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Fusion Ignition Experiment

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Abstract

A fusion ignition experiment (IGNITEX) is described, the original concept for which was proposed by Rosenbluth, Weldon, and Woodson. In this concept, a single-turn-coil tokamak device produces a self-sustained fusion reaction. The basic idea is to employ a very high magnetic field and a very high plasma current to heat the plasma ohmically to thermonuclear temperatures and then to produce a stable ignited plasma. The experiment will permit the scientific study of a new regime of physics: alpha-heated plasmas.

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BACKGROUND

For a self-sustained thermonuclear fusion reaction to take place in a plasma of deuterium (D) and tritium (T) confined magnetically, a mixture of a density of $n \sim 10^{20}$ particles/m^3 must be heated to thermonuclear temperatures, $T \sim 10^8$ °C. Under these conditions the power deposition in the plasma caused by fusion reactions can exceed the power lost from the system by radiation, convection, and conduction if most of the energetic alpha particles produced in the fusion reaction $T(d, n)^4He$ deposit their energy (3.5 MeV [megaelectronvolts]) into the plasma and the energy confinement time is $\tau_E \sim 1$ sec. The thermonuclear plasma is essentially transparent to the neutron radiation (14.1 MeV of energy), and then, the fusion chain reaction has to be maintained by the alpha particles. Controlled ignition is a plasma state in which fusion reactions occur at a steady rate without need for external heating and without explosive or disruptive processes. In an ignition experiment, physics and engineering considerations will constrain the speed at which ignition conditions can be attained.

A thermonuclear system will ignite when $\rho \geqslant 1$ (ρ is the ignition factor, defined as the quotient between the power deposited in the plasma by the alpha radiation and the total power lost from the plasma at the same instant of time). Since $\rho = \rho(n, T)$, the ignition condition defines a line in a density-temperature diagram of plasma operation above which ignition conditions are attained. When a plasma reaches ignition, fusion reactions occur at a fast rate and the plasma thermal energy increases rapidly. If this thermal excursion proceeds uncontrolled, the plasma will self-destruct when its pressure reaches the stability limits. Therefore, a fusion ignition experiment by magnetic confinement needs first to heat a plasma to thermonuclear temperatures and produce a self-sustained reaction and then to control the thermal instability sufficiently to maintain a stable ignited phase of the discharge.

A particular embodiment of a fusion ignition experiment (called IGNITEX) is considered in this report. The original concept was proposed by Marshall N. Rosenbluth, William F. Weldon, and Herbert H. Woodson [1] on the basis of Bruno Coppi's ideas for a compact fusion experiment [2] (also see Physics Today, May 1981, p. 17) and recent technological advances in pulsed-power systems with high currents. The IGNITEX concept proposes a simple experiment to produce a self-sustained fusion reaction. The proposed magnetic confinement device is a single-turn-coil tokamak with a 20-Tesla magnetic field on the plasma axis and plasma currents in excess of 12 megamperes (MA). The idea is to employ a very high magnetic field and a very high plasma current to heat the plasma ohmically to thermonuclear temperatures and then to produce a stable ignited plasma.

Among magnetic confinement schemes, tokamaks are at present thought to have the most immediate prospects of reaching fusion ignition. (The reader may refer to the article by H. P. Furth in the March 1985 issue of Physics Today.) High-field tokamaks have very successfully contributed to fusion research [3]. The ohmic regime of operation is the best known and simplest of the tokamak modes of operation; it has produced record values of energy confinement. In tokamaks, high plasma current is desirable because it improves energy confinement, stability, heating, and (possibly) alpha containment. The IGNITEX experiment is proposed to operate at low beta (β is the ratio of the plasma pressure to the magnetic pressure). Thus, the discharge will be far from stability limits. In addition, this feature makes possible the control of the thermal instability by cyclotron radiation emission. As shown later, the margin for ignition in this device is predicted to be very high on the basis of present knowledge.

THERMAL ENERGY PRODUCTION AND TRANSPORT

The study of a chain reaction in a fissionable system requires the solution of the equation of neutron transport because the number of neutrons is the essential quantity for the physical process of nuclear fission. In a fusionable system, on the other hand, the kinetic energy of the particles involved is the relevant physical quantity in the evaluation of the probability to produce a self-sustained thermonuclear reaction. The dominant processes of the thermal energy balance in a thermonuclear plasma confined in a tokamak with no auxiliary heating are: alpha heating, ohmic heating, electronic radiation emission by free-free interactions (bremsstrahlung radiation of soft X-rays) and by gyromotion (cyclotron radiation of microwaves), and conductive and convective energy transport. The ion and electron populations are coupled by the condition for plasma quasi neutrality and through Coulomb collisions of long range. Magnetohydrodynamic (MHD) fluctuations with very long wavelengths along field lines can strongly modify the plasma equilibrium and lead to fast plasma loss. These well-known instabilities can be avoided by imposing certain constraints on the tokamak plasma discharge, namely:

- The plasma pressure must be such that the beta value ($\beta = 0.8 \times 10^{-21}\ nT/B^2$, where n is in m^{-3}, T is in keV, and B is the magnetic field in Tesla) stays below the limits for stability of ballooning and external kink modes. (Ballooning modes are driven by the plasma pressure; they concentrate in the outside of the torus where the field lines are convex to the torus. External kink modes are driven by the plasma current and helically perturb the plasma-vacuum boundary.) The Troyon limit [4] (obtained after detailed numerical calculations) gives a reliable critical beta value; it can be written as 0.028 I/aB_t, where I is the plasma current (MA), a the minor radius (m) (see Figure 1), and B_t the toroidal magnetic field on axis (T).

- The plasma density must remain below disruptive limits. (Disruptions are gross instabilities accompanied by abrupt plasma temperature drop, expansion of the plasma column, negative loop voltage spike, and other characteristic effects, followed by the termination of the discharge.) The Murakami-Greenwald limit [5] (obtained empirically) is $0.75 \times 10^{20}\ \kappa\ \bar{J}\ (m^{-3})$, where κ is the elongation of the plasma cross-section and $\bar{J}$ is the average plasma current density (MA/m^2).

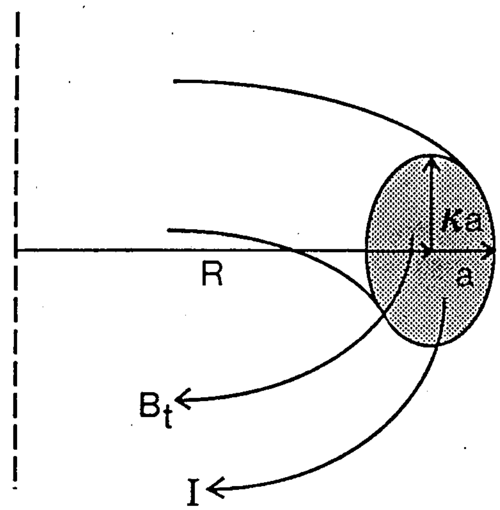


Figure 1. Sketch of a tokamak plasma indicating the major radius (R), the minor radius (a), the toroidal component of the magnetic field (B_t), and the toroidal component of the plasma current (I). The length of the major axis of the elliptical plasma cross-section is κa where κ is the elongation.

