Physics of energetic particles in ITER

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Summer school, 2011

Outline

- Introduction
- Progress in ITER project
- Classical confinement of energetic particles
- TF ripple effect on EP confinement
- Energetic particle instabilities
- Experimental tools for energetic ions
- Runaway electrons
- Summary and conclusion

Introduction

- Future tokamak reactors shall use DT reactions

$$D + T = \alpha(3.5MeV) + n(14MeV)$$

- Energetic alpha-particles must be confined in the plasma to provide main plasma heating in the burning plasmas and sustain plasma temperature

$$\frac{3nT}{\tau_E} = E_\alpha n_D n_T < \sigma v > + P_{aux}$$

- Parameter Q is used to measure fusion performance of the device

$$Q = P_{fusion} / P_{aux}$$

- ITER is designed to achieve Q = 10

- Fusion alpha particles is source of He ash in the plasma. An excessive accumulation of He ash could choke fusion reactions

Other energetic particles

- NB injection or ICRH minority heating are also based on generation of energetic ions
  - NBs shall produce 1 MeV Deuterium ions in ITER
  - ICRH minority heating will generate He3 ions with energy of few MeV

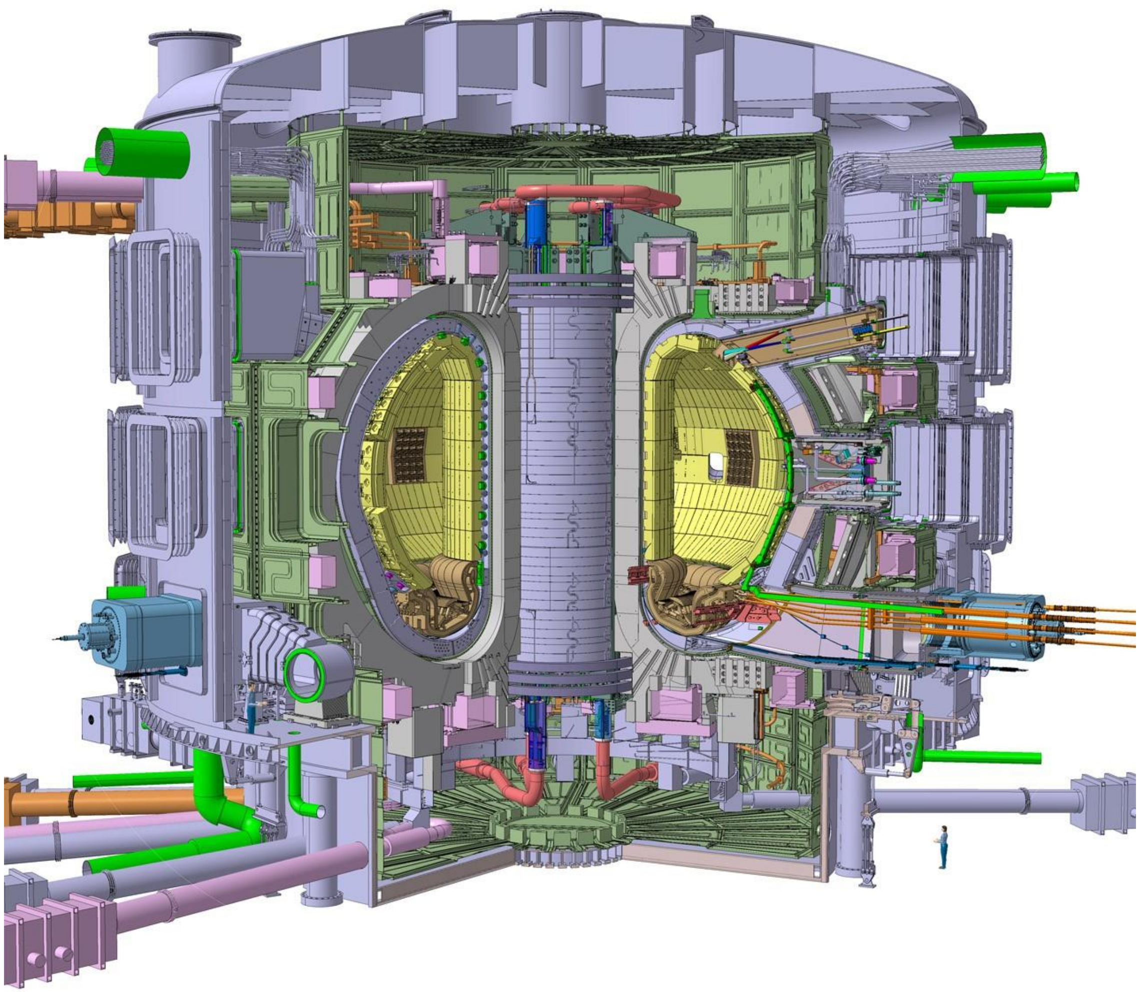
- I shall use fusion a-particles and ITER plasma parameters for the estimates in the following presentation

- Energetic electrons (runaway electrons) with energy 10-20 MeV can be produced at certain conditions in tokamaks. They are dangerous for the reactor plasma facing components and must be avoided

Introduction

- Good confinement of energetic ions and effective ash removal are the keys to the success of the fusion program

Introduction



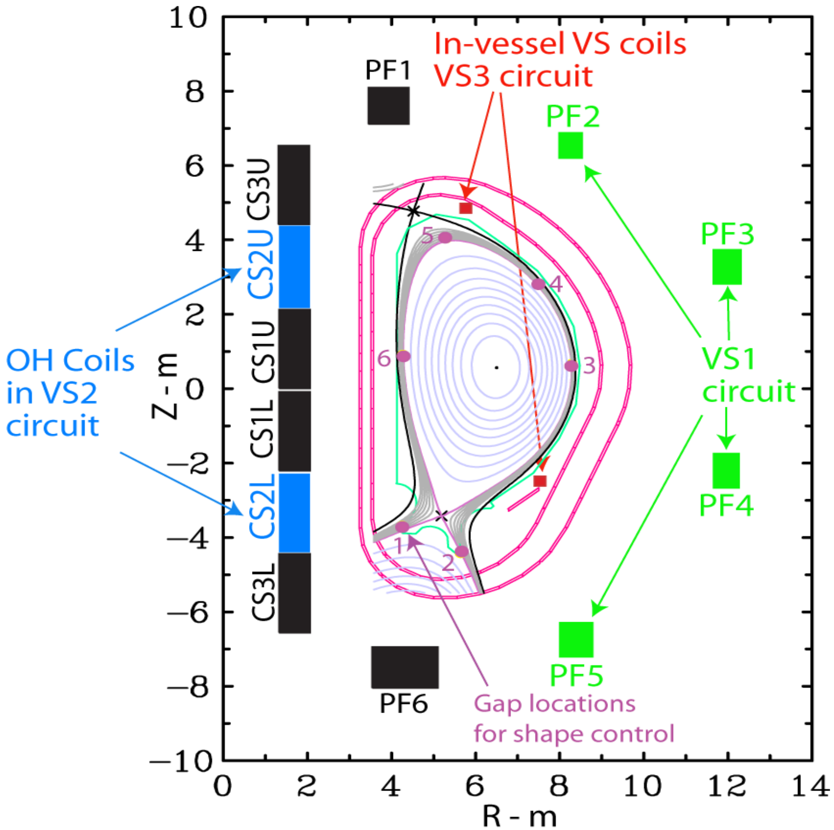
- ITER will be the first tokamak which has all essential elements of the future tokamak-reactors

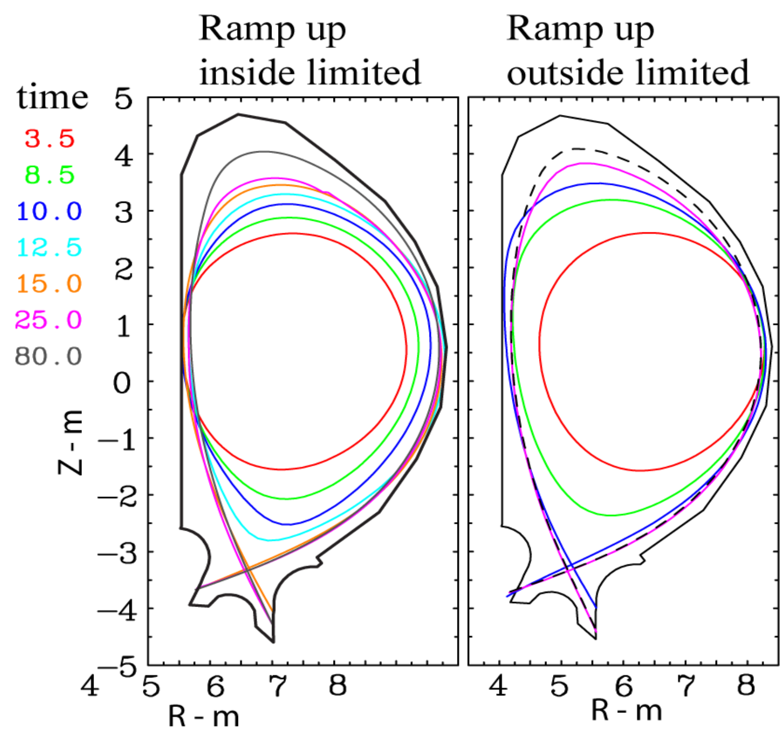
Introduction

Reference parameters of ITER plasma

Parameter	
Major radius (m)	6
Minor radius (m)	2
Toroidal magnetic field (T)	5.2
Plasma current (MA)	15
Plasma density (10^{20} m^{-3})	1
Plasma temperature (keV)	10
Energy confinement time (s)	3.5
Fusion power (MW)	500
$Q=P_{fus}/P_{aux}$	10

Plasma is controlled by PF system

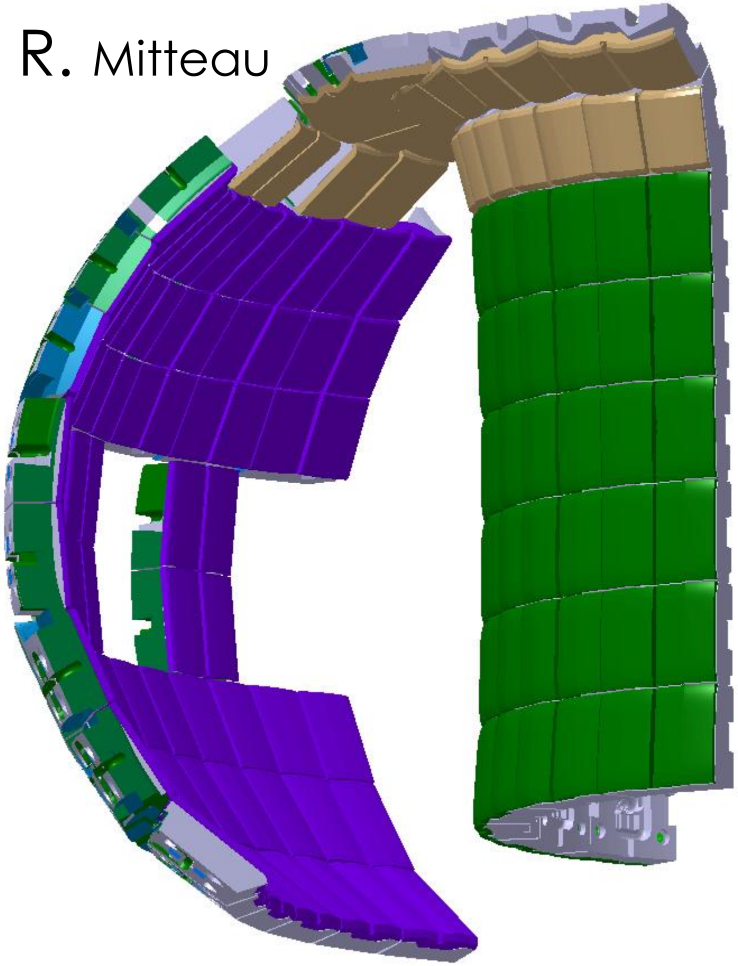




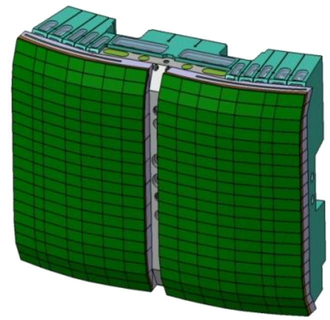
- Plasma has elongated cross section with a single null divertor configuration
- Poloidal field system must control gaps between plasma and wall with accuracy few cm

ITER First Wall Design

R. Mitteau

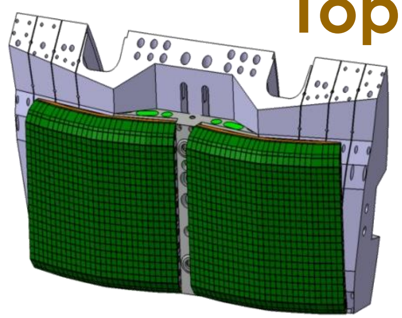


Inboard



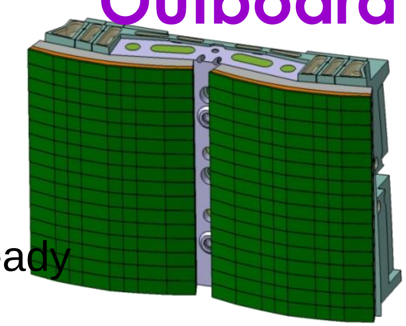
BM #1-6
Central column
HFS start-up
Toroidal & poloidal shaping

Top



BM #7-10
Secondary divertor region
Toroidal & poloidal shaping

Outboard



BM #11-18
Outboard
LFS start-up/ramp-down
Toroidal shaping

- All Be First Wall Panels shaped
- Shape & Power Handling (≤ 2 or 5 MWm^{-2})
- result of (on-going) optimization between steady loads and transients

ITER is nuclear facility

- ITER will be a nuclear facility and will require full nuclear licensing for operation
- 14 MeV neutrons will activate in-vessel components after the first shot. Remote handling is required for maintenance and repair
- Parts removed from the machine will be transported in special casks to the rad-waste building where they will be stored for cooling down and repair
- ITER will have very large Tritium facility for processing pumping exhaust
- At the same time ITER will be truly experimental device. It will be equipped with complex diagnostic system for experimental studies of burning plasmas

ITER will have many plasma diagnostics

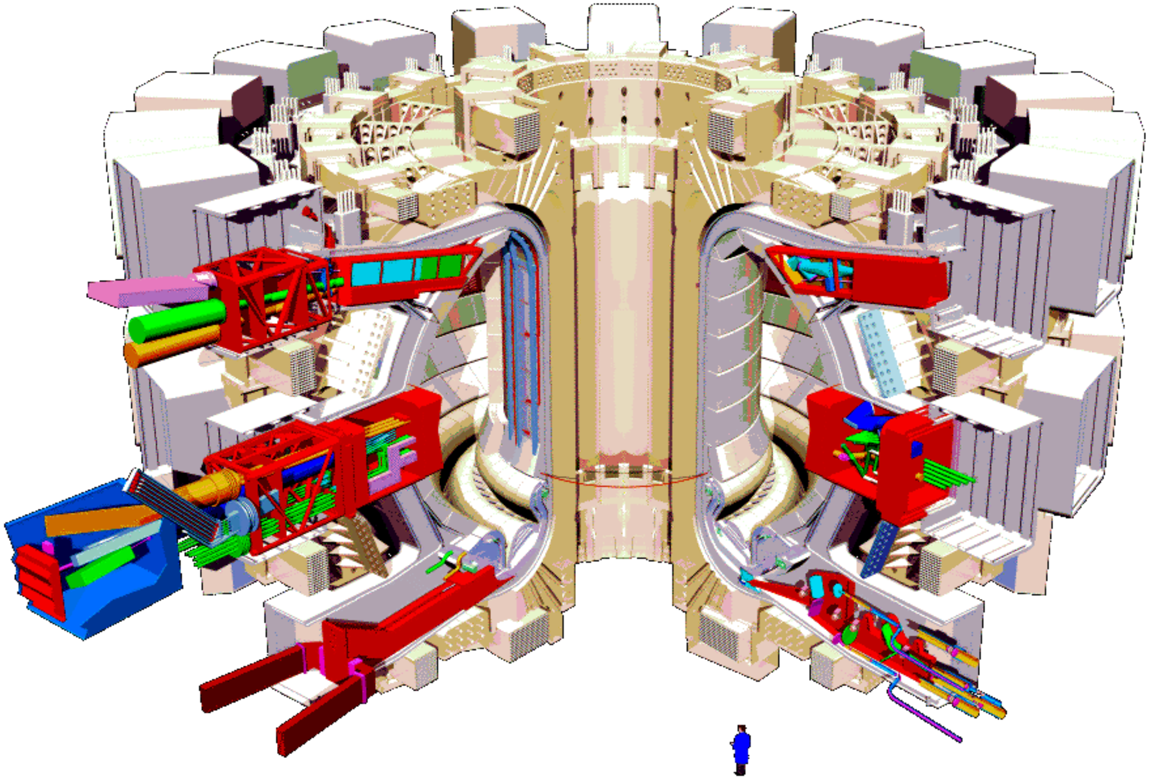
Facts - About 45 different diagnostic systems installed around ITER tokamak 1) for machine protection or basic control, 2) advanced performance control, and 3) evaluating the plasma performance and understanding important physical phenomena

Status – Systems are currently being integrated with the machine design. Several diagnostics have already successfully passed the Design Review stage and the first Procurement Arrangement are about to be signed.

