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Alternative approaches: concept improvements in magnetic fusion research

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Alternative approaches: concept improvements in magnetic fusion research

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While the conventional tokamak, as embodied in JET, is the front-runner in the magnetic-confinement approach to fusion, other concepts are being developed that might, in the long term, prove more attractive for power generation ('concept improvement' research). Also, in the shorter term, these concepts might offer a more-rapid and cheaper way of studying ignited or near-ignited plasmas. Their advantages, and disadvantages, are described. The stellarator offers steady-state operation at the cost of coil complexity, new advanced stellarators with overall size comparable to JET are either under construction or being commissioned in Japan and Germany. 'Spherical-torus' configurations, of which the most promising is currently the spherical tokamak, offer high-pressure containment in a compact device, but the high power density may mean power-plant technology will be more challenging (e.g. suitable materials). New spherical tokamaks will soon come into operation in the UK, USA and Russia and will test properties in larger higher-current longer-pulse plasmas. Meanwhile, other concepts such as the 'reversed-field pinch', magnetic-mirror systems and the dense Z-pinch, have their own advantages, though they are less well developed. The status of the various concepts are summarized as are their potential fusion applications that include electricity generation, acting as a fusion neutron source, and providing a driver for inertial fusion.

Keywords: stellarator; reversed-field pinch; compact toroids; spherical tokamaks; mirror machines; Z-pinches

1. Introduction

There are three related components of the international strategy for achieving electricity generation using magnetic fusion. These are

(a) pressing ahead with the tokamak (Keilhacker, this issue; Hawryluk, this issue), so that results from today's devices can be used to help design a sustained burning plasma experiment (e.g. the International Thermonuclear Experimental Reactor (ITER) (Aymar, this issue));

(b) development of the longer-term technology, especially materials (Ehrlich, this issue), required for a power plant; and

(c) investigation of 'concept improvements', i.e. alternative approaches that, although not as developed as the ITER-like tokamak, may in the long term be more attractive for power generation and/or for other fusion applications like testing prototype power-plant components in a fusion-neutron environment.

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Table 1. *Comparison of the current status of the tokamak with the alternative approaches*

((1) The conditions given in the table are approximate figures. They have not been achieved simultaneously. (2) $q(a)$ is the 'safety factor' at the edge of the plasma. It is the number of times the magnetic field line circuits toroidally for each poloidal circuit. $q(a)$ is negative in the reversed-field pinch (RFP) due to the reversal of the toroidal field near the edge. τ_E is the energy confinement time.)

approach	plasma configuration	steady state	current (MA)	T (keV)	n (10^{19} m^{-3})	β (%)	τ_E (ms)
tokamak	toroidal $q(a) > 1$	bootstrap current	< 7	< 40	< 100	< 12	< 1000
stellarator	toroidal $q(a) \sim 1$	inherent	0	< 5	< 10	< 2.2	< 250
RFP	toroidal $q(a) < 0$	helicity injection	< 1	< 1	< 10	< 20	< 6
spheromak	toroidal $q(a) \sim 0$	helicity	< 1	< 0.4	< 10	< 30	< 0.4
FRCs	toroidal $B_{tor} = 0$	rotating B fields? neutral beams?	< 1	< 3	< 500	< 95	< 1
spherical tokamak	toroidal $q(a) > 1$	bootstrap current	< 0.3	< 1	< 20	< 40	< 5
mirror	linear longitudinal B	inherent	0	< 10	< 1	< 15	< 600
Z-pinch	linear poloidal B	pulsed	20	< 30	10^7	*ca.* 100	10^{-5}

The more promising of these alternative approaches are described in this paper. Their properties and status are summarized in table 1. They range from well-established concepts, like the stellarator and Z-pinch, which are enjoying a new lease of life following the latest ideas and results, to more recent concepts like the spherical tokamak, pioneered on the START device at Culham in the past few years. There is great variation in geometry: from systems that are toroidal like the tokamak (the stellarator and reverse-field pinch), through much more compact, but still toroidal systems (spherical tokamak, spheromak), to linear devices (magnetic mirrors, Z-pinches). They also range in their potential applications: fusion power plants; facilities for testing power-plant materials and components; and drivers for inertial fusion.

As well as the differences, the similarities between the various systems should be stressed. The main challenge is the same: the sustained simultaneous optimization of plasma stability, confinement and exhaust. There is considerable cross-fertilization between the different approaches, with understanding of one line contributing to the understanding of the others. Closely related phenomena, such as magnetic reconnection, are observed on all the different systems (and in astrophysical plasmas such as the solar corona). The underlying physics is often surprisingly similar, despite very different geometry and timescales: for example, both spatial variations ('shear')

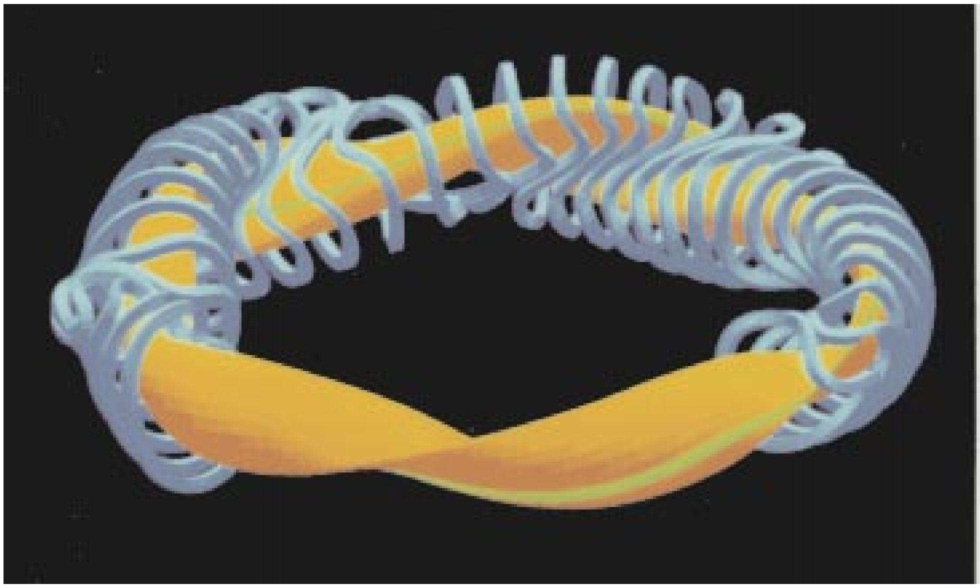

Figure 1. The superconducting modular coil set, and the plasma it will contain, for the advanced stellarator W7-X under construction at Greifswald, Germany. The major and minor radii of the plasma are 5.5 m and 0.52 m, respectively.

in plasma flow and magnetic field, and ion-orbit width effects, have an important influence on plasma stability and turbulence in devices as different as the tokamak and the Z-pinch.

