Available online at www.sciencedirect.com
SCIENCE DIRECT®
Fusion Engineering and Design 81 (2006) 113–119
Fusion Engineering and Design
www.elsevier.com/locate/fusengdes

# The tests of liquid metals (Ga, Li) as plasma facing components in T-3M and T-11M tokamaks

S.V. Mirnov [a,*], V.A. Evtikhin [b]

[a] TRINITI Troitsk, Moscow Reg. 142190, Russia
[b] State Enterprise, "Red Star" Electrolitnij Proezd 1a, 115 230 Moscow, Russia

Received 1 February 2005; received in revised form 4 October 2005; accepted 4 October 2005
Available online 27 December 2005

**Abstract**

The experimental study of liquid metals (Ga, Li) as tokamak Plasma Facing Component (PFC) was undertaken in Russian T-3M and T-11M tokamaks ($I_p \leq 100$ k A, $B_T \approx 1$ T). In T-3M droplet stream and film flow Ga limiters were tested. In T-11M the experiments with Li capillary pore systems (CPS) as rail limiter for investigation of real Li losses in tokamak boundary condition were performed. It was shown, that a liquid metal (Ga, Li) PFC can be used in tokamak as droplet and CPS structures. The main channel of lithium erosion looks like, as ion sputtering. The motion towards the tokamak-reactor with Li PFC seems possible and has no serious physical obstacles.


*Keywords:* Tokamak; Plasma facing components; Liquid metal; Splashing; Sputtering

## 1. Introduction

The plasma facing components (PFC) of tokamak-reactor (limiters, diverter plates and first wall) will be subjected to high fluxes of fast ions ($D^+$, $T^+$) and high electron heat loads (~10 MW/m$^2$). Melting, cracking, blistering, low cyclic fatigue of solid PFC may occur in such conditions. The problems of PFC degradation may be overcome by substitution of solid PFC for liquid metal (LM). This idea has long been suggested [1]. The best candidates for LM PFC are Li and Ga. Li was selected because it has two specific qualities. Firstly, the low melting temperature $T_M = 180\,^\circ$C and low vapor pressure in 200–600 °C temperature range (lithium boiling temperature $T_B = 1358\,^\circ$C) seem compatible with typical temperatures of reactor constructions. Secondly lithium as metal has the lowest nuclei charge $Z = 3$.

Ga with $T_M = 32\,^\circ$C and $T_B = 2400\,^\circ$C should be next perspective metal. The useful temperature range for Ga can be 900 °C, two and three times higher then for Li. Unfortunately, its $Z = 31$ is too high.

However, the evaporation is not a unique channel of LM loss and its emission to plasma. The practical

* Corresponding author. Tel.: +7 95 334 55 80;
fax: +7 95 334 55 10.
*E-mail address:* mirnov@triniti.ru (S.V. Mirnov).

0920-3796/$ – see front matter © 2005 Elsevier B.V. All rights reserved.
doi:10.1016/j.fusengdes.2005.10.003

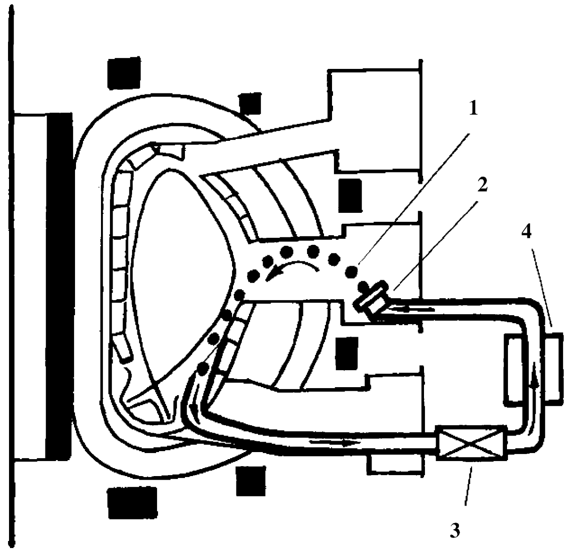


Fig. 1. Idea of the ballistic LM-droplet limiter for ITER-like tokamak-reactor. (1) droplet stream, (2) MHD-shaper, (3) MHD-pump, and (4) heat exchanger.

