JET–P(97)55

JET D–T Experiments and their Implications for ITER

M Keilhacker.

JET Joint Undertaking, Abingdon, Oxfordshire, OX14 3EA,

Preprint of a Paper to be submitted for publication in the
proceedings of the 17th SOFE Conference
(San Diego, USA, 6-10th October 1997)

December 1997

JET D-T Experiments and their Implications for ITER

The JET Team (presented by M Keilhacker)

JET Joint Undertaking, Abingdon, Oxon, OX14 3EA, UK

Abstract - This paper reports the first Deuterium-Tritium (D-T) fusion experiments in the geometry of the International Thermonuclear Experimental Reactor (ITER), with long pulse length and an ITER-like divertor. It first discusses the technical preparations for these D-T experiments and the issues of loading the vessel walls with tritium and of tritium clean-up. It then discusses important physics results of ELMy H-modes, the standard ITER mode of operation: three ITER reference ICRF heating schemes, second harmonic tritium ($2\omega_{CT}$) with and without additional He^3 and fundamental minority deuterium in a tritium plasma (ω_{CD}), have been successfully tested; the present ITER scaling for the L-H threshold power needs to be modified to include a favourable mass dependence (~1/A); and the $A^{0.41}$ mass dependence in the ITER scaling for the energy confinement time has to be removed. Finally, world records in fusion performance are reported: a record fusion energy (14MJ) in standard ITER ELMy H-mode and records of fusion power (13MW) and Q(0.6) in hot ion ELM-free H-mode, which also show clear signs of α-particle heating. In optimised shear mode, strong Internal Transport Barriers were established and 8.2MW of fusion power was produced. So far, no α-particle driven Alfvénic instabilities have been observed in these high fusion power discharges.

1. INTRODUCTION

After the closure of TFTR (in Princeton, USA) earlier this year, JET is now the only tokamak experiment world wide able to operate with deuterium-tritium fuel (D-T) mixtures. Furthermore, JET is the experiment closest in scale, geometry (plasma cross-section, bottom single null divertor) and operating conditions (ELMy H-mode) to the International Thermonuclear Experiment Reactor (ITER). D-T experiments in this ITER-like configuration are therefore one of the two main physics objectives of the JET Programme for the period to the end of 1999, the other being the development of a viable divertor concept for ITER. Two periods of D-T experiments are planned, DTE1 in 1997 and a more extensive DTE2 in 1999, or possibly later if the JET Programme continues beyond 1999. At present, JET is in the middle of its DTE1 series of D-T experiments. This paper summarises the most important results obtained so far and discusses their implications for ITER. The technical developments for D-T operation are not considered in detail but are reported in a companion paper at this conference [1].

The objectives of DTE1 can be divided into two areas, ITER modes of operation and high fusion performance. The first area comprises the isotope dependence of confinement and H-mode threshold power, ICRF heating methods in reactor-relevant D-T plasmas and high fusion power and Q in steady-state ELMy H-modes. The second area has the major objectives of producing maximum fusion power and Q in hot-ion ELM-free H-mode and optimised shear mode of operation, of observing α-particle heating and of studying the stability of Alfvén Eigenmodes.

The extent of the DTE1 experiments is constrained by the total number of D-T neutrons produced which has been limited to 2×10^{20} so that the subsequent activation of the JET vessel will not then prevent manned in-vessel intervention for more than a year (the activation decays with an e-folding time of 71 days).

The paper is structured as follows. Section 2 discusses the technical preparations for DTE1, including the issues of loading the vessel walls with tritium and of tritium clean-up. In Section 3, D-T results in ELMy H-mode, which is the standard mode of operation for ITER, are considered, with particular emphasis on ICRF heating in D-T, the L-H threshold power, the edge operational space, and energy confinement. Fusion power performance in the ELMy H-mode, the optimised shear regime and the hot-ion H-mode is considered in Section 4 which also deals with α-particle heating and Alfvén Eigenmode studies. The conclusions follow in Section 5.

2. TECHNICAL PREPARATIONS FOR DTE1

JET was designed from the outset for D-T operation and first experience in handling tritium was gained during the Preliminary Tritium Experiment (PTE) in 1991 [2]. The technical preparation for DTE1 consisted therefore of bringing existing systems to full tritium readiness and of commissioning the Active Gas Handling System (AGHS). All JET systems are now fully operational for D-T operation, as also is the AGHS which is a closed-cycle gas supply, retrieval and processing plant which pumps the torus in continuous operation and supplies deuterium and tritium to the torus via the gas introduction and NB systems. During DTE1, only one of the two NB injectors is operated with tritium, injecting up to 11.3MW of tritium for up to 5s. In addition, the tritium gas fuelling system has allowed >90% tritium plasmas. For DTE1 there is 20g tritium on site which is stored in U-beds and reprocessed in the AGHS to a purity of 99.4% by gas chromatography. By now, already 40g of tritium has been re-processed and is being re-used.

