The Book of Abstracts

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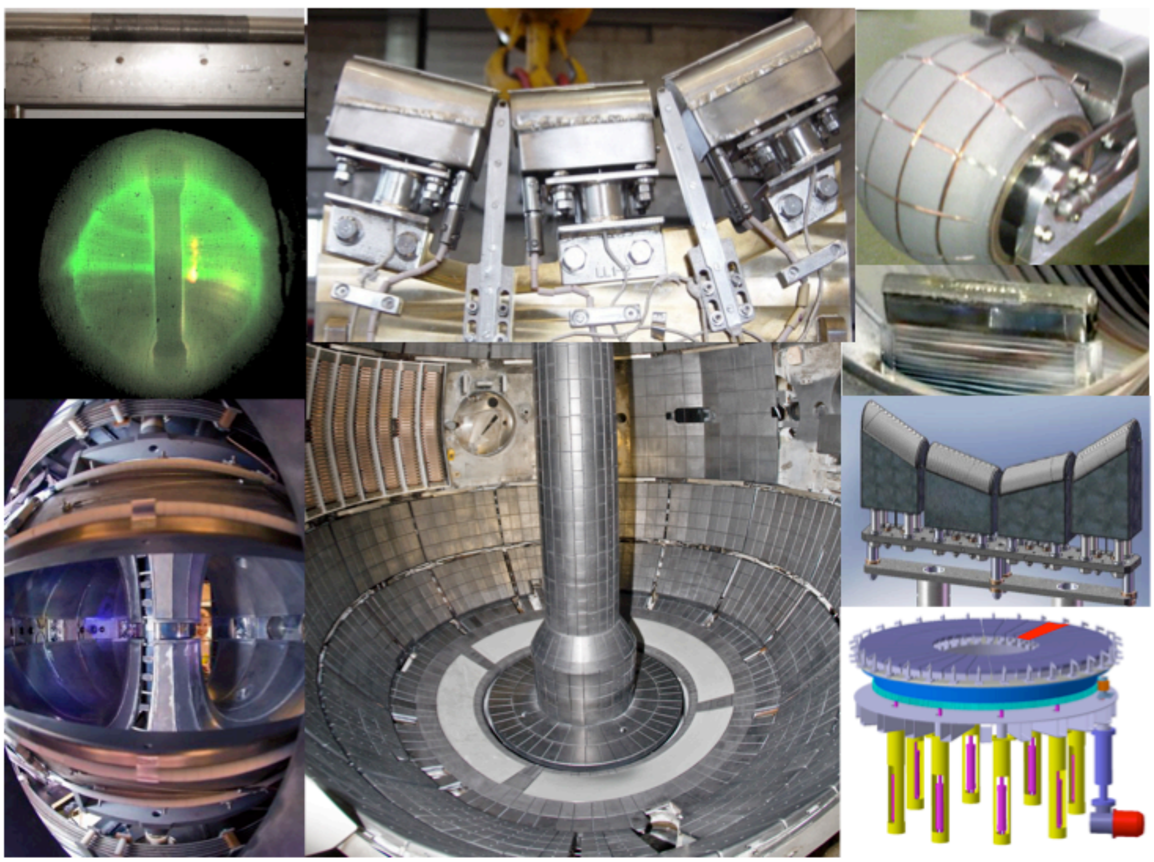

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NSTX Plasma Operation with a Liquid Lithium Divertor

H.W. Kugel, and NSTX Team

NSTX 2010 experiments were conducted using a molybdenum Liquid Lithium Divertor (LLD) surface installed on the outer part of the lower divertor. This tested the effectiveness of maintaining the deuterium retention properties of a static liquid lithium surface when refreshed by lithium evaporation as an approximation to a flowing liquid lithium surface. The LLD molybdenum front face has a 45% porosity to provide sufficient wetting to spread 37 g of lithium, and to retain it in the presence of magnetic forces. Lithium Evaporators were used to deposit lithium on the LLD surface. At the beginning of discharges, the LLD lithium surface ranged from solid to liquefied depending on the amount of applied and plasma heating. Noteworthy improvements in plasma edge conditions were obtained similar to those obtained previously with lithiated graphite, e.g., ELM-free, edge-quiescent, H-modes. During these experiments with the plasma outer strike point on the LLD, the rate of deuterium retention in the LLD, as indicated by the fueling needed to achieve and maintain stable plasma conditions, was the about the same as that for solid lithium coatings on the graphite prior to the installation of the LLD, *i.e.*, about two times that of no-lithium conditions. The role of lithium impurities in this result is discussed. Following the 2010 experimental campaign, inspection of the LLD found mechanical damage to the plate supports, and other hardware resulting from forces following plasma current disruptions. The LLD was removed, upgraded, and reinstalled. A row of molybdenum tiles was installed inboard of the LLD for 2011 experiments with both inner and outer strike points on lithiated molybdenum to allow investigation of lithium plasma facing issues encountered in the first testing of the LLD.

New progresses of lithium coating or plasma facing material in ASIPP

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Lithium coating were successfully carried out by various techniques, such by evaporation, associated by GDC or ICRF discharge, or actively coating from a NSTX type lithium dropper [1] or from a liquid lithium limiter [2] during plasma discharge in the last two years. On both of EAST and HT-7, compared with boronization and siliconization, lithium coating was testified as a best way to improve plasma performance, such as impurities and MHD suppression, recycling reduction, confinement improvement, and so on.

