A Steady State Spherical Tokamak for Components Testing

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Abstract. The possibility that a small, steady state spherical tokamak can provide the fusion conditions necessary for components testing is explored. Such a device would operate in parallel with DEMO, complementing and extending the anticipated database from IFMIF, and helping to reduce the risk of delay during this crucial phase of fusion power development. The paper primarily addresses the plasma physics and neutronics issues, to explore the feasibility of such a spherical tokamak design, and also discusses certain aspects of the engineering design.

1. Introduction

One of the crucial aspects of fusion research is the optimisation and qualification of suitable materials and components. IFMIF will provide rigorous testing of small material samples to enable the design and construction of DEMO. It is then envisaged that DEMO will prove the components and structures, to optimise the design of tritium-generating blankets for fusion power plants, for example. It is useful to explore additional options for components testing, which could be used in conjunction with DEMO, to reduce the risk of delay during this phase of fusion power development. To this end, we are exploring the possibility that a small, steady state spherical tokamak can provide the conditions for effective components testing [1,2].

It is worthwhile considering the conditions necessary for effective components testing in a fusion neutron environment. These were identified by an international study, and published in Ref [3]. Some of the main conclusions of that report are:

- The neutron wall loading should be in the range 1-2MWm^{-2}
- The device should operate in steady state
- The total neutron fluence should exceed 6MW-yrm^{-2} within a reasonable time (eg ~12yrs)
- The total test area should exceed 10m^2
- The magnetic field strength should exceed 2T

These are the targets that guide our design. An additional issue is whether or not the device should be reliant on tritium generation. We consider a low tritium-burning device, ~1kg per year, where breeding is unnecessary. De-coupling the elements required to operate the device from those it is testing is clearly desirable (though probably not essential). We therefore explore in this paper whether or not such a device is feasible. This complements the approach of [4].

The paper is organised as follows. In the following section, we provide some simple scaling calculations, which are useful to help us understand the roles of the various tokamak parameters that we can control. These scaling calculations then guide the particular choice of the equilibrium that we adopt, and study in some detail in Section 3. In Section 4, we describe some of the engineering aspects of the design, together with the neutronics study that we have performed. Finally, we close in Section 5 with a discussion of how well the design meets the objectives; what further work is required to advance the components test facility (CTF) design, and a possible strategy which could bring forward the operation of CTF.

2.Analytical scoping study

In order to illustrate the impact of the plasma parameters, we have performed an analytical scoping study. Although not quantitatively rigorous, such an exercise helps to guide our choice of design. For simplicity, we consider a fixed, tight aspect ratio.

The quantity we are interested in for the purpose of a CTF is the fusion power, P_{fus}, per unit surface area, S, which scales as

$$\frac{P_{fus}}{S} \sim \beta_N^2 \frac{I_p^2 I_R^2}{R^3} f(T) \tag{1}$$

where $\beta_N \propto \beta aB/I_p$, β is the ratio of plasma stored thermal energy to magnetic energy, $B \propto I_R/R$ is the toroidal magnetic field (assumed to dominate the poloidal field), with I_R the current in the centre rod. Finally, I_p is the plasma current, R is the major radius and $f(T)$ represents the variation of the fusion cross section with temperature T ($f(T)=\langle\sigma v\rangle/T^2$). Rather than β_N, we choose to work with the confinement enhancement factor H, which is the factor by which the confinement time is enhanced above the IPB98(y,2) scaling law prediction:

$$H \sim \beta_N \frac{I_p^{0.07} I_R^{0.85} \kappa^{0.22}}{n^{0.41} R^{0.82} P^{0.31}} \tag{2}$$

so that the fusion power becomes:

$$\frac{P_{fus}}{S} \sim \frac{H^2 P^{0.62} n^{0.82} I_p^{1.86} I_R^{0.3}}{\kappa^{0.44} R^{1.36}} f(T) \tag{3}$$

