Spherical tokamaks: Present status and role in the development of fusion power

A.W. Morris *, R.J. Akers, G.F. Counsell, T.C. Hender, B. Lloyd, A. Sykes, G.M. Voss, H.R. Wilson

EURATOM/UKAEA Fusion Association, Culham Science Centre, Abingdon, Oxfordshire OX14 3DB, UK

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Abstract

The spherical tokamak (ST) has triggered a fast-growing activity worldwide on account of its promising potential and its strong physics overlap with conventional tokamaks, including ITER. There has long been a view that it could have a key role as a component test facility, to complement ITER, IFMIF, and DEMO, and there are also interesting possibilities as an option for the fusion power source of an electricity plant. The experimental base is now considerably advanced from the time when these ideas were first raised, with the advent of the MA scale machines MAST and NSTX, and a growing theoretical and modelling base. Here, we describe the status of development on the key engineering and physics issues of the ST, considering in particular application to a component test facility and input to an accelerated programme towards deployed fusion power plants, the so-called “fast track.”

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1. Introduction

One present streamlined approach to fusion power [1,2] is to construct and operate ITER and in parallel construct and operate IFMIF [3] in order to provide information for an early decision on a DEMO. The precise objectives and scope of the DEMO stage are not yet fully defined, but the overall aim is to allow an early decision to construct the first commercial fusion power plant. This is done by demonstrating that the physics, materials, components and systems are viable and reliable. All of this can be achieved with the right programmes in the ITER, IFMIF and DEMO stages. It is, however, possible that the programme could be further accelerated, or performed with reduced risk of delay or technical problems, by incorporating a component test facility (CTF) to test items in between the small materials samples in IFMIF and the full size components of the present DEMO concepts. The experimental basis has advanced to the stage where an ST can be seriously considered for a CTF, since with the appearance of MAST [4,5] and NSTX [6], the ST is now on a par in many respects (plasma size, current and performance)
with existing medium-sized conventional-aspect-ratio tokamaks, and there are now many STs worldwide. The design of ITER was based on these medium-sized tokamaks as well as the large devices JET, JT-60U and TFTR. As will be seen below the size step to a compact CTF is much smaller than the size step to ITER.

The spherical tokamak displays tokamak properties (most of the physics is common with ITER), yet is very compact, given the small aspect ratio \( A = R/a \sim 1.5 \). There are a number of potential advantages, such as low magnetic field energy for a given performance, and potential engineering simplifications due essentially to the much smaller volume of the centre column (which although challenging is not superconducting).

This paper outlines the features of a possible compact ST-based CTF, and uses this to illustrate the physics and technology progress and issues in the spherical tokamak. A brief discussion of the ST as a power plant is included at the end.

2. Component test facility

The reliability and availability of a fusion power plant will depend on the behaviour of complex macroscopic components, such as breeding blanket modules in the combined presence of thermal, electromagnetic and neutron-induced effects. The structural materials properties under irradiation will have been tested in IFMIF, and many of the technology integration issues will have been studied in ITER, as a part of the test-blanket programme or in other areas (such as the divertor). DEMO is presently envisaged as proving full scale assemblies for functionality and reliability prior to commercial deployment. There could be a significant benefit from a smaller facility to test intermediate subassemblies and prototype elements with mixed materials, joints, cooling and breeding elements at high neutron flux and fluence, and also to test plasma facing components. This is the component test facility (CTF).

2.1. CTF design principles, parameters and engineering approach

Our approach has been to attain the required performance \([7]\) by setting quite demanding targets for some aspects and allowing freedom over others. The goal is a device capable of CW operation, generating the required 14 MeV neutron flux of 1–2 MW/m\(^2\) over an area of \( \sim 10 \text{ m}^2 \) or more, yet with modest tritium consumption (\( \sim 1 \text{ kg/year} \)). This has led us to the following design principles:

- Use standard tokamak physics where possible (to reduce extrapolation risk).
- Minimise the size (low capital cost, as well as T consumption).
- Minimise the plasma volume/test area ratio (low T consumption).
- Operate in a strongly driven mode, \( Q = 1 \) (minimise uncertainty when \( \alpha \)-heating appears).
- Adopt a simple design, rapid access (availability, maintainability).

To assist this we have allowed freedom (within reason) on:

- Running costs and power consumption.
- Waste, activation (maintenance needs will constrain, and waste will be minimised).
- The need to breed T (default is no breeding).
- Component lifetime (subject to achieving adequate availability).

