Compact Fusion Energy based on the Spherical Tokamak

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Abstract. Tokamak Energy Ltd, UK, is developing spherical tokamaks using high temperature superconductor magnets as a possible route to fusion power using relatively small devices. We present an overview of the development programme including details of the enabling technologies, the key modelling methods and results, and the remaining challenges on the path to compact fusion.

1. Introduction

Since the mid 1980s the spherical tokamak (ST) has been recognized as an important device for fusion research [1,2,3]. Such devices demonstrate all the main features of high aspect ratio tokamaks but are relatively small and inexpensive to construct. Moreover, research has shown that they have beneficial properties such as operation at high beta, lower H mode power threshold, reduced tendency to disrupt, and possibly higher confinement. Early attempts to design reactors based on STs, however, did not produce convincing designs, and until recently STs have been mainly seen as useful research devices and possibly as neutron sources for component testing. However, recent advances in both tokamak physics and superconductor technology have changed the situation, and relatively small STs operating at high fusion gain may now be possible. The key physics step is the realization that the power needed for high fusion gain may be considerably less than previous estimates, while the key technological step is the advent of ReBCO high temperature superconductors (HTS). In addition to operating at relatively high temperatures, HTS can also produce and withstand relatively high magnetic fields: both of these properties are beneficial in the design of magnets for fusion devices especially for STs where space is limited in the central column. Utilising these developments, Tokamak Energy (TE) is developing an alternative route to fusion power based on spherical tokamaks constructed using HTS magnets, and modelling and concept work is underway to determine the optimum power and size of an ST/HTS fusion module. This work identifies key aspects in the physics and technology that significantly affect the size, power and feasibility of such a module. In parallel, experimental work is underway addressing these aspects, including the construction and operation of a series of STs. In this paper, we present recent new results and the status of the development programme, and we outline the intended next steps.

The paper is divided into five main sections. In section 2 we summarise briefly our earlier modelling work that indicates that there is potentially a solution for a high fusion performance device at relatively small major radius and low aspect ratio. In section 3, we give an overview of the TE development programme; we include a brief description of the STs presently under construction and planned at TE. Our predictions of the performance of a candidate ST fusion module are extended and updated in section 4. Possibilities for modular fusion are discussed briefly in section 5. A summary is given in section 6.

2. Power and Size of Tokamak Pilot Plants and Reactors

Recent modelling with a system code based on an established physics model has shown that, when operated at reasonable fractions of the density and beta limits, tokamak pilot plants and reactors have a power gain, Q_{fus}, that is only weakly dependent on size; mainly it depends on P_{fus}, and H, where P_{fus} is the fusion power and H is the confinement enhancement factor relative to empirical scalings [4]. Frequently the ITER reference scaling (IPB98y2) is used and H is defined relative to that. When expressed in dimensionless variables this scaling has a significant inverse dependence on the plasma beta, ($\beta^{-0.9}$). However, dedicated experiments on several devices in which the dependence of the confinement time on beta has been probed directly, have shown that the confinement time is almost independent of beta; alternative beta-independent scalings have been developed, for example that by Petty [5]. These scalings are arguably more appropriate because they give consistency between the single device and multi-device experiments. Modelling with the system code has shown that the power needed for a given fusion gain is a factor of two to four lower with these scalings [4].

The dependence on P_{fus} implies that it is principally engineering and technological aspects, such as wall and divertor loads, rather than physics considerations, that determine the minimum device size. The lower power requirement arising from the beta-independent scalings is especially advantageous. Using the system code, a wide parameter scan was undertaken to establish possible regions of parameter space that could potentially offer high Q_{fus} with acceptable engineering parameters. In addition to the high aspect ratio, large tokamak solution, a region of parameter space at low aspect ratio and relatively small major radius, and hence small plasma volume, was identified (Figure 1). A candidate device

