Internal Transport Barriers in JET Deuterium-Tritium Plasmas

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The observation of internal transport barriers (ITBs) in which ion thermal diffusivity is reduced to a neoclassical level has been made for the first time in tokamak plasmas fueled with deuterium and tritium ions using a broad current density profile. The heating and current profiles required to obtain an ITB are similar in D-T and D-D plasmas. Central ion temperatures of 40 keV and plasma pressure gradients of 10^6 Pa/m were observed in a D-T plasma, leading to a fusion triple product $n_i T_i \tau_E = 1.10^{21}$ m^{-3} keV s and 8.2 MW of fusion power. There is potential for further optimization as a step towards the development of efficient tokamak fusion reactors. [S0031-9007(98)06367-4]

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Current tokamak reactor designs mainly assume operation in the so-called H mode where a transport barrier is formed at the plasma edge. However, scalings from existing experiments predict the need for such machines to operate at high plasma current in order to achieve ignition. While this is sufficient for long pulse operation, a significant effort is now being devoted to the development of operational regimes which would allow ignition to be achieved at lower current by also reducing the transport in the plasma core. With the corresponding reduced demand on noninductive current drive and the large fraction of the plasma current that can be provided by the neoclassical bootstrap effect, an economical route to a fully steady-state fusion reactor is feasible [1].

It has been proposed that the core energy confinement can be improved by modifying the plasma current profile. A flat or hollow current density profile produces a zero or negative value of the magnetic shear s [$s = r/q(dq/dr)$, q being the safety factor defined as the rate of change of toroidal magnetic flux with poloidal flux]. This is predicted to allow the plasma to remain stable at higher plasma pressure and to stabilize toroidal drift instabilities such as trapped particle and ion temperature gradient instabilities [2]. Such an improvement has been observed in various tokamak regimes. In particular, current profile control has allowed the production of plasmas where the transport is reduced to the neoclassical value in the center of the plasma within a region called the internal transport barrier (ITB) [3–6]. The understanding of the mechanism allowing an ITB to be triggered is progressing and it is likely that a combination of $\mathbf{E} \times \mathbf{B}$ shear and magnetic shear stabilizations are necessary conditions [7,8]. Scaling laws for such regimes are not yet established. An important issue is the effect of the fuel (deuterium and tritium to be used for fusion reactors) on the establishment of the ITB. The only data available so far in D-T plasmas have been from TFTR [9]. These indicated that the power threshold for the formation of an ITB was much higher in D-T than in D-D so that the D-T experiments in this regime were unsuccessful. In this Letter we will present the D-T experiments in JET, where ITBs were successfully produced, although with a modified scenario compared with that used in D-D.

In experiments where an ITB is present, the profile of the plasma current is usually modified by preheating the plasma during the current ramp-up phase so as to delay the inward diffusion of the plasma current. The main heating is then applied while the current profile is flat or hollow with low or negative magnetic shear. ITBs are produced in JET in the so-called optimized shear regime described in [10]. The plasma current is ramped at a rate of 0.4 MA/s with a single null X-point configuration, with the location of the strike points allowing maximum pumping from the divertor cryopump. In the first second of the discharge, 1 MW of lower hybrid current drive (LHCD) provides off-axis current drive and increases the electron temperature. Ion cyclotron resonance heating (ICRH), using fundamental hydrogen minority heating with the resonance near the plasma center increases the central electron temperature to about 6 keV. The preheating phase ends with increasing ICRH power and the start of neutral beam injection (NBI) at a power level up to 10 MW. The main heating phase starts about 0.4 s after the start of NBI injection. The use of a low target density and continued current ramp at the start of the main heating phase allows an ITB to be established before the onset of an H mode. The good core confinement maintains the plasma loss power below the level required to trigger an H mode, thus preserving a low pressure gradient at the edge in the so-called L mode. The highest fusion yield in D-D plasmas in JET has been achieved in this way [11].

