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JET Results in D/T Divertor Plasmas

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ABSTRACT

Recent results obtained during the first part of the JET D/T phase are presented. Emphasis is placed in conditions carefully set to be similar to those required in ITER. Wall-loading changeover from 100% deuterium to 90% tritium is achieved in about 20 discharges with 100% tritium injection. H-mode threshold power is found to have inverse isotopic mass dependence whereas there is little mass dependence in global energy confinement time in the ELM-free or ELMy H-mode discharges. Preliminary results are consistent with the gyro-Bohm physics form of global energy dependence. This form has a weak negative mass dependence but a stronger density dependence than the ITERH93-P form. The second harmonic ion-cyclotron resonance heating (ICRH) ITER reference scenario (2 ω_{CT} with and without the addition of a small amount of He^3) has been assessed in JET D/T plasmas. High performance combined heating (NB+ICRH) hot-ion and optimised shear regime experiments could only be carried out with half the installed NB power. Nevertheless, internal confinement barriers in optimised shear regime have also been demonstrated in D/T plasmas.

1. INTRODUCTION

Operation of JET to the end of 1999 has two main physics objectives: (i) to make direct contributions to the development and demonstration of a mature divertor concept for ITER and (ii) to strengthen the physics basis of ITER by carrying out ITER relevant experiments in D/T plasmas. JET has considerable flexibility that allows many different modes of operation. It can match ITER geometry and dimensionless parameters (except the normalised Larmor radius, ρ^*) and can study the effect of large variations around the ITER values.

The very first tritium experiments, the Preliminary Tritium Experiments (PTE), were carried out at JET in 1991 [7]. The programme was limited to two pulses in the hot ion regime. The tritium concentration was 11% far away from the optimum 50% mixture. We report here the results of some 70 full D/T pulses [1, 2] which include development of high performance regimes and physics experiments with various mixtures of deuterium and tritium (the DTE1 series of experiments) in conditions carefully chosen to be similar to those required in ITER. The two main aims of DTE1 are (i) to study physics of ITER modes of operation and (ii) to achieve high D/T fusion performance so that α-particle physics can be studied. The programme on ITER modes of operation includes a study of the effect of isotopic mass dependence on the energy transport and H-mode threshold power [3]. This will contribute to a more accurate assessment of the ignition margin and additional heating requirements for ITER. Furthermore, the ITER reference Ion Cyclotron Resonance Heating (ICRH) scenarios in D/T plasmas [2] and performance in ITER relevant quasi steady-state ELMy H-modes will be studied. The high performance regimes include the hot-ion ELM-free H-modes [4] and the optimised shear mode of operation

[5]. As a by-product, these regimes will allow the study of energetic particle physics including the α-particle physics and the instability threshold of Toroidal Alfvén Eigenmodes (TAEs) and their effect on plasma.

A budget of about 2 x 10^{20} total number of neutrons has been set for the DTE1 experiments to limit the subsequent activation of the vessel. This constrains the number of D/T shots consuming a large number of neutrons (long, high performance shots). The 20 g of tritium available on-site permits a significant number of D/T experiments during a period typically lasting a few days. The exhaust gases are entirely collected and reprocessed by the on-site closed-circuit Active Gas Handling System (AGHS) for subsequent experiments. The reprocessing takes four days. During the entire DTE 1 campaign, it is expected that about 80 g of tritium will be injected into about 200 full D/T pulses.

The main parameters of the JET tokamak are below (Table 1) which include the NB injection power of 21 MW (from 2 beam boxes with one box capable of injecting tritium), 17 MW of ICRH power and 7 MW of Lower Hybrid Current Drive power. JET has a single null (bottom X-point) divertor configuration. With a view to contributing to the design of a viable ITER divertor concept, JET has successively installed and tested two different divertors (presently operating with MkIIA). A third version (MkIIGB), the so-called Gas Box, will be installed after the DTE1 using remote handling systems available at JET. These three successive divertor configurations feature increasing ‘closure’, ie the fraction of recycled neutrals escaping from the divertor region is increasingly smaller [6].

Table 1. JET Parameters for Mk II D/T Divertor Experiments

Parameter	
Plasma minor radius, a (m)	0.95
Plasma half height, b (m)	1.75
Plasma major radius, geometrical centre, R_o (m)	2.85
Plasma volume (m^3)	85
Plasma aspect ration, R_o/A	3.0
Plasma elongation, b/a	1.85
Toroidal magnetic field (at R_o), B_{TO}(T)	3.8
Flat top pulse length, t(s)	10 to 25
Plasma current, I_p (MA)	≤ 6.0
Transformer flux, f (Wb)	42
Neutral Beam power at 80 keV and 140 keV, (MW)	21
Ion Cyclotron power at 25 to 55 MHz, (MW)	17
Lower Hybrid power at 3.7 GHz, (MW)	7

The technical installation for DTE1 and achievements are discussed in Section 2, followed by energetic particle effects and observation of TAEs (Section 3). ITER physics experiments including the isotopic effects on transport and H-mode power threshold are given in Section 4 which also contains results of the ITER-relevant ICRH scenarios. In Section 5, D/D and D/T fusion experiments including the hot-ion H-modes and optimised shear regimes are presented. Discussion and conclusions of the paper are contained in Section 6.

2. TECHNICAL INSTALLATIONS FOR DTE1

The programme of JET operation in D/T plasmas requires the use of tritium in close circuit and application of remote maintenance and repair techniques. These facilities are all integrated, into the machine conception, design, construction and operation.

During the technical preparations for DTE1, existing systems were brought to tritium readiness. In particular, the AGHS is fully operational on-site as a closed-circuit gas supply, retrieval and reprocessing plant (see Fig. 1). Deuterium and tritium are supplied to the gas introduction and neutral beam (NB) systems for injection into the torus which is pumped in continuous operation.

