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Overview of the JET Deuterium-Tritium Fusion Experiments

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ABSTRACT

After more than a decade of preparations, the JET device has in 2021 proceeded to a series of high power D-T experiments culminating in pulses producing ~10 MW of fusion power for a duration of 5 seconds. This power exceeds by a factor 2.5 the fusion power produced in the best similar 5 second pulse in the previous D-T experimental campaign in 1997. The most important achievements of the recent JET experiments however reside in the wealth of new physics, most of which remains to be analysed, harvested in hitherto unexperienced plasma conditions. The DT experiments were preceded by extensive experiments in deuterium plasmas for the purpose of developing plasma scenarios foreseen for ITER and completed by experiments in hydrogen, tritium and isotope mixtures for investigating the effect of isotope composition on plasma transport and confinement. In high confinement mode (H-mode) the energy confinement time was seen to scale favourably with isotope mass. Energetic ions, such as fusion-produced alpha particles, were also found to have a favourable effect on transport by reducing ion temperature gradient (ITG) mode turbulence. Important aspects relevant to the operation of a future fusion reactor were also addressed, such as tritium retention in the plasma facing components and post-experiment tritium clean-up. These JET results are invaluable for helping to deal with the challenges faced by the development of fusion energy.

INTRODUCTION

Construction of the JET tokamak as a Joint Undertaking of EURATOM member states started in 1978 at the site of the Culham Science Centre, Oxfordshire, UK, and the first plasma was produced in June 1983 [1]. JET was one of the two tokamaks designed to work with tritium (the other being TFTR [2] in the US, which operated until 1997). After a series of upgrades, including the installation of a divertor with in-vessel divertor coils, a first major deuterium-tritium experimental campaign (DTE1) was conducted in 1997, resulting in plasmas transiently producing up to 16 MW for fusion power for 25 MW of input power from neutral beam injection (NBI) and ion cyclotron heating (ICRH), i.e. the fusion gain was $Q_{fus}=P_{fus}/P_{heat}\approx0.6$, where P_{fus} is the power liberated by the D+T $\rightarrow$ ^{4}He (3.5 MeV)+n (14.1 MeV), P_{heat} being the power delivered to the plasma by external heating systems [3]. From 2000 to 2015 JET was operated under the European Fusion Development Agreement [4] with a vast participation of European research institutions. During this period, major upgrades enabling the performance of a second major DT experiment (DTE2) were initiated and completed [5], in particular the replacement of the carbon plasma facing components by tungsten and beryllium components [6], an upgraded NBI system [7] and numerous new or improved diagnostics [8]. From 2015 JET has operated under the Eurofusion Consortium [9], preparing for and performing the DTE2 experiments in 2021.

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DTE1: THE FIRST MAJOR JET DT EXPERIMENT IN JET

Fig.1 shows the configuration of the JET device, as built in 1983. The full NBI power became available in 1989. In this configuration, which didn't yet include a material divertor, JET was able to produce plasma currents up to 7 MA with the last close surface contacting the plasma facing components (PFC) on the vessel walls [1]. NBI power from both NBI systems (~20 MW) became available in 1989.

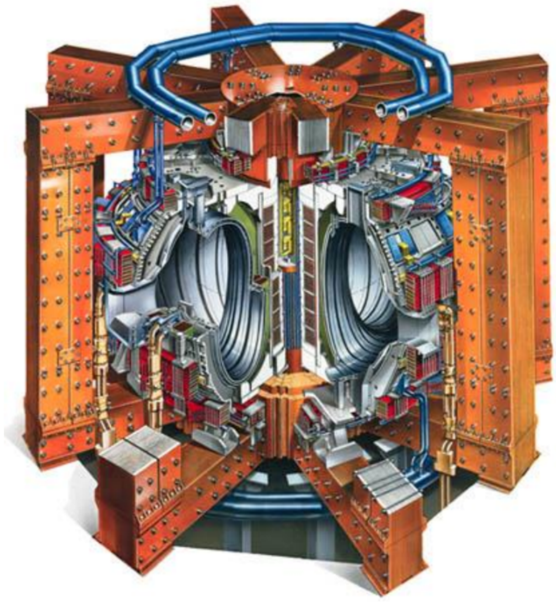

Fig.1. The JET device (1983)

