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JET Physics in Support of ITER: Results and Future Work

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JET Physics in Support of ITER: Results and Future Work

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The JET Programme to 1999 concentrates on issues that must be solved before a decision to construct ITER can be taken. The paper discusses three areas representative of the physics support provided: confinement studies, divertor studies and fusion performance. Due to its large size and proximity to reactor conditions, JET provides the most ITER relevant confinement data of all tokamaks. JET has an active divertor programme (Mark I, 1994/95; Mark IIA, 1996; Mark IIGB, 1997/98) aimed at developing a viable divertor concept for ITER, based on, for example, the gas target/radiative divertor concept. High performance ELM-free H- and VH-mode discharges, complemented by quasi-stationary ELMy H-modes in an ITER-like configuration will be developed further in 1994/95 and provide the basis for D-T experiments in 1996 and 1999.

1. INTRODUCTION

The approach to a fusion reactor based on magnetic confinement foresees a series of major facilities such as the Joint European Torus (JET) (start of operation in 1983), the International Thermonuclear Reactor (ITER) (~2005) and a demonstration reactor (DEMO) (~2025). JET is the European representative in the family of "Large Tokamaks" currently in operation and has always had a strong ITER and reactor relevance due to its large size, proximity to reactor plasma conditions and use of reactor relevant technologies (tritium and remote handling).

This paper focusses on the physics support that JET is providing to ITER. The JET Programme is proposed to continue to the end of 1999 and concentrates on those physics issues that must be solved before a decision on ITER construction can be taken. The main objectives of the JET Programme are therefore to develop a viable divertor concept for ITER, to improve plasma performance and to carry out D-T experiments in an ITER-like configuration (see JET Programme Schedule in Fig. 1).

JET is distinguished by its large scale (major radius~3m; plasma volume~80m^3; a plasma current capability up to 6MA), powerful heating and current drive systems (22MW of neutral beam injection; 20MW of ion cyclotron waves; 10MW of lower hybrid waves), efficient fuelling and pumping systems (a centrifuge delivering approximately forty x 2-3mm pellets a second; a divertor cryopump capable of pumping 170m^{3}s^{-1}) and a flexible divertor to control particle and energy exhaust and plasma purity. The divertor configuration will also permit a comparison between carbon fibre composite and beryllium target tiles.

The present paper concentrates on three areas which are representative of the physics support that JET provides for ITER: confinement studies, divertor studies and fusion performance.

2. CONFINEMENT STUDIES

Confinement studies on JET have recently concentrated on changes in transport in the different regimes of confinement often referred to as L-mode (low confinement), H-mode (high confinement) and VH-mode (very high confinement).

First, it is important to define the conditions under which a change in confinement occurs and, since the physics basis for such changes is not yet understood, empirical scalings have been derived from experimental results. For example, the threshold power required to enter the H-mode is available in a multi-device database to which JET has made substantial contributions [1]. Based on these data, dimensionally correct scalings have been constructed. So far, however, it has not been possible to distinguish between two of these scalings (A: $P_{thresh} \sim n_e^{0.75} B_T L^2$ and B: $P_{thresh} \sim n_e B_T L^{2.5}$) which fit the data equally well. This situation must improve since the two scalings give quite different projections to ITER (A$\rightarrow$ $P_{thresh} \approx 100$MW; B$\rightarrow$ $P_{thresh} \approx 200$MW).

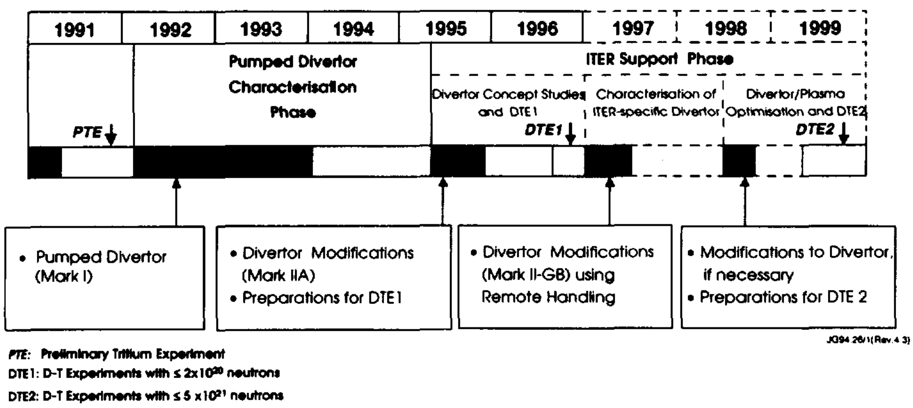


Fig. 1: The JET Programme Schedule to 1999

* See Ref [16]

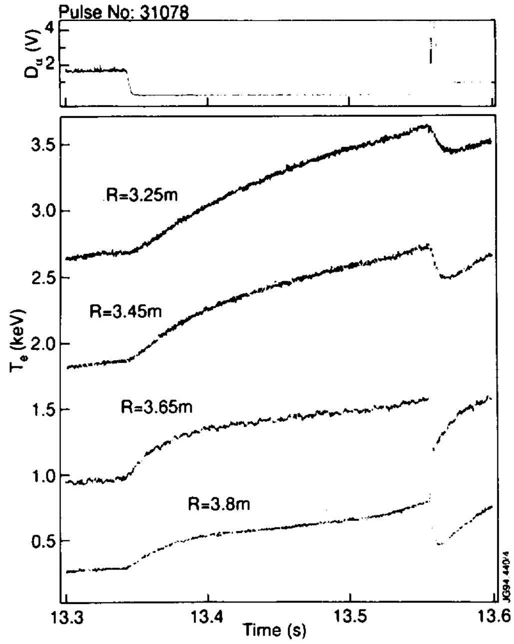


Fig. 2: Evolution of electron temperature (T_e) at the L- to H-mode transition across the plasma radius obtained using a 48-channel ECE heterodyne radiometer

