FUSION IGNITION EXPERIMENT

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Introduction

The IGNITEX project (started recently at The University of Texas at Austin as reported in TAERF Report No. 42) has as its objective to propose a fusion ignition experiment that can produce ignited fusion plasmas for scientific study in a simple, reliable, and cost-effective way.

A detailed study of the ignited plasma (Ref. 1) in the IGNITEX experiment, which includes consideration of several energy confinement scalings predicted to date, confirms that this experiment has an ample margin for ohmic ignition. The experiment operates within a wide ignition region. Numerical simulations of the discharge to ignition show that the ignited state (i.e., when the alpha particle heating is the dominant plasma heating mechanism and it surpasses the power losses in the plasma) is reached within 4 sec of the initiation of the discharge. This is 1 sec after the full current and magnetic field have been established in the plasma column.

After ignition is reached, the ion and electron temperatures increase drastically, and this thermal excursion lasts for about 4 sec. During the ignited phase of the discharge, all the stability criteria for plasma operation are satisfied, and the disruptive limits are not exceeded. After 8-9 sec of plasma formation the fields and currents are ramped down (in 2-3 sec). The basic mechanism for control of the thermal runaway is the cyclotron radiation emitted by the plasma. This emission of radiation is quite small before ignition since the electron temperature is low, but it becomes important when the plasma

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reaches ignition. This is then a natural mechanism for the IGNITEX plasma to control the thermal runaway.

The toroidal magnet system is a single-turn coil made of twelve pie-shaped sectors. Each sector will be fed by one homopolar generator (HPG) supplying 12.5 MA and 1 GJ at 10 volts. Since the modules are operated in parallel, insulation between the modules is not needed (it will be added, however, to prevent toroidal current flow). The terminals for the generators are distributed around the torus in the central plane. There is an internal inductor around the plasma vacuum vessel and within the bore of the single-turn coil that is made up of five pairs of single-turn coils. A layout of the system is shown in Fig. 1.

These coils provide for the magnetic flux required for ohmic heating during the discharge as well as for the plasma shaping and equilibrium. The current in the inductor is swung between maximum levels of +22 MA and -15.7 MA. Slits in the inductor allow penetration of the external toroidal field. The power is supplied to the internal inductor by five homopolar generators totaling 2.6 GJ and working at 60 volts. The terminals feeding the current to the internal inductor are arranged at two poloidal sections of the machine, 180° apart. The design minimizes the stray fields at the connections. A somewhat similar arrangement was used in the TTT tokamak at The University of Texas at Austin.

Thermal and mechanical stresses have been evaluated during the plasma discharge. The analysis provides for coupled electromagnetic and thermal diffusion, and it accounts for ohmic heating. The maximum attained level of stress in the single-turn coil is below the yield strength of the copper alloy employed (Al_2O_3 dispersion-strengthened Cu). The magnet system is immersed in liquid nitrogen and cooled down to 77°K before discharges. The temperature rises in the single-turn coil from 77°K to 145°C at the hottest spot. A steel compression bar is positioned in the central core. With it, the stress carried by the conductor itself is reduced substantially because the steel supports a major fraction of the stress. In addition, axial and radial preloads are applied by means of a press structure and a steel ring, respectively.

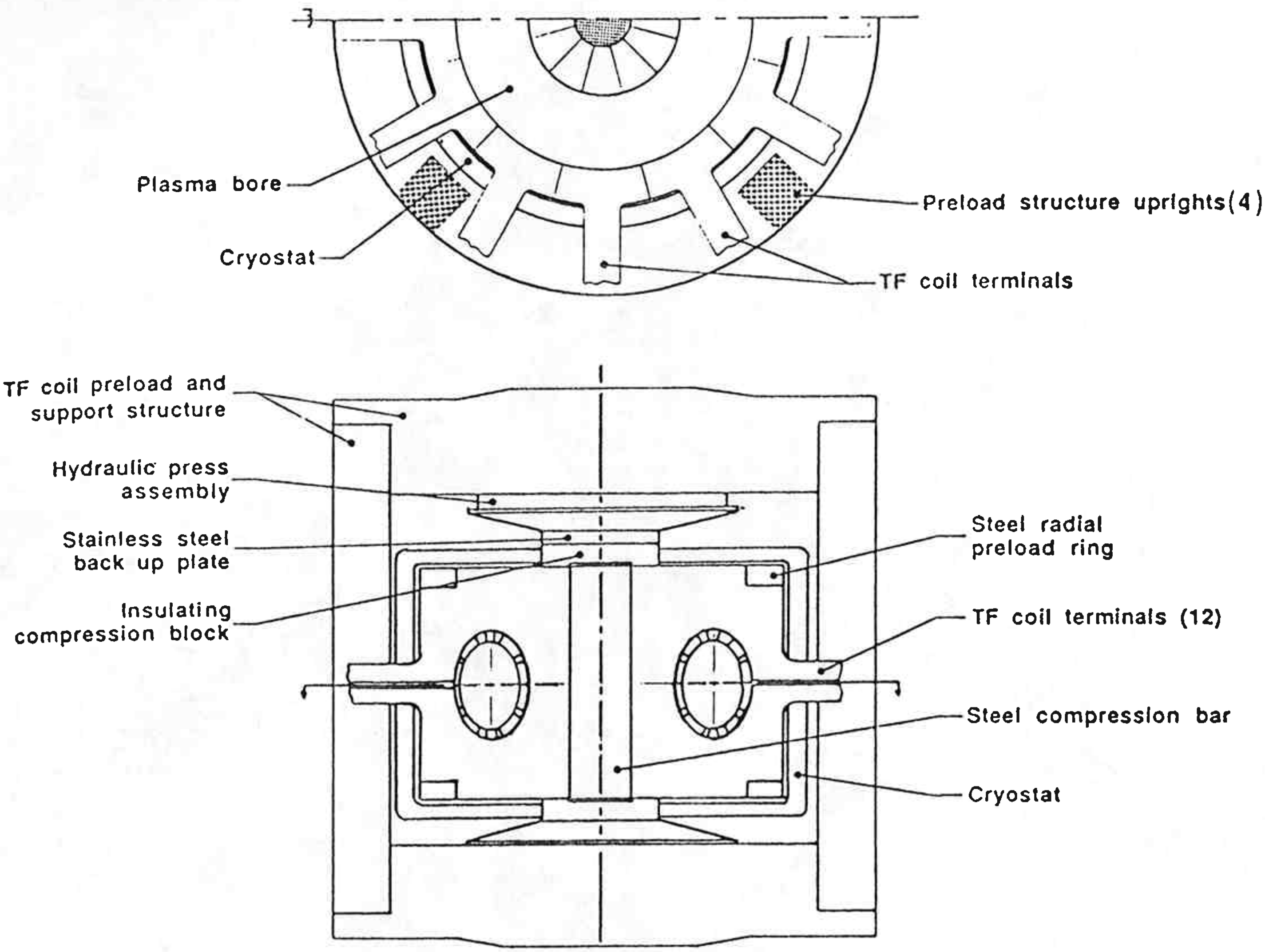


Fig. 1 IGNITEX conceptual layout

The basic idea of the power supply design is to generate voltage and current in a conventional iron-core homopolar generator and attach a filament-reinforced epoxy composite flywheel for energy storage. The generator voltage can be controlled during the length of the pulse by adjusting the excitation field.

Cost estimates for the construction of the IGNITEX device include considerations for the simplicity of the design. Single-turn coils and homopolar generators are much less expensive than their counterparts in conventional tokamaks. The fact that no auxiliary heating and divertor are needed contributes to the reduction of the experiment cost. A total cost of $110 million has been estimated for the construction of the experiment.

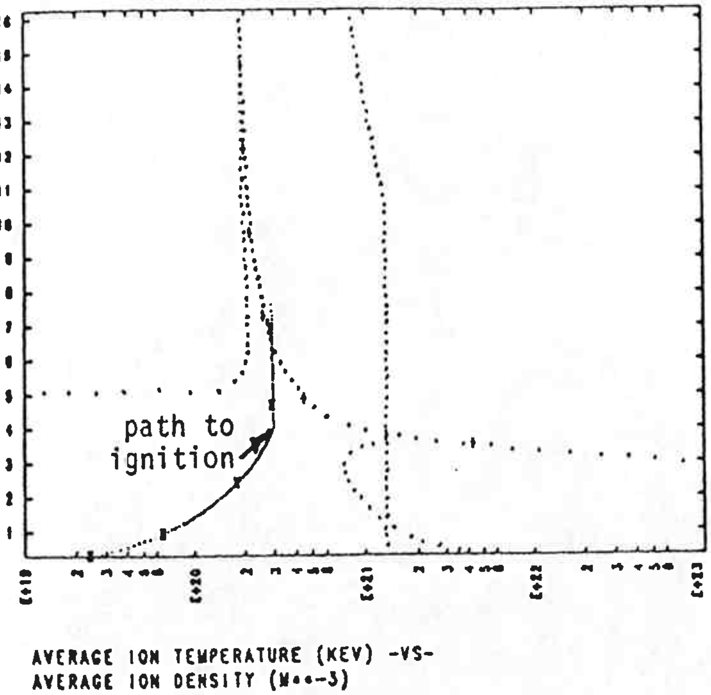
In this report, time-dependent numerical simulations of the plasma discharge to ignition and an analysis of the control of the thermal runaway at ignition are presented. In addition, the design of the poloidal field (PF) magnet system and cost estimates for the experiment are given.

