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Chapter 9: ITER contributions for Demo plasma development

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Abstract
The chapter summarizes the physics issues of the demonstration toroidal fusion power plant (Demo) that can be addressed by ITER operation. These include burning plasma specific issues, i.e. energetic particle behaviour and plasma self-heating effects, and a broader class of power-plant scale physics issues that cannot be fully resolved in present experiments. A critical issue for Demo is whether MHD and energetic particle modes driven by fast particles will become unstable and affect plasma performance. Self-heating effects are expected to be especially important for control of steady-state plasmas with internal transport barriers (ITBs) and high bootstrap current fractions. Experimental data from ITER will improve strongly the physics basis of projections to Demo of major plasma parameters such as the energy confinement time, beta and density limits, edge pedestal temperature and density, and thermal loads on in-vessel components caused by ELMs and disruptions. ITER will also serve as a test bed for fusion technology studies, such as power plant plasma diagnostics, heating and current drive systems, plasma facing components, divertor and blanket modules. Finally, ITER is expected to provide benefits for the understanding of burning plasma behaviour in other magnetic confinement schemes.

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1. Introduction

The overall programmatic objective of ITER operations is to demonstrate the scientific and technological feasibility of fusion energy. Specifically, ITER operations will prepare the physics basis for a fusion power plant based on tokamak magnetic confinement. The main physics goal of ITER is the achievement of extended ($\geqslant$400 s) burn of DT plasmas with the fusion power $P_{\rm fus}$ of 0.4–0.5 GW, fusion gain $Q = P_{\rm fus}/P_{\rm aux} \geqslant 10$ and $\beta_{\rm N} = \beta(100aB/I) \approx 1.8$, where $P_{\rm aux}$ is the auxiliary heating power, β is the normalized plasma pressure, $\beta = 2\mu_0 < p > /B^2$, B is the toroidal magnetic field and I is the plasma current. The steady state regime with $Q \geqslant 5$ will also be possible if an enhanced energy confinement with $H_{\rm H98(y,2)} \geqslant 1.3$ at $\beta_{\rm N} \geqslant 2.6$ and bootstrap current fraction $f_{\rm BS} \geqslant 0.5$ can be realized.

The first fusion device aiming at electricity production—usually called Demo—is the logical next step on the development path towards commercial power production. No concrete design for it yet exists, and even its exact strategic role has not been unambiguously defined. A consensus appears to be building, however, for a 'fast track' approach towards fusion, in which Demo should be at least scalable to a commercially acceptable power plant expected to generate about 1 GW of the net electrical power [1–4]. It should therefore incorporate only physics and technology solutions also usable in a first generation power plant, and differ from the latter essentially only in the way it will be operated, i.e. via its availability. To minimise the total costs of the fusion development path it may also be desirable to reduce Demo nominal power rating compared with later power plants, provided scalability to the latter can be ensured with high confidence.

Compared with ITER, a commercial power plant and Demo will need higher $P_{\rm fus}$, higher Q and higher operational reliability. To achieve higher $P_{\rm fus}$, a higher stored plasma energy ($\propto\beta \times B^2 \times V$ with V being the plasma volume) and higher density are required. Since the plasma size and, hence, an effective area of the divertor target plates in Demo is expected to be only slightly larger than that in ITER, higher radiative cooling is required. To increase Q, a higher $\beta_{\rm N}$, at constant or higher confinement improvement factor $H_{\rm H98(y,2)}$, and, in steady-state regime, higher $f_{\rm BS}$ are necessary. To improve operation reliability, large plasma perturbations, including disruptions and large ELMs, should be avoided. The major physics issues to be resolved in ITER and Demo in the frames of the fast track approach are illustrated qualitatively in table 1. Table 2 demonstrates in a quantitative fashion that the progression of plasma parameters from ITER to DEMO varies appreciably. In some cases (e.g. plasma current, machine size and values of some dimensionless plasma parameters), the step from ITER to Demo is small. However, a substantial increase in Demo relative to ITER will be necessary for fusion power, stored plasma energy and radiation power. The table also shows that the best values of individual normalized parameters achieved, not simultaneously, in present day experiments are close to or even better that those required simultaneously in ITER and Demo, with an important exception, i.e. the values of normalized ion Larmor radius, $\rho^* = \rho_{\rm i}/a$, at reactor-like $\beta_{\rm N}$ and collisionality ν^* differ significantly.

This chapter discusses potential contributions from ITER operation to the physics basis for Demo. Section 2 addresses the intrinsic characteristics of burning plasma such as behaviour of suprathermal α-particles and plasma self-heating effects. Section 3 deals with a broad class of power-plant scale physics issues that cannot be fully resolved in present experiments, including projections of the energy confinement time, beta and density limits, edge pedestal temperature and density, thermal loads on in-vessel components caused by ELMs and disruptions, and physics related fusion technology studies, such as power plant plasma diagnostics, heating and current drive systems, plasma facing materials, divertor and blanket modules. Potential benefits for other magnetic confinement schemes from burning plasma studies in ITER are also discussed in this section. A summary is given in section 4.

