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by

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ABSTRACT

Three years have passed since the Joint European Torus (JET) started operation in June 1983. Phase I of the scientific programme, devoted to ohmic heating studies, has been completed. Phase II, devoted to additional heating studies started in January 1985.

From the technical point of view JET has been entirely successful: indeed the plasma current, an important figure of merit for a tokamak, has reached 5.1 MA for 3s, (exceeding the design value of 4.8 MA). Ion Cyclotron Resonance Heating has added up to 6 MW to the plasma and Neutral Beam Injection has added up to 10 MW.

The energy confinement time in ohmic discharges has reached 0.8s; but degradation has been observed with additional heating. Recently, combined heating (P_{add} up to 14.5 MW) allowed achievement of ion and electron temperatures of ~7.5 keV at densities of ~3 x 10^{19} m^{-3}.

Several proposals for improvements of the JET scientific performance are reported in the paper and summarised in the new development programme.

INTRODUCTION

The JET Project is the key experiment of the European Fusion Programme, which has, as its ultimate aim, the development of a prototype fusion reactor.

In JET, the plasma is confined in a tokamak magnetic configuration, (see Fig 1). A hot plasma of hydrogen isotopes is contained in a large vacuum vessel of toroidal shape and is kept thermally isolated from the wall by a rather complex axisymmetric magnetic field configuration. The main component, the toroidal magnetic field, is produced by 32 toroidal field coils uniformly distributed around the vessel itself. The second component, the poloidal magnetic field, is produced by a current flowing in the plasma itself. This current is induced in the plasma through transformer action pulsing a current of appropriate shape in the primary transformer coils wound along the main axis of the machine. Other coils (equilibrium coils) are used to produce vertical, radial and shaping fields; their interaction with the plasma current controls the horizontal (radial) and vertical position as well as the elongation and the shape of the plasma. The transformer coils and the equilibrium coils constitute the poloidal field coil set.

* (See Reference [20] for definition)

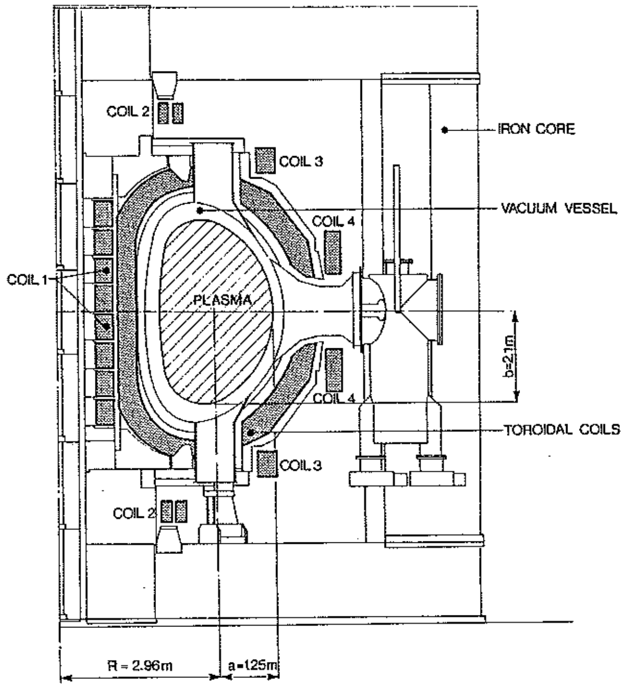


Fig 1: Cross Section of the JET Tokamak.

The plasma current has the purpose both to heat the plasma and to provide a stable equilibrium when below a threshold proportional to the main toroidal field and is a function also of the geometrical parameters. A safety factor, q, is defined as the number of revolutions in the toroidal direction (around the vertical axis) performed by a magnetic line of force to complete one revolution in the poloidal direction (around the magnetic axis).

The objective of JET [1] is to obtain and study plasma in conditions and dimensions which approach those needed for a fusion reactor. The four main aims of the experimental programme are shown in Table I.

Table I: Objectives of JET.

1. The scaling of plasma behaviour as parameters approach the reactor range;
2. The plasma-wall interaction in these conditions;
3. The study of plasma heating;
4. The study of α particle production, confinement and subsequent plasma heating.

Aim 1 is normally qualified by the energy confinement time τ_E, that is the figure of merit of the thermal insulation of the plasma: it is defined as the ratio between the plasma kinetic energy and the necessary power input to sustain it, in steady state conditions.

Aim 2 is normally qualified by the effective ion charge of the plasma, Z_{eff}; that is the figure of merit of the plasma purity. The impurities (all particles in the plasma different from the hydrogenic species and the fusion products) enhance the radiation losses for various reasons and dilute, for a given electron density the ion density that is the important parameter to reach reactor relevant conditions. Aims 3 and 4 are qualified by the comparison of the results obtained with the parameters needed to reach ignition (Lawson criteria). The three key parameters, the ion temperature, T_i, the ion density, n_i, and the energy confinement time have to satisfy the following conditions:

$$n_i \cdot \tau_E \geq 3 \times 10^{20} \text{ s m}^{-3}$$
$$T_i \geq 15 \text{ keV}.$$

Around these values the ignition domain in the Lawson diagram can be approximated with an hyperbola. Therefore only one parameter can be used, sometimes called fusion product $n_i \cdot \tau_E \cdot T_i$ that should reach $50 \cdot 10^{20}$ s m^{-3} keV for a plasma to enter the ignition domain.

The basic design choice of the JET machine defined in 1975 [1], aimed for operation with tritium, to study α-particle production and heating. As a consequence, the machine has the following characteristic features [2], shown in Fig 1 and given in Table II:

- high plasma current;
- large plasma volume and relatively low toroidal field;
- D shape for the toroidal coils, the vacuum vessel and the plasma;
- small aspect ratio (major radius/minor radius)
- poloidal coils all external to the toroidal magnet;
- large access ports;
- modularity of construction (octants);
- life: 10^5 pulses.

