

Recent progress in tokamak experiments

Plasmas confined by magnetic fields in a toroidal geometry can reach confinement times, densities and temperatures approaching those needed for producing useful fusion energy.

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At the 1968 conference on controlled fusion in Novosibirsk a group from Kurchatov, USSR, led by L. A. Artsimovitch, presented convincing evidence that one conceptually simple method for confining plasmas in a ring showed great promise for future developments. The name for their early machine, the tokamak, has now become the generic name for all such devices.¹ The Russian successes led to a rapid expansion of research with tokamaks, so that while in 1968 there were only nine of them, all in the USSR, there are now more than a hundred; they are in the USSR, the US, Europe, Japan, and elsewhere.

In a tokamak the plasma is confined in a toroidal vacuum chamber by a strong magnetic field running the long way around the torus together with a weaker field produced by a current flowing through the ring of plasma itself. A small vertical magnetic field serves to maintain the position of the plasma ring against inductive and kinetic expansion forces. The plasma current is induced by an external magnetic field, the plasma acting as a single-turn secondary winding of a transformer, and heats the plasma resistively. This ohmic heating is the simplest and most efficient method to create a relatively hot and dense plasma, but it is generally thought necessary to provide supplementary heating to bring the plasma to thermonuclear temperatures.

In this article we briefly review the status of tokamak experiments. Many of the effects of impurities and macroscopic instabilities that limited the performance of earlier tokamaks have been reduced in

recent studies. Neutral-beam injection and other auxiliary heating methods have also made considerable strides recently. Several parameters can be used to measure the performance of a fusion device: ion temperature T_i , confinement time τ , and plasma density, n . (The product $n\tau$ is called the Lawson number.) Very recently new records have been set for these parameters² (see PHYSICS TODAY, November 1978, page 17):

► $\tau = 100$ msec in the Princeton Large Torus

► $n\tau = 3 \cdot 10^{13}$ cm⁻³ sec in Alcator at MIT

► $T_i = 6.5$ keV $= 7 \cdot 10^7$ K in PLT with neutral-beam injection.

The values are approaching those needed for fusion reactors ($n\tau \approx 10^{14}$ cm⁻³sec, $T_i \approx 10$ keV). A particularly encouraging result is the one with neutral-beam injection in PLT, which has shown that ion and electron energy confinement did not deteriorate at low collisionality, contrary to predictions of the trapped-particle instability theory. This result combined with those obtained from many other tokamaks, extrapolates favorably to larger tokamak experiments and to a tokamak fusion reactor.

Supplementary heating

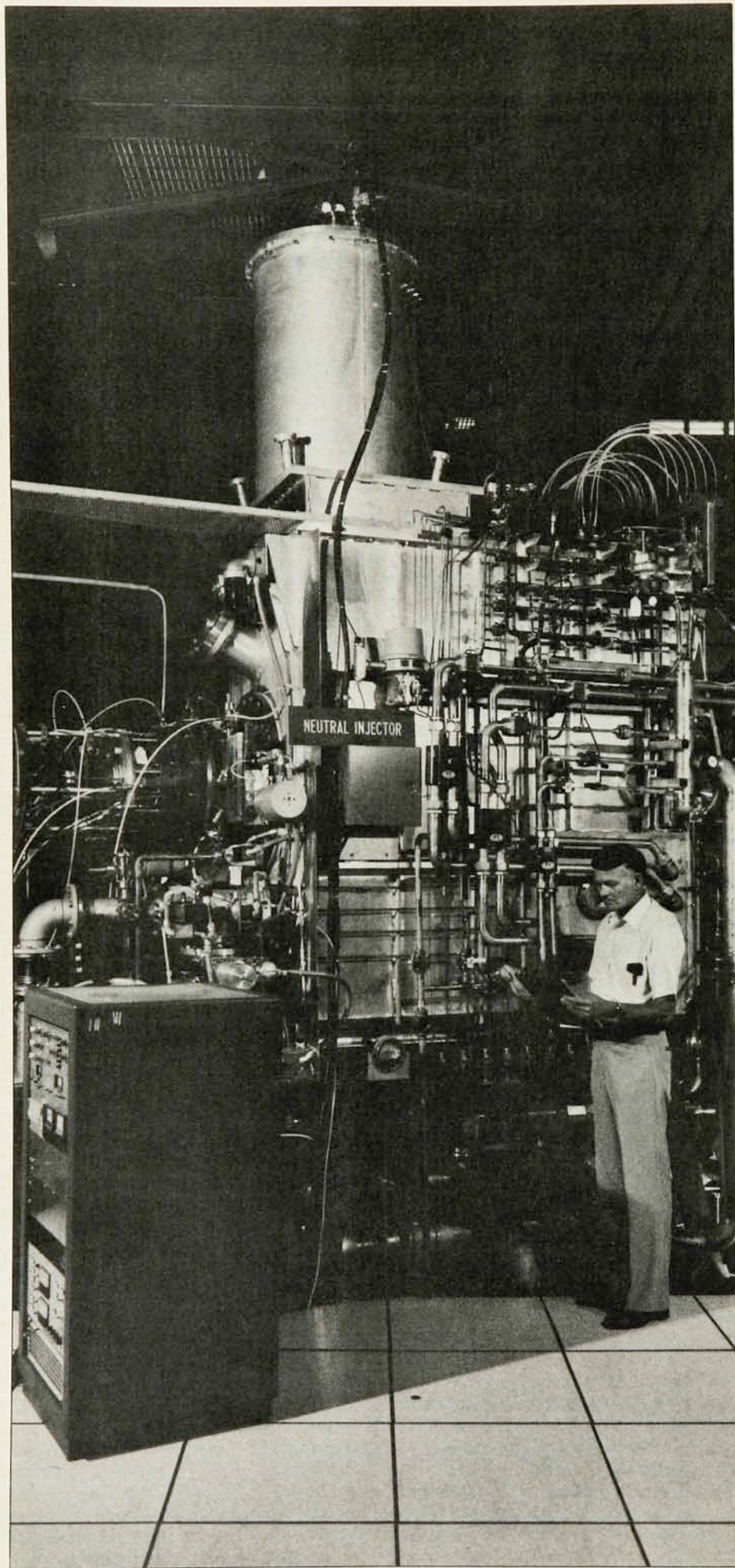
The heat supplied to the plasma must compensate for the energy lost from the plasma by radiation and by conduction and also raise its temperature to the desired level. The amount of energy that can be supplied by ohmic heating is limited by the resistivity of the plasma, which falls at high temperatures, and the current density that can be supported by the plasma without becoming unstable. Calculations indicate that classical ohmic heating alone will not be able to drive temperatures close to the values required for ignition, particularly in reactor-size

devices with the desired plasma densities of several times 10^{14} cm⁻³.