Here, we have defined the heating power, P (assumed to be dominated by the auxiliary heating power), the plasma density, n, and the elongation, κ. We can reduce the number of parameters by assuming that the total current is a combination of neutral beam current drive and bootstrap current. The current can then be expressed as

$$I_p = \beta_N I_R \left[h(\kappa) + c_\eta \frac{I_p P}{n^2 R^3} \right] \tag{4}$$

where the first term is the contribution due to the bootstrap current, and the second is the neutral beam current drive. The form of $h(\kappa)$ depends on the current profile, but is typically linear. The neutral beam current drive efficiency is assumed to scale approximately as T/Rn, so the coefficient c_η depends only weakly on the plasma parameters. Using Eq (2) to eliminate β_N in favour of H, and for simplicity supposing a weak scaling of $c_\eta=C(I_p/I_R)^{0.07}$, allows us to derive a scaling for the plasma current, and hence the fusion power:

$$\frac{P_{fus}}{S} \sim \frac{H^{3.73} P^{1.16} I_R^{0.56} R^{0.07} h(\kappa)^{1.74} f(T)}{\kappa^{0.83}} \frac{n^{1.53}}{\left(1-\lambda n^{-1.59}\right)^{1.73}}, \qquad \lambda = \frac{C P^{1.31} H I_R^{0.08}}{R^{2.18} \kappa^{0.22}} \tag{5}$$

The divergence at a critical value of density $n_c=\lambda^{0.63}$ corresponds to a divergence in the solution for the plasma current, which occurs due to the following feedback mechanism in the equations. The auxiliary current drive increases as the density is decreased. This leads to an improvement in confinement, and therefore an increase in temperature, which further amplifies the plasma current. At the critical density, the amplification leads to a runaway situation, and the current is infinitely large. [The divergence in Eq (5) is, of course, unphysical, but the equation is nevertheless a useful indicator of trends.] The fusion power has a minimum at a density n_0, given by

$$\frac{\lambda}{n_0^{1.59}} = 0.36 \tag{6}$$

Two optimal regimes therefore exist on either side of this minimum. The ratio $\lambda n^{-1.59}$ is simply the ratio of auxiliary current drive to total plasma current. Thus, at higher density the associated improved confinement leads to higher β_N and therefore higher fusion power; the bootstrap current fraction exceeds 64% in this regime. At lower density the NBI current drive becomes more efficient, resulting in higher total current (but lower bootstrap current fraction), again improving confinement and thus the fusion power. Because the current increases, β_N decreases. The high density, high bootstrap fraction regime is exploited by ST power plant designs [5], for which high bootstrap current fraction is a key requirement for economic reasons. We do not rule it out as a possible regime for CTF to operate in, but satisfying the constraints imposed by magneto-hydrodynamic (MHD) instabilities, such as the resistive wall mode, would require an extended research programme, which could delay construction of the device. We therefore consider the opposite, low plasma density regime, with modest β_N and bootstrap fraction. The limit for how far the plasma density can be lowered, and the plasma current increased, is set by the ideal MHD kink instability of the plasma (and the fusion power cross-section, which will begin to fall off through the $f(T)$ factor). The limit to the plasma current imposed by the kink mode can be raised by maximising the toroidal field, subject to engineering constraints.

3. Detailed plasma physics study

The analytical calculation of the previous section leads to the following guidelines for designing a compact CTF based on the spherical tokamak. First, the toroidal field should be maximised, subject to engineering constraints, to provide the maximum plasma current within kink stability limits. The maximum plasma current should then be chosen, limited either by the ideal MHD kink mode or the current drive efficiency. The plasma density should be low to optimise the auxiliary current drive efficiency, keeping in mind that we wish to remain within the regime where the fusion cross-section parameter $f(T)$ is approximately constant . Equation (5) reveals a very strong dependence of the fusion power on H, and identifying a regime with good confinement (relative to the IPB98(y,2) scaling law) is clearly crucial. Equation (5) also suggests a relatively weak dependence on the device size, but this is misleading. For example, a large device could accommodate a larger toroidal field current, but would also require more tritium: a compromise is required.

