Lecture 2: How to Design a Tokamak?

AP 4990y Seminar
Columbia University
Spring, 2011
## Summary: ITER's Parameters

\[ P_\alpha + P_{aux} + P_{oh} = 123 \text{ MW} \]

\[ P_{rad} = P_{brem} + P_{cy} + P_{line} = 48 \text{ MW} \]

### Table 5.1. Nominal parameters of ITER-FEAT in inductive operation

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Units</th>
<th>Reference Q = 10</th>
<th>High Q, high P_{ fus}</th>
</tr>
</thead>
<tbody>
<tr>
<td>R/a</td>
<td>m/m</td>
<td>6.2 / 2.00</td>
<td>6.2 / 2.00</td>
</tr>
<tr>
<td>Volume</td>
<td>m³</td>
<td>837</td>
<td>837</td>
</tr>
<tr>
<td>Surface</td>
<td>m²</td>
<td>678</td>
<td>678</td>
</tr>
<tr>
<td>Sep. length</td>
<td>m</td>
<td>18.4</td>
<td>18.4</td>
</tr>
<tr>
<td>S_{cross-sect.}</td>
<td>m²</td>
<td>21.9</td>
<td>21.9</td>
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<tr>
<td>B_\text{T}</td>
<td>T</td>
<td>5.3</td>
<td>5.3</td>
</tr>
<tr>
<td>I_p</td>
<td>MA</td>
<td>15.0</td>
<td>17.4</td>
</tr>
<tr>
<td>\kappa / \delta</td>
<td></td>
<td>1.86 / 0.5</td>
<td>1.86 / 0.5</td>
</tr>
<tr>
<td>\kappa_{95} / \delta_{95}</td>
<td></td>
<td>1.7 / 0.35</td>
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</tr>
<tr>
<td>l_{(3)}</td>
<td></td>
<td>0.86</td>
<td>0.78</td>
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<tr>
<td>V_{loop}</td>
<td>mV</td>
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</tr>
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<td></td>
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<td></td>
<td>1.77</td>
<td>1.93</td>
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<td>&lt;n_e&gt; / n_{e,GW}</td>
<td></td>
<td>10^{19} m^{-3}</td>
<td>10.14 / 11.56</td>
</tr>
<tr>
<td>\tau_e, s</td>
<td></td>
<td>3.7</td>
<td>4.1</td>
</tr>
<tr>
<td>W_{th}</td>
<td>MJ</td>
<td>325</td>
<td>408</td>
</tr>
<tr>
<td>W_{fast}</td>
<td>MJ</td>
<td>25</td>
<td>33</td>
</tr>
<tr>
<td>H_{H_{IPB98},(v,2)}</td>
<td></td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>\tau_{\alpha}/\tau_{E}</td>
<td></td>
<td>5.0</td>
<td>5.0</td>
</tr>
<tr>
<td>Z_{eff}</td>
<td></td>
<td>1.65</td>
<td>1.69</td>
</tr>
<tr>
<td>f_{He, axis}</td>
<td>%</td>
<td>4.1</td>
<td>5.9</td>
</tr>
<tr>
<td>f_{He, axis}</td>
<td>%</td>
<td>2.0</td>
<td>2.0</td>
</tr>
<tr>
<td>f_{C, axis}</td>
<td>%</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>f_{Ar, axis}</td>
<td>%</td>
<td>0.12</td>
<td>0.11</td>
</tr>
</tbody>
</table>

\[ \frac{W}{\tau_{E}} \sim 88 \text{ MW} \]

Performance calculations using the agreed physics guidelines yield a substantial operating window for Q ≥ 10 inductive operation for the selected parameter set.
Outline

- Fusion power and Q
- Plasma operational limits
- Technology limits
- ITER’s discharge targets
General tokamak design rules are now well-established


**The physics of magnetic fusion reactors**

John Sheffield  
*Oak Ridge National Laboratory, Oak Ridge, Tennessee 37831*

During the past two decades there have been substantial advances in magnetic fusion research. On the experimental front, progress has been led by the mainline tokamaks, which have achieved reactor-level values of temperature and plasma pressure. Comparable progress, when allowance is made for their smaller programs, has been made in complementary configurations such as the stellarator, reversed-field pinch and field-reversed configuration. In this paper, the status of understanding of the physics of toroidal plasmas is reviewed. **It is shown how the physics performance, constrained by technological and economic realities, determines the form of reference toroidal reactors.** A comparative study of example reactors is not made, because the level of confidence in projections of their performance varies widely, reflecting the vastly different levels of support which each has received. **Success with the tokamak has led to the initiation of the International Thermonuclear Experimental Reactor project.** It is designed to produce 1500 MW of fusion power from a deuterium-tritium plasma for pulses of 1000 s or longer and to demonstrate the integration of the plasma and nuclear technologies needed for a demonstration reactor.
Basics

