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Plasma Operation in ITER

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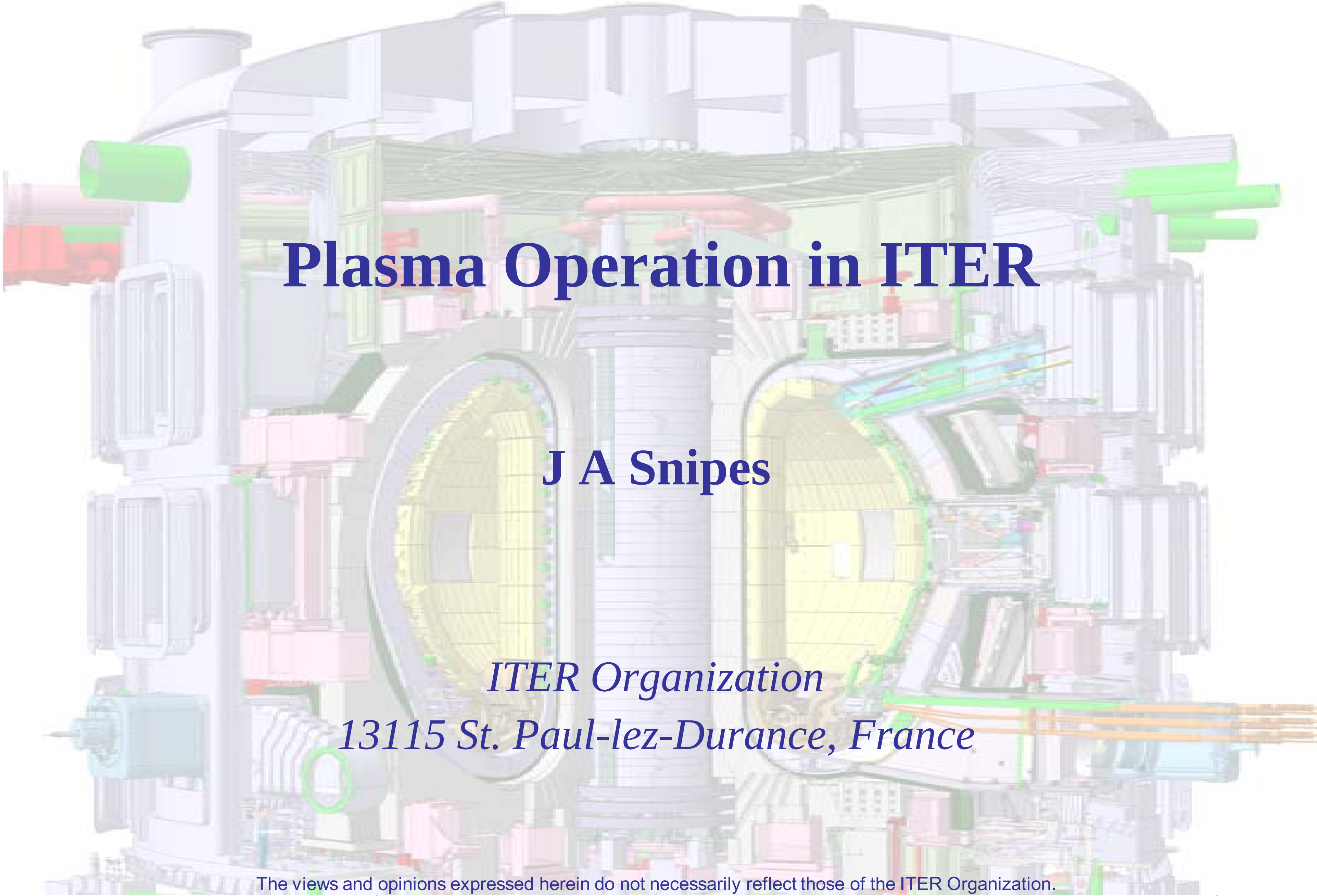

Plasma Operation in ITER

J A Snipes

ITER Organization

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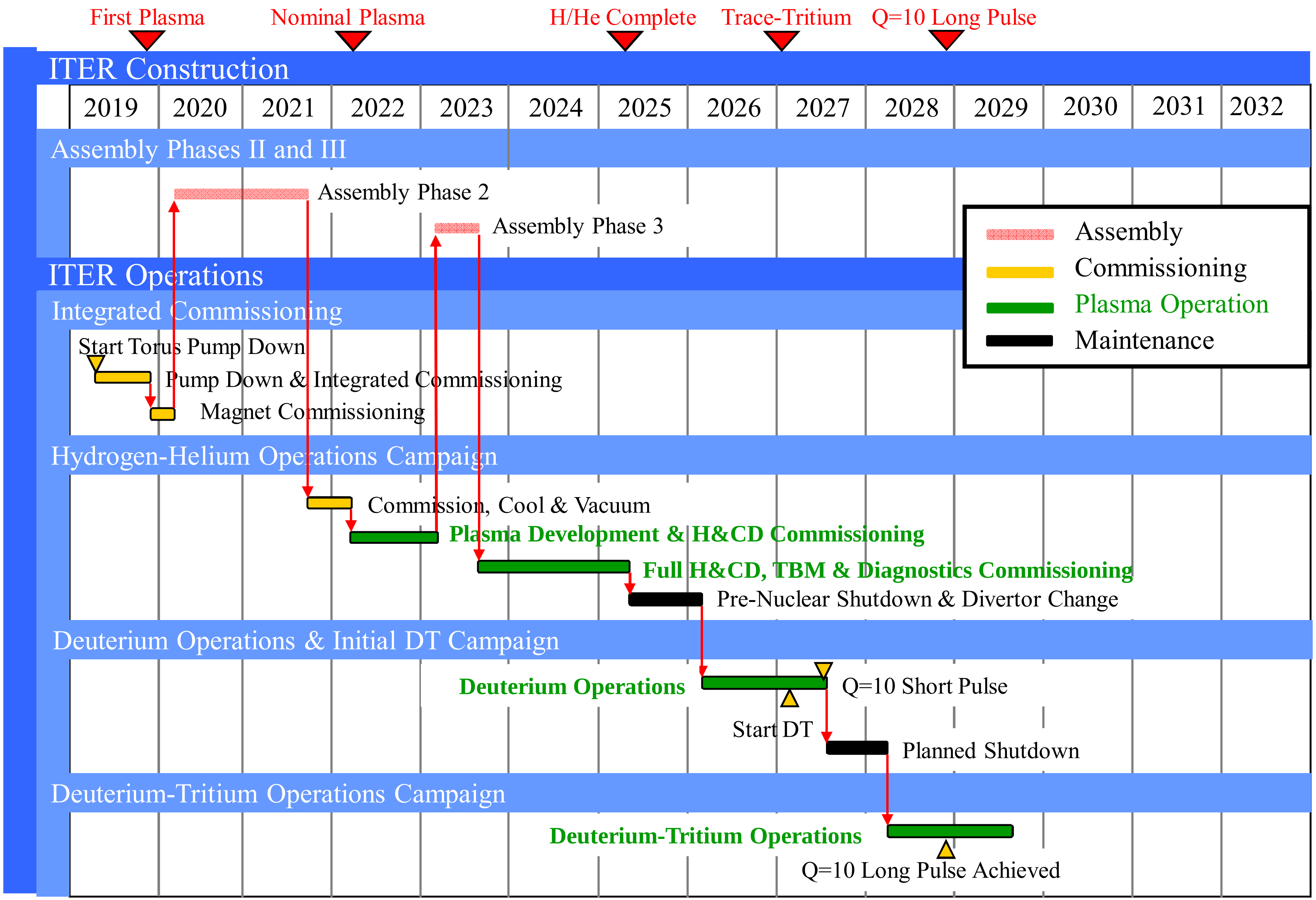
Outline

- ITER Experimental Program
- ITER Operational Scenarios
- ITER Plasma Control System (PCS) description
- Plasma control areas
  - 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



What are ITER plasmas designed to do?

$\Rightarrow$ ITER Operational Scenarios

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

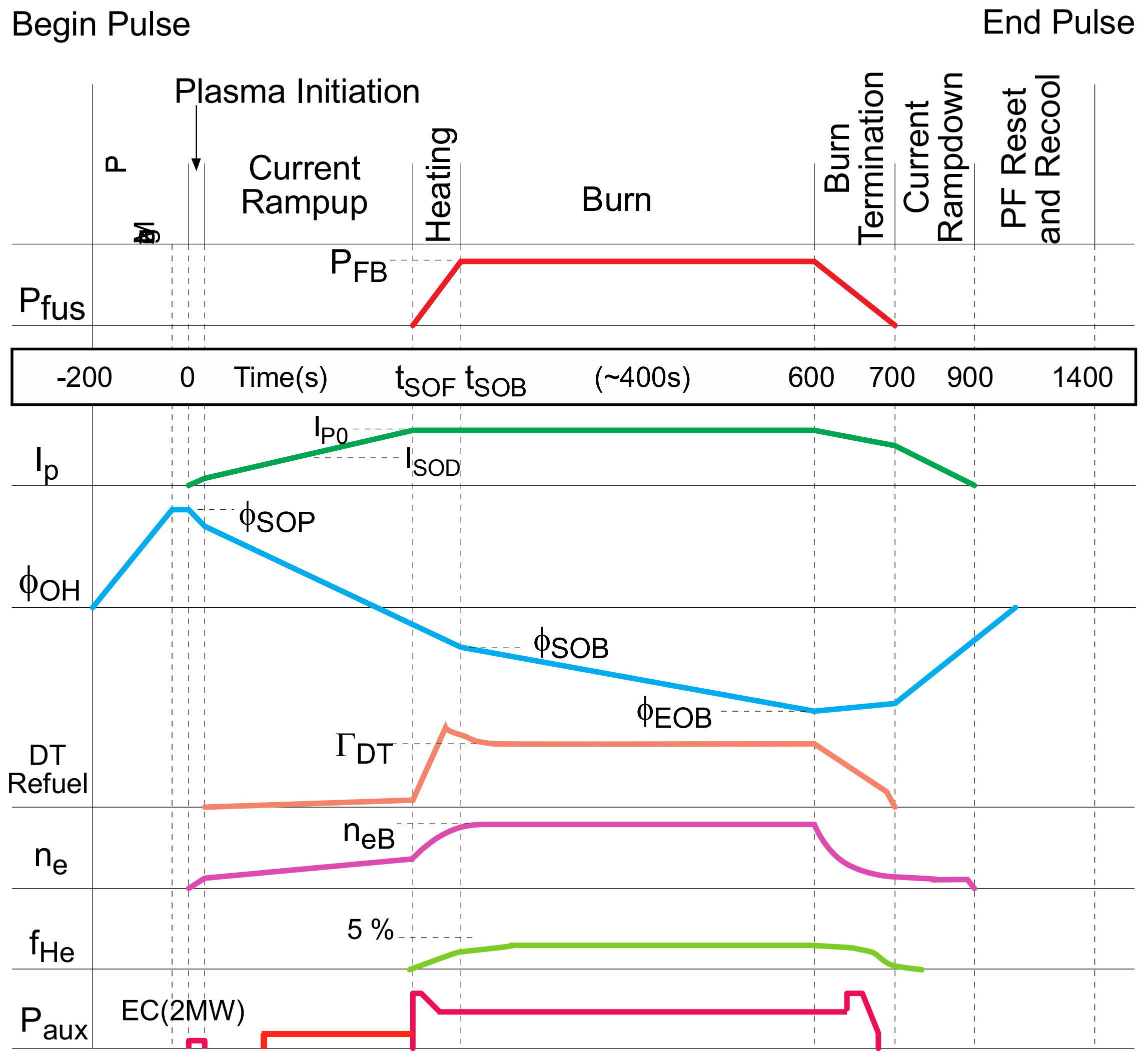
- **Advanced scenarios:**

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

15MA Inductive Scenario - Schematic

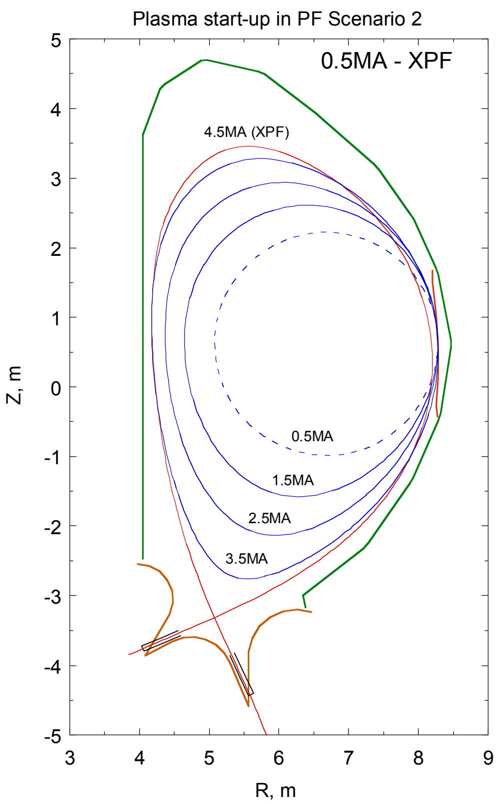
- **Typical 15MA Q=10 inductive scenario has:**
  - current ramp-up phase of 70-100s
  - heating phase of ~50s
  - burn phase of 300-500s
  - shutdown phase of 200-300s
- **Typical pulse repetition time ~1800s**
  - based on burn duty cycle of 25%

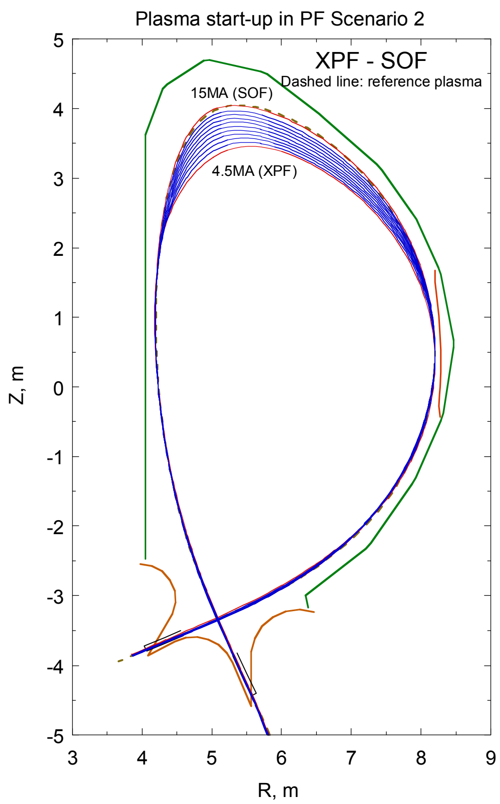


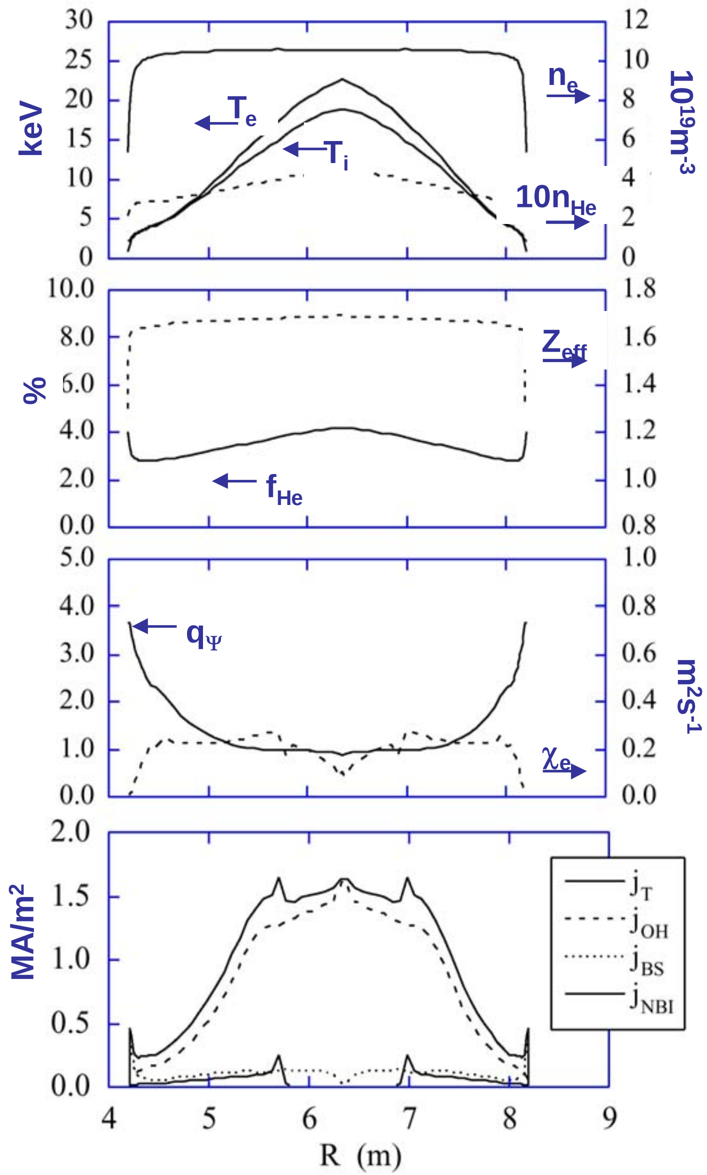
ITER Plasma Scenario - ELMy H-mode

A Q=10 scenario with:

I_p=15MA, P_{aux}=40MW, $H_{98(y,2)}$=1



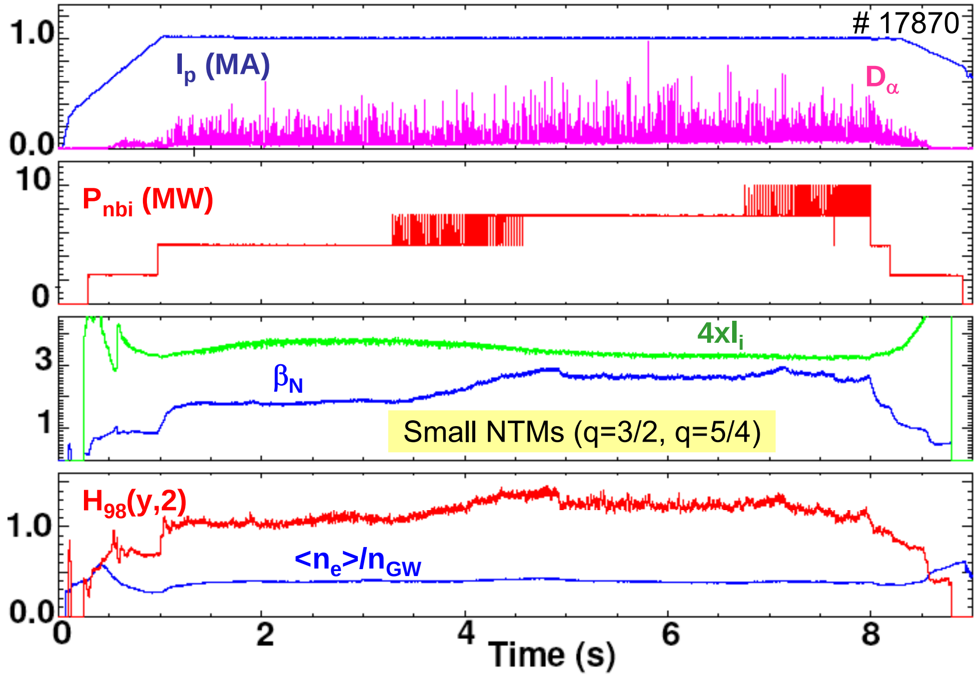




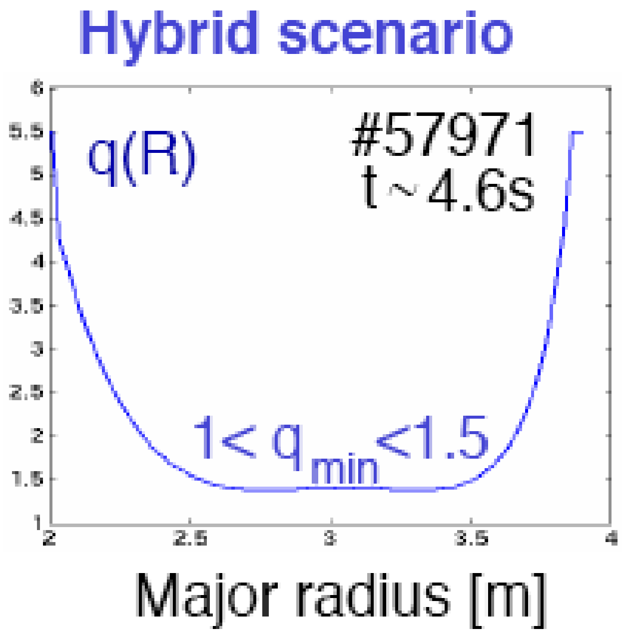
Current Ramp-up Phase

ITER Hybrid Scenario Operation

(O Gruber et al, 46th APS-DPP Meeting, Savannah, 2004)



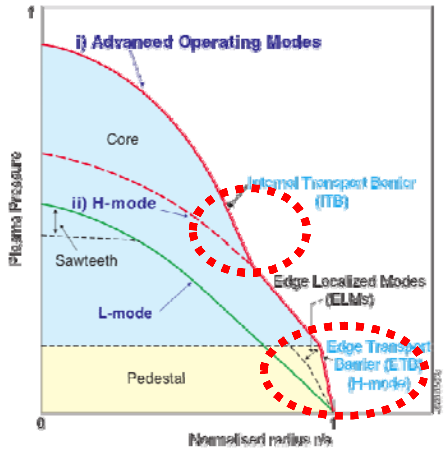
(M L Watkins et al, 21 IAEA FEC, Chengdu, 2006)



- **The so-called "hybrid" mode (improved H-mode) developed in recent years may allow ITER both to operate at higher fusion performance and for longer durations:**
  - Flat central q-profile with q(0) ~ 1 appears critical
  - R&D is ongoing to demonstrate extrapolability of regime to ITER

Steady-State Operation

- **Discovery of internal transport barriers ⇒ "advanced scenarios"**



plasma with reversed central shear + sufficient rotational shear

internal transport barrier
⇒ enhanced confinement

reduced current operation +
large bootstrap current fraction

active MHD control

reduced external current drive +
current well aligned for MHD stability and confinement enhancement

Steady-state operation +
High fusion power density

- **But development of an integrated plasma scenario satisfying all reactor-relevant requirements remains challenging**

ITER Plasma Control

Plasma Control System Has Five Control Areas

The ITER Plasma Control System (PCS) has five control areas:

1) **Wall conditioning and tritium removal:** clean in-vessel components and control tritium inventory
2) **Plasma axisymmetric magnetic control:** plasma initiation, plasma current, position, and shape
3) **Plasma kinetic control:** power and particle flux to the divertor and first wall, fuelling, non-inductive plasma current, plasma pressure & fusion burn
4) **Non-axisymmetric mode control:** sawtooth, neoclassical tearing mode (NTM), edge localized mode (ELM), Alfven eigenmode (AE), error field and resistive wall mode (RWM)
5) **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 ($\propto a^2 B_\phi / R I_p$) > 2, q = $d\varphi/d\theta$ = 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 $\propto p/B^2$), limited by various MHD instabilities

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

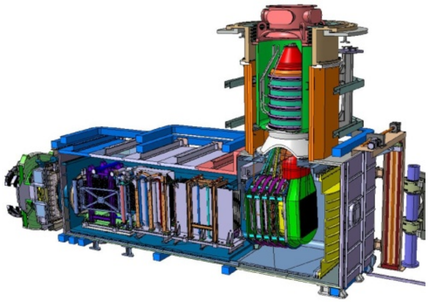
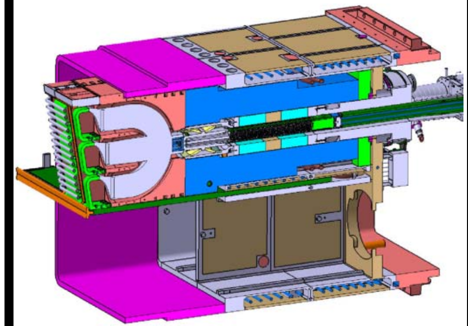
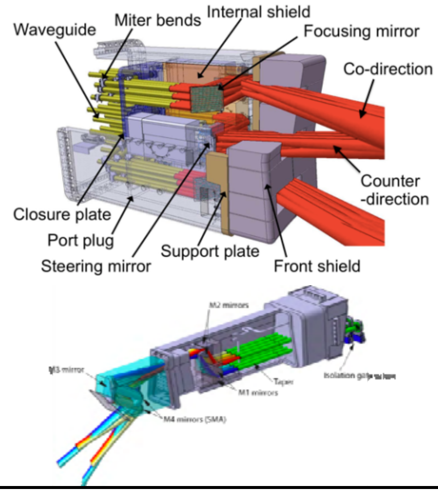
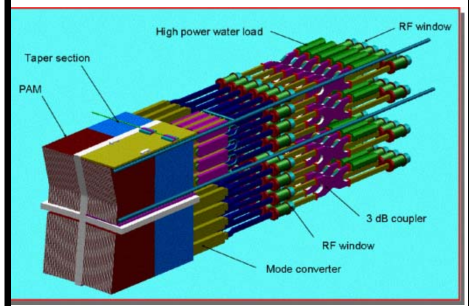
Operational Sequence Changes in Real-Time

- 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 mode 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

ITER Heating & Current Drive Systems

NB	IC	EC	LH
Neutral Beam - 1 MeV	Ion Cyclotron 40-55MHz	Electron Cyclotron 170GHz	Lower Hybrid ~5 GHz
			
33MW* +16.5MW#	20MW* +20MW#	20MW* +20MW#	0MW* +40MW#
Bulk current drive limited modulation	Sawtooth control modulation < 1 kHz	NTM/sawtooth control modulation up to 5 kHz	Off-axis bulk current drive

*Baseline Power
#Possible Upgrade

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

130 MW (max installed)
(110 MW simultaneous)

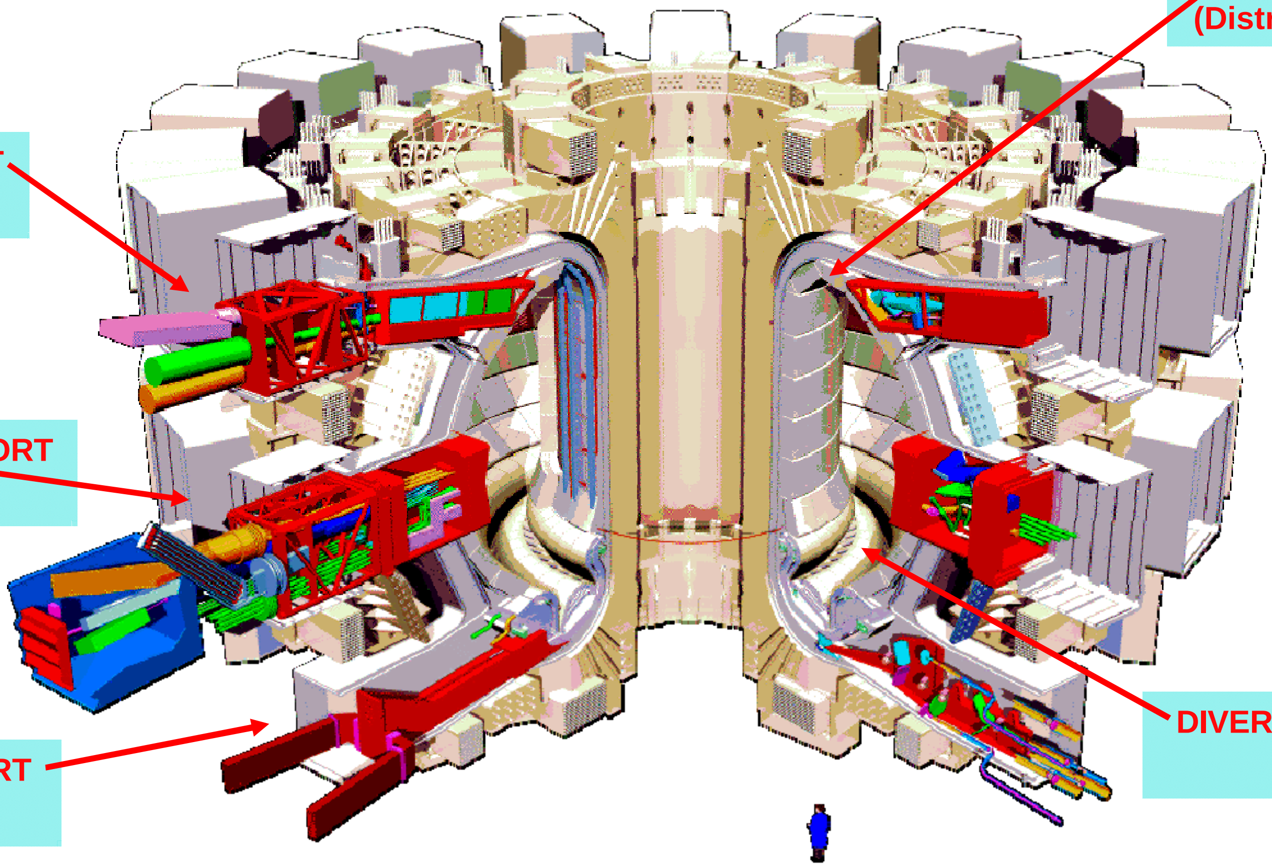
Why Four Heating Systems?

- **Technology:**
  - ICRF and LHCD fairly conventional
  - NBI and ECRH source technology challenging
- **Coupling to plasma:**
  - NBI and ECRH straightforward
  - ICRF and LHCD problematic: antenna design challenging due to difficulty in coupling wave through (evanescent) plasma edge
- **Radial localization:**
  - Resonance condition favours ECRH and ICRF radial localization
  - NBI and LHCD more global in effects
- **Current drive:**
  - NBI and LHCD most efficient
  - ECRH and ICRF used in more specialized applications where space localization important

PCS Requires Measurements for Control

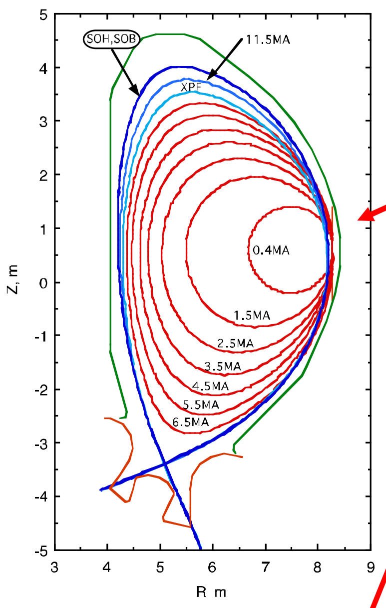
- **Wall conditioning and tritium removal** requires residual gas species and partial pressures on timescales of minutes and hours
- **Plasma axisymmetric magnetic control** requires neutral pressure, impurity radiation, stray fields, plasma current & position, poloidal field & flux, coil currents, toroidal 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 mode 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 ….)