Following the transition from L-mode to H-mode in JET, the evolution of the electron temperature across the whole plasma has been measured using a 48 channel ECE heterodyne radiometer [2]. These results are shown in Fig.2 and confirm the earlier result [3] of a very rapid temperature response over the entire plasma volume. The local transport model used previously to simulate L-mode plasmas [4]:

$$\chi_e = \frac{c^2}{\omega_{pe}^2}\frac{\varepsilon V_{T_e}}{qR} + \alpha_e \frac{|\nabla(nT_e)|}{eBn} aq^2,\ \ \chi_i = \chi_i^{neo} + \alpha_i \frac{|\nabla(nT_e)|}{eBn} aq^2$$

is unable to simulate this rapid response, even when a "barrier" of reduced transport is imposed at the plasma edge. A much more non-linear model would be needed.

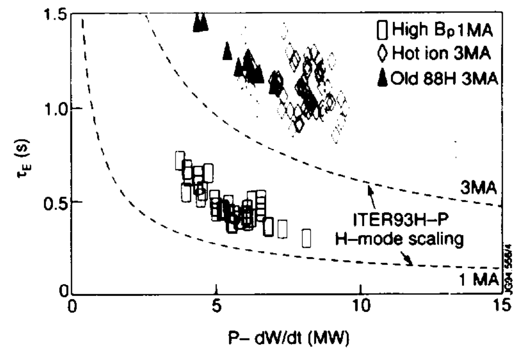
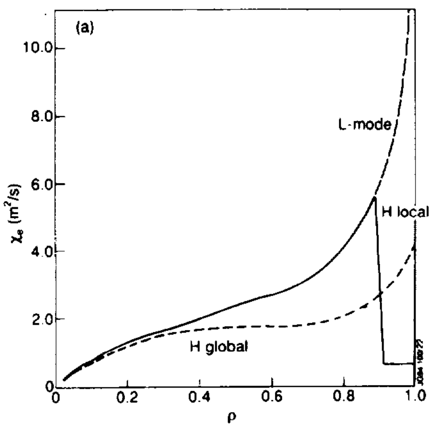


Fig. 4: JET ELM-free H-mode data exhibiting a range of enhancement factors above ITER 93H-P H-mode scaling at both 1MA and 3MA plasma currents

However, by reducing the predominantly L-mode (Bohm-like) contribution to the transport over the entire plasma (the numerical coefficients, α_e and α_i, are reduced from $\alpha_i^L=3\alpha_e^L=9.9\times10^{-4}$ in L-mode to $\alpha_i^H=3\alpha_e^H=5.4\times10^{-5}$ in H-mode [5]), the temperature response can be well-modelled (Fig.3). Furthermore, temperature profiles in both L- and H-mode phases of the discharge are then well described by this "global" model. Pedestals in temperature and density form spontaneously when the following edge boundary condition is used:

$$\chi_{e,i} n \nabla T_{e,i}\big|_{r=a} = \beta_{e,i} D T_{e,i} \nabla n\big|_{r=a} \quad \text{where } D = \frac{\chi_e \chi_i}{\chi_e + \chi_i}$$

The overall confinement of tokamak discharges are often described in terms of a scaling for the energy confinement time as a function of net input power, current, etc. For example, ELM-free H-mode data from JET and other tokamaks have been combined to provide a ITER93H-P H-mode scaling [6]. However, these scalings can be misleading, as shown in Fig. 4 where JET ELM-free H-mode data shows a range of enhancement factors above, for example, the ITER93H-P H-mode scaling at both 1MA and 3MA. Furthermore, there is no observable transition between a "normal" H-mode and the VH-mode with better energy confinement.



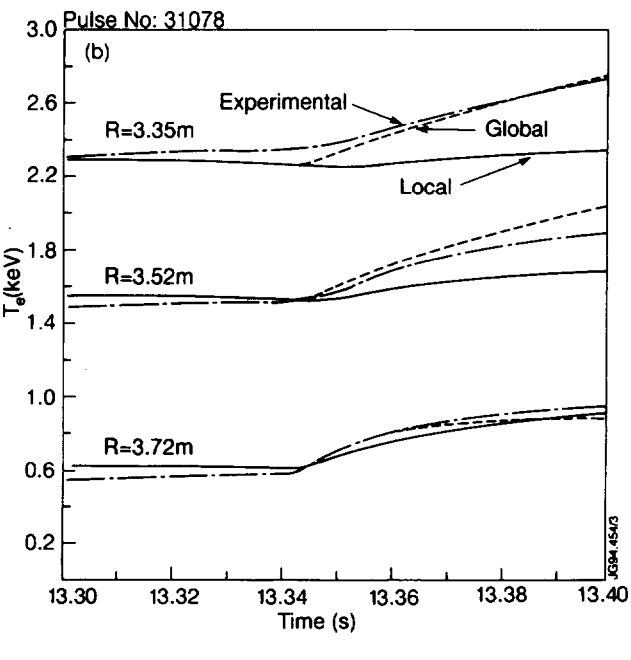


Fig. 3: Using the local and global models for χ_e shown in (a), the modelled temperature response at the L- to H-mode transition is compared with experimental results in (b)

