

Extended scenarios opened by the upgrades of the RFX-mod experiment

M.E. Puiatti, P. Agostinetti, M. Agostini, P. Bettini, T. Bolzonella, D. Bonfiglio, A. Canton, R. Cavazzana, S. Dal Bello, M. Dalla Palma, R. Delogu, G. De Masi, A. Fassina, P. Innocente, L. Grando, O. Kudlacek, G. Marchiori, L. Marrelli, E. Martines, M. Moresco, R. Paccagnella, N. Patel, S. Peruzzo, H. Sakakita¹, P. Scarin, M. Siragusa, P. Sonato, G. Spizzo, M. Spolaore, D. Terranova, M. Valisa, M. Vallar, P. Zanca, L. Zanotto and the RFX-mod team

Consorzio RFX, Padova, Italy

¹ *Innovative Plasma Processing Group, National Institute AIST, Tsukuba, Ibaraki 305-8568, Japan*

e-mail address: mariaester.puiatti@igi.cnr.it

Abstract. Thanks to the exploitation of its advanced active control system, during 10 years of operation RFX-mod reached its target design values, offering a cross-configuration point of view for issues such as 3D physics, dynamo and self-organisation, transport barriers, turbulence, isotopic effect, density limit, plasma-wall interaction.

In order to extend the explored operational scenarios, several upgrades, not involving radical machine modifications, are proposed and are presently in the design phase. This paper describes the scientific motivations and the expected improvements: reduction of residual magnetic chaos and achievement of more robust helical states in Reversed Field Pinch configuration; extension of high density regimes with a better density control; achievement of H-mode in Tokamak configuration to study ELM, sawtooth and runaway electron mitigation by magnetic perturbations; enhancement of diagnostic capability.

INTRODUCTION

RFX-mod is a toroidal device ($R/a=2\text{m}/0.46\text{m}$, [Sonato 2003]) characterised by great reliability and flexibility which allows performing experiments in a wide range of safety factor values $q(a)=aB_\phi/RB_\theta$ values. It can be operated as a Reversed Field Pinch (RFP), a magnetic configuration characterised by a q profile reversed at the edge ($q(a) < 0$) [Diamond 2014], as a Tokamak, getting access to regimes not explored in present devices, for example the very low q ($q(a) < 2$) scenario [Zanca 2012], and as a ultra-low- q device, with $0 < q(a) < 1$ [Bonfiglio 2008]. When operated as a RFP, thanks to an advanced MHD active control system based on 192 saddle coils independently driven, RFX-mod has explored the high plasma current regime up to 2 MA, with electron temperature up to 1.5 keV and densities in the range $1 \div 10 \cdot 10^{19} \text{ m}^{-3}$ [Puiatti 2015]. As a 0.1-0.2 MA ohmic Tokamak (with $B_0=0.55\text{T}$), it can produce both circular and shaped (Single and Double Null) plasmas, accessing the H-mode when the edge plasma is polarized by means of an insertable electrode [Spolaore 2016].

During more than 10 years of operation RFX-mod contributed to several critical physics topics for magnetic confinement fusion, including 3D physics, dynamo and self-organisation, MHD control, transport barriers, turbulence, isotopic effect, density limit, plasma-wall interaction.

To extend the explored parameter range and exploit the opportunity for a cross-configuration view of open issues, some enhancements for the RFX-mod device are presently proposed, which can be summarised as:

- 1) upgrade of the magnetic front-end to reduce the residual magnetic chaos;
- 2) change of plasma facing material (PFM) and upgrade of wall conditioning systems to improve density control and reduce wall recycling;
- 3) installation of a 1MW neutral beam aiming at a more reliable L-H transition in Tokamak configuration;

4) new/upgraded diagnostic systems to improve the understanding of plasma behaviour.

In the following, motivations, potentialities, perspectives and conceptual design principles of the modifications listed above will be discussed; the technical details of the design have been recently reported in [Peruzzo 2016]. The modified RFX-mod will be referred to as RFX-mod2.

