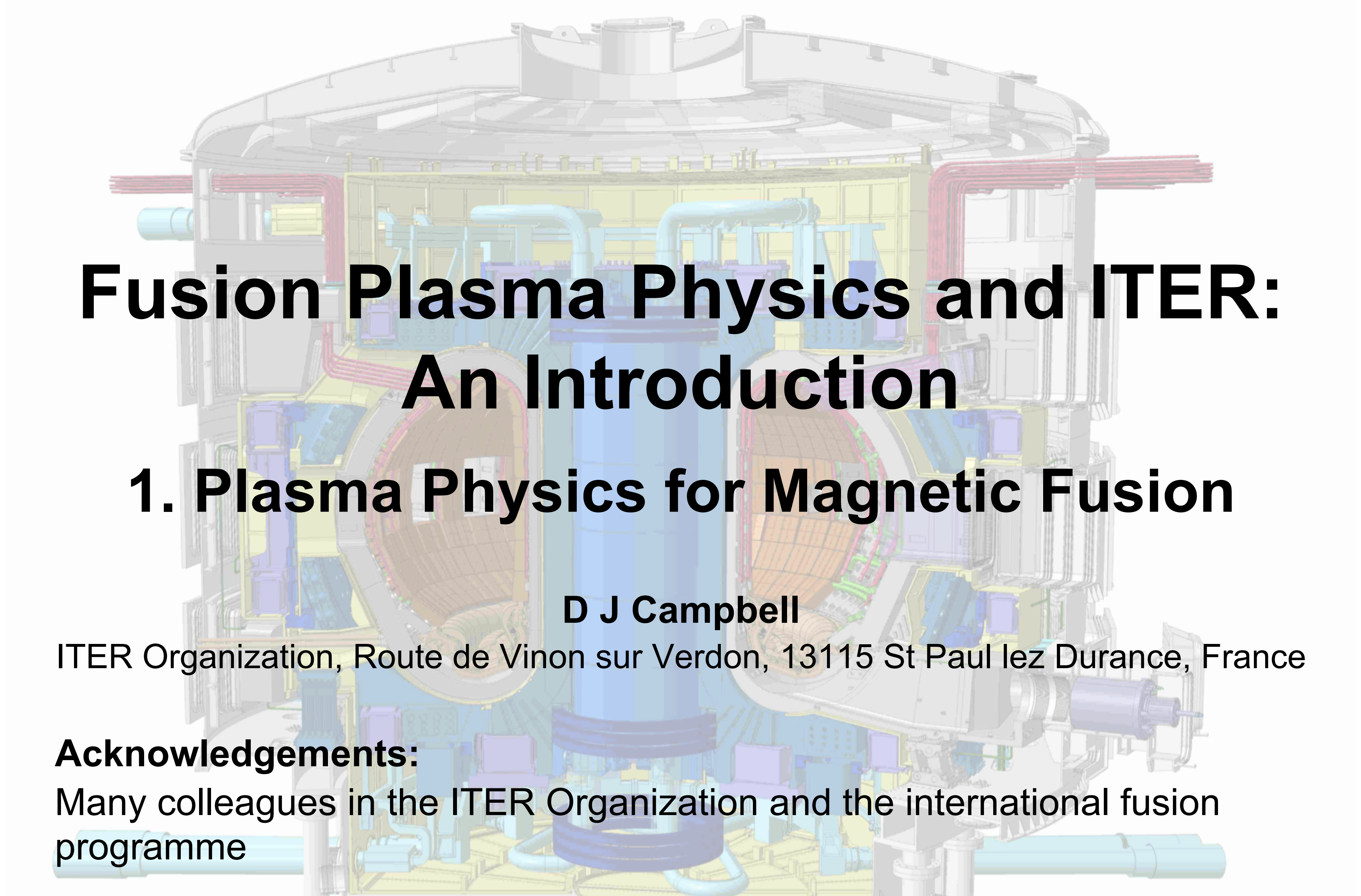

Fusion Plasma Physics and ITER: An Introduction

1. Plasma Physics for Magnetic Fusion

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Acknowledgements:
Many colleagues in the ITER Organization and the international fusion programme

Overview of Lectures

1. **Fusion Plasma Physics in Magnetic Fusion**
   DJ Campbell

2. **Physics of Tokamak Plasmas** **DJ Campbell**

3. **Fusion Technology for ITER and the ITER Project**
   G Janeschitz

4. **Further development towards a Fusion Power Plant**
   G Janeschitz

Lecture 1 - Synopsis

- **Introduction to thermonuclear fusion**
  - some aspects of inertial confinement fusion

- **Basics of magnetic confinement fusion – the tokamak**

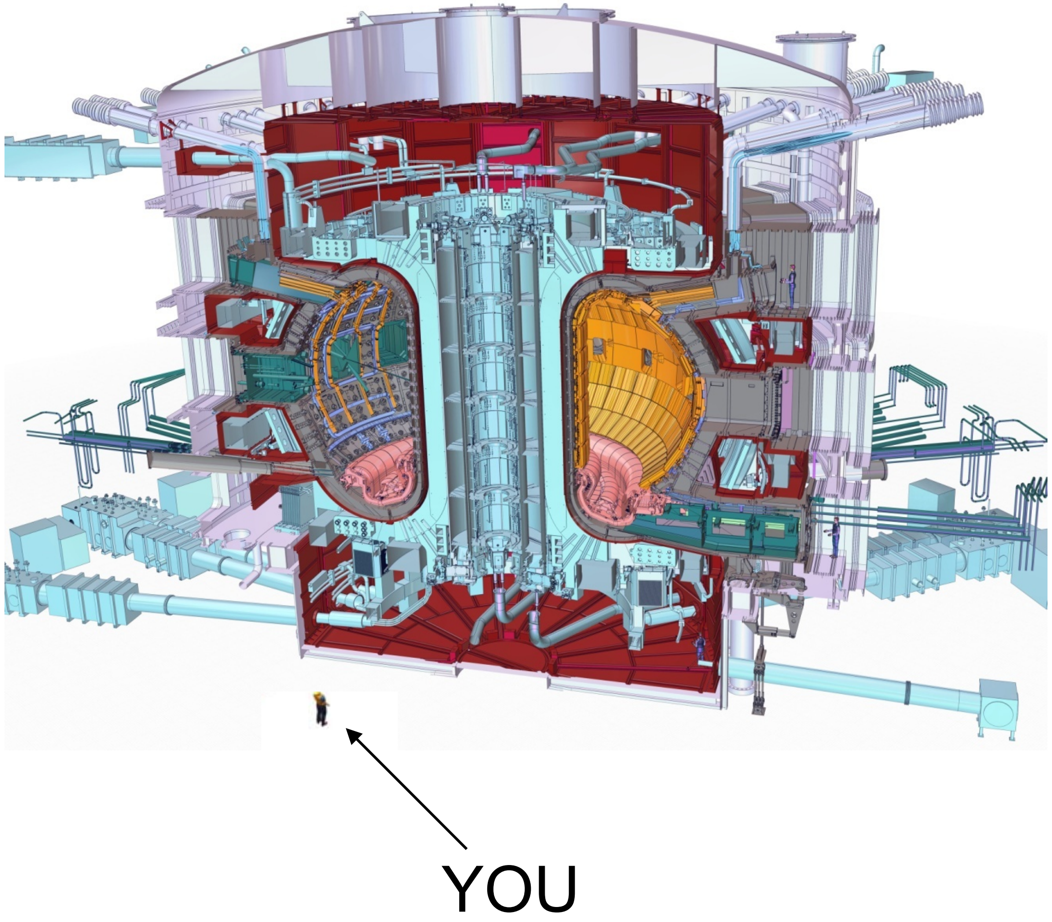
- **(Some) plasma physics for magnetic confinement fusion in tokamaks**

- **The ITER Project**

An Introduction - ITER

- 7 partners representing >50% of the world's population have embarked on the ITER project
- ITER is designed to produce 500MW of fusion power (tenfold power amplification) for extended periods of time (several 100s)
- 10 years construction
  20 years operation
  5 years de-activation
- These lectures will explain the background to ITER and how it fits into the development of fusion energy

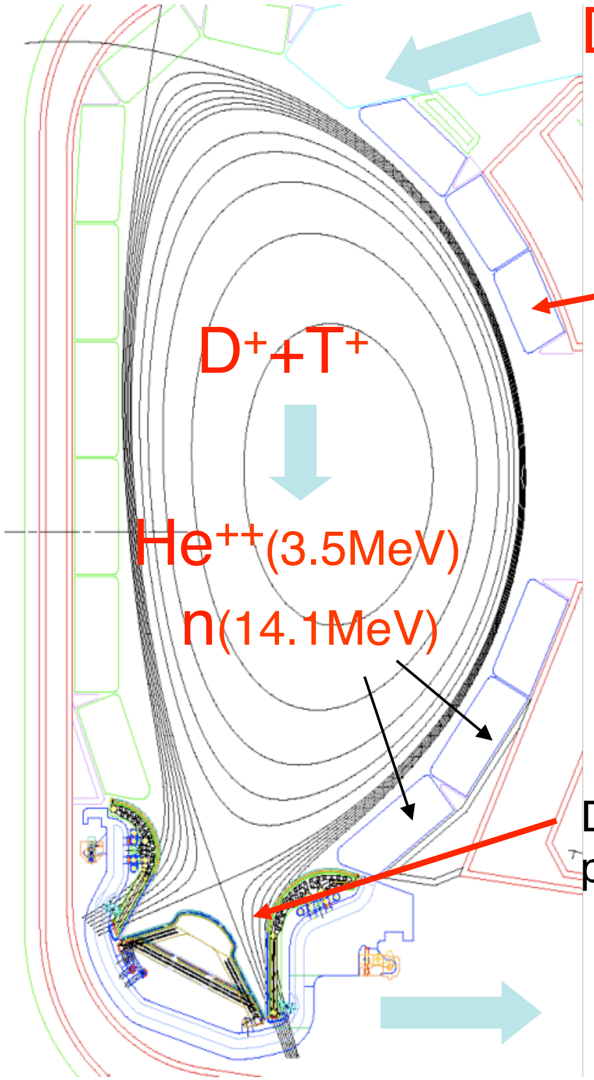
ITER tokamak



ITER: Fusion Power Production

ITER Plasma:

R/a:	6.2 m /2 m
Volume:	830 m^3
Plasma Current:	15 MA
Toroidal field:	5.3 T
Density:	10^{20} m^{-3}
Peak Temperature:	2×10^8 K
Fusion Power:	500 MW
Plasma Burn	300 - 500 s
(“Steady-state”	~3000 s)



D_2,T_2 Fuel

$D^+ + T^+$

He^{++}(3.5MeV)

n(14.1MeV)

Blanket:
neutron absorber

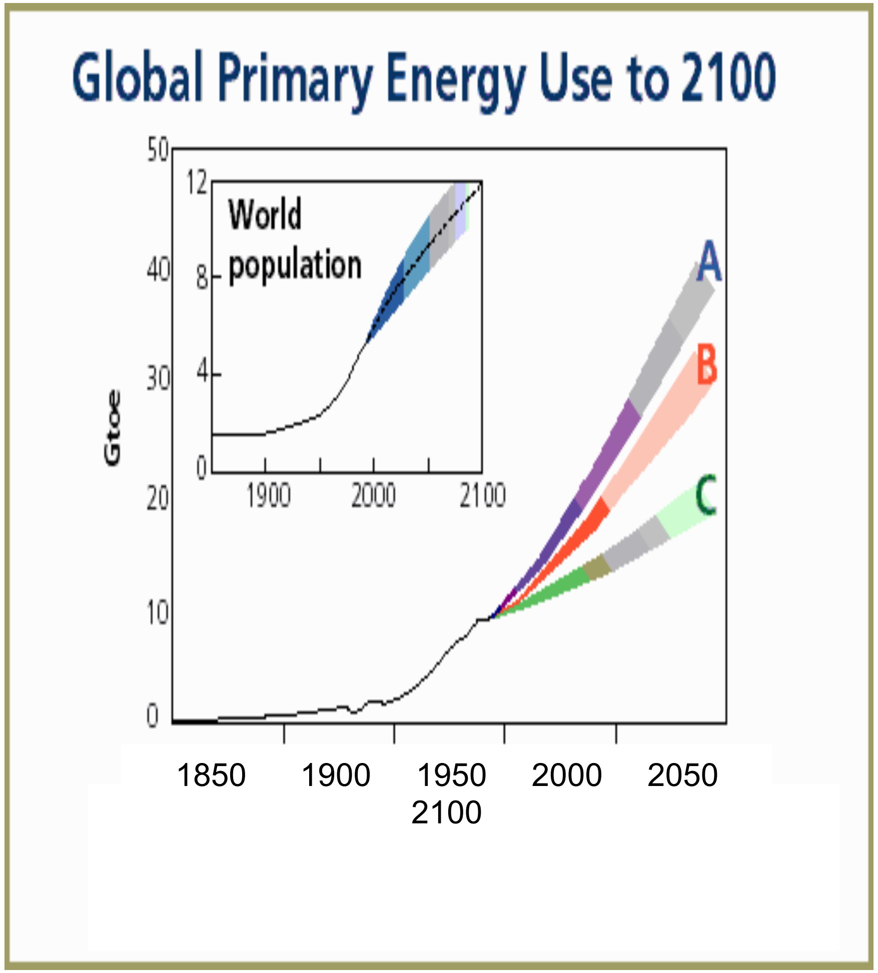
Power Plant
Li-->T
High temperature

Divertor:
particle and heat exhaust

He, D_2,T_2,
impurities

Introduction to Thermonuclear Fusion

Why Fusion ?