Fusion Plasma Diagnostics

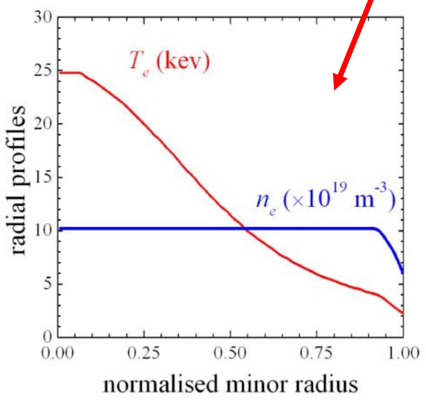


Plasma shape evolution (ITER)

Plasma-wall interaction (JET)

Plasma density and temperature (ITER)

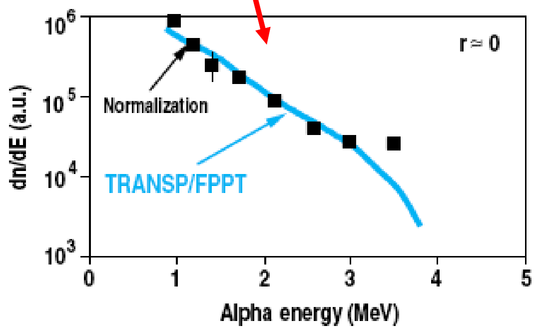
Plasma radiation (ASDEX-U)

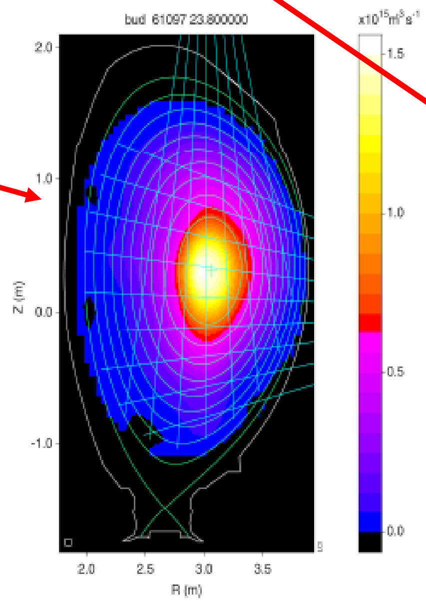


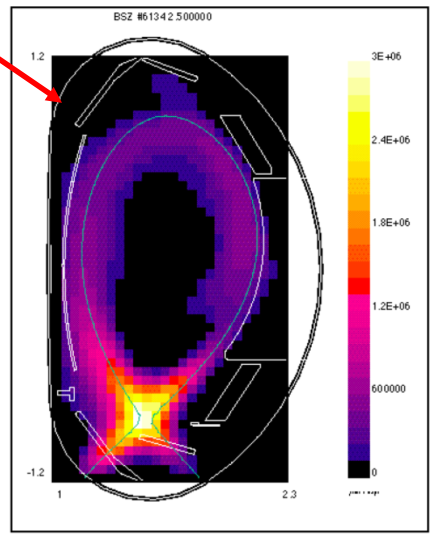
Fusion power:

14MeV neutron profile (JET)

α-particle spectrum (TFTR)



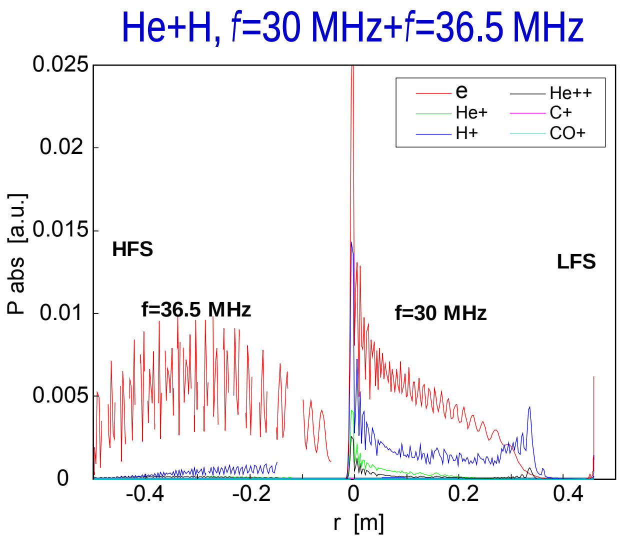


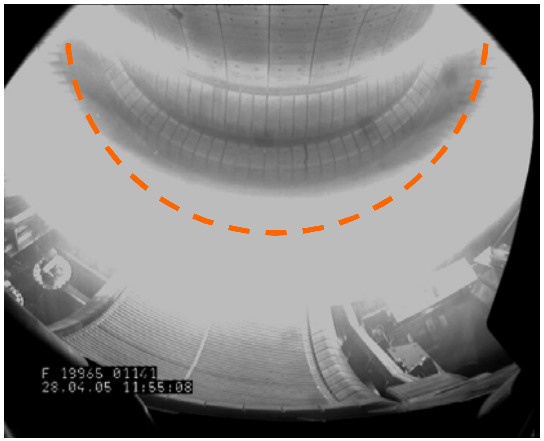


Five Plasma Control Areas of ITER

1) Wall Conditioning and Tritium Removal

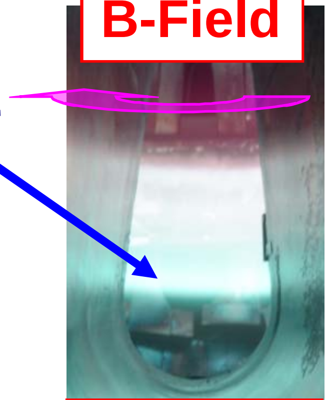
A. Lyssoivan, 18th PSI 2008





- ➢ 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

B-Field

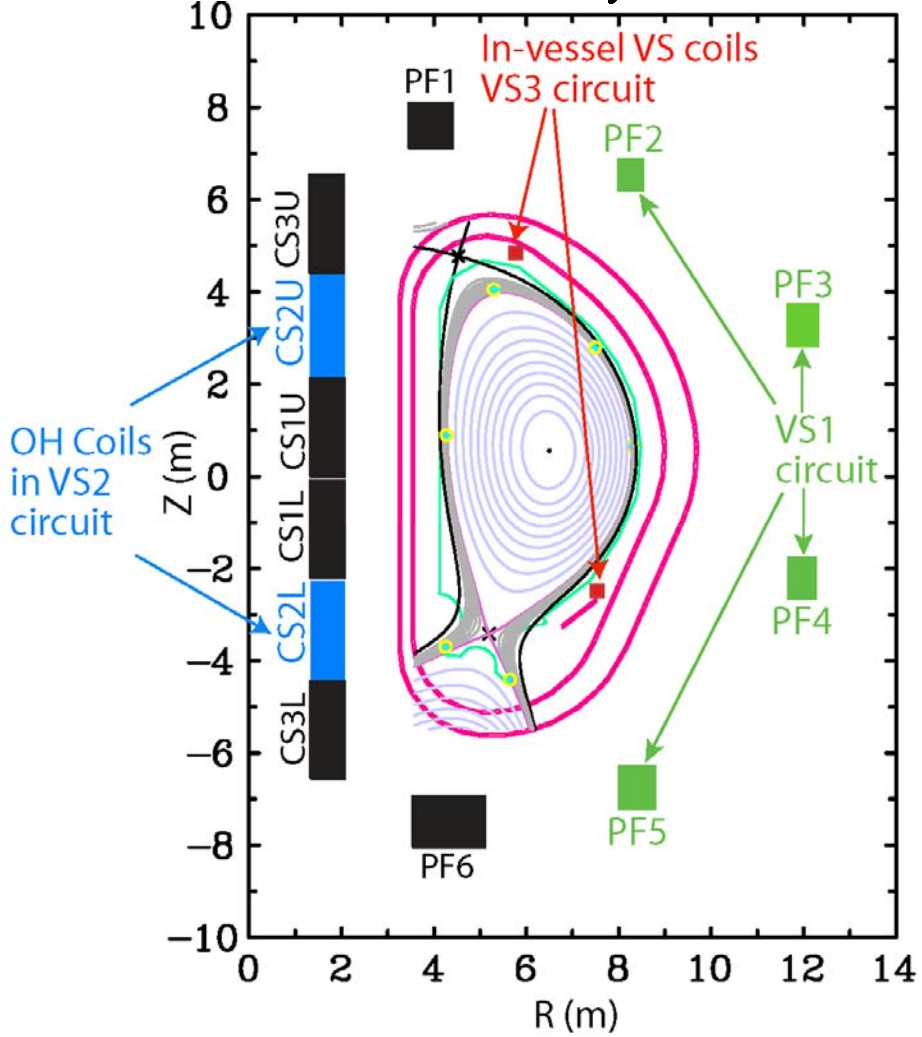


Vertical view top window

X Gong, J Li, PSI 2010

2) Axisymmetric Magnetic Control

Magnetic Actuators Showing Three Vertical Stability Circuits



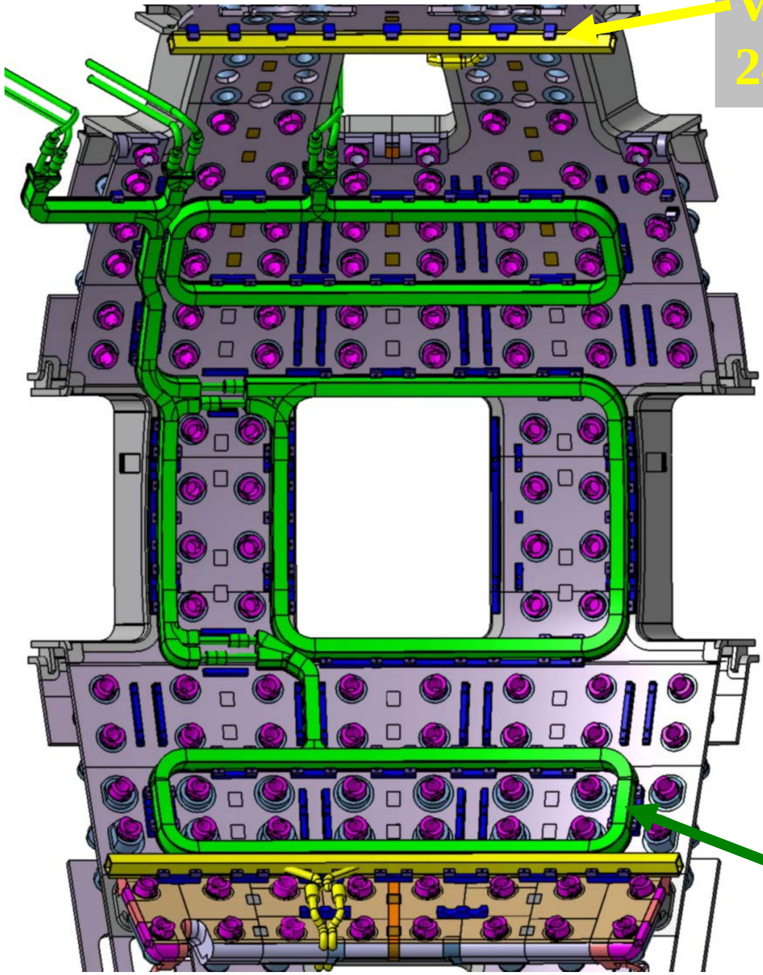
- Includes plasma initiation, inductive plasma current, position, and shape control
- PCS will control currents in CS, PF, and VS magnets, but not TF
- Plasma initiation will include several MW of startup ECH
- Inductive plasma current, shape, and radial position control will have a settling time of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

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 $l_i < 1.2$
- l_i is the plasma internal inductance $l_i = \dfrac{2\int_0^a B_\theta^2 r dr}{a^2 B_{\theta a}^2}$
- 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:

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

Magnetic Actuators Include In-Vessel Coils



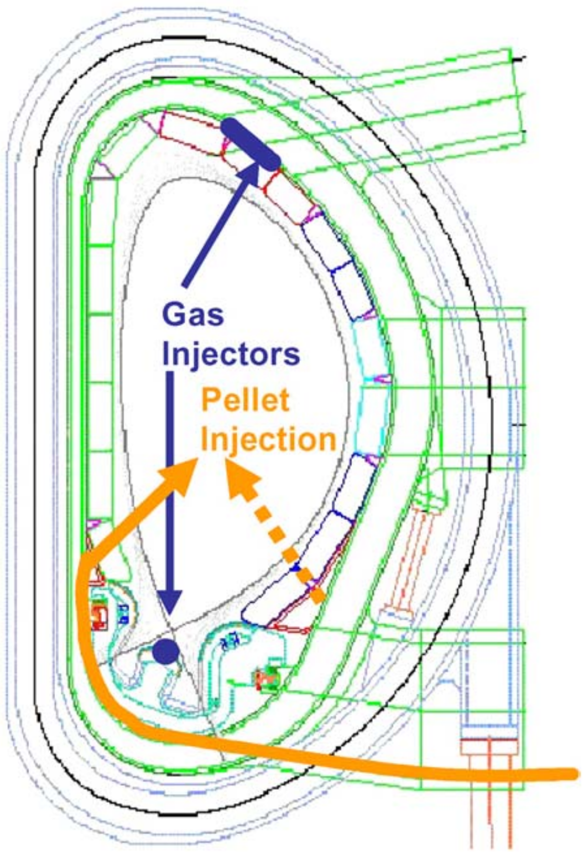
- **A set of in-vessel resonant magnetic perturbation (ELM) and vertical stability (VS) coils is being designed:**
  - 9 toroidal × 3 poloidal array on outboard internal vessel wall
  - vertical stabilization coils upper & lower loops form a saddle coil

3) Plasma Kinetic Control

- 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 $q_{min} > 2$
- Recall from introductory lecture q is the safety factor: $q = \dfrac{d\Phi}{d\Psi}$

Gas and Pellet Injection from Multiple Ports

Fueling Actuators

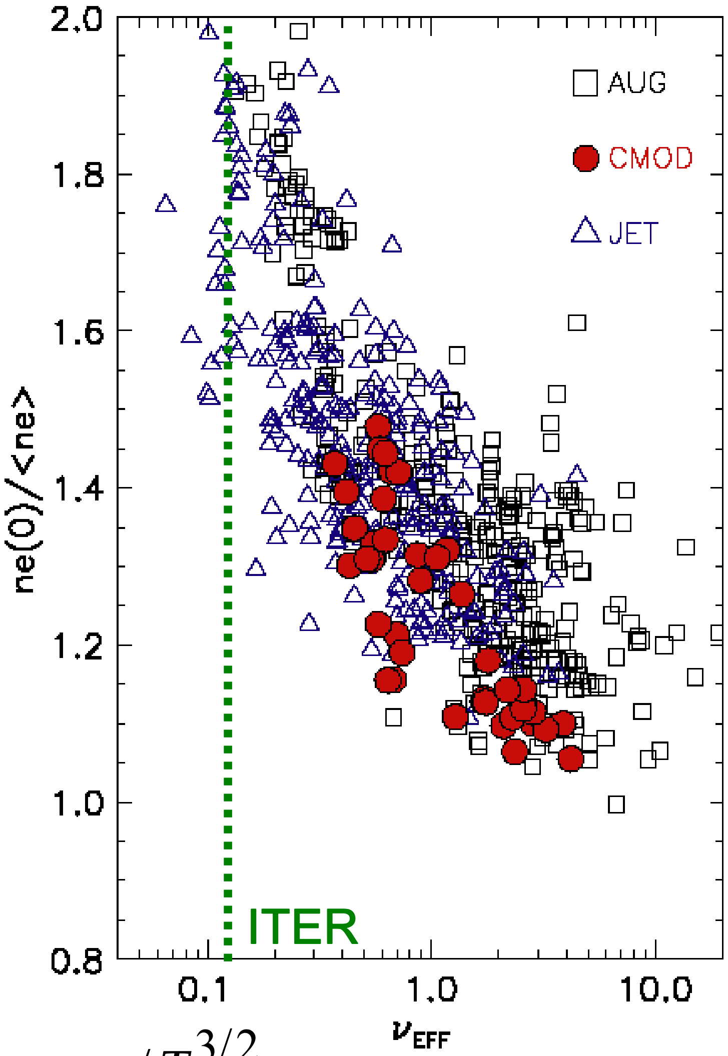


Baylor, NF 2007

- Gas fueling systems provide H, D, T, ^{4}He up to 100 Pa m^3/s except 10 Pa m^3/s for T
- Gas impurity injection provide N, Ne, Ar, and ^{3}He up to 10 Pa m^3/s
- 10 gas valve boxes in 4 upper and 6 lower ports each provide maximum throughput with a response time from < 1 – 3 s
- Frozen H, D, T, N, Ne, and Ar pellets provided from 3 lower ports with both high and low field side launch at up to 16 Hz
- ELM pellet pacemaking up to 48 Hz from 3 staggered low field side injectors
- Max throughput 120 Pa m^3/s for H, D, 111 Pa m^3/s for T, and 10 Pa m^3/s for impurities

What Will Core Fuelling be Like in ITER?

