FUSION TECHNOLOGY APPLICATIONS OF THE SPHERICAL TOKAMAK

D C Robinson, R Akers, S J Allfrey, A Bond[1], T C Hender, P J Knight, O J Kwon[2],K G McClements, G P Maddison, S Medvedev[3], J Mustoe[4], M R O'Brien, C M Roach, S E Sharapov[5], D V Sherwood[4], G M Voss and H R Wilson
EURATOM/UKAEA Fusion Association, Culham Science Centre, Abingdon, Oxon, OX14 3DB, UK

Abstract
Fusion technology applications of the spherical tokamak are presented, exploiting its high β capability, normal conducting TF coils, compact core, high natural elongation, disruption resilience and low capital cost. We concentrate here on two particular applications: a volume neutron source (VNS) for component testing and a power plant, addressing engineering and physics issues for steady state operation. The prospect of nearer term burning plasma ST devices are discussed in the conclusions.

1. VOLUME NEUTRON SOURCE

The high-β, compact nature of the spherical tokamak (ST) makes it an attractive volume neutron source for materials testing, for example. A range of candidate ST devices with aspect ratio of ~1.6, plasma elongation of 2.3, ratio of plasma current to toroidal field (TF) rod current I_p/I_{rod}~1 and constant rod current density of $50MAm^{-2}$ have been considered; examples are shown in Table 1. The main physics parameters have all been demonstrated on START (eg the record β=40% is more than twice that required; up to ~100% is possible theoretically[1]). The plasma is initiated by an induction/compression scheme (as on START) because there is no solenoid; this is followed by a current ramp using NBI. The main poloidal field coils are the vertical field and divertor coils, having Amp-turns of 60% and 50% of the plasma current, respectively. The central TF conductor is single turn and fabricated from dispersion strengthened copper, which resists radiation swelling, and there are 12 TF return limbs. Due to the large relative radius of these coils the peak TF ripple in the compressed plasma is negligible (<0.06%). The TF column is water cooled, with steel liners inserted into the cooling channels to prevent leaks. Sliding joints are used between the centre column and the TF return limbs to reduce the axial loads. Detailed thermohydraulic calculations show 29MW of resistive power is dissipated in the centre column and the peak temperature is relatively low ~130°C, which helps to improve resistance to radiation swelling.

In order to calculate the fluence an assessment of availability has been made. The main scheduled maintenance items are based on a conservative assumption of annual replacement of neutron damaged items (eg centre post and divertor targets) giving a total estimated scheduled annual maintenance time of 0.19 years. The total availability arising from unscheduled events is calculated to be 0.54 (using information from a NET study). This gives an overall availability of 0.44, which results in an acceptable fluence of $6.6MWam^{-2}$ over 10 years and a modest annual tritium consumption of ~0.9kg.

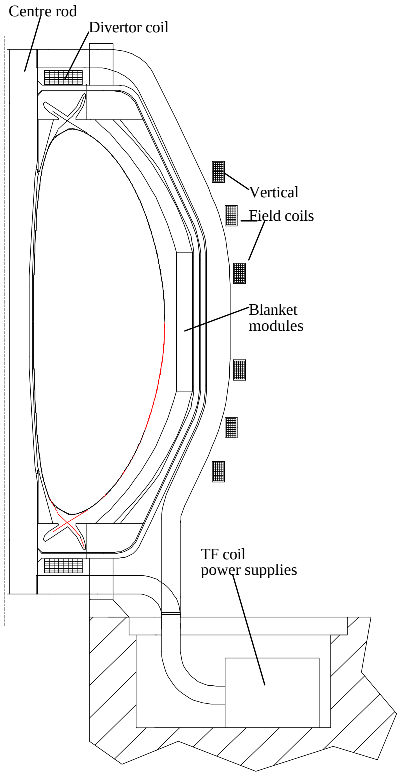
2. POWER PLANT DESIGN

A 1GWe steady state spherical tokamak power plant design is being developed based on conservative physics assumptions. The layout of one half of the machine cross section, which is not yet fully optimised, is shown in Fig 1, together with some of the system parameters in the adjacent table. This section describes how the physics and engineering designs are being integrated.

