case of eddies:**

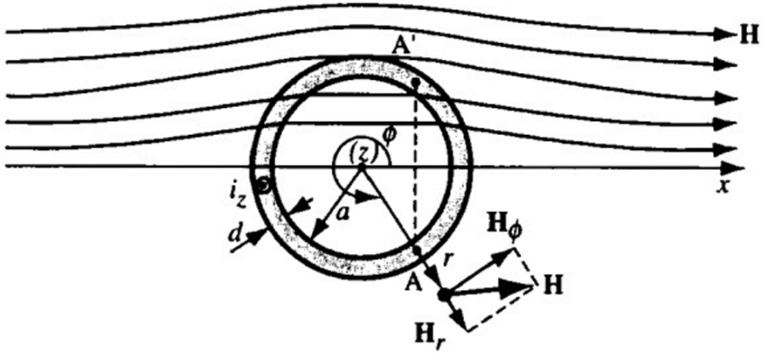
The case of eddies is similar to the case of halos with **the metal wall playing** (Lenz’s law) the same role as the halo region

What happens if a metal wall is present at the same time as halos or hiros ?

Some consideration about Hiro or surface currents (1)

from Knoepfel «Magnetic Fields»

If a step field Ho is turned on around a cylindrical metallic conductor the eddy current that arise on the conductor surface is:



$$\boxed{i_z = j_z d = -2H_0 e^{-t/\tau_d}\sin\phi}$$

with $\tau_d = \frac{\mu_o d a}{2\eta_w}$ being the diffusion time through the metal wall

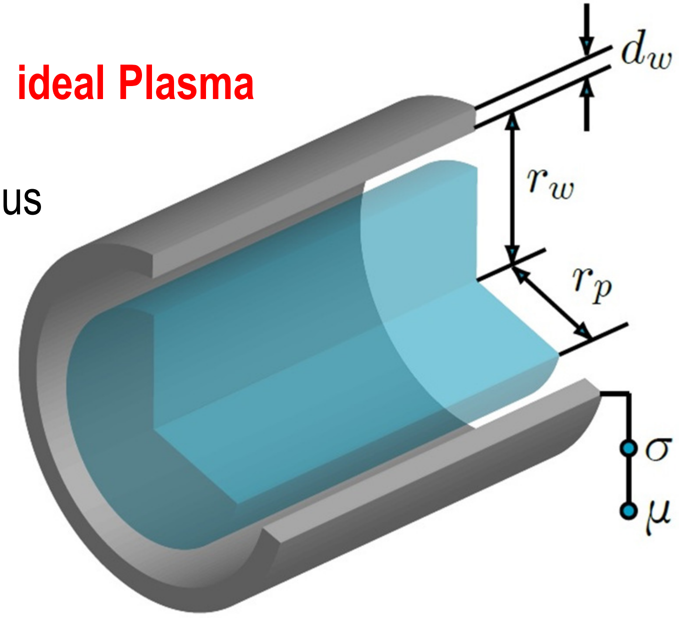
- Therefore it can be seen that the **current decays** in a time of the order of τ_d
- also τ_d is shorter for a narrow wall (dissipation increases)
- **similar things** can be expected to happen in a **«real» plasma** since **temperature is** high but **finite** and the **layers** containing the reaction currents **are** expected to be **thin**.

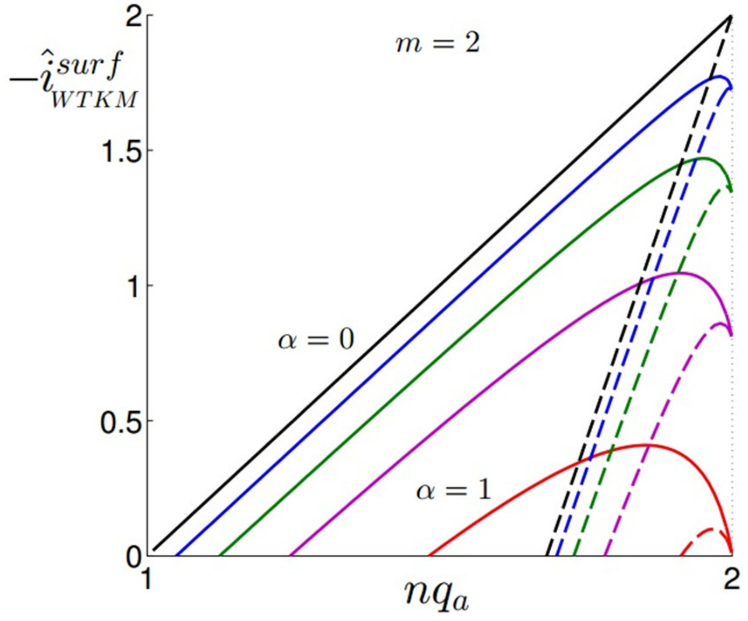
Assuming in a plasma d=1 cm a=1 m and T=10 eV what is τ_d ?

Some consideration about Hiro or surface currents (2)

Assuming a more **realistic** linear model **: Wall** + **Vacuum** + **ideal Plasma**

an **analytic dispersion relation** has been deduced for various current profiles (from flat $\alpha=0$ to parabolic $\alpha=1$)





Dashed lines have $r_w / r_p = 1.1$
For plain lines there is no wall

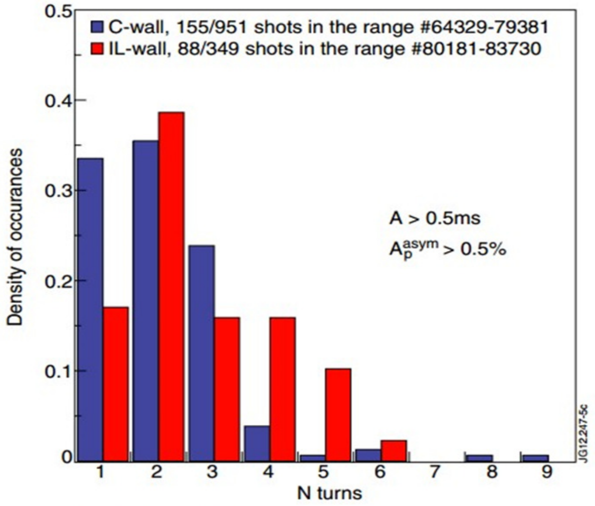
The main results are that the **surface currents** :

- depend on the equilibrium **J** profile
- are stongly reduced by the presence of a wall
- are linearly dependent on nq_a

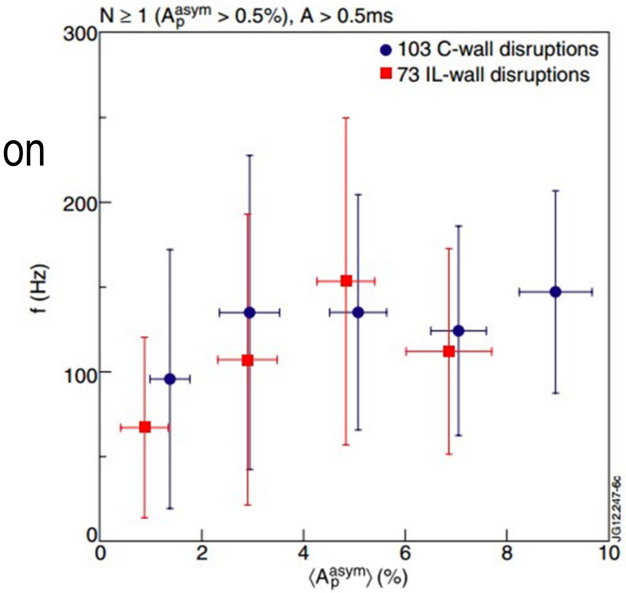
Rotation of the current asymmetry in JET

From S. Gerasimov et al NF (2014)

In JET rotation of the current asymmetry has been detected. The asymmetry is seen to make a few toroidal turns with a relatively low 100 Hz frequency



This effect is worring for ITER, in fact if the frequency will scale to 5-10 Hz it could resonate with **mechanical** structures **eigenfrequency** and produce force amplification



Rotation as been observed in **3D nonlinear MHD** (*Strauss, PoP (2014 and 2015*) , while the expanation is difficult considering KTM or passive wall models.
In **3D MHD** also the cause of the rotation is **not easily deconvolved** from simulations (I will come back later on this issue).

..again on the important role of the Virial theorem

CONSORZIO RFX
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From Pustovitov NF (2011)

Previously the role of the virial theorem for the force balance was discussed. In a more interesting and general form for what concern the conservation of angular momentum (in partiicular in toroidal direction) it can written as:

$$\boxed{\frac{\partial L_\phi}{\partial t} = T}$$

The interesting thing is that the torque T can be expressed completely by surface contributions:

$$\begin{array}{l} T = T_R + T_{em} + T_p + T_\Pi \\ \text{where :} \\ T_R = -\int R\rho v_\phi \mathbf{v}\cdot d\mathbf{S} \\ T_{em} = \int RB_\phi \mathbf{B}\cdot d\mathbf{S} - \int R\frac{B^2}{2}\mathbf{B}\cdot d\mathbf{S} \\ T_p = -\int Rp\hat{\phi}\cdot d\mathbf{S} \\ T_\Pi = -\int R(\mathbf{\Pi}\cdot\hat{\phi}\cdot d\mathbf{S} \end{array}$$

The momentum changes if there are non zero torques contributions at the wall either due to **kinetic or magnetic terms**

The normal **B** to the wall is very important i.e. **NONIDEAL (RESISTIVE) WALL bc**

with the viscous stress tensor given by: $\overleftrightarrow{\pi} = \rho\left\langle \boldsymbol{u}\boldsymbol{u} - \frac{\boldsymbol{u}^2}{3}\overleftrightarrow{I}\right\rangle$

where the < .. > is an average in velocity space over the particle distribution, and **u** the velocity

«Surface terms» and angular momentum conservation

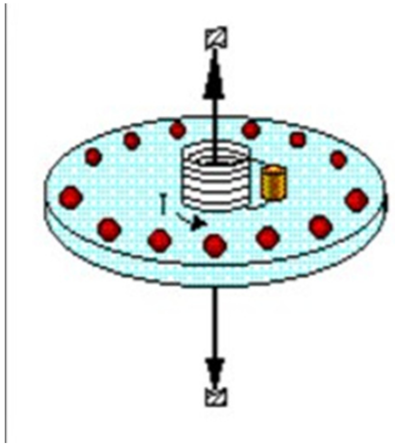
The concept of angular momentum conservation is exactly the same as for the boy on the revolving platform:

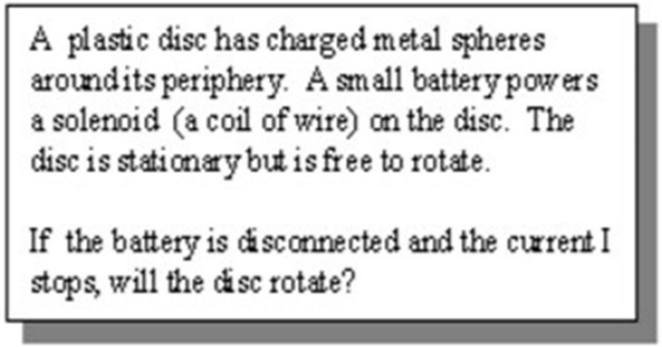
One central stack connected to the «earth» is necessary to change the angular momentum of the System.

Through the central stack «surface» torques are applied that are able to bring the system in rotation starting from rest.

..however is the situation so clear or are there complications that can arise in electromagnetism ?

Feynman disk paradox: the role of an electrostatic electric field (1)




A plastic disc has charged metal spheres around its periphery. A small battery powers a solenoid (a coil of wire) on the disc. The disc is stationary but is free to rotate.

If the battery is disconnected and the current I stops, will the disc rotate?

?

The answer is : …

YES

From the point of view of **electromagnetism the answer is quite clear** however the paradox arises because initially the disk is at rest and from a mechanical point of view apparently there are no applied torques.

The point is however that the **electromagnetic field has an intrinsic angular momentum** that is transmitted to the disk.

Feynman disk paradox: the role of an electrostatic electric field (2)

- **microscopic point of view: the electrons** providing the inital current in the wire **move in circle** and after the current is switched off they can transmit this loss of momentum to the disk through the wire (electrical resistance and collisions)

- **a macroscopic point of view:** from $-\nabla \times \boldsymbol{E} = \frac{\partial \boldsymbol{B}}{\partial t}$ an inductive electric field is generated that acts on the charges on the disk with a force : $\boldsymbol{F} = q\boldsymbol{E}$ that brings the disk in rotation

A more quantitative resolution of the paradox can be found in :

E. Corinaldesi, American Journal of Physics Vol. 48 (1980) 83.
G. G. Lombardi, American Journal of Physics Vol. 51, (1983) 213.

This paradox can however help in understanding that in tokamaks the edge conditions including the presence or the birth of **electrostatic electric fields** can be extremely important for the angular momentum balance and **therefore to understand plasma rotation** (even during disruptions).

In turn it should be remarked that such fields can be originated either by **transport phenomena** that can separate the electron and ion dynamics or even by any **charge accumulation** effects on wall gaps or divertor components .

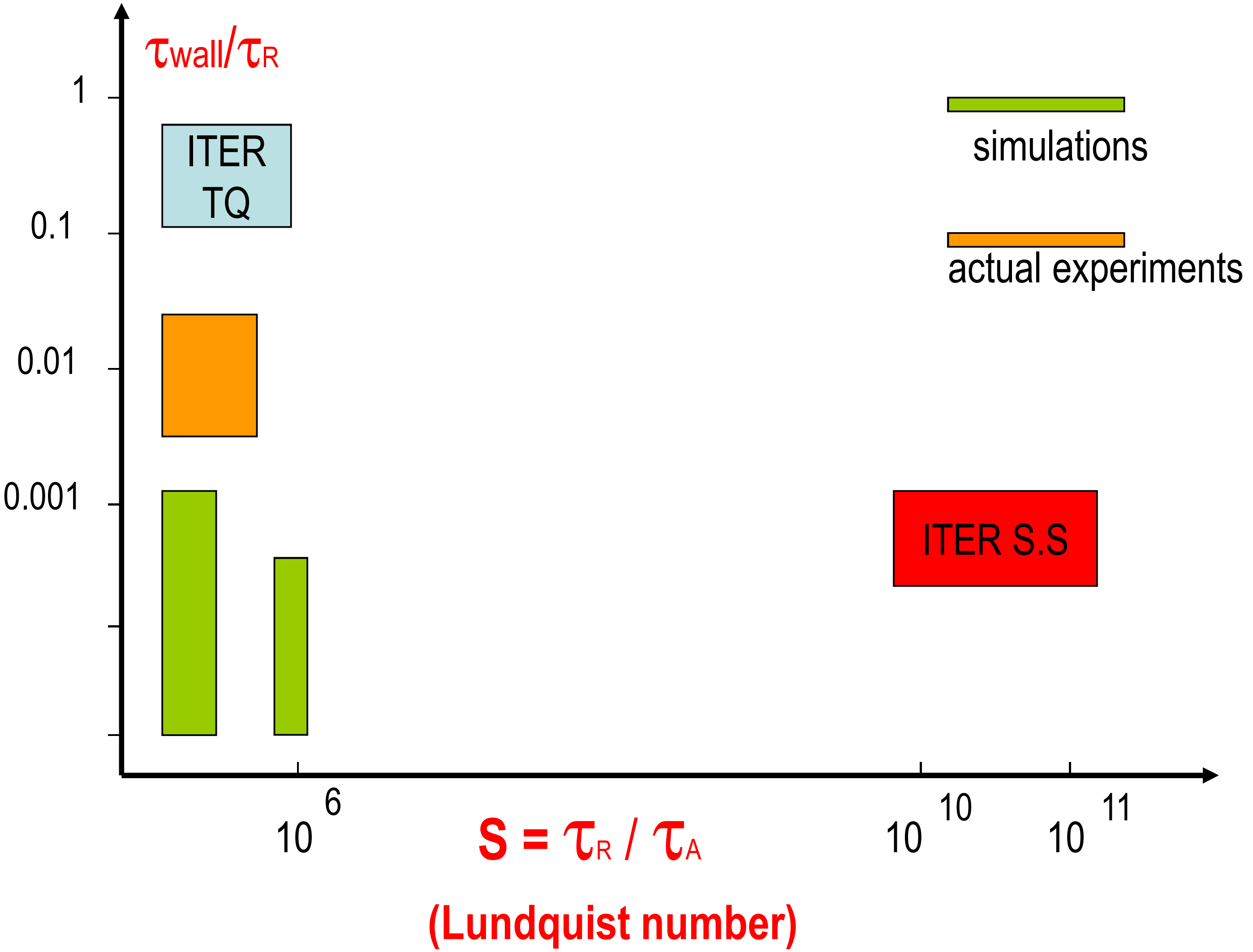
Summary of current asymmetries open issues for ITER

To summarize regarding non symmetric events **several points are still open** to predict the ITER behavior:

- the nature of the **current asymmetry: halos vs hiros vs eddies**
- **the role** of the **external conductor**
- **the duration** of the phenomenon i.e. the impulse transmitted to the structures
- the **nature/origin** and amount of **expected rotation**

Clearly also foreseeing the plasma conditions in ITER after the **thermal quench (TQ) (how fast? T?)** and during the **current quench (CQ) (how long?)** are extremely important to predict the following behavior and therefore to correctly estimate the consequences

ITER before and after the thermal quench



M3D code and nonlinear MHD simulations

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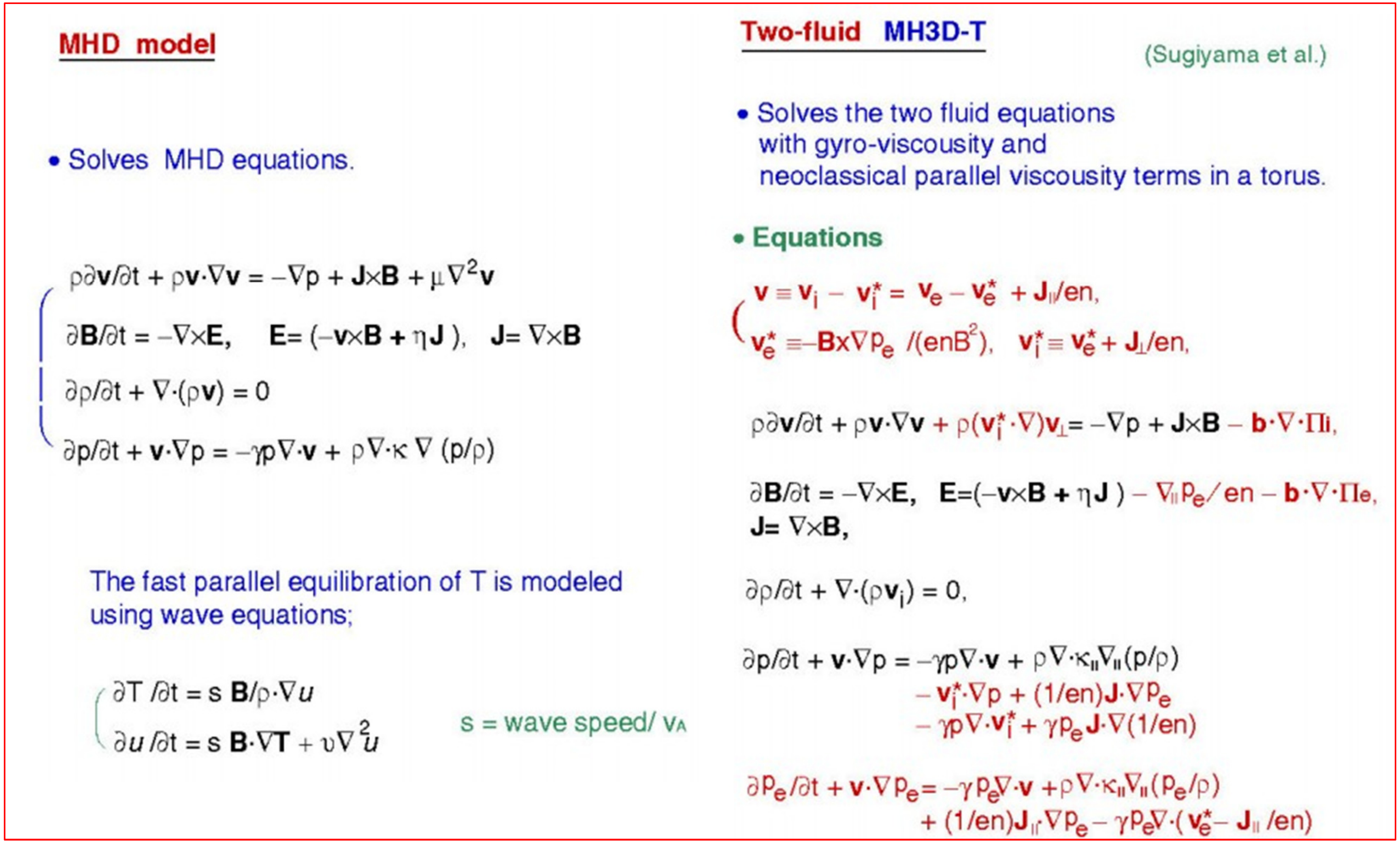
M3D is a **nonlinear (extended) MHD code** (with a peculiar model for the parallel transport) :



