JOINT EUROPEAN TORUS

JET–P(87)06

The JET Team
Presented by P.P. Lallia

Overview on 1986 JET Results

Overview on 1986 JET Results

The JET Team
Presented by P.P. Lallia

JET-EFDA, Culham Science Centre, OX14 3DB, Abingdon, UK

Preprint of Paper to be submitted for publication in
Plasma Physics and Controlled Fusion

Introduction

Tokamak operations in JET resumed in November 1985 after a major shutdown devoted to add new systems such as the first Neutral Beam Injection box, new carbon protection in the vessel, a third ICRF antenna and a single Deuterium pellet launcher. During all of 1986, experiments have been pursued and the results reported in particular at the EPS (Schliersee) and IAEA (Kyoto) meetings.

The main parameters of JET are summarised in Table I which shows also the achieved progress since the IV European Tokamak Programme Workshop in Copenhagen. The most significant improvements are

- the extension of 5 MA discharges to a flat top duration of 4.5 s

- the obtention of stable discharges at $q_\psi \approx 2.1$ with $B_T = 1.7T$ and $I_p = 3.5$ MA in preparation of future operations at 7 MA

- a total input power to the plasma reaching 18 MW

- a plasma energy content close to 6 MJ in both cases of limiter and separatrix limited plasmas

- an increase of the temperature of the plasma: up to 12.5 keV for the peak deuterium temperature in low density discharges with NBI and up to 8 keV for the peak electron temperature in H_e discharges with ICRF.

- a record value of the fusion parameter $n_i(o)\ T_i(o)\tau_E$ of $2.10^{20}m^{-3}$ keVs obtained during separatrix limited plasmas with enhanced confinement (H mode).

Section 1 of this summary presents the machine conditions, Section 2 addresses dedicated experiments performed on JET like pellet injection, ultra long sawteeth studies and magnetic separatrix operations. Section 3 is about global plasma behaviour such as impurities, density limit, plasma temperature, energy confinement time and thermonuclear reactivity. Section 4 describes briefly the future changes to be made to JET and the resulting prospects.

TABLE I SUMMARY OF MAIN JET PARAMETERS

(NOT NECESSARILY IN THE SAME PLASMA PULSE)

			JUNE 1985	NOV. 1986
TOROIDAL FIELD	B_T(T)	$\leq$	3.4	3.4
PLASMA CURRENT	I_p(MA)	$\leq$	5.0	5.0
DURATION OF MAX. I_p	t_p(S)	$\leq$	0.5	4.5
PLASMA MAJOR RADIUS	R_o(m)	$\leq$	3.0	3.0
HORIZ. MINOR RADIUS	a(m)	$\leq$	1.2	1.2
VERT. MINOR RADIUS	b(m)	$\leq$	2.0	2.0
ELONGATION	b/a	$\leq$	1.65	1.80
SAFETY FACTOR	q_{cyl}	$\geq$	1.8	1.5
AT				
PLASMA BOUNDARY	q_ψ	$\geq$	2.6	2.1
VOL. AVERAGE ELECTRON DENSITY	$\bar{n}$ ($10^{19}m^{-3}$)	$\leq$	4.0	5.0
CENTRAL ELECTRON TEMP.	T_e(keV)	$\leq$	5.0	8.0
CENTRAL ION TEMP.	T_i(keV)	$\leq$	4.0	12.5
GLOBAL ENERGY CONF. TIME	τ_E(s)	$\leq$	0.8	0.9
FUSION PERFORMANCE PARAM.	$n_i(0)T_i(0)\tau_E$ ($10^{19}m^{-3}$, keV.S)	$\leq$	5.0	20.
INPUT ICRF POWER	P_{RF}(MW)	$\leq$	5.0	7.0
INPUT NBI POWER	P_{NB}(MW)	$\leq$	-	9.0
TOTAL INPUT POWER	P_{TOT}(MW)	$\leq$	8.0	18.0
STORED PLASMA ENERGY	W_p (MJ)	$\leq$	3.0	6.1

