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Experiments with lithium limiter on T-11M tokamak and applications of the lithium capillary-pore system in future fusion reactor devices

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Abstract

The paper is an overview of recent results of Li limiter testing in T-11M tokamak. The lithium limiter is based on the capillary-pore system (CPS) concept. The Li erosion process and deuterium (D_2) and helium (He) sorption by Li first wall were investigated. The ability of capillary forces to confine the liquid Li in the CPS limiter during disruption was demonstrated. The idea of combined lithium limiter with thin (0.6 mm) CPS coating as a solution of the heat removal problem was realized. As a result the quasi steady-state tokamak regime with duration up to 0.3 s and clean ($Z_{\mathrm{eff}} = 1$) deuterium plasma has been achieved. The temporal evolution of the lithium surface temperature during discharge was measured by a IR radiometer and then was recalculated to the surface power load. For the estimation of the Li limiter erosion the Li neutral and ions spectral line emission were observed. The increase in lithium erosion as a result of limiter heating was discovered. The radial distribution of plasma column radiation measurements showed up to 90% of the total radiation losses in a relatively thin (5 cm) boundary layer and only 10% in a plasma centre during discharges with high Li influx. Oscillations of Li emission and saw-tooth-like oscillations of the limiter surface temperature have been detected in discharge regimes with highest Li limiter temperature ($> 600\,^{\circ}$C). A version of Li CPS first wall of DEMO reactor and Li CPS limiter experiment in the International Thermonuclear Energy Reactor are suggested.

(Some figures in this article are in colour only in the electronic version)

1. Introduction

The use of lithium as a low Z, self-recovery and renewable material for tokamak plasma facing components (PFC) has a number of potential advantages in comparison with solid materials such as carbon (C), beryllium (Be) and tungsten (W). The idea of using flowing Li films as a divertor plate was proposed in the tokamak-reactor UWMAK-1 project [1]. Later TFTR experiments with lithium injection into the hot DT plasma [2] discovered an absence of visible lithium accumulation in the plasma centre. These results motivated the use of lithium as a PFC material for a solution to steady-state tokamak-reactor operation without a significant increase in plasma Z_{eff}.

Since in the process of the reactor operation the PFC temperature should be higher than the lithium melting point (180.54 °C), Li should be liquid. This can face special technological problems such as (1) liquid metal splashing under the J × B forces during MHD instabilities and disruptions, (2) anomalous ion sputtering as a result of plasma–liquid lithium interaction and (3) heat removal and PFC thermal stabilization as prevention of strong lithium evaporation.

An additional problem of the lithium tokamak should be tritium removal from PFC.

The suggestion to use surface tension forces for suppression of lithium splashing was based on the idea of compensation of ponder motive forces in liquid metal by tension forces in capillary channels [3, 4]. These capillary channels from Mo, stainless steel (SS), V or W manufactured as pressed wire grids were called 'capillary-pore systems' (CPS). Two micrograph photos of typical CPS from 100 μm Mo-grids with (A) and without (B) Li filling are presented in figure 1. Self-recovery of the liquid Li surface by capillary forces (Li-wick PFC) is an intrinsic property of such structures. Lithium rail limiters based on CPS have been tested in the T-11M tokamak (1998–2005, TRINITI) as PFC in plasma conditions similar to the scrape-off layer (SOL) of the tokamak-reactor (T_{e} = 10–30 eV, $n_{\mathrm{e}} = 10^{19}\mathrm{m}^{-3}$). The compatibility of a liquid lithium PFC with tokamak plasma was the main subject of these investigations [5–10]. The first stage of the programme (1998–2001) was the demonstration of the ability of the capillary forces to confine the liquid lithium in porous structure under the J × B forces during tokamak ordinary and disruption regimes. The next step in the lithium T-11M programme in (2002–2005) was the CPS cooling by new limiter development with a thin lithium CPS coating and heat accumulator, which permit us to achieve the thermal quasi steady-state limiter mode [9, 11–15]. The solution of this problem gives us the principal scheme of heat removal for the reactor PFC. This paper is an overview of the recent results of the Li-limiter experiment in T-11M.

In section 2 conditions of the T-11M limiter experiments are presented. The main experimental results are described in section 3, discussed in section 4, and summarized in section 5.

2. Conditions of lithium experiment in the T-11M tokamak

2.1. The tokamak as a reactor plasma facing components test bed

The plasma density and the electron temperature near the tokamak first wall are considered to be the main parameters dictating plasma wall interaction. Two reasons for tokamak limiter use as a test bed for a reactor PFC are, first, the similarity of the power load of the reactor PFC (10–20 MW m^{-2}) and the heat load of real tokamak limiters, and second, a strong electron temperature (T_{e}) dependence of the electron heat flux on the limiter, which is the main part of this load. As is known, the electron heat flux along the magnetic field should be proportional to T_{e}^{α} where $\alpha = 3/2$ or 7/2 in the case of low and high electron collisions. That means electron

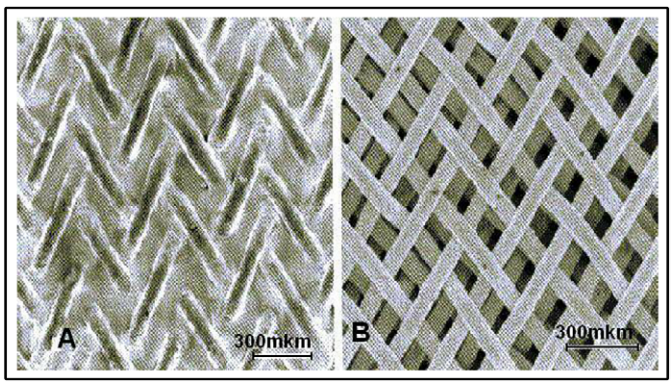

Figure 1. View of the 100 μm (mkm) CPS with (***A***) and without (***B***) Li filling (top).

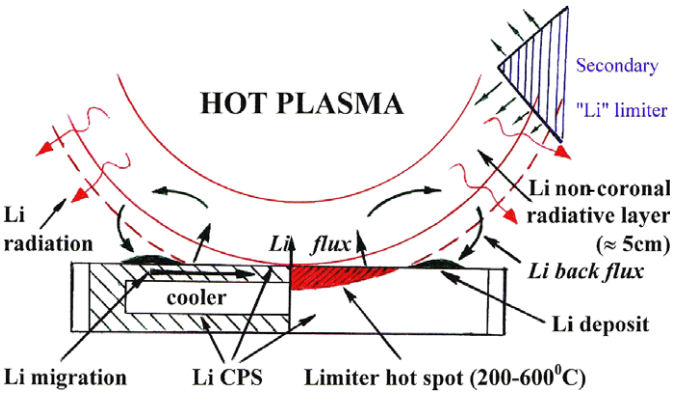

Figure 2. A principal scheme of lithium-limiter interaction with a tokamak plasma.

temperature T_e near limiters of the working tokamak and T_e near the future reactor PFC could be close.