- The plasma current density on the magnetic axis must be below the Mercier limit [6] (necessary for theoretical stability of interchange modes, which are pressure-driven internal models, similar to the Rayleigh-Taylor instabilities of fluid dynamics, in which the plasma interchanges position with the magnetic field). This condition is equivalent to having an MHD safety factor on magnetic axis greater than unity. Empirically it has been established that an MHD cylindrical safety factor at the plasma edge, $q(a) \gtrsim 2$, is needed for non-disruptive tokamak operation, where $q(a) = 2.5\, a^2\, B_t\, (1 + \kappa^2)/RI$ and R is the major radius (m) (see Figure 1).

- The total toroidal current in the plasma must be lower than the Kruskal-Shafranov limit [7] for stability of free-surface modes (in which the plasma-vacuum boundary is moved from its equilibrium position by an MHD perturbation); this condition is equivalently satisfied when $q(a) > 1$. This theoretical criterion ultimately gives the basis for proposing a high-field tokamak: the larger the toroidal magnetic field, the larger the current that can be driven through the plasma without destroying it.

- The elongation of the plasma cross-section must be moderate so that macroscopic modes do not become unstable. Specifically, the plasma will be subjected to a gross vertical instability during high elongations. An elongated plasma is otherwise desirable because the toroidal current-carrying capacity of the plasma is then larger than in comparable circular discharges. Stable discharges are routinely obtained with elongations $\kappa \lesssim 1.8$.

We now consider the thermal energy balance in a plasma which, for simplicity, we will assume to be of a conducting fluid with a Maxwellian distribution of D-T particles (equal D-T densities) with mean kinetic energy 3/2T. We assume that the fusion-product ions have sufficient time to transfer their energy to the electrons and to the fuel ions by Coulomb collisions, and that the ion and electron transport are similar, so that the various species' temperatures are equal. The nuclear reaction $T(d, n)^4H_e$ is considered to be the dominant fusion reaction. The plasma is assumed to be fully ionized (thus, radiation losses due to atomic excitation and recombination can be neglected); the mean ionic charge of the plasma is assumed to be 1.5 because of the presence of impurities (this level of discharge "cleanness" is routinely obtained in high-field tokamaks).

If perfect containment of the alpha particles is attained, the plasma heating due to fusion reactions can be written as

$$P_{FUS} = 1.4 \times 10^{-19}\ \overline{\langle\sigma v\rangle\ n_i^2}\ (\mathrm{MW/m^3})$$

where $\langle\sigma v\rangle$ is the spectrum-averaged fusion cross-section ($\mathrm{m^3/sec}$) and n_i is the ion density ($\mathrm{m^{-3}}$). The volume average of the quantity $f(r)$ is defined as $\overline{f} = (2/a^2\kappa)\int_0^{a\sqrt{\kappa}} f(r)rdr$ in the cylindrical limit of an axisymmetric toroidal plasma column; r is the radial variable in a poloidal cross-section of the plasma. A typical parabolic density profile is considered $n(r) = n_0(1 - r^2/a^2)$. A "consistent" temperature profile [8] is also considered, $T(r) \sim J^{2/3}(r)$ with $J(r) = J_0[1 + (r/r_J)^6]^{-4/3}$ and $r_J = 0.68a$. The principle of profile consistency [8] states the experimentally observed invariance of the radial profile of the electron temperature to changes in plasma conditions including the strength and shape of the heating profile, thus implying that the observed profiles are the result not only of microscopic local energy transport processes but of stronger and faster macroscopic equilibrium and stability constraints. Some sawtooth flattening of the central temperature distribution is implicit in the profile considered here. (Sawtooth activity in tokamak physics refers to characteristic relaxation oscillations near the magnetic axis which typically manifest themselves as internal helical structures of the electron temperature.) Absolute alpha containment is assumed when the plasma current is greater than a critical value obtained from neoclassical orbit theory [9], $I_\alpha = 7.5\ \kappa/A$ (MA) where $A = R/a$ is the plasma aspect ratio.

The ohmic heating contribution to the energy balance is given by

$$P_{OH} = \overline{\eta J^2}.$$

The plasma resistivity, η, is obtained from classical Coulomb collisional theory. A complicated expression that includes neoclassical toroidal effects like particle trapping and accounts for electron friction with impurities is obtained in Ref. 10 and is used here.

The total rate of plasma energy loss by conductive, convective, and radiative processes is given by Ref. 11 as

$$P_E = \frac{\overline{U}}{\tau_E}$$

where $\overline{U} = \frac{3}{4\mu_o} \overline{\beta}_t B_t^2$ is the plasma's stored thermal energy per unit volume, $\overline{\beta}_t$ is the volume-averaged toroidal beta value, and

$$\tau_E = \left(\tau_{E,OH}^{-2} + \tau_{E,AUX}^{-2}\right)^{-1/2}$$

is the global plasma energy confinement with $\tau_{E,OH} = 7.0 \times 10^{-22} \overline{n}aR^2$ (sec) being the ohmic scaling (neoalcator scaling [12]) that reproduces low-density experimental results in the ohmic regime; and

$$\tau_{E,AUX} = 3.5 \times 10^{12}\ I^{2.95}\ R^{2.54}/\kappa^{0.71}\ B_t^{0.21}\ a^{3.92}\ \overline{n}^{0.76}\ \overline{T}^{1.38}$$

being the auxiliary heating scaling [11]. This empirical expression for $\tau_{E,AUX}$ was obtained by regression analysis of an extensive data base of tokamak discharges with neutral-beam injection.

This formula satisfies the Connor-Taylor dimensional arguments that restrict the physical scaling laws to those invariant under any transformation that leaves the basic plasma equations invariant. The inverse quadrature relation of τ_E to $\tau_{E,OH}$ and $\tau_{E,AUX}$ seems to represent accurately the available data base of tokamak experiments, although no adequate theoretical basis has been given so that extrapolation into new regimes of plasma operation is required. We assume here that the Kaye-Goldston empirical scaling (given above) describes the global energy confinement in the plasma with ohmic and alpha heating powers degrading energy confinement as auxiliary heating does in present experiments with neutral-beam injection (and/or any form of wave heating). This assumption is required because the physics of alpha-heated plasmas is not presently known, as no experiment has yet operated in that regime.

