ITER:opportunities of Burning Plasma Studies

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ITER Joint Central Team

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1. How will particle and power exhaust be handled? How well will proposed components withstand the effects of plasma disruptions and other related "off-normal" operational events? (See Janeschitz’s view graphs)

1.a Particle Exhaust
Divertor with long legs (> 1m) and large pumps (200 Pam³/s, > 50m³/s for He)

\[ F_{\text{core}} < \frac{N}{\tau_E} \sim 2.5 \times 10^{22}/s \sim 50\text{Pam}^3/s, \quad F_{\text{divertor}} \sim 2 \times 10^{24}/s \sim 4000\text{Pam}^3/s \]

Detachment is not a necessary condition.

1.b Heat Exhaust
Plasma flow to divertor target < 60 MW/6m²
The present design 20 MW/m², CFC or W

Very high radiation cooling and detachment are not necessary conditions but will have to be studies for reactor plasmas.

1.c Divertor target material/Disruptions CFC (or W)
Early phase CFC because of its compatibility with disruptions
Later phase W because of its longer life time for normal erosion

High fluence Test : long pulse q₉₅ >3.5 operations with small ELMs
2. What types of heating and current drive are planned and what are the prospects for investigating "steady-state" plasma operation on the relevant plasma time scales?

<table>
<thead>
<tr>
<th>Heating and Current Drive System</th>
<th>Startup</th>
<th>Scenario 1</th>
<th>Scenario 2</th>
<th>Scenario 3</th>
<th>Scenario 4</th>
</tr>
</thead>
<tbody>
<tr>
<td>NB (1 MeV, D/H, Variable in. Angle)</td>
<td>33</td>
<td>33</td>
<td>50</td>
<td>50</td>
<td>50</td>
</tr>
<tr>
<td>IC (35-62 MHz)</td>
<td>20</td>
<td>40</td>
<td>20</td>
<td>40</td>
<td>20</td>
</tr>
<tr>
<td>EC (170 GHz, Steerable mirror)</td>
<td>20</td>
<td>40</td>
<td>40</td>
<td>40</td>
<td>20</td>
</tr>
<tr>
<td>LH (~5GHz)</td>
<td>0</td>
<td>20</td>
<td>20</td>
<td>0</td>
<td>40</td>
</tr>
<tr>
<td>Total Installed (MW)</td>
<td>73</td>
<td>133</td>
<td>130</td>
<td>130</td>
<td>130</td>
</tr>
</tbody>
</table>

Remarks: The total heating and current drive power ≤ 110 MW

A deep fuelling improves significantly the steady state operation.

- Increase boot strap current and reduce the requirement of active current drive in the outer region as well as improve confinement property.

100 s of burn duration is necessary to study inductive burn.

2000 s is necessary to achieve steady state of AT modes from conventional ones.

By optimizing current ramp-up, steady-state of AT can be achieved in 200 s.
3. What is the transport or confinement basis and MHD stability basis for reaching the burning regime and what are the uncertainties in reaching the projected operating regimes? How much margin exists for physics or hardware performance contingencies?

3.a Inductive operation for high Q
   ELMy H-mode: Empirical scaling
   \[ \beta_N < 2. \] If NTM, ECCD (20-40 MW)
   Margin: Plasma density, High field side pellet injection,
   Conservative assumptions \((\chi_i/\chi_e, n(r), P_{LH})\)
   and/or, Higher plasma current

3.b Steady state operation
   \( Q \approx 2 \) ELMy H-mode with \( H_H = 1 \)
   \( Q > 5 \) Advanced tokamak regimes (see 5)
4. What physics program is envisioned and how will the burning plasma scientific issues be addressed? Will planned diagnostic capabilities be commensurate with science program needs? Will the pulse rate and number and lifetime and provisions for the supply of tritium and maintenance, replacement and/or upgrade of activated components be commensurate with the proposed science program?

4.a High Q burning plasma

The flexibility of ITER will allow research in a large operation space. 

\( P_{\text{fusion}}, Q, n, \beta, \text{pulse length, } I_p \text{ ---} \)

(Confirm predictable operation → Explore frontier)

4.b Diagnostics (See Costley’s viewgraphs)

Large access ports and remote handling capability

4.c Pulse rate/number, Tritium, Maintenance and Upgrade

\( \leq 2 \text{ pulses/hr.} \quad > 30,000 \text{ full performance shots} \)

External tritium source (If necessary, tritium production in the later phase)

Maintenance: full remote for in-vessel components

Upgrade: in-vessel components/auxiliaries
5. What operational and/or hardware flexibility is incorporated into the design? What capability exists for studying burning plasma AT ("advanced tokamak") regimes? What are the scientific and technology issues involved in such "advanced" operation and how will they be addressed?

