Status and Perspectives of the Liquid Material Experiments in FTU and ISTTOK

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Abstract. The main goal of FTU lithium liquid metal experiments was to demonstrate the capability of the Capillary Porous System (CPS), firstly tested on T-11 tokamak, to confine liquid lithium avoiding sudden release as consequence of $j \times B$ electromagnetic force on the surface. These latter should be balanced by the surface tension forces associated to the liquid lithium in capillary channels. Very encouraging results have been obtained: a new ohmic plasma regime is obtained characterized by spontaneous peaking of the electron density profile with line average values of the density near or beyond the Greenwald limit. Furthermore these discharges exhibit an increase of the energy confinement over ITER97 L mode scaling up to a factor 1.4 after transport analysis with the JETTO code. Although the lithium liquid limiter is 2 cm away from the last closed magnetic surface and FTU is a fully metallic device the only impurity detected by the survey spectrometer is lithium, i.e. lithization is quite uniform over all the metallic part directly exposed to the plasma. In these conditions, the application of additional power (ECRH plus LH) has shown that the formation of an internal transport barrier (ITB) can be easier probably due, to an increase of LH current drive efficiency in plasma with lower Z_{eff} and/or different edge conditions with a reduction of the local turbulence.Furthermore a distinct approach is being investigated in the small size ISTTOK tokamak, where free flowing, fully formed, liquid gallium jets have been successfully exposed to the plasmas aiming at studying their power extraction capability as well as the effects of tokamak-like magnetic field on their behaviour. Preliminary results on these issues are presented.

1. Introduction

One of the most important question for DEMO and a future reactor is the first wall and divertor materials Up to now, only carbon has been extensively used and studied in tokamak device [1] while experimental results with tungsten are becoming available only in the last years [2]. But tritium retention for carbon and/or Carbon Fiber Composite and plasma operation with a high Z material like tungsten are important unsolved issues. So it is envisaged to test new approaches. Liquid metals could be a possibility perhaps, the only one. The crucial question to solve is how to counteract the Lorentz force as it was put in evidence in the DIII-D experiments with a lithium sample exposed in the lower divertor region [3]. The problem of the mechanical stabilization of liquid metal against $j \times B$ forces has been faced and solved in the Russian Federation where a new concept for the metal confinement based on a capillary porous system (CPS) in limiter configuration [4] has been developed and implemented using lithium as liquid element.

The use of free flowing liquid metal surfaces, in fusion devices, would be a key advantage since their use as plasma facing component would provide a highly efficient process to exhaust power from reactor chamber [5]. To explore this capability the interaction of a liquid gallium jet with plasmas is being studied in the small size ISTTOK tokamak. Although lithium presents incomparable benefits over gallium, due to its excellent compatibility with the plasma, its high reactivity with air and higher melting point turns it out to be a much more challenging material to generate liquid metal jets than gallium. Furthermore gallium has better thermodynamic properties (higher boiling point, lower vapour pressure [6]) which represent a significant benefit with regard to the power extraction feature. Studying this question is currently the main issue in the ISTTOK liquid metal experiment.

This paper describes the experimental results obtained on FTU with a liquid lithium limiter (LLL, section 2) and on ISTTOK using a gallium jets (section 3). The perspectives are briefly discussed in the conclusions (section 4).

2. FTU results

The LLL used on FTU employs the same CPS configuration previously tested successfully on T-11M [7]. This structure is realized as a matt from wire meshes of stainless steel 304, with pore radius 15 μm and wire diameter 30 μm that lead liquid Li to the side faced to the plasma from a liquid Li reservoir. The LLL system, composed by three similar units (shown in Fig. 1), is installed on a vertical bottom port of FTU.