It has been found that tritium concentrations in the plasma can be relatively easily controlled by loading the vessel walls

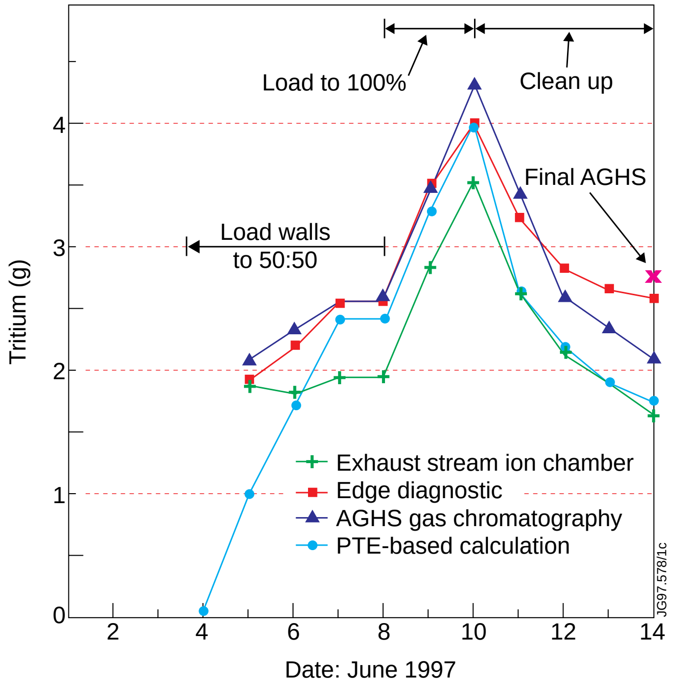

Fig. 1: Tritium inventory in the torus during the tritium wall loading and clean-up phases in June 1997.

with tritium in ohmic or low power (7MW) RF heated discharges. In fact, loading the walls using 100% tritium discharges allows >90% tritium concentrations in the plasma, while 50% tritium discharges result in 50:50 D-T concentrations. Furthermore, it has been found that the tritium concentrations in the core plasma, edge plasma and sub-divertor volume are very similar and are well simulated using a multi-reservoir model [3] which was developed on the basis of the PTE.

With regard to the clean-up of tritium from the torus following D-T operation in early June 1997, the measured tritium inventory in the torus was found to follow the expectations from the PTE (Fig. 1). The final AGHS figures indicated that after four days of clean-up, ≈2.7g tritium remained in the torus walls. More recently, however, the tritium torus inventory has been found to be larger than predicted by this model. It is believed that tritium trapped in films and flakes on cooled surfaces of the vessel has led to an additional inventory. This is important information for predicting the accumulation of tritium in the walls of ITER and will be assessed more quantitatively when the JET torus will be vented during the shutdown early in 1998 for the installation of the Mark II Gas Box divertor by remote handling.

3. ITER RELEVANT ELMY H-MODE RESULTS

The ELMy H-mode is the standard ITER mode of operation and a number of its key physics issues have been addressed during DTE1. In particular, the isotopic scaling of L-H threshold power and energy confinement time of “steady-state” ELMy H-mode discharges with the exact ITER values of normalised plasma pressure, β_n, and collisionality, ν, have been studied in D-T and in pure tritium. Following DTE1, similar experiments will be performed in hydrogen to complement these data.

3.1. ICRF heating of D-T plasmas

Three ITER reference ICRF heating schemes have been successfully tested in D-T plasmas: ω_{CD} with D minority in T plasmas, $2\omega_{CT}$ in 50:50 D:T and $2\omega_{CT}$ in 50:50 D:T with an additional 5 to 10% of He^3. All three schemes have demonstrated efficient plasma heating, with (D)T and (He^3) DT mainly heating