1. Machine conditions

1.1 Tokamak status

During the period 1985-86 the toroidal magnetic field (B_T) operated routinely at its maximum design value of 3.4T. The plasma current ($\pm < 3\%$), horizontal plasma position ($\pm \leq 10$ mm), plasma elongation $\varepsilon(\pm 5\%)$ and shape are all controlled by feedback circuits acting on poloidal field coils. Following considerable work on these systems, stable control has been obtained with elongations in the range 1.2 - 1.7. However, the plasma vertical position is naturally unstable due to both the quadrupole poloidal field necessary for the elongated plasma and the de-stabilising effect of the iron magnetic circuit. Loss of vertical position feed-back control at higher elongations can lead to large vertical forces on the vessel. Some additional vessel support has been introduced, but the plasma current, I_p, had still to be restricted within the operating regime given by $I_p^2(\varepsilon-1.2)<5.0(MA)^2$. At present, the full inductive flux (34 Vs) is not used as the maximum premagnetisation current creates stray fields, which inhibit reliable plasma breakdown.

The vacuum vessel is usually operated with wall temperatures at 250-300°C and with a base pressure of 10^{-7} mbar H and 10^{-9} mbar residual impurities. To reduce the level of metallic impurities and oxygen, the torus walls are carbonised by glow discharge cleaning in a mixture of hydrogen or deuterium and hydrogenic methane (CH_4). Difficulties in controlling the density during Tokamak discharges following carbonisation have limited the number of carbonisations performed in 1986. The vessel is also conditioned by glow discharge cleaning in hydrogen and/or deuterium.

Eight carbon plasma limiters are located symmetrically on the outer equatorial plane of the vessel. Since disruptions mostly terminate on the inner walls, these have been covered by carbon tiles to a height of ± 1 m around the mid-plane. Similar tiles also protect the frames of the RF antennae, eight octant joints, and the outer wall from neutral beam shine-through. Additional tiles have been installed to protect the top and bottom of the vessel during X-Point (Separatrix) operation. The total surface area covered is 45 m^2, corresponding to ~ 20% of the vacuum vessel area. The inner wall tiles used as limiters and those for X-Point protection have provided powerful pumping (with speeds up to 100 $mbar.l.s^{-1}$). This has allowed operation at low density near the plasma edge and was used to reduce the density after neutral injection to avoid disruptions. Recently, helium discharges prior to normal operation have improved the inner wall tiles pumping capacity.

1.2 RF Heating System

Three RF antennae have been installed at the outer equatorial wall. Power is transferred to the plasma at a radiation frequency (25-55 MHz) equal to the cyclotron resonance of a minority ion species (H or He^3). Each antenna is fed by a tandem amplifier delivery up to 3 MW in matched conditions. The three units have been regularly operated up to 7.2 MW for 2s pulses. Experiments with 8s pulse duration have also been performed delivering ~ 40 MJ to the plasma. Recently, a fourth RF generator has been installed so that two generators can be coupled to one antenna.

1.3 Neutral Beam System

A long pulse (~ 10s) neutral beam (NB) injector with eight beam sources has been operated on JET since early 1986. H beams have been injected into D plasmas with particle energies (in the full energy fraction) of up to 65 keV. The neutral power fractions were 69%, 23% and 8% in the full, half and one-third energy components, respectively, giving a total beam power of ~ 5.5 MW injected into the torus. D beams have also been injected into D plasmas, with particle energies up to 75 keV (injected power fractions of 76%, 17% and 7%) giving a total power up to 10 MW. Up to 40 MJ have been delivered to the plasma during a pulse.

2. Dedicated Experiments

2.1 Pellet injection

Preliminary deuterium pellet injection experiments have been performed with an injector delivering a single 3.6 or 4.6 mm diameter pellet per JET pulse at a speed of 1 to 1.2 km/s. This was done in various conditions of magnetic configuration (limiter or separatrix discharges) and heating (ohmic only, ICRF or NBI). Pellet injection allows to increase the density limit in JET and reduces the effective charge Z_{eff} of the plasma. Z_{eff} values close to 1 have been observed. In combined operations with NBI peak electron densities exceeding 10^{20} m^{-3} have been obtained lasting 0.5s after pellet injection, with a corresponding peak electron temperature down to 1 keV. "On axis" ICRF heating of high density plasma has not been really attempted but the observed compatibility between pellet injection and ICRF heating gives hopes for experiments with multiple pellet injection in 1987. The pellet ablation is fairly well described by a model taking into account shielding effects provided by neutral gas and plasma surrounding the pellet. Additional heating does not introduce any abnormality to the process.

