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1. INTRODUCTION

There have been many important advances in the Tokamak field since the last IAEA Fusion Conference held in London in 1984.

A number of new devices have been brought into full operation ranging from JT-60 an advanced large tokamak from Japan, a country with a well established and mature fusion programme, to HL-1 a medium sized tokamak from the Republic of China, a country who's fusion programme has recently embarked on a considerable expansion. We congratulate our Chinese colleagues on having achieved full ohmic tokamak operation, with all the usual features and on providing a useful plasma for future research.

The tokamak has always been an enchanted system. Through the years it has proceeded from success to success despite the ineptness of experimentalists and in spite of the cleverness of theoreticians. It may be that the secret of this success has been that, so far, fate alone has had control of the current density distribution: in the tokamak, it is this current density distribution which controls the magnetic shear, the stability properties and indeed the form of the magnetic configuration itself.

Many physicists now believe that further advances depend upon securing external control of this vital feature. That is, in being able to impose a current density profile rather than having to accept the one given by inductive current drive and ill-defined wall-boundary conditions. The obvious approach is to replace at least some of the plasma current with non-inductively driven current, the difficulty is that the available methods are rather inefficient.

One of the most important results to have been achieved on JT-60 is the demonstration of Lower Hybrid Current Drive on a large tokamak system. This is the first demonstration on a large tokamak and our Japanese colleagues are to be congratulated in having achieved significant results with 1.2 MW of Lower Hybrid power at 2 GHz, driving up to 1.5 MA of current, with substantial current drive observed at mean densities up to 2 x 10^{19} m^{-3}. An important, hoped for, benefit of operation in a large tokamak configuration is improved current drive efficiency. JT-60 has observed a synergetic effect of neutral beam heating on lower hybrid current drive and in this configuration has obtained 0.7 MA of non-ohmic current driven by 0.7 MW of lower hybrid power, in the presence of 7 MW of beam heating (see table 1). This gives a figure of merit $I\bar{n}_e R/P_{LH}$ of 2.8 x 10^{19} A/m^2 W which, indeed, is some 3 times the value obtained in smaller experiments.

These results will give increased impetus to ohmic current replacement experiments and we can expect new results in time for the 1988 meeting.

2. SAWTOOTH BEHAVIOUR

Let us now look at some of the benefits which might come from establishing control over the current density profile. The most impressive results to date concern the internal disruption relaxation - the so called sawtooth oscillation.

This form of instability has been a feature of Tokamak operation since the very first experiments were carried out in the USSR in the late 1950s and early 1960s. Only very recently have cases been reported where this oscillation is stabilised for long periods.

Fig. 1 shows an example of spontaneous sawtooth stabilisation in JET. It is spontaneous in the sense that, while there is a well defined recipe for its production, there is no control of a directly relevant physical quantity (e.g. current density distribution). The diagram shows the electron temperature profile at a number of times, indicated in the insert, during a pulse exhibiting sawtooth behaviour. Also shown is a profile during the ohmic phase of the same discharge, when the sawtooth amplitude is very small. During the normal sawtooth behaviour with neutral beam heating only, the flat profile at the bottom of the sawtooth is some 0.5 keV higher than the ohmic profile. At the top of the sawtooth the profile has become much more peaked and the central temperature exceeds the ohmic value by 1.5 keV. The application of Ion Cyclotron Heating (ICH) heating induces a stabilization of the sawtooth behaviour which lasts for 0.9 s (even longer in other cases). For the first 0.3 s of this period the central electron temperature continues to increase, reaching 7.5 keV (5 keV more than the ohmic value).

The JET authors believe that the mechanism for the sawtooth behaviour is associated with the existence of a very flat current density profile inside the inversion radius. The application of non-ohmic heating is then supposed to broaden the current profile very slightly, taking q_o just above 1 to stabilise the mode. There is evidence for this behaviour in that during the flat top of the "monster sawtooth", a very low level of MHD activity is observed indicating that the m=1 mode is stabilised. Furthermore at the sawtooth crash which terminates the "monster", the inversion radius has moved outwards suggesting again that j(r) has broadened.

However, Faraday rotation measurements on Textor appear to demonstrate conclusively that there are at least some instances of sawtooth behaviour where q on axis is and remains significantly below 1. The sawtooth behaviour in these discharges is clearly not amenable to an explanation based on small changes of the axis q value from 1. The Julich group suggest, instead, that the behaviour is determined by the current density profile near the q = 1 surface, (which is typically in the steep part of the temperature profile, between centre and edge). In extreme cases the flattening of the profile near q = 1 to stabilise the m=1 mode is actually measured and the sawtooth relaxation is supposed to hold the profile near this marginally stable condition. The Julich Group expect this process to lead to a universal current density j(r) profile in terms of a suitable combination of external quantities (i.e. j in units of $B_T/\mu_o R$ and r in units of $(\mu_o RI/B)^{1/2}$). They have shown by direct measurement that this is true for a range of ohmic profiles and for those profiles which have been measured in ICH discharges.

These measurements would seem to provide, at last, some of the direct evidence which is needed to guide the debate on profile establishment and profile consistency in tokamak discharges.

Stabilisation of the sawtooth mode by replacing some of the ohmic current by Lower Hybrid Current drive has been reported in many experiments (e.g. PETULA, ASDEX, PLT and ALCATOR). In PETULA and ALCATOR the experiments extend to densities in excess of 10^{20} m^{-3}. In these experiments also, it seems that a simple, small, modification of the central current density profile is not sufficient to explain the stabilisation. Thus there are often regions where an m=1 oscillation remains (implying q<1) and the inversion radius does not change as the Lower Hybrid power is increased to the threshold for stabilisation.

3. BOOTSTAP CURRENT

A further twist to this argument as to who shall control the current density profile is introduced by the first clear evidence for the neo-classical pressure driven current, the so called bootstrap current which was predicted 15 years ago by Bickerton, Connor and Taylor. Now careful comparison of the experimentally measured and theoretically predicted loop voltage in certain TFTR discharges gives a strong indication of the reality of the effect, an example is shown in Fig. 2. Note that in the discharge shown about 40% of the current is driven by this Neoclassical dynamo effect while 30% is driven ohmically and 30% by injected beams, this latter contribution having been minimised by using balanced injection.

4. PLASMA HEATING

Just as we may now be reaching the point at which the inherent tokamak current profile is no longer adequate for our purpose so we have already passed the point at which the inherent tokamak heating mechanism can no longer produce the required temperatures. For many years, many of the resources of fusion research have been devoted to establishing and understanding additional heating methods for the plasma. This work has come to full fruition at this Conference: many heating methods are routinely applied at the multi megawatt level to many tokamaks (see Table 2). These heating methods have now reached such a level of success that the emphasis has changed from understanding the heating to understanding the plasma behaviour. Examples are the study of energy confinement using modulated ICH by the Belgian Group on Textor and the use of Electron Cyclotron Heating on T-10 and ICH on JET as spatially localised heat sources to probe the energy transport.

