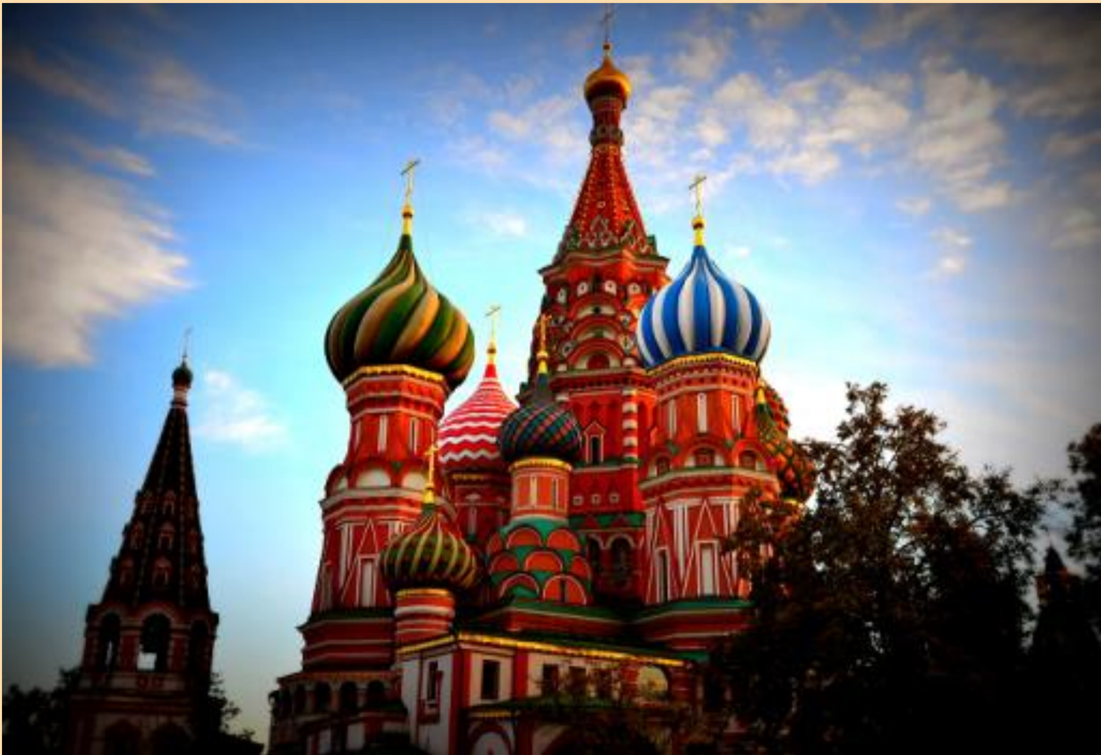
5th International Symposium on Liquid Metals Applications for Fusion

ISLA -5

25-27 September 2017 **Moscow, Russia**



PROGRAMME
ABSTRACTS

MEPhI

RSF Russian Science Foundation

ROSATOM

The ministry of Education and Science of the Russian Federation

National Research Nuclear University MEPhI
(Moscow Engineering Physics Institute)

Russian Science Foundation

5th International Symposium on Liquid Metals Applications for Fusion

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Editors:

V.A. Kurnaev

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S. A. Krat

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Dear colleagues,

We are glad to invite you to Moscow, 25-27 September 2017 to participate in the 5th International Symposium on Liquid Metals Applications for Fusion (ISLA-2017)!

This conference continues the series of conferences on lithium applications in fusion devices previously organized in Japan, USA, Italy, and Spain. This year, the International Organizing Committee decided to broaden the scope of the Symposium and invite reports on all liquid metals that could be possible candidates as plasma facing materials for fusion applications.

The Symposium will be held in the centre of Moscow in the Courtyard Marriott Hotel and we will have a chance not only for the fruitful discussion of the last scientific results, but also to spent a nice time in the historical part of our capital.

We are looking forward to see you in Moscow!

Local Organizing Committee of ISLA-2017

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G. Mazzitelli (Italy)

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- National Research Nuclear University MEPhI
- Russian Science Foundation
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- ROSATOM

INFORMATION

Registration

The conference registration will be open on Sunday evening (17.00-19.00) or on Monday morning (8.00-9.00) in the conference hotel “Hotel Courtyard by Marriott Moscow” (Voznesenskiy Pereulok, 7).

Welcome reception

On Sunday 24 September 2017 at 19:00 the welcome reception will take place in the conference hotel “Hotel Courtyard by Marriott Moscow”.

Poster presentation

The posters should have poster size of A0 in portrait format (841mm width x 1189mm height).

VENUE

The conference will be held at Courtyard by Marriott Moscow City Center Hotel (Voznesenskiy Pereulok, 7). It is located in the center of Moscow. You can get to it from Moscow subway system following these maps:

From Okhotniy Ryad Metro Station

From Arbatskaya Metro Station

From Tverskaya Metro Station

PROGRAMME OVERVIEW

	Monday	Tuesday	Wednesday
8:00	Registration		
8:30			
9:00	Opening	Ruzic	Vertkov Yang
9:30	Mirnov	Hirooka	Stemmley
10:00	Iafrati	Demina Van Eden	Zharkov Ni He
10:30	J.Hu	Rindt	**Coffee Break**
11:00	**Coffee Break**	**Coffee Break**	Giegerich Mossman
11:30	Tabares	Morgan	Final Discussion
12:00	Mazitelli	De Castro Krat	
12:30	Vershkov	Meng	Closing
13:00	Sun **Lunch**	W.Hu **Lunch**	
13:30			
14:00			
14:30	Sandefur Loureiro	Zakharov	
15:00	Pericoli Ridolfini	Pshenov Miloshevskiy	
15:30	Shcherbak **Coffee Break**	Deng	
16:00	Horacek	**Coffee Break** Benos	
16:30	Lazarev Tazhibayeva	Zanino	
17:00	Vasina	**POSTERS**	
17:30			
18:00			
18:30	Guided Tour		
19:00		**Conference Dinner**	

SCIENTIFIC PROGRAMME

Monday 2017/09/25

Section 1: Liquid Metals in Magnetic Confinement Experiments Chair: Y. Hirooka		
8:00 9:00	–	Registration
9:00 9:30	–	Opening
9:30 10:00	–	S. Mirnov – P/S tokamak limits and overcoming them by lithium use
10:00 10:30	–	M. Iafrati – Comparison of the experiment with liquid lithium and liquid tin in FTU
10:30 11:00	–	J. Hu – Summary of upgarde and experiments of lithium systems on EAST tokamak in the last two years
11:00 11:30	–	**Coffee break**
11:30 12:00	–	F. Tabares – Testing Liquid Metal-CPS prototypes by exposure to TJ-II fusion plasmas
12:00 12:30	–	G. Mazitelli – First experimental results of heat loads on tin liquid limiter on FTU
12:30 13:00	–	V. Vershkov – Experiments with lithium capillary porous structure in T-10 tokamak with tungsten limiters
13:00 13:20	–	Z. Sun - ELM Control by Lithium Aerosol and Granule Injection on EAST with Tungsten Divertor
13:20 14:30	–	**Lunch**

Monday 2017/09/25

Section 2 Liquid Metals in Magnetic Confinement Experiments Continued Chair: A.A. Pisarev	
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14:50 – 15:10	J.P. Loureiro - Spectroscopic imaging of liquid metal plasma facing materials on ISTTOK
15:10 – 15:30	V. Pericoli Ridolpfini - A comparative study of the effects of liquid Li and Sn as DEMO divertor targets on the heat loads and SOL properties
15:30 – 15:50	A. Scherbak - Investigation of dependence of lithium and hydrogen collection by collector target on temperature of target surface in emitter-collector system on T-11M tokamak
15:50 – 16:10	**Coffee break**
16:10 – 16:30	J. Horacek - Plans for liquid metal divertor module tests in the COMPASS tokamak
16:30 – 16:50	V. Lazarev – Observation of plasma processes in SOL region of T-11M tokamak with lithium limiter by fast video registration
16:50 – 17:10	I. Tazhibayeva - Technique of reactor experiments on study of lithium CPS interaction with deuterium under neutron irradiation
17:10 – 17:30	Ya. Vasina – Investigation of edge plasma parameters in T-11M tokamak with a mach probe
18:30 – 20:30	**Guided tour**

Tuesday 2017/09/26

Section 3: Liquid Metal Laboratory Tests **Chair: G. Mazitelli**		
9:00 9:30	–	D.N. Ruzic - Lithium loop and hydrogen isotope extraction technologies for use with a LiMIT-style PFC
9:30 10:00	–	Y. Hirooka - A Review of Recent Studies on Convection Effects on Particle Recycling from Liquid Metals under Plasma Bombardment
10:00 10:20	–	E. Demina - The study of the W-Li composite sample surface subjected to the 100-fold pulsed deuterium plasma irradiation
10:20 10:40	–	G.G. van Eden - Test of a Pre-Loaded Liquid Lithium Divertor Target on Magnum-PSI
10:40 11:00	–	P. Rindt - Performance of tin and lithium CPS targets in response to high heat-flux plasmas in Pilot-PSI and Magnum-PSI
11:00 **11:30**	**–**	**Coffee break**
Section 4: Liquid Metal Properties **Chair: G. Mazitelli**		
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12:00 12:20	–	A. De Castro - Temperature dependence of lithium film formation and deuterium retention on hot W samples by LIDS and TDS. implications for future fusion reactors
12:20 12:40	–	S. Krat – Co-deposition of Li-D layers at elevated temperatures
12:40 13:00	–	X. Meng - Study of the corrosion behaviors of 304 stainless steel exposed to static liquid lithium
13:00 13:20	–	W. Hu - Response of the Fe(001) surface and Fe(001)-Li solid-liquid interface under irradiation
13:20 **14:30**	–	**Lunch**

Tuesday 2017/09/26

Section 5: Liquid Metal Theory/Modeling Chair: D.N. Ruzic	
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15:00 – 15:20	A. Pshenov - SOLPS4.3 modelling of lithium transport and non-coronal radiation in T-15MD tokamak with lithium emitter-collector scheme in use
15:20 – 15:40	G. Miloshevsky - Analysis and modeling of lithium flows in porous materials
14:40 – 16:00	H. Deng - First-principles calculations on the wettability of Li atoms/clusters on the (111) surfaces of Fe, W and Mo substrates
16:00 – 16:20	**Coffee break**
16:20 – 16:40	E. Benos - Static arrangement of the cps: effect of capillarity overpressure and intermolecular forces with the substrate on the film thickness
16:40 – 17:00	R. Zanino - Quasi-2D analysis of a box-type Liquid Metal divertor for the DTT facility: Comparison Li vs. Sn
17:00 – 18:30	**POSTERS**
19:00 – 22:00	**CONFERENCE DINNER**

Wednesday 2017/09/27

Section 6: Liquid Metal Technology Chair: F.L. Tabares		
9:00 9:20	–	A. Vertkov - Liquid metal CPS based PFC for steady state operating tokamak
9:20 9:40	–	J. Yang - Experimental study on liquid metal film flows related with nuclear fusion engineering
9:40 10:00	–	S.A. Stemmley - Compact Self-Driven Liquid-Lithium Loop for Fusion-Relevant Neutron Generation
10:00 10:20	–	M. Zharkov - The devices for diagnostic and lithium collecting of tokamaks T-11M and T-10. First results
10:20 10:40	–	P. Ni He - Corrosion behavior of high-density helium plasma irradiated molybdenum in liquid lithium
10:40 11:00	–	**Coffee break**
11:00 11:20	–	T. Giegerich - Applications of Mercury in the Fusion Fuel Cycle.
11:20 11:40	–	A. Mossman - Experimental Study of Interactions Between Liquid Lithium Free Surface and Pulsed Currents and Magnetic Fields
11:40 12:30	–	Final discussion
12:30 13:00	**–**	**Closing**

POSTER SESSION

1. **Yu. Ponkratov** - Experimental technique for study of lithium-containing materials under neutron irradiation at the IVG1.M reactor (Kurchatov, Kazakhstan)
2. **E. Marenkov** - Radiation transport in edge lithium plasma
3. **D. Bulgadaryan** - MEIS analysis of Li layers deposition
4. **G.M. Apruzzese** - First experimental results with tin limiter on FTU plasma
5. **Z. Zaurbekova** - Analysis of reactor irradiation experiments of lead lithium eutectic Pb83Li17
6. **W. Ou** - Surface instability of liquid tin and gallium due to the formation and decomposition of volatile metal hydrides
7. **M. Poradzinski** - Assesment of the divertor power load for DEMO with liquid tin divertor
8. **A. Kapat** - Li-based chemical dynamics on porous W during D_2 irradiation under simulated fusion PMI conditions
9. **T. Kulsartov** - Study of hydrogen-containing sweep gas impact on the processes of tritium release from lead lithium eutectics
10. **Yu. Gordienko** - Reactor experiments on study of gaseous mixtures excitation by the products of 6LI(n,α)T nuclear reaction
11. **V. Budaev** - Arcs on the melted metal surface in fusion devices
12. **A. Zakharenkov** - Facility for investigation of heat exhaust efficiency in CPS based PFE of steady state operating tokamak
13. **X. Cao** - Interactions between helium plasma and lithiated graphite substrate coated with silicon carbide

ABSTRACTS

Section 1: Liquid Metals in Magnetic Confinement Experiments

P/S TOKAMAK LIMITS AND OVERCOMING THEM BY LITHIUM USE

S.V. Mirnov*
State Research Center of Russian Federation Troitsk Institute for Innovation and Fusion Research, Moscow, Russia

The comparative analysis of so-called “high performance tokamak regimes” gave [1] that during more as 50 years of tokamak operations the specific energy load of the first wall – P/S (where P is the total heating power of plasma which should be equal to the total energy losses for conventional tokamaks, and S – the inner area of the tokamak chamber, facing to the plasma) was approximately constant and equal 0.2+/-0.1 MW/m^2 without some kind dependence from the pulse duration Δt in the interval of "warm tokamak" operations 0.01 – 10 sec (see fig.1). However, already the first operations in superconducting tokamaks (TRIAM-1M, Tore Supra, for example) discovered that P/S should be decreased if one needs to increase the tokamak pulse duration ($\Delta t \sim 1/(P/S)1.7$) longer 10 sec (see fig.1). The mechanism of the electrical breakdown of sheath on the plasma-wall boundary can be suggested as the simplest explanation of P/S limit in conventional short pulse tokamak case [2]. This explanation can be extended at the case of Δt limiting, if we assume that the breakdown probability will increase during discharge with increase the thickness of deposition films of erosion products of limiters and chamber walls. That means the steady-state plasma process, which is necessary for the transition from laboratory devices to the fusion reactor, will require the online removal of the erosion products from the tokamak chamber. The use of the lithium circulation in the form of a closed loop between the wall and plasma promises a solution of this problem. In communication it will be discussed the last ideas and experimental achievements in this field.

