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Plasma-wall interactions and plasma behaviour in fusion devices with liquid lithium plasma facing components

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ABSTRACT

The application of lithium as a self-recovery and renewable material of plasma facing components (PFC) can be used to solve steady state fusion reactor PFC problems. This paper is a survey of liquid Li use in current tokamaks. Liquid Li as tokamak limiter material has been tested in T-11 M tokamak (TRINITI, RF), in FTU (Italy) and in CDX-U (USA). The idea of T-11 M and FTU liquid Li limiters is based on the Capillary–Pore System (CPS) concept. The main feature of CDX-U toroidal limiter was free liquid Li surface. The crucial issue of tokamak is impurity contamination. Lithium experiments in tokamaks discovered that poor lithium penetration into hot plasma core from its periphery (lithium screening) and the development close to plasma boundary lithium non-coronal irradiative blanket. Lithium screening can be physical ground of lithium 'emitter–collector' limiter concept with irradiated blanket and PFC prevention from a high local power load up to level of ITER parameters.

1. Introduction

1.1. Lithium as a new paradigm of fusion reactor PFC

The main demand of practical nuclear power engineering to magnetic fusion is ability of steady state neutron production. These neutrons can be used as for instance in the traditional DT-fusion reactor scheme with water boiling by 14 MeV neutrons, or in scheme of fuel breeding ([1] for example) or as transmutators of actinides–fission nuclear waste. For the last two purposes DT-tokamak reactor will be enough without burning and high neutron loads (more 2 MW/m^2), but a steady state volumetric 14 MeV-neutron source with neutron loads only equal to 0.3–0.5 MW/m^2. However, the current tokamaks cannot be used for steady state operation. Hydrogen isotopes retention and collection of dust, as a result of PFC erosion, should finally interrupt the plasma existence. The steady state tokamak TRIAM with full high Z (Mo + SS) PFC clearly demonstrated the long-term tokamak problems [2]. It terminated after every 5 h of operation. Obviously, the problem of dust accumulation should be solved in power DT-reactor. One of the solutions can be using of lithium as material for steady state tokamak PFC.

The lithium properties: a low melting point (180 °C), a high evaporation temperature (1343 °C, $P = 1$ atm.), low $Z = 3$, a large gap between the first (5.4 eV) and the second (76 eV) ionization potentials allows the application of lithium as a self-recovery and renewable PFC material.

Obviously, PFC temperature of tokamak- reactor should be more than 180 °C and lithium as PFC material should be liquid. Liquid lithium splashing and tritium retention are the traditional arguments against the liquid lithium use in tokamaks. However, several experiments which were performed in Russia (tokamak T-11 M [3]), USA [4] and Japan [5] showed that the trapped hydrogen isotopes can be removed from the lithium target by heating only up to 400–500 °C.

To prevent splashing the interruption of electrical current loops induced in liquid metal should be used. It can be droplets stream development [6], or electrical isolation of PFC liquid lithium surfaces. The most progressive method of splashing prevention was the proposal of lithium filled porous metals (Mo, SS or W) using the so-called capillary porous system (CPS, Evtikhin et al. [7–11]). This system was successfully tested in tokamaks T-11 M and in FTU (FEC1998–2006) as lithium limiters.

The successful experiment with electrical break (isolation) of liquid lithium surfaces used as tokamak limiter ('liquid lithium try limiter' [12,13]) was performed in CDX-U (2004).

These and early TFTR [14] (1996) experiments showed that lithium can be successfully used as tokamak PFC material ('lithium tokamaks'). The lithium coated tokamak vessel wall (lithiation) permits to obtain better plasma features (plasma confinement, purity, density limit) as compared with boronization. The optical spectral lines of high Z and low Z impurities practically disappear from plasma radiation after tokamak lithiation.

An important consequence of the tokamak and stellarator lithiation, is a suppression of hydrogen recycling. The hydrogen recycling coefficient R dropped after tokamak lithiation almost from 1 to 0.3.

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The crucial issue of 'lithium tokamaks' is lithium contamination of plasma center. The main surprise of all lithium experiments was the poor lithium penetration to the hot core from plasma periphery (lithium screening). The effective ion charge in plasma center-$Z_{eff}(0)$ which had been equal to 2 or higher (TFTR, T-11M, FTU, CDX-U, NSTX) dropped down to 1 after lithiaton. The mechanism of lithium screening is not fully clear.

The lithium screening effect can serve as a basis of the new paradigm of tokamak PFC material. It can be explained by a simple example of two lithium connected limiters (Fig. 1). One hotter lithium limiter is positioned in deep plasma SOL (scrape-of-layer) and works mainly as a lithium emitter. The second limiter is located in shadow of the first one and works mainly as a collector of the lithium diffused out SOL to chamber wall ('emitter–collector' model [15]). Lithium entering the plasma can produce close to the hot plasma boundary the non-coronal irradiated lithium blanket which will smooth the main plasma energy outflow by non-coronal lithium radiation at the all tokamak first wall and thus will protect it and a divertor from too high local power loads typical for tokamaks. Lithium collected by the second limiter can be turned again into emitter limiter and close the lithium loop in plasma boundary region.

In principle the 'helium ash' can be extracted from the plasma boundary and pumped out of the reactor chamber by the use of different retention properties of helium and hydrogen isotopes in lithium. It can in result a complete scheme of steady state 'lithium tokamak' with helium extraction.

The most serious technological problems, which historically stood in the way of active Li application in tokamak were:

1) liquid metal splashing under the JxB forces during MHD instabilities and disruptions,
2) problem of heat removal from lithium and prevention of its evaporation under heat load,
3) possible anomalous lithium erosion as a result of plasma–liquid lithium interaction,
4) problem of tritium removal from lithium.

Some solutions of these problems were found in current tokamak experiments with lithium limiters. Their short survey is presented in this paper. In Section 2 main conditions of liquid lithium limiter experiments in current tokamak are presented. The principal experimental features of plasma interaction with lithium are described in Section 3, one idea of liquid lithium use in ITER is proposed in Section 4 and main conclusions are summarized in Section 5.

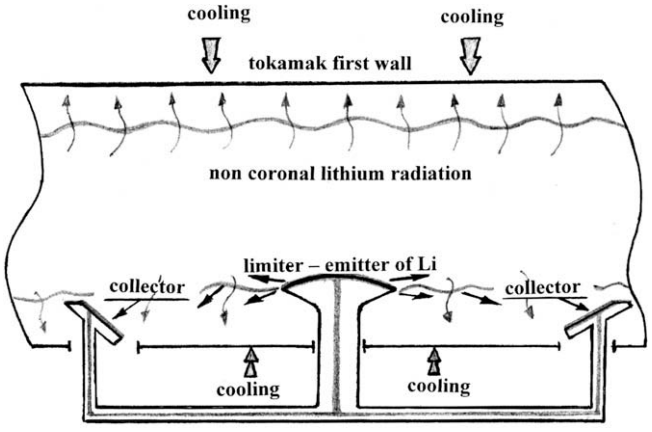

Fig. 1. [24]. The principal scheme of steady state tokamak with lithium emitter–collector limiter.

Table 1
Main parameters of tokamaks with liquid lithium PFC.

