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Physics Design of ITER-FEAT

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Abstract

The physics design of ITER-FEAT is outlined. ITER-FEAT is a tokamak experimental reactor capable of extended DT burn. The goal is to achieve Q (= output power/input power) of at least 10 for 300–500 s in inductive operation. The projection studies show that this mission is achievable in ELMy H-mode according to the methodologies described in ITER Physics Basis document. The main issue is to establish an operation scenario with good confinement ($H_H \sim 1$) at densities ~85% of Greenwald density with sufficiently low ELM amplitudes. If the confinement properties are favourable, controlled ignition will also be investigated. ITER-FEAT also aims at steady-state operation with Q of at least 5. An extended burn (> 1000 s) satisfying Q = 5 is feasible at modest confinement ($H_H \leq 1$) and beta ($\beta_N \leq 2.5$) at a plasma current of 12 MA and current drive power of 100 MW in a hybrid operation where ohmic current and non-inductive current are combined. Steady-state operation will also be possible in weak or reversed shear operation with modest current (8–9 MA), which will require challenging values of H_H (~1.5) and β_N (3.2–3.5). Investigation of such an operation will be facilitated with the flexibility in shaping, heating/current drive (NB, EC, IC and/or LH) and advanced features of ITER-FEAT, including ECCD for stabilising neoclassical tearing modes, saddle coils for stabilising resistive wall modes and profile diagnostics.

Keywords:

ITER, physics design, burning plasma, confinement, ELMy H-mode, non-inductive current drive, steady state, divertor, bootstrap current, helium exhaust

1. Introduction

The goal of ITER-FEAT is to obtain an extended DT burn in inductively driven plasma with $Q = P_{fus}/P_{aux} \geq 10$ and average neutron wall load of ≥ 0.5 MW/m^2 at a pulse length of 300–500 s which is sufficient to achieve stationary conditions on the time scale of the plasma processes. ITER-FEAT aims at demonstrating steady-state operation using non-inductive current drive with a Q value of at least 5. Also the regimes of controlled ignition will be investigated if the confinement properties are favourable at high densities. This paper outlines the physics design of ITER-FEAT under consideration [1].

The key physics issues relating to plasma performance in the ELMy H-mode regime are the maintenance of H-mode quality confinement at sufficiently high density, achieving adequate plasma β to produce the requisite fusion power, and hence Q value, the provision of satisfactory power and particle

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exhaust to ensure acceptable levels of helium and plasma impurities, and the demonstration of an efficient transfer of α-particle power to the thermal plasma while limiting anomalous α-particle losses, via TF ripple or collective instabilities, to prevent damage to the plasma facing components. At the same time, global magnetohydrodynamic (mhd) stability and plasma control capability must be such that the thermal and electromagnetic loads, as well as runaway electron currents, arising from disruptions are within acceptable bounds.

The physics rules established during ITER-EDA and described in ITER Physics Basis document [2] enabled projection of existing tokamak performances to ITER-FEAT. Table 1 lists the main parameters of ITER-FEAT and Fig. 1 shows the poloidal cross-section of the machine. To increase beta and density with good confinement in ITER-FEAT, relatively high triangularity and higher elongation are chosen. An increase in B/R implies operational densities lower than the Greenwald density. The relatively high triangularity, high elongation and high aspect ratio is beneficial for steady-state operation. ITER-FEAT is capable of operating with flexibility in shaping and in heating and current drive methods (33 MW NB, 20 MW EC, and 20 MW IC, totaling 73 MW initially and 100 MW full power). Fig. 2 shows a plan view of the neutral beam lines. Initially, two neutral beam lines at 1 MeV and 16.5 MW each are installed. The figure also shows a diagnostic neutral beam injector which shares a port with one neutral beam injector, and a space for a third neutral beam injector. The tangency radius of the neutral beam is 5.28 m. The vertical spread of the beam line is

