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The JET Joint Undertaking: Scientific Advances and Future Challenges

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ABSTRACT

JET, the flagship of the European fusion programme has obtained far-reaching results in all domains of tokamak physics. Results on confinement (scaling laws, confinement regimes), fusion performance (including D/T mixture), Divertor aspects and external current drive are described placing the emphasis on the consequence for the future JET programme and for the design parameters of the next step. After completion of major enhancement of its in-vessel components, JET will address the problems of combining high performance operation with radiating divertors and investigate advanced tokamak scenarios using profile control.

CONTENTS

Page No

I. INTRODUCTION

Maastricht, the home of the famous treaty which is close to the location of this conference, Carolus Magnus, the legendary European king, whose name was chosen by your school, and finally JET, a Joint European Undertaking, the subject of this special lecture are three European ventures very unequal in magnitude and degree of contemporariness. However, they share a common European motivation. On the modest JET side, I find it most pleasurable to speak in the context you have chosen and would like to thank you for offering such a possibility. I do not have to emphasise that European scientists working together can move ahead at considerable speed; it is even generally most enjoyable and I have yet to come across a complaint for loss of national identity!

JET is a joint undertaking of all EC members plus Sweden and Switzerland. It satisfied the principle of "subsidiarity" long before the word became fashionable: In mid 1978 when the construction started, it would have been difficult for a single nation to gather the knowledge and the resources for this enterprise. A machine was proposed with dimensions capable of confining α-particles at the birth energy for D/T reactions. This implies that I_P R/a > 7MA (I_P: plasma current, R: major radius, a: minor radius) in order that the orbits of the 3.5 MeV alphas do not interact with the vessel walls. The JET objectives were formally set to be:

a) the study of scaling of plasma behaviour as parameters approach the reactor range,
b) the study of plasma-wall interaction in these conditions,
c) the study of plasma heating and
d) the study of α-particle production, confinement and resulting plasma heating.

Objective d) requires operating in a Deuterium-Tritium mixture in high performance discharges eg with $Q = P_{fusion}/P_{input} > 1$ so that heating effects due to α-particles which involve only 1/5 of Pfusion can be distinguished from Pinput. A significant nuclear activation of the machine structures would occur after a small number of these high performance discharges, making maintenance difficult. Therefore objective d) is best pursued at the very end of the JET programme or during isolated experiments of a few shots as in the well known Preliminary Tritium Experiment performed by JET in November 1991.

Objectives a), b) and c) have been an integral part of the JET programme since it started plasma operation in June 1983 and a considerable amount of data is available. In this presentation, I will first make a short description of JET before highlighting the results which are most pertinent to future developments; I will underline the specific actions undertaken by JET to meet the challenges of the future.

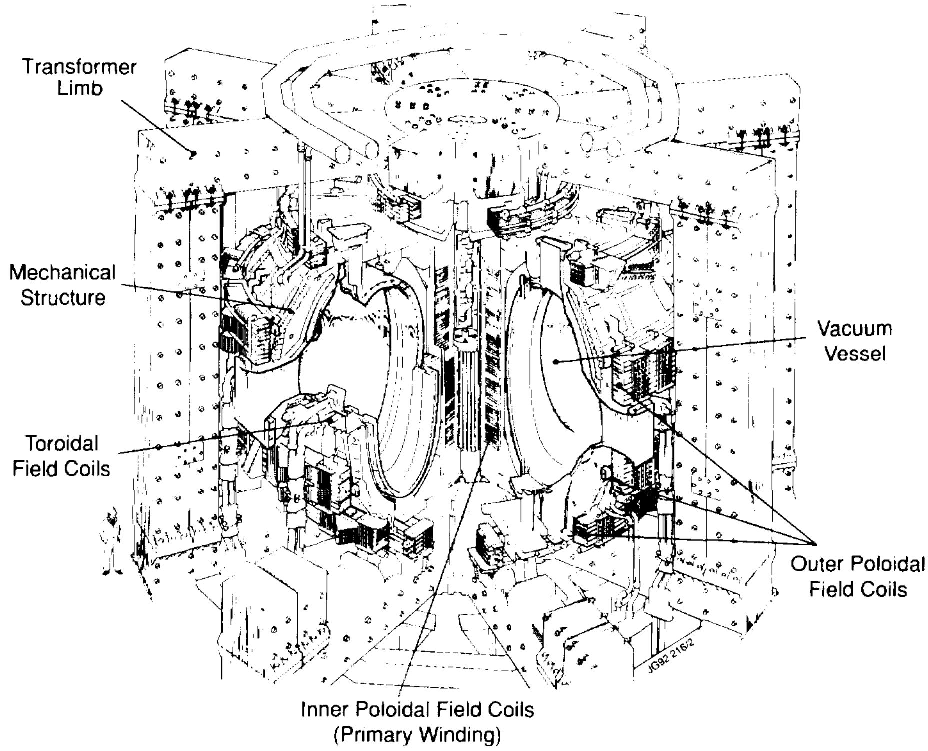

Fig 1. A diagram of the JET tokamak.

II. THE JET APPARATUS

The original conception of the JET apparatus[1] departed in many major elements from conventional tokamaks. In particular, circular cross-sections used to prevail at that time. The D-shape cross-section which was chosen (Fig 1) for JET provided a number of technical advantages (toroidal field coils in pure tension, larger volume to surface ratio, etc). It later turned out that a crucial advantage resulting from this choice was the flexibility in tailoring the magnetic configuration. In particular the possibility of creating a magnetic separatrix entirely contained in the vessel offers considerable advantages of enhanced confinement in the H-mode (an advantage which was not known at that time). Moreover, this configuration can be used to divert the plasma in order to control the exhaust of power and particles (magnetic divertor configuration). The magnetic divertor[2] allows to separate and to a large extent isolate the exhaust region (outside the separatrix) from the confinement (inside the separatrix).

