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ENERGY CONFINEMENT SCALING IN TOKAMAKS: SOME IMPLICATIONS OF RECENT EXPERIMENTS WITH OHMIC AND STRONG AUXILIARY HEATING

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ABSTRACT

Recent results from confinement scaling experiments on tokamaks with ohmic and strong auxiliary heating are reviewed. An attempt is made to draw these results together into a "low-density" ohmic confinement scaling law, and a scaling law for confinement with auxiliary heating. The auxiliary heating confinement law may also serve to explain the saturation in τ_E vs. $\bar{n}_e$ observed in some ohmic heating density scaling experiments.

I. INTRODUCTION

Scaling studies in tokamaks suffer from a host of serious difficulties. Early experiments were plagued by uncontrolled (and sometimes unmeasured) variations in Z_{eff} and radiated power. MHD activity, Z_{eff}, and P_{rad} even now may change uncontrollably with or without the variation of external parameters. Wall interaction is almost unavoidably altered in any single-machine size scaling study, while machine-specific effects (wall-conditioning techniques, limiter geometry, field symmetry) are necessarily involved in any size scaling study done by comparing results from different machines. Variations in input power or gas feed rate necessarily change wall and limiter interactions, in addition to producing the desired changes in plasma temperature and density.

In light of these difficulties, one might abandon the enterprise of scaling experiments entirely, except that it offers the possibility of a great deal of insight which cannot be obtained by any other means. Besides giving an overview of the "terrain," and some basis for extrapolation to future devices (both valuable goals in themselves), confinement scaling studies have the potential of giving critical information for understanding the underlying nature of radial transport. Indeed it can be argued that studying one plasma condition with ever increasing accuracy and with ever more detailed diagnostic techniques is as restricted in ultimate value as scanning widely over parameter space with very simple diagnostics.

In a properly conducted scaling experiment attention should be paid to measuring, and hopefully controlling, variations in MHD activity, Z_{eff}, and radiated power. To minimize uncertainties, size scaling experiments should be performed both within individual machines, and by comparing results from different devices. Experiments using different wall-conditioning techniques, and limiter or divertor geometries must be compared with caution. Finally, scaling studies must be designed to take into account the fact that the relevant plasma physics may not be well represented by

$$\tau_E \propto \Pi\ a^x b^y c^z \ldots \qquad (1)$$

where a,b,c... are plasma parameters, and x,y,z... are simple numerical exponents. As a result 2- and even 3-dimensional parameter scans may be required to understand the general shape of the "terrain" in the parameter range available.

In this paper we will review the results from recent scaling studies with ohmic and strong auxiliary heating which have met most of the criteria outlined above. In drawing conclusions from this new data base we must ourselves be wary of the possible pitfalls in comparing results from different machines. While the conclusions we will draw are of necessity tentative, and open to change with new experimental input, a certain consistency nonetheless emerges in the pattern. The fact that a consistent picture can begin to be drawn from such a wealth of data is a tribute to the care and hard work of all the experimental teams involved.

II. "LOW DENSITY" OHMIC REGIME

A. INTOR scaling

The attainment of very pure plasmas in the high-field Alcator A tokamak in 1976-77 permitted scaling experiments over a wider range in plasma density than had been previously explored (Gaudreau and co-workers, 1977). Gross energy confinement time was found to rise nearly linearly with plasma density, giving record values of $n_e(0)\tau_E$ of 2.0×10^{13} cm^{-3}sec. These results opened up the field of confinement scaling studies in ohmically heated tokamaks. Experiments of this sort were repeated in the TFR and Pulsator tokamaks (Equipe TFR, 1978a; Kluber and Murmann, 1982) again showing $\tau_E \propto n_e$. Experiments on the low field ISX-B tokamak confirmed the Alcator A results (Murakami and co-workers, 1979), and even showed that gross plasma confinement saturated at high density, consistent with 1-2 times the losses expected from neoclassical ion thermal transport (Hinton and Hazeltine, 1976). [This saturation has since proved to be nicely consistent with more accurate neoclassical calculations (Bolton and Ware, 1981; Chang and Hinton, 1982), which typically give ion thermal conductivities ~ 1.7 times the Hinton-Hazeltine result.]

Jassby, Cohn, and Parker (1976) compiled results from a number of experimental devices, and deduced an overall scaling law for ohmically heated tokamaks:

$$\tau_E \propto n_e a^2 q^{1/2} \qquad (2)$$

This result, in somewhat modified form, [$\tau_E(\mathrm{sec}) = 5 \times 10^{-19} n_e(\mathrm{cm}^{-3})\ a^2(\mathrm{cm})$] was taken over for the INTOR studies as a "benchmark" for confinement scaling. There were, however, some serious difficulties with this simple scaling law even at that time. Figure 1 shows τ_{Ee} ($\equiv W_e/P_{OH}$, the electron stored energy divided by the ohmic input power) for published results from the ST tokamak. There is a significant variation of τ_{Ee} with Z_{eff}, as shown here, which is ignored in the INTOR law for τ_E. This was also observed on PLT

(Bol and co-workers, 1978). More importantly, however, ST gave confinement times dramatically highe than those indicated by INTOR scaling. A further difficulty with the INTOR scaling law is that it does not obey Connor-Taylor (1977) constraints. This implies that it cannot arise solely from the usual equations of plasma physics, unless the Debye length plays an important role. Except in discharges with significant runaway content, this seems unlikely (Kadomtsev, 1975; Connor and Taylor, 1977).

B. Neo-Alcator scaling

In 1979 Pfeiffer and Waltz (1979) published the results of multiple regression fits to a large confinement data base, including the then recent Alcator A results. Expressed in terms of externally controllable parameters, they found $\tau_{Ee} \alpha n_e^{0.90} a^{0.98} R^{1.63} Z_{eff}^{0.23}$. Connor-Taylor constraints could be imposed, without substantially reducing the goodness of fit, giving $\tau_{Ee} \alpha n_e^{0.90} a^{1.14} R^{1.91} Z_{eff}^{0.14}$. This result is optimistic in giving a stronger than length^2 confinement scaling. The fit showed that τ_E depended more strongly on R than on a (consistent with the ST results). At the same time, because R had been varied less than a in the data set available, they found a stronger overall size scaling than Jassby, Cohn, and Parker (1976). The weak Z_{eff} scaling may reflect the mixture of discharges contaminated by high- and low-Z impurities in the data set. At the 1980 IAEA conference Leonov and co-workers (1980) showed size scaling results from T-11 (Fig. 2) indicating very little variation of τ_E with minor radius. In these experiments the electron density was kept low and constant at $1.5 \times 10^{13} cm^{-3}$ in order to minimize ion effects. The total radiated power and Z_{eff} were measured to be low. These high-quality results certainly raised some serious doubts about the a^2 component of INTOR scaling.

In 1981 the General Atomic Doublet-III team performed a set of plasma current, density, and minor radius confinement scaling studies with ohmically heated plasmas (Ejima and co-workers, 1982). These experiments were a major step forward in that they constituted an attempt at a systematic scan over three independent parameters (Fig. 3). Perhaps not surprisingly, such an ambitious undertaking ran into difficulties maintaining Z_{eff} and P_{rad} low over the entire parameter range. In addition, sawtooth MHD activity was not observed in the lower density regions of the lower current scans. Discharges without sawteeth tended to have higher central radiated power than those with sawteeth. In the context of the experiments mentioned above, however, and those performed subsequently on the Alcator C and FT tokamaks, the D-III data set is extremely valuable. The D-III group divided their results into a low density INTOR-scaling regime, and a high density saturated regime. Here we discuss only the low density data; we will return to the high density results in Sec. IV. The low density data showed confinement to scale roughly linearly with plasma density, and about as $q^{3/4}$. This is a somewhat stronger q scaling than observed in most other machines, and may reflect the measured increase in central radiated power as a function of rising plasma current.

The more critical issue in question, however, is the minor radius scaling of low density ohmic confinement indicated by these results. The GA group found that τ^G_{Ee} (= $\tau^1{}_{Ee}$ from ST) scaled as a^2 in going from the 23 cm data to the 32 cm data. The 44 cm data, however, fell below this scaling by a factor of 1.8. This was interpreted as being due to the different wall interaction of the large plasma, which made contact with an additional set of inner wall limiters. In light of the comments in Sec. I, this point of view must be taken seriously. On the other hand, looking at the data from other machines with carefully controlled Z_{eff} and P_{rad}, we find at most weak scaling with minor radius. The Pfeifer and Waltz results (1979) show this as well. Thus the correct interpretation may indeed be to simply take the three data sets as a single group. A line of $\tau_{Ee}^{G} \sim a$ can be fit through the error bars of the three low density ohmic data sets with no difficulty.

The Alcator C tokamak was designed to exploit the a^2 aspect of INTOR scaling, and high field magnet technology, in order to obtain $n_e(0)\tau_E$ in the range of 10^{14} cm^{-3}sec. At densities up to about $2 \times 10^{14} cm^{-3}$, τ_E scaled in agreement with the INTOR law. At higher densities, however, τ_E began to saturate (Fairfax and co-workers, 1980). Here again we will concentrate on the "low density" regime, and leave the saturated high density regime to Sec. IV. In order to investigate the a and R scaling of confinement in both regimes, the Alcator group constructed a set of full circle poloidally local limiters, which could be interchanged to make plasmas of different sizes (Blackwell and co-workers, 1982). At R_0 = 64 cm, a = 10, 13, and 16.5 cm were tried. At a = 10 cm, R_0 = 57.7, 64 and 70.5 cm were tried. Central radiated power and Z_{eff} were both kept low in these scans, and only sawtoothing discharges were included in the data set. Figure 4 shows the results from a = 10 cm, R_0 = 57.7 and 70.5 cm. The overall results showed $\tau_E \sim \bar{n}_e R^2 a$ in the "low density" regime. This scaling is nicely consistent with the higher $\tau_E/\bar{n}_e$ obtained in the similar high field FT tokamak (R_0 = 83 cm, a = 20cm) (Alladio and co-workers, 1982). Taking a survey over modern low Z_{eff}, low P_{rad}, moderate q results from other machines, the Alcator group found that their scaling fit the results from other machines quite well (Fig. 5). The recent preliminary results from TFTR (Young and co-workers, 1983) fit on the curve as well. Thus both inter- and intra-machine scaling experiments point to the strong R and relatively weak a scaling result shown.

Toroidal field and plasma current scaling have been explicitly excluded from the plot in Fig. 5. However the fact that machines with widely different toroidal fields and plasma currents fall onto the same scaling law suggests that τ_E will only depend on q, rather than on B_T or I_p separately. This is indeed the result observed tentatively on D-III (Ejima and co-workers, 1982). Scaling results on DIVA (DIVA group, 1980) showed $\tau_E \sim q^{1/2}$. A compilation of TFR data shows $q^{1/2}$ scaling as well (Equipe TFR, 1980a). The modest ohmic study performed on PDX in conjunction with the beam heating scaling experiments shows τ_{Ee}/n_e at $a/2 \underset{\sim}{\propto} q^{1/2}$, for B_T = 10 to 22 kG, and I_p = *200 to 500 kA (Grek, Johnson and Kaye, 1983). q-scaling results from FT* reported at this conference (Alladio and co-workers, 1983), combined with previous results (Pieroni and co-workers, 1980) seem to confirm the idea that τ_E depends on toroidal field and plasma current about as $q^{1/3}$ to $q^{2/3}$. Experiments by the Japan Atomic Energy Research Institute (JAERI) team on D-III (Nagami and co-workers, 1982) indicated that $\tau^G_{Ee}/\bar{n}_e$ in the low density ohmic regime scaled proportional to $q^{0.84}$. Furthermore, $\tau^G_{Ee}/\bar{n}_e$ is enhanced in low density elongated discharges roughly by the increase in q associated with operating at fixed plasma current and horizontal minor radius, but with finite elongation.

