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Formation of Internal Transport Barriers and Its Impact on the JT-60U Plasmas

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Abstract

Formation of internal transport barriers (ITBs) in a plasma is a key to access to improved confinement. In JT-60U, there are two types of plasmas which are accompanied with ITB. One is a high β_p plasma and the other one is a reversed magnetic shear (RS) plasma. In RS plasmas, interesting features can be seen in spatial profiles of plasma parameters. For example, the location of the ITB can be strongly related to the location where the safety factor (q) becomes minimum, and the radial electric field (E_r) has strong spatial variation at the ITB. These features can be related to the physics mechanisms of confinement improvement at the ITB. On JT-60U RS plasmas, experiments to modify the toroidal current density or the toroidal rotation profiles, which are expected to influence the ITB structure directly or indirectly, have been carried out. And these modifications were fond to cause changes in the ITB structure. The results are shown in this paper.

Keywords:

JT-60U, ITB, reversed magnetic shear, LHCD, toroidal rotation

1. Introduction

The main objectives of the JT-60U project are studies on confinement, stability, steady state operation, heat and particle handling by a divertor configuration and understanding of underlaying physics in order to establish physics basis for the international thermonuclear experimental reactor (ITER) and steady state reactor such as SSTR [1]. For this purpose high performance plasmas, such as H-mode, high β_p [2], reversed magnetic shear (RS) plasmas [3] and the latter two combined with the H-mode [4,5], have been developed. What is essential to keep improved confinement in those plasmas is to maintain the internal and/or the edge transport barriers (ITB [3,6] and/or ETB). Especially a role played by the ITBs is a key to access high confinement. The formation of the ITB and its dynamics can be seen more clearly in the RS plasmas than those in the high β_p plasmas in JT-60U. Therefore, topics concerning to the ITB structure and its modification in the JT-60U RS plasmas are reported in this paper. In the second section, the tokamak, the heating system and diagnostics which are closely related to RS experiments shown in this paper are briefly described. In the third section, some general characteristics of the ITBs in the JT-60U RS plasmas are introduced. Also some typical achievements in performance so far in the JT-60U RS plasmas are shown. Following that, in the fourth section, experimental results on toroidal current density (j) profile modification via lower hybrid current drive (LHCD) and its impact to confinement are presented. In the fifth section, experimental results on relation between the ITB gradient and the radial electric field profile modification via external torque input control are

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shown. Lastly, the summary will be presented in the last section.

2. The Tokamak and the Heating/Current Drive and Diagnostics Systems

The JT-60U tokamak is a single null divertor tokamak device, the plasma major radius (R_p) 3 – 3.5 m, the plasma minor radius (a_p) 0.6 – 1.1 m and the maximum toroidal magnetic field (B_t) <~ 4 T at R = 3.32 m. On the tokamak a variety of heating and current drive systems are equipped. They are the neutral beam injection (NBI), the lower hybrid range of frequencies (LHRF), the electron cyclotron range of frequencies (ECRF) and the ion cyclotron range of frequencies (ICRF) systems. Two out of the four systems, the NBI and the LHRF systems, were mainly used in the experiments described in this paper. Detail of the two systems are:

1) The NBI system: Two types of ion sources are used, one is based on standard positive ion based ion source of which acceleration voltage is $\leq$ 110 keV and the other one is based on negative ion based ion source of which acceleration voltage is $\leq$ 500 keV. The purposes of NBIs are heating, toroidal torque input and current drive. Due to higher acceleration voltage, the negative ion source based NBI (N-NBI) is better in driving current.

2) The LHRF systems: Two types of multi-junction (MJ) LHRF launchers are installed. One which is installed in a port at poloidal angle $\theta \approx 45°$, consists of four rows of eight MJ modules which have 3 sub-waveguides. The other one, installed in a horizontal port, consists of two rows of four MJ modules which have 12 sub-waveguides.