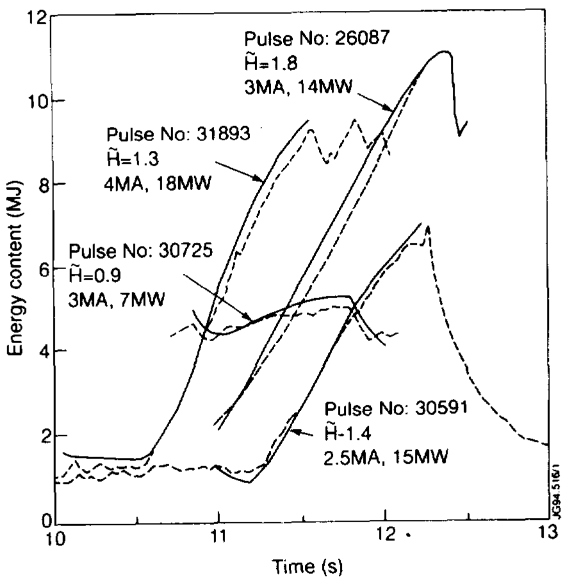


Fig. 5: Modelling (solid lines) of four H-mode discharges with widely different enhancement factors ($\tilde{H}$) over ITER93H-P scaling

It is reasonable to ask whether the transport is indeed different between the H-mode and VH-mode. Figure 5 shows the result of modelling four quite different H-mode discharges with a range of currents up to 4MA and input powers up to 18MW. Even though the four discharges have widely different enhancement factors over the ITER93H-P H-mode scaling, the energy content is well-matched in each case [5] using the same transport model with the same numerical coefficients described earlier. It is to be concluded that these ELM-free H-modes and VH-modes in JET show no difference in transport properties; the difference is rather in impurity radiation, power deposition profiles and recycling.

It should be noted that the above transport model is predominantly Bohm-like in L-mode and might be Gyro-Bohm-like in H-mode. This distinction is strongly supported by global scaling expressions (Goldston [7] and ITER89-P [8] in L-mode; and ITER93H-P in H-mode [6]) and specific experiments on JET, TFTR and DIII-D. Since Bohm-like and Gyro-Bohm-like models extrapolate quite differently towards ITER, further experiments to clarify this issue are planned on JET.

3. DIVERTOR STUDIES

3.1. JET'S DIVERTOR PROGRAMME

The functions of a divertor, as typified by the Mark I pumped divertor presently installed in JET (Fig.6), are to exhaust power at acceptable erosion rates and heat loads on the plasma facing surfaces (divertor target plates), to control the impurity content of the main plasma by removing the source of impurities from the main chamber, by minimising the production of impurities at the target and retaining them in the divertor region, and finally to remove the He ash. JET has an active divertor programme, consisting of Mark I, Mark IIA, and Mark IIGB (see JET Programme Schedule in Fig. 1). The programme is designed to ensure the success of JET's high performance programme, as well as to contribute to the development of the ITER divertor. These JET divertors use a four-element divertor coil system which makes possible a large variety of magnetic equilibria, covering a broad range of target to X-point heights, connection lengths, and magnetic flux expansion. A large capacity cryopump is used, in connection with gas puffing and/or repetitive shallow pellet fuelling, to induce a flow in the scrape-off layer (SOL) to purge impurities and He ash from the main chamber.

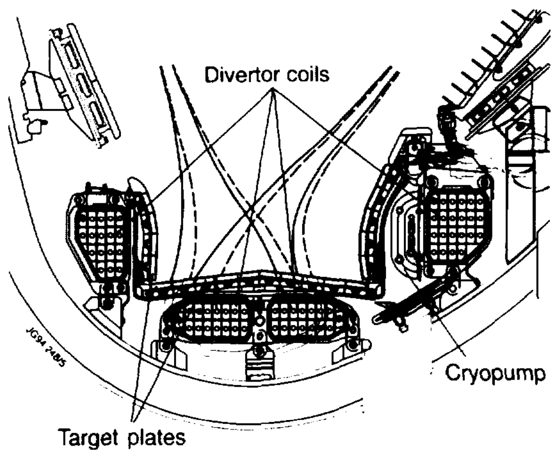


Fig. 6: A poloidal cross section through the JET Mark I pumped divertor, showing two positions of a moderately swept equilibrium

The JET divertor programme will investigate a large variety of divertor concepts relevant to ITER . The Mark I divertor is relatively wide, giving it great flexibility with respect to the variety of plasma equilibria it can accommodate, at the expense of being somewhat open to the escape of recycling neutrals. It will investigate both CFC and Be target tiles.

The Mark II divertors utilize a common "base structure" into which modules of tiles and tile carriers are installed and removed, allowing for relatively quick and inexpensive changes in divertor geometry, carried out fully by remote handling. Mark IIA is a moderate slot divertor which is more closed than Mark I and has improved static power handling capability (to eliminate the need for sweeping) and improved pumping. It allows high power, high current operation on both the bottom domed targets and the vertical sideplates. Following the next JET tritium experiment (DTE1) in late-1996, Mark II will be reconfigured, using remote handling, into the Mark IIGB geometry to provide a specific test of the gas box concept currently favoured by ITER.

