Tokamak Operational Scenarios

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• What is a ‘tokamak (operational) scenario’?
• Optimisation strategies for tokamak scenarios
• Conventional scenarios
• Advanced scenarios
• Summary and conclusions
What is a ‘tokamak scenario’?

A tokamak (operational) scenario is a recipe to run a tokamak discharge

Plasma discharge characterised by

- external control parameters: \( B_t, R_0, a, \kappa, \delta, P_{\text{heat}}, \Phi_D \ldots \)
- integral plasma parameters: \( \beta = 2\mu_0<p>/B^2, I_p = 2\pi \int j(r) \, r \, dr \ldots \)
- plasma profiles: pressure \( p(r) = n(r) \cdot T(r) \), current density \( j(r) \)

\[ \begin{align*}
\beta_p &= 1 \\
I_p &= 800 \text{ kA} \\
f_{NI} &= 14% \\
\end{align*} \]

\[ \begin{align*}
\beta_p &= 1 \\
I_p &= 800 \text{ kA} \\
f_{NI} &= 37% \\
\end{align*} \]

→ operational scenario best characterised by \textit{shape of} \( p(r), j(r) \)
Control of the profiles $j(r)$ and $p(r)$ is limited

- ohmic current coupled to temperature profile via $\sigma \sim T^{3/2}$
  $\rightarrow$ inductive current profiles always peaked, $q$-profiles monotonic

- external heating systems drive current, but with limited efficiency
  (typically less than 0.1 A per 1 W under relevant conditions)...

- pressure gradient drives toroidal ‘bootstrap’ current: $j_{bs} \sim (r/R)^{1/2} \nabla p/B_{pol}$

ITER Q=10 simulation

safety factor:

$$q \approx \frac{r}{R} \frac{B_{tor}}{B_{pol}} \propto \frac{r^2}{R} \frac{B_{tor}}{I_p}$$
Control of the profiles \( j(r) \) and \( p(r) \) is limited

Pressure profile determined by combination of heating / fuelling profile and radial transport coefficients

- ohmic heating coupled to temperature profile via \( \sigma \sim T^{3/2} \)
- external heating methods allow for some variation – ICRH/ECRH deposition determined by \( B \)-field, NBI has usually broad profile
- under reactor-like conditions, dominant \( \alpha \)-heating \( \sim (nT)^2 \)
- gas puff is peripheral source of particles, pellets further inside

*Shape of profile will strongly depend on (turbulent) heat conductivity and particle diffusivity (i.e. be ‘self-organised’)*
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Power $P_{\text{loss}}$ needed to sustain plasma

- determined by thermal insulation:
  \[
  \tau_E = \frac{W_{\text{plasma}}}{P_{\text{loss}}} \quad \text{('energy confinement time')}\]

Fusion power increases with $W_{\text{plasma}}$

- $P_{\text{fus}} \sim n_D n_T \langle \sigma v \rangle \sim n_e^2 T^2 \sim W_{\text{plasma}}^2$

Present day experiments: $P_{\text{loss}}$ compensated by external heating

- $Q = \frac{P_{\text{fus}}}{P_{\text{ext}}} \approx \frac{P_{\text{fus}}}{P_{\text{loss}}} \sim n T \tau_E$

Reactor: $P_{\text{loss}}$ compensated by $\alpha$-(self)heating

- $Q = \frac{P_{\text{fus}}}{P_{\text{ext}}} = \frac{P_{\text{fus}}}{(P_{\text{loss}} - P_{\alpha})} \rightarrow \infty$ (ignited plasma)
Optimisation of $nT\tau_E$: ideal pressure limit

Optimising $nT$ means high pressure and, for given magnetic field, high $\beta = 2\mu_0 <p> / B^2$

This quantity is limited by magneto-hydrodynamic (MHD) instabilities

‘Ideal’ MHD limit (ultimate limit, plasma unstable on Alfvén time scale $\sim 10 \mu s$, only limited by inertia)

