Lithium experiment in tokamak T-11M and concept of limiter tokamak-reactor

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Abstract. The paper suggests the addition or replacement of the magnetic divertor by lithium limiter as producer of irradiated blanket, which should prevent tokamak first wall and divertor plates from local high power loads. The main physical basis of this idea is lithium localization (screening) close plasma boundary and poor penetration into plasma center in process of its puffing, as was shown in tokamak experiments. The mushroom tokamak lithium limiter with capillary porous system (CPS) can be used for this aim. The hot part of this limiter can work as lithium emitter to tokamak plasma and cold parts can work as collectors of lithium flowing out of the plasma column. The surface capillary forces of limiter CPS will return the collected liquid lithium from cold to hot parts of limiter again and close the lithium circulation "limiter-plasma-limiter". The bulk of the power flowing out of the plasma core can be dissipated by non-coronal lithium radiation in the blanket and scrape-off layer. Such scheme was established partly in T-11M experiments with CPS rail limiters. In paper the main results of lithium behavior and its control in T-11M are discussed.

1. Introduction

An application of magnetic divertor to the tokamak configuration is well-known tool for the suppression of impurity efflux from the first wall of vacuum vessel into the plasma core due to the plasma-surface interaction (PSI). The impurity atoms in the scrape-off layer (SOL) are ionized by the electron impact, and return along the SOL magnetic lines to the wall (limiter or divertor plates) or diffuse into the plasma column.

The main function of the divertor is to enlarge the SOL thickness for effective capture of the impurity efflux from the wall, and its redirection into the divertor SOL. The "mushroom-type" limiters (Fig.1) also are able to perform the same function. Main advantage of such limiters is ability to catch the impurity flux from wall by "mushroom leg" area. Its disadvantage is an open PSI region at the top ("mushrooms hat") with high power load.

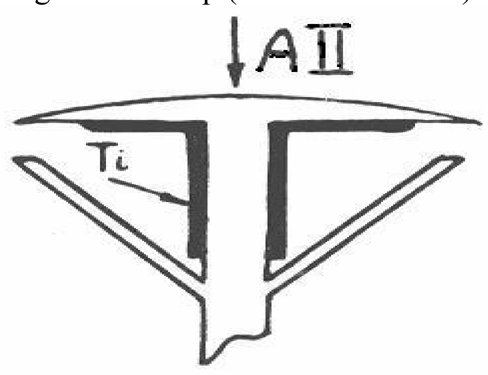

FIG.1. Scheme of "mushroom limiter" [1], AII array is direction to "hat" from the hot plasma.

The sputtered or evaporated impurity atoms have an opportunity of direct penetration into the plasma. Relatively low ionization potential and recombination rate of multi-ionized atoms in the low density plasma result in their localization at the plasma boundary and subsequent diffusion towards the hot center. The last process is enhanced by classic (neoclassic) accumulation phenomena resulted from the "friction" between the hydrogen isotope ions and the impurities with Z>1 (Braginsky's effect). This effect, in particular, explains the increase of Z_{eff} value in the plasma axis region after noble gas puffing (with Z>1), which is used sometimes for cooling of the peripheral plasma and reduction of the divertor heat loading.

The Z-decrease of tokamak PFC materials from Mo and W to C,B and Be was the main tendency of tokamak PFC development from 70^{th}. Next step in this direction should be Li with Z=3. Most serious arguments contra the lithium are: the high tritium retention and liquid lithium splashing. However the recent experiments, which were performed in Russia, USA and Japan [4], showed that the injected hydrogen releases the lithium target after its heating up to 400-500^{0}C only and for suppress of lithium splashing can be used the porous metals (Mo, SS or W) filled by lithium, so called capillary porous system (CPS) [2]. This idea was successfully tested in T-11M [3,4,5]. Ten years of successful use of lithium as tokamak PFC material (TFTR, T-11M, CDXU, FTU) allow to make some summary of lithium properties, as eventual material of first wall, limiters or divertor targets of tokamak-reactor.