With the above guidelines, we explore the properties of a CTF design with the parameters listed in Table 1. We have specifically opted for a compact device to minimise the tritium consumption. A minimum size is set by the aim to achieve a total test area exceeding 10m^2. The chosen aspect ratio of 1.6 represents a compromise between a desire to maximise the TF rod current (ie to maximise the radius of the centre column) and yet retain the beneficial feature of the spherical tokamak that relatively few neutrons are lost to the inboard vessel wall. Operation at high elongation permits higher plasma current to be achieved, but the precise reason for this depends on what sets the plasma current limit. If the limit is set by the kink mode, then the maximum current is proportional to $(1+\kappa^2)$ [6]. High elongation clearly has strong benefits in this case. If, on the other hand, it is the current drive efficiency that sets the limit, higher elongation is helpful as it provides a higher bootstrap current, leaving less required of the auxiliary current drive system. The value we have chosen, κ=2.5, provides a cylindrical safety factor of q^*=1.86, which is comfortably above kink stability limits for plasmas with optimised profiles [6].

Parameter	Value
Major/minor radius (m)	0.75/0.47
Elongation	2.5
Triangularity	0.4
Plasma current (MA)	8.0
TF rod current (MA)	10.5
β_N	3.5
Average density ($10^{20}m^{-3}$)	1.8
Average temperature (keV)	11.1
Press-driven current (MA)	3.0
CD power (MW)	50-60
$H_{y,2}$	1.3
P_{fus} (thermal+beam) MW	35+15
Neutron wall loading MWm^{-2}	1.6 or 1.4

Table 1: *Parameters of the CTF design in this study.*

The remaining global MHD parameter is β_N. This must be chosen to provide adequate fusion power, which needs to be in the region of 35MW to provide a neutron wall loading of $1MWm^{-2}$. A full 2-D equilibrium reconstruction, with an associated derivation of the fusion power, shows that $\beta_N=3.5$ is adequate. The calculations of Ref [6] indicate that this is well within the ideal MHD pressure limit for optimised plasma profiles. We have performed stability analyses of our equilibrium, where profiles are not necessarily optimised for stability, with both the MISHKA and KINX ideal MHD stability codes. We find that there tend to be strong internal modes if the central safety factor is below a critical value, $q_0<1.5$. This points towards a need for significant off-axis current drive, to which we shall return later. For $q_0=1.5$ the mode is stabilised with the wall on the plasma. Moving the wall to a large distance from the plasma (ten times the plasma minor radius) we find a weak instability, but this is suppressed by increasing the triangularity from 0.4 to 0.45 (see Fig 1). Thus, provided the current profile is sufficiently hollow, it seems that the $n=1$ resistive wall mode can be avoided. Future work will aim to identify equilibria that are more robustly stable.

The CTF must operate in steady state and we must therefore provide a scheme for non-inductive current drive. This is optimised at low density, but too low a density has penalties through reduced confinement and, at very low densities, a poor fusion cross-section. A line-averaged density of $1.8\times10^{20}m^{-3}$ and central density of $2.3\times10^{20}m^{-3}$ proves adequate to provide the required current drive efficiency from a positive ion neutral beam system. The corresponding central and volume-averaged temperatures are 25keV and 11 keV respectively. [For simplicity, we assume equal ion and electron temperatures, but, as we shall see later, transport calculations predict a somewhat higher ion temperature, so our calculations of fusion power may be pessimistic.]