Figure 2 shows the principal scheme of the test tokamak experiment with a Li rail limiter. The cooled horizontal lithium rod touches the plasma column. The plasma contact area of the limiter (hot spot) is the main source of the Li atom influx (Li emitter) into plasma. Sputtering and evaporated Li atoms are ionized and excited by electron impact and are diffused as ions (Li^+, Li^{++}, Li^{+++}) into the SOL and hot plasma column. Some parts of the outward ion flux can go back to the cold ends of the Li rod and collect there (Li collector). The capillary forces return this amount of lithium to the hot spot again. As a result the Li ions travelling in a SOL can recycle. If the travelling (or life) time τ of lithium ions is lower compared with their transient time to stationary ionized coronal balance, the total lithium radiation can significantly surpass the coronal limit and play a role as the main coolant of the plasma SOL. In this model the main radiated power flux goes to the broad area of the tokamak vessel surface (first wall) and spreads the limiter heat load. Secondary limiters (figure 2) should work as additional collectors of lithium atoms and as a secondary emitter, which can multiply the lithium non-coronal radiation due to a decrease in lithium confinement time τ. In the particular case of one Li-limiter (Li emitter) the vessel wall plays the role of the lithium and hydrogen flux collector (first wall lithization).

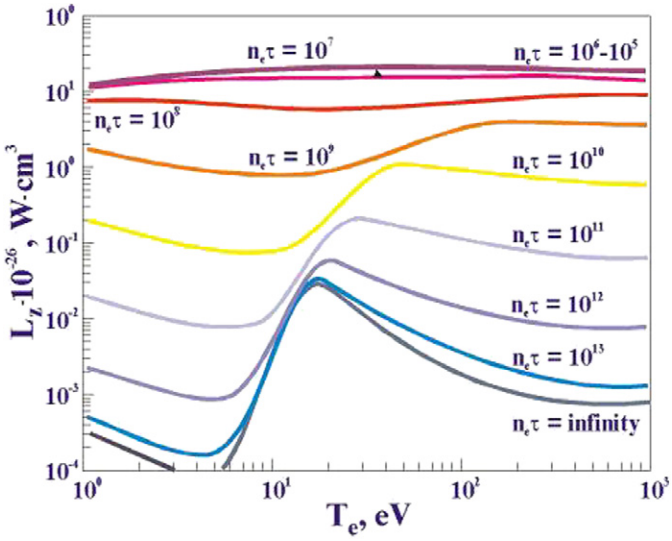

Figure 3. The Li non-equilibrium and coronal radiation ($n_e\tau$ = infinity) power per one atom and one electron as a function of electron temperature and 'non-stationary parameter $n_e\tau$'.

2.2. Lithium non-coronal radiation

Figure 3 [5] shows the total radiation power of lithium per one Li atom and per one electron, which was calculated for different electron temperatures (1–1000 eV) and non-stationary parameter $n_e\tau$ (n_e - electron density in cm^{-3}, calculations are carried out in coronal approximation; similarly for τ-Li ions confinement time in plasma column boundary [5]). The index 'inf' corresponds to the stationary coronal balance. In the electron temperature range of 30–300 eV the power of non-equilibrium (coronal) lithium radiation can exceed the coronal limit by 2–3 orders.

For estimations of the SOL cooling effect by lithium ionization and radiation we can use the so called 'energy cost of atom ionization'—a total electron energy loss during the transition of one neutral atom to coronal ionization balance. Figure 4 shows the 'energy costs' of Li, Be and C ions as a function of electron temperature. We can see that lithium is a more effective coolant of the plasma in the range of T_e = 13–30 eV in comparison with beryllium, for example.

2.3. Lithium limiter preparation

Figure 5 shows one of the typical Li CPS limiter designs with internal additional cooling and heating. The cooled molybdenum cylinder (Ø20 mm) was coated by thin ($\sim$ 0.6 mm) Li CPS (figures 5 and 6), which was touched by the plasma column. The hot spot area of the limiter was about 30 cm^2. The active surface of the CPS was coated during manufacturing by an additional SS foil, which was removed before experiments. The final cleaning of the CPS surface from oxygen and nitrogen films in tokamak was performed in several steps: discharge cleaning in He, heating in high vacuum and more effectively—heating up to 470 °C as an additional electrode of helium glow discharge. The last method gave us the cleanest CPS surface and minimized the impurity radiation losses from the plasma centre during experiments.

Two methods of Li limiter exposures during the experiments were used: 'cold'—without preliminary heating and 'hot'—with preliminary heating up to the melting temperature and

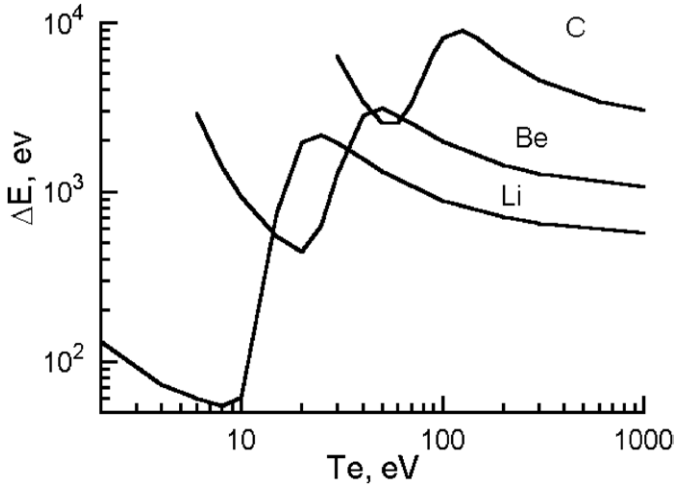

Figure 4. The 'ion energy cost' as a function of tokamak scrape-off layer electron temperature for Li, Be and C.