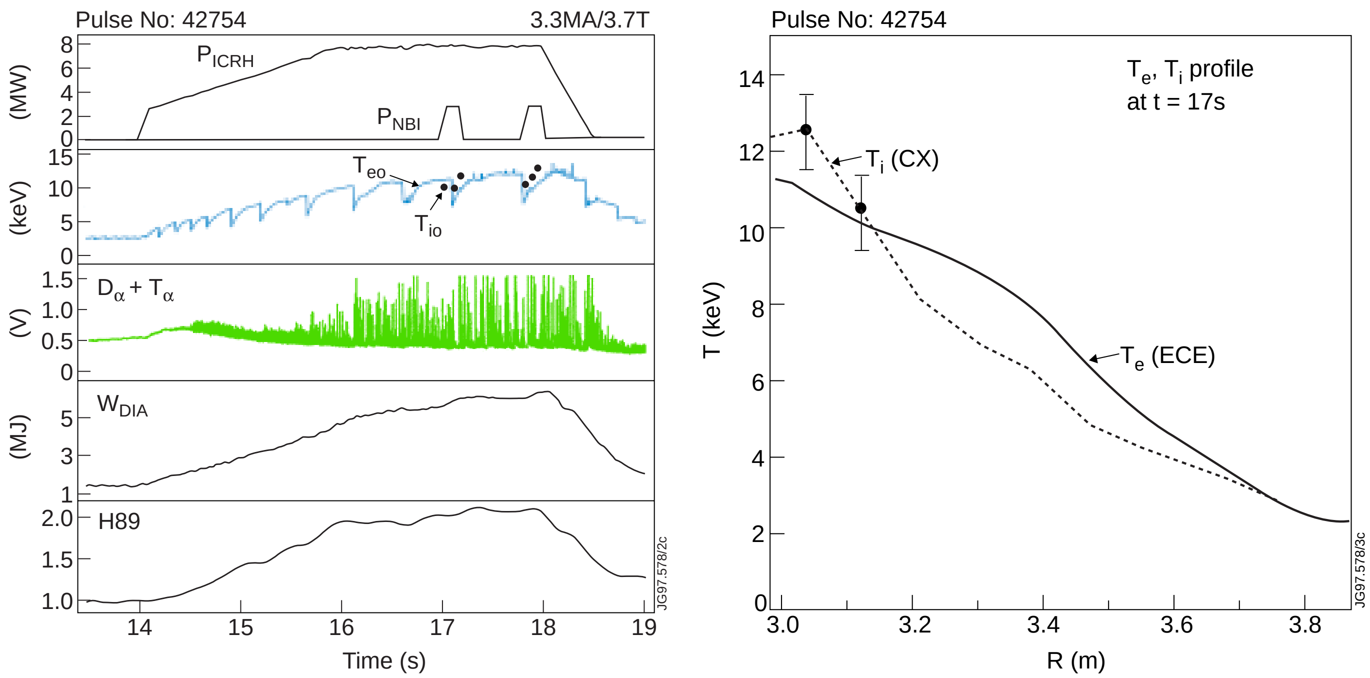

Fig. 2: Strong ion heating with ICRH at the second harmonic of tritium with an additional 5% He^3.

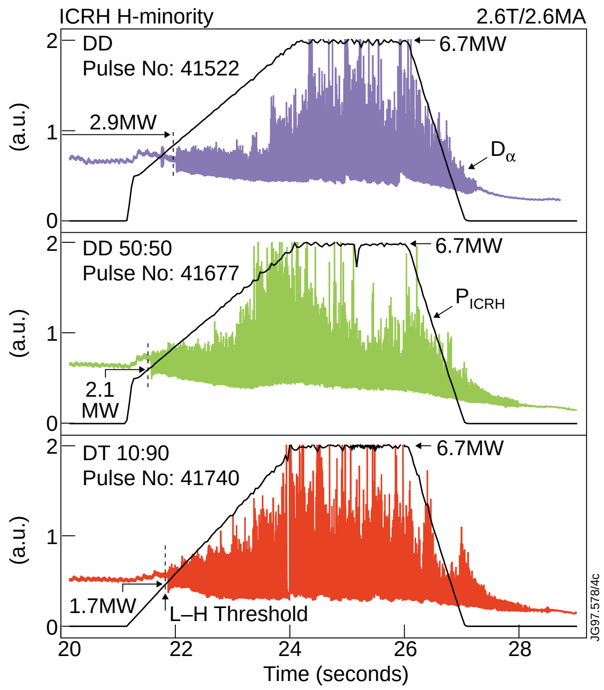

Fig. 3: For the same ICRH power waveform, the L-H threshold power decreases with increasing tritium concentration.

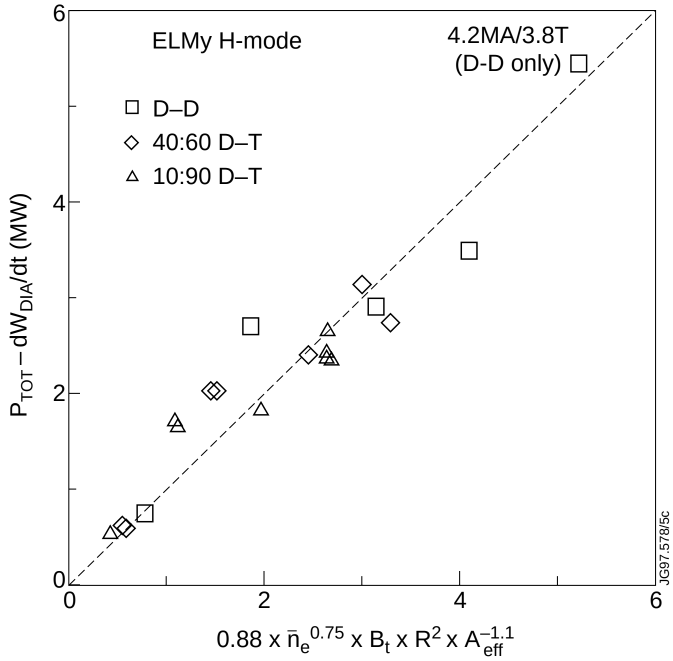

Fig. 4: JET L-H threshold power data for ELMy H-mode compared to the ITER scaling [5] modified by a mass dependence $A^{-1.1}$.

the ions, while the $2\omega_{CT}$ scheme with no He^3 preferentially heats the electrons. Figure 2 shows an example of ICRF heating (37MHz) at the second harmonic of tritium with 6.5% of He^3 added ($2\omega_{CT}=\omega_{CHe3}$). In this case, the RF power couples mainly to the He^3 minority ions which heat the plasma efficiently due to the formation of lower energy tails and higher E_{crit}, the energy at which fast ions slowing down in the plasma give energy at equal rates to electrons and to plasma ions. As the figure shows, a good ELMy H-mode is formed at 14.5s (P_{thr}=4 MW) and the central ion temperature reaches 13keV (the two blips of NBI are for the CXS ion temperature measurements), slightly above the electron temperature.