A major paradigm change in the fusion community occurred following the discovery of an enhanced confinement mode (H-mode) at the ASDEX tokamak [10], obtained above a threshold of input power and enabled by the use of a magnetic divertor with an X-point configuration, such as to avoid a contact of the plasma last closed flux magnetic surface (LCFS, effectively the plasma edge) with the vessel wall or limiters. JET was able to produce with an X-point configuration with the then available poloidal magnetic coils and achieved so-called hot ion H-modes with central ion temperatures $T_i(0)$ exceeding 20 keV and ion temperatures within 10 cm of the LCFS up to 5 keV [11]. The H-mode is characterised by an external transport barrier, referred to as a "pedestal", which at the time could not be resolved due to the low spatial resolution of the available diagnostics (~10cm). Recent measurements with a High Resolution Thomson Scattering System (HRTS) in JET have shown that the pedestal, which is the cause of the enhanced global thermal energy confinement time $\tau_E=W_{th}/P_{heat}$, (W_{th} is the total plasma thermal energy) in H-mode, is characterised by sharp temperature and density gradients within ~2cm of the LCFS [12]. Following a short preliminary DT campaign in 1991 with up to 11% tritium [13], JET was retrofitted with a cryo-pumped divertor and 4 divertor coils at the bottom of the vessel [14]. The divertor reduced the available volume for the plasma to ~80m^3, limiting the plasma current to a maximum of 5MA. The divertor version used for DTE1 is sketched in fig.2.

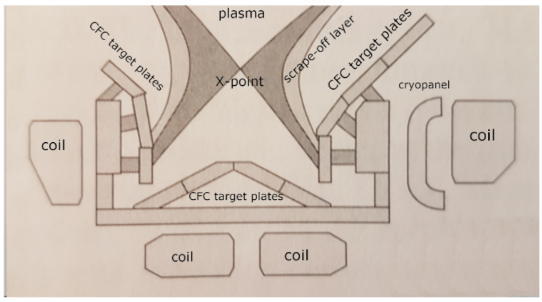

Fig.2 Schematic of the JET divertor as used for the DTE1 campaign (1997)

Several operating modes were developed in deuterium plasmas before the DTE1 campaign, including ELM-free hot-ion H-modes, plasmas with internal transport barriers and H-mode pules with regular Edge Localised Modes (ELMs). The latter allowed achieving stationary plasmas for the entire duration of the heating pulse (several seconds). By contrast, hot ion modes and internal transport barriers had higher peak performance, but the high performance phases were short-lived (~1-2 s). Fig.3 shows the time evolution of the DT fusion powers of the best performing plasmas in DTE1 (1997), as well as the best performing TFTR pulse in 1994 [1,3,15]. An example of a DT plasma from the preliminary DT experiment (1991), is also shown. The fusion power was inferred from the neutron rate measured by calibrated fission chambers [16]. The hot ion mode (labelled ELM free H-mode), after transiently achieving a peak fusion power of 16 MW, suffered an MHD event and an influx of carbon impurities [17]. By contrast, the ELMy H-mode offered a steady fusion power of ~4.5MW for nearly 5 seconds. Despite the lower Q_{fus}=0.2, it is an acceptable operating scenario for ITER and DEMO, now referred to as the baseline scenario [18]. The baseline scenario in ITER is predicted to achieve Q_{fus}=10.