The Heating methods have been very successful in increasing the plasma temperature. Two examples of successful heating from the very many available are shown in Figs. 3 and 4. First, in Fig 3 is shown Neutral Beam heating on JT60 with and without divertor: note the successful heating in each case ($\Delta T_e > 1$ keV), but note also the greatly reduced radiation loss and reduced particle influx (from H_α) in the divertor case. Second, in Fig. 4 is shown ion heating in TFTR by neutral beams. Temperatures of 20 keV are reached, thus exceeding by a wide margin the value necessary for fusion energy production.

5. WALL INTERACTION AND IMPURITY CONTROL

One of the keys to obtaining high temperature, and indeed generally good performance, in tokamaks, is the production of pure plasmas, especially plasmas in which the heavy ion impurity content is sufficiently low.

This has proved to be especially important in discharges with ICH where antenna interactions (as demonstrated in JET) or power coupled to the wall (as in ASDEX) can release metal impurities. One solution is to surround the plasma with a low Z environment. One technique, first developed at Julich (FRG) and demonstrated to be effective in JET, is carbidisation where a glow discharge is used to deposit a hard carbon layer on walls and antenna screens. Two examples of the effectiveness of the method are shown in Fig. 5. In JIPP TIIU the method is an essential prelude to high power ICH operation while the ASDEX result shows that the technique gives substantial gains even in a divertor tokamak.

An alternative technique to modify the edge plasma behaviour has been known theoretically for some years, but is now demonstrated for the first time by the group from the University of Texas (USA). A helical perturbation is used to produce an edge ergodic field layer as shown in Fig 6a. Measurement shows this to depress the edge temperature over a few cms and to steepen the edge temperature gradient (see Fig. 6b). This may turn out to have implications both for impurity control and for production of the H-mode confinement configuration.

The use of applied helical perturbations may also have applications to the stabilisation of MHD modes as was first reported by the PULSATOR group some years ago. Work reported at the conference by the groups from Tosca in the UK, HT6M in China and JFT-2 in Japan shows that the method is capable of further development.

6. CONFINEMENT

If the good news is that more heating does indeed give hotter plasmas, the bad news, until recently, has been that more heating has given worse confinement. This has been so to such an extent that the fusion figure of merit $n_D\tau_ET_i$ (product of deuteron density, energy replacement time and ion temperature) had until this week not been increased by additional heating. I could choose data from almost any experiment to demonstrate this degradation, but it would be invidious of me not to use JET data for this purpose. In Fig. 7 it can be seen that at each plasma current the τ_E degrades with increasing power according to an offset linear law for the plasma energy. The incremental energy replacement time (corresponding to the value at large power) can reach only 0.2 → 0.3s in JET compared to the best ohmic confinement times of $\sim$ 0.8s.

A way out of this dilemma was first found in ASDEX where it was shown that for divertor plasmas a second confinement mode - the so called H-mode could exist where the confinement time recovers to essentially the ohmic value and does not degrade with applied power. Experiments on DIII and later PDX showed that similar behaviour could be produced in tokamaks without a divertor, but including an internal separatrix.

Following in these footsteps, as was reported in a post deadline paper at the Conference, both separatrix and H-mode discharges have now been produced in JET. For the first time in JET the $n_D\tau_ET_i$ product has exceeded the ohmic value reaching a value of 20×10^{19}s keVm^{-3}. Some of the separatrix and H-mode points at 2 MA and 3 MA are indicated in Fig. 7. The H-mode τ_E values are better than those for similar limiter discharges by more than a factor 2. i.e. the H-mode in JET appears to be worth about 2.5 MA in Plasma current. Oddly enough it turns out that in JET mechanical forces will limit the ultimate plasma current capability with X-point to about 2.5 MA below that for limiter discharges!

It is not yet clear in JET whether or not the H-mode points will be free from power degradation as was found to be the case in ASDEX. However DIII-D a new magnetic limiter plasma, which has only been in operation for a short time at General Atomic (USA), has achieved H-mode discharges already and here τ_E does not appear to degrade with power (see Fig. 8).

The ASDEX group at this conference have given an account of the various techniques which can allow a tokamak discharge to make the transition to H-mode confinement. An increase in edge electron temperature, requiring a threshold in heating power appears to be necessary. One of the techniques which the ASDEX group advocates to achieve this, is a reduction in edge recycling. This may provide a link with the so called S-mode high confinement discharges which have recently been produced in TFTR. The method used is to reduce the edge hydrogen recycling by preconditioning with He discharges. So far, this regime can only be produced over a limited range of plasma current with roughly balanced beam injection. However, some extension of the range has been obtained by ramping the plasma current and further improvements are planned.

In this super shot regime in TFTR the energy confinement time is restored to the ohmic level and again shows no obvious degradation with additional heating power (see Fig. 9).

A further bridge between H and S mode discharges may be provided by JFT-2M which has obtained H-mode discharges in X-point configurations, but which also, remarkably, has produced H-mode discharges in a limiter configuration. As with the S-mode the technique used by JFT-2M is to restrict the edge recycling.

An important question to ask is: whether, as has been speculated, the H-mode confinement time scales as the product of plasma current and dimension, or whether it scales only as the current. Table 3 shows a comparison of the five devices which have reported high confinement modes at this conference. It will be seen that there is little to choose between the I_p and the I_pxR scaling and we must await further results to resolve this tantalizing question.

An important difficulty with the best H-mode discharges in ASDEX is that the plasma suffers from impurity accumulation effects which finally destroy it. This effect does not at present seem to occur in JET, TFTR or DIII-D, but further experiments are required.

7. DENSITY AND β LIMITS

We have discussed temperature and confinement time, two of the parameters which determine the approach to fusion conditions. The third parameter is plasma density. The first constraint is that on the plasma pressure. In order for the present generation of experiments (such as JET and TFTR) to reach fusion conditions and in order for future tokamak reactors to have a chance to be economic it is necessary that the plasma β shall reach the theoretical limit, established computationally by Troyon. The results in Fig. 10, from TFTR show that this limit can indeed be approached in the low q regime which is the one of interest for fusion relevant tokamaks.

In addition to the β limit, realistic parameters for an approach to fusion conditions require central densities in excess of $10^{20} m^{-3}$, say mean densities in excess of 7×10^{19} m^{-3}. The operational range of tokamaks is restricted by a density limit above which edge impurity radiation initiates a shrinking of the profile leading to instability and ultimately to disruption. For high field devices (such as ALCATOR and the Fascati tokamak) this limit is high enough to permit reactor relevant densities but for lower field devices there has been concern that, with achievable plasma purity, the density limit would be too low. At this conference we have seen that a number of low and intermediate field tokamaks (ASDEX, JT-60, TFTR & JET) can routinely produce values of the figure of merit $\bar{n}R/B \geq 6 \times 10^{19}$ $m^{-2}T^{-1}$. At this level, the necessary densities can be obtained. Some operating diagrams are shown in Fig 11. The higher values of $\bar{n}R/B$ are obtained with the help of pellet injection or neutral beam heating or both.

Once densities above those possible with purely ohmic discharges have been reached, however, some method of controlling the density down again is required. Success has so far been obtained by profiling the power input reduction and by introducing some in-vessel pumping such as, for instance, the suitability conditioned carbon tiles on the wall of JET.