There is evidently some modest disagreement on q-scaling between the data from different machines. This is perhaps not surprising, since input power depends strongly on plasma current, so wall interaction and plasma purity can be expected to vary significantly with I_p. The differences, however, are not great, and the scaling exponent does not clearly depend on another parameter, such as a or B_T. Thus, taking q = 3.5 as a typical "moderate" q for the interpretation of Fig. 5, we average over the different q-scaling results, and come to a rather simple estimated scaling law for low Z_{eff}, low P_{rad}, sawtoothing discharges:

$$\tau_E^{OH}(\text{sec}) = 7.1 \times 10^{-22}\ \bar{n}_e^{w}\left(\text{cm}^{-3}\right)\ a^{x}\left(\text{cm}^{-3}\right)\ R^{y}\left(\text{cm}^{-3}\right)\ q^{z} \qquad (3)$$

where $w=1$, $x=1.04$, $y=2.04$, and $z=0.5$. Connor-Taylor constraints for the collisional or collisionless high-β models, or for the collisional low-β model require $8w + 5-4(x + y) = 0$, which is certainly satisfied within reasonable error bars on w,x, and y. The constraints associated with ideal or resistive MHD models, however, $4(x + y) + 2w = 5$, and $w = 1/2$, are not satisfied. It is interesting to speculate on the physical processes which could give rise to a scaling result like Eq. (3). Clearly toroidicity plays a major role, perhaps through instabilities driven by particle drifts. It seems difficult to support the idea that ξ $(\equiv v_d/v_{th,e})$ is a driving factor, since it is not clear how anisotropy-driven modes would interact with toroidicity. Furthermore, ξ is peaked on axis, while detailed transport analyses, and the weak a scaling of Eq. (4) both suggest that the outer regions of the plasma have the stongest transport. Recent stellarator results call into question the scaling of τ_E with ξ (Atkinson and co-workers, 1980) and see little variation of τ_E with external rotational transform (Bartlett and co-workers, 1980; Atkinson and co-workers, 1980). Thus the possibility that transport is driven by $v_d/v_{th,e}$, but ameliorated by rotational transform does not seem to be supported by recent stellarator data. Indeed the similarity of the geometry of Cleo to that of ST, and the similarity of ohmic confinement results in the two machines suggests that external transform does not play a central role in gross confinement in ohmically heated stellarators. In general the $q^{1/2}$ scaling in Eq. (3) seems naturally to be tied more to macroscopic MHD activity than to microscopic processes.

C. Profile effects

The INTOR studies proposed not only a gross confinement scaling law, but also a "standard" electron thermal diffusivity for use in modeling studies.

$$\chi_e(\mathrm{cm^2/sec}) = 5 \times 10^{17}/n_e(\mathrm{cm^{-3}}) \qquad (4)$$

This form for χ_e was based on informal studies of data from Alcator A (Post, 1977; Gaudreau and co-workers, 1977) which gave a numerical coefficient on χ_e in the range of 5-7 × 10^{17}. Because other machines on the average found somewhat higher values of $\tau_E/(\bar{n}_e\ a^2)$ than Alcator A, the lower coefficient of 5 × 10^{17} was chosen. It is interesting to try out an alternate form for the low-density ohmic χ_e, based on the results described here. Scaling again from Alcator A, we might choose

$$\chi_e(\mathrm{cm^2/sec}) = 7 \times 10^{17}[56/R(\mathrm{cm})]^2\ (r(\mathrm{cm})/5)/n_e(\mathrm{cm^{-3}})\ . \qquad (5)$$

(Note that for Alcator A, R = 56 cm, a=10 cm). We have clearly obtained R^2 scaling for τ_E in a very simple way. Since we want τ_E to scale linearly with a, rather than as a^2, we need a means to increase transport in the outer regions of the plasma. Equation (5) is undoubtedly the simplest way to enforce linear a scaling. The coefficient on *r* is taken as 1/5 cm, because a

= 10 cm in Alcator A, and we do not wish to alter the average χ_e for Alcator A. Of course in this model χ_e goes to zero at the magnetic axis, so we need to impose a sawtooth model as well. In the simplest such model one strongly enhances χ_e in the region of $q < 1$, so as to maintain T_e essentially flat over that area.

The simple transport model of Eq. (5) has some appealing features. When combined with a Z = 1 Spitzer resistivity ohmic heating equilibrium model (neglecting ion effects and radiation), it gives better agreement with typical measured profiles than the INTOR model. The INTOR model achieves q = 1 on axis only at limiter q's below 3.3. At higher q's it gives a fixed electron temperature profile shape, about of the form $T_e \sim [1-(r/a)^2]^2$. The "neo-Alcator" model proposed here shrinks to q = 1 on axis at all values of the limiter q. Figure 6 compares $T_e(r)$ profiles for q = 4 in PLT as predicted by INTOR vs. the present model. The "neo-Alcator" profile is more realistic when compared to PLT data (Bol and co-workers, 1978). The scaling of $r(q = 1)/a$ vs. q(a) for the "neo-Alcator" model is shown in Fig. 7. Studies of this parameter by the DIVA group (1980) and by the JAERI D-III team (Nagami and co-workers, 1982) show $r(q=1)/a \approx 1/q(a)$ in the range of q=2 to 5.

It is interesting to note the scaling of τ_{Ee} with q(a) in Fig. 7. In this log-log plot the data form a curve, but in the most studied range of q = 2 - 4, τ_E varies about as $q^{1/2}$. In the INTOR model τ_E falls by only 13% in going from $q = \infty$ to q = 2. The difference in these results arises from the fact that the central sawtooth region is more valuable territory in the "neo-Alcator" model than in the INTOR model. Thus sawteeth have a greater effect on confinement. This simple calculated result must be taken with a certain caution, however. Detailed transport analyses on FT (Alladio and co-workers, 1982) and by the GA D-III group (Ejima and co-workers, 1982) have shown significant variation of χ_e with plasma current. One has to be careful in these studies, of course, to evaluate χ_e in the region outside the outermost reconnection point [~ 1.4r(q = 1)], and to use a time-averaged electron temperature gradient. The sawtooth modifies the time-averaged ohmic power input profile as well. Another option would be to model the sawtooth transport directly, in order to subtract it from the total transport losses using the method of Pfeiffer (1983). By the same token a more sophisticated sawtooth model should be included in the "neo-Alcator" confinement prescription, and radiation, ion effects, and possibly neoclassical resistivity should be included as well, before definitive conclusions are drawn. For example, a BALDUR transport simulation by Mikkelsen (1983) for present TFTR parameters (Young and co-workers, 1983) using neoclassical resistivity, INTOR-type transport, and a sawtooth model which repetitively flattens the current and electron temperature profiles out to the radius determined by Kadomtsev's helical flux model (Kadomtsev, 1976), reproduces the $q^{1/2}$ scaling shown in Fig. 7.

We conclude this section by observing that modern low Z_{eff}, low P_{rad}, "low density" ohmic scaling experiments have given results roughly consistent with Eq. (5). The observed peaked $T_e(r)$ profiles, and the relatively weak, but significant, q scaling may be results of the strong aspect ratio scaling of τ_E expressed in Eq. (5), coupled with the presence of rapid sawtooth mixing.

III. AUXILIARY HEATED REGIME

A. Early experiments

Early theoretical studies identified collisionality as the most crucial parameter controlling plasma transport in tokamaks (Düchs and co-workers, 1977), and toroidal β as a central parameter in determining the economic feasibility of a tokamak reactor. Thus the first generation of high power neutral beam injected tokamak experiments were directed towards studying low collisionality (PLT, TFR) and high β (DITE, ISX-B, JFT-2, T-11). The essential experimental technique was to choose target plasma parameters designed to optimize low collisionality, or high β, and then to perform power scaling scans. Thus PLT and TFR initially operated mainly at high plasma current, and relatively low density, while DITE, ISX-B, JFT-2, and T-11 tended to concentrate on low field operation.

The low-collisionality experiments were quite successful in producing high plasma ,temperatures, and low collisionalities (Eubank and co-workers, 1978, 1979; Equipe TFR, 1978b) with no sign of saturation as a function of beam power. The ohmic target plasmas had relatively poor confinement, because of their low density; with beam heating gross confinement was not significantly altered. This allayed initial fears of the theoretically predicted severely negative scaling of τ_E with temperature, which might have been previously hidden by the constraints of ohmic heating. Experiments at higher densities, however, gave mixed results. In general gross confinement was found to be degraded, although central confinement could be significantly improved (Stodiek and co-workers, 1980; Equipe TFR, 1980b; Goldston and co-workers, 1980, 1982). Experiments on PLT in particular showed a frustrating variability.

The high-β injection experiments at first showed very positive results also. The theoretical β limit for a circular cross section tokamak was approached rapidly (Murakami and co-workers, 1980; Leonov and co-workers, 1980; Suzuki and co-workers, 1980). With increasing beam power, however, it began to be clear that confinement was deteriorating as β_p increased (Swain and co-workers, 1981). Equilibrium field measurements (Swain and Neilson, 1982) for total plasma stored energy were very helpful in bringing these results into focus. Multiple regression analysis of the data base accumulated from magnetic equilibrium measurements on ISX-B (Swain and co-workers, 1981) showed that confinement correlated strongly with plasma current, and very little with density, in contradiction with INTOR scaling. These results, and the consequent widespread development of reliable equilibrium and diamagnetic measurements for total stored energy on beam-heated tokamaks, led to a new series of scaling experiments.

B. L-Mode scaling experiments

During the period of 1981 - 1983, the ISX-B, PDX, D-III (JAERI and GA), and ASDEX experimental teams participated in a series of scaling experiments which succeeded in elucidating many of the characteristics of gross confinement in beam-heated tokamak plasmas. In these experiments magnetic and profile measurements of total stored energy were performed (Goldston, 1982), as plasma parameters were systematically stepped across parameter space. Careful attention was paid to MHD activity, and P_{rad} and Z_{eff} were kept low. Early in this period the ISX-B group demonstrated that gross confinement does not depend significantly on toroidal field strength (Swain and co-workers, 1981). Figure 8 shows results from a careful toroidal field scan, with other parameters held fixed (Neilson and co-workers, 1983). This result was confirmed by the GA and JAERI D-III groups (Nagami and co-workers, 1982b). Even a slightly inverse toroidal field dependence was found on PDX (Johnson and co-workers, 1982). Equilibrium measurements on ISX-B gave a strong scaling with plasma current, $\tau_E \sim I_p^{1.5}$ (Swain and co-workers, 1981), while profile measurements gave a somewhat weaker scaling (Neilson and co-workers, 1983). On PDX equilibrium and diamagnetic measurements gave a scaling of $\tau_E \sim I_p^{1.15}$. The two D-III groups found $\tau_E \sim I_p$ in limiter plasmas, as did the ASDEX group. Figure 9 shows the magnetically determined τ_E from ASDEX as a function of plasma current, for a number of different modes of operation (Keilhacker, 1983). (The enhanced confinement of the H-mode, and its scaling, will be discussed in Sec. IIIc.)

Confinement scaling with density is a somewhat more difficult problem. Central core impurity radiation (Stodiek and co-workers, 1980) and beam shine-through can be limitations at low density, and beam penetration can, in principle, be a problem at high density. Magnetic measurements of stored energy must be treated with more care than usual, since the stored beam energy circulating in the plasma varies strongly with density. Figure 10 shows the results of a density scan by the JAERI D-III group, based on profile measurements of plasma stored energy (Nagami, 1983). Other groups have similarly found little or no confinement scaling with plasma density, in L-mode plasmas. The absence of a density scaling of any sort implies that <nT>, or average pressure, is independent of density at fixed input power. Since collisionality scales as n/T^2, this suggests that confinement is independent of collisionality over rather a wide range.

The two D-III groups and the ISX-B group have performed experiments studying the effect of plasma elongation. The D-III groups find that confinement scales approximately as $\kappa^{1/2}$, at fixed plasma current and beam power. The ISX-B group, however, found no scaling with κ independent of the current scaling (Neilson and co-workers, 1983). The stronger I_p scaling observed on ISX-B means that the enhanced current-carrying capability of elongated plasmas gives similar advantages in confinement in ISX-B to those found in D-III. At fixed q the ISX-B and D-III results give essentially the same confinement scaling with κ. Poloidal field limitations in ISX-B, however, have prevented low-q operation of elongated plasmas.

The final confinement scaling question we come to is the one we started with in Sec. IIIa, scaling with input power. Figure 11 shows normalized stored energy versus total input power from the GA D-III group for limiter and divertor (H-mode) plasmas (Burrell and co-workers, 1963). If, on either curve, we connect the 0.5 MW (ohmic heating) points with the points at 4 MW by a straight line, we find that the rest of the points fall on the line, to within the overall scatter. This suggests that the 1 MW of additional heating power between 3 and 4 MW caused about as much increase in plasma stored energy as the first MW. One is tempted then to conclude that some sort of "incremental" confinement time does not deteriorate with increasing input power. This might be the way the plasma moves gradually from ohmic confinement scaling to a beam-heated scaling. On the other hand, in general one finds that the points in the midrange in power tend to fall slightly above the straight line we "drew," and it is certainly true that the absolute confinement time represented by the high power points is significantly lower than typical ohmic confinement times at the same parameters. One may be justified then in concluding that $W_{tot}/(I_p\ \kappa^{1/2})$ scales as P_{tot}^{α}, and thus τ_E falls as $P_{tot}^{(1-\alpha)}$. Curves of this sort generally fit the data about as well as the offset straight line.