A diagram of JET is shown in Fig 1 and the principal parameters are given in Table I. The overall dimensions of the Tokamak is about 15 m in diameter and 12 m in height. At the heart of the machine, there is a toroidal vacuum vessel of a major radius of 3 m with the above-mentioned D shape cross-section 2.5 m wide by 4.2 m high. The machine is operated by introducing a small quantity of Hydrogen, Helium or Deuterium (Tritium at a later stage) which is ionised and heated by passing a large current through the gas. The poloidal magnetic field created by the plasma current combines with the field created by external coils mainly the toroidal field (3.4 T in JET) to produce helical lines of force which form nested toroidal surfaces confining the plasma.

Table I
Principal Parameters

Parameter	Value
Plasma current I_P	7 MA (limiter)
	5 MA (X-point)
Plasma minor radius (horizontal), a	1.25 m
Plasma minor radius (vertical), b	2.1 m
Plasma major radius; R_o	3.1 m
Toroidal magnetic field	3.45 T
Weight of iron core	2800 t
Weight of vacuum vessel	108 t
Weight of toroidal field coils	364 t

Originally the machine was designed to carry a plasma current of 4.8 MA but has already been modified and tested to achieve 7 MA. This current is induced by transformer action using the massive eight limbed magnetic circuit or created externally using travelling waves (frequency: 3.7 GHz for Lower Hybrid Current Drive, LHCD, or 23 to 57 MHz for Fast Wave Current Drive (FWCD). A set of coils around the centre limb forms the primary winding of the transformer with the plasma acting as the single turn secondary. The main heating power is provided by propagation and absorption of high power radio frequency waves in the Ion Cyclotron Resonance Heating (ICRH) and by the injection of beams of energetic neutral atoms into the Torus (Neutral Beam Injection - NBI). The main parameters of the heating and current drive systems are shown in Table II.

Table II
The Three Large Additional Heating Power Systems

High Power Auxiliary Systems	Achieved Power (MW)	Other Parameters
Ion Cyclotron Resonance Heating (ICRH), also used for Fast Wave Current Drive (FWCD)	21	• 23 to 57 MHz • 8 antenna modules • Uses 16 high power tetrodes
Neutral Beam Injection (NBI) mainly for ion heating	21	• Beam acceleration to 70-140 keV • 16 ion sources grouped into 2 large injection boxes
Lower Hybrid Current Drive (LHCD) for generating the plasma current and electron heating	2.7 (10 possibly in 1994)	• uses 24 Klystrons at 3.7 GHz • one launcher groups 384 phased waveguides • the launcher is moved during the shot under feedback control

The interior of JET as it was at the beginning of 1992 is shown in Fig 2. The sections of the vessel walls exposed to contact with the plasma are protected by beryllium tiles or by graphite tiles reinforced with carbon fibres. The plasma was programmed to interact either with two toroidal belt limiters located on the outside walls or with divertor plates located on the top and bottom of the vessel.

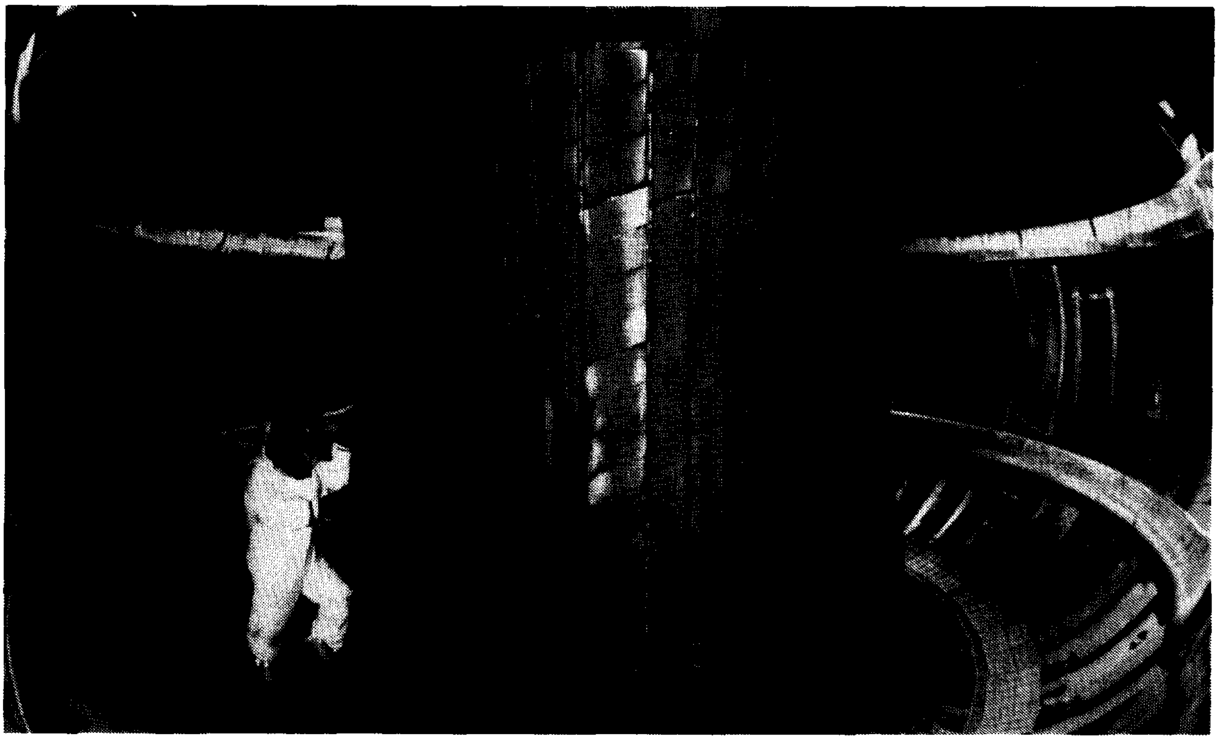

Fig 2: The interior of the JET vacuum vessel

The plasma cross sections in two configurations used in the experiments are represented in Fig 3. The figure also illustrates the nested magnetic surfaces generated in each case; in Fig 3b an X-point is formed near the top of the vessel and power and particle exhaust are channelled in the outer X-point region.

