Critical Physics Issues for Tokamak Power Plants

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Abstract. Analysis of tokamak fusion power plants such as the EU power plant conceptual study (PPCS) makes clear that, to ensure that tokamaks can be exploited to generate electricity at economically attractive rates, improvements beyond the ITER baseline performance levels are required. Recent studies within the EU which extend the results of the PPCS analysis have pursued the development of a more quantitative description of physics performance in such devices. The PPCS study showed that high beta and high density appear explicitly in the derived scaling for the cost of electricity (CoE). Additional critical factors include the ability to handle the power exhausted from the plasma while maintaining high energy confinement quality, high beta and acceptable plasma contamination, achievement of sufficiently high current drive efficiencies to maintain steady-state operation, the exploitation of auxiliary heating and current drive systems to maintain the current profiles required for high confinement and mhd stability, and the use of active feedback control techniques for mhd instabilities. Steady-state operation is considered to be the preferred operating scenario for fusion reactors following ITER and has formed the focus of these studies, but the development of the "hybrid" scenario, with enhanced plasma confinement and mhd stability characteristics, offers a fall-back approach to power plant operation. The paper presents the results of the latest modelling analysis and discusses the implications for tokamak physics R&D.

1. Introduction

During the ITER Design Activities, the physics communities in the ITER partners have developed an extensive physics design basis for the ITER reference scenario and the possible extension of ITER's performance towards steady-state operation [1,2]. This physics basis gives confidence that ITER will achieve its essential performance goals. However, the analysis of tokamak fusion power plants such as the EU power plant conceptual study (PPCS [3]) shows that further improvements are required to ensure that tokamaks can be exploited to generate electricity at economically attractive rates. Analysis of the cost of electricity (CoE) within the PPCS study showed that [4],

$$\mathrm{CoE} \;\propto\; \left(\frac{1}{A}\right)^{0.6} \frac{1}{\eta_{th}^{0.5}} \frac{1}{P_e^{0.4}\beta_N^{0.4} f_{GW}^{0.3}}\,, \qquad (1)$$

with A plant availability, η_{th} thermal efficiency, P_e electrical power output, β_N normalized plasma beta and f_{GW} the ratio of the plasma density to the Greenwald value. While plasma beta and density appear explicitly in equation (1), several additional factors are implicit in the analysis underlying this scaling, such as the ability to handle the power exhausted from the plasma while maintaining high energy confinement quality, high beta and acceptable plasma contamination, achievement of sufficiently high current drive efficiencies to maintain steady-state operation, the exploitation of auxiliary heating and current drive systems to maintain the current profiles required for high confinement and mhd stability and the development of plasma control capabilities to ensure an acceptable disruption frequency (typically estimated to be in the range 0.1-1 per operational year) and to suppress β-limiting instabilities.

Recent EU studies extending the results of the PPCS analysis have stimulated a review of critical physics issues which influence the design of tokamak reactors following ITER, i.e. devices of the DEMO class with net electrical power in the range 1 - 1.5GW in steady-state operation, and have highlighted key questions which must be resolved to establish a convincing physics basis for these devices. These studies aim to develop an improved characterization of physics performance in reactor-scale devices by validating the 0-D systems code analysis used previously against 1-D and 2-D modelling of the plasma core and edge transport and 2-D mhd stability analysis.

2. Scenarios for reactor plasma operation

Steady-state electricity production is accepted as an essential requirement of a fusion power plant, and it is generally assumed that the most cost-effective approach to this goal involves steady-state plasma operation. US studies and preliminary EU studies confirm this, but the cost disadvantages associated with pulsed plasma operation may be acceptable if certain physics challenges involved in achieving steady-state plasma operation prove more difficult, or more expensive, to resolve than assumed. The need for a more detailed comparison between steady-state and pulsed tokamak operation, encompassing both physics and technological constraints, therefore remains an important issue for future power plant studies.