2. Stellarators

The stellarator concept was 'invented' in the US by Spitzer (1958) at about the same time as the tokamak (the 1950s). Following much early research on stellarator systems, the focus of attention switched to tokamaks after the reports of very high temperatures on the Russian tokamak T-3, confirmed by a British team, in the late 1960s. However, interest has recently revived, and devices are operational in Spain, Germany, Japan, the US, Russia and Australia. A large device with superconducting coils has just commenced operation in Japan (LHD), and other devices are under construction in Germany (the superconducting W7-X) and the US (HSX). LHD and W7-X are comparable to JET (Keilhacker, this issue) in overall size though the plasma volume is significantly smaller; W7-X will have a plasma volume up to a third that of JET.

The basic difference from the tokamak is that the poloidal magnetic field required for equilibrium and confinement is provided by external coils and not by an externally driven current in the plasma, hence, key advantages of the stellarator plasma are that continuous operation is an intrinsic property and a source of free energy to drive instabilities (the plasma current) is eliminated. Several coil arrangements are possible: for example, coils that twist helically around the torus (torsatron, heliotron); coils lying in poloidal planes whose centres map out a helix of constant pitch (e.g. the Spanish heliac TJ-II); and distorted modular coils as on W7-X (figure 1). The resulting plasma (figure 1) is distorted in both poloidal and toroidal directions (i.e. the short and long ways around the torus), and so it is not axisymmetric as in most other toroidal systems: indeed the centre of the plasma itself often follows a helical path.

There are two main reasons for the renewal of interest in stellarators. Modern computing has meant that the very complicated calculations required to optimize the stellarator are now possible (in fact, these calculations now show that early stellarators had no closed magnetic surfaces). In these 'advanced stellarator' designs (Boozer 1998; Wagner 1998), as on the present and future German devices W7-AS and W7-X, the magnetic configuration maximizes the pressure-confining capability and minimizes the losses of heat and particles, while minimizing the effects of the internal plasma current driven by pressure gradients (the 'bootstrap current', which is beneficial in a tokamak, but undesirable in a stellarator). Another requirement is that the magnetic structure at the edge of the plasma should be suitable for handling the exhaust of particles and heat while restricting the influx of impurities (i.e. a magnetic 'divertor' action, as on modern tokamaks). These constraints lead to rather complex plasma and coil configurations (e.g. figure 1). The second reason for the renewal of interest is that advances in technology have meant that the high precision manufacture and positioning of the coil-sets required for these designs is now possible; for example, TJ-II was successfully constructed with a helical 3 m diameter copper coil with current density 100 MA m^{-2} and deviations from the helix of less than 1 mm.

The stellarator is a less-compact, lower-power-density device than the tokamak, as measured by the achievable value of β, the ratio of the plasma pressure to magnetic pressure ($\beta \lesssim 5\%$, cf. tokamak values less than *ca.* 10%). However, experiments show that stellarator plasmas are generally more stable, lacking the rapid terminations ('disruptions') to which the tokamak is prone. The energy-confinement time (figure 2, Stroth (1998)), temperatures and densities are similar in value to those on tokamaks, for example, temperatures on the W7-AS advanced stellarator have reached 5 keV, comparable to the value achieved on tokamaks with similar plasma volume. The challenges for the new stellarators, such as LHD and W7-X, are to show whether these properties are maintained in hotter larger plasmas, and to determine whether the advanced designs do indeed have the improved properties that theory and computer modelling predict. For LHD, for example, it is predicted that Q will reach *ca.* 0.35 (Q is the ratio of fusion power to heating power that would be achieved if the plasma had deuterium–tritium fuel). The challenge for stellarator theory, currently being addressed in the US, Germany and elsewhere, is to determine the scope for further optimization, to more-compact, higher-β systems.

3. Reverse-field pinches

Like the stellarator, the 'reversed-field pinch' (RFP) (Prager 1995) was one of the earliest systems to be studied in controlled-fusion research. A main distinguishing feature from the ITER-like tokamak (and stellarator) is that for the same plasma current, the toroidal magnetic field of the RFP is an order of magnitude smaller (and, remarkably, is reversed near the wall that surrounds the plasma). This property has the consequence that if the RFP had confinement properties comparable to that of the tokamak, then it would offer a power-plant designer the advantages of economy in magnetic energy (with $\beta \gtrsim 10\%$), low magnetic forces on external conductors, and, due to the high power density, the possibility of achieving ignition and sustained burn by ohmic heating alone (conventional tokamaks and stellarators require additional heating from neutral beams or radio-frequency (RF) waves). Other benefits are that non-superconducting coils can be employed, the aspect ratio can be chosen on engineering rather than plasma-physics grounds.

The reversal of the toroidal field, first explained by Taylor (1974), is a natural phenomenon that is due to the propensity of the plasma to seek, subject to constraints, a state of minimum magnetic energy (Taylor 1986). This is known as relaxation, and is a continuous process in that, provided the externally applied toroidal voltage and magnetic flux in the plasma are maintained, the equilibrium with reversed toroidal field is sustained. Simple theories assuming symmetric magnetic fields predict that the reversal must decay. In fact, the relaxation mechanism involves finite-resistivity, symmetry-breaking fluctuations and, because of the very strong analogies that exist between this and the way in which geophysical and astrophysical magnetic fields are thought to be generated, the autonomous production of internal electromotive force has come to be known as the dynamo effect. This three-dimensional, nonlinear process has been successfully simulated computationally and good fits to experimental measurements have been obtained (Ortolani & Schnack 1993). Recent measurements on the US device, MST, have shown that the Ohm's law equation parallel to the magnetic field is satisfied when the term due to the dynamo is included, i.e. $\boldsymbol{v} \times \boldsymbol{b}$ where $\boldsymbol{v}$ and $\boldsymbol{b}$ are the fluctuating velocity and field arising from 'tearing' instabilities (figure 3).