- **Fuel: abundant, world-wide distributed:**
  - sufficient deuterium in seawater for millions of years
  - tritium is produced from lithium - sufficient ore supplies for thousands of years (millions of years including seawater resources
- **Safety: no risk of major accidents:**
  - reactor contains fuel for only a few minutes burn
- **Waste: no long-term burden:**
  - low radio-toxicity after < 100 years
  - no CO_2

Fusion – the fundamental principle

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

  $E = \Delta mc^2$

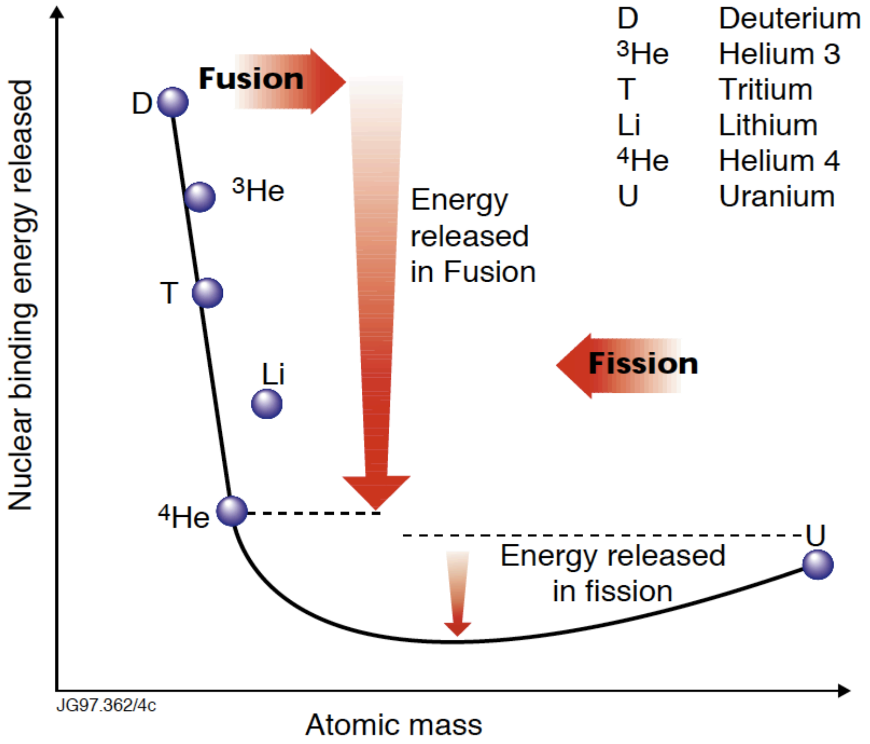
  – mass loss for DT reactions corresponds to ~ 0.4%

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

  – energy gain/ reaction:

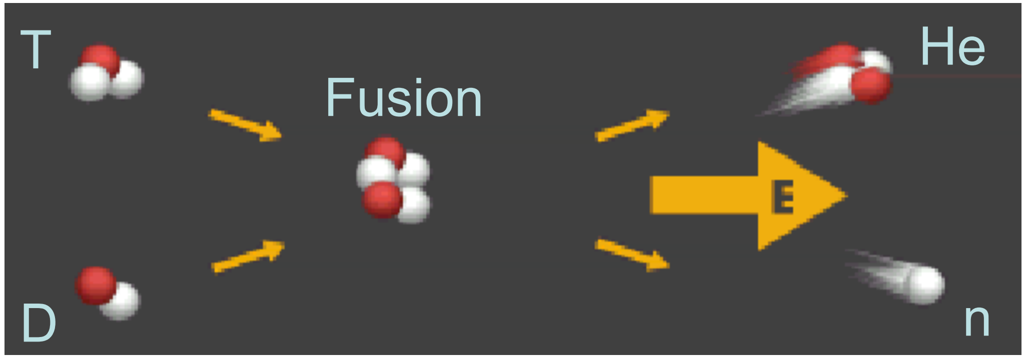
  DT fusion: 17.6 MeV

  U fission: ~200 MeV



D	Deuterium
^{3}He	Helium 3
T	Tritium
Li	Lithium
^{4}He	Helium 4
U	Uranium

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**

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

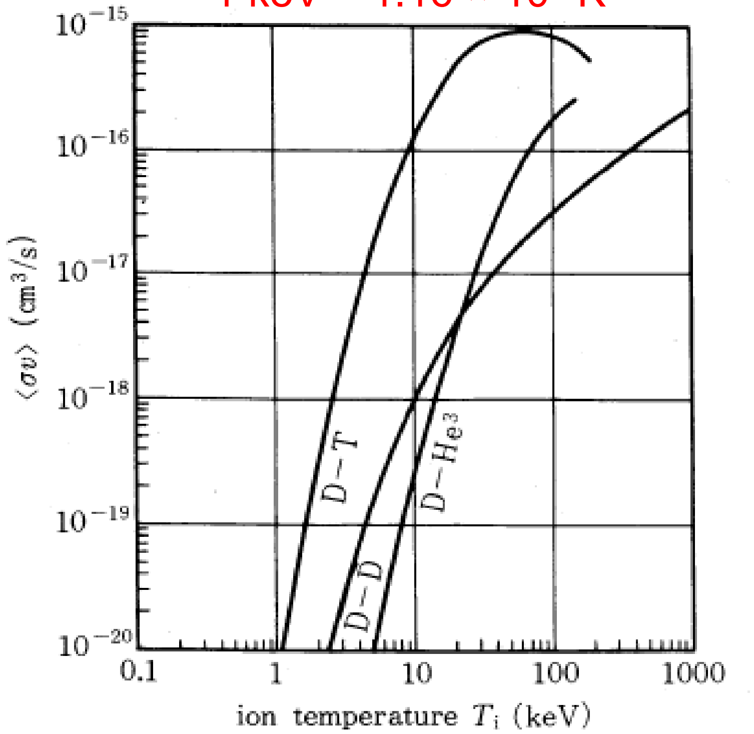
- **Two other important reactions for DT fusion:**

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

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

  – these reactions will allow a fusion reactor to breed tritium

1 keV = 1.16 × 10^7 K



Fusion Power Production

- **High temperatures (~10 keV) are required for significant thermonuclear fusion energy production (⇒ dealing with plasmas!):**
  - nuclei must overcome Coulomb barrier to approach close enough to fuse

    ⇒ T ~ 300 keV required to overcome Coulomb barrier of ~0.4 MeV
  - but, reaction rate is dominated by a small population of high energy nuclei which react by quantum tunnelling

    ⇒ for 100 keV nuclei, tunnelling probability, $w = 3.2 \times 10^{-2}$

- **Fusion power density for an optimal 50:50 D-T mixture:**

$$P_F = \frac{n^2}{4}\langle \sigma v \rangle E_F = \frac{p^2}{16}\frac{\langle \sigma v \rangle}{(kT)^2} E_F \quad (E_F = 17.6 MeV)$$

  - e.g. for magnetic fusion reactor parameters, with p ~ 10 atm, $P_F \sim 7.5\ \mathrm{MWm^{-3}}$

Power Gain – the Lawson Criterion

- **Obtaining the temperature required to produce fusion reactions involves heating the plasma**
  - for a net power gain, fusion power out must exceed input heating power (including correction for loss processes, e.g. bremsstrahlung, synchrotron radiation ….)

- **One can define a parameter, τ_E, the energy confinement time, which characterizes the rate of power loss:**

$$\tau_E = \frac{W_{th}}{P_{loss}} = 3\frac{\int nkT\, dV}{P_{loss}}$$

- **Then, the overall power balance can be written (P_{heat} = ext power):**

$$P_{heat} = \left(\frac{3\bar{n}k\bar{T}}{\tau_E} - \frac{n^2}{4}\langle\sigma v\rangle E_\alpha\right).V$$

- **Ignition (P_{heat} = 0) implies:** $\bar{n}\tau_E > \frac{12}{\langle\sigma v\rangle}\frac{k\bar{T}}{E_\alpha}$ $\Rightarrow$ $\boxed{Q_{DT} = \frac{P_F}{P_{heat}} \rightarrow \infty}$

Power Gain – the Lawson Criterion

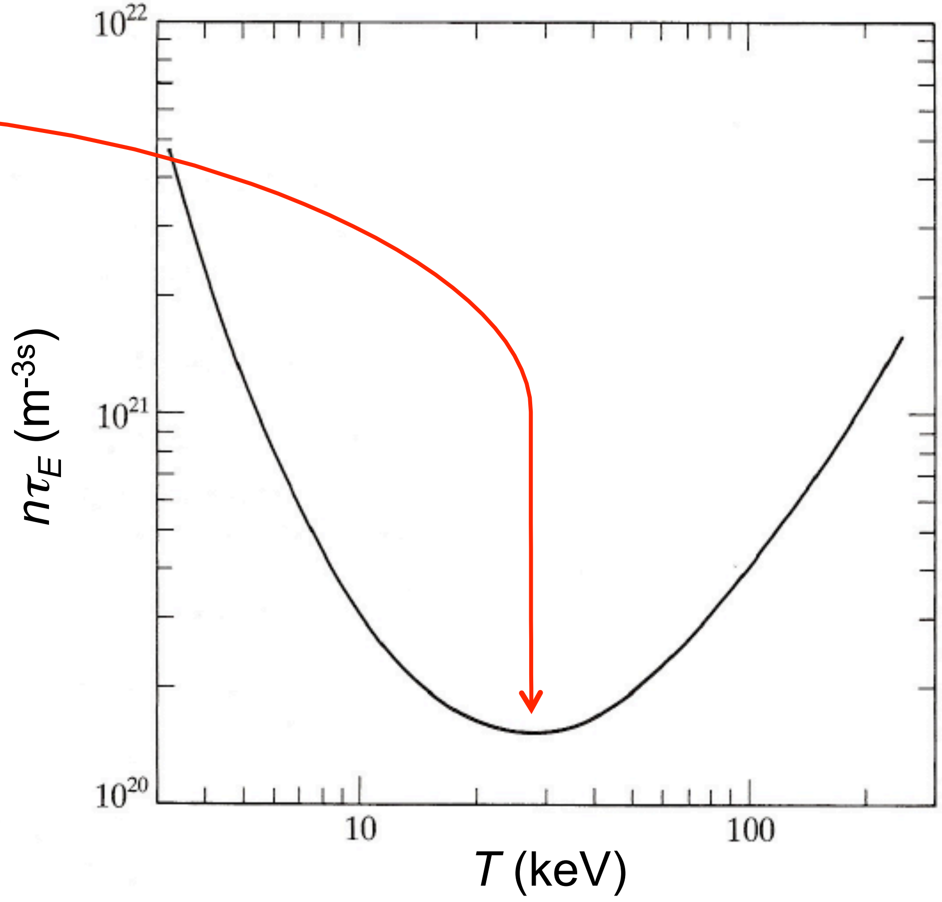
- **Evaluating the condition for ignition numerically, one finds:**

$$\bar{n}\tau_E > 1.5\times10^{20}\,\mathrm{m^{-3}s}$$

  - The minimum expressed by this “ignition criterion” occurs in the region 20-30 keV

