ITER-FEAT Operation

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Abstract. ITER is planned to be the first fusion experimental reactor in the world operating for research in physics and engineering. The first 10 years' operation will be devoted primarily to physics issues at low neutron fluence and the following 10 years' operation to engineering testing at higher fluence. ITER can accommodate various plasma configurations and plasma operation modes such as inductive high Q modes, long pulse hybrid modes, non-inductive steady-state modes, with large ranges of plasma current, density, beta and fusion power, and with various heating and current drive methods. This flexibility will provide an advantage for coping with uncertainties in the physics database, in studying burning plasmas, in introducing advanced features and in optimizing the plasma performance for the different programme objectives. Remote sites will be able to participate in the ITER experiment. This concept will provide an advantage not only in operating ITER for 24 hours per day but also in involving the world-wide fusion communities and in promoting scientific competition among the Parties.

1. Introduction

The technical requirements for the new ITER (ITER-FEAT) are as follows [1]:

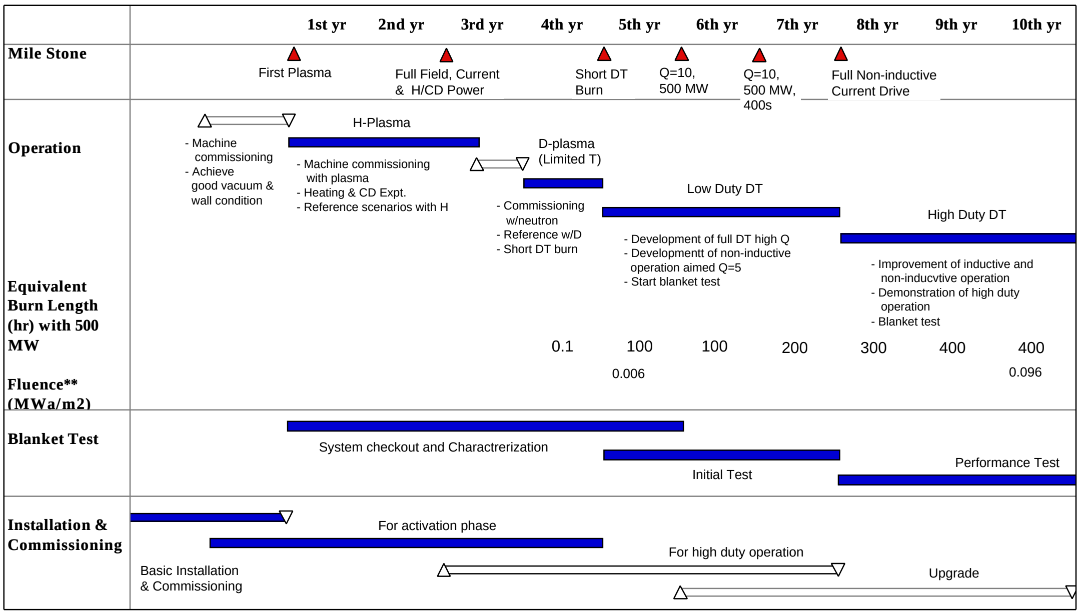
- Plasma Performance: 1) Extended burn in inductively-driven plasmas at the energy gain Q > 10, 2) Aim at demonstrating steady-state through current drive at Q > 5, and 3) Controlled ignition not precluded.
- Engineering Performance and Testing: 1) Demonstration of availability and integration of essential fusion technologies, and 2) Test of components for a future reactor including tritium breeding blankets with the average neutron flux > 0.5 MW/m^2 and the average neutron fluence > 0.3 MW a $/m^2$.

The design of the ITER-FEAT [2] has been developed not only to satisfy these requirements but also to have the flexibility of plasma operations for studying burning plasma and for the optimization of plasma performance for various objectives. The major characteristics of the ITER operation are discussed in this paper.