PELLETS

The use of injected solid pellets of deuterium ice as a fuelling method for tokamaks has facilitated high density operation and has lead to considerable performance improvements in many experiments. It can also apparently lead to changes in the basic transport processes, as was reported in a paper by the ALCATOR group. Comparisons of the observed ion thermal conduction loss with Neoclassical theory are always imprecise because of the experimental difficulty of establishing the various components of the ion energy balance. The ALCATOR C experiment however appears to show a clear transition between a regime with an anomalous ion thermal loss at about 2.5 to 6 times the neoclassical value to a regime with a level which appears not to exceed the neoclassical value (see Fig 12). The transition is induced by injection of a pellet which doubles the plasma density and steepens the density gradient. Improved particle confinement and neoclassical impurity concentration are said to occur concurrently. This appears to be a regime worthy of further elucidation.

8. PROSPECTS

Now that we have discussed each of the three components of the fusion figure of merit let me bring them together on a diagram to assess our progress and our prospects.

The diagram (fig 13) is a plot of the product $\hat{n}_D \hat{T}_i \tau_E$ vers $\hat{T}_i$, the density and temperature being central values while the energy replacement time is an overall value. Two computed curves are shown: the one marked "ignition" corresponds to dominant α-particle heating (taken as P_α/P_{LOSS} = 74% or fusion power/power input, Q = 14 ie a condition where the plasma temperature would continue to escalate due to the fusion power, even if the externally applied heating power were to be progressively reduced). The other corresponds to substantial α-particle heating (taken as P_α/P_{LOSS}= 14% or Q = 0.8)*. The value of n_D T_i τ_E on these curves is approximately constant in the important region of its minimum near $\hat{T}_i$ = 20 keV and so forms a figure of merit. Also shown on the diagram (in the box) are the best values so far obtained and reported at this conference. In addition the best values of $\hat{n}_D \hat{T}_i \tau_E$ reported at earlier times are indicated for some representative experiments. It will be seen that, in round terms, the figure of merit has increased by a factor 10 over the past 3 years, a factor 100 over the past 10 years and a factor 1000 over the past 20 years. The presently obtained best value of 2 x 10^{20} keV m^{-3} s needs to be increased by about 20 times to reach the level of "ignition" but by only some 3.5 times to reach the level of substantial α-particle heating. In a particular case the substantial α-particle heating curve would be reached by a modest increase of the parameters presently observed in JET i.e. by increasing n_D from 0.55 to 1 x 10^{20} m^{-3}; T_i from 6 to 12 keV and τ_E from 0.6 to 0.8 s.

Thus, there now seems every reason to expect that todays tritium compatible tokamaks (JET and TFTR) will reach the regime of substantial α particle heating within the next few years.

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*The computed curves are for a plasma with 50% of its ions as tritium and 50% deuterium. They are for specific profiles corresponding to n/n_o and T/T_o = $(1-\rho^2)^\alpha$ where ρ is the normalised radius and $\alpha_n + \alpha_T$ = 2, the subscripts denote the indices for density and temperature. The values are insensitive to the profiles as long as we consider broadly similar shapes, thus taking $\alpha_n + \alpha_T$ = 0.5 reduces the "ignition" curve minimum from 4 to 3 x 10^{21} keV m^{-3} s.

<u>TABLE 1</u>

JT-60 LOWER HYBRID CURRENT DRIVE

FREQUENCY	2 GHz
LOWER HYBRID POWER	up to 1.2 MW
DRIVEN CURRENT	up to 1.5 MA
PLASMA DENSITY	up to 2 x 10^{19} m^{-3}
EFFICIENCY $\left(\frac{I_{driven}.\bar{n}_e.R}{P_{LH}}\right)$	up to 1.7 A/m^2 W (no beams)
	up to 2.8 A/m^2 W (with beams)

TABLE 2

OPERATIONAL MW LEVEL HEATING SYSTEMS ON TOKAMAKS

Neutral Beam (NBI)	JT60 (20MW); TFTR (20MW) JET (10MW); DIII-D (6MW) ASDEX (4.4MW)
Ion Cyclotron (ICH)	JET (7MW), PLT (4.5MW) ASDEX (2.6MW) TEXTOR (2.5MW)
Lower Hybrid (LHRH)	ALCATOR C (1.5MW); JT-60 (1.2MW); PLT (1MW); ASDEX (1MW); PETULA (0.5MW)
Electron Cyclotron (ECRH)	T-10(2MW); TFR(0.6MW); JFT-2M (0.2MW); T-7 (0.4MW); CLEO (0.2MW)

ALFVEN WAVE TCA (0.6 MW)

TABLE 3 HIGH CONFINEMENT REGIMES (COMPARISONS)

EXPERIMENT	τ_E(s) NO RADIATION CORRECTION	I (MA)	R (m)	$\tau_E/I_p.R$ (normalised)	τ_E/I_p (normalised)
ASDEX	0.08	0.3	1.7	2	2
JFT-2M	0.034 0.033	0.22 0.28	1.3 1.3	1.5 1.2	1.2 0.9
BIG D	0.17	1.0	1.7	1.3	1.3
TFTR	0.15	0.9	2.5	0.8	1.9
JET	0.5 0.6 → 0.9	2 3	3 3	1.1 0.8 → 1.3	1.5 → 2.2

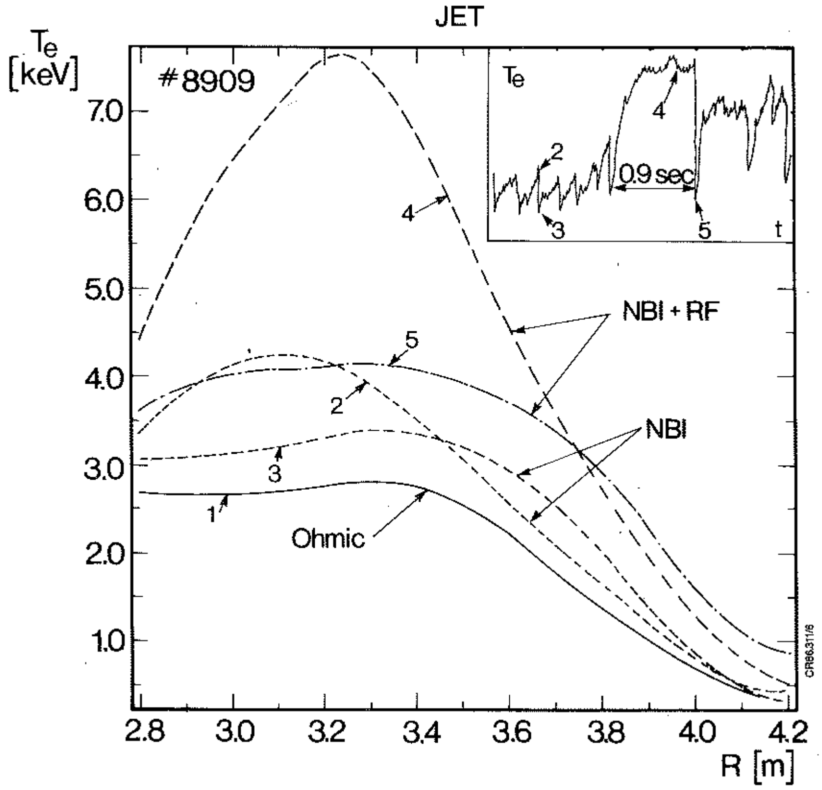


Fig 1. Spontaneous stabilisation of Sawtooth oscillation in JET.