Accepting the premise that some form of supplementary heating will be required to reach reactor conditions, what are the options? Heating methods which have been proposed and tested on tokamaks so far are compressional, radio frequency and neutral-beam heating. All have produced significant ion or electron temperature increases and sometimes both. Compressional heating was studied in considerable detail with the Adiabatic Toroidal Compression (ATC) device at Princeton: An increase in the vertical (equilibrium) magnetic field compressed the plasma in both major and minor radius. This study showed that, provided the compression time is sufficiently short compared to plasma-energy loss times, adiabatic compression increases the temperature, scaling as $C^{4/3}$, where C is the compression ratio. Large compression ratios, however, (ATC had a factor of 2.3) are expensive in terms of the required ratio of the volume of the toroidal magnetic field to that of the plasma. It appears that the future of compressional heating in tokamaks lies in providing a modest compressional boost in density and temperature to cross an ignition or breakeven threshold that has been approached by other means.

Several successful tokamak auxiliary heating studies have been carried out over the past few years using radio frequencies at the ion-cyclotron (tens of MHz) and lower-hybrid frequencies (hundreds of MHz). Recent development of high-power (about 200 kW) microwave devices operating at about 30 GHz also makes electron-cyclotron heating a possible contender. The ease of production and transport of multimewatts of rf power in the ion-cyclotron and lower-hybrid frequency ranges make these very ap-

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peeling heating processes. This is particularly true for lower-hybrid frequencies where modestly sized wave guides and coupling structures can be used effectively. Interpretation of the results of heating experiments done at these frequencies has been complicated because the tokamaks used have been small compared to the size of the suprathermal ion orbits generated by the rf fields. While it is beyond the scope of this article to give a detailed appraisal of rf-heating methods, it seems clear that rf heating, under the conditions of confinement and power necessary to enable direct extrapolation of the results to reactor-sized devices, has yet to be achieved. Several such experiments are in progress or planned, however. For ion-cyclotron heating the most notable of these are now in progress on the Princeton Large Torus, with up to 5 MW of generator power, and on TFTR-600 at Fontenay - aux - Roses. Lower - hybrid heating is being planned for PLT and the Frascati tokamak (Italy), at power levels above about 1 MW, and for Alcator C (MIT) at about 3 MW.

At the present time, the most successful, best understood and most widely used auxiliary heating method is that achieved by employing neutral beams of atomic hydrogen isotopes. These are produced by the acceleration of ions from large conventional ion sources; the accelerated ions are then neutralized by electron attachment from the unaccelerated atoms that flow along with the ions from the ion source. The technology of producing intense neutral beams is evolving rapidly. Systems are now available that produce megawatts of neutral power at particle energies of 50 to 100 keV, which would have been hard to believe possible a few years ago. These successes do not mean, however, that neutral-beam heating meets all the requirements for an ideal system. There are recognized problems with effecting penetration of the beam into plasmas of large, high-density tokamak systems of the future whose solutions are not yet clear. For example, because the cross section for electron attachment falls rapidly when the relative speed of the encounter is greater than 2×10^8 cm/sec, it is difficult to produce intense neutral beams with energies larger than about 100 keV. One alternative is to use negative-ion sources, so that the neutral beam is formed by electron stripping rather than attachment. Neutral-beam heating systems are still very complex and expensive

A neutral-beam injector developed by the Plasma Technology Section at Oak Ridge National Laboratory and used with ISX-B at ORNL. Four similar machines are used with the Princeton Large Torus at the Princeton Plasma Physics Laboratory. The device produces intense beams of neutral hydrogen or deuterium for injection into the hot plasma of a tokamak, off to the right of the photograph. Figure 1

(more than \$1 per watt of delivered power), and there may well be other heating methods, that are equally effective but cheaper.

Macroscopic stability

The first step of any magnetic confinement experiment is to obtain a macroscopically stable plasma. In a tokamak the plasma current plays a central role in the stability as well as the confinement and heating of the plasma. The poloidal magnetic field, B_p , produced by the plasma current is superposed on the toroidal field, B_ϕ , produced by an external current; the resulting field lines are helices. The field configuration can be characterized by the "safety factor"

$$q(r) = rB_\phi / RB_p$$

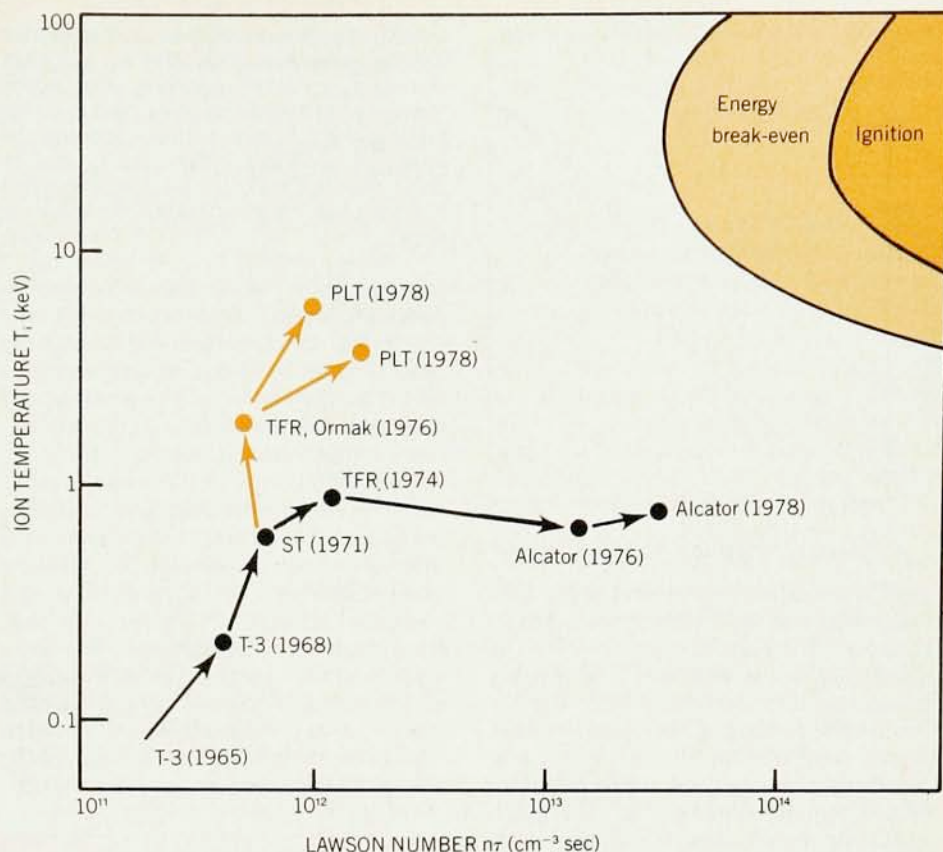
where R is the major radius of the torus and r is the distance from the center of the plasma column. The safety factor has a simple geometrical interpretation: A magnetic field line makes q transits in the toroidal direction while making one transit in the poloidal direction.

