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Introduction

The IGNITEX project has as its objective to produce and control ignited plasmas for scientific study in the simplest and least expensive way possible. The original concept for the fusion ignition experiment (IGNITEX) was proposed by M. N. Rosenbluth, W. F. Weldon, and H. H. Woodson [1] of The University of Texas at Austin on the basis of B. Coppi's (MIT) ideas [2] of a compact ignition experiment and recent technological advances in high-current systems at the Center for Electromechanics (UT). The IGNITEX device is a compact high-field single-turn coil tokamak capable of reaching and controlling fusion ignition with ohmic heating alone [3]. Homopolar generators (HPG) will supply the current and energy required for the experiment.

The analysis performed to date with regard to the IGNITEX experiment (as reported in TAERF Report Nos. 42 and 43 and in this report) has been focused on the basic plasma physics, magnet system and power supply designs, and cost estimates. The results of these calculations show the scientific interest of the experiment, the feasibility of the proposed technology, and the attractiveness of the cost.

A brief outline of some of the differences that distinguish the IGNITEX design from conventional designs that have been proposed following Coppi's ideas of compact high-field tokamaks is as follows:

- In IGNITEX, ignition is predicted to be possible with ohmic heating only and without auxiliary heating. This arrangement is considered to be an advantage because recent experimental tokamak results show that

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energy confinement appears to deteriorate gradually as the level of auxiliary heating is increased and also because of the additional complexity involved in auxiliary heating schemes.

- Ignition is predicted with all presently projected energy confinement scalings even in the standard L-mode of operation. Therefore, the problems associated with divertor designs and with the uncertainty of the H-mode of operation are nonexistent in the IGNITEX design.
- Ignition can be approached at low beta (and thus far from marginal stability) with the IGNITEX design. In addition, the plasma is closely surrounded by a conducting wall. These two characteristics anticipate stable plasmas; they give the IGNITEX design a high probability of performing as predicted and make ignition production and control possible.
- Ignition depends crucially on the ability of the experiment to confine the alpha particles produced in the fusion reactions. Alpha confinement in IGNITEX is expected to be much higher (as measured by the ratio of the plasma current to the minimum current required for most of the alpha particles to be confined) than in conventional designs. In addition, the familiar toroidal field ripple and its associated plasma losses are reduced in IGNITEX.
- Ignition triggers a thermal excursion of the thermonuclear plasma. In conventional experiments that are designed at marginal stability, reaching ignition means crossing the stability limits for the plasma, and then the discharge disrupts. In the IGNITEX design the plasma reaches the ignited state far from the stability limits, and the thermal excursion is stabilized by the plasma itself. The plasma conditions are such that once the plasma temperature begins to increase after ignition, the cyclotron losses become important and limit the thermal excursion. During the full period of plasma ignition, the disruptive limits are not exceeded.
- The IGNITEX design has much lower average current density in the inner leg of the toroidal field coil than conventional approaches (because of the high filling factors). The stress in the coil is thus tolerable. Moreover, because of the central compression bar, the wedging stresses in the toroidal field magnet are lower than usual.

- The IGNITEX design includes an internal poloidal field system configuration with single-turn-coil magnets that minimizes the power and energy requirements for plasma current induction, equilibrium, and shaping.
- The IGNITEX power supply operates at low voltage. Therefore, the magnet system has no severe insulation requirements. The problems associated with radiation damage, low strength, and nonavailability of appropriate insulators are minimized in IGNITEX. Furthermore, the IGNITEX magnet system design is simple, as it has no windings, turn-to-turn transitions, and so forth.
- The IGNITEX experiment should be less expensive to build than comparable conventional experiments because of its simplicity and unconventional technology.

In summary, the nonconventional IGNITEX experiment design emphasizes simplicity in order to maximize reliability and minimize costs. The IGNITEX concept has a high potential and it can offer a simple, reliable and low cost approach to fusion ignition. A proposal was written and presented in March to the Office for Fusion Energy (OFE) of the US Department of Energy. This proposal seeks support from DOE in the continuation of the IGNITEX project into a more expanded second phase of the design. Many members of the OFE and the fusion community have shown strong interest in the IGNITEX concept. After the presentation at DOE headquarters, the OFE requested an independent review of the concept. The review took place at The University of Texas at Austin recently. OFE also requested that a presentation of the IGNITEX concept be made to the Ignition Technical Oversight Committee during its last meeting at the Princeton Plasma Physics Laboratory. DOE is presently reviewing The University of Texas proposal for funding in fiscal year 1988.

