ITER-EDA PHYSICS DESIGN REQUIREMENTS AND PLASMA PERFORMANCE ASSESSMENTS

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ABSTRACT

Physics design guidelines, plasma performance estimates, and sensitivity of performance to changes in physics assumptions are presented for the ITER-EDA Interim Design. The overall ITER device parameters have been derived from the performance goals using physics guidelines based on the physics R&D results. The ITER-EDA design has a single-null divertor configuration (divertor at the bottom) with a nominal plasma current of 21 MA, magnetic field of 5.68 T, major and minor radius of 8.14 m and 2.8 m, and a plasma elongation (at the 95% flux surface) of ~ 1.6 that produces a nominal fusion power of ~ 1.5 GW for an ignited burn pulse length of ≥ 1000 s. The assessments have shown that ignition at 1.5 GW of fusion power can be sustained in ITER for 1000 s given present extrapolations of H-mode confinement ($\tau_E = 0.85 \times \tau_{\text{ITER93H}}$), helium exhaust ($\tau_{\text{He}}^*/\tau_E = 10$), representative plasma impurities ($n_{\text{Be}}/n_e = 2\%$), and beta limit [$\beta_N = \beta(\%)/(I/aB) \leq 2.5$]. The provision of 100 MW of auxiliary power, necessary to access to H-mode during the approach to ignition, provides for the possibility of driven burn operations at $Q = 15$. This enables ITER to fulfill its mission of fusion power (~ 1 – 1.5 GW) and fluence (~ 1 MWa/m²) goals if confinement, impurity levels, or operational (density, beta) limits prove to be less favorable than present projections. The power threshold for H-L transition, confinement uncertainties, and operational limits (Greenwald density limit and beta limit) are potential performance limiting issues. Improvement of the helium exhaust ($\tau_{\text{He}}^*/\tau_E \leq 5$) and potential operation in reverse-shear mode significantly improve ITER performance.

I. INTRODUCTION

The main physics objective for ITER, set by Special Working Group 1 (SWG 1),¹ is to produce an ignited, steady burn for a duration of 1000 s and a neutron wall loading of 1 MW/m² [and a fluence of at least 1 MWa/m²

for nuclear component testing]. Meeting this objective requires that ITER should be able to operate in regimes that are permitted by established principles of tokamak operation, for example those governing confinement, MHD stability (β limits), density limits, disruption avoidance, impurity control, particle transport, etc. In each of these areas, the physics specifications and design guidelines have been developed by the ITER JCT, the Home Teams, and the ITER Physics Expert Groups based on the present understanding and characterization of tokamak experiments and theory. The initial ITER physics design guidelines² was developed during the ITER Conceptual Design Activities (ITER-CDA). In Sect. II, we provide an update to these guidelines reflecting the developments during the ITER Engineering Design Activities (ITER-EDA) Interim Design.³ The main plasma and device parameters (Table I) have been derived from the performance goals using these guidelines.^{3,4} In Sect. III, we present results from plasma performance assessments and sensitivity of performance to changes in physics assumptions.

Table I
Representative ITER device and nominal plasma parameters

Parameter	Symbol	Value
Major radius	R	8.14 m
Minor radius	a	2.80 m
Plasma configuration	—	Single-null
Plasma elongation	κ_{95}, κ_x	$\sim 1.6, \sim 1.75$
Plasma triangularity	δ_{95}	~ 0.24
Nominal plasma current	I	21 MA
Toroidal field	B	5.68 T (at R = 8.14 m)
MHD safety factor	q_{95}	3.05
Fusion power (nominal)	P _{fus}	1.5 GW
Average wall loading	Γ_n	~ 1 MW/m ²
PF flux swing	$\Delta\phi_{\text{PF}}$	530 Wb
Flux swing for burn	$\Delta\phi_{\text{burn}}$	84 Wb

II. SUMMARY OF PHYSICS DESIGN GUIDELINES

The main physics issues have been assessed by the ITER JCT, the Home Teams, and the ITER Physics Expert Groups on the basis of data from present tokamak experiments. The physics design guidelines are developed from reasonable extrapolations of this data base and guidance of tokamak theory.^{3,4} A brief summary of these guidelines, related to plasma performance, is presented.

Here, units are mks, MA, MW, with κ, δ average values at 95% flux, and $n_{20} = n_e/10^{20} \text{ m}^{-3}$ = line-average electron density, $T_{10} = (T/10 \text{ keV})$ = density-weighted average temperature ($T \approx T_e \approx T_i$), A_i = ave. atomic mass. H-mode profiles: $n, T \sim (1 - r^2/a^2)^{\alpha_n, T}$, with $\alpha_n \approx 0.1$ and $\alpha_T \approx 1.0$ as nominal values.

Configuration: ITER design is based on well-controlled single-null (SN) poloidal divertor configuration with the X-point at the bottom.

- Average plasma elongation and triangularity: κ (95%) = $\kappa_{95} \leq 1.65$, δ (95%) = $\delta_{95} \geq 0.2$.
- Empirical relationships for SN X-point and 95%: $\kappa_x \approx 1.09 \kappa_{95}$; $\delta_x \approx 1.5 \delta_{95}$.

Operational modes:

Nominal Operational Mode:

Inductive pulse flat-top (ignited conditions)	1000 s
Fusion power	1.5 GW
Maximum fusion power excursion	$\pm 20 \%$
Nominal plasma current	21 MA
Pulse repetition time	2200 s

At 1500 MW, representative (nominal) plasma density and temperature are: $n_e \sim 1.3 \times 10^{20} \text{ m}^{-3}$, $T \sim 10.5 \text{ keV}$.

Long Pulse Operational Mode: ITER will have the capability to accommodate long pulse operational modes of 2000 s of burn with auxiliary powers of up to 100 MW.

Steady State Operational Mode: ITER will have the capability to assess the conditions and power requirements for the steady state mode in plasmas having alpha power at least comparable to the externally applied power ($Q = P_{\text{fus}}/P_{\text{aux}} > 5$). ITER shall have the flexibility to accommodate steady state "advanced" operational modes with an additional investment for auxiliary systems, for pulse lengths of $\sim 10,000 \text{ s}$ and a fusion power $\sim 1.5 \text{ GW}$.