Especially, in the autumn campaign of EAST in 2010, lithium coating with two upgraded ovens has been became a routine method for wall conditioning. By everyday coating with 10~30g lithium, plasmas with low impurities (Z_{eff}=1.5~2.5), low recycling and lower than 10% of the ratio of H/(H+D) were easily obtained. Lithium coating was base important for a few new milestone of plasma operation, such as the first H-mode plasma, 100s long pulse plasma and 1MA plasma, and is also beneficial for the improvement of the heating efficiency of ICRF. Repeatable H mode plasmas achieved by a relative low heat power of LHCD or ICRF were easily obtained on EAST either by Li coating by oven or by active Li powder injection.

These results encouraged us to start a new challengeable project of a flowing liquid lithium limiter with a long tray for HT-7, which would provide some techniques accumulation for a flowing lithium divertor for EAST.

This research is funded by National Magnetic confinement Fusion Science Program under contract 2010GB104002 and the National Nature Science Foundation of China under contract 11075185.

[1]D.K. Mansfield, et al., A simple apparatus for the injection of lithium aerosol into the scrape-off layer of fusion research devices, Fusion Engineering and Design 85 (2010) 890–895
[2]J.S Hu, et al., Investigation of lithium as plasma facing materials on HT-7, Fusion Engineering and Design 85 (2010) 930-934

Recent Results from the Lithium Tokamak eXperiment (LTX)*

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LTX is a newly commissioned, modest spherical tokamak with R=0.4 m, a=0.26 m, and elongation=1.5. Upgrades are in progress to produce a toroidal field of 3.2 kG, plasma current up to 400 kA, and a discharge duration of order 100 msec, although in 2010 the device operated at reduced parameters. LTX is the first tokamak designed to investigate modifications to equilibrium and transport when global recycling is reduced to 10 – 20%. To reduce recycling, LTX is fitted with a 1 cm thick heated (300 – 400 °C) copper shell, conformal to the last closed flux surface, over 85% of the plasma surface area. The plasma-facing surface of the shell is composed of thin stainless steel, explosively bonded to the copper, and is designed to be evaporatively coated with a thin layer of lithium. In addition, the lower sections of the shell are designed to retain up to several hundred cubic centimeters of liquid lithium, to form a lower liquid lithium limiter similar to that employed in CDX-U. [R. Majeski et al., Phys. Rev. Lett. **97** (2006) 075002] The shell is replaceable, and a second version has been constructed, which was plasma-sprayed with 100 – 200 microns of molybdenum to form a high-Z substrate for subsequent coating with lithium. LTX is the first tokamak designed entirely to accommodate high temperature walls and a large in-vessel inventory of liquid lithium.

In 2010 LTX was first operated with lithium wall coatings. Two new lithium evaporation systems were installed in the device. No traditional wall conditioning techniques (boronization or other low-Z coatings) were employed, and low-Z limiters are not installed, in order to prevent the formation of directly deposited or sputtered films on the inner wall of the shell, which could react with lithium and increase recycling. Early discharges against the uncoated stainless steel shells in LTX were therefore impurity-dominated, with plasma currents only in the 10 – 15 kA range, and short discharges of 4-6 msec duration. Wall conditioning with solid lithium films produced discharges with greatly increased plasma currents, up to 70 kA, and an increase in discharge duration to 20 msec. These parameters are similar to CDX-U discharges obtained with similar lithium wall coatings. Preliminary Thomson scattering data indicate core electron temperatures of 100 – 150 eV. Although good discharge parameters were obtained with room temperature, solid lithium wall coatings, operation with hot (300 °C) walls and presumably molten lithium films were not as effective. With hot walls, rapid passivation of the lithium coatings was observed. The performance of discharges limited on hot lithium coated walls was similar to the performance of discharges limited on uncoated, bare stainless steel walls.

Plans for the 2011 campaign, including operation with a liquid lithium fill, will also be discussed.

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Li collection experiments on T-11M and T-10 in framework of Li closed loop concept

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The concept of a steady state tokamak with the first wall and plasma facing components (PFC) on the basis of the closed loop of liquid Lithium circulation demands the decision of three tasks: Lithium injection to the plasma, Lithium ions collection before their deposition on the vacuum vessel and Li returning from collector to the zone of injection. For practical solution of these problems in T-11M and T-10 tokamak experiments have been applied Li, graphite rail limiters and special ring limiter-collector (T-11M). In this report the general attention has been paid to the investigation of the Lithium collection by different limiters and the studying of Lithium ions behavior close tokamak boundary. The behaviour of Lithium in the SOL area and efficiency of its collection by limiters in T-11M and T-10 tokamaks were investigated by sample-witness analysis and also (T-11M) by use of the heated mobile graphite probe (limiter) as a recombination target in relation to the stream of Lithium ions. It was measured, that characteristic depth of Lithium penetration in the SOL area of T-11M is about 2cm and 4 cm in SOL of T-10. That is equal proportional of their major radius R. The quantitative analysis of the sample-witnesses located on T-11M limiters showed, that nearby 60±20% of the Lithium injected during plasma operating of T-11M had been collected by limiters. It confirms a potential opportunity of collection of the main part of the injected Lithium by the limiters-collectors located in the SOL area of steady state tokamak.