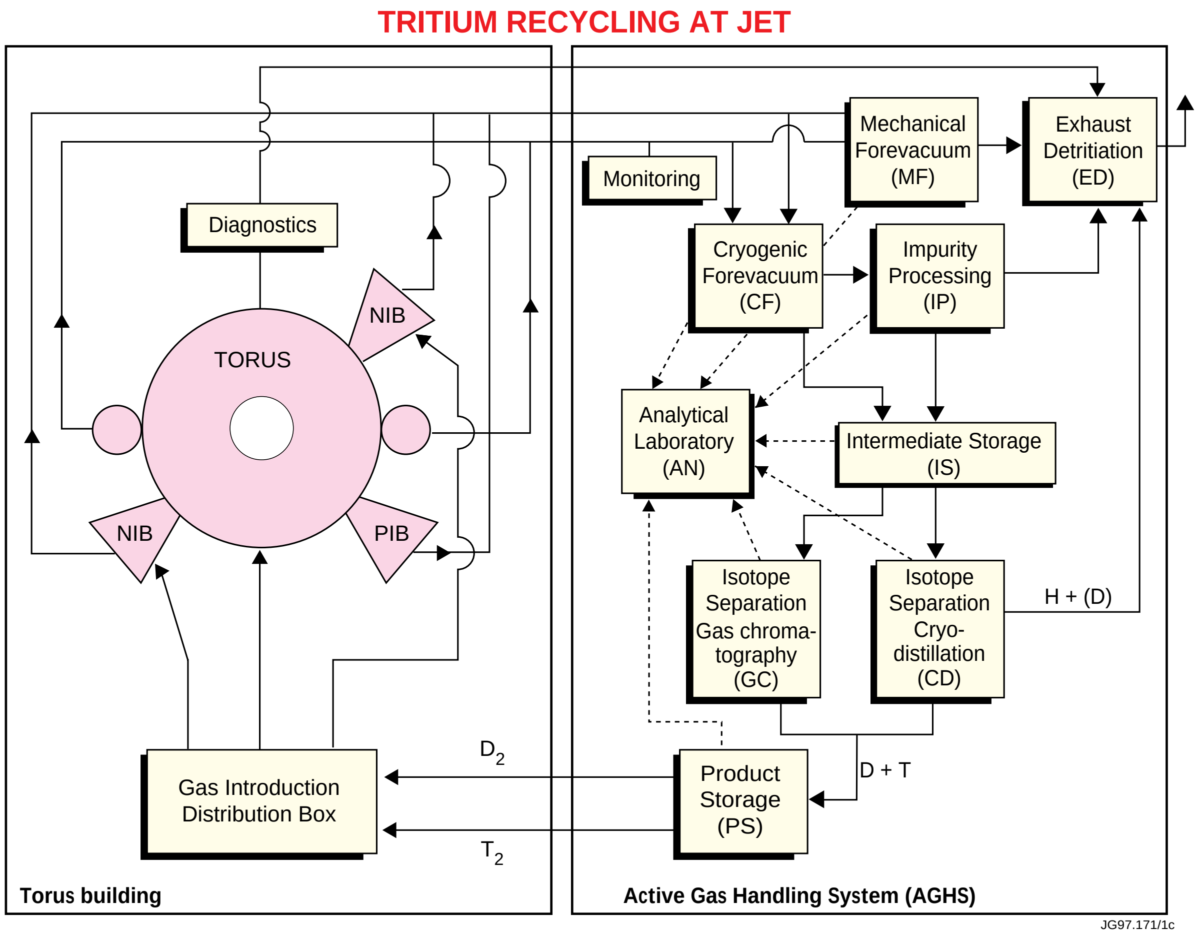

Fig. 1: Block diagram showing the essential elements of tritium reprocessing and gas introduction plant at JET.

A critical issue for both the high performance and ITER physics studies is the tritium concentration which can be maintained in the plasma. In the first part of the DTE1 campaign, tritium concentrations that have been achieved are shown in Fig. 2 where each point represents a D/T plasma shot. These experiments indicate that with 100% tritium fuelling of RF-heated discharges, it is relatively easy to obtain 80 - 90% tritium concentration in the plasma. Predictions of the multi-reservoir model [8] developed for the PTE are in good agreement with measured tritium concentrations in DTE1.

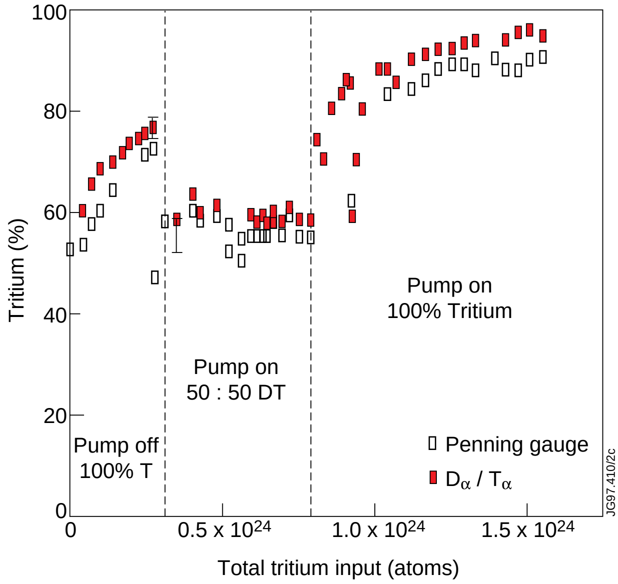

Fig. 2: Tritium concentration achieved, as measured by two edge diagnostics, (spectroscopy and Penning gauge) by fuelling successive discharges by 100% tritium with and without cryo-pumping as indicated. Similar concentrations were also measured in the core plasma.

During DTE1, one of the two NB boxes is operated with up to 100% tritium delivering 11 MW at 160 kV. During NB commissioning, 9.5 MW of tritium NB (at 155 kV) was injected for up to 5 s into a deuterium plasma producing 2.6 MW of peak fusion power and 7 MJ of fusion energy [1]. The neutron diagnostics was also calibrated during these tritium NB commissioning experiments.

Several enhancements and/or new diagnostics have been installed on JET for the DTE1. One such example is a system consisting of thin foil Faraday collectors to act as a lost α-particle diagnostic for high-yield D/T fusion plasmas [9]. In view of the limited neutron budget, automatic feedback real-time control systems have been implemented so that, if the desired performance is not achieved at the expected time during the discharge, the plasma shot is terminated with a soft landing saving neutrons. Also, certain given plasma parameters can be maintained at a programmed level by a system controlling in real time a number of auxilliaries (NB, ICRH, LHCD etc). Using digital techniques, power delivered to the plasma can be precisely controlled using pulse width modulation techniques of the NB or overall management of 16 ICRH amplifiers.

3. ENERGETIC PARTICLE EFFECTS

Alpha-particles will be the dominant plasma heating source for ITER. Therefore, the confinement of the energetic α-population and its impact on plasma performance are important issues. Moreover, any uncontrolled loss of this population either by collective instabilities such as Toroidal Alfvén Eigenmodes (TAE) or ripple induced loss could have adverse effects on the first wall.