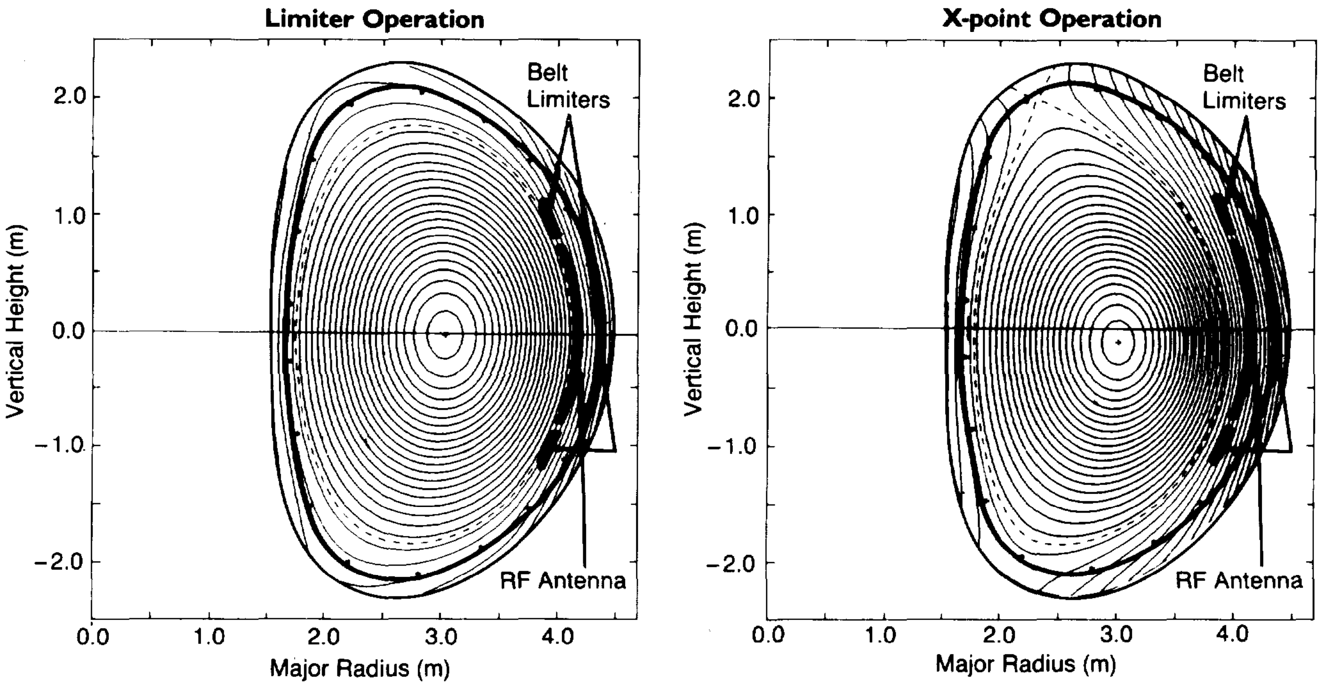

Fig 3: An illustration of the nested magnetic surfaces in limiter and X-point configurations.

The highest JET performance was obtained in the single null X-point configuration. The configuration has also been selected by ITER for its engineering design activities.

In 1992, JET entered a long shutdown to enhance the device capabilities in this mode and, to this effect, four internal divertor coils (Fig 4) have been constructed and installed in the bottom part of the machine. Divertor target plates capable of enhanced power loading will be precisely aligned in the divertor region. A large in-vessel cryogenic pump is also being installed. The new equipment will allow JET to operate at high current - 6 MA - with a single null X-point well above the divertor target plates, leaving a substantial divertor volume for the particle and exhaust processes to take place in a controlled way.

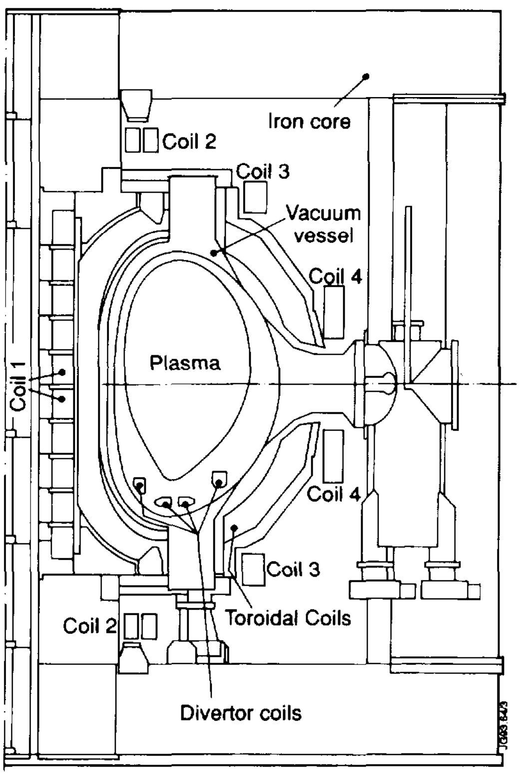

Fig 4: Cross-section of JET showing the four new internal divertor coils under construction.

