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Lithium divertor concept and results of supporting experiments

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Abstract
The ITER project development has shown that considerable difficulties are encountered when already known engineering solutions and materials are used for divertor and divertor plates for tokamaks of such a scale. We offer to use a Li capillary-pore system (CPS) as a plasma facing material for tokamak divertor. Evaporated Li serves as a gas target and redistributes thermal load. The heat flux from the plasma is transported to the first wall by Li radiation in the plasma periphery. This allows the divertor plate to reduce the heat flux. A solid CPS filled with liquid lithium has a high resistance to surface damage in the stationary mode and during plasma transitions (disruptions, ELMs, VDEs, runaways) to assure normal operation of the divertor target plates. These materials are not the sources of impurities giving rise to Z_{eff} and they will not be collected as dust in the divertor area and in ducts.

Experiments with lithium CPS in a steady-state mode (up to 25 MW m^{-2}) and in plasma disruption simulation conditions ($\sim$5 MJ m^{-2}, $\sim$0.5 ms) have been performed. High stability of these systems have been shown. Li limiter tests on T-11M tokamak have revealed the lithium CPS compatibility with the edge plasma for energy loads of up to 10 MW m^{-2}. In a stable discharge mode at lithium limiter temperature of 20–600˚C, no Li abnormal erosion and injection to plasma have been detected. A high sorption of D^+ and H^+ ions on the vessel walls was the main substantial result of the replacement of a graphite limiter by lithium one. He and D sorption was terminated by wall heating up to 50–100˚C and above 350˚C, respectively. T-11 tests on helium discharge allowed to reduce limiter heat load by a factor of two due to lithium radiation.

All the experimental results have shown considerable progress in the development of lithium divertor.

Introduction

Low Z solid materials are intensively investigated and are used as plasma facing materials in large fusion experimental machines with a short operation cycle. The work on the ITER reactor 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 introduce tungsten into the structure of plasma facing components. 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.

The idea to use liquid metals as plasma facing materials in fusion reactors with magnetic and inertial confinement attracted attention for a long time to control high heat and particle loads [1–6]. In particular, thick liquid metal films flowing on the wall were proposed for one of the first projects of tokamak reactor UWMAK-I [7]. This approach meets the main reactor requirements and ensures heat removal and self-regeneration of the plasma facing surface. Besides, the disposal of different impurities from the reactor vessel becomes a trivial matter.

Research work on liquid lithium films has been initiated to reveal these advantages appearing to be so important [8]. For instance, gallium-based liquid metal limiters [9, 10] have been also designed and tested in T-3M tokamak [10, 11]. The first 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.

A liquid metal jet-drop curtain [10, 11] appeared to be a more advanced option, it would eliminate the influence of induced forces on the liquid metal motion in the magnetic field. Now a new idea to make a stable lithium wall in tokamak due to the interaction of an intense lithium stream with magnetic field [12] is proposed and a number of others as well [13–16].

A new idea to use liquid metals in tokamaks was advanced based on the surface tension forces in capillary channels that may also be used to compensate forces induced in metals carrying electric currents in magnetic field. These capillary channels could be produced in the form of the so-called capillary-pore systems (CPSs) [17–19]. In contact with plasma self-regeneration is an intrinsic property of such a structure. This property becomes essentially important if we take into account that the ITER divertor plate will operate in the presence of frequent small disruptions—ELMs—which are the reason of an enhanced erosion. One may expect that surface self-regeneration will become the most important factor for the reactors next to ITER-FEAT.

Lithium compatibility with tokamak plasma becomes a special issue that is now intensively studied [20–27]. It could be subdivided into two parts: nature of lithium influx from the wall and lithium behaviour in the plasma. A 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 an abnormal lithium erosion.

Some of these concerns can be clarified at present, first, on the basis of positive experience of lithium injection into the hot plasma in tokamak TFTR [20, 21] and of T-11M tokamak operation experience with liquid lithium CPS limiter [26, 27].

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 protective layer between the hot zone and cold wall without increase of Z_{eff} in the plasma core. Thus, discharge regimes with maximal neutron yield and maximal triple product $n\tau T$ [20] have been obtained. Figure 1(a) [21, 22] shows characteristic wave form of plasma density in a typical TFTR discharge with lithium pellet injection followed by high energy D–T (NBI) injection heating.

At the same time D–T injector ($E > 100$ keV) served as a fuel feeder of the plasma core. This combination appears very convenient for the reactor—injection of lithium into the plasma periphery and of fuel into the central zone. A considerable gap in confinement time and, consequently, in density of D–T fuel and lithium in the plasma centre could be reached.

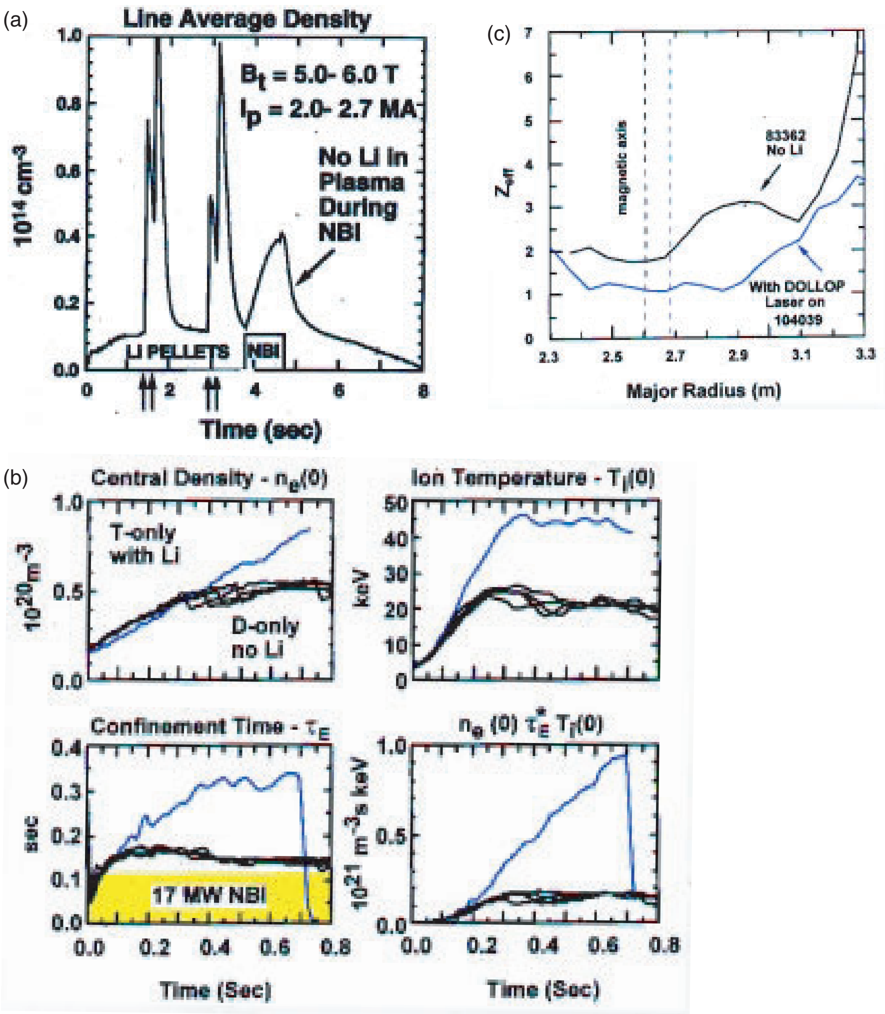

Figure 1. (a) Line-averaged density wave form in TFTR discharge. Li pellet injection and hot plasma generation by NBI. (b) Central density, ion temperature, confinement time and $n_e(0)\tau_\tau T_i(0)$ wave forms during NB heating. (c) Z_{eff} profiles with (DOLLOP) and without Li injection.

