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ABSTRACT.
The fusion performance of JET plasmas can be enhanced by the generation of internal transport barriers. The influence of the $q$-profile shape in the local and global plasma performance has been investigated in cases where the core magnetic shear ranges from small and positive to large and negative. Internal barriers extending to large plasma radii can be effective in raising the global performance of the plasma. It is found that such barriers tend to be generated more easily if the $q$-profile contains a region of negative magnetic shear. The formation is favoured by neutral beam injection compared with ion cyclotron resonance heating in scenarios where the two systems used together. The minimum power level required to observe a local transport reduction is significantly lower than the value at which very steep pressure gradients can be achieved. This results in a practical threshold in the power to access a regime of high plasma performance that is sensitive to the $q$-profile shape.

1. INTRODUCTION
The prospects for an economical steady-state tokamak fusion power plant could be significantly enhanced if plasma regimes could be developed to provide high plasma pressure at low plasma current in stationary conditions without the need for excessively high toroidal magnetic field strength or very large plasma volume. This combination of factors, known as the ‘Advanced Tokamak’ scenario, leads to conditions where a large fraction of the plasma current is driven by the neoclassical bootstrap mechanism which, in turn, results in a relatively modest requirement on the externally driven non-inductive current. This requires enhanced plasma confinement compared with the ELMy H-mode regime currently envisaged as the reference scenario for ITER. In recent years experiments have been performed on many tokamak devices to investigate the prospects for achieving such a scenario by the generation of a region of high plasma energy confinement in the plasma core, called an ‘Internal Transport Barrier’ (ITB). Indeed, the specification of ITER allows for the exploitation of regimes of this type and so has provided further impetus to the study of conditions where improved core plasma confinement is observed. A method commonly used to generate an ITB in a tokamak is the application of additional heating power during the current ramp-up phase at the beginning of a discharge before the plasma current has fully penetrated [2,3,4,5,6,7,8]. This can result in the safety factor, $q$, being significantly above unity throughout the plasma and the magnetic shear, $s (s = r/q(dq/dr))$, being low or negative in the plasma core. This magnetic configuration has been used for the production of ITBs that can locally reduce the radial transport of electron or ion energy, or both, and, in some cases, particle transport as well. A theoretical explanation for this localised reduction in transport has been proposed in terms of the suppression of plasma turbulence by sheared plasma flow, characterised by the $E \times B$ shearing rate, in both the toroidal and poloidal directions [9]. Indeed, ITB formation in JET experiments has been associated with the observation of turbulence suppression [10] and correlates with both magnetic shear and plasma flow shear [11].

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of this type, together with observations on other devices of the effect of varying the \( q \)-profile shape [12] and toroidal plasma flow [13,14,15] on the transport characteristics in the plasma interior, have motivated a further investigation of the effects of the \( q \)-profile and plasma flow shear on ITB generation in JET.

Experiments in JET have previously been reported where ITB generation in plasmas characterised by a monotonic \( q \)-profile with central \( q \) close to 2 has been compared with initial experiments using negative central magnetic shear [16]. It was observed that the location and evolution of the ITB, as well as the heating power required for its production, are sensitive to the \( q \)-profile shape. Indeed, many factors are thought to influence the formation of ITBs in tokamaks including MagnetoHydroDynamic (MHD) instabilities, plasma heating scheme, fuelling and applied torque. Recent experiments have been performed in JET to vary the \( q \)-profile shape and heating arrangements to further investigate the role of these different parameters on the generation of ITBs. At low heating power levels a local discontinuity can sometimes be seen in the plasma temperature gradient in the plasma interior indicating a local reduction in the energy transport. The confinement improvement is often transient and the correlation of such events with MHD phenomena has led to the interpretation that these instabilities can act as part of a trigger mechanism for transport barrier formation [17].

At higher levels of additional heating power such plasmas can exhibit a further large increase in the pressure gradient in the region of the ITB, which results in a significant enhancement in the plasma stored energy and fusion yield. The highest fusion yield produced in JET deuterium plasmas was obtained in this way using the low magnetic shear ITB regime [18]. High performance ITBs have now also been produced in plasmas with highly negative central magnetic shear, similar to the conditions exploited for high fusion gain in other tokamaks, such as JT-60U [19]. These experiments extend the range of conditions over which high confinement transport barriers can be studied in JET.

In this paper the experimental method used to vary the \( q \)-profile shape and heating method is described. The experimental results are presented in terms of the plasma conditions required for the appearance of an ITB. An assessment is made of the requirements to exploit the locally improved confinement for the achievement of high overall plasma performance. Finally, the advantages and limitations of different \( q \)-profile and negative magnetic shear regimes will be discussed.

2. METHOD TO VARY THE MAGNETIC SHEAR AND PLASMA FLOW SHEAR
The plasma \( q \)-profile obtained during the initial current ramp-up phase of a discharge in JET is sensitive to the plasma heating, current ramp rate and applied non-inductive current drive [20]. In the case where only Ohmic heating is used with a moderate current ramp rate (\( \approx 0.4 \text{MA/s} \)) the \( q \)-profile typically remains monotonic. It has been found, however, that the application of Lower Hybrid heating and Current Drive (LHCD) during the current ramp phase is effective to produce a negative magnetic shear region in the plasma core and high values of central \( q \). The effect is
partially due to off-axis current drive provided directly by the LHCD. This is assisted by the formation of an electron energy ITB, which, in the presence of strong electron heating at low density, raises the central electron temperature so as to temporarily prevent the penetration of plasma current to the core.

