Overview of the FTU results

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Abstract

Since the 2012 IAEA-FEC Conference, FTU operations have been largely devoted to runaway electrons generation and control, to the exploitation of the 140 GHz electron cyclotron (EC) system and to liquid metal limiter elements. Experiments on runaway electrons have shown that the measured threshold electric field for their generation is larger than predicted by collisional theory and can be justified considering synchrotron radiation losses. A new runaway electrons control algorithm was developed and tested in presence of a runaway current plateau, allowing to minimize the interactions with plasma facing components and tested in presence of a runaway current plateau, allowing to minimize the interactions with plasma facing components.
and safely shut down the discharges. The experimental sessions with 140 GHz EC system have been mainly devoted to experiments on real-time control of magnetohydrodynamic (MHD) instabilities using the new EC launcher with fast steering capability. Experiments with central EC injection have shown the onset of 3/2 and 2/1 tearing modes, while EC assisted breakdown experiments have been focused on ITER start-up issues, exploring the polarization conversion at reflection from inner wall and the capability to assure plasma start-up even in presence of a large stray magnetic field. A new actively cooled lithium limiter has been installed and tested. The limiter limiter was inserted in the scrape-off layer, without any damage to the limiter surface. First elongated FTU plasmas with EC additional heating were obtained with the new cooled limiter. Density peaking and controlled MHD activity driven by neon injection were investigated at different plasma parameters. A full real-time algorithm for disruption prediction, based on MHD activity signals from Mirnov coils, was developed exploiting a large database of disruptions. Reciprocating Langmuir probes were used to measure the heat flux e-folding length in the scrape-off layer, with the plasma kept to lay on the internal limiter the ITER start-up phase. New diagnostics were successfully installed and tested, as a diamond probe to detect Cherenkov radiation produced by fast electrons and a gamma camera for runaway electrons studies. Laser induced breakdown spectroscopy measurements were performed under vacuum and with toroidal magnetic field, so demonstrating their capability to provide useful information on the surface elemental composition and fuel retention in present and future tokamaks, such as ITER.

Keywords: tokamak, overview, FTU

(Some figures may appear in colour only in the online journal)

1. Introduction

FTU is a compact high magnetic field machine (toroidal field $B_T$ from 2 to 8 T, plasma current $I_p$ from 0.2 to 1.6 MA) with circular poloidal cross-section (major radius $R_0 = 0.935$ m, minor radius $a = 0.30$ m). Additional heating system are available, namely a 140 GHz electron cyclotron (EC) system with power up to 1.5 MW and a 8.0 GHz lower hybrid system with power up to 2.0 MW. The FTU stainless steel vacuum chamber is covered internally, in the high field side, by a toroidal limiter made of molybdenum tiles. In addition, there are an outer molybdenum poloidal limiter and a vertical poloidal lithium limiter [1]. Since the 2012 IAEA Fusion Energy Conference in San Diego, FTU ohmic operations were largely devoted to experiments on runaway electrons (RE), investigating the threshold electric field for their generation for a wide range of plasma parameter values and developing a new active RE control algorithm to minimize the RE beam interaction with plasma facing components (PFCs) during disruptions. The fast steering mirror of the 140 GHz EC launcher, recently installed on FTU for real-time experiments, underwent a series of tests and operations to fully characterize its dynamic performances and operational limits, and the experimental sessions with EC system were devoted to studies on real-time MHD control, neoclassical tearing mode onset by central EC injection and EC assisted plasma start-up in ITER-like configuration. Preliminary tests were performed to determine the maximum achievable thermal loads on the new actively cooled lithium limiter inserted in the scrape-off layer (SOL) and first elongated FTU plasmas with EC additional heating were obtained. Density peaking and controlled MHD activity driven by neon injection were investigated and a full real-time algorithm for disruption prediction was developed. Langmuir probes were used to measure the heat flux e-folding length in the SOL, and new diagnostics, as a Cherenkov probe and a gamma camera, were successfully installed and tested. Finally, laser induced breakdown spectroscopy (LIBS) measurements were performed, so demonstrating their capability to provide useful information on the surface elemental composition of PFCs.