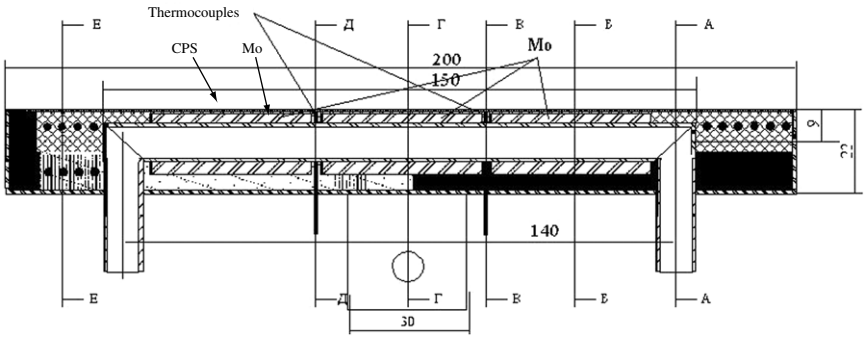

Figure 5. Cut view of one version of the lithium limiter with a thin CPS protection layer (→) and heat removal (backing) Mo tube (→). Black colour-liquid lithium reservoir.

Figure 6. View of rail CPS Li limiter after 'cold' exposures.

higher (>200 °C). In the first case the lithium back-flux along the limiter surface from the cold ends to the hot spot was suppressed, in the second, it was allowed. Figure 6 shows an image of the limiter after a series of 'cold' exposures. It is clearly visible: 'hot spot', grey spots of solid basis (SS) and lithium deposit in the cold area. On the other hand, after the 'hot' exposures (figure 7) the limiter surface seems inhomogeneous, and shows evidence of the efficiency of the longitudinal transport liquid lithium due to capillary forces. We worked in both regimes with visual control of the CPS surface and repaired its surface by additional heating, when it was needed.

Figure 7. View of rail CPS Li limiter after 'hot' exposures.

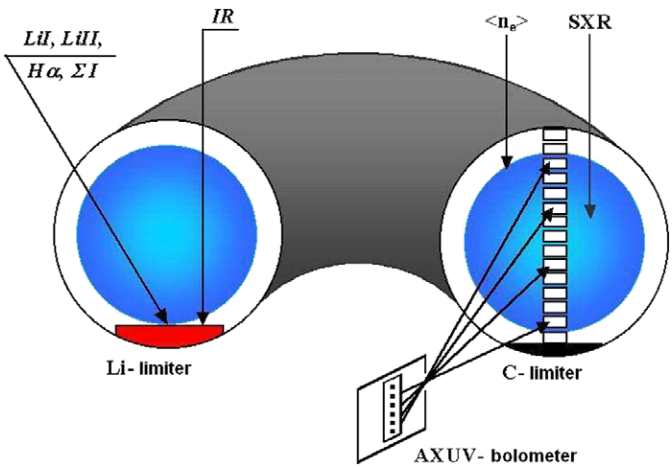

Figure 8. Geometry of Li experiment in T-11M. LiI, LiII and visible radiation ΣI measurements, SXR (soft x-ray) channels, IR (infrared)-observation and 16-channels of total radiation measurements.

2.4. T-11M experiment conditions

The main parameters of T-11M tokamaks are the following: $R = 0.7$ m, $a = 0.2$ m, $B_t = 1$ T, plasma current $J_p = 100$ kA, $n_e = (2\text{–}4) \cdot 10^{19} \text{m}^{-3}$ and $T_e(0) = 400$ eV [5]. First lithium erosion data and vessel sorption properties after lithization were obtained in the initial T-11M experiments (1998–2001) with the discharge duration being 0.1 s. The second experimental stage (2002–2005) was performed in T-11M with a discharge duration up to 0.3 s. The new limiter (figures 5–7) with a thin CPS protective layer was placed in the vacuum vessel on the radius of 19 cm (figure 8), thus limiting the plasma column aperture and determining the plasma current ($q(a) = 3\text{–}4$). The conventional (secondary) rail graphite limiter was placed in the opposite tokamak port for comparison with the lithium one. Two thermocouples were fitted in the lithium limiter close to its surface to measure the total energy absorbed by the limiter during discharge. The usual optical diagnostics of LiI, LiII and total visible light ΣI were applied to observe lithium flux into the plasma. A 16-channel AUXV-bolometer system [16] was set up and a special infrared radiometer was developed to measure the limiter surface temperature during discharge and to evaluate the deposited power values [10]. The typical thermal load on the Li-limiter surface (C-limiter was removed) was about 10 MW m^{-2} in normal discharges and achieved 100–200 MW m^{-2} during disruption. The limiter temperature addition during discharge was from 50 to 750 °C. Additional heaters incorporated in the Li-limiter structure allowed the initial limiter temperature to increase up to 400 °C. More then 4000 plasma shots with hydrogen, deuterium and helium were carried out on the T-11M tokamak and the plasma interaction processes with lithium CPS have been studied. The neutral lithium line (607.8 nm) emission in the limiter surface vicinity was chosen for the lithium influx estimations.

3. Experimental results

3.1. Lithium interaction with tokamak plasma

No catastrophic events leading to spontaneous lithium injection over the lithium temperature range (from 20 to 600 °C) have been observed in T-11M. This was the most important result of the first experimental campaign of Li CPS in T-11M. Lithium and graphite limiters worked roughly similarly. However, more detailed measurements (see below) showed that usually in the case of the Li-limiter $Z_{\mathrm{eff}}(0)$ was lower and T_{e} was higher as in the graphite limiter. Preheating of the Li-limiter gave rise to lithium injection into plasma detected by an increase in lithium lines intensity and total radiation in the limiter vicinity. It was known that lithium emission depends weakly on the energy of bombardment ions but reveals a clear visible increase in limiter temperature [5, 7, 9, 14].

The estimation of absolute lithium influx by electrical biasing of the limiter [7] has shown that for limiter temperatures $T_0 < 500$ °C it remains in the ranges expected for sputtering by D^+ and Li^+ ions with a sputtering yield from 0.5 to 1. This correlated with known data on Li-sputtering [17] measurements by ion beam bombardment. Note, however, that ion sputtering measurements are usually sensitive to the presence of impurity films on target surfaces. We made our erosion experiments under conditions of different kinds of limiter cleaning ('training' by 10–100 discharges, heating and glow discharge cleaning). In all cases the intensity of lithium emission was similar. In the best cleaning case (glow discharge cleaning) it was only a little higher. Figure 9 presents the results of lithium influx measurements in these cleaning experiments for three different shots of T-11M and sputtering data [17] as a function of limiter surface temperature T_{L} and Li target temperature T_{T}.

