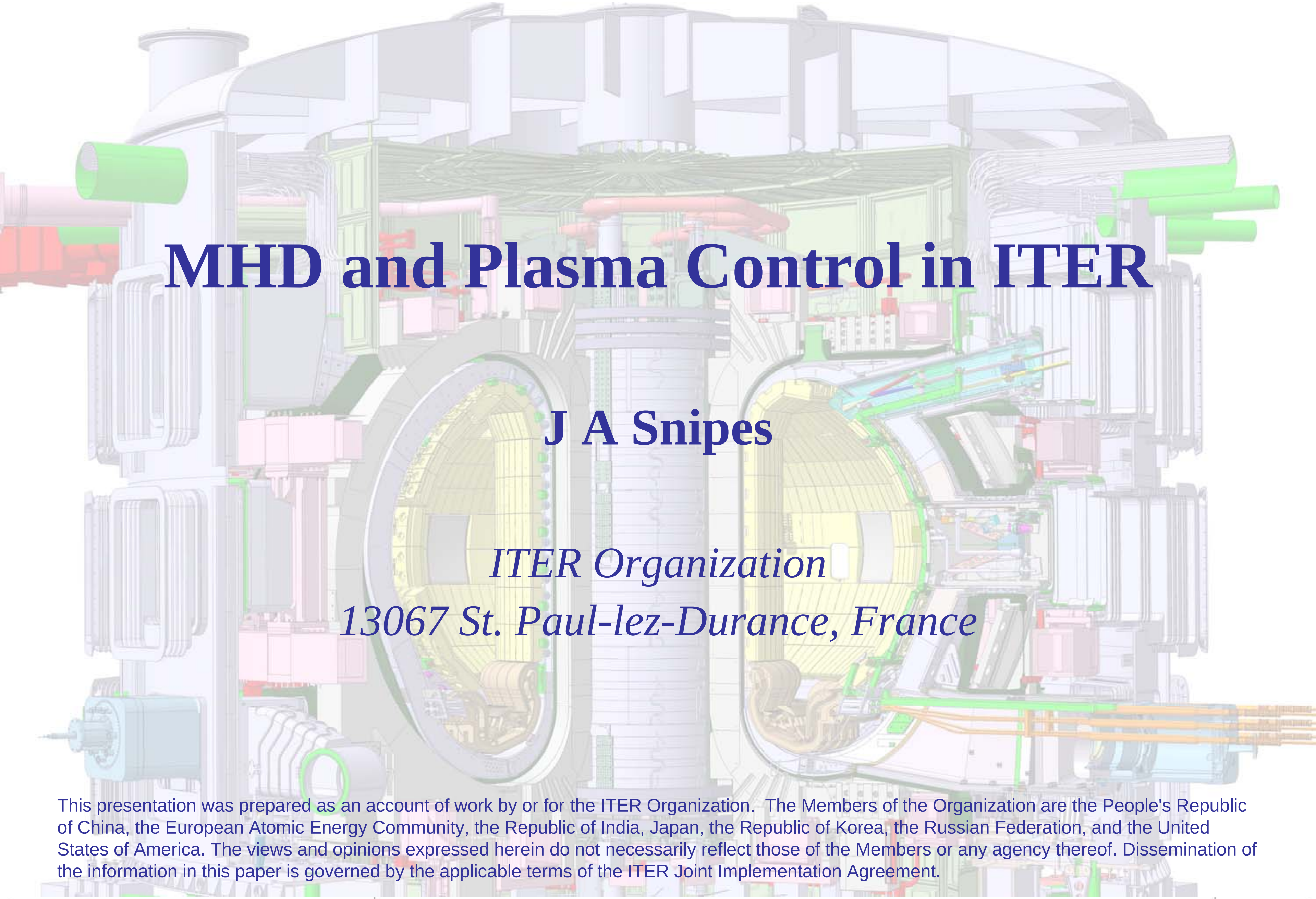


MHD and Plasma Control in ITER

J A Snipes

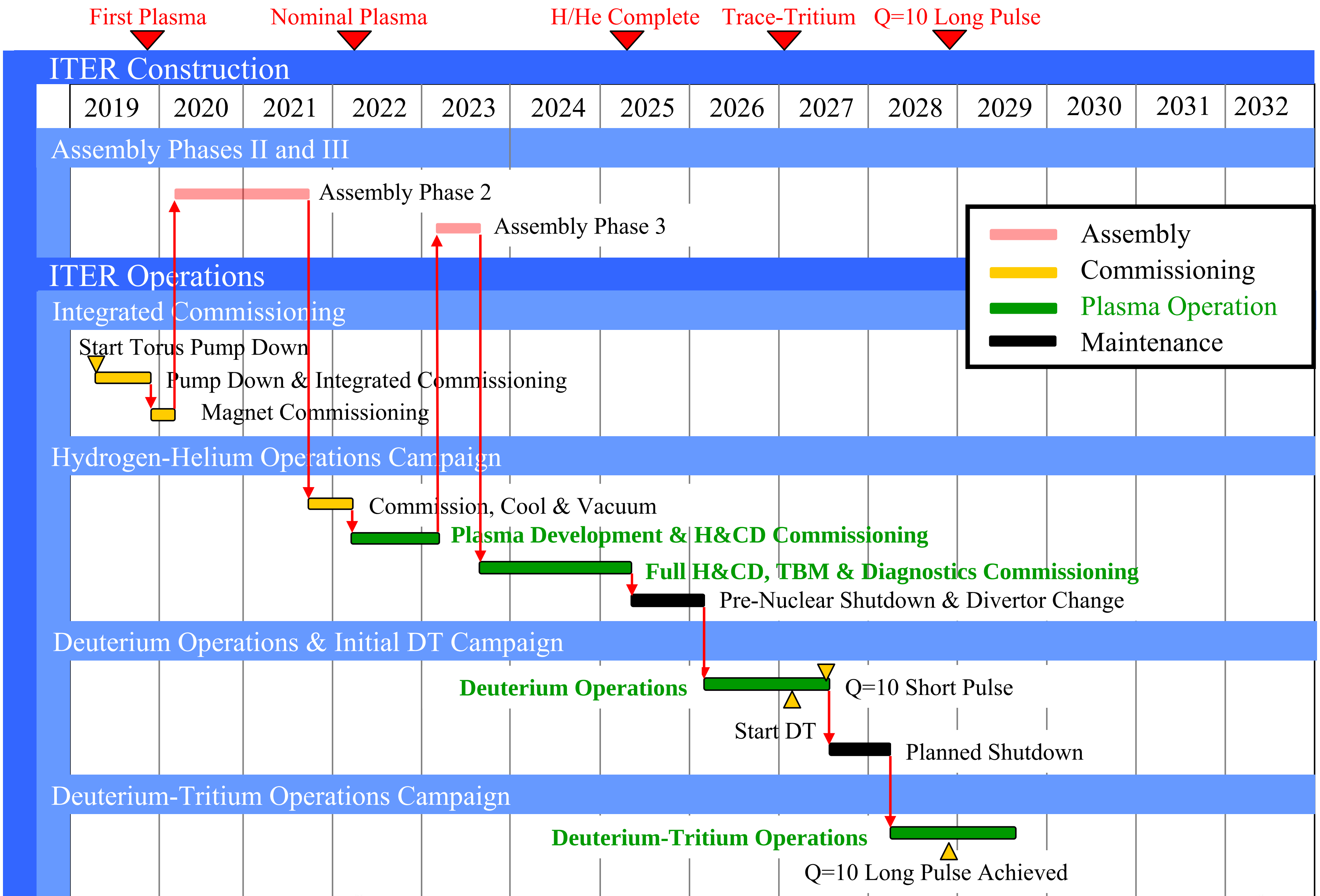
ITER Organization

13067 St. Paul-lez-Durance, France

Outline

- ITER experimental program schedule
- ITER Plasma Control System (PCS) description
- ITER operational scenarios
- Plasma control subsystems
    - Wall conditioning and tritium removal
    - Axisymmetric magnetic control
    - Kinetic control
    - Non-axisymmetric control – MHD instabilities and error fields
    - Event handling – disruptions
- Conclusion

ITER Experimental Program Schedule



Plasma Control System has Five Subsystems

The ITER Plasma Control System (PCS) has five subsystems:

- Some types of wall conditioning and tritium removal
- Plasma axisymmetric magnetic control: plasma initiation, plasma current, position, and shape
- Plasma kinetic control: power and particle flux to the divertor and first wall, fuelling, non-inductive plasma current, plasma pressure & fusion burn
- Non-axisymmetric control: sawtooth, neoclassical tearing mode (NTM), edge localized mode (ELM), Alfven eigenmode (AE), error field and resistive wall mode (RWM)
- Event handling: adaptive control to changing plasma and plant system conditions including disruption mitigation

PCS Must Navigate Within Plasma Operational Limits

Extensive R&D ➔ various stable plasma operational limits:

- current limit: edge plasma safety factor, q ($\propto a^2B_\phi/RI_p$) > 2, q = dφ/dθ = path of magnetic field lines around the torus, field lines close on themselves when q=m/n for integer m,n
- equilibrium limit(s): operating space q and ℓ_i (internal inductance)
- elongation limit: maximum elongation, κ, depends on plasma equilibrium & inductive coupling to the tokamak
- density/ radiation limit(s): maximum density/ radiation level depends on confinement regime
- pressure limit(s): β (= kinetic/magnetic pressure ∝ p/B^2), limited by various MHD instabilities

Plasma control system steers in operating space within these limits to ensure good confinement and high fusion power

Operational Sequence Changes in Real-Time

- Pre-programmed sequence and segment switching + real-time changes in operational sequence in response to faults or conditions

- Heating system fault during a pulse ➔PCS changes operational sequence to a backup experiment to save valuable plasma time

- Real-time integrated plasma modeling used to adjust plasma parameters based on expectations of the modeling

- Adaptive control algorithms use a database of previous plasma conditions to change the control scheme in real-time to achieve desired results (improve performance, avoid disruptions!)

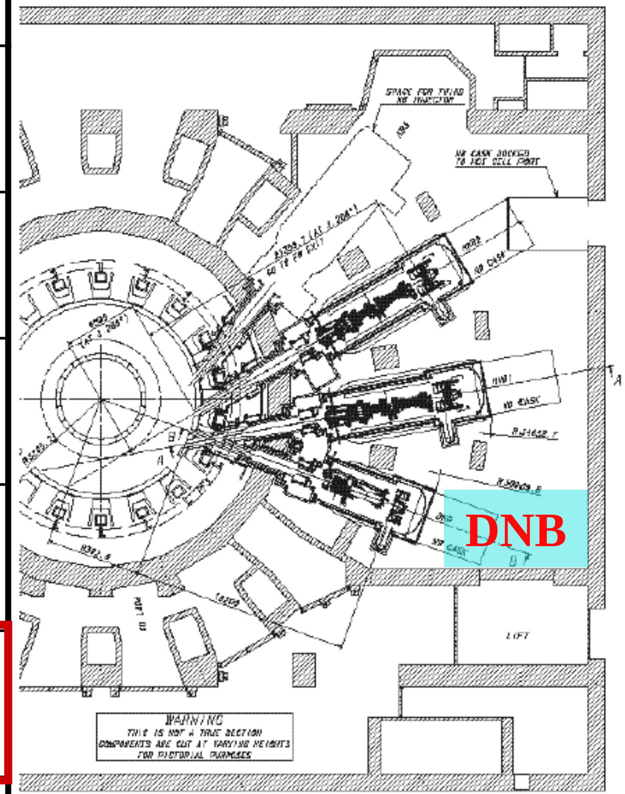
PCS Requires Multiple Actuators

- **Wall conditioning and tritium removal control** requires ion cyclotron (IC), electron cyclotron (EC), & high frequency glow discharge cleaning (HFGDC))
- **Plasma axisymmetric magnetic control** requires Central Solenoid (CS), Poloidal Field (PF), and internal Vertical Stability (VS) coils & power supplies
- **Plasma kinetic control** requires heating and current drive H&CD (IC, EC, & neutral beam injection (NBI)), Ar, Ne, H, D, & T gas and pellet injection, real-time pumping & strike point control
- **Non-axisymmetric control** requires H&CD systems, ELM coils and pellet pacing, gas and pellet fuelling, shape control, & external correction coils
- **Event handling** requires axisymmetric magnetic control & disruption mitigation

Heating & Current Drive Actuators

Heating System	Baseline (MW)	Possible Upgrades (MW)
NBI (1 MeV neg ion)	33	16.5
ECH&CD (170 GHz)	20	20
ICH&CD (40 – 55 MHz)	20	
LHH&CD (5 GHz)		20
Total	73	130 (max installed) (110 simultaneous)
ECH Startup (170 GHz)	> 2	
DNB (100 keV, H)	> 2	

NBI Layout

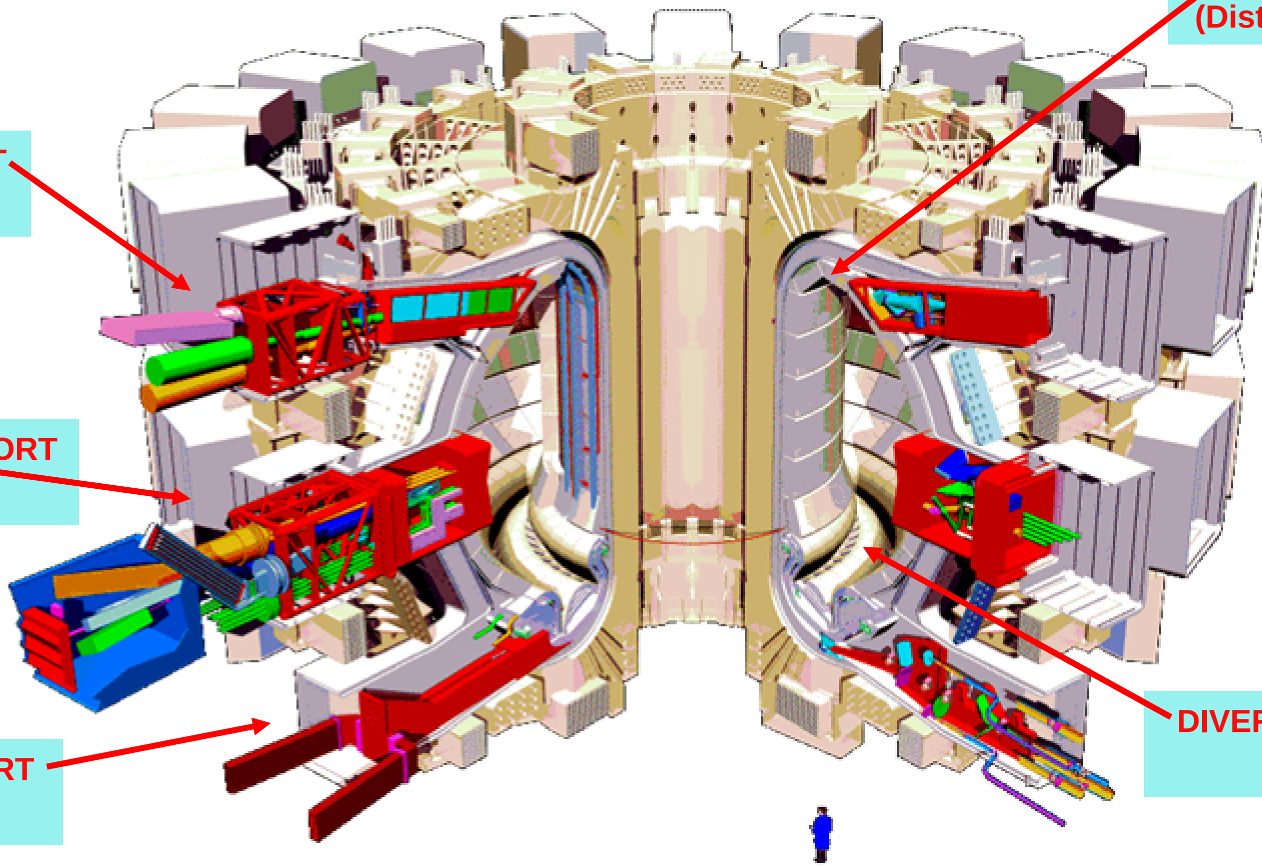


P_{aux} for Q=10 nominal scenario: 40-50MW

PCS Requires Measurements for Control

- **Wall conditioning and tritium removal** requires residual gas species and partial pressures on timescales of minutes and hours
- **Plasma axisymmetric magnetic control** requires neutral pressure, impurity radiation, stray fields, plasma current & position, poloidal field & flux, coil currents, toroidal field, and vessel eddy currents
- **Plasma kinetic control** requires particle flux and heat load on the first wall and divertor, impurity content, radiated power, D_α emission, neutral pressure, core and divertor helium content, electron, ion, and impurity densities, core DT mix, temperature & current density profiles
- **Non-axisymmetric control** requires measurements of sawteeth, ELMs, NTMs, error field characterization, RWMs, plasma rotation, and Alfvén eigenmodes
- **Event handling** requires measurements of plant system status, high first wall and divertor heat load, oscillating and locked modes, and runaway electrons