Key diagnostics indispensable to this paper are, the mortional Stark effect (MSE) measurement for the safety factor (q) or j profile measurement, Thomson scattering measurement using Ruby and YAG lasers for the electron temperature (T_e) and the density (n_e) profile measurement, and charge exchange recombination spectroscopy (CXRS) measurement for the ion temperature (T_i) and the toroidal rotation (V_ϕ) profiles measurement.

3. General Characteristics of Reversed Magnetic Shear Plasmas in JT-60U

In an RS plasma, the j profile is a hollow one. Then the q profile comes to have a minimum inside a plasma. In this paper, the minimum value in the q profile and its location are denoted as $q_{\min}$ and ρ_{qmin}. With this q profile, the magnetic shear $s = \rho/q\ \mathrm{d}q/\mathrm{d}\rho$ becomes negative inside ρ_{qmin}. Here ρ is the normalized flux radius.

In JT-60U experiments, an RS configuration is usually formed by NBI heating at fast plasma current (I_p) ramp-up to retard current penetration by raising T_e and with help of the bootstrap current. However it should be noted here that LHRF injection at the I_p ramp phase also could form an RS configuration with quite a larger ρ_{qmin} [7]. Moreover, even at the current flat-top an RS configuration formation was observed with LHCD or ECCD although ρ_{qmin} is not very large.

Usually a moderate to a high performance RS plasma is formed by injecting adequate NB power into an RS configuration. Typical profiles in an RS plasma are shown in Fig. 1. As shown in the top box, the q

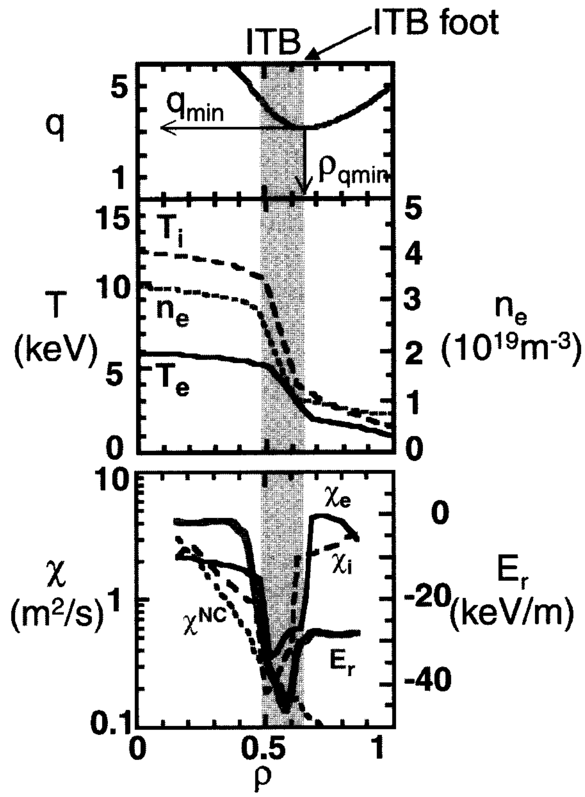

Fig. 1 Plasma parameter profiles in a typical RS plasma in JT-60U. The top box; the q profile and the minimum ($q_{\min}$) in the profile and its location (ρ_{qmin}) are schematically shown. The second box; the T_e, T_i and n_e profiles. The bottom box; the χ_e and the χ_i profiles with the neoclassical thermal diffusivity profile for a reference, and the E_r profile. The region of the ITB is schematically shown with the gray area. Also the ITB "foot" location is also indicated schematically by an arrow.

