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Li-CPS limiter in tokamak T-11M

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Abstract

This paper is a review of the experimental investigations of Li-behavior as limiter material in real tokamak. All experiments were performed in tokamak T-11M with main parameters: plasma current, $I_p = 100$ kA; duration of the discharge 0.1 s, toroidal magnetic field, $B = 1$ T; major radius, $R = 0.7$ m; minor radius, $a = 0.19$–0.23 m; average electron density, $n_e \sim (1$–$5) \times 10^{19}$ m^{-3} and electron temperature, $T_e(0) \sim 0.3$–0.5 keV. Two moving limiters with similar geometry were installed for comparison in chamber—the conventional graphite-boron limiter and Li-limiter on the basis of capillary pore system (CPS). The Li-influx to plasma was measured by visible, UV and SX plasma emission. The main experimental results were: (a) no dramatic bursts of lithium injection at heat load close to the tokamak-reactor level, ~ 10 MW/m^2, were observed, (b) the total lithium erosion from limiter can be explained by deuterium plus lithium ions sputtering (self-sputtering), (c) high lithium radiation during disruptions prevent Li-limiter from high power load, and (d) the solid basis of CPS limiter had no damages after more than 200 shots with disruptions. The main effect of lithium use in T-11M was the rise of the first wall getter properties, i.e. recycling reduction for not only H_2 and D_2 but for He too. The temperature of hydrogen isotopes desorption was 350–400 °C and He desorption was 50–100 °C.
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1. Introduction—towards liquid lithium divertor

It is well known that both limiters and tokamak divertor plates will be subjected to extremely high heat loads as well as to high fluxes of deuterium, tritium and impurities of different types. In such conditions, divertor plates melting with subsequent cracking is unavoidable. Also dangerous are cyclic thermal loads resulting in low-cyclic fatigue arising, for example, from edge local modes (ELM). The work on the ITER reactor project, for instance, showed that conventional design solutions of divertor and divertor plates for plasma burning, practically in steady state, in a tokamak of such a scale meet with serious difficulties. In particular, it was found necessary to replace the divertor modules after 1000 shots and to introduce tungsten into the structure of plasma-facing com-

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ponents. At the same time, high-Z materials were rejected in classical tokamaks because of plasma contamination by dust resulting from limiter erosion in MHD-unstable discharge conditions. We think that there is a principal possibility to move lower in the low-Z range, namely to develop a lithium divertor where dust and contamination problems would be solved naturally.

It is reasonable in extreme operational conditions to provide for periodic replacement of the plates, or else to make them movable, thus decreasing the local thermal loads. Separatrix sweeping during the discharge is one of the methods of spreading the power load over a larger surface. More radical solutions should be the moving tokamak limiters, namely rotating disks [1], or a stream of small metal or graphite balls intersecting the edge of the plasma column [2]. The rotating limiter found successful application on PLT [3]. However, the means for moving divertor plates in reactor-scale tokamaks faces severe problems due to the transmission of a mechanical motion in vacuum.

It is believed that these difficulties may be overcome when liquid metal flow will be used as a moving contact with plasma. The idea to use liquid metals as plasma-facing materials in fusion reactors with magnetic and inertial confinement to control high heat and particle loads attracted attention for a long time. In particular, thick liquid metal films flowing on the wall were proposed for one of the first projects of tokamak-reactor UWMAK-I [4]. This approach meets the main reactor requirements and ensures heat removal and self-regeneration of the plasma-facing surface. The hydrogen and helium ion absorption by moving lithium film was first investigated in Kurchatov Institute [5]. Gallium-based liquid metal film limiter [6,7] has also been first designed and tested in T-3M tokamak. The main result of these studies showed that in real tokamak conditions with fast variations of the magnetic field in time, it was practically impossible to make homogeneous flowing liquid metal film. Intensive splashing of liquid metal resulted from tokamak disruptions and softer MHD events.

Another variant, where a limiter plate based on a liquid metal jet-drop curtain was used, presents a modification of a small balls limiter. This provided by means of an MHD pump (Fig. 1 [8]) the successful transport of the liquid metal in tubes with electrical insulated coating. Jet decay into a regular drop flow (each drop being 2–4 mm in diameter) provided the possibility to achieve a high speed (4–10 m/s) of metal flow across the constant or variable magnetic field without interacting with it. Jets decaying into a drop flow may be achieved by means of a practically used [9] MHD shaper. The main idea of the shaper is the creation of a resonant resistance to the flow. Thus achieving decay of the flow into separate drops by Rayleigh instability excitation. The gallium-based jet-drop curtain limiter was successfully tested in tokamak T-3M [8]. However, for successful usage of such a scheme in a tokamak-reactor, we need to have the electrical insulating cover of the liquid metal guide tubes with high operational reliabilities.