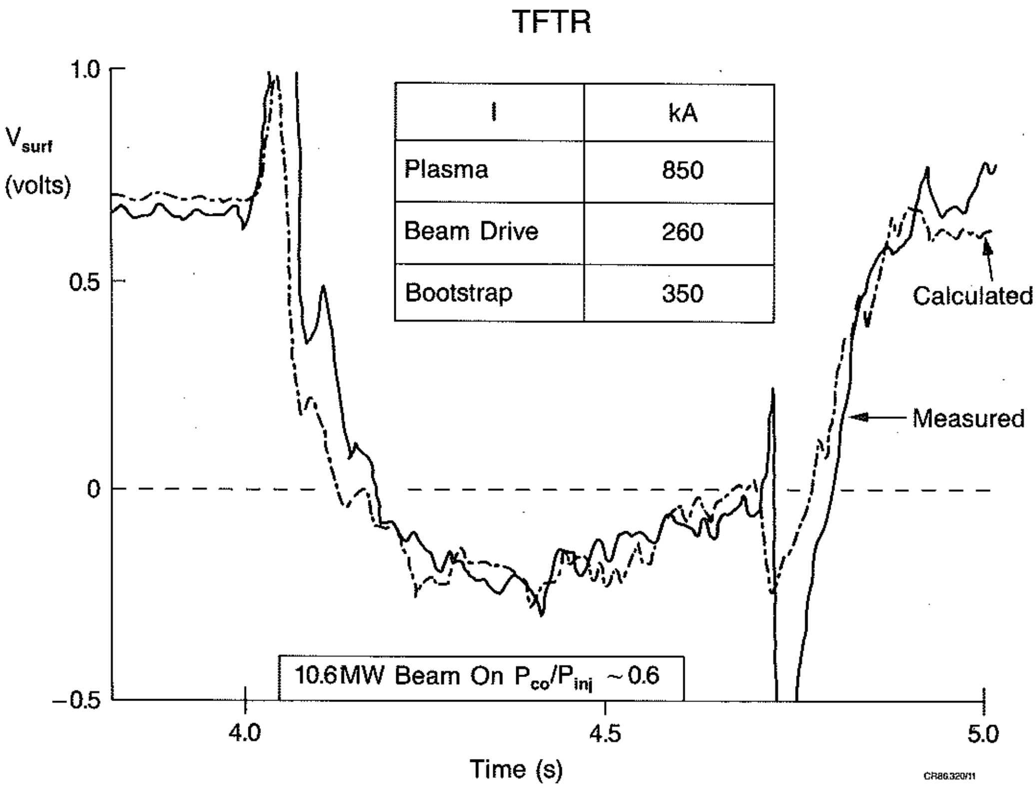


Fig 2. Evidence for the Bootstrap current in TFTR.

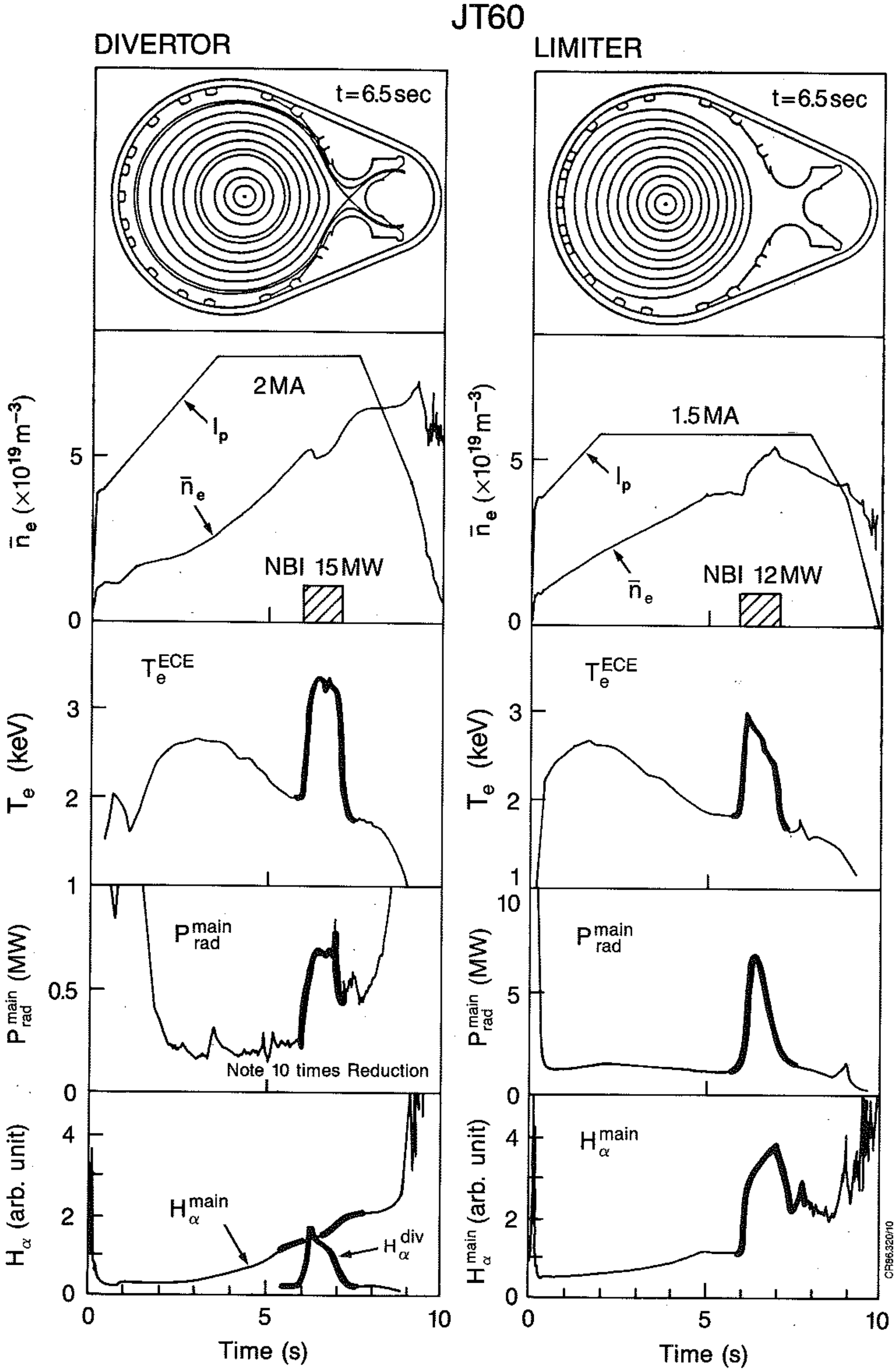
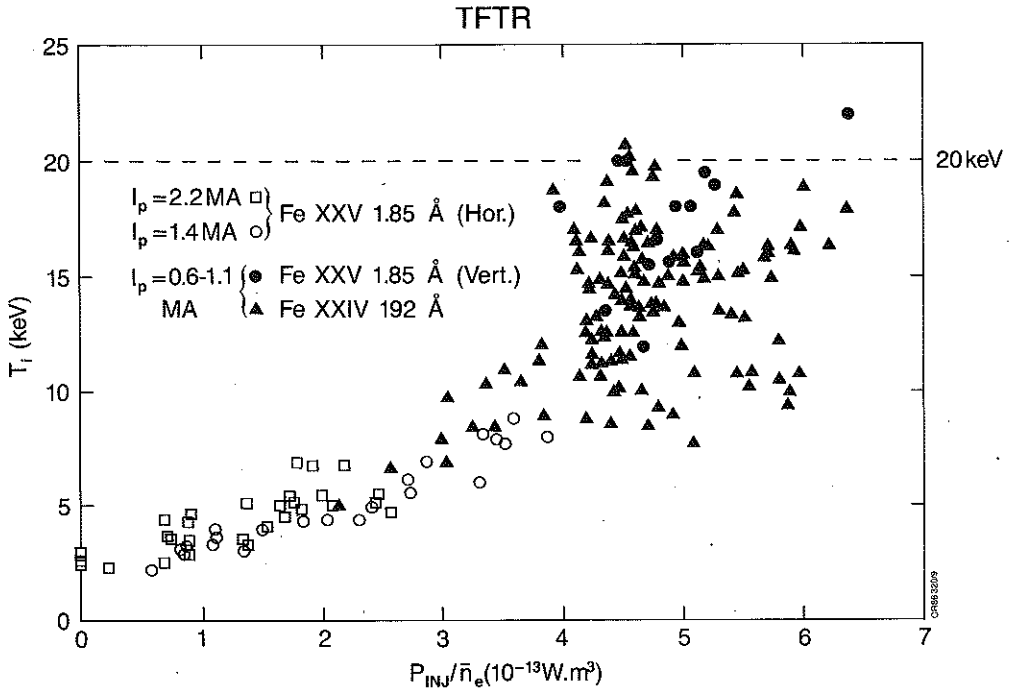


