PHYSICS DESIGN GUIDELINES FOR ESTIMATING PLASMA PERFORMANCE IN A BURNING PLASMA EXPERIMENT (FIRE)

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ABSTRACT

The physics design guidelines for a next step, high-field tokamak, burning plasma experiment (FIRE, Fusion Ignition Research Experiment) have been developed as an update of the ITER Physics Basis (IPB). The plasma performance attainable in FIRE (or any next-step device) is affected by many physics issues, including energy confinement, L-to-H-mode power transition thresholds, MHD stability/beta limit, density limit, helium accumulation/removal, impurity content, sawtooth effects, etc. Design basis and guidelines are provided in each of these areas, along with sensitivities and/or uncertainties involved. The overall basic device parameters and features for FIRE (R = 2 m, a = 0.525 m, κ_{95} ~ 1.8, δ_{95} ~ 0.4, q_{95} > 3, B = 10-12 T, I = 6.45-7.7 MA, P_{fus} ~ 100-200 MW, Q ~ 5-10) are consistent with these guidelines and uncertainties if the potential design upgrade option (12 T, 8 MA) is considered as part of the main design option.

I. INTRODUCTION

The physics design guidelines for a next step, high-field tokamak, burning plasma experiment (FIRE, Fusion Ignition Research Experiment) have been developed as an update of the ITER Physics Basis (IPB) [1,2]. The IPB [1] represented a comprehensive account of the scientific knowledge, as of mid-1998, relevant to the design of a reactor-scale tokamak. Physics design guidelines and methodologies for projecting plasma performance in FIRE or in any next-step/reactor-scale tokamaks are developed from extrapolations of various characterizations of the IPB database for tokamak operation and of the understanding that its interpretation provides [2].

FIRE [3] is envisioned as a near-term affordable burning plasma experiment that has enough flexibility to explore both conventional tokamak and advanced tokamak burning plasma physics. The overall device parameters are derived from the performance goals. The plasma performance attainable in FIRE (or any next-step device tokamak) is affected by many physics issues, including energy confinement, L-to-H-mode power transition threshold, MHD stability/beta limit, density limit, helium accumulation/removal, impurity content, sawtooth effects, etc. In addition to plasma performance determining physics issues, physics issues associated with disruptions, plasma control, and power and particle handling all impact plasma operation reliability and lifetime of in-vessel, torus vessel, and supporting subsystems. Design basis and guidelines are provided in each of these areas, along with sensitivities and/or uncertainties involved. Both conventional and advanced tokamak operating modes are considered and corresponding guidelines are summarized in the sections that follow.

The overall basic device parameters (Table I) considered for FIRE [3,4] are consistent with these guidelines and uncertainties if the potential design upgrade option (12 T, 8 MA) is considered as part of the main design option. Recently, a new physics performance point [4], FIRE*, has been identified by the design team (R/a = 2/0.565, κ_{95}/δ_{95} ~ 1.8/0.5, 10 T, 7.7 MA, ~125 MW of fusion power) and the engineering feasibility is being evaluated.

Table I. Representative FIRE parameters

Parameter	Symbol	Value
Major radius	R	2 m
Minor radius	a	0.525 m
Plasma configuration	—	Double-null
Plasma elongation	κ_{95}	~1.8
Plasma triangularity	δ_{95}	~ 0.4
Nominal plasma current	I	6.45 (7.7) MA
Toroidal field	B	10 (12) T (at R=2 m)
MHD safety factor	q_{95}	≥ 3.0 (at 6.45 MA)
Fusion power (nominal)	P_{fus}	100-200 MW
Burn duration	t_{burn}	~ 20 (12) s
Effective charge [with 3% Be+He($5\tau_E$)]	Z_{eff}	~1.4
Auxiliary heating power	P_{aux}	20-30 MW
Fusion gain, $Q = P_{fus}/P_{aux}$	Q	~ 5-10