5.a The flexibility of ITER will allow research in a large operation space

\( P_{\text{fusion}}, Q, n, \beta, \text{pulse length, } I_p \ldots \)

(Confirm predictable operation → Explore frontier)

- **Inductive operations**
  \[ 150 \rightarrow 700 \text{ MW}, n/n_G = 0.5\rightarrow 1, \beta_N = 1.2\rightarrow 2.4, Q = 5\rightarrow 10 \rightarrow 20 \rightarrow \infty \]

- **Hybrid operations**
  > 1000 s/500 MW/Q = 5 with reasonable parameters for blanket test (0.77 MW/m²)

- Research of fully non-inductive driven operation aiming at \( Q = 5 \)
  (higher \( \beta \)/higher confinement, methods included in ITER)

5.b The high repetition rate and the large number of pulses give flexibility in experimental operation

5.c Full remote maintenance of in-vessel components and large size of ports (1.8 m x 2.2 m) give flexibility in hardware.
5.d Advanced Tokamak Regimes

i) Tools involved
   Relatively close conducting wall ($r_p \sim 2m$, $d_{p,W} \sim 0.6m$)
   Saddle coils for stabilizing RWM
   Current drive and heating (NB/IC/EC/LH, 130/110 MW)
   Large plasma
   High field side pellet injectors

ii) Scientific issues
   General: Transports, mhd etc.
   Specific: Stabilization of high $\beta_N$ and low $l_i$ plasmas, and Deep fuelling
The technical requirements for the new ITER (ITER-FEAT)

1) Demonstrate inductively-driven plasmas at $Q_{\text{\geq}} 10$,
2) Aim at demonstrating steady-state at $Q_{\text{\geq}} 5$
3) Do not preclude ignition.
4) Demonstrate availability and integration of essential fusion technologies, and
5) Test components for a future reactor including blankets
   ($> 0.5 \text{ MW/m}^2, > 0.3 \text{ MW}\cdot\text{a}/\text{m}^2$.)

ITER is planned to be the first fusion experimental reactor.

- Flexibility is required to
  1) cope with uncertainties,
  2) study/optimize burning plasma for various objectives, and
  3) introduce advanced features
- Involvement of the world-wide fusion community is essential to
  1) use ITER efficiently and
  2) promote scientific competition among the Parties
Research on Burning Plasma

1. Inductive Operation
   High Q plasma \( Q \approx 5/10/20/50 \) ---- Based on standard ELMy H-mode
   Reduction of divertor heat load and erosion
      Radiation cooling, semi-detached, detached divertor operation modes with
      small ELMs by optimizing plasma, configuration, divertor, fuelling,
      impurity
   High density, Peaked density profile, Higher beta, Higher fusion power density,
   Higher fast \( \alpha \) pressure etc.
   Pulse reactor?

2. Long Pulse Operation for Blanket Tests
   \( \geq 1000 \) s, \( > 0.5 \) MW/m2
   Low divertor erosion for high fluence tests (Small ELM loss, W divertor target)

3. Advanced Tokamak Modes
   Steady state plasma
      Higher confinement/beta/density/bootstrap current and peaked density
      profile
   Interaction among
      burn, external H/CD, fuelling/pumping, impurity, transports etc.
   Steady State reactor?
ITER Poloidal Field Coils

Correction Coils
6x3, 100-150kA/coil
For Resistive Wall Mode
\(~10\text{G}/20\text{kA}\)
Upper launcher: poloidal steering = -60 ~ -70°
   toroidal angle = 24°
Equatorial launcher: toroidal steering = 20 - 45°
Equatorial port: standardized port plug for IC/EC/LH
Neutral Beam Injection for ITER

(1 MeV, 16.5 MW/Port)
Initial Installation 33 MW, Upgrade 50 MW

Beam Driven Current Profile

\[ P_{NB} = 17 \text{ MW} \]
\[ \gamma \approx 0.4 \times 10^{20} \text{ A/Wm}^2 \]

\[ j_{bd} \text{ [MA/m]} \]

\[ \rho \]

On-axis beam Tilting angle: 2.3°

Equatorial port

Intercoil structure

On-axis beam

\[ \Delta Z = 376 \]