Figure 1 shows the time evolution of two plasma pulses, similarly set up except that one had LHCD applied soon after the plasma initiation while the other had a purely Ohmic current ramp-up phase. Both plasmas were then heated with Neutral Beam Injection (NBI) and Ion Cyclotron Resonance Heating (ICRH) resulting in the formation of an internal transport barrier which is evident on both the ion and electron temperature profiles. The different time evolution of the neutron yield indicates that these ITBs were not formed at the same time in the two cases, nor did they result in the same degree of enhancement in the core plasma pressure. These differences and their implications are discussed in detail in the following section. LHCD was switched off at the beginning of the main NBI and ICRH phase so that the plasma density, heating, applied torque and non-inductive current drive were similar at this time in the two cases. The parameters of the main heating pulse were then varied in a series of pulses to study the conditions for ITB production in each case.

![Figure 1](image1.jpg)

*Figure 1. Time evolution of two similar discharges with and without LHCD in the early current ramp-up phase showing: the plasma current ($I_p$); additional heating power; internal inductance; line density ($n_e \ell$); and neutron yield. During the main heating phase the fusion yield is initially higher in the case without LHCD due to the formation of an ITB at $r/a = 0.6$. With LHCD a similar ITB is established at $t=6s$, which provides an even higher performance enhancement.*

![Figure 2](image2.jpg)

*Figure 2. Comparison of $q$-profiles prepared with and without an LHCD prelude in experiments of the type shown in figure 1. The shaded area represents range of $q$-profiles obtained in a series of such pulses while the lines show the average. The uncertainty in the determination of the absolute magnitude of $q$ is estimated to be of order ±15%.*
The $q$-profiles have been determined for these experiments using the EFIT equilibrium reconstruction code \cite{21,22} constrained by Motional Stark Effect (MSE) measurements \cite{23,24}. The results are consistent with analysis using measurements from a far-infrared polarimeter, but since the resolution of the MSE instrument allows a better determination of the magnetic shear in the plasma core, the MSE measurements are used throughout this paper. During the main heating phase the use of several overlapping neutral beams in the diagnostic line-of-sight complicates the MSE measurement, so a reduced power phase was included at start and end of the main heating pulse so that good measurements could be made of the $q$-profile. Figure 2 shows the comparison of $q$-profile shapes obtained at the start of the main heating pulse in the two discharge types illustrated in figure 1. It can be seen that the Ohmic preheat plasmas are characterised by a monotonic $q$-profile while the LHCD prelude case has a region of ‘weakly’ negative magnetic shear in the plasma interior. The $q$-profile shapes are represented in figure 2 by the average over each series of pulses. The accompanying shaded areas illustrate the full range of $q$ values obtained and therefore indicate the shot-to-shot reproducibility of the current profile in each series rather than the precision of the measurement technique. The uncertainty in the determination of the absolute magnitude of $q$ is estimated to be of order $\pm 15\%$.

A complimentary series of pulses were performed to vary the $q$-profile shape over a wider range than that shown in figure 2 followed by a high power main heating pulse. The conditions at the start of the main heating phase for this $q$-profile scan are illustrated in figure 3. The magnetic shear in the plasma interior was varied from small and positive in the Ohmic preheat case, through ‘weakly’ negative with low power LHCD, to ‘highly’ negative (extremely high central $q$) when the LHCD prelude was accompanied by an optimised plasma initiation sequence. The experimental technique to obtain this third class of $q$-profile is described briefly in section 4 of this paper. The narrow minimum in the $q$-profile shape in this case is thought to result from the localisation of lower hybrid current drive by an ITB. This feature is described in more detail elsewhere and quickly diffuses away when the LHCD is switched off\cite{25}. To distinguish this series of pulses from those illustrated in figures 1 and 2 in this paper it is hereafter referred to as the data is used in the analysis of both experiments.

Large plasma flows are generally linked to the application of additional heating power. The poloidal plasma flow is mainly driven by the radial plasma pressure gradient whereas shear in the toroidal plasma flow is largely provided by the torque applied by the NBI system. Consequently, the plasma heating parameters have been varied systematically in this experiment for the monotonic and ‘weakly’ negative magnetic shear target $q$-profile shapes shown in figure 2. The heating power was scanned to vary the poloidal flow shear through the plasma pressure gradient. Altering the ratio of the NBI and ICRH power also varied the toroidal flow shear. This technique has the disadvantage that, although it is effective for altering the applied torque at a given level of the total heating power, it results in significant modification of the core particle fuelling and, in the case of fundamental hydrogen minority ICRH, the ratio of ion and electron heating.
The effect on the power deposition profile of varying the ratio of NBI and ICRH heating at a constant total power is illustrated in figure 4. This shows the predicted fraction of the input power deposited in the plasma core (r/a<0.6) in the case of the plasma parameters and total heating power achieved in the Ohmic preheat pulse shown in figure 1. The ratio of NBI to ICRH was varied artificially over the whole range from NBI only to pure ICRH and the expected level of ion and electron heating was evaluated in the plasma core. The calculation was performed using the PENCIL NBI code \[26\] and the PION ICRH code \[26\], which include the effects of ICRH coupling to the neutral beam generated fast deuteron population. It can be seen that the overall core heating does not vary strongly in the range of NBI fractions used in this experiment (17% to 63% in the cases where it was systematically varied at constant total heating power) while the relative ion and electron heating is significantly altered. It is interesting to note that the highest fraction of the heating power is delivered to the plasma core in the case of combined heating. This scenario also allows the maximum additional heating power levels to be achieved in JET, and has been effective for the production of high performance ITBs. In this series of experiments, however, the effect of the various differences between the NBI and ICRH heating characteristics is to weaken the conclusions that can be drawn concerning the processes responsible for ITB formation. The arrangement of the NBI system, on the other hand, allows some modest variation of the injection geometry at reduced power. In this case the heating and fuelling characteristics can be maintained while only the applied torque is varied.
The JET NBI system [28] comprises sixteen separate beamlines: eight configured with a tangency radius of 1.31m (‘normal’ beams); and eight orientated with a tangency radius of 1.85m (‘tangential’ beams). The ‘normal’ beams are aligned so as to intersect the inner wall of the tokamak vessel whereas most of the ‘tangential’ beams pass through the plasma twice and strike to outer wall of the device. Part of the neutral beam power passes through the plasma without being absorbed. This shine-through fraction is highest for the ‘normal’ beams due to the single pass they make through the plasma. The shine-through power has been calculated for each of the pulses in this experiment using the PENCIL NBI code. The ‘coupled’ heating power was then estimated by subtracting the shine-through component from the total power. The applied NBI torque was determined in the same way. The uncorrected heating power has been quoted for the high performance experiments presented in section 4 of this paper due to the very large number of pulses involved. At the higher densities obtained in plasmas with high fusion yield the shine-through fraction is typically small (less than 10%).