From the time of conceptual studies, through the design phase to the construction phase, the experimental evidence gained with other tokamaks and the progress in understanding the physics have created a general consensus in considering the plasma current as a figure of merit for the performance of a tokamak. It is therefore important to stress that JET has, by far, the highest plasma current capability among existing tokamaks.

Aim 4 in Table I had a strong impact on the design concept of JET [3]: all services to the machine including the power supplies for the magnets and additional heating are located outside the Torus Hall. Long busbars or cable connections were therefore necessary.

The two additional heating methods selected were the most advanced at the time when the choice was made (neutral beam injection heating in 1978 and ion cyclotron radio frequency heating in 1981). Beams of neutral particles with very high kinetic energy (80 keV for hydrogen, 160 keV for deuterium) are required to penetrate the target plasma and deposit power in the central region (~ 25% of the volume). The production of powerful beams of neutral particles is rather complex: the basic elements of a Neutral Injection Beam Line are the ion source, the accelerating structure, the neutralizer, the ion dump, etc [4]. The related power supplies are also rather sophisticated because of the fast response time and of the high voltage required [5]. All components of the Neutral Beam Subsystem are designed for long pulse operation (~10s). Two injectors are foreseen in JET each with 8 sources with 4.8 MW of extracted ion beam power per source when operated at full voltage. The total power delivered from the two injectors into the torus is expected to be about 15 MW with 10 MW in the full energy beam component ("high grade power"), both at 80 kV hydrogen and 160 kV deuterium beams.

The Ion Cyclotron Radio Frequency Heating allows RF waves to be coupled to the plasma with large antennae. A wide frequency range was originally chosen (25-55 MHz) in order to heat the JET plasma at full toroidal field with the minority species of hydrogen, deuterium and helium 3. The first group of antennae, called A_0, are designed without active cooling for short pulse operation (1-3s) [6]. The two radiating elements of each antenna are supplied each through long trans- mission lines (83-84m) by 1.5 MW RF amplifier. Eight antennae are foreseen, in total, each driven with two RF amplifiers. The amplifiers will be upgraded to increase the power output up to 2.0 MW for a total of 32 MW, of which 16 MW ("high grade power") are expected to be deposited in the plasma centre (~25% of the plasma volume) and about 25 MW to be delivered into the torus [7].

From the electrical point of view, four different electrical loads can be identified in JET: the toroidal field coils, the transformer primary coils, the equilibrium and shaping field

TABLE II: Main design parameters of JET.

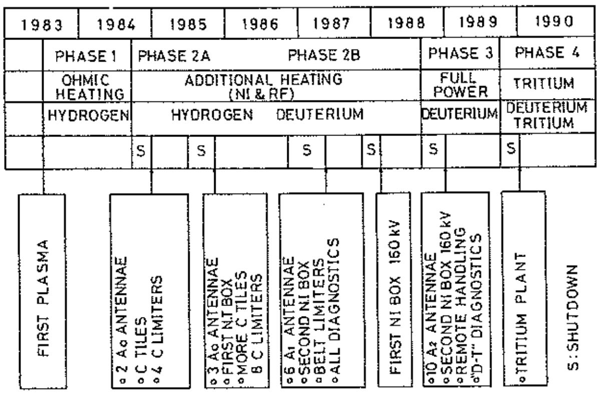
Parameter	Value
Plasma minor radius (horizontal), a	1.25 m
Plasma minor radius (vertical), b	2.10 m
Plasma major radius, R	2.96 m
Plasma aspect ratio, R/a	2.37
Plasma elongation ratio, e = b/a	1.68
Flat top pulse length	10 s
Toroidal magnetic field (plasma centre)	3.45 T
Plasma current, circular plasma	3.2 MA
D-shaped plasma	4.8 MA
Volt-seconds available	34 Vs
Toroidal field peak power	600 MW
Poloidal Field peak power	300 MW
Additional heating power (in plasma)	25 MW
Weight of vacuum vessel	100 tonnes
Weight of toroidal field coils	380 "
Weight of iron core	2700 "

coils and the additional heating equipment. All these pulsed loads require suitable current and voltage waveforms so that the power supply schemes will include AC/DC conversion and power modulation. In JET, as in the majority of the existing tokamaks, the magnets are made with copper coils, water cooled, so that their power and energy requirement are very high. When the additional heating requirements are added, the overall installed power capability is in excess of 1,000 MVA with 11,000 MJ per 20s pulse at a repetition rate of one pulse every 10 minutes. In spite of the large capabilities of the 400 kV supply line, it was necessary to use a combined scheme with two large dedicated flywheel generators that store energy between pulses and provide a large fraction of the peak power required in addition to several AC/DC convertors supplied directly from the mains which are able to provide the majority of the energy [8]. A detailed description of the magnet power supply system of JET is reported in [9], where the various steps in commissioning and operation up to full performance are described.

THE JET PROGRAMME

When JET was approved, five years were allocated to its construction starting in June 1978, and seven and a half years were foreseen to perform the experiments necessary to reach the main aims shown in Table I. The experimental programme of JET is given in Table III showing the four phases originally planned.

TABLE III: JET Development Programme.



Analysis of the table shows that the experimental programme is performed in parallel with a large manufacturing and development programme dedicated to:

- construction of the additional heating equipment;
- development of the remote handling facility and of the tritium plant, both essential ele ment to start phase IV of the programme [3];
- the progressive upgrade of the various diagnostics;
- implementation of the various modifications that the experimental results progressively require.

The first three items can be planned in advance whereas the last item requires some flexibility and re-adjustment of the programme itself.

