Main Results and Prospects of Lithium Capillary-Porous System Investigation as Tokamak Plasma Facing Material

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Abstract. Concept of lithium fusion reactor has been offered in FEC-16 and lithium **C**apillary-**P**orous **S**ystems (CPS) is a key element of its realization. Experiments in T-11M tokamak with lithium CPS based rail-type limiters were started in 1998 for the purpose of proving the compatibility of lithium CPS with tokamak boundary plasma. Further stage was the beginning of liquid limiter tests in Italian tokamak FTU in 2005. The main reasons of this investigation were confirmation of CPS ability to lithium surface renewal and lithium confinement during normal plasma operation and in disruption, possibility of withstanding of high power flux on the limiter surface without damage in real tokamak condition with ohmic plasma heating mode and with additional heating. Lithization effect on plasma discharge parameters of all metallic and carbon-free tokamak has been investigated. Experiments on inner-wall lithization of graphite containing camera of T-10 tokamak with application of CPS based unit were also started in 2006. All experiments have allowed for solving the problems on MHD stability of liquid lithium film in tokamak conditions, study of lithium CPS compatibility with tokamak boundary plasma, demonstrating the positive lithium effect on plasma parameters and determination of the next steps on further activity in this area. At the same time the promising results of experiments on the lithium CPS serviceability and possibility for withstanding of steady-state (up to 3 hours) high power flux (1- 10 MW/m^2) and plasma disruption effect have been demonstrated. The progress in development of lithium technology allows for deciding the problems in the development of projects of Steady-State Operating Lithium Limiters (SLL) for FTU and T-15, lithium divertor for tokamak KTM. The first stage of this activity is development and experimental study at a power flux up to 10 MW/m^2 of the single-element prototype of SLL/divertor with systems for surface temperature stabilization in the range of 350-550^{o}C and controllable lithium supply. SLL/divertor prototype structure, operating parameters and new tungsten-based CPS are presented. Safety analyses and critical aspects of prototype technology are considered. Investigations on structure material compatibility with lithium in DEMO-type tokamak conditions have allowed for proving the possibility of lithium application in fusion reactor.

1. Introduction

Concept of lithium fusion reactor was offered in 1996 at the 16th IAEA Fusion Energy Conference [1]. This concept has been supported by successful experiments in electron beam SPRUT-4 device at stationary high heat loads of the targets based on lithium **C**apillary-**P**orous **S**ystems (CPS) [2], design developments and a complex of compatibility test of materials and coatings with lithium in various conditions [3]. Lithium CPS is a key element providing realization of this concept technical and physical ideas. CPS different types due to porosity controllable parameters assure confinement of liquid metal under BxJ forces and surface self-regeneration of tokamak in-vessel plasma-facing elements in steady-state operation [4]. The results of lithium CPS resistance studies in tokamak and modeling conditions are presented in the paper.

2. Lithium CPS operating under modeling conditions

Research of lithium CPS under high heat load was conducted in the SPRUT-4 linear plasma device. Lithium CPS targets with vertical working surface was investigated under steady-state electron beam. The range of reactor relevant power loads 1-50 MW/m^2 was covered by the studies. Lithium CPS targets without thermo-stabilization has been successfully tested at a power load of 50 MW/m^2 during 5 s and at loads of 20 and 30 MW/m^2 during 60 and 30 s, respectively. In the further tests of targets with active water cooling the promising results on the lithium CPS ability to withstand steady-state (up to 3 hours) high power flux (in rage of 1-10 MW/m^2) effect have been demonstrated. The total amount of lithium evaporated from the CPS surface during this experiment was about 0.8 kg. Decrease in CPS serviceability has not been observed [5].

High serviceability of lithium CPS has also been demonstrated in plasma gun experiments on modeling of plasma disruption effects in plasma accelerators (QSPA, MK-200UG) and "Plasma focus" facility [6]. The modeling test conditions are presented in Table I.

TABLE I: HYDROGEN PLASMA PULSE PARAMETERS.

Parameter	Facility		
	Plasma focus	MK-200UG	QSPA
Energy density, MJ/m^2	60	15	4-5
Pulse duration, s	~10^{-6}	$4 \cdot 10^{-5}$	$(2\text{-}5) \cdot 10^{-4}$
Temperature, eV	10-100	100-200	30
Plasma density, cm^{-3}	10^{18}	$(2\text{-}6) \cdot 10^{15}$	$(2\text{-}5) \cdot 10^{16}$

Even under high-pulse power load a solid CPS structure (molybdenum, stainless steel) does not fail due to the formation of a protective layer of dense lithium plasma.

