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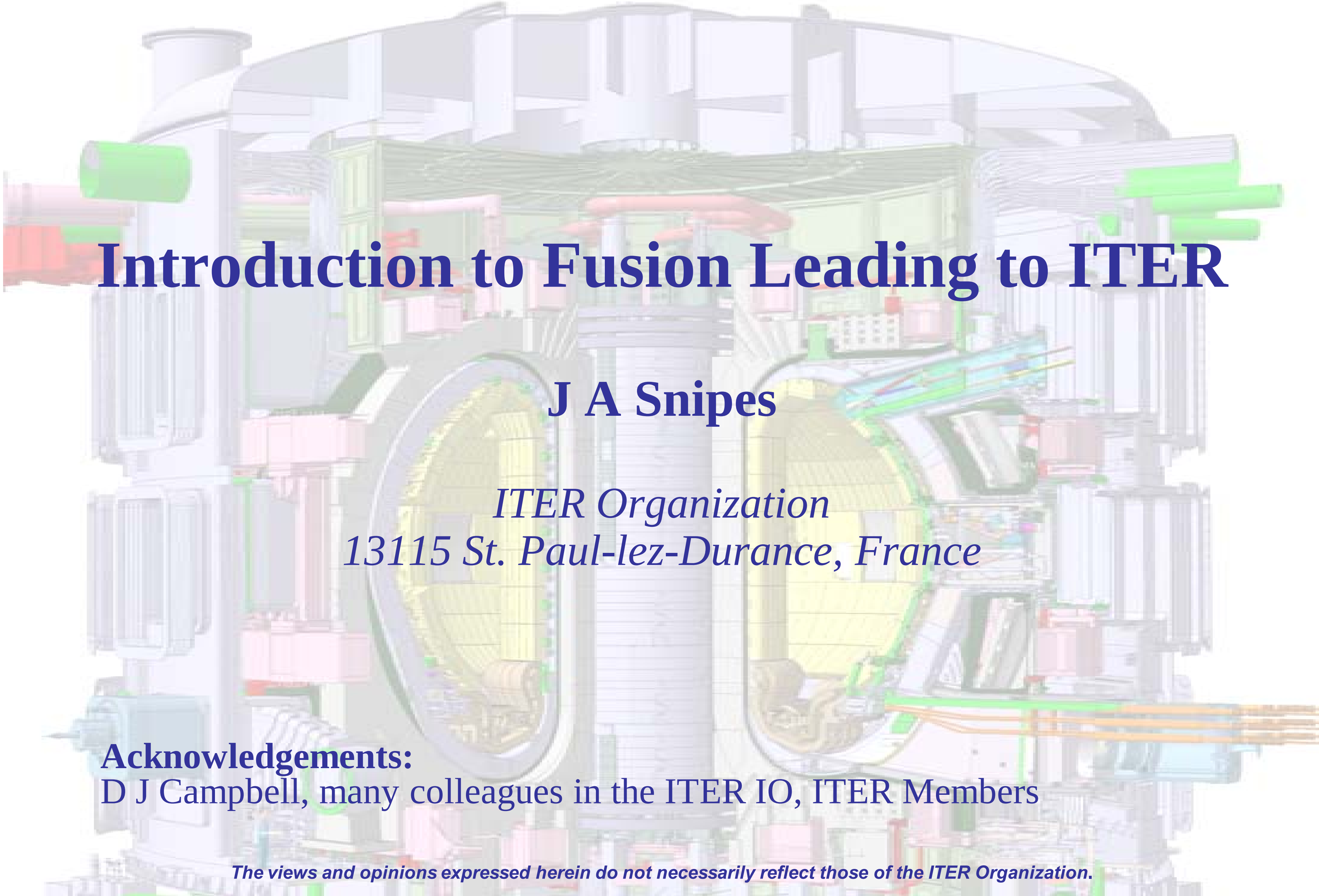
Introduction to Fusion Leading to ITER

SNIPES Joseph Allan
Directorate for Plasma Operation
Plasma Operations Group POP, Science Division
Building 523/023, Route de Vinon sur Verdon
13115 St Paul lez Durance
FRANCE



Introduction to Fusion Leading to ITER

J A Snipes

ITER Organization
13115 St. Paul-lez-Durance, France

Acknowledgements:
D J Campbell, many colleagues in the ITER IO, ITER Members

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

Synopsis

- Introduction to thermonuclear fusion
- Basics of magnetic confinement fusion – the tokamak
- Some key parameters for magnetic confinement fusion in tokamaks
- How the results from existing tokamaks led to ITER for the next step in fusion research

Fusion – the fundamental principle

- Energy gain from fusion, like fission, is based on Einstein's equation:

  $$E = \Delta mc^2$$

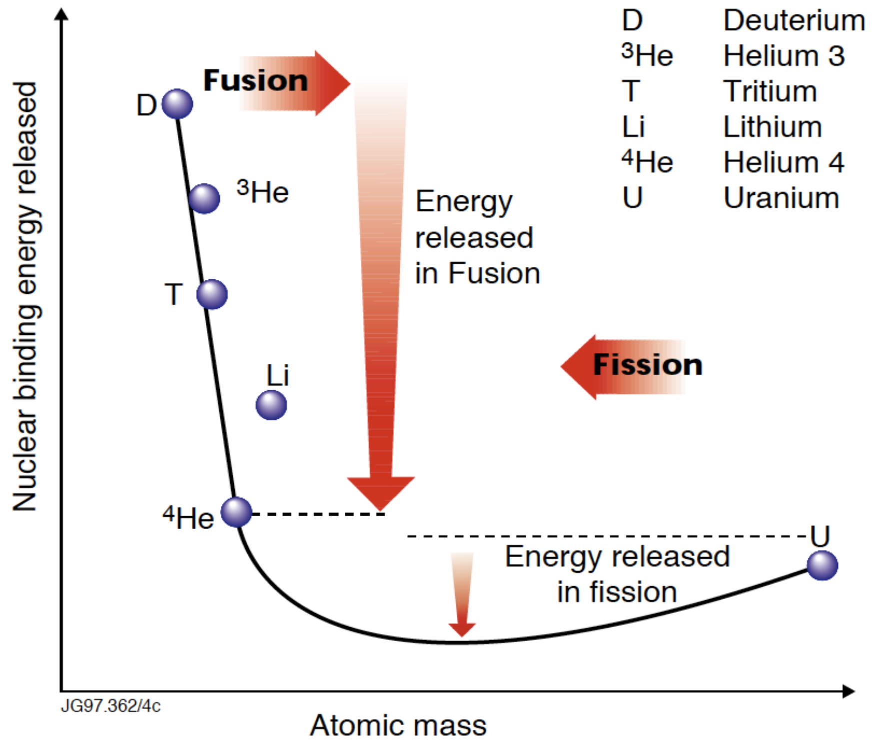
  − mass loss for DT reactions corresponds to ~ 0.4%

- As illustrated, energy gain per unit mass is greater for fusion

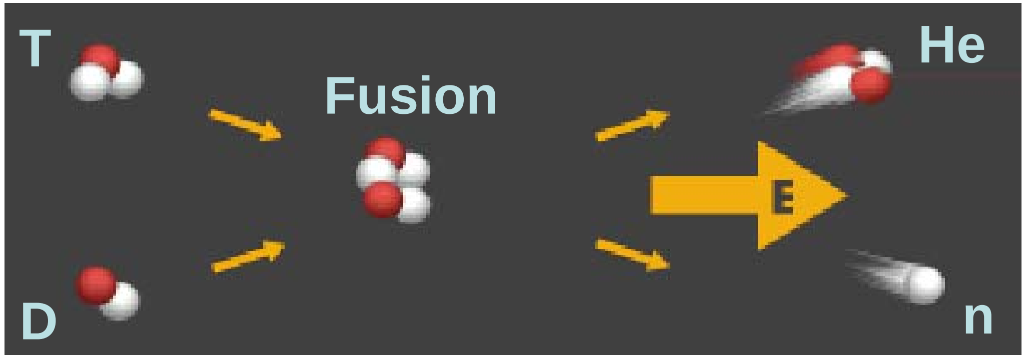
  − energy gain/ reaction:

  DT fusion: 17.6 MeV

  U fission: ~200 MeV



Essential Fusion Reactions



+ 20% of Energy (3.5 MeV)

+ 80% of Energy (14.1 MeV)

- **The D-T fusion reaction is the simplest to achieve under terrestrial conditions:**

$$^{2}D + {}^{3}T \Rightarrow {}^{4}He\ (3.5\ MeV) + {}^{1}n\ (14.1\ MeV)$$

- **Two other important reactions for DT fusion:**

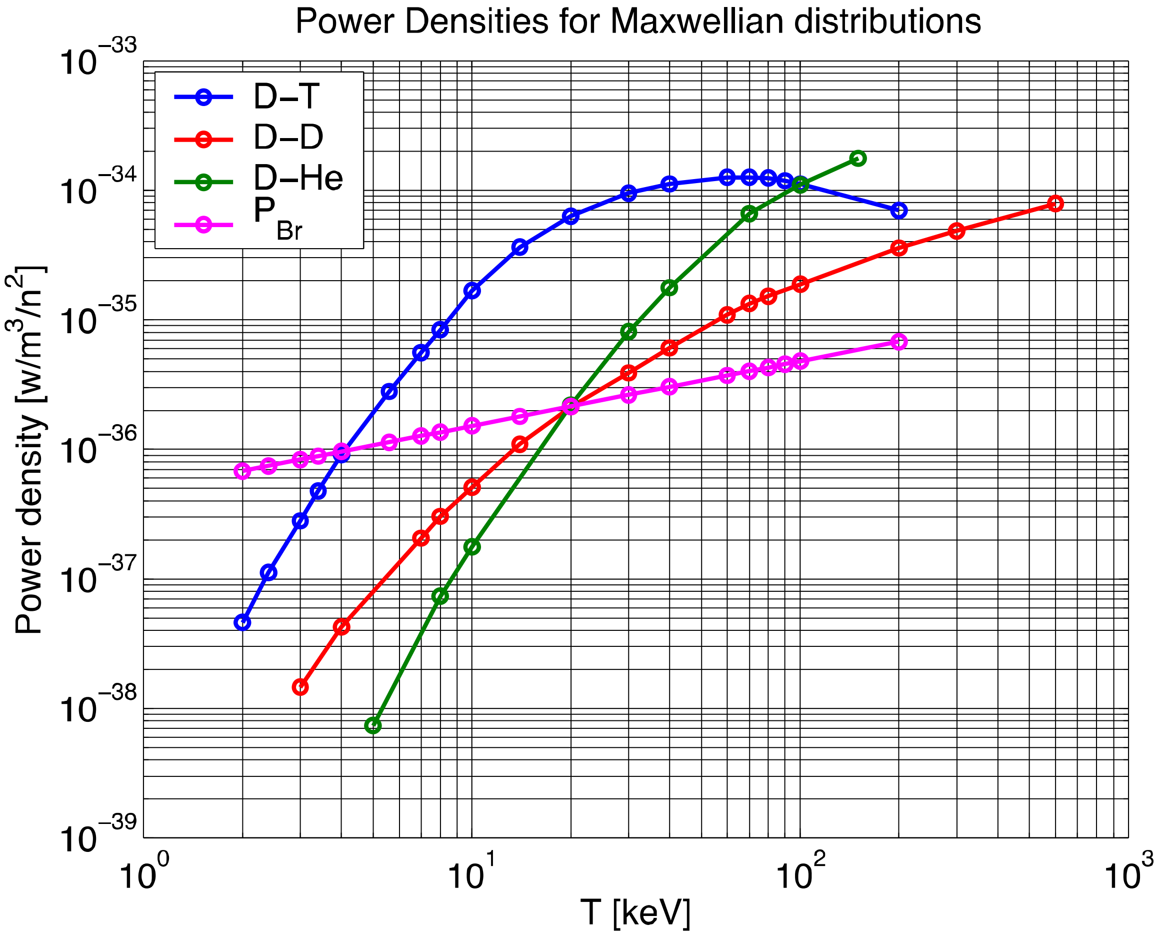
$$^{1}n + {}^{6}Li \Rightarrow {}^{4}He + {}^{3}T + 4.8\ MeV$$

$$^{1}n + {}^{7}Li \Rightarrow {}^{3}He + {}^{3}T + {}^{1}n - 2.5\ MeV$$

  - **these reactions will allow a fusion reactor to breed tritium**

Fusion Power Density vs Temperature

1 keV = 1.16 × 10^7 K



- **High temperatures (~10 keV) are required for significant thermonuclear fusion energy production ⇒ dealing with plasmas!**