Figures 1 and 2 show the time evolution of a D-D and a D-T pulse with similar power waveforms, along with the very peaked temperature profiles that result from the formation of an ITB. The main effect to overcome in D-T is the reduced power threshold for the L- to H-mode transition, which can be seen in Fig. 1. The early appearance

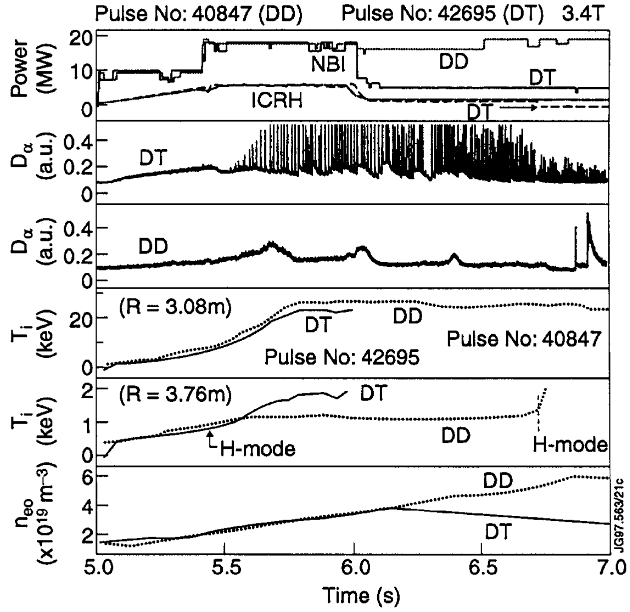

FIG. 1. Time evolution of signals for similar D-D and D-T optimized shear plasmas. For the D-T pulse, the target tritium concentration was 30% and the NBI tritium fueling concentration was up to 45%.

of an H-mode edge in the D-T pulse is seen from the edge pedestal in the ion temperature which is frequently interrupted by magnetohydrodynamics (MHD) events, called edge localized modes (ELMs) which are seen as sharp increases in the D_α emission. In D-D, the ITB is broad and has been found to expand continuously with time, whereas in this ELMy H-mode D-T pulse, it remains relatively narrow. These quasistationary conditions in D-T have also been achieved in D-D pulses [11] and offer a route to steady-state operation.

By modifying the power waveforms, plasmas with ITBs and L-mode edge have been achieved in JET D-T plasmas, as shown in Fig. 3. An ITB is established about 0.3 s after the start of the main heating phase. The plasma edge remains in an L mode until, at time $t =$ 6.7 s, a decrease in D_α emission indicates the formation of an ELM-free H mode. Subsequently, several large ELMs terminate the phase of improved confinement. This sequence of events is very similar to that in D-D. Record values of ∇T_i of 150 keV/m and ∇p_i of 10^6 Pa/m have been achieved in D-T as well as a record central ion temperature approaching 40 keV when the magnetic field and plasma current were increased, respectively, to 3.85 T and 3.4 MA. In Figs. 4 and 5 the corresponding ion temperature and density profiles are plotted against major radius. Such high ion temperatures can be explained by the combined action of NBI and ICRH. The ICRH frequency was 56.6 MHz which corresponds not only to the fundamental resonance of minority hydrogen ions but also to the second harmonic of deuterium ions and to the third harmonic of tritium ions. Therefore, as observed in D-D plasmas, a substantial part of the ICRH is damped directly on both beam ions and thermal ions. This allows the rapid formation of the ITB and also results in a large nonthermal component of the stored energy and a strong peaking of the plasma pressure. The pulse shown in Fig. 5 ends with a disruption which is attributed to a global $n = 1$ ideal kink mode driven by the strong pressure profile peaking. In similar D-D pulses, it was found that the disruption could be avoided by reducing the additional heating power until the ITB expanded. The larger the ITB, the higher the power which could be injected in the plasma core without causing a termination. The limited number of D-T pulses, about ten pulses similar to the one shown in Fig. 3, has not allowed such an optimization to be performed.