3. Interaction of Lithium CPS with tokamak plasma

Experiments with lithium CPS in T-11M tokamak (R/a = 0.7/0.2 m, B_T = 1 T, $J_p \approx$ 100 kA, τ=0.1-0.3 s, $T_e(0)$ = 400 eV, $n_e = (2\text{-}4) \cdot 10^{19}$ m^{-3}) were started in 1998. Step-by-step tests with four versions of lithium CPS based rail-type limiters were conducted for the purpose of proving the compatibility of lithium CPS with tokamak boundary plasma [7].

It has been successfully tested the following: (1) mechanical stability of liquid lithium surface to plasma effect, (2) temperature stability of lithium-plasma surface, (3) lithium erosion intensity versus heat influxes, (4) lithium accumulation effect in the plasma core, (5) technological features of vacuum vessel preparation related to lithium deposition on its first wall, (6) phenomena of lithium sorption and desorption of plasma-forming gases (H, D, He). Lithium sputtered by plasma from the limiter surface penetrates into its depths of 2-3 cm cascade - ionized and electronic - excited. Lithium non-coronal radiation created «radiative blanket» at plasma periphery distributing much of limiter power influx from plasma along the whole chamber. In experiments on T-11M tokamak lithium «blanket» total radiation power to the wall reached ~ 100 kW. Thus, lithium limiter transformed the local load into the distributed one simplifying radically a problem of local heat removal from in-vessel high heat flux elements (divertor, limiter).

A low level of lithium contamination in the tokamak plasma core is thought to be the most important result of lithium CPS-limiter application in T-11M tokamak. Z_{eff} was reduced from 2 to $\leq$ 1.2 [8].

Further stage of activity in this area was begun in 2005 on FTU tokamak with ITER relevant parameters (R= 0.93 m, a=0.31 m, B_T = ≤8 T, I_p ≤= 1.6 MA, n_e =(0.2-2.6)·10^{20} m^{-3}, τ = 1.5 s, P_{LH}≤2 MW, P_{EC}≤1.6 MW, P ~ 2-5 MW/m^2 at normal discharge). The main reasons of this investigation were confirmation of CPS ability to lithium surface renewal and lithium confinement during normal plasma operation and in disruption, capacity to withstand high power flux on the limiter surface without damage in real tokamak condition with ohmic plasma heating mode and with additional heating [5].

On lithization better plasma performances with Lithium than with Boron. In discharges with lithium limiter gas puffing should be increased >10 times to get the same electron density with respect to and fully metallic boronized walls. Operations near or beyond the Greenwald limit are easily performed. Discharges forming, their reproducibility and as well parameter recovery from plasma disruptions were greatly improved in LLL operating conditions. The LLL of applied design is able to withstand high power flux up to 5 MW/m^2 without CPS surface damage [10-13].

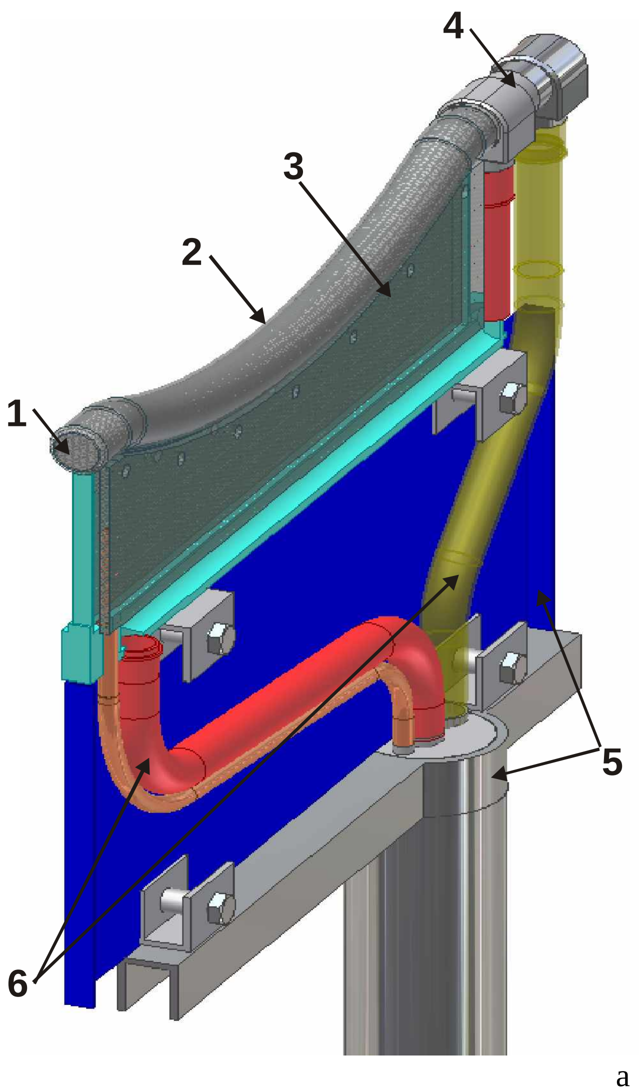
Experiments on inner-wall lithization of graphite containing camera of T-10 tokamak (R= 1.5 m, a=0.3m, B_T = ≤3 T, I_p ≤= 0.5 MA, τ ~ 1 s, P_{EC}≤3.0 MW) with lithium CPS application were also started in 2006 [14, 15]. Moderate wall deposition with lithium made it possible to decline deuterium recycling up to 0.7; to decrease contents of heavy impurities three times, oxygen 7-10 times, carbon by 20-30 %, bolometric losses by 20-30%, Z_{eff} by 10%. Experiments in discharge conditions with additional heating showed that lithization fit the reliable operation of T-10 power gyrotrones. The effect of lithization on light transmission of diagnostic windows has not been noted. In these experiments lithium CPS based devices have demonstrated the requisite parameters and servesability even after their long-term storage.

