Internal Transport Barriers in FTU at ITER relevant plasma density with pure electron heating and current drive

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Abstract - FTU significantly contributes to study ITER-relevant internal transport barriers (ITB), which require steadiness, high plasma density ($\overline{n}_e \geq 10^{20}$ m^{-3}), dominant electron heating and negligible momentum input. Its two radiofrequency systems, lower hybrid (LH, up to 1.9 MW at 8 GHz) and electron cyclotron (EC, up to 1.5 MW at 140 GHz), form and sustain steadily ITBs all along the heating pulse, $>35\tau_E$, and $\sim\tau_{R/L}$ at $n_{e0}>1.3\cdot10^{20}$ m^{-3} and $T_{e0}\approx5.5$ keV (τ_E=energy confinement time, $\tau_{R/L}$=ohmic current relaxation time, n_{e0} and T_{e0}=central density and electron temperature). Almost full current drive (CD) conditions are maintained with τ_E improved up to $1.6\cdot\tau_{E,ITER97\text{-}L}$. The proper safety factor profile q(r), almost flat or slightly reversed in the core ($q_{min}\approx1.3$) is established by off-axis LH-CD with important contribution of off-axis EC-CD or central counter ECCD. ITB radii r_{ITB}/a up to 0.67, and steadily up to ≥0.6 are obtained, by controlling the LHCD deposition profile. Peripheral LHCD is favored by lower q, and, at less extent, by broader $T_e(r)$ in turn established by off-axis EC heating. Good alignment of the bootstrap current is always obtained, with I_b/I_p up to 30%. The ITB strength is controlled by EC heating inside the barrier. Turbulence close to the ITB radius is depressed consistently with the improved confinement. Significant collisional ion heating inside the ITB ($\Delta T_{i0}/T_{i0}\geq35\%$, but still far from thermal equilibrium with electrons) and the almost neoclassical value of the ion transport indicate that e^--i^+ collisions do not affect the barrier dynamics. An anomalous inward particle pinch also at high density permits peaked profiles, $n_{e0}/\langle n_e\rangle\approx1.7$, in full CD plasmas despite the absence of the neoclassical Ware pinch.

1. Introduction

One of the most challenging goal for a tokamak reactor will be working steadily with a burning plasma, not only for a continuous energy production, but also to minimize the fatigue induced on the structure by pulse cycling. Most of the present effort in tokamak research is then devoted to find the best way to sustain indefinitely the plasma toroidal current in thermonuclear plasma without supplying magnetic flux from external coils, whose maximum flux swing is intrinsically limited. The external power that can be employed for this purpose is of course restricted because a high fusion gain Q (=fusion power/input power) must be maintained.

In the past years a plasma scenario, called ITB (Internal Transport barrier) other than the reference one for ITER [1], the ELMy H-mode (ELM=edge localized mode, H=high confinement) is proposing as a viable solution to this problem. Its higher energy confinement time, τ_E, allows relaxing the demands the plasma current, I_p, on which it depends almost proportionally, according to the commonly accepted scaling law [2]:

$$\tau_E^{98(y,2)}=H_H\cdot0.0562\cdot I_p^{0.93}\,B_T^{0.15}\,\overline{n}_e^{0.41}P^{-0.69}\,R_0^{1.97}\,\mu^{0.19}\,\kappa^{0.78}\varepsilon^{0.58}$$

H_H is the confinement quality factor, equal to 1 for the standard ELMy H-mode and usually >1 for ITBs. Units are s for τ_E, MA for I_p, T for the toroidal magnetic field, B_T, 10^{19} m^{-3} for the line-averaged plasma density, $\overline{n}_e$, MW for the total supplied power P, including external power and fusion produced α-particles, m for the tokamak major radius, R_0, while μ is the atomic mass of the main fuel, κ is the plasma elongation and ε is the tokamak inverse aspect ratio a/R_0 with a=minor plasma radius.