Plasma Discharge Simulation to Ohmic Ignition

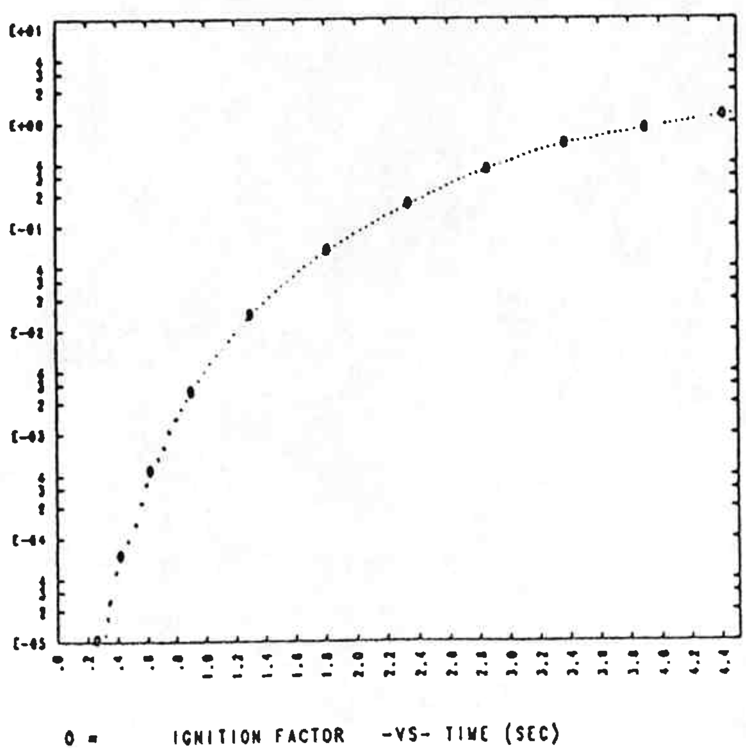
It has been shown (Ref. 1) that the proposed experiment can reach ignition ohmically with an ample margin. Naturally, next, one needs to evaluate how this state of ignition can be reached within the constraints of the experiment.

The physical mechanisms acting on the plasma in its path to ignition are described by a system of coupled ordinary differential equations that determine the time evolution of the densities of ions, electrons, and alpha particles and the ion and electron temperatures. This system of equations is integrated over the plasma volume. The solutions are subject to the considerations and constraints indicated in TAERF Report No. 42 for the steady-state calculations. It is assumed that the deposition of the alpha energy is instantaneous (this assumption is reasonable since the alpha slowing-down time is much smaller than the plasma energy confinement time).

Figures 2a-2f show a discharge simulation to ignition. Fields and currents are ramped in the first 3 sec of the discharge. Ignition is

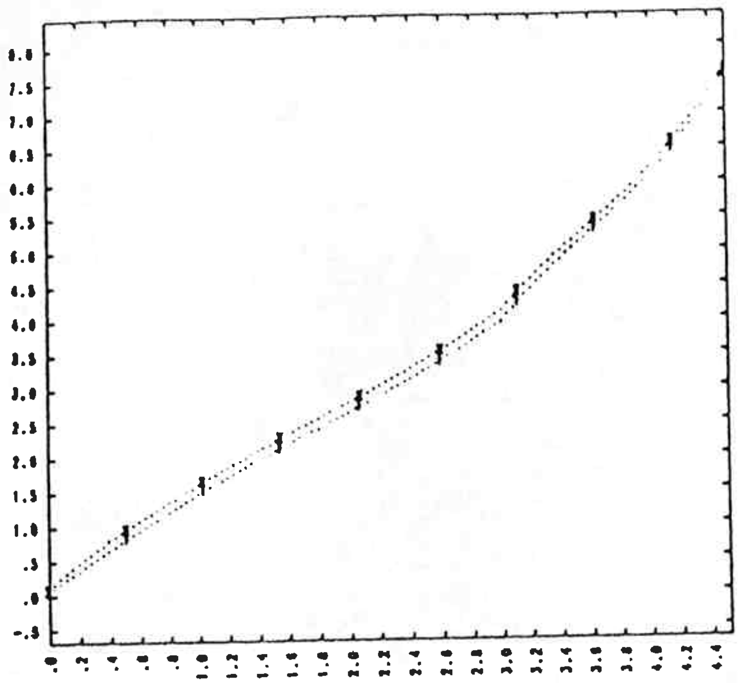


a. Plasma evolution in (n,T) diagram

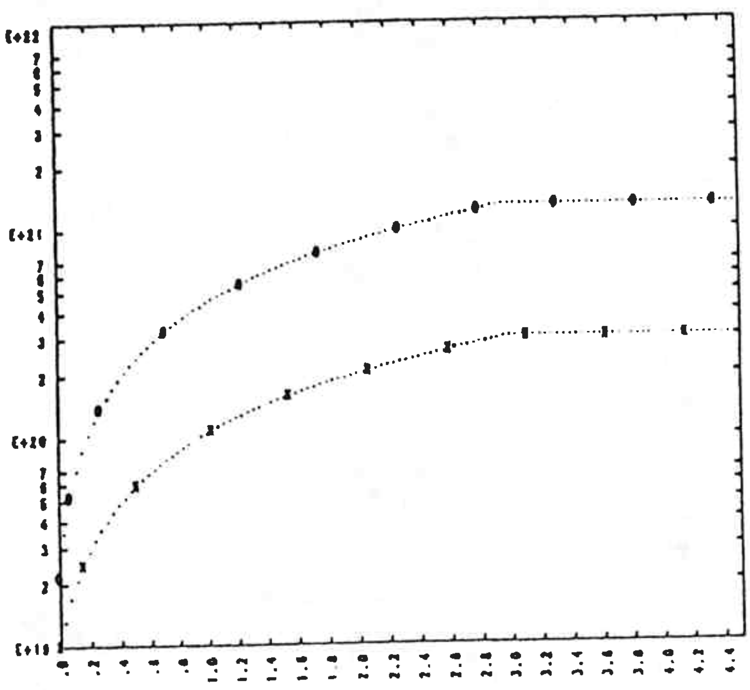


b. Ignition factor

Fig. 2 Discharge simulation to ohmic ignition

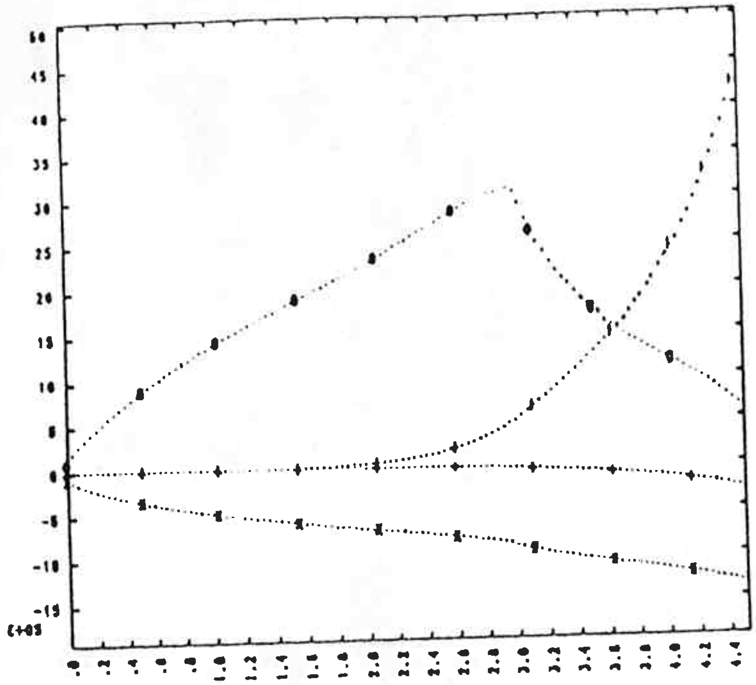


c. Ion and electron temperature

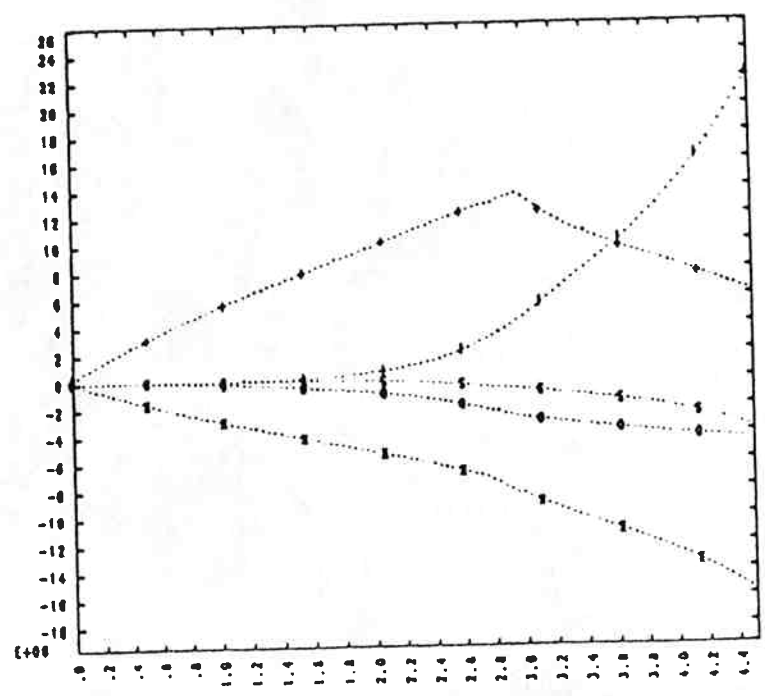


d. Plasma density and Murakami density limit

Fig. 2 (cont.) Discharge simulation to ohmic ignition



e. Ion power balance contributing terms



f. Electron power balance contributing terms

Fig. 2 (cont.) Discharge simulation to ohmic ignition reached in 4 sec after plasma initiation with only ohmic heating. A NeoAlcator scaling for the electron energy confinement has been considered. The ion energy is assumed to be lost from the plasma following a four times larger than neoclassical scaling. Fig. 2a shows the path to ignition in the (n,T) plane. Figs. 2b to 2d show the evolution of the ion and electron temperatures, plasma density, and ignition factor, respectively. Ignition is reached when $\rho = 1$. Figures 2e and 2f show the time variation of the terms contributing to the electron and ion power balance, respectively. It can be seen that the alpha heating power starts rising before the ohmic heating power into the plasma decreases significantly.