2. RE generation and control

2.1. Threshold electric field for RE generation

The determination of the threshold density value to be achieved by means of massive gas injection for RE suppression in ITER relies on the relativistic collisional theory of RE generation which predicts that, below a critical electric field ($E_{ER}$), no RE can be generated. No account of additional loss mechanism, that may increase the critical electric field, is usually made in the above treatment. However, past experiments by means of electron cyclotron resonance heating (ECRH) in the plasma current flat-top phase of FTU discharges had shown that RE suppression is found to occur at electric fields substantially larger than those predicted by the relativistic collisional theory of RE generation. This was found to be consistent with an increase of the critical electric field due to the electron synchrotron radiation losses, which lead to a new electric field threshold ($E_{R,rad}$) [3].

To verify such findings also in FTU discharges with no additional heating, the conditions for RE generation was systematically investigated for a wide range of plasma parameter values in the current flat-top of ohmic discharges: toroidal magnetic field ($B_T = 3.0–7.2$ T), plasma current ($I_p = 0.35–0.50$ MA) and effective charge ($Z_{eff} = 1.5–15$). The threshold electric field for RE generation was measured in two different types of experiments (RE onset and RE suppression), in the first using feed-forward gas programming to obtain a decreasing electron density until RE are generated and in the second creating a RE population during the $I_p$ ramp-up that is then suppressed by the density increase. The time of generation or suppression of RE is clearly indicated through the comparison of the time traces of a set of absolutely calibrated neutron detectors (BF3 chambers) with those of a liquid organic scintillator (NE213). The liquid scintillator is sensitive both to neutrons and hard x-rays (HXR) originated from in-plasma bremsstrahlung of RE, so in absence of a RE beam its signal overlaps with the neutron detectors signal, while in presence of a RE beam the two time traces diverge, as it is possible to see in figure 1 for the two different types of experiments: RE onset and RE suppression.
The experimental results indicate that the measured threshold electric field \( E_{\text{thr}} \) is larger than expected according to the purely collisional theory (up to \( \sim 5 \, E_R \)) and is very close to the new threshold calculated including synchrotron radiation losses (\( \sim 1.5 \, E_{R,\text{rad}} \)). The data are presented in figure 2, where the comparison with the classical collisional threshold and the radiation threshold is given, respectively, on the left and right panels.

2.2. RE control

One main problem in ITER operation is the need of reducing the dangerous effects of RE during disruptions. RE beams can be harmful for PFCs because their low pitch angle causes the deposition of a high amount of energy on small areas yielding serious and deep damages of the vessel structure. The present strategy to address this problem is to suppress RE generation during disruptions by means of massive gas injection of noble gas before the thermal quench (TQ), which however can require an effective disruption predictor and a long recovery time. An alternative possibility would be given by the implementation of dedicated RE active control routines for the dissipation of the RE beam energy and population, stabilizing the RE position to minimize the RE interaction with the PFC while achieving the RE suppression via the central solenoid by reducing the RE beam current.

Experiments on RE active control have been recently carried out in DIII-D, Tore Supra and FTU. A new tool was implemented in the FTU plasma control system for position and current ramp-down control of disruption-generated RE. Such real-time algorithm was tested in dedicated low-density plasma discharges in which a significant RE population is generated during the current flat-top and neon gas is injected to induce a disruption: the rapid variation of the resistivity and the increased loop voltage at the disruption accelerate the pre-existing RE population and lead, in some cases, to the formation of a RE current plateau. By means of the new RE active control, at the onset of the RE plateau, which is detected on-line by dedicated algorithms, the reference value of the plasma current \( I_p \) is linearly decreased to zero and the reference value of the plasma external radius \( R_{\text{ext}} \) is decreased in order to minimize the \(^{235}\text{U} \) fission chamber (FC) real-time signal via an extremum seeking scheme. Therefore, the RE beam position is controlled, in such a way to minimize
the interactions with PFC while the RE beam current is reduced and safely shut down the discharge. Some experimental results are shown in figure 3, where two discharges with similar plasma current drop and neutron signal level before the current quench (CQ) are compared. In discharge 36569 the RE active control is ON: we obtain \( I_p \) ramp-down when the RE plateau is detected, as well as a decrease of \( R_{\text{ext}} \), and this produces a reduced RE interaction with the PFC, as indicated by the reduced FC signal, which measures the photoneutron and photofission signal from HXR due to interaction of RE with the PFC [8].