Plasma behavior in presence of a liquid lithium limiter

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The liquid lithium limiter (LLL) is routinely used on FTU to obtain very clean plasma and to get very performing plasma discharges. But for using a liquid surface as plasma facing component in a future reactor it is also very important to assess the capability of the lithium liquid limiter to withstand heat loads.

The most significant results obtained of FTU will be reviewed with special emphasis on heat loads and plasma edge modification as the increase of the SOL electron temperature . This strong increase could be related to strong decrease on recycling coefficient. We are starting to simulate the plasma edge on FTU by using the B2-EIRENE code adapted to the circular FTU cross section.

Recycling and Sputtering Studies in Hydrogen and Helium Plasmas under Lithiated Walls in TJ-II.

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Up to date, TJ-II is the only stellarator routinely operated on lithiated walls, thus offering the possibility to address important issues concerning the possible design of a stellarator-based reactor under very low recycling conditions [1] In this work, the important issues of fuel retention and wall erosion for H and He plasmas are addressed. Concerning erosion and implantation, the energy of the ions reaching the wall could be strongly modified under pure NBI heating due to minimization of charge exchange loses and the concomitant flattening of edge Ti profiles [2]. However, the sputtering yield of lithium was found to be significantly lower than that expected from laboratory experiments and Trim code calculations. Moreover, the dependence of that yield on edge temperature is consistent with an energy threshold much larger than that of pure lithium. In order to assess the effect of material mixing, which appears a good candidate for the observed effect [3], several degrees of mixing of the Li layer with the underlying boron were induced by the conditioning plasma.

Another topic that has been recently investigated in TJ-II is particle retention and release under H/He operation. Recycling coefficients $R< 0.1$ and $R \sim 0.85$ for H and He, respectively, were measured, leading to good density control in ECRH and NBI heated plasmas and opening the possibility to strong He pumping by the lithium wall, as previously suggested [4]. The release of either species in the opposite plasma has also been investigated under several plasma conditions. It is concluded that thermal effects, possibly related to the diffusion of the released species across the lithium layer, can set a limit when isotope interchange is required, independently of the flux of impinging particles.

In this presentation, TJ-II as well as laboratory experiment results on Li sputtering and recycling in the presence of boron will be addressed.

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Lithization on RFX-mod reversed field pinch experiment

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RFX-mod experiment is a circular section Reversed Field Pinch (RFP) device with major/minor radius 2.0/0.46 m, maximum plasma current 2 MA and first wall entirely covered by graphite tiles.
Due to the high recycling related to the graphite wall, plasma-wall interaction (PWI) is an issue in RFX-mod for the operation at plasma current over 1 MA. At the highest plasma currents (1.5-2 MA) PWI influences the performance affecting both Z_{eff} and density and temperature control. In particular the improved single-helical-axis states (SHAx), spontaneously developing at high plasma current (Ip>1.2 MA), disappear when the density is increased to $n/n_G \geq 0.2$.
Following tokamak experience, in order to improve density and impurity control He glow discharge cleaning, high current He discharges, wall boronization and baking have been applied. All such techniques were effective in improving the operation reliability but none of them provided a strong improvement in term of plasma performance. As a further step ahead, based on good Tokamak and Stellarator results, we recently tested the effect of wall conditioning by Lithium.
As a first lithization method to deposit on the wall a controllable amount of Lithium we have used a room temperature pellet injector (max pellet diameter of 1.5 mm and max length of 6 mm). Coating deposition was optimized by adjusting plasma discharges used as target for lithium pellets, obtaining the best results with short 1 MA Helium discharges. Lithium coatings with a nominal thickness of about 10 nm were applied both directly to the graphite tiles and over a fresh boronization. The technique proved to be effective in maintaining Hydrogen wall influx very low Good indication on the lithization potential benefits have been obtained at plasma edge, where a lower density, higher temperature and an improved particle confinement time were observed. Yet such improvements are limited in amplitude and last only a small number of discharges. Graphite samples (and wall tiles) have been exposed to lithization and plasma discharges; the surface analysis indicated that after hundreds of plasma discharges lithium is still in place on the wall but it loses the capability to improve plasma-wall interaction.
Experiments with a Liquid Lithium Limiter (LLL) with a capillary porous system have been also started, in order to improve the lithization efficiency by producing a thicker lithium coating on the wall and providing a preferred path for plasma-wall interaction to a hot Lithium limiter. For this experiment a LLL on loan from FTU experiment of ENEA laboratories in Frascati has been used. The particular RFP feature of an edge magnetic field essentially poloidal has to be considered, as it makes difficult a toroidally uniform deposition. The LLL has been used both as limiter than as evaporator. The use as evaporator has been followed by He low current plasma discharges to spread Lithim. Till now the evaporator only has provided an evidence of the wall conditioning on plasma discharges: after conditioning by evaporation Hydrogen discharges showed a remarkably high adsorption capacity of the first wall. As a preliminary result, though on a single shot, a sensible reduction of the resistive loop voltage at $n/n_G > 0.15$ was observed. In addition, a Quasi Single Helical State associated to the formation of an Internal Transport Barrier appeared at 1.2 MA, whereas both usually develop at higher plasma current and lower density.