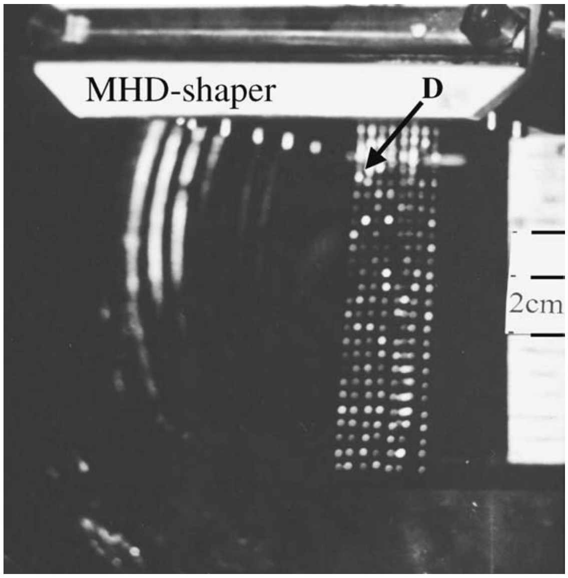


Fig. 2. View to Ga-droplet stream (D) and MHD-shaper in T-3M tokamak chamber.

use of liquid metals in real tokamaks met a number of serious difficulties:

- mechanical problems of liquid metal injection in a strong magnetic field,
- liquid metal splashing under MHD-forces in plasma instabilities,
- abnormal liquid metal erosion, like as unipolar arcs or local emission (like “carbon blooms”),
- abnormal sputtering by all kinds of bombarding ions and self-sputtering with yield higher than 1.

Today we have two practical solutions of LM injection and splashing problems. The first is a LM jet-drop curtain suggestion (Murav’ev [2]) that permits us to eliminate the influence of ponder-motive forces induced by liquid metal flow in magnetic field. The some idea of the ballistic jet-drop limiter for ITER-like reactors is presented in Fig. 1. Fig. 2 shows the realization of Ga-drop limiter in T-3M tokamak.

The second idea to use LM in tokamaks as PFC was based on the surface tension forces in capillary channels that may also be used to compensate ponder-motive forces (Evtikhin et al. [3,4]). These capillary channels from Mo, SS, V or W may be accomplished in the form of so called “capillary pore systems” (CPS). Self-regeneration of liquid metal surface, contacted with plasma, is the main property of such structures. Two micro photos of CPS from 100 μm Mo-grids with and without Li filling and scheme of CPS use as PFC with water cooling are presented in Fig. 3 (top and down).

Both methods permit us to have a stable LM PFC in real tokamak boundary conditions as it was validated by experiments. The idea of LM film limiter was unsuccessful. The steady state flow of LM-film has hindered by a tokamak MHD-perturbations.

The special program has been initiated in USSR and RF to reveal the features of LM PFC. The gallium-based LM limiters [2,5,6] have been designed and tested in T-3M tokamak (1989–1992, NPO “Energy”, Kurchatov Inst., Efremov Inst., Electrodynamic Inst. of Ukranien AS). The lithium rail limiters based on CPS have been tested in tokamak T-11M (1998–2005, TRINITI, “Red Star”). The LM PFC compatibility with tokamak plasma was the main topic of these investigations [7–9].

