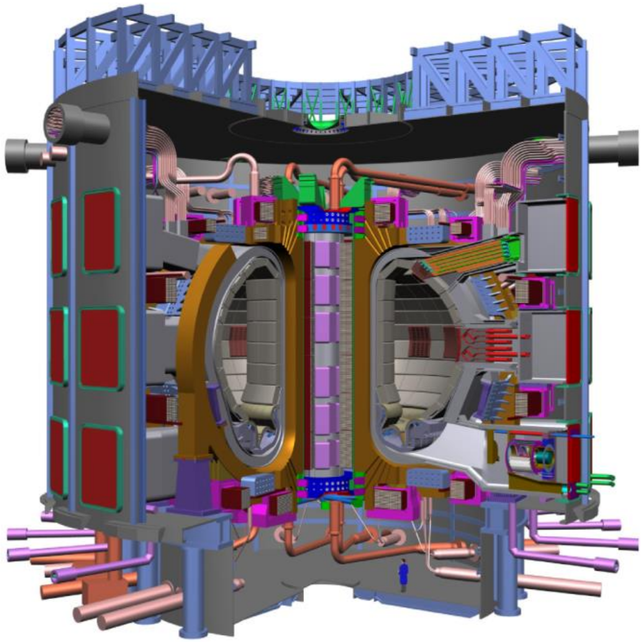


ITER Physics

D J Campbell
ITER Organization, Cadarache

Acknowledgements:
Many colleagues in the ITER IO, ITER PTs and ITPA

What is ITER ?

ITER is a major international collaboration in fusion energy research involving the EU (plus Switzerland, Romania, Bulgaria), China, India, Japan, the Russian Federation, South Korea and the United States

- **The overall programmatic objective:**
  - to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes

- **The principal goal:**
  - to design, construct and operate a tokamak experiment at a scale which satisfies this objective

- **ITER is designed to confine a DT plasma in which α-particle heating dominates all other forms of plasma heating:**

$\Rightarrow$ a burning plasma experiment

Synopsis

- **ITER Design:**
  - Key aims
  - Physics Basis
  - Principal aspects of device design
- **ITER Physics - Opportunities and Challenges:**
  - Operational scenarios and control issues
  - MHD stability issues
  - Power and particle exhaust
  - Burning plasma physics in ITER
- **ITER Status**
- **Conclusions**

The ITER Design

ITER Design Goals

Physics:

- ITER is designed to produce a plasma dominated by α-particle heating
- produce a significant fusion power amplification factor ($Q \geq 10$) in long-pulse operation
- aim to achieve steady-state operation of a tokamak ($Q = 5$)
- retain the possibility of exploring 'controlled ignition' ($Q \geq 30$)

Technology:

- demonstrate integrated operation of technologies for a fusion power plant
- test components required for a fusion power plant
- test concepts for a tritium breeding module

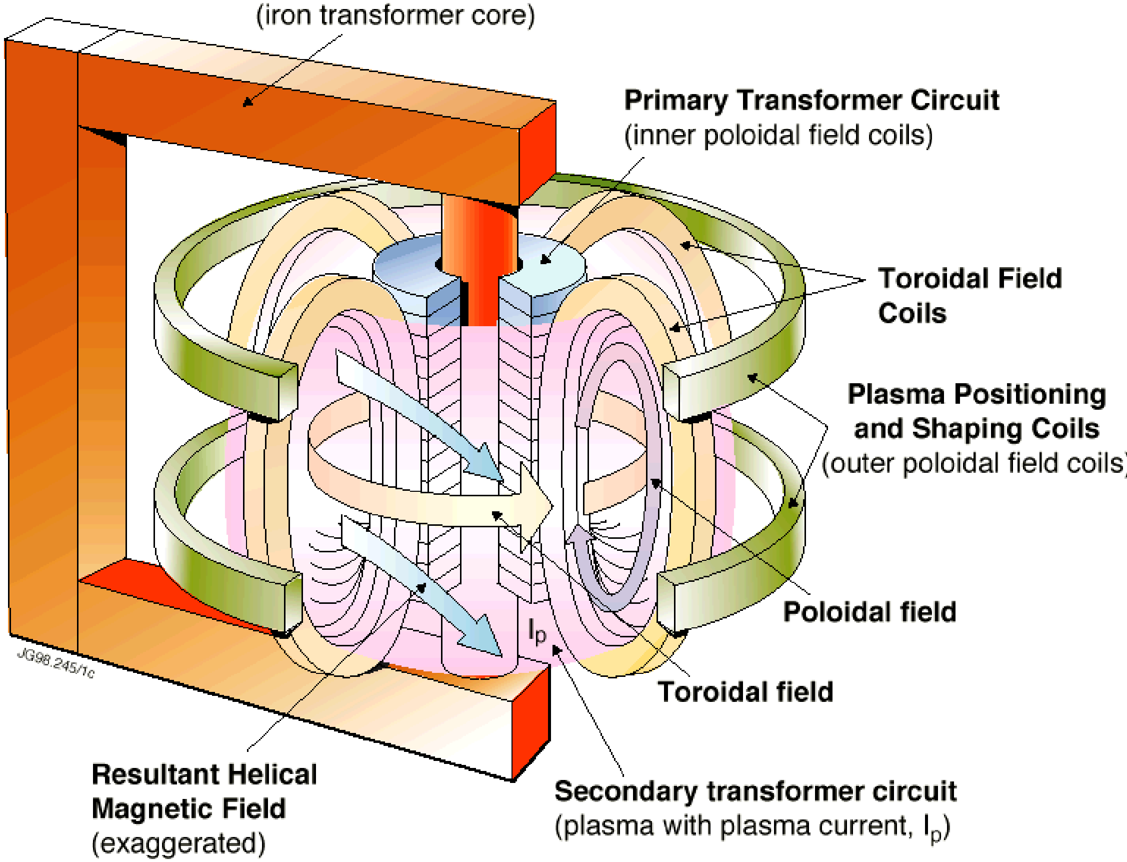
The Tokamak

The major <u>toroidal magnetic confinement</u> configuration

The Tokamak:

- operationally, is essentially an electrical transformer
- toroidal magnetic field is produced by external magnetic field coils
- plasma current produces poloidal magnetic field
- result is a set of nested helical surfaces

  ⇒ plasma confinement

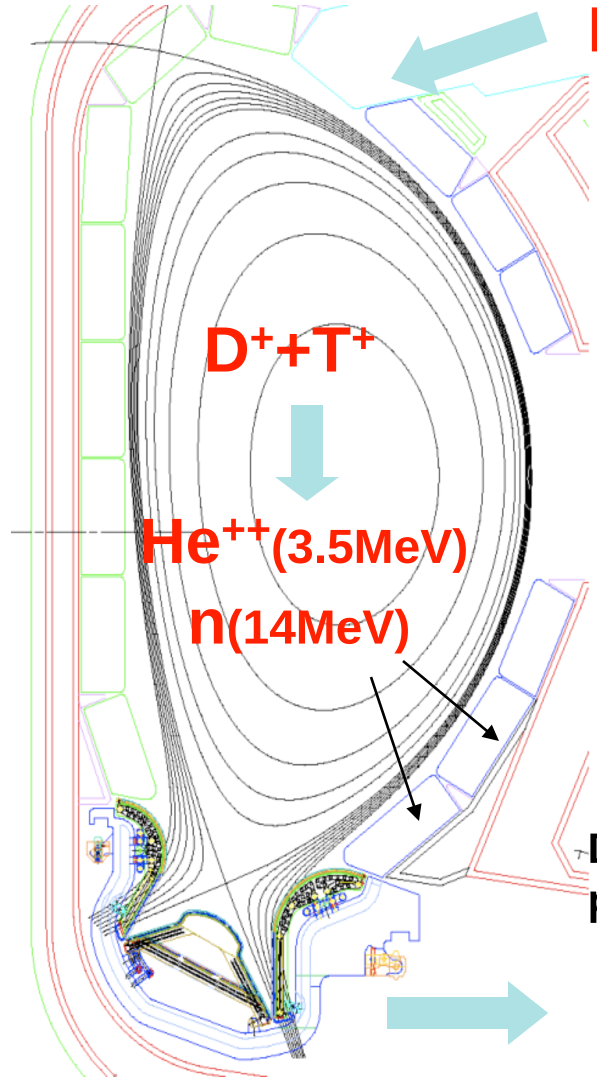


Fusion in a Tokamak Plasma

Toroidal Plasma:

Volume:	830m^3
R/a:	6.2m /2m
Plasma Current:	15MA
Toroidal field:	5.3T
Density:	10^{20}m^{-3}
PeakTemperature:	20keV
Fusion Power:	500MW

D_2,T_2 Fuel



Blanket:
neutron absorber

Power Plant:
Li $\Rightarrow$ T
high temperature

Divertor:
particle and heat exhaust

He, D_2,T_2,
impurities

Plasma Fusion Performance

Temperature (T_i): $1\text{-}2 \times 10^8$ °C (10-20 keV)
(~$10 \times$ temperature of sun's core)

Density (n_i): 1×10^{20} m^{-3}
(~10^{-6} of atmospheric particle density)

Energy confinement time (τ_E): few seconds
(plasma pulse duration ~1000s)

Fusion power amplification: $$Q = \frac{\text{Fusion Power}}{\text{Input Power}} \sim n_i T_i \tau_E$$

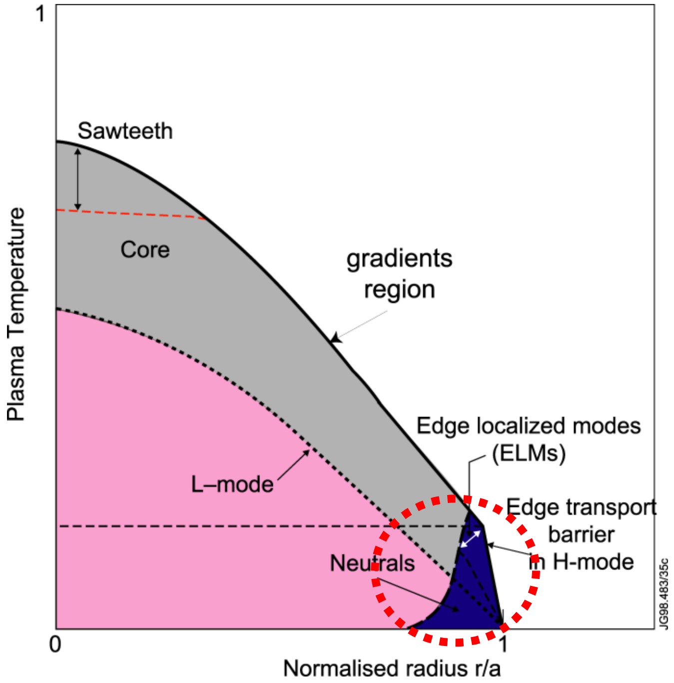
$\Rightarrow$ Present devices: $Q \leq 1$

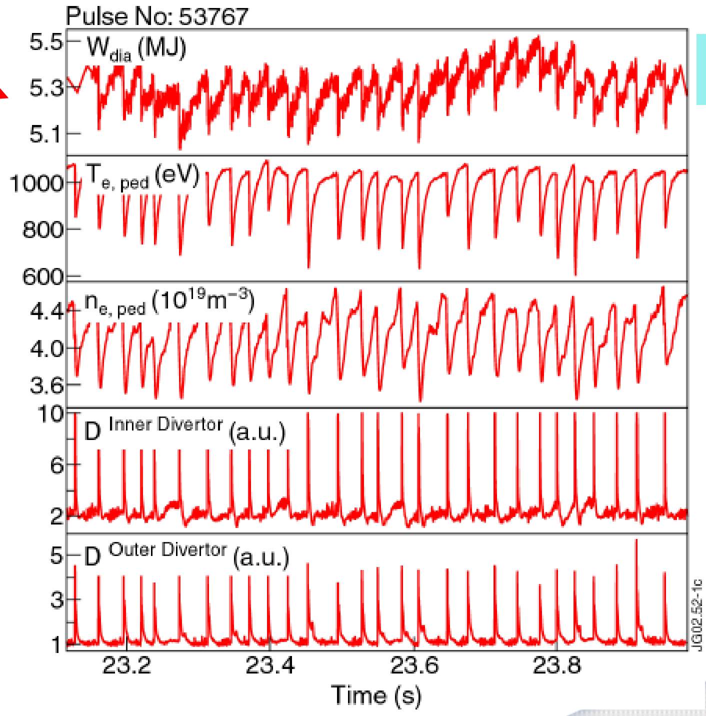
$\Rightarrow$ ITER: $Q \geq 10$

$\Rightarrow$ "Controlled ignition": $Q \geq 30$

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

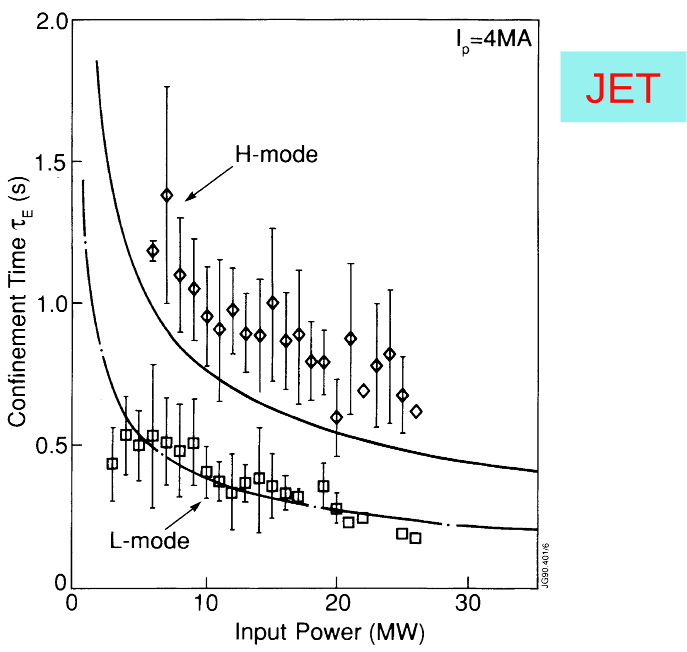


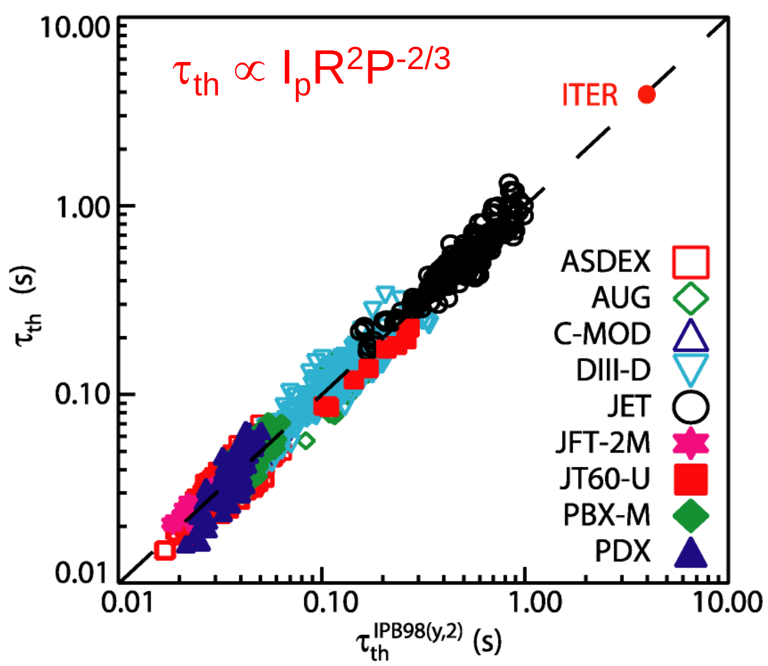


JET

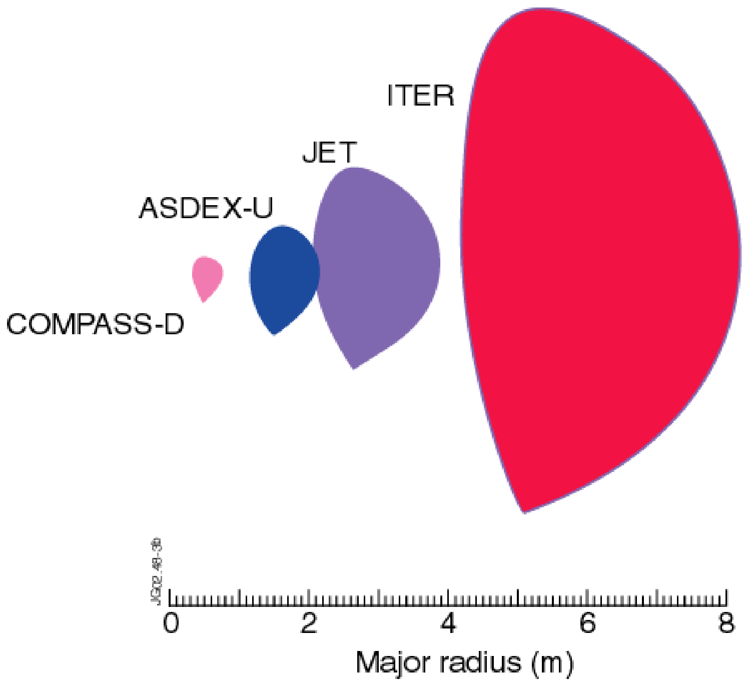
ITER Confinement Time - ELMy H-mode

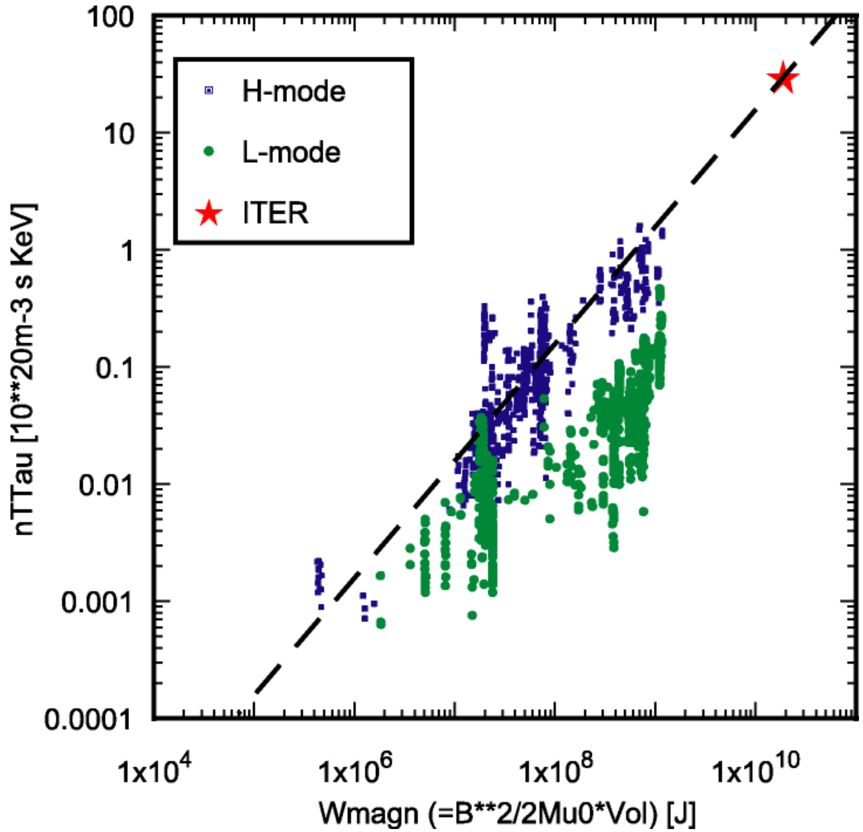
- **The ELMy H-mode is a robust mode of tokamak operation - ITER baseline scenario**
  - H-mode confinement time is approximately double that in L-mode
  - multi-machine database provides scaling prediction for ITER energy confinement time





How big should ITER be?