Figure 12 shows τ_E/I_p from L- and H-mode PDX plasmas (Kaye and co-workers, 1983). In the L-mode data set points with $B_T < 12$ kG and $P_{tot} > 3$ MW have been excluded in order to avoid conditions where "fishbone" beam ion losses cause signficant reduction in heating efficiency (McGuire and co-workers, 1982). In the range from 2.5 to 5 MW, in the L-mode, τ_E/I_p drops about as $P_{tot}^{-1/3}$. A similar plot for ISX-B (Murakami, 1983), in this case plotting $\tau_E/(I_p\ \kappa^{1/2})$, shows τ_E dropping as $P_b^{-1/2}$ in the range of 1 to 2 MW. The power law fit to the GA D-III limiter data in Fig. 11 gives $\tau_E/(I_p\ \kappa^{1/2}) \sim P_{tot}^{-2/3}$. If we were to choose an "average" power law fit, then we might settle on $P_{tot}^{-1/2}$. For understanding the relationships between current data sets, this sort of scaling law would seem to be as accurate as the offset linear fit, and somewhat more convenient. However, we cannot safely extrapolate to very much higher powers, since we cannot distinguish satisfactorily between the two scalings, and the difference in their implications at the power levels of next-generation devices is substantial.

The next step in putting together an overview of these L-mode confinement results is to try to deduce a size scaling. Confinement scaling results are available from ISX-B, D-III (JAERI and GA groups), PDX, and ASDEX in the form of $\tau_E/(I_p\ \kappa^{1/2})$ at a power level of 2 - 2.5 MW. On the basis of the above discussion, we might have chosen to scale τ_E with a very slightly higher power of I_p, however $\tau_E \sim I_p\ \kappa^{1/2}$ is the median result quoted, and so is certainly satisfactory. Figure 13 shows $\tau_E/(I_p\ \kappa^{1/2})$ for the different machines in different modes of operation. The particular scaling shown comes from a multiple regression fit to the four data points for L-mode deuterium target plasmas, but the scaling fits well for the other two conditions also.

In light of the discussion in Sec. I, and in light of the intrinsic variability of confinement indicated by the existence of the H-mode, by results from PLT, and by the recent enhanced confinement results from ISX-B (see Sec. IIIc), the data in Fig. 13 cannot be considered adequate for a solid scaling law. The strong R dependence, for instance, rests only on the difference between ASDEX and PDX, D-III. It would be desirable to have size scaling data with auxiliary heating from experiments on a single machine, and certainly data from next generation devices will provide an immensely improved lever arm on the scaling coefficients. It must be remarked, however, that the L-mode confinement observed on ISX-B, PDX, D-III, and ASDEX has a certain "rock-bottom" reproducibility, which makes it attractive for these studies.

If, even in the face of the uncertainties, we now combine the scaling results which we have assembled, we find

$$\tau_E(\mathrm{sec}) = 6.4 \times 10^{-8}\, I_p^v(\mathrm{A})\, P_{tot}^w(\mathrm{W})\, R^x(\mathrm{cm})\, a^y(\mathrm{cm})\, \kappa^z \qquad (6)$$

where $v = 1$, $w = -1/2$, $x = 1.75$, $y = -0.37$, $z = 1/2$, for L-mode deuterium target plasmas. Following Neilson and co-workers (1983), we can make use of the relation:

$$P_{tot} = 6\pi^2 R a^2\, \kappa \langle nT\rangle / \tau_E \qquad (7)$$

to eliminate P_{tot} in favor of $\langle nT\rangle \equiv \langle n_i T_i + n_e T_e\rangle/2$. This is only justified if we believe that the change in confinement scaling with neutral beam heating is associated with changes in the bulk plasma parameters, rather than with effects specific to the heating method. Results from ISX-B reported at this conference (Scott and co-workers, 1983) support earlier work on PLT which indicated that plasma rotation due to unbalanced neutral beam injection has little effect on confinement. The fact that confinement deteriorates more strongly at high plasma density than at low density with neutral injection indicates that the anisotropy of the ion distribution function does not degrade confinement, since n_b/n_i falls somewhat more rapidly than $1/n_i^2$. The strong current scaling of confinement with injection heating might suggest that finite-orbit-width effects were important. The fact that τ_E increases as the plasma current is raised even past the point where $q = 2$, and the $q = 1$ surface is near the plasma half-radius, suggests that poloidal field strength is playing a very important role, perhaps by reducing the fast ion orbit excursions. The very good confinement results obtained on PDX with counterinjection (Johnson and co-workers, 1982) could be taken to indicate that confinement is enhanced due to the outward radial current of beam ions associated with counterinjection. The inward radial current associated with co-perpendicular injection on PDX, or co-tangential injection on ISX-b, could then deteriorate confinement. The strong similarity of confinement results on PLT and PDX, however, where PLT operates typically with balanced tangential injection, severely contradicts this hypothesis. (The increased impurity concentration observed with counterinjection in PDX may enhance confinement, as on ISX-B.) Thus, while we await further results from other forms of plasma heating (see Sec. IIId), we are justified in applying Eq. (7). We arrive at:

$$\tau_E^{AUX}(\mathrm{sec}) = 4.3 \times 10^2 \; R^{2.5}\,(\mathrm{cm}) \; a^{-2.74}(\mathrm{cm}) \; I_p^2(\mathrm{A}) \; \langle nT\rangle^{-1}\left(\mathrm{eV\ cm^{-3}}\right) \,. \qquad (8)$$

The Connor-Taylor constraint (1977) for a finite beta, collisional plasma model, when applied to the coefficients of Eq. (6), gives 3w + v + 5 =

4(x + y) which is easily satisfied within the errors on these coefficients. [We apply the constraints to Eq. (6), rather than Eq.(8), in order to more easily assess the errors.] Interestingly, the coefficients in Eq. (6) also meet the additional constraint required when we restrict the plasma model specifically to resistive MHD: $2w + v = 0$. If we interpret the I_p term as expressed in Eq. (8) as being related to poloidal flux ($\sim I_p$), we find once again, as in Eq. (3), that confinement is strongly affected by aspect ratio. If we instead consider I_p as a source of rotational transform, $I_p \sim a^2B_T/(Rq)$, we still find a moderate effect of toroidicity, although now it is easily within the uncertainty in the coefficients.

Some modest insight into the radial profile of χ_e with auxiliary heating can be gained by studying gross features of $T_e(r)$, as we did with ohmic heating. Figure 14 shows central electron temperature divided by volume average electron temperature from ohmic and beam-heated experiments on PDX. Evidently the "peakedness" of the electron temperature profile depends essentially only on q(a), despite the quite different electron heating profiles of ohmic and beam heating as a function of q(a). An effect of this kind was observed on PLT as well (Goldston and co-workers, 1980). In the H-mode in PDX, profiles are slightly broader. This broadening may be more extreme in ASDEX (Wagner and co-workers, 1982b). We may interpret the thermal instability which leads to the sawtooth, then, as not only driven by the intrinsic tendency of ohmic heating power density to build up where T_e is high, but equally driven by the radial profile of transport losses, at least in the L-mode, which pushes even strongly beam-heated plasmas to $q \sim 1$ on axis.

C. The H-mode

The ASDEX group recently discovered a new confinement regime in neutral-beam-heated tokamaks, exhibiting enhanced particle and energy confinement properties (Wagner and co-workers, 1982a, 1982b, 1983; Keilhacker, 1983). This regime could be obtained in ASDEX only with divertor geometry. Qualitatively very similar results were subsequently obtained on PDX (Kaye and co-workers, 1983a, 1983b). In D-III an enhanced confinement mode was also observed by both the GA (Ohyabu and co-workers, 1983; Burrell and co-workers, 1983) and JAERI groups (Nagami and co-workers, 1983a; 1983b) in divertor geometry. The D-III results were qualitatively different in that they did not show an abrupt transition from L- to H-mode during the beam pulse. On ISX-B an enhanced confinement mode was obtained in limiter discharges by injecting small amounts of neon, or by abstaining from titanium gettering (Wootton and co-workers, 1982; Lazarus and co-workers, 1983; Scott and co-workers, 1983). On PDX, using an outside scoop limiter, enhanced confinement was found as well (Budny and co-workers, 1983).

It is by now a common idea that the most crucial element for the attainment of the H-mode may be a reduction in neutral recycling in the main plasma. In both PDX and D-III it was found to be crucial to operate in a geometry where neutrals emitted from the divertor plate could not return directly to the plasma. The role of recycling has been studied in some detail by both the JAERI and GA D-III teams. Some of the scatter in the PDX H-mode confinement data shown in Fig. 12 is due to experiments in which the neutral pressure in the divertor dome was raised to a high level and the pressure in the main chamber rose as well. Under these circumstances the global confinement time dropped to L-mode values. The strong reduction in confinement at higher powers can be explained by the large gas feed that was required to avoid disruptions in PDX, and by the presence of losses due to a macroscopic edge relaxation instability.

Perhaps the role of impurities in generating enhanced confinement comes from the fact that they can fuel the plasma with electrons more deeply than can hydrogen atoms. As a result a weaker gas puff is required to obtain a given plasma density, and so the hydrogen neutral density required for a given plasma density is substantially reduced by a modest admixture of light impurities. The reduction in main plasma neutral density in divertor geometry may come from the fact that the main plasma density is supported by density on the open field lines, reducing the need for neutral fueling to the plasma on closed field lines. This reasoning is consistent with the curious result that enhanced confinement in ISX-B is associated with narrower density profiles than in the L-mode, while in ASDEX and PDX the density profile is broader in the H-mode.

For the purpose of this discussion, perhaps the most critical feature of the H-Mode is that it indicates that gross confinement with auxiliary heating is quite sensitive to boundary conditions. This may be a useful clue in that it is a natural feature of any transport law with a negative temperature dependence, or with a positive dependence on local logarithmic derivatives. The ASDEX group, the PDX group, and both D-III groups have found a reduction in χ_e associated with the H-mode. The JAERI team finds this reduction to be consistent with an overall $T_e^{-1/2}$ scaling for χ_e which they have deduced from L-mode data.

It is probably too early at this time to discuss scaling in the H-mode, since there is some considerable controversy in this area. The ASDEX, PDX, and GA D-III groups find that confinement scaling with current and density in the H-mode is similar to what is observed in the L-mode, only enhanced by a significant factor. In Fig. 13 we have plotted H-mode data only from these three groups. (In the cases of the PDX and GA D-III data we have chosen values characteristic of the best group of H-mode points in the power range of 2-2.5 MW.) The JAERI D-III group finds quite different behavior in the H-mode. Essentially they find a return to INTOR scaling in this regime. The JAERI group uses a somewhat different divertor geometry than the GA group, which may explain some of the difference in results. The ISX-B group finds that τ_E in their enhanced confinement mode still scales strongly with current, but the degree of enhancement varies with plasma density as well. If the best high-density points from the JAERI and ISX-B enhanced confinement modes are plotted on Fig. 12, they fall close to the D-target H-mode line shown.

D. RF heating

Ion cyclotron (ICRF) and lower-hybrid (LHRH) radio frequency heating have both been used successfully on tokamaks. The goal of these RF heating experiments, however, has been largely to understand and optimize the heating process, rather than to study tokamak confinement. In addition, it has been necessary to pay considerable attention to limiting impurity influx associated with RF power. Thus confinement scaling studies with ICRF and LHRH have only just now begun. Hwang and co-workers (1982) studied ICRF and ICRF plus neutral beam heating, as a function of plasma density on PLT. They found a modest degradation in confinement with the application of up to 1.3 MW of absorbed ICRF power, under conditions with low central radiated power losses. The confinement results they obtained are very similar both in absolute magnitude and in scaling to the results of a similar density scan performed on PLT with 1.2 MW of neutral beam heating (Goldston and co-workers, 1982). While this is not an adequate data base for comparing confinement scaling with ICRF and neutral beam injection, it supports the arguments proposed above that the observed confinement degradation with injection heating is not specific to the heating method. Similarly, results of ICRF and neutral beam heating on TFR have been comparable, but in this case at high RF power impurity influx has played an important role, so direct comparison may not be justified (Equipe TFR, 1980c, 1982). Comparisons between ICRF and neutral beam heating have shown similarity on a number of other machines as well [ATC(Takahashi and co-workers, 1977) and JFT-2 (Yamamoto and co-workers, 1982)], but detailed scaling data are lacking. Higher RF power absorption, with low radiative losses, will be required for definitive tests.