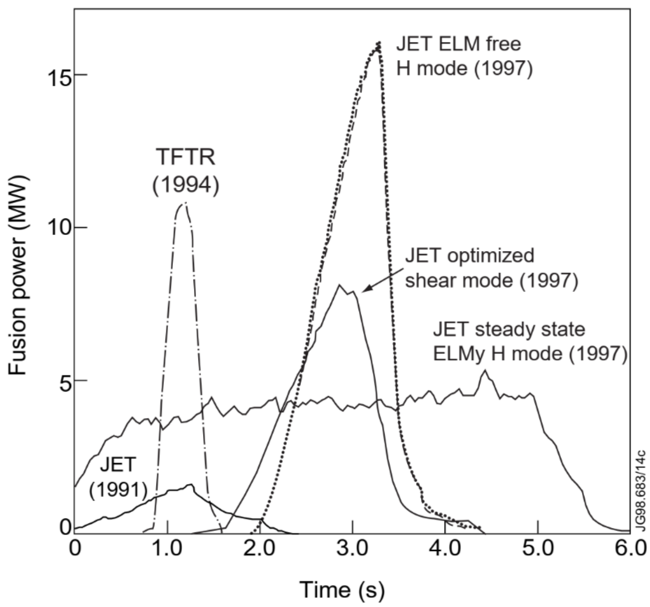

Fig.3 Fusion power evolution in JET plasmas (1991 and 1997) and TFTR (1994).

A major highlight of the DTE1 campaign was the identification of plasma electron heating by fusion alphas in a series of pulses where the fusion power was scanned by varying the D/T density ratio while keeping the total density and the heating power constant [19,20]. While there is no doubt that confined fusion alphas will heat the plasma as they slow down and will have to provide near 100% of the heating of a reactor, the observation of alpha heating can be seen as the holy grail of fusion research. High energy ions like fusion alphas almost exclusively heat the electrons by collisional energy transfer [21]. The central electron temperature increased from 10.3 keV to 12.2 keV when the alpha power was scanned from 0 to 1.5 MW. This experiment produced the plasma with the highest Q_{fus}=0.65 obtained in DTE1 with a heating power of only 10 MW. Doubts about the initial interpretation that the observed electron temperature increase was due solely to alpha heating were later brought forward, as the ion temperature increased even more in the presence of alpha particles, by some from ~12 to 17 keV [20, 22]. The explanation provided in [22], supported by gyrokinetic turbulence modelling [23], was that the high energy alphas were responsible for a reduction in ion heat transport by partly stabilising in temperature gradient modes (ITG). This work triggered a still ongoing paradigm change about the role of

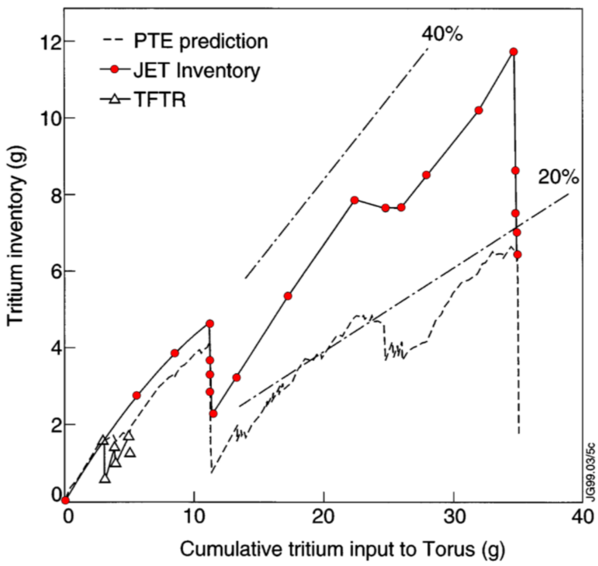

Fig.4 Evolution of the vessel tritium inventory vs injected tritium in DTE1. The sharp drops at 10g and 35g of cumulatively injected tritium are due to tritium recovery procedures [28].

energetic ions in fusion plasmas, which are now seen as having a mostly beneficial effect on ion heat transport [24-27].

Fig.4 shows the evolution of the tritium inventory in the JET vessel during DTE1 [28]. As seen in the figure, a fraction of ~40% of the tritium introduced into the vessel remained trapped in the form of carbon-tritium composite deposits, mostly in the divertor area [29]. Following the DTE1 campaign a series of clean-up methods were applied, which included ICRH, vessel baking and venting, resulting in the eventual recovery of all but 3.7 g of the 35 g of cumulatively injected tritium [28]. This, for safety and economic reasons, unacceptably high tritium retention rate by carbon PFC's was at the origin of the abandonment of carbon as first wall material for ITER and the motivation for the ITER-like wall project under EFDA [6].