3.2. THE "HIGH RECYCLING" DIVERTOR

"Next Step" divertor designs carried out prior to the ITER EDA phase relied upon the "high recycling" divertor concept to fulfil the functions described above. In this concept, a very dense, cool (~10eV) plasma develops adjacent to the targets by operating with a closed divertor, at moderately high SOL densities. In a narrow "high recycling" zone close to the target, the plasma temperature drops to the point that target sputtering becomes insignificant, reducing erosion and impurity production. The heat load is also somewhat reduced because re-ionisation of the recycling neutrals is accompanied by substantial amounts of hydrogen radiation, which is emitted over 4π steradians. Up to 40% of the power flow into the divertor can be re-directed in this way. In general, however, this is not sufficient to reduce the heat flux onto the targets to $\leq 5MW/m^2$, as required. For that reason, other methods of reducing the heat load to the targets are being developed. These

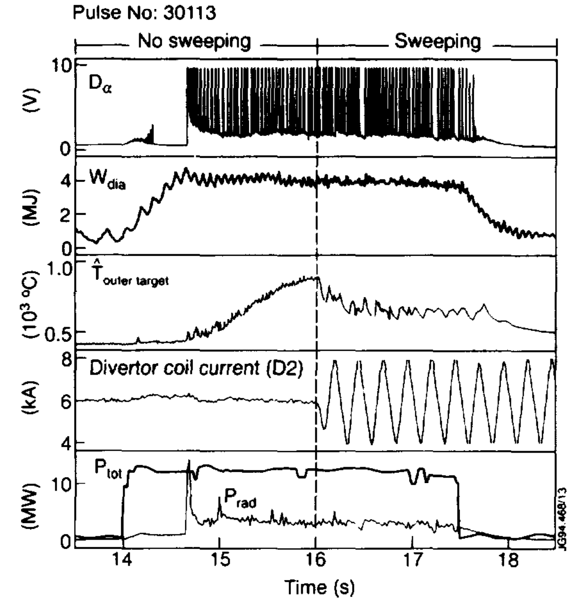


Fig. 7: Time traces for a quasi-steady H-mode pulse. Sweeping was initiated at 16 seconds, resulting in a reduction of the temperature of the outer target plates ($\hat{T}$). There was no effect on the ELMs (D_α), the stored energy (W_{dia}), or the radiated power (P_{rad})

include sweeping of the divertor plasma back and forth across the target plates, and creating a "detached" divertor plasma, where all power flowing into the divertor is dissipated by radiation and charge exchange before reaching the target plates (so-called gas target divertor).

3.3. SWEEPING

Sweeping has been employed very successfully in the JET Mark I divertor [9], where the plasma strike points are swept with an amplitude of up to 10cm at 4Hz by imposing an oscillating component on the divertor coil currents. Figure 7 shows an example where the sweeping was initiated about halfway through the neutral beam heating phase. It can be seen that this had no effect on the plasma stored energy, the ELM signal (D_α), or the radiated power, but that the temperature of the tiles, which had been rising continuously prior to initiation of the sweeping, was reduced to a steady level of about 700°C, well within the limits allowed. Using this technique, up to 140MJ of energy have been exhausted without evidence of tile

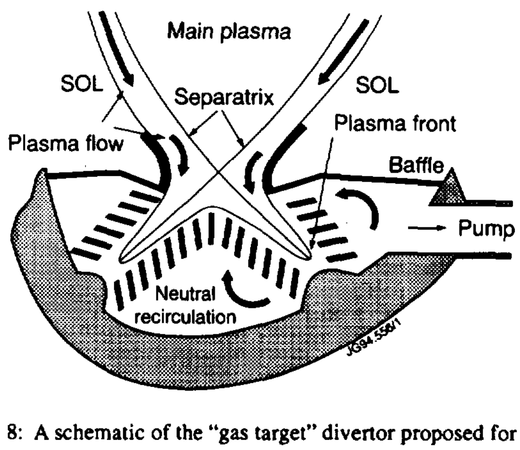


Fig. 8: A schematic of the "gas target" divertor proposed for ITER

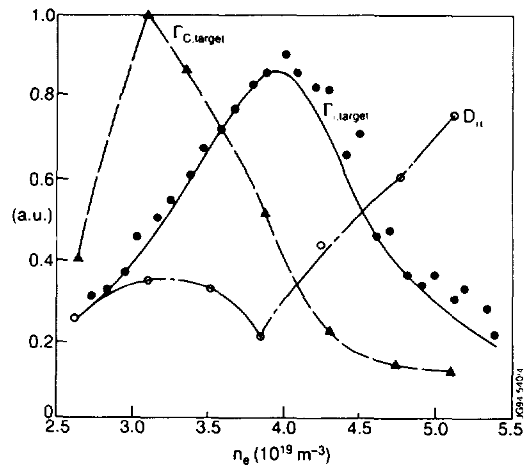


Fig. 9: Hydrogen flux to the target (Γ_i), carbon flux from the target (Γ_C), and D_α measured at the target, as a function of the main plasma density for an ohmic pulse, illustrating detachment as the density increases

overheating. This is a great improvement on the 1991-92 JET campaign, where carbon "blooms" typically limited exhaust energies to 15MJ or less.

Although sweeping has worked successfully in JET, there are doubts about its viability for ITER. The main concern is that lateral sweeping of the X-point is not compatible with the concept of a tightly "closed" divertor. This is required to reduce main chamber recycling to a tolerable level in the presence of high divertor neutral pressures associated with the gas target divertor concept. In addition, there are technical concerns about the difficulty of sweeping with superconducting coils.