3. ECW experiments

3.1. Real-time control of MHD instabilities

Experiments of real-time control of MHD instabilities using injection of electron cyclotron waves (ECWs) are being performed at toroidal field of 5.3–5.6 T and plasma current of 0.5 MA [9]. The control system was based on only three real time key items: an equilibrium estimator based on a statistical regression, an MHD instability marker (SVDH, ranging from 1 to 0 in presence of a growing mode) using a 3D array of pick-up coils [10] and the fast ECW launcher recently installed and tested on FTU [11]. One beam of 0.4 MW, 140 GHz, max pulse duration of 0.5 s, has been used for heating in the ordinary mode polarization (OM1), and the EC absorption volume has been controlled by the poloidal steering of the launcher (with \( 1^\circ/10\text{ms} \) maximum angular speed). The MHD instability (a low-order tearing mode, usually with \( m/n = 2/1 \)) has been deliberately induced either by neon gas injection or by a density ramp hitting the density limit. When the ECW power is switched ON, the instability amplitude shows a marked sensitivity to the position of the absorption volume (with an increase or decrease of its growth rate), with a considerable stabilizing effect when its position corresponds to the radial position of the mode rational surface. The control tools used are essential and based on a minimal set of diagnostics. Such experimental condition mimics the situation of a fusion reactor where reduced diagnostics capabilities are expected. Some experimental results are reported in figure 4.

In the left panel, the MHD behaviour is reported for two plasma discharges with and without ECRH and same amount of injected neon. A significant reduction of the MHD amplitude has been obtained during the ECW injection phase. However, the continued cooling by neon recycling that originates the instabilities does not allow their complete suppression at least at this ECW power level. The MHD control loop has been modified for the density limit experiments shown in the right panel, where the MHD behaviour is reported for two plasma discharges with and without ECRH in the high density regime. The automatic search of the steering angle producing the fastest instability reduction has been introduced, based on the evaluation of the time derivative of the MHD amplitude. Once such angle is reached the controller holds the position until the SVDH signal crosses the switching off threshold. This control criterion has led to the suppression of the instabilities, even if in some cases, after a prolonged heating, the instability newly start to increase, probably due to a change in the radial position of the mode rational surface with respect to the position of the ECW absorption volume.

3.2. Onset or amplification of (N)TM by central EC power

The onset of both the conventional and neoclassical tearing modes (N)TMs, in absence of explicit triggers as sawteeth and edge localized modes (ELMs), remains an important issue for the fusion plasma operations to avoid the degradation of the plasma confinement due to these resistive instabilities. It is well known that the ECW can destabilize the TMs when the power is absorbed off-axis around the main resonant surface \( q = 3/2 \) and \( q = 2 \) by local modification of the plasma current density. On the other hand, the understanding of the (N)TM onset driven by on-axis EC action, far from the resonant surface, is a field still not well understood and is a key issue for the MHD instability control. The effects of central electron cyclotron heating (ECH) and current drive (ECCD) on (N)TMs onset have been investigated in FTU high magnetic
field tokamak and in TCV elongated tokamak, for plasma configuration with low rotation in low confinement scenario (L-mode). The comparison of the response from these two different devices can give clear information about the main mechanisms leading to the mode destabilization due to the on-axis EC power. In both these devices the central EC power deposition affected the (N)TM dynamics. In TCV two possible concomitant driving mechanisms for these instabilities have been associated to the change of plasma current density profile upstream of the resonant location \( q = m/n \) (and of the mode stability parameter \( \Delta_i' \)) and to the change in sign of the local differential velocity between the toroidal plasma and the mode velocity due to the toroidal torque, originated from the EC driven current, allowing the destabilizing action of the ion polarization current. In FTU, the amplification of a mode that was present in a marginally stable state is due to the modification of the local plasma current density and to the increased bootstrap effect proportional to \( \beta_p \), while the frequency increase is due to torque action originated from the applied co-ECCD. No effect due to modification of rotation (ion polarization effect) are present, because of the amplified size of existing perturbation [12]. An example of the (N)TM dynamics on FTU in presence of central EC injection is reported in figure 5.