[1] Mirnov S.V. Plasma Phys. Contr. Fus. 55 (2013) 045003
[2] Mirnov S.V. Plasma Phys. Contr. Fus. 58 (2016) 022001

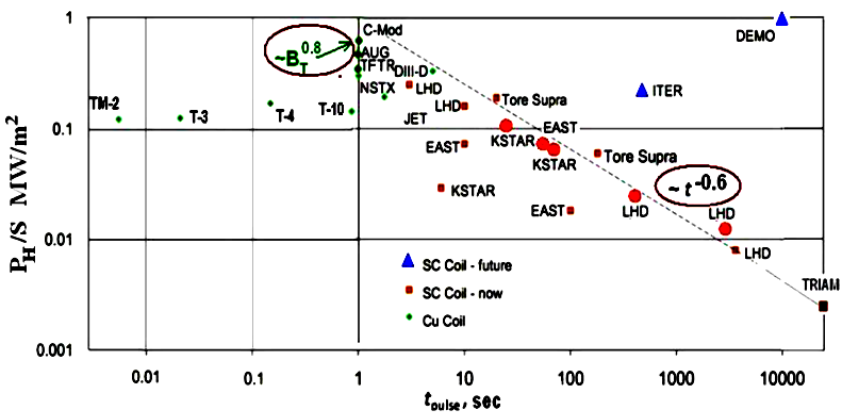

Fig.1

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COMPARISON OF THE EXPERIMENT WITH LIQUID LITHIUM AND LIQUID TIN IN FTU

M. Iafrati[1 *], G. Mazzitelli[1], M.L. Apicella[1], G. Apruzzese[1], J. P. S. Loureiro[2], I. Lyublinski[3], A. Vertkov[3], A. Berlov[3], and FTU team**
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[2]*Instituto de Plasmas e Fusão Nuclear, 1049-001 Lisboa, Portugal*
[3]*JSC "RED STAR", Moscow, Russia*

Liquid lithium and tin have been studied in the Frascati Tokamak Upgrade (FTU) a compact high magnetic field device able to achieve an high power flux close to the LCMS. The behavior of such materials is very dissimilar starting from the strong difference in the atomic weight. However the operating window for tin is much larger than that for lithium due to its lower vapor pressure at given temperature. The possibility to increase the operative temperature, and, thereby, the energy removal capacity, allows to increase the steady state heat load on the surface up to very high value[1]. FTU is the first tokamak in the world operating with liquid tin limiter and one of the pioneers in liquid metal applications. Furthermore a set of dedicated diagnostics aims directly at the limiters: four Langmuir probes monitor density and electron temperature close to the limiter, a fast IR-camera the surface temperature and one spectrometer detects visible radiation.

Both the limiters have been exposed to several plasma discharge and disruption without any permanent damage. The plasma parameters have been largely varied with $I_p = 0.3 \div 0.7$MA, $B_t = 2.5 \div 5.3$ T and $n_e = 0.4 \div 1.5 x 10^{20} m^{-3}$.

In this work a comparison between plasma performance with tin an lithium will be presented with particular regards to the plasma core modification.

[1] J.W. Coenen et al., Phys. Scr. T159 (2014) 014037

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SUMMARY OF UPGARDE AND EXPERIMENTS OF LITHIUM SYSTEMS ON EAST TOKAMAK IN THE LAST TWO YEARS

J.S. Hu[1,*], G.Z. Zuo[1], Z. Sun[1], W. Xu[1], R. Maingi[2], Q. X. Yang[1], M. Huang[1], Y. Chen[1], X.L. Yuan[1], X. C. Meng[3], J.G. Li[1], A. Diallo[2], R. Lunsford[2], D. Mansfield[2], T. Osborne[4], K. Tritz[5] , Z. Wang[6], and EAST team

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In the last two years, lithium (Li) systems, i.e. Li coating, flowing liquid Li limiter (FLiLi) and Li aerosol/granule injection, have been upgraded. Compared to the former systems, the Li coating system was upgraded for more uniform coating for more effective to suppress impurities and to reduce recycling. The FLiLi was upgraded with some new designs and technologies to increase the flowing liquid Li uniformity on stainless-steel plate, heat transfer capacity of its copper heat sink and also the resistance of the erosion. The Li granule injection was also upgraded to allow four kinds granules of various size with an easy control.

Using those upgraded system, new experiments have been successfully carried out in EAST tokamak with an ITER-like W/Cu divertor in 2016. Some new interesting results have been obtained. The uniform lithium coating using new dedicated ovens and real time Li powder injection could effectively suppress W impurity coming from W divertor to avoid W impurity accumulation in plasma core, which make a great contribution on the achievement of minute-scale H-mode operation [1]. In addition, reproducible higher performance ELM absent H-mode discharges with high power RF heating by Li powder injection and robust ELM pacing by Li granules injection were demonstrated in ITER-like wall plasmas, extending previous results on the graphite divertor [2]. Moreover, it should be noticed that using the upgraded FLiLi system, the coverage of flowing Li increase to 80% and no obvious damage of SS plate was found, which is much better than that in the first experiment in 2014 [3]. Due to that Li burst was effectively suppressed, which make it is more compatible with H-mode plasmas with high parameters.

Those interesting experiments results of new upgraded Li systems, coupled with better understanding of physics of injected mass in plasmas, would be beneficial for the enhancement of Li application for wall conditioning and ELM mitigation in tokamaks with W walls and for the increase of the possibility as alternative PFM in future fusion devices.

[1] B.N. Wan, Overview of EAST Experiments on the Development of High-Performance Steady-State Scenario, 26th IAEA FEC, OCT.17-22, 2016, Kyoto, Japan
[2] J.S. Hu, Z. Sun, H.Y Guo et al., Phys. Rev. Lett. 114 (2015) 055001
[3] J.S. Hu, G.Z. Zuo, J. Ren, et al., Nucl. Fusion 56 (2016) 046011

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TESTING LIQUID METAL-CPS PROTOTYPES BY EXPOSURE TO TJ-II FUSION PLASMAS

F.L. Tabares[1,*], E. Oyarzabal[1], D.Tafalla[1], A. de Castro[1] and the TJ-II Team

[1]*National Fusion Laboratory, CIEMAT, Madrid, Spain*

The use of liquid metals as plasma facing components (PFCs) in a future fusion reactor has been proposed as an alternative to solid metals, such as tungsten and molybdenum among others [1]. The expected advantages for the power exhaust issues, mainly arising at the divertor target at power densities of 10^{-20} MWm^{-2}, relay on the self-healing properties of liquid surfaces as well as the ability to in situ replacement of the surfaces exposed to the plasma by the effect of capillary forces (CPS design, [2]). Among the possible liquid metals (LM) presently considered as candidates for the development of an alternative solution to the Power Exhaust Handling in a future Fusion Reactor (Li, Sn), tin lithium alloys offer unique properties in terms of evaporation, fuel retention and plasma compatibility [3]. Recently, LiSn (20-30:80-70at.%) alloys have been exposed to ISTTOK and TJ-II and very promising results on D retention and surface segregation of Li were obtained [4,5]. Thus, a full campaign of comparative Li/ LiSn/Sn testing in TJ-II plasmas has been initiated. Solid and liquid samples have been exposed and a negligible perturbation of the plasma has been recorded in the Li and LiSn cases. The surface temperature of the liquid metal/CPS samples (made of a Tungsten mesh impregnated in SnLi or Li) has been measured during the plasma pulse with ms resolution by pyrometry and the thermal balance during heating and cooling has been used to obtain the thermal parameters of the LM/CPS arrangements as well as to calculate the thickness of the film interacting with the plasma. Temperatures as high as 1100K during TJ-II plasma exposure were observed for the LiSn case while temperature excursions consistent with the nominal value of the thermal conductivity of Li/W mixture were recorded for liquid Li exposure. On the other hand, the temperature dependence of the Li signal in the plasma indicate a binding energy of the atom much lower than that deduced from the vapor pressure values under vacuum for both LM's.

In this presentation a full account of the results obtained and their implications for the use of LM/CPS concepts in a future Fusion Reactor will be addressed. Particular attention will be given to the optimization of the thermal properties of the proposed designs.

[1] Nygren R and Tabarés F L 2016 Nucl. Mater. Ener. 9 6
[2] Mirnov S V et al 2011 Nucl. Fusion 51 073044
[3] M.A. Abdou et al,Fusion Eng. Des. 54 (2001) 181–247
[4] J. Loureiro et al. Fus. Eng. And Des. (2017)
http://dx.doi.org/10.1016/j.fusengdes.2016.12.031
[5] F. L. Tabares et al., Nuclear Materials and Energy 000 (2016) 1-6.
http://dx.doi.org/10.1016/j.nme.2016.11.026

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FIRST EXPERIMENTAL RESULTS OF HEAT LOADS ON TIN LIQUID LIMITER ON FTU

M.L. Apicella[1*], M. Iafrati[1], G. Mazzitelli[1], G. Apruzzese[1], A. Buscarino[2], C. Corradino[2], F. Crescenzi[1], L. Fortuna[2], G. Maddaluno[1], A. Mancini[1] and FTU team**

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At the end of 2016 and for the first time in a tokamak device, a Capillary Porous Tin Liquid Limiter (TLL) was exposed to the plasma on FTU (Frascati Tokamak Upgrade). Tin has been pointed as the best candidate due to its larger operational window ($300 < T_{SURFACE} < 1300$ °C), with low or negligible activation and low H retention. On the other side, tin is a high Z material (Z=50) that could be unfavorable for the plasma.

To characterize TLL under FTU plasma discharges, the following diagnostics have been used: 1) a fast infrared camera observing the whole limiter surface 2) 4 Langmuir probes for measuring the electron temperature and density on both sides of the limiter surface and at two different poloidal positions 3) thermocouples for the measurement of the TLL temperature 4) a visible spectrometer looking at the limiter for the monitoring of Tin and D line emission. A software tool was developed to correct the emissivity map of the IR surface temperatures by taking into account that liquid tin is not a black body and that the oxidation state of the tin surface can be different for experiments performed in different days. The most reliable correction map was extracted from the IR images at the phase transition from the solid to the liquid state as evidenced by the discontinuity of the temperature trend during the cooling phase of the limiter. Thermal analysis has been carried out by applying the ANSYS code to the real geometry of the TLL. In the simulation we have taken into account: the shape and the position of the FTU plasma as reconstructed by the equilibrium code, the net power to the SOL (P_{input}-P_{rad}) and the decay length λ of the energy as obtained by the Langmuir probes at different TLL radial positions. In the first experiments on FTU, the tin limiter, not actively cooled, was inserted progressively in scrape-off-layer (SOL) up to a distance less than 0.5 cm from to the last closed magnetic surface (LCMS). It was exposed in ohmic plasma discharges with I_p=0.5MA, B_t=5.3 T and n_e=6.0x10^{19}m^{-3}.

A maximum heat load of 10 MW/m^2 was found by ANSYS code with the tin limiter at the distance < 0.5 cm from the LCMS in agreement with the indications coming from the Langmuir probes measurements. In this case, despite the quite high surface temperature of 1300 °C over a limited central region of the limiter, not far from the expected strong evaporation of tin, the plasma performances did not significantly change and no increase of Z_{eff} value was also observed.

In his work, a comprehensive description of these first results will be given.