Diagnostic Systems:

Neutrons
Magnetics
Passive Spectroscopy
Active Spectroscopy
Infrared Thermography
Particle monitoring
Tritium & Dust
Density/Temperature

All integrated directly in to machine or port plugs

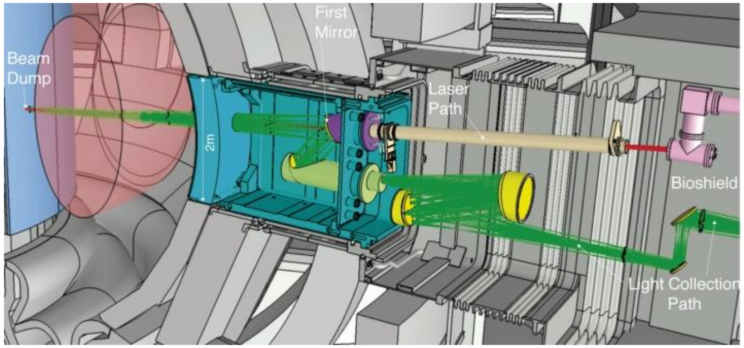


ITER systems with mirrors are challenging

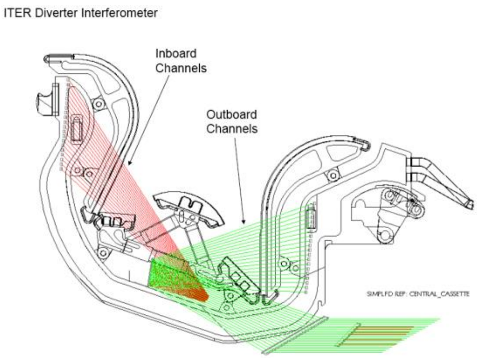
There are ~30 diagnostic systems on ITER involving first mirrors.

•They provide information on close to 100 plasma parameters

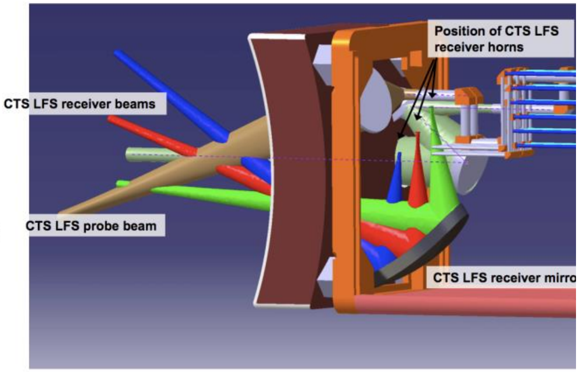
•They span the wavelength range from nanometer to millimeter and involve a wide range of solid angles and presumed fluxes.



Core LIDAR (300-1000nm)



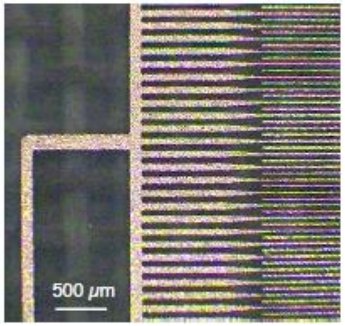
Divertor interferometer (10μ)



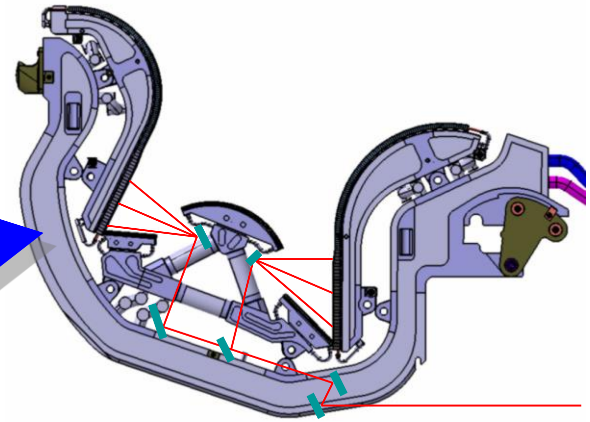
Low-field side CTS (1 mm)

Dust/Erosion/Tritium Diagnostics

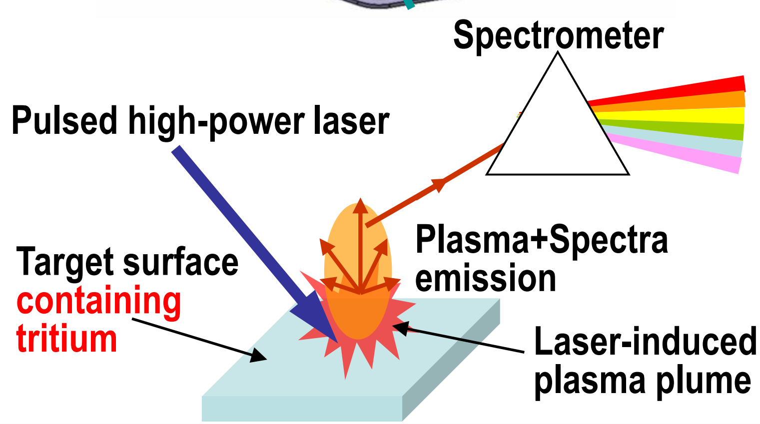
- ITER dust/tritium inventory strategy comprises measurement of
    - Local dust concentration
    - Divertor target erosion
    - Tritium retention
    - Hot dust



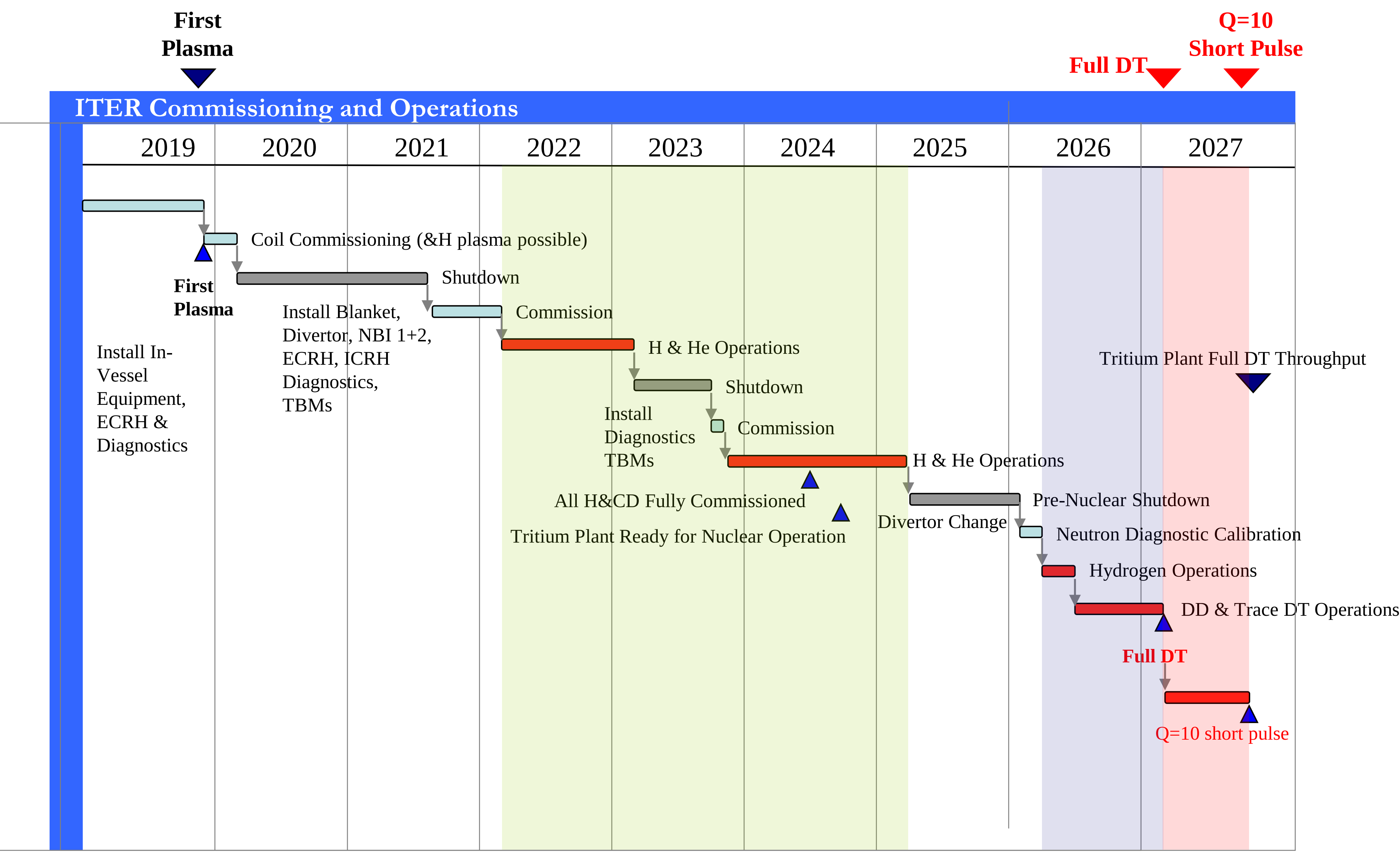
- For local dust concentration, 2 concepts:
    - Capacitive diaphragm microbalance
    - Electrostatic grid
- For divertor erosion, laser-based concep implemented at the divertor:
    - FM LADAR, Speckle interferometry, Digital holography



- For tritium retention, laser-induced spectroscopy is considered.
    - LIDS, LIAS, LIBS



ITER Experimental Schedule to DT



ITER Research Plan - Major Elements

- **H/ He Campaign I: March 2022 - January 2023**
  - System commissioning with plasma
  - H&CD short pulse commissioning to ~70MW input power
  - 15MA/ 5.3T technical demonstration
- **H/ He Campaign II: November 2023 - May 2025**
  - H&CD commissioning to long pulse
  - Disruption loads completed/ disruption mitigation implemented
  - ELM control commissioned in helium H-modes
- **D/ DT Campaign: May 2026 - August 2027**
  - Commissioning of Tritium Plant with tritium
  - Commissioning of tungsten divertor in H/ He plasmas
  - Development of H-mode scenarios in deuterium
  - Trace tritium experiments begin in January 2027
  - Full DT experiments begin in March 2027
  - Attempt at Q=10 short pulse in August 2027

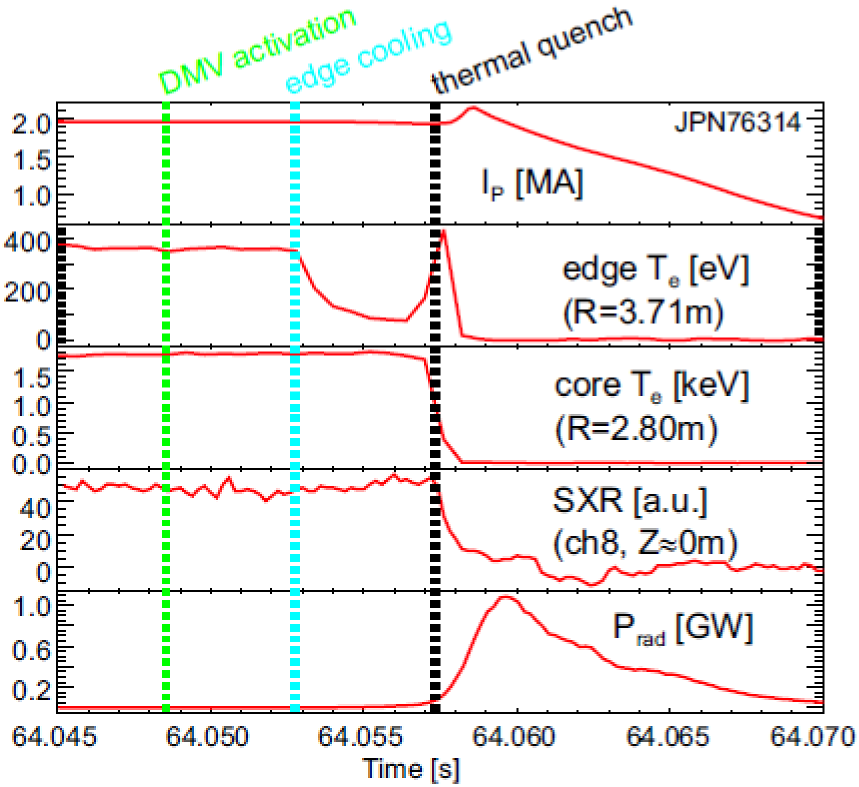
Open physics R&D issues

- Open R&D issues with major influence on designs for Baseline
  - Disruptions Loads and Disruption Mitigation (heat, forces and runaways)
  - ELM Heat Fluxes and ELM Control Schemes

- ITER Scenarios and Open R&D Issues
  - H-mode access (incl. I_p ramp), control of H-mode access and H = 1 sustainment
  - Helium H-modes
  - Fuelling of H-mode Plasmas
  - Control of plasma during transients
  - NTM control, RWM control, etc.

Plasma can abruptly disrupt in a tokamak

- This disruption is triggered by Ne injection and following edge cooling



M.Lehnen, EPS2009

Plasma disruption and disruption mitigation

- Critical for ITER design and performance
- Plasma disruption must be mitigated in ITER
  - High energy loads on divertor targets and FW during TQ of plasma disruption ~ R^2
  - RE can be readily produced in large machine by avalanche
  - Forces on conducting structures ~ plasma current

- Machine is designed mechanically to withstand 3000 major disruptions and 400 “hot” VDE

- However, energy loads from a few hundred disruptions can seriously damage divertor targets and FW panels

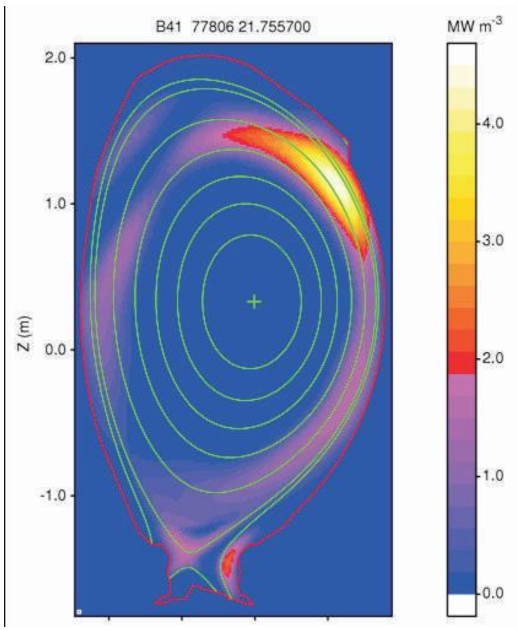
Approach to DMS in ITER

- Preventive Massive Gas Injection (MGI) for suppression and redistribution of the of thermal loads of TQ.
- A few kPa*m^3 (~ 2-5 10^{23} particles) of Ne would be sufficient for re-radiation and re-disrtibution of the energy flux over the wall
- This amount of gas can be digested by pumping system without much delay between pulses
- Suppression of RE collisionally (Rosenbluths density 10^{25} particles) will result in significant slowing down of ITER operation

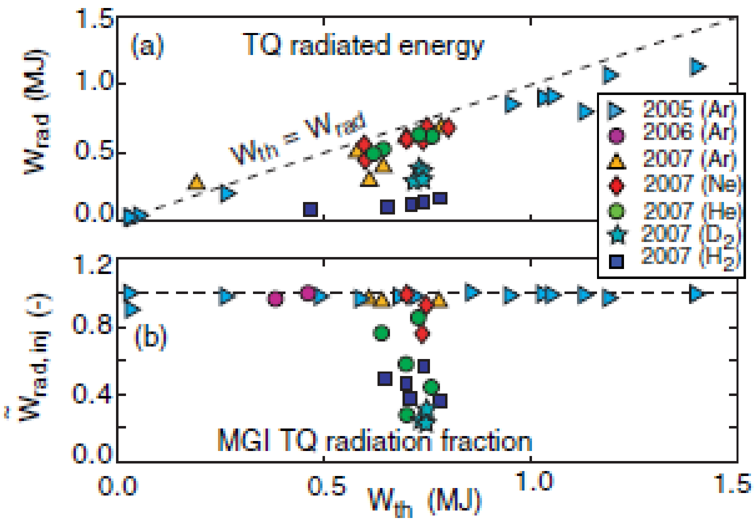
MGI can to re-radiate most of plasma thermal energy

