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SPRING COLLEGE ON PLASMA PHYSICS

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RESULTS OF THE JOINT EUROPEAN TORUS

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B J Green (May 1987)

1. Introduction

The Joint European Torus (JET) is the largest tokamak at present in operation. It was built and is operated by the JET Joint Undertaking whose members are organisations from the members states of the European Economic Community (EEC), and of Sweden and Switzerland.

1.1 Objectives of JET

The objectives of JET are:

i) To study the way the plasma confinement properties scale as the dimensions and parameters approach those necessary for a fusion reactor.

ii) To examine and control the plasma-wall interaction and impurity influx under these conditions.

iii) To demonstrate effective heating techniques, capable of producing high temperatures in JET in the presence of prevailing loss processes.

iv) To study alpha-particle production, confinement and subsequent plasma interaction and heating i.e. the physics of a reacting plasma.

1.2 Extrapolation of JET over Previous Small Tokamaks

When the size of JET was essentially determined (1975) the typical dimensions of tokamaks were

minor radius - 20 cm
major radius - 70 to 100 cm

and plasma currents were typically several hundred kiloamps ($\lesssim 0.4$ MA on TFR (France)) for durations of several hundreds of milliseconds.

This is to be contrasted with the JET parameters.

Plasma minor radius (horizontal)	a	1.25 m
Plasma major radius (vertical)	b	2.1 m
Plasma major radius	R_o	2.96 m
Plasma current (circular cross-section)		3 MA
Plasma current (full D-shaped aperature)		4.8 MA
Plasma duration		up to 20 s.

The performance of small tokamaks was as follows:

Best peak ion temperature (TFR) about 1 keV
Best peak electron temperature (TFR) about 3 keV
Best density-energy confinement time product about 10^{12} cm^{-3}s (TFR and Alcator (USA))

The extrapolation to JET from these smaller tokamaks in terms of size and performance was very large, and therefore predictions were considerably uncertain. Nevertheless the JET design was based on the improvement in plasma confinement with increasing size and plasma current observed in small tokamaks.

1.3 Developments in Tokamak Research between the JET Design Freeze (1975) and the JET Operation Start (1983)

The finalisation of the design (1975-1978), the establishment of the JET Joint Undertaking and the construction of JET (1978-1983) was a period of rapid development in tokamak research as medium-sized tokamaks were built and began operating, and as larger amounts of additional heating power was applied to these plasmas to boost their ohmic-heated performance. Clearly the design of JET had to be such as to allow for modifications which might arise as a result of greater understanding based on the results of medium-sized Tokamaks.

Many results were obtained in this period and in the following they are classified as 'good' or 'bad' for JET.

'Bad for JET'

1. Impurity control most effectively performed by divertors, not by limiters.

2. Ohmic-heated density in tokamaks is limited so that the mean density $\bar{n}$ is proportional to the mean current density $\bar{j}$. This is not good for large machines with necessarily moderate toroidal field (and hence moderate current density).

3. A limited current density with prevailing energy loss scaling implied verylimited plasma temperatures with Ohmic-heating.

4. The density limit is also dependent on the plasma impurity content.

5. The density limit is set by disruptions, which are dangerous for the machine.

6. Increased additional heating apparently decreases the energy confinement (so-called L (low) mode scaling)

'Good' for JET

1. The energy confinement time improves with plasma size (classically as a^2).

2. The energy confinement time improves with mean density ($\propto \bar{n}$,classically as $1/\bar{n}$).

3. The plasma confinement improves with atomic mass ($\propto \sqrt{m_i}$ (classically as $1/m_i$).

4. Additional heating at the megawatt level effectively increases the mean density ($\bar{n}$) and mean temperature ($\bar{T}$).

5. The prediction of very temperature-sensitive plasma losses (trapped-ion instabilities) at 'high' temperatures (7 keV on PLT (USA)) was not experimentally substantiated.

6. Cross-sectional plasma shaping allows increases in the stable plasma beta ($\beta = 4\mu_0 nkT/B^2$) which can be reached.