FIG. 1. Photograph of the three units of LLL

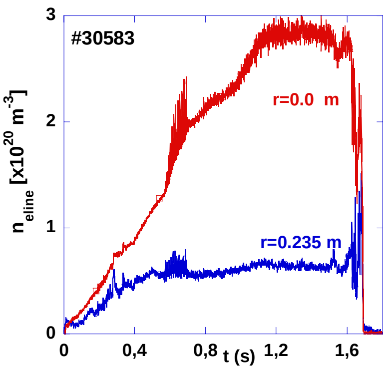


FIG.2.Line average density vs. time for two different chords

LLL has been exposed to the plasma in OH discharges at B_T=6T, I_p=0.5-0.7MA and $\bar{n}_e$ from 0.15 up to $3.0 \cdot 10^{20} m^{-3}$. New plasma regimes with highly peaked density profiles (peaking factor $pk_n = n_{e0}/\langle n_e \rangle > 2$) are spontaneously developed for $\bar{n}_e > 1.0 \cdot 10^{20}$ m^{-3}. In Fig. 2 the time evolution of two different radial line average electron densities are plotted; although there is a strong increase between 0.4 and 1.2 s of the electron densities at the center, in the periphery the signal is quite constant. Plasma densities near or beyond, the Greenwald density limit are achieved and easily reproduced. The origin of the steeper radial density profiles for many aspects similar to those obtained in pellet fuelled discharges, is probably due to the pumping effect of lithium leading to a particles depletion in the outermost plasma region. This is also confirmed by the higher electron temperatures at the plasma periphery and in the SOL measured by Langmuir probes [8].

A detailed transport analysis as well as a study of the confinement properties of the new scenario has been performed and the results obtained are discussed here, focusing in particular on the comparison of analogous studies performed on the bulk of ohmic “pre-Lithium” FTU discharges [9].

The transport analysis has been performed by using the JETTO code [10], taking as input the Z_{eff} evolution from visible bremsstrahlung signals and the electron temperature and density profiles as they result respectively from Thomson scattering and CO_2 interferometer diagnostics, whereas the ion temperature profile is derived predictively by assuming a neoclassical transport model with an "anomalous" multiplicative coefficient calibrated on the constraint of matching the experimental neutron rate. The radiation power density profile has been deduced from the bolometry taking into account the data obtained on the impurities concentrations (consisting for the great majority of Li and a very little fraction of O) by the spectroscopy.
In all the analyzed discharges the ion transport results to be fully neoclassical with an anomaly coefficient equal to 1. The study of the time evolution of high and peaked density profile discharges ($n_{e,lin}$=2.7÷2.8 $\times 10^{20}$ m^{-3}) shows a sharp transition from a low energy confinement regime to a higher one. The transition occurs as soon as the density profile effectively begins to peak, at the threshold value of 1.75÷1.8. The improved confinement can be observed both in the energy confinement time (τ_E) and in the enhancement on the ITER-97 L-mode scaling (H_{97}), which pass from about the average value of the typical FTU ohmic regime ($H_{97}\approx 0.92$) to $H_{97}\approx 1.25* \tau_{ITER97-L}$.
In the peaked, high density phase χ_e is about a factor 2 lower than in the not-peaked phase and takes the value of ~0.2 m^2/s which is typical of the saturated ohmic confinement (SOC) regime. However, in the peaked phase heat transport keeps being dominated by electron conductivity, whilst the ion conductivity remains very low and quite close to its neoclassical value.
The analysis of the entire available database of lithized discharges (with n_e ranging from 0.6 to 2.8 $\times$ 10^{20} m^{-3}) confirms the existence of the threshold in the peaking factor at which appears the improved confinement regime described above. The presence of a second regime of better confinement, up to a factor 1.4 above τ_{ITER97}, come out clearly as a general behaviour of the lithized discharges, at least at I_p = 500 kA. At higher plasma current (700 and 750 KA) few data are available but it is worthwhile to point out that the same peaked density profile above the threshold value are achieved, but more data should be considered in order to tell if an analogous transition to a better confinement regime also occurs at higher currents.
In Fig. 3 a statistical comparison of τ_E vs. line average electron density of the "lithized" database discharges with that of the "pre-Lithium" FTU ohmic, both gas-fuelled and pellet-fuelled discharges in the SOC phase, is shown.