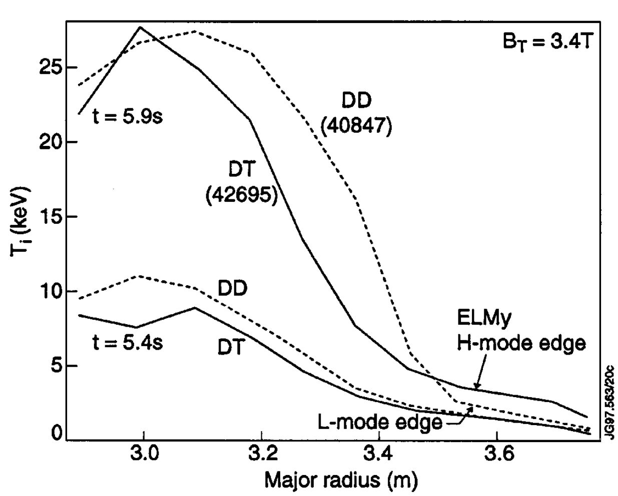

FIG. 2. Radial ion temperature profiles from charge exchange spectroscopy for similar D-D and D-T pulses as in Fig. 1.

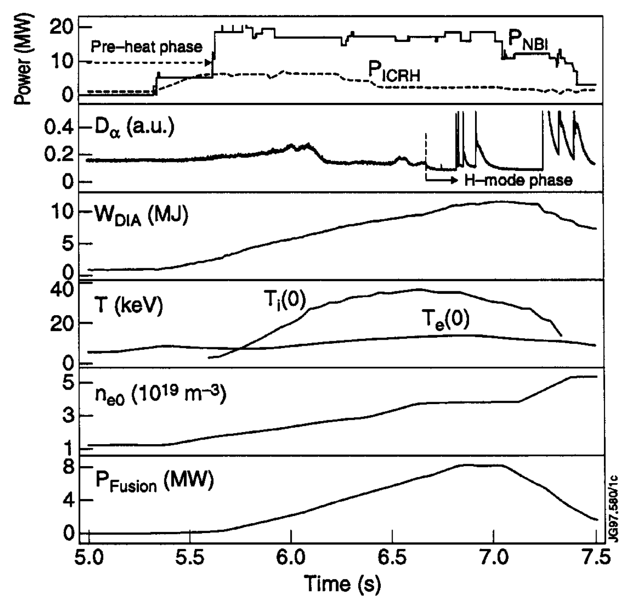

FIG. 3. Time evolution of typical signals for pulse 42746 in D-T at $B_T = 3.45$ T. I_p is increased at 0.4 MA/s up to 3.24 MA at $t = 7$ s, $f_{\mathrm{ICRH}} = 51.3$ MHz, $Z_{\mathrm{eff}} = 1.4$.

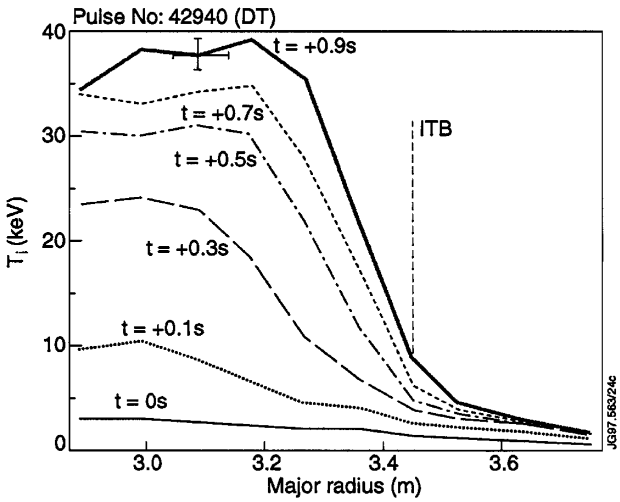

FIG. 4. Radial ion temperature profiles from charge exchange spectroscopy for pulse 42940 ($B_T = 3.8$ T, I_p up to 3.4 MA). An ITB is triggered 0.3 s after the start of the high power phase.