- **This inequality is essentially that first derived by Lawson in the 1950s: the Lawson Criterion**

- **One can take this analysis a step further, defining the ignition margin, C_{DT}, in terms of the fusion triple product, $n_i(0)T_i(0)\tau_E$:**


JA Wesson, *Tokamaks*, 3rd edition, OUP (2004)

$$C_{DT} = \frac{Q_{DT}}{Q_{DT}+5} = \frac{P_\alpha}{P_{loss}} = 2\times10^{-22}\,n_{DT}(0)T_{DT}(0)\tau_E \;\; (T_{DT}(0)\;\text{ in keV})$$

Plasma Fusion Performance

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

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

Energy confinement time (τ_E): few seconds ($\propto$ current $\times$ radius^2)
(plasma pulse duration ~1000s)

Fusion power amplification:

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

⇒ Present devices: $Q \leq 1$

⇒ ITER: $Q \geq 10$

⇒ "Controlled ignition": $Q \geq 30$

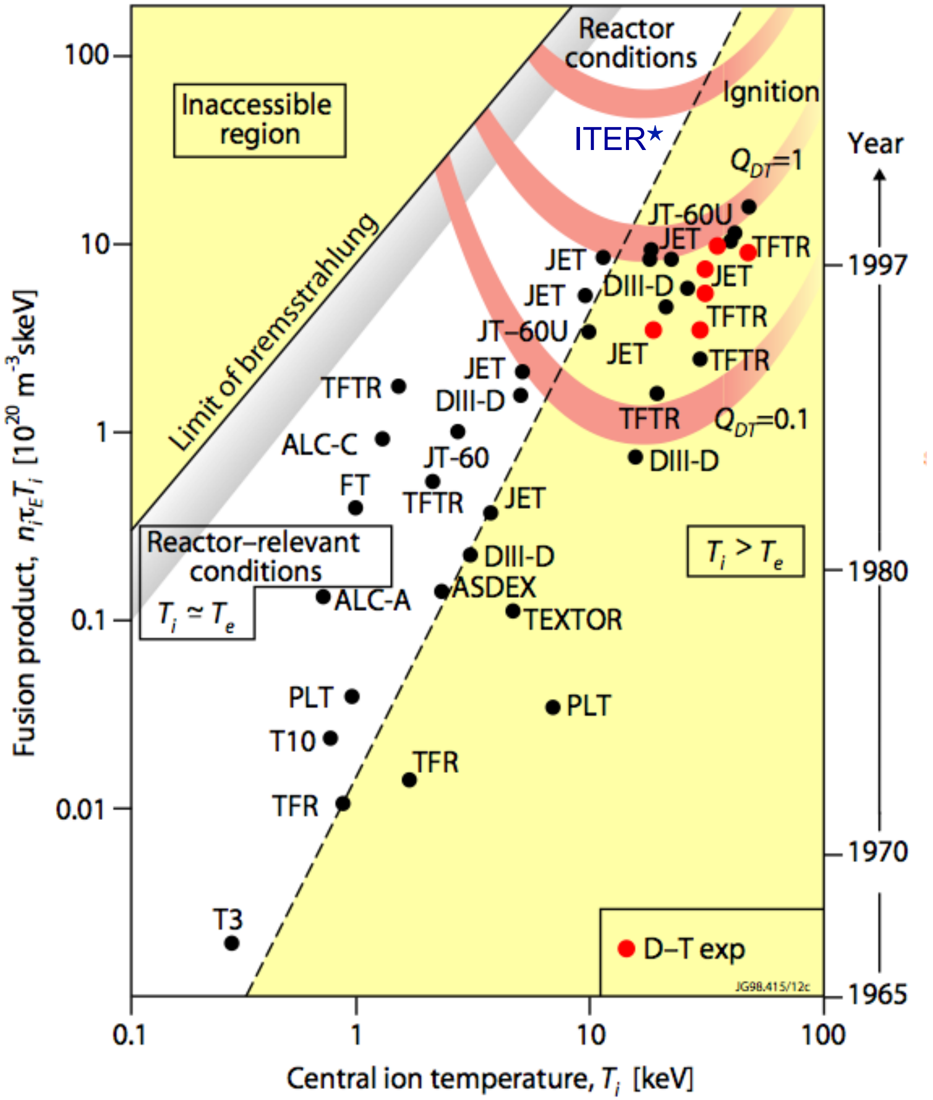
Plasma Fusion Performance – Tokamaks

Fusion Triple Product

- **Existing experiments have achieved nTτ values ~ 1×10^{21} m^{-3}skeV ~ $Q_{DT} = 1$**

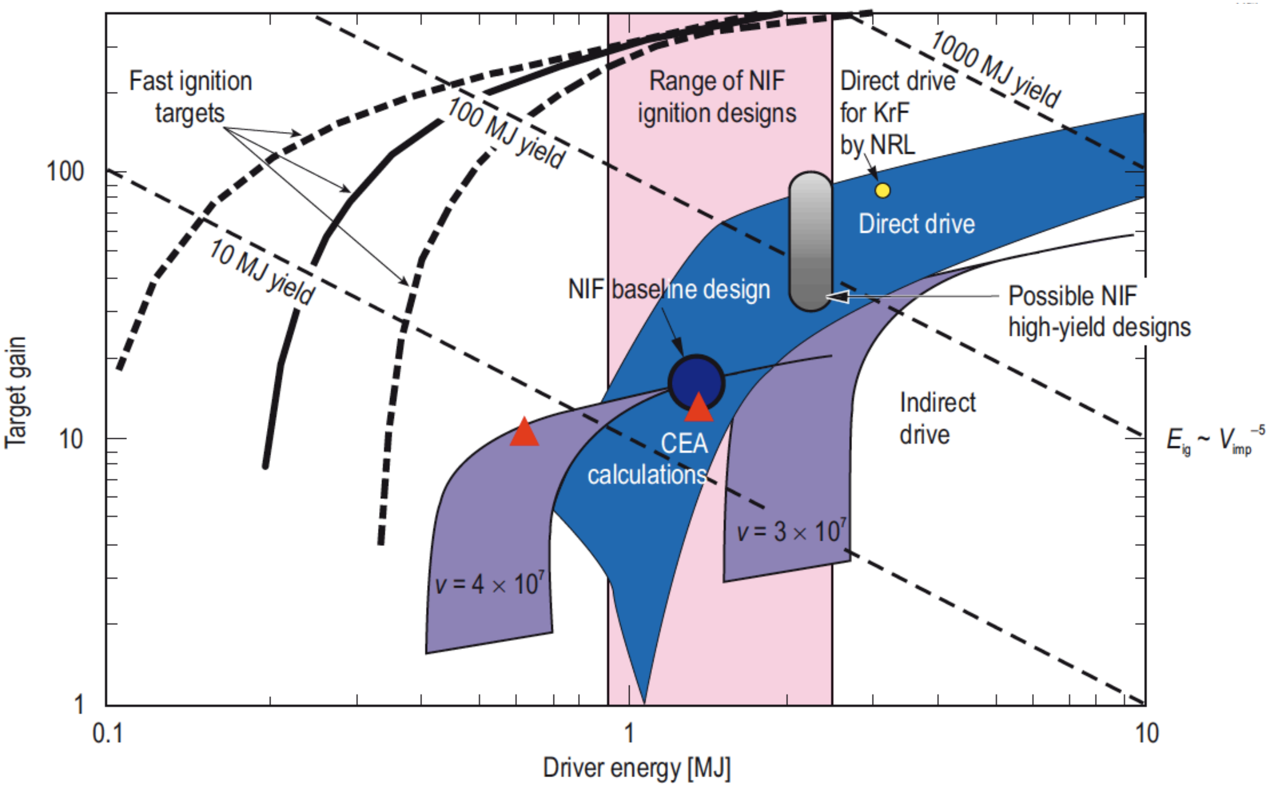
- **JET and TFTR have produced DT fusion powers of >10MW for ~1s**

- **ITER is designed to a scale which should yield $Q_{DT} \geq 10$ at a fusion power of 400 - 500MW for 300-500s**



Inertial Confinement Fusion

- **The US National Ignition Facility at Livermore is expected to achieve ignition within the next 2 years:**
  - uses high power lasers to compress a small DT capsule embedded in a “hohlraum” (indirect drive)



WJ Hogan, Ch. 8 in *Landolt-Börnstein-Handbook on Energy Technologies,* Springer Verlag (2005)

Inertial Confinement Fusion

- **Concept of ignition in MCF and ICF is somewhat different:**
  - in MCF, ignition criterion is based on power balance
  - in ICF, ignition criterion is based on burn propagation in capsule and fuel burn-up criterion

- **MCF criterion:**

  $nT\tau_E \sim 5\times10^{21}$ $(m^{-3}keV.s)$

  (fuel burn-up fraction ~ few %)

- **ICF criterion:**

  $\rho R \sim 2$ $(kg.m^{-2})$

  (fuel burn-up fraction ~ 30%)

	ITER	NIF
$n_i(m^{-3})$	1×10^{20}	1.1×10^{31}
ρ (kgm^{-3})	4.2×10^{-7}	5.7×10^{4}
<T> (keV)	~ 10	~ 10
<p> (atm)	3.3	4.5×10^{11}
τ_E (s)	~ 3.5	~ 10^{-10}
a (m)	2.0	3.5×10^{-5}
V (m^3)	830	1.8×10^{-13}
E_{plas} (J)	3.5×10^{8}	9.3×10^{3}
Output	500 MW	10-20 MJ

Basics of Magnetic Confinement Fusion:

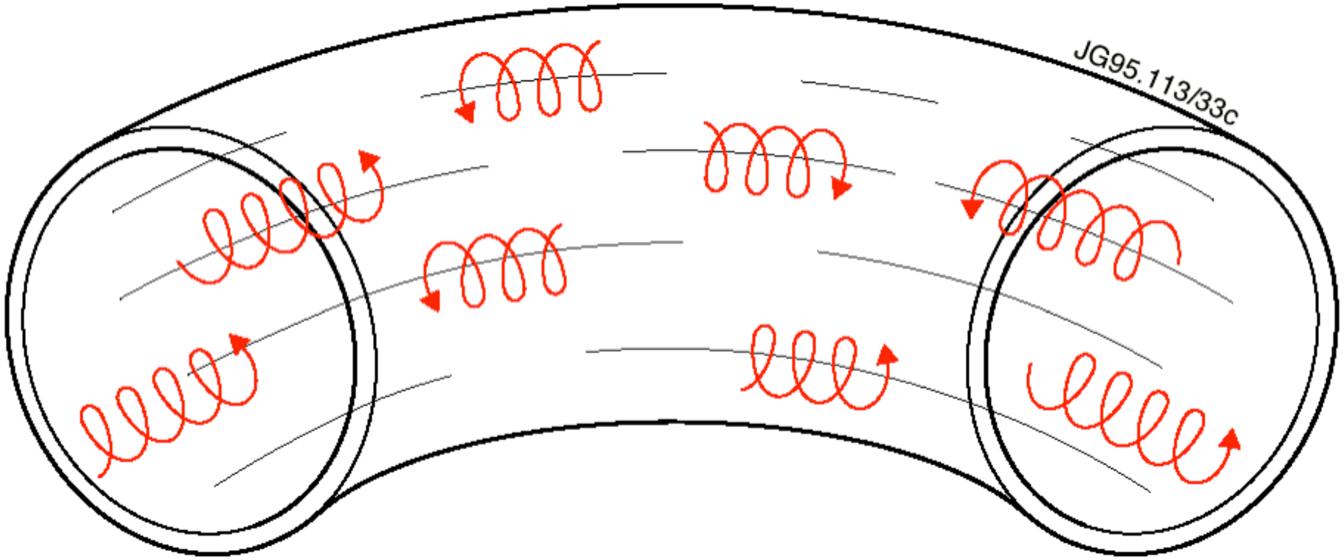
The Tokamak

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 tokamak:

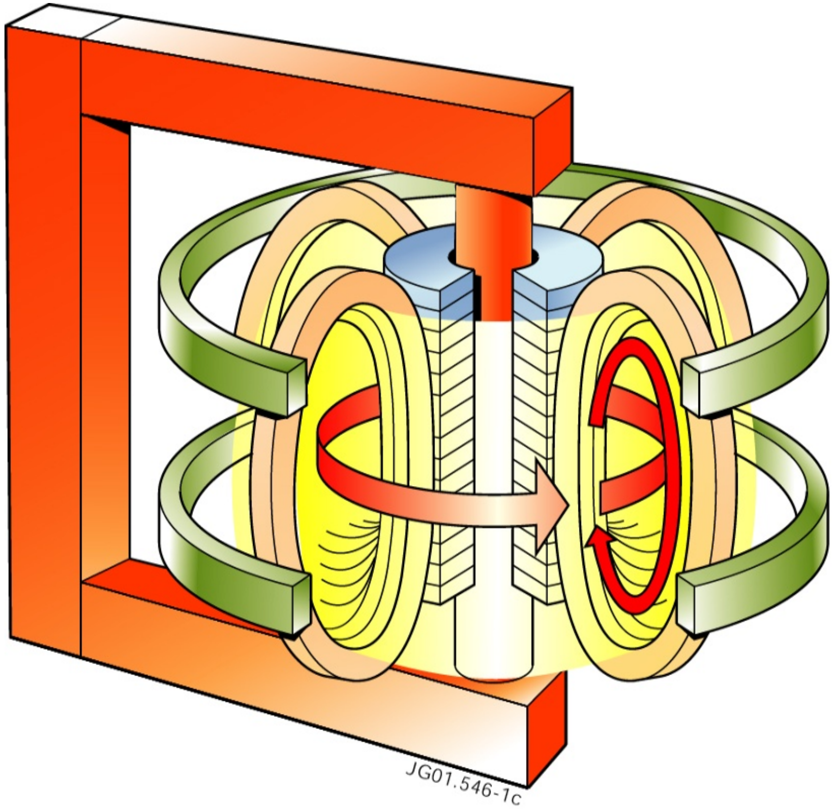
'toroidal chamber' and 'magnetic coil' (Russian)

Magnetic Confinement in a Tokamak

The Tokamak:

- **External coils**
  - to produce a toroidal magnetic field

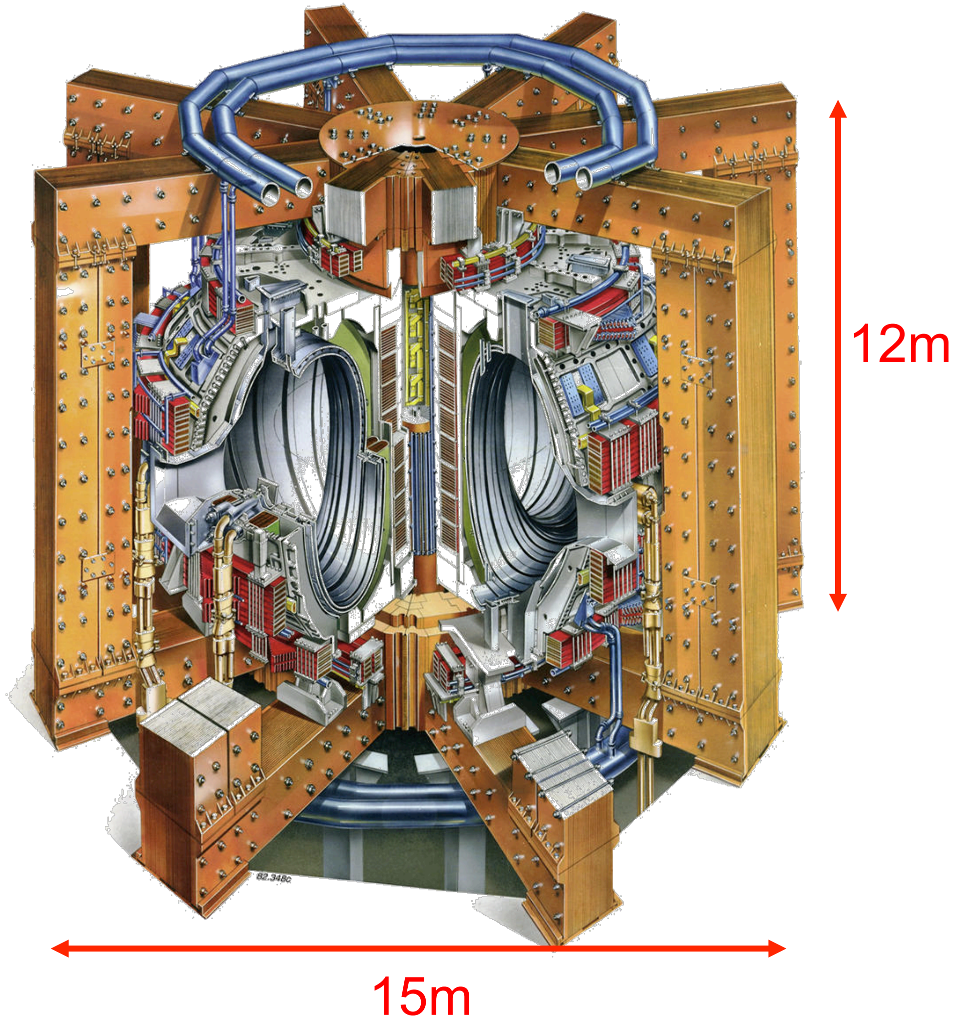
- **Transformer with primary winding**
  - to produce a toroidal current in the plasma
  - this plasma current creates a poloidal magnetic field



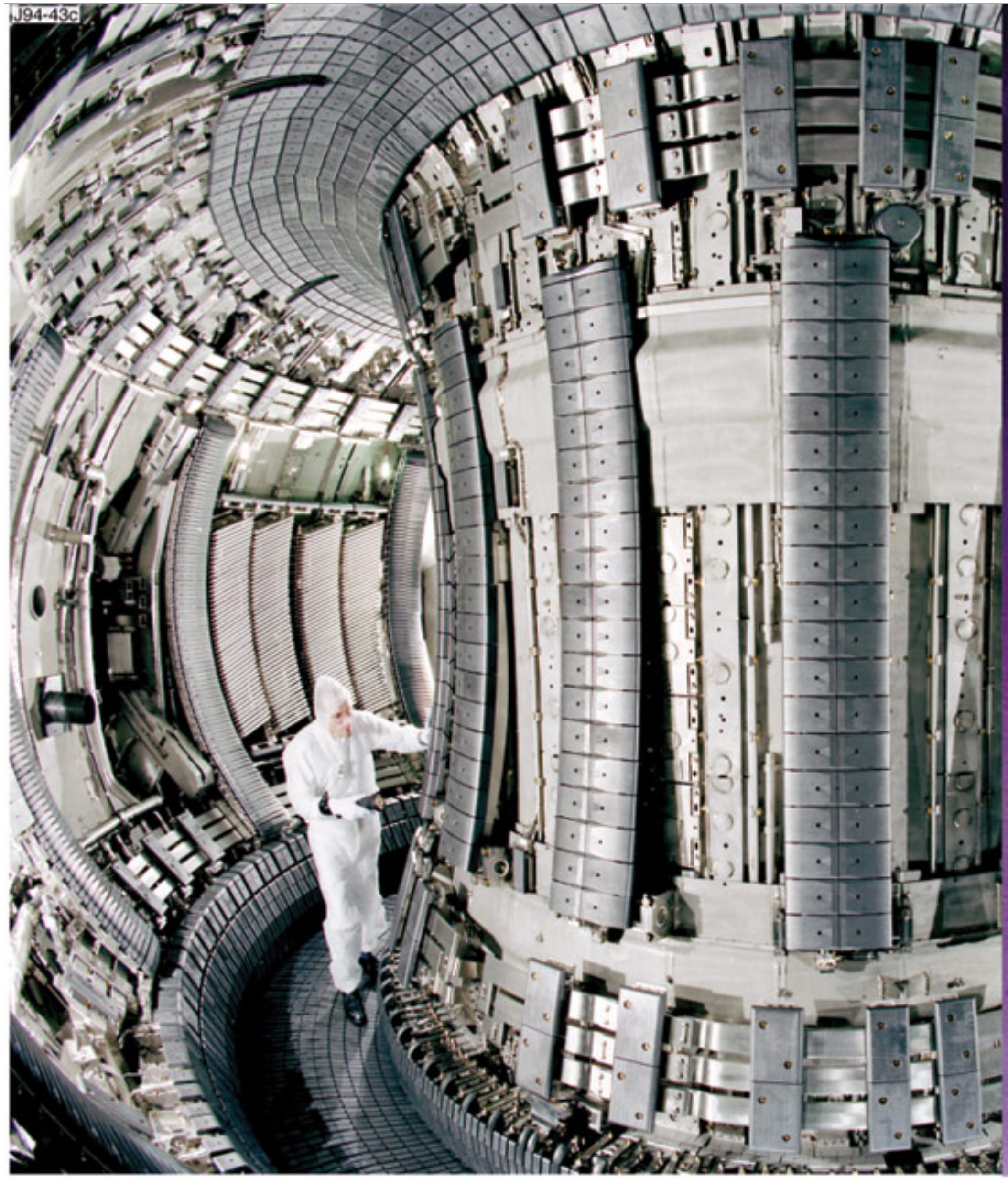
- **Finally, poloidal coils**
  - to control the position and shape of the plasma