Phase I dedicated to ohmic heating studies started in June 1983 according to plan. During 1984, the magnet power supplies were progressively commissioned up to their full performance. In summary, the 1984 experiments extended the plasma current to 3.7 MA with 5s flat top, the plasma elongation to 1.6, with full toroidal field 3.45 T; hydrogen and deuterium plasmas were produced with temperatures up to T_e~3keV, T_i~2.5keV, densities up to $3.5 \times 10^{19} m^{-3}$, maximum energy confinement time of 0.8s and Z_{eff} in the range 2.4 - 10. During Phase I, JET operated with 4 graphite limiters (total area $1.28m^2$) and inconel walls, "carbonisation" of the vessel walls has been the method used to control metallic impurities [10].

The planned shutdown took place between October and December 1984, during which two A_0 antennae were installed inside the vacuum vessel. Moreover, its inboard side was protected with about a thousand carbon tiles for a height of approximately 2m (total protected area $23m^2$). Finally, to cope with the stresses caused to the vacuum vessel by vertical instabilities (see next section), additional rigid supports were added at the bottom edge of each horizontal port together with shock absorbers on each vertical port (top and bottom).

In 1985, the experimental programme was partly devoted to additional heating studies with ICRF, (Phase II). The power deposited to the plasma was progressively increased up to 4.5 MW. A review of the results obtained during the early phase of operation with ICRF is given in [11].

The development programme was also continued to achieve the full performance of the machine (including IP = 5.0 MA) [10]. Operation with a magnetic separatrix was also established up to 2 MA [12].

The shutdown took place as planned between the mid-June and mid-November 1985. Four additional graphite limiters were installed increasing the total limiter surface to 2.56 m^2 with a heat load capability of 20 MW. Eight graphite belts, in the poloidal direction, have replaced the original Inconel octant joint protections, (12.9 m^2). A small fraction of the outboard wall in Octant 5 in the area of the main horizontal port has been protected with graphite tiles (3.5 m^2) against possible damage made by the "tangential" beams of the neutral beam injector located in Octant 8. A third A_0 antenna was also installed.

Since operation started, the main effort of the development programme was put on experiments with the plasma leaning on the limiter, on the inner wall or with the edge determined by a magnetic separatrix (magnetic limiter). Moreover, ICRF Heating used all 3 A_0 antennae. Finally, the first injector became operational: deuterium plasmas were injected first, with hydrogen beams (up to 70 keV), and then, with deuterium beams (up to 80 keV and up to 10 MW). A review of the experimental results obtained so far is given in [13].

A review of the status of the machine, of the experience gained in operation and a detailed analysis of the development and enhancement plans, as discussed within the JET Team, during Autumn 1985, are reported in [14]. Preliminary experiments with combined heating were also performed.

The experimental activity will continue with the present machine configuration up to December 1986. A major shutdown will then take place during which the activities foreseen in the JET Programme (Table III) will be implemented. During 1987 and 1988 operation (Phase IIB) the additional heating power will progressively be increased to its maximum capabilities (26 MW of high grade power, 40 MW in total).

TECHNICAL ACHIEVEMENTS

After almost three years of operation, the success of JET from the technical point of view is complete: Table IV shows a comparison between design and operational parameters: indeed the plasma current, already mentioned as an important figure of merit for a tokamak, has exceeded the design value: pulses with I_p = 5.1 MA with 3s flat top have been produced and with I_p = 4 MA for 6s flat top were performed routinely.

TABLE IV: Comparison between design and operational parameters.

Parameter	Design value	Operational values
Plasma minor radius a (horizontal)	1.25 m	0.8-1.2 m
Plasma minor radius b (vertical)	2.10 m	0.8-2.1 m
Plasma major radius R	2.96 m	2.5-3.4 m
Toroidal magnetic field at R = 2.96 m	≤3.45 T	≤3.45 T
Plasma current IP	≤4.8 MA	≤5.1 MA

The plasma current is controlled by feedback; the closed loop is softly brought into operation during slow rise [15],[16]. During flat top, the error is normally ≤3% with response time ~0.5s. The horizontal plasma position is controlled by feedback acting on the voltage applied to a section of the equilibrium coil P4 (see Fig.1) [15], [16]. The system is designed to control the distance from the inboard wall to the last closed magnetic surface that is touching the limiters at its outboard side. During the slow rise and flat top, the error is normally ≤10mm with response time ~30ms. The horizontal position control system can also move the plasma horizontally, decoupling it from the limiters and placing it on the inner wall.

The plasma elongation and shape is controlled by feedback acting on the current flowing in opposite directions on sections of P2 and P3 equilibrium coils [15],[17]. These currents produce a quadrupole poloidal field that helps in elongating the plasma in the range 1.4 - 1.7, at full aperture (maximum horizontal dimensions) and high plasma currents. During flat top, the error in elongation is normally 5% with response time ~20ms.

The plasma vertical position is naturally unstable because of both the quadrupole poloidal field necessary for an elongated plasma and the de-stabilising effects of the ion magnetic circuit, the two contributions are approximately equal, when the elongation is about 1.6. The plasma is stable when the elongation is below a critical value in the range 1.2-1.3. The natural instability (growth rate~50-150s^{-1}) must be stabilised by feedback control of the radial magnetic field produced by current flowing in sections of the equilibrium coils P2 and P3 (opposite direction in the upper and lower coils) [15]. Large vertical forces on the vessel can develop from a failure of this control system. This problem was somewhat under-estimated during JET design and became evident after pulse 1947 during which a vertical force of about 250 tonnes developed on the vacuum vessel. Extensive studies performed afterwards were used to improve the performance of the feedback system, reducing the response time from 4.0 to 2.0ms for small signals. Plasmas with elongations up to 1.9 were experimentally produced (and kept vertically stable)[17]. It was also necessary to strengthen the vessel (action taken in two steps: the first already implemented during the 1984 shutdown, the second will be implemented during the 1986/87 shutdown). With the existing vessel supports, the operating range has been restricted as follows:

$$I_p^2\ (b/a - 1.2) < 5.0\ (MA^2)$$

The final vessel supports will withstand forces up to 1600 tonnes and will allow operation up to I_p = 7 MA at full elongation e = 1.68.

