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# PROGRESS IN TOROIDAL CONFINEMENT AND FUSION RESEARCH

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# Progress in Toroidal Confinement and Fusion Research

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## ABSTRACT

During the past 30 years, the characteristic $T_i n\tau_E$-value of toroidal-confinement experiments has advanced by more than seven orders of magnitude. Part of this advance has been due to an increase of gross machine parameters. Most of the advance is associated with improvements in the "quality of plasma confinement." The combined evidence of spherator and tokamak research clarifies the role of magnetic-field geometry in determining confinement and points to the importance of shielding out plasma edge effects. A true physical understanding of anomalous transport remains to be achieved.

## 1. Introduction

When I first met Marshall Rosenbluth, exactly thirty years ago, the forefront of fusion research had reached approximately the point marked by an asterisk in the Lawson diagram of Fig. 1. The product $T_i(o)\ n(o)\tau_E$ of central ion temperature, central density, and global energy confinement time stood at about $10^8$ keV $cm^{-3}$ sec -- disturbingly far from the goal of $5 \times 10^{15}$ keV $cm^{-3}$ sec for an ignited D-T reactor. During the intervening decades, experimental progress has been fairly steady, so that the current $T_i(o)\ n(o)\ \tau_E$-product (end of 1986) stands at $2 \times 10^{14}$ keV $cm^{-3}$ sec. On this nostalgic occasion, it may be appropriate to reexamine the means whereby such substantial progress was made in toroidal confinement research.

Some favorite fusion-reactor candidates of the mid-50's were: the tokamak (invented by A. Sakharov and I. Tamm -- and independently by L. Spitzer, Jr.), the stellarator (invented by Spitzer), and the reversed-field pinch (invented by M. Rosenbluth[1]). In the meantime, MHD theory has evolved to include finite resistivity[2] and some other important effects.[3] There have been a number of stylistic refinements in the architecture of tokamaks (cross-sectional shaping and profile optimization), stellarators (reduction of connection lengths and pfirschlüter currents), and RFP's (utilization of the dynamo mechanism), but the favorite present-day reactor candidates are still: the tokamak, the stellarator, and the RFP. Contrary to a wide-spread expectation of the 50's, the evolution of magnetic architecture does not seem to have been the principal key to progress in the Lawson diagram.

The experimental results of the 50's and early 60's were mostly discouraging. Stellarators exhibited Bohm diffusion[4]

$$D_{Bohm} = (1/16)\ c\ T_e/eB \sim 6 \times 10^3\ T_e(eV)/B(kΓ)\ cm^2\ sec, \qquad (1)$$

$$\chi_{Bohm} \sim 3D_{Bohm}, \qquad (2)$$

over a broad range of plasma parameters. For stellarator configurations that obeyed the MHD stability conditions, there was no apparent further dependence of confinement on such architectural features as the rotational transform and shear. (One should note, however, that the available range of effective $\langle B_p \rangle$ was quite limited, corresponding to $\langle B_p \rangle / B_t < 10^{-2}$). In other types of toroidal configuration, such as the tokamak and the RFP, the observed confinement scaling was not obviously Bohmlike, but the observed magnitudes of anomalous transport seemed to offer no improvement relative to Eqs. (1) and (2).

At this stage of fusion research, the value of $T_i(o)\ n(o)\ \tau_E$ stood at about $10^{10}$ keV $cm^{-3}$ sec, and reactor extrapolations based on the Bohm formula called for unacceptably large plasma minor radii, in excess of $10^3$ cm. Theoretical interpretations of this state of affairs tended to appeal to drift waves and/or small-scale resistive MHD modes, but there was a lack of demonstrated correspondence between experiment and theory.

At this low point in the history of toroidal confinement research, the Ioffe "minimum-B" experiment,[5] first reported in 1961, struck a note of good cheer and served to revitalize the search for superior architecture. The potential advantages for toroidal confinement of a favorable magnetic well were recognized, and the invention of "minimum-average-B" configurations, with and without current-carrying rings floating inside the plasma, proved to be an exhilarating pastime.[6]

The relative architectural merits of some major toroidal-confinement options are illustrated in Table I. If one wishes to eliminate possible drivers of anomalous transport, the ability to operate without any net $J_{\parallel}$ is clearly an asset. Some other potentially favorable features are strong shear and a deep average magnetic well, with short connection lengths between regions of good and bad curvature. To reduce the possible threat from trapped-particle modes, there is a further benefit from avoidance of local mirror trapping in regions of bad curvature. In regard to these figures of architectural merit, floating-ring devices[7-10] are clearly most advantageous -- but unfortunately they are poorly suited for use in a D-T reactor. Marshall Rosenbluth played a key role in the invention of a powerful non-floating-ring solution of the stellarator type,[11] currently known as the heliac. Compared with these entries in categories 1 and 2 of Table I, the tokamak configuration has few and feeble architectural merits -- except for the virtue of simplicity.

The following section reviews the experimental data obtained during the late 60's and early 70's in architecturally optimal configurations of the spherator type (Fig. 2). The confinement results are related to the early stellarator experience and to the major advances introduced by the tokamak approach.[12] Section 3 briefly reviews the present state of tokamak research. Section 4 compares anomalous transport phenomena in spherators and tokamaks. Section 5 attempts to infer a general model for toroidal confinement.

## 2. Spherator Experiments

During the early 1960's, simple "levitron" experiments[9] were carried out with transiently free-floating copper rings, in order to test the ideal MHD stability theory. Later the emphasis switched to the study of near-vacuum-field low-$\beta$ confinement, and a number of superconducting-ring devices were built. The present discussion will make particular use of the results of the FM-1 device[13] of Fig. 3, which was operated by S. Yoshikawa et al. during the early 1970's.

The FM-1 contained a superconducting ring of 75 cm major radius, capable of carrying several hundred kA of ring current and remaining afloat in its own $B_p$-field for an 8-hour experimental shift in a room-temperature environment. The FM-1 had substantial flexibility for exploring the range of architectural effects referred to in Fig. 2 and Table I -- and incidentally pioneered the poloidal-field divertor concept.[14]

As illustrated in Fig. 4, the FM-1 studied relatively cold, low-density plasmas and achieved particle confinement times exceeding one second. Particle diffusion was found to decrease with rising $T_e$, up to roughly the point where the trapped-electron bounce frequency exceeded the collision frequency. Thereafter, diffusion increased again, following a Bohmlike scaling, but with geometrically determined proportionality constants that were as much as 300 times smaller than predicted by Eq. (1).

For a "tokamaklike" electron-cyclotron-wave-heated FM-1 plasma like that of Fig. 2b (with $B_t \sim B_p$), the particle and heat diffusion coefficients are shown in Fig. 5 as functions of $T_e$. [It should be noted that, both in the spherator studies and the earlier stellarator work, transport coefficients were determined <u>globally</u>, on the basis of $\tau^{-1}$, not <u>locally</u> as a function of plasma radius.] In the example of Fig. 5, the particle transport D was found to be about 100 times less than $D_{Bohm}$ and the heat transport $\chi$ (called $K_\perp$ in the figure) was about 10 times less than $D_{Bohm}$, or 30 times less than $\chi_{Bohm}$ [cf. Eq. (2)].

Turning to the impact of various architectural effects, the FM-1 experiments documented a very marked deterioration of confinement in the limit $B_t << B_p$ (Fig. 6) where the shear becomes weak and there is unfavorable magnetic curvature everywhere on the outer plasma surface. [FM-1 confinement was typically very much better on the inner surface of the plasma, facing the ring, which is in a true minimum-B situation.]

The particle confinement times associated with a variety of FM-1 configurations[15] are shown in Fig. 7. Optimum confinement, corresponding to Fig. 4 and to case (c) in Fig. 2, was obtained for ratios of ring current $I_p$ to vertical-field current $I_E$ around unity, and for moderately high ratios of toroidal-field current $I_T$ to ring current ($I_T/I_p \sim 3$ $B_t/B_p \sim 1$). At lower values of toroidal field, the low-shear regime of Fig. 6 is encountered. At higher values of $B_t/B_p$, where the configuration becomes increasingly tokamaklike (Fig. 5), confinement is also found to deteriorate. The lower curve in Fig. 7 shows the unfavorable effect of raising $I_p/I_E$, which produces an outward shift of the poloidal flux surfaces. [The extreme limit $I_p >> I_E$, shown as case (a) in Fig. 2, corresponds to the earliest levitron experiments,[9] which had no external vertical magnetic field.]