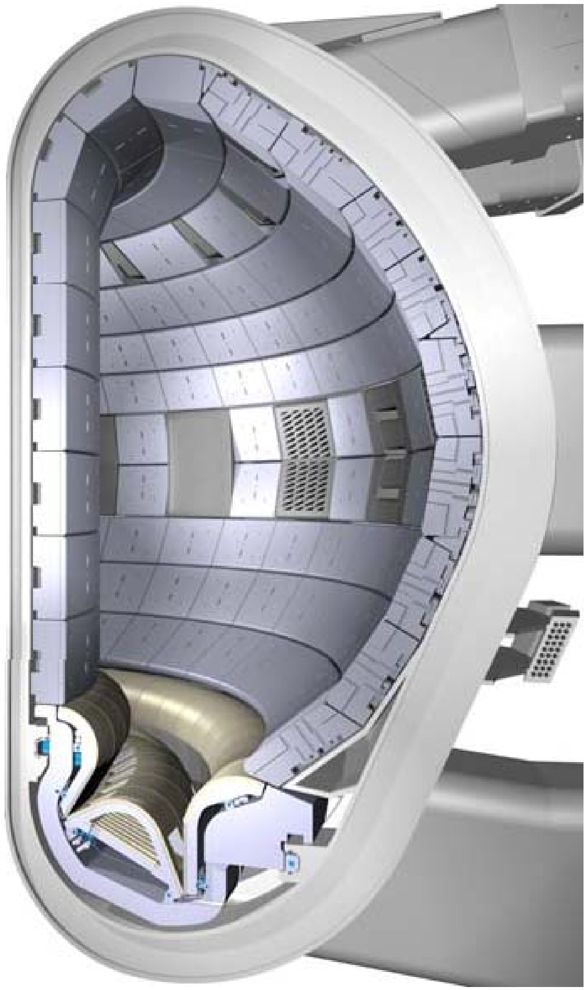
- Present cryopump design limit: $\Gamma_{pump} = 200$ Pa-m^3/s
- Expected recycling flux: $100 \times \Gamma_{pump}$
- Expect low central gas fuelling
  - ➔ flat density profiles
- Inward pinch at low ν* may lead to density peaking in ITER
- Could increase fusion reactivity
- But profile peakedness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



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

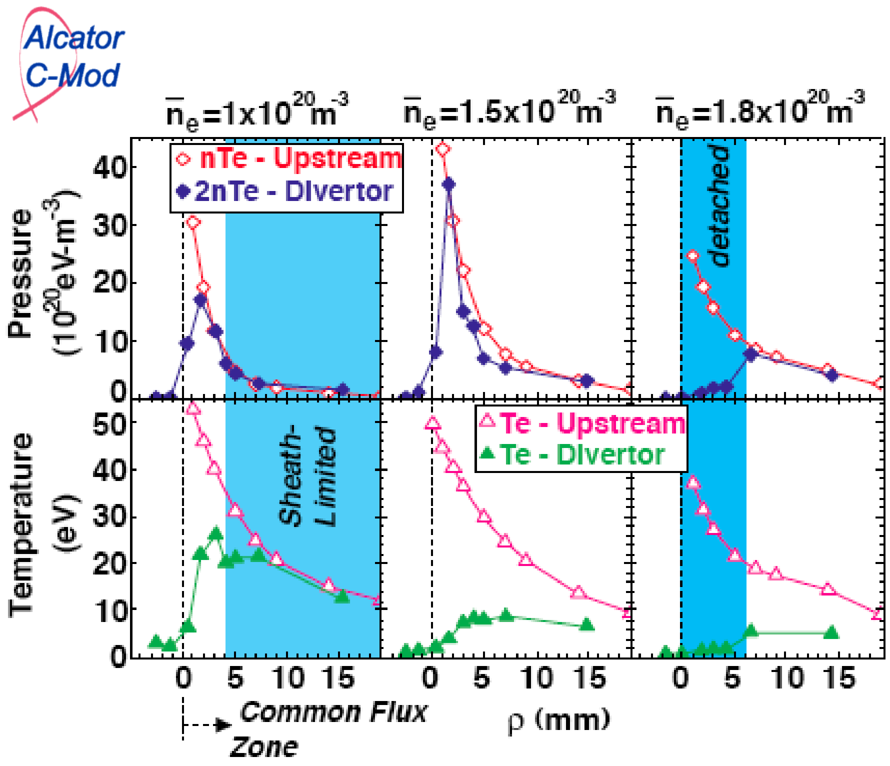
Greenwald, NF 2007

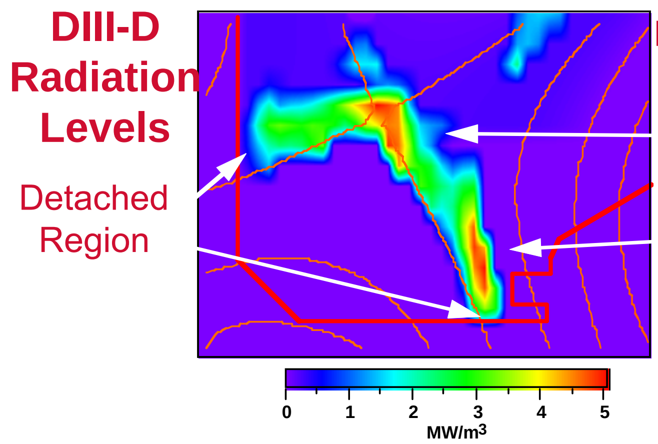
Power and Particle Flux Control is Essential

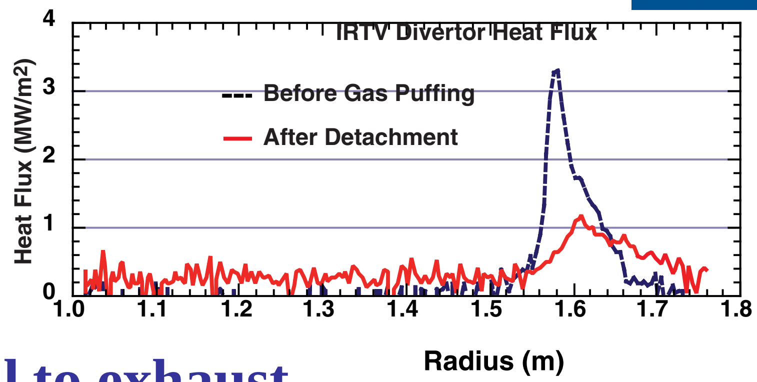


- Power and particle flux control to the first wall and divertor is essential to avoid damage and excessive impurity influxes
- Divertor melting can occur quickly (~1 s) at full performance
- Divertor detachment control with Ne/Ar puffing avoids excessive divertor heat load
- MARFE control will be required 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





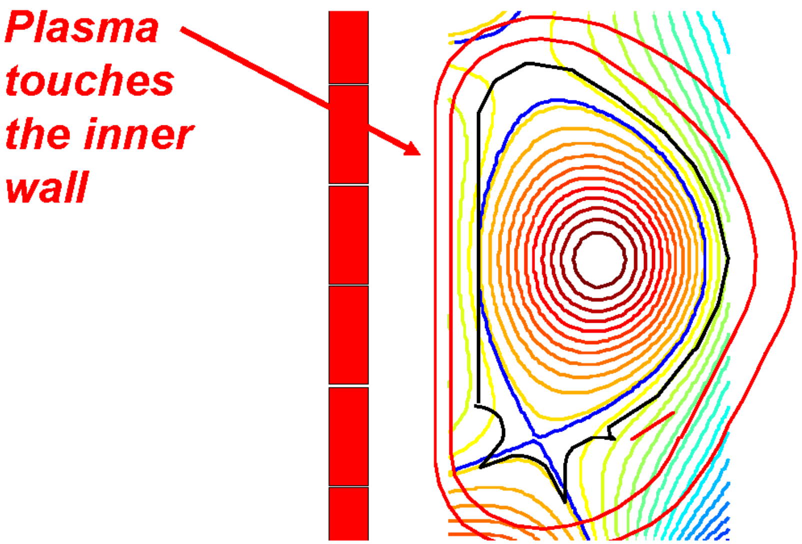


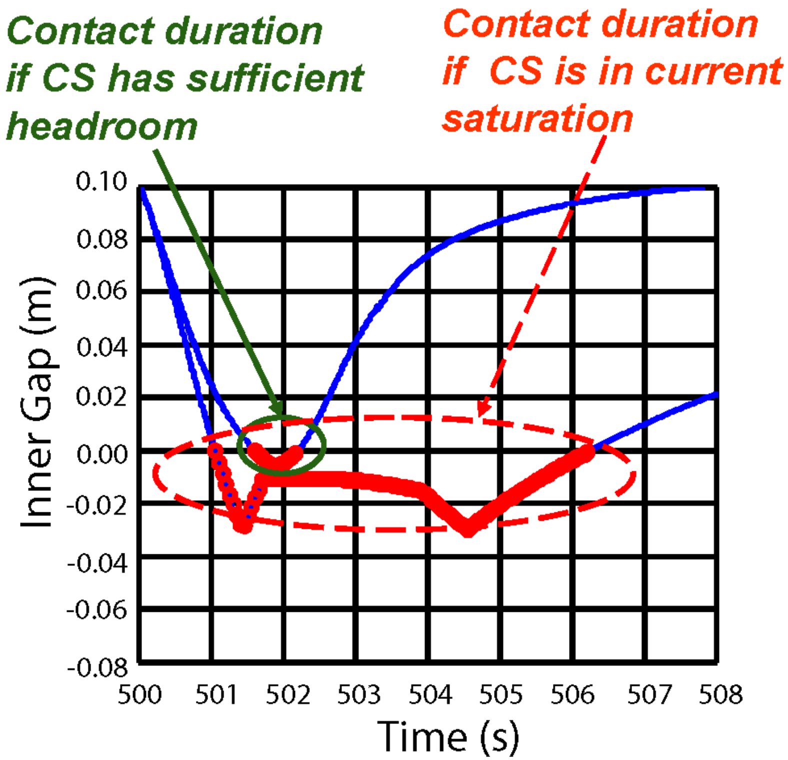
Divertor "detachment" is fundamental to exhaust power in a burning plasma environment:

- 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





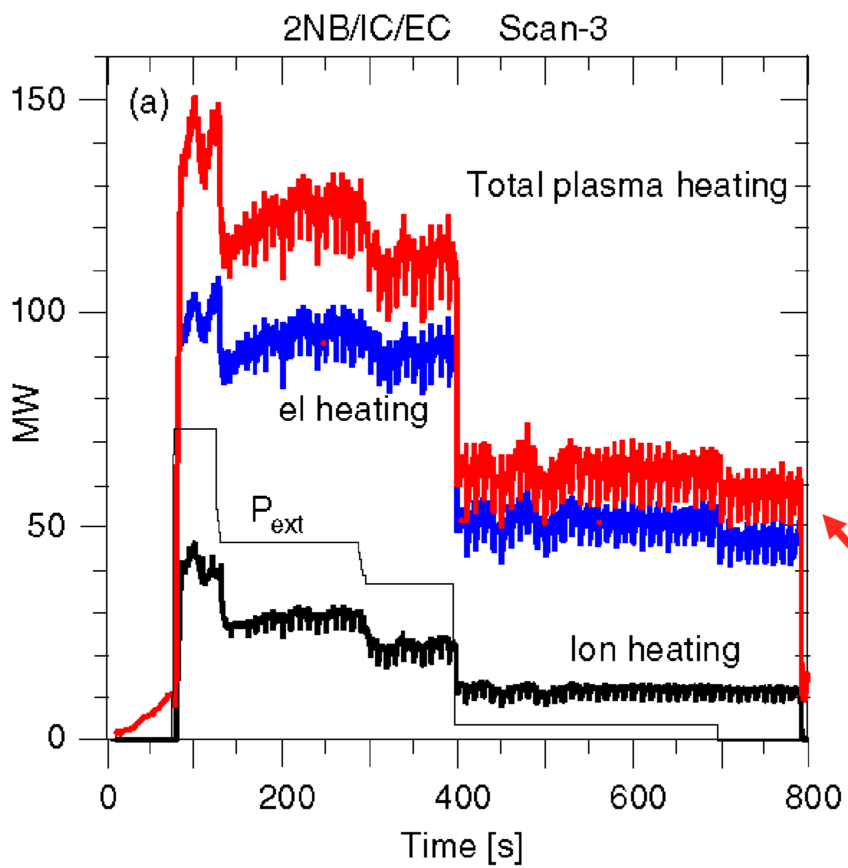
- Radial inward displacement can be ≥ 10cm ➔ contact with the inner wall
- Duration of inner wall contact depends on the central solenoid saturation state
- Peak engineering heat loads of ~40MW/m^2 ➔ Be tiles would melt in ~ 0.3 s!
- PCS must maintain large enough gaps or trigger the disruption mitigation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

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?

Simulations Show Fusion Burn is Stable in ITER

Simulated Burn Control in ITER



Budny, NF 2009

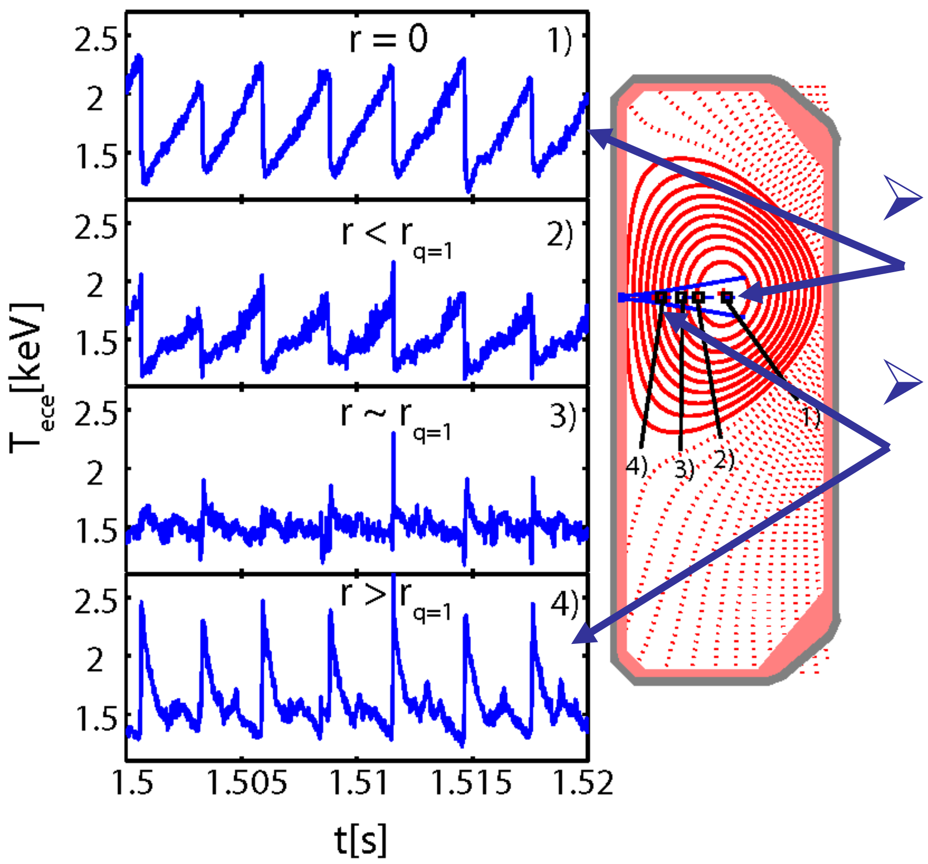
- Dominant α-particle heating at Q=10 requires reliable fusion burn control schemes controlling the core D/T mix with pellet injection, helium ash, and other core impurities
- Auxiliary heating power may also be used for secondary fusion burn control
- Simulations show that the fusion burn is stable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

4) Non-Axisymmetric Mode Control

- 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 are 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

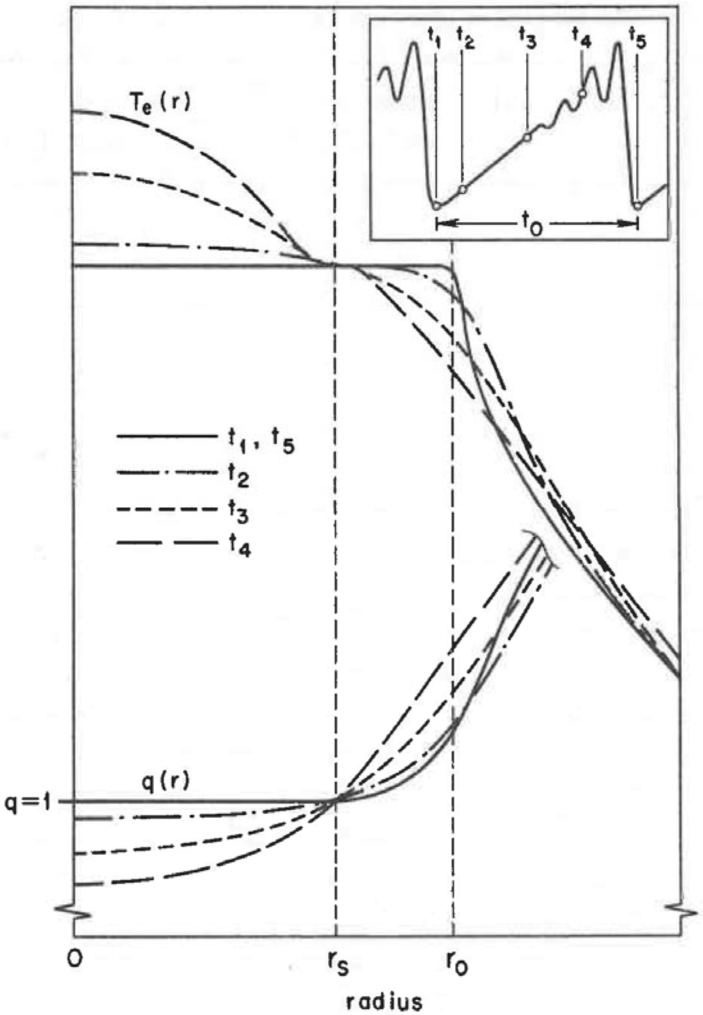


P Blanchard, PhD thesis, EPFL (2002)

- 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?