3.4. THE "GAS TARGET" DIVERTOR: ITER'S PREFERRED SOLUTION

Figure 8 shows a sketch illustrating the principles of the "gas target" divertor [10]. The baffle at the top is positioned and dimensioned to reduce the neutral pressure in the main chamber several orders of magnitude below that in the divertor, which may be a few tens of millitorr. The neutrals interact with the plasma to remove energy, by radiation and charge-exchange (CX), and momentum, reducing the plasma pressure near the target ("detachment"). Calculations suggest that a recycling impurity such as neon will need to be added to the divertor plasma in small amounts (≅ 1%) to radiate sufficient energy. The structure labelled "transparent wall" is intended to redirect hydrogen and impurity neutrals from the lower part of the divertor to the upper, while removing the energy of energetic CX neutrals before they re-enter the plasma. The divertor chamber is pumped to induce a SOL flow for the removal of He ash and intrinsic main chamber impurities. Several candidate geometries for detached, highly radiating divertors have been suggested, and will be tested at JET in the Mark IIA and IIGB series.

Detachment of the plasma from the target plates tends to occur in most divertor geometries, at a given exhaust power level, when the main plasma density is raised sufficiently. The role of a closed divertor is to promote detachment at moderate densities, where compatibility with H-mode operation can be demonstrated.

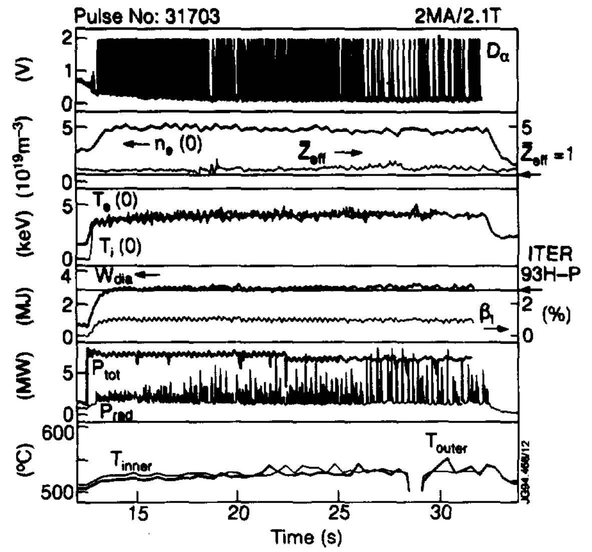


Fig.10: Time traces for an ELMy H-mode pulse. Steady conditions were maintained for 20s (more than forty energy confinement times)

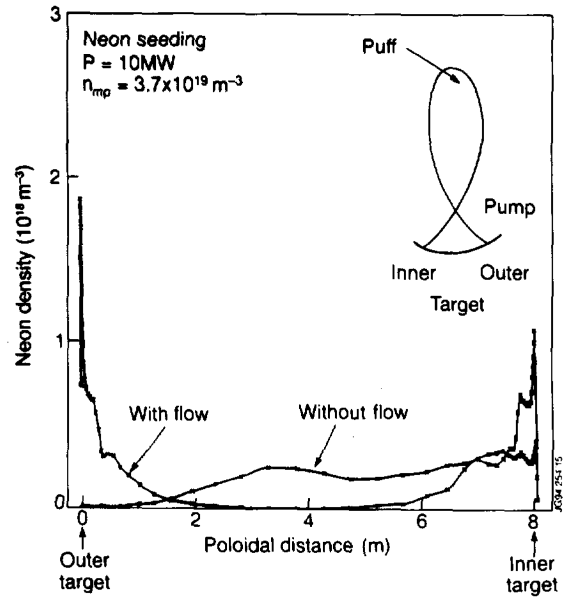


Fig.11: Neon density versus poloidal distance along the separatrix, from EDGE2D simulations of the JET Mark IIGB divertor with and without SOL flow induced by puffing in the main chamber and pumping in the divertor

In JET, detachment and the concomitant reduction of heat flux to the target plates was reported during the 1991-92 campaign [11]. With Mark I, these studies have been extended over a broader range of plasma conditions and equilibrium configurations. Figure 9 shows results from an Ohmic pulse where the density was ramped from 3 x 10^{19}m^{-3} to 6 x 10^{19}m^{-3} over a period of several seconds. As the density increased the particle flux density (Γ_i) to the target first increased (the high recycling regime), and then decreased dramatically as detachment set in. The carbon influx (Γ_c) from the target begins to decrease during the high recycling phase as the divertor plasma temperature falls below that required for significant sputtering, and then plunges further as the temperature and the particle flux decrease during the detached phase. Simultaneously, the recycling light increases as the number of excitations per ionisation increases rapidly at very low temperature.

The pattern of divertor radiation also changes as detachment proceeds. It varies from being localised near the target plates to being primarily in the region of the X-point, depending on the degree of detachment. The precise location of the radiating region and its control is of great importance, for example in connection with determining the H-mode power threshold, and is being studied in detail [12].

3.5. PARTICLE CONTROL AND STEADY-STATE H-MODES

In addition to handling the power exhaust, it is essential to maintain good particle control during a discharge. In the JET Mark I pumped divertor, the cryopump is observed to remove particles at a rate corresponding to several times the beam fuelling rate, facilitating density control [9]. This improved density control has made it possible to produce ELMy H-mode plasmas which are essentially steady state, as shown in Fig.10. In the pulse shown, the density, Z_{eff}, stored energy, recycling signal, plasma temperatures, β, and radiated power all remained constant, while the temperature of the divertor tiles, which are not actively cooled, increased at a very slow rate, remaining very far below their design limits. The length of the pulse corresponds to about forty times the energy confinement time, which was equal to that given by ITER93H-P H-mode scaling. The safety factor, q_{95}, was 3.3, giving an H/q value of 0.6. The demonstration of steady-state H-modes of this quality is a significant step along the road to successful operation of ITER.