Major Radius (m)	0.7	0.8
Plasma Current(MA)	10.3	13.4
Normalised Pressure, β_n	2.6	1.9
Current drive power (MW)	46	69
Required energy confinement time (s)	0.13	0.12
IPB98(y,1) scaling law confinement time (s)	0.11	0.14

Table 1. Examples of neutron sources with a wall load of $1.5MWm^{-2}$

[1] Reaction Engines Ltd; [2]Taegu University, Republic of Korea; [3]Keldysh Institute, Moscow, Russia; [4]NNC Limited, Knutsford, UK; [5]JET Joint Undertaking, Abingdon, Oxon, OX14 3EA, UK



System parameters

Plasma major radius	3.42m
Plasma aspect ratio	1.4
Plasma elongation	3.0
Plasma triangularity	0.45
Plasma current	31.0 MA
TF Current	31.0 MA
β_n	8.2
Vol avge density	$1.1\times10^{20}m^{-3}$
Vol avge temperature	19.2keV
Centre safety factor	3.1
Pressure-driven current	88%
NBI current-drive	3.7MA
NBI system	42keV, 12MW
	340keV, 29MW
	500keV, 33MW
Number of return limbs	16
Fusion power	3GW
Net output power	1GW
Gross thermal efficiency	41.5%
Average wall load	$3.7MWm^{-2}$

Figure 1 Section through ST load assembly

Physics design: The steady state capability relies on a large fraction of the plasma current being provided by the 'bootstrap' current (including diamagnetic current). The high bootstrap current fraction (~88%) is obtained by exploiting the high normalised pressure (β_n=8.2) and high elongation (κ=3) capability of the ST ($\beta_n \approx 6$ has already been achieved on START). Calculations demonstrate that this design is stable to both $n=\infty$ ballooning modes (accessing the second stable regime) and n=1,2 modes with a conducting wall positioned at 1.25a, where a is the plasma minor radius. The physics of the resistive wall mode remains uncertain. However, both theory and experiment suggest the mode is stabilised by plasma flow, as would be provided by the substantial neutral beam power allowed for in this design. The good curvature of the ST is expected to help stabilise neoclassical tearing modes; the risk is further reduced by maintaining $q_0>3$, which eliminates the (most dangerous) low order rational surfaces and sawtooth/fishbone trigger mechanisms.

The centre rod current is equal to the plasma current ($I_{rod} \leq I_p$ has already been demonstrated on START), to reduce the dissipation to an acceptable level (see below). The edge safety factor, and therefore bootstrap current fraction, is controlled by adjusting the elongation, κ, and at tight aspect ratio this can be high; for the class of equilibria considered here elongation $\kappa \approx 2.6$ is naturally stable without the need for vertical feedback stabilisation. Here we have exploited feedback stabilisation to achieve κ=3, providing a bootstrap current fraction of 88%. Should control of the plasma be lost, results from START for similar equilibria indicate that the consequences of a vertical displacement event would be much less severe than at conventional aspect ratio: halo currents on the centre column were less than ~3% of the plasma current.

The steady state current drive requirements can be provided by a neutral beam injection (NBI) system. Calculations show that a system of three beam-lines can provide the required current profile: a 42keV, 12MW system for the edge current drive, a 340keV, 29MW system for the mid-radius current drive, and a 500keV, 33MW system for the core current drive. The density at which the power plant is operated ($n_e = 1.1 \times 10^{20}m^{-3}$, $n_e/n_{Greenwald}$=0.7) represents a trade-off between current drive efficiency and the fusion reaction cross-section. The challenging current drive is the central current density, which needs to be relatively peaked to 'fill in' the missing bootstrap current there: a broader central current drive leads to a non-monotonic q profile close to the magnetic axis, which proves to be unstable to an internal n=1 mode (ie unaffected by wall position). Note that we have adopted the conservative view that no bootstrap current is driven on axis: if recent calculations of on-axis bootstrap current due to 'potato' orbits can be confirmed then this would reduce the neutral beam power required. We are also investigating the use of RF systems (eg ECRH and fast wave) to provide the central current drive: this would avoid the need for the negative ion high energy NBI systems.