Off-axis beam

\[ \Delta Z = 950 \]

Blanket opening

Beam source at 1341 from Machine Center Line

Mezzanine floor for upper port access

NB Elevation Layout

Tangent point at \( R_{tan} = 5.28 \text{ m} \)
## ITER Machine Capability

<table>
<thead>
<tr>
<th></th>
<th>Reference Performance</th>
<th>Flexibility</th>
</tr>
</thead>
<tbody>
<tr>
<td>$I_p$ (MA)</td>
<td>15 (flat top 400-500 s)</td>
<td>17 (flat top 100-200 s)</td>
</tr>
<tr>
<td>Fusion Power (MW)</td>
<td>500 (~2000s)</td>
<td>700 (100-200s)</td>
</tr>
<tr>
<td>$\kappa_x/\delta_x$</td>
<td>1.85/0.49</td>
<td>2.0/0.55($a=1.85$m)</td>
</tr>
<tr>
<td>Pumping</td>
<td>200 Pam$^3$/s</td>
<td>higher in shorter pulse</td>
</tr>
</tbody>
</table>

<table>
<thead>
<tr>
<th></th>
<th>Initial</th>
<th>Possible Upgrade</th>
</tr>
</thead>
<tbody>
<tr>
<td>NB (MW)</td>
<td>33</td>
<td>50</td>
</tr>
<tr>
<td>RF (MW)</td>
<td>40</td>
<td>80</td>
</tr>
<tr>
<td>ECCD for NT (MW)</td>
<td>(20)</td>
<td>(40)</td>
</tr>
<tr>
<td>Saddle coils for RWM</td>
<td>20KA/10G/2Hz</td>
<td>~50KA</td>
</tr>
</tbody>
</table>

**Divertor/Blanket**  
Exchangeable concept

**Large common ports**  
14 for blanket tests, RH, Diagnostics, H/CD
Standard Operations: ELMy H-mode

\[ P_{LH} = 2.84 M^{-1} B_T^{0.82} n_e^{0.58} R^{1.00} a^{0.81} \]

\[ \tau_{E,th}^{IPB98(y,2)} = 0.144 I_p^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_a^{0.78} \]

\[ \tau_E = H \tau_{E,th}^{IPB98(y,2)} \]

\( (s, MA, T, MW, 10^{20} m^{-3}, AMU, m \text{ and } \kappa_a = S_x / \pi a^2) \)
Conservative Assumptions in Standard Analysis

\( \zeta \) P Flat Density Profile

\( \zeta \) Q P = \( P_\alpha + P_{oh} + P_{aux} - (P_{brem} + P_{cycl} + P_{line}/3) \)

Radiation Loss \( \sim 30 \% \)

The confinement Data Base does not include this effect.

\( \tau_E \) is under estimated.

\( P_{LH} \) is estimated 30\% higher.

\( 3\bar{\Delta}\chi_i/\chi_e = 2 \)
Example:
Fusion Power 500MW  Q: 20-10
Alpha Heating 100MW  Additional Heating 25-50MW
Total Heating Power 125-150MW
Å Power required for L-H transition 50MW Å j Å
Radiation loss 50MW
Power in to Scrape-off-Layer 75-100MW
Power to Divertor Target 30-60MW Å < 60MW or Å 10MW Å m²Å
Å Maximum allowable heat load: 20MW Å m²Å
Detached plasma is not a necessary condition.
$I_p = 15 \text{ MA}, \quad H_H = 1.0, \quad \tau_{He^*/\tau_E} = 5, \quad \text{Divertor heat flux } \lesssim 10 \text{ MW/m}^2$

$n_G \left(10^{20}/\text{m}^3\right) = I_p (\text{MA}) / \pi a^2, \quad \beta_N = \beta (\%) / [I_p/aB_I]$
Lawson diagram of ITER with $n/n_G = 0.85$.

Hydrogen plasma is operated at 7.5 MA and 2.7 T because of difficulty of L-H transition. Data of the present machines are not ELMy H-mode.
Density is one of the most important parameters.

- $I_p=15$MA, $\tau_{\text{He}}^*/\tau_{\text{E}}\sim5$, $H_{\text{H98(y,2)}}=1.0$, flat density profile
- Argon impurity is seeded to limit the power to divertor region
A burn length of $\sim 60$ s is the minimum to study inductively driven plasma.