Extensive fluctuation studies were carried out in various spherator experiments,[13-16] mostly by means of Langmuir probes. The magnitude of the fluctuation level $\delta n/n$ was generally correlated with the magnitude of the transport coefficients: For example, fluctuations were found to be smallest in the well-confined case (c) of Fig. 2, and on the inner side (ring side) of the plasma. They were particularly large in case (a) and for weak shear ($B_t << B_p$). An important discovery was that "convective cells" -- i.e., fluctuations in space with $\omega = o$, could play a dominant role in accounting for anomalous transport in well-confined and seemingly quiescent spherator plasmas.[17]

The overall results found for energy confinement in the FM-1 experiments suggest the following simple model (which is based, with some added conservatism, on Ref. 18):

$$\chi = (1/30)\, G\, \chi_{\mathrm{Bohm}} = 600\, G\, T_e(\mathrm{eV})/B(\mathrm{k}\Gamma)\ \mathrm{cm}^2/\mathrm{sec}, \qquad (3)$$

where G is a number that depends on the magnetic-field geometry. There are three architectural cases of principal interest:

1. Optimal Geometry. *The flux surfaces are like those of case (c) in Fig. 2, with* $B_t \lesssim B_p \sim B$, giving high shear and trapping on the small-R side -- but only marginally favorable (or even slightly unfavorable) average magnetic well. In this case, G reaches its minimum value of

$$G_{\min} \sim 1/3. \qquad (4)$$

2. Approaching Pure Poloidal Field. The flux surfaces are shaped like those of case (b) or (c) in Fig. 2, but with $B_t \ll B_p \approx B$, so that the magnetic well is strongly unfavorable and the shear length increases as $L_s \propto B_p/B_t$. In this limit we have

$$G \rightarrow B_p/B_t. \qquad (5)$$

3. Approaching Tokamak Geometry. The flux surfaces are shaped like those of case (b) or (c), but with $B_p \ll B_t \approx B$, so that the magnetic well is favorable, but the particle trapping is on the large-R side, and the shear length increases as $L_s \propto B_t/B_p$. In this limit we have

$$G \rightarrow B_t/B_p. \qquad (6)$$

Let us first consider how the spherator confinement results can be related to the results of the early stellarators, which studied plasmas with somewhat similar parameters. On the basis of the spherator data, we should use Eqs. (3) and (6) to define a "neo-Bohm" formula

$$\chi_B^* = (B_t/30\ B_p)\ \chi_{Bohm} = 600\ T_e(eV)/B_p(k\Gamma)\ cm^2/sec, \qquad (7)$$

characterizing low-$B_p/B_t$ devices such as stellarators and tokamaks. For typical effective values of $\langle B_p\rangle/B_t < 10^{-2}$ achieved in early stellarator experiments,[4] the prediction of Eq. (7) is seen to be <u>more pessimistic</u> than the simple Bohm prediction. Since the Bohm formula can be identified theoretically as a kind of thermodynamic upper limit for anomalous cross-field transport in MHD-stable configurations, a plausible conclusion is that Eq. (7) can be valid only for $\chi^*_B \leq \chi_{Bohm}$, and should be replaced by the Bohm formula itself when $B_t/B_p$ is larger than 30.

In seeking to relate Eq. (7) to tokamaks, one notes, first of all, that the plasma parameters are greatly different. Since tokamak plasmas must carry their own "poloidal-field current," there is a minimum condition of the form $\sim 300 < v_{th}/v_s \propto nT_e^{1/2}/J$, which calls for typical plasma parameters that are far from those of Fig. 4. [This point is particularly well documented in Ref. 19, which finds that the addition of as little as 100 A of toroidal plasma current to a spherator plasma with 100 kA of ring current introduces a drastic (1/n)-dependent deterioration of confinement.] The requirement that tokamak plasmas must have certain minimum levels of density and temperature entails a second fundamental difference relative to the FM-1 plasmas: In tokamaks, the neutral density is typically "burned out" to a high degree within the plasma core, whereas the plasmas of Fig. 4 are weakly ionized in the low-$T_e$ regime and incompletely burned out even in the highest-$T_e$ experiments.

If we proceed, nonetheless, to make predictions for the tokamak on the basis of Eq. (7), we find, for representative TFTR parameters[20] ($B_p \sim 5$ kΓ, $B_t \sim 50$ kΓ, $T_e \sim 3$ keV), that the global energy confinement time is substantially underestimated. In modern tokamaks, where diagnostics are sufficiently good so that transport rates can be determined as a function of plasma radius, one also concludes that $\chi_e$ must increase strongly towards the plasma edge -- just the opposite of the trend that would be indicated by Eq. (7). In the cold edge region of TFTR ($T_e \sim$ 100-300 eV), Eq. (7) gives roughly the right answer, but the hot plasma core seems to be governed by different and far more favorable rules.

In summary, we see that spherator data did demonstrate strong architectural effects on confinement, more or less according to theoretical prejudice. That the early stellarator confinement results should have been very poor, is consistent with the geometric dependences inferred from the spherator experiments -- indeed one sees that the Bohm formula must represent a kind of benign upper limit on transport that kept Model-C confinement from being even worse than it was. The tokamak results, on the other hand, represent a clear-cut challenge to the notion that architectural optimization is important: Both "architectural theory" and the spherator studies themselves clearly show that the tokamak-type magnetic-field configuration is geometrically inferior -- whereas actual tokamak confinement is found to be dramatically better than the predictions of Bohm or neo-Bohm scaling. Clearly some non-architectural feature of the tokamak plasma regime must be exerting a *favorable effect that far exceeds the influence of the purely geometric* aspects of toroidal confinement.

## 3. Tokamak Experiments

The success of the T-3 tokamak served to advance $T_i n\tau_E$ to about $5 \times 10^{11}$ keV $cm^{-3}$ sec in 1969, using a facility of no greater magnitude than had been available in earlier fusion experiments, which only reached the $10^9$ - $10^{10}$ keV $cm^{-3}$ sec range. Subsequent years brought continuing progress in the quality of confinement: For example, during the late 70's and early 80's, Alcator A and C, which again were facilities comparable in scale to T-3, achieved $T_i n\tau_E$-values[21,22] of about $3 \times 10^{13}$ and $10^{14}$ keV $cm^{-3}$ sec, respectively, using ohmic heating at relatively high toroidal magnetic fields and densities.

These experiments pointed to the "neo-Alcator" scaling law

$$\tau_E \propto nqR^2a \quad , \tag{8}$$

where q is the "MHD safety factor" $2\pi/\iota \approx B_t a/B_p R$, and R and a are the major and minor plasma radii. Equation (8) successfully predicted the confinement scaling observed in the much larger TFTR device at low-to-moderate plasma densities (Fig. 8). Fairly high central densities up to $4 \times 10^{14}$ $cm^{-3}$, have proved to be achievable in TFTR by means of pellet injection,[23] but the favorable n-dependence of Eq. (8) appears to saturate at high densities (cf. Fig. 8).

One interpretation of these ohmic-heating results is that the low-n regime corresponds to the appearance of some extraneous energy-loss channel, similar to the adverse high-$v_s/v_{th}$ dependence reported for spherator plasmas in Ref. 19 -- though unlikely to be attributable to the same physical mechanism. The high-n portion of the ohmic data in Fig. 8 would then constitute the "normal" confinement regime of the tokamak and would be expected to conform with the general scaling observed for plasmas heated and fueled by a variety of techniques.

Goldston[24] has pointed out that this "normal" tokamak scaling can be approximated by

$$\tau_G \propto I \, L^{3/2} \, P_H^{-1/2} \tag{9}$$

or

$$\chi_G \propto P_H^{1/2} \, L^{-1/2} \, B_p^{-1} \quad , \tag{10}$$

where L represents linear size (R, a, $L_s$, etc.) and $P_H$ is the total plasma-heating power. The tokamak ohmic-heating regime has the special advantage that $P_H$ can be raised only by raising I. Pure ohmic heating generally gives the minimal level of $P_H$ for any given I, so that ohmic $\tau_E$-values must be optimal: As described by Eq. (9), they actually tend to "improve with rising $P_H$" (and I), unlike the typical auxiliary-heating results of Fig. 9, where $\tau_E$ deteriorates with rising $P_H$ (for fixed I).

Unhappily, the tokamak reactor regime calls for α-heating powers that greatly exceed the ohmic power level: For economically attractive reactor parameters, one typically requires $V_H \equiv P_H/I \sim (750\ \text{MW})/(15\ \text{MA}) \sim 50$ v -- in marked contrast with the ohmic regime, where $V_H \lesssim 1$ v. The development of practical tokamak reactors depends on the optimization of the high-$V_H$ plasma regime, so as to achieve somewhat more favorable confinement than predicted by the basic Goldston "L-mode" scaling of Ref. 24.

A major step of this kind was the discovery of the "H-mode" in the ASDEX tokamak.[25] Neutral-beam heating in ordinary L-mode operation typically depresses $\tau_E$ relative to the pure ohmic-heating case by some factor of order $(P_H/P_{OH})^{1/2}$. When a poloidal divertor separatrix is present, the thermal equilibrium of the tokamak discharge is found to become bi-stable at sufficiently high $P_H$, with a more favorable uppper branch (the H-mode). Recent results of this type, obtained in the DIII-D tokamak,[26] are shown in Fig. 10. The available range of heating powers in DIII-D has been insufficient, thus far, to determine whether entry into the H-mode improves Eq. (9) by a constant multiplicative factor of 2-3, or whether the adverse $P_H^{-1/2}$-dependence is fundamentally changed. In any case, the H-mode branch is evidently superior to the ohmic-heating regime of DIII-D, since the same $\tau_E$ is maintained at much higher $V_H$.

The happy discovery of the H-mode has served, incidentally, to rekindle interest in the architectural approach to confinement optimization. The presence of a poloidal-field separatrix creates a thin region of enhanced global shear, which also tends to be a region of improved magnetic well, provided that the null points of the separatrix are not located on the large-R side of the plasma. The essential improvements associated with the H-mode are, in fact, found to occur at the separatrix: Both D and $\chi$ undergo pronounced local decreases, thus permitting relatively high values of $n_{edge}$ and $T_{edge}$, with resultant increments of $n_{core}$ and $T_{core}$ relative to the L-mode results obtained at the same $P_H$.

While the strong influence of edge-localized phenomena on global confinement time may be surprising at first sight, it is consistent with a wide range of other "tokamak anomalies." Particularly relevant is the finding that the characteristic $T_e(r)$-profile shape in a given L- or H-mode regime with fixed values of q(o) and q(a) cannot be changed readily even by drastic changes in the profile of heating-power deposition.[27,28] This observation implies[29,30] that

$$\tau_E \propto (a^2/\chi_{edge})\, n_{core}/n_{edge} \quad , \tag{11}$$

where the "plasma edge," defined as the maximum radius within which the power deposition profile can be varied without producing much effect on $\tau_E$, turns out to be fairly close to a.