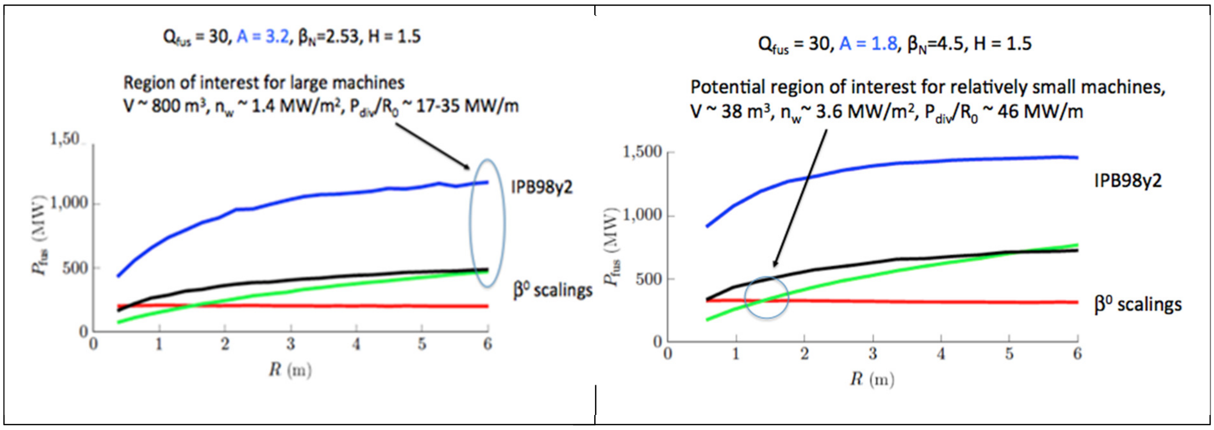

FIG. 1. P_{fus} as a function of R_0 at constant Q_{fus} = 30, H = 1.5 for both IPB98y2 scaling and beta-independent scalings for A = 3.2 and A = 1.8. Here P_{div} is the transported loss power that has to be handled in the divertor after allowance for radiation losses. Details given in [4]. The conventional large tokamak solution (left) and the potential low A solution (right) are indicated.

(ST135) with a major radius (R_0) of 1.35 m, aspect ratio (A) of 1.8 and magnetic field on axis (B_T) of 3.7 T operating at 185 MW with a Q_{fus} of 5 was suggested. The study that led to this proposal was mainly a physics study; engineering aspects were not investigated. Some important engineering and technological aspects are currently being developed and key results are presented below (Section 4).

3. Tokamak Energy Experimental Development Programme

3.1. HTS

Use of conventional low temperature superconductor (LTS) for an ST fusion device appears impractical because thick shielding (≥ 1 m) would be needed to prevent neutrons heating the superconductor above 4K. With shielding of this thickness on the inner central column, the device would be very large. The advent of HTS, however, appears to provide a solution. High temperature superconductors were discovered in the late 1980's and the 2nd generation ReBCO (where Re=Yttrium or Gadolinium) tapes have very promising properties; in particular, they are able to carry high currents under very high magnetic fields. Although superconductivity occurs at around 91K in zero magnetic field, far better performance is achieved when cooled to around 20 - 40 K. Thus, for constructing tokamaks, HTS has potentially two advantages relative to LTS: an ability to carry more current at high field, and less demanding cryogenics [6].