Heating by means of electron cyclotron resonance (ECRH) is in principle reasonably well-understood. In addition, the heat deposition can be made quite local, so ECRH is a nearly ideal tool for studying electron thermal transport. The observation on T-10 that the sawtooth instability can be suppressed by heating outside the q = 1 surface is especially interesting (Alikaev, 1983). The T-10 result that central region confinement in the absence of the sawtooth is extremely good is consistent with the analyses presented here.

RF heating is just now coming to the point, in terms of power capability, impurity control, and understanding, where we can hope to see significant contributions to the confinement scaling picture from these experiments.

IV. HIGH DENSITY OHMIC REGIME

Figures 3, 4, and 10 show a saturation in energy confinement time as a function of density which we have alluded to, but not discussed, up to this point. A qualitatively similar saturation was observed on ISX-B, as mentioned above, which could be explained in that case by losses due to neoclassical ion thermal transport. In the case of the GA D-III data (Fig. 3) and the data from Alcator C (Fig. 4) this explanation is not adequate. Neoclassical losses (Bolton and Ware, 1981; Chang and Hinton, 1982) have to be enhanced by a varying factor up to about 4 to explain these results. Indeed it is not even clear that ion transport is the source of the losses, although there are some indications that this is the case (Ejima and co-workers, 1982; Blackwell and co-workers, 1982).

An alternative explanation for the saturation may be derived from observing that our confinement time formula in Eq. (8) for τ_E^{AUX} eventually becomes comparable to or smaller than that of Eq. (3) for τ_E^{OH}, as plasma density is increased in an ohmic heating experiment. Since losses in beam-heated plasmas are dominated by electron thermal conductivity, τ_E^{AUX} must reflect losses in the electron channel. Thus this analysis is only valid under the assumption that electron losses cause the observed saturation in confinement. With this assumption, then, we proceed with a simple 0-dimensional analysis. The ohmic input power in a Z = 1 Spitzer resistivity plasma is approximately given by:

$$P_{OH}(W) = 0.167\ I_p^2(A)\ R(cm)/\left[a^2(cm)\ T^{3/2}(eV)\kappa\right] \qquad (9)$$

We now need a way to put together the confinement times expressed in Eqs. (3) and (8). If we combine them in the simplest way,

$$1/\tau_E = 1/\tau_E^{OH} + 1/\tau_E^{AUX} \tag{10}$$

and solve for τ_E vs. n_e (assuming $n_eT_e = n_iT_i$), we do not find an adequate $\tau_E \sim n_e$ region in most experiments. This is because the $1/\langle nT\rangle$ term has a noticeable effect even at low densities. Perhaps this problem could be alleviated by allowing $n_iT_i < n_eT_e$ in this region. If instead we choose to combine the confinement times in the form

$$(1/\tau_E)^2 = (1/\tau_E^{OH})^2 + (1/\tau_E^{AUX})^2 , \tag{11}$$

we find remarkably good agreement with experiment. Maybe this indicates that each transport mechanism requires control over the profiles to have its full effect. Figure 15a shows this 0-D, $Z_{eff} = 1$ simulation of the 44 cm D-III current and density scaling experiment, shown as panel a in Fig. 3. Figure 15b is a simulation of a full-bore Alcator C plasma, giving deviation from $\tau_E \propto \bar{n}_e$ scaling at the observed value of 2×10^{14} cm^{-3}.

A feature of the ohmic density scaling experiments which this simple calculation will not reproduce, is the weak saturation with density seen in some of the Alcator C size-scaled plasmas, and the weak or nonexistent saturation in the FT data. In both of these cases the density is found to have peaked profiles, as opposed to the broad profiles found in the full-bore Alcator C and D-III plasmas. Perhaps this implies that the high-density ohmic regime, dominated by τ_E^{AUX}, has an L- and H- mode similar to what is observed with auxiliary heating.

V. CONCLUSIONS

In this paper we have examined a large quantity of high quality tokamak confinement data. We have tried, without doing violence to the data, to develop an overview of confinement scaling in tokamaks with both ohmic and strong auxiliary heating. The fact that such different scalings are found in the "low density" ohmic regime [Eq. (3)] and the auxiliary heating regime [Eq. (8)], has made this a complicated task. Remember that the original hope was that auxiliary heating experiments would clarify the confinement scaling questions which could not be addressed with ohmic heating, rather than produce new questions. If we assumed with Connor and Taylor that fundamentally confinement scaled with n,T,B,and a (where B and a represent any magnetic field, and any size parameter), then the "low density" ohmic result of $\tau_E \propto \bar{n}_e$ suffered from no ambiguity associated with the constraints between T, B, and a imposed by ohmic heating. Auxiliary heating was simply called upon to untangle the ohmic restrictions on separating scalings with T, B, and a. In fact, of course, one of the early results of the scaling studies on ISX-B was that τ_E was found to be independent of $\bar{n}_e$, removing the one certainty which was then available.

It may be, however, that we have now come full circle. After examining confinement scaling with beam heating in some detail, we have constructed a tentative law for τ_E^{AUX} which is quite different from the "low density" ohmic scaling. However, when brought back to the ohmic arena and applied to the saturation of τ_E at "high density," it does seem to provide us with valuable insight.

As with all scaling studies, we must be careful in applying these results beyond the range where they have been tested. In particular, the scaling of confinement with input power which is expressed in Eq. (6) is open to serious question, and the size scaling of Eq. (8) rests on somewhat shaky ground. The next generation of large, high-power experiments should clarify these particular issues. In all probability, however, they will also raise some other completely different set of questions for us to puzzle out.

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FIGURE CAPTIONS

FIG. 1 Published confinement data from ST tokamak, R_o = 109 cm, a as listed.

FIG. 2 Minor radius scaling experiment from T-11, $\bar{n}_e$ 1.5 x 10^{13} cm^{-3}, B_T = 9.4 kG, q = 2.5.

FIG. 3 $\bar{n}_e$, I_p, and a scaling experiment from GA D-III group. B_T = 24 kG.

FIG. 4 Major radius scaling experiment from Alcator C.

FIG. 5 Regression fit to data from low Z_{eff}, low P_{rad}, moderate q, confinement experiments, from Alcator C group TFTR point new.

FIG. 6 Comparison of T_e profiles from standard INTOR χ_e α $1/n_e$ vs. "Neo-Alcator" χ_e α r/n_e; Spitzer resistivity and simple sawtooth model. PLT data obtained by symmetrizing Fig. 8a of Bol and co-workers, 1978.

FIG. 7 τ_{Ee} and r(q=1) vs. q(a) predicted from "neo-Alcator" χ_e, Spitzer resistivity, and simple sawtooth model.

FIG. 8 B_T scan from ISX-B showing little dependence of stored energy on B_T.

FIG. 9 Current scaling of magnetically measured confinement on ASDEX for H and D target plasmas, in L- and H-mode, B_T = 24 kG.

FIG. 10 Density scaling experiments by JAERI D-III team. D target, L-mode.

FIG. 11 Normalized magnetically measured stored energy for limiter and divertor (H-mode) plasmas by GA D-III team.

FIG. 12 Normalized confinement time versus total input power in PDX limiter and divertor plasmas.

FIG. 13 Regression analysis of size scaling of τ_E at fixed P_{tot} = 2.25 MW.

FIG. 14 Central peaking factor for T_e vs q(a) for ohmic and beam-heated plasmas in PDX.

FIG. 15 Simulations of saturation of τ_E vs. $\bar{n}_e$ in ohmically heated plasmas. τ_E from combining formulae for ohmic and auxiliary heating.

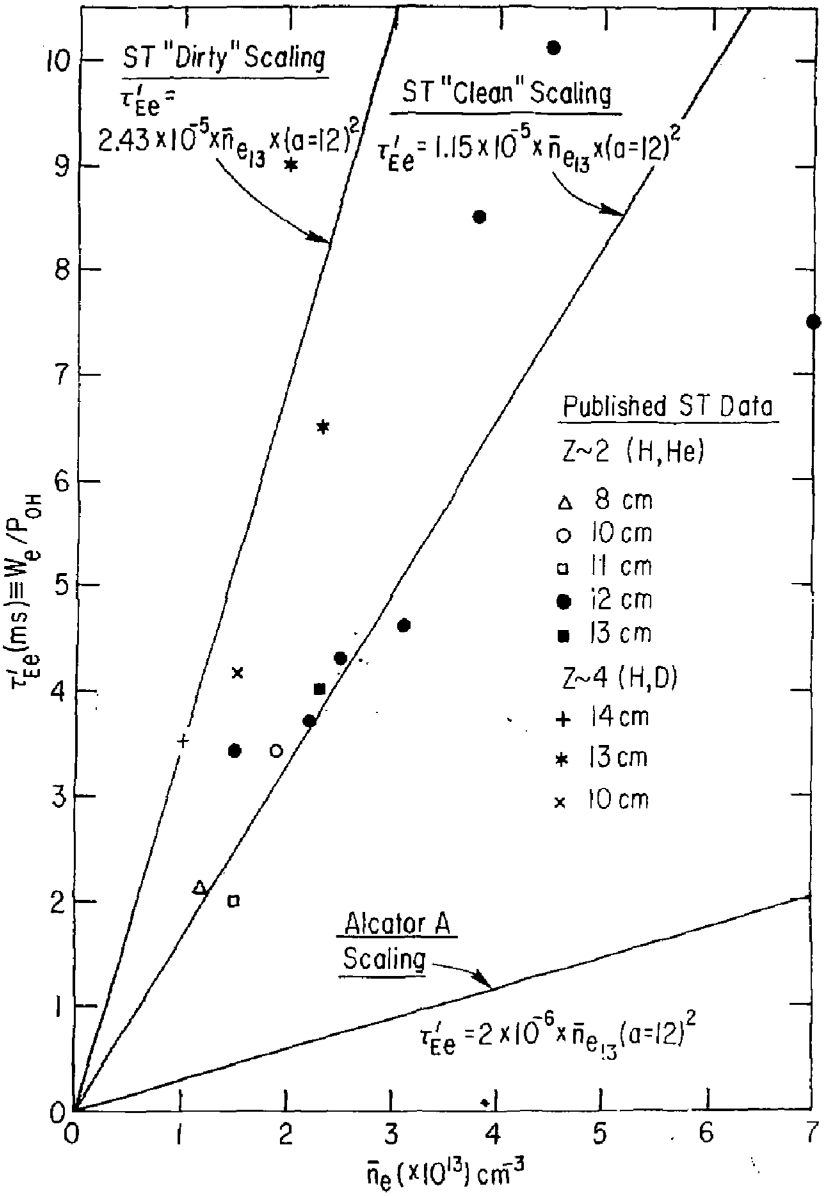

ST "Dirty" Scaling
τ'_{Ee} = $2.43\times10^{-5}\times\bar{n}_{e_{13}}\times(a=12)^2$
ST "Clean" Scaling
$\tau'_{Ee} = 1.15\times10^{-5}\times\bar{n}_{e_{13}}\times(a=12)^2$
Published ST Data
Z~2 (H,He)
△ 8 cm
○ 10 cm
□ 11 cm
● 12 cm
■ 13 cm
Z~4 (H,D)
+ 14 cm
* 13 cm
× 10 cm
Alcator A Scaling
$\tau'_{Ee} = 2\times10^{-6}\times\bar{n}_{e_{13}}(a=12)^2$
τ'_{Ee}(ms)$\equiv W_e/P_{OH}$
0
1
2
3
4
5
6
7
8
9
10
0
1
2
3
4
5
6
7
$\bar{n}_e$ ($\times10^{13}$) cm^{-3}