3.6. SIMULATION OF THE SOL/DIVERTOR

JET has developed a robust, two dimensional edge code, EDGE2D/NIMBUS, which treats hydrogen and impurity species in the fluid approximation, and the neutrals by a coupled Monte Carlo description [13]. This code is now used routinely for interpretation of edge data as well as for design-related prediction. An example of the latter is given in Fig. 11, which shows some results from an EDGE2D simulation of the JET Mark IIGB divertor [14]. This simulation used the full multi-species version of the code with Neon as the injected impurity. The figure shows that without pumping the Neon resides primarily in the SOL outside the divertor. With divertor pumping and puffing in the main chamber, the resultant SOL flow entrains the Neon in the divertor, and a majority of the radiation shifts to the divertor region, fairly equally balanced between inner and outer legs.

4. FUSION PERFORMANCE IN JET

4.1. HIGH PERFORMANCE OPERATION

The main high performance regimes currently pursued at JET are: (a) quasi-stationary, ELMy H-modes and (b) transient, hot-ion H- and VH-modes. So far, operation in the new pumped divertor configuration has been with currents up to 4MA, but will soon be extended to its full current capability of 6MA. In general, JET H-modes in 1994 are more ELMy than previously. As a result, steady-state conditions are more readily achieved

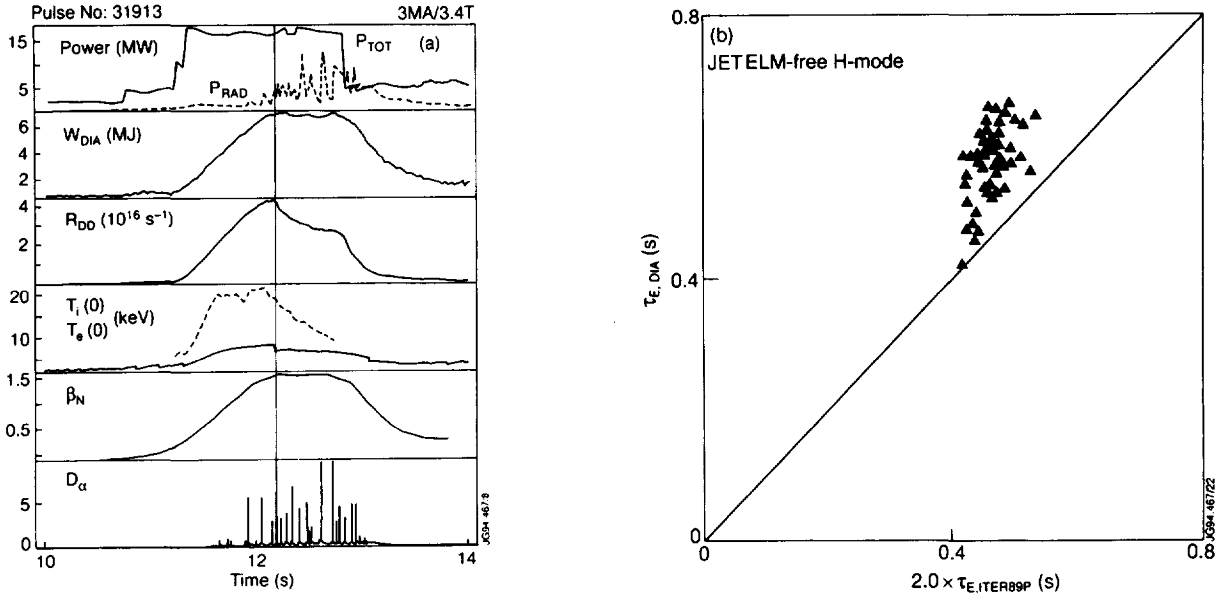


Fig.12:(a) A hot-ion H-mode (at 3MA) which reached a value for the triple fusion product of 5.6x10^{20}m^{-3}keVs; (b) the experimental energy confinement time ($\tau_{E,DIA}$) as a function of the calculated value of 2.0xτ_E(ITER89-P) for a number of JET hot-ion ELM-free H-modes

(see Fig.10) but performance so far is limited to below that achieved previously in 1991/92. Transiently, performance as measured by the fusion triple product $n_D(0)T_i\tau_E$ has reached 5.6x10^{20}m^{-3}keV in 3MA hot-ion H-modes, such as shown in Fig.12(a) [15]. It should be noted that, as shown in Fig.12(b), energy confinement during the ELM-free phase of similar discharges is well above normal H-mode expectations.

So far, recycling in the new JET configuration is higher than in 1991/92 and the performance is correspondingly lower. Only when the full cryopump is now used is recycling reduced to that of the earlier period of high performance. Special attention is being paid therefore to identifying and reducing the source of higher recycling which could be the cold divertor targets or larger surfaces of bare metal in the new JET configuration. In particular, the divertor targets and vacuum vessel have been baked to high temperatures (200°C and 320°C, respectively) and the cryopump used to deplete neutral reservoirs. The "plugging" of the divertor with plasma (by high flux expansion) to reduce the leakage path of neutrals into the main plasma has also been beneficial. This is seen in Pulse No:32867 (Fig.13), which features a clear ELM-free period and a distinct improvement in plasma performance when compared with Pulse No:31913 (Fig.12(a)). As an additional option the use of carbonisation, boronisation or lithium pellet injection is being assessed.

Figure 14 summarises performance achieved so far in 1994 in terms of the fusion triple product and central ion temperature [16]. Comparison is made with the best results from the 1991/92 campaign. At present, 1994 performance is about a factor of 1.5 below the best achieved in 1991/92 and approximately equivalent to that achieved in the two discharges of the Preliminary Tritium Experiment (see Section 4.3).