Table 1. Main parameters of ITER-FEAT

Parameter	Unit	Nominal	Max
Major radius, R	m	6.2	<=
Minor radius, a	m	2.0	<=
Plasma current, I_p	MA	15.0	17.0
Additional heating & CD power	MW	73	100
Fusion power	MW	500	700
Toroidal field at major radius, Bo	T	5.3	<=
Elongation at 95% flux, κ_{95}, elongation at the separatrix, κ_X		1.7, 1.85	<=
Triangularity at 95% flux, δ_{95}, triangularity at the separatrix, δ_X		0.33, 0.49	<=
Plasma volume	m^3	837	<=
Plasma surface	m^2	678	<=
MHD safety factor at 95% flux, q_{95}		3	2.6
Average neutron wall load at the first wall	MW/m^2	0.57	0.80

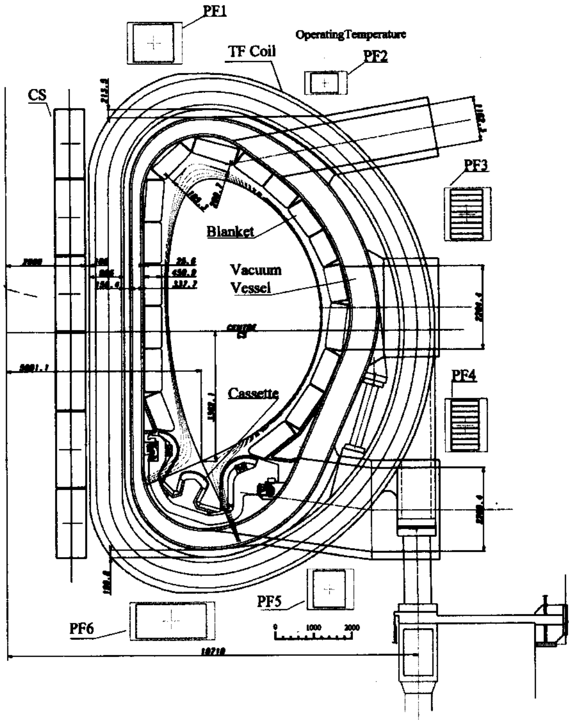

Fig. 1 Cross section of ITER-FEAT

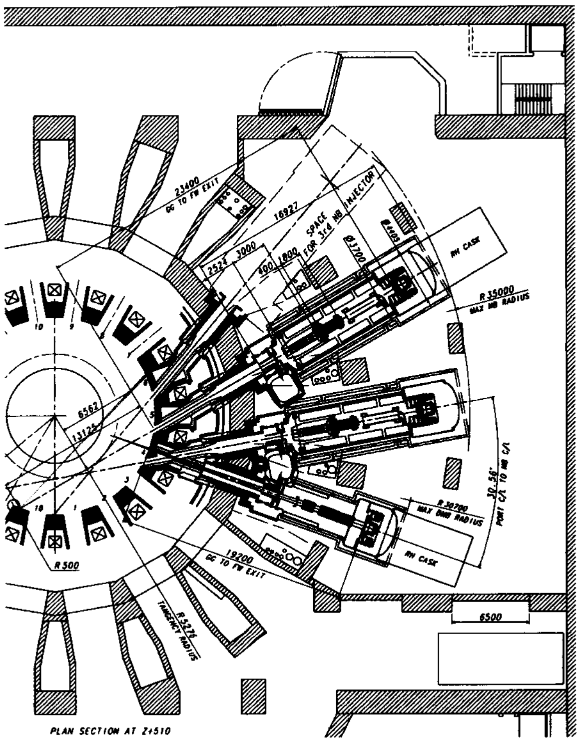

Fig. 2 NB system layout - plan view

± 0.3 m. Each neutral beam line can be independently tilted vertically from –0.9 m to –0.3 m from the plasma midplane at the tangency point. The diagnostics neutral beam injector is installed for charge-exchange recombination spectroscopy.

ITER-FEAT is also capable of investigating advanced operation modes such as weak or reversed shear modes with its ECCD for stabilising neoclassical tearing modes, saddle coils for stabilising resistive wall modes, and profile.