2. Burning plasma physics

The power-plant scale plasma physics to be addressed in ITER includes two classes of elements: (1) burning plasma physics, comprizing energetic particle behaviour and the effects of plasma self-heating; and (2) scale-dependent plasma physics [25]. ITER will also provide the only testbed before Demo where the plasma facing component (PFC) issues can be addressed in a fusion environment. In addition, it is anticipated that smaller, more versatile non-nuclear 'satellite' devices will be operated in parallel to ITER to address specific physics questions.

ITER affords a unique opportunity to study burning plasma physics, especially in the areas of energetic particle effects and plasma self-heating properties. The wide range of plasma parameters achievable in ITER gives a favourable opportunity to study the physics of toroidal burning plasmas in a new, currently inaccessible regime where power-plant like parameters, including the presence of fusion α-particles and self-heating, are achieved simultaneously.

Table 1. Physics issues and their resolution (adapted from [2]).

Issue	Today's experiments	ITER	Demo[a]	Power plant
Energetic particle effects	1	3	6	6
Self heating and burn stability	1	3	4	6
Understanding confinement	2	2	4	4
Fuelling	2	3	5	6
Density limits	2	3	4	5
Tearing mode stability	3	3	5	6
Resistive wall mode stability	3	2	5	6
Power and particle exhaust	1	3	5	6
Edge localized modes	3	3	5	6
Disruption avoidance/mitigation	2	3	6	6
Steady-state operation	1	3	5	5
Divertor performance	2	3	6	6
Burning plasma ($Q > 10$)	—	3	6	6
Power plant plasma performance	1	3	6	6
Diagnostics and neutrons	1	3	5	6

Note:
Key: 1 – Will help to resolve the issue
2 – May resolve the issue
3 – Should resolve the issue
4 – Confirmation of resolution needed
5 – Solution is desirable
6 – Solution is a requirement

[a] Risk would be reduced and options expanded by operating several alternative Demo plants in parallel.

Table 2. Comparison of ITER design parameters with proposed Demo parameters and the best individual parameters achieved in tokamak experiments.

Parameter	ITER inductive	ITER steady state	Demo example steady state [5]	Best achieved individual parameters in tokamak experiments
I (MA)	15	9	15	4.5 (JET) [6]; 5 (JT-60U) [7]
B/B_{max} (T)	5.3/11.8	5.18/11.8	6.8/14.6	8 (C-Mod) [8]; 4.8 (JT-60U) [7]
a (m)	2	1.85	2.1	1.05 (JT-60U) [7]; 0.96 (JET) [6]
R (m)	6.2	6.35	6.5	3.56 (JT-60U) [7]; 2.91 (JET) [6]
$T_i(0)$ (keV)	23	25	45	45 (JT-60U) [7]; 39 (JET) [9]
q_{95}	3.0	5.2	5.3	2.1 (DIII-D) [10]; 2.2 JT-60U [11]; 2.3 (JET) [12]
$\langle n_e \rangle / n_G$	0.85	0/75	1.0	1.4 (DIII-D) [13]; 0.8 JT-60U [14]
$H_{H98(y,2)}$	1.0	1.4	1.3	2.3 (JT-60U) [7]
ρ^* (10^{-3})[a]	1.9	2.5	2.1	4.6 (JET)[d] [12]; 5.6 (JT-60U) [7]
ν^*[b]	0.026	0.019	0.012	0.026 (JT-60U) [7]; 0.05 (JET) [12]
β_N	1.8	3.0	3.9	4.5 (DIII-D) [15]; 4.8 (JT-60U) [7]
f_{BS}	0.15	0.5	0.79	$\geqslant$0.9(JT-60U) [16]; 0.85 (DIII-D) [17]
f_{NI}[c]	0.21	1	1	1.0 (JT-60U) [18], (DIII-D) [19]
P_{fus} (MW)	400	350	3000	16 (JET) [20]; 10.7 (TFTR) [21]
P_{heat}(MW)[e]	120	140	654	40 (JT-60U) [7]; 39.5 (TFTR) [21]
W_{th} (MJ)	320	290	1215	12.9 (JET) [12]; 10.9 (JT-60U) [7]
$P_{rad,total}$ (MW)[f]	60	80	554	$\geqslant 0.9 P_{heat}$(AUG) [22], (JT-60U) [14], (TEXTOR) [23]
$Q = P_{fus}/P_{aux}$	10	5	54	0.8 (JET) [20]; 1.25[g] (JT-60U) [7]
$f_{disruption}$	~0.1(per pulse)	~0.1 (per pulse)	$\leqslant$1 (per year)	$\leqslant$0.01(TFTR) [24] (per pulse)

[a] $\rho^* = 4.57 \times 10^{-3} < T_i M_i >^{0.5} /(aB)$
[b] $\nu^* = 0.01 < n_e > q_{95} R Z_{eff} (R/a)^{3/2} / < T_i >^2$
[c] Non-inductive current fraction
[d] At ITER (inductive) like β_N and ν^*
[e] $P_{heat} = P_\alpha + P_{aux}$
[f] Radiation from core, SOL and divertor
[g] DT equivalent for D plasma

2.1. Energetic particle effects

Good confinement of α-particles generated in DT reactions is crucial for a fusion power plant. Single classical orbit α-particle confinement is well understood and predicted to be sufficiently good in ITER, with corrected toroidal magnetic field ripple, to allow attainment of the plasma performance goals. However, a high gradient of fast particle pressure in high-Q plasma can induce collective instabilities that may cause anomalous loss of α-particles [26]. Collective modes of concern include different types of Alfvén eigenmodes (AEs), kinetic ballooning modes and internal kink modes. In addition to these driven magnetohydrodynamic modes, there are also energetic particle modes (EPMs) characterized by strong dependence on the fast-ion distribution function.