The ion temperature increases very steeply when ignition is reached. Therefore, the plasma disruptive limits may easily be exceeded. In the next section a natural mechanism for limiting ion temperature in the IGNITEX experiment is shown.

Ignition Control

In this section the results of a plasma simulation of a full discharge are presented. In this simulation the electron energy confinement follows the NeoAlcator scaling, and the ion energy confinement follows the Kaye-Goldston scaling. The simulation assumes that the density can be ramped down appropriately during the shutdown phase of the discharge. A wall reflection coefficient for the synchrotron radiation $\rho_w = 0.8$ has been assumed. In a detailed calculation this coefficient will depend on the wavelength of the dominant radiation harmonics emitted by the plasma, the particular material structure of the first wall, and the specific reflective portion of the first-wall design. The basic conclusions of this section are not altered by perturbations of the assumed reflection coefficient.

The discharge proceeds as follows. The first 3 sec are used for ramping of currents and fields. Ignition is reached at 4 sec, and it lasts for 4 additional seconds. Then, the fields and currents are ramped down in 2 sec and the discharge is terminated. The IGNITEX design constrains the ion temperature (once ignition is reached) to within the region of plasma stability. Figures 3a-3n show the results of the simulation of a full discharge. First, the current and field time profiles are indicated. The evolution of ion and electron temperatures and densities are given next. In addition, the accumulation of alpha particles and impurities is shown. Note that the level of impurities is considered to have a sudden increase at the moment of ignition. Stability constraints are imposed along the discharge. The plasma evolves within the limits of the Hugill diagram. Moreover, the plasma beta is below the Troyon limit.

Estimates for the loop voltage and the magnetic flux along the discharge are given. The calculation assumes that the magnet system surrounding the plasma is a perfectly conducting wall. Finite resistivity effects will increase somewhat the value of the magnetic flux although it will still be much lower than normally found in tokamaks. In addition, the coupling of the plasma and the coil will be increased. Most of the inductive volt-sec consumption is internal flux.

The power production during the discharge is examined. The terms contributing to the ion and electron power balances are calculated, as well as the ignition factor. Finally, the power losses to the wall are given, including the neutron load, which in the IGNITEX design is maintained within quite reasonable limits even in the ignited phase of the discharge (3 Mw/m^2 at ignition with a maximum of 8 Mw/m^2 further into the discharge).

Poloidal Field Coils

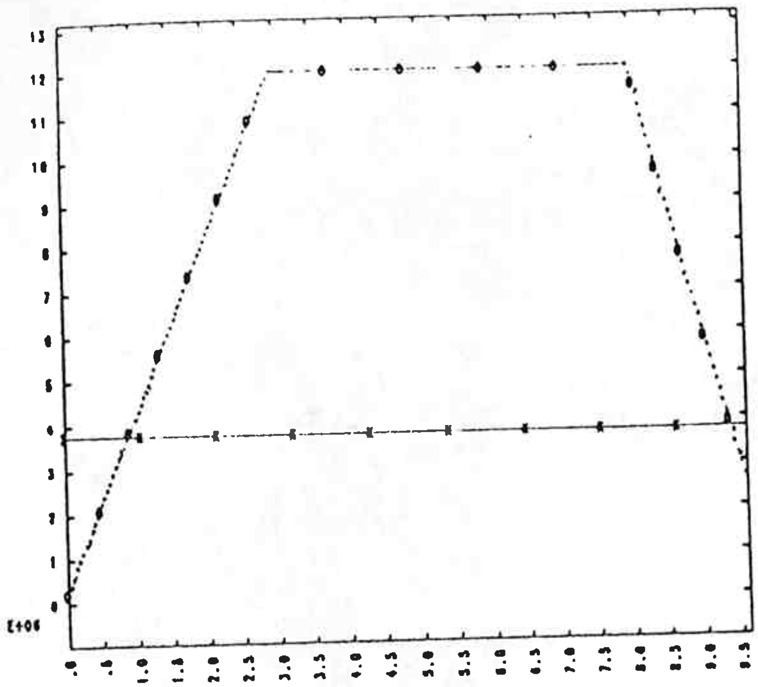
A key feature of the IGNITEX experiment that distinguishes it from conventional tokamak machines is the location of the entire poloidal field coil system within the bore of the torus. Advantages of this configuration are numerous. Proximity of the PF coils to the plasma increase coupling with it substantially, drastically reducing both the energy and power needed to induce the required plasma current and to stabilize and elongate the plasma. Since the coils must contain conductor joints for assembly into the toroidal coil, they are also of single-turn design. This feature makes their design and fabrication simple and cost effective when compared to conventional tokamak PF coils. Also as a consequence of the single-turn configuration of each coil, an inexpensive, low-voltage homopolar power supply will be used to drive each of the five pairs of coils.

Conceptual arrangement of the PF coils in the torus bore is shown in Fig. 4. Each of the coils is 8.5 cm thick. Coil pairs 1 and 2 are the primary ohmic heating coils, pairs 4 and 5 provide the vertical stabilizing field, and coil pair 3 is used to elongate the plasma. Operation of the poloidal field system uses the concept of "ringing" the circuit to reduce peak current and total energy requirements. This technique has been explored with HPG-charged cryogenically cooled inductors in previous electromagnetic launcher power supply development at the Center for Electromechanics (CEM) at The University of Texas at Austin. In these experiments, the HPG rotor energy was transferred to the inductor and then back into the rotor as the inductor current accelerated the rotor in the opposite direction. As conceptually designed, the peak coil currents are given in Table 1.

To allow them to be easily assembled, the PF coils are made of two 180° arcs that are joined as the two halves of the toroidal field (TF) coil are brought together. This split also provides an ideal location to feed the current into the individual coils. Since a total of ten coils are used, the terminals will be arranged so that four are located at one split and the remaining six penetrate the TF coil at the other split. This arrangement is shown conceptually in Fig. 5. Each terminal shown is actually a pair of parallel plate busbars in which current flows in opposite directions, thereby nearly self-cancelling the magnetic fields that could locally disrupt the toroidal field. Furthermore, by dividing the terminals between the two splits, the terminals are flatter and wider, so that the disturbance of the toroidal field will be minimized.

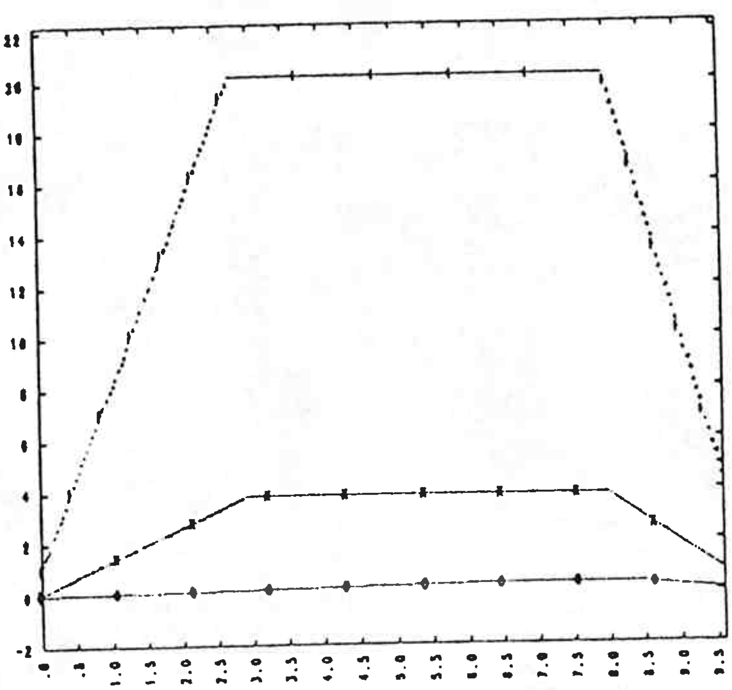
Plasma Current Induction

A major portion of the conceptual design effort has been spent in determining the effect of the single-turn TF coil on the operation of the PF system. If the single-turn TF coil were completely monolithic, the PF coils would be inductively coupled to it as well as the plasma, thereby reducing the efficiency of induction. To address this potential