The RFP dynamo, then, involves continuous or repetitive activity from plasma instabilities. As a result, the ambient magnetic fluctuation level in an RFP divided by the mean field is around 1%, one to two orders of magnitude higher than in a tokamak, leading to the relatively poor energy-confinement properties of the RFP. Although the confinement is poor, high power input has meant that relatively high values of temperature (*ca.* 1 keV) have been achieved. Scaling the RFP to an economic power plant requires that this magnetic turbulence decreases with increasing plasma temperature. Armed with the insight that comes from relaxation and dynamo theory, several schemes have been proposed to 'tame' the dynamo. In particular, experiments on MST using the external circuitry to help produce current profiles at the edge that are close to the relaxed state have reduced the fluctuation level and raised the energy-confinement time by a factor of five, which is an encouraging result. Improved confinement has also been observed on the Japanese device TPE-1RM20 at high values of the pinch parameter Θ (the ratio of the poloidal field at the edge to the average toroidal field). The dynamo can be made to work in the experimentalist's favour in various current-drive schemes, including the use of phased oscillating current wave-forms in the external coils that was tried with some success on the US machine ZT-40.

The RFP, which originated in the UK (ZETA device at Harwell), is currently being investigated in four principal experiments with plasma currents up to the MA level: MST in USA; RFX in Italy; Extrap-T2 in Sweden; and TPE-RX (TPE-1RM20's successor) in Japan. These experiments have complementary activities in that RFX and TPE-RX investigate confinement scaling, while MST studies transport improvement. The 'resistive wall' instability, which becomes active when pulse lengths are longer than the wall magnetic diffusion time (for shorter times the wall stabilizes the

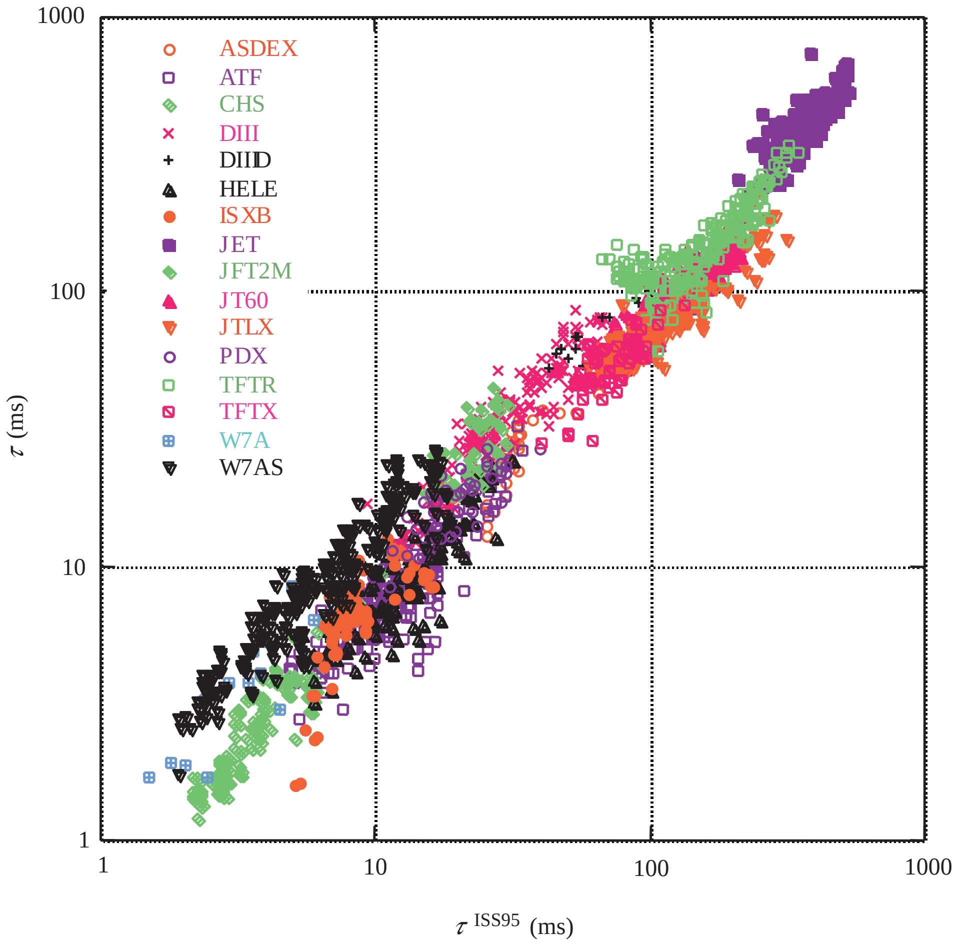

Figure 2. Comparison of energy-confinement time τ on tokamaks and stellarators. The stellarators are: ATF, CHS, HELE (Heliotron-E), W7-A and W7-AS. The plot uses the empirically based ISS95 scaling (International Stellarator Scaling (Stroth 1998)) and shows that stellarators and tokamaks with comparable parameters have comparable confinement times. Note that recently the new Japanese stellarator, LHD, has exceeded these values with $\tau \sim 250$ ms.

mode), is a threat to both the highest-β tokamaks and to RFPs, and is the principal focus of T2. Other issues include whether plasma performance improves as strongly with increasing plasma current as theory predicts, and whether there are satisfactory 'divertor' options for handling the exhaust of energy and particles.

4. Compact toroids

The compact toroid is a younger concept than either the tokamak, the stellarator or the RFP. 'Compact' here means there are no internal coils in the hole through the middle of the torus, and so near-spherical plasmas are possible. The spheromak was first proposed in the late 1970s by Rosenbluth & Bussac (1979), who extended Taylor's theory (Taylor 1986) to spherical geometry (figure 4). Like the RFP, its