3.2. ST25(HTS)

In order to gain experience with constructing tokamaks using magnets made from HTS, Tokamak Energy constructed a small but complete tokamak. This is the world's first demonstration of a tokamak magnet where all the magnets are made from HTS. All coils (toroidal and poloidal) are wound from YBCO HTS tape. The 6 limb cryostat is cooled 'cryo-free' to ~20 K using a single Sumitomo cold head seen above the vessel, thermal conduction from the HTS tape being provided by copper strips; the 2 PF coils being cooled by He gas to 20 - 50 K. A 29 hour run was obtained in June 2015, with a hydrogen plasma (Figure 2). The TF magnet in ST25(HTS) uses a continuous length of YBCO tape and, to save cost, is of low field (~ 0.1 T), chosen to permit current drive via 2.45 GHz microwave sources. This simple design is prone to single point failure at any of the several soldered joints, and also has a high inductance, which makes quench protection difficult. A high performance fusion ST will need a toroidal field of 3 – 4 T, which requires the development of HTS cables, and TE is currently developing HTS cable and joint technologies required to build and operate a larger device that will operate at higher field (section 3.4). A major challenge is the design of the centre column, and this is being addressed in the design work for ST135 (below).

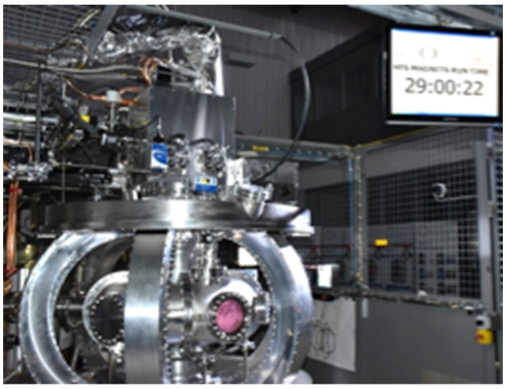

FIG.2. Demonstration of 29hr pulse in ST25(HTS) in June 2015

3.3. ST40

To date STs have operated at 1 T or below. For high fusion performance, devices operating at 3 T or above will be needed. In order to construct an ST that can operate at fields at this level innovative engineering solutions will be needed especially for the central column. To develop and demonstrate solutions to the key engineering aspects, TE is constructing a device (ST40) with copper magnets that is intended to operate at fields up to 3 T. Beyond this device TE is planning high field STs using HTS.

ST40 (Figure 3), will have a design field of B_{T0} = 3 T at major radius of R_0 = 40 cm. Use of copper for the TF coil (as in all existing STs, except ST25HTS at TE) has the advantages of combining structural strength with good conductivity (especially when cooled to liquid nitrogen temperature). Whereas existing STs have operated typically at 0.3 - 0.5 T, with the recent MAST, Globus-M and NSTX upgrades striving for 1 T, special design features are employed to enable ST40 to operate at up to 3 T. Principal amongst these is the use of Constant Tension Curve TF limbs, specially designed so that over the permitted temperature rise (whether starting from ambient or from liquid nitrogen temperature) the expansions of the centre post and the return limbs are matched, so that minimal movement occurs at the critical top and bottom joints, a simple robust flexi-joint being provided to accommodate the movement. At fields of 3 T, stresses are high; and an external support structure based on two steel rings (shown in grey above and below the magnet) accommodates in-plane and out of plane forces, such as those arising from tolerance errors in the radial position, and the JxB twists arising from TF-PF and TF-solenoid interactions.

