JOINT EUROPEAN TORUS

JET–P(93)02

E. Bertolini and the JET Team

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JET-Joint Undertaking, Culham Science Centre, OX14 3DB, Abingdon, UK

Preprint of a Paper to be submitted for publication in
Fusion Engineering and Design March 1999

ABSTRACT

The first ever production of controlled thermonuclear energy, using a mixture of deuterium and tritium plasma, was achieved with the JET Tokamak experiment on the 9th of November 1991. A fusion power of 1.7MW was obtained in a pulse 2s long. The construction of JET, the flagship of the Fusion Research Programme of the European Community, took place between 1978 and 1983. Then the experimental programme started. JET plasmas have already approached equivalent conditions to those needed in a thermonuclear reactor, although only transiently, due to the high influx of impurities from the vessel walls. This problem, the main threat to the development of a fusion reactor, will be addressed in the next phase of JET development (1992-96) using a new pumped divertor magnetic configuration. The expected experimental results will provide a further major contribution to the engineering design of the 'next step' device, ITER, a cooperative effort, presently under way, between Europe, United States, Russia and Japan. ITER will be a prototype thermonuclear reactor aimed at producing fusion power in excess of 1,000MW.

1. THERMONUCLEAR FUSION

The production of nuclear energy by *fission* is well known. This is the process of splitting heavy atomic nuclei, such as uranium, into lighter elements and is the process used in today's nuclear reactors. A further nuclear process is by *fusion* of nuclei of light atoms, such as hydrogen and its isotopes, deuterium and tritium. This is the process occurring in the sun and in the other stars. It is also the process used for the hydrogen bomb. The ultimate objective of fusion research is to release such energy in a controlled way so that it can be conveniently used to produce electricity in a power station.

1.1. Fusion reactions

There are several fusion reactions that can be considered (Fig. 1). The interacting nuclei must carry sufficient energy to overcome the repulsion forces between equally charged particles: therefore, they must be heated to a very high temperature T, of about of 10 keV (approximately 100 million ^{o}C) or more. The reaction with the highest cross-section at such temperatures is D (deuterium)+T (tritium) and is the one now considered most practicable for a fusion reactor. Moreover, the fuel mixture should have a suitable density n (above 10^{20} m^{-3}) to achieve the required reaction rate and should be maintained in these conditions for a time, τ_E (energy confinement time). This time must be sufficient for self-heating to occur, by the release to the fuel of the energy associated with the charged particles (helium, He), resulting from the fusion reactions. Due to the different reaction cross-sections, the *fusion parameter* $n\tau_E T$ is the lowest for the reaction D+T. A D+T fusion reactor requires a fusion parameter in excess of 5×10^{21} [m^{-3}skeV].

There are two main approaches to reach thermonuclear fusion conditions:
a) *Inertial confinement* is based on the same process of the hydrogen bomb. The source of energy for plasma heating and confinement is provided by powerful laser beams or by beams of heavy ions accelerated at high energy [1, 2];
b) *Magnetic confinement*, specifically developed for controlled thermonuclear fusion, is the one on which JET, most of the fusion experiments around the World and the concept of ITER are based.

1.2. Magnetic confinement

At the temperatures required for fusion, the fuel is in the *plasma* state, i.e. a mixture of positively charged nuclei and negatively charged electrons. Charged particles can be trapped by magnetic fields and free electrons can carry electric current. Magnetic confinement is based on the exploitation of these plasma properties. The most effective magnetic confinement device or *magnetic bottle* has proven to be the *Tokamak* (Fig. 2), (which stands for *toroidalnaya kamera magnitnaya*.). It was developed by Russian scientists and was made known during the 1960's. Since that time most of the magnetic fusion experiments throughout the World are tokamaks.

In a tokamak, the magnetic field for confinement and for keeping the plasma away from the vacuum vessel walls (a high vacuum is needed, for plasma purity) is provided by the combination of the high field produced by the toroidal coils, supplied by a DC current, with the more modest field associated with the DC current flowing in the plasma ring. This plasma current is induced by a modulated DC current flowing in the central solenoid, like in an ordinary transformer, where the solenoid plays the role of the primary winding and the plasma ring is the secondary winding. To produce and sustain the plasma current, a magnetic flux variation is needed to induce an electric field, which produces the gas breakdown, starts, increases and maintains the plasma current.

This makes the tokamak an inherently non-continuous machine. However, due to the high plasma conductivity (much larger than that one of copper), it is possible to sustain the plasma current in a reactor for half an hour or more. The plasma ring is maintained stable by position control magnetic fields produced by poloidal coils surrounding the machine.

Moreover, this current heats the plasma by joule dissipation (ohmic heating) up to temperatures of few keV. Additional external means are needed to further enhance the temperature, such as radiofrequency heating and beams of neutral particles (deuterium and/or tritium atoms) at high energy; neutral particles (atoms) are needed to penetrate the surrounding magnetic fields and reach the plasma centre. In addition neutral beam injection and radiofrequency methods may be used to accelerate the electrons and contribute to driving the plasma current, possibly indefinitely (current drive techniques).

1.3. Key tokamak and fusion parameters

The main parameters of a tokamak are the *major radius* of the plasma ring (R), the *minor radius* (a), i.e. the radius of the cross-section of the plasma ring, the *toroidal magnetic field* on the plasma axis (B_{T0}), and the *plasma current* (I_p). To avoid major Magneto-Hydro-Dynamic (MHD) instabilities, these parameters must satisfy a particular relationship, $B_{T0}/[(R/a)B_p] = q > 1$, where B_p is the poloidal magnetic field associated with the plasma current I_p, ($B_p = I_p/2\pi a$); q is called MHD safety margin. Therefore, in a tokamak, $B_p \ll B_{T0}$, where B_{T0} has a value of several Teslas and B_p is, usually, less than one Tesla. The plasma cross-section can be elongated (as in JET) and the *elongation* parameter is defined as b/a, where b is the halfheight of the plasma. In this case, the equivalent plasma radius is defined as $(ab)^{1/2}$.

Another key factor is the *fusion parameter* $n\tau_E T$, already mentioned. It is worthwhile recalling the definition of the energy confinement time, as the ratio between the energy stored in the plasma W_p and the total heating power P_{in}. In present experiments, plasma conditions are not steady state, hence $\tau_E = W_p/(P_{in} - dW_p/dt)$, where $P_{in} = P_\Omega$ (ohmic) + P_{NB} (neutral beams) + P_{RF} (radiofrequency).

The *energy gain* $Q = P_F/P_{in}$ is the ratio between the fusion power produced and the input power, where $P_F = P_\alpha + P_N$ is the sum of the powers associated with the alpha-particles (helium ions), (20%), and the neutrons, (80%). Q = 1 is called *breakeven* , and is the objective of JET to achieve this condition. For a reactor, Q must be infinite and this condition is called *ignition*. The plasma, once taken to ignition condition by the additional heating systems, is kept burning only by the power associated with the alpha-particles, produced by the fusion reactions.