The typical tritium concentration in the target plasma ranges from 18% to 30%, while the main tritium fueling from NBI is up to 45%, 10 MW of tritium being injected at 150 kV and 8 MW of deuterium at 75 kV. The overall fueling within the ITB was 10% to 15% less in D-T than in D-D for similar injected power, due to the higher average energy of injection. The estimated tritium concentration at the peak of fusion power is about 30%, which is significantly less than the optimum of 50% needed for maximum fusion yield. Nevertheless, a peak fusion power of 8.2 MW was obtained with low impurity concentration ($1.4 < Z_{\mathrm{eff}} < 1.8$) and no indication of impurity accumulation within the ITB. The ratio between central hydrogen densities and total central density remains above 0.8 throughout the discharge.

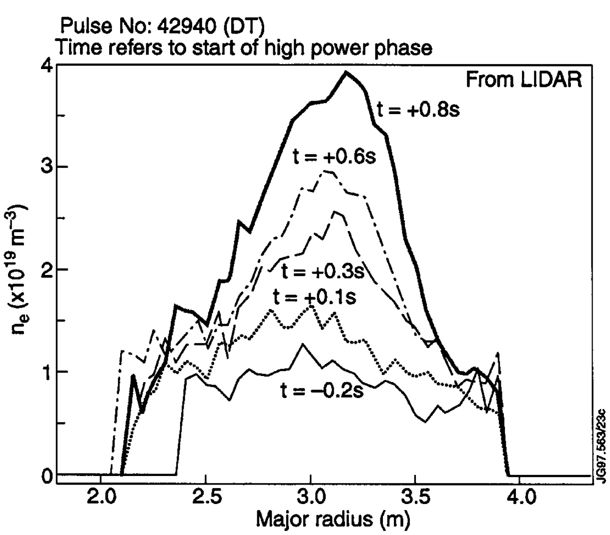

FIG. 5. Radial electron density profiles from LIDAR measurements for pulse 42940. Fusion yield reaches 7.2 MW with a stored energy of 10 MJ at $t = +0.9$ s.

In JET, triggering of ITBs requires, both in D-D and in D-T, a $q = 2$ magnetic surface in the plasma. The q profile is calculated from magnetic reconstruction, and the presence of the $q = 2$ surface is confirmed by $m = 2$, $n = 1$ MHD activity. Although central values of the magnetic shear cannot be accurately determined, the location of the $q = 2$ surface can be determined to within ± 10 cm for radius higher than $\rho = 0.4$, typical for optimized shear discharges. TRANSP [11] calculations using classic current diffusion confirm the location of the reconstructed $q = 2$ surface and indicate a link between the $q = 2$ location and the position of the ITB. These calculations also indicate low values of magnetic shear inside the ITB, but need to be confirmed with more sophisticated diagnostics to be installed in the future.

In comparing D-D and D-T pulses in similar conditions, the central ion temperature in D-T is several keV higher, as seen in Fig. 6. The reduced density in D-T plasmas partly explains the higher values of ion temperature in the plasma core. The TRANSP code has been used to check overall data consistency. The radial profiles of the ion thermal diffusivity χ_i, from TRANSP analysis are plotted in Fig. 7. Although error bars are large in the plasma center, these profiles are compared with the standard neoclassical values calculated assuming the ion banana width to be small compared with typical length scales. However, it is known that, owing to the high ion temperature and the low value of the poloidal field near the plasma center, this comparison becomes increasingly uncertain as the center is approached. Nevertheless, the values of ion thermal diffusivity approach the neoclassical level in the plasma center for both the D-D and the D-T cases.