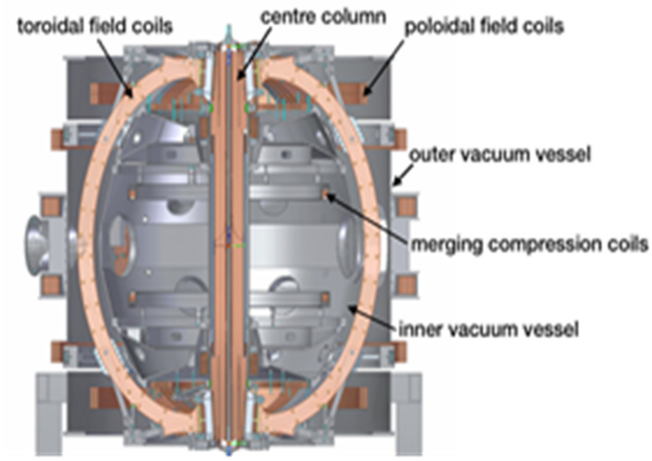

FIG. 3 Engineering drawing of ST40

An important aid to obtaining such a high field is the use of a minimal solenoid. The merging compression (MC) coils (indicated in Figure 3) operated successfully in START and MAST, and in ST40 are capable of generating a hot ST plasma of up to 2 MA without use of the central solenoid, which is thus only needed to maintain the flat-top current – assisted by the high bootstrap fractions expected, and current drive from NBI or RF. Hence, the solenoid is considerably smaller than in MAST and NSTX and their upgrades. This reduces JxB twisting stresses, takes less copper from the TF column thus reducing TF resistance and TF heating, and provides a stronger TF post.

The centre post is constructed from 24 wedges, each twisted by 15 degrees over their length thus obviating the need for a TF compensating coil. Except for the MC coils (which require a fast HV bank), the TF, solenoid and PF coils are powered by 'Supercapacitors' such as the Maxwell 125 V, 63 F, 0.5 MJ transport module, providing a very economic power supply from laboratory power supplies. Each unit has a limiting fault current ~ 7 kA even under dead short conditions providing safety; an important consideration in a 100 MJ capacitor bank.

In addition to the original objective of providing a high vacuum version of the pioneering START ST at a tenfold increase in TF, the specification has been extended: indeed it is predicted that the MC scheme will provide 1 – 10 keV plasmas in ST40. Details of its predicted performance are provided in a parallel paper at this conference [7]. The device is currently under construction and is expected to begin operation in early 2017.

3.4. Beyond ST40 (ST60)

As mentioned above, the intention is to combine the experience gained with the low field HTS device ST25(HTS) with that obtained with the high field copper device ST40 to design and construct a high field ST using HTS for the magnets. Currently the major radius being considered is 60 cm (ST60). The device will be used to develop physics understanding of a high field ST and to test HTS cable technology. It will use a design of the HTS core similar to that outlined in section 4, which is under study for the fusion power module (ST135). A neutron shield on the central column will probably not be included although the possibility of operation with DD or DT is not excluded. For physics tests without neutrons, the device could be operated in steady state using external current drive; in DD and DT operation, neutrons will heat the HTS core. To avoid large cryogenic cooling costs, adiabatic short-pulse operation is considered, whereby the HTS column is permitted to heat by 10 K before the discharge is terminated. Initial estimates indicate that this will give minutes of operation under DD operation; seconds if DT were to be used.

4. Conceptual Design of a Prototype Fusion Power Module: ST135

While from a physics perspective it seems that a compact fusion module may be possible (Section 2), the feasibility of such a device depends critically on there being satisfactory engineering solutions in a few critical areas. Three important components are the central column where it is necessary to handle the stress in this component at the same time as accommodating the HTS TF magnet; the divertor where it is necessary to handle high power loads; the inboard shielding that is needed to protect the HTS tape from bombardment from high energy neutrons so that it has an acceptable lifetime, and also to reduce the neutron heating to a level that can be handled with a reasonable cryogenic system. Possible solutions for these components are under study and development within TE.

4.1. HTS Central Column Design

One possible arrangement (Figure 4) utilises several significant features of HTS tape: namely, operation at 20 – 30 K that gives sufficient current carrying capability at high magnetic field, but at much lower cryogenic cooling cost than operation at 4K, and the property that tape aligned parallel to the local magnetic field can carry typically a factor 4 – 5 times more current than non-aligned tape. In this simple model the individual HTS tapes are bonded into multi-layer cables, and for the initial calculations we assume that the entire structure has the strength of half-hard copper. This layout is chosen to provide a uniform current density. Towards the geometric centre of the column, the magnetic field reduces and in consequence the current carrying capacity of the HTS tape increases. This makes it possible to reduce the number of tapes. The peak crushing pressure (σ_{cc}) occurs at the column axis. For this simplified model