The toroidal subsystem (toroidal field coils and power supplies) has reached the nominal performance of 67 kA that corresponds to 3.45 T on magnetic axis with nominal I^2t=9 $10^{10}A^2s$. The flat top is 12s; (8s are necessary for current rise because only 8 kV are used in order to not stress the interturn coil insulation). Longer flat tops could be obtained, increasing the voltage during current rise and extracting energy during current decay. The limitation on I^2t could also be made less stringent using a lower temperature on the demineralised cooling water (chillers are to be installed on the cooling system).

The poloidal subsystem (poloidal field coils and related power supplies) has been commissioned up to the nominal capabilities of its components: the nominal flux swing of ~34 Vs has been produced.

A dedicated generator is connected to the primary transformer coils (P1) in which, at the end of the premagnetisation phase, the current can reach the nominal 40 kA. This current is then diverted through resistors in order to provide the nominal 40kV across the primary winding. 12kV are normally produced in the experiments with 20kA premagnetisation. This voltage causes the plasma breakdown and fast rise, the generator is then reconnected to the primary transformer coils to raise slowly the plasma current and finally to maintain it constant (slow rise and flat top phases). At the end of the flat top phase the current in the primary coils normally reaches 40kA in the opposite direction (compared to premagnetisation).

In operation, the total flux swing available to raise and sustain the plasma current cannot be used because the nominal premagnetisation current (40 kA) would create stray fields too high for reliable breakdown. Moreover, even assuming successful breakdown and plasma formation, the current derivative would be too high and the plasma would show an excessive disruptive behaviour (probably due to skin currents).

The RF subsystem has three A_0 antennae operational at 3MW each.

The Neutral Injection subsystem has operated the first injector equipped with 8 sources up to 70 kV in Hydrogen with 50 A extracted current and subsequently has been converted to deuterium and operated at 80 kV with 40 A extracted current. The power ("high grade") delivered to the plasma was 10 MW. A new deuterium source has been tested in the JET Test Bed up to 160 kV with 40 A extracted current: this development is to be considered a great achievement both for the source and the power supply system.

The vacuum subsystem has been operated with the vessel above room temperature (main vessel temperature ~300°C, main port temperature ~150°C) The base pressure before the pulse is about 10^{-7} mbar. During the pulse, the density is progressively increased with feedback control on a dosing valve but the system becomes ineffective when degassing from the walls and/or the limiters is too high. A pumping effect from the walls (especially the inner portion covered with graphite tiles) has been observed but not yet understood.

ANALYSIS OF SIGNIFICANT PULSES.

Pulse 7293 gives the record performance in terms of plasma current using only 30 Vs of the 34 Vs available. The plasma current (see Fig.2) rises in 1s to 2.7 MA (fast rise) and in the following 6s reaches 5.1 MA (slow rise). The length of the flat top (3s) is not enough to reach steady state conditions. The average electron density, $\langle n_e \rangle$, continues to increase during current flat top and reaches the maximum ($4 \times 10^{19} m^{-3}$), 1.5s after the end of current flat top. Then the density drops because the plasma is moved to the inner wall. As usual the electron temperature on axis, T_{eo}, shows a maximum (~4keV) during slow rise when the current density profile is still broad. Afterwards, the current density profile peaks on the centre causing q<1 in the region inside the plasma half radius. As a consequence, a relaxation oscillation develops (a phenomenon observed in several tokamaks and known as sawteeth). This characteristic pattern is present also on the electron temperature on axis, T_{eo}, which time average during plasma current flat top is about

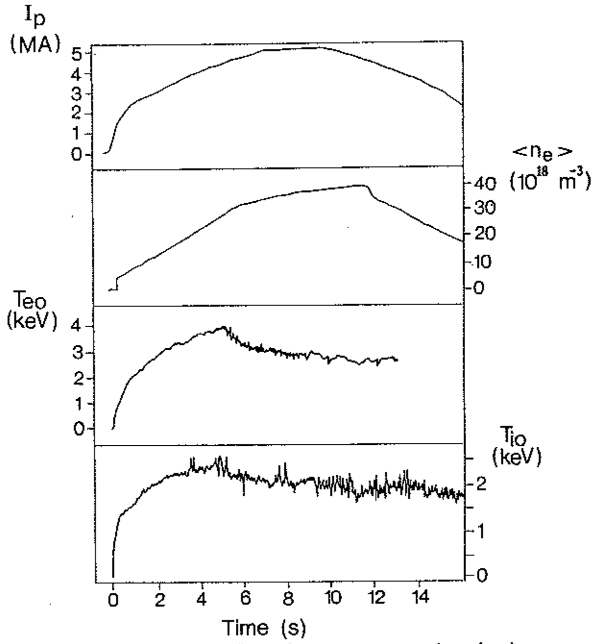


Fig.2: Pulse 7293: I_p plasma current, $\langle n_e \rangle$ average electron density, T_{eo} and T_{io} electron and ion temperature on axis.

2.8 keV. The ion temperature on axis, T_{io}, shows a similar behaviour with 2.2 keV during plasma flat top. The power input (ohmic only) reaches the maximum during flat top (~5 MW), but this number should be treated consciously since the plasma density profile has not yet reached steady state condition. Other significant parameters of this pulse were: toroidal field B_T=3.45 T, elongation e=1.4, energy confinement time τ_E~ 0.6s, impurity content Z_{eff}=2.5.