A new idea to use liquid metals as tokamak divertor plates protector was advanced, based on the surface tension forces in capillary channels that also compensates forces induced in the metal by the varying magnetic. These capillary channels could be produced in the form of the so-called capillary pore system (CPS) [10–12], which permit a liquid lithium replenishment during steady-state tokamak operations. This property of self-regeneration becomes essentially important, if we take into account that the ITER divertor plate will operate in the presence of frequent ELMs, which should be the reason of the enhanced erosion. One may expect that surface self-regeneration will become the most important factor for the post-ITER reactors. In [13] (Fig. 2) was proposed such kind suggestion of divertor plates and wall coating by CPS for ITER-like tokamak with very slow moving (<1 cm/s) of liquid metal. The key advantage of this divertor is the limitation of liquid metal splashing during minor disruptions and ELMs.

Their physical and technological analyses [13] do not show any principal limitation on such suggestions. But, we might expect practical obstacles concerning two points: the appearance of anomalous lithium influx from the wall and bed lithium compatibility with tokamak plasma. A

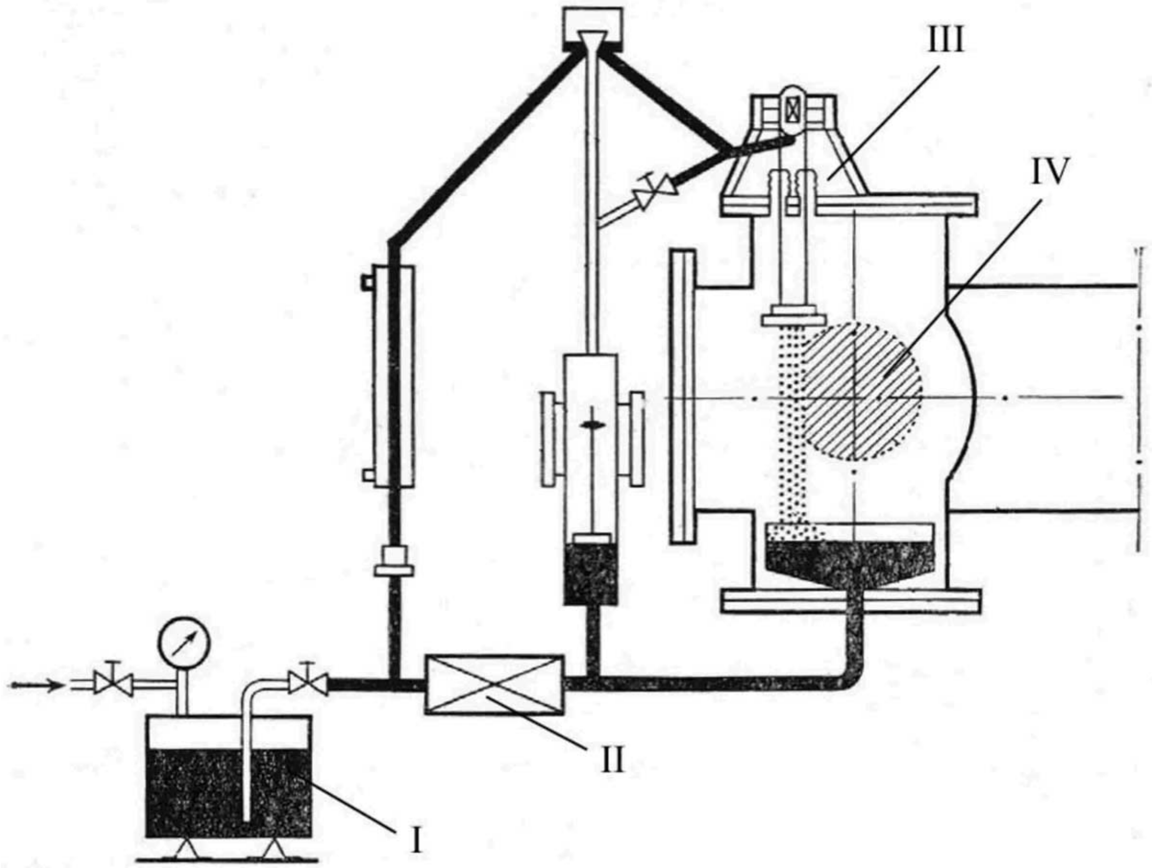

Fig. 1. Scheme of T-3M liquid gallium jet-drop limiter. I: gallium reservoir; II: MHD-pump; III: MHD shaper; IV: plasma column.

concern may arise that lithium having $Z = 3$ will concentrate around the axis of the plasma column and will come from the wall at random without a possibility to control it, for instance, by:

- powerful unipolar arcs,
- local emission bursts (like carbon blooms),
- development and splashing of micro-capillary waves at the plasma–liquid boundary,
- any mechanism of abnormal lithium erosion.

Some of these concerns can be clarified at present, first, on the basis of positive experience of lithium pellet injection into the hot plasma in tokamak TFTR [14–16] and our T-11M tokamak operation experience with liquid lithium CPS limiter [17,18].