The helical field configuration stabilizes the plasma against the simple "sausage instability," in which the plasma column is pinched off by its own magnetic field. The plasma can, however, develop other instabilities that reduce the effectiveness of the confinement. These instabilities are customarily divided into two classes: those derivable from a macroscopic theory, magnetohydrodynamics, and those that can only be derived from a microscopic, kinetic, theory. The latter are not yet as well understood as the former. There are three major forms of macroscopic disturbances observed in tokamak plasmas:

- ▶ disruptive instabilities
- ▶ Mirnov oscillations
- ▶ internal disruptions

The disruptive instability is an abrupt and generally unpredictable expansion of the plasma column accompanied by a large negative spike of the loop voltage around the plasma column, that is, of the voltage driving the plasma current. Mild disruptions may occur repeatedly in a given discharge, whereas a major disruption generally terminates the plasma current. This instability limits both the current and plasma density attainable for given discharge conditions. Below these limits disruptions do not occur spontaneously, although they can be induced by appropriate experimental conditions.

Under normal tokamak operating conditions without disruptive instabilities, small oscillating helical perturbations in the poloidal magnetic field can be detected at the plasma edge. These are the Mirnov oscillations, whose frequencies are in the range of 10–30 kHz. As the Mirnov oscillations become large, the energy and particle confinement time drops, indicating that the oscillations increase the rate at which energy and particles are



Progress of tokamak experiments. We plot the ion temperatures and Lawson numbers achieved in recent experiments. Black points indicate values achieved with purely ohmic heating; points in color are for experiments with auxiliary heating by neutral-beam injection. Figure 2

transported from the plasma column to the walls of the chamber.

Finally, the internal disruptions were first observed in measurements of soft x-ray emissions, and later in many other diagnostic signals. They appear as relaxation (or sawtooth) oscillations in the plasma temperature.³ (X rays in the 1 to 20 keV range serve as useful indicators of the electron temperatures, as discussed in the article on diagnostics by Charles B. Wharton on page 52). The effects of the internal disruptions are concentrated near the center of the discharge, and their effects on overall confinement are usually modest.

According to magnetohydrodynamic theory the plasma can be unstable to helical perturbations ("kinks") whose pitch is the same as that of the magnetic field. This can occur when the safety factor is a simple ratio,

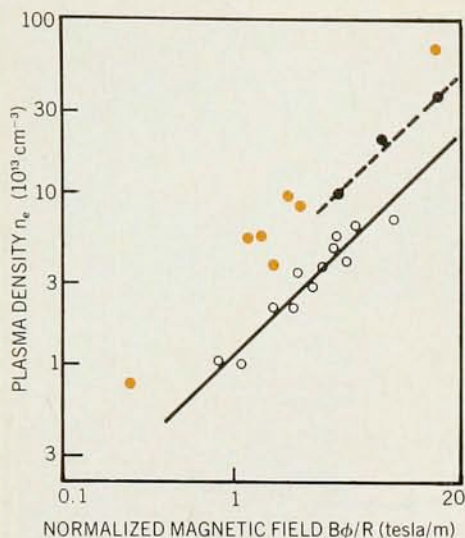
$$q(r) = m/n$$

If the safety factor drops below unity for any value of r the plasma is unstable. (The current for which $q = 1$ at the plasma edge is called the Kruskal-Shafranov limit.) Even if $q > 1$ everywhere, instabilities of higher helicity can set in, provided the resonance point falls in a region of the plasma that is poorly conducting. In that case, the magnetic field lines can break and reconnect to form magnetic islands near the resonance point. For this reason, such instabilities are often called "resistive" or "tearing" modes. Because

of rapid plasma transport along the field lines, the main effect of the island is to short circuit transport across the island, resulting in deterioration of plasma confinement.

Models based on the non-linear behavior of the tearing modes give a fairly good description of the observed features of the Mirnov oscillations and the internal disruptions.⁴ Such models also account for many features of the disruptive instabilities; however, computational and experimental difficulties make the identification less certain in this case. It now appears that the growth rate of a tearing mode (particularly the $m = 2, n = 1$ mode responsible for Mirnov oscillations) is large when the current density peaks sharply near the resonance point.

Various experimental techniques have been developed to keep the current channel in the plasma broad to reduce the growth rate of the tearing modes. These efforts have led to a significant expansion of the tokamak operating regime and a substantial improvement of plasma confinement. The increase of attainable maximum plasma density is illustrated in figure 3 by comparing the maximum densities reported for various tokamaks before mid-1970's and those attained in more recent tokamaks. A useful parameter⁵ against which to plot the maximum line-average density attainable without disruptive instabilities is the ratio of toroidal field B_ϕ to major radius R . As the graph shows, most of the points for steady



Scaling of plasma density with tokamak size and magnetic field. We plot the maximum line-averaged electron density n_e as a function of the ratio of toroidal magnetic field B_ϕ to the major radius R of the tokamak. Note that the steady-fill conditions (open circles) produce lower densities than the gas-puff techniques (solid circles). Recent results are shown in color. Figure 3

inserted into the vacuum chamber to prevent the plasma column from striking the walls. These atoms are not fully ionized even at the high temperatures found at the center of the plasma, and they can therefore cause significant radiation loss from the plasma core.