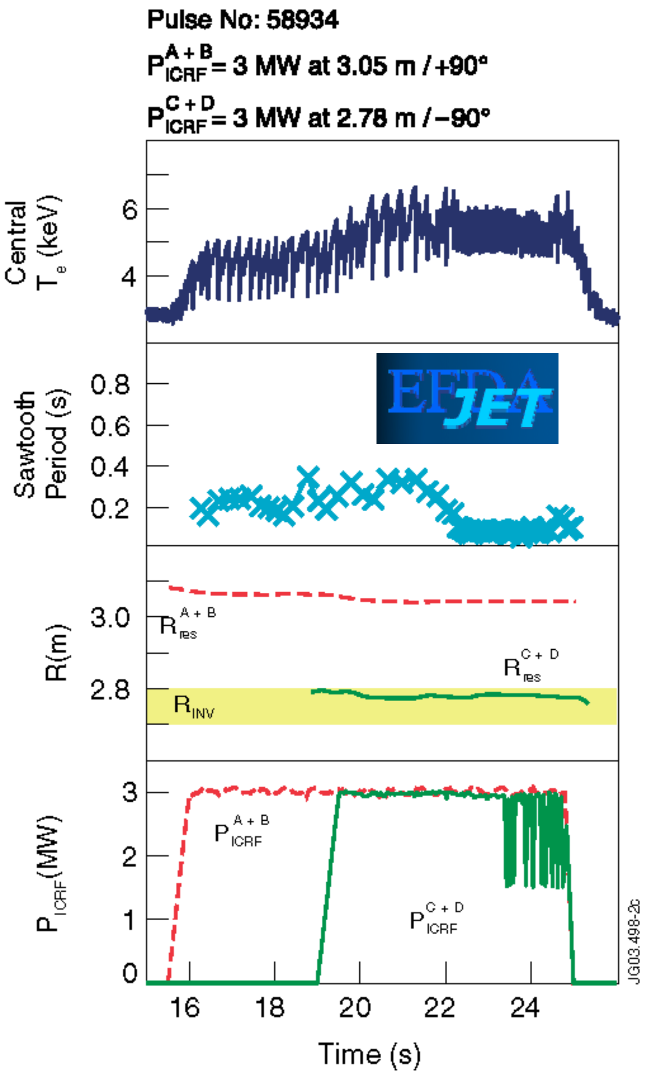
Model T_e and q Profiles During a Sawtooth



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

- 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
- 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

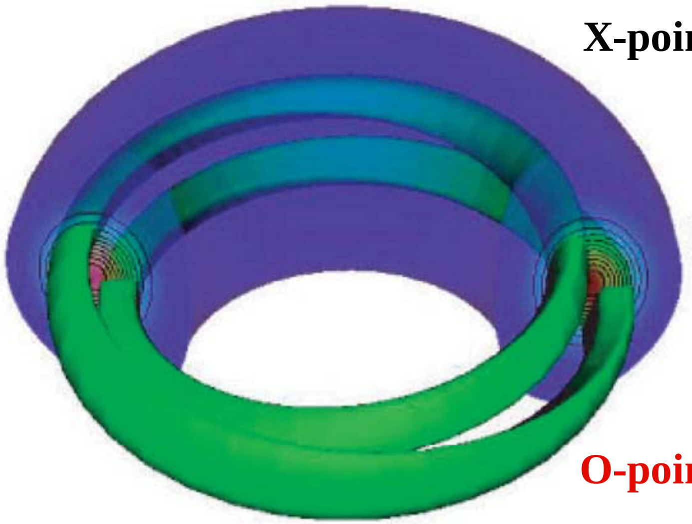
Sawtooth Control Has Been Demonstrated



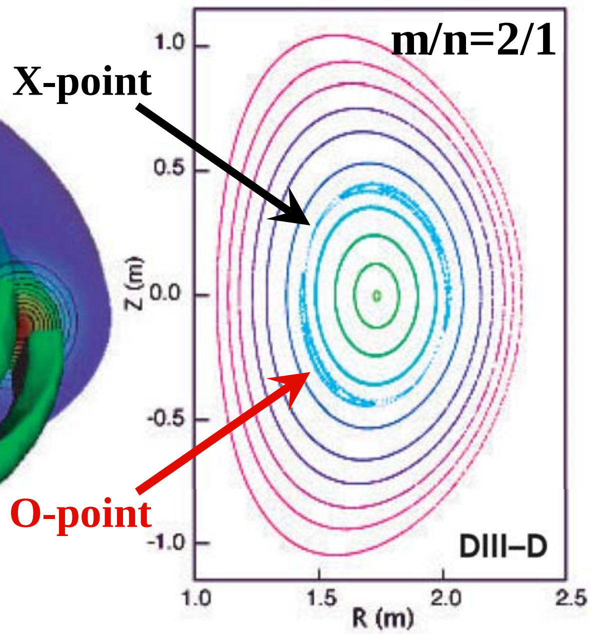
Pamela, et al., *NF* **45** (2005) S63

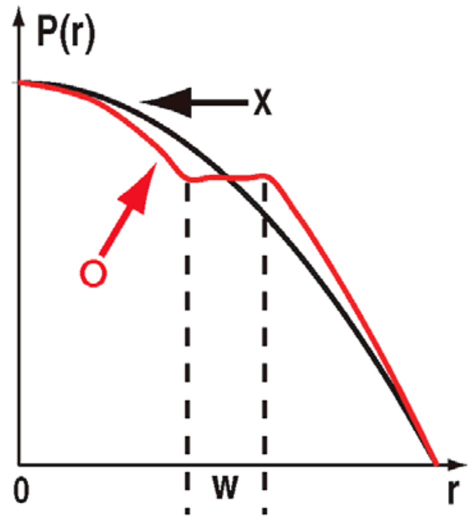
- Sawtooth control was demonstrated on JET with +90° ICRF phasing to create fast ions to partially stabilize sawteeth
  ➔ 'monster' sawteeth
- Then -90° ICRF phasing was added to destabilize sawteeth reducing the sawtooth period and amplitude
- ITER actuators for sawtooth control include ICRF and localized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

What are Neoclassical Tearing Modes?



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

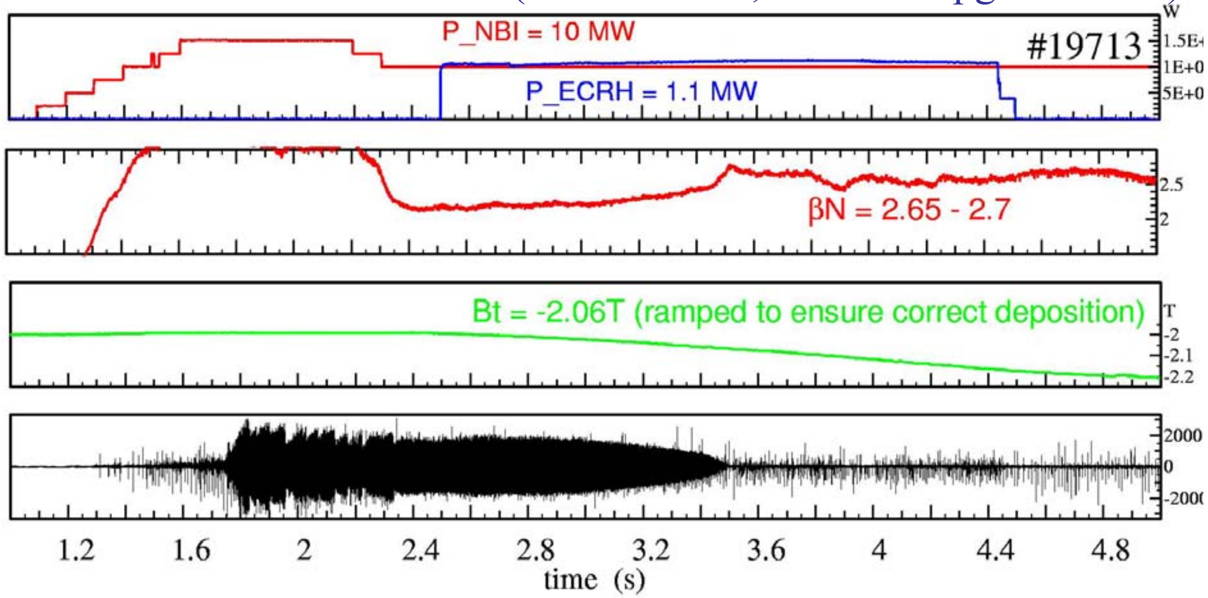




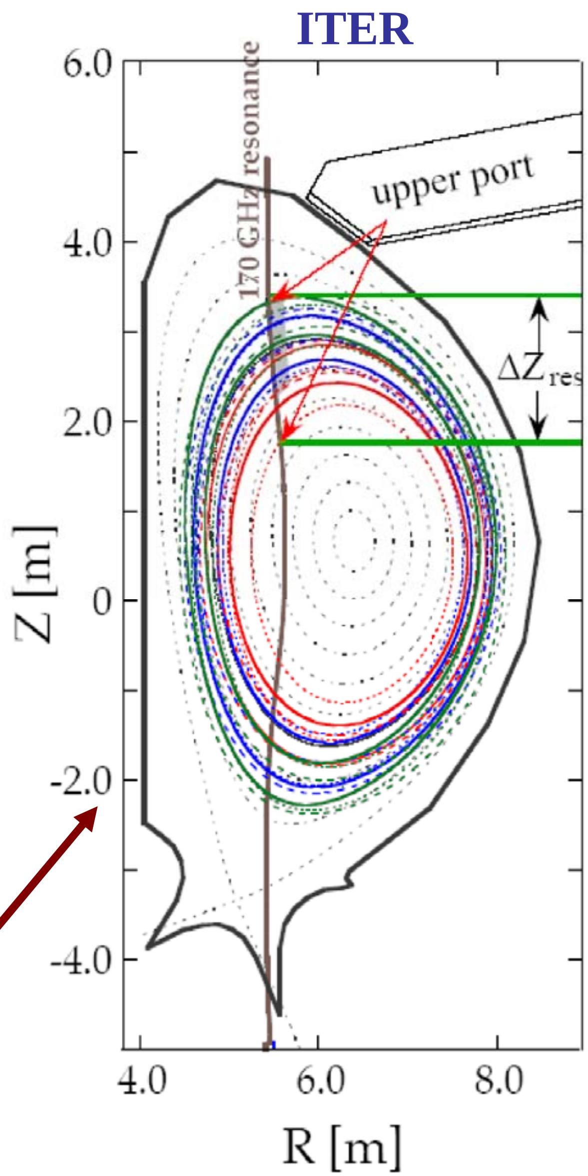
- Finite plasma resistivity allows toroidally non-axisymmetric helical currents to break or tear magnetic field lines at rational surfaces q = m/n (➔ a tearing mode)
- Field line reconnection creates magnetic islands and rapid energy transport along the field line flattens the pressure profile across 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

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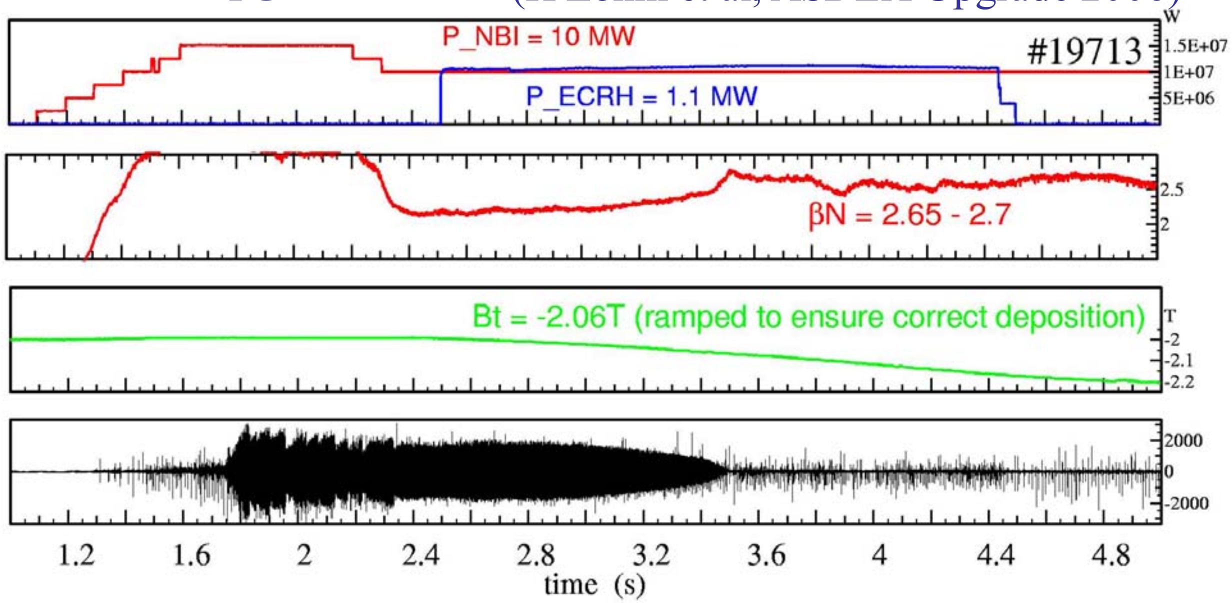
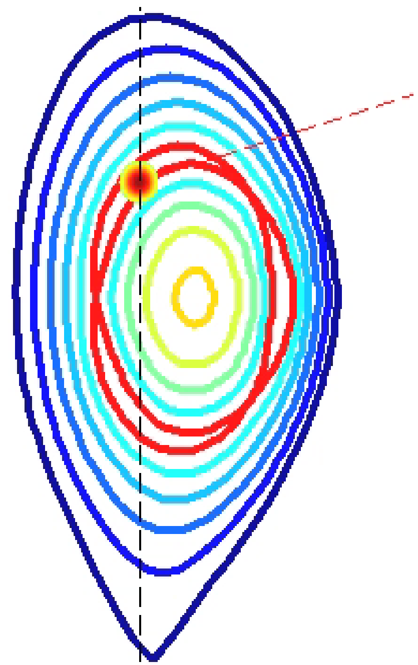


ITER



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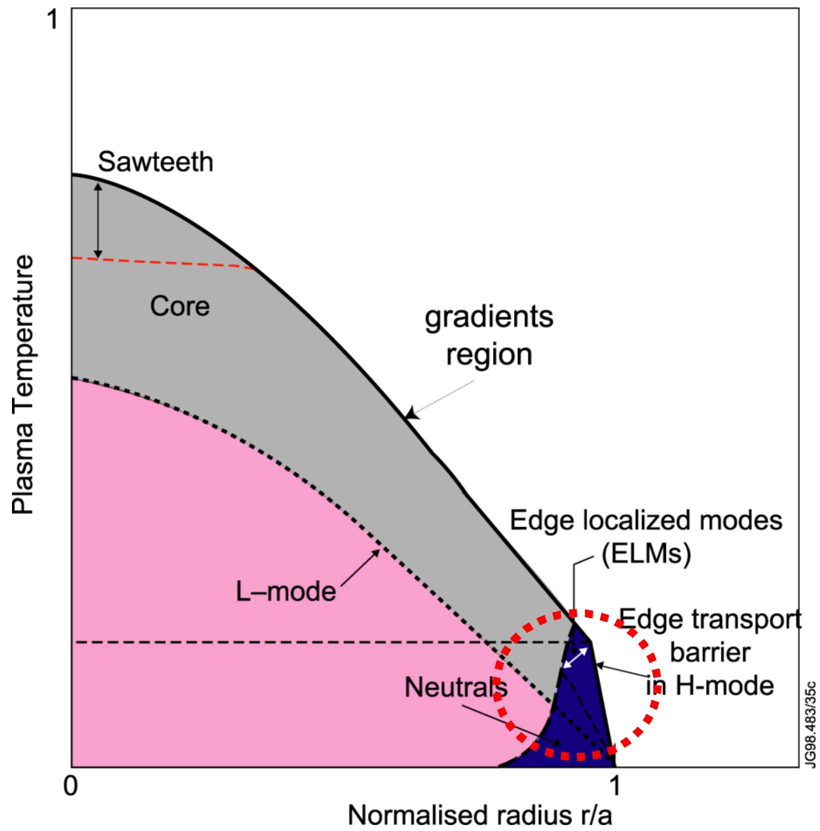
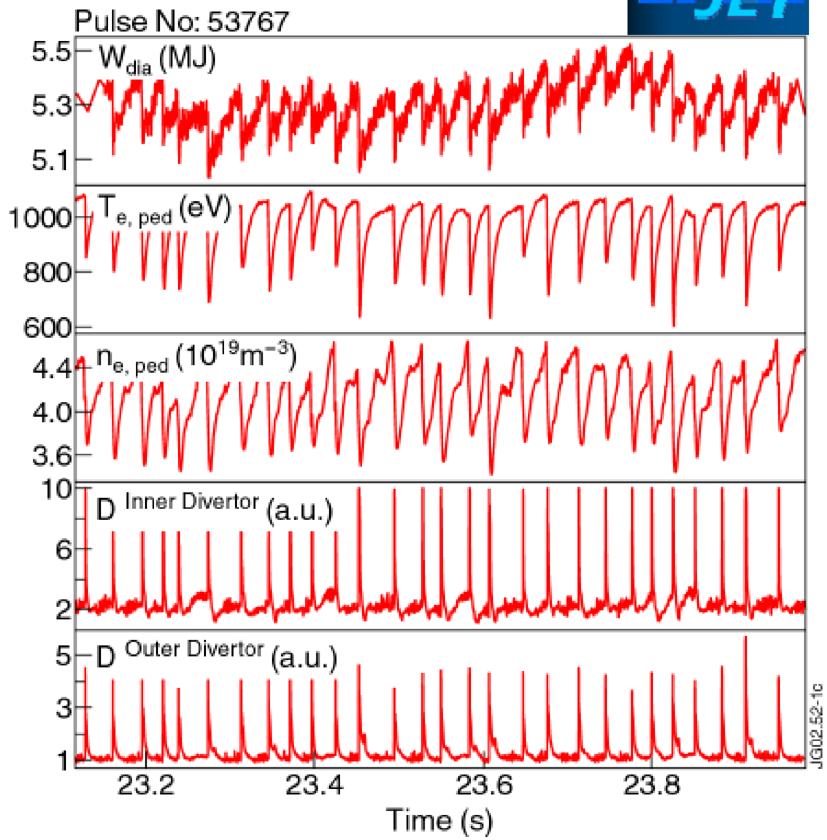
R LaHaye, APS 2005

What are Edge Localized Modes (ELMs)?