4.2. THE CONFINEMENT OF ENERGETIC IONS

Even when high fusion performance is achieved, α-particle heating in deuterium-tritium (D-T) plasmas will only be effective if the energetic α-particles are thermalised within the plasma volume. Given the importance of confining energetic particles, significant experiments have already been carried out to simulate in JET deuterium discharges some of the α-particle physics expected in D-T plasmas.

The thermalisation of tritons produced by the fusion process in deuterium discharges allowed the classical behaviour of individual energetic particles to be demonstrated [17], while a minority population of ions heated at the ion cyclotron resonance frequency (ICRF) to energies of a few MeV produced no deleterious collective effects for energetic particle pressures (β)

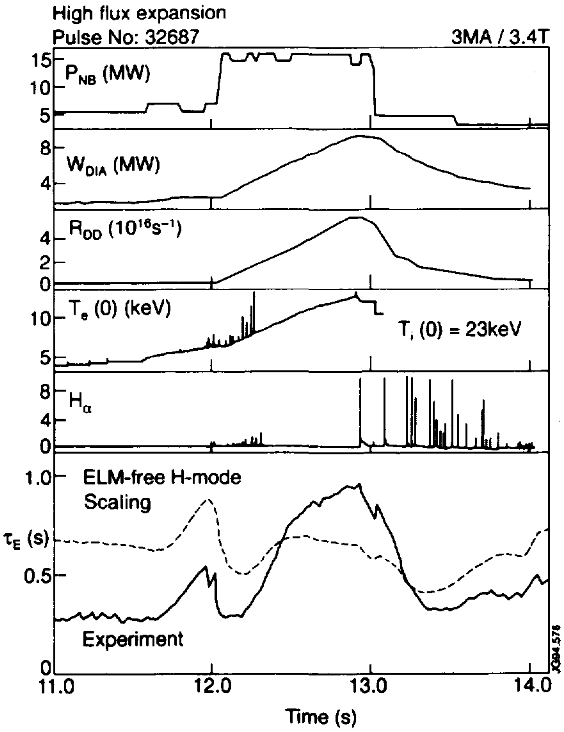


Fig.13: Characteristics of Pulse No:32687 showing the improved performance. In particular, the reaction rate (R_{DD}) reaches 6x10^{16}s^{-1}.

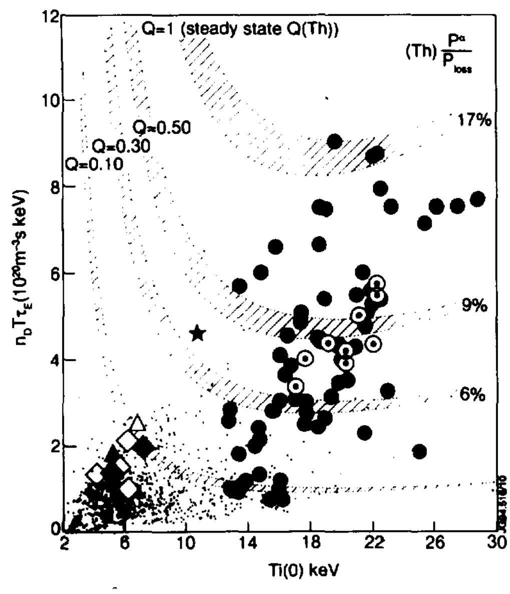


Fig.14: The fusion triple product of $n_D(0)T_i(0)\tau_E$ versus $T_i(0)$, obtained in 1991/92 (●) compared with 1994 quasi-stationary ELMy H-modes (at 2.5MA(♦), 3MA(◇), 3.5MA(▲), and 4MA(△)) and transient H- and VH-modes (at 2.5MA/3MA(o) and 4MA(★))

up to the investigated value of $\beta_{energetic} \approx 8\%$ [18]. Furthermore, by operating JET with only 16 toroidal magnetic field (TF) coils energised (rather than the standard 32 TF coils), the effect of increased TF ripple on energetic particles was investigated [19], showing a one to two order of magnitude reduction in the flux of MeV hydrogen ions. Furthermore, ELM-free H-mode operation (thought to be necessary for ITER) was prevented when only 16 TF coils were energised. This essential new design input to ITER will be augmented further by an improved TF ripple experiment when a continuous variation between 16 and 32 TF coils will be possible early in 1995.

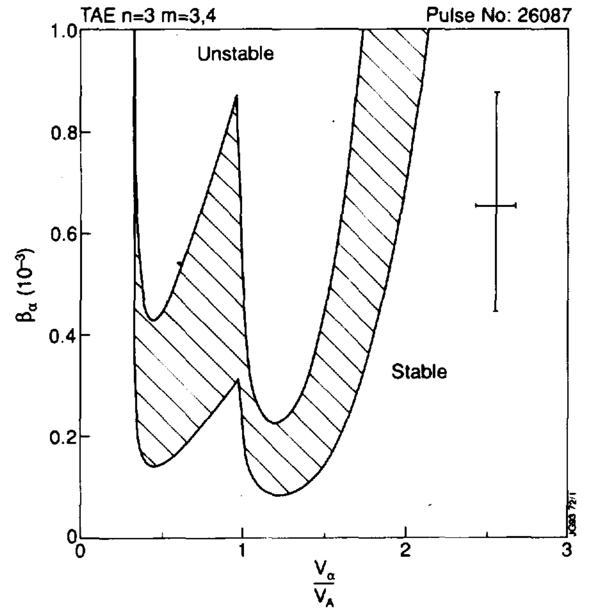


Fig.15: Stability boundaries for the most dangerous TAE modes (n=3, m=3, 4) for JET discharge Pulse No:26087 during the PTE series