3.1. TAE-mode excitation

Alfvén eigenmodes can be excited (or damped) by energetic particles such as fusion born α-particles, injected NB ions or fast-ions accelerated by ICRH. Plasma provides several damping mechanisms such as direct Landau damping on ions or electrons or mode conversion to Kinetic Alfvén Wave. Therefore, to assess the linear stability of these modes, both the driving and damping effects must be evaluated. External excitation of these modes in JET has been done in two ways: (i) by saddle-coils mounted inside the vacuum vessel and (ii) by beat waves generated by two ICRH antennas run master-slave at slightly different frequencies (100 - 200 kHz in 50 MHz). A coherent detection of AE modes in magnetic probes, Electron Cyclotron Emission (ECE) or reflectometry diagnostics provides a means to determine the wave characteristics (frequency, mode number and damping) [10].

The threshold power for ICRH produced fast-ion driven TAEs is determined from the magnetic fluctuations data as shown in Fig. 3. In this shot, ICRH power was increased in two steps as shown. During the first step (3 MW), no TAE modes could be identified. However, at the higher power level (7 MW), a clear observation of TAEs (n = 5 11) in the expected frequency range is made. The different toroidal wave numbers (n) are clearly separated by the Döppler effect resulting from plasma rotation. When this effect is subtracted, the modes follow accurately the Alfvén frequency dependence. Note also that a fine structure develops at the higher power. This remains to be explained.

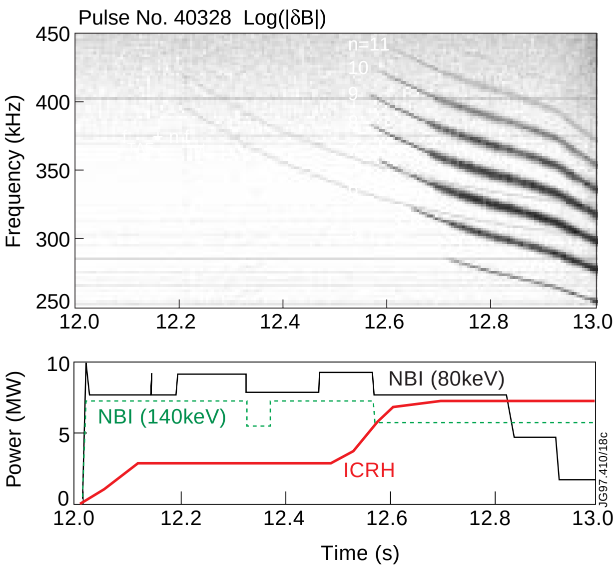

Fig. 3: Observation of power threshold of TAE modes excited by ICRH produced fast ions in high performance plasmas. ICRH power was increased in two steps as shown. The bands seen around each mode at higher power contains a fine structure which is not yet well understood.

Fig. 4 shows a spectrogram of magnetic fluctuations in a shot with 7.5 MW of ICRH and 11 MW of D-NBI into a 55:45 D/T plasma. TAE modes produced by ICRH led fast ions can be seen as usual in the expected 150 - 200 kHz frequency range. However, in this D/T shot, slowing down of a-particles are expected to drive TAE modes during the time interval 6.5 to 6.75 around 200 kHz. Calculated values of damping of TAEs and the α-drive is shown in the lower box. During the period, when the α-drive is calculated to be larger than damping, TAE modes can be seen in the after glow of beam heating. Six different modes can beidentified (toroidal mode number n = 5 - 10) and their frequency tending

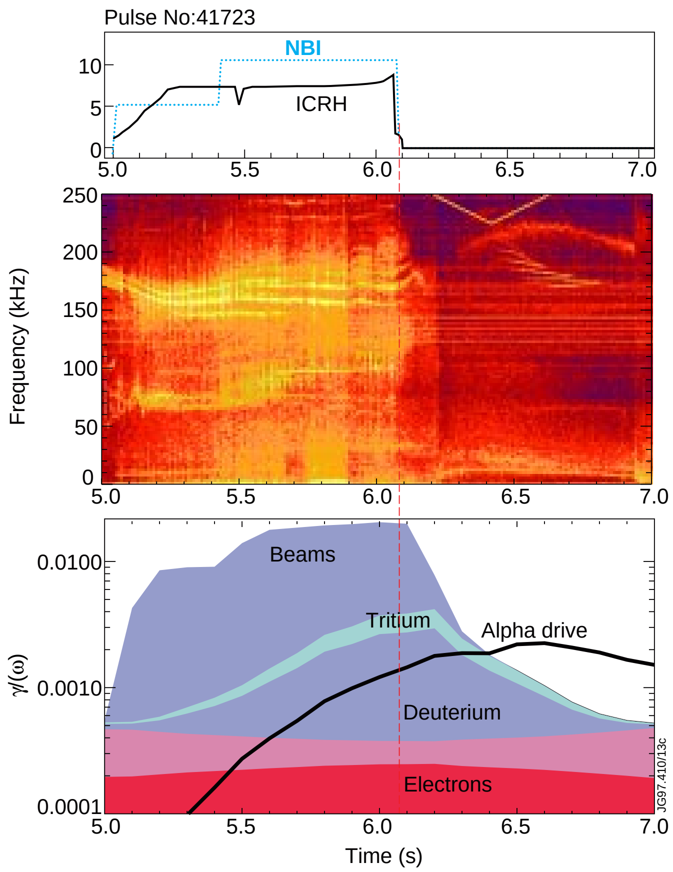

Fig. 4: Observation of TAE modes in the after glow of D-beam injected into 55 : 45 D/T plasma. Six discrete modes can be seen in the expected frequency range of 200 kHz in the 6.5 and 7 s time interval when the calculated damping rates becomes smaller than the α-drive. These modes could be driven by fusion born α-particles or by residual ICRH ions.

to merge as the toroidal rotation of the plasma due to beams comes to an end. However, the calculations do not take into account the drive of slowed down ICRH ions and additional experiments without (or with low power) ICRH are required to establish unambiguously the existence of the alpha drive.

3.2. Confinement of fast particles

Ion cyclotron minority heating has been used in JET earlier to produce 1 - 1.5 MeV He^3-ions with an energy content up to 2 - 2.5 MJ corresponding to almost 40 - 50% of the total plasma energy in such discharges [11]. The scaling of superthermal ion energy showed that the interaction of fast ions with the plasma is dominated by classical collisions. Further studies have been carried out of central ICRF heating at the third harmonic of deuterium in JET. The initial damping of the fast wave on deuterium is weak but a high energy tail soon develops thanks to the confinement of this energetic tail even at a plasma current of 2 MA. The thermal ion temperature was low (2.5 - 3 keV), but the D/D reaction rate in these discharges due to the energetic tail reached 1.8 x 10^{16}/s where the fast ion tail temperature of a population was measured to be about 0.8 - 1 MeV. These results were well reproduced by PION code calculations giving a confidence in the understanding of production and confinement of these energetic tails [12].

