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Towards a compact spherical tokamak fusion pilot plant

A. E. Costley

Tokamak Energy Ltd, 173 Brook Drive, Milton Park, Abingdon OX14 4SD, UK

AEC, 0000-0001-5661-5045

Review

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Author for correspondence:
A. E. Costley
e-mail: alan.costley@tokamakenergy.co.uk

The question of size of a tokamak fusion reactor is central to current fusion research especially with the large device, ITER, under construction and even larger DEMO reactors under initial engineering design. In this paper, the question of size is addressed initially from a physics perspective. It is shown that in addition to size, field and plasma shape are important too, and shape can be a significant factor. For a spherical tokamak (ST), the elongated shape leads to significant reductions in major radius and/or field for comparable fusion performance. Further, it is shown that when the density limit is taken into account, the relationship between fusion power and fusion gain is almost independent of size, implying that relatively small, high performance reactors should be possible. In order to realize a small, high performance fusion module based on the ST, feasible solutions to several key technical challenges must be developed. These are identified and possible design solutions outlined. The results of the physics, technical and engineering studies are integrated using the Tokamak Energy system code, and the results of a scoping study are reviewed. The results indicate that a relatively small ST using high temperature superconductor magnets should be feasible and may provide an alternative, possibly faster, 'small modular' route to fusion power.

This article is part of a discussion meeting issue 'Fusion energy using tokamaks: can development be accelerated?'.

1. Introduction

Research with tokamaks has been ongoing for more than 50 years and for most of that time it has generally been considered that in order to generate net fusion power tokamak fusion reactors will have to be large and powerful; major radius of ≥ 6 m, plasma volume

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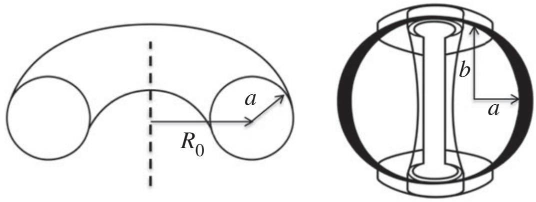

Figure 1. Schematic of conventional and spherical tokamaks. The aspect ratio $A = R_0/a$ and the elongation $\kappa = b/a$.

$\geq$1000 m^3, and operation with fusion power $\geq$1 GW, typically being considered necessary. The large scale ITER device currently under construction in France is the latest device in this line of approach [1], and designs of even larger and more powerful demonstration (DEMO) reactors are underway [2].

Recent work, however, has shown that an approach based on much smaller and lower power devices may be possible [3–5]. The approach is based on a re-evaluation of the empirical scaling of energy confinement time with machine parameters such as size and field, and the adoption of a relatively new technology, high temperature superconductors (HTSs) for the magnets. The shape of the plasma is important too. The work indicates that much smaller devices based on the spherical tokamak (ST) configuration, perhaps with a major radius of 1.5–2.0 m, volume of 50–100 m^3 and operating at relatively low power levels, 100–200 MW, may be feasible. Smaller devices would open the possibility of a modular approach to fusion power; that is one where single or multiple relatively small, low power devices would be used together to achieve the required power [6,7]. Smaller and less expensive fusion modules would enable faster development cycles and thereby speed up the realization of fusion power.

Spherical tokamaks have a much smaller ratio of plasma major radius (R_0) to plasma minor radius (a) than conventional tokamaks such as JET and ITER; they resemble the shape of a cored apple rather than the more conventional tokamak shape of a doughnut (figure 1). Research has shown that STs have beneficial properties from a reactor standpoint such as operation at high plasma pressure relative to the pressure of the confining magnetic field, and the generation of higher levels of self-driven current within the plasma [5,8,9]. This aspect is especially important. Auxiliary current drive systems are inefficient and thus can lead to substantial amounts of re-circulating power, and that power could be a major drain on the potential economics of a fusion reactor [10]. There are also indications that STs have higher levels of energy confinement relative to conventional shaped tokamaks [11]. STs share many of the challenges experienced in the development of the larger devices, for example the handling of the plasma exhaust in the divertor region where the power loads will be at the limit of available materials, and the installation of shielding on the inboard side necessary to protect the central column from the intense neutron and gamma radiation. The technical solutions being developed for the larger devices can be adapted and used on STs. The positive performance characteristics combined with potential solutions to the technical problems make STs particularly attractive for the compact approach.

In this paper, the work that is ongoing to realize this alternative approach to fusion power is reviewed. First, the question of size is addressed in general terms from a physics perspective. It is shown that it is not just size that is important; magnetic field and shape are important too, and the interplay between these parameters is developed. The question of fusion power is also important since that determines loads on the internal tokamak components and will limit the minimum possible device size. As shown in previous papers [3,4], the two key reactor performance parameters, i.e. the fusion gain, which is the power produced divided by the input power, and the fusion power are found to be directly linked. Using the latest empirical scalings for the energy confinement time, it is shown that the power needed for a useful fusion gain is three to

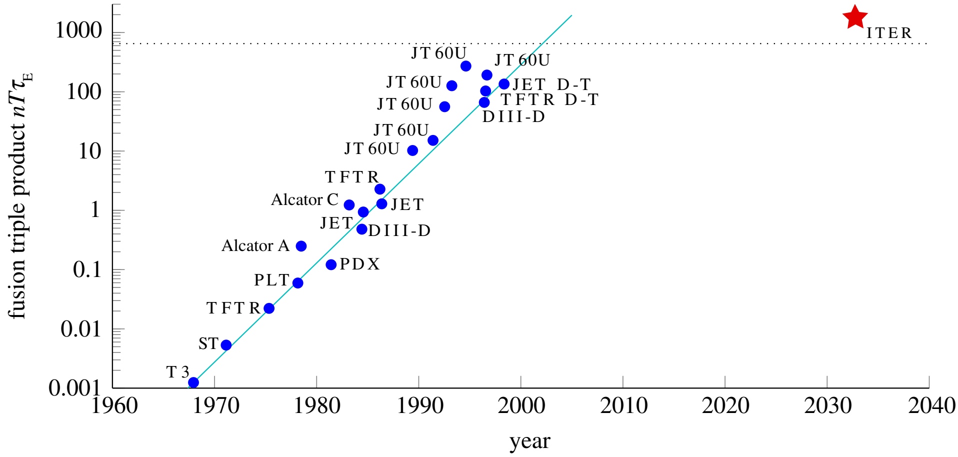

Figure 2. Improvement of the fusion triple product (relative units) with time as tokamaks of increasing, size, field and plasma current are constructed and brought into operation. (Online version in colour.)

four times lower than previously thought necessary. Taken together these findings indicate that smaller fusion devices based on the spherical tokamak should be feasible.

The realization of a relatively small, low power fusion module will depend on satisfactory solutions being developed to several key technical challenges such as the superconducting magnets that provide the plasma confining magnetic field, the inner shield that protects the central column from neutron and gamma radiation that potentially could cause material damage, and the divertor that handles the plasma exhaust. We present some details of these technical challenges and outline potential solutions.

A privately funded company in the UK, Tokamak Energy Ltd, is developing a compact approach using the spherical tokamak and magnets made with HTSs. The company has already constructed and operated developmental devices and has an ongoing R&D programme aimed at developing solutions to the key technical challenges. The company has developed a technology roadmap (TR) that charts a path through the physics and technical challenges that have to be met to realize modular fusion. In a final section, the main elements of the roadmap are presented. The paper concludes with a summary.

