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JET RESULTS AND THE FUTURE PROSPECTS FOR FUSION

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What are the scientific results from tokamak magnetic confinement systems and what prospects are there for their future? The main aim of such systems is to achieve plasma conditions as close as possible to those required for a reactor based on the deuterium tritium fusion reaction:

$$\underset{}{D + T = \underset{(3.5\text{Mev})}{He^4} + \underset{(14.1\text{Mev})}{n}}$$

This system would ignite when the energy liberated in α-particles balances the losses from the plasma by radiation, conduction and convection. The α-particles are assumed to be confined by the magnetic field in the plasma core. The 14 MeV neutrons will escape freely from the plasma; in an eventual reactor they would be absorbed in an external blanket, generating heat.

Once ignited the plasma is self-sustaining needing only fresh fuel and the helium ash removing. For a deuterium-tritium plasma the ignition conditions are that the central ion temperature $\hat{T}_i$ has to be between 10-20keV and the parameter $\hat{n}\ \hat{T}_i\ \tau_\epsilon$ has to equal $5 \times 10^{21}\ m^{-3}$keVs where n is the central ion density and τ_ϵ the global energy confinement time ($1\text{keV} = 10^7\,^\circ K$). τ_ϵ measures the thermal insulation of the plasma and is defined as

$$\tau_\epsilon = \frac{W}{p}$$

where W is the total kinetic energy content of the plasma and P the power required to sustain it in a steady state.

Magnetic Confinement

The plasma has to be hot and dense enough in the centre to produce the maximum number of reactions, and at the same time sufficiently rare and cool at the edges to be compatible with plasma wall contact. With toroidal magnetic confinement the plasma is held in pressure equilibrium by the plasma currents interacting with the magnetic field.

The system must be topologically toroidal or tyre-like because it is only in this geometry that the necessary onion-skin layers of magnetic surfaces can be produced. These are the surfaces traced out by field lines and they have the property that everywhere on a surface the component of the field normal to the surface is zero.

The tokamak (figure 1) is a particular example of such a toroidal magnetic confinement system. The safety factor q is a key parameter which at a radius r in the plasma is defined as

$$q(r) = \frac{\text{number of revolutions about major axis}}{\text{number of revolutions about minor axis}}$$

made by field lines tracing out a magnetic surface of radius r. A tokamak is defined as a system in which the safety factor substantially exceeds unity at the edge of the plasma, (typically 2-5).

The tokamak system was pioneered in the Soviet Union during the 1960s. Taken up enthusiastically by the West in the 1970s it is now virtually certain to be the first magnetic confinement system in which ignition conditions are achieved.

Breakeven Conditions

The so-called breakeven conditions for a tokamak is defined as

$$Q = \frac{P_{fus}}{P_{IN}} = 1$$

where P_{fus} is the total power output in neutrons and α-particles from fusion reactions and P_{IN} is the steady state power input sustaining the plasma. This breakeven condition is merely a useful benchmark. It does not imply system breakeven where the power used to drive the coils etc. would have to be included. Q=1 corresponds to an α-power into the plasma of 17% of the losses which is about the lowest level for detectable effects. For thermonuclear breakeven the temperature condition is the same as for ignition while the requirement on $(\hat{n}\ \hat{T}_i\ \tau_\epsilon)$ is relaxed by a factor five to 10^{21} m^{-3}keVs.

When the plasma is being heated by external power input, several methods in use create non-thermal ion velocity distributions, enhancing the reaction rate and further reducing the condition for breakeven - typically by a factor 2.

The Three Large Tokamaks

Three large tokamaks, which aim to reach breakeven conditions, have been built:

1) Tokamak Fusion Test Reactor (TFTR) in the US;
2) The Joint European Torus (JET) in the EEC;
3) JT-60 in Japan.

They started experimental operation in 1982, 83 and 85 respectively and the main parameters and the plasma cross-sections are shown in figure 2.

JET and TFTR are both designed to use deuterium-tritium (D-T) mixtures, demonstrating directly the influence of α-particle heating. JT-60 in contrast is to demonstrate 'equivalent' breakeven conditions using hydrogen plasmas.

Tokamak plasma heating

A number of different methods have been developed to heat the plasma. RF power can be launched into the plasma in frequency ranges which are absorbed resonantly in the plasma interior. The three main ranges used are 20-80MHz, 1-5GHz and 90-120GHz. Power sources, tetrodes, klystrons and gyrotrons respectively are available for these frequencies although for the highest frequency the unit size of presently available gyrotrons is rather small at 200kW. By contrast launching these electromagnetic waves into the plasma is more difficult for the two lower frequency ranges.