The similarity of both curves allows the assumption that the main mechanism of the Li-limiter erosion in the temperature range 200–500 °C has the same physical nature as liquid Li erosion during ion bombardment, which was observed in beam experiments with Li targets. For limiter temperatures higher than $T_{\mathrm{L}} = 500$ °C (figure 10) Li-evaporation appears to be the main mechanism of lithium emission, which has an approximately exponential increase over T_{L}.

It should be mentioned, however, that the main part (2/3 [18]) of the sputtered Li leaves the Li target as Li^+ ions and should turn back to the tokamak limiter by the electrical field of the Langmuir sheath. That means the real lithium losses from the tokamak limiter should be 2–3 times lower as was predicted by figure 9(*b*).

As the lithium temperature increased up to 650 °C bursts of lithium emission and sawtooth-like oscillations of the limiter surface temperature were detected (figure 11 [13]). The period of these oscillations was ∼20 ms and fell to 15–18 ms as the lithium temperature increased to ∼700 °C. The nature of these oscillations was not clear but it is possible that they were caused by the peripheral plasma area and cooling as a result of too high an influx of neutral lithium. The estimation of the total influx of lithium atoms from the limiter gives $\Gamma \sim 5 \cdot 10^{20}$ atoms s^{-1} and the mean flux, $\Gamma/S \sim 2 \cdot 10^{20}$ atoms $(\mathrm{s\,m}^2)^{-1}$, where S is area of limiter 'hot spot'. It is close to a threshold of 'ionization-condensation' instability with the formation of a MARFE-like region, which was suggested in [19]. But, it should be noted, the boundary instability was not finished by major disruption.

3.2. Experiments with thin Li–CPS limiter

High Li erosion (figures 9 and 10) at high temperatures (> 400 °C) demands solving a heat removal problem for Li CPS in order to restrict Li surface temperatures in the case of long time shots. The idea of a thin ($\delta = 1$–2 mm) CPS coated layer on a cooled backing [9] seems to be the solution of this CPS heat removal problem. This technique was tested on T-11M.

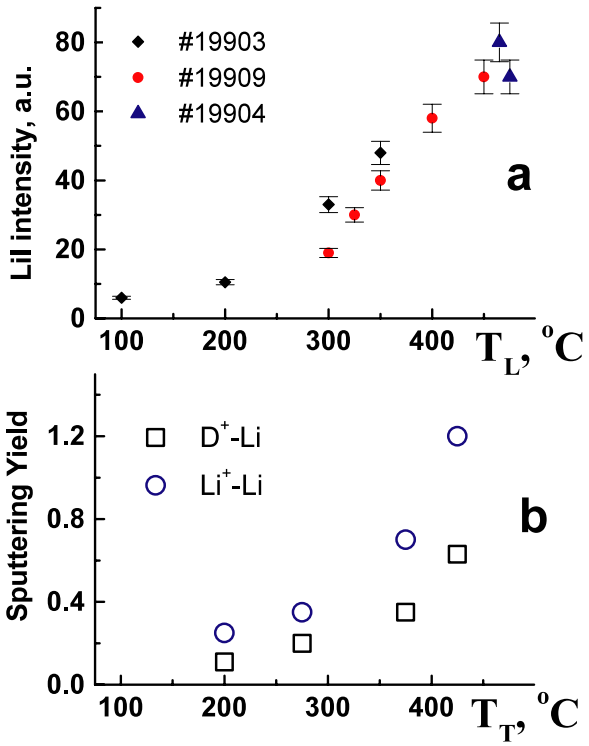

Figure 9. (*a*) the light emission of lithium (LiI) in limiter vicinity as a function of its temperature T_L for three shots (#) of T-11M with different initial limiter temperature (experiments with limiter cleaning by He glow discharge) and (*b*) yield of Li sputtering by D^+ and Li^+, as a function of Li target temperature T_T [17].

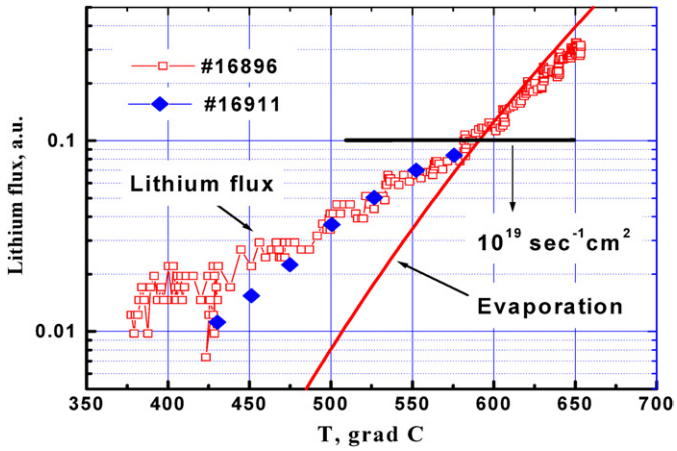

Figure 10. Measured and calculated fluxes of neutral lithium versus limiter surface temperature.

The ends of the CPS have a connection with the Li reservoir so that it works as a steady-state Li 'wick' (figure 5). Lithium can then flow to the limiter surface along the 'wick' due to capillary pressure and produce recovery of Li erosion damage on the limiter surface. The role of a cooled base (backing) is played by a thick Mo-tube (3 mm) (figure 5), which worked

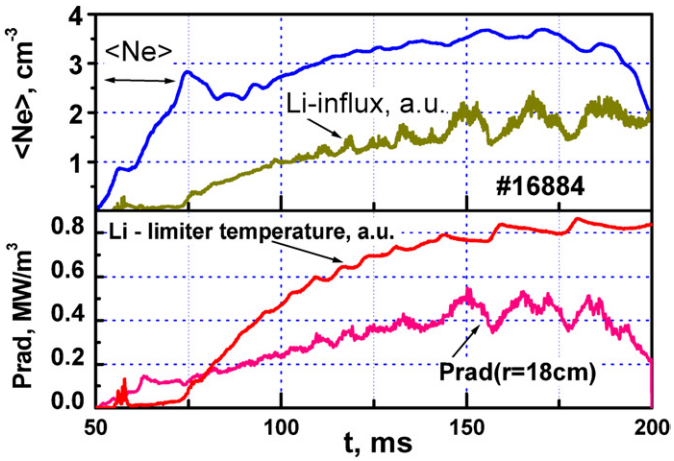

Figure 11. Oscillations of plasma density (<Ne>), Li emission, Li limiter surface temperature and total radiation (<Prad>), caused by the instability at the plasma edge in the discharge with a Li limiter temperature ∼600 °C.