The chance of lowering the plasma current not only decreases the request for external current drive (CD) power sources but also raises the self-generated bootstrap current I_b ($I_b/I_p \approx \varepsilon^{0.5} ß_p \propto I_p^{-2}$, $ß_p$ is the poloidal beta $=2\mu_0\langle p\rangle/\langle B_{pa}\rangle^2$, μ_0=vacuum magnetic permeability, $\langle p\rangle$ average plasma pressure, $\langle B_{pa}\rangle$ average poloidal field strength at the plasma boundary due to plasma current), so further reducing the amount of the current to be externally driven. In turn, the larger spatial gradients naturally associated to the slower radial transport generate higher bootstrap currents. The limit to reducing I_p is given at last either by stability, which becomes more and more challenging as the ratio $\langle p\rangle/I_p$ increases, or by the confinement time degradation. These advanced regimes, the interest in which has been ever growing in the past years [3], are however not yet in the full maturity state. Worldwide investigations are under way on how to fulfill the many simultaneous requests of a reactor: high density regimes, $\bar{n}_e \approx 10^{20}$ m^{-3} $\approx$ the Greenwald density limit $n_{GW}=I_p/(\pi a^2)$ [4] [units, respectively, 10^{20} m^{-3}, MA, m, with a=plasma minor radius]; low normalized collisionality ($\nu^*_e \approx 2\cdot 10^{-2}$); quite peaked density profiles, ratio of the central to volume averaged density $n_{e0}/\langle n_e\rangle \approx 1.5$; very low or even no momentum input and then no central particle fuelling; control of both strength and radial size of the ITB in order to optimize the stability and confinement; direct electron heating by far dominant (from α-particles in a reactor); electron and ion (e^-, i^+) temperature equipartition, $T_e \approx T_i$. Comprehensive overviews of the ITB studies, both experimental and theoretical, are given in Ref. [5] and more recently in Ref. [6].

The FTU distinctive location [7, 8] in this panorama dwells in the possibility to deal with two major problems of ITER-relevant ITBs, namely the high density, the electron heating only by means of two radio frequency waves systems, lower hybrid current drive (LHCD up to 1.9 MW at f_{LH}=8 GHz) and electron cyclotron heating and current drive (ECH/ECCD, up to 1.5 MW at f_{ECH}=140 GHz), with no momentum injection. Most present day tokamaks instead, either cannot access the high-density regimes and/or employ prevalently ion-heating methods. The most widely used of these latter, the neutral beam injection (NBI), delivers also a substantial angular momentum, which in turn strongly determines the ITB properties. The tokamak Alcator C-mod also is in the same exploration area as FTU [9], but FTU only can drive actively a large fraction of the plasma current through LHCD. This gives it the capability to act on one very important ITB parameter, namely the radial profile of the safety factor q. This latter is linked intimately to the current density profile j(r): in a large aspect ratio circular tokamak. It is given by $q \approx rB_T/R_0B_p$ with r=minor radius, B_p=poloidal magnetic field, $B_p(r)=\mu_0/r\cdot\int j(x)\cdot x\cdot dx$. Furthermore, the FTU electron cyclotron heating (ECH) system gives also the opportunity to heat electrons directly and locally. Details on the plasma auxiliary heating available in FTU are given in Ref [7] and references therein.

The items more relevant to an ITER like ITB addressed by the FTU program concern the effect of high density and electron-ion (e^--i^+) collisional coupling on the mechanisms that produce e^-ITBs, and the possibility for an ion transport barrier to be built even by e^- heating methods only. This not only would be very important for the heating mechanisms working in ITER, but also for the physics of transport: it could clarify the role of the many candidates to stabilize the ion turbulence. Conversely FTU cannot deal with some questions such as stability at high ß, due to its high magnetic field, the role of fast particles and the divertor physics, FTU being a limiter device. Further details on FTU can be found in Ref. [7].

In this frame, this paper describes the ITB dynamics at high density, sec. 2; energy confinement, transport of heat and particles, and turbulence suppression are illustrated in sect. 3; sec. 4 deals with the important issue of the control of ITB size and strength; sec. 5 draws the conclusions.

2. High-density ITB formation and main characteristics

ITBs in FTU are mostly produced by off-axis LHCD during the I_p plateau, when the amount of the driven current is enough to create a safety factor radial profile q(r) with a central value $1<q_0<2$, followed by a low or even weakly inverted magnetic shear region with $q_{min} \approx 1.2$-1.3 (shear is defined as $s=r/q\cdot dq/dr$). The foot of the ITB is usually very close to the point where $s \approx 0$, which in turn is near to the point where $q_{min} \approx 1.5$ [10]. This agrees well with the dynamics of the ITB radius in JET, which is always connected to a low-order rational q-value surface all along the discharge evolution [11]. When $s \approx 0$ is attained at $r \approx a/2$ the q(r) profile resembles those typical of hybrid regimes [12].