Basics of Magnetic Confinement Fusion:

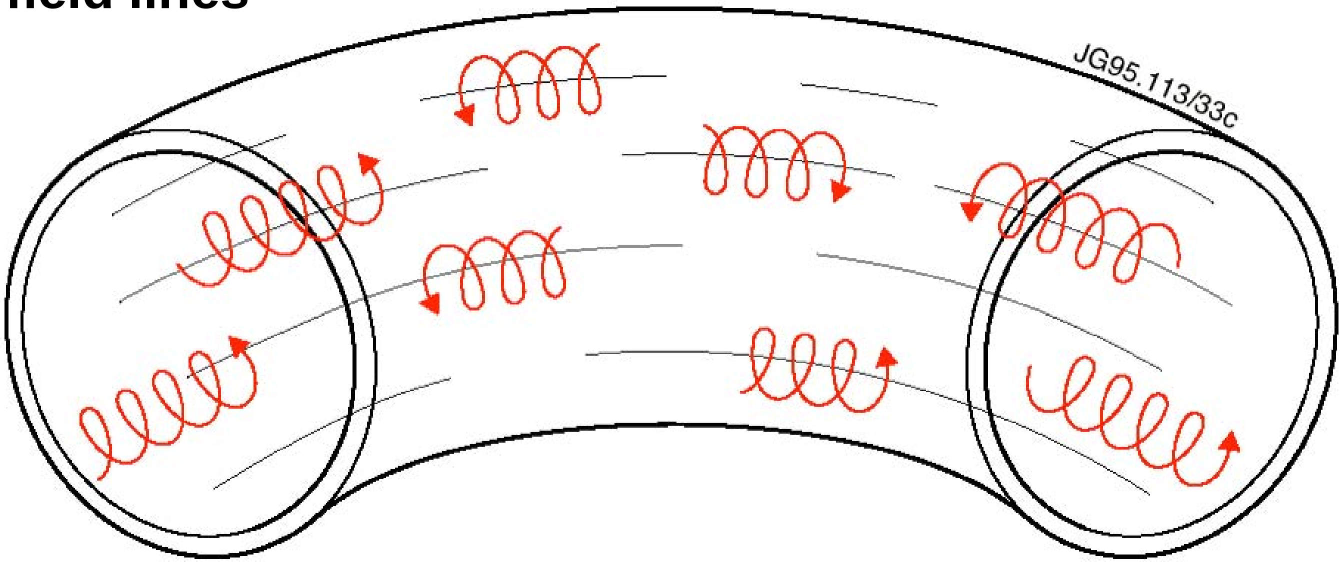
The Tokamak

Plasma Toroidal Magnetic Confinement

- **Magnetic fields cause ions and electrons to spiral around the field lines:**

$$F = q\left(E + v \times B\right)$$

  - **in a toroidal configuration plasma particles are lost to the vessel walls by relatively slow diffusion across the field lines**



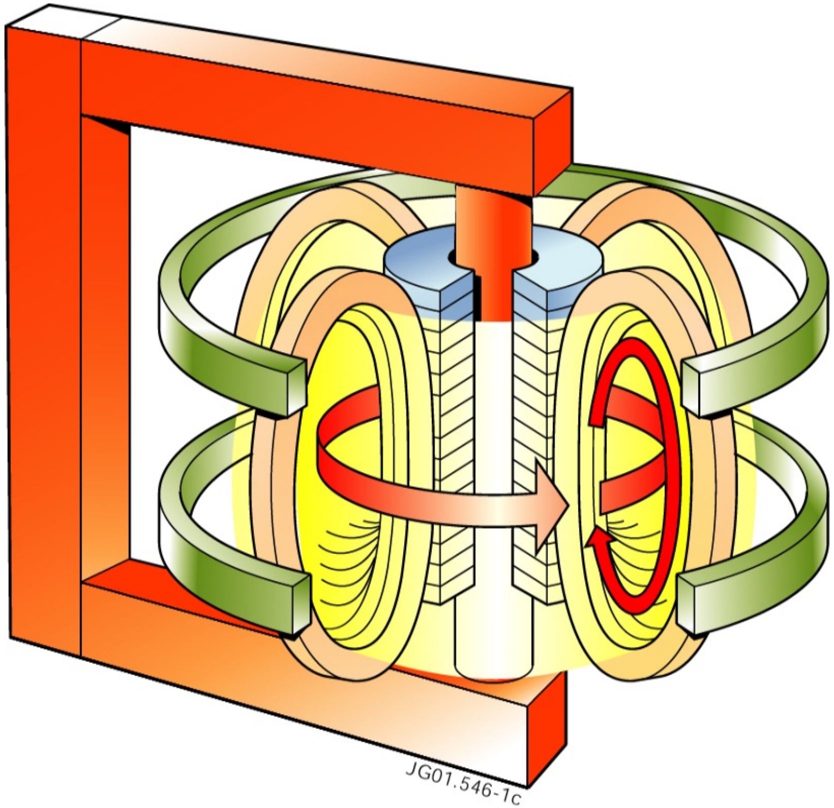
A special version of this torus is called a <u>tokamak:</u>

'toroidal chamber' and 'magnetic coil' (Russian)

Magnetic Confinement in a Tokamak

The Tokamak:

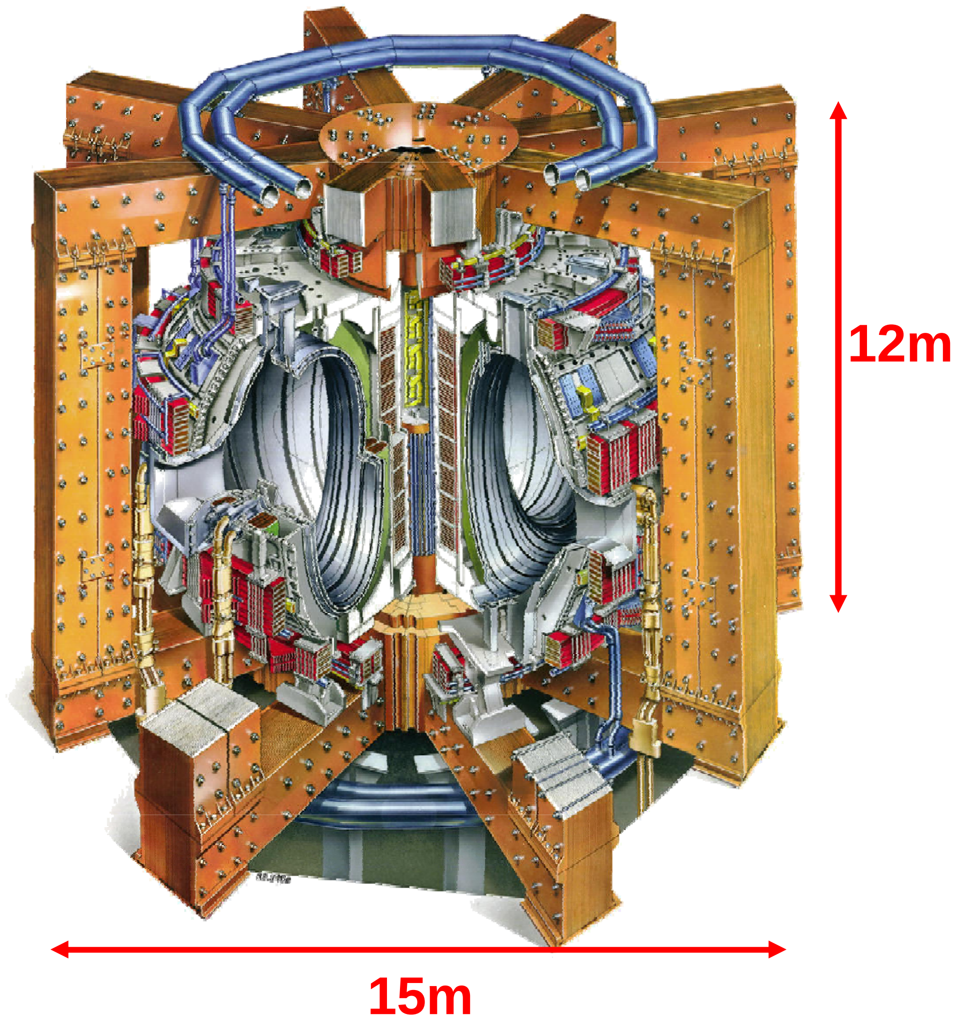
- **External coils**
  - **to produce a toroidal magnetic field**
- **Transformer with primary winding**
  - **to produce a toroidal current in the plasma**
  - **this plasma current creates a poloidal magnetic field**



- **Finally, poloidal coils**
  - **to control the position and shape of the plasma**