### 3.3. EC assisted plasma start-up

The intrinsic limited toroidal electric field in future large fusion devices with superconducting poloidal coils (0.3 V m\(^{-1}\) for ITER) requires additional tools, like EC waves, to initiate plasma current (breakdown) and to sustain it during the crossing of high radiative loss phase (burn-through). The FTU tokamak has contributed in the past to this subject together with a group of other tokamaks, by performing preliminary experiments focused on low electric field start-up using perpendicular injection of EC power [13]. Afterwards, a new experimental and modelling activity, addressing the study of EC assisted plasma start-up in a configuration close to ITER one (magnetic field, wave oblique injection and polarization) has been realized on FTU [14]. The results of these new experiments with oblique EC injection have demonstrated the importance of polarization conversion (from ordinary mode OX1 to extraordinary mode XM1) at the inner wall reflection, as confirmed by a lower toroidal electric field and a faster plasma current ramp-up when the waves reflect on the inner vessel surface. This effect is related to the lower plasma resistivity corresponding to the higher plasma temperature reached as a result of the better power absorption of extraordinary polarized waves generated at reflection, assuring a wider operational window in term of filling pressure and toroidal electric field. A comparison of the plasma current ramp-up for discharges with and without mode conversion is reported in figure 6. The optimization of the discharge with mode conversion allowed the start-up up to a toroidal electric field of 0.5 V m\(^{-1}\).
Dedicated experiments also showed the capability of EC power to sustain plasma start-up even in presence of relatively strong perpendicular magnetic field (10 mT, 3 times higher than the one expected in the ITER case, due to uncertainties of the currents flowing in the poloidal coils), with the field null outside the vacuum vessel (the variations of field null position was obtained via external vertical magnetic field). These results assume more and more importance considering that the first plasma in ITER will be likely obtained at half field (2.5 T), where the influence of stray magnetic field is doubled. The beneficial effect of EC power to sustain breakdown in case of low toroidal electric field or high stray magnetic field is partly counter balanced by a lower threshold for RE generation. These new experiments have been supported by a 0-code, BKD0, developed to model the FTU plasma start-up, solving a set of balance equations to estimate the temporal evolution of average plasma parameters, and linked to a beam tracing code, GRAY [15], to compute, in a consistent way, the EC propagation and absorption. Experimental results are in agreement with the BKD0 simulations, supporting the use of the code to predict start-up also in future tokamaks, like ITER.

