

PROGRESS TOWARD A TOKAMAK FUSION REACTOR

Tremendous advances have been made toward demonstrating the physics and technology required to develop magnetically confined deuterium–tritium tokamak plasmas as an environmentally attractive energy source.

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Nearly 90% of the world's energy needs are today supplied by fossil fuels.¹ Long-term reliance on fossil-fuel energy sources, however, is acknowledged² to be a dangerous strategy—despite the large reserves of coal available in the US and elsewhere. Use of fossil fuels exacerbates pollution and acid rain and heightens the risk of global warming by adding CO₂ to the atmosphere. Society will be served best if energy production in the next century uses environmentally attractive methods that do not involve the combustion of fossil fuels. (See the article by John H. Gibbons and Peter D. Blair in *PHYSICS TODAY*, July 1991, page 22.) The long time required to develop and implement new large-scale energy technologies—on the order of decades—underscores the urgency of the need to accelerate development of alternative energy sources.

Technical progress toward a tokamak-based magnetic-confinement fusion reactor has been dramatic in the last decade. Central ion temperatures of 35 keV (approximately 400 million kelvin) have now been attained, a fivefold increase since 1981. In the past decade, energy confinement times of strongly heated plasmas have increased from 0.02 sec to about 1.4 sec, and normalized plasma pressures have almost quadrupled (see table 1). Current-drive techniques adequate to support steady-state tokamak operation have been developed, including the theoretically predicted self-sustaining “bootstrap” current.

These results, together with the deuterium–tritium experiments that were begun at the Joint European Torus tokamak in Abingdon, England, in November 1991 and that will continue there and in the US, serve to demonstrate the scientific feasibility of fusion power: Enough

progress has been made in achieving key plasma parameters and in physical understanding that it is now possible to predict with good confidence the characteristics of the tokamak plasma required at the heart of a fusion reactor. The accomplishments of the last decade are driving the implementation of an internationally coordinated strategy for the development of fusion power, designed to address the remaining scientific and technological issues that will determine the economic feasibility of fusion power. The first steps in this direction are now being taken.

Fusion reactions and induced radioactivity

As far as is known, fusion is the primary source of energy in the universe. Stars like our Sun produce their energy through a succession of reactions beginning with fusion of protons into deuterium: $p + p \rightarrow D + e^+ + \nu + 1.44 \text{ MeV}$. Unfortunately this and succeeding reactions have cross sections far too small to permit their exploitation in a reasonably sized terrestrial fusion reactor. The following reactions, however, have cross sections large enough to be of interest:



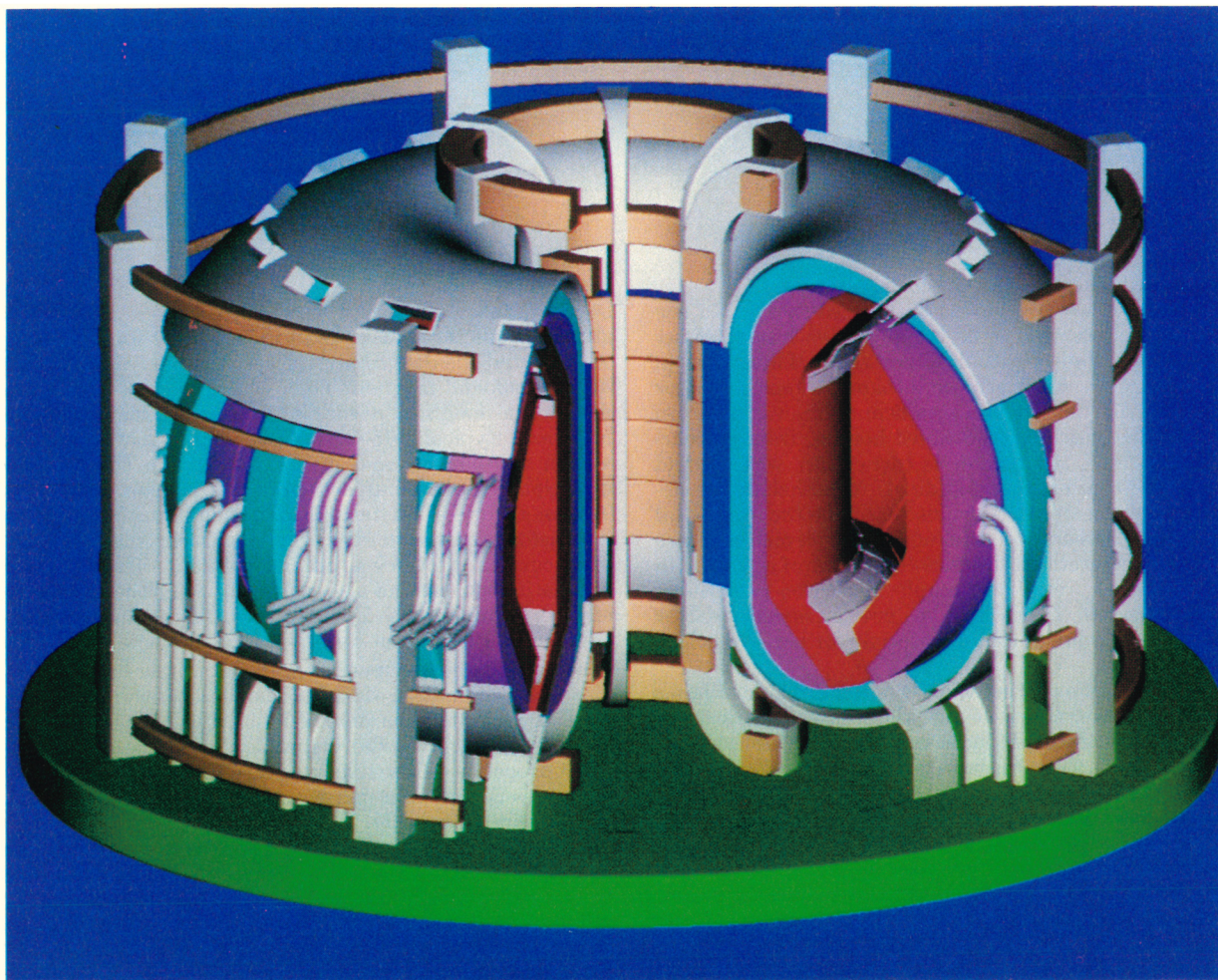
The fusion reaction rate parameter $\langle\sigma v\rangle$ (where σ is the cross section, v is the relative speed of the reactants, and the brackets indicate averaging over the Maxwellian velocity distribution of reacting species) is by far the largest for the deuterium–tritium reaction. Thus the first fusion reactors will almost certainly burn D–T fuel.