2. Performance Prediction

The reference plasma scenario for inductive Q = 10 operation, the ELMy H-mode, is a reproducible and robust mode of tokamak operation with a demonstrated long-pulse capability. The essential physics which enters into the prediction of plasma performance in ITER-FEAT therefore derives from the two principal ELMy H-mode scalings, i.e. the H-mode power threshold scaling [3], which defines the lower boundary of the device operating window in terms of fusion power, and the energy confinement time scaling. The recommended form for the former scaling is,

$$P_{LH} = 2.84 M^{-1} B_T^{0.28} \bar{n}_e^{0.58} R^{1.00} a^{0.81} \tag{1}$$

(rms err. 0.268)

in (MW, amu, T, 10^{20} m^{-3}, m), with M the effective isotopic mass of the plasma fuel. This scaling expression is based on the latest version of the threshold database (DB3) extended with results from the recent dedicated H-mode threshold experiments in Alcator C-Mod and in JT-60U, the latter using the new 'W' shaped divertor. For ITER-like devices, this scaling yields an H-mode power threshold prediction which is approximately a factor of 2 lower than that predicted by an earlier version IPB98(5) [2] with the 95% (i.e., 2σ) confidence interval of P_{LH}. There is, however, evidence from JET and JT-60U that the heating power should be 1.3–1.5 times higher than the H-mode threshold to obtain a good H-mode confinement. Therefore, a boundary corresponding to 1.3 × P_{LH} will also be shown in some Figures of the following section.

Thermal energy confinement in the ELMy H-mode is described by the IPB98(y,2) scaling,

$$\tau_{E,th}^{IPB98(y,2)} = 0.0562 I_p^{0.93} B_T^{0.15} P^{-0.69} n_e^{0.41} M^{0.19} R^{1.97} \varepsilon^{0.58} \kappa_a^{0.78} \tag{2}$$

(rms err. 0.145)

where the units are (s, MA, T, MW, 10^{19} m^{-3}, amu, m) and the elongation κ_a is defined as $\kappa_a = S_o/(\pi a^2)$ with So

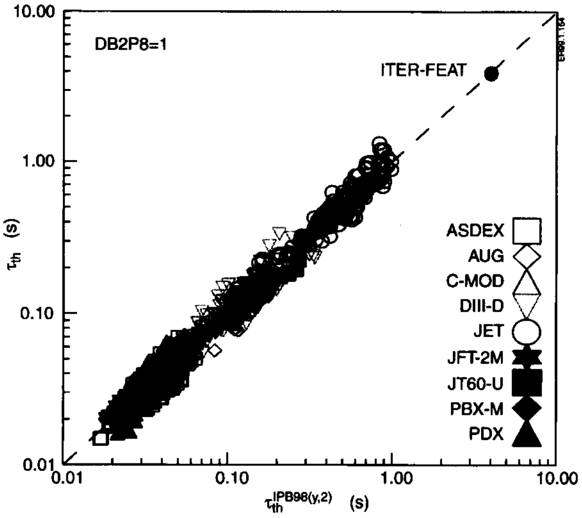

Fig. 3 Comparison of ELMy H-mode thermal energy confinement times with the scaling expression in Eq. (2). Also shown is the IPB98(y,2) scaling prediction for the energy confinement time in a nominal ITER-FEAT Q = 10 discharge.

being the plasma cross-sectional area. A comparison of the H-mode thermal confinement times with the scaling (2) for a subset of ELMy data in the ITER H-mode database is shown in Fig. 3. In the IPB report [2], five empirical log-linear (power law) scaling expressions for the energy confinement time are presented which are derived from different subsets of the H-mode global confinement database containing data from 13 tokamak devices. The expressions fall into two distinct groups, of which two expressions, IPB98(y) and IPB98(y,1), include the H-mode data from small tokamaks and predict ~20% higher confinement for an ITER-like machine than three others, IPB98(y,2) to IPB98(y,4), which exclude these data. In the IPB it is concluded that the available physical and empirical evidence is not strong enough to justify a preferential recommendation amongst these log-linear scalings. IPB98(y,2) has therefore been selected as a conservative option.