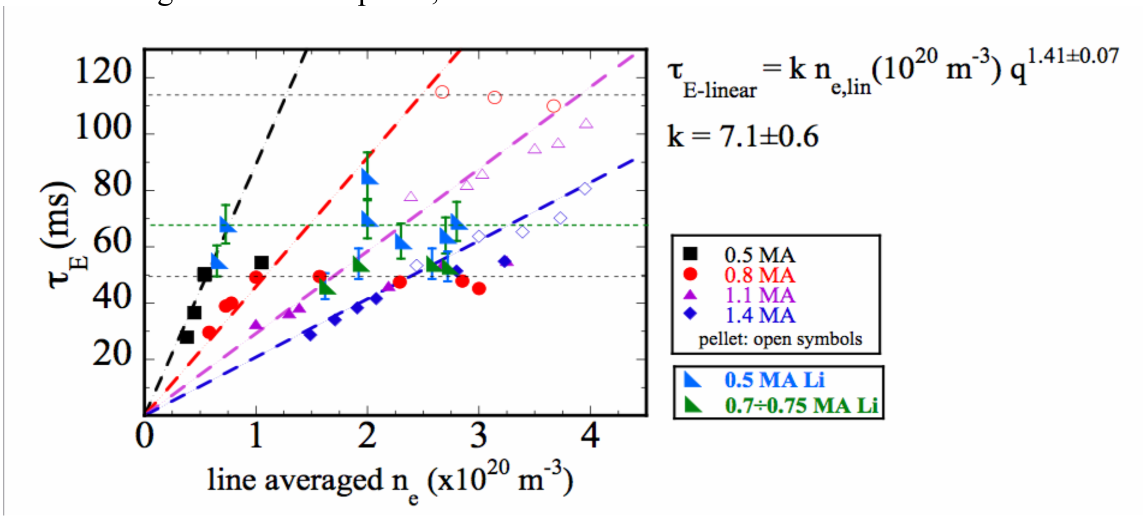


FIG. 3. Comparison between lithizated discharges and the ohmic database of FTU

Following the legend near the figure, lithizated discharges are indicate by the triangles in light blue at Ip=0.5MA and in green at Ip=0.7-0.75 MA while the other symbols for different plasma current are for the non lithizated discharges and open symbols are referred to pellet experiment. In presence of the LLL a rise of the saturated confinement threshold value from ~45-50 ms to ~65-70 ms is observed (the two horizontal dotted lines in the figure). Exactly as in the case of pellet fuelled discharges, as discussed in [11] and [12], this behavior is accompanied by a sharp increase in the peaking of the density profile and by a full neoclassical ion transport.

Generally, high density plasma discharges are clean, i.e. $Z_{eff}\approx 1$. This is true also for a lithizated machine but, in addition, lithium is the only detectable impurity into the plasma in all the plasma operation space. In Fig. 4 the VUV spectra recorded by a survey instruments are shown for two different discharges. The upper spectrum (red line) was taken in a non lithizated discharges and the Mo and O lines are prominent. In the lower spectra (blue line) only lithium line are present.

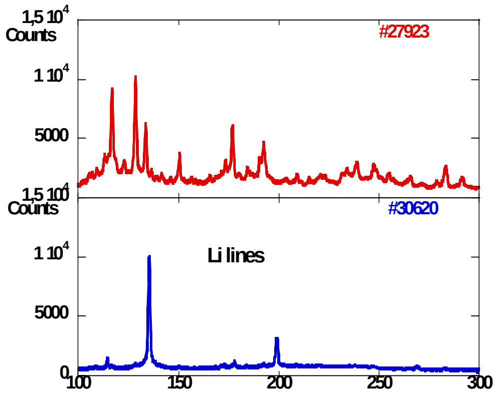
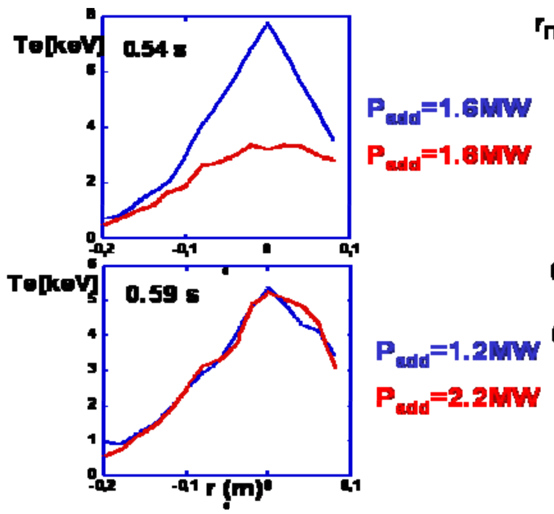