Strongly reversed shear plasmas, as in JET [11] or JT60-U [13] with $q_{min}>2$, have not been obtained since LHCD in the very beginning of the discharge, when the q(r) is very far from the relaxed one, is hindered by various constraints [14]. LHCD can be applied only in the late phase of I_p ramp-up when j(r) is near to full relaxation, but no significant peculiarity of the ITB dynamics is found in this case.

FTU ITBs are usually quite steady, lasting as long as the LHCD pulse, because no mechanism is needed to prevent the current diffusion, which is mostly fixed by an external drive. With this q(r) tailoring sawteeth are stabilized and no harmful MHD (magneto hydrodynamic) activity develops, as double tearing modes, easily excited instead at the q=2 surfaces in strongly reversed shears with $q_{min}<2$. Some Mild m=1 activity may persist with possible beneficial effect against impurity accumulation.

The range of plasma parameters suitable for building an ITB in FTU is rather wide, namely $0.3 \leq \bar{n}_e \leq 1.1 \cdot 10^{20}$ m^{-3} ($\bar{n}_e/n_{GW} \geq 0.7$ and $n_{e0}/n_{GW} \geq 0.9$), $0.35 \leq I_p \leq 0.7$ MA, $5.2 \leq B_T \leq 7.2$ T [15], corresponding to $4.5 \leq q_a \leq 8.2 (q_a=q(r=a))$.

To sustain proper q profiles becomes, anyway, more difficult as density and/or plasma current rise, because the driven current fraction falls as $1/(I_p \cdot n_e)$. ECCD may offer a twofold valuable assistance to LHCD in these cases by both increasing the plasma temperature and then enhancing the LHCD efficiency [16], and can provide a contribution to the shape of j(r), which is locally very elevated though small as a whole ($I_{ECCD} \approx 10$-15kA). Either a central counter current stream (ctr-ECCD) or a proper off-axis CD (co-ECCD) can be essential for establishing the q(r) profile suited for an ITB, at high density or high currents [17]. Only with the aid of central ctr-ECCD we built the so far highest density steady ITB in FTU, whose main features are illustrated in Fig. 1. Indeed, in the simple heating scheme a remarkably weaker ITB is obtained, whereas with co-ECCD even sawteeth reappear and no ITB is built. Off-axis co-ECCD has proved very helpful instead to establish wide steady ITB radii, even though in this case is not so straightforward to discriminate between the local heating and CD. The different use of the ECCD tool is mostly linked to the LHCD radial profile. For the reason clarified above, high density ITBs are usually at large q_a (low I_p), that makes the LHCD profile closer to the axis [10]. This may leave flowing there too much current that has to be removed by ctr-ECCD. On the contrary, wide ITBs must have more peripheral LHCD, obtained in FTU by lowering q_a (raising I_p) as detailed in Sect 4, and prefer to be sustained directly in this task rather than deprived of central current, already set at the right magnitude. Both LHCD and ECH are of course also the e^- heating sources for the bulk plasma.

Figure 1 shows that an almost full CD ITB (loop voltage, $V_{loop}<0.2$ V, ohmic (OH) current $I_{OH}<20\%$ of the total) is sustained for the whole duration of the ECH pulse, for about $35\tau_E$ and one resistive OH current relaxation time, $\tau_{R/L}$, at $n_{e0} \approx 1.3 \cdot 10^{20}$ m^{-3} ($n_{e0}/n_{GW} \approx 0.9$) with a density peaking factor $f_{pk}=n_{e0}/\langle n_e \rangle > 1.6$ ($\langle n_e \rangle$ is the volume averaged density) and central e^- temperature $T_{e0}>5$ keV. Collisional coupling to ions in turn raises T_{i0}, the central ion temperature, by more than 35% with a 10 fold increase of the neutron rate, but it is not enough strong to bring T_{i0} close to T_{e0}, $T_{i0} \leq 1.3$ keV, since the $e^- - i^+$ thermal equipartition time, $\tau_{th,ei}$, is too long respect to τ_E, typically $\tau_{th,ei}/\tau_E \geq 6$. In turn, the enhancement of this latter over the scaling valid for FTU, namely ITER97-L [18], exceeds 1.6, and the bootstrap current fraction I_b/I_p is close to 22%. The strength of the barrier in term of the normalized inverse temperature scale length $\rho^*_T=\rho_{L,s}/L_T$, where $\rho_{L,s}$ is the Larmor radius of the ions moving at the sound velocity and $L_T=T_e/(dT_e/dr)$, is well above 0.02, against the threshold value $\rho^*_{T,th}=0.014$ [19], and stays constant as all the other ITB relevant features. In particular also the ITB radius, taken as the radius where $\rho^*_T=\rho^*_{T,th}\equiv 0.014$, remains fixed at $r_{ITB}/a \approx 0.35$, where q is≈ 1.5. The lack in FTU of a sharp jump in $T_e(r)$, clearly bounding the ITB, leads to define in such way r_{ITB}, that, nevertheless, always well represents the region where j(r) is largely modified by LHCD and the magnetic shear falls to small values.