- Challenge for ITER DMS: re-radiate ~300 MJ of plasma thermal energy in about 3 ms and distribute it uniformly over FW

- Experimental results from present tokamaks are very encouraging



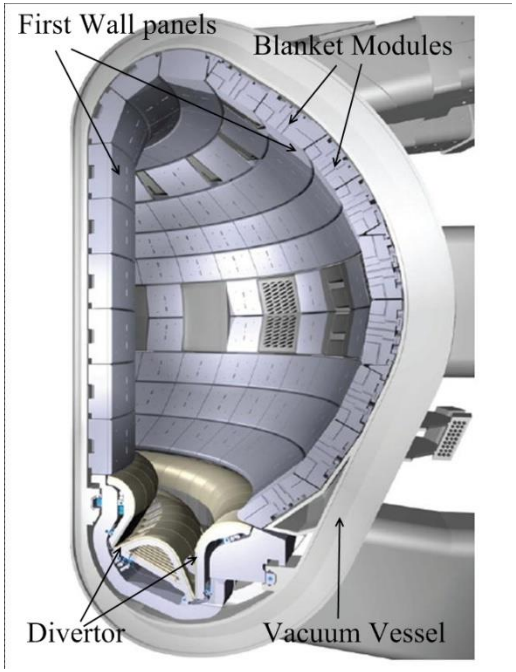
M. Lehnen, IAEA 2010

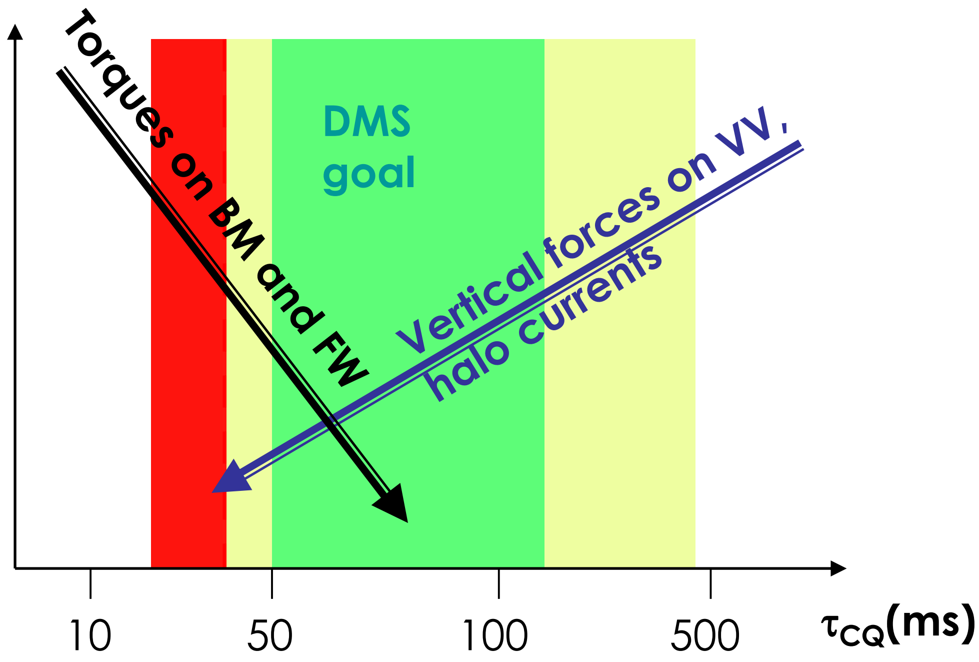


E.Hollmann, NF 2008

- ASDEX-U 60-100% G.Pautasso, Pl.Phys,2009
- Alcator C-mod ~75% R.S. Granetz, NF 2007
- JET ~ 90% M.Lehnen, EPS 2011

Forces impose constrain on maximum amount of gas

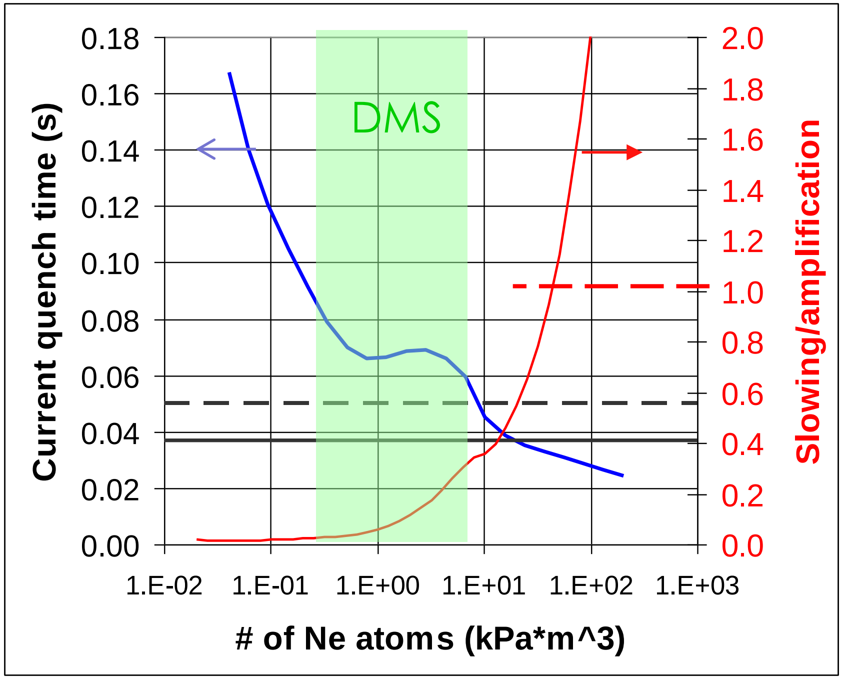




- The major EM loads on the VV and in vessel components occur during current quench of a disruption and following plasma VDE

Collisional suppression of RE is challenging in ITER

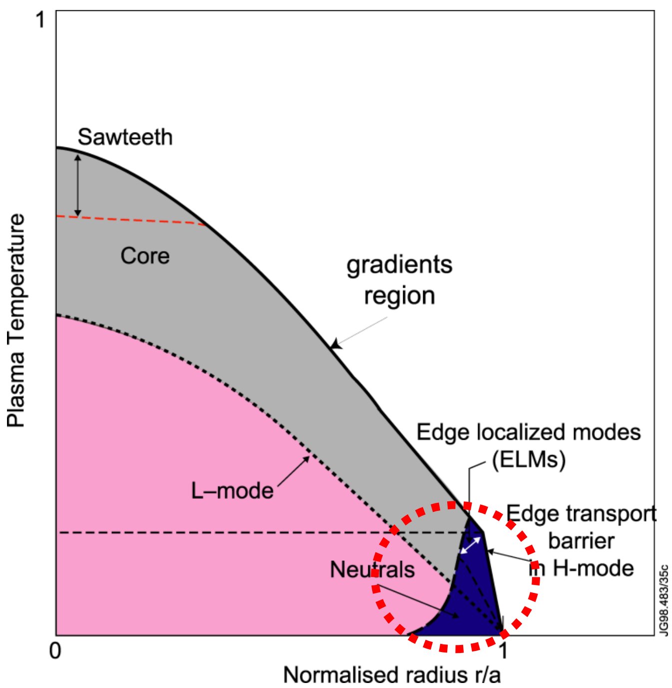
- Avalanche can be suppressed by:
  - increase of electron density to enhance collisional slow down of RE
  - enhancement of RE loss, $\gamma\tau_{loss}<1$

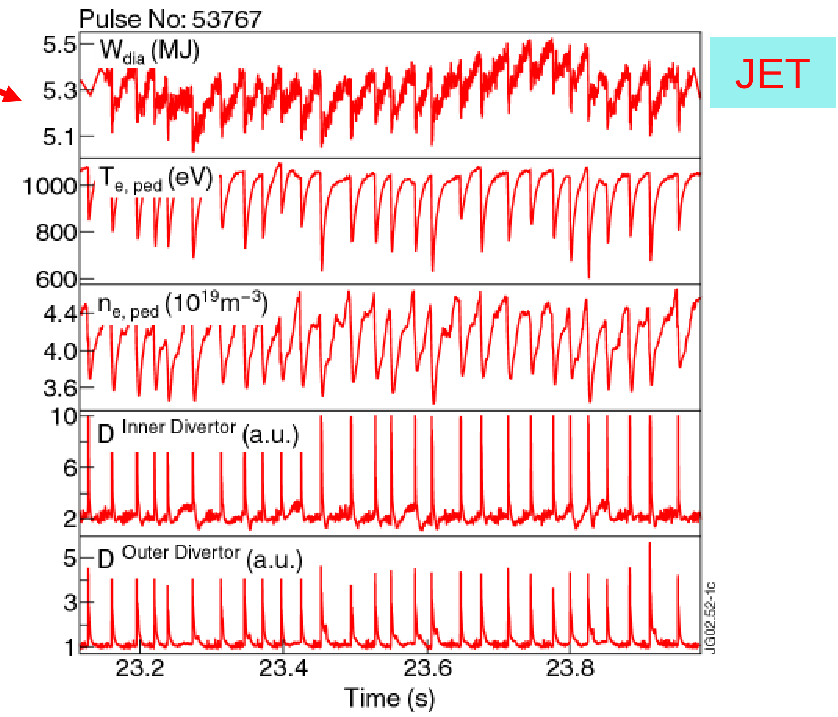


- Reaching critical density (with Ne) will likely be above capability of the machine
- Collisional suppression might work if RE will be suppressed at density less then 50% of Rosenbluth's density
- Suggested recently a new scheme for RE suppression could be an acceptable

ITER Plasma Scenario - ELMy H-mode

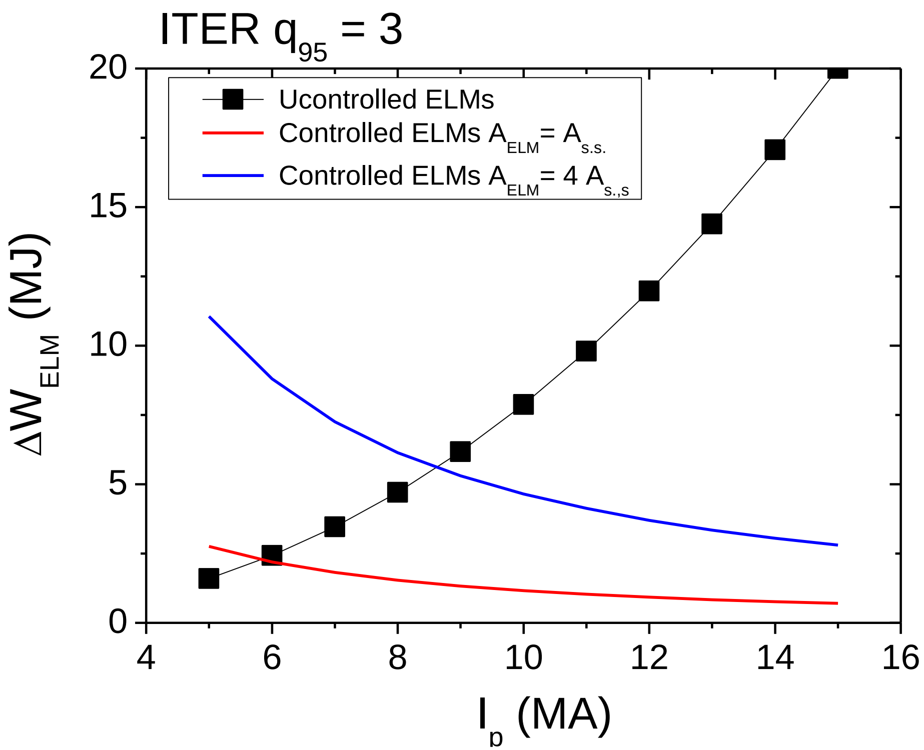
- Conventionally, plasma confinement regimes denoted L-mode and H-mode
  - The difference between these modes is caused by the formation of an edge pedestal in which transport is significantly reduced - edge transport barrier
  - edge localized modes maintain plasma in quasi-stationary state





ITER uncontrolled ELMs operation limited to → $I_p \leq 6 - 9$ MA

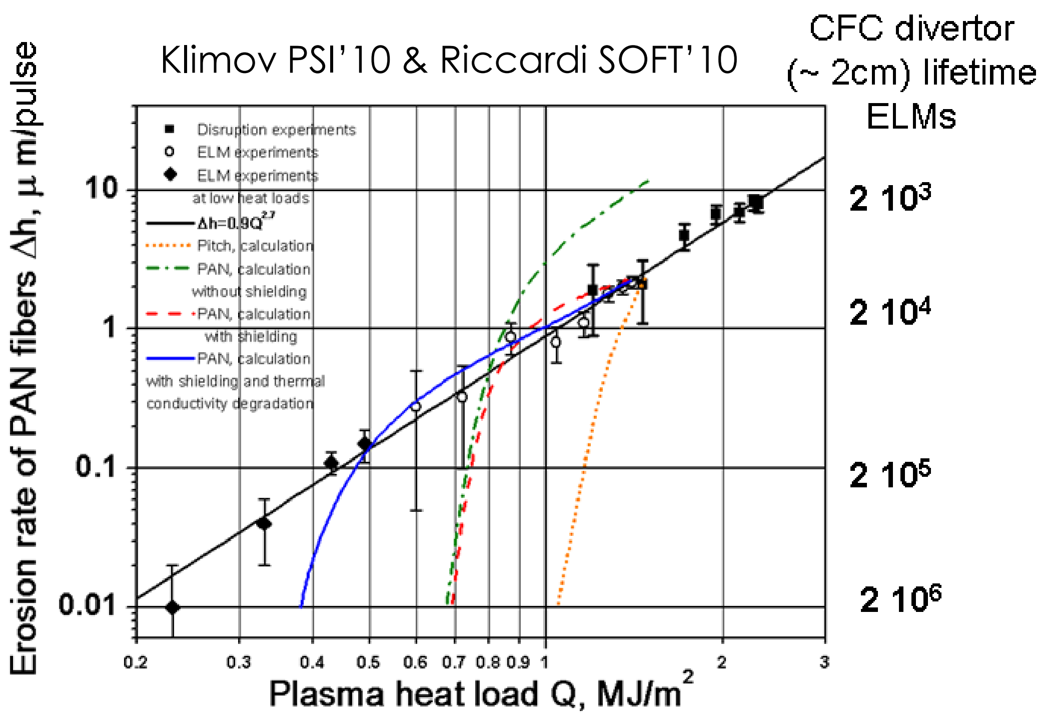
- ΔW_{ELM}^{uncont} determined by ELM physics and is ~proportional to I^2
- Material damage avoidance + ELM physics →
  - → $\Delta W_{ELM}^{controlled} \sim 0.7$ MJ (15 MA, $Q_{DT} = 10$, $A_{ELM} = A_{s.s.}$)



Uncontrolled ELM operation with low erosion possible up to $I_p = 6.0–9.0$ MA depending on $A_{ELM}(\Delta W_{ELM})$

ELM in ITER must be controlled

- ELMs must be controlled (ΔW_{ELM} reduced) to avoid excessive damage of the divertor targets



