Experimental Study on Tokamak Plasma Interaction with Lithium Capillary-Pore Systems

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Abstract. Behavior and influence of two lithium capillary-pore system - based rail limiter models on the discharge parameters at heat fluxes of about 10 MW/m^2 and discharge duration of 0.1 s have been experimentally investigated in a T-11M to substantiate the lithium capillary-pore system (CPS) application as a plasma facing material. Limiter thermal loads, plasma-limiter thermal balance, physics of lithium erosion and lithium accumulation in tokamak plasma have been studied. Reradiation effects in the lithium vapor at the plasma periphery has been detected and studied. The stability of lithium CPS under tokamak plasma conditions has been investigated and confirmed.

1. Introduction

Lithium filled capillary-pore system (CPS) application as a plasma facing material for a tokamak reactor has a number of advantages in comparison with other materials and, probably, will help in solving the important problem of a considerable increase of an operational resource of divertor plates without essential increase in Z_{eff} of a plasma column [1,2,3]. CPS with lithium proposed as a divertor plate surface structure in [1,2] provides the means for solving the liquid lithium film stability problem. However, the use of liquid lithium in tokamak raises some other problems: 1) energy removal ability from plasma by radiation, 2) lithium erosion due to: arcs, ion sputtering and evaporation (thermal emission), 3) lithium behavior in a plasma column, 4) lithium deposition on the surface of vacuum chamber, 5) modification of surface getter properties (gas recycling decrease) [2,3,4]. The tests (~ 2000 experimental shots) of liquid lithium surface stability have been performed and tokamak plasma interaction processes with CPS rail lithium limiter has been studied in a T-11M tokamak [3,5,6]. This paper contains data analysis of the last two experimental cycles of the research on lithium T-11M program with two types of lithium CPS-based limiters. The limiter thermal loads, plasma-limiter thermal balance, physics of plasma - lithium interaction and lithium accumulation in plasma are considered.

2. Lithium Limiter Design

Lithium CPS of two types (Table. I) was used as a rail limiter receiving element of T-11M tokamak (*FIG. 1*). It has a semicylindrical shape of surface formed by the lithium filled mat of meshes and was immersed to a 5 cm depth at the bottom of the plasma column. Effective area of plasma-limiter interaction was approximately equal to 25 cm^2. Special heating unit in case of a limiter provided the limiter initial temperature from 20 up to 400 ^oC and make possible restoring of lithium film at a CPS surface in experiments with the temperature of <180 ^oC.

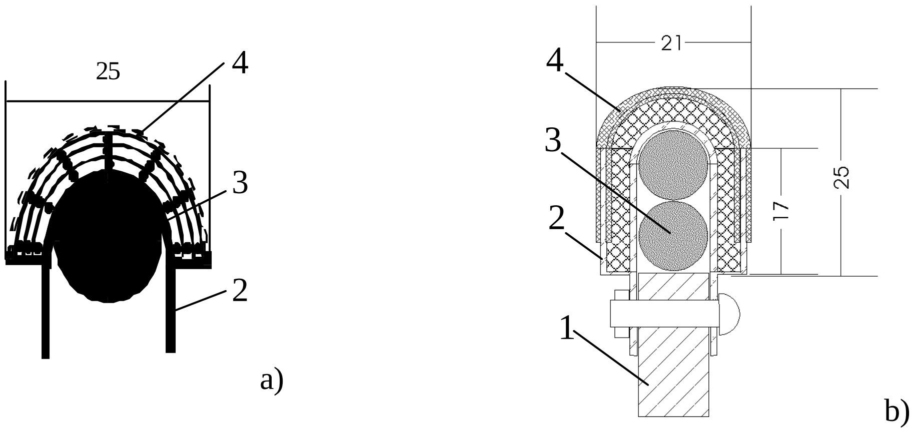

FIG. 1. Lithium limiter of T-11M tokamak:receiving element section of variant 1 (a) and variant 2 (b); 1-support, 2-case, 3-heater, 4- CPS.