PREPARATIONS FOR DTE2 AND ASSOCIATED EXPERIMENTS

The main aim of the JET programme from year 2000 was to engineer and experimentally test an ITER-like metal plasma facing wall and power handling divertor components and develop strategies to mitigate the potentially deleterious effects of plasma contamination by high Z impurities [5]. The geometry was optimised and the surfaces were in part made from bulk tungsten and in part by a tungsten coating applied to existing carbon tiles [6]. The main vessel limiters were made of beryllium and other surfaces in the main vessel were coated with beryllium by beryllium evaporation. The rationale for beryllium in areas not exposed to high power loads was that beryllium in the plasma, due its low nuclear charge, unlike heavier metals, contributes negligibly to plasma radiation losses.

The ILW reduced hydrogenic retention by an order of magnitude [30], validating the metal wall concept and allowed initiating the planning for the second major DT campaign in support of ITER operation [31,32]. The NBI systems were upgraded to operate at a total power of up to 32.5 MW with injection energies of up to 120 keV [7]. Together with the ICRH systems, the nominal installed heating power was ~39 MW. Numerous new or improved diagnostics were installed, including the fore-mentioned HRTS, a higher solution edge charge exchange system, several neutron spectrometers and gamma ray spectrometers [8].

Initial operation (in deuterium), however, showed that plasma confinement, due to a reduction of pedestal pressure, was lower in JET with the ILW (JET-ILW) than with the previous carbon PFC's (JET-C) [33,34]. Nitrogen injection restored confinement in some cases [33], but is not acceptable for DT operation, as it would be trapped in the uranium beds used for storing recycled tritium in the tritium plant. The role of light impurities in plasma transport is not well understood. Another difficulty was the propensity of the plasma to accumulate heavy impurities in the plasma centre. Scenario developers were able to mitigate this using core ICRH, gas puffing and injection of frozen deuterium pellets with shallow (~10cm) penetration into the plasma [35]. This development eventually led to baseline plasma confinement in deuterium comparable to that with the carbon wall [35]. Hybrid scenario preparation was successful in producing very promising plasmas for DTE2, with core ion temperatures up to 15 keV [35,36]. Unlike baseline scenarios, hybrid scenarios rely on a carefully tailored ramp-up procedure of the plasma current to create a current profile conducive to improved core confinement [37]. The fusion power P_{fus} predicted for DTE2 for these scenarios by a variety physics based modelling was in the range 10-17 MW for a total input power of 40 MW and in the range 7-10 MW for input powers near 30 MW [38,39].

The preparatory phase included experiments in hydrogen plasmas, tritium plasmas and isotope mixtures other than DT for the purpose of understanding the effect of isotope composition on plasma confinement and on the minimum power required for access to the

H-mode. Previous results, based on the DTE1 campaign, albeit on a small dataset, had shown a scaling of the energy confinement time on the ion mass as $A^{0.2}$, all other parameters being constant [40]. Two extensive matched series of experiments with wide parameter scans, one in hydrogen and one in deuterium baseline plasmas performed in 2016, show a stronger scaling of the energy confinement, $\tau_E \sim A^x$ with x in the range 0.4-0.53 [41,42]. This scaling is opposite to that from earlier theoretical expectations (gyroBohm scaling) and is attributed to the non-linear behaviour of fully developed turbulence [42]. An example of a regression for the thermal stored energy on this dataset is shown in fig.5. The thermal plasma stored energy, denoted Wmdh-f in the figure, was inferred from the plasma energy determined by the EFIT magnetic equilibrium code [43] after accounting for the fast ion pressure from NBI and ICRH [44]. The corresponding power law scaling for the thermal stored energy is given as