Another technique involves injecting high energy neutral atoms into the plasma. These cross the magnetic fields until they are ionised inside the plasma; there they are trapped and give their energy to the bulk ions and electrons by Coulomb collisions. Through the use of a multi-aperature ion source, neutral beams of 60A equivalent at 80keV are typical of today's technology.

Table I lists the additional heating capability of the three large tokamaks. None of the three uses the electron cyclotron resonance method, essentially because of the lack of suitable large gyrotrons.

Experimental results

All three large machines have been very successful technically, in fact, they have all exceeded their design rating in some respects. This is particularly so for JET where the plasma current has been raised to 6MA compared with the design figure of 4.8MA.

These large tokamaks have one key property that is the same as the smaller ones - a disruptive instability if the safety factor at the boundary is too low or if the plasma density is too high. This instability produces a rapid loss of plasma energy followed by an uncontrollable fall to zero of the plasma current. In a JET case, for example, 5MJ of plasma energy is lost in 300μs while the current falls from 5MA to zero in 10ms. Such disruptions induce eddy currents in surrounding metallic components giving rise to large pulsed forces on them, e.g. $\sim$1000 tonnes on the JET vacuum vessel.

This instability is a major long term problem for tokamaks. In principle it can be eliminated by avoiding extremes of high plasma current or particle density. In practice, however, there are hidden variables such as the impurity content of the plasma; a small flake of wall material falling into the plasma can trigger the instability even when the density and current are in the normally stable range.

Experiments on detecting an incipient instability in order to take feedback action to stabilise it or at least to reduce its severity will begin on JET in 1989.

The tokamak performance is characterised by Q - the ratio of the thermonuclear power (P_{th}) that would be generated in a deuterium-tritium plasma to the power (P_{IN}) required to sustain the plasma. At the point of ignition $Q \to \infty$.

All the present experiments are with deuterium, hydrogen or helium plasmas. However we can calculate what the D-T performance would be on the basis of reactions between the Maxwellian ion populations and the known cross-sections for D-T reactions.

Thus the results are presented in the $\hat{n}_D\hat{T}_i\tau_\epsilon$ against the $\hat{T}_i$ plane, where $\hat{n}_D$ and $\hat{T}_i$ are the central deuteron density and temperature. Contours of constant thermonuclear Q can be plotted in this plane using the range 0.1-1.0 for present experiments. JET results with only ohmic heating are shown at various discharge current levels in figure 3. You can see that the so-called fusion parameter, $X = \hat{n}_D\hat{T}_i\tau_\epsilon$ increases approximately linearly with the plasma current, but with only Ohmic heating the ion temperature is limited to values far below the optimum for maximising Q.

When additional heating is applied at the 5-20MW level using either the ion cyclotron resonance, neutral beam injection or both combined, the ion temperature is raised as shown in figure 4. The ion temperature has been increased to 14keV and the electron temperature to 10keV in this way at JET. In the so-called supershot experiments in TFTR, ion temperatures up to 28keV have been achieved using intense neutral beam heating in low density discharges.

Although the ion temperature in JET has increased by a factor of ~ 4, the fusion parameter X stays constant. This is due to the very important phenomenon of confinement degradation, first seen clearly on the smaller French tokamak TFR a decade ago. As the power input is increased so the energy confinement time is progressively reduced or degraded. Figure 5 shows the results for JET, the lines drawn through the data correspond to the empirical relationship,

$$\tau_\epsilon = \text{constant } I_p P^{-1/2} L^{\alpha}$$

where L is a length scale, constant in JET but which can be checked by comparison with smaller machines. This gives $1.5 > \alpha > 1.0$. The importance of τ_ϵ is highlighted by noting that

$$X \propto \frac{P \tau_\epsilon^2}{(\text{Plasma Volume})}$$

$$\propto \frac{I^2 L^{2\alpha}}{(\text{Plasma Volume})}$$

Thus X is a characteristic of the apparatus since the maximum current capacity is related to the toroidal field strength and the physical size. In this picture the function of the additional power is simply to take the central temperature into the optimum range of 10–20 keV.

The results so far were obtained with the plasma boundary determined by an outer magnetic surface intersecting a material limiter. It is possible, in principle, to use a magnetic separatrix to determine the boundary; two schemes are sketched in figure 6. Some years ago an improved mode of confinement was discovered on the German ASDEX tokamak. For this, the sufficient conditions were a separatrix-bounded plasma and an input neutral beam power above a certain threshold value. Although JET was not designed for this method it has been possible to change the currents in external coils so as to create plasmas with a magnetic separatrix at current levels up to 5MA. Thus the 'H' or 'high' mode of confinement has been achieved with neutral beam injection and the results show an approximate two-fold increase in the energy confinement time.