2. The question of size, field and shape

(a) The fusion triple product

The most important figure of merit of a fusion plasma is the product of the density (n), temperature (T) and energy confinement time (τ_E), $nT\tau_E$. This is known as the 'fusion triple product' and is derived from the work of John Lawson in 1957 [12]. For net fusion power, $nT\tau_E$ must be greater $1 \times 10^{21}\ \mathrm{m^{-3}}$ keVs [13]. The progress towards fusion can be measured with $nT\tau_E$. Figure 2 shows how $nT\tau_E$ has increased with time as larger tokamaks operating at higher magnetic field and higher plasma current were brought into operation. As can be seen, the rate of progress was very rapid from the late 1960s through to about 2000 but has slowed since, partly because of delays with ITER. Insight into key aspects of achieving net fusion power with tokamaks can be gained by looking closer at the fusion triple product.

The density and temperature are straightforward parameters but the energy confinement time is complicated. The energy confinement time characterizes the rate at which heat is transported from the hot central core of the plasma to the relatively cold surrounding material surfaces. Within a tokamak plasma there are multiple, interacting phenomena occurring simultaneously on a wide

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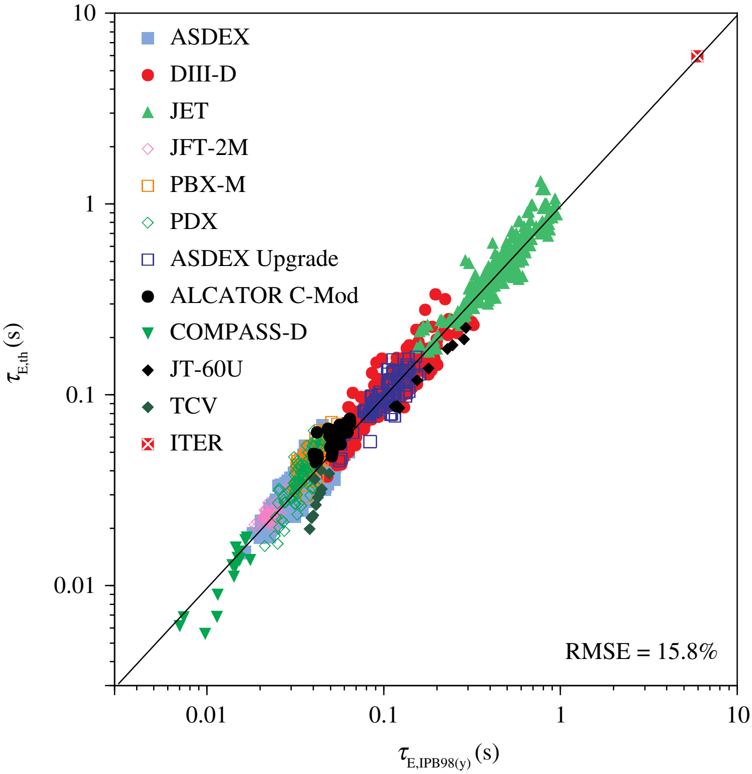

Figure 3. Measurements of the energy confinement time on 11 different tokamaks compared with the derived scaling. Reproduced from Wakatani *et al.* [15], with the permission of the IAEA.

range of temporal and spatial scales. These interactions lead to the transport of heat through processes that are essentially turbulent. While great progress has been made in understanding these processes it is not yet possible to determine the transport of heat through the plasma by a first principles approach. This is not an unfamiliar situation. In many areas of physics and engineering situations are too complex for a 'first principle' approach. In such situations, it is common to perform experiments on devices or structures of different scale and to determine how the parameter of interest scales with device parameters. Such experiments are carried out in fusion research on tokamaks, and scaling expressions, which relate the energy confinement time to the main device parameters, are determined.

The most extensive investigation of the scaling of the confinement time with tokamak parameters thus far carried out is that performed by the International Tokamak Physics Activity in the preparation for ITER [14]. Measured energy confinement times of plasmas created in multiple different tokamak devices located in different fusion laboratories were collated and analysed under standardized conditions. In total about 1000 plasma pulses were analysed and the scaling of the energy confinement time with the main device parameters was deduced. An example of the fit with the data is shown in figure 3 [15]. The most developed version of the scaling is known as the IPB98y2 scaling:

$$(\tau_E)_{IPB98y2} = 0.145\frac{I_P^{0.93}R_0^{1.39}a^{0.38}n^{0.41}B_T^{0.15}\kappa^{0.78}M^{0.19}}{P_L^{0.69}}. \tag{2.1}$$

Here I_p is the plasma current, B_T is the toroidal magnetic field, R_0 is the plasma major radius, n is the line average plasma density in units of $10^{-20}\,\mathrm{m}^{-3}$, a is the minor radius, κ is the elongation and M is the isotope mass normalized to the mass of a proton. P_L is the power transported from the plasma core to the surrounding surfaces. For a plasma in steady state, P_L is the sum of the input heating power supplied from external sources and the alpha power produced by the fusion reactions within the plasma minus any power radiated from the plasma core due to atomic and electronic processes.

Experiments on tokamak plasmas have shown that they are subject to operational limits and one important limit is the density limit. Attempts to raise the density above the limit usually result in loss of confinement and collapse of the plasma. While the physical processes giving rise to the limit are not fully understood, empirically it is found that the density limit is well represented by $(n)_{\mathrm{lim}} \propto I_P/a^2$. We note that this limit goes inversely with size and so to maintain the density as devices become larger the current has to be increased.

(b) Analytical derivation of $nT\tau_E$

Starting from a simplified version of the scaling of the energy confinement time, and taking into account the density limit, it is possible to derive an expression for the fusion triple product in terms of the main device parameters [3,4]. In the published derivation the plasma was assumed to be cylindrical and so the elongated shape of the plasma was not included. Repeating the derivation but including the shape characterized through the aspect ratio, $A = R_0/a$ and the elongation, κ, we find

$$nT\tau_E \propto \frac{H^2}{q^3} R_0^2 B_T^3 \left(\frac{\kappa^{7/2}}{A^3}\right). \tag{2.2}$$

The details of the derivation are given in appendix A. Here $H = (\tau_E)_{\mathrm{experiment}}/(\tau_E)_{\mathrm{scaling}}$ is a simple numerical factor included to allow for the fact that in experiments the actual confinement time, $(\tau_E)_{\mathrm{experiment}}$, can be above or below $(\tau_E)_{\mathrm{scaling}}$. Typically H is in the range 0.8–2.0. The parameter $q = R_0 B_T \kappa / A^2 I_p$ is an operational parameter related to the pitch of the magnetic field in the plasma and known as the 'safety factor'. Experiments have shown that q must be kept ≥ 2.5 and so for a given device it is essentially a constraint on the maximum current that can be used.

Equation (2.2) shows clearly how the fusion triple product depends on size (major radius), field and shape, and how the magnitude of one of these parameters can be traded for another. We note especially that the dependence on shape, $(\kappa^{7/2}/A^3)$, is strong. For example, going from the conventional tokamak ($A = 3.4$, $\kappa = 1.8$) to the spherical tokamak ($A = 1.8$, $\kappa = 2.7$), the major radius can be reduced by factor of approximately 5, or the field can be reduced by factor of approximately 3, for the same $nT\tau_E$.

Equation (2.2) demonstrates the origin of what might be thought of as the three main approaches to fusion: that is, the large device approach as pursued by ITER and the big DEMO reactors in which size (major radius) is emphasized; the high field approach as pursued notably at MIT and Commonwealth Fusion Systems, USA, and ENEA, Italy; and the highly elongated ST approach being developed for example at Princeton Plasma Physics Laboratory, USA, Culham Centre for Fusion Energy and Tokamak Energy Ltd, UK. Thus far most effort has been concentrated on the large size and high field routes.