MHD model

- Solves MHD equations.

$$\begin{cases} \rho\partial\mathbf{v}/\partial t + \rho\mathbf{v}\cdot\nabla\mathbf{v} = -\nabla p + \mathbf{J}\times\mathbf{B} + \mu\nabla^2\mathbf{v} \\ \partial\mathbf{B}/\partial t = -\nabla\times\mathbf{E}, \quad \mathbf{E} = (-\mathbf{v}\times\mathbf{B} + \eta\mathbf{J}), \quad \mathbf{J} = \nabla\times\mathbf{B} \\ \partial\rho/\partial t + \nabla\cdot(\rho\mathbf{v}) = 0 \\ \partial p/\partial t + \mathbf{v}\cdot\nabla p = -\gamma p\nabla\cdot\mathbf{v} + \rho\nabla\cdot\kappa\,\nabla\,(p/\rho) \end{cases}$$

The fast parallel equilibration of T is modeled using wave equations;

$$\begin{cases} \partial T/\partial t = s\,\mathbf{B}/\rho\cdot\nabla u \\ \partial u/\partial t = s\,\mathbf{B}\cdot\nabla\mathbf{T} + \upsilon\nabla^2 u \end{cases}$$

s = wave speed/ v_A

Two-fluid MH3D-T

(Sugiyama et al.)

- Solves the two fluid equations with gyro-viscousity and neoclassical parallel viscousity terms in a torus.

- **Equations**

$$\begin{cases} \mathbf{v} \equiv \mathbf{v}_i - \mathbf{v}_i^* = \mathbf{v}_e - \mathbf{v}_e^* + \mathbf{J}_{\parallel}/en, \\ \mathbf{v}_e^* \equiv -\mathbf{B}\mathrm{x}\nabla p_e\ /(enB^2), \quad \mathbf{v}_i^* \equiv \mathbf{v}_e^* + \mathbf{J}_{\perp}/en, \end{cases}$$

$$\rho\partial\mathbf{v}/\partial t + \rho\mathbf{v}\cdot\nabla\mathbf{v} + \rho(\mathbf{v}_i^*\cdot\nabla)\mathbf{v}_{\perp} = -\nabla p + \mathbf{J}\times\mathbf{B} - \mathbf{b}\cdot\nabla\cdot\Pi\mathrm{i},$$

$$\partial\mathbf{B}/\partial t = -\nabla\times\mathbf{E}, \quad \mathbf{E} = (-\mathbf{v}\times\mathbf{B} + \eta\mathbf{J}) - \nabla_{\parallel}p_e/en - \mathbf{b}\cdot\nabla\cdot\Pi\mathrm{e},$$
$$\mathbf{J} = \nabla\times\mathbf{B},$$

$$\partial\rho/\partial t + \nabla\cdot(\rho\mathbf{v}_i) = 0,$$

$$\begin{aligned} \partial p/\partial t + \mathbf{v}\cdot\nabla p = & -\gamma p\nabla\cdot\mathbf{v} + \rho\nabla\cdot\kappa_{\parallel}\nabla_{\parallel}(p/\rho) \\ & - \mathbf{v}_i^*\cdot\nabla p + (1/en)\mathbf{J}\cdot\nabla p_e \\ & - \gamma p\nabla\cdot\mathbf{v}_i^* + \gamma p_e\mathbf{J}\cdot\nabla(1/en) \end{aligned}$$

$$\begin{aligned} \partial p_e/\partial t + \mathbf{v}\cdot\nabla p_e = & -\gamma p_e\nabla\cdot\mathbf{v} + \rho\nabla\cdot\kappa_{\parallel}\nabla_{\parallel}(p_e/\rho) \\ & + (1/en)\mathbf{J}_{\parallel}\cdot\nabla p_e - \gamma p_e\nabla\cdot(\mathbf{v}_e^* - \mathbf{J}_{\parallel}/en) \end{aligned}$$

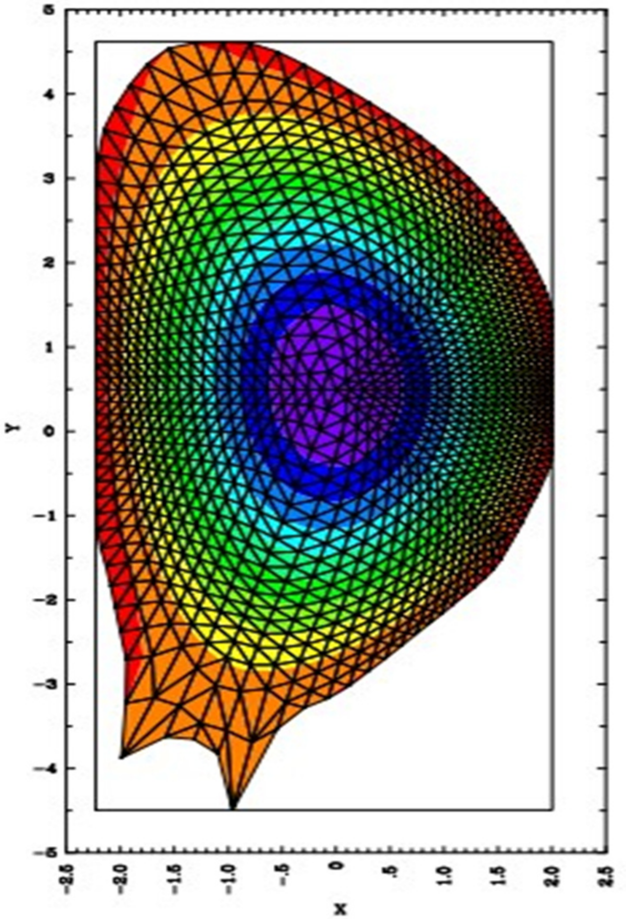
M3D mesh and equilibrium initialization:

Triangular piecewise linear element are used in the poloidal (R,Z) plane for an ustructured mesh, and a pseudospectral Fourier representation is used in toroidal (ϕ) direction

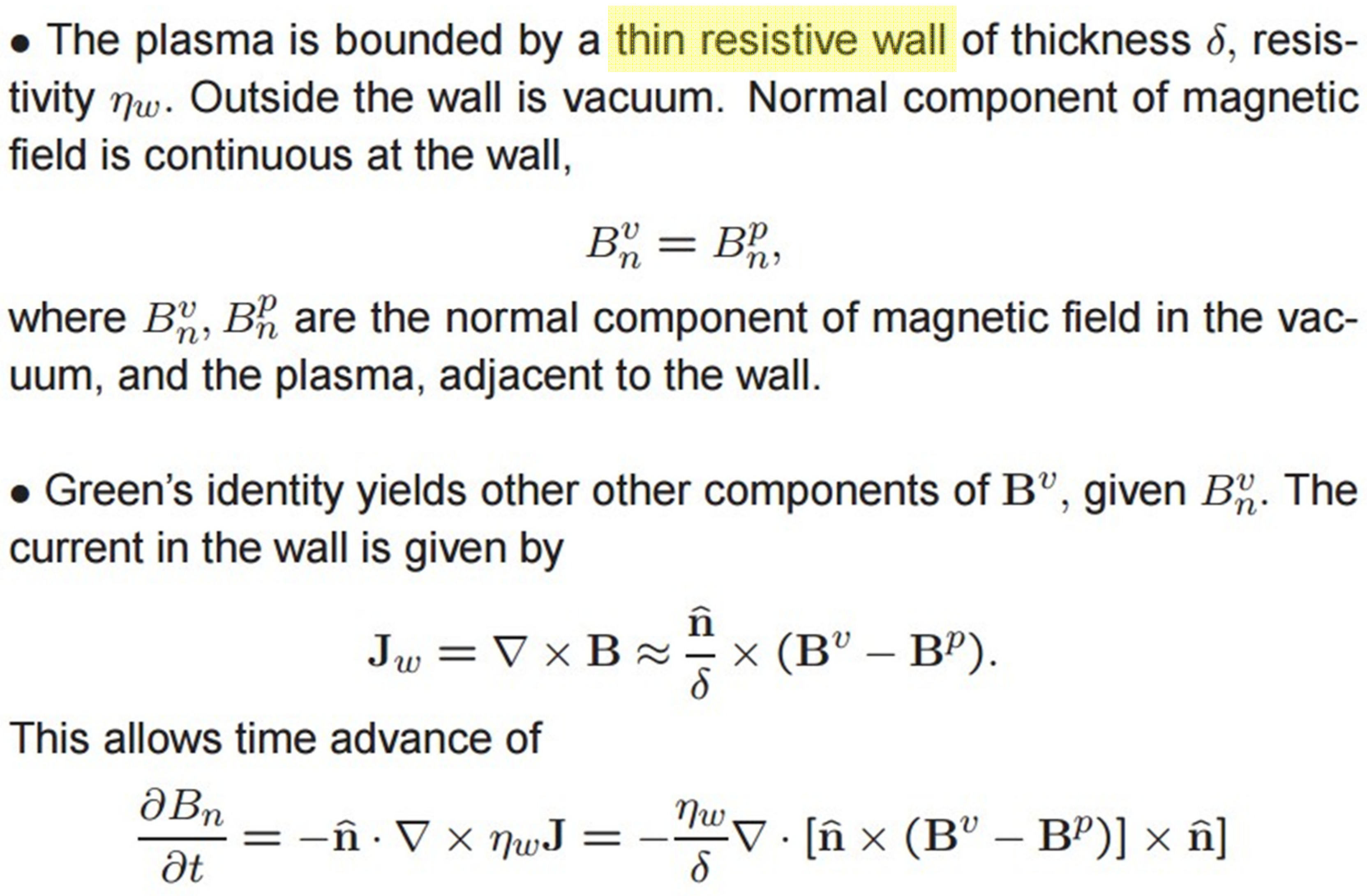
The open field line vacuum region surrounding the plasma is modeled as a low density high resistivity plasma.

Upwinding and dealiasing provide adeguate numerical stabilization.

The initial equilibrium can be read from an **eqdsk file** obtained from real data or as result of an **equilibrium MHD code**



M3D boundary conditions: (1)



- The plasma is bounded by a thin resistive wall of thickness δ, resistivity η_w. Outside the wall is vacuum. Normal component of magnetic field is continuous at the wall,

$$B_n^v = B_n^p,$$

where B_n^v, B_n^p are the normal component of magnetic field in the vacuum, and the plasma, adjacent to the wall.

- Green's identity yields other other components of $\mathbf{B}^v$, given B_n^v. The current in the wall is given by

$$\mathbf{J}_w = \nabla \times \mathbf{B} \approx \frac{\hat{\mathbf{n}}}{\delta} \times (\mathbf{B}^v - \mathbf{B}^p).$$

This allows time advance of

$$\frac{\partial B_n}{\partial t} = -\hat{\mathbf{n}} \cdot \nabla \times \eta_w \mathbf{J} = -\frac{\eta_w}{\delta} \nabla \cdot [\hat{\mathbf{n}} \times (\mathbf{B}^v - \mathbf{B}^p)] \times \hat{\mathbf{n}}]$$

M3D boundary conditions: (2)

$$\mathrm{B_v} = \nabla\psi_v \times \nabla\phi + \nabla\lambda + I_o \nabla\phi$$

Vacuum magnetic field

GRIN Solver:

$$\left(\frac{\partial \psi_v}{\partial n}\right)_i = \sum\nolimits_j K^o_{ij}\,\psi_{pj} + S_i \quad \rightarrow \quad \frac{\partial \psi_w}{\partial t} = \frac{\eta_w}{\mu_o \delta_w}\left[\frac{\partial \psi_w}{\partial n}\right]$$

[ ] → Jump

$$(\lambda^n)_i = \sum\nolimits_j K^n_{ij}\,(\mathbf{B}_p \cdot n)_j \quad \rightarrow \quad \frac{\partial B_{n\,pw}}{\partial t} = \frac{\eta_w}{\delta_w}\left[\frac{\partial B_{nw}}{\partial n}\right]$$

Virtual “case” $S_i = \frac{\partial \psi_p}{\partial n}$ @ t=0 to impose Ψ=0 at the wall

M3D important physical parameters:

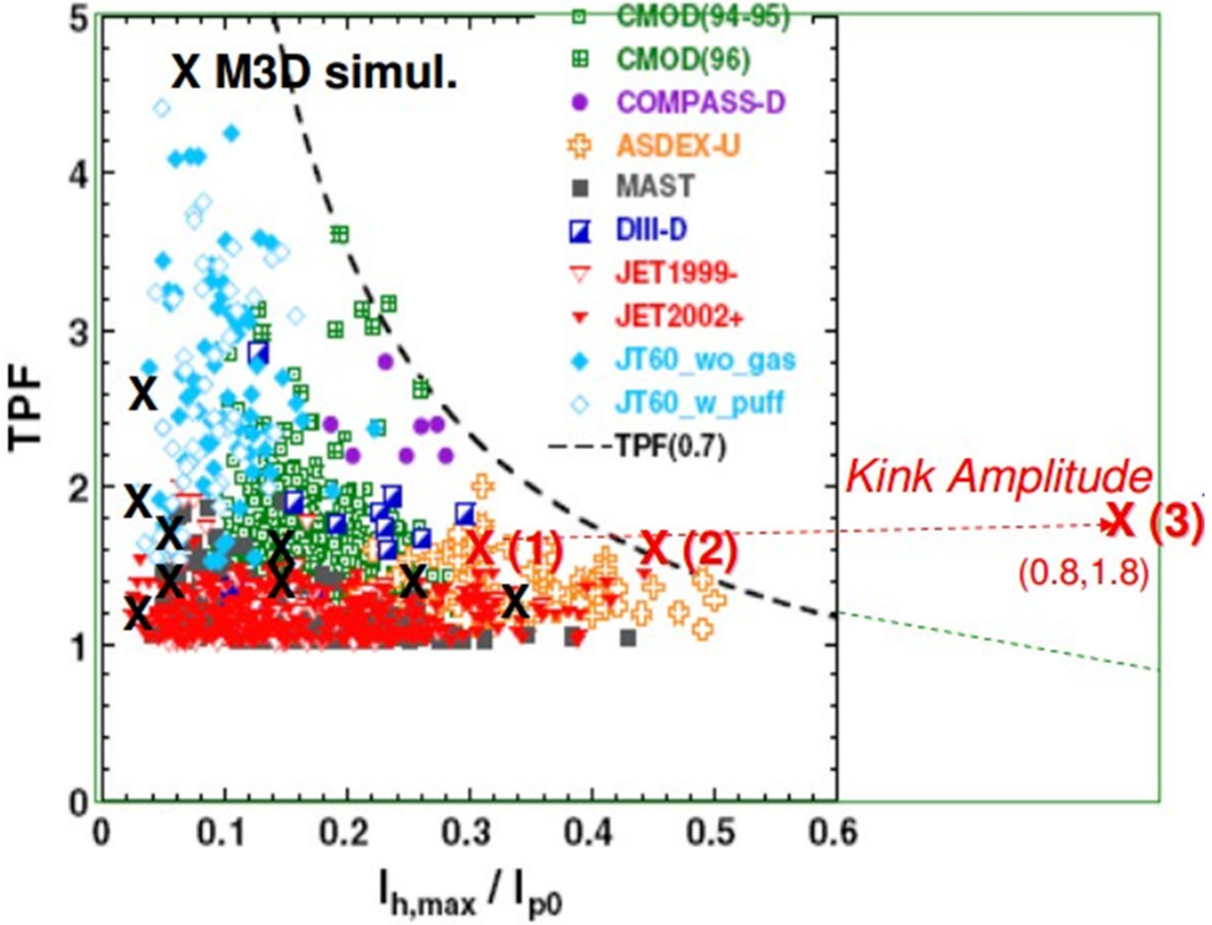
Some of the **free parameters** in the code are of **particular importance** for the disruption simulations. In particular:

- **S**, the Lundquist number (It is mainly limited by the achievable numerical resolution)
- μ the plasma viscosity (smooth the length scales of the turbulence)
- η_{out} the resistivity of the **outer plasma layer** (from the separatrix to the wall)
- η_{wall} the wall time constant (the longer the slower the penetration the longer the simulation time)
- **s** the sound wave related parameter (linked to the parallel tranport)
- χ_{perp} the perpendicular transport coefficient

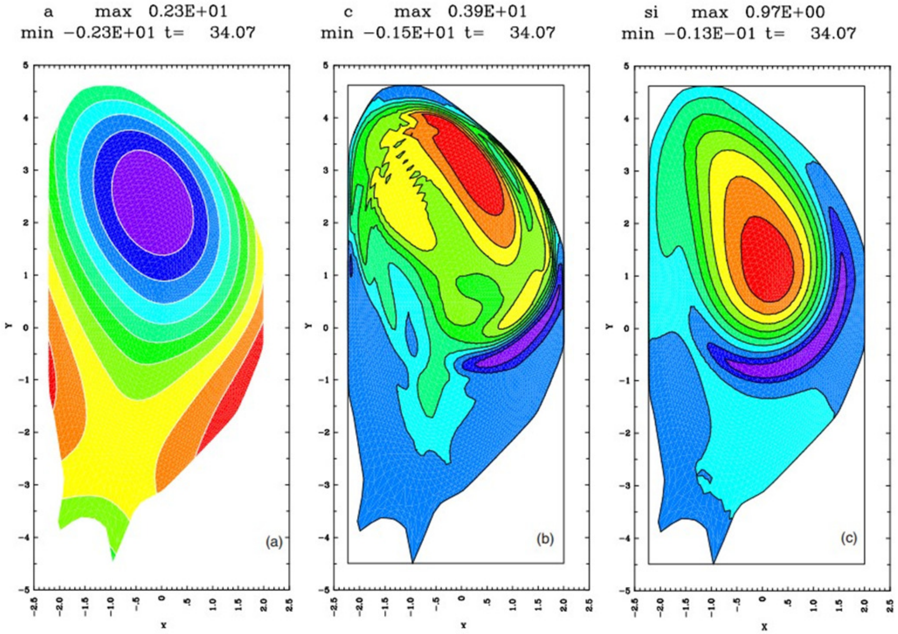
M3D code : TPF vs halo fraction

A relatively **fast kink** develops in numerical simulations.

