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REACHING IGNITION IN THE TOKAMAK

By

H.P. Furth

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PLASMA
PHYSICS
LABORATORY

PRINCETON UNIVERSITY
PRINCETON, NEW JERSEY

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REACHING IGNITION IN THE TOKAMAK

H.P. Furth

Princeton University, Plasma Physics Laboratory

P.O. Box 451, Princeton, New Jersey

During the early 1930s, the study of nuclear fusion reactions was at the forefront of high-energy physics. Fifty years later, fusion research has fallen behind by about eight orders of magnitude in particle energy. On the other hand, front-line magnetic-fusion devices, such as the Tokamak Fusion Test Reactor (TFTR) at Princeton (Fig. 1 and Table I) and the Joint European Torus (JET) at Culham, England, will have luminosities of $10^{42} \text{ cm}^{-2} \text{ sec}^{-1}$ -- some ten orders of magnitude beyond present-day high-energy particle storage rings. In these large tokamak devices, the D-T reaction rate is expected to be about 10^{19} events per second. In terms that are more familiar to fusion researchers, the nuclear power release will be about 30 MW -- sufficient to reach or exceed the energy input into the reacting particles.

A logical next step will be to produce a true thermonuclear burn, like that in the sun and stars, where the fraction of fusion energy released in charged particles (Fig. 2) is sufficient to maintain the temperature of the nuclear fuel against energy losses to the cold environment. To reach the ignition point with D-T fuel, the quality of heat insulation -- measured by the product of the central plasma density $n(0)$ times the energy confinement time τ_E -- must exceed $3 \cdot 10^{14} \text{ cm}^{-3} \text{ sec}$ at a central plasma temperature of about 20 keV (Fig. 3).

There is general agreement within the technical community -- as represented by the Magnetic Fusion Advisory Committee (MFAC) -- that the logical vehicle for an ignition experiment would be a next-generation tokamak device. The tokamak is a plasma torus located in a strong toroidal magnetic vacuum field. A toroidally circulating plasma current generates closed poloidal magnetic surfaces around the plasma and also serves to raise the plasma temperature to the 1-3 keV range by ohmic heating. Temperatures in the ignition range could be produced by auxiliary-heating techniques such as ionization-trapping of beams of energetic neutral atoms or damping of radio-frequency plasma waves. (For a more detailed discussion of the physics of burning plasmas and tokamaks see p. 5.)

The US fusion effort is currently pursuing a number of promising lines of research besides the tokamak. A major D-T facility of the mirror-machine type is another interesting option for the future, but its orientation would more likely be towards the development of nuclear technology, rather than towards ignited-plasma studies. The potential for achieving energy multiplication by means of the inertial confinement approach will be explored during the 1980s by a number of projects, including the NOVA laser at Livermore. The expected achievement of 1000-fold pellet compressions would open the way to ignition and propagating burns in a larger next-generation facility.

THE PHYSICS OF BURNING PLASMAS

The energetic alpha-particle population in an ignited D-T plasma will contribute a considerable part of the total plasma

pressure (typically 10-20%). Energetic-ion components with partial pressures in this range can be expected to have substantial impact on plasma stability properties. In this sense, the "physics of burning plasmas" has already begun to be studied: Substantial energetic-ion populations can also be injected in non-D-T experiments with neutral beams, or accelerated within the plasma by rf waves. In some high-powered tokamak experiments, the injected ions are predicted and observed to drive low-frequency magnetohydrodynamic (MHD) oscillations and to escape from confinement in the process¹ (Fig. 4). There are also numerous theoretical possibilities for driving high-frequency oscillations. The excitation of such modes by a thermalizing alpha-particle population is not expected to be very dangerous to the quality of overall plasma confinement, but the fluctuation level could be raised if lower-frequency loss mechanisms, such as that of Fig. 4, cause the alpha velocity distribution to become anisotropic.