TFTR experiments with lithium pellet and lithium aerosol injection (DOLLOP) during the discharge phase have indicated that lithium is well tolerated by fusion plasma contributing to the formation of a protective layer between the hot zone and cold wall without an increase of Z_{eff} in the plasma core. In this way, discharge regimes with maximal neutron yield and maximal triple product $n\tau T$ [14] have been obtained. In TFTR experiments, the lithium pellet injection was followed by high-energy D-T (NBI) injection heating. The D-T injector ($E > 100$ keV) was used as plasma heater and as D-T feeder to the plasma core simultaneously. This combination appears very convenient for the reactor: injection of cold lithium from the wall into the plasma periphery

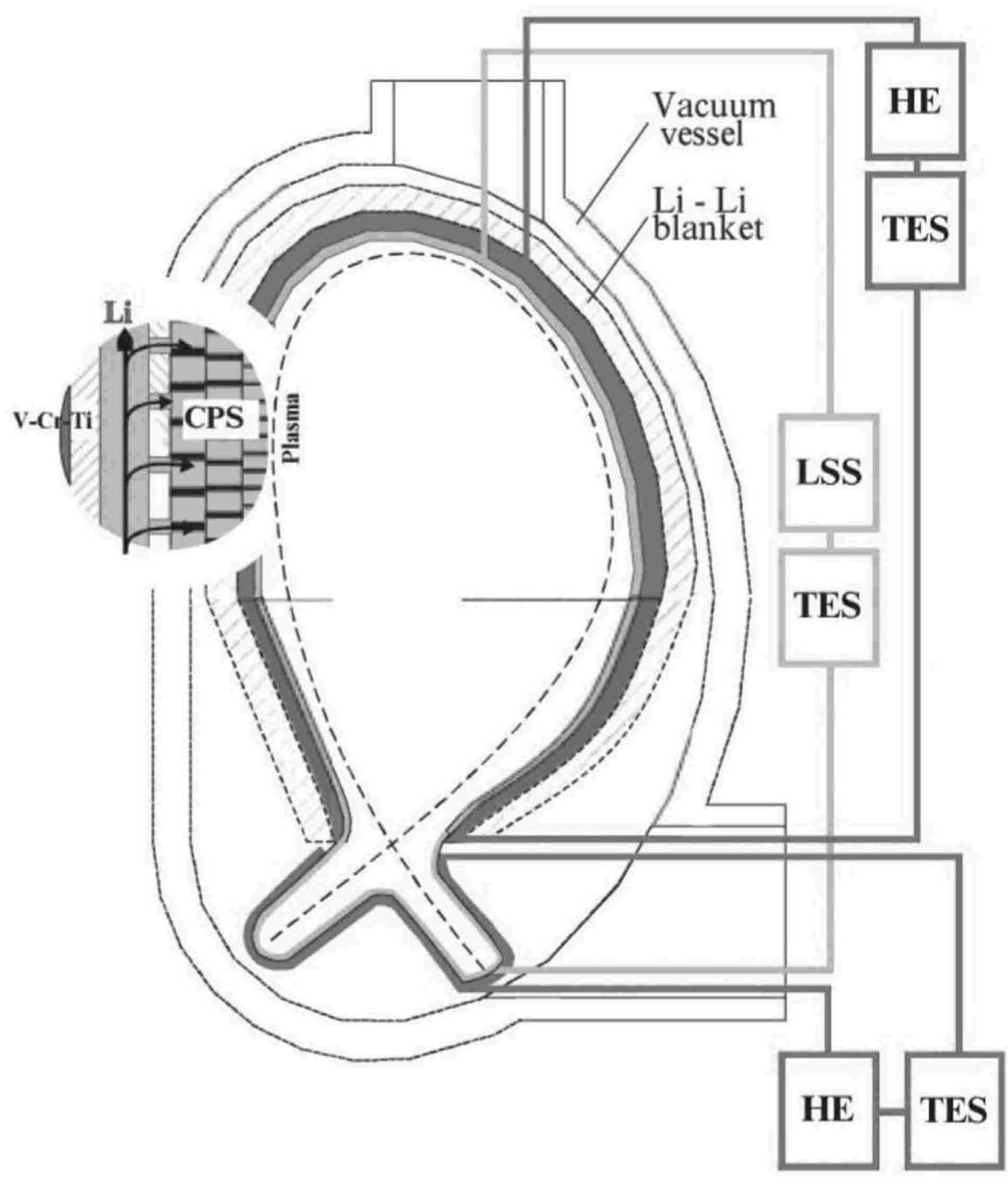

Fig. 2. Schematic view of fusion reactor with lithium divertor on CPS basis. HE, heat exchanger; TES, tritium extraction system; LSS, lithium supply system.

and D-T fuel into the central zone. A considerable gap in confinement time and, consequently, in D-T and Li density could be reached in the plasma center by such scheme of operation. The TFTR experiments have not revealed a tendency of lithium concentration in the center.

The first test of tokamak plasma compatibility with CPS-lithium limiter was performed in T-11M [19].

2. Interaction of plasma with lithium capillary pore structure in tokamak T-11M

Experiments in T-11M tokamak have been performed in order to prove compatibility of a lithium CPS with boundary plasma in tokamak conditions close to the quasi-stationary plasma parameters expected in reactor. The main task was to ascertain whether spontaneous lithium bursts from the liquid wall to the chamber volume were an important effect or not. Other then this, lithium interaction with working gases, lithium migration in plasma, technology of lithium application in tokamak and rehabilitation of the facility after lithium tests have been studied.