Controlling impurities in plasmas has received much attention in recent tokamaks, and these efforts have clearly paid off. Low-power discharge cleaning (sometimes called "Taylor discharge cleaning," after R. J. Taylor of UCLA, who originated the concept) has been employed to condition the wall surfaces and to pump away low-Z impurities. The low power of the discharge is intended to dissociate hydrogen while minimizing dissociation of water vapor and hydro-

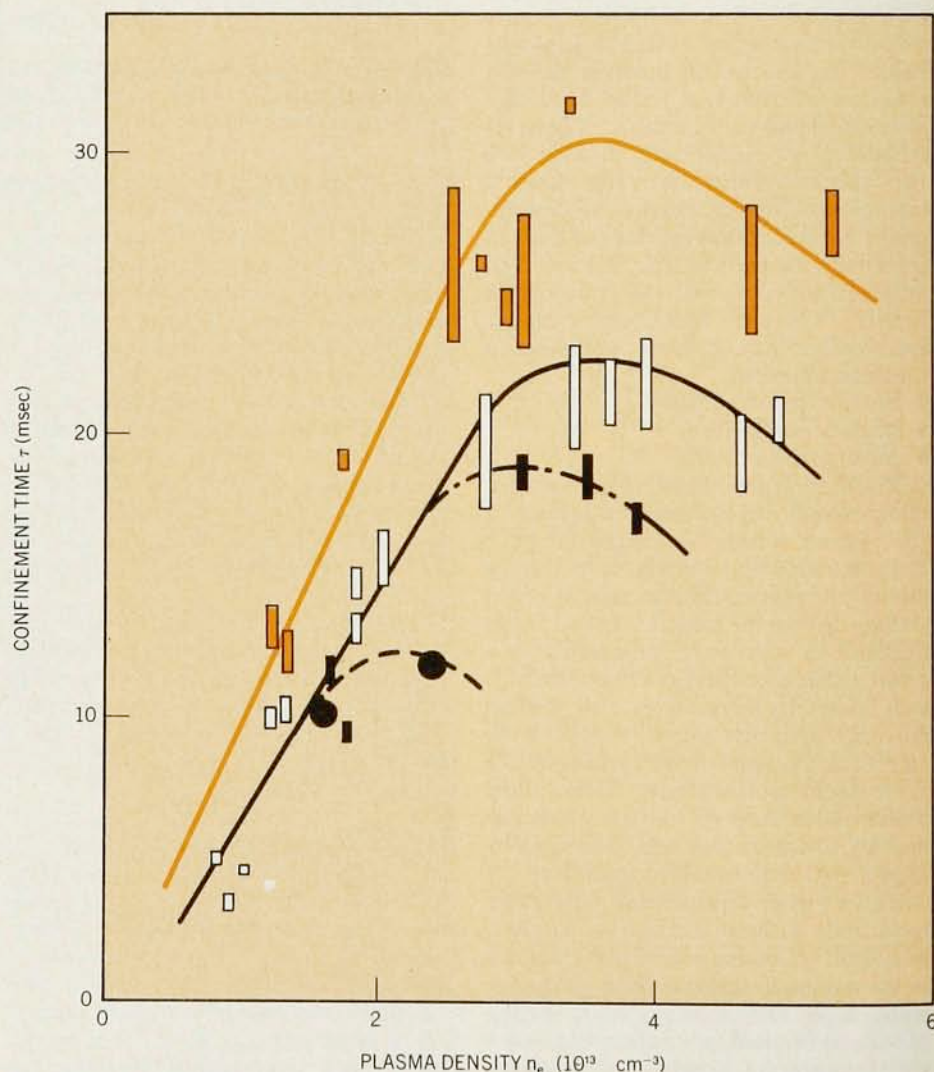
carbons. Many groups also periodically deposit a thin film of titanium to bury impurities adsorbed on the wall and to enhance the adsorption of new particles emerging from the edge of the plasma (a procedure called "titanium gettering"). The high-Z impurities have been reduced from the discharge volume by selecting low-Z limiter materials such as carbon or stainless steel.

Another technique for enhancing the stability of high-density plasmas, the "gas puffing" mentioned above, is being used with many tokamak devices. It involves injecting additional gas into the vacuum chamber to fuel the plasma after a discharge has been started in a low-density gas. The edge cooling that accompanies gas puffing appears to be effective in reducing sputtering and plasma-limiter interaction. Some tokamak groups have taken more active measures for controlling impurities. These include the poloidal divertors in DIVA and in Russia's T-12 in Kurchatov and the bundle director in DITE in the United Kingdom. Because of these successful measures, the effective charge obtained in recent toka-

gas-fill conditions lie on a straight line, suggesting a model in which the current-channel shrinkage is countered by the ohmic heating power, which is roughly proportional to the ratio B_ϕ/R . Experiments with Alcator, at MIT, and Pulsator, at the Institute for Plasma Physics in Garching, Germany, have shown that injecting supplemental gas in short puffs could produce densities significantly above those that can be achieved with steady-fill techniques. Recently experimenters have produced even higher densities by carefully programming the gas injections and maintaining as pure a hydrogen plasma as possible. An additional measure to prevent current-channel shrinkage has been taken in the Japanese tokamak JIPPT-2 at Nagoya and at Alcator by combining the gas injection with a positive ramp in the discharge current to produce "skin heating" in the plasma. Some of these techniques have also allowed other tokamak groups to operate successfully in regimes with $q(a)$ in the range 2.0–2.5. The DIVA tokamak at the Japan Atomic Energy Research Institute obtained stable operation at $q = 1.3$.

Plasma purity

Impurities can change almost any characteristics of the plasma including the macroscopic stability we have discussed. The effect of an impurity depends on its atomic number, Z . Typical low- Z impurities are oxygen and carbon, which mostly come from desorption from the wall of the plasma chamber, and they tend to be relatively abundant in the plasma. Their chief effect is to change the effective, or average, ionic charge of the plasma, thus affecting its resistance and its transport properties. Because these impurities are fully ionized at the center of the plasma column, the radiation loss they cause tends to arise mostly at the edge of the plasma. The resultant cooling leads to a shrinkage of the current channel, which in turn affects the macroscopic stability discussed above. High- Z impurities, such as tungsten and molybdenum, generally come from the limiters



Gross energy-confinement time as a function of line-averaged density in deuterium (color) and hydrogen discharges in ISX-A. The size of the bars indicates the uncertainty of the ion-temperature profile determinations. The hydrogen data are for magnetic fields of 15 (white) 12.5 (black) and 8 (black dots) kilogauss. The deuterium points were measured at 12.5 kG. Figure 4

maks is close to unity, and radiated power loss relative to input power is typically $1/3$.