profile is increasing toward the center from ρ_{qmin}. In the second box the T_{e}, the T_{i} and the n_{e} profiles are plotted. At a certain position in a plasma the gradient of all these profiles changes quickly, and the profiles build up toward the center. The position is referred to as the ITB "foot" (ρ_{foot}) in this paper. It should be noted that although ρ_{foot} could vary in each profile somehow it is found to be strongly related to ρ_{qmin} in the JT-60U RS plasmas [8]. However this steep gradient in the profiles usually does not continue up to the center, but becomes much less steep, or even flat, at a certain point (referred to as the ITB shoulder). A region between the ITB foot and the shoulder is called the ITB region. As shown in the bottom box, both the electron and the ion thermal diffusivities (χ_{e} and χ_{i}) show sudden drop at this region. Actually in this case, χ_{i} is found to be as small as the neo-classical value as shown in the figure. However, as expected from the less steep temperature and density profiles, the thermal diffusivities go up inside the ITB shoulder. Also, the radial electric field (E_r) profile is shown in the bottom box and indicates strong spatial variation at the ITB region.

Formation of the ITB can bring benefits from a view point of steady state advanced tokamak operation, such as high confinement and large bootstrap current fraction to I_{p} (f_{BS}). In JT-60U, there have been many achievement in view of plasma performance owing to ITB formation. Some of them are briefly described as follow;

1) High equivalent D-T fusion gain ($Q_{\mathrm{DT}}^{\mathrm{eq}}$)

The RS plasmas in JT-60U have been proved to have potential to achieve high fusion plasma performance, and in the campaign in 1998 the value of $Q_{\mathrm{DT}}^{\mathrm{eq}}$ has reached up to 1.25 at I_{p} = 2.6 MA and the toroidal magnetic field at the plasma center (B_{t0}) = 4.4 T [5]. The edge of the RS plasma was in the L-mode. In order to obtain high performance, it was found to be advantageous to form the ITB early in the I_{p} ramp-up phase. In order to maintain the ITB at large radial location which gives better performance, a certain amount of power needs to be applied. Otherwise, the ITB shrinks and the performance becomes lower or is even lost. On the other hand, since the current keeps penetrating into the plasma center, the minimum of the safety factor (q) profile (q_{min}) keeps decreasing.

2) High confinement RS plasma with H-mode edge

One of the key concept in the advanced tokamak operation is to utilize bootstrap current as much as possible for non-inductive operation [1]. On JT-60U, optimization of an RS plasma has been pursued to raise f_{BS}. This effort has succeeded in producing an ELMy H-mode RS plasma which has a large f_{BS} of about 80%, with I_{p} = 0.8 MA, B_{t0} =3.4 T, q at 95% flux (q_{95}) = 9.3 and the triangularity (δ) = 0.43 [9]. In the plasma, the confinement enhancement factor to the H-mode thermal confinement scaling ($HH_{98(y,2)}$) reached 2.1 – 2.3, and that to the L-mode scaling ($H_{89\mathrm{P}}$) was 3.3 – 3.8, with the normalized β (β_{N}) = 1.9 – 2.2. Owning to the large f_{BS} of 80% and the P-NB driving the rest 20% of I_{p}, full current drive was maintained with such a highly improved confinement state for 2.7 s (about 6 × τ_{e}).

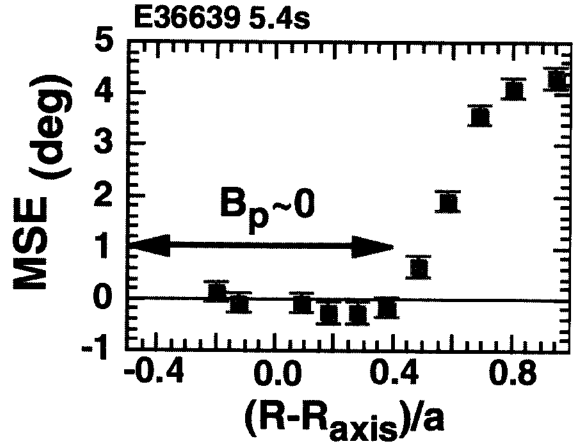

Fig. 2 The MSE polarization angle ($\gamma^{\mathrm{MSE}} \sim B_{\mathrm{p}}/B_{\mathrm{t}}$) vs radial location (distance from the magnetic axis normalized by the minor radius) of the measurement. Almost zero γ^{MSE} in the central region suggests almost zero B_{p} or j.