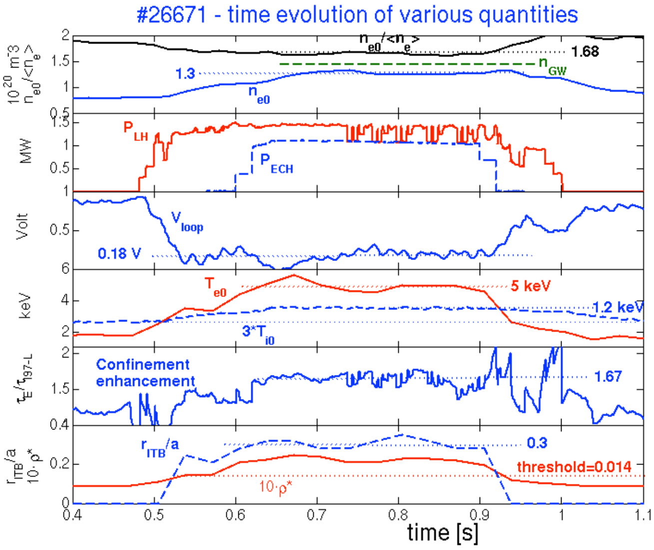

FIG. 1 – Time evolution of the most significant plasma parameters for #26671, the steady ITB at the highest density in ctr-ECCD configuration. I_p= 0.36 MA, B_T=5.3 T, q_a≈8

The ITB steadiness is a consequence of the almost stationary q(r) profile that in FTU is reached routinely in times definitely shorter than $\tau_{R/L}$, under almost full CD good conditions. Plasma resistivity then appears to affect only slightly the current diffusion when a local direct electromotive force dominates over the OH electric field. For the case of Fig. 1 ($\tau_{R/L}$≈0.3 s), a mildly reversed shape in the centre with q_0>1.5 and q_{min}≈1.2 is maintained all along the ITB phase, as derived from the profile of the hard X-ray (HXR, energy range 40-200 keV) perpendicularly emitted via bremsstrahlung of the LH generated fast e^- on the background ions.

This q(r) reconstruction is reliable close to full CD conditions [15, 20], and it is checked in FTU against the soft X-ray tomography (SXT) of the MHD activity [10, 21].

Equal stationarity is achieved for the quite wider ITBs (r_{ITB}/a≥0.65) that can be sustained at lower q_a (higher I_p). The relevant time evolution of the main quantities is given in Fig. 2 for discharge #27928, while in Fig. 3 the ITB e^- temperature profile is compared with that of the discharge presented in Fig.1. A co-ECCD configuration that shifts the deposition slightly off-axis (≈6-7 cm) with B_T fixed for on-axis EC resonance is requested. Density is lower than in Fig. 1 to balance the otherwise higher need of P_{LH} to attain a similar ratio of non-OH to the total current. This is due, beside directly to the magnitude I_p, to the more external LHCD deposition that inevitably spreads I_{LH} over a larger area, and then reduces the local weight of $j_{LH}(r)$, and to the decrease of I_b. The shape of both $j_{LH}(r)$ and $j_b(r)$ is shown in Fig. 4 for the discharges: in the wider ITB both currents are outward shifted. Automatically I_b aligns well to I_{LHCD}, supporting in turn the ITB, as requested for ITER, since gradients are built just around the LH deposition region.