3. Inductive Operation

A simple global power balance using the scalings discussed in sec. 2, together with appropriate rules on helium and impurity content and radiation losses, is incorporated in systems codes used to explore the range of design options satisfying the requirement that Q = 10 plasmas can be sustained for several hundred seconds. The impurities treated are helium, from fusion reactions, beryllium sputtered from the torus first wall, carbon sputtered from the divertor target, and argon, which is injected to increase radiation so that the peak power flux at the divertor target remains below 10 MWm^{-2}. The combination of impurities used generally results in a Z_{eff} in the range 1.6–1.9, yielding a DT fuel concentration ranging from 70% to 80% of the electron density.

Table 2. Nominal parameters of ITER-FEAT in inductive operation

Parameter	Units	Reference Q = 10	High Q, high P_{fus}
R/a	m/m	6.2 / 2.00	6.2 / 2.00
Volume	m^3	828	828
Surface	m^2	676	676
Sep.length	m	18.4	18.4
$S_{cross-sect}$	m^2	21.9	21.9
B_T	T	5.3	5.3
I_p	MA	15.0	17.0
κ_x / δ_x		1.86 / 0.5	1.86 / 0.5
κ_{95} / δ_{95}		1.7 / 0.35	1.7 / 0.35
$l_i(3)$		0.86	0.78
V_{loop}	mV	75	85
q_{95}		3.0	2.7
β_N		1.77	1.98
$<n_e>$	10^{19} m^{-3}	10.14	11.4
n/n_{GW}		0.85	0.85
$<T_i>$	keV	8.1	9.1
$<T_e>$	keV	8.9	9.9
$<\beta_T>$	%	2.5	3.2
β_p		0.67	0.62
P_α	MW	82	120
P_{aux}	MW	40	28
P_{ohm}	MW	1.3	1.7
P_{tot}	MW	123	149
P_{brem}	MW	21	29
P_{syn}	MW	8	10
P_{line}	MW	19	20
P_{rad}	MW	48	59
P_{fus}	MW	410	600
P_{sep} /P_{LH}	MW/ MW	75/48	90/53
Q		10	22
τ_E, s		3.7	4.0
W_{th}	MJ	325	401
W_{fast}	MJ	25	38
$H_{H-IPB98(y,2)}$		1.0	1.0
τ_α*/ τ_E		5.0	5.0
Z_{eff}		1.65	1.69
$f_{He,axis}$	%	4.1	5.6
$f_{Be,axis}$	%	2.0	2.0
$f_{C,axis}$	%	0.0	0.0
$f_{Ar,axis}$	%	0.12	0.12

On this basis, operating domains at specified levels of fusion performance (either Q or fusion power) can be mapped out within the defined operation limits. In addition, more accurate calculations of plasma performance are obtained from the PRETOR [4] and ASTRA [5] 1.5 D transport codes, which treat effects such as fuel dilution by impurities, radiation losses, and (in PRETOR) divertor plasma behaviour more accurately, and use transport coefficients normalized to yield the predicted global energy confinement.

Parameters of two representative plasmas in ITER-FEAT are listed in Table 2. The first column shows a reference Q = 10 discharge with a nominal plasma current of 15 MA and a fusion power of 400 MW, while the second column tabulates parameters for a regime with higher current, I_p = 17 MA (q_{95} ~ 2.7), that has the potential for a higher Q of ~25 and higher fusion power of ~600 MW, although with potentially higher risk of plasma disruption. In these simulations, Ar injection was controlled by a feedback loop used to limit the total power exhausted to the divertor target below 30 MW.