The level of *plasma purity* is defined with an equivalent atomic number for the whole plasma mixture, $Z_{eff} = \Sigma_j Z_j / n_i$, where n_i is the total density of the ions and n_j and Z_j refer to a specific ion species. $Z_{eff} < 2$ is a practical requirement for a reactor, since impurities have two main adverse effects. They increase the plasma energy losses by radiation and dilute the plasma, $n_{DT}/n_e < 1$, thus reducing the density of ions that can fuse. Therefore, low Z materials should be used, as far as possible, as plasma facing components.

2. THE JET PROJECT

The ultimate goal of the *European Fusion Programme*, is the industrial production and commercialisation of fusion energy for the generation of electricity (1971, Council of Ministers of the Community). The programme is based on a sequence of three devices: JET; ITER; and DEMO. JET (Joint European Torus) should prove the scientific feasibility of fusion; ITER (International Tokamak Experimental Reactor), now under detail design on a world-wide cooperation, should produce large amounts of fusion power in plasma ignition conditions; and DEMO (Demonstration Reactor), which would be part of the first thermonuclear power station, should produce electricity for the European Grid.

The hard core of the programme is supported by essential activities performed by the National Laboratories (*Associations*) directed to the study of specific physics issues making use of smaller machines and to the development of key technologies required, in particular, by ITER.

JET was designed between the end of 1973 and 1976. However, the start of construction was delayed due to political difficulties in reaching an agreement on the machine siting. Eventually, the British site was selected and the JET Joint Undertaking was established. Construction started in June 1978, and was completed, on time and within the allocated budget by June 1983. The experimental and development programme then started.

2.1. JET Objectives

The Resolution of the Council of Ministers of the European Communities, on the 30th of May 1978, establishing the JET Joint Undertaking, stated that JET was:...*to construct, operate and exploit, as part of the European Fusion Programme and for the benefits of its participants a large torus facility of tokamak type and its auxiliary facilities in order to extend the parameter range applicable to controlled thermonuclear fusion experiments up to conditions close to those needed in a thermonuclear reactor.*

Therefore the mission of JET is to investigate four main areas of research, the *JET Objectives*:

(a) plasma processes and scaling laws in operating regimes of reactor relevance;
(b) methods of plasma heating up to temperatures required in a reactor;
(c) control of plasma-wall interaction and influx of impurities in reactor relevant conditions;
(d) production and confinement of alpha-particles, resulting from D-T fusion reactions

The first three areas can be investigated by using pure deuterium to reduce unnecessary neutron induced activation of the vessel walls at plasma parameters of reactor relevance for a deuterium-tritium mixture.

2.2. The JET machine

A schematic view of JET [3,4] is shown in Fig. 3. The *vacuum vessel* and the *toroidal coils* have an elongated cross-section ('D-shape'), which allows production of plasmas of similar cross-section, with an appropriate energisation of the *poloidal coils*. This choice, aimed originally at reducing electromechanical stresses on the toroidal coils, has proven to be a great asset for JET. It has allowed use to be made of the magnet volume with large plasma currents; to operate with a special magnetic configuration called 'X-point', leading to plasma enhanced confinement properties; and, to install a pumped divertor at the bottom of the vessel. Both toroidal and poloidal coils are copper wound, and they operate in a quasi-adiabatic mode, since they are actively cooled after each pulse. The vacuum vessel has an inconel double-wall with the capability of hot gas flowing to control the temperature of the inner wall up to 350 ^{o}C. The *tokamak transformer* has an iron core. While making the vertical position control of the plasma more

difficult, it allows savings in Ampere-turns and a reduction of stray field around the machine, beneficial to diagnostics and neutral beam heating systems. The pulse time of over 20s flat top at currents of 3MA, allows conditions close to steady state to be explored.
The main parameters of JET are given in table 1.

The *additional heating systems* include 8 ICRF units (Ion Cyclotron Radio Frequency), operating at 25-55MHz, and 16 NBI (Neutral Beam Injectors) operating at 80 and/or 140-160kV. Each one of the two systems can deliver 20MW to the plasma [5, 6]. A 10MW LHCD (Lower Hybrid Current Drive) system, operating at 3.7GHz, allows control of the plasma current density profiles and to supplement the capability of the tokamak transformer for extended pulse length [7].

The *plasma fuelling*, i.e. the introduction of hydrogen isotopes into the vacuum vessel, is achieved in three ways: by injecting gas before and during the pulse; by neutral beam injectors; and by solid pellet injection at speeds up to 4km/s [8], during the plasma pulse.

About *50 diagnostics systems* allow measurements of plasma parameters with space and time resolution [9].

A large amount of *DC pulsed power* is required to operate JET, with a peak above 1000MW and an energy content per pulse that can reach 10,000MJ. More than 50% of this power is taken directly from the British Grid and the rest from two local flywheel generators. The main loads are the toroidal coils, the primary winding of the transformer, the outer poloidal coils, the additional heating systems and, in the near future, the divertor coils [10].

The operation of JET is managed by CODAS (*Control and Data Acquisition System*), a system of on-line computers to control the machine and to collect, organise and store machine and plasma diagnostics data [11].

JET has been equipped with the *nuclear systems*, required for operation in deuterium and tritium. This includes a set of remote handling tools and procedures [12] and the AGHS (Active Gas Handling System), a deuterium-tritium fuel processing plant [13].

An overall view of the JET tokamak is shown in Fig. 4.

2.3. JET Operation and main scientific and technical achievements

The JET experimental results, concerning a great variety of plasma physics issues, have been reported in a large number of papers presented at major international conferences and published in scientific journals. In the following, the progress toward plasma reactor relevant conditions will be briefly reviewed. This progress is marked by the increase of the key plasma parameters, namely, temperature, density, energy confinement time, level of impurities, fusion product and energy gain (table 2). These results have been achieved by an integrated physics and engineering approach, where physics results would suggest engineering development and its implementation would allow upgrading of plasma performance. It is clear that the first three JET objectives have to be tackled together since efficient plasma heating and control of plasma purity are essential for upgrading global plasma performance into the reactor regime.

Initially *ohmic heating* only was used, fuel was injected by gas puffing and only four small graphite limiters (the first points of contact with the plasma), where installed on the $200m^{-2}$ inconel vessel wall facing the plasma ('first wall'). Although the fusion parameter achieved was already well above those previously achieved with other tokamaks, only the value of the energy confinement time was of reactor relevance. The plasma temperature, a respectable value of 3keV, indicated that additional heating was required. Finally, the high level of impurities strongly suggested the move towards low-Z materials for the first wall [14].

During the following experimental campaigns both RF and NB *additional heating* systems came progressively into operation. In addition, more than 20% of the inconel vessel wall was covered with graphite tiles (Z = 6, much smaller than that of iron and nickel in the inconel alloy) and the full surface was covered, prior to a series of experiments, with a thin layer of carbon ('carbonisation' by means of special radiofrequency discharges in a CD_4 gas).
As a consequence, ion temperatures entered the reactor regime (above 10keV) and the level of impurities was significantly reduced. However, as shown in Fig. 5, the energy confinement time sharply degraded with increasing additional heating (i.e. the plasma temperature), preventing any significant increase of the

fusion product with plasma heating. However, at a given level of plasma heating, larger plasma currents showed an increase in energy confinement time.

