PHILOSOPHICAL TRANSACTIONS OF THE ROYAL SOCIETY A MATHEMATICAL, PHYSICAL & ENGINEERING SCIENCES

Generation and Confinement of Plasmas in Tokamak Systems [and Discussion]

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Generation and confinement of plasmas in tokamak systems

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The tokamak field configuration is based on plasma current flowing along a predominantly toroidal magnetic field. The joule dissipation associated with this current heats the plasma ions to typical temperatures of about 1 keV. Auxiliary heating by means of neutral atom beams has been used to raise the ion temperature to about 6 keV; high-frequency wave heating appears equally promising. The magnetohydrodynamic stability of the tokamak has been demonstrated experimentally for values of β (ratio of mean plasma pressure to magnetic pressure) up to 3%, confirming theoretical expectations of stable reactor operation. Ion-energy confinement is found to be close to the optimal theoretical prediction. Electron-energy confinement is anomalous, but its observed scaling is compatible with a moderate-sized tokamak reactor. The principal remaining plasma physics problem is believed to be the control of 'impurity' ions associated with wall interactions and with the burn-up of deuterium–tritium fuel.

Introduction

During the decade after the dramatic achievements of the T-3 device (Artsimovich *et al.* 1969), tokamak research has been done worldwide, and progress has continued at an encouraging rate. The present paper describes the basic tokamak concept, and a brief review is given of some recent experimental results in the U.S. tokamak programme. Subsequent papers of this symposium (Roberts; Robinson; Paul *et al.*; Riviere *et al.*; Hastie; Sweetman *et al.*; Peacock & Burgess; Hancox) present more detailed theoretical treatments of tokamak features, and more extensive experimental information. Futher information and references can be found in review articles by Artsimovich (1972) and Furth (1975).

1. Basic configuration and gross stability properties

The basic tokamak configuration is illustrated in figure 1. A strong toroidal magnetic field B_t, generated by an external toroidal solenoid, is supplemented by a weak poloidal field B_p, generated by a toroidally directed plasma current I which is induced by transformer action. The resultant helical magnetic field lines generate nested axisymmetric magnetic surfaces that confine the plasma particles.

The strength of the plasma current is limited by m.h.d. stability considerations which are discussed by Robinson (this symposium). It is useful to define a 'safety factor', $q = 2\pi/\iota \approx rB_t/RB_p$, which measures the margin relative to a critical rotational transform $\iota = 2\pi$ (cf. figure 1) for which a helical magnetic-field line closes on itself. Suitably m.h.d.-stable tokamak operation is obtained typically for $q(0) \approx 1$ on the axis $(r = 0)$ and $q(a) \approx 3$ at the plasma edge $(r = a)$.

The equilbrium magnetic-field configuration of the plasma requires appropriate shaping of the poloidal field by external, toroidally wound coils. In particular, the tendency of the plasma to expand in R must be controlled by an inward Lorentz force due to an externally generated vertical component of B_p. The maximum plasma pressure that can be sustained stably in this manner is given by $\beta^* = 8\pi\langle p^2\rangle^{\frac{1}{2}}/B^2 \lesssim ka/Rq^2$, where k depends sensitively on the detailed cross-sectional shape. According to current theory (see Hastie, this symposium), β-values in the range 5–10% should be obtainable under m.h.d.-stable conditions.

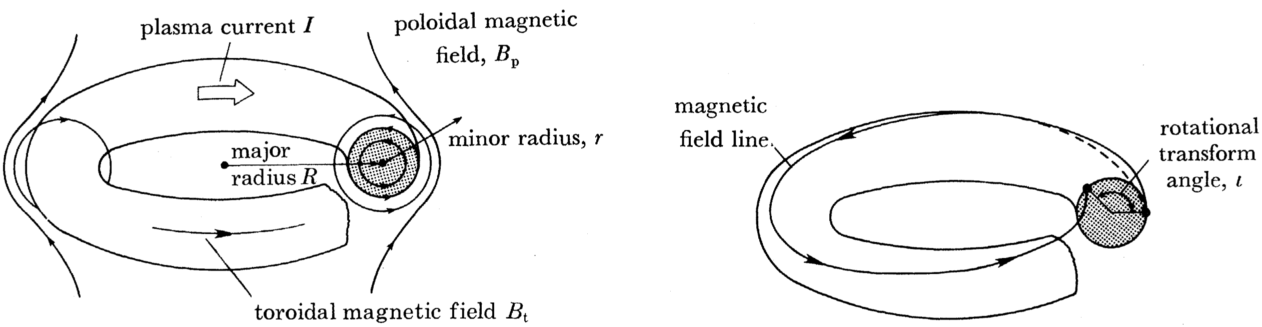

FIGURE 1. The tokamak configuration. Rotational transform angle $\iota = 2\pi$ gives fundamental kink instability; $\iota = 2\pi/m$ gives the harmonics.