Fast particle effects have been studied in many tokamak experiments using energetic ion tails produced by NBI and ICRF plasma heating, as well as fast α-particles in JET and TFTR DT experiments. However, direct extrapolation from these experiments to burning plasma is limited, since the values of characteristic dimensionless parameters are different. Also, the relative concentration of energetic particles will be less in a power-plant scale device, and the distribution function of fast α-particles in burning plasma will be nearly isotropic whereas in present experiments with auxiliary heating it is anisotropic.

Available codes predict that the α-particle pressure with $\beta_\alpha < 1\%$ in ITER inductive scenarios only marginally excites toroidal Alfvén eigenmodes (TAEs) with $n = 10$–12, while an extended range of unstable modes ($n = 8$–17) is expected due to the contribution of high energy ions generated by high energy NB injection [27]. Scenarios with reversed magnetic shear are expected to be more prone to fast particle instabilities. Initial analysis of non-linear dynamics of EPMs in configurations with reversed shear predicts rapid energetic particle redistribution, albeit with a relatively small global particle loss [28].

Fast particles can transiently suppress the internal $m/n = 1/1$ MHD mode resulting in a large amplitude 'monster' sawtooth crash that redistributes fast particles and plasma parameters inside the affected zone (somewhat larger than the radius of the $q(r) = 1$ magnetic surface). The sawtooth crash can induce seed magnetic islands initiating the growth of neoclassical tearing modes (NTMs). On the other hand, fast particles can destabilize the internal $m/n = 1/1$ MHD mode through the resonant wave-particle interaction at the magnetic precession frequency of the trapped fast ions resulting in the 'fishbone' oscillations that can also redistribute the fast particles within the central zone [26].

Given the large number of instabilities and the complexity of the problem, the numerical codes predicting α-particle driven instabilities in burning plasma are unavoidably approximate and need to be checked against experiments. Although partial tests of the codes are being made in present day experiments, thorough tests of the codes will be possible only in a power-plant scale experiments like ITER. For the science and the code verification mission of ITER it is important that possible operating scenarios include—according to linear theory—both nearly stable but also clearly unstable situations.

2.2. Self-heating and thermal stability

A significant plasma self-heating by α-particles raises the issue of thermal stability of the operating point for burn. While one can conceive a burn control simulation experiment in present devices making a fraction of heating power to be proportional to W_{th}^2 ($\propto P_{\mathrm{fus}}$ in a power plant plasma), thermally stable operating modes for the power plant must be demonstrated at a power-plant scale facility heated by fusion reactions. The plasma burn in ITER and Demo will be globally stable since their operation areas will be located near the stable (right) branch of the ignition curve [29] where the power loss increases faster with temperature than the fusion power. Increase in fuel dilution by helium ash with rizing fusion power is an additional stabilizing factor. However, the consistency between plasma pressure profile, current density profile, including bootstrap current, and the plasma stability requirements may raise multi-faceted feedback and profile control issues that could be important, especially for steady-state operation with internal transport barriers. In such a case, a simple global control scheme may not be adequate and a sophisticated control of plasma current and pressure profiles may be required. ITER will provide a unique environment in which to develop plasma control necessary in a Demo.

3. Power-plant scale physics

In addition to the effects discussed above, the large scale of the device and the associated power densities create additional opportunities for testing Demo issues in ITER.

3.1. Energy confinement

The energy confinement time is of primary importance for the design of future tokamak power plants and should be known with a much higher accuracy and confidence level than is required for the design of ITER, which is still an experimental device. Extrapolation in τ_{E} from present experiments to power-plant like devices is rather far, e.g. a factor of 7 from the best JET shot to the ITER reference scenario. In dimensionless physics, in particular, parameters the extrapolation involves are a factor of 3 or higher in ρ^*, whereas ITER's operating range in β and ν^* has more or less been covered in present experiments. It is probably even more important that ITER (and Demo) will approach the Greenwald density, $n_{\mathrm{G}} = I/(\pi a^2)$, at a collisionality significantly smaller than that in present devices, as—at constant plasma shape and q_{95}—the ratio n/n_{G} can be written as $n/n_{\mathrm{G}} \propto B^{1/5}(\beta^{4/5}\nu^{*1/5}/\rho^{*4/5})$, implying a reduction in ν^* like ρ^{*4} (as $B^{1/5}$ will change by less than 20% compared to present medium-size or JET/JT-60U class experiments).