Fig 3. Electron Temperature Heating in JT-60.



$\frac{\bar{n}_e \, \Delta T_i}{P_{inj}}$ $\leq 3.8\times10^{13}\,\mathrm{keVW^{-1}m^{-3}}$ in supershots ($\leq 1.5\times10^{13}$ in other shots)

Fig 4. The production of high Ion Temepratures in TFTR.

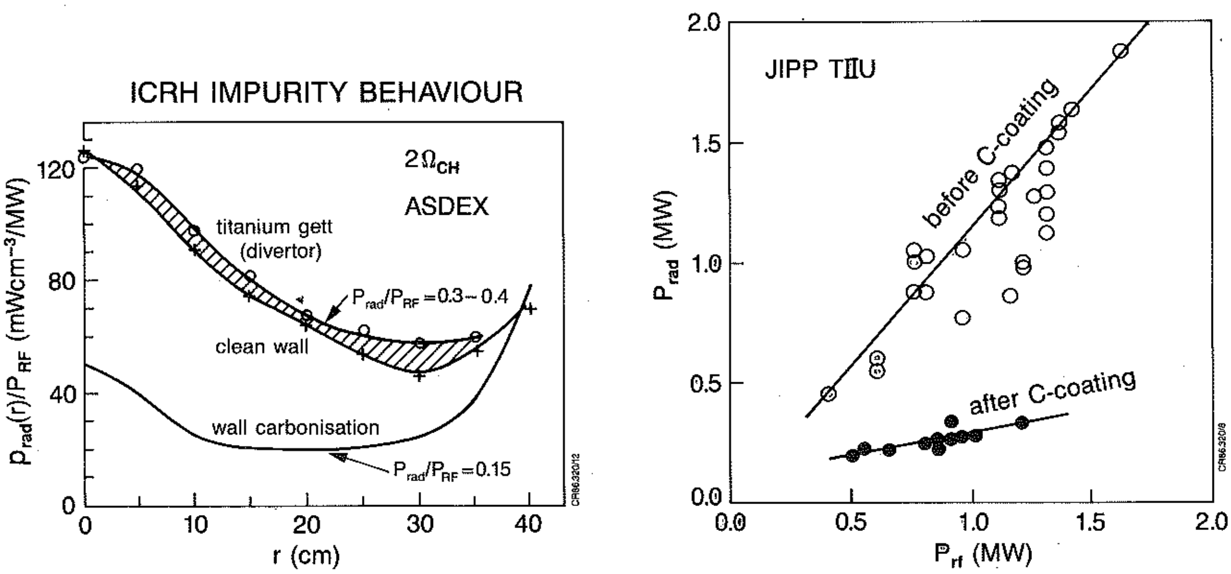
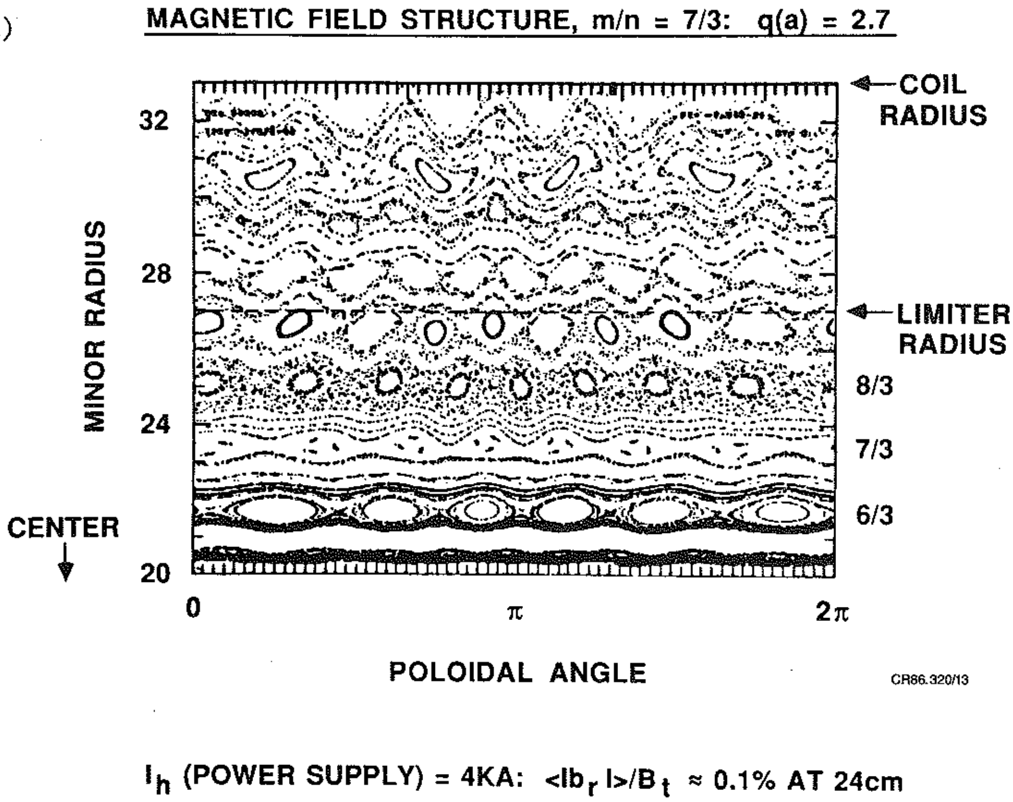


Fig 5. The use of wall carbidisation for impurity control JIPP TIIU and ASDEX.