A different kind of "enhanced confinement regime" has been obtained with intense neutral-beam heating in the absence of a divertor separatrix in TFTR[20] (Fig. 11). Again, there is an improvement of $\tau_E$ by factors of 2-3 relative to L-mode prediction. The "best data points," corresponding to balanced injection of the tangential neutral beams (so as to avoid driving a rapid toroidal plasma rotation), show little evidence of any unfavorable $P_H$-dependence. As in the case of the H-mode, the key to this "supershot" regime is an edge effect: the suppression of recycling at the TFTR limiter by using specially conditioned hydrogen-absorbing graphite tiles. The central particle-fuelling associated with neutral-beam injection is then able to produce n(r) profiles with large values of $n_{core}/n_{edge}$, along with an improvement in global confinement, as predicted by Eq. (11). The success of pellet-injection experiments in Alcator C,[22] TFTR,[23] and ASDEX[31] seems to be due to the same phenomenon.

The TFTR supershot and pellet-injection results, along with initial H-mode results in JET,[32] have pushed the frontier of fusion research to the line marked $T_i n\tau_E \sim 2 \times 10^{14}$ keV cm$^{-3}$ sec in Fig. 1. There is every reason to believe that significant further advances will occur during the coming year. Break-even plasmas seem likely to be attainable in TFTR, JET, JT-60,[33] and other large tokamaks -- particularly since the relevant break-even condition in energetic-ion-heated plasmas is somewhat relaxed by comparison with the Lawson curve.[34] If an "enhanced confinement" factor of at least 1.5, relative to the L-mode, can be obtained in the proposed next-generation CIT experiment,[35] the empirical extrapolation to ignition and equilibrium-burn conditions would seem to be favorable.

In this context of programmatic success, it is sobering to reflect that identification of the specific physical phenomena responsible for anomalous transport in toroidal configurations has not progressed decisively since 1956. The physical understanding of Bohmlike transport has never reached the stage of clear, detailed correspondence between experiment and theory The tokamak confronts us with the additional mystery of a low-level transport mechanism within the plasma core that serves to maintain the $J(r)$-$T_e(r)$ profile shape. There is some satisfaction in noting that the $J(r)$-constraints prescribed by resistive MHD theory[36] seem to be respected by the experimental plasma -- but the transport mechanism whereby the tokamak selects a stable profile has yet to be identified.

## 4. Comparison of Spherator and Tokamak Results

During the past two decades, a great deal of detailed tokamak experimental data has accumulated. The translation of data into physical understanding is hampered, however, by the narrowly limited range of architectural and plasma parameters accessible within the inherent tokamak constraints: *The ratio $B_p/B_t$ is limited by kink stability, while $B_v/B_p$ is* predetermined by the plasma equilibrium in R; the magnitude of J follows from $B_p$ and in turn imposes limits on n and $T_e$.

By way of contrast, in the spherator device discussed in Sec. 2, the parameters $B_t$, $B_p$, $B_v$, J, n, and $T_e$ can be varied essentially independently. An appropriately designed spherator could produce an edge-plasma region that would be effectively identical with the tokamak edge plasma -- and the spherator could then carry out significant parameter variations. Diversion of about one percent of the world tokamak effort to such a study might prove both relevant and enlightening to tokamak research. In the absence of a contemporary spherator project, we may still hope to draw useful inferences from the "fossil" data deposits of the 70's.

Why did the tokamak emerge as a markedly superior confinement device in the late 60's? The answer seems to be that, while transport in the tokamak edge plasma is actually rather worse than in an architecturally optimal spherator configuration, the tokamak has a relatively well-confined plasma core -- while spherator (and early stellarator) plasmas in some sense were "all edge." In this view, the only architectural merit of the T-3 tokamak was its simplicity, which allowed plasmas of relatively large minor radius and high field strength (high plasma density) to be reached with a device of moderate cost. This conjecture is strengthened by the success of modern stellarators in achieving tokamaklike quality of confinement: larger, denser stellarator plasmas exhibit weaker anomalous transport -- which is found to resemble tokamak transport in peaking at the plasma edge.[37] (The objection that reversed-field pinches have long had "large dense plasmas," without encountering notably good confinement, can be set aside on the basis that RFP plasmas suffer from an additional energy loss channel: they are not truly MHD-quiescent.)

In this spirit, the neo-Alcator scaling law of Eq. (8) can be seen as representing an increase in the effectiveness of "tokamak plasma-core shielding" as a function of density and size. If we adopt a simple model where $\chi_{edge}$ is given by Eq. (7) and the width of the edge region $\Delta r_{edge} = a - r_{edge}$ is inversely proportional to $n_{edge}$, the result is

$$\tau_E \propto (n_{core} L^3)(B_p/T_{core}). \tag{12}$$

Noting that $B_p/T_e$ is roughly constant in the data base for ohmic-heated tokamaks, Eq. (12) conforms reasonably well with the neo-Alcator scaling. The ad hoc model $\Delta r_{edge} \propto n_{edge}^{-1}$ used above corresponds, for example, to the conjecture that the plasma core should be protected against penetration by neutral atoms -- which might help to generate surface noise by irregular deposition.[38] More generally, one would expect the singularity of d log T/dr and/or d log n/dr at the plasma edge to drive a variety of noisy phenomena. A specific physical argument in terms of the surface-noise concept has been made by Kadomtsev [39] (who draws the pleasing analogy between anomalous tokamak transport and the turbulence excited by fluid flow through a rusty pipe).

Tokamak confinement data such as that in Fig. 8 indicate that, for sufficiently high plasma density and size, there is no further benefit in raising the ratio $L/\Delta r_{edge}$ -- presumably because a more fundamental trouble is encountered. In this "normal" tokamak regime, we have seen (Eq. 11) that maximizing the ratio $n_{core}/n_{edge}$ becomes advantageous for increasing the global $\tau_E$.

In summary, the identification of the spherator plasmas of Sec. 2 with the edge plasmas of modern tokamaks fits in rather neatly with prejudices about tokamak confinement that have evolved in recent years.[24,29,38] If the confinement of these low-Ln plasmas can indeed be characterized in terms of some pervasive form of anomalous transport, we should now inquire as to its nature and parameter dependence.

The simplified account given above has failed to distinguish between the edge heat outflows due to convection ($TD\nabla T$), and conduction ($n\chi\nabla T$), or between the electron and ion channels of heat conduction. Equilibration is sufficiently rapid in the edge region of large tokamaks so that the relative importance of the ion and electron channels is not easy to resolve. In tokamaks, as in other toroidal-confinement systems, the particle transport coefficient D is generally smaller than $\chi$, but since convection is enhanced by recycling at the plasma edge, the corrective term cannot normally be neglected in the local heat-flow balance.

While keeping these complexities in mind, we can think in terms of an "effective $\chi_{edge}$" and ask how it scales in various cases. On the basis of the spherator results, $\chi$ should follow the "neo-Bohm" scaling of Eq. (7) in tokamaklike geometry. Since Eq. (7) refers to the average global $\chi \propto L^2 \tau_E^{-1}$, we can eliminate the explicit dependence on T by means of $T = P_H \chi^{-1} n^{-1} L^{-1}$, obtaining

$$\chi^*_B \propto P_H^{1/2} L^{-1/2} B_p^{-1/2} n^{-1/2}. \tag{13}$$

[This equivalent global expression for neo-Bohm scaling has the incidental merit of offering a much more plausible model for the local $\chi(r)$.] There does seem to be a marked similarity between the Goldston scaling of Eq. (10) and the neo-Bohm scaling of Eq. (13):

$$\chi_G/\chi^*_B \propto B_p^{-1/2} n^{1/2}. \tag{14}$$

Since the tokamak data base shows a strong correlation between high-$B_p$ and high-n points, the apparent discrepancy factor in Eq. (14) is never very large.

## 5. Summary and Conclusions

What explanation can we offer for the major historical event in toroidal confinement research -- the advance from Bohmlike transport in early stellarators and spherators to superior confinement in the T-3 tokamak? An explanation in terms of better magnetic architecture is irreconcilable with the spherator data. The most plausible remaining hypothesis is that Bohmlike transport is driven by edge effects (cf. Kadomtsev's "flow through a rusty pipe") and that the real innovation of the T-3 was to provide a thick, dense, MHD-stable plasma. The tokamak edge plasma actually bears a strong resemblance to the smaller, lower-density toroidal plasmas of earlier times and seems to exhibit similar anomalous transport. The favorable nL-dependence of neo-Alcator scaling can be seen as resulting from "better shielding," i.e., reduced relative size of the edge-plasma region ($\Delta r_{edge}/L$).

The normal high-density tokamak regime is characterized by: (1) a thick plasma-core region, with a rigid $T_e(r)$-profile shape that is maintained by somewhat mysterious but essentially benign transport processes; (2) a thin Bohmlike outer region that calibrates $dT_e/dr$ in terms of the local $\chi_{edge}$ and the global heat throughput $P_H$, controlling the magnitude of the global energy confinement time according to some model of the form

$$\tau_E \approx (L^2/\chi_{edge})\, n_{core}/n_{edge}$$

$$\chi_{edge} = G_{edge}\, P_H{}^{1/2}\, L^{-1/2}\, B_p{}^{-1/2}\, n^{-1/2}. \qquad (15)$$

The quantity $G_{edge}$ refers to geometric dependences: If the tokamak edge plasma resembles the spherator plasma, then one would expect reductions in $G_{edge}$ to result from higher edge shear, improved location of particle trapping and weaker adverse magnetic curvature.