The applied torque can be varied up to the ratio of the tangency radii by comparing the effect of the two beam configurations with the NBI power restricted to half the routinely available level of about 18MW. All beams are orientated for co-injection (in the direction of the plasma current) for this series of experiments. The complete range of heating power and NBI torque applied with two $q$-profile shapes in figure 2 and the $q$ profile scan in figure 3 is illustrated in figure 5.

![Figure 5](image1.png)

Figure 5. Range of additional heating power coupled and NBI torque applied to plasmas with and without an LHCD prelude and in the $q$-profile scan.

![Figure 6](image2.png)

Figure 6. Central $q$ plotted against the main heating power level coupled for each pulse before ($t_{\text{start}}$) and after ($t_{\text{end}}$) the main heating phase ($t_{\text{end}}-t_{\text{start}} \approx 2.5s$). The solid symbols represent cases with ‘weakly’ negative central magnetic shear while the open symbols are for monotonic $q$-profiles.
In this type of experiment the $q$-profile evolution during the main heating pulse depends on the level of heating power applied, partly due to variations in the achieved electron temperature. The noninductive current drive provided by the neoclassical bootstrap effect, which is sensitive to the poloidal magnetic field strength and the plasma pressure [29,30], and the beam driven current also vary as the heating system parameters are altered. The noninductive fraction of the plasma current is estimated to be up to about 50% in these pulses and this may significantly influence the $q$-profile evolution during the main heating phase.

The evolution of the $q$-profile during the short main heating phase ($\Delta t \approx 2.5s$) is illustrated in figure 6, which shows the central value of $-q$ before and after this phase as a function of applied heating power. The rate of fall is greatest for the pulses following the use of LHCD during the current ramp-up phase as the value of central $-q$ is higher at the start of the main heating phase and therefore furthest from the stationary condition. The rate of fall of central $-q$ increases at the lowest power levels because the electron temperature is lower and allows more rapid current penetration. Also the fraction of the plasma current provided by the offaxis bootstrap mechanism, which is roughly proportional to $\beta_{\text{poloidal}}$ [31], is reduced at low power. In the case of low power heating following an LHCD preheat phase that generates only a ‘weakly’ negative magnetic shear target the $q$-profile can become completely monotonic by the end of the main heating phase. These cases are indicated in figure 6. The effect of varying the NBI current drive on the $q$-profile evolution when comparing the high and low torque cases is small compared with the overall sensitivity of the central diffusion to the plasma heating. The appearance of integer $q$ surfaces in the plasma interior varies in time with the applied main heating power level as well as the initial $q$-profile generated by the different preheating techniques. This affects the time evolution of the discharges as they can play a role in the formation of ITBs.

3. EFFECT OF MAGNETIC SHEAR, HEATING POWER AND APPLIED TORQUE ON ITB FORMATION

ITBs have been identified in previous JET experiments from discontinuities in the gradients of the ion and electron temperature as well as the plasma density and toroidal rotation velocity [32]. Such discontinuities can be interpreted in terms of a local reduction in the energy, particle and momentum transport. In cases where the $q$-profile is monotonic, ITBs have generally been observed to form near integer $q$ magnetic surfaces [16,33]. A mechanism has been proposed to explain the link between ITB formation and rational $q$ surfaces based on a local enhancement to the plasma flow shear due to the coupling of core and edge MHD modes [17]. The power required to generate such a transport barrier at the $q=2$ surface, where the most extensive database has been established, was seen to increase with toroidal magnetic field strength (or plasma current, which has the same dependence) [34]. The ITB access power for JET (in MW) scaled roughly as $5B_t$ (in T). Initial ITB experiments on plasmas with a region of negative magnetic shear in the core showed that ITBs could be obtained at much lower power levels. In this scenario barriers
were generated in the negative magnetic shear region at a location that did not appear to be related to the location of a particular \( q \) surface and even during the preheating phase with very low levels of predominantly electron heating. It was possible to obtain both types of ITB simultaneously during high power heating, but at different plasma radii.

