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D-T Plasmas in the Joint European Torus (JET): Behaviour and Implications

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* - see Appendix I

Abstract

This paper reports the first Deuterium-Tritium (D-T) fusion experiments in the geometry of the International Thermonuclear Reactor (ITER) *(R. Aymar, V. Chuyanov, M. Huguet, R. Parker, Y. Shimamura, and the ITER Joint Central Team and Home Teams, Proceedings of the 16th International Conference on Fusion Energy (International Atomic Energy Agency (IAEA), Vienna), Fusion Energy **1** (1996) 3)*, with long pulse length and an ITER-like divertor. Physics aspects, such as the isotope dependence of confinement, the H-mode threshold, shear optimisation, heating methods, high fusion performance and alpha particle heating, are discussed together with their implications. The technology aspects of tritium wall loading and clean-up, the close coupled tritium plant and the future remote handling divertor target exchange are also mentioned.

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I. INTRODUCTION

This paper reports the first results from a very recent series of deuterium-tritium (D-T) experiments on the Joint European Torus (JET). Analysis of the data from these experiments is still in progress and minor revisions can be expected when the analysed data is published in more detail in due course.

Significant fusion power was first produced in JET in 1991[1], when a hot-ion plasma containing 11% tritium in deuterium produced 2MJ of fusion energy with a peak fusion power of 1.7MW and a transient Q = 0.15 (P_{fuse}/P_{in}=0.12). These plasmas were terminated by MHD events leading to wall overheating and large carbon influxes.

Since 1991, JET has carried out a number of configuration changes, including the installation of a sequence of divertor types. The present Mark IIAP divertor, with its large tiles and accurately aligned surfaces, is extremely effective at accepting plasma heat loads so that large impurity influxes no longer occur. However the duration of the highest performance is still limited by MHD events.

Reliable, repeatable high performance D-D plasmas are now routinely produced in an ITER-like divertor configuration in a variety of plasma modes. Consequently, a more extensive but closely focused D-T programme has been carried out to address specific issues of D-T operation. The results from this series of experiments are particularly relevant to the proposed ITER[2] experiment because JET has very similar geometry to ITER. JET is essentially a one third scale model for ITER and is the best available system for extrapolation and for the evaluation of ITER scenarios.

2. OBJECTIVES

The objectives for the experiments are listed below under three headings:

A. Fusion Performance

1. Attempt to obtain maximum fusion power and ratio of fusion power to power absorbed in the plasma in both the hot-ion edge localised mode-free (ELM-free) high confinement-mode (H-mode) [3, 4, 5] and the optimised shear [4, 5, 6, 7].
2. Attempt to obtain high values of these quantities in steady-state in the ELMy H-mode [5].
3. Attempt to observe alpha particle heating.

B. ITER Relevant

1. Study isotope effects on energy confinement time and H-mode threshold in ITER-like ELMy H-mode D-T plasmas.
2. Evaluate ICRH D-T Heating Schemes.
3. Examine whether or not alpha driven instabilities appear.

C. Technology

1. Demonstrate the operation of an ITER scale tritium processing plant operating close coupled to a tokamak.
2. At the end of the experiments, replace the divertor target plate assembly using fully remote handling techniques inside the activated torus.

3. FUSION PERFORMANCE

A. The Optimised Shear Mode of Operation

Regions of high confinement known as internal transport barriers are produced in JET by optimising the magnetic shear. A target plasma, with low magnetic shear in the core, is formed by preheating the plasma during the current ramp phase. A combination of Lower Hybrid, Ion Cyclotron (ICRH) and Neutral Beam (NBI) heating is used. When the current profile is such that the q=2 surface has a reasonable size (about one third of the minor radius), full heating power is applied; typically this is 16-18 MW of NBI plus 6MW of ICRH. This scenario allows the early formation of a region of high confinement which appears to be largely coincident with the low shear region.

In D-T, the lower H-mode threshold makes it difficult to avoid an early edge H-mode pedestal, which inhibits beam penetration and prevents internal barrier formation. Nevertheless, a narrow window of parameters has been found where an internal barrier can be formed. At B_T = 3.4T, after careful tuning, a similar internal barrier to that in D-D was formed and sustained to a similar H-mode termination. A particular discharge (Pulse No: 42746) produced a fusion power, P_{fuse} = 8.2MW for an input power of 19MW, at the first edge localised mode (ELM). This is ~20% less than could have been expected from the equivalent D-D pulse, because the D-D pulse had a longer L-mode phase with high core density. Further work is needed to combine optimum fuelling with the other conditions for barrier formation in the D-T case.

B. High Fusion Power 'Steady State' ELMy H-mode

Near steady state ELMy H-mode plasmas have been obtained with ITER magnetic geometry and q value (q_{95} = 3.4). Pulse No: 42982 had I_p=3.8MA; B_T=3.8T; P_{IN} =24MW (mainly NBI with ~10% ICRH). The density profile was flat with a central value $n_e(0)=8x10^{19}m^{-3}$. The central ion and electron temperatures, $T_i(0)$ and $T_e(0)$, were both about 8keV. The plasma energy, energy replacement time and normalised β were: W_p=10MJ; τ_E= 0.45s and β_N=1.3.