TABLE I. LIMITER CPS PARAMETERS

Parameter	Data	
	Variant 1	Variant 2
CPS material	Molybdenum mesh	304 SS mesh
Cell size, mm	0.15×0.15	0.03×0.1
Wire diameter, mm	0.1	0.03
Limiter length, mm	160	150
Limiter diameter, mm	25	21
Amount of lithium, g	13	8.5

3. Experimental results on CPS Damage and Splash Resistance

Experiments have been conducted in the T-11M tokamak. Power load at a limiter contact surface determined on the data of infrared radiometry and of special experiments for thermodynamic parameter evaluation had a magnitude of ~10 MW/m^2 (t ~ 50 ms) with normal discharges and achieved 100÷200 MW/m^2 (t ~ 80 μs) in the development of disruption instability.

The lithium drop losses have been observed on tests of limiter type 1 [6] due to the effect of MHD forces on liquid metal in the CPS (the pressure pulses up to $P = 2\cdot10^4$ Pa caused at a current I_d = 600 A through limiter and at a toroidal magnetic field of 1.3 T under disruption). But the CPS type 1 surface material has not shown degradation after the experiments (~ 10^3 discharges) under plasma impact. The advanced lithium limiter design was based on the CPS type 2. The ends of new type limiter was closed by membranes. The new limiter has demonstrated the full damage-resistance and splash-resistance at ~ $2\cdot10^3$ discharges. The lithium confinement conditions in the CPS ($P< 7\cdot10^4$ Pa) have been met in all test modes and lithium drop losses have been completely prevented in this case.

The limiter surface observation during experiments has shown, that lithium film has been evident on the CPS surface constantly at the initial temperature of > 180 ^{o}C (liquid state) and in experiments at initial temperature of < 180 °C (solid state) has been completely restored by limiter preheating above lithium melting temperature. Thus, the lithium limiter is self-restored.

4. Calorimetric Measurements of Limiter Thermal Flux and its Reradiation

The thermal energy flux to the limiter Q_{lim} have been determined from the data on limiter heating ΔT_{lim} during plasma discharge and its thermodynamic parameters. The total ohmic energy of discharge E_{oh}, has been estimated by electromagnetic measurements. The behavior of the relation Q_{lim}/E_{oh} as a function of limiter preliminary temperature for the He experiments is given in *FIG. 2.* It shows that the ohmic contribution E_{oh} has a weak variety while the relation Q_{lim}/E_{oh} changes approximately twice on limiter heating. This is well consistent with a quantity of the local specific thermal flux on the limiter surface $q_s(t)$. Such behavior may be explained by magnification of radiative losses from a plasma due to lithium. In prolonged pulse testing it may be established a stationary equilibrium between the total radiative losses of plasma, limiter thermal flux and lithium emission from limiter. This equilibrium should be the most important feature of the radiative divertor modeling.

5. Erosion of Lithium Limiter Surface

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 emission 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 the negative voltage of 400 V between the limiter and tokamak chamber during discharge. Simultaneously, we registered a pulse of ion current from plasma to limiter (100-140 A), growth of electron density $\Delta n_e(t)$ and LiI intensity (*FIG. 3*). According to the rate of electron density growth dn_e/dt under the assumption that practically all lithium is captured by plasma, we can estimate an

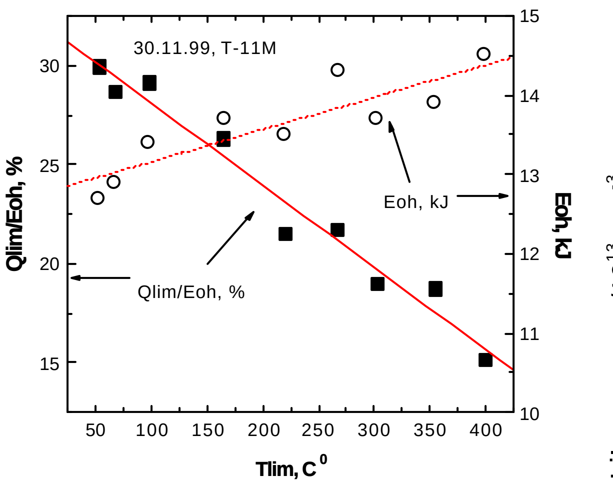

FIG. 2. Behaviour of relative energy incoming to limiter Q_{lim}/E_{oh} versus preheating temperature of limiter in ohmic mode. E_{oh} – total ohmic energy of discharge.