Recycling, Pumping and Divertor plasma-Material Interaction studies with evaporated lithium coatings in NSTX

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In the National Spherical Torus Experiment (NSTX) solid lithium coatings on graphite plasma-facing components (PFCs) are studied for impurity and density control. Access to reduced core collisionality is important for the development of plasma scenarios with a high-non-inductive current fraction and for adequate NBI current drive efficiency, as well as for understanding transport, stability, and non-inductive start-up and sustainment relevant to potential spherical tokamak (ST)-based fusion and nuclear science facilities. Plasma regimes with lower pedestal and scrape-off layer (SOL) collisionality also enable studies of edge and divertor transport and heat flux mitigation techniques directly relevant to next step STs which are predicted to operate in sheath-limited divertor heat transport regimes with high peak heat fluxes.

Significant modifications in the SOL and divertor conditions with lithium coatings were evident in NSTX. The lower divertor, upper divertor and inner wall recycling rates were reduced by up to 50 %. Core ion density (and inventory) has also been reduced by up to 50 %, and this reduction was sustained up to 1.2 s discharge duration. Analysis of relative PFC recycling coefficients and lithium fluxes indicated that 1) Recycling was reduced over a full poloidal extent of the PFCs, except in the strike-point region, where marginal reduction, if any, was observed; 2) Increased lithium evaporation, and cumulative applications, correlate with further recycling reduction and increased divertor lithium fluxes; 3) Observation of pronounced peaking of divertor lithium flux and recycling trends in the strike point region suggested that lithium layer could be melted and evaporated. Zero-dimensional particle balance equation indicated un-saturated and transient pumping by lithium coatings. The pumping effect longevity was found to disappear in 1-3 discharges without additional evaporations.

With reduced recycling the outer SOL transport regime changed from the high-recycling, heat flux conduction-limited with $\nu_e^* \sim 10$-40 to the sheath-limited regime with a small parallel T_e gradient and higher SOL T_e with $\nu_e^* < 5$-10. Reductions in SOL neutral pressure (density) and electron density were observed, leading to the re-attachment of the normally detached inner divertor region, and disappearance of occasional X-point and inner divertor MARFEs.

An elimination of ELMs and an improvement in particle confinement caused impurity accumulations and an increase in core P_{rad} up to 2-3 MW. Spectroscopic measurements of carbon fluxes due to physical sputtering suggested that the wall and divertor sources did not increase with lithium, implying an increased inward transport effect. Lithium core concentration was found to be low $n_{Li3+}/n_e \sim 0.001$, expected from divertor screening and prompt redeposition of sputtered lithium in the divertor. In a dedicated experiment, divertor D_2 injection was demonstrated to reduce core carbon concentration by up to 30 %, suggesting a method for controlling the divertor carbon physical sputtering source.

The effectiveness of divertor heat flux mitigation techniques, such as the radiative divertor with D_2 or CD_4 seeding, and the snowflake divertor configuration, were demonstrated to be compatible with lithium coatings. Peak divertor heat fluxes have been reduced by up to 80 %, albeit e.g. higher gas injection rate requirements for the radiative divertor.

This work was performed under the auspices of the U.S. Department of Energy under Contracts DE-AC52-07NA27344, DE-AC02-09CH11466, DE-AC05-00OR22725, W-7405-ENG-36, and DE-FG02-04ER54758.

2nd International Symposium on Lithium Applications for Fusion Devices Abstract

Modification of the Electron Energy Distribution Function during Lithium Experiments on the National Spherical Torus Experiment

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Lithium coatings on plasma-facing components have found wide use in experiments in the National Spherical Torus Experiment (NSTX). To date, a number of empirical observations have been made that indicate macroscopic changes in plasma performance including a broadening of the plasma profile[1] and the disappearance of ELMs[2,3] with lithium wall conditioning. In an effort to assess the performance of liquid lithium PFCs, the Liquid Lithium Divertor (LLD) was installed for the 2010 run campaign.

Plasma measurements are made with a number of diagnostics including a new high-density Langmuir probe array (HDLP) [5]. The LLD presented a general challenge due to difficulties in diagnosing the state of the lithium during plasma operations. During one set of experiments where the LLD was heated by plasma bombardment, observations were made indicating the possible activation of the lithium present on the LLD (e.g. drops in fueling efficiency and increased PFC heating). The HDLP signals provide data indicating that local changes in plasma conditions also occurred.

The probes operate in the thin-sheath regime greatly simplifying analysis by removing significant sheath growth effects[6]. Initial analyses have been made with the 'classical' interpretation method[5,7] which relies on data in the ion current portion of the I-V characteristic only. These first analyses enable the identification of the separatrix strike point so that more detailed comparisons may be made on specific magnetic surfaces. Detailed comparison is accomplished by examining the electron energy distribution function (EEDF) measured by the swept probes. It is found that the EEDF is well-described by a bi-modal Maxwellian distribution, similar to measurements made on the CASTOR tokamak[7]. The analysis indicates that the relative fraction of the hot electron population increased during the discharges. This increase helps explain depressed floating potentials as well as the increase in heat flux. A transition to a higher, single-temperature Maxwellian was predicted with a kinetic code for a non-recycling boundary condition by earlier researchers[8]. The increase in the hot population fraction is consistent with the predictions for a lower recycling system.

The Langmuir probe analysis methodology and results will be presented in detail.

