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Tokamak experiments

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ABSTRACT

With the advent of the new large tokamaks such as JET, JT-60, TFTR, important advances in magnetic confinement have been made. A number of small and medium sized tokamaks have also come in to operation, each addressing specific problems which remain for the future development of the system.

The tokamak has moved from success to success over a period of some 25 years and this has been achieved without any control of the current distribution. Many research workers now believe that the final step can be achieved by providing that control with the application of non-inductive current drive. Lower hybrid current drive has now been used very effectively and efficiently on JT-60 and electron cyclotron resonance heating current drive has been demonstrated on smaller devices. The control of the current distribution has led to the stabilisation of the internal sawtooth oscillation and m=2 activity. Sawtooth control has led to impressive rises in central electron temperature. A key feature of recent experiments has been the possible demonstration of the existence of the bootstrap current which is a neoclassical prediction for a toroidal device.

Ion and electron cyclotron resonance heating together with neutral beam heating have been highly successful, however in most cases they lead to some degradation in energy confinement time. This degradation can be avoided with the so called H-mode which occurs in magnetic configurations with a separtrix. The best nT τ product for magnetic fusion has been achieved on the JET device with an H-mode. An alternative method of achieving enhanced confinement has been established with the super shots on TFTR. Density and β limits still pose important restrictions on tokamak operation. Methods of overcoming the density limit by using applied helical perturbations are being investigated on smaller tokamaks. Shaping and profile control should improve the β limit. Pellet refueling has also proven to be a highly satisfactory technique for improving confinement and decreasing the impurity concentration.

It now seems quite likely that todays tritium compatible tokamaks will demonstrate substantial α particle heating within the next few years.

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1. INTRODUCTION

With the advent of the new large tokamaks JET, JT-60 and TFTR important advances in magnetic confinement have been made. These include the exploitation of radio frequency and neutral beam heating on a much larger scale than previously, the demonstration of regimes of improved confinement and the demonstration of current drive at the Megamp level. A number of small and medium sized tokamaks have also come into operation recently such as WT-3 in Japan with an emphasis on radio frequency current drive and HL-1 a medium sized tokamak in China. Each of these new tokamaks is addressing specific problems which remain for the future development of the system. Of these particular problems: β, density and q limits remain important issues for the future development of the tokamak. β limits are currently being addressed on the DIII-D device in the USA. The anomalous confinement that the tokamak displays is being explored in detail on the TEXT device in the USA. Two other problems areas are impurity control and current drive. There is significant emphasis on divertor configurations at the present time with their enhanced confinement in the so called H mode. Due to improved discharge cleaning techniques and the ability to repetitively refuel using pellets, purer plasmas can be obtained even without divertors. Current drive remains a crucial issue for quasi of near steady state operation of the tokamak in the future and many current drive schemes are currently being investigated.

The tokamak has been able to evolve very successfully over a period of some 25 years and this has been done without any control of the current distribution apart from that which naturally occurs through sawtooth activity and programming the external current. Many research workers in the tokamak field believe that the final problems associated with the tokamak can be overcome by providing sufficient control of the current distribution through localised and global non inductive current drive. This is currently being explored on small, medium and large devices.

Progress towards ignition can be conveniently summarised in a plot of the product in $\hat{n}_i\hat{T}_i\tau_E$ versus $\hat{T}_i$ with the ion density, $\hat{n}_i$ and the ion temperature, $\hat{T}_i$, being central values while the energy replacement time is a global value.

Figure 1 shows the progress from the 1960's to the 1980's for tokamaks though some stellarator points are also included. The ignition region is shown corresponding to dominant α particle heating, the Q_{DT} = 1 curve corresponds to substantial α particle heating. It is to be noted that both JET and TFTR have Q equivalent to Q_{DT} values in excess of 0.1. The figure shows clearly that the figure of merit has increased by a factor of 1000 over the last 20 years. The best value of 2 x $10^{20} m^{-3}$ s keV obtained on JET, has to be increased by ~ 20 times to reach the level of ignition but only by about 4 times to provide substantial α particle heating.

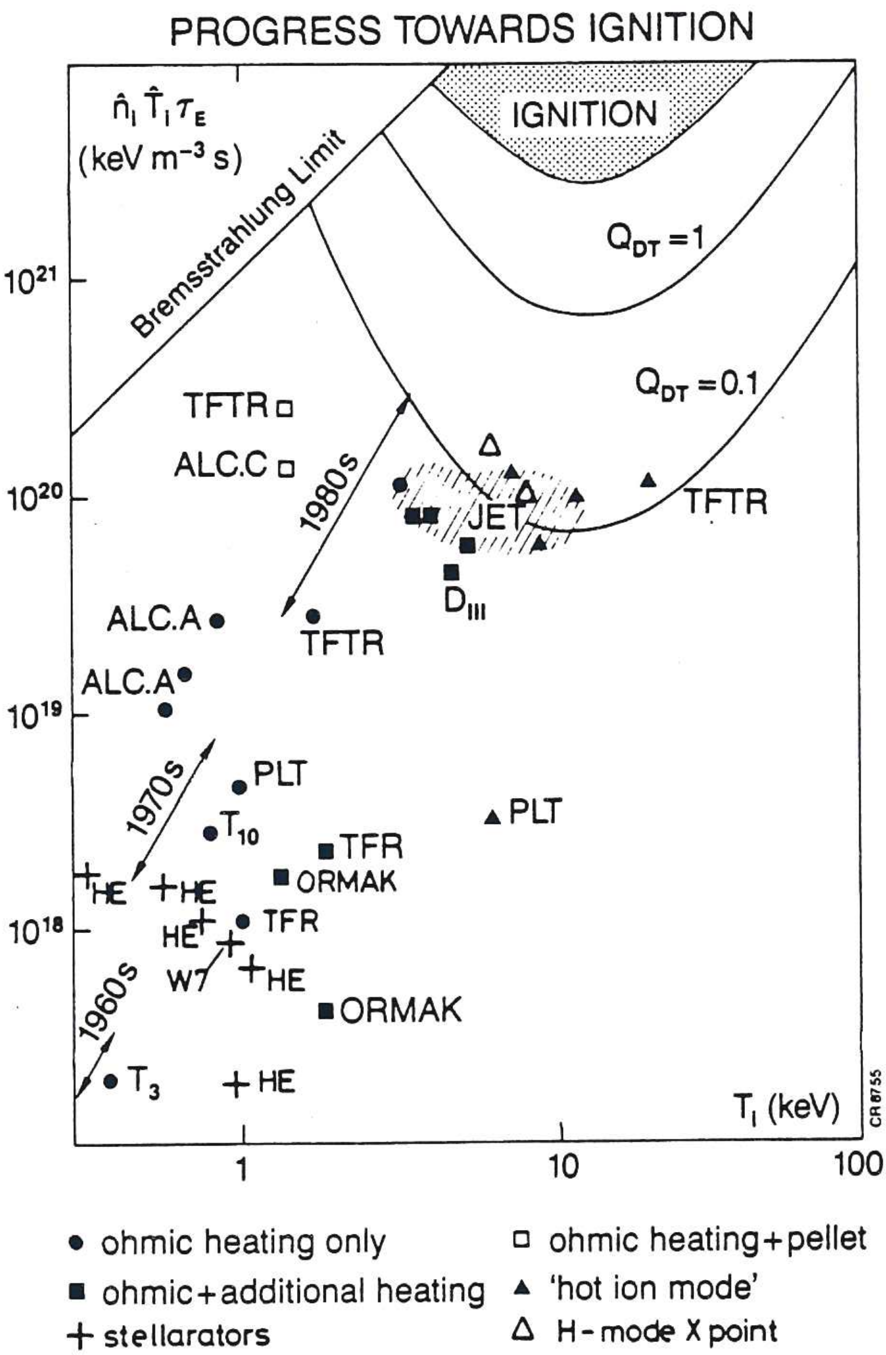

Fig 1. Progress towards ignition in the $\hat{n}_i \hat{T}_i \tau_E$ - $\hat{T}_i$ plane over the last 20 years.