The confinement scaling laws derived from the conventional aspect ratio tokamak database provide a range of predictions for the ST power plant design: $H_{ITER97ELMy}$=1.6, $H_{ITER93H}$=1.82, and $H_{IPB98(y,1)}$ =1.24 (*H* is the factor by which the confinement must be improved). For comparison, START data gives $H_{ITER97ELMy}$=1.6, $H_{ITER93H}$=1.9 and $H_{IPB98(y,1)}$=1.2. An important role for the next generation of STs (eg MAST and NSTX) will be to refine confinement scaling laws at tight aspect ratio.

Calculations of direct orbit loss of fusion α-particles show it to be small (less than ~2%) primarily because the banana orbits of trapped fast ions are squeezed on the outboard midplane due to the non-monotonic dependence of the magnetic field on major radius. The TF ripple is negligible, less than ~1%. There is limited information from modelling fast particle instabilities in STs as yet, but there is much experimental evidence from START of instabilities driven by the fast ions from the NBI. Nevertheless there is no evidence to suggest that these modes have a detrimental effect on the confinement of the fast particles. Significantly, there were no Alfvénic instabilities observed in the highest β START discharges, possibly as a result of strong Landau damping, which provides further optimism that the α-particle confinement will be good in high β power plants.

Engineering design: The low aspect ratio and high TF coil current combined with the effects of neutron irradiation on the structures lead to compromises in the centre column dimensions and choice of materials. The baseline design of the centre column consists of a dispersion strengthened copper conductor material which was selected based on its resistance to swelling and ability to maintain its electrical and mechanical properties under irradiation. The conductor is flared at both ends and water cooled with a 15% cooling fraction at its mid-plane and an inlet water temperature of 30°C in order to minimise the resistive power losses. The copper conductor is partially protected by a water cooled martensitic steel inboard shield. This shield operates at a higher temperature than the copper conductor and removes a significant fraction of the nuclear heating which would otherwise be absorbed by the centre conductor, raising its operating temperature and hence its electrical power dissipation. The thermal gradients and hence thermal stresses produced in the embrittled outer regions of the conductor are also reduced by the shield. In addition, this shield reduces the nuclear damage to the conductor, which would raise the resistance due to the production of nickel, zinc and cobalt transmutants from the copper base.

The electro-magnetic loads induced on the water-cooled copper return limbs are reacted by the main vessel which has steel webs welded to its external and internal surfaces. Strain isolation between the limbs and the vessel and between the limbs and the centre conductor, is achieved by using sliding electrical joints. These sliding joints are based on the use of Feltmetal™ which has been developed for a similar application for MAST [2] and ALCATOR C-MOD. These joints prevent axial loads from being transmitted to the centre rod from the TF limbs and allow axial and radial relative movements as well as in-plane rotation between the sections of the TF coils. The Feltmetal material, about 1mm in thickness, has a large number of contact points on one face and can be soldered or brazed to copper conductors on the other face. A tongue and groove geometry is adopted in which the contact pads are brazed to the outer surfaces of two tongues at the end of the TF limbs. These tongues locate into grooves in the centre rod and operate at an average current density of less than 1 kA/cm^2. Steel spring plates are inserted axially to provide the required contact pressure and are easily removed and replaced remotely during maintenance.

The design of the TF coils has considerable influence on the design and performance of the entire plant, due mainly to its electrical power requirements which are dominated by the centre rod conductor. Since the electrical conductivity of metals increases with falling temperature, this offers a route for reducing the resistive power dissipation in the coils. However the coils now need refrigerating which also consumes power, adding to the total dissipation. A centre column and TF coil design based on cryogenic temperature operation has been developed as an alternative system. This uses pure aluminium as the conductor and offers advantages in terms of power consumption, activated waste and long lifetime (~ plant life), but requires a thick water cooled tungsten carbide shield and more complex structural supports.