Time behaviour of the $n\tau T$ product for two discharges with and without Li pellet injection is given in figure 1(b). Difference almost by an order is mainly due to better plasma confinement. This may be attributed to a higher concentration of lithium in the plasma periphery, it does not penetrate into the core thus favouring the current density peaking. This assumption is proved by figure 1(c) where Z_{eff} distribution along the major radius R is shown. The Z_{eff} value decreases down to $\sim$1.0 and grows to the boundary, that is exactly what one could expect if lithium is concentrated in the periphery. The TFTR experiments have not revealed a tendency of lithium concentration in the centre.

We propose to use CPSs to take advantage offered by lithium as a plasma facing material and on this basis to develop a fusion reactor divertor. Taking into account the gained experience, we consider different aspects of this proposal and report the results of modelling experiments.

1. Lithium CPSs

The following basic considerations were the starting point when an idea to use lithium CPS for protection of the divertor target plates was proposed and developed [17–19, 27–29]. Materials taken for target elements are practically non-serviceable because of extremely high local heat loads, so a natural way to lower them is to redistribute them over a larger area. It is known [31] that the most efficient means of heat transfer in high temperature machines for energy conversion are evaporation–condensation elements with liquid metal as a coolant. This method of heat removal provides the highest performance with an appropriate choice of the working fluid; for lithium it is hundreds of MW m^{-2} at temperatures below the boiling point (figure 2). The second efficient mechanism to decrease local heat loads is lithium radiation. While puffing of (heavy) gases is envisaged in the gas target concept, here in lithium divertor such a target is formed naturally by operating temperature control. Thus, energy will be redistributed over a larger area by radiation, and the heat flux to the divertor plate will be decreased. Finally, power removal from the divertor will be carried out by thermal conductivity to cooling loops and further to energy conversion system without overheating.

CPS is used in the target plate design to confine liquid metal in a given configuration and to feed the evaporating surface with liquid metal. Characteristics of CPSs (changing porosity,

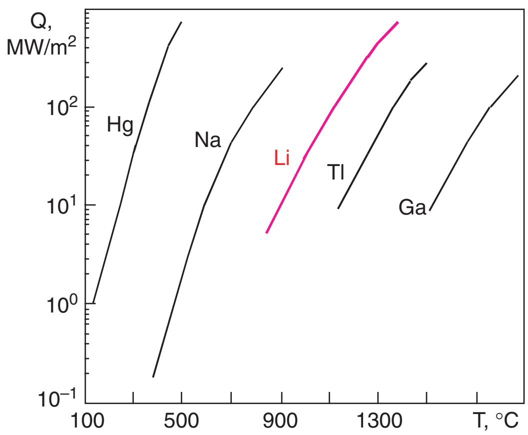

Figure 2. Limit values of heat flux vs temperature for evaporation from a surface to vacuum.

anisotropic permeation, working surface geometry etc) may vary significantly depending on fabrication technology. As is shown later, CPS structure ensures sufficient pressure level in the feeding system due to capillary pressure with no need of external pressure source. The system is self-sustaining and self-regenerating because the CPS working fluid pressure distribution is extremely sensitive to changes in local heat load distribution on its surface. This principle is realized in heat pipes of different applications [29, 31].

Lithium makes the proposed divertor concept highly efficient and it has a number of principally new features. The concept is feasible for the following reasons:

(a) lithium has a low Z that determines its minimal effect on the main plasma in comparison with any other materials,
(b) high latent heat of lithium evaporation, radiation and ionization of lithium vapour lead to redistribution of the important part of incoming energy, thus decreasing power load density on the divertor,
(c) lithium fits well the reactor design with self-cooled lithium–lithium blanket; service systems could be used both for the blanket and the divertor; tritium extraction technology can be the same for both components; the same structure material can be used in those systems—low activated vanadium alloys that are well compatible with lithium at temperatures below 700˚C.

Lithium interacts actively with hydrogen isotopes and forms solutions and hydrides [29]. Helium and other noble gases don't interact with lithium in ordinary conditions. Based on these properties, it is principally possible to separate helium from hydrogen isotopes and to evacuate it through pumping system.

Long life time of target elements will be ensured by the following properties of liquid metal divertor:

- erosion of the target plate is compensated due to constant feeding with liquid lithium,
- thermal gradients will not give rise to stresses in the lithium-filled CPS; consequently, contrary to the solid divertor surface no cracking and no fatigue cracks will occur on the target plates;
- no need to elaborate technology of attachment of the capillary-pore structure to divertor supporting structure which is a problem for the case of solid materials;
- the problem of radiation resistance practically does not arise for CPS;
- tritium accumulation in the target elements of the divertor may be controlled, tritium content may be maintained at a needed level in the circulating liquid metal;
- contrary to gases lithium vapour may be easily condensed; so it is possible to control its flow from the divertor to the main vessel; the condensed liquid metal will come from condensation zone to circulation system and will not be accumulated in the divertor and around it as dust as in case with the solid material divertor;
- low speed of lithium flow and insulating self-healing coatings [29, 32] covering the inner surface of liquid metal loop will significantly reduce MHD effects.

Let us consider the basic physical, technical and technological aspects that make the considered concept physically acceptable and technically feasible.

2. Main properties of lithium CPSs

For low melting metals, lithium has the best physical and thermal properties for application in a liquid metal reactor [29, 31]. Liquid metal divertor CPS operation efficiency will depend on its design, on solid structure wetting properties and on capillary pressure.

CPS serves

- to confine and to redistribute homogeneous protecting lithium film on the target plate;
- to feed the surface with lithium in quantity corresponding to evaporated one—self-regulation of the system;
- to drive lithium circulation in the internal divertor cavity and to link hydraulically evaporating and condensing surfaces with no need of a special pumping system.

Self-regulation is an intrinsic property of CPS. It is based on the dependence of capillary forces on meniscus radius $R(R_{\mathrm{eff}}, Q)$ at the evaporating surface depending, in turn, on pore radius R_{eff} and on incident power flux Q [29, 33]. Capillary pressure is defined by the following expression:

$$P_{\mathrm{c}} = \frac{2\sigma(T)\cos\theta}{R(R_{\mathrm{eff}}, Q)}, \tag{1}$$

where $\sigma(T)$ is lithium surface tension, θ is the edge wetting angle.

Figure 3 gives capillary pressure as a function of CPS pore efficient radius and temperature. CPS is able to confine lithium due to capillary forces at a surface of any configuration and orientation with no cavities in lithium film and in the absence of non-controlled surface flows. Moreover, there is no effect of lithium film separation from the surface and baring of CPS solid structure under plasma impact.

Confinement of lithium in CPS in stationary conditions is defined by the following inequality for all points of evaporation surface:

$$P_{\mathrm{c}} \geqslant \Delta P_{\mathrm{t}} + \Delta P_{\mathrm{f}} + \Delta P_{\mathrm{m}} + \Delta P_{\mathrm{h}} + P_0 + P_{\mathrm{p}}, \tag{2}$$

where ΔP_{t}—liquid–vapour phase transition pressure difference, ΔP_{f}—hydrodynamic pressure loss of lithium flow in CPS, ΔP_{m}—MHD pressure loss in CPS in the magnetic field, ΔP_{h}—hydrostatic pressure drop in CPS, P_0—pressure in the supply system, P_{p}—pressure of incident plasma on the CPS surface.