- ‘Troyon’ limit $\beta_{max} \sim I_P/(aB)$, leads to definition of $\beta_N = \beta/(I_P/(aB))$
Optimisation of $nT\tau_E$: resistive pressure limit

Optimising $nT$ means high pressure and, for given magnetic field, high $\beta = 2\mu_0 \langle p \rangle / B^2$

This quantity is limited by magneto-hydrodynamic (MHD) instabilities

‘Resistive’ MHD limit (on local current redistribution time scale $\sim 100$ ms)

Neoclassical Tearing Mode (NTM)
Empirical 'Greenwald-limit' describes well maximum density

- seems to be linked to a change in edge transport at $n \sim n_G$
- can be overcome if density profile shape is varied (peaked)

$n_G = \frac{l_p}{\pi a^2}$
Operation at $n/n_{GW} > 1$

Stable operation at $n/n_{GW} = 1.5$ using pellets.
Edge density stays below $n_{GW}$ in all cases (up to $n_{e0} = 4 \times n_{GW}$!)

- H-mode density limit = edge density limit
Empirical confinement scalings show linear increase of $\tau_E$ with $I_p$

- note the power degradation ($\tau_E$ decreases with $P_{\text{heat}}$!)
- ‘H-factor’ $H$ measures the quality of confinement relative to the scaling

Empirical ITER 98(p,y) scaling:

$$\tau_E \sim H I_p^{0.93} P_{\text{heat}}^{-0.63} B_t^{0.15} \ldots$$
BUT: for given $B_t$, $I_p$ is limited by current gradient driven MHD instabilities

Limit to safety factor $q \sim (r/R) \left( B_{tor}/B_{pol} \right)$

- for $q < 1$, tokamak unconditionally unstable $\rightarrow$ central ‘sawtooth’ instability
- for $q_{edge} \rightarrow 2$, plasma tends to disrupt (external kink) $\rightarrow$ limits value of $I_p$
Optimising for $Q=P_{\text{fus}}/P_{\text{ext}}$ drives operational point close to operational limits
For steady state tokamak operation, high $I_p$ is not desirable:
Tokamak operation without transformer: current 100% noninductive
- external CD has low efficiency (remember less than 0.1 A per W)
- internal bootstrap current high for high $j_{bs} \sim (r/R)^{1/2} \nabla p/B_{pol}$
  $$\rightarrow f_{Ni} \sim j_{bs}/I_p \sim p/B_{pol}^{2} \sim \beta_{pol}$$

‘Advanced’ scenarios, which aim at steady state, need high $\beta$, low $I_p$,
have to make up for loss in $\tau_E$ (e.g. through transport barriers)

Without the ‘steady state’ boundary condition, a tokamak scenario
is called ‘conventional’
Optimising for $Q = \frac{P_{\text{fus}}}{P_{\text{ext}}}$ drives operational point close to operational limits
Including the steady state constraint emphasizes the $\beta$-limit (and de-emphasizes current limit)
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The (low confinement) L-mode scenario

Standard scenario without special tailoring of geometry or profiles

- central current density usually limited by sawteeth
- temperature gradient sits at critical value over most of profile
- extrapolates to very large ($R > 10 \text{ m}, I_p > 30 \text{ MA}$) pulsed reactor
The (high confinement) H-mode scenario

With hot (low collisionality) conditions, edge transport barrier develops

- gives higher boundary condition for ‘stiff’ temperature profiles
- global confinement $\tau_E$ roughly factor 2 better than L-mode
- extrapolates to more attractive ($R \sim 8$ m, $I_p \sim 20$ MA) pulsed reactor
H-mode is ITER standard scenario for $Q=10$…

The design point allows for…

- …achieving $Q=10$ with conservative assumptions
- …incorporation of 'moderate surprises'
- …achieving ignition ($Q \to \infty$) if surprises are positive
...but some open issues remain...