Approach to ELM control in ITER

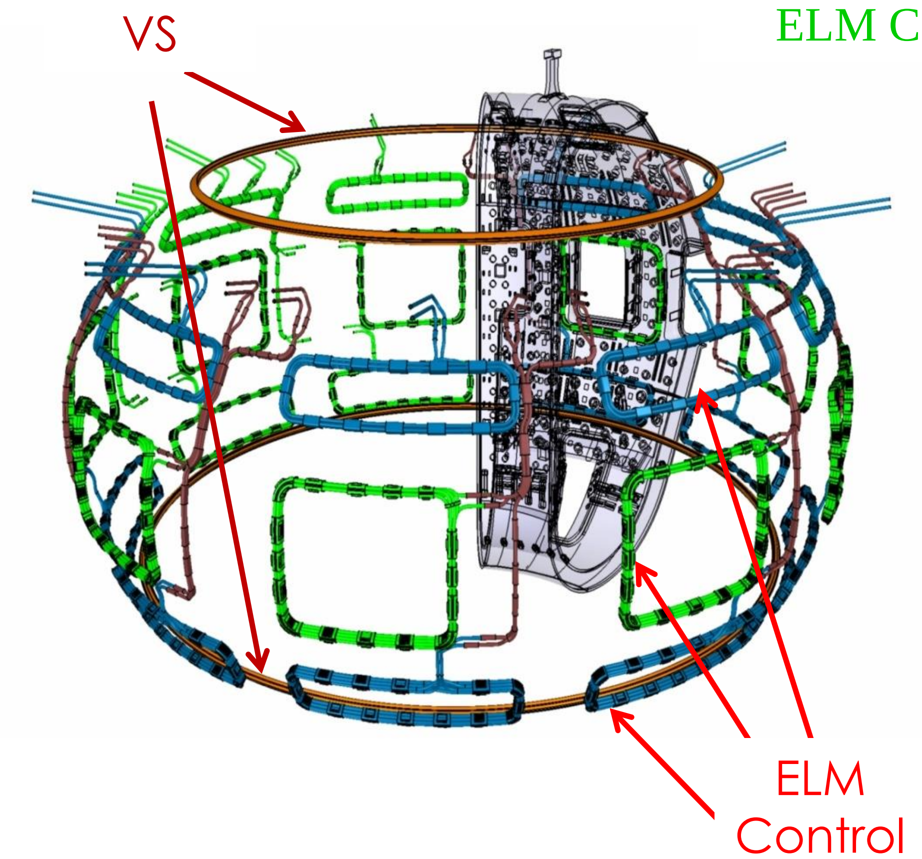
- ELM coils shall be installed. ELM coils shall create helical magnetic perturbations increasing transport at the edge

- Pellet pacing: periodic injection of cryogenic pellet will trigger smaller ELMs

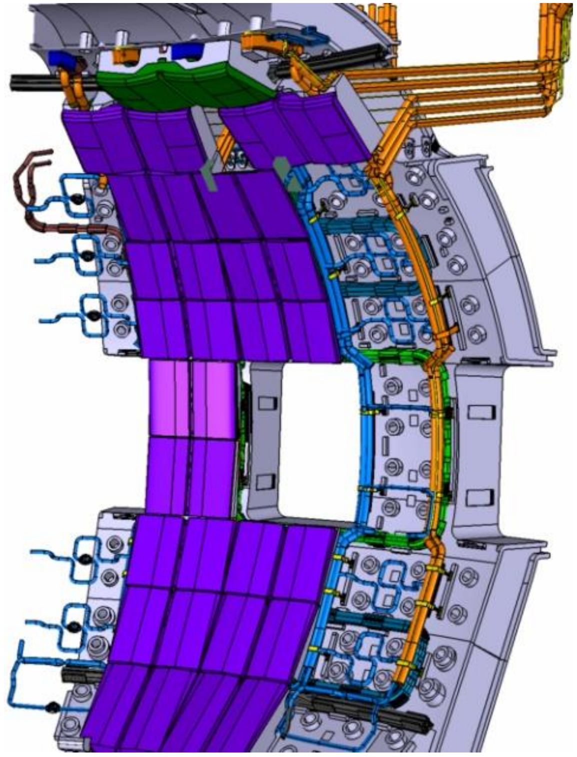
- Both ways have been successfully tested in the present experiments

ITER ELM control coils

- Three coils per Vacuum Vessel Sector (40º) → 27 ELM Control Coils
- ELM Control Coils located between Blanket Modules and Vacuum Vessel
- Every coil powered independently → required for flexibility across ITER operational scenarios (advanced $Q_{DT} = 5 \rightarrow q_{95} \sim 5$)



ELM Control Coils



Progress in ITER construction

ITER site in the future

Pit for the tokamak building

Pit for tokamak building, closer view

Pit for tokamak building, even closer view

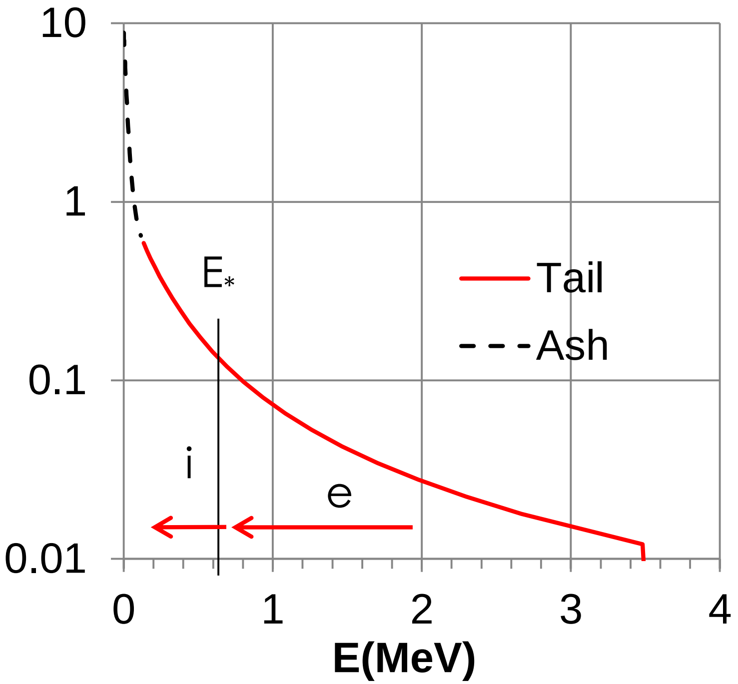
Building for winding PF coils

Building for winding PF coils

Classical confinement of α-particles

Energetic α-particles slow down on electrons and ions

- Alpha-particles have isotropic and almost monoenergetic source. Doppler broadening of the source is small ($dE \sim (TE_\alpha)^{1/2} << E_\alpha$)



Energy distribution of α-particles

$$f_\alpha = \frac{S\tau_s}{4\pi(v^3 + v_*^3)}$$

- Collisions with ions and electrons slow down α-particles

$$\frac{\partial f_\alpha}{\partial t} = \frac{1}{\tau_s}\left\{\frac{1}{v^2}\frac{\partial}{\partial v}(v^3 + v_*^3) f_\alpha + \frac{v_{**}^3}{2v}\Delta_\theta f_\alpha\right\} + \frac{S\delta(v - v_0)}{4\pi v^2}$$

where v_* is critical velocity

$$v_* = \left\{\frac{3\sqrt{\pi}}{4}\frac{m_e}{n_e\lambda_e}\sum_\beta \frac{Z_\beta^2 n_\beta \lambda_{\varepsilon\beta}}{m_\beta}\right\}^{1/3}\left(\frac{2T_e}{m_e}\right)^{1/2}$$

- and τ_s is slowing down time on electrons

$$\tau_s = \frac{3}{8\sqrt{2\pi}}\frac{m_\alpha T_e^{3/2}}{m_e^{1/2} n_e \lambda_e e^4 Z_\alpha^2}$$

Typical parameters of energetic α-particles in ITER

- Some numbers for DT plasmas with T ~ 10 keV:

- Slowing down time for alpha particles at T = 10 keV and n = 10^{20} m^{-3} ,

 τ_s ~ 0.3-0.4 s

- E^*/T_e ~ 30 -> E^*/E_0 = 0.1 <<1 -> about 80% of energy goes to electrons in a typical DT plasmas

- α-particle pressure: $W_\alpha = n_\alpha \langle E_\alpha \rangle \sim S \tau_s E_0/2$. Using power balance in the burning plasma

$$\frac{3nT}{\tau_E} = S_\alpha E_0 \longrightarrow \frac{p_\alpha}{p_{pl}} \approx \frac{3\tau_s}{4\tau_E} < 1$$

Formal τ_E scaling allows significant loss of alphas

- How sensitive is reactor performance to the loss fraction of energetic particles (λ)?

$$P(1-\lambda) = \frac{3nT}{\tau_E}$$

- Energy confinement time in a tokamak scales as

$$\tau_E = H\tau_{E,ELMY} \propto H / P^{0.69}$$

- Combining the above equations

$$H = \frac{H_0}{(1-\lambda)^{0.31}}$$

- 5-10% margin in H allows 15-30% loss of energetic alpha particles

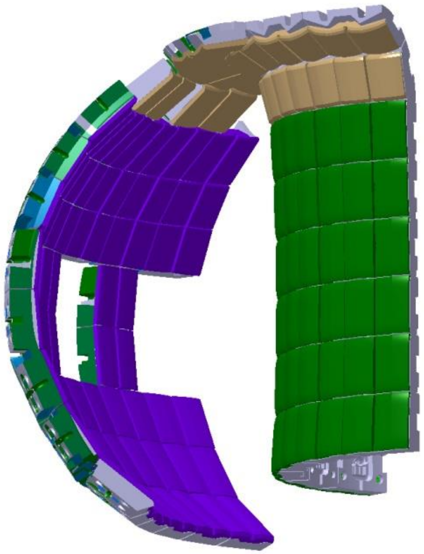
- H-mode requires a certain power crossing the separatrix. H-mode power threshold in ITER is about 70 MW and α-particle heating is 100 MW. About 10% of α-particle loss (CW) seems acceptable

- Thermal inertia of the plasma (τ_E ~3-4 s) and fast recovery of population of energetic alphas (τ_s ~0.3-0.4 s) allow very large single loss event (up to 100%) without interruption of the burn

Energy loads on the FW limit EP loss

- Orbit following Monte-Carlo calculations of particle loss predict ~ 0.3 MW/m^2 at 1% loss assuming 36 section of the FW

- At 10% loss ITER will have additional 3 MW/m^2 local power loads on the FW which is dangerously close to the maximum power handling capability of the FW panels in ITER (4.5 MW/m^2)



- Near the port area power loads could be larger because of fewer wall panels

- Power loads on the wall are more restrictive then global power balance. 5% loss can probably be tolerated

Classical confinement of energetic particles

- In axysimmetrical plasmas confinement of energetic particles is defined by conservation of energy

$$mv^2/2 + Ze\varphi = const$$

- magnetic moment

$$v_\perp^2/B = \mu = const$$

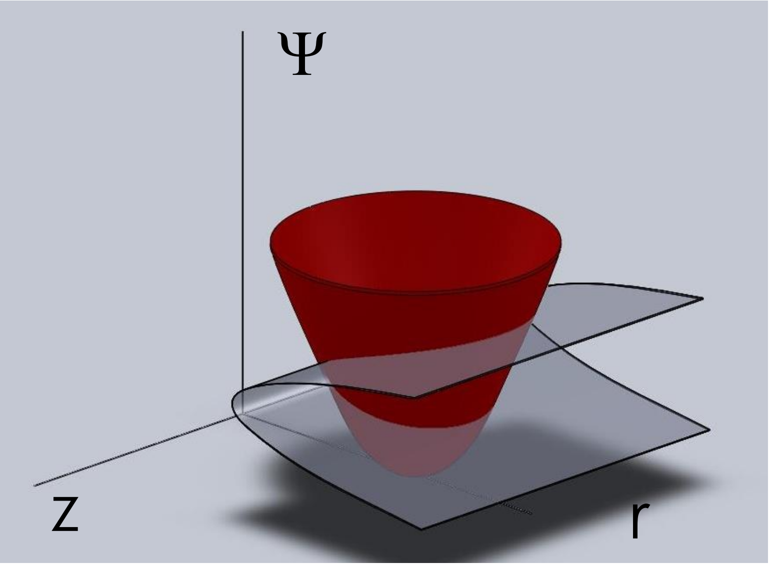
- and canonical angular momentum, P_φ (toroidally symmetrical field). In the drift approximation:

$$P_\varphi = mv_\| r - Ze\Psi/2\pi = const$$

- Assuming that B ~1/r and electric field is small the above equation can be reduced to

$$r^2 - rr_t - \left(\frac{Ze}{2\pi mv}\right)^2 (\Psi - \Psi_0)^2 = 0$$

$$\Psi = \Psi(r, z)$$



- Typical size of the orbits in ITER

ρ_L = Vm/ZeB ~ 5 cm;

Δ_{banana} ~ ρ_Lq/ε1/2 ~ 20 cm << a

Classical confinement of energetic particles

- α-particles do experience NC diffusion. However, diffusion displacement during τ_s is smaller than width of banana orbits

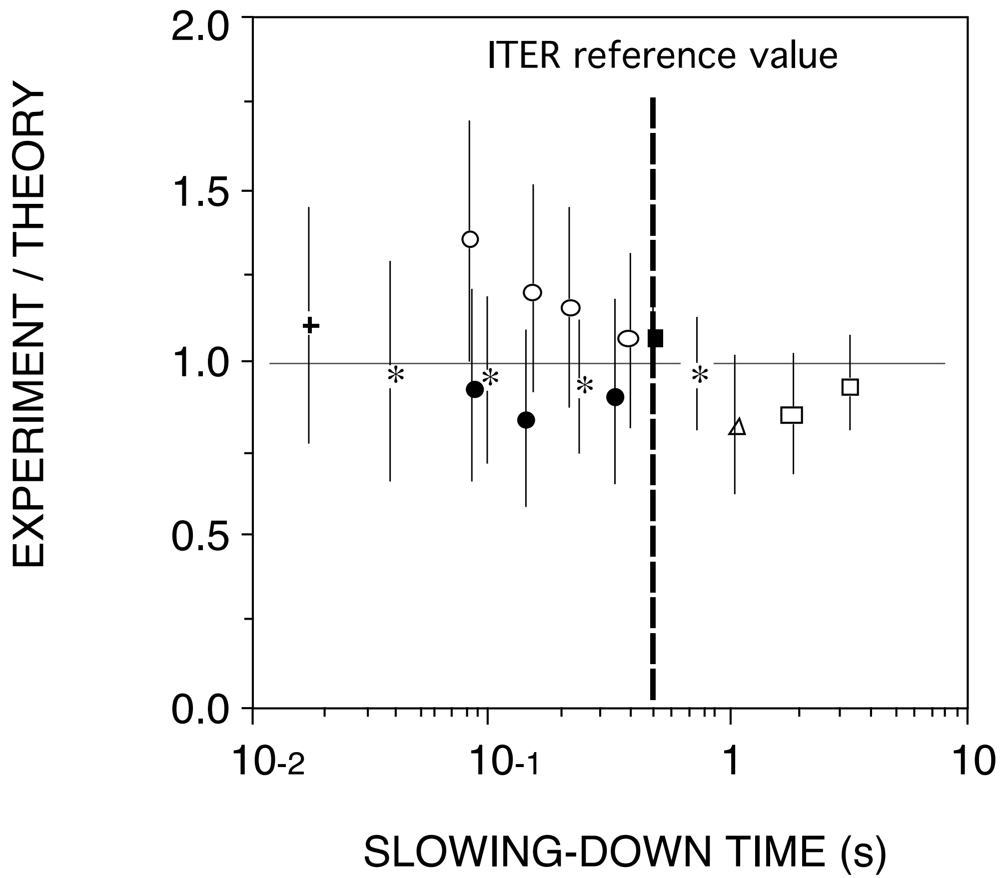
$$D_\vartheta \approx \frac{v_*^3}{v^3}\frac{1}{\tau_s} << \frac{1}{\tau_s}$$

- By the time when v -> v* and ~80% of energy transferred to the plasma average orbit displacement is smaller than the initial banana width

- Large orbit width of energetic particles results in an averaging small scale turbulence (such as ITG) responsible for turbulent transport of the main plasmas

- Spatial diffusion of a-particles starts when they cool down to ash

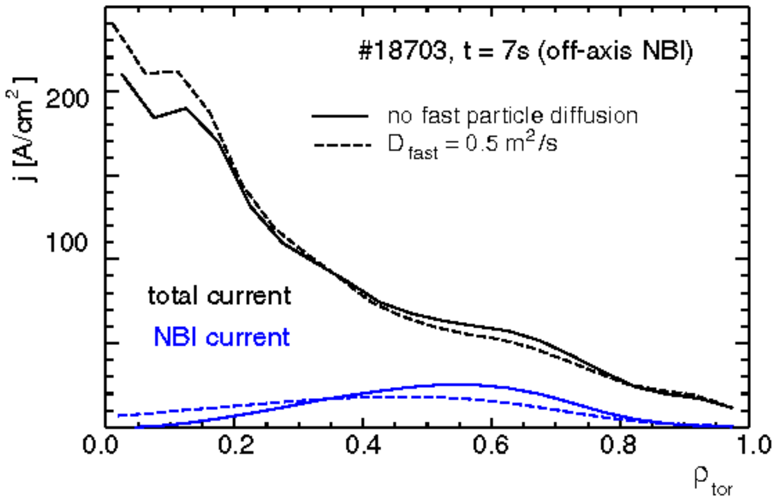
Classical slowing down usually observed



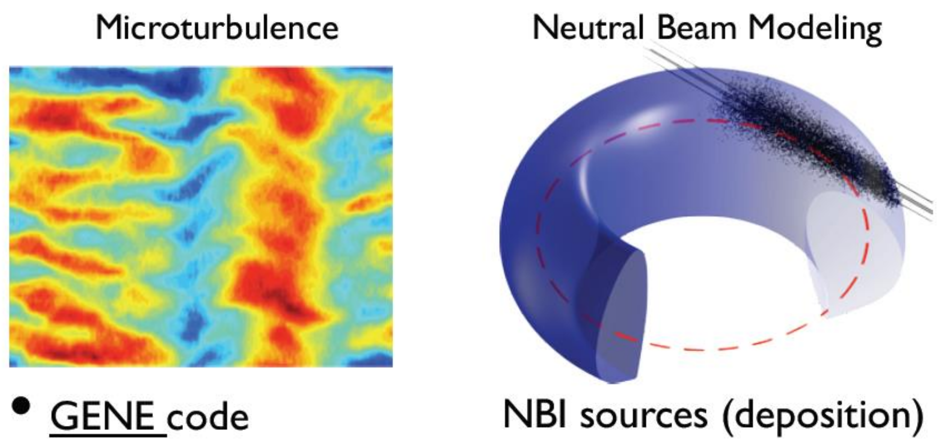
W W Heidbrink, G J Sadler, Nucl Fusion **34** 535 (1994)