At present, projections of power-plant plasma parameters based on *ab initio* turbulence models still suffer from a lack of adequate transport models for the plasma pedestal. Experimental data from ITER will substantially improve the accuracy of empirical confinement scalings and extend the database for testing theoretical models, in particular also for the edge region. The need for ITER data points is particularly high regarding ITBs because, at least for ion transport barriers, there are only two size generations (as measured in ρ^*) on which extrapolations can be based (C-Mod/ASDEX Upgrade/DIII-D and JET/JT-60U), whereas the data base for the standard H-mode includes also a smaller generation (Compass, TCV, JFT-2M, ASDEX). ITER has a unique mission to contribute to the physics of confinement at low ρ^* and to provide the basis, together with smaller machines, of ρ^* scaling to Demo.

3.2. Density limits and plasma fuelling

Fusion power plants are pushed to operate close to, or even beyond, the Greenwald density by the desire to stay with the ion temperature close to the maximum of the ratio $\langle\sigma_{\rm fus}v_{{\rm th},i}\rangle/T^2$ to fully benefit from the higher plasma pressure, although this is less so at steady-state operation. This, and the need to replenish the plasma core with deuterium and tritium independently, to compensate for fuel burn-up, implies that ITER has to provide reliable, quantitative information on the achievable plasma density and on the efficiency of different refuelling methods in power-plant scale devices.

As discussed in detail in [30, 31], two major basic mechanisms have been proposed for the density limit in high confinement regimes: (a) divertor plasma detachment causing a saturation of the separatrix density, which, through pedestal transport, produces a limitation in the main plasma density with gas-puffing [32]; and (b) enhancement of the particle transport limiting the plasma density and leading to overall energy confinement degradation [33]. Scalings based on models incorporating these mechanisms can reproduce the present experimental evidence with a similar levels of success, but lead to sizeable differences in the predicted density limit in ITER and Demo, and this can significantly influence their foreseen fusion performance [32]. ITER will allow, for the first time, investigation of the physics processes that determine the density limit in tokamaks in power-plant like conditions (i.e. high density but low collisionality, which are not simultaneously achievable in present day experiments) and will thus provide a firm basis on which to determine the maximum density at which Demo will be able to operate.

Particle fuelling techniques for a power-plant scale device may also need to be different from the gas puff method used in contemporary tokamaks. Modelling of SOL and divertor plasma in ITER using the B2-Eirene code [34] with a purely diffusive model of particle transport predicts saturation of the separatrix density with an increasing gas puff rate at a rather low level because of strong screening of neutrals by the SOL plasma. Therefore, a significant core fuelling using pellet injection is planned in ITER and will be necessary in Demo to obtain the required core plasma density. Pellet injection from the high-field side (HFS) of the torus seems the most appropriate tool for this purpose. A caveat for the use of pellets for plasma fuelling is their capability to generate seed magnetic island and trigger NTM when they reach a corresponding resonant flux surface [35]. This is only observed in experiments with large fuelling pellets; therefore, optimization of pellet size, their penetration depth and injection frequency will be needed [36] (see section 3.5).

There are theoretical mechanisms of anomalous particle pinch supported partially by experiments that could provide core density peaking in the absence of a central particle source [37]. Results of edge simulations appear to indicate a particle pinch acting at the edge, which would alleviate the problem of replenishing the plasma fuel by gas puffing, but also suggest that it would be difficult to decouple the fuelling from divertor conditions. Extrapolation of these effects to power-plant scale plasmas is still uncertain and can significantly affect the complexity of the fuelling systems required in ITER—even more so in Demo. Studies of particle transport and of fuelling efficiency with all the techniques available in ITER will provide the necessary answers for the design and operation of Demo.

3.3. Operational beta limit

Fusion power plant designs assume β-values significantly higher than the baseline scenario of ITER. The achievement of such values can in principle also be tested in smaller devices, but there are a number of elements—even apart from the presence of energetic α-particles—where experiments on a power-plant scale device are needed to confirm their size independence and their controllability under power-plant conditions.

As discussed in [38, 39], the ideal no-wall $\beta_{\rm N}$ limit set by kink/ballooning instabilities in the ITER inductive scenario with positive magnetic shear and $q_{95}\sim 3$ is high, $\beta_{\rm N,no-wall}\approx 4l_{\rm i}\sim 3$ and independent of the machine size ($l_{\rm i}$ is the internal plasma inductance). The $\beta_{\rm N}$ limit above which NTMs can be destabilized by a large enough seed perturbation has been found experimentally to scale almost linearly with $\rho^*\equiv\rho_{\rm i}/a$ [40] and baseline operation in ITER is thus predicted to be in a regime in which NTMs are metastable although understanding of these phenomena is quite recent. Suppression of NTMs using localized electron cyclotron current drive (ECCD), which has been demonstrated experimentally, will therefore be important in ITER, although the ECCD power required remains to be further refined [41]. NTMs can also be avoided by seed island reduction through control of the sawtooth amplitude. The amplitude of the 3/2 NTM can be reduced by transition to the frequently interrupted regime (FIR) of NTM, utilizing non-linear coupling of this mode with 1/1 and 4/3 ideal MHD modes [42]. Both these techniques have been demonstrated in experiments, but their efficiency under power-plant conditions will have to be confirmed in ITER.