2.1. Experimental conditions

The performance data of the small tokamak device T-11M [17] are the following: $R = 0.7$ m, $a = 0.2$ m, $B_T = 1$ T, plasma current, $J_p \approx 100$ kA ($q(a) = 3$–4), discharge pulse duration about 0.1 s. The heat load to limiter is about 10 MW/m^2, similar power density is expected to be on the ITER divertor plates. Taking into account the strong dependence of the heat load on electron temperature (as $\propto T_e^{\alpha}$, where α can vary between 7/2 and 3/2 depending on plasma collisionality near divertor plates), one may suppose that boundary plasma temperatures $T_e = 20$–30 eV that are characteristic of modern tokamaks will be of about the same level or lower (for higher density) in a reactor machine. All negative effects occurring at the wall that are known at present, namely arcs, emission bursts, ion sputtering, micro-capillary waves, etc., are functions of sheath potential and, finally, of T_e. High recycling condition regime could not be simulated exactly in T-11M. However, it corresponds to lower $T_e < 5$ eV that seems preferable for lithium divertor plates. Therefore, we suppose that T-11M modeling experiments were carried out in the conditions close to or even more severe than those of a reactor periphery.

A schema of the T-11M experiment is presented in Fig. 3. Movable rail limiter (Fig. 4) with plasma-contacting surface made of lithium CPS (two versions of CPS were studied, with pore radius $R_{eff} = 100$ and 30 μm) was inserted into plasma to about 5 cm thus limiting plasma column aperture.

The study of the first version limiter showed that induced electromagnetic forces appearing at the limiter edges during disruptions were underestimated. As a result, splashing of lithium across the B_T-field lines was observed. This effect was suppressed in the second limiter version ($R_{eff} =$ 30 μm), where liquid lithium confinement condition was satisfied with a good margin.

Conventional graphite limiter was placed in the opposite port for comparison with the lithium one. Two fast thermocouples were fitted in lithium limiter close to its surface to measure total energy absorbed by the limiter during the discharge. Standard optical diagnostics were applied to observe lithium penetration to the plasma near and far from limiter. A 15-channel bolometer system was also set up and special infrared diagnostics were developed to measure the limiter surface temperature during the discharge and to calculate the deposited power [17,18]. The local heat deposited was shown to be 10 MW/m^2 in a quasi-stationary discharge for effective heat pulse length equal to 50 ms. The limiter temperature rise during this discharge was 100–250 °C. A special heater incorporated in the limiter structure enabled higher temperatures to be obtained (up to 400 °C by pre-heating).

2.2. Lithium erosion

No catastrophic events leading to abundant lithium injection in the MHD-stable discharge conditions within the whole lithium temperature range (T_L from 20 to 600 °C) was observed in those T-11M experiments, an important result of this work. Lithium and graphite limiters worked practically in a similar way if additional heater was not used [17,18]. Heating of the lithium limiter gave rise to lithium injection into plasma detected by an increase of lithium line radiation (LiI, 670.8