3.3. Ripple experiments

The effect of high toroidal field ripple on plasma behaviour and fast particle losses was studied in the previous experimental campaign of JET. The ripple was varied in the ITER relevant range of 0.1 to 2%. Ripple induced losses of thermal and high energy particles (125 keV NB ions and 1 MeV tritons) were less than 1% and were consistent with theoretical estimates. However, the observed losses of particles of intermediate energy (thermal to a few tens of keV) were higher than predicted [13]. Furthermore a strong reduction of plasma rotation was observed at high ripple.

4. ITER PHYSICS EXPERIMENTS IN D/T

4.1. H-Mode threshold power

A database on H-mode threshold power has been established using ICRH (three different ion cyclotron resonance scenarios) and NBI (80 keV deuterium) heating at magnetic fields between 1 and 3.8 T. Plasma conditions were set to match precisely the ITER shape and q. Auxiliary heating power is ramped over 3 s and D_α-signal is monitored for the appearance of threshold ELMs. The slow ramp permits an accurate determination of threshold power. Time traces in Fig. 5 illustrate three ICRH shots in three different D/T gas mixtures 100:0, 50:50 and 10:90. These plasmas at 2.6 T/2.6 MAwere heated with H-minority ICRH at 42 MHz. As indicated in the figure, H-modes occur at lower power levels when tritium concentration in the plasma increases from 0 to 90%.

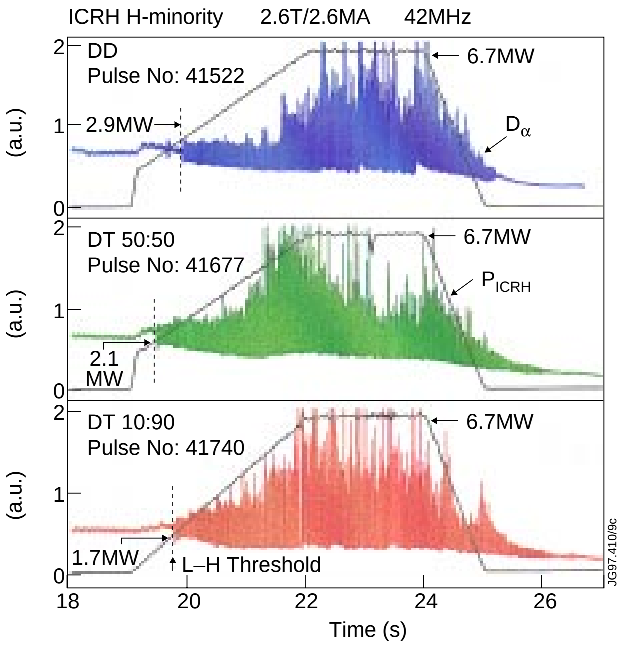

Fig. 5: Observation of mass dependence of H-mode thresholds with D_α-signal in three similar shots with three different mixtures of D/T concentrations as indicated. These plasmas were heated by ICRH with a slow power ramp up as shown.

In Fig. 6, we plot the loss power at H-mode threshold as a function of the Montreal [14] H-mode scaling for a range of plasma current and magnetic field in D/D and two D/T mixtures as indicated in the figure. The 45°-line is drawn for the Montreal scaling which does not include a mass dependence. Note that the D/T data points lie clearly below the line while the deuterium shots are close to it. In order to include the isotopic mass dependence in the scaling, a regression analysis has been done using the same power exponents as in [14] but with an isotopic mass parameter A_{eff} which is weighted by the relative D/T concentration [15]. The threshold power data shows roughly an inverse mass dependence (see scaling expression on the horizontal axis in Fig. 7). Such a dependence had already been found previously in Hydrogen and Deuterium discharges in ASDEX [16]. The radiated power from the bulk is generally 15-20% of the input power. If the radiation from the bulk is included in the loss power (P_{TOT} - $\dot{W}$ - P_{rad}), the strong isotopic effect on the power threshold remains essentially unchanged.

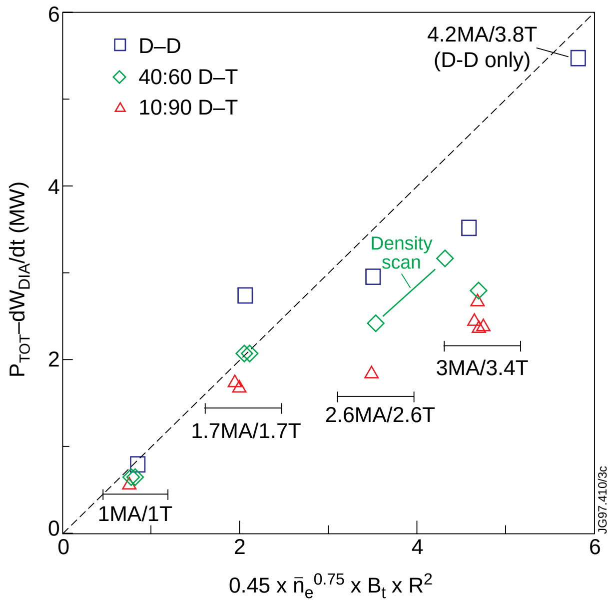

Fig. 6: Plasma loss power at the L-H transition as a function of the Montreal ITER scaling [14] for three different D/T mixtures in JET plasmas as indicated. The D/T data points are below the line suggesting the need of a new scaling including an ion mass parameter.

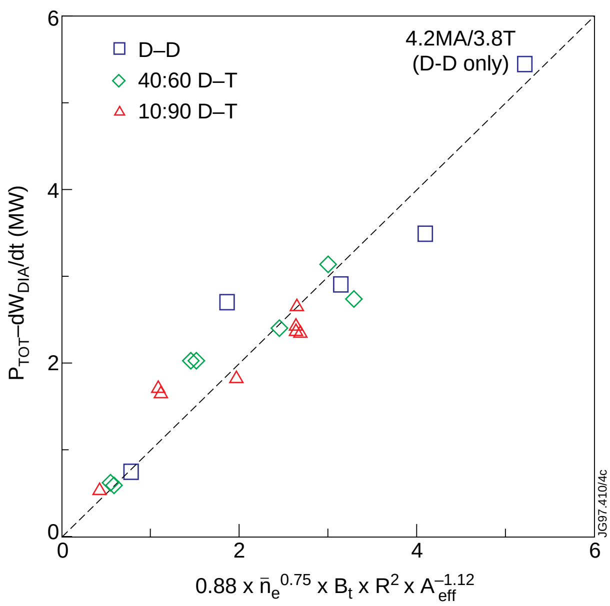

Fig. 7: A new fit to the JET H-mode threshold power data (see Fig. 6) with the same power coefficients as in [14] but with a mass dependence. Approximately, an inverse mass scaling is inferred similar to the one found on ASDEX with Hydrogen and Deuterium.