2. Overview of ITER Operation

After commissioning the ITER system, the tokamak discharges with hydrogen will start and the plasma performance and operation space will be gradually increased. Operation phases over the first 10 years is illustrated in Fig.1 and each phase is as follows:

The hydrogen phase is a non-nuclear phase, mainly planned for full commissioning of the tokamak system in a non-nuclear environment where extensive remote handling is not required. The discharge scenario of the full DT phase reference operation such as plasma current initiation, current ramp-up, formation of a divertor configuration and current ramp-down can be developed or simulated in this phase. The semi-detached divertor operation for DT plasma can also be studied since the peak heat flux onto the divertor target will be of the same order of magnitude as for the full DT phase by using the initial full heating power of 73 MW. Characteristics of electromagnetic loads due to disruptions or vertical displacement events (VDEs), and heat loads due to runaway electrons, will be basically the same as those of the DT phase. Studies of the design-basis physics will significantly reduce the uncertainties of the full DT operation. Mitigation of severe disruptions, VDEs and ELMs or better control of these events in the later phase will become possible, leading to a more efficient DT operational phase. However, some important technical issues will not be fully tested in this phase because of the smaller plasma thermal energy content and the lack of neutrons and energetic alpha-particles. For example, evaporation of the divertor target surface expected at the thermal quench phase of disruption, effects of neutron irradiation of the in-vessel materials, and alpha-particle heating of the plasma, will not be tested.



*Average Fluence at First Wall (Neutron wall load is 0.56 MW/m2 in average and 0.77MW/m2 at outboard midplane.)

FIG. 1. ITER-FEAT operation plan for the first ten years.

In the deuterium phase, although the fusion power is low, the activation level inside the vacuum vessel will not allow human access after a few deuterium discharges with powerful heating. Characteristics of deuterium plasma behavior are very similar to those of DT plasma except for the amount of alpha heating. Therefore, the physics uncertainties will be significantly reduced. Since tritium is already produced from DD reactions, addition of a small amount of tritium from an external source will not significantly change the activation level of the machine. Fusion power production at a significant power level for a short period of time will be demonstrated in the 4^{th} year without fully implementing cooling and tritium-recycle systems. By using a limited amount of tritium in a deuterium plasma, the integrated commissioning of the device's nuclear systems is possible, in particular, the shielding performance can be assessed.

DT operation can be divided into two phases predominantly oriented towards physics and engineering goals respectively. During the first full DT phase the fusion power and burn pulse length will be gradually increased until the inductive operational goals are reached (Fig.1). Hybrid and non-inductive, steady-state operation will also be developed. The tritium-breeding blanket modules will also be tested whenever significant neutron fluxes are available, and a reference mode of operation for that testing will be established. The total average neutron fluence in this first full DT phase (5^{th}-10^{th} year) will be about 0.1 MW a/m^2 and the net consumption of tritium will be about 5 kg.

The next 10 years of the second full DT operation will emphasize improvement of the overall performance and the testing of components with higher neutron fluences. This phase should address the issues of higher availability of operation and further improved modes of plasma operation. Implementation of this phase should be decided following a review of the results from the first 10 years' operation and assessment of the merits and priorities of programmatic proposals. A decision on incorporating tritium breeding during the course of the second DT phase will be based on the availability of tritium from external sources, the results of breeder blanket testing, and experience with plasma and machine performance. At present, it is considered that 30 kg tritium can be supplied from external sources if required. This value corresponds to an average neutron fluence of about 0.6 MW a/m^2 (two times the minimum requirement) and a neutron fluence at the first wall of the test blanket sections of 0.82 MW a/m^2.

The ITER has a large flexibility in fusion power, pulse length and plasma current. For example, the nominal fusion power for the design is 500 MW but 700 MW can be accomplished for about 200 s. The ITER construction will follow a "staged" approach to maximize the opportunities for deferring cost and reducing the peak demand for funding in any single year. Technical requirements of equipment will be qualified better by the early experimental results. In particular, the ability to study steady-state operation, various advanced scenarios, high β operation with active stabilizers, high density operation with advanced fuelling will, if necessary, be improved through additional investment. Table I shows the start-up set of heating and current drive system and possible upgrade scenarios.

TAB. I: POSSIBLE UPGRADE SCENARIOS OF HEATING AND CURRENT DRIVE SYSTEMS.