$I_p = 17$ MA, $\langle n_e \rangle = 1.1 \times 10^{20}/m^3$ ($n_e/n_G = 0.81$) and $P_{ADD} = 73$ MW (t = 10-13.5 s)
1.5D Simulation of Thermal Stability

(a) 17 MA, $n_e = 1.1 \times 10^{20} / m^3$

$$Q \sim \infty$$

(b) 16 MA, $n_e = 1.0 \times 10^{20} / m^3$

$$Q \sim 20$$

(c) 15 MA, $n_e = 1.0 \times 10^{20} / m^3$

$$Q \sim 10$$

(d) 12 MA, $n_e = 0.81 \times 10^{20} / m^3$

$$Q \sim 5$$

($\tau_{He}^*/\tau_E = 5, H_{H98(y,2)} \sim 1.0$ after H-mode transition)
<table>
<thead>
<tr>
<th>Ion Temperature (keV)</th>
<th>Electron Density (10^{20}/\text{m}^3)</th>
</tr>
</thead>
<tbody>
<tr>
<td>2.0</td>
<td>2.0</td>
</tr>
<tr>
<td>1.5</td>
<td>1.5</td>
</tr>
<tr>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>0.5</td>
<td>0.5</td>
</tr>
</tbody>
</table>

- Pf = 0.4 GW
- Pf = 0.8 GW

- \(\beta_N = 0\) (He = 1%)
- \(\beta_N = 3\) (He = 5%)
- \(Q = \infty\) (He = 1%)

- \(n_e/n_G = 1.0\)

**Thermal Instability with 17 MA**

Diagram showing the relationship between ion temperature, electron density, and plasma parameters, with various lines indicating different conditions and parameters such as Pf, \(\beta_N\), and Q. Points A, B, C, D, A', B', and C' are marked on the graph, indicating specific conditions or states.
Control of power excursion by impurity injection

$I_p = 17$ MA, $\tau_{\text{He}}^*/\tau_E = 3$, $H_{\text{H}}(y,2) = 1.0$ and 73 MW of heating power ($P_{\text{ADD}}$) is added from 10s to 13.7s: solid line - with argon (Ar) impurity seeding, dotted line - without impurity seeding
# Examples of ITER-like Discharges

(Example)

<table>
<thead>
<tr>
<th>JET</th>
<th>Pellets</th>
<th>Ar seeded</th>
<th>q(_{95})~3.4</th>
<th>q(_{95})~3</th>
<th>ITER 15 MA Q =10/20</th>
<th>ITER 16/17 MA Q =20/50</th>
</tr>
</thead>
<tbody>
<tr>
<td>(H_H)</td>
<td>1.0-0.8</td>
<td>1.0</td>
<td>0.98</td>
<td>1.0</td>
<td>1.0</td>
<td>1.0</td>
</tr>
<tr>
<td>(\beta_N)</td>
<td>1.8</td>
<td>1.78</td>
<td>2.17</td>
<td>1.8</td>
<td>1.7/1.75</td>
<td>1.7/1.8</td>
</tr>
<tr>
<td>(n_e/n_G)</td>
<td>1.2-1.0</td>
<td>0.9</td>
<td>0.92</td>
<td>0.95</td>
<td>0.85/0.95</td>
<td>0.8/0.85</td>
</tr>
<tr>
<td>(Z_{eff})</td>
<td>1.7</td>
<td>1.8</td>
<td>1.4</td>
<td>1.7</td>
<td>1.7/1.6</td>
<td>1.6/1.6</td>
</tr>
<tr>
<td>(\delta_X)</td>
<td>0.39</td>
<td>0.23</td>
<td>0.34</td>
<td>0.43</td>
<td>0.5</td>
<td>0.5</td>
</tr>
<tr>
<td>(q_{95})</td>
<td>2.8</td>
<td>2.8</td>
<td>3.4</td>
<td>3.1</td>
<td>3</td>
<td>2.85/2.69</td>
</tr>
</tbody>
</table>

Q = 10-20/400MW, Q = 50/500MW,
Ar seeded: ~0.1%,
Divertor heat load < 10 MW/m²
\[ H_H = 1 \text{Å} \]