Fig. 1

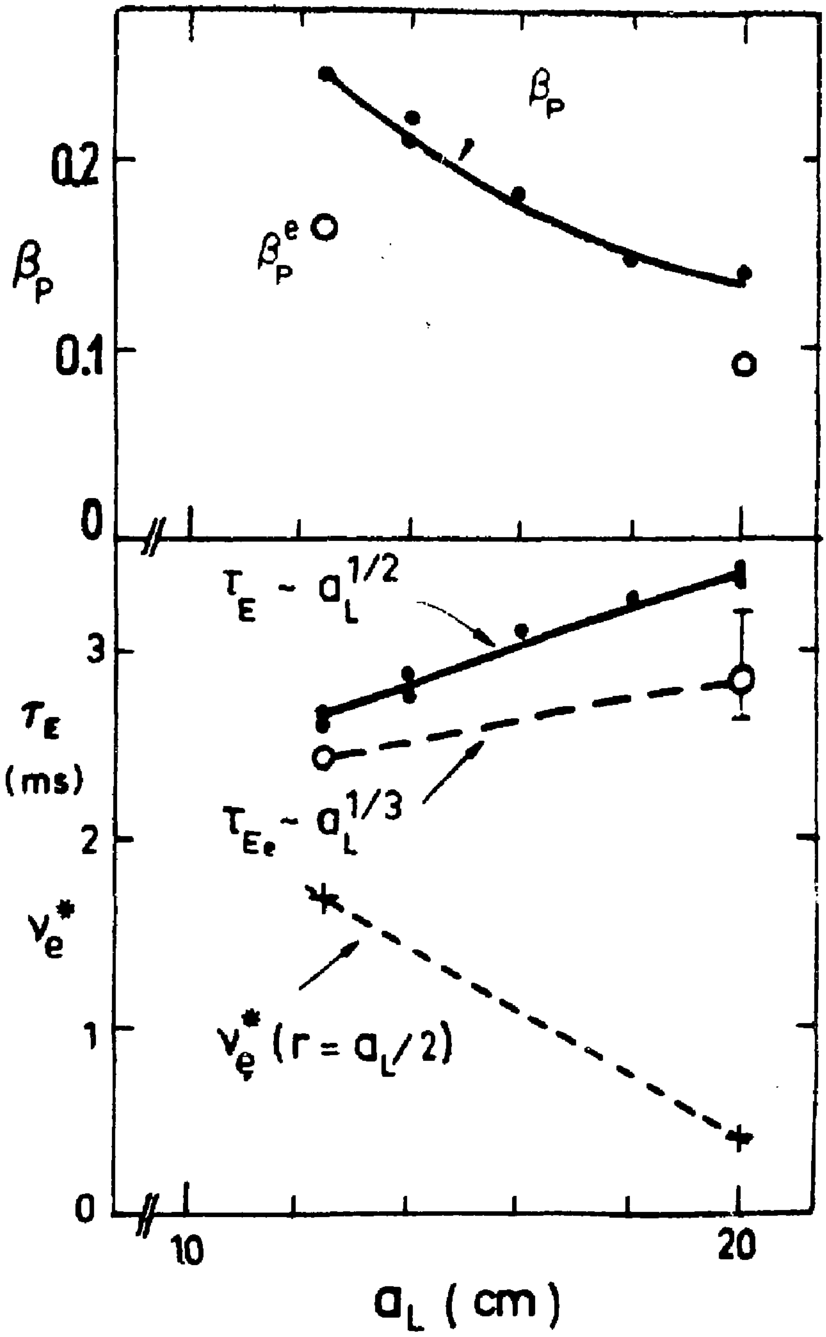
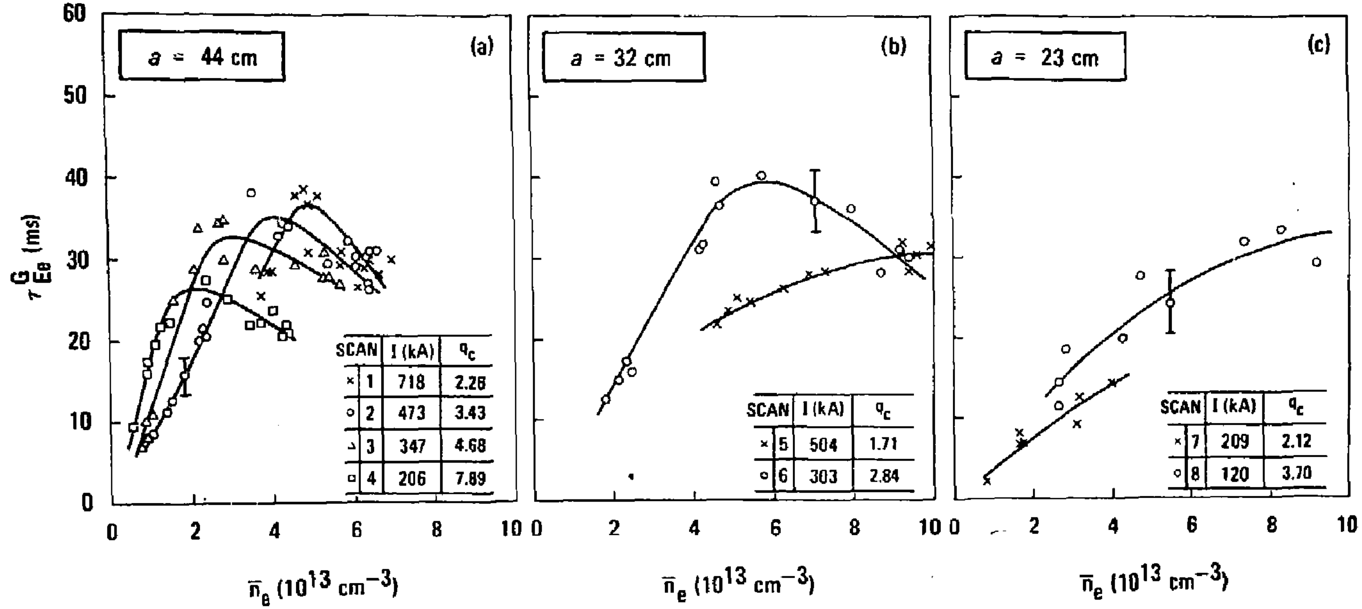

β_p
0.2
0.1
0
β_p
β_p^e
τ_E (ms)
3
2
1
0
ν_e^*
$\tau_E \sim a_L^{1/2}$
$\tau_{Ee} \sim a_L^{1/3}$
$\nu_e^*(r = a_L/2)$
10
20
a_L (cm)

Fig. 2



SCAN		I (kA)	q_c
×	1	718	2.26
○	2	473	3.43
△	3	347	4.68
□	4	206	7.89

SCAN		I (kA)	q_c
×	5	504	1.71
○	6	303	2.84

SCAN		I (kA)	q_c
×	7	209	2.12
○	8	120	3.70

Fig. 3

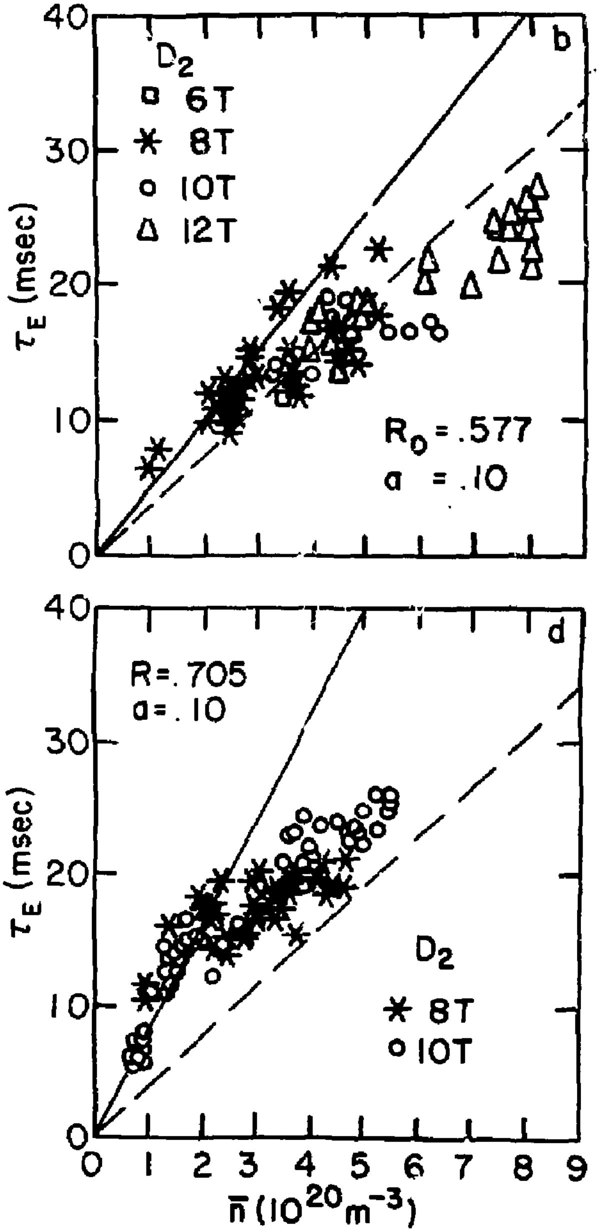

b
D_2
□ 6T
✱ 8T
○ 10T
Δ 12T
R_0 = .577
a = .10
d
R=.705
a=.10
D_2
✱ 8T
○ 10T
τ_E (msec)
0
10
20
30
40
0 1 2 3 4 5 6 7 8 9
$\bar{n}$ ($10^{20}m^{-3}$)

Fig. 4

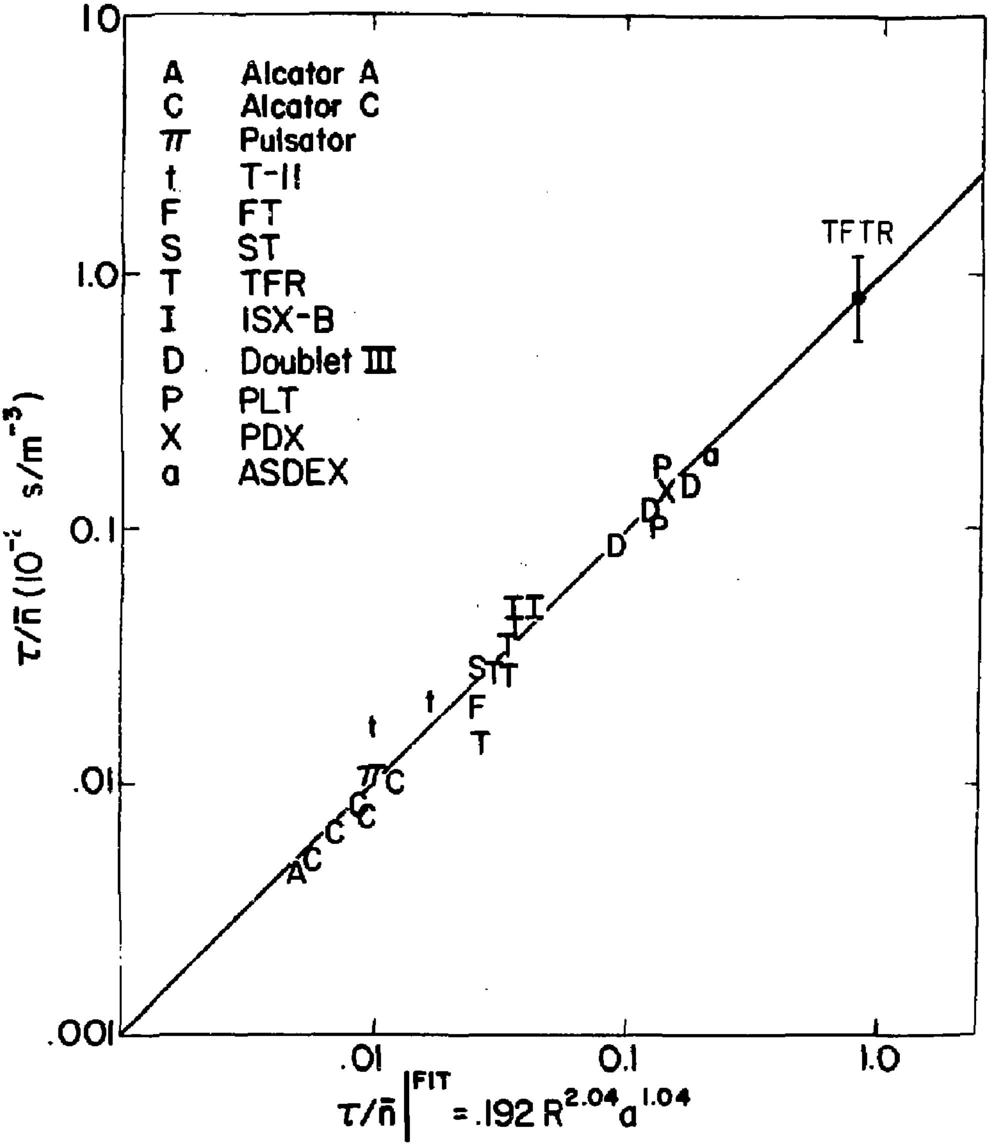

A Alcator A
C Alcator C
π Pulsator
t T-11
F FT
S ST
T TFR
I ISX-B
D Doublet III
P PLT
X PDX
a ASDEX
TFTR
10
1.0
0.1
.01
.001
.01
0.1
1.0
$\tau/\bar{n}$ (10^{-20} s/m^{-3})
$\tau/\bar{n}\Big|^{FIT} = .192\,R^{2.04}a^{1.04}$