	Start-up		**Scenario 1**		**Scenario 2**		**Scenario 3**		**Scenario 4**	
	Power [MW]	No. of Equat. ports	Power [MW]	No. of Equat. ports	Power [MW]	No. of Equat. ports	Power [MW]	No. of Equat. ports	Power [MW]	No. of Equat. ports
NB	33	2	33	2	50	3	50	3	50	3
IC	20	1	40	2	20	1	40	2	20	1
EC	20	$1^{(1)}$	40	$1^{(2)}$	40	$1^{(2)}$	40	$1^{(2)}$	20	$0^{(2)}$
LH	0	0	20	1	20	1	0	0	40	2
Total	73	4	133	6	130	6	130	6	130	6

Note: NB (1 MeV, H/D), IC (~ 50 MHz), EC (~ 170 GHz), LH (~ 5 GHz). 1) 20 MW of EC will be used either (i) in 2 upper ports to control neo-classical tearing modes, or (ii) in one equatorial port for main heating or current drive; 2) EC will be able to use 4 allocated top ports for the power upgrade. No additional equatorial ports are therefore foreseen for this system. The total installed power is given in the table and the total maximum power into the torus is limited to 110 MW.

In order to use ITER efficiently, to involve the world-wide fusion communities and to promote scientific competition among the Parties, remote experimental capabilities are foreseen. The initial operation, i.e., the hydrogen phase, is the real ITER commissioning phase with plasma and the initial learning phase to develop operation and train future operational groups. Therefore, in this phase, working at one site is fundamental and moderate operational shifts, i.e., two experimental shifts plus one night shift of limited activities like discharge cleaning, may be most appropriate for this phase. After this phase, remote experimental sites could be introduced. An example is as follows:

- 3 shifts/day on site: Most people will work during the day, i.e., 1 or 2 shifts for experimentation. A limited number of people will work during the night shift for minimum machine monitoring and to support experiments from remote sites.
- 1 shift (or 2 shifts)/day on remote experimental site(s): Remote experimentation will be done within the envelope of machine parameters and conditions agreed to in advance or given by the on-site control room.

3. Burning Plasma Operation

ITER can accommodate various types of operations, such as inductive ELMy-H mode operation, hybrid operation with very long pulse, and non-inductive steady-state operation with large ranges of plasma current, current profile, plasma density, beta and fusion power. In this section, typical burning plasma operations with long pulses or steady state are discussed.

Inductive high-Q operation

Inductive high-Q operations are based on the ELMy H-mode because of its reproducibility and robustness with a demonstrated long-pulse capability and a well established database. The plasma confinement time (τ_E), H_H factor, L-H transition power (P_{LH}), Greenwald density (n_G), and normalized beta (β_N) are given as follows [3,4]:

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

where the units are (s, MA, T, MW, 10^{20}m^{-3}, AMU, m and $\kappa_a = S_x / \pi a^2$),

$$P_{LH} = 2.84 M^{-1} B_T^{0.82} \bar{n}_e^{0.58} R^{1.00} a^{0.81}, \quad n_G = I_P/\pi a^2 \ \ and \ \ \beta_N = \beta(\%)/[I_P/aB_T].$$

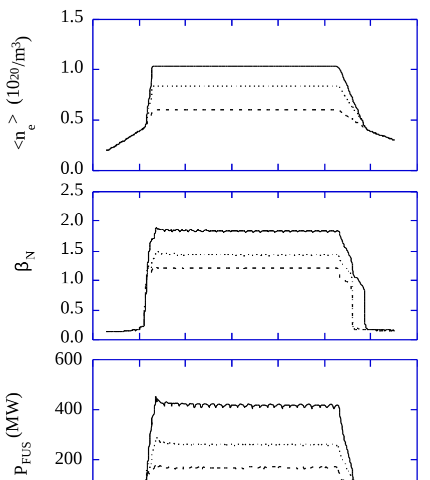


FIG. 2. Time history for n_e/n_G = 0.5 (---), 0.7 (···), 0.85 (¾). Here, I_P = 15MA, H_H = 1.0, τ_{He}^*/τ_E=5, P_{LOSS}/P_{LH} ≥ 1.3 and peak divertor heat load < 10 MW/m^2. Q = 6, 10 and 10 when n_e/n_G = 0.5, 0.7 and 0.85, respectively.