(c) Impact of the density limit

As mentioned above, tokamak plasmas are subject to a density limit, which is inversely dependent on size. The density limit has a significant impact on the size dependence of $nT\tau_E$. This can be seen by generalizing the limit to the form $(n)_{\mathrm{lim}} \propto I_P/(A/R)^y$ and then repeating the analysis. The result is shown in equation (2.3).

$$nT\tau_E \propto \frac{H^2}{q^3} R_0^{4-y} B_T^3 \left(\frac{\kappa^{7/2}}{A^{5-y}}\right). \tag{2.3}$$

Of course, when we take $y = 2$ we obtain the former result. The analysis shows that the size dependence in $nT\tau_E$ is substantially reduced due to inverse size dependence of the density limit.

The limit also reduces the impact of shape through the reduction of the exponent of A but the influence of shape remains strong.

3. The question of power

(a) The fusion gain

An important parameter of a fusion plasma is the fusion gain, Q_{fus}. Tokamak plasmas require some external power (P_{aux}) to be applied, for example to drive current in the plasma, and it is partly through this externally applied power that the device operators have control of the plasma (there are other means as well). Q_{fus} is the ratio of the fusion power to P_{aux}: $Q_{\text{fus}} = P_{\text{fus}}/P_{\text{aux}}$. Obviously, in order to achieve net fusion power, $Q_{\text{fus}} > 1$. In practice, because of balance of plant power consumptions, a reactor will require $Q_{\text{fus}} \sim 10$–20 to produce net power.

For a tokamak plasma operating in steady state, there is a direct relationship between Q_{fus} and $nT\tau_{\text{E}}$:

$$Q_{\text{fus}} = \frac{5cnT\tau_{\text{E}}}{5 - cnT\tau_{\text{E}}}, \tag{3.1}$$

where c is a constant. Using the expression for $nT\tau_{\text{E}}$ from equation (2.2) we can obtain the relationship between Q_{fus} and the main device parameters:

$$Q_{\text{fus}} = \frac{5cH^2R_0^2B_{\text{T}}^3\kappa^{7/2}}{5q^3A^3 - cH^2R_0^2B_{\text{T}}^3\kappa^{7/2}}. \tag{3.2}$$

Here again we can see the interplay between size, field and shape. The safety factor, q, is present too.

In a tokamak, the plasma is contained by the magnetic field and the ratio of the plasma pressure to the magnetic field pressure known as beta, β, is an important quantity: $\beta = nT/B^2$. As in the case of the density, there is a limit in the amount of pressure that can be contained with a given magnetic field, and in this case also the limit depends on the plasma current and inversely on the plasma size: $(\beta)_{\text{lim}} \propto I_{\text{P}}/aB_{\text{T}}$. It is convenient to use a normalized version of beta defined as $\beta_{\text{N}} = \beta RB/I_{\text{P}}A$. The beta limit has the effect of linking the current, field and power, and it can be readily shown that $P_{\text{fus}} \propto \beta_{\text{N}}^2B^4R^3\kappa^3/q^2A^4$. We can use this relationship to eliminate B in equation (3.2) and we find

$$Q_{\text{fus}} = \frac{5cH^2P_{\text{fus}}^{3/4}\kappa^{5/4}}{5\beta_{\text{N}}^{3/2}q^{3/2}R^{1/4} - cH^2P_{\text{fus}}^{3/4}\kappa^{5/4}}. \tag{3.3}$$

Thus we see that the two most important parameters of a fusion plasma, the fusion gain and the fusion power, are directly linked. This linkage has significant consequences. For example, as the fusion power is raised, the denominator in equation (3.3) goes to zero and Q_{fus} goes to infinity, a situation known as 'ignition' in fusion language. It also means that there is a limit to the amount of power that can be obtained from a reactor of given size, shape and performance parameters q and β_{N}. In consequence, reactor designs aiming at high power often have to include an additional loss mechanism to generate the power required at a manageable Q_{fus}, for example by adding impurities to enhance core radiation. But probably the most noteworthy point is that the dependence on size is very weak, and in this case also that weak dependence can be traced back to the inverse size scaling in the density limit. The absence of a significant size dependence means that, in principle, small devices could have a high fusion performance, but of course there will be technical limitations as to how small a device can be constructed and still have tolerable loads (as discussed below). A significant dependence on shape, however, remains through κ and favours devices with high elongation.

(b) System codes

The analysis carried out thus far has used simplified expressions for the shape and volume of the plasma, energy confinement time etc., and has ignored an important aspect: the internal self-driven plasma current known as the bootstrap current. This was done so that the equations could be solved analytically and the underlying physics behaviour elucidated. In particular, it highlights the interplay between size, field and shape, and the important impact of the density limit. For more accurate work, codes that use the full expressions are used: these are known as system codes. System codes go further than just handling the physics aspects; they also include expressions for the main engineering aspects; for example, the power load in the divertor and the stresses in the magnet. Thus they permit the interplay between the main physics and engineering aspects to be studied, and optimized device designs developed to meet given high-level performance requirements. Tokamak Energy has developed a system code, which has been benchmarked against established codes and the details published [3].

(c) Beta dependence of the confinement scaling

As described above, equation (2.1) presents the ITER IPB98y2 scaling for the energy confinement time in engineering variables, that is variables such as B_{T}, R and I_{p}. The confinement time scaling can also be expressed in what are termed 'physics variables', and the principal physics variables are normalized Larmor radius, ρ_* (basically size), normalized collisionality, υ_* (basically time), the plasma beta, β, and the safety factor, q. There are established definitions of the physics variables in terms of the engineering variables and so the transformation between the two types of expression is straightforward. In terms of the physics variables, the IPB98y2 scaling is

$$(\tau_{\mathrm{E}})_{IPB98y2} \propto \rho_*^{-2.70}\beta^{-0.9}\upsilon_*^{-0.01}. \tag{3.4}$$

Note, that it depends inversely on beta.

In a coordinated international effort, the ITPA (International Tokamak Physics Activity) Topical Group on Confinement have carried out a series of experiments on different tokamaks to probe directly the dependence of the energy confinement time on the individual physics variables. One notable observation has been that the confinement time in single device scans has a low to zero dependence on beta, especially at low collisionality where reactor plasmas will operate. This was confirmed recently in a paper from JET that reviewed and summarized 10 years of work in support of ITER [16].

Researchers have developed scalings for τ_{E} in which the beta dependence is constrained to be zero. These are known as beta-independent scalings and several have been developed [17–19]. The fit to the experimental data is found to be almost as good as in the case of the beta-dependent scaling. Since the beta-independent scalings give consistency between the single-device and multi-device experiments they are arguably more appropriate. When the beta-independent scalings are used to predict reactor performance rather than the beta-dependent scalings, the results are very different.

In figure 4 we show some results obtained with the system code. We plot P_{fus} as a function of R_0 at constant $Q_{\mathrm{fus}} = 30$, $H = 1.5$ for both IPB98y2 scaling and beta-independent scalings for both $A = 3.2$, $\kappa = 2.2$ and $A = 1.8$, $\kappa = 2.6$. The broad characteristics as described earlier are repeated, and especially we note the weak dependence of P_{fus} on size when Q_{fus} is fixed, in line with the results of the analytical derivations. The most significant finding, however, is that the absolute level of the power needed to achieve a given Q_{fus} is a factor ∼2–4 lower in the case of the beta-independent scalings: the reduction factor depends on which beta-independent scaling is used. This is an important result. It suggests that there could be solutions for high performance fusion devices at much smaller size than currently envisaged. Included in the figure are the magnitudes of a few key technical parameters—the field on the conductor in the centre column, B_{cond}, the load on the first wall due to the neutron flux, n_{w}, and the value of the ratio P_{div}/R which is an indication of the power load in the divertor. P_{div} is the power flowing into the divertor region.