Fig. 5

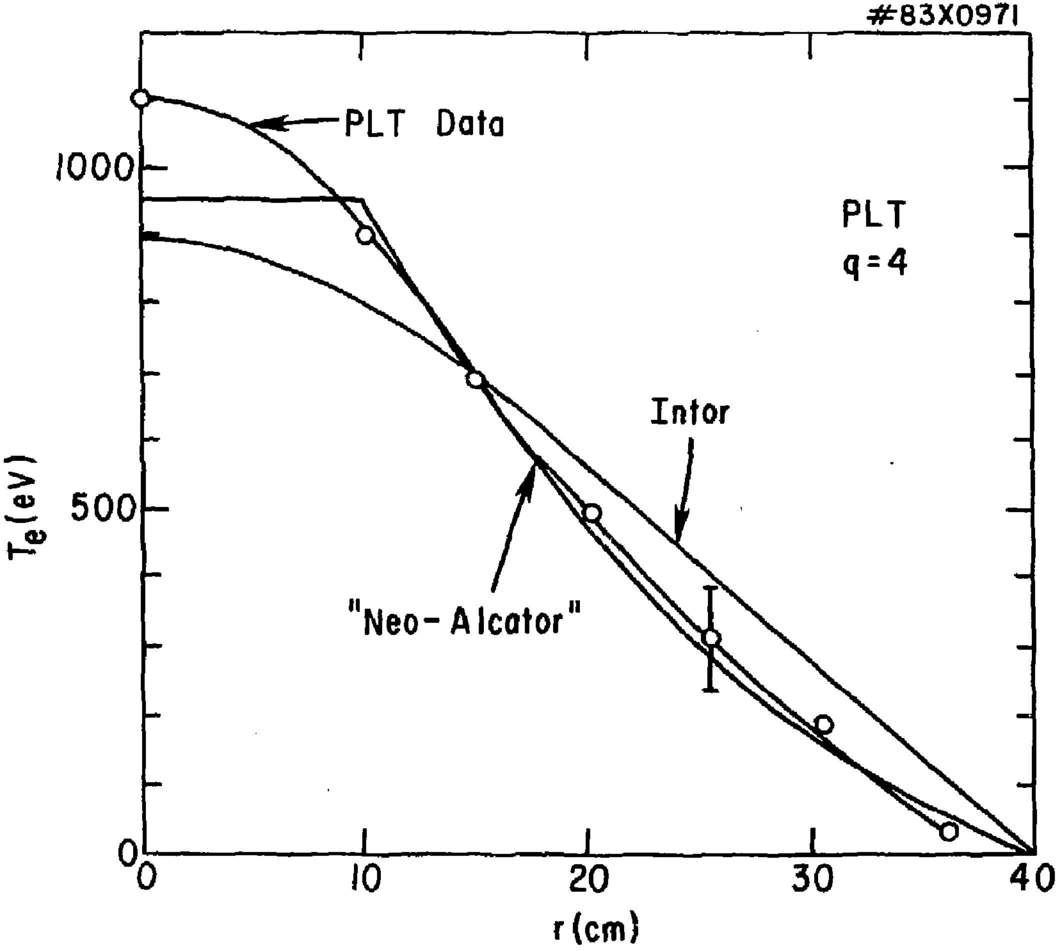

#83X0971
PLT Data
PLT
q=4
Intor
"Neo-Alcator"
T_e(eV)
1000
500
0
0
10
20
30
40
r (cm)

Fig. 6

#83X0771

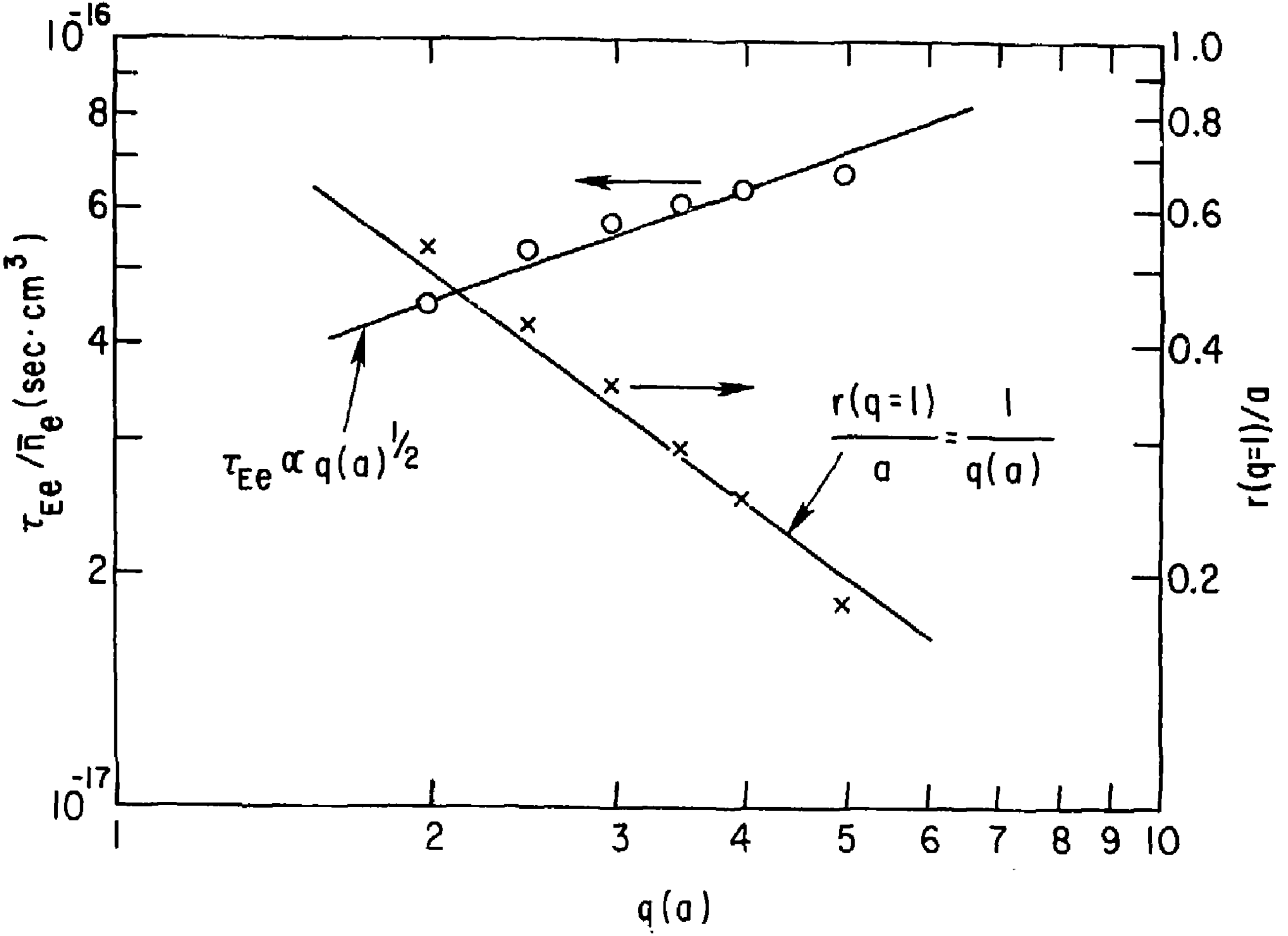


Fig. 7

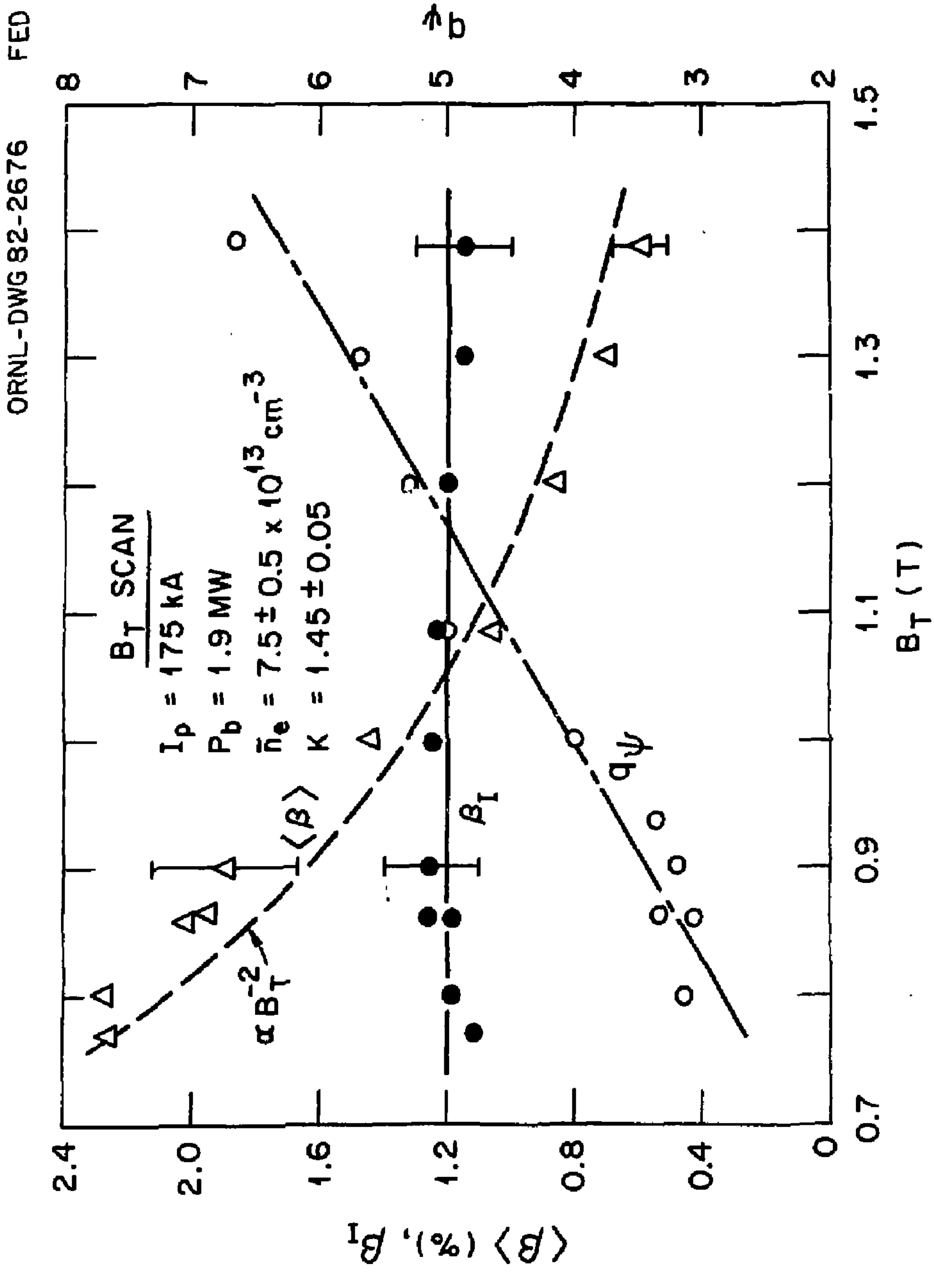
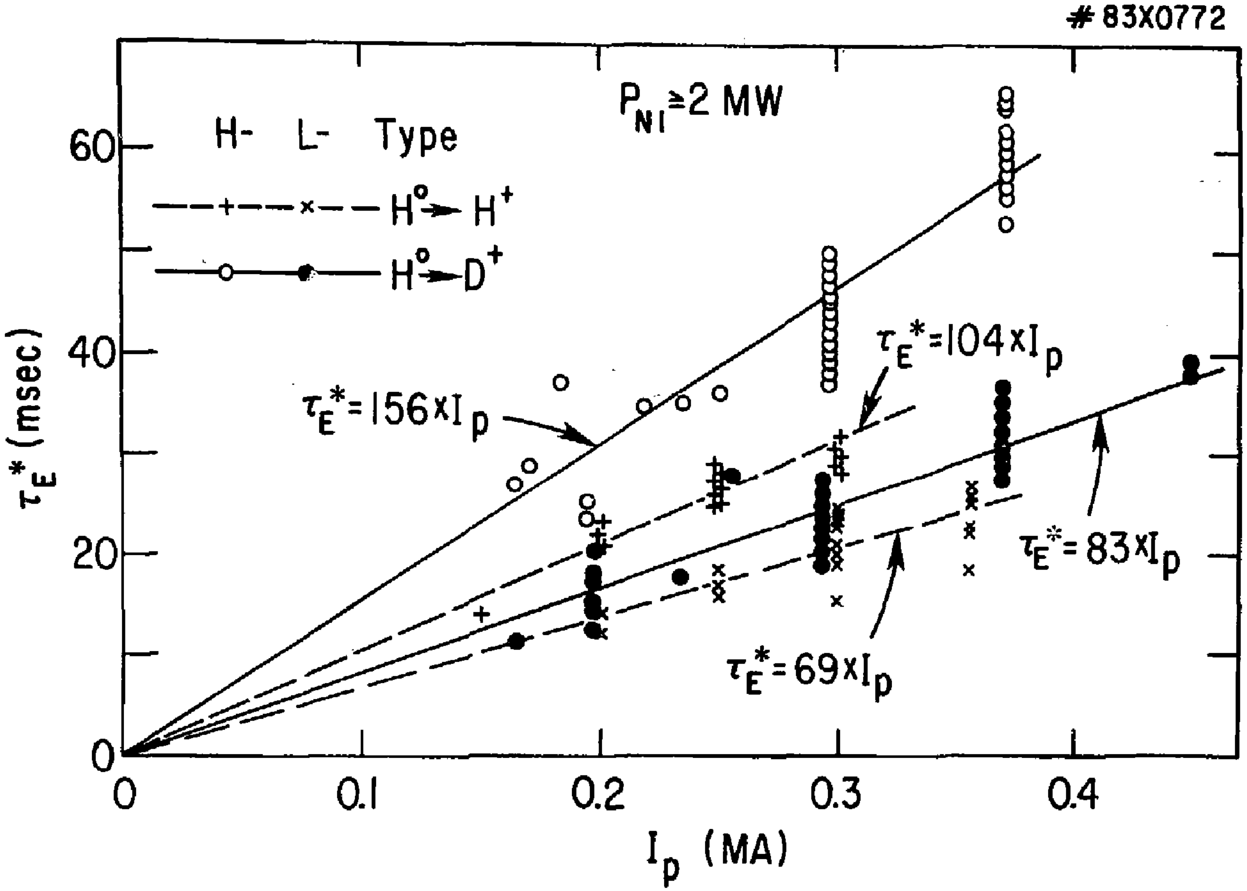