2. Lithium properties as tokamak PFC material

2.1. Lithium erosion

Fig.2 presents the principal scheme of use Li CPS rail limiter in tokamak T-11M. The cooled horizontal rod (rail limiter), coated by thin (1-2mm) CPS with lithium, touches the plasma column. The CPS edges are plunged into the liquid Li reservoir for the refilling of evaporated Li. The plasma contact area of the limiter (hot spot) is the main source of the Li atom influx (Li emitter) into plasma. Sputtered and evaporated Li atoms are ionized and excited by electron impact and are diffused as ions (Li^+, Li^{++}, Li^{+++}) into scrape-off layer (SOL) and hot plasma column. The main part of the outward lithium ions flux can go back to the cold end of Li rod and can be collect by him (Li collector). The T-11M experiments show, that capillary forces can return the deposited lithium to hot spot again. As result the closed Li circuit created on the boundary of plasma core. The intensity of LiI, LiII, LiIII spectral lines and total visible light ΣI was observed. Particularly, LiI intensity was selected as indicator of lithium emission (erosion) during discharge [3].

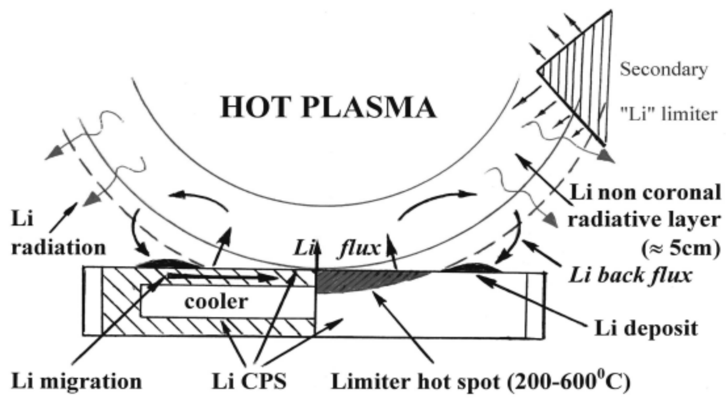

FIG. 2. Scheme of Li-rail limiter interaction with plasma [5].

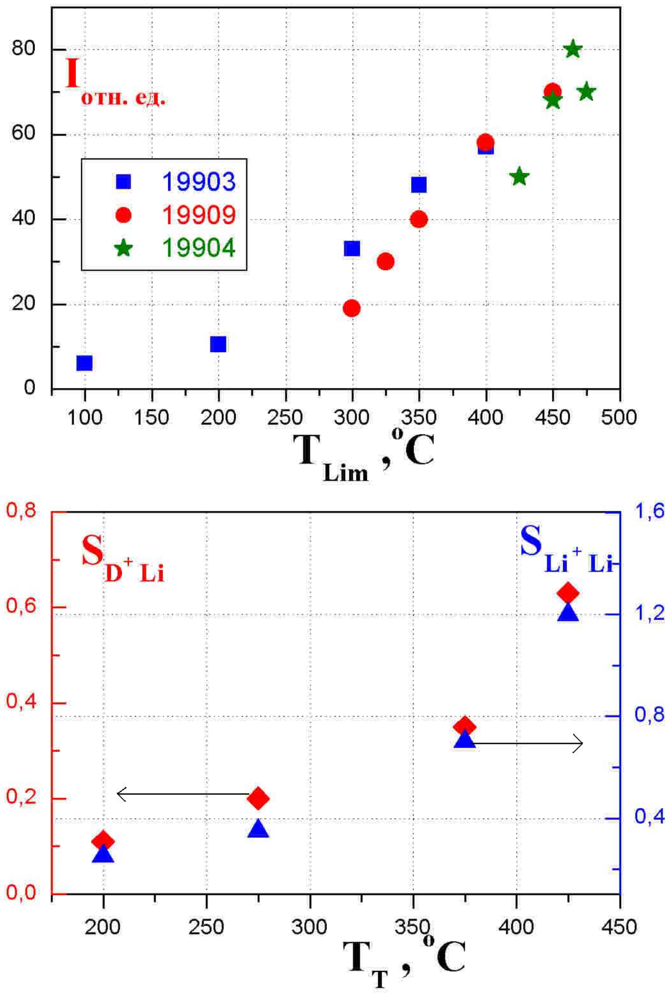

FIG.3. I – intensity of Li(I) emission (arb..un.) as function of limiter temperature and sputtering coefficients of liquid lithium as function of target temperature T_T.