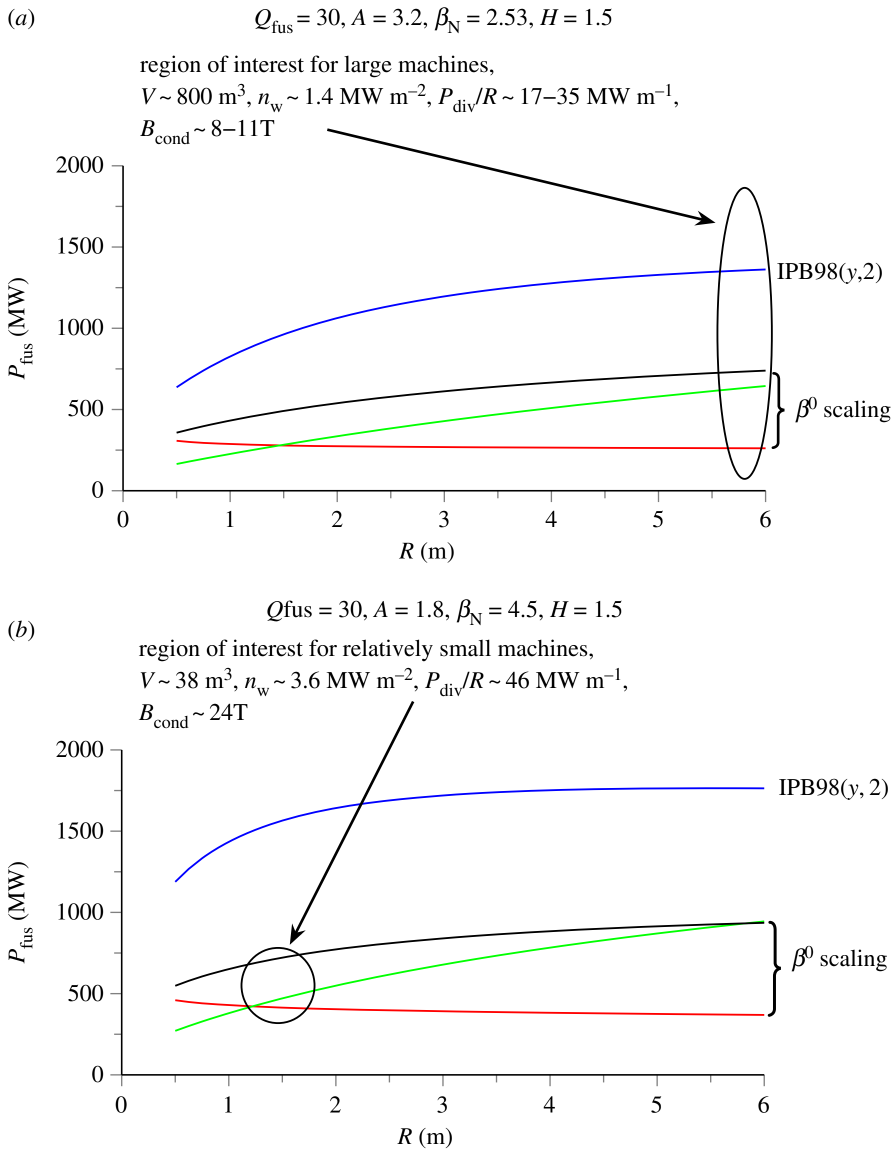

Figure 4. P_{fus} as a function of R_0 at constant $Q_{fus} = 30$, $H = 1.5$ for $A = 3.2$ and $A = 1.8$ for both IPB98y2 scaling and beta-independent scalings. Here P_{div} is the transported loss power that has to be handled in the divertor after allowance for radiation losses. Details are given in [3]. The conventional high A, large tokamak solution (*a*) and the potential low A spherical tokamak solution (*b*) are indicated. Reproduced from Costley *et al.* [3], with the permission of the IAEA.

The values of these parameters suggest that there might be technically feasible solutions at small major radius especially in the low aspect ratio case. Whether they could be realized in practice will depend on the engineering and technology, and we turn next to these.

4. Technical aspects

In order to realize a compact fusion device based on the spherical tokamak feasible designs for several key technical aspects have to be developed. Four areas are especially important—the magnets, the inner radiation shield, the divertor and the overall mechanical structure. The solutions adopted in these areas will have a significant impact on the overall device size. Fortunately, much of the R&D and technical solutions being developed in support of the much more powerful and large DEMO reactors are directly relevant and can be used; in other areas, customized R&D is required.

(a) HTS magnets

Powerful electromagnets are needed to confine the plasma and thus far in fusion research the conductors employed have been either copper or low temperature superconductors (LTSs) operating at or near the temperature of liquid helium, approximately 4 K. Because of their very high power consumption copper magnets are not suitable for a fusion device that is intended to make net power. LTS magnets require substantial shielding and powerful cryogenic systems and so are not suitable for small compact devices. HTSs, on the other hand, appear promising for compact devices. In addition to operating at higher temperature—up to approximately 77 K although optimum performance is usually achieved in the 20–30 K range—they can also operate at higher current densities and higher magnetic fields. Thus they offer substantial improvement in the three critical magnet performance parameters—operating temperature, current density and magnetic field—with factors of two to five improvements being typical. The principal design considerations then become those of mechanical support, especially supporting the conductor to keep the stress and strain in the conductor within acceptable limits; protecting the conductor against events that could potentially lead to a loss of superconducting performance known as a 'quench', and protecting the superconductor against radiation—neutrons and gamma rays (photons with energy > 1 MeV)—emitted by the plasma core, which could significantly change the performance characteristics of the conductor and possibly irreparably damage it.

The HTS conductor is supplied in the form of tape, typically 10 mm $\times$ 0.1 mm in cross-section; the thickness of the HTS layer is typically only approximately 1 µm. The step from these tapes to the construction of magnets potentially involving thousands of kilometres of tape is substantial. Fusion devices are not the only possible application of HTS magnets: other possible applications include high power transmission lines, high field accelerators, high efficiency motors and advanced MRI machines. Hence there is significant motivation for commercial development of HTS tape and HTS based magnets, and that development is underway. Nevertheless, the fusion application has specific requirements and so dedicated R&D is needed, and is on-going in some fusion establishments and private companies.

There are several aspects to the needed R&D. The performance of the tapes needs to be fully characterized: in particular the critical current density and maximum tolerable magnetic field need to be determined as a function of the operating temperature for candidate HTS conductors. The stress and strain limits need to be determined and how the performance of the tape changes with these parameters. Methods of supporting the tapes to create cables are required, and for fusion applications cables that can carry currents approximately 50–100 kA are needed. The step from cable to the magnet requires structural support of the cable, appropriate insulation and provision for cryogenic cooling.

As mentioned above, under certain conditions, the tape can lose its superconductor property and then potentially the high level of stored energy in the magnet could be deposited in a specific location leading to severe local damage. Methods to detect the fault early and dissipate the energy rapidly, generically known as 'quench protection', are required and are under development.

The neutron and gamma radiation emitted by the plasma can change the properties of the HTS tape and the insulators and other structural elements supporting the tape. Research is on-going to determine the impact of the radiation on the tape performance (e.g. [20]), but more needs to be done in this area.

Currently there are several groups, institutions and companies working on the specific problems of HTS magnets for fusion. It is a growing field, ripe with innovation and rapid progress is being made.