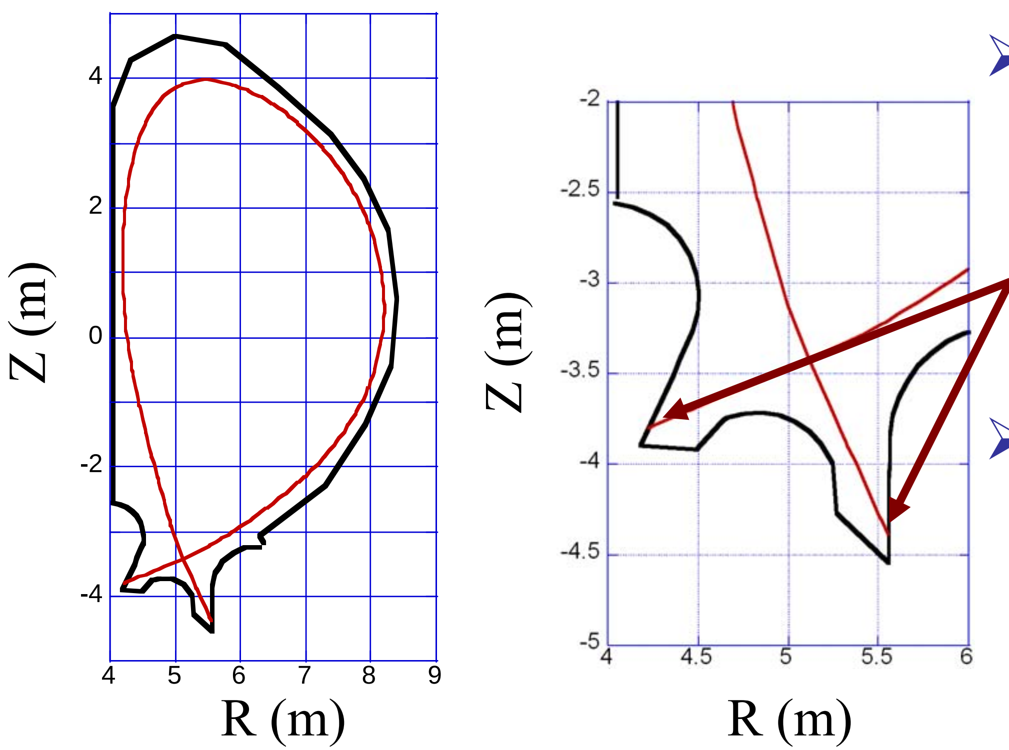
Analyzing the Plasma - ITER Diagnostics



- **About 50 large scale diagnostic systems are foreseen:**
  - Diagnostics required for protection, control and physics studies
  - Measurements from DC to γ-rays, neutrons, α-particles, plasma species
  - Diagnostic Neutral Beam for active spectroscopy (CXRS, MSE ….)

PCS is Designed for Three Reference Scenarios

Nominal 15 MA target separatrix
Q = 10 D-T plasma



- Control requirements apply over timescales from quasi-stationary to rapid (~1ms) disturbances
- Magnetic control based on 15MA target separatrix to limit first wall quasi-stationary heat loads & maintain divertor strike point locations
- PCS designed for three reference scenarios:
  - Inductive operation: Q=10, 15 MA, 500 MW
  - Hybrid operation
  - Non-inductive operation

ITER Scenarios

- **Baseline scenarios:**

Single confinement barrier

- ELMy H-mode:
  - Q=10 for ≥300s
  - well understood physics extrapolation to:
    - control
    - self-heating
    - α-particle physics
    - divertor/ PSI issues
  - physics-technology integration
- Hybrid:
  - Q=5 - 50 for 100 - 2000s
  - conservative scenario for technology testing
  - performance projection based on extension of ELMy H-mode

- **Advanced scenarios:**

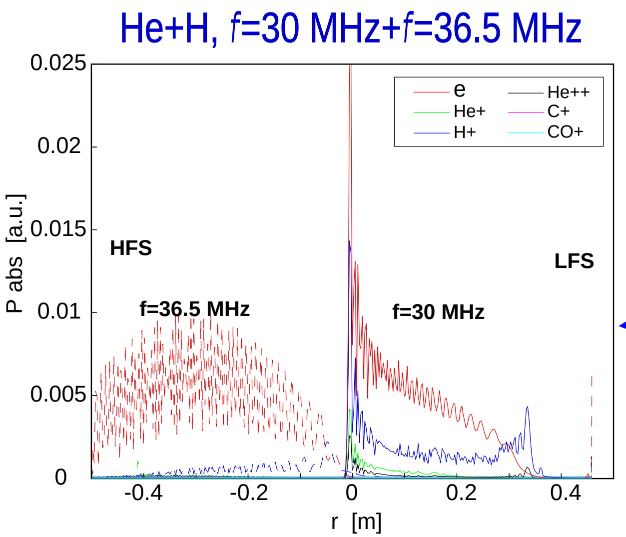
Multiple confinement barriers

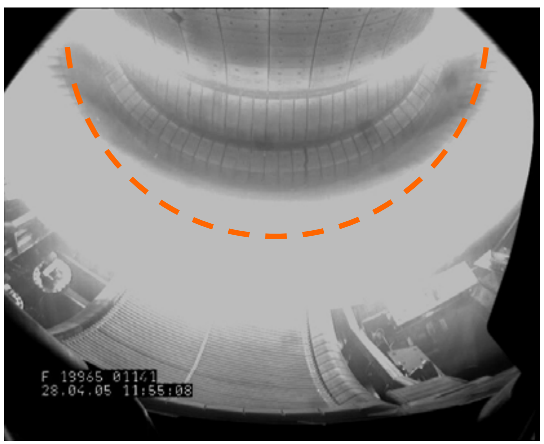
  - satisfy steady-state objective
  - prepare DEMO
  - develop physics in a range of scenarios:
    - extrapolation of regime
    - self-consistent equilibria
    - MHD stability
    - controllability
    - divertor/ impurity compatibility
    - satisfactory α-particle confinement

Litaudon: Tuesday AM

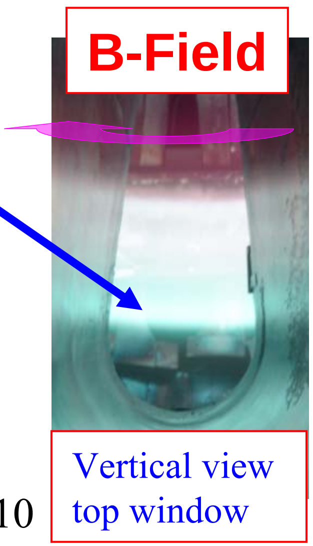
Wall Conditioning and Tritium Removal Subsystem

A. Lyssoivan, 18th PSI 2008





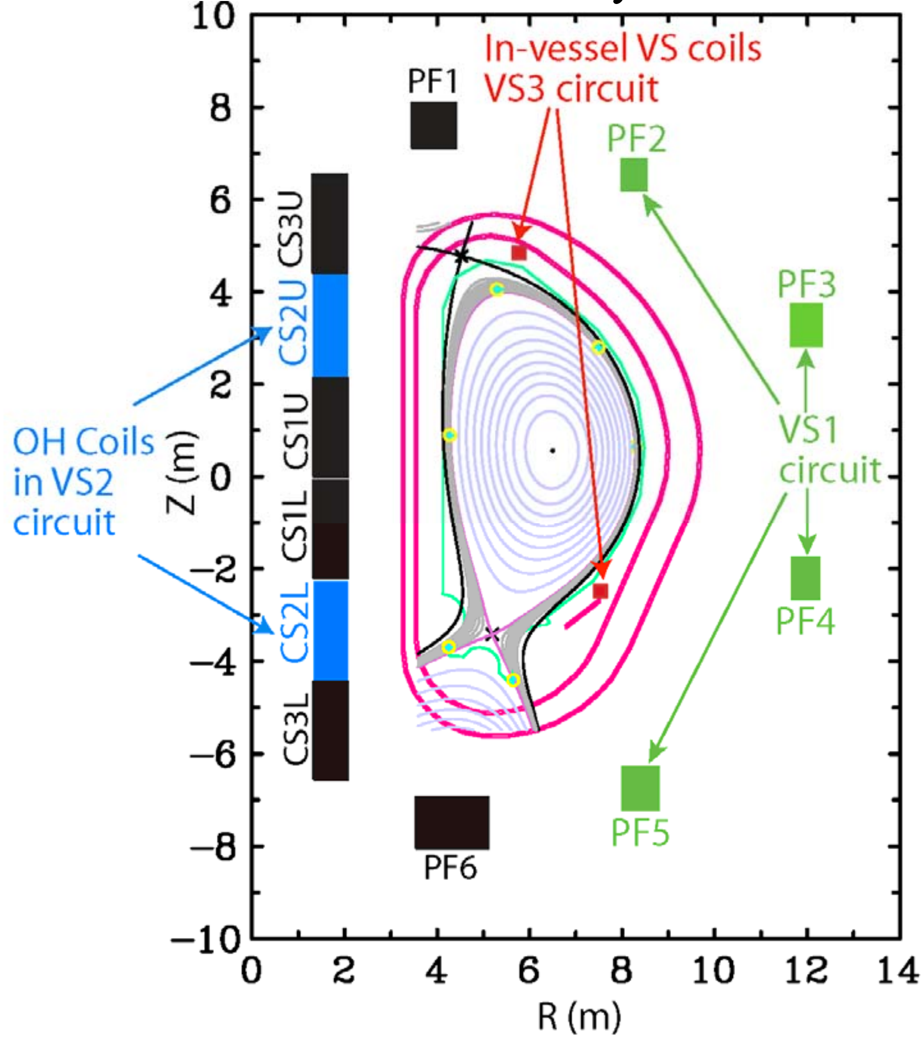
- PCS will control plasma wall conditioning(WC) during the TF including PF control
  - for D and DT plasmas to reduce adsorbed H isotopes from the first wall
  - ICWC and possibly ECWC techniques
  - homogeneous ICWC on AUG with dual frequencies, He+H, & vertical field
  - High frequency glow discharge cleaning with toroidal field
  - 20 – 100 kHz HFGDC with B_T demonstrated on EAST with stable uniform glow toroidally, over wide range of pressure
  - removal rates similar to ICWC



X Gong, J Li, PSI 2010

Axisymmetric Magnetic Control Subsystem

Magnetic Actuators Showing Three Vertical Stability Circuits



- Includes plasma initiation, inductive plasma current, position, and shape control
- PCS will control currents in CS, PF, and VS magnets, but not TF
- Plasma initiation will include several MW of startup ECH
- Inductive plasma current, shape, and radial position control will have a settling time of ~ 5 s
- Vertical position control with VS1+VS3 coils will have a settling time ~ 0.1 s
- VS2 possible backup system

de Tommasi: Wednesday

Vertical Position Control Based on VS1+VS3 Circuit

- Baseline system for stabilizing plasma vertical displacements (ΔZ) (VS1+VS3) capable of restoring the plasma vertical position after a maximum uncontrolled vertical drift ~ 16 cm for $\ell_i < 1.2$

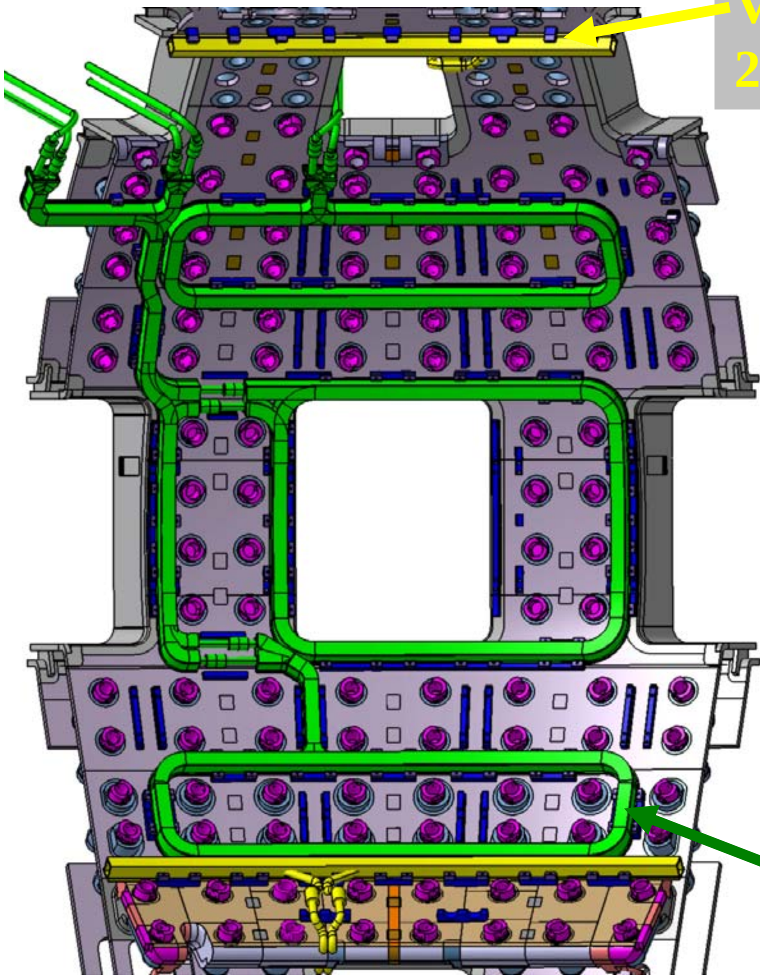
- Assumed dZ/dt RMS noise ~ 0.6 m/s with 1 kHz bandwidth

- Timescales > vacuum vessel radial field penetration time (~0.2 s)