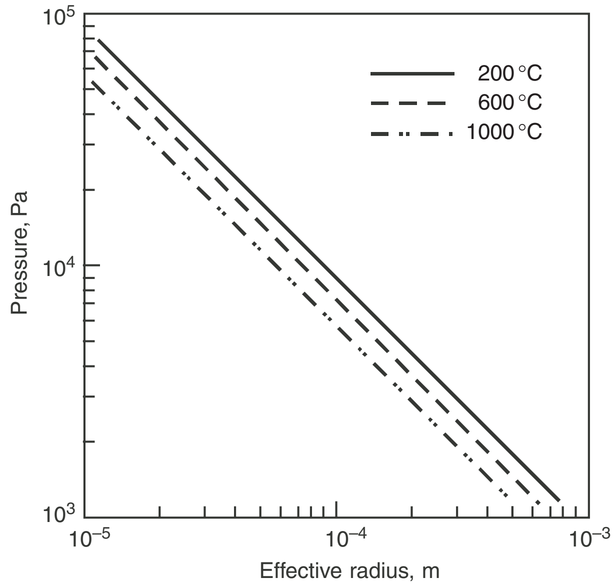

Figure 3. Liquid lithium capillary pressure vs effective pore size.

Different solid materials can serve as CPS basis—metal wire cloth, metal felt, sintered powders etc. The choice of CPS structure material is determined by operating conditions according to the requirements here below:

- low sensibility of mechanical properties to radiation damages,
- low dependence of CPS serviceability on radiation resistance and on mechanical properties,
- high resistance to high temperature gradients,
- ability to keep properties in conditions of partial damage,
- compatibility with liquid metal,
- good wetting with liquid metal,
- acceptable fabrication technology of structure elements.

3. Stability of lithium CPSs under stationary and pulsed power loads

3.1. Experimental study of lithium CPSs in stationary conditions

CPS-based mock-ups of lithium targets have been designed and tested under stationary high power heat load to validate the concept of liquid lithium divertor and to prove the possibility of using CPSs as target elements. Experiments have been performed in the beam-plasma simulator SPRUT-4 [35]. The device has an electron injector with maximal power 40 kW. Electron beam passes along the axis of linear magnetic field (0.2 T) providing power flux of 1–200 MW m^{-2} at the target position. The target response effect to the beam action was evaporation of lithium depending on the surface temperature corresponding to the incident power flux value. The beam energy was partially absorbed in lithium vapour and lithium plasma was generated in front of the target. It was shown that this part of energy was not important in the studied experimental conditions so that power flux on the CPS surface during experiment remained constant and this was important for interpretation of the experimental results.

Balance of lithium lost from the target and collected in condenser and energy balance were studied in the range of 1–25 MW m^{-2}. Thermocouple was placed in the target body close to its surface to evaluate the temperature of the CPS in the hot spot. Different target modifications have been tested [37]. They were equipped with thermocouples to measure distribution of heat flux in the target structure during irradiation. The last modification ensured stabilization of thermal conditions in the target by forced water cooling (figure 4). The heat fraction absorbed by the target during irradiation was removed by water and a calorimetric system was used to measure this value.

The first experiments have shown [34–38, 43] that a simple model of lithium CPS target ($S = 3\,\mathrm{cm}^2$) can carry long-duration heat loads up to 25 MW m^{-2} and short excursions to 50 MW m^{-2}. Further experiments have been carried out with liquid lithium open loop that provided sufficient lithium supply in stable thermal conditions in the target during prolonged irradiation tests. This target modification has been shown to operate in a steady-state regime during a long time period (up to 3 h) at power flux from 1 to 10 MW m^{-2} [36]. Figure 5 shows measured values of lithium loss rate from the CPS target in this range of incident power. Note that CPS surface temperature depended on the incident heat flux and it was in the 350–970°C range.

Lithium plasma generated in front of the target spread along the magnetic field (figure 6) and it was studied by probes, plasma radiation was analysed by spectroscopy. Plasma density and electron temperature were measured to be in 10^{12}–10^{14} cm^{-3} and 1–17 eV range,

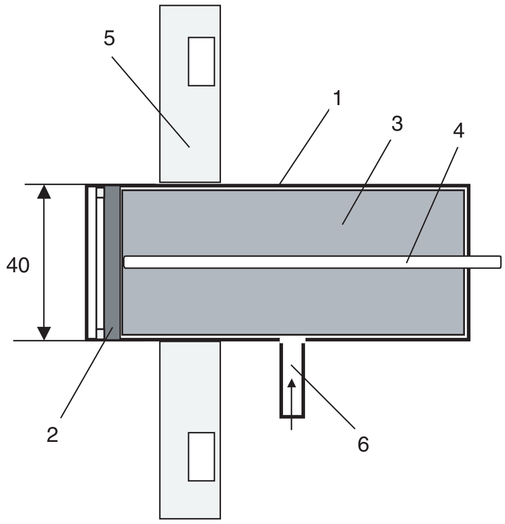

Figure 4. Lithium CPS model for tests on SPRUT-4 beam-plasma device: 1—target casing, 2—CPS ($D = 40$ mm), 3—Li-filled structure, 4—thermocouple, 5—water-cooled flange, 6—lithium inlet.

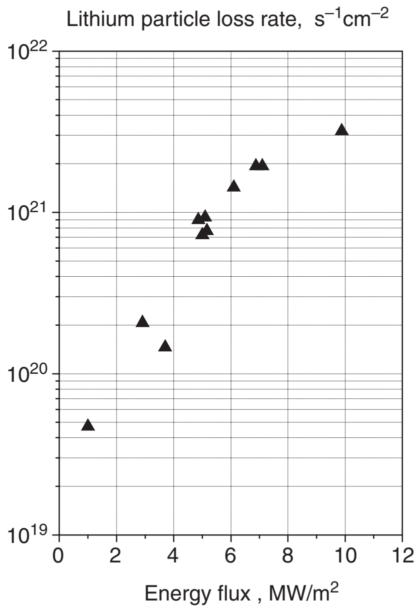

Figure 5. Lithium loss rate from the CPS surface under steady-state power load.

respectively. Higher densities corresponded to lower electron temperatures [35]. Shown in figure 7 is a typical lithium plasma spectrum. Lithium neutral and ion radiation has been identified and no indications of CPS structure material lines (molybdenum) have been found in the studied spectra. These observations give evidence of a low sputtering effect in our conditions.

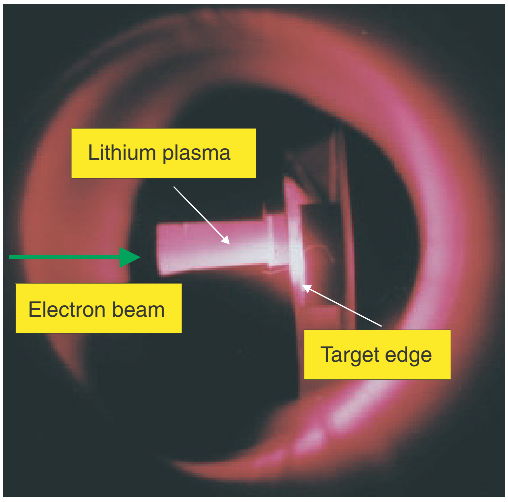

Figure 6. Lithium plasma in front of lithium CPS target.

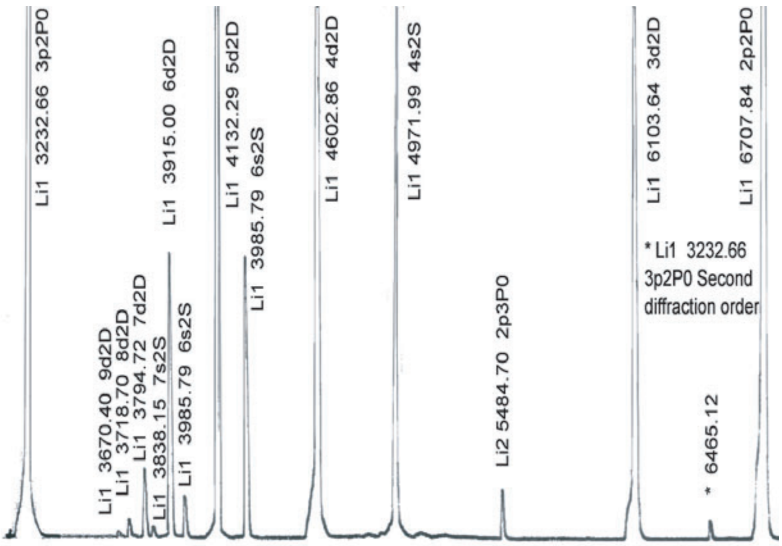

Figure 7. Optical spectrum of plasma radiation at the surface of CPS target.