4.2. Operational window in edge parameters

Plasma edge temperature and density are expected to play a crucial role in the behaviour of H-mode threshold power and characteristics of edge localised modes (ELMs). It is possible to make a characterisation of L-H threshold and types I and III ELMs in the (T_{es} - n_{es})-plane where T_{es} and n_{es} are the edge electron temperature and density respectively. The upper boundary for type I ELMs [17] as shown in Fig. 8 is fitted by the expression $T_{es} = 2.8 \times 10^{29}\ n_{es}^{4/3}$ (eV, m^{-3}). This is interpreted as an edge pressure gradient (ballooning) limit with the scale length varying as $\rho_i^{1/2}$ or $T^{0.25}$, which is representative of a mixed transport model [18]. Data points of type I ELMs obtained with 12 MW of NBI heating in deuterium follow the ballooning limit line (Fig. 8). Similar data obtained in 50:50 and 10:90 D/T mixtures with 8 MW NBI or ICRH are also shown which extend somewhat into the D/D ballooning limit region. The L-H threshold points seen in the lower left hand corner do not seem to be very different in various isotopic D/

T mixtures even though they are obtained at lower power in tritium. With 8 MW of NBI in D/D-plasmas, type I ELMs could not be achieved. In contrast, clear type I ELMs are obtained with D/T mixtures at the same power.

4.3. ρ*-scaling confinement experiments

JET is carrying out dedicated experiments with NBI and ICRH to verify the ITER confinement scaling (ITERH93-P) [19] for ELM-free and ELMy discharges. This study is mainly aimed at determining the mass scaling which is poorly documented in the ITER database and at improving the accuracy of the size scaling. The plasma shape, the safety factor q, in some cases the plasma pressure β and the collisionality ν* are kept constant in these experiments where the isotopic composition and the normalised radius are varied.

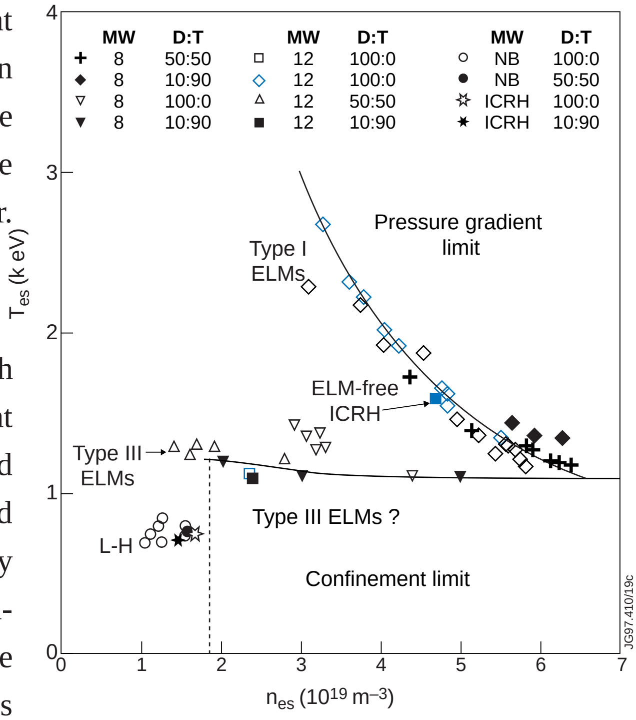

Fig. 8: Operational window in edge electron temperature and edge electron density defined by type I ELMs pressure gradient limit line and confinement degradation with type III ELMs. The 12 MW NBI type I ELMs data points in D/D lie on the above line. The 8 MW data points in D/D do not reach type I ELMs whereas in 50:50 and 10:90 D/T mixtures, type I ELMs are obtained at this power as shown. Despite the power difference at H-mode threshold, edge parameters for the L-H transition are very similar in the three D/T mixtures both for NBI and ICRH.

The magnetic field and plasma current combinations are varied from 1 T/1 MA to 3.5 T/3.5 MA. This study was completed earlier in D/D discharges [3] and at present is being extended to D/T and T discharges to examine the effect of isotopic mass in the scaling. Due to the influence of isotope mass on ELM behaviour (longer ELM-free periods with tritium-rich mixtures), it is not always possible to operate at the same density for the same input power. An illustrative comparison of time traces of two shots of D-NB injection into D and T-plasmas respectively (at 2.6 T/2.6 MA) are shown in Fig. 9 where the density was similar but the power input was about 10% different. The isotope scaling is found to be very weak, namely $\tau_E \propto A^{0\pm0.2}$. This is also found to be the case for similar discharges obtained with ICRH. Thus, the strong mass dependence included in ITERH-93P, ie $A^{0.41}$ does not agree with either the ELM-free or ELMy data.

This disagreement of ITERH-93P scaling with JET most recent data is shown in Fig. 10. Our mass dependence of the ELM-free data shows that $\tau_E \propto A^{0\pm0.2}$ (close to gyro-Bohm) and for the ELMy data $\tau_E \propto A^{0.1\pm0.2}$. In terms of dimensionless parameters, the data is also compatible with a form based on gyro-Bohm physics [20]