Following pellet injection in JET a zone of lower electron temperature (by 20%) and higher density (by up to 100%) has been observed at the radius of sawtooth inversion by the Soft X-ray camera and the ECE temperature measuring system. This non axisymmetric zone (nick-named "snake") which extends 0.15 m radially and 0.25 m poloidally can be associated with a m = 1, n = 1 perturbation (m and n being the poloidal and toroidal wave numbers respectively). It can probably be identified with a magnetic island located at the radius where the safety factor q is equal to unity. The perturbation can be observed for as long as 1 second after the pellet injection, implying that the insulation with the surrounding plasma is good and that it survives the sawtooth crash.

2.2 Ultra long sawteeth

Sawtooth oscillations occur in almost all JET discharges. With central deposition of additional power, especially ICRF, sawteeth may develop large amplitudes (up to doubling the central electron temperature) and long periods (up to 0.6s). It has been observed that ultra long sawteeth (nick-named "monster") can occur, i.e. with a duration up to 1.6s. During this time period the central electron temperature can reach values above 7 keV, usually the ion temperature rises too and the thermonuclear reactivity is improved. A strong reduction of the low m, n numbers MHD activity is observed and there is no apparent impurity accumulation. These ultra long sawteeth have initially been observed in combined heating with Hydrogen minority at 2 MA. Since then this phenomenon has been observed in various conditions: NBI only, ICRF only (in Helium plasma) even after the injection of a deuterium pellet, "limiter" or "separatrix" discharges and at values of plasma current up to 5 MA. The only necessary condition seems to be a threshold in the input power per particle of ~ 5 $MW/10^{19}\ m^{-3}$ in Deuterium plasma but down to 3 $MW/10^{19}\ m^{-3}$ in Helium plasma, where charge exchange losses and outgassing are strongly reduced. The mechanism holding the central electron temperature at high temperatures and preventing the usual crash characteristic of sawtooth oscillations is still conjectured. The most plausible one would be a slight change in the current profile, possibly caused by bootstrap current, raising the central value of the safety factor q(0) above unity.

2.3 Magnetic Separatrix operations

Stable discharges with magnetic separatrix (or X-point) inside the vessel have been maintained in JET for several seconds, at plasma currents up to 3 MA with a single null and up to 2.5 MA with a double null. The single null discharges have an elongation of 1.65 as compared to 1.80 for the double null and are therefore more stable against vertical displacements. While interaction of the discharges with the limiters was curtailed, localised power deposition on the top and bottom target plates has up to now limited the total input power to 8 MW.

Without additional heating and when compared to discharges leaning on the limiters or on the inner wall, the Magnetic Separatrix discharges show the existence of a dense plasma near the X-point with an average density 1-2 x 10^{20} m^{-3}, i.e. an order of magnitude higher than the average plasma, and an improved confinement time.

With a Neutral Beam injection power larger than 5.5 MW, a transition to enhanced plasma confinement (H mode) has been obtained in single null operation, at B_T = 2.2 T. The usual features of the H mode are observed; decreased D_α light emission at the plasma boundary, reduced broadband magnetic fluctuations near the X-point, a rise in the plasma density and in energy content, a sudden increase of the electron temperature near the separatrix producing a pedestal to the temperature profile, a flat density profile with steep gradient near the separatrix. The H phase can be sustained for durations approaching 2 seconds. The continuous density rise increases the radiated power from the bulk plasma and is likely the cause of the termination of the H phase though there is no indication of peaking of the impurity profile. While the energy content reaches a quasi steady state for time approaching 1 second, the electron temperature may present a maximum before the end of the H phase. In those conditions a plasma energy content of 6 MJ has been obtained with 8 MW of NBI in addition to an ohmic power of 2 MW and for a plasma current of 3 MA. The global confinement time exceeds by more than a factor of 2 the value obtained in "limiter" or inner wall discharges. It has been observed that increasing the magnetic field raises the power threshold required to get an H mode, which has up to now prevented a transition to occur above 2.8 T. H modes have not yet been achieved in "double null" operations.