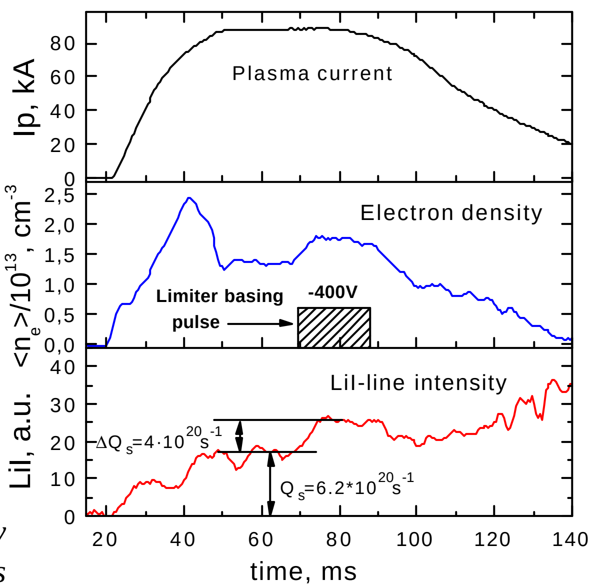

FIG. 3. Experimental evaluation of liquid Li sputtering yield by D^+ ions from limiter.

absolute value of additional stimulated
flux of lithium atoms from the limiter of $\Delta Q_s \approx 4\cdot10^{20}$ s^{-1}. The magnitude of lithium emission from a surface under conventional condition (without an additional potential) we may estimate under the assumption 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 of $Q_e \approx 6.2\cdot10^{20}s^{-1}$. The magnitude of total ion flux to the limiter is $Q_i \approx 8.8\cdot10^{20}s^{-1}$ (ion saturation current). Suppose that the main mechanism of lithium emission is ion sputtering then the $k_s=Q_e/Q_i \approx 0.7$ ratio is the averaged sputtering yield of lithium atoms from a surface of liquid lithium by ions with 100-400 eV energy. During a start period of the T-11M discharge at the initial limiter temperature of ~100 ^oC after 45 ms the surface temperature achieves a melting point of ~180 ^oC and then is stabilised at a level of ~300 ^oC. Comparison between the obtained value k_s and liquid lithium ion sputtering coefficient by D^+ [7] shows rather good agreement. Since the effective area of plasma-limiter interaction estimated according to lithium melting spots, was equal to $S \approx 25$ cm^2 we found a specific emission as $q_e = Q_e/S \approx 2.5\cdot10^{19}s^{-1}\cdot cm^2$. This flux is measured at the surface temperature of T_s~300 ^oC. At this temperature the thermal emission of lithium will be equal to q_{th} (300 ^oC) $\approx 1\cdot10^{16}$ $s^{-1}\cdot cm^2$ [8]. The level of thermal emission $q_{th} \approx 1\cdot10^{19}s^{-1}\cdot cm^2$ for the lithium is achieved already at T_s~500 ^oC. Therefore, we can assume, that at T_s>500 ^oC the main channel of lithium erosion should be the thermal emission. Really, early in the test of limiter preheated up to 300 ^oC, we observed doubling approximately the electron plasma density [3]. It shows that under our conditions at T_s<500 ^oC the main physical mechanism of lithium erosion is likely to be ion sputtering. It should be noted, that an lithium accumulation during discharge and step by step substitution of deuterium by it may be explained by a monotonic growth of lithium emission during discharge [3], since the coefficient of sputtering of liquid lithium by Li^+ ion exceeds k_s for deuterium approximately 5 times [7].