Plasma profiles typical of ITER-FEAT operating conditions are illustrated in Fig. 4 for three values of the auxiliary heating power (i.e., 10, 39 and 80 MW). The

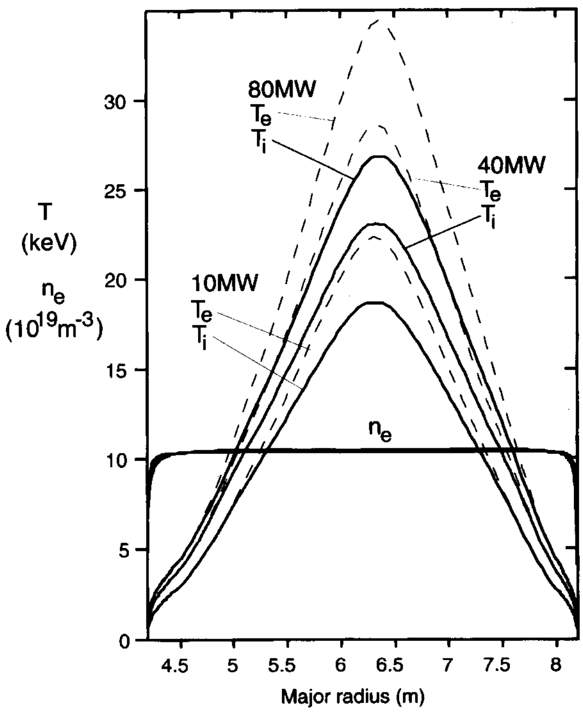

Fig. 4 Temperature and density profiles at a nominal plasma current I_p = 15.1 MA and P_{aux} = 10, 40 and 80 MW.

estimated fusion powers are 260, 400 and 510 MW, respectively. The temperature profiles would correspond to those expected near the end of a sawtooth period (or persisting during the saturated phase for sawtooth periods which are long compared to the energy confinement time). While, the precise shape of the temperature and density profiles is determined by the form of the transport model in PRETOR (or ASTRA) code, the electron density profile is virtually flat and therefore has a conservative (though realistic) influence on fusion performance. Moreover, for a given energy confinement time (or β) less peaked temperature profiles would, in fact, be beneficial for fusion performance. The temperature profiles at different heating powers are similar, with the electron temperature in the plasma core typically ~20% higher than the ion temperature,

A range of plasma parameters satisfying Q = 10 can be shown by an operational domain in terms of fusion power and confinement enhancement factor (Fig. 5). In this diagram the various operational boundaries (P_{loss} = 1.3 P_{LH}, $n = n_{GW}$, and β_N = 2.5) can also be traced. This figure (as well as Figs. 6–8) is calculated using a 0-D code and including impurity contributions from He (τ_{He}/τ_E = 5), Be (2%) and Ar (0.12%).

From Fig. 5, the following conclusions can be drawn. Firstly, for operation at q_{95} = 3, the fusion output power from the ITER-FEAT design is in the region of 200–600 MW (at $H_{H(y,2)}$ = 1), corresponding to a mean separatrix neutron flux ('mean neutron wall loading') of 0.29–0.86 MWm^{-2}, so that the device retains a significant capability for technology studies, such as tests of tritium breeding blanket modules. Secondly, the margin in H-mode threshold power (at $H_{H(y,2)}$ = 1) is significantly greater than the predicted uncertainty derived from the scaling. Thirdly, the device has a capability for Q = 10 operation at n/n_{GW} ~0.6 and β_N ~ 1.5 (when $H_{H(y,2)}$ = 1). Although operation at higher current (q_{95} = 2.7), would entail a shorter burn duration (though still in excess of 100 s), Fig. 6 illustrates the flexibility of the design, its capacity for responding to factors which may degrade confinement while maintaining its goal of extended burn Q = 10 operation, and, by implication, its ability to explore higher Q operation as long as energy confinement times consistent with the confinement scaling are maintained.