Work supported by DOE contract No. DE-AC02-09CHI1466.

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Determination of Effective Sheath Heat Transmission Coefficient in NSTX Discharges with Applied Lithium Coatings

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Recycled particle flux can be a significant contributor to edge plasma density and lead to reductions in edge temperature. Previous measurements in NSTX have shown that solid evaporated lithium coatings can lead to lowered edge recycling, corresponding decreases in edge plasma density, and a broadening of the electron temperature profile [1]. During the 2010 run campaign, NSTX operated with both solid and liquid lithium coatings on its plasma-facing components, with two LITER evaporators providing the lithium input. In preparation for this campaign, a 99-tip dense Langmuir probe array was installed in the outboard divertor to measure scrape-off layer density and temperature [2,3]. The first row of outboard divertor tiles are ATJ graphite, while the second and third rows have been replaced with copper plates underneath a stainless steel shield layer which is coated with a surface layer of flame-sprayed porous molybdenum. While the lithium coatings on the graphite remain solid, the plates can be heated to render the evaporated lithium into a liquid state. The probe array was located so as to radially span these two different divertor surfaces and measure their respective effects on the edge parameters. The array is capable of measuring radial spatial scales of 3mm and temporal scales of 500 Hz in swept-probe mode and 250 kHz in triple-probe mode. A dual-band fast IR camera was also installed to provide surface temperature and heat flux measurements. The use of two-color IR thermography allows for an assessment of effects due to the uncertain, phase- and purity-dependent emissivity of the lithium coatings. Although these diagnostics measure at different toroidal locations, they view the same radial region and thus can provide cross-calibrated measurements of heat flux to plasma surfaces.

The present study compares the derived heat fluxes from these diagnostics to determine an effective classical sheath heat transmission coefficient γ_{eff}. The Langmuir probe heat flux is expressed as: $\gamma_{eff} kT_e \Gamma$, where the net flux is obtained from the saturation current density. This value is compared to the theoretical classical result, which is a sum of the electron and ion contributions. The electron term includes the forward going Maxwellian flux as well as the energy gained through the sheath and plasma potential drops that the electrons encounter. The ion term is simply the forward going energy of a Maxwellian drifting at the sound speed. Although the probes can measure many of the quantities necessary for this comparison, including the floating potential, the IR camera comparison can help elucidate other unknowns such as the ion temperature and effective Z of the scrape-off layer. Finding γ_{eff} is also an important intermediate step for utilizing Langmuir probe and IR data in quasi-1D simulations of the plasma edge.

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Liquid Lithium Divertor surface temperature dynamics and edge plasma modification under plasma-induced heating and lithium pre-heating

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The Liquid Lithium Divertor (LLD) was installed in the National Spherical Torus Experiment (NSTX) for exploration of density and impurity control, and edge plasma modification throughout the 2010 campaign [1,2,3]. Experiments were run where the LLD surface was heated up to and beyond the melting point of Lithium, 180.54 °C, both prior to a plasma discharge by electrical or hot air heating of its copper substrate, and by exposure to successive plasma discharge in which the bulk temperature of the LLD rose by ~5-10 °C per shot.

In order to remove the influence of variable emissivity – a potential variable due to phase change and contamination of the Li surface – a pioneering fast dual-band infrared (IR) camera was developed [4] and used for regular measurement of radial 1-D and areal 2-D temperature dynamics (*T* [K]) and heat flux (*q* [MW/m2]) on the NSTX lower divertor surface (~0.27 m < R < ~0.85 m, ~210° < φ < ~228°).

Results from the fast IR camera demonstrate that extended dwell of the outer strike point (OSP) on the LLD caused an incrementally larger area of the LLD to be greater than the Li melting point through the discharge. Comparison of $T_{surface}$ averaged over the near-OSP LLD surface to that over a Li-coated graphite tile at the same major radius demonstrates a significant clamping of the LLD surface temperature associated with the presence of liquid Li. Extrapolation of this result to Li temperatures >200°C suggest that Li evaporation is playing a significant role in reducing the power flux to the divertor surface. During post-discharge cooling of the LLD surface, the latent heat of fusion is demonstrated by a thermal decay transition to the Li melting temperature on the LLD surface, compared with simultaneous decay in Li-coated graphite temperature below the Li melting temperature.

Modification of the edge plasma and the presence of ELMs in the scrape-off-layer due to changes in recycling associated with the LLD melted state will be investigated and discussed.
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Thermal Modeling of the Surface Temperatures on the Liquid Lithium Divertor in NSTX

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Abstract: This paper summarizes thermal analyses of the Liquid Lithium Divertor (LLD) in NSTX. The objective was to identify the evolution of surface temperatures of the lithium on the LLD for various loading conditions, e.g. location of the strike point and power. Two unknowns in the calculations were (1) the emissivity of the surface of the LLD and (2) the thermal conductivity of the lithium-filled layer of porous plasma-sprayed molybdenum on the LLD. The thermal calculations used parametric variations of these unknowns within reasonable limits and attempts were made to extract values of emissivity based on the cooling of the LLD after shots. Extraction of the emissivity and thermal conductance of the Li-Mo layer were not very successful, but the approach taken may provide some insights for other experiments in the future. However, a planned investigation of the emissivity of lithium in a separate experimental collaboration by Sandia and Purdue University will use infrared thermography to measure the surface temperature of lithium in the PRIHSM, a specialized high vacuum chamber at Purdue with instrumentation to characterize the surface chemistry of solid and liquid lithium surface with impurities present.