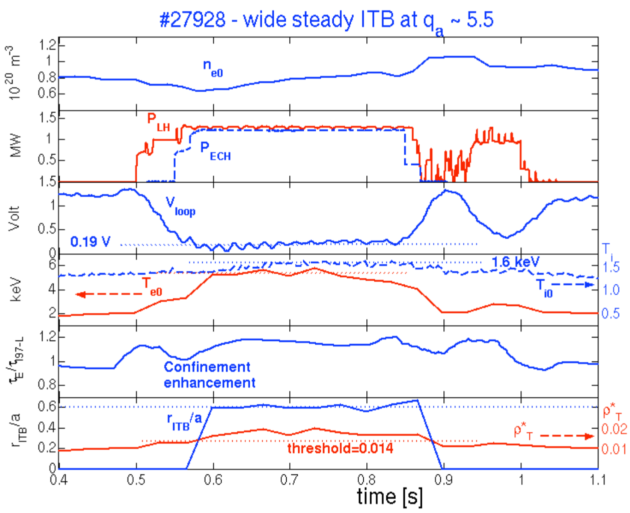

FIG. 2 - Time evolution of the most significant plasma parameters for #27928, the largest steady ITB. co-ECCD configuration slightly off axis (≈6 cm). I_p=0.51 MA, B_T=5.3 T q_a≈5.5

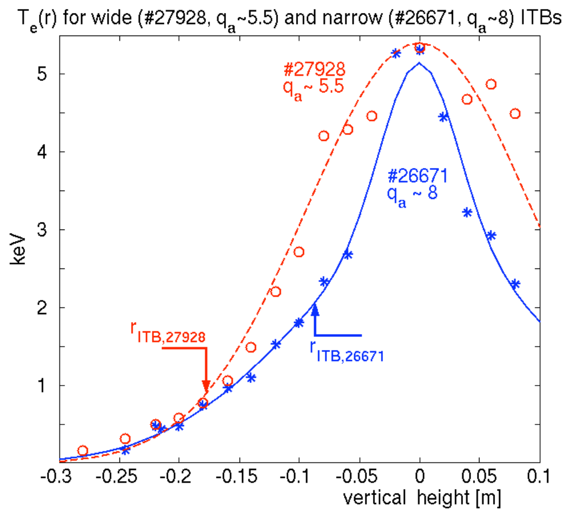

FIG. 3 – Comparison between the ITB $T_e(r)$ of discharges #26671 (Fig. 1) and #27928 (Fig. 2) with quite different ITB radius

3. Transport and confinement in ITBs

The drop of the electron heat conductivity in e^-ITBs is now well assessed [14, 10] from energy transport analysis. AS discussed in Ref. [15] a reduction of the gyroBhom (gB) coefficient in the frame of the mixed Bohm-gB model [22] is necessary for the strong ITBs to explain the experimental temperature profile. The possible onset of some process stabilizing turbulence explaining the transport reduction has been recently obtained in FTU [23]. Figure 5 shows the measurements carried out with a 2-channels fixed frequency reflectometer looking along two chords poloidally apart by $\Delta\theta$=5 deg, The fluctuation and coherence power spectra are compared respectively in top and bottom frames for a discharge very similar to that of Fig. 1 and one identical but for the fact that ITB is only marginal, if any, due to the lack of ECH power. Reflection occurs for both at the same radius equal to $r_{ITB,\#26669}$. Reduction of the fluctuation power and coherency for the ITB is evident.

A careful comparative analysis of the turbulence with and without ITB, given in Ref. [23], shows that for strong enough ITBs ($\rho^*_T \geq 0.02$) the suppression of turbulence affects a wide part of the frequency spectrum. For weaker ITBs, instead, the long wavelength low-coherence components, (f<20 kHz, $k_\theta\rho_i \approx 0.2$-$0.3$ (ρ_i is the ion Larmor radius)) are little affected whereas the high frequency quasi-coherent components are spread over a wider frequency interval extending up to 300 kHz. Poloidal rotation speeds up in both cases. In terms of modal structure, the ITB phase is richer in the higher poloidal numbers m (m=50 $k_\theta\rho_i \approx 50$ for f>80 kHz) and poorer in the low ones [24].