MAGNETIC CHAOS REDUCTION THROUGH AN IMPROVED CONTROL OF RADIAL MAGNETIC FIELD IN RFP PLASMAS

In RFP configuration, RFX-mod has discovered the so called Quasi Single Helicity (QSH) states, theoretically predicted [Cappello 2000] and then experimentally observed [Lorenzini 2009],

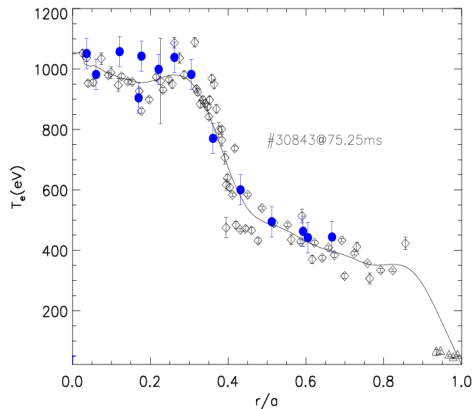


Fig.1 : example of electron thermal transport barrier in RFX-mod; data from Thomson scattering (black) and soft X-rays (blue) diagnostics: Edge points (triangles) are from a Thermal Helium beam system [Agostini 2015]

spontaneously occurring at high plasma current ($I_p > 1\text{MA}$) when a single tearing mode (TM) ($m/n=1/7$ in the case of RFX-mod) grows up in amplitude becoming dominant over the other (secondary) modes which instead decrease. During QSH states, thermal electron internal transport barriers develop (ITBs), featuring strong gradients (an example is shown in fig. 1). The width of the hot region encompassed by strong gradients is found to increase with decreasing amplitude of the secondary modes, so that a further reduction of secondary modes is expected to produce more robust and wider ITBs [Lorenzini 2016]. At the barrier location, the thermal conductivity is decreased and magnetic field stochasticity is no longer the main mechanism driving transport. In particular, ITG microturbulence and microtearing modes become

important [Predebon 2015, Zuin 2013]. Helical states are not stationary in RFX-mod, since they are subjected to reconnection processes when the plasma is perturbed by bursts of MHD activity. However, the persistence during the current flat-top increases with increasing Lundquist number, i. e. with increasing plasma current [Piovesan, 2009]. On the other hand, a reliable operation at very high current requires a reduction of the localized Plasma Wall Interaction (PWI) by a decrease of the non-axisymmetric deformation of the Last Closed Magnetic Surface (LCMS) [Puiatti 2013]. A significant improvement of the control of the magnetic boundary has been obtained by the Clean Mode Control algorithm [Marrelli 2007]. In particular a reduction of the

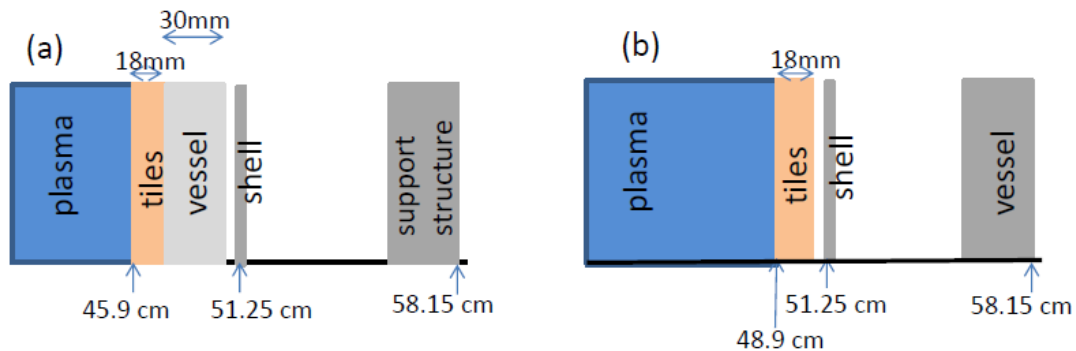


Fig. 2: (a) present layout of the magnetic front-end in RFX-mod; (b) new layout for RFX-mod2

edge radial field and consequently a forced rotation of TM (though at frequencies lower than the natural rotation frequency) have been observed avoiding the stationary localized plasma wall interaction of the bulge produced by their locking in phase and to the wall. Tearing Mode dynamics under feedback controlled conditions have been successfully modelled by the RFXLOCKING code [Zanca2009], used to optimize the feedback parameters [Piron2010] and to