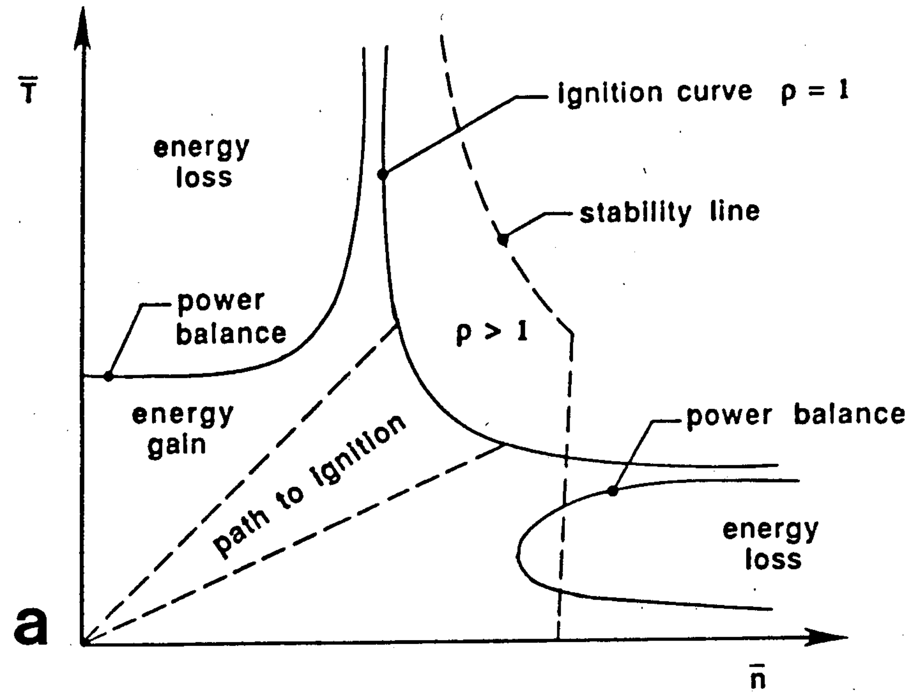
The present understanding of energy confinement in hot plasmas is based on empirically obtained global energy confinement time scalings (rather than on detailed local transport theories). Thus a volume-averaged calculation of the thermal energy balance seems to be a reasonable approach to the prediction of the ignition margin of a fusion ignition experiment.

EVALUATION OF THE IGNITION MARGIN

The results of solving the equilibrium equation for the volume-averaged rate of production of thermal energy in a thermonuclear toroidal plasma column,

$$P_{FUS} + P_{OH} = P_E,$$

can be conveniently represented in a diagram $(\overline{n}, \overline{T})$. A conceptual representation of a solution in which ohmic ignition is possible is given in Figure 2a. The locus of the roots of the power balance defines a region of energy gain, where $P_{FUS} + P_{OH} > P_E$ (i.e., a plasma within that zone will heat up as long as the ohmic heating is maintained). The line $\rho(\overline{n}, \overline{T}) = 1$ is the ignition curve. The region above this line is the ignition zone, where $P_{FUS} > P_E$. The disruptive stability limits for the plasma beta (Troyon) and the plasma density (Murakami-Greenwald)



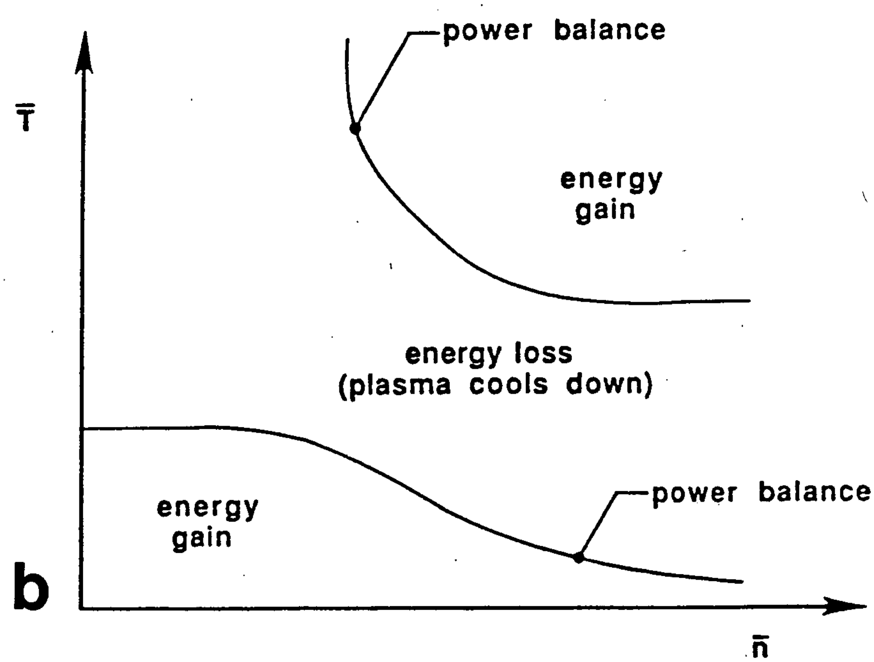


Figure 2. Conceptual representation of the locus of the roots of the equation for the volume-averaged thermal energy balance in a thermonuclear plasma confined in a tokamak: a) when ohmic ignition is possible; b) when ohmic ignition is not possible.

define a zone within the ignition region where stable ignited plasmas are predicted to be produced. A steady self-sustained fusion reaction is possible for plasmas located in this zone if a mechanism to control the thermal instability exists.

The IGNITEX experiment is designed to reach the zone of plasma ignition and stability and stay there during most of the flat-top phase of the discharge, thus producing a stable ignited plasma. A path to ignition with an appropriate plasma heating rate is shown later in a time-dependent simulation of the discharge. In order to maximize the probability of reaching and controlling ignition, an ignition region as wide as possible is desirable. A measure of this "width" can be obtained by defining the ignition margin, $\Delta\rho$, as the ignition factor evaluated at marginal stability. This definition makes $\Delta\rho$ dependent on temperature. Here, $\Delta\rho$ is evaluated (arbitrarily) for a temperature on axis of 12 keV (kiloelectronvolts). We note that $\langle\sigma v\rangle/T^2$ is maximum at $T \approx 12$ keV and therefore, for fixed beta and magnetic field values the fusion power density of a fusion system with deuterium and tritium is maximum at about 12 keV. For completeness, we give in Figure 2b a representation of the $(\overline{n}, \overline{T})$ diagram when ohmic ignition is not possible. In this case, a gap develops between the high- and low-temperature energy-gain regions. A plasma located in this gap will cool down because the induced ohmic heating is not sufficient to reach thermonuclear temperatures.