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2nd International Symposium on Lithium Applications for Fusion Devices Abstract

Surface reflectivity and carbon source studies with the Liquid Lithium Divertor in NSTX

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Lithium evaporative coatings on graphite tiles are routinely applied between discharges on the National Spherical Torus eXperiment (NSTX) to reduce deuterium recycling at the plasma facing components (PFCs) [1]. In addition, in 2010 a Liquid Lithium Divertor (LLD) module was installed on NSTX to exploit the deuterium retention properties of molten lithium [2]. A liquid lithium reservoir on a molybdenum porous substrate in proximity of the strike point can help extend deuterium pumping capabilities of solid coatings. A clean lithium surface is essential in order to exploit the LLD beneficial effects. However, several issues can complicate the ability to maintain a pristine lithium surface on the LLD. These include the current LLD filling method (several hours of evaporation), the high chemical reactivity of liquid lithium with vacuum impurities and the presence of graphite PFCs. In particular, lithium reacts with residual vacuum components (e.g. H_2O and CO_2) to form compounds such as LiOH, Li_2O and Li_2CO_3 [2].

In order to try to diagnose PFCs surface conditions, the reflectivity of the lower divertor after lithium evaporations was routinely monitored throughout the LLD experiments using two divertor fast cameras. First observations included a gradual decrease in surface reflectivity after the Li melting temperature was achieved on the LLD plates. On the other hand, a drastic increase in reflectivity was observed after overnight lithium evaporations. Laboratory tests on a LLD sample are planned in order to study these surface reflectivity trends observed on NSTX with changes in evaporated amount of lithium, bulk temperature and reaction to vacuum impurities.

Lithium-conditioned H-mode ELM-free discharges in NSTX are generally affected by core carbon accumulation. Significant core carbon concentrations were also observed in ELMy discharges with outer strike point on the LLD. This stressed the importance of understanding carbon sources in NSTX and the purity of lithium on the LLD surface. The divertor fast cameras are equipped with several narrow band pass filters for impurity influx studies. Carbon influxes at the outer strike point are derived from the camera brightness measurements using the S/XB method [3] and T_e and n_e measured by a high density Langmuir probe array [4]. Initial analysis of carbon emission profiles from the divertor area indicated that carbon sourced from the graphite diagnostic tiles located between the LLD segments provides a significant contribution to the carbon brightness at the outer strike point location. The carbon emission from the LLD surface itself is indicative of surface contamination, and its origin from the erosion of graphite PFCs is being investigated.

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Effect of Lithium Coatings on Edge Plasma Profiles, Transport, and ELM Stability in NSTX

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Lithium coatings have been shown to improve energy confinement mainly through reduction of electron transport [1] in the National Spherical Torus Experiment (NSTX). When 'thick' coatings are applied between discharges, edge localized modes (ELMs) are completely suppressed [2,3]. The resulting post-lithium discharges are ELM-free with a 50% increase in normalized energy confinement, up to the global $\beta_N \sim 5.5$-6 limit [4]. Stability calculations have shown that the ELM suppression is caused by broadening of the pressure profile and the corresponding edge bootstrap current, owing mainly to a modification of the density profile [4].

The pressure profile broadening originated mainly from reduced recycling and edge fueling, which relaxed the edge density profile gradients inside the separatrix, effectively shifting the profile inward by up to 2-3 cm. In contrast, the edge electron temperature profile was unaffected in the H-mode pedestal steep gradient region at constant plasma stored energy; however, the region of steep gradients extended radially inward by several cm following lithium coatings. The measured edge profiles in both the pre-lithium and post-lithium discharges were simulated with the SOLPS code package, which indicated that both a reduction in recycling *and* a drop in the edge and SOL cross-field transport for $\psi_N < 0.95$ was required to match the post-lithium profiles. Indeed the edge fluctuations from reflectometry and BES were substantially reduced.

Calculations with the PEST and ELITE codes have confirmed that the post-lithium discharge pressure profiles were farther from the stability boundary than the reference pre-lithium discharges, which were relatively close to the kink/peeling boundary. Indeed low-n (n=1-5) pre-cursors were observed prior to the ELM crashes in the reference discharges, consistent with the PEST and ELITE predictions. While these ELM-free discharges otherwise suffer radiative collapse, pulsed 3-d magnetic fields were used to trigger ELMs for impurity control [5].

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Characterization of transient particle loads during lithium experiments on the National Spherical Torus Experiment

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Transient events such as Edge Localized Modes (ELMs) or disruptions can lead to large power loads in the divertor plates of tokamak experiments. These events can cause significant erosion and are detrimental to the lifetime of the plasma facing components in that area. The material response is determined by the particular characteristics of the transients, such as amplitude and duration. This makes understanding the impact of ELMs a complex problem and a major challenge. In this study, an effort is made to characterize these ELMs and other transients based on their properties. This is achieved by making use of the High Density Langmuir Probe (HDLP) array installed in the divertor region of National Spherical Tokomak eXperiment (NSTX).