TOKAMAKS	T-11 M	CDXU	FTU
R/a [cm/cm]$k = b/a$	70/20, $k = 1$	34/22, $k = 1.6$	93/30, $k = 1$
B_T (T)	1	0.2	6-8
J_p (kA)	100	<80	500–800
Δt (ms)	250	<25	1500
$\langle n_e \rangle$ (10^{19} m^{-3})	4	1	5–20
T_e (eV)	400	100	2000
Kind of Li limiter	CPS	Free Li surface	CPS

2. Experimental basis of liquid lithium experiments in tokamaks

Three tokamaks with liquid lithium PFC (limiters) operated successfully during the last 10 years. These were: T-11M (TRINITI, Russia), CDX-U (PPPL, USA) and FTU (ENEA, Italy). Their main parameters are listed in Table 1. The idea of T-11M and FTU liquid Li limiters was based on the Capillary–Pore System (CPS) concept. The scheme of T-11M experiment with lithium and graphite rail-limiters is presented in Fig. 2. FTU scheme with lithium CPS limiter and Mo bumper limiter is presented in Fig. 3. Fig. 4 shows a three-section FTU limiter after an exposure in plasma. The idea of CDX-U limiter (Fig. 5) was based on the free surface (lithium mirror) concept. To form fully toroidal liquid lithium limiter (broken electrically into two parts) the liquid lithium mirror was contained in the stainless steel tray (10 cm wide and 0.64 cm deep) mounted with resistive heater (Fig. 6). Unfortunately, as we can see from Table 1, CDX-U experiments were performed in non typical short

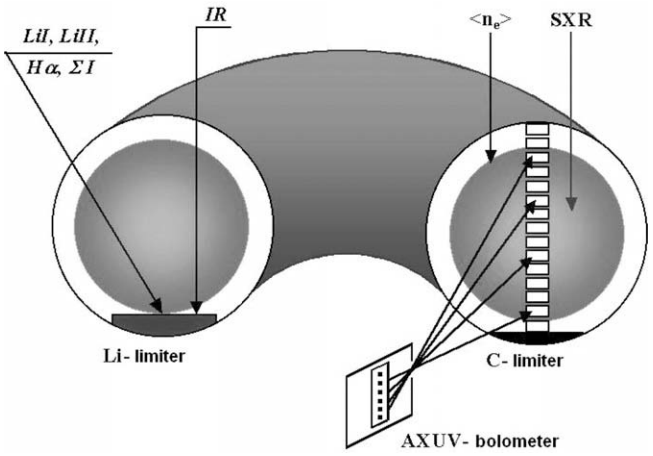

Fig. 2. Geometry of Li experiment in T-11 M.

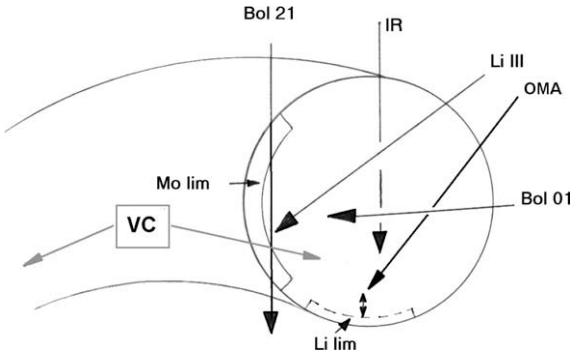

Fig. 3. Scheme of FTU experiment.

Fig. 4. View of FTU limiters.

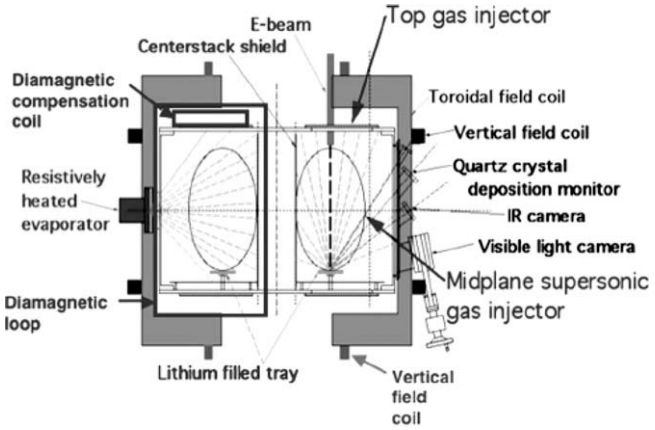

Fig. 5. CDX-U scheme [26].

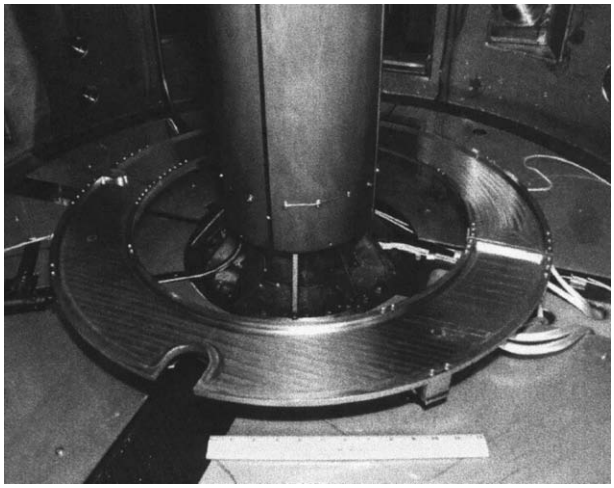

Fig. 6. View of CDX-U 'try limiter'.

(20 ms) tokamak regimes yet. They demonstrated all main features of 'lithium tokamaks', but they do not permit to make any predictions of steady state behaviour of liquid lithium limiters mirror type. On the contrary, the experiments with CPS limiters (T-11M, FTU) with up to 1 s pulse duration permit to have quasi steady state tokamak discharges and to make extrapolations up to a steady state. Our review focuses on the lithium CPS concept.

2.1. Lithium CPS properties

The proposal to use the surface tension forces in capillary channels for suppression of lithium splashing was made in [7,8]. The capillary channels from Mo, stainless steel (SS), V or W were manufactured as pressed wire grids. Two photos of typical CPS from 100 μm Mo-grids after plasma exposition with (A) and without (B) Li filling are presented in Fig. 7. As is seen, the thin lithium film (~10 μm) coats the solid structure. A self-recovery of the liquid Li surface by the capillary forces (Li-'candlewick' PFC) is an intrinsic property of such structures.

As it was shown [5], the penetration depth of 600 eV hydrogen in liquid Li is not more than 1 μm. The experiments with steady state electron beam [11] showed that the maximum power load of such CPS in lithium recovery regime can be as high as 20 MW/m^2. Thus the solid basis of CPS has a thick shielding from the plasma bombardment during steady state tokamak operation with typical tokamak power loads lower than 10 MW/m^2.

2.2. Li CPS cooling

The heat removal is a serious problem for all tokamak limiters and divertors. Relative low lithium thermal conductivity complicated the lithium use as tokamak limiter material. The idea of thin ($\delta \approx 1$–2 mm) lithium CPS coated layer on the cooled backing [16] seems to be a solution of CPS heat removal problem in a short pulse (~1 s) experiment. Thin CPS was successfully tested on T-11M and FTU. Fig. 8 shows the typical scheme of such limiter. The ends of thin CPS had a connection with Li reservoir so that it works as a steady state Li 'candlewick' (Fig. 8). The role of the cooled base (backing) was played (Fig. 8) by a thick (3–7 mm) Mo-tube (heat accumulator). An auxiliary cooling of Mo-tube should be used in a steady state experiment (>3 s) for this purpose. Water can be chosen as a coolant in small and middle scale experiments.