ORNL-DWG 82-2676
FED
B_T SCAN
I_p = 175 kA
P_b = 1.9 MW
n̄_e = 7.5 ± 0.5 x 10^13 cm^-3
K = 1.45 ± 0.05
⟨β⟩
β_I
q_ψ
∝ B_T^-2
⟨β⟩ (%), β_I
0
0.4
0.8
1.2
1.6
2.0
2.4
q_ψ
2
3
4
5
6
7
8
B_T (T)
0.7
0.9
1.1
1.3
1.5

Fig. 8



83X0772
$P_{NI} \geq 2$ MW
H- L- Type
$H^o \rightarrow H^+$
$H^o \rightarrow D^+$
$\tau_E^* = 156 \times I_p$
$\tau_E^* = 104 \times I_p$
$\tau_E^* = 83 \times I_p$
$\tau_E^* = 69 \times I_p$
τ_E^* (msec)
0
20
40
60
0
0.1
0.2
0.3
0.4
I_p (MA)

Fig. 9

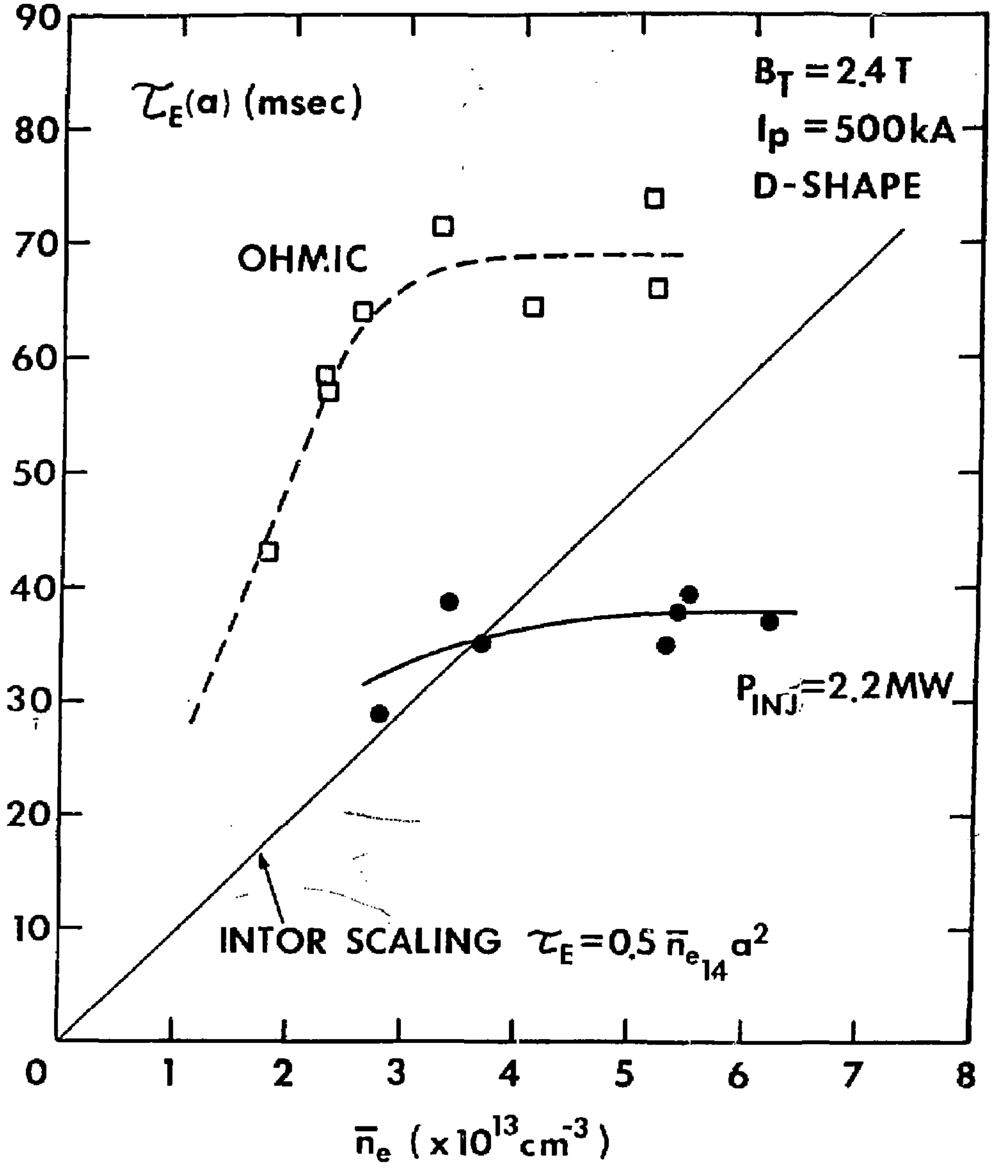

$\tau_E(a)$ (msec)
$B_T = 2.4$ T
$I_p = 500$kA
D-SHAPE
OHMIC
$P_{INJ} = 2.2$MW
INTOR SCALING $\tau_E = 0.5\, \bar{n}_{e_{14}} a^2$
90
80
70
60
50
40
30
20
10
0
1
2
3
4
5
6
7
8
$\bar{n}_e$ ($\times 10^{13}$cm^{-3})

Fig. 10

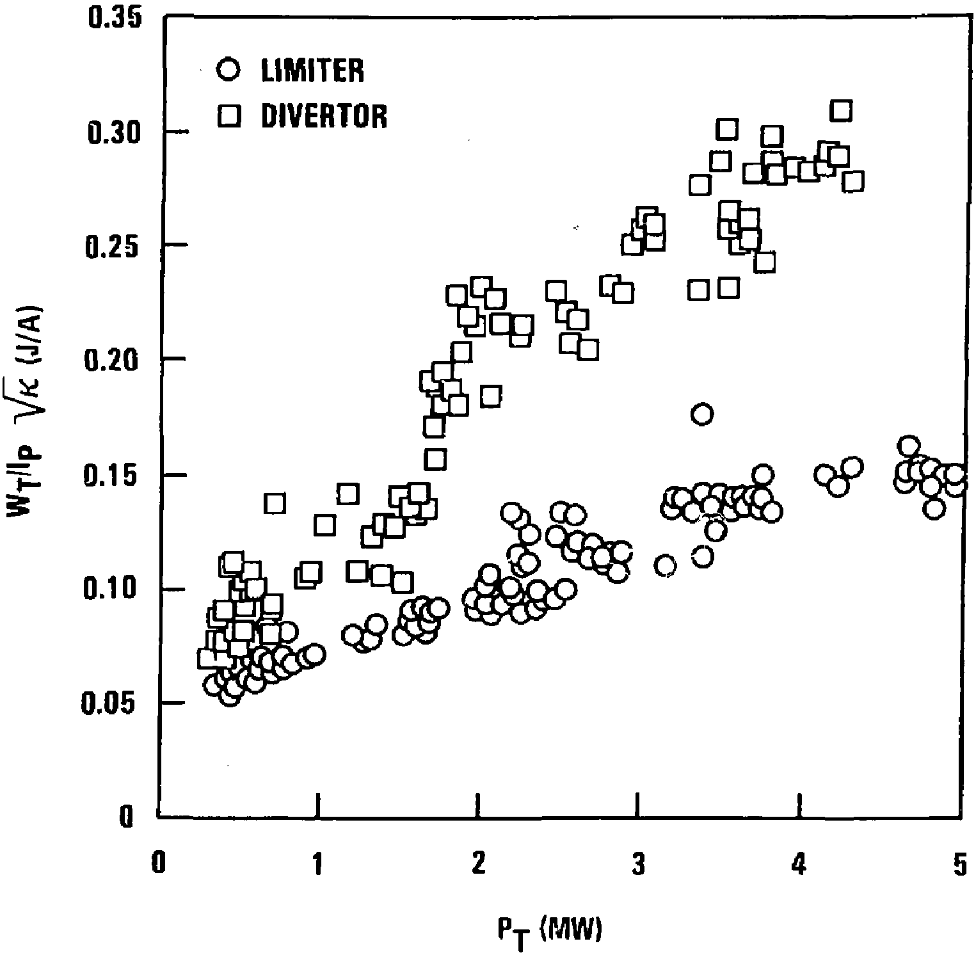
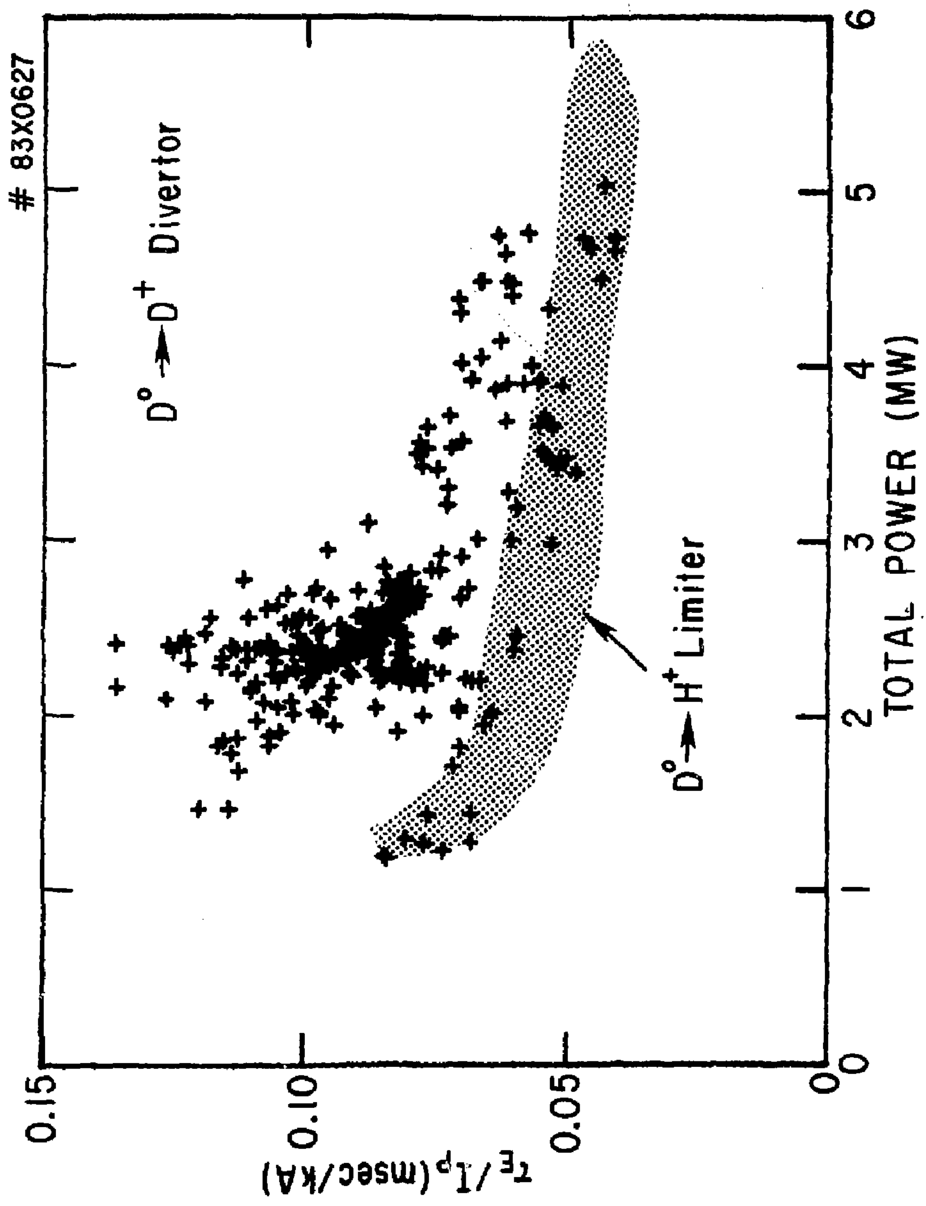

