Development and Experimental Study of Lithium Based Plasma Facing Elements for Fusion Reactor Application

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Abstract. Results of experimental study of lithium capillary-porous systems as the plasma facing material in tokomak conditions are considered. Impressive effect of lithium application in plasma performance improvement for domestic (T-11M, T-10) and foreign (FTU, NSTX, LTX, HT-7) fusion installations are observed. New results and plans on lithium experiments in FTU, KTM, T-11M tokamaks and stellarator TJ-II are presented.

1. Introduction

Lithium capillary-porous system (CPS) is the key element providing implementation of lithium fusion reactor concept [1]. The promising results on lithium experiments in T-11M [2], FTU [3], T-10 [4], have clearly shown that the compatibility of the lithium CPS with the plasma under real tokamak conditions, the capacity of the CPS to self-regenerate the exposed surface and to confine lithium during normal plasma operations and at disruptions, and the capability to withstand high heat loads without damage to the CPS surface have been confirmed. A strong pumping capability of the wall, low values of Z_{eff} and radiation losses lead to better parameters of the tokamaks plasma. Proceeding from this fact, the lithium CPS can be considered as a new advanced material for the plasma-facing elements (PFE) of tokamak reactors, which allows all the advantages of liquid metals to be realized and the engineering problems of their application to be overcome. The results in plasma parameters improvement are correspond to TJ-II [5], HT-7 [6], NSTX, LTX, CPX-U experimental studies of lithium effect. New liquid lithium plasma facing components being under development for effective impurity and particle control in a tokamaks KTM (divertor plate with systems for active thermal stabilization and lithium supply), FTU (limiter with systems for active thermal stabilization and lithium supply) and stellarator TJ-II (two poloidal limiters) are presented.

1. Experiments in T-11M and T-10 tokamaks

Five version of lithium limiters (FIG. 1) based on different porous materials (stainless steel and molybdenum meshes with pore size 30 – 150 μm) have been tested in T-11M from 1998 till now [7, 8]. The compatibility of Li CPS with boundary plasma and its stability have been confirmed. About 100g of Li have been deposited to the tokamak T-11M wall during experiments. Similar CPS based element for tokamak wall lithiation (FIG. 2) has been used in T-10 from 2006 till now.

Very similar behavior of lithium and it effect on tokamak plasma performance have been found for T-11M and T-10. No catastrophic events leading to lithium injection in MHD stable discharge conditions within the lithium temperature range of 20-600^{o}C have been observed.

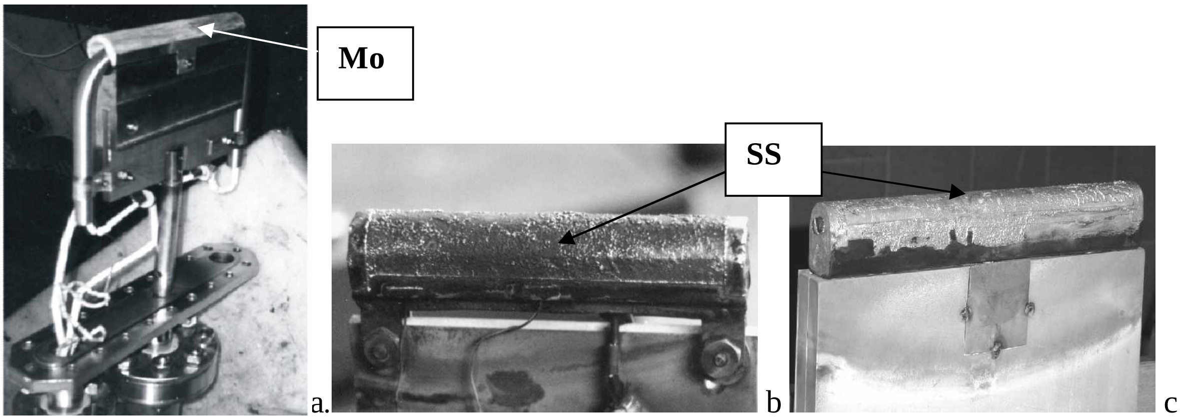


FIG. 1. Liquid lithium limiters of T-11M with CPS based on Mo (a) and stainless steel meshes(b c)