The following processes related to lithium divertor have been observed and studied in this experiment:

- high rate lithium evaporation was measured, lithium loss at the surface was compensated by lithium feed due to surface tension forces, the ratio of the removed power by evaporation to incident power achieved 0.7,
- lithium vapour was ionized in front of the target, radiation intensity was high in the region ~10 cm from the target,
- part of the heat absorbed in the target was removed by water cooling system,
- diffusion and recombination of lithium plasma occurred as well as vapour condensation on the wall in the beam transport channel, these processes being intensive enough to provide

necessary conditions in the electron gun zone situated upstream for operation without break-down in all tested regimes.

The obtained results confirm that lithium target with CPS surface structure can operate efficiently in stationary conditions at high power loads.

3.2. Disruption simulation experiments

The effects of disruptions in tokamaks have been simulated in quasi-stationary plasma accelerator QSPA by magnetized plasma flows interacting with lithium capillary target [28, 29, 33, 37, 38]. Experimental models of CPS have been designed, manufactured and tested in order to investigate their behaviour at disruptive high heat loads. In principle, the lithium-filled CPS has the capability of absorbing the energy under plasma disruptions without failure. The target was placed in the central part of the magnetic solenoid. The plasma flow parameters were: plasma density $n_e \approx (2–5) \times 10^{16}\,\mathrm{cm}^{-3}$, temperature $T_e + T_i \approx 30\,\mathrm{eV}$, magnetic field in plasma $B \approx 1\,\mathrm{T}$, energy flux $Q = 4–5\,\mathrm{MJ\,m^{-2}}$, pulse duration $\tau = 200–500\,\mu\mathrm{s}$, diameter of plasma flow $d = 40–80\,\mathrm{mm}$, plasma pressure $P \approx 4 \times 10^5\,\mathrm{Pa}$.

Two effects were observed to occur during irradiation: shielding layer formation near the target surface and droplet erosion.

Shielding layer. A dense plasma layer was formed in front of the target during interaction. Iinterferograms of the process have shown that at $5\,\mu\mathrm{s}$ the plasma density reached $n_e = 10^{17}\,\mathrm{cm}^{-3}$. Then it decreased and an opaque layer $\delta \approx 10–15\,\mathrm{mm}$ thick was formed in front of the target. Turbulent processes in this layer may explain it. The curve of plasma density distribution 2 (figure 8) was also measured. Plasma density spatial distribution and the line

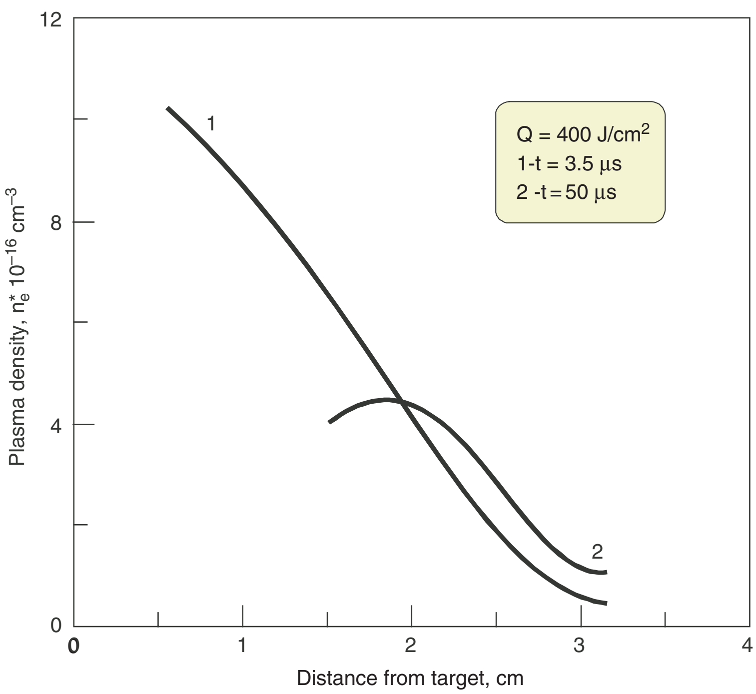

Figure 8. Plasma density distribution near the target.

LiI $\lambda = 61.036$ nm intensity in front of the target were measured in time. The measurement results of neutral lithium layer thickness in front of the target are given in figure 9. One can see that the evaporated neutral lithium appears in about 10 μs after start-up of the plasma interaction with the target at a distance of about 10 mm from it. The layer becomes 40–50 mm thick by the moment $\tau \approx 200\,\mu$s. The pulsation of radiation can be explained by changes in the plasma density, since the intensity of radiation also depends on electron density.

Thus, the experiments show that a dense plasma layer, 10–15 mm thick, $n_e = 10^{17}$ cm^{-3}, is formed in front of the target. The major part of the plasma energy, ~97–99%, is absorbed and radiated in this layer which plays the role of a shielding layer. A small part of liquid lithium was evaporated from the target at every shot. The target itself remained undamaged even after 22 plasma shots (figure 10). In contrast, a special target made of a molybdenum mesh without lithium was destroyed by the plasma flow after a single shot. 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 part of energy coming to limiter achieved 50% of the total flow to the wall [39].

Droplet erosion. Lithium erosion by evaporation in the first 5–10 μs of plasma pulse was just a small fraction of total mass loss (~5–10 μm) including periods of shielding layer development and evolution. Much higher erosion of liquid lithium surface was induced by splashing. Splashing could arise as a result of 'wind waves', Kelvin–Helmholtz hydrodynamic instabilities and volume bubble boiling. Droplet ablation erosion rate was measured for a free lithium surface and it reached ~1–3 mm per pulse at heat fluxes up to 3 GW m^{-2} that agreed well with estimations. For porous structures, the erosion was considerably suppressed by

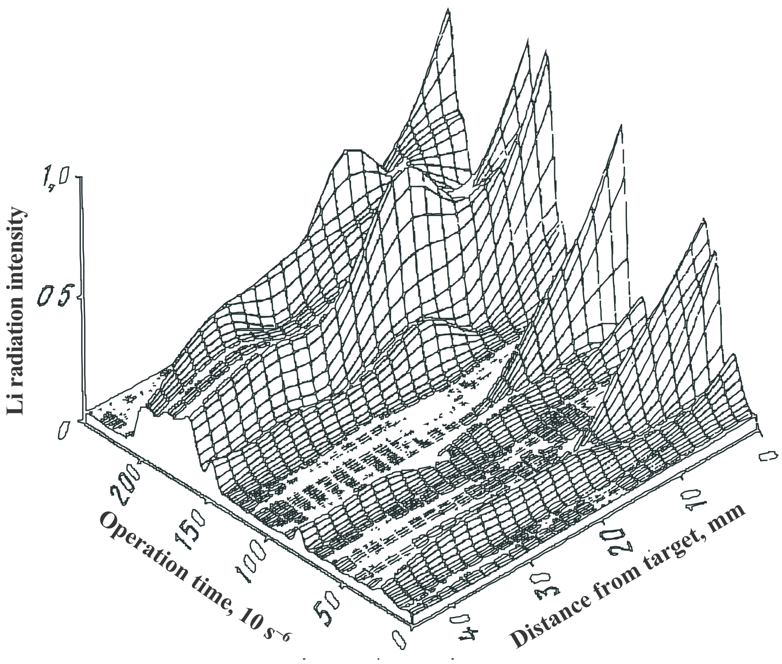

Figure 9. Radiation intensity of neutral lithium as a function of distance from the target and time.