About 1 tonne of deuterium would be required to produce 1 gigawatt-year of electrical energy in a fusion reactor, whereas 2×10^6 tonnes of carbon are required for a coal-fired power plant. This disparity reflects the relative scale of nuclear and molecular potentials. The high fuel-

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ARIES-I tokamak reactor design developed at the University of California, Los Angeles. The D-T plasma circulates in the evacuated region inside the torus, surrounded by the breeding blanket (red), the shield (purple) and the toroidal magnetic field coils (light blue). The copper-colored coils produce the poloidal magnetic field. Divertor plates (gray) are visible at the top and bottom of the plasma region. The dark blue element just inside the "doughnut hole" is the bucking cylinder, which resists the centripetal forces experienced by the toroidal coils. (From ref 6.) **Figure 1**

mass efficiency of nuclear energy production translates into substantial economies in the mining and transportation of fuel and dramatic reduction in the production of waste.

While deuterium is stable, and one deuterium atom occurs naturally for every 6700 atoms of hydrogen, tritium is a β emitter with a half-life of 12.3 years and so is not found in significant quantities in nature. Consequently tritium must be generated in a thick blanket surrounding the plasma (see figure 1), using reactions such as $n + {}^6\text{Li} \rightarrow {}^4\text{He} + \text{T}$. The heat deposited by the slowing down of the 14-MeV neutrons in the blanket and by the exothermic reactions that breed the tritium is transferred to a coolant such as high-temperature helium. The hot helium is subsequently used to drive turbines to generate electricity. Thus deuterium and lithium, which would be extracted from abundant sources such as saline lakes, the ocean and geological deposits, are the raw fuels required for an electricity-generating D-T fusion reactor.

Fusion fuels and their reaction products involve no long-lived radioactive isotopes. However, both D-T and D-D reactions produce multi-MeV neutrons that are

absorbed in the blanket surrounding the reacting plasma. At a typical flux of about 3 MW/m^2 at the first wall of a D-T fusion reactor,³ these neutrons induce nuclear transmutations in the blanket and support structure. The strength and consequence of the induced radioactivity depend on the structural materials employed. The calculated decay of radioactivity following shutdown of D-T fusion reactors constructed of various materials is compared⁴ with that of a fission breeder reactor in figure 2. Fusion reactors using advanced structural materials such as silicon carbide are expected to produce far less and much-shorter-lived radioactive waste than their fission counterparts.

Development and testing of materials with desirable structural and nuclear properties for use in fusion reactors is important for increasing the environmental attractiveness of fusion power. The longevity of first-wall components in the fusion reactor environment is an important economic issue as well. Substantial progress has been made in developing and characterizing candidate fusion reactor materials using fast-fission reactors, but a high-flux 14-MeV neutron source is needed to

support a full development program.

A D-T reactor will have a tritium inventory of a few kilograms, which represents a radioactive hazard many orders of magnitude less than that of the fuel inventory in a fission reactor. Furthermore a tokamak fusion reactor carries the risk neither of a significant runaway reaction nor of severe afterheat (continued heating from radioactive decays after the reactor is turned off), which could damage the containment building. Nonetheless the design of a D-T fusion reactor must preclude the accidental release of tritium and volatile radioactive elements. Since there is no requirement for fissile materials of any form in a fusion reactor, relatively simple inspection procedures should make it impossible to use a fusion reactor clandestinely to breed fissile materials for use in nuclear weapons.

Fusion reactions other than D-T require much higher plasma parameters than we now know how to achieve, but they offer substantial reductions in radioactive hazards. For the D-D reaction, tritium generation is not required, and the tritium produced via the triton branch of the reaction can be burned by D-T reactions. The $D + {}^3\text{He} \rightarrow p + {}^4\text{He}$ reaction has the further advantage that only stable nuclei are involved in all steps of the fuel cycle. While collateral D-D reactions produce neutrons, optimizing the ratio of D to ${}^3\text{He}$ and adjusting the ion temperature can reduce the energy production in neutrons to 1% or less. The proposed eventual source of ${}^3\text{He}$ is the regolith of the lunar surface, which contains about 10 parts per billion of ${}^3\text{He}$ as a result of deposition by the solar wind.⁵

Tokamak fusion reactors

The recently completed conceptual design of the ARIES-I D-T tokamak reactor,⁶ specified to produce 1000 MW of electric power, is illustrated in figure 1. The magnetic

system of a tokamak reactor is dominated by the large superconducting toroidal field coils, which produce a strong magnetic field in the toroidal direction (the long way around the doughnut), typically in the range of 5–10 T. The poloidal magnetic field, whose lines of force wrap around the doughnut in the short direction, is produced by a current (typically 15–25 MA) flowing around the torus within the plasma, and by coils wound in the toroidal direction and distributed around the plasma periphery. The solenoidal coil in the hole of the doughnut functions as a transformer to induce the toroidal current in the plasma. This current provides a crucial element of the overall magnetic topology and also ohmically heats the plasma to electron temperatures T_e of 2–3 keV. Above this temperature range the plasma resistivity (which varies as $T_e^{-3/2}$) becomes too low to support significant ohmic heating.

The combination of the toroidal and poloidal magnetic fields provides the characteristic sheared helical magnetic structure that gives the favorable stability and confinement properties of the tokamak, as discussed in the article by James D. Callen, Benjamin A. Carreras and Ronald D. Stambaugh on page 34.