Some of the indirect effects of bulk-plasma heating by energetic-ion components are also likely to have some impact on energy confinement in the burning-plasma regime. The maximum plasma pressure in a magnetic confinement geometry is defined by MHD stability conditions that limit the value of beta -- the ratio of plasma pressure nT to magnetic-field pressure $B^2/8\pi$. In tokamaks, a variety of MHD perturbations can become unstable, depending on the detailed shape of the minor cross section of the torus and the radial profiles of the plasma pressure and the rotational transform l (i.e., the angle through which a magnetic field line advances in the poloidal direction when passing once

around in the toroidal direction). Tomographic soft-X-ray diagnostics have shown that the central profiles of the tokamak configuration maintain themselves by a cyclical MHD relaxation process²: slow growth of the peak plasma pressure and transform, typically interrupted at 10-100 msec intervals by a finite-resistivity MHD perturbation that grows to nonlinear amplitude, reconnects the poloidal magnetic flux and cools the plasma center. The understanding and control of this type of relaxation phenomenon is likely to be important in a burning plasma, where the temperature dependence of the D-T reaction cross section favors thermal instability.

Aside from tending to drive the plasma pressure profile to its MHD stability limit, the thermonuclear heat source will play a major role in determining the detailed profiles of the plasma temperature and density, as well as the electrostatic potential in the plasma interior -- factors that are known to influence the local plasma transport coefficients. As D-T plasma particles are transmuted into helium, the need to replace this "fusion ash" with fresh D-T fuel will constitute a challenging new transport problem in itself. While all these phenomena can be addressed by computer modelling techniques, past experience with high-temperature plasma confinement suggests that direct experimental tests will be needed to sort out the relevant physics and find the most practical approaches to a quasi-steady-state burning-plasma regime.

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PLASMA PHYSICS REQUIREMENTS FOR REACHING IGNITION

An irreducible rate of plasma heat and particle transport across magnetic field arises from small-angle Coulomb scattering events between plasma particles moving in different orbits along magnetic field lines. In the case of the tokamak,³ the dominant channel for this "classical" energy loss process is the ion thermal conductivity. The associated ion-energy confinement time τ_{Ei} scales as $I^2 T_i^{1/2} / n$ in the relevant range of high ion temperature T_i . When the tokamak plasma current I is large enough so that the associated poloidal field B_p provides good confinement for the 3.5-MeV alpha particles of the D-T reaction, the same current ($I \sim 2$ MA) is also typically sufficient to raise the classically predicted nT_E value above the ignition condition of Fig. 3. Although the ohmic heating power generated by the plasma current diminishes with rising plasma temperature, the classical expectation is that the temperature scales as $T \propto B_p^2 / n$ -- so that the tokamak plasma can always be ohmic-heated to ignition at sufficiently low plasma density.

From an a priori classical point of view, both TFTR and JET could be ignition machines -- but experimentalists have known since the early 1960s that tokamak confinement is strongly anomalous. Experimental contact with the classical transport model has actually been made in TFTR at low plasma current (0.5 MA), where the ion heat-loss channel dominates. As the current is raised into the 1-2 MA range, the confinement of ion energy becomes better than that of electron energy. This usual tokamak regime is called "anomalous," because the classical transport of electrons is supposed to be very small.

Extensive confinement studies on TFTR are consistent with an empirical scaling law for the total plasma energy confinement time τ_E that had been proposed on the basis of ohmic heating experiments with smaller devices: namely, $\tau_E \propto nR^2 a q(a)$, where R and a are the major and minor radii of the plasma and $q(a)$ is the value of the so-called MHD safety factor $q(r) \equiv 2\pi/l = rB_t/RB_p$ at the edge of the plasma. The observed cubic dependence on linear size implies inversely size-dependent transport coefficients and allows the empirical τ_E scaling to be interpreted in terms of physical models based on low-frequency microinstability theory. (It should be noted, incidentally that the formula is not unique: there are equivalent expressions where the temperature is allowed to appear explicitly.) The exponents in the formula are currently undergoing more precise determination, and the parameter range is being extended.⁴ Maximum τ_E values of 0.4 sec and 0.8 sec have been measured, respectively, at the three-quarters- B_t level (4 T) in TFTR and the full- B_t level (3.5 T) in JET. As indicated in Fig. 3, full ohmic heating performance in both TFTR and JET is expected to be around 3 keV and $n(0)\tau_E \sim 6 \cdot 10^{13} \text{ cm}^{-3} \text{ sec}$.