The target device is small, to minimise T consumption without breeding (a larger breeding CTF is studied in [9]). For small devices we expect the plasma pressure to be limited by confinement and power exhaust rather than stability. There are several ways to determine the device parameters: e.g. prescribe the neutron flux and deduce the plasma parameters based on confinement scalings, or iterate via other constraints, such as bootstrap fraction, externally driven current. Each of these is based on a number of assumptions, but all point
towards a similar device if low tritium consumption is a key goal.

After an optimisation process [11], including current drive requirements and confinement scaling, we arrived at a device comparable in size to MAST and NSTX, but at substantially higher current and field (Fig. 1, Table 1). Tritium consumption is low at ∼1 kg/year, setting aside tritium retention in the walls (an issue common to all fusion devices). The actual usage and testing rate depends on the device availability (allowing for some unreliability of the test modules) estimated to be about 35% (some issues discussed in [8]).

The main machine would preferably be constructed from a low activation steel, for example EUROFER, noting that in the low field of an ST it may not all be magnetised to saturation. Normal conducting coils are chosen for simplicity and to save on shielding volume. The toroidal field coil is novel in comprising a single central conductor (and without a solenoid to drive the plasma current) with the outer return limbs connected in parallel. This has the key advantage of eliminating insulators in the centre column, and a thin inboard shield gives a reasonable lifetime of the central copper conductor. A rather unusual power supply is needed, ∼10 MA at a few volts. In common with design studies for an ST power plant [12] a high voltage thyristor system with multiple step-down transformers and rectifiers has been suggested.

Test modules are installed on the equator (the region of highest neutron flux) and in polar ports above and below the midplane. The typical size of the front face of an equatorial test module is 0.85 m × 0.7 m (0.6 m²) and about 0.36 m² for the polar modules (matching the suggested lower limit on component size [7]). The version in Fig. 2 has 12 sectors, and allowing for heating and current drive systems and diagnostics there could be as many as 10 equatorial and 16 polar testing stations, giving a total test area of about 12 m². Each module can in principle be independent, with different coolants. No breeding is assumed, although some test modules may well be breeding elements, so could contribute to the tritium cycle.

A full neutronics calculation has been performed with MCNP [13] using the model in Fig. 2 to determine the power density to the test modules (Table 1). Neutrons arrive at the test module from the whole plasma aside from the small solid angle screened by the narrow centre column: this is key to the efficiency and low T consumption of the compact ST.

The focus here is on a neutron source, however, a compact very long pulse/cw ST would also contribute to the testing of plasma facing components in conditions of high heat and particle flux even in the absence of substantial neutron flux, building on linear devices such as the proposed MAGNUM-PSI [14].

2.2. State of the physics basis and issues for a compact non-breeding CTF

There are four main areas to consider: (i) the nominal main plasma parameters, (ii) power exhaust, (iii) heating, fuelling and current drive, (iv) start-up.

2.2.1. Nominal plasma properties

The basic features/design targets are as follows and the device is based on conservative physics and parameters as far as possible:

- H-mode-like confinement scaling, as seen on present STs, noting that L-mode and H-mode scaling are quite close at low A, although not necessarily H-mode profiles and properties.
• Fully non-inductive scenario, based on a mix of bootstrap current (\(\sim 30\%\)), central and off-axis NBCD (although microwave schemes, such as core ECCD, edge EBWCD are possibilities when further developed).

• Double null divertor configuration (which works very well on MAST) to minimise the power to the inboard targets (where there is very limited access for innovative solutions).

• Modest elongation (for an ST): \(\kappa = 2.5\) for broad current profiles, to maximise vertical stability (allowing natural stability in some cases).

• Triangularity—maximised, but limited by constraints on the PF coil positions and the divertor power density, so only modest values (\(\sim 0.3-0.4\)) are expected.

• Modestly peaked density profile (optimising fusion power and off-axis current drive)—although the central value is high (e.g. \(\sim 2 \times 10^{20} \text{ m}^{-3}\)), the normalised density (Greenwald number, \(n_n a^2/I_p\)) is modest, \(<0.2 (10^{20} \text{ m MA})\).

• Plasma pressure in the first stability regime for ideal MHD high-\(n\) modes and stable to low-\(n\) modes without any conducting wall (\(\beta_N = 3.5\)).

• High \(q(0) (\gtrsim 1.5, \text{ ideally } >2)\) to minimise neoclassical tearing modes and consistent with a broad current profile to assist high elongation. The absence of Ohmic current drive will help keep \(q(0)\) high.

• High \(q(a)\) to assist bootstrap current as well as stability.

• \(Z_{eff} \sim 3\) to allow for a radiative region at the edge (from intrinsic or seeded impurities).

• Fusion power gain \(Q \sim 1\), minimising the nonlinearity and control issues from \(\alpha\)-heating.