4. Lithium limiter experiments

4.1. Thermal load on the new lithium limiter

Power load on the divertor is one of the main problems to be solved for steady state operation on the future reactors and liquid metals (Li, Ga, Sn) could be a viable solution for the target materials. Since 2006, experiments by using a capillary porous system (CPS) liquid lithium limiter (LLL) were successfully performed on FTU, indicating a good capability of the system to sustain power loads up to 1–2 MW m⁻² without any damage to the limiter surface [16]. However, experiments have clearly pointed out that if the surface temperature exceeds 500 °C, lithium evaporation becomes very strong, thus limiting its use in a reactor. In order to prevent the overheating of the liquid lithium surface and the consequent strong lithium evaporation for T > 500 °C, an advanced version of LLL has been realized and installed on FTU by using the same vertical bottom port of the previous limiter. This new system, named cooled lithium limiter (CLL), is an actively cooled limiter optimized to demonstrate the limiter capability to sustain thermal loads as high as 10 MW m⁻² with up to 5 s of plasma discharge duration, reachable in FTU by operating at low magnetic field (Bₜ = 4 T). Water circulating at high pressure (30 bar) at the temperature of about 200 °C plays a double role: it heats lithium up to the melting point and removes the heat during the plasma discharge. The heat load on the CLL is evaluated by three different means: a fast infrared camera observing the whole limiter, the temperature measurements of inlet and outlet water as detected by the thermocouples and finally the measurements of the local electron temperature and density by Langmuir probes placed on the CLL. As first step, CLL has been tested in the FTU SOL to identify the best plasma conditions for a good uniformity of the thermal load. Thermal loads analysis was performed by applying ANSYS code, that has been adapted to the real CPS geometry and to the active cooling conditions of the new limiter, pointing out that heat loads as high as 2–3 MW m⁻² for 1 s have been withstood without problems for the limiter surface [17]. New plasma discharges will be performed in the near future with CLL inserted deeper in the FTU SOL and close to the last closed magnetic surface (LCMS) to provide a clear understanding of CLL behaviour under plasma discharges and to determine the maximum sustainable heat load.

4.2. Elongated plasmas

In 2013, first elongated FTU plasmas heated by 500 kW of ECRH have been obtained with the new CLL. The magnetic configuration (Bₜ = 5.5 T, Iₚ = 200 kA, elongation k ~ 1.2, triangularity (βₚ) = 0.18, magnetic shear ~40% higher than in circular plasma) presents the X-point close to the first wall with the 3λ surface (λ, energy e-folding length, ~1 cm in FTU) opened on the CLL, as it is possible to see in the left side of figure 7, where the flux map for FTU discharge 37869 is reported. In these discharges the CLL does not act as first limiter; the D-shaped FTU plasma is indeed in contact with the outer molybdenum poloidal limiter [18]. These discharges aim at investigating the access to H-mode, as done at JET with MarkIIGB divertor by moving the X-point up to 5 cm inside the septum [19]. No L-mode to H-mode transition was observed so far, consistent with the threshold being above the injected power at Bₜ = 5.5 T [20]. Neither damage on CLL nor plasma pollution were observed despite the CLL was located near (~1 cm) the LCMS. Further experiments are planned in the near future, considering discharges with a lower value of the toroidal magnetic field (2.7 T) to reduce the power requirement for accessing H-mode, thus having the possibility to study the impact of ELMs on the CLL used as first limiter. An alternative connection scheme for the poloidal field coils in FTU has been preliminary analysed, with the aim of achieving a true X-point configuration with a magnetic single null well inside the plasma chamber and strike points on the lithium limiter. X-point plasma scenarios, with current up to 300 kA and duration up to 2.5 s, have been studied [21]. In the right side of figure 7, the configuration during Iₚ flat-top, as calculated by MAXFEA equilibrium code, is shown. A first engineering analysis has been also carried out, showing the
structural, thermal and electromagnetic compatibility of this alternative connection scheme with the load assembly structure and with the existing power supply and cooling plants.

5. Plasma response to neon injection

5.1. Peaked density profiles

The attainment of high density in the plasma hot core, which is an essential condition to obtain high fusion reaction rate is facilitated by density peaking. To this purpose, it is worth noting that a stable radiative edge seeded with light impurities has beneficial effects and provokes density peaking without any undesirable central impurity accumulation [22, 23]; on the other hand, edge cooling by too large an amount of impurities can lead to a disruptive MHD activity. In order to maintain the positive effect of the edge radiation it is important to fix the conditions of a strong increase of particle confinement while minimizing the amount of impurities needed. Recent experiments were devoted to characterize the plasma response to neon injection for different plasma parameter values: toroidal magnetic field $B_T = 5.2–6.0$ T, plasma current $I_p = 360–700$ kA, line-averaged density $\bar{n}_e = 0.2–1.2 \times 10^{20} \text{ m}^{-3}$ [24]. On FTU the neon injection causes a spontaneous increase of the line-averaged density up to a factor 2, in the absence of additional deuterium gas puffing. In figure 8 a neon doped discharge is compared with a reference one that reaches the same density by D gas puffing in the absence of neon injection. The comparison shows a more peaked density profile for the neon doped discharge. A qualitative estimate from UV spectroscopy measurements indicates that the density behaviour cannot be attributed simply to the stripped electrons from the puffed neon, but a modification of particle transport should be invoked in order to explain the spontaneous rise of the density and the higher density peaking. The observed density peaking was analysed in terms of electron diffusion coefficient and pinch velocity, in the framework of a simple particle transport model [25], confirming the presence of a higher inward pinch (with respect to the reference case) inside half radius of the plasma, where the assumption of negligible sources is valid and the method described is more reliable. A micro-stability analysis will be performed in the near future to investigate the role of the ion and electron gradient driven modes on particle transport.