The fusion energy produced was W_{fuse} = 21.7MJ with a fusion power P_{fuse} ~4MW, which was steady for 3.5s. One third of the fusion power came from thermal processes and two thirds from beam plasma and beam-beam interactions. The ratio of the fusion energy produced to the energy supplied to the plasma, W_{fuse}/W_{in}, was 0.18 over a 3.5s period or eight energy replacement times. The time development of parameters of interest is shown in Fig.1.

The duration of the discharge was limited only by the beam duration and could have been extended. The energy confinement time τ_E is the same as for a similar D-D discharge indicating that τ_E has little isotope dependence. The ELM frequency was somewhat lower in D-T than in D-D.

C. Hot Ion ELM Free H-Mode

The highest fusion performance has been obtained in this type of pulse. The highest performance pulses are, however, always transient with plasma parameters and fusion power increasing strongly over an energy confinement time (τ_E ~1s).

The pulses typically end in a giant ELM after 1 to 2s of heating. The energy released in the ELM can be >1MJ, up to 15% of the plasma energy. The ELM is provoked by steepening edge gradients as the central ion temperature increases. Typically the confinement degrades and the rate of increase of plasma energy slows in the 0.2 to 0.3s before the ELM.

Four such JET pulses have produced between 12 and 16MW of fusion power each. Table I shows the main parameters for the pulse which produced the highest fusion power. Figure 2 shows the time development of parameters of interest. It can be seen from Fig. 2, that in the final 0.15s of the pulse before the ELM, the ratio of fusion power to absorbed power P_{fuse}/P_{abs} = 0.66 (where P_{abs} is defined as the sum of the power deposited by NBI, ICRH and ohmic power with losses due to beam shinethrough and beam charge exchange subtracted). Over this interval the plasma energy (W_{dia}) increases from 16.4 to 17.0 MJ, a mean rate of 4.0 MW; i.e. 17% of the absorbed power is being used to increase the plasma energy. In order to quantify the possible performance of a similar plasma in steady state, a quantity $Q_{transient}$ is defined[1] which takes into account this rise in stored energy for transient discharges but which reverts to P_{fuse}/P_{abs} in steady state.

$$Q_{transient} = Q_{th} + Q_{nonth}$$

$$Q_{th} = {}^{P_{fuse}}\!\!/_{P_{abs}} \cdot f_{th} \cdot \left[\frac{P_{abs}}{P_{abs} - \dot{W}}\right]$$

$$Q_{nonth} = {}^{P_{fuse}}\!\!/_{P_{abs}} \cdot (1 - f_{th})$$

where $P_{abs} = P_{NB} + P_{RF} + P_{OH} - P_{SH} - P_{BCX}$

The thermonuclear fraction f_{th} is the ratio of thermonuclear yield (computed by the TRANSP code from kinetic measurements) to the total measured yield and is approximately 0.6 for this pulse. The non-thermonuclear fraction ($1 - f_{th}$) is made up of beam-beam and beam-plasma reactions and includes the contribution from ICRF acceleration of deuterons.

$Q_{transient} \approx 0.74$ averaged over the final 0.15s before the ELM (t=13.26 to 13.41s). Earlier in the pulse, before the confinement degradation, $P_{fuse}/P_{abs} = 0.59$ (averaged over a similar 0.15s interval from 13.05 to 13.20s) but now the mean rate of increase of plasma energy is >40% of the absorbed power and $Q_{transient} \approx 0.84$. $Q_{transient}$ can also be computed as a function of time, and such a calculation is shown in fig. 2 using a smoothed backward derivative of plasma energy. Further analysis is in progress to refine and validate these calculations, but it is clear that in this pulse $Q_{transient}$ is greater than about 0.8 for 0.4s or half an energy replacement time.

If a discharge could be obtained in steady state with the same density, temperature and energy confinement time, then it would have the same Q_{th} (which is a function only of n, T and τ_E [9]) and the same Q_{nonth} (which for beam heating is a function only of beam energy and plasma parameters - mainly T_e [9, 10]). Such a steady state plasma could be obtained by stepping down the beam power and thus would have $P_{fuse}/P_{abs} \approx 0.8$. Step down experiments have been performed in deuterium plasmas [11, 12], but to achieve the step down at high performance before the occurrence of the ELM requires development over many pulses. This was not possible in this series of D-T experiments within the total allowed neutron budget which was set to facilitate repair in the event of a future in-vessel failure.

4. ITER RELEVANT RESULTS

A. Mass Scaling of Energy Confinement Time

JET data has been obtained in ELM-free and ELMy H-mode discharges with 100%D; 50%T and >80%T, corresponding to approximate effective ion masses of A=2; 2.5 and 3. These discharges all have the same configuration with q_{95} in the range 3.3 to 3.6.