The minority deuterium (ω_{CD}) heating scheme has also been very successful. In a discharge at 3.3MA/3.7T with a species mix of about 15% D and 85% T, a fusion power of 0.85MW has been produced for an input RF power (28MHz) of only 4MW(Q=0.2 for 3s). The fusion yield is mainly due to the suprathermal deuterons (~90keV) reacting with the majority tritium ions. The plasma density is high and the deuterium energy is near E_{crit} and near the peak fusion cross-section. The results are in excellent agreement with the PION code [4] which gives confidence in predicting ICRF heating for ITER.

3.2. H-mode threshold power in D-T plasmas

The H-mode threshold power was determined by observing the onset of ELMs in the D_α/T_α light emission as the ICRF power was ramped up over a 3s period. As can be seen in Fig. 3, the L-H threshold power clearly decreases with increasing tritium concentration. The same behaviour is found for both ICRF and NB heated discharges. The JET L-H threshold power database, which has been extended to 4.2MA/3.8T in deuterium and includes discharges with 60% and 90% tritium concentrations, shows that the scaling presently used for ITER ($P_{thr}=0.45n_{e20}^{0.75}B_TR^2$[5]) needs to be modified to include a mass dependence approximately proportional to $A_{eff}^{-1.1}$ (Fig. 4). This implies a reduction of the H-mode threshold power for ITER by more than 25% in D-T and by more than 50% in a pure tritium plasma.These predicted reductions in threshold power offer improved operational flexibility for ITER.

3.3. Edge operational space for ELMy H-modes

The edge parameters of electron density and temperature (n_e,T_e) in steady-state ELMy H-modes show a cyclic behaviour during Type I ELMs and do not exhibit hysteresis between the L-H and H-L transitions [6]. Furthermore, the superposition of (n_e,T_e) trajectories for discharges with increasing D_2 gas rates defines an upper boundary in edge parameters for Type I ELMs which is best fitted by $T_e=2.8\times10^{29}n_e^{-4/3}$(eV, m^{-3}). This is interpreted as an edge pressure gradient limit with the scale length varying as $(a\rho_i)^{1/2}$ or $T^{1/4}$ which is representative of a mixed transport model [7] and is favourable for ITER. As the gas rate increases, the energy confinement degrades and the plasma density ultimately decreases. Thus, a region of acceptable operation is defined and, in JET, this operational window in edge parameters shrinks with increasing I and B.

As can be seen from Fig. 5, which shows the upper boundary in edge parameters for Type I ELMs in deuterium and tritium plasmas, the maximum edge pressure increases with isotope mass.

3.4. Mass scaling of energy confinement time

A scan in the isotope mix from 100% deuterium to 100% tritium shows that in tritium the stored energy, density and edge pressure

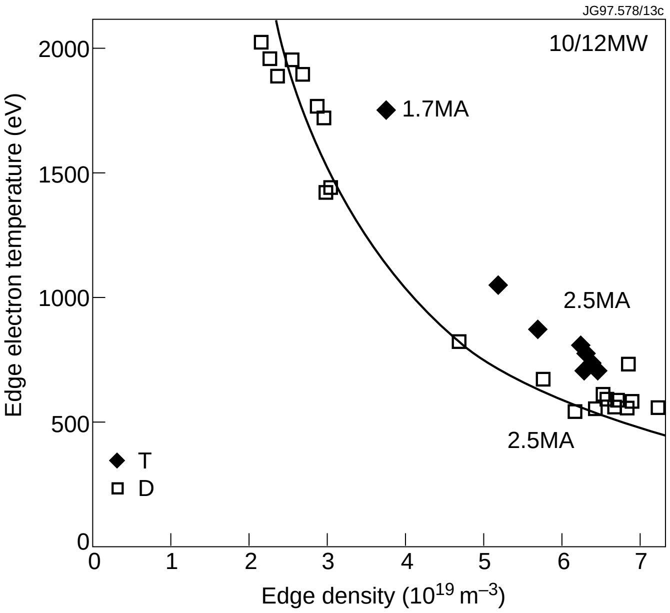

Fig. 5: Upper boundary in edge parameters for Type I ELMy H-mode discharges in D-D and D-T. The solid line shows the maximum edge pressure for I_p=1.7MA. Data from discharges at 2.5MA are scaled assuming p_e~I_p^2.

are higher and the ELM frequency is lower. A heating power of 8MW is sufficient to produce Type I ELMs in tritium, compared to 12MW needed in deuterium.