as a heat accumulator during 200 ms of a T-11M typical shot. Obviously a critical element of this scheme is the heat contact between the CPS and the Mo-tube backing. To estimate the heat removal efficiency of the thin limiter we made several series of experiments in T-11M. Figure 12 shows two T-11M shots with different thin limiter designs: without the heat contact CPS–Mo tube and with good heat contact. In the first case the surface limiter temperature increases up to 650 °C and remains almost constant until the discharge is finished and also after its finish. In the second case the surface heated only up to 200 °C and slowly decreased during the shot. This decrease is a result of the increase in lithium radiation and the decrease in the limiter power load from 10–20 MW m^{-2} to 1–3 MW m^{-2} during the discharge. After the shot the surface temperature decreased faster according to the model of good heat contact between Li and Mo. The shots with thin cooling CPS, when the limiter temperature and main plasma parameters were almost constant in the time interval longer as 4–5 τ_E, seem to be an example of quasi steady-state Li-limiter discharge. One of such typical shot is presented in figure 13 [14], where the total plasma current $J_p(t)$, loop voltage $U_p(t)$, total radiation flux from plasma centre, mean electron density $N_e(t)$, limiter temperature $T_{lim}(t)$, parameter $Z_{eff}(0)/q(0)$ and electron temperature $T_e(0,t)$ are shown. The most important feature of the quasi steady-state phase (150–175 ms) is the constancy of the parameter $Z_{eff}(0)/q(0) = 1$, which was measured by comparison of the real plasma electrical conductivity ($\sigma(0) = 2\pi R J(0)/U_p = !0^5 B_t/U_p q(0)$) with Spitzer's prediction of pure deuterium plasma. If we suppose (from sawteeth activity), that $q(0)$ is approximately equal 1, $Z_{eff}(0)$ should be close to 1 as well. In the condition of high Li influx from the limiter into the boundary plasma we should assume the existence of lithium screening mechanism, which maintains a clean deuterium plasma in the centre.

The measurements of total radiation distribution across the plasma column (figure 14) shows the main source of total radiation losses (up to 80 ± 10%) during Li-limiter discharges localized in a rather thin plasma surface layer (≈ 5 cm) near the boundary. The estimations of lithium density from the remaining central radiation corresponded to $n_{Li}/n_D \leqslant 2\%$ ($Z_{eff} = 1.1$). By contrast in C-limiter shots the high level of radiation from the centre (>50%, figure 14) and $Z_{eff}(0) = 1.2$–1.4 were measured.

In experiments with preliminarily-heated thin Li limiter the total plasma radiation power-to the wall increased up to 100 kW, more than 80% of P_{OH}. The limiter hot spot surface

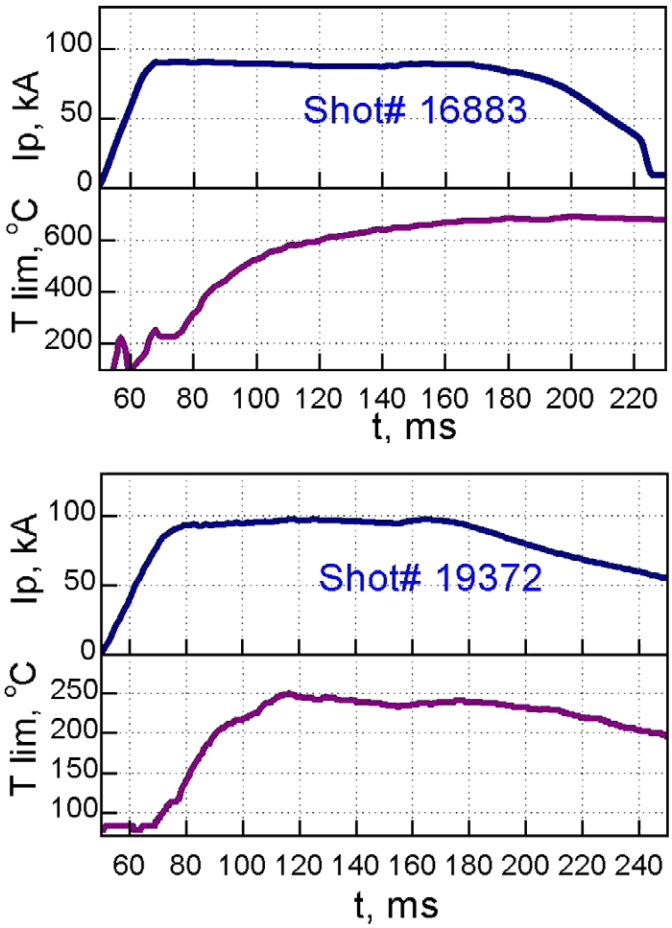

Figure 12. Two shots: without (#16883) and with (#19372) a good heat contact with backing.

(figure 6) was equal to 30 cm^2. This means that the Li-limiter worked in our experiments as a heat tube with the highest equivalent thermo load—more than 30 MW m^{-2}. Obviously, this was the result of non-coronal radiation of Li ions, which circulated in the limiter SOL.

3.3. Deuterium retention and removal

The common feature of all discharges with Li limiters and the first wall covered by lithium (T-11M [5–8], TFTR [2], CDX-U [20, 21]) was very low hydrogen recycling and, as a result, high gas puffing, which we needed to use for plasma density support in quasi steady-state regimes. The total deuterium amount, which was puffed during the T-11M discharge, enhanced approximately five times the total amount of deuterium ions in plasma column. In conditions of low hydrogen recycling (<0.3–0.5) this corresponds to a plasma confinement time of $\tau_p \approx 20 \pm 5$ ms.