In addition to the performance improvement shown above, a very interesting structure is found in the RS plasmas, that is a current hole [10,11]. In an RS plasma, almost zero poloidal field (B_{p}) is observed in the central region (Fig. 2). This in other words means that j is very small or even zero in the region. It should be stressed that this finding suggests a possibility of a stable tokamak operation with very small or even zero toroidal current in the core region.

4. Modification of the Current Profile and the ITB Location

As noted in the previous section, the ITB foot is found to be strongly related to the position of q_{min} in JT-60U RS plasmas. It was also found that expansion of ρ_{qmin} (or ~ ρ_{foot}) gives improvement of confinement performance [12]. Therefore it is an interesting question what happens when the q_{min} location is changed via some active external scheme. Modification of the ITB location by modifying target q profile by means of LHCD was already reported [7]. Also quasi-steady state sustainment of the RS configuration and ITB by external LHCD was demonstrated [13]. On the other hand, quasi-steady sustainment of the RS configuration and ITB with high bootstrap current fraction (~ 80%) was also reported as described in the previous section [12]. However, in the former cases the bootstrap current fraction was not as high as expected in the future machines, and in the latter case there was small room for external NBCD to work and also q_{95} is too high in view of economical operation of a future rector. Therefore, in order to explore current drive and current modification effect with high (but not too high) bootstrap current fraction at lower q_{95}, LHCD was applied to high β_N RS plasmas at low q_{95} regime.

The experiment was done at B_{t0} = 2.5 T with deuterium gas. The waveforms of typical plasma parameters are shown in Fig. 3. As shown in the figure, the target RS plasma was formed by P-NB heating. Then P_{NB} was reduced and kept constant for clearer measurement of the LHCD effect. It is noted that just at the P_{NB} step-down a mini collapse occurred and a large drop in β_N at ~ 6.3 s is attributed to the mini collapse. After P_{LH} injection, the surface loop voltage (V_ℓ) started decreasing, which indicated that the inductively driven plasma current was being replaced with non-inductive current, that is LH driven current. In the bottom box n_e at ρ ~ 0.2, 0.6 and 0.8 are shown, which correspond to the central region, near the ITB foot and peripheral region respectively. The temporal evolution of n_e at ρ ~ 0.6 starts increasing at about 6.7 s, which indicates this position is coming into the ITB region, or in other words the ITB region is expanding. Change in the profiles is shown in Fig. 4. In Fig. 4 (a), the q profiles at 6.4 and 7.5 s are shown. As shown in the figure, the location of the q_{min} shifts outward. This shift is attributed to off-axis LHCD. In Fig. 4 (b), n_e profiles at 6.5 and 7.5 s are plotted. As mentioned before, expansion of the ITB foot is seen in the change in the profile. It is noted that although the central density decreases, the expansion of the ITB foot compensates to keep high electron population in the core region. Furthermore it should be noted that expansion of the ITBs in the T_e and T_i profiles is also observed.

In the discharge, V_ℓ reached very closed to zero but not completely, as shown in Fig. 3. Demonstration of full current drive in this regime is very interesting and important in relevance to the ITER steady state operation. In order to access full current drive, I_p was

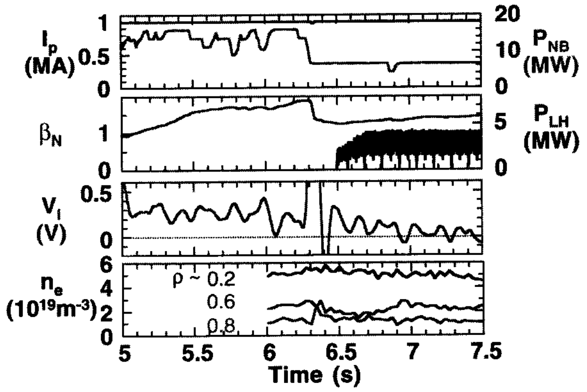

Fig. 3 A typical waveforms of LHCD on a high β_N RS plasma at I_p = 1 MA and B_t = 2.5 T. The LH power was modulated 0 to 100% with the duty of 90%.