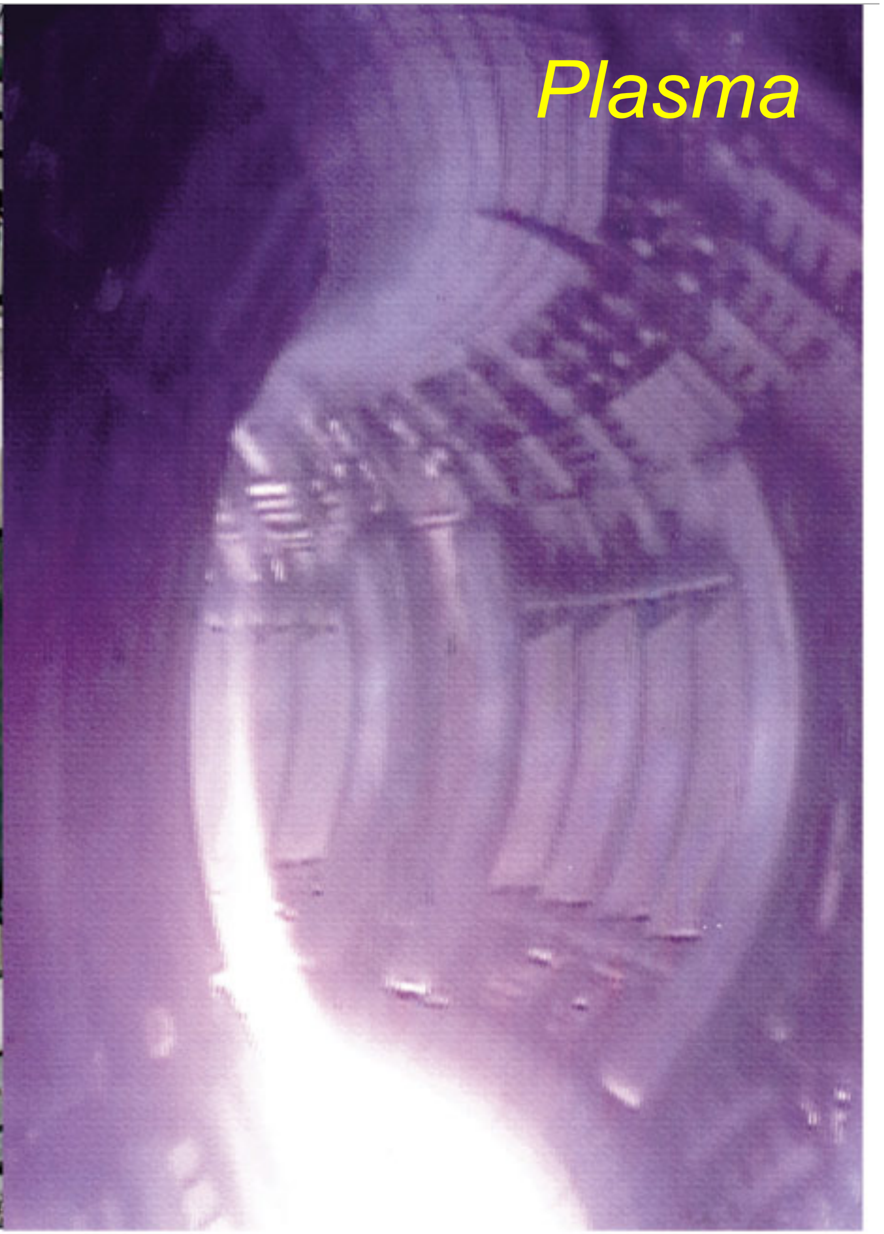
JET: Joint European Torus

- **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 (Q_{DT} ~ 0.6)**

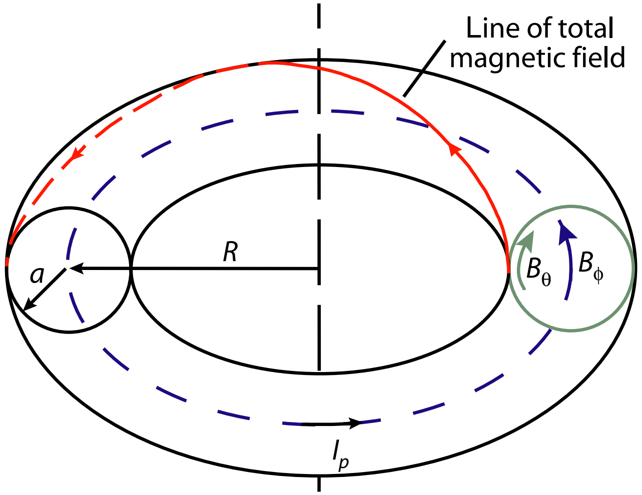


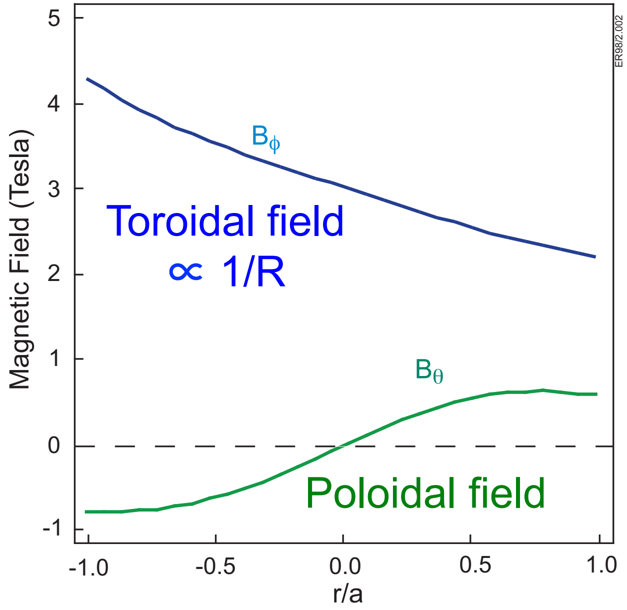
JET - Internal





Magnetic Confinement in a Tokamak



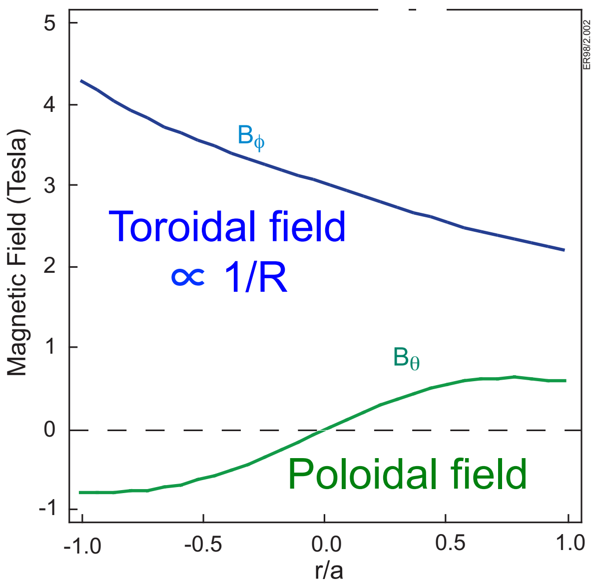


- **In configurations with only a toroidal field, ions and electrons drift vertically in opposite directions:**
  - caused by field gradient and curvature
  - resultant electric field destroys plasma
- **An additional poloidal field allows particles to follow helical paths, cancelling the drifts**
- **"Winding number" of helix is an important stability parameter for the system:**

$$q_c = \frac{aB_\phi}{RB_\theta}$$

  - q = safety factor
  - R/a = aspect ratio

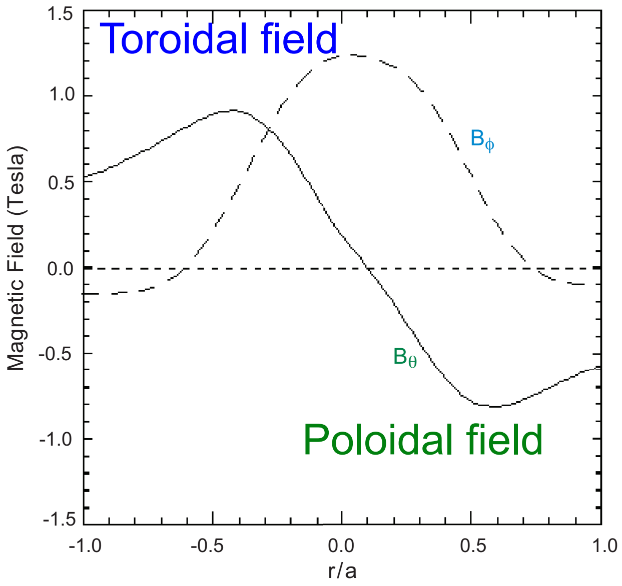
Toroidal Magnetic Confinement Systems



Tokamak

Strong B_ϕ

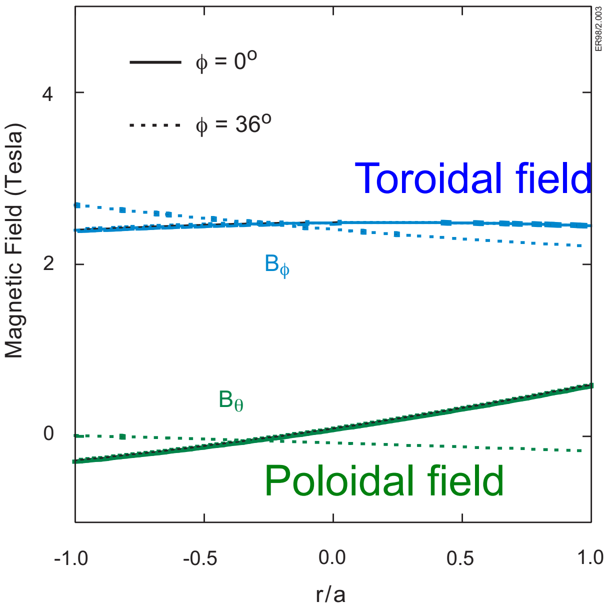
Strong I_p



Reversed Field Pinch

Weak B_ϕ

Strong I_p



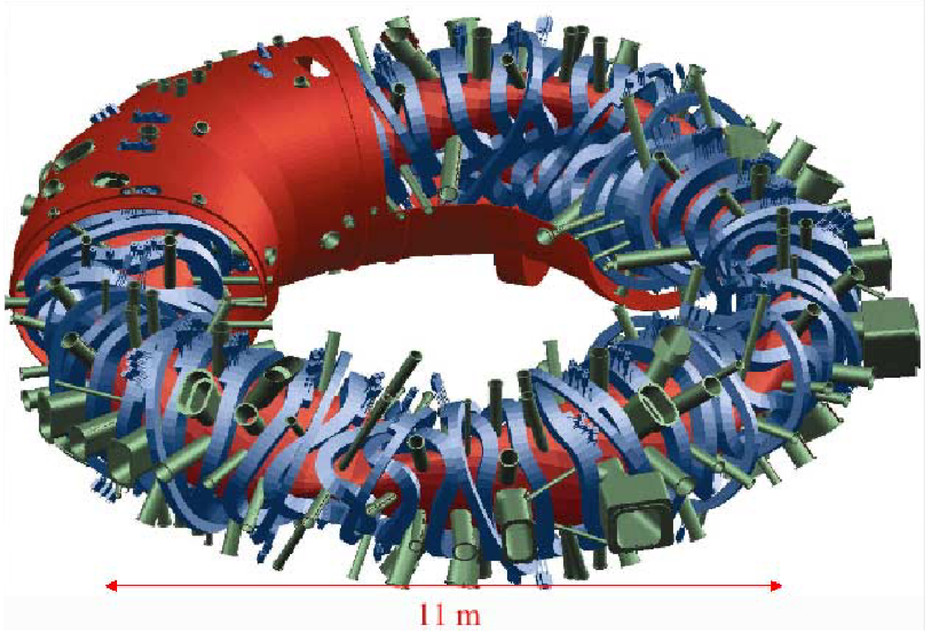
Stellarator

Strong B_ϕ

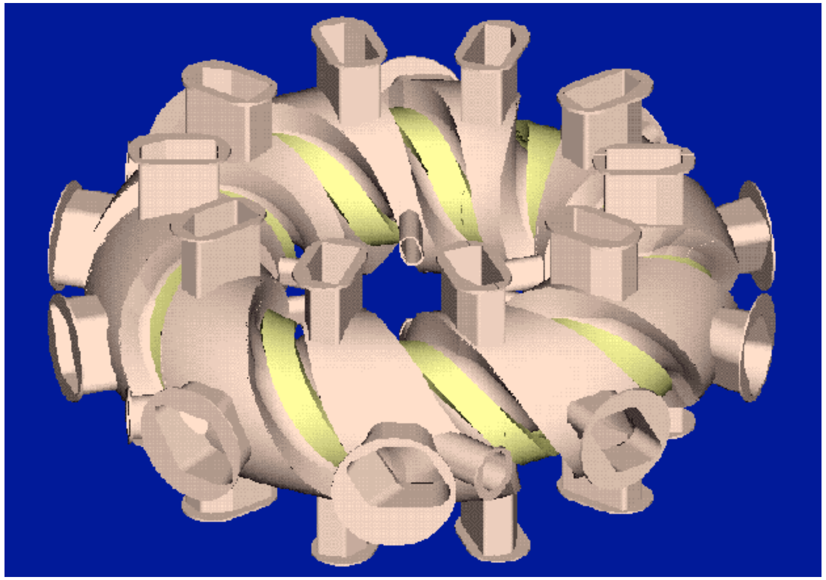
Zero I_p

- **Numerous toroidal confinement configurations are being studied:**
  - Tokamak has progressed most rapidly and is ready for the “thermonuclear” step
- **Note that in the stellarator, the helical magnetic surfaces are produced entirely by external coils:**
  - Radial profiles of B_ϕ, B_θ vary periodically in toroidal direction