The best results in the $\hat{n}\,\hat{T}_i\,\tau_\epsilon$ vs $\hat{T}_i$ plane for each of the three large machines are summarised in figure 7. Conditions corresponding to a thermonuclear Q ∿ 0.2 have been achieved in JET and with further additions to the machine to control radial profiles of current, density and temperature Q ∿ 0.5 can be expected. There will also be an additional α-particle production, due to the presence of fast injected or accelerated ions, which is expected to double the total Q so that it approaches unity. This should be sufficient to enable JET to reach its declared aim - the study of α-particle confinement and heating in the eventual operation with a deuterium-tritium plasma.

Theoretical understanding

Losses of energy from the plasma by conduction and convection are significantly larger than predicted by stable plasma theory in which transport is solely due to interparticle collisions. However the theory does predict that a confined tokamak plasma will be unstable to a range of instabilities. Their non-linear behaviour however is difficult to calculate and measurement of fluctuations and their correlation is difficult in the interior of hot plasmas. The net result is that, although there is a wide range of linear instabilities to choose from, it is not possible using theory alone, to predict the performance of a particular system. Empirical scaling laws based on experimental results and some theoretical constraints must be used. By contrast the theory of heating physics and of limits on the ratio of plasma to magnetic pressure is in relatively good agreement with the experiments.

Compared with the range of earlier experiments the extrapolation from JET to the next step is relatively small. However, in view of the costs involved, the accuracy required in predicting performance is high if a wasteful design is to be avoided.

Ignition experiments

The next logical step in the development is to demonstrate a burning plasma, i.e. one sustained energetically by thermonuclear reactions. Intense and controversial discussions are taking place on the best course to take. At one extreme are very high field (>10T), small physical-size copper coil experiments pioneered by Professor Bruno Coppi of MIT. A new design study on this 'Ignitor' line has just started in Italy while a similar Compact Ignition Tokamak (CIT) is under consideration in the USA. Such copper-coil tokamaks may be the cheapest ways to demonstrate ignition but they are limited in pulse length and of debatable relevance to an eventual reactor.

This is envisaged as using superconducting coils and a blanket ∿ 2m thick to absorb neutrons, generate tritium and shield the magnetic field coils. This leads to moderate fields 5T, large physical size and steady state operation.

Thus the Next European Torus (NET) and ITER proposals are for 20MA plasma current, superconducting coil machines, designed to have an ignited plasma and a high availability for radiation damage and other technological studies. NET is a joint venture in the European Economic Community (EEC), while ITER is a four bloc collaboration initiated by the Reagan-Gorbachov summits (EEC, USA, USSR and Japan). Both are in the conceptual design phase at present, are likely to cost several billion dollars and due to start operation early in the next century - they would not generate electricity. The precise aims of NET & ITER are still being defined but in general they are to operate with ignited, deuterium-tritium plasmas for long pulses. The energy confinement must be adequate and the problems of plasma fuelling and helium ash removal solved. The main technical elements of an eventual fusion reactor would be tested including large scale superconducting coils, blanket and shield engineering, remote maintenance, tritium handling, power handling elements in the vacuum vessel, disruption forces and probably non- inductive methods of driving the plasma current in steady state. Years of operation with high availability would be needed to achieve the required neutron fluence for radiation damage studies.

Since the aims of ITER and NET are similar it seems that NET will be dropped if ITER goes ahead. The construction of ITER would involve collaboration between four very different political, economic and industrial systems. It is clear that a major factor will be the relationship between the two superpowers. The ITER team is required to produce a machine design by the end of 1990 and so the decision to build or not could be taken in 1991. With an estimated construction time of 7 years it seems unlikely that ITER (or NET) could be operating before the year 2000. An incidental point is that with JET scheduled to stop work in 1992 there will be a significant interruption in experimental studies of reactor grade plasmas.

Before such large projects are launched there will quite properly be a requirement to assess the long term potential of fusion for electricity generation. Fusion has some clear intrinsic advantages in safety over fission and a reduced environmental impact compared to fission or coal. The capital cost of the nuclear core of a fusion reactor seems likely to be high in view of its complexity. However with negligible fuel costs the final unit cost of electricity should be in the same range as that for the other systems.

If successful, the NET/ITER generation of machines would be followed by a demonstration power generating system. Following a sequential time scale of 20 years for each stage means that commercial exploitation cannot be expected before the middle of the next century. However the programme could be speeded up if the forecast shortage of oil and gas materialises, together with objections to near-total reliance on nuclear fission and coal.

Conclusions

The JET tokamak has achieved plasma conditions which in a deuterium-tritium plasma would give a thermonuclear output equal to 20% of the power input to the plasma. With the planned modifications to the apparatus, this is expected to rise to at least 50% in the next two years. Reactions between non-thermal ions will contribute another 50% enabling JET to fulfil one of the main aims laid down in 1973, namely to study α-particle confinement and heating.