From the point of view of energy transport in tokamaks it is convenient to regard the plasma as consisting of an interior region and the plasma edge. At the plasma edge, the dominant loss mechanisms are those related to atomic processes, such as impurity radiation, charge exchange and ionization losses. Recent tokamak experiments have been reasonably successful in confining the atomic effects into a narrow outer region by effective measures for the control of impurities. Energy flow into the outer region is principally through transport losses consisting of heat conduction and convection from the plasma interior. In a macroscopically stable discharge, these transport losses are caused by classical Coulomb collisions (neoclassical transport) and microscopic instabilities (anomalous transport).

Energy confinement

In the neoclassical theory,⁶ the collisional transport processes in a uniform magnetic field are modified by particle orbits specific to the nonuniform tokamak field geometry. The toroidal field increases toward the center of the ring, so that the helical field lines bunch together near the center of the torus and spread apart near the outside (away from the "doughnut hole"). Such a field configuration produces shallow magnetic wells for the plasma particles. Particles that have a sufficiently large component of the velocity along the magnetic field become "passing particles" and circulate relatively freely around the torus, while particles whose velocity is more nearly transverse to the field are "trapped particles," oscillating back and forth between regions of high magnetic field. The ability of the trapped particles to experience a number of reflections between the magnetic mirrors is determined by the collisionality parameter, ν^* , which is a normalized measure of the collision frequency. For small ν^* the plasma is said to be "collisionless," and particles are trapped in the mirror wells for many reflections (for that reason $\nu^* \ll 1$ is also called the "trapped-particle" regime). The other extreme ($\nu^* \gg 1$) is the collisional regime, where the mean free path of particles is small compared to the circumference of the plasma ring. The latter regime dominates at a low temperatures and high densities; as the temperature increases or the density decreases, ν^* generally decreases.

A key question is how close experimentally observed values of ion and electron energy transport are to the values predicted by neoclassical theory. Many tokamak experiments have studied ion-energy losses in the intermediate regime (ν^* in the vicinity of 1), and their results generally lie within a factor of 3–5 of the neoclassical theory, and some lie within a

factor of 2 of the theory. The collisional regime (ν^* up to 8) has been explored in Alcator with densities up to 10^{15} cm^{-3} , and the results for ion-energy losses are, again, in reasonable agreement with the neoclassical theory. While the observed ion-energy transport is consistent with the neoclassical theory, this is not true of electron-energy transport. The neoclassical theory predicts that energy loss via electrons should be much less significant than thermal conduction due to ions. In practice, the opposite has been seen in most tokamak experiments at modest densities. Experimentally observed electron heat conduction losses are larger than the neoclassical predictions by a factor ranging from 10 to 500.

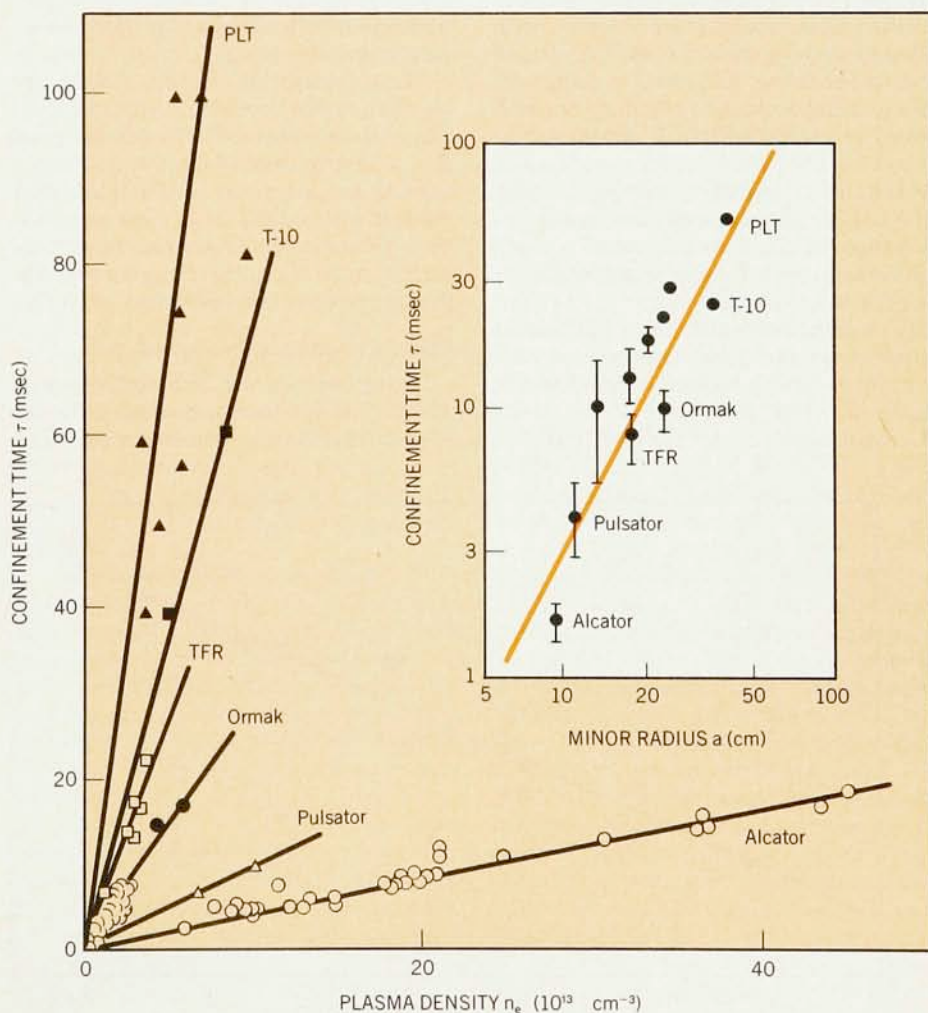
The transport of energy and particles out of the plasma column can be enhanced by turbulence caused by a variety of fine-grained (microscopic) plasma instabilities. Particularly dangerous instabilities that are theoretically predicted are the so called "trapped particle modes."⁷ These arise from small fluctuations of the electric fields in the plasma that are amplified due to the presence of trapped particles in the collisionless re-

gime, and they have been predicted to lead to virulent mass and energy diffusion out of the plasma column. These modes have not yet been identified in experiments.