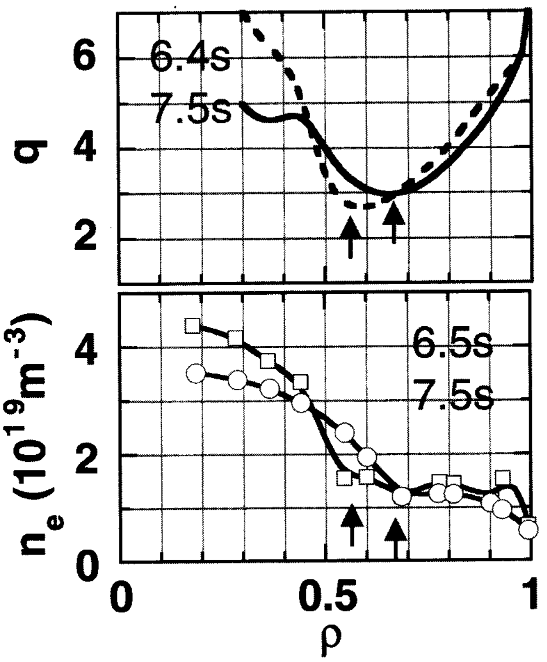

Fig. 4 Profiles of q and n_e at 6.4 s/6.5 s and 7.5 s (solid curve, circles).

reduced to 0.9 MA and also N-NB was injected. The role of N-NB injection is not only current drive but also effective central heating. The waveform of the discharge is shown in Fig. 5. Injection scenario of P-NB and LHW is similar to that shown previously, but N-NB injection started prior to LHCD by 0.2 s. As shown in the figure, V_ℓ dropped to zero and then even negative, showing full current or even over drive. Similar to the previous case, temporal evolution of n_e near the ITB foot suggests expansion of the ITB foot. As the results of the

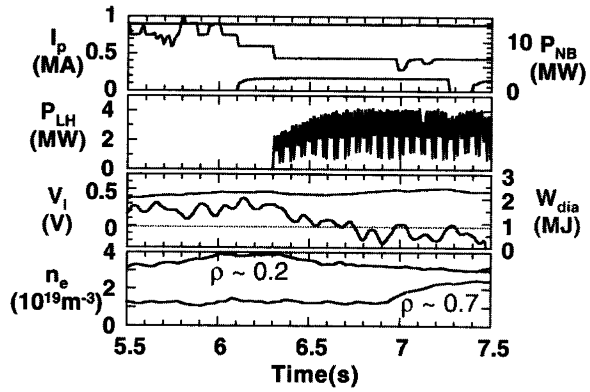

Fig. 5 Wave forms of the discharge in which full current drive was achieved simultaneously with high confinement ($HH_{98(y,2)}$) ~ 1.4 at high normalized electron density ($\bar{n}_e/n_{GW}$). Also in this shot P_{LH} was modulated 0 to 100% with the duty of 90%.

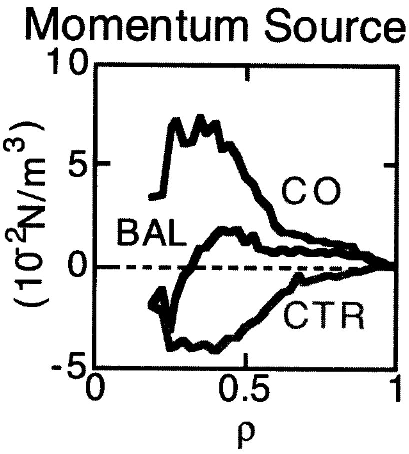

Fig. 6 Typical torque (momentum) input profile, balanced, co- and counter directional, in the discharge shown in Figs. 7 and 8.

