Main Directions and Recent Test Modeling Results of Lithium Capillary-Pore Systems as Plasma Facing Components

V.A. Evtikhin[1], I. E. Lyublinski[1], A. V. Vertkov[1], E. A. Azizov[2], S. V. Mirnov[2], V. B. Lazarev[2], S. M. Sotnikov[2], V. M. Safronov[2], A. S. Prokhorov[2], V. M. Korzhavin[3]

[1] FSUE "Red Star"-"Prana-Center" Co, Moscow, Russian Federation

[2] FSUE TRINITI, Troitsk, Moscow region, Russian Federation

[3] RF Ministry for Atomic Energy, Moscow

Abstract At present the most promising principal solution of the divertor problem appears to be the use of liquid metals and primarily of lithium Capillary-Pore Systems (CPS) as of plasma facing materials. A solid CPS filled with liquid lithium will have a high resistance to surface and volume damage because of neutron radiation effects, melting, splashing and thermal stress-induced cracking in steady state and during plasma transitions to provide the normal operation of divertor target plates and first-wall protecting elements. These materials will not be the sources of impurities inducing an increase of Z_{eff} and they will not be collected as dust in the divertor area and in ducts.

Experiments with lithium CPS under simulating conditions of plasma disruption on a hydrogen plasma accelerator MK-200 [$\sim$ (10 - 15) MJ/m^2, $\sim$ 50 μs] have been performed. The formation of a shielding layer of lithium plasma and the high stability of these systems have been shown.

The new lithium limiter tests on an up-graded T-11M tokamak (plasma current up to 100 kA, pulse length ~0.3 s) have been performed. Sorption and desorption of plasma-forming gas, lithium emission into discharge, lithium erosion, deposited power of the limiter are investigated in these experiments. The first results of experiments are presented.

Keywords: lithium, divertor, plasma facing material

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1 Introduction

The development of high-performance divertor target plates and other high heat flux components is one of the most serious concern for a tokamak reactor [1]. In real tokamak-machines C, W, Be are used as plasma facing materials and in ITER they are also considered as materials for first wall, limiter and divertor plate. Yet, the operation performance of the above materials under ITER conditions will be limited practically. In next-step fusion reactors a great increase in fusion energy loads is allowed. A rise in the erosion rate of the protective materials of high heat flux component leads to their serious damage and to the considerable plasma contamination resulting in an increase of Z_{eff}.

Unlike free-flowing metal film concept [2,3], the use of lithium Capillary-Pore System (CPS) as a plasma facing material is the most promising concept [1]. Solid CPS filled with liquid lithium will have a high resistance to radiation damage, thermal stresses and cracking, melting in steady state and during plasma transitions (disruptions, ELMs and the others), and possess surface self-regeneration through surface tension forces, which are basically different from the solid material divertor concept. Lithium has a low atomic number ($Z = 3$), a high latent heat of evaporation, a capacity to be interacted with hydrogen isotopes to enable removal of tritium, and a low melting temperature ($\sim 180°$C) to allow for the operation of divertor target plates and first wall protective elements. The ability of lithium to be effectively condensed at ~450°C eliminates its accumulation in the

Table 1. Plasma pulse parameters

Parameter	Facility		
	Plasma focus	MK-200UG	QSPA
Energy density, MJ/m^2	60	15	$4 \sim 5$
Pulse duration, s	$\sim 10^{-6}$	4×10^{-5}	$(2\sim5)\times10^{-4}$
Temperature, eV	$10 \sim 100$	$100 \sim 200$	30
Plasma density, cm^{-3}	10^{18}	$(2 \sim 6)\times 10^{15}$	$(2\sim5)\times10^{16}$
Axial magnetic field, T		2	1

reactor plasma. This material will not be a source of impurities inducing a rise in $Z_{\rm eff}$ and they will not be collected as dust in the divertor area.

The concept of lithium fusion reactor has been proposed at FSUE "Red Star" [4]. At present, its successful development together with TRINITI and RRC "Kurchatov Institute" [5~8] allowed a closed concept of lithium liquid-metal divertor to be developed and substantiated[9]. The main directions of further lithium CPS study on protection of high heat flux component of tokamak are as follows: (1) investigation of their resistance to high energy loads and (2) investigation of compatibility with tokamak plasma. The recent study results are presented.