The scale length of the suppressed coherent modes is located at the boundary between the ITG (i^+ temperature gradient) and TEM (trapped e^- mode) modes. Calculations with KINEZERO code with collisional effect included show that collisionality is necessary to have stabilization in our case and that ITG should prevail on TEM [25].

The global confinement is affected of course by the onset of the ITB. The reduction of the electron transport following the turbulence stabilization is well illustrated in Ref. [10], where it is also put into evidence that the ion transport is less affected by the measured changes of the fluctuation: the same improvement inside the ITB is found for the cases shown in Fig. 5. However so far no insight has been possible on the turbulence status in regions where the ion transport is reduced.

The quite good confinement properties of the ITB region are well illustrated in Fig. 6 where the time traces of the ITB radius and strength and of the global confinement time are plotted together to P_{LH} and P_{ECH}. This latter, being absorbed well inside the ITB, affects its strength, but not the radius and leaves unchanged τ_E, despite a degradation is expected due to the total power increase, as shown by the trace of the ITER97-L scaling

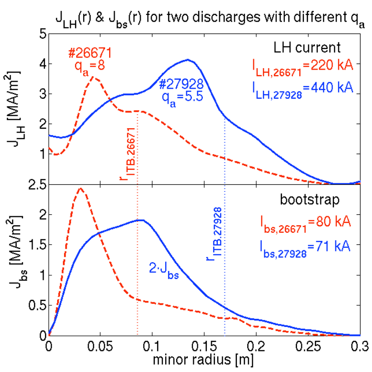

FIG. 4 - Radial profiles of the LH and bootstrap current for the same two discharges of fig. 3. The two currents are well aligned in both cases.

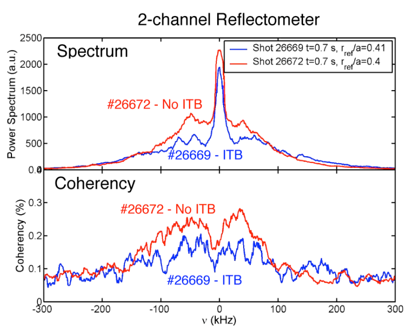

FIG. 5 - Fluctuation (top) and coherence (bottom) spectra for: #26669 without ITB (red) and #26672 with ITB (blue); $\bar{n}_e \approx 0.85 \cdot 10^{20}$ m^{-3} for both

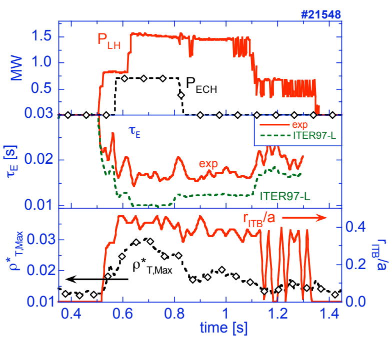

FIG. 6; #21548 time evolution; a) LH and ECH power; b) good energy confinement inside ITB; c) ITB radius (steady) and strength

The improvement of τ_E consequent to the transport reduction of both ions and electrons is shown in Fig. 7 as $H_{97}=\tau_E/\tau_{ITER97-L}$ versus $\rho^*_{T,max}$. The H_{97} values grow with $\rho^*_{T,max}$ reaching a maximum >1.6, and locate stably above 1 when $\rho^*_{T,max}$ exceeds the threshold value 0.014. No clear correlation has been found so far with the ITB size, i.e. with r_{ITB}, probably because it should be taken into proper account the balance of the power deposited inside or across the ITB border, i.e. in a good or a not very good confining region, as fig. 6 strongly suggests.