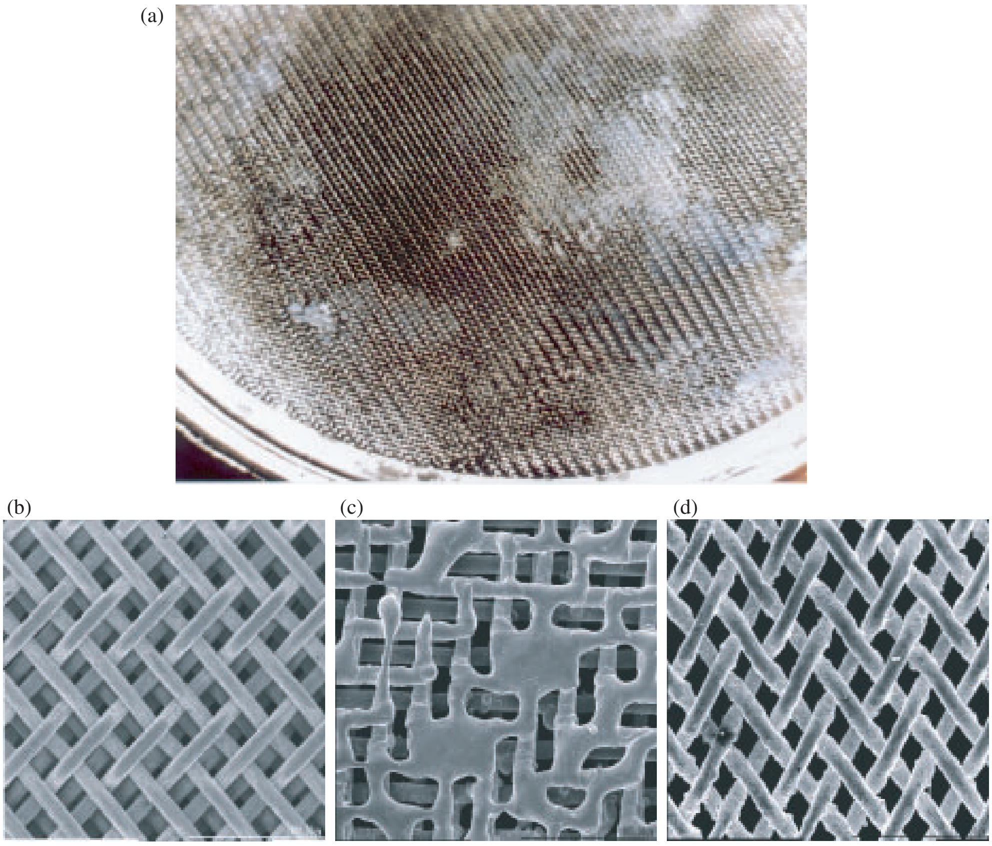

Figure 10. CPS view after hydrogen pulse irradiation at $Q = 4\,\mathrm{MJ\,m^{-2}}$, $t = 250\,\mu\mathrm{s}$. (*a*) lithium-filled CPS general view, 22 pulses; (*b*) initial state; (*c*) CPS without lithium, one pulse; (*d*) lithium filled CPS, 22 pulses.

capillary forces (from 100 to 5 μm for effective pore radius from 200 to 5 μm, figure 11). No structure damage was observed since the lost lithium layer was restored immediately after each shot.

Laser scattering technique was applied to estimate the total amount of droplets and their size distribution. The large size fraction (0.5–1 mm) depended strictly on the CPS parameters (it increased with pore radius) and on the CPS surface orientation with respect to the incident plasma flow (increased with incident angle). The main particle loss was observed in the surface plane. The droplet expansion velocity was 0.1–$10\,\mathrm{m\,s^{-1}}$. CPS with initially solid lithium ($T < T_{\mathrm{melt}}$) showed an increase of erosion rate for increasing number of pulses. This was not the case for CPS with initially liquid lithium ($T > T_{\mathrm{melt}}$). This effect was attributed to wave relief observed on the solid lithium to be formed with increasing number of shots thus causing higher erosion at every next shot compared to initially smooth surface. No waves and relief were formed on the surfaces with $T > T_{\mathrm{melt}}$. This effect proves one of the advantages of CPS with liquid metal in comparison with a solid target.

Ablation erosion may be efficiently suppressed by an optimal choice of CPS parameters (pore radius $\sim$10–100 μm) and the conditions may be obtained when the CPS base material is not eroded, damaged or melted.

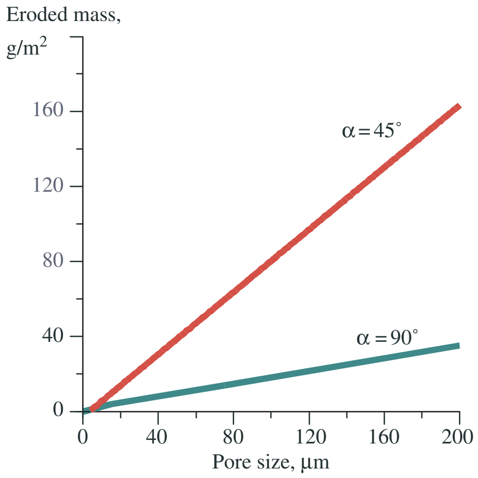

Figure 11. Lithium erosion from CPS surface vs pore size.

These results indicate that lithium CPSs have evident advantage compared to the solid targets because they don't practically loose their mass; their geometrical characteristics and capillary properties are well conserved under the studied experimental conditions.

4. Interaction of plasma with lithium capillary-pore structure in tokamak

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 quasi-stationary conditions expected in reactor. The first task was to ascertain whether spontaneous lithium bursts from the liquid wall to the chamber volume were an important effect or not. Besides, 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.

The performance of the small tokamak device T-11M are the following: $R = 0.7$ m, $a = 0.2$ m, $B_t = 1$ T, plasma current $J_p \approx 100$ kA, discharge pulse duration about 0.1 s. Heat flux to limiter is about 10 MW m^{-2}. The similar power density is expected to be on the ITER divertor plates. Taking into account 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 even some lower (for higher density) in a reactor machine. All negative effects occurring at the wall that are known at present, namely, arcs, emission bursts, sputtering, micro-capillary waves etc are functions of sheath potential and, finally, of T_e. High recycling condition could not be simulated exactly in T-11M. However, this regime corresponds to lower $T_e < 5$ eV that seems preferable for lithium divertor plates. Therefore, we suppose that T-11M modelling experiments were carried out in the conditions close to or even more severe than those of a reactor.

A diagram of the T-11M experiment is presented in figure 12. Movable rail limiter (figure 13) with plasma contacting surface made of lithium CPS (two versions of CPS were studied—with $R_{\mathrm{eff}} = 100$ and 30 μm) was inserted into plasma to about 5 cm thus limiting plasma column aperture and determining plasma current ($q(a) = 3$–4).