- If VS3 fails, possible backup: VS1 up to 9 kV & VS2 up to 6 kV VS1+VS2 alone capable of vertical position control after a maximum uncontrolled vertical drift given by:

$$Z_0(cm) = 160\, e^{-3.7\ell_i(3)} + 1.8$$

Magnetic Actuators Include In-Vessel Coils



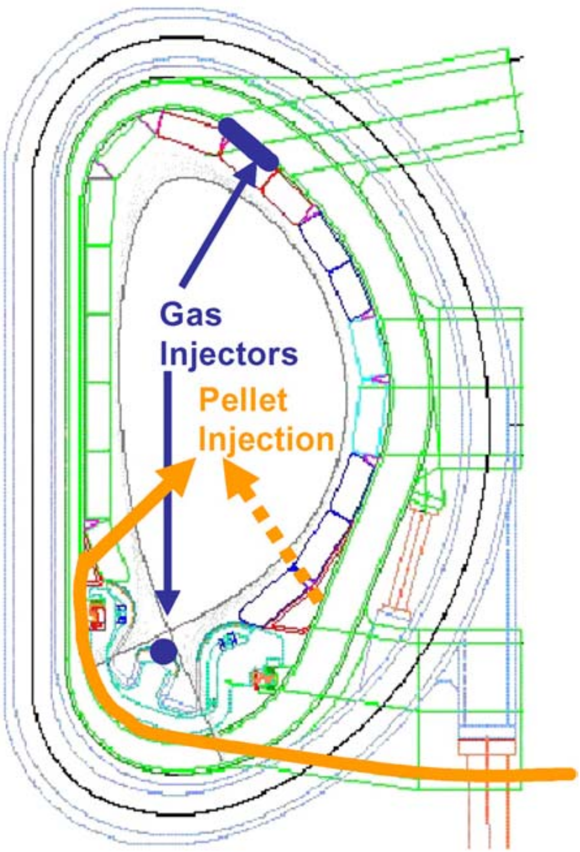
VS coils (4 turns, 40 kAturns RMS, 240 kAturns peak, 2.3 kV)

- **A set of in-vessel resonant magnetic perturbation (ELM) and vertical stability (VS) coils is being designed:**
  - 9 toroidal × 3 poloidal array on outboard internal vessel wall
  - vertical stabilization coils upper & lower loops form a saddle coil

ELM coils (3 sets of 9 coils)
6 turns up to 90 kAturns

Plasma Kinetic Control Subsystem

Fuelling control actuators



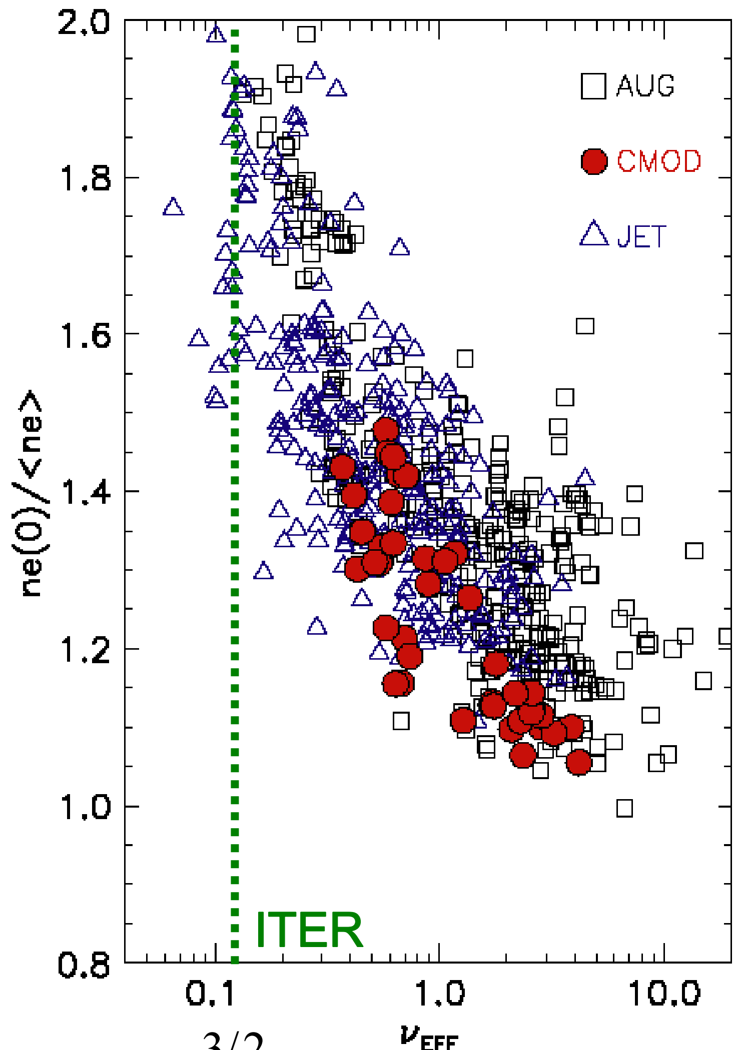
Baylor, NF 2007

- Plasma kinetic control includes power and particle flux, fuelling, heating and current drive, plasma pressure and fusion burn control
- Power and particle flux control: first wall & divertor protection and MARFE (edge radiative instability)
- Fuelling control: main ion species mix, electron density, and injected impurity density
- Impurity density control: Ne/Ar and helium ash
- Heating & current drive power and deposition
- Current density profile control for hybrid and long pulse steady-state scenarios for $q_{min} > 1$ or $q_{min} > 2$

Kikuchi: Monday AM

What Will Core Fuelling be Like in ITER?

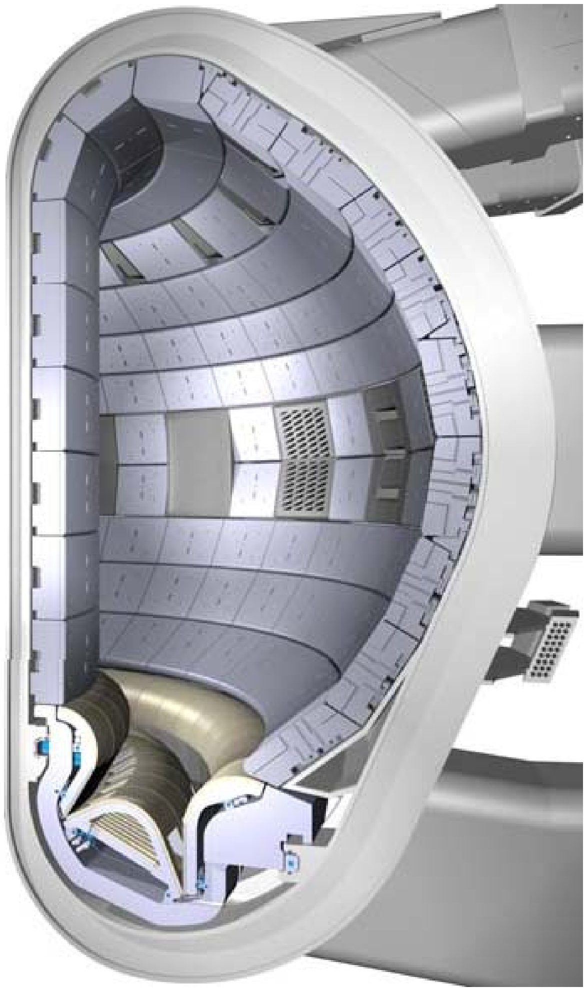
- Present cryopump design limit: $\Gamma_{pump} = 200$ Pa-m^3/s
- Expected recycling flux: $100 \times \Gamma_{pump}$
- Expect low central gas fuelling
  ➔ flat density profiles
- Inward pinch at low ν* may lead to density peaking in ITER
- Could increase fusion reactivity
- But profile peakedness must be carefully controlled to avoid He ash and other impurity peaking



$\nu_{eff} \sim n_e / T_e^{3/2}$

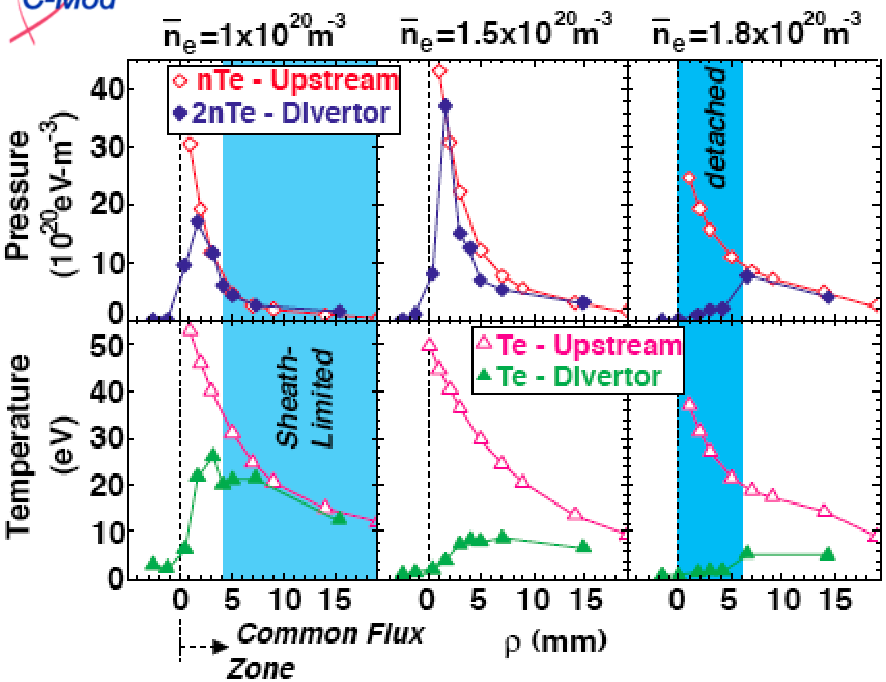
Greenwald, NF 2007

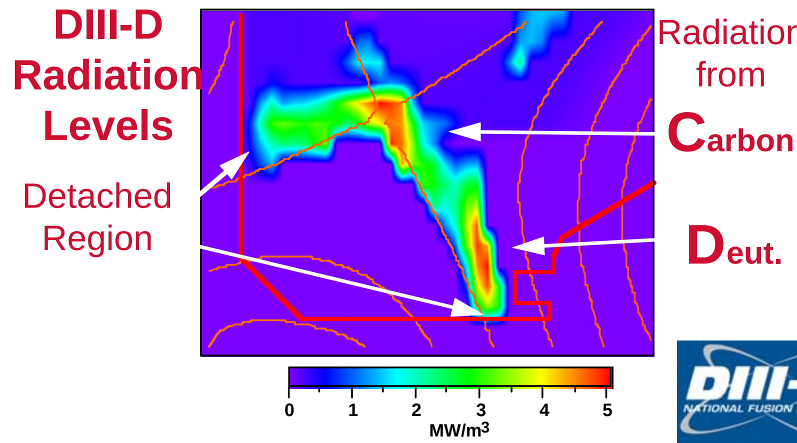
Power and Particle Flux Control is Essential

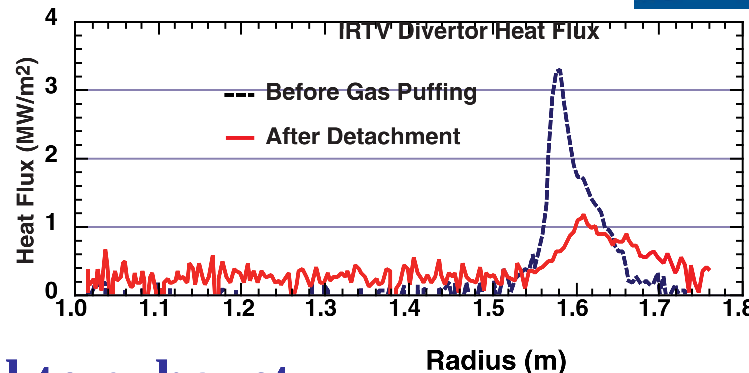


- Power and particle flux control to the first wall and divertor is essential to avoid damage and excessive impurity influxes
- Divertor melting can occur quickly (~1 s) at full performance
- Divertor detachment control with Ne/Ar puffing avoids excessive divertor heat load
- MARFE control will be required at high density to maintain good confinement
- Unmitigated ELM and disruption heat loads will severely limit the divertor lifetime
- Fusion performance requires core helium ash control with divertor cryopumping, strikepoint position, and H&CD profile control

Power Exhaust Control Through Divertor Detachment





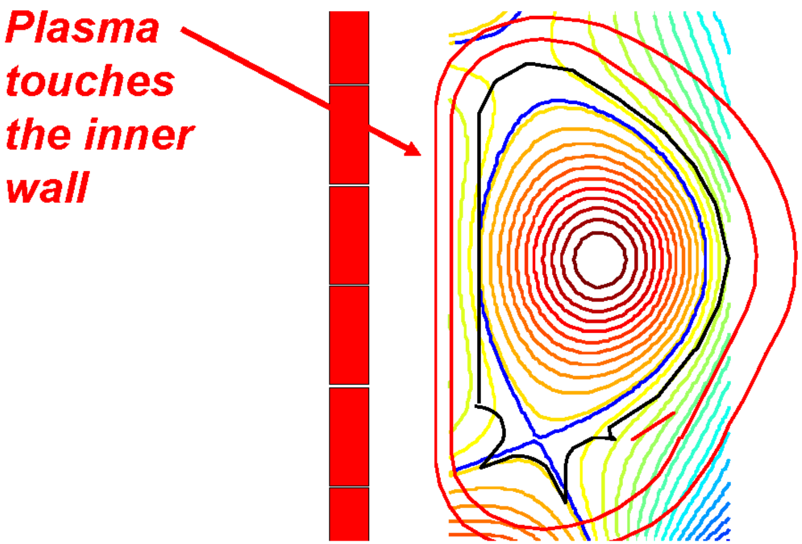


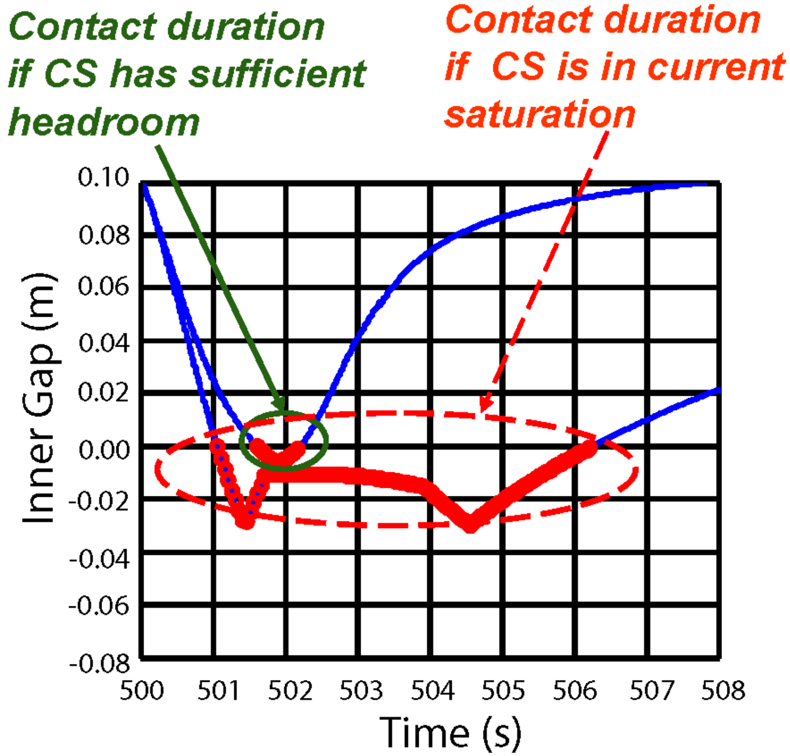
Divertor "detachment" is fundamental to exhaust power in a burning plasma environment:

- large pressure gradient develops along field lines into the divertor
- at high density, divertor plasma temperature falls to a few eV
- large fraction of plasma exhaust power is redistributed by radiation from impurities injected into the divertor and ion-neutral collisions

