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# Technological aspects of lithium capillary-pore systems application in tokamak device

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**Abstract**

The paper concerns two problems. First, the usage of a lithium material basing on capillary-pore systems as a material of rail limiter working in conditions of considerable thermal ($\sim 10$ MW/m$^2$) and electromechanical loads caused by interaction with a tokamak plasma in a T-11M tokamak. The second problem is the influence of a modification of recycling for hydrogen, deuterium and helium due to lithium sputtering and deposition on the discharge conditions. 

*Keywords:* Lithium; Capillary-pore system; Plasma; Tokamak

## 1. Introduction

Application of liquid lithium as a plasma facing material for a tokamak reactor has a number of potential advantage in comparison with other materials and, probably, will help to solve the important problem of a considerable increase of an operational resource of the divertor plates without essential increasing $Z_{\text{eff}}$ of a plasma column [1–3]. However, the use of liquid lithium causes some other problems: (1) mechanical stability of liquid lithium films; (2) ion sputtering and evaporation (thermal emission) of lithium; (3) accumulation of lithium in plasma column; (4) lithium deposition on the surface of vacuum chamber; (5) modification of getter properties of a surface (gas recycling decrease) [2–4]. Capillary-pore systems (CPS) with lithium were proposed as a surface structure for divertor plates in [1,2] as a method of solving the problem of liquid lithium film stability.

During 1998–2000 [3,5,6] tests of liquid lithium film stability have been performed and tokamak plasma interaction processes with CPS rail type lithium limiter have been studied on T-11M tokamak. About 2000 experimental shots were carried out with hydrogen, deuterium and helium for the following parameters of tokamak discharge: $I_{\text{p}} \approx 90$ kA, $\Delta t \approx 0.1$ s. The CPS rail type lithium limiter had a semicylindrical form of surface with $r_0 \approx 2$ cm and was immersed to 5 cm depth at the bottom of the plasma column. Effective area of

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plasma–limiter interaction was approximately equal to 25 $cm^2$. Special heating unit provided an initial temperature of limiter from 20 up to 400 °C. Thermal load on limiter contact surface had a magnitude of $\sim 10$ MW/$m^2$ ($t \sim 50$ ms) in normal discharges and achieved $100 \div 200$ MW/$m^2$ ($t \sim 80$ μs) at the moment of disruption instability development. The results are partially published in [3]. Some data on physics of plasma–limiter interaction are presented in papers [5,6]. Fig. 1 shows a typical time behaviour of surface limiter temperature $T_s(t)$ measured by a IR radiometer and specific thermal load $q_s(t)$ calculated from $T_s(t)$. This paper contains data analysis of two experimental cycles of the research on lithium T-11M program. These data are: limiter erosion, getter effects and modifications of CPS properties under tokamak plasma conditions.

## 2. Properties of lithium capillary-pore structures and their stability under plasma impact

The first series of experiments on T-11M tokamak was carried out on CPS-based lithium filled limiter made of multi-layer molybdenum mesh with effective pore radius ($R_{eff}$) of $\sim 75$ μm. In these tests [6] lithium drop losses have been observed due to the effect of MHD forces on liquid metal in the CPS. The CPS surface material has not shown degradation after the first series of experiments ($\sim 10^3$ discharges) under plasma impact. The design of an advanced lithium limiter was based on a stainless steel mesh (304 type wire, diameter 30 μm, $R_{eff} \sim 15$ μm). As the previous one, the new CPS-based lithium diaphragm demonstrated the total stability of geometric characteristics and chemical composition under plasma conditions ($\sim 2 \times 10^3$ discharges). Surface temperature of CPS structure material does not reach a melting point for steel because of efficient cooling by lithium evaporation. In this case $R_{eff}$ was 15 μm, so that conditions of lithium deduction in the CPS was carried out in all modes of tests and lithium drop losses were completely prevented.

## 3. Erosion of lithium under plasma conditions on T-11M

Limiter surface erosion (losses of material) in a tokamak is determined by balance of two opposite processes during discharge: by an emission of atoms and ion deposition on the limiter. In our experiments we could register only lithium emis-