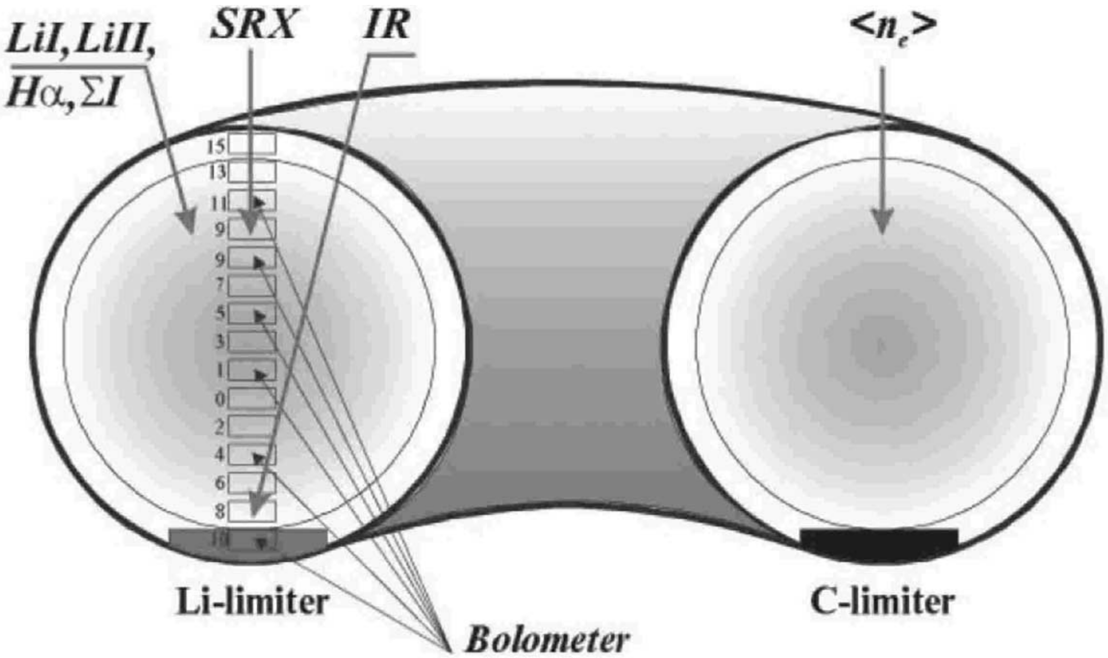

Fig. 3. Scheme of the T-11M experiment with lithium CPS-limiter. 1–15, fast bolometer measurement channels; IR, channel of infrared measurements; SXR, soft X-ray measurements channels; LiI, LiII, H_α, ΣI, spectral line and total light measurements channels; n_e, electron density micro-wave measurements.

nm) and visible integral light emission (proportional, as usually to LiI intensity) in the vicinity of the limiter. Temporal dependence of integral light emission for three discharges with different initial limiter temperatures (T_0) is presented in Fig. 5. It is evident that while T_0 increases, lithium flux begins to grow in time. Lithium light peak during the current decay ($T_0 = 300$ °C) may be explained by recombination process (MARFE) because it coincides with decrease of plasma heating and it is not followed by growth of plasma density and cannot be explained by a simple additional lithium influx to the plasma.

Estimation of total lithium emission (erosion) from limiter was performed by use of electrical biasing method [20]. It is shown that for limiter temperatures, $T_0 < 500$ °C, erosion remains in the limits expected for sputtering by D^+ and Li^+ ions with sputtering yield from 0.5 to 1 (for $T_L >$ 500 °C, conventional evaporation appears to become the main channel of lithium emission).

This is in correlation with the known data on Li-sputtering [21]. The monotonic rise of lithium flux during the discharge for $T_0 > 200$ °C may be attributed to self-sputtering by Li^+ ions accumulated in the plasma periphery. However, the last PISCES-B experiments [22] and some measurements of Allain [23], where Li^+ accumulation effect was excluded, show the same kind of fast-increasing dependence of lithium erosion from initial target temperature as was in T-11M experiments. May be that is the result of some new mechanism of enhanced liquid lithium sputtering, which depends on the lithium temperature [22].

The T-11M results support this explanation. In Fig. 6, in arbitrary units, we have presented the intensities of total light and LiI emission in vicinity of T-11M CPS limiter, as a function of its surface temperature T_L for middle part of discharge. The comparison, Fig. 8 with result [22,23], permits us to conclude that lithium emission from CPS limiter in tokamak discharge behaves much like pure lithium sputtering in simulated experiments. Future investigations are planned for T-11M, because these results are very important for choosing of Li-divertor parameters.