In Fig.3, the data is compared with the scalings derived by the ITER Database Group[13, 14] and given in Table II. Further analysis should reduce the scatter in the JET data points and experiments with hydrogen plasmas are planned as an anchor point for the data. However, it is clear that for both ELM-free and ELMy discharges, the JET data does not support any scaling with ion mass and $\tau_E \propto A^{0\pm0.2}$ would be an acceptable fit to both. Scalings of the gyro-reduced Bohm type suggested by a number of turbulence theories for core confinement in tokamaks could thus be accommodated by the data.

The JET ELM-free data does not support the $A^{0.41}$ dependence in the H93P scaling. If the constant in the ITER scaling relation in Table II is fitted to this JET data and compared to a similar fit but with an A^{0} dependence (instead of $A^{0.41}$) imposed, the projected value for ITER is reduced by 13%. Similarly, scaling the JET ELMy data as A^{0} instead of the EPS97, $A^{0.2}$ results in a ~4% reduction in the projected ITER τ_E. Actual projections of the multi-machine database to ITER is the subject of much discussion and interpretation [14]. It is clear, however, that the present database projections will have to be modified to take account of the lack of isotope scaling in the JET data, especially since theory also supports a weak dependence on ion mass. This is not a trivial task but these initial considerations suggest that it may not lead to any substantial change in the projected energy replacement time for ITER.

B. Change in H-mode Threshold between D-D and D-T plasmas

The H-mode threshold was determined by observing the onset of ELMs in the divertor target D_α light emission as the applied ICRF power was ramped up over a 3s period. The threshold clearly decreased from D-D to T-T discharges (from 2.9MW to 1.7MW in a 2.6MA/2.6T pulse). Comparing D-D, D-T and T-T discharges indicates threshold power approximately proportional to A^{-1}. The same behaviour is found in both ICRF and beam heated discharges.

In Fig. 4, the threshold loss power observed in JET is compared with the ITER database scaling [15],

$$P_{thres} \propto \bar{n}^{0.75} B_T R^2$$

and with the same scaling with an A^{-1} scaling imposed. Clearly, the A^{-1} dependence is a better fit.

Using the A^{-1} scaling fitted to the JET deuterium data in this set gives, for ITER parameters, threshold powers of 70MW for a D-T plasma and 58MW for a pure tritium plasma. These values are to be compared with the fit with no A dependence, which gives 88MW. These predicted reductions in threshold power offer improved operational flexibility for ITER.

C. Comparison of Fusion Power from Similar D-T and D-D Discharges

It is important to verify that the fusion power gain (calculated from the reaction rate coefficients) between similar D-D and D-T discharges is realised in practice. Comparisons have been made of JET discharges in the Hot-Ion ELM-free H-mode in D-D and in approximately 50:50 D-T discharges. These comparisons typically show that the calculated ratio [16] (about 88 times in neutron rate and 210 times in fusion power) is achieved between discharges with similar parameters. In a particular case (D-D Pulse No:40305 and D-T Pulse No:42676 at 3.8MA; 3.4T with central ion temperature ~25keV, predominately neutral beam heating and an approximate 50:50 D-T mixture), the values of the ion temperature, the electron density, the thermal ion energy content and the total applied heating power in the two pulses were the same to within 10% during the 0.3s high performance phase. The ratio of the neutron rates in this period was in the range 85 to 100.

The TRANSP[16] code was used to model JET D-D Pulse No:40305, the beam sources were then changed to those of the corresponding D-T pulse (Pulse No:42676). The predicted neutron rate then matched the observed neutron rate in the D-T pulse which had similar parameters to those in the TRANSP simulation.

These results show that the fusion power to be expected from a D-T pulse can be predicted confidently from a D-D pulse with the same parameters. They do not imply that the behaviour of a D-T pulse can be inferred from a D-D pulse with the same machine settings since many effects, particularly the difference in H-mode threshold, can mean that the behaviour of D-T and D-D discharges can differ strongly even if input parameters are identical.

D. ICRF Heating of D-T Plasmas

ICRF heating of D-T plasmas has been demonstrated for three schemes in ITER geometry:

a) minority D heating (ω_{CD}) in T plasmas (+10%D);

b) $2\omega_{CT}$ in 50:50 D-T;

c) minority He^3 heating (ω_{CHe3}) in 50:50 D-T (+ 5% to 10%He^3).

All three schemes are effective.

The ω_{CD} scheme has given the highest fusion performance. In a particular discharge (Pulse No: 43015) at I_p =3.7MA; B_T = 3.7T; f=28MHz (central resonance), for an input ICRF power of 6MW, 1.6MW of fusion power was obtained. This corresponds to a peak P_{fuse}/P_{in} = 0.25 and W_{fuse}/W_{in} = 0.22 for 2.7s which corresponds to three energy replacement times.

The neutron rate is mainly (~95%) from supra-thermal ions at ~120keV. The results are in agreement with the PION code(17) which shows that the ICRF power gives 45% to electrons and 55% to ions. The plasma density is high, the D concentration is 10% and the D energy is near E_{crit} and near the peak fusion cross-section. Here, E_{crit} is the energy at which fast ions slowing down in the plasma give energy at equal rates to electrons and to plasma ions.