FIG.4. VUV Spectra

In these discharges with low Z_{eff} and n_e in the range from 0.5 up to $0.8\cdot 10^{20}$ m^{-3} (B_T=5.3T, I_p=0.5MA), LH and ECR heating at total power levels up to 1.5 MW have been injected. Preliminary results indicate that strong internal transport barrier (ITB) can be obtained with lower additional power than for pure metallic or boronized walls. The low Z_{eff} value, that increases the LH current drive efficiency, could help the formation of a proper current radial profile, as well as that of the reduced recycling. The relative importance of the two processes is under investigation.



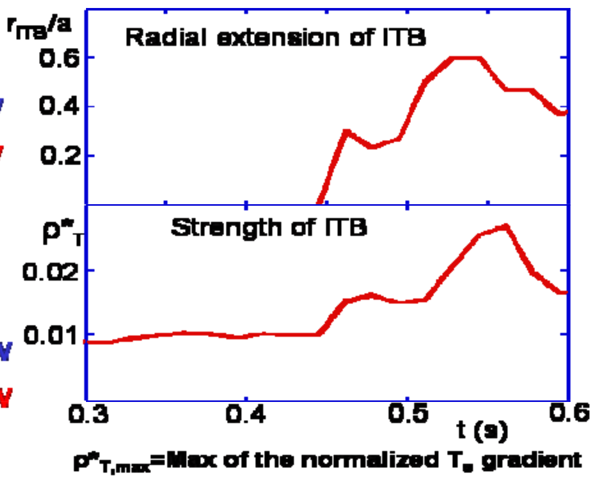


FIG.5. Electron temperature profile *FIG.6. Radial extension and strength of ITB*

In Fig. 5 the electron temperature profiles for two discharges have been compared with lithized (#30620 blue line) and metallic (#27923 red line) walls characterized by similar main plasma parameters. In both cases additional power has been applied during the plateau of the plasma current. For #30620, P_{LH}=0.75 MW and P_{ECH}=0.8 MW starting respectively at 0.5 s and 0.45 s, while for #27923 P_{LH}=1.5 MW and P_{ECH}=1.2 MW respectively starting at 0.5 s and 0.55 s. For #30620, obtained with LLL, a strong ITB with very high central temperature of 8 keV develops to be compared with the lower value of 5 keV reached in the metallic case, in spite of the higher additional power injected. The same profile and central temperature are obtained, in a lithizated discharge injecting half additional power. Although it is short life, the barrier is quite high (Fig. 6), since its strength as given by the normalized temperature gradient, $\rho^*_{T,Max}$ [13] is about 0.026, well above the ITB threshold at 0.014. Its radial width exceeds half the minor radius, similarly to what previously obtained in FTU for the same current and field [13]. In these discharges a large dilution of the plasma with lithium particles up to 50% accounted for by the partial decrease of neutron rate signal, is a consequence of the too low plasma density operations, where higher Li flux is expected into the plasma. The dilution effect has not been observed for medium and high- density discharges by comparing the neutron production of similar shots with different wall conditions.

3. ISTTOK Results

A detailed description of the full liquid metal loop installed on ISTTOK to inject gallium at the plasma edge has been done elsewhere [14]. Fig. 7 shows the implemented setup with details of the experiment in the vicinity of the plasma-jet interaction region. The jets are generated by hydrostatic pressure, have a 2.3 mm diameter and 2.5 m/s flow velocity. The liquid metal injector has been built from a ¼" stainless steel pipe reduced to a suitable shaping nozzle and allows the positioning of the jet inside the tokamak chamber, within a 13 mm range (59<r<72 mm). The pressure required to generate a stable, vertical jet is generated by a 1.3 m height liquid metal column. The setup parameters have been chosen to ensure a 13 cm break-up-length (continuous part of the jet, before its spontaneous decomposition into droplets, due to Rayleigh instability). A detailed characterization of the produced jets is presented in [15].

