26th EPS Conf. on Contr. Fusion and Plasma Physics, Maastricht, 14 - 18 June 1999 ECA Vol.23J (1999) 229 - 232

The Physics of Spherical Tokamak Power Plant Designs

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Introduction

The physics of steady state spherical tokamak (ST) power plant designs is discussed, employing an aspect ratio *A*=1.4 equilibrium to demonstrate theoretically that the beneficial features of STs can be combined self-consistently, taking account of constraints imposed by engineering considerations. Many of these features (eg high β, good confinement, disruption resilience, high natural elongation, etc) have already been demonstrated on START [1,2], while others, for example confinement scaling laws at tight aspect ratio, will benefit from additional data from the larger MAST device at Culham. Thus the spherical tokamak programme at Culham has a dual role: (1) to supply data to test theoretical models of tokamak plasma behaviour in extreme regions of parameter space (eg aspect ratio and elongation scaling of confinement) to assist the development of a device like ITER, and (2) to explore the possibilities for the ST as a route to fusion power production.

Bootstrap current in STs

The key to steady state tokamaks is the current drive, and therefore the need to optimise the fraction of pressure-driven current guides the development of the ST equilibrium to a large extent. There are two contributions to the pressure-driven current: the bootstrap current and the diamagnetic current, both of which can have a significant toroidal component in the ST. The bootstrap current fraction is typically the larger, and can be expressed as:

$$\frac{I_{\mathrm{bs}}}{I_p} \sim A^{1/2} h(k) b_N q_{\mathrm{cyl}} \quad (1)$$

where $\beta_N=\beta_v(\%)aB_v/I_p$, with *a* the plasma minor radius, B_v the vacuum magnetic field at the geometric axis R_0 and I_p is the plasma current; $q_{cyl}=I_{rod}/I_pA^2$ is the cylindrical safety factor with *A* the aspect ratio. The precise definition of β_v we use is $\beta_v=2\mu_0\langle p\rangle/B_v^2$ where $\langle p\rangle$ is the volume-averaged pressure. The function h(κ) is an increasing function of the elongation, which depends on the details of the plasma shape and current profile but is typically linear for plasmas with triangularity ~0.4.

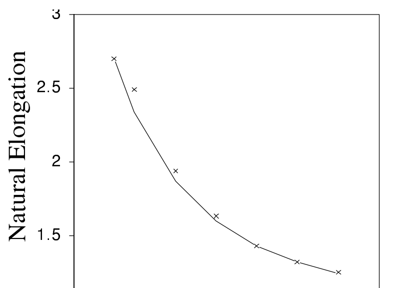

Figure 1: Natural elongation as a function of aspect ratio for a series of equilibria with constant vertical field and low internal inductance l_i=0.15 (crosses) compared to a fit $k=1+2.94A^{-3}$ (full curve).

Equation (1) is an important result for the design of any steady state tokamak, and shows that to obtain the full benefit of the bootstrap current in STs it is important to take account of their high natural elongation. This is shown in Fig 1, together with a fit demonstrating the strong dependence of natural elongation on aspect ratio, particularly for low internal inductance l_i, where $l_i = 2\int B_p^2\, dV/\left(m_0^2 R_0 I_p^2\right)$. Exploiting the high natural elongation one can readily demonstrate the existence of ST equilibria with approaching ~100% bootstrap current fraction at high β_N. Operation at high β_N is limited by MHD instabilities, and a detailed theoretical survey has demonstrated that by placing a wall at 1.3 minor radii, β_N approaching 9 can be achieved in an aspect ratio *A*=1.4 ST for plasma elongations exceeding $\kappa \approx 3$ [3]. Indeed these equilibria have up to 100% pressure-driven current, and we use this study as a basis for the power plant design presented here. However, we back off from the β_N-limit by ~10%, to allow some head-room for control of plasma current density profiles. The resulting equilibrium has a bootstrap current fraction approaching 90%, β_N=8.2, aspect ratio A=1.4, elongation κ=3.0 and the plasma current equal to the rod current. High triangularity, δ, is known to be beneficial when optimising MHD stability [3], yet low triangularity allows the ends of the centre column to be flared, thus reducing the power dissipated in it; we choose δ=0.45 for this study which represents a compromise between the conflicting requirements of the physics and engineering.