Some diffusion remains in spite of averaging

Agreement of measured profiles with theory: only if diffusion of fast particles assumed (D_{fast} ~ 0.5 m^2/s)



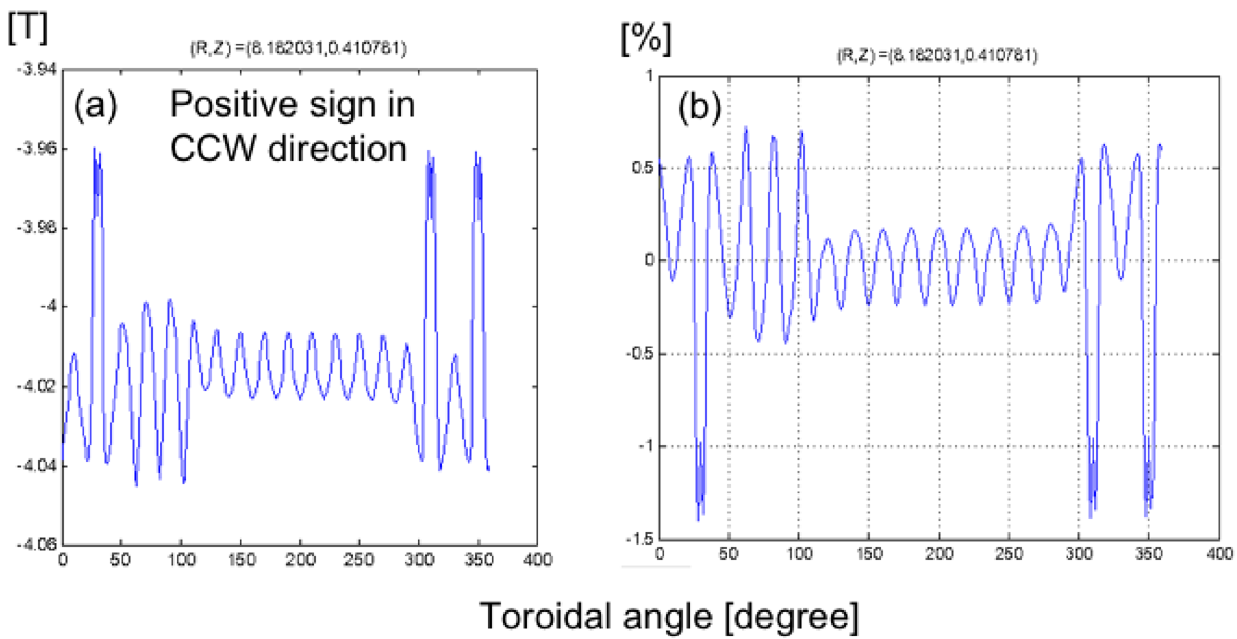
AUG experiments
(S.Guenter)



Modeling for ITER shows that effect of microturbulence on 1 MeV beam is small
(M.Albergante et al.)

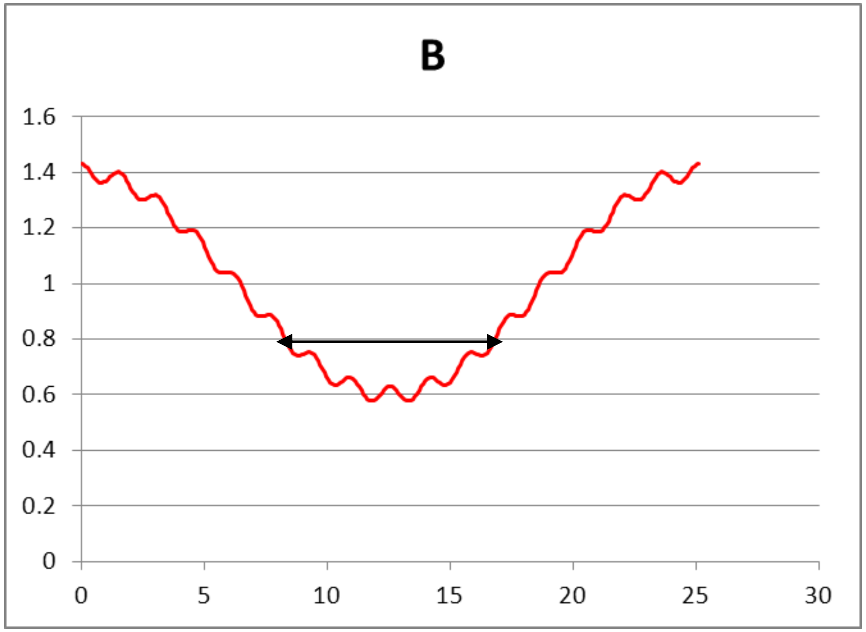
Effect of DC magnetic perturbations

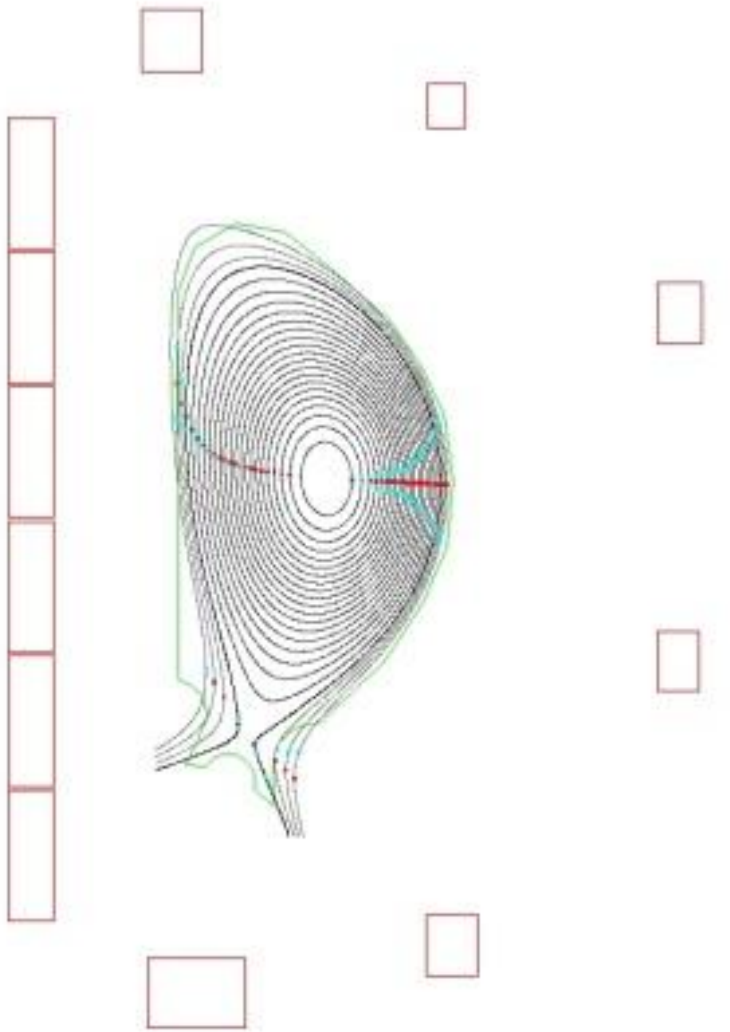
- Periodic variation of magnetic field due to discreteness of the TF coils always exists in a tokamak



ITER toroidal magnetic field

Particle trapped in a local well are lost





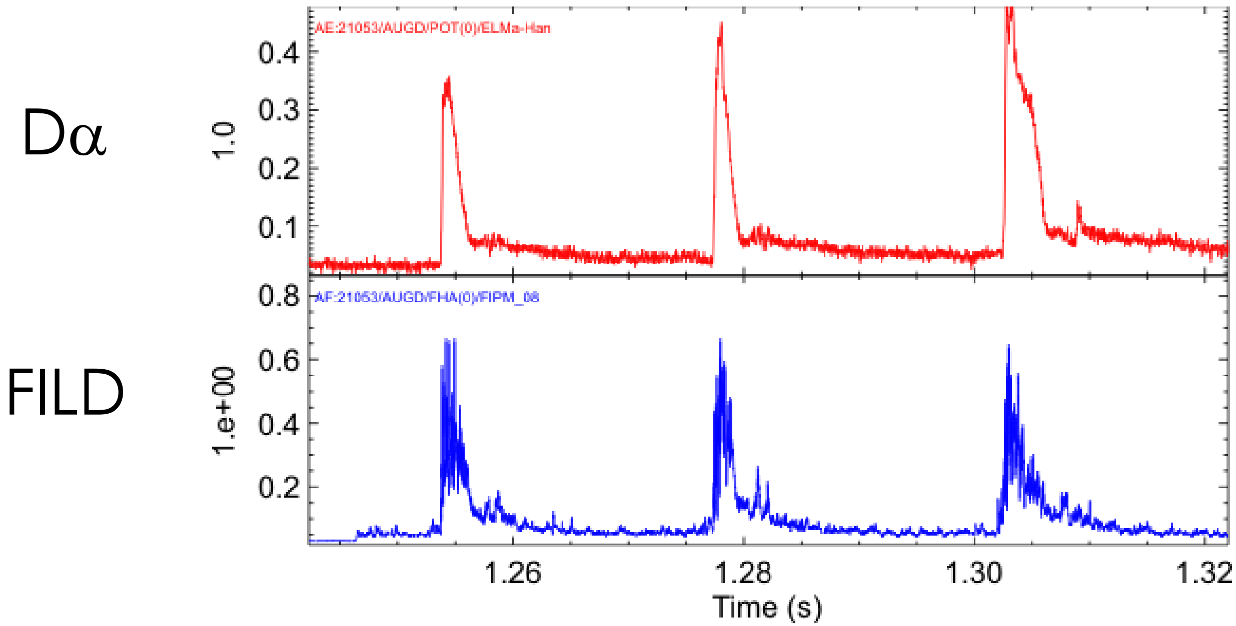
- Local wells exist only $\theta = 0$

- Local well region in ITER is very small and loss fraction of alpha-particles is less than 0.1%

Effect of ELMs and ELM coils

- It has been discovered recently that helical field produced by ELM coils can result in a significant loss of NB ions (up to 20% of 1 MeV in ITER). The work is in progress to optimize coil current to reduce loss

- ELM themself can result in EP-loss

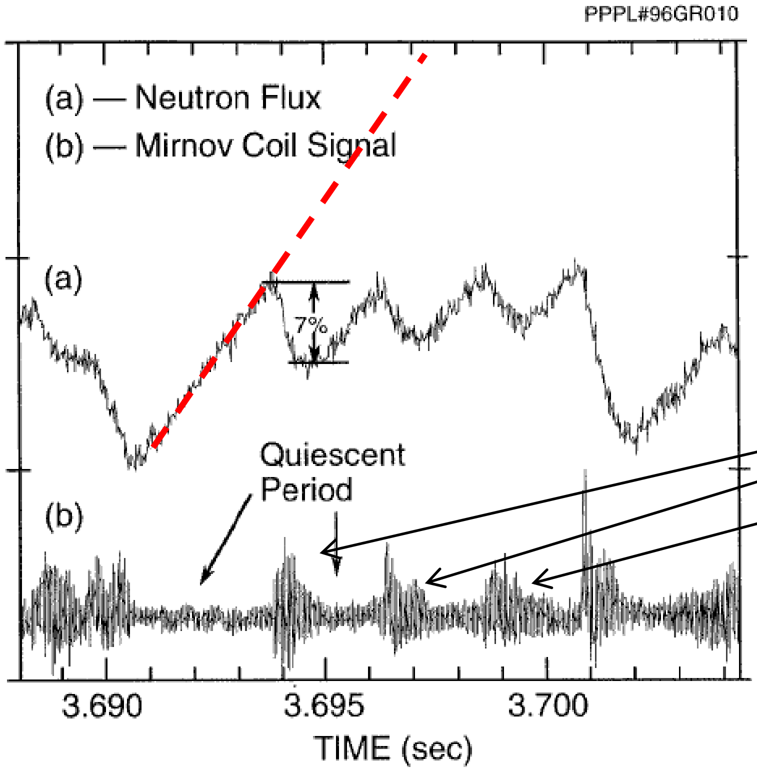


An example from ASDEX Upgrade, 2011.
Peak loss = 15 of prompt loss

- Alpha-particle loss is small due to small source of alphas at the edge

Significant loss can be caused by Alfvén modes

TFTR
K.L.Wong
1999



Bursts of Alfvén modes

a) Emission rate of neutrons produced by energetic D

b) Mirnov coil signal measuring Alfvén modes

The stored beam energy is at 40% of classical slowing down value!

Alfven modes

- An impressive progress has been made in understanding of Alfven modes in tokamaks during last 30 years
- It has been shown theoretically and in experiments that TAE can be made unstable by energetic particles and the instabilities can result in large loss of EP’s
- New plasma diagnostic techniques such as *magnetohydrodynamic (MHD) spectroscopy* have been developed to probe plasmas in tokamaks
- Sophisticated numerical codes for analysis of these modes and their effect on EP have been developed and benchmarked on experimental data

Toroidal Alfvén Eugenmodes (TAE)

- Alfvén wave is one of fundamental MHD waves

$$\omega = k_{\|} V_A = \frac{m - nq}{qR} V_A$$

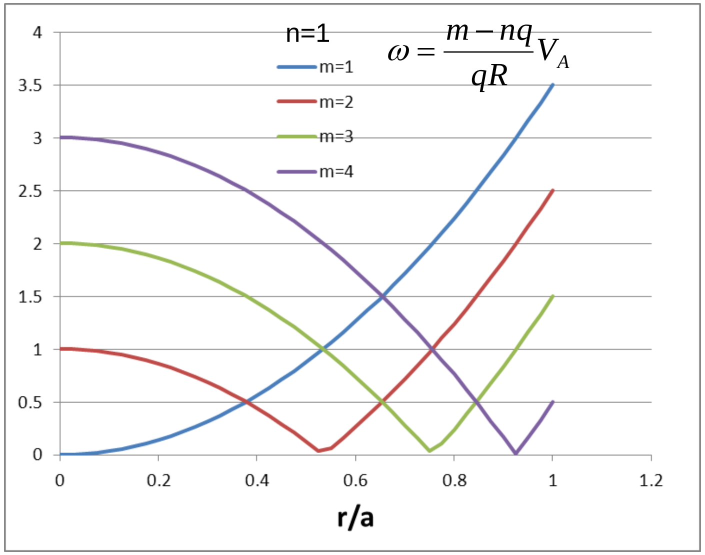
- where q is safety factor $q=rB/RB_\theta$ and V_A is Alfvén velocity

$$V_A = \frac{B}{(2\mu_0 \rho)^{1/2}}$$

- The waves propagate along the magnetic field lines

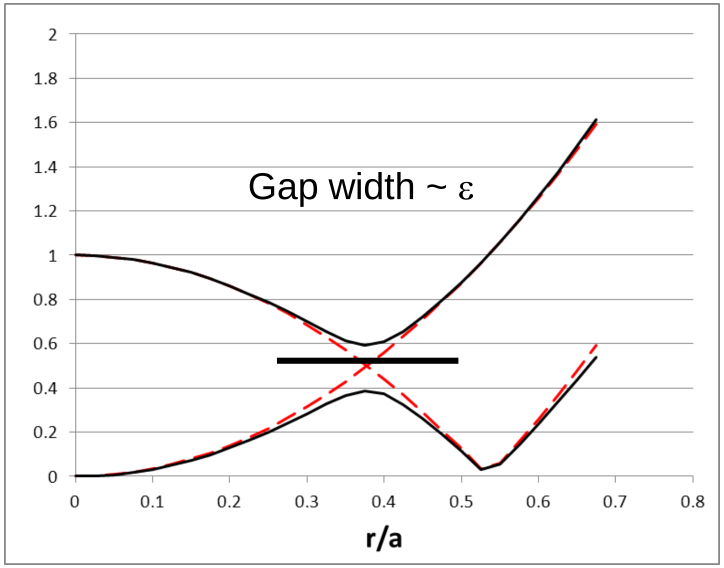
- In a cylinder, with $q=q(r)$, $k_{\|}$ varies with r resulting in continuous spectrum. Continuous modes are very localized and have large damping rate $\gamma \sim \omega$

Gap modes



In torus

B~1+εcos(θ)



$$(\omega^2 - \omega_m^2)\xi_m + \varepsilon\omega_m^2\omega_{m+1}^2\xi_{m+1} = 0$$
$$(\omega^2 - \omega_{m+1}^2)\xi_{m+1} + \varepsilon\omega_m^2\omega_{m+1}^2\xi_m = 0$$

- A gap in continuous spectrum appears near the point where $\omega_m=\omega_{m+1}$ (and $\omega=V_A/2qR$)
- Plasma ellipticity results in the gaps at $\omega_m=\omega_{m+2}$ near frequency $\omega=V_A/qR$, *etc.*
- The damping rate of this modes by plasma is small γ/ω ~1-2%

Alfvén Eigenmodes

- A series of discrete frequency <u>Alfvén eigenmodes</u> can exist in these gaps
  - Toroidal Alfvén Modes (TAE) — gap created by toroidicity
  - ellipticity-induced (EAE) — gap created by elongation
  - triangularity-induced (NAE) — gap created by additional non-circular effects
  - beta-induced (BAE) — gap created by field compressibility
  - kinetic toroidal (KTAE) — gap created by non-ideal effects such as finite Larmor radius

… and others!

