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THE PPPL TOKAMAK PROGRAM

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The economic prospects of the tokamak are reviewed briefly and found to be favorable -- if the size of ignited tokamak plasmas can be kept small and appropriate auxiliary systems can be developed. The main objectives of the Princeton Plasma Physics Laboratory tokamak program are: (1) exploration of the physics of high-temperature toroidal confinement, in TFTR; (2) maximization of the tokamak beta value, in PBX; (3) development of reactor-relevant rf techniques, in PLT.

1. Tokamak Reactor Prospects

The historical objective of the magnetic fusion research effort has been the achievement of adequately good plasma confinement to allow ignited equilibrium burn in a reactor of moderate size. The tokamak configuration is well suited to attain this objective: Classical transport theory permits a wide range of tokamaks to reach ignition by the spontaneous ohmic-heating process, at plasma current levels as low as 3 MA. Present experiments indicate that nonclassical transport phenomena are likely to raise this minimum current requirement severalfold, and that auxiliary heating will be needed to achieve ignition at practical levels of magnetic field strength. The sensitivity of the ignited plasma regime to impurity effects places additional demands on the technology of edge-plasma handling. Even so, there is a worldwide consensus that ignited long-pulse operation should be achievable in a tokamak device of a size and cost not greatly exceeding the standards set by the present generation of large tokamak research facilities.[1] Some illustrative parameters might be: $I \sim 10$ MA, $R \sim 3$ m, $P_{fusion} \sim 300$ MW, $\tau_{burn} \sim 300$ sec.

Now that the reactor plasma confinement objective is virtually within reach, its historical role in stimulating interconcept competition has diminished. Other aspects of fusion reactor design -- such as fusion power density, burn-pulse length, and blanket geometry -- have been receiving increased emphasis. In this contest for "economic attractiveness," the low-beta, iron-core, copper-shell tokamak of fifteen years ago was not an instant favorite, but the pressure of recent competition has driven tokamak evolution a long ways towards optimal reactor potential.

The projected MHD-stable beta value of the tokamak was originally in the range of only a few percent. Tokamak experiments[2] have now advanced beyond 4% and theoretical predictions range at least to the 20% level -- possibly higher.[3] The situation has actually become rather disconcerting to the plasma physicist: Further reductions in the projected cost of fusion power seem to be more sensitive to improvements in the geometry and lifetime of first-wall materials than to the achievement of higher beta values. If the ultimate high-beta limit on fusion power density is to be found in nuclear-engineering considerations, one may also conclude that all reactor concepts using a first wall and blanket of toroidal topology must attain roughly similar economic performance at their optimal beta values.

As long as the tokamak current is driven by a transformer, the burn-pulse length is necessarily finite. Frequent pulsing is an undesirable reactor feature, since it implies increased thermal fatigue as well as added conventional costs for maintaining the reactor heat flow between pulses. The experimental success of rf current drive[4,5] has given substance to the possibility of steady-state tokamak reactor operation,[6] but the associated recirculation of electric output power is an economic liability. An approach that would seem to meet all economic concerns is the use of rf current ramp-up[4,5] to conserve transformer volt-seconds and facilitate efficient long-pulse operation in a set of moderate-sized tokamak reactor units sharing common facilities.

Economies of scale tend to force the minimum size of magnetic fusion reactors up into the 1000 MW range.[6,7] In the case of the tokamak, the projected plasma confinement capabilities of a single 1000-MW electric power reactor exceed the requirements for ignition, so that one could well envisage subdivision into smaller units[8]: for example, four individual tokamaks operating with their burn cycles out of phase to produce a total of 2000 MW. The obvious penalty for subdividing the mechanical part of the reactor can be offset by the common utilization of auxiliary power supplies and other assets -- particularly if tokamak reactor design is optimized *ab initio* with a view to multiunit operation. For example, rf current ramp-up and rf heating to ignition are tokamak design features that lend themselves well to multiunit economies.

In summary, the tokamak reactor approach turns out to have favorable prospects, not only for demonstrating the experimental reality of fusion power, but also for economic competition. The following two sections provide a brief review of PPPL efforts to improve tokamak plasma parameters and develop economically promising auxiliary techniques. The final section gives a brief outline of future plans.

2. Stability and Confinement

2.1 Ohmic-Heated Regimes

The constraint of ohmic heating tends to ensure that tokamak plasmas cannot exceed the MHD-stability beta limit. There remains a wide spectrum of resistive kink modes that can disrupt the discharge or seriously degrade plasma energy confinement; however, the stability of these modes depends on the safety-factor profile $q(r)$ in a manner that is fairly well understood and subject to external control.[9]

Kink modes tend to be stabilized by nearby conducting walls or by neighboring plasma regions in which the kink-mode-number ratio m/n does not match the local $q(r)$. In particular, the specification $q(o) \lesssim 1$, which tends to be met naturally in tokamaks, has a strong stabilizing influence on all the higher-m, $n = 1$ modes and is found to be associated with optimal confinement conditions. This result may seem somewhat paradoxical since the "safety factor" q was originally meant as a figure of merit in regard to MHD stability (i.e., $q \gg 1$ was supposed to be safest). The dreaded ideal MHD mode $n = 1$, $m = 1$, which is strictly localized within the hot plasma core, has turned out to cause much less energy loss than the resistive MHD modes $n = 1$, $m = 2, 3, \ldots$, which link the plasma core with the cold plasma edge.

The main test bed of ohmic-heating research at PPPL is currently the TFTR device[10] (Fig. 1). A total of about 60,000 discharges has been produced to date, including vacuum-conditioning discharges and some 1,000 documented "high-quality" tokamak plasmas. Energy confinement has been studied at currents up to 1.5 MA and magnetic fields up to 2.7 T -- about half the machine design levels. Some typical results are shown in Figs. 2-4.

The pulse shapes of the plasma current and density in Fig. 2 were specified by the experimenters and imposed on the plasma by feedback control. A one-second plateau is needed to ensure that transients in the plasma current-density distribution and loop voltage have time to decay. The measurements of electron temperature in Fig. 3 were based mainly on electron cyclotron emission and X-ray spectra; recently the TV Thomson scattering system has come into operation and has provided an additional check. The ion (deuteron) temperature has been inferred from the neutron yield and also from the Doppler broadening of impurity X-ray lines. At lower plasma densities, passive charge-exchange measurements have been used for confirmation of T_i.

The sawtoothlike excursions of the central electron temperature in Fig. 3 correspond to the desirable tokamak operating mode, where $q(o) \lesssim 1$. The sawtooth cycle begins with the slow diffusion of poloidal flux into the high-conductivity region around the magnetic axis, accompanied by a drop of q(o) below unity, and the quasi-static growth of an $n = 1$, $m = 1$ island around a second magnetic axis with $q = 1$ and lower T_e. There follows a process of rapid magnetic reconnection during which the old magnetic axis is replaced by the new one. The maximum period τ_s observed for the sawtooth cycle is already quite long in the case of Fig. 3. The period τ_s should scale up with the plasma electrical conductivity and should ultimately approach one second in the beam-heated plasma regime of TFTR. Since the energy confinement time τ_E is likely to be shorter, the TFTR experiment will explore an interesting transition from the regime $\tau_E >> \tau_s$, which has characterized tokamak experiments to date, to the regime $\tau_E << \tau_s$, which is characteristic of tokamak reactors. The latter regime is attractive, since anomalous transport due to the sawtooth "dynamo-mechanism" must, on the average, diminish with $1/\tau_s$. The best instantaneous values measured for τ_E should therefore be enhanced. On the other hand, if sawtooth activity is allowed to persist at all, the $n\tau_E$-value will now fluctuate in time -- possibly resulting in an unstable burn cycle and periodic extinction of the fusion reaction.

The small differential between T_e and T_i in Fig. 3 is indicative of a high degree of electron-ion equilibration and a long ion-energy confinement time τ_{Ei} -- 500 msec or more in the best cases. An accurate determination of the magnitude of ion heat flow relative to neoclassical expectation has been difficult, since the loss rate is so small, but the multiplication factor is believed to be generally less than 5. The dominant energy loss mechanism in TFTR, as usual in tokamaks, is the anomalous electron thermal conductivity.

In the initial phase of magnetic fusion research, when energy confinement times were typically in the 10-μsec range, considerable comfort was derived from the idea that even an anomalous τ_E should scale up in the classical manner, as the square of the plasma size. Originally, this conjecture was criticized on the basis that heat transport might turn out to be convective rather than diffusive, resulting in a weaker size-dependence. More recently, a number of tokamak researchers[11,12] including the Alcator C group,[13] have suggested that τ_E might actually have a <u>cubic</u> size dependence -- something like $\tau_E \propto nR^2a$. This "neo-Alcator" scaling was shown to provide a fairly good fit to all available tokamak data in the ohmic-heating regime. The cubic size dependence has now been confirmed by the TFTR results as well (Fig. 4). There is a remaining uncertainty factor of perhaps $(R/a)^{\pm 1/2}$ in regard to the exact trade-off between the R- and a-dependences of τ_E, and there is also a multiplier of about $q^{0.8}(a)$, reflecting an increase in MHD activity when the ratio $q(a)/q(o)$ (i.e., the measure of average tokamak shear) becomes too small.