7. Empirical β limit favours low aspect ratio ($A = R_0/a$) machines (β_{lim} 1/A).

8. A H (high) mode of confinement where the degradation with increasing additional heating power disappears. This was found in diverter (PDX(USA)) and an expanded plasma boundary i.e. magnetic limiter or X point (D-III(USA)) experiments.

1.4 Types of JET Operation

There are many different types of JET operation which are tabulated below. The basic form of plasma heating is ohmic heating (OH) associated with the large plasma current (required also to produce its poloidal field which is an essential contribution to the toroidal magnetic confinement).

The additional forms of heating are

Neutral beam (NB) heating: one box of 8 ion sources producing beams of hydrogen (energy 65 kV, total 5 MW), or of deuterium (energy 80 kV, total 10 MW).

Radiofrequency heating: 3 antennae (1986 operation total 8 MW) launching electromagnetic radiation in the JET plasma in the ion cyclotron frequency (ICF) range (25-55 MHz). The ICF heating mode is either

a) at the fundamental of the majority gas

b) at the second harmonic of the majority gas

c) at the fundamental of the minority gas (Minority heating).

Types of JET Operation

Heating	Refueling	Configuration	Gas
OH	Gas puff	outer limiter	Majority: H,D,He
OH+ NB	Pellet	inner wall	Minority: He^3,H (for ICF)
OH+ RF		double null (2 X points)	
OH+ NB + RF		single null (X point)	Trace (for diagnostics) Ar,Kr.

The initial fill of gas is ionised to form the plasma and is replenished by gas puffing (either pre-programmed or controlled by a feedback system). A pellet injector has been used to feed pellets of solid deuteruim (or hydrogen, or gas mixtures) at high speed (1 km/s) into the discharge.

The magnetic configuration can be so arranged that the plasma is maintained in contact with the outer limiters, or moved into contact with the inner wall, or in full aperture cases it may be in contact with both. The shaping coil fields can be so chosen to form a magnetic separatrix (a surface containing a region of closed magnetic surfaces but outside of which there is a region of open magnetic surfaces). In the up-down symmetric case, two stagnation ($\underline{B} = 0$) or X-points are formed, but it is also possible to form a single X-point.

The choice of major-minor gas combinations is one determined by RF heating considerations. Sometimes small amounts of 'trace' gas is puffed into the plasma to assist in specialised studies (e.g radial temperature profile control and energy accounting).

1.5 Aim of this Lecture.

This lecture aims to present a description of JET only to the extent of indicating its place in tokamak research and of making clear how the experiments have been carried out.

It aims to present a restricted number of results of a few major studies performed on JET in the period mid 1983 to end 1986. There is no attempt to make a complete summary. Detailed results are available as JET reports and as papers in the published literature. If this lecture has "given a taste" of the experimental work and achievements of JET, then it will have been successful.

2. Impurities

For a reactor plasma mixture of hydrogen isotopes (e.g. deuterium and tritium), non-hydrogenic material (e.g. helium, carbon, oxygen, metals etc.) are impurities and have a deleterious effect on performance because:

a) the impurities dilute the concentration of reacting ions (reduction of fusion power)

$$n_i/n_e = (Z_i + 1 - Z_{eff})/Z_i$$

where Z_i is the charge number of the principal impurity and Z_{eff} is the effective charge number of all impurities $(= \Sigma n_i Z^2_i / \Sigma n_i Z_i)$

b) the impurities enhance the radiation losses (Bremsstrahlung line and recombination radiation) i.e. they degrade the energy confinement.

c) the impurities <u>may</u> accumulate in the plasma centre

$$Z e n_Z E_r = \frac{\partial p_Z}{\partial r}$$

$$\frac{1}{n_{Z1} Z_1} \frac{\partial p_{Z1}}{\partial r} = \frac{1}{n_{Z2} Z_2} \frac{\partial p_{Z2}}{\partial r}$$

$$\frac{p_Z}{p_H} = \left(\frac{n_Z}{n_H}\right)^Z$$

so that fusion conditions e.g. the attainment of sufficient ion density, become more difficult to achieve.