The plasma of the IGNITEX single-turn-coil tokamak [13] has a major radius R = 150 cm, minor radius a = 47 cm, cross-section elongation $\kappa = 1.6$, toroidal magnetic field on axis $B_t = 20.2$ T, and plasma current

$I = 12$ MA. Large dimensions are desirable in order to improve energy confinement during the approach to ignition conditions and to support the thermal and mechanical stresses in the inner leg of the single-turn coil. However, size is constrained by cost considerations. The volume of the plasma chamber in IGNITEX is 10.5 m^3, and the first-wall area is 39.5 m^2. In the IGNITEX device, ignition is possible by ohmic heating alone when magnetic fields of approximately 15 Tesla or greater on axis are produced. Ignition can occur with density $\overline{n} = 3.6 \times 10^{20}\ m^{-3}$ and temperature $\overline{T} = 6.5$ keV. Then, the plasma pressure is $\overline{p} = 9.9$ atm, the toroidal beta is $\overline{\beta}_t = 0.6\%$, and the poloidal beta is $\overline{\beta}_p = 15.5\%$. The resistive loop voltage is $\overline{V}_{loop} = 0.7$ volts, and the current density is $\overline{J} = 11.0\ MA/m^2$. At the time of reaching ignition, the total fusion power produced is 149 MW (including alpha particles plus neutrons), the neutron production rate is 5.3×10^{19} neutrons/sec, and the neutron wall load is 3.0 MW/m^2 (assuming uniform deposition on the plasma chamber wall). The alpha gyroradius on plasma axis is $\rho_\alpha = 0.03a$, and the alpha containment factor has a high value, $I/I_\alpha = 3.2$. About 6 MW and 30 MW of ohmic and alpha heating, respectively, will be produced at this point in the discharge. The global energy confinement time is $\tau_E = 0.54$ sec, the Lawson product is $n_o\tau_E = 3.9 \times 10^{20}\ sec/m^3$, and the McNally product is $T_o n_o \tau_E = 4.7 \times 10^{21}\ keV/m^3/sec$. The ignition factor is $\Delta\rho = 3.4$.

Plasma equilibrium calculations indicate that the last magnetic surface is shifted outward about 3.0 cm with respect to the magnetic axis at the time of reaching ignition. The MHD safety factor is approximately unity on axis, it is rather flat within the $q(r) = 1$ rational surface [$r(q = 1) \simeq 0.37a$], and it has an edge value $q(a) = 2.2$ [which corresponds to $q_\psi(a) = 2.72$ where q_ψ is the MHD safety factor, defined as the derivative of the toroidal magnetic flux with respect to the poloidal magnetic flux].

The energy confinement in ignited plasmas is not yet known. This is one of the issues to be addressed by an ignition experiment. The neoclassical transport theory [14] includes geometrical effects that increase the basic classical transport by Coulomb collisions. This well-developed theory gives ignition margin predictions about one order of magnitude larger than the empirical Kaye-Goldston scaling. Moreover, empirical scalings themselves give rather disparate predictions. One limit example is the Mirnov scaling [15] in which $\Delta\rho = 14.0$. Another limit example is the Goldston scaling [16] that predicts $\Delta\rho = 1.1$ for the IGNITEX experiment if the plasma is elongated to $\kappa = 1.8$ (reducing the minor radius to $a = 42$ cm), the density distribution is peaked (by pellet injection, for instance) such that the parabolic exponent is 2.5, and, finally, $q(a) = 2.0$ is considered.

Flat density profiles (with parabolic exponents of 0.5 or less) can prevent ignition because the fusion energy production is lowered substantially and because excitation of ion mixing modes (electrostatic sound waves driven unstable by the gradient of the ion temperature transverse to the magnetic field) is possible [17]. A larger elongation and triangularity of the plasma cross-section as well as a lower edge safety factor permit more than 12 MA to be carried by the plasma (the poloidal field system in IGNITEX has been designed to induce up to 14 MA), which increases the predicted ignition margin.

The ignition margin can be increased even further by making the density profile more peaked in the plasma axis [18], because this change is expected to increase reactivity and confinement. A separatrix configuration of the equilibrium magnetic surfaces may produce a hot, low-density plasma edge during the alpha heating phase of the discharge. Thus energy confinement may increase [19]. Using its elongation coils, the IGNITEX device can produce an equilibrium configuration of the magnetic field with two X points. An ignited phase in the so-called H-mode of operation [19] may then be possible. The ultimate increase of ignition margin by this approach in a compact ignition experiment is not yet clear because of the plasma current reduction associated with separatrix configurations. Wall preconditioning may be also important to increase energy confinement [20].

NUMERICAL SIMULATION OF IGNITION DISCHARGES

The calculations discussed so far are based on a steady-state analysis of the volume-averaged thermal energy balance in a thermonuclear plasma confined in a tokamak device. As a result, the possibility for the ohmic ignition in the IGNITEX experiment has been shown, and a large ignition region in $(\bar{n}, \bar{T})$ space has been identified. We are now interested in solving the plasma dynamics equations to study the path to ignition and the control of the thermal instability in the ignited phase of the discharge. We consider here a system with two conducting fluids, ions and electrons, having temperatures T_i and T_e, respectively. The ion and electron energies are calculated along the discharge by solving the approximate energy balances:

$$\frac{\partial}{\partial t}\left(\frac{3}{2}\overline{n_i T_i}\right) = P_{\alpha i} + P_{ei} - P_D - P_{Ei},$$

and

$$\frac{\partial}{\partial t}\left(\frac{3}{2}\overline{n_e T_e}\right) = P_{\alpha e} + P_{OH} - P_{ei} - P_{RAD} - P_{Ee}.$$

$P_{\alpha i,e} = \overline{\frac{n_i^2}{4} <\sigma v> E_{\alpha i,e}}$ are the rates of alpha particle energy deposited in ions and electrons, respectively [21]. P_{ei} is the rate of energy deposited in the ions because of the electron thermal equilibration [22]. $P_D = \frac{3}{4}\overline{n_i^2 <\sigma v> T_i}$ is the power lost by fusion depletion of ions. P_{OH} is the ohmic heating power (deposited mainly in the electrons) given before.