Unfortunately, the water cooling circuit of Li PFC seems to be a serious problem for large fusion devices due to Li and water incompatibility. A double circuit system with an intermediate heat conductor (Fig. 9) may be suggested to solve this problem [17].

As a prospective method of Li PFC cooling in DEMO as it seems today is the use of low temperature eutectics, for example, Pb + Mg or Pb + K with melting temperature 245–275 °C. To solve the problem of liquid metal motion in high magnetic field the long-term electrical insulation should be developed. The best coolant of Li CPS is liquid Li.

2.3. Li CPS resistance to disruptions

Resistance of Li CPS to disruption was tested in the special simulators [11] as well as in tokamak T-11M disruptions. In plasma gun experiments the disruption effects were simulated by a plasma accelerator with energy load $Q = 4$–5 MJ/m^2 and pulse duration $\tau = 0.2$–0.5 ms. It was shown that a dense plasma layer of 10–15 mm thickness with $n_e = 10^{23}$ m^{-3} is formed in front of the CPS target. The major part of the plasma energy (~97–99%) was absorbed by this layer and then radiated mainly by UV. This layer plays the role of a shielding layer [11]. This result was confirmed qualitatively in T-11M experiment. The solid basis of CPS limiter had no damages after more than 2×10^3 of plasma shots with 5–10% of disruptions. The intensity of iron spectral lines in Li limiter vicinity was almost zero during disruption. A relatively small amount of lithium in the simulator experiment was evaporated from the target during test pulses. The main channel of lithium loss was splashing.

The universal method of the suppression of liquid lithium splashing is a decrease of CPS cells. Fig. 10 shows dependence of capillary pressure limit on the CPS cells dimensions and lithium

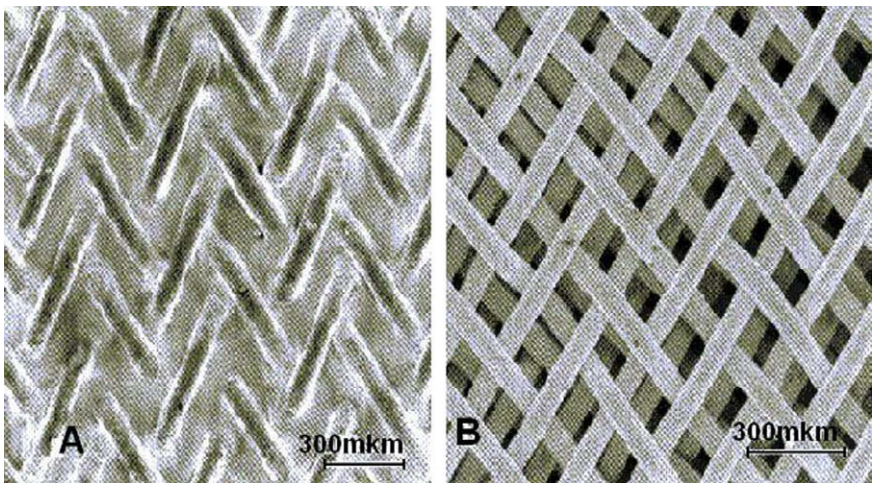

Fig. 7. View of the 100 m (mkm) CPS with (A) and without (B) Li filling (top).

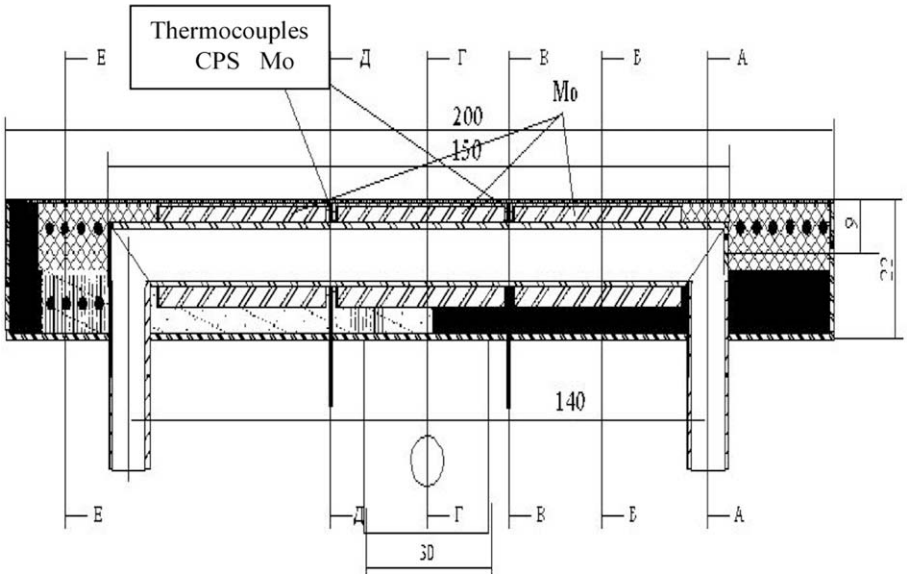

Fig. 8. Cut view of typical T-11 M limiter with Mo heat accumulator (3–7 mm), thin (<1 mm) Li CPS layer, lithium reservoir (black region) and preheater (black points).

temperature [11]. The exceeding of this limit is the main reason of lithium splashing during a disruption. Such pressure exceeding can happen during liquid metal shocks as a result of plasma MHD instabilities and electrical current J excitation in lithium limiters. The $J \times B$ forces can be the cause of such pressure shocks. The universal method of their suppression is a break of the electrical circuits. For example, the electrical connection of half parts of CDX-U limiter try was the reason of high lithium splashing in early CDX-U experiments. A similar situation with disruption hallo currents happened in the first T-11M experiments too. At the initial stage of FTU and T-10 experiments the main cause of liquid lithium splashing were the mechanical oscillations of CPS elements during tokamak operation. This kind of splashing was suppressed by hard mechanical fastening of the limiter.

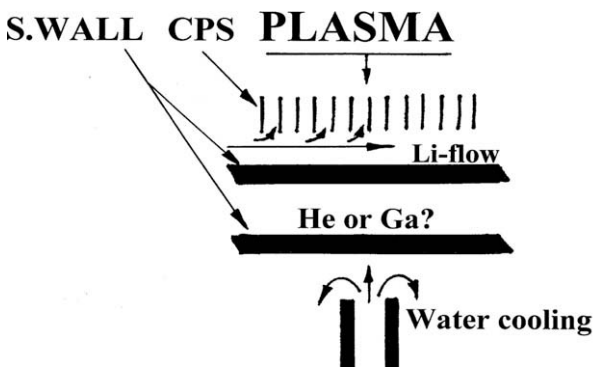

Fig. 9. The principal scheme of double water cooling circuit of PFC with lithium CPS.

Fortunately, the lithium splashing has not drastic consequences for tokamak plasma performance, as compared, for example, with liquid Ga splashing [6].