- These modes can be driven unstable by the free energy arising from energetic particle populations with velocities above the Alfvén velocity, eg α-particles

Alpha-particle drive

- To transfer energy from a-particle to the mode a resonance between particle and the wave is needed

- For transit particles: $v_{||} = V_A$, for banana particles $\omega = \omega_{bounce}$ -> $v = V_A/\varepsilon^{1/2}$ etc

- The free energy comes from pressure gradient of energetic particles

$$\frac{\gamma}{\omega} \approx \beta_a \left(\frac{\omega_{*\alpha}}{\omega_0} - \frac{1}{2} \right)$$

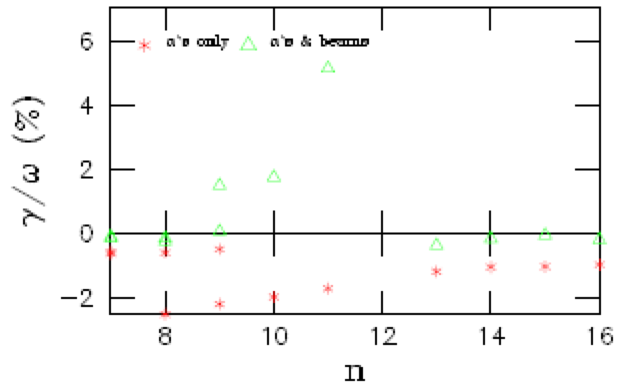
- where

$$\omega_{*\alpha} = -\frac{m}{rZeBn_a} \frac{dp_\alpha}{dr}$$

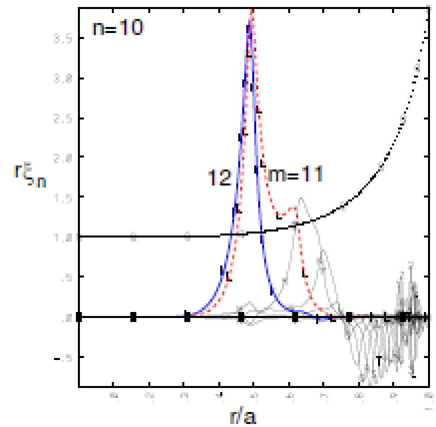
- One can expect that TAE will result in a flattening or loss of a-particles

ITER 15 MA, Q=10 plasma is marginally stable

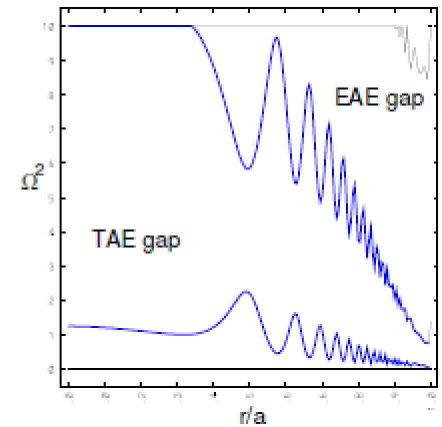
Growth rates vs. n



Most unstable n=10 TAE



AE continuum



N.Gorelenkov, ITPA, 2006

- An increase of plasma temperature above 10 keV results in a fast increase of the EP beta and can make TAE unstable
- Activity in this area is targeted on mapping unstable regimes in operational ITER space

ITER will be the first device to study burning plasmas

- Dimensionless parameters for confinement of energetic particles

$$\frac{V_\alpha}{V_A} \qquad \frac{\rho_\alpha}{a} \qquad \beta_\alpha$$

- The first two can be combined to yield

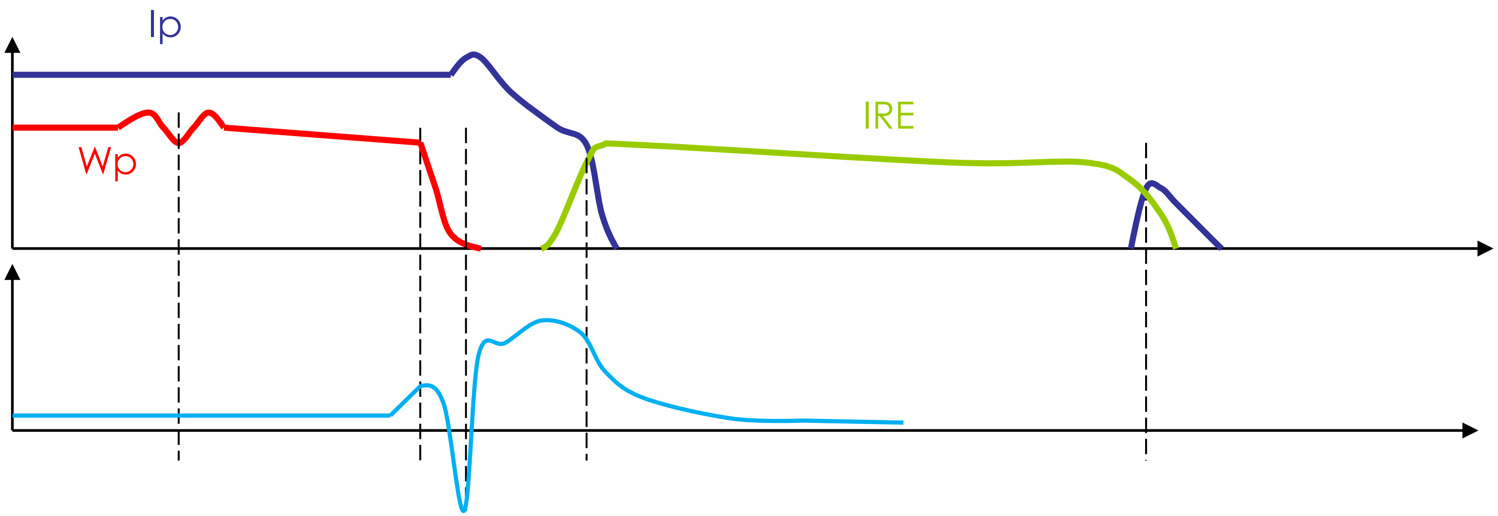
$$\frac{V_\alpha^2}{V_A^2} \propto \frac{\rho_\alpha^2}{a^2} n a^2$$

- Alpha-particle confinement in burning plasmas of a reactor can be simulated in device with the same na^2 as the reactor

- Present experiments provide better understanding of the physics but the integrated test can be done only in ITER scale device

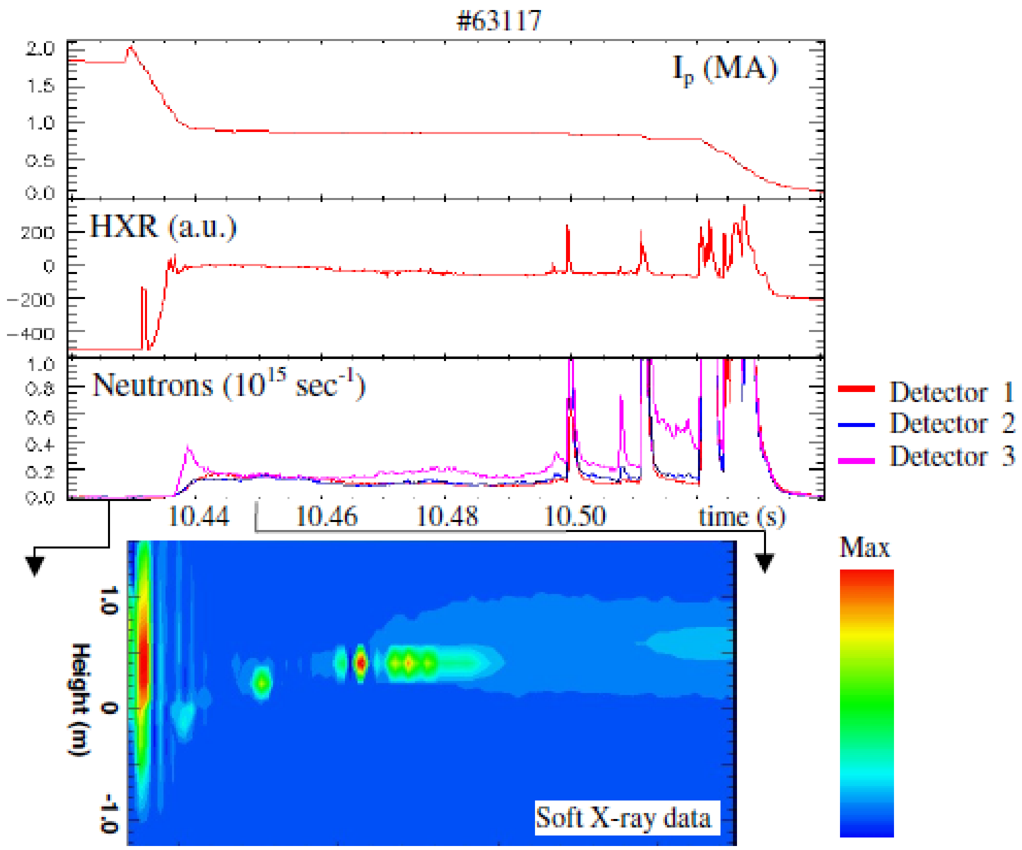
Runaway Electrons

Runaway electrons can be produced during CQ



- Typical time traces during plasma disruption in a tokamak
- The high loop voltage occurs during Current Quench when plasma is dirty and resistive
- The high loop voltage at certain conditions can accelerate electrons
- Runaway electrons are dangerous for the plasma facing components because of their long range in materials

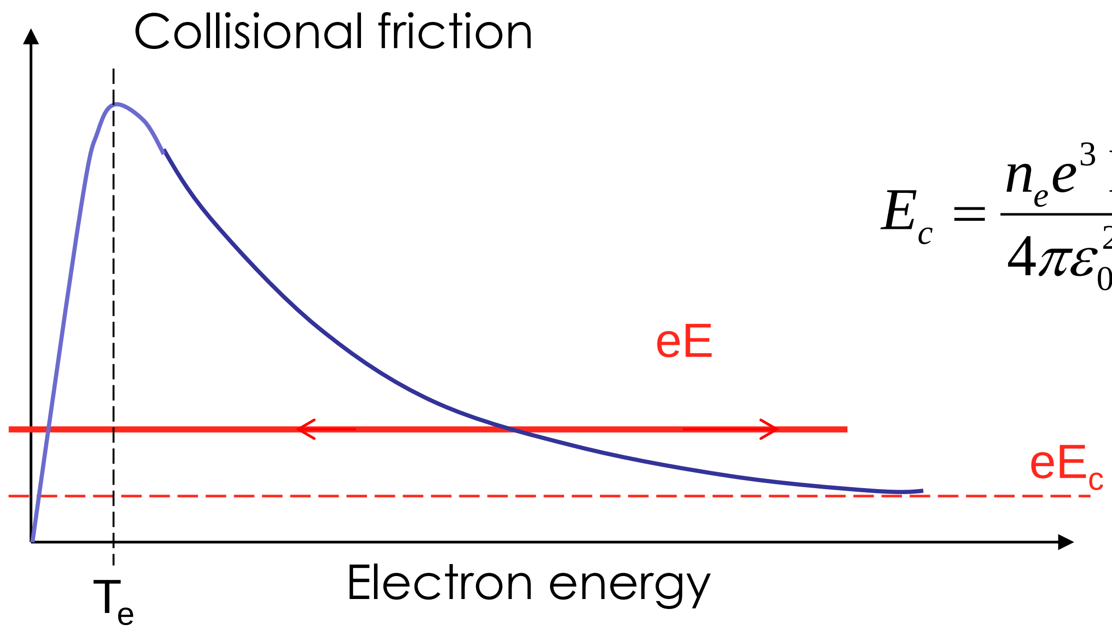
Runaway electrons are often observed during plasma disruptions



- Runaway electrons in JET (Pluschin, NF,1999

- Large loop voltage can accelerate electrons to > 10 MeV
- Plasma resistive current is replaced by current of relativistic electrons
- Hard X-rays and photoneutrons are typical signature of energetic electrons
- Soft x-rays from chord array show that RE current is peaked near magnetic axis

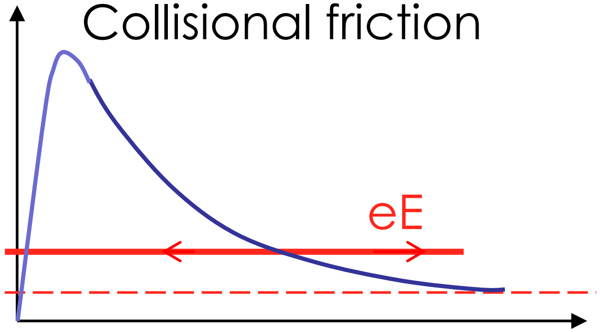
Physics of RE generation



$$E_c = \frac{n_e e^3 \ln(\Lambda)}{4\pi\varepsilon_0^2 m_e c^2}$$

- Acceleration of energetic electron when electric field larger than friction force from background electrons
- At $E < E_c \sim n_e$, the runaway electrons can not be produced

Dreicer acceleration



- Introduce electric field equal to maximum friction force (Dreicer field):

$$E_D = \frac{n_e e^3 \ln(\Lambda)}{4\pi\varepsilon_0^2 T_e}$$

- At electric field much smaller than maximum friction force only electrons from far Maxwellian tail can accelerate

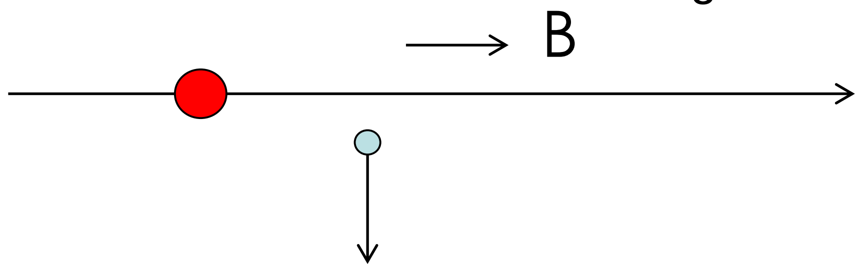
- In this case the runaway generation rate can be calculated from kinetic equation

$$\dot{n}_r = \frac{n_e}{\tau}\left(\frac{m_e c^2}{2T_e}\right)^{3/2}\left(\frac{E_D}{E}\right)^{3/8}\exp\left\{-\frac{E_D}{4E}-\sqrt{\frac{2E_D}{E}}\right\}$$

- This is Dreicer source of runaway electrons

Electron avalanche

- Close Coulomb collisions of “heavy” energetic electron with electron of the background plasma can double number of energetic electron per collision



- A new source proportional to RE current will appear in the kinetic equation with a predominant direction of velocity normal to the magnetic field