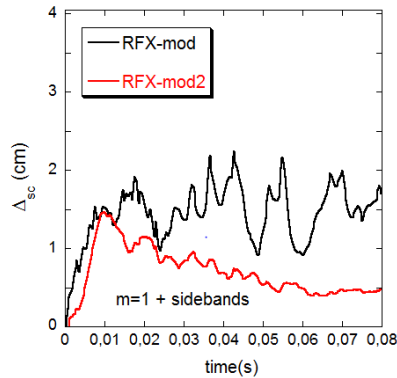


Fig.3: deformation of LCMS, with vacuum vessel removed (red) and in present configuration (black)

highlight the critical role of passive structures in determining the limits of improvements possible by feedback control [Zanca2012]. A further decrease of the residual deformation of the last magnetic surface still remains a crucial point to produce stable plasmas with stationary helical states in the 2MA range. Fig. 2a shows a sketch of the present configuration of the magnetic front-end in RFX-mod. It includes an Inconel vessel with a honeycomb structure surrounding the plasma (thickness 3.0 cm), a 0.3cm, $\tau_w \approx 100$ ms copper shell (radius $b=51.25$ cm) and a support structure (radius $c=58.15$ cm), the latter also supporting the control coils. Based on RFXLOCKING simulations the edge radial field and the related LCMS deformation are reduced if the plasma is moved closer to the copper shell. Accordingly, the new magnetic front-end will be modified as shown in fig. 2b: the vacuum vessel will be removed and the copper shell will become the first conductive surface for the plasma. The RFXLOCKING code has been applied to quantify the effect of such change, and the result is shown in fig. 3, where the deformation of the LCMS is compared in the two configurations. The calculation includes the effect on $m=1$ modes and the impact of coil sidebands on measurements and on plasma surface distortion is taken into account. The figure indicates a reduction of a factor about 2-3 for the LCMS deformation. This simulation is performed with the pessimistic assumption that the internal amplitude of Tearing Modes will remain the same as in RFX-mod; indeed, according to DEBS simulations [Paccagnella 2007], the reduced shell proximity affects the amplitude of $m=1$ TM: if the mode energy decreases by 50%, an additional reduction of the mode amplitude of about 25% can be expected.

REDUCTION OF $m=0$ MODES

The new configuration is expected to have a positive effect also on $m=0$ modes: RFXLOCKING simulations indicate that the removal of the vacuum vessel leads by itself to a faster rotation of the LCMS bulging related to $m=1$ TM phase locking, which non-linearly interact with $m=0$ modes.

In addition, an upgrade of the toroidal power supply system in terms of latency and synchronisation among different inverters by which it is composed, is expected to improve the effectiveness of $m=0$ control.

Indeed, it has been observed that the RFP plasma performance is improved at very shallow q value, when the amplitude of the $m=0$ TM is the lowest [Carraro 2013]. At shallow field reversal the amplitude of $m=0$, low n modes decreases, as they are intrinsically stabilized by the shift of their resonance closer to the external conducting shell [Paccagnella 1998]. When the amplitude of $m=0$ modes is low, the non-linear coupling with secondary $m=1$ modes decreases, allowing the growth of the dominant one.

The topology of the chain of islands due to $m=0$ TM resonating at the $q=0$ surface, also affects the operation in the high density regime, where they are associated to the development of convective cells leading to localised density accumulation and finally to the onset of MARFE-like poloidal structures leading to a soft termination of the plasma current. In particular, in [Spizzo 2014] it has been shown that MARFE develops when the amplitude of $m=0$ modes exceeds a critical value reached when the island intercepts the wall. Interestingly, a similar behaviour with

respect to the $m=2, n=1$ islands has been observed in RFX-mod operated as a tokamak and in the FTU tokamak.