For the last decade or so, the “advanced scenario”, based on a plasma with reversed central magnetic shear and an internal transport barrier (ITB) producing higher confinement than a conventional H-mode, allowing access to high values of β_N via active control of resistive wall modes (RWM) and generating a significant non-inductive current via the bootstrap mechanism, has been the favoured approach to steady-state operation of a reactor. Table 1 lists key parameters of several recent power plant studies which rely on this approach – the physics parameters listed illustrate several of the physics challenges, in terms of mhd stability, plasma confinement and density, which must be addressed in developing a robust steady-state scenario by this route. A candidate ITER scenario for steady-state operation is also given, highlighting the direction in which ITER’s steady-state capabilities will need to be developed to advance the physics basis for this class of power plants.

More recently, the so-called “hybrid”, or improved H-mode, regime has also achieved higher confinement than the conventional H-mode and has several attractive features which may simplify the development of a reactor scenario. In current devices, hybrid discharges exhibit a q-profile with q_0~1 sustained by central mhd instabilities or non-inductive currents. The regime has achieved plasma performance, in terms of the confinement factor, $H_{89}\beta_N/q_{95}^2$, and the normalized (to the resistive diffusion time, τ_R) pulse duration, which is superior to, and typically more reproducible than, that obtained in advanced scenarios, as illustrated in Fig. 1 [5]. Moreover, the plasma parameters obtained in present experiments suggest that this regime might have the potential for achieving reactor-relevant values of β_N without active RWM control and that high confinement can be sustained at densities approaching n_{GW}.

Initial studies of a DEMO device utilizing a hybrid scenario indicate that, although initially considered an option for a more cost-effective long-pulse tokamak design, this regime might offer an alternative approach to steady-state operation, at the cost of a higher current drive requirement, but without the need for operation beyond the no-wall ideal mhd stability limit. In this analysis, using the PROCESS and HELIOS 0-D codes and assuming only moderate gain in β_N from RWM, 1GWe DEMO-class devices equivalent in size to the PPCS-C option

Table 1: Tokamak power plant studies and a candidate ITER steady-state scenario

Parameter	ITER-ss	ARIES-AT[6]	A-SSTR2[7]	PPCS-C[3]	VECTOR[8]	STPP[9]
B_ϕ (T)	5.2	5.8	11	6.4	5	1.8
I_p (MA)	9	12.8	12	20.1	14.6	31
R, a (m)	6.35, 1.85	5.2, 1.3	6.2, 1.5	7.5, 2.5	3.2, 1.4	3.4, 2.4
A	3.4	4.0	4.1	3.0	2.3	1.4
V (m^3)	800	350	470	1750	281	1120
P_{fus} (MW)	340	1720	4000	3400	2500	3300
P_{aux} (MW)	68	35	85	112	60	50
Mode	ss	ss	ss	ss	ss	ss
$H_{98(y,2)}$	1.3	2.0 (H_{89})	1.6	1.3	1.4	1.5
β_N	2.6	5.4	4.0	4.0	6.0	8.2
n_e/n_{GW}	0.83	0.9	1.2	1.5	1.0	0.66

are obtained with ~50% bootstrap current fractions. The penalty in auxiliary heating power associated with non-inductive hybrid operation appears small, ~20%, though the analysis requires further development.

Confinement properties of plasma scenarios foreseen for power plants remain uncertain and the assumptions made in modelling plasma performance, in particular in relation to the confinement quality relative to that of the reference H-mode database, $H_{98(y,2)}$, simply reflect target values. Our analysis of plasma confinement properties of candidate power plant scenarios in terms of first principles based transport models [10] are in agreement with an extensive analysis of transport properties of hybrid plasmas [11] which found that the predicted performance is sensitive to the boundary conditions, i.e. the H-mode pedestal height. Additional uncertainties are associated with the fact that reactor scenarios will have $T_e \approx T_i$ and have little momentum input, so that their rotation is likely to be small. A more robust predictive capability for these regimes is therefore required. Moreover, although it is generally assumed that the high α-power in a power plant guarantees access to the H-mode, uncertainties in present scalings for the H-mode power threshold imply that, when allowance is made for necessary core radiation losses from a reactor plasma, access to the H-mode could be more marginal than generally assumed, particularly in a DEMO where the fusion power density may (through choice) be lower than in a commercial scale device [12].