2.4. Lithium radiation

Lithium radiation plays a very important role in the energy balance of a liquid lithium limiter in steady state and disruptive regimes. Fig. 11 [18] shows the total radiated power per one Li atom and per one electron, which was calculated as function of electron temperatures ($1 \div 1000$ eV) and non-stationary parameter $n_e\tau$ (n_e-electron density in cm^{-3}, τ is Li ions confinement time in plasma column). The calculations were carried out in a coronal

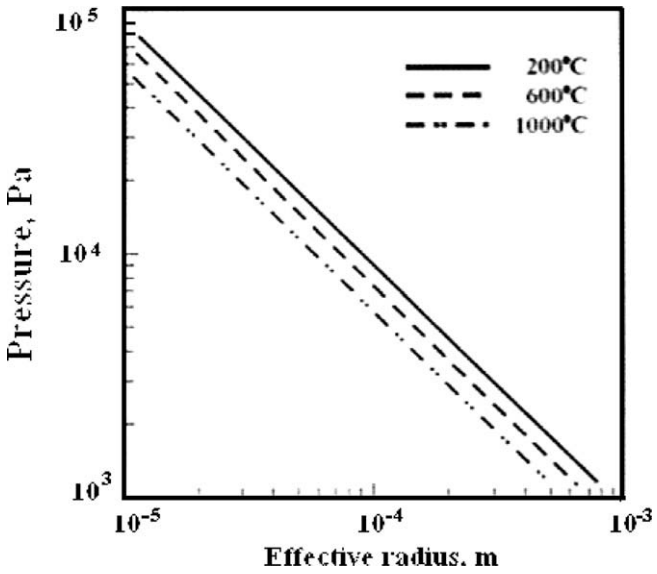

Fig. 10. The capillary pressure as function of CPS cells dimensions and lithium temperature [11].

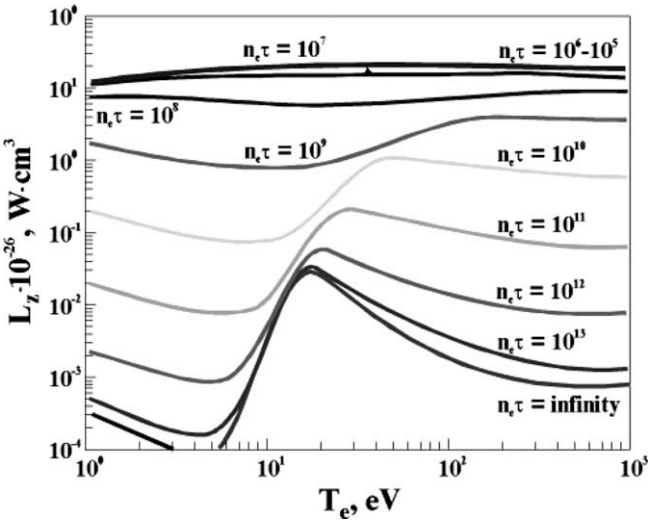

Fig. 11. The Li radiation power per one atom and one electron in coronal equilibrium ($n_e\tau = \infty$) and non-equilibrium regimes.

approximation [18]. Index 'inf' corresponds to coronal equilibrium ($\tau \approx \infty$). In the electron temperatures range of 30–300 eV the level of non-equilibrium lithium radiation can exceed the coronal limit by 2–3 orders.

For estimations of SOL cooling effect by lithium ionization and radiation we can also use the so-called 'energy cost of atom ionization'–a total electron energy losses during transition of one neutral atom to coronal ionization equilibrium. Fig. 12 shows the 'energy costs' of Li, Be and C ions as the function of electron temperature [15]. We can see that in the range of T_e = 13–30 eV lithium is a more effective radiator in comparison with beryllium.

3. The liquid lithium compatibility with tokamak plasma

The first test of liquid lithium compatibility with tokamak plasma was performed in tokamak T-11M experiments [9–11,16–24] and then in CDX-U [12,13,25,26] and FTU [27–29]. The plasma condition in T-11M limiter SOL was similar to one in of tokamak reactor (T_e = 20–30 eV, $n_e \approx 10^{19}$ m^{-3}). The first stage of the test

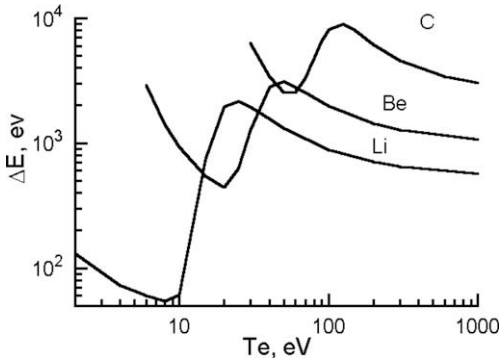

Fig. 12. The 'ion energy cost' as function of electron temperature for Li, Be, C [15].

program (1998–2001) was the demonstration of capillary forces ability to confine the liquid lithium by porous structure under the effect of $J \times B$ forces during tokamak regular and disruption regimes. The next step of the lithium T-11M program (2002–2005) was the CPS cooling by new limiter development with a thin lithium CPS coating and Mo heat accumulator, which permitted to decrease the lithium temperature and to achieve the thermal quasi steady state limiter mode [16,18–23]. The most part of power flux from hot plasma to the wall was smoothed in such tokamak regime by non-coronal lithium radiation.

3.1. T-11M and FTU experiments

Fig. 13 [15] shows the physical scheme of a limiter test experiment in T-11M and in FTU. The cooled horizontal lithium rod (rail limiter) coated by Li CPS touches the plasma column (T-11M) or plasma SOL in the shadow of main Mo-limiter (FTU).

The plasma contact area of the limiter (hot spot) is the main source of the Li atom influx (Li emitter) into plasma. Sputtered and evaporated Li atoms are ionised and excited by electron impact and penetrate as ions (Li^+, Li^{++}, Li^{+++}) into SOL and into hot plasma column. A part of the outward ion flux can go back to the cold ends of Li rod and is collected there (Li *collector*). The capillary forces give this part of lithium back to the hot spot and close the loop of Li recycling. In T-11M test experiment two methods of the Li limiter exposure were used: 'cold'-without preheating with initial $T_{0lim} < T_{melt}$ = 200 °C and 'hot'–with preheating to the melting temperature and above it ($T_{0lim} > 200$ °C). In the first case the lithium flux along the limiter surface from the cold ends to the hot spot was suppressed, while in the second case it could take place.

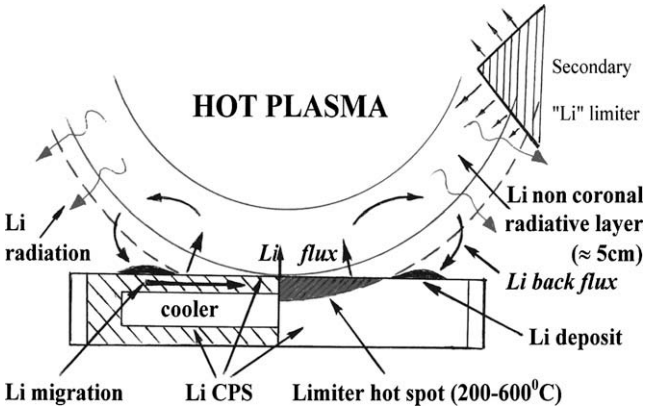

Fig. 13. A principal physical scheme of lithium limiter interaction with tokamak plasma.