## 2. Test conditions

The main parameters of test tokamaks were the following: T-3M – $R/a = 0.9/0.15$ m, $B_T = 1$ T,

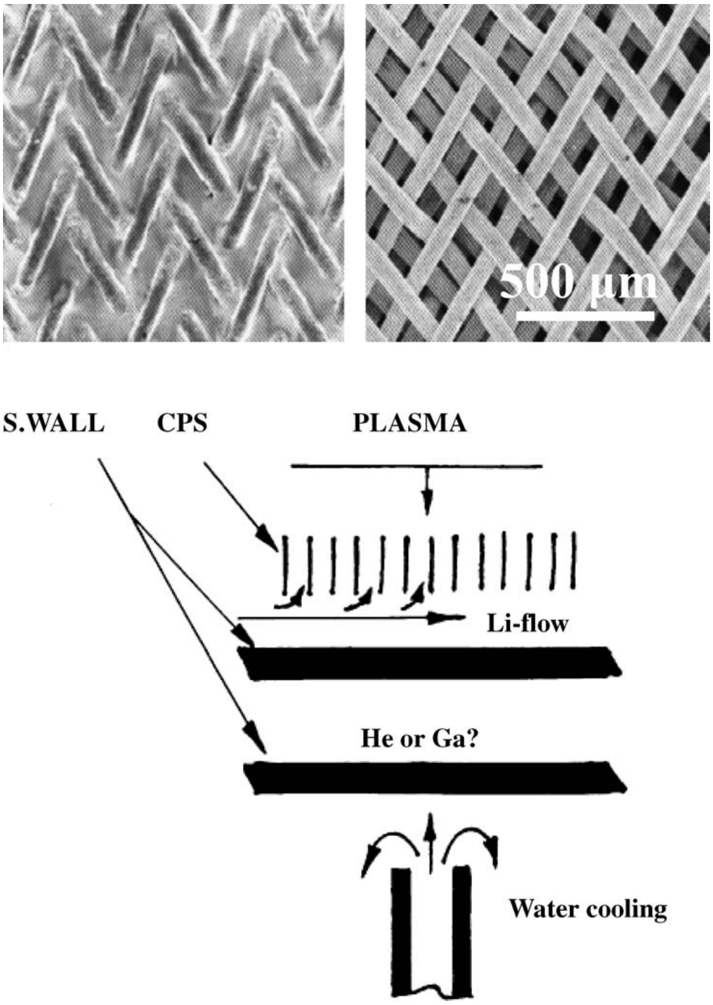


Fig. 3. CPS with (left) and without (right) Li filling (top). Scheme of CPS use as PFC with water cooling (down). S.WALL – double steel or Vanadium walls with divided gap (He or Ga).

plasma current $J_p \approx 30$–$40$ k A, $n_e = (1$–$2) \times 10^{19}$ m$^{-3}$, $T_e(0) = 250$ eV [5], discharge pulse length was about 0.1 s; T-11M – $R/a = 0.7/0.2$ m, $B_T = 1$ T, plasma current $J_p \approx 100$ k A, $n_e = (2$–$4) \times 10^{19}$ m$^{-3}$, $T_e(0) = 400$ eV [7], pulse length was 0.1–0.3 s. The density of heat flux to limiter in both cases was about 5–10 MW/m$^2$. The similar power load is expected on the ITER diverter plates.

The patterns of the T-3M (droplets) and T-11M (CPS) experiments are the similar. In T-3M Ga fell as two rows droplet stream with equivalent depth 4 cm. The droplets (Fig. 2) were created by special MHD shaper [5], which excited the Rayleigh instability in the LM flow. A droplets diameter was 2–4 mm. Its velocity (2–5 m/s) depended on MHD-pump. The droplet stream was sunk 1.5–2 cm into plasma column like vertical rail limiter. The regimes of discharge with graphite and Ga-limiters might be compared. The calculation showed that droplet surface heated up to 200 °C during pass through plasma.