Scaling laws

In view of the theoretical uncertainties involved in making reliable predictions of energy loss from plasmas at conditions approaching those necessary for fusion reactions, tokamak physicists have turned to empirical scaling laws to extrapolate from established experimental data to as yet unexplored regimes. Because the ranges of parameter variation available on individual tokamaks are limited, derivation of scaling laws also involves comparison of the data from various tokamaks. Such scaling laws are particularly important with regard to anomalous electron transport, where the underlying physical mechanisms have not yet been identified. The most interesting scaling laws are those that relate electron-energy confinement time to parameters such as the density, the temperature, or the size of the apparatus.

Recent results of tokamak experiments



Scaling of confinement time with plasma density and tokamak size. The main part of the graph shows the gross energy-confinement time as a function of electron density for several devices. The inset shows the confinement time as a function of the minor radius of the tokamak. We plot data for plasma densities of $3.5 \times 10^{13} \text{ cm}^{-3}$ as interpolated from actual experimental results. The straight line on the log-log plot indicates that confinement time scales as a^2 . Figure 5

have proven very useful in establishing such scaling laws. As an example, the data shown in figure 4 come from an experiment with the Impurity-Study experiment, ISX-A at Oak Ridge, in which the plasma density was varied systematically; the heating was entirely ohmic.⁸ The uppermost curve is for deuterium; the lower curves are for hydrogen, and each curve represents data for a different magnetic field. The gross energy-confinement time for ions plus electrons, τ , increases linearly with density, but then reaches a maximum and perhaps even decreases. The behavior of the ion temperature in this experiment was, as usual, in agreement within a factor of 2 with the neoclassical theoretical prediction. Given this approximately neoclassical ion-energy confinement one can calculate that the electron-energy confinement time continues to rise approximately linearly with density. Although the electron-energy time, τ_e scales with density, the overall confinement is limited by the (essentially neoclassical) confinement of ions. The Alcator group at MIT observed similar behavior of the gross energy-confinement time evaluated at the plasma center. The data indicate that electron-confinement time scales linearly with average density up to $5 \times 10^{14} \text{ cm}^{-3}$. In the forthcoming Alcator-C experiment, the increased value of available toroidal fields (up to 120 kG) will permit an extension of the data to larger densities and presumably longer confinement times. The experiments will attempt to explore a range of Lawson numbers $n\tau$ approaching $10^{14} \text{ cm}^{-3} \text{ sec}$, corresponding to conditions near ignition.

The results of experiments at a large number of tokamaks^{2,9} and with a wide variety of other parameters are shown in figure 5. The main part of the graph demonstrates that, for the most part, the

gross energy-confinement time scales as the plasma density, with, of course, different proportionalities for the different devices. The linear dependence of the energy-confinement time is the clearest discernible scaling for tokamak plasmas. Other scaling laws, depending on other variables (such as temperature), are harder to determine. In purely ohmically heated plasmas, for example, there is an implicit dependence of temperature on plasma current which disguises the actual temperature dependence. Independent variation of the temperature can be obtained only with substantial supplementary heating, and such studies have only recently begun.

The inset of figure 5 shows the scaling for devices of various sizes. The slope of the line on the log-log plot indicates that the confinement time is proportional to the square of the minor radius of the tokamak. Because most of the data in the figure are from situations in which the electron-energy losses dominate over other processes, one can conclude¹⁰ that the electron-energy confinement time scales as $n_e a^2$. The a^2 scaling is, in fact, what would be expected if the dominant loss mechanisms are turbulent processes, which tend to scale as a^2 for short-wavelength perturbations.

These empirical scaling results for electron losses combined with the neoclassical behavior of ion losses suggest that we may ultimately be able to produce conditions for igniting fusion in tokamak devices with small—or at least practicable—values of the minor radius, a . Recent results from the Princeton Large Torus reinforce this optimism.

Plasma heating by neutral beams

These recent PLT experiments were able for the first time to produce sufficient plasma heating to produce the collision-

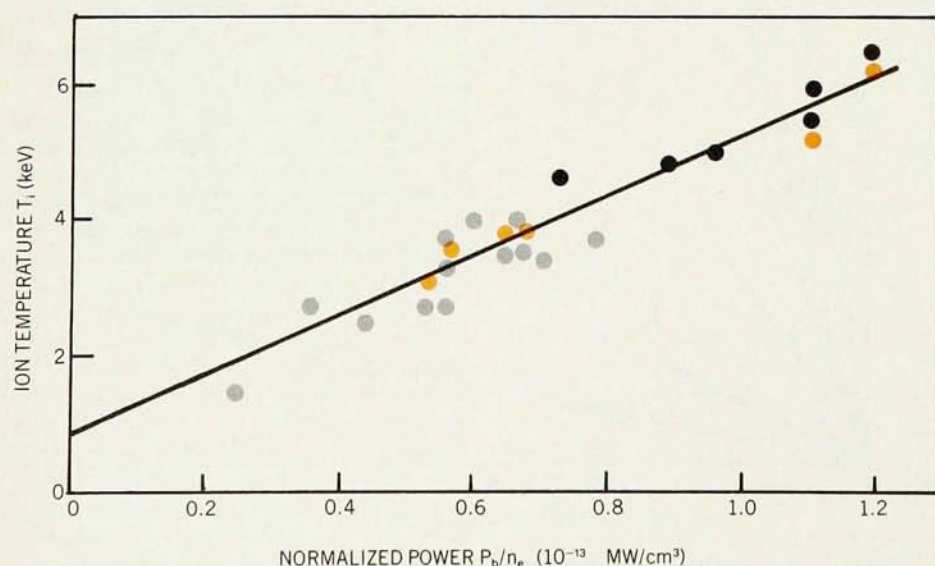
less regime required for an operating fusion reactor and observed no enhanced transport losses of either ions or electrons. Combined with results from other tokamaks,¹¹ this result extrapolates favorably to future experiments and fusion reactors.

