Filamentary transport, pedestal structure, and alternative divertor concepts

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Understanding the edge region of fusion plasmas is of primary importance for fusion research and a fascinating scientific challenge. In this presentation for the 2015 IUPAP Young Scientist Prize in Plasma Physics, we will focus on physics processes in three distinct regions of the edge plasma. In the scrape-off layer (SOL), where field lines intercept the vessel wall, turbulent cross-field transport of particles and heat is driven to a large extent by filamentary structures called blobs. We will present progress achieved on the toroidal basic plasma physics experiment TORPEX in the understanding of blob formation and propagation and demonstrate possible methods to control them. We will focus next on the closed field line region just inside the confined plasma, where different physics are at play. In high-confinement regimes, an edge transport barrier forms in this region with pedestal-like profiles usually in both temperature and density and strongly suppressed turbulence. We will present new insights on the poloidal and radial structure of the pedestal obtained on the Alcator C-Mod tokamak. Eventually, one of the main challenges for fusion will be the controlled exhaust of the plasma loss power in the divertor, the region downstream along the magnetic field from the main SOL. Different techniques are used in the divertor to distribute the exhaust power over a sufficiently large area to limit peak heat fluxes and material erosion. We will discuss experiments planned to tackle this issue with alternative divertor magnetic geometries on the TCV tokamak.

1. Introduction

The processes in the edge region of magnetically confined fusion plasmas are of primary importance. They determine the overall plasma confinement, the heat fluxes on the divertor plates and the first wall, main chamber recycling, impurity production and influx, tritium retention, and helium ash removal. We will discuss aspects of the related physics in three distinct regions of the plasma edge, the scrape-off layer (SOL), the pedestal region, and the divertor, as outlined in the following.

2. Blob formation, propagation, and control on TORPEX

Particles and heat can be efficiently transported across the magnetic field in the form of filaments or *blobs*. Blobs are structures of enhanced pressure compared to the surrounding plasma that are elongated along the field but localized in the perpendicular plane. They can propagate radially as coherent entities over distances several times their size, driven by a self-generated ExB drift. Blobs are responsible for the intermittent, non-diffusive transport not only in the tokamak SOL, but in a variety of magnetized plasmas [1].

We will present progress in the understanding of blob physics achieved on the TORoidal Plasma Experiment (TORPEX) [2-3] at EPFL. TORPEX plasmas combine important aspects of SOL physics, such as open field lines and magnetic field curvature, with a relatively simple geometry, high flexibility, and full diagnostics access. This makes TORPEX an ideal experiment to investigate basic aspects of SOL physics and to validate plasma turbulence codes [4-6].

In Fig. 1 (a)-(d), we show a snapshot of Langmuir probe (LP) measurements of a plasma blob across a plane perpendicular to the magnetic field on TORPEX. The blob is clearly visible in the ion saturation current and density profile, Fig. 1 (a) and (b). Fig. 1 (c) reveals the self-generated potential dipole, resulting in an outward directed ExB drift of the blob. Fig. 1 (d) shows that the blob also carries an excess electron temperature compared to the background plasma. We will discuss the mechanism by which blobs on TORPEX are generated from ideal interchange waves [7-9] and subsequently propagate radially according to the following scaling law for blob velocity [10]:

$$v_{blob} = \frac{\sqrt{\frac{2a}{R}}c_s}{1+\frac{1}{\rho_s^2 L}\sqrt{\frac{R}{2}}a^{5/2}+\frac{\upsilon_{in}\sqrt{Ra}}{\sqrt{2}c_s}}\frac{\delta n}{n} \qquad (1)$$

Here, a is the vertical blob size and the three terms in the denominator represent damping of the blob velocity due to ion-polarization currents, currents to the sheath (L being the blob connection length), and ion-neutral friction (υ_{in} being the ion-neutral collision frequency), respectively. A comparison of Eq. (1) with experimental data, in normalized blob size and velocity [10], is shown in Fig. 1 (e). This reveals a transition of the blob propagation regime from a situation where blob velocity is mainly limited by ion-polarization currents ($\tilde{a} \leq 1$) to a regime where parallel currents to the sheath are the dominant damping mechanism ($\tilde{a} \geq 1$) [10-12]. Predictions based on Eq. (1) to actively control blob motion [12] will be presented together with techniques relying on bias-induced convective cells [13-14].