ITER PCS is Critical to Avoid Melting First Wall

Modeling of an H-mode to L-mode Transition at Q=10 with 15 MA

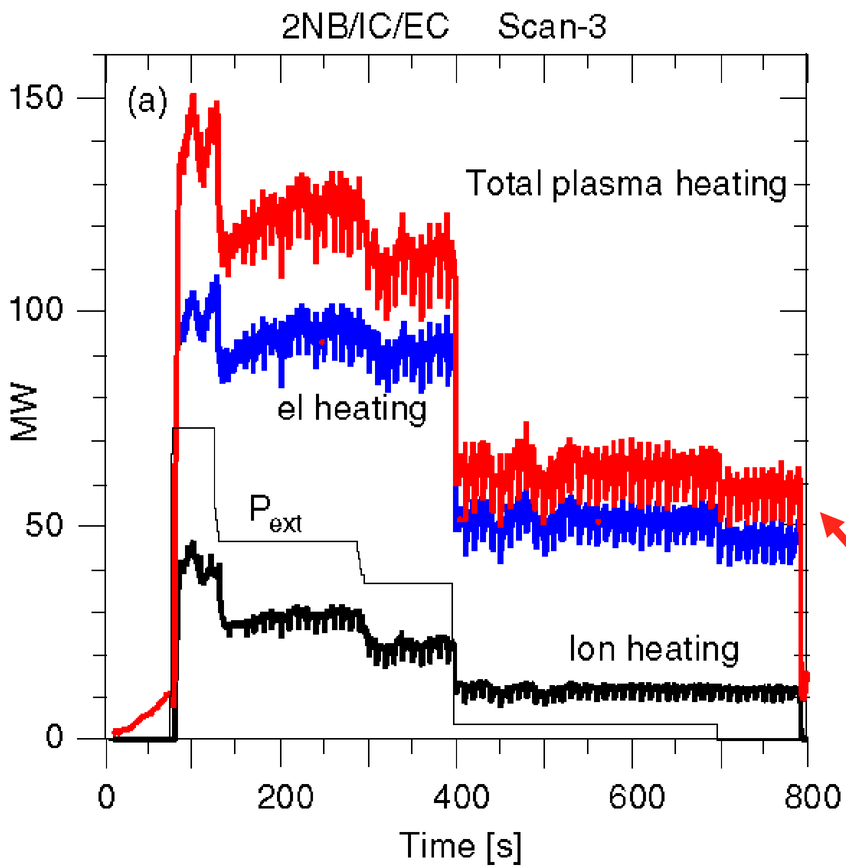




- Radial inward displacement can be ≥ 10cm ➔ contact with the inner wall
- Duration of inner wall contact depends on the central solenoid saturation state
- Peak engineering heat loads of ~40MW/m^2 ➔ Be tiles would melt in ~ 0.3 s!
- PCS must maintain large enough gaps or trigger the disruption mitigation system

Simulations Show Fusion Burn is Stable in ITER

Simulated Burn Control in ITER



Budny, NF 2009

- Dominant α-particle heating at Q=10 requires reliable fusion burn control schemes controlling the core D/T mix with pellet injection, helium ash, and other core impurities
- Auxiliary heating power may also be used for secondary fusion burn control
- Simulations show that the fusion burn is stable in a 15 MA Q=10 DT ITER plasma

ITER Will Enter New Fusion Burn Control Regime

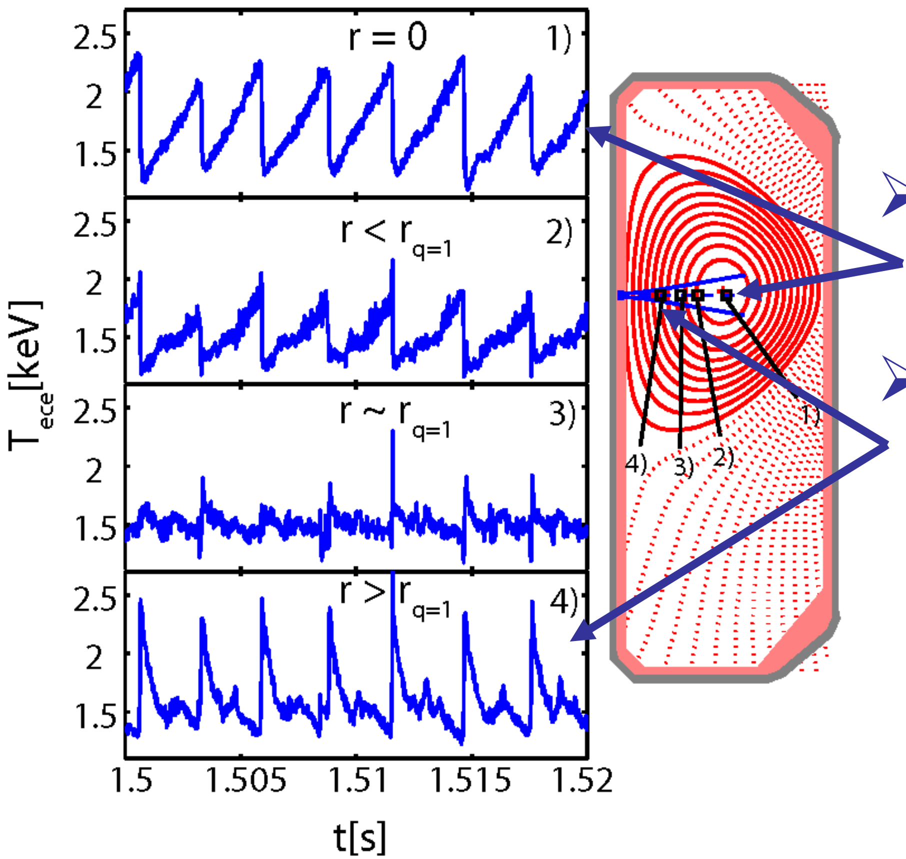
- Novel aspects of burning plasma physics are key to the ITER research program
- α-particle/energetic particle physics:
  - energetic particle confinement at low $\rho^*(= r_L/a \sim (T^{1/2}/B)/a)$, influence of self-heating
  - nonlinearly coupled MHD with Alfvén eigenmodes (AEs)
  - enhanced heat loads with high fusion power
- Burning plasma control scenarios:
  - burn control through D/T mix profile control
  - dominant core pellet fuelling is also a new regime
  - transport barriers and their control (isotope effects in DT?)
  - non-linear interactions between α and auxiliary heating, plasma pressure, rotation and current density profiles
  - can Alfvén eigenmode stability be used for burn control?

Non-Axisymmetric Control Subsystem

- Non-axisymmetric control includes sawtooth, neoclassical tearing mode (NTM), edge localized mode (ELM), Alfvén eigenmode (AE), error field and resistive wall mode (RWM) control
- Sawtooth and NTM control required at high performance with ion cyclotron range of frequency (ICRF) and localized and steerable electron cyclotron current drive (ECCD)
- ELM control critical to reduce divertor erosion with pellet pacing (30 – 50 Hz repetition rate) and in-vessel ELM coils
- Alfvén eigenmode control may be required at high performance for burn control and to avoid enhanced localized fast particle losses
- Error field control is required to avoid locked modes and RWMs
- RWM control upgrade may be required at high β using ELM coils

What are Sawteeth?

T_e at Four Radial Locations in TCV

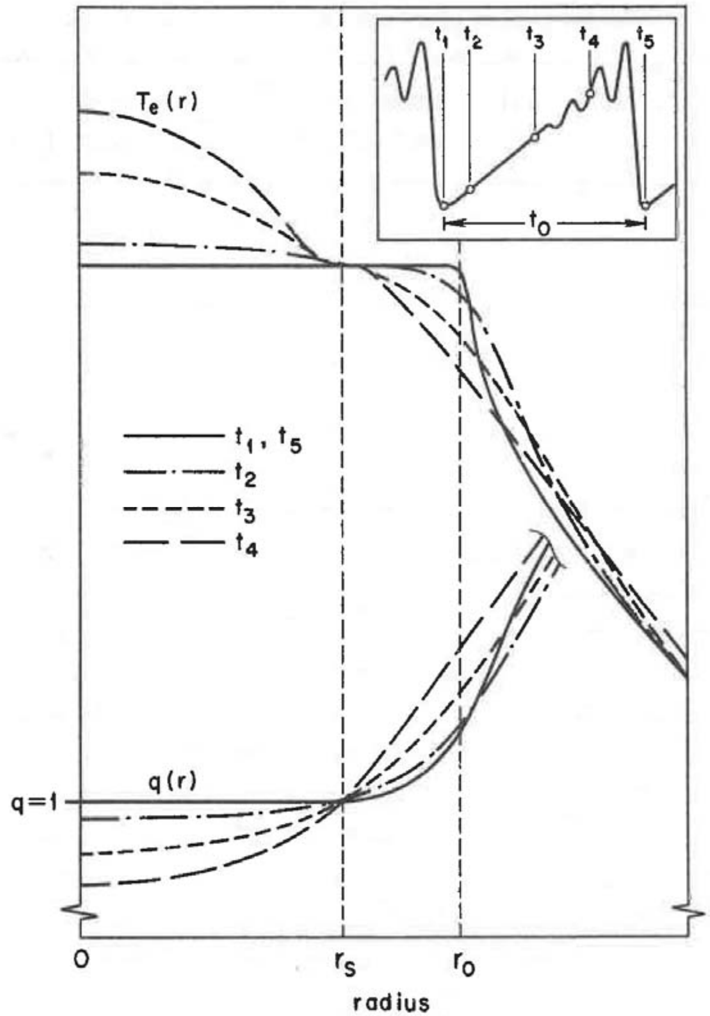


P Blanchard, PhD thesis, EPFL (2002)

- Sawteeth are periodic oscillations in the plasma temperature with a characteristic sawtooth shape
- Slow rise in the core temperature followed by a rapid crash
- Outside the q=1 ($q \sim rB_T/(RB_\theta)$) 'sawtooth inversion' radius, the temperature rises rapidly and then falls slowly

What are Sawteeth?

Model T_e and q Profiles During a Sawtooth



Jahns, et al., *NF* **18** (1978) 735

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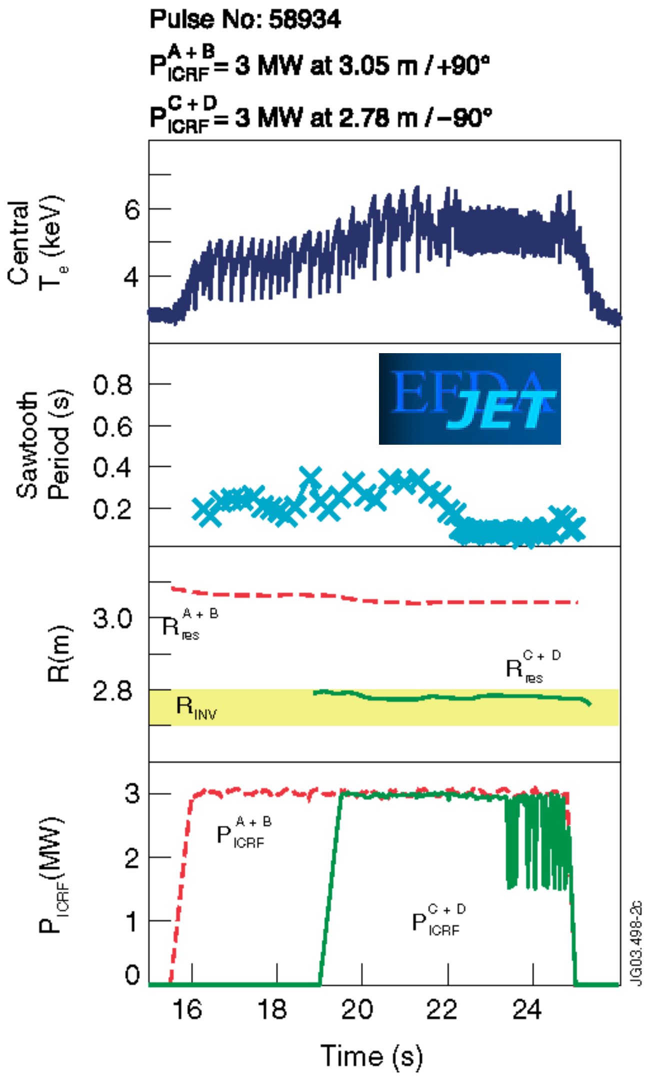
- Slow rise in the core temperature followed by a rapid crash
- Outside the q=1 ($q \sim rB_T/(RB_\theta)$) 'sawtooth inversion' radius, the temperature rises rapidly and then falls slowly
- Model shows how T_e and q profiles change during a sawtooth
- Large sawteeth provide seed islands that could lead to unstable NTMs and reduced confinement

Sawtooth Control Has Been Demonstrated

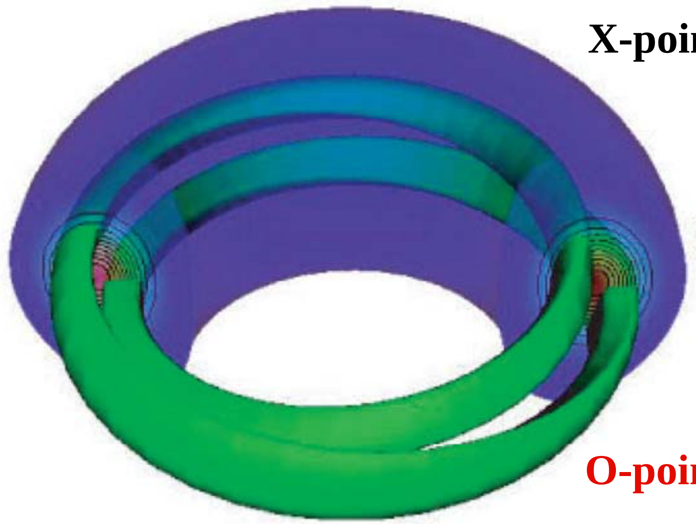


Pamela, et al., *NF* **45** (2005) S63

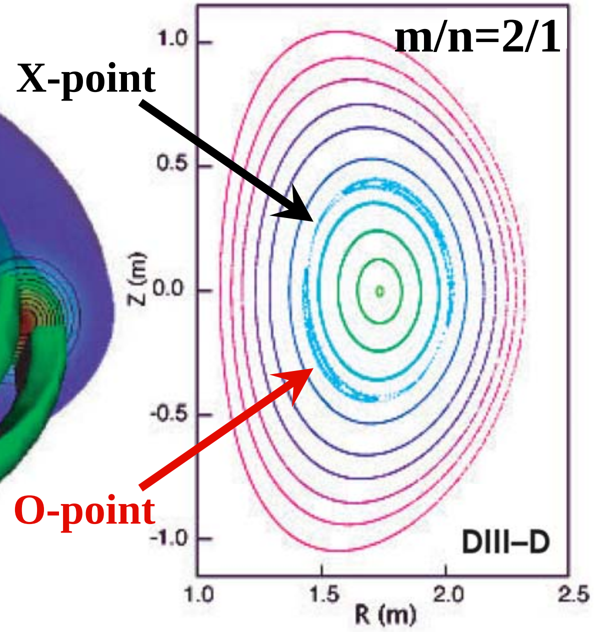
- Sawtooth control was demonstrated on JET with +90° ICRF phasing to create fast ions to partially stabilize sawteeth ➔ 'monster' sawteeth
- Then -90° ICRF phasing was added to destabilize sawteeth reducing the sawtooth period and amplitude
- ITER actuators for sawtooth control include ICRF and localized ECCD near the q=1 surface
- Current drive techniques will also be used to maintain q > 1 for long pulse scenarios to avoid sawteeth

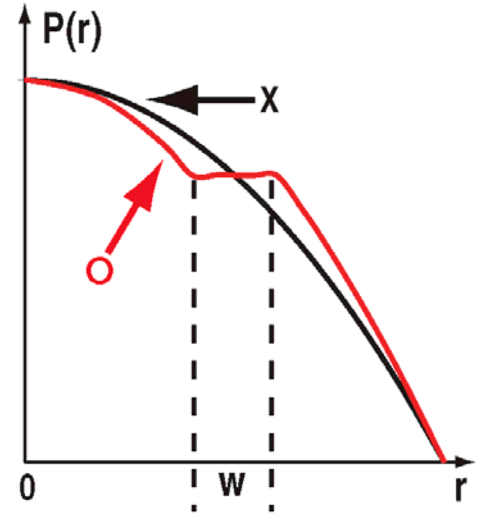
Graves: Thursday PM

What are Neoclassical Tearing Modes?



