Plasma Potential Well and Velocity Shear Layer at the Edge of Reversed Field Pinch Plasmas

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(Received 30 June 1997)

The first evidence of an electrostatic potential well at the edge of plasmas confined in a reversed field pinch configuration is reported, based on measurements made on the RFX experiment. The radial width of the well decreases with plasma current, whereas the potential drop is a few times the electron temperature at the edge, almost independent of plasma current. The resulting radial electric field points inward at the edge and this behavior has been related to finite Larmor radius losses as in tokamaks. Because of the spatial structure of the plasma potential, a naturally occurring double velocity shear layer has been identified at the edge with a shear value comparable to that of tokamaks and stellarators. [S0031-9007(97)04797-2]

PACS numbers: 52.55.Hc, 52.35.Ra, 52.40.Hf, 52.70.Ds

In recent years, the structure of the radial electric field at the edge of tokamaks and stellarators has been investigated to establish the stabilizing effect on the transport driven by electrostatic fluctuations [1]. Since in most reversed field pinch (RFP) experiments [2] the particle transport is driven mainly by electrostatic fluctuations [3–6], there is nowadays a growing interest on the structure of the plasma potential and of the radial electric field at the edge to study the possibility of achieving enhanced confinement regimes also in RFPs.

A common feature of tokamak and stellarator experiments is a radial electric field pointing inward right inside the last closed flux surface (LCFS) and outward in the scrape off layer (SOL) generated by limiters (see, for example, [7]). In tokamaks, this radial behavior has been interpreted with different mechanisms, including ion losses due to finite Larmor radius (FLR) effects [7,8]. As a result, a naturally occurring $\mathbf{E} \times \mathbf{B}$ velocity shear layer, with thickness $\sim$5–10 mm, has been identified in tokamaks and stellarators in the region across the last closed flux surface, with velocity shear $\sim 10^5$–10^6 s^{-1}. In tokamaks, nonambipolar ion losses near the plasma periphery are also proposed as a possible mechanism for the establishment of transport barriers at the edge with scale lengths comparable to the poloidal gyroradius [1]. In RFPs, the structure of the plasma potential at the edge, measured in different devices, reveals that the radial electric field is directed outward in the SOL [3,9] and inward right inside the plasma surface [5,9–11].

This behavior was already pointed out to exhibit a surprising analogy to that found in tokamaks and stellarators [11], despite the different physics expected in a RFP configuration. In fact, the large magnetic fluctuation level ($b/B \sim 1\%$) and the wide spectrum of unstable MHD modes characteristic of this configuration result in a wide stochastic region where the higher electron diffusivity should give rise to an outward directed radial ambipolar electric field to restrain the electrons. According to the theory [12], in a completely stochastic plasma with Maxwellian electron distribution function in the absence of other mechanisms of nonambipolar transport, the radial ambipolar electric field is related to the electron temperature (T_e) and density (n_e) gradients by

$$E_r = -\frac{T_e}{e}\left(\frac{\nabla n_e}{n_e} + \frac{\nabla T_e}{2T_e}\right). \tag{1}$$

It is worth mentioning that in REPUTE 1, where the plasma potential in the plasma core has been measured, the radial electric field is outward directed inside the toroidal field reversal surface, confirming that the magnetic field is fully stochastic [13].

It is still an open question if the outer region, between the toroidal magnetic field reversal surface and the wall, is also stochastic. On one hand, the inward directed radial electric field found at the edge could be interpreted as a proof of a nonstochastic behavior of the magnetic field lines. On the other hand, in some experiments the presence of superthermal electrons at the edge has been interpreted as a proof of the extension of this stochastic region up to the wall [9,14,15].

In this Letter, we report the edge structure of the plasma potential in the RFX reversed field pinch experiment ($R = 2$ m, $a = 0.457$ m) [16]. Our results suggest that the negative E_r at the edge is due to FLR losses. As a consequence, it appears to be a general property of magnetically confined plasmas, not related to magnetic field stochasticity.

In RFX the structure of the electric field at the extreme edge was measured in the past at $I_p \sim 500$ kA and a first indication of a change in sign of the electric field at the edge was reported, based on the behavior of the plasma flow direction [11] and on the inversion of impurity drift velocity measured by the Doppler shift of different ionization state impurity lines [17]. In [11], the velocity shear value at the extreme edge was also reported. This value was found comparable to that reported for tokamaks and stellarators [7].