ICRF heating has been applied during X-point discharges, alone or in combination with NBI. In optimum conditions to get an H mode with NBI, the distance of separatrix to antenna is close to 10 cm and the coupling resistance of the ICRF antenna is halved when compared to "limiter" discharges. The plasma was moved radially to decrease this distance but no H modes were observed in D plasma with ICRF alone up to the operational limit of the input power. Furthermore, at moderate power ($\geq$ 1 MW), ICRF usually provokes the termination of an H phase previously triggered by NBI. This difference between heating methods may be linked to the usual increase in charge exchange and in radiation losses at the plasma boundary during ICRF pulse. Indeed, when these losses are reduced, as in ^3He plasma with proton minority, some of the features of an H mode are observed with ICRF only, namely reduction of broadband MHD fluctuation level, decrease in light emission at the plasma periphery while the density is rising, attainment of $T_e(0)$ and $T_i(0)$ close to 8 and 6 keV respectively for an ICRF power of 6.5 MW at a volume averaged density of 1.5 10^{19} m^{-3}.

3. Global Plasma Behaviour

3.1 Impurities and effective charge of the plasma

In most of the cases, impurity radiation losses are mostly caused by carbon and oxygen and originate mostly from near the plasma edge. In limiter discharges, the metal concentration was only significant (i.e. > 0.1% n_e) if the carbon limiters were metal-coated subsequent to accidental melting and evaporation of wall material. The release of metals from the limiters can be explained by a combination of sputtering by deuterium and by light impurities. The metal fluxes decrease as the plasma density increases and the plasma current decreases. Those fluxes are inversely correlated with the light impurity behaviour. At high plasma density radiation losses were mainly caused by oxygen.

The effective plasma charge Z_{eff} ranges usually between 2 and 3 for line averaged electron density $\bar{n}_e$ greater than 3.10^{19} m^{-3}. Z_{eff} was reduced and approached unity for a time duration exceeding 0.5 s after the injection of a deuterium pellet.

3.2 Density limits

In ohmic plasmas, the density limit is n_c (OH)(m^{-3})=1.2 x 10^{20} B_T (T)/qR(m). This limit depends on plasma purity. In RF heated discharges, it is only slightly increased, possibly because the effect of the extra power is cancelled by an increase in impurities. In neutral beam heated plasmas, the limit is substantially increased to $n_c(NB)(m^{-3})=2.0x10^{20}B_T(T)/qR(m)$. Switching off neutral beams at high density causes the plasma to disrupt, which indicates that the power input plays an important role in the disruption mechanism. Preliminary experiments with a single-shot pellet injector have also exceeded the OH density limit as mentioned in Section 2.1.

3.3 Plasma Temperature

Electron and ion temperature have been observed to react differently on the application of additional power. At a power input per particle $P/\bar{n} > 4$ MW/10^{19} m^{-3} the ion temperature on axis can greatly exceed the electron temperature and can reach 12.5 keV in JET. The ion temperature profile is broadened in cases of off-axis ICRF heating.

The peak electron temperature can be raised significantly above the value reached in the ohmic phase as exemplified by the ultra long sawteeth where $\Delta T_e(0)$ has exceeded 4 keV. By contrast, the electron temperature at the sawtooth inversion radius (q ~1 surface) shows only a weak dependence with the power input per particle $P/\bar{n}$ for a given plasma current and toroidal field; T_e is increased from 2.5 keV to 4 keV when $P/\bar{n}$ varies from 2 to 8 MW/$10^{19}m^{-3}$. The normalized temperature profile outside the inversion radius can be very well described by a Gaussian type expression, $T_e(R) = T_e$ $(R = 4m)$ exp $\alpha_T[1 - (R/4)^2]$, where R is the radial position relative to the vertical axis. It is found that $\alpha_T = 1.8 \pm 0.1$ for the vast majority of the JET plasmas. Only very highly additionally heated and high q discharges are fitted by a different value of α_T. In that respect the electron temperature profile while stiff, is not rigidly "consistent".

3.4 Energy confinement

The definition of the total energy confinement time used at JET is $\tau_E = W_k/[P_t - dW_k/dt]$, where W_k is the kinetic energy and P_t is the total input power to the plasma without subtracting radiation losses. Reported values of τ_E are quasi-stationary.

Independently of the type of additional heating, whether RF, NB or combined, the confinement time degrades with increasing input power as seen in a number of other experiments. The rate of increase in W_k with $P_t (=\Delta W_k/\Delta P_t)$ appears to reach a limit of 0.1 - 0.3 MJ/MW (=s) at high powers. This suggests a lower limit to the global confinement time, τ_E, of 0.1 - 03s in JET. The confinement time depends weakly on plasma density but scales favourably with plasma current.