All experiments have allowed for solving the problems of liquid lithium film MHD stability in tokamak conditions, study of lithium CPS compatibility with tokamak boundary plasma, demonstration of the positive lithium effect on the plasma parameters and determination of further activities in this area [16].
Investigations on structure material compatibility with lithium in DEMO-type tokamak conditions have confirmed the possibility of lithium application in the systems (plasma facing component, heat transfer, self healing coatings ect.) of fusion reactor [3, 17].

4. Prospects of lithium CPS component development for tokamaks

The progress in development of lithium technology and also activity in lithium area in the tokamaks NSTX, CDX-U and LTX, stellarator TJ II permit of solving the problems on the development of projects of steady-state operating in-vessel lithium components for operative and under construction tokamaks.

At present the proposals on steady-state lithium limiter for T-15 tokamak (R/a = 2.43/0.7 m, B_T = 3.6 T, J_p ≈ 1 MA, τ=1.5, «Kurchatov Institute», Russia), lithium divertor elements for KTM tokamak (R/a = 0.9/0.45 m, B_T = 1 T, J_p ≈ 0.75 MA, τ=4-5 s, Republic Kazakhstan) and others are considered. Among them the most elaborated proposal is believed to be the creation of lithium limiter single-element prototype with the systems for surface temperature active stabilization and lithium feeding in FTU tokamak.



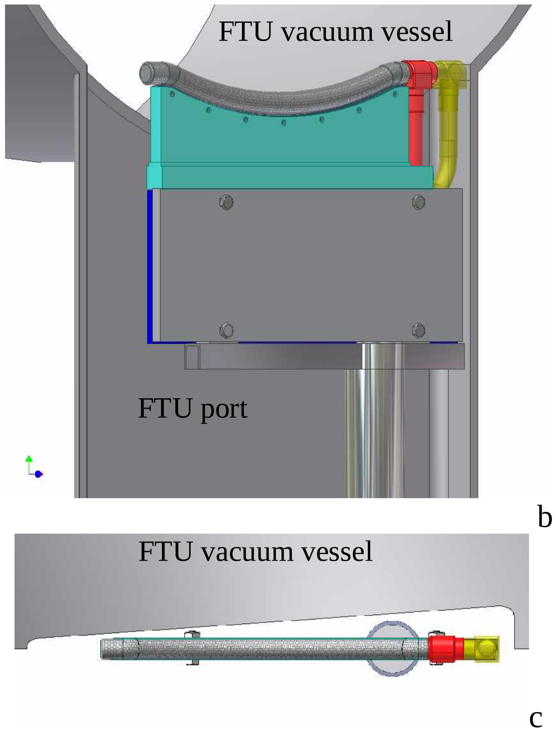


FIG. 1. General view of CLL in-vessel element (a) and view of CLL in-vessel element in FTU lower port window (b, c).

1 – tubular element трубчатый приемный элемент, 2 – lithium filled CPS, 3 - intermediate lithium volume, 4 – coolant inlet/outlet collector, 5 – supporting elements, 6 – channels for heat transferring liquid flow

The single-element prototype of Cooling Lithium Limiter in-vessel element (Fig. 1) is offered to be created on the basis of Field's-type thin wall (~1 mm) tubular element (1) made from molybdenum with inlet/outlet collector (4) and channels for heat transferring liquid flow (6). Limiter surface is covered by the lithium filled CPS made from fiber-based tungsten (2). Limiter surface supply with lithium is ensured by external secondary system through intermediate lithium volume (3) hydraulically connected to the CPS on the limiter surface. Initial heating of limiter up to 200^{o}C and its temperature stabilization during plasma interaction in the range of 350-550^{o}C is provided by special thermal stabilization external system due to circulation of the overheated water under pressure.