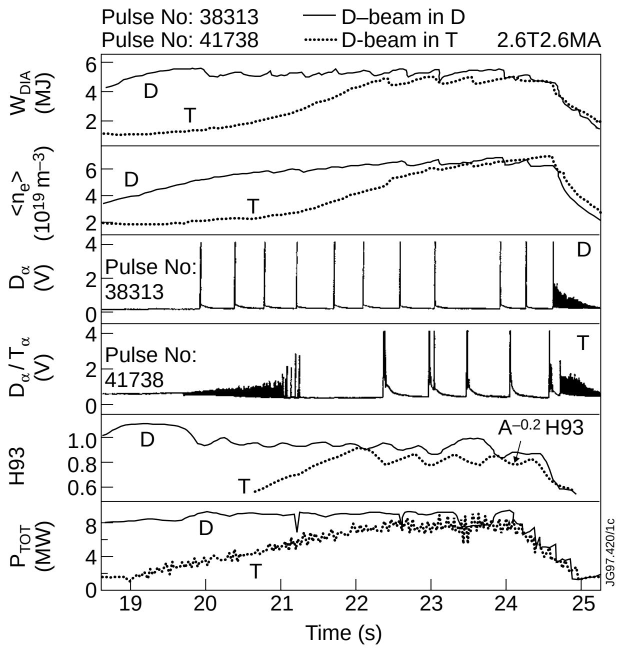

Fig. 9: A comparison of two similar shots, one with D-beam in D plasmas and the other with D-beam in T-plasmas, is shown. Time evolution of diamagnetic stored energy (W_{DIA}), volume averaged density $< ne >$, and D_α/T_α signals are shown. The ITERH93 confinement factor in DD and that multiplied with a $A^{-0.2}$ mass factor in D/T are also shown.

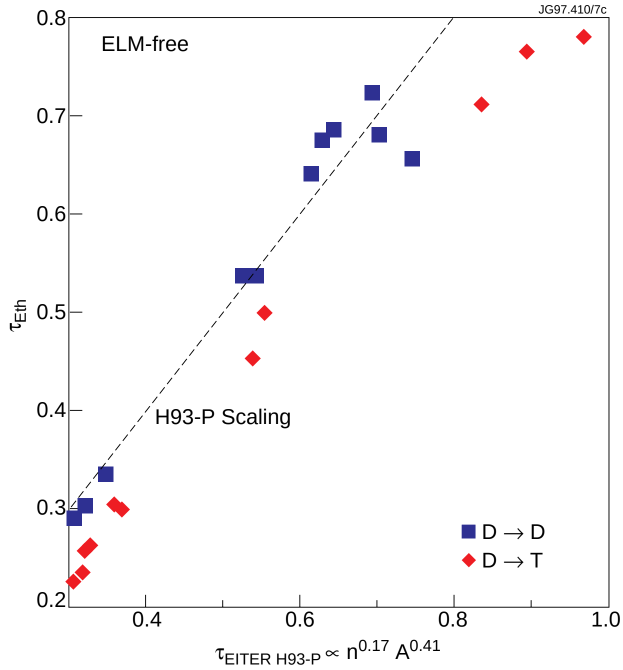

Fig. 10: Normalised thermal confinement time obtained in discharges with D-beam in D plasmas and D-beams in T-plasmas (ELM-free) are plotted against the ITERH93-P confinement scaling. The D/T data does not fit the scaling well. $\tau_{EPHYS} \propto \omega_c^{-1}\ \rho^{*-3}\ \nu^{*-0.3}\ \beta^0 \propto A^{-0.25}$

A comparison of this scaling with JET data is shown in Fig. 11. For ITER conditions, the unfavourable mass scaling in the energy confinement time is expected to be compensated by the better density scaling found. The higher density scaling is embedded in r*.n* where the density exponent in ITERH93-P was +0.17 as compared to +0.31 of the physics form.

New D/T experiments are planned in the near future to consolidate these important preliminary results. Emphasis will be placed in pure tritium plasmas which exhibit the largest isotopic effect.

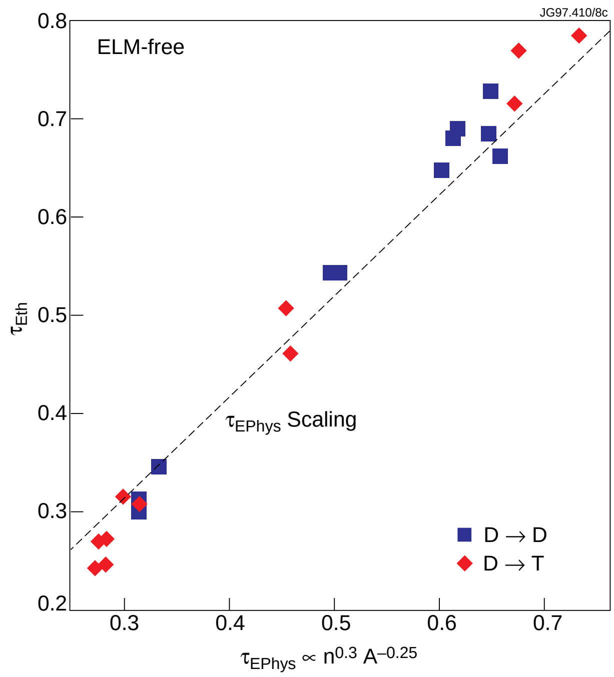

Fig. 11: Normalised thermal confinement time is plotted as a function of gyro-Bohm physics scaling with a weak mass dependence ($A^{-0.25}$). A reasonably good fit with both D/D and D/T data is obtained.

4.4 ICRF Heating Regimes

The JET ICRH system couples power via four antennas each made of four current straps which can be phased independently. Experiments have been performed in D/T plasmas with up to 95% tritium in ITER-relevant ICRH scenarios such as T second harmonic heating ($2\ \omega_{CT}$), D-minority heating in T (ω_{CD}) and He^3 minority in T. All heating schemes achieved H-modes at power thresholds described by the new mass dependent scaling law. The data obtained with the ITER reference scenario ($2\ \omega_{CT}$) is seen to be sensitive to the presence of a small amount of He^3.

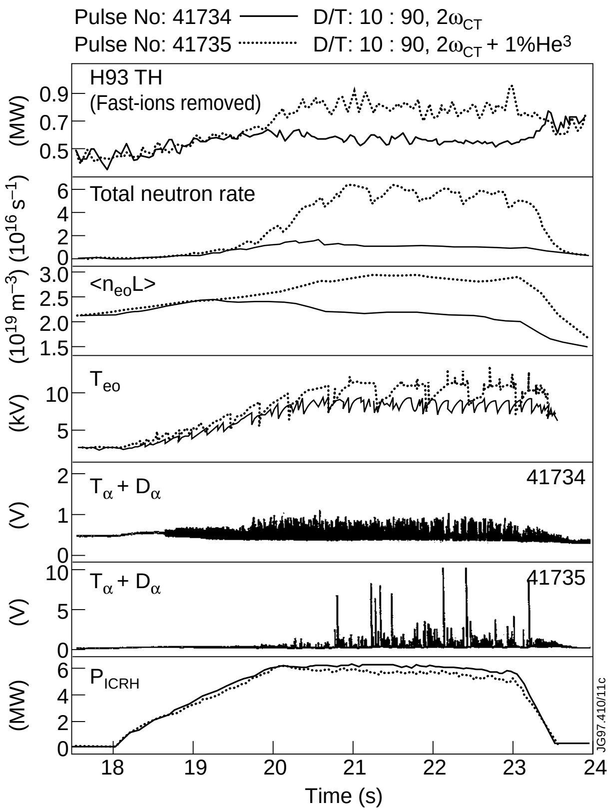

Fig. 12: A comparison of two similar ICRH discharges: one without and the other with He^3 added (about 1%) in the $2\ \omega_{CT}$ heating scheme. Enhanced performance with He^3 is obtained due to more peaked power deposition profile resulting from stronger wave damping and smaller orbits of the resonating ions.