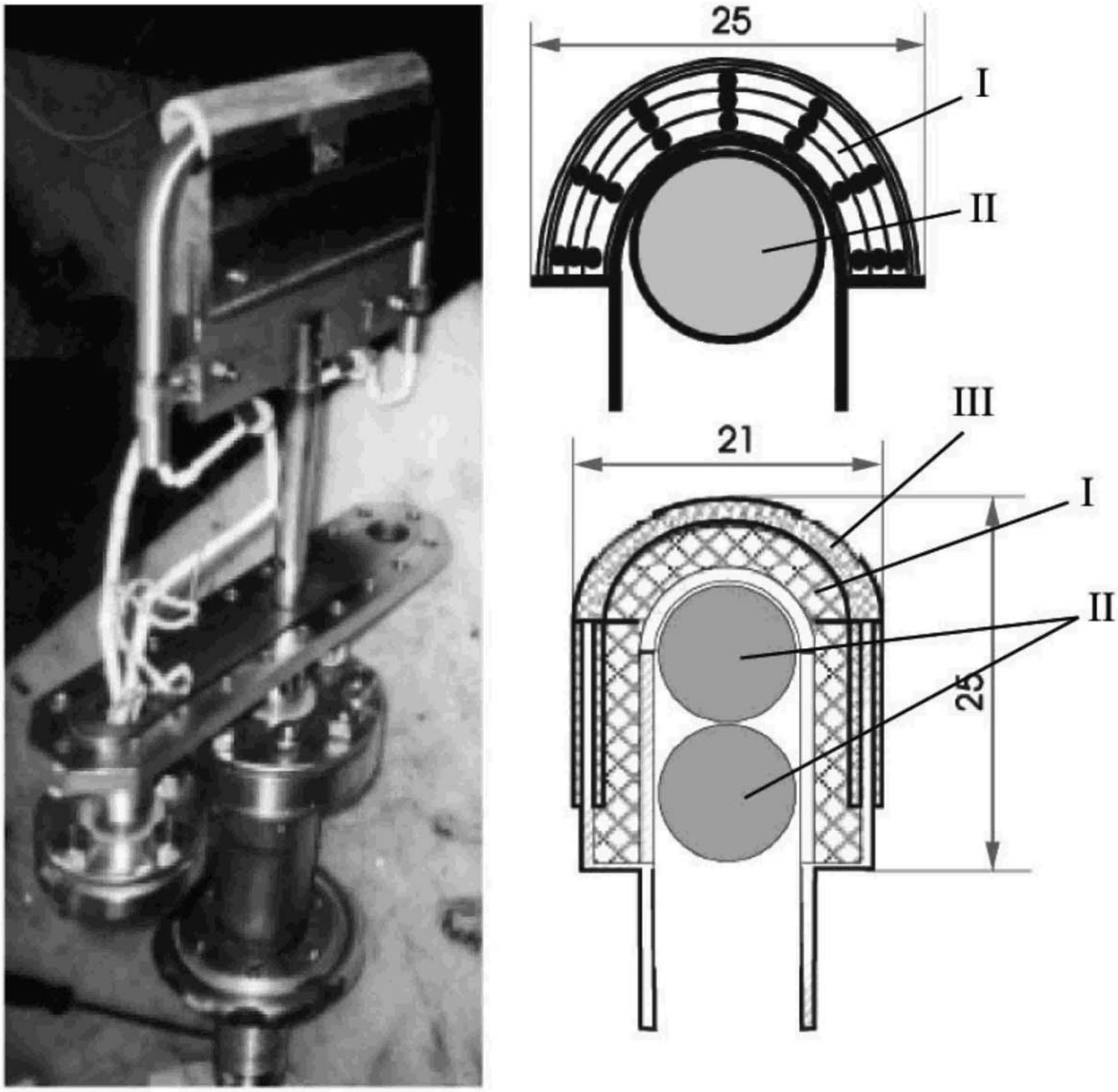

Fig. 4. Lithium rail limiter: (a) general view; (b) CPS lithium limiter (cross-section); I: CPS layer with pore radius 100 μm; II: limiter pre-heaters; III: CPS layer with pore radius 30 μm. All dimensions are in mm.

2.3. Radiating cooling

Therefore, lithium emission into the discharge could be controlled by an increase of initial limiter temperature in T-11M. One could expect to obtain a growth of periphery radiation and, by this, to reduce the heat load to the limiter. It was really reduced by approximately a factor of 2 by these manipulations in the helium discharge [17,18]. Even larger fraction of the heat flux is supposed to be radiated with the increase of heat pulse duration and that will be closer to the limit of lithium radiating mantle. Thus, a step to radiation improved (RI)-conditions with a smaller impurity contamination of the center in actually operating tokamaks [24] and to radiating divertor in a reactor may be done. Lithium confinement time in the periphery layer (τ) may be taken as a

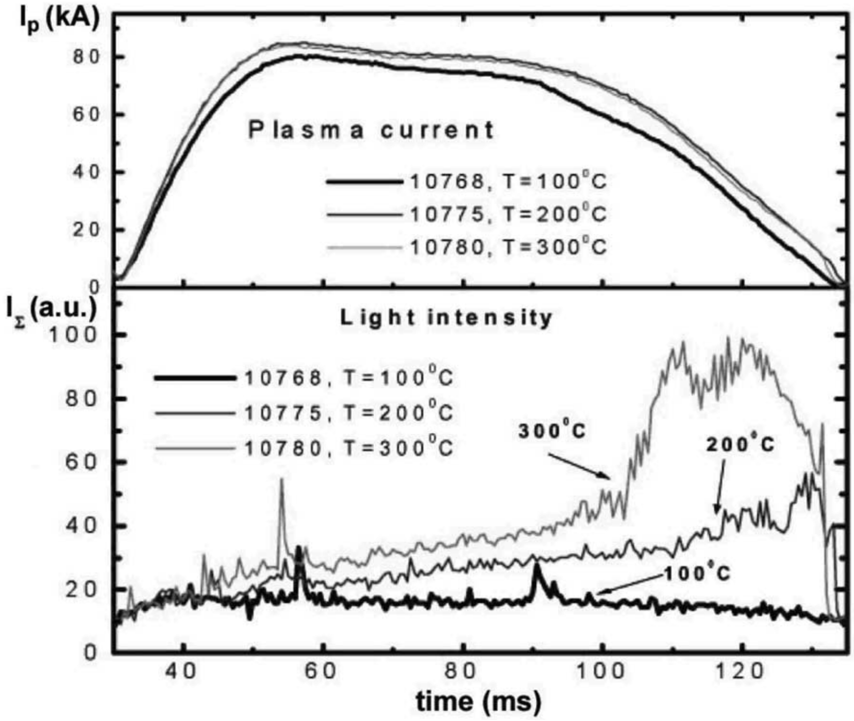

Fig. 5. Temporal behavior of integral light emission from the lithium limiter for different initial temperatures (100–300 °C).