Aside from \(Q\) and steady state current drive, essentially all the above requirements have been demonstrated separately on existing STs—research is needed.
to combine them in a single plasma. An important feature of the compact ST CTF is that the normalised Larmor radius $\rho^* \sim T^{0.5}/B$ hardly changes between MAST/NSTX and this CTF: the temperature rise is essentially cancelled by increase in field. The collisionality ($\nu^*$) is however much lower in the CTF. While there are indications that confinement improves with reducing collisionality [15], this is a topic for research. Lower collisionality may also have an impact in the scrape-off layer and thus the divertor physics.

2.2.2. Plasma power balance and exhaust

The small size and high power density of the CTF lead, potentially, to very high power fluxes to the divertor targets. Three approaches are under consideration. First, maximise the radiated power from the main plasma whilst maintaining adequate purity (the small volume means that highly radiating species, such as tungsten probably have to be used). Second, use the experience from MAST where a double null configuration leads to the vast majority of the power going to the outboard targets (>95\% in L-mode and during ELMs [16]). Third, develop a flowing divertor target (e.g. a curtain of “pebbles”) to intercept most of the outboard divertor power. Using recent scalings [17], we deduce power decay widths of $\sim 8$ and $\sim 6$ mm for inboard and outboard midplane, respectively. With reasonable flux expansion and modestly angled target plates this results in power density $\sim 10$–15 MW/m$^2$ on the inboard side, which is considered manageable. The situation on the outboard side is, however, likely to be more demand-
Fig. 3. Comparison of the target current profile for a CTF (i.e. that used for the equilibrium reconstruction, dashed curve) with the predicted neutral beam current drive profile (full curve), edge NBCD shown as a second dashed curve.

2.2.3. Heating, fuelling and current drive

The CTF is a fully driven system, so the heating systems have to be chosen to generate both fusion power and current (with the right profile). Neutral beams seem suitable (Fig. 3), and the injection energies are modest for a small device, and accessible with today’s technologies and may allow either positive or negative ions to be used. For central current drive 20 MW of ∼150 keV beams injected on the midplane are proposed. Off-axis current drive is provided by 40 MW D injection at 150 keV, directed downwards from upper ports at ∼40° from the horizontal to decrease the impact of trapping (this angling may not be essential). NBI technology needs further development (e.g. steady state capability), but this issue is common with other fusion devices. The modest energy also leads to significant fuelling, which helps peak the density profile, and reduce the tritium injection (gas fuelling is less efficient). Additional fuelling will be needed and pellets will penetrate these small plasmas. The momentum injection from the beams will lead to strong plasma rotation, which is likely to assist formation of transport barriers, as on MAST [18]. There will be a significant bootstrap fraction, assisted by the high q, high elongation operation, however it is not dominant. This could in fact be an advantage, as it makes the device more robust (the bootstrap current is quite sensitive to changes in the plasma profiles). Preliminary data from MAST suggests that both neutral beam and bootstrap currents behave as expected [19]. Electron cyclotron and electron Bernstein wave (EBW) current drive (core and off-axis/edge respectively) are being investigated as alternatives and for control [20,21].

2.2.4. Start-up

The compact design precludes sufficient shielding for insulators in the centre column, so a solenoid is not included. This means that start-up and current ramp have to be accomplished using the large radius PF coils and the auxiliary heating and current drive systems. This is a key uncertainty, but there are a number of ideas some of which could be combined, many of them being tested on MAST and other STs, already with some success:

(i) Breakdown and initial current rise at a field null from the PF coils, using a conventional single high-order null, or multiple nulls, followed by merging of the resultant seed plasmas. Low order, e.g. quadrupole nulls also appear to work [5,22].

(ii) Preionisation and initial current drive using EBW (reflected waves from the centre column) allows high efficiency current drive as observed on COMPASS-D [23].

(iii) Injection of plasma from a plasma gun (if the insulators can be remote) [24].

(iv) NBCD assisted by bootstrap current using a seed plasma.

(v) Inductive drive during a beta ramp, using flux from the vertical field ramp [25,26], a major contribution in an ST.

Each scheme is specific to the exact set-up on the particular tokamak, and given the uncertainty it may be prudent to have an initial low activation phase of operation of a CTF with a small solenoid whilst reliable techniques are developed.