**See Appendix A- Overview of FTU Results "IAEA Conference 2016 Paper OV/P-4

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EXPERIMENTS WITH LITHIUM CAPILLARY POROUS STRUCTURE IN T-10 TOKAMAK WITH TUNGSTEN LIMITERS

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For a long period of time the T 10 tokamak was equipped with rail and circular limiters made both of graphite. However the level of light impurities in the plasma was found to rise steeply when the ECRH power exceeded 2 MW. In order to reduce the impurity level in regimes with powerful heating and to get experience of operation with tungsten PFCs in 2015 both limiters were replaced with tungsten ones. However it is well known that high Z materials can have harmful consequence to the plasma operation due to neoclassical impurity accumulation. In order to mitigate negative influence of the tungsten a movable lithium limiter based on capillary porous structure [1] was mounted on the upper manifold. It can operate in two regimes. One is the covering of the chamber before discharges with lithium by means of evaporation. Second is the insertion of the lithium limiter into the plasma at different radii during discharge. It was also found that the amount of lithium, accumulated in chamber during the campaign is also influenced the plasma purity.

Experiments showed that the use of lithium limiter resulted in the decrease of tungsten density in the plasma core more than order of magnitude, while W influx into plasma decreased 3-5 times. Drastic drop of light impurities in plasma was observed together with improvement of τE and density limit values. Nevertheless lithium level in plasma remained low in both OH and ECRH regimes. In spite of strong reduction of impurities, Li density in the core as low as 0.5% of ne was obtained.

[1] EVTIKHIN, V.A., et al., Plasma Phys. Control. Fus. 44 (2002) 95.

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ELM CONTROL BY LITHIUM AEROSOL AND GRANULE INJECTION ON EAST WITH TUNGSTEN DIVERTOR

Z. Sun[1], R. Maingi[2], J.S. Hu[1*], R. Lunsford[2], T. Osborne[3], K. Tritz[4], G.Z. Zuo[1], W. Xu[1], Y. Ye[1], Q.Q. Yang[1], R. Chen[1], A. Diallo[2], D. Mansfield[2], J. Canik[5], Z. Wang[6], G.S. Xu[1], X.Z. Gong[1], X. C. Meng[1], M. Huang[1], J.G. Li[1] and EAST team

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A reproducible, fully non-inductive high-confinement H-mode regime devoid of large ELMs has been achieved by continuous Li aerosol(Dia. ~45μm)injection in EAST into the upper 'ITER-like' tungsten divertor, extending previous results on the graphite divertor[1]. These discharges did not suffer from density or impurity accumulation, and maintained constant core radiated power. The new results extend the energy confinement multiplier H98(y,2)~1.2, as compared to H98(y,2)~0.75 previously on the graphite divertor. One possible explanation is the reduced recycling which leads to an inward shift of the electron density profile relative to the electron temperature profile close to the separatrix, similar to profile shifts stabilized peeling-ballooning modes in NSTX[2]. Another possible explanation is that a low-n electromagnetic coherent mode at ~40kHz became stronger in amplitude and also more coherent. In addition, Li granules with diameters ranging from 300 microns to 900 microns are radially injected into the edge of H-mode plasmas with ITER-like wall at velocities of 50 to 150 m/sec for ELM destabilization and elimination. The experimental observations indicate that ELM triggering efficiency depends on many interwoven parameters, such as granule size, penetration depth, and heating scheme. It was also observed Li granules injection shifted the density profile outward, which changed the characteristics of the edge fluctuations. ELMs were mitigated to 'grassy-like' ELM phase was observed with high frequency(~500Hz) small Li granules(Dia.~0.3mm) injection with enhanced edge fluctuation.

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Section 2: Liquid Metals in Magnetic Confinement Experiments Continued

INVESTIGATION OF A TIN-LITHIUM EUTECTIC AS A LIQUID PLASMA-FACING MATERIAL

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Sn-Li is a low melting-point eutectic that has been identified as a material with favorable performance in plasma material interaction (PMI) studies. While lithium is a low Z material with a demonstrated ability to absorb impinging ions, pure lithium is plagued by high evaporation rates in the liquid phase. The Sn-Li eutectic is a more stable alternative that provides a lower rate of evaporative flux due to the high vapor pressure of tin. In addition, in the liquid phase, the bulk segregation of lithium to the surface of the material has been observed. This surface segregation prevents the high Z tin from entering the plasma and results in a lithium film that can act as a low-recycling wall [1].

While the eutectic is of considerable interest to the PMI community, little data has been collected on its surface chemistry in a plasma environment. In order to expand the existing body of knowledge in this area, samples of an 80% Sn—20% Li eutectic were prepared and analyzed in both the solid and liquid phase in order to assess the surface composition and degree of lithium segregation in the liquid phase. The eutectic's rate of hydrogen retention was also investigated.

The Angle-Resolved Ion Energy Spectrometer (ARIES) at Sandia National Laboratories was used to probe the surfaces of the materials using direct recoil spectroscopy (DRS) [2, 3]. The deuterium coverage dependence was investigated as a function of temperature, and molecular dynamics (MD) simulations were used to analyze the scattering patterns and measure the height of the D atoms above the material surface. In addition, the outer surface oxide layers were analyzed using X-ray photoelectron spectroscopy (XPS).

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Spectroscopic imaging of liquid metal plasma facing materials on ISTTOK

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Handling high power loads impinging on the divertor region of fusion reactors, and generally on the first wall, is a problem upon which the success of nuclear fusion depends [1]. An alternative to solid plasma facing components (PFC) is the use of liquid metals such as lithium, gallium or tin due to the self-healing properties of the liquid surface and the vapour shielding effect. However, the use of these materials in fusion reactors depends among other factors on the discharge performance degradation induced by the enhanced impurity contamination.

Samples of pure Sn (99.999% Sn) and Li-Sn alloy (30 at.% Li) were recently [2] exposed to deuterium plasmas at the ISTTOK tokamak [3,4]. The edge of the plasma column was characterized by spectroscopic measurements. The spectroscopy setup is composed of a spectrometer (CVI 1/4 Digikröm), a fast EMNCCD camera (iXon Ultra 888) and optical alignment system fitted to the vessel horizontal port. The optical connection between port and spectrometer was achieved using a bundle of ten optical fibers, aligned with the poloidal plane of the vessel. This 10 chord geometry allows simultaneous measurements of the spectral emission spanning from the scrape-off layer to the inner part of the plasma core. The system is focused such that each sampled volume is separated 5 mm from the adjacent. A search for the characteristic lines from the exposed elements was done in visible range. In particular, Li I line (670.8 nm) and a Sn II line (579.9 nm) were studied. Peculiarly no lines were found compatible with Sn I even at the most potentially severe conditions.

In this work the radial decay of the line intensity for these two elements is observed. Simultaneously the effective nuclear charge of the plasma was estimated and was mostly unaffected by the sample exposition. These observations suggest that the sputtered ions from samples remain mostly in the region where the sample is positioned instead of migrating to the plasma core. While this was observed in other pure Li exposure experiments [5] it was the first time for pure Sn and Li-Sn alloy.

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A COMPARATIVE STUDY OF THE EFFECTS OF LIQUID LI AND SN AS DEMO DIVERTOR TARGETS ON THE HEAT LOADS AND SOL PROPERTIES

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The behaviour of the European DEMO SOL when either liquid Lithium or liquid Tin are used as target materials is analysed for the standard single null configuration by means of the 2D edge code TECXY. The code has been recently upgraded in order to consider liquid targets, which are modelled as liquid metal layer superimposed on a tungsten substrate. The liquid faces the plasma and then is hit directly by the plasma particles while the W bottom is kept at a fixed temperature by the cooling system. By solving the heat transport equation the temperature at the target top is calculated and the evaporation rate derived. The total impurity source strength is then estimated by including also sputtering. The impurity concentration and the involved radiative losses are calculated self-consistently by solving the multifluid plasma transport equations. The input power in the edge region coming from the core is fixed to 150 MW as suggested for the present DEMO version. Different situations are considered for both Li and Sn in terms of the coolant temperature and thickness of the W substrate. Mitigation is quite sensitive to the chosen boundary conditions. The cooling effects may be quite strong, even exceeding a total of 90 MW radiative losses, for both liquids if proper parameters are chosen. However for achieving conditions close to detachment additional impurities seem to be necessary, at least for Li. Contamination of the core plasma appears to maintain acceptable in the parameter range where a stable solution is obtained.

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INVESTIGATION OF DEPENDENCE OF LITHIUM AND HYDROGEN COLLECTION BY COLLECTOR TARGET ON TEMPERATURE OF TARGET SURFACE IN EMITTER-COLLECTOR SYSTEM ON T-11M TOKAMAK

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One of the key elements of the closed lithium loop concept near the first wall of the steady-state fusion neutron source (FNS) is a lithium collector for the collection of lithium "waste" and unused "fuel" (isotopes of hydrogen) for subsequent return to the plasma column. An important feature of the lithium collector efficiency is the degree of trapping of the falling lithium ions flow on collector. For example, for a smooth metal collector of stainless steel in the case of cooling it with liquid nitrogen, this coefficient could be considered close to 1, at least during the duration of exposure of 600 operating pulses of T-11M (it is equivalent to about 150 s of continuous operation of the collector). Under steady-state operating conditions, the collector target cooling with liquid nitrogen becomes unprofitable, and it should be ready for the transition to other ways of cooling and, respectively, to other working temperatures of lithium collectors surfaces. Therefore, the question of the tolerance limit of surface temperature deviations from the cryogenic level is important in real conditions, when the collector surface have been already covered with the active layer of the absorbed lithium. Thus, the goal of this work was to study the efficiency of a smooth metal coating and capillary porous structure as a collector with respect to the incident lithium and hydrogen ions, namely, the determining of the acceptable temperature range for lithium and hydrogen collection by the lithium target.

Experiments carried out on T-11M tokamak in typical discharges: I_p = 70 kA, B_T = 1.4 T, t = 200 ms, $n_e = 3 \cdot 10^{19}$ m^{-3}. A vertical lithium limiter of T-11M was used as a Li emitter and a longitudinal lithium limiter was used as a Li collector. A movable collector target placed in the "shadow" of the vertical lithium limiter in the SOL zone was introduced in the tokamak chamber to a distance of about 10 cm. In addition, in the shadow of the main emitter a graphite limiter is located, which serves as an indicator of the lithium flow at the periphery of the plasma column.

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PLANS FOR LIQUID METAL DIVERTOR MODULE TESTS IN THE COMPASS TOKAMAK

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The COMPASS tokamak (R_0=0.56m, a=0.2m, B_0=1.3T, I_p~400kA, t_{pl}=0.3s) operates in ITER-relevant plasma shape in H-mode with Type-I ELMs. In 2018/19, we plan to install into the divertor a capillary porous system module – a mesh sponge filled with liquid lithium/tin/lithium-tin alloy on a single plate with internal liquid reservoir at melting point (200°C). This single target will be inclined toroidally in order to be exposed to significant plasma heat flux. Preliminary calculations predict (for 25° inclination) the surface temperature reaching values up to 700°C within 100ms of the standard ELMy H-mode heat flux with ELM filaments above 10^7W/m^2. Significant lithium vapor generation is then expected, with back lithium condensation monitored by a quartz micro-balance system, the target surface observed by a lithium-line- filtered fast visible camera, fast infrared camera, fast divertor probe array and thermocouples. The scientific programme will be focused on operational issues (window coatings, ejection of droplets, etc.) as well as on the effect on the plasma physics (e.g., change in plasma confinement, L/H power threshold, global Z_{eff}, etc.). The contribution describes the technical and scientific objectives as well as design of the liquid metal systems.

After 2024, a full LM divertor will be installed into the planned COMPASS Upgrade tokamak (R_0=0.84m, a=0.3m, B_0 = 5T, I_p = 2MA, t_{pl}~2s). In this device, a full toroidal closed liquid metal divertor will be exposed to ITER-relevant heat fluxes.

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OBSERVATION OF PLASMA PROCESSES IN SOL REGION OF T-11M TOKAMAK WITH LITHIUM LIMITER BY FAST VIDEO REGISTRATION

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A new high-speed color camera with exposition time up to 4 μs and 300-1000 fps (frames per second) and narrowband interference filters was installed for fast video recording of plasma-surface interaction with a Lithium limiter on the base of capillary-porous system and target-collector in T-11M tokamak vessel. This new technique gave us a possibility to observe many new phenomena near the surface of the lithium limiter especially when viewed fast transient processes and also to observe in SOL region and near the surfaces of the target-collector. So the subject of our study was:

1.Determination of radial distribution of Lithium flux in the SOL by neutral Li line emission near the target;

2.Effect of MHD instabilities on CPS Lithium limiter and lithium drops generation;

3.The distribution of Lithium Li+ ion line emission in the plasma boundary and filamention and MHD instability development;

4.Generation and evolution of Lithium bubbles on the surface of liquid lithium limiter.

We observed an interesting new phenomenon on the surface of the capillary-porous system with liquid lithium during a stable tokamak discharge, which can be called an analogue of the blistering.

Video recording of the recombination target-collector (probe) installed in the tokamak vessel with a narrow band optical filter allows to determine the radial profile of the flux of Lithium and Hydrogen ions during time exposition up to 40÷100 μs.

The paper presents new results of the study of tokamak plasma interaction (frame exposure time up to 4 microsecond) with CPS Lithium limiter in a stable stationary phase, unstable regimes with internal disruption and results of processing of the image of the light emission around the probe, i.e. e-folding length for neutral Lithium penetration (λ ~1.3 cm). and e-folding length for Lithium ion flux in SOL region (λ ~2÷3 cm).