II. SUMMARY OF PHYSICS DESIGN GUIDELINES — CONVENTIONAL TOKAMAK OPERATION

Here, the units are mks, MA, MW, with (κ,δ) average values of elongation and triangularity at 95% flux surface, and n_{20} = $n_e/10^{20}$ m^{-3} = line-average electron density [or $n_{19} = n_e/10^{19}$], T_{10} = T/10 keV = average temperature ($T \approx T_e \approx T_i$), M = A_i = ave. atomic mass. For simple estimates, profiles can be represented as: $n \sim (1 - r^2/a^2)^{\alpha_n}$, $T \sim (1 - r^2/a^2)^{\alpha_T} + T_{ped}$ with $\alpha_n \approx$ 0-0.1, $\alpha_T \approx$ 1-1.3 and T_{ped} ~ 2-3 keV = H-mode pedestal temperature, as nominal values. Note that $\alpha_n \approx$ 0-0.1 is consistent with IPB and H-mode profiles with gas or outside-launch pellet fueling in the absence of appreciable internal fueling (e.g., NBI), and that more peaked H-mode profiles, which will improve burn performance (Q), may arise with inside pellet launch (which an on-going R&D). Note that ELMy H-mode temperature profiles are characterized by a large edge temperature pedestal that needs to be included in fusion performance projections and the way one parameterizes the edge pedestal in various models (0-D or 1-D simulations) would impact the ‘effective’ value of α_T.

Configuration: Plasma size and basic device parameters are determined from both physics and engineering considerations. FIRE [3,4] is envisioned as a near-term affordable burning plasma experiment that has enough flexibility to explore both conventional tokamak and advanced tokamak burning plasma physics. The geometry of FIRE is based on a double-null (DN) poloidal divertor configuration (a design choice, not a physics requirement) with a fixed X-point. Plasma cross-section is strongly shaped, with average plasma elongation and triangularity:

$$\kappa(95\%) = \kappa_{95} \approx 1.8$$
$$\delta(95\%) = \delta_{95} \geq 0.4.$$

Internal control coils and wall stabilization capabilities should be part of the configuration design.

Energy Confinement: Plasma energy confinement must be sufficient to achieve fusion alpha heating dominated burn conditions ($Q = P_{fus}/P_{aux} = 5P_\alpha/P_{aux} > 5$), for times long compared to the plasma characteristic time scales ($t_{burn} \geq 20\tau_E$, $\sim\tau_{skin}$), under ELMy H-mode confinement and corresponding He and impurity concentrations. Three methods of extrapolations should be used in parallel: empirical (global) scalings (obtained from ITER H-mode database), dimensionless parameters scaling technique, and 1-D local transport models. The first one, empirical scalings, is commonly used in the design of next-step tokamak burning plasma experiments, and was the recommended method by the ITER confinement expert groups to extrapolate the energy confinement time [1]. This is because present-day experiments have attained values of the relevant dimensionless physical parameters that are sufficiently close to those needed in a next-step tokamak that empirical extrapolation is considered to be the most reliable method of projecting performance.

Empirical (global) thermal energy confinement scaling:

$$\tau_{E,th}(\text{ELMy H-mode}) = H_H \times \tau_E^{IPB98}(\text{database fit})$$

where

$$\tau_E^{IPB98} = C(10^{-2})\, I^{\alpha_I} B^{\alpha_B} P^{-\alpha_P} n_{19}^{\alpha_n} M^{\alpha_M} R^{\alpha_R} \varepsilon^{\alpha_\varepsilon} \kappa_a^{\alpha_\kappa}$$

Exponents for this empirical (log-linear, power law) scalings from H-mode database (ITERH.DB3) are:

Scaling	C	α_I	α_B	α_P	α_n	α_M	α_R	α_ε	α_κ
IPB98(y)	3.65	0.97	0.08	0.63	0.41	0.20	1.93	0.23	0.67
IPB98(y,1)	5.03	0.91	0.15	0.65	0.44	0.13	2.05	0.57	0.72
IPB98(y,2)	5.62	0.93	0.15	0.69	0.41	0.19	1.97	0.58	0.78
IPB98(y,3)	5.64	0.88	0.07	0.69	0.40	0.20	2.15	0.64	0.78
IPB98(y,4)	5.87	0.85	0.29	0.70	0.39	0.17	2.08	0.69	0.76

κ_a = plasma elongation = b/a in IPB98(y) and $\kappa_a = S/\pi a^2$, S = plasma cross-sectional area, in IPB98(y,1-4).