Fig. 7 illustrates the window for higher Q operation (Q = 50, representative of 'controlled ignition') in ITER-FEAT, showing that operation at Q values as high as 50 can be attained if $H_{98(y,2)}$ ~ 1.2 is achieved in an

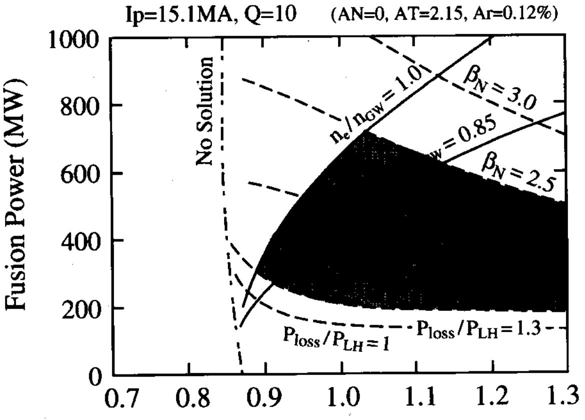

Fig. 5 Q = 10 domain (shaded) for I_p = 15.1 MA (q_{95} = 3.0).

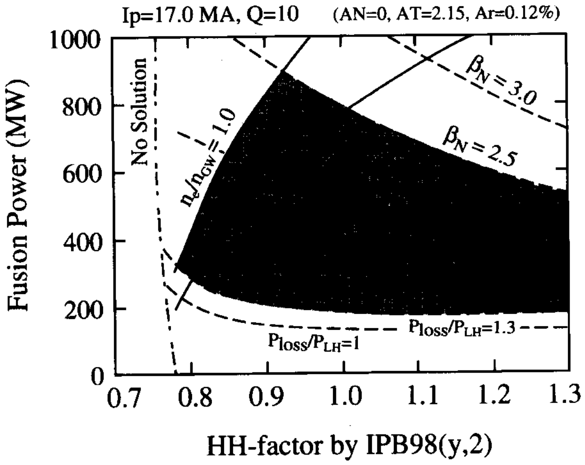

Fig. 6 Q = 10 domain (shaded) for I_p = 17.0 MA (q_{95} = 2.7).

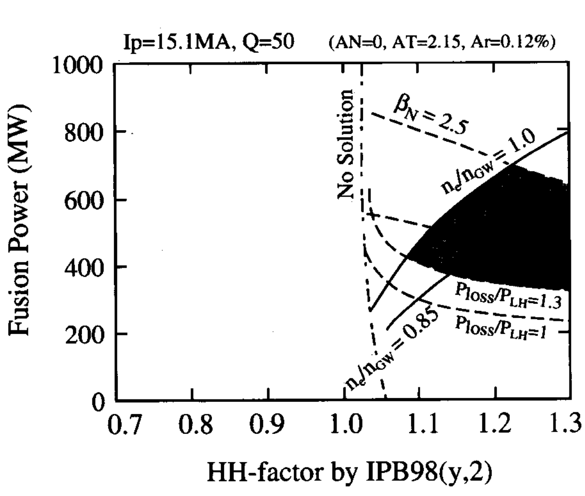

Fig. 7 Q = 50 domain for I_p = 15.1 MA (q_{95} = 3.0).

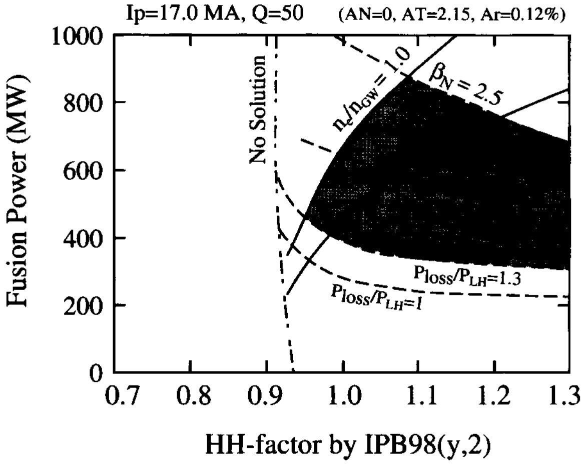

Fig. 8 Q = 50 domain for I_p = 17.0 MA (q_{95} = 2.7).