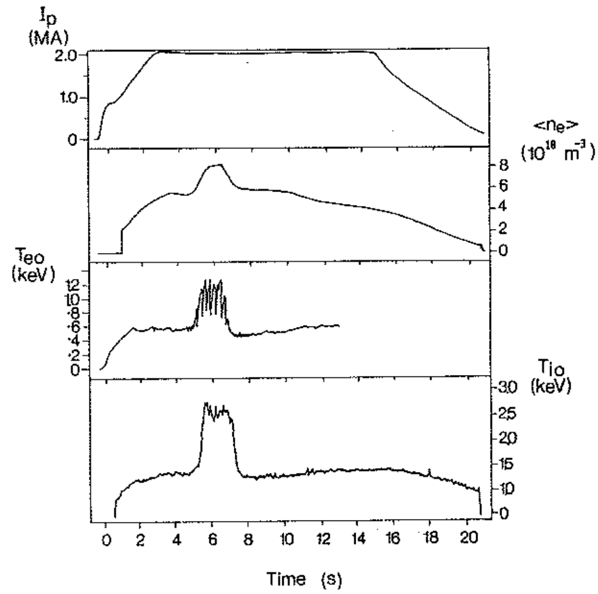


Fig.3: Pulse 7220: I_p plasma current, $\langle n_e \rangle$ average electron density, T_{eo} and T_{io} electron and ion temperature.

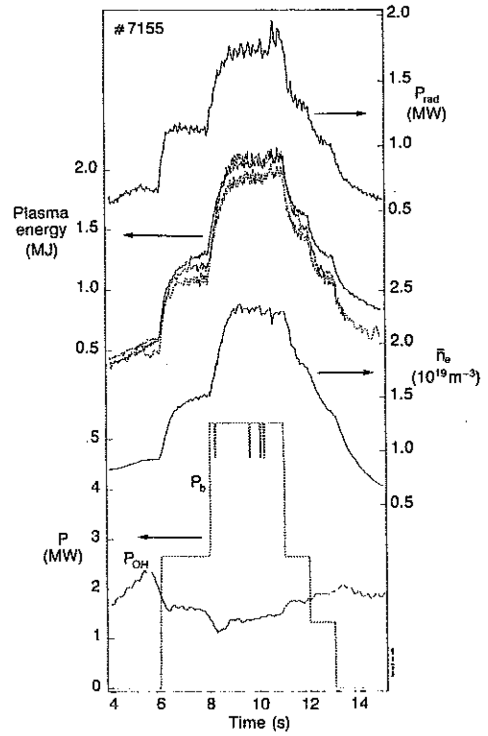


Fig.4: Pulse 7155: P_{OH} ohmic power, P_b neutral beam power, n_e average electron density, plasma energy, P_{rad} radiated power.

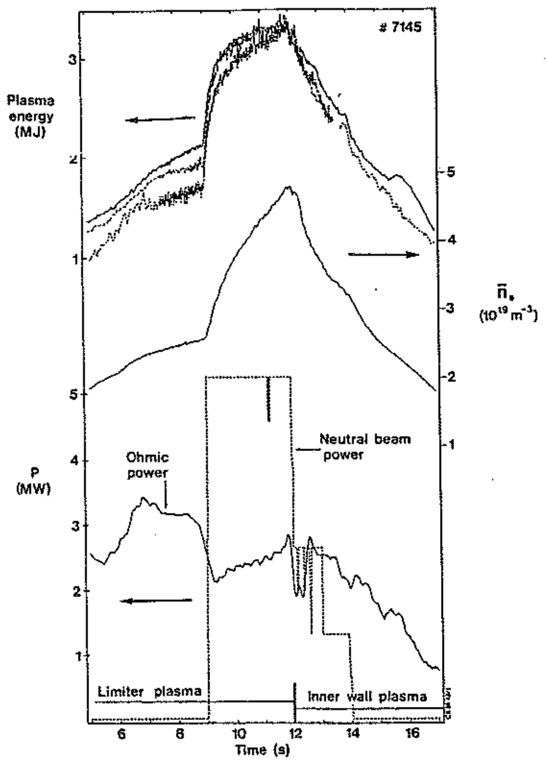


Fig.5: Pulse 7145: ohmic power, neutral beam power, n_e average electron density, plasma energy.

Pulse 7220 shows the consequence of 5.5 MW RF additional heating power (coupled to the hydrogen minority) in a deuterium plasma with 2 MA and 2.3 T. Fig.3 shows the plasma current, the average electron density, the electron and ion temperature on axis. The total power input reaches about 6.0 MW during the 2s RF pulse during which the average electron density increases (about 30%). Giant sawteeth are clearly visible on both electron and ion temperature on axis. The time average temperatures roughly double from the initial ohmic level. The key result is that the plasma stored energy rises from 0.72 to 1.8 MJ only, whereas the total input power rises from 1.5 to 6.1 MW. In other words, the energy confinement time τ_E drops from 0.48s to 0.3s.

Two different plasma regimes can be established with NBI according to the time evolution of the density of the target plasma. The two pulses described in the following are typical of both regimes [18].

Pulse 7155 shows the effect of 5.5 MW NBI additional heating power (hydrogen neutrals at 60 keV) on a low density deuterium plasma with 3 MA and 2.8 T. Fig.4 shows respectively ohmic and neutral beam power, the average electron density, the plasma kinetic energy and the radiated power. The total power reaches a maximum of approximately 7 MW for 3s and is then dropped in steps, the average electron density rises more than 50%. The central electron and ion temperature rise from 4.1 to 4.8 keV and from 2.2 to 6.5 keV. These results are obtained at relatively low density ("hot ion regime"); in order to achieve this, the plasma was moved to the inner wall during current rise. The inner wall pumping effect ensures low initial density when the injector is turned on and mainly minimum density rise during beam pulse.