In steady-state scenarios with a weak or negative magnetic shear foreseen in many power plant studies, the most dangerous NTMs with $m/n = 3/2$ and 2/1 can be avoided by keeping $q(r) > 2$, and the β-limit will be determined by the ideal kink mode instability. The no-wall β-limit appears to be close to or below the required $\beta_{\rm N}$ for non-inductive steady-state operation in ITER, so that not only will wall stabilization be essential, but the resistive wall modes (RWM) should also be controlled. Since both ITER and a fusion power plant will have limited means for producing and controlling plasma rotation, demonstration of the successful application of a magnetic feedback scheme for the suppression of RWMs [43, 44] at the power-plant scale will be an important aspect of the ITER research program. Indeed ITER's contribution will be fundamental to understanding the extent to which active control of RWMs will permit extension of the accessible $\beta_{\rm N}$ in steady-state scenarios under power-plant relevant conditions.

Exploration of the experimental β-limits for inductive and steady-state scenarios in ITER using active feedback control techniques for suppression of β-limiting instabilities will therefore be an essential contribution to the physics of power-plant plasmas.

3.4. Plasma energy exhaust and core-boundary interface issues

The most distinctive feature of Demo is a much higher fusion power (by at least a factor of 4) and stored plasma energy (by at least a factor of 2) than those of ITER. Since the divertor target may be only slightly larger than that of ITER, the power handling limits set by technological constraints on the PFCs (the maximum acceptable heat flux lies in the range 10–15 MW m^{-2}) implies that a substantial fraction, perhaps $\geqslant$80% of the plasma exhaust power must be distributed over the first wall by radiation. While the partially detached divertor scenario proposed for ITER should satisfy the power handling requirements for the ITER reference operating scenarios, developing very high radiation fraction scenarios (with high energy confinement) will be an important aspect of the ITER experimental programme towards Demo. This problem may not be as severe as it first appears: with higher-Z impurity seeding large radiative power losses can be generated inside the separatrix—if it turns out that profiles are stiff and thus insensitive to central power losses, the impact on plasma performance may be minimal. A potential approach to addressing this issue in ITER (i.e. the establishment of highly radiating plasmas with core parameters characteristics of a Demo device) would be to reduce $P_{\rm fus}$ by changing the tritium fraction $n_{\rm T}/(n_{\rm T}+n_{\rm D})$ in the plasma to $\sim$0.2 or $\sim$0.8 and to increase $P_{\rm aux}$ to compensate reduced α-particle heating [45]. This should permit a demonstration of the interplay of plasma radiation at the divertor and the plasma edge with plasma energy confinement and particle exhaust in conditions close to those expected in Demo.

Theory-based transport models self-consistently describing SOL, pedestal and core regions, are being tested against experimental data from the present machines but much work needs to be done before they can be used for extrapolations to a power plant. Available integrated models [46, 47] are not yet sufficiently mature. The least reliable features of these models are associated with transport in the edge pedestal and SOL regions. Both existing experiments and ITER are needed to mature these models. Due to the higher temperature and lower ν^* expected at $\bar{n}_{\rm e}/n_{\rm G}\sim 1$ in ITER (a regime not accessible in any current device), the physics processes determining the behaviour of the plasma edge in ITER might differ significantly from those of present experiments. Only ITER can make definitive experimental tests of the plasma edge physics for power-plant scale devices.

3.5. Pulsed loads and anomalous events

The very large plasma energy content in a power plant requires avoidance of any significant transient release of plasma energy on short time scales, such as those associated with ELMs and disruptions in present devices.

Pulsed energy fluxes through the plasma separatrix into the SOL due to type I ELMs are tolerable in present devices. However, in ITER and especially in Demo their amplitude is expected to increase to the level at which they could significantly shorten the divertor target lifetime [48]. As discussed in [30], a variety of mitigation techniques for type I ELMs have been suggested and tested with different levels of success [49]. One such technique, the frequent injection of small hydrogenic pellets, with each pellet triggering ELM, has been applied successfully in ASDEX Upgrade [36] demonstrating the reduction of the energy loss during ELM with increasing pellet frequency of the form $\Delta W_{\rm ELM}\propto f_{\rm p}^{-1}$. This technique appears promizing for ITER and Demo. Although simulations show that the HFS injection of DT pellets with parameters required for core fuelling in ITER (diameter $\phi_{\rm p}\approx 0.7$ cm, $f_{\rm p}\approx 4$ Hz, and velocity $\upsilon_{\rm p}\approx 500$ m s^{-1}) could reduce the ELM energy loss to an acceptable level [50], an independent ELM pace making by the low-field side injection of smaller pellets would be preferable to alleviate simultaneous ELM mitigation and plasma density control. It should be noted that application of this technique, which is still not fully optimized, is accompanied by a moderate reduction of the energy confinement time ($\tau_{\rm E}\propto f_{\rm p}^{-0.16}$) [36]. Type I ELM suppression has also been demonstrated using plasma edge ergodization by resonant magnetic perturbations produced by external coils [51]. This technique seems to require the coils located in the vicinity of the plasma surface that could be difficult in a reactor environment. A quiescent H-mode regime has also been shown on DIII-D, ASDEX Upgrade, and JT-60U. Clearly, more experiments, modelling and design studies are required to understand the physics mechanisms and to demonstrate the applicability of these methods in ITER and power-plant scale devices.