D.P. Brennan, et. al., 2004





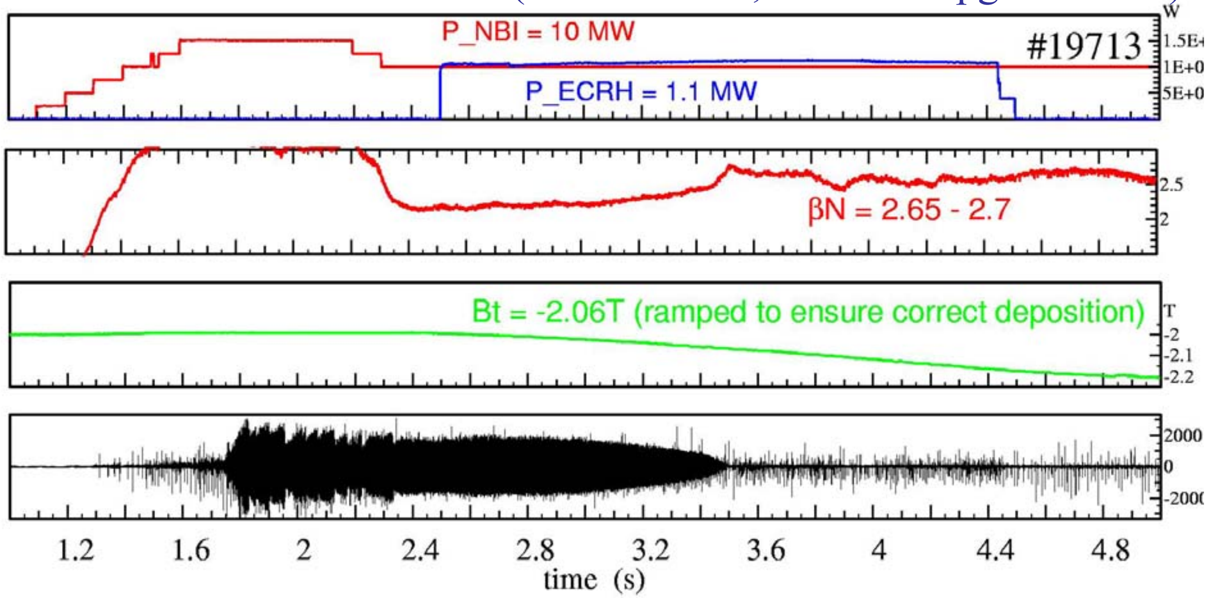
Fitzpatrick: Thursday AM

- Finite plasma resistivity allows toroidally non-axisymmetric helical currents to break or tear magnetic field lines at rational surfaces q = m/n (➔ a tearing mode)
- Field line reconnection creates magnetic islands and rapid energy transport along the field line flattens the pressure profile across the island width W
- Toroidal effects produce a pressure gradient driven bootstrap current $j_{bs} \sim -\frac{\varepsilon^{\frac{1}{2}}}{B_\theta}\frac{dp}{dr}$
- Reduced gradients in the island produce a helically perturbed bootstrap current
- Neoclassical Tearing Modes (NTMs) are excited by seed islands above a critical β

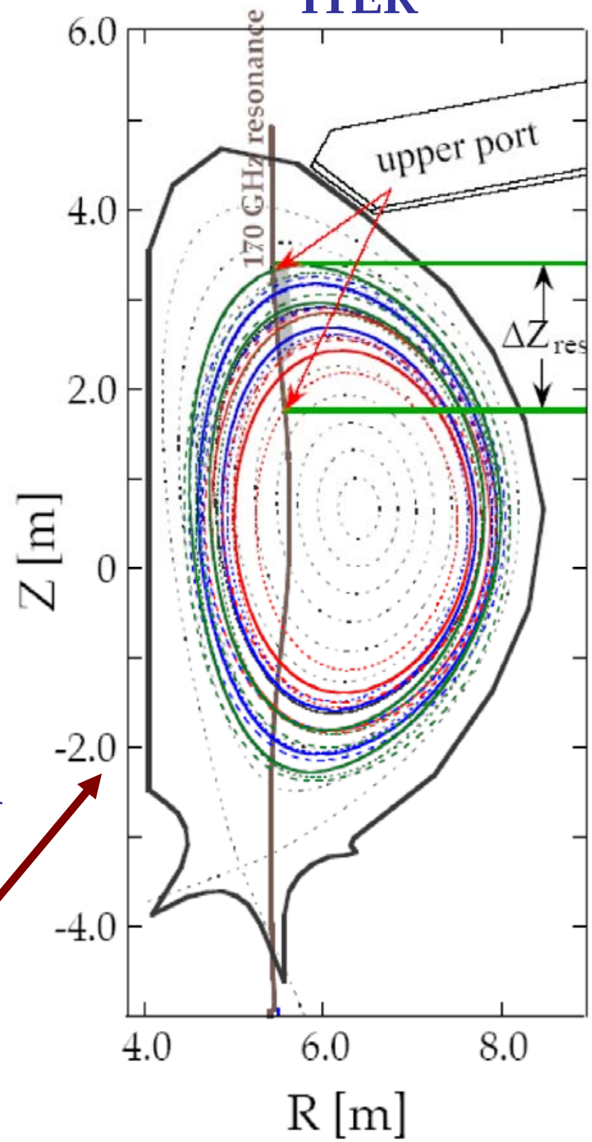
Localized ECCD Controls NTMs

ASDEX Upgrade

(H Zohm et al, ASDEX Upgrade 2006)



ITER

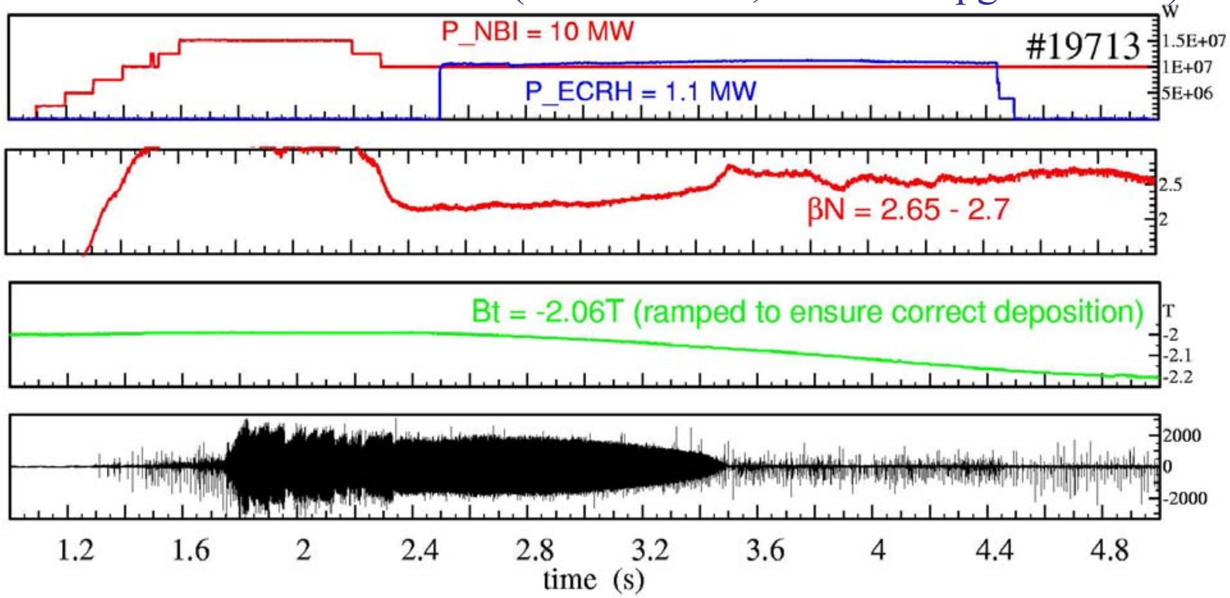


- **Electron cyclotron waves can produce localized current drive inside magnetic island**
  - exploited in present experiments to suppress NTMs
- **ITER: 4 steerable launchers in upper ports injecting 20MW ECCD power in phase with the NTM up to 5 kHz modulation frequency**

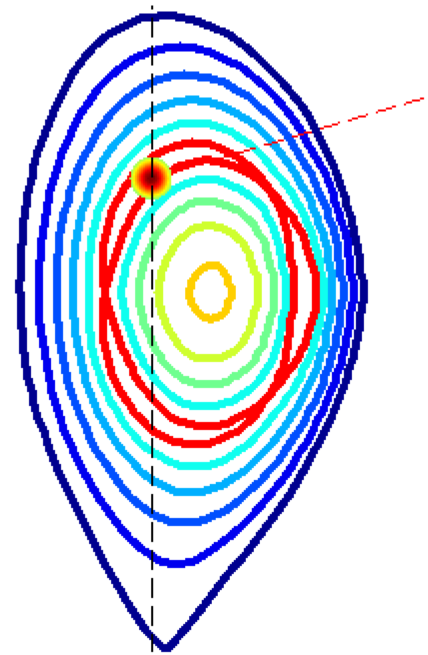
Localized ECCD Controls NTMs

ASDEX Upgrade

(H Zohm et al, ASDEX Upgrade 2006)



ITER



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- ➢ **ITER: 4 steerable launchers in upper ports injecting 20MW ECCD power in phase with the NTM up to 5 kHz modulation frequency**

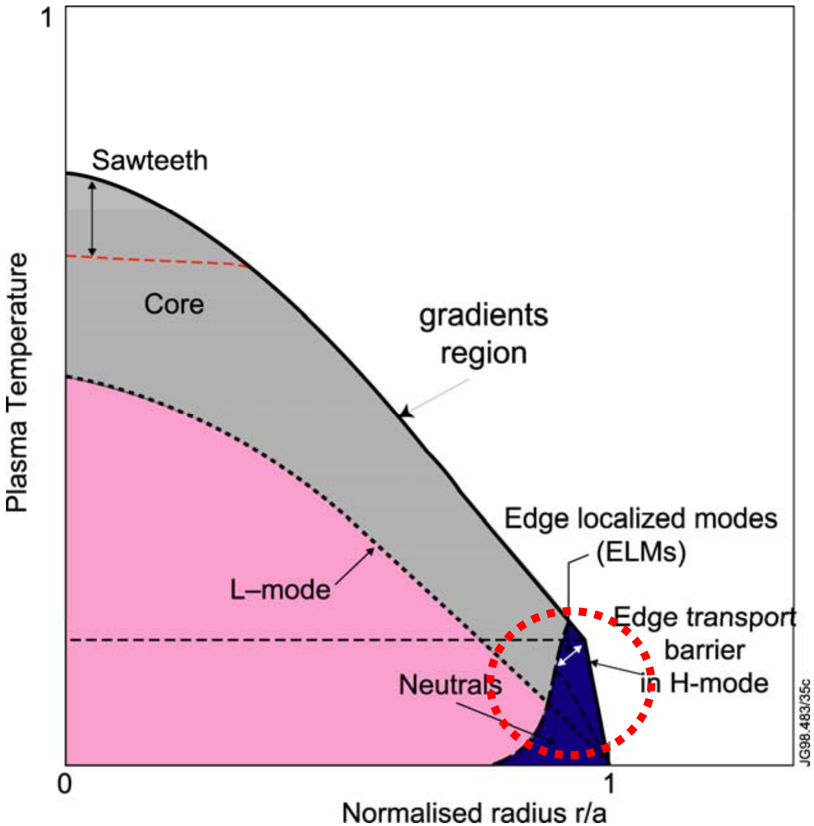
R LaHaye, APS 2005

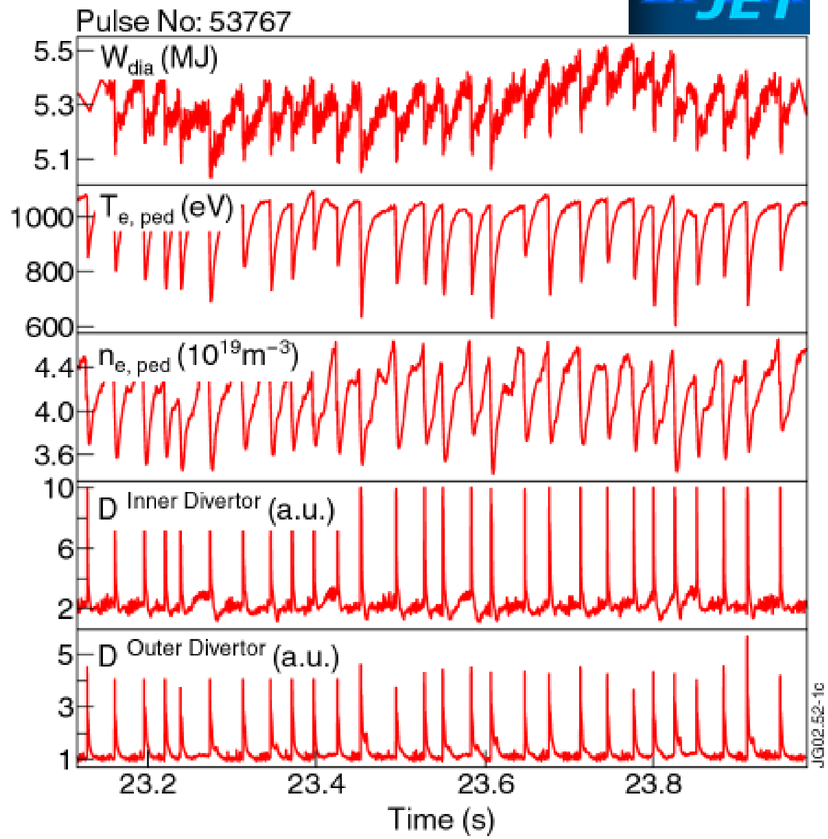
Sen: Thursday PM

What are Edge Localized Modes (ELMs)?