Confinement: Plasma energy confinement must be sufficient to achieve ignition and sustained burn at fusion power of $\sim 1.5 \text{ GW}$, under empirical scalings for ELMy H-mode plasma energy confinement and corresponding He and impurity concentrations ($Z_{\text{eff}} \sim 1.5$, $n_D T/n_e \sim 0.6$). Reference assumptions/models are as follows:

Energy confinement: ELMy H-mode

$$\begin{aligned}\tau_E(\text{required}) &= \tau_E(\text{ELMy H-mode}) \\ &= H_H \times [0.85 \times \tau_E(\text{ELM-free H-mode}) \\ &\quad \sim H_L \times \tau_E(\text{ITER89P L-mode})]\end{aligned}$$

Here H_H = H-mode scale factor with respect to 0.85 times ELM free H-mode [$H_H = 1$ ELMy H-mode, $H_H < 1$ degraded H-mode], H_L = $H_{89} =$ L-mode enhancement factor.

$$\tau_E(\text{ELM-free H-mode}) = \tau_E(\text{ITER93H}) = \tau_{E-93H} = 0.053 I^{1.06} R^{1.9} a^{-0.11} n_{20}^{0.17} B^{0.32} A_i^{0.41} \kappa_x^{0.66} p^{-0.67}$$

$$\tau_E(\text{ELMy H-mode}) = H_H \times 0.85 \times \tau_{E-93H}$$

$$\begin{aligned}\tau_E(\text{L-mode}) &= \tau_E(\text{ITER89-P L-mode}) = \tau_{E-89P} \\ &= 0.048 A_i^{0.5} I^{0.85} R^{1.2} a^{0.3} \kappa_x^{0.5} n_{20}^{0.1} B^{0.2} p^{-0.5}\end{aligned}$$

Particle confinement: ELMy H-mode with $\tau^*_{pHe}/\tau_E = 10$. Here $\tau^*_{pHe} = \tau_{pHe}/(1 - R_{He})$, where τ_{pHe} = He particle confinement time and R_{He} = He recycling coefficient.

Impurity content: Beryllium $n_{Be}/n_e = 2\%$

Auxiliary power: $P_{\text{aux}} \geq P_{\text{thr}}$ (L-H threshold power)

Beta Limit: $\beta_{\text{max}}(\%) = g(I/aB) = \beta_N(I/aB)$

- $\beta_N \leq 2.5$ nominal operation (ignition studies)
- $\beta_N (> 2.5?)$ advanced scenarios with reversed shear

Current profile control is possible for steady state, advanced scenarios (with reversed shear), resulting in a potentially higher β_N -value. See comments in $q_{\psi 95}$.

Safety factor: $q_{\psi 95} \geq 3.0$

Note: Confinement partly deteriorates and disruptivity increases with low-q operation. For low-q, experimentally observed effective stability limit is typically $\beta_N/q_{\psi} \sim 0.7$ – 0.9 , indicating that ITER parameters ($\beta_N/q_{\psi} \sim 2.5/3$) are marginal. Thus, it would be difficult to attain high- β_N values (> 3) without raising $q(0)$ and q_{ψ} substantially.

H-mode Threshold Power: H-mode is reached above a certain threshold power ($P_{\text{thr}} = P_{L-H}$) that is observed to depend on plasma and device parameters. A reverse H-to-L-mode transition will occur when the power crossing the separatrix falls below a value roughly half of P_{L-H} .

L-to-H transition: Initially ITER used $P_{\text{sep}} \geq P_{L-H0} = 0.044 n_{20} B S$, where $P_{\text{sep}} (= P_{\text{heat}} - P_{\text{rad}} - \partial W/\partial t; P_{\text{heat}} = P_{\alpha} + P_{OH} + P_{\text{aux}})$ is the power crossing the separatrix and S is the plasma surface area. For nominal ITER parameters, $P_{L-H0} \sim 300 n_{20} \text{ (MW)}$. During startup, $n_{20} \sim 0.3$ – 0.5 , access to H-mode will require 100–150 MW. Recent analysis of the ITER H-mode threshold database, assuming various experimentally observed dependence in a dimensionally correct form, or for a simple fit to data, suggest the following expressions:⁵

$$\begin{aligned} P_{L-H1} &= 0.3 n_{20} B R^{2.5} & P_{L-H2} &= 0.016 n_{20}^{0.75} B S \\ P_{L-H3} &= 0.036 n_{20} B^{0.6} S & P_{L-H4} &= 0.025 n_{20} B S. \end{aligned}$$

At $n_{20} \sim 0.5$, the extrapolation of these expressions to ITER gives about $P_{L-H1} = 150$ MW, $P_{L-H2} = 65$ MW, $P_{L-H3} = 60$ MW, and $P_{L-H4} = 85$ MW. Isotope scaling of the threshold power is not yet well established. A lower threshold in tritium than in deuterium is possible. Note that P_{L-H0} expression is the most conservative for ITER..

H-to-L transition: $P_{H-L} \sim 0.5 \times P_{L-H}$. For nominal ITER parameters, P_{H-L} (MW) $\sim 150 \times n_{20}$ [about 180–190 MW at $n_{20} \sim 1.3$, indicating that radiated power fraction, P_{rad}/P_{α} , cannot exceed ~40% (for 1500 MW nominal fusion power) without likely loss of H-mode and likely loss of sustained burn].

Density Limit:

High-density limit — imposes an upper limit on the plasma edge density. A tentative characterization of the Borrass density limit adapted for a single-null divertor configuration of ITER-EDA takes a form

$$n_{20} = \frac{(n_e / n_{es})}{f_{rad}^{div} 11/16 / (q_{\psi} R)^{1/16}} C [Q_{\perp}^{5/8} B^{5/16} (1 -$$

where n_{es} = upstream plasma electron density at the separatrix, $Q_{\perp}$ (MW/m²) = mean power flux crossing the separatrix, $f_{rad}^{div} = P_{rad}^{div} / (4\pi^2 R a k^{0.5} Q_{\perp})$ is the divertor impurity radiative fraction, $q_{\psi} = q_{\psi}(95\%)$, and $C \approx 2.37$. The value of n_{es}/n_e (~ 0.4 – 0.7) depends on particle transport and fueling at the plasma periphery.

For a comparison, Greenwald density limit (based on tokamak operational experience with gas puff fueling) is: $n_{20}^{GR} = \kappa J(\text{MA/m}^2) = I/(\pi a^2)$. For ITER, this yields to $n_e = 0.85 \times 10^{20} \text{ m}^{-3}$, significantly below the nominal value of $1.3 \times 10^{20} \text{ m}^{-3}$. In several experiments with pellet fueling it was possible to exceed the Greenwald limit by as much as a factor of 2. Test of density limit in H-mode experiments with ITER like geometry is an area of active physics R&D.

Low-density limit — defines the heating power required to get the H-mode. This limit is also disruptive and associated with the error-field induced instability. For example: $n_e = 3 \times 10^{20} \text{ m}^{-3}$ corresponds to H-mode power threshold of ~100 MW and requires $m=2/n=1$ component of the radial vacuum error field $\delta B_{r21} \leq 3$ G at $q = 2$ surface.