This programme should be completed in the next four years and it is becoming increasingly urgent to launch the next stage machines if the momentum is not to be lost. International collaboration in this field is already excellent and is expected to be extended further in order to spread costs as widely as possible. The evidence from JET and other large tokamaks suggests that from the confinement physics viewpoint a tokamak fusion reactor of reasonable size ($\sim$ 1GW electric) is feasible.

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TABLE 1

Plasma Heating Systems

	JET	TFTR	JT-60
Ion Cyclotron Resonance Heating	20 MW	8 MW	2.5 MW
Lower Hybrid Resonance Heating	10 MW	None	7.5 MW
Neutral Beam Injection Heating	20 MW 80 kV Deuterium	27 MW 120 kV Deuterium	20 MW 100 kV Hydrogen

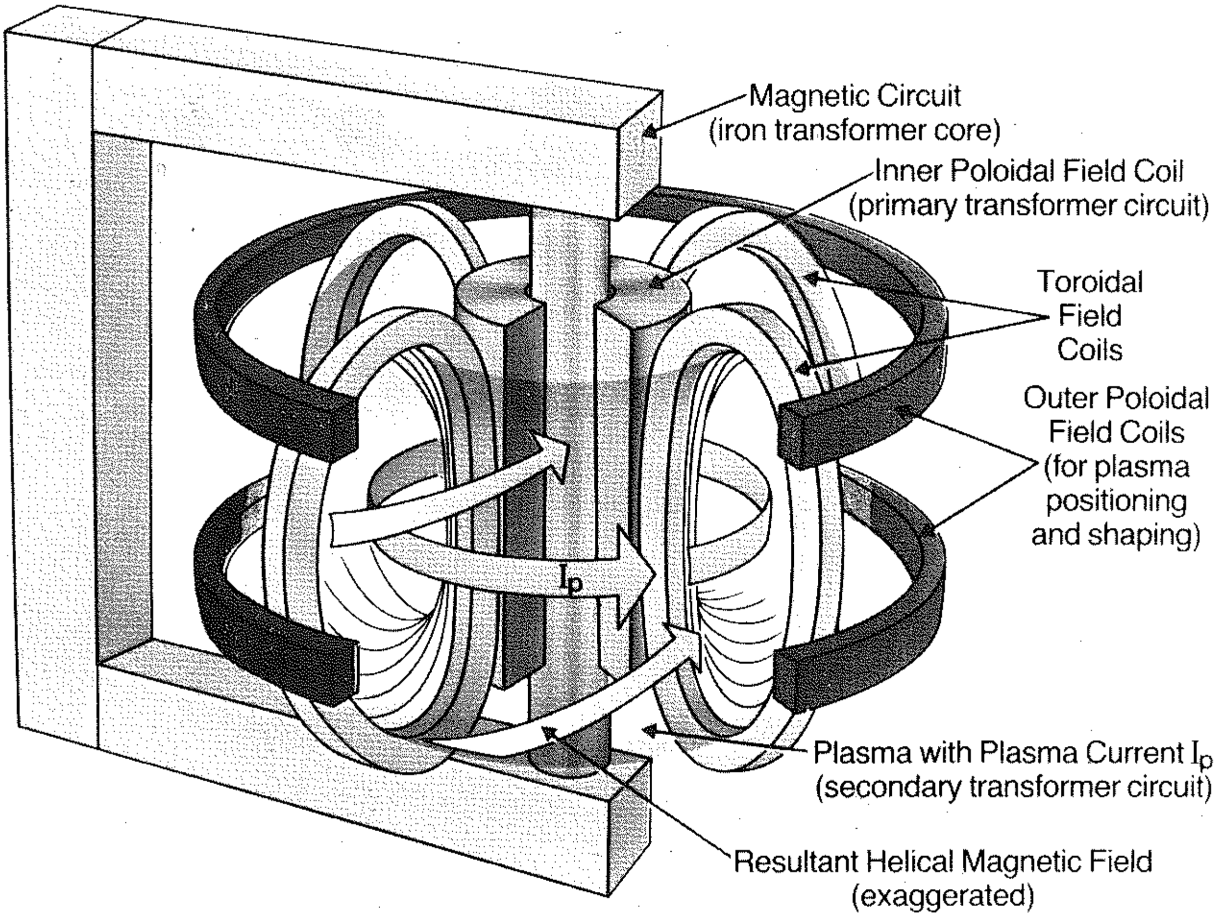

Fig. 1 Schematic view of a tokamak. This figure shows the pulse transformer used to induce a plasma current I_p parallel to the large externally applied toroidal field, B_ϕ. Outer poloidal field coils shape the cross-section of the current-carrying plasma ring and maintain it in radial and vertical equilibrium. Not shown is the vacuum vessel inside the toroidal field coils.