• Tokamak geometry (surface, volume, stability, field: $B$, $a$, $\kappa = b/a$, $\varepsilon = a/R$, ...)

• Plasma stability (pressure limits, density limits, current limits: $q$, $\beta$, $\beta_N$, $n_G$, ...)

• Nuclear reactivity ($\sigma$, ...)

• Power balance (thermal conduction, radiation: $\tau_E$, $\chi$, $\rho_{brem}$, $\rho_{cyc}$, ...)

Thursday, February 3, 2011
D-T ($^6$Li) Fusion: Easiest Fuel for Laboratory Power

$$3D + 3(^6Li) \rightarrow 6(^4He) + 3n + (10.5 \text{ MeV plasma}) + (56.4 \text{ MeV blanket})$$

- D-T fusion has largest cross-section and lowest $T \sim 170,000,000^\circ$.
- Tritium is created from $^6$Li forming a **self-sufficient fuel cycle**. Practically no resource limit ($10^{11}$ TW y D; $10^4(10^8)$ TW y $^6$Li)!
- **Notice:** ~ 80% of energy as fast neutrons (~ 1.5 m shielding).

⇒ the source of fusion’s technology & materials challenge.
Other fuel cycles are possible, but more challenging, e.g.

**D-D (³He) Fusion**

\[ 6D \rightarrow 2(4\text{He}) + 3H + e^- + n + (41.5 \text{ MeV plasma}) + (2.45 \text{ MeV shield}) \]

- Significantly reduced fast neutron flux!! Most energy to plasma and then first wall. **Simplifies fusion component technologies.**
- Next easiest fusion fuel cycle, but requires confinement \(\sim25\) times better than D-T(Li) and \(T\) extraction from plasma (i.e. only MFE).
- Equally challenging, but exciting, D-D options exist for IFE.

Thursday, February 3, 2011
Self-Sustained Fusion Burn

- $\tau_E$ is plasma energy confinement time
- Lawson’s condition:
  $$\eta(P_{fus} + P_{aux}) > P_{aux}$$
- $Q = \frac{P_{fus}}{P_{aux}}$
- Ignition: $Q \rightarrow \infty$, or ...

$$\frac{W_p}{\tau_E} + P_{rad} = \text{(Charged Particle Fusion Power)}$$

Neutrons escape and heat surrounding blanket
Parameters of importance in characterizing the fusion plasma are the energy confinement time ($\tau_E$), which equals the stored energy in the plasma ($W$) divided by the heat ($P$) leaving the plasma (excluding the neutrons),

$$\tau_E = \frac{W}{P}(s) \quad (2.1)$$

and beta ($\beta$), which is the ratio of the kinetic pressure ($n_e k T_e + n_i k T_i + \Sigma n_x k T_x$) of the plasma divided by the magnetic pressure,

$$\beta = \frac{(\text{pressure} \times 100)}{(B^2/2\mu_0)} \% . \quad (2.2)$$
Basics: Fusion Reactivity

The D-T-Li fuel cycle is the most attractive, because a.

\[ ^6\text{Li} + n \rightarrow ^4\text{He} + T + 4.80 \text{ MeV} \]  \hspace{1cm} (2.3)

\[ ^7\text{Li} + n = ^4\text{He} + T + n - 2.47 \text{ MeV} \]

**FIG. 3.** Maximum charged-particle power density release vs. temperature for the principal fusion fuels in thermalized plasmas at \( n_e = 10^{20} \text{ m}^{-3} \) and \( n_i/n_j = Z_j/Z_i \). Power output scales as the square of the electron density (McNally, 1982).

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TABLE IV. Nuclear fusion reactions of greatest relevance to magnetic fusion. The final two quantities in each line refer to the total nuclear energy release in one reaction and to the energy release in the form of charged particles, respectively.