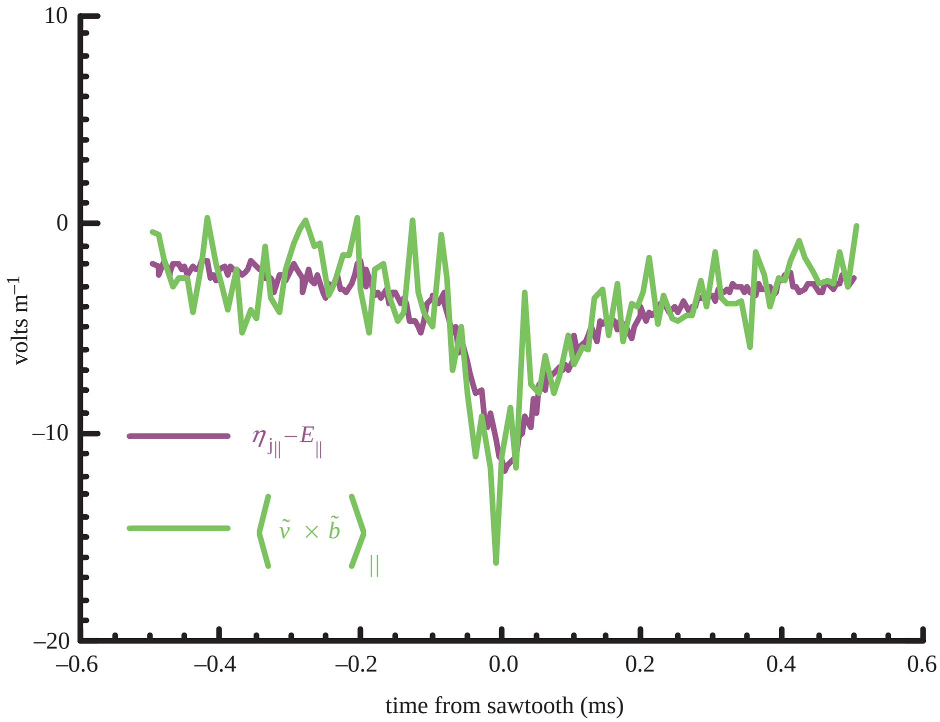

Figure 3. Measurements on the MST 'reverse-field pinch' so that Ohm's law parallel to the magnetic field is satisfied with the inclusion of the dynamo term, i.e. $\eta\langle j\rangle_{\|} + \langle E\rangle_{\|} = \langle \boldsymbol{v} \times \boldsymbol{b}\rangle_{\|}$ where $\boldsymbol{v}$ and $\boldsymbol{b}$ are the fluctuating velocity and field, η is the resistivity, E the electric field and $\langle\cdot\rangle$ denotes a time average.

plasma relaxes towards a minimum-energy state, but this time the toroidal field is generated by the plasma current alone (i.e. not by coils); a limiting case of the RFP magnetic configuration in which the toroidal field at the wall is zero, i.e. there is no field reversal. It was hoped that this scheme would combine the stability and good confinement properties of the tokamak with the efficiency of the RFP, with the added advantages of simplicity and compactness.

The first spheromaks were built in the 1980s and included CTX at Los Alamos (US) and SPHEX at the University of Manchester Institute of Science and Technology (UK) (Jarboe 1994; Robinson 1999; Hooper & Fowler 1996). Experiments showed that the plasma is in a partly relaxed state. Peak values on CTX of temperature and energy-confinement time were *ca.* 400 eV and 0.2 ms, respectively, and β at the plasma centre exceeded 20%. There are theoretical reasons for expecting that energy confinement might improve with temperature, as in RFPs, and this will be tested on the 'sustained spheromak physics experiment' (SSPX), being built at the Lawrence Livermore National Laboratory (US), which will be sustained by driving plasma currents using the established technique of injecting magnetic 'helicity' by applying voltages to external electrodes. Even with good confinement scaling, substantial currents (*ca.* 20 MA) would be needed for ignition in such systems.

The 'field-reverse configuration' (FRC) (Steinhauser 1996) is a compact toroid variant in which there is negligible toroidal magnetic field and the poloidal field comes

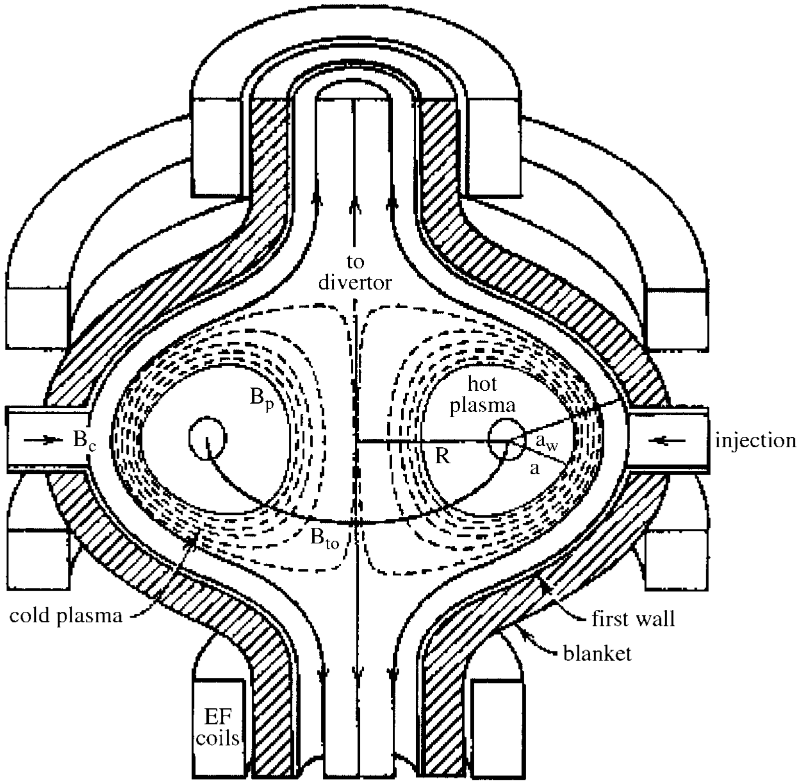

Figure 4. The spheromak geometry (Ortolani & Schnack 1993). The plasma is very close to spherical but remains toroidal. The toroidal field B_{toroidal} is generated by currents in the plasma, not by external coils.

from the toroidal plasma current, which is maintained solely by diamagnetism. It is a simple system, with only a low-field linear solenoidal magnet confining a plasma with toroidal geometry. Its potential advantages are compactness, high power density ($\beta \sim 50$–100%), no central components and a natural plasma exhaust action out of the ends of the solenoid. There are numerous FRC facilities, with plasma formation by various methods, notably in Russia, Japan and the US. Parameters achieved include densities comparable to tokamaks (up to *ca.* 10^{20} m^{-3}), and temperatures up to 3 keV (ions) and 500 eV (electrons). While the plasmas are, so far, only comparatively brief (*ca.* 1 ms), they are stable for longer than the Alfvén time that theory would suggest. A key parameter is the ratio of the plasma size to the radius of the ion Larmor orbit, which has been comparatively low so far. An important challenge for future research is to determine whether the unexpected stability is due to finite-orbit width effects, which would be smaller in a larger device, or to other beneficial effects like spatially varying ('sheared') plasma flow. Another key issue is the best way to heat and drive currents in such plasmas: neutral-beam-injection systems very similar to those used in other fusion concepts are being tested, for example on the FIX device at Osaka University. Reactor systems based on FRCs have been considered, including the use of colliding beams of high-energy ions to enhance fusion reactivity (Rostoker *et al.* 1997).

