ITER Physics

D J Campbell
ITER Organization, Cadarache

Acknowledgements:
Many colleagues in the ITER IO, ITER PTs and ITPA
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 $\alpha$-particle heating dominates all other forms of plasma heating:

  $\Rightarrow$ a burning plasma experiment
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 $\alpha$-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:

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

Fuel: $\text{D}_2, \text{T}_2$

Blanket: neutron absorber

Power Plant: $\text{Li} \Rightarrow \text{T}$ high temperature

Divertor: particle and heat exhaust

He, $\text{D}_2, \text{T}_2$, impurities
Temperature ($T_i$): \(1-2 \times 10^8\) °C (10-20 keV) 
\((\sim 10 \times \text{temperature of sun’s core})\)

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

Energy confinement time ($\tau_E$): few seconds
\((\text{plasma pulse duration } \sim 1000\text{s})\)

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”: \(Q \geq 30\)
• 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
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
Confinement scaling studies provide the robust approach to determining ITER’s size:
- detailed design relies on numerical codes combining engineering and physics constraints
• Predictions of fusion performance in ITER rely essentially on a small number of physics rules:
  • Energy confinement scaling (IPB98(y,2)):
    \[
    \tau_{E,th}^{98(y,2)} = 0.144 I^{0.93} B^{0.15} P^{-0.69} n^{0.41} M^{0.19} R^{1.97} \xi^{0.58} \kappa^{0.78} \text{(s)}
    \]
    \[
    \tau_E \propto I R^2 P^{-2/3}
    \]
  • H-mode threshold power:
    \[
    P_{LH} = 2.84 M^{-1} B^{0.82} \bar{n}_{20}^{0.58} Ra^{0.81} \text{(MW)}
    \]
• MHD stability:

\[
q_{95} = 3 \quad \text{and} \quad q_{95} = 2.5 \frac{a^2B}{RI} f(\varepsilon, \kappa, \delta)
\]

\[
\frac{n}{n_{GW}} \leq 1 \quad \text{and} \quad n_{GW}(10^{20}) = \frac{l(\text{MA})}{\pi a^2}
\]

\[
\beta_N \leq 2.5 \quad \text{and} \quad \beta_N = \beta(\%) \frac{aB}{l(\text{MA})}
\]

\[\kappa, \delta \text{ determined by control considerations}\]

• Divertor physics:

Peak target power \( \sim 10\text{MWm}^{-2} \)

Helium transport: \( \frac{\tau_{He}^*}{\tau_E} \sim 5 \)

Impurity content: \( \frac{n_{Be}}{n_e} = 0.02 (+ \sim 0.1\% \text{ Ar for radiation}) \)
ITER Design Parameters

<table>
<thead>
<tr>
<th></th>
<th>ITER</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius</td>
<td>6.2 m</td>
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<tr>
<td>Minor radius</td>
<td>2.0 m</td>
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<tr>
<td>Plasma current</td>
<td>15 MA</td>
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<td>Toroidal magnetic field</td>
<td>5.3T</td>
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<tr>
<td>Elongation / triangularity</td>
<td>1.85 / 0.49</td>
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<tr>
<td>Fusion power amplification</td>
<td>$\leq 10$</td>
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<tr>
<td>Fusion power</td>
<td>~400 MW</td>
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<tr>
<td>Plasma burn duration</td>
<td>~400 s</td>
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</table>

A detailed engineering design for ITER was delivered in July 2001
ITER Main Features

- Central Solenoid
- Outer Intercoil Structure
- Toroidal Field Coil
- Poloidal Field Coil
- Machine Gravity Support
- Blanket Module
- Vacuum Vessel
- Cryostat
- Port Plug (EC Heating)
- Divertor
- Torus Cryopump
### Heating System