<table>
<thead>
<tr>
<th>Reaction</th>
<th>Charged particle (MeV)</th>
<th>Total (MeV)</th>
<th>Total equivalent energy (kWh/g)</th>
</tr>
</thead>
<tbody>
<tr>
<td>(1) D+T \rightarrow ^4\text{He}(3.52 \text{ MeV})+n (14.06 \text{ MeV})</td>
<td>3.52</td>
<td>17.58</td>
<td>94,000</td>
</tr>
<tr>
<td>(2a) D+D \rightarrow ^3\text{He}(0.82 \text{ MeV})+n (2.45 \text{ MeV})</td>
<td>0.82</td>
<td>3.27</td>
<td>22,000</td>
</tr>
<tr>
<td>(2b) D+D \rightarrow T(1.01 \text{ MeV})+p (3.03 \text{ MeV})</td>
<td>4.04</td>
<td>4.04</td>
<td>27,000</td>
</tr>
<tr>
<td>(3) D+^3\text{He} \rightarrow ^4\text{He}(3.67 \text{ MeV})+p (14.67 \text{ MeV})</td>
<td>18.34</td>
<td>18.34</td>
<td>98,000</td>
</tr>
</tbody>
</table>
The power balance may be written separately for each species. For the electrons, a simplified power balance is
\[
\frac{\partial}{\partial t} \left[ \frac{3}{2} n_e T_e \right] = \frac{1}{r} \frac{\partial}{\partial r} \left[ n_e e T_e \right] - n_e X_e \frac{e \partial T_e}{\partial t} + \frac{3}{2} D_e T_e \frac{\partial n}{\partial r} + p_{\Omega} + p_{el} - p_{LR} - p_{br} - p_{s} + p_{r} + p_{ae} + p_{ae} \quad (W m^{-3}).
\]
(2.11)

For the ions
\[
\frac{\partial}{\partial t} \left[ \frac{3}{2} n_i e T_i \right] = \frac{1}{r} \frac{\partial}{\partial r} \left[ n_i e T_i \right] - n_i X_i \frac{e \partial T_i}{\partial t} + \frac{3}{2} D_e T_i \frac{\partial n}{\partial r} + p_{el} - p_{cx} + p_{ai} + p_{ai} + p_{ai} \quad (W m^{-3}).
\]
(2.12)

Note that temperatures are given in electron volts. In

For a toroidal plasma, major radius \( R \), minor radius in the median plane \( a \), and ellipticity \( \kappa \),
\[
\tau_E = \frac{3}{2} \pi R a^2 \kappa \frac{e \langle n_e T_e + n_i T_i \rangle}{P} \quad (s).
\]
(2.22)

For a plasma sustained by the fusion alpha power, and \( T_{10C} < T_{10} < 2T_{10C} \), a parabolic temperature profile \((\alpha_T = 1)\) and square-root parabolic density profile \((\alpha_T = 0.5)\), from Eq. (2.7) we have \( T_{10C} = 0.75 \) and
\[
P_a = 4.9 \times 10^{-42} \langle n_{DT} T_i^2 \rangle^2 R a^2 \kappa \quad (W),
\]
(2.23)

where \( n_{DT} \) is the density of deuterium plus tritium ions. Substituting for \( P_a \) in Eq. (2.22), with \( T_e = T_i = T \), leads to a requirement for a self-sustaining pure D-T plasma \((n_e = n_i)\)
\[
\langle n_{DT} T_i \rangle \tau_E \approx 1.93 \times 10^{24} \quad (m^{-3} \text{ eV s}).
\]
(2.24)

For example, if \( \langle n_{DT} T_i \rangle = 1.5 \times 10^{20} \text{ m}^{-3} \times 10 \text{ keV} \), we require \( \tau_E = 1.3 \text{ s} \).
The line radiation is given by (Jensen et al., 1977)

\[ p_{LR} = \sum_z n_e n_z f(z)(W \text{ m}^{-3}) \]  \hspace{1cm} (2.14)

This is the dominant radiation term in present-day tokamaks (typically 20–40% of the power is radiated), and it is particularly important at the plasma edge. As shown in Fig. 5, \( f(z) \) is a strongly increasing function of \( Z \) as the impurities become more massive. Consequently small amounts of heavy materials such as molybdenum and tungsten can have a disproportionately large effect.