The compatibility of ISTTOK plasmas with gallium jets has been discussed in [14] where it was shown that the liquid metal has a highly localized effect on the plasma characteristics essentially due to the low amount of atoms produced in the interaction. Gallium release from the jet surface can be due both to particle sputtering and evaporation. The jet is heated during its exposure to plasma and this increase in temperature is the main aspect that influences the evaporation rate. The temperature rise in a planar surface submitted to a power flux density q(t) can be written [16]:

$$\Delta T(t) = \frac{1}{\sqrt{\pi \rho C_P \kappa}} \int_0^t \frac{q(t-t')}{\sqrt{t'}} dt' \qquad (1)$$

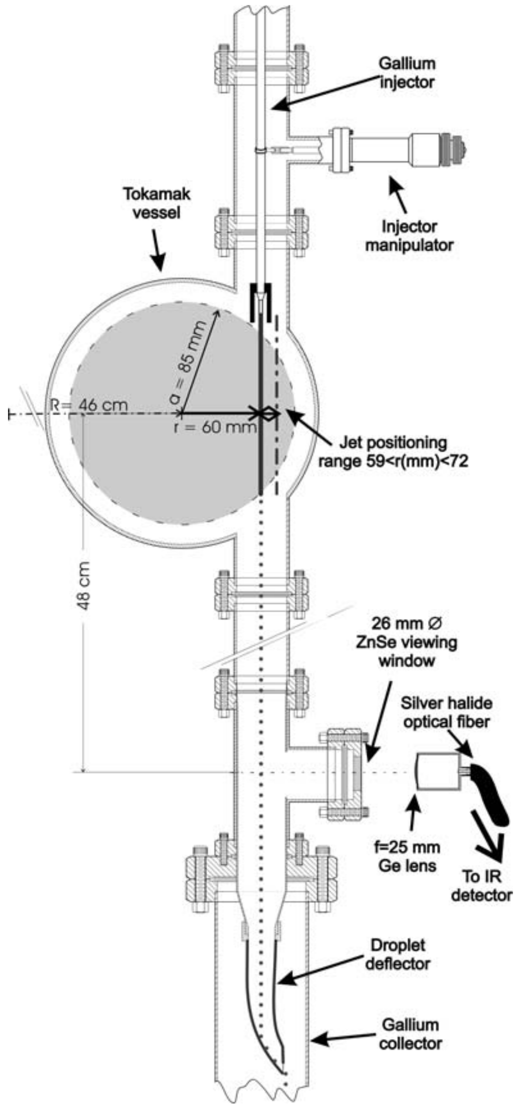


FIG.7. Cross-section of the experimental setup in the vicinity of the plasma-jet interaction region.

where C_P is the material specific heat, ρ its density and κ the thermal conductivity. It is possible to obtain the expected temperature increase of the gallium jet surface, while passing through the chamber, provided the heat fluxes along its path are known. These parameters have been measured, in ISTTOK using a copper probe [14]. The heat flux profile shown in Fig. 8 a) was obtained for 9 kW power input ohmic discharges. It is possible to integrate equation (1) using the best fit function indicated in this figure and performing the variable transformation: r → $\sqrt{(z^2+0.06^2)}$ → $\sqrt{((z_0+v_{jet}\,t)^2+0.06^2)}$, where z is a coordinate along the jet, z_0 the position of an element of fluid, at t = 0 s, and it is assumed that the injector is at a r = 60 mm. Since ISTTOK discharge is short duration the effect of the gravity acceleration has a small contribution to the change in the position of a gallium element of fluid present in the chamber and, for simplicity, has been disregarded. The results of these calculations for 16 kW discharges and several flow velocities are presented in Fig. 8 b). It is seen that the maximum expected temperature increase on the jet surface, in ISTTOK experiment, is about 98 ºC. Since at the input the liquid metal is at 75 ºC, the maximum temperature it could reach would be 173 ºC, for which gallium still has a low vapor pressure (~10^{-22} mBar!). It is also important to stress the behavior of the jet temperature as the flow velocity changes: as expected there is a clear decrease on the liquid metal surface temperature when velocity increases.