The main reason for the decrease in recycling is the high probability of hydrogen and deuterium retention on the lithium-covered vessel wall. Moreover, helium retention was discovered in the T-11M experiment as well [7, 8] but with a slow removal from the

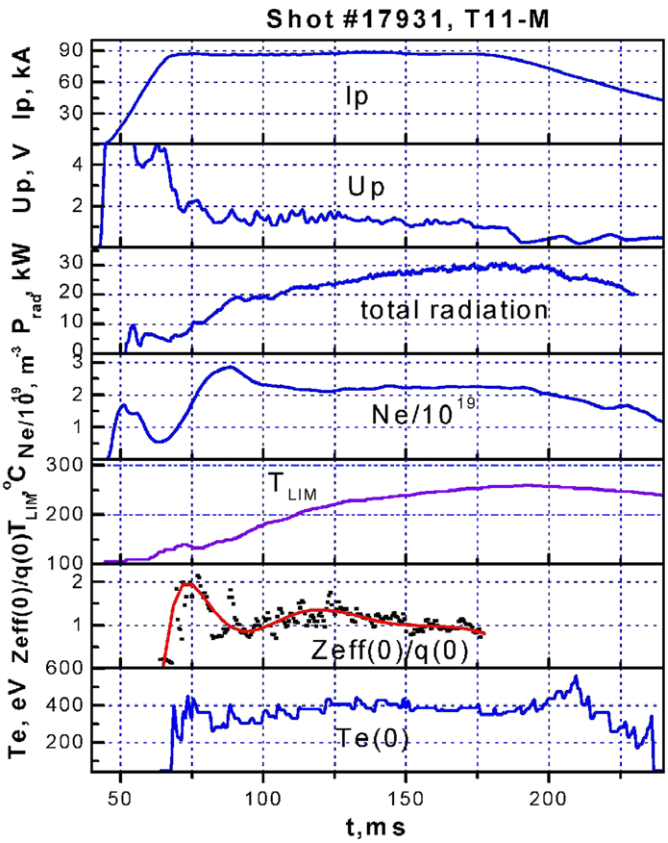

Figure 13. Dynamics of the main plasma parameters in the T-11M quasi steady-state regime with a thin CPS (0.6 mm) lithium limiter.

lithium-covered vessel wall (during 20–100 s after shots). Figure 15 shows the typical dynamics of helium removal after shot with the room temperature of the chamber wall. In order to avoid helium retention it was sufficient to heat the T-11M vessel wall to 50–100 °C. For deuterium even the highest attainable wall temperature of 250–300 °C turned out to be insufficient. However (figure 16 [7]), the Li-limiter heating up to 450 °C after plasma experiments showed the start of deuterium removal from lithium at temperatures higher than 320 °C. Lithium hydrides are supposed to be decomposed at temperatures higher than 600 °C. Therefore, one may conclude that a considerable part of the deuterium was not captured by lithium in the form of hydrides (deuterides). Perhaps it was just dissolved in lithium? As was shown later in USA [22] and Japan [23] direct heating to 400–500 °C of lithium after its bombardment by deuterium ions seems to be sufficient to remove all deuterium and, probably, tritium also. The difference between helium and deuterium removal temperatures may be used, in principle, for the separation of helium and hydrogen isotopes in a reactor.

3.4. Disruption resistance of lithium CPS

The resistance of Li CPS against disruption was tested in the special simulator experiment [8] and in T-11M. In the simulator experiment, using a plasma gun, the disruption effects were

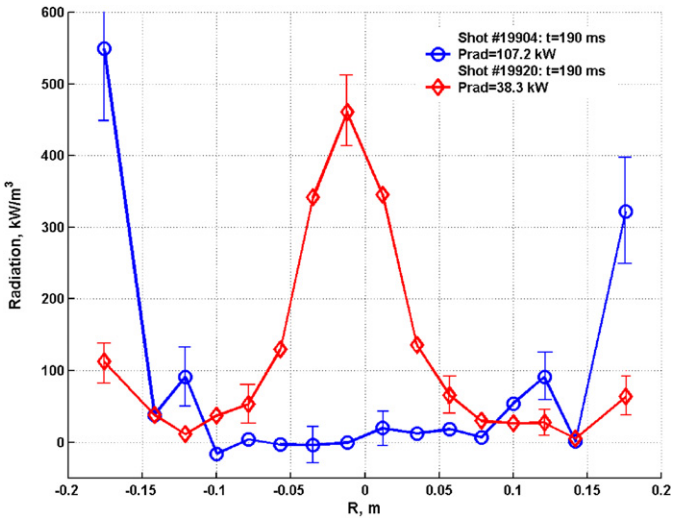

Figure 14. Radiative loss profiles at the moment $t = 150$ ms for two similar C-limiter and Li limiter shots. The average plasma density $<n_e> \sim 2.2{\cdot}10^{19}\mathrm{m}^{-3}$ for both cases. ⋄-C limiter, ○-Li limiter.

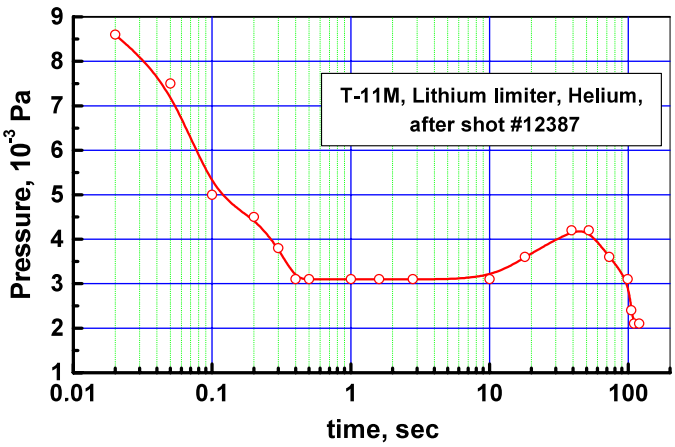

Figure 15. The behaviour of the gas pressure (He) in the tokamak vacuum vessel after the He discharge.

simulated with an energy load $Q = 4$–$5\,\mathrm{MJ\,m^{-2}}$ and pulse duration $\tau = 0.2$–0.5 ms. It was shown that a dense plasma layer, 10–15 mm in thickness with $n_e = 10^{23}\mathrm{m}^{-3}$, is formed in front of the target. The major part of the plasma energy (∼97–99%) was absorbed by this layer and then radiated mainly in the UV. This layer plays the role of a shielding layer [8]. This result has been confirmed qualitatively in the T-11M experiment: only 30–50 J of about 0.7 kJ of the total plasma energy loss has been found to reach the Li limiter during disruptions. The solid ground of the CPS limiter had no damages after more than $2 \cdot 10^3$ of plasma shots with 5–10% of disruptions. The intensity of the iron spectral lines in the Li-limiter vicinity was almost zero during disruption. A relatively small amount of lithium is evaporated from the target during test pulses. The main reason for lithium loss was splashing [8]. We have to