The need of peaked density profiles [6] for increasing the fusion reaction rate, $\propto n_e^2$, puts into evidence the importance of the particle transport if no significant direct central particle fueling is available, as it will be most probably for ITER. In the steady state scenario the inward particle transport will miss the important contribution of the Ware pinch, because the toroidal electric field, responsible for it, is almost vanishing. The first FTU results in full CD conditions are encouraging: in a high density ITB with negligible residual electric field, the density profile flattens only slightly respect to the OH phase, as shown in Fig. 1. The peaking factor $n_{e0}/\langle n_e \rangle$ remains above 1.7 higher than that quoted ($\approx$1.3) in other devices as JET [26], or Asdex U [27] for similar values of the non-dimensional collisionality parameter: $\nu_{eff}\approx 2$, $\nu^*_{ei}\approx 0.2$ at r_{ITB} in FTU, with $\nu_{eff}=\nu_{ei}/\omega_{De}$, where ν_{ei} is the e^--i^+ collision frequency and ω_{De} the curvature drift angular frequency and $\nu^*_{ei}=\nu_{ei}/\omega_b$, ratio of ν_{ei} to the trapped particle bounce frequency. The presence of an anomalous (turbulent) inward particle pinch in FTU is object of the paper [24] and has been found also in Tore Supra low-density plasmas [28]. Here we point out only the importance of the temperature gradients for the outward particle thermo-diffusion independently of the way they are built: only LH or LH+ECH. This can be seen in Fig. 8, where the temporal evolution of the applied RF power, the density peaking factor, the normalized temperature gradient and the line density are compared for two discharges. In #26669, $n_{e0}/\langle n_e \rangle$ falls to the level of #26670 only during the very short ECH pulse, after which it almost recovers the OH value. Correspondingly, $\rho^*_{T,max}$ after an initial jump reclines to a remarkably lower level, while density is very much similar to #26670.

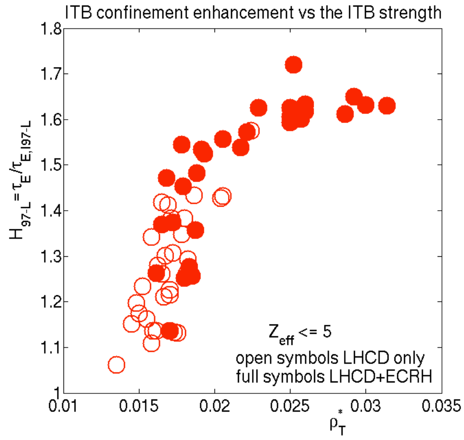

FIG. 7 Global energy confinement of the ITBs, shown as the enhancement over the ITER97-L scaling versus the barrier strength

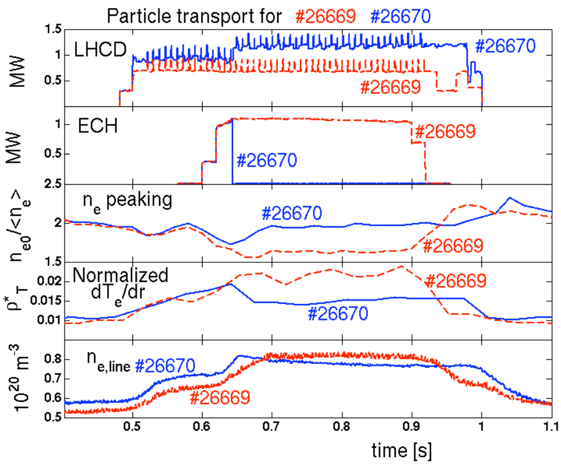

FIG. 8 - Time evolution of the most significant quantities for the particle transport for two discharges with different density peaking, but almost equal averaged density

4. Control of the ITB strength and radius

A full control of the ITB would require the ability to prefix both radius and strength of an ITB and it is highly desirable in a tokamak reactor [3, 6]. ITBs with a larger radius would enclose a larger plasma volume inside the improved confinement region and then increase the fusion performance. In addition, moderate gradients would enhance the MHD stability at large normalized beta, $\beta_N=\beta_T \cdot B_T/(I_p/a)$ ($\beta_T=2\mu_0\langle p \rangle/B_T^2$) and the bootstrap would be better aligned with a desirable broad total current profile [6]. In the frame of continuous maintenance of the optimal ITB profile all along the discharge it is very valuable to learn how to realize a desired r_{ITB} and $\rho^*_{T,max}$ during the current plateau, as we did in FTU, whereas the most ITBs are built only in the ramp-up phase.

The link of r_{ITB} with a low rational q value associated to a magnetic shear close to zero, pointed out above, provides a tool for controlling the ITB radius. Since in FTU the only way to modify the current profile at the extent and on the spatial scale requested is through LHCD, tuning the LHCD absorption profile just on the chosen q-surface is essential. This in turn can be done only by changing either the q(r) or $T_e(r)$ targets.