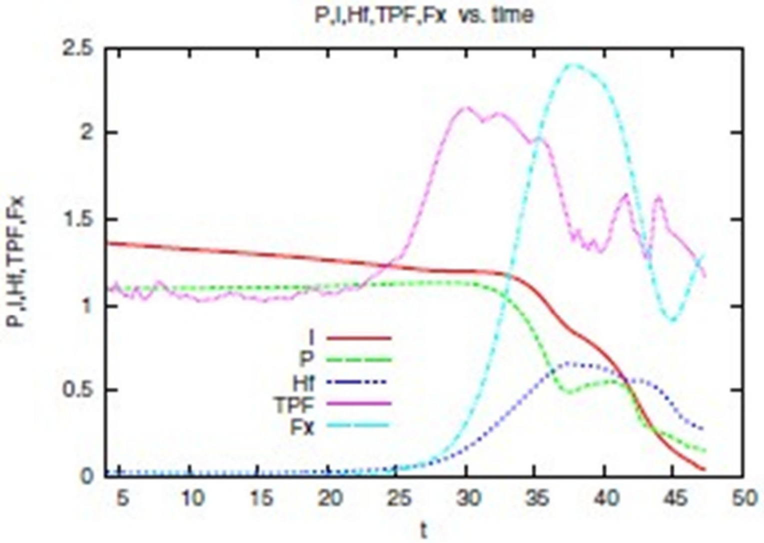
TPFs and halo fractions are consistent with the experimental database.

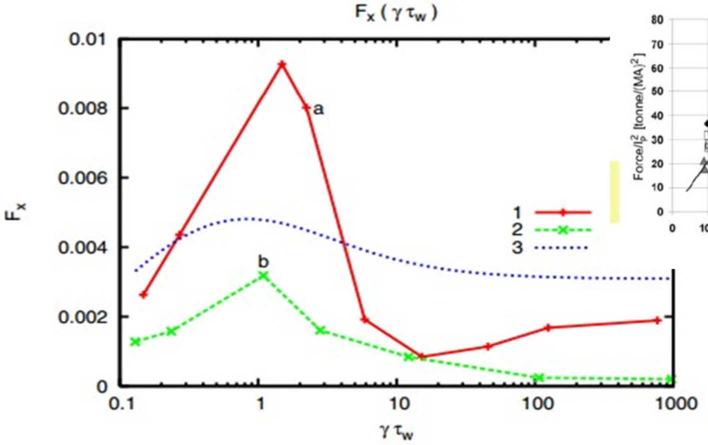


M3D simulations results : horizontal force scaling

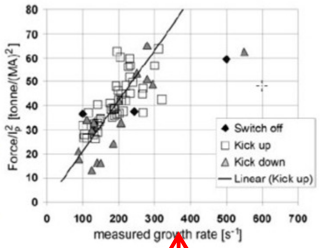


From Strauss et al PoP (2010)

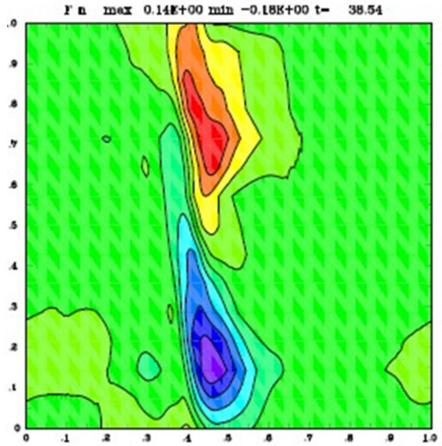




Force vs growth time



$\gamma\tau_w$=2

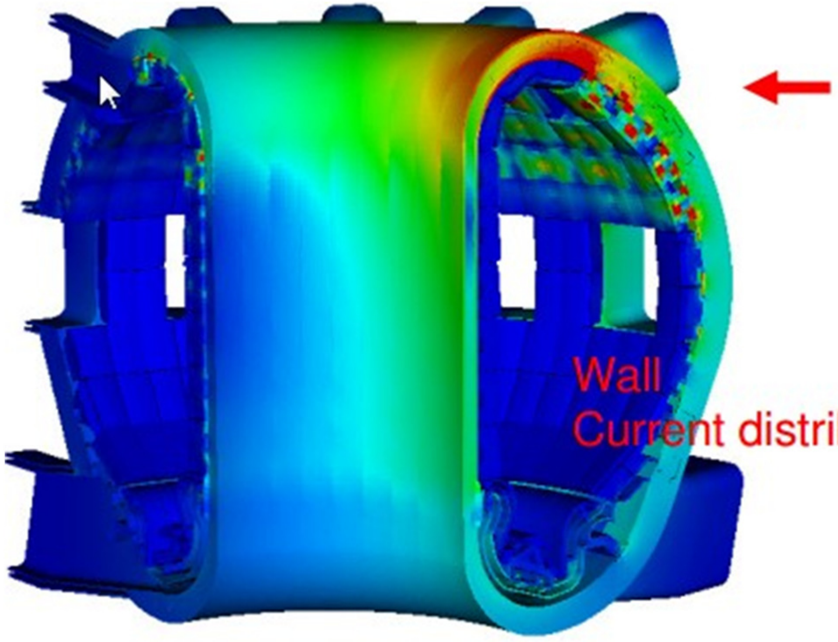


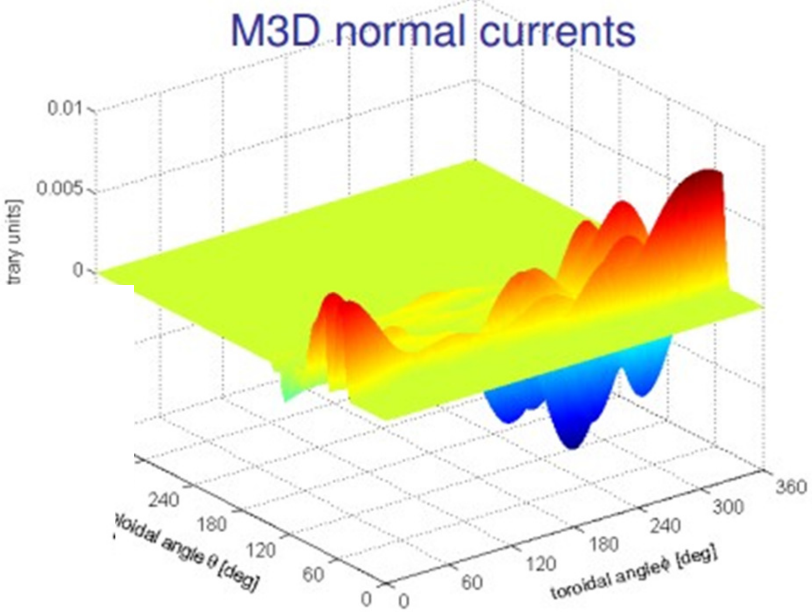
Normal force at the wall vs poloidal and toroidal angles (x and y axis)

n=1 structure

Realistic wall effects and horizontal forces

Realistic model of ITER wall





Clearly the forces depend also on the distibution of wall currents and therefore on wall real geometry

M3D simulations : current and mag. moment correlation (1)

From Strauss et al PoP (2010)

By assuming: $J_\phi = J_{\phi 0}(r - \xi_{\mathrm{VDE}} \sin\theta) + J_{\phi 1}(r - \xi_{\mathrm{VDE}} \sin\theta)\cos(\theta + \phi)$

The **perturbed toroidal current** can be calculated as:

$$\boxed{\begin{aligned} I_{\phi 1} &= -\int dr r d\theta \frac{dJ_{\phi 1}}{dr} \xi_{\mathrm{VDE}} \sin\theta \cos(\theta + \phi) \\ &= -\pi \xi_{\mathrm{VDE}} \int dr J_{\phi 1} \sin\phi, \end{aligned}}$$

The magnetic moment is instead:

$$\boxed{\begin{aligned} M_{IZ} &= \int d\theta dr r^2 \sin\theta J_{\phi 1} \cos(\theta + \phi) \\ &= -\pi \int dr r^2 J_{\phi 1} \sin\phi. \end{aligned}}$$

Therefore assuming $J_{\phi 1} = K_a \delta(r - a)$:

$$\boxed{\frac{dI_\phi}{d\phi} = \frac{\xi_{\mathrm{VDE}}}{a^2} \frac{dM_{IZ}}{d\phi}.}$$

M3D simulations : current and mag. moment correlation (2)

On the other hand from $\nabla \cdot \boldsymbol{J} = 0$ a simple relation can be deduced:

$$\partial I / \partial \phi = - \oint J_n R dl = - \tilde{I}_{halo}$$

Which shows a 90 degree phase shift between the toroidal variation of the toroidal current and the halo current (as noted in experiments at JET).
Analogously from:

$$\nabla \cdot \mathrm{B} = 0, \partial \Phi / \partial \phi = - \oint R B_n dl$$

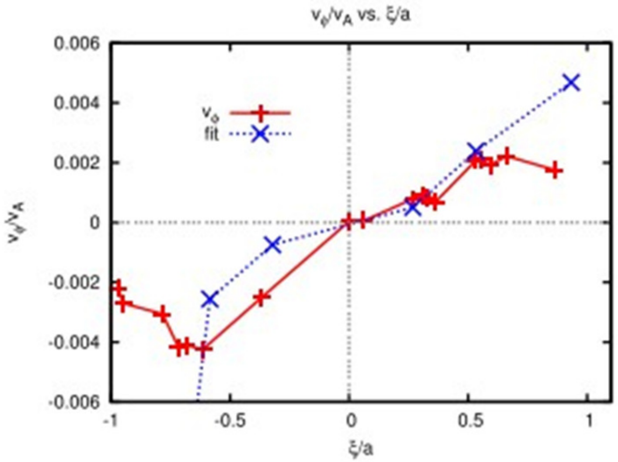
And assuming : $J_n \approx B_n / a$ it can be seen that $\Delta \boldsymbol{\phi} \approx \Delta \boldsymbol{I}$

again **similarly** to what observed **in experiments at JET**.

M3D simulations: rotation

From Strauss et al NF (2014) and Strauss PoP (2015)

A correlation has been found in simulations between the **VDE vertical displacement and the plasma rotation**



And also an analytical theory has been developed:

$$\boxed{\dot{L}_\phi = -\frac{R}{B_{\phi 0}} \oint \frac{\partial \psi_0}{\partial \theta} p \, d\theta \, d\phi.} \quad \longrightarrow \quad \boxed{\dot{L}_\phi = \frac{\pi^2}{8} \beta_N' \xi_{10}^3 R r \sum_{mn} \frac{\partial}{\partial r} \left\{ \frac{[1 + m(m+1)] B_{\theta mn} B_{\theta (m+1) n}}{(m - nq)(m + 1 - nq)} \right\}.}$$

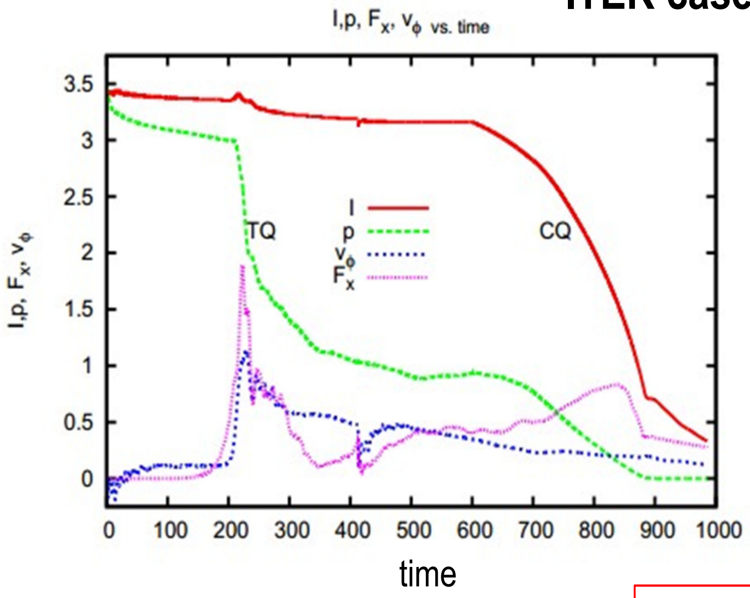
Taking into account that:

$$\frac{\partial \psi_0}{\partial \theta} = \xi_{10} \cos\theta \, B_{\theta 0},$$

and that at the second order in perturbation : $p_2 = \frac{p_0'}{2r^2} \frac{\partial}{\partial r} \left(\frac{\partial \xi}{\partial \theta} \right)^2.$

M3D simulations: sustained current

ITER case



In MHD simulations **TQ and CQ are** quite **coupled**

TQ and CQ can be decoupled in sustained cases, where an external electric field is applied to sustain the plasma current.

IN CONCLUSION:

- **progresses have been done in simulating AVDE's time behavior , forces and also rotation**

however

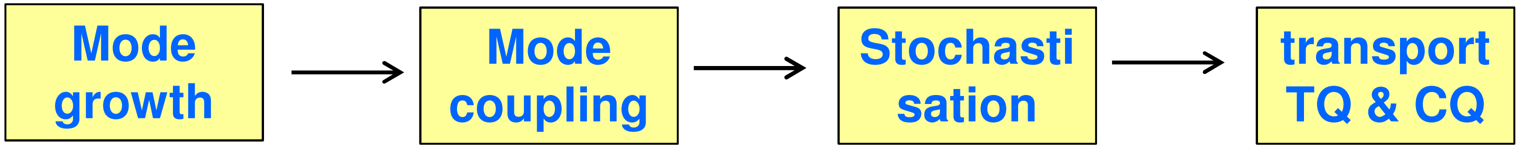
- numerical resolution is generally low (up to n = 6-8)
- simulations could not reach realistic collisionality regimes
- kinetic effects are completely neglected
- flow is generally absent from the initial equilibrium
- transport is likely not realistically modelled

Is the fast experimental thermal quench a mistery?

- In experiments the thermal quench is a fast phenomenon sometime **without clear precursor**, or at least **without** from the **outside measurable** big **MHD modes**

- plasma internal energy is **suddently released in msec timescale (or faster)**

- **in simulations** (apart sustained cases) **TQ and CQ are simultaneous** and **follow the modes growth**:


Mode growth → **Mode coupling** → **Stochastisation** → **transport TQ & CQ**

So the question is :

Are there different mechanisms that can explain the fast experimental TQ ?

Stochastic transport (1)

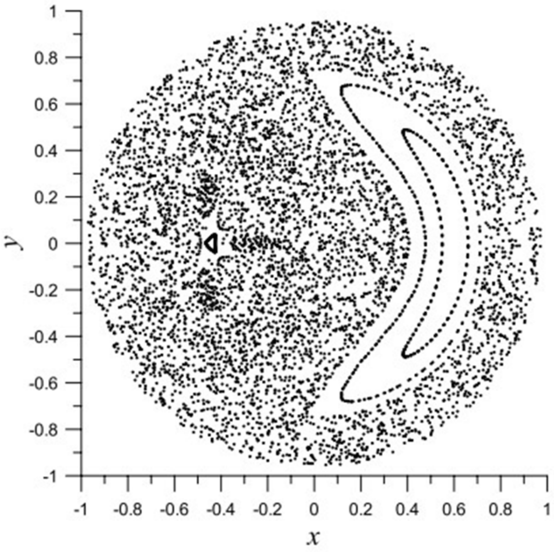
An **axi- symmetric tokamak has well conserved (2D) flux surfaces**.
However if there are non symmetric perturbations the magnetic filed can be described by a perturbed hamiltonian like:

$H_{tot} = H_o(J) + H(J, \alpha, \varphi)$ with: $$H(J,\alpha,\varphi) = \sum_{m,n} H_{m,n}(J)\cos[m\alpha + (m-n)\varphi]$$

Harmonics overlapping can lead to field line stochasticity

In turn harmonic overlapping depends on the locations and amplitudes of the modes at the resonant radii (determined by the q profile). For 2 modes the threshold is obtained for **s >1** with:

$$s = \frac{1}{2}\frac{(w_{m,n} + w_{m1,n1})}{(|r_{m,n} - r_{m1,n1}|)}$$ *(Chirikov parameter)*



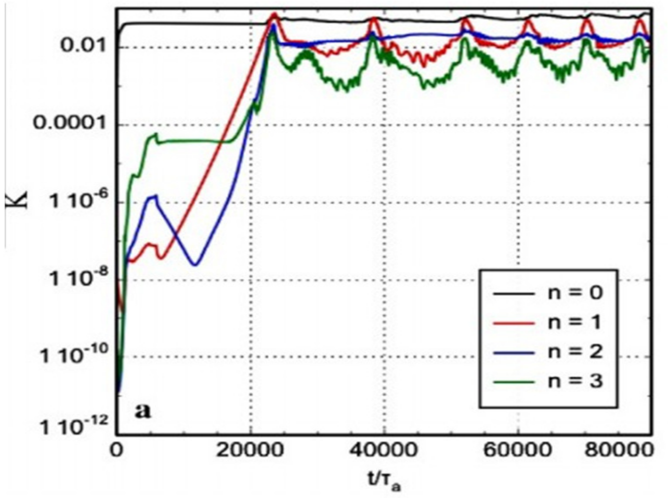
The **electron thermal diffusion in stochastic fields** can be estimated (collisionless) as:

$\boldsymbol{\chi_e = D_{st}\, v_{th,e}}$ with $D_{st} = \langle (b_r/B)^2 \rangle\ L_c$ and $L_c \approx \pi R$

Stochastic transport (2)

- Although the stochastic transport could be quite fast (if $\frac{\mathrm{b}}{\mathrm{B}} \approx 10^{-3}$ and T_e= 3 Kev , $\chi \approx 100\,\frac{m^2}{s}$) compared with standard transport **, a quantitative estimate is difficult** since often the **q profile is only approximately reconstructed**, **the spectrum and the amplitudes of the modes** are also **not very well known** from external measurements.

- Not just for the **TQ**, but even for more **standard phenomena** in tokamaks, like the **sawtooth crashes** that are observed in the core plasma region when the q on axis approaches 1, **there is no firm agreement** about what is determining the temperature crashes and if they can be linked to an enhanced stochastic thermal diffusion through higher harmonic generation, as found in some simulations.