ITB expansion, the confinement in the plasma was improved further and some of the plasma parameters reached to the region which is relevant to the ITER with full current drive condition. The confinement improvement factor to the H-mode ($HH_{98(y,2)}$) is about 1.4 and the fraction of line averaged density ($\bar{n}_e$) to the Greenwald density (n_{GW}) is about 0.82. It should be stressed that simultaneous achievement of full current drive at high confinement and high density is a new regime in JT-60U operation. Although q_{95} = 6.9 in the discharge is still not very low, it is approaching to the range that is expected in the ITER steady state operation scenario, that is q_{95} = 4 – 5.

5. Modification of the Toroidal Rotation Profile and the ITB Strength

As described in the second section, a large spatial variation in the radial electric field profile is often observed in RS plasmas in JT-60U. Theoretical predictions show that competition between the shearing rate ($\omega_{E\times B}$) and linear growth rate (γ_L) of a micro-turbulence is a key in the turbulence suppression. Here, $\omega_{E\times B}$ is written [14];

$$\omega_{E\times B} = \frac{(RB_\theta)^2}{B} + \frac{\partial}{\partial\psi}\frac{E_r}{RB_\theta} . \qquad (1)$$

Here, B_θ is the poloidal magnetic field and ψ = constant is a magnetic flux surface. Therefore, spatial derivative of E_r (it will be expressed as E_r' later on in this paper) would be a key parameter. On the other hand the radial electric field is written as;

$$E_r = \frac{\nabla p_i}{Z_i e n_i} + V_\phi B_\theta - V_\theta B_\phi . \qquad (2)$$

Here, p_i, Z_i, n_i, V_ϕ and V_θ are the pressure, the charge number, the density, the toroidal and the poloidal rotation velocities of an ion species, and B_ϕ is the toroidal magnetic field. As noticed in the equation, there would be a room to modify the first and the second terms externally. That is, control of the heating to modify the first term and the control of the toroidal torque input to modify the second term. On the other hand, the third term is expected to be determined mostly by neo-classical force balance. The idea is to utilize four tangential P-NB lines on JT-60U to control torque input profile to act on the radial electric field via modification in the second term in the eq. (2).

The experiment was carried out with I_p = 1.5 MA and B_t = 3.5 T and the working gas was deuterium [8]. In the discharge, the input toroidal torque profile was changed by changing combination of tangential beam units. Three input momentum profiles that were used in this experiment are shown in Fig. 6. The combination which gives co-directional torque is referred to the "CO" injection. On the other hand, the combination which gives counter-directional torque is referred to the "CTR" injection. The combination which gives almost zero torque relative to the other two cases is referred to the "BAL" injection.

The target RS plasma was initially formed with the BAL injection, then switched to the CO injection, then the CTR injection and finally to the BAL injection again. Since the interest is on the ITB structure, the gradient of T_i at the ITB is taken as a measure of the ITB characteristics. In Fig. 7, the temporal evolution of the T_i gradient at the ITB is shown, and the combination of the torque input is also shown schematically at the top of the figure. As indicated in the figure, at just after 6.25 s the T_i gradient started decreasing, and at just after 6.65 s the gradient started increasing again, or recovering. Note that the NB power was kept almost constant. The confinement improvement (H_{89P}) was evaluated as 2.2, 1.6 and 1.8 at 6.25, 6.6 and 7.0 s respectively. The change in the toroidal rotation profile is shown in Fig. 8 (a). As shown in the figure, in the case of the CO injection V_ϕ builds up in the central region while the profile flattens at the ITB foot just before the degradation in the T_i gradient starts (~ 6.25 s). On the other hand, in the case of the CTR injection followed by the BAL injection, a flat V_ϕ profile at the ITB foot comes to have a positive gradient just before the recovery of the T_i gradient starts. The gradient of E_r at each timing was calculated and shown in Fig. 8 (b). Together with E_r', gradients of the first and the second terms in eq. (2) are shown, labeled with "∇p" and V_ϕ respectively. As seen in Fig. 8 (b), towards the degradation in T_i gradient ,what changes most clearly in the E_r' profile is a decrease in E_r' at the ITB foot. On the other hand, when T_i gradient is about to recover, increase of E_r' at the ITB foot is observed. Both of them are attributed to the change in the "V_ϕ" due to change in the V_ϕ profile. These results suggest that the E_r gradient at the ITB foot is a key parameter to control the confinement of whole the ITB region.