2. Theory of tokamak plasma confinement

Since the ion plasma cross section for Coulomb scattering always exceeds that for fusion, a critical attribute of a fusion reactor is its ability to confine particles during many scattering times. The outstanding effectiveness of the tokamak in this regard was pointed out in Tamm & Sakharov (1961): axisymmetry implies conservation of toroidal canonical angular momentum, which constrains the maximum excursion of a scattered particle from its initial poloidal flux surface.

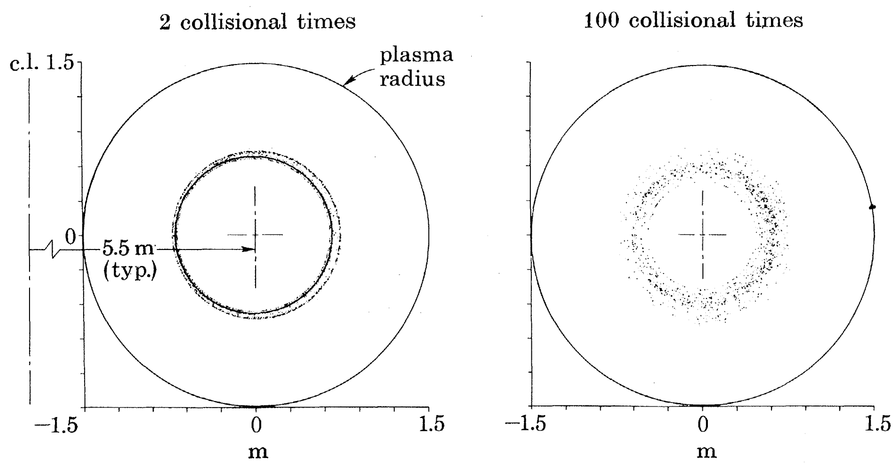

FIGURE 2. Monte-Carlo computer study of the classical diffusion of a representative ion in an Intor-like tokamak plasma at 5 MA, during the approximately 100 collisions required to regenerate the thermal energy content by D–T α-particle production. Projections are given of the particle orbits in the poloidal plane for plasma radius/Larmor radius = 428, B = 5T and T = 15 keV.

Monte-Carlo computer studies made by Dr G. Petravic and Dr A. H. Boozer (figure 2) for representative parameters of a small tokamak reactor illustrate the quality of the particle confinement.

The irreducible plasma transport due to Coulomb scattering is called 'classical', or sometimes 'neoclassical', to distinguish more sophisticated recent transport analyses from older ones (see Riviere *et al.*, this symposium). The principal classical energy-loss mechanism is the ion conductivity χ_i; the coefficients χ_e and D for electron heat conduction and particle transport are smaller than χ_i by $(m_e/m_i)^{\frac{1}{2}}$. Classical predictions of the minimum tokamak reactor size required to reach ignition in a deuterium–tritium plasma are very favourable: a reactor plasma half the size of that in figure 2 would be ample.

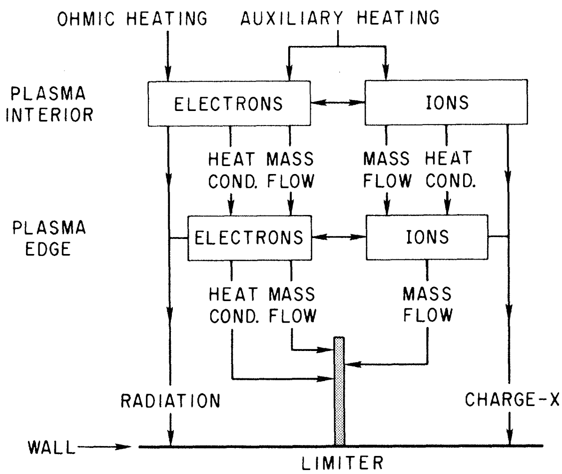

FIGURE 3. Schematic diagram of the typical flow of energy and mass in a tokamak plasma.

As the history of controlled-fusion research has so well documented, there are also 'anomalous' transport mechanisms, which proceed through cooperative effects (cf. Riviere *et al.*, this symposium). The great empirical attraction of the tokamak has been that its anomalous phenomena appear to be relatively harmless. The observed values of χ_i are usually close to classical prediction, although the accuracy of measurement typically could not exclude anomalous enhancements by factors of less than about 3. While the observed χ_e-values are clearly much enhanced relative to the classical expectation for χ_e, they are comparable with the classical χ_i. The observed particle diffusion coefficient D resembles χ_e, but its magnitude is smaller and the associated 'convective' energy loss is generally unimportant. The net result is that classical expectation for overall energy transport from the tokamak plasma is only slightly too optimistic. On the basis of the tokamak experiments, a reactor plasma such as that in figure 2 would be expected to reach ignition.