A Digression: Stellarators



Wendelstein 7-X (Germany)
Modular Stellarator

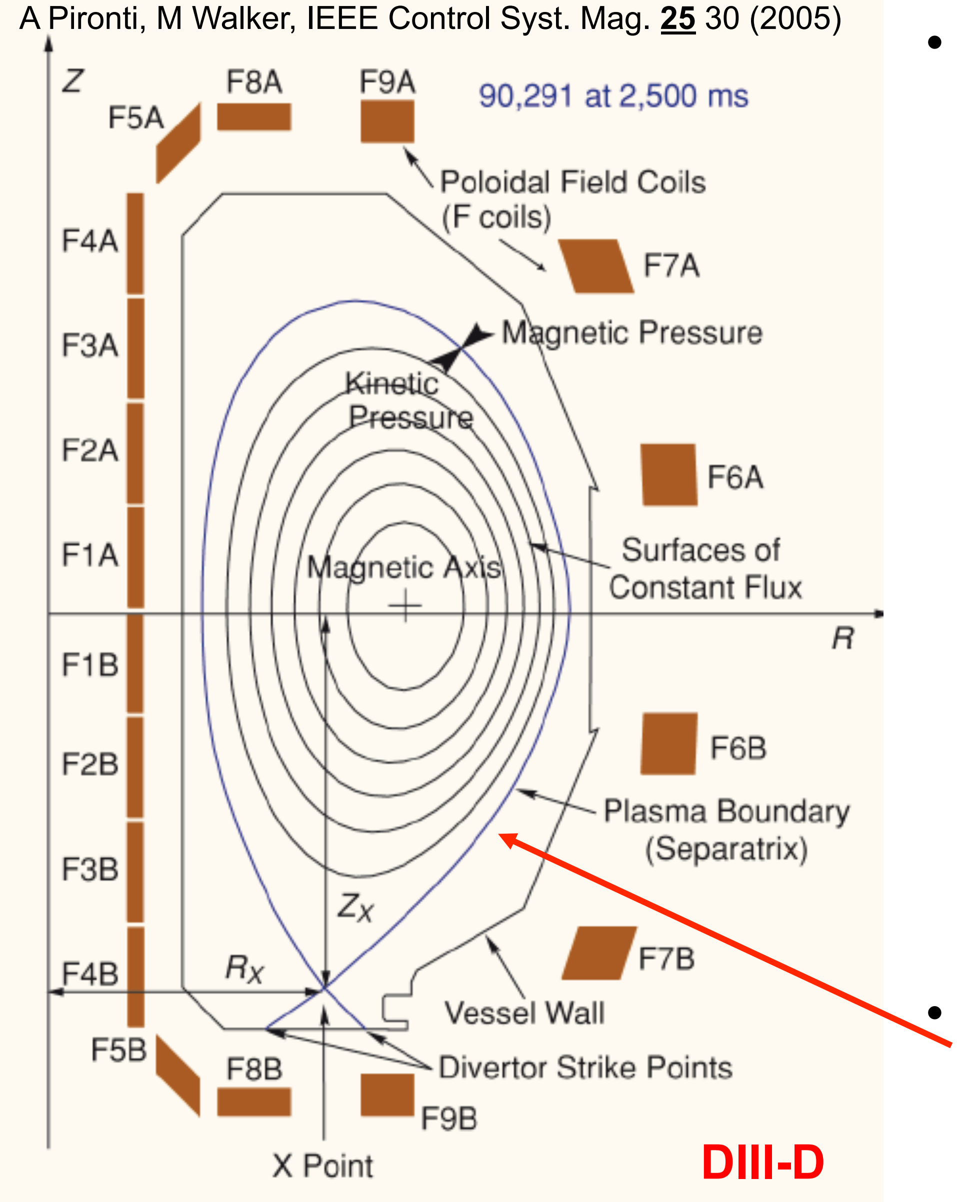


Large Helical Device (Japan)
Heliotron

- **Operation without a plasma current has some advantages (eg steady-state operation), but coil configuration is more complicated**
  - LHD is already in operation, while W7-X will enter operation in middle of decade
  - overall, stellarator energy confinement is similar to that in “equivalent current” tokamaks

Plasma Equilibrium in a Tokamak

A Pironti, M Walker, IEEE Control Syst. Mag. **25** 30 (2005)



- **Plasma is force-free, ie "in equilibrium":**
  - implies both internal and external force balance
  - ignoring internal flows and electric fields, force balance equation takes form:

$$\mathbf{j} \times \mathbf{B} = \nabla p$$

  - It follows that

$$\mathbf{B} \bullet \nabla p = \mathbf{j} \bullet \nabla p = 0$$

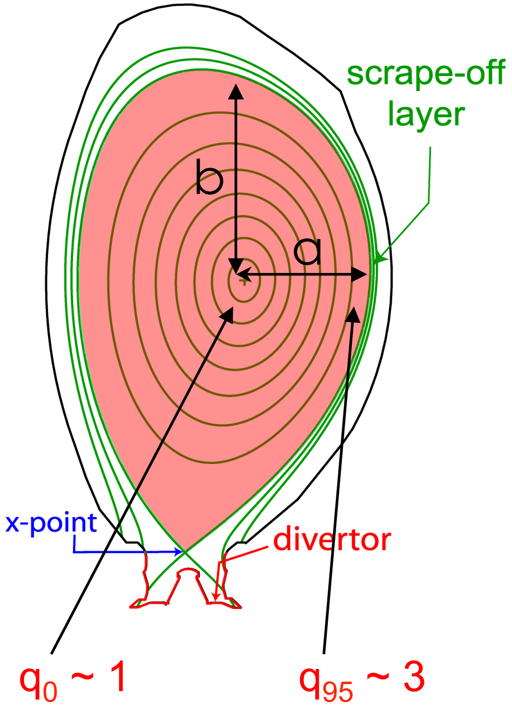
  ⇒ pressure is constant on magnetic surfaces

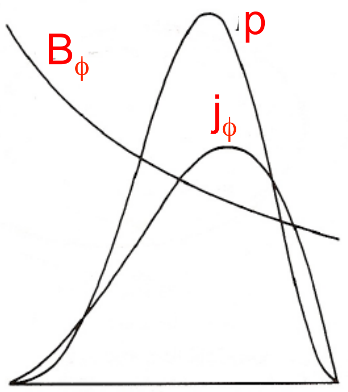
  ⇒ current lines lie within magnetic surfaces

- **Can define poloidal magnetic flux function, Ψ, satisfying,**

$$\mathbf{B} \bullet \nabla \Psi = 0$$

Plasma Equilibrium in a Tokamak





- **Formal definition of safety factor:**

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

toroidal flux

poloidal flux

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

  – define magnetic shear as:

$$s = \frac{r}{q}\frac{dq}{dr}$$

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

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

- **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

Plasma Equilibrium in a Tokamak

- **Since there internal force balance between the plasma pressure and the magnetic field, it is conventional to work with a normalized pressure, poloidal beta:**

$$\beta_p = \frac{2\mu_0 \int p.dV}{\left\langle B_\theta(a) \right\rangle_{line}^2 .V}$$

- when $\beta_p < 1$, plasma is paramagnetic
- when $\beta_p > 1$, plasma is diamagnetic

  - equilibrium condition limits β_p to approximately $\beta_p < R/a$

- **The plasma beta, i.e. pressure normalized to the toroidal field, is an important measure of plasma stability and of efficient use of field:**

$$\beta(\%) = 100 \times \frac{2\mu_0 \int p.dV}{B_\phi^2(0).V}$$

- **The plasma internal inductance characterizes how peaked the current profile is and is also a significant factor in plasma stability:**

$$\ell_i = \frac{\left\langle B_\theta^2 \right\rangle_{vol}}{\left\langle B_\theta^2(a) \right\rangle_{flux}} = \frac{4U_p}{\mu_0 R_0 I_p^2}$$

stored poloidal field energy inside plasma

- typically, $0.75 \le \ell_i \le 1.25$

(Some) plasma physics for magnetic confinement fusion in tokamaks

Some Basic Plasma Physics

- **Plasmas studied in fusion research are essentially, quasi-neutral, but there is a characteristic scale length for shielding of the potential due to individual charges, the Debye length:**

$$\lambda_D = \left(\frac{\varepsilon_0 T}{ne^2}\right)^{1/2} = 2.35\times 10^5 \left(\frac{T}{n}\right)^{1/2} \quad (T \text{ in keV})$$

  - the assumption of quasi-neutrality is satisfied if $n\lambda_D{}^3 >> 1$ (~10^8 in tokamak)

- **Characteristic plasma frequency:**

$$\omega_{p,e} = \left(\frac{ne^2}{\varepsilon_0 m_e}\right)^{1/2} = 56.4 n^{1/2}\ \mathrm{s}^{-1}$$

- **Motion of charged particles in confining magnetic fields can be characterized as a gyro-motion around of Larmor radius, ρ_L, around a guiding centre:**

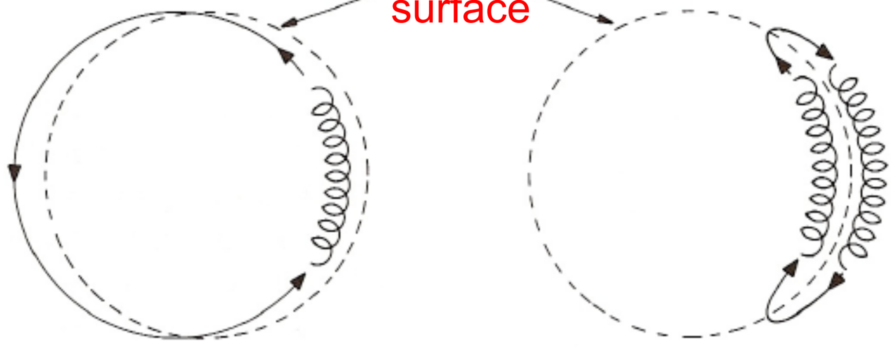
$$\rho_{Lj} = \sqrt{2}\,\frac{m_j v_{Tj}}{|e_j| B}$$

~ 100μm for electrons at 10keV and 3T

~ 5mm for protons at 10keV and 3T

Particle Orbits in the Tokamak

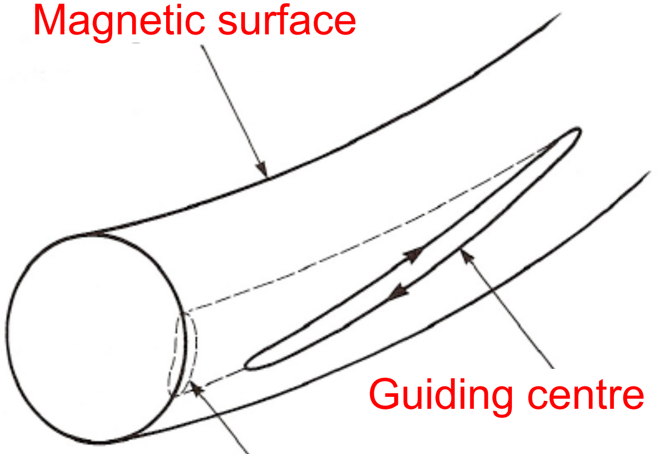
JA Wesson, *Tokamaks*, 3rd edition, OUP (2004)



Flux surface

Passing

Trapped



Magnetic surface

Guiding centre orbit

Projection onto poloidal plane

- **Gradients and curvature in the magnetic field lead to modifications in the particle trajectories:**
  - "Passing" particles – orbit shift: $\delta r_p \sim \varepsilon.\rho_{L\theta} = q.\rho_{L\phi} \quad (\varepsilon = a/R)$
  - "Trapped" particles – "banana" width: $\Delta r_t \sim \varepsilon^{0.5}\rho_{L\theta}$
  - Guiding centre orbit of trapped particles bounce back and forth on outer half of torus due to magnetic mirror formed by toroidal field ($B_\phi = R_0B_0/R$)
  - At low collision frequencies, a fraction of particles are trapped: $f = \sqrt{2r/(R_0+r)}$

Plasma Resistivity

- **How does one calculate the current achievable in a tokamak plasma (ignoring stability considerations)?**
  - Ohm's law for magnetized plasma:

resistivity

$$\mathbf{E} + \mathbf{v} \times \mathbf{B} = \eta \mathbf{j}$$

inductive electric field

current density

fluid velocity

magnetic field

  - average ionic charge number, "effective charge": $Z_{eff} = \dfrac{\sum_i n_i Z_i^2}{n_e}$
  - Coulomb logarithm characterizes average over electron-ion interactions:

$$\ln \Lambda_{ei} = 15.2 - \frac{1}{2} \ln \left(\frac{n_e}{10^{20}} \right) + \ln \left(T_e \right) \qquad (T_e \text{ in keV})$$

  - "Classical" (or Spitzer) parallel resistivity:

$$\eta_{par} = 1.65 \times 10^{-9} f(Z_{eff}) Z_{eff} \frac{\ln \Lambda_{ei}}{T_e^{1.5}} \ \Omega m \quad (T_e \text{ in keV}) \qquad (f(Z_{eff}) \sim 1)$$

NB!