The term P_{RAD} represents the power emitted by the plasma because of bremsstrahlung and cyclotron radiation of the electrons. Note that the plasma is not in radiative equilibrium. The rate of bremsstrahlung energy emission including quantum mechanical effects is

$$P_B = 5.4 \times 10^{-43} Z_{eff} \overline{n_e^2 T_e^{1/2}} \ (\mathrm{MW/m^3})$$

where n_e is in m^{-3}, T_e is in keV (in this expression the Gaunt factor has been evaluated at high temperatures), and $Z_{eff} = \Sigma n_i Z^2/n_e$ is the mean ionic charge of the plasma. The sum in Z_{eff} refers to all the ion species, Z is the atomic number of the various ion species, and

$n_e = \Sigma n_i Z$ (quasi-neutrality condition). It is assumed that the bremsstrahlung radiation is absorbed completely by the vessel wall. The cyclotron emission is [23]

$$P_C = 1.74 \times 10^{-17} B_t^{5/2}\left[(1 - \rho_W)/a\right]^{1/2} \overline{n_e^{1/2} T_e^{10/4} (T_e^{1/2} + 18.0\ a/R)^{1/2}}$$

with the usual units. This formula includes the effects of line broadening due to magnetic field inhomogeneity and Doppler shift. The wall reflectivity, ρ_W, depends on the wavelength of the dominant harmonics emitted by the plasma, the material structure of the wall, and the geometric details of the chamber. Here, for simplicity, we assume $\rho_W = 0.8$. The basic conclusions of our calculation are not fundamentally altered by considering more elaborate reflectivity expressions.

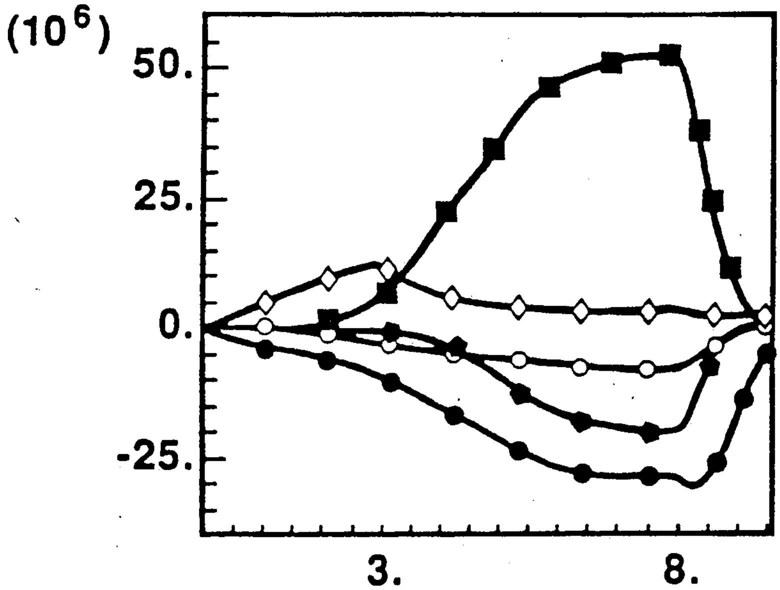
The terms $P_{Ei,e} = \frac{3}{2} \frac{\overline{n_{i,e} T_{i,e}}}{\tau_{Ei,e}}$ represent the rates of energy loss ions and electrons, respectively, due to conduction and convection. Here we will consider that the electron transport follows the neoalcator empirical scaling [12] and the ion transport follows the auxiliary heating component of the Kaye-Goldston scaling [11]. Energy confinement increases with density before ignition and saturates during the ignited phase. Ion and electron transport are competitive at ignition.

The slowing-down time of the alpha particles is much shorter than the energy confinement time. Then, we can assume that these particles deposit their energy from the fusion reactions instantaneously into the plasma. The description of the thermonuclear system includes equations for the variation of the number of fuel and impurity ions and alpha particles in addition to the equations given earlier. The particle confinement time can be approximated by the energy confinement time in high-field tokamaks [3]. The electron density is obtained from the condition for plasma quasineutrality. Radial profiles of densities, temperatures, and plasma current are considered as given earlier, and they are assumed not to change during the discharge. The system of equations just described is volume averaged (the plasma volume is assumed constant with time) and solved subject to the stability constraints imposed on the equilibrium solutions obtained earlier.

In the IGNITEX experiment the ignited plasmas will be thermally stable. The mechanism for stabilization is the following. The electron temperature will increase because of the intense ohmic heating. Then, the ion temperature will increase because of the collisional coupling to the electrons. Therefore, alpha particles will be produced. They will supplement the ohmic heating to the electrons when the plasma conductivity becomes significantly high. Because of the steep temperature dependence of the nuclear fusion reaction cross-section, the alpha heating of the plasma will increase very fast once ignition conditions have been attained. At that point, the electron temperature increases rapidly. For high electron temperatures, the intensity of the electron cyclotron radiation emission in the high harmonics becomes important. The plasma is rather transparent to this radiation, and the cyclotron losses increase strongly. The radiation cooling of the plasma damps the thermal excursion. The discharge proceeds far from the disruptive limits to the end of the ignited phase.

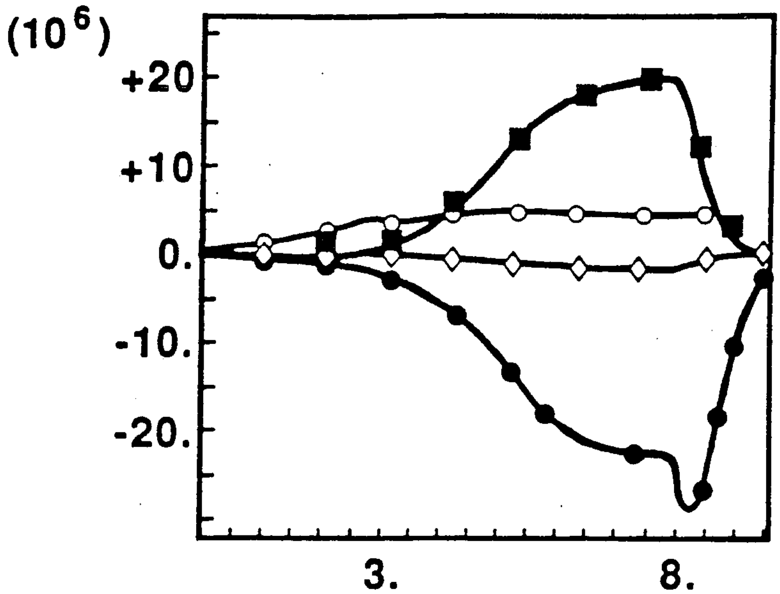
Some results of numerical simulation of a typical discharge in the fusion experiment (IGNITEX) are given in Figures 3 to 8. The fields and currents are considered to be ramped to their full value during the first 3 seconds of the discharge. After a flat-top of 5 seconds, the plasma is shut down in 2 seconds. Our simulation assumes that the plasma density can be decreased appropriately during the shut-down phase. Discharges with somewhat longer pulses may be possible in IGNITEX. These might be needed if longer ignited phases and softer terminations turn out to be desirable. Ignition is reached shortly after full plasma current has been established in the discharge. It can be seen in Figures 3 and 4 that the alpha heating produced by the fusion reactions becomes an important heating mechanism before ohmic heating decreases substantially as a result of the high conductivity of hot plasmas.