In order to firm up the confinement predictions for ITER a number of carefully designed ITER similarity pulses (so-called ρ* scans) with the exact ITER values of q_{95}, ν and - as far as JET's heating power permits - β_n have been carried out in D-T. As shown in Fig. 6, they fit well the EPS97 ELMy H-mode scaling

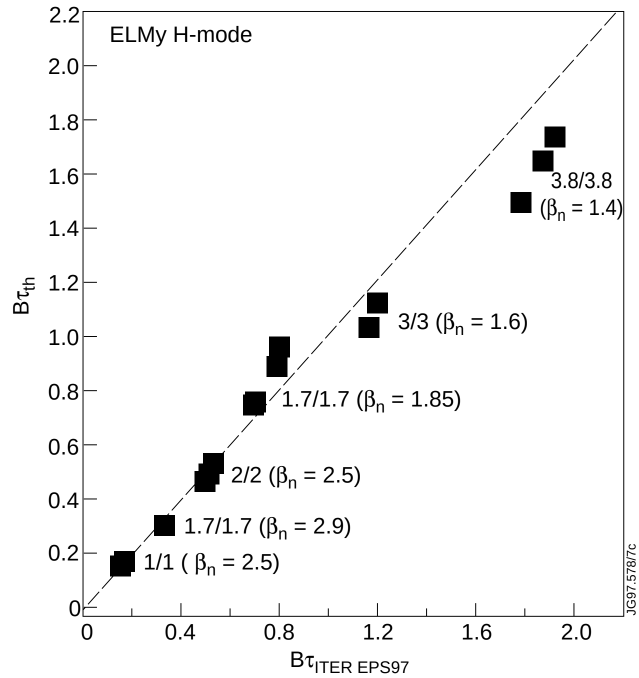

Fig. 6: ITER similarity pulses in D-T showing a good fit to the EPS97 energy confinement scaling [8] for ELMy H-modes.

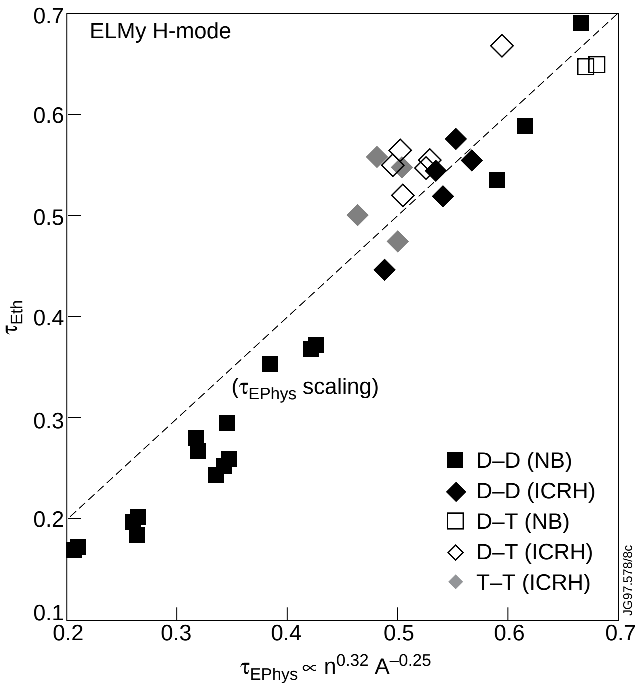

Fig. 7: Gyro-Bohm fit to JET energy confinement time data in D-D, D-T and T-T.

[8], which has a weak mass dependence (τ_E~$A^{0.2}$). (They fit also reasonably well pure Gyro-Bohm scaling with τ_E~$A^{-0.25}$).

An analysis of the whole JET D-T data set for ELMy H-mode discharges shows clearly that the $A^{0.41}$ dependence of the energy confinement time in the ITERH93-P scaling [9] is too strong and does not fit the JET D-T data. On the other hand, a pure Gyro-Bohm scaling has a slightly too negative mass dependence (Fig. 7). The best fit for ELMy H-modes is τ_E ~ $A^{-0.1\pm0.2}$ and this could be indicative of Gyro-Bohm transport in the bulk plasma and Bohm type transport at the edge. In predicting the confinement time for ITER, the weak mass dependence is compensated, however, by a more favourable density dependence provided ITER can be operated at or above the Greenwald density limit [10]. Further experiments in pure tritium (and later also in hydrogen) are planned to substantiate these findings.

4. HIGH FUSION PERFORMANCE

4.1. High fusion power "steady state" ELMy H-mode

During DTE1 the best ever fusion performance in an ELMy H-mode was achieved with the production of 14MJ of D-T fusion energy. As can be seen in Fig. 8, regular Type I ELMs kept the discharge (3.8MA/3.8T) "steady state" for more than 3s with the ratio of the fusion power to the input power, Q, being close to 0.2.