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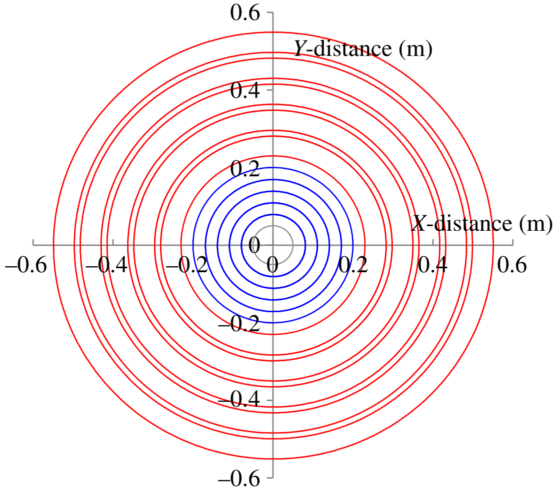

Figure 5. Schematic of inner radiation shield (in red) comprising concentric annular volumes composed of tungsten carbide or boride alloy, separated by water channels. The shield surrounds the central core (in blue), which consists of HTS tape, cooling channels and structural support. Reproduced from Windsor & Morgan [23], with the permission of the IAEA.

(b) Inner radiation shield

Neutrons emitted by the plasma, and gamma rays that arise from collisions of the neutrons with the structure of the tokamak, will penetrate and be partly absorbed by the HTS magnet; the resulting heating could terminate the superconducting properties. The neutrons and gammas can also potentially damage the HTS tape and that damage could lead to a shortening of the operating lifetime of the magnet. To deal with both these potential problems an effective radiation shield is needed. The shield would be mounted on the central column and so, for a device of fixed aspect ratio, the thickness of the shield adds directly to the major radius. The volume of the plasma scales as $\sim R_0^3$ and so for a compact device it is important that the shield is of minimum thickness to provide the required shielding.

Much is known about the interaction of neutrons and gammas with atoms and so accurate calculations can be made of the effectiveness of candidate neutron absorbing materials. For example, it is known that hydrogen is particularly effective at reducing the energy of neutrons, and that boron is an effective absorber of neutrons especially at low energies. Atoms of high atomic number such as tungsten have high neutron interaction cross sections. However, such interactions can create high-energy gamma rays and these can give rise to heating and potentially material damage. Gamma rays are absorbed or scattered to lower energies by the electrons in high atomic number materials, so tungsten is a good gamma ray attenuator. Based on the knowledge of these interactions, neutron and gamma ray absorbing shields can be designed, and their effectiveness in a compact spherical tokamak can be modelled.

Windsor, Morgan and colleagues at Tokamak Energy have carried out such calculations. Using a model of a spherical tokamak fitted with an inner radiation shield, they have examined the effectiveness of possible shield configurations and materials [21,22]. They have found that a shield comprising concentric annular volumes composed of tungsten carbide or boride alloy, separated by water channels constitutes a very effective neutron absorbing shield with a thickness in the range 0.3–0.4 m being sufficient to reduce the neutron heating to a reasonable level (figure 5) [23].

While much is known about the interaction of neutrons and gammas with atoms, little is known about the possible degradation of the superconducting properties of HTS under neutron and gamma irradiation. Some experiments have been carried out using neutrons in fission reactors but the conditions of the experiments are not the same as will be experienced in the application in fusion: in particular, the temperature of the HTS samples was approximately 290 K

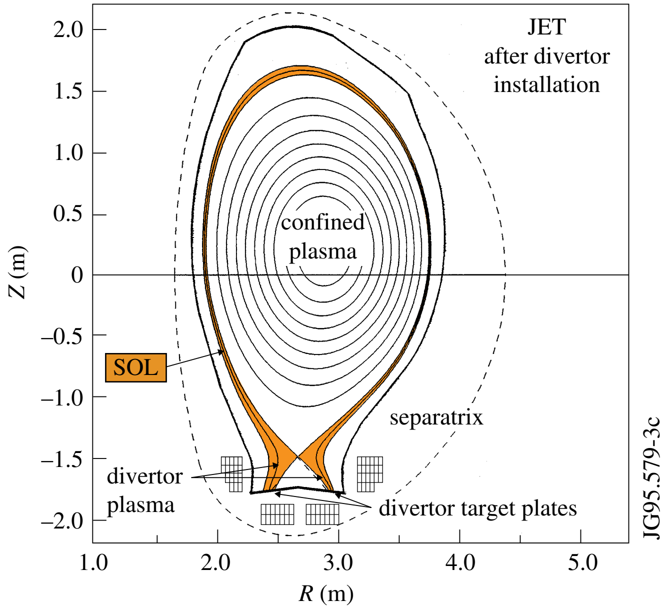

Figure 6. Schematic of the JET divertor illustrating the power flow to the divertor target plates. Copyright © Courtesy of EUROFusion.

rather than 20–30 K as is likely in an application in fusion reactors. Nevertheless, using the available data Windsor and Morgan and colleagues have attempted to estimate the lifetime of the magnet when protected by the tungsten carbide shield and found it to be reasonable [23].

Experimental work in which the candidate shields are manufactured and the attenuation of the neutron flux is measured is an obvious next step. Similarly, measurements of the impact of energetic neutrons and gamma rays on HTS tape with the tape held at the planned operating temperature are required. Several groups are carrying out research in this area and new and relevant results can be expected in the near term.

(c) Divertor

The plasma in a tokamak is contained by the magnetic field and in the poloidal cross-section the lines of constant magnetic field approximate to concentric ellipses (figure 6). The energy in the core of the plasma transports outwards across the lines of magnetic field essentially by a conductive diffusive process. Towards the edge of the plasma a line of constant magnetic field inevitably comes into contact with a solid surface, which is usually a metal. In consequence, the plasma comes into contact with a solid surface and that can lead to erosion and generate impurities, which can penetrate the plasma and cause fuel dilution and other deleterious effects.

By changing the magnetic configuration, it is possible to arrange the plasma/solid interaction to occur at some distance from the plasma core so that strong local pumping can be employed to reduce the amount of eroded material that permeates back to the plasma. Usually it is arranged that the contact is at the top or bottom of the vacuum vessel in a specially designed mechanical arrangement known as a divertor (figure 6). The region of the plasma that interacts with the solid surface is known as the scrape off layer (SOL) and the solid surface is known as a divertor plate. The transport of energy along lines of constant of magnetic field is much faster than it is across the lines and so the SOL is very narrow, typically approximately 1 cm. This can lead to very high power loadings on the divertor plate. In consequence, erosion of the plate and contamination of the plasma can occur.

As an example, we can consider the case of a plasma operating essentially in steady state conditions with a fusion gain of 10 and at a fusion power of 200 MW. The externally applied power

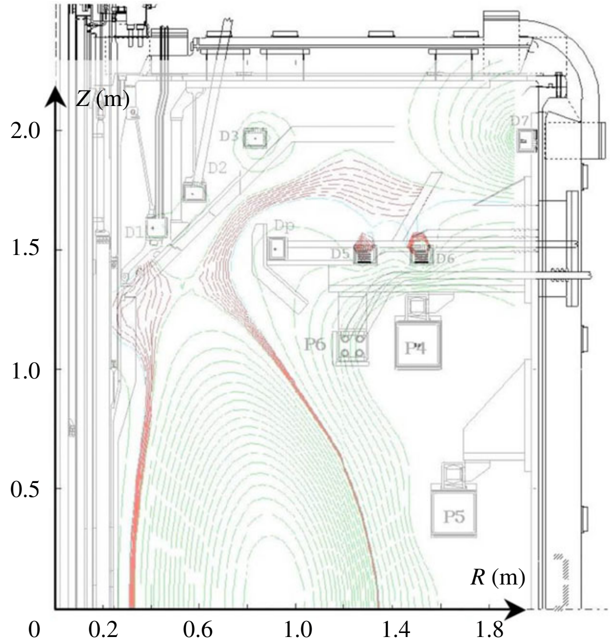

Figure 7. Schematic of a Super X divertor illustrating the approximately five times increase in the plasma/target contact area. Reproduced from Valanju *et al.* [25], with the permission of AIP Publishing.

is 20 MW. Twenty per cent of the fusion power is in the form of energetic alpha particles, which are confined in the plasma and act as the source of power that sustains the fusion reactions. Thus in steady state the power transported across the lines of magnetic field is $0.2 \times 200 + 20 = 60\,\mathrm{MW}$ and, if not attenuated, this power could impinge on a very narrow area (less than $1\,\mathrm{m}^2$) of the divertor plate leading to very high power loads, far in excess of what is generally regarded as acceptable for a solid surface, which is in the region $5–10\,\mathrm{MW\,m^{-2}}$. Clearly mitigating measures are required.