The details of the use and implementation of the HDLP array can be found in ref. [1, 2]. Briefly, it is connected to custom designed electronics system that allows biasing of the probes and collecting the signals. The electronics enable signal amplification and noise reduction, and permit the array to be configured both as a set of single Langmuir probes and triple Langmuir probes (TLP). The HDLP array has a radial spatial resolution of 3 mm and temporal resolution of 4 μs when operated in TLP mode. This high spatial and temporal resolution of the HDLP array thus provides unique capabilities for characterizing ELMs.

Typically, the evolution of an ELM is characterized by a steep rise and a gradual decrease of current signal. This burst like structure is seen by Langmuir probes as a rise in the ion saturation current with a width of a few microseconds. Despite previous experience during lithium experiments showing the elimination of ELMs [3,4], LLD experiments have been performed when transients occurred. This study entails gathering statistics of typical ELM-like events for various shots, including those with the strike point on LLD. Also, the time evolution of the ion saturation current as measured by four different triple probes which are radially separated may allow investigation of radial propagation of ELM events in the Scrape off Layer (SOL) region. The details of this analysis are also presented.

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High Density and Pellet Injection Experiments with Lithium Coated Wall on FTU Tokamak

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Abstract. Experiments with a Lithium coated wall in FTU have given encouraging results for a liquid metal wall to be considered as a Plasma Facing Component in a fusion reactor. Vaporization caused by contact of a Capillary Porous System with the SOL plasma produces, after 2-3 discharges, a wall coating of about 10 monolayers of Li atoms and is accompanied by the reduction of most impurity lines (O, Mo, Fe) and by a strong reduction of the particle recycling. Very peaked density profiles have been produced with gas puffing only. Energy transport shows a transition to an improved regime, i.e. 1.2-1.4 times ITER.97-L scaling, when the density peaking factor (peak/volume average) exceeds the threshold value of 1.8. Deuterium fuelling pellets were injected for the first time in presence of a significant amount of Lithium. A preliminary analysis of particle transport shows a Bohm gyro-Bohm type diffusion coefficient as well as the existence of an inward particle pinch which is needed to explain the further density peaking taking place on a diffusion time scale (~50 ms) after the completion of the pellet ablation process. The issue of pellet ablation and particle deposition will be addressed comparing code predictions with density profile evolution measured by a fast, high resolution CO_2 interferometer. Finally, an MHD analysis of the post-pellet phase will also be presented in comparison with previous observations

Status and prospect for the development of Liquid Lithium Limiters for Stellarotor TJ-II

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Stellarator concept is considered as an encouraging approach for fusion reactor development because it basically free from extreme thermal load events. However, the potential problem of impurity accumulation must be taken into account. In the last years, the TJ-II has been operated with lithium coated wall that is provided by vapor deposition method with ovens. In comparison with other high and low Z plasma facing materials very promising results in density control, plasma reproducibility and confinement characteristics have been obtained, significantly enlarging the operational window of the machine even when only partial wall coverage with Li was achieved.

The next step in the improvement of TJ-II Heliac plasma performance is the development of two mobile poloidal liquid lithium limiters (LLL) allowing further progress in achievements of enhanced energy confinement owing to effective impurity and particle control. Experimental possibilities, design, structural materials and main parameters of LLL based on capillary-pore structure (CPS) filled with liquid lithium are considered. Status of LLL creation is presented.

Understanding in hydrogen isotope interaction with liquid lithium surface is an important aspect of lithium technology development for fusion reactor application. Study of deuterium sorption / desorption process on lithium surface is stipulated in experiments with LLL and investigation method is considered.

The development of lithium CPS based devices decreasing intensity of plasma-wall interaction on the central "groove" of TJ-II vacuum camera is proposed as the further step in plasma performance improvement owing to decrease in impurity flux from the wall.

Effect of Lithium Wall Conditioning and Impurities in LTX

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The Lithium Tokamak Experiment (LTX) is the first magnetic confinement device designed to have lithium plasma-facing components (PFC's) that surround nearly the entire plasma. Lithium coatings are evaporatively deposited onto the stainless-steel surfaces of shells that are designed to be conformal to the last closed-flux surface.

Two filterscopes study impurity optical line emission: one is on the mid-plane aligned to view the center-stack, and another is aligned to view one of the small Molybdenum limiters on the edges of the lower shells. Filters allow measurement of oxygen, carbon, and lithium emission, as well as H-alpha light. Small spectrometers with ~0.1nm resolution are employed as broad survey instruments in the 377-592nm range to complement the filterscopes. In addition, an AXUV diode array is used as a bolometer to detect the total radiated power. Finally, an existing XUV spectrometer is being reconditioned to measure emission from high-Z impurities and higher ionization states in the 5-40nm range.

Plasma operations within several hours after evaporating about 4g of lithium produced dramatic reduction in the OII emission, which typically continued to decrease over the course of the run day. Depending on the amount of lithium evaporated, the solid coating could take several days to lose the capability to pump hydrogen. Subsequent plasma operation without additional lithium evaporation showed substantial increase in OII emission with each shot, finally recovering the levels seen during bare wall operation.

Fueling of LTX Plasmas with Lithium Plasma Facing Components

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The Lithium Tokamak eXperiment (LTX) is designed to reduce particle recycling with lithium coverage on the plasma facing components (PFCs). A conformal shell with a stainless steel surface surrounds ~90% of the plasma edge, and when coated in lithium, acts as an efficient sink for impurities, protons, and atomic hydrogen. This sink prevents “recycling”, where large numbers of low-energy neutrals re-enter the plasma, raising the plasma edge density and lowering edge temperatures. With lithium PFCs suppressing the recycled particle source, the edge neutral density is expected to fall, and external fueling requirements are expected to increase dramatically.