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TECHNIQUE OF REACTOR EXPERIMENTS ON STUDY OF LITHIUM CPS INTERACTION WITH DEUTERIUM UNDER NEUTRON IRRADIATION

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Problems of plasma-facing materials degradation and in-vessel element destructions, tritium accumulation and plasma pollution can be overcome by the use of liquid metals with low atomic number. The best candidate as a material for divertor receiving plates and other in-vessel devices is lithium. Liquid lithium advantages as a plasma facing material have been confirmed by a large number of the experiments at the plasma-physical facilities being operated worldwide.

One of the problems associated with the use of such liquid lithium systems in the fusion reactors is to determine the parameters of the working gases interaction with plasma facing surfaces under conditions simulating real operation, i.e. under conditions of neutron and gamma radiation. Therefore, one of the important goals is to research the processes of hydrogen isotopes interaction with lithium capillary-porous systems (CPS) under reactor irradiation. The first phase of the studies was to develop a technique for lithium CPS sorption experiments at deuterium pressure of 0.1 – 100 Pa and samples' temperatures of 100 – 800°C.

This paper describes a technique of the reactor experiments to study lithium CPS interaction with deuterium under neutron irradiation. The radiation source was IVG 1.M reactor of National Nuclear Center RK. In particular, the neutron-physical calculations were performed by using the MCNP software, which allowed to estimate the generation rates of 3H and He in lithium CPS during its irradiation in IVG.1M reactor, and to determine energy release during ions thermalization. ANSYS software packages were used for thermophysical calculations of an irradiator for various temperature regimes of lithium CPS, and the distribution of temperature fields in the experimental cell was determined for different reactor power levels. The neutron-physical and thermophysical calculations were the basis for development of the design and further manufacture of a unique irradiation ampoule device with a lithium CPS sample. A number of the experiments was performed to calibrate deuterium fluxes through an experimental cell with lithium CPS and preliminary results of these experiments were obtained. The new design of the experimental cell allows to carry out the experiments on tritium generation and release from lithium CPS under conditions of reactor irradiation at the temperatures above 500°C. The research is carried out in the framework of ISTC K-2204 project.

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INVESTIGATION OF EDGE PLASMA PARAMETERS IN T-11M TOKAMAK

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Lithium is being considered as a plasma-facing material, and its properties are being actively studied at present time. New experiments with lithium have been conducted on the T-11M tokamak.

In previous experiments on T-11M, the radial distribution of the lithium ion flux onto a collector target placed in the shadow of the main lithium limiter was obtained [1].There was a strong asymmetry between the “ion” and “electron” sides of the collector target. That asymmetry could be explained by the assumption that the plasma rotated toward the ion drift direction with a velocity close to that of the lithium ion sound.

To test this assumption, a Mach probe was installed in T-11M. It is composed of two single electrical probes separated by a metal plate. The probe’s angle of rotation and insertion depth with respect to the vacuum chamber were controlled with a feedthrough.

The angular and radial distributions of the ion saturation current and the radial distribution of the ion and electron temperature in L- and H-regimes were obtained. In L-regimes, the radial distributions of the ion saturation current and the electron temperature were found to have maxima at the insertion depth of 4 cm. This could be connected with a magnetic island formation on the vertical limiter. In H-regimes, no such maxima were observed.

The angular distributions of the ion saturation current showed that the ion saturation current from the “ion” side of the probe was higher than that from the “electron” side.

Finally, using the formulae from [2], the plasma flow velocity was calculated. It was found to be close to half the sound velocity of lithium ions.

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Section 3: Liquid Metal Laboratory Tests

LITHIUM LOOP AND HYDROGEN ISOTOPE EXTRACTION TECHNOLOGIES FOR USE WITH A LiMIT-STYLE PFC

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As lithium has grown in popularity as a plasma-facing material, efforts have been placed on examining its viability as a first wall candidate. Lithium has proven over numerous studies to improve core confinement, while allowing access to operational regimes previously unattainable while using solid, high-Z divertor and limiter modules [1, 2]. These benefits are due to the fuel retention capabilities of lithium, which allow it to be an almost ideally absorbing boundary. While this important characteristic of lithium is known to be advantageous, it is also detrimental with regards to tritium inventory and safety concerns for reactor scale operations. As such, extraction technologies for the recovery of hydrogenic isotopes captured by lithium require development and testing at both the pilot scale and the demonstration scale. Any extraction technology must also be verified within the scope of a larger scale lithium loop system that separates lithium impurities, recovers deuterium and tritium, and recycles clean liquid lithium back to the plasma-material interface.

Efforts at the University of Illinois at Urbana-Champaign (UIUC) have been dedicated to determining the most efficient method for fuel reclamation and lithium recycling for eventual installation in a full LiMIT-style [3] loop system. These technologies will take advantage of the thermophysical properties of the lithium- hydrogen-lithium hydride system as the driving force for recovery. Previous work reported by the Center for Plasma-Material Interactions [4] indicates that hydrogen gas release from the thermal decomposition of pure lithium hydride reaches a maximum rate of 7×10^{18} s^{-1} at 665 ^{o}C. While this is appreciable, even higher evolution rates from the Li-H-LiH solution are needed for full reactor-scale operation. The ratio of isotope dissolution to hydride precipitate formation must therefore be determined, along with the energy needed to recoup hydrogen isotopes from this system within the context of a continuously-flowing liquid lithium loop. Technologies for liquid lithium flow control developed at UIUC will be included in the loop system in order to maximize lithium treatment and minimize spilling and dryout concerns. Lithium flow velocities only reach 1-10 cm s^{-1} in large scale systems (B > 1 Tesla), therefore external electromagnetic pumps will be implemented to aid in increasing throughput. Extraction and loop technologies for use with a LiMIT-style plasma-facing component will be discussed, and results will be presented.

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A REVIEW OF RECENT STUDIES ON CONVECTION EFFECTS ON PARTICLE RECYCLING FROM LIQUID METALS UNDER PLASMA BOMBARDMENT

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Power and particle handling in the plasma edge region is one of the key issues, affecting the successful operation of a steady state magnetic fusion power reactor. Tungsten has widely been used for plasma-facing components in existing fusion experiments and is envisaged to be employed for the ITER divertor, perhaps with seeded impurity for radiation detachment. Unfortunately, conventionally available tungsten is known to suffer from cracking due to its exceptionally high DBTT (for the Ductile-Brittle Transition Temperature). Most recently, nonetheless, efforts have been devoted to develop ductile tungsten and also W-W composite materials.

To resolve the mechanical property issue with solid divertor materials, over the past decade the use of liquid metals has been proposed and implemented in a number of medium-sized confinement devices. Experimental data so far have been encouraging with improved confinement performance. However, there are tremendous uncertainties and yet-to-be explored nature about the behavior of free-surface liquids and vapors, interacting with the edge plasma, particularly under off-normal conditions such as disruption. Fluid dynamics simulation has begun only recently to understand the effect of liquid convection on hydrogen recycling, for example.

Presented in this paper are a review of the recent work on the interactions between plasmas and liquid metals such as molten lithium, the most widely used for plasma-facing components in magnetic fusion experiments, and a future perspective of the application of liquid metals for future fusion power reactors.

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THE STUDY OF THE W-Li COMPOSITE SAMPLE SURFACE SUBJECTED TO THE 100-FOLD PULSED DEUTERIUM PLASMA IRRADIATION

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Investigation of the lithium capillary-porous systems (CPS) as a plasma-facing material for fusion devices with magnetic and inertial confinement of plasma has been carried out. Composite W-Li specimens of 20×12×1 mm in size were irradiated with deuterium plasma with the use of “plasma cannon” setup in Ioffe PTI RAS (St. Petersburg). Deuterium plasma energy density was 0.25 and 0.43 MJ/m^2 (power density 22 and 41 GW/m^2 respectively), plasma pulse duration is 15 ms, and the distance between the specimen and plasma source — 220 mm. Specimens were preliminary heated up to 200°C before irradiation, the temperature of specimen during irradiation was measured by thermocouple and the temperature of irradiated surface was controlled with the help of two-color pyrometer (3.3/4.2 μm). The composite structure was studied by optical and scanning electron microscopy, local X-ray, and X-ray diffraction analyses. It is ascertained that irradiation leads to partial evaporation of Li from the surface of W and formation of a cork-like surface layer from Li_2CO_3 compound in the process of specimen cooling. Li_2CO_3 compound is formed as a result of radiation-induced interaction of lithium with the components of air environment. The surface morphology is depending on the experiment conditions including the use of spatial filter aperture of ion beam. In some cases, a continuous film of Li_2CO_3 compound or film with the parts of W net which was partially blasted under the action of a local thermal effect were observed on the surface.

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TEST OF A PRE-LOADED LIQUID LITHIUM DIVERTOR TARGET ON MAGNUM-PSI

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A novel liquid lithium (Li) divertor target has been tested in the linear plasma device Magnum-PSI [1], which was recently upgraded with a 2.5 T superconducting magnet as originally intended [2]. This is the first divertor target featuring an internal lithium reservoir. It is designed for the National Spherical Torus Experiment Upgrade (NSTX-U) [3] and provides a larger amount of Li than previous designs [4]. In the target Li from the reservoir is supplied to a textured TZM surface via a capillary channel. Targets have been tested with a surface textured with electrode discharge machining (EDM), a wire mesh, or a combination of both. The objective of the design is to enable experimentation in high-performance fusion plasmas.

The targets were exposed to 10 s pulses of helium plasma with increasing power density while actively water-cooled. An IR camera and pyrometer monitored the surface temperature while plasma radiation was monitored by spectrometry, bolometry and a filtered (Li I 670.8 nm) visible camera. Damage to the target was assessed in post-mortem inspection. The power flux density to the surface is estimated from Thomson scattering (TS) near the plasma source and the temperature rise on a reference (i.e. not filled with lithium) TZM target.

We firstly observed that, where the reference target shows a continuously increasing surface temperature exceeding 1400 °C, the Li-loaded target reaches a plateau temperature (Tp) in the range 600 – 900 °C, the level of which depends on the power load density. During steady state load conditions quasi-periodic fluctuations of Tp are observed. The phenomenology is very similar to that in experiments with liquid tin [5,6], where it is attributed to 'vapour-shielding'

Secondly, we observe that the targets with a wire mesh withstood a power density of 9-14MW/m^2, before a hole is created in the mesh. Interestingly, it appears this is merely a technical issue, and the vapour-shielding continues to be observed with the damaged mesh. In a 80 s long discharge it was seen that the temperature plateau persists as long as the lithium supply stretches.

In the paper we shall also show results where the recorded 670.8 nm light was used to investigate the distribution of Li in the plasma and the possible interaction with the oscillating surface temperature mentioned earlier. Using the spectroscopic data it is also investigated whether the lithium prevents erosion of the TZM substrate. Finally, data from calorimetry, TS, and a newly installed bolometer is used to determine the radiated power from the near surface plasma due to the presence of lithium.

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PERFORMANCE OF TIN AND LITHIUM CPS TARGETS IN RESPONSE TO HIGH HEAT-FLUX PLASMAS IN PILOT-PSI AND MAGNUM-PSI

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The use of liquid plasma facing components (PFCs) is gaining interest in solving one of the most pressing issues in fusion engineering: providing an efficacious plasma facing surface between the extreme plasma heat exhaust and the structural materials of fusion devices. For solid PFCs, preserving long-term integrity in response to energetic transients such as ELMs are a tremendous challenge [1]. Although the CPS concept is promising as a liquid PFC in terms of the relative simplicity of implementation and testing, understanding its power handling capabilities, notably for Li, is still limited. Especially lacking for liquid CPS targets is the investigation of resilience to ELM-loading and the role of the vapour shielding phenomenon during these events.

Sn and Li CPS experiments performed in both the Pilot-PSI and Magnum-PSI linear plasma generators are discussed. This entails both steady-state high-power exposures (1-20 MW m^{-2}) by He plasmas on Li CPS targets and ELM-simulated plasmas in H on Sn CPS samples. A self-regulatory heat flux mitigation phenomenon resulting from vapor shielding has been previously observed for Sn in steady-state [2,3]. Similar studies of the potential of vapour shielding in reducing the power impinging the liquid surface are now conducted in the case of Li.

Magnum-PSI is currently equipped with a newly developed bolometry system which allows for investigating the radiated power produced by the Li vapour cloud in front of the target. The conversion fraction of incoming power from the plasma into radiated power by the Li vapour in the vicinity of the CPS target is compared to solid reference targets using the aforementioned diagnostic. In addition to that, the role of vapour shielding during millisecond ELM-like transients on Sn CPS targets is addressed. The integrity of the capillary structure and solid substrate upon receiving such transients is finally discussed.