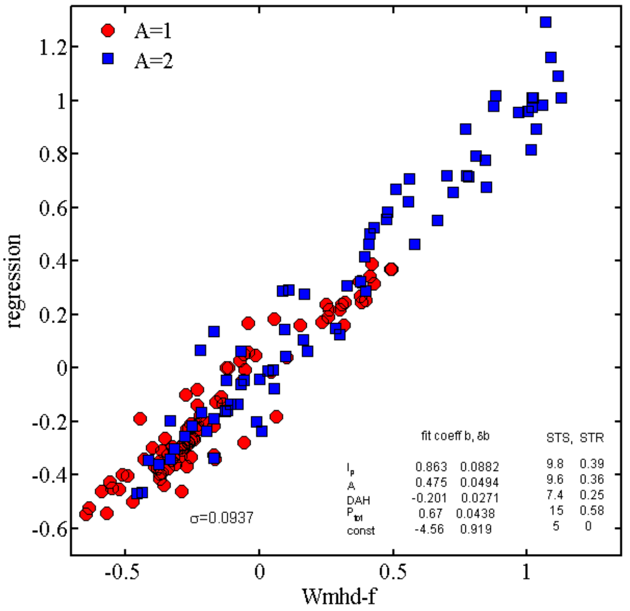

Fig.5 Regression for thermal confinement in H and D plasmas. Dataset as in ref [42], where the significance of the legend is explained.

$$W_{th} \propto A^{0.48\pm0.05} P^{0.67\pm0.04} I_p^{0.86\pm0.09} \Gamma^{-0.2\pm0.03} \qquad \text{(eq.1)}$$

Here P the total heating power, I_p the plasma current and Γ a proxy for the particle fuelling based on visible divertor Balmer-alpha light emission. Dedicated experiments in T, T-H, D-H and D-T mixtures have added several hundred samples to this dataset. Preliminary analysis shows that the favourable isotope scaling persists through the whole range $1 \leq A \leq 3$.

DTE2 RESULTS

The results from DTE2 are still to be analysed in depth and the ones presented here are preliminary in nature. Operation in D-T and T plasmas was found to differ from D plasma operation in several ways [45]. The power necessary to obtain H-mode was lower for lower effective ion atomic mass defined as $A = \sum A_i n_i / \sum n_i$ [46]. Plasma pedestal and overall densities increase with A, at near constant temperature, as already seen in H and D plasmas [47]. Because of technical issues, the heating power available was also lower than anticipated for most of the duration of DTE2. These difficulties implied that significant readjustments of the scenarios were necessary.

Nonetheless, a hybrid scenario [48] pulse with B_T=3.45 T, I_p=2.3 MA and $T_i(0)$≈11 keV was successfully developed with sustained high fusion power for 5s (the duration of the heating pulse) and P_{fus} in the range 7 to 8.6 MW for a total heating power of ~30 MW. This was also the record for fusion energy production (42 MJ) from a 50/50 D/T plasma in a single pulse. The fraction of thermal-thermal fusion power $P_{b\text{-}th}/(P_{b\text{-}th}+P_{th\text{-}th})$, calculated using the ASCOT and AFSI codes [49,50] was about 35%. $P_{b\text{-}th}$ is the power resulting from fusion reactions between the energetic beam ions with thermal plasma ions. P_{th-th} is the thermonuclear fusion power and will dominate in a reactor. The achieved fusion power was in the range predicted by first principles modelling, taking into account the lower than expected power available in DTE2 [39]. Most importantly, the development of the hybrid scenario in DT was able to overcome the difficulties with heavy impurity accumulation, as called for at the start of the ILW project [5] and bodes well for long pulse operation in ITER [51].

A higher fusion power pulse with P_{fus}~10 MW on average for 5.5s (transiently up to 13 MW) and a fusion energy of 59 MJ was obtained with a variation of the hybrid scenario, in which deuterium beams from both beam boxes were injected into a tritium plasma target. Subsequently, due to beam fuelling by the NBI, a D/T mix of ~15/85 was established. This scenario is dominated by beam-thermal fusion with $P_{th-th}/(P_{b-th}+P_{th-th})$~0.1. While this pulse produced the highest fusion power and energy, it is not a reactor scenario as beam driven fusion is energetically inefficient.