A more general lesson can also be drawn from the historical experience of toroidal confinement research: During the first part of the 60's, Bohm diffusion was seen as imposing a prohibitive upper limit on confinement -- but as of the late 60's, the prevalence of Bohm diffusion has seemed more likely to reflect a thermodynamic lower limit for confinement in MHD-stable plasmas. In the same way, the emergence of Goldston L-mode scaling during the early 80's was taken to impose a somewhat painful constraint on the prospects of tokamak confinement. Once again, the more likely significance of Goldston scaling is now seen to be that it sets a lower limit for global energy confinement in MHD-stable high-density tokamak plasmas. Improvements in the architecture of the tokamak plasma edge have already served to decrease $G_{edge}$ substantially in H-mode operation. Increases in the factor $n_{core}/n_{edge}$ have yielded similar benefits for TFTR supershots.

Figure 1 illustrates that toroidal-confinement research, and particularly tokamak research, has made substantial progress towards its goal -- even without the guiding light of a true physical understanding. The potential rewards for a rigorously scientific optimization of tokamak confinement, however, remain very large: Achieving ignition at a minimal plasma-current level will be the key to cost-effective steady-state operation (i.e., with noninductive current-drive); moderate reactor unit size and cost is a precondition for timely progress through the developmental phase of fusion power. One good way to achieve the desired quality of insight into the tokamak will be to pursue the search for a consistent overall physics of plasma transport in toroidal geometry.

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## Table I

## Search for an Ideal Architectural Solution

|  | No $J_{\parallel}$ | Strong Shear | Magnetic Well | No Unfavorable Trapping |
|---|---|---|---|---|
| **1. Best Possible Configuration (incompatible with D-T)** |  |  |  |  |
| Multipole | ✓ | -- | ✓ | ✓ |
| Multipole with $B_t$ | ✓ | ✓ | ✓ | ✓ |
| Spherator | ✓ | ✓ | ✓ | ✓ |
| **2. Ideal Reactor-Compatible Solution** |  |  |  |  |
| Heliac | ✓ | ✓ | ✓ | -- |
| **3. Nonideal But Simple Architecture** |  |  |  |  |
| Tokamak | -- | -- | ✓ | -- |

## Figure Captions

1. Progress of toroidal-confinement research in the Lawson diagram.
2. Three basic types of axisymmetric toroidal-confinement geometry: (a) the levitron, where $B_p$ is generated by a ring current and $B_t$ by external coils; (b) the spherator, where a weak external $B_v$ is added, giving a tokamaklike configuration on the outer plasma surface away from the ring; (c) a spherator with $B_v \sim B_p$ and $B_t \lesssim B_p$, where the mirror-trapping region moves to the small-R side of the plasma.
3. The FM-1 spherator.
4. Particle confinement in FM-1 (and its smaller prototype LSP).
5. Average global transport coefficients in a tokamaklike FM-1 plasma. (The higher set of points is the thermal conductivity K or $\chi$.)
6. Shear-dependence of transport coefficients in low-$B_t$ FM-1 regimes. (The quantity "TF" refers to the ratio of total TF-coil current to ring current $I_T/I_P$.)
7. Survey of geometric dependences of FM-1 confinement. The parameter $I_T/I_P$ corresponds to roughly 3 times the "average $B_t/B_p$." The quantity $I_P/I_E$ measures the ratio of the ring current to the current producing the external vertical field.
8. Dependence of energy confinement on the neo-Alcator scaling parameter in TFTR ohmic-heating regimes.
9. Dependence of total stored plasma energy $W_{tot}$ on total input power $P_{tot}$ in TFTR neutral-beam-heating experiments. The ratio $\tau_E = W_{tot}/P_{tot}$ is seen to decrease with rising $P_{tot}$, consistent with the Goldston L-mode prediction.

10. Scaling of energy confinement time with $P_{tot}$ in DIII-D neutral-beam-heating experiments. The L-mode plasmas follow a Goldston-type scaling. For $P_{tot} \gtrsim 3$ MW, an H-mode branch appears.
11. Scaling of energy confinement time with neutral-beam power in the TFTR supershot regime. Beams are injected tangentially in the "co" and "counter" directions relative to the plasma current, which is in the range 0.9-1.0 MA.