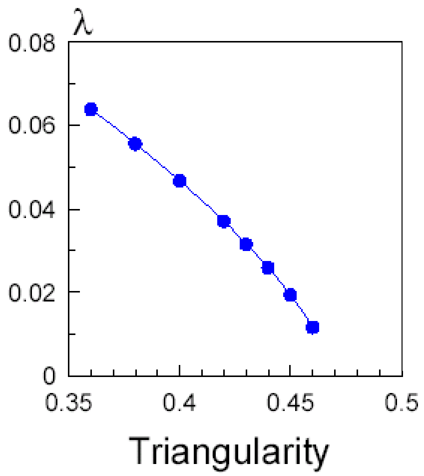

Fig 1: *The effect of increasing triangularity on the n=1 ideal MHD stability of CTF, with the wall far from the plasma. λ is proportional to the growth rate.*

The current drive efficiency is determined as follows. We have used the ion gyro-orbit code, LOCUST, to calculate the collisional evolution of the fast ions from the neutral beam system. The current near the magnetic axis is most effectively provided by a 200keV neutral beam with 10MW of power. This would require negative ion beam technology, building on the developments for ITER. An alternative is to use conventional, positive ion beams at 150keV, though higher power is then required (20MW). The beam is injected along the mid-plane, at a tangency radius of R_T=0.83m. For the off-axis current drive, a 150keV system is optimal, and 40MW is sufficient to provide the required current. These

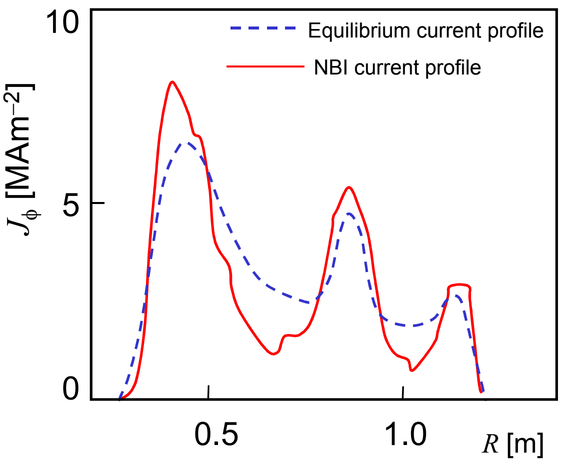

***Fig 2:** Plot of "target" current profile used for MHD stability analyses versus the current profile predicted by the 150keV NBI system (full curve), across the plasma mid-plane.*

off-axis beams would be injected at 40^0 to the horizontal at R_T=1.1m, accessing the plasma through ports above the mid-plane (see Fig 3). We shall see in Section 4 that the mid-plane has the highest neutron wall loading, so it is important to leave as much space here as possible for the testing modules. The predicted current-drive profile is similar to that required for the MHD stability, as shown in Fig 2, but more work will be required to develop a fully self-consistent scenario.

We have also explored the possibility of using electron cyclotron waves for current drive at the magnetic axis. The cut-off density is usually very low in a spherical tokamak, but the relatively high magnetic field (for a spherical tokamak) in CTF does permit 160GHz waves to be employed in the O mode, exploiting the second harmonic resonance. Calculations with the ray-tracing, Fokker-Planck code, BANDIT 3-D, predict that ~20MW could provide the required on-axis current drive.

The final plasma physics issues we address are related to confinement, fuelling and exhaust. The fusion power from the thermal ions is calculated to be 35MW (assuming equal ion and electron temperature). We have used the LOCUST code to predict the additional fusion power arising from the beam-thermal interaction and find that it is quite significant ~15MW. Thus there is a total of ~10MW of α-heating in addition to ~60MW of auxiliary heating. Calculations of α-particle losses, retaining finite Larmor radius effects, show that these are acceptably low, at the level of a few %, even allowing for the toroidal field ripple. Although the banana orbits are large, the effect of the poloidal magnetic field (which is significant on the outboard side in an ST) acts to "pinch" the orbits on the outboard mid-plane, thereby reducing losses. For our plasma parameters, we find that we need to achieve a confinement enhancement factor above the prediction of the IPB98(y,2) scaling law of H=1.3. This corresponds to an absolute confinement time of 98ms. This H factor has been comfortably exceeded in rotating MAST discharges, where an increase of H with rotation is observed [7].