Performance Margins:

Confinement margin defines how much the confinement can be decreased while maintaining the power balance:

$$M_{conf} = [(\langle nDT \rangle \tau_E)_{(predicted)} / (\langle nDT \rangle \tau_E)_{(req.)}] = 1/H_{Hmin}$$

where 'predicted' denotes predicted value of triple product $[\langle nDT \rangle \tau_E]$ using the physics rules and 'required' denotes the minimum value required to satisfy the power balance. Confinement margin is evaluated at the nominal fusion power (1500 MW). For $P_{aux} = 0$, confinement margin = ignition margin, $M_{conf} = M_{ign}$. [$P_{aux} \neq 0$ represent driven operation with fusion gain $Q = P_{fus}/P_{aux}$.]

Beta limit margin shows how close the operating point is to the beta limit, $\beta_{Nmax} = 2.5$: $M_{\beta} = 2.5 / \beta_N$.

Burn duration margin compares the actual burn duration to the required 1000 s: $M_{burn} = t_{burn} / 1000$ (s).

H-to-L transition defines how close an operating point is to the H-to-L transition: $M_{H-L} = P_{sep} / (0.5 \times P_{L-H})$.

Fusion Power: The nominal value of the fusion power is 1500 MW, of which 1200 MW is the neutron power (300 MW alpha power). A power excursion of $\pm 20\%$ lasting for ~10 s is anticipated. Fusion power is obtained by integrating D-T fusion reactivity over plasma profiles $[P_{fus} \sim \int n^2 \langle \sigma v \rangle E_{fus} dV]$. Expression for $\langle \sigma v \rangle$ is⁶

$$\begin{aligned} \langle \sigma v \rangle &= 10^{-6} \times C1 \times \theta \times \exp(-3\xi) \times [\xi / (m_r c^2 T^3)]^{0.5} \\ \theta &= T / (1 - T[C2 + T(C4 + T \times C6)] / [1.0 + T[C3 + T(C5 + T \times C7)]]) \\ \xi &= [(B_G)^2 / (40)]^{1/3} \end{aligned}$$

where $\sigma v (\text{m}^3/\text{s})$ = reactivity, $T(\text{keV})$ = temperature. For D-T: fuel, $B_G[(\text{keV})^{0.5}]$ = parameter = 34.3827, $m_r c^2 (\text{keV})$ = rest mass energy = 1.124656×10^6 and C's [C1 through C7] are the fitting vectors: $C1 = 1.17302 \times 10^{-9}$, $C2 = 1.51361 \times 10^{-2}$, $C3 = 7.51886 \times 10^{-2}$, $C4 = 4.60643 \times 10^{-3}$, $C5 = 1.35 \times 10^{-2}$, $C6 = -1.06750 \times 10^{-4}$, $C7 = 1.366 \times 10^{-5}$.

III. PLASMA PERFORMANCE ASSESSMENTS

The plasma performance attainable in ITER is affected by many physics issues, including energy confinement, L-to-H and H-to-L-mode power transition thresholds, MHD stability/beta limit, density limit, helium removal, impurity content, sawtooth effects, etc. The performance margins of ITER are examined for a number of operational scenarios, using the ITER physics guidelines given in Sect. II. The sensitivity of ITER performance to changes in physics guidelines and uncertainties is investigated.

Most of the projections of ITER plasma performance were done using the 1.5-D transport modeling (PRETOR code) and 0-D models (simple power balance, as well as SUPERCODE). While 0-D studies provides a quick survey and assessment for given profiles and confinement scaling, 1.5-D allows dynamic evolution and determination of (n, T) profiles and magnetic configuration consistent

with local power deposition and particle sources. The 1.5-D modeling can also be constrained to follow a specified scaling for τ_E ($0.85 \times \text{ITER93H}$). This ensures that the projected confinement time is in agreement with the scaling deduced from the tokamak database.

A. Reference Operational Mode and Performance Margins

The main plasma and machine parameters (Table I) along with physics rules (from Sect. II) are used in the modeling. Figure 1 shows typical temperature and density profiles. Figure 2 displays the plasma operation contours (POPCONs). Sustained burn is possible on any point of the ignition curve indicated by $P_{\text{aux}} = 0$. Table II summarizes the key plasma parameters in the reference (ignited) operational mode. The results in Table II show that ITER can achieve ignited operation ($P_{\text{aux}} = 0$) and sustain burn at He concentration as high as 14%.

Figure 1. Typical ITER temperature and density profiles.

Figure 2. ITER-EDA POPCON diagram for reference ignition scenario assuming ELMMy H-mode ($H_H = 1$) with $\tau^*_{\text{He}}/\tau_E = 10$ and $n_{\text{Be}}/n_e = 2\%$. Ignition ($P_{\text{aux}} = 0$), beta limit ($\beta_N = 2.5$), H-L transition ($P_{\text{H-L}} = 0.022 \text{ nBS}$) and $P_{\text{fus}} = 1.5 \text{ GW}$ contours are shown.

Table II: ITER plasma parameters for ignited burn with "reference" modeling assumptions

Parameter	Symbol [unit]	Value
Volume-average temperature	$T = 0.5(T_e + T_i)$ [keV]	10.5
Volume-average density	n_e [10^{20} m^{-3}]	1.3
Impurity fractions	$n_{\text{Be}}/n_e, n_{\text{He}}/n_e$ [%]	2, 14
Effective charge	Z_{eff}	1.5
Radiated power fraction	$P_{\text{rad}}/P_{\alpha} = f_{\text{rad}}$	0.36
Confinement time	τ_E [s]	6.2
Normalized confinement	$H_H = \tau_E/0.85\tau_{E93H}$	1.0
Normalized beta (total)	β_N	2.4
Poloidal beta (total)	β_{pol}	0.9
DT Triple product	$\langle nDT T \tau_E \rangle$ [$10^{21} \text{ m}^{-3} \text{ keV s}$]	3.3
Internal inductance	$l_i(3)$	0.9
Loop voltage	V_{loop} [mV]	72
Burn duration	t_{burn} [s]	1160
Plasma thermal energy	W_{th} [GJ]	1.1
Plasma magnetic energy	W_{mag} [GJ]	1.2

Table III summarizes the margins. In Table III, all margins are above 1, showing that the plasma parameters projected for ITER provide sufficient confinement, remain below the beta limit, provide sustained burn, and are compatible with H-mode operation.

Table III. Physics margins for the ITER reference (1.5 GW, ignited) operational point

Margin	Value
M_{conf}	1.11
M_{β}	1.04
M_{burn}	1.09
$M_{\text{H-L}}$	1.06

Energetically accessible regimes are determined from a simple power balance using physics models and guidelines discussed in Sect. II with representative profiles and ELMMy H-mode confinement. Results are summarized in Fig. 3, displaying POPCONs in (n-T) space.