- Theory of this phenomena results in equation for RE current density

$$\frac{1}{j_{RE}}\frac{dj_{RE}}{dt} = \gamma\left(\frac{E}{E_c} - 1\right) \sim \gamma\frac{E}{E_c} \sim \frac{eE}{3m_c c\ln(\Lambda)}$$

- At $E = L(dI/dt)/2\pi R$ the number of e-folds in avalanche amplification of RE is proportional to initial plasma current

$$\ln(I_{RE} / I_{RE,0}) \approx 2I_{pl}$$

RE avalanche is very important in ITER

- The avalanche will continue until all resistive current will decay and E will drop to E_c.

$$I_{RE} = LI / L_{RE}$$

- The RE energy is either not sensitive to the plasma parameters

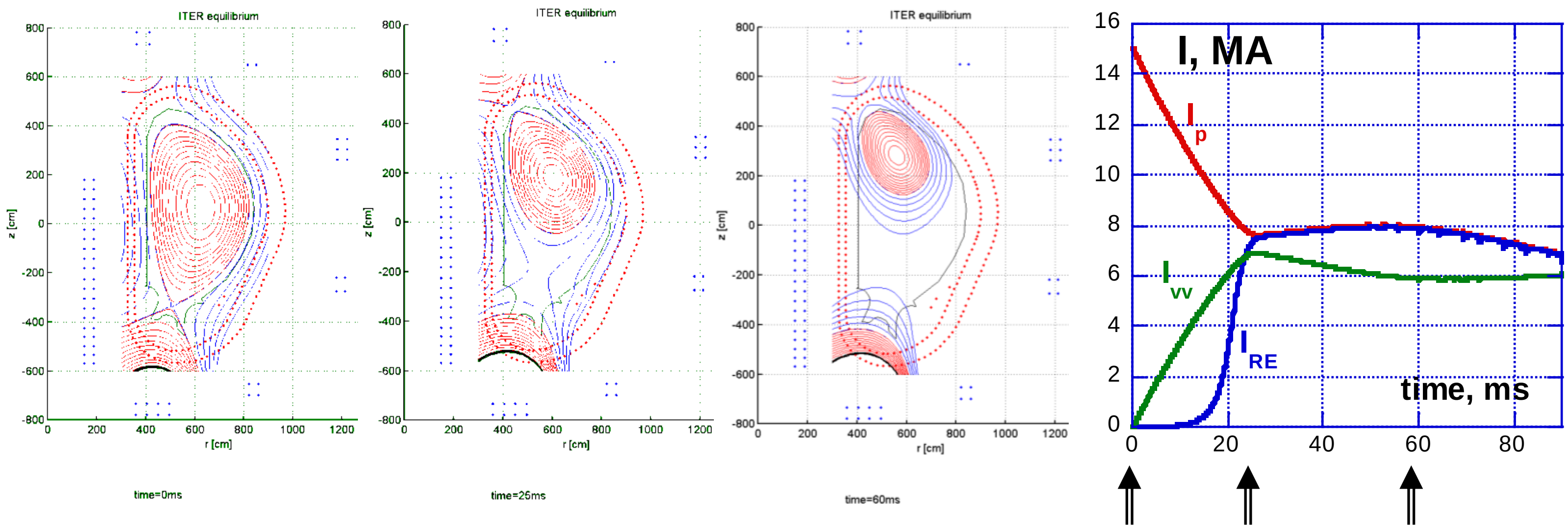
$$\frac{d\varepsilon}{dt} = eEc - \frac{\dot{n}_{RA}}{n_{RA}}\varepsilon$$

$$\varepsilon = eEc\frac{n_{RA}}{\dot{n}_{RA}} \approx mc^2\left(\frac{2}{\pi}\right)^{1/2} 3\ln\Lambda \approx 10-20\ MeV$$

- These results are corroborated by numerous numerical simulations

Runaway electrons must be suppressed in ITER

- 2D simulation of CQ in ITER by code DINA. $n_{DT}=5 \cdot 10^{19}$ m^{-3}, Ar impurity, 7%



- Very large peaking of RE loads and long range of 10-20 MeV electrons in FW materials can result in a deep melting of FW Be panels

- To avoid melting of the wall the RE current must be suppressed to less then 2MA in ITER

- Several schemes for RE suppression are under development

Movie

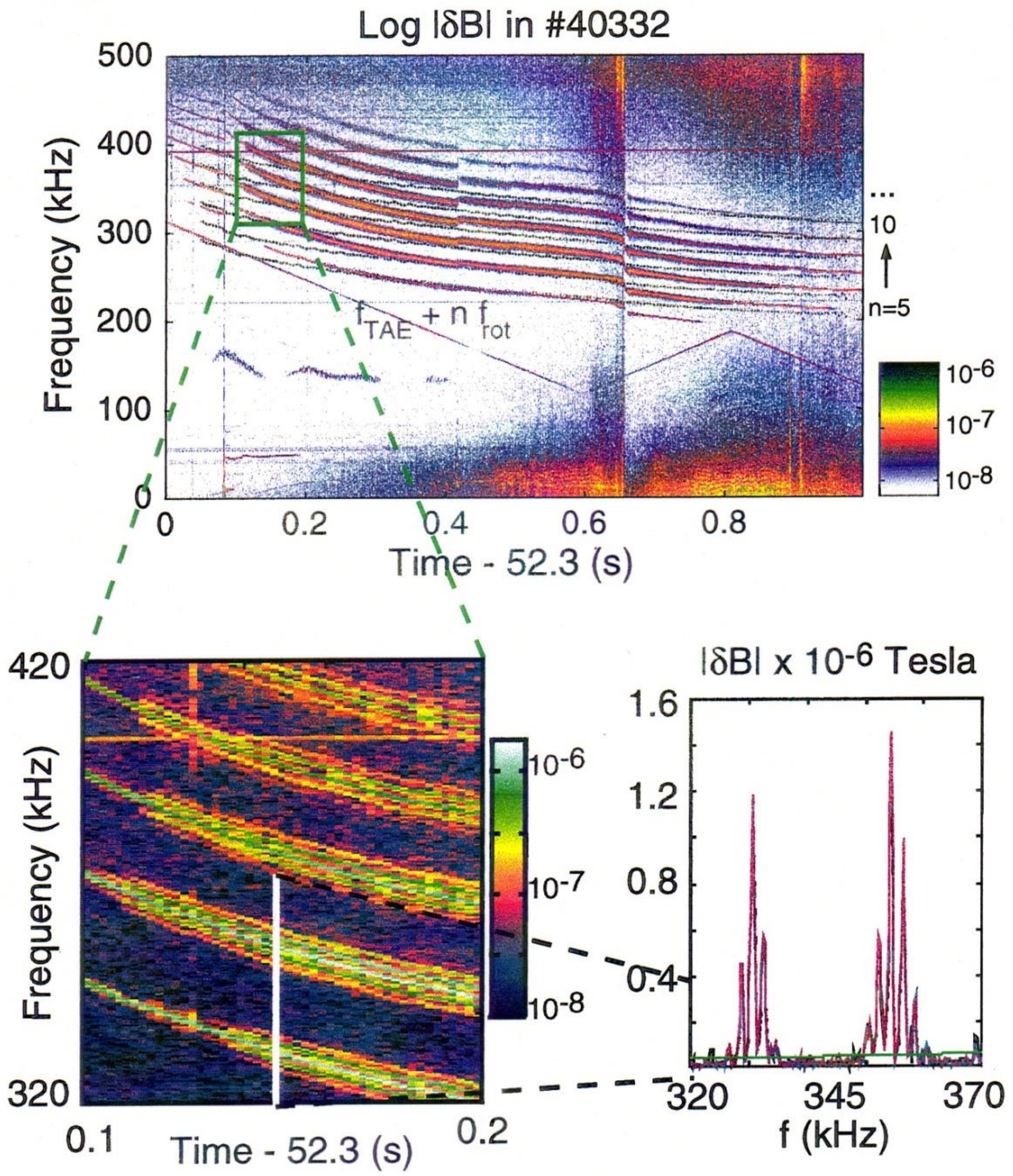
Summary and conclusions

- Energetic particles is essential component of the burning plasmas
- Good confinement of alpha-particles is a key to the success of the fusion program
- Not all energetic particles are welcome. RE electrons must be suppressed

ITER site in the future

Additional slides

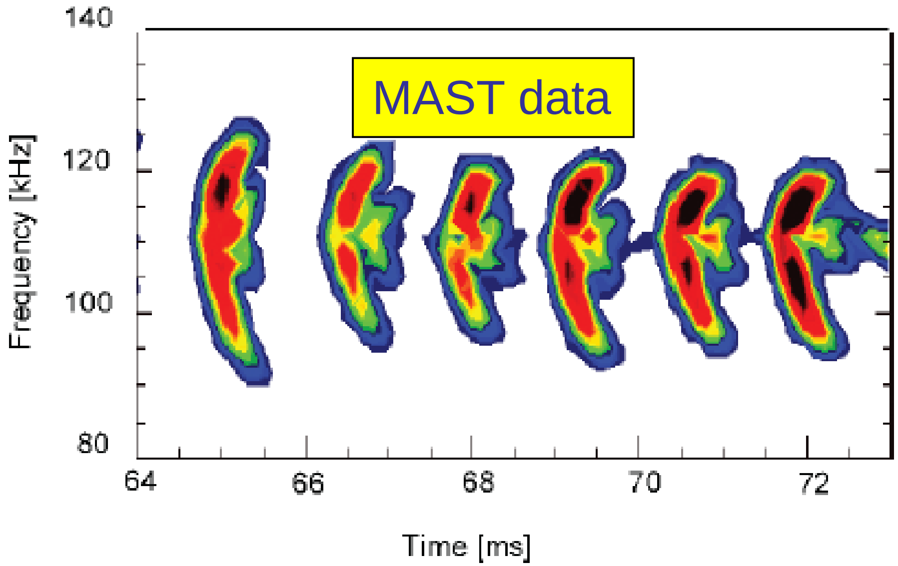
Nonlinear Splitting of Alfvén Eigenmodes in JET

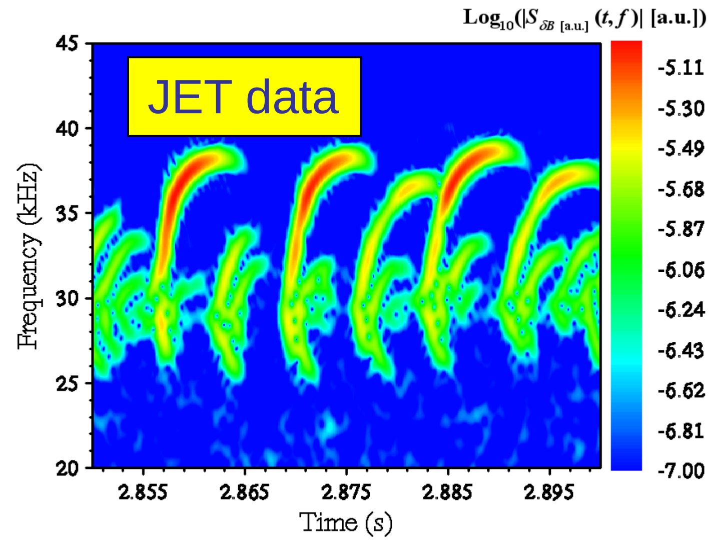


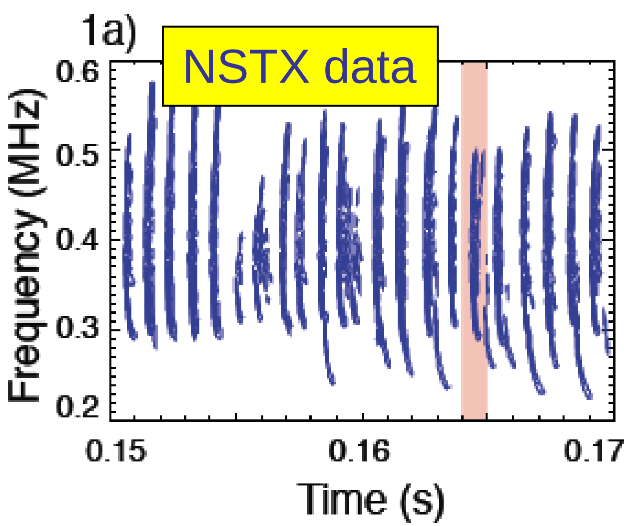
A. Fasoli, et al.,
*PRL **81**, 5564 (1998)*

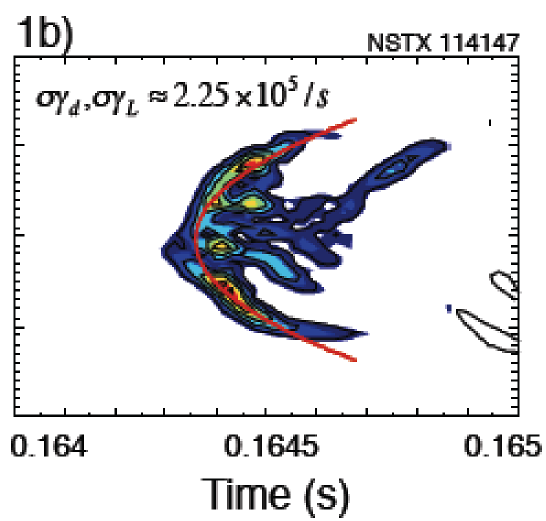
R. F. Heeter, et al.,
*PRL **85**, 3177 (2000)*

Rapid Frequency Sweeping Events





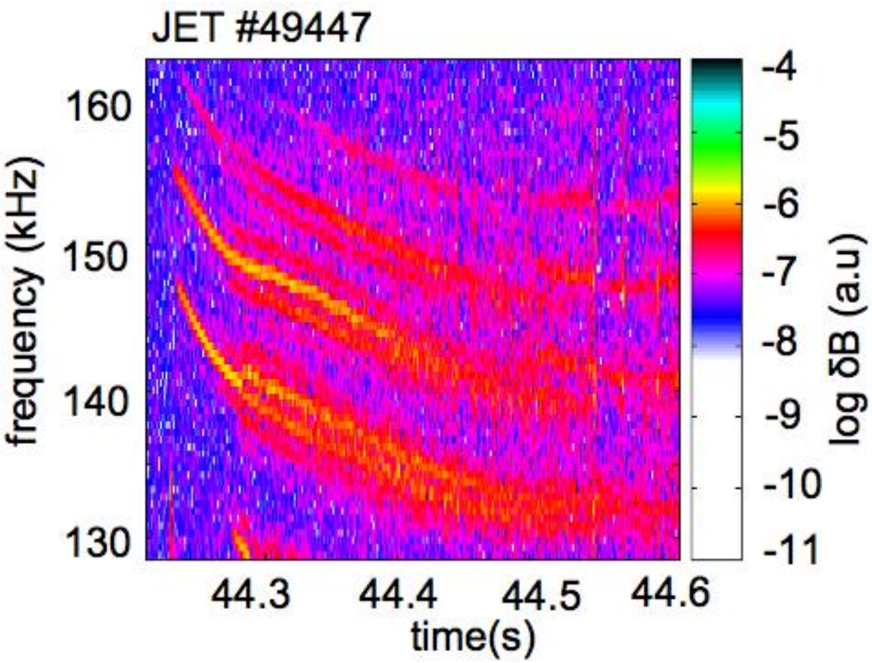




The ms timescale of these events is much shorter than the energy confinement time in the plasma

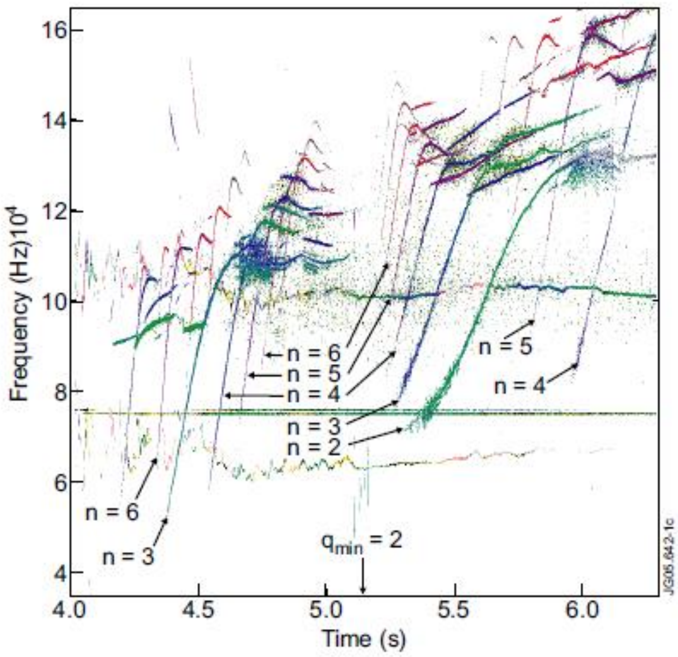
Fast Particle Driven TAEs

ICRH drive (JET)