Figure 2 shows typical examples of the time evolution of inductive, high Q pulses with various plasma densities at the nominal plasma current of 15 MA. The flat top is about 400 s and the quasi-steady state is achieved within 100 s after the burn started. The density of 50 % of Greenwald density (n_G) with the auxiliary heating power of 27 MW give a fusion power P_{fus} of 160 MW, Q = 6 and the total heating power including alpha particles of 60 MW with β_N = 1.2. This type of operation should be easily achieved and is a suitable operation in the initial study. Then, the plasma density and the fusion power will be increased gradually. Q = 10 will be achieved at P_{fus} = 250 MW with n_e/n_G = 0.7 and β_N = 1.5 or at P_{fus} = 400 MW with n_e/n_G = 0.85 and β_N = 1.8 (Table II). These parameters are still in conservative ranges. If necessary, a high field side pellet injector, stabilization of neo-classical tearing modes by EC current drive and /or mitigation of heat load due to ELMs will be applied for high fusion power operation up to 700 MW with higher beta, higher density and higher stored energy.

TAB. II: TYPICAL PARAMETERS OF Q = 10 INDUCTIVE OPERATION.

Parameter		**Parameter**	
R/a (m/m)	6.2/2.0	P_{fus} (MW)	400
Volume (m^3)	828	P_{aux} (MW)	40
B_T (T)	5.3	P_{OH} (MW)	1
I_P (MA)	15.0	P_{RAD} (MW)	47
κ_x/δ_x	1.85/0.49	$P_{pedestal}/P_{LH}$(MW)	87/48
κ_{95}/δ_{95}	1.7/0.33	P_{SOL} (MW)	74
li (3)	0.84	$P_{divertor}$ (MW)	30
q_{95}	3	τ_E (s)	3.7
β_N	1.8	H_H	1
$<n_e>$ ($10^{19}m^{-3}$)	10.1	$\tau_{He}{}^*/\tau_E$	5.0
n_e/n_G	0.85	$Z_{eff.axis}$	1.69
$<T_e>$ (keV)	8.8	$f_{He.axis./ave}$ (%)	4.3/3.2
$<T_i>$ (keV)	8.0	$f_{Be.axis}$ (%)	2.0
$<\beta_T>$ (%)	2.5	$f_{Ar, axis}$ (%)	0.12

Operation branches with n_e/n_G = 0.85 in the Lawson Diagram are shown in Fig. 3 for a fixed fusion power at 400 MW and fixed auxiliary heating powers at 40 MW / 20 MW. The intersection of Q = 10, H_H = 1, P_{fus}= 400 MW and P_{aux} = 40 MW is the operation point shown in Table II, and Fig. 2. The Q value increases for decreasing heating power with fixed H_H, e.g., Q = 10 at P_{aux} = 40 MW and Q = 17 at P_{aux} = 20 MW with H_H = 1. However, the maximum Q is limited to less than 20 by the H-mode transition condition. The Q value varies from 5 to 50 with H_H = 0.8-1.2.

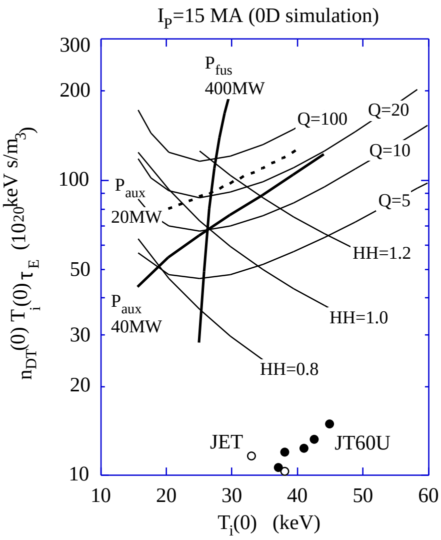


FIG. 3. ITER operation branches with n_e/n_G = 0.85 in the Lawson diagram. The intersection of lines P_{fuS} = 400 MW and P_{AUX} = 40 MW is the operation point shown in II. Experimental data in JT60U (●) [4] and JET (○) [5] are also shown but with high ion temperature modes.