In Fig.12, we show a comparison of $2\ \omega_{CT}$ heating with about 1% He^3 and without any addition in which case about 0.2% is intrinsically contained in the injected tritium due to radioactive decay. The input power traces are very similar in the two cases. The ITERH93-P thermal confinement factor H93TH (fast-ion energy subtracted) is found to be significantly higher with He^3 added. Also, with additional He^3, discrete ELMs with higher amplitude, in addition to the higher frequency ELMs typical of ICRH, are found. The neutron rate, line-averaged plasma density and electron temperature are also shown. From other analysis, it is found that in discharges with He^3, neutrons are of thermal origin. Performance with added He^3 is expected to be better, for in this case power deposition profile is calculated to be more peaked than the $2\ \omega_{CT}$ trace He^3 case. In the latter case where the density was relatively low, it is estimated that about 15 - 20% of the fast ions (tritium and He^3) are lost due to large banana orbits and the effective power available to build up the H-mode thus remained marginal in this case. A PION code [21] simulation of observed D/T neutrons and the fast ion energy in a $2\ \omega_{CT}$ heating discharge with +90° antenna phasing are shown in Figs. 13(a) and (b) respectively. It is found that the simulation agrees well with experimental observations. Significant losses due to large banana orbits are not expected in ITER due to the much lower value of ρ*.

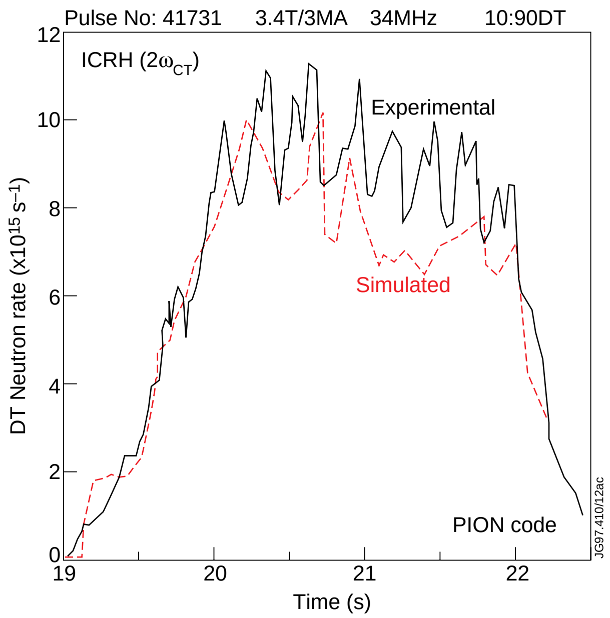

Fig. 13(a): A PION code simulation of observed neutron rate in a 2 ω_{CT} heating scheme with 0.2% trace He^3 that exists in the injected tritium due to radioactive decay. Trace He^3 can absorb about half the power during the early heating phase.

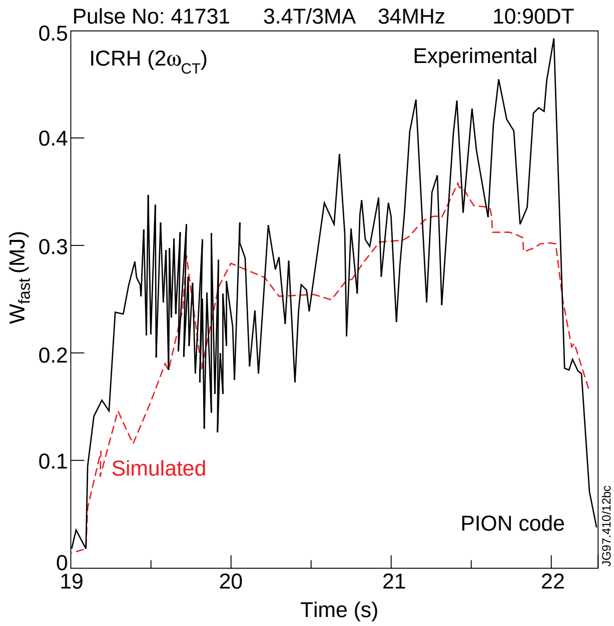

Fig. 13(b): A PION code simulation of observed fast-ion energy in the same shot as in Fig. 13(a).

5. D/D AND D/T FUSION PERFORMANCE EXPERIMENTS

JET is pursuing two scenarios for high fusion performance: (i) Hot-ion H-mode and (ii) Optimised shear mode.

5.1. Hot-ion H-mode and Optimised Shear Regimes

In the hot-ion H-mode operation, a low electron density target plasma with high triangularity (0.25) and high flux expansion in the divertor is heated with strong NB beam injection heating. A good confinement barrier is formed at the edge. More recently, the performance of hot-ion H-modes have been improved considerably by an addition of ICRH power at a level of about 10 - 20% of NB [22].

In JET, optimised shear experiments are carried out at the end of the current rise phase of the discharge where advantage is taken of the natural delay in the current diffusion to the plasma centre as the current is ramped. The current diffusion is further delayed by electron heating, first using LHCD then ICRH at low powers (~ 1 MW). The target plasma has q > 1 everywhere. High power NB and ICRH are injected at optimised times in the low target density plasmas. In such discharges, an internal transport barrier is established early in the discharge which expands outwards to approximately 2/3 of the plasma radius. These discharges have a very peaked plasma temperature and density profile where the central plasma pressure has reached up to 3 bar [1] and the ion temperature up to 34 keV.

Fig. 14 shows a comparison of the JET database [23] of hot-ion H-mode and optimised shear regimes both with and without ICRH. In this figure, the maximum neutron yield in D/D discharges is plotted as a function of plasma stored energy. It is seen that the addition of ICRH to the maximum available NB power extends the neutron yield significantly. The maximum neutron yield is slightly higher in the optimised shear regime and achieved at lower values of stored energy than in the case of hot-ion H-modes. This is due to the fact that optimised shear regimes feature more peaked plasma profiles than hot-ion H-modes. It has not been possible so far to combine the quality of edge confinement barriers of the hot ion H-mode with the internal confinement barrier of the optimised shear.