No catastrophic events leading to spontaneous lithium injection from the Li-limiter in temperature range from 20°C to 600°C have been observed in T-11M. Preheating of the Li-limiter gave rise to lithium emission in the limiter vicinity. Electrical biasing experiment showed, that lithium emission depends weakly on energy of bombardment ions, but reveals a clear visible increase with limiter temperature [4].

Fig.3 presents the results of lithium influx measurements for three different shots of T-11M with different level of limiter preheating as a function of limiter surface temperature T_{Lim} and sputtering data [6] of liquid lithium target by $D^+$and Li^+ as function temperature T_T.

The similarity of both curves allows assumption, that the main mechanism of Li-limiter erosion in 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 $T_0 \approx 500$°C Li-evaporation appears as the main mechanism of lithium emission, which has approximately exponential increase with T_{Lim}. That means the best surface temperatures T_s for Li PFC tokamak-reactor to be 300-500°C, until total Li influx is not too high and increases with T_s. This nonlinear dependence is ground of the negative feed back between T_s and local energy flux to the first wall.

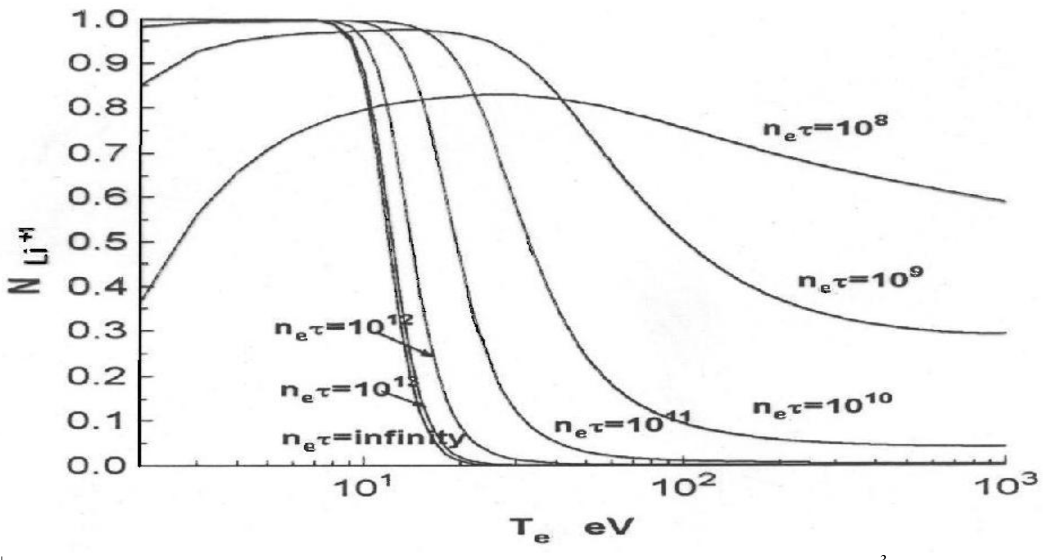

FIG.4. Li^+ fraction as function of lithium confinement time τ, plasma density (in cm^{-3}.) and electron temperature Te.

2.2 Lithium screening

The crucial issue for lithium use in tokamaks is its contamination of plasma center. The main surprise of all previous experiments with lithium injection to tokamaks (TFTR, T-11M, FTU, CDXU) was the poor lithium penetration to hot plasma core (lithium screening).
In all lithium experiments Zeff(0) is close 1. This visible mechanism of lithium screening is not clear today. Very high second ionization potential (~75.6 eV) and relatively low first one (~5.4 eV) are an outstanding property of lithium in comparison to other impurities. If we take to account the finite time τ of lithium existence close the plasma boundary (typically $\approx$ 1ms) we can estimate its ionization state [3]. The results of such mathematical modelling of lithium behavior are presented in Fig.4. For $\tau\approx$1ms and $n_e=10^{19}m^{-3}$ lithium should have mostly a singly ionized state at the plasma periphery with typical $T_e\approx$15…20 eV, and therefore it is much less subjected to the Braginsky's effect. Perhaps, this lithium property is one of reasons of low lithium penetration into the plasma core after peripheral injection. Maybe the lithium screening is results of high gas puffing, which accompanies all lithium experiments. That is the important subject of future investigations.