The other two schemes are equally efficient in heating the plasma. In the $2\omega_{CT}$ scheme with no added He^3, high energy tritons form and mainly heat electrons. In the (He^3) D-T scheme, He^3 minority ions heat the plasma ions efficiently due to smaller tail energies and higher E_{crit}. This scheme differs from that used in the Tokamak Fusion Test Reactor (TFTR), where an admixture of He^3 was used to improve the second harmonic ($2\omega_{CT}$) heating of tritium(18).

E. Alpha Particle Heating

The best JET plasmas have about 3MW of plasma heating due to alpha particles compared to total power inputs of ~24MW. Since the alpha-power is centrally peaked and goes mainly to electrons it should be observable, despite competition from the other inputs to the electrons. In order to identify alpha heating and separate it from possible isotopic effects on heating, equipartition and confinement, scans were carried out of similar discharges with varying percentages of tritium but with the external power supplied to the plasma kept roughly constant.

A version of the 3.8MA, 3.4T high fusion yield pulses with 10.5MW of neutral beam power was used. This power was dictated by the need to keep both applied power and neutral particle sources as constant as possible across the scan. A hot-ion H-mode was obtained and a maximum of 6.7MW fusion power obtained at the optimum D-T mixture.

An example of the data is shown in Fig. 5 where the central electron temperature variation with tritium fraction γ_T is shown together with the total power delivered to the plasma. The tritium fraction is obtained from TRANSP code runs which use the actual injected beam parameters but adjust the wall recycled tritium fraction to match the observed neutron rate. This procedure was used because, for this scan, the tritium fractions estimated spectroscopically and from neutral particle analysis were not reproducible.

The observation that highest T_e occurs in optimum mixtures while the observed T_e increases with increasing alpha power is a clear indication of significant alpha heating. In these plasmas, the thermal neutron yield is about half the total and the optimum is for γ_T somewhat bigger than 0.5.

In preparing this experiment, pulses were performed using ICRF minority heating of deuterium plasmas with the heating power proportional to the measured D-D neutron rate (to simulate alpha heating). Increases of electron temperature were observed similar to those in the D-T experiment and the efficiency of electron heating was similar for the ICRF power coupled in deuterium and for the alpha particle heating observed in D-T in both cases $\Delta T/\Delta P \sim 1$keV/MW.

F. Alfven Eigenmode Behaviour

In D-D discharges with ICRF power above ~5MW, magnetic fluctuation spectra obtained from magnetic pick-up coils clearly show TAE modes driven by the ICRF generated fast particles.

In the D-T high performance hot-ion mode (Pulse No:42676), which had 2.6MW of alpha power, there was no indication of any TAE modes. Specific TAE stability diagrams for these pulses have still to be calculated. However, comparison with the stability plot for a generically similar pulse, indicates that Pulse No:42676 marginally approaches the instability region when the alpha particle pressure is at its highest.

5. TECHNICAL ASPECTS

A. Changeover from D-D to D-T Recycling

It has proved relatively straight forward to change from D-D to D-T discharges in JET. High plasma tritium concentrations require both tritium gas puffing and tritium wall loading.

1. Method

A series of RF heated discharges (at a few MW level, preferably in H-mode with ELMs) was performed with Tritium only puffing. The tritium fraction, $\gamma_T = n_T/(n_T + n_D)$, was measured at the edge and in the core using four different methods. The methods show good agreement for a wide range of ohmic discharges and show differences consistent with the expected tritium radial profiles in other discharges. The methods used were as follows:

(a) Neutral Particle Analysis by electrostatic deflection and time of flight, this gives a measurement typical of a zone 20 to 40cm in from the plasma edge;

(b) High resolution Balmer alpha line spectroscopy, this gives a measurement typical of the plasma edge region;

(c) Balmer alpha spectroscopy of a Penning discharge in neutral gas evolved from the divertor plate;

(d) Neutron rate from a short deuterium neutral injection pulse, this gives a measurement of tritium concentration in the core region.

2. Results

After five discharges, $\gamma_T > 0.6$ both before and after heating and there is no appreciable variation of γ_T between the core and the edge. After ~20 pure tritium discharges with the admission of ~5g of tritium, $\gamma_T > 0.9$ and the wall inventory is ~2g of tritium. These results for plasma tritium concentrations are consistent with the multi-reservoir model developed during JET's 1991 D-T experiments and with the wall loading seen at that time[19, 20].

B. Removal of Tritium from Torus Walls (Clean-Up)

Two series of clean-up experiments have been performed, the first in June 1997, after about 11.5g of T_2 had been introduced into the torus (0.05g from beams, the rest as gas puff). Clean-up began with about 4.4g of tritium retained on the walls.

The most effective clean-up method used ohmic and ICRF pulses with various strike points. The use of NBI was restricted to limit neutron production. The wall load was reduced to ~2.9g in a four-day period with ~120 pulses. The plasma tritium fraction fell from around 80% to between 1 and 2%. Little further tritium was evolved during a two month period with the torus filled with nitrogen gas at room temperature. During the following 1 month of operation with 700 D-D pulses the wall load was reduced to ~2.2g.