New measurements of electron temperature, ion saturation current (I_s), and floating potential (V_f) have been performed in RFX hydrogen plasma discharges using

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Langmuir probes mounted on three different diagnostic systems inserted through a port in the equatorial plane. One of the systems has already been described [11]; for the purpose of the present Letter, it is worth mentioning that the second one is an equipment analogous to that in [11] with the protecting head made of boron nitride instead of graphite and the third one is a small size fast insertion arm. The discharges had currents ranging between 300 and 650 kA, and at all plasma currents the values of the pinch parameter Θ and reversal parameter F were between 1.4 and 1.5 and between -0.2 and -0.1, respectively, kept constant at different currents. As a consequence, the toroidal magnetic field reversal surface was located at $r/a \approx 0.85$–0.9, independent of the plasma current. In discharges with $I_p \geq 500$ kA (referred to as high current discharges in the following) a graphite mushroom-shaped limiter equipped with Langmuir probes was used [18] and temperature and density profiles were obtained up to 10 mm inside the first wall. The maximum insertion was limited in order to protect the equipment from the high energy fluxes (~ 100 MW/m^2) in long pulses (~ 100 ms). The floating potential up to 40 mm insertion was measured in high current discharges by a reciprocating Langmuir probe diagnostic, made of carbon fiber, which sweeps the plasma in some tens of milliseconds minimizing the thermal load on the graphite and the plasma perturbation. The plasma potential at high current has thus been derived from direct V_f measurements and from the T_e profiles extrapolated from the temperature gradient in the extreme edge region. At low plasma current, in the range of 150–400 kA, a boron nitride head has been inserted up to 60 mm inside the first wall. T_e, I_s, and V_f have been simultaneously measured by Langmuir probes operated in single probe configuration with a sweep frequency of 1 kHz. The local plasma perturbation during these measurements has been already discussed in [11]. Indeed, since the head was kept floating, no major perturbation on the plasma potential structure is expected. The fact that measurements at the extreme edge at 500 kA made with all three systems, including the fast insertion probe, show similar values supports this argument. It should be pointed out that the measurements at the lowest currents (150–200 kA) are taken during the final part of decaying discharges with 300–400 kA of peak value. The values of the reversal parameter F and of the pinch parameter Θ (which describe the internal distribution of the magnetic field [2]) were constant also during the current decay.

All of the measurements were made keeping the probe protected from the superthermal electron flow [9], that is, where the electron energy distribution function was not distorted. The plasma potential was derived as $V_p = V_f - \alpha T_e$, where V_f is the floating potential. The factor α, calculated by neglecting secondary electron emission and taking into account the different collection surfaces for electrons and ions, is $\alpha = -2.5$ [9]. The resulting V_p profiles are shown in Fig. 1 as a function of the normalized radius r/a for three different plasma current ranges. The $r/a = 1$ surface has been identified as the location where the probe is at the same potential of the wall, that is, where $V_f = 0$, which represents the transition point from the SOL to the confined plasma. This choice allows comparison of V_p profiles for discharges which had slightly different plasma column horizontal shift. The edge electron temperature profiles for the different plasma current regimes are shown in Fig. 2, showing that the temperature gradient rises with plasma current. Data from interferometer and Langmuir probes show that the density profile is flat in the plasma core [9], so that the pressure gradient develops mainly in the edge region.

The data in Fig. 1 show the presence in the confined plasma near the first wall of a plasma potential well. The depth of the well ΔV_p, almost independent of plasma current, has an average value of $(4.5 \pm 0.4)T_e^*$, where T_e^* is the electron temperature measured at the bottom of the potential well, as shown in Fig. 3. The distance Δx of the

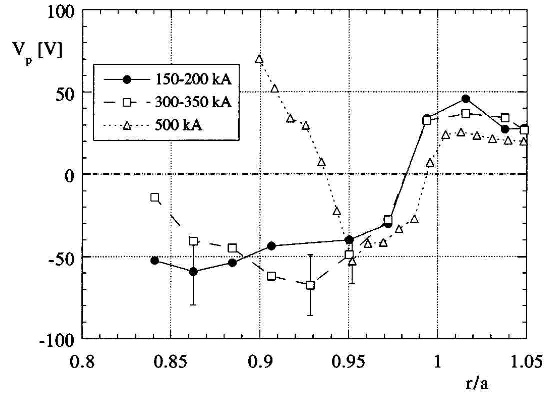

FIG. 1. Plasma potential plotted as a function of the normalized radius for three ranges of plasma current. Each point is an average over multiple measurements. For each curve a typical error bar is plotted, which gives the standard deviation of the data.