- **Confinement scaling studies provide the robust approach to determining ITER's size:**
  - detailed design relies on numerical codes combining engineering and physics constraints

ITER Physics Basis I

- **Predictions of fusion performance in ITER rely essentially on a small number of physics rules:**
  - Energy confinement scaling (IPB98(y,2)):

$$\tau_{E,th}^{98(y,2)} = 0.144\, I^{0.93} B^{0.15} P^{-0.69} n^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa^{0.78} \quad \text{(s)}$$

$$\tau_E \propto I R^2 P^{-2/3}$$

  - H-mode threshold power:

$$P_{LH} = 2.84 M^{-1} B^{0.82} \bar{n}_{20}^{0.58} R a^{0.81} \quad \text{(MW)}$$

ITER Physics Basis II

- MHD stability:

$$q_{95} = 3 \qquad q_{95} = 2.5\,\frac{a^2 B}{RI} f(\varepsilon,\kappa,\delta)$$

$$n/n_{GW} \le 1 \qquad n_{GW}(10^{20}) = \frac{I(MA)}{\pi a^2}$$

$$\beta_N \le 2.5 \qquad \beta_N = \beta(\%)\frac{aB}{I(MA)}$$

κ, δ determined by control considerations

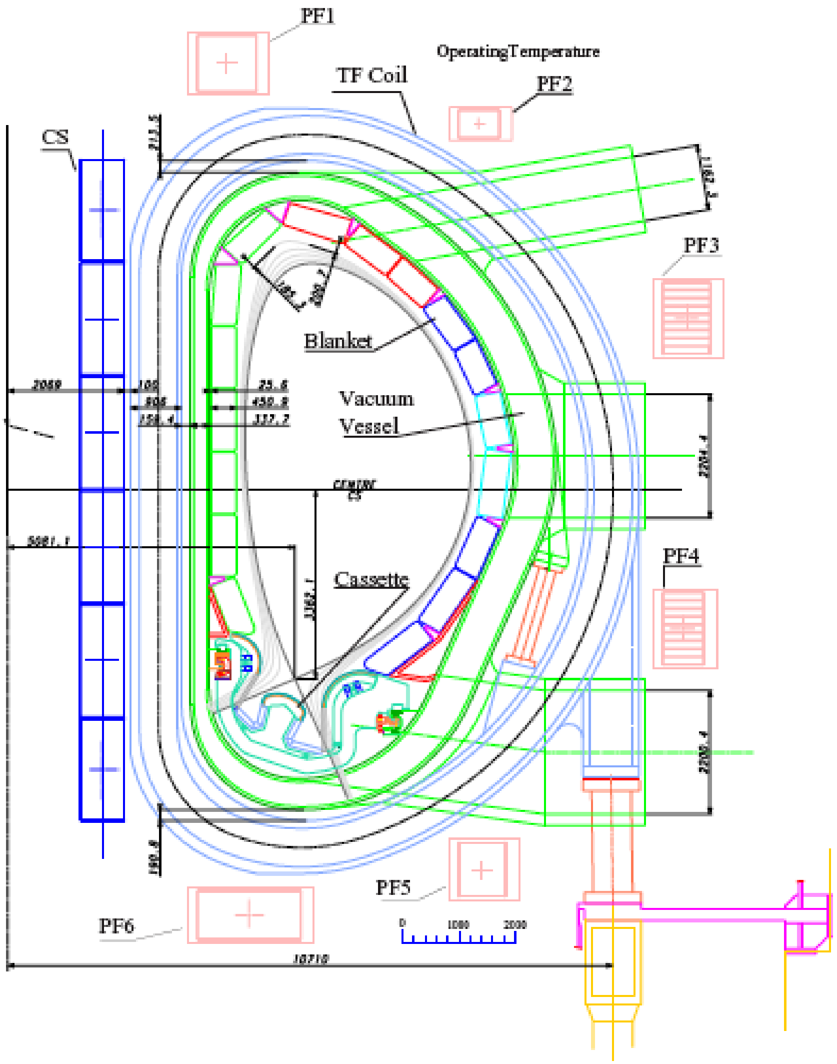
- Divertor physics:

Peak target power ~ $10 MWm^{-2}$

Helium transport : τ^*_{He}/τ_E ~ 5

Impurity content : $n_{Be}/n_e = 0.02$ (+ ~0.1% Ar for radiation)

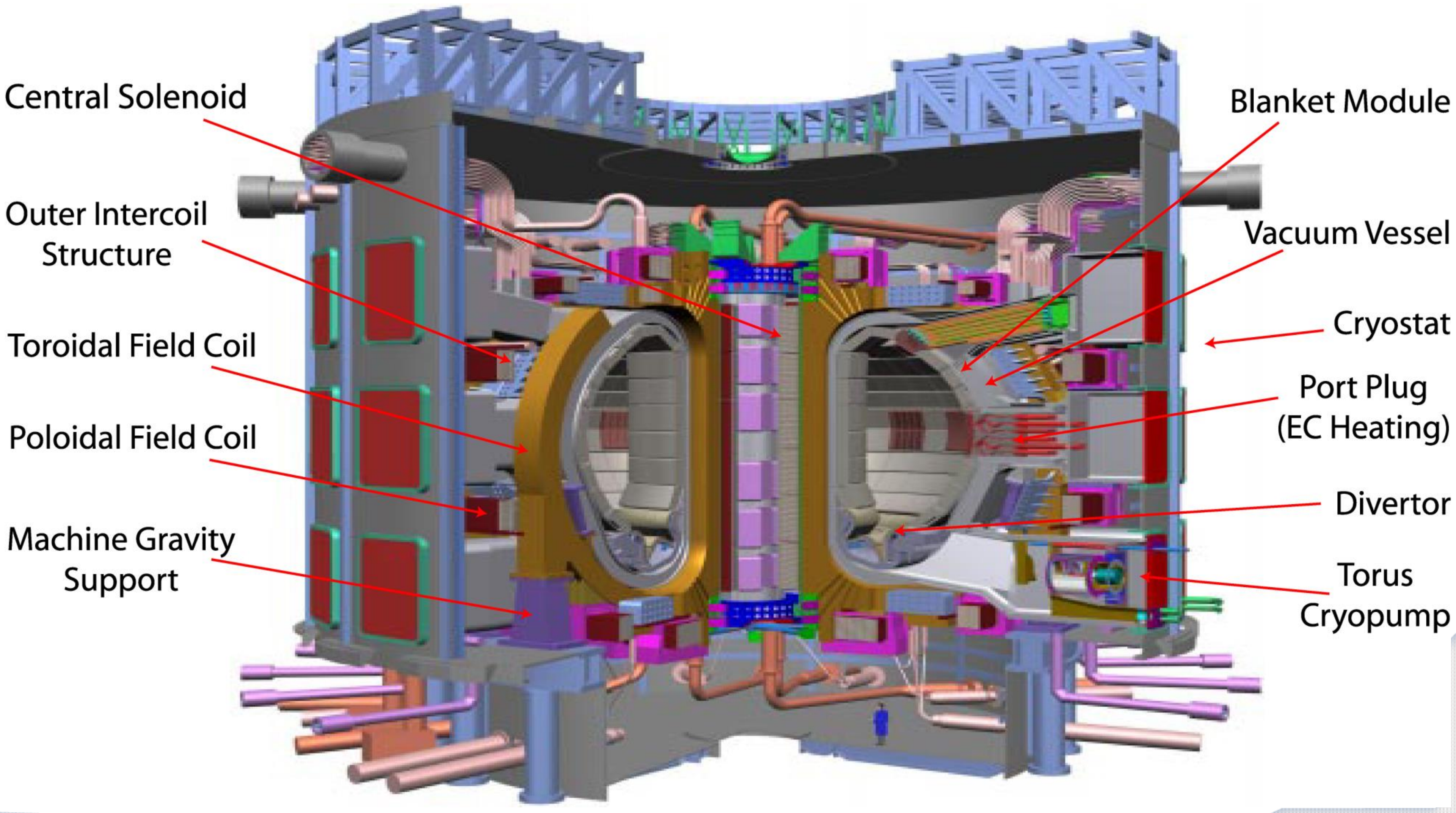
ITER Design Parameters



	ITER
Major radius	6.2 m
Minor radius	2.0 m
Plasma current	15 MA
Toroidal magnetic field	5.3T
Elongation / triangularity	1.85 / 0.49
Fusion power amplification	^{3} 10
Fusion power	~400 MW
Plasma burn duration	~400 s

A detailed engineering design for ITER was delivered in July 2001

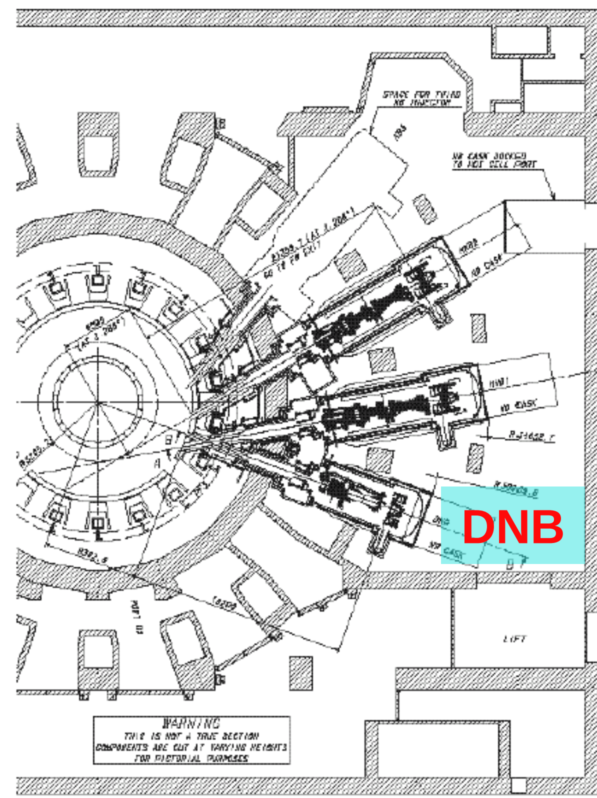
ITER Main Features



ITER Heating and Current Drive

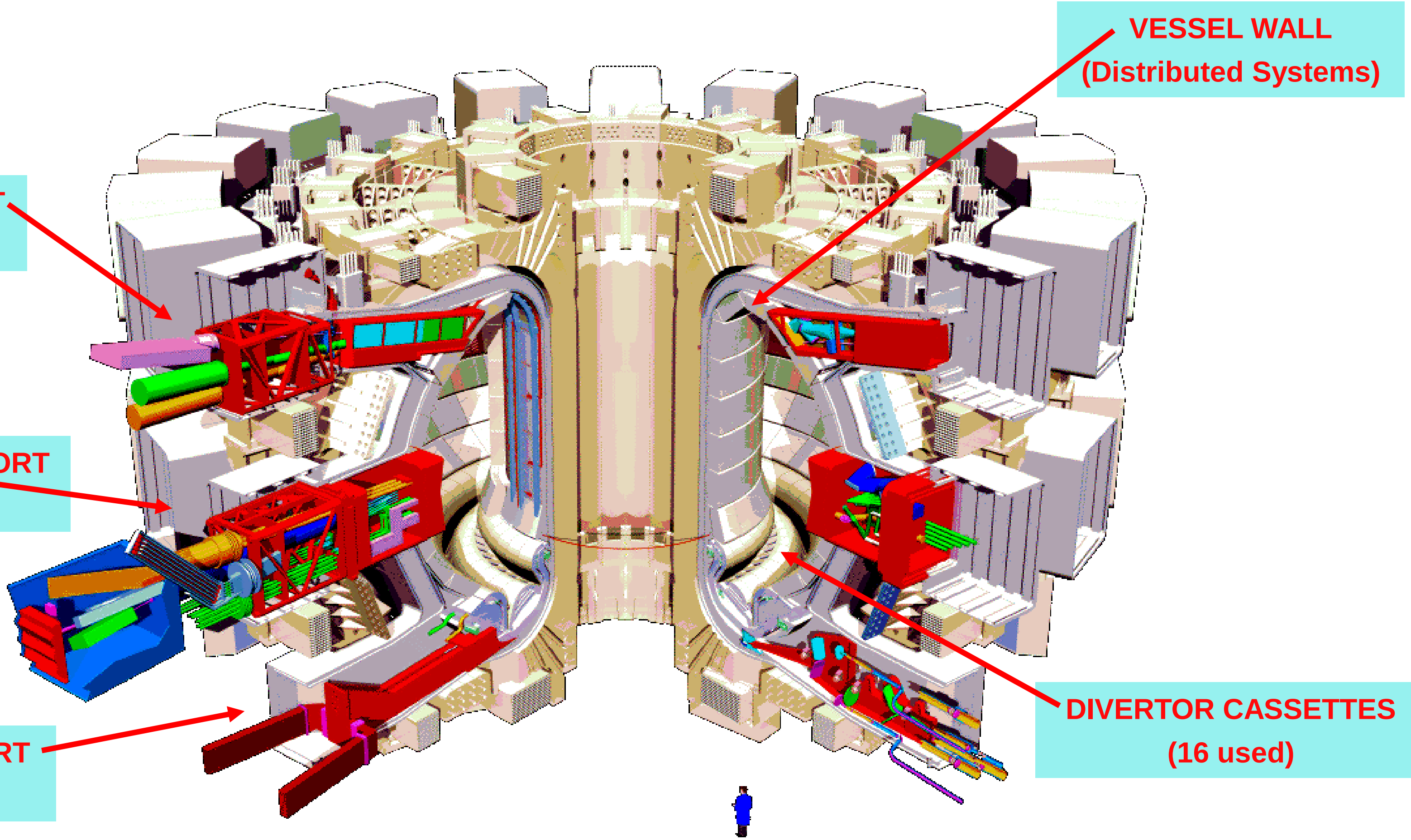
Heating System	Stage 1	Possible Upgrade	Remarks
NBI (1MeV Ğve ion)	33	16.5	Vertically steerable (z at Rtan -0.42m to +0.16m)
ECH&CD (170GHz)	20	20	Equatorial and upper port launchers steerable
ICH&CD (40-55MHz)	20		$2\Omega_T$ (50% power to ions Ω_{He3} (70% power to ions, FWCD)
LHH&CD (5GHz)		20	$1.8<n_{par}<2.2$
Total	73	130 (110 simultan)	Upgrade in different RF combinations possible
ECRH Startup	2		120GHz
Diagnostic Beam (100keV, H^-)	>2		

NBI Layout



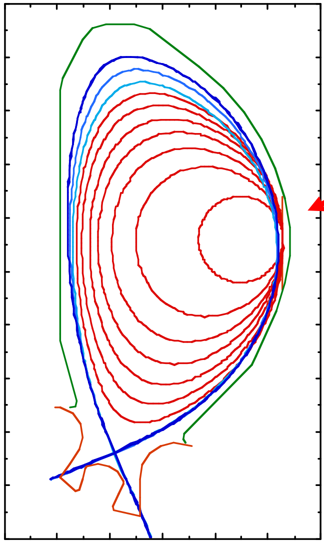
P_{aux} for Q=10 nominal scenario: 40-50MW

ITER Diagnostics



- **About 40 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 ….)