<table>
<thead>
<tr>
<th>Heating System</th>
<th>Stage 1</th>
<th>Possible Upgrade</th>
<th>Remarks</th>
</tr>
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<tbody>
<tr>
<td>NBI (1MeV Čive ion)</td>
<td>33</td>
<td>16.5</td>
<td>Vertically steerable (z at Rtan (-0.42)m to (+0.16)m)</td>
</tr>
<tr>
<td>ECH&amp;CD (170GHz)</td>
<td>20</td>
<td>20</td>
<td>Equatorial and upper port launchers steerable</td>
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<tr>
<td>ICH&amp;CD (40-55MHz)</td>
<td>20</td>
<td></td>
<td>(2\Omega_\text{T} (50%) power to ions (\Omega_{\text{He}3} (70%) power to ions, FWCD)</td>
</tr>
<tr>
<td>LHH&amp;CD (5GHz)</td>
<td>20</td>
<td></td>
<td>(1.8&lt;n_{\text{par}}&lt;2.2)</td>
</tr>
<tr>
<td><strong>Total</strong></td>
<td>73</td>
<td>130 (110 simultan)</td>
<td>Upgrade in different RF combinations possible</td>
</tr>
<tr>
<td>ECRH Startup</td>
<td>2</td>
<td></td>
<td>120GHz</td>
</tr>
<tr>
<td>Diagnostic Beam (100keV, H(^-))</td>
<td>&gt;2</td>
<td></td>
<td></td>
</tr>
</tbody>
</table>

\(P_{\text{aux}}\) for Q=10 nominal scenario: 40-50MW

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**Notes:**
- **NBI Layout**
- **DNB**
• About 40 large scale diagnostic systems are foreseen:
  • Diagnostics required for protection, control and physics studies
  • Measurements from DC to $\gamma$-rays, neutrons, $\alpha$-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)
$\alpha$-particle spectrum (TFTR)
An ITER Plasma

A Q=10 scenario with:

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

Current Ramp-up Phase
ITER Physics - Opportunities and Challenges

Operational scenarios and control issues
• **Baseline scenarios:**

**Single confinement barrier**

- **ELMy H-mode:**
  - Q=10 for ≥300s
  - well understood physics
  - extrapolation to:
    - control
    - self-heating
    - $\alpha$-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 $\alpha$-particle confinement
Physics Design Rules

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

2 MHD stability:
- $q_{95} = 3$
- $\kappa$, $\delta$ determined by control requirements
- $n \leq n_{GW}$
- $\beta_N \leq 2.5$

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

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

First ITER Summer School, Aix-en-Provence, 16-20 July 2007
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

Conservative scenario for technology testing

Conservative scenario for technology testing

Conservative scenario for technology testing
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
- 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
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”
  - $I_p=9$MA, $q_{95}=5.3$
  - $H_{98(y,2)}=1.3$, $\beta_N=2.56$
  - $P_{LH}+P_{NB}=34+34$MW, $P_{fus}=340$MW

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

- $H_{98}^\beta_N/\varphi_{95}^2$ provides a performance figure of merit for extrapolation to ITER-scale plasmas

- a major challenge to extend high performance to long pulses
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 turbulence and magnetohydrodynamic instabilities (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
  - 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 \times \text{burn time}
** limited by external cooling capacity (at present)

- Current diffusion requires:
  \[ \tau \sim \tau_R \]
- But execution of control implies
  \[ \tau >> \tau_R \]

\[ \Rightarrow \] 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

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

- 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)
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)

JET Complex Plasma Control
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, $\kappa$, has a maximum value which depends on plasma equilibrium and its inductive coupling to tokamak structure (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, $\beta \propto p/B^2$, 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

- $l_i$-$q$ 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)
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:
  \[ n_{GW} = \frac{I(\text{MA})}{\pi a^2} \]
- operational figure of merit
Maximum value of normalized plasma pressure, $\beta$, is limited by mhd instabilities:

$$\beta(\%) = 100 \frac{<p>}{B^2 / 2\mu_0}$$

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

Typically, “Troyon” limit describes tokamak plasmas:

$$\beta_N \leq 2.8-3.5$$

More generally, “no-wall” limit:

$$\beta_N \leq 4 |I_i|$$
• NTMs can determine the \( \beta \)-limit below ideal limit
  • experiments show that \( \beta_{N(\text{critical})} \propto \rho_i^* \)
• 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))