Fig. 14. View of rail CPS Li limiter after 'cold' plasma exposure.

Fig. 14 shows a limiter picture after a 'cold' exposure. One can see 'hot spot' with small gray spots of solid basis (SS) and thick lithium deposits in the cold ends area. On the contrary, after 'hot' exposure (Fig. 15) the limiter surface seems homogeneous. It can be the evidence of the efficiency of the longitudinal transport of the liquid lithium from the end areas of CPS to 'hot spot' due to capillary forces.

If the confinement time τ of lithium ions before their return to the limiter or to the vessel wall is lower or comparable with their transient time to coronal ionization balance, the intensity of total lithium radiation can surpass the coronal limit (Fig. 11) and can play the active role of SOL plasma coolant. In this model the main radiated power flux goes to the broad area of tokamak vessel surface (first wall) and decreases the main limiter (or divertor) heat load. The movable secondary limiters (C and Mo) in T-11M case and main Mo-limiter in FTU case (Fig. 13) should work as additional collectors of lithium ions and as secondary emitters of lithium atoms, which can multiply the lithium radiation due to the decrease of the lithium confinement time τ. In the simplest case of one Li limiter (T-11M) the vessel wall plays the role of lithium and hydrogen collector (first wall lithiation in-situ). Lithiation is used successfully in many tokamak experiments, but we should have to assume that in the steady state regime the thick lithium deposit on the vessel wall can be the source of non controlled lithium injection and the ensuing plasma instabilities. This problem can be solved with the help of Li CPS coating of the whole tokamak first wall [8], or by using the emitter–collector scheme of Li limiters with a lithium collector (Fig. 1 [24]). Li should circulate in this scheme between lithium emitter (limiter, divertor plate) and collector (secondary limiter, first wall) delivering the main SOL energy to the vessel wall by lithium non-coronal radiation. The key problems of such a concept are:

1. Li limiters (emitter, collector) steady state cooling,
2. intensity of lithium radiation,
3. intensity of lithium erosion,
4. lithium penetration to the plasma center.

These questions were investigated in T-11 M and FTU.

3.2. Lithium radiation in T-11M and FTU

No catastrophic events leading to a spontaneous lithium injection in MHD stable discharges within the main lithium temperature range (from 20 °C to 600 °C) were observed in T-11M and it was the most important result of the initial experiments [18]. (The Li-droplets during disruptions were suppressed by a special construction of limiter ends and by a decrease of hallo currents).

The T-11M operation with thin cooling CPS made possible the tokamak discharges with an almost constant limiter temperature and main plasma parameters during equal 50–70 ms ($\tau_E \approx 7$ ms). These discharges seem to be as quasi steady state. The measurements of radiation distribution across plasma column in such discharges (Fig. 16) showed that the main source of plasma radiation (up to 80 ± 10%) is localized in a rather thin surface layer (<5 cm) near the plasma boundary. The remaining central radiation corresponded to $n_{Li}/n_D \leqslant 2\%$ ($Z_{eff} \approx 1.1$). In C-limiter shots the high level of plasma radiation from the center (>50%, Fig. 11) and $Z_{eff}(0) \approx 1.2$–1.4 were measured.

In experiments with preheating Li limiter up to 300 °C the total plasma radiation flux to the wall increased to 100 kW, which accounts for more than 80% of P_{OH}. Simultaneously, the direct plasma heat flux to the limiter dropped to 5% of P_{OH}. That was the example of limiter screening by lithium radiation. The limiter hot spot surface (Fig. 5) was equal to 30 cm^2. That means the Li limiter spread energy flux like a heat tube with the equivalent thermal load – over 30 MW/m^2.

Likewise, radiation redistribution was observed in FTU shots with high level lithium injection [29]. The scheme of FTU experiment is shown in Fig. 3. The liquid lithium limiter (LLL) was positioned in the shadow of the main Mo bumper limiter. Visible light emission observed by Video Camera (VC, Fig. 3) was the indicator of lithium injection. The intensity of LiIII spectral line was the indicator of lithium contact with Mo bumper limiter. The bolometer signals B21 and B01 indicated hot plasma radiation. Without lithium both bolometric signals had similar amplitudes (Fig. 17 dotted line). That corresponded to the symmetrical shape of plasma radiation probably as a result of high Z impurities penetration from the wall and Mo-limiter to the plasma center. When lithium injection was increased (by the sinking of LLL into plasma SOL), the lithium content close to Mo-limiter drastically increased (Fig. 17, increase of LiIII intensity) and the total plasma radiation lost its initial symmetry. The main radiation source shifted, supposedly, to the lower part of the vacuum vessel, close to LLL. The pictures of visible radiation confirm this assumption (Figs. 18 and 19). The reason of this transformation can be the suppression of the high Z impurities production on Mo-limiter during its shielding by Li.

In both cases-in T-11M and FTU we see a visible transformation of plasma energy flux to Li–radiation and mitigation of limiter power loads.

3.3. The Lithium screening

The main surprise of all experiments in the 'lithium tokamaks' field was the poor lithium penetration to hot plasma core (lithium screening) and the decrease of $Z_{eff}(0)$ correctly measured by two methods – Spitzer electric conductivity and visible bremstrallung. In some tokamaks $Z_{eff}(0)$ dropped from 2–1.5 to 1 during lithiation (T-11M, FTU, for example [16,28]) and increased again shot per shot after its finish. The mechanism of lithium screening is not completely clear so far.

The reactor heat removal problem will be significantly simplified by radiation transfer the main energy flux to vessel wall (Fig. 1), if the lithium screening mechanism will be universal.

3.4. The liquid lithium erosion

The mechanism of liquid lithium erosion is the next key question of the lithium limiter concept. The main result of all experiments on lithium erosion by ion beams [29,30] or by tokamak

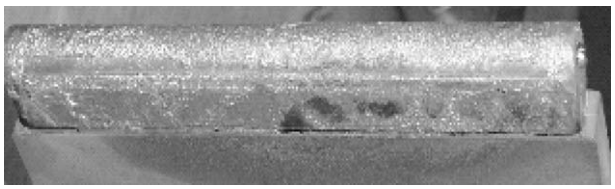

Fig. 15. View of rail CPS Li limiter after 'hot' plasma exposure.