INCREASE OF CURRENT TRESHOLD FOR TEARING MODE SPONTANEOUS ROTATION

In RFX-mod, recent experiments showed the recovery of the fast spontaneous rotation branch of TM at low amplitude [Innocente 2014], at a plasma current below 120 kA. Such threshold depends on the conductivity of the first conducting structure, which will change in RFX-mod from Inconel to copper. As a consequence, the range of RFP operations with spontaneously rotating tearing modes will be extended. In a numerical study conducted in [Zanca 2009] the locking threshold has been evaluated as a function of the conductivity of the shell located close to the plasma, with a shell to plasma radius ratio similar to the new RFX-mod2 one (RFX-mod2 will be the RFP with the highest conducting stabilizing wall, similar to MST whose wall is made by Aluminum). This study predicts an increase of the locking threshold in terms of mode amplitude by a factor 3-5. Assuming a linear increase of mode amplitude with plasma current, we expect a locking threshold of 300-500 kA in RFX-mod2. Moreover, as already mentioned, a further increase of such threshold could be related to the decrease of mode amplitude due to the reduced shell proximity (factor 25%), leading to a new threshold in the range 375-625kA.

DENSITY AND WALL RECYCLING CONTROL

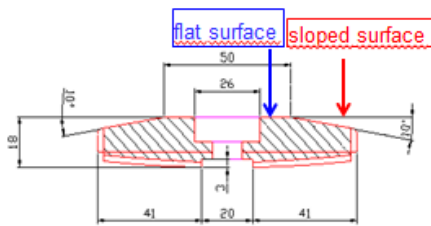


Fig.4: sketch of present RFX-mod tile shape

In RFX-mod the Inconel vacuum vessel is fully covered by polycrystalline graphite tiles as Plasma Facing Material (PFM) [Puiatti 2013a], with a flat central surface and two flat 10° sloped surfaces (fig.4). No limiter or divertor is present. Consequently, during a plasma discharge, the density behavior strongly depends on the high deuterium desorption from the wall and on the localized PWI rather than on the amount of the fuelling gas (no neutral beam is installed, so that fuelling is only by gas puffing or pellet injection). In RFX-mod operation high power loads are involved, which, due to local error fields and mode locking, can lead to localised peak power deposition of the order of $\approx 10\text{-}50 \text{ MW/m}^2$ [Puiatti 2013], thus producing large and uncontrolled particle release, in particular when the graphite is saturated with deuterium. This motivated the proposal of replacing the present graphite tiles to mitigate this effect. Model simulation and experimental tests have been performed to select the most convenient option, in terms of ability in

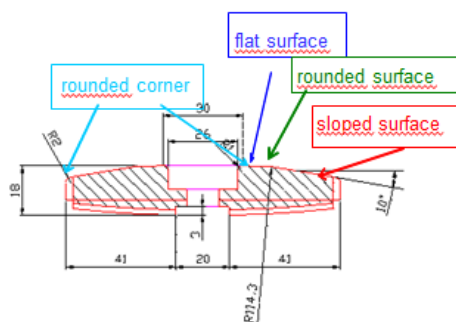


Fig.5: optimised tile shape

supporting high power loads, deuterium retention capability, optimization of the shape to avoid very sharp localized interactions. In particular, a new shaping has been designed, featuring a reduced flat central surface connected by rounded surface to the edges [Canton 2016]. This is sketched in fig. 5. The comparison of the effect of the new shape in terms of power deposition, made by a simple calculation assuming parallel flux with an exponential decay length of 1mm (as evaluated for RFX-mod in [Innocente 2016]), gives the result shown in fig.6.