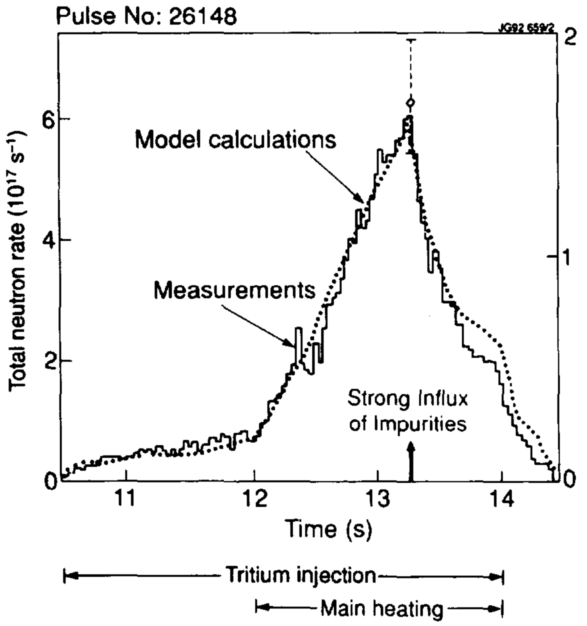


Fig.16: The total neutron rate and the fusion power as a function of time for the Preliminary Tritium Experiment (PTE)

Theory predicts that energetic α-particles can lead to the resonant excitation of global Alfvén Eigenmodes (TAE) which can, in turn, induce α-particle losses. These losses can lead to unacceptably high heat loads on the first wall of fusion devices. Figure 15 shows that a high performance deuterium discharge in JET when operated with equal concentrations of deuterium and tritium would be stable to the most dangerous TAE modes (n=3, m=3,4) [20]. However, TAEs could be excited spontaneously in future D-T experiments on JET if operated at a lower density, since β_α would increase, and v_α/v_A would decrease. There are two loss channels. Prompt α-particle losses scale linearly with the amplitude of the applied TAE perturbation, which, for the case above, leads to 1.5% α-particle losses. Stochastic α-particle diffusion losses on the other hand set in above a threshold field perturbation which is fairly high. For the case above, the threshold is about $\delta B_r/B \approx 1 \times 10^{-3}$ at the position of the gap for the considered case of the n=3 TAE mode. In that case, $\delta B_r/B \approx 3 \times 10^{-3}$ would lead to an additional 1% α-particle loss [21]. However, in the perhaps more realistic case that a large number of TAE modes is unstable with n in the range between 3 and 10, the stochastic regions associated with each mode may overlap leading to significantly higher losses.

In the meantime, the stability of TAEs and their interaction with energetic ions is being investigated in active excitation experiments using internal saddle coils and fast magnetic coils to excite and detect TAEs. First results have been reported [22].

4.3. DEUTERIUM-TRITIUM OPERATION

JET started its programme of deuterium-tritium (D-T) operation in 1991, when performance had reached the level which warranted the use of tritium for the first time in a laboratory plasma experiment. In this Preliminary Tritium Experiment (PTE), the activation of the machine structure had to be strictly controlled so that only two PTE discharges were planned and executed with a tritium content limited to 10% (rather than the optimum ~50%). As seen in Fig.16, each of two practically identical PTE discharges produced peak fusion power of 1.7MW averaging 1MW over a two second period [23]. The duration of the high performance phase of this and similar high performance discharges was abruptly terminated by MHD effects followed by an unacceptably high influx of carbon impurities (the so-called carbon "bloom") originating from divertor targets.

D-T operation on JET is next scheduled for the end of 1996 with the DTE1 series of experiments (see Fig. 1), which will have the physics objectives:

- to demonstrate long pulse fusion power production ($Q \approx 1$ with more than 10MW lasting for several energy confinement times) and to study α-particle effects; and
- to make important contributions to D-T physics in an ITER relevant divertor configuration (H-mode threshold, ELM and confinement behaviour, and RF heating studies).

The technical objectives are:

- to demonstrate the ability to process tritium (using the Active Gas Handling System) while supporting a reacting tokamak plasma; and
- to demonstrate a reactor-relevant remote handling operation.

The fusion performance expected during DTE1 has been assessed on the basis of high performance deuterium Pulse No. 26087 obtained during the PTE series of discharges. Assuming that the carbon "bloom" can be avoided (as with the Mark I divertor), TRANSP calculations show that a 50:50 D-T plasma would produce 14MW of fusion power and $Q \approx 1$ ($Q_{thermal} \approx 0.67$). α-particle heating would then lead to a clear increase in the central electron temperature (from 13keV to 16.5keV).

The neutron budget for DTE1 will be limited to 2×10^{20} D-T neutrons (TFTR has produced 1.2×10^{20} neutrons so far) corresponding to 0.6MJ of fusion energy. The programme would then continue with the remote handling exchange of the divertor target structure. If manual intervention became necessary, the experimental programme could still continue after a delay of at most about one year.

More extensive D-T experiments (DTE2) are then planned to follow in 1999.

5. CONCLUSIONS

Following a shutdown that lasted nearly two years, JET with its new pumped divertor configuration is again contributing to ITER and a fusion reactor. The present experimental programme is already addressing the most critical scientific issues (such as confinement and divertor physics and high performance operation) that must be solved before ITER construction can proceed. The proposed JET Programme to the end of 1999 will complement this work. This should ensure that Europe maintains its present lead in fusion research.

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