2. PLASMA HEATING

A number of heating methods have now been routinely applied at power levels > 1 MW. These include neutral beam injection at levels up to 20 MW on TFTR and JT-60. On TFTR in the so-called supershots [1] ion temperatures as high as 30 keV have been obtained as shown in Fig 2. These temperatures exceed by a wide margin the values necessary for fusion energy production. Ion cyclotron

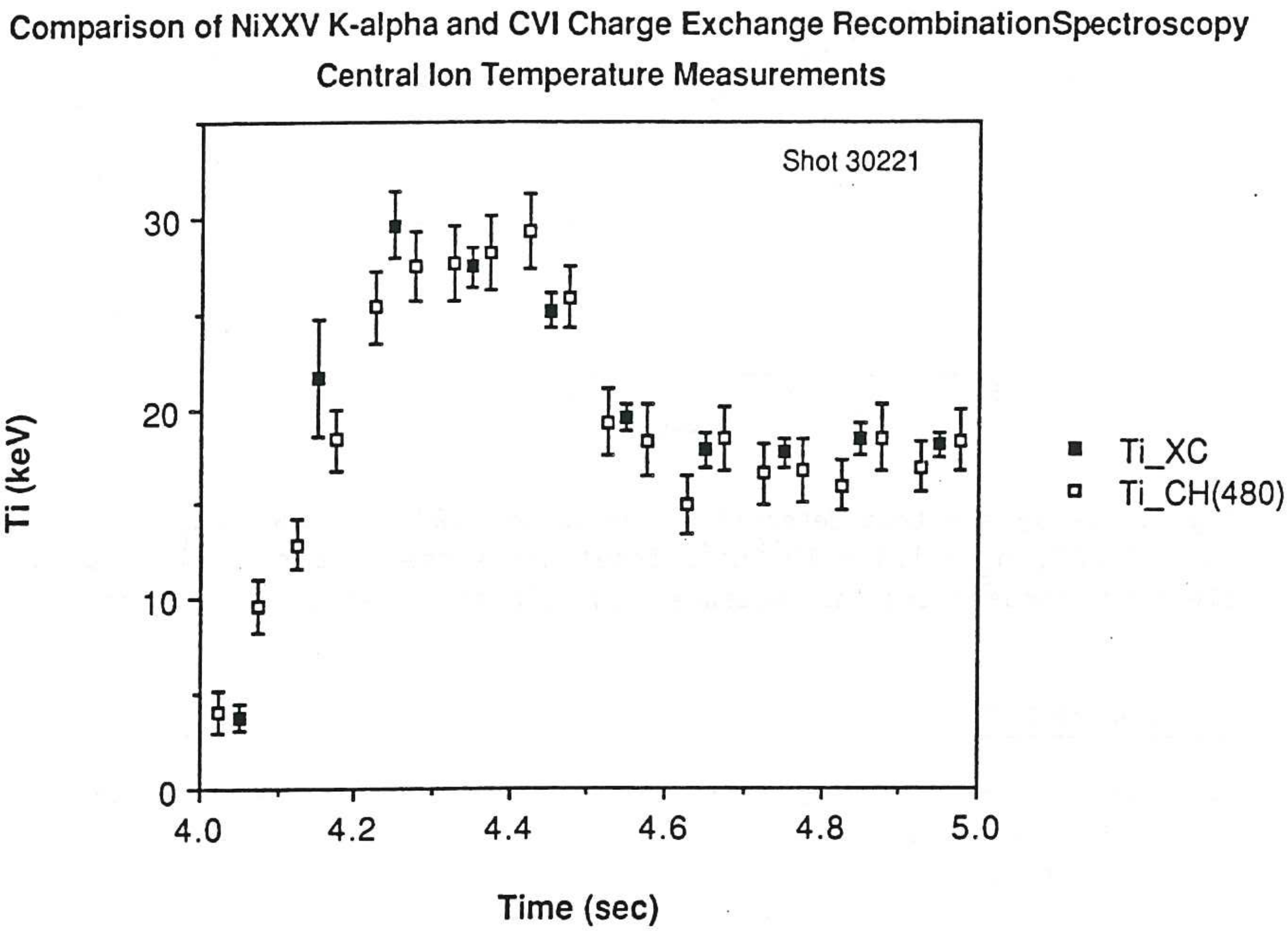

Fig 2. Central ion temperature as a function of time on TFTR supershot with neutral beam heating.

resonance heating has been deployed on JET at a level of up to 7 MW [2] and its spatially localised properties as a heating source have been used to probe energy transport. Electron cyclotron resonant heating has been used on the T10 tokamak [3] at a level in excess of 2 MW and its spatially localised heating potential has also been exploited in an effort to understand the energy confinement time. Figure 3 shows recent results obtained with high power ECRH on T10 where electron temperatures close to 10 keV have been obtained. Lower hybrid resonance heating [4] and Alfven wave heating [5] have also been used on a number of medium sized tokamaks at levels up to 2 MW.

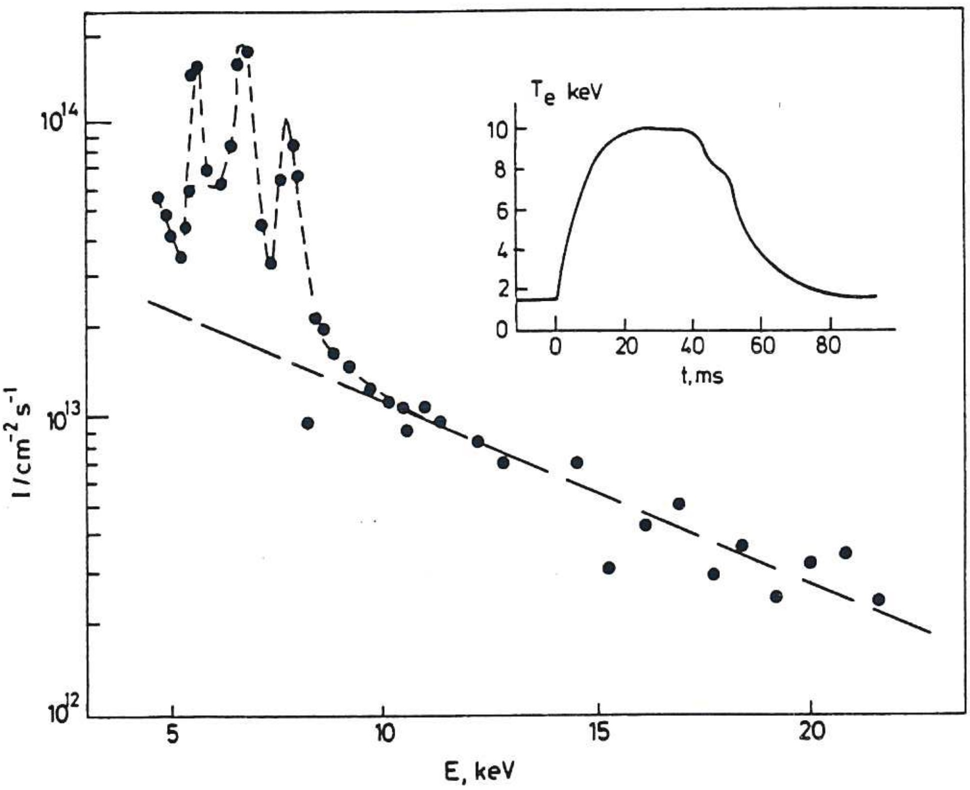

Fig 3. X-ray spectrum detected during strong ECRH on T10 with I_p = 210 kA, B_T = 2.82T, n_e = 1.5 x $10^{19}m^{-3}$. Inset shows the temporal behaviour of electron temperature as measured by electron cyclotron radiometer.

3. CONFINEMENT

Degradation of the energy confinement time with additional heating power has been a feature of most tokamaks. Unfortunately this is so strong that the fusion product figure of merit, $\hat{n}_i\hat{T}_i\tau_E$ does not increase with additional heating. Figure 4 demonstrates the decrease in energy confinement time with additional heating power on JET and the associated current scaling which characterises the so-called L mode [6] confinement regime. This behaviour can be described by the Goldston relation $\tau_E \propto I/P^{\frac{1}{2}}$ where P is the power, or alternatively by using the concept of an incremental energy confinement time [2]. This incremental time may be only 0.2 - 0.3 seconds on JET compared with the best ohmic confinement time of ~ 0.8s.