Neo-Alcator scaling has many attractive features. A transport coefficient $\chi_e \propto a/nR^2$, which can be viewed as reflecting unfavorable dependences on the mirror ratio along field lines and the steepness of the plasma edge, is conducive to explanation by microscopic turbulence theory. From the point of view of dimensional analysis, the apparent proportionality of τ_E to the large dimensionless quantity nR^2a is also intriguing: If this were precisely true, then the tokamak confinement time scale would have to be some product of fundamental constants, such as e^2/mc^3. According to Ref. 14, the options allowed in terms of real plasma-physical mechanisms are more limited: The neo-Alcator formula would have to be modified at least slightly -- perhaps in the direction of the scaling law advocated in Ref. 12. Ohmic-heating experiments, of course, are notoriously ill-suited to establish true physical parameter-dependences, since the ohmic power balance is present as an implicit constraint. For example, at the Murakami limit,[15] a relationship between ohmic heating and radiation cooling can be expressed in the approximate form $B_p/a \propto n$ (valid when q(a) is not too small), and the neo-Alcator formula can then be rewritten as $\tau_E \propto B_p R^2$ -- similar to the GMS scaling[16] found by the Kurchatov tokamak group in the 1960s. (See also Fig. 5.)

In the initial phases of auxiliary tokamak heating, there was an optimistic expectation that the ability to vary the plasma temperature independently of the confining magnetic field should help to clarify the physics of tokamak transport. As will be discussed in Sec. 2.2, actual confinement phenomena have turned out to be far more complex in the auxiliary-heated case, so that even the nature of the empirical scaling is still controversial. The investigation of the ohmic-heating regime may thus turn out, after all, to be the simplest approach to the understanding of basic tokamak transport mechanisms. Large, well-diagnosed ohmic heaters such as TFTR and JET offer some significant new experimental opportunities. In this context, the adiabatic-compression capability of TFTR (cf. Fig. 1) is particularly promising. The plasma can be removed from limiter contact in a time that is considerably shorter than τ_E, and the process of free expansion into the surrounding "vacuum region" can be followed by the TV Thomson scattering system. The initial results of TFTR compression experiments[10] have confirmed the older evidence from limiter-bounded tokamak plasmas that χ increases sharply towards the plasma edge. Observation of the detailed mechanisms of tokamak free-expansion transport should turn out to be instructive. For example, one might search at the low-T_e edge for evidence of the resistivity-gradient-driven "rippling" mode (a neglected near-relative of the resistive kink mode), which is theoretically expected to become unstable for $T_e \lesssim 50$ eV.

Ohmic heating has also gained some ground as a practical option for reaching ignition temperatures in a tokamak plasma,[17] since neo-Alcator scaling allows the plasma temperature to increase not only in proportion to $B^{0.8}$, but also as $R^{0.4}$. For example, starting from the data of Fig. 3, one finds that the TFTR plasma temperature could be raised to the 5-keV level by increasing the field strength to 10 T. Cross-sectional shaping may turn out to give further improvements. The practical issue is, whether it is less expensive to ohmic-heat the plasma in a large high-field device or to auxiliary-heat it at lower field strengths. Present opinion in the fusion-engineering community favors the latter alternative, but as long as the scaling of τ_E in the non-ohmic-heating regime remains controversial, the existence of the ohmic-heating mode provides a scientifically interesting fall-back option.

2.2 Nonohmic Regimes

The demonstration of substantial nonohmic heating began about a decade ago with the injection of 100 kW of 20-keV neutral beams into 200-kW ohmic-heated ATC plasmas.[18] It progressed to 2.5 MW at 40 keV on PLT[19] and 7.5 MW at 50 keV on PDX.[20] The TFTR objective is to inject 27 MW of 120-keV neutral beams by 1986. The TFTR compression capability can be used to double the beam energy in about 30 msec, corresponding to an equivalent beam power imput of order 100 MW. Two of the TFTR neutral beam lines are now in place, and plasma-heating experiments are scheduled to begin in the second half of 1984.

Thus far, neutral beam injection has served as the main tool for the study of auxiliary-heated tokamak confinement. It has the advantage of being compatible with reasonably broad ranges of parameter variation in the target plasma. When optimal injection and limiter techniques are used, the influx of gas and impurities can generally be kept from dominating the experiments. Available neutral-beam-power levels have been more than enough to drive tokamak beta values to their MHD limit. Stability and energy confinement in high-powered PDX neutral-beam-heating experiments are reviewed in a separate paper [21]; the present discussion is therefore limited to more general remarks.

The overall pattern of present tokamak scaling laws is illustrated schematically in Fig. 5. In the ohmic-heating regime, there is always some maximum energy confinement time that can be obtained for a given poloidal field strength B_p in a plasma of size L. As was noted in Sec. 2.1, when neo-Alcator scaling is obeyed (the usual case), the Murakami limit implies $\tau_{E\ max} \propto B_p L^2$, which is shown as line A in Fig. 5. During the late 1970s, when the currently operating generation of neutral-beam-heating experiments was just getting under way, there was an expectation that increased heating power would allow plasma densities to rise well beyond the Murakami limit, giving correspondingly large improvements of τ_E, as indicated by line B of

Fig. 5. In the subsequent experiments,[2,20] the ability to exceed the Murakami limit was readily demonstrated, but the observed τ_E values have been disappointing (line C). Energy confinement has been found to follow a scaling of the form $\tau_E \propto B_p L^2$ without any consistent dependence on plasma density and with generally less favorable -- rather than more favorable -- maximum values of τ_E than in the ohmic-heating regime.

There are two fundamentally different ways to interpret these auxiliary-heating results: (1) The plasma confinement deteriorates as the plasma pressure is raised, even at beta values well below the MHD stability limit. (2) Perturbation of the normal tokamak equilibrium by intense auxiliary heating serves to enhance anomalous transport. The choice between these interpretations will have important consequences for the tokamak program. A brief discussion of the two alternatives is given below.

The conventional view of the finite-beta limitation in tokamaks is illustrated in Fig. 6. Plama pressures not much below those of a reactor plasma can be attained by ohmic heating, as in Alcator C, but to approach limiting beta values, powerful auxiliary heating is needed. In high-beta experiments to date, the confining magnetic fields have had to be derated as well. Standard tokamak plasmas of round minor cross section should have MHD beta limits around 2-3%, (corresponding to the "Troyon limit" $B_p \lesssim 0.14\ q(a)\ R/a$ for $q(a)\ R/a \sim 5$-7). "Beta-pushing" experiments with such plasmas[2,20,22] seem to encounter increasing difficulty as the MHD limit is approached -- but there has been no direct evidence implicating theoretically predicted pure MHD modes. The only high-beta loss mechanism that has been identified specifically thus far is the "fishbone" instability,[20] which involves a drift resonance of the injected energetic ions with MHD kinks.[23] Perhaps the most gratifying result for MHD theory has been that experimental plasmas with noncircular cross sections do seem to show a marked improvement in the limiting beta values.[2]

A disturbing aspect of the experimental evidence is that, even when the magnetic fields are raised and beta is brought well below its critical value, auxiliary-heated plasmas continue to show a marked deterioration of confinement with increasing heating power. In Ref. 24 this trend was interpreted in terms of a fundamentally adverse plasma-pressure dependence of roughly the form $\tau_{E\ aux} \propto L^2/\beta_p$ (a formula that would have the attractively simple feature that the "ignition margin" $M_{ig} \propto n\tau_E T$ scales as I^2.) Regimes exhibiting this non-MHD type of pressure limitation should be accessible even in ohmic-heated plasmas (i.e., when $\tau_{E\ \text{neo-Alc}} > \tau_{E\ \text{aux}}$), and experimental evidence for the occurrence of such a high-density saturation effect[13] was cited in support of the general thesis of Ref. 24.

An alternative interpretation is that the fragile equilibrium of the ohmic-heated tokamak plasma regime is upset as the auxiliary power becomes large compared with the ohmic power. The quality of tokamak confinement is known to respond to minor changes in any one of the functions $J(r)$, $T_e(r)$, $n(r)$, and $n_{\text{impurity}}(r)$. The radial gradients of some of these functions are theoretically capable of driving relevant instabilities. In addition, these gradients are found to be interdependent through the ohmic-heating/radiation-cooling balance, which is never unimportant in tokamaks. The general rule that at least one-third of the tokamak heat outflow must take the form of radiation cooling presumably plays some necessary role in tokamak profile-shaping, along with the Murakami prescription for density optimization, which has been discussed in Sec. 2.2. In view of this well-documented record of confinement sensitivity to the profile and profile sensitivity to the heat-flow pattern, there is some logic in ascribing related effects to the auxiliary-heating term.

If tokamak confinement is sensitive to details of power input, rather than to the fundamental issue of plasma energy content, then there is hope of discovering special techniques for restoring favorable τ_E scaling. Considerable progress has been made already in this regard -- notably in the case of the ASDEX experiment,[25] which has used divertor control over the plasma-edge conditions to raise confinement times back towards their ohmic-heating values. Similar improvements have been achieved in other divertor and magnetic-limiter experiments,[26,27] as well as in experiments with controlled impurity-ion admixture.[28] During the past year, the saturation of τ_E at high plasma densities in Alcator C, which had been observed in the earlier ohmic-heating experiments,[13] was also remedied by a technique related to conditions at the plasma edge: in this case, the use of pellet-fuelling in place of intensive neutral-gas feed.[29]

In the struggle against the pessimistic scaling law of Ref. 24, a clear factor of four has already been gained on behalf of the proportionality constant standing between L^2/β_p and τ_E. One may hope that in the course of the next several years the critical high-temperature experiments on TFTR and other large tokamaks will permit further gains to be made. Unhappily, by the very nature of a saturation theory, it can never be defeated -- it can only be made to recede. Furthermore, there are some grounds for apprehension in the present pattern of experimental advances beyond the τ_E values allowed by Ref. 24: The best results seem to have been achieved in regimes that are not too close to the MHD beta limit.