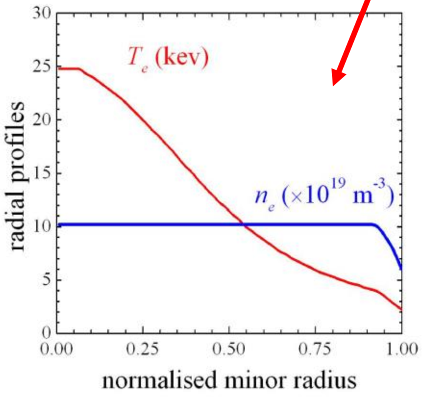
Fusion Plasma Diagnostics



Plasma shape evolution (ITER)

Plasma-wall interaction (JET)

Plasma density and temperature (ITER)

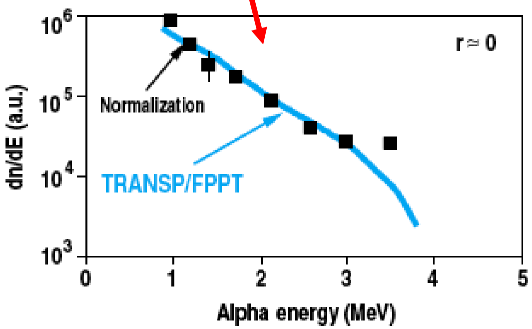


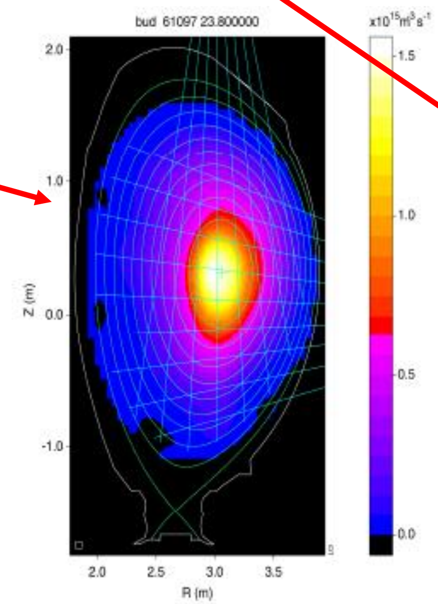
Plasma radiation (ASDEX-U)

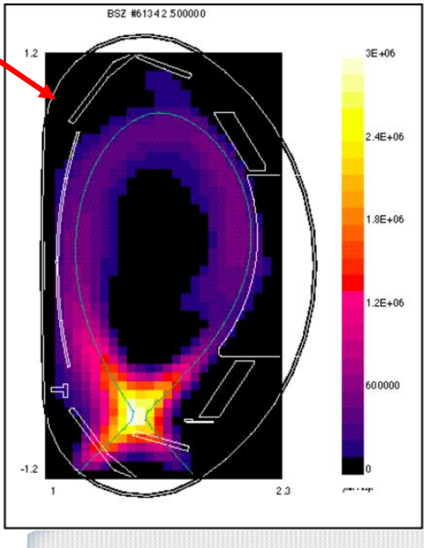
Fusion power:

14MeV neutron profile (JET)

α-particle spectrum (TFTR)



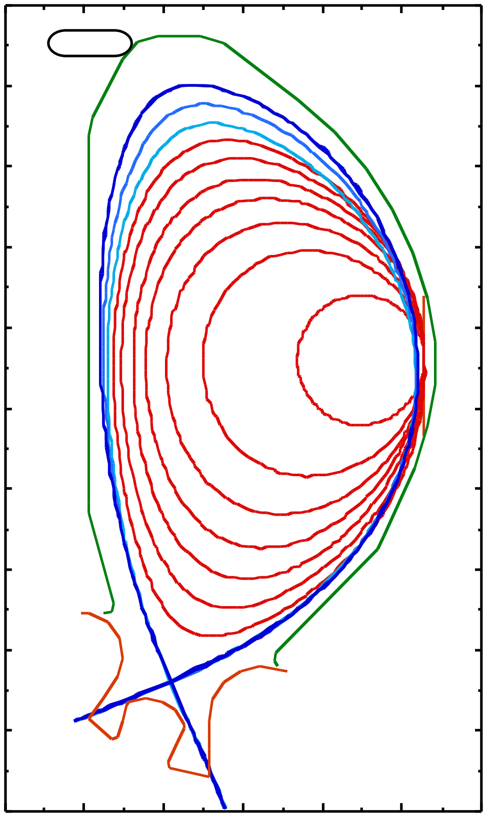


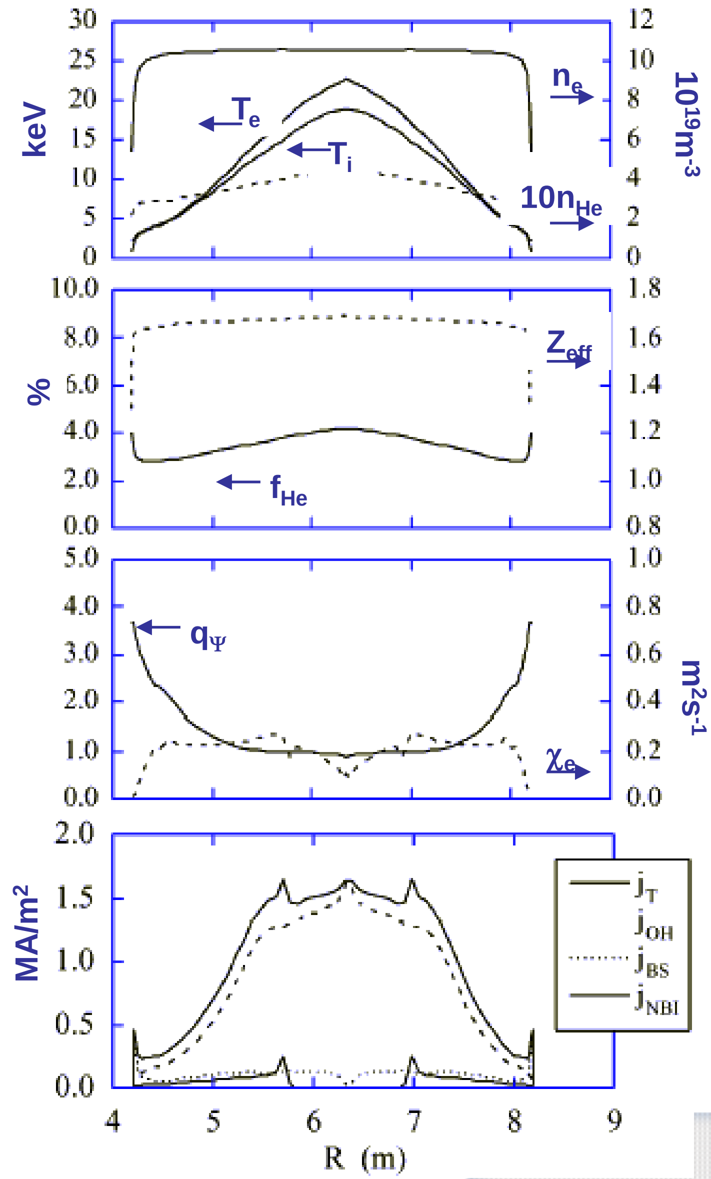


An ITER Plasma

A Q=10 scenario with:

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





Current Ramp-up Phase

ITER Physics -
Opportunities and Challenges

Operational scenarios and control issues

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

ITER Baseline Inductive Scenario

Physics Design Rules

1 Confinement:

- τ_E scaling: IPB98(y,2) $\Rightarrow H_{98(y,2)}$
- ITER H-mode threshold scaling

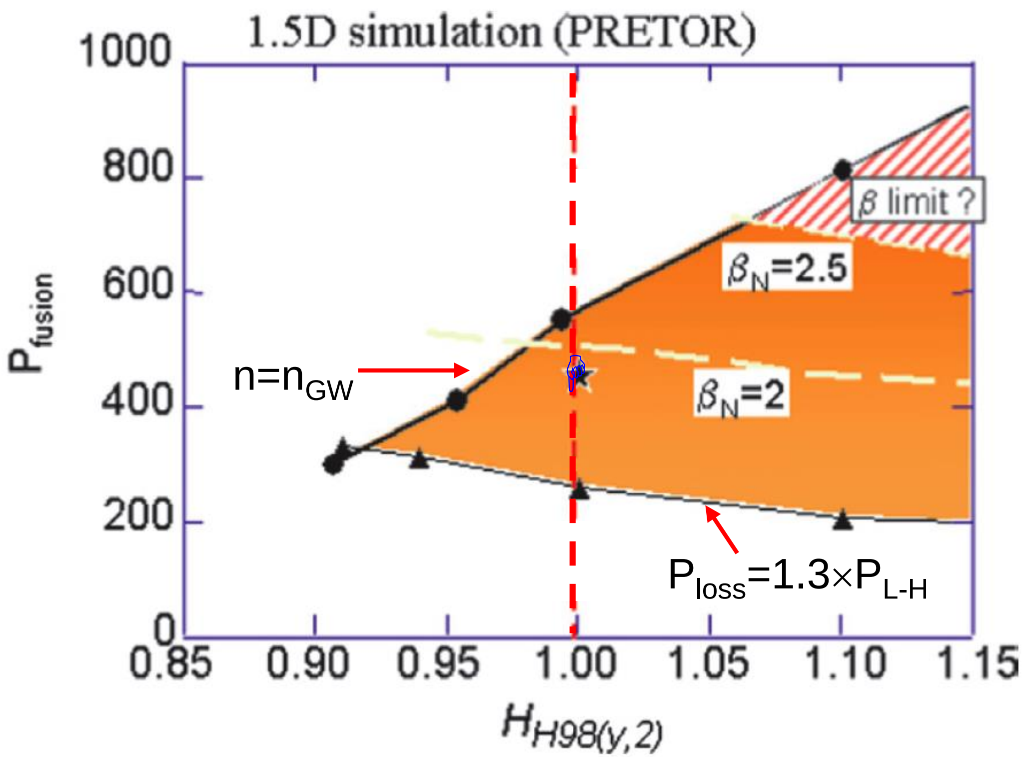
2 MHD stability:

- $q_{95} = 3$
- κ, δ determined by control requirements
- $n \leq n_{GW}$
- $\beta_N \leq 2.5$

3 Divertor:

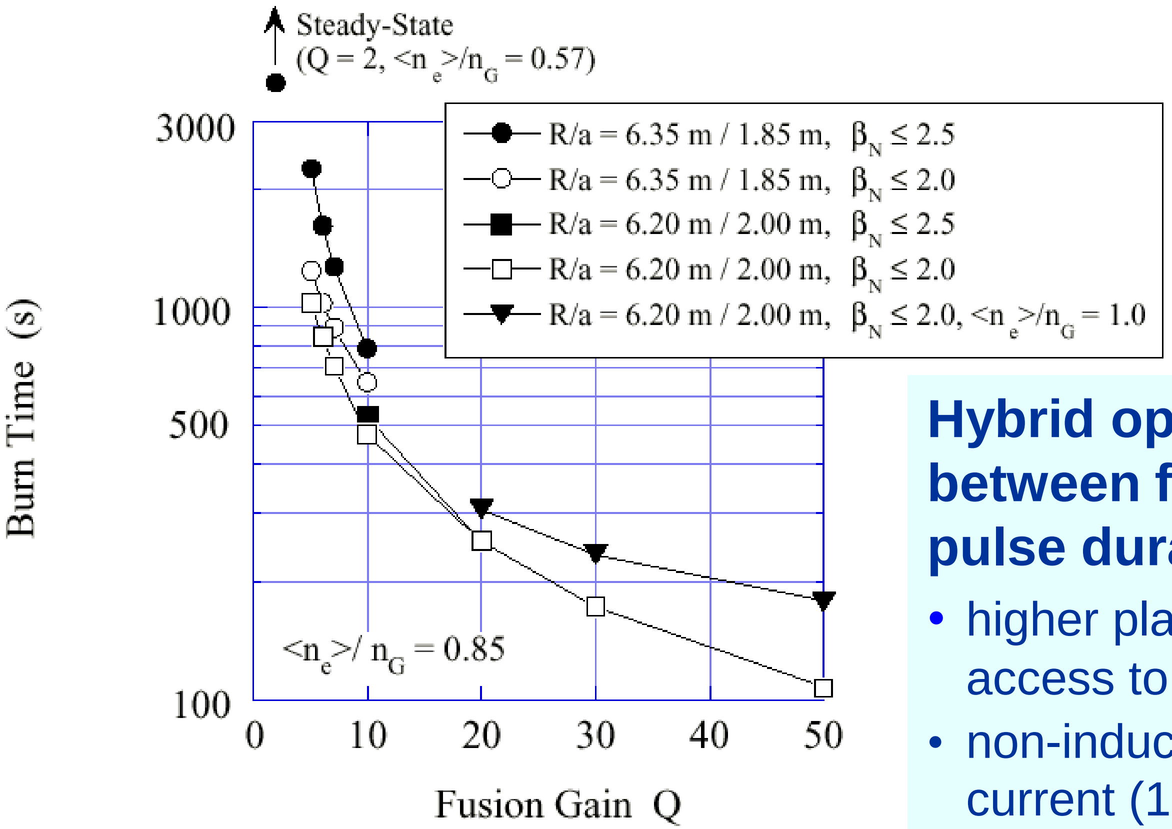
- Peak target power ≤ $10 MWm^{-2}$
- $\tau_{He}^*/\tau_E \sim 5$
- $n_{Be}/n_e = 0.02$

Q=10 at 15MA (q_{95}=3)



Hybrid Operation: Q>5

⇒ Conservative scenario for technology testing

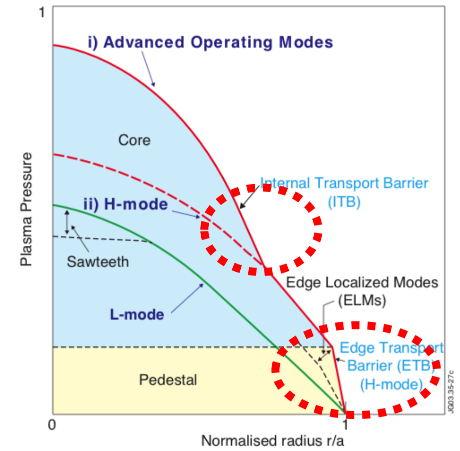


Hybrid operation allows trade-off between fusion performance and pulse duration:

- higher plasma current (17MA) allows access to higher fusion performance
- non-inductive current drive at lower current (12 - 14MA) allows pulse lengths >1000s

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

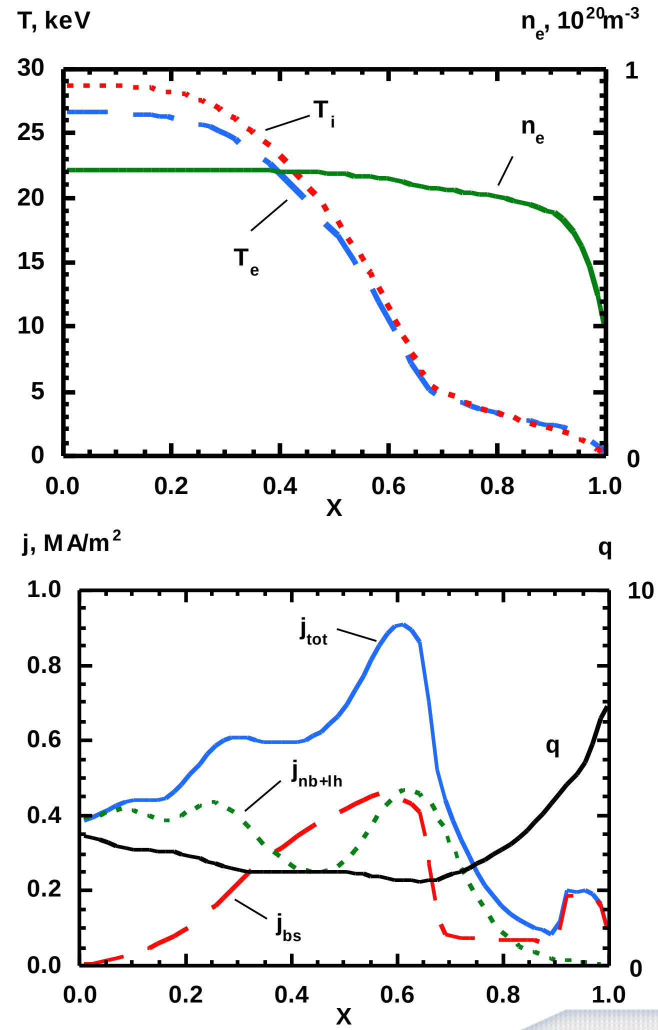
- **But development of an integrated plasma scenario satisfying all reactor-relevant requirements remains challenging**

ITER Steady-State Operation

A possible Q=5 steady-state scenario in ITER

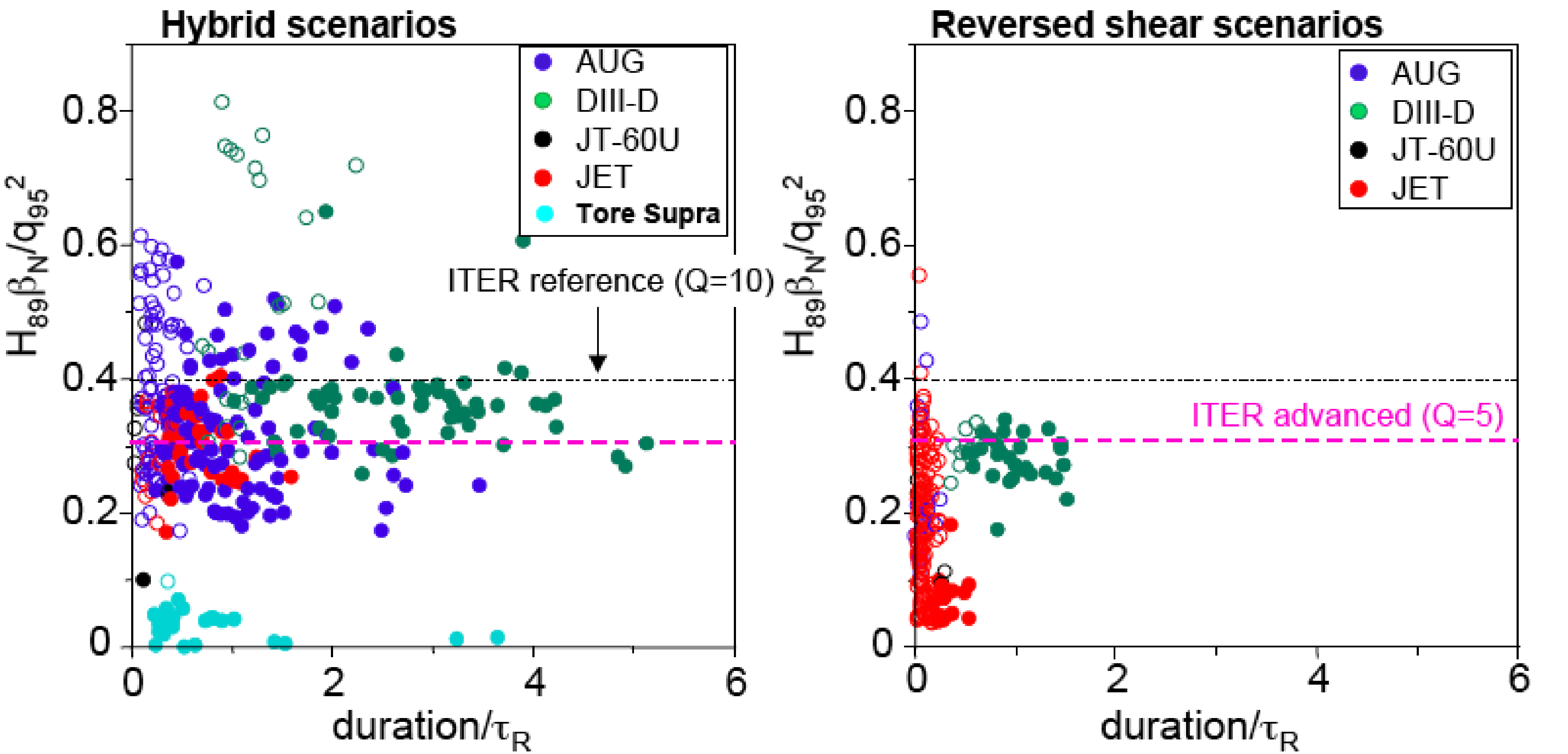
- **A range of scenarios has been explored with varying assumptions on core shear**
  - results are illustrative
  - possibilities for operation at higher Q have also been analyzed

- **ASTRA calculations of plasma profiles for an ITER Q=5 steady-state scenario:**
  - "weak central shear"
  - I_p=9MA, q_{95}=5.3
  - $H_{98(y,2)}$=1.3, β_N=2.56
  - $P_{LH}+P_{NB}$=34+34MW, P_{fus}=340MW



(A Polevoi et al, IAEA-CN-94/CT/P-08, IAEA FEC2002)