2 Pulsed plasma flows - lithium CPS interaction

The interaction of lithium CPS with plasma flows were investigated in [6 ~ 8]. The modeling test conditions are provided in Table 1. In work [6] the CPS target was irradiated in accelerator QSPA by hydrogen plasma flow under high heat loads in simulating conditions of the divertor plate operation during plasma disruption. The formation of a dense plasma layer and lithium neutral atoms in the vicinity of the target surface is the distinguishing feature of the interaction of plasma flow with liquid lithium surface. Fig. 1 presents a spatial distribution of plasma density at different instants from the pulse onset. At the first instant the density reached $10^{17}cm^{-3}$. An increase in density was caused by lithium ionization and lithium plasma formation. Then, the plasma density was slightly reduced (up to maximum $4\times10^{16}cm^{-3}$). In front of the lithium target a great number of neutral lithium appeared.

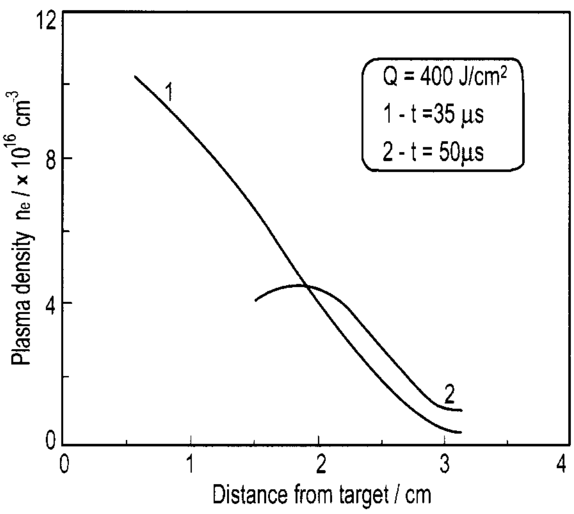

Fig.1 Plasma density distribution near the target

The experiments show that the plasma energy is absorbed in the layer of lithium plasma and neutral atoms, which plays a role in a shielding layer for the lithium target. A major part of the absorbed energy is reirradiated by the excited atoms. Experiments were continued in plasma accelerator MK-200UG. A CPS target made of a stainless steel mesh without lithium was destroyed by the hydrogen plasma after a single shot and remained undamaged in case of being filled with liquid lithium at 250°C. CPS for target protection at a heating temperature up to 250°C provided the formation and retention of lithium film on the frontal surface of target during tests (17 shots) as this occurred on tests in QSPA and "Plasma focus" facility [8]. Experiments in plasma accelerators and facilities at significantly different parameters indicate that the plasma near the CPS surface is lithium plasma and the electron temperature is low. Even under a high-pulsed power load a solid CPS struc-

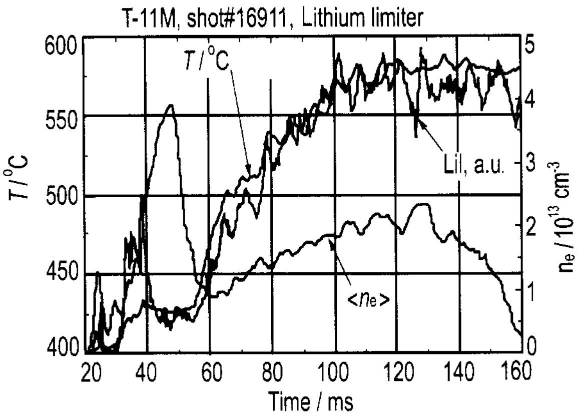

Fig.2 The temporal behavior of limiter surface temperature LiI line and electron density

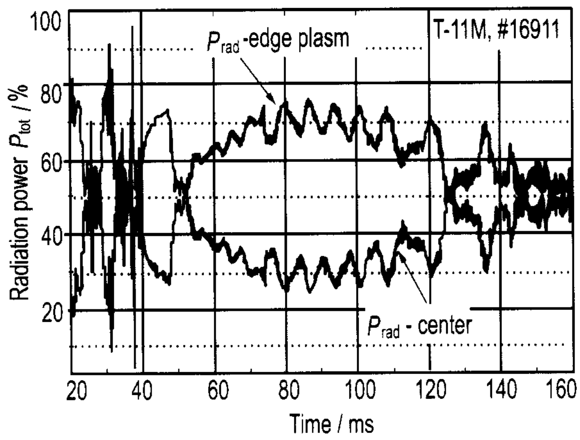

Fig.3 Behavior of relative radiation losses in discharge with "hot" lithium by limiter(400°C) from central ($r < 15.2$ cm) and peripheral plasma ($r > 15.2$ cm)

ture does not fail due to the formation of a protective layer of dense lithium plasma.