PLASMA CURRENT (AMP) -VS- TIME (SEC)
O=PLASMA CURRENT
X=MINIMUN CURRENT FOR ALPHA CONFINEMENT

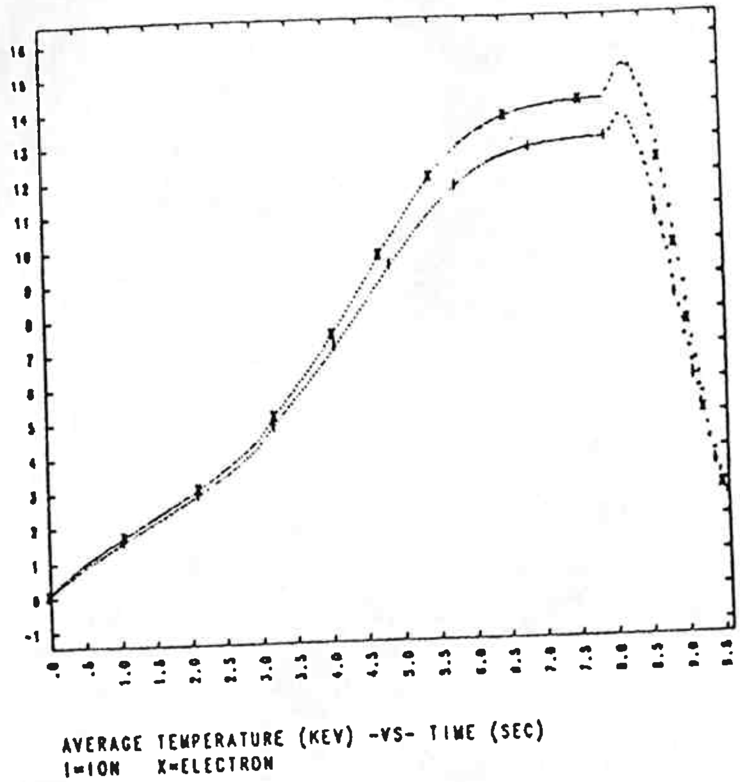
a. Plasma current



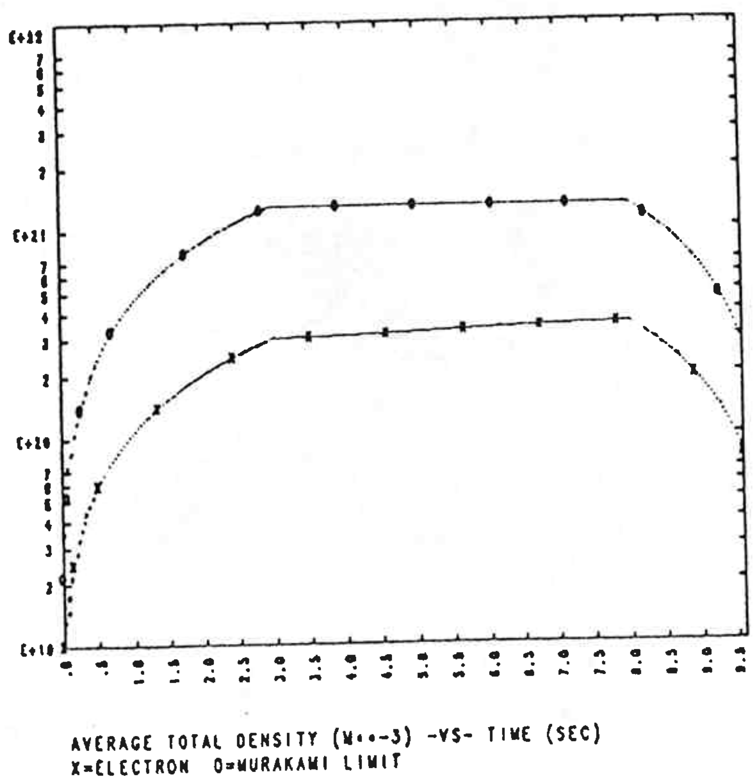
MAGNETIC FIELD (TESLA) -VS- TIME (SEC)
I=TOROIDAL X=POLOIDAL
O=VERTICAL

b. Toroidal magnetic field on the plasma axis

Fig. 3 Simulation of a full plasma discharge

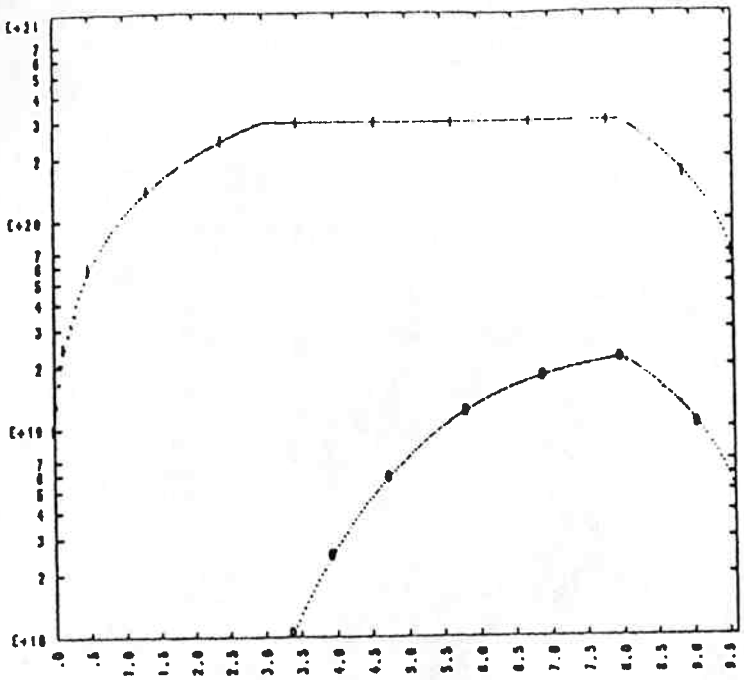


c. Ion and electron temperatures

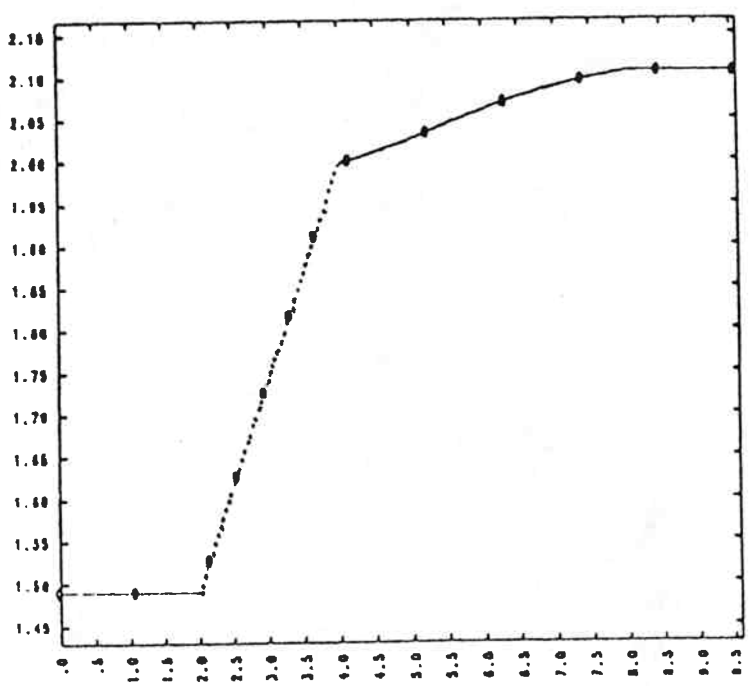


d. Plasma density and Murakami density limit

Fig. 3 (cont.) Simulation of a full plasma discharge

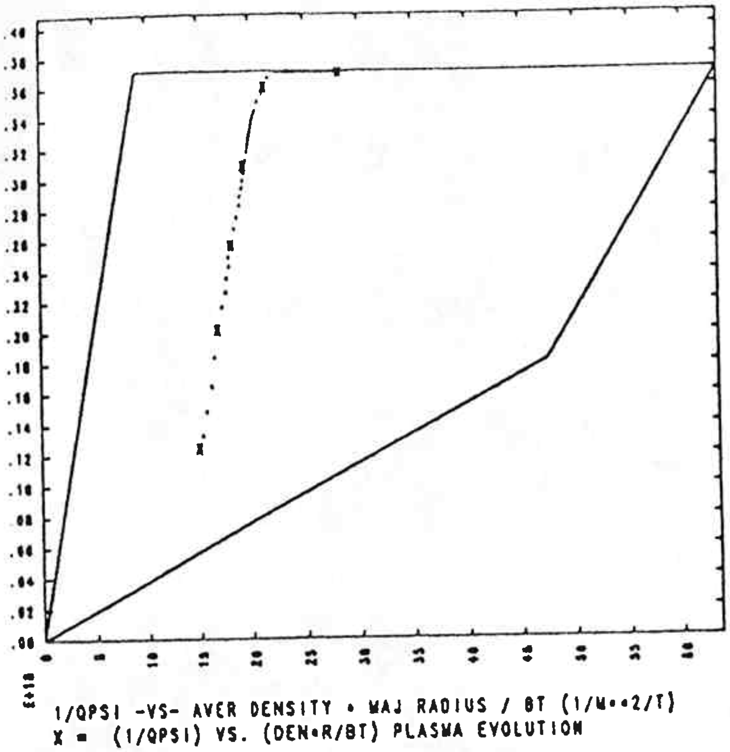


e. Ion and alpha particle densities

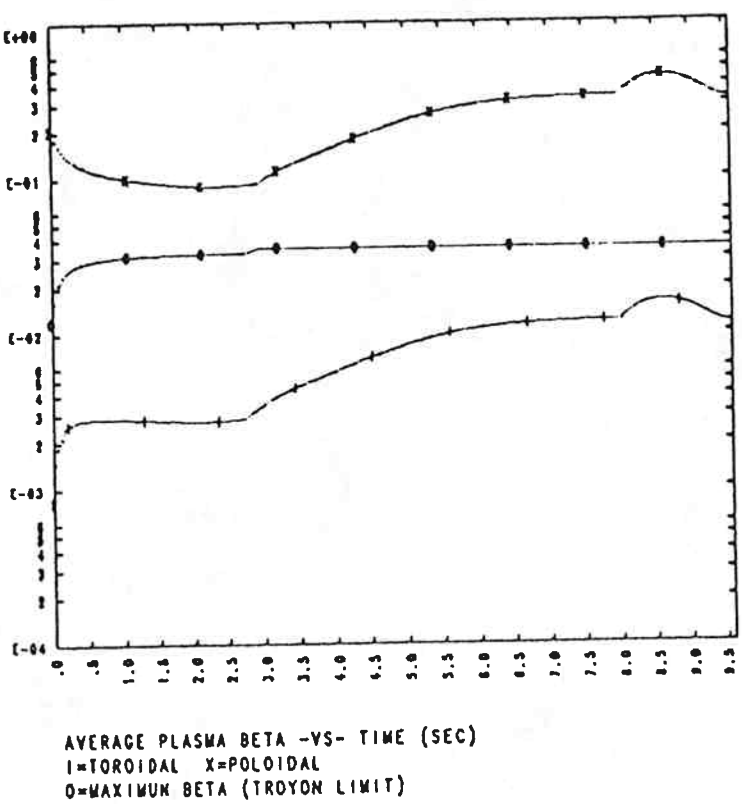


f. Mean ion charge in the plasma