Confinement and auxiliary current drive

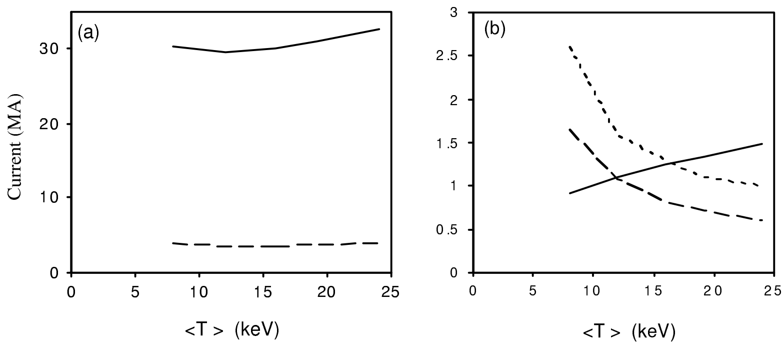

Figure 2: For fixed A=1.4, β_N=8.2, κ=3, δ=0.45, $I_p=I_{rod}$, R=3.4m, P_a=0.7GW, $n_{e0}/<n_e>=1.3$, $T_0/<T>=1.2$ (a) plasma current (full) and auxiliary current drive (dashed); (b) $<n_e>$ (dotted, in units $10^{20}m^{-3}$), density normalised to Greenwald value (dashed) and τ_E relative to ITER scaling law IPB98(y,1) (full curve).

The bootstrap current fraction is optimised for a broad pressure profile. Figure 2a shows both the plasma current and the auxiliary current drive which must be provided as a function of volume average temperature $<T>$ (assuming broad profiles for both density and temperature) to achieve 0.7GW of α-particle power keeping all other parameters fixed (given in the caption). We see that there is a shallow minimum in the plasma current required for the fusion power at $<T>$~11keV (this is higher than for more peaked temperature profiles), but current drive efficiency from the neutral beam injection system is increased at higher temperature where the density is lower (we are exploring alternative current drive schemes employing electron Bernstein waves, which may be more efficient at higher density). For example, neutral beam current drive calculations demonstrate that 110MW are required at $<T>$~16keV ($<n_e>=1.3\times10^{20}m^{-3}$), while 76MW are required at $<T>$~19keV ($<n_e>=1.1\times10^{20}m^{-3}$), which is the value chosen for this study. The density is then ~70% of the Greenwald limit, and an improved confinement regime, approximately 40% above the prediction of the ITER98(y,1) scaling law, is required (START has already exceeded ITER98(y,1) by ~20% and data from future STs such as MAST and NSTX will shed more light on this issue). The desired current profile is determined from MHD stability constraints and can be achieved with the neutral beam system: a monotonic q profile is adopted for simplicity, and we maintain $q_0 \approx 3$ to avoid low order rational surfaces and therefore reduce the risk due to the most dangerous neoclassical tearing modes (though these are expected to be less important in STs). The resulting current density and safety factor profiles are shown in [4]. The resulting parameters of the power plant are summarised in Table 1.

MHD Stability

MHD stability calculations show the plasma equilibrium is stable to high toroidal mode number, n, ballooning modes (second stability) and requires a close-fitting wall to stabilise n=1,2 kink modes [4]. The localised edge current density may drive higher n kink modes, localised at the plasma edge (ie 'peeling' modes). Such localised modes would not be expected to provide a hard operational limit, but may affect operation through driving Edge-Localised Modes (ELMs). On the other hand peeling modes are stabilised by a magnetic well, and this is expected to be large in this class of equilibria: work is under way to develop the numerical tools required to address these localised instabilities. Vertical stability calculations show that the stability index for this equilibrium is f_s=1.82, which demonstrates that there would be no problem maintaining control of the vertical instability and, indeed, higher elongation would be possible, providing a larger bootstrap current fraction (calculations show that 95% bootstrap fraction is obtained for κ=3.2, resulting in ~50% reduction in the NBI power required).

Parameter	Value	Comments
Major radius	3.4m	Set by neutron wall loading for 3.2GW fusion power
Aspect ratio	1.4	Minimises dissipation in centre rod
Elongation	3.0	High bootstrap current fraction, good vertical stability properties
Triangularity	0.45	Good MHD stability; some centre column "flaring"
Plasma current	31MA	Determined by fusion power
Centre rod current	31MA	$I_p=I_{rod}$ to avoid high recirculating power fraction
Pressure-driven current Total Bootstrap+Pfirsch-Schlüter Diamagnetic	 27.3MA 21.1MA 6.2MA	
β_N	8.2	Close to MHD stability limit
β_v, $\bar{b}$	58%, 41%	
l_i	0.14	
Electron density: vol avge central	1.1×10^{20} m^{-3} 1.4×10^{20} m^{-3}	Trade-off between optimum fusion power and current drive efficiency
Temperature: vol average central	19.2 keV 24.0 keV	
$H_{IPB98(y,1)}$	1.4	START measures values up to 1.2
t_E	1.9s	
$n/n_{Greenwald}$	0.71	
NBI CD power: on axis mid-radius edge	29MW, 500keV 32MW, 340keV 15MW, 50keV	On-axis CD may be provided by "potato" orbit bootstrap current [5] (not included). RF alternatives being explored (eg EBW)
q_0, q_{95}	3, 10	monotonic q-profile
τ^*_{He}/τ_E	4	
Z_{eff}	1.6	
Total fusion power	3.2 GW	
Avge neutron wall loading	3.7 MWm^{-2}	