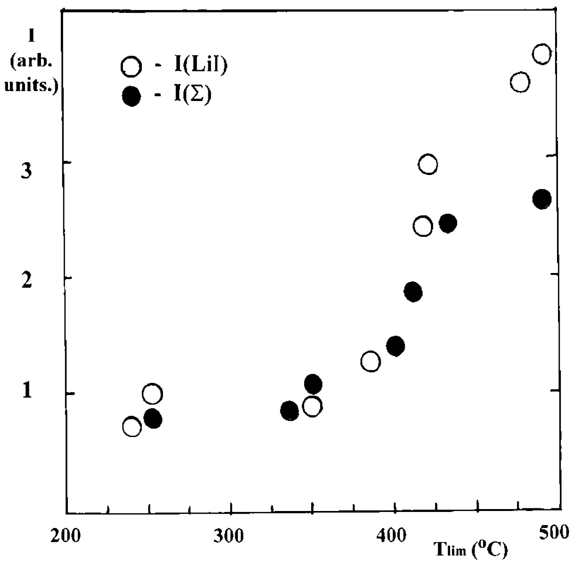

Fig. 6. Total light and LiI intensity as function of current CPS temperature T_{lim}.

governing parameter. If it is small, then lithium ions will not reach a coronal equilibrium before they return to wall. In this case, the lithium radiation intensity becomes much higher than that expected for coronal equilibrium. Fig. 7 [26]

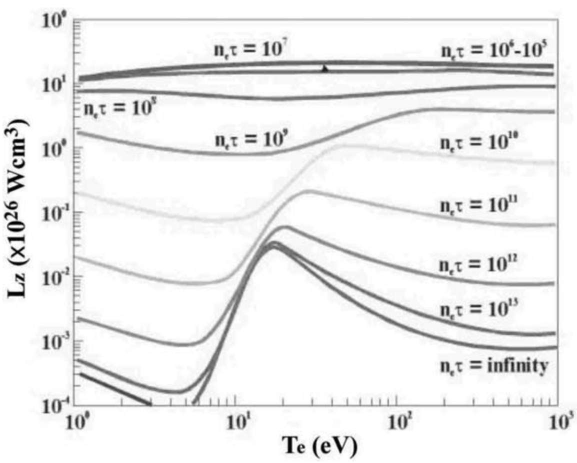

Fig. 7. Lithium radiation date as function of $n_e\tau$ and T_e.

shows the calculated evaluations of such radiation (per atom and per electron) as a function of T_e and of $n_e\tau$ (cm^{-3} s), which is a characteristic factor of deviation from the coronal equilibrium conditions ($n_e\tau$—infinity). For example, the lithium radiation may be by two orders higher than the coronal equilibrium level for $T_e = 30$ eV, $n_e = 10^{13}$ cm^{-3} and $\tau = 10^{-3}$ s that is quite realistic for plasma periphery. The radiating mantle also appears realistic in these conditions. Principle possibilities of such $n_e\tau$ control are just known: ergodic magnetic fields at the plasma boundary, controlled ELMs, local excitation of MHD activity, etc. Further development of these methods is needed.

2.4. Deuterium capture in lithium and desorption effect

One of the most evident, though expected, consequences of lithium introduced into real tokamak machines (TFTR, T-11M, CDX-U) was the high growth of sorption of hydrogen species D^+ and H^+ on the wall [20,25,26]. Moreover, helium sorption was discovered in T-11M experiments as well [20] with a slow desorption during 20–100 s after the discharge (Fig. 8). However, in order to avoid this effect of helium sorption, it was sufficient to heat the T-11M vessel wall to 50–100 °C. For deuterium, even highest attainable wall temperature 250–300 °C turned out to be insufficient. At the same time, the lithium limiter could be heated up to 450 °C. The result of the limiter heating cycle after experimental campaign is illustrated in Fig. 9. Shown is deuterium pressure as a function of limiter temperature. One can see that the captured deuterium is desorbed from lithium at temperatures higher than 320 °C. Lithium hydrides are supposed to decompose at temperatures about 600 °C. Therefore, one may conclude that considerable part of deuterium was captured by lithium not in the form of deuteride but it was dissolved in lithium. It means that a simple heating to 370–500 °C [27] seems sufficient to desorb deuterium. The character of lithium interaction with hydrogen isotopes should be studied in more detail. The observed difference of helium and deuterium desorption properties may be used for tritium–deuterium separation from helium in the reactor lithium loop.

2.5. Disruption shielding

As was shown in special disruption simulation experiments [28] through plasma accelerators, a dense plasma layer was formed during high-energy plasma interaction with CPS target. The major part of the plasma energy (~97–99%) is absorbed and radiated in this layer which plays the role of a