While the global confinement time results could be fitted by a Goldston type law, the measured radial propagation of heat pulses consecutive to sawtooth crashes strongly supports a linear offset relation between W_k and P_t in the form

$$W_k \, (P_t) = W(0) + \tau_{inc} \, P_t$$

The best fit to the data set has been found by use of standard regression techniques in case of "limiter" or "inner wall" discharges. It gives $W(0) = 0.225 \, n^{0.6} \, I^{0.5} \, B^{0.4}$ and $\tau_{inc} = 0.047 \, I$, where the units are MJ, $m^{-3} \times 10^{-19}$, MA, T, s. Note that, in order to be consistent with the dimensionless variables of plasma physics, a dependence on size and temperature should exist and τ_{inc} look like a sort of poloidal Bohm time.

In the H mode, the energy confinement time was observed to be more than twice the one obtained in limiter discharges. A degradation is also observed when the input power is increased and a similar offset linear scaling of W_k with P_t is quite likely. It is too early to say how the corresponding incremental confinement time compares with the one obtained in limiter discharges, but in both cases higher plasma current looks beneficial.

3.5 Thermonuclear Reactivity

The best performances regarding thermonuclear reactivity for various conditions are summarized in Table II. The record value of the fusion product $< n_i(o) T_i(o) \tau E > \approx 2 \; 10^{20} \; m^{-3}$ keVs has been achieved with 10 MW of Neutral Injection during an H mode. For limiter or inner wall discharges the values of the fusion parameter are similar for ohmic heating only, RF, NBI and combined

TABLE II

Maximum Values of $< n_i(0)\tau_E T_i(0)>$

Experimental Programme	Peak Density $n_i(0)$ ($x10^{19}m^{-3}$)	Energy Confinement time τ_E (s)	Ion Temperature $T_i(0)$ (keV)	Fusion Parameter $<n_i(0)\tau_E T_i(0)>$ ($x10^{19}m^{-3}$.s.keV	Q_{DT} Equivalent Q_{DT}	Plasma Current I_p (MA)
Ohmic (4.6 MW)	4.2	0.8	3.0	10	0.010	5
ICRF (7 MW)	3.7	0.3	5.4	6	0.012	3
NBI (6 MW)						
High "	4.4	0.4	4.0	7	0.10*	3
Low "	1.5	0.4	10	6	0.20*	3
Combined NBI + RF (14 MW)	5.0	0.4	3.5	7	0.10*	3
X-point (NB - 10 MW)	5	0.65	6	20	0.15*	3

*Beam-Plasma reactions are dominant

heating cases, degradation in confinement time with additional heating off-setting gains in the other parameters. Neutron yields up to $3x10^{15}$ s^{-1} have been obtained in "hot ion mode" with Neutral injection, but mainly from beam plasma D-D reactions. The best ratio of fusion power to input power obtained was $Q_{DD} = 3.5x10^{-4}$ which is equivalent to $Q_{DT} \sim 0.2$ and would have corresponded to a fusion power production of above 1 MW.

4. Future plans and prospects

JET is presently in a major shut-down phase during which new equipment and diagnostics will be installed. In particular strengthening of the vacuum vessel, modifying the primary winding and improving the power supply will allow an enhancement of the plasma current to 7 MA at the full elongation of $\varepsilon = 1.7$ and to 4 MA in magnetic separatrix operations. The coverage of the inner wall by protection tiles will be extended and two toroidal 'belt' limiters will replace the existing ones. The presently installed tiles are made of Carbon while Beryllium will be used later on. A second Neutral Injection Box and the installation of 8 water cooled RF antennae will allow application of up to 44 MW of additional heating power. A multipellet injector, built in collaboration with Oak Ridge National Laboratory (U.S.A.), will be also available at the restart of plasma operations.

The present results give confidence that α particle production in JET will be significant as the equivalent Q_{DT} should be close to 1 in both configurations, namely 7 MA with limiters and 4 MA in single null operations. In these conditions 2/3 of the fusion power would result from beam plasma reactions and further improvements are needed to increase the genuine thermonuclear output. Further actions on the plasma profiles by use of Lower Hybrid Current Drive, pump limiters and high velocity pellet injectors are actively prepared for installation after 1988.