Pulse 7145 is a typical example of a high density plasma obtained with the same NBI parameters of the previous pulse on a deuterium plasma with 4 MA and 3.4 T which was leaning on the limiters during the beam pulse and was moved to the inner wall during decay. Fig.5 shows the ohmic input power and the beam power respectively. The drop in ohmic power during beam pulse is due to both beam-driven current and a drop in plasma resistivity because of the electron temperature increase and of the impurity content decrease. The average electron density rises sharply during beam pulse from about 2.5 to 4.8 x $10^{19}m^{-3}$ without showing any indication of saturation. The ion temperature shows an increase of 1 keV. The stored plasma energy doubles, reaching 3.3 MJ.

EXPERIMENTAL RESULTS.

Fig.6 shows the fit of the energy confinement time, τ_E, with a widely used scaling law. All JET data obtained in 1984 and 1985 (hydrogen and deuterium) with ohmic heating only are plotted [19]. The strong influence of the plasma geometrical dimension is clearly indicated by the scaling law that confirms the validity of choice made in JET of large plasma volume. An increase in density will be beneficial to increase the energy confinement time. The record performance so far

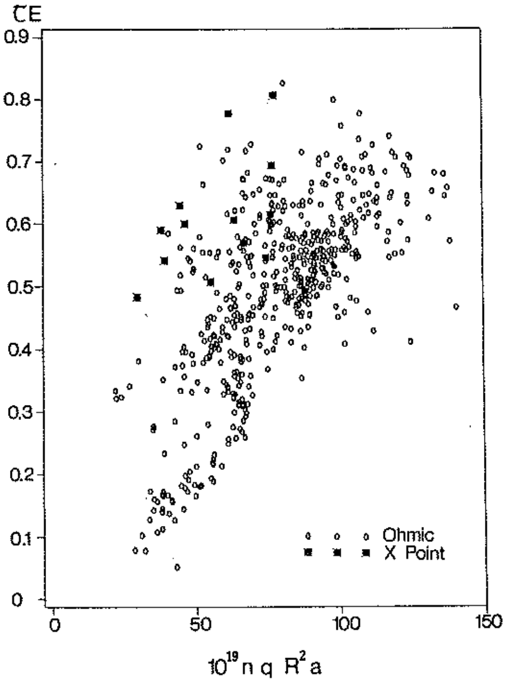


Fig 6: Scaling law of the energy confinement time τ_E

obtained in JET (τ_E=0.8s) with ohmic heating only should be noted. The black points show the results with magnetic separatrix: they seem to indicate a better behaviour of the plasma.

The degradation of the energy confinement time with additional heating already observed in other tokamaks is confirmed by the JET results. To analyse better this phenomenon the plasma kinetic energy is plotted (Fig.7) versus the total input power [20]. From present results a linear behaviour can be extrapolated; an increase with plasma current is to be pointed out. The pattern shown in Fig.7 can be represented by the equation:

$$W = \tau_{Eo} P_{\Omega o} + \tau_a (P_{TOT} - P_{\Omega o});$$

in other words the plasma kinetic energy increases with two different rates, τ_a related to the degradiation of confinement is approximately τ_a=0.3 τ_{Eo}.

Fig 8 [19] shows the effective ion charge versus the ratio of density to current density: an increase in density reduces the impurity content, as confirmed by the experiments with neutral injection. On the contrary, experiments with ICRF show an increase in metal impurity content coming from the RF antennae.

Use of low Z materials for the limiters and wall protection has always been in the JET programme. The two alternatives carbon and beryllium have both been studied. Tiles of both materials are being manufactured so that the final choice can be delayed for the time being. By the end of the 1987 shutdown, 48% of the total vessel area will be covered with low Z material. [21].

The experimental results obtained with many different Tokamaks and confirmed by JET show that

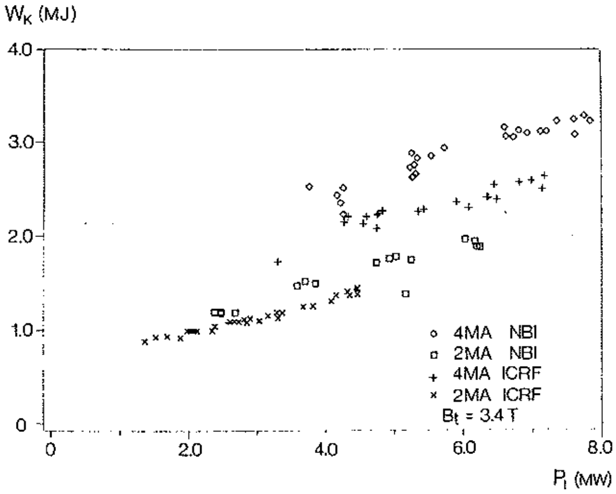


Fig 7: Plasma kinetic energy versus total power input.

the electron density is limited. JET results plotted in the form of the Hugill-Murikami diagram are shown in Fig.9 [18]: the electron density is limited for ohmic and RF heating discharges to

$$n_L \sim 10^{20} \frac{B}{Rq} \quad .$$

With neutral injection heating n_L has been increased by a ~70% showing that the electron density can be increased if the plasma is fuelled in the centre.

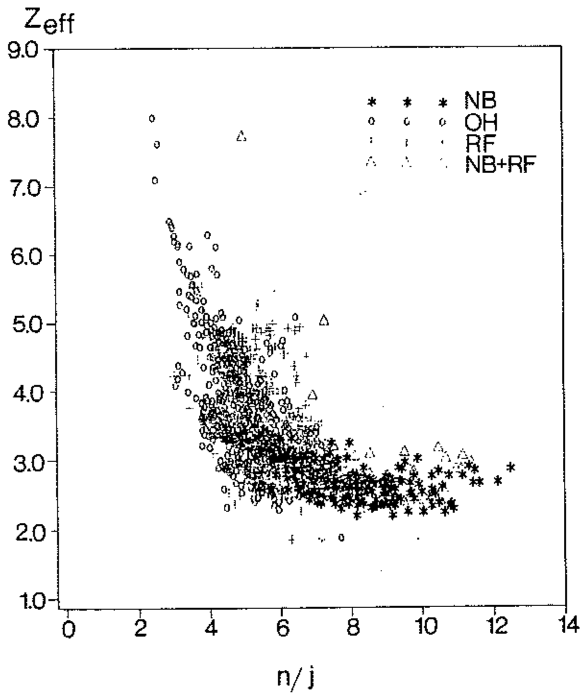


Fig.8: The effective ion charge versus the ratio of average density to average current density.