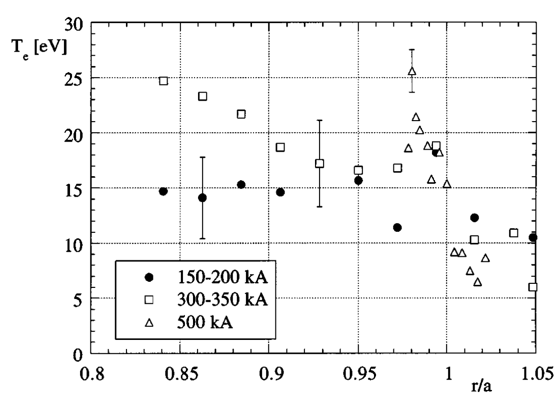

FIG. 2. Electron temperature plotted as a function of the normalized radius, for three ranges of plasma current. Each point is an average over multiple measurements. For each curve a typical error bar is plotted, which gives the standard deviation of the data.

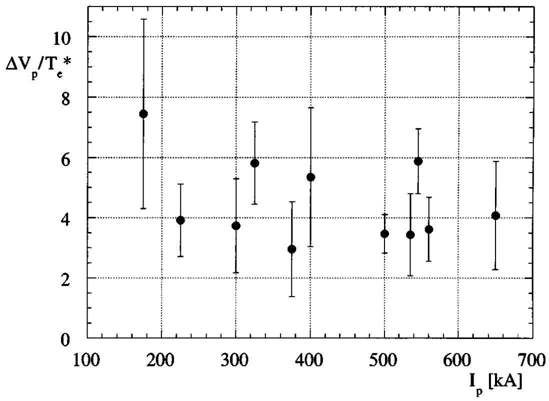

FIG. 3. Depth of the potential well normalized to the edge electron temperature at the well bottom $e\Delta V_p/T_e^*$ plotted as a function of plasma current. Each point is an average over multiple measurements. Error bars give the standard deviation of the data.

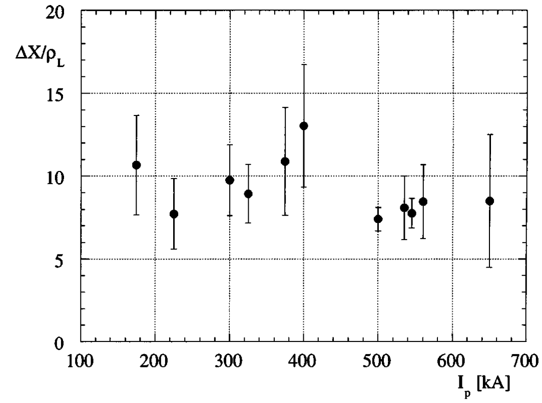

FIG. 4. Distance of the potential well bottom from the first wall Δx normalized to the hydrogen Larmor radius ρ_L plotted as a function of plasma current. Each point is an average over multiple measurements. Error bars give the standard deviation of the data.

potential minimum from the wall is of the order of a few centimeters in the explored current range, and is almost inversely proportional to the plasma current. This result proves that Δx is not related to the location of the toroidal magnetic field reversal surface, which was constant. It is worth noting that the dependence of ΔV_p on T_e^* and the value of Δx show remarkable analogy with tokamak results in the region across the last closed flux surface [8].

Concerning the plasma potential profiles measured in RFX in the region right inside the last closed surface, where the radial electric field is negative, a possible interpretation is that FLR effects cause an increased ion loss to the wall, inducing the formation of a negative electric field to preserve ambipolarity. In order to make a proper comparison, it is worth noting that in RFPs the ion Larmor radius is 1 order of magnitude larger than in tokamaks of comparable size and plasma current, since the magnetic field at the edge is mainly poloidal. An obvious consequence is that the ion Larmor radius changes with plasma current. On the other hand, while in tokamaks the banana orbit width is the dominant parameter for determining direct wall losses, in RFPs neoclassical effects hardly matter [2] and the ion Larmor radius is the relevant quantity. Since in tokamaks the negative E_r region has an amplitude comparable to the ion banana orbit width, it is reasonable to expect for RFPs a width of this region of the order of the ion Larmor radius.

The ratio of Δx to the hydrogen Larmor radius ρ_L derived from the temperature T_e^* is shown in Fig. 4. The ratio is independent of plasma current, and has an average value of 9.2 ± 0.5. This result supports the interpretation described above, even if the ratio turns out to be somewhat higher than expected from tokamak results. Possible explanations could be higher ion temperature and carbon impurity dilution at the edge due to the graphite first wall of RFX. It must also be pointed out that an accurate and self-consistent treatment of the cross field sheath formed right in front of the first wall would require the solution of the Poisson equation taking into account the ion density depletion caused by FLR losses as well as the ion orbit squeezing originated by an electric field with a radial characteristic variation length comparable to the ion Larmor radius. At present, the structure of the plasma potential deeper inside the plasma, in the negative gradient region, has not a clear analogy to tokamaks. Such behavior could be interpreted as an indication that this region is dominated by losses of electrons streaming along the stochastic magnetic field lines.