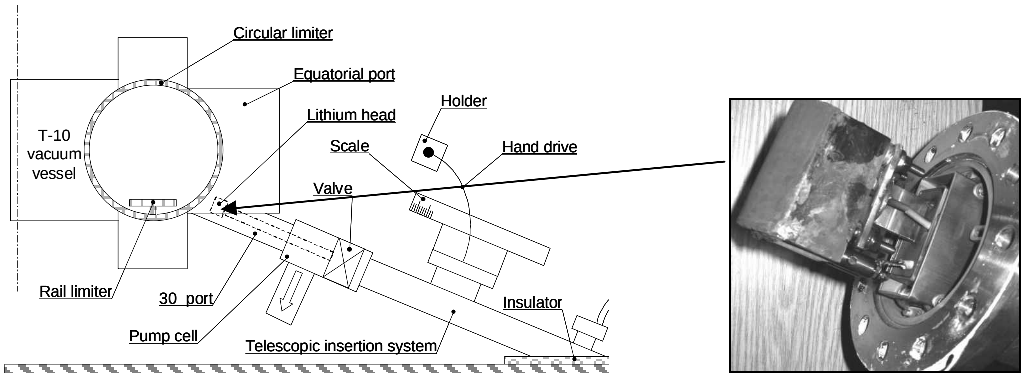


FIG. 2. Liquid lithium evaporator – limiter of T-10

The total lithium erosion was close to the level of sputtering by deuterium and lithium ions with a sputtering yield from 0.5 to 1. The fact of limiter surface protection owing to noncoronal lithium radiation from high power load during stable discharges and disruptions have been detected. Only 30-50 J of about 0.7 kJ of total plasma energy loss has been found to reach the rail limiter in disruption. The recovery temperature of hydrogen isotopes from Li is 320-500^{o}C and 50-100^{o}C for helium. Therefore, at high temperatures (400-500^o C) a tritium capture can be minimized. The solid basis of CPS limiter had no damages after more than $2{\cdot}10^3$ of T-11M plasma shots. Decrease in recycling, Zeff (to 1.1÷1.2) and in total radiation losses (in 3.5 – 5 times) have been observed. The energy confinement times is increased.

2. Experiments with liquid lithium limiter in FTU.

The key stage of lithium CPS investigation under tokamak plasma conditions was the experimental program on all metallic medium size high magnetic field (up to Bt = 6 T) tokamak FTU started at the end of 2005 [3]. The main technological aims of the experiment were to test: the compatibility of lithium CPS made from stainless steel mesh and tungsten felt in real plasma tokamak conditions; the capacity of CPS to self-regenerate the exposed surface and to confine lithium during normal plasma operations and disruptions; the capability to withstand high heat fluxes without damage of limiter surface. The limiter has been installed into the lower vertical port of FTU (FIG. 3) and its radial position can be varied against the last closed magnetic surface. After plasma exposure it can be moved to a separated volume where an optical window permits the lithium surface to be observed.

The ability of capillary force for Li surface renewal and liquid Li confinement are confirmed for high magnetic field up to 6 T and power load exceeding 5 MW/m^2. No catastrophic events