By making use of the inherent flexibility of the JET design, a new magnetic configuration was implemented, following the interesting results already obtained with a medium size tokamak, ASDEX, Germany. This configuration is called 'X-point' or magnetic limiter, because the key feature is a *null* of the poloidal magnetic fielf at the bottom and/or at the top of the plasma cross-section and because the plasma does not touch the first wall limiters. In this condition ('High confinement or H-mode'), as shown in Fig. 5, at a given plasma current and heating power level, the energy confinement time is about twice the one obtained in the material limiter configuration ('Low-confinement or L-mode') [15, 16].

These results had a remarkable impact on the JET programme to follow and on tokamak research around the World. Tokamaks have been modified accordingly (D-III, USA and JT-60, Japan), and ITER is conceived to operate in the magnetic limiter configuration.

A *major upgrading* of the JET machine and its auxiliary facilities was then undertaken, extending the parameter range to higher plasma currents in both material (up to 7MA) and magnetic limiter (up to 5MA) configurations, well above the design values. The lower current achievable in the X-point configuration is due to the need to use some of the vessel volume to establish such a configuration. At this time, the graphite first-wall tiling was extended to 50% of the surface and the pellet injector, for deep fuelling into the plasma was made operational. In addition, JET could now be operated in the three configurations shown in Fig. 6. A significant improvement in plasma parameters became possible. In separate experiments, but for the first time with the same machine, all parameters required for a reactor were individually achieved: ion density above $10^{20}m^{-3}$, ion temperatures of 20keV and energy confinement time exceeding 1s [17,18,19,20]. However, the level of impurities still exceeded the values acceptable for a reactor, leading to large radiation losses and to a dilution value below 0.5. It was evident that the main obstacle for further progress was poor plasma purity.

In the following experimental campaign, *passive control of impurities* was successfully attempted using beryllium as a first-wall material. Beryllium has two advantages over carbon; lower Z (Z=4 instead of 6) and an excellent capability to reduce oxygen. The carbon tiles of the main limiters and those facing the X-point area (where most of the plasma energy would be released in L-mode and H-mode operation, respectively), were replaced by beryllium tiles. Moreover, frequent coating of the remaining inconel and carbon first wall material was achieved by means of four beryllium evaporators. The present configuration of the vacuum vessel is shown in Fig. 7 [21]. The main scientific achievement [22,23,] was the success in keeping impurities to a level where $Z_{eff} <2$ in many discharges, and in increasing the dilution factor n_D/n_e above 0.8. This allowed a further increase in plasma parameters, achieved individually, to values well within the reactor regime. The fusion product increased by a factor of about four to a global plasma performance in deuterium equivalent to D-T 'breakeven' (Q ~1). This is shown in Fig. 8, where the progress in JET performance is compared with the other worldwide tokamaks, particularly the largest American (TFTR and D III-D) and Japanese (JT-60) tokamaks. It is worth mentioning that close to breakeven conditions have been obtained in JET both in the so called 'hot ion' regime (low plasma density and ion temperature T_i much larger than the electron temperature T_e), produced by intense NB heating and in the 'thermal' regime ($T_i \sim T_e$), with predominant RF heating and higher densities: the latter is the reactor regime, since alpha-particles (the sole source of heating in ignition conditions), tend to heat the electrons preferentially.

3. THE FIRST JET THERMONUCLEAR EXPERIMENT

The success in bringing plasma performance well into the reactor regime, leading to high neutron emission from D-D reactions ($4\times10^{16}s^{-1}$), made it possible to conceive an early D-T experiment , the Preliminary Tritium Experiment (PTE), not planned originally. This was aimed at providing a firm basis for the D-T phase of JET operation; gaining experience in handling tritium and in tritium recovery from in-vessel components and NB systems; and at the production a fusion power exceeding 1MW for one second. This limitation in power was due for a number of reasons. The total neutron production had to be $\leq 1.5\times10^{18}$, to limit the vessel wall activation to $\leq 100\mu Sv/hour$ at the start of the long shutdown in February 1992. In addition, the amount of tritium on site was limited to 0.2g, because the AGHS was not completed. Finally JET was operated at about half of its best performance. As a consequence a D-T mixture with 10% of T only was used, allowing two pulses to be performed, preceded by few pulses with ≤1% of T for calibration purposes. It was efficient for this experiment to inject tritium into a deuterium plasma by means of two NB injectors converted from deuterium to tritium operation and to use neutral injection only for additional heating. JET was operated in magnetic limiter configuration.

The two pulses were performed successfully, on November 9th 1991, showing almost identical results. The total neutron rate during one of the pulses is shown in Fig. 9. When the energy associated with the alpha-particles is added, this correspond to a peak power of 1.7MW and at an average power of 1MW for a period of over one second [24].

After the PTE, JET resumed operation in pure deuterium, while the tritium recovery process was in progress. By the start of the main shutdown in February 1992, all tritium was recovered and the activation of the vessel wall was within the predetermined limits. In conclusion, all the goals set for the experiment were successfully accomplished [25].

4. FUTURE DEVELOPMENT OF JET

Although the JET results are outstanding, they suffer one important limitation. High performance can only be achieved transiently for one or two seconds, due to the production and accumulation of impurities in the X-point region. The impurities are generated at the vessel wall by localised deposition of power and particles from the plasma.

For further progress in JET performance, for the design of ITER and of a future fusion reactor, *active control of impurities* is needed. To tackle this problem, - perhaps the last hurdle in the path leading to a fusion power reactor - the JET Project has been further extended from 1992 to 1996.

The aim of this new phase is to demonstrate a concept of impurity control, by means of an *axisymmetric pumped divertor* configuration.
The divertor configuration allows the channelling of power and particle exhaust to appropriate target plates and to the pumping region (Fig. 10). This configuration is created by four copper coils, now under construction at the bottom of the vacuum vessel. Other key components of the divertor are the target plate, installed at the top of the coils at a suitable distance from the X-point, and the cryo-pannels to pump away the gas.

It is the aim of the divertor programme to extend the duration of the H-mode regime, allowing plasma parameters to approach steady-state, in breakeven conditions. The success of this programme would represent an additional major contribution to the final design of ITER [26, 27, 28].

5. CONCLUSIONS

The operation of JET has brought tokamak plasmas well within the fusion reactor regime. This has been made possible by a constructive interaction between a well focused physics programme supported by engineering development, which has allowed the operating range to be extended well beyond the original design parameters.

The first two objectives of JET (scaling laws and plasma heating) have been fulfilled. This provides a sound basis for the choice of the key parameters of ITER, a cooperative design activity between Europe, United States, Japan and Russia, aimed at the construction of a prototype fusion reactor, with plasma operating in the ignited regime, capable of delivering a fusion power exceeding 1000MW.

Concerning the third objective (control of impurities), significant progress has been made on passive control, using carbon and beryllium as first-wall materials. However, the real solution of this problem, the only remaining threat to the successful development of a tokamak fusion reactor, will be the purpose of the JET pumped divertor programme.

Work on the last objective (alpha-particle production, confinement and subsequent plasma heating) has been initiated with the preliminary tritium experiment. A full scale deuterium-tritium programme, to be performed at the end of the new development phase of JET (1992-96), with the machine operating in the pumped divertor configuration, should allow completion of the JET mission.