THE THREE LARGE TOKAMAKS IN THE WORLD

TFTR USA	
Minor Radius	0.85m
Major Radius	2.48m
Toroidal Magnetic Field	5.0 T
Plasma Current	3.0 MA
Pulse Length	2 s.
Additional Heating Power	27 MW (total)
D-T Operation Capability	
1st Plasma	Dec 1982

JET EUROPE	
Minor Radius	1.25m (Horz) 2.1m (Vert)
Major Radius	2.96m
Toroidal Magnetic Field	3.5 T
Plasma Current	5.0 MA
Pulse Length	20 s
Additional Heating Power	44 MW (total) [25 MW high grade]
D-T Operation Capability	
1st Plasma	June 1983

JT-60 JAPAN	
Minor Radius	0.95m
Major Radius	3.0m
Toroidal Magnetic Field	4.5 T
Plasma Current	2.7 MA
Pulse Length	5-10 s
Additional Heating Power	54 MW (total) [30 MW high grade]
Magnetic Limiter Configuration	
1st Plasma	April 1985

Fig. 2 This figure shows the cross-sections of the plasmas in the three large tokamaks together with the main parameters.

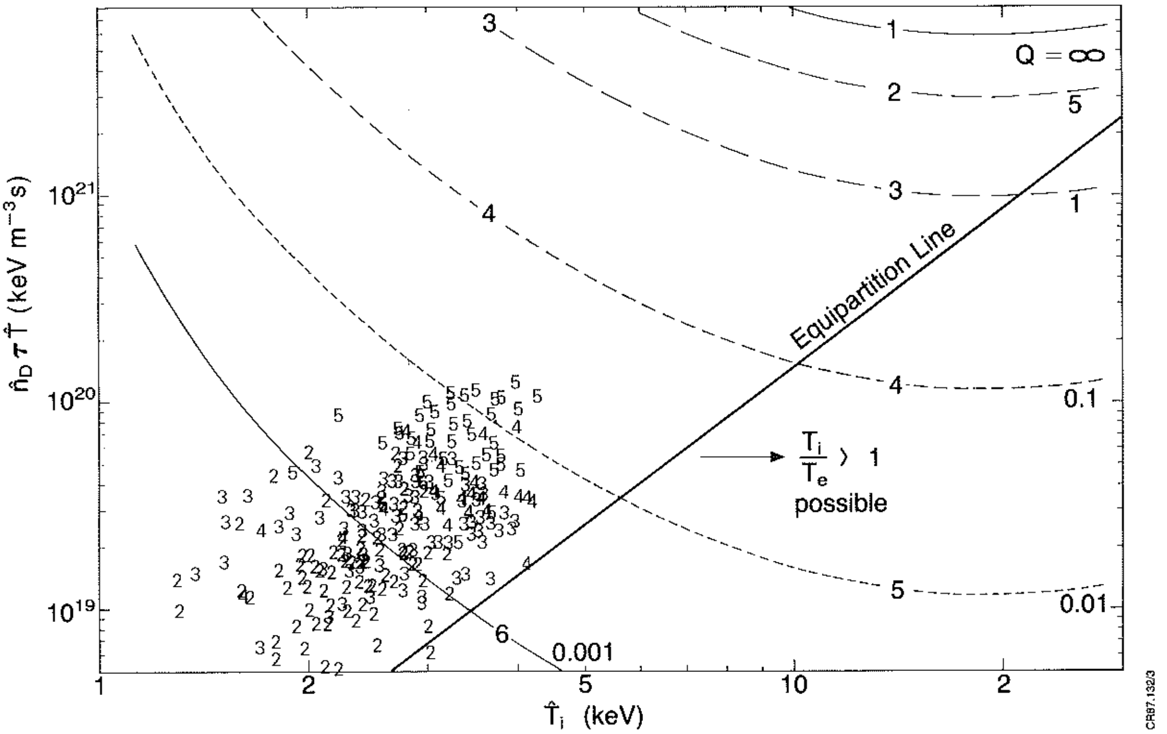

Fig. 3 Data for ohmically-heated plasmas in the $\hat{n}_D\ \tau_E\ \hat{T}_i$ vs $\hat{T}_i$ plane. Numbers refer to the current level in megamperes. Contours show lines of constant thermonuclear Q for a deuterium-tritium plasma.