Subsequently, a further 24g of T_2 has been introduced into the torus. Measurements during cryo-regeneration indicate that most hold-up is in the torus rather than in the Neutral Injection system and it is estimated that there was ~11.5g of tritium on the torus walls when the second clean-up sequence began in November 1997. After 300 deuterium pulses over a two-week period, the plasma tritium fraction was reduced to less than 1% and the wall inventory had fallen to 8.5g.

These estimates indicate that about 40% of tritium introduced into the JET torus is retained in the wall material until displaced by clean-up discharges. The wall material is largely carbon fibre composite, mainly at 300°C but with some colder areas where carbon flakes are known to form. In deuterium operation, these flakes have been found to contain large quantities of embedded deuterium.

C. The Tritium Processing System (AGHS)

Using this close coupled active gas handling system (AGHS), the 20g of T_2 on the JET site has been reprocessed and reused allowing about 100g to be supplied to JET.

The plant has two isotope separation systems.

(a) Gas Chromatography which is used for the separation of relatively rich tritium exhaust mixtures, recovered during torus and Neutral Beam Injection System cryo-regenerations. 10 to 20% T_2 input mixtures are routinely separated to >99.5% T_2 as required for the beam systems.

(b) Cryodistillation which is used for the pre-enrichment of weak tritium mixtures (typically < 0.5%) from tailings and other arisings and for clean-up of hydrogen and deuterium containing traces of tritium. 2.5m^3 of gas has been processed by this system. Almost 1g of T_2 has been recovered. The processed gas routinely has <1ppm tritium and can be discharged to atmosphere.

D. The Remote Handling Target Exchange

DTE1 will be followed by about two months of D-D and H-H operation, after this, the present Mark IIAP divertor target assembly will be removed and replaced by a new gas-box divertor using entirely remote handling techniques inside the activated torus. The Remote Handling equipment is fully proven and available. The methodology is based on a 'man-in-the-loop' strategy with coarse positioning operations carried out by computer 'teach' files and final operations executed by a man using a force reflecting servo system. This strategy allows checking and inspection of all planned operations and gives an intelligent response capability for unforeseen events. The remote handling operations procedures are all complete and have been proven using the actual remote handling equipment operating inside a full size torus mock-up. This remote installation operation on JET is crucial to establish the credibility of maintenance on future fusion devices.

6. SUMMARY

A. Important Results for ITER

1. There is no mass scaling of energy confinement time between similar JET discharges in D-D and D-T. Direct scaling from JET suggests only a small effect on the predicted τ_E for ITER.
2. There is a reduction in H-mode threshold approximately $\propto A^{-1}$. Direct scaling from JET suggest significant (20 to 30%) reductions in threshold power for ITER which should increase operational flexibility.
3. The increase in fusion yield between similar D-D and D-T discharges agrees with that expected from the reaction rate coefficients.
4. Three important ICRF heating schemes have been shown to be effective in D-T plasmas in ITER geometry: ω_{CD}; $2\omega_{CT}$ and $2\omega_{CT}$ (He^3).
5. There is clear evidence of alpha particle heating.
6. TAE modes driven by high energy RF tails are clearly seen in D-D plasmas but no alpha particle driven modes have been seen in D-T plasmas.

B. Fusion Performance

1. Optimised Shear

The window for optimised shear operation in JET is narrower in D-T than in D-D. 8.2MW of fusion power has been obtained and it is clear that further optimisation is possible.

2. Steady State ELMy H-mode

This is a preferred ITER operation mode. It is an important advance that it has now been demonstrated in ITER geometry in D-T. A particular JET pulse has P_{fuse} = 4MW steady for 3.5s with W_{fuse} = 22MJ and $Q=W_{fuse}/W_{in}$ = 0.18 for 3.5s or eight energy replacement times.

3. Hot-Ion ELM Free H-mode

This is the highest performance mode in JET and produces P_{fuse}=16MW (>10MW for 0.7s) and W_{fuse}=14MJ with P_{fuse}/P_{abs}=0.66 for ~0.15s. At this time, the plasma energy is increasing at a rate of 4MW. If a discharge with similar plasma parameters could be obtained in steady state, P_{fuse}/P_{abs} should reach 0.8.

C. Progress with Fusion Power Production

Figure 6 summarises developments. The diagram encompasses: 10% T in D experiments in JET in 1991; a result from the extensive and fruitful D-T studies on TFTR (1993 to 1997)[10]; and High Fusion Power and Long Fusion Pulse, quasi-steady state operation, in the present series of JET experiments.

Future work planned for JET includes : the study of divertors similar to the ITER proposal; the development of the optimised shear and long pulse regimes; improvements to the scaling of JET data towards ITER. This work, together with the demonstration of reactor relevant technology at JET will form the basis of future progress with fusion power production. This progress will require a new device to study plasmas near ignition. The results reported in this paper set the scene for such a device for which plans are being developed by the ITER Design Team[2].

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TABLE I

Provisional Parameters for the JET High Fusion Power D-T Pulse

Pulse No. 42976 on 31st October 1997
(values at t=13.3s into pulse)

Configuration: Single Null (bottom): Ion ∇B drift downwards.