2.1 Impurity Control Methods

a) Glow discharge cleaning.

b) Pulse discharge cleaning.

c) Bakeout and hot vessel walls.

d) Low Z material for in-vessel elements facing the plasma.

e) Carbonisation.

f) Density screening.

Knowledge of impurity concentration is based mainly on the analysis of resonance line intensities in the visible vacuum ultra violet (VUV) and information from soft X-ray emission.

2.2 Results of Wall Conditioning

The wall conditioning methods tried were 2.1 a) - e) above. The mechanism of impurity removal for a) and b) is the chemical reaction of low Z impurities with atomic hydrogen to form volatile compounds which are then pumped away.

The methods a) to e) are briefly reviewed in the following:

Glow discharge cleaning (GDC) overnight and at weekends. Two electrodes situated in ports in the vessel are used. Gas is continuously introduced at typically 5×10^{-3} mbar and a DC current of about 8 A between the electrodes (anodes) and the vessel (cathode) is maintained. The effects of GDC have not been analysed in detail but are mixed:

i) there is considerable gas (hydrogen or deuterium) absorbed in the wall/limiter surface which leads to strongly enhanced recycling in subsequent tokamak discharges. In some cases this effect has led to difficulty in achieving a sustained breakdown in the first few discharges following GDC (in particular with carbonisation - see below).

ii) there is a strong suspicion that GDC causes the migration of some metal from the wall near the electrodes to the limiters. This metal can be easily released in subsequent high power tokamak discharges.

iii) the density limit achieved in reference discharges after GDC is a measure for the level of impurities which are present. This density limit increases somewhat (i.e. impurity concentration is decreased) following GDC, but the effect is not dramatic and is complicated by i).

Pulse discharge cleaning (PDC): In mid 1984, 12000 hydrogen pulses were performed with:

Toroidal field B = 0.15 T.

Plasma current = 30-40 kA.

Plasma duration = 0.5 s.

Pulse frequency = 3-4 per min.

The influxes of carbon and oxygen in reference discharges were somewhat lower after PDC. The overall effect of PDC (Z_{eff}) essentially unaltered and radiation power loss still about 80% P (ohmic) was not significant in improving the impurity situation and so has not been repeated.

Bakeout and hot vessel walls. After the vessel has been opened (for in-vessel installation etc), it is pumped at high temperature (350^0C) to desorb volatile impurities. Tokamak operation is normally carried out with the double-walled vessel at 300^0C while the single-walled ports are maintained at about 150^0C. No systematic study has been performed on the effects of varying the vessel temperature. Nevertheless a typical base pressure is 10^{-7} mbar hydrogen with 10^{-9} mbar residual impurities (at these vessel conditions).

Low Z materials for in-vessel elements facing the plasma. Original operation of JET was with an inconed (metal) wall surface facing the plasma and 4 discrete graphite limiters. In an effort to reduce metallic (high Z) contamination which arose mostly as a result of disruptive discharges terminating on the inner wall, graphite tiles were installed to cover this wall to a height of 1 m around the mid-plane. Subsequently the amount of installed graphite tiling has been increased to cover most of the vessel elements facing the plasma. Indeed the 4 graphite limiters have now been replaced by toroidal (belt) limiters.

The covering of carbon reduces the amount of metal which reaches the plasma but because of carbon's affinity for hydrogen there is considerable hydrocarbon chemistry involved and the wall acts as a reservoir for hydrogen/deuterium. Special protective tiles have been strategically placed to cope with neutral beam shine-through and X-point localised energy deposition.

Carbonisation or carbidization is the name given to the process whereby the wall is covered by a layer of carbon or carbides. This is achieved by glow discharge cleaning in a mixture of hydrogen and methane (CH_4). After 'light' carbonisation (12% CH_4,6 hr GDC) the metal radiation drops by a factor 5 and the radiated power drops from 70% to 50% of the Ohmic input. The effects of this carbonisation are lost after about 20 tokamak pulses. After heavy carbonisation (17% CH_4, 48 hr GDC), metal radiation drops by a factor 100. The effects of this carbonisation are lost after about 200 pulses. The reduction in impurity level increased the ohmic density limit somewhat (about 10%).