#87A0049

LAWSON DIAGRAM

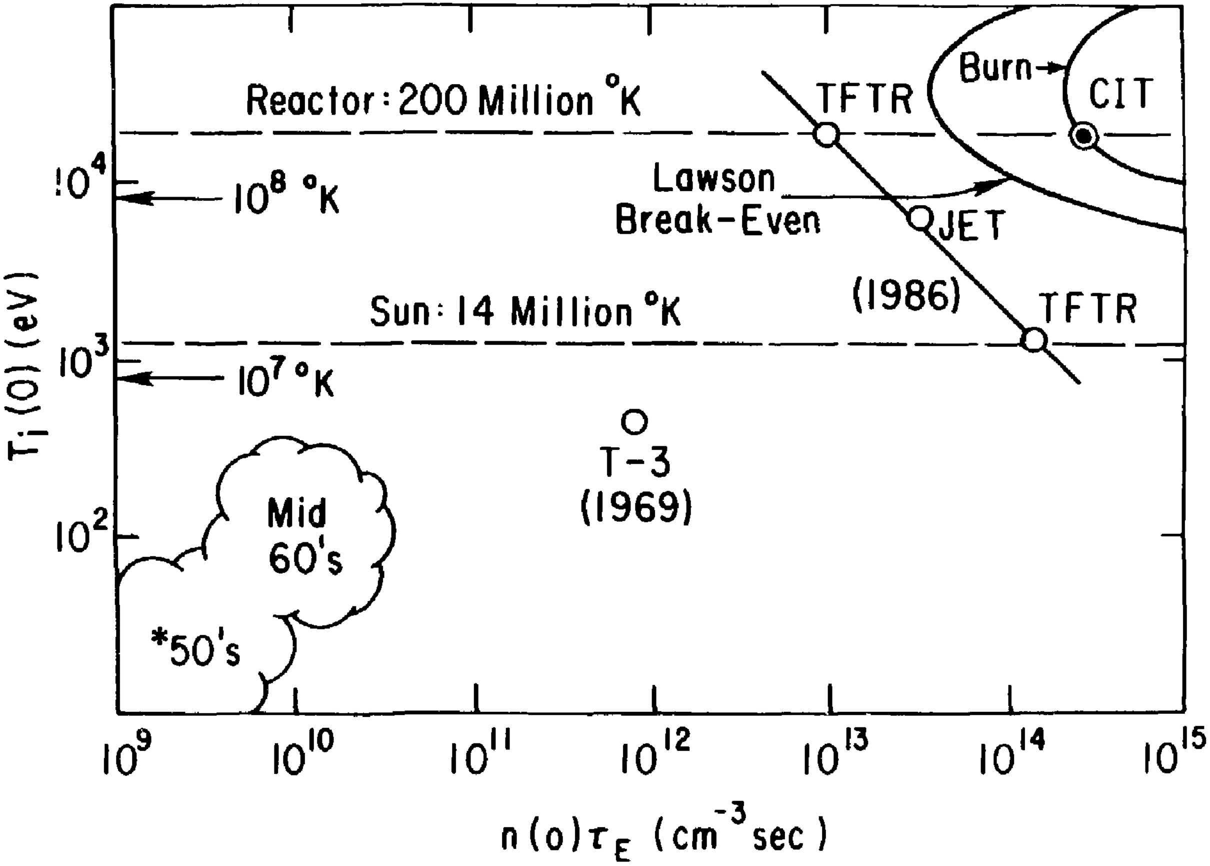

Fig. 1

#87A0024

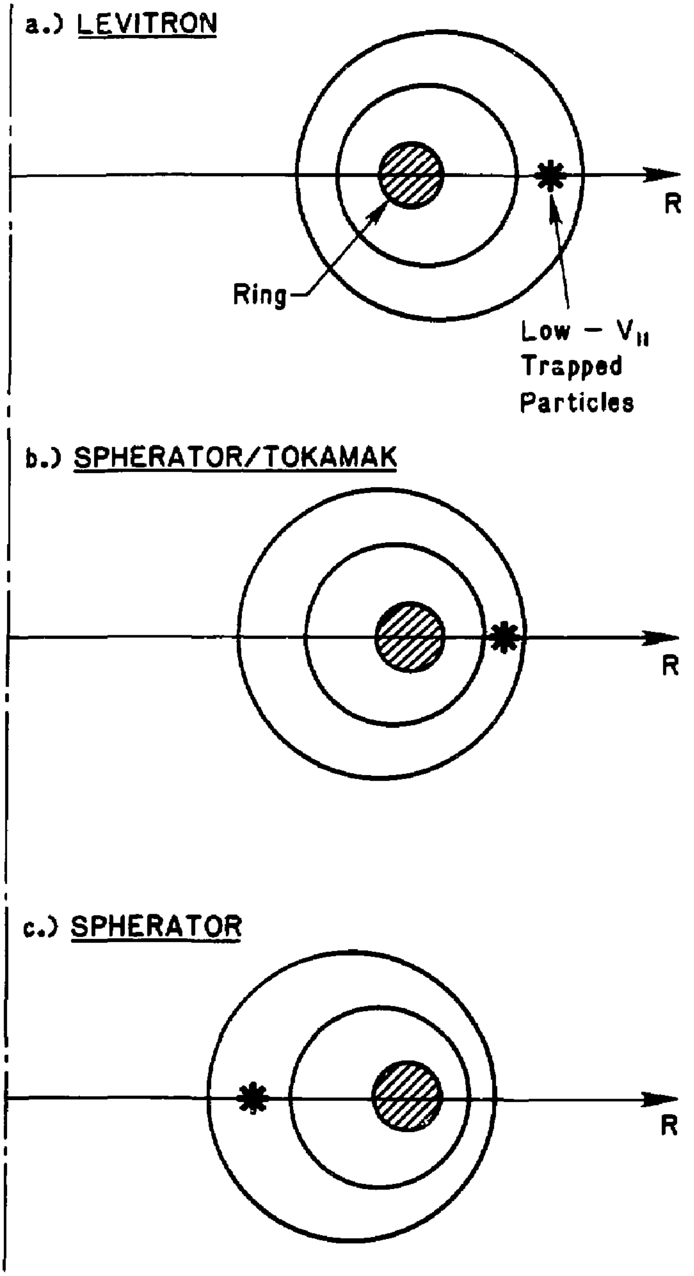

Fig. 2

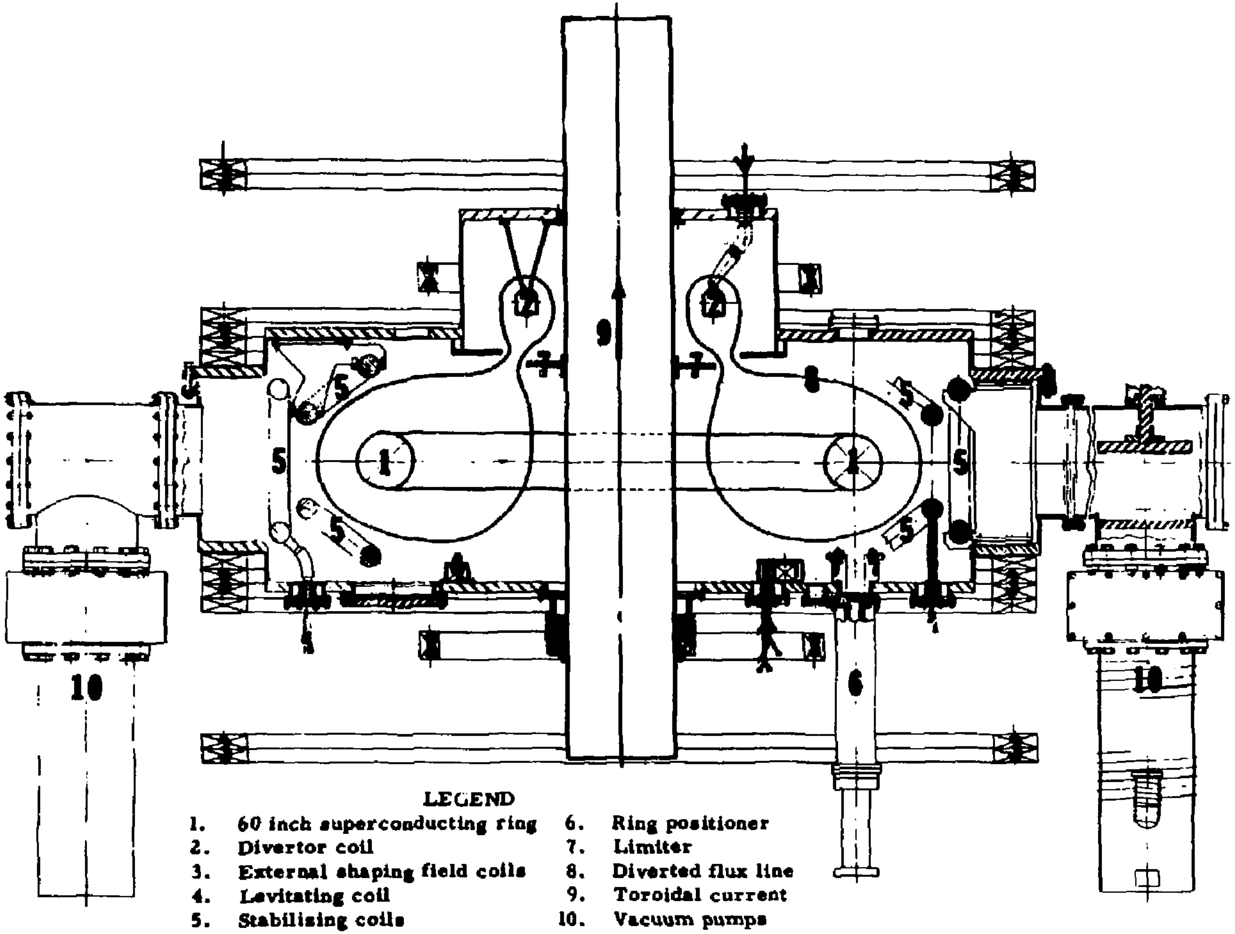
1
1
2
2
5
5
5
5
5
5
7
7
8
9
10
10
6
LEGEND
1. 60 inch superconducting ring
2. Divertor coil
3. External shaping field coils
4. Levitating coil
5. Stabilizing coils
6. Ring positioner
7. Limiter
8. Diverted flux line
9. Toroidal current
10. Vacuum pumps

Fig. 3

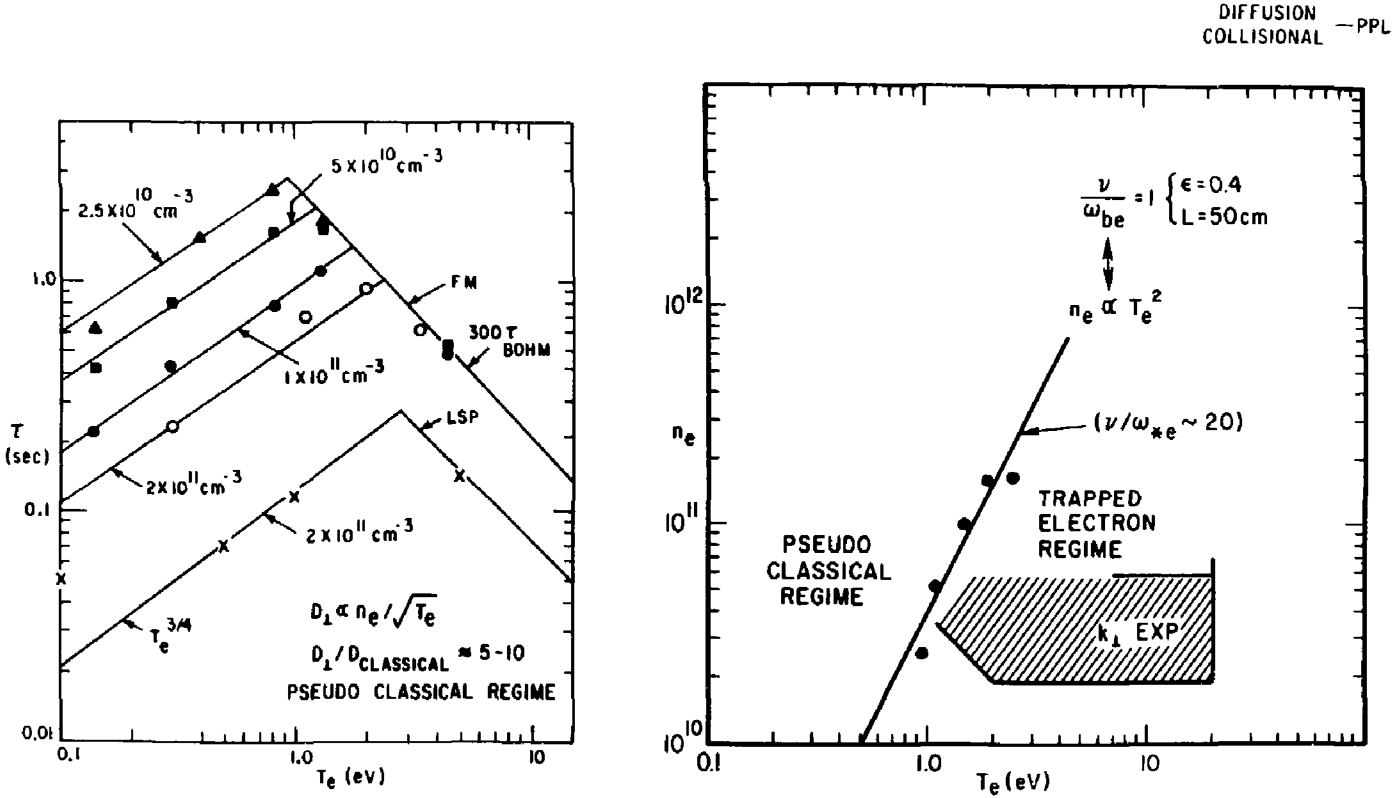
DIFFUSION
COLLISIONAL —PPL
τ (sec)
1.0
0.1
0.01
0.1
1.0
10
$T_e$ (ev)
$2.5 \times 10^{10}$ cm$^{-3}$
$5 \times 10^{10}$ cm$^{-3}$
$1 \times 10^{11}$ cm$^{-3}$
$2 \times 10^{11}$ cm$^{-3}$
$2 \times 10^{11}$ cm$^{-3}$
FM
300 $\tau_{BOHM}$
LSP
$T_e^{3/4}$
$D_\perp \propto n_e/\sqrt{T_e}$
$D_\perp/D_{CLASSICAL} \approx 5-10$
PSEUDO CLASSICAL REGIME
$n_e$
$10^{12}$
$10^{11}$
$10^{10}$
0.1
1.0
10
$T_e$ (eV)
$\frac{\nu}{\omega_{be}} = 1 \begin{cases} \epsilon = 0.4 \\ L = 50\text{cm} \end{cases}$
$n_e \propto T_e^2$
$(\nu/\omega_{*e} \sim 20)$
TRAPPED ELECTRON REGIME
PSEUDO CLASSICAL REGIME
$k_\perp$ EXP