This reduction in transport is often linked [7] with $\mathbf{E} \times \mathbf{B}$ shear stabilization of the turbulence. In JET, toroidal

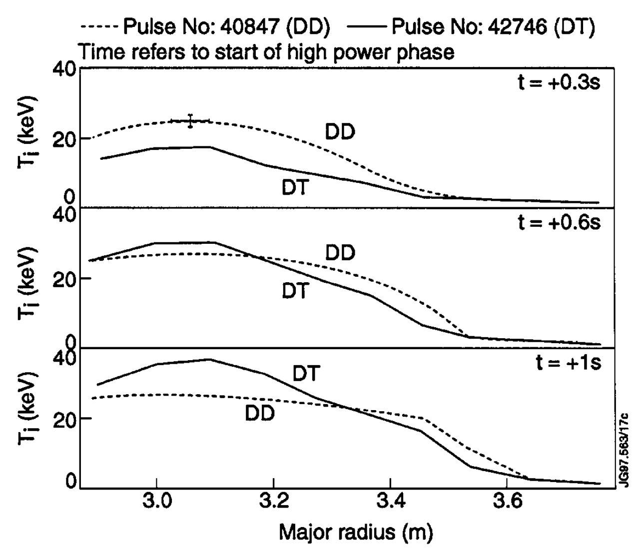

FIG. 6. Comparison of ion temperature radial profiles for D-D and D-T optimized shear pulses with similar power waveforms and target q-profiles. Typical uncertainty in ion temperature measurement is ± 1 keV.

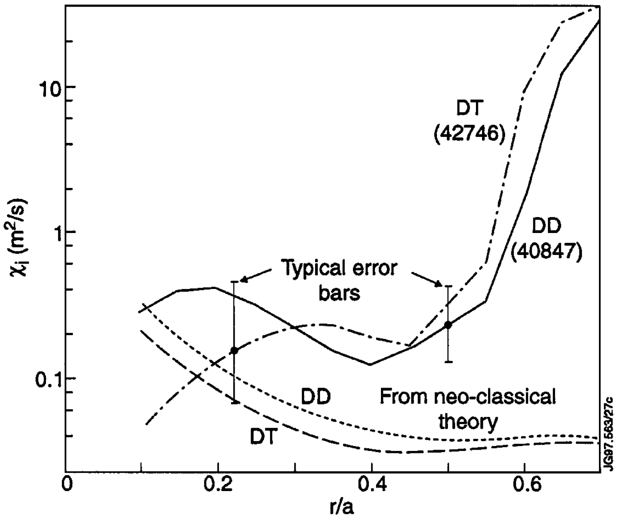

FIG. 7. Comparison of ion thermal diffusivity with typical error bars from TRANSP analysis for the pulses in Fig. 4 at $t = +1$ s.

rotation is measured from charge exchange spectroscopy, but poloidal rotation can only be calculated from a neoclassical model. This gives values of up to 150 kV/m, just inside the ITB for both D-D and D-T plasmas, which produces a shear flow rate in excess of the growth rate for the toroidal branch of the ion temperature gradients driven instability.

In summary, internal transport barriers have been produced in D-T plasmas with similar additional heating power levels and similar plasma current profiles to those in D-D. Optimized shear plasma scenarios in JET have been modified for D-T in order to take into account the consequences of a lower L- to H-mode power threshold compared with D-D plasmas. Plasmas have been produced with significant fusion production: 7.2 MW of fusion power at the end of the L-mode edge phase and 8.2 MW during the subsequent H-mode edge phase with low impurity concentration. Central ion temperatures of 40 keV, ion temperature gradients of 150 keV/m, pressure gradients of 10^6 Pa/m, and a triple product $(n_i T_i \tau_E)$ of 1×10^{21} m^{-3} keV s (L-mode phase) have been achieved in D-T plasmas. The limited available 14 MeV neutron budget has prevented development of several optimization routes, including control of central pressure and plasma edge, which may lead to further performance improvement. The demonstration of internal transport barriers in deuterium-tritium plasmas with ion thermal diffusivities comparable with neoclassical levels is a further step in the direction of using current profile control as a tool to improve confinement in a thermonuclear fusion tokamak reactor. In addition, the coexistence of internal transport barriers and edge transport barriers in D-T plasmas is important for the future of the method.

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