Hybrid and Advanced Operation



- $H_{98}\beta_N/q_{95}^2$ **provides a performance figure of merit for extrapolation to ITER-scale plasmas**
  - a major challenge to extend high performance to long pulses

A C C Sips et al, PPCF2005

Plasma Control

- **A magnetically confined plasma is a complex state**
  - it cannot be created and sustained without a sophisticated feedback control system
- **Free energy which is available within the plasma tends to generate <u>turbulence</u> and <u>magnetohydrodynamic instabilities</u> (mhd) which reduce plasma confinement quality**
  - much effort is expended within plasma control to sustain a high quality plasma capable of producing significant fusion power
- **In a thermonuclear plasma with significant fusion power, additional control requirements are imposed in ensuring the fusion power can be sustained for extended periods**
  - failures of the control system lead to conditions which tend to reduce fusion power or extinguish the plasma
- **At this stage of R&D, significant excursions in fusion power appear unlikely**
  - measures exist to control and suppress such excursions

Plasma Control in ITER

- **Plasma control in ITER is foreseen to consist of several main elements**
  - Plasma equilibrium control: (routine and robust)
    - plasma shape, position and current
  - Basic plasma parameter control: (routine and robust)
    - plasma density
  - Plasma kinetic control: (exploratory to robust)
    - fuel mixture, fusion power, radiated power …
  - Control for advanced operation: (exploratory)
    - current profile, temperature profile
  - Active MHD stability control: (exploratory to established)
    - error field modes (EFMs), edge localized modes (ELMs), neoclassical tearing modes (NTMs), resistive wall modes (RWMs)
  - Disruption/ vertical displacement event (VDE) avoidance and mitigation (exploratory to robust)
- **All of these elements have been developed and demonstrated to varying extents in existing tokamak devices**

Plasma Control Timescales

Scenario	Burn (s) *
Inductive (reference)	500
Hybrid	1000
Steady-state	3000 **

* repetition time = 4 × burn time

** limited by external cooling capacity (at present)

- **Current diffusion requires:**

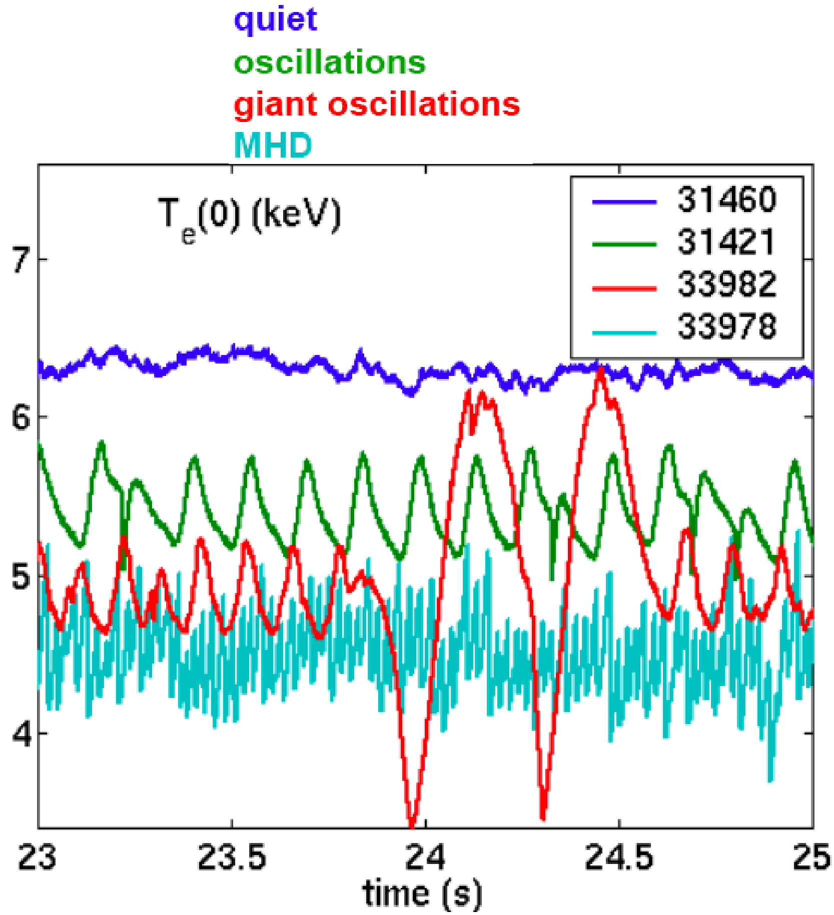
$$\tau \sim \tau_R$$

- **But execution of control implies**

$$\tau \gg \tau_R$$

⇒ pulse lengths in ITER will allow plasma control issues to be studied

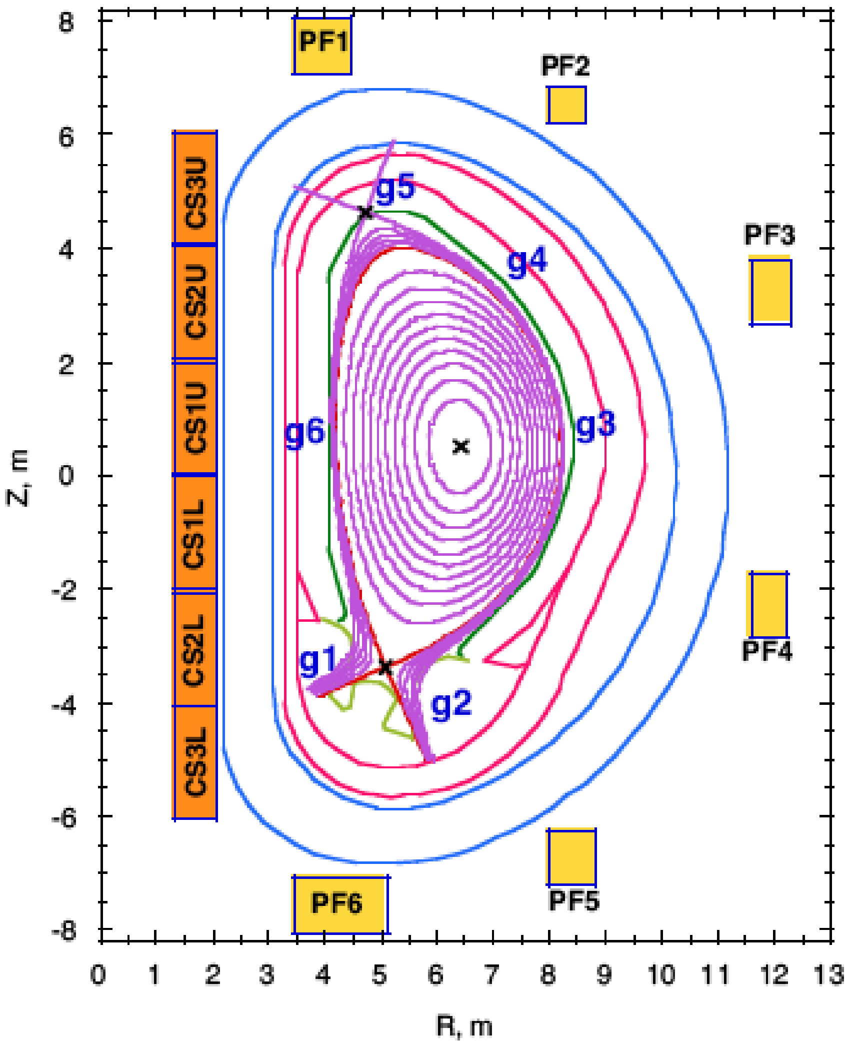
Tore Supra experiments at V_L=0 indicate sensitivity of plasma behaviour to q-profile



G Giruzzi et al, Phys Rev Lett **91** 135001 (2003)

Plasma Equilibrium Control

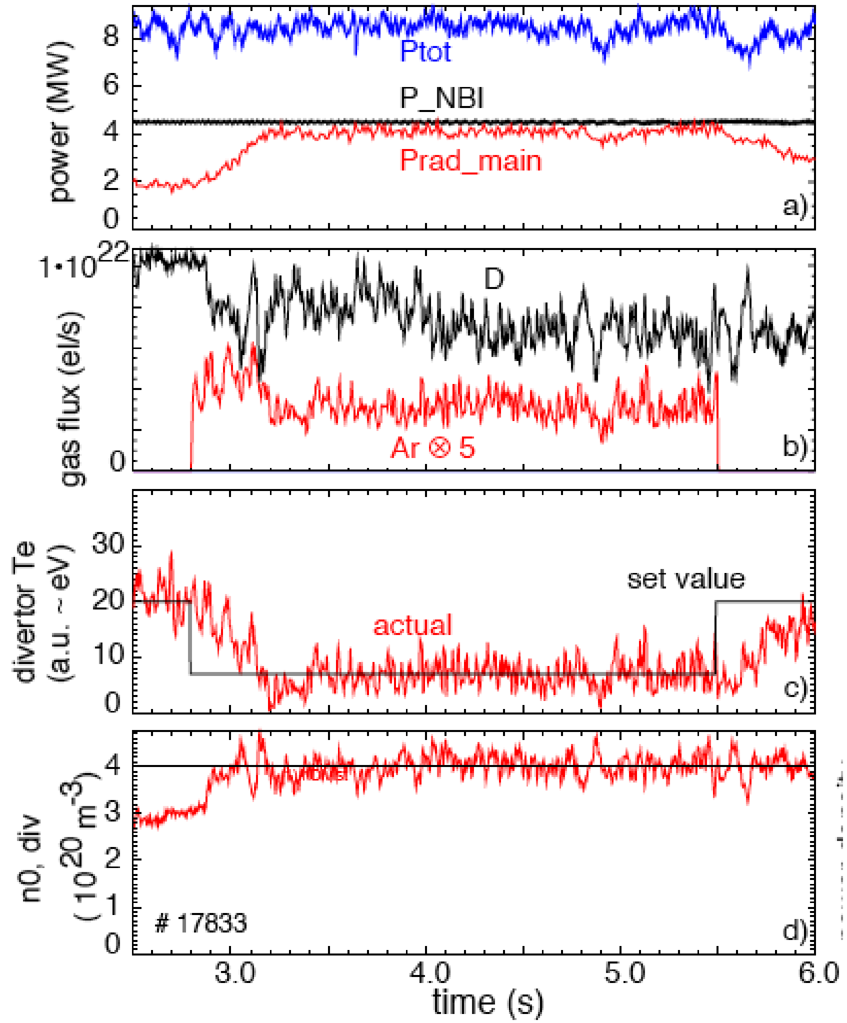
- **Plasma equilibrium control is routine:**
  - plasma shape, position and current are kept under feedback control
  - system is based on magnetic sensors which provide signals to reconstruct plasma boundary or equilibrium
  - feedback signals control voltages to PF and CS coils to maintain required plasma equilibrium parameters
  - very long pulses require particular care to avoid drifts in magnetic diagnostic signals



Exhaust Power Control

ASDEX Upgrade

- **Exhaust power flowing to the divertor can be controlled by injection of selected impurities:**
  - noble gases usually chosen
  - limits heat flux to target
  - allows divertor temperature to be kept low to minimize erosion
- **Feedback control of impurity gas flow allows radiated power level to be actively adjusted**
  - heat flux to target can be adjusted in response to variations in loss power (fusion power) in plasma

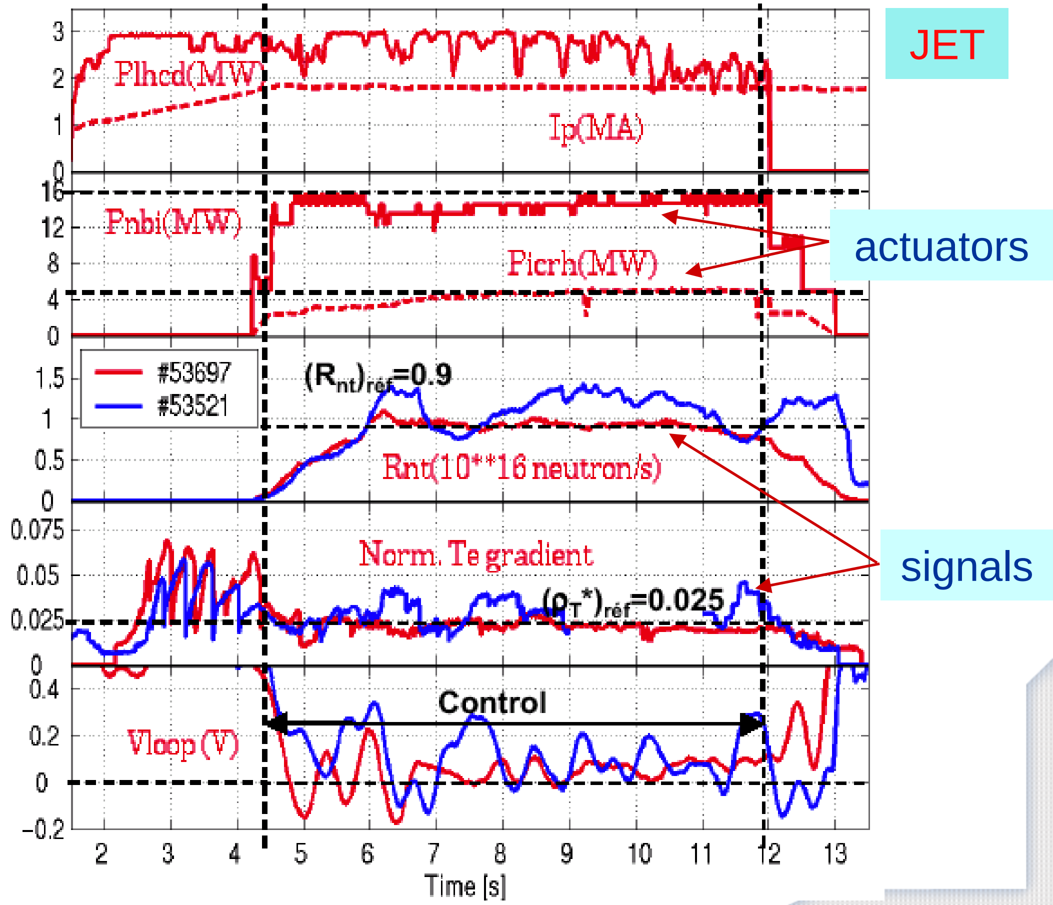


(H Kallenbach et al, ASDEX Upgrade 2002)

Complex Plasma Control

ITER's extensive H&CD system, Diagnostic capability, together with Control coils and Pellet injection will allow the exploitation of sophisticated control techniques in a burning plasma environment

- plant monitoring and safety control
- machine protection during plasma operation
- real-time control of plasma equilibrium and auxiliary systems
- efficient real-time data analysis and archiving
- active feedback control loops between diagnostic measurements and auxiliary systems (H&CD, fuelling etc)



D Mazon et al, 28th EPS Conference, Funchal (2001)

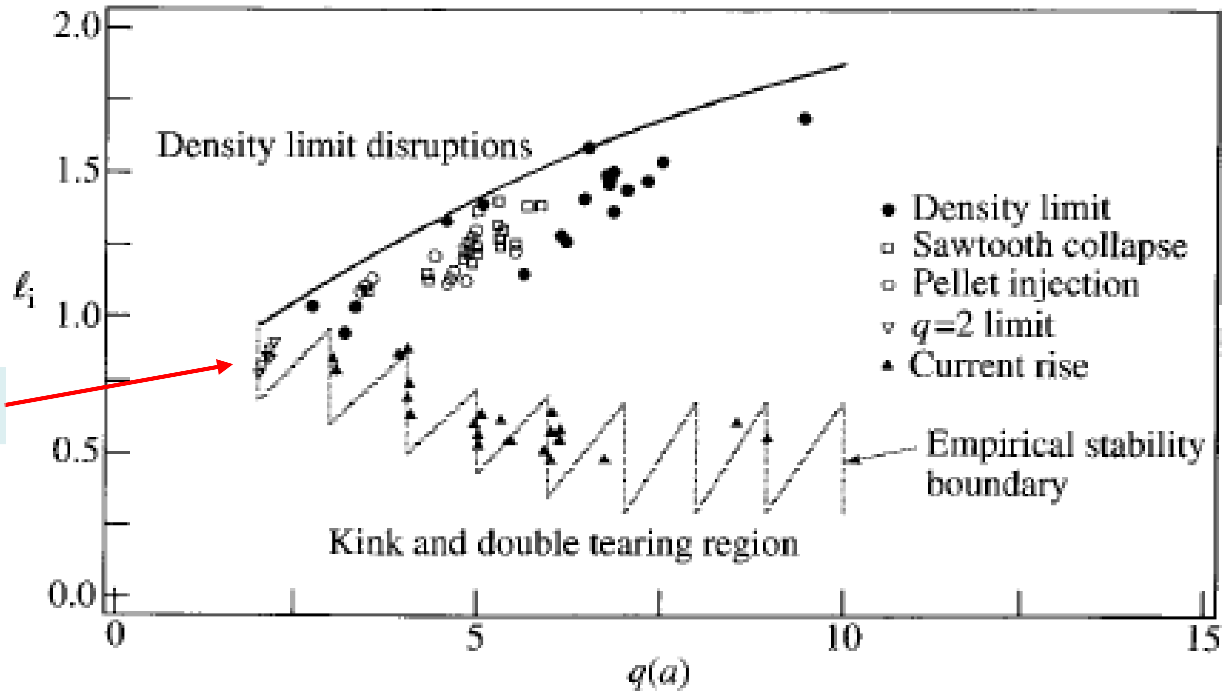
ITER Physics - Opportunities and Challenges

MHD Stability Issues

Plasma Operational Limits

- **Extensive R&D has defined various operational limits within which a stable plasma can be sustained:**
  - Plasma current limit:
    - plasma safety factor, q ($\propto a^2 B_\phi / R I_p$) > 2 (hard limit)
  - Plasma equilibrium limit(s):
    - an equilibrium operating space can be defined relating q and l_i (internal inductance) (hard limits)
  - Plasma elongation limit:
    - plasma elongation, κ, has a maximum value which depends on <u>plasma equilibrium</u> and its <u>inductive coupling to tokamak structure</u> (hard limits)
  - Plasma density/ radiation limit(s):
    - the plasma can sustain a maximum density/ radiation level which depends on confinement regime (soft or hard limits)
  - Plasma pressure limit(s):
    - plasma normalized pressure, β ($\propto p/B^2$), is limited by various mhd instabilities (soft or hard limits)
- **Plasma control system steers plasma in operating space within these limits to ensure good confinement and high fusion power**