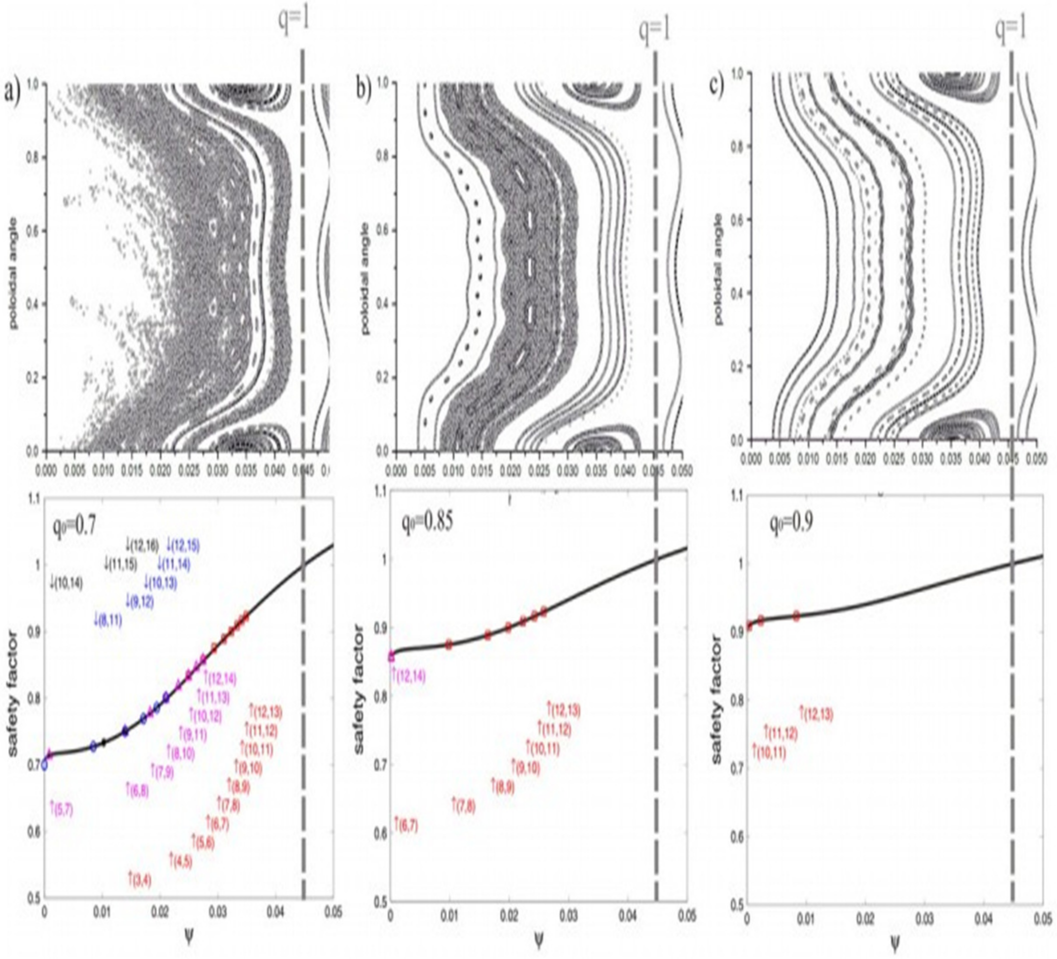
Nonlinear XTOR simulations showing the generation of high n mode numbers

The physical parameters are not extremely realistic:

$S = 10^6, \chi_{\parallel} = 100\ , \chi_{\perp} = 10^{-5}\ ,\ Pr = \frac{\nu}{\eta} = 1\ (?)$

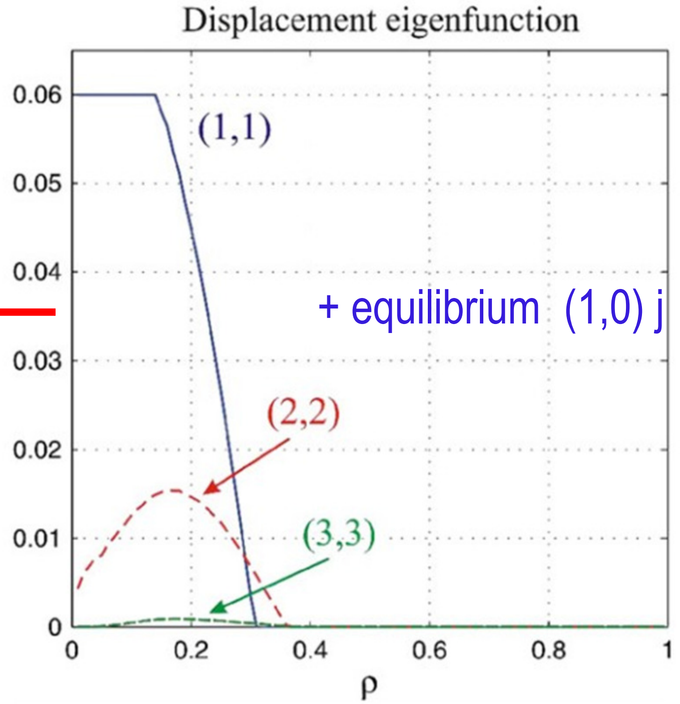
From Luetjens et al JCP (2010)

Stochastic transport (3)



From Igochine et al NF (2008) and PoP(2010)

Experimental results in AUG tokamak have shown that the appearance of chaos **during sawtooth activity** is very sensitive to the q profile near the axis

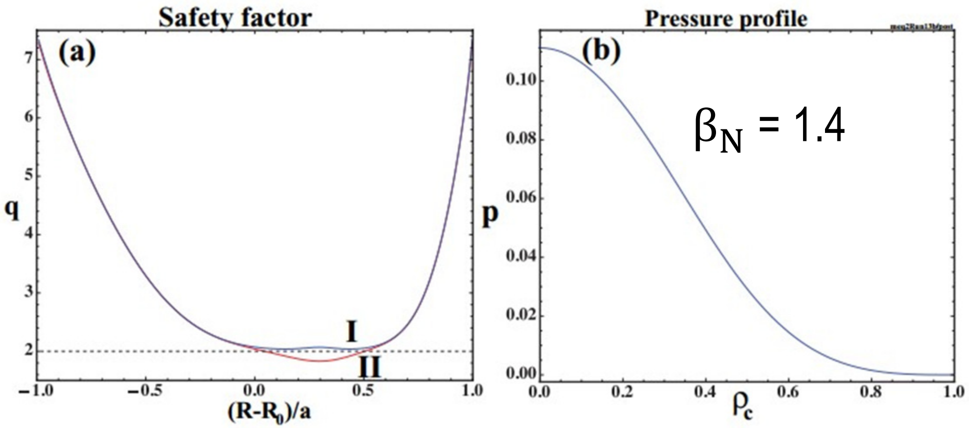


+ equilibrium (1,0) j

TQ possible alternatives: explosive instability at relatively low β_N

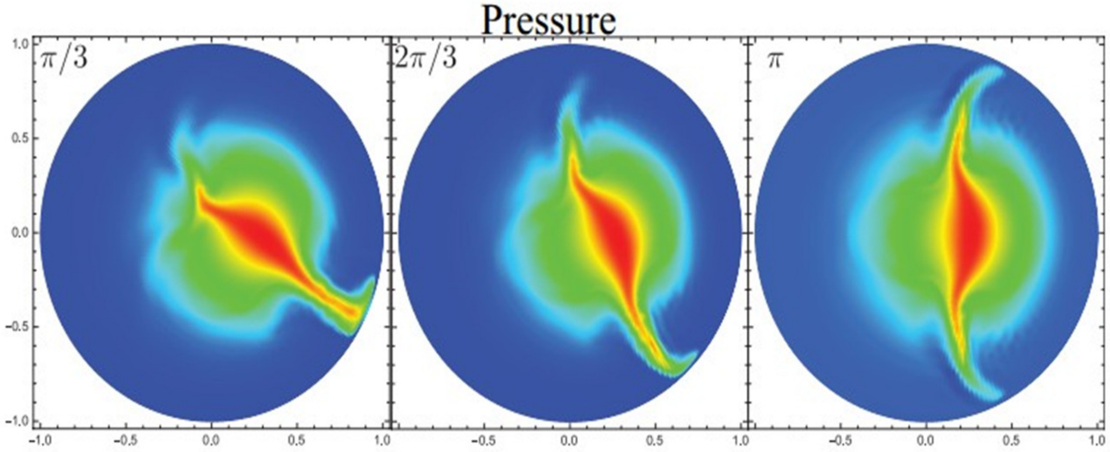
From A Y Aydemir et al, NF 56 (2016)

CTD spectral toroidal code



I, II both ideally unstable (effect of geometry?)

Explosive pressure fingers development



High harmonics **up to n=30**
in simulation
High numerical resolution

ITER and **Fusion Relevant (FR)** scenarios and disruptions

- Do **FUSION RELEVANT (FR)** low disruptivity scenarios exist ?
- Is it possible to **classify FR** scenarios according to **disruptivity**?
- what we know about **scalings to larger devices** ?

Fusion relevance and critical limits

- Fusion power scales $\mathrm{p}^2 \sim (nT)^2$
- Plant efficiency scales $\sim \beta \sim \frac{n\,T}{B^2}$

Density is limited by the so called Murakami/Greenwald limit:

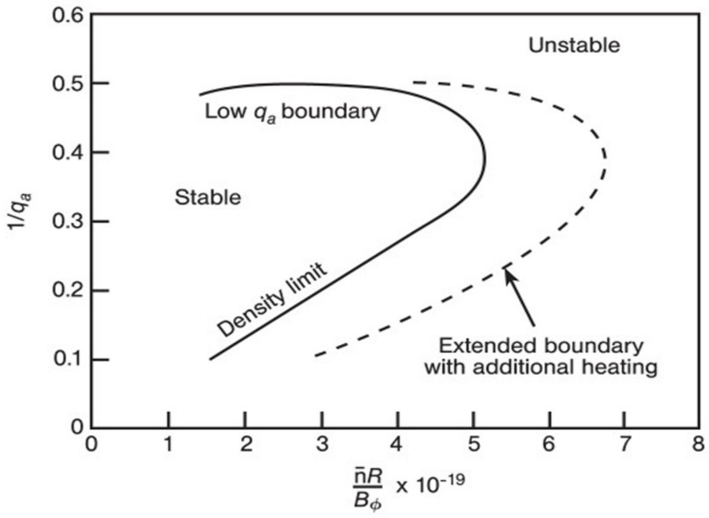
$n_G\ [m^{-3}]$ = **I** [MA] / (π a^2 [m])

..not well understood (but likely connected with input/output energy balance)
.. experimentally clear → ***DISRUPTIONS ABOVE GL*** *(or near to it..)*

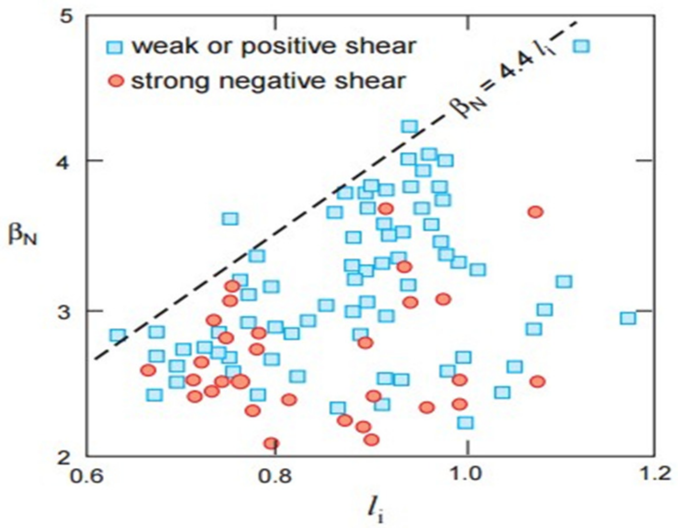
β limited by ideal MHD instabilities (ISL): $\beta_T \leq C\ (\frac{I}{a\ B})$ (or $\beta_N\ \leq$ C (3-4)
.. experimentally reasonably confirmed (standard non rev. shear plasmas)
→ ***DISRUPTIONS ABOVE or NEAR ISL***

The limits in terms of simple macroscopic parameters

Hugill plot for density limit and q(a)



IPB-NF 1999 (DIIID data)



From Stacey:Fusion Plasma Physics (Wiley 2012)

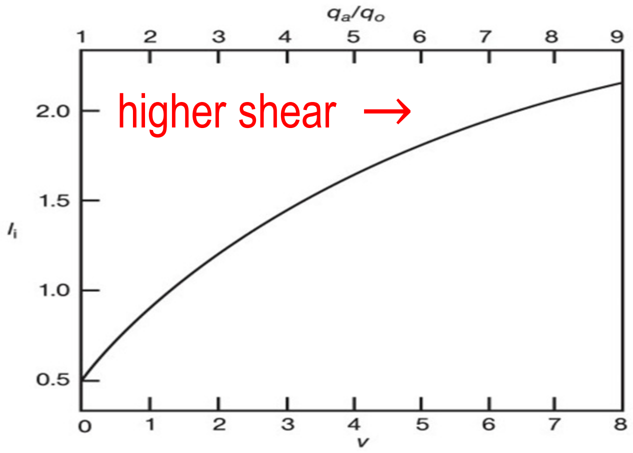
$$\boxed{\begin{aligned} p &= p_0\left(1-\frac{r^2}{a^2}\right) \\ j &= j_0\left(1-\frac{r^2}{a^2}\right)^{\nu} \end{aligned}} \rightarrow$$

$$l_i = \frac{\bar{B}_\theta}{B_{\theta a}^2} = \frac{2\int_0^a B_\theta^2 r\, dr}{a^2 B_{\theta a}^2}$$

$$\boxed{l_i = \ln(1.65 + 0.89\nu)}$$

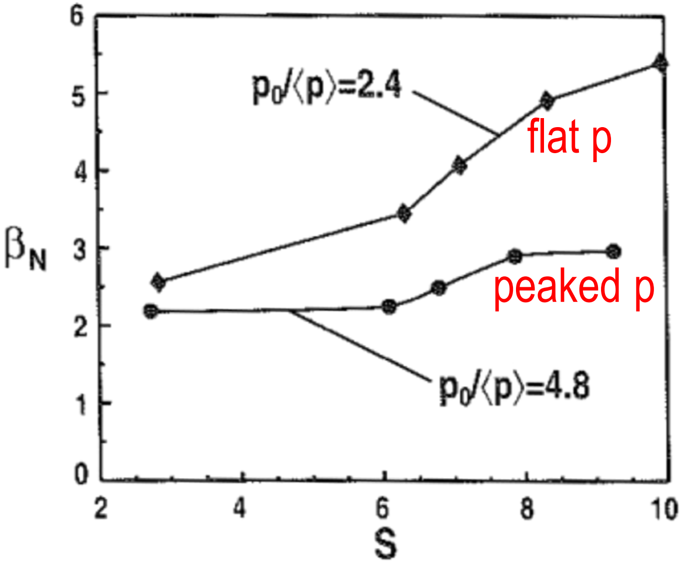
and

$$\boxed{q_a/q_0 = \nu + 1}$$



ISL in toroidal geometry and shaped plasmas

Turnbull et.al. NF (1998), β limits

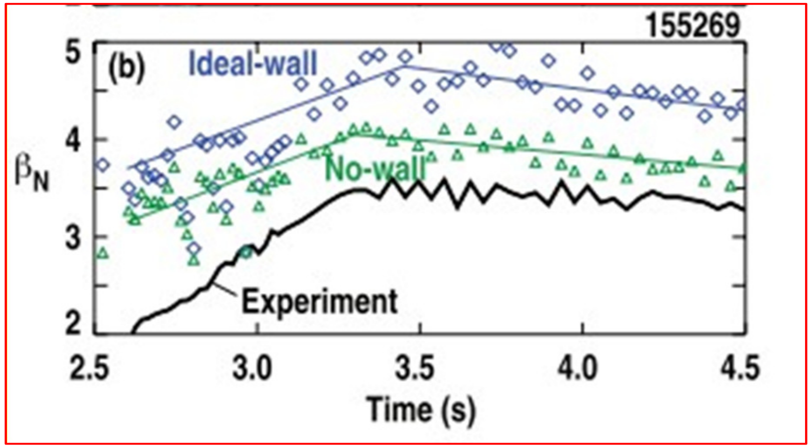


$$shape\ factor\ S = \left(\frac{I}{aB}\right)\ q_{edge}$$

However limitation are due to:

- broad p enhances f_{bs} at edge and limits achievable l_i
- therefore high l_i also depends on possibility of reducing outer pedestal height (lower f_{bs}) (..with ELMs control)
- low q(0) could maximize l_i but sawth. limits q(0) to 1

ITER like plasmas in DIIID



from Ferron et.al. NF (2015)

Conditions for high β and performing plasmas (i e FR)

- High β_N , High l_i , high shear
- High shaping S
- as low as possible q(0)/q_{min}
- low outer pedestal h.(by ELM control)
- flat pressure

→

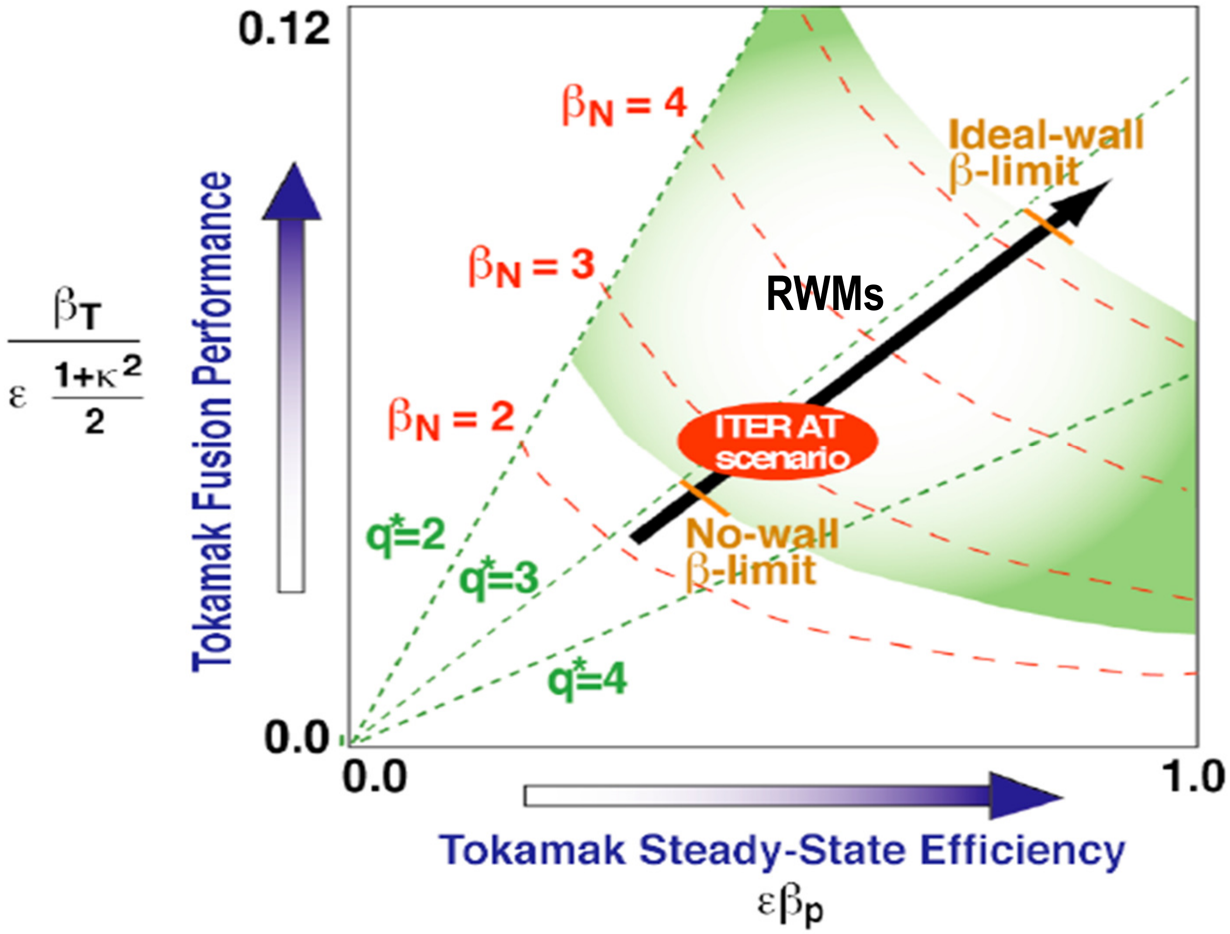
Clearly quite contradictory with:
disruption limits/avoidance and safe tokamak operation!

not to speak of:

- *Low plasma rotation (..eventually)*
- *Plasma wall proximity (see later)*

in larger devices

Fusion performances and β limits



YQ Liu, Peking University, Feb 16-20, 2009

What is a Resistive Wall Mode (RWM) ?