ITER has a possibility to reach ignition or a very large Q especially at Ip = 17 MA with H_H = 1 as shown in Fig. 4. After reduction (Fig.4-a) or termination (Fig.4-b) of the auxiliary heating power, increase of the fusion power, i.e., the thermal instability, starts due to the low average temperature, typically 6.5 keV, but the temperature is suppressed at 9-10 keV and 600-650 MW mainly due to the helium accumulation. With a nominal condition of the helium pumping, i.e., $\tau_{He}{}^*/\tau_E$ = 5, Q = 50 is maintained (Fig.4-a) but the ignition condition is terminated within about 40 s because of the accumulation of helium (Fig.4-b). If the confinement time is 5-10% longer than the predicted value, i.e., H_H = 1.05-1.1, the ignition condition can be reached and maintained. Another possibility is the reduction of helium. Normally, the divertor pumping rate is < 150 Pam^3/s and is limited to 200 Pam^3/s due to the limit of the amount of hydrogen in the cryopump system in the divertor ports. For short pulse operation, the pumping rate can be increased by increasing the fuelling rate, e.g., with high field side pellet injections, and the

$\tau_{He}*/\tau_E$ could be reduced. With $\tau_{He}*/\tau_E = 4$, a quasi-steady-state ignition condition can be maintained (Fig. 4-b). The flat-top duration of 17 MA is limited by the available volt-seconds and is 100-200 s which would be sufficient to study the major characteristics of the ignited or very high-Q plasma because a quasi-steady-state operation is achieved 50 s after start of heating.

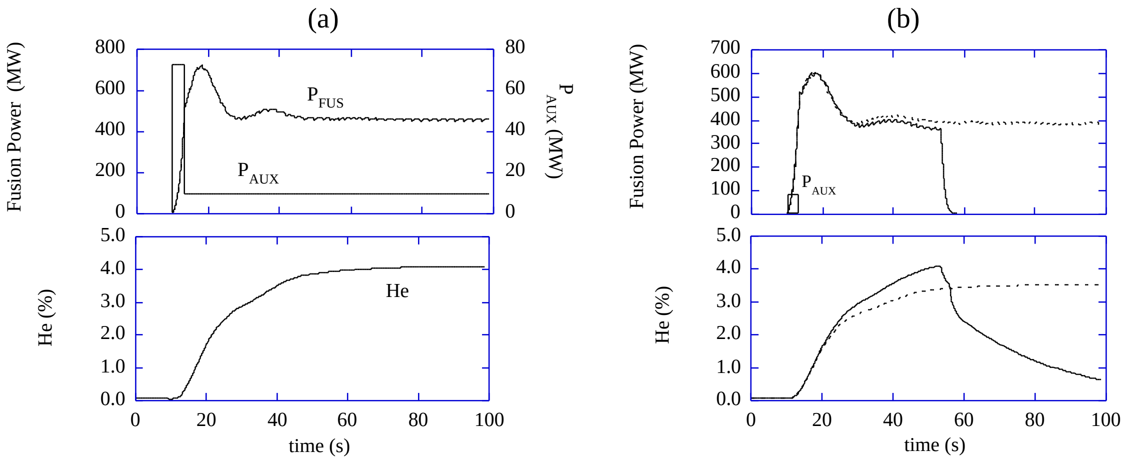


FIG. 4. I_p = 17MA operations. (a) High-Q operation, (b) Ignition operation with different He accumulation. Here, $<n_e> = 1.1\times10^{20}/m^3$ ($<n_e>/n_G = 0.81$), $\tau_E = \tau_E(y,2)$ and P_{AUX} = 73 MW at 10-13.5s. ---- $\tau_{He}*/\tau_E = 4$ and — $\tau_{He}*/\tau_E = 5$.

Hybrid and steady state operation

Very long pulse operation can be achieved even with $H_H = 1$ by increasing non-inductive current drive and decreasing the plasma current. Consequently, the Q value is reduced (Fig.5). A pulse length of about 1000 s is needed to achieve a thermal quasi-steady state of the front part of the breeding blankets and would be suitable for tests of breeding blankets. One example of this scenario is given in Table III and the set of plasma parameters is in a reasonable range with an acceptable heat load on the divertor. A much longer pulse operation requires a higher beta and a smaller plasma size with $H_H = 1$. Explorations in this direction will be done not only to improve the operation for the engineering tests but also to find possible operation modes of the steady-state plasma with non-inductive current drive. Only a small Q (< 2) and a small bootstrap current (I_{bs} < 15%) are expected in a steady-state plasma with a conventional ELMy H mode, i.e., H_H = 1and a flat density profile. Therefore, a further increase of confinement is necessary in a higher beta plasma with a high bootstrap current.