With regards to the material, both graphite with higher conductivity and W-deposited graphite have been considered. In fact higher conductivity

graphite could in principle guarantee faster heat dissipation and therefore a lower surface temperature. On the other side, a metallic PFM would be related to lower deuterium retention. As an experimental test, during the last campaign different tiles have been exposed to high current plasmas: a) optimized shape as in fig.5; b) higher conductivity graphite (by a factor 3); c) 2-3 μm W deposited on graphite [Ruset 2011], with the layer thickness optimized to minimize the differential thermal expansion between the deposited layer and the substrate, thus mitigating delaminations. After the exposition to about 100s of 1.5 MA plasma in RFP configuration no damage has been found on tiles of optimised shape and higher conductivity graphite with a W film deposited. However, this test is not conclusive, because during the campaign no extremely severe interaction in the region where the tiles were placed has been observed by CCD cameras. The

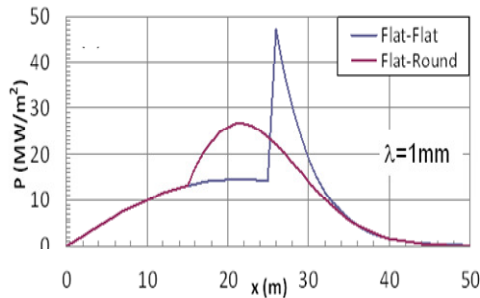


Fig.6: effect of shaping on the power deposition profile

behavior of different type of tiles has also been investigated by a model inspired by [Pitcher1990], which calculates the particle influxes and therefore the expected plasma electron density in different conditions of the plasma and of the PFM. In summary, experimental tests and calculations show that to minimize influxes from the wall and therefore improve density control critical factors are: a) as expected, a low power deposition, in particular avoiding localized peaks; b) a low amount of deuterium trapped in the wall before the discharge, and therefore efficient wall conditioning techniques operating on a shot-by-shot basis; c) high conductivity of the first wall, to keep the tile temperature low. Based on these considerations, extruded graphite is presently under consideration as a new tile material. Indeed, though characterized by less optimized mechanical properties than Carbon Fiber Composite (CFC), it is much cheaper and offers a conductivity comparable to the CFC one, higher by a factor about 3 than the present polycrystalline graphite tiles of RFX. Additional laboratory tests are planned to definitely qualify the thermal mechanical properties of this material as PFM in RFX-mod2.

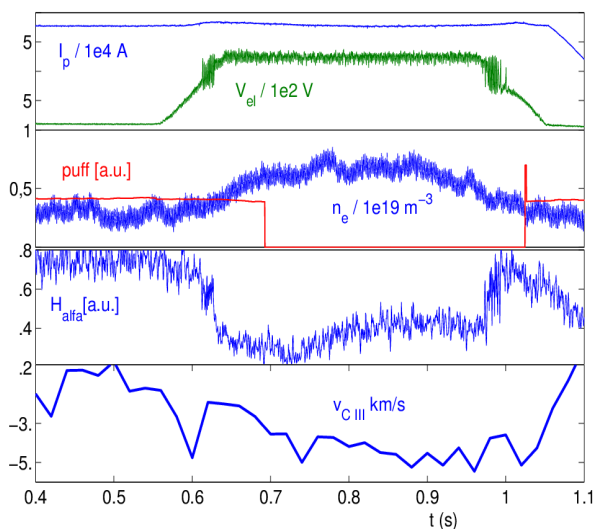


Fig.7: during the application of an electrode in a tokamak discharge, from top to bottom: plasma current and electrode voltage; electron density and puffing pulse; H_{α} signal; rotation velocity from C III

TOKAMAK OPERATION: IMPROVED PLASMA SHAPE CONTROL AND L-H TRANSITION

In Tokamak configuration, the RFX-mod control system allowed the stabilisation of the $m=2, n=1$ RWM and the exploration of the $q(a)<2$ regimes [Zanca 2012]; in addition, experiments with feedback-controlled magnetic perturbations contributed to the studies on error field control, tearing mode mitigation and disruption avoidance, sawtooth mitigation [Piovesan 2013], runaway electron decorrelation [Gobbin 2016], helical flow behaviour [Piron 2013].