Achievement of adequate current drive efficiencies allowing steady-state operation to be sustained with economic levels of auxiliary heating power while maintaining current profiles consistent with high confinement and mhd stability is a further challenge. Power plant studies at conventional aspect ratio have typically aimed to generate 80-90% of the plasma current via the bootstrap effect, while low aspect ratio designs such as STPP are designed to achieve pressure driven current fractions of greater than 90%. The EU PPCS and DEMO studies have taken a less aggressive approach, with bootstrap current fractions of 40-60%, leaving a substantial requirement for auxiliary current drive.

The current drive efficiency of auxiliary heating systems, $\gamma_{CD}=n_{e,20}R_0I_{CD}/P_{aux}$ ($AW^{-1}m^{-2}$), is expected to increase with electron temperature [13], a prediction confirmed in numerous

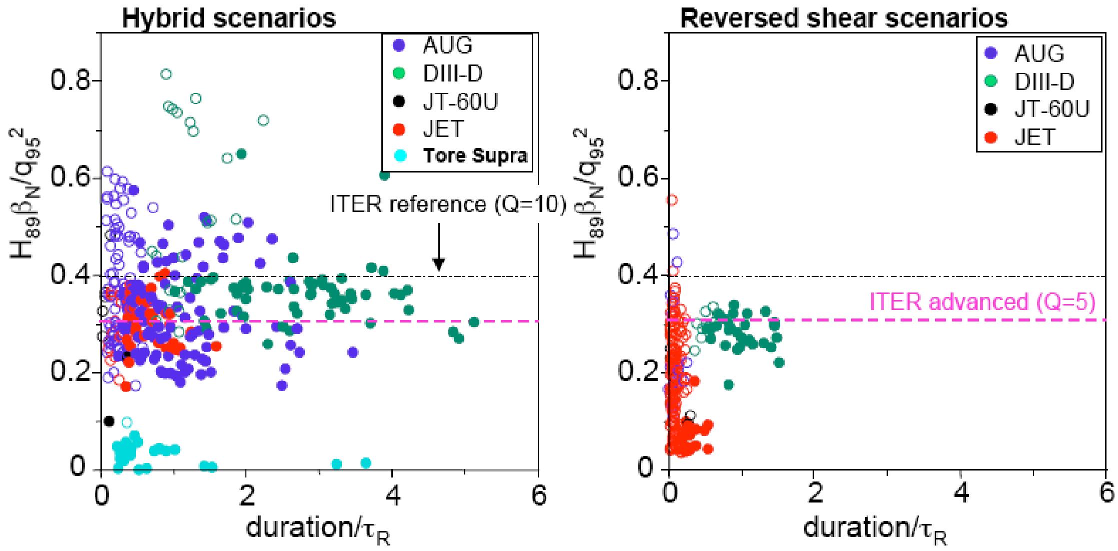

Figure 1: Comparison of the figure of merit $H_{89}\beta_N/q_{95}^2$ for hybrid and reversed shear scenarios as a function of regime duration normalized to the resistive diffusion time, τ_R, derived from the ITPA multi-machine database [5].

experiments, e.g. [14, 15]. However, an extension of the observed scalings by a factor of 2-10, reflecting the higher temperatures in reactor plasmas, is required to sustain an adequate level of auxiliary current drive in power plant scenarios. Current drive systems must achieve these efficiencies while controlling the current profile to sustain high confinement and to maintain mhd stability. This gives rise to additional complications: the absorption conditions for lower hybrid waves imply that in a reactor plasma, the current will be deposited in the outer 20% of the minor radius (Fig. 2). In addition, recent experiments in ASDEX Upgrade indicate that at high power levels, neutral beam injection achieves the predicted global current drive efficiency, but that the off-axis component of the driven current is lower than expected [16]. The relevant ICRF bulk current drive mechanism will limit this technique to on-axis current drive, and the lower efficiency of ECCD (γ_{ECCD}~0.1 compared with γ_{NBCD}~0.6 at T_e=20keV - see Table 2) will probably restrict its use to mhd control, where it can be invaluable, or to minor adjustments of the current profile.

Technological developments also have an important influence on the importance of auxiliary current drive in reactor scenarios and cannot be decoupled from the physics analysis, since plug-to-plasma power efficiencies of ~60% are often assumed, RF systems (LHCD, ICRF) must demonstrate efficient and reliable power coupling in the reactor environment, and neutral beam systems operating at 1.5-2MeV are incorporated in many analyses. A more ambitious approach to operation at high-β_N, and hence high (>80%) bootstrap fraction, partially alleviates these concerns, but at the cost of careful tailoring of equilibrium shape and plasma profiles to allow the required β_N values to be sustained (e.g. ARIES-AT in Table 1).