ELMs are rapid disturbances of the edge temperature and density

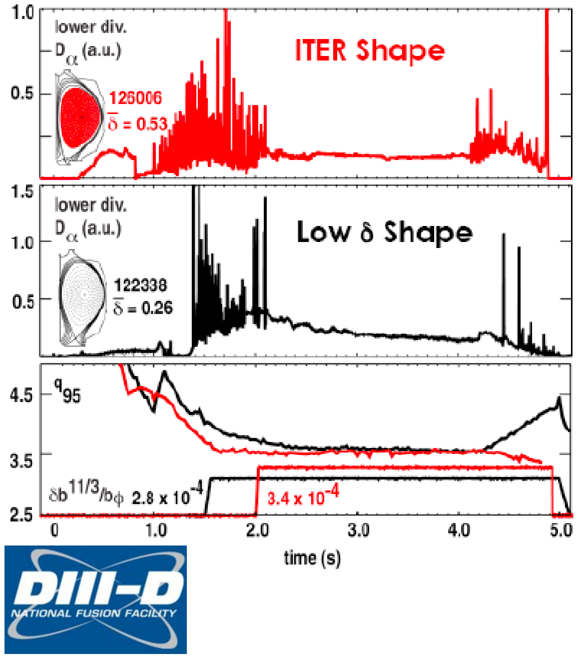
- destabilized when the edge pressure gradient becomes too steep
- yield very high transient heat and particle flux on wall and divertor
- maintain the plasma in a quasi-stationary state



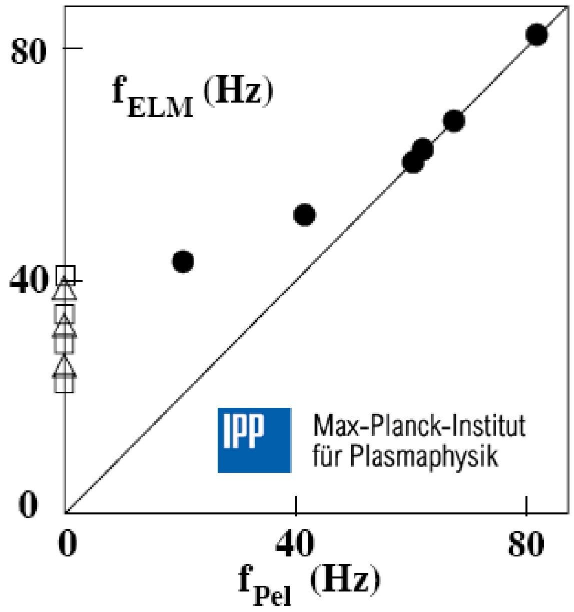


First Wall Heat Load: ELM Control/ Mitigation is Critical

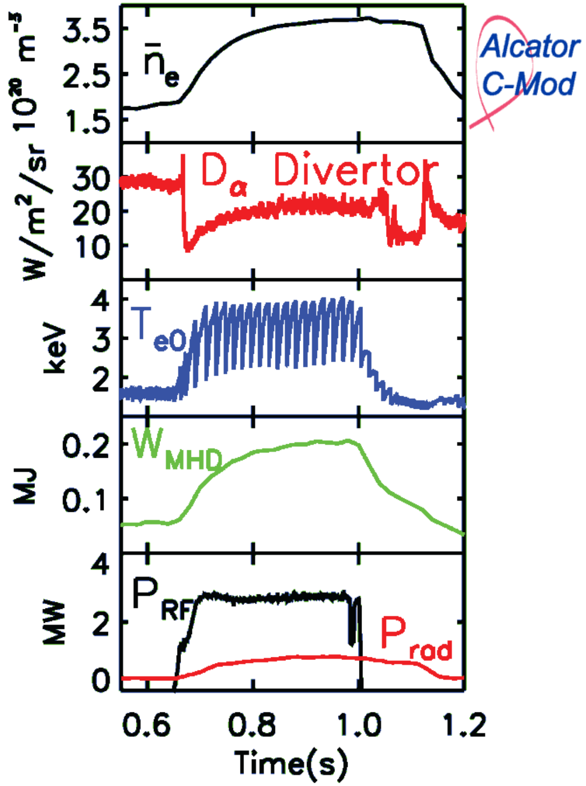
DIII-D Magnetic Control



AUG Pellet Pacemaking



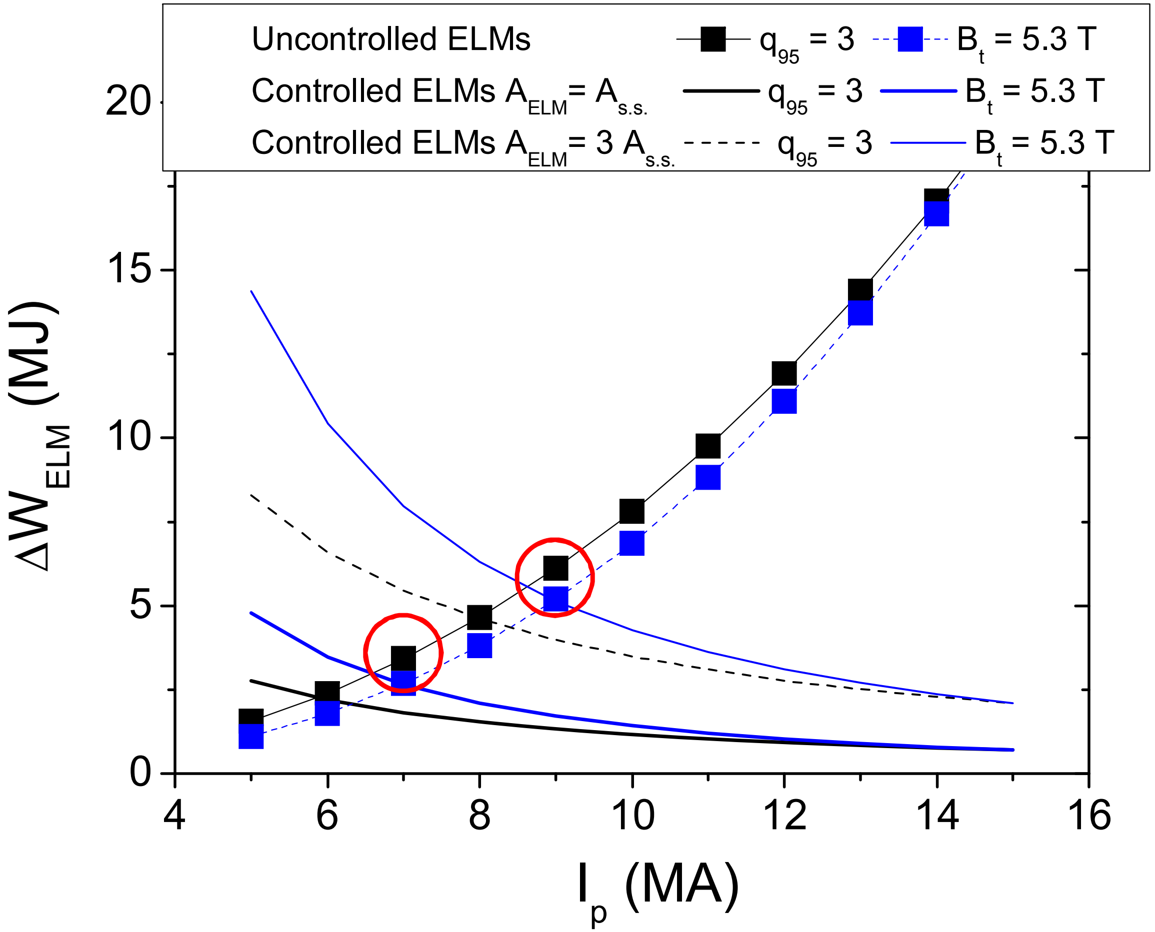
C-Mod EDA H-mode



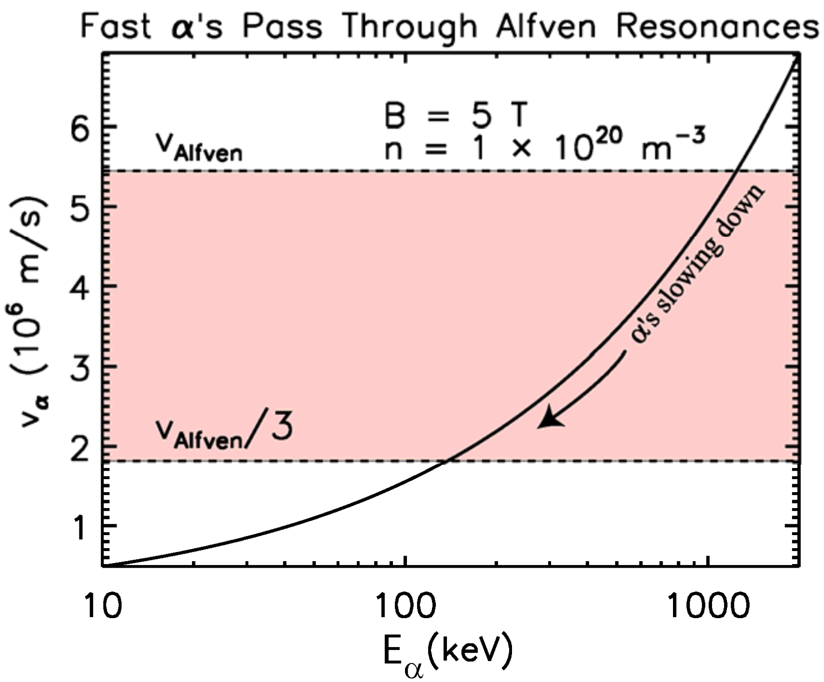
- ELM control is needed to substantially reduce divertor heat loads to enhance the divertor lifetime
- ITER will use in-vessel ELM coils and pellet pacing for ELM control
- Steady-state ELM-free regimes may also be found on ITER

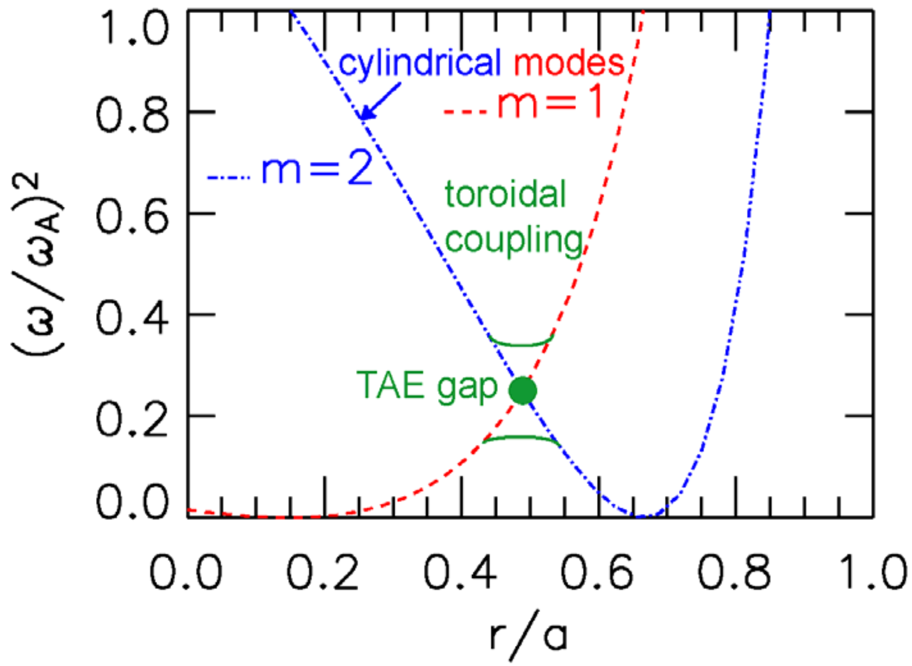
ELM Control Required for High Current Operation

- 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?



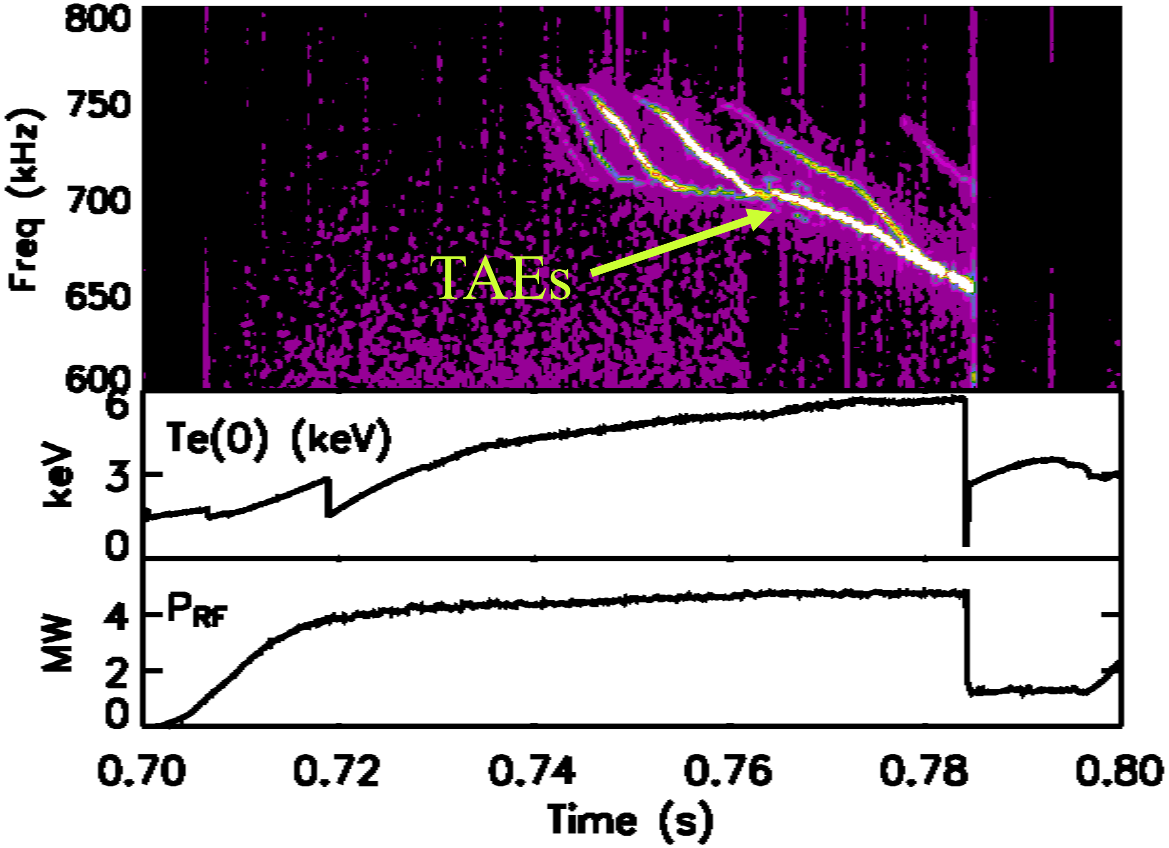


- Energetic particles with specific resonances (v_A, $v_A/3$) e.g., α particles slowing down excite Alfvén modes in gaps in the continuum spectrum where damping is weaker ➔

$$\omega^2(r) = k_{\parallel}^2(r) v_A^2(r)$$
$$\omega_A = v_A(0)/(q_a R_0)$$
$$\propto B_T / (q_a R_0 \sqrt{n_i m_i})$$