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Section 4: Liquid Metal Properties

UNDERSTANDING THE PERFORMANCE LIMITS OF LIQUID METAL BASED PLASMA FACING COMPONENTS FOR POWER PLANT DIVERTORS USING LINEAR PLASMA DEVICES

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A wide variety of innovative concepts for liquid metal (LM) utilizing plasma facing components (PFCs) have been proposed for DEMO and future fusion power plants [1-6]. These proposals rely on the inherent benefits of a liquid plasma facing surface, including avoiding erosion or surface damage as a lifetime limit, providing additional heat loss channels (e.g. via fast flow, evaporative cooling and/or radiative protection) as well as resilience against neutron loading. For Li improvements in plasma performance due to wall conditioning and Z_{eff} reduction are also well documented [7-10]. On the other hand most designs are pre-conceptual and stringent limits on tritium retention, core impurity accumulation and material compatibilities provide strict power and temperature ranges which are not clearly established. In order to develop a LM PFC as a timely and viable alternative for the DEMO divertor these parameters must be satisfactorily addressed. Linear plasma devices can give insight into many of these questions under well diagnosed reactor relevant exposure conditions.

In order to better understand these performance limits a variety of experiments have been carried out recently at the DIFFER linear plasma devices, including on power handling [11], vapour shielding [12], retention [13] and erosion [14]. This work will provide an overview of recent results, and discuss how they can help in extrapolating the potential performance range and limits of LM PFCs in a reactor divertor.

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TEMPERATURE DEPENDENCE OF LITHIUM FILM FORMATION AND DEUTERIUM RETENTION ON HOT W SAMPLES BY LIDS AND TDS. IMPLICATIONS FOR FUTURE FUSION REACTORS

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The development of magnetic fusion reactors needs to solve the challenging power/particle exhaust issues avoiding unacceptable damage to the Plasma Facing Materials (PFM's) that would limit their useful life and the feasibility of such power plants. Although nowadays the use solid tungsten (W) components constitute the main investigated option for these requirements, there are serious concerns over their limitations, especially for the case of unmitigated heat flux and transients handling on the divertor. Liquid metal (LM) divertor concepts explore an alternative solution to the challenging power/particle exhaust issues in future magnetic fusion reactors. Among them, lithium (Li) is the most promising and studied material [1]. Its employment has shown important advantages in terms of improved H-mode plasma confinement and heat handling capabilities [2]. In such scenario, a possible combination of tungsten at the first wall and liquid Li at the divertor could be an acceptable solution, but several issues related with this material compatibility must be investigated. In particular, the co-deposition of Li and hydrogen isotopes on W components could increase the associated tritium retention and might represent an important safety risk. In this work, the co-deposition of Li and deuterium (D) on tungsten at different surface temperature (200ºC-400ºC) has been studied by exposing W samples to Li evaporation under several D_2 gaseous environments. Deuterium retention in the W-Li films has been quantified by using Laser Induced Desorption Spectroscopy (LIDS). Additional techniques as Thermal Desorption Spectroscopy (TDS), Secondary Ion Mass Spectrometry (SIMS), profilemetry and Flame Atomic Emission Spectroscopy (FAES) were implemented to corroborate the retention results and for the qualitative and quantitative characterization of the W-Li films. The results show a negligible (below the limit of detection) D uptake by the W-Li layer at $T_{surface}$=225ºC, when it is exposed to simultaneous Li evaporation and low pressure (0.67 Pa) D_2 gas exposition. Pre-lithiated samples were also exposed to higher D_2 pressures (133.3 Pa) at different superficial temperatures (200ºC-400ºC). A non-linear drastic reduction in the D retention was found for increasing temperatures on the W-Li films that determined D/Li atomic ratios lower than 10^{-4} at 400ºC. Based on the obtained results, an extrapolation of the D co-deposition on W-Li first wall areas in DEMO reactor designs is performed, showing that the associated fuel retention in a hot first wall may be compatible with the tritium inventory limitations.

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CO-DEPOSITION OF Li-D LAYERS AT ELEVATED TEMPERATURES

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Co-deposition of lithium with hydrogen isotopes from plasma can lead to accumulation of large quantities of radioactive tritium in future fusion devices, and can affect functionality of liquid wall using installation if solid precipitates of hydrides form in the liquid. It was previously shown that large amounts of deuterium, up to tens of atomic percent can accumulate in co-deposited layers. Accumulated deuterium is mostly released from Li-D co-deposits at relatively low temperatures of about 700 K [1,2]. These experiments were performed at room temperature. In this work, we studied co-deposition of lithium and deuterium at elevated temperatures of the substrate.

The experiments were carried out in the MD-2 installation. Li-D films were formed onto a heated-up molybdenum substrate using DC magnetron discharge in deuterium. The substrate temperature was varied from about RT to 773 K. After that, in-vacuo thermal desorption spectroscopy (TDS) analysis was performed to observe D content in the obtained layers. The deposition rate was controlled by quartz microbalbance, placed near the sample. To account for possible evaporation of lithium, the thicknesses of the films were measured outside the chamber in reference experiments.

It was found that in absolute amounts D accumulation via co-deposition drops significantly at temperatures of about 500-550 K, however, the major peak in TDS spectra was still about 700 K. At substrate temperatures above 700 K, there were still some amount of D remaining in the film, and this deuterium desorbed during TDS at 800-1100 K temperature range.

High temperature release (>700 K) of deuterium from Li-D films, but with a small contribution, was observed also in our earlier experiments with deposition at RT. A series of experiments with a step-wise TDS has been carried to get more careful data about this part of spectra. At each stage, the sample was linearly heated up to a given temperature, higher at each stage, and held there for thirty minutes. After that, the sample was cooled to room temperature, removed from the analysis chamber, which was baked off for several minutes, and then placed back in it for the next stage of the analysis. It was found that from 5 % to 10 % of all deuterium contained in Li-D layers deposited at room temperature is released at temperatures above 800 K in our conditions.

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STUDY OF THE CORROSION BEHAVIORS OF 304 STAINLESS STEEL EXPOSED TO STATIC LIQUID LITHIUM

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The investigation of corrosion behaviors of stainless steel served as one kind of structure materials exposed to liquid lithium (Li) is one of the keys to apply Li as potential plasma facing materials (PFM) [1,2] or blanket coolant in the fusion device [3]. The corrosion experiments of 304 stainless steel were carried out in static liquid Li at 600 K and 640K for 1320 hours under high vacuum with pressure less than 10^{-4} Pa. The results show that the weight loss rates were 5.3×10^{-4} g/m^2/h at 600 K and 2.2×10^{-3} g/m^2/h at 640 K, respectively, equivalent to 0.59 μm/a and 2.45 μm/a. The SEM and XRD analysis results show that the specimens produced a non-uniform corrosion damage by preferential grain boundary attack because of the formation of Fe and Cr carbides grains and/or Li compound at grain boundaries. Moreover, the surface damage at 640 K was more serious than that at 600K. The line scans of the cross-sections revealed a slight depletion of Cr and Fe near the surface, the thickness of corrosion layers at 600 K and 640 K were about 2 μm and 5 μm. The surface hardness of specimens increased by 49HV at 600 K because lots of carbides grains adhered on the surfaces, but the hardness of specimens decreased by 42HV at 640 K because C vast dissolved into liquid Li. The results show that corrosion behaviors of 304 stainless steel in static liquid Li were sensitive to temperature. This study is of important significance for further guiding the design and application of the components with liquid Li in fusion device.

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RESPONSE OF THE FE(001) SURFACE AND FE(001)-LI SOLID-LIQUID INTERFACE UNDER IRRADIATION

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In the future nuclear fusion reactors, the stability, especially under irradiation, of the solid-liquid interface between the first wall structure material and the liquid tritium breeder is closely related to the lifetime and safety of the reactors. In this work, the response behavior of the Fe(001) surface and the Fe(001)-Li solid-liquid interface under the irradiation of a 20keV Fe atom was comparatively studied with molecular dynamics simulation at 500K. Analysis suggests that the evolution processes of the interface and surface structure and the defects in the Fe blocks are very similar. The irradiation regions all suffered an evident damage in the interface and surface cases, where a certain number of Fe atoms were sputtered out, and then a crater-shaped pit surrounded with a “ridge” formed there. In these two cases, the retention rate of the irradiation defects is very low after the Fe blocks experienced a sufficient relaxation, irradiation did not have cause a plenty of Li atoms diffusing into the Fe block under the interface situation. The existence of liquid Li is beneficial for the sputtered Fe atoms’ clustering, and a 4.13 to 8.95Å thick liquid Li layer can partly weaken the damage, caused by the high energy Fe atom after passing through the liquid Li layer, to the Fe block.

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Section 5: Liquid Metal Theory/Modeling

PLASMA BOUNDARY AS A KEY FACTOT IN TOROIDAL MAGNETIC CONFINEMENT

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The energy confinement is the major parameter of the toroidal plasma which determines the size, magnetic field, plasma current, the external heating, the total fusion powers, and many other aspects of potential fusion reactors. Traditionally, the energy confinement is associated with the transport properties of the core plasma. Unfortunately, the reliance on the properties of the core transport is essentially exhausted. Its requirements for large size of burning plasma devices, the highest possible magnetic field and plasma current, big external heating power lead to difficult problems of disruptions, power extraction and very survival of the plasma facing components.

In reality, the confinement of the high-temperature plasma is much more sensitive to the boundary conditions, and specifically to recycling, which determines the rate of plasma edge cooling by the neutral atoms [1,2]. While the always turbulent core transport is uncontrollable, the recycling can be controlled and reduced to the level below 50 % by arranging the liquid lithium flow, in fact, creeping (1 cm/s, 0.1 mm thickness, 1 g/s flow rate). The key technology elements for it were recently invented and the first limiter, which implements the concept of continuously flowing liquid lithium (24/7FLiLi) was tested on EAST tokamak [3]. At present there are numerous experimental evidences of high sensitivity of confinement to the plasma boundary: H-mode with poloidal divertors, TFTR supershots with lithium conditioning, 4-fold enhancement of confinement on CDX-U [4] with plasma absorbing liquid lithium.

From the plasma physics point of view, the reduction of recycling below 50 % manifests the transition to a new, the best possible confinement regime, where the energy is lost only due to plasma diffusion [2] (radiation neglected). In its turn, the high edge and core plasma temperatures are determined exclusively by boundary conditions, i.e., by the ratio of the heating power to highly reduced particle flux from the plasma boundary. The thermal conduction in core (and anomaly of electrons) does not play any role in confinement. The plasma diffusion, if assessed realistically, is of the order of the ion neo-classical thermal conduction and sma This understanding implemented into a simple Reference Transport Model (RTF), described very well the energy confinement in CDX-U machine [4]. The DIII-D tokamak with RMP experiments (and then many others) confirmed the independence of the edge temperature on transport coefficients by demonstration that the magnetic perturbations cannot shake the edge pedestal electron temperature.

The entire plasma physics of the tokamaks is simplified dramatically in such a low recycling, called the Li Wall Fusion (LiWF), regime. First, the stabilization of the plasma edge was predicted [5] and confirmed in NSTX by suppression of ELMs by lithium conditioning. With flattened core temperature, there is no tendency for excitation of sawtooth oscillations and associated neo-classical tearing modes. The NBI can externally control the particle source in the core and the plasma pressure profile, thus, allowing for the control of the global MHD.

The expected ion-neoclassical core confinement could be sufficient for maintaining the plasma density by NBI. For the burning plasmas and fusion DEMO [6] this opens the opportunity for a new high-temperature/low-density LiWF regime where the fusion alpha particle energy is delivered to the side wall by synchrotron radiation of electrons, and only minor energy from NBI is directed to the divertor target plate. The implementation of such a LiWF regime also suggest a solution to severe problem of power extraction from the burning plasma.

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SOLPS4.3 MODELLING OF LITHIUM TRANSPORT AND NON-CORONAL RADIATION IN T-15MD TOKAMAK WITH LITHIUM EMITTER-COLLECTOR SCHEME IN USE

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A system consisting of two lithium limiters, one primarily emitting lithium into the edge plasma and the other primarily collecting it has been successfully tested at both T-11M and T-10 tokamaks [1]. Now with construction of the first Russian poloidal divertor tokamak T-15MD at NRC "Kurchatov Institute" [2] possible application of the lithium emitter-collector scheme to the medium-sized divertor tokamak should be considered. First and foremost the optimal position for both lithium emitter and collector should be defined. The emitter should provide as wide as possible lithium distribution through the scrape-off layer (SOL) and lithiation of the first wall. Whereas the lithium collector position should be chosen in accordance with the lithium migration pattern. The later in turn depends on the lithium transport in SOL plasma, its deposition and erosion from the first wall surface.

Another important issue that should be addressed during lithium emitter-collector application analysis is the peak heat load mitigation capabilities of lithium impurity in the tokamak edge that is directly connected to the lithium radiation loss. Despite lithium's cooling rate in coronal equilibrium is rather weak [3] non-coronal effects arising due to the short lifetime of lithium impurity ions in the edge plasma can significantly enhance lithium radiation [4]. However up to now this non-coronal enhancement factor was estimated with crude zero- and one-dimensional models only. In-depth analysis with a proper transport models is necessary to conclude whether lithium radiation loss could contribute to power exhaust in T-15MD-sized tokamak or not.

We use the SOLPS4.3 code package [5] to study lithium transport and associated radiation loss in the edge of T-15MD tokamak for different edge plasma conditions (i.e. heat flux from the core, deuterium inventory in the edge and lithium source intensity). For this purpose SOLPS4.3 was recently upgraded to incorporate lithium atomic process rates (ionization, recombination, excitation) taken from the ADAS database. In our work we define the best possible positions for the lithium emitter and collector in T-15MD and the lithium impurity inventory needed to achieve noticeable energy dissipation in the edge.