ELMs are rapid disturbances of the edge temperature and density

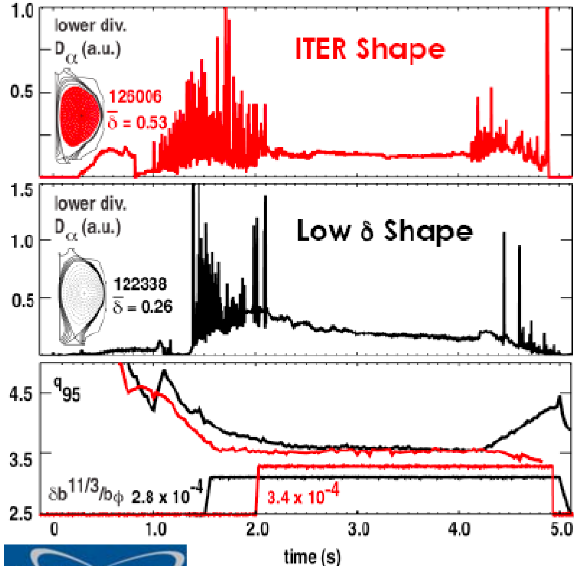
- destabilized when the edge pressure gradient becomes too steep
- yield very high transient heat and particle flux on wall and divertor
- maintain the plasma in a quasi-stationary state

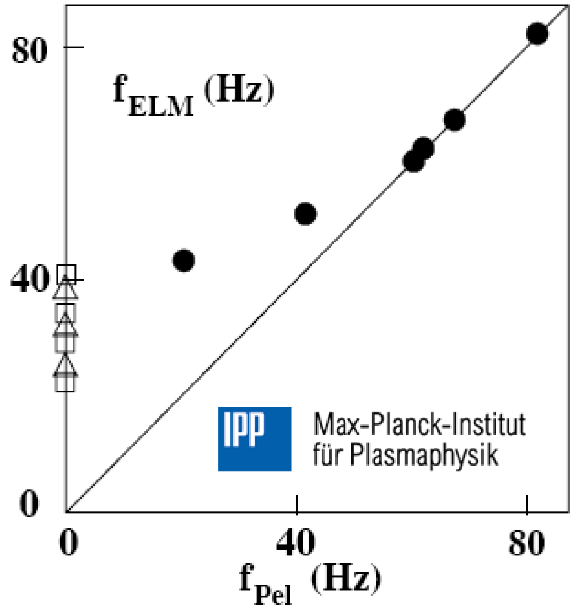


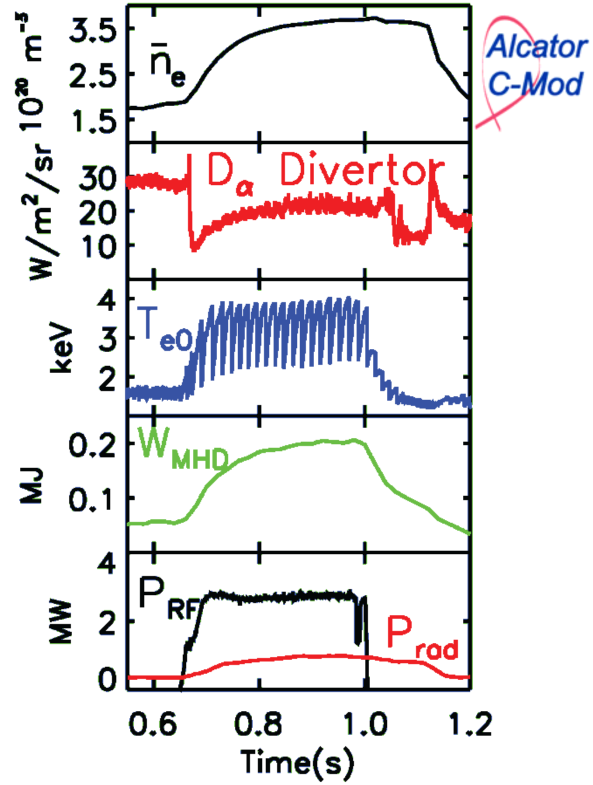


First Wall Heat Load: ELM Control/ Mitigation is Critical

DIII-D Magnetic Control	AUG Pellet Pacemaking	C-Mod EDA H-mode







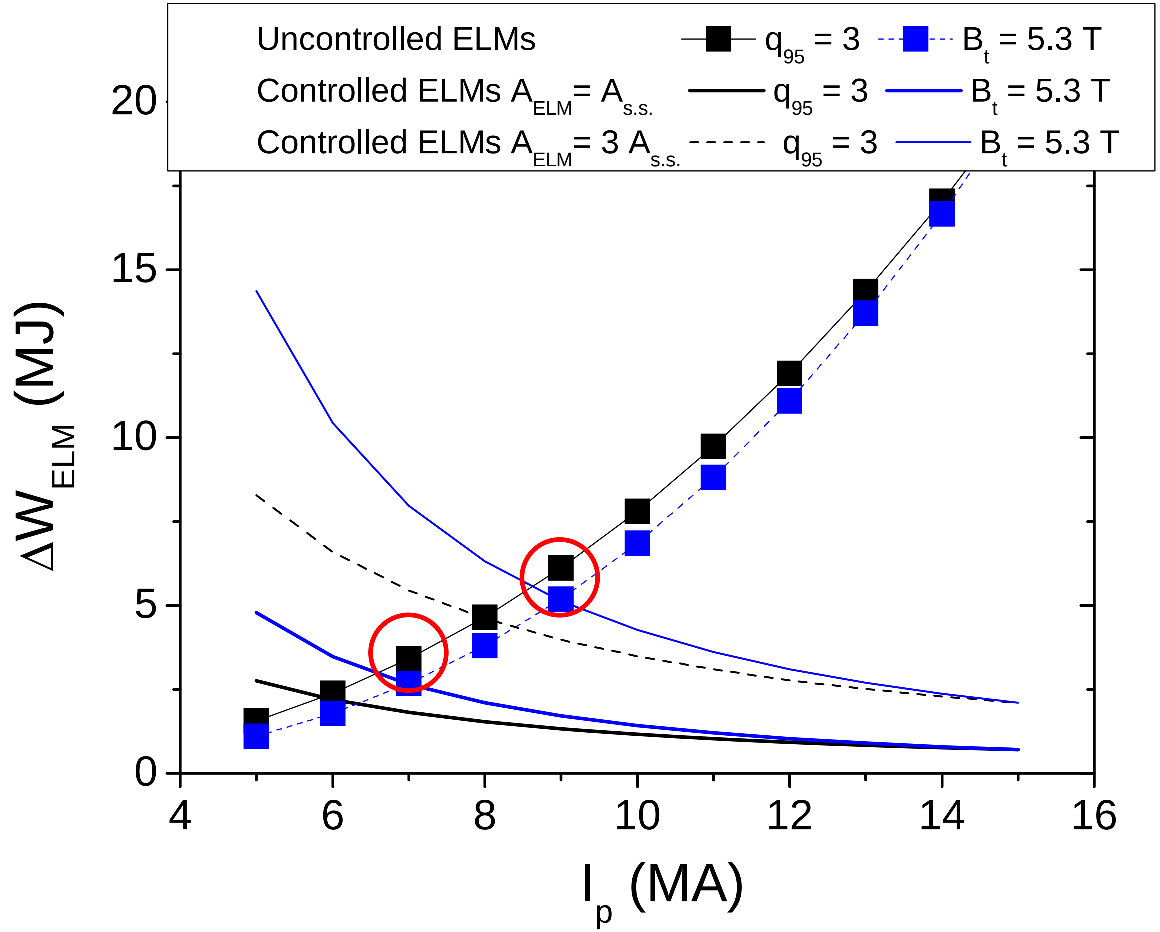
- ELM control is needed to substantially reduce divertor heat loads to enhance the divertor lifetime

Liang: Friday AM

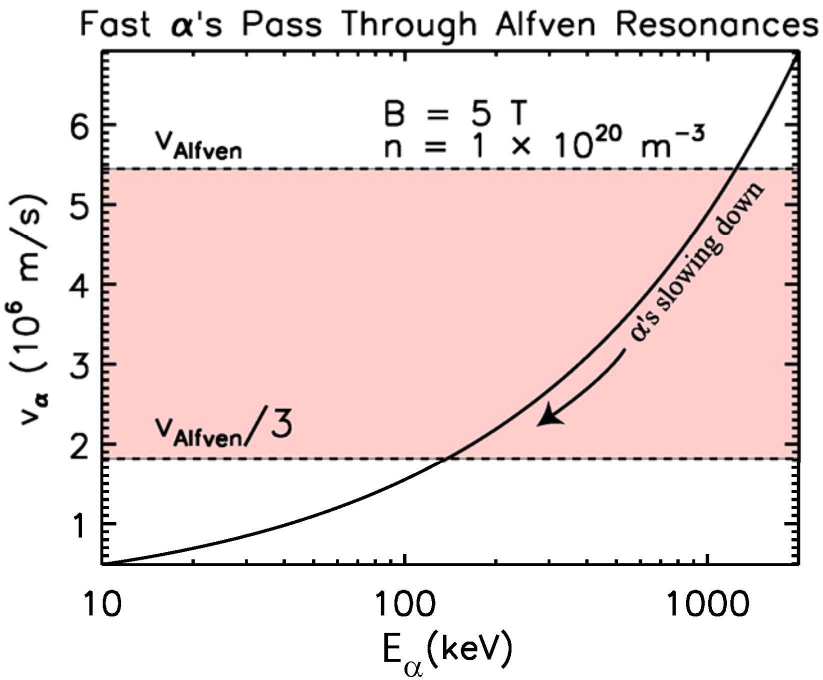
- ITER will use in-vessel ELM coils and pellet pacing for ELM control
- Steady-state ELM-free regimes may also be found on ITER

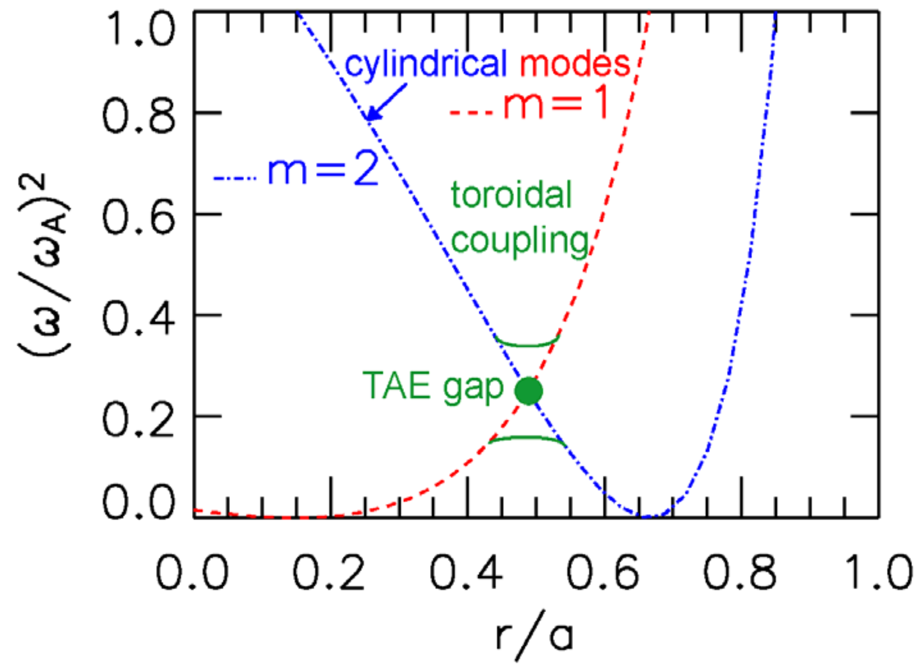
ELM Control Required for High Current Operation

- Operation with uncontrolled ELMs is possible in ITER for I_p < 9 MA
  → ELM control required from H-mode transition (in I_p ramp) through burn and H-L transition for 15 MA Q_{DT} = 10



What are Alfvén Eigenmodes?





- Energetic particles with specific resonances (e.g., v_A, $v_A/3$) e.g., α particles slowing down excite Alfvén modes in gaps in the continuum spectrum where damping is weaker ➔

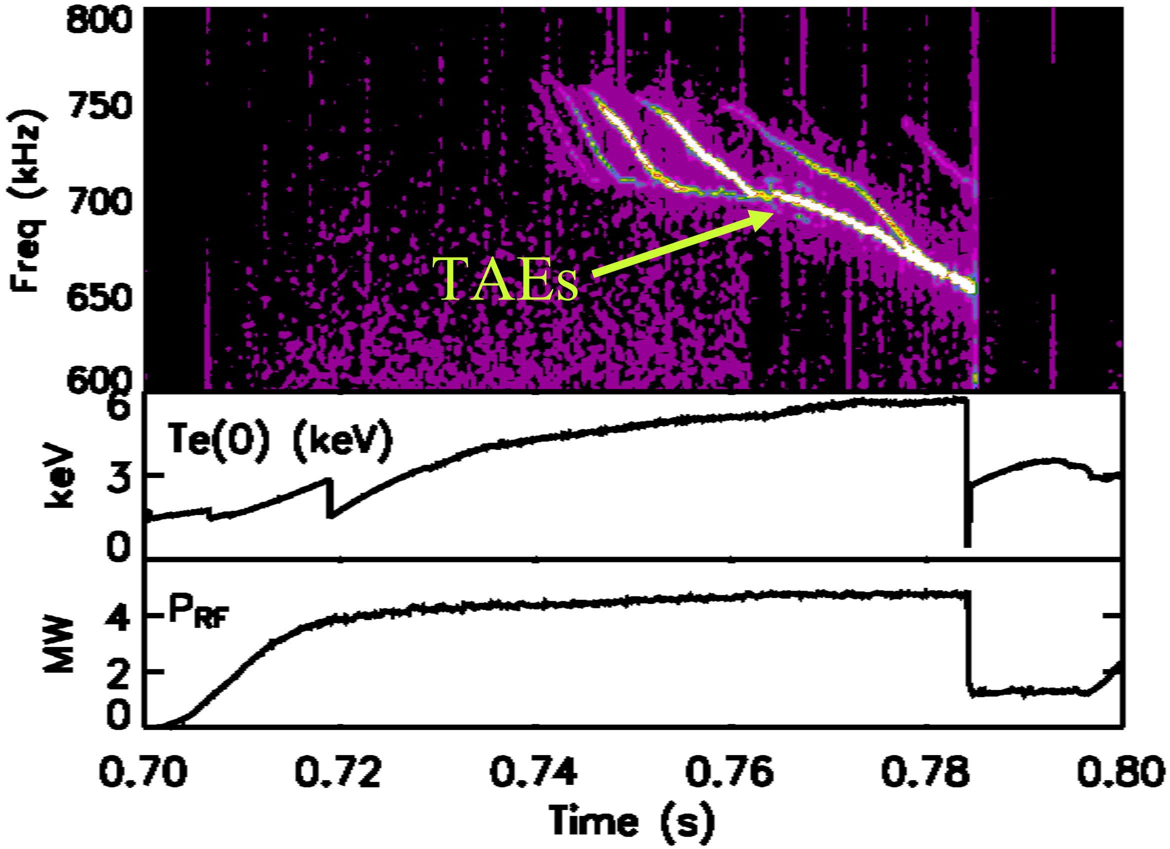
$$\omega^2(r) = k_{||}^2(r) v_A^2(r)$$

$$\omega_A = v_A(0)/(q_a R_0)$$

$$\propto B_T/(q_a R_0 \sqrt{n_i m_i})$$

- Toroidal Alfvén Eigenmodes (TAEs), Elliptical AEs (EAEs), etc
- Overlap of multiple AEs may enhance α particle loss before thermalizing

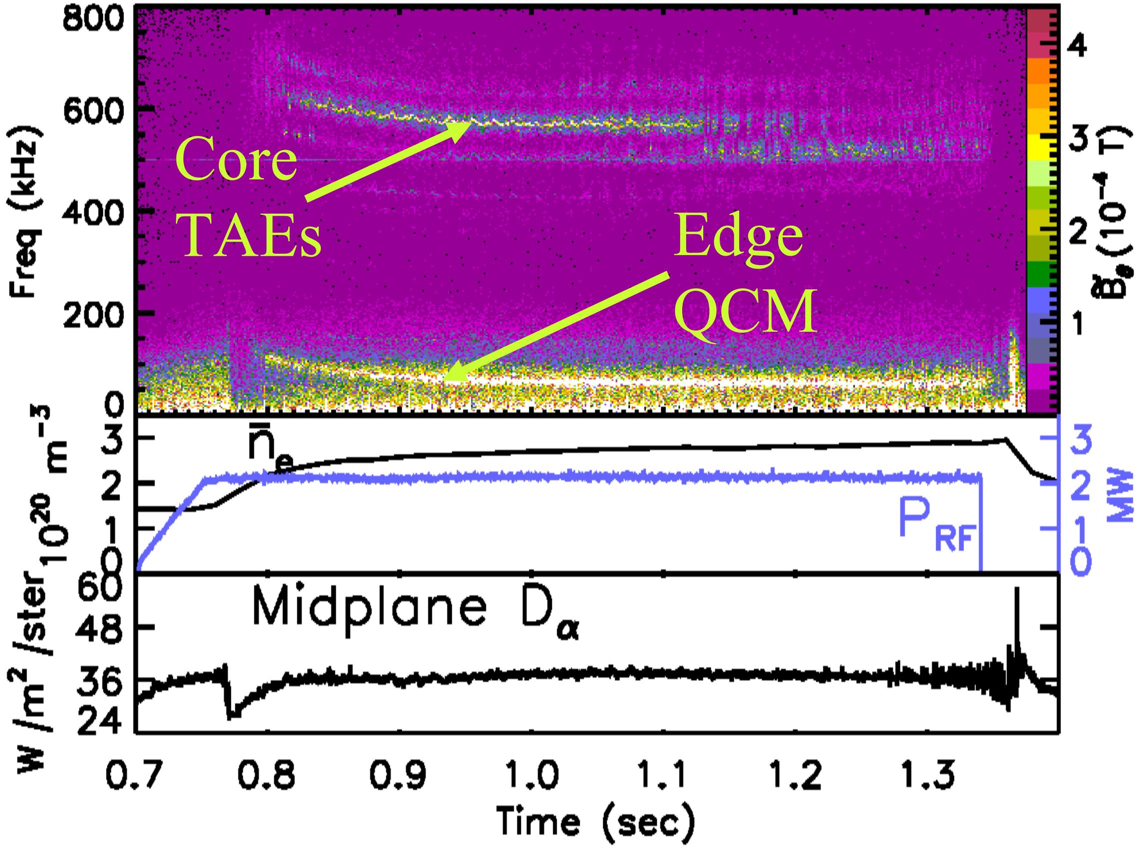
How Will Fast α-particles Affect Sawtooth Stability?