Whether one believes that the tokamak beta value will ultimately be limited by orthodox MHD modes or by some anomalous high-beta mechanism, there is a clear incentive to develop tokamak configurations with strong poloidal fields and good MHD stability properties. The outstanding candidate appears to be the bean-shaped tokamak of Fig. 7. The advantage of this shape for satisfying the Mercier criterion up to relatively high-beta values was pointed out in Ref. 30. More recently, extensive computer studies have been carried out to assess the stability of bean-shaped plasmas against ballooning modes[3] and kinks.[31] Bean shaping has turned out to be theoretically advantageous in both these respects. The practical question is whether the value of the high-beta stability features of the bean will be sufficiently great to outweigh the added complications of the required poloidal field system.

In deciding this issue, a good starting point is to review the trade-off between MHD-stability advantages and engineering difficulties in the case of the "conventional" D-shaped tokamak reactor plasma. Use of the D-shape permits the attainment of very high ballooning-stable beta values[32] -- but only at inconveniently low aspect ratios. If the minimization of reactor unit size is considered to be important, then the provision of adequate space for nuclear systems on the small-R side of the plasma implies considerably higher aspect ratios and lower MHD beta limits for a D-shaped tokamak reactor plasma than can be reached in present-day D-shaping experiments.

The special attraction of the bean shape consists in its relatively favorable potential for entering the "second stability regime" of Fig. 8, and operating there relatively free from MHD modes. This potential is insensitive to the plasma aspect ratio, so that the conventional reactor-engineering drawbacks of high-beta operation are removed. Instead, there is the potential inconvenience of the poloidal "bean-shaping coil." In present-day experiments, such as the PBX device of Ref. 33, a trapped bean-shaping coil presents no special problems. In a reactor, a hard decision will have to be made between permitting a trapped poloidal field coil (inside the tokamak-field coils but outside the blanket and shielding), or accepting a considerable increase in poloidal-field-coil system requirements and toroidal-field-coil torques.

The level of engineering hardship that reactor designers may be willing to accept for the sake of very high MHD beta limits will be strongly influenced by the degree of difficulty that beta-pushing experiments encounter during the next several years and the degree of success that D-shaping and bean-shaping experiments will have in overcoming these troubles. Experiments on PBX have just begun and have already demonstrated that the basic low-beta bean shape can be formed and maintained. Initial neutral-beam injection experiments are now underway. The ultimate PBX injection capability of 7 MW should be sufficient to provide a significant test of high-beta stability and confinement in the course of the present year.

3. Auxiliary Features

3.1 Impurity Control

During the initial phase of experimental tokamak research, the plasma edge was defined by a simple material limiter (Fig. 9a). The main improvement during this phase was the progress from heavy-metal limiters, which caused impurity-radiation losses from the hot plasma core, to much lighter limiter materials, notably graphite, which are associated with impurity radiation from the plasma edge. The latter type of "radiation loss" does not directly affect the total energy confinement time, and may actually be advantageous in helping the plasma achieve radial profiles conducive to the minimization of anomalous transport.

In TFTR ohmic heating experiments to date, movable graphite limiters have been used (Fig. 1). In the initial experiments[10] they were coated with titanium carbide. As illustrated in Fig. 10, fairly low values of Z_{eff} were typically achieved at densities in the range $\bar{n} \sim 3 \cdot 10^{13}$ cm^{-3}, but operation at the highest current levels was found to be associated with an increased influx of titanium, as well as other impurities, and a corresponding rise in Z_{eff}. The titanium-carbide limiter coating has now been removed, and the influx of titanium has diminished markedly. Whether improvements in the titanium-carbide coating technique would have yielded more favorable results has not been determined; perhaps an even more intriguing question is whether the spontaneous coating processes that go on during tokamak operation and discharge cleaning could be used to regenerate an optimal carbide coating on a continuing basis.

In regard to the key issue of heavy impurity radiation, the TFTR limiters have been giving satisfactory performance for some time: bolometric measurements indicate that the contribution of radiation cooling to τ_E can be neglected.[10] The critical challenge, of course, will come in the neutral-beam-heating phase. The TFTR strategy for impurity control in this phase has been to minimize the total energy input required to reach the plasma-temperature objectives, by using plasmas of moderate size and providing an option for fast adiabatic compression (cf. Fig. 1).

The limiter approach to tokamak-plasma edge control is clearly incompatible with longer-term reactor requirements, where the dilution of fuel by light impurities can have a critical impact on the maintenance of ignition conditions, and where the removal of helium "ash" through pumping apertures of limited size will require some form of high-density plasma outflow, as in a pumped limiter or divertor (Figs. 9b and c). Experimental divertor studies at PPPL have been carried out mainly on PDX. The effect of the divertor on impurity control and plasma pumping has been encouraging. The additional benefits of magnetic divertors for energy confinement, first reported in Ref. 25, have been confirmed in PDX[26] and extended to the pumped-limiter regime.[27] For future use in a reactor, the pumped limiter would offer substantial cost advantages, but would also raise serious questions about the ability of the plasma-contact surface to handle the heat load. A helpful step in the direction of smoothing out local heat deposition has recently been taken by means of a rotating pumped limiter[34] in PLT (see Fig. 11).

A central element of impurity-control studies at PPPL has been the computer-modeling effort,[35] which is aimed at optimizing divertor and pumped-limiter design in future tokamak devices. Thus far, divertor modeling studies have concentrated on ordinary thermalized edge plasmas. There is some experimental evidence, however, that the tokamak plasma is capable of sustaining anomalously large gradients along magnetic field lines. The inspiration of tandem mirror physics thus may perhaps lead to improved ideas for tokamak edge plasma control.

3.2 Radio-Frequency Techniques

As has been discussed in Sec. 1, the economic viability of a tokamak fusion reactor is expected to be sensitive to the issues of pulse length and current drive. During the past several years, the use of lower-hybrid waves in low-density plasmas[4,5] has provided a striking demonstration that noninductive current drive can be made simple and energetically efficient. In particular, the utilization of the lower-hybrid technique for the initial ramp-up of the tokamak current has some important potential advantages: Transformer volt-seconds can be saved without shortening the current plateau phase and the burn time, thus reducing the size and cost of the mechanical part of a tokamak reactor. The first wall can be made simple and solid because there is no need for "voltage gaps."

A serious potential drawback of the rf-current ramp-up method has recently been studied at PPPL: The induced back-emf tends to drive an ordinary plasma current in the opposite direction. Still worse, when the conductivity of the rf-driven hot-electron component is taken into account,[36] the back-current becomes larger and imposes a rather damaging constraint on the efficiency and time scale for rf-driven ramp-up. Fortunately, there are also simple methods to escape from the back-current problem. The ordinary plasma current can be held in check by arranging for a relatively short initial τ_{Ee} and correspondingly low electron temperature and bulk conductivity. The hot-electron back-current can be kept from developing by operating near the Dreicer runaway limit: The back-emf then simply pushes the rf-driven electrons back into the bulk distribution instead of building up a population of backwards-moving hot electrons, as would be the case for a more collisional plasma, where "hot-electron conductivity" would indeed develop.

During the evolution of these theoretical considerations, a certain amount of comfort was drawn from the PLT experimental results shown in Fig. 12. (See also Fig. 11 for a view of the 800 MHz grille.) On the basis of the hot-electron conductivity treatment, the efficiencies observed in PLT would have been anomalously high. When the more accurate runaway analysis is used, the experimental and theoretical results fit remarkably well. The normal choice of parameters for PLT current ramp-up experiments is found to be close to the theoretically recommended optimum.

The current-drive experiments can be extended to higher density by using stronger toroidal fields. Progress in this direction will also be made in PLT by means of the recently installed 2.8-GHz, 1.5-MW lower-hybrid system. The ultimate prospects for achieving true steady-state lower-hybrid current drive at economically attractive levels of tokamak reactor-power recirculation, however, are fairly marginal. The maintenance of energetic-particle currents is intrinsically incompatible with the achievement of a very high reactor Q-value, and the accessibility of lower-hybrid waves in a high-beta reactor plasma is also subject to debate. In the Starfire study,[6] these problems were bypassed by driving current only within the low-density plasma edge. This approach has been viewed with some skepticism because it seems certain to excite MHD activity. In the context of the "dynamo current-drive" method[37] advocated in conjunction with a steady-state RFP reactor, however, tokamak surface-current drive gains renewed interest. The basic idea of the tokamak dynamo cycle would be to increase the helicity slowly by rf-driving a skin current, then let the skin current relax rapidly by the precise inverse of the sawtooth mechanism described in Sec. 2.1, while conserving the helicity and increasing the poloidal flux. In this current-driven sawtooth mode, the plasma core region is replaced cyclically by a skin region with <u>higher</u> transform, but lower temperature. The net effect of dynamo current-drive on energy confinement should be no worse in the tokamak than in the RFP -- and the process is easier to understand explicitly, since only the poloidal flux needs to be regenerated.

Once the tokamak reactor plasma has been established, alpha-particle heating is by far the most desirable method of maintaining the thermal equilibrium. As discussed in Sec. 1, ignition lends itself naturally to economical multiunit operation for tokamak reactors. The best method for <u>entering</u> the ignited regime, however, remains subject to debate. As has been mentioned in Sec. 2.1, ohmic heating tends to fall short of ignition, assuming practical choices of toroidal field strength. If there is a rf-current ramp-up system in place, there will be a strong economic incentive to use it also to reach plasma ignition parameters. Among the possibilities are: (1) The rf current-drive system becomes increasingly inefficient as the density is raised, but the waves still penetrate;(i.e., bad current drive implies good plasma heating). (2) RF current-drive and/or transformer induction is used to store kinetic energy in the current carriers during the low-density phase, and this energy is then transferred to the bulk plasma as the density rises.