The scheme of Li—experiment is shown in Fig. 4. Movable (from shot to shot) Mo-horizontal rail limiter with 1 mm lithium CPS shielding [7–9] was inserted into plasma approximately up to 5 cm, thus limiting plasma column aperture. Conventional graphite limiter was placed in the opposite port relative to the

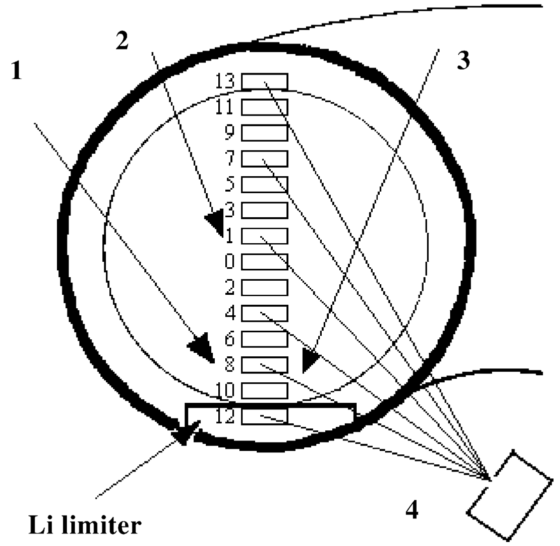


Fig. 4. Geometry of Li—experiment in T-11M. (1) LiI, LiII and visible radiation ΣI measurements, (2) SXR (soft X-ray) channels, and (3) IR (infrared)-observation, four-multi-channels of total radiation measurements.

lithium one. Standard thermocouples, soft X-ray and optical diagnostics (LiI, LiII and total visible light ΣI) were applied to observe lithium flux into the plasma. Besides a 15-channel bolometer system was set up and infrared diagnostics (IR) was developed to measure the limiter surface temperature. The typical limiter temperature addition during discharge was 250 °C. The preliminary heater incorporated in the limiter structure permitted us to increase the final limiter temperature up to 700 °C. To prevent the impurity (O, N) influence upon erosion results the limiter surface was cleaned in experiments with preliminary electron bombardment by special glow discharge.

## 3. Ga, Li interaction with plasma

In Fig. 5 [6] we can compare two discharges with Ga (exactly, GIS alloy = 67% Ga + 20.5% In + 12.5% Sn) and graphite limiters. The plasma parameters of both experiments are the similar. We can see the sharp distinction in total radiation behavior only. Ga-limiter "loses" in initial stage of discharge, but "wins" in the main part. The electrical conductivity ($J_p/V_p$) is the same and $n_e$ in case of Ga-limiter has no visible increase during discharge. The last result gives us some evi-

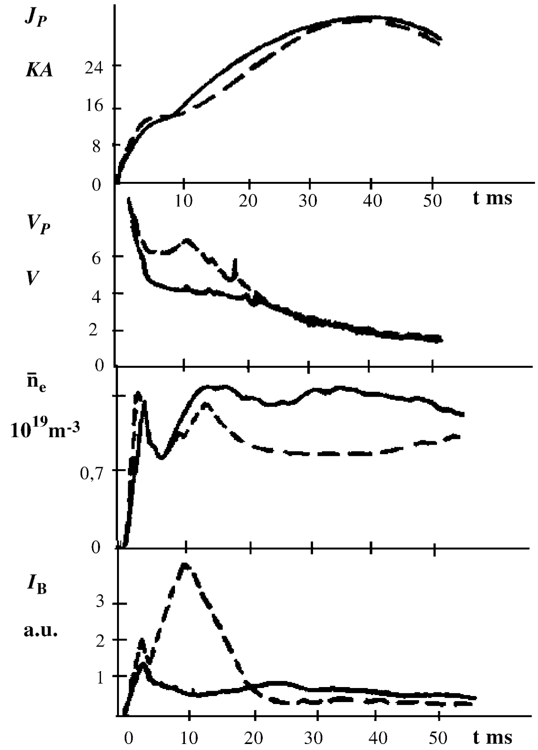


Fig. 5. Comparison of C (solid lines) and Ga (dotted) experiments in T-3M. Plasma current-$J_p$, voltage-$V_p$, mean electron density-$n_e$, total radiation-$I_B$.

dence, that self-sputtering yield $Ga^+ + Ga$ is not higher 1 in T-3M boundary condition ($T_e \approx 20$ eV).