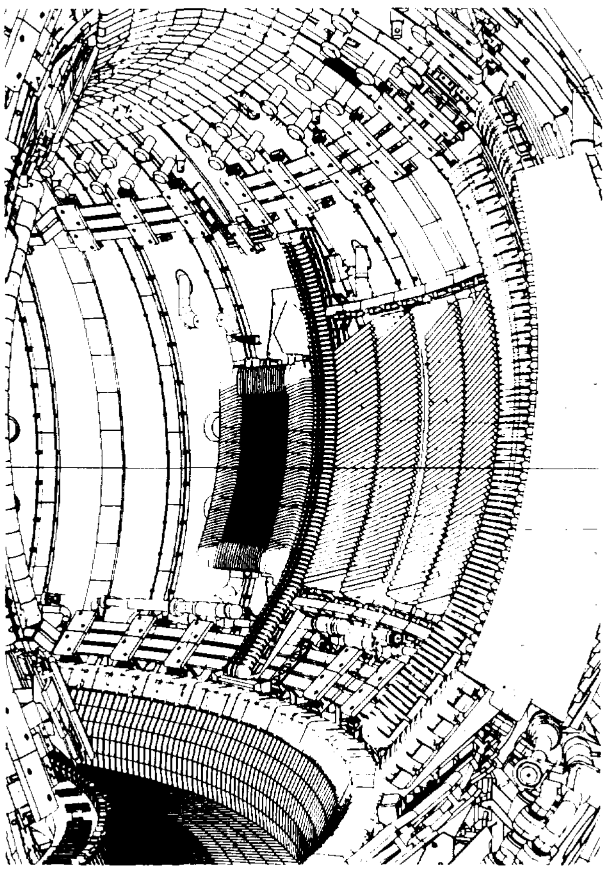

Fig 5: A CAD representation of the JET vacuum vessel after completion of the present shutdown. Divertor target plates are seen in the bottom, one of the four ICRH modules is on the right adjacent to the LHCD launcher (centre).

The two high power RF systems (ICRH and LHCD) have also been upgraded to match the antenna shape to the new plasma cross-section. The power launched in the form of travelling waves will be tripled in order to increase the capability for generating the plasma current and, more precisely, to control the profile of the plasma current density which is expected to influence the plasma confinement properties. A view of the new in-vessel equipment is shown in Fig 5. The experimental programme is planned to start in 1994.

Finally, I will end this brief introduction of the JET apparatus by showing (Fig 6) wave forms of the plasma current in JET's most interesting modes of operation. The longest plasma "flat top" reached one minute with reduced current and field (2 MA, 2.2 Tesla). The largest plasma current was 7 MA with limiter plasma. However, in this condition, the plasma purity was mediocre with adverse effects on the performance. The highest performance corresponding to an equivalent QDT ~ 1 was obtained in 3 MA single null discharge which gave both excellent confinement and low impurity content. Finally, 5 MA single null discharges gave an inferior performance as compared to the 3 MA case. Several aspects (eg sawtooth control and correct divertor operation at high current) will have to be developed in the new JET phase in order to release a reserve in performance expected from operation at higher current, larger heating powers and with control of the plasma current profiles.

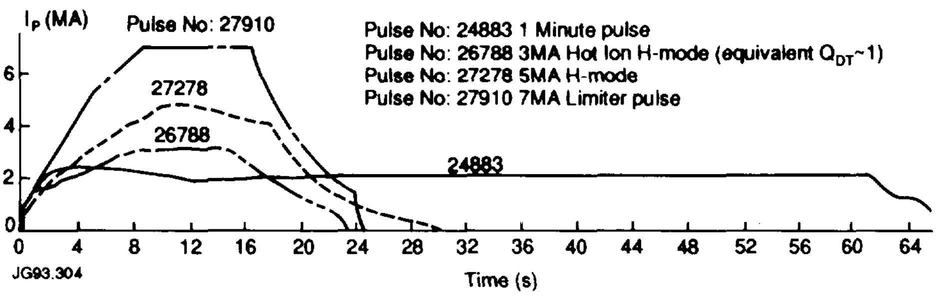

Fig 6: Plasma current waveforms achieved in several of the most interesting modes of operation of JET.

III. HIGHLIGHTS OF JET RESULTS

In 1992 alone JET prepared 436 publications; 118 of these were in the form of journal publications. It would be presumptuous to give here even a brief account of all the topics where significant progress was made. It is possible to find such a summary in the 1992 JET Progress Report (EUR-15081-EN-C, EUR-JET-PR-10, 1992) which also includes a useful list of references. The young readers (and not so young alike) should be reminded of the dreadful consequences for his scientific reputation of a superficial varnish of knowledge and is advised to read the original publications. In this ultra-brief review performed at a cosmic altitude, I will only pick up a few topics which have obvious impact on future work.

III.A. Plasma Confinement

Tokamaks offer a confinement quality which is only approached by Stellarators. Nevertheless a tokamak plasma is a turbulent fluid with a complex interplay of electric and magnetic activities fuelled by the free energy arising from numerous departures from thermodynamic equilibrium. Ordinary fluid turbulence which develops in the water flowing in a pipe is still far from being understood after many years of study. Therefore, it is not surprising that a unified physical interpretation of the local plasma transport properties is not available. Phenomenology (sometimes botanic) appears as a necessary disease in this field. Globally, the energy confinement time $\tau_E = W/(P - \partial W/\partial t)$ (W: stored energy; P: input power to the plasma) is observed to increase with plasma current and to degrade with P, a behaviour called L-mode. However, such a simple behaviour would be terribly boring, but to the experimentalist's delight, the plasma bifurcates, when certain conditions are met to higher confinement modes. The H-mode (ASDEX, 1982) is a sudden bifurcation which doubles τ_E due to a significant increase of the confinement near the edge. PEP modes (JET, 1987) and supershots also called hot ion modes (TFTR and JET, 1989) show increased central confinement. VH-mode, also called hot ion H-mode (DIII-D and JET, 1992) combine the enhancements in the edge and to that achieved in the centre. On the physics side, it emerges that the shear of the plasma flows creates the stabilising influence on edge turbulence in the H-mode and that a low (possibly inverted) magnetic shear could be responsible for the improvement of the central confinement. These ideas are far from being solidly proven but they serve to guide new research. In particular they emphasize the need to control the magnetic shear and I will come back to this aspect later.