In the absence of a current-ramp-up method that lends itself also to the heating of dense plasmas to ignition, a special auxiliary-heating system is required. Following the success of the PLT experiments[19] of the late 1970s, the effectiveness of neutral-beam heating in this role has been considered well-established. There is now substantial confidence that the 27-MW TFTR injection system will serve to produce plasmas in the reactor parameter range. From the point of view of practical reactor operations, however, rf-heating of the plasma ions appears more promising than beam-heating -- and this would be particularly the case for multiple tokamak reactor units sharing a common plasma-heating system. In the case of the neutral-beam heating system, the cost of a set of beam lines attached to an individual reactor unit is comparable to the cost of the shared power supply. In the case of rf heating, the cost of the individual antennas is considerably smaller -- and the economies that can be achieved by multiunit operation are correspondingly large.

During recent years, PLT operations have been devoted primarily to rf heating, and particularly to heating in the ion cyclotron range of frequencies (ICRF). Ion temperature increases of order 3 keV have been achieved[38] at heating efficiencies comparable to those for neutral-beam injection. An extensive and meticulous process of ICRF antenna development has been carried out during the past five years. Most of the work has been done with half-turn loops on the large-R side of the plasma, but there are also half-turn loops on the small-R side, as well as center-fed quarter-turn loops on the large-R side, as in Fig. 11. A horizontal loop was installed recently to study electrostatic coupling effects.

As a result of the antenna-development work, voltage-handling capabilities have improved greatly. A set of six antennas now in place is expected to deliver a total of about 6 MW of 30-MHz heating power to the plasma for minority heating of ^3He ions. PLT will also have 1.5 MW of 80-MHz power for second-harmonic heating of hydrogen. The principal remaining challenge is to control the influx of hydrogen and impurities during an extended rf-heating pulse. Since the escape of rf-accelerated plasma ions appears to be an important causative factor in this process, the use of ICRF heating in conjunction with high plasma currents is particularly appropriate. In PLT, some initial success has been achieved along this line, and there are favorable prospects for ICRF heating on the larger next-generation tokamaks.

The PLT rf-facility includes about 400 kW of 60-GHz gyrotron power for electron cyclotron heating (ECH). The principal objectives of the ECH system are to assist in the initiation of purely rf-driven tokamak currents and to help optimize the radial T_e profile in auxiliary-heated tokamak discharges.

Both the ICRF and ECH systems have important potentials for current-drive[39,40] as well, which remain to be explored. In both cases, there is also the potential to store energy in low-density fast-particle populations that could help to induce ignition as the plasma is densified.

4. Future Plans

The principal goal of the next two years will be to bring TFTR up to its full operating parameters: $B_t \gtrsim 5T$, $I \gtrsim 25$ MA, and plasma temperatures above 10 keV. Even on the basis of fairly pessimistic expectations about energy confinement and MHD stability (Sec. 2.2), the achievable $n\tau_E$ values should be sufficient for a demonstration of approximate break-even conditions ($Q = P_{fusion}/P_{beam} \sim 1$). The relevant plasma-physics documentation is scheduled to be obtained in deuterium plasmas during 1986. Since the TFTR facility is designed to operate with tritium plasmas as well, an explicit $Q \sim 1$ demonstration could then be carried out. Alternatively, there are proposals for further enhancement of TFTR performance, by extending the pulse length of the neutral-beam system and introducing auxiliary rf techniques, prior to initiation of the D-T-burning phase.

The experience gained with the large tokamaks of the TFTR generation should provide a fairly accurate basis for estimating the minimum parameters of a tokamak ignition experiment, and determining the most appropriate hoice of auxiliary features, such as impurity control methods and rf systems. PPPL has been directing a national U.S. preconceptual design effort aimed at exploring the probable hardware options and costs of such a device (the Tokamak Fusion Core Experiment, TFCX). During the coming year, these studies will be refined and extended within the overall context of seeking a coordinated international plan of advance to the tokamak reactor objective.

Acknowledgment

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Figure Captions

Figure 1. Minor cross section of the TFTR, showing movable limiters and three representative plasma positions. The vertical field system can displace the plasma rapidly in major radius, thus effecting an adiabatic compression.

Figure 2. Plasma current, loop voltage, and line density for a representative ohmic-heating discharge in TFTR at $B_t = 2.7$ T, $a = 0.83$ m, $R = 2.55$ m.

Figure 3. Plasma temperatures for the TFTR discharge of Fig. 2. The electron temperatures are based on X-ray pulse height analysis (PHA) and electron cyclotron emission data (ECE). The ion (deuterium) temperature is based on neutron emission and impurity X-ray line broadening.

Figure 4. Total energy confinement times measured in the ohmic-heating regimes of TFTR and PLT appear to follow a cubic size-scaling.

Figure 5. The maximum τ_E values attainable with ohmic heating define the Neo-Alcator-Murakami limit. The ability to raise n by auxiliary heating was originally expected to produce "trans-Murakami" levels of confinement, but has been found instead to reduce τ_E at high input powers P_{aux} relative to the N-A-M limit. Some special techniques (H-mode) have largely succeeded in restoring the quality of energy confinement to ohmic-heating levels.

Figure 6. Schematic illustration of the plasma beta requirements of tokamak reactors, relative to theoretical expectations and experimental results.

Figure 7. The P<u>D</u>X has been converted to the P<u>B</u>X by a minor rearrangement of internal poloidal field coils. Bean-shaped plasmas at the 400-kA level are currently being studied.

Figure 8. MHD stability analyses[41] indicate that a moderately bean-shaped plasma (having a horizontal half-width a and an indentation d relative to the D-shape) should have high-β stability against ideal ballooning modes. To suppress kinks, a conducting shell is needed at radius a_w.

Figure 9. Schematic of plasma recycling at a simple limiter (a), pumped limiter (b), and partial divertor (c). The black arrows refer to plasma flow, the white arrows to neutral gas flow.

Figure 10. The measured effective Z in TFTR reaches satisfactorily low values at the high end of the plasma density range.

Figure 11. An interior view of the PLT vacuum vessel shows, from left to right, the 800-MHz grille, the rotating limiter, and two quarter-turn center-fed ICRF antennas. An ordinary movable graphite limiter is seen below the rotating limiter.

Figure 12. PLT plasma current is ramped up [illegible] means of 200 kW of 800-MHz LH current drive (B_T = 2.0 T, a [illegible]4 m, R = 1.32 m).