The major radius R of the plasma in tokamak reactor designs is typically 6–7 m, the minor horizontal radius a is in the range 1.5–2.0 m, and the plasma cross section is elliptical or D-shaped, with an elongation $\kappa \equiv b/a$ of about 2, as shown in figure 3. The experimental achievement of elongated and even somewhat triangular plasma shapes, which permits optimization of energy confinement and of plasma pressure, has been a major accomplishment of the last decade of fusion research.

The plasma boundary is defined by the magnetic separatrix (see figure 3). Each magnetic field line on the plasma side of the separatrix stays within this region, wrapped helically and endlessly around one of a set of toroidally nested surfaces containing a given magnetic

Table 1. Key plasma parameters: Achieved and required

Parameter	1971	1981	1991	Steady-state D-T reactor
Central ion temperature T_{i0} (keV)	0.5	7	35	30
Central electron temperature T_{e0} (keV)	1.5	3.5	15	30
Energy confinement time τ_E (sec)	0.007	0.02	1.4	3
Triple product $n_{i0} T_{i0} \tau_E$ (keV sec/m ³)	1.5×10^{17}	5.5×10^{18}	9×10^{20}	7×10^{21}
Plasma pressure $\beta = 2\mu_0 \bar{p}/B^2$	0.1%	3%	11%	5%
Current-drive parameter $\eta_{CD} = n_{e0} R I_p / P_{CD}$ (A/m ² W)	—	8×10^{18}	3.4×10^{19} + bootstrap	3×10^{19} + bootstrap
Fusion reactivity:				
D-D reactions/sec	—	3×10^{14}	1×10^{17}	—
D-T reactions/sec	—	—	$\sim 6 \times 10^{17}$	10^{21}

flux. Field lines outside the separatrix leave the vicinity of the plasma within a single poloidal transit. Heat and particles flow quickly along magnetic field lines, but are relatively well confined on the flux surfaces within the separatrix. Ultimately, however, cross-field transport processes cause the plasma to diffuse across the internal flux surfaces and eventually across the separatrix. The plasma then follows the field lines to actively cooled divertor plates, which absorb its energy. Plasma flowing into the divertor is also neutralized at these plates, and the gas this produces, including the He ash of the D-T fusion reaction, is pumped away through nearby ducts.

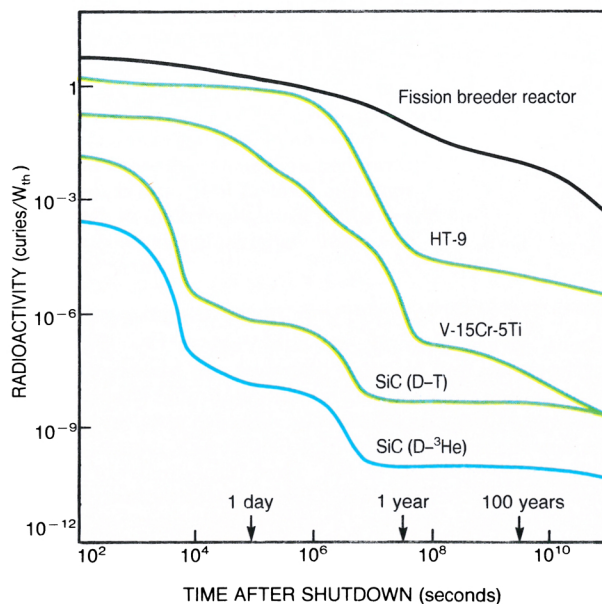
The main plasma-wall interaction takes place at the divertor plates. Because the heat and particle fluxes are substantial, the divertor represents one of the most challenging and important problems facing reactor designers. Substantial advances have been made in the development of reactor divertor concepts, but further innovation is needed. For example, researchers are actively investigating concepts that involve converting the very-high-power-density plasma of the divertor region into a denser but much cooler neutral gas that would spread the heat load over a larger area in the divertor.

A steady-state tokamak reactor will require substantial auxiliary heating and current-drive power (about 100 MW) to bring the 2–3-keV ohmically heated plasma to ignition temperatures (about 30 keV) at the beginning of each burn cycle, and to augment the capability of the solenoidal transformer to drive the plasma current. Although pulses lasting as long as several hours can be sustained by transformer action, eventually the solenoid reaches its maximum magnetic flux, and consequently a true steady state—which would be desirable for a fusion reactor—requires an auxiliary current-drive system.

The plasma itself provides fortuitous assistance in this process, since a confined toroidal plasma creates its own toroidal “bootstrap” current.⁷ In a plasma with a finite pressure gradient and an applied magnetic field the superposition of the Larmor orbits of the plasma particles creates a diamagnetic current perpendicular to both the magnetic field and the pressure gradient. In a roughly analogous manner, the special orbits in a tokamak plasma, in the presence of the radial pressure gradient, generate a substantial current parallel to the helical magnetic field. A small “seed” current must still be driven in the center of the plasma, where there is no pressure gradient. This seed current must be externally driven by radiofrequency waves or by MeV ions injected into the plasma initially as neutrals, but the bootstrap current can efficiently generate the majority of the plasma current.

Reactor plasma performance requirements

A D-T tokamak plasma must meet certain fundamental requirements to function as the core of an economically attractive, power-producing fusion reactor. It must produce net output power with sufficient gain compared with any input power required to sustain the plasma, and it must produce this power at sufficiently high power density at a realistic magnetic field strength.



Radioactivity after shutdown per watt of thermal power (W_{th}) in D-T fusion reactors made of various structural materials: HT-9 ferritic steel, V-15Cr-5Ti vanadium–chromium–titanium alloy and silicon carbide. Note the millionfold advantage of SiC over steel a day after shutdown. For comparison, radioactivity levels after shutdown are also given for a liquid-metal fast breeder reactor and a SiC fusion reactor using the neutron-reduced D-³He fuel cycle. (Adapted from a figure provided by the ARIES group; see ref. 4.)