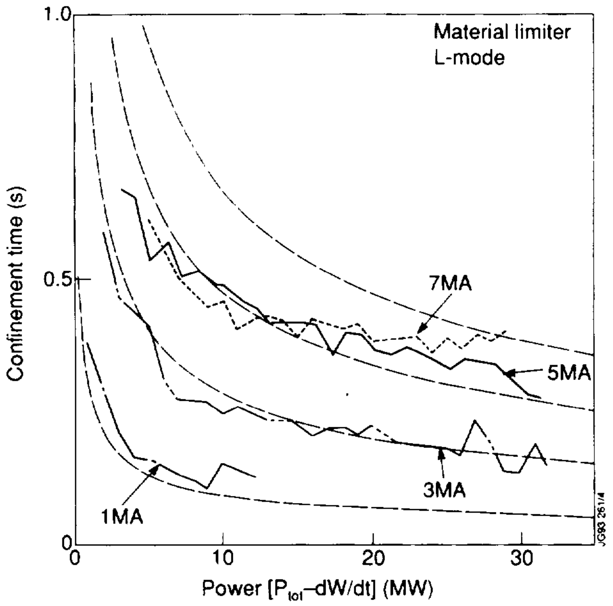

Fig 7a: Energy confinement time as a function of the loss power for several plasma currents in the L-mode material limiter operation.

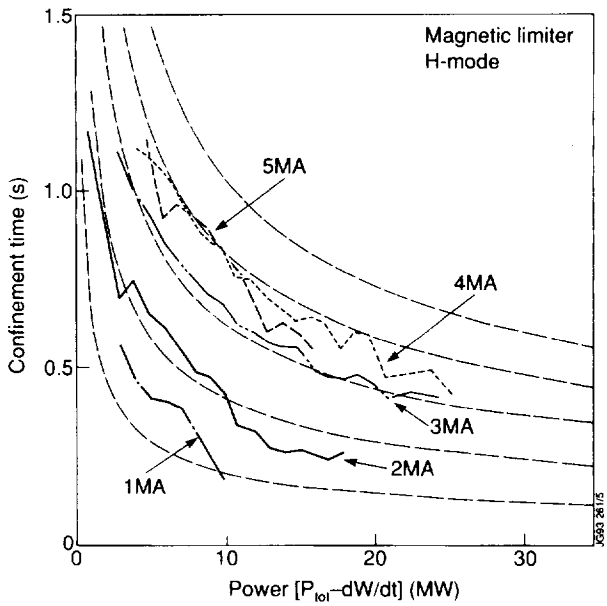

Fig 7b: Same as in (a) but for H-mode in magnetic limiter (X-point) operation.

Fig 7 summarizes the JET data base of τ_E versus P when material limiters are used (Fig 7a) or with a magnetic limiter or divertor (Fig 7b). The degradation of the confinement with power is obvious in both cases. In the divertor case, the H-modes are triggered at around 5 MW and the confinement improves at this threshold; at higher powers the trend of the degradation is similar to the L-mode case albeit with twice the global confinement. The smooth curves on diagrammes represent "scaling laws" obtained by statistical regression fitted to a simple power law on a world-wide data base (including of course JET). The scaling laws in the L and H modes have been found to be:

For the ITER 89 Power law in L-mode (ITER 89-P, L-mode):

$$\tau_E = 0.048\ A^{0.5}\ I_p^{0.85}\ R^{1.2}\ a^{0.3}\ \kappa^{0.5}\ (n/10)^{0.1}\ B^{0.2}\ P^{-0.5}$$

For the ITER 92 Power law in H-mode (ITER 92 H-P, H-mode):

$$\tau_E = 0.032\ A^{0.45}\ I^{0.95}\ R^{1.9}\ a^{0.05}\ \kappa^{0.65}\ n^{0.3}\ B^{0.2}\ P^{-0.65}$$

where A is the atomic number, I_P (MA) is the plasma current, R(m) is the major radius, a(m) is the minor radius, κ is the elongation, B(T) is the magnetic field, n(10^{19} m^{-3}) is the density and P(MW) is the total input power. Extrapolation from the scaling laws form the basis for the parameters of the ITER machine which, with its present dimensions could tolerate a slightly degraded H-mode to achieve ignition.

The work on scaling laws has recently evolved using the approach of dimensional similarity (also called the wind tunnel approach) with the aim of deriving the scalings based on the underlying physics equations. The results[4] presented by both TFTR and JET at the last IAEA conference in Würzburg (1992) suggest that the fundamental processes may be associated with coherent structures (long wavelength turbulence) rather than microscopic processes with a turbulence wavelength comparable to the ion Larmor radius.

III.B. High Fusion Performance in Deuterium and Deuterium-Tritium Mixture

The confinement of the H-mode has recently been surpassed by about another factor of 2 by DIII-D, JET and JT60-U. The improved confinement (VH-mode - also called hot ion mode in some cases) follows a second, more gradual transition during the H-mode. The improvement from L-mode to H-mode coincides with the development of steep gradients in the temperature and density profiles near the edge. The further improvement from H-mode to VH-mode coincides with a broadening of the region where the steep gradients develop. The large pressure gradients in turn generate a plasma current (bootstrap current) as predicted by the classical description of the parallel plasma flows. The new equilibrium enters part of the plasma into a second domain of stability against ballooning modes predicted by theory. At this early stage of the research it is not possible to distinguish between causes and effects in the sequence of events. Many other aspects will need to be investigated before proposing to operate a fusion reactor in this mode; most notably one will need to demonstrate steady state operation and also that the ratio of impurity to particle confinement remains acceptable ($\tau_P < 8\ \tau_E$) to prevent poisoning by helium ash accumulation.