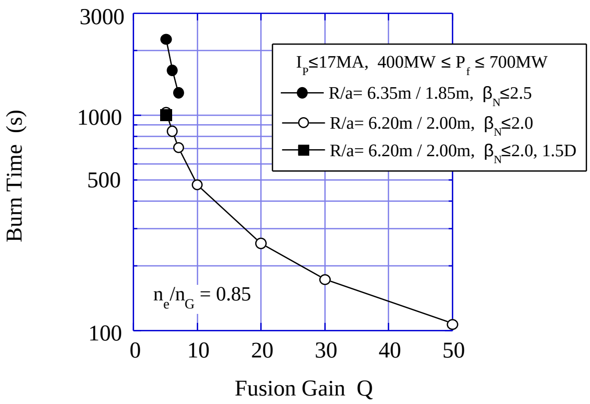


FIG. 5. Burn pulse length vs. Q. The plasma current is adjusted to satisfy Q and $H_H = 1$.

Figure 6 shows examples of required parameter ranges to achieve Q = 5 steady-state plasmas with non-inductive current, typically $H_H = 1.5$ and $\beta_N = 3.5$ and I_{bs} ~ 50%. These parameters would not be achieved in a conventional ELMy H mode. Different improved modes will have to be explored. One of the possible candidates is a weak negative or shallow shear profile with an internal transport barrier. For example, a stable operation with $H_H = 1.6\text{-}1.7$, $q_{95} = 4.1$, $\delta_{95} = 0.34$, $\beta_N = 2.2\text{-}2.5$, a weak negative shear, an internal transport barrier at r~(3/4) a and 70-80% non-inductive current drive (1/3 neutral beam current drive and 2/3 bootstrap current) is demonstrated in experiment [7]. Further extensive studies in tokamaks are needed to predict ITER performance with an internal transport barrier. Figure 6-f and Table III show one example with an internal transport barrier, $H_H = 1.5$, $q_{95} = 4.2$, $\delta_{95} = 0.4$, and $\beta_N = 3.5$. In this case, the plasma height is the same as in typical inductive operation, but the minor radius is reduced to have a more favorable condition for the steady-state operation. This gives a high elongation but the vertical position control is relatively easy because of the low internal inductance and the high normalized beta. The minimum safety factor is larger than 2 and the neo-classical tearing modes would not be excited but the resistive wall modes would have to be stabilized by the set of saddle loops installed on the toroidal field coils. Relatively large impurity seeding is also required because a large fraction of the total heating power, typically 200 MW, should be radiated since the divertor target heat load should be less than about 60 MW. A conservative analysis of the impurity radiation loss is included in this analysis. The divertor target heat load is sufficiently reduced with 0.45 % Ar. A more detailed analysis of the cooling in the scrape-off layer and the divertor is under way [8].

If fuelling near the internal transport barrier is feasible, a favorable density profile is obtained, which increases the bootstrap current and decreases the required H_H factor. In this case, the conditions for the steady-state operation with the full non-inductive current would be significantly relaxed. These issues are under study.