The consequence that can be drawn is that in RFPs, as in other magnetic confinement configurations, the presence of a negative radial electric field in the plasma region across and right inside the LCFS is a natural feature occurring because of ion FLR losses to the wall. As a consequence, no conclusions concerning the edge magnetic field stochasticity can be drawn from the simple observation of a negative

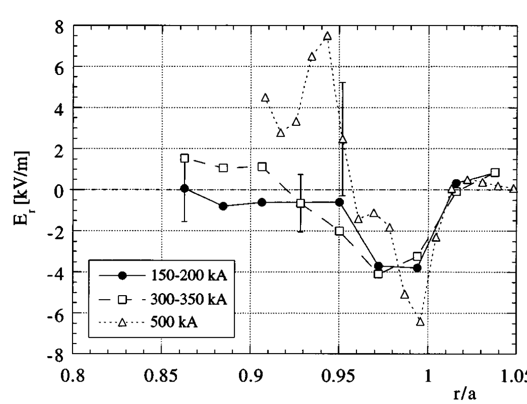

FIG. 5. Radial electric field plotted as a function of the normalized radius for three plasma current ranges. Each point is an average over multiple measurements. For each curve a typical error bar is plotted, which gives the standard deviation of the data.

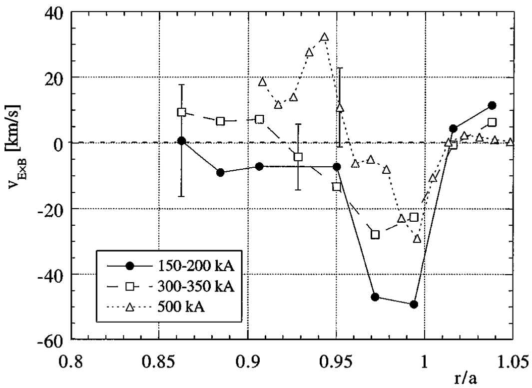

FIG. 6. $\mathbf{E} \times \mathbf{B}$ flow velocity plotted as a function of the normalized radius for three plasma current ranges. Each point is an average over multiple measurements. For each curve a typical error bar is plotted, which gives the standard deviation of the data.

electric field, since Eq. (1) does not hold due to the nonambipolar contribution of the lost ions. Moreover, in the specific RFX case, the presence of superthermal electrons in the edge region [19] is in favor of a stochastic region extending up to the wall.

From the plasma potential profiles of Fig. 1, the radial electric field has been derived and is shown in Fig. 5. Typical values of the electric field of a few kV/m are found. The resulting $\mathbf{E} \times \mathbf{B}$ velocity is depicted in Fig. 6, and reveals the presence of a naturally occurring velocity shear layer in the region crossing the plasma boundary. This result is in agreement with previous direct measurements of the plasma and impurities flow velocity, which have given for 500 kA discharges a velocity shear of the order of 10^6 s^{-1} as measured by Langmuir probes across the $r/a = 1$ surface [11] and have shown a further change in the sign of the velocity around $r/a = 0.9$ as deduced from the impurity lines Doppler shift [17]. The velocity shear right inside the last closed flux surface is independent of plasma current and has a value of the order of some 10^6 s^{-1}, comparable to the naturally occurring velocity shear observed in tokamaks and stellarators [7]. A remarkable feature is the presence of a spontaneous second velocity shear region where the gradient of the radial electric field is negative. The velocity shear in this second region increases almost linearly with the plasma current. The presence of this second shear layer is related to the fact that the radial electric field is positive in the plasma core, due to the magnetic field stochasticity. It is worth noting that while in tokamaks the second layer is not found in Ohmic discharges, it has been observed in L-mode and H-mode discharges [1], where a toroidal momentum input is present (due, for example, to injection of neutral beams).

The role of the double velocity shear structure in the stabilization of electrostatic fluctuations through radial decorrelation, as well as the comparison with the velocity shear layer in tokamaks, are presently under investigation. As in other toroidal configurations, the complete characterization of the plasma potential structure and of the electrostatic fluctuation properties at the edge opens the possibility for an active control of the particle confinement. The similarities with tokamaks and stellarators in terms of electric field radial structure and velocity shear layer suggest the possible existence of enhanced confinement regimes yet to be explored, which could be achieved through plasma edge biasing.

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