#82X0290

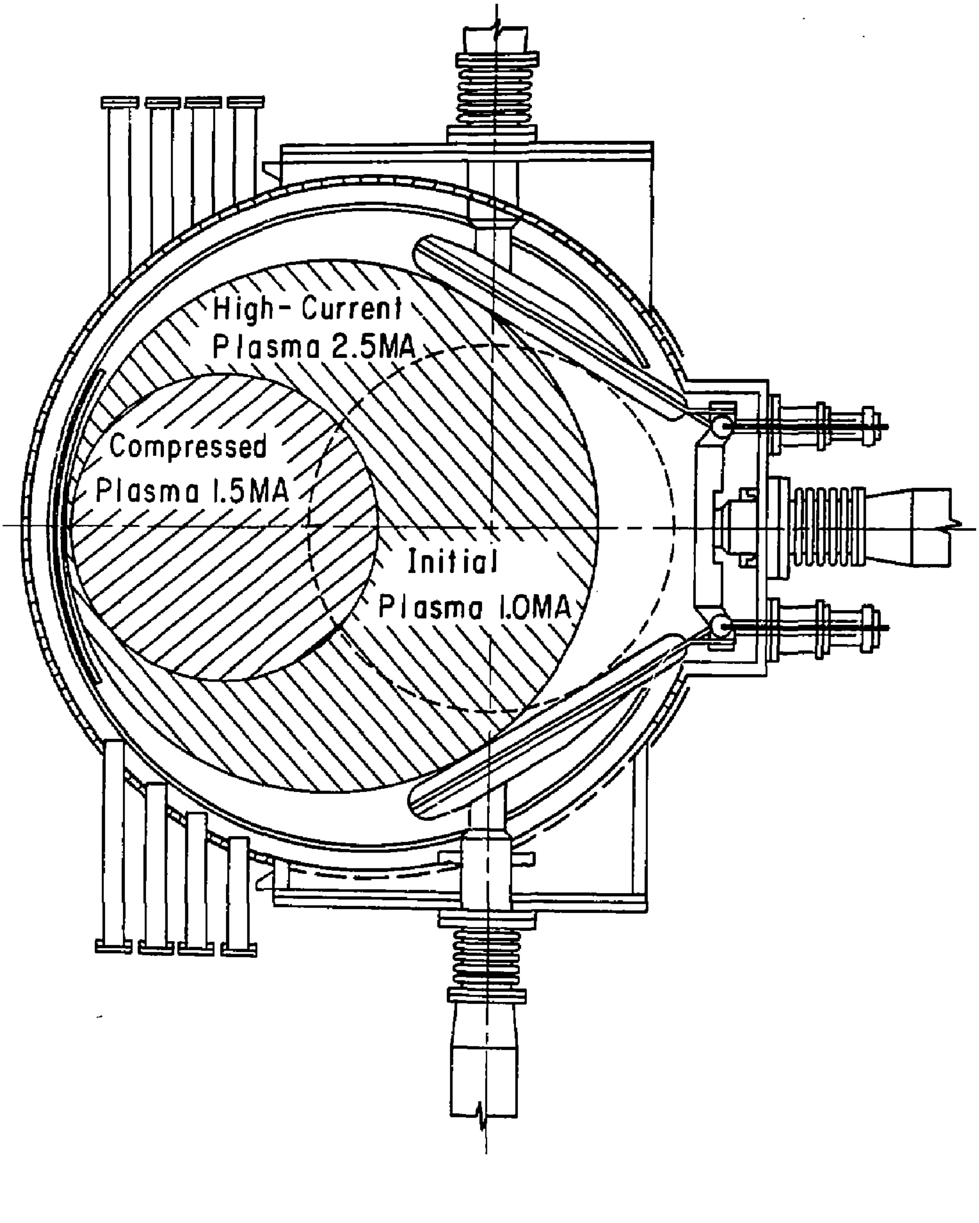


Figure 1

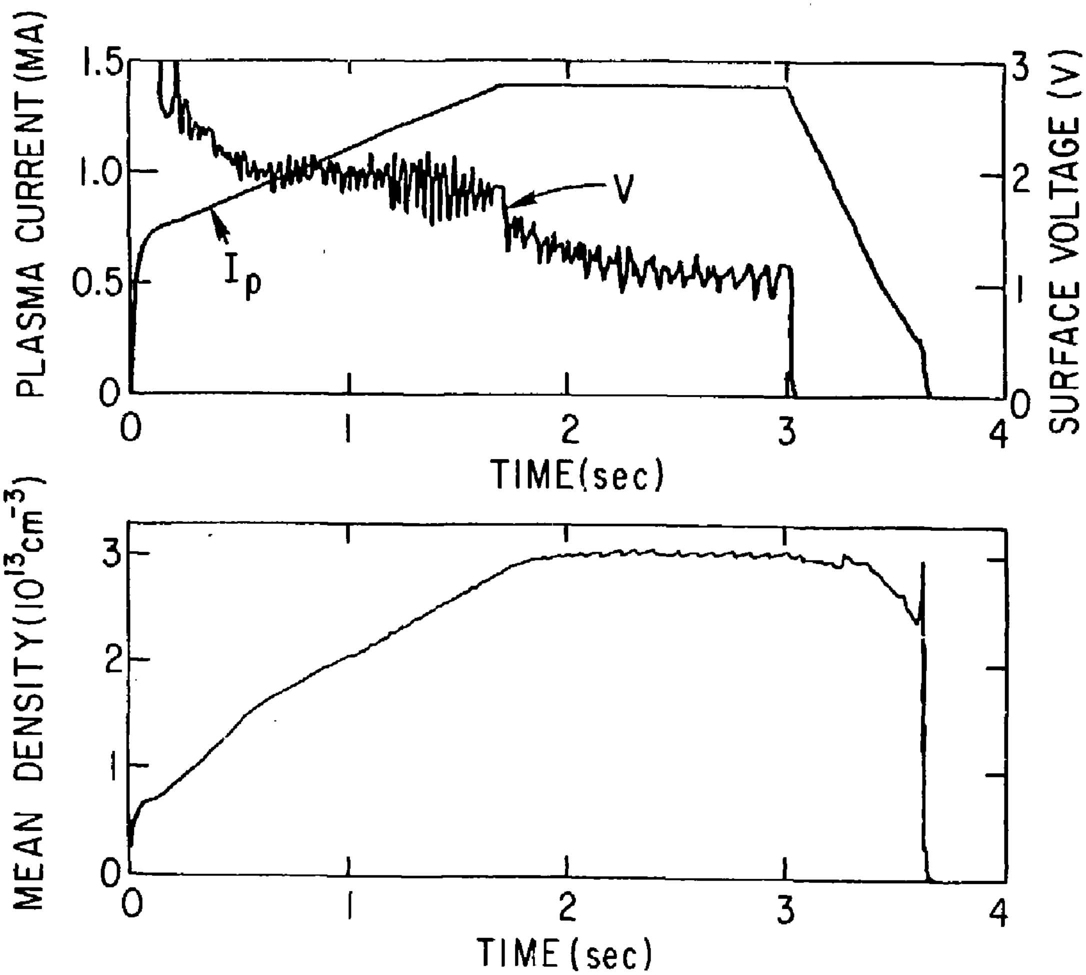

PLASMA CURRENT (MA)
1.5
1.0
0.5
0
I_p
V
SURFACE VOLTAGE (V)
3
2
1
0
1
2
3
4
TIME(sec)
MEAN DENSITY($10^{13}cm^{-3}$)
3
2
1
0
0
1
2
3
4
TIME (sec)

Figure 2

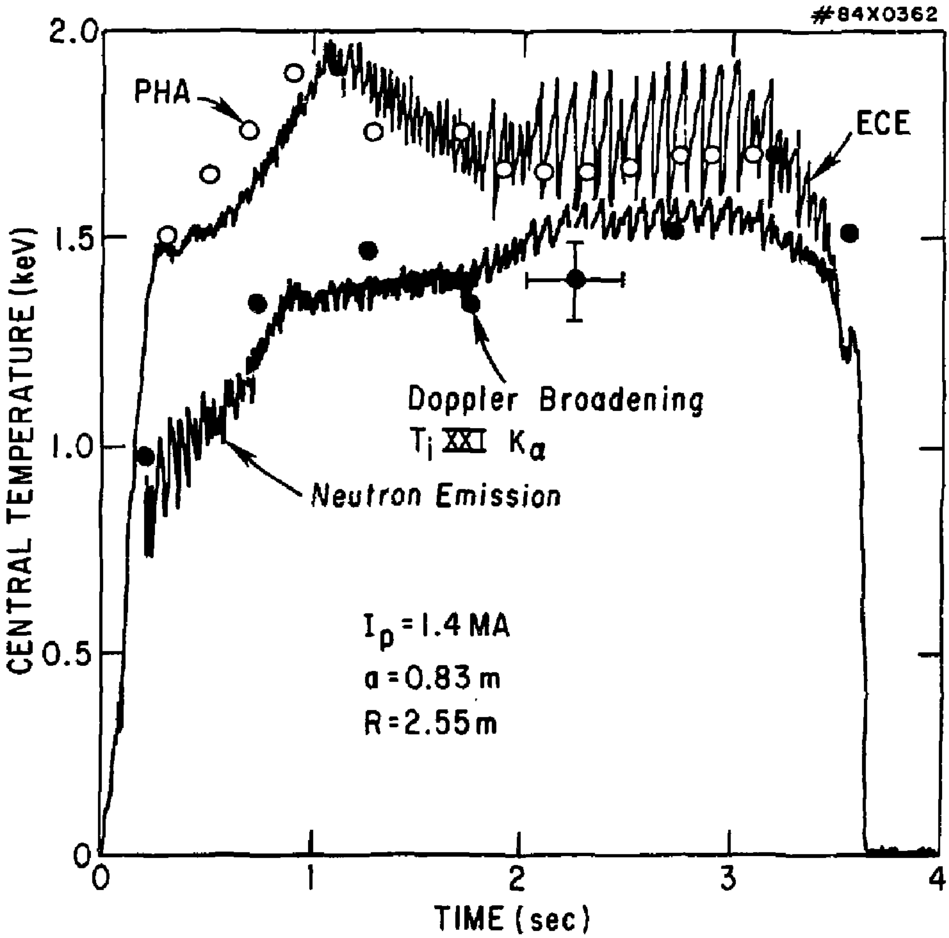

#84X0362
CENTRAL TEMPERATURE (keV)
2.0
1.5
1.0
0.5
0
PHA
ECE
Doppler Broadening
Ti XXI Kα
Neutron Emission
Ip = 1.4 MA
a = 0.83 m
R = 2.55 m
0
1
2
3
4
TIME (sec)