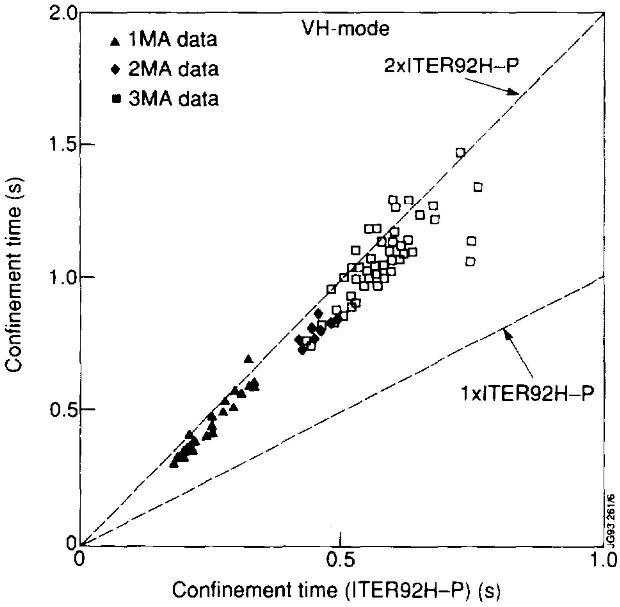

Fig 8: Energy confinement time for the VH-mode plotted versus the ITER(92 H-P) scaling for plasma current values of 1, 2 and 3 MA.

Fig 8 shows the confinement time for the VH-mode obtained in JET with NBI, ICRH and combined heating for the VH-mode versus the ITER (92H-P) scaling for plasma currents of 1, 2 and 3 MA. The best performances of JET have been obtained in this mode and the PEP mode (not discussed here) at 3 MA. The maximum fusion triple product reached $9\ 10^{20}\ m^{-3}\, s\, keV$. Using this type of discharge a tritium injection experiment was performed for the first time by injecting with Neutral Beam a mixture of deuterium and tritium $n_T/(n_D + n_T) \simeq 0.11$. The total fusion power released was 1.7 MW at peak power and 2 MJ of energy.

The consistency of the experimental data was established with simulations using the TRANSP code. The good combination between measurement and simulation gives confidence in the accuracy of extrapolation from existing discharges. Assuming an optimum tritium concentration ≈ 0.6 on the same

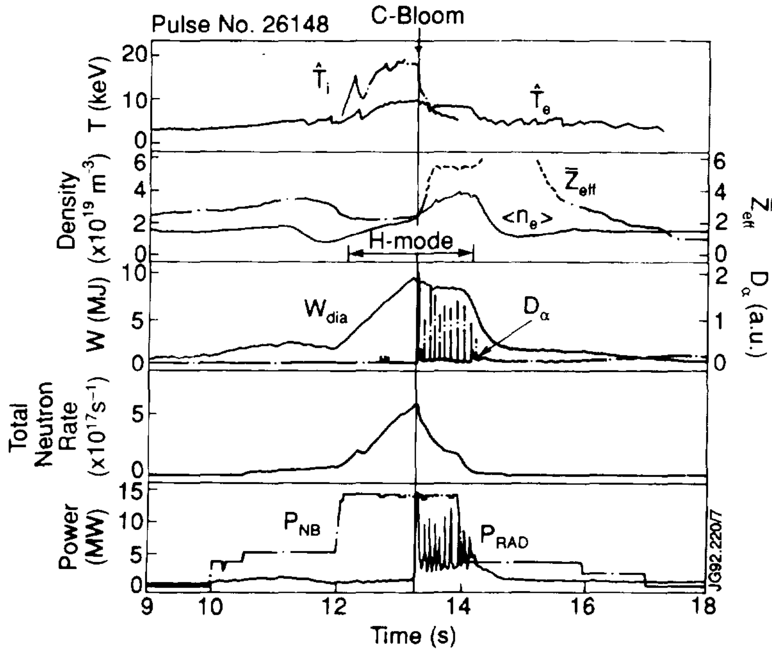

Fig 9: The time development of the central electron and ion temperatures, the volume-averaged electron density, the line-averaged density Z_{eff}, the plasma diamagnetic energy, the D_α emission, the total neutron rate, and the NB and radiated power for Pulse No 26148. After the "carbon bloom", the Z_{eff} measurement is affected by black body radiation emanating from the targets.

discharge (# 26148) would have produced a fusion power of 5 MW ($Q_{DT} \approx 0.46$). The same extrapolation for one of the best JET shots (# 26087) gives 11 MW and $Q_{DT} \approx 1.14$.

Fig 9 shows the time development of the plasma parameters during the tritium experiment. The sharp reduction of fusion yield at 13.2 s is associated with excessive thermal loading of the divertor tiles and MHD instabilities. It emphasises the need to control both the plasma exhaust with improved divertor concepts and the plasma current profile for improved stability margin.