Fig. 6 shows the wave forms: plasma current $J_p(t)$, intensity of lithium light Li($I$), as indicator of lithium erosion, IR-signal, as indicator of limiter surface temperature ($T_L \leq 250$ °C), and mean plasma density $n_e(t)$ during $D_2$ discharge in T-11M. All parameters achieved some quasi-state regime which is the result of CPS cooling due to heat contact with Mo-heat accumulator [9]. As we can see catastrophic-like events leading to spontaneous lithium injection are absent. That is a common feature of all MHD stable T-11M discharges within the whole lithium temperature range (from 20 to 600 °C). That means an absence of intensive unipolar arcs or lithium blooms. Lithium and graphite limiters worked roughly similar with one difference: the total radiation losses of plasma center in Li-case are several times lower than in C-case. Preheating of the lithium limiter increases a lithium flux into plasma that was detected by lithium, SX and total radiation in the vicin-

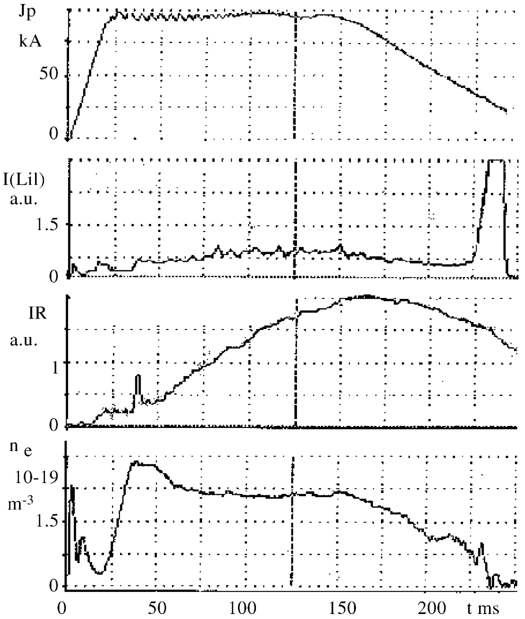


Fig. 6. Wave forms of plasma current-$J_p(t)$, Li($I$) emission, IR($t$) emission and $n_e(t)$-electron density for quasi-state discharge in T-11M.

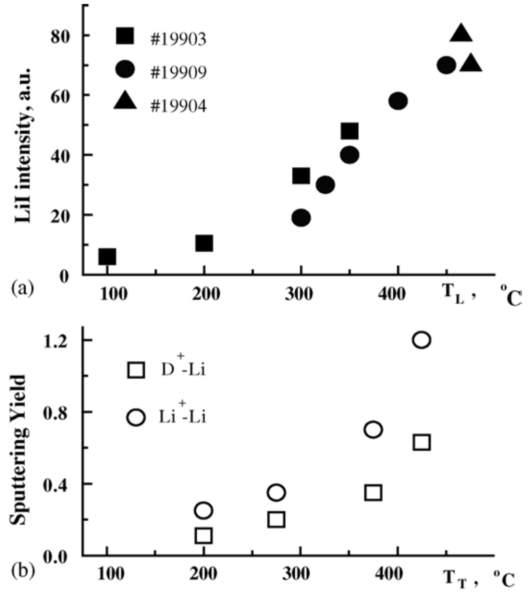


Fig. 7. (a) Lithium emission (LiI) as function of limiter temperature $T_L$ for three different shots (#) of T-11M in experiments with preliminary electron cleaning and (b) yield of Li sputtering by $D^+$ and $Li^+$, as function of Li target temperature $T_T$ [10].

ity of the limiter [7–9]. Fig. 7a shows this dependence in $T_L$ ranges 100–500 °C (from solid to liquid). The estimations of absolute lithium emission has shown that for $T_L < 500$ °C it remains close to expected for Li sputtering by $D^+$ and $Li^+$ ions with sputtering yield from 0.5 to 1. That correlates with the known data on Li-sputtering [10] by ion beams bombardment (Fig. 7b). For $T_L \geq 500$ °C Li-evaporation seems like the main channel of lithium emission. Note, that lithium emission was almost insensitive to energy of bombardment ions [7–9]. Obviously the best working temperatures of Li PFC should be 300–500 °C until Li flux is not very high but already the negative feed back condition between heat flux to PFC and plasma cooling by Li-injection exists.