In the sections that follow, plasma physics and engineering issues relevant to IGNITEX are analyzed. A number of different energy confinement scalings have been studied to predict the ignition margins of this experiment. The energy confinement scaling of thermonuclear ignited plasmas is not yet known (this is one of the problems that could be addressed with the IGNITEX experiment). Therefore, these calculations assume that alpha heating will degrade confinement as auxiliary heating does in present experiments. IGNITEX has been designed to satisfy the requirements of the most pessimistic scalings. The thresholds for ohmic ignition and the sensitivity of the ignited state have been studied. The equilibrium at ignition has been analyzed (assuming fixed boundary conditions). In addition, the plasma radial dynamics during the plasma discharge has been investigated. The fabrication and assembly of the magnet system are summarized here. The power supplies for the poloidal field magnet system have been designed, and their mode of operation has also been studied. As part of the proposal to DOE, a scaled single-turn-coil prototype experiment (PROTEX) has been proposed. Preliminary calculations for PROTEX are described in this report. Finally, some concluding remarks are given.

Ignition Margin Predictions

The results of numerically solving the plasma power balance equations (as explained in TAERF Report No. 42) can be conveniently represented in the plane of average plasma temperature versus average plasma density. An idea of the margin available for reaching ignition for each of the different energy scalings considered here is obtained by calculating the quantity $\Delta\rho$ (ignition margin). For a given temperature, $\Delta\rho$ is defined as the ignition factor (ρ) evaluated at marginal stability; $\Delta\rho$ is a function of temperature.

Table 1 gives the ignition margins obtained for several energy confinement scalings. The calculations considered a plasma elongation $\kappa = 1.6$, a cylindrical safety factor at the plasma edge $q_{cyl}(a) = 2.2$, and an exponent of the parabolic density profile $\alpha_n = 1$. The case with Goldston scaling has been solved with $\kappa = 1.8$, $q_{cyl}(a) = 2$, and $\alpha_n = 2.5$ (pellet injection). These last parameters are more demanding than the ones used in the IGNITEX design, but still they are within the range of generally accepted values.

All presently projected confinement scalings predict ohmic ignition of the IGNITEX experiment. Specifically, the Kaye-Goldston scaling (which is the scaling that fits the largest number of tokamak experiments to date) predicts an ignition margin $\Delta\rho = 3.4$.

Table 1

IGNITEX IGNITION MARGIN PREDICTIONS

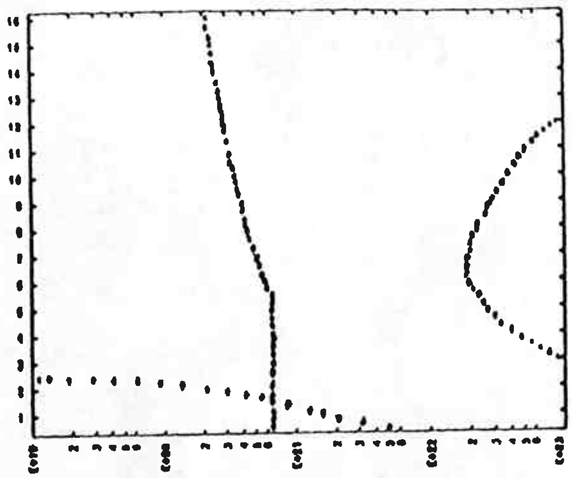
Energy Confinement Scaling	Ignition Margin
NeoAlcator	4.5
Kaye-Goldston	3.4
Mirnov	14.0
NeoAlcator with density saturation (gas puffing)	3.0
NeoAlcator with density saturation (pellet injection)	5.3
Doublet-III	6.4
Perkins	4.0
TFTR-Goldston	1.4
Goldston	1.1
Sawtooth	2.3
Ion mixing modes	3.8