Figure 3

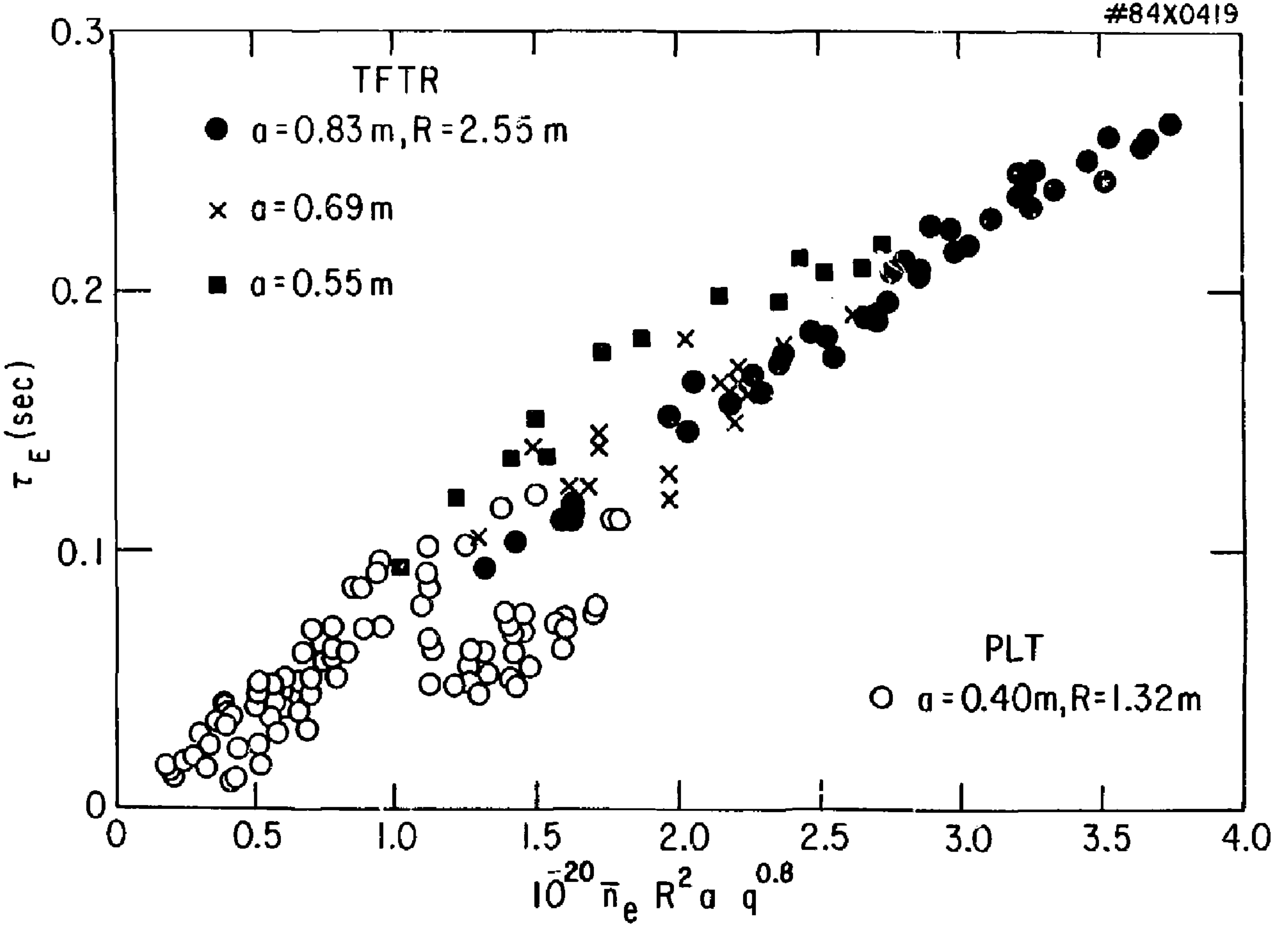

#84X0419
TFTR
a = 0.83 m, R = 2.55 m
a = 0.69 m
a = 0.55 m
PLT
a = 0.40m, R = 1.32m
τ_E (sec)
$10^{-20}\bar{n}_e R^2 a q^{0.8}$
0.3
0.2
0.1
0
0.5
1.0
1.5
2.0
2.5
3.0
3.5
4.0

Figure 4

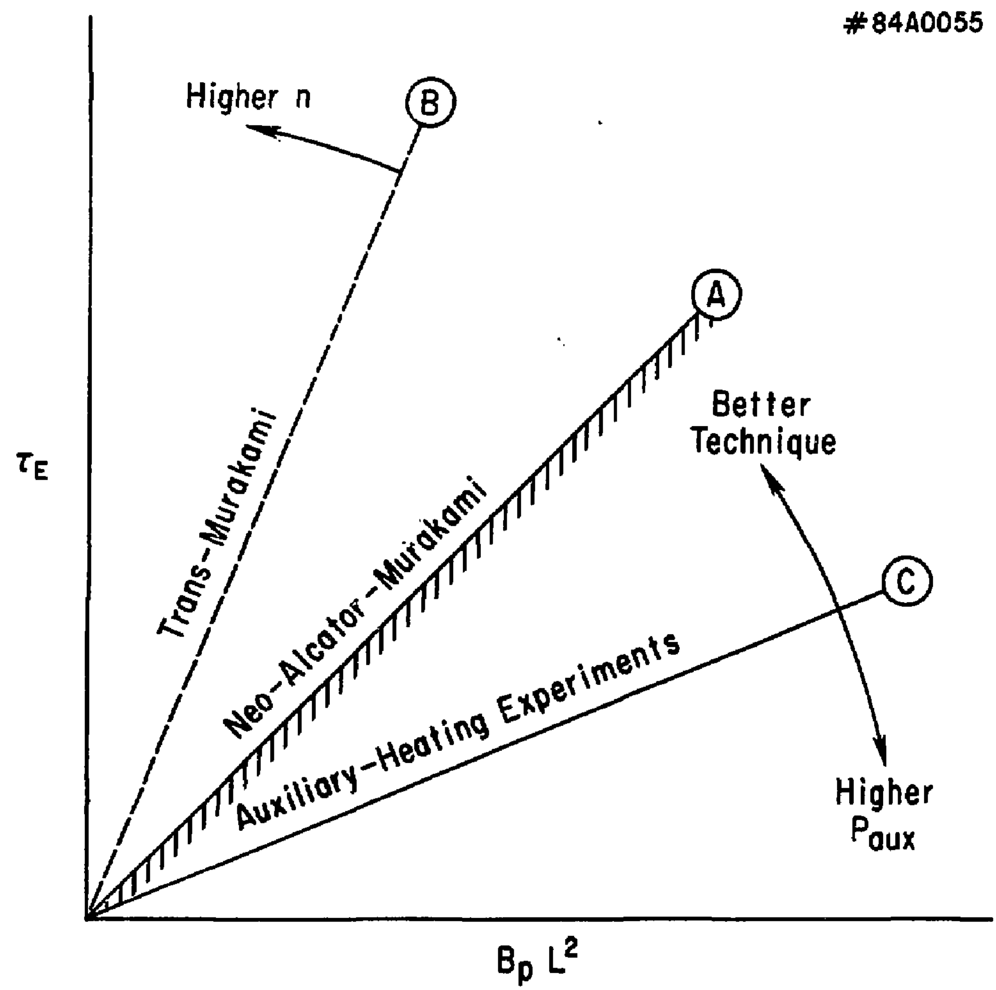

#84A0055
Higher n
B
A
C
Trans-Murakami
Neo-Alcator-Murakami
Auxiliary-Heating Experiments
Better
Technique
Higher
P_{aux}
τ_E
$B_p L^2$