Figure 5 shows that the plasma temperature increases very rapidly after ignition. Thus, the plasma disruptive limits may easily be exceeded. If a disruption takes place, the plasma column will be destroyed. In addition to the loss of the whole discharge, the device may be damaged (first wall, vacuum vessel, and so forth). However, as explained before, this thermal excursion is damped in the IGNITEX experiment. Before ignition, bremsstrahlung dominates cyclotron emission. During the ignited phase of the discharge, however, the cyclotron radiation is much more important than bremsstrahlung because



AVERAGE ELECTRON POWERS (WATTS) -vs- TIME (SEC)
■ = FUSION ● = ELECTRON ENERGY CONFINEMENT
O = BREMSSTRAHLUNG ⬟ = CYCLOTRON
◇ = OHMIC HEATING

Figure 3. Electron power balance contributing terms.



AVERAGE ION POWERS (WATTS) -vs- TIME (SEC)
■ = FUSION ● = ION ENERGY CONFINEMENT
O = ION-ELECTRON EQUILIBRATION
◇ = LOSS DUE TO FUSION REACTIONS

Figure 4. Ion power balance contributing terms.

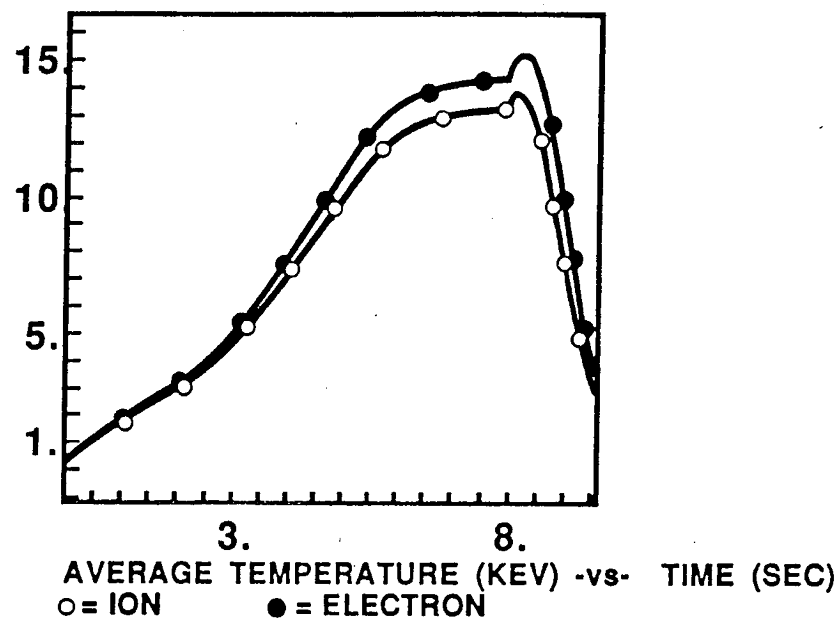


Figure 5. Ion and electron temperatures.

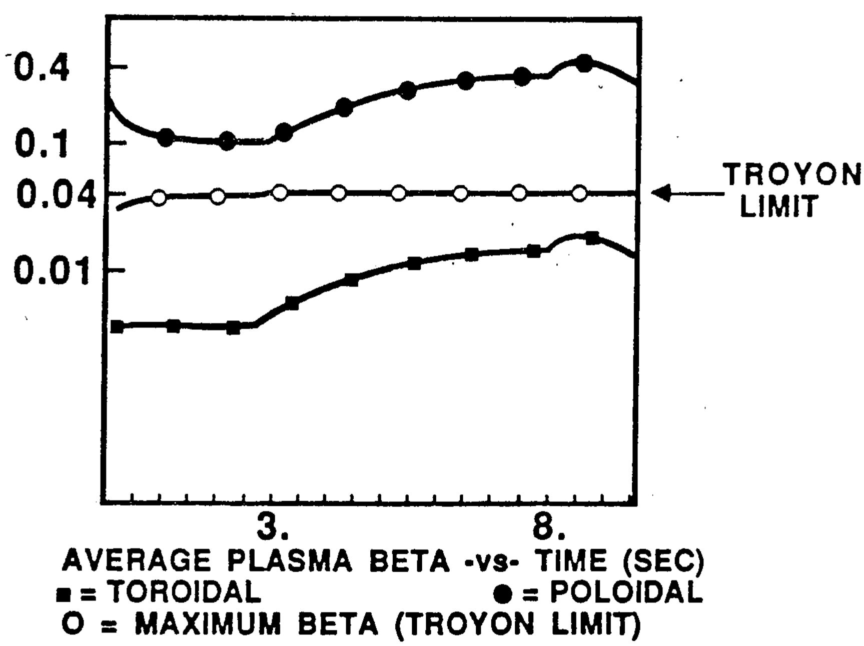


Figure 6. Volume-averaged toroidal and poloidal beta values and Troyon limit.

of the high electron temperature. Passive control of the thermal instability is possible because of the cyclotron emission by the electrons.

During the discharge, no known plasma instabilities are excited. As illustration, Figure 6 shows the evolution of volume-averaged toroidal beta value together with the variation of the disruptive Troyon limit. In addition, it can be seen in Figure 7 that the discharge evolves within the empirically obtained Hugill diagram for stable operation of tokamaks. (Disruptive instabilities can be avoided completely and optimum operating conditions achieved by operating well within this island.) Detailed stability calculations indicate that IGNITEX discharges have an ample margin for stability to MHD tokamak modes. Low toroidal beta, moderate density and elongation values, and plasma-wall stabilization are important ingredients for the good stability characteristics of IGNITEX. A low-plasma-temperature approach to ignition maximizes ohmic heating, energy confinement, and ignition margin while minimizing beta and neutron wall load. Low beta values imply low energy confinement degradation by fusion power, sufficient cyclotron emission to control the thermal instability, and better plasma stability. A moderate neutron wall load reduces damage and wall activation and also reduces impurity influx into the plasma. When ignition conditions are attained, the neutron wall load in IGNITEX will be 3.0 MW/m^2. During the ignited phase it will reach a maximum value of 8.0 MW/m^2. Other thermal loads to the chamber wall, due to other forms