2.3 Density screening

The effective charge number of the plasma is reduced as the average density of plasma increases. The metal impurity influx is more effectively screened at high density, but the oxygen impurity was relatively insensitive to plasma density but more dependent on the state of the vessel. At high plasma density, radiation losses were mainly caused by oxygen. Z_{eff} usually ranges between 2 and 3 for n_e 3 x 10 m^{-3} but approached 1 for a time duration of about 0.5 s after the injection of a deuterium pellet.

At higher density the plasma temperatures are reduced which leads to a lower sputtering yield of metals and carbon from the limiter. The oxygen content is roughly independent of density. No accumulation of impurities has been observed in JET.

2.4 Effects of Additional Heating

The result of RF heating is small but there is a measurable increase in metal impurity which came from the antennae screens. This direct influx from an antennae was observed only when power was applied to the antenna. Indirectly screen material enters the plasma via the limiter.

With neutral beam heating the relative metal concentration is reduced and the absolute metal concentrations decrease progressively during a sequence of stable neutral beam heated discharges.

2.5 Magnetic Limiter

When a magnetic separatrix is formed there is a reduction in carbon content compared with similar limiter discharges. However this leads to only a minor reduction in Zeff. Neutral beam heating leads to an increase in carbon concentrations in such plasmas which is in contrast to the decreased carbon concentration for NB heating in limiter discharges. As in OH high density discharges oxygen was the main impurity in the NB heated plasmas.

3. Density Limit and Disruptions

3.1 Density Limit

As in other tokamaks, the plasma density in JET is limited. For OH discharges (before carbonisation) the limit was

$$n_L(m^{-3}) = 10^{20}\, B(T)/(R(m)q_{cyl})$$

where $q_{cyl} = \frac{A}{2\pi R} \cdot \frac{B}{I}$

A is the cross-sectional area of the plasma and I is the plasma current.

This limit was increased by about 10% after carbonisation.

The density limits are 2 kinds.

a) a low q limit

b) a high density limit

3.2 Major Disruptions

In both cases, attempts to break through these limits result in major disruptions. The disruption is a dramatic event in which plasma confinement is suddenly destroyed, followed by a complete loss of current. Such disruptions not only limit the range of operation but they lead to large mechanical stresses and heat loads on the vessel.

The low q limit is found in terms of q_ψ (defined in terms of magnetic surfaces $\psi(r)$ = constant i.e. $q_\psi = dT/d\psi$ where T is the toroidal magnetic flux bounded by $\psi(r)$ = constant and ψ is the poloidal magnetic flux bounded by $\psi(r)$ = constant). The lower q_ψ limit is q = 2.

The high density limit depends on the plasma purity and on the power input. Large amounts of neutral beam heating push the density to twice the OH density limit. This shows that internal fuelling of the discharge is important because RF power does not have the same effect Removal of the additional heating at high density almost inevitably results in a disruption. One way to avoid this is to control the density decay. In JET one successful way has been to push the plasma to the inner wall where the pumping power of the graphite tiles has been shown to be very effective in reducing the plasma density. This large pumping power of the tiles is not yet understood.

In the high density disruption there are essentially four phases:

i) Pre-precursor phase: in which usually there is a change in the underlying conditions leading towards a more unstable configuration, such as an increase in total current or plasma density.

ii) Precursor Phase: in which, following the underlying change, a critical point for the onset of MHD instability is reached and growth of m = 2 mode magnetic oscillations occurs (timescale of about 10 ms). If sawtooth oscillations (m = 1) are present prior to this stage they are brought to a halt.

iii) Fast Phase: in which, following considerable growth of the MHD instability the central temperature collapses on a rapid timescale (1 ms) and a fast flattening of the radial current profile occurs.

iv) Quench Phase: in which, the plasma current decays to zero with a decay time depending on particular conditions (decays of up to 50 MA/s have been observed in JET).