Recently a number of ways of improving the L mode confinement degradation have appeared. On TFTR preconditioning with helium has led to reduced hydrogen recycling at the edge and for plasma currents less than ~ 1 MA, a regime of improved confinement is obtained with the energy confinement time being comparable with the ohmic value ~ 0.16s and showing little degradation with increasing power. Unfortunately as the plasma current is raised further MHD activity is induced and this degrades the confinement time [7].

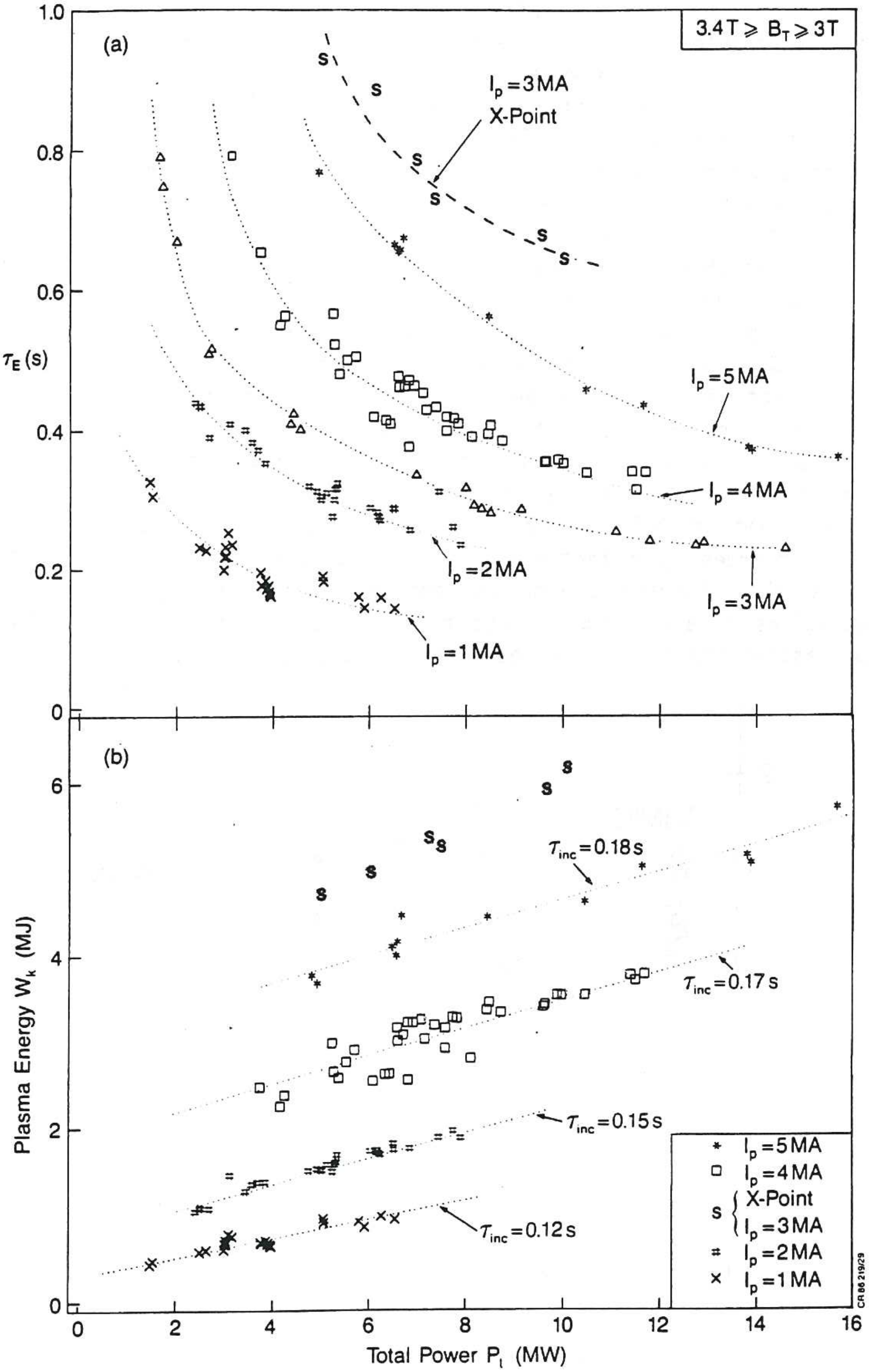

Fig 4. Energy confinement time as a function of additional heating power on JET and energy content as a function of total input power showing the incremental confinement time. X point discharges at plasma currents of 3 MA are shown by the S.

A particularly important and effective way of overcoming this confinement degradation problem was discovered on ASDEX with a closed divertor configuration. Here a new mode of operation called the H mode was observed in which the confinement time recovered to its ohmic value and did not degrade with increasing power. The key feature of this mode of operation is sufficient additional heating power in the presence of a magnetic limiter configuration with an X point and with the last closed flux surface some 2 cm away from a limiter. Later experiments on DIII and PDX showed that a similar regime could be obtained in tokamaks with an open separatrix. In the DIII-D device at GA Technologies with a magnetic limiter configuration the plasma shows H mode behaviour with no confinement degradation [8] with increasing power - Fig 5. However similar experiments on JET, Fig 4 do show confinement degradation with increasing power though in these experiments the density and radiation increase in an uncontrolled manner possibly because of the absence of edge localised modes. On JET for the first time the $\hat{n}_i\hat{T}_i\tau_E$ product has exceeded the ohmic value and Fig 1 indicates H mode points at 3 MA. τ_E values obtained in JET in the H mode are better than those obtained in limiter discharges by a factor of 2-3. Unfortunately mechanical forces in JET limit the ultimate plasma current carrying capability for X point configurations to about 2.5 MA less than limiter discharges thus offsetting the substantial advantage obtained with H mode discharges.

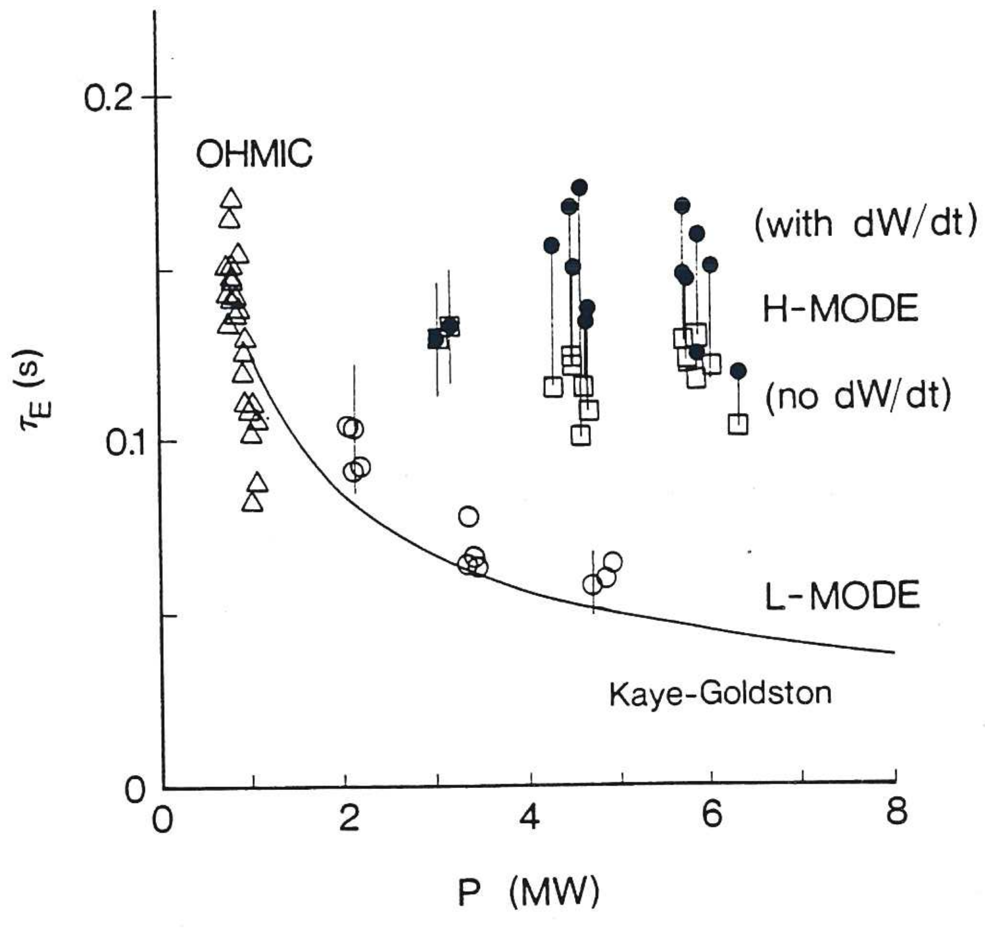

Fig 5. Confinement time for L mode and H mode discharges and comparison with the Kaye-Goldston scaling for divertor discharges with a plasma current of 1 MA on DIII-D.