*R. F. Heeter, et al., PRL **85**, 3177 (2000)*

ICRH drive, Reverse Shear (JET)



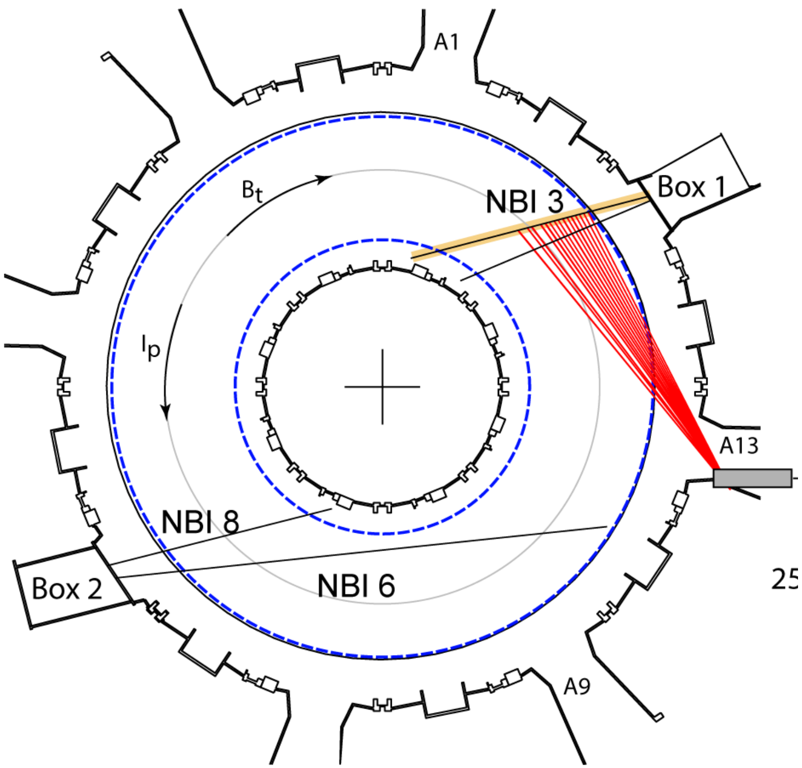
S.Sharapov, 2002

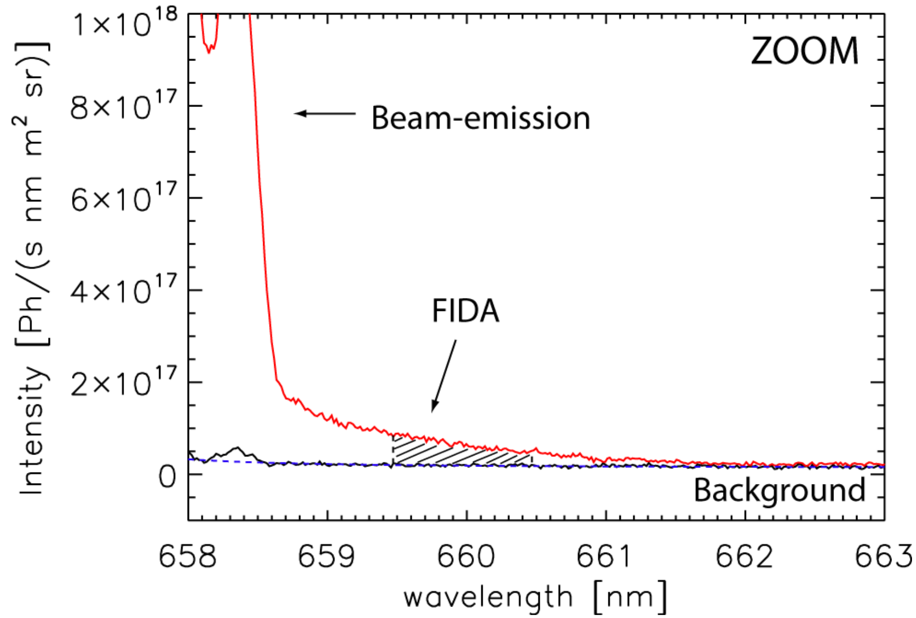
Experimental tools for measuring EP

- Mirnov coils (100th kHz range) for AE signal
- Reflectometry for measuring density perturbations created by AE
- Fast-Ion D-Alpha (FIDA)
- Fast Ion Loss Detectors
- Tritium burn up and neutron measurements
- Charge-Exchange Recombination Spectroscopy (CHERS)
- Others

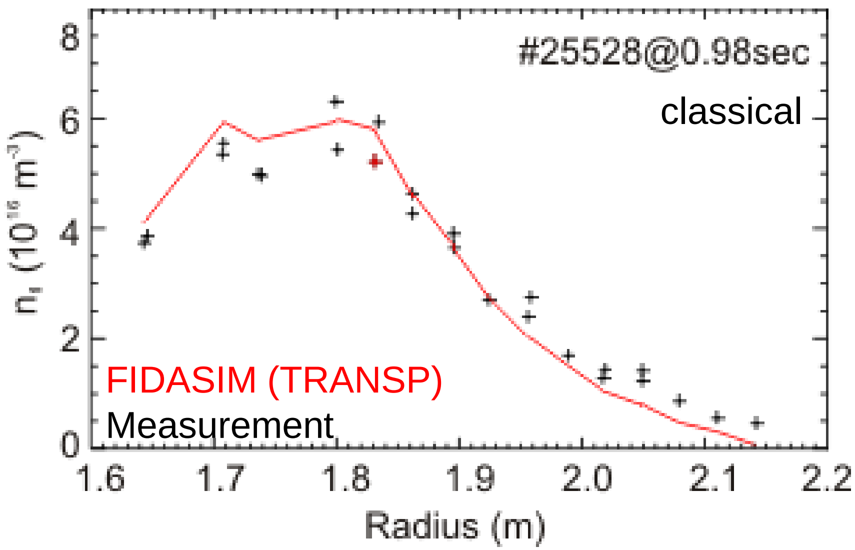
Fast-Ion D-Alpha (FIDA) Diagnostic at AUG

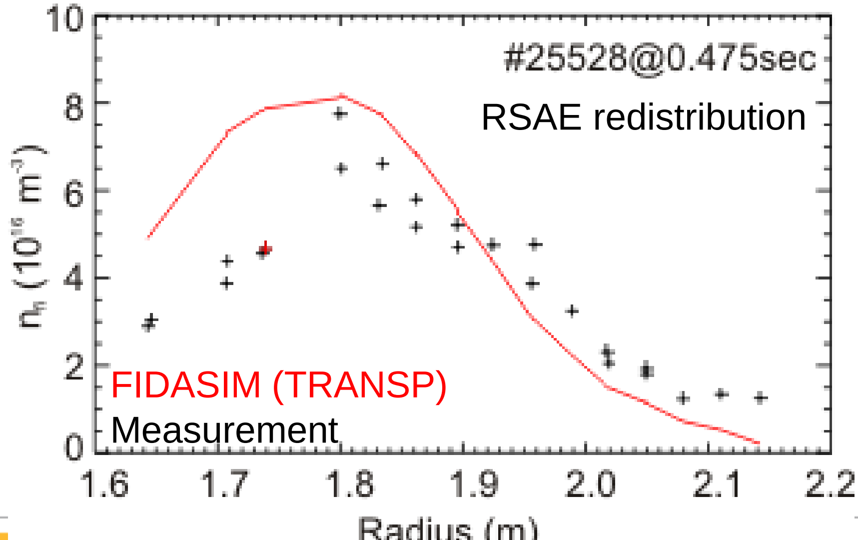
FIDA measures Fast-Ion D-Alpha emission from neutralized fast-ions as they CX in diagnostic beam neutral cloud

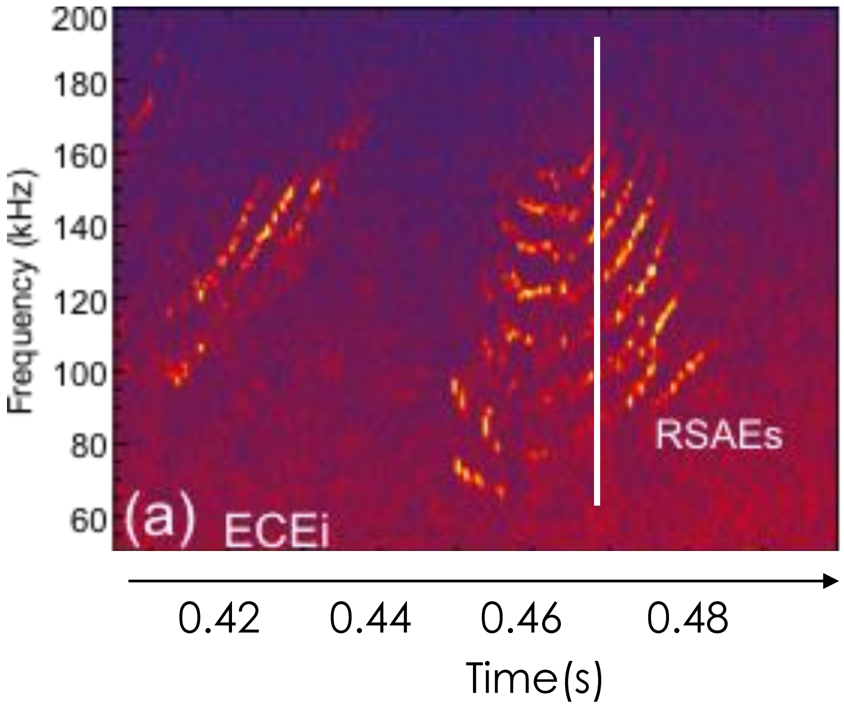




FIDA Measures a Drop in Central Fast-Ion Population as q_{min} passes through an integer (*M. Garcia-Munoz*)





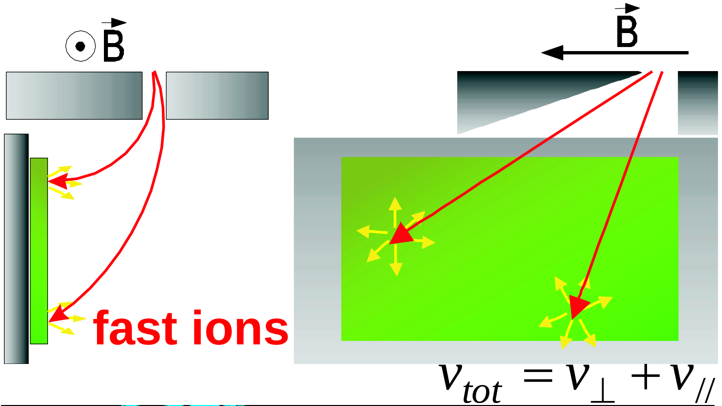


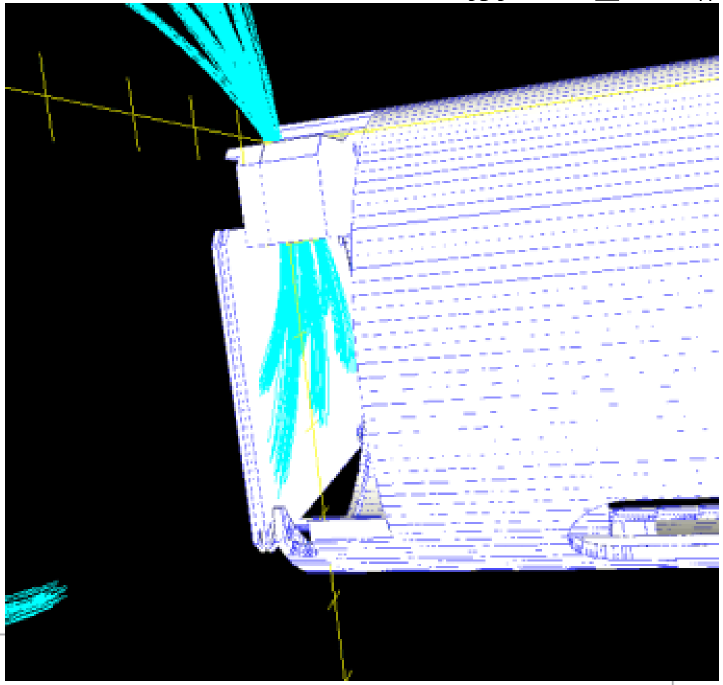
FILD; Energy and pitch angle resolution

• FILD will give absolute fluxes at 3 positions

• Time-resolved Energy and Pitch Angle measurement to understand loss mechanism and better estimate total heat load

M. Garcia-Munoz et al, RSI **80** 053003 (2009)



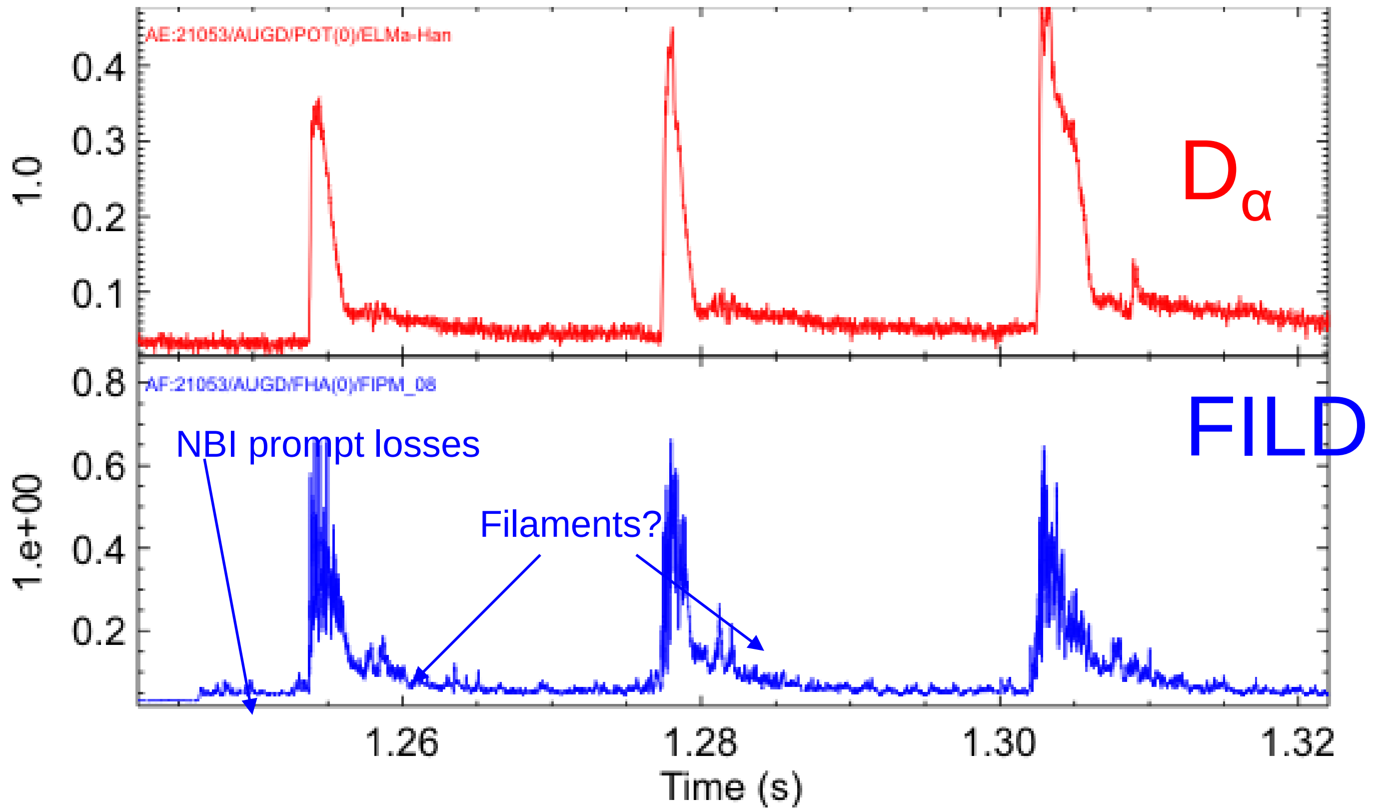


• The strike points of the ions on the scintillator plate depend on their gyroradius and pitch angle (~magnetic spectrometer)

• Novel scintillator material ($SrGa_2S_4$: Eu^{2+}) with high efficiency and short decay time

Edge Localized Mode Induced Fast-Ion Losses

Overall ELM induced fast-ion losses appear correlated with D_α signal with some fluctuating losses within ELM



ELM induced fast-ion losses up to 15 x NBI prompt losses