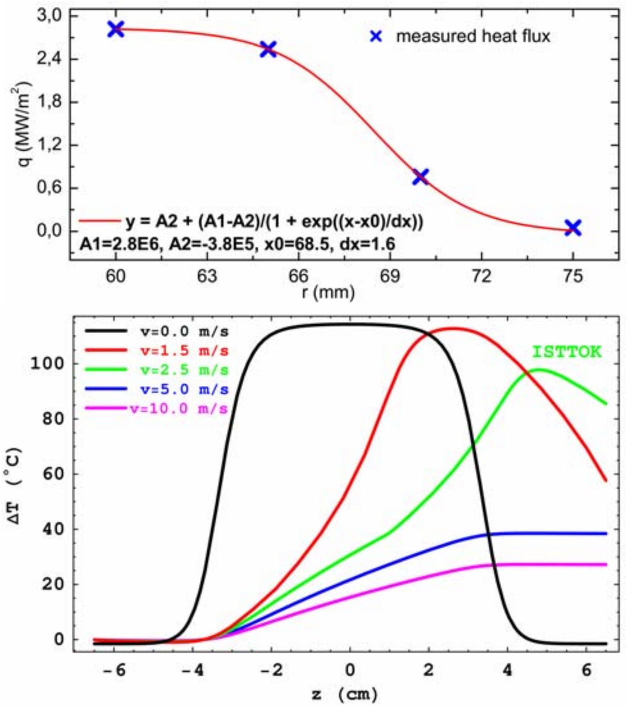


FIG. 8. a) Measured plasma heat flux profile in a 9 kW ISTTOK discharge and b) calculated temperature increase on the jet's surface, for several flow velocities, in a 16 kW discharge.

The emission of infrared radiation has been successfully used to measure the temperature of liquid lithium surfaces in FTU tokamak [17]. A HgCdTe infrared sensor, with a 6.7 μm cutoff wavelength, is being used in ISTTOK to monitor the radiation emitted by gallium droplets, after their interaction with the plasma, aiming at studying the power exhaustion capability of such thin jets. Measurements are performed at the viewing window schematically represented on the lower part of Fig. 7. The observation direction is perpendicular to the poloidal plane (in fact perpendicular to the direction represented in Fig. 7). At that distance from the chamber (~48 cm from the equatorial plane) the jet is already in droplet form and has reached thermal equilibrium since heat propagates about 3 mm (> droplet radius) before reaching that viewing chord. As such, the average power extracted by the gallium jet can be deduced from the specific heat definition by:

$$\bar{q} = \bar{m} C_{PGa} \Delta T \qquad (2)$$

Where $\bar{q}$ is the average input power, $\bar{m}$ is the liquid metal mass flow rate and ΔT is the temperature rise. A germanium meniscus lens, with broadband antireflection coating, 25.4 mm focal length and 24 mm diameter was used to focus the radiation emitted by gallium droplets on a 1 mm core diameter silver halide optical fiber which transmits the radiation to the cryogenically cooled (78 K) sensor.

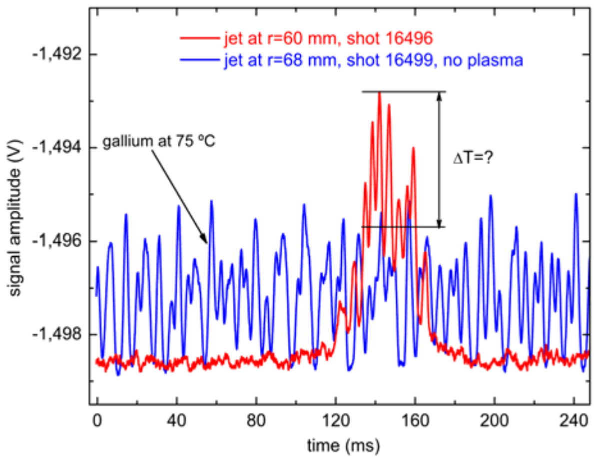


FIG. 9. Infrared sensor signal with and without plasma.

