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Introduction to Fusion Leading to ITER

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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:

\[ DT \text{ fusion: } 17.6 \text{ MeV} \]
\[ U \text{ fission: } \sim 200 \text{ MeV} \]
Essential Fusion Reactions

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

\[ ^2D + ^3T \rightarrow ^4He \ (3.5 \text{ MeV}) + ^1n \ (14.1 \text{ MeV}) \]

Two other important reactions for DT fusion:

\[ ^1n + ^6Li \rightarrow ^4He + ^3T + 4.8 \text{ MeV} \]

\[ ^1n + ^7Li \rightarrow ^3He + ^3T + ^1n - 2.5 \text{ MeV} \]

- 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:

\[ F = q(E + v \times B) \]

- 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)
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 m³
  - 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

with plasma
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
Formal definition of safety factor:

\[ q = \frac{d\Phi}{d\Psi} \]

- 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 \( q \):
  \[ \kappa = \frac{b}{a} \]
- \( \kappa \) 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 \( q(0) \sim 1 \)
Many Plasma Shapes Have Been Investigated

- Plasma shape affects confinement and stability properties

R Stambaugh, APS (2000)
Plasma fusion performance

**Temperature -** $T_i$: $1 - 2 \times 10^8$ K (10-20 keV)

($\sim 10 \times$ temperature of sun’s core)

**Density -** $n_i$: $1 \times 10^{20}$ m$^{-3}$

($\sim 10^{-6}$ of atmospheric particle density)

**Energy confinement time -** $\tau_E$: few seconds ($\propto$ current $\times$ radius$^2$)

(ITER plasma pulse duration $\sim$1000s)

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

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

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

$\Rightarrow$ “Controlled ignition”: $Q \geq 30$
• Existing experiments have achieved $nT\tau$ values
  $\sim 1 \times 10^{21} \text{ m}^{-3}\text{skeV}$
  $\sim Q_{\text{DT}} = 1$

• JET and TFTR have produced DT fusion powers of $>10\text{MW}$ for $\sim 1\text{s}$

• ITER is designed to a scale which should yield $Q_{\text{DT}} \geq 10$ at a fusion power of 400 - 500MW for 300-500s
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
  - \( E_b \sim 100 \text{ keV}, P_b \text{ up to } 40\text{MW} \) in TFTR
  - very effective:
    - heating
    - current drive
    - fuelling
    - rotation drive

• For ITER:
  - \( E_b \sim 1 \text{ MeV} \) is required to penetrate plasma/deserve current
  - \( \rightarrow \) negative ion source technology
  - higher energy \( \Rightarrow \) little fuelling, little rotation drive
Radiofrequency Heating

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

• **Electron Cyclotron Resonance Heating (ECRH):**
  - launched at frequencies ~ $\omega_{ce}$ ⇒ $f > 100$ 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: $f \sim 5$ 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 \(\eta_{CD} = \text{driven current/input power}):
  - typically increases with \(T_e\)
  - for beams, also increases with \(E_b\)
  \(\Rightarrow\) favourable for ITER

Plasma Confinement: H-mode

• It is found that the plasma confinement state ($\tau_E$) 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
• Predictions of fusion performance in ITER rely essentially on a small number of physics rules:
  • H-mode energy confinement scaling (IPB98(y,2)):
    \[
    \tau_{E,th}^{98(y,2)} = 0.144 I^{0.93} B^{0.15} P^{-0.69} n^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa^{0.78} \text{ (s)}
    \]
    \[
    \tau_E \propto IR^2 P^{-\frac{2}{3}}
    \]
    NB: \( H_{98(y,2)} = \frac{\tau_{E,th}^{exp}}{\tau_{E,th}^{98(y,2)}} \)
  • H-mode threshold power:
    \[
    P_{LH} = 0.098 M^{-1} B^{0.80} n_{20}^{0.72} S^{0.94} \text{ (MW)}
    \]
  (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 ($\tau_{He}^*/\tau_E$) 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 scaling prediction for ITER energy confinement time, $\tau$.

$$\tau_{th} \propto I_p R^2 P^{-2/3}$$
ITER is twice as large as our largest existing experiments

Tore Supra
\[ V_{\text{plasma}} = 25 \, m^3 \]
\[ P_{\text{fusion}} \approx 0 \, MW \]
\[ t_{\text{plasma}} \approx 400 \, s \]

JET
\[ V_{\text{plasma}} = 80 \, m^3 \]
\[ P_{\text{fusion}} \approx 16 \, MW \, 1s \]
\[ t_{\text{plasma}} \approx 30 \, s \]

ITER
\[ V_{\text{plasma}} = 830 \, m^3 \]
\[ P_{\text{fusion}} \approx 500 \, MW \, 300 – 500 \, s \]
\[ t_{\text{plasma}} \approx 600 – 3000 \, s \]
ITER Physics Basis II

• MHD stability:

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

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

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

\( \kappa, \delta \) determined by control considerations

\[ \beta = \frac{\text{plasma kinetic energy}}{\text{plasma magnetic energy}} \]

• Divertor physics:

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

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

Impurity content: \( n_{\text{Be}} / n_e = 0.02 (+ \sim 0.1\% \text{ Ar for radiation}) \)
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
• \( l_i - q_a \) 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)

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

Greenwald 'limit':
\[ n_{e,\text{lim}}(10^{20} \text{ m}^{-3}) = \frac{I(\text{MA})}{\pi a^2} \]
Plasma MHD Stability – Pressure Limit: $\beta$

- Maximum value of normalized plasma pressure, $\beta$, is limited by MHD instabilities:

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

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

- Typically, “Troyon” limit describes tokamak plasmas:

$$\beta_N \leq 2.8-3.5$$

- More generally, “no-wall” limit:

$$\beta_N \leq 4 \times |i|$$
• 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

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

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 $\alpha$’s in plasma
  - response of plasma to $\alpha$-heating
  - influence of $\alpha$-particles on MHD stability

- Experiments in existing tokamaks have already provided some positive evidence
  - “energetic” particles (including $\alpha$-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 $I_p \geq 3\text{MA}$ 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

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 $\alpha$-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 $\beta$ operation $\Rightarrow$ robust MHD stability
  • Effective disruption avoidance and control
  • Power (and particle) exhaust with relevant PFCs
  • Tritium efficiency
  • $\alpha$-particle confinement
  • Reactor-relevant auxiliary systems (H&CD, diagnostics, fuelling, control …)
## ITER on the Path to Fusion Energy

<table>
<thead>
<tr>
<th>When?</th>
<th>Fusion Power</th>
<th>Burn Duration</th>
<th>Q</th>
</tr>
</thead>
<tbody>
<tr>
<td>1997</td>
<td>16 MW</td>
<td>~1 second</td>
<td>0.65</td>
</tr>
<tr>
<td>2027-2028</td>
<td>500-700 MW</td>
<td>~7 minutes</td>
<td>10</td>
</tr>
<tr>
<td>~2040</td>
<td>2-2.5 GW</td>
<td>days/ steady-state</td>
<td>30</td>
</tr>
</tbody>
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References: Tokamak Fusion Physics


[http://www.iter.org](http://www.iter.org) - and associated links