The high density limit disruption is invariably preceded by a peaking of the temperature (current) profile (a "radiative" collapse). In this disruption the radiated power is roughly equal to the total input power.

The plausible hypothesis for this type of disruption is that, following the parameter changes in the pre-precurser phase leading to a more unstable configuration, tearing mode oscillations grow (particularly m = 2 modes) on the resonant surfaces q = m/n. The magnetic islands produced grow rapidly with the oscillations and when these islands of different helicity have grown to the extent that they overlap, a region of ergodic field is created within the region of overlap. The surfaces are destroyed and plasma energy is lost rapidly by electron thermal conduction along field lines (the fast phase). The consequent central temperature collapse produces a fast flattening of the current profile. The current then decays rapidly to zero (quench phase).

Low q disruptions are not preceded by a pre-precursor phase (radiative collapse and current peaking) and the growth of oscillations in the precursor phase is more rapid than for the high density disruptions. Apparently triggered by sawtooth collapse (temperature flattening) and a relatively 'soft' quench.

There is another type of disruption (skin-effect-induced e.g. during current rise and decay).

3.3 Sawtooth Oscillations (Minor Disruptions)

A feature of almost all discharges are the sawtooth oscillations (or internal disruptions). In OH plasmas the modulation of the central electron temperature is up to 20% and the period is between 30 and 250 ms. With ICF heating or co-injected neutral beams the modulation is increased up to 50% and the period is increased to about 600 ms. With counter-injected neutral beams the modulation and period are reduced. With combined heating, monster sawtooth of long duration (1 s) are sometimes produced. After collapse of the monster, the plasma remains in such a state as to prevent the growth of a second monster, although there may be more normal sawteeth.

4. Energy Confinement

4.1 Global Considerations

The global energy confinement τ_E has been calculated from measured quantities (assuming steady state conditions and always assuming an ion temperature profile and impurity model). For the OH JET discharges a regression analysis has been performed on data, assuming a power law for $_E$ as a function of the "independant" variables $\bar{n}$,B,q,κ(= b/a). The analysis then determines the exponents in a 'best fit' sense: e.g.

$$\tau_E(s) = 0.027\bar{n}(10^{19})^{0\cdot4} q_{cyl}{}^{0\cdot3}\ B(T)^{0\cdot3} R(m)^{1\cdot7} a(m)^{1\cdot3} \kappa^0$$

although this fit for density hides the fact that there is a saturation of τ_E with increasing density. Note that τ_E scales as L^3 where L is a dimension, which is clearly good for large devices.

The τ_E values observed are significantly less than those based on neoclassical (binary collision)transport. This difference (anomalous transport) is believed to be due to a leval of plasma turbulence. Measurements of edge magnetic field and visable light fluctuations have been made on JET but a self-consistant theory linking them with the anomalous transport is lacking.

With additional heating is quickly became apparent that, as on smaller tokamaks, τ_E decreased with increasing applied power, although τ_E improves with increasing I_p and is only weakly dependent on $\bar{n}$.

$$\tau_E(s) = 0.34 \sqrt{\frac{I(MA)}{P(MW)}} \cdot B(T)^{0.25}$$

or $$= (0.29/P(MW) + 0.95) \sqrt{I(MA)} \cdot B(T)^{0.25}$$

There is no difference for H and He^3 the minority heating. $\bar{n}$ scaling is not included as $\bar{n}$ increases with RF heating in a way dependent on the type of RF heating. One difficulty here is to evaluate in P the amount of RF power actually coupled to the plasma.

With the introduction of NB heating on JET it was found that the scaling of τ_E with P_{tot} was independent of the type of additional heating. This scaling agrees pretty well with the so-called L mode scaling (as do, somewhat surprisingly the JET OH results).