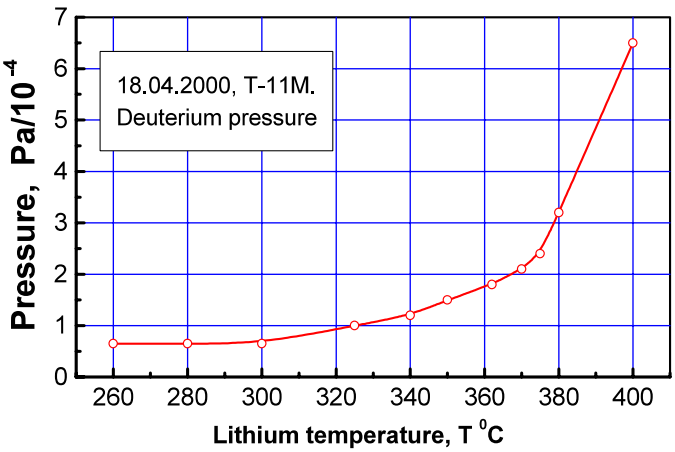

Figure 16. The increase in deuterium pressure in the tokamak vacuum vessel in the process of lithium limiter heating.

note that Li-splashing during disruption did not have dramatic consequences for the plasma performance in subsequent shots in T-11M and CDX-U [21].

4. Discussion of results and perspectives of Li CPS as reactor PFC

The T-11M limiter experiment showed good compatibility of liquid Li and SOL plasma, an absence of spontaneous Li bursts and plasma-Li instabilities except in the high Li temperature region (>650 °C). A very important feature of the T-11M experiments was the existence of the Li screening mechanism, which permitted us to have clean deuterium plasma in the centre. We do not know today the reason for such behaviour, but the Li screen effect was observed in TFTR experiments [2] and later in CDX-U [21] as well. Perhaps the physical origin of such screening is the large difference between the first (5.3 eV) and second (75 ev) ionization potentials of lithium. This means that the main part of the lithium close boundary should be only the single ionized and friction force between deuterium and lithium, which is the main reason for the neoclassical impurity accumulation effect, is absent in lithium. The reactor heat removal problem will be significantly simplified if this mechanism is universal. In this case lithium ions will circulate between the lithium emitter (limiter, divertor plate) and collector (secondary limiter, first wall) past the main SOL energy to the vessel wall by radiation. One can suppose that this mechanism for heat removal was achieved in quasi steady-state T-11M regimes with a high limiter temperature (>400 °C) and boundary layer radiation up to 80% of ohmic power P_{OH}. The power load of the limiter drops in these regimes from the conventional 20–30% P_{OH} to 3–5%.

However, the mean radiation flux on the wall was pretty low (2–3 $\times$ 10^{-2} MW m^{-2}) in T-11M discharges. In the ITER-like reactor it should be higher by 3–5 times. But the mean boundary T_e in high radiation regimes T-11M, as shown in the electric basing experiments [7], was too low −10–15 eV. If we turn to figure 4, we can see that increase in T_e to 20–30 eV should multiply several times the 'energy cost of lithium ion' and the corresponding radiation losses. On the other hand, we can increase the radiation level by decreasing the Li ions confinement τ (figure 3) in SOL or by a traditional injection of a high Z impurity like Ar. The calculation shows that for transformation of ITER total heat out flux (100 MW) to Li radiation

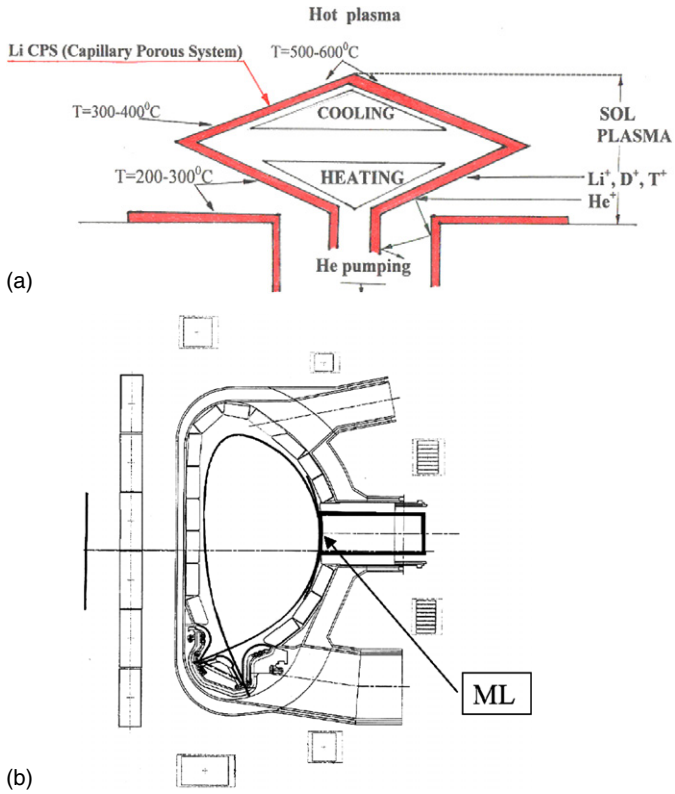

Figure 17. (*a*) A principal scheme of the ITER Li-mushroom limiter experiment. (*b*) Mushroom limiter (**ML**) position in the equatorial port of ITER chamber [27].

it would be enough to have the lithium radiating layer thickness $\sim 10^{-1}$ m, $n_e = 2 \times 10^{19}$ m^{-3}, $n_{Li} = 10^{19}$ m^{-3} and $\tau = 10^{-3}$ s. In T-11M experiments τ of Li ions near the boundary might be equal to $1-3 \times 10^{-3}$ s based on our estimations. It is possible to assume that such a small τ for ITER SOL would be the result of probable ELM activity. An additional decrease in τ can be achieved in principle by magnetic stochasticity (ergodic divertor), or by installation of special limiters. Figure 17(*a*) presents the idea of such a lithium limiter placed (figure 17(*b*)) into the ITER divertor SOL. It should be a pumped mushroom limiter [24], covered by a thin (<1 cm) Li CPS. The Li 'wick' should connect the 'hot' (>550 °C) top part of the mushroom and its 'cold' (<350 °C) down part, such as in 'hot spot' and 'cold' limiter ends in figure 2. This means that the top part of the limiter plays the role of a lithium emitter and down-collector of lithium and hydrogen isotopes. The vertical temperature gradient can be controlled by cooling of the limiter top and down parts. Such a limiter can work as a transformer of ITER SOL power flux to the radiation power; it decreases the divertor power load and increases the power load on the tokamak vessel wall. Our estimations show that total radiation efficiency of this type of Li-limiter can be more than 20 MW.