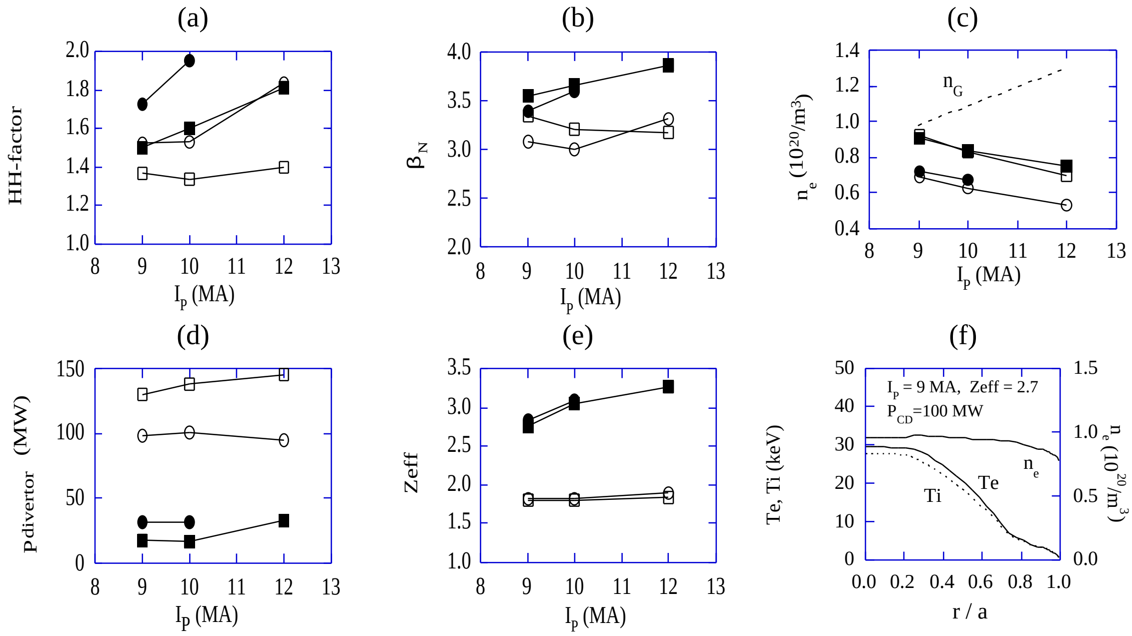


FIG. 6. Non-inductive operation parameters for ITER-FEAT (Q = 5). R/a/_a = 6.5 m/1.7 m/2. 0.4-0.5% of argon and 0.5-1.2% of carbon impurities are seeded for ● and ■ to reduce the divertor heat load $P_{divertor}$. 60 MW of $P_{divertor}$ gives about a peak heat flux of 10 MW/m^2 on the divertor target. The bootstrap current is about 50 % at I_P = 9 MA and 40% at I_P = 12 MA.

TAB. III : TYPICAL PARAMETERS OF STEADY STATE OPERATION

Parameter		**Parameter**	
R/a (m/m)	6.5/1.7	P_{fus} (MW)	500
Volume (m^3)	729	P_{aux} (MW)	100
B_T (T)	5.06	P_{OH} (MW)	0
I_P (MA)	9	P_{RAD} (MW)	86
κ_x / δ_x	2.0 / 0.55	$P_{pedestal}/P_{LH}$ W)	144 / 40
κ_{95}/δ_{95}	1.9 / 0.4	P_{SOL} (MW)	114
li (3)	0.59	$P_{divertor}$ (MW)	20
q_{95}	4.2	τ_E (s)	2.5
β_N	3.5	H_H	1.5
$<n_e>$ ($10^{19}m^{-3}$)	9.1	$\tau_{He}*/\tau_E$	5
n_e/n_G	0.92	$Z_{eff,\ axis/\ ave}$	2.8 / 2.7
$<T_e>$ (keV)	12.6	$f_{He,\ axis/\ ave}$ (%)	4.5 / 3.3
$<T_i>$ (keV)	11.7	$f_{Be,\ axis}$ (%)	2.0
$<\beta_T>$ (%)	3.7	$f_{C,\ axis}$ / $f_{Ar,\ axis}$ (%)	0.4 / 0.45

4. Conclusions

The flexibility of ITER will allow exploration in a large operation space of fusion power, plasma density, beta, pulse length and Q value, e.g., the nominal fusion power for the design is 500 MW but 200-700 MW can be accommodated. A low fusion power operation with a low plasma density should be easily achieved and is a suitable operation for the initial study. ITER has a capability to achieve a quasi-steady state of Q = 10 for about 400 s with a reasonable condition and a high possibility to achieve a quasi-steady state of a high value of Q or the ignition condition for 100–200 s in a high plasma current operation. A higher fusion power would be achieved by stabilizing the neoclassical tearing mode and/or using high field side pellet injection. ITER has also a capability of very long pulse operation for about 1000 s with reasonable plasma parameters by increasing the non-inductive current drive and decreasing the plasma current and reducing Q, typically 5, at 400-500 MW which gives a neutron flux of 0.62 –0.77 MW/m^2 at the first wall of the test blankets. A fully non-inductively driven plasma at Q = 5 requires a high beta and a high confinement. Methods to develop this kind of operation are also included in the ITER experimental programme.

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