4.2. Optimised shear regime

The optimised shear regime is established after following a particular scenario [11]. First there is a fast current ramp and early plasma expansion to full aperture during which LHCD is

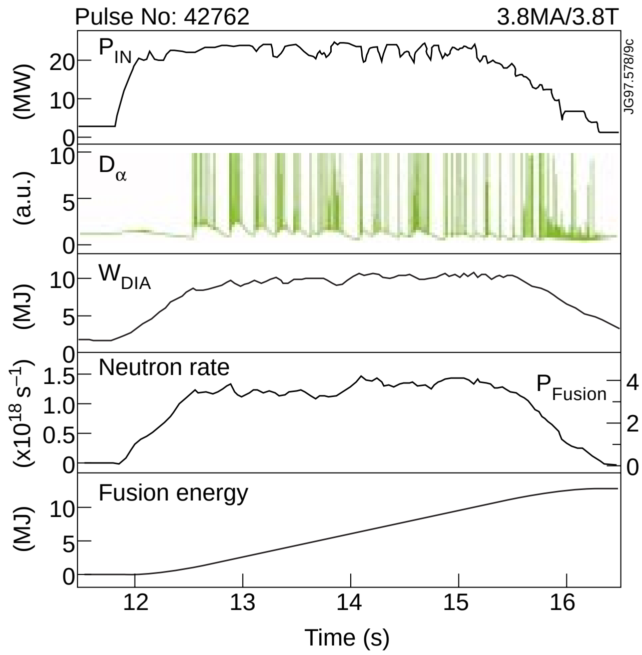

Fig. 8: Record fusion energy of 14MJ in a "steady-state" ELMy H-mode discharge.

applied to control the current profile. This is followed by ICRH pre-heating to slow down current diffusion. When the current profile is such that the volume within the q=2 surface is reasonably large (r/a ≈ 0.3-0.4), the full heating power, typically 16-18MW of NBI plus 6MW of ICRH, is applied. This leads to the formation of an Internal Transport Barrier (ITB) and plasma profiles become very peaked. The plasma pressure in the dense plasma core reaches 3bar at 3.45T, and even higher at 3.8T. In fact, the plasma in these optimised shear discharges is close to the ideal MHD stability limit for most of the heating pulse [12], with β_N increasing with time as the ITB moves outwards to ≈2/3 of the plasma radius and the pressure profile becomes less peaked. The highest performance has been achieved with small or slightly reversed central shear and q(0)=2-3.

The first D-T experiments on JET in the optimised shear regime have shown that the details of the scenario in deuterium could not be transferred directly to D-T. As a result, the scenario used for delaying the onset of the H-mode and for avoiding a β-limit disruption had to be reworked for D-T. None-the-less, after only two days of D-T operation, a refined scenario had been developed leading to the establishment of strong Internal Transport Barriers. In the best of these discharges (3.3MA/ 3.45T), 8.2MW of fusion power have been produced (Fig. 9). Further optimisation in D-T is planned.

4.3. Hot ion ELM-free H-mode

The highest fusion performance has been obtained in this type of discharge [13]. The best pulses are, however, always transient with fusion power increasing strongly during the ELM-free period which typically ends in a giant ELM after 1 to 2s of heating.

In order to optimise the D-T mixture with respect to performance and to gauge the relative importance of beam fuelling in the particle balance, the beam mix was first varied from 100% deuterium to 100% tritium for pure deuterium gas, walls and target plasma. From this it was concluded that the walls, target plasma and gas injection contribute about twice as much to the fuelling as the beams. After having loaded the walls and divertor