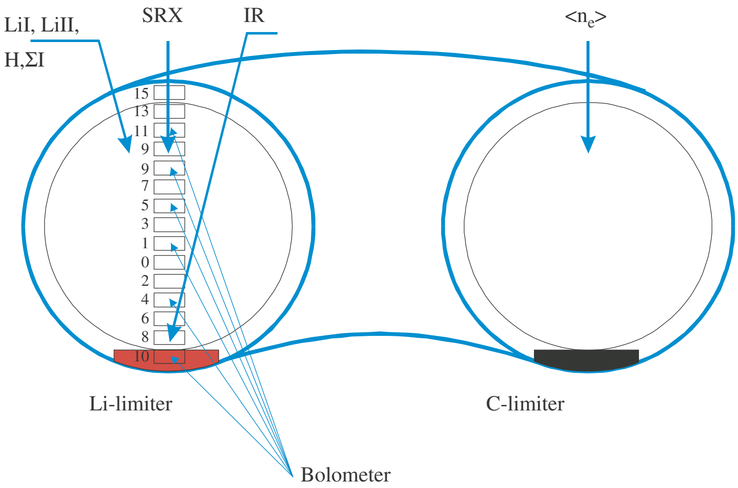

Figure 12. A scheme of the T-11M experiment with lithium limiter.

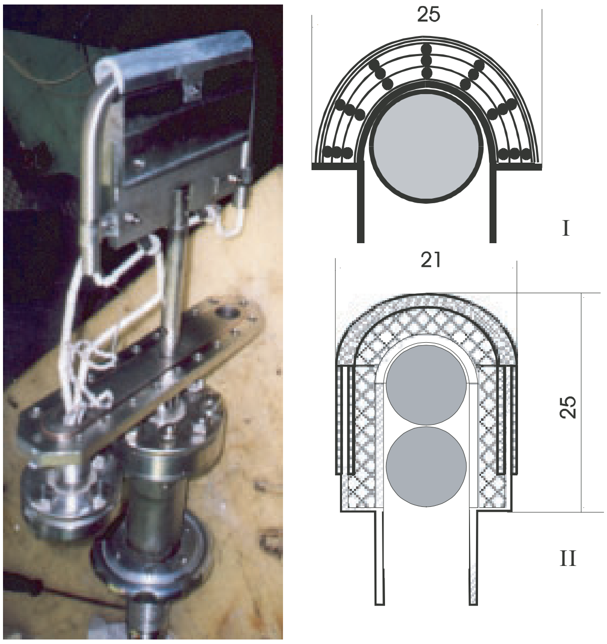

Figure 13. Lithium rail limiter: (*a*) general view; (*b*) CPS lithium limiter (cross section).

The study of the first version showed that induced forces appearing at the limiter edges in disruption conditions were underestimated. As a result, splashing of lithium across the field lines was observed. This effect was suppressed in the second limiter version ($R_{\text{eff}} = 30\,\mu$m) where confinement condition similar to (2) was satisfied for liquid lithium 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 was applied to observe lithium penetration into the plasma. Besides, a 15-channel bolometer system was set up and a special infrared diagnostics was developed to measure the limiter surface temperature during the discharge and to evaluate the deposited power values [26, 27]. The local heat deposit was shown to achieve 10 MW m^{-2} in a quasi-stationary discharge for effective heat pulse length of no more than 50 ms. The corresponding temperature rise was 250˚C. A special heater incorporated in the limiter structure enabled higher temperatures to be obtained (up to 600˚C by preheating).

No catastrophic events leading to lithium injection in MHD stable discharge conditions within the whole lithium temperature range (from 20˚C to 600˚C) have been observed in T-11M experiments and it was the first important result of the work. Lithium and graphite limiters worked practically in a similar way if additional heater was not used [26, 27]. Heating of the lithium limiter gave rise to lithium injection into plasma detected by an increase of lithium line radiation and of integral light emission in the vicinity of the limiter. Temporal dependence of integral light emission from the lithium limiter region for three discharges with different initial limiter temperatures (T_0) is presented in figure 14. It is evident that while T_0 increases lithium flux begins to grow in time. However, estimations of absolute lithium emission have shown [40] that for limiter temperatures $T_0 < 500$˚C it remains in the limits expected for sputtering by D^+ and Li^+ ions with sputtering yield from 0.5 to 1. This is in correlation with the known data on sputtering [41]. The 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. Lithium light peak after the end of the heat pulse may be explained by recombination process (MARFE) because it is not followed by growth of plasma density and cannot be explained by a simple additional lithium influx to the plasma. For temperatures higher than $T_0 \approx 500$˚C, evaporation appears to become the main channel of lithium emission.

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,

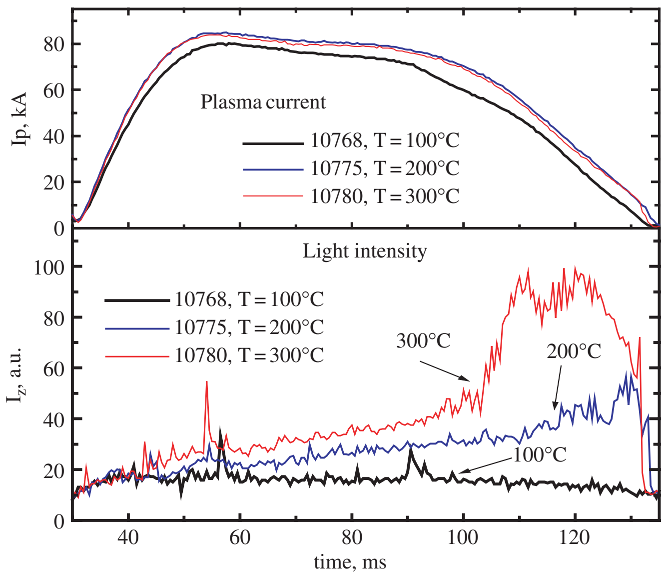

Figure 14. Time dependence of integral light emission from the lithium limiter region.

by this, to reduce the heat load to the limiter. It was really reduced by approximately a factor of two by these manipulations in the helium discharge [26, 27] (figure 15). 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 centre in actually operating tokamaks [44] and to radiating divertor in a reactor may be done. Lithium confinement time in the periphery layer (τ) may be taken as a governing parameter. If it is small then lithium will not reach equilibrium ionization condition. In this case, radiation intensity becomes much higher than that expected for coronal equilibrium. Figure 16 [26] shows calculated evaluations of such

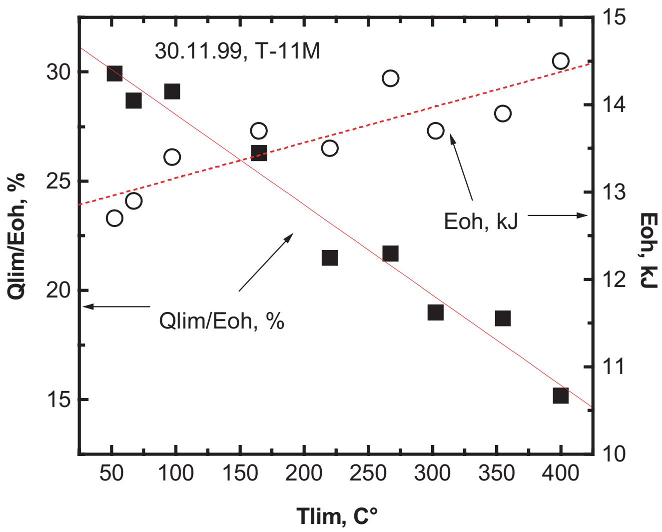

Figure 15. Relative energy coming to lithium limiter $Q_{\mathrm{lim}}/E_{\mathrm{oh}}$ vs its initial temperature in ohmic mode.