Thanks to the flexible system of power supplies combined with field shaping windings, discharges with non-circular cross section have been operated, and enhanced confinement regimes have been obtained

with an insertable polarised electrode (fig.7) [Spolaore 2016]. A new configuration of magnetic sensors is planned for RFX-mod2, in particular with increased number of poloidal sensors. In fact the present number of poloidal and radial measures is quite adequate to determine the plasma horizontal and vertical position in circular discharges, with the magnetic equilibrium reconstructed by a fully model-based approach, as described in [Marchiori 2016] and [Kudlacek 2015]. Instead, in shaped discharges, the contribution of higher order harmonics is not negligible and has to be numerically evaluated. Their direct calculation is limited by the present 4 (toroidal) x 8 (poloidal) array of probes. Considering the presence of 28 tiles, an array of 12 (toroidal) x 14 (poloidal) sensors is foreseen as the minimum requirement for a measurement-based and reliable equilibrium reconstruction. This magnetic diagnostic improvement will allow a strong enhancement in the

| | Present layout | Upgraded layout |
|----------------------------------|----------------------------|------------------------------|
| Bp,Bt,Br triaxial pick-up | 0 (4 x 48 biaxial) | 4 x 72 |
| Br saddle probes | 4 x 48 | 4 x 48 |
| Bp,Bt,Br dense poloidal array | 8 x 4 (4 shared per array) | 14 x 12 (2 shared per array) |
| Br at poloidal and toroidal gaps | 0 + 48 | 24 x 2 + 48 |

Table 1: increase of magnetic sensor number in RFX-mod2

control of magnetic equilibrium and plasma shape, thus increasing the flexibility in the position of the X-point; also negative triangularity shaping will be tested.

Two additional arrays will be also installed to measure the phase and rotation direction of the $m=3, n=2$ mode.

It is worth mentioning that also the number of other magnetic sensors is going to be increased, aiming at improvements both for Tokamak and RFP (some are listed in table 1).

In RFX-mod the active control system can be exploited to apply in a controlled way magnetic perturbations and study their effect on several instabilities. Of particular interest is to study their effect on ELMs. Indeed, different strategies for ELM suppression or mitigation are foreseen in ITER on the basis of the experimental results obtained in the presently operating Tokamak devices, e.g. pellet pacing and application of resonant magnetic field perturbation (RMP). Regarding the latter technique, several issues are open, such as the perturbation spectrum, the range of the safety factor, the magnitude of the perturbed field. The RFX-mod flexible MHD active control system offers the opportunity for ELM mitigation and edge transport control experiments by means of RMP techniques, easily testing a variety of mode control strategies and possibly contributing to investigate the above-mentioned open issues. Of course this implies the achievement of the H-mode, where ELMs are observed. To this end, a NBI system, available at

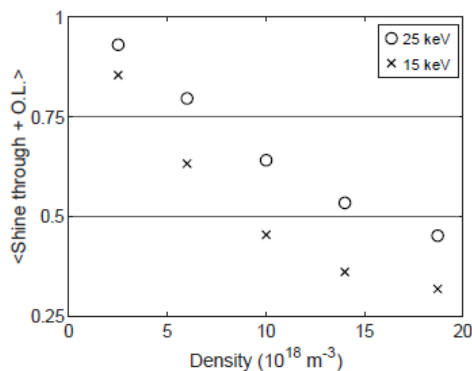


Fig. 8: 1 MW NBI in RFX-mod2: simulation of the losses (shine-through + orbit losses) with 25 keV and 15 keV energy

RFX (25 keV, 50 A) and originally developed at AIST (Japan) will be installed. The duration of the beam pulse, when operated at 25 keV, is limited to 30ms, which should be sufficient to induce the L-H transition, but could be short to perform RMP experiments for ELM control. However, if operated at 15 keV the pulse duration can be extended to 100 ms. While a tangential injection appears to be not feasible, due to the port dimensions and other mechanical constraints, a perpendicular injection can guarantee an adequate beam penetration into the plasma. Simulations by the METIS code with a beam energy of 15 keV have shown that in the density parameter range of RFX-mod tokamak the losses (shine-through + orbit losses) are below 50% (fig.8) and scenarios with enhanced temperature and

$Ti/Te > 1$ can be explored [Vallar 2016]. Indeed, according to the international tokamak scaling the power threshold for an RFX-mod single-null plasma is about 0.1-0.2 MW, comparable with the ohmic power. The injection of the beam is therefore important to make the transition easier and more robust.