Blanket Test: >1000s, 500 MW Å Test area Å 0.77 MW/m² Å

- **R/a = 6.35 m / 1.85 m, \( \beta_N \leq 2.5 \)**
- **R/a = 6.35 m / 1.85 m, \( \beta_N \leq 2.0 \)**
- **R/a = 6.20 m / 2.00 m, \( \beta_N \leq 2.5 \)**
- **R/a = 6.20 m / 2.00 m, \( \beta_N \leq 2.0 \)**
- **R/a = 6.20 m / 2.00 m, \( \beta_N \leq 2.0, n_e/n_G = 1.0 \)**

\[ n_e/n_G = 0.85 \]

\[ Q \]
# Long Pulse Operation

<table>
<thead>
<tr>
<th></th>
<th>Hybrid #7</th>
<th>Hybrid #2</th>
</tr>
</thead>
<tbody>
<tr>
<td><strong>$I_p$ (MA)</strong></td>
<td>13.3</td>
<td>14.4</td>
</tr>
<tr>
<td><strong>$q_{95}$</strong></td>
<td>3.5</td>
<td>3.2</td>
</tr>
<tr>
<td><strong>$\langle n_e \rangle$ (10^{20} \text{m}^{-3})</strong></td>
<td>0.90</td>
<td>1.0</td>
</tr>
<tr>
<td><strong>$n_e / n_G$</strong></td>
<td>0.85</td>
<td>0.85</td>
</tr>
<tr>
<td><strong>$\beta_n$</strong></td>
<td>1.9</td>
<td>2.2</td>
</tr>
<tr>
<td><strong>$P_{\text{FUS}}$ (MW)</strong></td>
<td>350</td>
<td>500</td>
</tr>
<tr>
<td><strong>$P_{\text{NB}}, P_{\text{RF}}$ (MW)</strong></td>
<td>73</td>
<td>60</td>
</tr>
<tr>
<td><strong>$Q = P_{\text{FUS}} / (P_{\text{NB}} + P_{\text{RF}})$</strong></td>
<td>4.8</td>
<td>5.0</td>
</tr>
<tr>
<td><strong>$\tau_E$ (s)</strong></td>
<td>2.62</td>
<td>2.40</td>
</tr>
<tr>
<td><strong>$f_{\text{He, axis / ave}}$ (%)</strong></td>
<td>2.9 / 2.2</td>
<td>3.9 / 2.7</td>
</tr>
<tr>
<td><strong>$Z_{\text{eff, ave}}$</strong></td>
<td>1.73</td>
<td>2.03</td>
</tr>
<tr>
<td><strong>$P_{\text{Separatrix}}$ (MW)</strong></td>
<td>100</td>
<td>129</td>
</tr>
<tr>
<td><strong>$I_{\text{CD}} / I_p$ (%)</strong></td>
<td>28</td>
<td>32</td>
</tr>
<tr>
<td><strong>$I_{\text{BS}} / I_p$ (%)</strong></td>
<td>18</td>
<td>20</td>
</tr>
<tr>
<td><strong>Burn time (s)</strong></td>
<td>1280</td>
<td>1220</td>
</tr>
<tr>
<td><strong>Shot # for 0.2 MWa/m²</strong></td>
<td>12800</td>
<td>9400</td>
</tr>
</tbody>
</table>

* Neutron fluence at test area = 0.28 MWa/m²
* Neutron flux at test area = 0.55 MW/m² at 350 MW, 0.78 MW/m² at 500 MW
* $R (m) / a (m) = 6.2 / 2.0$, $\kappa_{95} / \delta_{95} = 1.7 / 0.33$, $\tau_{\text{He}}^* / \tau_E = 5$, $HH(y,2) = 1.0$

