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The Physics Basis of a Spherical Tokamak Components Test Facility

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In an accelerated, "fast track" route to fusion power, the identification, development and testing of suitable materials in a fusion neutron environment is on the critical path. The accelerator-based IFMIF device will test small samples of materials, but it will not be able to test components and composite structures, such as scale models of breeding blankets. This will be one of the objectives of DEMO, the device envisaged to demonstrate the commercial viability of fusion power, which will follow ITER. Greater flexibility and a more rapid blanket testing capability would be possible if DEMO were supported by a dedicated, small-scale components testing facility (CTF). In order to remove the reliance on tritium-breeding, which could reduce availability (particularly in the earlier years of operation, before blanket designs are fully optimised), such a device should be sufficiently small that its tritium consumption is below about 1kg per year. We explore the possibility that a spherical tokamak (ST) can provide a suitable design for such a CTF.

We first consider scaling arguments to identify those plasma physics parameters that have most impact on the design. Working in a temperature (T) regime where the fusion cross-section scales as T^2, and adopting the IPB98(y,2) confinement scaling law (enhanced by a factor, H), we derive a scaling for the fusion power, P_{fus}, per unit plasma surface area, S:

$$S^{-1}P_{fus} \sim H^2 P^{0.62}\overline{n}^{0.82} I_p^{1.86} I_R^{0.3}\kappa^{-0.44}R^{-1.36} \qquad (1)$$

[heating power P (dominated by auxiliary heating), line-averaged density $\overline{n}$, plasma current I_p, toroidal field rod current I_R, elongation κ, major radius R]. P_{fus}/S is adopted as a measure of the neutron wall loading. The pressure is assumed to be confinement-limited rather than stability limited (appropriate for a small device). The device is envisaged to operate in steady state, so the current is a combination of bootstrap current and auxiliary current drive:

$$I_p = \frac{\overline{H}}{I_p^{0.07}}\left[1+\frac{c_\eta}{h(\kappa)}\frac{I_p P}{\overline{n}^2 R^3}\right]; \qquad \overline{H} = HP^{0.31}R^{0.82}\overline{n}^{0.41}I_R^{0.15}h(\kappa)\kappa^{-0.22} \qquad (2)$$

where the bootstrap current has a dependence on elongation, h(κ), which is typically linear, and c_η is determined from the current drive efficiency (assumed to depend only weakly on plasma parameters). Taking $c_\eta = C(I_p/I_R)^{0.07}$ for simplicity, Eq (2) has solution:

$$I_p^{1.07} = \overline{H}\left[1 - \frac{C}{h(\kappa)}\frac{P}{\overline{n}^2 R^3 I_R^{0.07}}\overline{H}\right]^{-1} \tag{3}$$

Substituting Eq (3) in Eq (1), we derive a scaling for the fusion power with plasma density:

$$\frac{P_{fus}}{S} \propto \frac{\overline{n}^{1.53}}{\left(1 - \lambda\overline{n}^{-1.59}\right)^{1.74}}; \qquad \lambda = \frac{CP}{h(\kappa)R^3 I_R^{0.07}}\left(\frac{\overline{H}}{\overline{n}^{0.41}}\right) \tag{4}$$

The quantity $\lambda\overline{n}^{-1.59}$ is the ratio of auxiliary current drive to total plasma current. Thus, bootstrap fraction can be adopted as the relevant dimensionless measure of the density. There is a minimum in P_{fus}/S when the bootstrap fraction is 64%. As the bootstrap fraction (density) is increased, P_{fus}/S increases; this is due to an increase in stored energy as the confinement improves. Thus the density, and P_{fus}/S, can only be increased until the β-limit is reached. At lower density (bootstrap fraction <64%) the auxiliary current drive efficiency improves, so the plasma current increases. This improves confinement, and so increases P_{fus}/S; the divergence corresponds to $I_p\to\infty$, as $T\to\infty$ (the auxiliary current drive is assumed to scale as $\overline{n}/T^2$). Kink modes would eventually limit performance as the density is reduced.

We shall find that the scaling favours high heating power, which therefore places us in the high auxiliary current drive, low density regime. Thus we use the kink stability limit to provide a scaling for $I_p\sim\kappa^2 I_R$. Eq (3) then provides a density scaling, so that Eq (1) becomes:

$$S^{-1}P_{fus} \sim H^{2.52}P^{1.3}I_R^{2.2}\kappa^{3.17}R^{-2.49} \tag{5}$$

We have assumed the auxiliary current drive dominates the bootstrap current. The achievable I_R increases with R (typically $\sim R$ for stress limits, $\sim R^{1.5}/\kappa^{0.5}$ for heating limits). The heating power would also increase with R (eg, more space to accommodate beams), so Eq (5) predicts larger devices are better. The maximum R is set by the tolerable tritium consumption. High confinement and high elongation are also strong drivers (the high elongation providing a higher current-carrying capability). [Recall that we assumed we are in a regime where fusion cross-section scales as T^2, so some care should be taken when interpreting these scaling results, which are only intended as a guide.]

We now describe a more detailed, quantitative study of a particular design, with parameters guided by the results derived above. An ST with aspect ratio 1.6 is our starting point. We should maximise R subject to the desire to maintain tritium consumption below 1kg per year, and yet maintain an acceptable neutron wall loading for testing. This guides our choice of R=0.75m. There is a benefit to operating at high κ, and we choose a value of κ=2.5, which satisfies vertical stability constraints (stability index f_s=2.0). The toroidal field current is the maximum that can pass through the centre column: 10.5MA for a single-turn copper centre rod, set by a tensile stress limit. No-wall kink stability limits [1] then suggest that a plasma current of 8.0MA should have a comfortable margin at modest normalised pressure, β_N=3.4.