DIAGNOSTIC IMPROVEMENTS

In addition to the upgrade of the magnetic sensor system, several new/upgraded diagnostics will be installed, to improve the understanding of the plasma behaviour and to test advanced systems providing informations for applications to future devices, including for example DTT and DEMO. Among them, a new reflectometry system has been designed, based on three reflectometers located in three poloidal positions at the same toroidal angle: two on the equatorial plane, on the Low Field Side (LFS) and on the High Field Side (HFS), the third one on the upper side. In RFP, working in the O-mode, the covered density range is $3-8 \cdot 10^{18} \text{ m}^{-3}$, allowing detailed measurements of the edge density profile and, in particular, a direct probing of the HFS/LFS asymmetries and a deeper characterisation of the $m=0$ islands. In tokamak configuration, the upper X-mode cutoff frequency will be exploited to probe the edge density both on the HFS and on the LFS; this condition is also suitable for measurements in the upper position. Main aim of the diagnostic is in this case the real-time control of plasma position, which is a critical issue for future devices, where the use of magnetic sensors will be limited by high neutron fluxes. For this reason a specific work package has been set-up for DEMO by the EUROfusion Consortium. Indeed a full reflectometry based plasma position control has never been conceived up to now, so that RFX-mod2 will be an excellent test bench for this kind of diagnostic.

At present, edge measurements by probes are limited in RFX-mod to a maximum current of 0.5 MA. A fast reciprocating manipulator will be installed, to be combined with the measurements from Thermal Helium Beam (THB) and Gas Puffing Imaging in order to fully characterise the plasma edge in terms of current filaments, plasma pressure, parallel and perpendicular flow, vorticity and Reynolds and Maxwell stresses.

Given the 3D nature of RFX-mod edge, which requires the implementation of spatially distributed diagnostics, a tomographic light impurity system is proposed for the detailed reconstruction of main gas and impurity influxes. Such system is based on 7 CCD detectors coupled to optics equipped with interchangeable interference filters. In combination, the addition of a third THB in the HFS equatorial plane (two THBs are presently available, one in the LFS equatorial plane, the other on the top) will allow a poloidal reconstruction for the edge temperature and density, to study the poloidal asymmetries of pressure profile and also necessary to calculate the influxes.

For RFX-mod2 tokamak plasmas, where Thomson scattering measurements are difficult due to the rather low density, an improvement of the signal-to-noise ratio will be obtained by modifying the present system (which provides temperature and density profiles on 84 radial points with 10 ms repetition rate) in order to allow a double-pass of the laser beam.

SUMMARY

The scientific motivations of the improvements proposed and designed for the RFX-mod device and based on a relatively limited set of modifications have been presented. Main deliverable is to enhance the contribution to the study of 3D physics in fusion devices, in particular self-organisation, helical magnetic equilibria, transport barriers, development of MHD control tools and algorithms, effect of applied magnetic perturbations on plasma instabilities and turbulence.

To advance the performance in RFP configuration a modification of the magnetic front-end increasing the plasma-shell proximity will allow a significant reduction of the residual magnetic chaos and TM spontaneous rotation at higher plasma current. Main upgrades for the Tokamak operations will be instead the increase of the number of poloidal magnetic sensors and the installation of a NBI system to favour the transition to H-mode allowing studies of ELM control

by magnetic perturbations. A better control of plasma density, which is a crucial issue in some operational regimes, will be guaranteed by a new first wall, made by high conductivity carbon tiles with optimised shape. Finally, several new innovative diagnostic systems will allow a deeper comprehension of plasma behaviour in all regimes.