Threshold for Ohmic Ignition

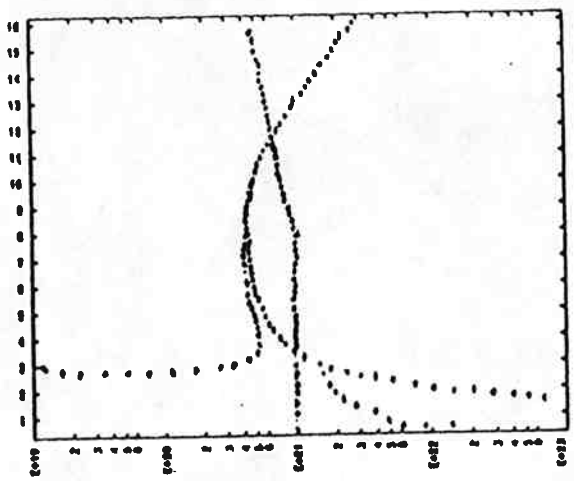
The high field of IGNITEX implies good confinement and stability and high plasma current. These beneficial effects are very important. In fact, plasmas with on-axis magnetic fields below 15 Tesla cannot achieve ohmic ignition, as shown in Fig. 1. This threshold value corresponds to Kaye-Goldston scaling. At this same scaling, it has been obtained that a minor radius lower than 25 cm will preclude ohmic ignition (with 20 T field operation). Of course, these threshold values change for different energy confinement scalings. If scalings better than the more pessimistic ones turn out to prevail for IGNITEX plasmas, ignition may be reached with less than maximum toroidal field. For instance, Kaye-Goldston scaling will allow ignition at 75% of maximum field.

The IGNITEX experiment as described in TAERF Report No. 42 has values of $\eta_i(r)$ lower than the critical value for excitation of ion mixing modes. Calculations have been done to find the broadest density profile that permits ohmic ignition under the assumption that η_i modes might be a mechanism for ignition inhibition. The threshold for ohmic ignition is $\alpha_n = 0.5$. Below this value the region of η_i modes excitation covers most of the plasma, and ignition is prevented. This threshold value applies approximately for both flat and parabolic temperature profiles.

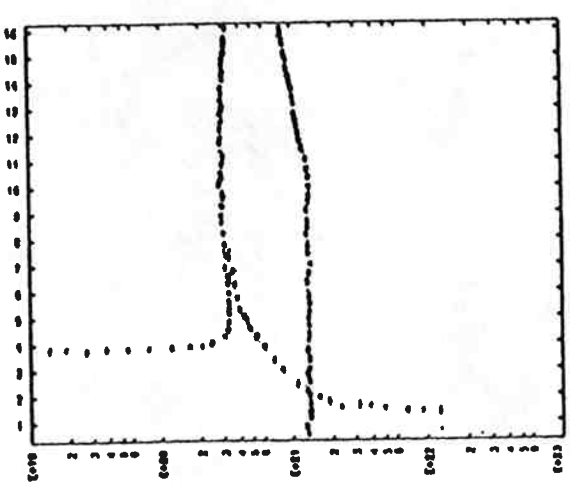
If a mechanism exists that maintains the value of $\eta(r)$ close to the critical value for η_i mode excitation [i.e., does not allow $\eta(r)$ to increase too much], then ignition is possible even with broad density profiles. Experimental evidence indicates that broad density profiles are detrimental for energy confinement in the plasma and that pellet injection is beneficial in providing better confinement. Some results related to this issue are presented in the next section. The IGNITEX experiment designers will consider the use of a pellet injector for improving plasma performance. Because high velocities will be required to penetrate ignited plasmas, electromagnetic railgun technology will be considered for pellet injection.



a. B_t = 10 T



b. B_t =15.1 T



c. B_t = 20.2 T

Fig. 1. Effect of the variation of the toroidal magnetic field on the plasma power balance for Kaye-Goldston scaling

Sensitivity of the Ignited State

The plasma cross-section of IGNITEX is elliptical, with an elongation $\kappa = 1.6$. When the results obtained in this case are compared with those of a circular discharge with the same minor radius, the current in the elongated plasma is much larger and therefore its ohmic heating, stability, and confinement are superior to the circular case. Even if the total plasma current is maintained constant (with the radius of the circular plasma larger than the minor radius of the elliptical plasma), an elongated plasma cross-section gives better confinement and stability than a circular plasma cross-section. Noncircular cross-section plasma discharges have a better chance of reaching ignition. In addition, elongated plasma bores are beneficial in reducing magnet stress.