4.) **Figure 2**

In steady state the fusion power gain is given by $Q \equiv P_f/P_i$, where P_f is the fusion power and P_i is the input power. The fusion power is given by

$$P_f = 17.58 \text{ MeV} \times \int n_D n_T \langle \sigma v \rangle dV \quad (1)$$

where n_D and n_T are the deuterium and tritium number densities and the integral is over the volume V of the plasma.

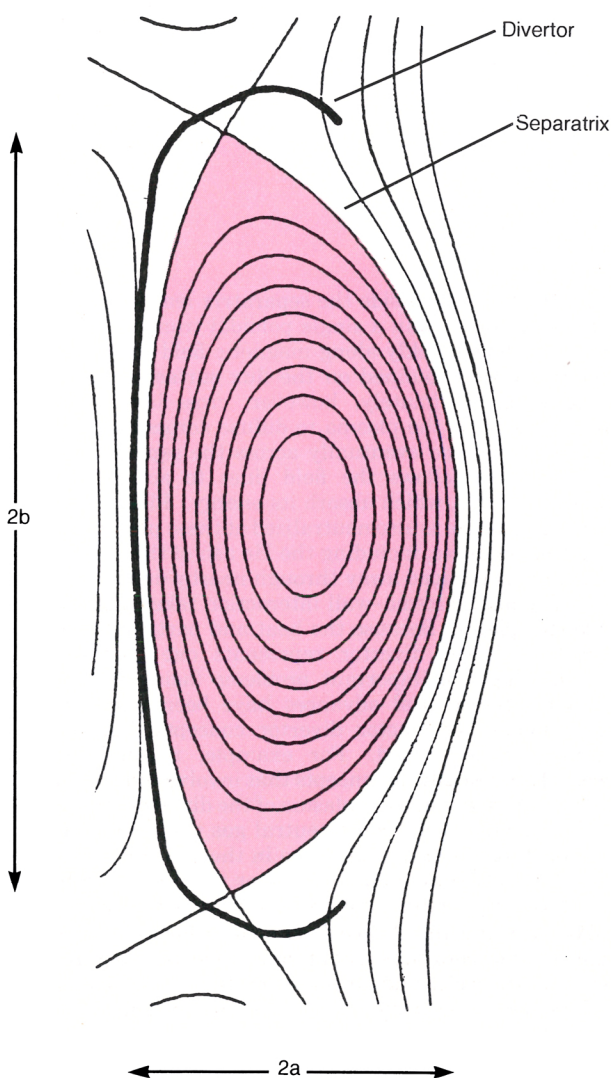
The externally supplied input power must make up the difference between the power lost from the plasma by cross-field plasma transport and radiation and the power gained due to the ⁴He or α particles that are generated in the D-T fusion reaction and captured in the plasma:

$$P_i = \frac{3}{2\tau_E} \int (n_i T_i + n_e T_e) dV - 3.52 \text{ MeV} \times \eta_\alpha \int n_D n_T \langle \sigma v \rangle dV \quad (2)$$

The energy confinement time τ_E characterizes the rate at which energy is lost from the plasma by cross-field transport and radiation. The efficiency η_α with which the α power is transferred to the bulk plasma is generally taken to be close to unity. T_i and T_e are the ion and electron temperatures (including an implicit factor of the Boltzmann constant, giving them units of energy), and n_i and n_e are the ion and electron number densities. The input power must also sustain the plasma current, but for inductive current drive this is a very small term.

To get a feeling for the main dependencies of the

Modern tokamak plasma cross section. The lines indicate surfaces containing constant magnetic flux; magnetic field lines are constrained to move along these surfaces. The "separatrix" surface marks the boundary between plasma field lines, which are confined to helical trajectories on closed magnetic surfaces (red), and divertor field lines, which leave the vicinity of the plasma on "open" surfaces and intercept divertor plates. **Figure 3**



fusion power gain Q , we can neglect the differences between ion and electron densities and temperatures, as well as the temperature and density profile shapes, which affect the integrals differently. We also note that for D-T fusion reactions in the $T_i = 10\text{--}30$ keV range, $\langle\sigma v\rangle$ is roughly proportional to T_i^2 . Then it follows from dividing each term by nT/τ_E that Q depends dominantly on the fusion "triple product," $nT\tau_E$. Detailed calculations indicate that $n_{i0} T_{i0} \tau_E$ (where the "0" subscript indicates the value in the center of the plasma) must exceed about 7×10^{21} keV sec/m³, with central ion temperatures near 30 keV, for the terms in the denominator to cancel, giving $Q = \infty$, or ignition. Figure 4 illustrates these requirements and shows the values achieved over the past 25 years. Ignition in D-D or D-³He requires peak temperatures of 50–100 keV and more than an order of magnitude increase in $n_{i0} T_{i0} \tau_E$.

Another key parameter describing tokamak performance is $\beta \equiv 2\mu_0 \bar{p}/B^2$, the ratio of the volume average of the plasma pressure $p \equiv n_i T_i + n_e T_e$ to the magnetic pressure $B^2/2\mu_0$. The maximum fusion power density that can be obtained in a D-T plasma in the temperature range of interest scales as $n^2 T^2$, which is proportional to $\beta^2 B^4$. The strength of the magnetic field is limited by practical considerations such as stresses in the magnetic support structure and the critical field and current density of the superconducting cable. To provide a neutron flux of about 3 MW/m² at the reactor walls, a β value of about 5% is required. High-aspect-ratio designs ($R/a \sim 5$) require somewhat lower values of β than low-aspect-ratio designs ($R/a \sim 3$), since in the former the field strength in the plasma is a larger fraction of the peak field at the magnet.

The final key measure of tokamak performance is the current-drive "efficiency" $\eta_{CD} \equiv n_{e0} R I_p / P_{CD}$, where n_{e0} is the central electron density, I_p is the driven plasma current, and P_{CD} is the current-drive power. For a tokamak reactor with noninductive current drive, the requirement to provide input power to drive the toroidal current sets a lower limit on the value of P_i and consequently sets an upper limit on the achievable value of Q . A current-drive efficiency η_{CD} of about 3×10^{19} A/m² W and a bootstrap current making up about 70% of the total are required to obtain $Q > 20$, as is desirable for an economically attractive steady-state fusion reactor.