The first method is now well assessed: as the q_a of the discharge decreases the ITB expands due to the different LHCD absorption. Comparison between figs. 1 and 2 and fig. 4 provide a clear example. A multi-shot based analysis is done in Refs. [10, 15]. Here it is shown how r_{ITB} shrinks from $r_{ITB}/a \approx 0.6$ at $q_a \approx 4.5$ to $r_{ITB}/a \approx 0.3$ at $q_a \approx 8$. It is also pointed out the importance of a central counter OH current, induced to balance an over LHCD, in agreement with the assessment made above on the role of central ctr-ECCD, see also Ref. [17].

More recently also the possibility to shift the LHCD power radial deposition by modifying $T_e(r)$ has been recognized, even though at present it appears much less effective than varying q_a. This is inferred by comparing Fig. 9 with the HXR and T_e radial profiles for two discharges with same $q_a \approx 4.7$. The overall hotter profile of #27266 is due both to the higher total power ($P_{LH}/P_{ECH} \approx 1.2/0.8$ MW) and lower density ($\bar{n}_e \approx 0.64 \cdot 10^{20}$ m^{-3}) than in #27267 ($P_{LH} \approx 1.2$ MW, no ECH, $\bar{n}_e \approx 0.85 \cdot 10^{20}$ m^{-3}). This shift, though not big, may however become important for the quite low n_e and high T_e during I_p ramp-up phase, and it is indeed the main reason for the correspondingly wider ITBs in some cases at q_a=5.8 with very strong ITBs ($\rho^*_{T,max} \approx 0.5$) [10].

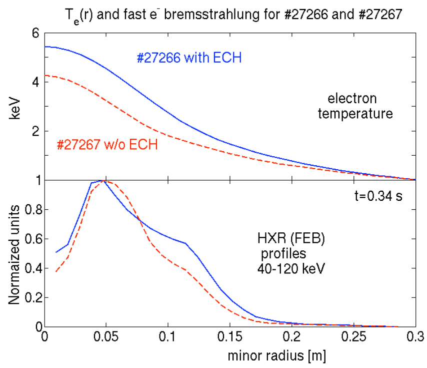

FIG. 9 - Radial profiles of electron temperature and FEB (40-120 keV) for two discharges. The higher and wider $T_e(r)$ shows an outward shift of the LHCD deposition

The attempt to actively control $T_e(r)$ by off-axis ECH succeded at low density, where it can lift still enough the temperature despite the larger heated volume. The strength of the barrier is instead controlled by direct heating inside the barrier that can be easily done in FTU with ECH for a proper value of B_T. This is evidenced by the frame b) of Fig. 6 showing a strong increase of the barrier strength, i.e. $\rho^*_{T,max}$, while the barrier size stays unchanged.

5. Conclusions

Stationary electron ITB, lasting longer than $35\tau_E$ and longer than $\tau_{R/L}$ are obtained in FTU up to $n_{e0} \approx 1.3\ 10^{20}$ m^{-3}, $B_T \approx 5.3$ T, indicating that operations close to n_e and B_T of ITER do not prevent ITBs to be achieved. Good relation between r_{ITB} and the weak shear region is found, χ_e for $r<r_{ITB}$ is reduced during the main heating.

The barrier size grows if q_a is reduced as a consequence of the outward shift of the LH deposition. Radial ITB sizes up to rITB/a≥0.6 are steadily sustained. To a smaller extent, the LH deposition can be varied also by varying the temperature profile.

The energy confinement time exceeds up to 1.6 times the ITER97-L thermal scaling. Ions are heated mainly by collisions, $\Delta T_{i0}/T_{i0}>35\%$ and about 10 times increase of the neutron fusion yield, but T_i stays substantially lower than T_e because $\tau_{ei,th} \approx 6\tau_E$ (typically 180 against 20 ms). The ion thermal diffusivity is reduced in high density ITBs respect to the OH phase, according to transport analysis. Ion confinement then remains good even under important collisional heating and the electron-ion collisions do not prevent e$^-$ITBs to be achieved.

A reduction of the amplitude of the fluctuation close to the barrier is observed to be well correlated to the onset of the ITB, together with a drop in the coherence between two separate reflectometer channels that is consistent with a reduction of the turbulent spectrum in the medium-low k_θ range. Code stability calculations indicate that the modes stabilized are ITG.

The particle transport studies are quite encouraging for achieving high-density, rather peaked ITBs plasmas in ITER without direct central fuelling and neoclassical inward pinch.

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