❑**External ideal kink instability** (time scale = microseconds)

❑Normally **pressure-driven** (above no-wall beta limit)

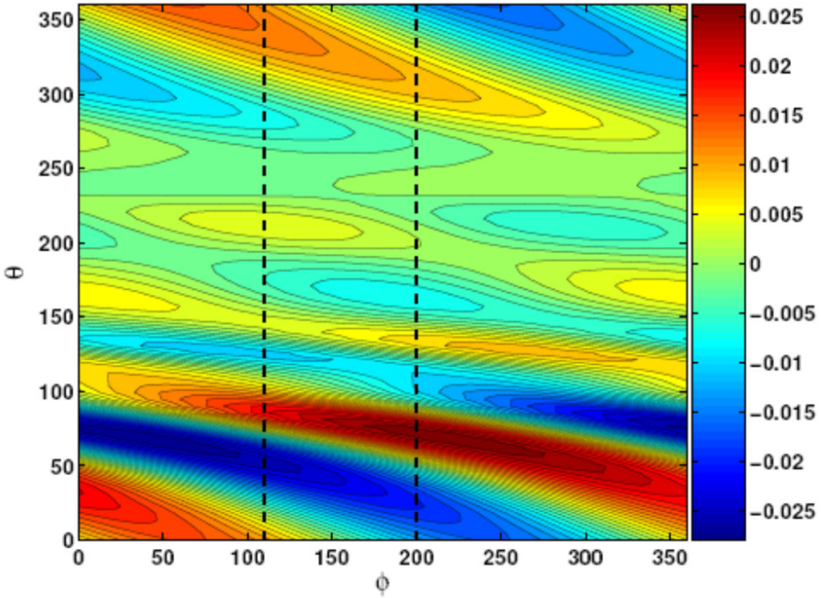
❑**Resistive wall slows down** kink instability to time scale of wall eddy current decay time ➔ RWM (typically milliseconds)

❑At high pressure, mode **located towards low-field side** (kink-ballooning)

❑Low toroidal **mode number n=1,2,3**

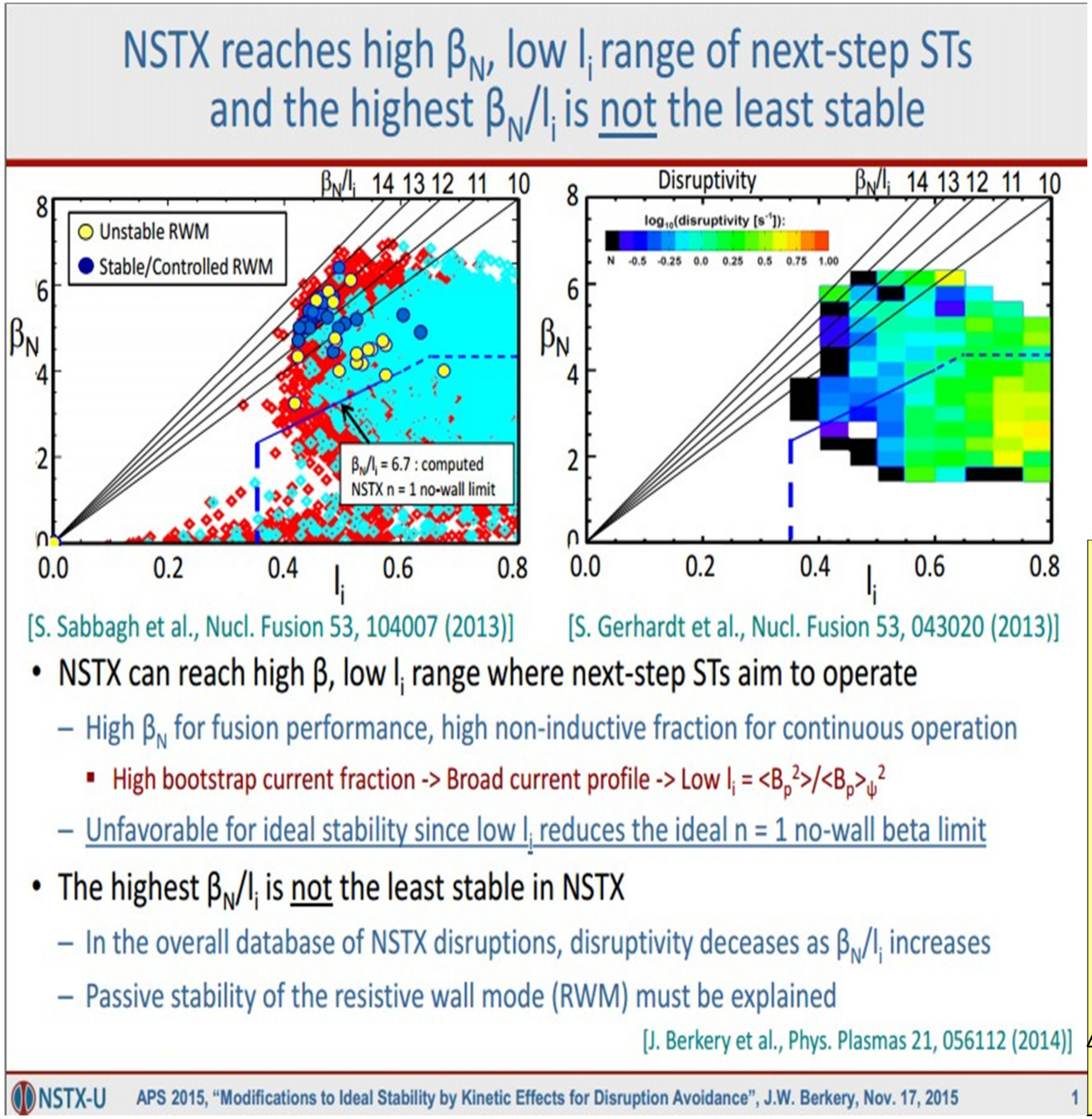
❑Similar to vertical instability (RWM with n=0)

❑Three consequences of slowed down
  - ➢Still unstable ➔ eventually causes disruption
  - ➢Time scale feasible for feedback control
  - ➢Kinetic effects become important



YQ Liu, Peking University, Feb 16-20, 2009

Recent NSTX results seems **to enlarge** the parameter space



NSTX reaches high β_N, low l_i range of next-step STs and the highest β_N/l_i is not the least stable

[S. Sabbagh et al., Nucl. Fusion 53, 104007 (2013)]

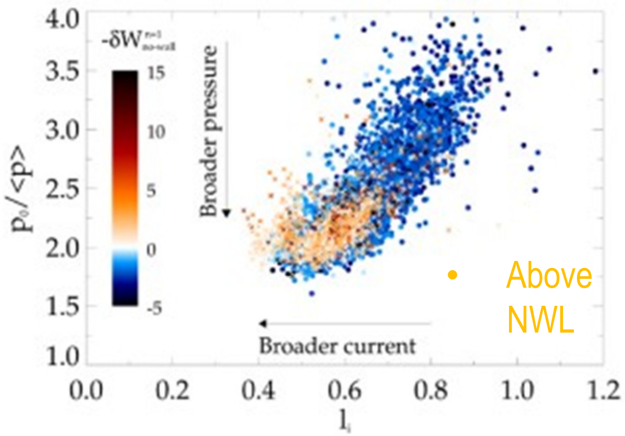
[S. Gerhardt et al., Nucl. Fusion 53, 043020 (2013)]

- NSTX can reach high β, low l_i range where next-step STs aim to operate
  - High β_N for fusion performance, high non-inductive fraction for continuous operation
    - High bootstrap current fraction -> Broad current profile -> Low $l_i = \langle B_p^2 \rangle / \langle B_p \rangle_\psi^2$
  - Unfavorable for ideal stability since low l_i reduces the ideal n = 1 no-wall beta limit
- The highest β_N/l_i is not the least stable in NSTX
  - In the overall database of NSTX disruptions, disruptivity deceases as β_N/l_i increases
  - Passive stability of the resistive wall mode (RWM) must be explained

[J. Berkery et al., Phys. Plasmas 21, 056112 (2014)]

NSTX-U APS 2015, “Modifications to Ideal Stability by Kinetic Effects for Disruption Avoidance”, J.W. Berkery, Nov. 17, 2015 1

J. Berkery et al, NF 55 (2015)



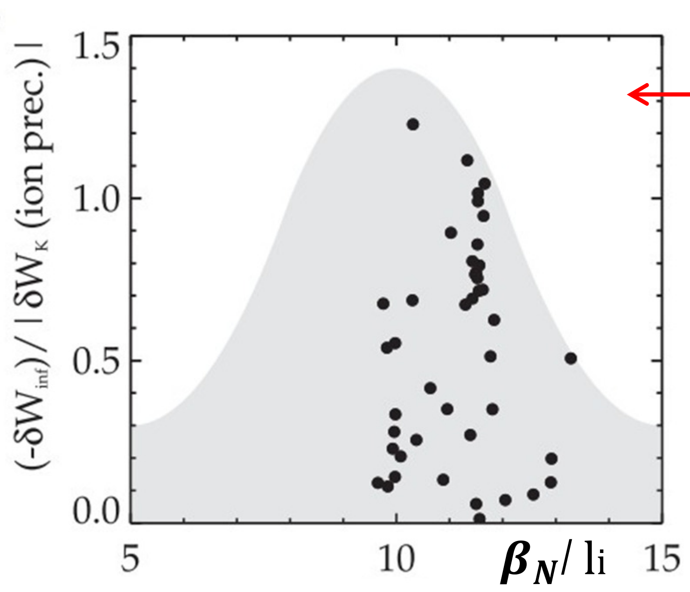
RWM disruption rate from 45% to 14% at low li and high β_N

Interpreted by the combined effect of rotation and kinetic stabilization

BUT ..*low collisionality plasmas are also susceptible to* ***sudden instability when kinetic profiles change..***

Kinetic stabilization of the RWM

J. Berkery et al, PoP 21 (2014)



Misk code calculated kinetic terms for experimental data showing larger effect at high β_N/ li

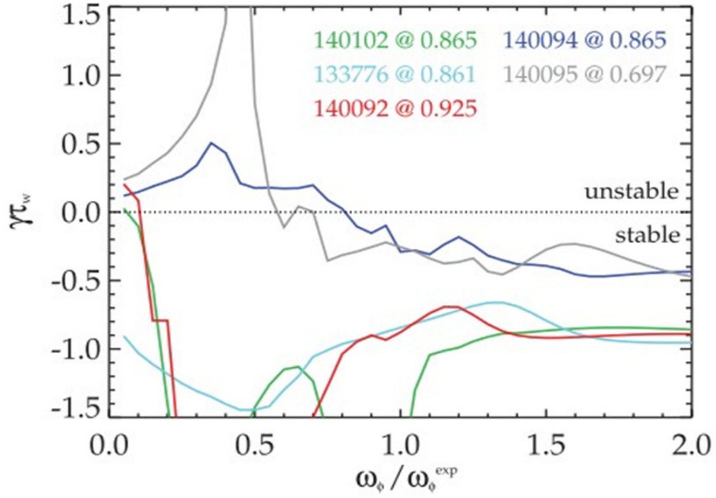
The **relevant resonance** is between the slowly rotating mode and the thermal ions in their precession drift motion:

$$|\omega_E - \omega_D| \approx 0$$

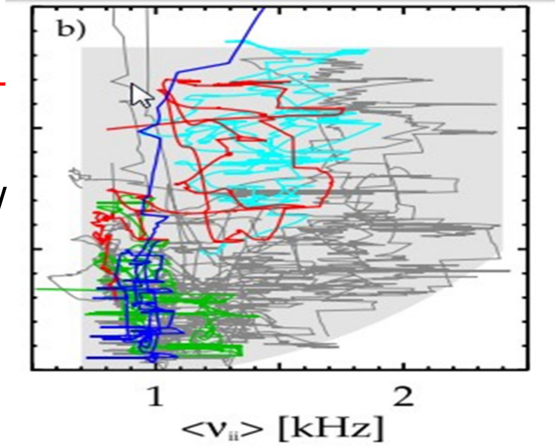
with:

$$\omega_E = \omega_\phi + \frac{1}{en_i}\frac{d}{d\psi}(n_i T_i)$$

and ω_ϕ is the plasma rotation



Collisionality seems also to play a beneficial role for stable cases: **(not for the unstable cases)**

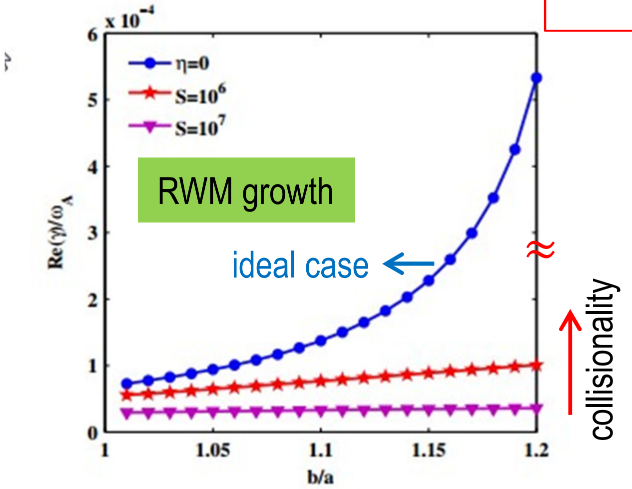


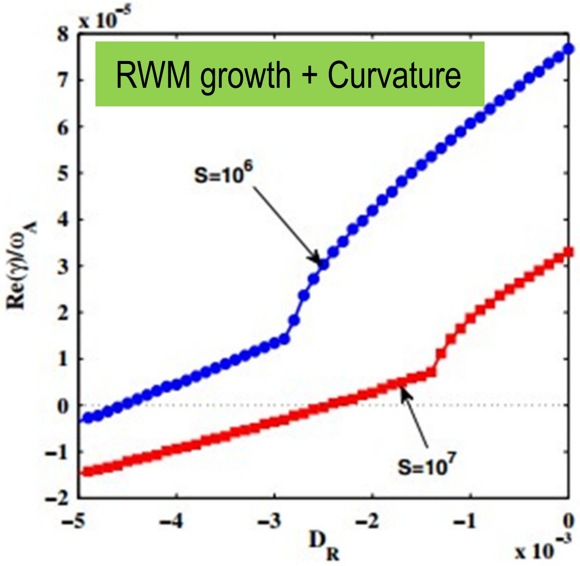
Low collisionality stabilization of RWMs due to resistivity ?

He et.al., PRL (2014)

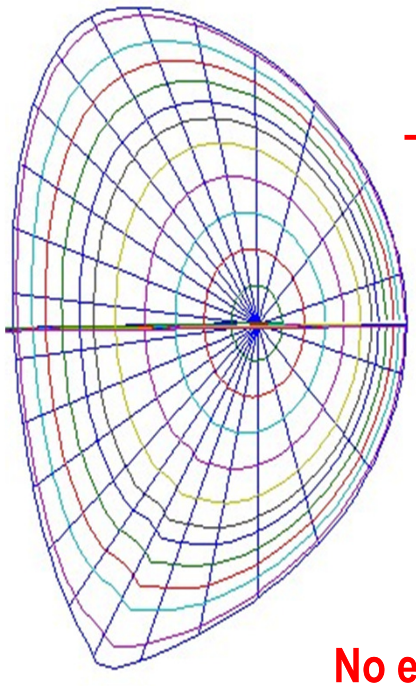
$$\gamma\tau_w^* = -\frac{\delta W^\infty + \delta W_k + \cancel{\delta W_{\mathrm{RL}}}}{\delta W^b + \delta W_k + \cancel{\delta W_{\mathrm{RL}}}},$$

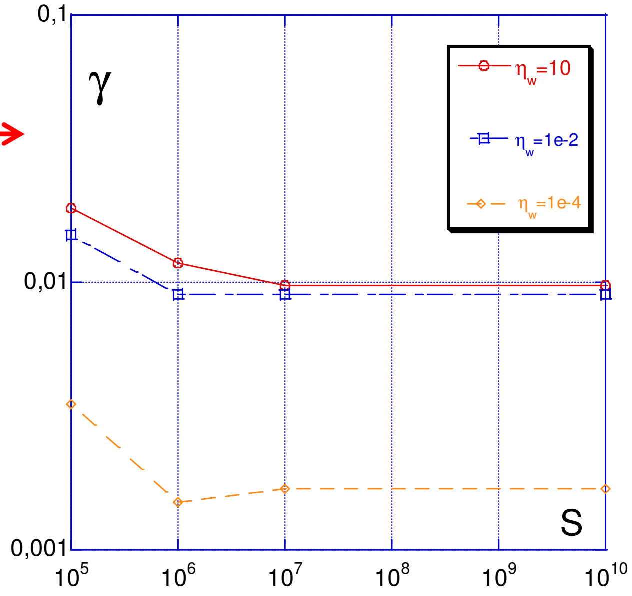
Ideal terms





ITER equilibrium (β_N=4.2 b/a=1.25) **CHEASE&MARS-F**





No effect of resistivity above interesting S (Lundquist) numbers (no kinetic effects here)

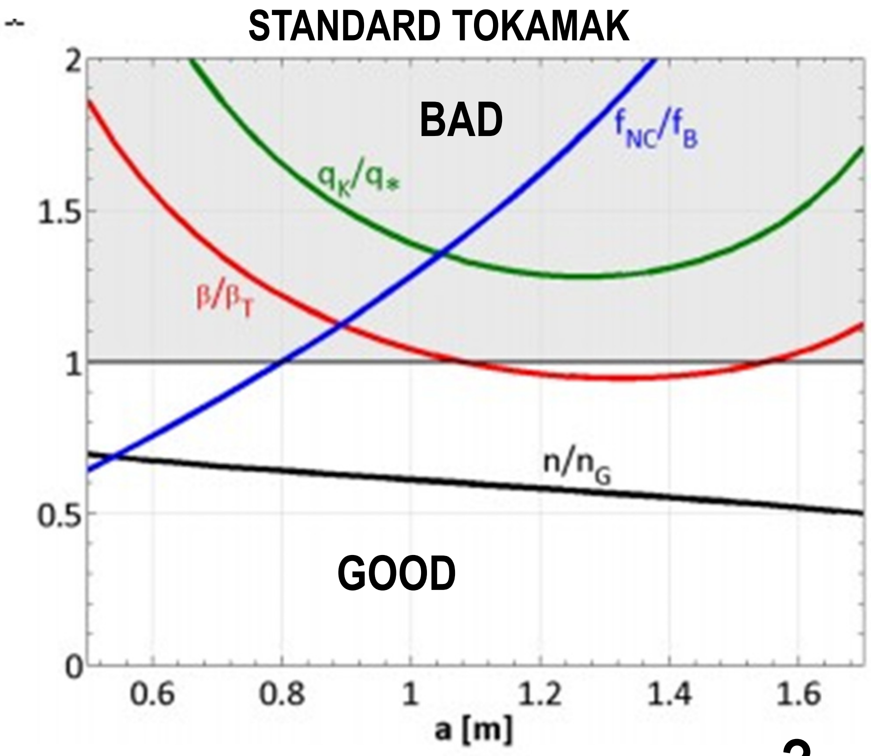
Low li versus High li operation

- low aspect ratio devices operate at larger f_{bs} and broad current (i.e. low li)
- in this case however β_{NW} **<<** β_{IDW} => **wall stabilization is needed to increase** β
- On the other hand **at higher li** (peaked current) $\beta_{NW} \approx \beta_{IDW}$ => **wall** stab. is **less important**

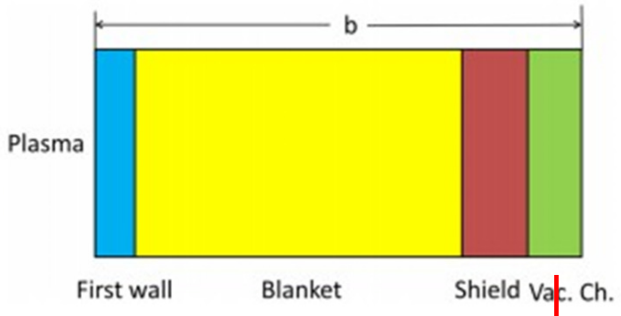
- **Which is the situation in a FR device regarding the wall stabilization effectiveness?**

SCALING OF THE SHELL PROXIMITY IN **FR CASES** ..IS IT A RELEVANT ISSUE ?