Several methods of reducing the power load are under development: for example, tilting the divertor plate increases the area of interaction; adding a controlled amount of impurities in the plasma edge can convert some of the power to electromagnetic radiation, which is then distributed over a large area in the vacuum chamber. Use of liquid lithium, which is an alkali-metal, to form the plasma/divertor interaction surface has potentially very high power load capability and can give rapid self-repair [24].

Modifications of the magnetic configuration in the divertor region are also under development. In one case, known as the Super X divertor (figure 7) the interaction region is substantially increased and, in consequence, it is expected that the power load will be reduced by a factor of 5–10 [25]. Two divertors can also be used; one at the top and one at the bottom. This arrangement is known as double-null divertor. The double-null arrangement is particularly favourable for STs because the enhanced curvature of the magnetic field leads to the bulk of the transported power going to the outer divertor plate where the plasma/surface interaction region can be extended.

It is unlikely that one single measure will solve the problem entirely; rather a few of these measures will be employed. The optimum arrangement will depend on the individual device design.

(d) Mechanical arrangement

The high magnetic field in the complex tokamak structure with some components, especially the toroidal field (TF) coils, carrying high currents inevitably leads to strong forces in the structures,

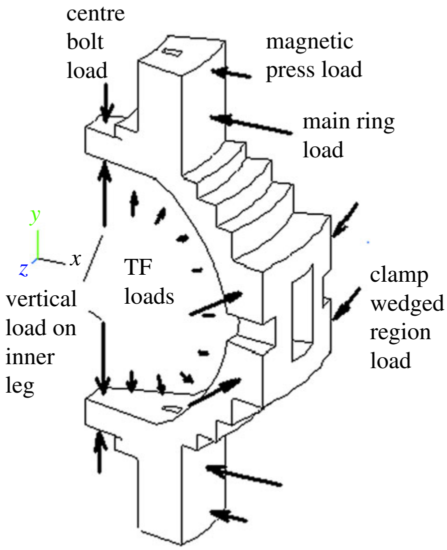

Figure 8. External 'C-Clamp' structure proposed for IGNITOR to react the strong forces arising from the high currents in the toroidal field magnets. Reproduced from Titus [26], courtesy P. Titus.

and care must be taken in the mechanical design to prevent stresses and strains from exceeding limits. Over the 50 plus years that tokamaks have been constructed, device designers have developed several different ways of reacting the forces to cope with this problem. For example, in one case, the TF coil bears, or is 'bucked' against the central solenoid (CS), which supports the TF and loads the CS in compression. An example of this approach is the JET tokamak.

For compact devices, this aspect of the design is potentially difficult because of the requirement to keep the structure to a minimum especially in the radial direction. On the other hand, one possible solution for a small device is to react some of the forces against an external structure. Such an arrangement has been proposed for the high field ignition tokamak, Ignitor [26]. In the proposed design a large radial compression ring is used in combination with an external clamp/case to offset the vertical separating forces (figure 8). In principle, an arrangement along these lines could be used for a small modular ST. The ST would have a HTS magnet and so a key design decision would be whether to have the external structure at cryogenic temperature or at ambient temperature.

(e) Other technical issues

There are other technical areas that will have to be addressed in the design of a compact ST fusion module. Key areas are the start-up, ramp-down and control of the plasma for the required long pulse duration; the selection and optimization of the additional heating, current drive and diagnostic systems; remote handling and maintenance; tritium handling and ultimately tritium breeding; site location, licensing and safety. The same aspects occur in the publicly funded programmes to prepare and construct large and powerful DEMO reactors, and much development has occurred in these areas. Some of the solutions developed for the large devices can be adopted for the ST. In other areas, dedicated solutions will be required.

5. Integration: design of a compact ST fusion module

The output of the various technical activities must be integrated to produce a design of a compact fusion module and the tool for the initial integration is the system code. Once the broad

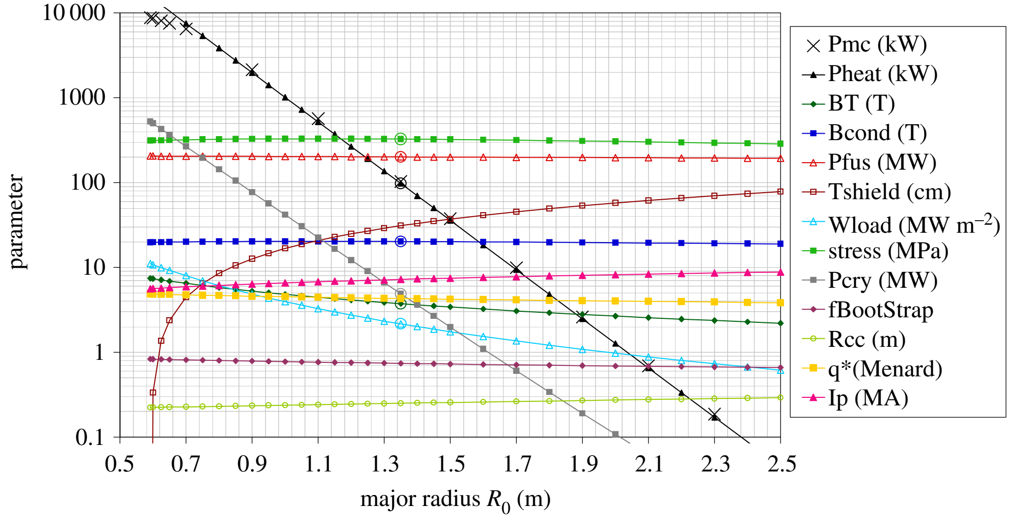

Figure 9. Heating power deposited in the superconducting core and the power needed in the cryo-system to handle this heating and other key parameters, as a function of plasma major radius. The extra space made available by increasing the major radius has been divided in the ratio 92% to the shield thickness T_{shield} and 8% to the HTS core radius R_{cc} across the plot. Reproduced from Sykes *et al.* [27], with the permission of the IAEA.

parameters of the device, such as size, field, shape, plasma current and power are determined with the system code, then a detailed design can be undertaken.

As an example, a scoping study has been undertaken by Sykes *et al.* [27]. The authors took as a reference plasma $Q_{\text{fus}} = 5$, $P_{\text{fus}} = 200$ MW, $H(IPB98y2) = 1.9$, $A = 1.8$, $\kappa = 2.64$, $\beta_N = 4.5$, and examined the engineering feasibility in relation to a few key parameters, for example the cryopower needed to maintain the temperature of the HTS magnet in the required range (20–30 K). A summary of the results of their study is shown in figure 9. In the study, as the major radius increases, the extra space is given to the inner radiation shield and to the superconducting core in the ratio 92% to 8% respectively. This ratio maintains the peak radial stress below its limiting value of 320 MPa, and generates a rapid reduction in the cryopower needed to maintain the HTS at the operating temperature as the major radius increases. We see that at the major radius $R_0 = 1.35$ m the shield thickness is 0.31 m, the peak field on the conductor in the central column is 20.2 T, the plasma current is 7.2 MA, the neutron heating to the central column is 97.7 kW and the wall load due to the fusion neutrons is 1.88 MW m^{-2}. To handle the neutron heating in the HTS, it is estimated that a cryogenic plant of 3.0 MW wall-plug power would be needed. These values, although demanding, are similar to those expected in much larger and more powerful DEMO reactors and are regarded as achievable.