$P = P_\alpha + P_{OH} + P_{aux} - P_{rad\text{-}core}$, $P_{rad\text{-}core}$ = core (bremsstrahlung + synchrotron) radiation, and H_H = H-mode scale factor representing range of data and uncertainty. Improved H-mode conditions ($H_H > 1$) are possible in advanced tokamak modes, whereas operation near density and/or MHD stability limits yield $H_H < 1$. Amongst the five empirical log-linear scaling expressions, recommended ‘reference’ scaling is IPB98(y,2), representing a conservative option:

$$\tau_E(y,2) = 0.0562 I^{0.93} B^{0.15} P^{-0.69} n_{19}^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_a^{0.78}$$

Although this empirical confinement scaling does not fully satisfy the dimensional constraints of plasma physics, it is very close to a version that it does. However, this dimensionally correct version exhibits an explicit degradation of confinement with beta, at fixed values of the other dimensionless parameters, and this deterioration is generally not observed in (JET, DIII-D, C-mod β-scan) experiments except in cases where neoclassical tearing modes (NTMs) arise. So, in this sense, the scaling being recommended could be pessimistic if NTMs can be avoided and/or stabilized.

Dimensionless parameter scaling: Analysis of the H-mode database show that the dimensionless confinement time $\Omega\tau_E$ $B\tau_E$ $(\rho^*)^{-(2+\alpha)}$ with $\alpha \sim 1$ (close to gyro-Bohm) from database/ITER demonstration discharges (discharges with ITER's β_N and ν^*, except ρ^*) showing good agreement with global scaling predictions. Here, ρ^* is the normalized ion gyroradius (ρ_i/a), ν^* is the collisionality, β_N is the normalized plasma beta.

<u>1-D local transport models</u>: There is a range of transport models, both purely theoretical and semi-empirical. Many are successful in reproducing the core plasma properties, edge region ($0.9 < r/a < 1$) remains an active area of development. These models are being developed and tested against experimental data, and, at present, there is no reliable model that can be used in next-step and/or reactor-scale extrapolations describing the transport across the entire plasma profile.

Particle Confinement: Reference particle diffusivity (from experiments) is $D/\chi_i \sim 1$ with D independent of charge or mass for fuel, He, low-Z impurities. Range of uncertainty: $D/\chi_i \sim 0.3$–1 for design of fueling and pumping systems.

Impurity Content: Governed by a combination of beryllium (first wall): $n_{Be}/n_e \leq$ 2-3% and helium content determined from: $\tau^*_{He}/\tau_E \sim$ 4-10 should be considered with $Z_{eff} < 1.5$. *Nominal baseline case*: 3% Be and $\tau^*_{He}/\tau_E = 5$ can be considered for simple performance predictions. Note those small quantities of recycling of high-Z impurities from the divertor plates (tungsten) and/or those medium-Z (neon, argon) impurities added to promote pre-divertor radiation of thermal power could significantly increase Z_{eff} values. Important to maintain a balance between control of impurity sources and transport to maximize edge radiation while maintaining core cleanliness.

H-mode Threshold Power: Transition from L-to-H-mode is reached above a certain power threshold, P_{thr}, which depends on plasma conditions and machine size. The threshold power is about a factor of 2 lower for the single null (SN) configuration with the ion ∇B drift towards the X-point than for the opposite direction or double null (DN) configuration. The threshold is about a factor of 2 lower in deuterium than in hydrogen. Recent scaling expression for P_{thr}, obtained from the latest version of the ITER H-mode threshold database (DB3) is [5,6]:

$$P_{thr}(\mathrm{MW}) = 2.84\,(n_{20})^{0.58}\,B^{0.82}\,R\,a^{0.81}\,M^{-1}$$

The new database includes results from dedicated H-mode threshold experiments from C-mod and JT-60U. For FIRE (or ITER), this scaling yields a power threshold by a factor of two lower than that predicted by an earlier IPB(98) version [1.2]. Design basis guideline is:

$$P_L = P_{heat} - P_{tot\text{-}rad}(\mathrm{core}) - \partial W/\partial t \geq 1.3 \times P_{thr}$$

where

P_L = power crossing the separatrix,
P_{heat} = net (alpha+OH+auxiliary) heating power
$= P_\alpha + P_{OH} + P_{aux}$,
$P_{tot\text{-}rad}$(core) = total radiated power inside separatrix.