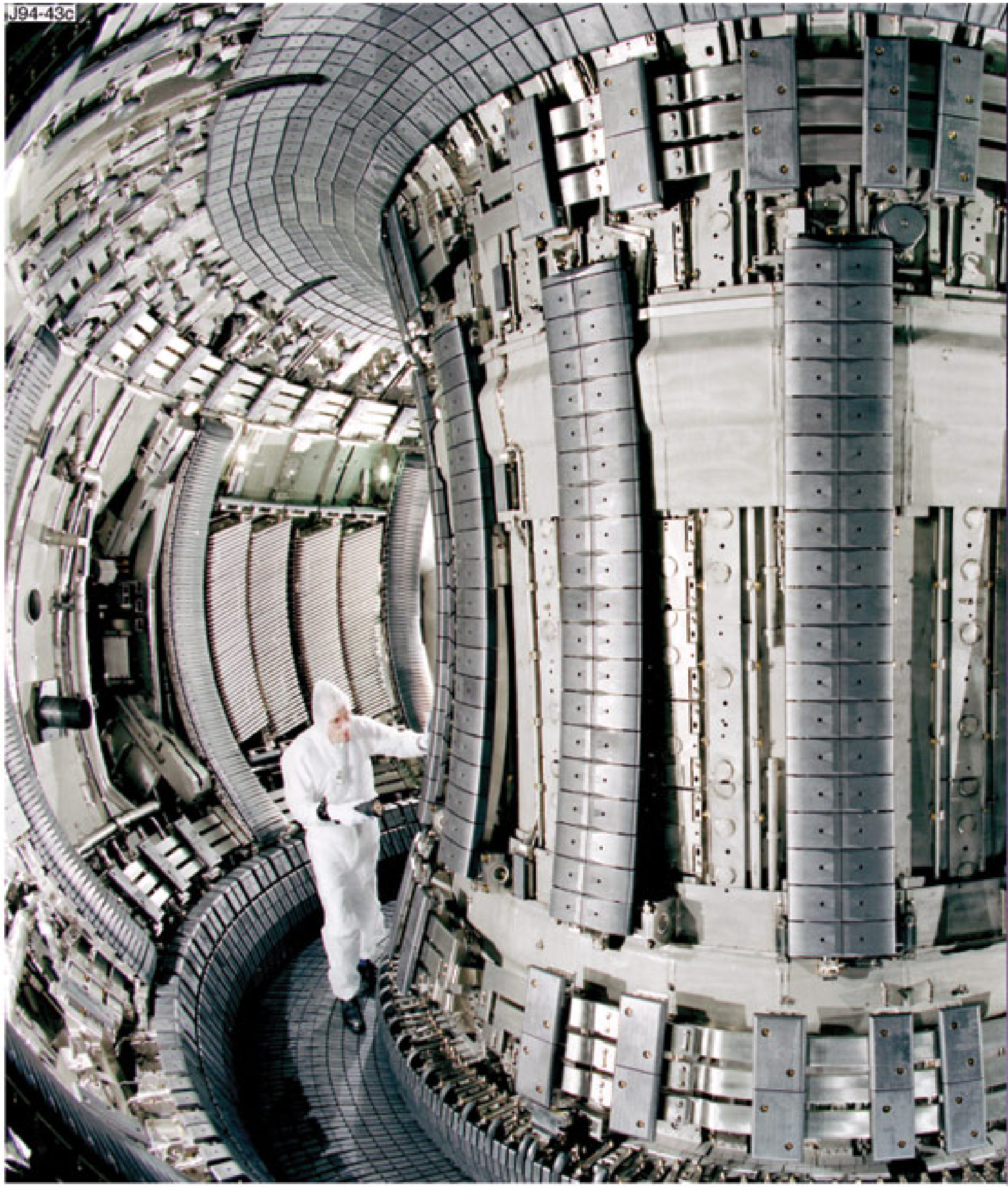
JET: Joint European Torus

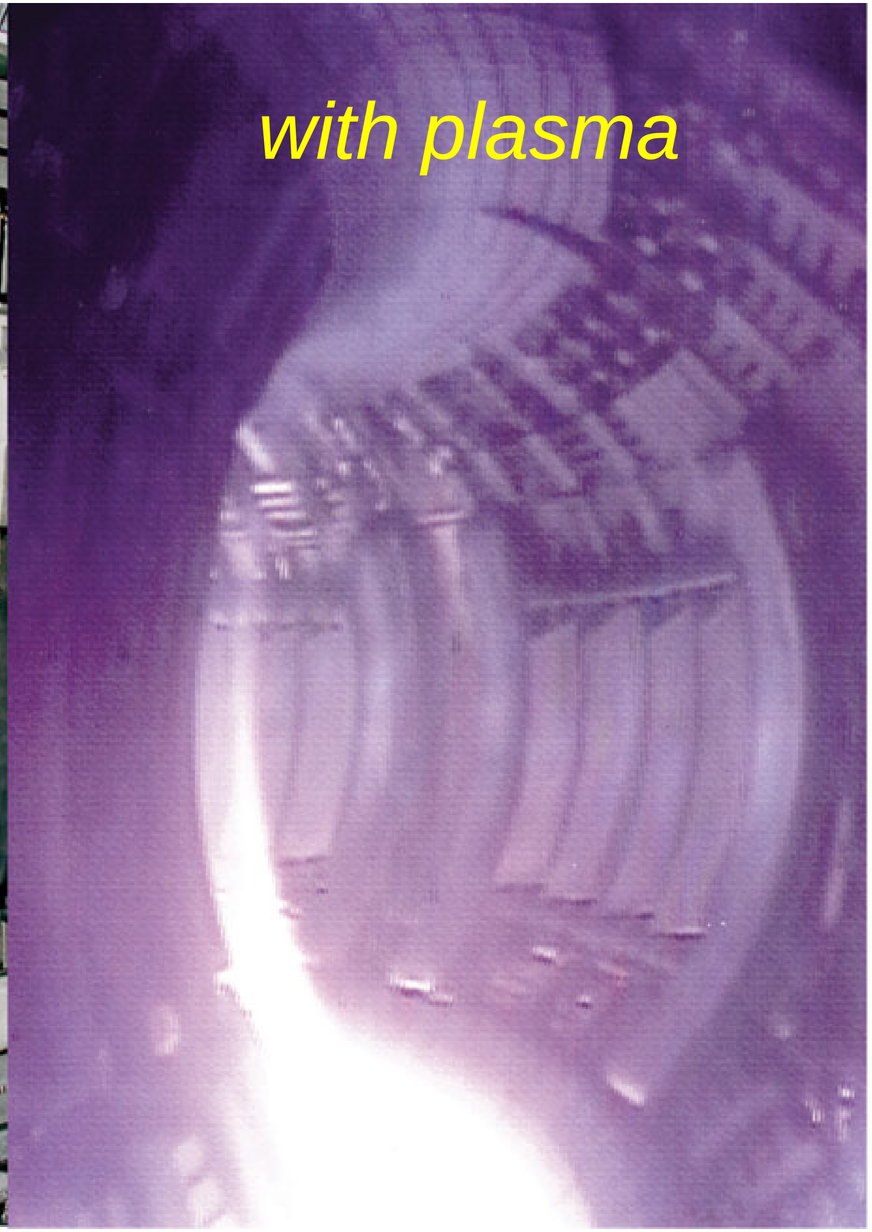
- **JET is currently the largest tokamak**
  - **Major/ minor radius: 3 m/ 1 m**
  - **Plasma volume ~100 m^3**
  - **Toroidal field: 3.4 T**
  - **Plasma Current: 7 MA**
- **In DT experiments in 1997, a peak fusion power of 16 MW was produced**



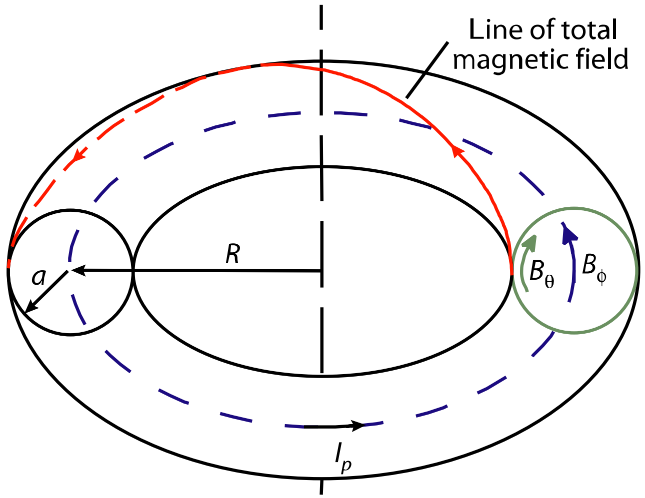
JET - the largest existing Tokamak

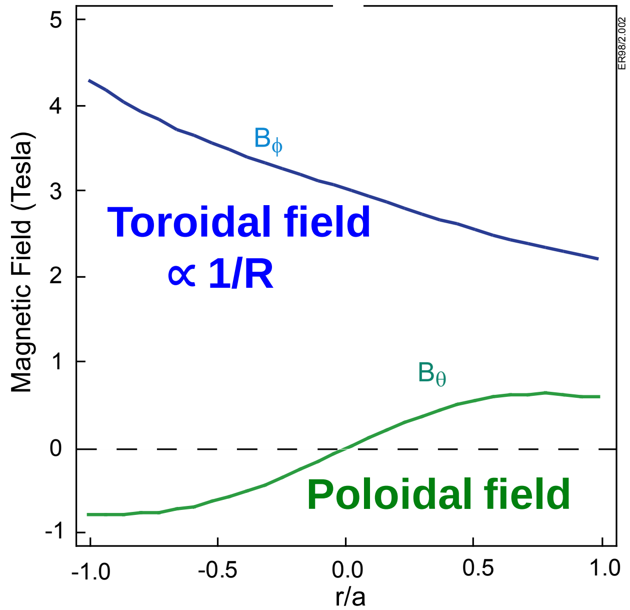
Internal View





Magnetic Confinement in a Tokamak





- In configurations with only a **toroidal field**, ions and electrons drift vertically in opposite directions:

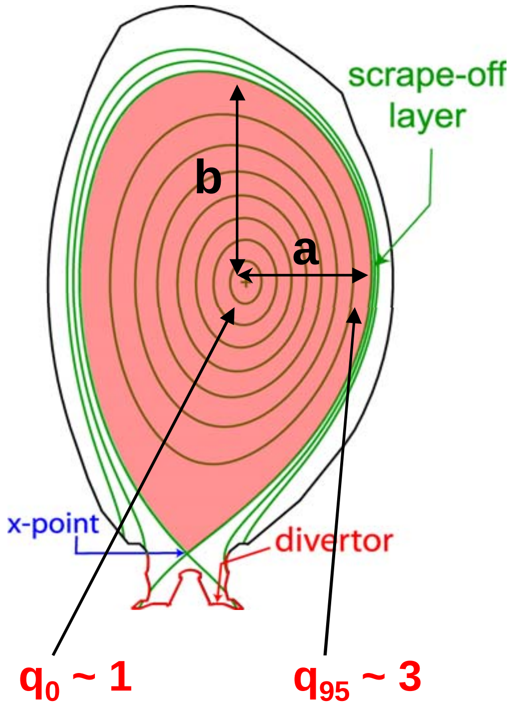
- An additional **poloidal field** allows particles to follow **helical paths**, cancelling the drifts

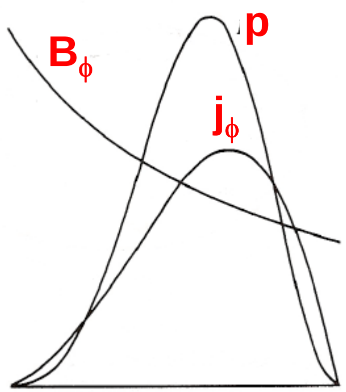
- "Winding number" of helix is an important **stability parameter** for the system:

$$q_c = \frac{aB_\phi}{RB_\theta} \sim \frac{a^2B_\phi}{RI_p}$$

  - q_c = "cylindrical" safety factor
  - R/a = aspect ratio

Plasma Equilibrium in a Tokamak





- **Formal definition of safety factor:**

$$q = \frac{d\Phi}{d\Psi}$$

toroidal flux ($d\Phi$)

poloidal flux ($d\Psi$)