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

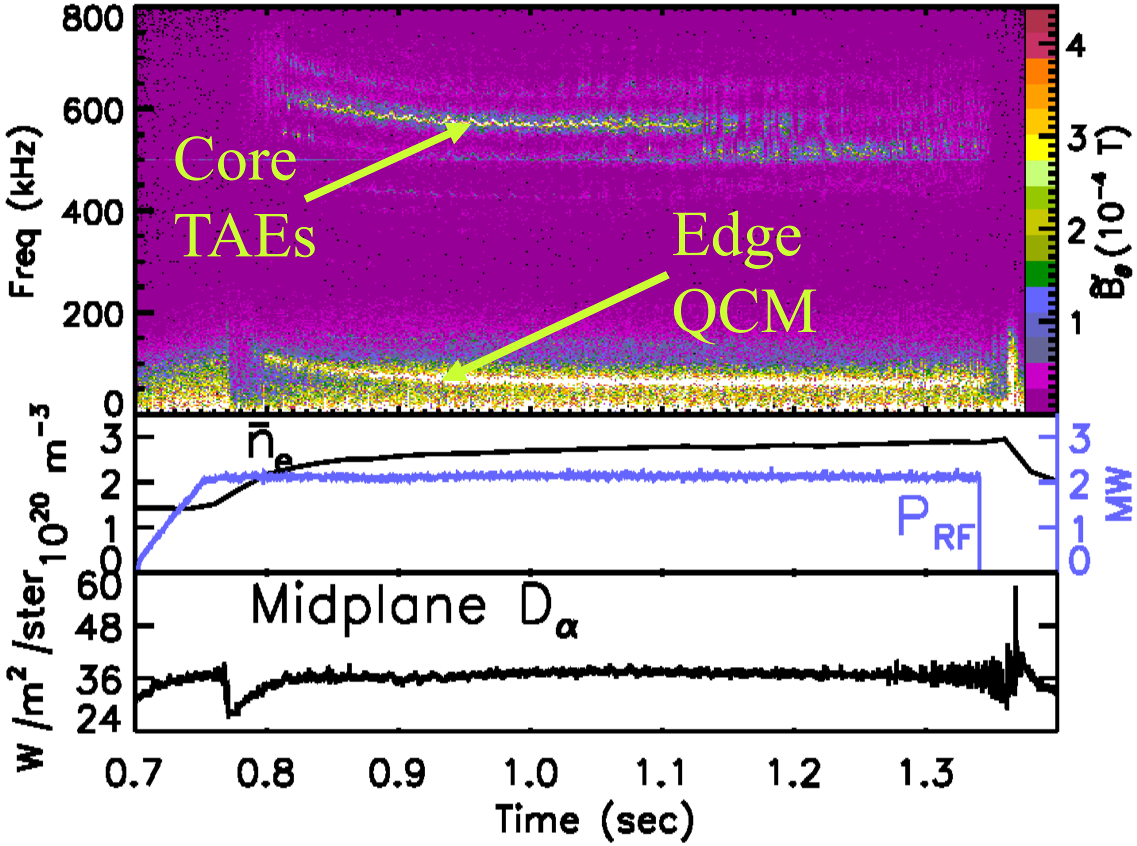
How Will Fast α-particles Affect Sawtooth Stability?



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

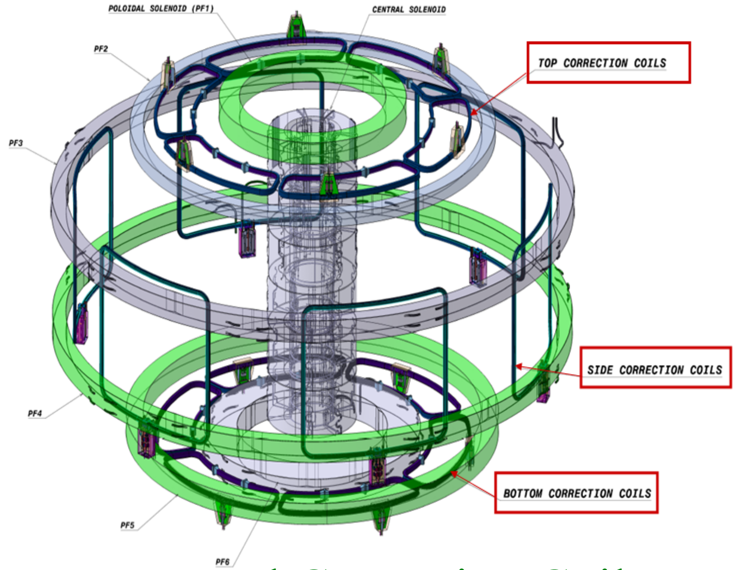
Will Fast α's Strongly Couple Modes Nonlinearly?



- 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?

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

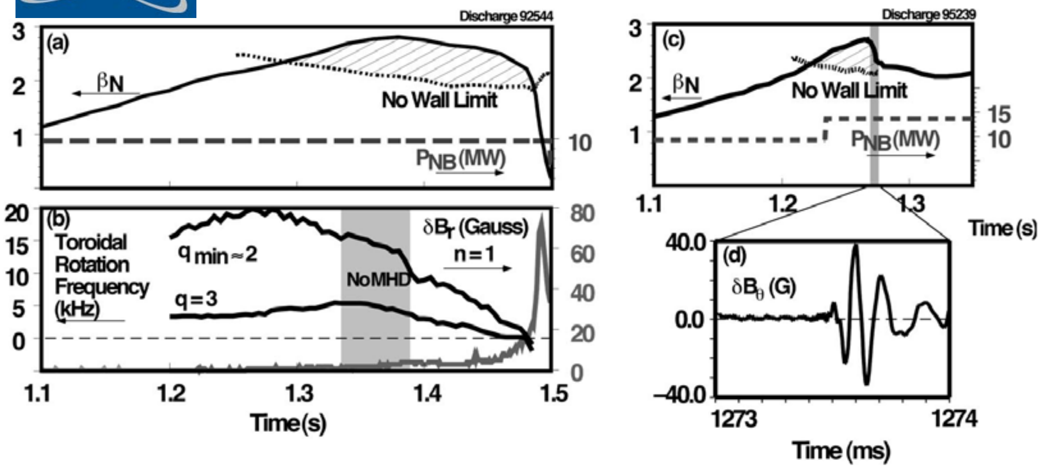


External Correction Coils

- 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 within the 320 kAt top & bottom and 200 kAt side current limits together with in-vessel ELM coils to correct a broad error field spectrum

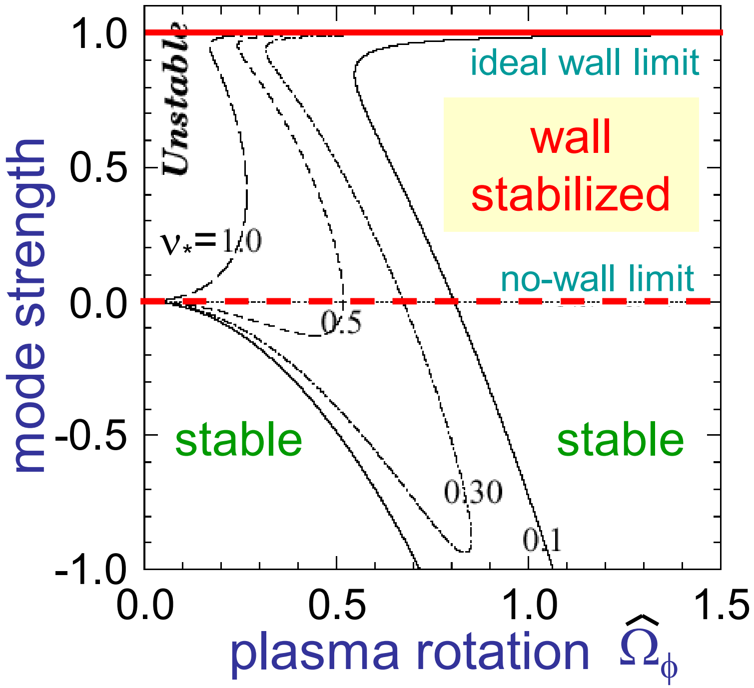
What are Resistive Wall Modes?

Garofalo, Phys Plasmas 1999



Fitzpatrick-Aydemir (F-A) stability curves

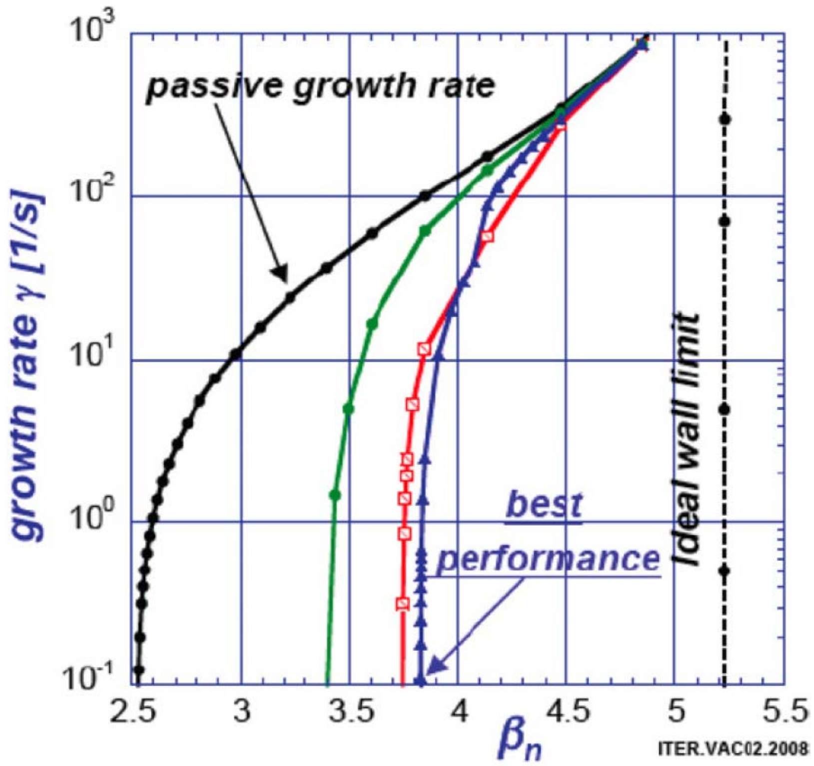
Phys. Plasmas **9** (2002) 3459



- 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

Resistive Wall Mode Control Allows High β Operation

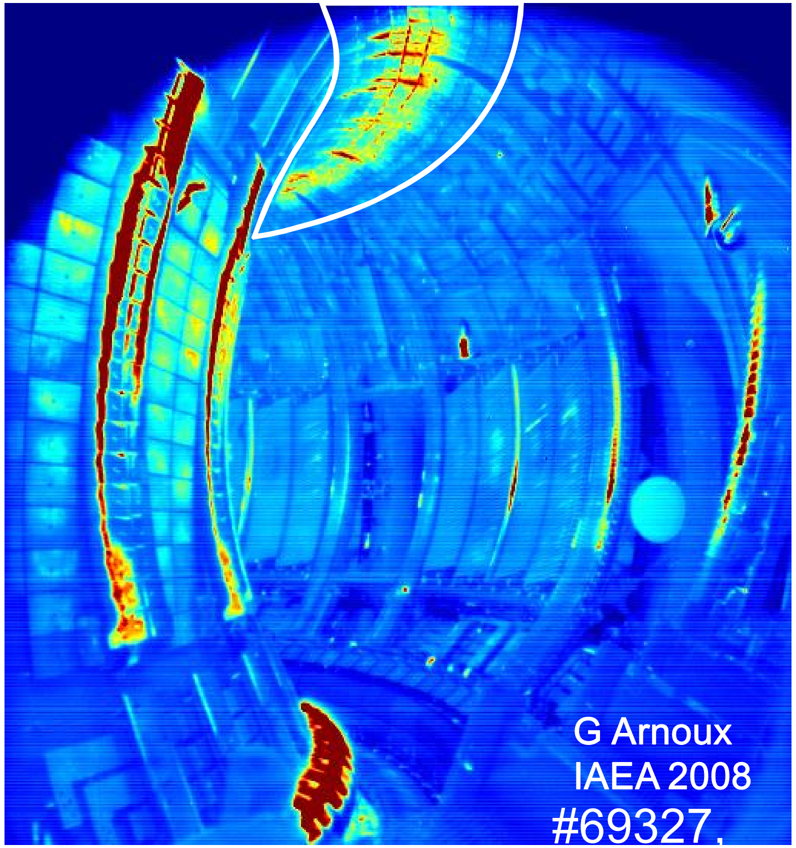
RWM Control: Hawryluk, NF 2009



- 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

Event Handling

Real-time Hot Spot Detection
Infrared View of JET Plasma



- 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 for machine protection
  - bridge segments – automatically switch to alternate control segments if the 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

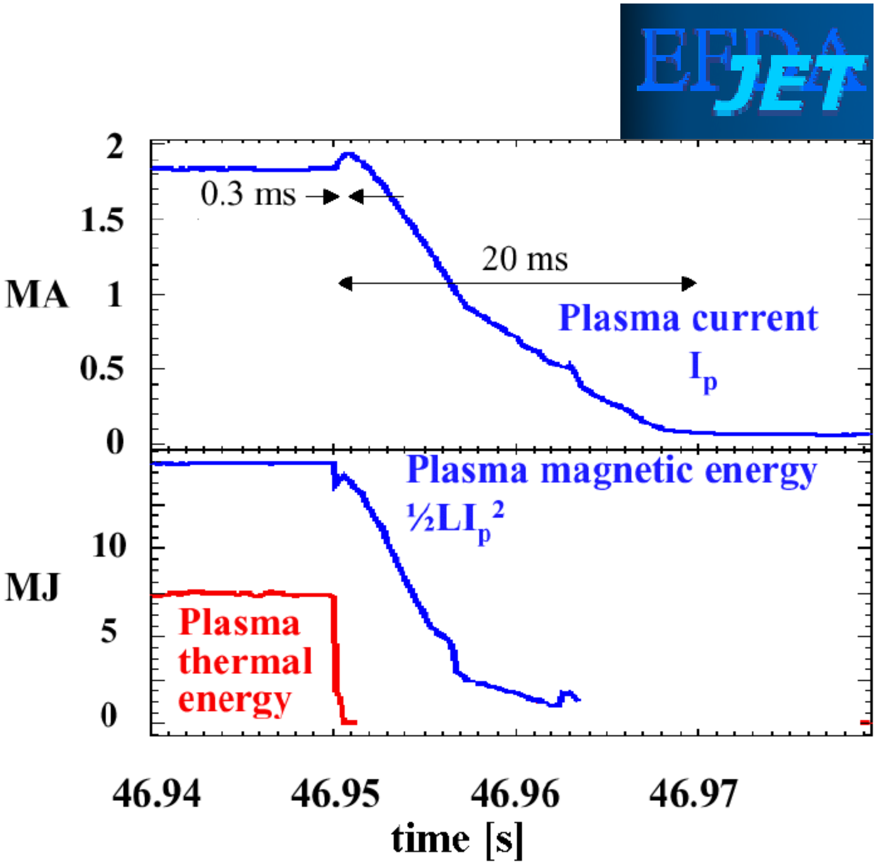
What are Disruptions?