(received on 20 January 1993)

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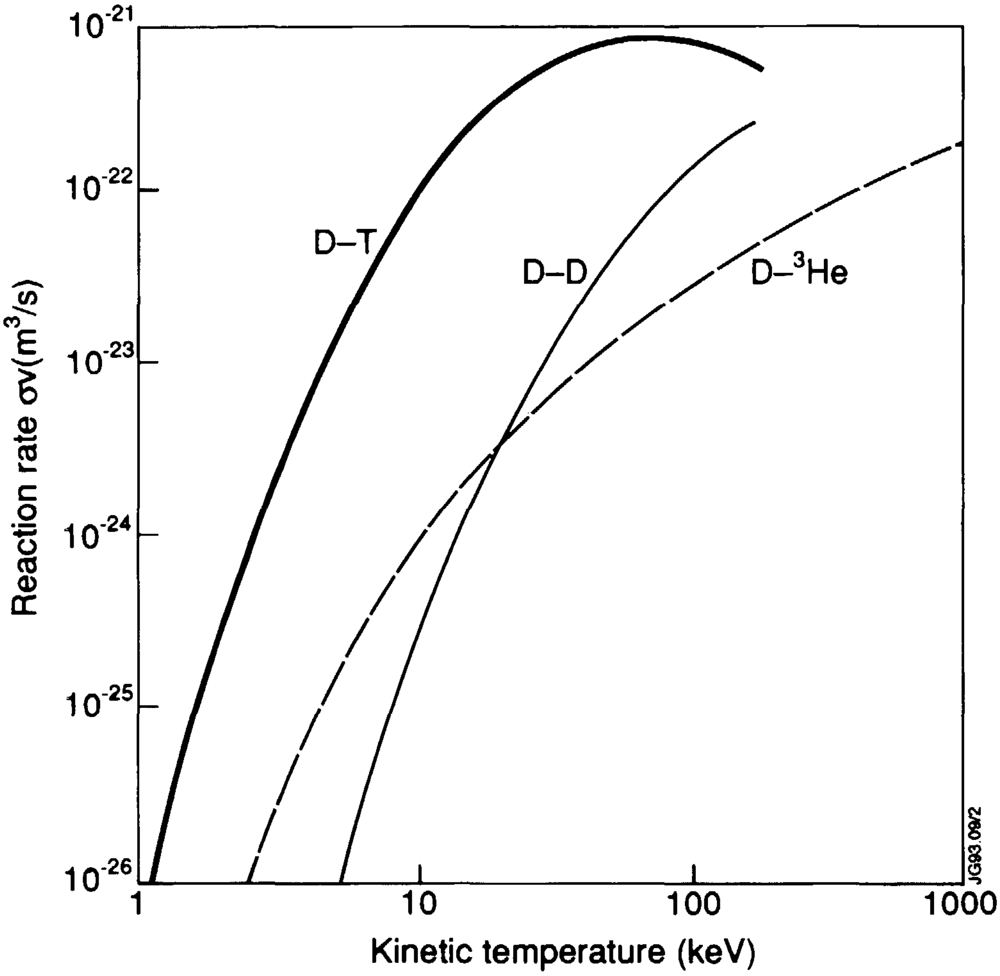

$$D + D = {}^3He + n + 3.27MeV$$

$$D + D = T + H + 4.03MeV$$

$$D + T = {}^4He + n + 17.6MeV$$

$$D + {}^3He = {}^4He + p + 18.3MeV$$

Fig. 1. Reaction rate versus kinetic temperature for the three main fusion reactions