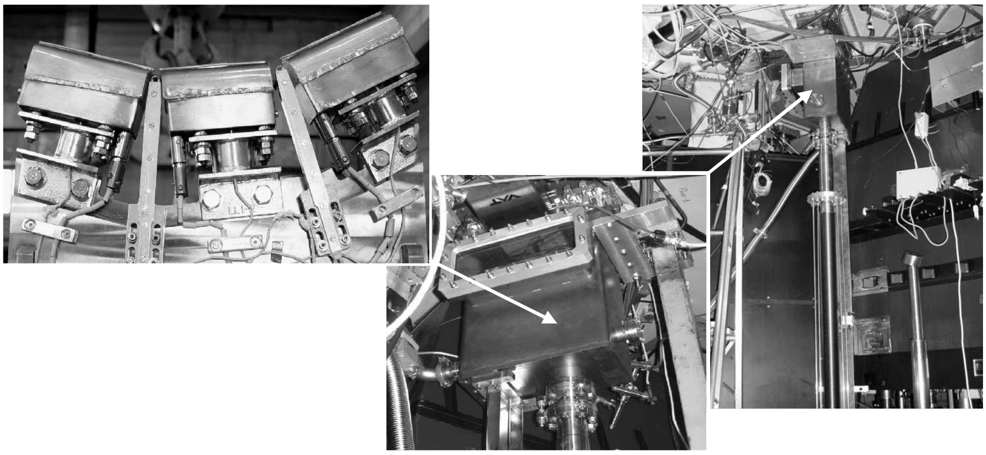


FIG. 3. Liquid lithium limiter of FTU

leading to lithium injection in MHD stable discharge conditions within the lithium temperature range of 200-600 ^{o}C have been observed. No damages of CPS surface and no spectral lines of CPS material (Fe, Cr, Ni) in UV spectra have been found. About of 1.10^{21} Li atoms have been deposited to the wall per discharge owing to sputtering and evaporation. Ten time increase of D puffing is required for programmed plasma density ($Ne=2.10^{20}$ m^{-3}) providing in experiments with LLL. The protection of limiter surface from high heat flux by non-coronal lithium radiation at plasma periphery has been detected. Application of lithium has led to decrease in hydrogen recycling. Lithium application have resulted in reduction of Zeff (from 2.5-3.5 to 1.2-1.5) and radiation losses (two times lower), increase in energy confinement times leading to improved plasma operations. The operations near or beyond the Greenwald limit are easily performed. Lithium application is compatible with ECR and LH plasma heating.

3. Progress in development of lithium technology for fusion application.

Lithium filled CPS from stainless steel mesh has shown the basic opportunities and advantages of this class of plasma facing materials but it not satisfy a requirements for a long-time operation of PFE under power flux more than 5 MW/m^2. A new CPS from tungsten fiber-base material has developed to achieve the increase in thermal conductivity (about 6 times) and mechanical strength (up to 6 times) comparing to stainless steel mesh. Porous mat of 1 mm thick made from tungsten fiber with diameter of 30 μm has pore radius of 15 μm and excellent flexibility (FIG. 4). Investigations of lithium limiters (FIG. 5) from this material in T-11M and FTU tokamaks are in progress now and from the first results have shown the advanced capability with respect to stainless steel based one [8].

Promising results on tokamak plasma improvement are resulted in grows of interest on lithium application to overcome the problem in impurity accumulation for stellarators [9]. Development of two movable poloidal limiters based on lithium CPS (FIG. 6) has been started in 2010. Designing of limiters, lithium element mock-ups manufacturing and test are in progress now. The total area of lithium surface is about $2.24 \cdot 10^{-4}$ m^2. Study of deuterium sorption / desorption process is supposed. Installation of liquid lithium limiters to the camera of TJ-II is expected in 2011.

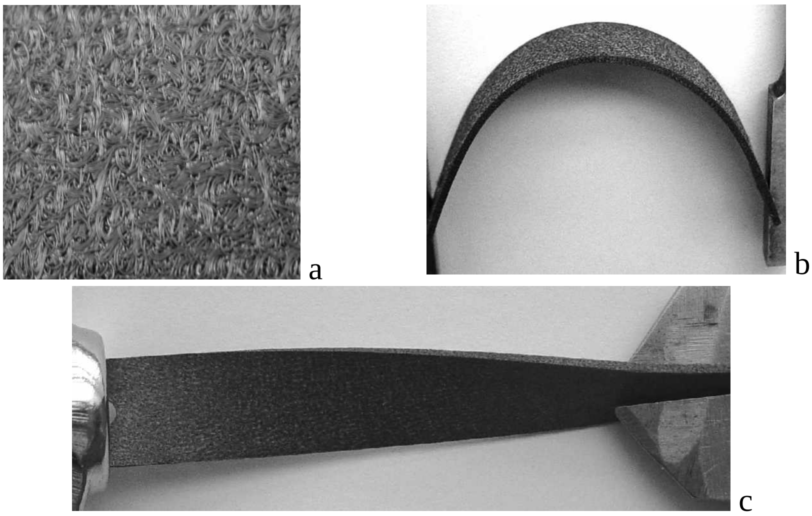
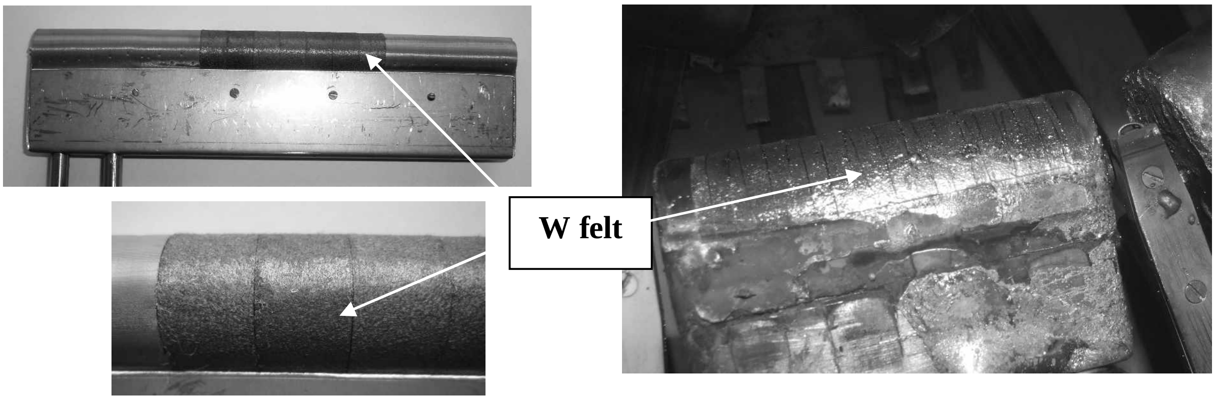