*from J. Freidberg et al PoP **22** (2015)*



?



$b \geq 1.2$ m

stabilizing wall

..but for stability with an ideal wall :

$$\frac{b}{a} \leq 1.3 \div 1.4$$

=>

$$a \approx 3 \div 4 \ m$$

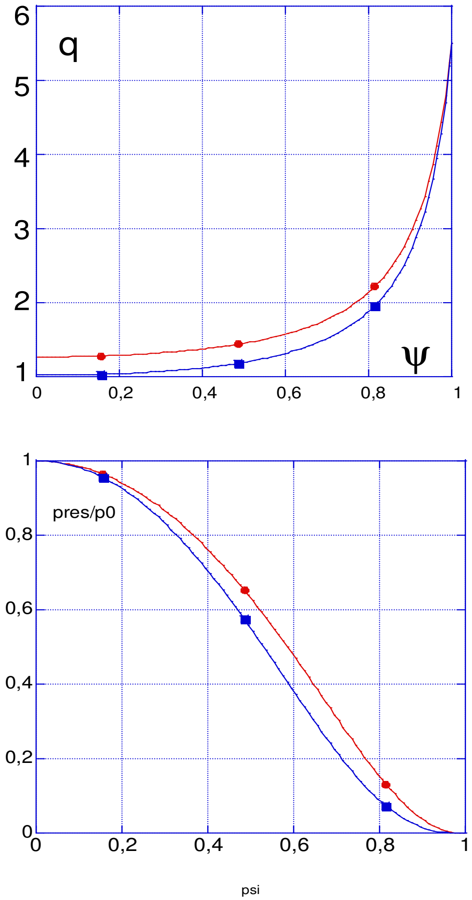
This is a serious constraint to minimum a (and R) !

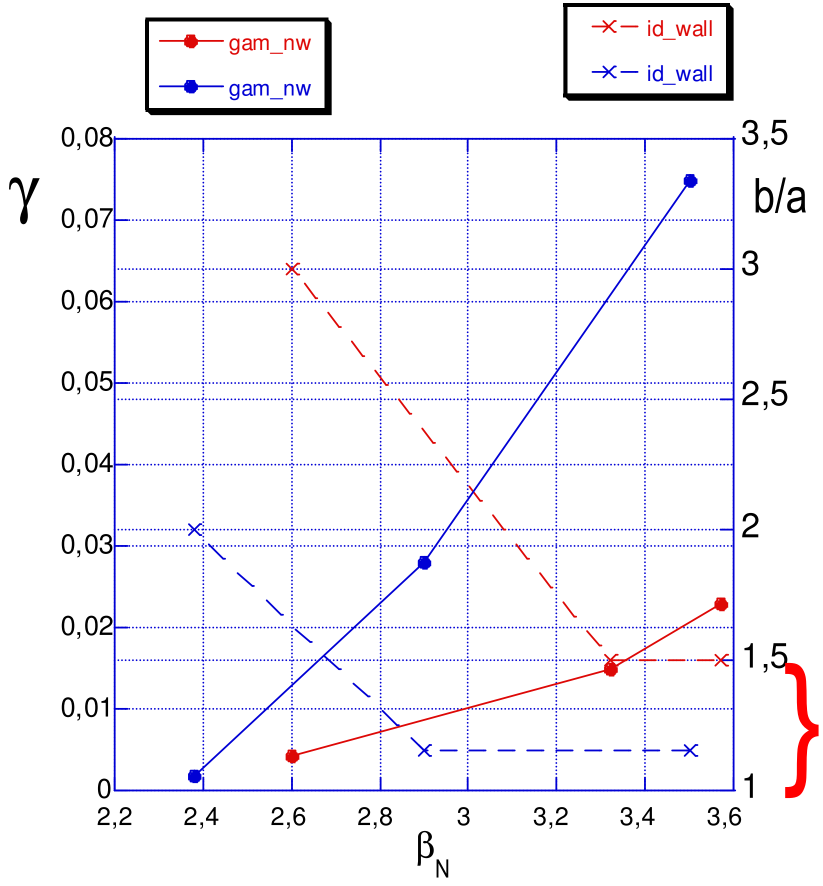
(this is true also for vertical n=0 stability !)

Growth vs. β_N and ideal wall stability

R/a=2, elong.=1.7 tri=0.2

just as an example..





range of
wall position

Plasma rotation misteries.. and its role in stabilization

CONSORZIO RFX
Ricerca Formazione Innovazione

from J. E. Rice, Experimental observations of driven and intrinsic rotation in tokamak plasmas PPCF 58 (2016):

- a substantial fraction of the rotation observed following **NBI** is not due to direct drive from the beams

This calls into question the traditional method of determining momentum transport coefficients from observed rotation profiles assuming momentum input (calculated) from the beams

- Regarding **LH** : once the q profile is modified, the observed rotation is in the opposite direction to the momentum input from the LH waves!

- **ICRF** waves in the minority heating scheme, observations show rotation in both directions, with complicated profile shapes and agreement with theoretical models isn't even qualitative.

These results indicate that momentum input from RF waves is not well understood

- For momentum sinks due to **locked modes**, magnetic braking and NTV the agreement between experiment and theory is **often very good**

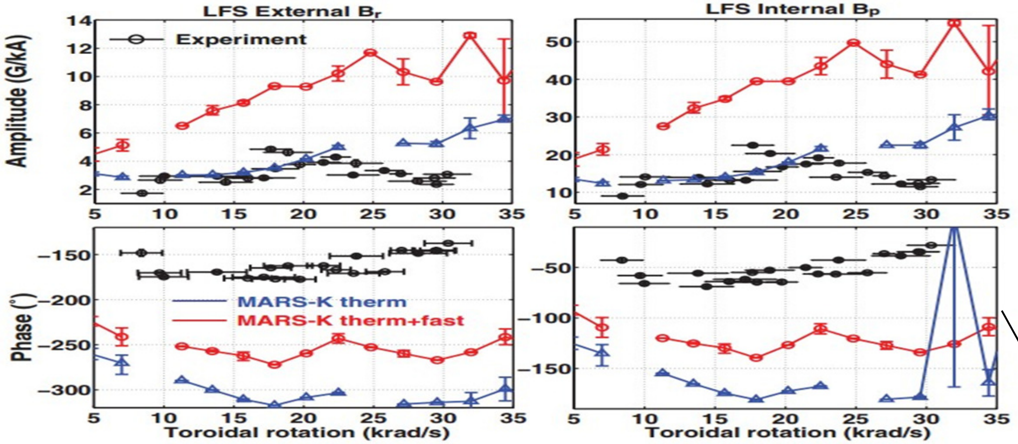
- the comparison between observations and the predictions of neo-classical theory show a **huge range of agreement/disagreement** from excellent quantitative comparisons to complete disparity! ..***not understood residual stresses ?!***

plasma rotation & mode stabilization:

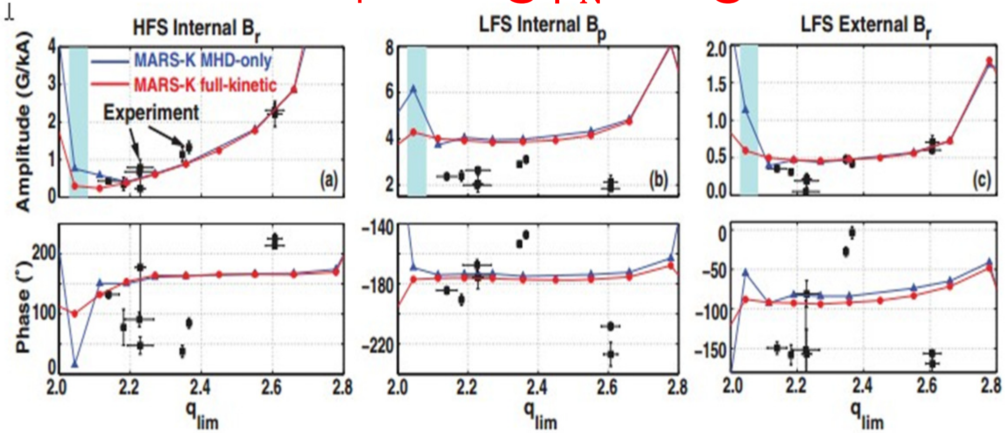
from F. Turco et al, NF 55 (2015)

Plasma response: the model shows a significant discrepancy at the highest β_N points ..***still missed physics !***

MARS-K vs Exp. DIIID @ β_N=2.4 @ I-coil 20Hz



MARS-K vs Exp. DIIID @ β_N=1.9 @ I-coil 20Hz



- **plasma rotation stabilize RWM**
  Bondeson&Ward PRL (1994)

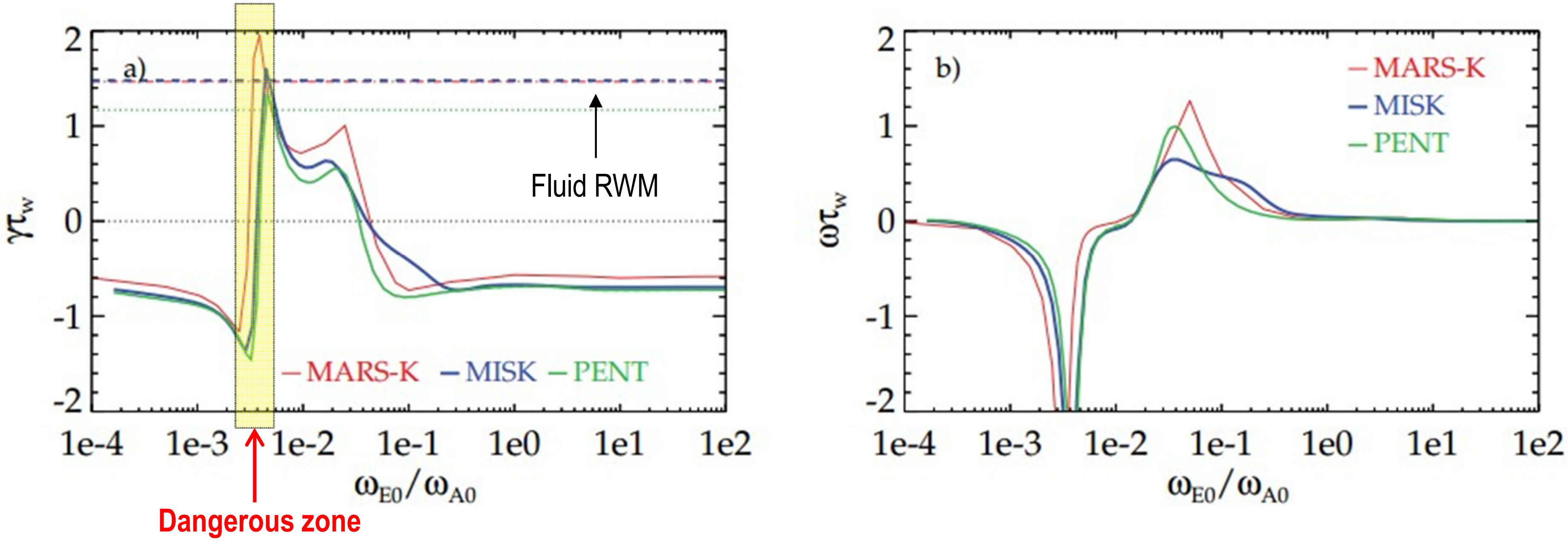
However later:

- the threshold is at relatively low plasma rotation
  $\omega_{crit}\,\tau_A$= 0.3% at the q=2 surface
  (T.*Strait et al, PoP 14(2007)*)

- kinetic drifts therm. & fast. ions are important but seem not to fully describe the physics in DIIID

Kinetic effects and predictions for ITER

from J. Berkery et al, " Benchmarking kinetic calculations of resistive wall mode stability" PoP 21 (2014)



Dangerous zone

One must recall that this is an incomplete calculation for ITER, however, as various simplifications have been made in the benchmarking process, including, most notably, ***the lack of collisions and energetic or alpha particles****. Nevertheless, the codes agree in the basic underlying calculation of kinetic effects and all support the present understanding that both high and low rotation kinetic resonances are stabilizing to the RWM, but* ***intermediate plasma rotation is potentially susceptible to instability***

Assuming as in Parra et al, PRL (2012) : $V_\phi = k\frac{T}{I}$ *and* $\omega_{Eo} \approx \frac{V_\phi}{a}$ *ITER@20Kev,10MA will likely be in the dangerous zone (or.. near t to it) !!*

RWMs critical issues

- Rotational stabilization of RWM
  - **Mode damping physics**: ω_*, ω_d resonances; reactive closure model (Weiland), neoclassical viscosity (Shaing)
  - **Effect of rotation profile**: rotation shear, damping distribution, role of edge rotation, global parameter for rotation threshold
  - **Toroidal momentum damping** due to RWM
  - **Nonlinear coupling** between RWM, RFA, and rotation: (rotation damps mode, mode damps rotation via RFA)
  - **Plasma rotation enough in ITER for RWM stabilization?**
- Feedback stabilization of RWM
  - **Systematic toroidal study of RWM dynamics** (PRM) vs. β_N and ω_{rot}
  - **Control issues for ITER**: choice of feedback coils, non-ideal effects (voltage saturation, noise, ac-losses of super-conducting coils)
  - **3D wall effect** on RWM control in ITER
  - RWM control for $n \geq 2$

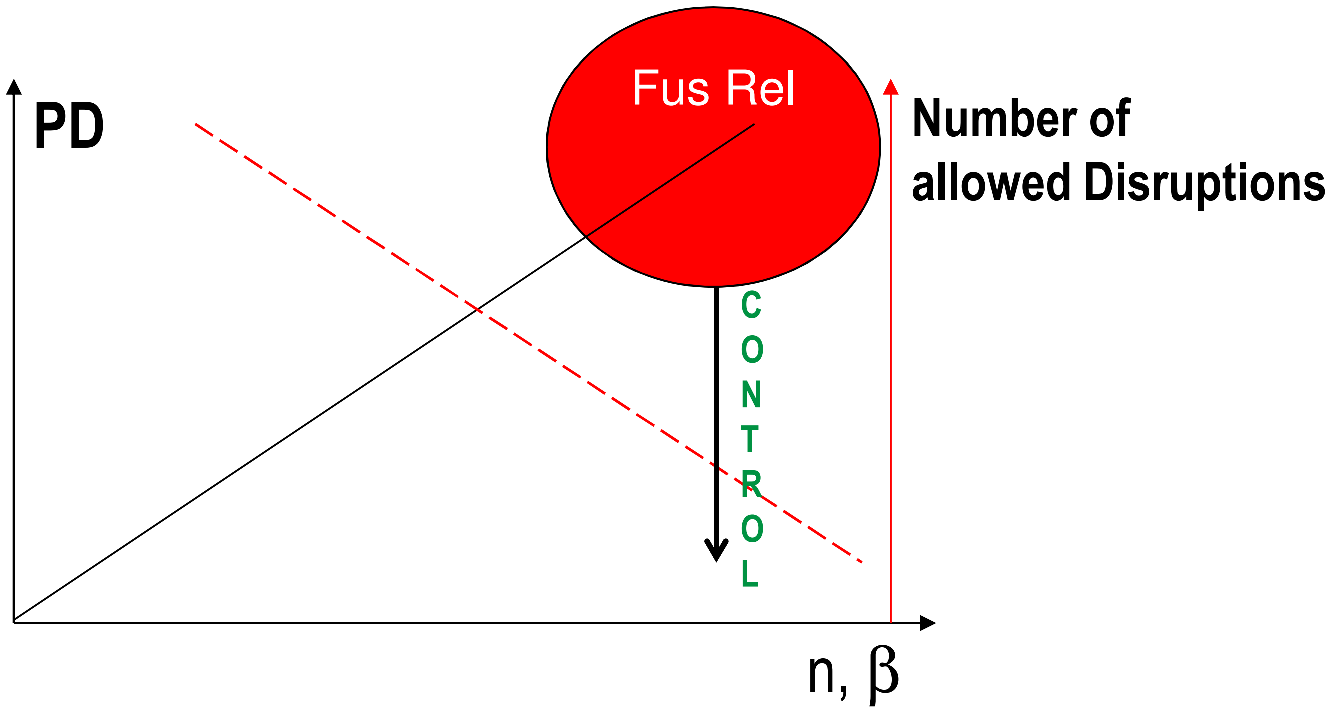
YQ Liu, Peking University, Feb 16-20, 2009

Probability of disruption at high β and in FR plasmas

The request for high performances i.e. high n and high β

is equivalent to operate near to the DISRUPTION LIMITS

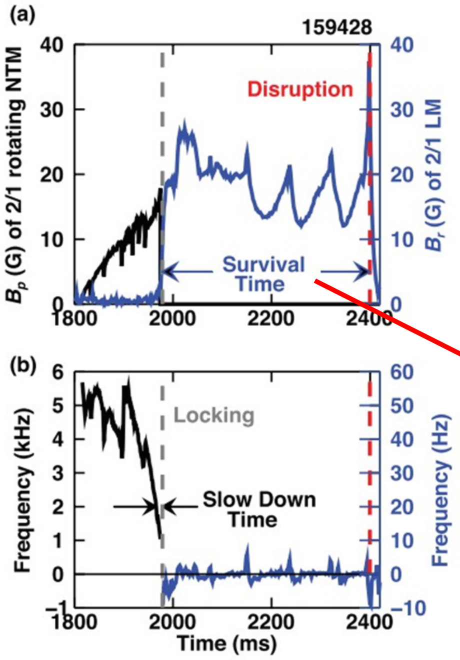
and increase the PROBABILITY of DISRUPTIONS (PD)