Until recently the JT-60 magnetic configuration with its X point in the region of unfavourable curvature of the magnetic field on the outside of the torus, showed no evidence for H mode behaviour possibly indicating the importance of the region in which the X point is situated.

A further potentially important observation is that on JFT-2M [9] where H-mode-like discharges have been obtained in non-X point configurations with the plasma on the inboard wall and again reduced edge recycling is found to be very important.

It is not yet clear in the different devices in which the energy confinement time is improved, whether τ_E increases simply with the plasma current or as the product of major radius and plasma current. Alternative explanations in which the incremental confinement time is found to scale simply with the minor radius squared have also been suggested.

4. CURRENT DRIVE

Recent results on JT-60 have demonstrated lower hybrid current drive with very high efficiency [4] at the 2 MA level with 2 MW of lower hybrid waves at a frequency of 2 GHz. Figure 6 shows a current ramp-up case. The current drive efficiency is improved by combining neutral beam heating with LHCD. This gives an efficiency of 2.8A per watt per m^2 which is about 3 times larger than that obtained in smaller experiments.

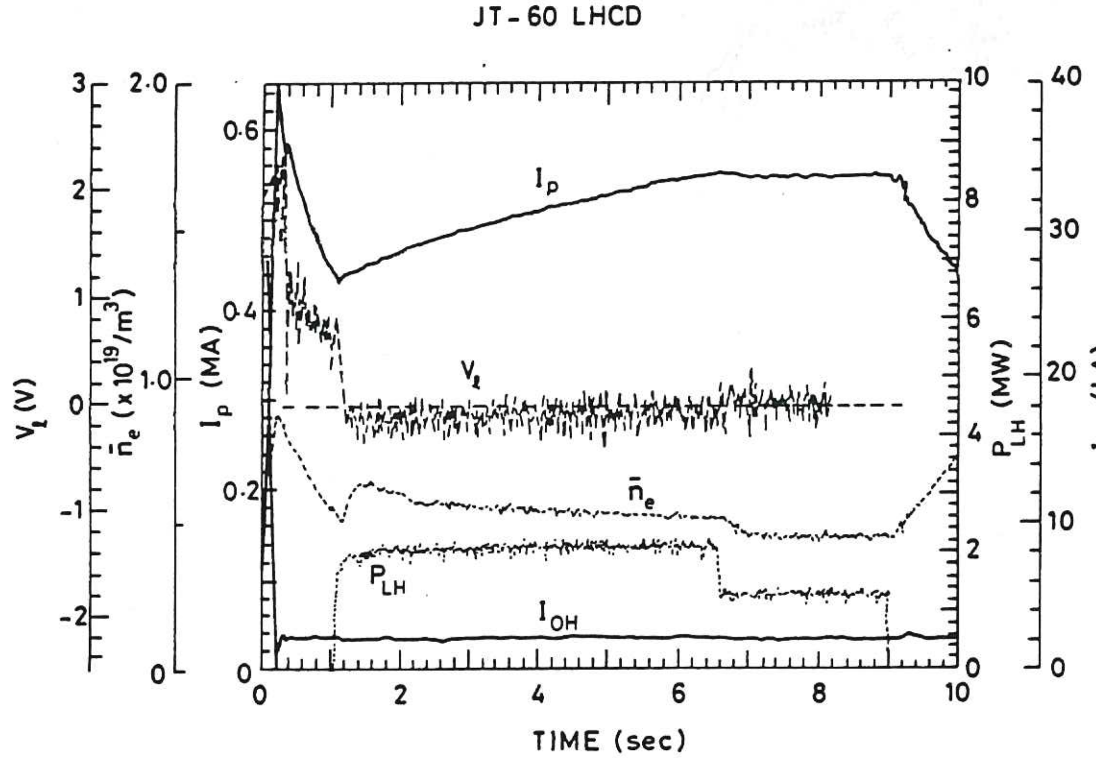

Fig 6. Lower hybrid current drive on JT-60 showing the application of 2 MW of lower hybrid power to ramping up the plasma current to 0.5 MA over a duration of ~ 8s.

On the smaller tokamaks, electron cyclotron current drive has been demonstrated both in near thermal plasmas on the CLEO experiment [10,11] and nonthermal plasmas in the WT-2 and 3 devices [12,13]. Figure 7 shows the loop voltage obtained in the CLEO experiment for plasmas in which the electron cyclotron waves are directed co and counter to the plasma current direction. 50% of the plasma current has been driven in this way. The figure also shows the expected loop voltage variation if the neo-classical bootstrap current is also present in these discharges. It seems from the measured variation in loop voltage that there is little evidence in these hot electron plasmas for the bootstrap current.

However experiments in TFTR, in the hot ion regime, from analysis of the loop voltage variation do indicate evidence for the bootstrap current [14] in high poloidal β regimes similar to those obtained on CLEO. In this case about 40% of the plasma current might be driven by the neoclassical dynamo effect with some 30% being driven by injected beams.