** High triangularity gives $q_{95} = 3.7$ instead of 3.5.
Operation Parameters of ITER
15 MA, $H_{98(y,2)}=1.0$

<table>
<thead>
<tr>
<th></th>
<th>$Q = 20$</th>
<th>$Q = 10$</th>
</tr>
</thead>
<tbody>
<tr>
<td>$&lt;T_i&gt;$ (keV)</td>
<td>7.1</td>
<td>8.0</td>
</tr>
<tr>
<td>$&lt;T_e&gt;$ (keV)</td>
<td>7.7</td>
<td>8.8</td>
</tr>
<tr>
<td>$&lt;n_e&gt;$ ($10^{19}/m^3$)</td>
<td>11.3</td>
<td>10.1</td>
</tr>
<tr>
<td>$&lt;n_e&gt;/n_G$</td>
<td>0.95</td>
<td>0.85</td>
</tr>
<tr>
<td>$f_{He}$ (axis)</td>
<td>%</td>
<td>4.6</td>
</tr>
<tr>
<td>$P_{FUS}$ (MW)</td>
<td>400</td>
<td>400</td>
</tr>
<tr>
<td>$\beta_N$</td>
<td>%</td>
<td>2.4</td>
</tr>
<tr>
<td>$&lt;\beta_T&gt;$</td>
<td>%</td>
<td>2.4</td>
</tr>
<tr>
<td>$&lt;\beta_{f\alpha}&gt;$</td>
<td>%</td>
<td>0.13</td>
</tr>
<tr>
<td>$\beta_{f\alpha}$ (axis)</td>
<td>%</td>
<td>0.9</td>
</tr>
<tr>
<td>$n_{f\alpha}/n_e$ (axis)</td>
<td>%</td>
<td>0.5</td>
</tr>
<tr>
<td>$\tau_{s,f\alpha}$ (axis)</td>
<td>s</td>
<td>0.7</td>
</tr>
<tr>
<td>$\tau_{e,i}$ (axis)</td>
<td>s</td>
<td>0.8</td>
</tr>
<tr>
<td>$\tau_E$</td>
<td>s</td>
<td>4.5</td>
</tr>
<tr>
<td>$W_{thr}$ (MJ)</td>
<td>310</td>
<td>320</td>
</tr>
<tr>
<td>$W_{f\alpha}$ (MJ)</td>
<td>18</td>
<td>24</td>
</tr>
</tbody>
</table>

* $n_{f\alpha} \sim n_D n_T <\sigma v>_{DT} \times \tau_{s,f\alpha}$ ( $n_{f\alpha}/n_e$ has a weak dependence on $n_e$ )
Fast Alpha Particle Parameter in ITER (17MA)

High fast alpha particle pressure can be accessed only with better confinement
Onset of the first sawtooth crash
(a) 20 MW of RF heating power is added at $t = 77s$
(b) 20 MW of RF power is added at $t = 34s$ to reduce current penetration.
QÅ 5.4, I_p = 9.5 MA Steady State Operation

\[ P_{\text{FUS}} \div P_{\text{CD}} = 340 \text{ MW} \div 63 \text{ MW}, \]
\[ Z_{\text{eff}} \sim 1.9 \ (\text{He} / \text{Be} / \text{Ar} = 4\% / 2\% / 0.16\%), \ P_{\text{sep}} = 100 \text{ MW}, \ H_I = 1.45, \beta_N = 2.7 \]
\[ r = 0-0.5 \ (\text{NB:28 MW}), \ r = 0.5-0.8 \ (\text{EC and LH : 35MW}) \]
Non-inductive Operation with Internal Transport Barrier from a Conventional Operation

\[
\langle n_e \rangle = 0.67 \times 10^{20} / \text{m}^3 \ (\langle n_e \rangle / n_G = 0.8), \ HH = 1 \rightarrow 1.4, \ P_{\text{FUS}} / P_{\text{CD}} = 280 \text{ MW} / 80 \text{ MW} \ (Q = 3.5)
\]

- On axis (EC): 20 MW, \( \gamma_{20} (\text{EC}) = 0.15 \)
- Off axis (NB): 20 MW, \( \gamma_{20} (\text{NB}) = 0.18 \rightarrow 0.28 \)
- Far off axis (LH): 40 MW, \( \gamma_{20} (\text{LH}) = 0.3 \)
Evolution of plasma parameters for the WNS steady-state operational scenario. X-point formation corresponds to $t=15.7$ s, start of flat-top corresponds to $t=40$ s, start of burn corresponds to $t=40$ s.
Time response of fusion power to increase of fuelling rate ($S_{\text{FUEL}}$) for various $\tau_{\text{DT}}/\tau_{\text{E}}$ values. Here, $H_{\text{H98(y,2)}} = 1.0$, $P_{\text{NB}} = 33$ MW, $P_{\text{RF}} = 7$ MW and $\tau_{\text{He}}/\tau_{\text{E}} = 5$.

Evolution of plasma parameters as a reaction on the 25% testing decrease in the plasma gas-puff fuelling ($G_{n0}$) for the WNS steady-state scenario.
I_p = 10 MA, <n> ~ 7 x 10^{19} m^{-3} (0.8n_e), \beta_N \sim 3.2, P_{\text{fusion}} = 400-500 MW, a = 2m
If P = 100 MW, H_{\text{H}} \sim 1.6,
Bootstrap Current Fraction in ITER (PRETOR simulation)

\( I_{BS}/I_p \)

\( \beta_N \)

\( \beta_P \)

\( q_{95} \)

\( Q \)

\( P_{FUS} \)

\( P_{FUS} \)