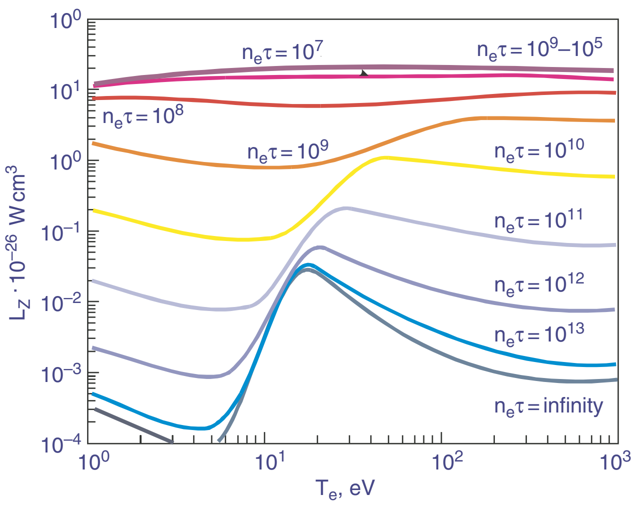

Figure 16. Lithium radiation evaluation data.

radiation (per atom and per electron) as a function of T_e and of $n_e\tau$ [cm^{-3} s] which is a factor characterizing deviation from the equilibrium conditions. Lithium radiation may be by two orders higher than the equilibrium one for $T_e = 30$ eV, $n_e = 10^{13}$ cm^{-3} and $\tau = 10^{-3}$ s that is quite realistic for plasma periphery. Radiating mantle also appears realistic in these conditions. Unfortunately, there are no reliable methods of plasma confinement control at the plasma boundary. Principle possibilities of such a control are just known: ergodic magnetic fields at the boundary, controlled ELMs, local excitation of MHD activity etc. Further development of these methods is needed.

5. Tritium 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 [23, 40, 45]. Moreover, helium sorption was discovered in T-11M experiments (figure 17) as well [40] with a slow desorption during 20–100 s after the discharge (figure 18). 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 figure 19. 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 the lithium. It means that a simple heating to 370–500˚C 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.

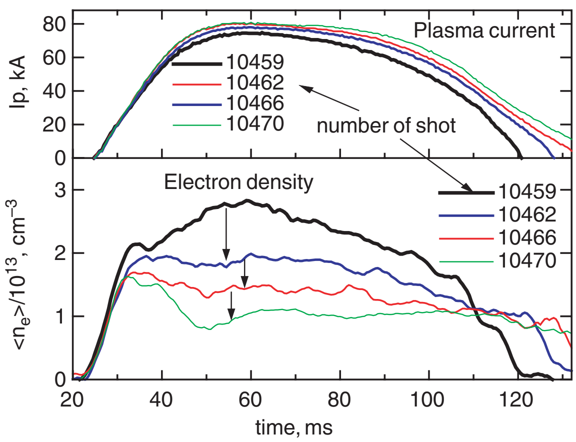

Figure 17. Electron density evolution from shot to shot in discharges with high initial temperature at lithium limiter (about 500˚C); helium taken as working gas.

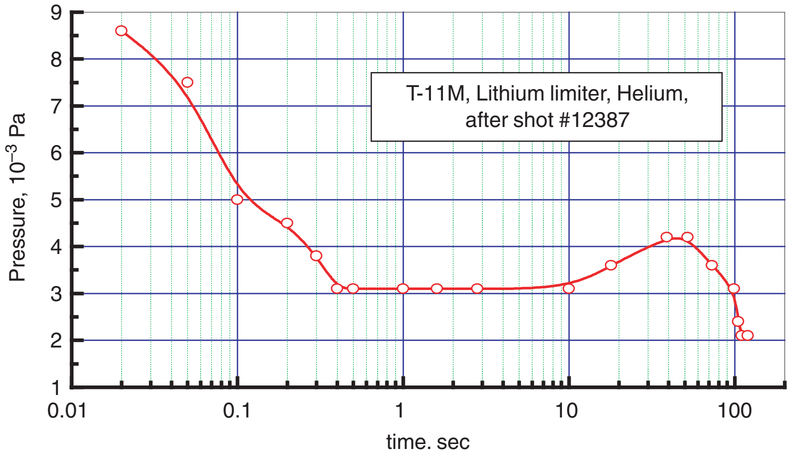

Figure 18. Behaviour of helium pressure in the tokamak chamber after discharge.

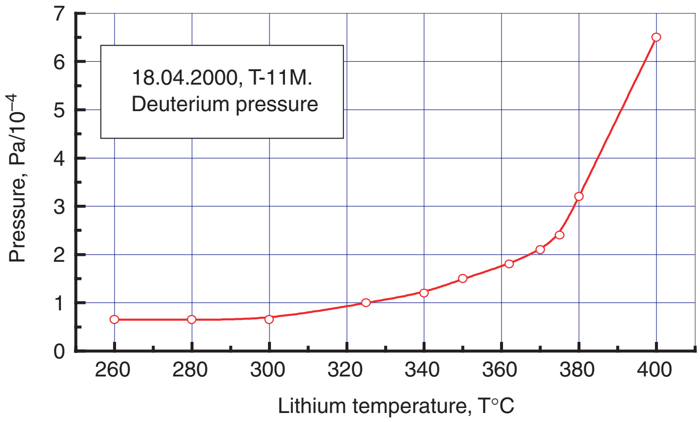

Figure 19. Deuterium pressure in the tokamak chamber during heating of lithium limiter.

6. Lithium divertor general scheme

Figure 20 shows a schematic view of lithium divertor on CPS basis. Reactor has two independent heat transfer circuits. Internal lithium circuit covers the divertor plates and the vessel wall with CPSs. External circuit of heat transfer from the wall and divertor plates to energy conversion system may be also lithium-filled. The choice of cooling fluid will be determined by a required temperature interval and by chemical compatibility with lithium to ensure safe operation of the reactor. Lithium cooling is the best at wall temperatures higher than 350˚C. At lower temperatures Na–K eutectics may be used (solidification point -11˚C) as well as organic coolants (e.g. diphenil mixture with solidification point 12˚C) etc.

Different operation conditions are possible in the divertor. For a reactor of the ITER-FEAT size, the total needed lithium circulation rate (evaporation–condensation) will be about 5 kg s^{-1} of lithium to remove 100 MW. A radiating lithium mantle may be realized to relieve the heat load of target plates. In this case, a part of the heat from the plasma will be radiated to the wall (about 0.15–0.20 MW m^{-2}) and then removed by the cooling system. Then divertor target

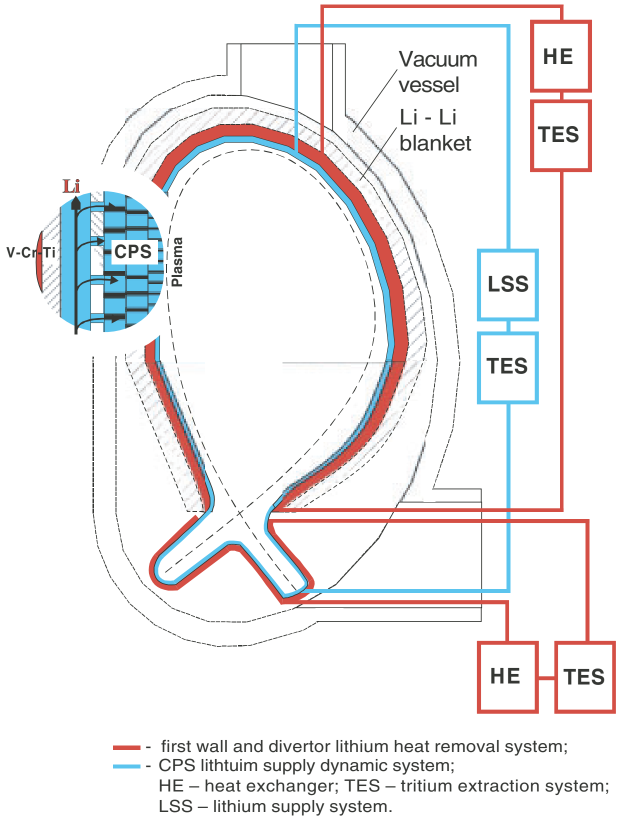

Figure 20. Schematic view of fusion reactor with lithium divertor on CPS basis.

plates will play the role of lithium supply in a needed quantity to the SOL. The wall temperature will be 250–350˚C in this condition. If higher temperatures are adopted at the level of 400˚C then lithium evaporation from the wall will exceed its flow from plasma, therefore, lithium circulation will change direction.