2.3. Lithium irradiative blanket

Lithium impurity localized at the plasma periphery provides so-called irradiative blanket around the plasma. If the average Li ion lifetime τ in plasma boundary region is much less than the time of lithium transition to ionization equilibrium state (coronal equilibrium) the intensity of lithium radiation (non coronal radiation) could be 100-1000 times higher in comparison to the equilibrium (coronal) level. The level of the blanket radiation could be controlled by the variation of τ [3]. For example, in T-11M experiment the Li impurity at the plasma periphery provided the re-radiation level up to 80% of total heat power in Ohmic mode (~130 kW for 0.2 s plasma discharge). The typical spatial distributions of plasma radiation for discharges with Li and C limiters are shown in Fig.5 [5]. In contrast with C case the main radiation in Li case was localized in thin (5cm) layer close plasma boundary.

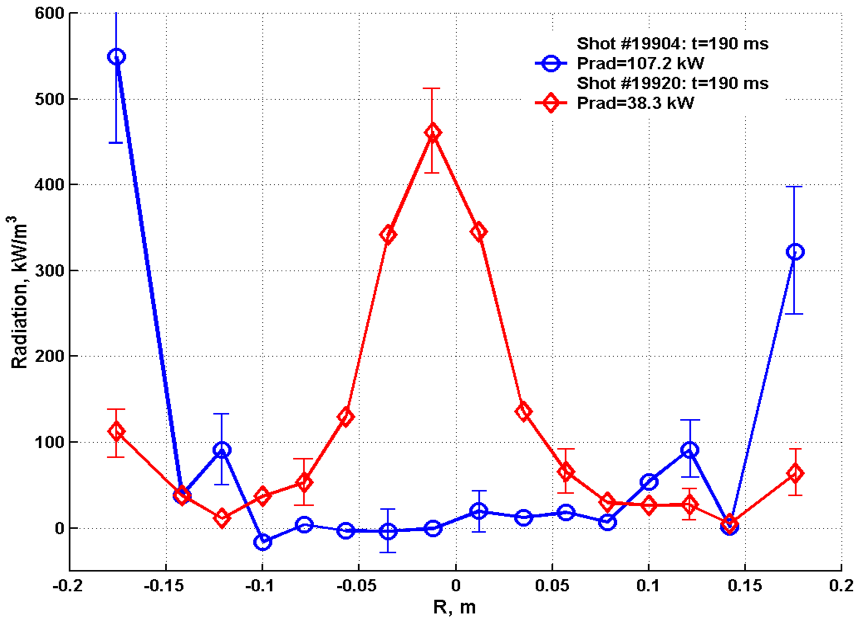

FIG.5. T-11M. Radiative loss profiles at the moment t = 150 ms for two similar C-limiter and Li limiter shots. The average plasma density $<n_e> \sim 2.2\cdot10^{19}m^{-3}$ for both cases. Diamond (red) points-C limiter, circle (blue) points -Li limiter [5].

The lithium spectral line radiation (LiI, LiII, LiIII) and SXR from plasma center showed the quasi steady state character and very low impurity concentration at the plasma axis ($Z_{eff} \approx 1$). The calculation shows, that for transformation of ITER total heat out flux (100 MW) to Li radiation it would be enough to have the lithium radiating layer thickness ~10^{-1}m, n_e=2x10^{19} m^{-3}, $n_{Li}=10^{19}$ m^{-3} and $\tau = 10^{-3}$s. In T-11M experiments τ of Li ions near boundary might be equal 1-3x10^{-3}s based on our estimations.

2.4 Electrical biasing of Li-limiters

One of the most critical point of radiation cooling of plasma boundary is control of Li ion lifetime in irradiative blanket (τ control) for dissipation of the power flowing out of the plasma. In the conditions of constant power flow, the greater is τ, the more should be lithium density close a plasma periphery and increase of lithium influx to plasma core. The well known method of magnetic field line ergodization [7] as method of τ (and lithium density) decrease can be used. But the special helical coils in or close plasma chamber will be needed for its realization. Unfortunately it can't be used really in reactor condition. The helical coils can be replaced by currents, induced in plasma periphery between lithium limiters. Main objection to this suggestion was misgiving of high lithium out flux to plasma during such operation. The test experiment was performed in T-11M. The pulse voltage U_{Li} (≤300V, 30ms) was applied between Li and Mo limiters. The Mo rail limiter (d=0.2m) was situated in opposite of Li port of T-11M ($\Delta\varphi=180^0,\Delta\theta=180^0$). The saturated electrical current equal to 100A was measured between both rail limiters during voltage pulse. The indicator of lithium emission from limiter - Li(I) intensity – was almost non sensitive to 100A current pulse. The alteration of applied voltage polarity does not change remarkably the current and lithium emission too. That means the electrical biasing of lithium limiters can be used for generation of such helical currents in tokamak chamber to control plasma periphery.