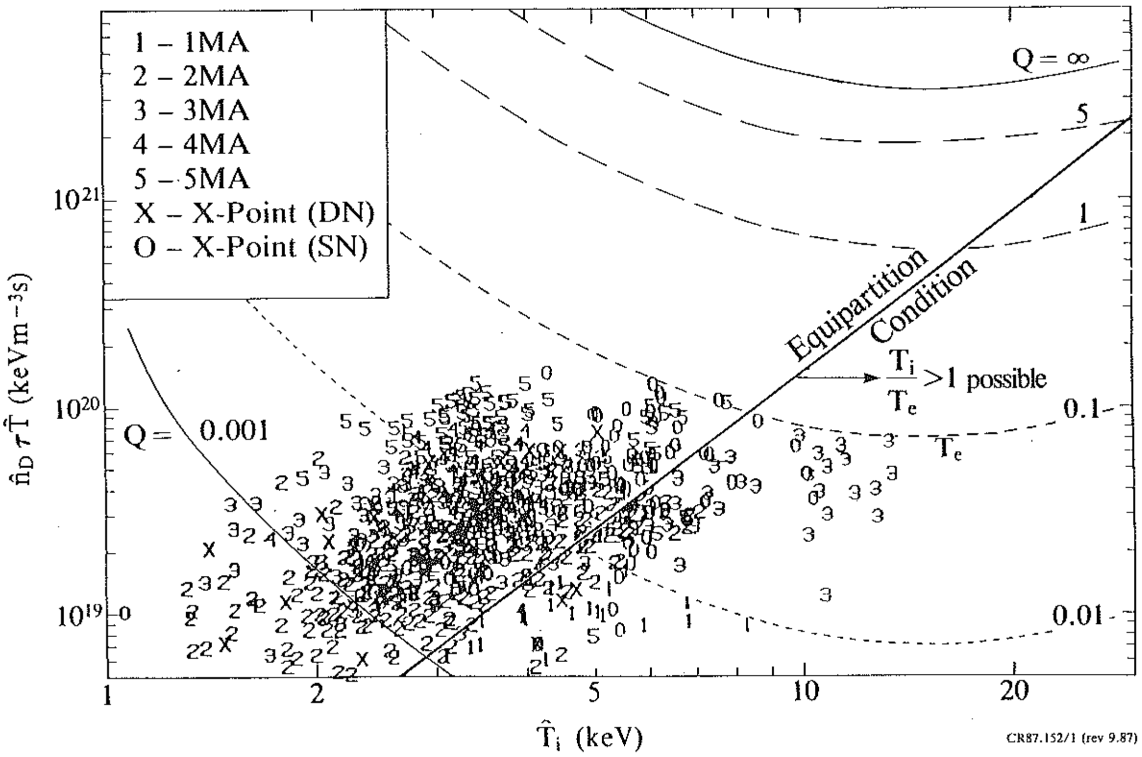

Fig. 4 Data for additionally heated discharges in the $\hat{n}_D\ \tau_E\ \hat{T}_i$ vs $\hat{T}_i$ plane. Numbers denote the plasma current in megamperes. Points with symbol (o) are for separatrix-bounded plasmas at 1 or 2MA.