2.3. Engineering solutions and issues for CTF

The key ST-specific issues are briefly described here. Most of these also apply to the ST power plant, and some are being addressed already in the frame of ITER and DEMO.
2.3.1. Centre rod

The single turn centre rod and the unusual TF supplies have already been mentioned. The TF rod current limit is set by several factors: (a) continuous cooling requires a substantial fraction of the rod to be drilled and filled with coolant (which incidentally reduces neutron damage), (b) tensile stress mainly due to electromagnetic end effects as well as non-uniform radial and hoop stresses from the self pinching current in the rod, (c) expansion and elongation due to damage-induced swelling. The maximum allowable stress is taken to be 1/4 of the ultimate tensile stress (UTS) of the unirradiated material, to give margin for radiation-induced embrittlement. This is the same criterion as adopted by the ARIES team [27], and more conservative than the ASME-VIII limit of 1/3 of the UTS. The present design assumes dispersion-strengthened copper (such as Glidcop™) which also reduces the radiation-induced swelling. The centre rod is protected by a thin steel shield which reduces the damage by a factor of around 5 which is compatible with a lifetime of about 2 full power years (recycling of the material may be an option after some development).

2.3.2. Insulating the divertor coils

The present design calls for two divertor coils with limited shielding (to be close to the plasma). These could be removed and replaced with the centre column, but are still a critical item since they are so closely coupled to the divertor design. It is suggested that the coil be insulated with cyanate-ester (CE) impregnated glass fibre [28] which has improved radiation resistance compared to conventional epoxy-resins [29]. The gamma flux in the CTF design above is, however, higher than the ITER reference level for coil insulation. To address this a substantial reduction has already been achieved by iterative design including neutronics calculations. Tests have also been made of the strength of CE which show it is superior to conventional epoxy-glass insulation, in particular regarding operation at high temperatures (e.g. 70–100 °C). CE has a (non-compressed) shear stress limit of 40–60 MPa at 100 °C compared to conventional epoxy resins which have dropped to ~20 MPa at this temperature [28].

2.3.3. Divertor targets

A combination of high heat flux fixed targets and a “fluid” curtain composed of small pebbles is proposed, as for an ST power plant [12,30]. This has also been considered for conventional tokamaks [31]. The fluid consists of ~3 mm diameter SiC balls cascading through the two outboard divertors under gravity, removing 75% or more of the incident power leading to power density at the fixed targets similar to that at the inboard divertor. Such fluid-like divertor targets are still at an early stage of development (tests are intended on MAST), but are much better suited to the ST than to a conventional tokamak due to the large ratio of poloidal to toroidal field in the ST. This reduces the parallel power density and power asymmetry on individual pebbles. The SiC has many good properties, and erosion and dust production can be reduced by tungsten coating, and there could be other materials. The static targets are presently taken to be tungsten brushes attached to low activation steel plates cooled by gaseous helium. This avoids the tritium retention issues associated with CFCs, at the expense of some uncertainty in the behaviour of W brushes.

3. ST power plant

The primary differences from the CTF are: (a) to be economically attractive a much lower recirculating power fraction is needed (some coils need to be superconducting) and (b) tritium self-sufficiency. This pushes the design towards a rather more uncertain territory with high bootstrap fractions and very high beta, probably requiring control of resistive wall modes. The main issues are summarised and treated in a recent study [32]. It is thus necessary to advance the physics understanding and parameter range of STs more than for a CTF, although CTF results will be used if available (especially CTFs based on more advanced physics [9]. The device in [32] is larger (R=3.4m, a=2.4m) with higher current (~30 MA) but otherwise has a lot in common with the CTF design, e.g. maintenance by removal of the whole central column allowing a simple tower-and-arm remote handling system.

4. Summary and conclusions

Over the last few years, the experience with MA-level spherical tokamaks and the rapid growth of the field (capabilities and number of devices) has com-
firmed their good performance and thus increased interest in their possible exploitation as fusion devices, in addition to a significant input to the physics of ITER-like tokamaks. One such application is for a component test facility to supplement IFMIF and DEMO, in order to test macroscopic composite components, including plasma-facing items, in support of an accelerated development of fusion power. The ST version of a CTF has a number of attractive features, such as the efficient use of tritium due to the low neutron absorption by the slim centre column allowing a large testing area. Later, the ST might be suitable as an option for a fusion power plant. An overall physics and engineering description of a compact non-breeding ST CTF exists based on fairly conservative plasma physics, and most description of a compact non-breeding ST CTF exists. An overall physics and engineering description of the technology would overlap with the CTF, and the physics to an acceptable level, although much could soon be possible to complete the basic design of an ST CTF, especially if some engineering contingency and phased operation were included. Full performance operation will depend on development of some technologies, many of which are needed for any fusion plant. In particular, the CTF described here would benefit from development of a fluid-like divertor target, continuous neutral beam injection and a suitable shield/insulator combination (perhaps based on the promising cyanate ester insulators) for the divertor coils. The ST power plant is based on more ambitious performance, and it would take longer to develop the physics to an acceptable level, although much of the technology would overlap with the CTF, and conceptual outline physics and engineering designs exist.

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