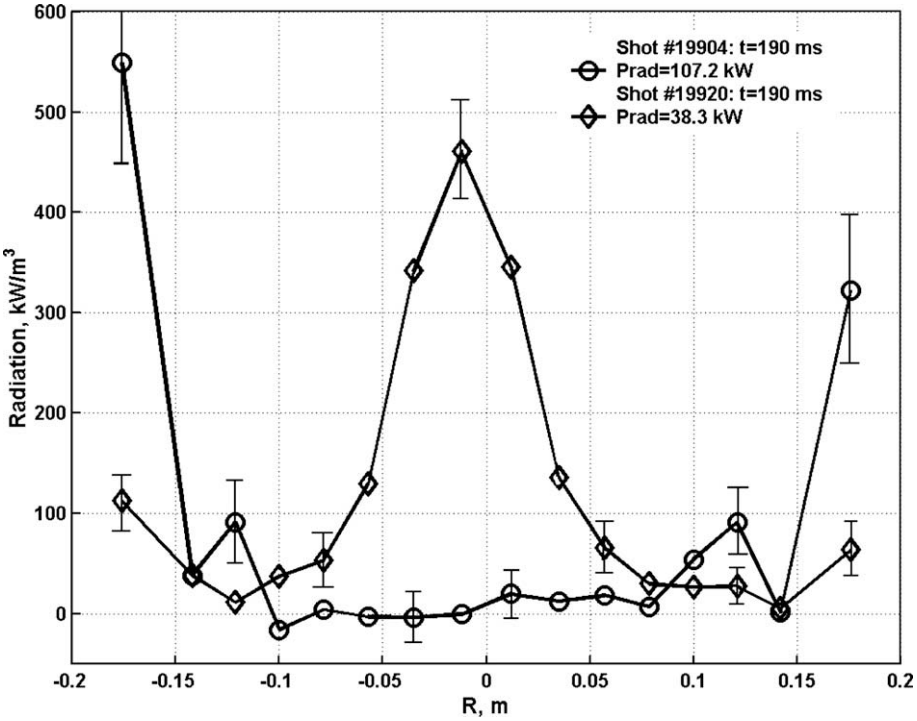

Fig. 16. The radiation profiles at the moment $t = 150$ ms for two similar C-limiter and Li limiter in T-11 M shots. Diamond points-C limiter, circle points -Li limiter.

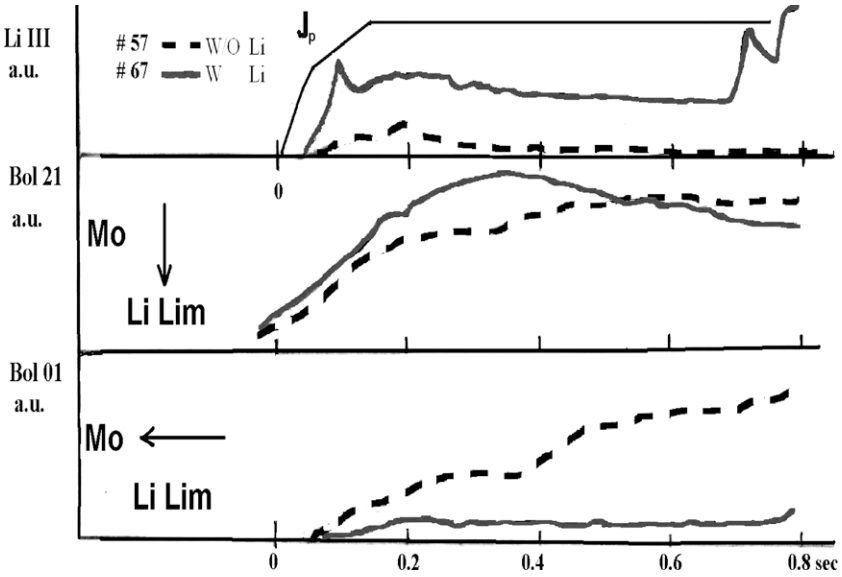

Fig. 17. FTU [29]. Wave forms of plasma current J_p, LiIII intensity and the bolometer signals B21 and B01 (Fig. 3).

plasma is a visible increase of lithium emission with a rise of liquid lithium temperature and weak dependence on energy of impinging ions [18,10,16,23].

Fig. 20 represents the results of lithium influx measurements (by lithium emission) in three different shots of T-11 M as a function of limiter surface temperature T_L [15] as well as lithium sputtering yield [30,31] under ion (D^+, Li^+) bombardment as a function of Li target temperature T_T. The behaviour of both parameters can be approximated by the function $\sim \exp\text{-}E_k/T$. The characteristic lithium sputtering energy E_S in temperature interval 200–450 °C was equal to 0.22 ± 0.02 eV and emission energy E_L was equal to 0.2 ± 0.02 eV. The practical equality of these values permits us to conclude that the erosion of liquid lithium in tokamak limiter condition has the same physical nature as liquid Li erosion under ion bombardment.

In limiter temperatures interval $T_L \approx 500$–700 °C E_L increased up to 0.6 eV. This increase can be understood, if one takes in to account that Li-evaporation energy E_{0VP} in the range of these temperatures is equal to 1.5–1.6 eV. Increased lithium evaporation should increase the total lithium emission as well as visible E_L.

One of the results of the beam sputtering experiment can be particularly important for tokamaks. The main part of sputtered Li (0.6–0.7) leaves the target as charged particles [30,31]. In the tokamak magnetic field they must come back to the limiter. This

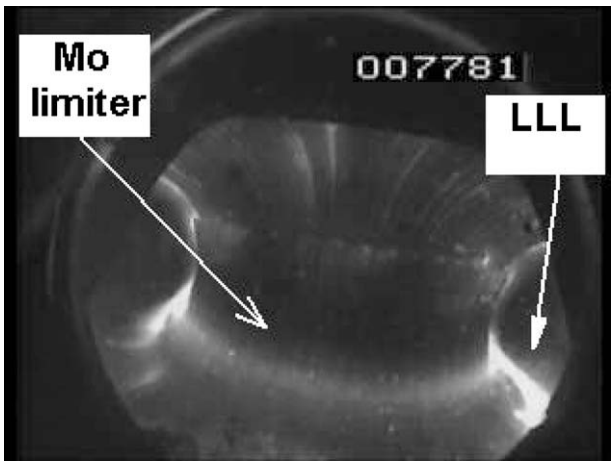

Fig. 18. FTU [29]. Visible light emission in regime with 'small' level of Li injection.

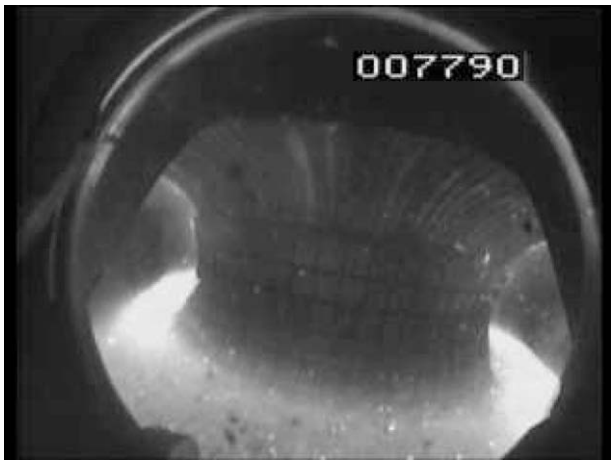

Fig. 19. FTU [29] Visible light emission in regime with 'high' level of Li injection.

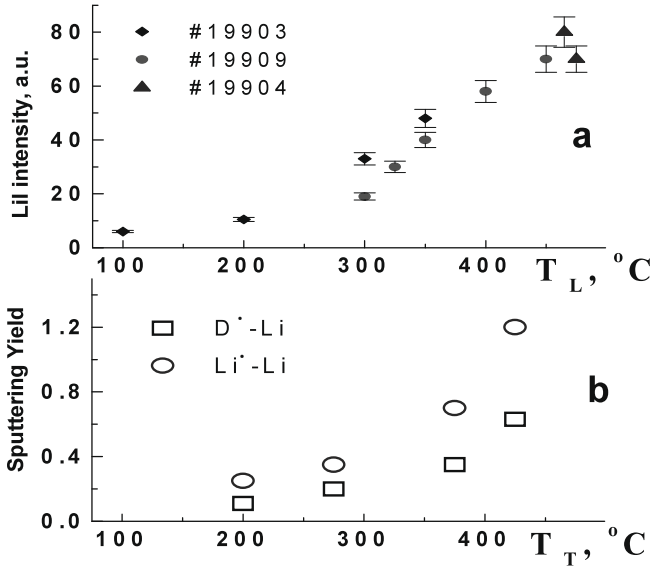

Fig. 20. (a)-the light emission of lithium (LiI) in limiter vicinity as function of its temperature T_L,(b) –the yield of Li sputtering by D^+ and Li^+, as function of Li target temperature T_T [30,31].

additional back flux of lithium to the limiter can improve the homogeneity of CPS lithium coating.