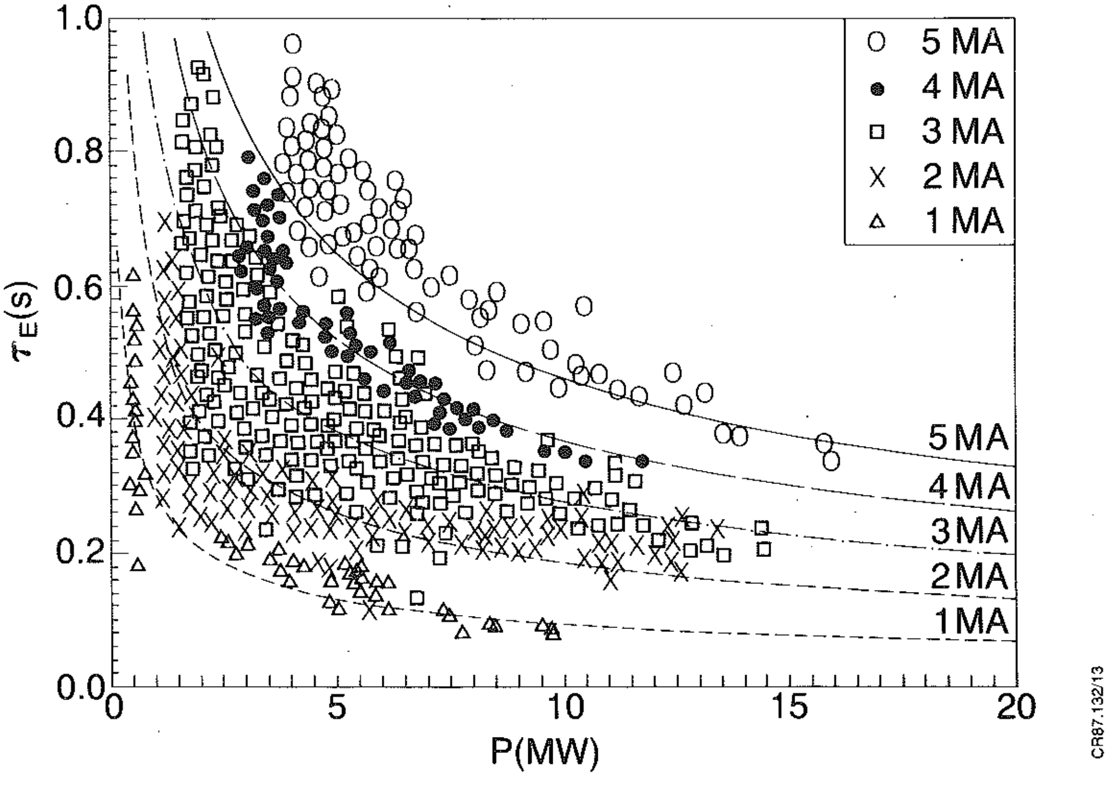

Fig. 5 Confinement time τ_E (s) versus input power P(MW) for various plasma currents. Continuous lines correspond to empirical scaling $\tau_E \propto I P^{-1/2}$.

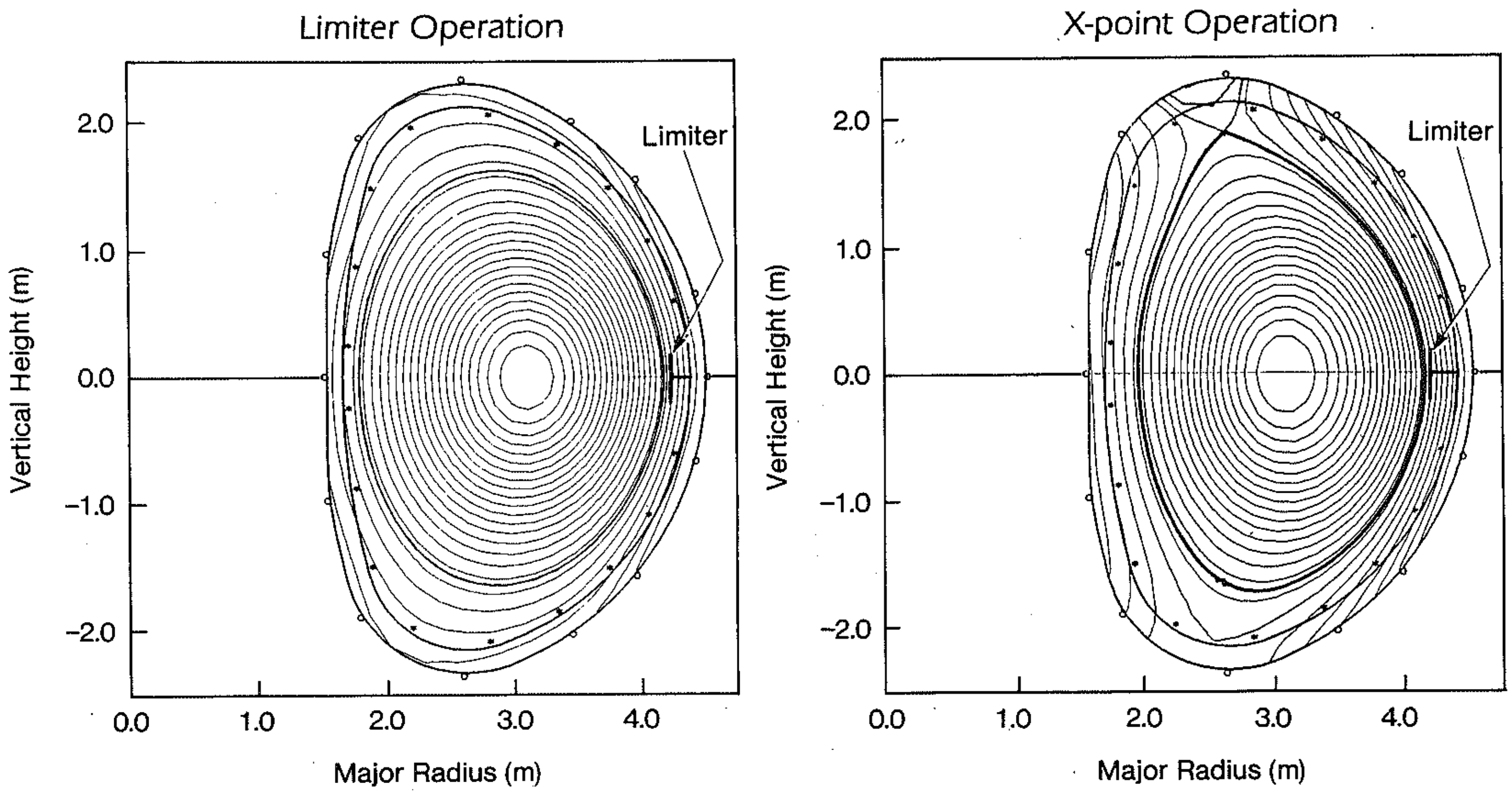

Fig. 6 Cross-section of magnetic surfaces for the two cases of limiter-bounded and separatrix-bounded plasma. In the limiter-bounded case the last-closed surface intersects the limiter. In the separatrix case the boundary is determined by the X-point (separatrix) in the magnetic surfaces.