In the preceding section it was seen that broad density profiles might be detrimental to the plasma performance. Pellet fueling has been shown to improve plasma behavior in ohmic discharges. This phenomenon has been attributed to the suppression of drift modes driven by ion temperature gradients. Pellet injection improves refueling and makes the density profile more peaked on axis. Peaked density profiles increase the ignition margins predicted for IGNITEX to even higher values.

Finally, variations of the major radius around the design value $R = 150$ cm have been considered. The conclusion of these calculations has been that the selection of the major radius is basically a matter of engineering considerations (stress vs. cost), with no major consequences for the plasma performance when the major radius is varied around the design value.

Plasma Equilibrium at Ignition

Plasma equilibrium calculations at ignition have been carried out using the VMONS code. This code solves the Grad-Shafranov equation in an axisymmetric tokamak plasma. The procedure of solution considers a variational formulation and uses a spectral representation of the flux function. A system of coupled ordinary differential equations is obtained by requiring the variational principle to be stationary with respect to the Fourier amplitudes.

The equilibrium configuration for the ignited state is shown in Fig. 2. Figure 3 gives the plasma pressure, current, and safety factor radial profiles. It also shows the radial variation of the shift of the magnetic surfaces with respect to the geometric center of the plasma bore and the elongation of the magnetic surfaces. These results have been used in defining the stability constraints in volume-averaged calculations.

Plasma Radial Dynamics to Ignition

The radial dynamics of the plasma discharge has been studied with the PROCTR code. Time-dependent spatial profiles are evolved for plasma relevant quantities including the densities and temperatures of the various species. The numerical scheme includes equations for the scrape-off layer, limiter, and wall. An approximate pressure equilibrium is maintained by assuming shifted elliptical flux surfaces.

The calculations presented here simulate plasma current penetration by assuming an undetermined mechanism operating in the plasma that relaxes the current distribution so that it satisfies the principle of profile consistency. The poloidal field is ramped to maximum intensity in 3 sec, whereas the toroidal field is considered here to be at full strength since discharge initiation. The electron energy confinement is assumed to follow the NeoAlcator scaling, and the ion energy confinement is considered to be four times smaller than neoclassical. Alpha heating and cyclotron radiation losses are included in the calculations.

Detailed radial dynamics calculations confirm the results of the volume-averaged, time-dependent calculations given in TAERF Report No. 43. Figures 4.a-4.g give the evolution of ion and electron temperature profiles and plasma density and current density radial profiles. Finally, the shift of the magnetic surfaces profile, the ion and electron temperatures on axis along the discharge, and the variation of the average plasma density are given.