  - e.g. T_e = 1keV, $\eta_{par} \sim 2 \times 10^{-8}\ \Omega$m – room temperature copper

"Neoclassical" Plasma Resistivity

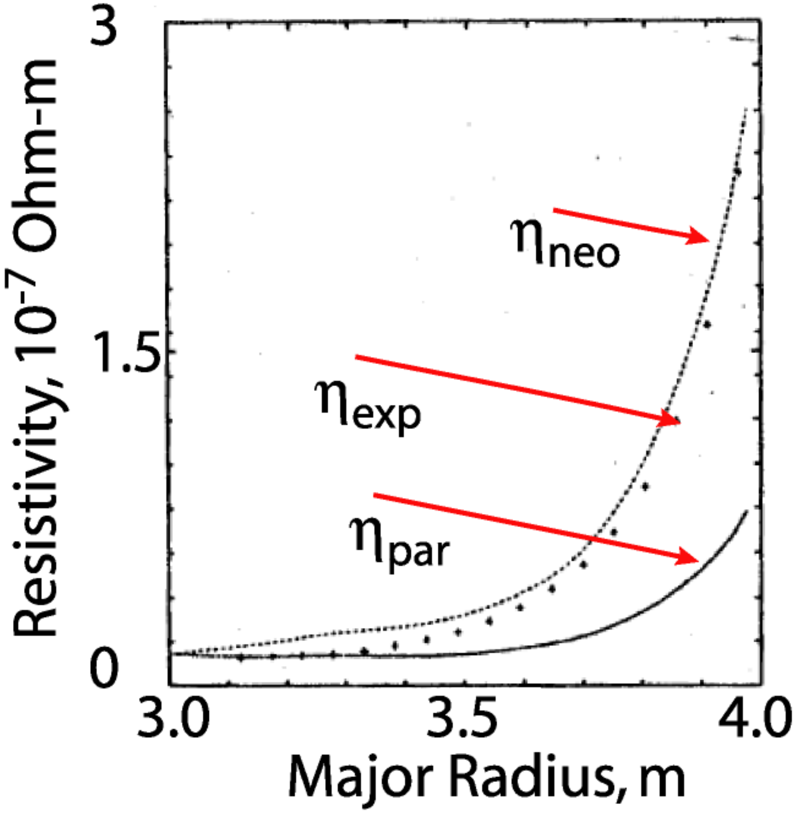
- **"Trapping" of particles in toroidal magnetic mirrors leads to an enhancement of the plasma resistivity:**

$$\eta_{neo} \approx \frac{\eta_{par}}{(1-\varepsilon^{0.5})^2} f(\nu^*,\varepsilon,Z_{eff})$$

collisionality

  - this effect first became detectable in the hot "collisionless" plasmas characteristic of JET scale devices

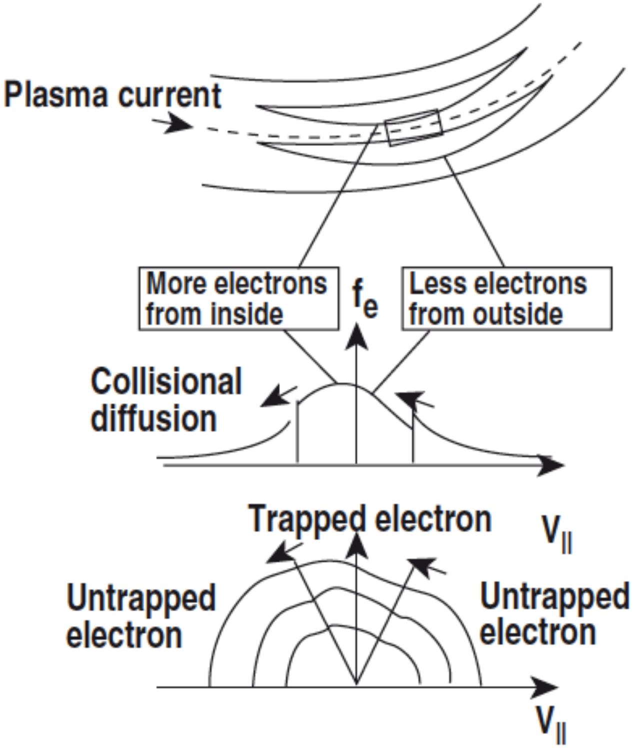
- **Comparisons between resistivity profiles calculated from T_e measurements and from resistive diffusion analysis of plasma current showed better agreement with the neoclassical resistivity**

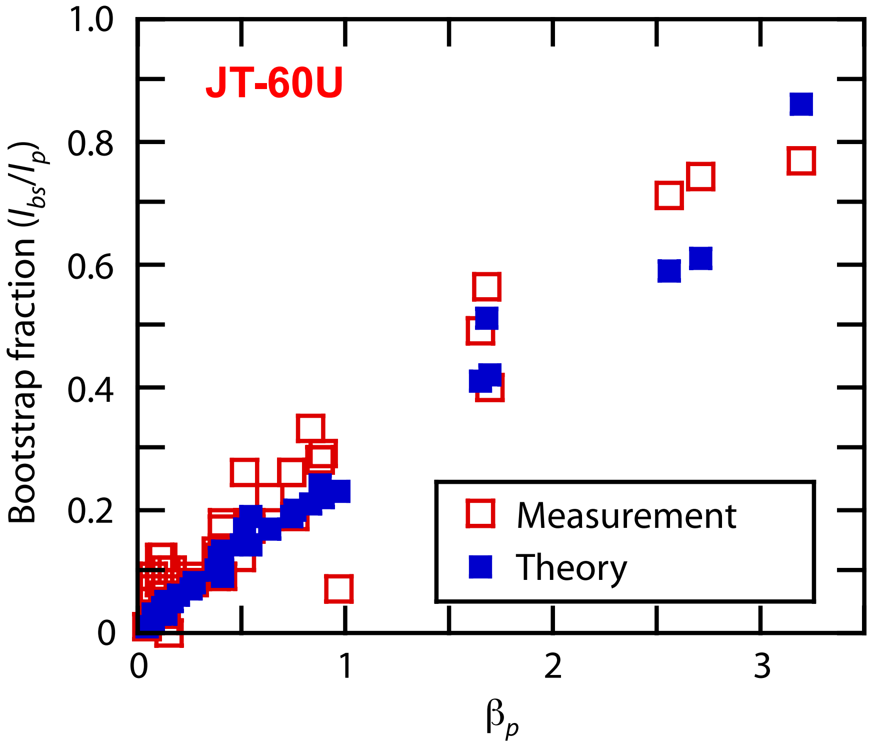


D J Campbell et al, Nucl Fusion **28** 981 (1988)

“Bootstrap” Current

M Kikuchi, M Azumi, Plasma Phys Control Fusion **37** 1215(1995)

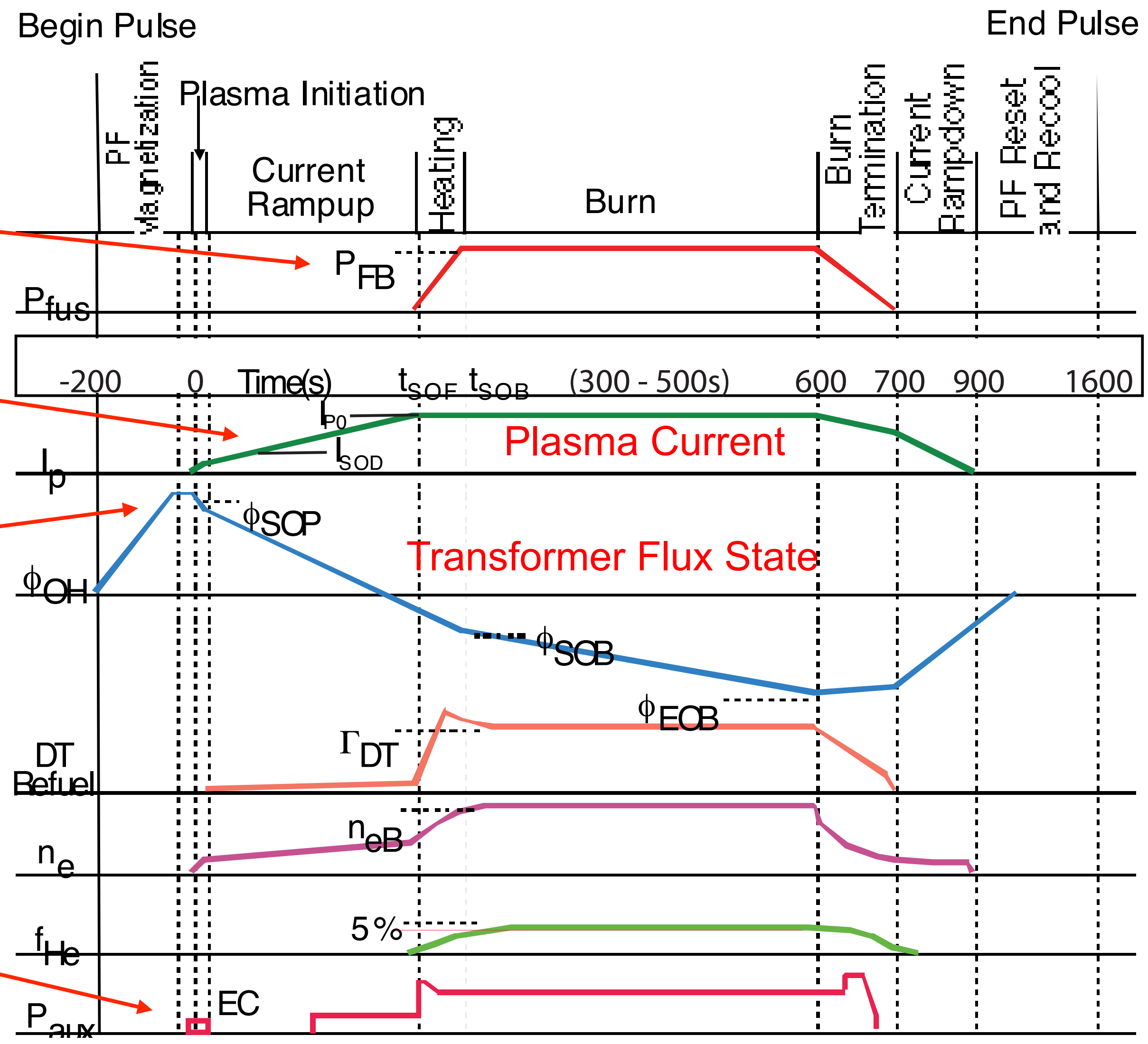




- **The “bootstrap” current is a further “neoclassical” consequence of the presence of trapped particles:**
  - momentum exchange between trapped and passing particles, together with density and temperature gradients, lead to an additional component of current:
  - locally: $j_{bs}(\varepsilon \to 1) = -\dfrac{1}{B_\theta}\dfrac{dp}{dr}$ globally: $I_{bs} = C\varepsilon^{1/2}\beta_p I_p \quad (C \sim 1/3 - 2/3)$

Overview of a Tokamak Plasma Pulse

Simulated ITER plasma pulse



4. Fusion burn produced during the current flat-top

3. Plasma current ramps up in 60-100s and is held constant for ~400s

1. Central solenoid is initially charged and then discharged to initiate plasma and ramp current up

2. External heating (EC) is used to assist ionization and “burnthrough” of radiation barrier

Tokamak Plasma Pulse – Flux Consumption

- **Flux consumption during an ITER plasma pulse:**

$$\Psi_{tot} = \Psi_{bd} + \Psi_{ramp} + \Psi_{ind} + \Psi_{res}$$

Ψ_{bd} = breakdown loss ~ 5 - 10 Wb

$\Psi_{ramp} = C_E \mu_0 R_0 I_p$ ~ 25 Wb (C$_E$ ~ 0.4 – 0.5 is an empirical coefficient)

$\Psi_{ind} = L_p I_p$ ~ 180 Wb

$$\left[L_p = \mu_0 R_0 \left(\ln \frac{8R_0}{a} + \frac{\ell_i}{2} - 2 \right) \approx 2R_0 \ (\mu\mathrm{H}) \right]$$

Ψ_{res} = resistive loss = $V_l I_p$ ~ 30 - 40 Wb

Ψ_{tot} ~ 240 - 260 Wb

- **During the current flat-top at 15 MA, the single turn loop voltage, V_l < 100 mV, due to:**
  - the high plasma temperature ($T_e(0)$ ~ 25 keV)
  - a bootstrap current contribution of ~10%
  - external “non-inductive” current drive of ~10%

The ITER Project

What is ITER?