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ANALYSIS AND MODELING OF LITHIUM FLOWS IN POROUS MATERIALS

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Liquid lithium (LL) walls have been identified as a potential solution to many of the engineering problems in a magnetic fusion reactor. The flowing liquid metal could provide heat removal, elimination of erosion via constant renewal of the wall, and possible stabilization of MHD modes. The NSTX-U experiments at the Princeton Plasma Physics Laboratory are planned for testing a pre-filled LL divertor system with capillary wicking channels for supply of lithium from an internal reservoir. However, the erosion of the LL surface can be induced by plasma flow or Lorentz forces that may lead to its instability and undesirable effects such as waving, de-wetting, and lithium droplet formation. The theoretical analysis and computational modeling of the LL surface and the interaction between the plasma and lithium including transition regimes and disruption events are very important for understanding the physical mechanisms of lithium waving, splashing, and droplet formation.

The two approaches that we have developed and used to study the coupled liquid metal-plasma flows are linear stability analysis and computational modeling [1, 2]. A linear stability analysis includes the thermal effects, viscosity, and material's porosity. The expression for the relative velocity of the viscous instability in a porous material is derived. The critical plasma velocity, development and growth of waves on the LL surface as a function of wavelength, and the onset of instability are predicted. It is found that the plasma viscosity has a destabilizing effect on the LL flow. The LL flow modes are investigated as a function of material's porosity and permeability. The increase in porosity has a stabilizing effect on the LL flow. The LL flow with long wavelengths becomes more stable with the decrease of permeability.

The mathematical model is developed for treating the LL flow with free surface in a porous material as well as the coupled LL-plasma flow under the influence of an external magnetic field. The model is based on a volume of fluid (VoF) approach implemented within the OpenFOAM toolbox. We have also implemented the effects of thermal conduction and magnetic field (MHD model) into the interFoam solver. The VoF-MHD modeling is carried out for different values of plasma and LL velocity and three cases: 1) no magnetic field; 2) magnetic field is parallel to the LL flow; and 3) magnetic field is perpendicular to the LL flow. In the absence of plasma flow, the development of running waves with large wavelengths is observed on the LL surface. The parallel magnetic field damps these surface waves. It is observed that the LL waves are generated much faster at higher plasma speeds and their wavelength decreases accordingly. The Lorentz force induced by a perpendicular magnetic field accelerates the LL flow when the plasma stream becomes well coupled to the LL motion. The results of this research advance the understanding of physical processes that may limit, prevent or eliminate the LL losses from PFCs in fusion devices.

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FIRST-PRINCIPLES CALCULATIONS ON THE WETTABILITY OF LI ATOMS/CLUSTERS ON THE (111) SURFACES OF FE, W AND MO SUBSTRATES

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As a candidate material for plasma facing components in fusion devices, the flowing liquid lithium (Li) wall can continuously remove high particle and heat fluxes from plasma bombardment and self-heal the damaged wall materials. However, the steady flowing Liquid film is very difficult to get experimentally because of the bad wettability of Li on the surfaces of supported substrate metals. To improve the wettability, it is very important to study the interactions between Li atoms/clusters and the metal surfaces.

There exist several typical sites for Li atom occupancy, such as top, bridge, fcc- and hcp-hollow sites on the bcc (111) surface. In the present work, the wettability properties of Li atoms and clusters on the (111) surfaces of Fe, W, and Mo substrates have been studied with first-principles Density-Functional Theory (DFT) calculations. It is found that the adsorption energy of single Li atom in the threefold deep-hollow site (or the fcc site) is lower than those of other possible ones and therefore the fcc site is the most favored one for Li atom occupancy. Two Li atoms occupy two nearest-neighbor fcc sites and form a dimer; three Li atoms form a equilateral triangle and each Li atom occupies one fcc site and the mass center is on the top site. When the number of Li atoms in a Li cluster is more than four, the planar or flat construction is more stability than the stacking or island one on the surfaces of the supported substrates. Based on the above thermo-dynamical calculations, it can be concluded that the liquid Li is “wetting” intrinsically on the surfaces of the substrates under considered. It is possible that the impurities in liquid Li (such as H, C, N and O) and/or the oxide of the substrates make the liquid Li “unwetting”, as observed experimentally on the surface of the substrates. The effects of impurities and oxide on the wettability have also been discussed with the first-principles calculation approaches.

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STATIC ARRANGEMENT OF THE CPS: EFFECT OF CAPILLARITY OVERPRESSURE AND INTERMOLECULAR FORCES WITH THE SUBSTRATE ON THE FILM THICKNESS

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One of the most promising methods to circumvent problems concerning the plasma-wall interaction is the Capillary Porous System (C.P.S.) [1,2]. In this arrangement, a porous mesh acts as a capillary pump that replenishes the liquid metal protecting the divertor region from extreme plasma activity. In order to assess the stability threshold of the porous structure, there is need to investigate the static arrangement of the liquid metal that coats the porous matrix as a function of the liquid metal physical properties, operating conditions and the topography of the porous mesh.

To this end, our previous parametric study [3] was extended to cover a wide range of overpressures between the liquid metal reservoir and the surrounding plasma. A numerical solution was obtained via the finite element methodology that solves the Young-Laplace equation, which incorporates the interfacial tension between liquid metal and plasma, gravitational and pressure forces, and intermolecular forces between the liquid metal and porous substrate. The aforementioned intermolecular interactions are assembled in a disjoining pressure term. Two different kinds of intermolecular potential were investigated, namely, a long range attractive short range repulsive potential and a purely repulsive potential.

It was thus seen that, for relatively large overpressures, the balance between pressure and capillarity gives rise to a drop-like arrangement of the liquid metal resting on the porous substrate. Upon reducing the overpressure a liquid metal film is established with the gravitational forces determining its thickness. Further reduction of the overpressure, approaching vacuum conditions, provides a coating that almost uniformly covers the entire plasma facing region. At realistic conditions micron or even submicron coatings are established as a result of the balance between pressure and intermolecular forces. In this regime the pore size and mesh topography start playing an important role in the static arrangement. Currently the effect of substrate topography on the static arrangement is investigated, along with the impact of Lorentz forces. It is anticipated that the interplay with capillarity and intermolecular forces, in the presence of Lorentz forces that tend to pull the liquid metal out of the porous structure, will complement previous relevant studies [4] and provide the critical pore size for stability of the CPS against drop ejection.

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QUASI-2D ANALYSIS OF A BOX-TYPE LIQUID METAL DIVERTOR FOR THE DTT FACILITY: COMPARISON LI VS. SN

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The DTT facility [1], which is currently being planned in Italy, should assess the effectiveness of alternative solutions for the steady-state power exhaust problem in future fusion reactors (e.g. DEMO). The present study concerns one among the strategies which are foreseen to be tested, namely the use of liquid metals (LM) as plasma-facing materials.

Based on [2], a first conceptual design of a "box-type" LM divertor for the DTT machine was proposed in [3], considering Li as the LM, including a LM evaporation chamber (EC), which is open towards a second divertor box, i.e., the differential chamber (DC), in turn connected to the main plasma chamber (MC). A simple model of the EC and DC, including the incoming plasma heat load and a very basic treatment of the interactions of Li vapor with the plasma –e.g. the reduction of the Li vapor efflux due to ionization by the plasma- was presented, allowing to roughly determine the operating range of the system, in terms of surface temperatures and vapor pressures.

Here the conceptual design of the "box-type" DTT LM divertor is further developed, considering both the inner and the outer targets as two separate ECs, connected to a common DC. In the ECs, "condensation drivers" are foreseen to increase the LM condensation surface and to efficiently collect the condensed fluid which is to be recirculated and re-injected in the LM pool. In the paper, the new configuration in analyzed using a 2D FE model for the chamber walls and condensation drivers, coupled to a 0D description of the LM. The effect of adopting Li or Sn as the LM is investigated in terms of: 1) effectiveness of the LM vapor in radiating the parallel heat flux incoming in the divertor, 2) reduction of the LM vapor efflux towards the MC through the DC, and 3) total amount of LM to be recirculated by the external recirculation loop.

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Section 6: Liquid Metal Technology

LIQUID METAL CPS BASED PFC FOR STEADY STATE OPERATING TOKAMAK

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Application of capillary-porous system (CPS) with liquid metal (lithium or lithium - tin alloy) and effective power exhaust with high specific density (20-30 MW/m^2) at appropriate surface temperature is the promising concept of plasma facing components (PFC) for realization of steady-state fusion reactor.

Experience in development, creation and experimental study of lithium CPS based models of steady-state operating PFC with heat removal systems for T-11M, FTU and KTM is presented. The option of liquid metal, structural materials and coolant media are analyzed. The possible scheme of liquid metal PFC for DEMO type and tokamak based neutron source reactors is considered.

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EXPERIMENTAL STUDY ON LIQUID METAL FILM FLOWS RELATED WITH NUCLEAR FUSION ENGINEERING

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As plasma-facing components (PFCs), it need to endure the high heat flux from a burning plasma. Unfortunately, traditional PFCs made of solid materials such as carbon and tungsten can only remove a maximum heat of 5–10 MW m^{-2}[1]. And it is still far from the needs of fusion engineering. The newly concept liquid metal PFCs which states using a thin flowing liquid metal film to cover on a traditional solid PFCs has attracted a great number attentions. It has the advantage to removal a higher heat flux and has the potential to solve some issues in PFCs. However, due to the bad wettability of liquid metal on solid surface and some related magnetohydrodynamic (MHD) effects which happened at the situation that the liquid metal flow though the magnetic field, it is still a great challenge to form the liquid film which can flow stable and cover the solid surface completely. Some preliminary liquid metal film flow studies have been carried out both from experimental and numerical aspects [2, 3], but it also needs more quantity researches.

In the present study, we built two liquid metal loops (a low temperature Galinstan loop and a high temperature Lithium loop) to study the liquid metal film flows under the influence of a uniform horizontal magnetic field which can adjust the amplitude of magnetic field from 0 to 2T. The high-speed camera was used to capture the surface contour of film flow, while a laser displacement sensor which can emit a slice of laser on liquid film surface was used to obtain the thickness of liquid film on a fixed line. Through changing the experimental conditions such as flow rate, magnetic field strength and temperature, plenty of quantity data have been got. Results show that the horizontal magnetic field has effect of smoothing the surface of liquid metal film and adding an extra resistance to liquid metal film flow. The thickness of liquid film is increased linearly with the increasing of amplitude of magnetic field which also consistent with some previous publication data. However, through conducting film flow using Lithium and Galinstan at the same experimental condition respectively, we found that the MHD effects on Lithium film flow appear more obviously than that on Galinstan film flow.

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COMPACT SELF-DRIVEN LIQUID-LITHIUM LOOP FOR FUSION-RELEVANT NEUTRON GENERATION

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In D-T fusion devices, the neutron-induced damage to the first wall will be significant. Currently, producing 14 MeV neutrons to study this phenomenon requires a large facility such as IFMIF or the use of tritium. A compact, closed liquid lithium loop has been developed at the University of Illinois to test and utilize the Li-7(d,n) reactions. The compact, yet scalable, loop is comprised of a stainless steel trench system with embedded heating and cooling and was designed to handle large heat and particle fluxes for use in neutron generators as well as fusion devices. It operates utilizing thermo-electric MHD [1]. The benefits of this device are two-fold, 1) produce high energy neutrons for materials testing and radiography and 2) provide a self-healing, low Z, low recycling wall material. The flowing lithium will keep a fresh, clean surface allowing Li-7(d,n) reactions to occur as well as enhance the deuterium adsorption in the fluid increasing the overall neutron output. Expected yields of this system would be 107 n/s for 13.5 MeV neutrons and 10^8 n/s for 2.45 MeV neutrons. Previous work has shown that using a tapered trench design allows for an increase in velocity of the fluid at the particle strike point. The result is better heat removal, which prevents dry out of the lithium. For heat fluxes greater than 10 MW/m^2, COMSOL fluid models have shown that high enough velocities (~70 cm/s) are attainable to prevent significant lithium evaporation. Future work will be aimed at addressing wettability of lithium on stainless steel at lower temperatures to protect the permanent magnets in the system, experimentally determine the necessary velocities to prevent dry out, and determine the neutron output of the system. The preliminary results and discussion will be presented.

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THE DEVICES FOR DIAGNOSTIC AND LITHIUM COLLECTING OF TOKAMAKS T-11M AND T-10. FIRST RESULTS

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Because of the application of lithium as a tokamak plasma facing material the problem of the creation of appropriate diagnostic tools becomes actual. Devices allowing to study the transport of lithium particles in the tokamak SOL, the dynamic of lithium deposition at different temperatures of collecting surface in real time, adsorption and desorption of plasma-forming gases by lithium, effect of electric field on the collecting lithium were designed and manufactured for T-10 and T-11M tokamaks. Langmuir probes provide the scanning of the plasma parameters. These devices can be used to evacuate lithium, which deposits on the vacuum chamber walls without breaking of vacuum condition due to gateway and vacuum input based on innovative liquid-metal coupling.