3. OHMIC-HEATING EXPERIMENTS

The presence of the tokamak plasma current serves incidentally to heat the plasma resistively to typical electron temperatures of 1–2 keV. The pattern of tokamak power flow is outlined in figure 3. The electrons in the plasma interior lose their energy by heat diffusion and convection, by radiation cooling (if there are imperfectly stripped heavy impurity ions), and by equilibration with the plasma ions; these, in turn lose their energy through conduction, convection, and diffusion by multiple charge exchange. At the edge of the tokamak plasma, the heat outflow is typically collected on a metallic ring 'limiter'.

Ohmic heating in tokamaks is most effective for high magnetic fields and small dimensions, since the current density is then maximized. The most successful device of this kind has been the Alcator A at the Massachusetts Institute of Technology (Gondhalekar *et al.* 1979), which has operated at fields up to 8 T, with $R = 0.54$ m, $a = 0.1$ m and $I \approx 200$ kA. The intense ohmic heating in Alcator A has permitted an unusually wide range of plasma densities, up to $n(0) \approx 10^{15}$ cm^{-3}, to be attained at temperatures $T_e(0) \approx T_i(0) \approx 1$ keV. As illustrated in figure 4, the experimental values of χ_i are in good agreement with classical expectation ($\chi_i^{\text{n.c.}}$). The observed χ_e, however, is far from classical: it follows an empirical scaling $\chi_e \propto \bar{n}_e^{-1}$ which has come to be known as Alcator scaling.

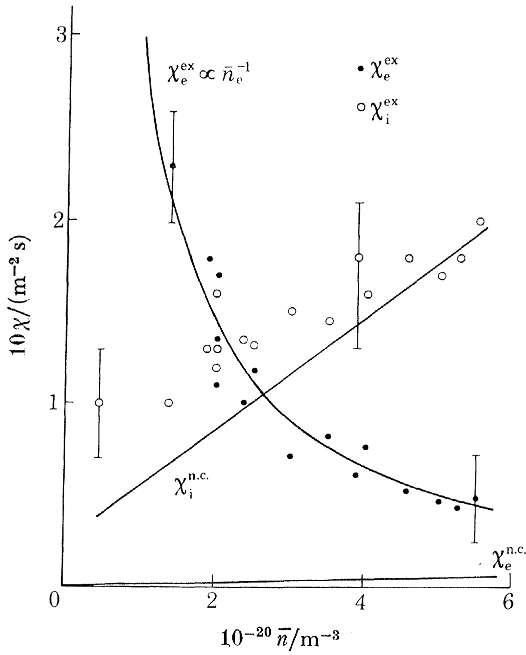

FIGURE 4. Comparison of experimental data χ_i^{ex}, χ_e^{ex} of Alcator A with theoretical classical ion thermal conductivity $\chi_i^{\text{n.c.}}$ and empirical anomalous electron conductivity $\chi_e^{\text{n.c.}} \propto \bar{n}_e^{-1}$. (A. Gondhalekar, to be published.)

Figure 5 shows that the energy confinement scaling $\tau_e \propto a^2 \bar{n}$, which would be expected from the Alcator empirical χ_e-formula when electron thermal conductivity dominates the power balance, provides a reasonably good fit to a collection of experimental data from other devices. Unfortunately, numerous other scaling laws for χ_e have also been shown to offer good empirical fits. To provide better information on the true physical parameter dependence of χ_e, as well as to bring the tokamak experiments into the temperature range of reactor interest, it is necessary to apply some form of auxiliary plasma heating.

4. Auxiliary heating experiments

The most effective non-ohmic heating method that has been applied to tokamaks to date is the injection of powerful beams of neutral atoms, which become trapped in the form of circulating superthermal ions. These ions then thermalize gradually with the bulk plasma ions (see Sweetman *et al.*, this symposium). The P.L.T. device at Princeton (figure 6) illustrates both the basic architecture of a conventional ohmic-heated tokamak and the added feature of tangential neutral-beam injection. The machine parameters are: $B_t \approx 3.5$ T, $R = 1.3$ m, $a = 0.45$ m, $I \approx 500$ kA.

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In ohmic-heating operation, the P.L.T. device is unable to exceed an ion temperature of 1 keV. Its large size gives rise to τ_E-values as high as 0.1 s (figure 5), but the $n\tau_E$-value, which is of critical interest for fusion-reactor purposes (cf. Roberts, this symposium), is slightly lower than that in the small Alcator A device.