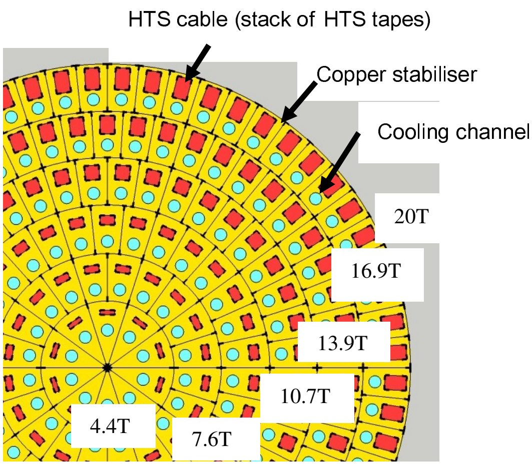

FIG. 4. Example of a monolithic HTS centrepost.

$$\sigma_{cc} = 0.8B_{T0}^2(R_0/R_{cc})^2 \text{ MPa} \quad \ldots\ldots(1)$$

where B_{T0} is in T, and major radius R_0 and the HTS magnet radius R_{cc} are in m. For the reference ST135 design, $R_0 = 1.35$ m, $R_{cc} = 0.25$ m, $B_{T0} = 3.7$ T, A = 1.8, and so the peak field at the edge of the HTS magnet is 20 T and the centre column current is 25 MA. The calculated peak radial stress is 320 MPa. This is high, but is in the form of uniform hydrostatic compression. There will undoubtedly be stress raisers in a real structure, but there is a high degree of symmetry and more detailed FEA calculations indicate that these can be accommodated.

Expression (1) shows that forces increase as the square of the toroidal field, but reduce as the square of the central column radius. Hence for example, a 0.05 m addition (20%) to the HTS core radius (accompanied by a 0.05 m decrease in shield thickness, if it is desired to maintain the aspect ratio of 1.8, and a reduction to 16.7 T at R_{cc}), reduces the peak stress to 205 MPa whilst maintaining a field of 3.7 T at R = 1.35 m.

Note that whereas it is conventional to twist superconductor cables to minimize AC losses these will not be significant in the TF magnet of an ST power plant as this will have a slow rise to reach a constant peak current; so with a suitable design, use can be made of the substantial increase in performance afforded by aligned operation, giving a substantial reduction in cost.

4.2. Divertor Loads

High power plasmas in relatively small devices would impose very large divertor loads if operated in single-null (SN) configuration, especially in an ST where the inner strike point is at low radius and space to mitigate the power load by angled strike points or long divertor legs is limited. However the use of Double-Null Divertor (DND) operation, as studied extensively on the START and MAST STs [8], can considerably improve the loading, as the DND configuration is very favourable for the ST concept. Firstly, the inner SOL is now (largely) isolated from the outer, and it is found that most SOL power escapes through the outer segment and so hits the outer strike points; the ratio varies widely, dependent on plasma conditions. During ELMS the ratio can be over 20 times higher; during inter-ELM periods when the core heating is partially retained, the ratio can fall to 4, approximately the ratio of the inner and outer SOL areas; but the average ratio is typically taken as 10 in MAST [9].

An extensive study of divertor loads in the FNSF design [3] which is similar to ST135 estimates peak divertor loads for both the inner and outer DND strikepoints to be within 10 MW/m^2, and it is instructive to compare ST135, designed to have P_{fus} = 200 MW and Q_{fus} = 5, with the FNSF design, where P_{fus} = 160M W and Q_{fus} = 2. The load on the outer divertor plates is P_{sol}/strike area, where the strike area is proportional to R_{trg} x f_x x λ_q where R_{trg} is the radius of the strike point, f_x the flux expansion from the midplane to the target, and λ_q represents the width of the SOL at the midplane as given by Eich scaling [10]. The heat entering the plasma is the combination of alpha heating and auxiliary heating making a total of 112 MW in FNSF and 80 MW in ST135, and after radiation losses due to impurity radiation, bremsstrahlung and cyclotron this will enter the SOL. The strike points R_{trg} are at about 20% larger radius in FNSF; expansion f_x should be very similar; and Eich scaling predicts λ_q varies as $B_{pol}^{-1.2}$ and B_{pol} is a factor 1.3 higher in FNSF and so λ_q is a factor 1.3 larger in FNSF. Overall, we conclude that strike areas should be very similar; and since P_{sol} is approximately 30% less, peak power on each outer divertor in ST135 should be ~ 7 MW/m^2.