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CORROSION BEHAVIOR OF HIGH-DENSITY HELIUM PLASMA IRRADIATED MOLYBDENUM IN LIQUID LITHIUM

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Molybdenum is presently considered as the primary potential substrate materials for liquid lithium plasma-facing components for its advantages of corrosion resistance and wettability for liquid lithium. However, when liquid lithium is introduced into a practical magnetic confined fusion device, the underlying substrate may be exposed to the plasma due to the destruction of lithium layer. Consequently, it is necessary to evaluate the synergy effects of the helium plasma irradiated and liquid lithium corrosion on molybdenum.

In this work, the compatibility of molybdenum, exposed to high-density helium plasma for 0, 30, 60 and 90 minutes, respectively, with liquid lithium at 623 K for 1300 hours was investigated. The high density linear plasma generator at Sichuan University had been utilized to replicate the high-density helium plasma environment. The corrosion characteristics were examined by SEM/EDS, XRD, fine scale weight measurement and Vickers hardness test.

The experimental results of helium plasma irradiation show that the morphology of molybdenum surface are from roughness, bubbles, porousness to nano fuzz by increasing the expose time. It is also confirmed that the lithium droplets are absorbed in the nano fuzz of molybdenum surface. The corrosion test results show that the molybdenum, without exposing to helium plasma, is prone to pitting corrosion. After exposing to helium plasma, the surfaces of molybdenum undergoes general corrosion and the localized corrosion is avoided. With increasing plasma irradiation times, the surface roughness of molybdenum increases, but the corrosion resistance of molybdenum does not decrease significantly.

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APPLICATIONS OF MERCURY IN THE FUSION FUEL CYCLE

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In the early 20th century, mercury was an important fluid in science and technology: It has been used extensively as working fluid in vacuum pumps and in chlorine production facilities. In gold mining, until nowadays mercury is extensively used as extraction fluid of gold. Also in the nuclear field, mercury has been considered as coolant for fission reactors and as working fluid in high temperature steam circuits.

Nowadays, mercury is banned almost completely from most of these applications as non-toxic replacement fluids or replacement processes are available. The Minamata Convention on Mercury signed in 2013 banned mercury on global scale from a large number of applications and gave very strong restrictions to e.g. mining and treatment.

In spite of this trend, there are some features of mercury which are so unique that mercury is still the first choice. A good example is the use of mercury in DT nuclear fusion experiments because of its excellent tritium compatibility. In the European DEMO reactor concept, mercury will be used in the divertor pumping system as working fluid in vapour diffusion vacuum pumps and liquid ring vacuum pumps. These pumps - needed for tritium processing in the fusion fuel cycle - are currently under development at KIT Karlsruhe.

Another application that will become important in fusion technology is the separation of lithium isotopes. In a self-sufficient fusion reactor, tritium is generated in breeding blankets out of lithium. In these blankets, a large amount of lithium-6 is needed, which has to be produced in an enrichment process from natural lithium (which is a mixture of two isotopes at an atom% ratio of Li-7/Li-6=92.6/7.4). The reference industrial process for lithium enrichment is based on the fact that Li-6 has a greater affinity than Li-7 for the element mercury. Nowadays, to the best of our knowledge, no facility that could supply the demand of a fusion reactor (some ten tons) is operated and would be accessible, which is a high threat to the success of fusion technology.

This paper outlines the importance of mercury in fusion technology and gives an overview of technologies in nuclear fusion where mercury will play a major role. Furthermore, it outlines gaps on the way to safe and reliable mercury processes and gives an overview of ongoing activities in this field.

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EXPERIMENTAL STUDY OF INTERACTIONS BETWEEN LIQUID LITHIUM FREE SURFACE AND PULSED CURRENTS AND MAGNETIC FIELDS

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General Fusion is developing a magnetized target fusion reactor in which a spherical tokamak plasma target is injected by a magnetized Marshall gun into a cavity lined with liquid metal. A compression system will then collapse the cavity to compress and heat the target. Development is currently at the component level, where plasma targets are injected into a solid metal flux conserver with standard plasma diagnostics. Experiments in liquid metals and cavity formation and collapse are underway.

One such experiment is focused on the interaction between liquid lithium and pulsed currents and magnetic fields. In this subscale experiment, a 20 cm diameter spherical vacuum vessel containing a static inventory of liquid lithium with a free surface will be exposed to pulsed currents of several hundred kA, with rise time of 50 microseconds, representative of plasma operation. Modelling suggests the lithium will be strongly accelerated due to differential magnetic pressure, and managing the resulting flow without quenching the plasma is critical to reactor design. The experiment will run at temperatures up to 500C, and with a variety of current waveforms and liquid levels. The experiment will also be used to evaluate features aimed at controlling the flow and protecting the plasma and the vessel, enabling repetitive operation.

Preliminary results for the experiment will be shown compared with MHD model predictions. Next steps will also be discussed, including baffles and other protection features.

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POSTERS

ISLA -5

EXPERIMENTAL TECHNIQUE FOR STUDY OF LITHIUM-CONTAINING MATERIALS UNDER NEUTRON IRRADIATION AT THE IVG1.M REACTOR (KURCHATOV, KAZAKHSTAN)

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In advanced fusion facilities, such as ITER and DEMO, tritium will be used as a fuel. Since tritium inventory in nature is very limited, the production of tritium in the fusion power plant itself is one of the priority research tasks. To solve this problem it is planned to use lithium-containing materials (LCM) of different classes: liquid metals, solid metallic materials and ceramics in the fusion reactor blanket. It should be noted that in addition to tritium generation, there is one promising field of application of lithium and lithium capillary-porous systems (CPS) in the controlled fusion devices - their use as a plasma-facing material [1]. LCM as structural elements will work in neutron fields, so there is a problem associated with elucidating the mechanisms of hydrogen isotopes interaction and models development that allow describing the processes of tritium generation, diffusion and release under neutron irradiation.

Studies of LCM under conditions of complex impact of temperature, hydrogen isotopes and neutron irradiation, existing during fusion reactor operation have a high actuality. There is a very few experimental data available on LCM under such conditions, what shows necessity to create techniques for conducting of reactor experiments and to develop the existing ones. The presented study included a description of methodological and hardware base creation for providing reactor studies of LCM, carrying out the calculations and methodical experiments confirming the possibility of using the proposed techniques in the experiments at the IVG1.M reactor. The developed techniques are based on sorption, desorption and permeability methods. The paper also describes the schemes of experiments, experimental facilities, reactor ampoule devices and experimental conditions. The results of experiments confirmid the applicability of the developed techniques for conducting studies of LCM under neutron irradiation at the IVG1.M reactor.

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RADIATION TRANSPORT IN EDGE LITHIUM PLASMA

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Radiation transport in edge plasmas contributes both to energy fluxes coming to the first wall materials and plasma composition through effects of photon excitation, photon ionization etc. E.g., it is known that divertor plasma in typical tokamak conditions is opaque for hydrogen Layman-alpha line carrying the most of the radiance. Another important case where radiation transport has to be treated carefully is shielding phenomena of the first wall materials. Intense particle and energy fluxes to which the divertor elements may be subjected during transient events such as ELMs or disruptions induce evaporation of the material. The vapor quickly becomes ionized and the resulting plasma serves as a shield for the target. The shielding plasma has strongly nonuniform density and temperature profiles influencing radiation transport.

These phenomena should be considered for the lithium tokamak first wall for determination of conditions of plasma-wall interactions.

We develop a radiation transport model based on radiative-collision approximation and perform corresponding calculations for background lithium plasma parameters obtained by SOLPS code. Three ionized states are considered. Population of every state is taken as a background from SOLPS modeling. Within every state the radiation transport is described by multilevel model taking into account electron collision excitation/deexcitation processes, spontaneous photon emission, photon absorption, and induced photon emission. The transport in wings of line profile is treated employing collisional approximation for line profile shape. Influence of both Doppler and Stark effects on the line broadening are considered.

Calculations show that the radiation transport tends to be noticeably asymmetric. The preferential direction of the radiation energy flux is defined by the density and temperature gradients influencing photon mean absorption length and variation of the line profile half-width. This effect can potentially reduce the first wall energy loads. We also notice that there are approximations used in radiation transport code EIRINE which is a part of SOLPS package which are not justified in our case. Firstly, it is supposed that the population of excited states is governed only by electronic collisions while contribution of photon absorption can be neglected. Secondly, as in the most of Monte-Carlo codes dealing with radiation transport, only limited amount of “photons” is sampled resulting in underestimation of transport in wings of the line profile. Finally the computational cell size used is often too large to account for radiation transport carefully, only 3-5 cells are taken within the scrape-off layer. Our model lacks these deficiencies as it uses solution of ordinary differential equations instead of Monte-Carlo simulation on a fine mesh with sufficient resolution of the line profiles and fully includes contribution of photons in excitation.

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MEIS ANALYSIS OF LI LAYERS DEPOSITION

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The spectroscopy of hydrogen keV-energy ions scattering can be used as non-destructive method of determination of thickness of ultrathin (1-10 nm) surface layers with very different atomic mass from that of substrate [1]. In the presented work this method is used to analyze the process of lithium deposition on tungsten.

Lithium piece was heated to temperatures up to 450° C, the dynamics of evaporated lithium deposition on tungsten target was investigated in situ by scattering of 9 keV hydrogen ions from the target at 38° angle. Evolution of energy spectra during the experiment is shown in fig. 1.

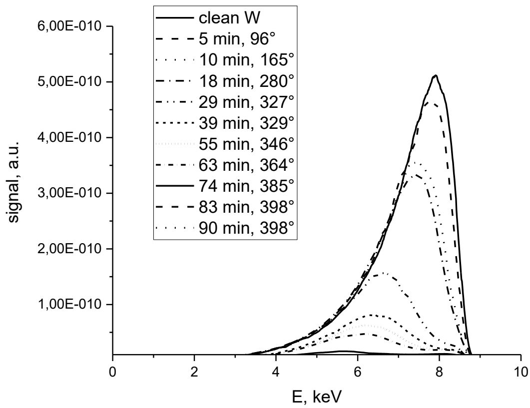

Fig. 1. Evolution of scattered H^+ energy spectra depending on the time and the heated Li temperature

During the experiment partial pressures of residual gas components were controlled by quadrupole mass-spectrometer. The dynamics of their change combined with lithium temperature and the data from energy spectra can provide information about lithium deposition rate and its interaction with hydrogen, oxygen and other residual gas components.

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FIRST EXPERIMENTAL RESULTS WITH TIN LIMITER ON FTU PLASMA

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In the framework of the liquid metal research it has been taken into account the use of liquid metals as plasma facing materials in fusion devices. Two different materials have been studied in the Frascati Tokamak Upgrade (FTU): lithium and tin. Since 2006, a liquid lithium limiter has been tested on FTU and the successful experiments and the promising results have pointed out the importance to explore other liquid metal materials for the limiter such as tin whose operating window is much larger than lithium. The possibility to increase the operation temperature allows increasing the steady state heat load on the limiter surface up to very high values. The tin limiter has been tested on FTU in the last experimental campaign, started at the end of 2016. FTU is the first tokamak in the world operating with liquid tin limiter and one of the pioneers in liquid metal application. The Sn Limiter, without active cooling for these first experiments, has been inserted in the FTU plasma at different distance from the last closed magnetic surface (LCMS) both in ohmic regime and with LH and ECRH additional heating systems. The experiments have been performed in repeated discharges characterized by very clean conditions, due to the only presence of Mo impurity, coming from the TZM toroidal limiter. The preliminary analysis of the experimental data has been focalized to detect the presence of tin in the discharge: suitable monitors are the spectroscopic diagnostics in the visible and UV ranges. The experimental observation of the tin spectral lines represents a new goal for extending the database of atomic nuclear data in the plasma tokamak research. In particular, 607.8 nm and 645.3 nm spectral lines of SnII have been observed. In addition, all the expected spectral lines in UV range have been detected, 20.4 nm of SnXXI and 21.9 nm and 27.6 nm of SnXXII. Another analysis has been addressed to evaluate the Sn limiter surface temperature, by using the infrared fast camera data. Numerical simulation performed with ANSYS code shows that a maximum heat load of 10 MW/m^2, in ohmic discharges, has been achieved with the Sn limiter very close to LCMS. In this condition, although Sn limiter surface reaches, in a limited central region, temperatures up to 1300 °C, not far to the evaporation temperature, the plasma performances do not significantly change and a slight decrease of the Z_{eff} value is observed. The preliminary results obtained during these experiments with the tin limiter in FTU will be presented.

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ANALYSIS OF REACTOR IRRADIATION EXPERIMENTS OF LEAD LITHIUM EUTECTIC Pb83Li17

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At present time, the lead lithium eutectic is considered as a breeder material for DEMO reactor intended to demonstrate the commercial attractiveness of fusion power engineering [1]. Works in the direction of creating the optimal DEMO blanket design are being carried out all over the world. Several concepts of Test Blanket Modules (TBM) are being developed with their further testing at the ITER reactor: a) Helium-Cooled Lead Lithium blanket (HCLL) in the EU [2]; B) Water-Cooled Lead Lithium blanket (WCLL) [3]; C) a Dual-Coolant Lead Lithium blanket (DCLL) in the US [4]. Moreover, Japan is working on the creation of a lead lithium eutectic blanket for the KOYO-FAST laser fusion reactor [5].