Figure 5

#84X0990

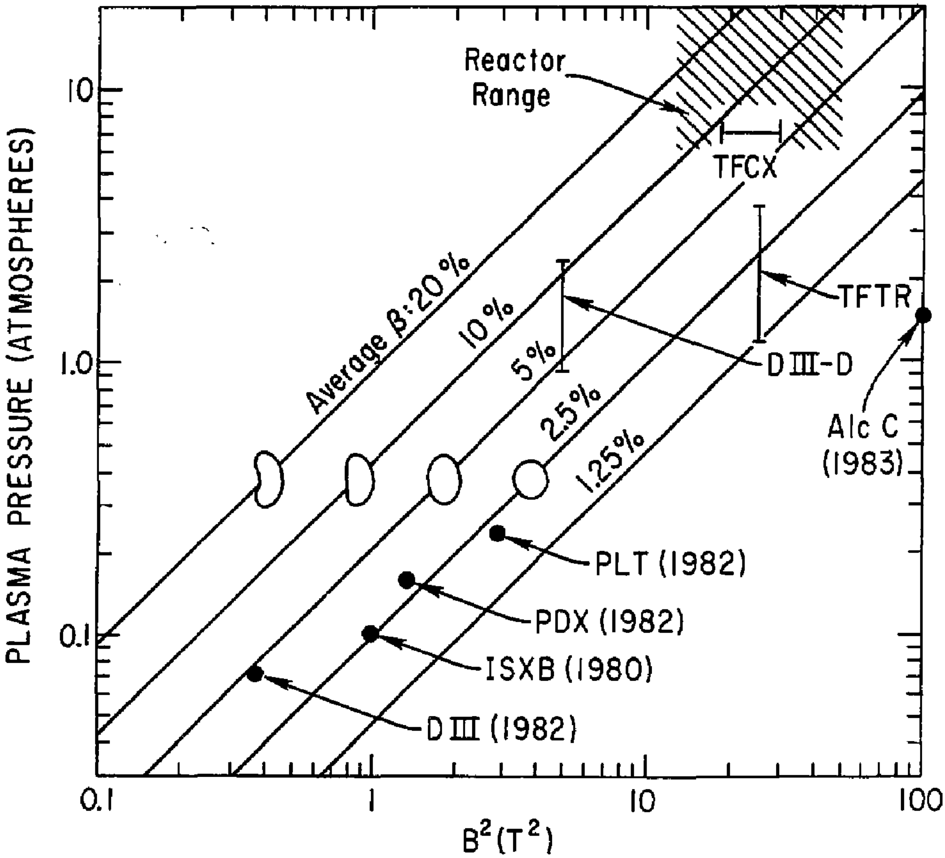


Figure 6

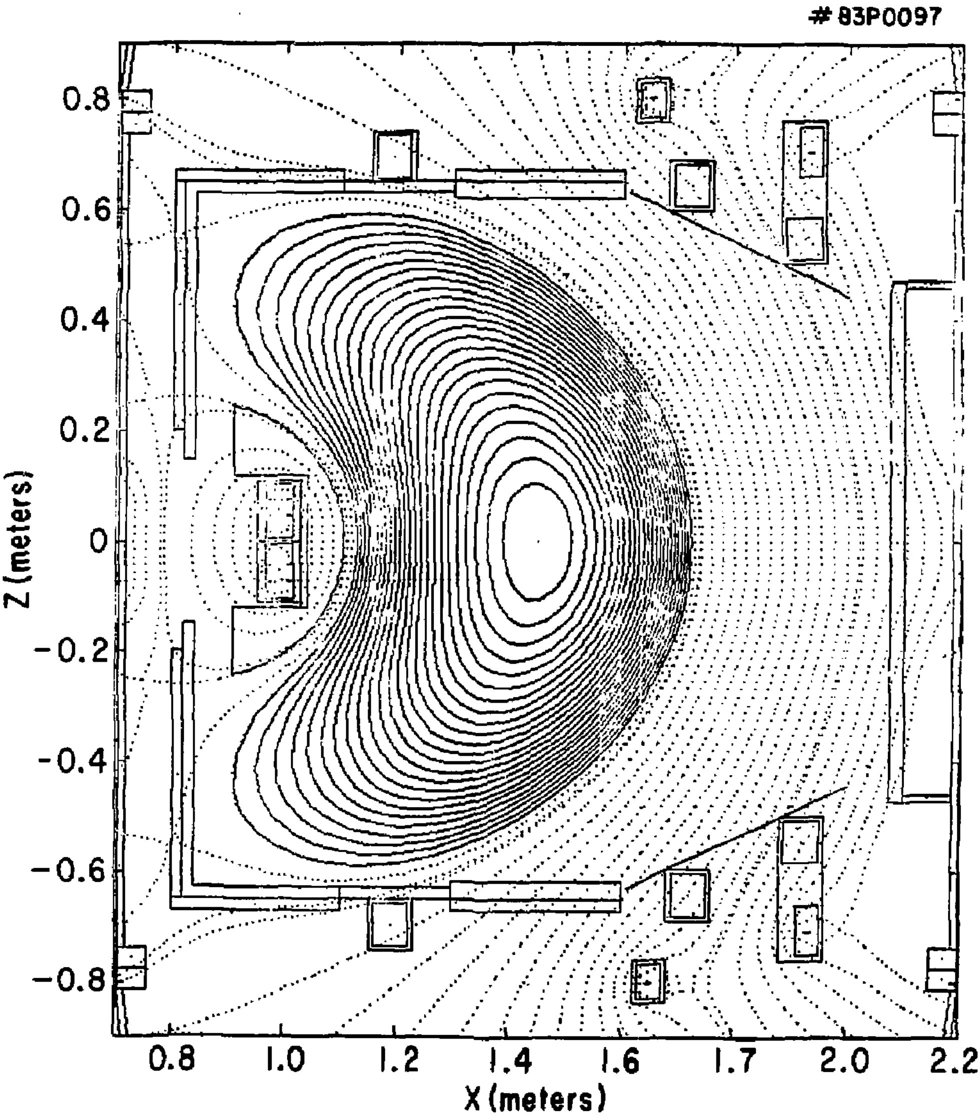
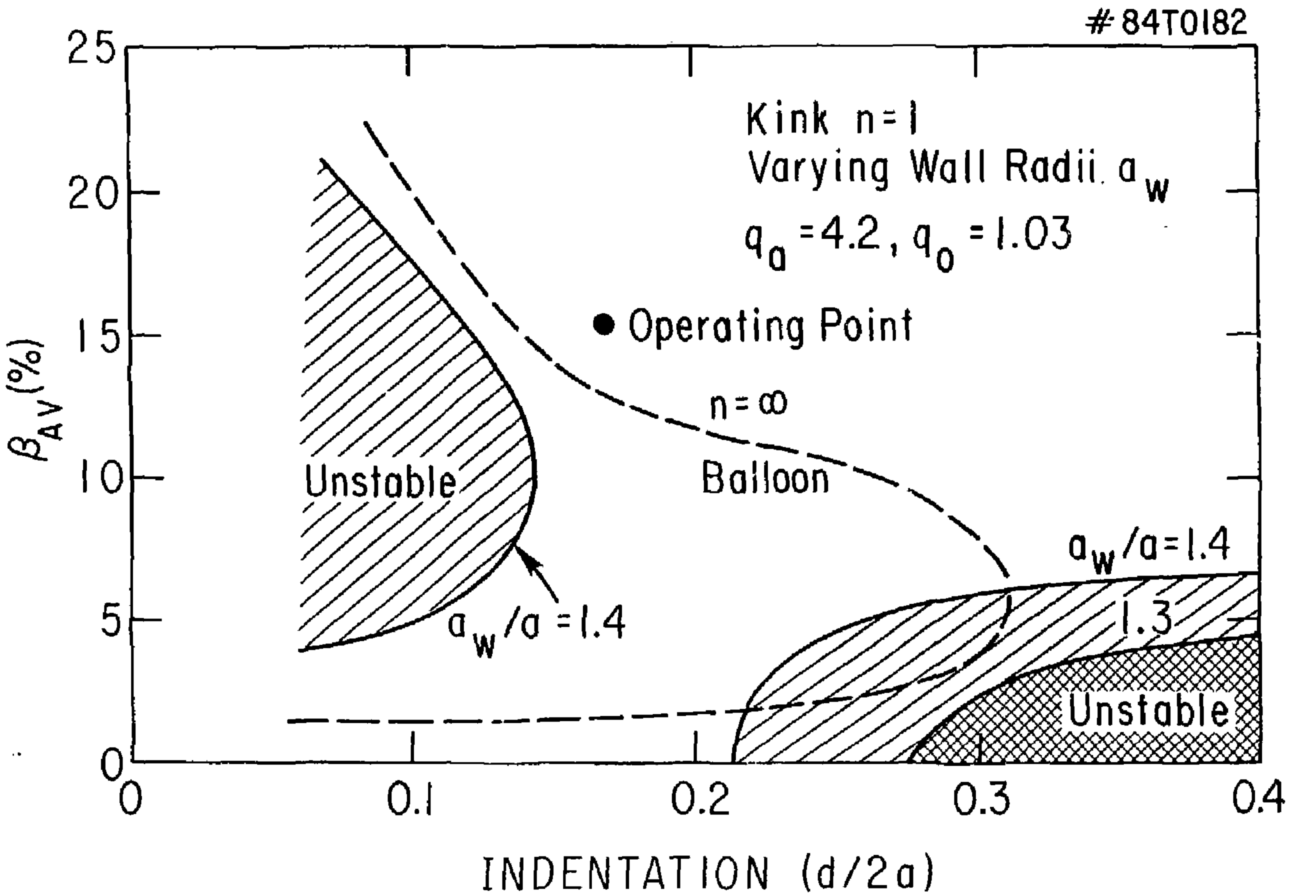

#83P0097
0.8
0.6
0.4
0.2
0
-0.2
-0.4
-0.6
-0.8
Z (meters)
0.8
1.0
1.2
1.4
1.6
1.7
2.0
2.2
X (meters)

Figure 7



#84T0182
Kink n=1
Varying Wall Radii a_w
$q_a = 4.2, q_0 = 1.03$
Operating Point
n=∞
Balloon
Unstable
$a_w/a = 1.4$
$a_w/a = 1.4$
1.3
Unstable
β_{AV}(%)
INDENTATION (d/2a)
0
5
10
15
20
25
0
0.1
0.2
0.3
0.4