The partly classical, partly empirical transport model that describes the tokamak ohmic heating experiments is substantially favorable to reaching ignition --- but the optimum strategy is different from the purely classical case. With the empirical scaling, high $n\tau_E$ values are most easily obtained at maximum plasma density. This is illustrated by the Alcator C device at MIT,⁵ which has used very strong magnetic fields ($B_t \geq 10 \text{ T}$) and correspondingly high ohmic power densities to overcome radiation

cooling at high plasma density, up to $n(0) \sim 1.5 \cdot 10^{15} \text{ cm}^{-3}$ and $n(0)\tau_E \sim 8 \cdot 10^{13} \text{ cm}^{-3} \text{ sec}$ (Fig. 3). In this approach, however, the maximum ion temperature has remained below 2 keV -- whereas a threshold of at least 5 keV would have to be reached in order for the thermonuclear self-heating mechanism to begin to take over.

To achieve high plasma temperature, the introduction of non-ohmic heating is clearly helpful. During the late 1970s, the PLT device at Princeton reached $T_i = 7 \text{ keV}$ at $n\tau_E \sim 10^{12} \text{ cm}^{-3} \text{ sec}$, using 2.5 MW of neutral-beam heating.⁶ More recently, the D-III tokamak⁴ at GA Technologies has extended the $n\tau_E$ value to $10^{13} \text{ cm}^{-3} \text{ sec}$ (Fig. 3). TFTR has just begun neutral-beam injection up to the 3-MW level and is expected to apply about 30 MW of auxiliary heating in 1986. The JET device will ultimately have 15 MW of neutral beam, plus 15 MW of ion-cyclotron heating power. TFTR has an additional feature that will be helpful if impurity radiation becomes important in limiting τ_E : The plasma can be displaced in major radius, so as to provide compression-heating ($\geq 50 \text{ MW}$) and improved control of wall-interaction effects. On the other hand, JET has a D-shaped minor cross section, which should permit operation up to higher beta values in case MHD stability is the limiting factor. As indicated in Fig. 3, both machines are designed to reach central ion temperatures of 20 keV or higher. The scaling of $n\tau_E$ in this new parameter range will provide a definitive checkpoint for ignition-machine designs.

The present state of tokamak physics clearly still holds some potential for major surprises. As the magnetic Reynolds number (ratio of resistive to dynamic time scale) rises into the

10^9 range, one may hope for improvements in the character of the resistive MHD modes that have dominated configurational stability in the lower-temperature tokamak regimes. On the other hand, there could be a damaging upsurge of collisionless microinstabilities of the trapped-particle type. The experimental evidence from auxiliary-heated tokamaks is equivocal: An increase in heating power is generally accompanied by a marked reduction in τ_E relative to the pure ohmic heating regime, but good confinement has been partly or wholly restored in many cases by improvements in operating technique -- particularly by careful control of neutral-gas inflow at the plasma edge.⁷ The present results can be interpreted either in terms of a curable sensitivity of τ_E to details of the power-deposition profile, or a fundamental tendency towards (non-MHD) deterioration of tokamak confinement as a function of rising beta value.⁸

Tokamak computer models based on moderately pessimistic points of view indicate that a three-to-sixfolding of the classically required current level for ignition should be sufficient to compensate for transport anomalies. The conceptual design studies carried out to date have tended to concentrate within this plasma-current range -- on the practical basis that reasonably good high-temperature confinement will be a precondition for any actual next step along the tokamak line.