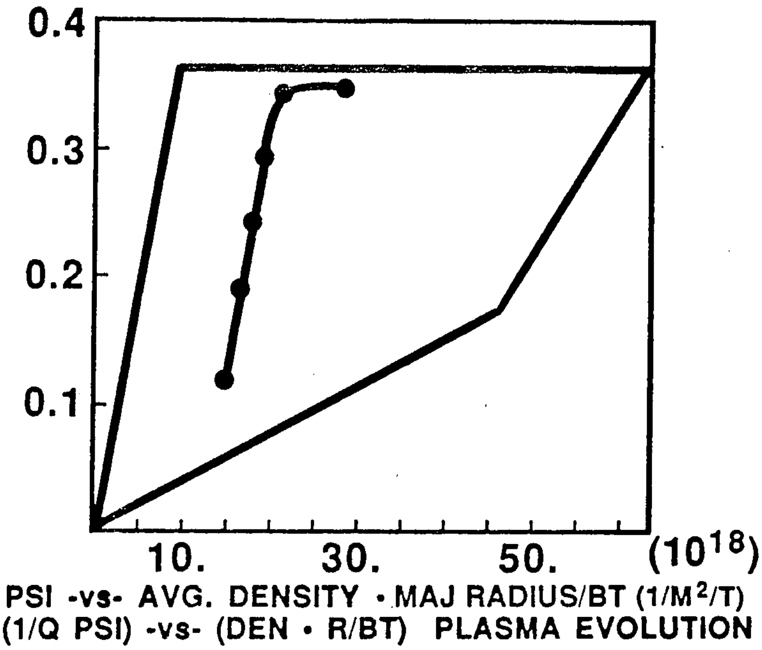


Figure 7. Hugill diagram.

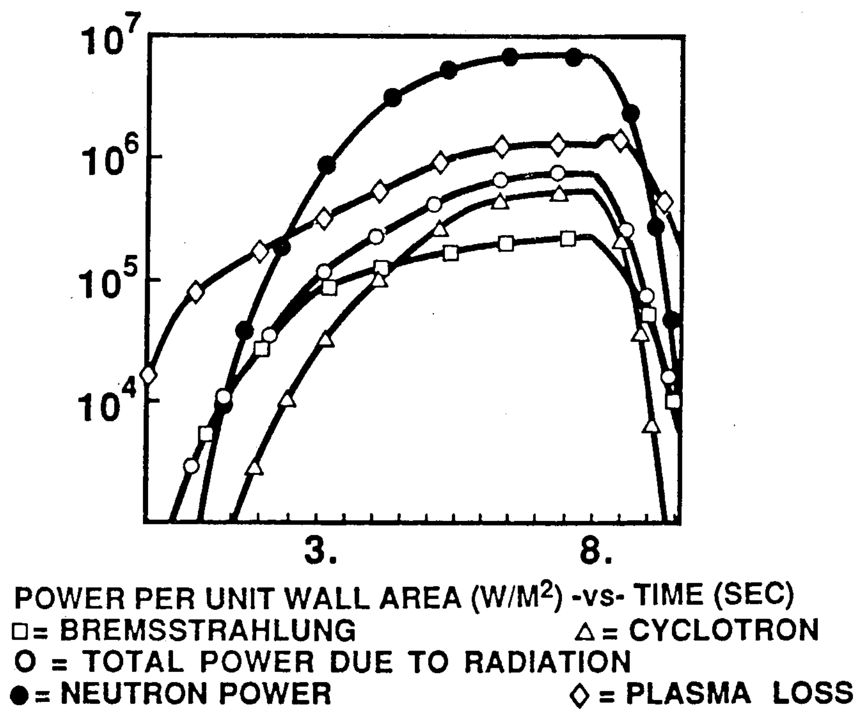


Figure 8. Radiative power losses to the first wall.

of radiation and charged particle losses, are about five times lower than the load due to the neutrons (Figure 8).

More detailed transport analysis of the radial plasma dynamics confirms the results of the volume-averaged calculations presented here. Cyclotron radiation transport in the plasma has also been studied in detail to further analyze the thermal stability control. The conclusion of the radiation transport studies is that the volume-averaged results (obtained using standard analytical theories for cyclotron emission and absorption in plasmas at low and high temperatures) give reasonably accurate rates of cooling of the bulk of the plasma. (The radial distribution of the net radiation emission is not well approximated in the external plasma region.) Discharges in the H- and hot-ion modes of operation have been also simulated in IGNITEX. (H-mode stands for high-energy-confinement mode, and hot-ion mode refers to plasmas in which the ion temperature exceeds the electron temperature; both modes of operation are desirable but not commonly attainable.)

Many uncertainties are associated with the predictive analysis of any ignition experiment. In the case of the IGNITEX concept, these uncertainties are reduced by:

- Operating in the best-known and best-performing regime of operation in magnetic confinement devices
- Maintaining the plasma discharge far from marginal stability
- Maximizing ignition margin and alpha containment predictions
- Using passive means of thermal runaway control
- Simplifying experiment operation

IGNITEX DEVICE

The experiment described in this report can be carried out in a tokamak device that uses unconventional fusion technology [13]. The high level of stress present in the magnet system can be tolerated by a single-turn toroidal coil (Figure 9) with a very high filling factor (i.e., minimum requirement for insulation). Because of the large surface available to carry current in the inner zone of the coil (which is the magnet region subjected to the worst thermo-mechanical stresses), the average current density there will be fairly low (57 MA/m^2), considering the magnitude of the field being generated. This fact makes the peak thermo-mechanical stresses in the magnet lower than those in conventional tokamaks. Stresses are further reduced by: a compression bar in the central hole of the torus supporting the radial collapsing forces; an elliptically shaped plasma bore that reduces bending moments; axial preload to support the vertical separating forces; and radial preload that prevents structural gaps. This approach is efficient for the production of the high fields required (in terms of stress and heating minimization). The copper-alloy magnet will be precooled to liquid nitrogen temperature (77K) to extend the duration of the pulse and decrease resistive losses. The impedance of this system is very low ($< 1 \mu\Omega$). A set of direct-current homopolar generators can handle the currents ($\sim$ 150 MA), voltages ($\sim$ 10 volts), and energies ($\sim$ 12 GJ) required in the experiment.