Plasma Equilibrium Limits



JET

Limiter plasmas

q=2 limit

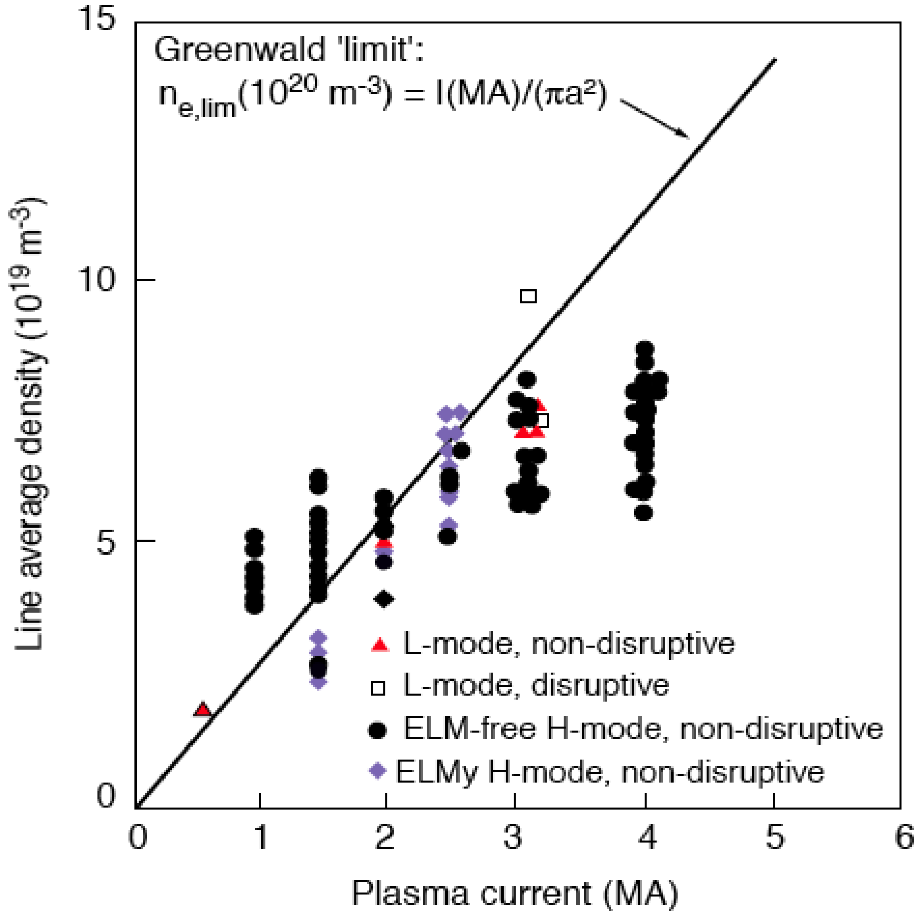
- **l_i-q diagram describes stable plasma operating region, limited by disruptions:**
  - low l_i typically has to be negotiated during the plasma current ramp-up
  - high-l_i limit typically occurs due to excessive radiation at plasma edge, resulting in cold edge plasma and narrow current channel (eg at density limit)

Density Limits

- **Experiments have shown that tokamak plasmas can sustain a maximum density:**
  - limit depends on operating regime (ohmic, L-mode, H-mode …)
  - limit may be determined by edge radiation imbalance or edge transport processes
  - limit can be disruptive or non-disruptive
- **Comprehensive theoretical understanding still limited**
  - "Greenwald" density:

    $n_{GW} = I(MA)/ \pi a^2$
  - operational figure of merit

JET



Plasma MHD Stability Limit: β

- **Maximum value of normalized plasma pressure, β, is limited by mhd instabilities:**

$$\beta(\%) = 100 \frac{<p>}{B^2/2\mu_o}$$

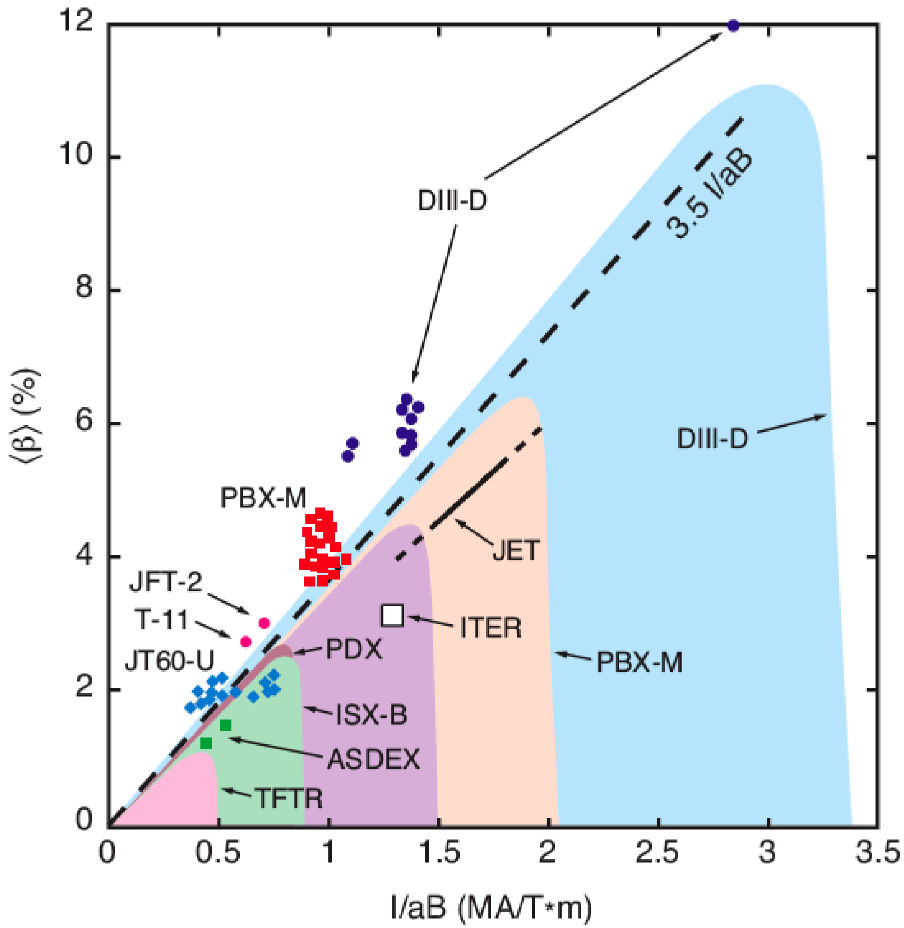
$$\beta_N = \frac{\beta(\%)}{I_p(MA)/aB}$$

- **Typically, “Troyon” limit describes tokamak plasmas:**

$$\beta_N \leq 2.8\text{-}3.5$$

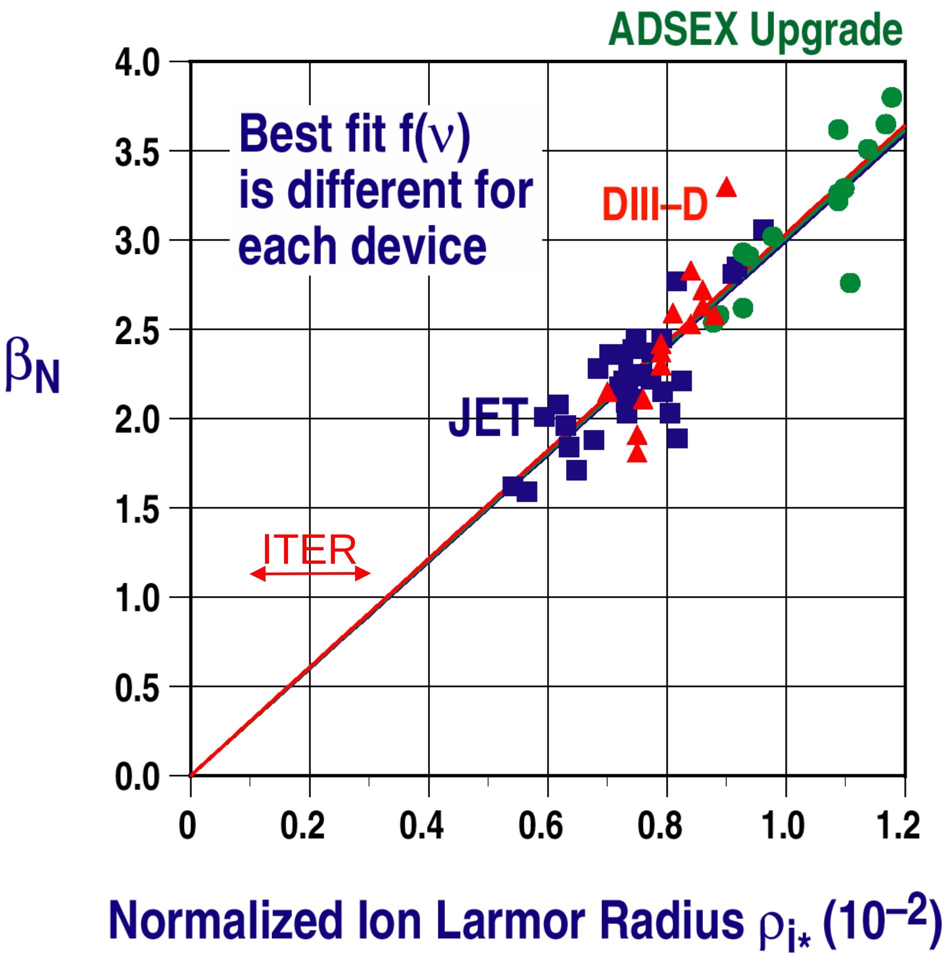
- **More generally, “no-wall” limit:**

$$\beta_N \leq 4 \times l_i$$



β-Limit in Inductive Scenarios:
Neoclassical Tearing Modes

- **NTMs can determine the β-limit below ideal limit**
  - experiments show that $\beta_{N(critical)} \propto \rho_\iota{}^*$
- **Several successful approaches developed which allow expansion of inductive operating regime:**
  - active ECCD feedback stabilization
  - sawtooth control of seed island trigger by ICCD / ECCD
  - self-limitation via FIR NTMs (AUG & JET)

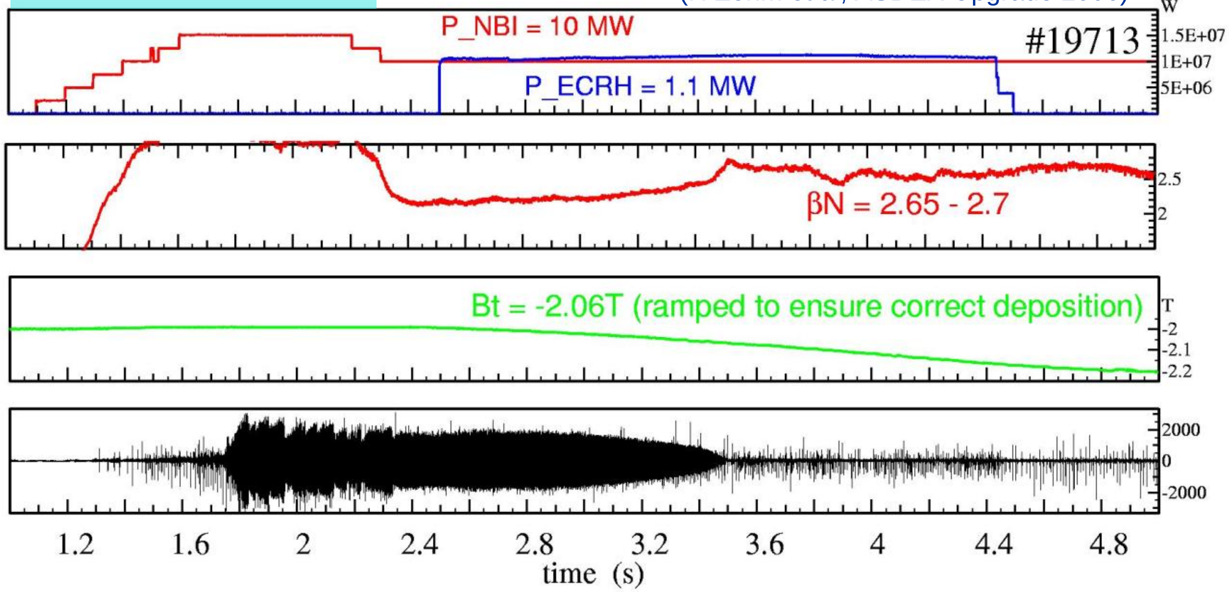


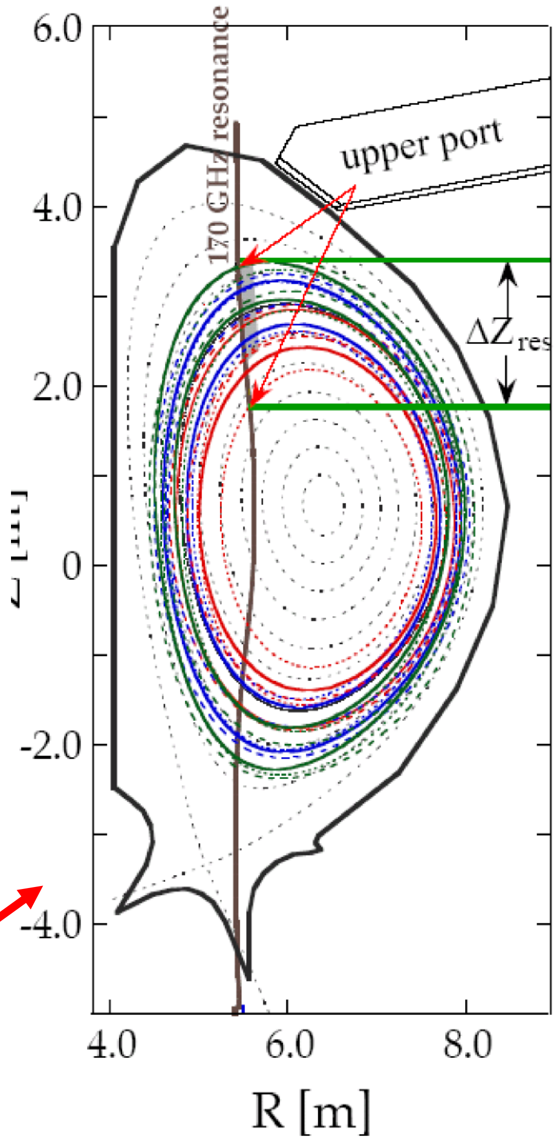
(R J La Haye et al, Phys Plasmas, **7** 3349 (2000))

ECCD Control of NTMs

ASDEX Upgrade

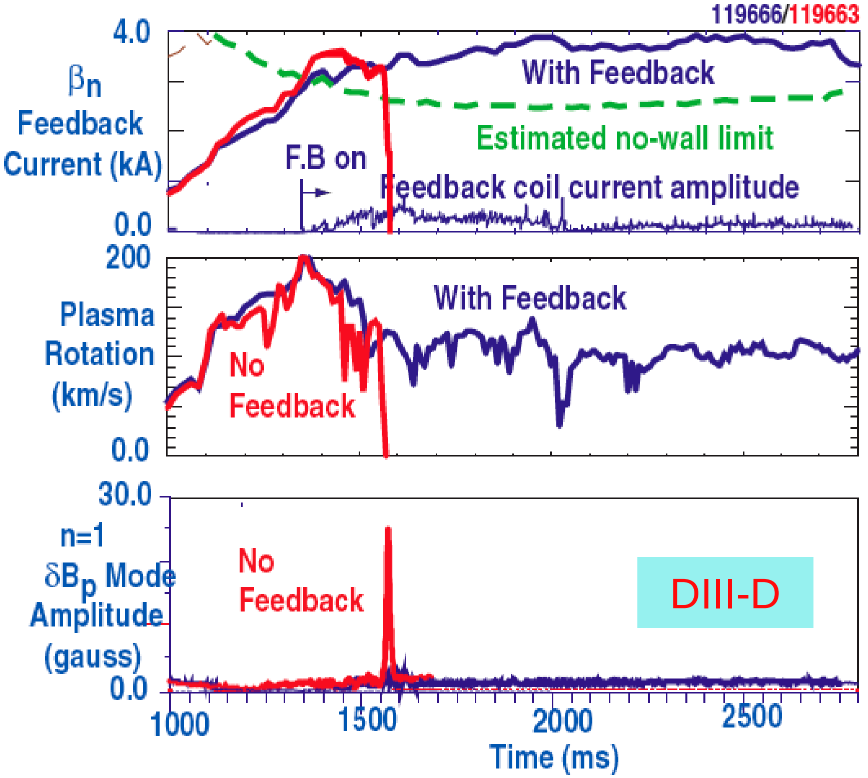
(H Zohm et al, ASDEX Upgrade 2006)

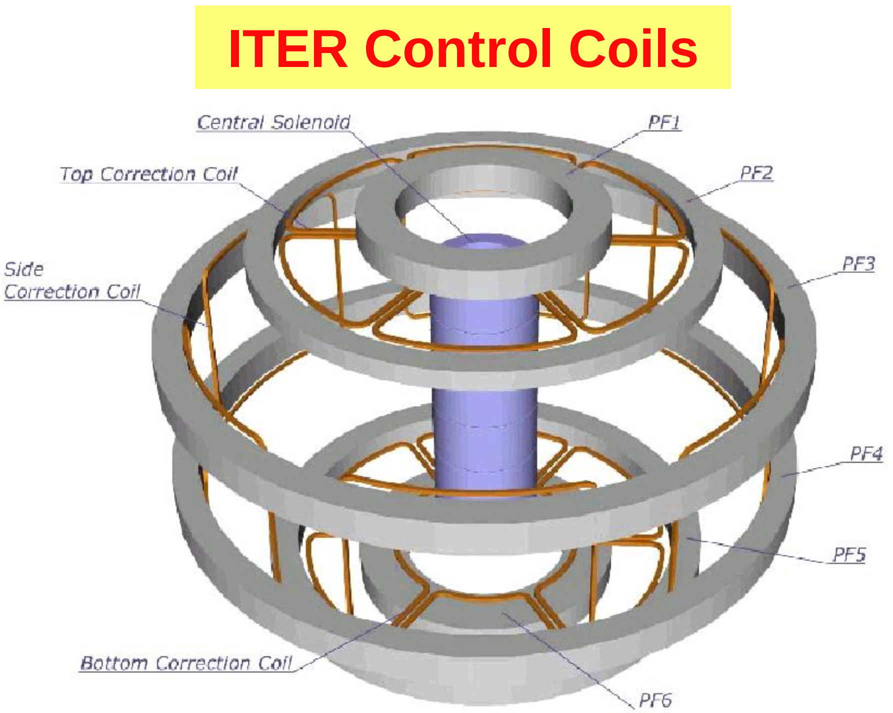




- **Electron cyclotron waves can produce localized current drive inside magnetic island**
  - this is exploited in present experiments to suppress NTMs
- **ITER: 4 steerable launchers in upper ports injecting 20MW total ECCD power**