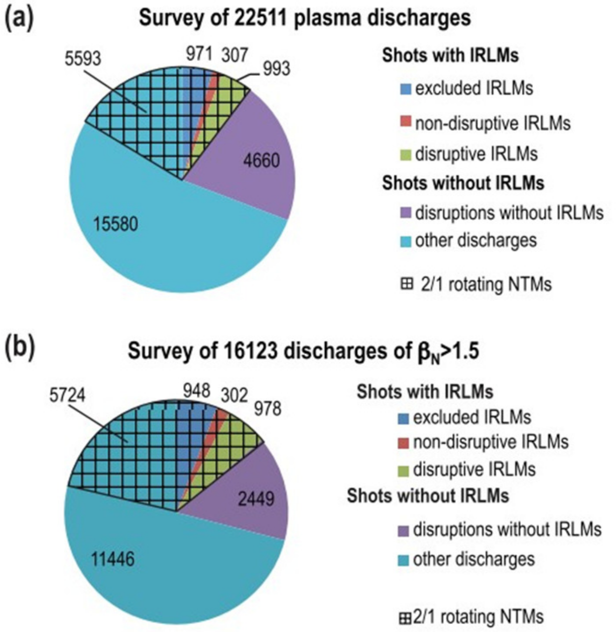


Mode Locking and disruptions

From Sweeney NF (2017)

Typical sequence of mode locking





Shots with IRLM ended 76% of the time in a disruption

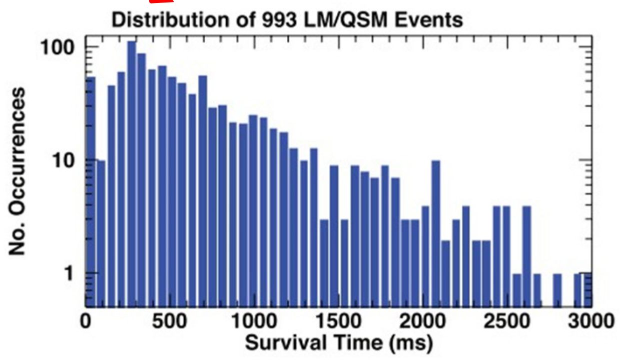


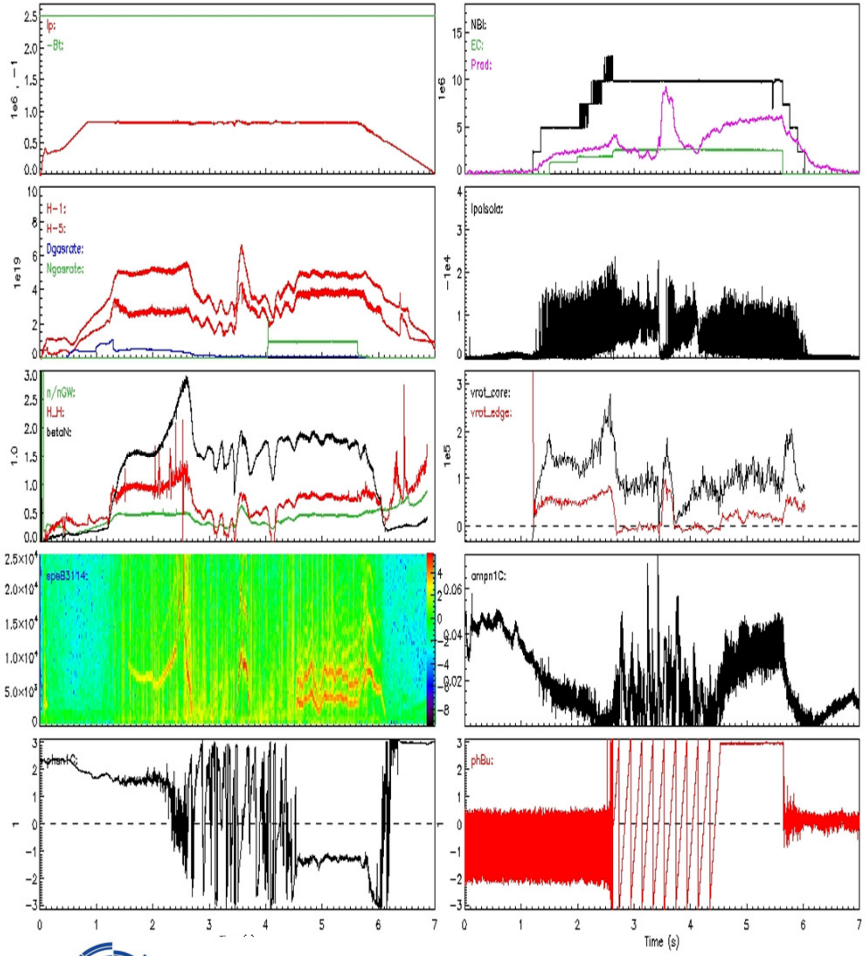
Figure 6. A histogram of the survival time, defined as the duration of a locked mode that ended in a disruption. Less than 2% of events survive for more 3000 ms.

At high β 28% of the disruptions are caused by a detected IRLM (18% at low β)

Recent experiments in AUG-U of disruption control with **RMP**

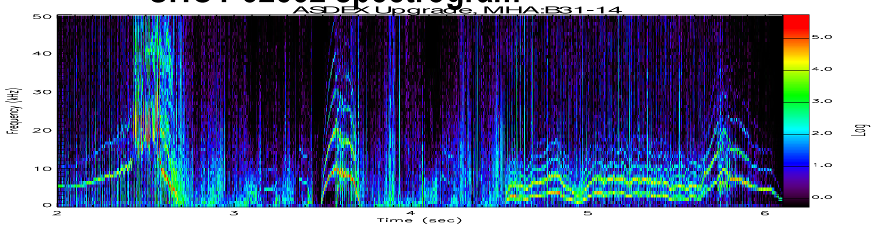
From R. Paccagnella et al, EPS P1.027 (Leuven, 2016)

AUG #33197 10 MW NBI with mode entrainment

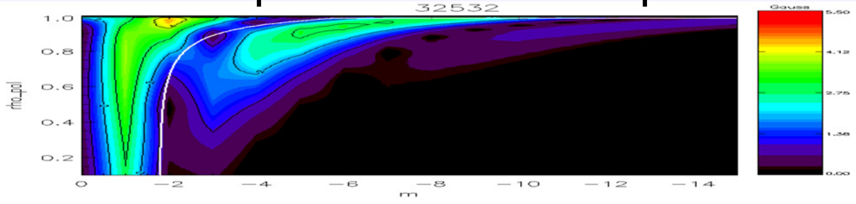


2/1 (NTM?) tearing excited by a kink aligned RMP

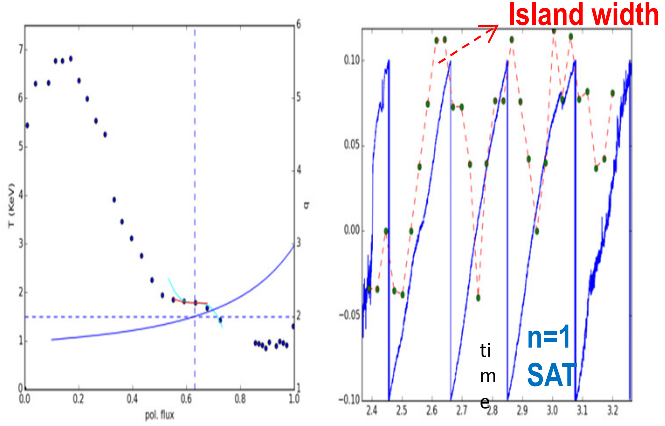
SHOT 32532 spectrogram



SHOT 32532 amplitude of vacuum field vs poloidal mode m



SHOT 32532 : ECE confirms resonant 2/1 island structure In phase with B coils



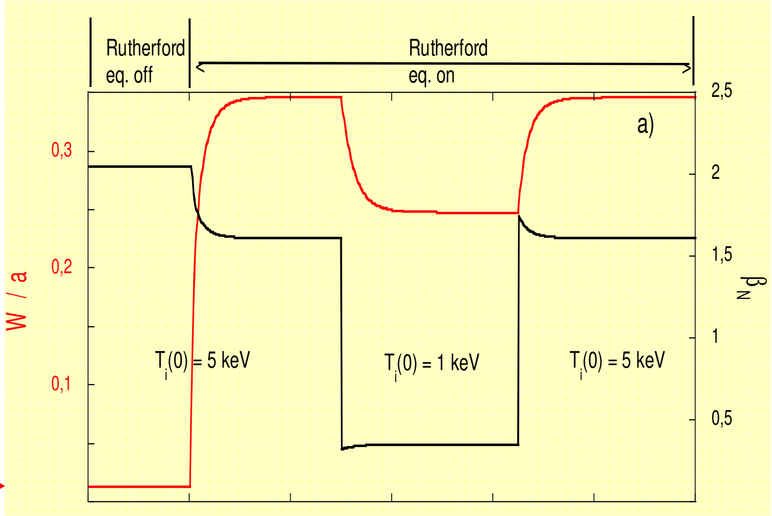
What about the modelling of this interesting case ?

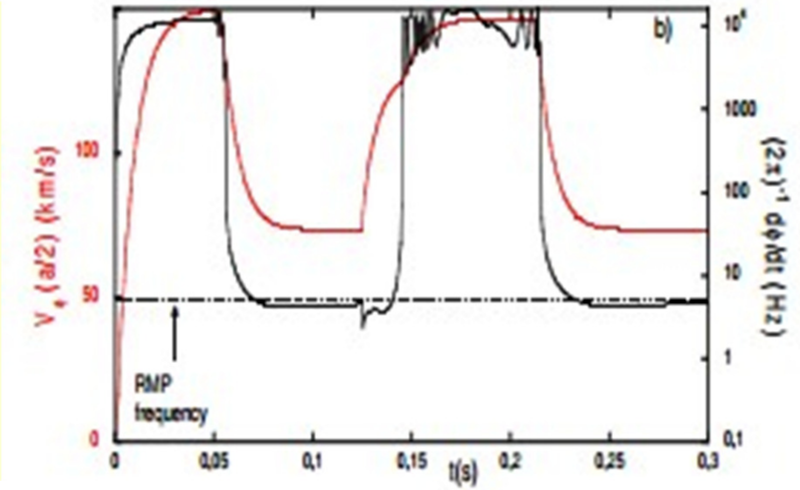
The dynamics of the m=2, n=1 tearing mode is simulated by **the cylindrical,** spectral **RFXlocking code** [1]

- Equations of motion
- Newcomb Equation
- NTV from island determined as in [2]
- Rutherford Equation
- No-slip condition
- Wall resistive diffusion

[1] P. Zanca et al Nucl. Fusion **55** (2015) 043020
[2] A. J. Cole, C. C. Hegna, J. D. Callen, PoP **15** (2008) 056102

Modelling in toroidal geometry would be necessary (..but very difficult) Reduced Models are important





RFXlocking Code (1)

TM dynamics is simulated by the cylindrical, spectral RFXlocking code, solving:

- **Single-fluid motion equations** with perpedicular viscosity μ and em. torque δT_{EM} localized at the resonant surface $r_{m,n}$

$$\rho \frac{\partial \Omega_\phi}{\partial t} = \frac{1}{r}\frac{\partial}{\partial r}\left(\mu r \frac{\partial}{\partial r}\Omega_\phi\right) + \frac{\delta T^{m,n}_{EM,\phi}}{4\pi^2 r R_0^3}\delta\left(r - r_{m,n}\right)$$

$$\rho \frac{\partial \Omega_\theta}{\partial t} = \frac{1}{r^3}\frac{\partial}{\partial r}\left(\mu r^3 \frac{\partial}{\partial r}\Omega_\theta\right) - \frac{\rho}{\tau_D}\Omega_\theta + \frac{\delta T^{m,n}_{EM,\theta}}{4\pi^2 r^3 R_0}\delta\left(r - r_{m,n}\right)$$

+ NTV like torque:

$$\frac{\partial \Omega}{\partial t} = -\mu_\parallel \left(\frac{\delta B_{\mathrm{eff}}}{B_0}\right)^2 (\Omega - \Omega_*)$$

- Em. Torque, due to interaction with the passive structures, is modelled exploiting **Newcomb's equation** **…also for the NTV**
  (parabolic equilibrium current distribution)

RFXlocking Code (2)

- **Newcomb equation**

From Zanca P PPCF (2010)

$$\frac{\partial}{\partial r}\left[\frac{r}{H^{m,n}}\frac{\partial}{\partial r}\psi^{m,n}\right]-\left[\frac{1}{r}+\frac{rG^{m,n}}{H^{m,n}F^{m,n}}\frac{\mathrm{d}\sigma}{\mathrm{d}r}+\frac{2mn\varepsilon\sigma}{(H^{m,n})^2}-\frac{r\sigma^2}{H^{m,n}}\right]\psi^{m,n}=0,$$

$$F^{m,n}(r)=mB_{\theta 0}-n\varepsilon B_{\phi 0},\quad G^{m,n}(r)=mB_{\phi 0}+n\varepsilon B_{\theta 0},$$

$$H^{m,n}(r)=m^2+n^2\varepsilon^2,\qquad \varepsilon=r/R_0.$$

with $\psi^{m,n}(r,t)\equiv -\mathrm{i}rb_r^{m,n}(r,t)$

- **Rutherford equation** for the island width

$$\frac{\tau_R}{r_{m,n}}\frac{dW}{dt}=1.22\,\Delta'(W)$$

- **Diffusion equations** for radial field penetration across the passive structures

$$\mu_0\,\sigma\frac{\partial b_r^{m,n}}{\partial t}=\frac{\partial^2 b_r^{m,n}}{\partial r^2}$$

- Island phase determined by the **no-slip** condition

(for the present simulations we neglect diamagnetic term)

$$\frac{d\varphi^{m,n}}{dt}=n\;\Omega_\phi\left(r_{m,n},t\right)-m\;\Omega_\theta\left(r_{m,n},t\right)$$

RFXlocking estimates for ITER

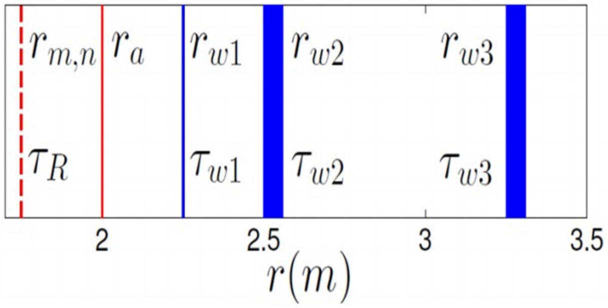
TM locking during CQ in ITER

Cylindrical model and walls:

Blanket is treated as a EM-thin wall* $\tau_{w1} = 2ms$

VV is treated as 2 EM-thick walls : $\tau_{w2} = 94ms$

$\tau_{w3} = 94ms$



* Villone F. *et al 2010 Nucl. Fusion* ***50 125011***

m/n=2/1 TM locking during CQ

τ_{CQ} - Current quench time

τ_R - Resistive time

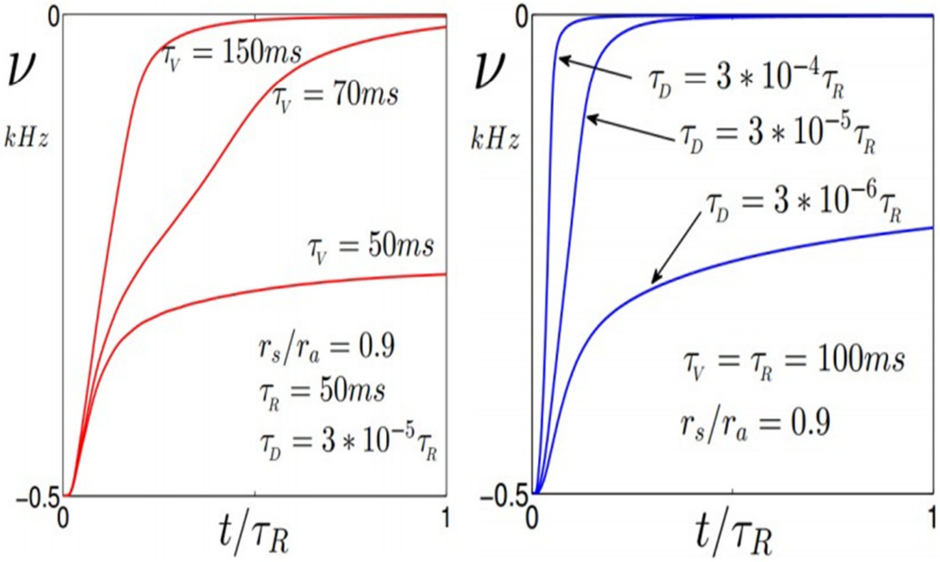
τ_V - Viscous time

τ_D - Poloidal damping time

$\tau_R \approx \tau_{CQ}$

$\tau_V \geq \tau_{CQ}$

$\tau_D \approx \tau_{ii}$



The role of radiation in disruption mitigation

- Disruptions can be **triggered by** a sudden increase of the **radiation losses**
- However **radiation can** also be used to **mitigate disruption effects:** reducing divertor heat loads, asymmetric stresses and runaway electrons

From Lehnen et al NF (2013) and (2015)

JET

Radiation asymmetry

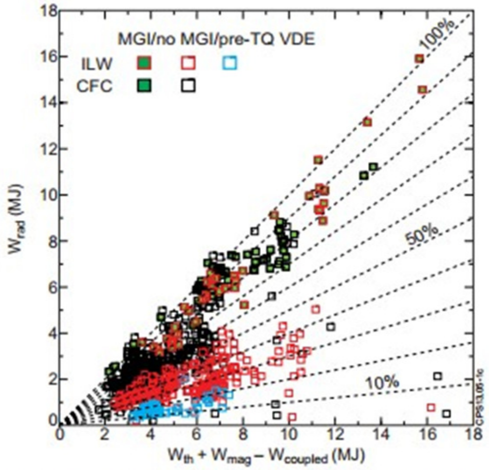


Figure 4. Radiated energy as a function of the energy available in the plasma.

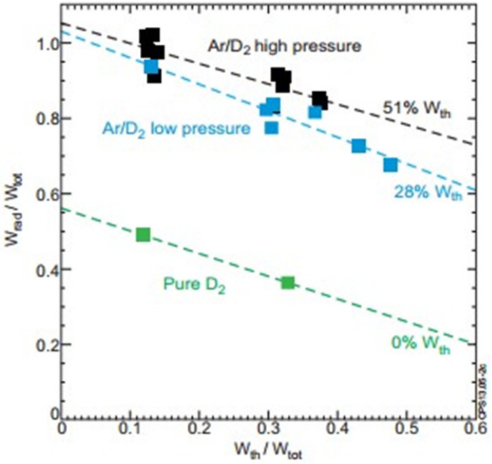
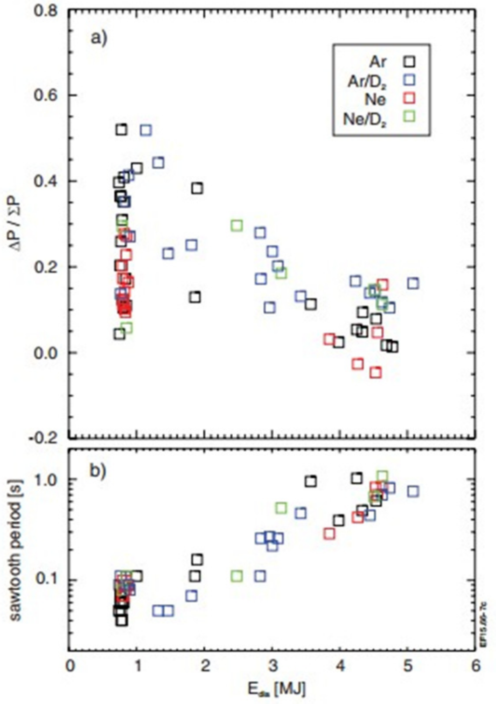


Figure 16. Radiated energy during MGI as a function of the fraction of thermal energy stored in the plasma before injection.