\[ f(z) = \frac{p}{n_{\text{e}z}} \text{ (W m}^{-3}) \]

FIG. 5. Line radiation factor \( f(z) \) as a function of electron temperature for representative impurities (Jensen et al., 1977).
Simple Fusion Power Conditions

\[ f_\alpha P_{fus} + P_{aux} = \frac{W}{\tau_E} + P_{brem} + P_{rad} \]

\[ Q \equiv \frac{P_{fus}}{P_{aux}} \]

\[ P_{fus} \propto P_{brem} \propto P_{rad} \propto n^2 \]

\[ nT\tau_E = \frac{n^2V}{(f_\alpha + 1/Q) P_{fus} - P_{brem} - P_{rad}} \]

a function of T only
Simple Fusion Power Conditions

\[ nT\tau_E = 6.3 \times 10^{-6} \left( \frac{\beta}{\chi} \right) \times B^2 a^2 \]

\[ nT\tau_E = \frac{n^2 V}{(f_\alpha + 1/Q) P_{fus} - P_{brem} - P_{rad}} \]

**TABLE XI. Representative modern tokamak parameters.**

<table>
<thead>
<tr>
<th>Tokamak</th>
<th>Tore Asdex</th>
<th>TFTR</th>
<th>JET</th>
<th>Supra</th>
<th>U</th>
<th>T-15</th>
<th>JT-60U</th>
<th>TRIAM</th>
<th>TPX</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius ( R ) (m)</td>
<td>1.65</td>
<td>2.45</td>
<td>3.10</td>
<td>2.25</td>
<td>1.65</td>
<td>2.43</td>
<td>3.40</td>
<td>0.80</td>
<td>2.25</td>
</tr>
<tr>
<td>Aspect ratio ( R/a )</td>
<td>2.75</td>
<td>2.90</td>
<td>2.82</td>
<td>3.75</td>
<td>3.3</td>
<td>3.5</td>
<td>4.0</td>
<td>5.0</td>
<td>4.5</td>
</tr>
<tr>
<td>Ellipticity ( K )</td>
<td>-2.0</td>
<td>1.0</td>
<td>1.8</td>
<td>1.0</td>
<td>1.6</td>
<td>1.0</td>
<td>1.6</td>
<td>1.5</td>
<td>2.0</td>
</tr>
<tr>
<td>Current ( I ) (MA)</td>
<td>2.0</td>
<td>3.0</td>
<td>6.0</td>
<td>2.0</td>
<td>2.0</td>
<td>1.4 (2.0)</td>
<td>6.0</td>
<td>2.0</td>
<td></td>
</tr>
<tr>
<td>Magnetic field ( B_{TFJ} ) (T)</td>
<td>2.0</td>
<td>5.2</td>
<td>3.4</td>
<td>4.5</td>
<td>4.0</td>
<td>3.5 (5.0)</td>
<td>4.2</td>
<td>8.0</td>
<td>4.0</td>
</tr>
<tr>
<td>Pulse length (s)</td>
<td>10+</td>
<td>2+</td>
<td>20+</td>
<td>;5600</td>
<td></td>
<td>10+</td>
<td>1.5+</td>
<td>20+</td>
<td>00</td>
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<td>No</td>
<td>Yes</td>
<td>No</td>
<td>Yes</td>
<td>No</td>
<td>Yes</td>
<td>Yes</td>
<td>Yes</td>
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<tr>
<td>Coil type</td>
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<td>Cu</td>
<td>Cu</td>
<td>SIC</td>
<td>Cu</td>
<td>SIC</td>
<td>Cu</td>
<td>SIC</td>
<td>SIC</td>
</tr>
</tbody>
</table>

\( \text{Cu} = \text{copper}; \quad \text{SIC} = \text{superconducting.} \)

**FIG. 31. Performance of tokamaks, JET Team, 1992.**

Confinement

5.3. Global scaling

Global scaling expressions for the energy confinement time ($\tau_E$), or the stored energy ($W$), are powerful tools for predicting the confinement performance of burning plasmas. These expressions are described using engineering parameters, such as the major radius ($R$), minor radius ($a$) or inverse aspect ratio ($\varepsilon = a/R$), elongation ($\kappa$), toroidal magnetic field ($B_t$), plasma current ($I_p$), electron density ($n_e$), heating power ($P$) or loss power ($P_L \equiv P - dW/dt$) and ion mass number ($M$).