Fig. 4

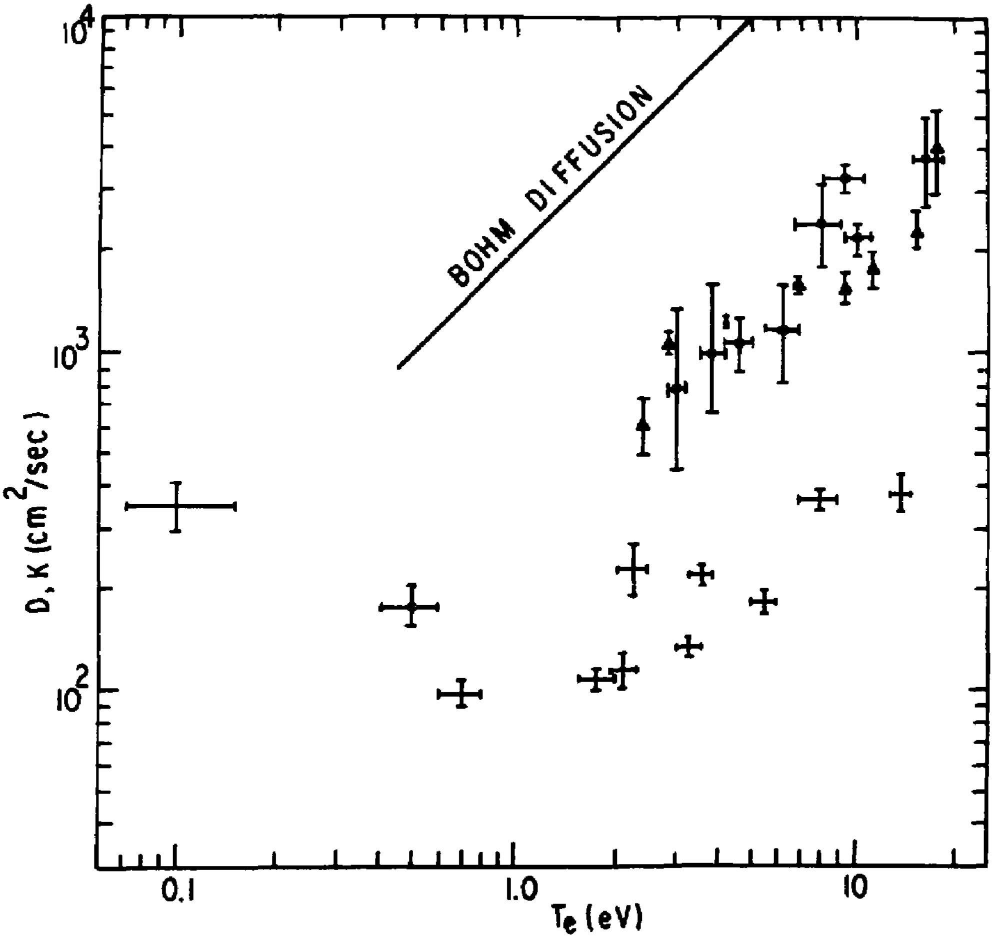

Fig. 5

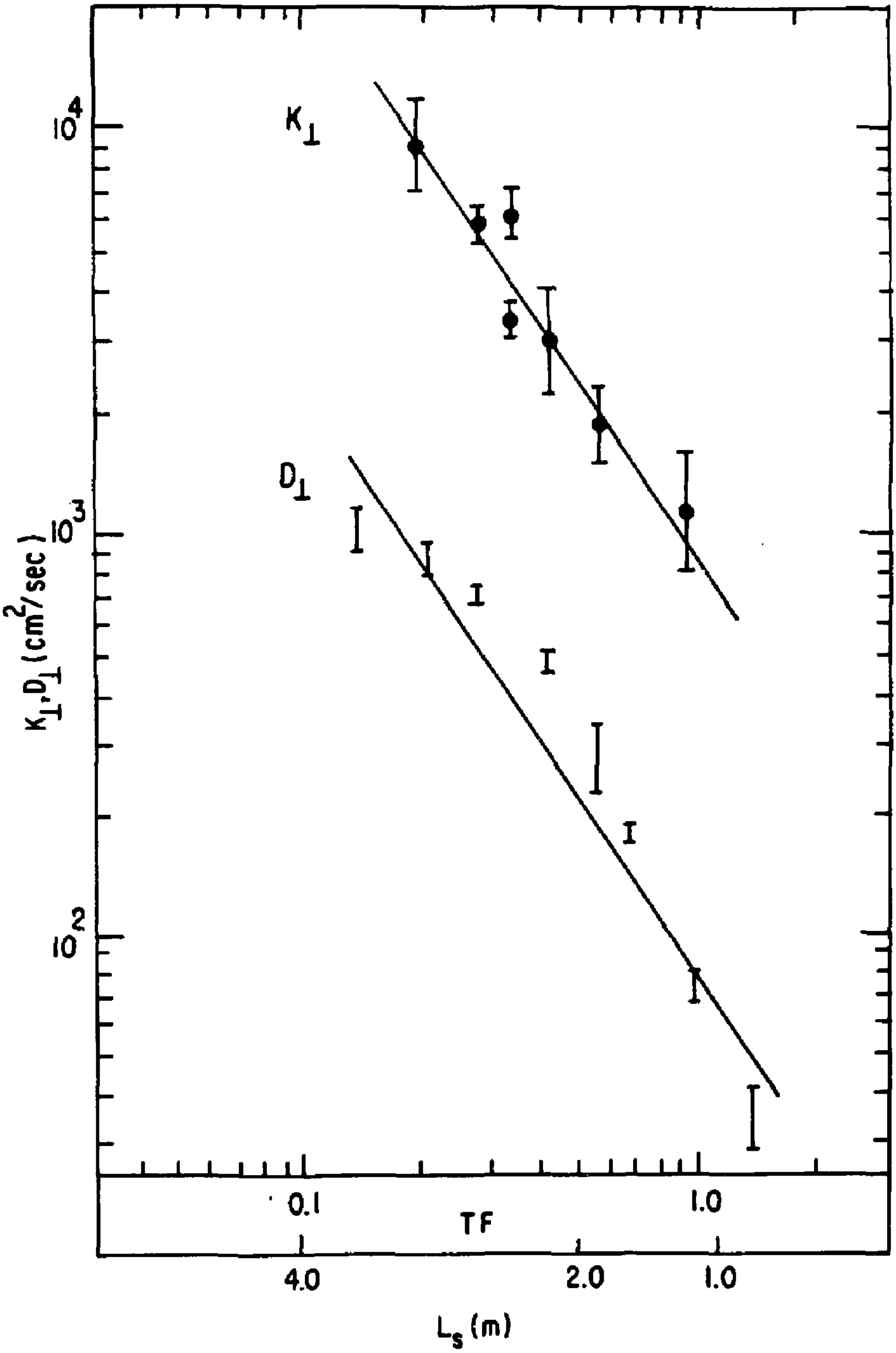
$10^4$
$10^3$
$10^2$
$K_\perp$
$D_\perp$
$K_\perp, D_\perp$ (cm$^2$/sec)
0.1
1.0
TF
4.0
2.0
1.0
$L_s$ (m)