The estimation of absolute lithium influx based on experiments with electrical biasing of limiter [11] showed that for the limiter temperatures $T_0 < 500$ °C it remains in the range expected for the sputtering by D^+ and Li^+ ions with 0.5–1 sputtering yield.

Unfortunately, T-11M and FTU liquid lithium experiments were performed under short plasma exposure (<1.5 s). Some long-term liquid lithium erosion mechanisms (blistering, for example) may be supposed. They should be tested in the course of future steady state tokamak experiments.

3.5. Deuterium retention and removal

The common feature of all discharges with Li limiters and the first wall litiation (T-11M [18,11], TFTR [14], CDX-U [12,13], FTU [27,28], NSTX [25], T-10 [32]) is a very low hydrogen recycling and the resulting high gas puffing, which is needed to support the control plasma density. The total amount of deuterium, which was puffed during T-11M discharge, for example, exceeded the total amount of deuterium ions in plasma column up to 5×.

The helium retention was found in T-11M experiment as well [10,11] with a removal of He from the lithium covered vessel wall during 20–100 s after shots. In order to avoid helium retention it was enough to heat the T-11M vessel wall to 50–100 °C, as for deuterium even the possible highest wall temperature 250–300 °C proved insufficient. However, the Li limiter heating to 450 °C after plasma experiments showed the start of deuterium removal from lithium at temperatures higher than 320 °C. Fig. 21 shows the temperature dynamics of deuterium removal from liquid lithium for three known experiments [3–5] in semi-logarithmical scale. In temperature interval 300–500 °C we can approximate its behaviour by $\sim \exp\text{-}E_k/T$ again. The characteristic deuterium removal energy (E_{0R}) is equal to1.1–1.3 eV. It is close to energy of lithium evaporation – E_{0VP} = 1.5–1.6 eV, but it is very far from the characteristic energy of lithium hydrides decomposition ($\approx$ 2 eV, temperatures higher than 600 °C). Therefore, one can conclude that a considerable part of deuterium was not captured by lithium in the form of hydrides (deuterides), but it was just dissolved in lithium and probably removed during its evaporation. As is shown in Fig. 21 lithium heating up to 450–500 °C seems to be sufficient to remove all deuterium and perhaps tritium too. The difference between helium and deuterium removal temperatures can be used for separation of helium and hydrogen isotopes in DT-reactor.

4. The proposal of liquid lithium experiments in ITER

The tokamak limiter experiments showed a good quality of Li CPS as PFC. Calculations showed that the lithium radiating layer thickness $\sim 10^{-1}$ m, $n_e = 2 \times 10^{19}$ m^{-3}, $n_{Li} = 10^{19}$ m^{-3} and $\tau = 1$ ms would be enough for the transformation of the ITER total heat outflow (100 MW) to Li radiation. It can be supposed that such small τ for ITER plasma boundary can be the result of natural ELM activity, or of auxiliary magnetic stochastisation, or can be achieved by installation of special limiters.

Fig. 22 represents the idea of ITER lithium limiter. It can be a pumped mushroom limiter covered by a thin (<1 cm) Li CPS. The Li 'wick' must connect the 'hot hat' of a mushroom limiter (>550 °C) and its 'cold' (<350 °C) lower part like 'hot spot' and 'cold' limiter ends in Fig. 13. This means that the top part of the limiter must play the role of the lithium emitter while the lower part – the role of the collector of lithium and hydrogen isotopes. The vertical temperature gradient of the limiter can be controlled by the cooling of its top and lower parts. Such a limiter can work as trans-

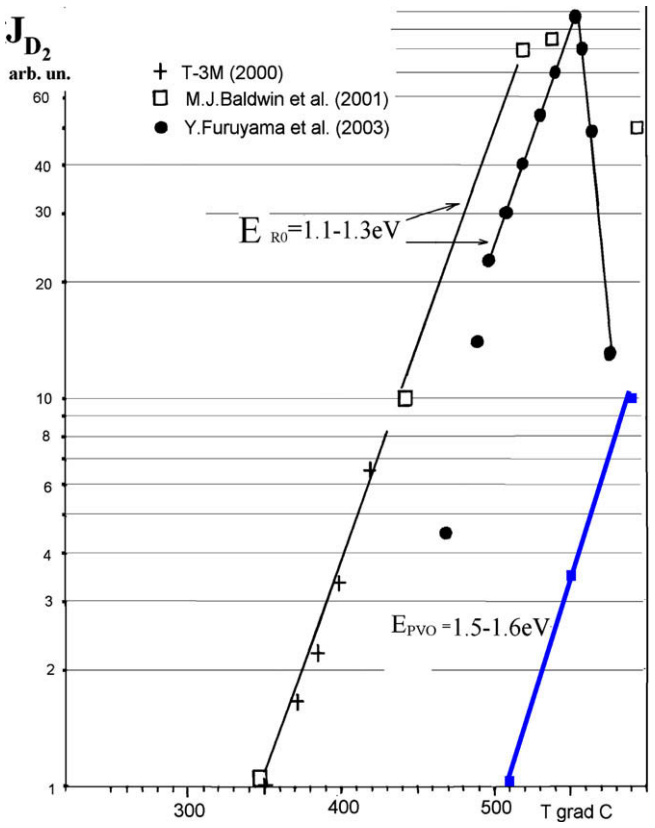

Fig. 21. The temperature dynamics of deuterium removal from liquid lithium for three experiments [3–5] in semi-logarithmical scale. Black squares – dynamics of Li evaporation.

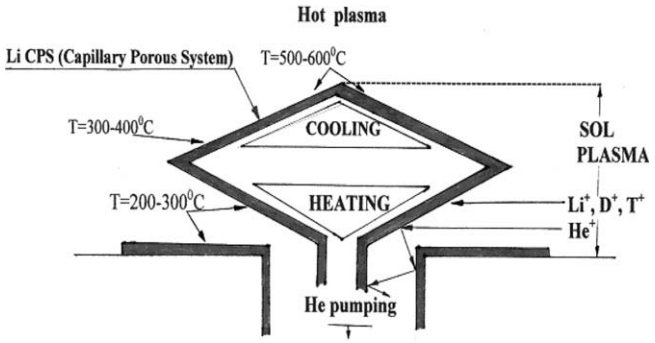

Fig. 22. Suggestion of CPS liquid lithium mushroom limiter with functions of lithium 'emitter–collector' and helium ash separation [15].

former of ITER SOL power flux to the radiation power: it decreases the local divertor power load and increases the broad power load on the tokamak vessel wall. The estimations show that the total power radiated by one such Li limiter can be more than 20 MW. The SOL width of such a limiter in approximation of Bohm diffusion must be equal to 10 cm at T_e = 20–30 eV.