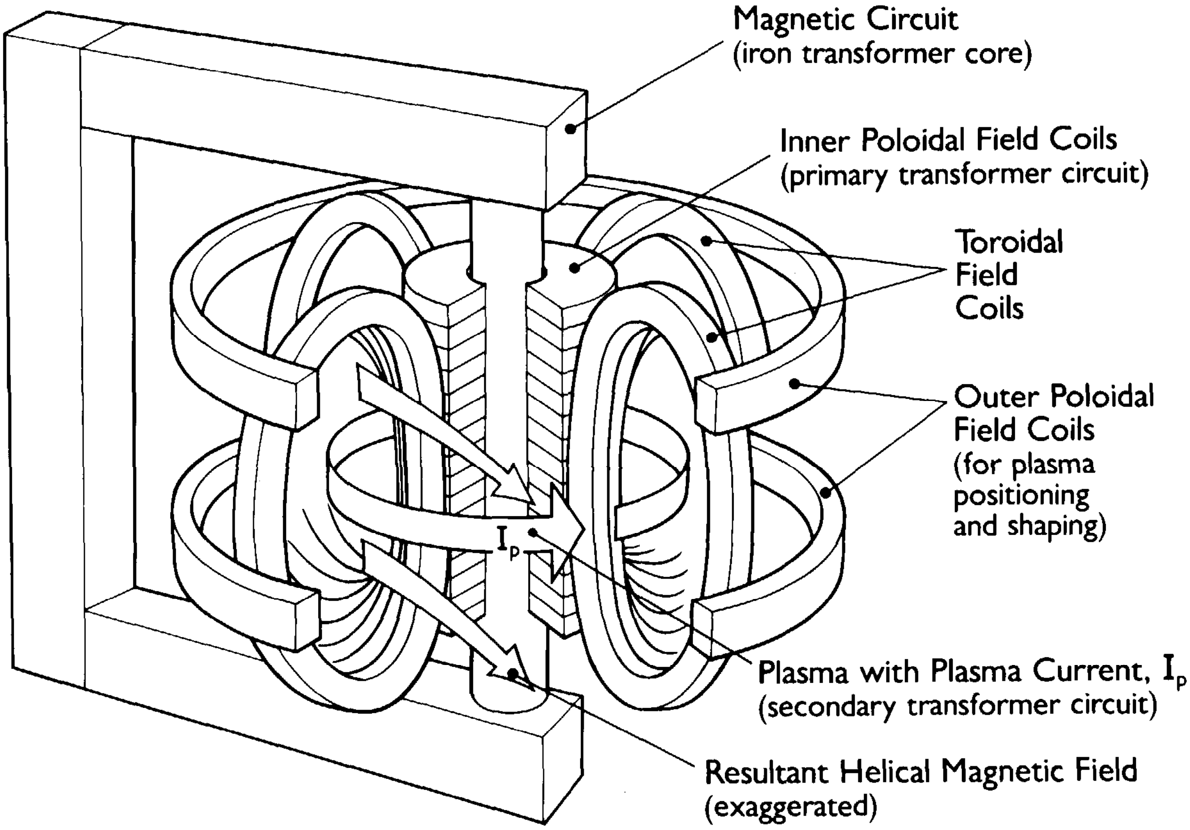

Fig. 2. The tokamak concept

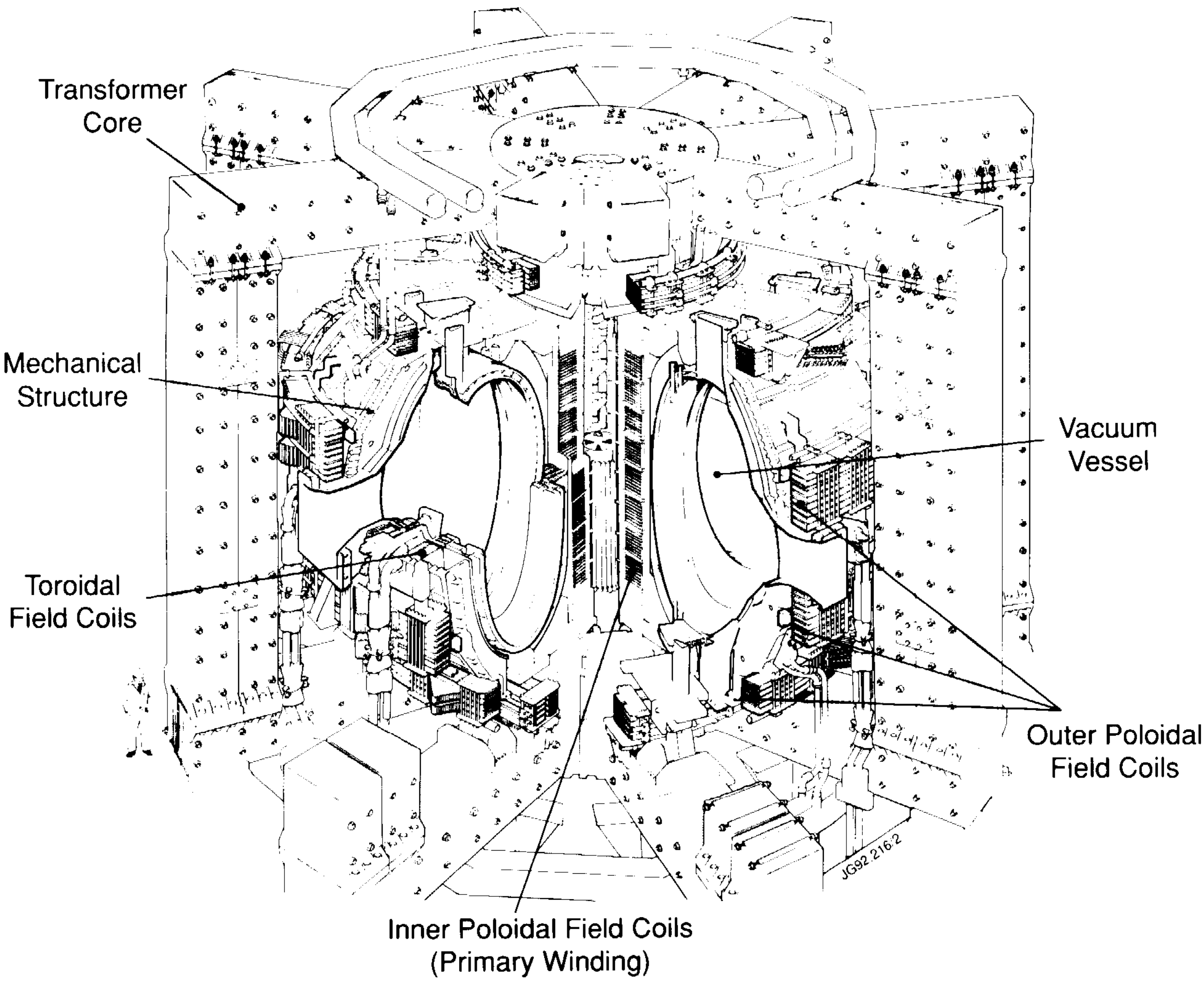

Fig. 3. Schematic view of JET showing main components and key features

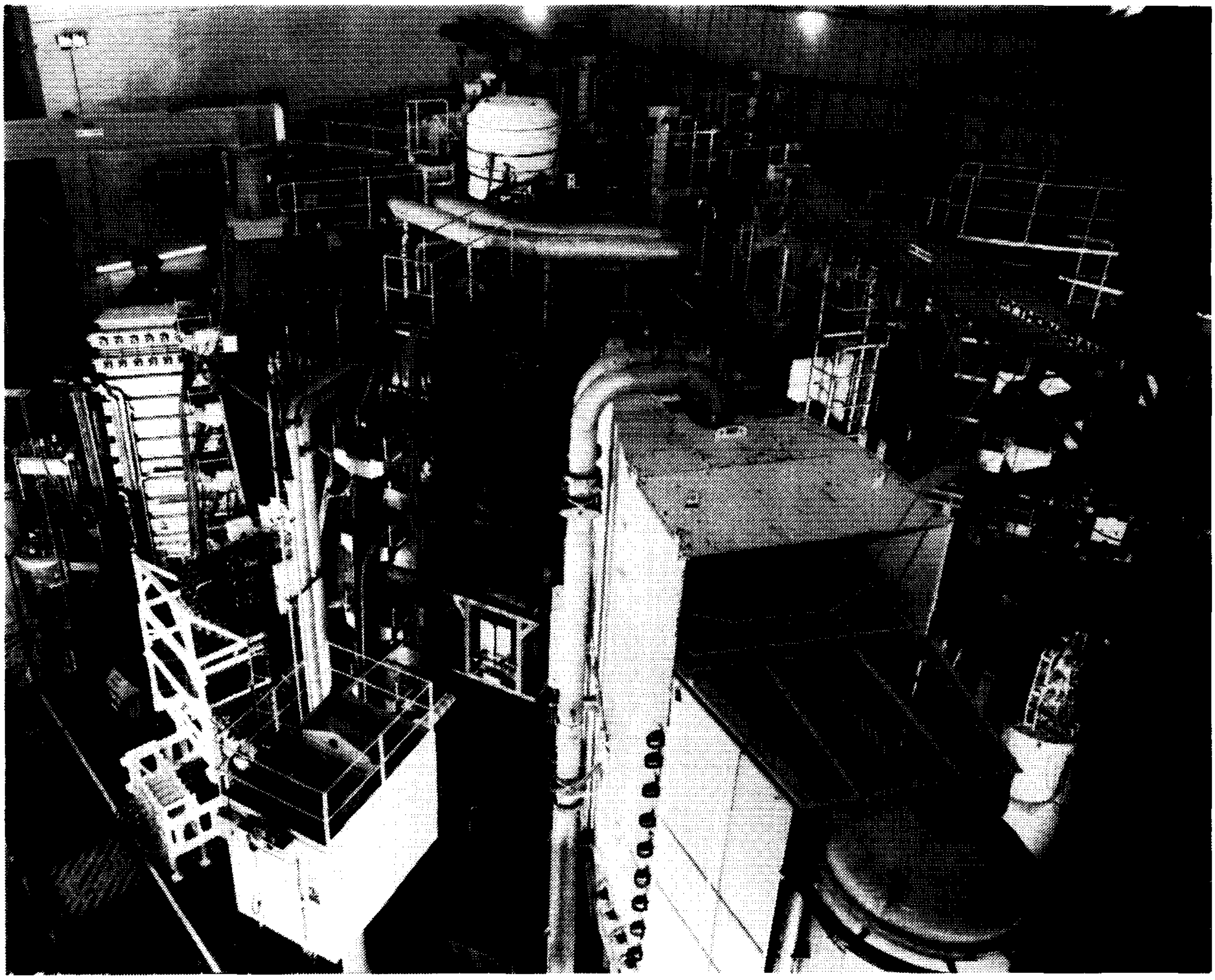

Fig. 4. Overall view of the JET tokamak

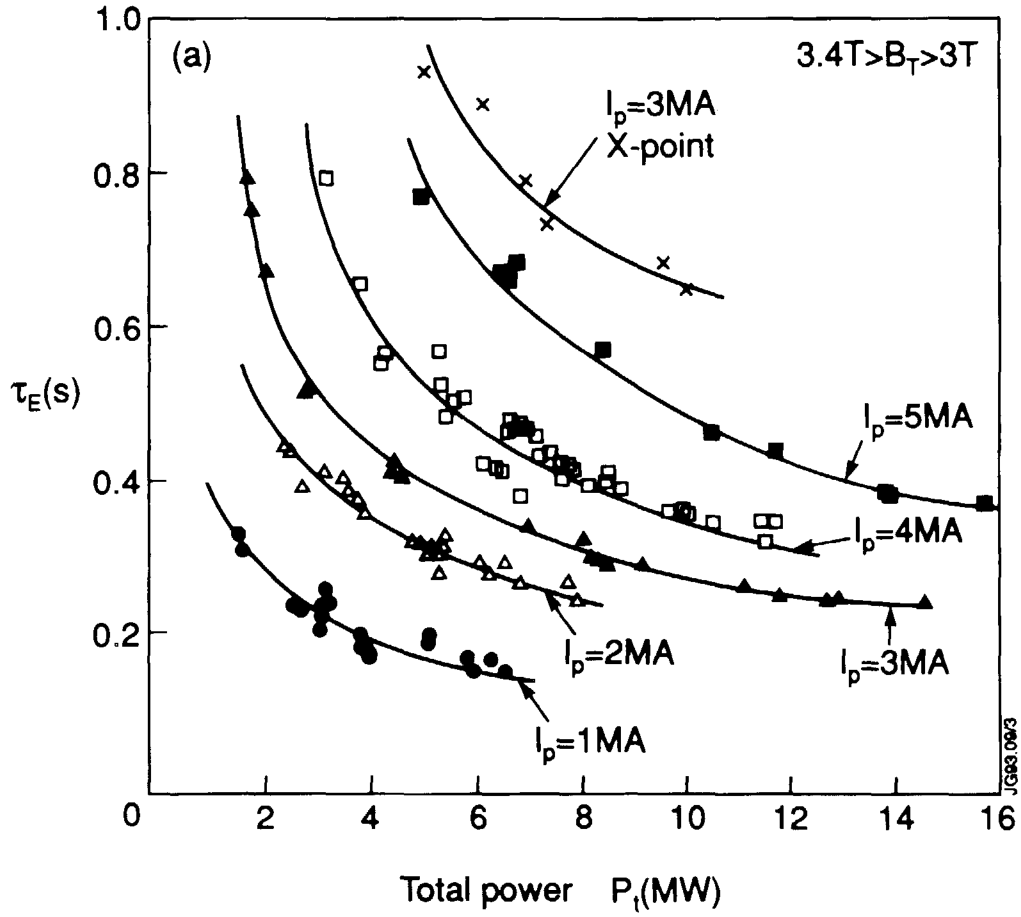