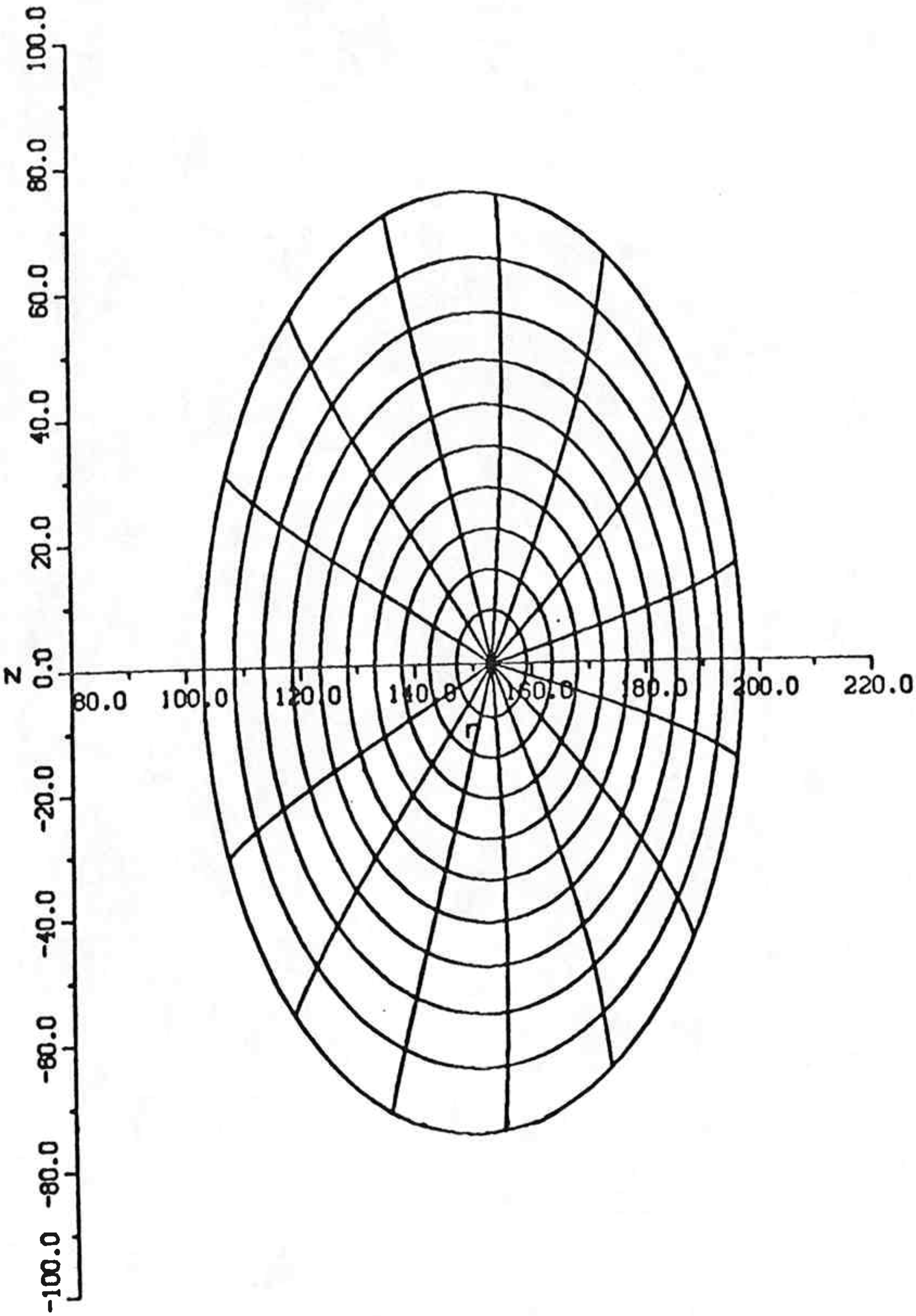
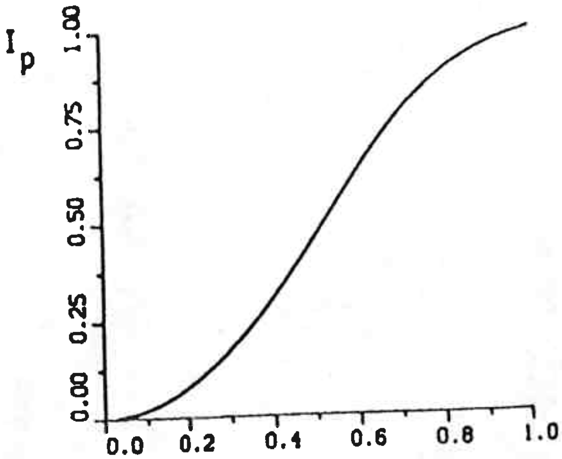
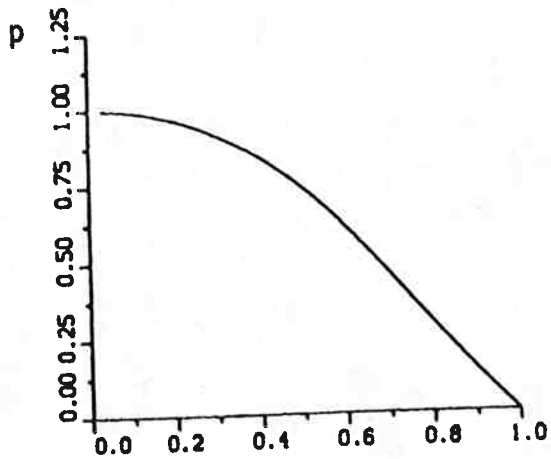