One of the most reliable scaling expressions since 1998 for the ELMy H-mode thermal energy confinement time ($\tau_{th}$) is the so-called IPB98($y,2$) scaling [2]:

$$\tau_{th,98y2} = 0.0562I_p^{0.93}B_t^{0.15}n_e^{0.41}P_L^{-0.69}R^{1.97}\varepsilon^{0.58}\kappa_a^{0.78}M^{0.19}$$

(30)

(in s, MA, T, $10^{19}$ m$^{-3}$, MW, m). The effective elongation is defined as $\kappa_a = S_e/\pi a^2$, where $S_e$ is the plasma cross-sectional area. The interval estimation of $\tau_{th}$ in the ITER FDR with the use of such scaling expressions was studied in detail [2, 701]. Later estimation for the present ITER design using the extended database ITERH.DB3 showed a smaller interval of a 95% log-linear uncertainty (+14%/-13%) than that for ITER FDR (+25%/-20%) [704].

Figure 92. $HH$ factor ($\equiv \tau_{th}/\tau_{th,98y2}$) versus $n/n_G$. Reprinted with permission from [698].

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*Nucl. Fusion* 47 (2007) S18–S127
Simple Fusion Power Conditions

\[ nT\tau_E = 6.3 \times 10^{-6} \left( \frac{\beta}{\chi} \right) \times B^2 a^2 \]

\( \beta/\chi_\perp \equiv \beta 2\tau_E/a^2 \)

Turbulence Suppression & Shape Optimization:
Lazarus, Navratil, et al. PRL, 1996

HBT-EP:
- \( P_{aux} = 14 \text{ kA} \times 7 \text{ V} \)
- \( W = 106 \text{ kW} \)
- \( \tau_E = 1 \text{ msec} \)
- \( \beta = 0.004 \)
- \( a = 0.14 \text{ m} \)
- \( Ba = 0.045 \text{ T} \cdot \text{m} \)
- \( (1/312 \text{ smaller than ITER}) \)
- \( \beta/\chi = 0.0004 \)
- \( T = 50 \text{ eV} \)

ITER

Turbulence Suppression:
MST, PRL, 1997
RFX, PRL, 1999

Sheffield Confinement Parameter

\begin{align*}
&0.001 \quad 0.01 \quad 0.1 \quad 1 \quad nT\tau_E = 6.3 \times 10^{-6} \left( \frac{\beta}{\chi} \right) \times B^2 a^2 \quad (\text{s/m}^2) \\
&0.5 \quad 1 \quad 5 \quad 10 \quad 50 \quad T_i \text{ (keV)}
\end{align*}
Summary: Fusion Power

- $T \approx 9$ keV
- $n \approx 10^{20}$ m$^{-3}$
- $\beta = 0.025$ (with $\beta_N = 1.8$)
- $B \approx 5.3$ T
- $\epsilon = a/R = 0.32$ (with $\kappa = b/a = 1.7$)
- $a = 2$ m ($aB = 14$; with $I_p = 15$ MA and $q = 3.0$)
- $\tau_E = 3.7$ s (with $\beta/\chi \approx 0.027$)
- $Q \geq 10$
Summary: ITER's Parameters

\[ P_\alpha + P_{aux} + P_{oh} = 123 \text{ MW} \]

\[ P_{rad} = P_{brem} + P_{cyc} + P_{line} = 48 \text{ MW} \]