Fig. 6

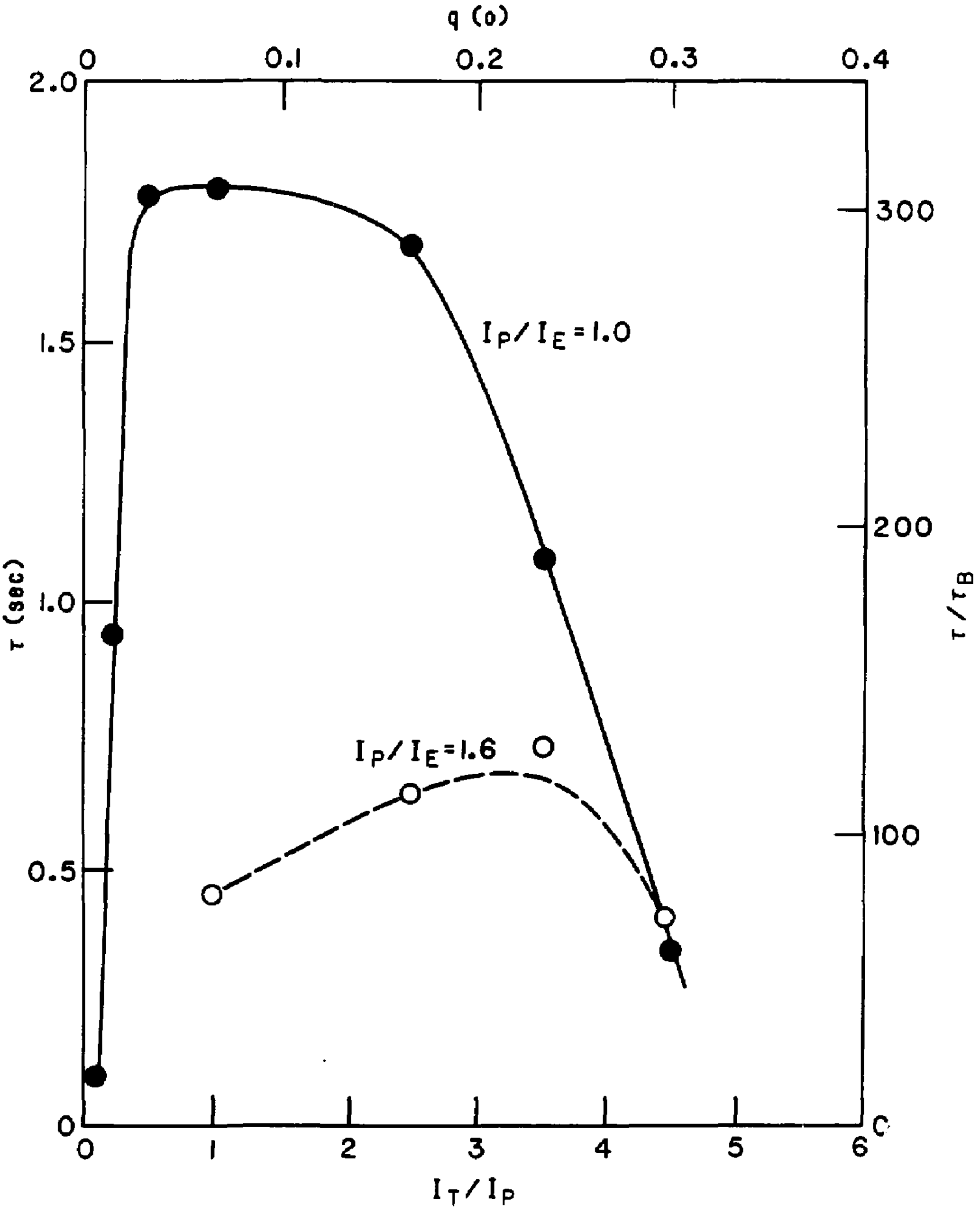
q (0)
0
0.1
0.2
0.3
0.4
2.0
1.5
1.0
0.5
0
τ (sec)
300
200
100
0
τ/τ_B
I_P/I_E=1.0
I_P/I_E=1.6
0
1
2
3
4
5
6
I_T/I_P

Fig. 7

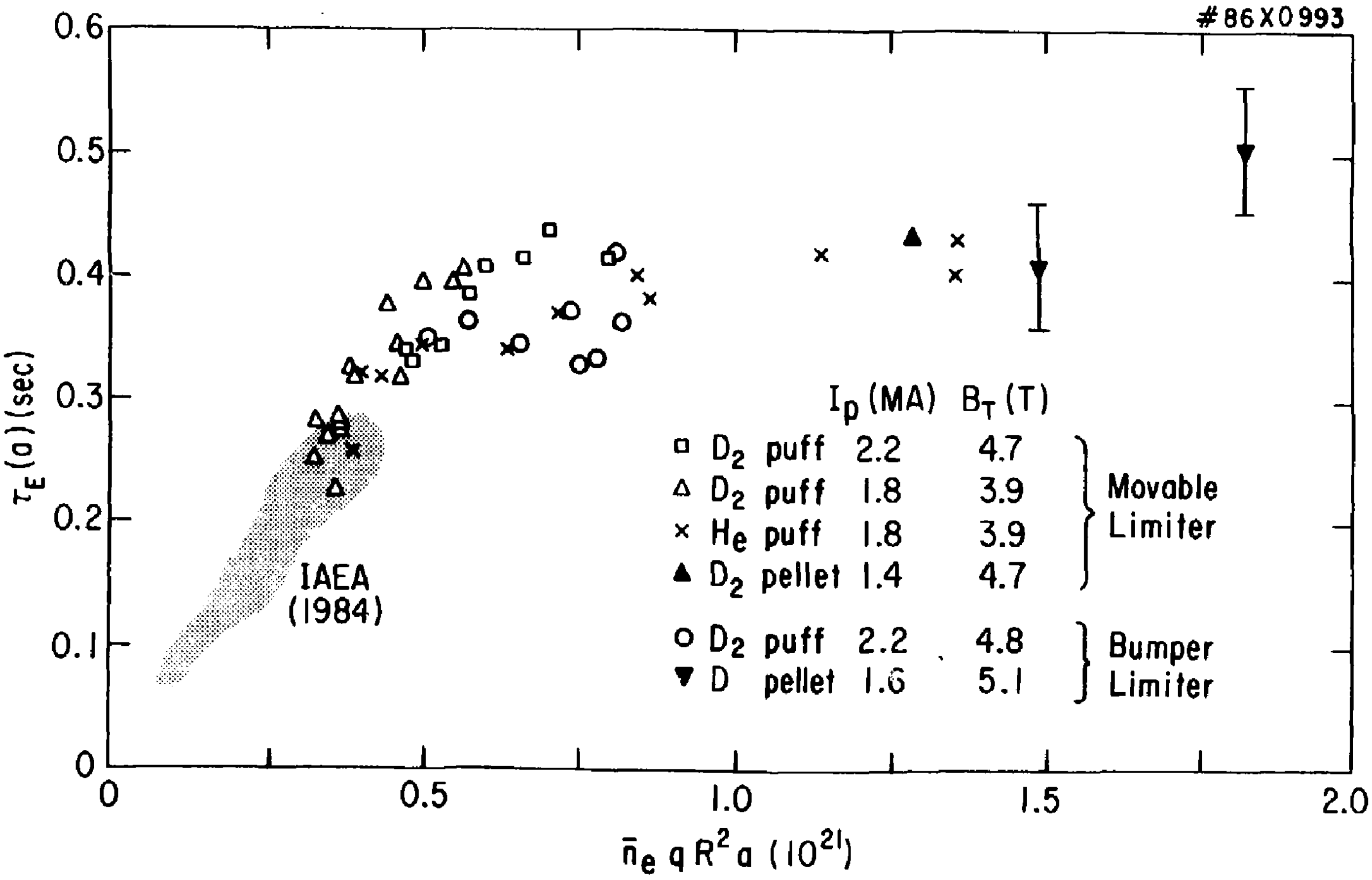
#86X0993
$\tau_E(a)$ (sec)
0.6
0.5
0.4
0.3
0.2
0.1
0
0
0.5
1.0
1.5
2.0
$\bar{n}_e q R^2 a$ ($10^{21}$)
IAEA (1984)
$I_p$ (MA) $B_T$ (T)
□ $D_2$ puff 2.2 4.7
△ $D_2$ puff 1.8 3.9
× He puff 1.8 3.9
▲ $D_2$ pellet 1.4 4.7
Movable Limiter
○ $D_2$ puff 2.2 4.8
▼ D pellet 1.6 5.1
Bumper Limiter

Fig. 8

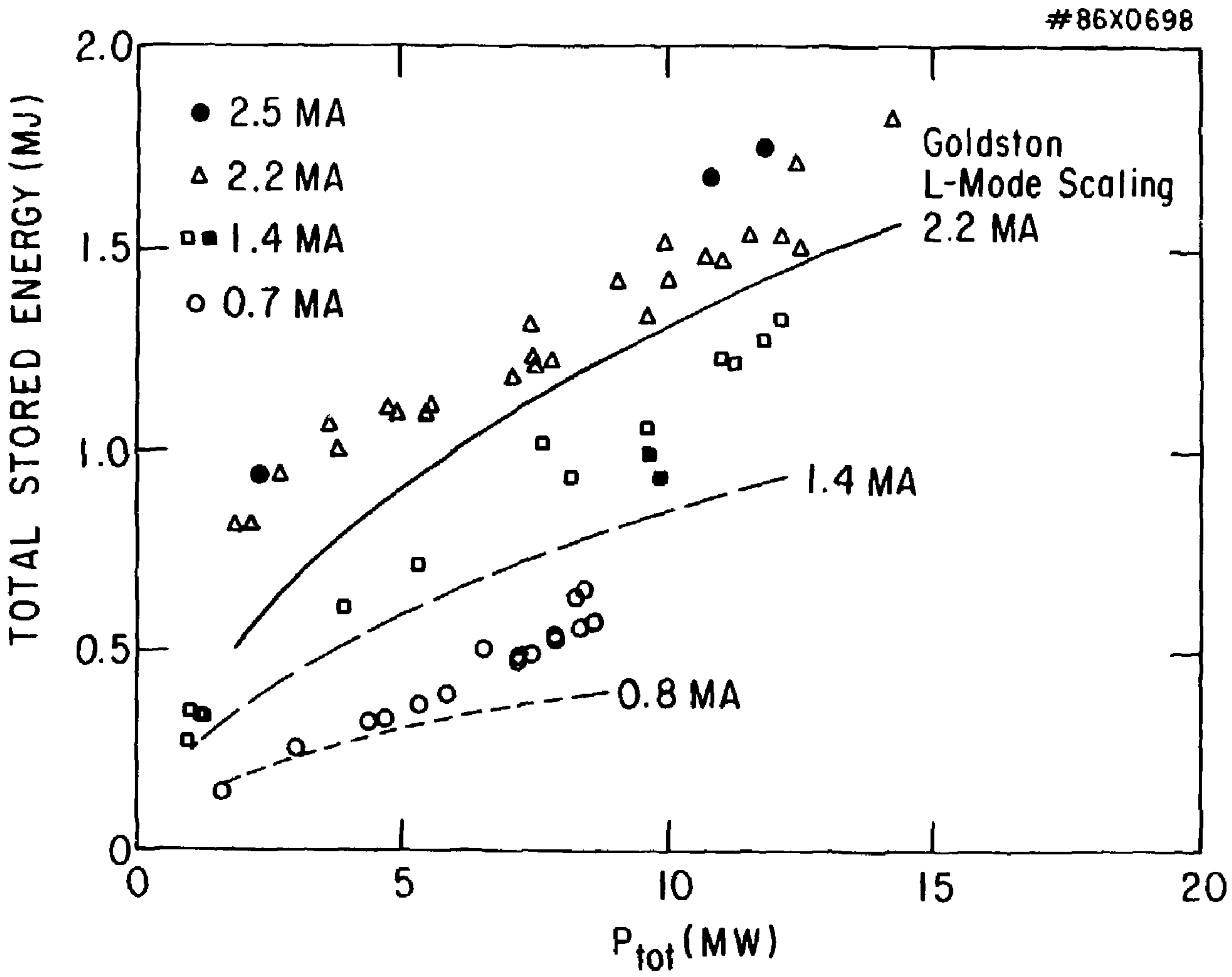
#86X0698
TOTAL STORED ENERGY (MJ)
2.0
1.5
1.0
0.5
0
● 2.5 MA
△ 2.2 MA
□ ■ 1.4 MA
○ 0.7 MA
Goldston
L-Mode Scaling
2.2 MA
1.4 MA
0.8 MA
0
5
10
15
20
P_tot (MW)