4.3. Inner Shield: Heat Deposition to Central Core and Optimisation of Shield Thickness

An extensive investigation of candidate materials for the inner shield has been carried out and tungsten carbide with water cooling has been identified as a promising configuration [11]. MCNP calculations of the attenuation due to this shield have been carried out. The attenuation of the neutron flux, and associated heat deposition in the central core, as a function of shield thickness have been parameterised and included in the TE System Code [11]. The heat deposited will have to be removed actively with a cryoplant and an estimate of the power requirement is also included.

In order to determine the optimum shield thickness several factors have to be taken into account simultaneously. For a device of given Q_{fus}, H factor and aspect ratio A, it is necessary to consider each of the attenuation due to the shield, the magnetic field on the HTS tape and the radial stress in the central column. The TE system code has been extended so that these different aspects can be taken into account simultaneously.

As an example, for a reference plasma (Q_{fus} = 5, P_{fus} = 201 MW, H(IPB98y2) = 1.88, A = 1.8, β_N = 4.5), we present in Figure 5 the variation of key parameters with major radius. The extra space in the radial build as the major radius increases is used to increase both the thickness of

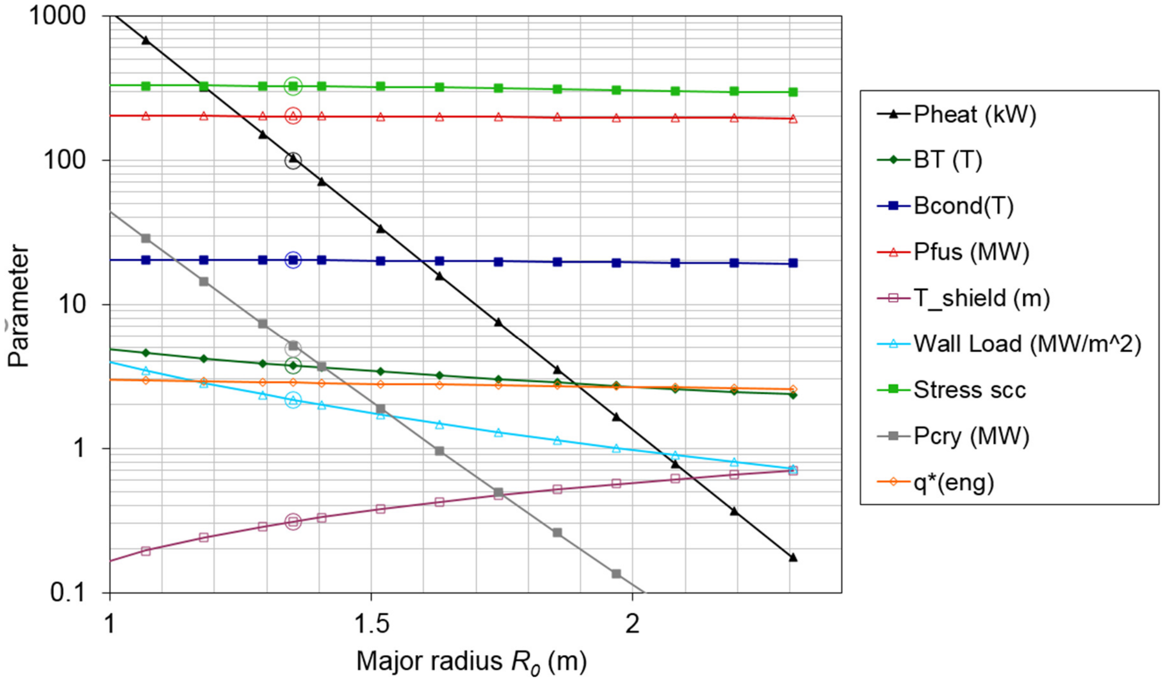

FIG. 5. Heating power deposited in the superconducting core, and other key parameters, as a function of R_0. The scan has been performed with the wall loading adjusted to give a constant HIPB98y2= 1.9, the central temperature adjusted to give 0.8 of the Greenwald density limit, and the toroidal field adjusted to give 0.9 of the beta limit. The circles show the reference design at 1.35 m major radius. Note that Qfus, Pfus and H all remain constant as R increases.

the shield and of the HTS core in the ratio: 92% to shield, 8% to the HTS core radius R_{cc} which approximately maintains constant stress. At the reference major radius for ST135 (R_0 = 1.35 m), the HTS core has radius 0.25m, gaps total 0.05m, the shield thickness is 0.3 m, the field on the conductor is 20 T, the neutron heating to the central column is 100 kW, and the wall load is 2 MW/m^2. To handle this level of neutron heating we estimate that a cryogenic plant of 5 MW wallplug power would be needed. It is clear from the figure that as the shield thickness increases the heating of the central column reduces rapidly. Using expression (1), the stress in the central column is 320 MPa. As discussed in section 4.1, if the scan was repeated with a 5 cm exchange increasing R_{cc} at the expense of shield thickness, the stress would be reduced to 205 MPa but P_{heat} and P_{cry} would increase. From the comparison with FNSF (section 4.2) the peak divertor load would be ~ 7 MW/m^2. For all values of R_0, q*eng (defined as $5(1+\kappa^2)/2\ a^2B/(RI)$) is ≥ 2.8, the value recommended for avoidance of disruptions [3].