Quantity	Value	Unit
I_p (plasma current)	4.0	MA
B_T (toroidal field - on-axis)	3.6	T
P_{NB} (NB power)	22.3	MW
P_{ICRF} (ICRF power)	3.1	MW
P_{abs}*	23.8	MW
$n_e(0)$ (central electron density)	4.2	$10^{19}m^{-3}$
$n_D(0)+n_T(0)$ (central D & T ion densities)	3.5	$10^{19}m^{-3}$
$\bar{Z}_{eff}$ (effective ion charge)	2.1	
$T_i(0)$ (central ion temperature)	28	keV
$T_e(0)$ (central electron temperature)	14	keV
W_{dia} (plasma energy)	17	MJ
$\dot{W}_{dia}$	~4	MW
τ_E	0.75	s
$[n_D(0)+n_T(0)]T_i(0)\tau_E$	7.5±10%	$10^{20}m^{-3}$keVs
$(n_D+n_T)/n_e$	0.8	
$n_T/(n_D+n_T)$	0.6	(from NPA)
Neutron Rate	5.7±10%	$10^{18}s^{-1}$
Fusion Power	16.1±10%	MW
	>15MW for 0.22s	
	>10MW for 0.66s	
Fusion Energy	13.8±10%	MJ
$f_{th} = P_{fuse}(\text{thermal})/P_{fuse}$	= 0.6 - 0.7	From TRANSP code[8]

* - P_{abs} is the input power less shinethrough and beam charge exchange losses

TABLE II

Scalings derived by theITER Database Group[13, 14]

$$\tau_{Eth} = c\, I^{i}\, B^{b}\, P^{p}\, n^{n}\, R^{r}\, \varepsilon^{e}\, \kappa^{k}\, A^{a}$$

Exponent	ELM FREE (H93P)	ELMy (EPS 97)
	ITER data base	ITER data base
i	+1.06	+0.90
b	+0.32	+0.20
p	-0.67	-0.66
n	+0.17	+0.40
r	+1.79	+2.03
e	-0.11	+0.19
k	+0.66	+0.92
a	+0.41	+0.20
	c=0.036	c=0.029

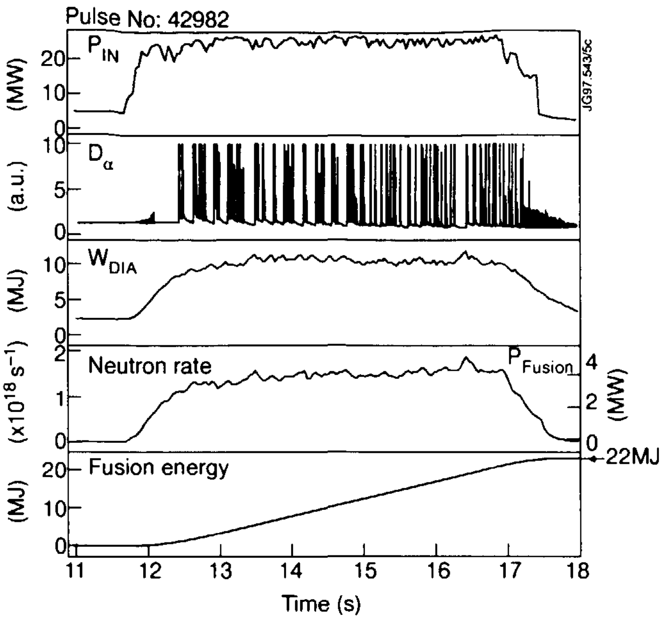

Fig. 1. A number of quantities for the steady state ELMy H-mode (Pulse No:42982) are shown. The traces from top to bottom are: Input Power; Divertor D_{α} light; plasma diamagnetic energy; 14 MeV neutron rate and the corresponding fusion power; and fusion energy. The neutron rate increases slowly throughout the pulse because the plasma density slowly falls. The plasma energy remains constant while the ion and electron temperatures increase giving increased reactivity.