Measurements with this device have provided the results shown in Fig. 9. These were obtained, with the collection lens 85 mm away from the droplets position. Each one of the spikes shown in the curves corresponds to single droplets. The blue curve is a reference shot (no gas feed) showing the droplet behavior without plasma. This has been obtained with gallium injector at r = 68 mm. Another important outcome of the measurements, obtained from the analysis of Fig. 9, is that the jet suffers a small (<10 mm) radial displacement due to the influence of the plasma. This is noticed since the droplets are not observed prior to the shot (red curve in Fig. 9, injector at r = 60 mm) and they appear in the collection optics FOV during the discharge time (with the corresponding delay). This shows that there is a small tilt in the injector direction since the jet should be centered at r = 60 mm (it should be stressed that this corresponds to an angular shift less than 0.9º) and within the FOV (Fig.7). Previously acquired movies of the jet inside the chamber, during its interaction with the plasma, have not put this shift in evidence since at this location it is too small to be clearly identified. The observed increase in the signal amplitude, when compared to the shot without plasma, could only be related to an increase in the gallium jet temperature. Unfortunately, at this time, it is not possible to provide a value for this parameter since its measurement requires a calibration of the detector which procedure is under development. But it is worthy to mention that for both curves in Fig. 9 gallium, at the injector output, was at a temperature close to 75 ºC. Unfortunately the sensor response is not linear [17] and this value is not enough to obtain the droplets temperature increase due to the plasma interaction. The low signal amplitude obtained for the data presented in Fig. 9 (a few mV) is connected to two factors: 1) the low voltage output of the infrared sensor was connected directly to the tokamak data acquisition system without being amplified; 2) the sensor response was not the most recommended for the temperature range under consideration.

4. Conclusions and Perspectives

An extended analysis, although far from being exhaustive, has been performed among lithizated discharges, mainly at I_p=0.5 MA. An improved energy confinement behavior of lithizated discharges (up to 1.4 of τ_{ITER97}) has been confirmed on the whole range of plasma density. The analysis indicates that in lithizated discharges the threshold of SOC confinement rises from ~45÷50 ms up ~65÷70 ms. This behavior is accompanied by an increase of the volume density peaking factor above a threshold value of ~1.8 and by a heat transport largely dominated by electron conductivity, with the ion conductivity near to its neoclassical level even at high density. At higher plasma current (700-750 kA) more data are needed to confirm the behavior we have found at I_p= 500 KA.
A high purity of the plasma increases the efficiency of lower hybrid so that a strong and wide ITB develops into the plasma. The dilution due to the low electron density measured in these discharge should be overcome with a more careful pre-programmed gas fuelling. More data are needed to a better evaluation of the beneficial effects of lithizated discharges in presence of a relevant additional power.
Preliminary results about the power extraction capability of liquid metal jets in the ISTTOK tokamak have been presented while putting in evidence a radial displacement in the jet that has been shown to be due to its interaction with the plasma.
The experimental results obtained on ISTTOK, FTU and other devices like NSTX [18],TJ-II [19], CDX-U [20] and T-11M [7] point out, in a strong and clear evidence, that liquid metals are more than a potential candidate to solve the problem of plasma wall interaction. At least the only, but impressive results obtained when the devices are "lithizated" are sufficient to plan future experiments in divertor device and to test liquid metal in presence of strong heat load (≥ 10 MW/m^2). On FTU we are designing and projecting a new liquid lithium limiter able to act as main limiter to verify its capability to withstand high thermal load for few second.

The temperature calibration required to provide absolute value for the power extracted by the thin gallium jet is an ongoing task in ISTTOK. A new three-channel infrared sensor is planned for the next measurements, using a more sensitive detector (suitable for lower temperature values, cutoff wavelength of 9 μm) and higher spatial resolution which will allow following the motion of the gallium droplets along the referred radial displacement.
The exposure of liquid gallium jets in the FTU SOL is being envisaged but the observation performed in ISTTOK clearly shows that a thorough modellization is required before this can be implemented.

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