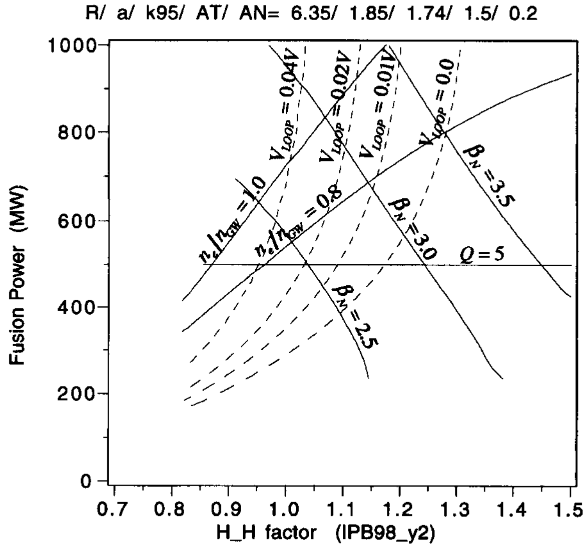

Fig. 9 Operation space for ITER-FEAT for hybrid (long pulse and steady-state) operation. Here, I = 12 MA and P_{CD} = 100 MW.

improved confinement mode, or high density operation can be extended beyond the Greenwald value. Fig. 8 shows a much wider window of Q = 50 operation with higher current (I_p = 17 MA) if a discharge at lower q_{95} (~2.7) can be sustained without confinement degradation.

4. Steady State and Hybrid Operation

An operation space, in terms of fusion power versus confinement enhancement factor, showing the transition from hybrid to true steady-state operation is illustrated in Fig. 9 for I_p = 12 MA and P_{CD} = 100 MW. Contours of constant n/n_{GW} and β_N are indicated, as is the threshold for Q = 5 operation. The plasma minor radius is reduced to 1.85 m by shifting the magnetic axis outward (R = 6.35 m) and optimistic plasma profiles are assumed (density ~$(1-\rho^2)^{0.2}$, temperature ~$(1-\rho^2)^{1.5}$). The plasma elongation at 95% separatrix flux is 1.74. For a given value of fusion power (and hence Q), as the confinement enhancement factor, $H_{IPB98(y,2)}$, increases (simultaneously decreasing plasma density and increasing β_N), the plasma loop voltage falls towards zero. For example, operation with V_{loop} = 0.02 V (the bootstrap current fraction is estimated to be 31%) and I_p = 12 MA, which corresponds to a flat-top length of 2500 s, is expected at $H_{IPB98(y,2)}$ = 1, Q = 5, n_e/n_{GW} = 0.7, and β_N = 2.5. True steady-state operation at Q = 5 can be achieved with $H_{IPB98(y,2)}$ = 1.2 and β_N = 2.8. The bootstrap current fraction is estimated to be 39%. This analysis indicates that a long pulse mode of operation is accessible in ITER-FEAT.

Investigation of steady-state operation will also be possible in weak or reversed shear operation with modest current (8–9MA), and a bootstrap current fraction of ~50%. For this operation, a strongly shifted configuration (R/a = 6.6 m/1.6 m) and strong shaping (plasma elongation ~2.0) is employed. This operation will require challenging values of HH (~1.5) and β_N (3.2–3.5) but offers an attractive scenario for DEMO (a demonstration fusion power reactor) with a high fraction of bootstrap current (50–60%). ITER-FEAT is capable of investigating this operation regime with flexibility in shaping, variety in heating/current drive (NB, EC, IC and/or LH) and advanced features, including ECCD for stabilising neoclassical tearing modes, saddle coils for stabilising resistive wall modes, and profile diagnostics.

5. Divertor Characteristics

Detailed modelling of divertor behaviour for ITER-FEAT is in progress, but extensive analysis has been performed for related design variants [6]. The quantitative similarity of the predictions for these variants indicates that they can also be taken as representative of ITER-FEAT. These results indicate that the peak power loading of the divertor targets for ITER-FEAT device can remain in the range of 5 to 10 MWm^{-2} for a range of upstream (separatrix) plasma densities, 3.3 to 3.8 × 10^{19} m^{-3} (Fig. 10). In such a density range, the upstream helium concentration can be reduced to 3% (Fig. 11).