If the temperature of the lower part of the mushroom limiter is higher than 100 °C it should work as a reflector and compressor for He atoms (ash)—the result of fusion reaction. The simple pumping of the limiter port can allow the avoidance of the visible He ash part from the reactor chamber. This can be an example of the lithium absorption properties used for segregation of ash and fuel in ITER and DEMO.

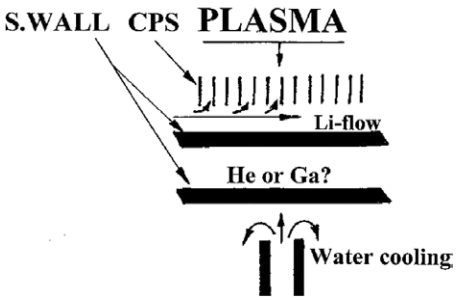

Figure 18. A scheme of CPS use as PFC with water cooling (down). S.Wall-double steel or vanadium walls with divided gap (He or Ga).

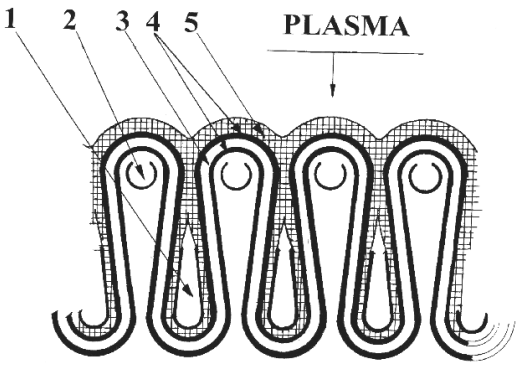

Figure 19. Bellows version of Li CPS first wall with the water-cooling. 1-Li flow channels, 2-water channels, 3-gap with He or Ga filling, 4-steel or V double bellows and 5-Li CPS.

The results of the Li CPS experiments as a tokamak limiter in T-11M permitted us to define the temperature framework for the use of Li PFC in fusion reactors. As we can see from figure 9, the best surface temperatures T_s for the Li PFC tokamak-reactor to be 300–500 °C, when the total Li influx is not high but increases rapidly with T_s. This nonlinear dependence forms the grounds for the negative feed back between T_s and the local energy flux to the first wall.

Unfortunately the cooling circuit of Li PFC seems to be a serious problem for the Li-reactor due to the incompatibility of Li and water. The double circuit system with an intermediate heat conductor (figure 18) may be suggested for its solution. The other version of such a double bellows wall is presented in figure 19 [15]. Its main parameters are a thin (<1 cm) CPS layer (5), Li- channels (1), water cooling (2) and the SS or V (Vanadium) [25] double bellows $\delta \approx 2$ mm (4), separated by the gap $\delta \approx 0.3$ mm, filled by He or Ga (3).

This gap plays the role of a heat conductor between the two bellows and should work simultaneously as a separator of water and lithium circuits during possible accidents. The internal bellows contact the CPS and Li while the external one contacts the water flow. A simple calculation shows that if the Li surface temperature is 450 °C and water 200 °C, the passing heat flux in the case of He filling should be equal to 0.4 MW m^{-2}. This is equivalent to 1.6 MW m^{-2} of the 14 MeV neutron load for DEMO. The heat flux can be increased up to 2 MW m^{-2}, if the gap is filled with Ga. These heat fluxes look suitable for the reactor first wall, but not for the ITER-like divertor plate. This means that smoothing of the heat load between

the divertor and the wall by non-coronal lithium radiation is the most realistic approach towards realization of the Li-tokamak reactor concept. The use of Ga as an intermediate heat conductor has two additional advantages:

- low permeation of hydrogen isotopes in Ga can prevent tritium diffusion into the water circuit, and
- Ga ability to create the compound LiGa with a melting point $T_M = 740$–$760\,^\circ$C [26] can be used for self-recovery of internal bellows (Li-circuit) in cases of cracks or possible damages.

Finally, it should be emphasized again that the key question of Li use in future reactors is lithium penetration to the plasma centre. The TFTR, T-11M and CDX-U experiences show existence of the Li screen mechanism, but it is not certain that it will exist under reactor conditions. The final answer should give rise to experiments on different large tokamaks and ITER as well.

5. Summary

The T-11M experiments with Li limiter on CPS have shown good compatibility of liquid lithium PFC with tokamak plasma. The main features of the behaviour of liquid lithium PFC on CPS in tokamak plasma can be summarized as follows.

1. The surface tension forces in CPS may be used to solve the problem of ponderomotive forces ($J \times B$) which are responsible for the splashing and can suppress it. Also surface tension forces may use surface regeneration of PFC in the tokamak.
2. In experiments with hydrogen (deuterium) and helium plasmas on T-11M tokamak with Li–CPS limiter the following have been shown.

- No spontaneous bursts of lithium injection under a heat flux to the limiter up to the power load 10–20 MW m^{-2}.
- Total lithium erosion of Li PFC during the interaction with tokamak SOL plasma is close to the level of hydrogen and lithium ions sputtering.
- The lithium non-coronal radiation protected the limiter from high power load during disruptions.
- In ordinary tokamak regimes a non-coronal radiation of Li ions, circulated in limiter SOL removes up to 80% P_{OH} power to the vessel wall. So the plasma SOL with a Li limiter together worked as a heat tube of the highest equivalent thermal load – more than 30 MW m^{-2}.
- The solid basis of the CPS limiter had no damages after more than $2{\cdot}10^3$ of plasma shots.
- The recovery temperature of the hydrogen isotopes from Li 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 circuits with lower PFC temperatures.

The lithium ion behaviour in tokamak plasma (TFTR, T-11M, CDX-U) leads to speculation on the existence of a lithium screening mechanism. This assumption should be tested in large ITER-like tokamaks.

3. These results make a convincing case for the application of liquid lithium PFC for steady-state tokamak-reactors such as ITER and DEMO. The following problems of such

tokamaks might be solved: wall and divertor plates erosion (by self-recovery), 'dust' accumulation and redeposition, tritium recovery, radiative cooling of the core plasma due to the high Z impurity, heat removal during the steady-state regime and disruption.

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

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