The up-down symmetric, double-null equilibrium can be provided with 3 pairs of poloidal field (PF) coils, which are far from the plasma, provided the current profile is not too hollow (Fig 1). In this case, a central safety factor of q_0=1.46 was achieved through a slightly hollow current profile. [We are also beginning to explore a design with a more hollow current profile (with q_0=2.07, f_s=2.5), which has better Mercier and high toroidal mode number MHD stability properties near the core. The safety factor profile is monotonic in both cases.]

We have explored the no-wall stability to kink modes for the toroidal mode number n=1 and found this case to be stable provided the triangularity δ>0.47. This can just be achieved with the free boundary equilibrium, but is marginal and therefore weak resistive wall modes may be a possibility. Employing higher κ may be a beneficial option.

For the current drive, we rely mainly on neutral beam injection (NBI), which we find has an adequate current drive efficiency at a line-averaged density $\bar{n}=1.8\times10^{20}\text{m}^{-3}$, well below the Greenwald limit. The pressure-driven current is found to be 3MA. There are two components required from the auxiliary system: one off-axis and one on-axis. The off-axis component can be provided by 40MW, 150keV beams at a tangency radius of 0.83m angled at 40^0 to the mid-plane. There is a choice for the on-axis current drive: a 10MW, 200keV beam, a positive ion 20MW, 150keV beam, or 20MW electron cyclotron (EC) waves (O-mode) at 160GHz (2nd harmonic). One advantage of such an RF scheme is that it frees up space for components testing.

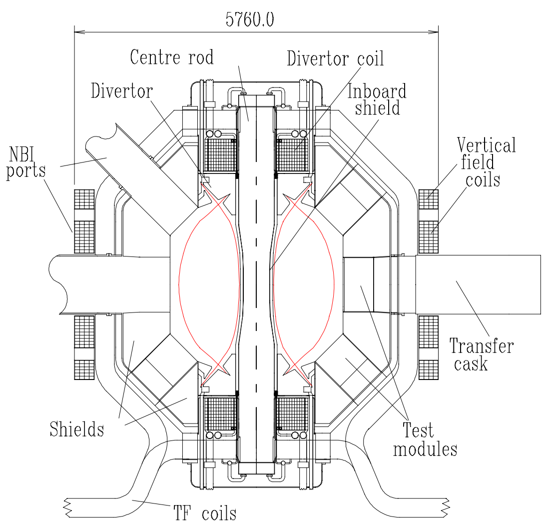

Fig 1: Load assembly for baseline CTF

NBI calculations predict ~12MW of fusion power from beam-thermal interactions, in addition to 35MW from the thermal ions, (ie Q~1). Full orbit calculations show negligible prompt loss of α-particles. The resulting confinement time is 98ms, requiring H=1.3. One route to achieving such confinement may be through controlling or eliminating sawteeth. The fuelling rate from the 60MW beam system is significant: 2.5×10^{20} particles per second. This suggests a further advantage for (EC) waves: to decouple heating from fuelling. Estimates for tritium pellets predict that firing five 2mm pellets per second into the plasma at $250ms^{-1}$ provides one option for core T fuelling if all D beams were employed.

We estimate the scrape-off layer (SOL) thickness as in [2]: 8mm (inboard) and 6mm (outboard). Assuming 90% of the heat goes to the outboard side (see [2]), we find the inboard target heat load is $10MWm^{-2}$ if the target plates are inclined at 15^0. The outboard target load requires a more novel solution. We are exploring the possibility of a cascading curtain of SiC pebbles, 2-3mm in diameter, which fall from a hopper at the top of the device, in front of the target plates. The high ratio of poloidal to toroidal magnetic field in the ST is key to the viability of this scheme. Approximately ¾ of the power can be removed by the curtain, again leaving $\sim 10MWm^{-2}$ to be handled at the outboard target plates.

We have performed neutronics calculations using the MCNP radiation transport code [3], with an accurate model for the CTF geometry. The result is that the mid-plane test modules (providing $6m^2$ of test area) receive $1.6MWm^{-2}$, while the others ($6m^2$ of test area) receive $1.4MWm^{-2}$. These figures are for the energy passing through the wall carried by the direct (ie without collisions) neutron flux. Thus, a goal to achieve a total neutron test fluence of $6MWyrm^{-2}$ within 12 years [4] is possible with a realistic availability of 31%.

In conclusion, a first analysis of the detailed plasma physics properties of a CTF design based on the ST suggests that such a device is feasible. Compared to MAST and NSTX, CTF plasmas will have: ~20× lower collisionality (normalised Larmor radius and β are close), higher power/momentum injection, higher fast particle content, hollow current profile. Experiments and modelling should aim to reduce uncertainties associated with these. In addition, demonstration of non-solenoid start-up and non-inductive current drive is crucial. The design meets all of the criteria that were identified in an international review of needs and requirements of a CTF [4]. While there remain a number of uncertainties, and the design is not yet fully self-consistent and optimised, we conclude that the ST does provide a possible option for a steady state components test facility which is worthwhile pursuing.

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