In the experiments described in the previous section a wide range of ITB phenomena have been observed. ‘Narrow’ ITBs can be seen on the electron temperature profile during the low power LHCD prelude phase in the case of ‘strongly’ negative magnetic shear. These are discussed more fully elsewhere [35]. In this paper ‘narrow’ ITBs are defined as those located entirely within \( r/a<0.5 \). The core electron ITB can persist during the main heating phase depending on the parameters and timing of the main heating pulse [36]. Even in the monotonic \( q \)-profile cases ‘narrow’ ITBs are sometimes seen on the ion temperature profile during the main heating pulse depending on the location of low order rational flux surfaces. These have also been generated at power levels much lower than \( 5B_T \) and illustrate that this power scaling, obtained from a database of similar ITBs associated with \( q=2 \), is not universally applicable to JET ITB formation even in cases with Ohmic preheating. This is thought to be due to a sensitivity to local conditions, such as the magnetic shear and the presence of MHD instabilities [37], and it is therefore concluded that a power threshold scaling based on global parameters alone is unlikely to adequately describe the complete ITB phenomenology. However, these ‘narrow’ ITBs obtained at low heating power levels do not strongly improve the global plasma performance in terms of total stored energy or fusion yield. At higher main heating power levels, approaching \( 5B_T \), ITBs were obtained in these experiments at wider radius (\( r/a=0.6 \)) in both the LHCD and Ohmic preheat cases. These ITBs, seen on both the ion and electron temperature profiles, were able to provide a significant global plasma performance enhancement. Figure 7 shows examples of the different classes of ITB obtained during this series of pulses for comparison. The electron temperature profiles were determined from Electron Cyclotron Emission (ECE) using a heterodyne radiometer with high spatial resolution, of order 2cm, and the ion temperature profiles were measured using charge-exchange recombination spectroscopy.

The ‘wide’ ITBs generated during this investigation were obtained near the location of the \( q=2 \) surface. The term ‘wide’ is used to describe ITBs characterised by a steep gradient that extends into the plasma region \( r/a>0.5 \). Although the electron temperature gradient is typically localised in the outer part of the plasma, the ion temperature gradient sometimes extends into the plasma core, as illustrated by case (iv) in figure 7, which has ‘weakly’ negative magnetic shear. The equivalent example with a monotonic \( q \)-profile (profile (iii) in figure 7) exhibits the same increased gradient around \( R=3.5m \) compared with the reference profile, but the enhancement does not extend as far into the plasma centre. This contributes to the difference in the peak fusion yield observed in figure 1. However, there is no evidence of a significant pressure gradient within the very high \( q \) region at the core of LHCD prelude cases with optimised plasma initiation, such as example (v) in figure 7. The appearance of these ‘wide’ ITBs was delayed in the case of the LHCD prelude due to the absence of any \( q=2 \) magnetic surface at the start of the main heating pulse. The resulting delayed improvement in
fusion yield is seen in the LHCD prelude example shown in figure 1. The delay in the formation of this ITB increased with additional heating power due to the slowing of the current profile evolution illustrated in figure 6. This means that, despite the variations in current penetration dependence on applied power, the \( q \)-profiles were similar at the moment of ITB formation. This, together with the relatively slow current profile evolution in the cases with an Ohmic preheat, provided reproducible \( q \)-profile shapes for the assessment of heating variations.

The identification of this ‘wide’ ITB has been made from a criterion based on the ratio of the inverse temperature gradient scale length to the ion Larmor radius at the sound speed (\( \rho^*_s = \rho_s/L_T \)), where a value of \( \rho^*_s T > 0.014 \) corresponds to a clear ITB [38]. Figure 8 shows the peak value of \( \rho^*_s \), applied to the electron temperature profile in the region of the ‘wide’ transport barrier, as a function of coupled additional heating power. ITBs become evident when the power level exceeds about 9MW, depending on the \( q \)-profile shape and heating arrangements. It was seen in figure 2 that a major difference between the Ohmic and LHCD prelude cases is that the magnetic shear at the location of the ‘wide’ ITB was much lower, or negative, in the cases where LHCD was used.

\[
\rho^*_s \equiv \frac{\rho_s}{L_T}
\]

Figure 7. ‘Narrow’ ITBs evident on electron temperature profile during LHCD prelude (i), and ion temperature profiles during low power heating following Ohmic preheat (ii). ‘Wide’ ITBs illustrated by the ion temperature profile: during high power heating following Ohmic preheat (iii); during high power heating following LHCD prelude (iv); and during very high power heating following an optimised LHCD prelude (v). Open symbols show the reference conditions before a clear ITB is seen. The error bars indicate the uncertainty in the measurement location and absolute magnitude. In the case of \( T_e \), the radial resolution is much less than this, of order 2cm.