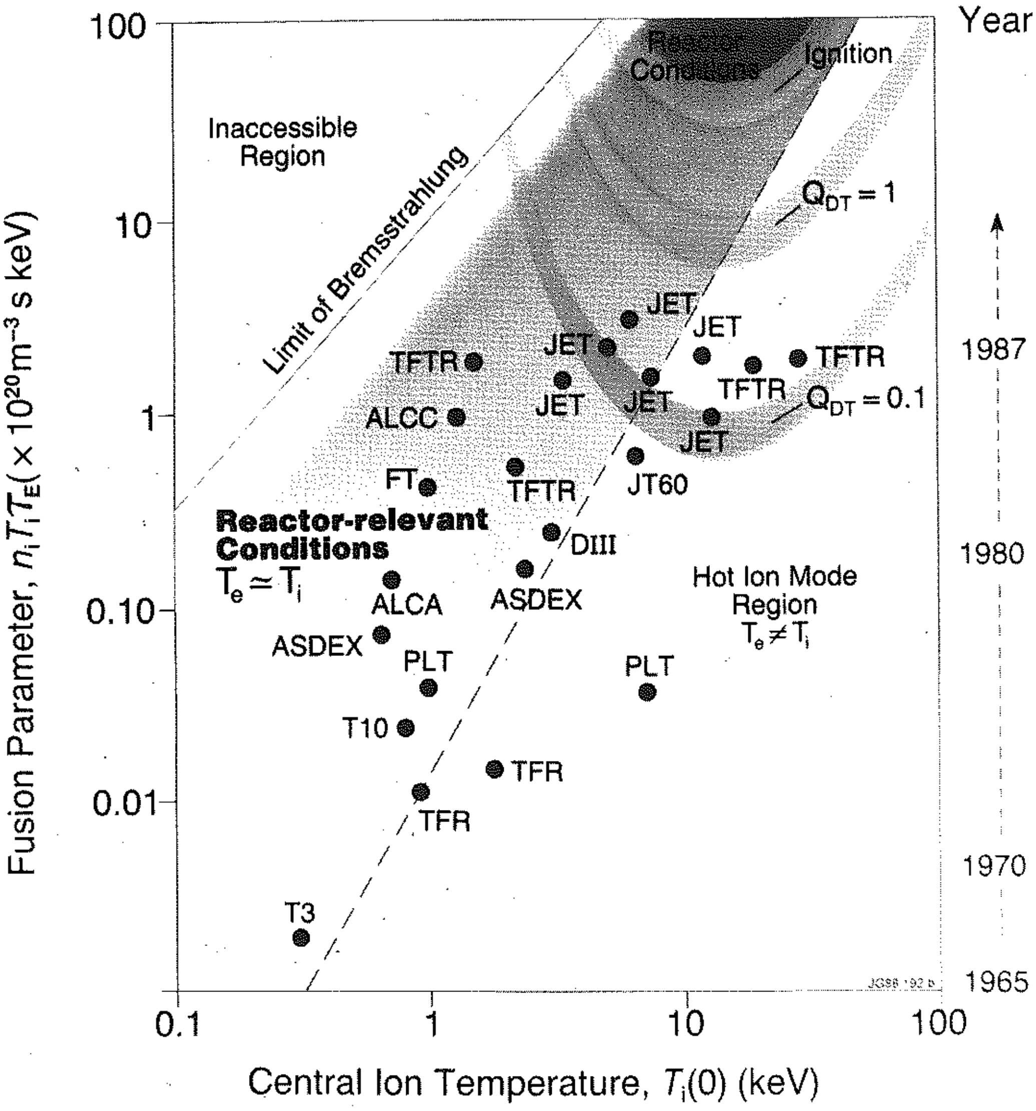

Fig. 7 Summary figure showing tokamak performance in the $\hat{n}\ \tau_E\ \hat{T}_i$ vs $\hat{T}_i$ plane. Apart from JET, TFTR and JT-60 the results from many smaller and earlier tokamaks are plotted. From the right-hand scale it can be seen that $\hat{n}\ \tau_E\ \hat{T}_i$ has been increased by three orders of magnitude over twenty years.