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<td>m(^3)</td>
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<tr>
<td>Surface</td>
<td>m(^2)</td>
<td>678</td>
<td>678</td>
</tr>
<tr>
<td>Sep.length</td>
<td>m</td>
<td>18.4</td>
<td>18.4</td>
</tr>
<tr>
<td>S(_{\text{cross-sect.}})</td>
<td>m(^2)</td>
<td>21.9</td>
<td>21.9</td>
</tr>
<tr>
<td>B(_l)</td>
<td>T</td>
<td>5.3</td>
<td>5.3</td>
</tr>
<tr>
<td>I(_P)</td>
<td>MA</td>
<td>15.0</td>
<td>17.4</td>
</tr>
<tr>
<td>(\kappa_x / \delta_x)</td>
<td></td>
<td>1.86 / 0.5</td>
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<td>(\kappa_{95} / \delta_{95})</td>
<td></td>
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<tr>
<td>(l_{(3)})</td>
<td>mV</td>
<td>0.86</td>
<td>0.78</td>
</tr>
<tr>
<td>(V_{\text{loop}})</td>
<td>mV</td>
<td>89</td>
<td>98</td>
</tr>
<tr>
<td>(q_{95})</td>
<td></td>
<td>3.0</td>
<td>2.7</td>
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<tr>
<td>(\beta_N)</td>
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<td>1.77</td>
<td>1.93</td>
</tr>
<tr>
<td>(&lt;n_n&gt;)</td>
<td>(10^{19} \text{ m}^{-3})</td>
<td>10.14</td>
<td>11.56</td>
</tr>
<tr>
<td>n/(n_{GW})</td>
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<td>0.85</td>
<td>0.84</td>
</tr>
<tr>
<td>(&lt;T_e&gt;)</td>
<td>keV</td>
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<td>9.1</td>
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<tr>
<td>(&lt;T_i&gt;)</td>
<td>keV</td>
<td>8.9</td>
<td>9.9</td>
</tr>
<tr>
<td>(&lt;\beta_T&gt;)</td>
<td>%</td>
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<td>3.2</td>
</tr>
<tr>
<td>(\beta_n)</td>
<td></td>
<td>0.67</td>
<td>0.62</td>
</tr>
<tr>
<td>(P_\alpha)</td>
<td>MW</td>
<td>82</td>
<td>120</td>
</tr>
</tbody>
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<tbody>
<tr>
<td>(P_{aux})</td>
<td>MW</td>
<td>40</td>
<td>23</td>
</tr>
<tr>
<td>(P_{ohm})</td>
<td>MW</td>
<td>1.3</td>
<td>1.7</td>
</tr>
<tr>
<td>(P_{tot})</td>
<td>MW</td>
<td>123</td>
<td>144</td>
</tr>
<tr>
<td>(P_{brem})</td>
<td>MW</td>
<td>21</td>
<td>29</td>
</tr>
<tr>
<td>(P_{syn})</td>
<td>MW</td>
<td>8</td>
<td>10</td>
</tr>
<tr>
<td>(P_{line})</td>
<td>MW</td>
<td>19</td>
<td>20</td>
</tr>
<tr>
<td>(P_{rad})</td>
<td>MW</td>
<td>48</td>
<td>59</td>
</tr>
<tr>
<td>(P_{fus})</td>
<td>MW</td>
<td>410</td>
<td>600</td>
</tr>
<tr>
<td>(P_{sep}/P_{LH})</td>
<td>MW/ MW</td>
<td>75/48</td>
<td>84/53</td>
</tr>
<tr>
<td>Q</td>
<td></td>
<td>10</td>
<td>24</td>
</tr>
<tr>
<td>(\tau_{E}, S)</td>
<td></td>
<td>3.7</td>
<td>4.1</td>
</tr>
<tr>
<td>(W_{th})</td>
<td>MJ</td>
<td>325</td>
<td>408</td>
</tr>
<tr>
<td>(W_{fast})</td>
<td>MJ</td>
<td>25</td>
<td>33</td>
</tr>
<tr>
<td>(H_{H,IPB98(v,2)})</td>
<td></td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>(\tau_{\alpha}/\tau_E)</td>
<td></td>
<td>5.0</td>
<td>5.0</td>
</tr>
<tr>
<td>(Z_{\text{eff}})</td>
<td></td>
<td>1.65</td>
<td>1.69</td>
</tr>
<tr>
<td>(f_{\text{He, axis}})</td>
<td>%</td>
<td>4.1</td>
<td>5.9</td>
</tr>
<tr>
<td>(f_{\text{Be, axis}})</td>
<td>%</td>
<td>2.0</td>
<td>2.0</td>
</tr>
<tr>
<td>(f_{\text{C, axis}})</td>
<td>%</td>
<td>0.0</td>
<td>0.0</td>
</tr>
<tr>
<td>(f_{\text{Ar, axis}})</td>
<td>%</td>
<td>0.12</td>
<td>0.11</td>
</tr>
</tbody>
</table>

\[ \frac{W}{\tau_E} \sim 88 \text{ MW} \]

Performance calculations using the agreed physics guidelines yield a substantial operating window for \(Q \geq 10\) inductive operation for the selected parameter set.
Density, Beta, H-Mode Limits

**Figure 1.** The operating space is shown on the basis of calculations using the HELIOS code for the baseline 15 MA, 5.3 T ELMy H-mode scenario. For reference, the baseline heating power is 73 MW. At an operating density of 0.85 of the Greenwald limit, the projected $Q$ is 10 with 40 MW of heating power and $\tau_E = 3.8$ s. The accessible operating regime in white is bounded by the estimated power required to achieve an H-mode, the Greenwald density and the available auxiliary heating power.