A good confinement regime with neutral beam heated plasmas has been observed (the so-called H-mode) as in other tokamaks (ASDEX (FRG), PDX (USA) and Double III (USA)). It occurred in the single X-point configuration and is characterised by higher electron temperature near the separatrix with a steep gradient. The L to H mode transition is signalled by a reduction in edge recycling of atomic hydrogen. In the H mode the τ_E is typically 2 to 3 times greater than the corresponding L mode value, it improves with plasma current, although it appears that degradation with increasing applied power still exists. In the JET H mode the best fusion parameter $\hat{n}_i \tau_E \hat{T}_i = 2 \times 10^{20}\ m^{-3}s$ keV has been obtained.

4.2 Local Energy Transport

A great deal of effort has been expended in attempting to model the local energy transport. This is done mainly with numerical (predictive) numerical codes in which the local transport coefficients (e.g thermal conduction and resistively) are modelled (from the results of interpretive calculations) to see whether agreement in the form of the radial profiles of plasma quantities and the scaling of quantities as others are varied, can be reproduced.

An experimental study of particle fluxes has been undertaken to understand the overall particle balance and recycling. Ultimately these studies should help to elucidate the fundamental processes involved in particle transport and plasma-wall interactions.

One transport parameter for which theory and experiment agree quite well is the parallel electrical resistively $\eta_\parallel$. The Spitzer (classical) value is

$$\eta_\parallel = \frac{m_e^{\frac{1}{2}} e^2}{3 \varepsilon_0^2 \pi^{\frac{3}{2}}} Z_{eff}\, \alpha(Z_{eff}) \frac{\ln \Lambda_{ei}}{Te^{\frac{3}{2}}}$$

The neoclassical (trapped partical effects included) value is

$$\eta_\parallel^* = g\, \eta_\parallel$$

note

$$\alpha(Z_{eff}) = 0.29 + \frac{0.46}{1.08 + Z_{eff}}$$

and

$$1/g = 1 - f_T / ((1 + \xi \nu^*)$$

where f_T, ξ and ν^* are given by S.P. Hirshman, D.J. Sigmar: Nuclear Fusion 21, 1079 (1981)

Near the end of the current flat-top values of plasma resistance are obtained V_ℓ/I_p from $I = \int (E_\phi/\eta_\parallel)ds$ using the measured electron temperature profile. From this resistance (assuming uniform Z_{eff}), values of Z_{eff} are obtained : Z_{spi} and Z_{neo}. These two values are compared with the value Z_{vis} obtained from a measurement of visible Bremsstrahung

$$Z_{vis} = C B_V \left(\int n_e^2 g_{ff} Te^{\frac{1}{2}} dl\right)^{-1}$$

where C is a constant, B_V the measured brightness, the integral is taken along the viewing line and g_{ff} is the Gaunt factor. Z_{spi}/Z_{neo}

values usually lie in the range 1.5 to 2. The neoclassical values are much closer to the Bremsstrahlung-derived values.

A further study of $\eta_{/\!/}$ was carried out by examining the field penetration following a current ramp say from 1 to 2 MA. Equilibrium calculations allowed the flux, toroidal current density and toroidal electric field within the plasma to be calculated. This allowed the resistivity $\eta_{/\!/}$ to be calculated ($= E_\phi/j_\phi$) as a function of space and time. A comparison with $\eta_{/\!/neo}$ and $\eta_{/\!/spi}$ (where Z_{eff} was taken as Z_{vis}) was made, and once again the neoclassical value fitted best.

5. The Future

To achieve the JET objectives (see 1.1) various developments are planned for the future operation:

5.1 Impurity Control

The process of covering the metal surfaces with low Z material continues. For the next period of operation all poloidal bellows will be covered with graphite tiles (in addition to the octant joints) and the limiter area will be increased (from 8 discrete limiters to 2 toroidal limiters). Extra protection will exist for the new neutral beam injection and the X-point protection will be upgraded to handle the full additional heating power. There is the option to replace the limiter graphite with beryllium ones and to install Be evaporators to cover the in-vessel surface with beryllium.