Figure 8. Maximum value of \( \rho^*_s \) (\( \rho/L_{Te} \)) versus coupled power for various \( q \)-profile and heating scenarios at B=2.6T. Analysis is restricted to barriers at wide plasma radius. An ITB is deemed clear if \( \rho^*_s > 1.4 \times 10^{-2} \). Annotations (a) to (d) refer to the pulses illustrated in figure 3.
The plasmas with an LHCD prelude tend to provide a ‘stronger’ electron ITB (i.e. larger value of $\rho^*_{Te}$) than the Ohmic preheat cases. This observation could be attributed to the reduction in magnetic shear at the location of the $q=2$ surface, which is expected to be favourable for ITB formation, although it is interesting to note that ITBs are observed on the electron temperature profile in JET in regions of positive or negative magnetic shear. The $q$ profile scan at high power also shows the generation of ‘steeper’ electron temperature gradient in the cases with and LHCD prelude (points (b), (c) and (d) in figure 8), when compared to the Ohmic preheat plasma (point (a)). However, in the LHCD prelude cases there is no correlation between the value of $\rho^*_{Te}$ and the value of the magnetic shear in the plasma core. This is not entirely surprising since the peak value of $\rho^*_{Te}$ for the ‘wide’ ITB occurs outside the region where ‘highly’ negative magnetic shear is obtained. It also suggests that the achievement of ITBs that are both ‘wide’ and ‘strong’ is favoured by the presence of an off-axis minimum in the $q$-profile, while it may not be sensitive to the value of central $q$.

Figure 8 also shows that plasmas heated predominantly with NBI generate ITBs much more effectively than plasmas with a larger fraction of ICRH. The poorer performance with the increased fraction of ICRH, in terms of the local electron temperature gradient achieved at a given power level, cannot be unambiguously explained due to the simultaneous variation of several parameters. Most of the power from the ICRH is delivered to the plasma electrons through a highly energetic ion tail generated from the thermal plasma ion velocity distribution. This is characteristically similar to the main heating mechanism in a fusion power plant, where fusion produced $\alpha$-particles will principally heat the plasma electrons without providing fuelling or applying torque. The use of combined NBI and ICRH, however, provides the possibility that the effects of the toroidal and poloidal rotation might be tending to cancel with high fractions of ICRH, since are associated radial electric fields are of opposite sign when co-injected NBI is used. Indeed, calculations of the $E \times B$ shearing rate generated in the characteristically similar high performance ITB plasmas described in section 4 show that the toroidal rotation terms dominate in this scenario when mainly high power NBI heating is used [39]. This is qualitatively consistent with the observations on TFTR that turbulence suppression associated with an ITB was momentarily lost when making the transition from balanced NBI heating to dominantly co-injection. This was attributed to the temporary reduction in the magnitude of the radial electric field, and hence shearing rate, as it changed sign during the increase in toroidal rotation [13]. Experiments have also been reported on DIII-D where wider ITBs were obtained using counter-injected NBI compared with coinjection [14], and on JT-60U where both the level of reduced transport and the radial extent of ITBs was changed by varying the NBI momentum input using co-, counter- and balanced injection [15]. The relative inefficiency of the ‘high ICRH’ experiments presented here to form ITBs capable of enhancing the global plasma performance indicates that this scenario requires further study, especially with pure ICRH where the radial electric field cancellation can be discounted. Such investigations are necessary to establish a basis for extrapolation to power plant regimes. The comparison of pulses with the same ratio of NBI and ICRH, but varying the applied torque through the NBI geometry, was inconclusive, perhaps due to the relatively small variation in torque that could be achieved with the JET NBI geometry.
Figure 9 shows the peak value of $\rho_{ie}^*$ evaluated for the ion temperature profile in the same plasma region as for the data presented in figure 8 for the electron temperature profile. The pulses with a high fraction of ICRH during the main heating, as indicated on figure 5, have been omitted from this figure because the relative reduction in the ion to electron heating ratio makes it difficult to draw conclusions directly from the data. It should also be noted that the charge-exchange recombination spectroscopy measurements are made on a sparse radial grid ($\Delta R \approx 10$ cm) so that local enhancements in the ion temperature gradient with a radial extent comparable or less than the measurement separation cannot be fully resolved. Results for cases with an Ohmic preheat are consistent with the previous JET scaling of $5B_T$ which would give a power requirement of about 13 MW for ITBs in this case. Clear ion ITBs are observed at lower power levels following an LHCD prelude. The disparity in the resolution of the ion and electron temperature measurements makes it difficult to determine whether any slight differences power level required to obtain clear ITBs on the two profiles are due to different conditions for barrier formation in the ion and electron energy transport channels. ‘Strong’ ITBs are again seen in the negative magnetic shear cases with the application of high power heating in the $q$-profile scan. The lack of systematic correlation between the value of the negative magnetic shear in the plasma core and the ITB gradient supports the conclusion that the presence of an off-axis minimum in the $q$-profile is the critical component for optimising ‘wide’ ITB generation.

![Figure 9](image9.png)

**Figure 9.** Maximum value of $\rho_{ie}^*$ (= $\rho/L_{ni}$) versus coupled power for various $q$-profile and heating scenarios at $B=2.6T$. Analysis is restricted to barriers at wide plasma radius. An ITB is deemed clear if $\rho_{ie}^*>1.4 \times 10^2$. Annotations (a) to (d) refer to the pulses illustrated in figure 3.