  - absolute value of q and its variation across the plasma radius are important in plasma stability

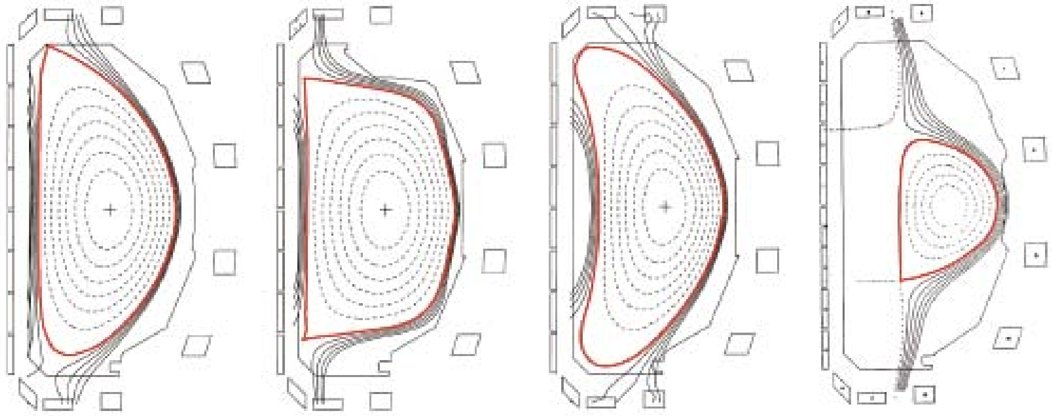
  - by elongating the plasma, more current can be squeezed into the plasma ring at fixed q:

$$\kappa = \frac{b}{a}$$

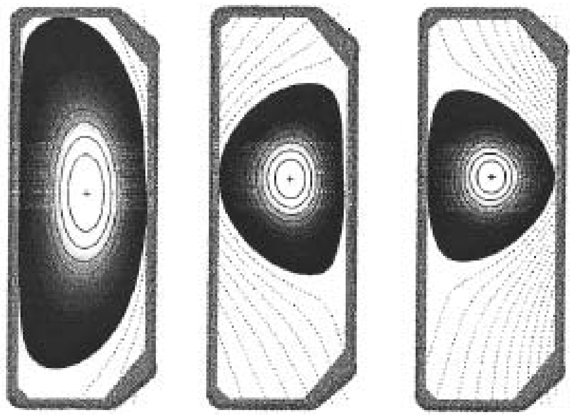
  - κ also turns out to have important consequences for plasma stability

- **Typically the pressure (temperature, density) and current profiles are peaked on the plasma axis:**
  - the profile of q is then the inverse, with q(0) ~ 1

Many Plasma Shapes Have Been Investigated

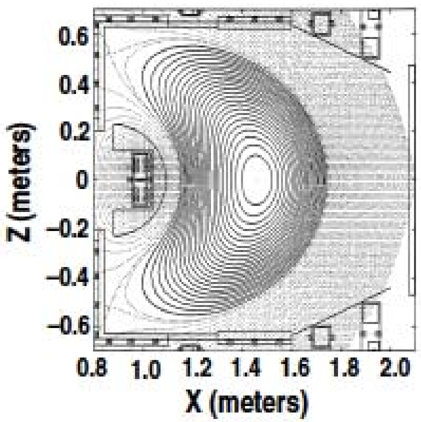


DIII–D

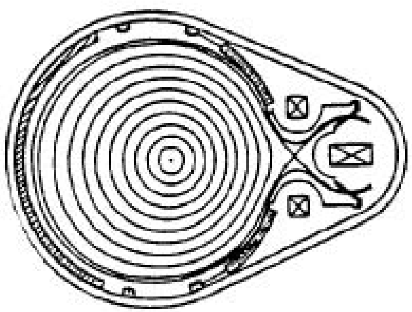


TCV

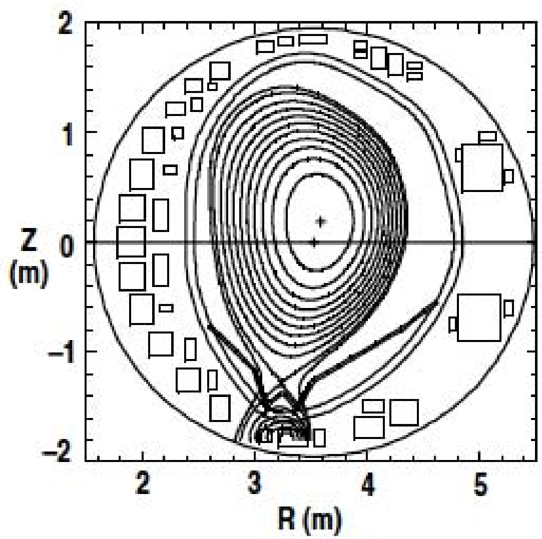
R Stambaugh, APS (2000)



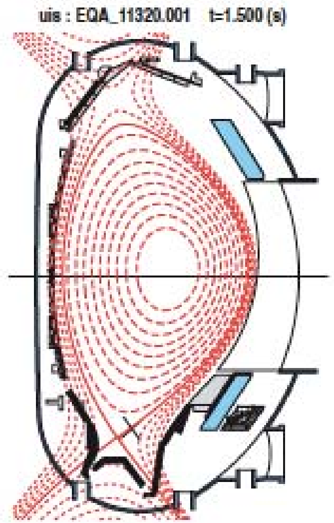
PBX–M



JT-60



JT–60U



ASDEX UPGRADE

- **Plasma shape affects confinement and stability properties**

Plasma fusion performance

Temperature - T_i: *$1\text{-}2 \times 10^8$ K (10-20 keV)*
(~10 × 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 ($\propto$ current × radius^2)*
(ITER plasma pulse duration ~1000s)

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

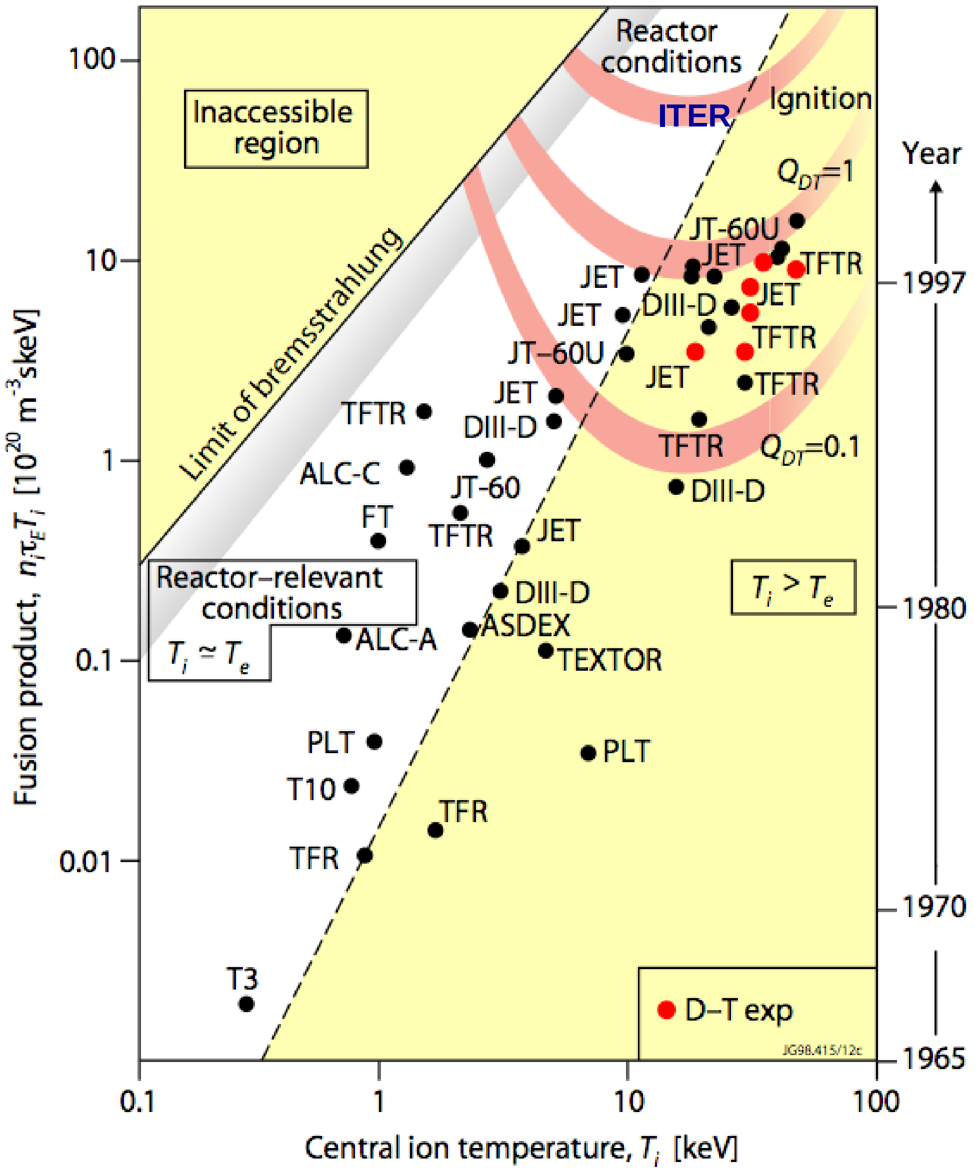
⇒ Present devices: Q ≤ 1

⇒ ITER goal: Q ≥ 10

⇒ "Controlled ignition": Q ≥ 30

Fusion Triple Product

- Existing experiments have achieved nTτ values ~ 1×10^{21} m^{-3}skeV ~ Q_{DT} = 1
- JET and TFTR have produced DT fusion powers of >10MW for ~1s
- ITER is designed to a scale which should yield Q_{DT} ≥ 10 at a fusion power of 400 - 500MW for 300-500s