3. High density, highly radiating plasmas

High plasma density - typical cases which we have studied have a Greenwald fraction above unity - is required to allow efficient use of the plasma beta and efficient radiation of exhaust power to the reactor walls. Table 1 is indicative of the results of analyses of power plant scenarios: only the spherical tokamak study, STPP, can operate significantly below the Greenwald density, so that the establishment of high confinement scenarios at high density is

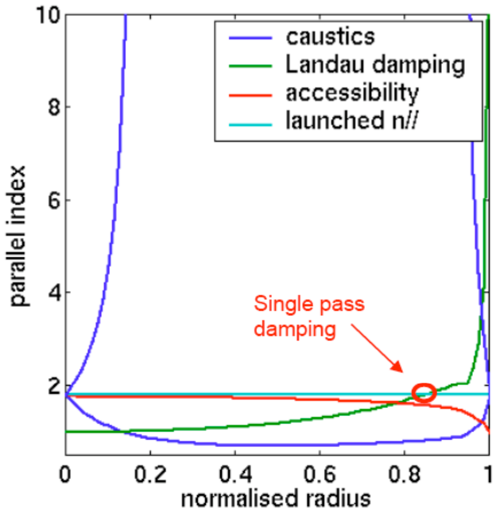

Figure 2: Accessibility and propagation diagram for LHCD in a DEMO-class device (R=7.4m): the waves are damped on the first pass and the deposition layer corresponds to the outer 20% of the plasma radius.

Table 2: Estimates of current drive efficiency for a DEMO-class device with $\langle T_e \rangle \sim$ 20keV (γ_{CD} as defined in text).

H&CD System	**γ_{CD} $AW^{-1}m^{-2}$**	**Deposition radius**
NBCD	0.6 – 0.7	$\rho \sim 0$
LHCD	~ 0.3	$\rho > 0.8$
FWCD	0.3 – 0.4	$\rho \sim 0$
ECCD	~0.12	$\rho \sim 0$

fundamental to the realization of a viable tokamak power plant with conventional aspect ratio (i.e. in the range 3-4). The maximum sustainable level of plasma density in a tokamak burning plasma remains uncertain, since the existing understanding of density limits, particularly in high confinement regimes, is essentially empirical, with several competing theories awaiting definitive experimental confirmation.

Early observations in which the density limit was explained in terms of a balance between input power and radiative losses have been superseded by more complex phenomena in diverted plasmas and in improved confinement regimes, where confinement degradation at high density is a common phenomenon. The relative importance of edge density and edge collisionality, as well as the contributions by bulk plasma and scrape-off layer/ divertor physics processes to the density limit in high confinement regimes remains open, e.g. [1]. The Greenwald density provides a good figure of merit for H-mode plasmas, but the higher confinement scenarios on which reactor regimes are based generally operate at lower densities and focussed efforts are required to demonstrate that the attractive confinement properties of these regimes can be sustained in the density range of interest to power plant operation. Some relaxation in the Greenwald factor might be obtained if the density peaking which has been observed recently in low collisionality plasmas [17] turns out to be a robust phenomenon. However, any gains might be offset by fuel dilution if high-Z impurity accumulation results (the probable need for tungsten plasma facing components is discussed below). The development of appropriate fuelling strategies, and the relevant technology, will also have an important role to play, since, at the reactor scale, the screening of neutrals by the SOL plasma is expected to decouple the core plasma density from that at the separatrix.

Handling of the exhaust power from a reactor plasma implies that significant progress must be made in combining high confinement with high radiated power fraction at the required level of plasma purity: in the absence of impurity radiation, the power crossing the separatrix in a power plant would be a factor of 5-10 greater than that in ITER, while the area of the divertor high heat flux components will probably differ by less than a factor of 2 from ITER's - and engineering constraints essentially limit the peak power flux which can be handled at divertor PFCs to $10MWm^{-2}$. At present, the favoured approach to power handling in a reactor combines tungsten plasma facing components, to ensure an acceptable erosion rate (and interval between PFC replacements), with a highly radiating edge/divertor plasma to distribute the exhaust power over a sufficiently large wall area. Impurity seeding must therefore be exploited to generate sufficient radiation, an extension of the strategy developed for ITER. However, our analysis confirms a requirement for a fractional radiated power (including bremsstrahlung and synchrotron radiation) of 80-90% of the total loss power.