Turning to the fuelling, more than half of this can be provided by the beams. In particular, a 20MW, 150keV core beam supplies sufficient on-axis fuelling to provide a central density ~50% higher than the pedestal density, taking a particle diffusion coefficient of $0.4\text{m}^2\text{s}^{-1}$. This does not take account of any particle pinch, which may be operative. The density peaking we have assumed is broadly consistent with this. The main advantage of such density peaking is that it improves the current drive efficiency of the edge neutral beam system.

We are beginning to perform more detailed transport modelling with the ASTRA transport code. An interesting result is that the majority of the neutral beam power goes into the ions, resulting in a predicted ion temperature ~50% higher than the electron temperature. All the calculations described above make the apparently pessimistic assumption of equal ion and electron temperatures. Future iterations of the design will exploit the higher ion temperature, and should permit operation at reduced β_N (close to 3) and lower H (close to 1.1).

Finally, we turn to consider the exhaust properties, which are particularly demanding. Double null operation has the advantage that the majority of the exhaust heat goes to the outboard side, where it is easier to handle. We assume ~50% of the heat can be radiated (and have included a tungsten impurity at the level such that Z_{eff}=3), and that 95% of the heat flows to the outboard side, as observed in MAST. Using an average of those physics-based scaling laws for the scrape-off layer (SOL) width that provide a satisfactory fit to the MAST data, we derive a mid-plane width of 8mm (6mm) for the inboard (outboard) SOL. Angling the target plates at 20^0, we then predict a tolerable $10MWm^{-2}$ on the inboard side. On the outboard edge, the SOL magnetic flux surfaces expand by a factor 3.5 between the midplane and the target plates. This results in a heat load of $\sim 50MWm^{-2}$, which cannot be handled by conventional schemes. A possible solution is proposed in the next section. If only 20% of the heat can be radiated, a higher flux expansion factor ~6 would be required on the outboard SOL to limit the power to $\sim 50MWm^{-2}$.

4. Engineering design and neutronics study

The basic cross-section of the device is shown in Fig 3. The design has many features in common with our ST power plant design (refer to Ref [8] for details). The coils are all water-cooled, normal-conducting copper. The toroidal field is generated by a single-turn coil consisting of a shielded central rod, carrying 10.5MA, and 10 return limbs. The poloidal field coils have positions and currents deduced from a free boundary equilibrium calculation. During maintenance, the divertor coils, P1, can be removed complete with the centre column in a relatively straightforward process. One consequence of the hollow plasma current profile is that it complicates the design of the poloidal field coils. In particular, the coils labelled P2 must be placed inside the toroidal field coil return limbs, but outside the vessel, otherwise the plasma tends to elongate uncontrollably. Equilibria with more peaked current profiles can be constructed with both outboard poloidal field coils, P2 and P3, outside the TF return limbs, which is better for maintenance, and therefore availability. Vertical stability calculations for the derived hollow current profile, with a suitable representation of the vessel structure, shows that the plasma is easy to control, having a stability index, f_s=2.0.

The vessel is primarily constructed from a low activation steel (eg EUROFER or F82H). There are ports on the mid-plane for the blanket test modules (right hand side Fig 3) and the on-axis neutral beam injection system (left hand side). Above and below the mid-plane is a second set of ports for the off-axis beam system, diagnostics and additional component testing modules. The design provides a total of $12m^2$ of testing area. A cask, shown fitted to the mid-