First ITER Summer School, Aix-en-Provence, 16-20 July 2007
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-$\beta_N$ Stability: Resistive Wall Mode Control

Advanced scenarios at high $\beta_N$ ($\beta_N \sim 4l_i$) 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
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%.
• **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
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 \( \Rightarrow \) ELMs
  □ \( \Delta W_{ELM} \) is small fraction of \( W_{plasma} \) (\(< 10 \%\)) in \( \sim 200 \, \mu s \)
  \( \Rightarrow \) Large Energy Flux

ASDEX Upgrade Herrmann

Outer divertor

Inner divertor
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
Disruptions

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

  \[ \implies \text{MHD unstable modes grow} \]
  \[ \implies \text{plasma confinement is destroyed (thermal quench)} \]
  \[ \implies \text{plasma current vanishes (current quench)} \]

Typical timescales

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

Typical values for ITER current quench

- \( W_{\text{poloidal}} \approx 1 \text{ GJ} \)
- \( \tau_{\text{c.q.}} \approx 20-40 \text{ ms} \)
- \( q_{\text{rad}} \approx 35 – 70 \text{ MWm}^{-2} \)
- \( A_{\text{wall}} \approx 700 \text{ m}^{2} \)
- \( q_{\text{rad}} \tau_{\text{c.q.}}^{1/2} \approx 7–10 \text{ MJm}^{-2}\text{s}^{-1/2} \) (no Be melting)
When a loss of vertical position control takes place

- plasma impacts wall with full plasma energy
- high localized heating
- mitigation required

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
• 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
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
Physics constrained by experimental goals:

- Scaling of transport/confinement etc to reactor scale
- MHD stability at the reactor scale (e.g., sawteeth, NTM, RWM)
- Power and particle exhaust (e.g., edge/core integration, transients)
- Development of steady-state operation (e.g., 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 $\alpha$-heating
  - influence of $\alpha$-particles on MHD stability
In existing experiments single particle theory of energetic ion confinement confirmed:
- simple estimate, based on banana orbit width shows that $I_\rho \geq 3M\AA$ required for $\alpha$-particle confinement

Classical slowing down of fast ions well validated:
- data range 30keV NBI (ISX-B) to 3.5MeV $\alpha$-particles (TFTR)

Energetic ion heating processes routinely observed in additional heating experiments

Electron heating by $\alpha$-particles validated in JET and TFTR DT experiments:

- in JET, maximum electron temperature correlated with optimum fuel mixture for fusion power production

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
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 $\alpha$-particles
Alfvén Eigenmode Stability

Linear stability:
• Pressure gradient of resonant particles with $v \sim v_A$ 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 toroidal field ripple 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

<table>
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<tr>
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<th>Inductive</th>
<th>Weak RS (#4)</th>
<th>Strong RS</th>
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<tbody>
<tr>
<td></td>
<td>No FI</td>
<td>With FI</td>
<td>No FI</td>
</tr>
<tr>
<td>Total particle loss fraction (%)</td>
<td>2.15</td>
<td>negligible</td>
<td>6.5</td>
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<tr>
<td>Total power loss fraction (%)</td>
<td>0.65</td>
<td>negligible</td>
<td>2.5</td>
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<tr>
<td>Peak FW heat load (MWm⁻²)</td>
<td>&lt; 0.1</td>
<td>negligible</td>
<td>0.23</td>
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<tr>
<td>Plasma current (MA)</td>
<td>15</td>
<td>10</td>
<td>10</td>
</tr>
</tbody>
</table>

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

- Normalized half-width of fast ion banana orbit:
  \[ \frac{\delta_f}{a} = \frac{q}{\varepsilon^{0.5}} \frac{r_f}{a} \]