5.2. Tearing mode instability

The possibility of driving controlled MHD activity by neon injection was investigated for different plasma parameter values (toroidal magnetic field, plasma current, electron density) during stationary conditions and in absence of strong MHD activity [26]. In all the analysed discharges, after the neon puffing, a weak $m/n = 3/1$ tearing mode is observed at first, followed by a $m/n = 2/1$ mode. It is possible to separate the dynamics of the 2/1 mode in three phases (see top panels of figure 9). In the first phase the mode amplitude grows above the very low pre-neon value and the mode frequency increases from the value at formation (usually 5–8 kHz) up to around
15 kHz. The second phase is characterized by large oscillations of the mode frequency that result in a broad spectral peak with nearly stationary average mode frequency, and by increasing mode amplitude. In some cases, the reaching of a critical amplitude leads the start of a third phase with rapid mode lock and subsequent disruption. The formation and dynamics of MHD activity strictly depend on the amount of puffed neon. This allows controlling the duration of the three phases of the mode dynamics; for this reason, neon injection represents a very efficient method to deliberately induce formation of different repeatable MHD targets (low-order tearing modes) for different experimental needs. The mode onset is associated to the increase of the current density gradient around \( q = 2 \) radius due to the plasma cooling by neon radiation, while the continuous increase of the mode frequency is correlated with the progressive growth of pressure gradient at the \( q = 2 \) radius, induced by neon injection. This mode frequency increase is in a quantitative agreement with the increase of the electron diamagnetic drift frequency.

6. MHD signals as disruption precursors

6.1. Single Mirnov coil method

The definition of suitable disruption precursors in order to trigger actions for avoiding or at least mitigating disruptions is currently being investigated in FTU, as in many other tokamaks. Typical FTU disruptions have a phase dominated by a strong MHD activity preceding the CQ, so a full real-time algorithm for disruption prediction, based on MHD activity signals from a single Mirnov coil, was developed exploiting a database of about 2000 FTU discharges (400 disruptions, 1600 regular terminations) covering a wide range of physical parameters. The Mirnov coil signals were integrated to obtain the poloidal field perturbation whose amplitude is compared with a preset threshold, and the threshold parametrization in terms of plasma parameters \( (B_T, I_p) \) was optimized in order to maximize the number of correctly predicted disruptions (85% of right alerts) while simultaneously minimizing the number of alerts for non-disruptive discharges (12% of false alerts) [27]. The resulting threshold is given by \( Th(G) = 9.5/(q_a - 1) \), where \( q_a = 0.533 \times B_T(T)/I_p(\text{MA}) \). The optimized distribution of the right alerts as a function of the time difference between the CQ time \( (t_{CQ}) \) and the alert time \( (t_{AL}) \) is reported in the left side of figure 10.