FIG. 4. Porous mat from tungsten felt with pore radius of 15 μm.



A. Lithium limiter of T11-M *B. Lithium unit of FTU limiter after tests*

FIG. 5. View of lithium limiters based on CPS from tungsten felt

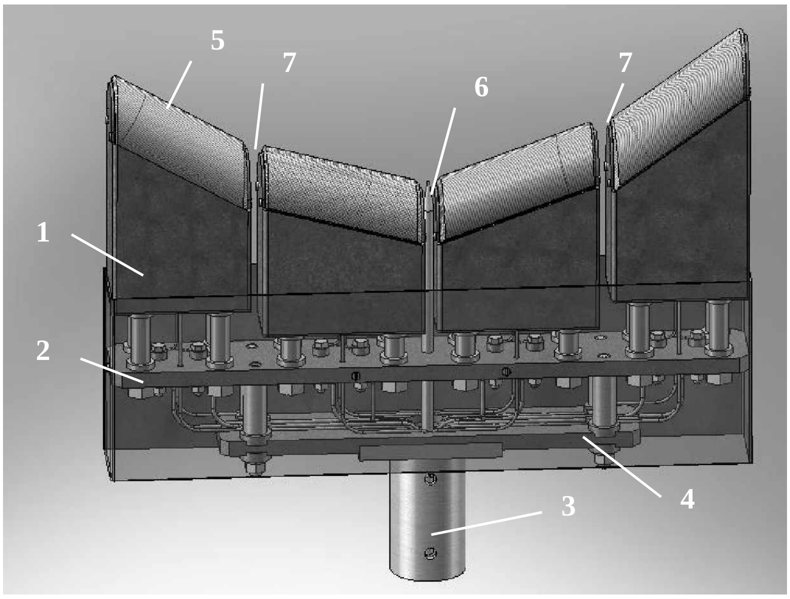


FIG. 6. General view of liquid lithium limiter for TJ-II. 1 –lithium CPS based element with heater and thermocouples ; 2- supporting plate; 3- rod of positioning mechanism, 4- connector with electric insulation, 5- lithium CPS based surface; 6- pipe for gas puffing; 7- two Longmuir's probes

Problems related to PFE (first wall, limiter, divertor) for a new generation of fusion reactors can be overcome by development and experimental study of steady-state operating prototypes based on lithium CPS and including systems for temperature stabilization, lithium supply and tritium concentration control. New projects of steady-state operating lithium limiters for FTU (under development, FIG. 7) and T-11M (experimental tests, FIG. 5A) are providing plasma facing surface temperature stabilization by water heat transferring liquid and permanent lithium supply. Development of lithium divertor element for KTM (ISTC project #K1561 supported by EU) with Na-K eutectic alloy as heat transfer liquid have been started in 2009. Versions of in-vessel lithium element design for KTM are presented on FIG. 8.