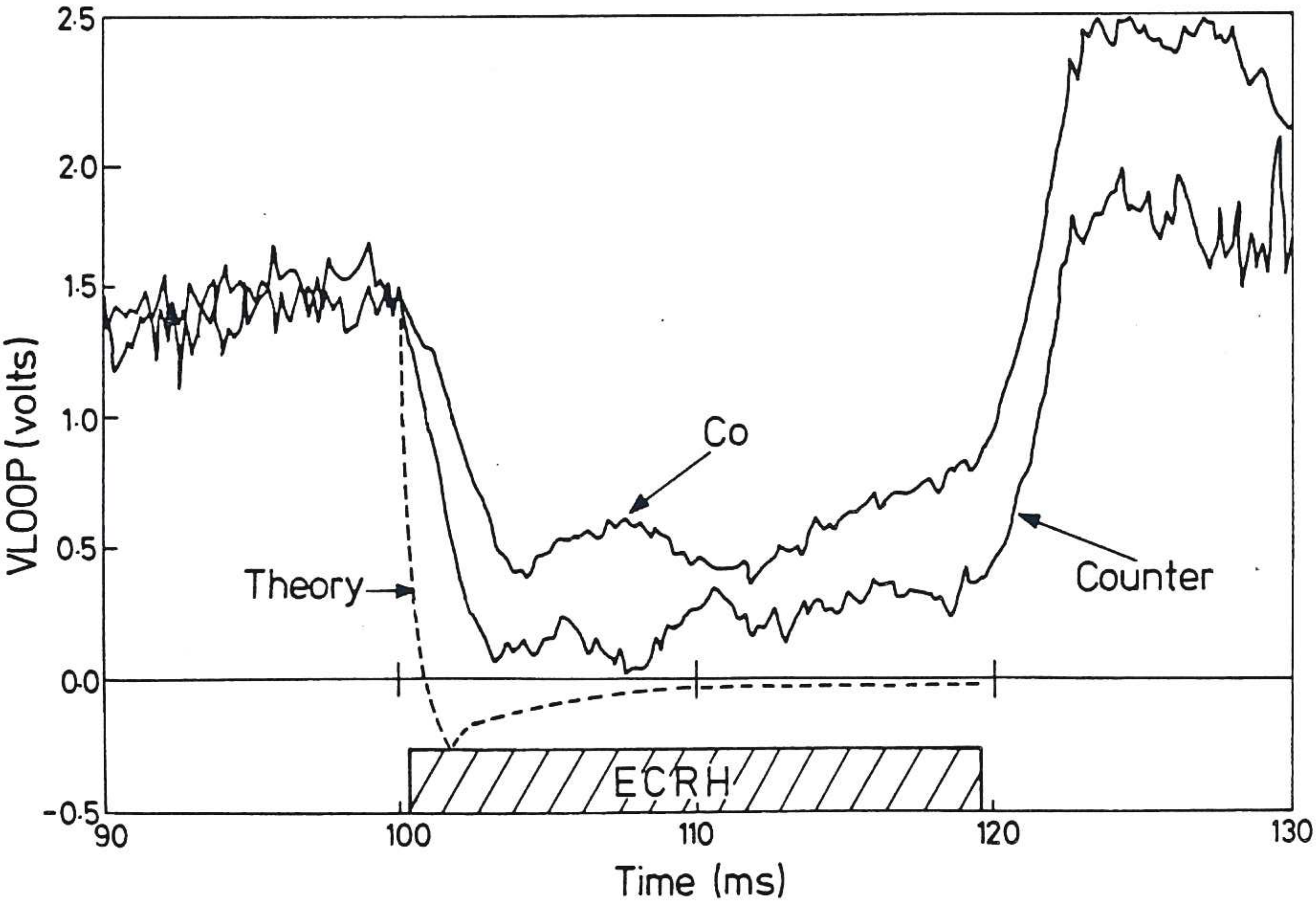

Fig 7. Loop voltage on CLEO for clockwise and counter-clockwise plasma currents and the calculated voltage including the bootstrap current for the counter case. I_p = 10 kA, $\bar{n}_e$ = 3.5 x 10^{18} m^{-3}, T_{eo} = 1.3 keV.

The impact of these driven currents on the current density profile and stability of the discharge has been noted in many experiments. The most impressive results have been associated with the stabilisation of the internal disruption relaxation, the so-called sawtooth activity. Figure 8 shows the results of partial sawtooth stabilisation achieved on the JET device [15]. The result of this stabilisation is that the electron temperature profile peaks with central values close to 7 keV. The application of centrally resonant ion cyclotron heating induces partial sawtooth stabilisation on JET for times which last for more than 1s. It is believed that the sawtooth behaviour is controlled by a very flat current density profile in this case.

Sawtooth stabilisation by lower hybrid current drive and by ECR heating and current drive has been reported in many experiments. In a number of these experiments a small broadening of the current density profile has been measured which has been invoked to explain stabilisation, however continuous m=1 oscillations often remain and there may be no change in the radius, where q is believed to be ≈ 1, seeming to indicate more localised changes in the current profile may be responsible.

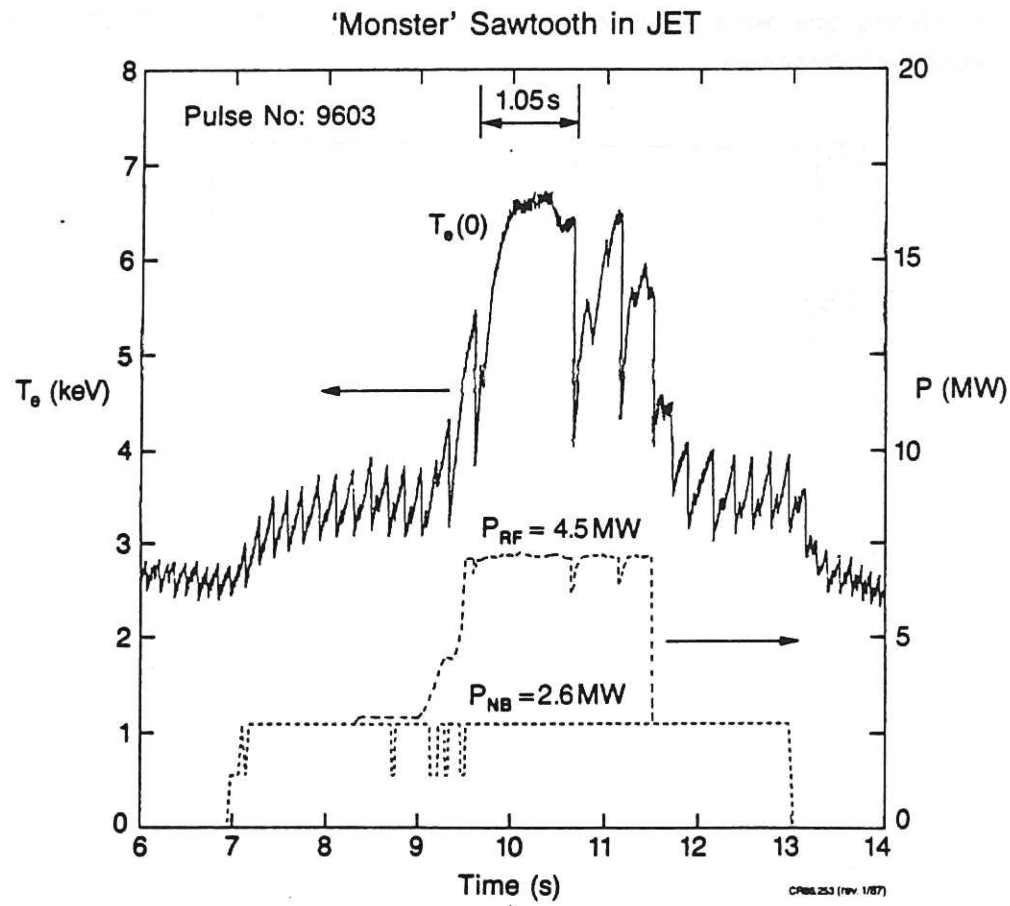

Fig 8. Central electron temperature on JET showing the increase obtained between the giant sawtooth with 4.5 MW of ion cyclotron heating.

5. ANOMALOUS CONFINEMENT

It is believed that the anomalous confinement in a tokamak is due to turbulence induced transport which affects both the confinement of energy, momentum and particles. A key question is whether this transport process is due to electrostatic or electromagnetic fluctuations. It is particularly difficult to relate the fluctuations to transport, however studies with probes in the outer regions of small tokamaks, eg TOSCA [16] have demonstrated that convective electrostatic turbulence is adequate to account for the particle transport. In addition the magnetic fluctuations measured in the outer regions of such devices by probes, appear to be inadequate to account for the heat transport. Detailed studies further into the core of the plasma on the TEXT tokamak [17] of the density and potential fluctuations show that they do not obey the Boltzmann relation ie density fluctuation level not proportional to potential fluctuations normalised to electron temperature - Fig 9. This was also observed to be the case on TOSCA using probes [18]. This would seem to indicate that collisional drift waves, dissipative trapped electron modes, modes driven by electron and ion temperature gradients which are too high relative to density gradients in these plasmas are not responsible for the anomalous transport but that instabilities of a fluid origin such as those driven by resistivity or pressure gradients are more likely to be associated with the fluctuations and transport which is observed.

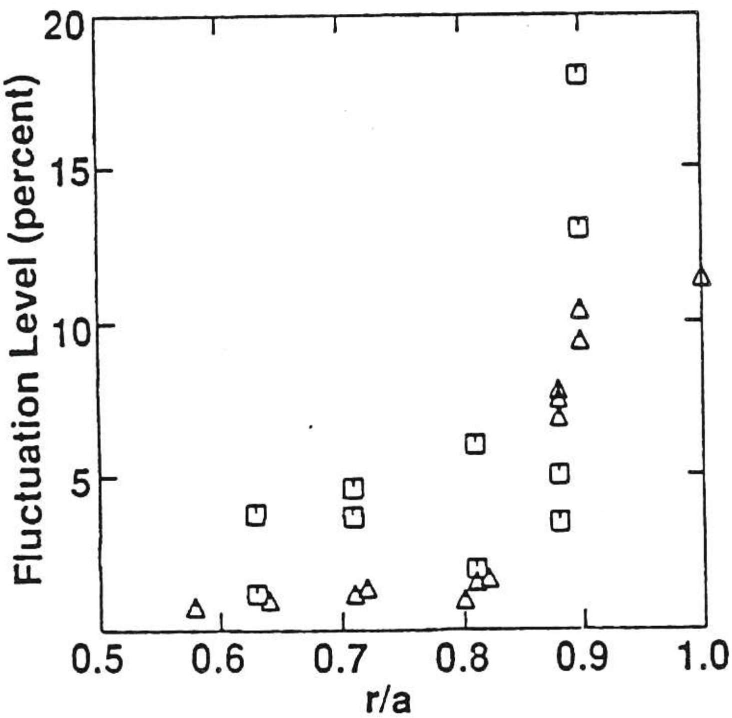

Fig 9. Normalised fluctuation levels on TEXT, boxes are $\tilde{\Phi}/kT_e$ and triangles are $\tilde{n}/n$. Spectra are integrated from 50 kHz to 500 kHz to eliminate MHD effects. The density point at r/a = 1.0 has substantial power in the 0 to 50 kHz range which is not included in this measurement.