![Figure 10](image10.png)

**Figure 10.** Maximum value of thermal stored energy as a function of coupled heating power. The open symbols represent plasmas with no clear ITB, shaded symbols are marginal cases and solid symbols represent clear ITBs. Annotations (a) to (d) refer to the pulses illustrated in figure 3. The solid line illustrates the approximate energy expected by the IPB98(y,2) scaling. The stars show the energy representative of the edge pedestal and the broken line is a fit to this data.
The effect of these ‘wide’ ITBs on the global plasma performance is illustrated in figure 10, which shows the peak thermal plasma stored energy as a function of coupled additional heating power. The thermal stored energy has been evaluated by integrating the measured plasma density (using LIDAR Thomson scattering and far-infrared interferometry) and temperature profiles (using LIDAR Thomson scattering and charge-exchange recombination spectroscopy) over the plasma volume. Up to 6MW of coupled heating power the plasma exhibits an L-mode edge (i.e. no large pedestal exists in the plasma periphery). At higher power Edge Localised Modes (ELMs) are observed as frequent (50-100Hz), small amplitude bursts in the Dα emission outside the plasma. Conventional ELMy H-modes exhibit improved confinement compared with L-mode plasmas, largely due to a reduction in transport localised at the plasma edge. To investigate the relative effects of the plasma core and edge confinement in this series of experiment, the magnitude of the thermal pressure pedestal has also been determined in the plasma periphery. The electron temperature was determined from ECE measurements at a plasma radius of 3.78m in the plasma equatorial plane (within 10cm of the last closed flux surface) and the plasma density was measured using the far-infrared interferometer, which has a vertical measuring chord at 3.75m. The ion temperature is assumed equal to the electron temperature, typically a good assumption outside the ITB radius. The effective pedestal energy is calculated by integrating the pedestal pressure over the entire plasma volume.

Figure 10 shows the effective pedestal energy as a function of heating power. The pedestal energy rise is roughly linear with heating power over the whole range of pulses, regardless of the presence or absence of ELMs. The absence of an abrupt transition from L-mode to H-mode confinement in this power range indicates that the achievement of confinement levels higher than expected by the ITER Physics Basis IPB98(y,2) ELMy H-mode scaling \[40\] at high power is largely attributable to the presence of an ITB rather than a large H-mode edge pressure pedestal.

In the pulses with up to 6MW of coupled heating power the overall confinement is about half of the value expected from the ELMy H-mode scaling. This data includes some ‘narrow’ ITBs generated at low power levels (e.g. case (ii) in figure 7) and confirms the previous assertion that they do not significantly improve the global plasma performance. In the region 8-13MW, where the ‘wide’ ITBs illustrated in figures 8 and 9 become evident, the global confinement time approaches the value given by the scaling. The exception being cases with a higher fraction of ICRH, which did not generate plasma performance equivalent to the NBI cases in these particular pulses. Above 15MW the confinement is comparable to or better than expected from the ELMy H-mode scaling due to the reduced transport at the ITB.

Plasmas with negative magnetic shear and coupled heating power in the range 15-16MW can exhibit extremely high confinement improvement. In this regime the pressure gradient continues to improve after the ITB is triggered to provide transient high plasma performance which can eventually destabilise MHD modes if the heating power level is not rapidly reduced. In these cases large ELMs develop, which destroy the ITB. The global confinement in this case
far exceeds the ELMy H-mode scaling. It is noteworthy that the power levels required to achieve ITBs capable of providing very high plasma performance is somewhat larger than the values at which an increase in the global confinement is first evident. At even lower power levels ITBs can be identified that hardly enhance the overall plasma performance at all. This leads to the conclusion that, although the power required to form ITBs in JET varies over more than an order of magnitude depending on the shape of the $q$-profile and the location of the ITB, a practical power requirement does exist for the class of ‘wide’ ITBs capable of delivering high plasma performance. This power requirement is lowest in plasmas with an off-axis minimum in $q$, but is less sensitive to variation in values of $q$ and magnetic shear near the plasma centre.

4. HIGH FUSION PERFORMANCE ITB PLASMAS

Previous JET high fusion performance ITB experiments have exploited the regime where a monotonic $q$-profile, or very weakly negative core magnetic shear, was generated with central $q$ slightly below 2 using Ohmic or low power on-axis ICRH preheating during the initial current ramp-up phase. This led to the production of 8MW of fusion power in plasmas using deuterium and tritium fuel [41]. In this scenario the level of additional heating power required to generate a localised transport reduction in JET increased with toroidal magnetic field strength. An additional increase in heating power was required to produce a region in the plasma interior of sufficiently low energy and particle transport that a substantial enhancement in the fusion yield was achieved. The power required to obtain high fusion yield in this way also increased with magnetic field strength and was typically about 30% higher than the lowest power level at which an ITB could be identified [42]. A graph showing the fusion yield achieved in a large number of JET deuterium plasmas in this regime at two values of magnetic field strength (2.55T and 3.45T) is shown in Figure 11, where it is plotted against the peak additional heating power used. The performance of the best pulses at $B=2.55$T with additional heating in the range 14-20MW easily exceeds the cases at $B=3.45$T with the same heating power, despite having a lower plasma current. This is due to the production of very steep internal pressure gradients at an ITB, which encloses a substantial plasma volume. The highest fusion yield was still achieved at high magnetic field and plasma current due to the improved MHD stability and confinement. Figure 11 shows, however, that the application of a high heating power level was required to access the high fusion performance domain at high magnetic field in the this regime.