High-β_N Stability:
Resistive Wall Mode Control





Advanced scenarios at high β_N (β_N ~ $4l_i$) require RWM feedback stabilization

(M Okabayashi et al, IAEA-CN-116/EX/3-1Ra, IAEA FEC2004)

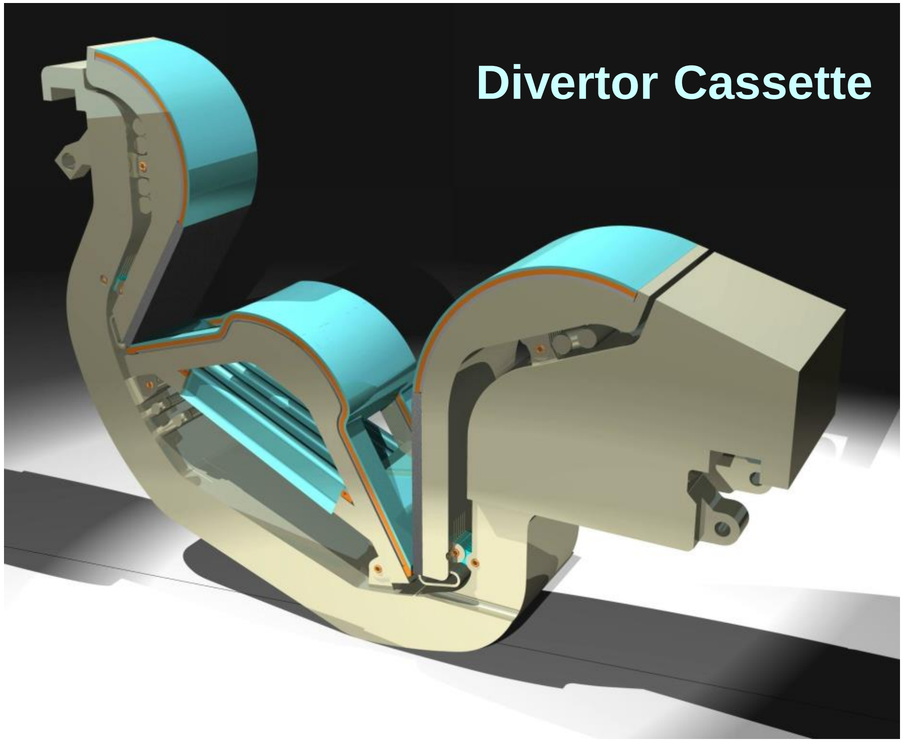
ITER error field correction and RWM control coils:

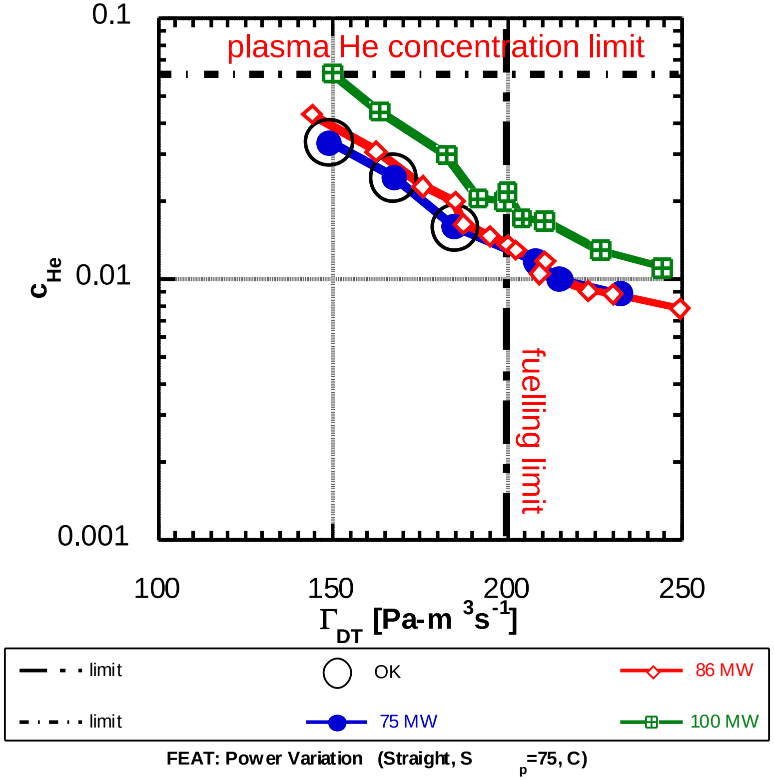
- additional coil systems under investigation

ITER Physics - Opportunities and Challenges

Power and Particle Exhaust

Power and Particle Exhaust

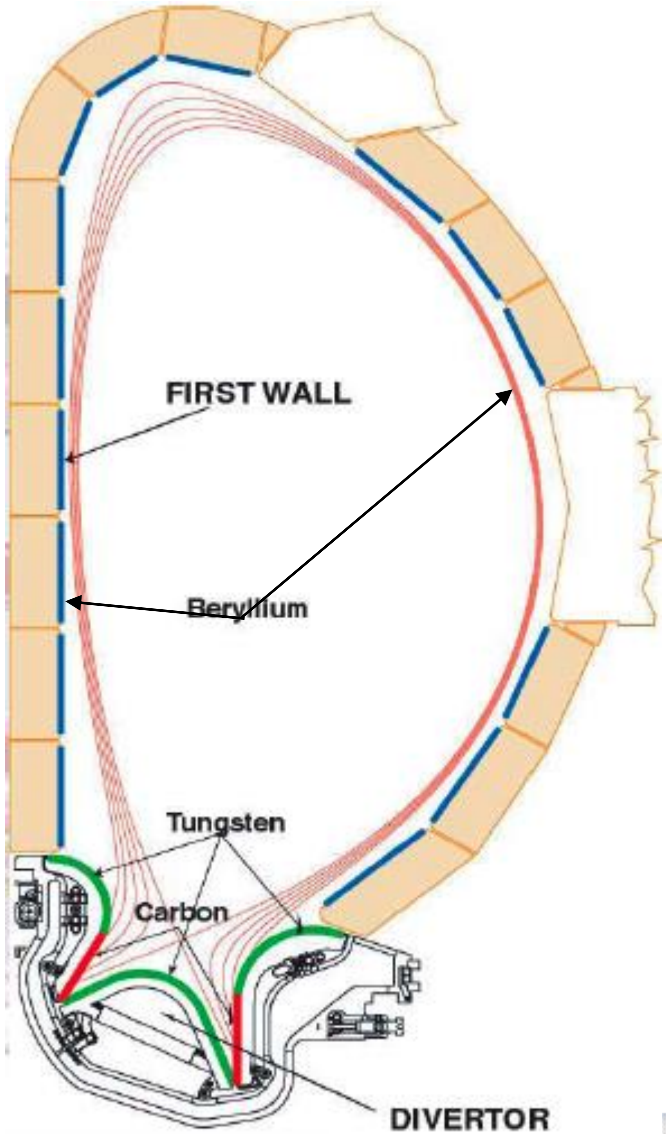




- **Extensive modelling of power and particle exhaust gives confidence in ITER divertor performance:**
  - peak target power can be limited below 10MWm^{-2} in reference scenarios
  - installed fuelling and pumping capacity should ensure that core helium capacity can be held below 6%

Plasma Facing Components - Challenges

- **CFC divertor targets (~50m²):**
  - erosion lifetime (ELMs!) and tritium codeposition
  - dust production
- **Be first wall (~700m²):**
  - dust production and hydrogen production in off-normal events
  - melting during VDEs
- **W-clad divertor elements (~100m²):**
  - melt layer loss at ELMs and disruptions
  - W dust production - radiological hazard in by-pass event



Plasma Transients

- **Several types of transient event can occur in plasmas, only some of which need to be controlled:**
  - Sawteeth:
    - a repetitive mhd instability which modulates central plasma parameters (principally benign)
  - Edge localized modes (ELMs):
    - a repetitive mhd instability which modulates edge plasma parameters (principal impact on lifetime of plasma facing components)
  - MARFEs:
    - a radiation instability which can lead to localized heating of the first wall (first wall designed to handle estimated heat loads)
  - Disruptions:
    - mhd instabilities trigger a rapid termination of plasma energy and current (can produce enhanced erosion of PFCs; generates eddy current forces in structures)
  - Vertical Displacement Events (VDEs):
    - loss of plasma vertical position control causes loss of energy and current (can produce localized surface melting/ ablation of PFCs; generates eddy and halo current forces in structures)

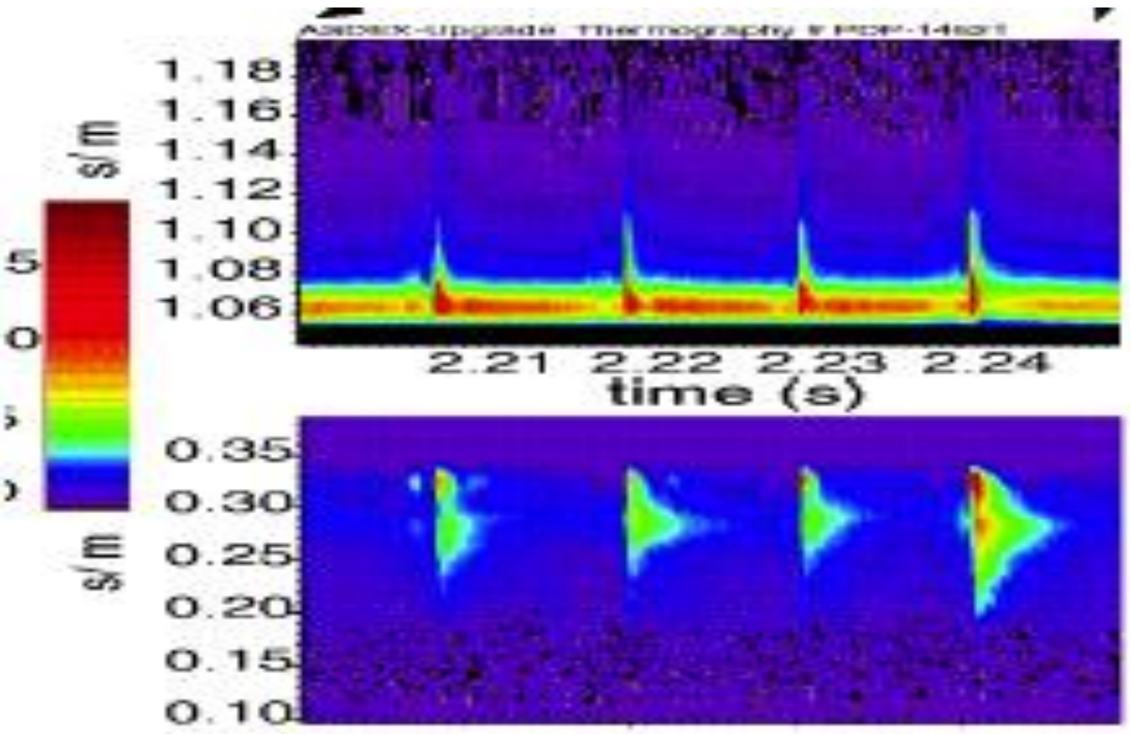
Edge Localized Modes

- **ELMs are a repetitive instability of the edge plasma in H-mode:**
  - Edge plasma experiences quasi-periodic relaxations ⇒ ELMs

  □ ΔW_{ELM} is small fraction of W_{plasma} (< 10 %) in ~ 200 μs

  ⇒ Large Energy Flux

ASDEX Upgrade Herrmann



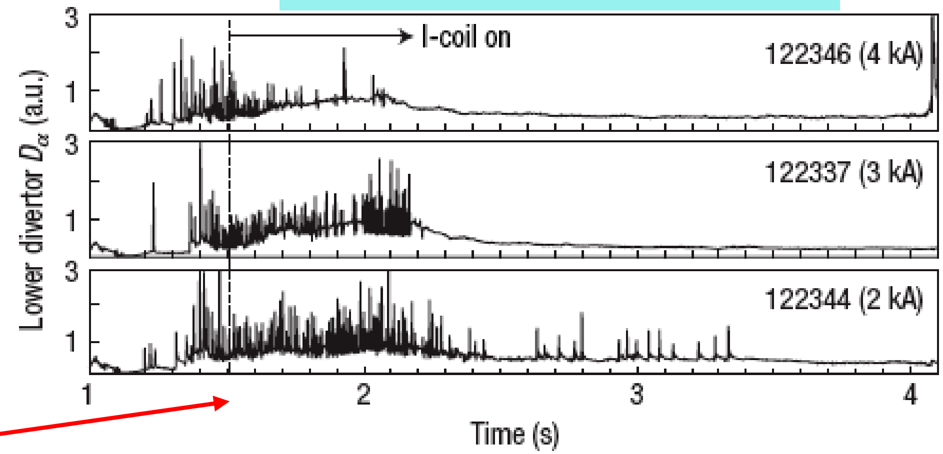
Outer divertor

Inner divertor

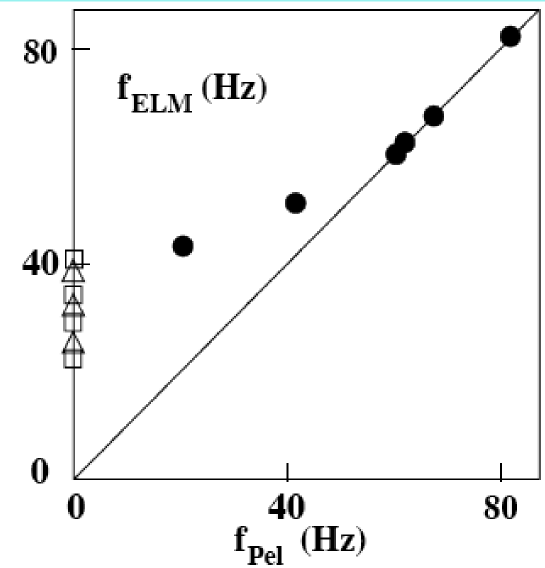
ELM Control/ Mitigation

- **Several options are being investigated for ELM control:**
  - problem is sufficiently important for ITER that they are all being pursued
- **Results with magnetic control look promising:**
  - studies underway to design control coil system for ITER
- **“ELM pacemaking” using pellet injection also effective:**
  - quantitative basis for application in ITER being studied

DIII-D Magnetic Control



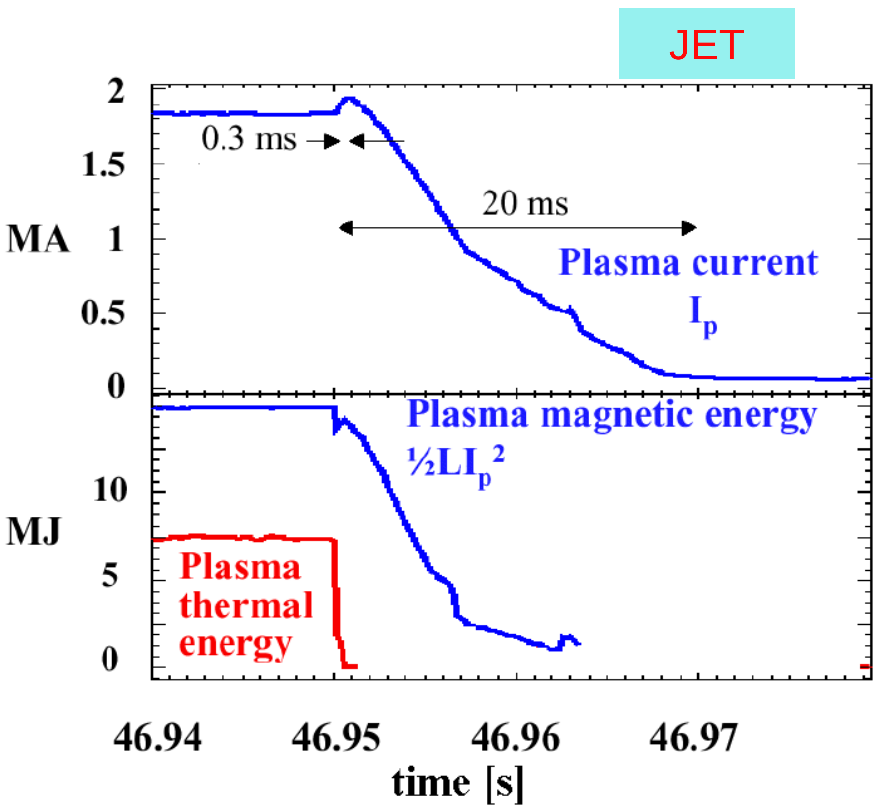
AUG Pellet Pacemaking



Disruptions

- **Disruptions occur in tokamak plasmas when unstable p(r), j(r) are developed**

  ⇒ MHD unstable modes grow
  ⇒ plasma confinement is destroyed (thermal quench)
  ⇒ plasma current vanishes (current quench)



Typical timescales

- Thermal quench < 1ms ⇒ deposition of plasma thermal energy on PFCs
- Current quench > 10 ms ⇒ deposition of plasma magnetic energy by radiation on PFCs & runaway electrons

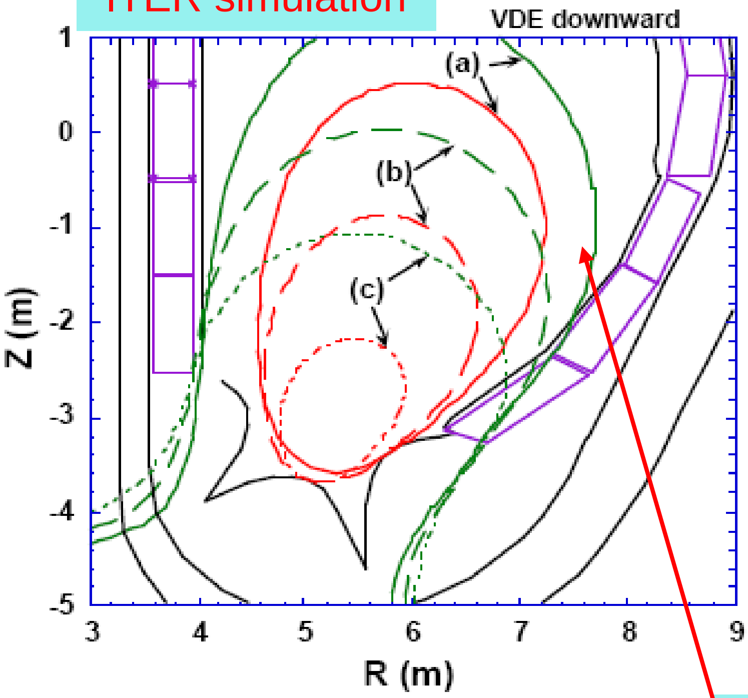
Typical values for ITER current quench

- $W_{poloidal}$ ~ 1 GJ
- $\tau_{c.q.}$ ~ 20-40 ms
- q_{rad} ~ 35 – 70 MWm^{-2}
- A_{wall} ~ 700 m^2
- $q_{rad}\,\tau_{c.q.}^{1/2}$ ~ 7–10 $MJm^{-2}s^{-1/2}$ (no Be melting)