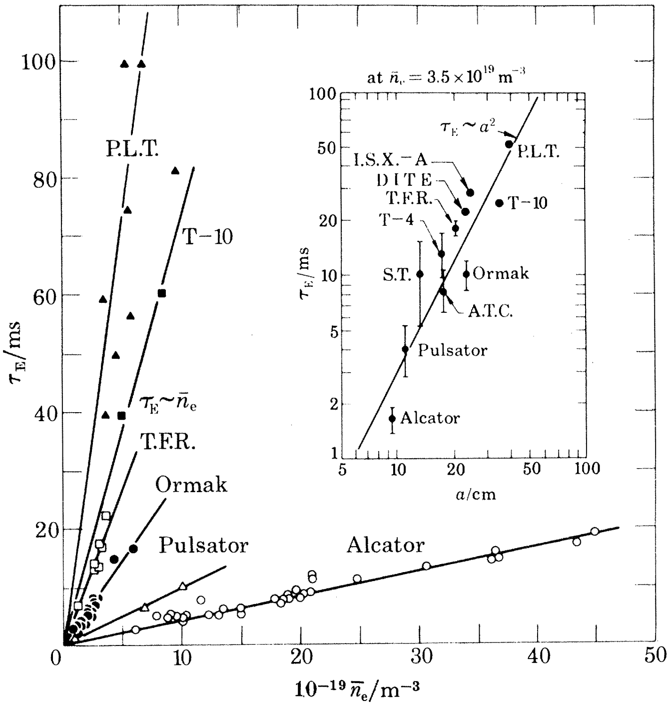

FIGURE 5. Approximate consistency of the 'Alcator scaling' $\tau_E \propto \bar{n}a^2$ with a wide range of ohmic-heated tokamak régimes.

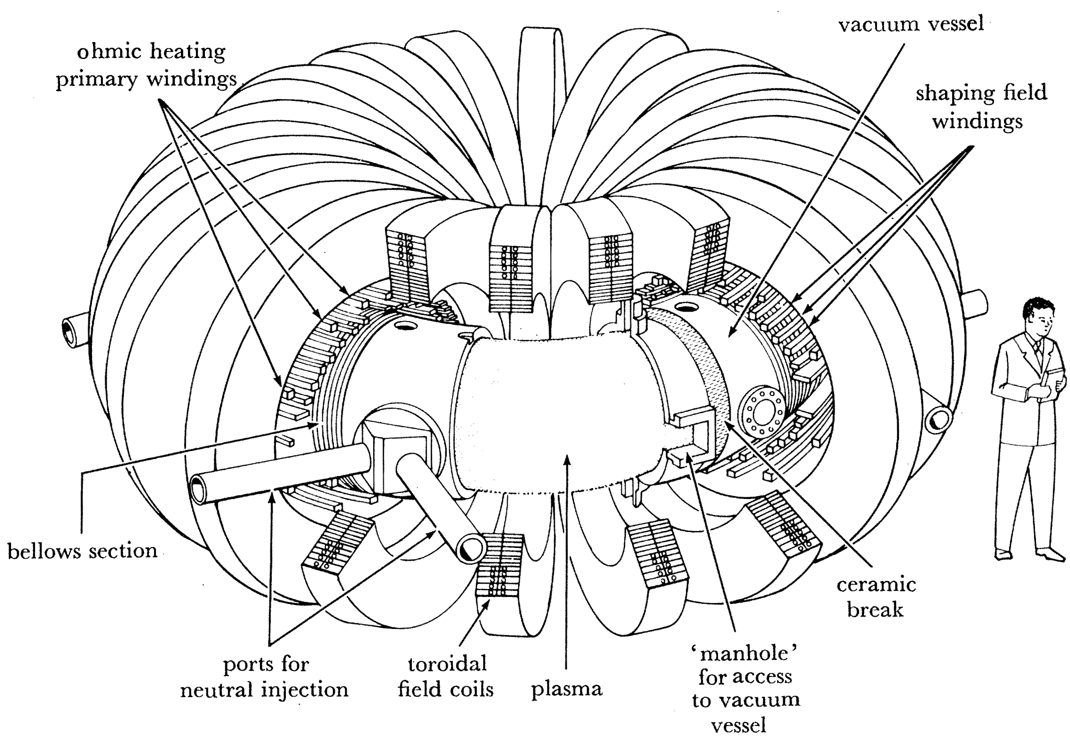

FIGURE 6. The P.L.T. device.