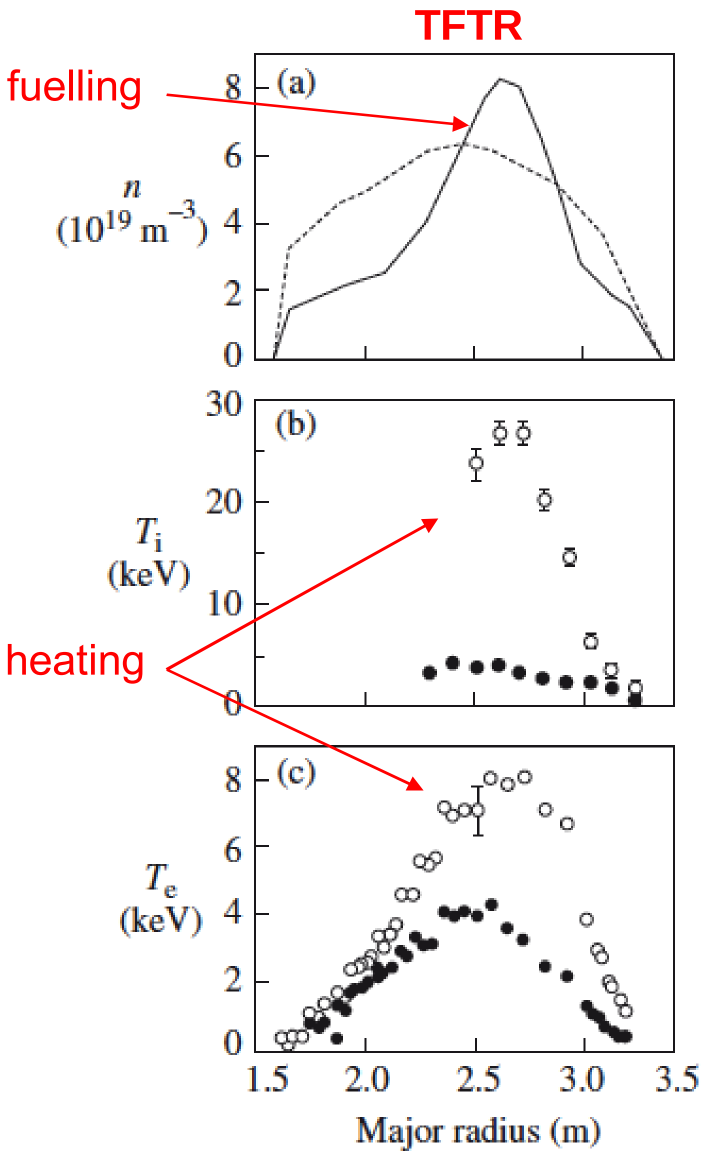
Plasma Heating

- **Tokamaks have a built in heating scheme: "Ohmic" heating by the plasma current**
  - but plasma resistivity varies as $T_e^{-3/2}$, so heating power declines with increasing T_e
  - so Ohmic plasma temperatures of several keV are possible, but additional heating is required to achieve 10-20 keV
- **Two basic heating schemes:**
  - injecting neutral particle beams
  - injecting radiofrequency waves – because the plasma refractive index depends on density and magnetic fields, several RF options are possible
- **Each heating technique also provides some current drive**

Injection of Neutral Particle Beams

- **Neutral beam injection (NBI):**
  - intense particle beams are accelerated, neutralized and injected into plasma
  - E_b ~ 100 keV, P_b up to 40MW in TFTR
  - very effective:
    - heating
    - current drive
    - fuelling
    - rotation drive

- **For ITER:**
  - E_b ~ 1 MeV is required to penetrate plasma/ drive current
  - ➔negative ion source technology
  - higher energy ⇒ little fuelling, little rotation drive



Radiofrequency Heating

- **Ion Cyclotron Radiofrequency Heating (ICRF):**
  - launched at frequencies ~ $\omega_{ci} \Rightarrow$ f ~ 50 MHz
  - technology conventional
  - wave coupling to plasma problematic – penetration through edge

- **Electron Cyclotron Resonance Heating (ECRH):**
  - launched at frequencies ~ $\omega_{ce} \Rightarrow$ f > 100 GHz
  - source technology non-conventional: "gyrotrons"
  - coupling, absorption, space localization very good

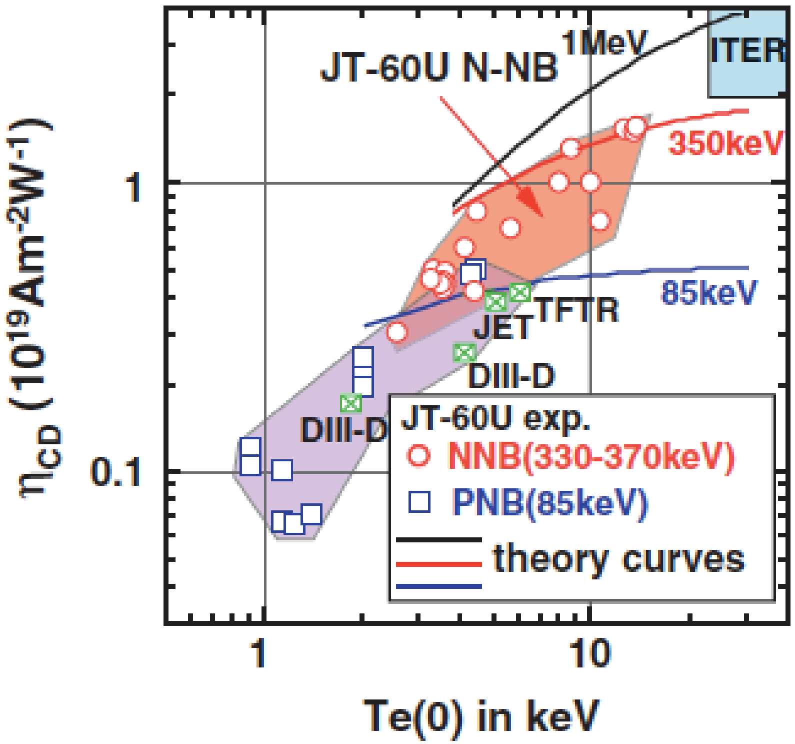
- **Lower Hybrid Heating/ Current Drive (LHCD):**
  - "lower hybrid" a complex wave resonance in plasma: f ~ 5 GHz
  - technology fairly conventional (source: klystrons)
  - wave coupling to plasma problematic – penetration through edge

Current Drive

- **Current drive provides:**
  - replacement of the transformer drive $\Rightarrow$ towards steady-state plasma
  - manipulation of the current profile to improve confinement/ stability
  - direct suppression of plasma instabilities

- **Current drive efficiency (η_{CD} = driven current/input power):**
  - typically increases with T_e
  - for beams, also increases with E_b

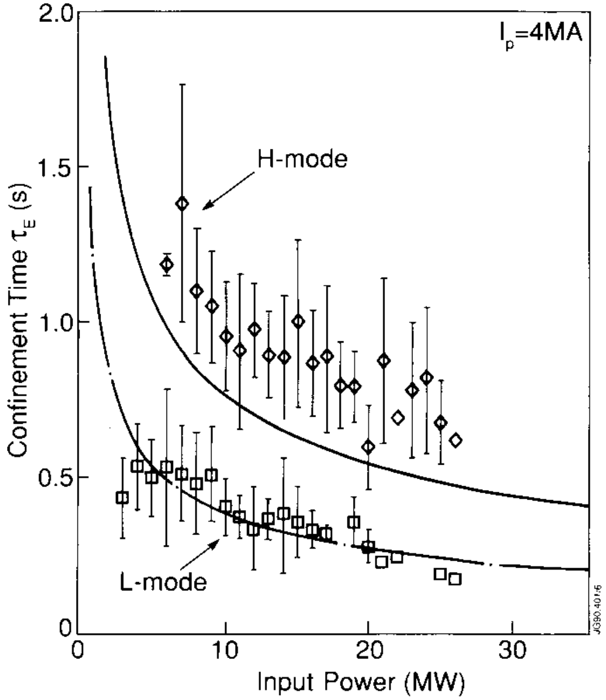
    $\Rightarrow$ favourable for ITER



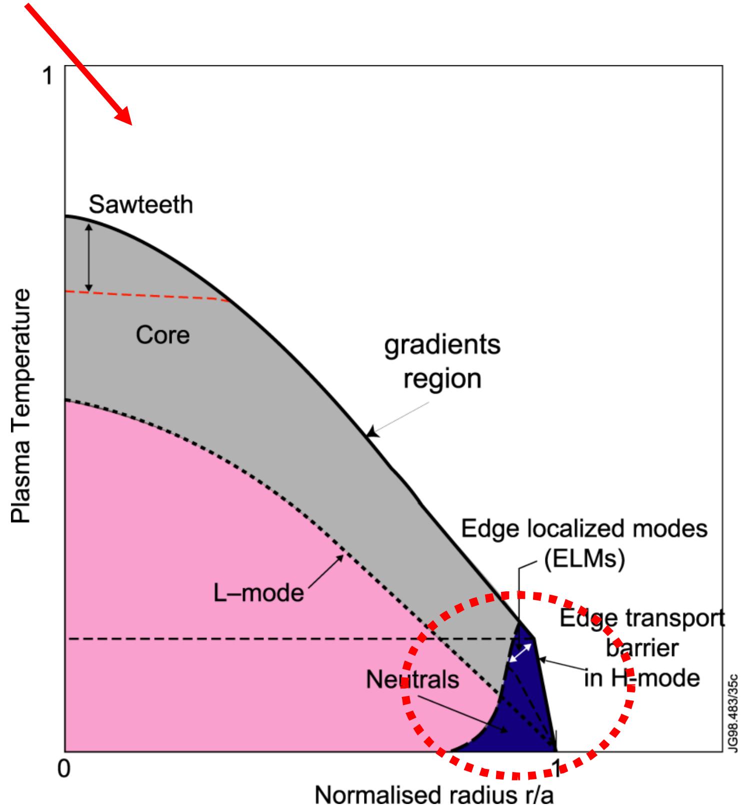
C Gormezano et al, Nucl Fusion **47** S285 (2007)

Plasma Confinement: H-mode

- **It is found that the plasma confinement state (τ_E) can bifurcate:**
  - two distinct plasma regimes, a low confinement (L-mode) and a high confinement (H-mode), result
  - this phenomenon has been shown to arise from changes in the plasma flow in a narrow edge region, or pedestal



JET



ITER Physics Basis I

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

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

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

$$\text{NB:}\ H_{98(y,2)} = \tau_{E,th}^{exp} / \tau_{E,th}^{98(y,2)}$$

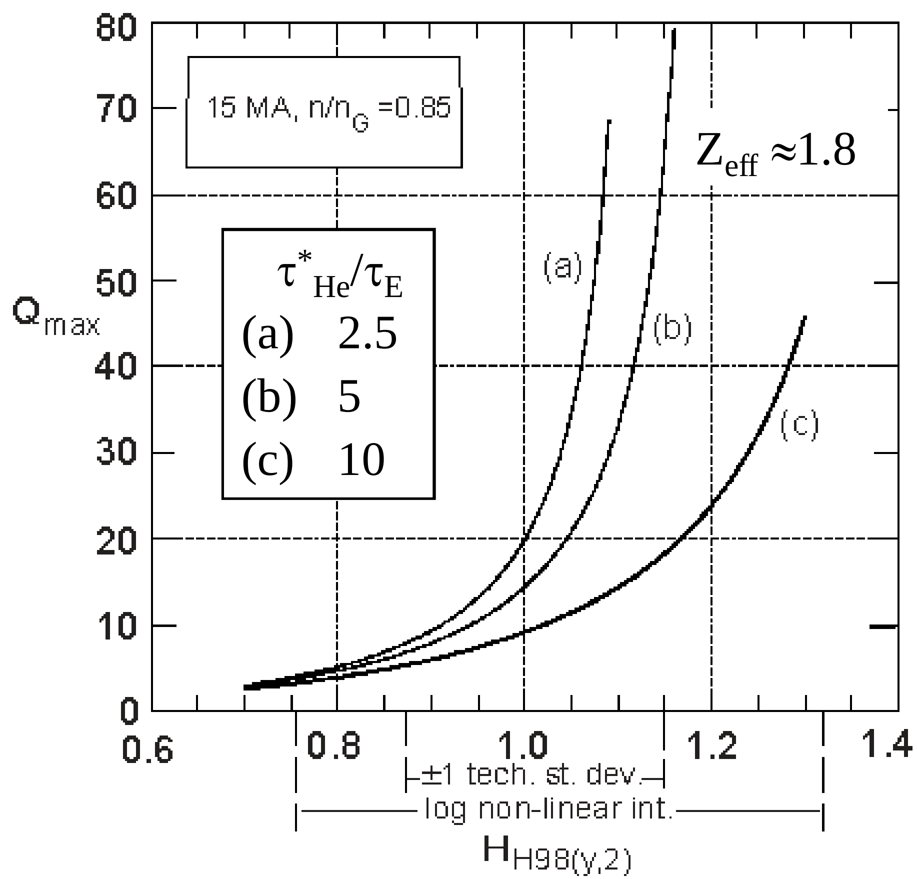
  - H-mode threshold power:

$$P_{LH} = 0.098 M^{-1} B^{0.80} \bar{n}_{20}^{0.72} S^{0.94} \quad (MW)$$

(i.e., a certain level of power needs to flow across the plasma boundary to trigger an H-mode)