Figure 8

#83X0502

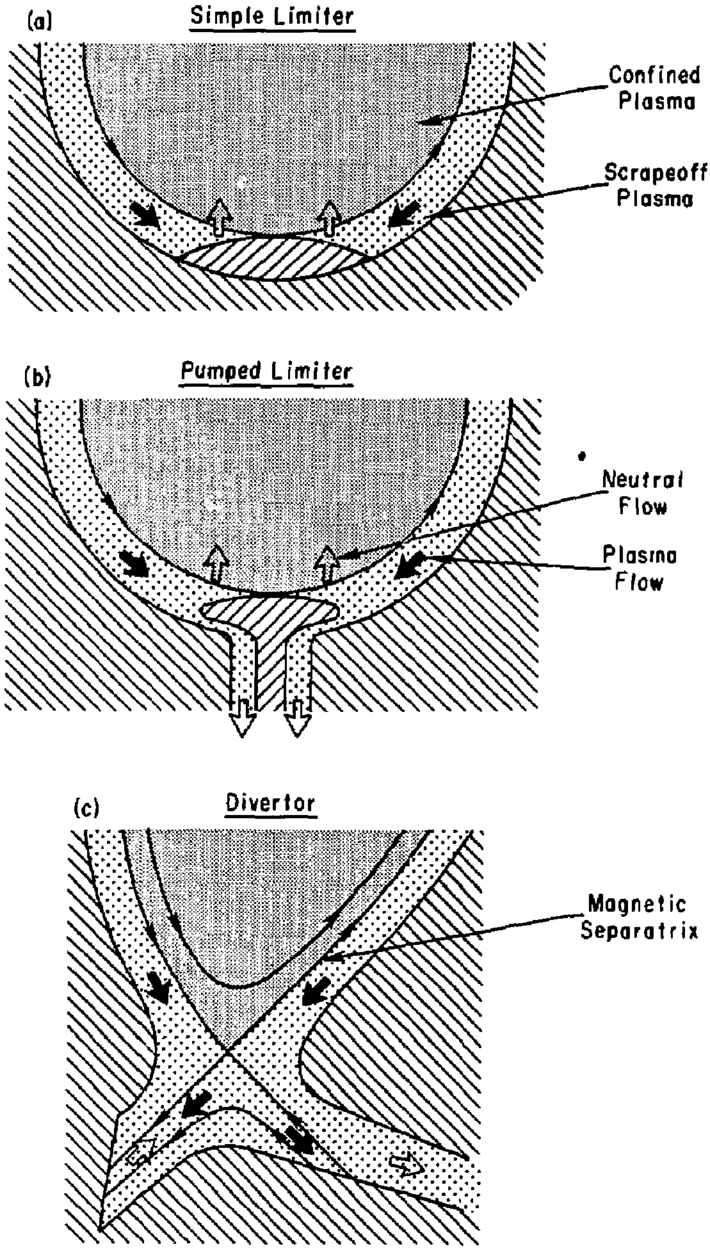


Figure 9

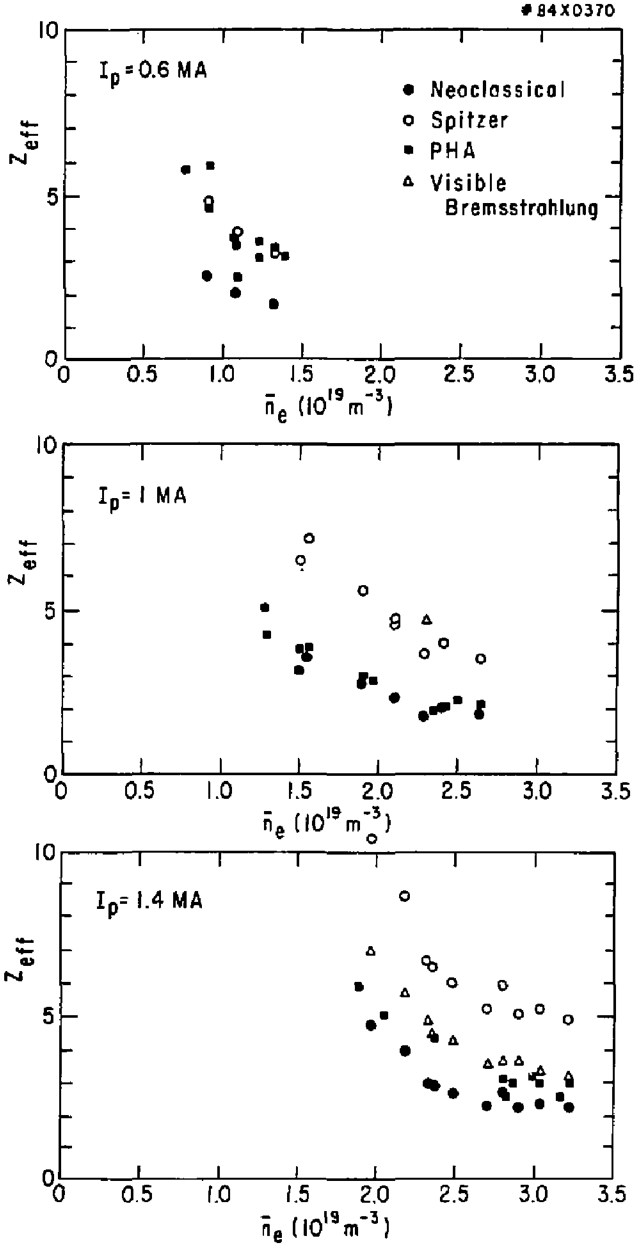

#84X0370
I_p = 0.6 MA
● Neoclassical
○ Spitzer
■ PHA
△ Visible Bremsstrahlung
Z_{eff}
$\bar{n}_e$ (10^{19} m^{-3})
I_p = 1 MA
I_p = 1.4 MA

Figure 10

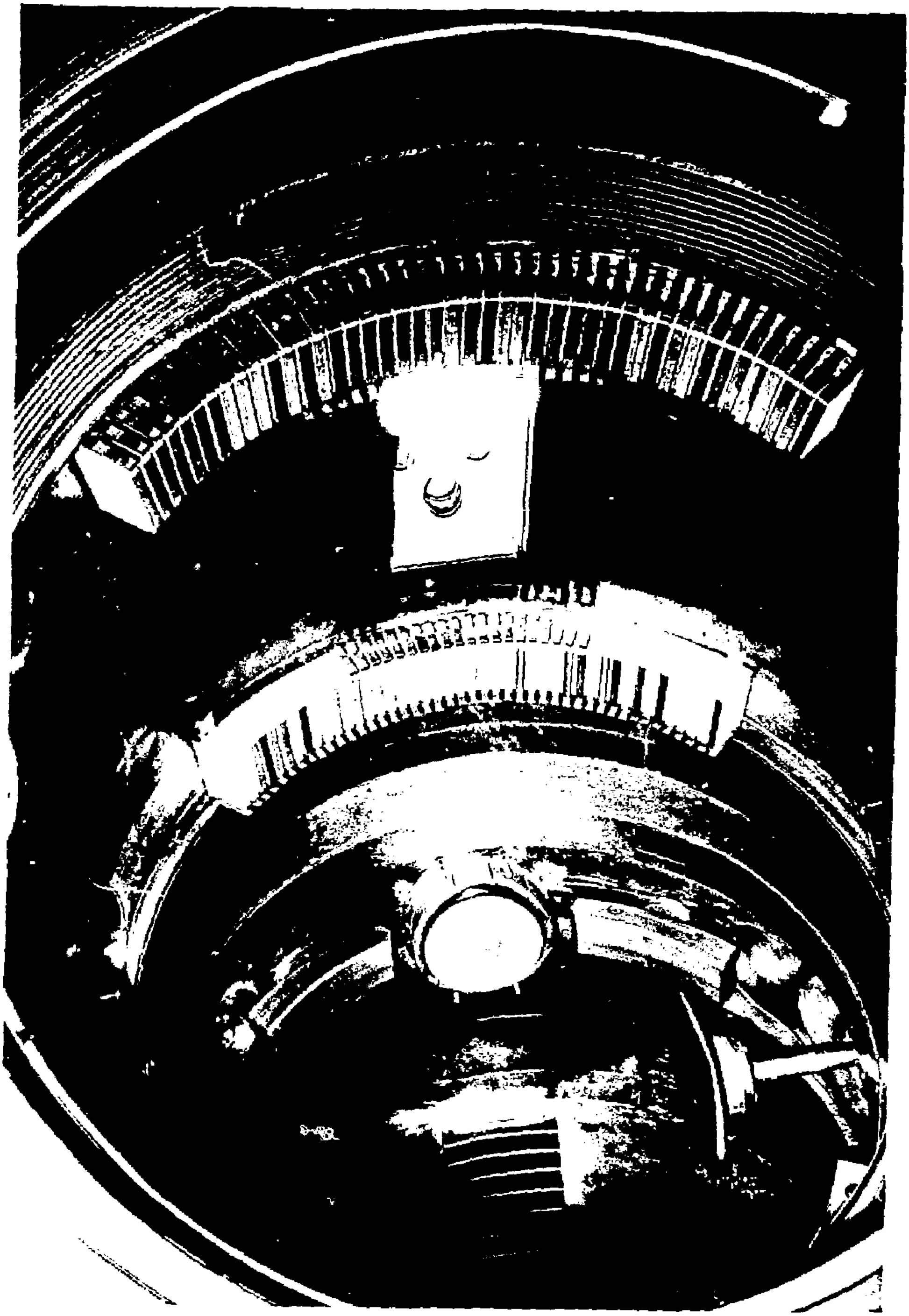


Figure 11

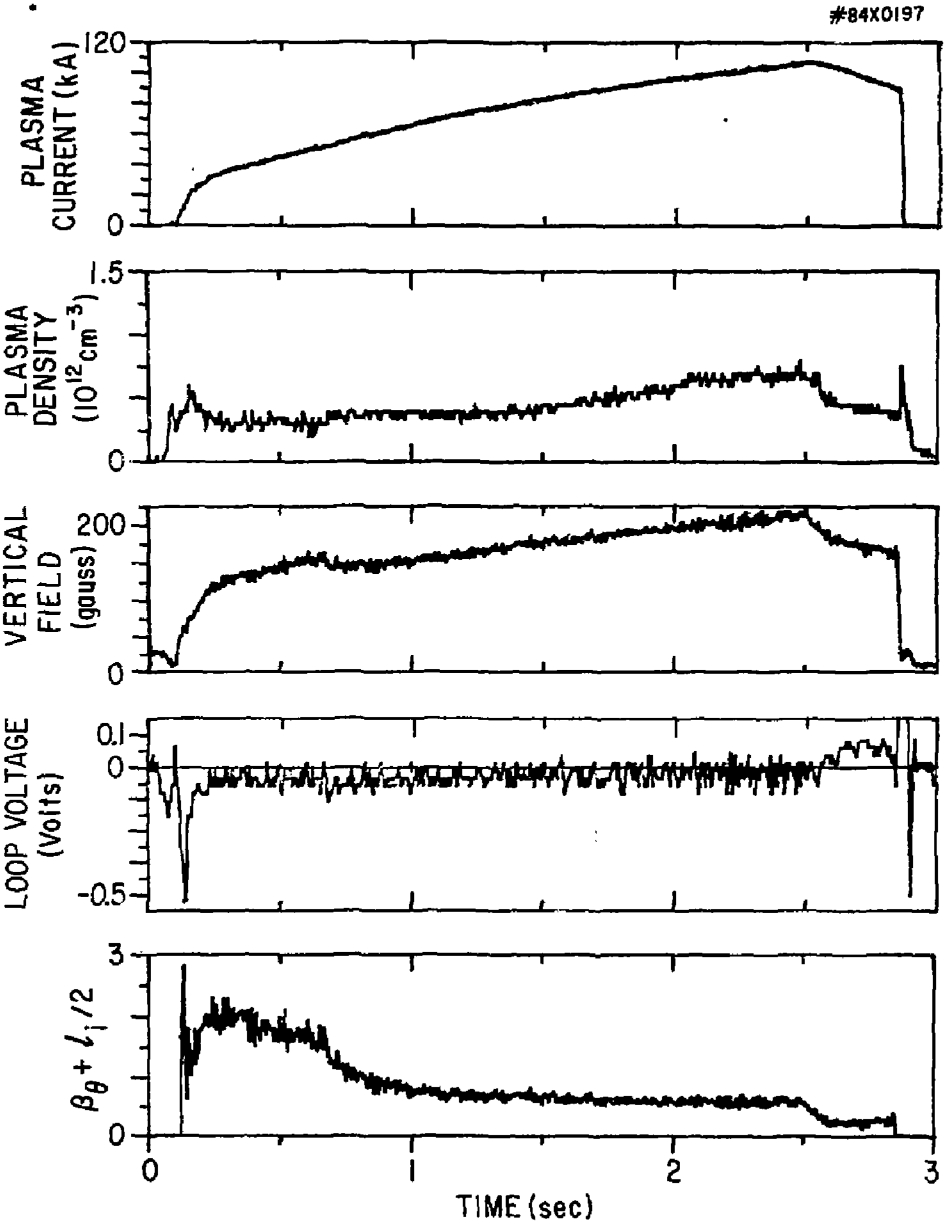

#84X0197
PLASMA CURRENT (kA)
120
0
PLASMA DENSITY ($10^{12}cm^{-3}$)
1.5
0
VERTICAL FIELD (gauss)
200
0
LOOP VOLTAGE (Volts)
0.1
0
-0.5
$\beta_\theta + \ell_i/2$
3
0
0
1
2
3
TIME (sec)

Figure 12

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