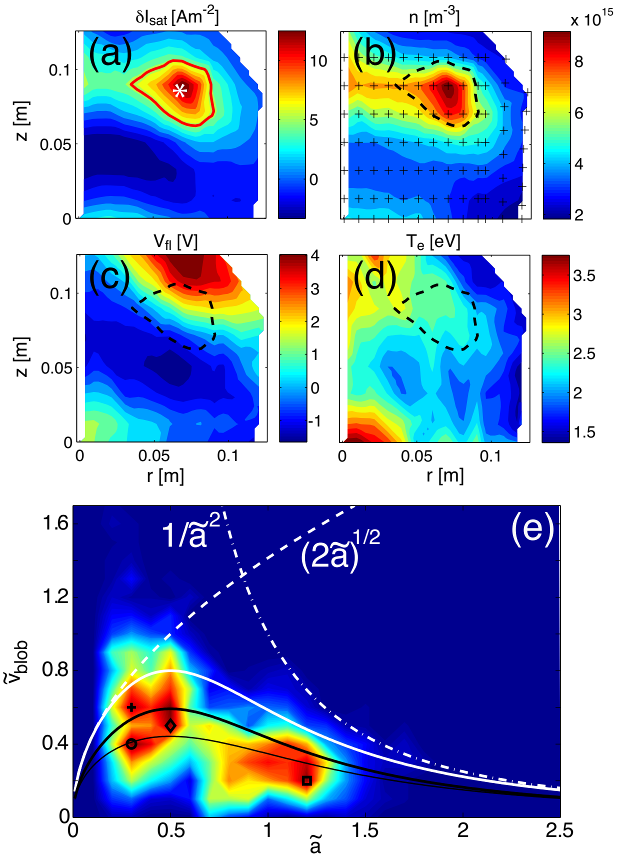

Fig. 1: (a)-(d): Snapshot of cross-field profiles of blob ion saturation current, electron density, floating potential, and electron temperature obtained using conditionally averaged LP data [12,15]. (e): Joint probability of measured blob normalized size and velocity together with a comparison with the scaling law in Eq. (1) [10]. As limiting cases, the scaling laws for blobs in the inertial and sheath connected regimes [1] are also shown (dashed curves).

3. Inboard-outboard asymmetries on Alcator C-Mod

While most measurements on TORPEX are obtained with LPs, different diagnostics are required to explore the substantially hotter and denser pedestal region of tokamaks. Gas puff charge exchange recombination spectroscopy (GP-CXRS) [16] proved to be a powerful diagnostic to obtained new insights on the pedestal in the Alcator C-Mod Tokamak [17], a high field (B≤8T), all-metal walled device located on the MIT campus. The GP-CXRS diagnostic is similar to standard CXRS, with the main difference that it uses a simple thermal gas puff instead of a high-energy neutral beam to locally induce CX reactions. As gas puffs can be installed almost everywhere around the periphery of a tokamak, this opens up the possibility to investigate the poloidal structure of the pedestal.

The GP-CXRS diagnostic thus allowed for the first comparison of inboard-outboard measurements of impurity density [18], temperature, and radial electric field [19] in the pedestal. Those measurements reveal that plasma potential and impurity temperature are not simultaneously flux functions in these pedestals, see Fig. 2. In addition, an inboard localization of impurity density in H-mode plasmas is identified, with values in the inboard pedestal exceeding the values at the outboard side up to six fold. Possible effects of such poloidal asymmetries on radial impurity transport and core impurity peaking and a comparison with simulations [20] will be presented.

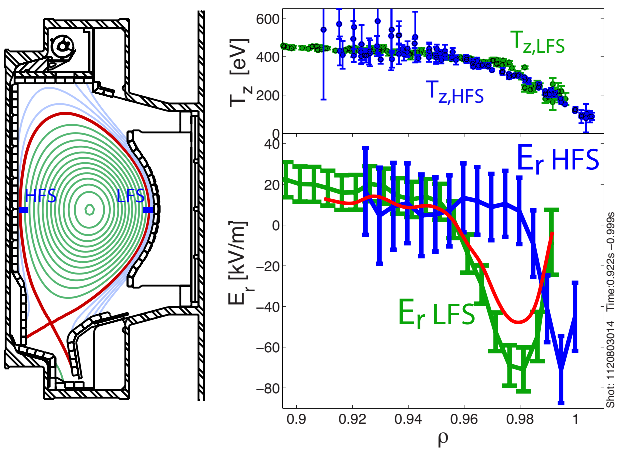

Fig. 2: Left: poloidal cross-section of an Alcator C-Mod plasma with the inboard (HFS) and outboard (LFS) measurement locations of the GP-CXRS diagnostics indicated. Right: HFS and LFS pedestal profiles of impurity temperature (T_z) and radial electric field (E_r) in an H-mode plasma. The red curve shows the HFS E_r profile expected from the LFS measurement and the assumption that plasma potential is a flux function. While the exact alignment of HFS profiles relative to the LFS ones is subject to uncertainty, these measurements demonstrate that T_z and plasma potential are not simultaneously flux functions [19].

4. Alternative divertor experiments on TCV

Above a certain density threshold, plasma pressure is no longer constant along a magnetic field line in the SOL, but drops towards the divertor plate. The plasma is said to (partially) detach. In these conditions, a large part of the exhaust power can be radiated before reaching the divertor target, such that the heat is distributed more evenly over the wall and peak divertor heat loads are strongly reduced. Unfortunately, in standard divertors, as the radiated power increases, the cold radiative region moves from the divertor plate to the X-point and further into the confined plasma. This generally results in a degradation of confinement. Therefore, while a large radiated power fraction in the divertor is a requirement for a fusion reactor, this might not be compatible with good core confinement. The vertical plate divertor, first implemented on Alcator C-Mod and visible in Fig. 2, is expected to be adequate for ITER. However, extrapolation to DEMO is currently uncertain.

For these reasons, alternative exhaust solutions are currently being explored on a number of tokamaks. The TCV tokamak [21], located on the EPFL campus, is well suited for this task, due to its flexible magnetic geometry. Over the past few years, pioneering work has been performed in this respect on TCV (see e.g. [22-23]). Recently, it has been hypothesized that both the increasing poloidal flux expansion near the target in an X-Divertor and the total flux expansion in a Super-X Divertor have the potential not only to lower the detachment threshold, but also to keep the detachment front localized near the target as detachment progresses. Such a stabilization of the detachment front promises compatibility of a fully detached divertor with good core performance. Upcoming experiments on TCV will explore these ideas. The role of poloidal and total flux expansion on detachment access and stability will be investigated using dedicated parameter scans. The benefits of introducing an additional X-point near the divertor plate to generate a X-point target divertor will also be assessed. Both density ramps and impurity seeding will be performed to reach detachment, and different external heating sources, including the new 1MW neutral beam, will be used to vary the SOL power.

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