\( n_e/n_G = 1.5, \quad H_{H98(y,2)} = 1.6, \quad f_{Be} = 2\%, \quad f_{Ar} = 0.12\% \)

\( /a = 6.35m/1.85m, \quad P_{NB}/P_{RF} = 17 \text{ MW}/40 \text{ MW} \)

* Typical Density Profile
D-III-D/H

ITER/H

a : 100m/s , b:300m/s,c:1000m/s, d: 3000m/s
D = h = 1 cm

By A. Polevoi (Kuteav/Parks/Strauss, ablation/cloud size/mass relocation)
Photograph sets for 10-mm pellets shot through 80-cm-radius curved guide tube.
(Upper 285 m/s, Lower 315 m/s)

Pellet Fuelling

Assumption: \(T_{eb} = 1 \text{ KeV}, n_e = 10^{20} \text{m}^{-3}, 10 \text{ mm pellet}\)
Low field side: 0.2 \(a\) at 1 km/s
High field side: (0.2a) at 0.3 km/s
(Simple extrapolation from ASDEX-U)
Model will have to be developed for high field side injection

100 Pam\(^3\)/s, 1 cm\(^3\)/s, 0.27 g/s or 5.5 x 10\(^{22}\)/s of tritium extraction has been achieved (\(n_{DT} \text{ VP} = 6.3 \times 10^{22}\) IN ITER-FEAT). A total of 36 g T\(_2\) and 28 g D-T runs.

Test data summary for 2.7-mm pellets shot through curved guide tubes of different radii

Range of Pellet Speed in ORNL Experiments

<table>
<thead>
<tr>
<th>Pellet Speed (m/s)</th>
<th>Range of Curvature (cm)</th>
</tr>
</thead>
<tbody>
<tr>
<td>285 m/s</td>
<td>0 - 100</td>
</tr>
<tr>
<td>315 m/s</td>
<td>100 - 1000</td>
</tr>
</tbody>
</table>

Intact Pellets Observed
Fractured Pellets Observed

Photograph sets for 10-mm pellets shot through 80-cm-radius curved guide tube.
(Upper 285 m/s, Lower 315 m/s)
### Preliminary Analysis with High Field Side Pellet Injection

<table>
<thead>
<tr>
<th>Parameter</th>
<th>Value</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major radius (m)</td>
<td>6.35</td>
</tr>
<tr>
<td>Minor radius (m)</td>
<td>1.85</td>
</tr>
<tr>
<td>Elongation (95% flux)</td>
<td>1.85</td>
</tr>
<tr>
<td>Plasma current (MA)</td>
<td>12.5</td>
</tr>
<tr>
<td>Toroidal field on axis (T)</td>
<td>5.18</td>
</tr>
<tr>
<td>Safety factor, q95</td>
<td>3.77</td>
</tr>
<tr>
<td>Normalised beta $\beta_{N,max}/\text{li}_3$</td>
<td>3.56</td>
</tr>
<tr>
<td>Bootstrap fraction fbs</td>
<td>0.46</td>
</tr>
<tr>
<td>Confinement coefficient, HH</td>
<td>1.25</td>
</tr>
<tr>
<td>Plasma density, $&lt;n&gt;$ ($10^{20}$m$^{-3}$)</td>
<td>0.92</td>
</tr>
<tr>
<td>$n_{\text{line}}/n_{\text{GW}}$</td>
<td>0.77</td>
</tr>
<tr>
<td>$n_0/n_{\text{ped max}}$</td>
<td>1.65</td>
</tr>
<tr>
<td>Av. Electron temperature (keV) $&lt;nT&gt;/&lt;n&gt;$</td>
<td>16.4</td>
</tr>
<tr>
<td>Aux. Heating power (MW) (NBI)</td>
<td>100</td>
</tr>
<tr>
<td>Fusion power (MW)</td>
<td>780</td>
</tr>
<tr>
<td>Q</td>
<td>7.85</td>
</tr>
</tbody>
</table>

**ITER:** High field side pellet injection: 2Hz, 2r=h=0.7 cm, $v_p = 0.5$ km/s. $\chi_i = \chi_e = D$ with parabolic profile
Net consumption of tritium

The first ten years ~ 5kg
Average 0.3/Blanket test area 0.4 MWa/m² ~ 15 kg (Minimum requirement)
Average 0.5/ Blanket test area 0.7 MWa/m² ~ 25 kg (Design value)
~30kg of tritium could be supplied with external sources
Phased Operations