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The ion-temperature history for neutral-beam heating in P.L.T. (Eubank *et al.* 1979), as shown in figure 7, begins 0.4 s after initiation of ohmic heating. At 0.45 s, four tangential beams are turned on, which inject a total of 2.4 MW of deuterium atoms at 40 keV into the hydrogen plasma. The hydrogen ion temperature, as determined from the charge-exchange neutral spectrum, then rises to 6.5 keV. (Iron impurity ions, which are present with a relative density of approximately 10^{-3}, are actually seen to become hotter, as is expected from the theory of injected-ion thermalization.)

The quality of energy confinement observed under these high-temperature P.L.T. plasma conditions is of critical importance for the prospects of the tokamak reactor. In ohmic-heated tokamak plasmas, the collisionality parameter (the ratio of Rq to the scattering mean free path λ_c) had always been substantially greater than it would be under tokamak reactor conditions, so that one could not assess the deleterious effects of various micro-instabilities predicted for the low-collisionality régime. In neutral-beam-heated P.L.T. plasmas like that of figure 7, the collisionality is about the same as in a reactor, yet the ion thermal conduction continues to agree with the classical theory within the error of measurement (a factor of three to five). Even more remarkable, the electron thermal conductivity, which had been predicted to be greatly enhanced at low collisionalities by the trapped-electron micro-instability mode (cf. Riviere *et al.*, this symposium), turns out to decrease with rising T_e in the hot central region of the plasma (figure 8), relative to the standard Alcator scaling law for χ_e.

The P.L.T. confinement results thus showed that reactor-like temperatures were readily achievable by neutral-beam heating and that ignition conditions are highly likely to be attainable in a tokamak plasma no larger than that of figure 2. At this point, one last item of basic physics remained to be tested experimentally before the tokamak could be certified as a prospect with a high probability of success as a reactor: in ohmic heated tokamaks, the plasma β^*-value had never exceeded about 1% – yet a β-value of at least 5% would be desirable in a commercially attractive power reactor (cf. Roberts, this symposium).

During the past year, the I.S.X. device at the Oak Ridge National Laboratory and the T-11 at the I.V. Kurchatov Institute have been able to obtain values of β^* of about 3%, by

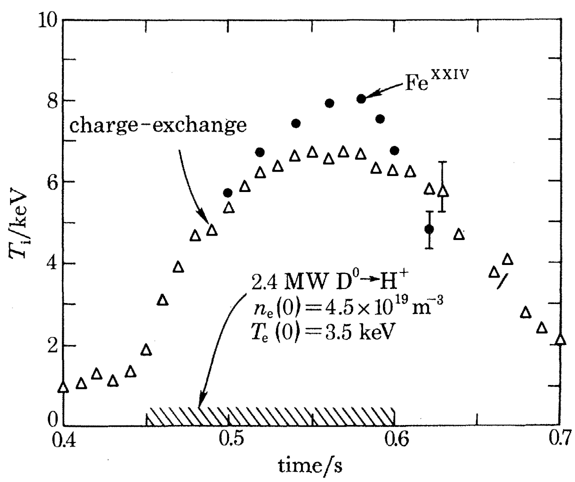

FIGURE 7. During a P.L.T. ohmic-heating discharge of approximately 1 s, neutral-beam heating is applied for 0.15 s to raise the ion temperature.

means of intense neutral-beam heating in relatively low-field tokamak régimes (less than about 1 T). These results have been particularly encouraging, because the theoretically predicted stability limits on β^* for the experimentally studied geometries ($R/a \approx 4$, $q \approx 3$, in circular minor cross section) were only of order less than about 2%. As is illustrated in figure 9, the quality of energy confinement in I.S.X. is not found to deteriorate, up to the highest β^*-values that have been attained with the available neutral-beam power. Continuing studies on I.S.X.

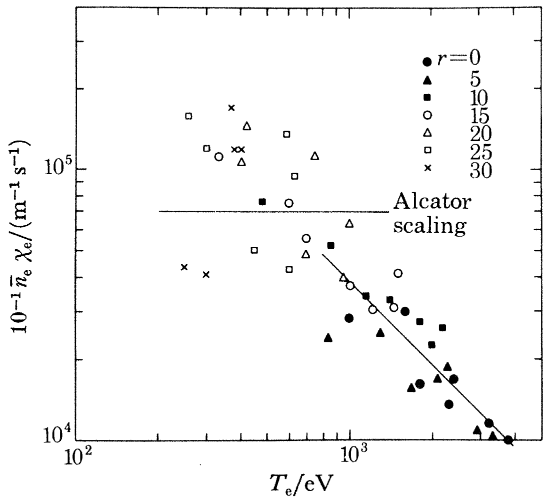

FIGURE 8. Data from P.L.T. discharge with neutral-beam heating in P.L.T. showed that the electron thermal conductivity decreases with rising T_e in the central high-temperature region.

and other devices are needed to document the ultimate β-limits in various geometries – especially for the vertically-elongated D-shaped cross section, which is believed to be of principal reactor interest – but the current outlook seems quite reassuring.