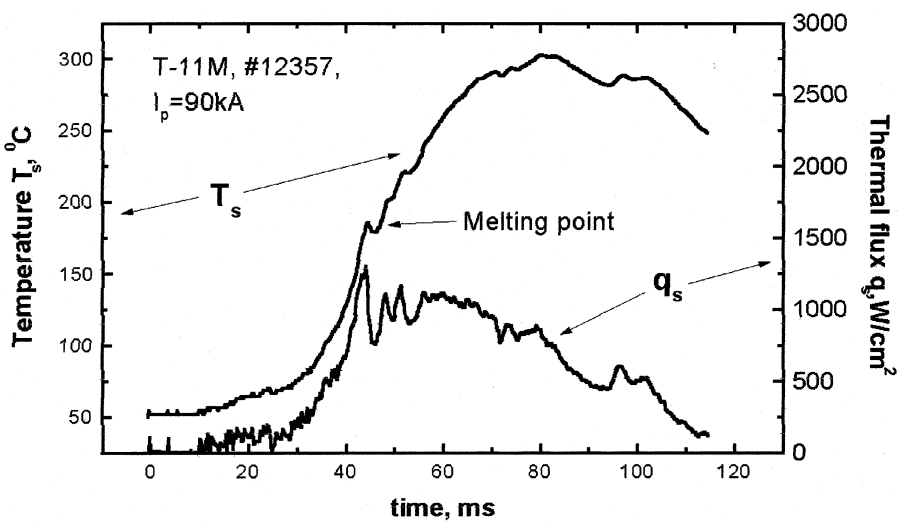


Fig. 1. Behaviour of surface temperature and specific thermal power flux in time during discharge.

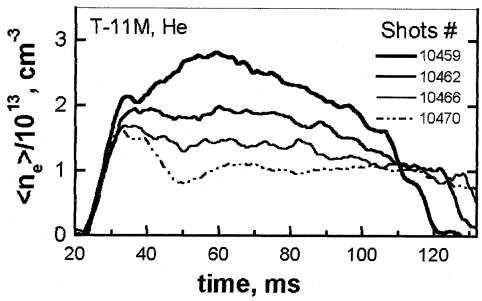


Fig. 2. Electron density evolution from shot to shot in discharges with high temperature at lithium limiter (near 500 °C); working gas—He.

sion process by the intensity of neutral lithium line LiI (670.8 nm). For a quantitative estimation of lithium emission from the limiter surface we used a method of calibration of the spectrometer by a stimulated emission. For this purpose we applied a pulse of negative voltage of $-400$ V between the limiter and tokamak chamber during discharge. Simultaneously, we registered pulse of ion current from plasma to limiter (100–140 A), growth of electron density $\Delta n_e(t)$ and LiI intensity (Fig. 2). From the rate of electron density growth $dN_e/dt$ supposing that practically all lithium is captured by plasma, we can estimate an absolute value of additional stimulated flux of lithium atoms from the limiter, $\Delta Q_s \approx 4 \times 10^{20}$ s$^{-1}$. We can estimate the magnitude of lithium emission from a surface in a conventional condition (without an additional potential) supposing that LiI intensity is approximately proportional to neutral lithium flux from the limiter. In such a case we obtain the value of total emission flux of lithium atoms from the limiter $Q_e \approx 6.2 \times 10^{20}$ s$^{-1}$. Magnitude of total ion flux to the limiter is $Q_i \approx 8.8 \times 10^{20}$ s$^{-1}$ (ion saturation current). If we suppose that the main mechanism of Li emission is ion sputtering then the ratio $k_s = Q_e/Q_i \approx 0.7$ is the averaged sputtering yield of Li atoms from a surface of liquid lithium by ions with 100–400 eV energy. During the early period of the T-11M discharge with an initial limiter temperature of $\sim 100$ °C (Fig. 1) the temperature of surface achieves a melting point of $\sim 180$ °C in 45 ms and then is stabilised at a level of $\sim 300$ °C. Comparing the obtained value $k_s$ with beam experimental data [7] by measurement of ion sputtering coefficient for liquid lithium by $D^+$ and $Li^+$ ions, we have revealed a good enough correspondence with $D^+$ sputtering processes. Since the effective area of plasma–limiter interaction estimated according to spots of lithium melting, has a value of $S \approx 25$ cm$^2$, we find specific emission $q_e = Q_e/S \approx 2.5 \times 10^{19}$ s$^{-1}$ cm$^2$. This flux is measured at a surface temperature about $T_s \sim 300$ °C. At this temperature the thermal emission of lithium according to [8] will be equal to $q$th (300 °C) $\approx 1 \times 10^{16}$ s$^{-1}$ cm$^2$. The level of thermal emission qth $\approx 1 \times 10^{19}$ s$^{-1}$ cm$^2$ for the lithium is achieved already at $T_s \sim 500$ °C. Therefore, we can assume, that at $T_s > 500$ °C the main channel of lithium erosion should be thermal emission. In practice, beginning with a limiter preheated up to 300 °C we observed approximate doubling of the electron plasma density [3]. It shows that in our conditions at $T_s < 500$ °C the main physical mechanism of lithium erosion is probable to be the ion sputtering. It should be noted, that an accumulation of lithium during discharge and the step-by-step substitution of deuterium by it may explain the monotonic growth of lithium emission during discharge [3], because the coefficient of sputtering of liquid lithium by $Li^+$ ion exceeds $k_s$ for deuterium approximately five times [7].