6. Contamination of Plasma by Lithium

Taking into account, that during the discharge a considerable amount of lithium incomes to plasma, the step by step substitutions of deuterium by lithium may be expected. It should show up as plasma Z_{eff} growth from 1 up to 3 and magnification of SX-continuum intensity. However, proceeding from the SXR measured electron temperature $T_e(0)$ at the center of the plasma cord and electrical conductivity estimation in the guess of q (0) =1 it has been detected an unexpectedly low level of Z_{eff}. In the low density range ($n_e < 2\cdot10^{19}$ m^{-3}) of discharges Z_{eff} has not exceeded 1.2 and has made up no more than 1.4 - 1.5 at n_e= 4-5$\cdot10^{19}$ m^{-3}. Thus $T_e(0)$ was within the limits of 270-400 eV. Even after disruption, when the intensity of the continuum increased by the order and almost complete substitution of deuterium by lithium was possible Z_{eff} (0) did not increased above 1.8-2. It should be assumed, that lithium is not accumulated at the center of the plasma column and remains mainly at the periphery or an electrical plasma conductivity in these modes is noticeably determined by the accelerated electrons ("slow runaway") not detected in the experiment because of a weak roentgen radiation. In experiments with helium Z_{eff} (0) did not essentially exceed 2. However, for stationary lithium distribution in the plasma column section the discharge duration of 0.1 s was not enough.

7. Gas Retention

In experiments on T-11M it has been revealed [9], that the desorption of deuterium captured by limiter lithium surface has been taken place during the discharge at the heating temperature of above 350 ^oC. Thus, coupling between lithium and deuterium proved to be weak.

8. Summary

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

1. Lithium CPS material has not failed and has not lost a noticeable amount of lithium over a long period of device operation at the temperatures from 20 up to 400 ^{o}C at thermal load of ~10 MW/m^2. CPS have demonstrated full damage- and splash-resistance which has been provided with the appropriate geometric parameters.
2. Lithium surface erosion under T-11M conditions at T_s<500 ^{o}C is caused mainly by ion sputtering and it is at a level of $\sim 2 \cdot 10^{19}\ s^{-1} \cdot cm^{2}$ for T_s<500 ^{o}C; in the case when T_s>500 ^{o}C a thermal emission is probably a main erosion mechanism. The liquid lithium sputtering coefficient by D^+ ions is about 0.7 and is in an agreement with experimental data of Allain.
3. The measurement of thermal flux on the limiter shows a decrease of limiter thermal load (2 times) as a result of lithium emission increase. That may be a step towards to RI mode and a radiative divertor.
4. The plasma center contamination by lithium was low in T-11M experiment.

References

[1] EVTIKHIN V.A., et al., Liquid Lithium Tokamak Reactor, Proc. of 16th IAEA Fusion Energy Conference, 7-11 Oct. 1996, Montreal, Canada. Fusion Energy 1996, IAEA, Vienna, 1997, vol. 3, p. 659-665.

[2] MIKHAILOV V.N., et al., Lithium for fusion reactors and space nuclear power systems of XXI century, Energoatomizdat, Moscow, 1999 (in Russian).

[3] LAZAREV V.B., et al., Compatibility of the Lithium Capillary Limiter with Plasma in T-11M, 26th EPS Conf. on Contr. Fusion Plasma Physics, ECA, 1999, vol. 231, p. 845.

[4] HOGAN J.T., et al., Lithium effects in plasmas, Nucl. Fusion, **37** (1997) 705.

[5] EVTIKHIN V. A., et al., Design, calculation and experimental studies for liquid metal system main parameters in support of the liquid lithium fusion reactor. Proc. of 17th IAEA Fusion Energy Conference, 19-24 Oct. 1998, Iokohama, Japan, Fusion Energy 1998, IAEA, Vienna, 1999, vol. 4, p. 1039-1313 (CD-ROM file FTP/27).

[6] EVTIKHIN V. A., et al., Energy removal and MHD performance of lithium capillary-pore system for divertor target application, Fus. Engg. Design. (in press).

[7] ALLAIN J. P., et al., Measurement and Modelling of Li Sputtering, 40th Plasma Physics Div. APS, 1998, R8P3.

[8] KOVALENKO V. F., Thermophysics processes and electrovacuum devices, Moscow, The Soviet wireless, 1975 (in Russian).

[9] EVTIKHIN V.A., et al., Technological Aspects of Lithium Capillary-Pore Systems Application in Tokamak Device, Book of Abstracts of the 21st SOFT, Sept. 2000, Madrid, Spain, p. 292.