III.C. Divertor Aspects

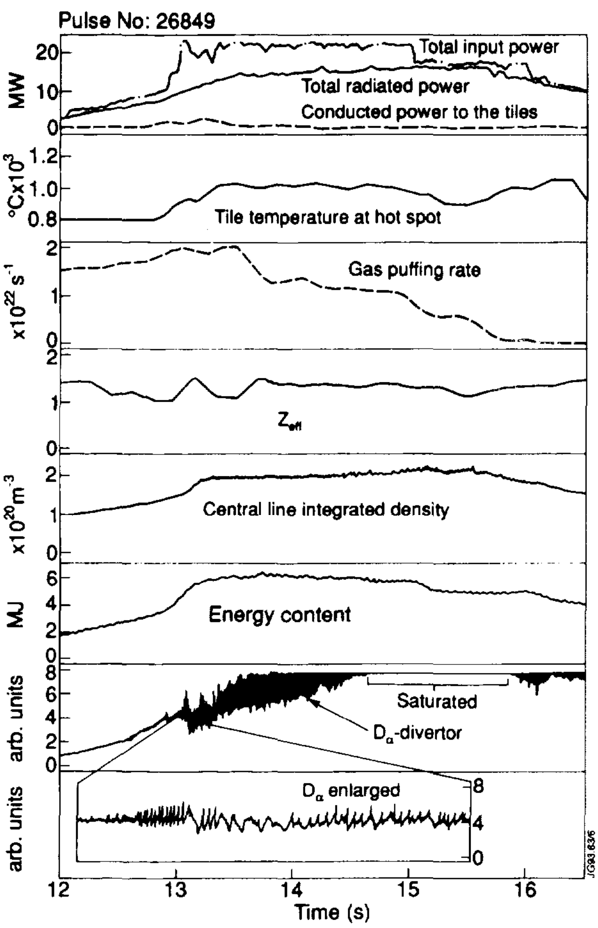

Fig.10: A radiative divertor discharge with 22MW of input power (ICRH and NBI) and only 1MW of power conducted to the target tiles. Quasi steady-state operation is demonstrated.

Power and particle exhaust appears as the most challenging problem for the magnetic fusion reactor. The problem arises from the fact that the power scrape-off layer does not increase appreciably with the tokamak dimensions; consequently the power per flux tube area diverted away along the magnetic field line reaches, for reactor sizes, a value that exceeds acceptable ratings of the divertor material. It is therefore mandatory that most of the exhaust power should be radiated across the field before reaching the divertor plates. Code simulations predict that the divertor plasma goes into a high recycling regime when the plasma density in the divertor exceeds a threshold value. It is not clear at this stage that the enhanced radiating properties of the recycling regime are sufficient to meet the ITER requirement. The divertor geometrical shape plays a key role in determining the properties of the recycling regimes. Progress will require a systematic investigation of divertor physics with various geometries on several tokamaks.

The most successful attempts to establish a radiative divertor were performed in single null X-point configuration on beryllium targets. Fig 10 shows that a stable high power discharge can be maintained in quasi steady state with less than 5 % of the power conducted to the divertor tiles. A key of the success was a feedback control loop regulating the gas-puff rate from measurements of the thermal radiation emitted by the divertor target: more gas was injected if the tile temperature increased thus avoiding the instability of the exhaust power burning through to the target plates. The demonstration of this regime is of major significance for the development of reactor-relevant scenarios. However, it has not yet been possible to combine the radiative divertor with high confinement H-modes, since the plasmas either remained in the L-mode or exhibited ELMy H-mode behaviour with modest confinement improvement. The new divertor structure which is being installed will allow a better separation of the divertor region from the bulk plasma and it will be a crucial test to try to achieve good confinement with radiative divertor operation.

III.D. Profile Control and Progress Towards Steady State

The plasma current can be generated by three sources: by induction from the flux swing of the primary ohmic current, by bootstrap resulting from classical diffusion of the pressure gradients and by external current drive systems. Only the last two sources can be used in steady state tokamaks. The bootstrap is very attractive as a free source of current and most steady state reactor scenarios are based on producing more than 70 % of the current by this way. Since the bootstrap current fraction can be written $I_{bs}/I_p \sim \sqrt{a / R}\, \beta_p$ where β_p is the plasma pressure normalised to the poloidal field pressure, it is necessary to operate with $\beta_p \gtrsim 1$. Both JET and JT60-U have recently demonstrated high confinement operation with more than 70 % of bootstrap current. The result is highly encouraging; however, in both JET and JT60-U the current profile evolves towards a broader profile which becomes MHD unstable. An external source of current is therefore required not only to drive part of the total current but also to maintain a stable profile.

Lower Hybrid Current Drive (LHCD) has been so far the most successful external source of current drive. JT60-U has recently reported considerable success in driving large currents of up to 3.5 MA and on the benefits of profile control on high performance discharges with large bootstrap current fraction. JET started experiments with its prototype system delivering up to 2.7 MW. Full current drive has been achieved at 2 MA. Combined operation with the ICRH system led to the discovery of synergistic acceleration of the fast electrons beyond the range accessible to LHCD only. The current drive efficiency defined as $\gamma = \langle n \rangle R$ I_{CD}/P_{CD} reached a record value of 0.410^{20} (m^{-2} A W^{-1}). This result which is important for reactor extrapolation is illustrated in Fig 11 and compared to demonstrated and anticipated efficiency of several other methods. Assuming that no more than 20 % of the fusion power output is used by the current drive system, the required current drive efficiency would have to exceed 0.510^{20} m^{-2} A/W for high bootstrap reactors or 2 10^{20} m^{-2} A/W for the standard operating mode.

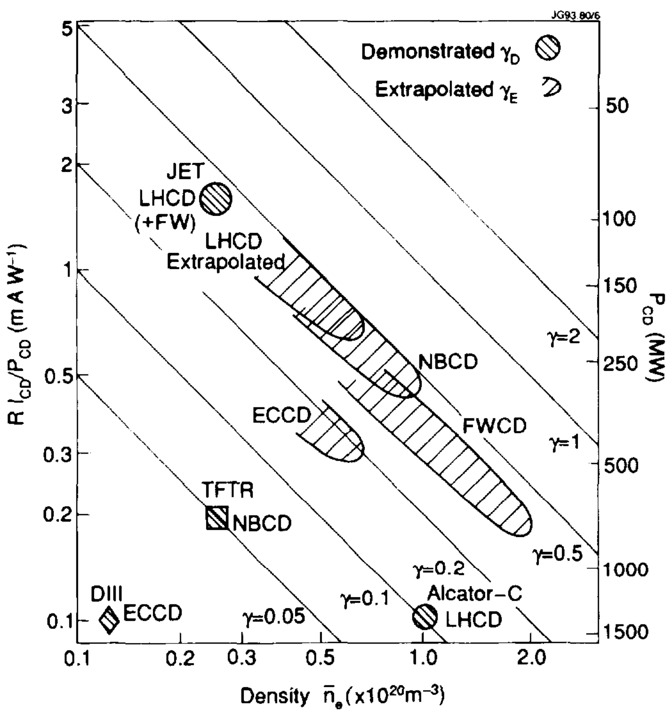

Fig 11: A comparison between demonstrated and extrapolated current drive efficiency for several current drive schemes. P_{CD} refers to the required current drive power for a conventional reactor.