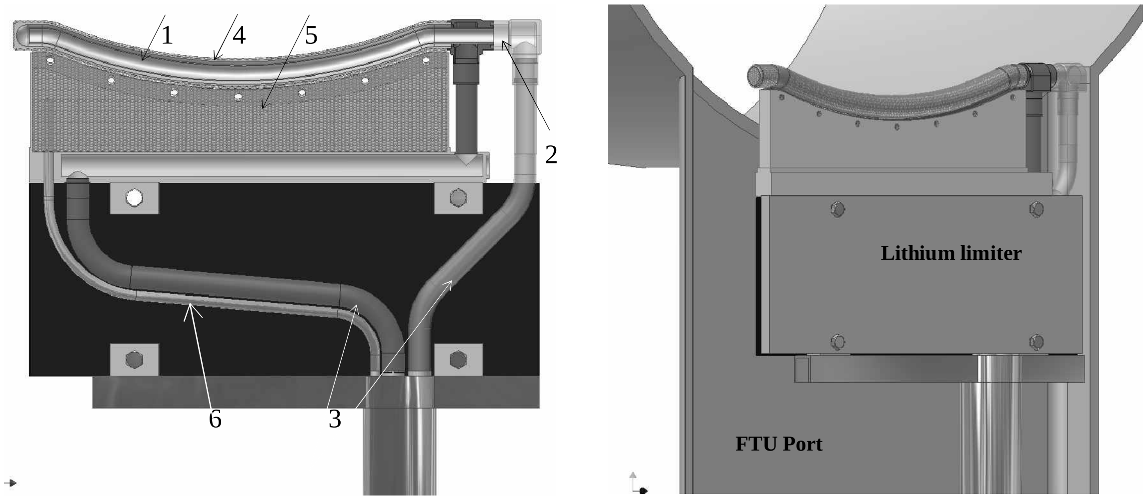


FIG.7. View of single-element prototype of lithium limiter for FTU. 1- molybdenum tube for heat transferring liquid flow; 2- inlet/outlet collector;3- channels to out-vessel systems; 4- lithium filled CPS made from fiber-based tungsten; 5- intermediate lithium supply volum;6 – tube for lithium supply from out-vessel system

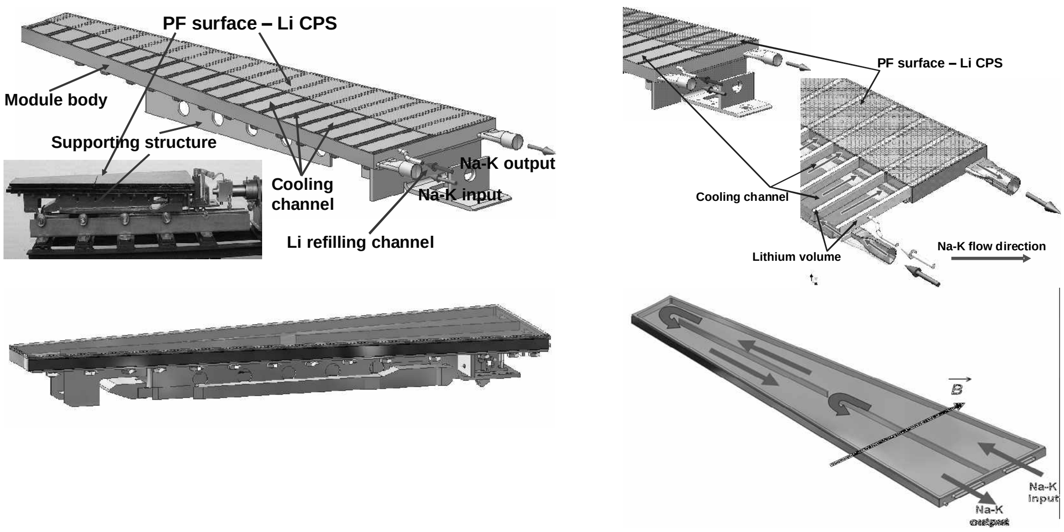


FIG. 8. Design of the lithium in-vessel element of KTM divertor

4. Conclusion

Development of lithium technology for fusion application provides new possibilities in plasma performance improvement and is considered as a promising approach for progress in creation of new fusion device generation.

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