Fig. 3 (cont.) Simulation of a full plasma discharge

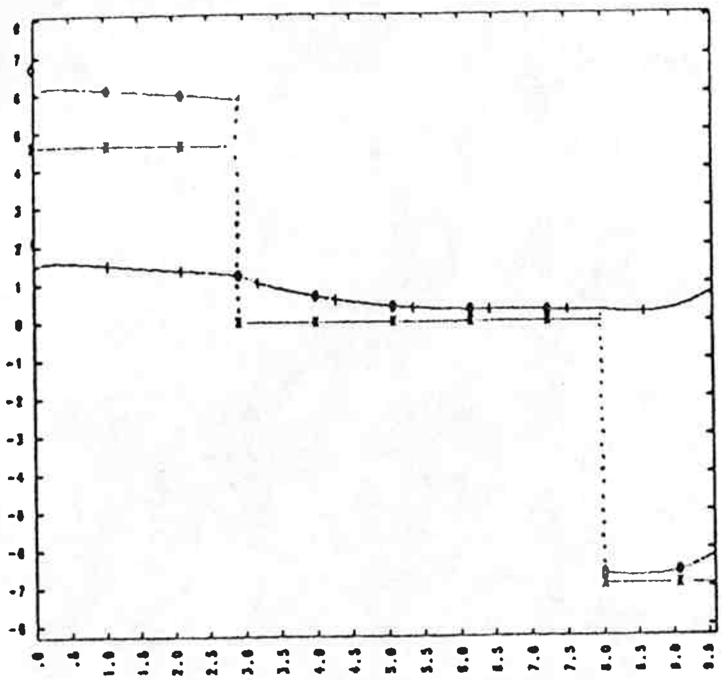


g. Hugill diagram



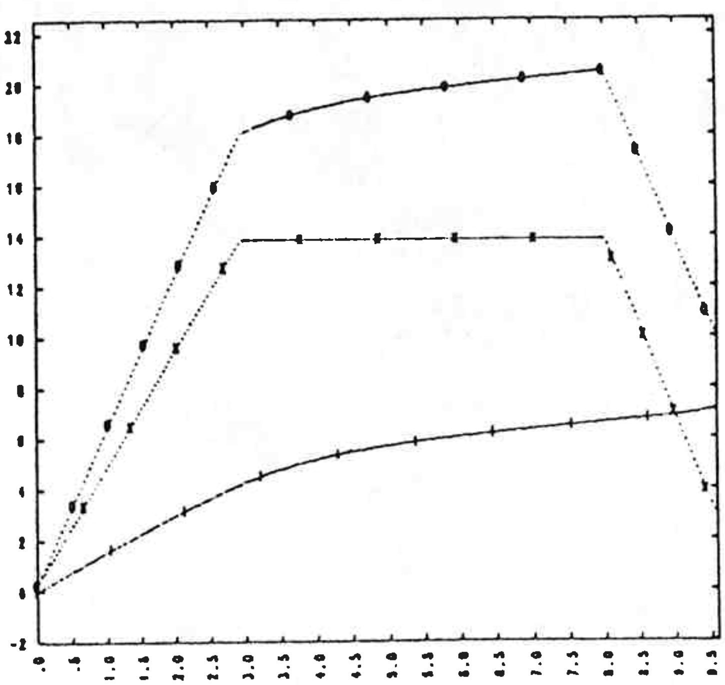
h. Volume-averaged toroidal and poloidal betas. Troyon beta limit.

Fig. 3 (cont.) Simulation of a full plasma discharge



LOOP VOLTAGE (VOLTS) -VS- TIME (SEC)
I=RESISTIVE X=INDUCTIVE
O=TOTAL

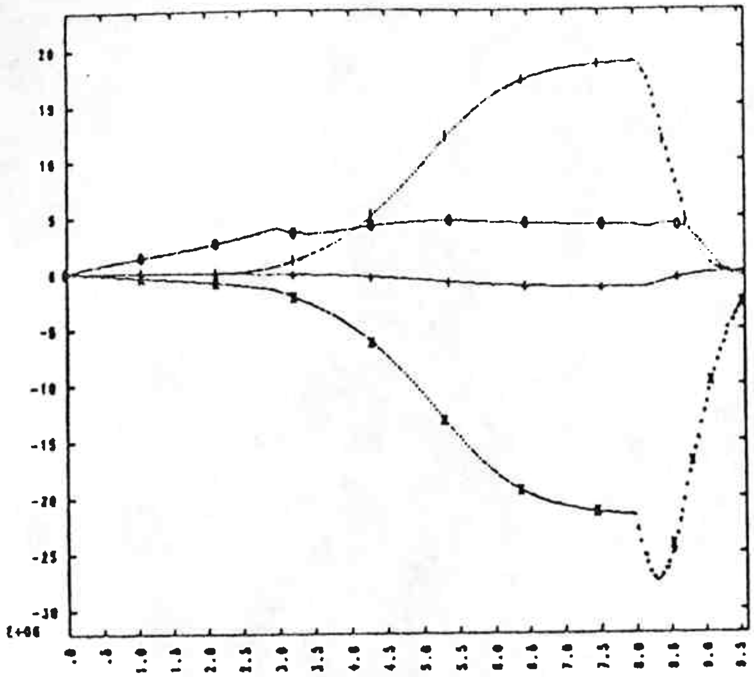
i. Average loop voltage



MAGNETIC FLUX (VOLT-SECONDS) -VS- TIME (SEC)
I=RESISTIVE X=INDUCTIVE
O=TOTAL

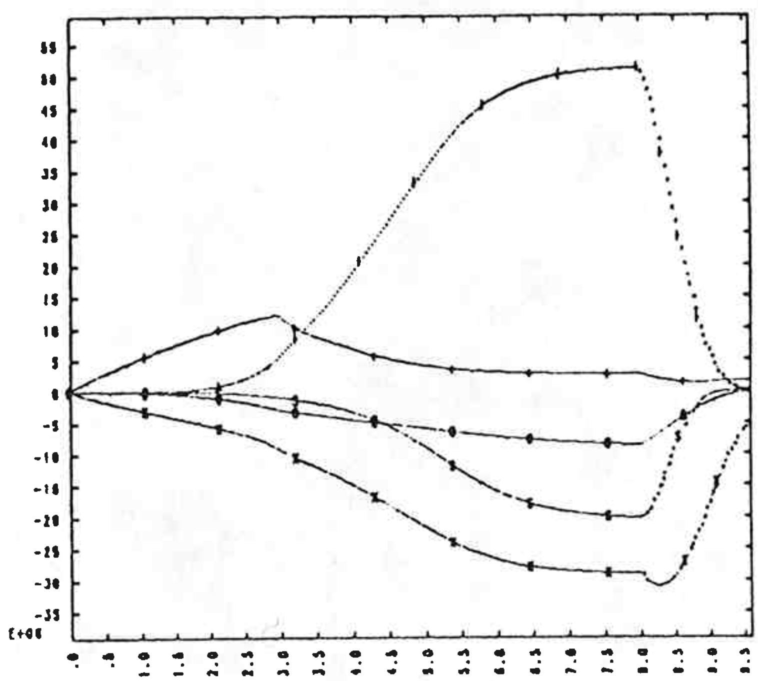
j. Magnetic flux

Fig. 3 (cont.) Simulation of a full plasma discharge



AVERAGE ION POWERS (WATTS) -VS- TIME (SEC)
I = FUSION X = ION ENERGY CONFINEMENT
O = ION-ELECTRON EQUILIBRATION
+ = LOSS DUE TO FUSION REACTIONS