Progress toward reactor-grade plasmas

The results of the last two decades' dramatic progress in plasma properties are summarized in table 1. We now discuss each aspect in more detail.

Plasma heating. Several different plasma heating techniques have been successfully developed. They can be divided into two main classes: heating by injected energetic particles, and heating by radiofrequency electromagnetic waves.

The highest ion temperatures ($T_{i0} = 34.6$ keV) have been achieved in the Tokamak Fusion Test Reactor⁸ at the Princeton Plasma Physics Laboratory through the injection

of 30 MW of energetic ions (about 110 keV per ion), using injectors developed at the Lawrence Berkeley Laboratory. In this technique, called neutral-beam injection, or NBI, deuterium ions are generated in an arc-chamber plasma and accelerated by multiaperture electrostatic grids. They are then neutralized via charge exchange with deuterium gas. The resulting energetic atoms cross the magnetic field into the plasma, where they are collisionally reionized and then thermalized, transferring their energy to the bulk plasma.

The physics of this heating scheme is well understood. Detailed measurements have confirmed that under most conditions the fast ions are well confined by the tokamak magnetic fields and transfer their energy to the background thermal plasma—usually mostly to the ions—via cumulative small-angle Coulomb collisions ("classical" slowing down). Fusion neutrons produced by reactions between energetic beam ions and the bulk plasma ions provide both an excellent diagnostic of the thermalization process and a significant enhancement of the total fusion output power from present experiments. At the highest plasma pressures, however, and under conditions where the pressure of the thermalizing ions is a substantial fraction of the total pressure, interactions between the fast ions and modes of plasma oscillation have been observed to cause losses of the energetic ions from the core of the

Approaching break-even and ignition: Fusion "triple products" $n_{i0} T_{i0} \tau_E$ and ion temperatures achieved in experiments over the past 25 years. Also indicated are the requirements for deuterium-tritium fusion power gain $Q_{DT} = 0.1, 1$ (break-even) and ∞ (ignition). See table 2 for key to the various experiments. **Figure 4**

plasma. The most virulent case involves a low-order resonant interaction with the fast-ion orbital motion. The physics of this interaction is well understood analytically and has been modeled numerically in both the linear (small amplitude) and the nonlinear (large amplitude) phase.

One of the most advanced of the rf-wave heating techniques is ion cyclotron resonance heating. In this scheme a deuterium plasma is doped with a small component of hydrogen or helium, which is heated by the rf waves to energies up to the MeV range. Unlike NBI, which usually supplies energetic ions with their velocity nearly parallel to the magnetic field, ICRH energizes ions in the perpendicular direction. At MeV energies the minority ions transfer their energy by Coulomb collisions mainly to the background plasma electrons. In the Joint European Torus tokamak, central electron temperatures exceeding 15 keV have been obtained using this technique.

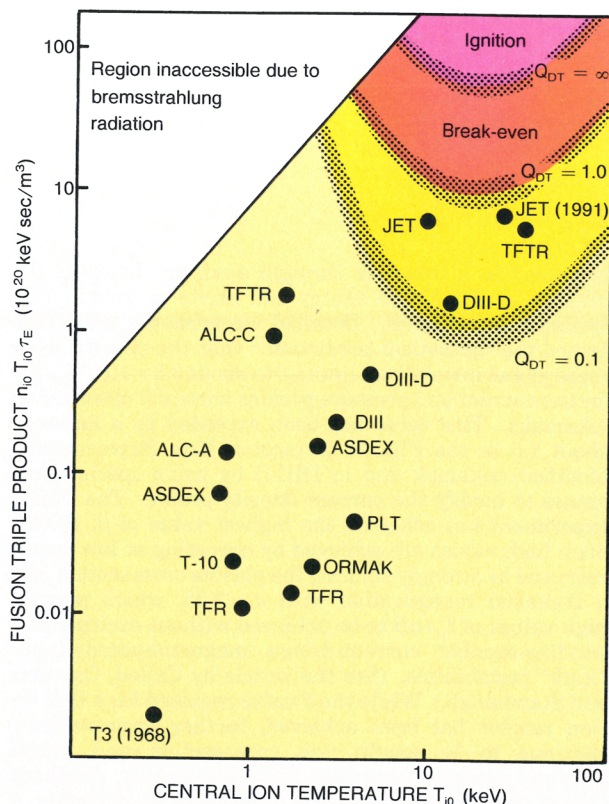
Both NBI and ICRH heating processes are similar to the collisional heating of a D-T reactor plasma by the fusion-product α particles. The fact that these heating processes are generally classical encourages the belief that 3.5-MeV α particles will also transfer their energy classically to the background plasma. If under some circumstances the α heating efficiency η_α were less than unity, however, then somewhat higher values of $n_{i0} T_{i0} \tau_E$ would be required to attain a specified value of Q . A key issue in this area is that fusion α particles have velocities parallel to the magnetic field that are greater than the Alfvén speed, a natural propagation speed of perturbations in the plasma, and thus may interact resonantly with plasma modes associated with shear Alfvén waves, which are generally inaccessible to the fast ions associated with NBI or ICRH.

Energy confinement. "Anomalous" (that is, turbulence-driven) cross-field energy transport in tokamaks is not well understood, and indeed providing a solid theoretical understanding of cross-field transport is one of the main outstanding challenges of tokamak plasma physics. However, in the early 1980s comparison of results from heating experiments on the available small and medium-sized tokamaks ($R = 0.9\text{--}1.6$ m, $a = 0.25\text{--}0.45$ m, $I_p = 100\text{--}600$ kA, auxiliary heating power $P_{aux} = 0.2\text{--}6$ MW) established⁹ an empirical scaling of the energy confinement time:

$$\tau_E \text{ (sec)} = 3 \times 10^{-5} I_p^{1.75} a^{-0.37} \kappa^{0.5} A_i^{0.5} / P_{aux}^{0.5} \quad (3)$$

where A_i is the ion mass in atomic mass units, and all other quantities are expressed in base SI units. The high quality of the experimental data yielded consistent scaling results across different tokamaks, as shown in figure 5. This permitted development of a meaningful overall confinement scaling, including parameters such as R and a , which were not varied significantly within individual devices.