Alcator C-Mod

- Energetic α-particles are expected to stabilize sawteeth
- α-driven TAEs may redistribute the fast ions ➔ 'monster' sawteeth
- RF H&CD will be used to control such 'monster' sawteeth

Will Fast α's Strongly Couple Modes Nonlinearly?

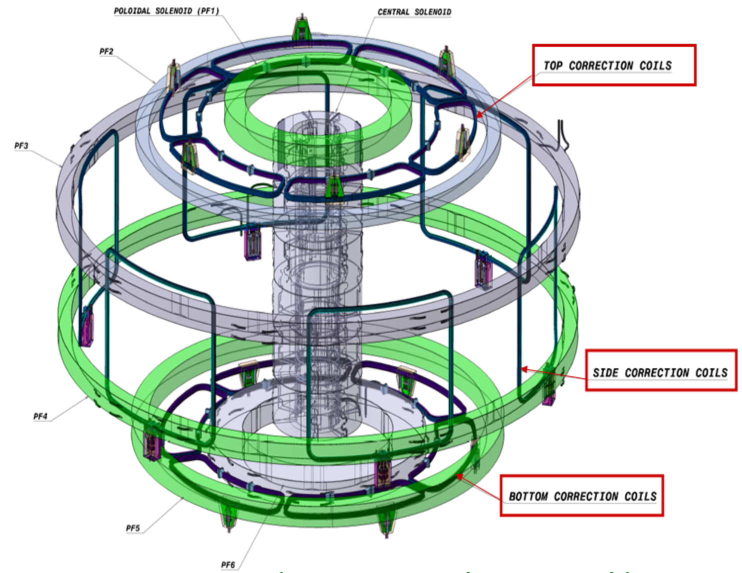


- Alfven eigenmodes may couple the core plasma to the edge
- Will nonlinear mode coupling then greatly enhance transport?
- What new nonlinear control schemes will be required?

Breizman: Thursday PM

Error Field Control with External Correction Coils

- Error fields come from CS, PF, and TF coil misalignments and feeds
- Error fields also from ferromagnetic materials especially Test Blanket Modules (TBMs)
- Error fields induce a torque slowing down the plasma toroidal rotation



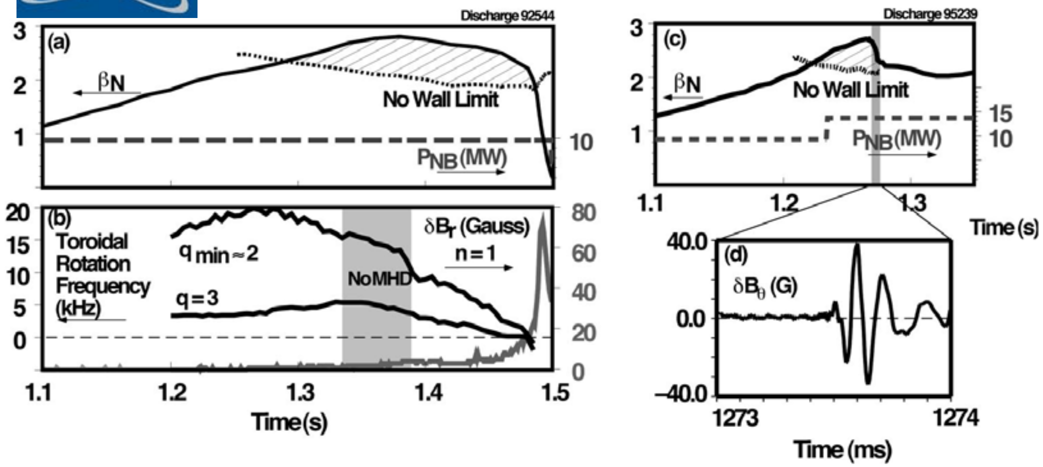
External Correction Coils

- Reduced rotation can lead to more locked modes and disruptions
- Error fields also enhance resistive wall modes (RWMs) at high β
- Three sets of 6 top, bottom, and side external correction coils will be used to minimize the (m,n) = (1,1), (2,1), (3,1) components within the 320 kAt top & bottom and 200 kAt side current limits

Reimerdes: Friday AM

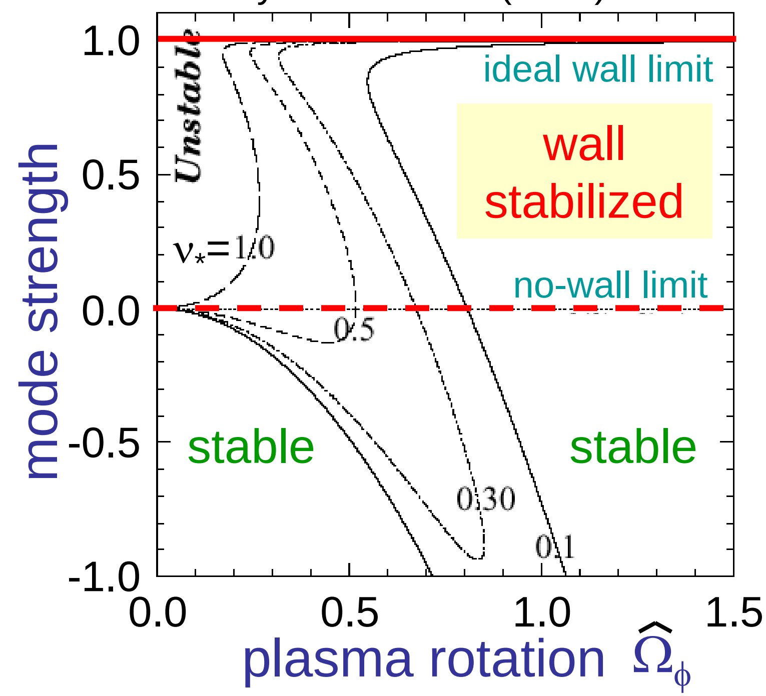
What are Resistive Wall Modes?

Garofalo, Phys Plasmas 1999



Fitzpatrick-Aydemir (F-A) stability curves

Phys. Plasmas **9** (2002) 3459

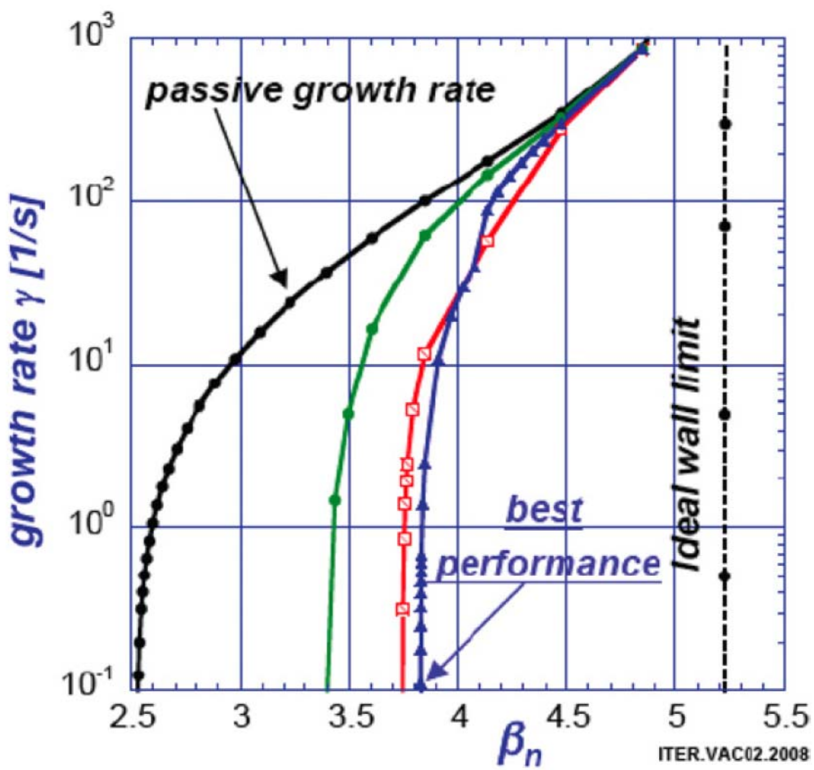


- Image currents in a conducting wall tend to stabilize external kink modes
- Image currents decay on a resistive eddy current decay time (τ_W ~ 200 ms in ITER)
- At high β_N, RWMs leak through wall with exponential growth time ~ τ_W
- RWMs grow in gap between no-wall and superconducting wall β limit
- Plasma rotation helps stabilize RWMs by maintaining image currents

Hegna: Monday PM

Resistive Wall Mode Control Allows High β Operation

RWM Control: Hawryluk, NF 2009

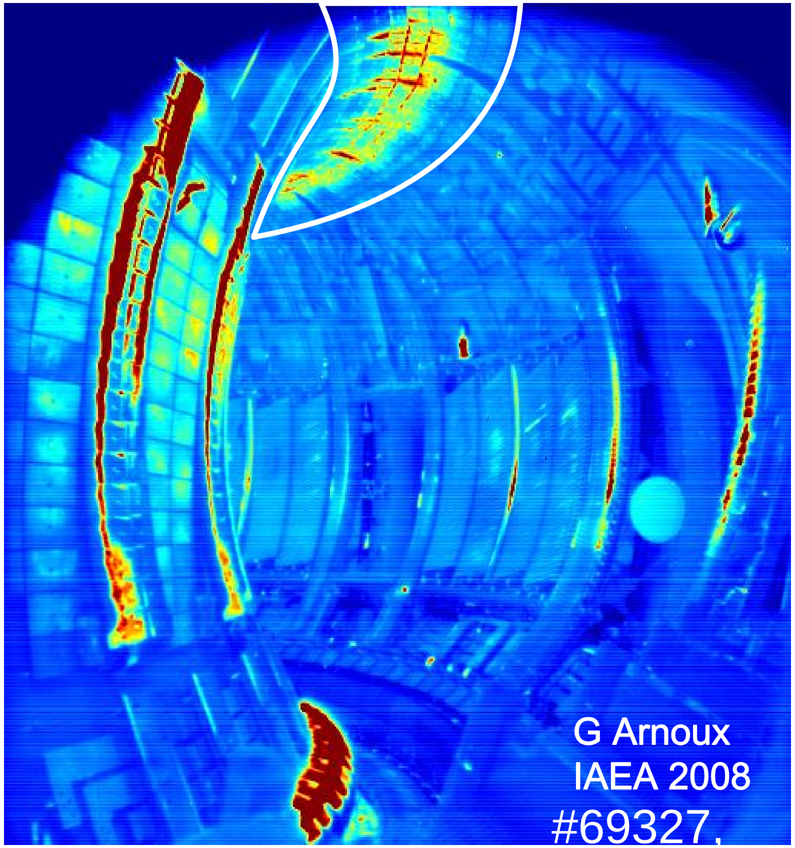


- RWM control may be required as an upgrade at high β using internal ELM coils to reduce RWMs and external correction coils + ELM coils to reduce error fields
- VALEN code calculations indicate that the ELM coils can stabilize RWMs for $\beta_N < 3.7 – 3.8$ in ITER
- The ELM coils will be phased with the slow rotation of the RWM
- Power supply characteristics will be defined after initial ITER operation

Boozer: Monday PM

Event Handling Subsystem

Real-time Hot Spot Detection



- Crucial for machine protection
  - PCS is first line of defense to avoid triggering central interlock system
  - to save valuable plasma time
  - e.g., hot spot detection
- Adaptive control in real-time
  - change algorithm to maintain performance or reduce machine damage
  - bridge segments – automatically switch to alternate segments if initial objective cannot be met
- Implement real-time forecasts
  - real-time modeling of performance
  - predict plasma regime changes
  - predict and avoid MHD instabilities
  - predict, avoid, and mitigate disruptions

Jardin: Thursday PM

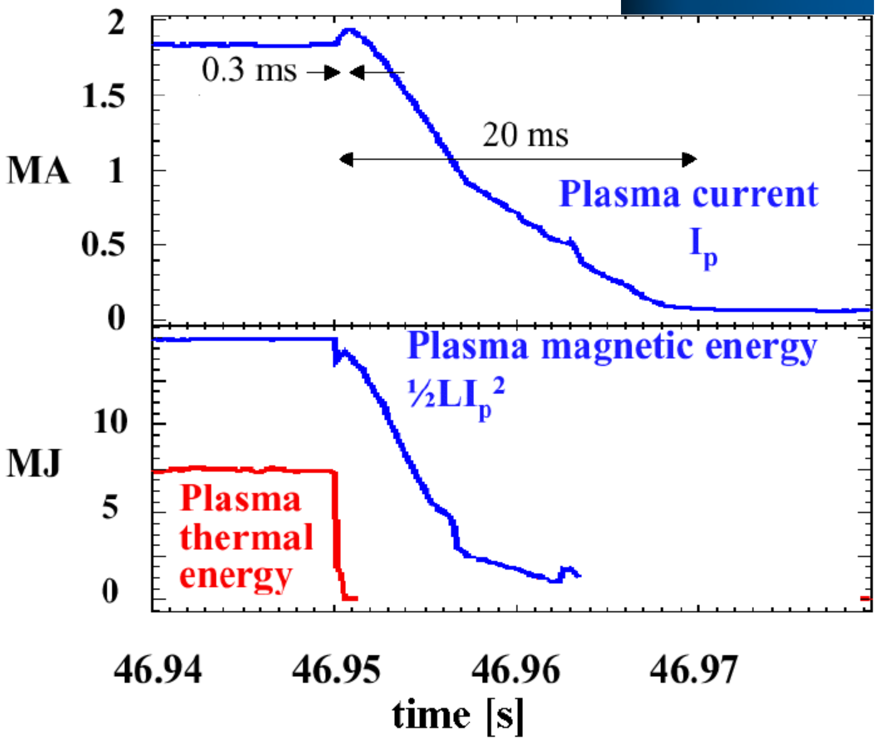
What are Disruptions?