Hydrogen Phase
Confirmation of the machine performance and increase of reliability of the operation
  Full commissioning of the ITER system in a non-nuclear environment
  Development of operation scenarios with semi-detached divertor and ~70 MW
  Better control/mitigation of disruptions/VDEs/ELMs/runaway electrons
  Characterization of dusts
  Build-up of experimental groups in the world wide fusion community

Deuterium Phase
Nuclear commissioning and confirmation of the basic plasma characteristics
  No human access into the vessel

Deuterium Tritium Phase
  Research of long burning plasmas
  Optimization of operations for various objectives
  Engineering tests including blanket tests for the next step
Participations

1. The maximum numbers of the ILE staff from the different Parties will be set in consideration to the contributions of the Parties. The complement of the directly employed ILE staff should be kept to a minimum necessitated in the project implementation.

2. The ILE staff arrangements should encourage mobility between the project and the Parties’ domestic programmes.

3. To ensure wide scientific participation in the Project, the short-term (less than a year) participation of qualified personnel from universities and other institutions (“Guest Researcher”) will be encouraged.

4. The operation of ITER will be conducted by the ILE staff only. The scientific exploitation of ITER will, beside the ILE staff, also involve the participation of researchers from fusion laboratories, universities or other research institutes of the Parties, which will be only part-time on site.
Physics Issues of Burning Plasmas in Inductive Mode

1. Thermal instability/Burn control
   - Normal operation: Stable
   - $Q_{\text{10}}$: Thermal instability could be triggered.

2. Collective fast alpha particle effects
   - Normal operation: $\beta_{\text{axis}} \leq 1\%$
   - Higher beta ($\leq 2\%$): $H_H > 1$, $P_{\text{ad}} > 100$ MW, $I_P > 15$ MA

3. Thermal beta effects
   - Normal operation: $\beta_N \leq 2$
   - Higher beta: $n > n_G$, $H_H > 1$, $P_{\text{ad}} > 100$ MW

4. Sawtooth effect
   - Normal operation: with sawtooth
   - No sawtooth: $q_0 > 1$, $\sim 100$ s

5. Pedestal/Edge Plasma/Divertor

Additional Issues of Burning Plasmas in Non-inductive Mode

6. Interaction among burn, $I_b/I_{\text{ad}}$, fuelling/pumping, impurity, transport, etc.

How can appropriate plasmas (high $Q$, $\beta_N$ and $I_b/I_p$) be obtained?

Requirements are clear but predictions are not reliable.

Stabilization: high $\beta_N$ with low $I_i$
High $n$, peaked $n$: deep fuelling
High $H_H$: ITB, deep fuelling
The flexibility of ITER will allow research in a large operation space.

\[ (P_{\text{fusion}}, \ Q, \ n, \ \beta, \ \text{pulse length}, \ I_p \----- ) \]

(Confirm predictable operation ⇒ Explore frontier)

- Predictable operations and extended operations with inductive current drive
  \[ 150 \rightarrow 700\text{MW}, \ \frac{n}{n_G} = 0.5 \rightarrow 1, \ \beta_N = 1.2 \rightarrow 2.4, \ Q = 5 \rightarrow 10 \rightarrow 20 \rightarrow \infty \]
  ~ 100 s burn is necessary to study plasma behavior.

- Hybrid operations
  > 1000 s / 500 MW/Q=5 with reasonable parameters for blanket test (0.77 MW/m²)
  If necessary, q95 > 3.5 scenarios is available.

- Research of fully non-inductive driven operations aiming at Q=5
  (higher \( \beta \)/higher confinement, methods included in ITER)
  ~ 2000 s is necessary to achieve steady state of AT mode from conventional one.
  By optimizing current ramp-up, steady state of AT can be achieved within 200 s.

The experimental concept will increase efficiency, involve the worldwide fusion community and promote scientific competition.
Ip=17 MA, \( <n_e> = 1.15 \times 10^{20}/m^3 \) \( (<n_e>/n_G=0.85) \), \( \tau_{he}/\tau_E \sim 5 \) and argon is seeded to limit the power to the divertor region to 30 MW.
Simulation for feedback control of target plate temperature by impurity seeding in case of sudden increase of fusion power $P_{\text{FUS}}$. 
Plasma parameter profiles at the current flat-top ($t > 1000$ s) for the steady-state WNS operational scenario.

$H_H = 1.57$, $Q = 6$, $\beta N = 2.95$, $P_{NB} + P_{LH} = 30 + 30$ MW, $R/a = 6.35/1.85$