The limited available data on degradation of HTS tape performance with neutron irradiation suggests that the tape lifetime corresponds to a total neutron fluence of about 10^{23} m^{-2} [20]. Taking this as a lifetime limit, it is possible to estimate the lifetime of the magnet (figure 10) and how this changes as the device major radius and inner shield thickness are increased. For many scientific objectives, the actual running time would be composed of many relatively short pulses and so the magnet lifetime would, in effect, be much longer than the plasma running time.

6. Technology roadmap

Potentially there are many paths through the multiple scientific and technical development steps needed to realize a compact ST fusion module. A similar situation arises in the development of ground breaking commercial products employing state-of-the-art technology and in that

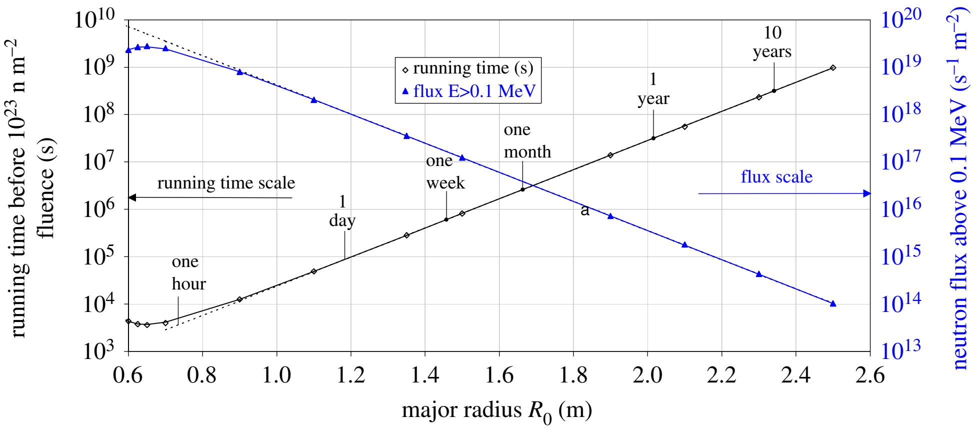

Figure 10. The neutron flux across the outer surface of the mid-plane region of the superconducting core for neutron energies above 0.1 MeV is shown by the full blue triangles (right-hand scale). The number of seconds of continuous running which correspond to a total neutron fluence of 10^{23} m^{-2} are shown by the open diamonds (left hand scale). Reproduced from Sykes *et al.* [27], with the permission of the IAEA.

area the technique of technology roadmapping has been developed to determine the optimum development path and is now widely applied [28]. In technology roadmapping, the challenges are broken down into distinct steps, potential methods of resolution are identified, links between the steps are identified, and the optimum path selected through a process of review, analysis and risk assessment. Some challenges have to be dealt with by dedicated in-house R&D; for other challenges the solution can be imported because of relevant work ongoing elsewhere. Estimates are made of the timing and need of resources—manpower, buildings, funding etc.—and these in turn are linked to the technical work. In this way, the TR is built to give a dynamic holistic overview of the necessary steps to achieve the required objective.

In collaboration with the Open University and the University of Cambridge, Tokamak Energy has applied this process to develop a TR for the realization of commercial fusion power based on compact ST fusion modules [29]. Fourteen distinct areas where development is needed have been identified, and a resolution path for each developed. The four key technical areas summarized in §§4a–d are included. In some cases, for example the HTS magnets, dedicated R&D projects with specific milestones are required and are incorporated. Necessary supporting and enabling aspects, such as funding, manpower, buildings, site, safety and licensing are included. The TR goes beyond the development of the fusion module and extends to the commercial realization of fusion based on compact ST devices. Potentially fusion has applications other than the generation of electricity—for example, the energy in the fusion neutrons could be used to produce high temperature, which in turn could be used for the production of hydrogen to support a hydrogen economy. Such additional potential applications are included in the TR.

7. Summary

In summary, progress in fusion research is usually measured by the fusion triple product $nT\tau_E$. It has long been known that in order to increase $nT\tau_E$, tokamaks of increasing size and/or higher magnetic field are needed but the benefit of plasma shape on $nT\tau_E$ has not been previously highlighted. Starting from a simplified form of the empirical scaling of the energy confinement time, we have derived the separate dependences on major radius, magnetic field and shape, and using these dependences it is possible to see how one aspect can be traded against another. Significantly, the impact of shape is substantial: for example, an elongated plasma in a spherical tokamak with $A = 1.8$ and $\kappa = 2.7$ would have the same $nT\tau_E$ as a conventional tokamak with $A = 3.4$, $\kappa = 1.8$ but the major radius could be approximately five times smaller or the magnetic field approximately three times lower. Further, our analysis shows that the two principal performance parameters of a tokamak—the fusion power gain and the output fusion power—are closely linked but perhaps more significantly the size dependence in the relationship is weak, and we show that that is primarily because of the inverse size scaling in the density limit. Hence from the perspective of this relationship, there is no need or benefit in building devices of increasing size. Calculations with a system code have confirmed these findings. We have also shown that the fusion power that corresponds to a given fusion gain can be up to factor of four lower if the energy confinement time is independent of beta, as indeed it has been found to be in current experiments, most notably on JET. These three physics aspects—favourable impact of elongated shape, weak size dependence between fusion power and fusion gain, and lower fusion power corresponding to a given fusion gain—open the possibility of a route to fusion power using relatively small spherical tokamaks.

The feasibility of this route depends on developing engineering solutions particularly in a few key areas; notably the magnets, the inner radiation shield, the divertor and the structure needed to react the forces in the magnets. HTSs appear to offer a solution to the magnets: certainly the basic performance of HTS tapes appear to be sufficient but obviously there is a substantial step from samples of short length to magnets of the scale needed for fusion reactor. MCNP calculations of the effectiveness of candidate shield materials have shown that a shield based on tungsten carbide and water should be effective. Construction and testing of such a shield is required. The divertor is a critical area but there are several different approaches to the problem under development. If the device can be made small enough then an external structure that reacts the main forces can be considered. There is significant overlap between the technical needs of the spherical tokamak approach and that being pursued for much larger DEMO conventional shape tokamaks, and many of the technical solutions, materials and supporting systems, for example heating, current drive and diagnostic systems, developed for that programme can be imported.

Tokamak Energy Ltd, UK, is a privately funded company that aims to realize this route to fusion power. It has an ambitious, multifaceted programme focused on solving the key technical issues on a near to mid-term time scale, and of demonstrating that success on spherical tokamaks of enhancing fusion performance. Applications of fusion power in addition to the generation of electricity, for example the generation of hydrogen using high temperatures created by fusion neutrons, are also under consideration.

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Disclaimer. All views expressed in this paper are personal.

Appendix A. Derivation of expression for $nT\tau_E$ with plasma shape included

For an elliptical shaped plasma, the plasma volume $V_P \propto R_0 a^2\kappa \propto R_0^3\kappa/A^2$. We assume the plasma temperature is constant in the range approximately 10–20 keV, so $P_{fus} \propto n^2T^2R_0^3\kappa/A^2$. The power loss by conduction from the plasma core is the stored energy divided by the confinement time; $P_L \propto nTR_0^3\kappa/A^2(\tau_E)_{storedenergy}$. The density limit, known as the Greenwald density, is proportional to the plasma current and inversely to the minor radius; $n_{lim} \propto I_PA^2/R_0^2$. By definition, the plasma beta $\beta \propto nT/B^2 \propto \beta_N I_P A/RB$. Here β_N is the normalized beta, $\beta_N = \beta/(I_P/aB_T)$. The safety factor $q \propto B_T R_0\kappa/A^2 I_P$.