Table 1: Physics parameters for ST power plant design $\bar{b} = 2m_0 \int p\, dV / \int B^2\, dV$

Exhaust and α-particle losses

To handle the heat loads from the plasma exhaust, we envisage operating in a double null configuration, with near equal power loading to the upper and lower target plates. The main problem is then associated with the inner strike points, where there is physically not much room to spread the power. On the other hand, the outer strike point is at a comparable radius to the inner strike point of a conventional aspect ratio tokamak, and can even be extended to reduce power loadings to a level similar to those in conventional tokamak power plant designs. It is therefore important in an ST that the majority of the power flows to the outboard side. Two features of the ST assist this: the ratio of outboard-to-inboard surface area of the plasma is large, and the high β operation results in a large Shafranov shift of the equilibrium, which increases the pressure gradient (and therefore heat flux) to the outboard side. To quantify this, we note that the ratio of heat to the outboard scrape-off layer to the heat to the inboard scrape-off layer, is given by:

$$R_H = \frac{\int_{\text{outer}} R^2 B_p \, dl}{\int_{\text{inner}} R^2 B_p \, dl} \qquad (2)$$

where the integrals are over the poloidal arc length between the maximum and minimum heights of the boundary. In deriving Eq (2) we have assumed that the plasma pressure and diffusion coefficient are flux surface quantities: any "ballooning" of the turbulent diffusion would tend to further increase R_H. Evaluating this ratio for the power plant equilibrium one obtains a value approaching ~25; that is 25 times more power flows to the outboard scrape-off layer than the inboard. The quantity R_H is plotted as a function of the inverse aspect ratio of the power plant equilibrium flux surfaces in Fig 3. In addition, scaling from existing data suggests a scrape-off layer width comparable to that of ITER; data from MAST will improve confidence in these predictions.

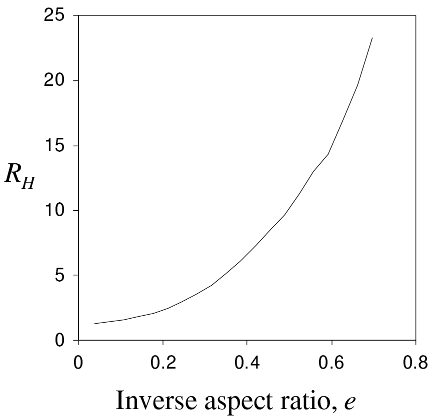

Figure 3: Ratio of outboard to inboard heat flux, R_H*, as function of a/R power plant flux surfaces.*

Calculations of fast particle losses show these will be small (<1%), largely because their orbits are pinched on the outboard side, as $|B|$ is increasing with major radius there (a feature of high β STs). In addition, data from START on fast particle driven instabilities suggests these may occur less in high β discharges.

Summary

We have described a steady state scenario for STs which is suitable for fusion power generation, concentrating on the physics issues but keeping in mind constraints imposed by engineering. This is compatible with our present understanding of MHD stability, confinement, fast particle effects, exhaust and current drive. These results suggest that the ST can provide a route to fusion power, and may be an alternative to the advanced tokamak. Nevertheless one should keep in mind that ST research is a relatively 'young' subject, which is evolving rapidly, and it is likely that future developments will modify the operational constraints assumed, and therefore some of the conclusions drawn here. Particular areas which require further development include current drive, MHD stability, confinement and exhaust. Data from the next generation of STs (eg MAST and NSTX) will help benchmark theoretical models, providing greater confidence in their use for predicting the performance of future tokamaks, including the more conventional approach as well as the ST. In addition, a number of alternative innovative concepts are emerging which further improve on the ST power plant design presented here [6]. These would have little or no rod current, which offers the opportunity for further improvements in fusion technology applications of magnetic confinement systems.

Acknowledgement This work was supported by the UK Department of Trade and Industry and Euratom

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