## 4. Lithium getter effects on T-11M

A decrease of electron density from discharge to discharge during tokamak operation with lithium was observed earlier on deuterium [4,9]. In our experiments we have detected a similar appearance on helium too (Fig. 2). During sequence of 10 shots we can see an electron density decreasing shot by shot as a result of helium absorption. On helium the effect of a drop recycling is a little bit more feeble, than on hydrogen and deuterium. To be convinced, that helium has been absorbed by walls of the vacuum chamber we have carried out a series of discharges with the hot tokamak chamber. The absorption of helium has appeared to be reduced at a chamber temperature of above 50 °C and practically disappeared at above 150 °C. By operation with vacuum

chamber temperature up to 200 °C helium recycling is completely recovered. Measurements of gas pressure in the tokamak chamber during 100 s just after discharge with helium plasma at room temperature (Fig. 3) show that after an usual pressure drop there is some plateau and even up rise of pressure, i.e. outcome of absorbed helium from the walls of the chamber surface covered with lithium. The total amount of helium approximately corresponds to amount of gas in the tokamak vacuum chamber before discharge. Thus, the helium is not stored in walls and in phase between discharges, there is an inverse process of a desorption. By operation on deuterium with tokamak vacuum chamber heated up to 250 °C we have not detected recycling to restore. However, the mass-spectrometer analysis of the gas from the heated limiter, which has been carried out after a series of discharges on deuterium, has shown a sharp growth of deuterium emission at a temperature of above 325 °C (Fig. 4). At a further raise of temperature up to 420 °C the deuterium pressure decrease has been exponential with time of $\sim 5$ min. The total number of deuterium molecules has made up approximately $1.6 \times 10^{19}$ that exceeds the total amount of deuterium puffed in the tokamak chamber during discharge approximately three times. Thus, we may assume that restoring of recycling on hydrogen or deuterium is possible by heat of the tokamak vacuum chamber up to 350–400 °C.

The data on the modification of recycling for deuterium one can use to estimate the amount of

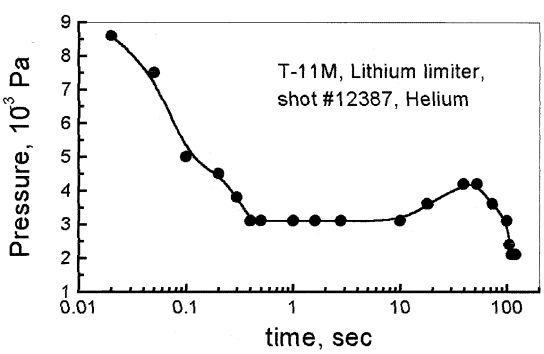


Fig. 3. Behaviour of pressure in the tokamak chamber just after discharge.

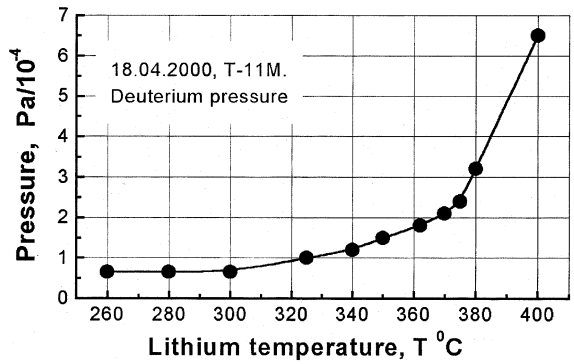


Fig. 4. Behaviour of deuterium pressure in the tokamak chamber under lithium limiter heating.

lithium deposited on a vacuum wall of the chamber of a tokamak. Supposing coefficient of implantation of deuterium in lithium is $\sim 1$ [10] and $(n_e/n_e$ is $\sim 0.2$ (relative drop of an electron density per one discharge from our data) we shall receive an amount of lithium atoms deposited on the chamber walls per one discharge $N_d \sim 10^{19}$ or $\sim 0.2$ monolayers of lithium.

In conclusion we should note that a complete rehabilitation (restoring of getter properties) walls of the chamber after long-lived lithium deposition is achieved by usual boronization procedure.

## 5. Conclusions

Experiments on T-11M tokamak with lithium CPS limiter have shown:

1. Lithium CPS material does not fail and does not loose a noticeable amount of lithium in long operation of the device at temperatures from 20 up to 400 °C at thermal load of $\sim 10$ MW/m$^2$.
2. Lithium surface erosion in T-11M conditions at $T_s < 500$ °C is caused mainly by ion sputtering and it is at a level of $\sim 2 \times 10^{19}$ s$^{-1}$ cm$^2$ for $T_s < 500$ °C; the $T_s > 500$ °C case corresponds to thermal emission as the main mechanism of lithium loss. The liquid lithium sputtering coefficient by $D^+$ ions is about 0.7 and is in an agreement with experimental data reported by Allain [7].

3. Lithium deposition on the walls of tokamak vacuum chamber reduces recycling of the working gas and can lead to a considerable density drop.
4. The complete rehabilitation (restoring of getter properties) of the vacuum chamber walls after long lithium deposition is achieved for helium and deuterium by vacuum chamber heating up to 100–150 and up to 350–400 °C, respectively.

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