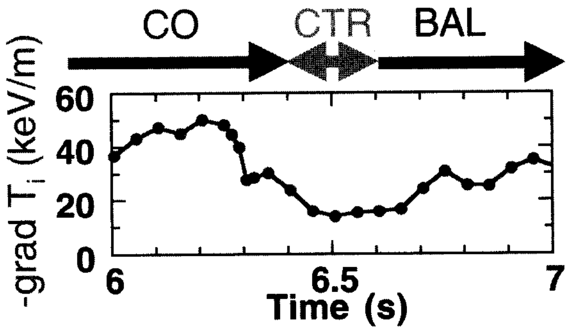

Fig. 7 Temporal evolution of the T_i gradient at the ITB when torque input profile was changed as "CO" ⇒ "CTR" ⇒ "BAL".

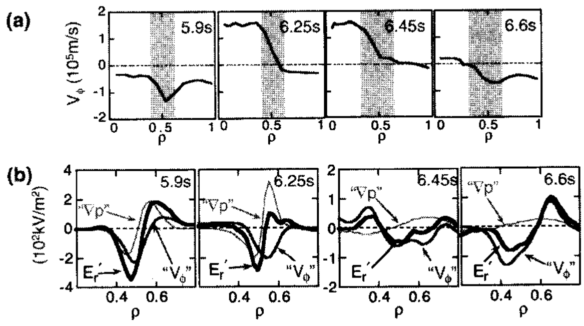

Fig. 8 (a); The V_ϕ profile at 5.9, 6.25, 6.45 and 6.6 s in the discharge shown in Fig.7. (b); The gradient of E_r profile and its terms dominated by ∇p and V_ϕ, labeled with "∇p" and "V_ϕ" respectively, at 5.9, 6.25, 6.45 and 6.6 s in the same discharge.

6. Summary

General feature of the reversed magnetic shear plasma in JT-60U are descried;

- the location of the ITB foot is strongly related to the location at which q becomes minimum,
- strong spatial variation in the E_r profile is often found to be formed at the ITB region,
- the temperatures and density profiles often become flat in the central region.

Two topics relating to the first and second feature are described in this paper. The first one is concerning to the current profile and the ITB location. It was shown that by extending the location of the minimum q by means of non-inductive off-axis current drive using lower hybrid waves the location of the ITB foot expanded outward. By combining the off-axis LHCD and the current drive by N-NBI, full current drive of I_p = 0.9 MA with high confinement of $HH_{98(y,2)}$ of about 1.4 at high density normalized to the Greenwald density $\bar{n}_e/n_{GW}$ of about 0.82 and q_{98} of 6.9.

The other one is concerning to modification of the E_r profile via control of the toroidal torque input in order to affect the transport characteristics of the ITB. It was demonstrated that by changing the toroidal torque input profile by choosing combination of four tangential NB lines the T_i gradient at the ITB changed. As the result of the modification in the V_ϕ profile, the E_r profile and thus the gradient of E_r profile was changed. It was fond that prior to the degradation of the T_i gradient, the E_r gradient at the ITB foot became small due to flattening of the V_ϕ profile at the ITB. On the other hand, prior to the steepening of the T_i gradient increase of the E_r gradient at the ITB foot was observed. These results suggest that the E_r gradient at the ITB foot is a key parameter to control the confinement of whole the ITB region.

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