DESIGN STUDIES FOR IGNITION DEVICES

The existence of a potential for reaching ignition in the tokamak configuration had become obvious by the late 1970s, particularly on the strength of plasma-confinement results from

Alcator A at MIT and high-temperature results from the Princeton Large Torus (PLT). The four large tokamak projects⁹ of Table I, which were under construction at that time, offered the prospect of reliable intermediate checkpoints to test the required plasma physics and technology. In 1978, the International Atomic Energy Agency organized the International Tokamak Reactor Workshop to discuss the nature and scope of an appropriate follow-on device -- the INTOR -- which was envisaged as a collaborative project.

Beginning in February, 1979, the INTOR Workshop has been meeting in Vienna on a quarterly basis for multiweek sessions, attended by technical delegations from the US, Europe, Japan, and the USSR. The results have been published by the IAEA in a series of massive and technically detailed reports¹⁰: INTOR Zero Phase (1980) reviewed the scientific and technological data bases, outlined a provisional INTOR concept, and identified critical problem areas. INTOR Phase One (1982) presented a conceptual design for the INTOR device. INTOR Phase Two A, Part I (1983) addressed the critical design issues in greater depth, carried out optimization studies, and outlined a set of specific R&D tasks that would be required in support of an INTOR construction project. Current INTOR activities are directed towards reassessment and further optimization of the conceptual design.

The major parameters and structural elements of the INTOR are outlined in Table II and Fig. 5. The superconducting magnet coils are housed in a common cryostat, which is protected against the fusion neutron flux by a blanket and shield about one meter in thickness. The toroidal geometry permits segments of the vacuum vessel and blanket to be replaced by remote handling

techniques without disassembly of the magnet-coils and cryostat. For initial plasma heating to ignition, neutral-beam injection has been the reference option, as in Fig. 5, but recent advances in heating at the ion cyclotron frequency⁴ have been shifting the balance in favor of an rf-heated INTOR. A single-pull poloidal divertor is the preferred choice to protect against plasma impurities and promote removal of the helium ash.

The overall objective of the INTOR project is not merely to achieve ignited quasi-steady-state operation, but to create a scientific and technological basis for a real demonstration reactor (DEMO). The INTOR plan calls for high-duty-cycle operation, with 600 MW of fusion power and 100-sec burn times. Generation of electric power is proposed only on a small-scale test-module basis, but the expected annual tritium consumption (5-10 kg) is sufficiently large so that in-blanket breeding of a significant fraction of the tritium supply would be a desirable option. The integrated neutron fluence through the first wall is intended to reach the multi-MW-year/m² level, where useful materials-damage information begins to be generated. On the basis of detailed costing studies -- believed to be generally conservative -- the total capital cost of the INTOR project has been estimated at about \$3.7B in 1984 dollars. The construction period, from the time of authorization to the start of experimental work, would be seven years.

The INTOR study phase is generally acknowledged to have been a success. The intellectual challenge of the multinational interaction has helped to set high standards of technical quality, yet has proved compatible with convergence on a common

design. The prospect that the INTOR study would actually turn into a quadrupartite INTOR construction project, however, has been viewed with some skepticism from the outset. Each of the four participating parties has maintained its own independent version of the INTOR design, and the long-range energy plans of Europe, Japan, and the USSR currently include domestic INTOR-scale projects. For example, the FER study in Japan and the NET study in Europe are timed for the initiation of construction projects in 1989 and 1991, respectively.