Now that the tokamak approach appears to be clearing the three critical 'physics' hurdles on the way to fusion power – heating, confinement, and stability – attention is beginning to focus

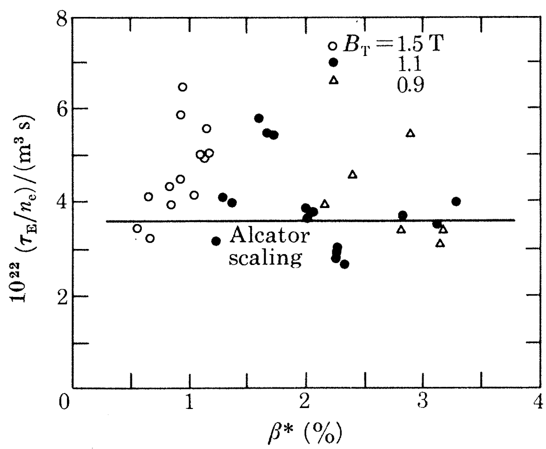

FIGURE 9. Neutral-beam heating at low B_t in I.S.X.-B has been used to explore high-β régimes. At values of β^* exceeding the expected ideal-m.h.d. stability limits (approximately 2%), energy confinement appears unimpaired. (M. Murakami, to be published.)

on a large number of important subsidiary issues that lie on the interface of physics and engineering. In the area of auxiliary heating, one concludes that neutral-beam injection will serve to provide ignition conditions in a first-generation reactor, but that plasma heating with various radio-frequency waves is likely to prove a technically more convenient approach eventually.

Currently, successful heating of the plasma electrons by waves at the electron cyclotron frequency has already been demonstrated in a number of tokamaks, notably in the T-10 device at the I.V. Kurchatov Institute. Unfortunately, plasma waves propagating at the cyclotron frequency of the plasma ions ordinarily lack the proper polarization for effective heating. This difficulty can be surmounted by heating at the cyclotron frequency of a 'minority' ion species.

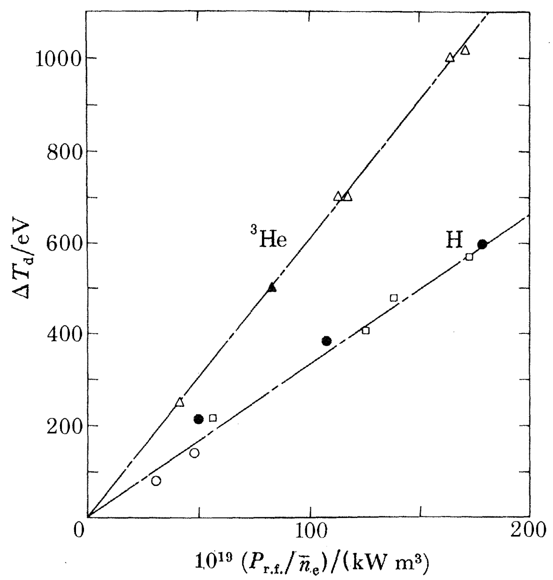

FIGURE 10. Radio-frequency heating at the ion cyclotron frequencies of various minority ions in a P.L.T. deuterium plasma produces deuteron temperature rises ΔT_d that compare favourably with those obtained by neutral-beam injection at the same power.

Recent experiments of this kind on the P.L.T. have generated minority-ion populations with energies of about 10 keV, which have thermalized with the bulk-ion population to give ΔT_i-values of the order of 1 keV. This heating process is basically similar to that of neutral-beam injection; the magnitudes of ΔT_i that have been obtained for given r.f.-input powers (figure 10) have exceeded those obtained for neutral-beam injection.

5. Plasma fuelling

The maintenance of the plasma density during a tokamak discharge does not ordinarily present a technical problem. While the confinement of the plasma particles is far from classical, it tends to be better than the confinement of the plasma energy content; furthermore, when plasma particles arrive at the outer limiter they are typically reinjected as neutral atoms (with energies in the 3–30 eV range) into the plasma edge region. This 'recycling' process maintains the plasma density at a level determined by the gas initially filling the vacuum chamber and the surface chemistry of the vacuum walls. When exceptionally high plasma densities are desired, a pulsed gas value can be used to inject additional particles during the discharge.

In a tokamak-reactor plasma, the maintenance of the desired plasma density is of an altogether different character: the principal objective becomes to flush continuously the helium ash of the D–T reaction from the central plasma region, and to replace it with fresh D–T fuel (cf. Roberts, Hancox, this symposium). In this context, one may be thankful that the tokamak particle transport has *not* turned out to be classical, since the outward diffusion of central-region ions – particularly minority ions such as helium, with nuclear charge $Z > 1$ – would then be too slow to allow an ignited state to maintain itself. The actual observed 'anomalous' transport rate appears to be of a convenient magnitude for purposes of ash removal.