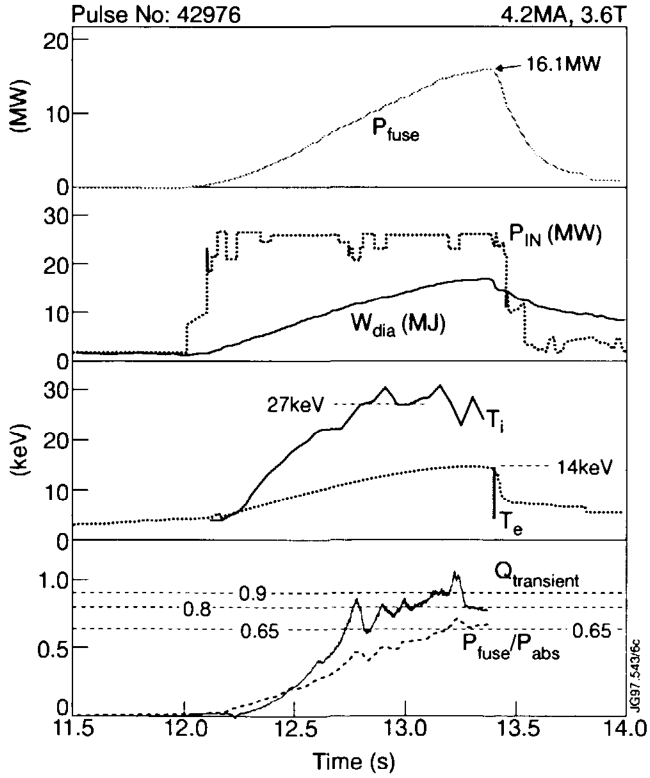

Fig. 2. Traces for the High Fusion Yield Hot-ion discharge (Pulse No:42976). From top to bottom, the traces are Fusion Power, Input Power and Plasma Energy (W_{dia}); Central Ion and Electron Temperature; the quantity $Q_{transient}$ defined in the text and the ratio of Fusion Power to Input Power absorbed by the plasma.

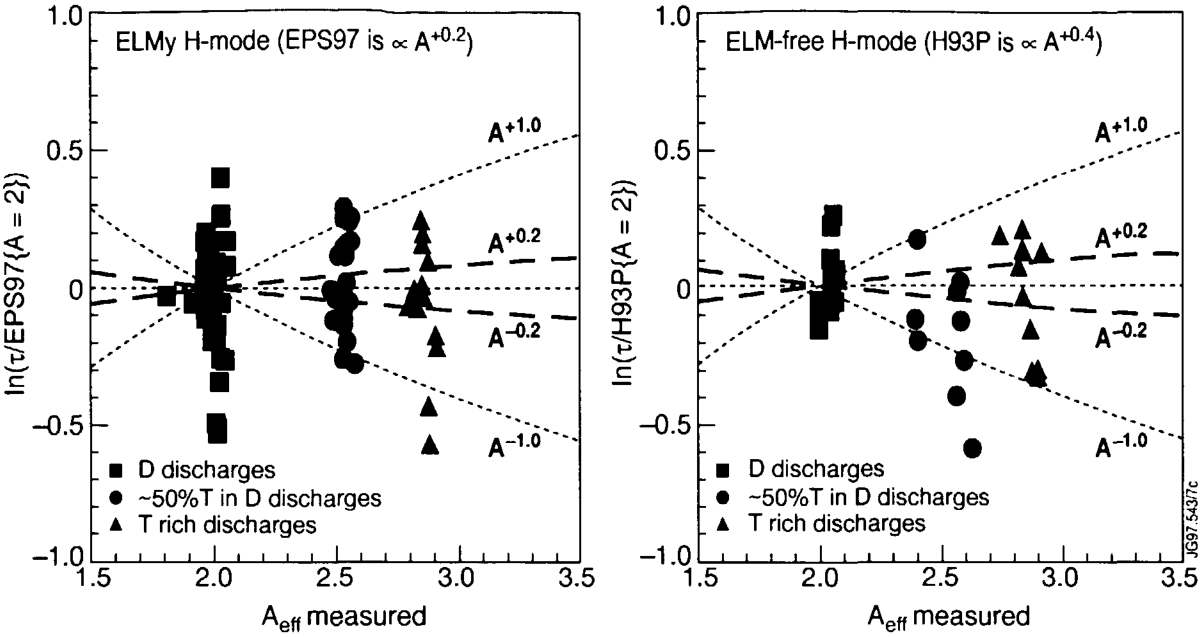

Fig. 3. The ratio of the observed JET energy replacement time to the ITER Database prediction, with A=2 for both ELMy and ELM-free H-mode discharges. The data is plotted on a natural logarithmic scale against the plasma mass number. The curves show the behaviour to be expected for the A dependence indicated.