Five years ago, the United States took a strong lead by enacting the Magnetic Fusion Energy Engineering Act of 1980, which called for a near-term doubling of the overall fusion program, with the Fusion Engineering Device (FED) as the centerpiece of the new effort. Following the recommendations of the Energy Research Advisory Board (ERAB) Fusion Review Panel of 1980, the Department of Energy's Office of Fusion Energy (OFE) initiated an FED project aimed at somewhat more limited objectives than the INTOR, along with a relatively early start of construction (mid-1980s). Its capital cost was expected to be about \$2.5B. Pending selection of the FED site, a Fusion Engineering Design Center (FEDC) was established at Oak Ridge.

During the early 1980s, the transition from energy crisis to budget crisis led to the replacement of the FED by a less demanding concept. The Magnetic Fusion Advisory Committee (MFAC) recommended that near-term efforts should be focussed on a Tokamak Fusion Core Experiment (TFEX) with the primary mission of investigating the physics of quasi-steady-state burning plasmas. The development of "reactor-relevant technology" remained as a

significant secondary goal, but was not to be a major driver of project cost. To help address the nuclear-engineering issues, the use of independent irradiation facilities was considered, including a mirror machine with a modest fusion-power multiplication factor and various non-plasma neutron sources. The TFCX strategy was examined by the ERAB Fusion Review Panel of 1983 and was strongly endorsed by ERAB in February, 1984. The specific recommendation was that OFE should begin to fund the TFCX project within the existing fusion budget level, while also seeking international participation.

The TFCX study group consisted of ANL, FEDC, INEL, LANL, LLNL, MIT, ORNL, and others, with overall coordination by PPPL. Two typical design options developed during the 1983-84 period are illustrated in Table II and Fig. 6. To ensure achievement of the scientific objectives of the TFCX, the plasma size and hypothetical ignition margin (ratio of alpha power to rate of plasma heat loss) were kept comparable to the INTOR design, but the capabilities in the nuclear engineering area were cut back sharply. As a result, the neutron shielding requirements relaxed and the plasma major radius became smaller. Since superconducting coils are believed to be ultimately needed for economical fusion power production in a tokamak reactor, some preference was still given to the use of superconductor technology. The only formal requirement, however, was that the magnet on-time should be sufficiently long so as to allow study of quasi-steady-state conditions: i.e., alpha-particle thermalization, helium plasma buildup, skin-current relaxation, and so forth. As indicated in Table II, this requirement could be satisfied equally well by a

water-cooled copper-coil device of the type labeled "TFCX C," or by a superconducting "TFCX S." The TFCX Preconceptual Design Report issued in June, 1984 described two copper coil and two superconducting coil options, with estimated capital costs in the range \$0.9B-\$1.3B (assuming location at one of the existing fusion research sites).

From the very outset of the ignition-studies activity, there had been a vigorous contest between two equally reasonable -- but contrary -- points of view: (1) A compact, science-oriented device could test the basic viability of the burning-plasma regime at a relatively early time and moderate cost. (2) A high-performance, technology-oriented device would come much closer to a real fusion reactor and would put the required hardware development effort to more fruitful use. The TFCX study sought to find a satisfactory compromise solution, but convergence on a single TFCX design proved difficult to achieve. By the time of the MFAC meeting of July 17-20, the budgetary outlook had further evolved, and OFE concluded that a near-term TFCX project at the \$1B level was no longer appropriate. MFAC then recommended adoption of a two-pronged approach: (1) reconsideration of opportunities for innovative design and maximum use of existing facilities, with a view to lowering the cost of a science-oriented ignition experiment into the range below \$0.5B; (2) design of an attractive technology-oriented machine in the sub-INTOR size-range, to be constructed as a multinational project.