Establishing a physics basis for highly radiating scenarios in power plants requires experimental demonstrations that the key elements, in particular high confinement and acceptable fuel dilution, can be sustained in plasmas with 80-90% radiation fractions. Experiments to date have focussed on H-mode plasmas with seeded impurities, but with predominantly carbon PFCs, and show evidence of confinement degradation at high radiation fractions [18]. These experiments must be extended to the advanced and hybrid scenarios favoured for power plants - promising results from initial studies in this direction have been reported by JT-60U [19] - and to experiments with high-Z PFCs, an aspect now being addressed in ASDEX Upgrade (e.g. [16]). An improved predictive capability for the power decay length in the SOL is also essential.

Recent EU studies of DEMO have addressed the relationship between radiated power fraction and plasma purity. Initial modelling of DEMO scenarios with the ICPS code has integrated first principles based core transport models with edge parameters defined by B2/Eirene edge/ divertor simulations and has obtained acceptable solutions, in terms of plasma purity and divertor power loading, for devices capable of generating 1GWe [20]. This analysis must be extended to incorporate more sophisticated SOL physics and to deal with high-Z PFCs. An alternative approach using the CRONOS suite has highlighted the important contribution made by bremsstrahlung and synchrotron radiation in such scenarios [21] and the need for more accurate treatment of these radiation loss channels (including significant central electron cooling and radiative energy transport by synchrotron radiation) in reactor plasmas (Fig. 3).

Such modelling and the power plant studies which accompany them assume implicitly that the amplitude and frequency of transient power excursions to the plasma facing components associated with ELMs and disruptions in existing tokamaks can be reduced to a level sufficient to avoid excessive erosion of the PFCs. Already a focus of substantial R&D for ITER, the mitigation and control of such transient loads (e.g. the acceptable disruption frequency is typically estimated in the literature at 0.1-1 per year) is essential.

4. MHD stability

It is recognized that increasing the plasma beta towards the limits predicted by ideal mhd theory allows the development of more compact and economic power plants. Operation at the highest attainable value of β_N also generates a significant bootstrap current, an essential element in steady-state operation of a tokamak reactor. In advanced scenarios, this implies operation beyond the "no-wall" β-limit, commonly characterized as $\beta_N < 4l_i$, requiring the active control of resistive wall modes, most likely via a close-fitting conducting wall and magnetic feedback control, as pioneered in the DIII-D tokamak (e.g. [22]). Considerable care has been taken in optimizing plasma equilibria and profiles in support of the development of power plants operating at $\beta_N > 5$ [23] in which the bootstrap current profile is well aligned with the optimum current profile, thereby simplifying the task of the external current drive

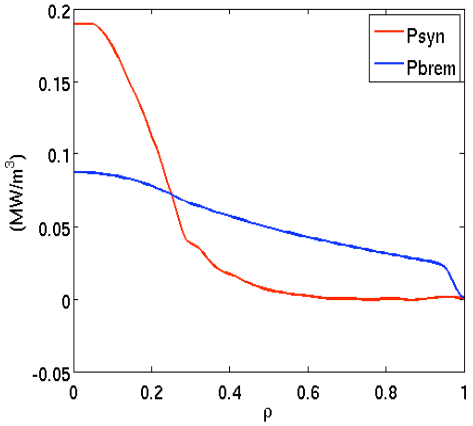

Figure 3: Profiles of synchrotron and bremsstrahlung emission analyzed for a DEMO class device with R=7.5m. An improved model is used for synchrotron radiation and transport, and wall reflectivity of 0.7 is assumed [21].