Working from the constituent equations governing turbulent transport in a fully ionized plasma, one can show that the exponents in any scaling relation for



confinement time must obey an algebraic constraint.¹⁰ The experimental result (equation 3) easily obeys this constraint within its error bars. Such arguments also indicate that progress in confinement should be expressed in terms of dimensionless quantities such as $\omega_c \tau_E$ (where ω_c is the cyclotron frequency of the ions or the electrons), as in figure 5, which plots the proportional quantity $B \tau_E$. In the present generation of large tokamaks (TFTR, JET and the JT-60 tokamak at Naka, Japan), which have $R \sim 3$ m, $a \sim 1$ m, I_p up to 7 MA and P_{aux} up to 30 MW, energy confinement was found to be predicted quite well by this empirical scaling relation. The mean error of the extrapolation was 4%, with an rms spread of 12%. This result provides confidence in the further extrapolation to tokamak reactors.

Of equal importance are a number of special techniques developed during the last decade that improve energy confinement by significant factors beyond the established scaling. For example, the "high mode," or "H-mode," confinement¹¹ was developed on the medium-sized ($R = 1.6$ m) ASDEX divertor tokamak in Garching, Germany, and has been replicated on the larger DIII-D tokamak at General Atomics in San Diego and on JET, as well as on other medium-sized tokamaks. The scaling of H-mode confinement is consistent with equation 3 multiplied by an overall factor of about 1.85. Confinement times of up to 1.4 sec have been achieved in JET at the highest currents. Continuing research in this area has shown that regimes exist with even higher enhancement factors. Thus, while detailed cross-field energy transport processes are not yet well understood theoretically, empirically the confinement-time scaling is reliable enough to permit confident predictions of the performance of reactor-regime plasmas. Furthermore, the scaling is favorable enough that it predicts that such plasmas can be achieved with $R \sim 6\text{--}7$ m.

Plasma pressure. When the plasma pressure is

increased in a tokamak through auxiliary heating, the pressure-gradient free energy can excite magnetohydrodynamic instabilities. Detailed experiments have confirmed the theoretical prediction¹² that the threshold for these global instabilities limits β to about $3.5 \times 10^{-8} I_p / aB$ for the current and pressure profiles normally observed on tokamaks. This value has been exceeded by a factor of about 1.5 in the PBX-M (Princeton Beta Experiment—Modified) tokamak and in DIII-D by using special techniques to modify the current density profile. The DIII-D experiment has achieved the highest value of β , 10.6%. Such high values are achieved by operating at low aspect ratio and by strongly shaping the plasma cross section into a D-shaped configuration with $\kappa = 2.35$, which permits high values of I_p / aB to be achieved without exciting low-winding-number current-driven magnetohydrodynamic “kink” instabilities. (See the article by Callen, Carreras and Stambaugh.) While the β value required for a D-T fusion reactor has been achieved, further improvements continue to be sought (via cross-section and profile shaping, perhaps accessing the theoretically predicted second-stability regime¹³ that should exist at higher β values) in order to increase the fusion power density and make attractive D-D and D-³He fusion reactors possible.

Current-drive efficiency. Current can be driven in a tokamak by supplying momentum to the plasma electrons through interaction with traveling electromagnetic waves,¹⁴ or by the unidirectional ion beams produced by tangential neutral-beam injection. In either case, the flow of the current-carrying species is impeded by Coulomb collisions with the bulk plasma. It follows that the current-drive parameter η_{CD} has an upper limit due

strictly to “classical” binary Coulomb collisions. Values of η_{CD} of approximately $3.4 \times 10^{19} \text{ A/m}^2 \text{ W}$ have been achieved, in good agreement with theoretical prediction, in the JT-60 tokamak. Triam-1, a small superconducting tokamak in Kyushu, Japan, has achieved pulse lengths in excess of one hour. Both devices, however, employed lower-hybrid waves (that is, waves with frequency $\omega \sim (\omega_{ci} \omega_{ce})^{1/2}$, where ω_{ci} and ω_{ce} are the ion and electron cyclotron frequencies, respectively), which may not be able to penetrate to the core of a hot fusion reactor plasma. Waves with frequencies in the ω_{ci} range are now being tested for use in reactor current drives.

Very high values of η_{CD} would be required to drive the total current in a D-T tokamak fusion reactor at high Q , but the bootstrap effect can provide the majority of the plasma current. In 1986 experiments on TFTR first verified the 1971 prediction of the bootstrap current in a tokamak, and other experiments confirmed the result soon afterwards. The current-drive efficiency that has already been achieved would, in combination with the bootstrap effect, be adequate to provide the current required for a steady-state tokamak reactor.