Inspections of Figs. 2,3 and Table III, for the reference operating scenario, show that (based on the physics models and assumptions), margins for achieving the ITER (SWG 1) objectives (ignition and sustained burn for 1000 s with 1.5 GW) are finite, but small. However, further analysis considering 100 MW of auxiliary power shows that driven, high-Q regimes are possible with considerably more confinement margin. Estimated confinement margin is most sensitive to edge temperature profiles, impurity levels, and auxiliary heating. These are discussed next.

Figure 3. Summary of POPCON diagrams for ITER-EDA assuming ELMy H-mode ($\tau_E = H_H \times 0.85 \times \tau_{E-93H}$) with $\tau^*_{He}/\tau_E = 10$ and 2% Be. Curves passing through open points are ignited operating contours energetically permitted by H-mode confinement ($H_H = 1$) and effect of degradation ($H_H = 0.95, 0.9 \dots$). Curve labeled MSI corresponds to the 'marginally sustained ignition'. Curve labeled $\beta_N = 2.5$ corresponds to a beta limit. Pulse length is constant at the value indicated for curves labeled 800 and 1000 s. $P_\alpha = 400$ MW, 350 MW, ... curves correspond to contours of constant alpha power. Curves labeled H-L= 50%, 40% ... show contours where the power crossing the separatrix equals 0.5, 0.4, ... of the L-H transition threshold [Here conservative scaling, $P_{L-H} = 0.044 n_{20} B S$ is used.]

B. Sensitivity to Physics Assumptions

A sensitivity study for the reference parameters was performed [5] with the aim to assess the operations space for ITER and to determine the confidence level for fulfilling the ITER mission. The following areas were investigated:

Sensitivity to impurity level. At least three main impurity species will be present in ITER plasma. Helium will be produced by fusion reaction and Beryllium will originate from plasma facing components. The main effect of low Z impurities is dilution of DT mixture and as result decrease of the fusion power. In addition, it is likely that a trace amount of medium-Z species such as Neon or Argon used for radiation in SOL plasma/divertor and Tungsten (W) may be present--these impurity levels must be kept very small in the main plasma because of Bremsstrahlung.

Sensitivity of ignition (at 1.5 GW) to Be level is estimated and the results are summarized in Table IV. The Be concentration is varied, other physics rules are held unchanged. At constant P_{fus} , a higher Be fraction requires a higher electron density, resulting in a lower He levels. The confinement margin drops by 5% with added 0.7% Be.

Table IV : Confinement/Ignition margins vs Be concentration at constant fusion power (1.5 GW)

	He concentration	M_{conf}
$n_{Be}/n_e = 2\%$ (ref.)	14%	1.11
$n_{Be}/n_e = 3.4\%$	12.5%	1.00

The He level is varied by varying τ^*_{He}/τ_E and thereby the He pumping efficiency, at constant fusion power, with the nominal Be fraction and with $H_H = 1.0$ (unless H_H is varied to study the confinement margin). All other plasma parameters are then deduced self-consistently. The results of the He level scan are shown in Fig. 4. The He fraction increases with τ^*_{He}/τ_E from 8.7% at $\tau^*_{He}/\tau_E = 4$ to 15% at $\tau^*_{He}/\tau_E = 12$. The ignition margin decreases, and sustained ignition is lost when $\tau^*_{He}/\tau_E > 12.5$.

At higher He levels ($\tau^*_{He}/\tau_E > 12.5$), 1.5 GW can be attained in a driven mode. For example, with 50 MW of auxiliary power, $Q = 30$ operation in ITER is possible for $\tau^*_{He}/\tau_E \sim 15$ [corresponding to $n_{He}/n_e \sim 16-17\%$].

Figure 4. Sensitivity to He concentration: confinement margin ($M_{conf} = M_{HH}$) vs He level (τ^*_{He}/τ_E) at 1.5 GW. Sustained ignition is lost for $\tau^*_{He}/\tau_E > 12.5$.

Note that the lower He levels correspond to higher operating temperatures and longer burn duration, i.e. 2000 s at $\tau^*_{He}/\tau_E = 4$. The loop voltage varies from ~40 mV at $\tau^*_{He}/\tau_E = 4$ to ~100 mV at $\tau^*_{He}/\tau_E = 12$. The normalized beta is not sensitive to the He level.

As indicated earlier, trace amount of medium-Z species such as Neon used for radiation in SOL plasma/divertor may be present. The sensitivity to Ne level is estimated

by direct variation of neon concentration [$n_{Ne}/n_e \sim 0-0.5\%$]. At neon level of 0.2%, ignition (at 1.5 GW) under reference physics assumptions [$H_H = 1$, $\tau_{He}/\tau_E = 10$, 2% Be, $\beta_N < 2.5$] is possible with a very small operating window/margin. At 0.5% Ne, 1.5 GW of fusion power is attained with ≤ 40 MW of auxiliary power ($Q \sim 40$).

Transient ignition: Transient ignition occurs as long as the nominal fusion power can be sustained without auxiliary heating. Helium will not be present during the early phase of the discharge, and confinement requirements to reach ignition will be less stringent. As the H factor is decreased, the duration at which ignition is sustained decreases, and becomes a single point in time for $H_H \sim 0.83 = H_{Hmin}$ ($M_{conf} = 1.2$). For burn duration beyond 100 s, auxiliary heating is required if $H_H < H_{Hmin} = 0.89$, corresponding to marginal ignition.

Sensitivity to temperature profiles (edge pedestal): Plasma profiles can substantially influence the ignition margin. Peaked (L-mode like) profiles are more favorable for ignition than H-mode type profiles (which are relatively broad, Fig. 1), because they produce larger fusion power at the same beta (but edge densities or temperatures are lower for L-mode profiles). L-mode profiles were compared with the H-mode like ITER profiles (edge pedestal defined by ballooning limit) and the impact of sawteeth was assessed (peaked versus flat profile at plasma center). Confinement margins are: $M_{conf} = 1.11$ for reference H-mode profiles, whereas $M_{conf} = 1.22$ for L-mode profiles. The edge temperature pedestal decreases the confinement margin by about 10% while sawtooth stabilization increases the confinement margin by about 4% when fusion power and H_H factor are fixed at the nominal values. Note that sawtooth stabilization would lead not only to more peaked profiles but would probably also improve the overall confinement, further increasing the margin.