LIMITER
DIVERTOR
$W_T/I_P\ \sqrt{\kappa}$ (J/A)
0.35
0.30
0.25
0.20
0.15
0.10
0.05
0
0
1
2
3
4
5
P_T (MW)

Fig. 11



83X0627
D° → D+ Divertor
D° → H+ Limiter
TOTAL POWER (MW)
0
1
2
3
4
5
6
τE/Ip (msec/kA)
0.15
0.10
0.05
0

Fig. 12

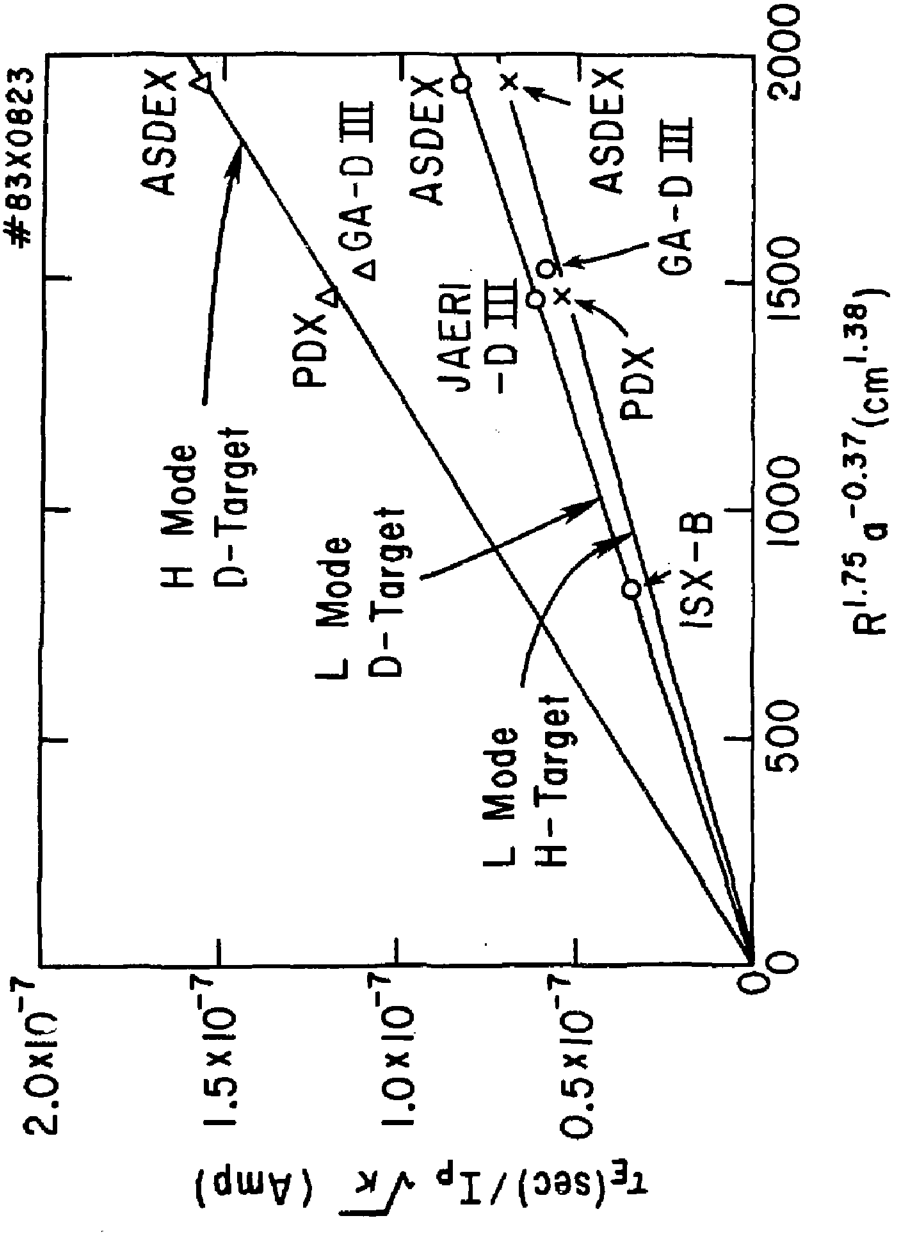

#83X0823
ASDEX
H Mode
D-Target
PDX
GA-D III
ASDEX
JAERI
-D III
L Mode
D-Target
ASDEX
GA-D III
PDX
ISX-B
L Mode
H-Target
τ_E(sec)/$I_p \sqrt{\kappa}$ (Amp)
2.0x10^{-7}
1.5x10^{-7}
1.0x10^{-7}
0.5x10^{-7}
0
0
500
1000
1500
2000
$R^{1.75} a^{-0.37} (cm^{1.38})$

Fig. 13

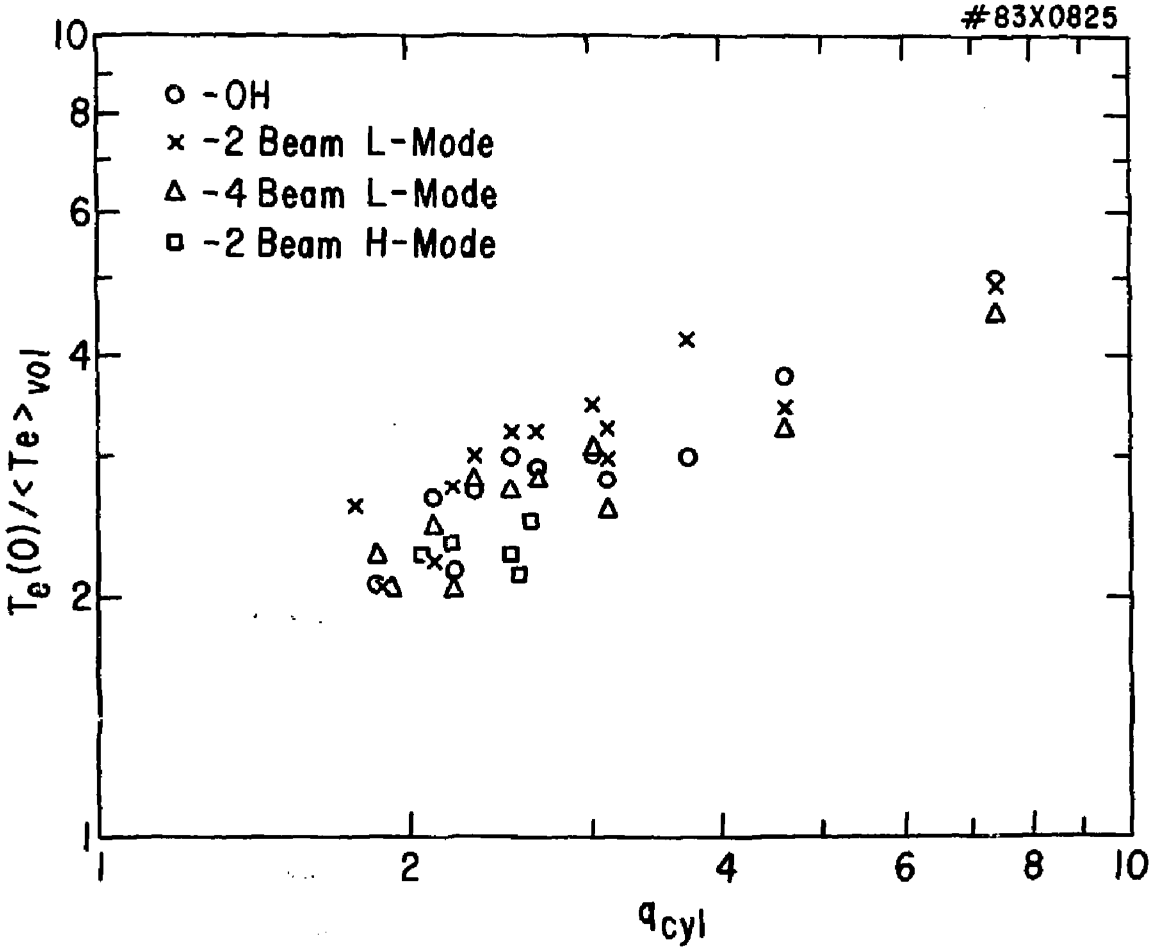

#83X0825
○ -OH
× -2 Beam L-Mode
△ -4 Beam L-Mode
□ -2 Beam H-Mode
$T_e(0)/\langle T_e\rangle_{vol}$
10
8
6
4
2
1
1
2
4
6
8
10
q_{cyl}

Fig. 14

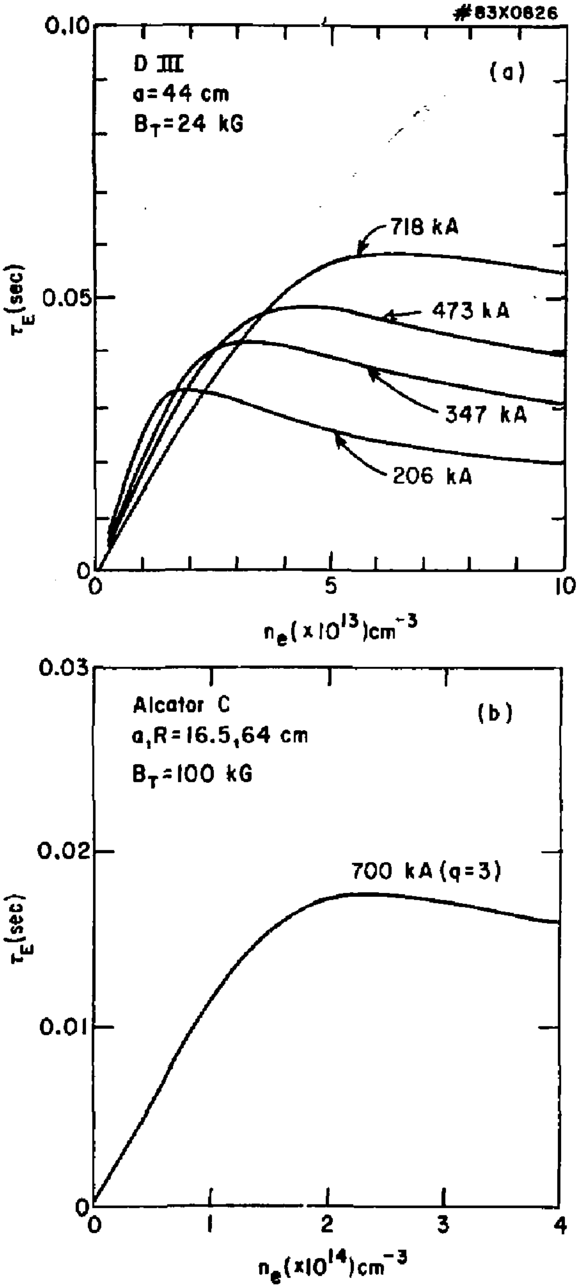

#83X0826
0.10
D III
a=44 cm
B_T=24 kG
(a)
718 kA
473 kA
347 kA
206 kA
τ_E(sec)
0.05
0
0
5
10
n_e(×10^13)cm^-3
0.03
Alcator C
a,R=16.5,64 cm
B_T=100 kG
(b)
0.02
700 kA (q=3)
τ_E(sec)
0.01
0
0
1
2
3
4
n_e(×10^14)cm^-3

Fig. 15

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