Vertical Displacement Events

- **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

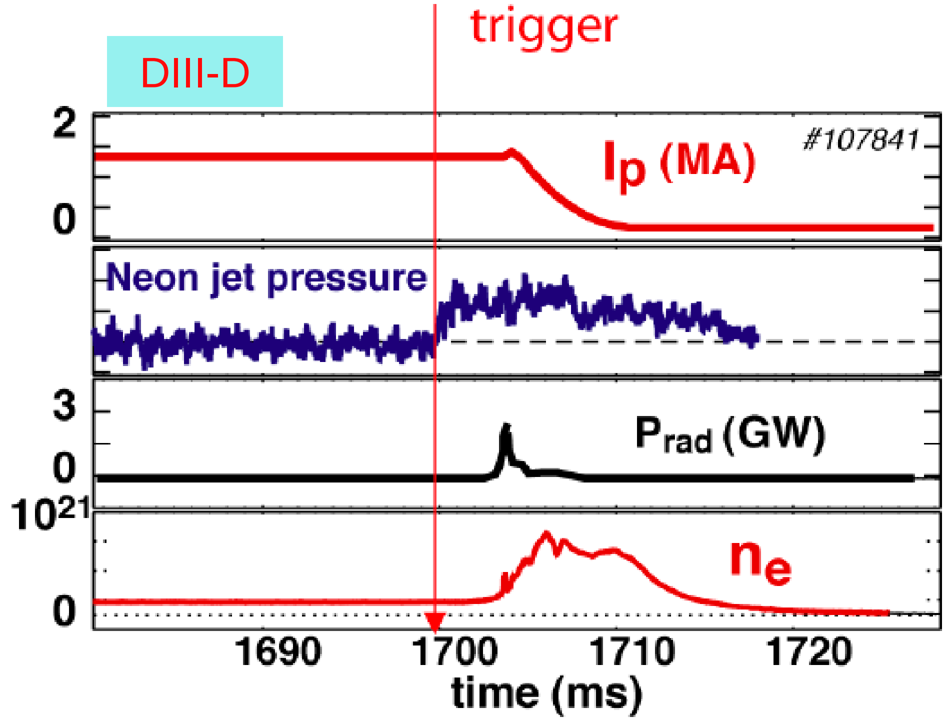


Halo current layer

Control issues

- Detection of loss of vertical position control
- Fast stop of plasma by massive gas injection, killer pellets, etc.
- Issues of effectiveness, reliability of mitigation method, as well as additional consequences (runaway electrons) need to be addressed in experiment

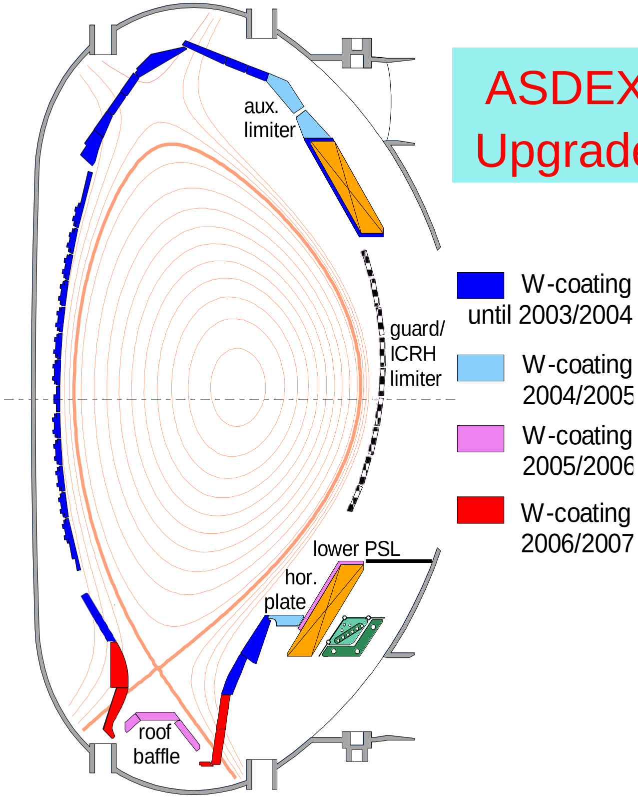
Disruption VDE/ Mitigation



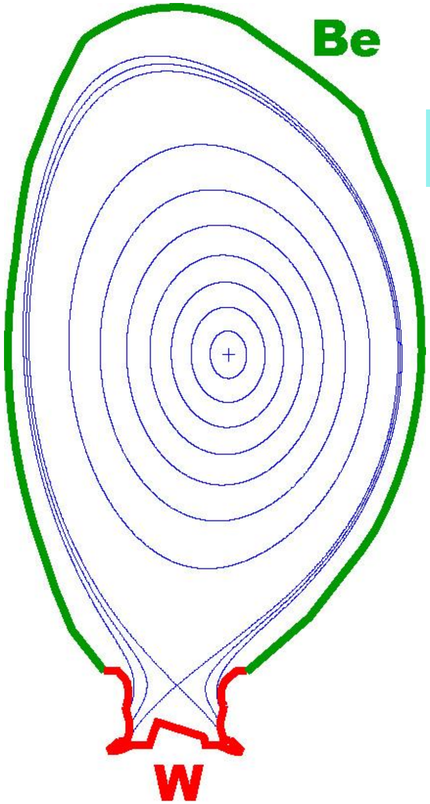
- **The development of high pressure impurity gas injection looks very promising for disruption/ VDE mitigation:**
  - efficient radiative redistribution of the plasma energy - reduced heat loads
  - reduction of plasma energy and current before VDE can occur
  - substantial reduction in halo currents (~50%) and toroidal asymmetries

PFC R&D in EU Tokamaks

Tore Supra



ASDEX Upgrade



JET

- **Major R&D projects are underway in EU tokamaks to investigate PFC performance issues:**
  - Tore Supra: long pulse operation with CIEL CFC pumped limiter
  - ASDEX Upgrade:conversion to all tungsten PFCs complete
  - JET: installation of beryllium wall and tungsten divertor in 2009

ITER Physics -
Opportunities and Challenges

Burning Plasma Physics in ITER

Burning Plasma Physics in ITER

Physics constrained by experimental goals:

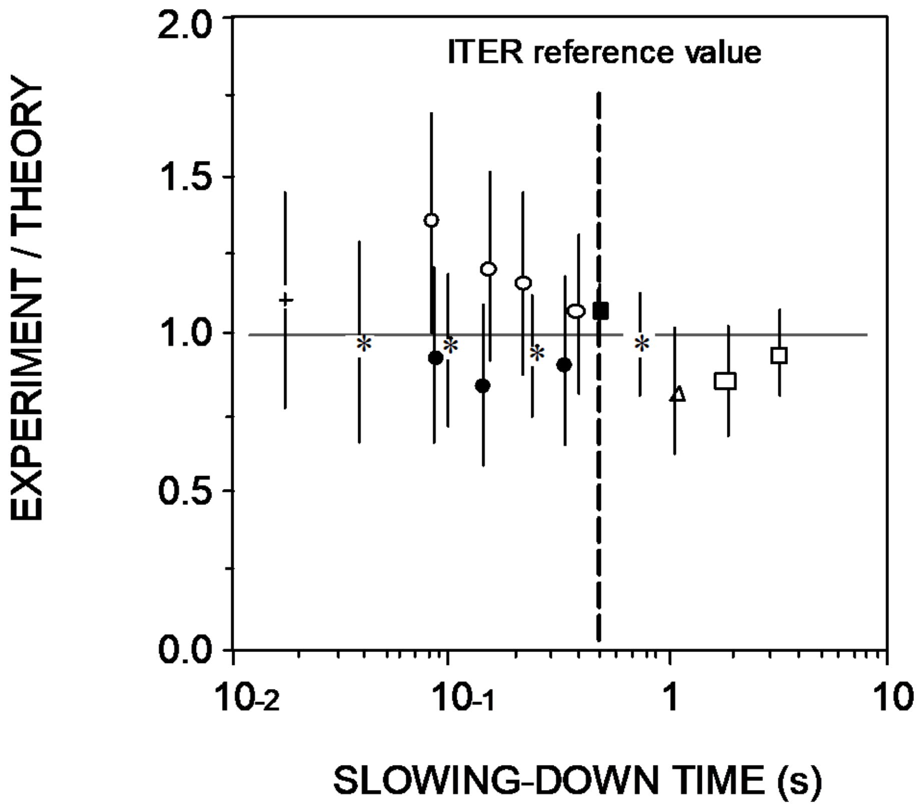
- **Scaling of transport/confinement etc to reactor scale**
- **MHD stability at the reactor scale (eg sawteeth, NTM, RWM)**
- **Power and particle exhaust (eg edge/core integration, transients)**
- **Development of steady-state operation (eg control of non-linear equilibrium, heating and current drive physics)**

Physics accessible only in a burning plasma experiment:

- **Burn control**
- **Alpha-particle confinement, slowing-down and heating:**
  - response of plasma to α-heating
  - influence of α-particles on mhd stability

Energetic Ion Confinement

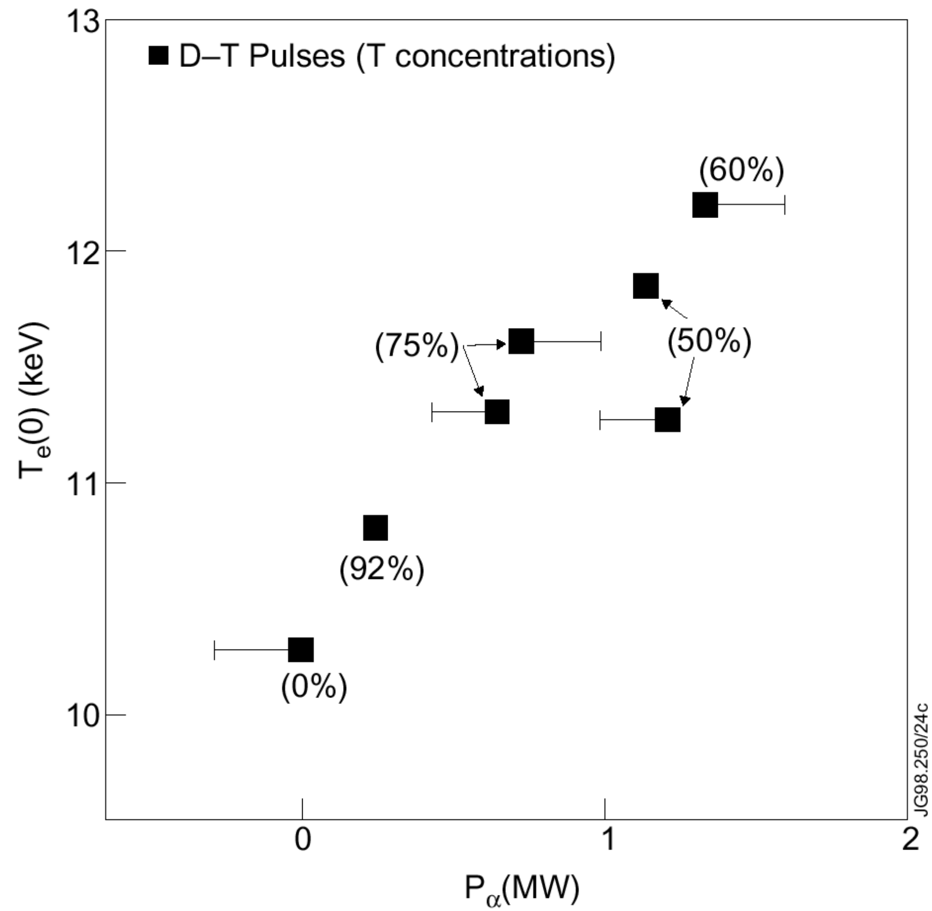
- **In existing experiments single particle theory of energetic ion confinement confirmed:**
  - simple estimate, based on banana orbit width shows that $I_p \geq 3MA$ required for α-particle confinement
- **Classical slowing down of fast ions well validated:**
  - data range 30keV NBI (ISX-B) to 3.5MeV α-particles (TFTR)
- **Energetic ion heating processes routinely observed in additional heating experiments**



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

Electron Heating by α-Particles

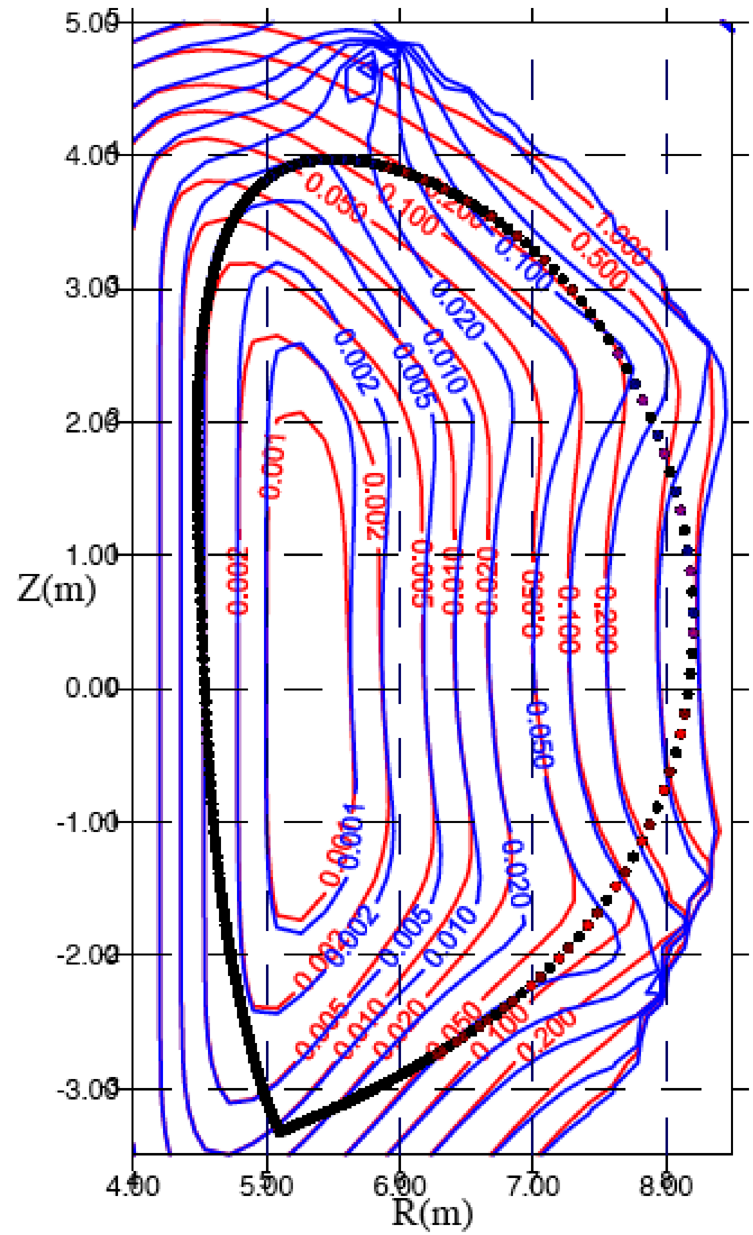
- **Electron heating by α-particles validated in JET and TFTR DT experiments:**
    - in JET, maximum electron temperature correlated with optimum fuel mixture for fusion power production



P R Thomas et al, Phys Rev Lett **80** 5548 (1998)

TF Ripple Losses

- **TF ripple causes anomalous losses of fast particles - “trapping” or “orbit drift”:**
  - sophisticated numerical codes developed and validated
- **Key issue at ITER scale is to limit localized heating of first wall:**
  - maximum allowable first wall loading typically 0.5MWm^{-2}
- **Ferromagnetic inserts beneath coils reduce TF ripple to acceptable levels:**
  - TF ripple typically reduced from 1.4% to 0.5% at separatrix



Alfvén Eigenmodes

- **In a tokamak plasma, the Alfvén wave continuum splits into a series of bands, with the gaps associated with various features of the equilibrium:**
  - a series of discrete frequency Alfvén eigenmodes can exist in these gaps**:**

• toroidicity-induced (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**

Alfvén Eigenmode Stability

Linear stability:

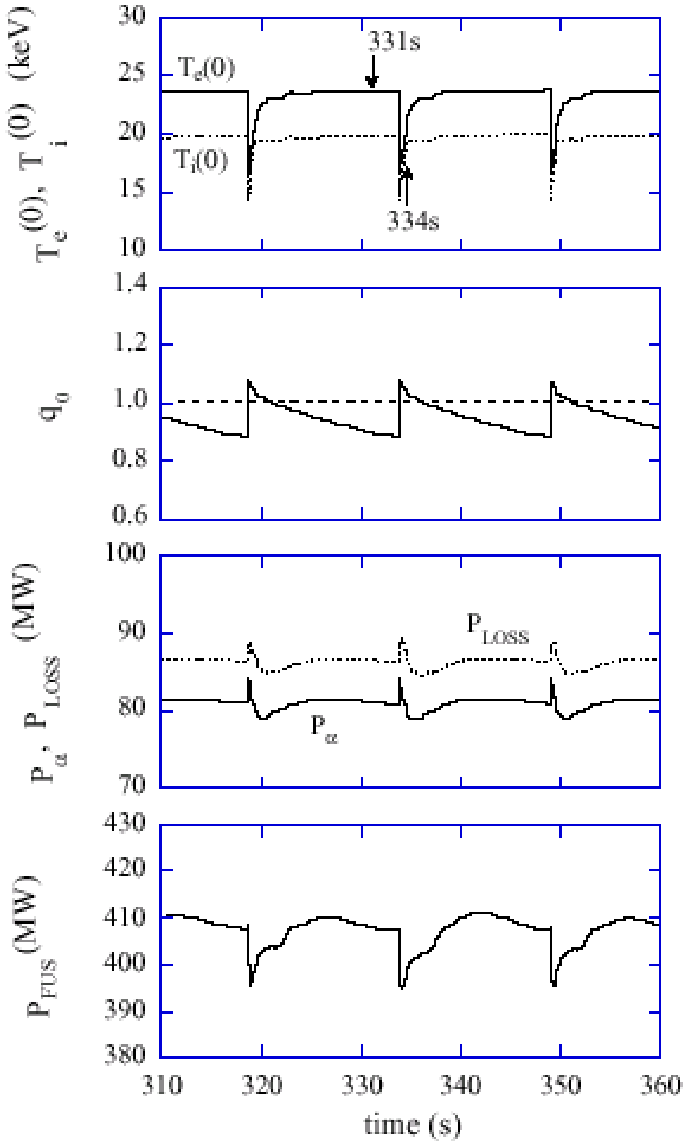
- **Pressure gradient of resonant particles with v ~ v_A provides source of free energy which excites the mode:**
  - both passing and trapped particles can resonate with the AE
  - a resonant sideband also exists at $v_A/3$ for TAEs
  - several damping mechanisms exist which complicate estimation of instability thresholds
  - recent experiments in "advanced scenarios" with non-monotonic q-profiles show a rich population of AEs can be excited