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Compact toroid research is also being undertaken for reasons other than the long-term aim of a fusion device. Experiments add to the understanding of basic phenomena, like turbulence and magnetic reconnection, which occur in a wide range of terrestrial and astrophysics plasmas. FRCs allow the study of high-β plasmas, which are of great interest in their own right. Also, the injection of accelerated spheromak plasmas into tokamaks is being investigated as a potential method for plasma fuelling, current drive and making the plasma rotate. Experiments on fuelling have already been undertaken on the Canadian Tokamak de Varennes and at the Lawrence Livermore National Laboratory, and an injector is being installed on JFT-2M in Japan prior to possible use on the stellarator LHD and the JET-sized tokamak JT-60U.

Finally, for many years it has been recognized that compact-toroid plasmas may also be used in 'magnetized target fusion' systems (Kirkpatrick *et al*. 1995), in which the densities and timescales are intermediate between those of magnetic fusion (10^{14} cm^{-3}, seconds) and inertial fusion (10^{25} cm^{-3}, nanoseconds). The basic idea in MTF is to implode a metal or gaseous shell or liner onto a preformed magnetized plasma, e.g. a compact toroid, to give very high pressures via near-adiabatic $p\,\mathrm{d}V$ compression. The main potential advantage of 'magnetized target fusion', as compared with standard inertial fusion, is reduced power requirement because the magnetic field both reduces thermal conduction and increases the deposition of energy in the plasma from charged fusion products. There are a range of concepts with different targets and different implosion methods (chemical explosives, magnetic acceleration of high-Z liners, etc.). Research is principally in the US and the Russian Federation. For example, on the MAGO system at Nizhni Novgorod, Russia, a plasma target of radius *ca.* 10 cm is accelerated by a cylindrical plasma gun into a metallic flux trap, then chemical explosives with energy *ca.* 500 MJ would be used to compress the metal shell to increase the plasma density and temperature to ignition. Suitable targets have been trapped but the final ignition experiment is yet to be undertaken. Reactor concepts have been considered, for example, repetitive ignition of magnetized targets in a blast chamber, using the greater than *ca.* 10 GJ generated per ignition in a Rankine cycle involving an MHD generator (Logan 1993).

5. Spherical tokamaks

The idea behind the spherical tokamak (ST) concept (Robinson 1999; Robinson *et al*. 1996) is to combine the attractive features of the conventional tokamak (good confinement and stability) with those of the RFP and the spheromak (compactness, high β). The ST has a very low aspect ratio ($R/a \sim 1.3$, compared with $R/a \sim 3$ for JET, where R and a are the mid-plane radii of the torus and of its poloidal cross-section). This means that the plasmas are close to spherical (they resemble cored apples, see figure 5*a*), and there is room for only a very slender current-carrying central rod to generate the toroidal magnetic field and a solenoid to induce and sustain the current. The high-β capability arises because, in tokamaks, the theoretical limiting value of β for which plasma stability is maintained scales as the normalized plasma current (I_{P}/aB where I_{P} is the current and B is the toroidal field). High normalized plasma current is possible in an ST because the magnetic-field line paths stay mainly in the high-field region of good curvature.

Research on STs contributes to the understanding of conventional tokamaks by testing understanding of tokamak behaviour in a new regime. There is also consider-

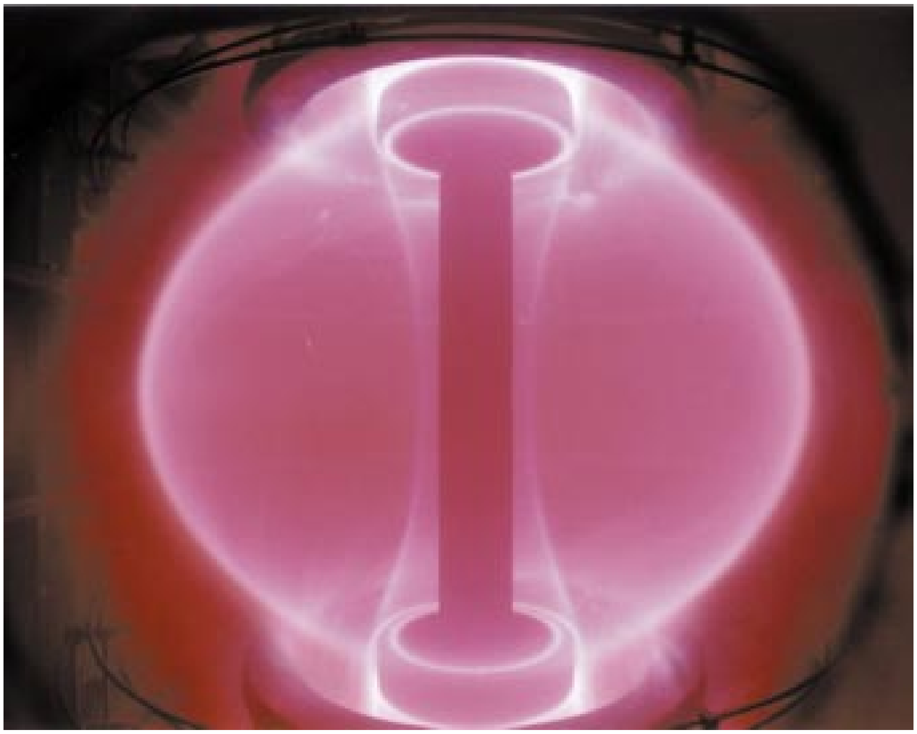

Figure 5. (*a*) Photograph of a high-β START plasma, showing that it is very nearly spherical and has a very-well-defined edge, indicating a confinement barrier. The diameter of the plasma exceeds 1 m. The 'X-points' at the top and bottom of the plasma result from the nearby divertor coils, which look like collars at the top and bottom of the current-carrying centre rod.

able overlap with spheromak research. Indeed, the first ST experiment was done by inserting a rod in the Heidelberg spheromak and the Rotamak at Flinders, Australia, and some of the more recent ST experiments have used the same technique, such as the FRC studies on NUCTE-ST (Nihon University, Japan). It has been found that the addition of the toroidal field from the rod does indeed improve performance. Also, the successful use of helicity injection on spheromaks translates readily to STs, and is the basic method of current formation and sustainment in the Helicity Injection Tokamak (HIT, US) and Helicity Injection Spherical Tokamak (HIST, Japan) devices.