References:

- [Agostini 2015] M. Agostini et al., Rev. Sci. Instrum. 16 (2015) 123513
- [Bonfiglio 2008] D. Bonfiglio, S. Cappello, R. Piovan, L. Zanotto, M. Zuin, Nucl. Fusion 48 (2008) 115010
- [Cappello 2000] S. Cappello, D.F. Escande, Phys. Rev. Lett. 85 (2000) 3838
- [Carraro 2013] L. Carraro et al., Nucl. Fusion 53 (2013) 073048
- [Diamond 2014] P.H. Diamond, X. Garbet, P. Ghendrih, Y. Sarazin, *Rotation and Momentum Transport in Magnetized Plasmas*, Vol. 2 (World Scientific 2014), chapter 9 “What is a Reversed Field Pinch?” by D.F. Escande
- [Gobbin 2016] M. Gobbin et al., “Runaway electrons mitigation by applied magnetic perturbations in RFX-mod tokamak plasmas”, accepted Nucl. Fus.
- [Innocente 2014] P. Innocente, P. Zanca, M. Zuin, T. Bolzonella, B. Zaniol, Nucl. Fusion 54 (2014) 122001
- [Innocente 2016] P. Innocente et al., “Heat flux measurements and modeling in the RFX-mod experiment”, presented at 22nd Plasma Surface Interaction in Controlled Fusion Devices, 2016 Rome May 30-3 June 3, submitted Nucl. Mat. and Energy
- [Kudlacek 2015] O. Kudlacek et al., Phys. of Plasmas 22 (2015) 102503
- [Lorenzini 2009] R. Lorenzini et al., Nature Physics 5 (2009) 570
- [Lorenzini 2016] R. Lorenzini et al. Phys. Rev. Lett. 116 (2016) 185002
- [Marchiori 2016] G. Marchiori et al., Fusion Eng. Des. 108 (2016) 81-91
- [Marrelli 2007] L. Marrelli et al Plasma Phys. Control. Fusion (2007) 49 B359
- [Paccagnella 1998] R. Paccagnella, Nucl. Fusion 38 (1998) 1067
- [Paccagnella 2007] R. Paccagnella R. et al Nucl Fus 47 (2007) 990
- [Peruzzo 2016] S. Peruzzo et al., “Design of machine upgrades for the RFX-mod experiment”, presented at 29th Symposium on Fusion Technologies, Prague 5-9 September 2016, submitted Fusion Eng. Des.
- [Piovesan 2009] P. Piovesan et al., Nucl. Fusion 49 (2009) 085036
- [Piovesan 2013] P. Piovesan P. et al., Phys. Plasmas 20 , 056112 (2013)
- [Piron 2010] L. Piron et al., Nucl. Fusion 50 (2010) 115011
- [Piron 2013] L. Piron . et al., Nucl. Fusion 53 (2013) 113022
- [Pitcher 1990] C.S. Pitcher, Fusion Eng. Des. 12 (1990) 63-71
- [Predebon 2015] I. Predebon, P. Xanthopoulos, Phys. Plasmas 22, (2015) 052308
- [Puiatti 2013] M.E. Puiatti et al., Plasma Phys. Control. Fusion 55 (2013) 124013
- [Puiatti 2013a] M.E. Puiatti et al. Nucl. Fusion **53** (2013) 073001
- [Puiatti 2015] M.E. Puiatti et al., 2015 Nucl. Fusion **55** 104012
- [Ruset 2011] C. Ruset et al., Fusion Eng. Des. 85 (2011) 1677-1680
- [Sonato 2003] P. Sonato et al., Fusion Eng. Des. 66–68 (2003) 161
- [Spizzo 2014] G. Spizzo et al., Nucl. Fusion 54 (2014) 043007
- [Spolaore 2016] M. Spolaore et al., *H-mode achievement and edge features in RFX-mod Tokamak operation*, this conference, paper EX/P5-24
- [Vallar 2016] M. Vallar et al., “Requirements and modelling of fast particle injection in RFX-mod tokamak plasmas”, presented at 29th Symposium on Fusion Technologies, Prague 5-9 September 2016, submitted Fusion Eng. Des.
- [Zanca 2009] P. Zanca , Plasma Phys. Control. Fus. **51** (2009) 015006
- [Zanca 2012] P. Zanca et al., Plasma Phys. Control. Fusion 54 (2012) 094004
- [Zuin 2013] M. Zuin et al., Phys. Rev. Lett. 110 (2013) 055002