(a)



(b)

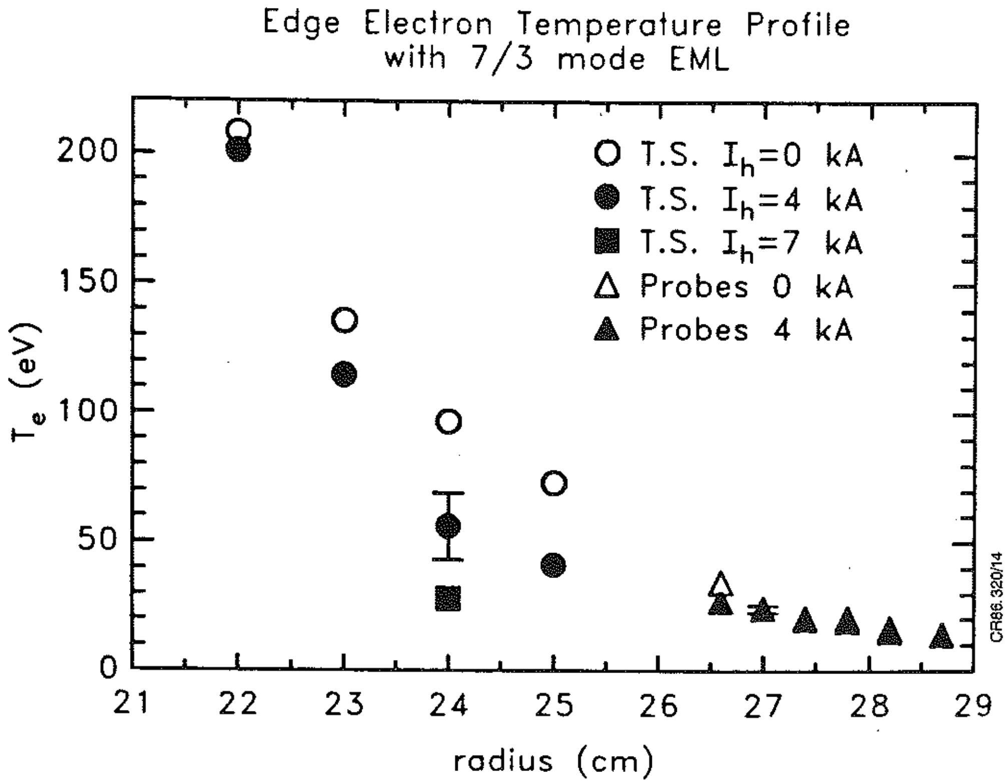


Fig 6. The effect of an Ergodic Limiter on Edge Temperature Profiles in TEXT. (a) Computed magnetic configuration in edge region, using actual coil configuration. (b) measured edge temperature profiles for various currents in the helical col.

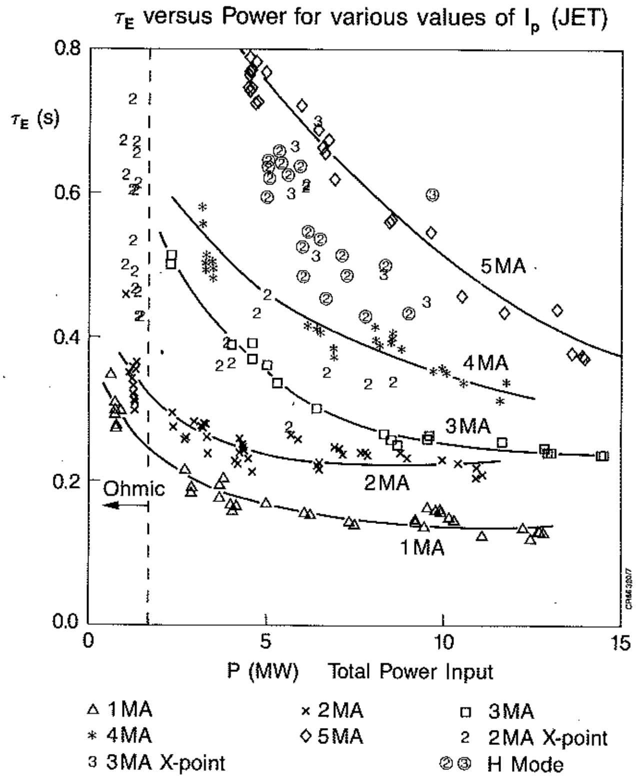


Fig 7. Degradation of Energy Replacement Time with Additional Heating Power in JET.

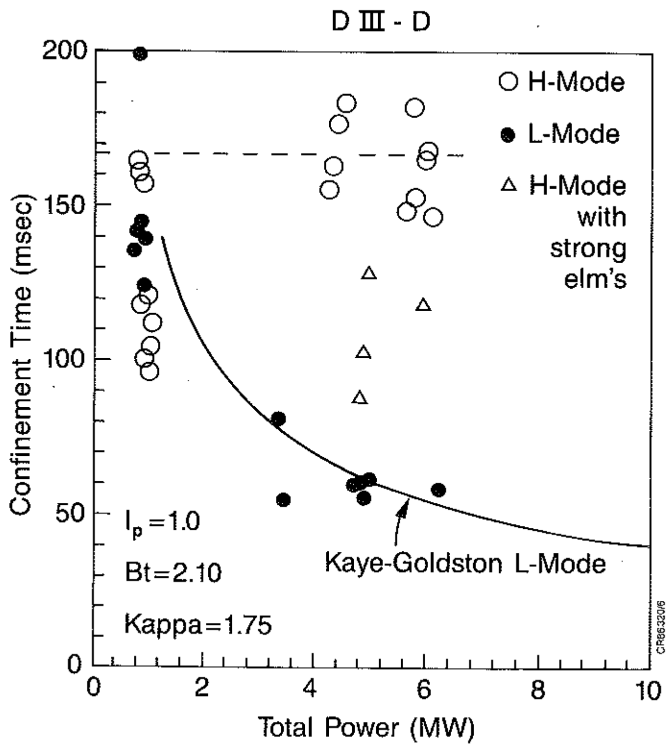


Fig 8. H-mode behaviour in DIII-D.