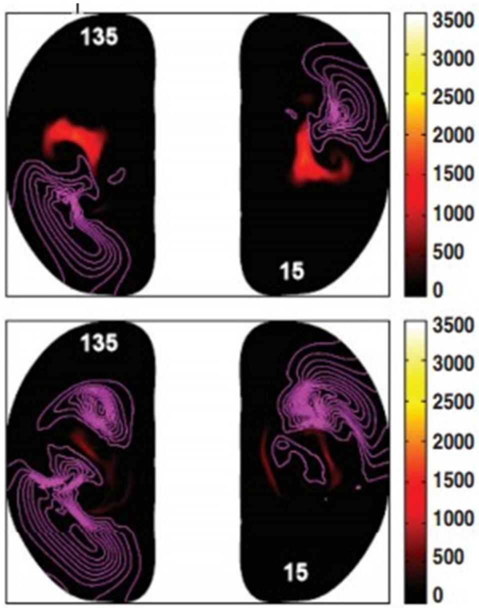


Nonlinear simulations of disruption mitigation (1)

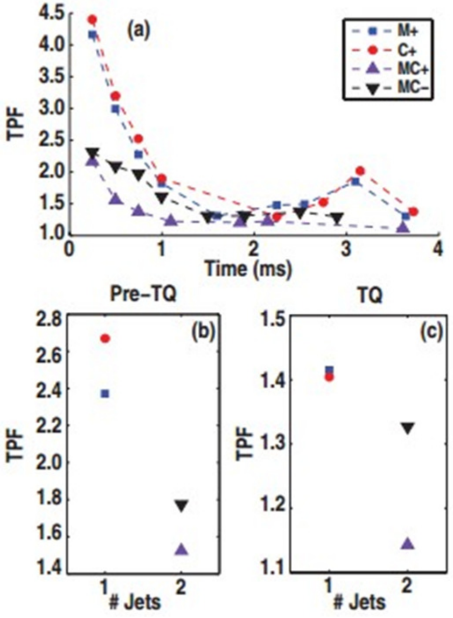
from Izzo V et al, NF 55 (2015)

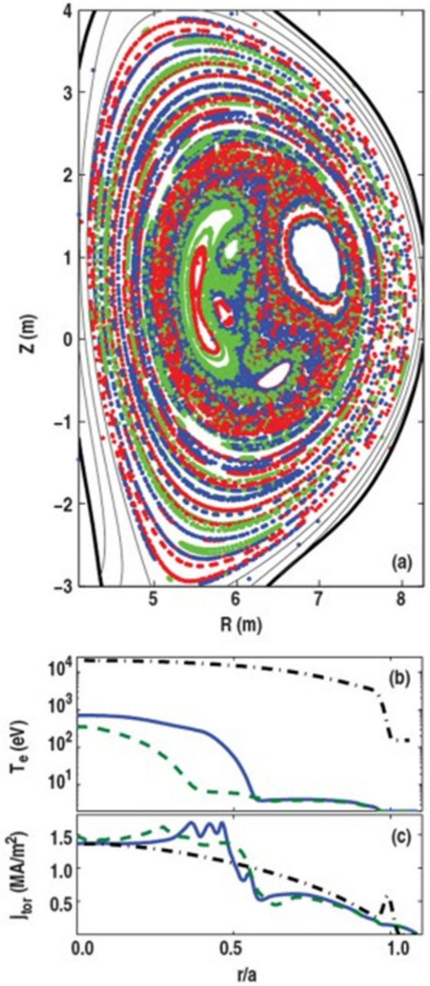
Runaways in ITER
10^{-2}(s) est. conf. time

MHD produces radiation asymmetries



NIMROD SIMULATIONS





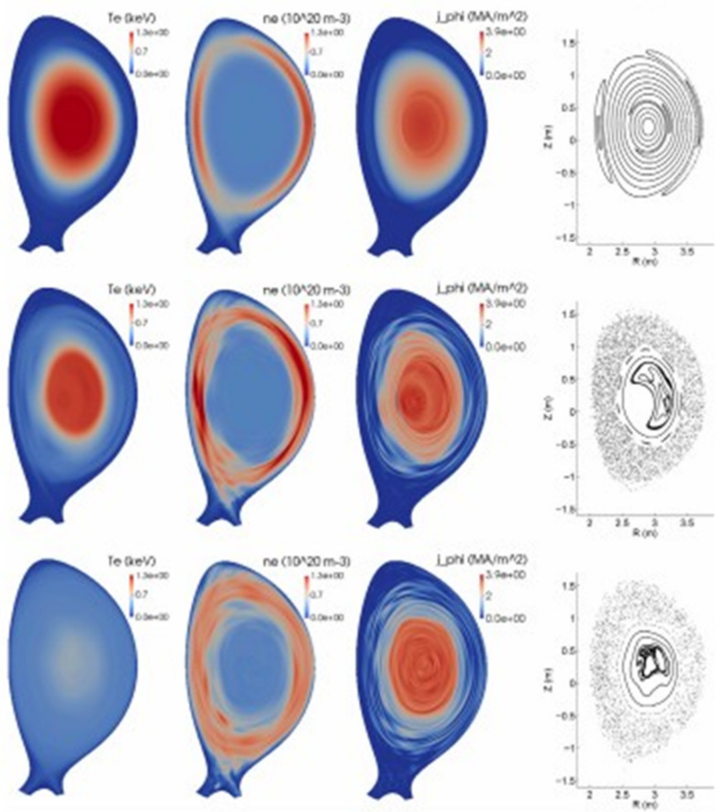
from Izzo V et al, NF 51 (2011)

Weak points:

- Simplified physics,radiation models, plasma-wall int.
- transport
- Collisionality
- Num. resolution

Nonlinear simulations of disruption mitigation (2)

from Nardon et. al. PPCF (2017)

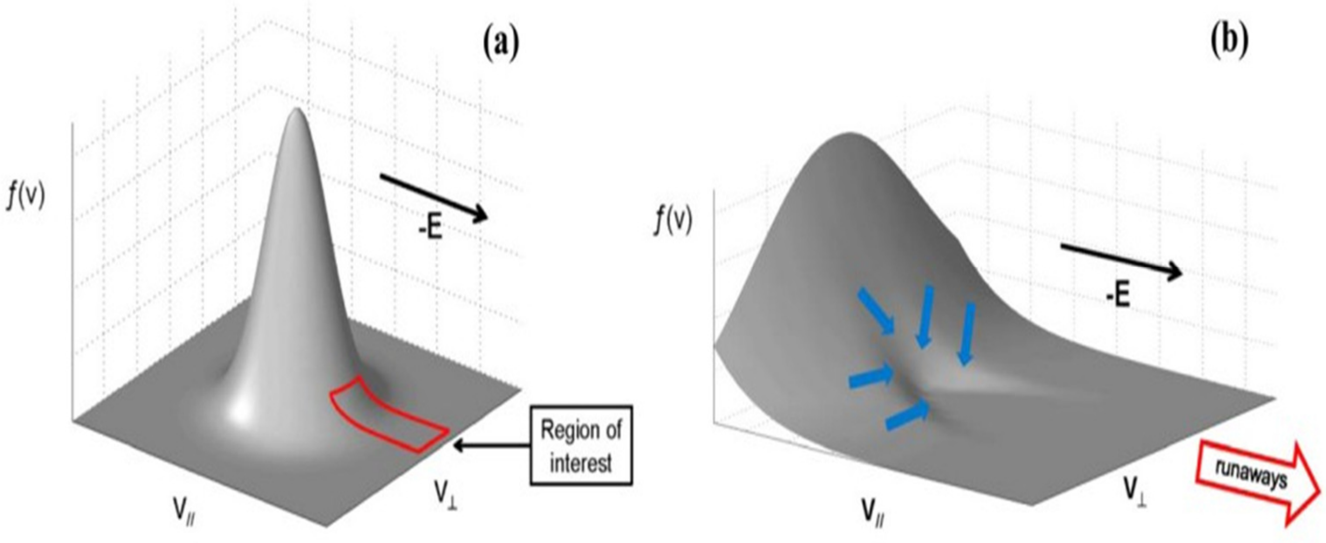


Recent JOREK simulations
(at high S number > 10^7)

- 2/1 induced by resistivity drop du to MGI
- 3/2 destabilized by current flattening
- nonlinear coupled modes triggered
- plasma stochastisation and TQ

About Runways electrons (1)

From Granetz PoP (2014)

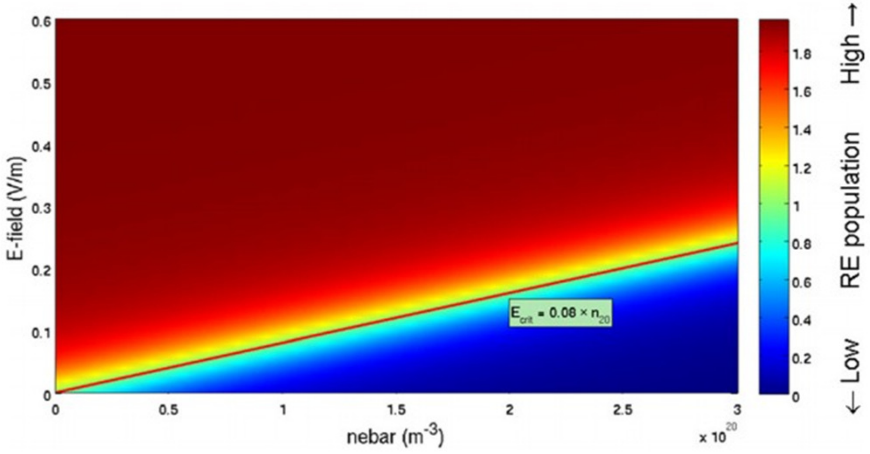


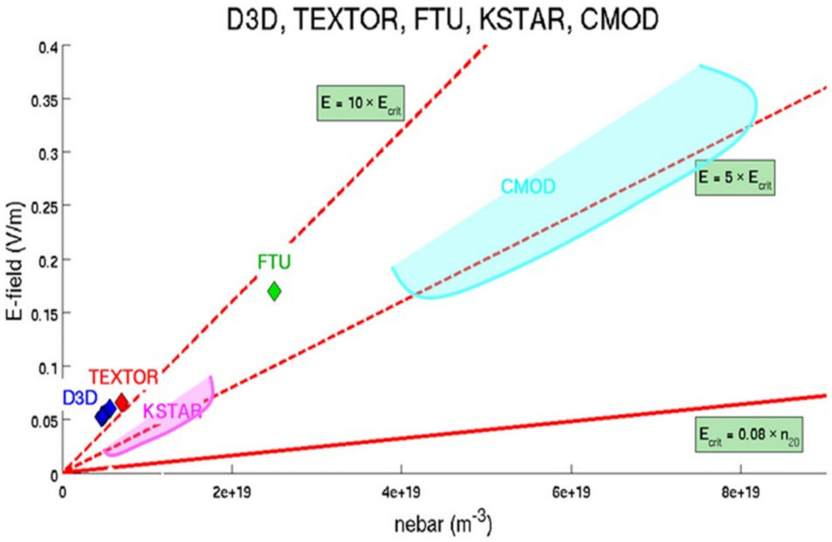
$$E_{\mathrm{crit}} = \frac{n\,q^3 \ln\Lambda}{4\pi\varepsilon_0^2 m c^2}$$

The critical electric field depends on plasma density (more weakly on temperature)

About Runways electrons (2)

From Granetz PoP (2014)





- The **critical electric field is quite small according to theory**

- the experimental data show a **much higher electric field threshold :**
  interpreted as an **extra loss mechanism** beside the collisional drag (e.g synchrotron rad.)

- **Avalanche mechanism** likely dominant in ITER at difference with actual experiments

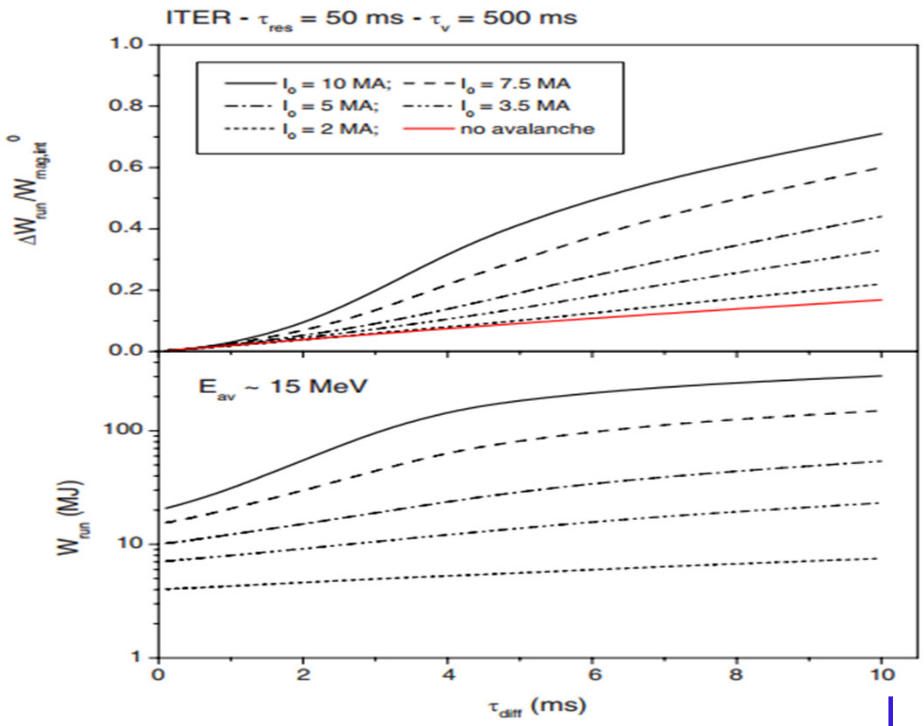
$$\frac{dn_{\mathrm{RE}}}{dt} = \underbrace{\left(\frac{dn_{\mathrm{RE}}}{dt}\right)^{\text{primary}}}_{\textit{linear}} + \underbrace{\gamma_{\text{sec}} n_{\mathrm{RE}}}_{\textit{exponential}}$$

and

$$\gamma_{\text{sec}} \propto \left(\frac{E}{E_{\text{crit}}} - 1\right)$$

About Runways electrons: (3)

From Martin-Solis et al NF (2014)



↓ **Nimrod estimate**

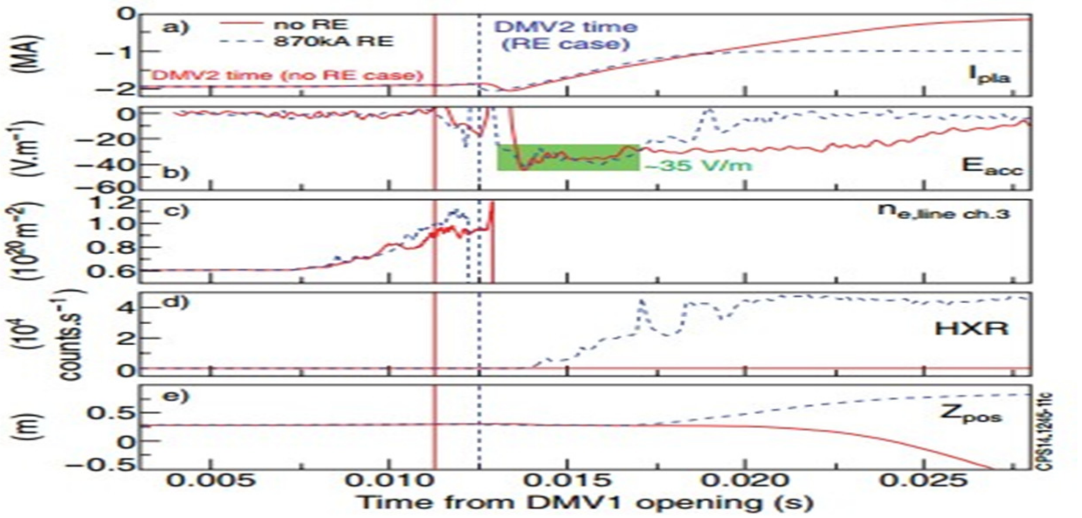
- **Predictions for ITER** are quite **uncertain**
- **the ratio** between the plasma **resistive diffusion time** (after TQ) and the **RE loss time** is critical
- **Avalanche** (for long duration of CQ) **could be an issue**
- **large fraction of plasma magnetic energy** could be converted to **RE energy**

RE represent a serious issue for ITER therefore mitigations and/or control systems are mandatory

About Runways electrons : beam penetration (4)

From Reux et al NF (2015)

In recent JET experiments with ILW:



- RE are suppressed by early (before TQ) gas injection
- are instead produced for a later gas injection

- Runaway mitigation **after the beam has been accelerated** has been proven **unsuccessful at JET**, with injections of 663 Pa.m3 to 4340 Pa.m3 of argon, krypton or xenon
- These results confirm globally **that runaway physics are similar with a metallic wall and with carbon wall**, and that runaway electron suppression should be attempted **before the beam is fully developed**.

CONCLUSIONS (1)

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There are several MHD related physical OPEN issues related to disruptions:

- the nature and detailed mechanisms of the **TQ**
- the duration of the **CQ** (residual temperature after TQ & RE)
- The **halo** structure (**2D**) in symmetric VDEs
- the **halo** structure (**3D**) in non symmetric VDEs
- **halos** vs. **hiros** (the role of surface currents)
- the role of the **passive structures** (and eddy currents)
- the nature and origin of **plasma rotation** and the residual **slow mode rotation**
- Interaction between **plasma** and **external MPs**
- **Thermal loads** and **RE electrons**

… for all this issues **EXISTING MODELS ARE GENERALLY LACKING**

CONCLUSIONS (2)

- a lot of **interesting physics** is related / linked to **DISRUPTIONS**

- our understanding is still quite **incomplete** and our modelling capabilities need to be further **extended (physics) and increased in capability (resolution)**

- **Pathological cases** (**like disruptions**) can be **very helpful** also for the understanding of **healthy plasmas**: the physics of plasma rotation, mode locking, plasma relaxation and reconnection, transport in stochastic fields are only few examples

- for their effects on the structures and on the containing wall material plasmas **completely avoiding** them are **needed in a fusion plants**

- **disruptions** are really the **most serious showstopper** for **fusion**

Runaways and localised plasma wall interactions :
could represent also **very serious issues for fusion** even in presence of mitigation systems as **MGI** or fast and massive **pellets launchers: NO DISRUPTIONS → NO SIDE EFFECTS**