H-modes with small (type II) ELMs have been observed at high triangularity ($\delta\geqslant 0.4$) and high safety factor ($q_{95}\geqslant 3.5$) in most divertor tokamaks [30, 39]. In addition, ELM-free H-mode regimes with low-amplitude continuous magnetic fluctuations which increase particle transport at the plasma edge and prevent uncontrolled density increase have been observed [52]. ITER will provide access to the power-plant-relevant H-mode edge conditions in which such techniques can be tested and an acceptable approach to the operation of high performance long-pulse/steady-state plasmas with acceptable edge pedestal behaviour can be developed.

Plasma disruptions are a major concern for a power-plant scale device. Early projections to ITER based on a limited experimental database showed that during an unmitigated disruption the thermal plasma energy, $W_{\rm th}\sim 320$ MJ, will be lost in $\sim$1 ms, mainly to the divertor, producing a heat load up to 25 MJ m^{-2}, that is more than one order of magnitude higher than the vaporization threshold of the divertor target plates. Recent multi-machine studies suggest that the divertor heat load in ITER could be substantially lower, i.e. average values are expected to be of $\sim$3.3 MJ m^{-2} over a timescale of 2.3 ms [53]. This is mainly due to the finding that the energy of the plasma at the thermal quench is generally much smaller than that of full performance plasma, except of purely vertical disruption events and high-β disruptions in discharges with ITBs. Also, the profiles of the power fluxes at the thermal quench are a factor 5–10 broader than that at normal plasma operation. Nevertheless, the resulting power flux remains higher (by a factor of 3–4) than that required for carbon ablation or tungsten melting.

During the subsequent current quench phase, in-vessel components experience high mechanical loads caused by eddy and halo currents [38], and the runaway electrons can be generated by the avalanche process. The number of e-foldings supported by the avalanche mechanism is proportional to the plasma current and could be $\sim$40 in ITER at 15 MA. This is sufficient to ensure that the plasma will transfer a significant fraction (up to 80%) of its current to a runaway population, in contrast to present experiments where the generation of runaway electrons is mild. The runaway electrons could transfer up to 25 MJ to the first wall providing a highly localized heat load. It will be important to verify these predictions in ITER and to demonstrate disruption mitigation techniques capable of effectively mitigating halo current, runaways and power loads simultaneously.

Disruption-free operation is a prerequisite for Demo and power plant and is important for ITER. However, even with a high quality control system to avoid disruptions, some will occur. Therefore, understanding of the triggering mechanisms, identification of the disruption precursors and mitigation of disruption consequences are necessary. Current research indicates that massive gas injection could be successful in mitigating all the major consequences of disruptions, although the depth of the gas jet penetration into plasma and localization of radiation energy deposition on the first wall need further studies. A future variant of this technique applicable to larger devices might be the liquid jet. ITER must demonstrate a disruption mitigation method both for its own operation and for Demo.

3.6. Integrated operational scenarios

For the successful operation of a fusion power plant, it is not sufficient to maximize an individual parameter: it has to be demonstrated that all essential requirements can be satisfied simultaneously in an integrated scenario. As power-plant values of the relevant plasma physics parameters ρ^*, ν^*, β cannot simultaneously be achieved in smaller scale devices, such a demonstration requires an ITER-class device. The fact that nuclear (fusion) and atomic physics processes introduce additional dependences that cannot be expressed in terms of the above three plasma physics parameters further underlines the importance of a burning plasma at a scale close to that of a Demo/power plant plasma for the exploration of such integrated scenarios and their demonstration. The two scenarios outlined below constitute arguably the best current candidates for tokamak power plant operation in a long-pulse or a truly steady-state mode. In addition to bringing these (and other) scenarios into the power-plant range of parameters, ITER will also test their controllability and stationarity under conditions where the available power for non-inductive current drive will be only a relatively small fraction of the total heating power.

Improved hybrid scenarios. Hybrid operation with reduced plasma current ($q_{95} \approx 3.3$) driven by a combination of inductive and non-inductive means is planned in ITER to extend the duration of the fusion burn to allow testing of some technological systems. It was discovered in recent years that ELMy H-mode discharges with $q_{95} = 3.2$–4.5 and $q_0 \geqslant 1$ at low central magnetic shear exhibit improved confinement ($H_{98(y,2)} > 1$) and stability ($\beta_N > 2.5$) [39]. These regimes, known as 'improved H-mode' in ASDEX Upgrade, 'hybrid regimes' in DIII-D and 'high β_p regimes' in JT-60U, if realized in ITER, could allow $Q \geqslant 10$ to be reached at a reduced plasma current (10–14 MA) with a pulse length up to 3000 s. However, the physics of these favourable regimes is not yet fully understood. In particular, it is important to understand the current density evolution and the need for active current profile control. It is also important to assess the operational space for these regimes, especially at reactor-relevant β_N, ρ^* and ν^*, though this will be possible in full only in ITER.