If the temperature of the lower part of a mushroom limiter is higher than 100 °C, it must work as a reflector and compressor for He atoms (ash) – the product of fusion reaction. The pumping of limiter port allows a removal of a considerable amount of He ash from the reactor chamber. That can be an example of the use of absorption properties of lithium for segregation of ash and fuel in ITER and DEMO.

The main goals of the ITER test must be investigation of:

1. lithium penetration into the plasma center in reactor conditions,
2. efficiency of the heat flux transformation into radiation,
3. boundary plasma control.

5. Conclusions

1. The surface tension forces in CPS may be successfully used for solving the problem of liquid lithium splashing during MHD-events
2. The surface tension forces may ensure the PFC regeneration in steady state devices.
3. The experiments with hydrogen (deuterium) and helium plasmas with Li – CPS limiter can be summarized as follows:
   - no serious spontaneous lithium ejection events under a heat flux to the limiter up to 10–20 MW/m^2 and lithium temperature lower than 600 °C were observed;
   - a total lithium erosion of Li PFC during the interaction with tokamak SOL plasma is close to the level predicted by hydrogen and lithium ion sputtering;
   - the lithium non-coronal radiation protected the limiter from high power load during quasi steady state (1.5 s) regime and disruptions;
   - in the conventional tokamak regimes the Li ions circulating in limiter SOL allows the removing of as many as 0.8 P_{OH} power (with 2–3 W/cm^2) to the vessel wall by non-coronal lithium radiation;
   - the solid basis of CPS limiter had no damages after more than 2×10^3 plasma shots with 10% of disruptions;
   - the temperature of hydrogen isotopes recovery from Li after hydrogen plasma bombardment is 320–500 °C (for helium 50–100 °C). Therefore, at high PFC temperatures (400–500 °C) tritium capture can be minimized;
   - the separation of helium and hydrogen isotopes is possible in lithium circuit at lower PFC temperatures.

The lithium ion behaviour in 'lithium tokamaks' permits us to believe in the existence of a lithium screen mechanism. It should be tested in ITER-like tokamaks and in stellarators too.

4. For successful use of lithium PFC in future steady state tokamaks the following should be studided:
   - problem of lithium cooling in steady state mode;
   - the lithium self-sputtering and lithium sputtering by hydrogen isotopes during a long-term PFC exposure in tokamak plasma;
   - physical origin of lithium screening effect.
5. The current experiments let one hope that liquid lithium PFC can be used for steady state tokamak reactor like ITER and DEMO.

The ITER Li limiter experiment with a combined Li-emitter and collector which could decrease the local power load to divertor plate can be suggested today.

References

[1] P.-H. Rebut, Plasma Phys. Control. Fus., 48 (12B) (2006) B1.
[2] H. Zushi, M. Sakamoto, M. Ogava, et al., Nucl. Fus. 45 (2005) S142.
[3] V.A. Evtikhin et al., in: Eighteenth IAEA Conference on Fusion Energy Sorrento, Italy, 2000 Exp 4/21.
[4] M.J. Baldwin, R.P. Doerner, S.C. Luckhard, R.W. Conn, Nucl. Fus. 42 (2002) 1318.
[5] Y.J. Furuyama, J. Nucl. Mat. (313–316) (2003) 288.
[6] S.V. Mirnov, V.N. Demjanenko, E.V. Muravi'ev, J. Nucl. Mat. (196–198) (1992) 45.
[7] V.A. Evtikhin, L.G. Golubchikov, Divertor of fusion reactor, J. Nucl. Mat. (1996) 667. 1995 RF Patent 2051430, February 7.
[8] V.A. Evtikhin et al., in: Sixteenth International Conference on Fusion Energy, Montreal, October 7–11, IAEA, Vienna 3, 1997, p. 659.

[9] V.A.Evtikhin et al., in: Seventeent IAEA Conference on Fusion energy 1998, IAEA, Vienna, 1999, IAEA-CSP-1/P, 4, p. 1309.
[10] V.A. Evtikhin, I.E. Lyublinski, A.V.Vertkov, et al., 2001 SOFT-21 Madrid, 2001, p. A-37.
[11] V.A. Evtikhin et al., Plasma Phys. Control. Fus., 44 (2002) 955.
[12] R. Majeski, R. Kaita et al., in: Eighteenth IAEA Conferece on Fusion Energy, Sorrento, Italy, Book of abstracts, 2000, p. 102.
[13] R. Majeski, R. Kaita, et al., in: Twentieth IAEA Conference on Fusion Energy Vilamoura, Portugal, 2004, EX/P5-24.
[14] D.K. Mansfield et al., Phys. Plasmas 3 (1996) 1892.
[15] S.V. Mirnov, E.A. Azizov, V.A. Evtikhin, et al., Plasma Phys. Control. Fus., 48 (2006) 821.
[16] S.V. Mirnov et al., Fus. Eng. Des. 65 (2003) 455.
[17] S.V. Mirnov, V.A. Evtikhin, Fus. Sci. Technol. 47 (2005) 698.
[18] V.B. Lazarev V B et al., in: Twenty-sixth EPS Conference on Control Fusion & Plasma Physics, 1999, ECA 231, p. 845.
[19] V.B. Lazarev et al., Plasma Phys. Reports 28 (10) (2002) 802.
[20] V.B. Lazarev et al., in: Thirtieth EPS Conference on Contr. Fusion and Plasma Phys., St. Petersburg, 7–11 July, 2003, ECA 27A, p. 3.162.
[21] V.A. Evtikhin et al., Plasma Sci. Technol. 6 (3) (2004) 2292.
[22] V.B.Lazarev et al., in: Thiryt-first EPS Conference on Control Fusion and Plasma Physics, July 2004, London, p. 4.152.
[23] S.V.Mirnov, E.A. Azizov, V.A.Evtikhin, et al., Twentieth IAEA Conference on Fusion Energy, 2004, Vilamoura, Portugal, EX/P5-25.
[24] S.V.Mirnov,et al., in: Twenty-first IAEA Conference on Fusion Energy, 2006, Chengdu, China, EX/P4-17.
[25] J.Menard J.et al., in: Twenty-first IAEA Conference on Fusion Energy, 2006, Chengdu, China, OV/3-1.
[26] R.Majeski et al., in: Twenty-first IAEA Conference on Fusion Energy, 2006, Chengdu, China, EX/P4-23.
[27] V.Pericoli Rodolfini et al., in: Twenty-first IAEA Conference on Fusion Energy, 2006, Chengdu, China, OV/3-4.
[28] G. Mazzitelli et al., in: Twenty-first IAEA Conference on Fusion Energy, 2006, Chengdu, China, EX/P4-16.
[29] M.L.Apicella, G.Apruzzese, G.Mazzitelli, et al., in: Thirty-fifth EPS Conference on Control Fusion and Plasma Physics, 2008, Create, P4.004.
[30] J.P. Allain et al., Nucl. Fus. 42 (2002) 202.
[31] P.J. Allain et al., Nucl. Fus. 44 (2004) 655.
[32] V.A.Vershkov, S.V.Mirnov, V.A.Evtikhin, et al., in: Thirty-fourth EPS Conference on Control Fusion and Plasma Physics, 2007, Warsaw, P1.059.