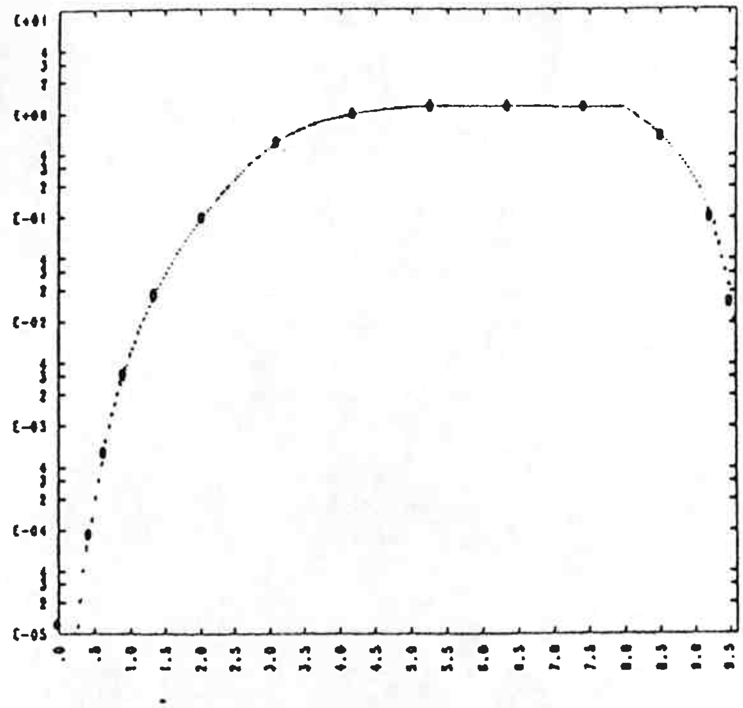
k. Ion power balance contributing terms



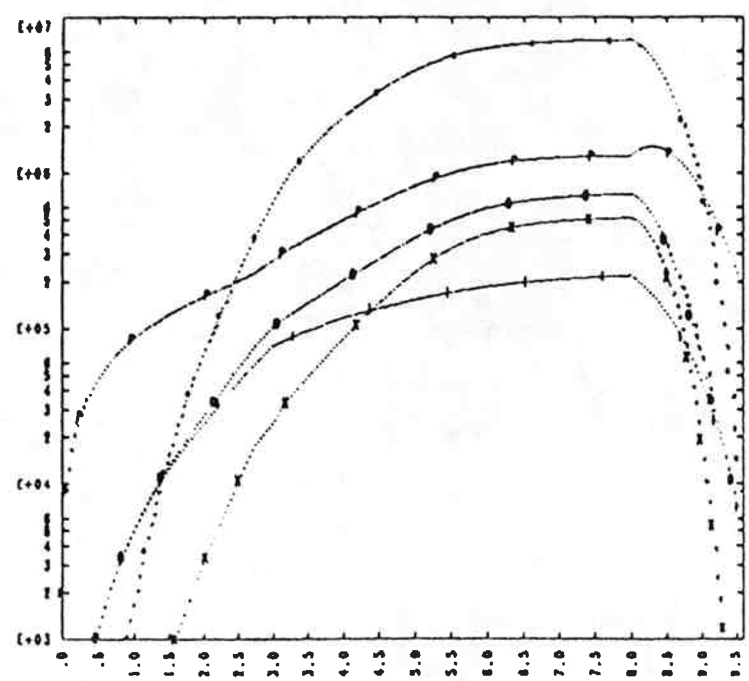
AVERAGE ELECTRON POWERS (WATTS) -VS- TIME (SEC)
I = FUSION X = ELECTRON ENERGY CONFINEMENT
O = BREMSSTRAHLUNG S = SYNCHOTRON
+ = OHMIC HEATING

l. Electron power balance contributing terms

Fig. 3 (cont.) Simulation of a full plasma discharge



m. Ignition factor



n. Power losses to the first wall

Fig. 3 (cont.) Simulation of a full plasma discharge

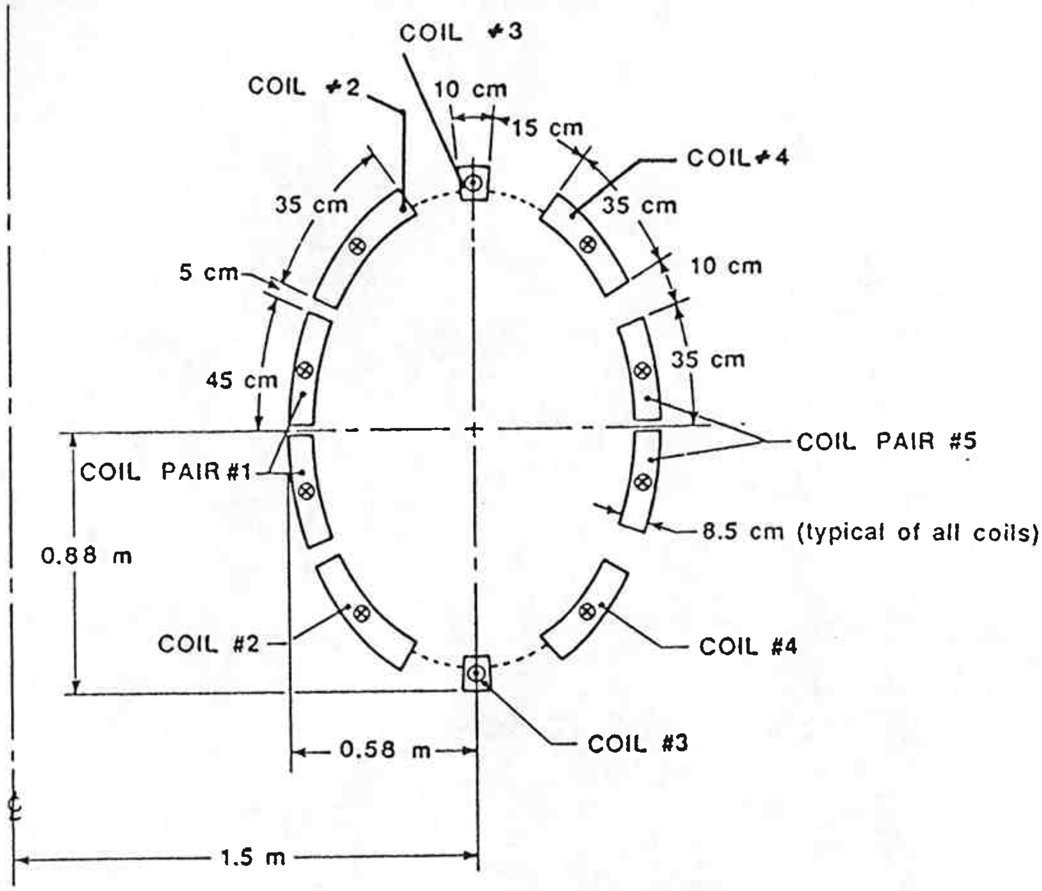


Fig. 4 Poloidal field coil system

Table 1

PEAK CURRENTS AND CURRENT DENSITIES IN THE PF COILS

PF Coil	Peak Current (MA)	Peak Current Density (MA/m^2)
1	3.61	94.5
2	2.55	85.7
3	0.37	39.9
4	2.24	75.0
5	2.24	75.0

Note: The current in coil pair 3 is in the toroidal direction of the plasma current and opposite to the current in the other coil pairs.

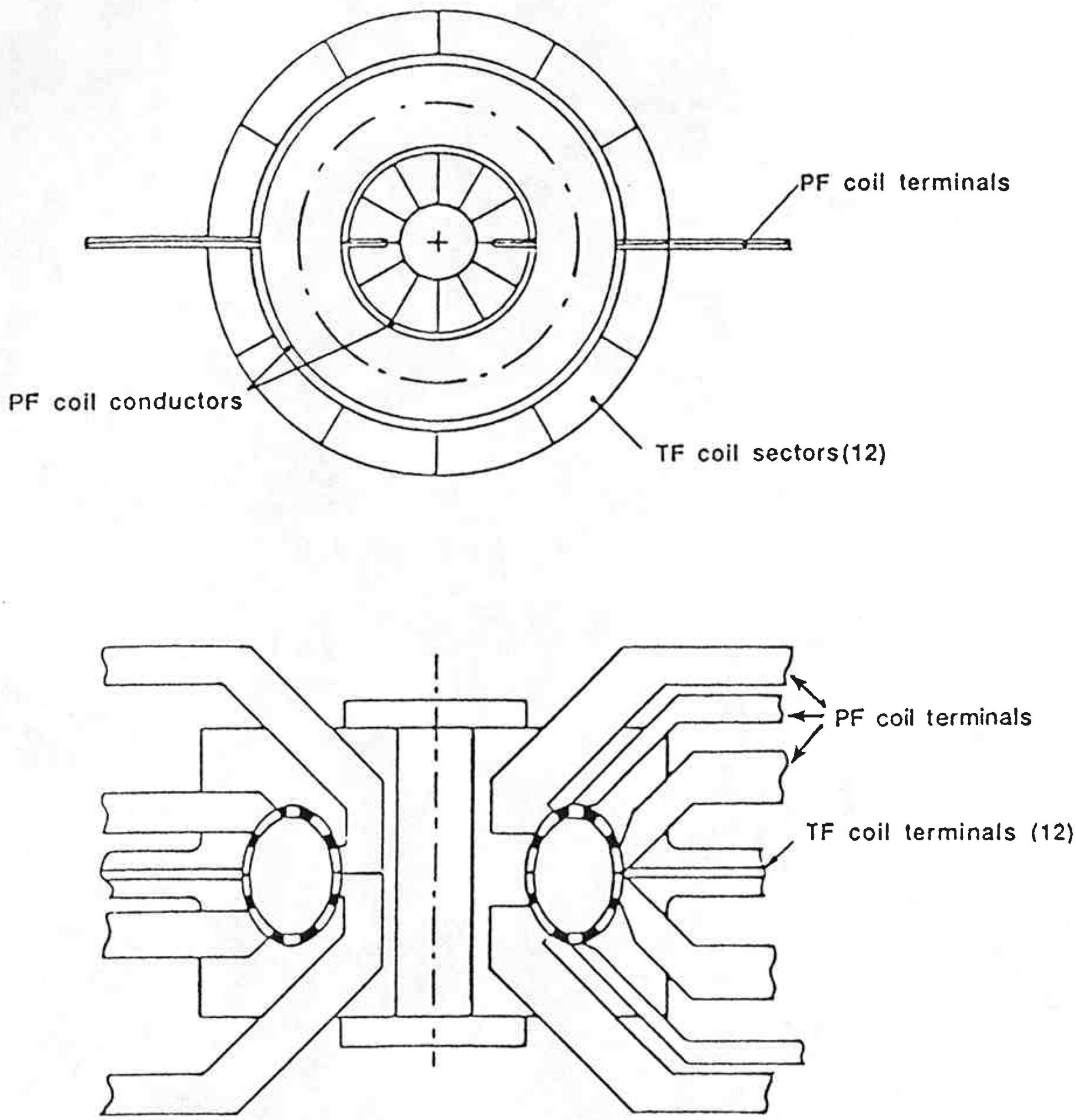


Fig. 5 Arrangement of poloidal field coil terminals

problem, two baseline cases have been analyzed. Each one uses the profile consistency requirement for the plasma current diffusion, and satifies the classical equations for eddy current diffusion into the TF coil. A linear current ramp from 22 MA to -15.7 MA in the PF coils over discharge has been assumed. The first case assumes that the TF coil is made of an infinite number of laminates with insulation between them. The resulting induced plasma current is 15.7 MA, or 31% more than the nominal value in the design. The second calculation assumes that the TF coil is monolithic copper, with no insulating slits to break up the eddy current path. A plasma current of 10.2 MA results. These results indicate that some slitting of the TF coil is required; however, as a result of its laminated construction, including the insulating slits is very easily accomplished.