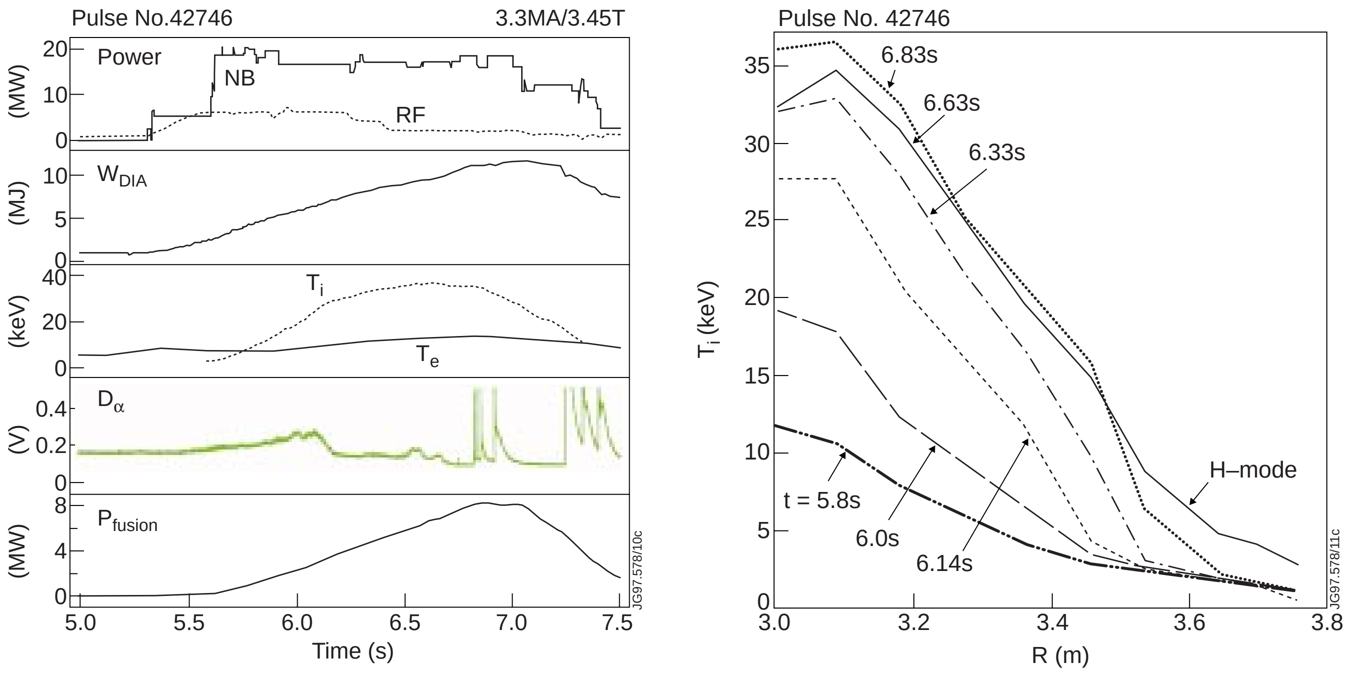

Fig. 9: Refined scenario for optimised shear discharges in D-T establishing a clear Internal Transport Barrier and producing a fusion power of 8.2MW.

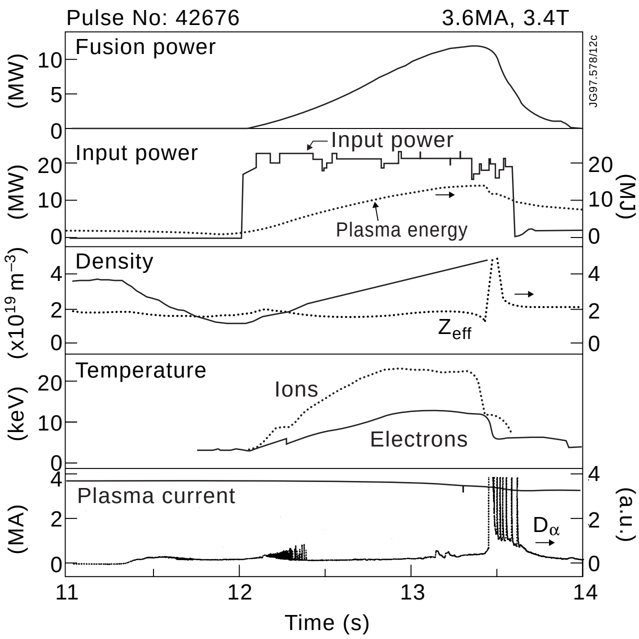

Fig. 10: Record fusion power of 13MW and record Q=0.6 in a hot ion ELM-free H-mode discharge.

TABLE I
Parameters for the High Fusion Power D-T Pulse
Pulse No. 42676 on 22nd September 1997
(values at t=13.3s)

Quantity	Value	Unit
I_p	3.6	MA
B_T	3.4	T
P_{NB}	18.7	MW
P_{ICRF}	3.6	MW
$n_e(0)$	3.8	$10^{19}m^{-3}$
$n_D(0)+n_T(0)$	3.0	$10^{19}m^{-3}$
Z_{eff}	2.1	
$T_i(0)$	23	keV
$T_e(0)$	14	keV
W_{dia}	15.3	MJ
$\dot{W}_{dia}$	6	MW
τ_E	0.9	s
$[n_D(0)+n_T(0)]T_i(0)\tau_E$	6	$10^{20}m^{-3}keVs$
$[n_D(0)+n_T(0)]/n_e$	0.8	(bremsstrahlung)
$[n_D(0)+n_T(0)]/n_e$	0.9	(axis CX)
$n_T/[n_D(0)+n_T(0)]$	0.6	(from NPA)
Neutron Rate	4.6	$10^{18}s^{-1}$
Fusion Power	12.9±10% >10MW for 0.53s	MW
Fusion Energy	12.5±10%	MJ

using 50:50 D:T gas fuelled ohmic pulses a discharge with 18.7MW of NB heating power augmented by 3.6MW of ICRH (Fig. 10) produced records in D-T fusion power (12.9MW peak, with more than 10MW for 0.5s), fusion energy (12.5MJ) and Q=0.6. Detailed parameters for this pulse are given in Table I. A similar result was obtained with 21.5MW of NB alone.

4.4. Alpha particle heating

In experiments with 11MW of tritium NB into a deuterium plasma and deuterium walls, Q (with dW/dt correction) exceeded 0.7 for about 1s. The temperatures and stored energy were higher than in the equivalent case with pure deuterium and this is most likely due to α-particle heating since the isotope effect on confinement has been found to be very weak (see section 3.4). In the near future, it is planned to conduct experiments which have been designed specifically to separate these two effects.