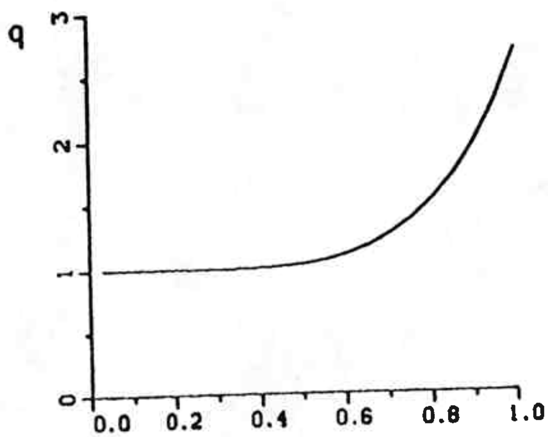
Fig. 2. Equilibrium configuration at ignition



a. Plasma current

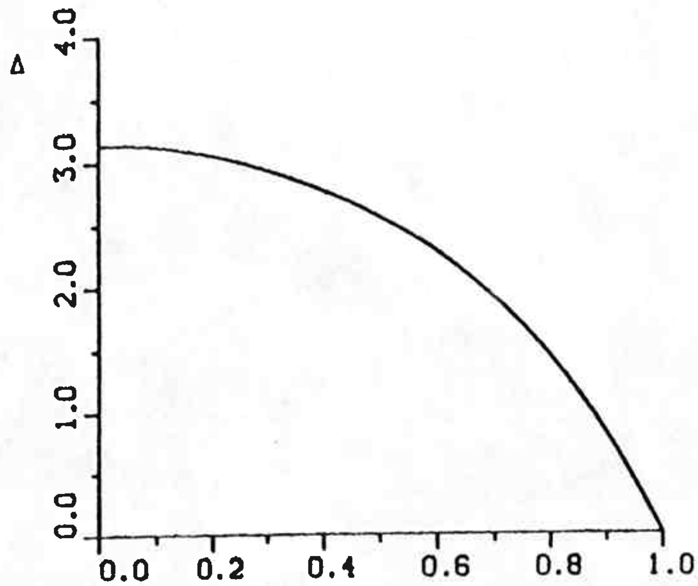


b. Plasma pressure

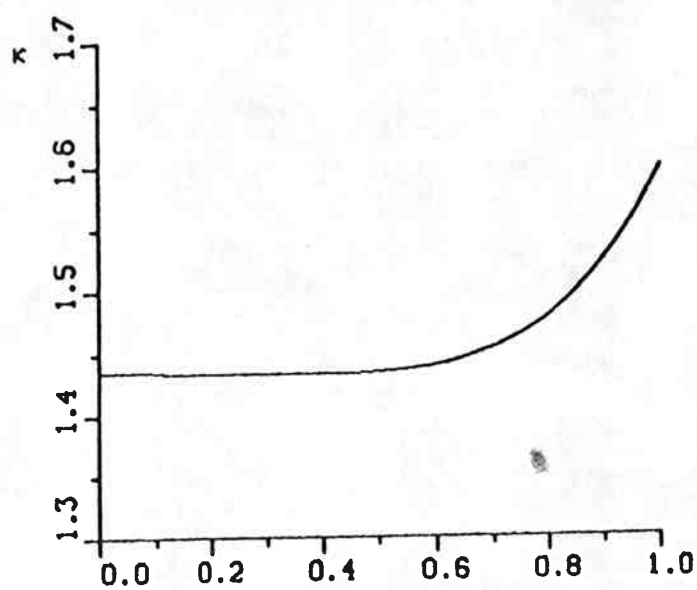


c. Safety factor

Fig. 3. Equilibrium radial profiles