Experimental confinement times are typically of the form:

$$(\tau_E)_{\text{scaling}} \propto \frac{I_p R_0^{3/2} a^{1/2} n^{1/2} \kappa^{3/4}}{P_L^{1/2}} \propto \frac{I_p R_0^2 n^{1/2} \kappa^{3/4}}{A^{1/2} P_L^{1/2}}. \tag{A 1}$$

$(\tau_E)_{\text{stored energy}} = \tau_E = H(\tau_E)_{\text{scaling}}$ where H is a simple multiplier and so

$$\tau_E \propto H\left(\frac{I_p R_0^2 n^{1/2} \kappa^{3/4}}{A^{1/2}}\right)\left(\frac{A\tau_E^{1/2}}{n^{1/2} T^{1/2} \kappa^{1/2} R_0^{3/2}}\right). \tag{A 2}$$

Hence

$$\tau_E^{1/2} \propto \frac{H I_p R_0^{1/2} \kappa^{1/4} A^{1/2}}{T^{1/2}}, \tag{A 3}$$

so

$$nT\tau_E \propto H^2 I_p^2 R_0 n \kappa^{1/2} A. \tag{A 4}$$

For operation at fixed fraction of the density limit, we can eliminate n using, $n \propto I_p A^2/R^2$. We can eliminate I_P using the expression for the safety factor, $I_P \propto BR_0\kappa/qA^2$, and so

$$nT\tau_E \propto \frac{H^2 B_T^3 R_0^2 \kappa^{7/2}}{q^3 A^3}. \tag{A 5}$$

References

1. Aymar R, Barabaschi P, Shimomura Y. 2002 The ITER design. *Plasma Phys. Control. Fusion* **44**, 519–565. (doi:10.1088/0741-3335/44/5/304)
2. Maisonnier D *et al.* 2007 Power plant conceptual studies in Europe. *Nucl. Fusion* **47**, 1524–1532. (doi:10.1088/0029-5515/47/11/014)
3. Costley AE, Hugill J, Buxton P. 2015 On the power and size of tokamak fusion pilot plant and reactors. *Nucl. Fusion* **55**, 033001. (doi:10.1088/0029-5515/55/3/033001)
4. Costley AE. 2016 On the fusion triple product and fusion power gain of tokamak pilot plants and reactors. *Nucl. Fusion* **56**, 066003. (doi:10.1088/0029-5515/56/6/066003)
5. Menard JE *et al.* 2016 Fusion nuclear science facilities and pilot plants based on the spherical tokamak. *Nucl. Fusion* **56**, 106023. (doi:10.1088/0029-5515/56/10/106023)
6. Gryaznevich MP, Chuyanov VA. 2015 Advancing fusion by innovations: smaller, quicker, cheaper. *J. Phys Conf. Ser.* **591**, 012005. (doi:10.1088/1742-6596/591/1/012005)
7. Windridge M. 2019 Smaller and quicker with spherical tokamaks and high-temperature superconductors. *Phil. Trans. R. Soc. A* **377**, 20170438. (doi:10.1098/rsta.2018.0438)
8. Sykes A *et al.* 1999 The spherical tokamak programme at Culham. *Nucl. Fusion* **39**, 1271–1282. (doi:10.1088/0029-5515/39/9Y/305)
9. Ono M, Kaita R. 2015 Recent progress on spherical torus research. *Phys. Plasmas* **22**, 040501-1–040501-74. (doi:10.1063/1.4915073)
10. Chuyanov VA, Gryaznevich MP. 2017 Modular fusion power plant. *Fusion Eng. Des.* **122**, 238–252. (doi:10.1016/j.fusengdes.2017.07.017)
11. Valovic M *et al.* 2009 Scaling of H-mode energy confinement with I_p and B_T in the MAST spherical tokamak. *Nucl. Fusion* **49**, 075016. (doi:10.1088/0029-5515/49/7/075016)
12. Lawson JD. 1957 Some criteria for a power producing thermonuclear reactor. *Proc. Phys. Soc. B* **70**, 6. (doi:10.1088/0370-1301/70/1/303)
13. Wesson J. 2011 *Tokamaks*, 4th edn, pp. 2–13. Oxford, UK: Oxford University Press.
14. Doyle EJ *et al.* 2007 Progress in ITER Physics Basis, Chapter 2: Plasma confinement and transport. *Nucl. Fusion* **47**, S18–S127. (doi:10.1088/0029-5515/47/6/S02)
15. Wakatani M *et al.* 1999 ITER Physics Basis, Chapter 2: Plasma confinement and transport. *Nucl. Fusion* **39**, 2175–2249. (doi:10.1088/0029-5515/39/12/302)
16. Litaudon X *et al.* 2017 Overview of the JET results in support to ITER. *Nucl. Fusion* **57**, 102001. (doi:10.1088/1741-4326/aa5e28)
17. McDonald DC et al. 2004 The beta scaling of energy confinement in ELMy H-modes in JET. *Plasma Phys. Control. Fusion* **46**, A215–A225. (doi:10.1088/0741-3335/46/5A/023)

18. Cordey JG *et al.* 2005 Scaling of the energy confinement time with β and collisionality approaching ITER conditions. *Nucl. Fusion* **45**, 1078–1084. (doi:10.1088/0029-5515/45/9/007)
19. Petty CC. 2008 Sizing up plasmas using dimensionless parameters. *Phys. Plasmas* **15**, 080501. (doi:10.1063/1.2961043)
20. Prokopec R, Fischer DX, Weber HW, Eisterer M. 2014 Suitability of coated conductors for fusion magnets in view of their radiation response. *Supercond. Sci. Technol.* **28**, 014005. (doi:10.1088/0953-2048/28/1/014005)
21. Windsor CG, Morgan JG, Buxton PF. 2015 Heat deposition into the superconducting central column of a spherical tokamak fusion plant. *Nucl. Fusion* **55**, 023014. (doi:10.1088/0029-5515/55/2/023014)
22. Windsor CG *et al.* 2017 Modelling the power deposition into a spherical tokamak fusion power plant. *Nucl. Fusion* **57**, 036001. (doi:10.1088/1741-4326/57/3/036001)
23. Windsor CG, Morgan JG. 2017 Neutron and gamma flux distributions and their implications for radiation damage in the shielded superconducting core of a fusion power plant. *Nucl. Fusion* **57**, 116032. (doi:10.1088/1741-4326/aa7e3e)
24. Kaita R *et al.* 2007 Low recycling and high power density handling physics in the Current Drive Experiment-Upgrade with lithium plasma-facing components. *Phys. Plasmas* **14**, 056111. (doi:10.1063/1.2718509)
25. Valanju PM, Kotschenreuther M, Mahajan SM, Canik J. 2009 Super-X divertors and high power density fusion devices. *Phys. Plasmas* **16**, 056110-1–056110-7. (doi:10.1063/1.3110984)
26. Titus P. 2003 Structural Design of High Field Tokamaks, PSFC/JA-03-9. See https://core.ac.uk/download/pdf/78059375.pdf.
27. Sykes A *et al.* 2018 Compact fusion energy based on the spherical tokamak. *Nucl. Fusion* **58**, 016039. (doi:10.1088/1741-4326/aa8c8d)
28. Phaal R, Farrukh C, Probert D. 2010 Roadmapping for strategy and innovation. University of Cambridge, Institute for Manufacturing.
29. Pearson RJ *et al.* In preparation. A technology roadmap to the realisation of compact fusion energy.

royalsocietypublishing.org/journal/rsta *Phil. Trans. R. Soc. A* **377**: 20170439