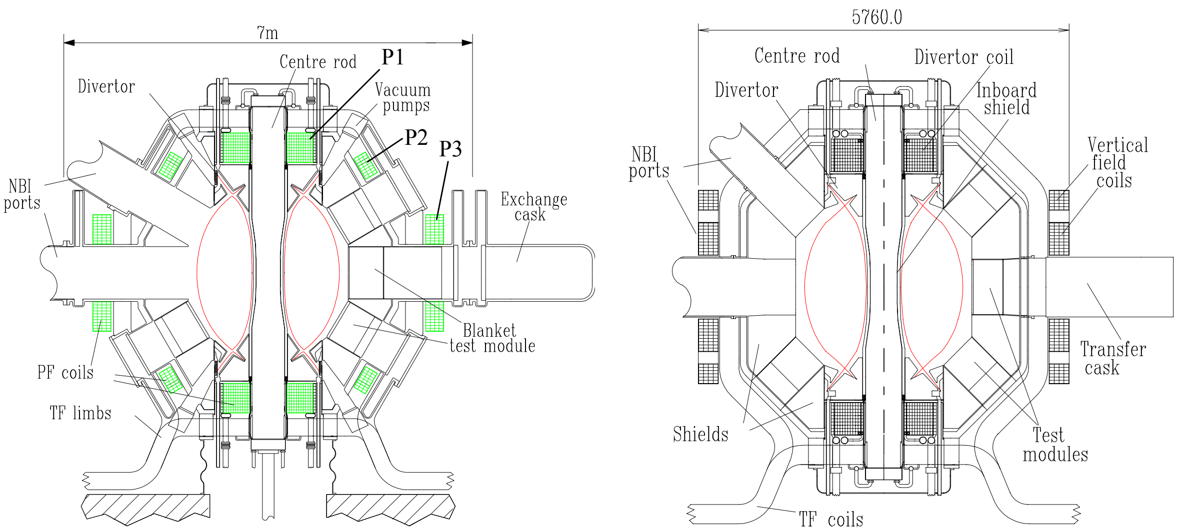

Fig 3: *Cross-sections of the design for the components test facility, showing the main components. On the left is the design required for hollow current profiles; for more peaked profiles the P2 coils can be placed outside the TF return limbs (right).*

plane port on the right of Fig 3, can be temporarily attached to allow removal of activated modules for assessment without exposing either them or the vessel to air.

We have performed a neutronics study of the CTF design using the MCNP Monte-Carlo radiation transport code. The components which we model are shown in Fig 4 and include: the centre rod and its steel casing, the inboard shield, the divertor structure and coil, the first wall, the test blanket modules and the vacuum vessel. The geometry is divided into 63 cells, each uniformly filled with a material representative of the actual materials in that vicinity. The low activation steel, F82H, has been used for this study (eg for the vessel, blanket components and divertor structure).

Calculations of the nuclear heating demonstrate that all components have a tolerable level, with the exception of the present design for the divertor coils. This is subject to 2×10^{-4}MW/kg from the neutrons and 0.47×10^{-4}MW/kg from gamma rays. It is the gamma radiation that is the biggest concern for the insulator, which is anticipated to be a cyanate ester resin. While tests indicate that this has good radiation properties, future work on the design will need to provide improved shielding for these coils.

Turning to neutron wall loading on test blanket modules, we find that this is highest on the equatorial test blanket modules (see Fig 4), at 1.63 MWm^{-2}, but also quite acceptable on the polar modules, at 1.40MWm^{-2}. At these levels the fluence of 6MW-yrm^{-2} could be achieved within 12 years, assuming an availability of 30%. The tritium burnt would be ~0.9kg per year.

We now return to consider a possible solution for handling the divertor heat load on the outboard target plate. We are exploring the possibility of using a cascade of small (~3mm diameter) spherical SiC pebbles (possibly coated with tungsten). The pebbles would fall under gravity in a continuous toroidal curtain, ~2cm thick, in front of the upper and lower outboard divertor target plates, taking ~75% of the heat load. This leaves a tolerable 10MWm^{-2} to be handled at the fixed target plates behind the curtain. The pebbles are collected below the vessel and cooled. A fraction are removed and replaced to allow tritium recovery and any damaged pebbles to be recovered before being transported back to the top of