2.5 Hydrogen and Helium retention and removal

The common feature of all Li tokamak experiments is very low hydrogen recycling and a high level of gas puffing, which is needed for control of steady state plasma density.

The main reason of recycling decrease is the high probability of hydrogen and deuterium retention on the lithium covered vessel wall. Moreover, helium retention was discovered in T-11M experiment as well [3,4], but with a slow removal from lithium covered vessel wall (during 20-100s after shots). In order to avoid helium retention it was sufficient to heat the T-11M vessel wall to 50-100^{o}C. For deuterium even highest attainable wall temperature 250-300^{o}C turned out to be insufficient. However the Li-limiter heating after plasma experiments showed start of deuterium emission from lithium at temperatures higher than 320^{o}C with fast increase up to 450^{0}C. As it was shown later in USA [8] and Japan [9] the direct heating of lithium target to 400-500^{o}C after its bombardment by deuterium ions seems to be sufficient to remove all deuterium and, probably, tritium also. Lithium hydrides are supposed to be decomposed at temperatures higher than 600^{o}C. Therefore, one may conclude that considerable part of deuterium wasn't captured by lithium in the form of hydrides (deuterides).

3. The CPS lithium limiter as alternative or addition of magnetic divertor

The lithium screening, erosion and sorption dynamics can be used for design of pumped ("mushroom") CPS lithium limiter, which can carry out the main functions of tokamak magnetic divertor: the energy exhaust by non coronal lithium radiation to chamber wall and He ash removal due to different sorption properties of D,T and He ions in process of their interaction with liquid lithium.

Fig.6 presents the one of ideas of such kind of lithium limiter placed in ITER divertor SOL. It might be a mushroom kind limiter [5], covered by a thin (<1cm) Li CPS. The Li "wick" should connect the hot "hat" (>550^{o}C) of mushroom and its cold (<350^{o}C) down part- "leg", like as "hot spot" and cold limiter ends in Fig.2. That means the top part of the limiter plays the role of lithium emitter and down – collector of lithium and hydrogen isotopes. The vertical temperature gradient can be controlled by cooling of limiter top and down parts.

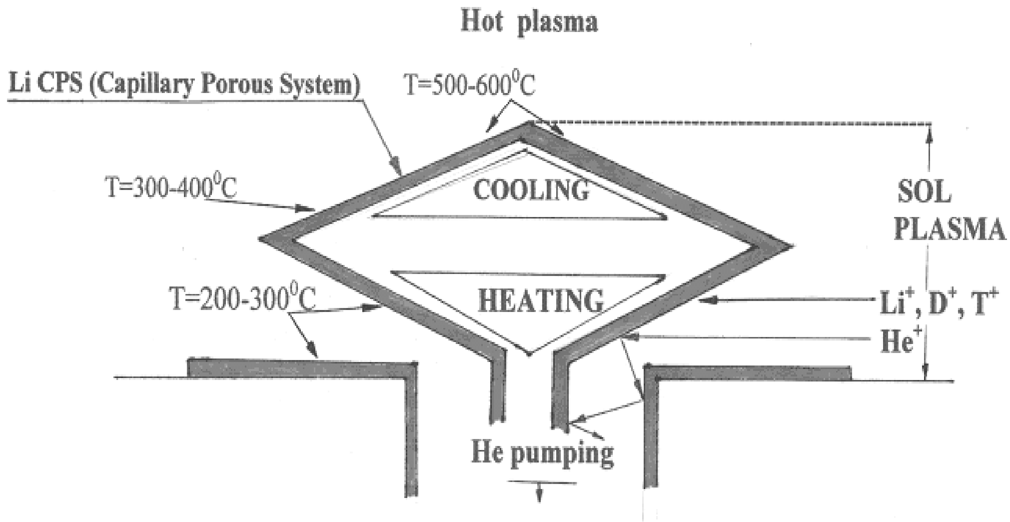

FIG.6. The principal scheme of the CPS Li-mushroom limiter[5].