Driven operation/auxiliary heating: Auxiliary heating power available in ITER is an effective way to increase the confinement margin. Each 50 MW of heating corresponds to an increase of margin by 10%. $M_{conf} = 1.23/1.31$ for $P = 50/100$ MW. The margin provided by the 100 MW of auxiliary power can provide long pulse operation at nominal fusion power (1.5 GW) even when the impurity content is higher and sustained ignition is not possible. The same conclusion applies if the impurity content remains at the nominal value but the confinement is degraded to $H_{99} = 1.8$ for ITER89P L-mode scaling (representative of Bohm type scalings) corresponding to $H_H = 0.76$. Driven operation/auxiliary heating are also discussed under sensitivity to impurities, beta and density limits, and other scenarios.

Sensitivity to density and beta limits: The density limit is predicted to be either $0.85 \times 10^{20} \text{ m}^{-3}$ for Greenwald scaling or $\sim 2 \times 10^{20} \text{ m}^{-3}$ for Borraass scaling. The nominal operation density is $1.3 \times 10^{20} \text{ m}^{-3}$, which is well above the Greenwald limit but comfortably below the Borraass density limit. The Borraass density limit occurs whenever the power across the separatrix can not sustain the losses in the divertor plasma and is therefore an edge density limit. Other mechanisms associated with plasma edge density might be responsible for the widely observed Greenwald limit. Experimentally, operation at average densities above the Greenwald limit has been achieved with peaked density profiles (e.g. deep pellet injection). Should the Greenwald limit be confirmed and should deep pellet injection not be feasible or be unable to provide peaked profiles in ITER (penetration, ELMs) the operating density might be limited to $0.85 \times 10^{20} \text{ m}^{-3}$.

The maximum normalized β (thermal + fast particles) in ITER is taken as $\beta_N = 2.5$. For $q_\psi \sim 3$, experimentally observed effective stability limit is typically $\beta_N/q_\psi \sim 0.7-0.9$, indicating that ITER parameters ($\beta_N/q_\psi \sim 2.5/3$) may be marginal. Under nominal confinement ($H_I = 1$), reducing β -limit to $\beta_N \sim 2$ prevent reaching fusion powers above 1.1 GW. For cases with degraded confinement ($H_{99} < 1.8$ or $H_H < 0.76$), the beta limit is not an issue, they all require auxiliary power and operate with $\beta_N < 2$.

Table V summarizes ITER performance for the case of density being limited to the Greenwald limit or lower ($0.8 \times 10^{20} \text{ m}^{-3}$). Auxiliary heating power is varied from the minimum value needed to sustain H-mode (~ 30 MW) to the maximum design value (100 MW). The scenario with $P_{aux} = 50$ MW corresponds to the same fluence as the nominal reference scenario (1.5 GW sustained for 1000 s).

Table V. ITER performance at a density below the Greenwald limit ($n_e \sim 0.8 \times 10^{20} \text{ m}^{-3}$).

	Minimum P_{aux} [Max. Q]	Equivalent Ref. Case Fluence	Maximum P_{aux}
P_{aux} (MW)	30	50	100
Fusion power (MW)	800	1000	1300
$Q_{th} = P_{fus}/P_{aux}$	27	20	13
β_N	1.9	2.2	2.5
Burn time (s)	1300	1500	1700
P_{sep}/n_e (MW/ 10^{20} m^{-3})	160	200	300

Reducing the plasma density, leads to a reduction of the fusion power and of the ratio P_{sep}/n_e (power crossing the plasma separatrix over averaged density). Therefore 30 MW of auxiliary heating would have to be applied to

sustain the H-mode. Higher fusion powers could be reached with more auxiliary heating, but the nominal value of 1.5 GW would not be attained even for the maximum auxiliary heating (100 MW). However, a longer burn duration - compared to the reference case - is possible because the plasma operates at higher temperature.

Figure 5 shows a POPCON diagram calculated for nominal ITER parameters for low density ignition scenario assuming ELMy H-mode ($H = 1$) with $\tau^*_{He}/\tau_E = 10$ and $n_{Be}/n_e = 2\%$. Ignition ($P_{aux} = 0$), beta limit ($\beta_N = 2.5$), Greenwald density limit ($I/\pi a^2$), H-L threshold ($P_{H-L} = 0.022nBS$) and $P_{fus} = 0.8$ GW contours are shown. At the H-L transition, $P_{fus} \sim 0.8$ GW, and $P_{aux} \sim 30$ MW ($Q \sim 27$). With the maximum available power (100 MW), $P_{fus} \sim 1.3$ GW ($Q \sim 13$). Performance can be improved dramatically and Greenwald density limit can be overcome if it is possible to lower He level to $\tau^*_{He}/\tau_E = 5$.

Figure 5. ITER-EDA POPCON diagram for low density ignition scenario assuming ELMy H-mode ($H_H = 1$) with reference He and Be levels. Ignition ($P_{aux} = 0$), β -limit ($\beta_N = 2.5$), Greenwald density limit ($I/\pi a^2$), H-L threshold ($=0.022nBS$) and $P_{fus} = 800$ MW contours are shown.

Other sensitivities and scenarios: For nuclear testing purposes ITER will require scenario that produce largest fluence during one pulse. Figure 6 shows the maximum fluence as a function of plasma current calculated for driven plasma scenario with auxiliary power $P_{aux} = 100$ MW. It was assumed that $\beta_N = 2.4$, and auxiliary power drives current with a current drive efficiency of $\gamma = 0.2 \times 10^{20}$ A/W-m². Note that the optimal plasma current that maximizes the fluence is $I \sim 15$ -18 MA and maximum fluence (for one pulse) is ~ 3200 MWs/m². Maximum pulse length is about 4000 s and $P_{fus} \sim 1$ GW.

Figure 6. Fusion power and neutron fluence during one pulse vs plasma current for driven scenario with auxiliary heating and current drive power of 100 MW. Plasma parameters are optimized to obtain maximum fluence.

ITER configuration has been shown to support true steady-state scenarios based on negative central shear with $I \sim 12$ MA and a large bootstrap current fraction (~ 70 -80%) assisted by on- and off-axis current drive (CD). In this mode $P_{fus} \sim 1$ -to-1.5 GW and $Q > 10$ [$P_{aux} = 100$ MW].^{3,7}

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ITER-EDA Physics Design Requirements and Plasma Performance Assessments

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Summary:

Physics design guidelines, plasma performance estimates, and sensitivity of performance to changes in physics assumptions are presented for the ITER Interim Design.

The overall ITER device parameters have been derived from the performance goals using physics guidelines based on the physics R&D results.