The achievement of high fusion performance has been reported for JET plasmas with ‘highly’ negative central magnetic shear at $B=3.45$T [43,44]. The time evolution of such a pulse is illustrated in figure 12. In this scenario LHCD was applied soon after the plasma initiation until the main heating pulse at a typical power level of 2-3MW. As with the $B=2.6$T experiments described in the previous section, it was necessary to optimise the plasma initiation in order to achieve highly negative core magnetic shear [35]. This optimisation is firstly necessary to avoid a very fast initial current rise which can lead to MHD instabilities (e.g. external kink modes)
associated with the presence of a substantial current density in the plasma periphery as \( q \) reaches low order rational values near the plasma edge. However, it has been found that a significant fraction of the current can penetrate to the plasma centre if the current ramp is too slow. This may be due to a delay in generating a sufficiently hot plasma to prevent current diffusion into the core region. With the plasma initiation optimised in this way, and the subsequent application of low power LHCD, it is possible to generate a region of extremely large negative magnetic shear near the plasma centre. Figure 13 shows the \( q \)-profile generated at the end of the LHCD phase for the plasma presented in figure 12. In this class of discharge the current density at the plasma centre is close to zero [45].

As with the \( B=2.6T \) LHCD prelude case shown in figure 1, the formation of an ITB at wide radius is delayed with respect to the start of the main heating pulse, and is thought to correspond to the time when the minimum value of \( q \) reaches a low order rational value [44,37]. Correlation between ITB development and the appearance of particular \( q \) surfaces in the plasma has been observed in other devices [46,47]. A core ITB is visible on the electron temperature profile during the LHCD prelude phase, which persists during the early part of the main heating phase. Following the trigger event at \( t \approx 5.9s \) very steep gradients develop on the profiles of the ion and electron temperature, density and toroidal rotation. The plasma profiles before and after this event are illustrated in figure 14.

The plasma illustrated in figure 12 achieved high transient confinement compared with the IPB98(y,2) scaling (up to \( H_{IPB98(y,2)} \approx 1.9 \)) due to the steep pressure gradient at the ITB and the relatively large volume enclosed within it.
Such values are in the range required for currently envisaged ‘Advanced Tokamak’ scenarios for a fusion power plant. In fact, the local transport at the barrier may be too low in plasmas of this type. The tendency towards neoclassical particle confinement is expected to lead to the accumulation of high $Z$ impurities and helium ash in reactor conditions. Evidence already that the plasma impurity density increases in the core of this class of ITB plasma [48].

The operation at relatively low plasma current (up to 2.5MA) and the generation of a wide ITB with very low transport resulted in the achievement of the highest values of $N$ in JET at $B=3.45T$ ($\beta_N$ up to 2.4) since the introduction of the pumped divertor in 1992/3. This corresponds to $d_{st}=5$, which is of the order considered appropriate for ‘Advanced Tokamak’ regimes, but the values of $\beta_N$ still need to be further increased to fully investigate the desirable domain for a reactor. The high performance phase of the pulse illustrated in figure 12 ended with a large ELM, followed by a disruption. In previous high performance ITB plasmas without an LHCD prelude the achieved $\beta_N$ increased as the ratio of peak to average ion pressure ($p(0)/<p>$) was reduced$^{49}$. A reduction of this parameter is obtained if the radius of the ITB can be increased. Figure 15 show the ion pressure profile peaking factor plotted against $\beta_N$ for selected ITB plasmas at $B=3.45T$ for which the high performance phase was terminated with a disruption. The pulses shown without an LHCD prelude, as well as the LHCD case with highly peaked

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Figure 13. Target $q$-profile provided for main heating in high fusion yield experiments using an optimised plasma initiation and an LHCD prelude phase.

Figure 14. Radial profiles of ion and electron temperature and density at $t=5.63s$ and $t=6.63s$ (before and after the main ITB trigger event in the pulse shown in figure 12). The electron density was evaluated using the LIDAR system. The error bars indicate uncertainty in the absolute magnitude of each measurement and the radial smoothing.
pressure, were subject to a global pressure driven $n=1$ kink mode instability. The high $\beta_N$ examples are of the type shown in figure 12. MHD stability analysis indicates that these are also near to the stability limit and that, even if the large ELM could be avoided, a significant increase in $\beta_N$ would probably also require the pressure profile to be further broadened. Previous attempts to generate ‘wide’ ITBs that are capable of reaching high $\beta_N$ in roughly steady conditions have been successful without the use of an LHCD prelude, but so far only at lower magnetic field [50].

High performance ITBs in JET are observed at the same radial location on the ion and electron temperature profiles as well as those of the electron density and toroidal rotation. This conclusion is difficult to draw directly from the profile measurements at a particular time, as illustrated in figure 14, due to: the sparse radial measurements of the ion temperature and toroidal plasma velocity, both measured using charge-exchange recombination spectroscopy; and the limited spatial resolution and temporal frequency of the LIDAR Thomson scattering system. It is possible, however, to determine the location of the discontinuity in the gradient of a plasma profile associated with a transport barrier if the ITB expands such that the outer edge of the barrier region crosses a diagnostic measuring location. This technique locates the ‘foot-point’ of

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Figure 15. Ion pressure profile peaking factor plotted against $\beta_N$ at the disruption of selected high performance ITB plasmas.