The projection of ELM energy in ITER is associated with a large uncertainty. According to the present scaling [2] based on discharges with modest density (n/

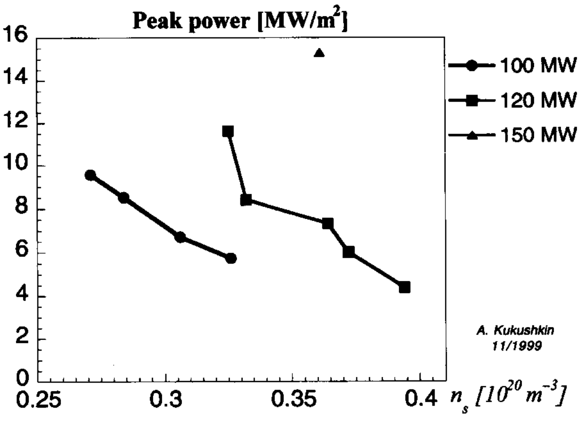

Fig. 10 Peak power loading vs. upstream density for different values of the estimated power flowing to the SOL in a device similar to ITER-FEAT. The sharp drop of the peak power from the leftmost point on the 120 MW curve corresponds to the onset of partial detachment in the outer divertor.

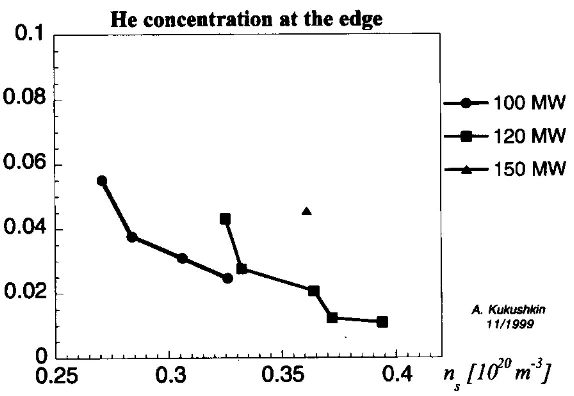

Fig. 11 Helium concentration at the plasma edge vs. upstream density in a device similar to ITER-FEAT for different values of the power conducted into the SOL. The fusion power, which determines the helium production rate was assumed to be 570 MW for the 100 MW and 120 MW cases, and 700 MW for the 150 MW case. The onset of partial detachment in the outer divertor is also seen here.

n_{GW} ~ 0.4), the projected ELM energy is 3–4 times the ablation threshold for nominal operation. There will be a window of acceptable ELM energy for fusion power of 250 MW, if the ELM deposition area is twice the steady-state heat load. Moreover, recent DIII-D experiments show [7] that high density discharges (n/n_{GW} ~ 0.7) are associated with ELM amplitudes lower by a factor of four and high-quality confinement characteristics (H_H ~ 0.9).

6. Conclusions

The assessment of plasma performance based on the scaling laws recommended by the ITER Physics Expert Groups shows that a heating power of 50 MW is sufficient to achieve L-H transition and Q = 10 operation with 13% margin in the energy confinement time. The main issue is to establish an operation scenario with good confinement (H_H ~ 1) at densities ~80% of Greenwald density with sufficiently low ELM amplitudes. Ignition in ITER-FEAT is also possible within the uncertainties of the extrapolation from the present database but with a significantly smaller probability than in the FDR ITER. It has been shown that hybrid operation provides a scenario for a long pulse (≥ 1000 s) with a modest requirement on confinement ($H_H \le 1.0$) and beta ($\beta_N \le 2.5$). Steady-state operation will also be possible in weak or reversed shear operation with modest current (8–9 MA) , which will require challenging values of H_H (~1.5) and β_N (3.2–3.5). ITER-FEAT is capable of investigating this regime with flexibility in shaping, heating/current drive (NB, EC, IC and/or LH) and advanced features, including ECCD for stabilising neoclassical tearing modes, saddle coils for stabilising resistive wall modes and profile diagnostics.

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