- The assessments have shown that ignition at 1.5 GW of fusion power can be sustained in ITER for 1000 s given present extrapolations of H-mode confinement ($\tau_E = 0.85 \times \tau_{93H}$), helium exhaust ($\tau_{He}^*/\tau_E = 10$), representative plasma impurities ($n_{Be}/n_e = 2\%$), and beta limit [$\beta_N = \beta(\%)/(I/aB) \leq 2.5$].
- The plasma performance attainable in ITER is affected by many physics issues — power threshold for H-L transition, confinement uncertainties, operational (density and beta) limits, He removal, and impurity content are potential performance limiting issues.
- The provision of 100 MW of auxiliary power provides for the possibility of driven burn operations at $Q = 15$. This enables ITER to fulfill its mission of fusion power ($\sim 1\text{--}1.5$ GW) and fluence (~ 1 MWa/m²) goals if physics (confinement, impurity levels, density/beta limits, etc.) prove to be less favorable than present projections.
- Improvement of the helium exhaust ($\tau_{He}^*/\tau_E \leq 5$) and potential operation in reverse-shear mode significantly improve ITER performance.

ITER Objectives

- “Ignition and controlled burn in DT plasmas,...
...with steady-state as an ultimate objective”

 $t_{\text{burn}} (\text{ignited}) \geq 1000 \text{ s}$; $Q_{\text{SS}} \geq 15$; H&CD capability (to 100 MW)
- “Integrated demonstration of technologies necessary for a fusion reactor”

SC magnets, steady-state plasma heating/fueling,
in-vessel shield/blanket,
maintainable divertor and remote handling capability,
high operational reliability
- “Testing of prototype nuclear and in-vessel high-heat-flux components”
 $\Phi_n = 0.2 \rightarrow 1 \text{ MWa/m}^2$; permanent components = 3 MWa/m^2

These objectives and the related physics, technology and cost constraints
define the ITER design

ITER Interim Design

<u>Nominal Parameters</u>	<u>Design Features</u>
<ul style="list-style-type: none"> • $P_{fus} = 1.5 \text{ GW (+ 20\%)}$ • $R = 8.14 \text{ m}$, $a = 2.80 \text{ m}$ ($A = 2.91$) • $k = b/a = 1.75$ (1.60 at 95% flux); SN divertor plasma configuration • $B = 5.68 \text{ T}$ (12.5 T at coil); $\leq 2\%$ ripple • $I = 21 \text{ MA}$ ($q_{95} \geq 3.0$ at end-of-burn) • H/CD options: EC, IC, LH, NBI • PH/CD = 100 MW (two or more systems, selected by end of 1996) • $t_{burn} = 1000 \text{ s}$; $t_{rep} \approx 2200 \text{ s}$ (40% duty cycle) • $N_{pulse} = 10^5$; end-of-life fluence (permanent components): 3 MWa/m^2 • Availability = 10 – 25% 	<ul style="list-style-type: none"> • Bucked TF/CS construction for maximum B_T and $\Delta\phi_{PF}$; $N = 20$ • Pancake-wound TF with case; nearly bending-free outer leg • SN divertor: $l_{channel} \sim 2 \text{ m}$, modular cassette configuration • Modular in-vessel shield (water-cooled) with separable FW; replaceable with breeding blanket • Toroidally-continuous resistive VV; 5 bar containment • 7-coil PF system; robust plasma position and configuration control by ex-vessel PF coils

Representative ITER Device and Nominal Plasma Parameters

Parameter	Symbol	Value
Major radius	R	8.14 m
Minor radius	a	2.80 m
Plasma configuration	—	Single-null
Plasma elongation	κ_{95}, κ_x	$\sim 1.6, \sim 1.75$
Plasma triangularity	δ_{95}	~ 0.24
Nominal plasma current	I	21 MA
Toroidal field	B	5.68 T (at $R = 8.14$ m)
MHD safety factor	q_{95}	3.05
Fusion power (nominal)	P_{fus}	1.5 GW
Average wall loading	Γ_n	~ 1 MW/m ²
PF flux swing	$\Delta\phi_{\text{PF}}$	530 Wb
Flux swing for burn	$\Delta\phi_{\text{burn}}$	84 Wb

Plasma Performance and Physics Considerations

- Obtaining sufficient plasma performance is critical to meeting the ITER objectives, and to providing the physics database needed for future magnetic fusion reactors
- The key ITER objective is sustained burn: demonstration of time-independent burn conditions with a fully-equilibrated current profile, stationary levels of He and other impurities, stable fusion power and steady-state plasma-facing-component conditions
- The ITER objectives specify ignited operation as a goal, and the ITER design parameters are chosen to provide a credible chance, subject to cost constraints, of achieving an ignited/sustained burn.

However, driven-burn operation ($Q \geq 15$) provides a major means for offsetting physics or engineering shortfalls, and will still allow ITER to achieve its physics and technology testing objectives

- Operation with finite auxiliary power may also be needed for profile or rotation control, and will be needed for extended-pulse or true steady-state operation

Physics Guidelines for Plasma Performance Assessment

Confinement:

Plasma energy confinement must be sufficient to achieve ignition and sustained burn at fusion power of ~ 1.5 GW, under empirical scalings for ELMy H-mode plasma energy confinement and corresponding He and impurity concentrations ($Z_{\text{eff}} \sim 1.5$, $n_{\text{DT}}/n_e \sim 0.6$). Reference assumptions/models are as follows:

- Energy confinement: ELMy H-mode

$$\tau_E(\text{reference}) = \tau_E(\text{ELMy H-mode}) = H_H \times [0.85 \times \tau_E(\text{ELM-free H-mode}) \\ \sim H_L \times \tau_E(\text{ITER89P L-mode})]$$

$$\tau_E(\text{ELM-free H-mode}) = \tau_E(\text{ITER93H}) = \tau_{E-93H} \\ = 0.053 I_{1.06} R_{1.9} a^{-0.11} n_{20}^{0.17} B_{0.32} A_i^{0.41} k_x^{0.66} P^{-0.67}$$

$H_H = 1$: Reference ELMy H-mode, $H_H < 1$: degraded H-mode

- Particle confinement: ELMy H-mode with $\tau_{\text{pHe}}^*/\tau_E = 10$.
- Impurity content: Beryllium $n_{\text{Be}}/n_e = 2\%$
- Auxiliary power: $P_{\text{aux}} \geq P_{\text{thr}}(\text{L-H threshold power})$

Profiles:

Profile information is required to assess ITER performance margins with sufficient accuracy. Dynamic evolution of the profiles is also required to:

- develop operation scenarios,
- study stability and control of the fusion power
- assist design of ITER in particular PF system by specifying plasma wave form and perturbations associated with various plasma events (sawteeth, ELMs, ..)