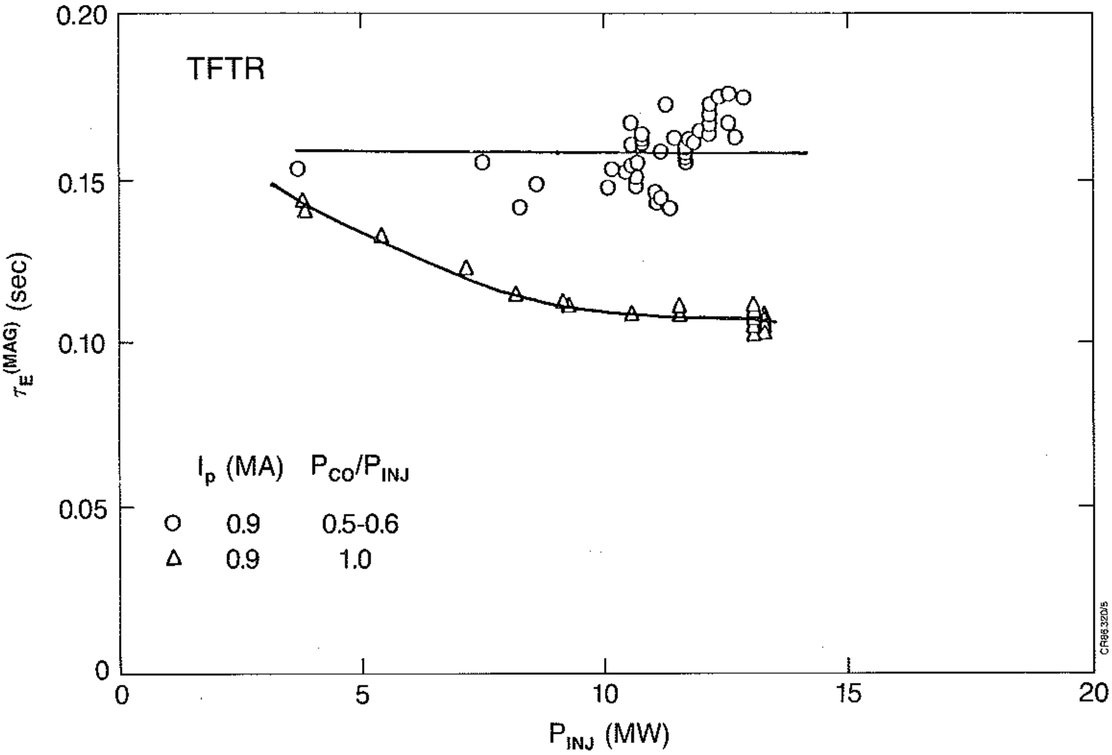


Fig 9. An Improved Confinement Regime in TFTR.

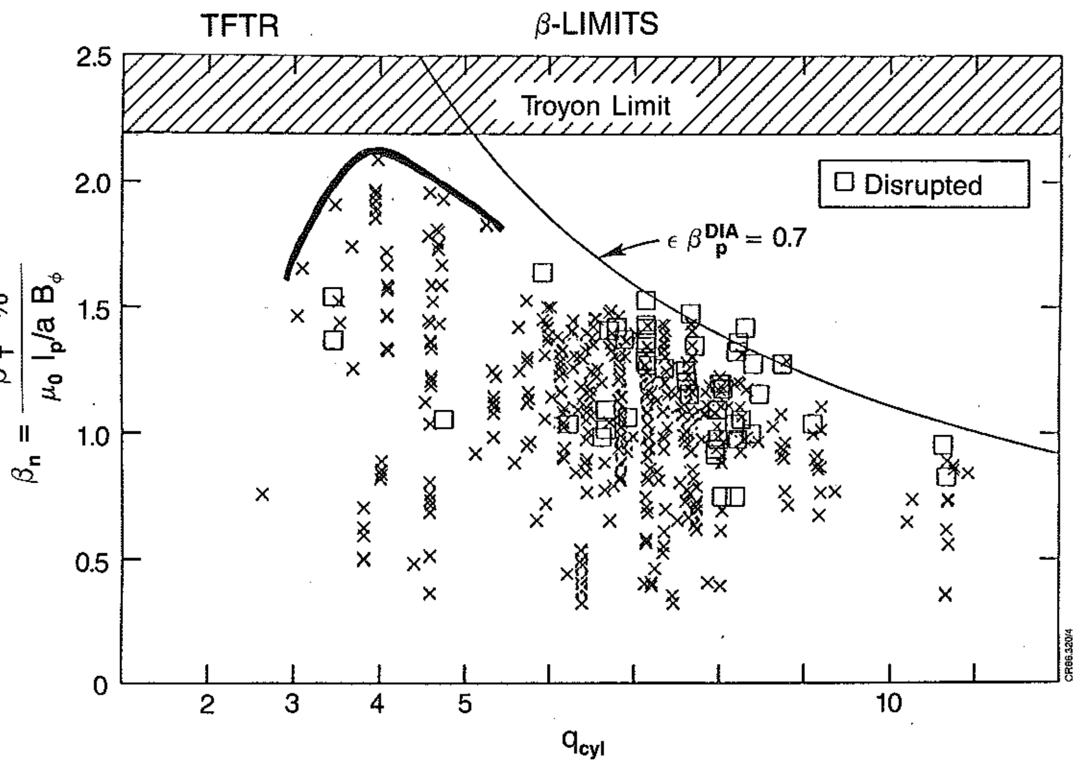


Fig 10. Approach to the Theoretically Predicted β limit in TFTR.

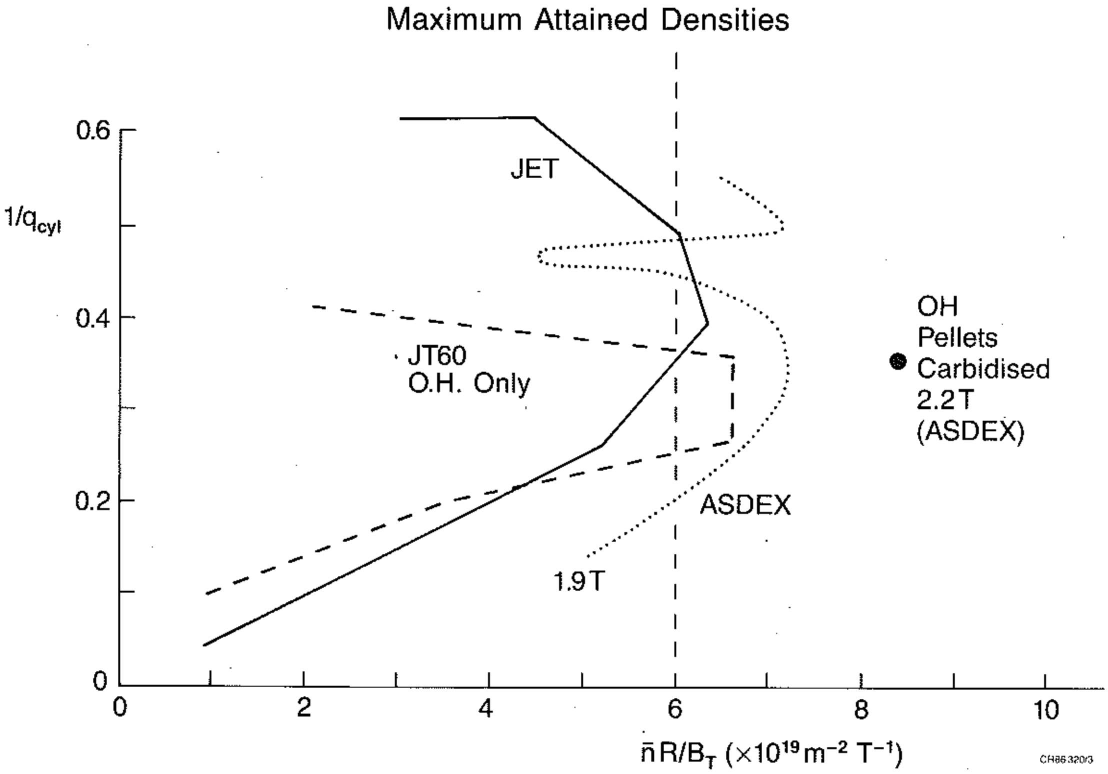


Fig 11. Density Limits in a Number of Tokamaks.