The toroidal field magnet system does not require the conventional windings, insulators, and turn-to-turn transitions of typical tokamak

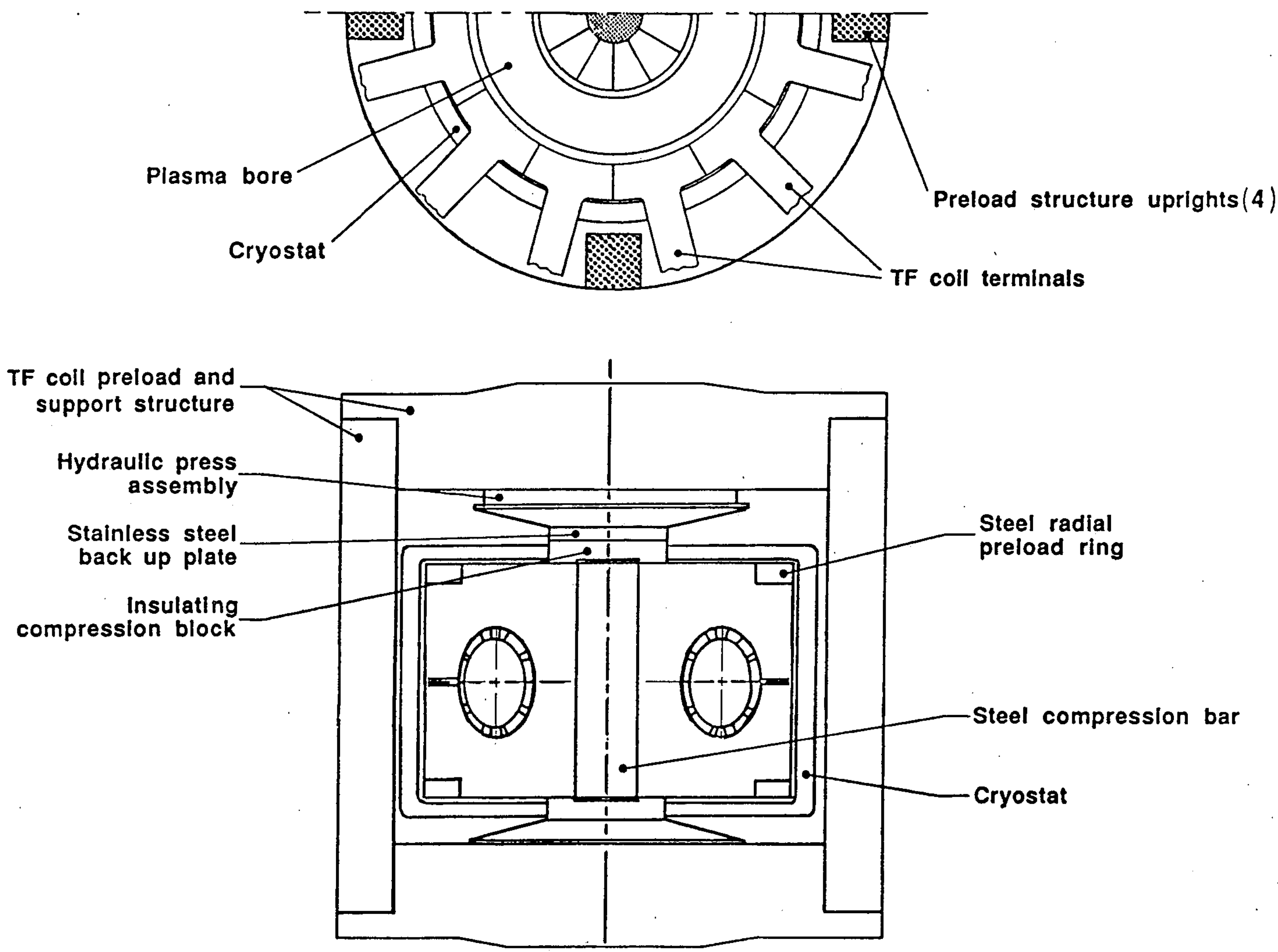


Figure 9. IGNITEX conceptual layout.

designs. The poloidal field system is internal to the single-turn coil. Plasma coupling is, then, very strong so that the energy requirements for current induction are reduced and plasma control is facilitated. Because of the simplicity and compactness of the device and because of the new magnet and power supply tokamak technologies to be employed, initial estimates suggest a relatively low cost for the construction of this experiment [13]. The scientific and engineering effort put into the analysis of the ideas described in this work is of a preliminary nature. Therefore, some aspects of the experiment described here need further study.

PHYSICS AFTER IGNITION

A thermonuclear ignited plasma differs from the plasma produced in present-day experiments primarily in that it will contain fusion reaction products. In the case of deuterium-tritium fusion, the new component is alpha particles (helium nuclei, He^4), which can have a significant influence on the behavior of the burning plasma. Therefore, ignition opens up a new regime of physics to be investigated.

Thermonuclear reactions in an ignited plasma are a peculiar heating source since their intensity depends nonlinearly on the plasma state, which is itself evolving in time. In addition, the time for alpha thermalization can be comparable to the time scale for macroscopic plasma changes. In an ignition experiment, the contribution of the alpha particles to the thermal energy balance in the plasma is dominant. Thus, alpha particles will modify the balance of particle density, momentum, and energy in an ignited plasma through the fusion energy that is generated and through mechanisms whereby alpha particles couple the large-scale and small-scale behavior of the bulk fusion plasma. It is unknown whether the transport processes will be essentially the same as those with no alpha particles present or whether they will be dictated by a new kind of scaling due to the presence of the alpha particles. Because of its very high energies (MeV range), the alpha component of the plasma must be studied kinetically rather than by the usual fluidlike treatment.

Problems of interest to be investigated in the ignition experiment described here include the effect of alpha particles in the stability, heating, and confinement of ignited plasmas, the control of the thermal instability, the accumulation of alpha particles and impurities in the plasma, and the physics of the plasma edge during ignition. The investigation of neoclassical and collective mechanisms by which alpha particles are lost from the plasma will be also of interest.

Alpha particles, with their high energies and nonequilibrium thermodynamic distribution, can cause new thermonuclear instabilities that may lead to alpha losses on the fast MHD time scale, as well as to degraded plasma confinement. Empirical energy confinement scalings exhibit degradation of bulk plasma confinement with auxiliary power and possibly with ohmic power. Because alpha heating and ohmic heating are intrinsic to the plasma and both primarily heat electrons, the alpha heating power may also contribute to the degradation of confinement. It is expected that the distortion of the electron distribution function (due to isotropic slowing-down of the alpha particles) will be less able to excite fluctuations than the distortion due to the anisotropic ohmic heating mechanism.

The physics phenomena that will be discovered and investigated in a fusion ignition experiment like the one described here will be determine the basic design and operation of future fusion energy systems. The production of a self-sustained fusion reaction and the study of the novel ignition regime of fusion plasma physics will significantly contribute to proving the scientific feasibility of controlled thermonuclear fusion.

In conclusion, the IGNITEX concept proposes an experiment with a compact, high-field, single-turn-coil tokamak to reach and control fusion ignition with ohmic heating alone. The predicted margin for ignition is high. Ignition can be reached far below stability limits. The ignited phase of the discharge can be maintained passively by the plasma itself without exceeding disruptive limits. Initial calculations indicate that this system can provide a simple means of producing ignited plasmas for scientific study at a relatively low cost.

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