- Reference Profiles: Flat

Typical ITER temperature and density profiles

Experimental: $n_z(r)/n_e(r) \sim \text{constant}$. No preferential peaking of low-Z impurity or He.

Beta limit: $\beta_{\max}(\%) = \beta_N (I/aB)$

β_N -- reference maximum normalized beta toroidal (thermal + fast): 2.5

Current profile control is possible for steady state, advanced scenarios (with reversed shear), resulting in a potentially higher β_N -value (see note).

Safety factor: $q_{\psi 95} \geq 3.0$

Note: Confinement partly deteriorates and disruptivity increases with low- q operation. For low- q , experimentally observed effective stability limit is $\beta_N/q_{\psi} \sim 0.7\text{--}0.9$, indicating that ITER parameters ($\beta_N/q_{\psi} \sim 2.5/3$) may be marginal. Thus, it would be difficult to attain high- β_N values (>3) without raising $q(0)$ and q_{ψ} substantially.

Density limit:

Borass density limit for a SN divertor configuration:

$$n_{20} = (n_e/n_{es}) C [Q_{\perp}^{5/8} B^{5/16} (1 - f_{\text{rad}^{\text{div}}})^{11/16} / (q_{\psi} R)^{1/16}]$$

with n_{es}/n_e ($\sim 0.4\text{--}0.7$) depends on particle transport and fueling at the plasma edge.

Greenwald density limit (tokamak operational experience with gas puff fueling):

$$n_{20}^{\text{GR}} = \kappa J(\text{MA/m}^2) = I/(\pi a^2)$$

H-mode power threshold:

- Transition from L-mode to H-mode is reached above a certain threshold power ($P_{\text{sep}} > P_{\text{thr}} = P_{\text{L-H}}$) that is observed to depend on plasma and device parameters.
- A reverse H-to-L-mode transition will occur when the power crossing the separatrix falls below a value roughly half of $P_{\text{L-H}}$

* L-H transition:

'unfavorable' $P(\text{L} \rightarrow \text{H}) \sim 0.044 n_{20} B S$

'unfavorable' $P(\text{L} \rightarrow \text{H}) \sim 0.3 n_{20} B R^{2.5}$ (close to $0.044 n_{20} B S$)

'favorable' $P(\text{L} \rightarrow \text{H}) \sim 0.016 n_{20}^{0.75} B S$

'favorable' $P(\text{L} \rightarrow \text{H}) \sim 0.036 n_{20} B^{0.6} S$

ITER projections at $n_{20} \sim 0.5$: $P(\text{L} \rightarrow \text{H}) \sim 60 \text{ MW}$ for 'favorable'
 $\sim 150 \text{ MW}$ for 'unfavorable'

* H-L transition:

$$P(\text{H} \rightarrow \text{L}) \sim 0.5 P(\text{L} \rightarrow \text{H})$$

Four Indicators of Plasma Performance — Margins

- Confinement margin:

$$M_{\text{conf}} = [\langle n_{\text{DT}} \rangle \langle T \rangle \tau_{\text{E}}]_{\text{predicted}} / [\langle n_{\text{DT}} \rangle \langle T \rangle \tau_{\text{E}}]_{\text{min}}$$
$$= \frac{HH=1 \text{ from Physics Rules}}{\text{Minimum HH factor required to satisfy power balance}}$$

For $P_{\text{aux}} = 0$, $P_{\text{fus}} = 1.5$ GW this corresponds to 1.5 GW ignition margin.

- Beta-limit margin:

$$M_{\beta} = 2.5/\beta_{\text{N}} \quad [\beta_{\text{N-max}} = 2.5]$$

- Burn duration margin:

$$M_{\text{burn}} = t_{\text{burn}}(\text{s})/1000 \quad (t_{\text{burn}} = \phi_{\text{burn}}/V_{\text{OH}})$$

- H- to L-mode transition margin:

$$M_{\text{H-L}} = P_{\text{sep}}(\text{MW})/[0.5 \times P_{\text{L-H}}] = P_{\text{sep}}(\text{MW})/[0.5 \times 0.044 \text{ B(T)} n_{20} \text{ S(m}^2\text{)}]$$

Interim Design Reference Case

with $\tau_E = 0.85 \times \tau_{E-93H}$ ($HH = 1$), $\tau_{He}^* / \tau_E = 10$, $n_{Be}/n_e = 2\%$

POP CON diagram for reference ignition scenario.

Ignition ($P_{aux} = 0$), beta limit ($\beta_N = 2.5$), 'unfavorable' H-L transition and $P_{fus} = 1.5 \text{ GW}$ contours are shown.

ITER Plasma Parameters for Ignited Burn with “Reference” Modeling Assumptions

Parameter	Symbol [unit]	Value
Volume-average temperature	$T = 0.5(T_e + T_i)$ [keV]	10.5
Volume-average density	n_e [10^{20} m^{-3}]	1.3
Impurity fractions	$n_{Be}/n_e, n_{He}/n_e$ [%]	2, 14
Effective charge	Z_{eff}	1.5
Radiated power fraction	$P_{rad}/P_\alpha = f_{rad}$	0.36
Confinement time	τ_E [s]	6.2
Normalized confinement	$H_H = \tau_E/0.85 \tau_{E93H}$	1.0
Normalized beta (total)	β_N	2.4
Poloidal beta (total)	β_{pol}	0.9
DT Triple product	$\langle n_{DT} T \tau_E \rangle$ [$10^{21} \text{ m}^{-3} \text{ keV s}$]	3.3
Internal inductance	$l_i(3)$	0.9
Loop voltage	V_{loop} [mV]	72
Burn duration	t_{burn} [s]	1160
Plasma thermal energy	W_{th} [GJ]	1.1
Plasma magnetic energy	W_{mag} [GJ]	1.2

Physics Margins for ITER Reference Operational Point (1.5 GW, ignited scenario)

Margin	Value
M_{conf}	1.11
M_{β}	1.04
M_{burn}	1.09
$M_{\text{H-L'unfavorable'}}$	1.06

In above table, all margins are above 1, showing that the plasma parameters projected for ITER provide sufficient confinement, remain below the beta limit, provide sustained burn, and are compatible with H-mode operation.