A key aspect not yet included is any change in tape performance due to irradiation by neutrons. Inevitably the performance will degrade but information on the extent of the degradation is limited. Dedicated R&D is needed in this area.

5. A Modular Power Plant

If a relatively small fusion module is feasible, then a possible alternative supply of fusion power based on a modular concept may be available. The economics and operational advantages of this concept, utilizing perhaps 11 small 100 MW units, (10 working and 1 undergoing maintenance) have been outlined [12]. The advantages include improved availability; cyclic maintenance; the need for only a relatively small hot cell; a sharing of start-up facilities and energy conversion; the possibility of pulsed operation to provide loads variable in time and the economics of mass-production. STs can exhibit the combination of high bootstrap fraction and high beta - important both for maximizing power gain and in obtaining/maintaining the plasma current, especially in the absence of a central solenoid. In this latter respect, recent predictions that RF techniques can provide full plasma current initiation and ramp-up [13] are encouraging; it is expected that these techniques will perform well in a high field (3 - 4 T) ST as then the densities can be sufficiently high.

6. Summary

The Tokamak Energy programme aimed at developing the spherical tokamak as a future power source is outlined. Areas that have a high leverage on the feasibility of this approach have been identified and are under study in current R&D. Two such areas are the energy confinement scaling at high field (3 – 4 T), and the impact of fusion neutron irradiation on the properties of HTS rare earth tape at 20 - 30 K. Both are under investigation and the data should be available in the near future. Favourable results could lead to economic fusion based on modular high gain STs of small size; less favourable results could lead to larger but still economic ST fusion power plants of around 1.5 - 2 m major radius. In either case, the small scale of the fusion modules should lead to rapid development and make possible the resolution of the remaining key outstanding physics and technology steps that are needed for the realisation of fusion power.

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