- Fractional density of fast ions:
  \[ \frac{n_f}{n_e} \]

- Normalized axial fast ion pressure:
  \[ \beta_f(0) \]

- Dimensionless fast ion pressure gradient:
  \[ \text{max} |R.\nabla \beta_f| \]

- Ratio of fast ion velocity to central Alfvén velocity:
  \[ v_f / v_A(0) \]

- Central Alfvén velocity:
  \[ v_A = \frac{B}{\sqrt{\mu_0 \rho_{\text{mass}}}} \]
• Excitation of AEs and their influence on $\alpha$-particle confinement is a central question for viability of advanced scenarios

• $\alpha$-particle parameters in ITER allows access to relevant range where $\alpha$-driven instabilities and their influence on $\alpha$-particle transport can be studied:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>$\alpha$‘s (TFTR)</th>
<th>$\alpha$‘s (JET)</th>
<th>$\alpha$‘s (ITER)</th>
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<tbody>
<tr>
<td>$P_f(0)$ [MWm$^{-3}$]</td>
<td>0.3</td>
<td>0.16</td>
<td>0.44</td>
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<tr>
<td>$\delta/a$</td>
<td>0.3</td>
<td>0.34</td>
<td>0.08</td>
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<tr>
<td>$n_f(0)/n_e(0)$ [%]</td>
<td>0.3</td>
<td>0.17</td>
<td>0.8</td>
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<tr>
<td>$\beta_f(0)$ [%]</td>
<td>0.26</td>
<td>0.3</td>
<td>1.1</td>
</tr>
<tr>
<td>$\langle \beta_f \rangle$ [%]</td>
<td>0.03</td>
<td>0.04</td>
<td>0.16</td>
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<tr>
<td>$\max</td>
<td>R. \nabla \beta_f</td>
<td>$ [%]</td>
<td>2</td>
</tr>
<tr>
<td>$v_f / v_A(0)$</td>
<td>1.6</td>
<td>1.4</td>
<td>1.8</td>
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</table>

• Higher power density in reactor could be even more challenging

$\Rightarrow$ ITER programme aims to move in this direction
ITER 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
ITER Agreement Signature

Ceremony ITER Agreement Signature, Elysee Palace, 21 November 2006
**CENTRAL SOLENOID MODEL COIL**
- Radius 3.5 m
- Height 2.8 m
- \( B_{\text{max}} = 13 \, \text{T} \)
- \( 0.6 \, \text{T/sec} \)

**REMOTE MAINTENANCE OF DIVERTOR CASSETTE**
- Attachment Tolerance ± 2 mm

**DIVERTOR CASSETTE**
- Heat Flux 20 MW/m²

**TOROIDAL FIELD MODEL COIL**
- Height 4 m
- Width 3 m
- \( B_{\text{max}} = 7.8 \, \text{T} \)

**VACUUM VESSEL SECTOR**
- Double-Wall, Tolerance ± 5 mm

**REMOTE MAINTENANCE OF BLANKET**
- 4 t Blanket Sector
- Attachment Tolerance ± 0.25 mm

**BLANKET MODULE**
- HIP Joining Technology
- Size: 1.6 m x 0.93 m x 0.35 m

**ITER Technology R&D Advanced**
## Project Schedule

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**COILS**

- Bid
- Vendor Design
- Contract
- 1st PFC
- 1st TFC
- Fabrication
- Last TFC
- Last PFC
- Last CS

**VACUUM VESSEL**

- Bid
- Vendor Design
- Contract
- Fabrication
- First sector
- Last sector

**Site Preparation Contract**

- Clearing/Leveling
- Excavation

**Construction License Process**

- Site Prep Contract

**TOKAMAK ASSEMBLY**

- 1st VV/TF/TS Sector
- Complete VV
- Complete BLK/DIV
- Install Cryostat
- Install CS
- 1st PFC
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