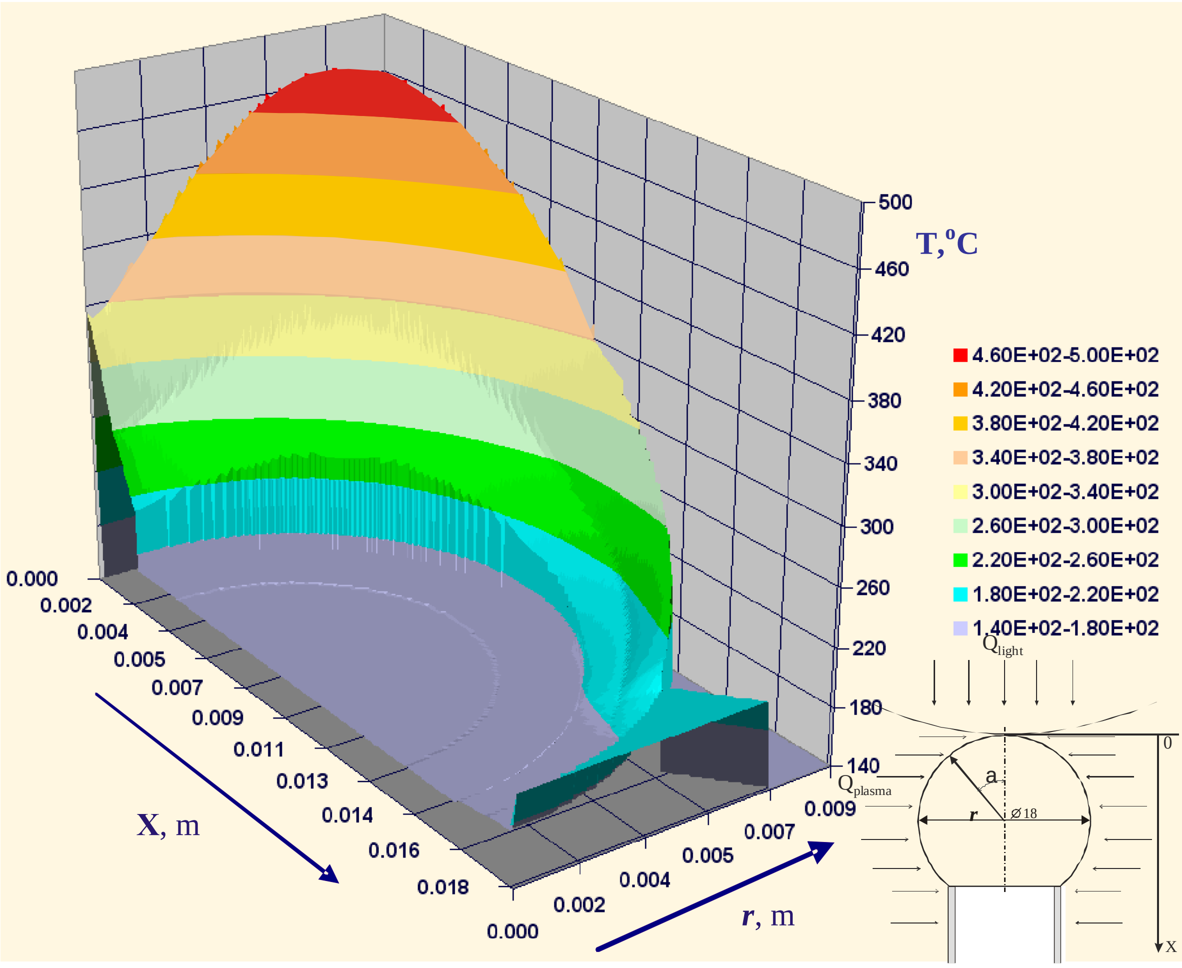


FIG. 2. Field of temperatures in input coolant cross-section of CLL working part. Q_{max} – 10 MW/m^2, 5 s from plasma shot start, input coolant temperature - 140^{o}C, coolant rate – 0.5 kg/s, coolant velocity – 7-8 m/s

Thermo-hydraulic analysis of limiter design with appropriate cooling system operating parameters has shown (Fig. 2) that the temperature of lithium limiter surface is not excess of 490^{o}C in FTU conditions at power flux up to 10 MW/m^2 during 5 s. It allows for providing a controllable flux of lithium atoms to the plasma column, long-term operation of the limiter without damages at high thermal loads. Thermal-stress and safety analyses of limiter design specify its high damage resistance and an opportunity of safe operation during lithium experiments.

Thus, a principal feasibility of design under development was demonstrated. In case of successful experimental studies FTU full-scale steady-state limiter based on lithium CPS is thought to be created.

To continue and advance the activities on physical aspects of lithium use in tokamaks, to decide the problems on in-vessel element creation are planned with lithium CPS on T-11M and T-10 tokamaks.

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