4.5. Alfvén Eigenmode Studies

Fast particles involved in plasma heating (NBI, ICRH, α-particles) can resonate with Alfvén waves, leading to particle and energy losses. In the case of Toroidal Alfvén Eigenmodes (TAEs) excited by the saddle coils and/or driven by ICRH, the measured damping rates for n=1 TAEs are found to be reduced with increasing ICRF heating power, indicating a net fast particle drive [14]. Low or intermediate n TAEs are unstable for relatively low ICRF heating power (>2.5-5MW), in line with theory. Magnetic fluctuation spectra indicate clear TAE activity driven by ICRH when the power is above a threshold (4-5MW at $5x10^{19}m^{-3}$). Different bands in the spectra correspond to different toroidal mode numbers (n=5-11). There is, however, no significant effect on overall performance.

The fluctuation spectra in D-T, on the other hand, show no evidence of Alfvénic instabilities driven by α-particles - or even a combination of α-particles and ICRH, which is just below the threshold - in the record fusion power hot ion H-mode pulses. Furthermore, although TAEs are observed in the afterglow of high fusion power optimised shear discharges, the identification of these as α-particle driven TAEs is difficult since Alfvénic instabilities driven by low power ICRH (≈1MW) have also been observed under similar conditions.

5. SUMMARY

5.1. D-T Operations

All major systems have been fully commissioned for D-T operation.The Active Gas Handling System has already supplied and reprocessed 40g of tritium and the tritium NB box has operated reliably up to 11.3MW for up to 5s. Measured tritium concentrations in the plasma and wall are well simulated using a model based on the Preliminary Tritium Experiments, thus providing a good basis for ITER predictions although an additional investigation of flakes etc. will be needed. In over 100 full D-T pulses, 340MJ of fusion energy has been produced already.

5.2. Standard ITER Mode of Operation: ELMy H-mode

For "steady state" ELMy H-mode discharges, the standard mode of operation foreseen for ITER, a number of physics issues have been studied in D-T.

Three ITER reference ICRH schemes have been successfully tested: second harmonic tritium ($2\omega_{CT}$), with and without additional He^3, and fundamental minority deuterium in a tritium plasma (ω_{CD}). The D(T) and (He^3)DT schemes show strong ion heating and D(T) has already produced steady state Q values of≈0.2.

The D-T H-mode data show that the present ITER scaling for the L-H threshold power needs to be modified to include a favourable mass dependence (≈1/A). This reduces the predicted threshold power for ITER by at least 25% in D-T and even more in pure tritium.

Finally, the JET D-T data suggest that the $A^{0.41}$ mass dependence in the ITER scaling for energy confinement in ELMy H-modes should be removed. This, however, is compensated by a more favourable density dependence provided ITER can be operated at or above the Greenwald density limit.

5.3. Fusion Power Performance

A world record fusion energy (14MJ) has been produced in a standard ITER ELMy H-mode.

The optimised shear scenario used in D-D had to be modified in D-T due to the early onset of the H-mode. None-the-less, after only two days of operation, internal transport barriers were established and 8.2MW of fusion power was produced.

World records of fusion power (13MW) and Q (0.6) have been set in an hot ion ELM-free H-mode. Further optimisation is planned for the future. The higher temperatures and stored energies observed in these D-T discharges are most likely due to α-particle heating.

So far, no α-particle driven Alfvénic instabilities have been observed in discharges with high fusion power.

5.4. Conclusion

In conclusion, a wealth of new data for ITER has been produced during the first half of DTE1. More D-T results are expected within the next weeks before JET continues with an ITER physics programme in hydrogen and deuterium.This will then be followed, early in 1998, by the fully remote handling installation of the Mark II Gas Box divertor in an activated environment.

ACKNOWLEDGEMENTS

The author wishes to thank all his colleagues who have contributed to this paper and in particular ML Watkins, P Andrew, JG Cordey, A Fasoli, C Gormezano, L Horton, J Lingertat, PJ Lomas, DF Start and P Thomas for their contributions and clarifying discussions and BE Keen for his assistance in editing the paper.

REFERENCES

[1] The JET Team (presented by D Stork), this conference
[2] The JET Team, 1992 Nucl. Fusion **32** 187
[3] Horton LD, 1992 Journ. Nucl. Mat. **196-198** 13
[4] Eriksson LG et al, 1993 Nucl. Fusion **33** 1037
[5] Takizuka T et al., Proc. 16th IAEA Fusion Energy Conference, Montreal (Canada) IAEA-CN-64/F-5 (1996)
[6] Lingertat H et al., Europhysics Conference Abstracts (Proc. of the 24th EPS Conference on Controlled Fusion and Plasma Physics, Berchtesgaden, Germany 1997), Vol21C (to be published)
[7] Wilson HR et al., Ibid
[8] Greenwald M et al., Ibid
[9] ITER H-mode Database Working Group, Controlled Fusion and Plasma Physics (Proc. 20th Eur. Conf. Lisbon, 1993), Vol. 17C, Part I, European Physical Society, Geneva (1993) 15
[10] Greenwald M et al., 1988 Nucl. Fusion **28** 2199
[11] Söldner F et al., Europhysics Conference Abstracts (Proc. of the 24th EPS Conference on Controlled Fusion and Plasma Physics, Berchtesgaden, Germany 1997), Vol21C (to be published)
[12] Huysmans G et al., Ibid
[13] Rimini F et al., Ibid
[14] Fasoli A et al., Ibid