An optional scheme of liquid lithium divertor design for DEMO-S reactor [4] is given in figure 21 [42]. Each sector has two independent circulation (cooling) systems. The first one is formed by a capillary-pore structure which covers the divertor target and evaporates lithium. The vapour interacts with the plasma flow coming to the divertor volume and reaches the cooler zone. Lithium is condensed on its surface and cooled. Via a closed CPS, it is further returned to the evaporating panels. Estimation shows that the area of condensation surface should be 20 times as large as the evaporation one, to enable effective lithium vapour condensation and to minimize its penetration to the zone of the main plasma. It is capillary pressure that drives circulation in this system. This circuit is equipped with its own supply system. It serves to vary the pressure in the system, to make up for a lithium loss and to purify lithium from deuterium, tritium and other impurities including those of corrosion–erosion origin.

The second system contains cooler-condenser piping channels and it is meant for heat removal from the divertor (figure 21). Liquid lithium coolant in cooler-condenser is provided.

An optimal choice of liquid metal divertor operation condition will be finally made based on experimental research. Important is that optional conditions are technically feasible though ample physical research and design studies are certainly needed.

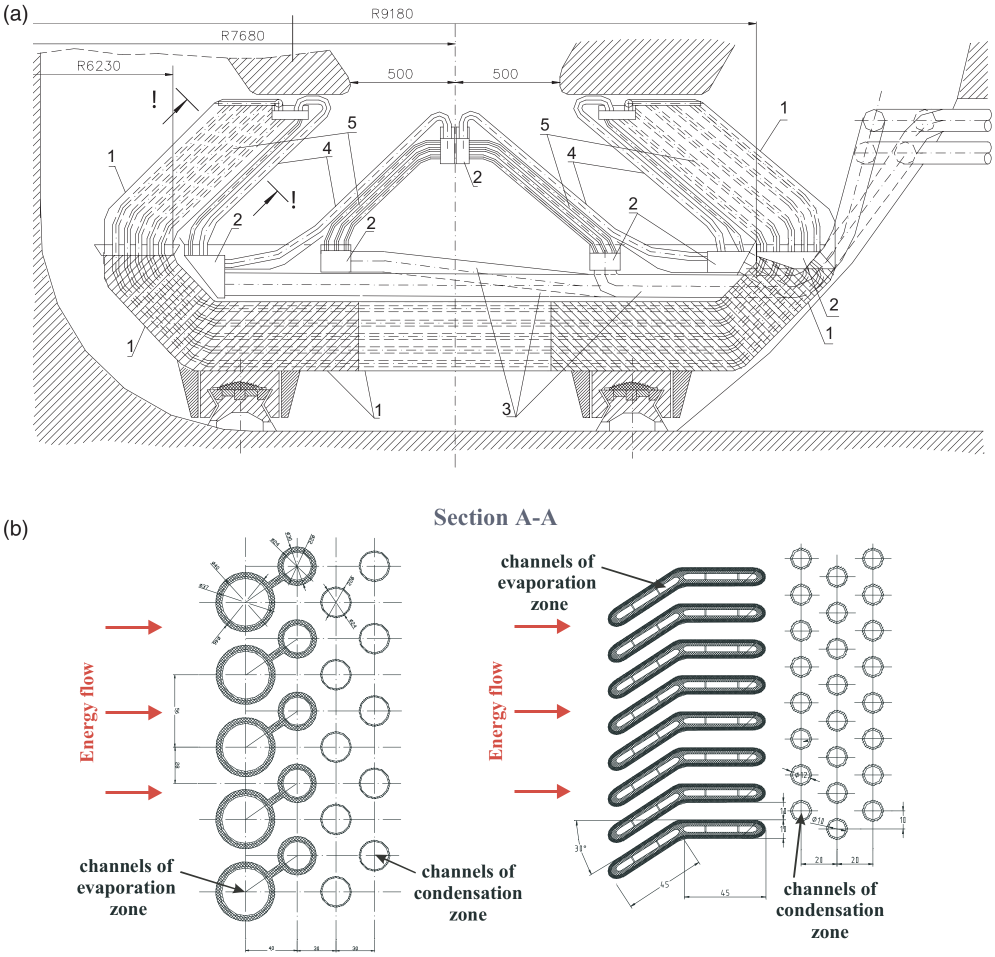

Figure 21. DEMO-S lithium divertor design option. (*a*) cross section (1: radiation shield, 2: collectors, 3: inlet piping, 4: evaporation zone, 5: condensation zone) (*b*) section A-A.

7. Conclusion

The use of liquid lithium as a divertor target material in a tokamak has a high potential. First, note that lithium injected into plasma periphery even in considerable quantity does not cause catastrophic consequences for plasma column. On the contrary, the experience of lithium injection into the hot plasma (TFTR) showed that it is favourable for plasma confinement and contributes to the decrease of Z_{eff} in the plasma core to reactor level of 1.2–1.5. The main effect of lithium injection in TFTR and other tokamaks (T-11M, CDX-U) was the rise of the first wall getter properties, i.e. reduction of the working gas recycling. The total impurity flow to the plasma core also fell. These results form a convincing basis to continue the use of lithium.

The surface tension forces may be used to form a free liquid metal surface with CPSs thus solving the problem of ponderomitive forces applied to the vessel and divertor surface layers, i.e. the problem of sufficient confinement of liquid metal and regeneration of its surface in contact with plasma.

A complex of experimental studies has been carried out on the research of CPS having different characteristics (effective pore radius, material etc) under high stationary (up to 25 MW m^{-2}) and under high power plasma pulses. Their performance without failure has been demonstrated in these experiments. Proceeding from these results, a long service life of CPS-based wall elements in stationary and disruption conditions may be predicted.

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. High energy loss by 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 of desorption temperature was shown to exist for hydrogen isotopes (320–350˚C) and helium (50–100˚C). These effects may be used to minimize tritium capture at higher wall temperatures. On the other hand, separation of helium and hydrogen isotopes is possible in lithium circuit for lower wall temperatures, tritium content may be maintained at a minimal level in reactor systems. It is difficult to do the same in the case of solid plasma facing materials—tungsten, beryllium and CFCs.

A number of unique properties of lithium determines its high potential for application for heat removal at the plasma–wall boundary:

- high latent heat of evaporation allows to redistribute efficiently the energy coming into divertor by evaporation–condensation processes,
- lithium flow from divertor region to the main vessel may be controlled by an appropriate divertor design due to efficient condensation (unlike gases),
- radiative emissivity of lithium may achieve 1000–1500 eV per one Li atom at T_e = 20–50 eV according to estimations of stepwise ionization process and it may be used for protection of the wall and divertor plates by radiation in disruption events and in normal conditions.

Wide range of operation temperatures may be reached in principle in lithium divertor (∼200˚C and higher) satisfying safety conditions by an appropriate choice of fluid in the systems of heat removal from CPS. These may be lithium, sodium–potassium eutectics, organic coolants etc. An extensive experience of safe operation and maintenance of sodium systems has been accumulated for the last decades of successful service of large fast nuclear reactors (BN-350, BN-600, Phoenix, Super-Phoenix).

CPS-based liquid lithium divertor appears feasible based on the experimental, calculation and design studies and the present day technological experience. 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 progress of the considered approach needs further experimental, calculation, design research and technological developments. The following studies seem necessary:

- calculations of lithium migration and radiation in divertor region and SOL on the basis of existing codes,
- experimental work with lithium in divertor tokamaks,

- lithium experiments in linear plasma simulators,
- extension of data base on lithium interactions,
- development of experimental devices to model lithium behaviour in the divertor including evaporation, condensation, ionization processes etc.

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