A detailed correlation of the density and potential fluctuations at about half radius on TEXT leads to a particle flux and a heat flux sufficient to account for both particle and energy confinement from the electrostatic turbulence. This would seem to indicate that in ohmically heated plasmas magnetic fluctuations leading to stochastic magnetic fields and anomalous transport for example as a result of micro tearing modes, is not central to the anomalous confinement process. However as the plasma pressure increases most devices show a substantial rise in the magnetic fluctuation level and thus it becomes more likely that the magnetic fluctuations may be sufficient to lead to strong losses along the stochastic field lines. Direct measurement of the magnetic fluctuations in the core of a plasma, where a level $\delta b/B \sim 10^{-4}$ would account for the anomalous transport, has not yet been made. Indirect deductions from runaway electrons show small scale lengths ~ 1mm comparable in magnitude and scaling to the ion Larmor radius for the magnetic turbulence [19].

Heat and particle pulse propagation experiments form an important method of ascertaining the local heat conduction and particle diffusion coefficients. The standard method in many tokamaks both small and large, is to measure the heat conduction coefficient by using the heat pulse produced by the abrupt central temperature decrease at the end of a sawtooth period.

In most experiments where this has been performed this coefficient is found to be between 2 and 3 $m^2 s^{-1}$. Similarly on some experiments [20] the density pulse produced at the end of a sawtooth period also gives a particle diffusion coefficient of $2\text{-}3 m^2 s^{-1}$, possibly indicating close coupling between particle and heat diffusion. However, the thermal conduction coefficient is always somewhat larger than that which would be expected from the global confinement properties of the discharge and seems to be related to so-called incremental confinement time.

The thermal conductivity can also be measured by localised heating using electron cyclotron waves and this has been performed on the DIII [21] and DITE devices. Both devices seem to indicate heat conduction coefficients ~ $2\text{-}3 m^2 s^{-1}$ comparable with or greater than the global energy confinement time. Given the complex relation between a temperature perturbation, the heat conduction and particle diffusion, particle and heat pinch effects etc it is not clear in a turbulent plasma precisely what this observation relates to. It seems unlikely that it can be directly translated to a heat conduction coefficient.

Anomalous transport remains a complex and difficult subject in the tokamak and there certainly is a need for coordinated studies on both turbulence and transport in such devices.

6. β AND DENSITY LIMITS

Many tokamaks using additional heating have now demonstrated that the plasma pressure or β is limited to that given theoretically by Troyon [22], Sykes [23] and others [24] with an expression of the form $\beta = 2.8\ I(MA)/a(m)B_\phi(T)\%$. Figure 10 shows this relation for a variety of experiments with the PBX and DIII devices producing values up to 5%. Very recently the DIII-D device has achieved a value of 6% [25] with a value of $I/a\ B_\phi$ of 2.5 which is somewhat below the limit. However exploration of the β limit on TFTR in the so-called supershot regimes shows that at high q appropriate to these regimes the limit is set by $\varepsilon\ \beta_p \sim 0.7$ [26]. It should be noted that these limits are sensitive to the current distribution and that a number of experiments have shown that as the current profile evolves so the limiting value of β may decrease. The approach to the β limit leads to significant degradation in confinement. It is important in future devices to have sufficient β potential so that operation without impaired confinement can be obtained.

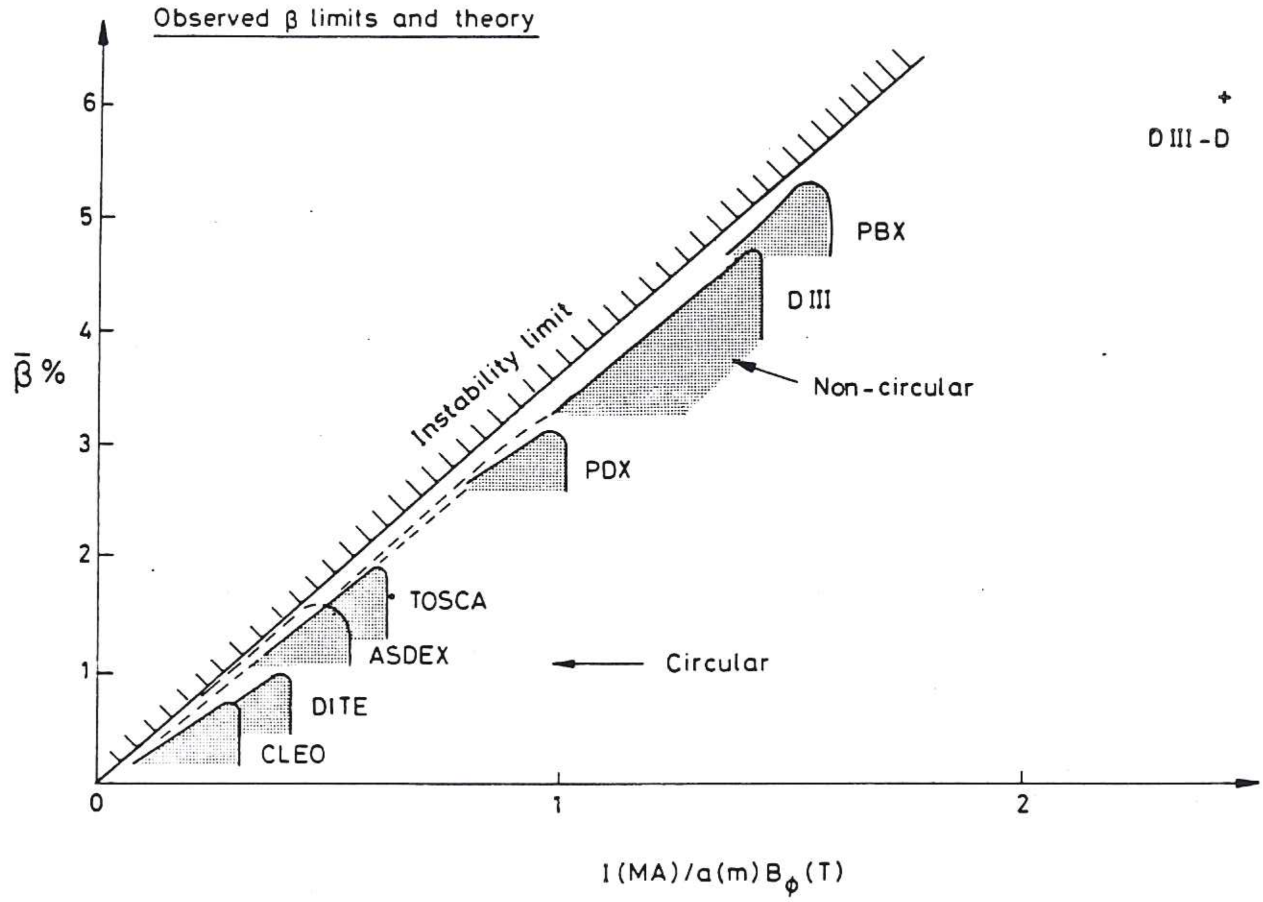

Fig 10. Average β value as a function of I/aB_ϕ for a number of circular and non circular tokamaks. The instability limit is given by $\beta = 2.8\ I\ (MA)/a(m)B_\phi(T)\%$.

The density limit has always posed a considerable constraint on the operation of all tokamaks. Typically this limit is given by $I/N = 10^{-14}$ Am - where N is the area density - the number of particles per m [27]. When the area density reaches this value the current distribution tends to contract and the current gradient inceases in the region of the q=2 surface which drives an m=2 tearing mode unstable which then leads to a disruption. In high field devices with a relatively small cross-section, high absolute densities ($\sim 10^{21} m^{-3}$) can be obtained. The density limit is found to be sensitive to the purity of the plasma. Both neutral beam heating and pellet refuelling tend to decrease the impurity content and both allow some improvement in the density limit to be obtained. However once these higher densities are obtained it becomes important to control the density decrease once the heating or fuelling source is switched off in order to terminate the discharge in a stable manner. Stabilising the instabilities at the density limit by controlling the current profile through current drive or by other means has not yet proved successful. However, a number of experiments on smaller devices, eg with ECRH on CLEO [28] and the use of resonant helical fields, have led to an improvement in the density limit of up to 70% under some circumstances.