The divertor consists of a deep outboard leg since a large proportion of the power is deposited here (data from START indicate outboard power flow is ~ 6 times the inboard flow). This deep slot forms

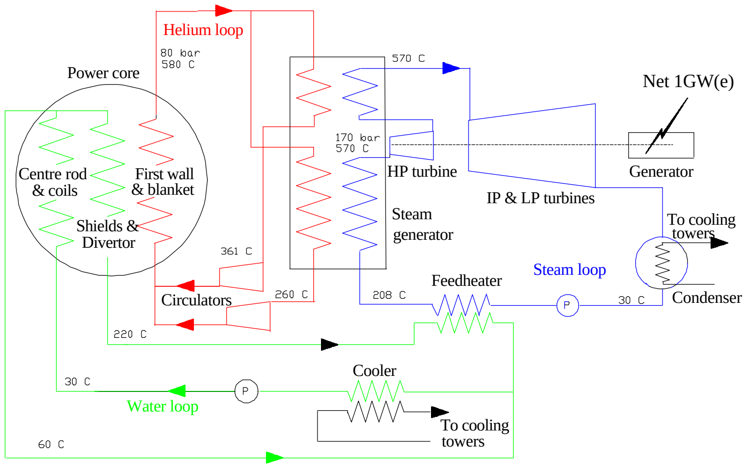


Figure 2 ST Power Cycles

a closed unattached gas target with sufficient surface area to reduce the heat flux to acceptable levels. The inboard leg is much shorter to avoid intercepting the centre column and alternative systems are being investigated to reduce the heat flux in this attached divertor slot to acceptable values e.g. applying an AC field to sweep the strike point over the surfaces. Alternative approaches, dispersing the power using ~100% radiating edge plasmas seeded with impurities, are also being studied.

The divertor coils and structures are integrated into the centre column assembly which can be quickly removed from the machine by lowering it vertically downwards into a basement for refurbishment. The 48 breeder blanket modules can be removed internally and lowered through the hole left by the removal of the centre column into the basement before new blanket modules are lowered through the top hole and fitted into position from within the plasma chamber. A complete new centre column assembly is then lifted into position from the basement. This system provides a simple and effective maintenance strategy, allowing on-site refurbishment and re-use of the activated structures. Separate routes for new components and activated materials minimises the potential for contamination.

The power conversion cycles are shown schematically in Fig 2. The steam cycle conditions are based on AGR power stations and gives a gross thermal efficiency of 41.5%. The cycle has been modified to utilise the low grade heat produced by the water cooling systems that cool the centre conductor, TF limbs, inboard shield and divertor to pre-heat the feedwater to over 200°C. The steam is re-heated back to 570°C after the high pressure turbine which maintains the dryness at the low pressure turbine exit at about 93%, which is necessary to limit blade erosion.

3. CONCLUSION

The recent progress in understanding ST physics and technology through the pioneering experiments on START, inventive engineering and theoretical modelling, provide a persuasive argument for taking the ST concept forward, not only to improve our understanding of the tokamak, but to develop it both for burning plasmas and technological applications, such as a VNS and a power plant. Recent calculations suggest that the physics of burning plasmas could be studied in a modest-sized ST device; 1-D transport modelling indicates an attractive device with: R=1.32m, a=0.94m, κ=2.6, δ=0.4, B_{total}=4T, I_ρ=14.5MA, $H_{IPB98(y,1)}$=1.2 so that high Q or even ignition can be achieved with 40MW of auxiliary heating. Furthermore, H-mode power threshold scalings suggest that such a device would easily access H-mode. There remain key issues to address with the next generation of MA-level STs, MAST and NSTX, supported by theoretical modelling and engineering studies: an extended ST confinement data-base, confirmation of the steady state potential, energetic particle physics, possible halo currents in large devices, centre post engineering, start-up and methods of dealing with exhaust.

Acknowledgement This work is funded by UK Department of Trade and Industry and EURATOM.

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