During the current year, option (1) is being addressed by a set of design studies for compact copper-coil ignition devices, carried out independently by GA Technologies, MIT, ORNL, and PPPL. One promising approach is to shorten the burn-time requirement and raise the magnetic field strength. The cost-effectiveness of the Alcator high-field tokamak experiments at MIT has stimulated a number of ignition schemes over the years: notably the Ignitor¹¹ and the LITE¹² designs, which correspond loosely to the "SCC" and "CC" entries in Table II and Fig. 6. Recent tokamak successes in augmenting ohmic heating with rf heating⁴ have improved the technical outlook for achieving ignition temperatures along this line. While the lack of reactor-relevance of high-field copper-coil technology is almost¹³ universally accepted, it should not be a disqualifying factor in the choice of a science-oriented ignition experiment. The key issue in determining the appropriate field strength and magnet-cooling technique is the pulse length needed to achieve meaningful quasistatic conditions.

A desirable alternate approach to the reduction of machine cost, both in an ignition experiment and in subsequent power reactors, is to raise the MHD-stable beta value. Tokamak experiments with mildly elongated plasmas in the D-III device have already demonstrated stability up to the 4.5% level,¹⁴ in fairly good agreement with MHD code predictions. Based on similar calculations, the TFCX studies envisaged betas of about 6% in D-shaped plasmas (Fig. 6). According to MHD theory, there are two avenues for raising the tokamak beta value into a substantially higher range: strong vertical elongation with low

aspect ratio R/a , and "bean shaping" of the minor cross section with moderately large aspect ratio (cf. Fig. 7). Stable beta values in the 20% range or better are theoretically possible in either case, and the moderate-aspect-ratio approach would be particularly convenient from the point of view of future reactor design. Tokamak "beta-pushing" experiments currently in progress (which are passing the 5% level with the bean shape) will support a clearer assessment of the prospects along this line.

Design work for the technology-oriented option (2) is being done mainly at the FEDC, with participation of LLNL and coordination by MIT. A number of superconducting coil designs are being investigated. In this case, the possibilities for a major size reduction are more limited, because of the minimum requirements on shielding thickness inside the coils. For a low-fluence ignition experiment, the most critical consideration is the pulsed heating of the superconductor by the fusion neutron flux. Considerable progress towards a minimum-sized device has been made by introducing forced liquid-helium cooling and high-density shielding materials. Increasing the plasma beta value also offers cost savings in superconducting-coil ignition machines, but the impact is not as strong as for copper coils. Particularly in the event that some major multinational cost-sharing arrangement can be negotiated, the optimization process is more likely to progress in the direction of enhanced performance: longer pulses, higher duty cycle, and neutron fluences that begin to rise back towards the INTOR level. The US INTOR team is providing close linkage between the FEDC studies and the multinational design group in Vienna.

PROSPECTS FOR AN IGNITION PROJECT

While we have no assurance as yet that the new data from the TFTR generation of large tokamaks will open the door to an ignition experiment, there is a sense of favorable expectation in the plasma physics community. During the past two decades, maximum energy confinement times in the tokamak have moved from a few milliseconds to 0.8 sec, relative to an ignition target around 1-2 seconds; n^*_E values have moved from a few times $10^{10} \text{ cm}^{-3} \text{ sec}$ to $8 \cdot 10^{13} \text{ cm}^{-3} \text{ sec}$, relative to a target value of $3 \cdot 10^{14} \text{ cm}^{-3} \text{ sec}$; central ion temperatures have moved from a few hundred eV to 7 keV, relative to a target of 15-20 keV. Unless there is a sharp discontinuity in these trends, the simultaneous achievement of the full set of ignition parameters looks like a reasonable goal for the mid-1990s. Encouragement can also be drawn from a more sophisticated argument: Front-line tokamak experiments are now stimulating a rapid growth in our physical understanding of plasma confinement phenomena -- which should serve us well in guiding future experimental progress.

Before a specific next-step proposal is made, the priority issue between the two basic types of ignition projects will have to be resolved. International collaboration may help to bring this problem to a constructive solution. There are obvious advantages in an overall strategy that combines a science-oriented pilot project with a high-performance technology project developing in parallel on a somewhat slower time scale. A bilateral or multinational effort would have the resources to pursue this program in a timely fashion: The smaller project could start in 1987-88, with facility completion in 1992-93.