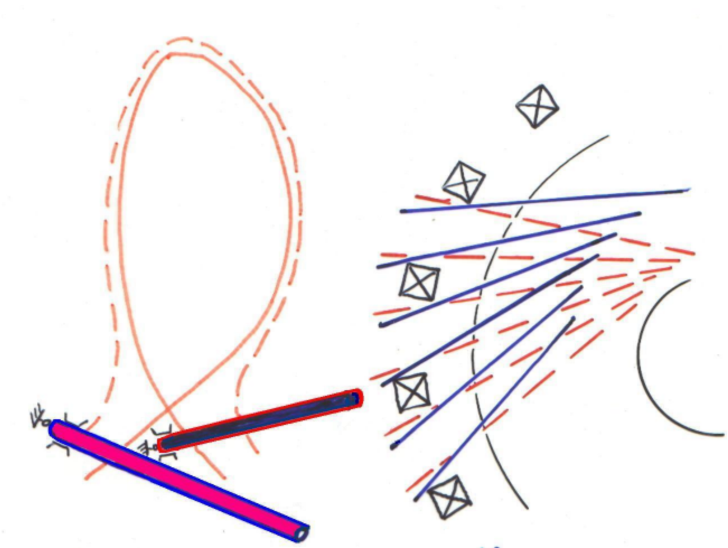

FIG.7 Schema of the rod divertor targets

Such limiter can work, for example, as transformer of ITER SOL power flux to the radiation power. It should decrease the divertor power load and increase the power load on tokamak vessel wall. Our estimations show that total radiation efficiency of this type of Li-limiter in ITER experiment can be more than 20MW.

If the temperature of down part of mushroom limiter will be higher than 100^{o}C it should work as a reflector and compressor for He ash – result of fusion reaction. The compression degree of such pumped limiter can be equal to100 and simple pumping of limiter port can allow to avoid significant part of the He ash from reactor chamber. That can be an example of the lithium use for segregation of ash and fuel in ITER and DEMO.

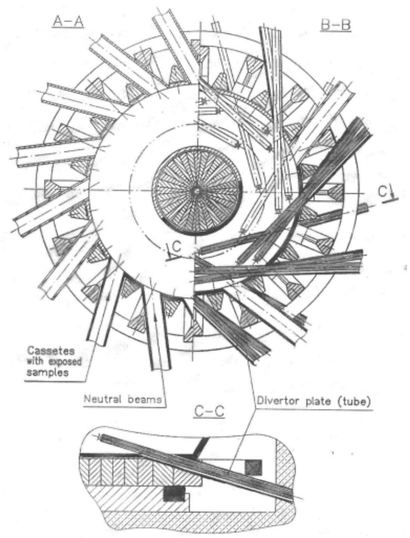

FIG.8Plan of divertor system of VNS [10]

The different idea to use the lithium CPS as additional divertor targets is presented in Fig.7. The CPS lithium divertor targets (rods) like as rail limiter, showed in Fig.2, can be inputted in several ports of divertor chamber to cross the separatrix. Obviously, the hot part of these rods, which should develop in crossing area, will work as lithium emitters and cold parts as its collector. As results we have possibility to close again the lithium circuit: “hot spot”-divertor plasma- cold ends of roads-CPS-“hot spot”. The non coronal radiation of lithium ions during its travelling in divertor and SOL plasma should work as its cooler and smoother of divertor energy loads to first wall of reactor. In condition of intensive plasma boundary cooling we can hope to decrease the energy flux to rods up to low magnitude, which can be removed by conventional cooling.
As an example of the rod divertor targets design in Fig.8 is presented the view of one of versions of Volumetric Neutron Source tokamak [10].

Conclusions

The main lithium properties as material of tokamak PFC, which were offered in tokamak experiments: lithium screening, erosion and sorption dynamics, can be used for design of pumped (“mushroom”) CPS lithium limiter, which can carry out the main functions of tokamak magnetic divertor.

Lithium emitter-collector scheme allows the producing in tokamak boundary plasma the closed lithium circuit, which can play role of plasma cooling by lithium non-coronal radiation. This can solve the problem of high power load of tokamak divertor plates and limiters.

Liquid lithium sorption properties can be used for segregation of ash and fuel in ITER and DEMO.

The concept of tokamak-reactor with CPS lithium limiter or divertor targets can be tested in ITER experiment.

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