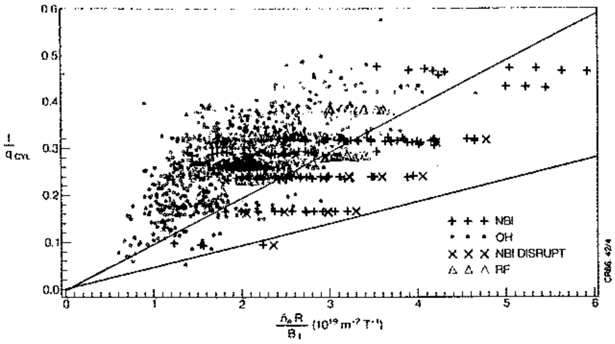


Fig.9: Hugill-Murikami diagram.

Fig 10 [22] shows the JET results plotted in the Lawson diagram. The results obtained with additional heating are only marginally better than the ohmic heating results. The increases in ion temperature and ion density due to additional heating are, at least in part, offset by the degradation in the energy confinement time. The highest value of the fusion product is 6×10^{19} s m^{-3}keV with neutral beam heating only marginally higher than with ohmic heating.

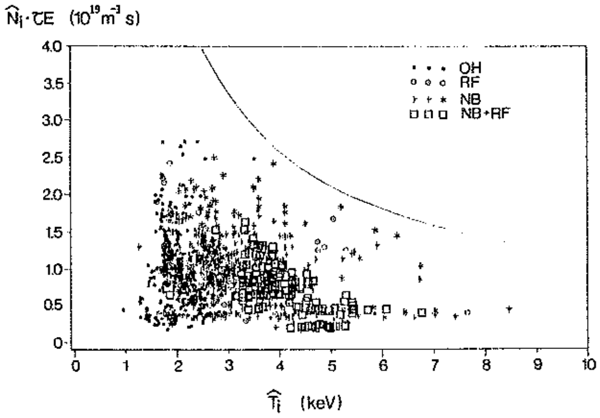


Fig.10: Lawson diagram.

PROPOSALS TO IMPROVE THE JET PROGRAMME

In order to make full use of the total flux swing available (~34 Vs) the primary transformer coils will be increased from 8 to 10, making use of existing spare coils. In this way, the stray fields present inside the vacuum vessel, at full premagnetisation current (40 kA), will be reduced to about the same values that are now produced by 20 kA. Moreover, the voltage across the transformer coils will be reduced in steps to decouple plasma build-up from the current rise. Proper plasma build-up will require high loop voltage (~40 V, 50ms) for fast multiplication of ionised particles (breakdown) followed by intermediate voltage (10-5 V, 300ms) to provide an ohmic power input sufficient to burn through the radiation barrier (I_p~300-500 kA, T_e~T_i>50 eV). The voltage will then be reduced further to meet the requirements on plasma current rise (≲ 1 MA/s). From breakdown onwards, the plasma horizontal position must be kept under control with a vertical field approximately proportional to plasma current, requiring a voltage across the vertical field coils, P4, to be equal to half of the transformer voltage. The new nominal values are 24 kV for P1 and 12 kV for P4.

A non uniform current distribution in the P1 coils (about 40 kA in the central six coils and approximately zero current in the upper and lower two coils) will cause a large stray flux in the top and bottom area of the vessel when the current in P1 (30-40 kA) saturates the iron. This effect combined with appropriate current in the shaping coils (sections of P2 and P3) causes a magnetic configuration with a separatrix inside the vacuum vessel, for plasma current up to 4 MA [23]. Both single null and double null configuration are actively studied: the single null seems more attractive at this stage because plasma currents up to 4.5 MA could be achieved with the X-point 0.2m inside the vessel. A plasma, limited by a magnetic separatrix instead of a material limiter, should show a transition to H-mode behaviour with improved energy confinement time. Preliminary experiments on a magnetic limiter plasma with neutral beam heating have shown improved plasma behaviour.

The following technical implications are carefully analysed:

- localised stresses on TF coils and on the mechanical structure;
- power deposition on inner wall near the X-point (top and bottom of the vessel): additional vessel protection with dump plates will be necessary; water cooled metal plates covered with low Z material are being considered;
- a new power supply configuration to allow different currents in the different sections of P1 is under study.

Preliminary modifications will be performed during the 1986/87 shutdown, whereas the full configuration should become available at the end of the 1988 shutdown.

It is expected that standard plasma discharges in excess of 6 MA, possibly up to 7 MA, are feasible in JET. A task force will be set up in the near future to analyse the various subsystems as built. The aim will be to evaluate how many pulses can be made at increasing performance, taking into account fatigue degradation and built in safety margins. Even in this case, non uniform current distribution in P1 coils will be considered. Current higher than nominal (possibly up to 60 kA) in the central six coils could be accepted since these coils are compressed during the pulse by the toroidal coils working at full field (an aspect not considered in the design of P1 coils).

The performance of the TF subsystem will also be reviewed with the aim of increasing both the amplitude of the toroidal field and its flat top length. These studies should be concluded before Phase III starts (devoted to optimisation studies, see Table III).

It will be necessary to increase the density at the centre of the plasma well above the present level to reach reactor relevant conditions. Experiments performed with other machines (TFTR, ASDEX, ALCATOR, PDX) indicate that this aim could be reached with multipellet injection.