The predicted advantages of the ST have now been confirmed experimentally, in particular on the START device at Culham (UK), but also on a range of smaller STs: CDX-U, HIT and MEDUSA, all in the US, TS-3 and TST-M (Japan) and ROTAMAK-ST (Australia). START plasmas have confinement and operating space at least as large as comparable conventional tokamaks, and improved stability properties. They are resilient to the disruptions that larger aspect ratio tokamaks are prone to; when an instability does occur (an 'internal reconnection event') there is a loss of energy and a jump in plasma current, but although the plasma changes shape it generally recovers. The use of high-power-beam heating equipment, loaned by the US Department of Energy, has led to the most impressive result from START: β up to 40% (figure 5*b*), which is three times the record for a conventional-aspect-ratio

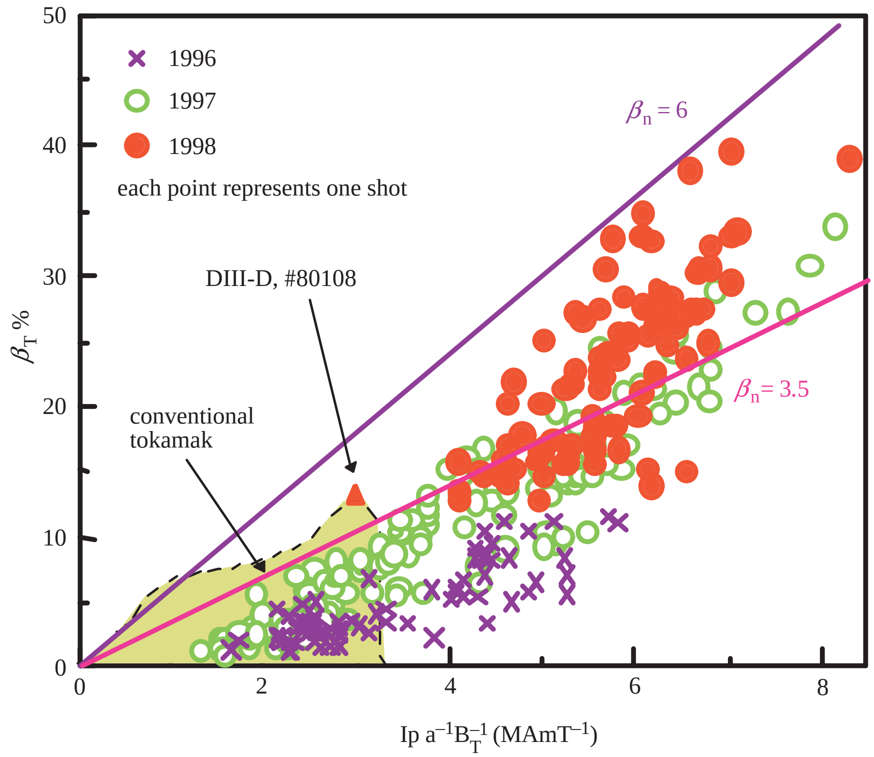

Figure 5. (*Cont.*) (*b*) Diagram showing the high-β values obtained on the START spherical tokamak using neutral-beam heating, more than three times the conventional tokamak record, and greatly exceeding the Troyon scaling for the β limit.

tokamak. Moreover, β_N, the ratio of β to the normalized plasma current I_P/aB, has been raised to *ca.* 5.7 compared with the more usual tokamak limit of $\beta_N \sim 3.5$ (figure 5*b*). Also, the high-β plasmas show no sign of a deterioration of confinement, indeed the reverse is true, the plasmas enter a new improved confinement regime with many of the characteristics of the 'H-mode' observed on conventional tokamaks such as JET (Keilhacker, this issue).

These very promising results have led to the design and construction of the next generation of STs: MAST (UK), NSTX and PEGASUS (US), GLOBUS-M (Russian Federation), ETE (Brazil) and TS-4 (Japan). A key objective of MAST, for example, is to establish whether or not the improved properties are maintained in much high-current longer-pulse larger plasmas than in START (volume increase up to *ca.* 20×). A second key objective is to use neutral-beam current drive to determine whether theoretically predicted high-β regimes with the potential for steady-state operation can be accessed (START is a short pulse machine with little non-inductive current-drive capability).

These new devices should confirm whether or not the ST is suitable for a fusion device. Its compact high-β nature means that it is potentially attractive both for power generation and for a low-tritium-usage facility to test the performance of power-plant components and materials in a fusion-neutron environment. A key com-

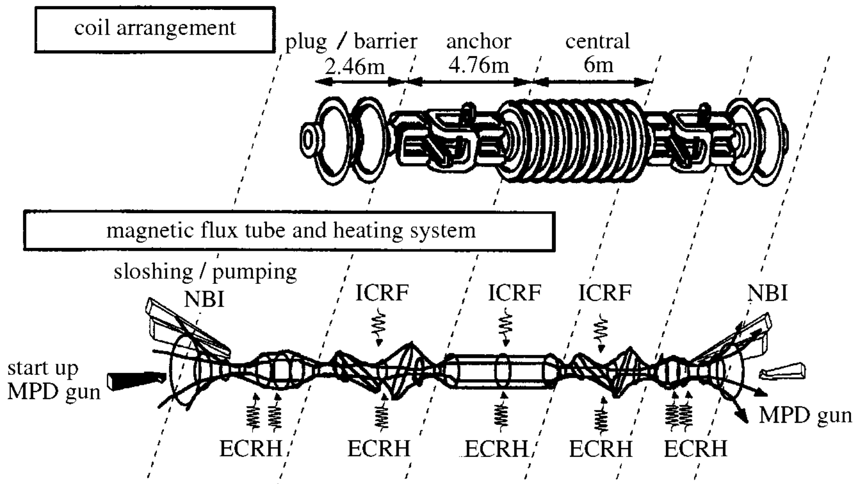

Figure 6. Coil arrangement and heating systems for the GAMMA-10 tandem mirror experiment at the University of Tsukuba, Japan. Note the dimensions of the device and the combined use of 'neutral-beam-injection' heating, and 'electron' and 'ion cyclotron resonance heating'.

ponent in either application is the central rod, which will have to carry a high current and withstand neutron flux; the lack of room for a substantial shield means that this rod cannot be superconducting. Possibilities include a shielded copper centre post, which would have to be replaced every few years in a power plant, and a cryogenic aluminium post, which, while requiring larger cooling power, would require much less ohmic power and might last the life of the power plant. Italian researchers have suggested a more radical solution, i.e. replacing the rod with a current-carrying plasma arc, though the feasibility of this system has still to be tested. Present confinement scaling studies from the first generation of STs, when combined with the conventional tokamak database, indicate that a plasma in a device with a current of *ca.* 15 MA and major radius of *ca.* 1.3 m would ignite.