6.2. Singular values decomposition method

An MHD instability marker, named SVDH, was derived from a singular value decomposition factorization applied to the normalized signals from a 3-dim array of Mirnov coils [10]. It represents the average information associated with the singular values of the factorization of the matrix of the poloidal magnetic perturbation signals sampled at multiple space positions (as much as the number of coils) and multiple time instants (equivalent to the number of samples in a given control cycle time). The SVDH value tends to a minimum (ideally close to zero) when there is perfect phase coherence among the coils signals and then the magnetic perturbations are developed; on the contrary, it is close to unity when there are no coherent perturbations. Experimentally it can be seen that the time evolution of the SVDH value indicates clearly the birth of the instabilities and usually well in advance with respect to the increase of their amplitude to the level to which the disruption happens, suggesting that the SVDH signal analysis can be applied to real-time disruption prediction. The analysis of the time behaviour of the SVDH value for 800 discharges of the FTU database resulted in the identification of 82% of disruptions in advance with respect to the plasma CQ, with 29% of false alarms, the percentages weakly depending on the set of chosen coils. The nature of the considered signals implies that the time window in which the analysis is performed is well before the mode locking appearance. The optimized distribution of the right alerts as a function of the time difference between the CQ time and the alert time is reported in the right side of figure 10.

7. SOL studies

Understanding how transport in the SOL is affected by macroscopic and local parameters and how it is linked to heat transport on limited/diverted tokamaks is still under debate. Despite being a limited tokamak, FTU results can be applied to diverted machines before X-point formation and far from the X-point, where the transport in the SOL is expected to be quite similar. Data collected by two arrays of reciprocating Langmuir probes, located 180° poloidally away from each other, were processed to determine the heat flux e-folding
length $\lambda_q$ in the SOL for a wide range of toroidal magnetic field (2.7–7.5 T), plasma current (250–500 kA), line-averaged density ($0.3–1.3 \times 10^{20} \text{ m}^{-3}$) and safety factor (3.5–14) values. All the plasma discharges were ohmic and during the experiment the plasma column was kept to lay on the internal toroidal limiter to resemble the ITER start-up phase in order to contribute to the definition of the still uncertain value of $\lambda_q$ in that phase. In view of adding information to predict the flow channels width for ITER during the pre X-point formation phase, we searched for an empirical dependence of $\lambda_q$ on the main macroscopic plasma parameters and the main findings were a strong dependency on the plasma current ($\sim I_0^{0.6}$) and power crossing the last closed magnetic surface ($\sim P_{\text{SOL}}^{0.8}$) and a little dependency on the line-averaged density [28]. This experiment contributed to the multi-tokamak scaling of SOL heat flux width of ITER limiter start-up plasma.

8. Diagnostics

8.1. Cherenkov probe

In the 2013 experimental campaign, a novel optical diagnostic system, based on the Cherenkov effect, was installed on FTU, in collaboration with the NCBJ group (IPPLM Association), to study the dynamics of non-thermal electrons escaping the plasma [29]. The Cherenkov probe consists of a diamond detector mounted on a titanium–zirconium–molybdenum (TZM) head inserted in the FTU vessel (in the limiter shadow). The Cherenkov radiation emitted within the diamond by electrons escaping from the plasma is coupled to a VIS/UV fibre connected to a high-gain photomultiplier tube. The diamond probe is not sensitive to background electromagnetic radiation such as visible, synchrotron or gamma radiation, so only superluminal electrons impinging on the probe contribute to the detected optical signal. Data from the Cherenkov probe detector were correlated with those from several other diagnostics, including ECE, liquid organic scintillator (NE213), Mirnov coils and soft X-ray cameras, providing evidence of loss of confinement of fast electrons in presence of high-amplitude magnetic islands and demonstrating that the modulation of the Cherenkov signal is due to the rotation of the magnetic island (see figure 11). Future work is planned to enable energy discrimination of the non-thermal electrons escaping from the plasma, through the implementation of a multi-channel probe, as well as an increase of the spatial information by installing a second probe. Modelling and simulations with a hybrid MHD gyrokinetic code, HMGC [30], are also planned in the near future.
8.2. Gamma camera

A digital upgrade of the analogue electronics of the FTU neutron camera [31] has been carried out in order to enable studies on REs by measuring the HXR produced in the bremsstrahlung interactions between RE and plasma ions. The gamma camera system is based on six radial lines of sight equipped with liquid organic scintillators (NE213) capable of $n/\gamma$ discrimination also in conditions of very high count rate (MHz range) [32]. HXR profiles can be obtained for energies greater than 0.1 MeV to study the runaway population during the current ramp-up, flat-top and ramp-down phases with sub-ms time resolution. Measurements of HXR radial profiles at two different times are reported in figure 12 for a standard FTU discharge, showing that REs are created mainly in central part of the plasma. Disruption events can also be analysed, although in such phases the data are affected by the presence of a strong HXR background not originating in the plasma.