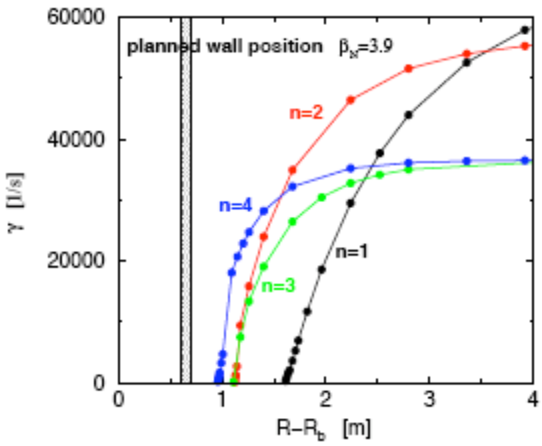

Figure 4: Calculated growth rates of ideal low-n modes (by CASTOR_FLOW [10]) for a plasma with R=8.1 and an optimized negative shear equilibrium such as developed in [23]. R is the radius of the stabilizing wall and R_b that of the plasma boundary.

systems in tailoring the required profile. However, the lack of adequate current drive and RWM control tools in existing experiments has prevented the full theoretical gain in β_N from being realized. A convincing empirical case for sustained operation at theoretically achievable values of β_N therefore must be developed by investment in the necessary auxiliary systems and feedback control tools. Such experiments would also allow the role of higher-n RWMs in limiting β_N to be addressed - Fig. 4 illustrates that modes with n>1 are likely to be most unstable as the β-limit is approached [10]. Additional issues include the role which rotational stabilization might play in a reactor (thought to be inadequate), how close to the plasma a stabilizing wall can be located, where the required control coils can be situated, and the importance of neoclassical tearing modes with m/n > 2.

The monotonic q-profile characteristic of hybrid operation will not allow access to the high values of β_N predicted for advanced scenarios with active control, implying lower bootstrap current fraction and a greater demand on external current drive (together with a reliance on the temperature scaling of current drive efficiency discussed previously). Moreover, active control of NTMs at the q = 3/2 and/ or 2 surfaces will probably be required to sustain operation close to the predicted β-limit. Stabilization via localized ECCD inside the NTM island, to compensate for the loss of bootstrap current associated with the flattened pressure profile in the island, has now been demonstrated in several experiments (e.g. [24]), but the development of a routine control tool is ongoing, and establishing a robust prediction for the required ECCD power in a reactor probably relies on experiments at the ITER scale.

The existence of a population of energetic α-particles, the novel feature of burning plasma experiments, will have a profound effect on the mhd stability of reactor plasmas [1]. Experiments in existing devices with energetic particle populations created by additional heating systems, or produced in initial DT experiments in JET and TFTR, have allowed many aspects of interactions between such populations and mhd instabilities to be studied and code predictions to be validated. ITER will provide a much more comprehensive access to the exploration of this aspect of plasma stability and α-particle confinement, a study which is critical for the development of self-consistent plasma regimes for tokamak power plants.

5. Conclusions

A tokamak power plant will require physics parameters which are simultaneously close to the limits of what is achievable on the basis of our experimental and theoretical understanding. The design studies underway are invaluable in establishing targets for plasma performance which must be verified in existing and planned tokamak experiments and in highlighting key directions for future research. The almost universal acceptance of the necessity of steady-state electricity production, together with the engineering advantages of steady-state tokamak operation, underline the need for an extensive programme of R&D to confirm the key physics elements underpinning this approach. In addition to the need for efficient non-inductive current drive, current profile control, mhd stability control and confinement enhancement (beyond H-mode levels), steady-state operation in a burning plasma will require plasma density of the order of the Greenwald value, radiated power fractions of 80-90% (with high-Z wall materials) and robust confinement of α-particles. Significant efforts are required in the current experimental, theory and modelling programmes to lay the basis for a successful demonstration of steady-state burning plasma operation in ITER, paving the way to DEMO. Combining all of these elements in a reliable operational scenario implies extensive R&D in developing and demonstrating the relevant plasma control methodology and tools. Finally, the recently developed hybrid scenario offers a potential fall-back operating regime, if it proves too difficult, or too expensive (in cost of electricity), to establish a viable reactor design based on the advanced scenario. Experiments to date suggest that the hybrid scenario provides a candidate regime for extended operation of tokamak reactors. albeit at a higher cost in terms of external current drive than the optimized advanced scenario.

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