6. Magnetic mirrors

Mirror devices (Ryutov 1991; Tamano 1995; Post & Ryutov 1995) are topologically different from the toroidal concepts discussed above. They are linear systems in which 'magnetic mirrors' plug the ends, i.e. a large ratio of the magnetic field at the ends to that at the centre of the device, so that only a small fraction of particles is lost through the ends. The Russian Federation and Japan have the largest mirror programmes, with GAMMA-10 at Tsukuba (Japan) and a 'gas dynamic trap' (GDT) at Novosibirsk the largest facilities, and Korea has the Hanbit device. There are a number of different designs for mirror systems (e.g. figure 6 shows the coil and heating arrangements for GAMMA-10), each aiming to suppress the particle losses including scattering into the loss-cone. Electron conduction losses remain a key issue for the mirror, to date the electron temperature has been rather low (*ca.* 100 eV), and to make the mirror a credible competitor for an ignition device requires methods of increasing this temperature. The theory of mirror systems predicts a number of attractive properties for fusion applications: continuous operation; stable high-β

operation; lack of disruptions; and a natural channel (through the ends) for removal of impurities and ash from fusion reactions. Also, if high enough temperatures can be attained for use of aneutronic 'advanced fuels', charged fusion products that leave through the ends could be used for 'direct-conversion' power production (while in deuterium–tritium power plants the energy of the neutrons would generate electricity indirectly, they would heat a tritium-breeding blanket to raise steam to drive a turbine).

Mirror devices require systems to stabilize low-order 'flute'-like MHD instabilities. Active systems include the use of RF waves (as on GAMMA-10), to heat the electrons and ions in different regions of the device, to give a stabilizing ponderomotive variation of electric potential along the system. In contrast, the GDT has a passive stabilization system: its high mirror ratio (less than *ca.* 70), and large length (*ca.* 10 m, greater than a particle mean free path), mean that plasma losses through the ends are determined by gas-dynamic equations (hence the name). This results in a significant stabilizing plasma in the favourable magnetic-curvature region between the mirror throats and the 'guns', which inject plasma into the system, allowing it to benefit fully from this region of favourable stability.

While confinement times of mirror devices are relatively low, tokamak-like densities and (separately) temperatures have been achieved. For example, in its 'hot-ion operating mode' very high fusion-relevant ion temperatures, greater than 10 keV, have been achieved in GAMMA-10, albeit at low density. At lower temperatures, in the 'high potential operating mode', the density of a 4 keV ion-temperature plasma has been improved by 50% by adjusting the pattern of the electron and ion cyclotron heating systems. In this mode, ECH is used to give a positive electrostatic potential plug in the mirror cells, reducing longitudinal ion losses, while radial losses remain low. A challenge for future work is to combine the hot-ion and high-potential operating modes, and to sustain the plasma for at least a second.

Like the spherical tokamak, the mirror is an attractive concept for a low-tritium-usage component test facility. The Novosibirsk group has proposed a device based on the 'gas dynamic trap'. This would be *ca.* 10 m long with a neutron irradiation test cell, just before the mirror coil, of 1–2 m^2 wall area. The mirror ratio would be *ca.* 28 T/1.8 T, i.e. *ca.* 15, it would be heated by high-energy neutral beams, and the neutron flux would exceed 1 $\mathrm{MW\,m^{-2}}$. Operation for a decade or more would give neutron fluences sufficient to test material properties for a power plant.

7. *Z*-pinches

In the *Z*-pinch (Haines 1994), a high voltage is applied to the ends of a fibre or, using electrodes, a gas in a vessel. The intense current pinches, ionizes and heats the plasma column. The Lawson conditions for fusion, allowing for end losses, can occur for currents close to the Pease–Braginskii current (*ca.* 1 MA), at which (a) the balance between the pressure gradient and the pinching force (Bennett relation), and (b) the power balance between Joule heating and bremsstrahlung losses, coincide. Above the Pease–Braginskii current, calculations predict a rapid collapse to much greater than solid densities, so that high fusion yields could occur. Although the simplest MHD theory predicts that the plasma column should be unstable to gross modes, theoretical work (Haines 1994; Coppins 1997) has indicated that there are a number of stabilizing effects that reduce their growth rate. These include spatial variation

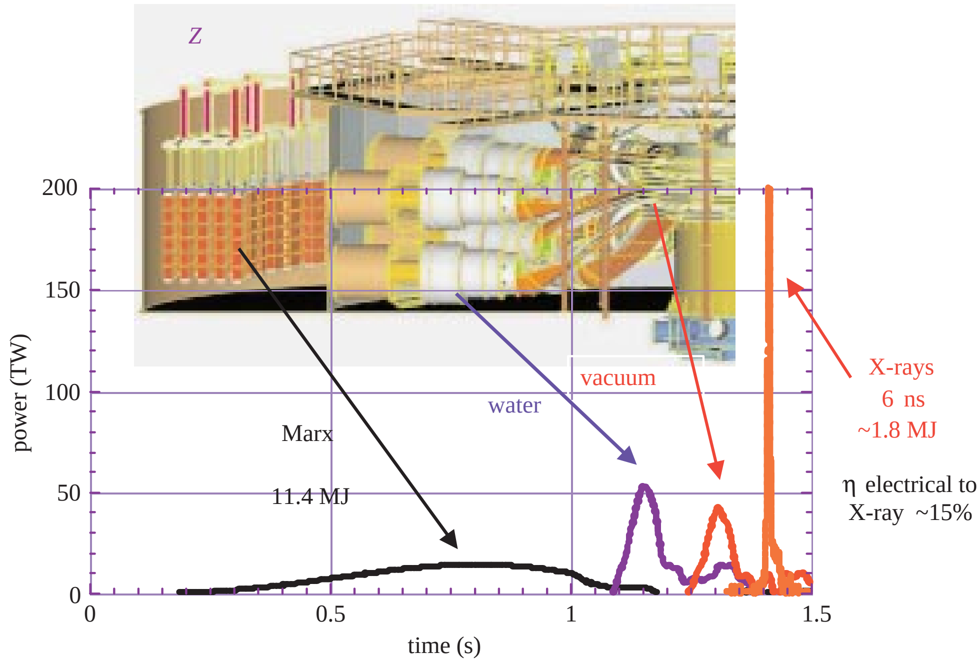

Figure 7. Production of high-power X-rays using the PBFA pulsed power facility at Sandia driving a Z-pinch wire array. In another shot, the X-ray power reached 290 TW.