There is a very limited number of works devoted to tritium release from the lead lithium eutectic. Most of the studies on the evaluation of diffusion, solubility and release parameters of hydrogen isotopes (in particular, tritium), for this material were performed under ordinary conditions, i.e. these were experiments with non-irradiated materials, or experiments with samples after irradiation [6, 7].

This paper describes the analysis of experiments on study of helium and tritium generation and release from lead lithium eutectic under irradiation at the IVG1.M reactor. A phenomenological model for helium and tritium release from the eutectic has been developed, and the main parameters of the model have been determined.

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SURFACE INSTABILITY OF LIQUID TIN AND GALLIUM DUE TO THE FORMATION AND DECOMPOSITION OF VOLATILE METAL HYDRIDES

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Large scale formation of liquid tin and gallium droplets was observed with free liquid surfaces (FLS) exposed to a H-admixed plasma using the nano-PSI linear plasma device. Typical plasma parameters were ne~10^{18} m^{-3}, Te~ 0.5 eV, Γ~ $5x10^{20}$ m^{-2} s^{-1}. Addition of only 5% H_2 (gas flow ratio) to Ar and He plasmas induced intense droplet ejection, while once this H was removed droplet formation ceased. Camera recordings indicate that the production of these droplets should be due to bubble bursting. Two hypotheses for bubble production involving hydrogen are proposed: (1) the recombination of hydrogen atoms and ions into molecular hydrogen; (2) the formation and decomposition of volatile metal hydrides. It is known that hydrogen radicals reacting with tin and gallium can generate two hydrides (stannane and gallane) which are very unstable and easy to decompose to hydrogen gas at room temperature [1-3].

As a comparison, liquid tin was exposed to nitrogen-admixed plasma (5% to 50% N_2) where recombination of atoms and ions into molecular form may also be expected. In this case there was no clear formation of tin droplets. This indicates volatile hydrides (hypothesis 2) are the dominant driver of droplet formation. Experimental results show that temperature has a weak influence on this process from melting point to 600 cC.

Two different capillary porous system (CPS) targets with surface pore size of 100 μm diameter filled with tin were also exposed to H-admixed Ar plasma (5% H_2). In one case gaps between the edge of the target and the edge of the mesh were present and no droplets were ejected. In the second case the mesh layers were present up to the edge of the sample. In this case the mesh was observed to expand upwards, leading to a massive sudden ejection of droplets. This indicates that a pressure difference can easily build between mesh layers as more and more hydrogen gas gathers due to SnH_4 decomposition. In summary this indicates that CPS use can prevent such droplet formation but that CPS design must be carefully considered when designing liquid metal divertor components.

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ASSESEMENT OF THE DIVERTOR POWER LOAD FOR DEMO WITH LIQUID TIN DIVERTOR

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A divertor in DEMO will be exposed to high heat fluxes as well as neutron, deuterium, tritium and helium fluxes which will compromise its lifetime. In order to extend a divertor lifetime a Liquid Metal (LM) lithium divertor was proposed [1,2]. The purpose of liquid metal flowing on the tungsten base is to limit surface erosion and material stresses. Vapor shielding may additionally reduce high heat fluxes during transients. Liquid tin (Sn, Z=50) was brought to attention recently as a high Z material. A direct consequence of using high Z material for divertor capillary porous system (CPS) is high tin radiation both in the core and in the SOL, which in turn carries the necessity of an integrated core-SOL modelling.

Aim of this paper is to investigate a divertor power load in DEMO reactor with liquid tin divertor dependence on perpendicular SOL transport coefficient and density at the separatrix. The simulation is performed with the COREDIV code, which self-consistently solves radial 1D energy and particle transport equations of plasma and impurities in the core region and 2D multifluid transport equations in the SOL. Influence of the sputtering, evaporation and prompt redeposition is taken into account. The evaporation rate depends on the energy flux to the divertor target and the amount of tin released is, thereby, coupled with the plasma in the divertor region. The heat deposition due to re-condensation of Sn is presently neglected as the Sn vapour distributes the heat over a large area or is absorbed in the pumps. Increased heat flux leads to more evaporation and hence more Sn in the SOL. Vapors radiate and redistribute significant part of the energy flux to the plate creating a vapor shield.

First studies [3] show that DEMO operation in presence of tin impurity is not excluded although power across the separatrix (PSOL) is slightly below the LH threshold, but still within the uncertainty limit. Argon seeding increases PSOL extending an operational space. Argon is a good radiator in the SOL region, which radiation depends on the perpendicular transport model in the SOL.

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LI-BASED CHEMICAL DYNAMICS ON POROUS W DURING D2 IRRADIATION UNDER SIMULATED FUSION PMI CONDITIONS

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Tungsten is the material of choice for the plasma facing components in the divertor region for its favorable thermomechanical properties such as its high melting point, thermal conductivity, low sputter yield, stability to neutron irradiation, and high D recycling [1]. However, mechanical failure and ion induced surface morphology transformation such as fuzz [2], can expose the bulk W to energetic ion fluxes and heat fluxes increasing the probability for high-Z metal dust formation. In addition, transient events can cause micro cracks, fractures that can propagate throughout the W armor, compromising its performance [3]. The intrinsic healing nature of a liquid metal wall mitigates these concerns but are limited by drawbacks including but not limited to macro ejection, liquid-induced degradation, and evaporation/redeposition. Previous work has shown higher grain boundary density can increase the threshold for He-induced surface morphology in W and net radiation damage resistance. Extending that same reasoning, it is posited that the increased surface-area-to-volume ratio of a porous substrate can act as a defect sink, and thus be more resistant to detrimental morphology [4]. The porous nature allows for the escape of migrating He. Additionally, porous W has demonstrated favorable thermomechanical properties in inertial confinement systems [5].

A hybrid system combining multiple phases (e.g. solid and liquid) may address some of these concerns by exploiting the external stimulus of particle and heat fluxes and utilizing a “healing/shielding” agent i.e. Li, to delay the onset of mechanical failure and mitigate its propagation [6]. The incoming flux can induce segregation of Li embedded in a W structure as a protective layer between exposed W and form a protective vapor cloud via evaporation above the PFC to reduce incident heat flux through radiative dissipation [7]. Porous W would be an ideal scaffold for a liquid metal, with the porosity tunable to a desired balance of surface wetting and evaporation to reach optimal edge plasma conditions. Two additional systems we will study are porous W based systems with Sn and SnLi, which can provide control over fuel retention. Surface chemistry of liquid Li on a porous surface changes with D irradiation is studied and its effect on retention. These hybrid systems, as well as traditional W samples, were bombarded with 500eV D2+ and Ar+ at 230^{o}C and 300^{o}C. The Li, O, and C XPS peaks were examined and compared to controls.

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STUDY OF HYDROGEN-CONTAINING SWEEP GAS IMPACT ON THE PROCESSES OF TRITIUM RELEASE FROM LEAD LITHIUM EUTECTICS

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Interest in studying the hydrogen isotopes (including tritium) interaction with liquid metal lithium-containing materials is associated with their use in fusion reactors. Firstly, of course, is their use in blankets as breeder materials (as tritium-reproducing material and coolant at the same time). On the other hand, it should be mentioned that since recent time lithium has been considered as a possible plasma-facing material. Various versions of the walls lithification of fusion facilities are discussed, including using lithium capillary porous systems (CPS).

To correctly take into account the balance of tritium in the fusion reactor nodes, it is necessary to estimate the parameters of tritium interaction with functional materials (in particular, with liquid metal lithium systems), determine the mechanisms of tritium transport, and investigate the processes on the surface of tritium containing materials.

The investigation of tritium release processes is of interest now: not only under conditions of its generation because of neutron irradiation of liquid lithium metal systems, but also under conditions of the presence of a hydrogen-containing sweep gas. According to general approaches, the presence of a hydrogen-containing sweep gas should lead to an increase of tritium release rate.

In this paper, the results of reactor experiments on the investigation of tritium release in the presence of hydrogen above the sample surface are presented.

The experiments were carried out according to the following scheme: under reactor irradiation conditions at a given temperature of the eutectic, a quasi-equilibrium tritium release into the plenum above the eutectic (which is continuously evacuated) is achieved. Further, a hydrogen stream is fed into the cell with eutectic and the kinetics of flux change of tritium release from the eutectic is registered.

The obtained results made it possible to determine the dependence of the change in tritium release flux from the eutectic and to make conclusions about the mechanisms of the hydrogen-containing sweep gas influence on the processes of tritium release.

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REACTOR EXPERIMENTS ON STUDY OF GASEOUS MIXTURES EXCITATION BY THE PRODUCTS OF 6LI(n,α)T NUCLEAR REACTION

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Direct conversion of nuclear reaction energy into energy of optical radiation is a promising direction in the field of obtaining a large amount of light radiation, including its perfect form - coherent light. Such exceptional features of nuclear energy sources, such as high specific power, compactness, ability to efficiently excite and ionize large amounts of active gaseous media, give certain potential advantages over other traditional methods of gaseous mixtures excitation. In addition, studies of gaseous mixtures excitation by products of nuclear reactions are of interest to develop a method for energy removal from nuclear and fusion reactors, as well as for reactor's parameters monitoring and control.

Currently, in the Institute of Atomic Energy, the investigations of spectral-luminescent characteristics of a nuclear-excited plasma produced by products of nuclear reactions are performed at the IVG1.M reactor (Kurchatov, Kazakhstan). This activity is aimed to select the gaseous mixtures with a high conversion factor of nuclear energy into optical radiation. The reaction ^{6}Li(n,α)T almost unused previously for this purpose was chosen as the reaction resulting in excitation of gaseous mixtures.

This paper presents the results of reactor experiments on study of the spectral-luminescent characteristics of gaseous mixtures (on the basis of He, Ne and Kr inert gases) excited by products of nuclear reaction ^{6}Li(n,α)T under irradiation at the IVG1.M reactor.

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ARCS ON THE MELTED METAL SURFACE IN FUSION DEVICES

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Erosion and cracking of the materials such as tungsten and beryllium exposed to extreme thermal plasma load have been recently observed in fusion devices [1,2]. Several effects such as sputtering, cracking, melting of the surface layer, molten metal motion over the surface, drop erosion, evaporation and boiling, recrystallization and solidification of the molten layer lead to embrittlement and change in the surface relief. Such processes result in the stochastic clustering and formation of porous layers on the surface on scales from tens of nanometers to hundreds of micrometers forming the surface roughness with extremely high specific area. Such surface is favourable for an ignition of unipolar arcs and sparks. Recent experimental observations support the point of vew that the contribution of arcs and sparcs processes to the total erosion of plasma-facing materials can be significant in fusion reactor including ITER. Numerous arc craters were detected on the berillium plates after high heat flux test in the QSPA-T fusion facility [3]. Snowflake-like craters and traces produced by arcs were detected on the tungsten surface of the poloidal limiter plates in the T-10 tokamak after plasma experiments (Fig.1).

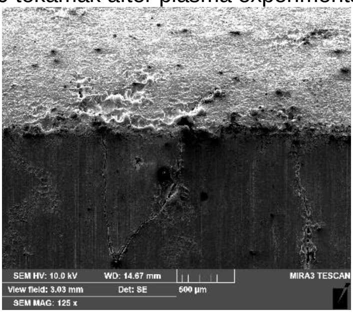

Fig. 1. Arc tracks on the tungsten limiter edge at the LCFS in the T-10 tokamak

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FACILITY FOR INVESTIGATION OF HEAT EXHAUST EFFICIENCY IN CPS BASED PFE OF STEADY STATE OPERATING TOKAMAK

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Heat removal potential of PFE based on CPS with liquid metals is one of the critical aspects to be investigated for realization of steady-state operation in projects of DEMO type fusion reactor and fusion neutron source.

The experimental facility for the study of power exhaust process at high specific density (~10^{-20} MW/m^2) in CPS based mock-ups with water-air spray coolant is described. The first result of mock-ups tests with scanning e-beam source of power load is presented.

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INTERACTIONS BETWEEN HELIUM PLASMA AND LITHATED GRAPHITE SUBSTRATE COATED WITH SILICON CARBIDE

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Due to the properties of stable refractory surface, high toughness, thermal shock resistance, low neutron activation energy, good chemical compatibility and similar thermal expansion coefficient with the graphite substrate, the silicon carbide coating is employed as protective coating on graphite in fusion devices. Due to advantages of absorption impurity and good compatibility with inner plasma, lithiation of graphite has been confirmed to effectively improve the plasma characteristics in the EAST[1]. However, many experimental results indicate that lithiation of graphite coated with silicon carbide resulted in enhancing erosion by plasmas.

In order to understand the erosion mechanism, the linear plasma device built in Sichuan University was utilized to perform helium plasmas interacting lithiated graphites in this study. The optical emission spectroscopy, fast reciprocating probe and thermocouples was adopted to examine the effect of the lithium from the lithiated sample on incident plasma beam. The scanning electron microscope, energy dispersive X-ray detector, X-Ray diffraction, X-ray photoelectron spectroscopy and laser-induced breakdown spectroscopy was used to characterize the morphology, composition change, phase transformation and lithium penetration depth in the irradiated samples by plasmas. The experimental results show that the lithiated sample was serious sputtered by plasmas. And due to the corrosion of lithium, the partially decomposed and phase transformation from α-SiC to β-SiC were observed on the the lithiated silicon carbide coating during the plasma irradiation.

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