The conceptual design presently calls for the insulating slits to be placed every 5 cm around the torus, and they will extend approximately 15 cm radially out from the plasma bore. With this arrangement, the assumed current ramp will induce 13.8 MA in plasma, about 15% over the 12 MA design value. The magnetic surfaces within the plasma bore remain closed for the entire duration of the discharge using this particular geometry. An example of the magnetic surfaces generated is shown in Fig. 6.

Electromechanical Analysis of the Poloidal Field System

Once the electromagnetic and plasma current induction requirements had been determined, a mechanical analysis of the PF coil system was performed using a 2-D finite element code. With current profiles assumed as described above, the stress and temperature distributions and energy requirements were calculated. It was assumed that the coils were unsupported and made of dispersion-strengthened copper alloy in the fully work-hardened condition. Coil stress can be reduced by using the TF coil to support some of the PF coil loading. Obviously, coil pair 1 will experience the highest temperature and stress since it operates at the highest current density; thus only the results of the analysis for it are presented. Shown in Figs. 7 and 8, the peak stress in the coil reaches 483 MPa (69.8 ksi) and the coil hot spot is 87°C if the coil is

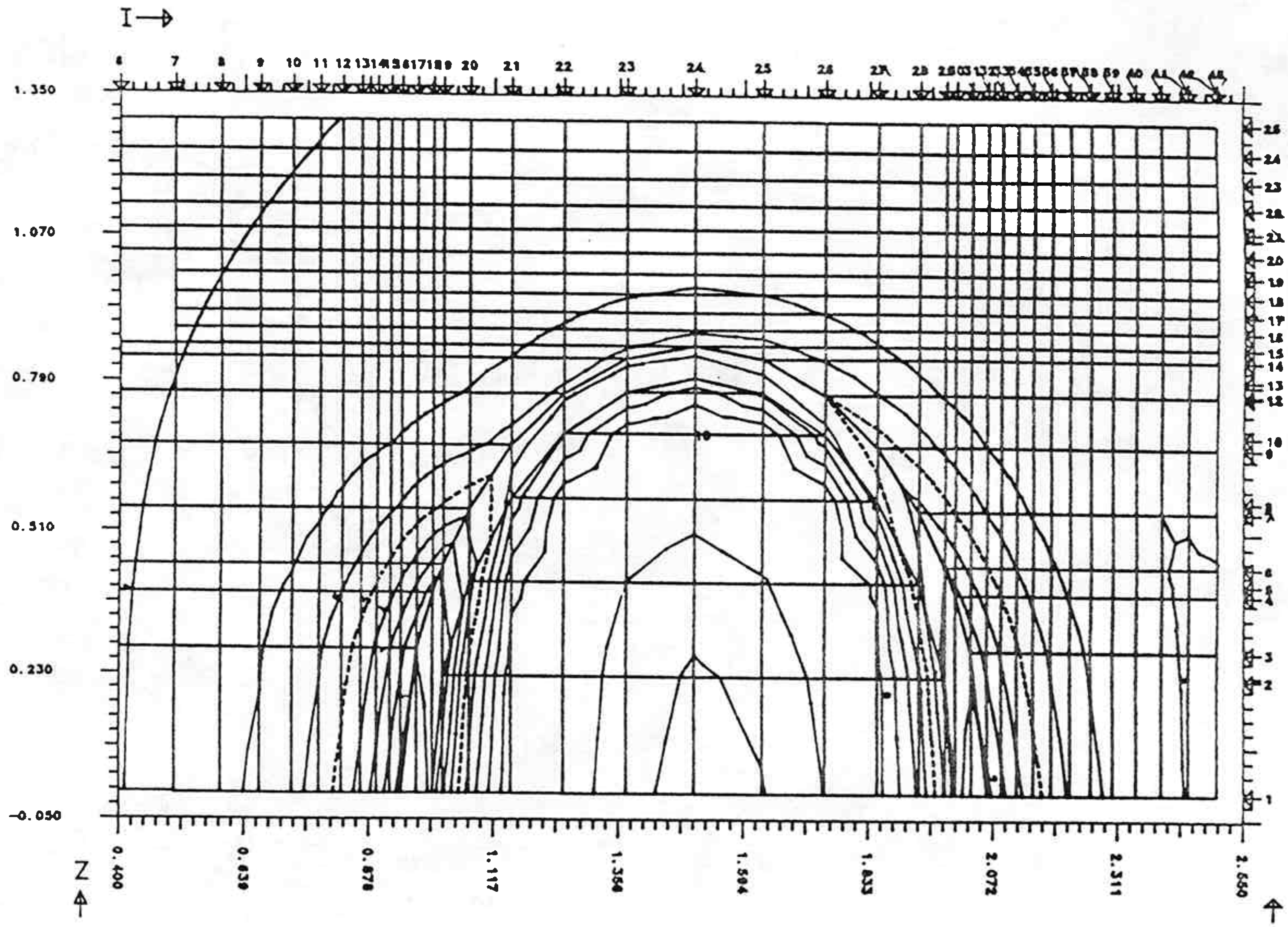


Fig. 6 Magnetic surfaces in and around the plasma bore at 2 sec after discharge initiation

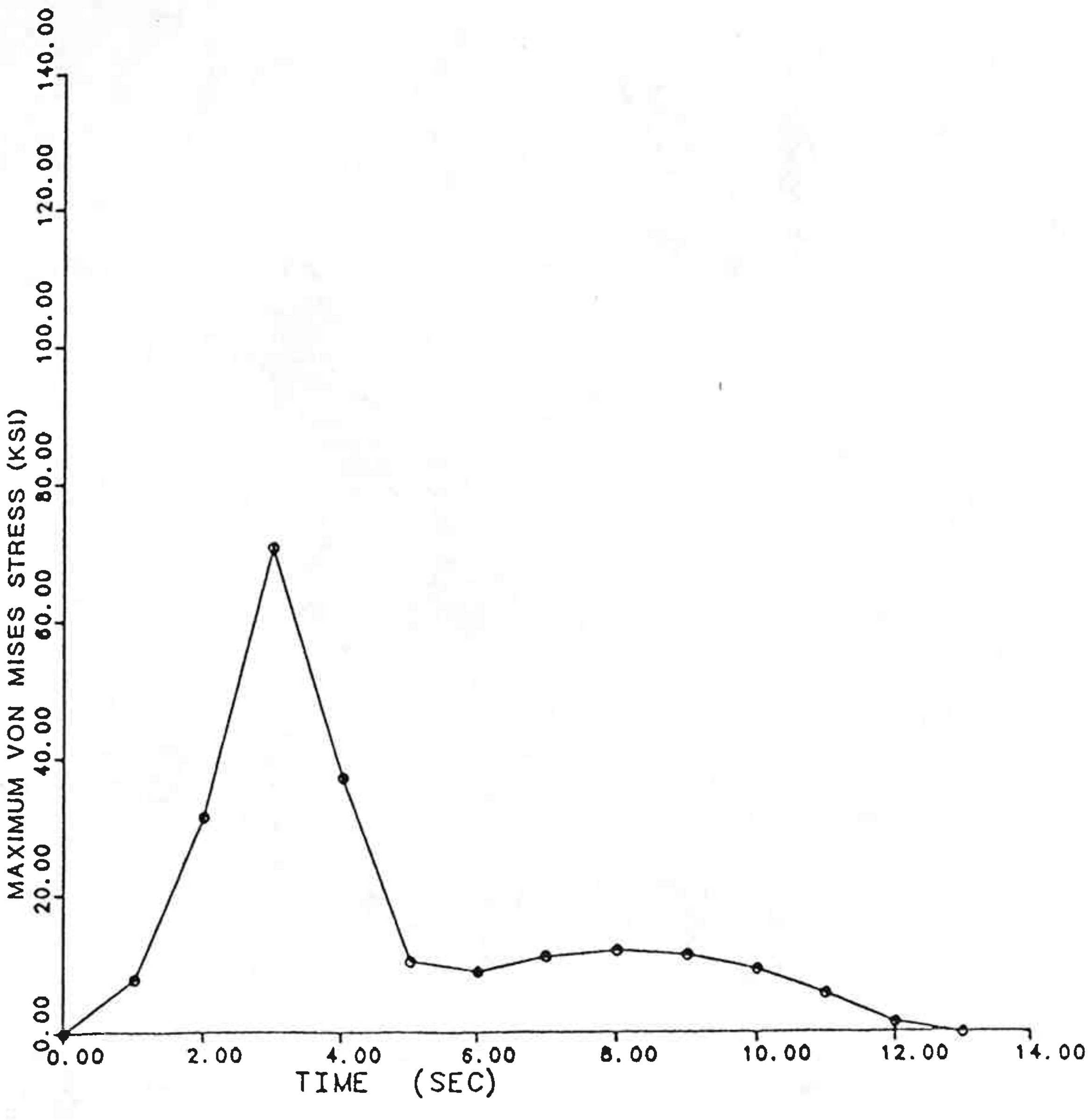


Fig. 7 Evolution of the maximum Von Mises stress in the unsupported poloidal field coils