8.3. Laser induced breakdown spectroscopy

Monitoring the changes in the composition of the PFC surface layer, as a result of erosion and re-deposition mechanisms, can provide useful information on the possible plasma pollution and, above all, on the fuel retention in present and future tokamaks, such as ITER. Laser-based methods, relying on the heating of wall materials, with subsequent desorption or ablation of target components and the analysis of their emitted light, are the most promising candidates for these in situ measurements. Among these laser-based methods, the LIBS can be used in between plasma shots, the localized plasma plume produced by the intense laser radiation providing, under suitable conditions, all the needed information, without affecting plasma operations. LIBS measurements have been performed on FTU, by focusing the radiation of a Q-switched Nd–YAG laser to tungsten samples, coated with thick Al–C–W mixed layers, and collecting the light emitted by the plasma plume generated by the laser pulse. Experiments were carried out under vacuum, with a toroidal magnetic field up to 4 T. In figure 13 the Al and W spectral lines from the Al–C–W mixed sample are shown as an example. These experiments have demonstrated the feasibility of in situ LIBS diagnostic of surface layer composition, with the possibility to provide useful information on the fuel retention in present and future tokamaks, such as ITER [33].

9. Conclusions and perspectives

Experiments on runaway electrons have shown that the measured threshold electric field for their generation is larger than predicted by collisional theory and can be justified considering synchrotron radiation losses. These studies will be extended to a wider range of plasma parameters values. A new runaway electrons control algorithm has been developed and tested in presence of a runaway current plateau, allowing to minimize the interactions with plasma facing components and safely shut down the discharges. Future activities will be focused on optimizing the scenario for reliable runaway plateau generation at disruption (using, for example, argon instead of neon to induce the disruption) and the performance of position control algorithm using new control system for disruption-generated runaway electrons, based on real-time fission chamber and loop voltage signals.

Experiments on real-time control of MHD instabilities using the new EC launcher have shown the capability to stabilize low-order tearing modes induced by neon gas injection or by a density ramp hitting the density limit. Future activities in this area will explore the possibility of overcoming the density limit by means of ECRH applied to heat the edge magnetic island appearing in FTU plasmas in the high density regime and rapidly growing just before the disruption for density limit. EC assisted plasma start-up experiments have been focused on ITER start-up issues, exploring the polarization conversion at reflection from inner wall, showing the capability to assure plasma start-up even in case of low toroidal electric field or in presence of a large stray magnetic field, with the field null outside the vacuum vessel. Further experiments are planned with a toroidal magnetic field of 2.5 T, which will be likely used for the first plasma in ITER.

Preliminary test were performed with the new actively cooled lithium limiter inserted in the scrape-off layer. Thermal loads analysis have pointed out that heat loads as high as $2 - 3 \text{ MW m}^{-2}$ for 1 s have been withstood without any damage to the limiter surface. New plasma discharges will be performed in the near future with the limiter inserted deeper in
the FTU scrape-off layer and close to the last closed magnetic surface to determine the maximum sustainable thermal load. First elongated FTU plasmas with EC additional heating were obtained. Further experiments are planned, considering discharges with a lower value of the toroidal magnetic field \(B_T = 2.7\, T\) to reduce the power requirement for accessing H-mode, thus having the possibility to study the impact of edge localized modes on the lithium limiter used as first limiter. An alternative connection scheme for the poloidal field coils in FTU has been preliminary analysed, with the aim of achieving a true X-point configuration with a magnetic single null well inside the plasma chamber and strike points on the lithium limiter.

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