Summary POPCONS for ITER

with $\tau_E = H_q \times 0.85 \times \tau_{E-93H}$, $\tau_{He}^* / \tau_E = 10$, $n_{He}/n_e = 2\%$

Estimated Confinement Margin is Sensitive to the Following Rules

Edge temperature profiles:

Flat H profile: $M_{\text{conf}} = 1.11$

Peak L profiles: $M_{\text{conf}} = 1.22$

Auxiliary heating	Impurity level
50 MW: $M_{\text{conf}} = 1.23$ 100 MW: $M_{\text{conf}} = 1.31$	Lower Helium level: $\tau_{\text{He}}^* / \tau_{\text{E}} = 5$: $M_{\text{conf}} = 1.22$ Higher Beryllium level: $n_{\text{Be}} / n_{\text{e}} = 3.7\% (0.7 \cdot 10^{19} \text{ m}^{-3})$ $M_{\text{conf}} = 1.00$

The minimum H_{H} factor required to sustain 1.5 GW of fusion power with 100MW of auxiliary heating ($H_{\text{H}} = 0.76$) corresponds to [with respect to ITER89-P] $H_{89\text{P}} = 1.8$.

Simulations with $\tau_E = H_H \times [0.85 \text{ ITER93-H mode scaling}]$
with varying values of H_H

Performance with Greenwald Density Limit

- $\langle n_e \rangle$ limited to $0.85 \times 10^{20} \text{ m}^{-3}$

At the H->L-mode transition:
(0.022 n B S 'unfavorable' scaling)

- Fusion power: $\sim 0.8 \text{ GW}$
- $P_{\text{aux}} = 30 \text{ MW}$ ($Q_{\text{th}} \sim 27$)

At the maximum additional power:

- 100 MW auxiliary heating
- Fusion power: $\sim 1.3 \text{ GW}$ ($Q_{\text{th}} \sim 13$)

Notes:

- Greenwald density limit likely related to fueling: neutral recycling and particle fueling in ITER different than experiments.
- Greenwald density limit overcome by Pellet fueling (ref. ASDEX -U)
- Greenwald density limit overcome if we manage to lower the impurity level:

$$\langle n_e \rangle = 8.8 \times 10^{20} \text{ m}^{-3} \text{ at } \tau_{\text{He}}^* / \tau_E = 5$$

ITER Backup Scenarios/Cases under Alternative Assumptions

Change in Physics rules vs [$P_{fus-max}/P_{aux}$]

	Optimistic	Nominal	Pessimistic
Confinement		$H_H = 1$ [ignited--1500 / 0]	$H_{89P} = 1.8$ [1500/100]
Density limit		Borrass [ignited--1500 / 0]	Greenwald [1300 / 100]
Beta limit	$\beta_N > 2.5$ [ignited--1500 / 0]	$\beta_N = 2.5$ [ignited--1500 / 0]	$\beta_N = 2$ [1050 / 14]
Additional confinement degradation at high density	20% improvement [ignited--1500 / 0]	0% [ignited--1500 / 0]	- 20% [1500 / 50]
Impurities (He, Be, Ne)	$\tau_{He}^* / \tau_E = 5$ Be: 2%, Ne: 0% [ignited--1500 / 0]	$\tau_{He}^* / \tau_E = 10$ Be: 2%, Ne: 0% [ignited--1500 / 0]	same + 0.5% Ne [1500 / 37]
H-L power threshold	$P \sim n^{0.75}BS$ [ignited--1500 / 0]	$P \sim nBR^{2.5}$ [ignited--1500 / 0]	
Profiles	peak [ignited--1500 / 0]	flat [ignited--1500 / 0]	

Changed one assumption at a time, while keeping others at nominal value: shown for each assumption are maximum fusion power produced (constrained to 1500 MW max) with 100 MW or less aux power.

Pessimistic Physics rules vs [$P_{\text{fus-max}}/P_{\text{aux}}$]

Change two (2) assumption at a time: for each, maximum fusion power produced (constrained to 1500 MW or less) with 100 MW or less auxiliary power

	$H_H = 1.8$	Greenwald limit	$\beta_N = 2$	20% conf degradation	+0.5% Ne
$H_H = 1.8$	1500 / 100 Q = 15	850 / 100 Q = 8.5	1300 / 100 Q = 13	500 / 100 Q = 5	900 / 100 Q = 9
Greenwald limit		1300 / 100 Q = 13	900 / 30 Q = 30	1100 / 100 Q = 11	1100 / 100 Q = 11
$\beta_N = 2$			1050 / 14 Q = 75	1400 / 40 Q = 35	1000 / 35 Q = 28
20% conf degradation				1500 / 50 Q = 30	1300 / 100 Q = 13
+0.5% Ne					1500 / 37 Q = 40

If one assumption changed [from nominal to pessimistic], ITER would still fulfill its mission of fusion power (1-to-1.5 GW), but would have to be operated in driven mode with $Q \sim 15$.

If two assumptions changed simultaneously [from nominal to pessimistic], but fusion power can fall significantly below its mission for strong confinement degradation ($H_{89P} = 1.8$ plus 20% reduction, corresponding to $H_{89P} \sim 1.5$) or combination of $H = 1.8$ and Greenwald density limit. For most other combinations $Q > 10$.

Long Pulse Scenario

- Reduction of plasma current
- Current drive with 100 MW input power.
($\gamma = 0.2 \times 10^{20}$ MA/MW/m²)
- Fusion power input adjusted to satisfy β -limit ($\beta_N < 2.5$)
- H-mode power threshold always satisfied
- Max. pulse length is ~4000s
@ 1 GW and
 $I_p \sim 15$ to 17 MA

$$H_H = 1.0, \quad \beta_N (= g_{\text{Troyon}}) < 2.5$$

- 100 MW current drive
- $\beta_N < 2.5$
- Steady state non-inductive operation, i.e. fluence = ∞ , requires $H_H > 1.25$

ITER is Compatible with Non-Inductive Operation at 12 MA [Reverse Shear/Advance Scenario]

- High- β_p plasma:
 $\kappa \sim 1.9$
 $\delta \sim 0.4$
- PF control similar to reference ($m_{stab} \sim 2$)
- MHD stable for $\beta_N < 5$ @ wall ($r/a \sim 1.25$)

ITER Reverse-Shear Configuration

Optimised pressure profiles:

Well aligned currents:

Reverse Shear/Advance Scenario Parameters/Profiles

R_o (m) / a (m)	~8.5 / ~2.4
κ_{95}/δ_{95}	1.92/0.37
B_o (T)	5.39
q_{ψ}	4.70
I_p (MA)	12.0
P_{fusion}/P_{CD} (MW)	1500/100
Q_{CD}	15
f_{BS} (%)	78
γ_{cd} (1020MA/MW/m ²)	0.20
β_N/β_{N^*} (%)	3.96 / 4.52
$\langle n_e \rangle / n_{eo}$ (1020m ³)	1.0 / 1.05
$\langle T_e \rangle_n / T_{eo}$ (keV)	12.7 / 27.4
τ_E (s)	2.44
$\tau_E/\tau_{ITER93-H}$	1.28
$\tau_E/\tau_{ITER89-P}$	2.33
τ_{He}^*/τ_E	10

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