3 Lithium CPS - tokamak plasma interaction

Experimental study on lithium CPS as a plasma facing surface of limiter under real tokamak plasma conditions was begun on T-11M in 1998. In the initial stage the goal was a test of liquid lithium stabilization by CPS. Mechanism of lithium influx from the liquid metal surface into the plasma chamber has been investigated. At this stage, the experiments were performed with plasma discharges in He, D and H at $I_p \sim 90$ kA, $P_{Ohm} \sim 100$ kW, $\Delta t \sim 0.1$ s. Experiments showed that (1) the capillary forces of porous structure with liquid lithium in T-11M conditions were sufficient for mechanical stability of lithium in spite of MHD events, (2) the lithium CPS based on Mo and SS structures withstood > 2000 discharges without failure at a limiter power flux of 10 MW/m^2 and (3) lithium was compatible with tokamak edge plasma, i.e. anomalous erosion of lithium and its influx processes were not observed [10,11].

T-11M up-grade (discharge duration $\Delta t \sim 0.3$ s, $I_p \sim 80$ kA, toroidal field on an axis 1 T - 1.2 T) and the use of a new lithium limiter with a lithium-surface-temperature stabilizer during the plasma discharge allowed the testing to be conducted under quasi-stationary conditions. The initial temperature of lithium limiter was controllable from 20 °C to 400°C. The surface temperature of limiter was measured during the discharge. The neutral lithium line emission from the plasma edge near the limiter surface was determined. The radial distribution of radiation losses was measured by a bolometer. A superhigh-frequency interferometer was used for the density measurement on different chords. The behavior of the limiter surface temperature and the calculated heat flux on the surface during discharge were determined. In the first 30 ms ~ 40 ms after the start the power reaches a stationary level (70 ms ~ 100 ms), and the evolution of the temperature is described by the well-known law $T_s(t) \sim t^{1/2}$. The absolute magnitude of the thermal load during the initial phase is approximately 18 MW/m^2. After 30 ms ~ 40 ms the temperature begins to increase more slowly than $t^{1/2}$, resulting in a decline in q roughly by a factor of two to the end of the discharge.

During operation with lithium limiter, atomic lithium begins injecting into the plasma in the initial stage by the ion sputtering [9], and then at a higher temperature ($T > 500$°C) starts to predominate over thermal evaporation, strongly depending on the temperature. The discharge with lithium high initial temperature (400°C) is given in Fig. 2. After about 60 ms the temperature has gone up to a steady-state value of about 570°C. The electron temperature on an axis has been about 400 eV and slightly changed during the discharge in spite of the fact that the flowing atomic lithium from the limiter surface has in-