Fig. 9

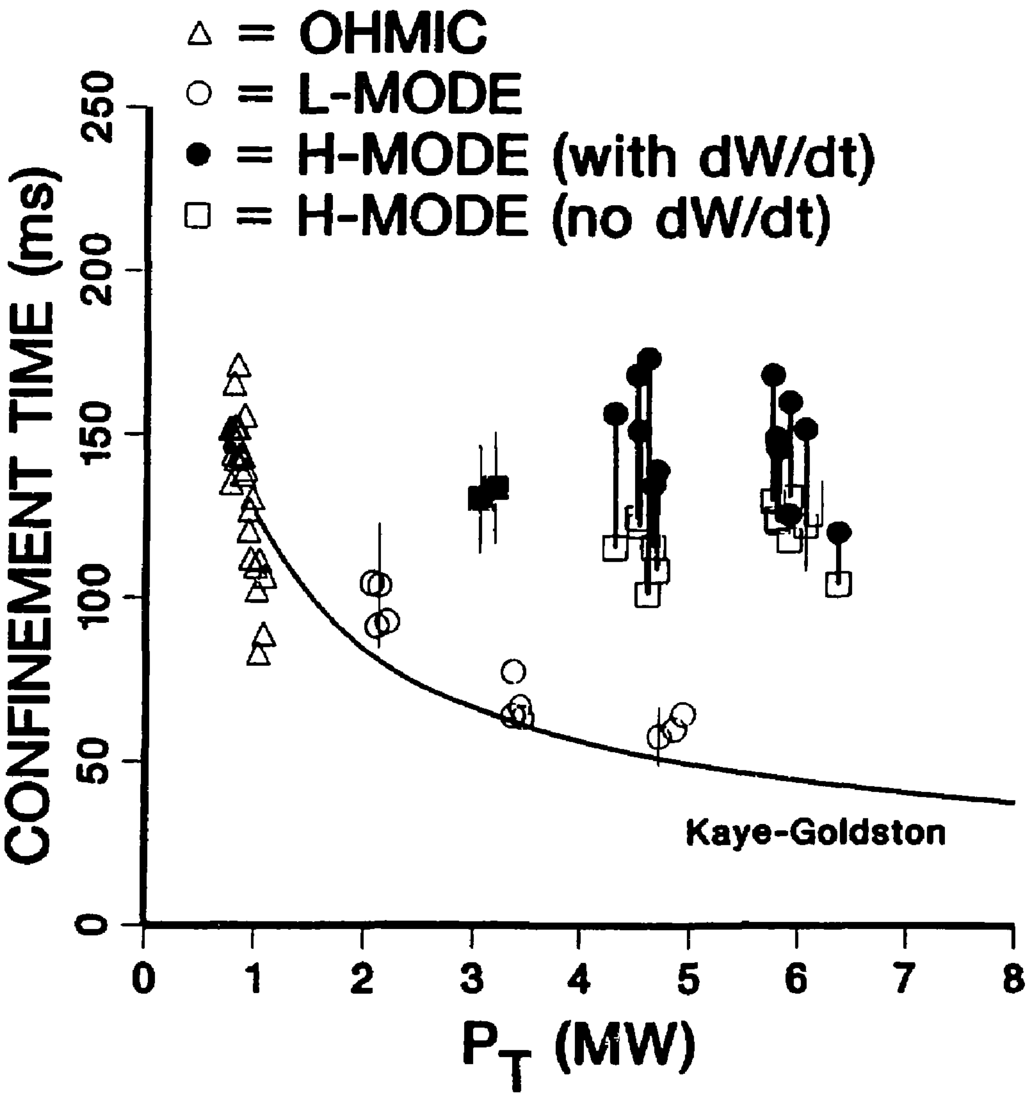
△ = OHMIC
○ = L-MODE
● = H-MODE (with dW/dt)
□ = H-MODE (no dW/dt)
CONFINEMENT TIME (ms)
0
50
100
150
200
250
$P_T$ (MW)
0
1
2
3
4
5
6
7
8
Kaye-Goldston

Fig. 10

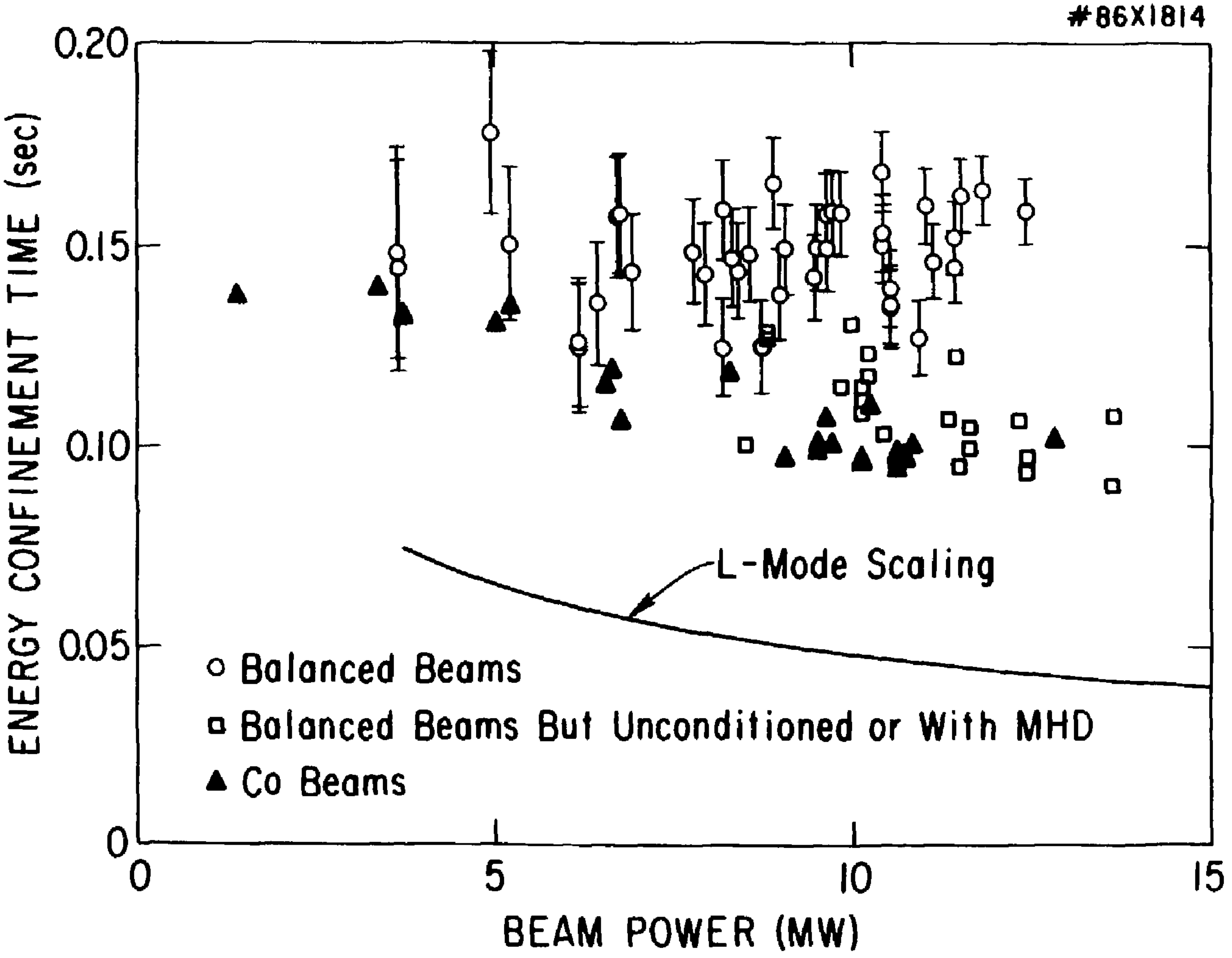
#86X1814
ENERGY CONFINEMENT TIME (sec)
0.20
0.15
0.10
0.05
0
BEAM POWER (MW)
0
5
10
15
L-Mode Scaling
○ Balanced Beams
□ Balanced Beams But Unconditioned or With MHD
▲ Co Beams

Fig. 11

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