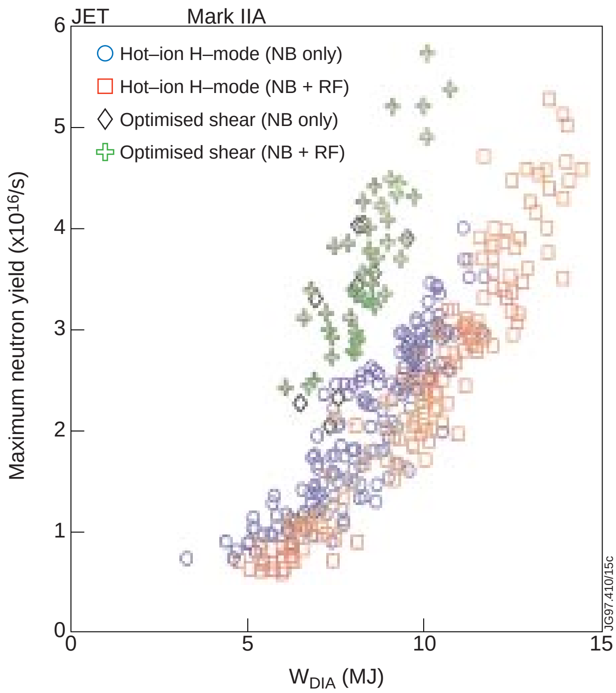

Fig. 14: Maximum neutron yield obtained in D/D plasmas is plotted as a function of the diamagnetic stored energy in hot-ion H-modes and optimised shear regimes with NB alone and combined NB and ICRH heating. The difference in the two regimes is due to more peaked pressure profiles in the optimised shear case. Neutron rates are significantly enhanced when NB is supplemented by ICRH.

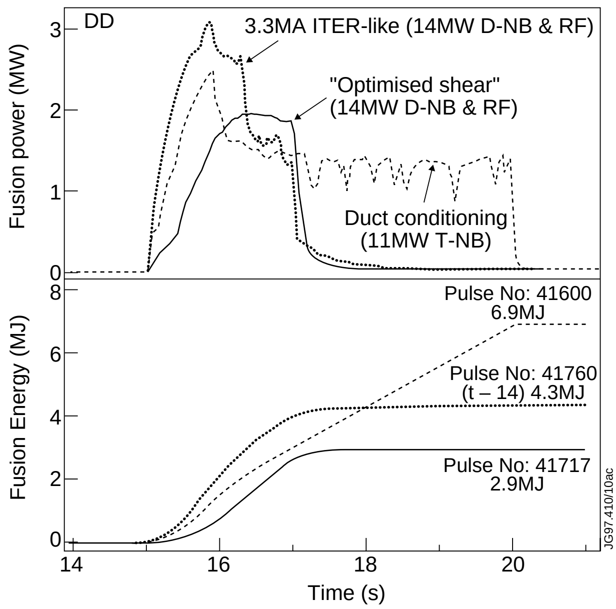

Fig. 15: A comparison of fusion power and energy in the three best D/T shots in the first part of the DTE1 campaign where T-beam power was unavailable for optimised shear and ITER-like plasmas. D-beams were not used in the duct conditioning shot.

5.2. Present status in fusion power and energy

In the first part of the D/T campaign of JET, due to the unavailability of tritium NB, full-scale D/T experiments with beams have not yet been carried out. Nevertheless, a significant part of the D/T programme has seen wall change-over from D to T, neutron calibration, ITER ρ*-scaling experiments, some trace tritium and ICRH experiments.

Fig. 15 compares fusion performance in the best shots of three regimes albeit with half the installed beam power. An NB duct conditioning shot in which 11 MW T-beams were injected into a deuterium plasma produced 2.5 MW of peak fusion power and 6.9 MJ of fusion energy. The best optimised shear discharge with 10.5 MW of D-NBI and 8 MW of ICRH produced 2 MW of fusion power and 2.9 MJ of fusion energy. However, the highest performance in terms of peak fusion power (3 MW) was achieved in a 3.3 MA discharge with ITER shape and q with 11 MW of D-NB and 4 MW of ICRH. In all cases, there has been no attempt to optimise the fusion yield by controlling the D/T mixtures.

6. DISCUSSION AND CONCLUSIONS

We have summarised in this article the results obtained during the initial phase of the present JET programme using tritium. This initial phase produced a total of $2\ 10^{19}$ neutrons (eg 60 MJ of fusion power) in 71 discharges. This corresponds to 10% of the neutron budget foreseen in the on-going DTE1 experimental campaign. A number of important results have already emerged.

i) A mass scaling will have to be introduced in the presently accepted ITER scaling law for the H-mode power threshold. The present data is compatible with an inverse mass dependence. This result reduces by more than 25% the power requirement to reach the high confinement regime in ITER and widens quite favourably the ITER route to ignition.

ii) The preliminary results on energy confinement show a very weak mass scaling for a fixed density. This appears different from the TFTR supershot results [24] which show a significantly favourable scaling with mass. The reason for this difference is likely to be that JET (as ITER) operates in the ELM-free or H-mode regimes which exhibit gyro-Bohm confinement physics where the turbulence size scales like the gyro radius. The initial JET results are also compatible with a scaling law for energy confinement derived from gyro-Bohm physics constraints (eg $\tau_E \propto \omega_c^{-1}\beta^{\circ}\ \rho^{*-3}\ \nu^{-0.3} \propto A^{-0.25}$). This implies a stronger density dependence which, when extrapolated to ITER conditions, compensates approximately the unfavourable mass scaling.

iii) Internal confinement barriers could be produced in D/T plasmas using optimised current profiles despite the fact that only half the NBI power was available.

iv) The ICRH reference scenario selected by ITER has good heating efficiency (H93-P = 0.9, fast ions removed) and maintains moderate ELM relaxations typical of ICRH. The results show that modelling should include trace He^3 that is generally present in tritium due to radioactive decay. Adding a small amount of He^3 improves the localisation of power deposition.

The 90% neutron budget which remains available for experiments to be performed in coming weeks will be used to consolidate these results which are essential for predicting accurately ITER performance and to demonstrate fusion power production above the 10 MW level in a divertor tokamak with ITER geometry.

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