The baseline scenario was hampered by excessive impurity radiation, only transiently achieving ~8 MW of fusion power, as well as by a lack of development time [45]. A neon seeded baseline scenario successfully demonstrated the benefit of increasing peripheral radiation for the purpose of reducing the power loads on the divertor PFC's, allowing steady operation of the latter at a constant 450 °C surface temperature with Ne, as compared to a continuously increasing PFC temperature up to 900 °C without Ne. The challenge in a reactor will be to achieve a high edge radiated fraction of the entire fusion power (and additional heating if applicable) while maintaining optimal DT temperatures for thermonuclear fusion (10-20 keV) in most of the bulk plasma.

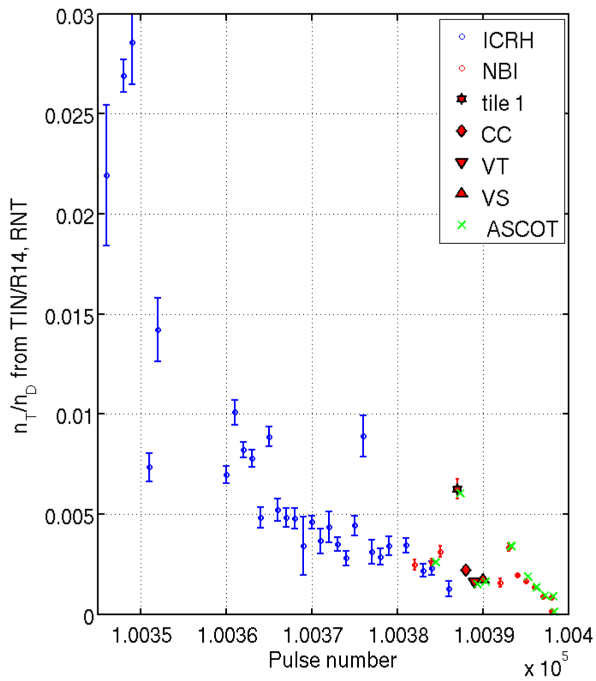

Fig.6 Decrease of the tritium fraction in plasmas after one week of vessel baking, ICWC and GDC. The pulse range shown covers a 3-day operational period.

Alfvén Eigenmodes (AE) , driven unstable by fast ions from ICRH and in some conditions NBI, are regularly observed in JET and other devices [53]. These instabilities, if driven by fusion alphas in a reactor, have the potential of increasing alpha particle losses and heat loads on the walls [54]. Alfvén modes driven by fusion born alphas from the D + ^{3}He → ^{4}He(3.6MeV)+p(14.7MeV) reaction had already been observed during the DTE2 preparation phase [55], however not in DTE2. This non-observation cannot be extrapolated to ITER, as alpha particle populations in JET DTE2 experiments were low and AE's are stabilised by the comparatively low energy ions from NBI. We note that fusion alphas and Alfvén Eigenmodes are now believed to have a beneficial effect on ion heat transport by stabilising ion turbulence [27].

The DTE2 campaign was followed by ~1 month of operation in tritium and a clean-up procedure including baking at 320°C, Ion Cyclotron Wall Cleaning (ICWC) and Glow Discharge Cleaning (GDC), followed by 2 days of plasma operation in deuterium with using ICRH and one day with NBI and ICRH. During the three plasma days, the residual tritium concentration in the plasma was seen to drop from ~2.5% to near 0.1%, much faster than expected, as seen in fig.6, boding well for the remainder of the clean-up, which involved a further month of plasma operation. The measurement was based on the ratio of DT to DD neutron rates from neutron spectroscopy and on modelling using ASCOT/AFSI [49,50].

CONCLUSIONS

Despite having been hampered by technical issues and lower than expected heating power, DTE2 has returned a rich harvest of unique results, in particular validating the usage of a metal wall for future reactor designs. We, however, wish to point out that some important experiments could not be performed in a satisfactory manner. The baseline scenario, which is an ITER operating scenario, did not, in the short available experimental time, demonstrate that it could overcome the issues with heavy impurity radiation. Other experiments were cut short, such as very promising ICRH schemes unique to DT operation [57]. It is the opinion of the author that the questions left open by DTE2 should be addressed in a third major DT campaign, considering, in particular, that a significant neutron budget can still be made available.

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