Beta Limit: MHD stability plays a defining role in determining the accessible parameter space and relate to maximum plasma pressure, $\beta_{max}(\%) = \beta_N\,(I/aB)$

$\beta_N \leq 4l_i$	Ideal MHD limit
$\leq f(\nu^*,\rho^*) < 2.5$	'neoclassical' modes
≤ 2.5	nominal baseline

Potential limitation in β_N (and degraded energy confinement) is typically correlated with the growth of neoclassical tearing modes (NTMs) associated with depletion of bootstrap current. The NTM β-limit is not hard (soft beta limit or beta saturation) and prospects for stabilization with electron cyclotron current drive (ECCD), or potentially with lower hybrid current drive (LHCD), are being tested in experiments. Because of the frequency requirements, ECCD may not be feasible in FIRE or other high-field next-step devices.

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

Sawtooth: Sawtooth activity is associated with an instability at q = 1 surface, resulting in a flattening of the central plasma profiles. Although little impact on global plasma performance is foreseen, sawteeth may produce seed islands needed to trigger neoclassical tearing modes. For next-step/reactor-scale plasmas, typical IPB characterization is as follows:

mixing radius:	$r_{mix}/a \leq 0.5$
min. central q:	$q(0)_{min} \geq 0.8$
period & range:	tbd (to be determined)

ELMs: Edge localized modes are instabilities of the plasma edge associated with H-mode confinement. ELMs result in regular relaxations of the edge temperature and density and limit the maximum edge pressure gradient within the edge pedestal region.

energy:	$\Delta W_{th}/W_{th} \approx 3\%$
frequency:	$f_{ELM} \approx 1.4$ Hz (range 4–0.5 Hz]

Density Limit: For gas fueling and ohmic and auxiliary heating, density limit is usually described by the Greenwald value, $n_{GR}\,(10^{20}\ \mathrm{m}^{-3}) = I(\mathrm{MA})/\pi[a(\mathrm{m})^2]$. Design basis guideline:

$$n_{20} \leq 0.75 \times n_{GR}$$

There are two limits. Low-density limit defines the heating power required to get the H-mode (L-H transition). This limit is disruptive and associated with the error-field induced instability. At high density, typically, approaching the Greenwald value leads to loss of H-mode confinement and H-to-L back transition. Experimental results indicate that inside (high-field) pellet launch can promote improved penetration and fueling efficiency and good confinement up to $1.5 \times n_{GR}$.

Power and Particle Control: Power and particle control is central to successful operation of tokamak reactor plasma. Experimental database and comprehensive divertor modeling codes provide the framework for extrapolations. FIRE specifications: double null (design choice) detached divertor with $\leq$ 10 MW/m^2 peak heat loads on the divertor plates. Radiating most of the heating power to first wall and divertor chamber walls during detached divertor operation will reduce the power. The radiation is distributed among the central plasma, edge plasma, SOL and divertor. FIRE example for distribution of plasma radiation losses and the power loads on the divertor plates can be found in Ref. [3].

Disruptions: Disruptions and their consequences significantly impact the design and operational planning for next-step DT burning plasmas. Plasmas with sufficient performance to achieve DT burn also have enough thermal (W_{th}) and magnetic (W_{mag}) energy to put in-vessel (plasma facing components, PFCs) and torus vessel systems at risk from disruptions and/or loss of vertical equilibrium control (vertical displacement events, VDEs). Disruption and disruption-related design basis recommendations (adapted from IPB) are as follows:

- frequency: 10% (range: 10-30%) per pulse,
  30% for plasma development;
  ≤ 10% for repetitive operation
- number: 10% at full W_{th}, W_{mag}
  balance at $\leq 0.5 \times W_{th}$ and full W_{mag}
- thermal quench time: $\tau_{TQ} \sim 0.2$ (range: 0.1–0.5) ms
  single or multiple step
  thermal quench (TQ)
- thermal energy: $W_{th} \approx 33$ MJ for 200 MW baseline
- W_{th} distribution: 80–100% to divertor by conduction
  up to 2:1 toroidal asymmetry
  ≤ 30% to first wall (FW)
  by radiation or
  by conduction to baffle
- in-divertor partition (inside/outside)-in/out split:
  SN~1.1 (2:1-1:22);
  no DN data
- up/down split for DN: depends on symmetry;
  design basis assumption 1:1-2:1
- poloidal localization in divertor: 3×normal SOL;
  (range: 1× - 10×) incident energy
  up to 2:1 toroidal asymmetry,
  plasma shielding & re-radiation
  likely redistribute in-divertor energy.
- magnetic energy: $W_{mag} \approx 35$ MJ for 6.5 MA baseline
- current quench time: $\tau_{CQ} \geq 6$ ms (range: 2-600 ms)
  ≥30ms: more-severe VDE & halo current
- max. current decay rate: 3 MA/ms fastest part of CQ,
  typical max rate ~ 1MA/s
- W_{mag} distribution: 80–100% to FW by radiation
  with poloidal peaking factor ~2
  0–20% to FW by conduction

Vertical Displacement Events (VDEs):

- frequency: tbd (1% of pulses, or 10% of disruptions)
  presently uncertain, may be able to maintain
  vertical position control after TQ.
  control failure result in VDE or
  loss of after-TQ control
- max halo current: $I_{h,max}/I \approx 0.4$ (range: 0.01-0.5)
  highest value may apply
  (depends on passive stabilizer)
- toroidal peaking factor: 2 ($1.2 \leq$ TPF ≤ 4)
  TPF≤2 yields 'sinφ' distribution;
  TPF>2 yields 'localized filament'
- halo fraction × TPF: 0.5 typical [≤0.75 max.]
  note: data shows inverse TPF–halo fraction correlation

Runaway electrons:

- runaway current: ~0.5 × I (range: 0-50%)
- energy: 15 MeV, limited by knock-on avalanche
- localization of runaway deposition: ≤ 1 m2
  poloidal: ~0.1m of FW, or divertor target?
  toroidal: depends on PFC and FW alignment
  to toroidal field

Poloidal Field Capability:

- breakdown near the limiter in the equatorial port
- separatrix control with $0.7 \leq l_i \leq 1.1$ and $\beta_p \leq 1.2$
- more than 20 s burn
- operation at large scale plasma disturbances:
  $\Delta l_i = -0.1$, $\Delta\beta_p = -0.2$

Toroidal Field Ripple: $\delta_{TF} = \Delta B/B \leq 0.5\%$
Here, $\delta_{TF} = (B_{max} - B_{min})/(B_{max} + B_{min})$ = peak-to-average field ripple amplitude.

III. ADVANCED TOKAMAK (AT) PHYSICS RULES

Since the preparation of the IPB, the world tokamak program continued to make steady progress in improving tokamak performance and understanding. Advanced tokamak (AT) modes broadly refer to tokamaks operating in steady state (or for long pulses, long compared to many plasma characteristic times) with significantly improved performance and substantial bootstrap current (hence high poloidal and normalized beta, β_N). Examples of AT modes include: internal transport barriers, pellet enhanced performance, high-beta poloidal mode, D-alpha (or enhanced D-alpha) mode, LH enhanced performance, high-l_i mode, edge radiation enhanced RI-mode, reverse shear, negative central shear, etc. However, next-step/reactor plasma relevance and 'uncertainty' are the issues. At present, all these modes are transitory and are not yet achieved with all the relevant dimensionless parameters simultaneously. The international tokamak community striving toward such a goal and one of the missions of FIRE (and ITER) is to explore such AT modes in alpha-heating dominated regime.

At present, tokamak facilities are implementing radio-frequency (RF) current-profile control (utilizing bootstrap current), plasma density control (e.g., divertor and fueling), and exploring ideas for internal transport barrier control (e.g., RF). Within the next couple of years, the following advanced physics rules may become plausible:

Confinement: $H \le 1.4$; $\tau^*_{He}/\tau_E < 5$

Operational Limits [q, β_N, n]:

$q_{95} \ge 4$ with $q(0)$, $q(min) \ge 1.5$

$\beta_N \le 3.5$–4 50% better than baseline

$\langle n \rangle \le 1.5 \times n_{GR}$ range: $n/n_{GR} \sim 1$–1.5

Plasma current profile control: off-axis RF

Other physics rules remain the same as conventional tokamaks.

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