Non-linear behaviour:

- **Redistribution of resonant particles can occur by finite amplitude waves**
- **Overlap of multiple modes can lead to enhancement of energetic ion transport**

α-Particle Effects in Inductive Scenarios

- **α-particle confinement:**
  - classical confinement good - ferromagnetic inserts reduce <u>toroidal field ripple</u> to <0.5% (small areas >0.1%)
  - Alfvén modes: for reference scenario (monotonic q-profiles), calculations show:
    - linearly stable, or
    - weak redistribution of α-particles
  - Fishbones: (marginally) unstable for nominal parameters
- **Sawteeth:**
  - period extended by α-particle stabilization
  - 30% excursion in T(0)
  - small effects on fusion power (~3%) and heat flux



α-Particle Physics in Advanced Scenarios

- α-particle confinement and heating more challenging than in conventional scenarios:
  - ferromagnetic inserts ensure good classical confinement

	Inductive		Weak RS (#4)		Strong RS	
	No FI	With FI	No FI	With FI	No FI	With FI
Total particle loss fraction (%)	2.15	negligible	6.5	0.08	21	0.75
Total power loss fraction (%)	0.65	negligible	2.5	0.04	9.3	0.13
Peak FW heat load (MWm^{-2})	< 0.1	negligible	0.23	0.005	0.8	0.025
Plasma current (MA)	15		10		10	

- **Response of plasma to α-heating is a key issue for advanced scenarios:**
  - predominantly electron heating & α-heating profile essentially determines pressure profile
  - high performance plasmas must be maintained in non-linear equilibrium involving pressure, current, thermal diffusivity profiles

Key Energetic Ion Parameters

The influence of energetic ion populations on plasma stability can be expressed through a small number of parameters

- $\dfrac{\delta_f}{a} = \dfrac{q}{\varepsilon^{0.5}} \dfrac{r_f}{a}$ **- normalized half-width of fast ion banana orbit**
- n_f / n_e **- fractional density of fast ions**
- $\beta_f(0)$ **- normalized axial fast ion pressure**
- $\max \left| R.\nabla \beta_f \right|$ **- dimensionless fast ion pressure gradient**
- $v_f / v_A(0)$ **- ratio of fast ion velocity to central Alfvén velocity**

$$v_A = \frac{B}{\sqrt{\mu_o \rho_{mass}}}$$

α-Particle Physics in Advanced Scenarios

- **Excitation of AEs and their influence on α-particle confinement is a central question for viability of advanced scenarios**
- **α-particle parameters in ITER allows access to relevant range where α-driven instabilities and their influence on α-particle transport can be studied:**

Parameter	α's (TFTR)	α's (JET)	α's (ITER)
$P_f(0)$ [MWm^{-3}]	0.3	0.16	0.44
δ/a	0.3	0.34	0.08
$n_f(0)/n_e(0)$ [%]	0.3	0.17	0.8
$\beta_f(0)$ [%]	0.26	0.3	1.1
$\langle\beta_f\rangle$ [%]	0.03	0.04	0.16
max $\lvert R.\nabla\beta_f\rvert$ [%]	2	1.6	8
$v_f / v_A(0)$	1.6	1.4	1.8

Increase in reactor

- **Higher power density in reactor could be even more challenging $\Rightarrow$ ITER programme aims to move in this direction**

ITER Status

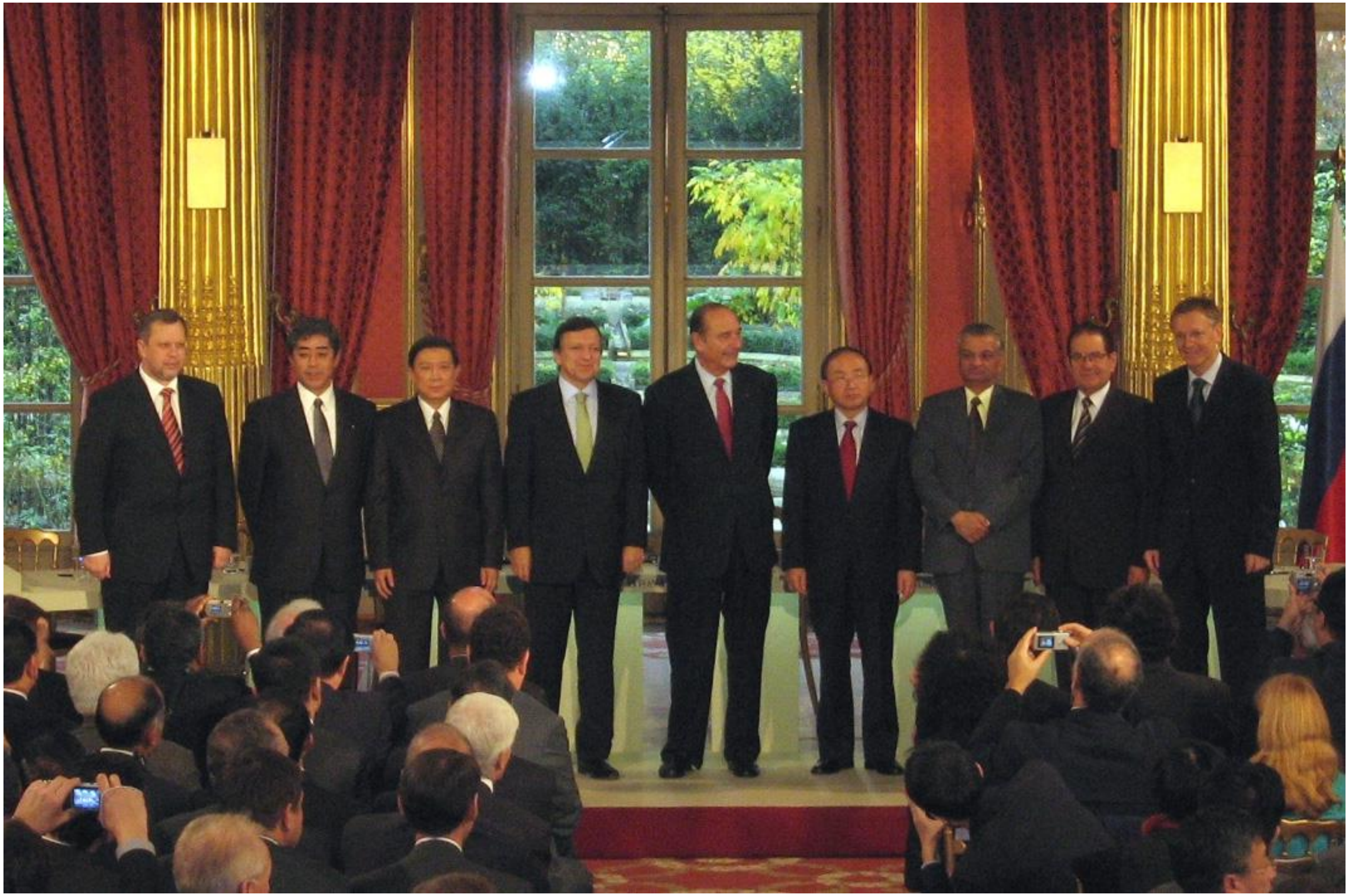
ITER - Present Status

- The ITER Team has been established on the Cadarache site
  ~200 personnel on site
- Provisional ITER Organization is now in operation
- Design Review is underway to revise Baseline by the end of 2007
- 10 years construction,
  20 years operation
- Cost: 5 billion Euros for construction, and 5 billion for operation and decommissioning

Cadarache Site

The current ITER building

ITER Agreement Signature

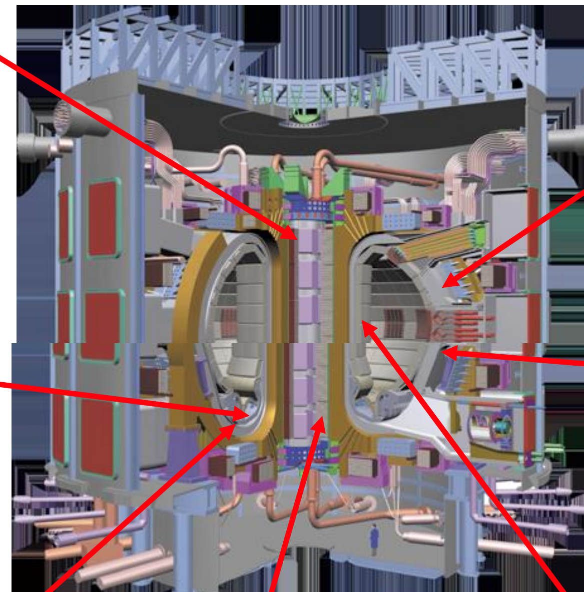


Ceremony ITER Agreement Signature, Elysee Palace, 21 November 2006

ITER Technology R&D Advanced

CENTRAL SOLENOID MODEL COIL

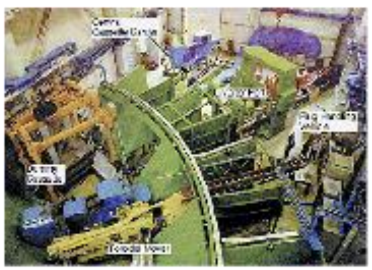
Radius 3.5 m
Height 2.8m
B_{max}=13 T
0.6 T/sec



VACUUM VESSEL SECTOR

Double-Wall, Tolerance ± 5 mm

REMOTE MAINTENANCE OF DIVERTOR CASSETTE



Attachment Tolerance ± 2 mm

BLANKET MODULE

HIP Joining Technology
Size : 1.6 m x 0.93 m x 0.35 m

DIVERTOR CASSETTE

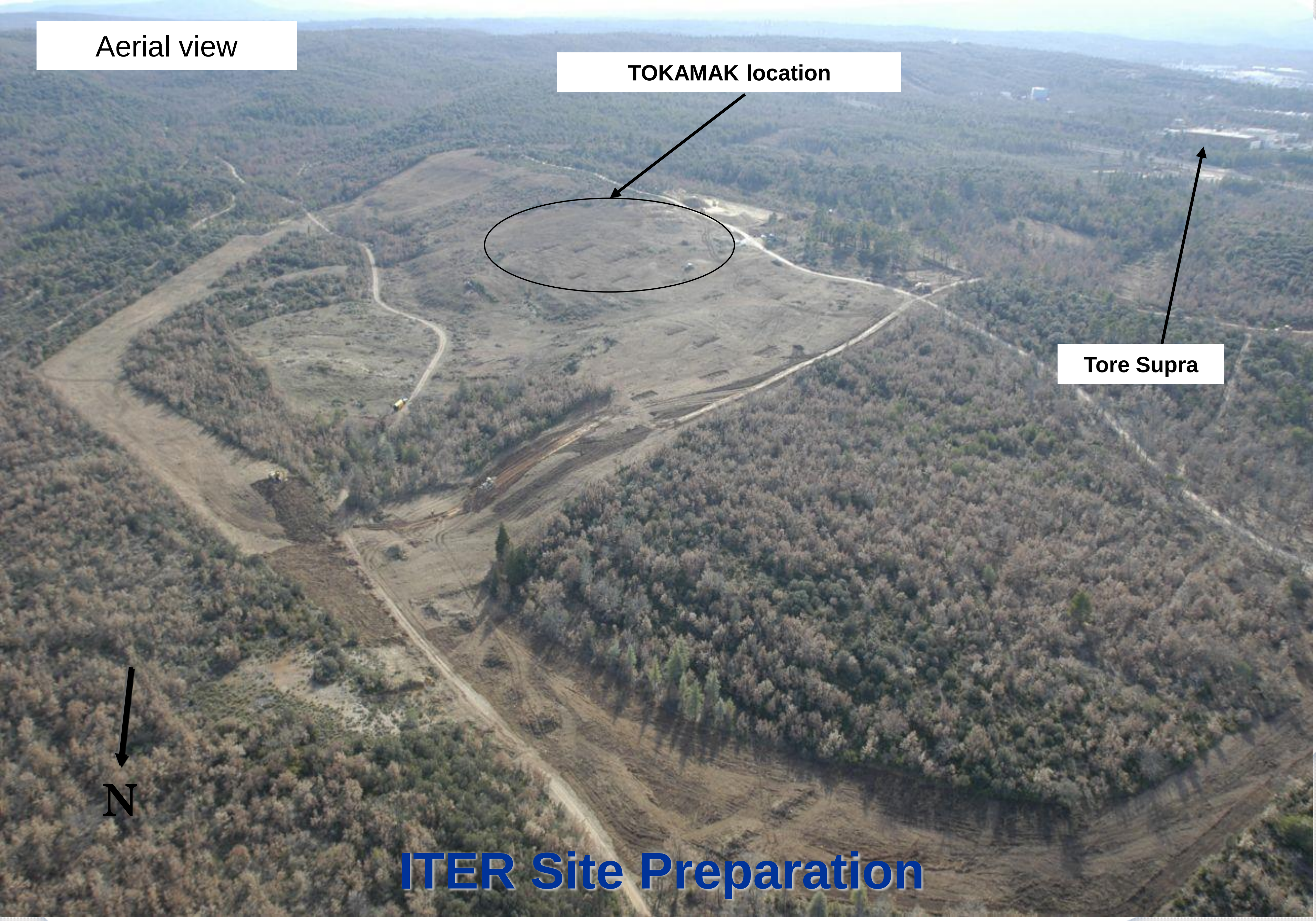
Heat Flux 20 MW/m^2

TOROIDAL FIELD MODEL COIL

Height 4 m
Width 3 m
B_{max}=7.8 T

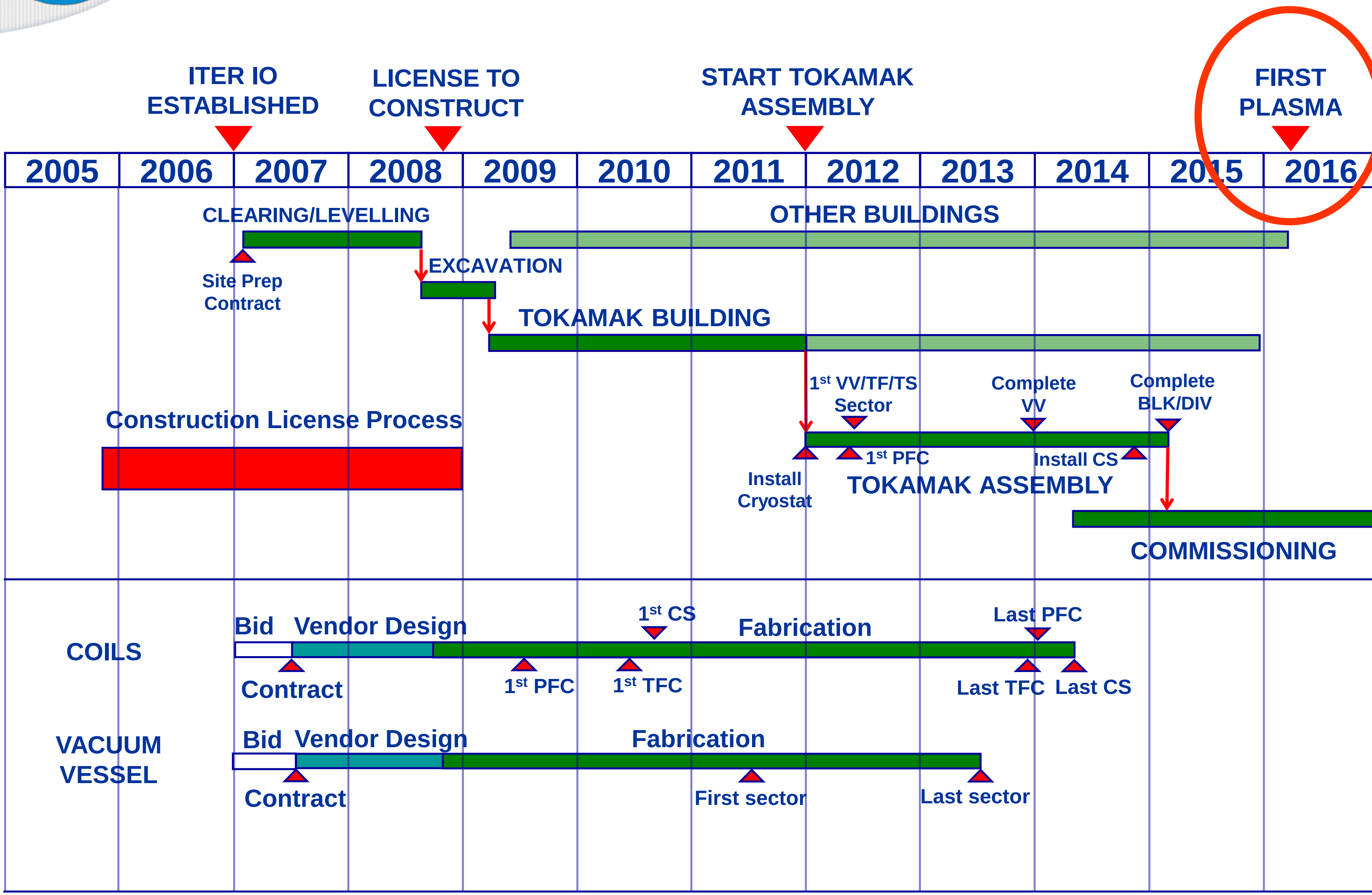
REMOTE MAINTENANCE OF BLANKET

4 t Blanket Sector
Attachment Tolerance ±0.25 mm



Aerial view
TOKAMAK location
Tore Supra
N
ITER Site Preparation

Project Schedule



Conclusions

ITER has many assets as a burning plasma experiment and the key step towards the realization of fusion energy

- **To fulfill its missions ITER must carry out an ambitious and exciting physics programme**
- **Its essential design features give it the capability to do this:**
  - pulse length and duty cycle
  - flexible heating and current drive systems
    - total power
    - variety of systems
  - diagnostic access and facilities
  - additional plasma engineering systems
    - inside pellet launch
    - sawtooth, NTM, RWM and Error Field control
  - equilibrium shape flexibility
  - divertor and first wall exchange capability