Steady state scenarios. Steady-state operation is preferred for a fusion power-plant. Therefore, ITER aims at demonstrating steady state operation with $Q \geqslant 5$. This aim is more demanding than the main ITER goal of achieving $Q \geqslant 10$ in pulsed inductive operation, especially in terms of the requirements on β_N/l_i. Given the low efficiency of external non-inductive current drive techniques, steady-state operation requires minimizing the plasma current and maximizing the bootstrap current fraction (f_{BS}), which, in turn, requires maximizing β_N and improving τ_E relative to the scaling for the standard ELMy H-mode. Some degree of active RWM control is also likely to be needed [38, 39]. Significant progress has been achieved in present experiments in realizing the combination of enhanced normalized energy confinement, higher β_N/l_i and high non-inductive current fraction in discharges with moderate degrees of sustainment. In particular, a steady-state scenario with $q_{95} = 5.4$, $\beta_N = 3.4$, $H_{89} = 2.3$ and $f_{BS} = 0.58$ that projects to $Q \approx 4.7$ in ITER has been sustained for 0.8 s ($\sim 5\tau_E$) in DIII-D [54], albeit at relatively high l_i and under conditions where the plasma profiles are still continuing to evolve. Typically, these discharges can have weakly or strongly reversed magnetic shear with $q_{min} > 1.5$. The essential issue for both present-day experiments and ITER is the control of the current and pressure profiles. This control is a challenge because of strong non-linear coupling of the q profile, pressure gradient, bootstrap current and fusion power as these evolve in time. Studies in ITER at modest $Q \geqslant 5$ will be a necessarily prelude to the realization of the more-sophisticated control needed for Demo and beyond at $Q \geqslant 20$.

3.7. Diagnostics and H&CD for burning plasma experiments

Even routine operation of a power plant will be based on a large number of feedback and control systems, requiring continuous diagnostic input measuring key physics and engineering parameters [55, 56]. These diagnostics will have essential features tested only on ITER: not only will they have to provide measurements of standard tokamak parameters, but also information characterizing the thermonuclear burn process. These diagnostics need to work in a harsh environment, i.e. at high levels of neutron flux and fluence, nuclear heating and gamma radiation, which will give rise to the occurrence of radiation induced phenomena such as electromotive force, conductivity and absorption in the materials of diagnostic components, generate spurious signals and shorten the diagnostics lifetime. The diagnostic systems also have to satisfy stringent requirements on tritium confinement, remote handling maintainability and reliability. During ITER operation, extensive information will be accumulated that will provide the basis on which diagnostics for Demo can be selected and designed. After ITER operation, extrapolations associated mainly with higher fusion power and longer duty cycle will be needed.

ITER will equally be a key testing ground for long pulse high power H&CD systems with a performance capability and overall efficiency close to that required for a Demo device [39]. As in the case of plasma diagnostics, ITER

H&CD systems will operate in a hostile environment requiring a significant extension of current technology. In addition, and perhaps even more challenging, the requirements on launched power and pulse duration far exceed performance levels achieved in present H&CD systems. An extensive program of R&D is underway to develop H&CD systems to the level of performance required for ITER, but it is the routine operation of these systems and the demonstration that they can be used routinely in feedback loops to control plasma conditions which will establish the basis for the application of these systems in a Demo-class device.

3.8. Development of experience with power-plant relevant materials and in-vessel components in ITER

The mix of materials selected for ITER PFC surfaces (carbon, beryllium and tungsten) was based on balancing current experience (carbon in most fusion experiments) with concerns about tritium retention and power-plant relevance of the materials [31]. This approach allows ITER access to a wide range of burning plasma scenarios while making best use of the experience in plasma-wall interactions from current devices. There is a widespread consensus that, in the absence of significant new developments in the technology of PFCs for fusion devices, considerations such as PFC lifetime and power plant availability indicate that Demo will need to operate with high-Z PFCs, and tungsten appears at present to be the most suitable material for satisfying power plants requirements. ITER will be the only possible facility for the proof-of-principle testing of tungsten PFCs in advance of Demo and it can therefore be expected that ITER will ultimately address the issue of operation with tungsten PFCs in a power-plant environment. Such testing will include studies of power handling issues discussed above, the control of plasma contamination by tungsten and the extent to which tritium is retained in tungsten. Installation of a complete tungsten first wall in ITER will also necessitate the testing of tungsten's mechanical properties, particularly under irradiation, its resilience to electromagnetic loads induced by disruptions and the long term retention of its material properties when operated continuously at the high wall temperatures which would be required if helium were selected as the primary coolant for Demo.

The important mission of ITER is to test the blanket and divertor designs [57]. Functional tests of blanket modules and divertor cassettes will be conducted early in the experimental programme. Nuclear tests of Demo relevant blanket modules requiring long burn pulses ($\geqslant$1000 s) include the demonstration of a breeding capability that would lead in Demo to tritium self-sufficiency, extraction of high grade heat, and electricity production. The ITER blanket system is designed to make possible conversion (outboard area only) of the shielding blanket to the breeding blanket at a later stage of ITER operation.