('shear') of plasma flow, kinetic stabilization arising because the ion Larmor radius is a significant fraction of the pinch radius, and nonlinear effects. However, experiments with the MAGPIE generator at Imperial College (UK), with up to 1.2 MA through initially cryogenic deuterium fibres led to early instabilities when the plasma was relatively cold.

The development of multi-terawatt pulsed-power machines has revitalized Z-pinch research, so that conditions much closer to those required for fusion can be attained. Facilities include those at US laboratories, notably the SATURN and PBFA systems at Sandia National Laboratory (US) (Matzen 1997), MAGPIE and Russian devices (notably ANGARA-5), and France is building a new facility. Sandia has obtained some spectacular results, including ion temperatures up to *ca.* 36 keV and electron temperatures exceeding 1 keV using neon–argon plasmas in the SATURN facility (Wong *et al*. 1998).

A main aim of the Sandia experiments is to investigate the use of wire-array Z-pinches as X-ray source hohlraums for a driver for inertial fusion (Matzen 1997). In the study of the wire-array Z-pinch, a multiple aluminium- or tungsten-filament array, arranged to form a cylinder, acts as the load of the generator and implodes to its axis. The magnetic energy is transferred to radial kinetic energy of ions, which, at stagnation, converts via ion–ion collisions to an ion temperature and, via equipartition, to electron heating, further ionization and soft X-ray emission. With the PBFA-Z 11 MJ generator, which gives a current rising to 20 MA in 100 ns, up to 290 TW of soft X-rays in a 4.5 ns pulse have been produced using over 300 wires (figure 7). The efficiency of conversion of stored electrical energy is over 15%. With these X-rays, a hohlraum has been created with a radiation temperature of 145 eV: hence the interest as a possible driver for inertial fusion. The growth of Rayleigh–Taylor instabilities is an important limiting factor to the X-ray power, and a heuristic model developed at Imperial College (UK) shows that the initial seed level for these instabilities to grow is reduced by a factor $n^{-1/2}$, where n is the number of wires. A marked change in the scaling of X-ray power with wire number (or gap between wires) is shown both in experiments at Sandia and by this model. Experiments are planned in which a wire array is imploded onto a central deuterium fibre, which will hopefully allow exploration of conditions closer to those required for fusion. There is also design work at Sandia on a scaled-up machine ('X-1') with current about 60 MA which might yield X-ray powers up to *ca.* 1000 TW.

8. Conclusions

The stellarator has now become well established as the principle alternative to the main-line tokamak. New large experiments are commencing operation or are under construction, there are promising results from existing experiments, and scope for further optimization is indicated by large-scale modelling calculations. The reverse-field pinch and spheromak represent systems in which magnetic reconnection and relaxation are an integral part of the performance, and minimization of the associated fluctuations, through, for example, profile control, might allow improved performance. The recent rapid progress with the spherical tokamak has shown that compact systems can possess good confinement properties together with high β, and could provide a shorter less-expensive route for the development of fusion. Many of the other approaches are less well developed and have significant problems to overcome if they are to be developed into fusion-energy systems. All the alternative approaches add to our understanding of magnetic confinement systems and conversely have benefited from the focused R&D activities on physics and technology as part of the ITER process over the last few years. The main challenge for all the concepts is the simultaneous optimization of stability, confinement and exhaust to produce a more attractive power-plant concept.

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Discussion

J. Jacquinot (*JET Joint Undertaking, UK*). What does Dr Robinson think about fission/fusion hybrids—best of two worlds, or the worst of two worlds?

He mentioned that spherical tokamaks could be at a disadvantage with respect to TAE instabilities presumably because of the high β of the α-particles. Are the other physics aspects different from the standard tokamak regimes?

D. C. Robinson. Such systems raise concerns over safety and proliferation, and therefore are the worst of both worlds. Nevertheless, they are investigated in China for strategic reasons concerning future energy supply.

Though TAEs are predicted to be worse in spherical tokamaks, they have so far been observed to be benign. There are differences from the standard tokamak because at high β and when poloidal and toroidal fields are comparable, a magnetic well is created in the low magnetic curvature region that modifies the particle behaviour, rather like in stellarators. This will impact on transport, heating and current drive and on neoclassical tearing modes, though early considerations would indicate that the effects are positive.

G. H. Wolf (*Institute for Plasma Physics, Jülich, Germany*). My question addresses RFPs. The dynamo effect is vital to maintain the field reversal. However, it reduces turbulence (followed by reduced confinement) and it consumes flux swing to drive toroidal current thus limiting the pulse length from inductive current drive. Could this limitation in pulse length be overcome?

D. C. Robinson. The dynamo action allows the conversion of poloidal flux to toroidal flux by suitably oscillating the poloidal and toroidal fields with appropriate phases via helicity conservation: 'F–Θ' pumping. This was demonstrated on ZT-40 and could in principle sustain the RFP; however, the edge turbulence excited by this process could further degrade the confinement, making its application problematical.

R. S. Pease (*West Ilsley, Newbury, UK*). Do 'spherical tokamak' systems offer a quicker and cheaper route to a fusion ignition demonstration than conventional tokamaks?

D. C. Robinson. These devices can be built quickly and cheaply due to their simplicity and small size, and could permit a low cost route to an ignited plasma. However, the existing database on confinement, operating limits and exhaust plasmas is relatively small, and until we have results from the next generation of experiments such as MAST (UK) and NSRTX (US) at the MA level, in more sustained collisionless plasmas, we cannot proceed with confidence to devices at the 10–15 MA level which the potential to ignite. This potential has been recognized by others and for example Academician Velikhov of the Russian Federation has sought to build just such as a device in collaboration with other countries.

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