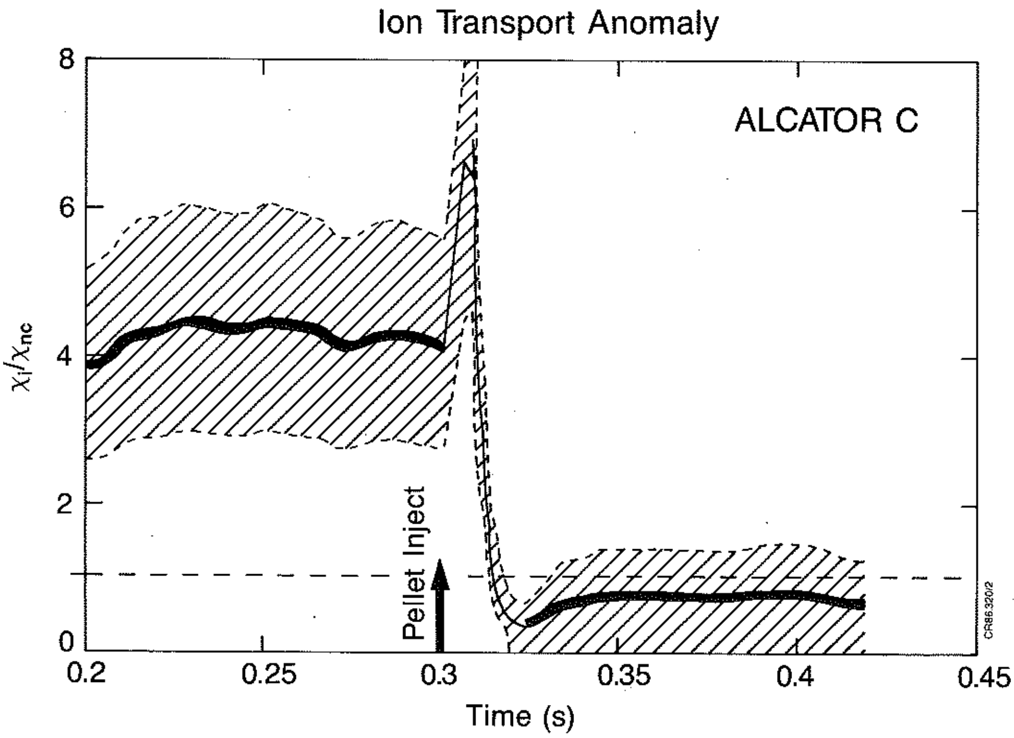


Fig 12. Neoclassical Transport Behaviour in Alcator-C.

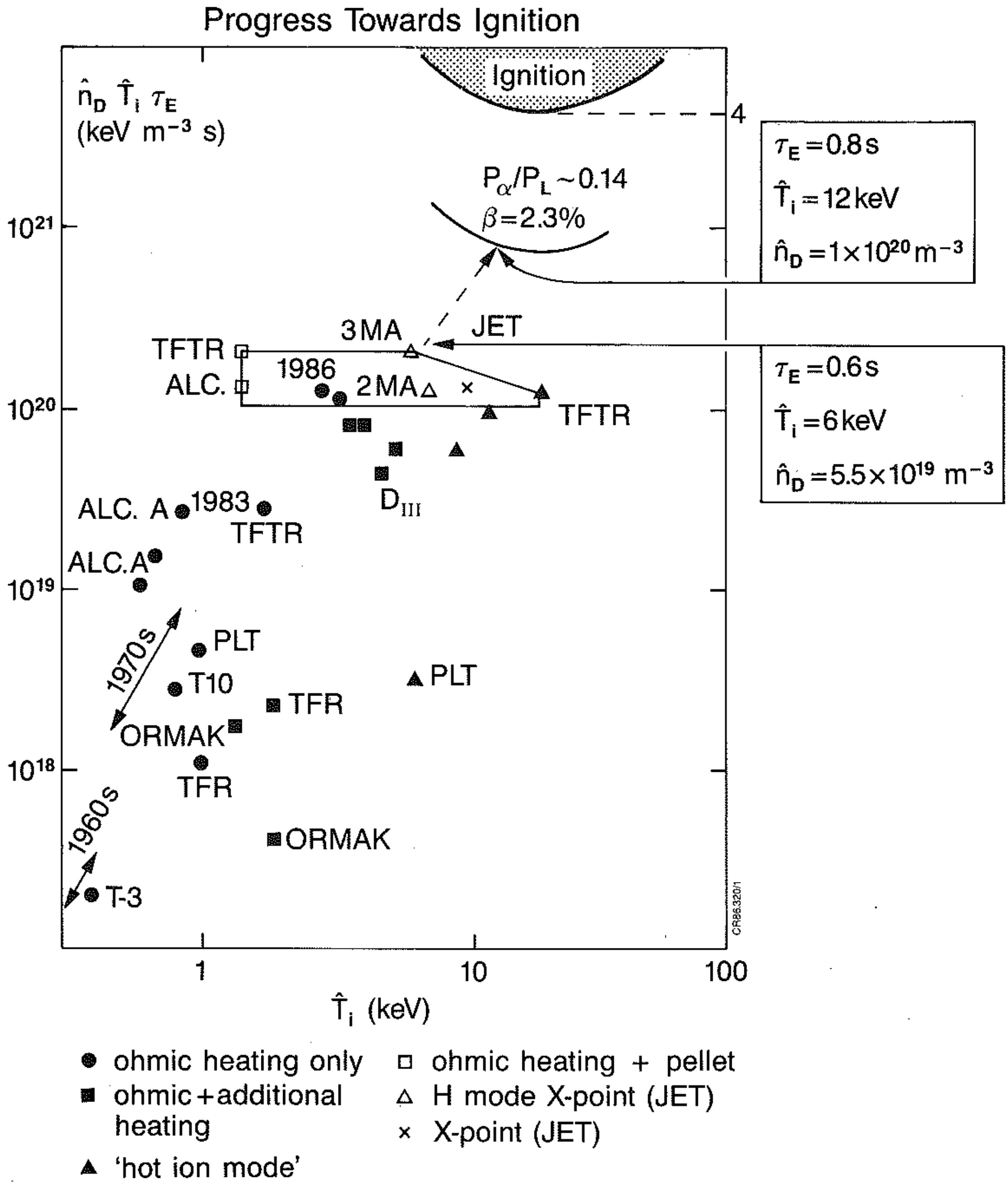


Fig 13. The fusion figure of merit $(\hat{n}_D\ \hat{T}_i\ \tau_E)$ versus central ion temperature $(\hat{T}_i)$ showing Progress towards Ignition Conditions with Tokamak Systems.