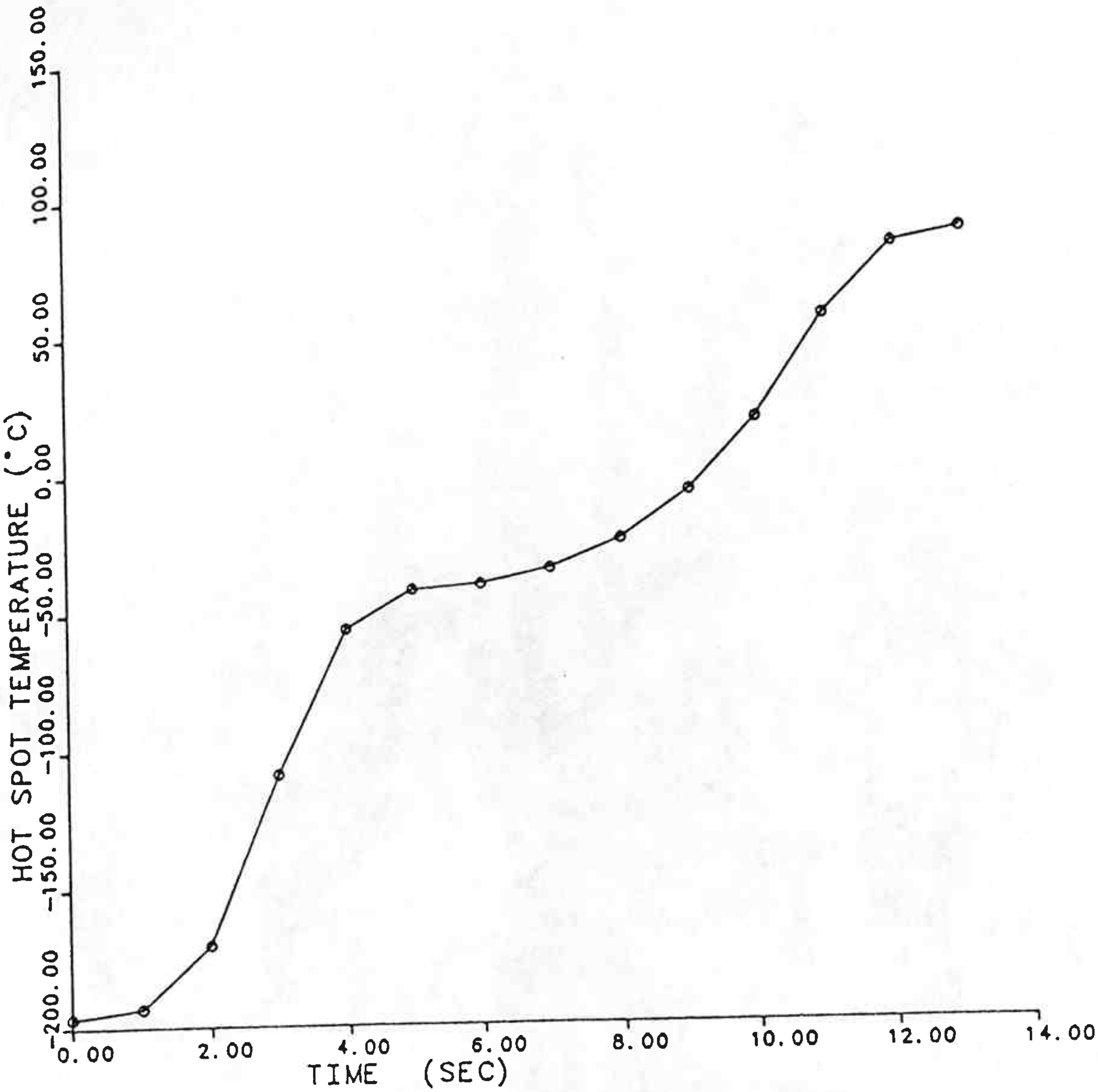


Fig. 8 Evolution of the maximum temperature in the poloidal field system

initially precooled to liquid nitrogen temperature. With these stress levels, the dispersion-strengthened copper alloy is well suited for the application. Sizeable forces will also be experienced by the terminals of the coils; however, by trapping the terminals within the TF coil, these forces can be sustained by the structure of the single-turn coil.

Energy required for the poloidal field system is divided as follows:

Magnetic energy	0.300 GJ
Coil resistive losses	1.320 GJ
HPG resistive losses	0.474 GJ
TOTAL ENERGY REQUIRED	2.094 GJ

As with the TF power supply, the energy is initially stored in the HPG rotors. Five homopolars will be used for maximum flexibility of operation, and of these, two will store 700 MJ each, two will store 600 MJ each, and the remaining one will be sized at 40 MJ. The generators are capable of storing more energy than is needed, and then the operating point for these power supplies can be conveniently optimized.

Cost Estimates

The proposed IGNITEX experiment offers a straightforward, cost-effective approach to achieving controlled fusion of a D-T plasma in a timely manner. Several features of the experiment make possible a relatively low-cost estimate for the device. One of the best cost-saving features, however, is dedication of the project to the single goal of achieving ignition for scientific study of ignited plasmas, and using the most direct appoach available to achieve it.

An initial cost breakdown is shown in Table 2. The costs are based on the assumption that the experiment is located at the Balcones Research Center (BRC), a research campus of The University of Texas at Austin located in north Austin. BRC facilities include 85 MW of available grid power; the CEM machine shops and assembly areas; the Center for High Performance Computing, which includes a Cray supercomputer available for use on the project; a library service that accesses all university libraries; and various other support utilities including chilled water for cooling. An added attraction is the

Missouri-Pacific railroad, which runs right through the BRC campus and could be used to transport large equipment to the site.

Other factors contributing to the cost effectiveness of the project include the simplicity of the TF coil construction, the use of in-bore PF coils which drastically reduce their size and power supply requirements, the relatively inexpensive HPG power supplies, and the absence of any auxiliary plasma heating requirements and divertor system.

For purposes of cost estimation, the program to design, fabricate, and test the IGNITEX experiment has been set up on a six-year schedule. The first phase is a one-year conceptual design in which a complete analysis of the IGNITEX experiment is performed and a proof-of-principle experiment is designed, fabricated, and tested. Phase 2, also one year, is devoted to detailed design of the IGNITEX experiment and continued testing of prototype experiments. Approximately three years are planned for construction of the experiment, and one year for commissioning and testing. Project and administrative costs for this six year period are estimated to be $20 million.

Cost of the single-turn TF coil is based on material costs of $10 million (one million pounds of dispersion-strengthened copper at $10/pound), fabrication and machining costs of $8 million, and assembly tooling costs of $2 million. The cost of the PF coils includes 30 tons of dispersion-strengthened copper at $10/pound, machining and tooling costs of $1.6 million, and assembly costs of $600,000. The TF coil support system is made from mild steel uprights and cross members which together weigh approximately one million pounds. Cost for the mild steel is appoximately $1.50/pound. The hydraulic press to apply the compressive preload will cost about $4 million, and the fabrication costs are estimated at $2 million.

Power supply costs shown in Table 2 include fabrication and installation into the IGNITEX building. The costs shown also include instrumentation and control systems for the generators. The PF power supply cost is inexpensive compared with the TF system because it uses virtually all of the design work and fabrication tooling used to build the TF power supply. Facility costs are based on a building that

Table 2

ESTIMATED COSTS OF THE IGNITEX EXPERIMENT

Title	Cost ($ million)
Project	20
Single-turn TF coil	20
PF coil system	2.8
TF power supply	22.5
PF power supply	5
TF coil support structure	7.5
Facility	8
Vacuum vessel	2.5
First wall	1.5
Shielding	1.0
Diagnostics	2.0
Instrumentation and control	3.0
Cryogenics	2.5
Fueling	0.5
Vacuum pumping	3.5
Disposal	4.3
Remote maintenance	3.0
ESTIMATED COST	$109.6 million

includes, among others, a test bay for the experiment, an assembly area, and office space to support the project. A 100-ton polar bridge crane over the test bay and a 50-ton parallel bridge crane for the assembly area are included in the costs.

The additional costs associated with the IGNITEX experiment are based on the collective experience of engineers at The University of Texas at Austin. CEM has accumulated a great deal of experience in the design and manufacture of large high-vacuum systems, cryogenic cooling, and insulation of pulsed coils. Other cost estimates are as indicated in Table 2.

Conclusion

The work presented here (together with the work reported in the preceding TAERF Report No. 42) addresses the basic design issues in the IGNITEX fusion ignition experiment. The analysis of the predicted plasma behavior indicates that this device offers a simple means of producing ignited plasmas. The electromechanical design produces the magnetic field and current needed by the plasma to ignite. Furthermore, the cost of the experiment is very attractive.

A number of clearly defined advantages of the IGNITEX design with respect to previously proposed ignition experiments make the IGNITEX project worthy to be further pursued. With this experiment The University of Texas has a unique possibility to make an important contribution to thermonuclear fusion research even within the present strongly restricted budget for fusion research.

A proposal is being written to seek funding from the U.S. Department of Energy for the next phase of the IGNITEX project. In that phase a conceptual design of the experiment will be made. This effort will include the study of plasma behavior problems related to equilibrium, transport, and stability. Refinements of the electromechanical and power supply designs will be carried out. In addition, a small prototype of the 20 T single-turn coil will be designed, fabricated, and tested using power supplies operating at the Center for Electromechanics. Issues dealing with the nuclear safety analysis of the proposed facility, the plasma breakdown, the diagnostics and operation of the plasma experiment, the first wall and limiter designs, and the remote maintenance of the device will be studied. As a result, the conceptual design will give a consistent and quite complete picture of the IGNITEX experiment.

References

R. Carrera, E. Montalvo, and M. N. Rosenbluth, Bull. Am. Phys. Soc. 31, 1565 (1986).