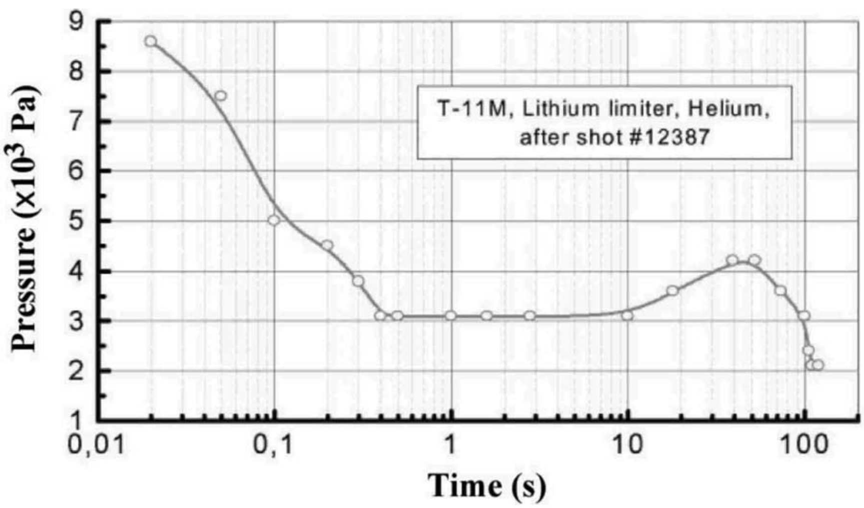

Fig. 8. Temporal behavior of He pressure in the tokamak chamber after helium discharge (pump speed is decreased).

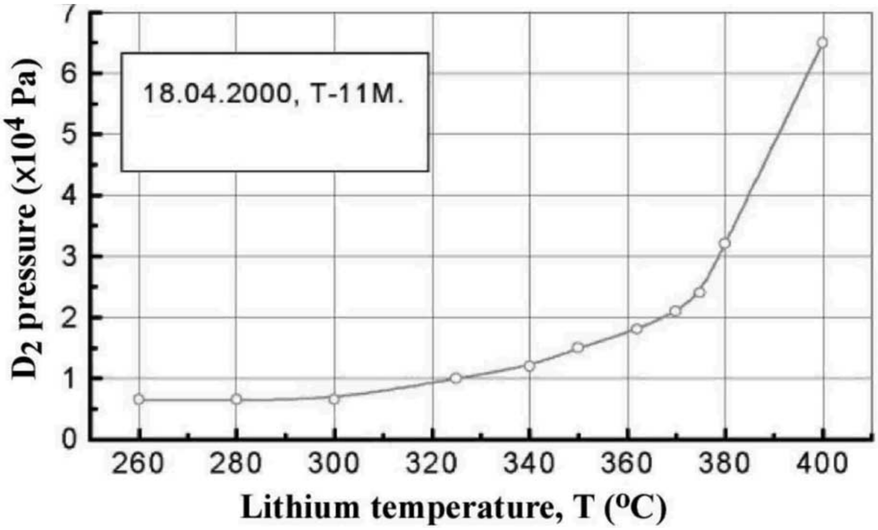

Fig. 9. Evaluation of D_2 pressure in the tokamak chamber during lithium limiter heating after D_2-discharge operations.

shielding layer. This result has been confirmed later experimentally in a T-11M tokamak: only 30–50 J of about 0.7 kJ of total plasma energy loss has been found to reach the rail limiter during disruption events, while under normal discharge condition the energy loss to the limiter is equal to 50% of total energy flux from plasma column [29]. The solid basis of CPS limiter had no damages after more than 200 shots with disruptions.

2.6. Future plans

The main physical problem, which should be resolved for realization of stationary liquid lithium divertor, is the controlled lithium exchange between plasma boundary and divertor plates during all reactor regimes. As we can see from Fig. 8, it is possible in principle, over a lithium temperature range from 350 to 500 °C, where a plasma heat flux increase should produce no proportional lithium emission and probably a plasma cooling. As a result, we hope to have the feedback stabilization of boundary conditions. However, today we have information only about lithium emission processes in short duration T-11M regimes (0.1 s). We plan to repeat our investigations in upgraded T-11M with enlarged discharges up to 0.3 s and transition by water or organic (diphenil, for example) cooling to stationary Li-limiter regime. Main feature of stationary CPS-limiter should be relative think CPS-slab (1.5–2 mm), which will have a good heat contact between it and cooling matter. That will be the next step of our program.

3. Conclusion

1) CPS-based liquid lithium divertor appears feasible based on the experimental, calculation and design studies and technological experience of today. The following problems could find solution:
   - wall and divertor plates erosion,
   - ‘dust’ accumulation and redeposition,
   - tritium recovery,
   - low $Z_{eff}(0)$,
   - heat removal in stationary conditions.The physical and technological analysis does not show today any principal limitation on CPS liquid lithium divertor.
2) A series of experiments on T-11M tokamak has proven compatibility of lithium CPS limiter with plasma in all operating conditions. No spontaneous burst injection of lithium at heat load close to that of reactor level 10 MW/m^2 has been observed.

3) High level of lithium radiation has been detected including the case of disruption events so that solid basis of CPS limiter had no damages after more than 2×10^3 of plasma shots.These experiments have shown that hydrogen (deuterium) and helium ions bombarding lithium wall or limiter in normal conditions in tokamak periphery ($T_e \approx 10$–30 eV) are captured by lithium. Difference in desorption temperature was shown to exist for hydrogen isotopes (320–350 °C) and helium (50–100 °C). These effects may be used for D-T separation from He.
4) By the next step of T-11M program, it will be the repetition of investigations in upgraded T-11M regimes with enlarged discharges up to 0.3 s and transition by water or organic cooling to stationary Li-limiter regime.

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