7. MHD INSTABILITY CONTROL

Both density and q limits are characterised by a rise in low mode number MHD activity typically from a background level $\delta B/B \sim 3\ 10^{-6}$ rising to $\sim 10^{-3}$ or $\gtrsim 1\%$ of the poloidal field which corresponds to magnetic island sizes $\gtrsim 10\%$ of the plasma radius. This leads to catastrophic behaviour of the discharge - a disruption, and usually an abrupt termination of the current pulse. In many circumstances the mode associated with this activity has poloidal mode number m=2 and toroidal mode number n=1, though closely coupled with this due to toroidal effects, is activity with the m=1, n=1 and m=3, n=1. Also present is nonlinearly driven activity, very often with m=3, n=2 and lower and higher harmonics with this toroidal mode number. A particular feature noted on larger tokamaks is that this mode activity is observed to initially oscillate, associated with the rotation of the electron fluid, then to slow down and finally lock [29]. This is shown in *Fig 11* for the JET device. This locking phenomenon may be associated with the magnetic forces created by the rotating electron fluid interacting with the resistive wall of the vacuum vessel so that the rate of change of frequency is related to the square of the perturbed field, and a factor which is related to the penetration of the oscillating mode into the vacuum vessel. On the JET device this translates into a magnetic perturbation at the boundary of ~ 10G being sufficient to produce the stationary mode behaviour. For mode activity at higher amplitude the discharge ceases to rotate even in the presence of strong neutral beam injection with injected momentum. The mode may grow in amplitude until some

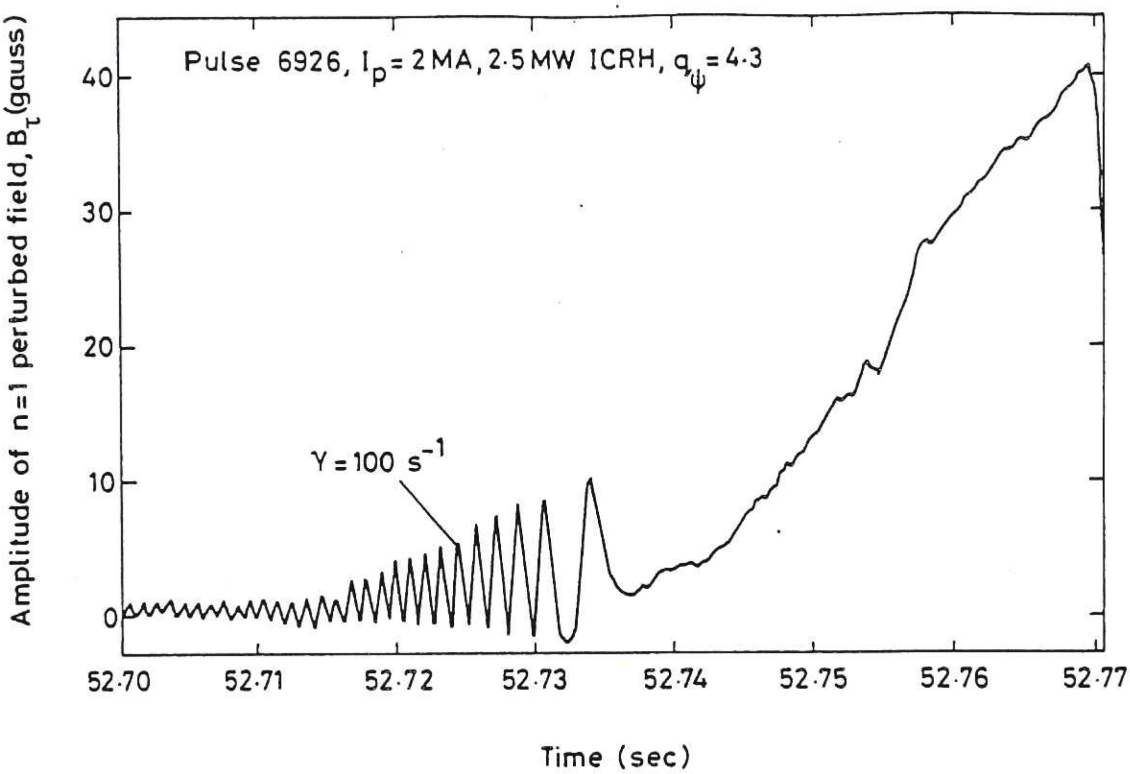

Fig 11. Amplitude of the n=1 component of the perturbed tangential field inside the JET vacuum vessel in gauss as a function of time demonstrating mode lock and the growth of a stationary helical structure to an amplitude $B_\tau/B_\theta \sim 1.5\%$ leading to disruption at $t = 52.77$sec.

considerable time after the mode has locked, and then lead to a disruption. This activity is in general not seen on smaller tokamaks because the mode growth is sufficiently rapid that it can reach the disruption perturbation level before the magnetic braking associated with the wall currents can stop the mode rotation.

Control of the m=2 activity has been attempted by a variety of means. Localised heating with ECRH in close proximity to the q=2 surface has been successful in a number of devices, T10 [30], TFR [31], CLEO [10] using electron cyclotron resonant heating. However in a reactor it is necessary to achieve control of the m=2 mode by localised current drive rather than by localised heating and so far this has not been demonstrated on a tokamak. An alternative method is to use stationary helical fields of sufficiently small size to modify the transport locally in the region of the q=2 surface and thereby alter the current gradient. This method has been demonstrated on small tokamaks [32] but has not yet been pursued on larger tokamaks. The third method exploited on the TO-1 device [33] in the Soviet Union is to directly control the growth of the m=2 mode by magnetic feedback. Such a system is planned to be used on JET in an attempt to minimise the impact of approaches to density and q limits on the mechanical and electrical integrity of the device.

An important instability which limits the central plasma parameters of the tokamak is the m=1,n=1 mode associated with the conventional sawtooth activity. There is considerable debate at the present time as to whether the central q value in a tokamak is ≈ 1 and fairly uniform with little shear in the core of the plasma, as is thought to be the case on JET [34]. In such circumstances the abrupt temperature decrease at the end of a sawtooth period is associated with a very rapidly growing ideal interchange mode in the low shear region. The more conventional picture is that the central q value decreases significantly below unity and drives a resistive tearing mode with m=1,n=1. Recent studies in toroidal geometry have demonstrated that by decreasing the current gradient in the region where q=1 that it is possible to stabilise the m=1,n=1 tearing mode by toroidal effects. Figure 12 shows the results by Hender et al [35,36] for the stabilisation of this mode using

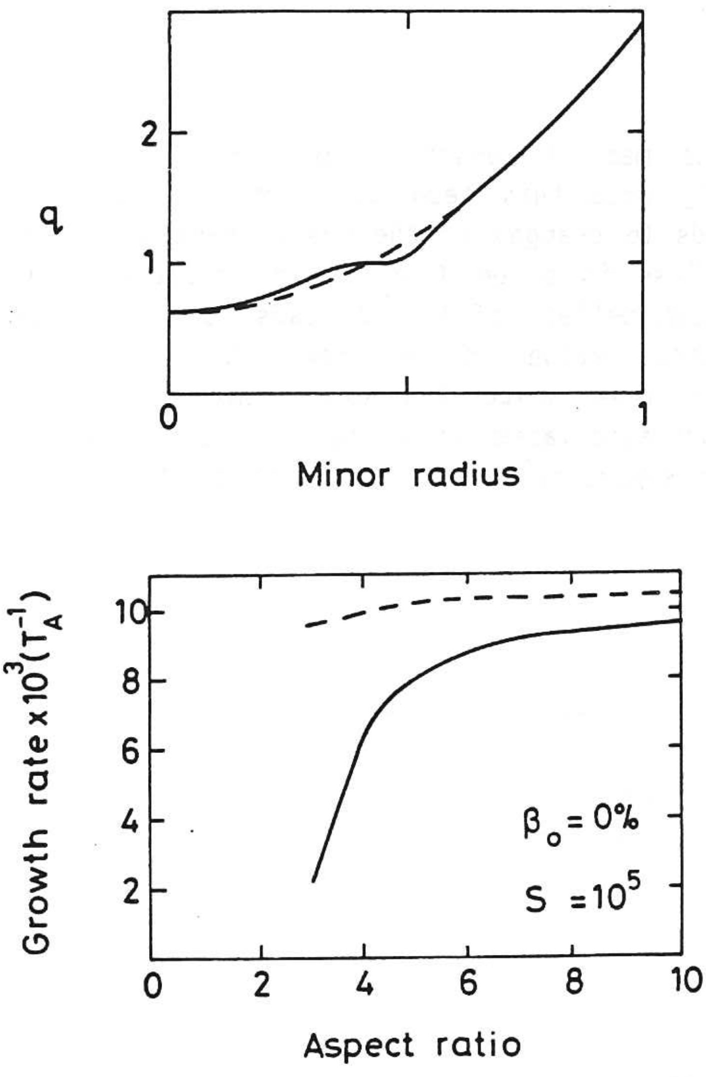

Fig 12. Stabilisation of the m=1, n=1 resistive mode by toroidal effects for a localised current perturbation around the q=1 surface. The dashed curve is for no perturbation. S is the magnetic Reynolds number. There is no plasma pressure in this case.