Fig. 5. Decay of the energy confinement time with additional heating and increase of energy confinement time with plasma current in limiter configuration (L-mode) and in magnetic limiter configuration (H-mode)

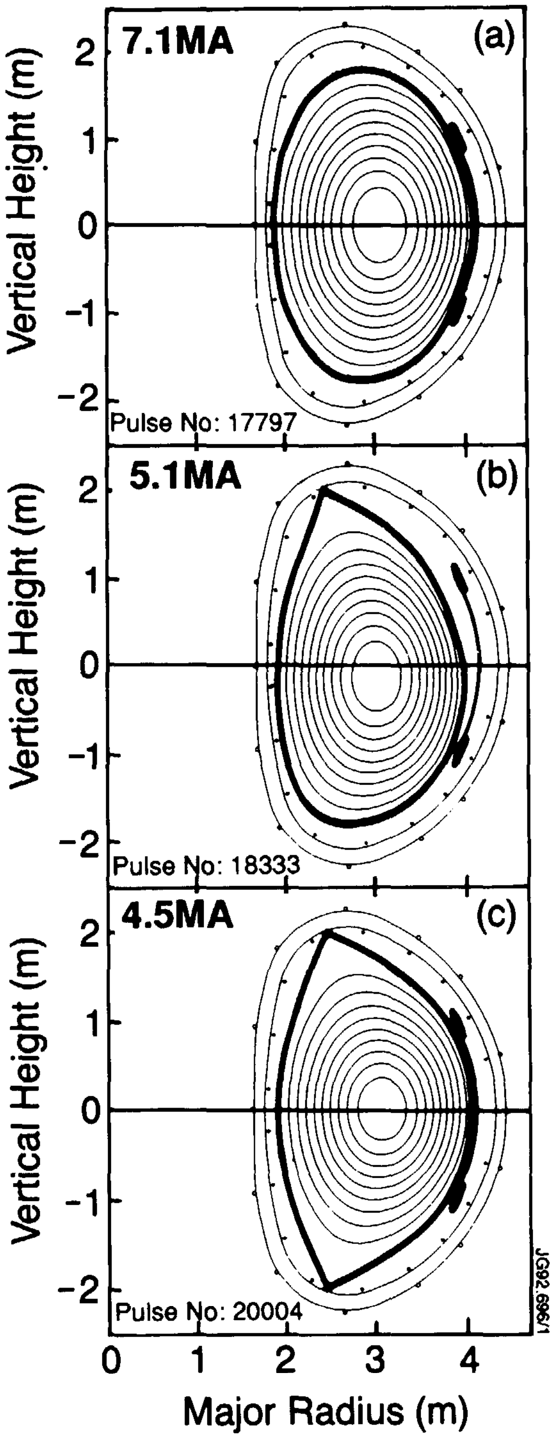

Fig. 6. The three main magnetic configurations available in JET: limiter (a), 7.1MA, single null (b), 5.1MA, and double null (c), 3.0MA