3.9. Benefit of a tokamak burning plasma experiment for other magnetic confinement schemes

Among the magnetic confinement schemes so far investigated, the tokamak is currently best developed to allow confident extrapolation into the burning plasma regime. Other toroidal plasma configurations have, however, specific advantages—for example the intrinsic stationarity, high density limit and disruption-free operation of the stellarator, or the high achievable β in spherical tokamaks—that might still make them attractive as fusion power plants or as component test facilities. The commonality of many physics phenomena, but also the apparent differences (e.g. the appearance and the parameter regime of the density limit) promise to contribute substantially to our physics understanding.

The pioneering role of ITER in the exploration of burning plasma and power-plant scale physics will therefore be beneficial to all toroidal confinement schemes. Apart from the evident benefits from the development of radiation resilient and specifically burning plasma relevant diagnostics, H&CD systems and PFC materials, these benefits will be realized, in particular, through the development of a comprehensive theoretical understanding, which will allow the combination of information coming from distinct sources and the extrapolation to novel situations. Cases in point will be plasma confinement and α-particle heating. With computers contemporary to ITER operation, it should be possible to implement gyrokinetic simulations with coverage of the space scales from the ion-gyroradius to the full plasma dimensions. The variety of available experimental configurations (stellarators of the LHD, W7X and NCSX generations, spherical tokamaks and base-line tokamaks including ITER) will allow the predictions of such simulations to be tested over a broad range of geometries and characteristic plasma parameters. This should significantly strengthen the credibility of the extrapolation of simulation results towards novel scenarios, such as stellarators with values of ρ^*, ν^*, β characteristic of a burning plasma, and could provide confidence in the design of a thermonuclear follow-up experiment based on a stellarator with (depending on the aggressiveness of the roadmap followed) JET, ITER or Demo–like characteristics. A similar situation holds in the area of α-particle driven instabilities and their non-linear consequences, where existing stellarators, including those under construction, can explore the behaviour of fast particles produced by ICRF or NBI, and the ITER experience will provide key information on the relevance of such results for the extrapolation to α-heated cases.

A strong synergy between ITER and future stellarator operation can also be expected in all areas related to first wall and divertor physics and technology, where ITER will clearly contribute unique experience in the handling of high, stationary power fluxes, tritium retention and neutron radiation compatibility, but where superconducting stellarators like LHD or W7-X, due to their capability to operate quasi-steady-state at high plasma density, can also serve as relevant test platforms for many plasma/wall interaction issues. Evidently, other magnetic fusion configurations could also profit from the knowledge and the techniques developed and tested on ITER in the areas of plasma control, heating, fuelling and pumping, as well as ITER will benefit by physics and control studies and diagnostic development in other magnetic confinement schemes. Furthermore, engineering in the areas of superconducting magnets, heating devices, blankets, etc is common across all the magnetic configurations.

4. Summary

ITER, together with smaller satellite experiments operating during the same time frame, and a strong theory and modelling effort, has to provide the full physics information to allow the design and construction of a Demo device, with a confidence level commensurate with its investment and operation costs. A specific role of ITER is, of course, clearly defined by the first operation with dominant thermonuclear heating, and hence with a substantial contribution of suprathermal α-particles to the plasma pressure. Equally important is, however, its function of providing the ultimate scaling points in areas like transport scaling, operating limits and access requirements to improved confinement regimes. Thermonuclear power is also a surprizingly cost-efficient method of testing the feasibility of converting high fractions of the total heating power into electromagnetic radiation in this regime. The high power to the PFC surfaces, together with the nuclear environment, will provide key tests of the use of tungsten in a power-plant environment before taking the step to Demo.

Long pulse or steady-state operation of Demo (and ITER) will also differ from that in present devices, as the plasma must be sustained at high fractions of the Greenwald density, with only a modest fraction of total heating power available for direct, non-inductive current drive. Ultimately many transient effects (ELMs, sawteeth & fishbones, disruptions) on ITER and Demo can potentially be associated with such large pulsed energy and force loads that a very high confidence in avoidance strategies, and a reliable capability for mitigating their consequences, in case of malfunctions, has to be established. This is also related to some areas where the situation on ITER and Demo will involve even qualitatively new physics compared with present devices, such as the effects of α-particles or of runaway electron generation by knock-on collisions during disruptive current decay.

A commercially attractive power plant, and—according to the fast track logic—also Demo, will also need further physics progress in areas not strictly linked to size or α-particle heating, such as the achievable values of the stability parameter β_N/l_i, normalized plasma density n/n_G, radiation power fraction P_{rad}/P_{heat}, and normalized ELM amplitude W_{elm}/W_{ped}. Techniques to ensure this progress have to be developed by the physics programmes currently underway and by those which will accompany ITER in moderate sized devices. It is evident, however, that new results emerging from the experimental and theory programmes will also have to be validated on ITER before their consequences can be incorporated into the final design of Demo.

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