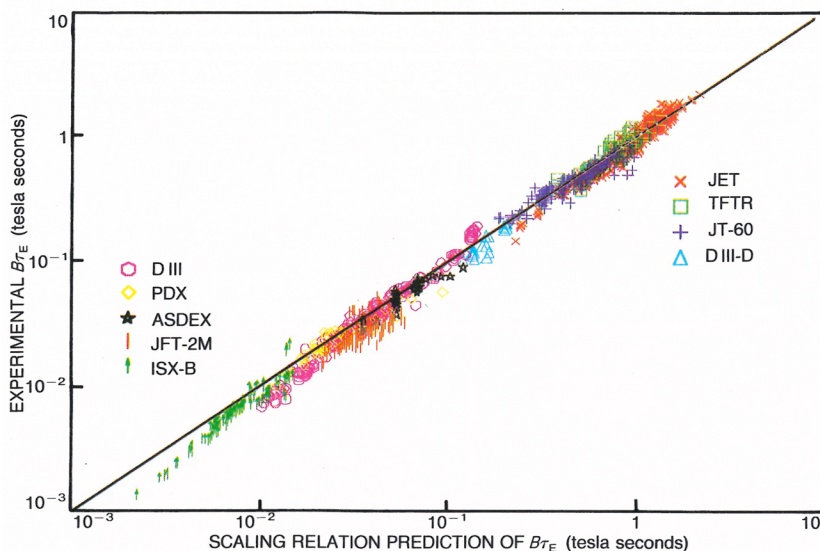
Steady-state operation, however, still presents some difficulties. Acceptable values of η_{CD} are obtained at high plasma temperatures, which tend to result in unacceptably high sputtering and erosion rates at the divertor plates. Tokamak plasmas also suffer from sudden current “disruptions.” These occur when I_p / aB is close to the kink limit, β is close to its limit, or the current profile is strongly distorted due to excess plasma cooling by line radiation from partially ionized impurities at the plasma edge that arise from abnormal plasma-wall interactions. Current disruptions terminate the plasma and rapidly dump large amounts of energy on the divertor plates. There are reasons to believe that fully current-driven plasmas can be made less susceptible to disruptions, owing to the enhanced current profile control that is possible in such plasmas. However, all these effects and the uncertainties associated with them add to the need for innovative solutions to the divertor problem and for an integrated test of steady-state tokamak operation.

Fusion reactivity. Returning to the fusion performance of present experiments, it can be seen from figure 4 that both TFTR and JET have achieved plasma parameters close to the “scientific break-even” condition, $Q_{DT} \sim 1$. JET has obtained $n_{i0} T_{i0} \tau_E \sim 9 \times 10^{20} \text{ keV sec/m}^3$, with $n_{i0} \sim 3.5 \times 10^{19} \text{ m}^{-3}$, $T_{i0} \sim 25 \text{ keV}$ and $\tau_E \sim 1 \text{ sec}$. Only a decade ago $n_{i0} T_{i0} \tau_E$ values in high-temperature tokamaks such as the Princeton Large Torus were about 200 times lower, with similar density but much lower T_{i0} and τ_E .

The TFTR and JET experiments generally operate using deuterium, with their neutral-beam systems also injecting deuterium. The fusion performance of these devices can be assessed by measuring their neutron yields. The highest D-D fusion reaction rate measured so far is 10^{17} neutrons/sec in TFTR. In preliminary experiments JET introduced tritium into two of its 16 neutral-beam injectors, and achieved D-T fusion power approaching 2 MW, or approximately 6×10^{17} neutrons/sec. This

Table 2. Fusion experiments

ALC-A	Alcator-A, Plasma Fusion Center, MIT
ALC-C	Alcator-C, Plasma Fusion Center, MIT
ASDEX	Axially Symmetric Divertor Experiment, Max Planck Institute for Plasma Physics, Garching, Germany
ATC	Adiabatic Toroidal Compressor, Princeton Plasma Physics Laboratory
BPX	Burning Plasma Experiment, PPPL (proposed)
DIII	Doublet III, General Atomics, San Diego
DIII-D	Doublet III-D, General Atomics, San Diego
ISX-B	Impurity Studies Experiment B, Oak Ridge National Laboratory
ITER	International Thermonuclear Experimental Reactor (location under negotiation)
JET	Joint European Torus, Abingdon, England
JFT-2M	Japan Atomic Energy Research Institute, Tokai, Japan
JT-60	Japan Atomic Energy Research Institute, Naka, Japan
ORMAK	Oak Ridge Tokamak, Oak Ridge National Laboratory
PDX	Princeton Divertor Experiment, PPPL
PLT	Princeton Large Tokamak, PPPL
T-3	Kurchatov Institute, Moscow
T-10	Kurchatov Institute, Moscow
TFR	Centre d'Etudes Nucleaire, Fontenay-aux-Roses, France
TFTR	Tokamak Fusion Test Reactor, PPPL



Empirical scaling relation developed in 1983 from experimental results on small- and medium-sized tokamaks accurately predicts the confinement times of subsequent larger devices. The toroidal magnetic field B multiplied by the confinement time τ_E is proportional to the dimensionless quantity $\omega_c \tau_E$, where ω_c is the cyclotron frequency of either the ions or the electrons. **Figure 5**

result confirms the theoretical extrapolations from D-D operation that indicate that both TFTR and JET will be able to perform experiments in the $Q = 0.5$ –1 range when they begin D-T operation in earnest in 1993–95. These experiments should generate more than 10 MW of fusion power, due predominantly to fusion reactions between energetic beam ions and background plasma ions. They will provide the first information on fast-ion collective effects associated with α particles but are not likely to achieve the substantial levels of α -particle plasma heating needed for measurement of the value of the α -heating efficiency η_α . Figure 6 shows the maximum fusion power output in each tokamak device worldwide, illustrating the dramatic progress in fusion reactivity achieved over the last two decades.

The next steps

Development of fusion power in the next century will require improved physics understanding and engineering technique in a number of areas. A variety of US¹⁵ and international studies have identified the key remaining scientific and technological issues that must be resolved:

- ▷ confinement of D-T plasmas at high $n_{i0} T_{i0} \tau_E$
- ▷ collective stability of super-Alfvénic α -particle plasma components
- ▷ heating by α particles and stable burn control of the self-heated plasma
- ▷ power and particle handling in divertors
- ▷ steady-state current drive and disruption control
- ▷ helium ash removal
- ▷ long-life, low-activation structural materials
- ▷ large superconducting magnet systems
- ▷ tritium breeding blankets.