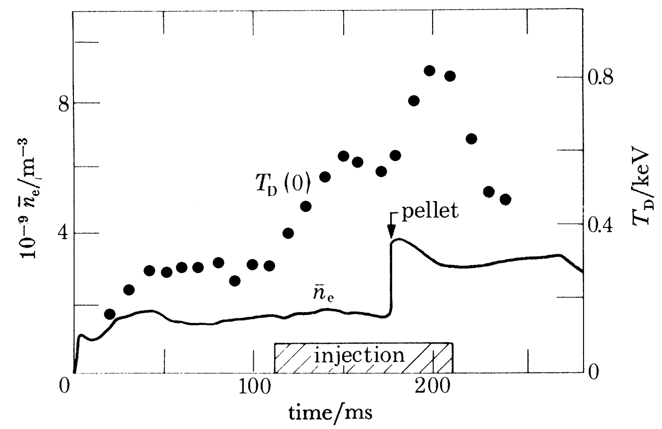

Figure 11. The I.S.X.-B plasma has been 'refuelled' in a reactor-relevant manner by injection of a 1 mm pellet of frozen hydrogen, travelling at approximately 10^3 m/s. The pellet was injected into a deuterium plasma with $I_p = 120$ kA, $B_t = 1.15$ T during heating by a 0.5 MW hydrogen neutral beam. The central ion temperature $T_D(0)$ is maintained after pellet injection by a sharp increase in the thermalization rate of the energetic ion population. (S. L. Milora, to be published.)

The injection of fresh fuel into a toroidal reactor has long been envisaged in terms of high-velocity D–T ice pellets, which are supposed to evaporate and turn into plasma after penetrating into the plasma interior. There has, however, been considerable uncertainty about both the pellet ablation process, and the possible impact of a highly localized density perturbation on the quality of tokamak confinement. These concerns have recently been alleviated considerably by the successful injection into I.S.X. of millimetre-sized solid-hydrogen pellets moving at 10^3 m/s. In the example given in figure 11, even though the mean plasma density $\bar{n}$ was doubled by the pellet injection process – a far larger perturbation than would be envisaged during reactor plasma fuelling – the central ion temperature of the neutral-beam-heated I.S.X. plasma was scarcely reduced and the gross energy content of the discharge maintained itself throughout the refuelling process. The principal remaining obstacle to reactor fuelling appears to be a technical one: the pellet velocity must be increased towards 10^4 m/s.

6. Impurity control

In current tokamak experiments, the most serious impurities are not fusion-reaction products but metallic ions evolved from the limiter and vacuum-chamber wall. Even though the plasma edge is relatively cold (typically a few hundred electronvolts or less) and overheating of exposed surfaces is not yet a critical problem, there are various sputtering and arcing processes that can easily produce relative metallic ion densities of 10^{-3} or more within the plasma. Since such ions tend to remain imperfectly stripped even at electron temperatures in the multi-kiloelectronvolt range, they constitute a highly effective mechanism for radiation cooling of the central plasma region.

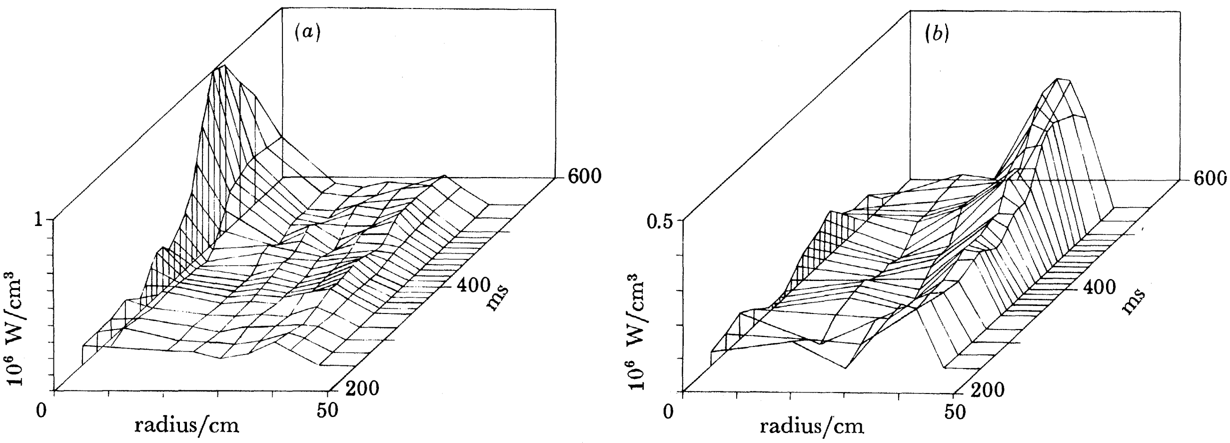

FIGURE 12. Bolometer measurement of radiated power from P.L.T. during neutral-beam heating for (*a*) steel limiters (0.4 M.W. hydrogen counter beam injection into deuterium for 0.4–0.5 s) and (*b*) carbon limiters (1.4 M.W. hydogren c.o. plus counter beam injection into deuterium for 0.4–0.5 s).