5.2 Density Limites and Disruptions.

To refuel the plasma more efficiently, high-speed multiple pellet injection has been planned and will be upgraded over the next few years. Increased additional heating should also improve the density limit achievable.

Improved plasma performance (temperatures and energy confinement) can possibly be obtained by control of the sawtooth oscillations in the plasma interior and plans for exercising such control exist. The main remedy which is foreseen is an alternative form of plasma current drive (e.g. lower hybrid resonance heating) which may allow the radial profiles of important plasma parameters (e.g. current density and electron temperature) to be adjusted independently. This control will probably require to be of feedback type.

Finally, special discrete limiters with associated pumping (pump limiters) are planned for future installation. This particle exhaust system will assist in controlling plasma terminationss following additional heating and should control the edge density during additional heating.

5.3 Heating Power.

In order to achieve the thermonuclear temperatures required, a certain level of additional heating was included in the original JET design. This level has not yet been reached. In the next period of operation the RF heating system will include 8 antennae (up until now there were 3), and this, together with RF generator improvements, should increase the power radiated by the antennae from 7 to 32 MW.

The neutral beam heating system will be doubled in size with the addition of a second injector box. This will take the deuterium injection power from 10 to 20 MW. Development is underway to increase the energy of these deuterium beams from 80 kV, which would lead to improved beam penetration.

Finally, the ohmic heating performance has been upgraded because plasma currents of 7 MA will be possible from the magnet coils and power supplies point of view. Modifications to the primary coil should improve the plasma build-up (through reduction of stray fields) and allow higher current-carrying X-point plasmas (4 MA for a single X point c.f.3 MA at present). Previous limitations on plasma elongation at high plasma current should now be removable since extra vessel supports have been installed. These limitations were imposed to safeguard the vessel against damage resulting from disruptions or loss of plasma position control. Extra in-vessel protection (see 5.1) should protect the vessel from large energy fluxes.

5.4 Energy Confinement

The most promising plasma confinement with additional heating has been observed in single X-point JET plasmas (H-mode), and so higher performance, X-point (magnetic limiter) target plasmas are desirable. This is one of the aims of an improved poloidal field magnet and power supply configuration which should achieve a 4 MA single X-point plasma (see 5.3). The increased plasma current should also result in improved plasma betas.

If some or all of the above measures are successful in improving the thermonuclear fusion index $\hat{n}_i \tau_E \hat{T}_i$ to values greater than 2×10^{21} $m^{-3}s$ keV (the best value achieved is at present 2×10^{20} $m^{-3}s$ keV), then alpha particle production in deuterium-tritium plasma should be sufficient to allow worthwhile studies of alpha particle heating. These studies will lead to some activation of the machine and will complicate operation because maintenance and repair of systems on the torus will have to be done by remote handling methods. However, the results of these studies are essential to complete the design of the next tokamak experiment in the European Fusion Programme which is required to investigate the more technological problems of a fusion reactor.

6. Acknowledgement

This lecture represents the efforts of a large number of people, that is, the JET Project Team. The contributions of all those involved in this European co-operative venture are gratefully acknowledged.

General on JET. — Fusion Technology, Jan 1987 Vol 11, No 1.

Suggested Reading

1. JET Contributions to 11th International Conference on Plasma Physics and Controlled Nuclear Fusion Research Kyoto, Japan 13-20th Nov. 1986 — JET-P(86)44 JET Preprint but will appear in IAEA Proceedings. Good for latest results.

2. Comparison between Experiment and Theory — R.J. Bickerton et al — JET-P(86)28 Good picture of our "lack" of understanding.

3. Review of JET Diagnostics and Results — P. Stott — JET-P(87)04 Good survey of how plasma parameters are measured

4. Current and Temperature Profile Evolution in JET — ~~[illegible]~~ J. Campbell et all — JET-P(86)46 Results on the η_e vs η (neoclassical) study.

5. Physics models to describe plasma confinement and heating — J.D. Callen — JET-IR(87)07 Series of lectures with ~~many~~ much theory related to JET & tokamaks.