Earlier tokamak experiments have, as we indicated above, demonstrated significant ion heating as well as significant increases in the electron temperature from beam heating, although the interpretation of the electron-heating results is, to some extent, complicated by possible changes in equilibrium or in the efficiency of the ohmic heating due to impurity-generated changes in plasma resistivity. The Princeton Large Torus offers, by virtue of its increased size (major radius $R = 130 \text{ cm}$, minor radius $a = 40 \text{ cm}$), an improved energy-confinement time attributable to the a^2 scaling and the opportunity to substantially increase the neutral-beam input power beyond that previously obtained.

The PLT neutral beam-injection system consists of four beam lines and 40-kV ion sources designed and supplied by the Fusion Energy Division of the Oak Ridge National Laboratory.¹² Two sources inject their beams parallel to the plasma current (co-injectors) and two inject anti-parallel to the plasma current (counter-injectors). The Princeton group chose this geometry several years ago because they were then concerned about problems that might arise from imparting toroidal momentum to the plasma with an unbalanced system. Although the experiments have, in fact, observed toroidal-plasma rotational velocities up to 10^7 cm sec^{-1} from an unbalanced injector system, no detrimental effects have been associated with the rotation. The group has been able to inject, under optimum conditions, neutral hydrogen-atom beams of 400–500 kW per injector. If deuterium is used as the source gas, the neutralization efficiency is improved, and beam powers of up to 600 kW per injector are obtained.

Capture of the neutral-beam power by the target plasma involves all the binary processes that lead to ion formation, that is, charge-exchange with plasma ions and impact ionization by encounters with electrons and ions. These processes include such interactions with impurity species as well. In fact, until recently, beam penetration was thought to be rather strongly governed by impurity concentrations, but current measurements and theory indicate a somewhat weaker dependence on impurity concentrations. However, at very high beam energies, impurities present in the plasma can still be expected to play a non-negligible role in determining the effectiveness of neutral-beam penetration.

Following ionization and capture of the energetic neutral species, power is transferred to plasma ions and electrons by



Ion temperature as a function of beam power divided by the line-average electron-density. Data in black are from charge exchange, those in color from iron ions. Note that the linear relation between temperature and P/n_e holds both when there are strong density fluctuations (dark color, solid black) and when there are none (light color, gray).

Figure 6

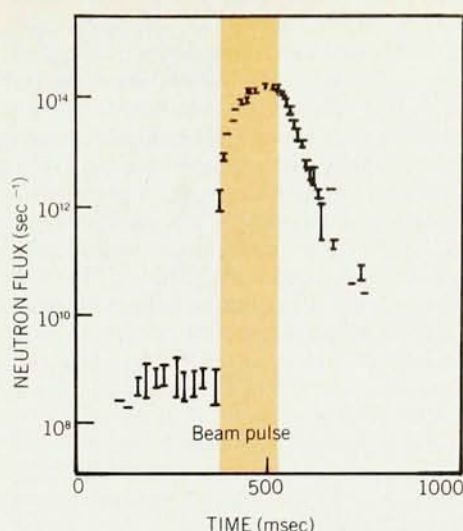
multiple small-angle Coulomb collisions. This power transfer is frequently expressed in terms of E_c , the energy for which the rate of energy transfer to electrons and ions is equal. The critical energy E_c is given by

$$E_c = 15kT_e K$$

where T_e is the electron temperature, k is Boltzmann's constant, and K is a dimensionless constant whose value, which is in the vicinity of 1, depends on beam and plasma properties. For beam energies E the rate of energy transfer to ions is proportional to $E_c^{3/2}E^{-1/2}$, while the rate of energy transfer to electrons is simply proportional to E . When $E = E_c$, about 75% of the beam total energy is transferred to plasma ions. For most present day neutral-beam heating experiments, $E \approx E_c$ and the bulk of the beam power is transferred to plasma ions. Thus, neutral-beam heating has been most effective in heating ions.

The techniques for measuring ion temperature on PLT fall into three categories: mass and energy analysis of the escaping fast neutral species generated by charge exchange, measurements of the Doppler broadening of impurity-line radiation in the x-ray and ultraviolet spectral regions, and thermonuclear neutron emission measurements for H^0 injection into D^+ plasmas. The article by Charles B. Wharton on page 52 discusses these diagnostic techniques in some detail. A multiplicity of methods for determining the ion temperature is desirable because each technique has intrinsic uncertainties, and the agreement between the different measurements increases our confidence in them. Generally, the ion-temperature assessments agree to within 10%. The earlier results reported in *PHYSICS TODAY* in November have now been extended; the PLT group has recorded hydrogen ion temperatures up to 6.5 keV using charge-exchange neutrals. They also measured an impurity-ion temperature of 8.1 keV using the Doppler broadening of a vacuum-ultraviolet line of Fe-XXIV. Because the iron ions are strongly heated by beam ions, one expects the latter temperature to be higher than the hydrogen ion temperature; the calculated difference comes to 1.7 keV, so the two results in fact agree.

While the achievement of an ion temperature in excess of that required for ignition in an ideal DT fusion reactor (about 4 keV) is noteworthy, the true significance of the data lies in the linear relationship of ion temperature to beam power as the temperature moves well into the collisionless regime, where trapped particle modes were predicted to produce enhanced energy transport. As one proceeds into the collisionless regime by increasing the ion temperature, there is a marked enhancement in the level of density fluctuations as measured by microwave scattering. So far, however, no



Neutron emission as a function of time for a 2.2-MW neutral-deuterium pulse injected into a deuterium plasma. The peak fusion output for this experiment was 170 watts. Figure 7

observable effect on the ion energy balance nor on the circulating fast beam-particles has been seen. The ion temperature increase remains proportional to beam power and inversely proportional to line-average electron density as shown in figure 6, even though at the highest power to density ratios (solid points), one observes up to an order of magnitude increase in density fluctuations.

Injection of 2.2 MW of approximately 40 keV D^0 into D^+ plasmas has produced a flux of 1.6×10^{14} neutrons/sec or 2×10^{13} neutrons/pulse, as shown in figure 7, in good agreement with theoretical calculations. Calculations show that these neutrons arise about equally from beam-plasma and beam-beam interactions with less than a 10% contribution from the thermal particles by themselves. The fusion-power production from both branches of the D-D cross-section is 170 watts. The D-T equivalent power, obtained by simply scaling the fusion cross sections, is 50 kW.