Figure 16. Time evolution of the electron temperature at various locations and of other plasma parameters at an effective radius $R = 3.48$ m for a high performance ITB with no LHCD prelude. The expansion of the ITB ‘foot-point’ is indicated by the abrupt rise in $T_e$ sequentially at increasing radii. The abrupt rise in all parameters at $t = 6.25$ s and $R = 3.48$ m indicates the coincidence of the ITB ‘foot-point’ on the corresponding transport channels.
the steep gradient region at the ITB uniquely at a point in space and time. This method has been applied to data from the heterodyne radiometer and charge-exchange recombination spectrometer, but the resolution and frequency of measurements from the LIDAR instrument is insufficient for this to be used in the analysis of the density profile. Instead data from the far-infrared interferometer is used which has 8 chords passing through the plasma at different locations. The time evolution of measurements from each of these diagnostics, as well as a collimated neutron camera, is shown in figure 16 for a high performance ITB generated without an LHCD prelude. The expansion of the ITB is evident on the electron temperature where data from several radial locations are shown. Data are also shown for each diagnostic at the same effective plasma radius of $R \approx 3.48\,\text{m}$ (all data mapped to the outboard equatorial plane of the plasma). The ITB is evident at this radius on all profiles at $t = 6.25\,\text{s}$, signifying that the ‘foot-points’ of the barriers are coincident in space (within a few centimetres, which corresponds to the resolution of this technique) at this time.

The same technique has been applied to the plasma shown in figure 12. In this case the ITB location has been identified at various times using different measurement channels for each instrument. The results are shown in figure 17 where it can be seen that the barrier appears simultaneously on all channels, within the resolution of the measurements, and expands coincidentally on all profiles. The coincidence of measurement at various radii at the moment of ITB formation does not imply a very rapid expansion, but rather the appearance of an ITB at the outer edge of that region which results in a sudden increase in plasma parameters everywhere within it. The observation that the location of discontinuities on the various plasma parameter profiles are correlated can only be made in the case of ‘strong’, expanding barriers. It does not necessarily imply that the observation is general to all types of ITB observed in JET.

![Figure 17. Time and space evolution of the ITB ‘footpoint’ seen on various plasma parameters for the pulse illustrated in figure 12.](image1)

![Figure 18. Peak neutron yield plotted against maximum heating power for pulses with early heating at $B>3.3\,\text{T}$ with and without LHCD prelude (data from experiments with the JET MkIIGB divertor)](image2)
From figure 11 it can be seen that heating power in the range ≥20MW was required to generate transport barriers capable of producing high fusion performance at $B \approx 3.45T$ without an LHCD prelude. Figure 18 shows the peak neutron yield achieved as a function of the peak applied heating power compared for plasmas at $B \approx 3.45T$ with and without an LHCD prelude phase. With the LHCD, high fusion yield was obtained at lower heating power levels ($\approx 16$MW), consistent with the observation of a reduced power requirement to achieve an improved ion temperature gradient in the lower magnetic field experiments described in section 2. This observation confirms that the LHCD prelude is effective to modify the $q$ profile in such a way as to reduce the heating power requirement for access to ITB regimes that can deliver a substantial plasma performance improvement.

5. CONCLUSIONS
The influence of the $q$-profile shape on plasma performance has been explored on the JET tokamak for regimes that produce internal transport barriers. The $q$-profile was varied using LHCD during the initial current ramp-up phase of the plasma discharge. Experiments have been performed at magnetic fields of $B\approx 2.6T$ and 3.45T using combined NBI and ICRH over a wide power range to investigate the behaviour and fusion performance capability of the ITBs produced.

The investigations varying the $q$-profile shape and heating system parameters at $B\approx 2.6T$ show that ITBs can be generated in a wide variety of conditions. They can appear at various plasma radii, on various plasma parameter profiles, at various power levels and with various plasma heating systems and $q$-profile shapes. It is consequently very unlikely that a universal power threshold can be specified for the formation of internal transport barriers based on global plasma parameters such as those used to describe the conditions for the L-mode to H-mode transition[40]. However, only ITBs at large plasma radius contribute significantly to the enhancement of global plasma performance indicators such as fusion yield and stored energy. In this case some systematic trends can be seen in the conditions for ITB generation. In general barrier formation is favoured by $q$-profiles with a core region of negative magnetic shear and NBI dominated heating. The sensitivity to these parameters is consistent with theories that such transport reductions are due to the suppression of plasma turbulence by a combination of magnetic shear and plasma flow shear.

Heating power above about 10MW is required to access this class of ITB at $B \approx 2.6T$ in JET. It is possible to generate transport barriers with extremely large pressure gradients if the power level is raised significantly above the level at which the local confinement improvement is first observed. Although only a small number of pulses have been performed with a large fraction of ICRH, it is significant that the performance of the ITBs in these plasmas was inferior to cases with dominant NBI heating. ICRH on present devices is characteristically similar to the main heating mechanism in a fusion power plant, where fusion produced $\alpha$-particles will principally heat the plasma electrons without providing fuelling or applying torque. Consequently, further investigation of these differences is a critical issue if a basis is to be established for extrapolation to reactor scenarios.
At $B \approx 3.45$T a similar picture is seen where high fusion yield plasmas have been obtained at $\geq 16$MW in cases where an LHCD prelude has been used to generate negative magnetic shear near the plasma centre, while $\geq 20$MW is required in the cases with low central magnetic shear. The generation of plasmas with ‘wide’ ITBs giving good confinement at modest current in the negative magnetic shear regime has allowed an extension of the accessible $\beta_y$ at high magnetic field. This regime has the potential for exploitation in JET for high fusion performance and work has already begun to develop steady ITB scenarios following the LHCD prelude [51]. Issues such as impurity accumulation and $\alpha$-particle confinement in regions of very low poloidal magnetic field have still to be addressed, and such considerations may eventually rule out plasmas with ‘highly’ negative magnetic shear as candidates for steadystate application in an ‘Advanced Tokamak’. Nevertheless, they may have a role to play in transiently lowering the access power for ITBs if the conditions for barrier generation are challenging in a reactor.

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