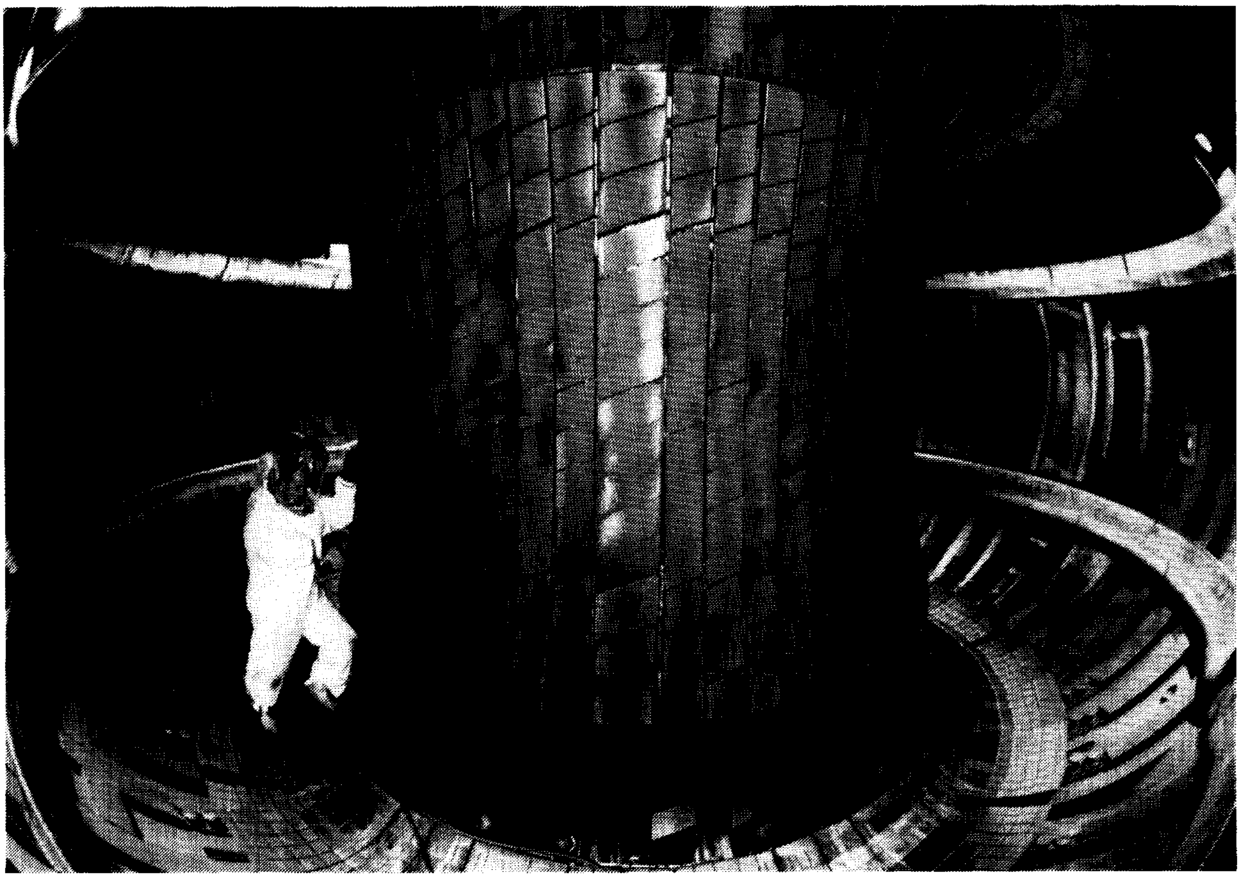

Fig. 7. Vacuum vessel walls in the present JET configuration

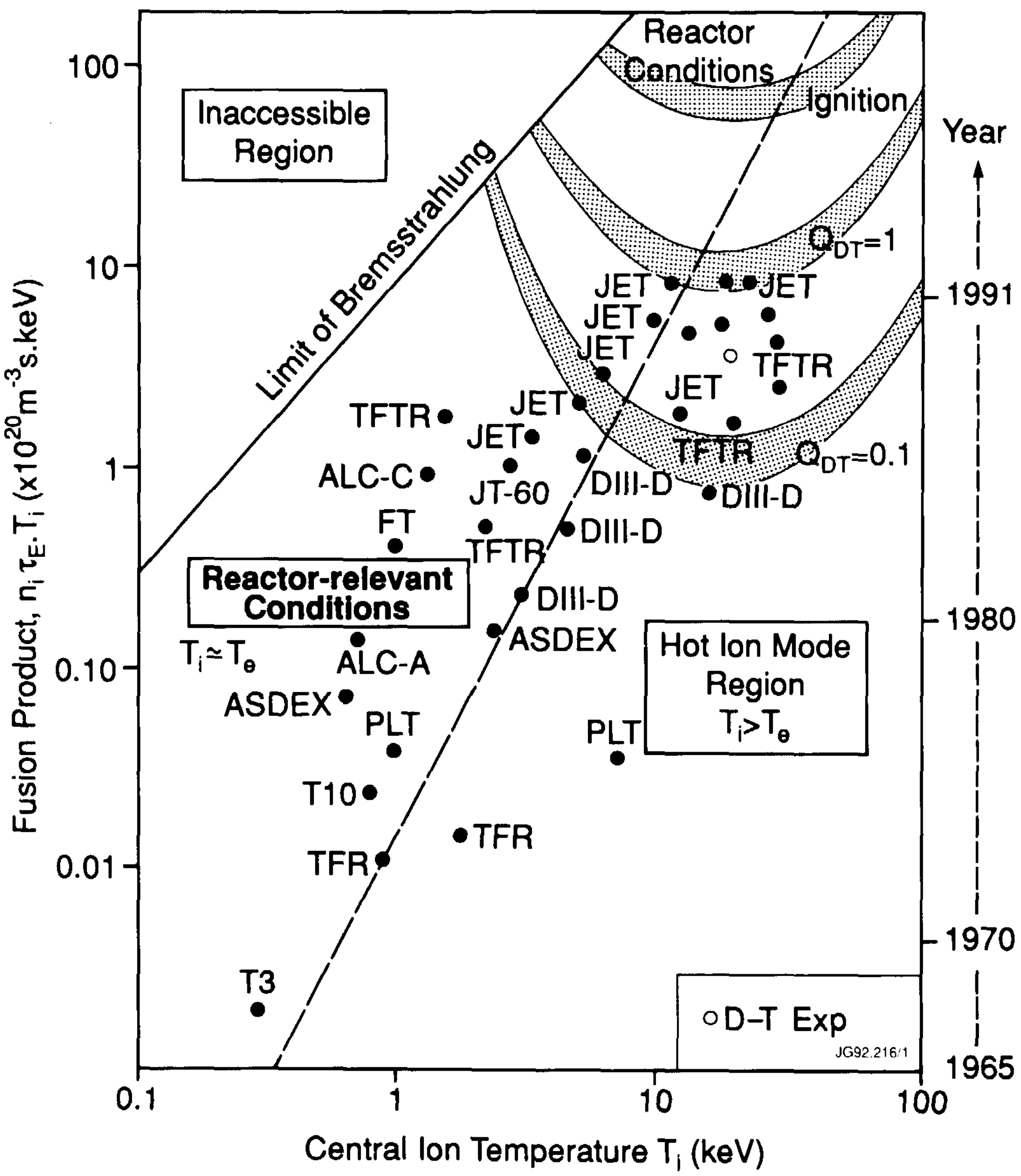

Fig. 8. JET results in the $n_i \tau_E T_i$ - Ti diagram

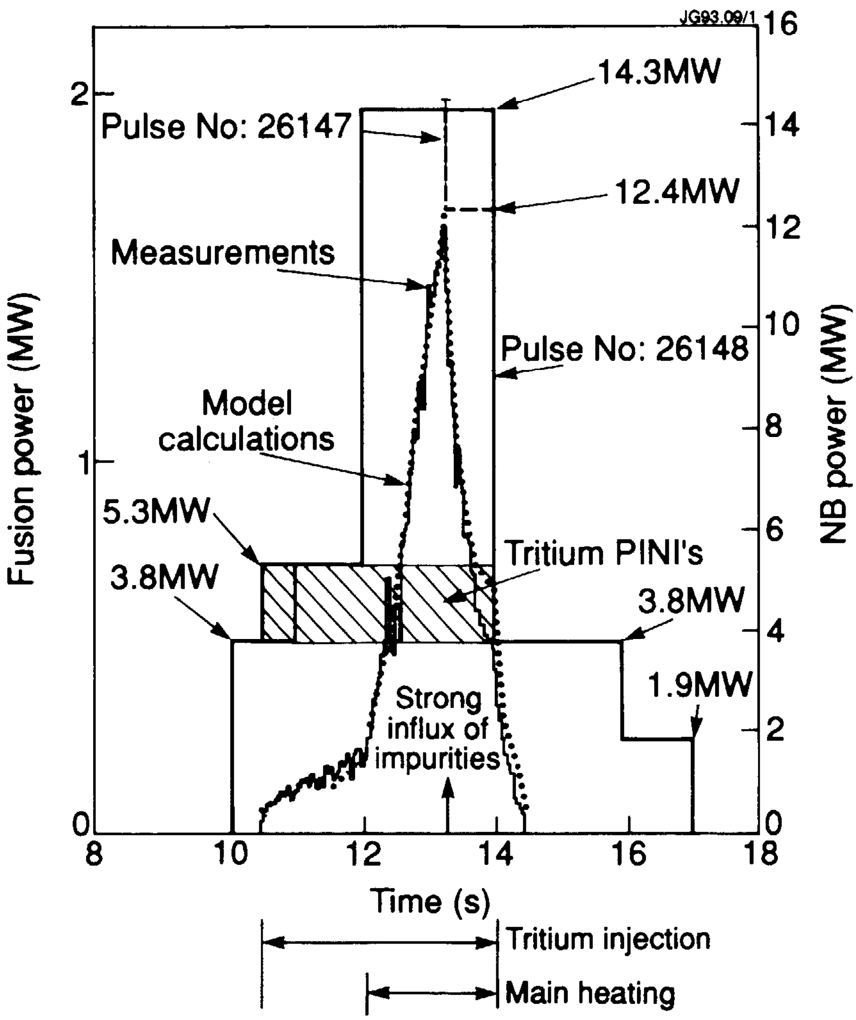

Fig. 9. Neutral beam heating power and 14MeV neutron rates for the Preliminary Tritium Experiment

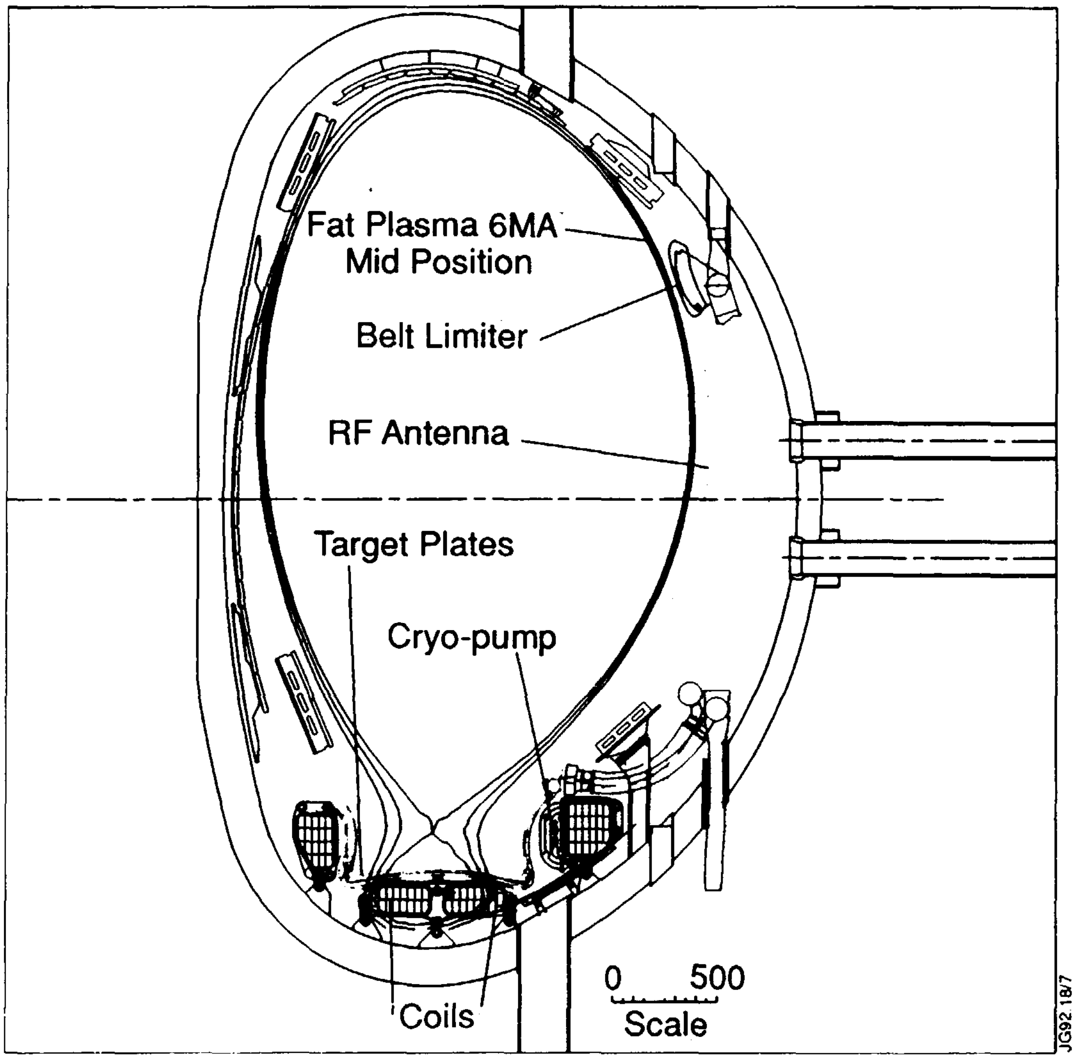

Fig. 10. The key elements of the JET pumped divertor

Table 1 – Main Design Parameters of JET

Parameter	Value
Major radius, R(m)	2.96
Minor radius, a(m)	1.25
Elongation, b/a	1.68
Magnetic field, B_T(T)	3.45
Plasma current, I_p(MA)	4.80
Operating time (at 3MA), t(s)	20
Plasma volume, V_p(m^3)	150

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Table 2 – Progress in JET Performance

Experimental programme	Peak ion* densities $n_i (10^{20} m^{-3})$	Peak ion* temperature T_i (keV)	Energy* confinement time τ_E(s)	Z_{eff}	Fusion product $n_i \tau_E T_i$ $(10^{20} m^{-3} s\ keV)$	Equivalent Q_{DT}
Ohmic heating (1983 -1984)	0.4	3.0	1.0	3–10	1.2	0.01
Additional heating (1984 -1986)	0.5	12.0	0.9	2–5	2.0	0.3
Machine upgrading (1986 -1991)	1.2	20.0	1.2	2–3	2.5	0.3
Passive control of impurities (1988 -1991)	4.0	30.0	1.2	1–2	9	1.07

*Not achieved simultaneously

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