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

⇒ unstable MHD modes grow

⇒ plasma confinement is destroyed (thermal quench)

⇒ plasma current vanishes (current quench)



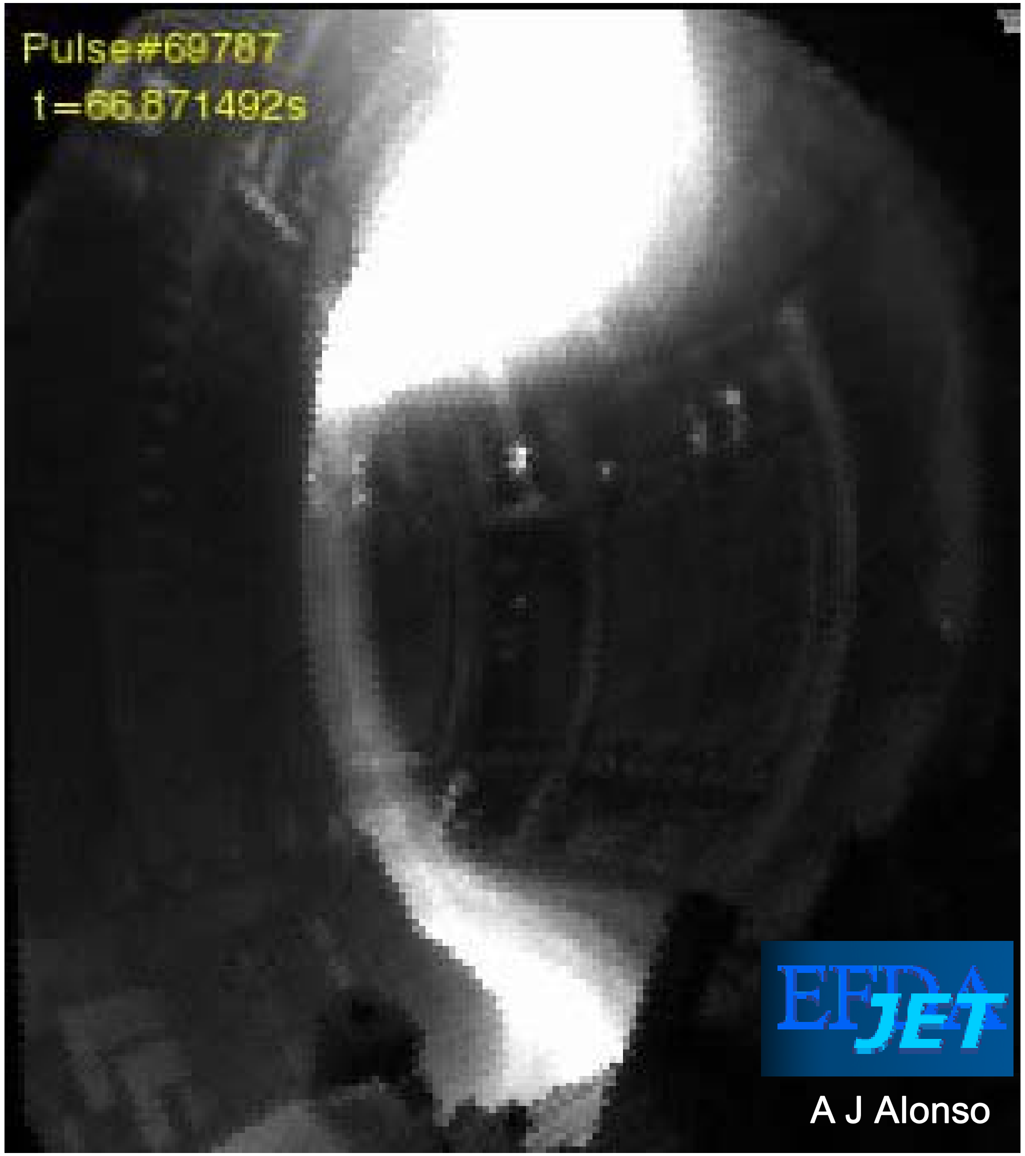
Typical JET timescales

- Thermal quench < 1ms ⇒ deposits plasma thermal energy on plasma facing components (PFCs)
- Current quench > 10 ms ⇒ 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

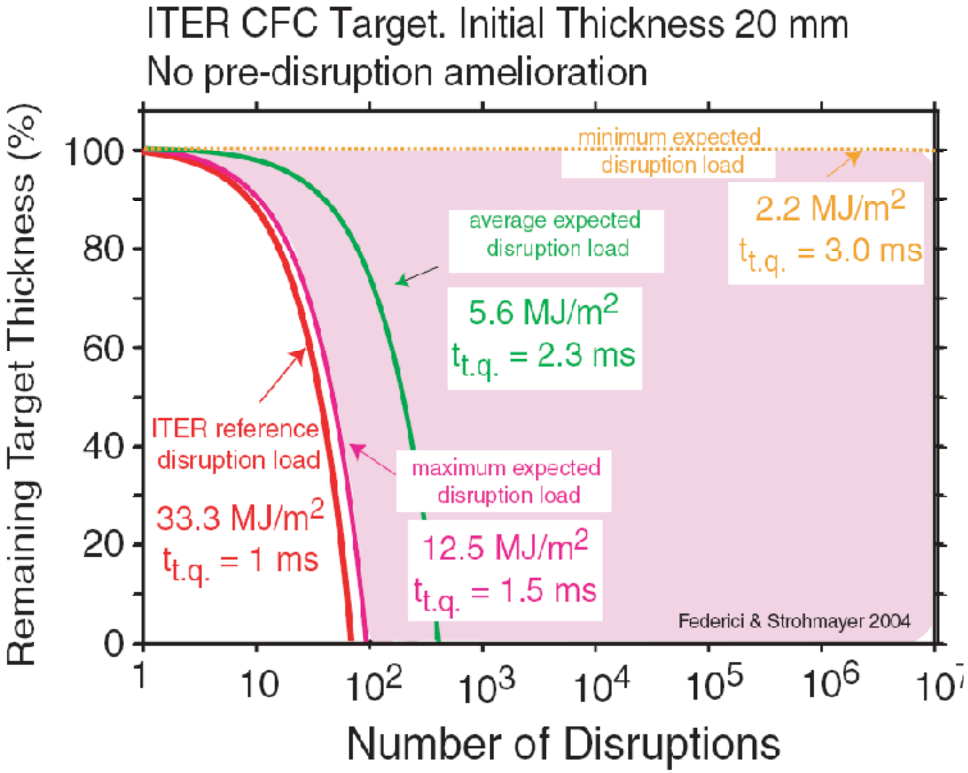
Current quench:

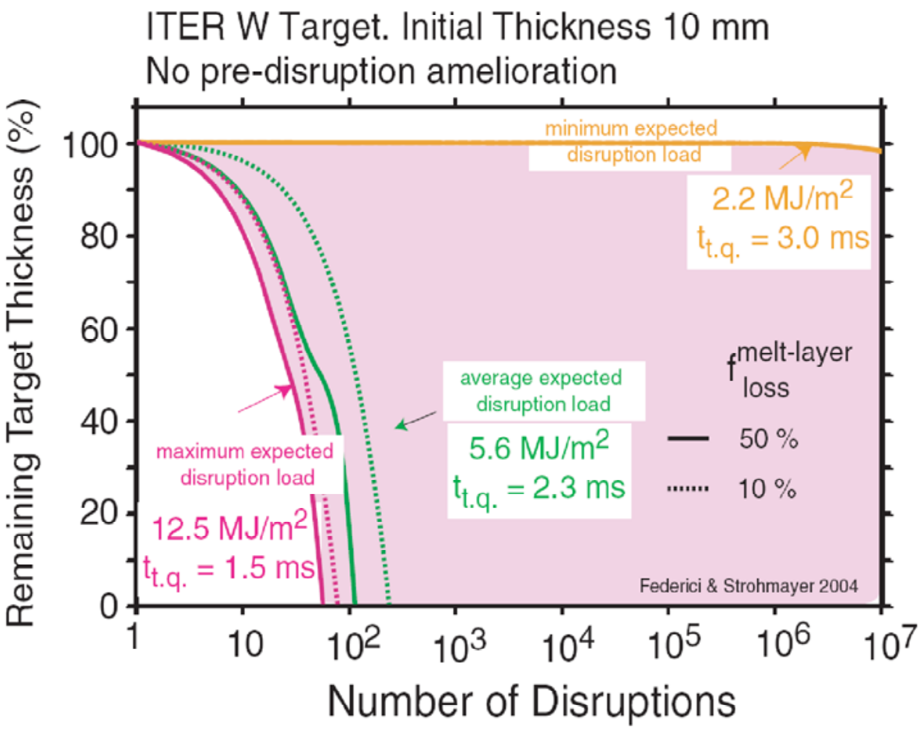
Large electromagnetic forces on the vacuum vessel and in-vessel components

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



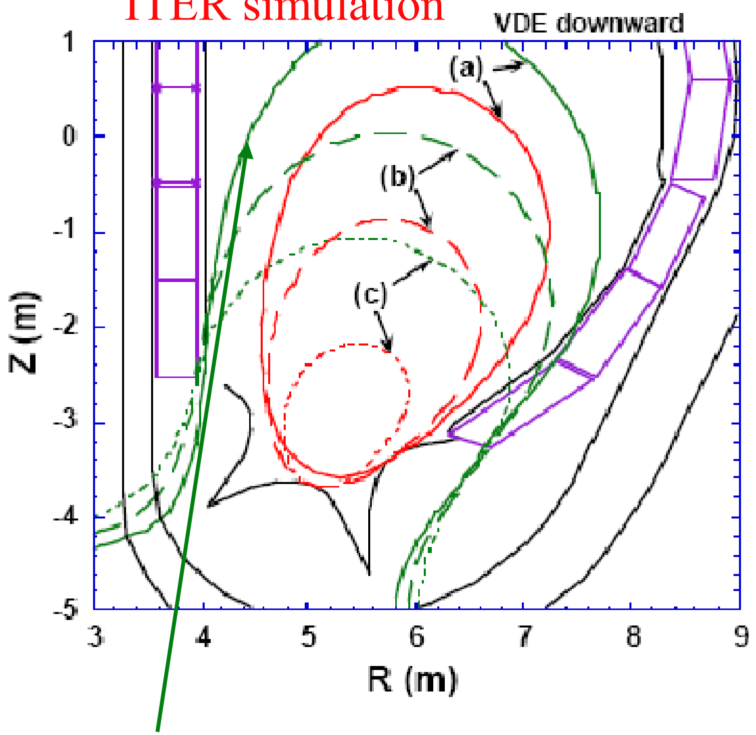


- ➢ The divertor may only withstand a (few) hundred Q=10 disruptions!

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



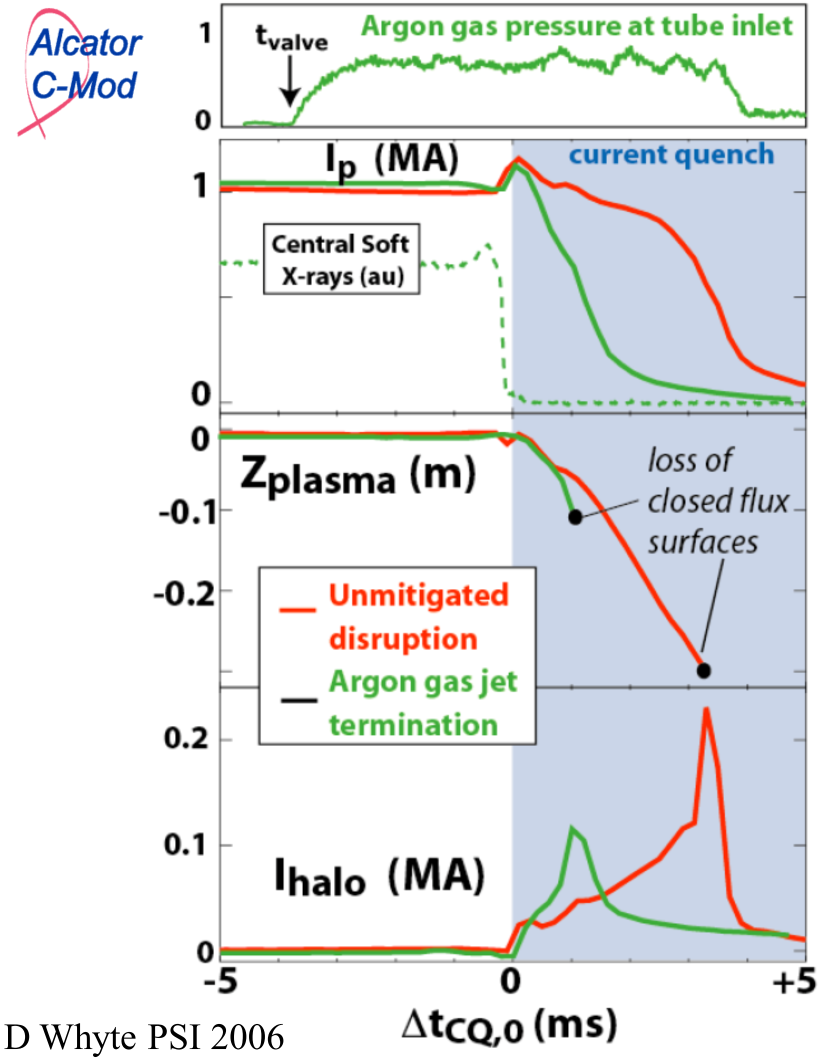
Halo current layer

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

How Can Disruption/VDE/Runaways be Mitigated?

Disruption Mitigation with massive impurity injection

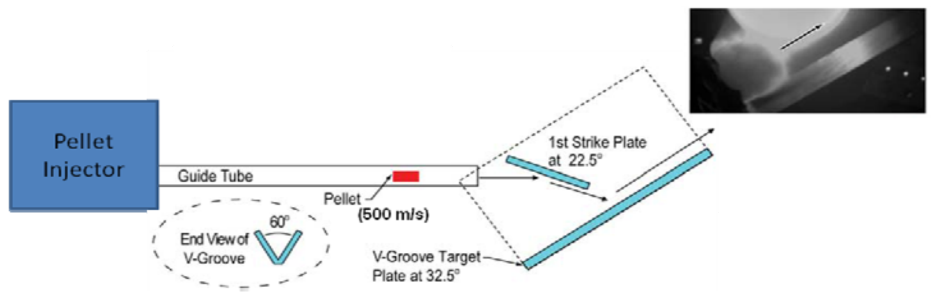


D Whyte PSI 2006

High pressure impurity gas or pellet 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

Pellet Injector Design for Disruption Mitigation



Pipe-gun concept with shattered pellets

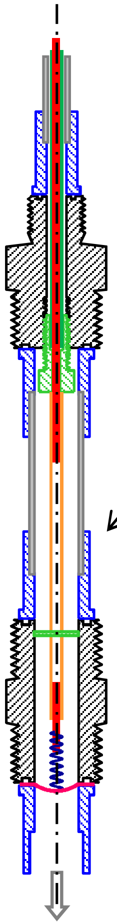
CAD model of top port multi-barrel injector

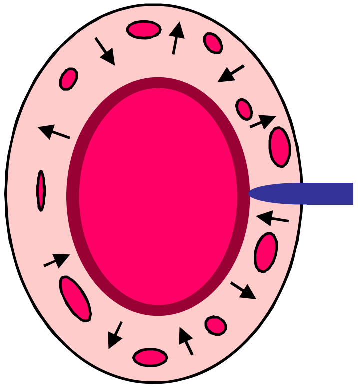
- Injector of large Ne or Ar cryogenic pellets is under development at ORNL
- Pellets injected in pre-thermal quench plasmas to mitigate energy loads
- Pellets shattered upon entry to vacuum vessel to improve impurity distribution
- The concept has been successfully tested on DIII-D

Suppression of RE electrons by repetitive gas injection

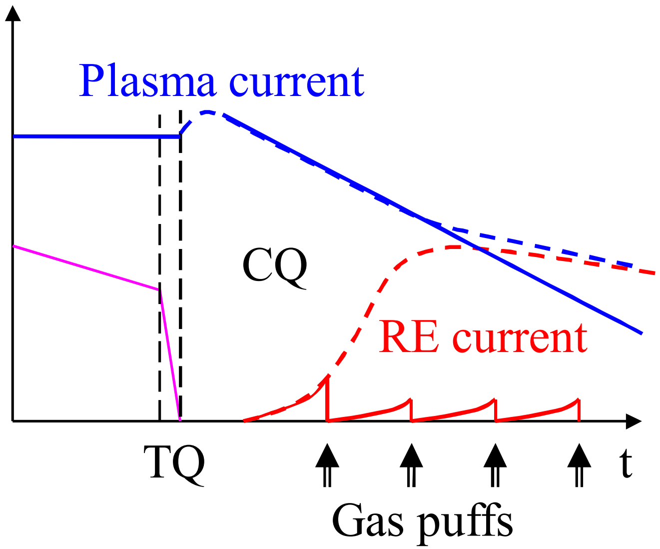
Large magnetic perturbations can be produced by dense gas jets injected repetitively into the current quench (CQ) plasma

Tore Supra valve with rupture disk





Dense and resistive gas jet contracts current channel



- Required gas pressure > 1 atm, gas amount ~1 kPa*m^3, 5-6 jets during CQ (staggered in time by 5 – 10 ms)
- Based on estimates the total amount of gas can be 10 times less than for collisional damping!
- Experiments are planned to test this scheme in Tore-Supra, ASDEX-Upgrade, and T-10

Conclusions

- ITER plasma operation 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 ~ 2027 ➔ today's students will make Q=10 and long pulse steady-state fusion regimes a reality