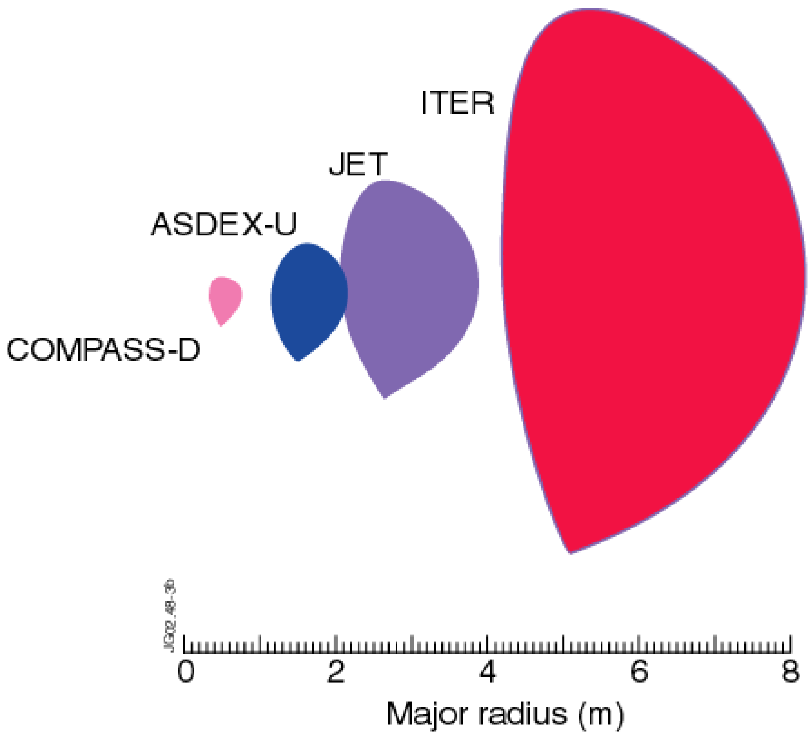
Fusion Performance Depends on Confinement

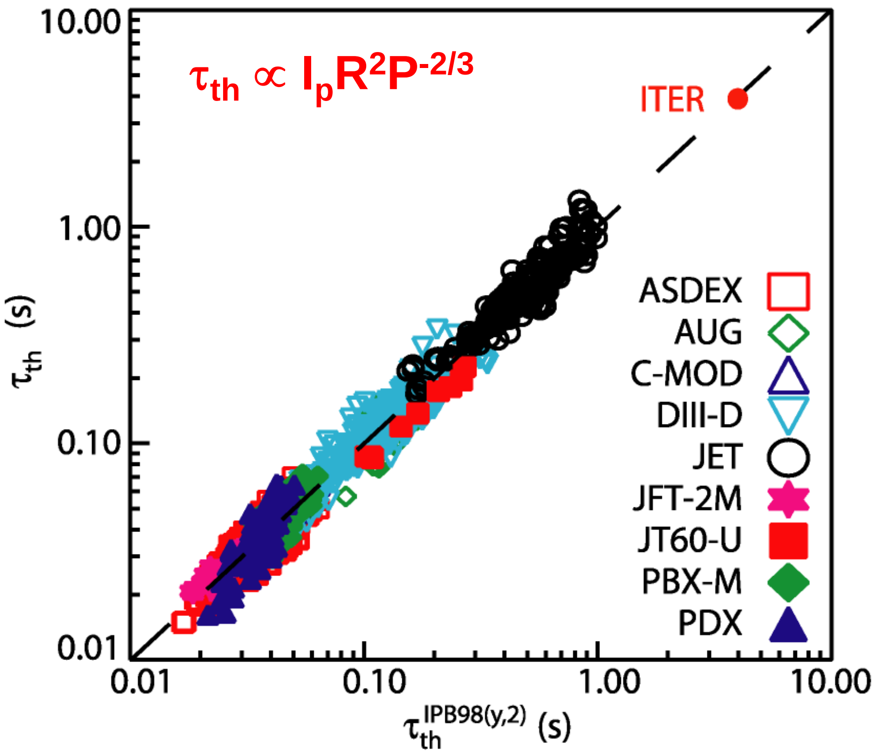
- Uncertainty in achievable energy ($H_{H98(y,2)}$) and helium particle (τ^*_{He}/τ_E) confinement gives a large uncertainty in resulting fusion performance
- Q=10 Inductive Scenario uses $H_{H98(y,2)} = 1$ and $\tau^*_{He}/\tau_E = 5$ based on empirical data from existing tokamaks
- Too much core helium ash accumulation could reduce fusion performance
- Q > 50 is not excluded within the uncertainty



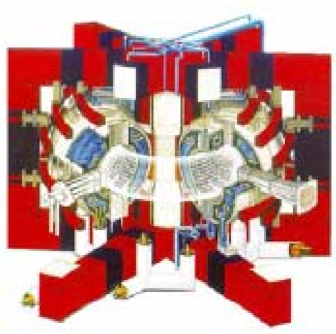
How is ITER scale determined ?

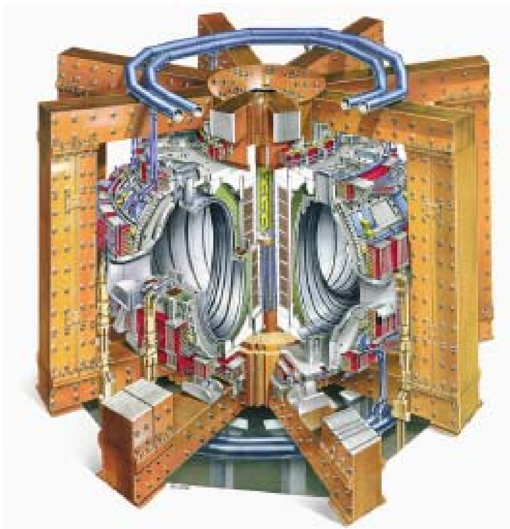
- **Energy confinement time is one of many parameters studied in a wide range of tokamak experiments**
  - **multi-tokamak experimental database provides <u>scaling</u> prediction for ITER energy confinement time, τ**

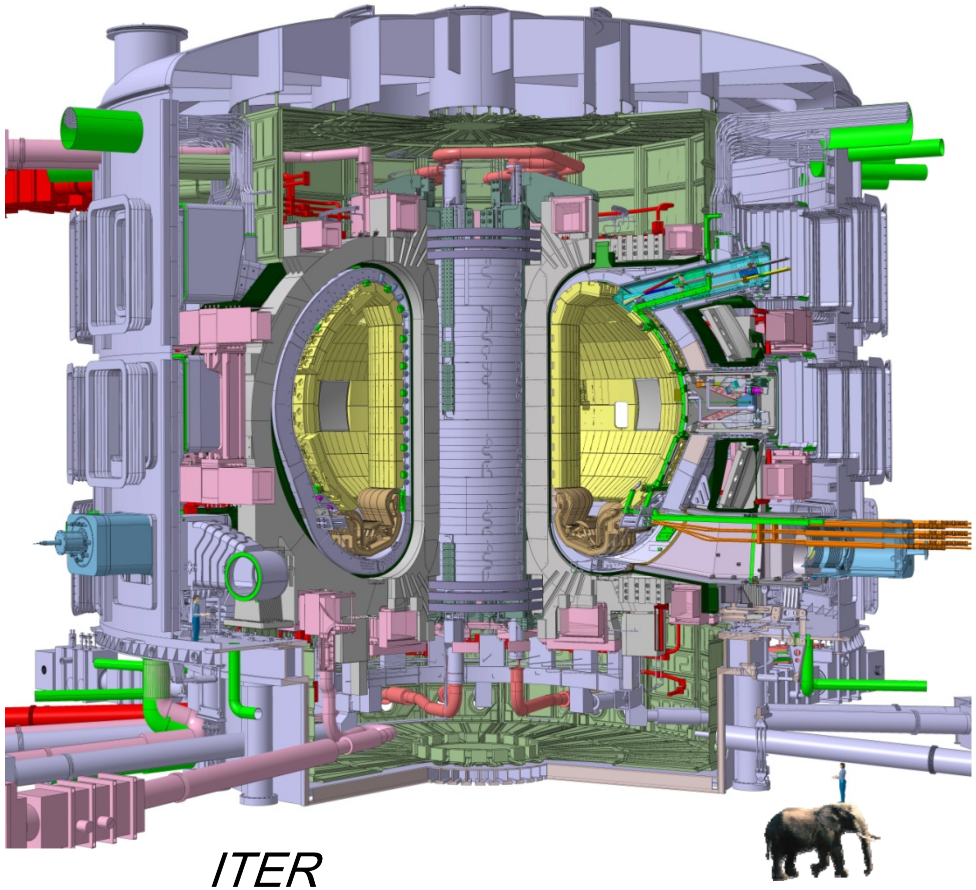




ITER is twice as large as our largest existing experiments







Tore Supra

V_{plasma} 25 m^3

P_{fusion} ~0 MW

t_{plasma} ~400 s

JET

V_{plasma} 80 m^3

P_{fusion} ~16 MW 1s

t_{plasma} ~30 s

ITER

V_{plasma} 830 m^3

P_{fusion} ~500 MW 300 – 500 s

t_{plasma} ~600 – 3000 s

ITER Physics Basis II

- MHD stability:

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

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

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

κ, δ determined by control considerations

β = (plasma kinetic energy)/(plasma magnetic energy)

- Divertor physics:

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

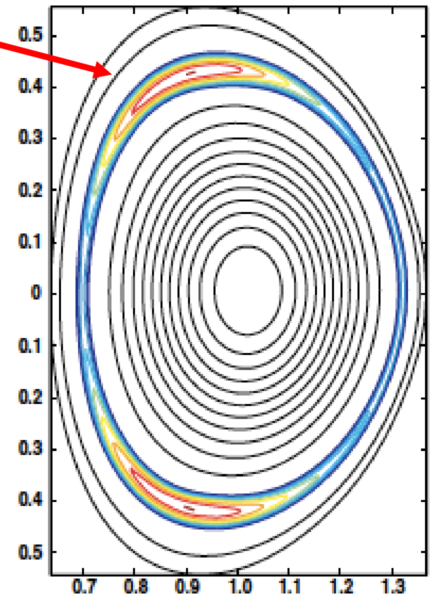
Helium transport: τ^*_{He}/τ_E ~ 5

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

MHD Stability - Plasma Operational Limits

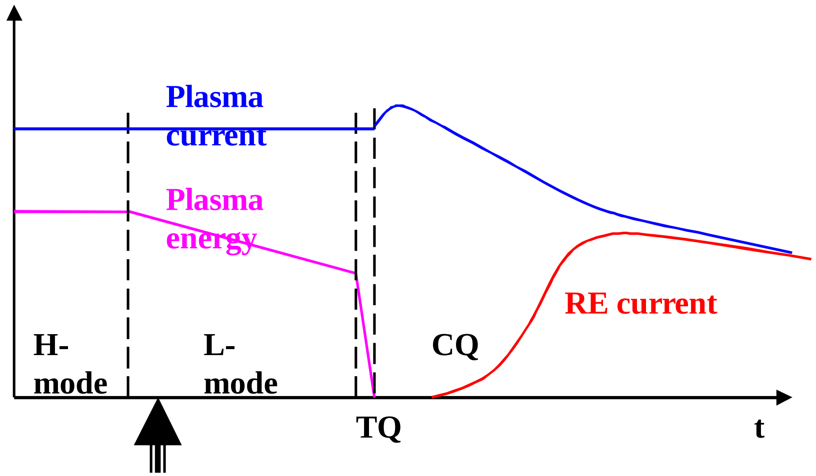
- **The interaction of the plasma fluid and the magnetic field is described by magnetohydrodynamic (MHD) stability theory**
  - provides a good qualitative, and to a significant extent quantitative, description of stability limits and the associated instabilities

- **There are two basic types of instability:**
  - "ideal" instabilities produce field line bending – can grow very rapidly
  - "resistive" instabilities cause tearing and reconnection of the magnetic field lines $\Rightarrow$ formation of "magnetic islands"



- **Plasma control techniques are being applied to suppress or avoid the most significant instabilities**
  - Neo-classical tearing modes (NTMs)
  - Edge localized modes (ELMs)
  - Disruptions and vertical displacement events
  - Allows access to higher fusion performance

MHD Stability: Disruptions

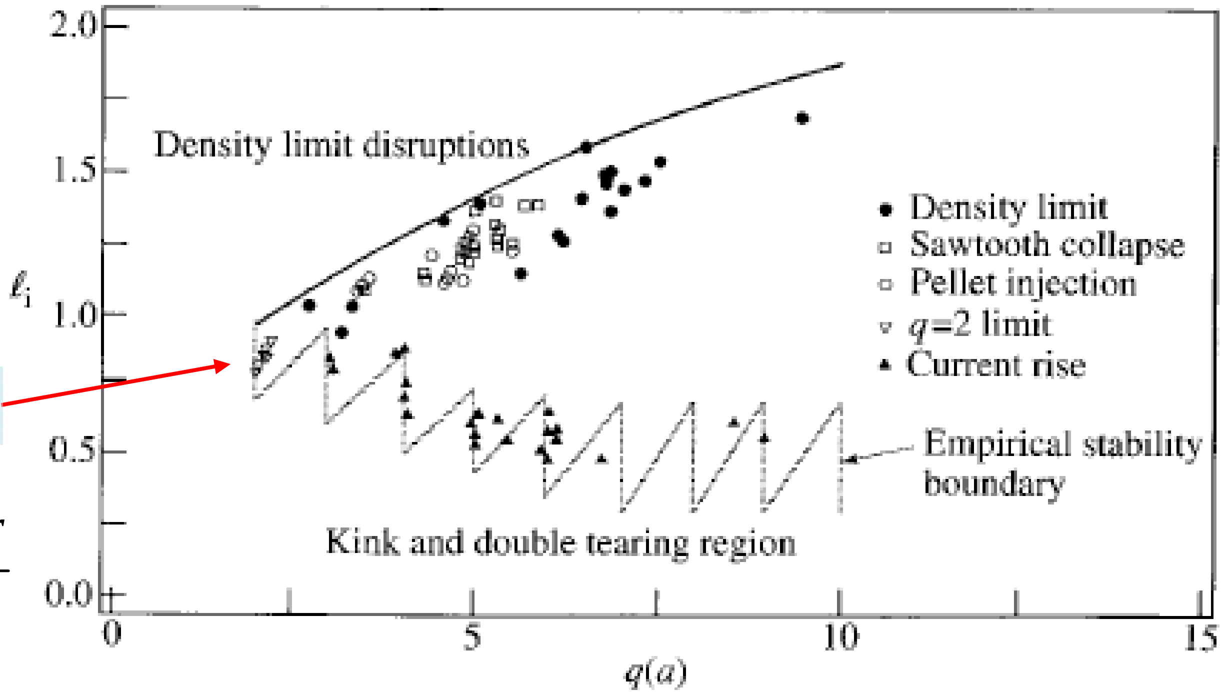


Typical chain of events during a plasma disruption

- **The ultimate stability limit in tokamak plasmas is set by major disruptions: large scale MHD instabilities**
  - loss of plasma energy in milliseconds (thermal quench – TC)
  - plasma current decays in 10s of milliseconds (current quench – QC)
- **Produces:**
  - very large heat loads on plasma facing surfaces
  - significant electromagnetic forces in vacuum vessel
  - large runaway electron beam

Mitigation techniques essential

MHD Stability - Plasma Equilibrium Limits



q_a=2 limit

$$l_i = \frac{2\int_0^a B_\theta^2 r dr}{a^2 B_{\theta a}^2}$$

JET

Limiter plasmas

Snipes, et al, Nucl Fus **28** (1988) 1085

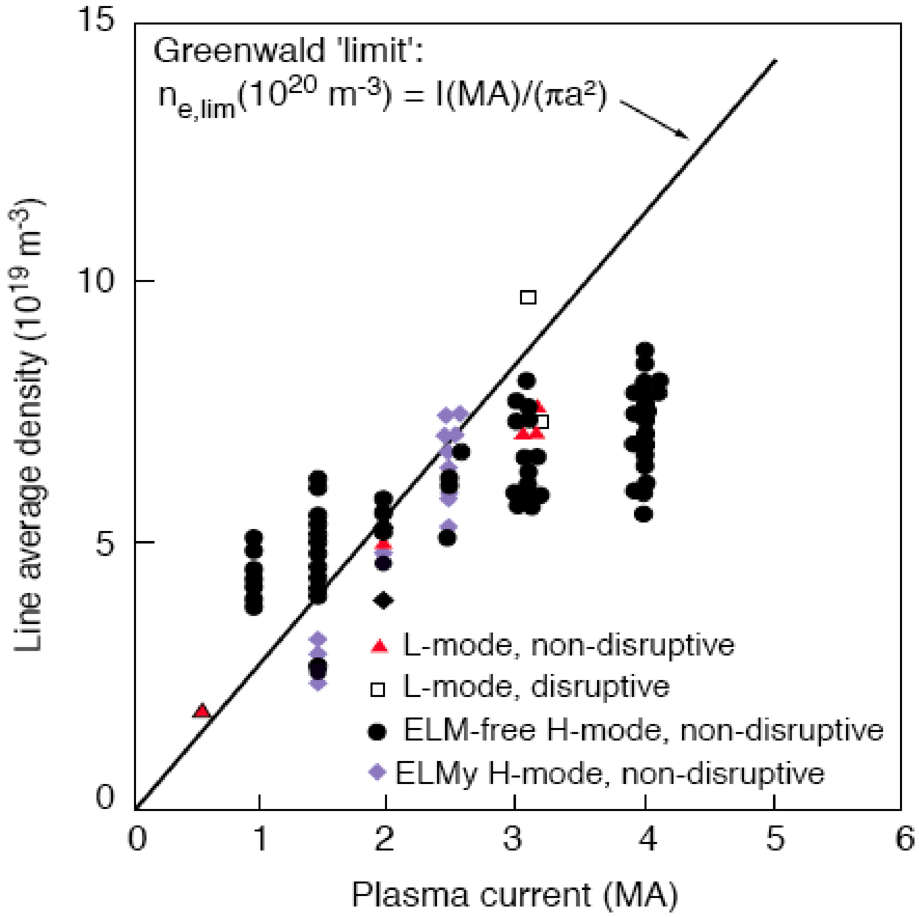
- **l_i-q_a diagram describes stable plasma operating space of internal inductance vs safety factor, limited by disruptions:**
  - low l_i typically has to be negotiated during the plasma current ramp-up
  - high-l_i limit typically occurs due to excessive radiation at plasma edge, resulting in cold edge plasma and narrow current channel (e.g., at density limit)

MHD Stability - 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 – Pressure Limit: β

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

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

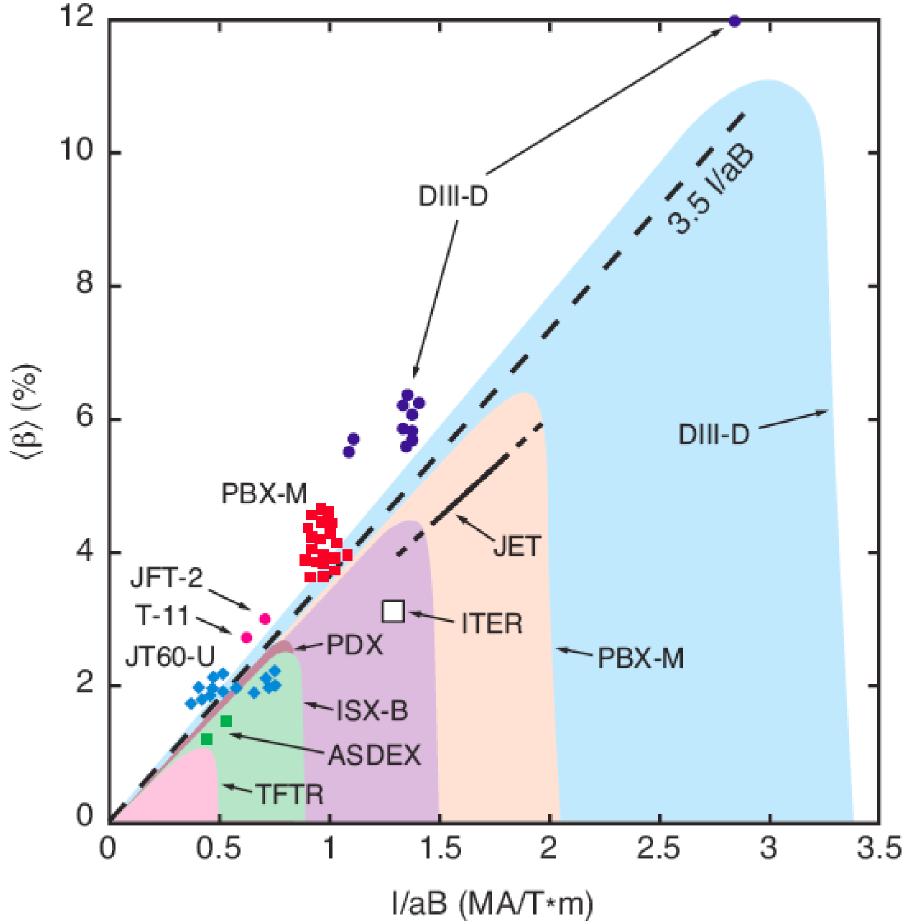
$$\beta_N = \frac{\beta(\%)}{I_p(MA) / aB}$$