Operating experience in the burning-plasma regime would begin to be gained on a low-duty-cycle basis in the mid-1990s. Construction of the larger project could start in 1990-91, with facilities completion in 1997-98. A high-duty-cycle operating mode approximating that of a real fusion reactor could begin around the turn of the century.

While the success of fusion power is rightly said to be "always twenty years away," one must keep in mind that a great deal of progress has been made in the definition of the word "success." During the infancy of the fusion program, passage through the break-even point was considered roughly equivalent to accomplishing the program goals. Now that a break-even demonstration is scheduled (1988 in TFTR) and even an ignition demonstration looks promising in the fairly near term, the definition of success has evolved to include engineering optimization; by the year 2005, it may acquire its full commercial significance. If this rate of progress can be sustained, then by the year 2025 the standards for the success of fusion power will be set at a very high level: readiness to support a significant part of the world's energy needs.

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Table I

	TFTR	JET	JT-60	T-15
Location	Princeton US	Culham UK	Tokai-Mura Japan	Moscow USSR
Start of Experiment	December 1982	July 1983	March 1985	1986
Major Radius R(m)	2.50	2.96	3.00	2.40
Minor Radius a(m)	0.85	1.25	0.95	0.70
B _t (T)	5.2	3.5	4.5	5.0
I (MA)	3.0	5.0	2.7	2.3
Aux. Heating (MW)	30*	30	30	15
Working Gas	H, D, D-T	H, D, D-T	H, D	H
Special Features	Adiabatic Compression	D-Shape, Ion Cyclotron Heating	Divertor, Lower Hybrid Heating	Super- conducting Coils, Electron Cyclotron Heating

*Not including compression.

Table II

	INTOR	"TFCX S"	"TFCX C"	"CC"	"SCC"
Major Radius R	5.2 m	4.1	2.6	1.5	1.0
Minor Radius a	1.2 m	1.5	1.0	0.6	0.4
B_t	5.5 T	3.7	4.5	8.0	12.7
I	6.4 MA	11.2	10.4	10.0	10.0
β	5.6%	5.5	6.0	5.5	5.0
Burn Time	100 sec	620	300	10	2
Fusion Power	620 MW	270	200	320	300
Duty Cycle	70%	3	3	2	1
Fluence Goal	3 MWa/m ²	---	---	---	---
"Ignition Margin:"					
Class. $\propto a^{3/2} R^{1/2} B_p^2$	16	40	34	32	32
Emp. $\propto a^2 R B_p^3 B_t^3 \beta^2$	3.7	2.0	1.6	2.5	3.9

FIGURE CAPTIONS

1. Vacuum vessel of the TFTR showing installation of movable limiters.
2. Deuterium-tritium fusion reaction.
3. Tokamak experiments are progressing towards the fusion reactor regime. A non-Maxwellian ion energy distribution, such as that produced by neutral-beam injection, facilitates the achievement of break-even. The full-performance ohmic-heating point for TFTR and JET can be projected with some confidence; the scaling of nT_E with auxiliary heating remains to be explored.
4. The precession of neutral-beam-injected ions in the PDX tokamak drives a synchronous MHD kink mode ($m = 1, n = 1$). In this process, the energetic ions escape (charge-exchange signal) and the neutron emission drops. The time scale is shown expanded on the right side of the figure.
5. The INTOR is a long-pulse, high-duty-cycle ignition machine, with extensive reactor-engineering features.
6. Plasma cross sections for various conceptual ignition devices.
7. The plasma beta value can be raised relative to the standard tokamak design by bean-shaping or by vertical elongation at low R/a . The option of still greater elongation at intermediate R/a is also being studied.

Figure 1

THE D-T FUSION REACTION

(1 eV = 12,000 Degrees K)

Figure 2

Figure 3

Figure 4

INTOR

June 1981

Figure 5

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Figure 6

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Figure 7

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