Need to minimise ELM impact on divertor

- reduce power flow to divertor by radiative edge cooling
- special variants of the scenario (Quiescent H-mode, type II ELMs)
- ELM mitigation – pellet pacing or Resonant Magnetic Perturbations

Need to tackle NTM problem

- NTM suppression by Electron Cyclotron Current Drive demonstrated, but have to demonstrate that this can be used as reliable tool
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Advanced scenarios aim at stationary (transformerless) operation

- external CD has low efficiency (remember less than 0.1 A per W)
- internal bootstrap current high for high $j_{bs} \sim (r/R)^{1/2} \nabla p/B_{pol}$
  
  \[ f_{NI} \sim I_{bs}/I_p \sim p/B_{pol}^2 \sim \beta_{pol} \]

Recipe to obtain high bootstrap fraction:

- low $B_{pol}$, i.e. high $q$ – elevate or reverse $q$-profile ($q=(r/R)(B_{tor}/B_{pol})$)
- eliminates NTMs (reversed shear, no low resonant $q$-surfaces)
- high pressure where $B_{pol}$ is low, i.e. peaked $p(r)$

Both recipes tend to make discharge ideal MHD (kink) unstable!

In addition, $j_{bs}$ profile should be consistent with $q$-profile
A self-consistent solution is theoretically possible

- reversing $q$-profile suppresses turbulence – internal transport barrier (ITB)
- large bootstrap current at mid-radius supports reversed $q$-profile
Problems of the Advanced tokamak scenario

Broad current profile leads to low kink stability (low $\beta$-limit):

- can partly be cured by close conducting shell, but kink instability then grows on resistive time scale of wall (Resistive Wall Mode RWM)
- can be counteracted by helical coils, but this needs sophisticated feedback

Position of ITB and minimum of $q$-profile must be well aligned:

- needs active control of both $p(r)$ and $j(r)$ profiles – difficult with limited actuator set (and cross-coupling between the profiles)
Close-by conducting wall can stabilise external kink instability

- usually not an issue for conventional scenario, but advanced scenarios prone to external kink due to broad current/peaked pressure profile

for conventional H-mode: modest need and gain

for advanced ITB scenarios: strong need and significant gain
...but no wall is ideally conducting!

When ideal kink is wall stabilised, RWM can grow on wall time scale

- rotation w.r.t. wall can stabilise the RWM if $\omega_{\text{rot}} >> 1/\tau_W$
- balance between wall drag and (rotating) plasma drag on mode

Ideal regime: Ideal kink stable if wall is close enough

RWM regime: RWM is stable when slipping between mode and wall is large enough

When ideal kink is wall stabilised, RWM can grow on wall time scale
Feedback control using RMPs shows possibility to exceed no-wall $\beta$–limit

- rotation plays a strong role in this process and has to be understood better (ITER is predicted to have very low rotation)
- fast particle stabilisation can help substantially even at low rotation
Good performance can only be kept for several confinement times, not stationary on the current diffusion time ($10 – 50 \tau_E$ in these devices)
A compromise: the 'hybrid' scenario

Reversed shear, ITB discharges
+ very large bootstrap fraction
+ steady state should be possible
- low $\beta$-limit (kink, infernal, RWM)
- delicate to operate

Zero shear, 'hybrid' discharges
+ higher $\beta$-limit (NTMs)
+ 'easy' to operate
- smaller bootstrap fraction
- have to elevate $q(0)$
  (also avoids sawteeth, NTMs)

Hybrid operation aims at flat, elevated q-profile discharges with high $q(0)$
Not clear if this projects to steady state, but it will be very long pulse…
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A variety of tokamak operational scenarios exists

- **L-mode**: low performance, pulsed operation, no need for profile control
- **H-mode**: higher performance, pulsed operation, MHD control needed
- **Advanced modes**: higher performance, steady state, needs profile control
ITER aims at operation in conventional and advanced scenarios
• demonstrating Q=10 in conventional (conservative) operation scenarios
• demonstrating long pulse (steady state) operation in 'advanced' scenarios

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<th>Hybrid</th>
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One mission of ITER and the accompanying programme is to develop and verify an operational scenario for DEMO
• DEMO scenario must be a point design (no longer an experiment)
• actuators even more limited (e.g. maximum of 2 H&CD methods)