the localised current perturbation near q=1. Recent results from TEXTOR [37] using polarimetry indicate that the central q value is significantly below unity and that there is a localised perturbation in the region of the q=1 surface which may be in accord with the theoretical predictions. These two pictures of the core of a tokamak indicate 2 methods of control which could be exploited. If the central q value is significantly below 1 then localised current drive in the region of the q=1 surface could remove the sawtooth activity whereas in configurations where q is slightly larger than 1, then driving the current substantially outside the q=1 radius to broaden the current profile as is probably the case in lower hybrid current drive experiments, may be sufficient to remove the sawtooth activity. It is possible that control of the sawtooth activity could lead to higher values of $\hat{n}_i \hat{T}_i$ and thereby improve the chances of achieving core ignition in future large devices.

8. PELLETS

Pellet injection has made it possible to operate tokamaks at very high densities. Not only does this lead to improvement in the purity of the plasma but also leads to changes in the basic transport processes. The ion thermal conduction loss is often found to be anomalous by 2-5 times the neoclassical value but pellet refuelling leads to a transition to a regime where the neoclassical value is restored [38]. There appears to be significant evidence from scattering experiments that this enhanced ion transport anomaly is associated with the ion temperature gradient driven instability which is suppressed by the changing density gradient produced by the pellet.

9. FUTURE TRENDS

Figure 1 shows that there are 2 approaches to obtaining ignition. There is the high density, low temperature approach exemplified by TFTR pellet refuelled discharges which come close to the Bremsstrahlung limit and ohmic heating in high field devices such as Alcator C and FT. In contrast there is the hot ion mode of operation where low density high temperatures are obtained by beam heating, such as on TFTR and JET. The embodiment of the high density approach to ignition is in the Ignitor type of devices where for toroidal fields in excess of 10T it becomes possible to conceive of devices with some elongation which might ignite by ohmic heating alone on present scaling laws. A proposed machine which is similar in approach is the CIT device currently under intensive design study in the USA at present which will use a modest amount of radio frequency heating in addition to strong ohmic heating to possibly achieve ignition in the early 1990s.

There are two other variants of the tokamak which are suitable for small or medium scale investigation in the near future. The first is the tight aspect ratio approach embodied in the STX proposal from Oak Ridge which in principle allows much higher values of β than on current tokamaks, possibly $\bar{\beta} \sim 20\%$. This magnetic configuration has a relatively small toroidal field and could produce a compact reactor configuration not dissimilar to the spheromak and magnetically somewhat similar to the reversed field pinch in that the paramagnetic toroidal field variation is dominant. The tight aspect ratio however does necessitate almost complete current drive in such a device which will require significant improvement in the current drive efficiencies achieved to date. Key problems in the tight aspect ratio approach are the unfavourable neo-Alcator scaling for the ohmic background plasma and the presence of a large number of trapped particles and the impact that this will have on both transport and ohmic heating.

An alternative approach is to work at large aspect ratio whereby it is easier to access the second region of stability in a circular cross-section plasma using much higher values of β, possibly as is planned in the SRX proposal.

A further alternative is to increase the elongation of the tokamak device beyond a height to width ratio of two, to a level where the poloidal and toroidal fields become comparable as in the belt pinch. This may make it possible to achieve ohmic ignition. In this particular case the stability and control of the current distribution in highly elongated equilibria is a key problem to be overcome.

10. CONCLUSIONS

Using combinations of different heating methods which permit both global and localised heating and control of the temperature profile, using pellet refuelling to control the density profile, using a variety of methods to control impurities and thereby minimising the depletion together with current drive to minimise the magnetic activity and possibly with the potential to produce X point configurations, would seem to indicate considerable potential for further improvement of the present tritium compatible tokamaks, JET and TFTR. The potential control systems available on these devices together with developments being conducted and pursued on smaller tokamaks strongly suggest that regimes of operation with substantial α particle heating will be obtained within the next few years.

References

[1] Hawryluk, R J et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p51.

[2] Altmann, T et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p449.

[3] Alikaev, V et al, Invited paper 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987.

[4] Abe, T et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p563.

[5] Besson, G et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p617.

[6] Goldston, R J, Plasma Phys and Contr Fus, Vol 26, 1984, p87.

[7] Bitter, et al, Invited paper Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987.

[8] Luxon J, et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p159.

[9] Odajima, K, et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p151.

[10] Robinson, D C et al,Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p575

[11] Robinson, D C et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p814

[12] Tanaka, S et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p553.

[13] Tanaka, S et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p886

[14] Zarnstorff, M C et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p144.

[15] Campbell, D J et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p433.

[16] Howling A and Robinson, D C, Turbulence and anomalous transport in magnetised plasmas, Cargese Workshop, July 1986, Ed D Gresillon & M A Dubois, p115.

[17] Schoch, P M et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p 126.

[18] Robinson, D C, Turbulence and anomalous transport in magnetised plasmas, Cargese Workshop, July 1986, Ed D Gresillon & M A Dubois, p21.

[19] Wagner, F et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p222.

[20] Brower, D L et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p1314.

[21] Jahns, G L et al, Nucl Fus, Vol 26, 1986, p226.

[22] Troyon, F, Gruber, J, Phys Lett A110, 1985, p29.

[23] Sykes, A et al, Proc of 11th Euro Conf on Contr Fus and Plasma Phys, Aachen, 1983) Part 2, EPS 1983, p363.
[24] Chance, M et al, Phys Rev Lett, 51 (1983) p1963.
[25] Shimada, M et al, Invited paper, 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987
[26] McGuire, K et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p421.
[27] Robinson, D C and Todd, T N, Culham Laboratory Report CLMR-222 (1982).
[28] Todd, T N, Proc of 12th Euro Conf on Contr Fus and Plasma Phys, Budapest, 1985, supplement EPS 1985, p76.
[29] Thomas, P R et al, Plasma Phys and Contr Nucl Fus Res, 1984 (Proc 10th Int Conf, London, September, 1984) Vol 1, IAEA Vienna (1985)p353.
[30] Alikaev, V V et al, Plasma Phys and Contr Nucl Fus Res, 1984 (Proc 10th Int Conf, London, September, 1984) Vol 1, IAEA Vienna (1985) p419.
[31] De Esch, H P L et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p599.
[32] Ellis, J J et al, Plasma Phys and Contr Nucl Fus Res, 1984 (Proc 10th Int Conf, London, September 1984) Vol 1, IAEA Vienna (1985) p363.
[33] Arsenin, V V et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 7th Int Conf, Innsbruck, 1978) Vol 1, IAEA Vienna (1978) p233
[34] Wesson, J, Plasma Phys and Contr Fus, Vol 28, 1986, p243.
[35] Hender, T C et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p298.
[36] Hender, T C et al, Proc of 14th Euro Conf on Contr Fus and Plasma Phys, Madrid, June 1987, Vol II, p1145.
[37] Soltwisch, A et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p272.
[38] Greenwald, M et al, Plasma Phys and Contr Nucl Fus Res, 1986 (Proc 11th Int Conf, Kyoto, 1986) Vol 1, IAEA Vienna (1987) p139.