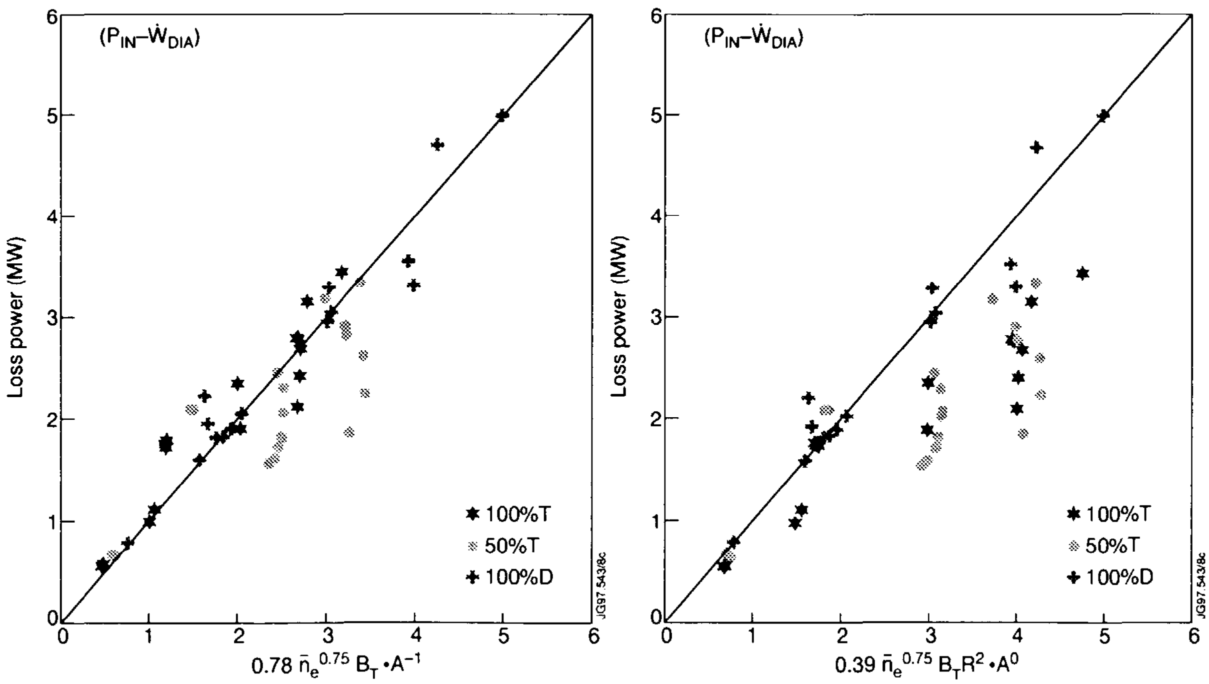

Fig. 4. JET H-mode threshold data compared to the ITER Database scaling fitted to the A=2 data and the same data fitted with an A^{-1} introduced into the scaling. The line shown in each case is the best fit to the JET deuterium pulses in this data set.

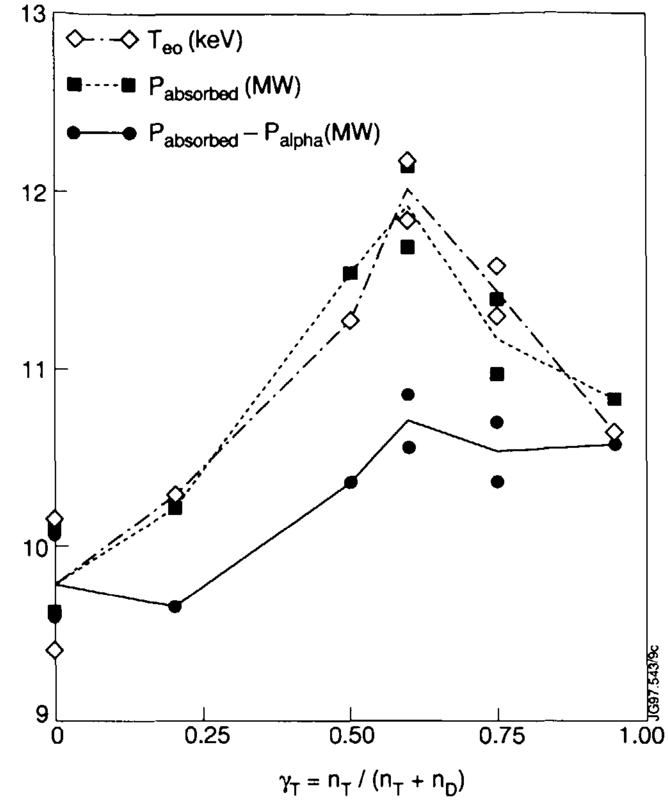

Fig. 5. The close correlation of the central electron temperature with total power delivered to the plasma (including the alpha particle power) as the tritium fraction (estimated from TRANSP simulations) is varied. The lower curve shows the absorbed externally supplied power (i.e. the alpha power is not included). The fact that this power is not constant as γ_T is varied complicates the analysis of the data.

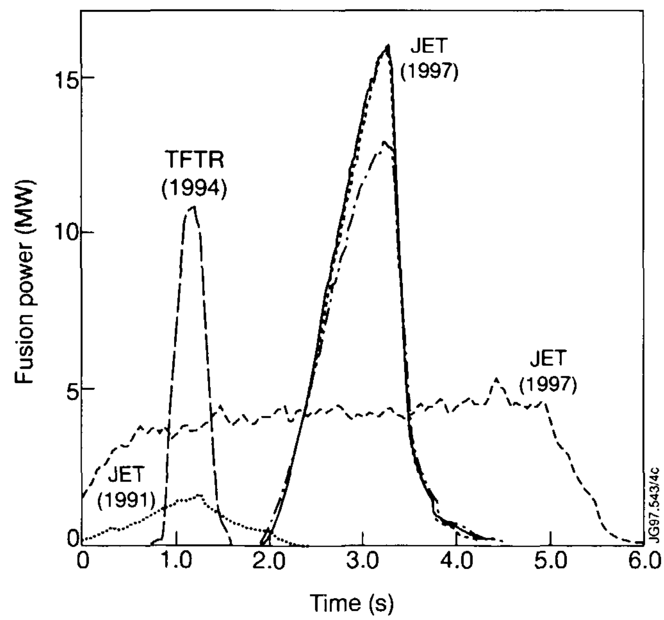

Fig. 6. The development of fusion power generated in different D-T tokamak discharges.

Appendix I

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