Tokamak disruption resistance of Li CPS was tested in model experiments and in T-11M disruptions [8]. The major part of the plasma energy (∼97–99%) was absorbed and reradiated by no coronal radiation mechanism in thin (1 cm) shielding layer. The solid basis of CPS limiter (Mo, SS) had no damage after more than $2 \times 10^3$ of plasma shots with 5–10% disruptions. The main reason of lithium erosion during disruptions was splashing [8].

But Li-splashing did not have some dramatically consequences for the T-11M plasma performance. In contrary sometimes after intensive Ga-splashing in T-3M we needed to use the boronization procedure to suppress a high plasma radiation. That is argument pro Li and contra Ga, as PFC material.

Finally we have to note that the common feature of all experiments with Li as tokamak PFC not only in T-11M but in US-experiments too (TFTR, CDXU), was a very low hydrogen recycling.

## 4. Deuterium retention and release

The main reason of recycling decrease in the all Li experiments is the high growth of sorption of hydrogen species $D^+$ and $H^+$ on the wall. Moreover the helium sorption was found in T-11M experiments as well [8] but with a flabby desorption during 20–100 s after discharge. To avoid helium sorption it was sufficient to heat the T-11M chamber wall up to 50–100 °C. The Li-limiter heating after plasma experiments showed a start of deuterium desorption from lithium at temperatures higher than 320 °C [8]. As it was shown later by beam experiments [11] and in process of limiter cleaning in T-11M, the lithium heating up to 500 °C seems to be sufficient to remove all deuterium and probably tritium also. The difference of helium and deuterium sorption may be used for its separation in the future reactor.

## 5. Li and Ga use in PFC of tokamak-reactor

The ITER project development had difficulties in choice of materials for first wall and diverter plates. The Li capillary-pore system seems today a possible candidate for ITER PFC. The no coronal radiation [7] of sputtered and evaporated Li will play role of a reradiated blanket and help to smooth local thermal load of the first wall.

Unfortunately the cooling circuit of Li PFC seems to be a serious problem for Li-reactor due to Li and water incompatibility. The double circuit system with intermediate heat conductor (Fig. 3, down) may be suggested for its solution. Some version of such double bellows wall is presented in Fig. 8. Its main features are: a thin (1 cm) CPS layer (5), Li-channels (1), water cooling (2), the SS or V double bellows ≈2 mm (4), divided gap ≈0,3 mm, filled by He or Ga (3).

This divided gap plays role of heat conductor between two bellows. Internal bellows will contact with CPS and Li. The external one will contact with water. The simple calculation shows that if the Li temperature

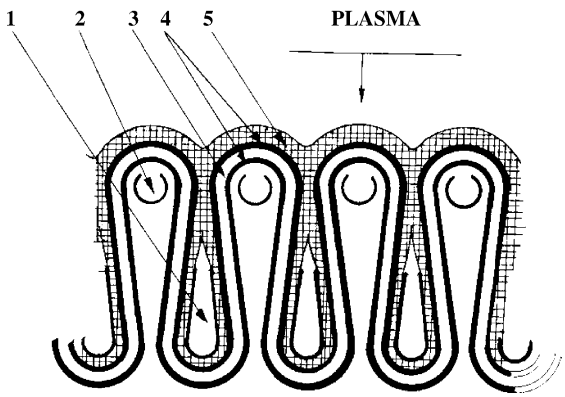


Fig. 8. Bellows version of Li CPS first wall with water cooling. (1) Li flow channels, (2) water channels, (3) gap with He or Ga filling, (4) steel or V double bellows, and (5) Li CPS.

will be 450 °C and water 200 °C, the passing heat flux in case of He filling should be equal to 0.4 MW/m$^2$. The heat flux can be increased to 2 MW/m$^2$, if the gap is filled with Ga. These fluxes are looking suitable for the reactor first wall, but not for ITER-like diverter. That means that smoothing of heat flux between divertor and wall may be useful for Li-tokamak concept.