- **Typically, "Troyon" limit describes tokamak plasmas:**

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

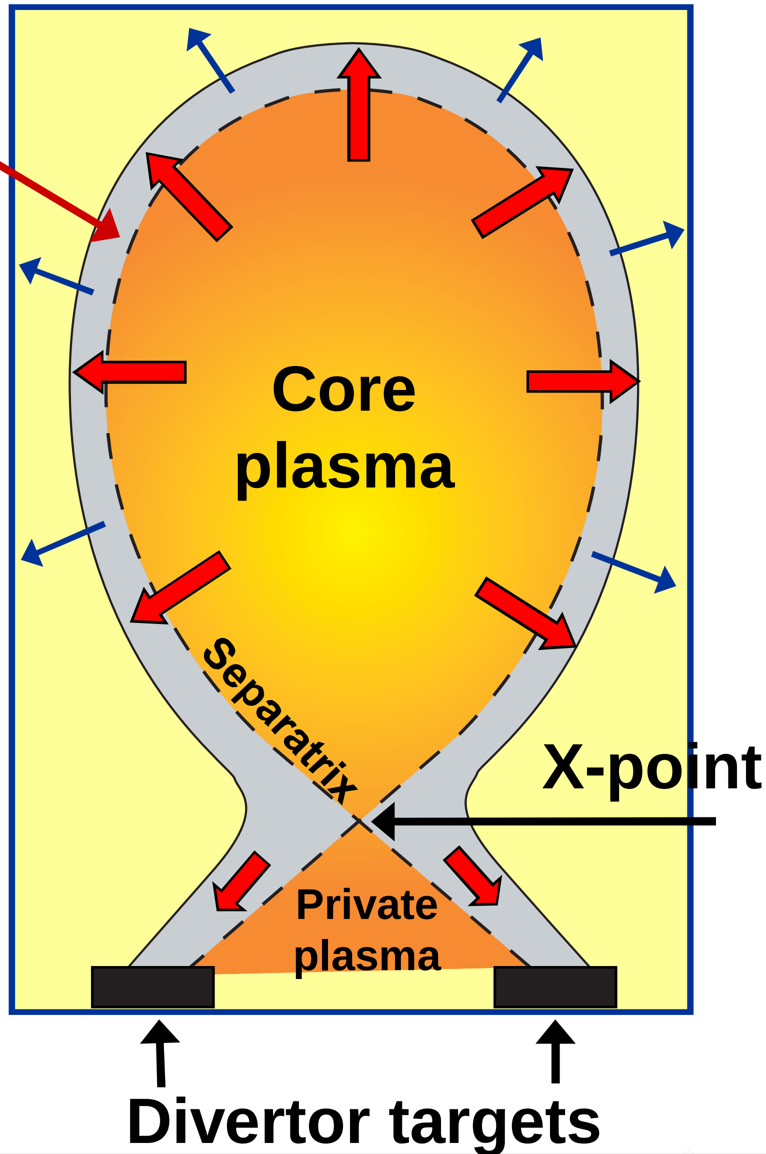
- **More generally, "no-wall" limit:**

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



Power and Particle Exhaust

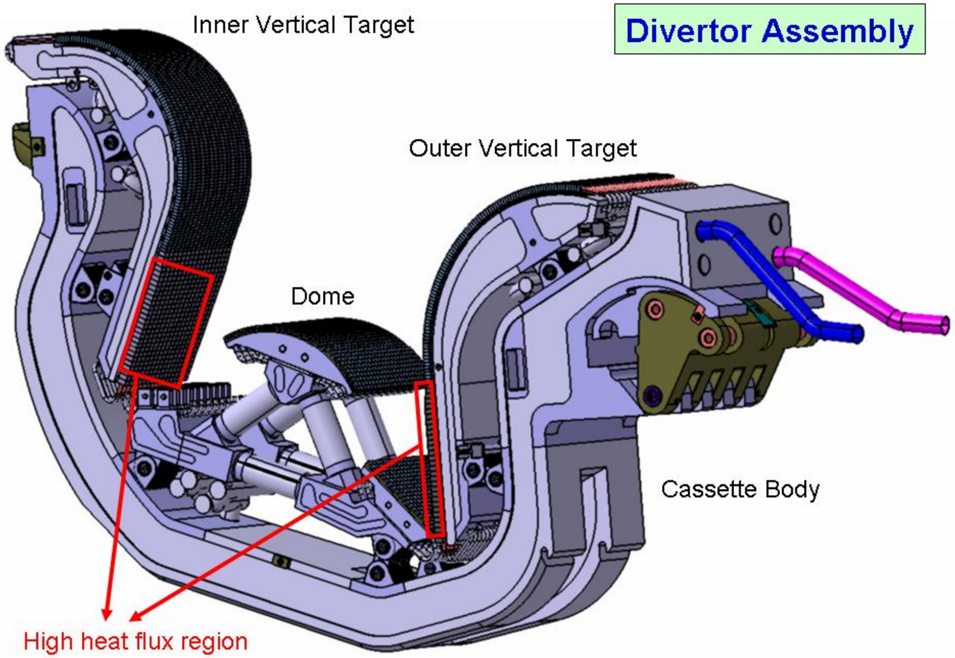
Scrape-off layer (SOL) plasma: region of open field lines



- **Essential problem is:**
  - handle power produced by plasma with (steady-state) engineering limit for plasma facing surfaces of 10 MWm^{-2}
  - extract helium from the core plasma to limit concentration below ~6%
  - prevent impurities from walls penetrating into plasma core
  - ensure plasma facing surfaces survive sufficiently long

Power and Particle Exhaust

- **The divertor is a significant element of the solution**
  - surfaces for high heat fluxes (10 MWm^{-2})
  - cryopumping to extract particles leaving the plasma, including helium



ITER divertor cassette – 54 cassettes make up the complete toroidal ring

- **The divertor is fundamental to exhaust power from a burning plasma:**
  - **impurities are added to the edge plasma to increase radiation**
  - **a large pressure gradient develops along the field lines into the divertor**
  - **the divertor plasma temperature falls to a few eV**
  - **a large fraction of the plasma exhaust power is redistributed by radiation and ion-neutral collisions**

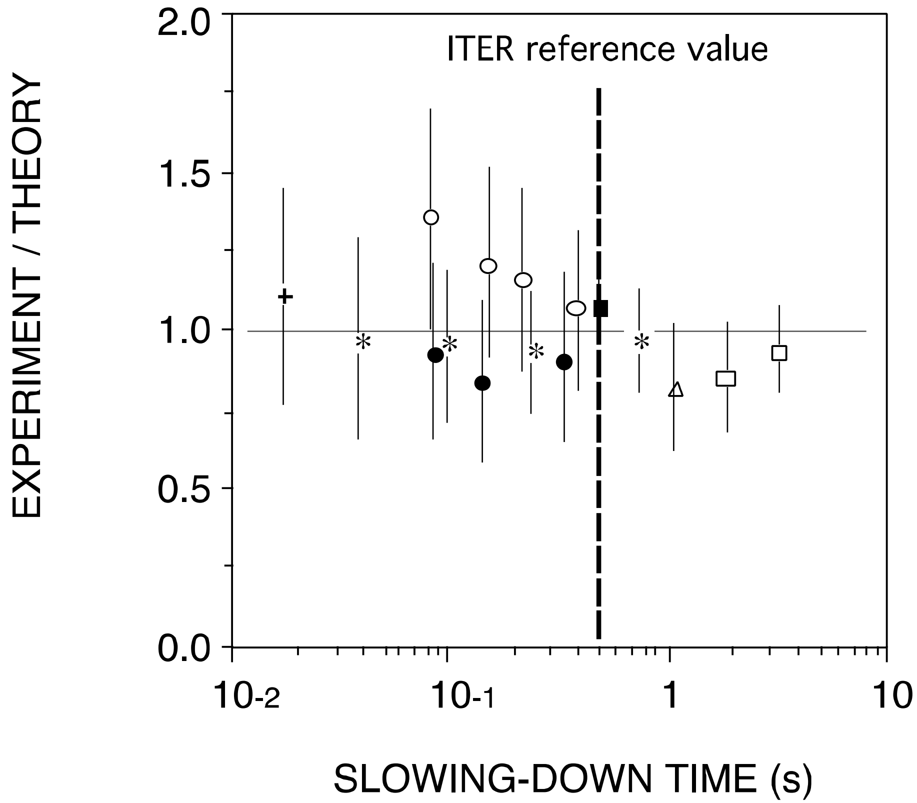
Burning Plasma Physics

- **Access to plasmas which are dominated by a-particle heating will open up new areas of fusion physics research, in particular:**
  - confinement of α's in plasma
  - response of plasma to α-heating
  - influence of α-particles on MHD stability

- **Experiments in existing tokamaks have already provided some positive evidence**
  - "energetic" particles (including α-particles) are well confined in the plasma
  - such particle populations interact with the background plasma and transfer their energy as predicted by theory
  - but energetic particles can induce MHD instabilities (Alfvén eigenmodes) - for ITER parameters at Q=10, the impact is expected to be tolerable

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)

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!

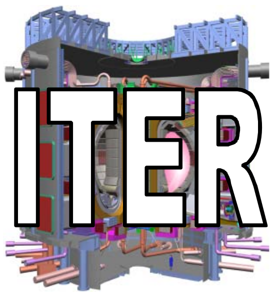
- **These modes can be driven unstable by the free energy arising from energetic particle populations with velocities above the Alfvén velocity, eg α-particles**

Physics for Fusion Power Plants

- A fusion power plant requires physics parameters that are simultaneously close to the limits of what might be achievable on the basis of our (experimental and theoretical) understanding

- Several key issues in (burning) plasma physics for a tokamak power plant must be developed in the current programme and demonstrated (and extended) in ITER:
  - Operating scenario - steady-state ?
  - High confinement at high density and high radiated power fraction
  - High fusion power $\Rightarrow$ high β operation $\Rightarrow$ robust MHD stability
  - Effective disruption avoidance and control
  - Power (and particle) exhaust with relevant PFCs
  - Tritium efficiency
  - α-particle confinement
  - Reactor-relevant auxiliary systems (H&CD, diagnostics, fuelling, control …)

ITER on the Path to Fusion Energy

	When?	Fusion Power	Burn Duration	Q
	1997	16 MW	~1 second	0.65
	2027-2028	500-700 MW	~7 minutes	10
	~2040	2-2.5 GW	days/steady-state	30

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http://www.iter.org - and associated links