**Figure 2.** Operating scenario assuming a 10% reduction in toroidal field (4.77 T) and plasma current (13.5 MA) relative to figure 1. For the nominal operating point at 0.85 of the Greenwald density, $Q \sim 6$ and $\tau_E = 3.3$ s.
Density Limit

4.1.3. Understanding of density-limiting processes. There are several density limits in tokamaks [1,431]. Two of them, i.e. the H-mode density limit associated with a back transition from H- to L-mode and the ultimate L-mode density limit related to plasma disruption are the most important for reactor like devices. The figure of merit for the L-mode density limit is the Greenwald density [1,431],

\[ n_G = \frac{I_p}{\pi a^2} \approx 1.59 g \frac{B_t}{q_{95} R} \left(10^{20} \text{ m}^{-3}\right), \text{ MA, T, m}, \] (12)

where \( g = q_{95}/q_{cyl} \) is the plasma shaping factor with \( q_{cyl} = 5a^2B_t/(RI_p) \). Typically, at operation in the Type I ELMy H-mode with gas puff fuelling, an increase in density above some limit leads to a transition from Type I to Type III ELMs accompanied with reduction of the stored plasma energy by 15–40% [503]. A further increase in the gas fuelling rate leads to a back transition to the L-mode, correlated with complete divertor detachment and/or divertor/X-point MARFE formation [1]. At even higher fuelling rates, the L-mode density limit disruption occurs, terminating the discharge.
Stability of high beta tokamak plasmas*

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Stability at high beta (the ratio of plasma pressure to magnetic field pressure) is an important requirement for a compact, economically attractive fusion reactor. In large tokamak experiments, where the best performance is achieved rather than by energy transport. The past decade has seen major advances in our understanding of the stability of high beta tokamak plasmas, as well as in the theory of beta. Ideal magnetohydrodynamic (MHD) theory has been remarkably successful in predicting stability limits, and the scaling of maximum stable beta  seen in many experiments,21,48,55-62 the pressure gradient reaches values up to $6I/aB$ to be achieved experimentally. Recent modeling have shown that the beta limit depends on the pressure profiles, and modification of the current density to create centrally peaked profiles in DIII-D (Ref. 49). Probes 170° Apart Toroidally

FIG. 4. Time evolution of an ideal $n=1$ kink mode disruption at high beta ($\beta_N=3.5$) in DIII-D (Ref. 49).
**β Limit**

**FIG. 2.** Comparison of experimental beta limits to Troyon scaling, showing the operational envelopes for several tokamaks.

**FIG. 3.** Operational limits for HBT. Open circles indicate stable discharges; closed circles indicate transient, unstable cases with a growing \( n=1 \) kink instability. The \( q=2 \) limit and \( \beta_N=2.8 \) (Troyon) limit are shown (Ref. 46).

β and Density Limit

\[ \langle \beta \rangle \sim 10^{-2} \beta_N \left( \frac{I \text{ (MA)}}{aB} \right) \]

\[ \langle n \rangle \sim 10^{20} \text{ m}^{-3} n_G \left( \frac{I \text{ (MA)}}{\pi a^2} \right) \]

\[ \langle T \rangle \propto \frac{\langle \beta \rangle}{\langle n \rangle} = 0.4 \text{ keV} (aB) \frac{\beta_N}{n_G} \]

\approx 10 \text{ keV (ITER) and 30 eV (HBT-EP)}
Neoclassical Tearing Modes (NTMs)

Figure 2. DIII–D discharges with (114504, dotted lines) and without (114494, solid lines) ECCD suppression of an \( m/n = 3/2 \) neoclassical tearing mode. (a) Neutral beam power, (b) \( \beta_N \), (c) \( n = 2 \) Mirnov \( |B_i| \), (d) \( n = 1 \) Mirnov \( |B_i| \). The degradation in energy confinement due to the NTM from 3/2 and 2/1 NTMs can be seen in the effect on \( \beta_N \).