A staged approach [24] has been chosen, as it fits best the JET development programme. A single pellet injector able to launch pellets of 2.6 to 4.6mm diameter at speeds of 1.2-1.5 km/s is being installed and will be operational during the second half of 1986.

During the 1986/87 shutdown, a multipellet injector will be installed: its main features are:

- sizes: 2.6, 4, 6 mm diameter (same length);
- speed: 1.5 km/s for all deuterium pellets;
- fuel: deuterium, hydrogen;
- number of pellets per pulse: 10, 8, 6 for respective diameters;
- propellant gas: hydrogen up to 120 bar.

Finally a development programme has been undertaken to increase pellet speeds towards 10 km/s. A single shot pellet injector with speed in the range 2-5 km/s is expected for installation during the 1988 shutdown, whereas a multipellet injector with speed in the range 5-10 km/s should be installed during the 1989 shutdown.

An increase of central density should be obtained without increasing the edge density. Therefore, a particle exhaust system is required to control the edge density during pulses with additional heating and pellet injection. Moreover, a controlled pulse termination will be achieved reducing the density to ohmic levels at the end of additional heating, during which the density increases well above the ohmic level.

For this purpose a development programme [25] has started, both of a pump limiter system and of pumping panels. Two steps are foreseen for the pump limiter: firstly a prototype will be installed in the 1988 shutdown: the design implies a limiter blade movable radially by about 20-30mm and able to withstand a power loading of about 10 MW/m^2 for 1-2s.

Assuming a density of $10^{20}m^{-3}$, it will decay with a time constant of ~14s. Second, an actively cooled pump limiter should withstand ~30 MW/m^2 for 10s. Swirl tubes covered with 2.5mm carbon or beryllium will be used. Installation is foreseen during the 1989 shutdown.

Pumping panels have been considered because of the pumping effect of the JET walls, especially of the inboard wall (already mentioned). A systematic study of available data has started, and ad hoc experiments will be performed. According to present estimates, 50 m^2 of pumping panels, radiation cooled, would be required. Installation is foreseen during 1988 shutdown.

Control of current density profile [26] could remove the strong correlation now existing between the current density itself and the electron temperature now caused by Ohm's law ($J \propto T_e^{3/2}$). A flat current profile could keep $q > 1$ everywhere in the plasma, thus avoiding the sawteeth. Higher central electron temperatures could be reached.

Radio frequency waves at lower hybrid resonance (~2-4 GHz) have already been used for this purpose in different experiments. Electron cyclotron resonance (60-100 GHz) should produce current drive (according to theory), but it has not yet been observed. Ion cyclotron resonance could also be used; the JET system is now being prepared to attempt some experiments along these lines: phase locked operation of the 8 A_1 antennae is now planned, and will be attempted after the 1986/87 shutdown. Current drive has also been produced with neutral injection.

Among the various options, JET is proposing to make current drive with lower hybrid; 10 MW would be produced with 24 klystrons (phase locked) at 3.7 GHz. Such a system would be capable of driving 1 MA in the plasma.

Broadening of the current density profile in JET would be achieved using the two injectors available (~0.4 MA capability) in counter injection to reduce the current density in the central region together with the envisaged lower hybrid system able to drive up to 1 MA (positive) in the outer region.

CONCLUSIONS: THE NEW DEVELOPMENT PROGRAMME

The various improvement proposals integrated with the already ambitious JET programme, could clearly not be completed by the end of 1990. So a new development programme has been prepared and the necessary steps have been started to seek approval from the various supervisory bodies as required by the JET Statutes.

Table V shows the new JET development programme [13] that should be considered provisional, for the reasons mentioned above. Although two years longer than the original programme (Table III), the new development plan is very tight. One major shutdown per year is foreseen, during which all necessary modifications needed by the following period of operation, are implemented.

Phase II dedicated to the additional heating studies with progressive increase of performance in terms of power added to the plasma, will be reduced by three months compared to the original programme.

Phase III devoted to optimisation studies with full additional heating is now approximately doubled and is divided in two parts A and B with a major shutdown between, during which all necessary modification for the Tritium Phase should be made.

The last shutdown in the middle of 1991 should only be dedicated to further modifications as required by the tritium operation.

The Tritium Phase (IV) is now envisaged to last for one year. During such a period, about 3000-5000 pulses would be performed to study the alpha-particle production, confinement and heating of the plasma.

A large and solid data base should become available for NET (Next European Torus) [27] that will then be the key experiment of the European Fusion Programme.

TABLE V: JET PROGRAMME — SCENARIO 1

1983	1984	1985	1986	1987	1988	1989	1990	1991	1992

PHASE I	PHASE IIA	PHASE IIB	PHASE IIIA	PHASE IIIB	PHASE IV
Ohmic Heating Studies	Additional Heating Studies		Full Power Optimisation Studies		Tritium Phase

5	13	6	13	6	14	7	12	3	12

FIRST PLASMA	OPERATIONAL 8 C Limiters	OPERATIONAL Belt Limiter (C or Be)	OPERATIONAL	OPERATIONAL	OPERATIONAL
	3rd A_0 Antenna	8 A_1 Antennae			
	First NI Box (80kV)	Second NI Box (80kV)	One NI Box at 160kV	Both NI Boxes at 160kV	
		ORNL Multiple Pellet Injector 1.5km/s	Prototype Pellet Injector 2 to 5km/s	Final Multiple Pellet Injector 5 to 10km/s	
		Profile Control (Preliminary System ICRH)		Profile Control (Final System possibly LHCD)	
		X-points— Additional P1 Coils	X-points— Cooled Dump Plates		
			Prototype Pump Limiter	Cooled Pump Limiter	
				Main Modification for Tritium Operations	Final Tritium Modifications
		Vacuum Vessel Restraints	Disruption and Sawteeth Control	All Remote Handling Systems	

The encircled figures show approximate durations in months.

The shaded areas represent shutdowns.

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