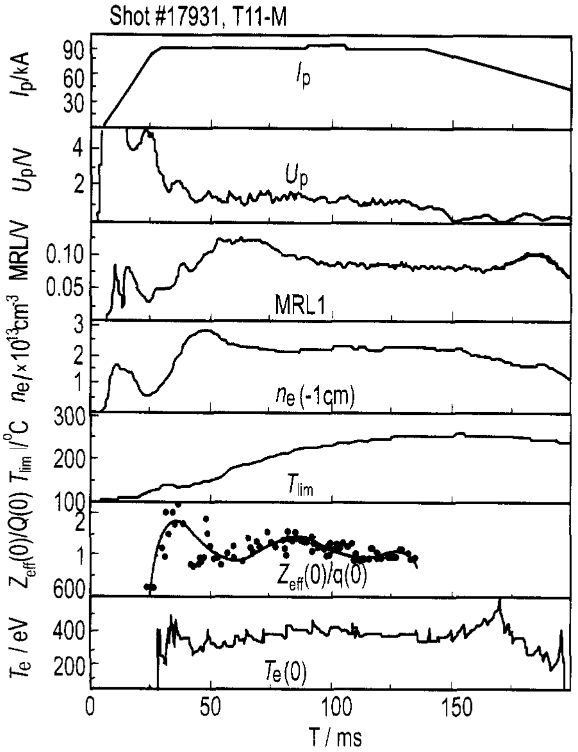

Fig.4 Shot with quasi-stationary parameters

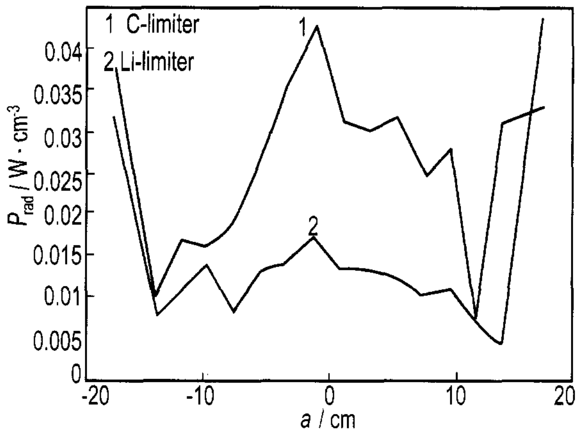

Fig.5 Radiation losses profile for shots with lithium and graphite limiters

creased approximately five times. The behavior of the relative radiation losses is presented in Fig. 3.

From these data it follows that up to 70% of power is radiated in a surface layer about 5 cm thick and only 30% from a central area. Such a distribution of radiation losses in the plasma column explains why there is no radiation cooling in the core plasma at high lithium influx and why a rather fast stabilization of limiter surface temperature takes place. Calculations of $Z_{\rm eff}$ on the axis give a quantity of ~ 1.2 for this mode. The most successful result has been achieved in the recent experiments with thin (≤ 1 mm) CPS-layers which rest on the heat accumulator. Using such a limiter in the deuterium plasma at a current of $I_{\rm p} = 90$ kA with a duration of ~ 0.2 s (Fig. 4) a quasi-stationary discharge regime has been obtained with parameters at a steady-state level ($< n_{\rm e} > = 2 \times 10^{19}$ m^{-3}, $T_{\rm e} = 400$ eV, $\tau_{\rm E} = 8$ ms, $T_{\rm lim} = 300$ °C, $P_{\rm lim} \leq 10$ MW/m^2). This mode was accompanied by litization of discharge chamber walls that reduced sharply deuterium recycling and resulted in its strong puffing. Parameter $Z_{\rm eff}(0)/q(0)$ declined to a level of 1.1 under these conditions. It is indicated that $q(0) \sim 1$ is usually such a low value of $Z_{\rm eff}(0)/q(0)$ which points to the production of practically pure deuterium plasma on the column centre.

Measurements of total radiation from the central zone of the plasma by AXUV detectors confirm this conclusion as well. Fig. 5 shows radiation losses from the cross-section of plasma column for two discharge modes: with Li and C limiters at plasma fixed density ($< n_{\rm e} > = 2 \times 10^{19}$ m^{-3}). The comparison between the two discharge modes concludes that Li version is preferable since the intensity of the radiation from the column centre has been suppressed 3 $\sim$ 5 times as opposed to a graphite limiter.

Lithium erosion is proportional to Li line intensity and integral light emission in the vicinity of the limiter [12] and has the same increased dependence on the limiter temperature as sputtering yield of lithium specimen by H and Li ions obtained in the simulating experiments. This agreement confirms that the main mechanism of Li erosion in edge plasma is the same as the fundamental one which is observed on its ion sputtering in the above experiments [13]. Lack of Li-eroded abnormal channels at temperatures up to 500°C makes it possible to anticipate a success in lithium tokamak reactor development. In this case, the problems of first wall erosion in tritium recuperation, metallic dust and high Z at the column centre could be solved.

4 Conclusions

The plasma accelerator experiments have shown that in the vicinity of the CPS surface a dense protective layer of lithium plasma is formed, due to which a solid CPS structure will not be damaged under a short-term high thermal load.

During operation with "hot" lithium limiter, a mode with an external radiation layer radiating up to 70% of total radiation losses was detected. Results of previous experiments on T-11M are confirmed. On the T-11M tokamak in the mode with a lithium limiter the pure deuterium plasma with $Z_{\text{eff}} = 1.1$, $I_{\text{p}} = 90$ kA and all the parameters at a quasi stationary level has been obtained. Lack of lithium-eroded abnormal channels up to 500°C confirms possible realization of lithium tokamak reactor concept.

One of the current directions of further activities is the development and testing of high heat flux components based on lithium CPS under large tokamak (T-10, FTU) conditions.

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E-mail address of I. E. Lyublinski:

lyublinski@mtu-net.ru