Figure 3. Sketch of the time evolution of the island growth rate as given by equation (6) at the onset of the NTM when the critical seed island size \( (W_{\text{crit}}) \) is exceeded and an NTM forms at \( \beta_{p,\text{onset}} \). A slow decrease in beta from \( \beta_{p,\text{onset}} \) to \( \beta_{p,\text{marg}} \) (when max(dW/dt) = 0) is assumed, as in power ramp-down experiments, such that dW/dt \( \approx 0 \) (reproduced from [54] ‘Marginal \( \beta \)-limit for neoclassical tearing modes in JET H-mode discharges’).

\[
\frac{\tau_R}{r_s} \frac{dW}{dt} = r_s \Delta'(W) + r_s \beta_p (\Delta'_{BS} - \Delta'_{GGJ} - \Delta'_{pol}) + r_s \Delta'_{CD}.
\]

(6)
ITER (Fusion) Requires Large Size (aB) and High Power

- Very strong magnets and large forces
- Very high power flux to limiters
4.1.1 Toroidal Field Coils

The toroidal magnetic field value on the plasma axis is 5.3T, which leads to a maximum field on the conductor ≤ 12 T. Because of this high field value, Nb3Sn is used as superconducting material, cooled at 4.5K by a flow of supercritical helium at ~ 0.6 MPa. The total magnetic energy in the toroidal field is around 40 GHz, the confinement of which leads to significant forces on each coil restrained by a thick steel case to resist circumferential tension (≈ 100 MN) and by constructing a vault with the inboard legs of all 18 coils (the large centripetal forces are due to the I/R variation of the toroidal field). The compressive stress levels inside this vault are large, and therefore the side surfaces of each coil should match one another as perfectly as possible.

The coils are connected together (Figure 4.1.2) by bolted structures, and by two compression rings made of unidirectional glass fibres, that provide an initial inward radial force on each coil (2 x 30 MN).

This very robust assembly is provided mainly to resist the toroidal forces induced by interaction of the TF coil current with the transverse poloidal field from plasma and poloidal field coils. These forces produce a distribution of torque around the TF coil proportional to the magnetic flux crossing unit length (the net torque is thus 0). These local forces are pushed, and therefore mechanical fatigue is a concern for the highly stressed structural steel of the coils. These forces, due to the highly shaped plasma, are largest across the inboard coil legs (in particular at their lower curved region) where they are resisted by the friction between coil sides (under high compression) and by specific keys.

kip = 1,000 lbs force
0.1 x GWB

22 million pounds tension
4.2.5 In-vessel Component Water Cooling

Each divertor cassette is separately cooled by water, with feeder pipes connecting to the manifold outside the vessel and cryostat. Groups of two or three blanket modules are similarly fed by separate pipes installed on the plasma side of the inner shell of the vacuum vessel. This arrangement leads to handling a large number of small size pipes, but (e.g. by “spiking” specific coolant channels with tracer elements) allows the identification of possible modules or cassettes leaking water, from tests outside the cryostat, a crucial procedure to be able to rapidly localise the leaks in vacuum.
Thin SOL & Elms

Chapter 4: Power and particle control

**Figure 1.** Comparison of ELM-averaged IR, LP (inter-ELM) and TC (histograms, ELM-averaged) derived heat flux profiles on the JET outer divertor target for a plasma with 16 MW NBI, 2.5 MA/2.4 T and ion $B \times \nabla B$ towards the X-point. The scale of the electron heat flux (LP) is four times smaller in the high power case. Also shown are the profiles obtained for a 12 MW Type I H-mode (same field and current), using the shot-by-shot TC method (——). For comparison the poloidal gyro-radius at the outer mid-plane is shown for three values of the ion energy [21].

**Figure 8.** Temporal evolution of the divertor surface temperature, deposited ELM power and energy onto the JET outboard divertor target for a typical Type I ELM [162].
Summary

- Fusion cross-sections determine the scale (Ba ~ 14 T·m) of burning plasma experiment
- $\beta/\chi \sim 0.027 \text{ s/m}^2$ is based upon existing data
- $\beta_N$ and $n_G$ are conservative limits
- ITER’s size scale necessitates state-of-the-art engineering and technology.