Figure 12*a* illustrates the radiation (mostly in the far ultraviolet region) from a neutral-beam-heated P.L.T. plasma with a steel limiter. In the central region, almost the entire beam input power is radiated away, and the plasma temperature then tends to be clamped well below the level shown in figure 7. By substituting graphite limiters (figure 12*b*), one can avoid the radiation problem, since carbon ions are perfectly stripped on the plasma interior. For reactors, however, it appears that much more sophisticated limiter materials and designs will be required – or that a completely different approach must be taken: the magnetic divertor.

As will be described in detail by Paul *et al.* (this symposium) the idea of the divertor is to define the boundary of the plasma confinement region by means of a magnetic separatrix, beyond which the edge-plasma exhaust can be guided into a separate divertor chamber for pumping. There are two principal forms of tokamak divertor: the 'bundle divertor' of Paul *et al.* (this symposium) and the 'poloidal-field divertor', which is currently under study in the P.D.X. facility at Princeton. Initial P.D.X. experiments, in ohmically heated plasmas, have demonstrated the effectiveness of the poloidal divertor in guiding plasma particles and energy into the divertor chamber. A neutral-beam-injection system, with an ultimate capability of 6 MW, has just begun to become operational.

7. Conclusion

Traditionally, three major obstacles have stood in the way of magnetic fusion power: insufficiently powerful plasma heating, excessive plasma transport, and defective magnetohydrodynamic stability. The tokamak research programme has addressed these obstacles and has now, for the first time, provided experimental evidence that they could be overcome simultaneously in a magnetic fusion reactor of the tokamak type (figure 13).

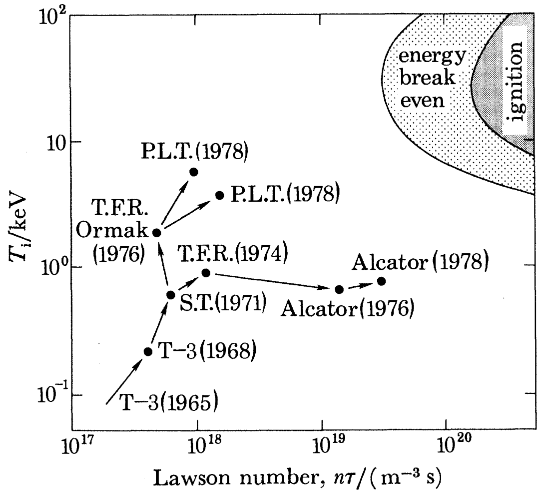

FIGURE 13. Progress of tokamak experiments towards parameter values of reactor interest.

The recent advent of reactor design efforts aimed at proposals for next-generation machines of the tokamak type has brought to light a whole new generation of less fundamental, but nonetheless very important problems at the interface of physics and engineering, for which practical solutions need to be found. The problem of plasma impurity control is a prime example.

The solution of these practical problems seems likely to call for a new round of inventions and advances in physical understanding at a level of creativity that will be at least comparable with past performance, and at a level of expenditure that will be considerably greater. The prospects of ultimate success, however, seem increasingly favourable.

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Discussion

I. J. Spalding (*U.K.A.E.A. Culham Laboratory, Abingdon, Oxfordshire OX*14 3*DB*, *U.K.*). The author discussed pellet-injection experiments at velocities of 10^3 m/s into I.S.X. which suggest that velocities of *ca.* 5×10^3 m/s might be required for refuelling a D–T toroidal reactor. Does the author know of any satisfactory alternative to refuelling conceptual tokamak reactors by pellet injections?

H. P. Furth. Does Dr Spalding have other suggestions?

I. J. Spalding. No, I am interested in the physical problems associated with accelerating cryogenic pellets to such high velocities.

H. P. Furth. Perhaps I should qualify my suggestion that high velocities will be desirable for pellet injection into a tokamak reactor. There are reactor-studies experts who believe that pellet velocities of about 2×10^3 m/s will prove to be adequate. In this case, the fuel will be deposited near the plasma edge and will then mix by diffusion into the central plasma.

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