Disruptions occur in tokamak plasmas when unstable p(r),j(r) develop

$\Rightarrow$ unstable MHD modes grow

$\Rightarrow$ plasma confinement is destroyed (thermal quench)

$\Rightarrow$ plasma current vanishes (current quench)



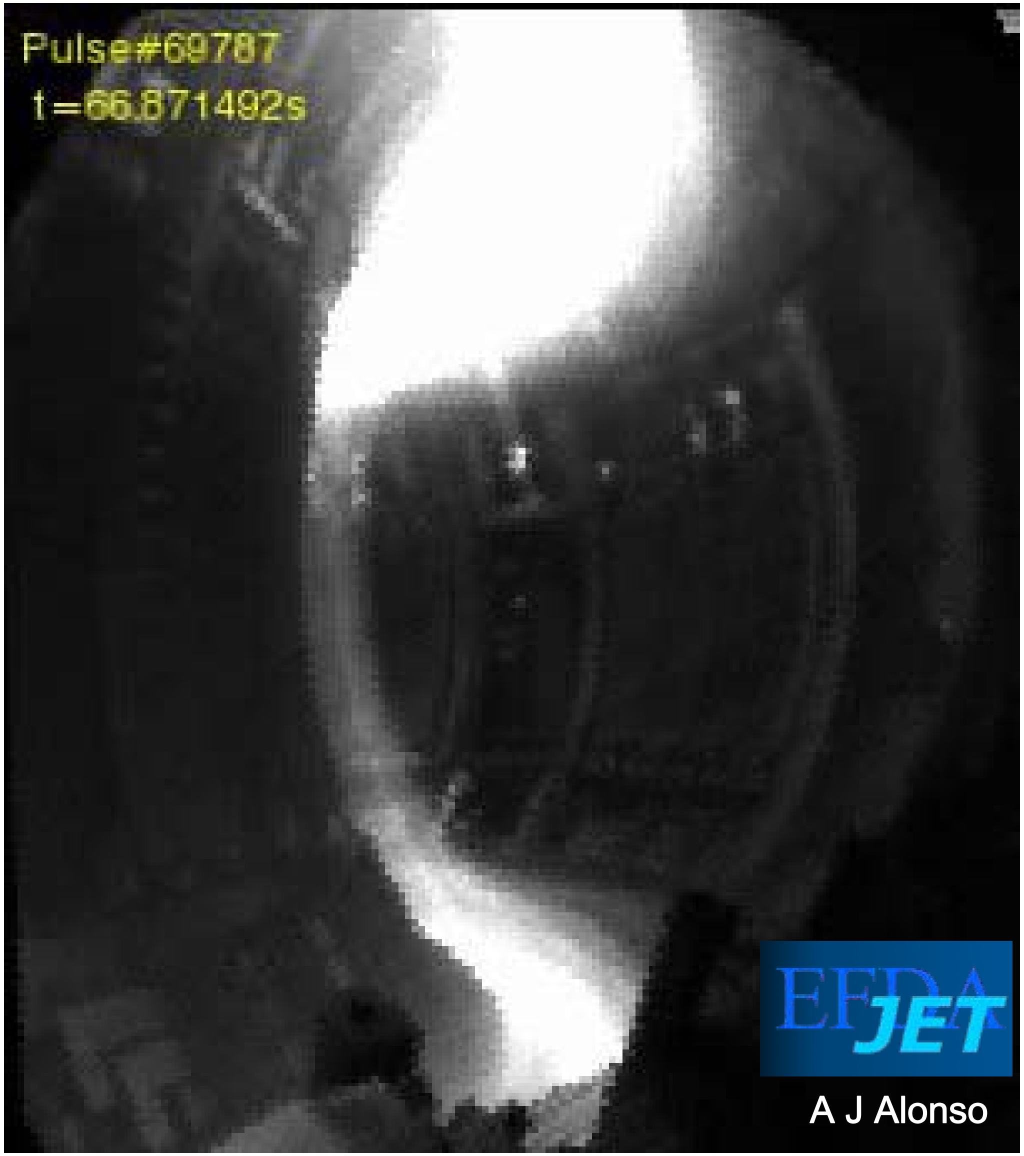
Typical JET timescales

- Thermal quench < 1ms $\Rightarrow$ deposits plasma thermal energy on plasma facing components (PFCs)
- Current quench > 10 ms $\Rightarrow$ deposits plasma magnetic energy by radiation on PFCs & runaway electrons

Expected values for ITER

- Thermal energy ~ 300 MJ
- Magnetic energy ~ 600 MJ
- Thermal quench time ~ 1.5 - 3 ms
- Current quench time ~ 20 - 40 ms

Disruptions Produce High Thermal and Mechanical Loads



Fast video taken in the visible at 250 kHz frame rate for 50 msec for a planned high performance density limit disruption in JET

Thermal quench:

High concentrated heat loads on plasma facing components

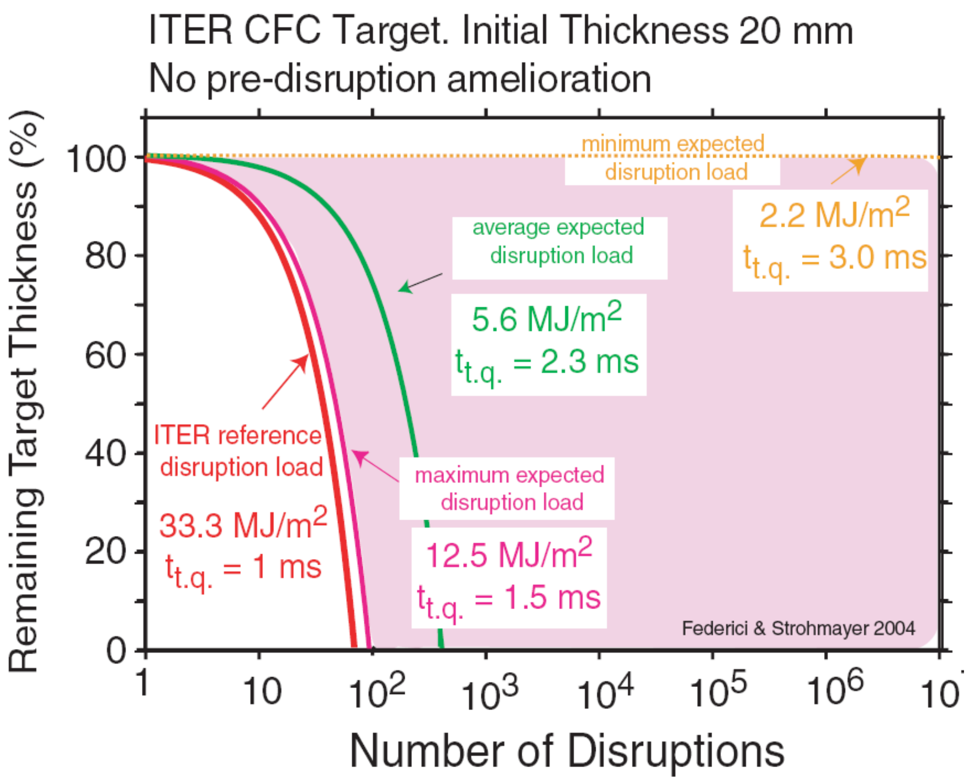
Current quench:

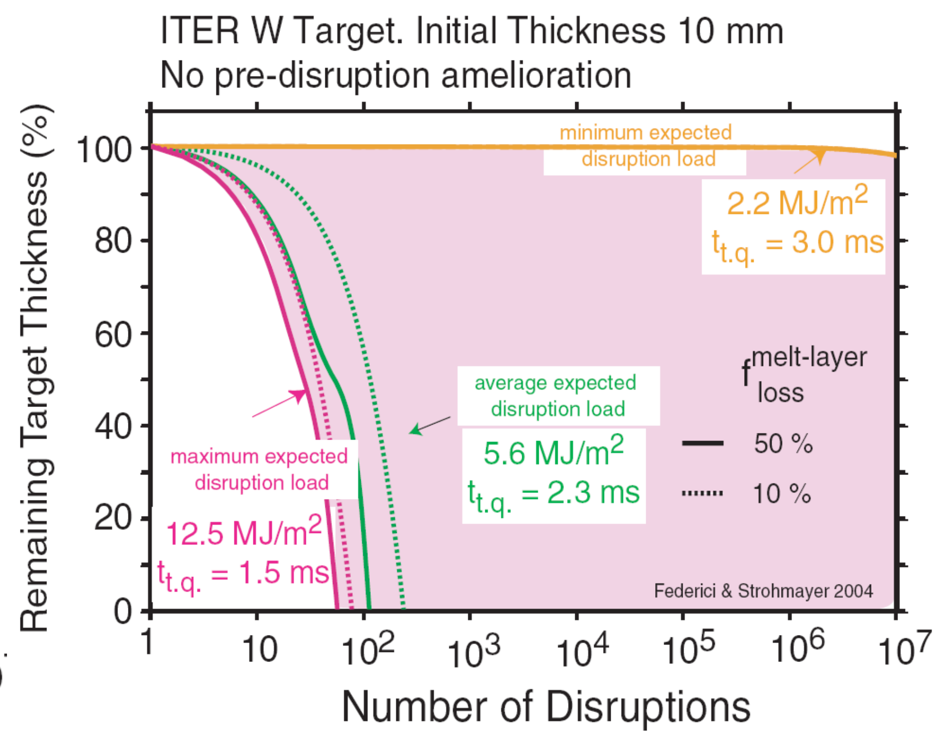
Large electromagnetic forces on the vacuum vessel and in-vessel components

Disruption forces shake the camera support several cm!

Disruptions Limit the Divertor Lifetime in ITER

- ➢ Expected energy loads on the divertor and first wall in ITER may exceed material limits (sublimation + melting)
- ➢ Dynamics of plasma and materials in these conditions is very complex
  → major uncertainties in consequences of disruptions for PFCs in ITER





- ➢ The divertor may only withstand a (few) hundred Q=10 disruptions!

What are Vertical Displacement Events – VDEs?

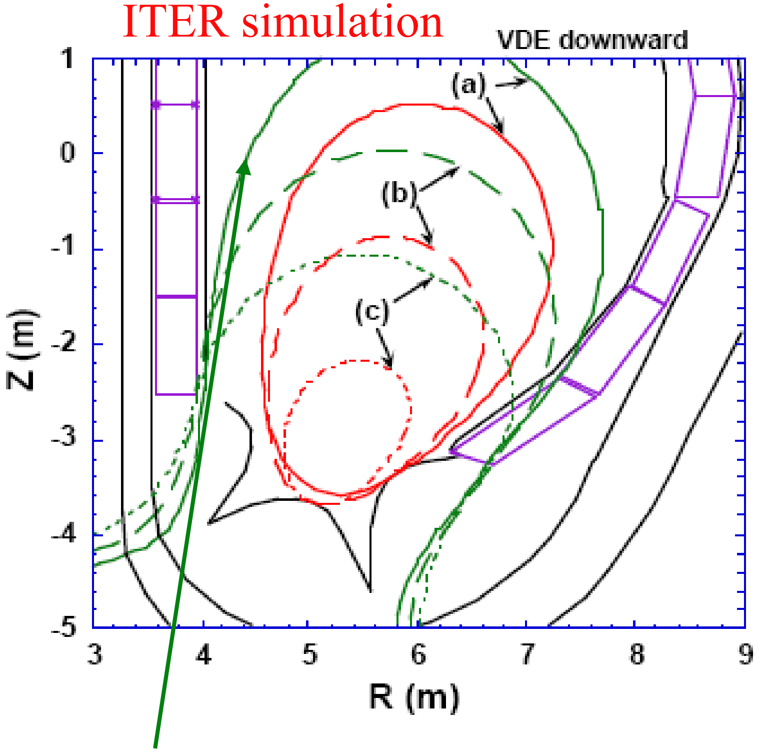
- **When a loss of vertical position control takes place:**

  $\Rightarrow$ plasma impacts wall with full plasma energy

  $\Rightarrow$ high localized heating

  $\Rightarrow$ mitigation required

ITER simulation



Halo current layer

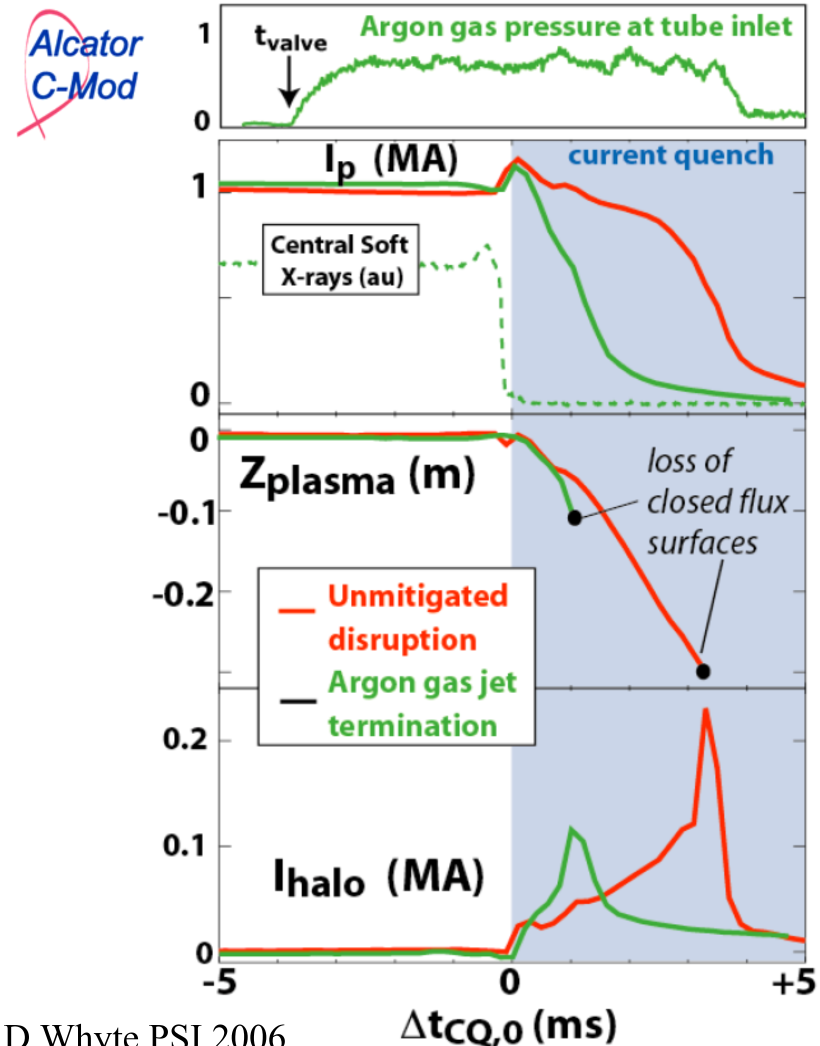
Control issues

- Detection of loss of vertical position control
- Fast stop of plasma by massive gas injection, killer pellets, etc.
- Effectiveness, reliability of mitigation
- Runaway electron plasma must be controlled and safely eliminated to avoid localized wall damage
- Need R&D in existing experiments

Humphreys: Tuesday AM

How Can Disruption/VDE/Runaways be Mitigated?

Disruption Mitigation with massive impurity injection



D Whyte PSI 2006

High pressure impurity gas injection looks promising for disruption/ VDE mitigation:

- efficient radiative redistribution of plasma energy - reduced heat loads
- reduction of plasma energy and current before VDE can occur
- substantial reduction in halo currents (~50%) and toroidal asymmetries
- Separate disruption and runaway mitigation systems may be necessary
- Multiple high pressure gas injection may shrink runaway current channel

Conclusions

- ITER plasma control will be based on present tokamaks but:
  - must be very reliable including pre-pulse validation with simulations
  - also requires divertor power exhaust and fusion burn control
  - requires effective multiple parameter control with shared actuators
  - will develop adaptive control based on previous conditions and real-time plasma modeling simulations
  - needs a sophisticated event handling system for machine protection
- Substantial R&D on existing machines is required to establish effective plasma control techniques for ITER
- MHD control in ITER must be very flexible to control the expected modes found in existing devices and unexpected modes discovered in new high performance burning plasma regimes
- DT in ITER will be ~ 2026 – 27 ➔ today's students will make Q=10 and long pulse steady-state fusion regimes a reality