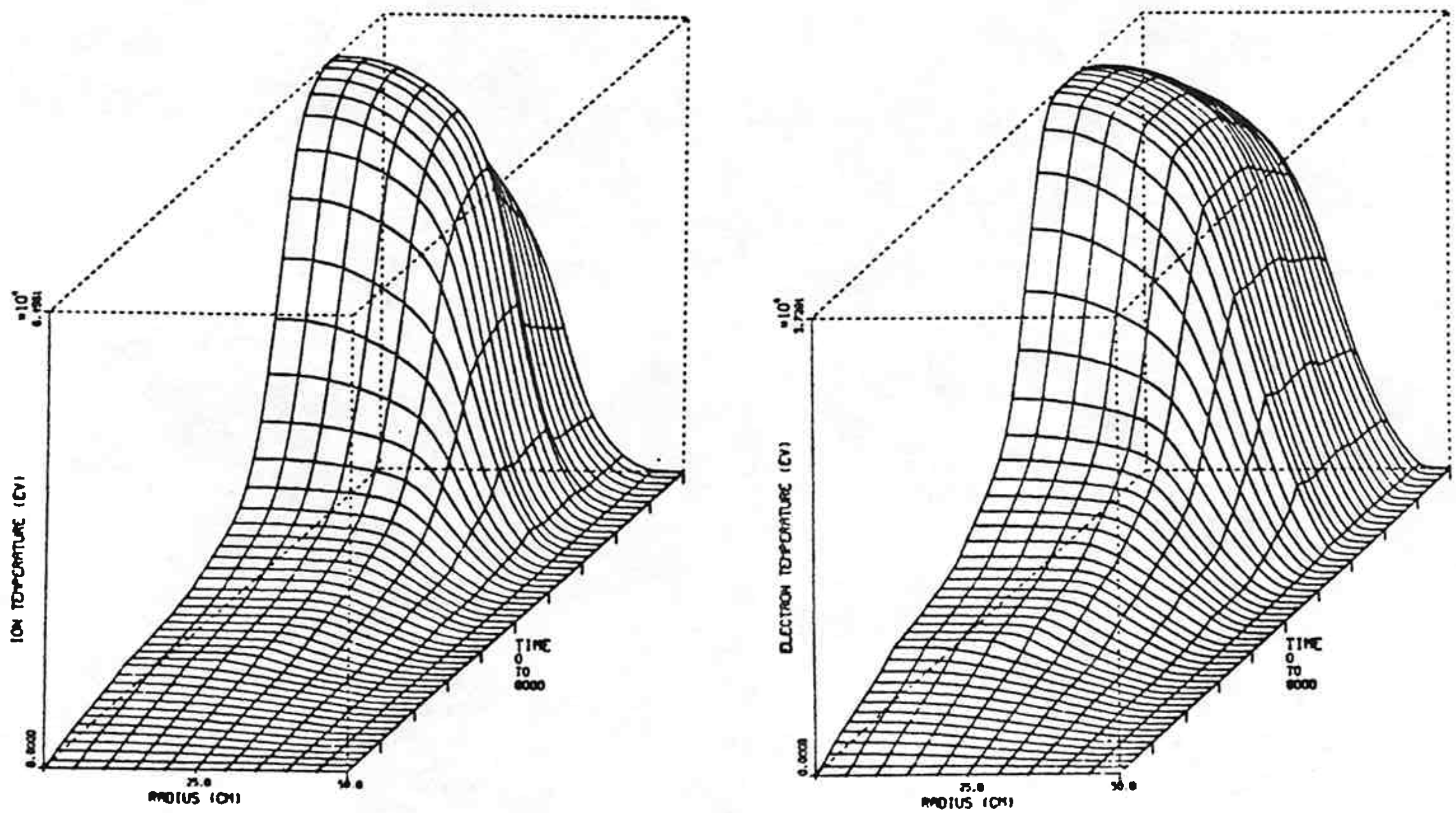


d. Shift of the magnetic surfaces



e. Elongation of the magnetic surfaces

Fig. 3. (cont.) Equilibrium radial profiles



a. Ion temperature profile

b. Electron temperature profile

Fig. 4. Time evolution of plasma characteristics