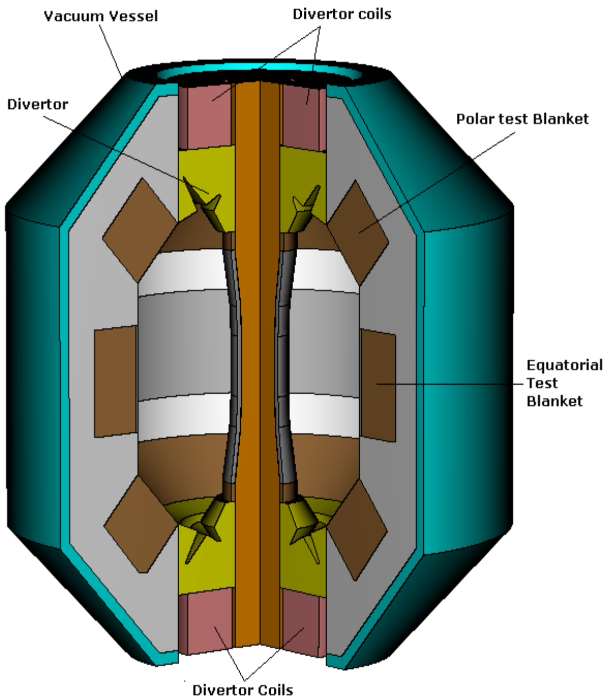

Fig 4: *Geometry of the CTF design used for the neutronics study.*

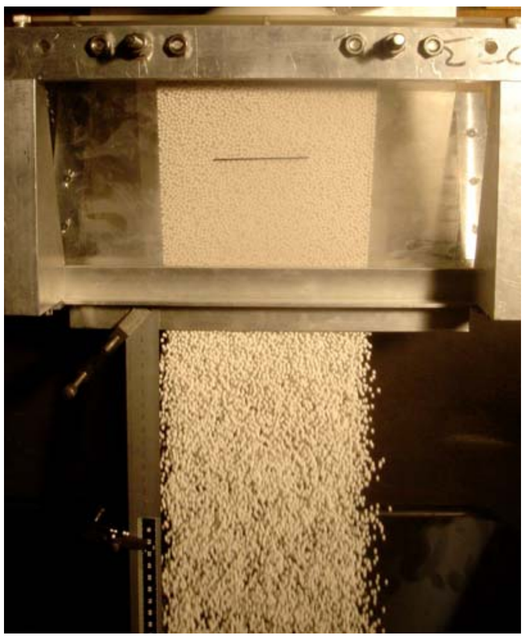

Fig 5: Apparatus testing the cascading curtain of pebbles in the laboratory.

the vessel and the process repeated. Initial tests, exploring the flow of the pebbles in the curtain are encouraging (see Fig 5), and experiments are now being designed for testing on MAST in the future.

5. Discussion

We have considered a range of issues that need to be addressed to improve the scientific and technological basis for a components test facility based on the spherical tokamak. There are challenges, which will require further research, but there are no obvious show-stoppers. Principle challenges include: demonstration of steady state off-axis current drive; design of the divertor; choice of first wall material, and non-inductive start-up (we have not addressed start-up issues here, but we do have possible schemes in mind that will be presented elsewhere). There are also uncertainties associated with the confinement of such plasmas with large momentum input and high fast particle content. However, the plasma parameters that are required are all relatively modest and, while future research should aim to minimise the uncertainties, there is a degree of confidence that an ST could provide a components test facility meeting the requirements of the Abdou report [3].

It is difficult to envisage that a components test facility can be constructed in time to provide data for use in the main design of DEMO (in the context of a fast track route to fusion power). One way to bring the CTF forward is to adopt the following strategy. First, address as many of the key issues as possible by taking full advantage of existing spherical tokamak facilities world-wide. Then construct the CTF as a DD device, anticipating an upgrade to full DT operation once the issues have been fully resolved. This first stage of the device could have modest pulse length and, because CTF only has Q~1, it would be possible to study performance in conditions very similar to the full DT device. Finally, the device would be upgraded to full DT operation and multi-week pulse lengths. With such a strategy, CTF could be a valuable facility, providing a flexible components testing capability in conjunction with DEMO. The availability of a flexible, relatively simple device, operating in parallel with DEMO, would help to reduce technical risk, as well as helping to speed up the final stage to commercial fusion power plants.

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