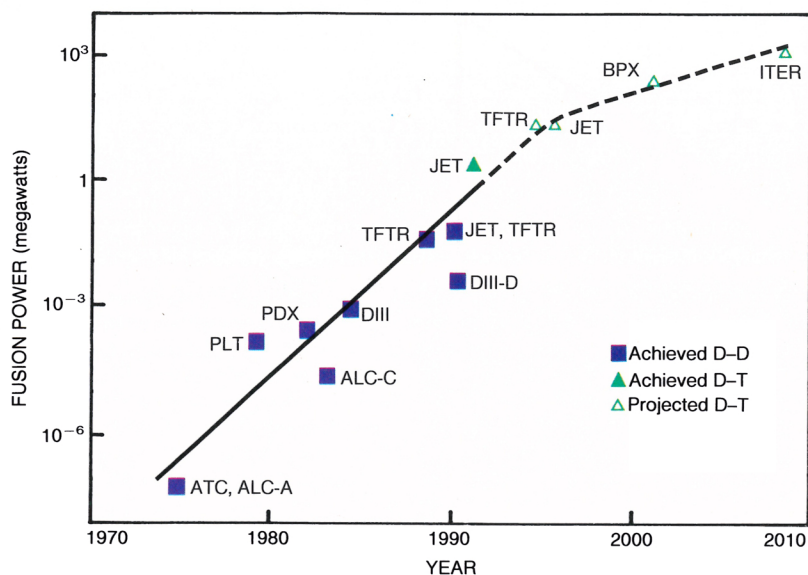
In 1990 the US Secretary of Energy, James D. Watkins, commissioned an independent Fusion Policy Advisory Committee to advise the Department of Energy on a sound long-term policy for fusion research. After six months of deliberation, FPAC prescribed in its final report¹⁶ a magnetic fusion energy development program whose largest elements were a burning-plasma experiment to address the first three of these items, a long-pulse or steady-state advanced tokamak physics experiment to address the next three, a 14-MeV neutron source to address the matter of fusion reactor structural materials, and an engineering test reactor to integrate solutions to all the issues listed. We now briefly describe these

facilities, as conceived by FPAC and discussed in President Bush's National Energy Strategy.

Burning Plasma Experiment. The US magnetic fusion program has completed the conceptual design of a high-field, relatively compact, D-T-burning plasma physics experiment that is projected to produce 100–500 MW of fusion power and to attain Q in the range of 5 to ignition for short pulses. The major parameters of the BPX are $R = 2.6$ m, $a = 0.8$ m, $\kappa = 2.2$, $I_p = 12$ MA and $B_T = 9$ T. In many ways this device represents the culmination of the US fusion research effort: The Doublet tokamaks at General Atomics have demonstrated the advantage of vertical elongation, the Alcator tokamaks at MIT have developed the physics and engineering of high-magnetic-field copper-coil tokamaks, and the tokamaks at Princeton have developed the physics and engineering of strong auxiliary heating. Budget constraints, however, make it unlikely that BPX will be constructed, with the result that the scientific issues associated with α -heated plasmas will first be addressed in the ITER device (see below).

Steady-state advanced tokamak physics experiment. It is anticipated that a steady-state tokamak reactor with $Q > 20$ can be built using noninductive current drive enhanced by the bootstrap effect. A key plasma physics question that must be addressed by experimentation is whether the auxiliary current drive and the bootstrap effect can support steady-state, non-disruptive, self-consistent current and pressure profiles. In addition, this device should explore extensively the significant enhancements of confinement and β that should be achievable in a plasma where the current profile can be externally controlled. Other important objectives for a steady-state device are the demonstration of innovative high-heat-flux divertor concepts that generate a minimum of plasma impurities, and the development of efficient techniques for removing helium ash.

International Thermonuclear Experimental Reactor. At the 1985 Geneva summit General Secretary Mikhail Gorbachev proposed to President Ronald Reagan that a major tokamak experiment should be built by a collaboration of the world's four major fusion programs. Beginning in 1987 a design team with participants from the US, the Soviet Union, Europe and Japan met for extended periods in Garching, Germany. A conceptual design for ITER was completed in 1990, with parameters $R = 6$ m, $a = 2.15$ m, $\kappa = 2.2$, $I_p = 22$ MA and $B_T = 4.85$ T.



Fusion power generated and expected in past and future tokamak experiments worldwide. See table 2 for key to experiments. (Adapted from ref. 16.) **Figure 6**

The device is designed to achieve ignition and steady-state burn for very long pulses and to function subsequently as an engineering test reactor. In March 1992 the four ITER partners are scheduled to sign an agreement committing to the first two years of a six-year engineering design phase. A subsequent decision to begin construction of ITER in about 1996 would result in first plasma in about 2005, with D-T experiments beginning in about 2008.

High-flux 14-MeV neutron source. To develop the full economic and especially environmental potential of fusion it will be necessary to develop and test new structural materials. ITER will not provide a neutron fluence high enough to permit full-lifetime tests of fusion materials. Thus a high-flux 14-MeV neutron irradiation facility is required so that the next-generation tokamak device after ITER can demonstrate not only the economical production of power but also the environmental and safety advantages of fusion.

Tremendous progress has been made toward a tokamak fusion reactor in the last two decades. In the early 1970s ion temperatures of 500 eV, confinement times of 0.007 sec, and fusion power of 50 milliwatts inspired the construction of new experimental tokamak devices worldwide. In the late 1970s increased ion temperatures and β values motivated the construction of large tokamaks in the US, Europe and Japan: TFTR, JET and JT-60. Now, two generations of devices after the original Soviet tokamaks, the key physical parameters required for a D-T fusion reactor have been demonstrated, including ion temperatures of 35 keV, confinement times of 1.4 sec and required values of β and current-drive efficiency. The first D-T experiments on JET have demonstrated the production of significant amounts of fusion power; D-T experiments in the next few years on TFTR and JET will raise the fusion power into the 10-megawatt range.

A consensus has emerged in the US and the international fusion communities that the field is ready to address the issues of physics and technology that will ultimately determine the economic attractiveness of fusion power. Positive results will make an abundant, environmentally attractive new energy source available to the world at a time when it will surely be needed. We hope that the political resolve to support both large-scale international fusion research and strong national research efforts will follow from the recent technical successes.

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