Fig 11 also illustrates the need to extend the operating domain towards the high density required by reactors (1 to 2 10^{20} m^{-3}). The penetration of lower hybrid waves is limited at around 7 10^{19} m^{-3} and current drive in the centre needs to be provided by other means. This could be achieved with Neutral Beam Current Drive (NBCD) requiring expensive development of negative ion beams at 2 MeV in order to reach the plasma centre with the required efficiency. Fast Wave Current Drive is also an attractive candidate which requires less technical development but the experimental observation of the current drive efficiency is in its infancy. The fast waves have no density cut-off and this property has been demonstrated routinely using fast waves in the ICRH domain. Current drive in the minority ion heating regime has also been demonstrated by JET. The missing link is the generation of a well defined travelling wave with strong damping on the plasma electrons (and not on the ions) in order to avoid the loss of wave directivity due to multiple reflections in the Tokamak. DIII-D has obtained the first encouraging results in this mode. JET is presently completing the construction of its 4-strap modules in order to launch the well defined travelling waves. The system which replaces a 2-strap system should enhance the current drive potential by a factor of 3 allowing new experiments at increased power in synergy with LHCD, Fast Wave Current Drive in the electron damping regime as well as in the ion damping mode.

IV. CONCLUSION

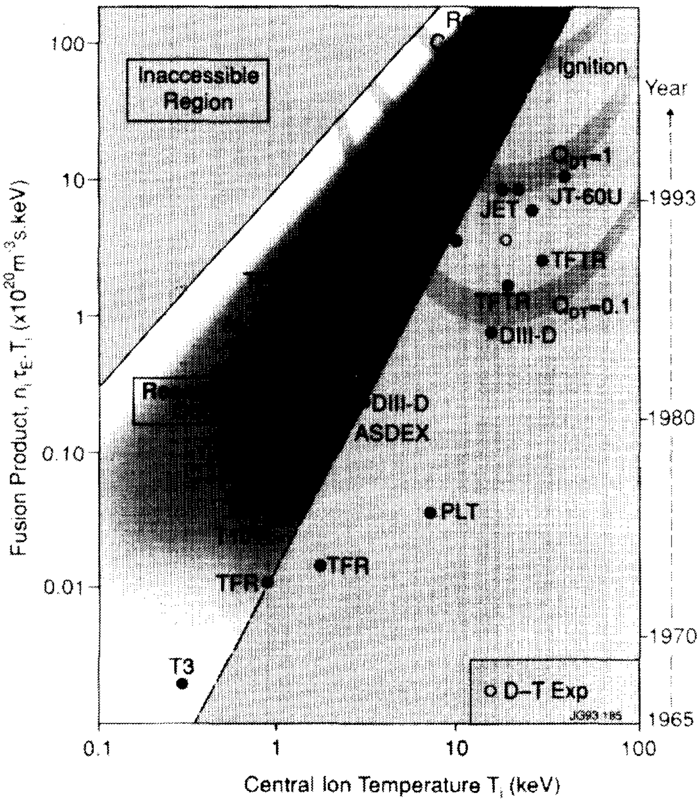

Fig 12: Triple fusion product ($n_D\tau_ET_i$) versus central ion temperature for a number of machines world-wide in the period 1965 - 1993.

The human and capital resources assembled in JET by Euratom and the Associated laboratories have placed Europe on the front line of fusion research. JET has obtained far-reaching results in all domains of tokamak physics.

On confinement, JET has provided the "top right corner" of the confinement data base which plays a key role in the extrapolation to a next step device capable of ignition. This research needs to be continued most notably to investigate the scaling of the H-mode threshold and to create a data base for the newly discovered VH-mode.

On High Fusion performance, JET still holds the highest value of the equivalent fusion amplification factor $Q_{DT} \sim 1.14$. JT60-U has now achieved the largest triple product $n_D\ \tau_E\ T_i$ (Fig 12). However this does not translate into the highest Q_{DT} because the ion temperature exceeds the optimum value. JET and JT60-U have considerable potential to extend the present performance by developing with profile control a VH-mode discharge at higher current and heating power. Such performance developments will have a considerable impact on the Deuterium-Tritium phase which is planned at the end of the JET plasma operations.

The Divertor aspects now receive first priority in the JET programme not only because they impact on the JET performance but also because they address the most difficult challenge on the way forward toward a fusion reactor. JET has operated at high power with a radiating divertor which now needs to be made compatible with high performance operations. The optimisation of the divertor geometry will require testing of 2 or perhaps 3 divertor generations. In particular, it is apparent that ITER will require the demonstration of an ITER-like divertor on large devices such as JET before entering a construction phase.

Profile control with external current drive is also receiving increased attention in the JET programme. JET has developed current drive systems using both high power slow waves (LHCD) or fast waves (FWCD) and achieved the highest current drive efficiency. Steady state tokamak reactors will not only require external current drive but also the appropriate localisation of this current to prevent the occurrence of MHD instability. LHCD and FWCD are prime candidates for this task.

JET is presently completing a major enhancement of its internal machine structure to install 4 new divertor coils, a cryogenic pump, new ICRH and LHCD antennae and saddle coils for disruption control. JET will restart operation in 1994. Its new equipment has the potential to maintain the European leadership in fusion for many years to come.

ACKNOWLEDGEMENT

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