The use of Ga as an intermediate heat conductor has two additional advantages:

- the low permeation of hydrogen isotopes in Ga can prevent tritium diffusion into the water circuit,
- the Ga ability to create the connection LiGa with $T_M$ = 740–760 °C [12] can be used for self-recovery of internal bellows (Li-circuit) in cases of its cracks or possible damages.

## 6. Conclusions

1. Ga – can be used as tokamak PFC. The LM-droplet limiter seems some version of reactor limiter.
2. Ga seems intermediate heat conductor between Li CPS and cooling system. Main Ga advantages are: the tritium screening and probability of Li first wall cracks self-recovery.
3. The plasma experiment on T-11M tokamak with Li CPS as tokamak PFC has shown:
   - No spontaneous bursts of lithium erosion under heat flux up to level 10 MW/m$^2$.
   - Total lithium erosion is close to the level of lithium sputtering by deuterium or lithium ions.
   - The no coronal lithium radiation protected the Li limiter from high power load.
   - The recovery temperature of hydrogen isotopes from Li is 320–500 °C, for helium 50–100 °C.

This difference may be used for separation of helium and hydrogen isotopes.

## References

[1] UWMAK-I, A Wisconsin Toroidal Fusion Reactor, UWFDM-68, University of Wisconsin, 1974.
[2] E.V. Murav'ev, Contact devices for divertor, Vopr. Atom Nauk. i Tekhn, Fusion (Russia) 2 (1980) 57–65.
[3] V.A. Evtikhin, L.G. Golubchikov, Divertor of fusion reactor, RF Patent 2,051,430 (February 7, 1995).

[4] V.A. Evtikhin, I.E. Lyublinski, A.V. Vertkov, V.I. Pistunovich, L.G. Golubchikov, V.M. Korzhavin, et al., Proceedings of the 16th Conference on Fusion Energy, Montreal, IAEA, Vienna, October 7–11, 1996, The liquid lithium fusion reactor, Fusion Energ. 3 (1997), 659–663.

[5] V.O. Vodyanuk, V.N. Dem'yanenko, A.F. Kolesnichenko, S.V. Mirnov, E.V. Murav'ev, Liquid metal limiter of tokamak, Fiz. Plas. (Russia) 14 (1988) 628–631.

[6] S.V. Mirnov, V.N. Dem'yanenko, E.V. Murav'ev, Liquid-metal tokamak divertors, J. Nucl. Mater. 196–198 (1992) 45–49.

[7] V.B. Lazarev, E.A. Azizov, S.V. Mirnov, A. Alekseev, V.A. Evtikhin, I.E. Lyblinski, A.V. Vertkov, Proceedings of the 26th EPS Conference on Contr. Fusion, ECA, Compatibility of the lithium capillary limiter with plasma in T-11, Plasma Phys. 231 (1999), 845–849.

[8] V.A. Evtikhin, I.E. Lyblinski, A.V. Vertkov, S.V. Lazarev, N.P. Petrova, S.M. Sotnikov, et al., Lithium divertor concept and result of supporting experiments, Plasma Phys. Contr. Fusion 44 (2002) 955–977.

[9] S.V. Mirnov, V.B. Lazarev, S.M. Sotnikov, V.A. Evtikhin, I.E. Lyblinski, A.V. Vertkov, Li-CPS limiter in tokamak T-11M, Fusion Eng. Des. 65 (2003) 455–465.

[10] J.P. Allain, D.N. Ruzic, M.R. Hendrics, Measurements and modeling of D, He and Li sputtering of liquid lithium, J. Nucl. Mater. 290–293 (2001) 180–184.

[11] Y. Furuyama, K. Ito, S. Dohi, A. Taniike, A. Kitamura, Characteristics of lithium thin films under deuterium ion implantation, J. Nucl. Mater. 313–316 (2003) 288–291.

[12] S.P. Yatsenko, Gallium Interaction with Metals, Nauka, Moscow, Russia, 1974, p. 86.