The electron heating obtained in PLT with neutral injection is very sensitive to the choice of limiter material and to the conditions of the vacuum-vessel wall. In early experiments with tungsten limiters and even in more recent experiments with steel limiters, counter injection into low-density plasmas has, possibly because of particle orbits which impinge on the limiter, consistently resulted in metallic line radiation from the plasma core greater than the input beam power, quenching any significant electron temperature rise. Installing carbon limiters has permitted the Princeton group to routinely maintain a discharge relatively free of metallic impurities, with a central radiation significantly less than the input power, and obtain strong electron heating with beam injection at low plasma densities.

These experiments at Princeton have not displayed the large increase in energy

transport that had been feared for the conditions achieved. In the low-density, high-temperature regime of the experiments, charge-exchange losses are quite large, whereas the neoclassical ion-energy transport is small by comparison, as we have said. At higher densities where charge-exchange losses are reduced (due to greater plasma opacity), the ion-energy confinement time appears essentially unchanged from its ohmic heating value of roughly 100 msec but the ion energy transport still constitutes a rather small term in the ion power-balance picture. The experiments have also not been able, with the available beam power, to routinely produce large density fluctuations in the higher density regimes. One cannot, therefore, rule out an anomalous enhancement of the ionic thermal conduction under these beam-heated conditions. At the same time it is important to note that the results are also consistent with completely neoclassical ion-energy transport, even in the low-density, highest-power cases with strongly enhanced density fluctuations.

In examining the electron power balance for many beam-heated plasmas in PLT, we observe that, although the volume-integrated net electron-energy confinement-time is unchanged during injection, the electronic thermal conduction in the core of the plasma appears to decrease as $(n_e T_e)^{-1}$. It seems unlikely, however, that the evident reduction in thermal transport in the central plasma region is due to a simple scaling with T_e , because the transport was not reduced in the outer regions of the plasma, which were also heated. Nonetheless, despite the limited precision of the analysis, the improved confinement does appear to be a clearly defined effect.

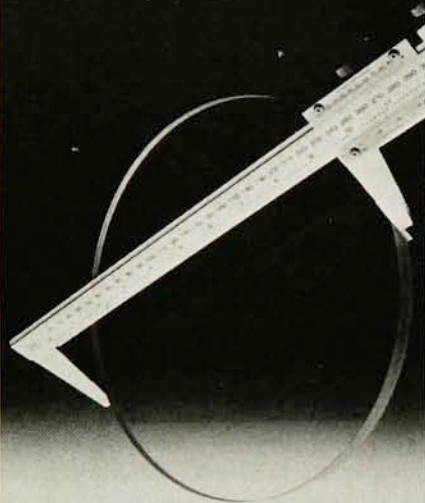
Extrapolation

The results from PLT provide a very optimistic view for the behavior of future tokamaks. One can conclude that ion-energy transport is in reasonable agreement with the neoclassical prediction, and electron-energy transport, although anomalous, diminishes with rising density and possibly with electron temperature as well. Extrapolating these results to the TFTR tokamak, currently under construction at Princeton, one can expect it to achieve "breakeven" operation.

As plasma heating and confinement in tokamaks has progressed, increasing attention has been directed towards the broader questions of engineering and economic feasibility of tokamak fusion reactors. One key issue for economic feasibility has been the efficiency with which the tokamak uses the field produced by the toroidal coils. One measure of that efficiency is the parameter β , which is the ratio of the plasma pressure to the energy density of the magnetic field. Values of about 5–10% are considered appropriate for a tokamak reactor.

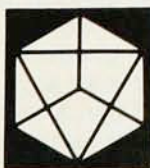
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Both experimental and theoretical efforts are being emphasized to optimize β and to determine the fundamental limitation on β . One of the recent encouraging trends is attainment of favorable energy confinement and relatively high β -values in tokamaks with modest toroidal fields. This has resulted from success in reducing both impurities, which increases the ratio $n_e/(B_\phi/R)$, and macroscopic instabilities which allows operation at low q . In fact, the average β in ohmically-heated tokamaks is found to scale with the product of $n_e/(B_\phi/R)$ and $1/q$. TOSCA at Culham has attained a β of 1% at a q of 2, and DIVA in Japan achieved 0.8% with $q = 1.7$. However, as electron temperature increases in clean plasmas, the resulting reduction of the ohmic heating power input limits the values of β that can be achieved. Neutral-beam injection has proved to be effective in achieving higher β by increasing the input power, increasing the maximum density achievable, and allowing operation at lower q than with ohmic heating alone. Experiments with ISX-B at Oak Ridge obtained values of β of 1.5% averaged over the plasma volume, and central values of about 7% with modest injection power (about 300 kW). The T-11 tokamak at Kurchatov obtained similar results.

Achieving average β -values of 5–10% is a critical step in demonstrating the viability of relatively compact tokamak reactors with relatively low toroidal fields.¹⁴ However, a number of problems remain. For example, linear magnetohydrodynamic theory predicts that a new class of macroscopic instabilities, called "ballooning modes," becomes important at large β , and may limit the values of β that can be attained. The theoretical threshold for this instability can be increased from values of β of about 3% to about 5 or 10% by changing the cross-sectional shape of the plasma column. Several laboratories, including Oak Ridge, Princeton and General Atomic, are planning experiments to confirm this prediction.

There are many other problems that will require attention, such as

- ▶ the buildup of impurities in the plasma during pulses with high-power wall loading and longer than about 10 sec
- ▶ the development of superconducting coils for generating the magnetic fields
- ▶ improving the efficiency of high-voltage beams
- ▶ developing viable refueling schemes, such as pellet injectors
- ▶ understanding and decreasing the flow of energy and particles out of the plasma.

The successful demonstration of a fusion reactor will also require the solution of engineering problems involving the reliability of components, remote handling of parts, and so forth.

Some of these problems are specific to tokamaks, and some are generic fusion problems. Progress in tokamak experi-

ments has been paced by the development of the specialized technologies needed to heat and confine thermonuclear plasmas. At the same time, technological development has benefited from focusing on specific experiments as their target. We have seen a good example of this relationship between physics and technology in the development of neutral-beam injection. We hope that the future will bring other such successes.

The authors would like to acknowledge the help from their colleagues at ORNL and PPPL in preparing this article, in particular the assistance of J. L. Dunlap, J. T. Hogan and J. L. Lyon, to Murakami, and R. Goldston to Eubank.

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