ITER is a major international collaboration in fusion energy research involving the EU (plus Switzerland), 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 Deuterium-Tritium plasma in which α-particle heating dominates all other forms of plasma heating:**

⇒ a burning plasma experiment

ITER Scope - Mission Goals

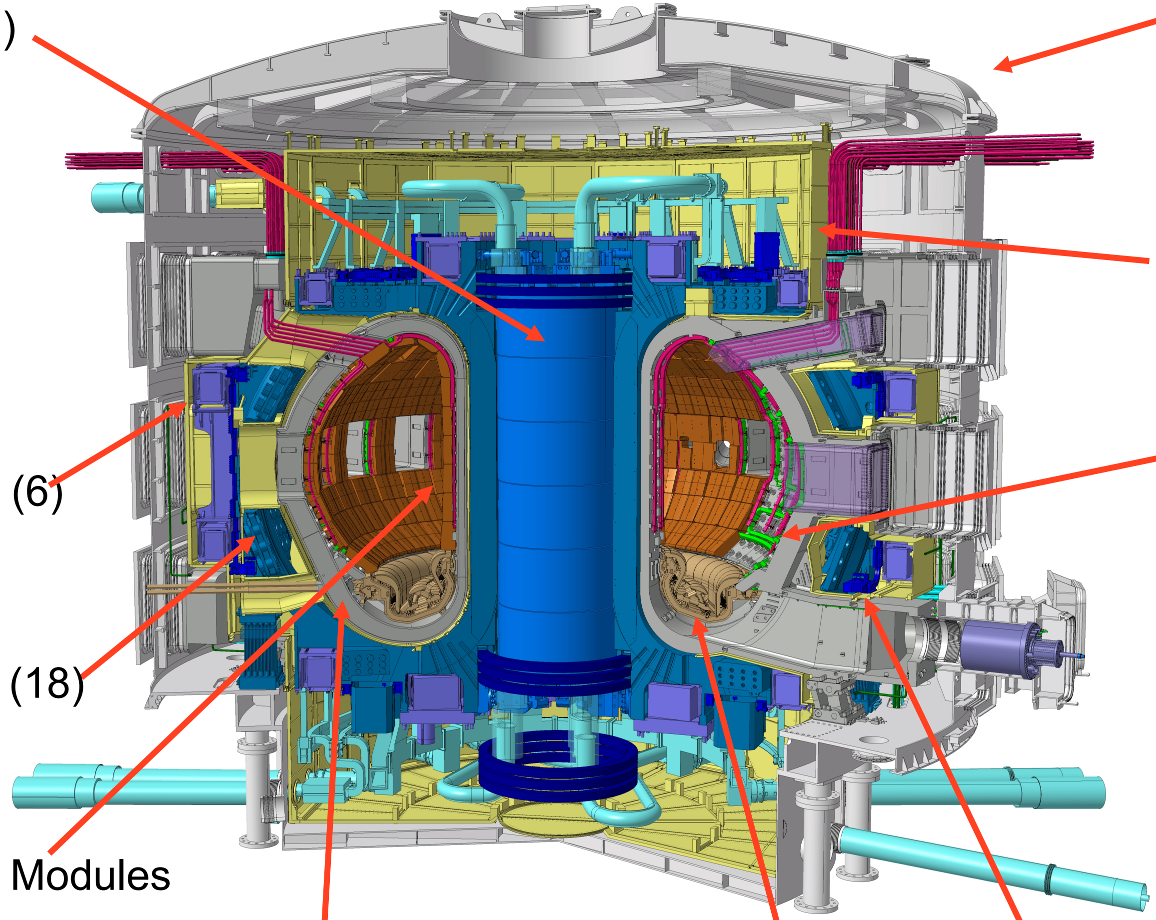
Physics:

- 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

ITER - Major Components



Central Solenoid (6)
(Nb_3Sn)

Cryostat

Thermal Shield

Poloidal Field Coils (6)
(NbTi)

Internal Coils
(Cu)

Toroidal Field Coils (18)
(Nb_3Sn)

Shielding Blanket Modules
(SS/ Be)

Correction Coils (18)
(NbTi)

Vacuum Vessel
(SS)

Divertor
(SS/ CFC-W)

The ITER Project - Current Status

- **Spring 2006: ITER Joint Work Site established in Cadarache - design teams arrive from Naka and Garching**
- **November 2006: ITER Agreement signed in Paris**
- **Late 2006: Design Review begins**
- **Early 2007: Construction activities launched**
- **October 2007: ITER Organization formally established**
- **July 2010: ITER Baseline (scope, schedule, cost) approved by ITER Council**
- **July 2010: New Director-General, Osamu Motojima appointed by ITER Council**
- **August 2010: Building construction begins on-site**

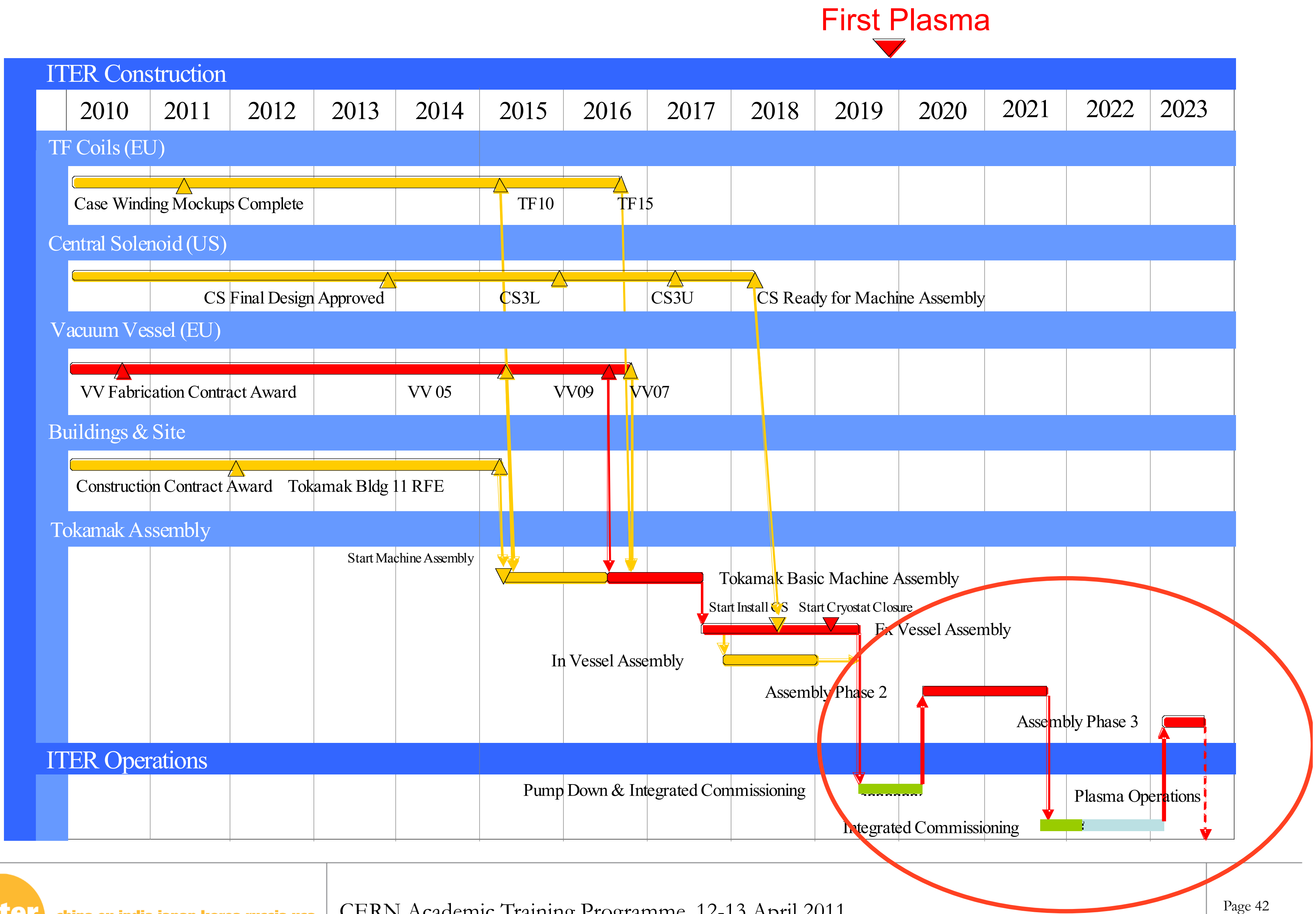
ITER Overall Project Cost (OPC)

- **The total cost for the Construction Phase approved in July 2010 is 4584.7 kIUA**
- **Table shows total cost over lifetime of project:**

Construction Phase	4584.7 kIUA
Operation Phase	188 kIUA per year
Deactivation Phase	EUR 281 Million
Decommissioning Phase	EUR 530 Million

NB: 1 kIUA = 1M $US (1989) = 1.5M Euro (2010)

ITER Construction Schedule



The ITER Project Team - Domestic Agencies

- **90% of ITER components will be supplied “in-kind” by the Members through their Domestic Agencies**

Itinerary of ITER Components

ITER Site

🟩 = Itinerary of ITER Components

ITER Construction at Cadarache

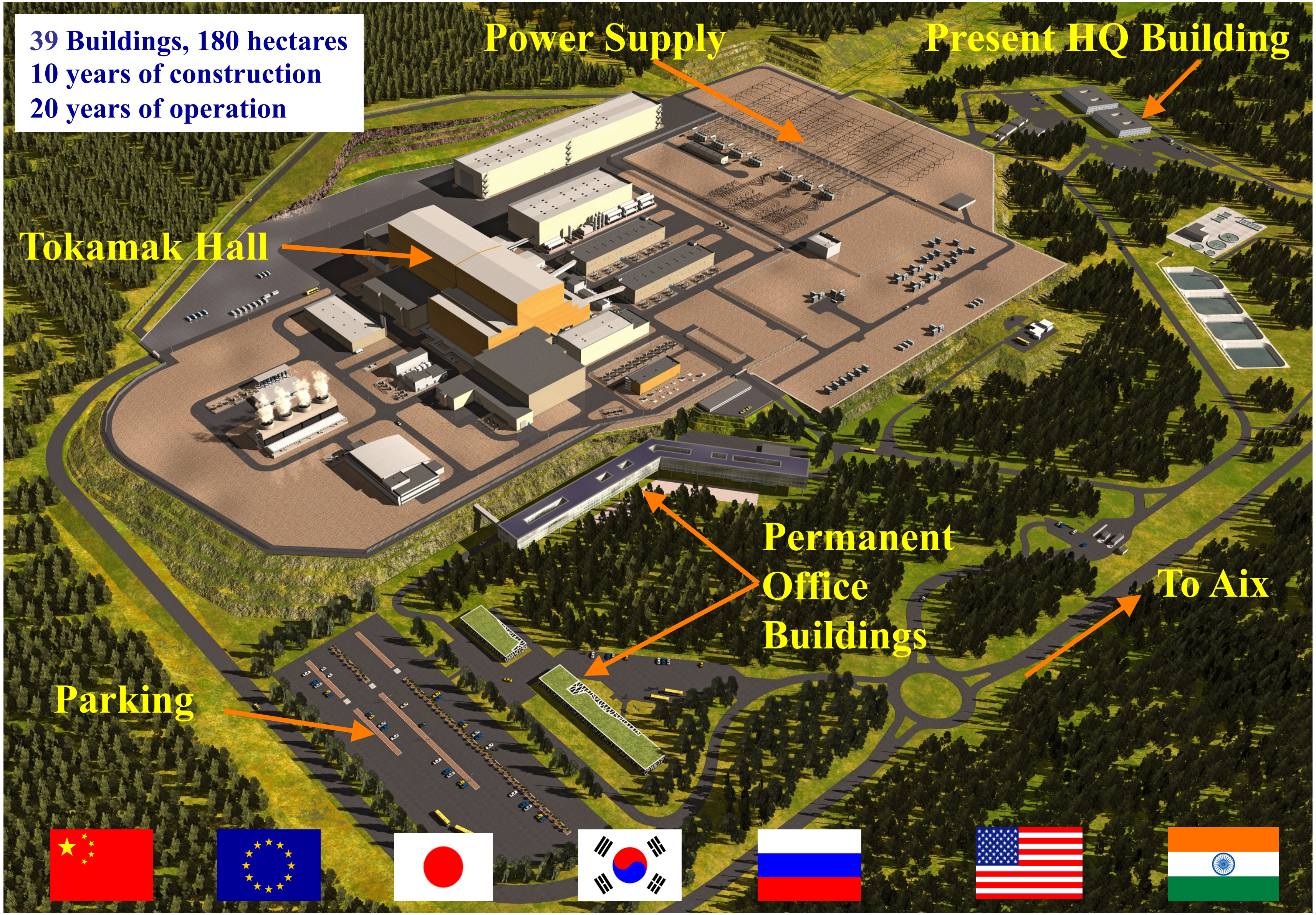
ITER Site platform levelling complete and construction underway

CEA Cadarache Site

ITER Temporary Headquarters

ITER site

ITER Site after Construction



39 Buildings, 180 hectares
10 years of construction
20 years of operation

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