LTX has a diagnostic suite designed to study these predictions. Two fast neutral pressure gauges provide global measurements of the neutral particle inventory, yielding the net neutral particle flux into the plasma. Two individual H_{alpha} viewing chords and a pair of Ly_{alpha} arrays provide relative neutral density measurements over a substantial fraction of the plasma volume. Additionally, a fast visible camera can be equipped with an H_{alpha} filter to monitor the penetration of injected fueling particles. A scanning 2mm interferometer and a fixed 1mm interferometer provide electron density measurements with a fast time response.

Initial experiments, using cold lithium coatings evaporated on to the shell surface, indicate at least a seven-fold increase in the fueling requirements over the high-recycling discharges produced with bare stainless steel PFCs. In the absence of additional external fueling, the neutral emission is reduced to low levels, and subsequent fueling pulses produce emission that is quickly burned out, indicating a substantial reduction in recycling. Higher performance discharges in LTX require near constant fueling from a conventional gas puffer, which sources a large number of particles into the plasma edge. The current state of the diagnostic and fueling hardware will be presented, and the design and characterization of a fueling system that will provide a larger fraction of core fueling will be discussed.

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"Plasma facing surface composition during Li evaporation on NSTX and LTX"

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Evaporated lithium coatings can react with water in the base vacuum to produce lithium hydroxide and hydrogen.

$$2\ Li + 2\ H_2O \rightarrow 2\ LiOH + H_2$$

Since tokamaks typically do not have ultra high vacuum conditions, this process can occur in the time interval between lithium evaporation and the next discharge. The resulting PFC surface should be considered as a mixed material rather than a pure 'lithium coating'. We present calculations of the flux of water from the residual vacuum to PFCs in NSTX and LTX under various conditions. To avoid reactions with residual vacuum gasses an ultra-high vacuum (≤1e-8 torr) is required and may be achievable by a large-scale lithium getter pump.

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Summary of current R&D efforts for liquid metal based blankets and ITER TBM

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The core focus of the liquid metal research and development activity primarily derives from the dual coolant lead lithium (DCLL) blanket concept. The DCLL concept proposes the use of a flow channel insert (FCI) to electrically insulate the PbLi flow from current closure paths in the ferritic steel (FS) walls to reduce the MHD pressure drop, while thermally insulating the self-cooled PbLi breeder region from the helium cooled FS walls hence achieving high temperature for a higher thermal efficiency. At present, the concept is applied to the US reference DEMO blanket design and to the ITER TBM. Despite a significant reduction in MHD pressure drop by the FCI, the PbLi MHD velocity profile is unstable. It has been shown numerically that the non-uniform heating in PbLi breeder region can lead to strong buoyancy forces and drive secondary flow. Flow reversal is found to occur near the gaps between the FCIs. A significant research effort is on the modeling development of the 3D unsteady MHD code HIMAG. The code utilizes consistent and conservative numerical schemes for determination of current density on an unstructural collocated mesh, and has simulated several practical design problems including flow distribution characteristics in blanket PbLi manifold. The interrelated behavior between the MHD flow, heat transfer, temperature and temperature gradient compels recent modeling effort taking into account this coupling effect and addressing thermofluid-MHD flow dynamics as a whole. Modeling development was further expanded to study the mass transfer including corrosion and tritium transport under the impact of magnetic field in concert with thermofluid flow and chemical potentials.

Flow distribution among the multiple parallel channels plays an important factor for the design. Experimental investigations were conducted to understand the mechanisms that determine the division of flow from a single supply channel to a series of parallel blanket channels. Empirical correlations were obtained to express the flow rate distribution among the parallel channels though the magnetic interaction parameter. Experiments were also conducted to quantify factors that govern MHD flow transition from 3D to 2D.

In the tritium transport area, experiments were conducted under the US-JA Titan collaborations, specifically to characterize solubility and diffusivity of tritium in PbLi. Physical database in this area has been updated. In addition, models were developed to address the effectiveness of the use of the vacuum permeator for tritium extraction from PbLi. Safety analysis and modeling were performed and led to the completion of the RPrS for the DCLL TBM. In this paper, recent progresses made in the aforementioned research areas are summarized.

Improvement of compatibility of liquid metals Li and Pb-17LI

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Liquid metals Li and Pb-17Li are considered as a coolant and a tritium breeder for the blanket systems of fusion reactors [1, 2]. The important issue is the compatibility of liquid metals with structural materials. The non-metal impurities such as oxygen, nitrogen and hydrogen, dissolved in the melts increase the activity of the liquid metals. The solubility of the metal elements of steels in the liquid metals is larger at the higher concentration of the non-metal impurities in the melts [3]. The stable oxides such as Er_2O_3 and Al_2O_3 possessed corrosion resistant in the liquid metals [4]. The structural materials can improve the compatibility by the coating of these oxides on the surface. The purpose of the present study is to summarize the corrosion data based on the impurity control and the coating technology toward the improvement of the compatibility.

The initial impurity of the liquid metals Li and Pb-17Li was determined by the chemical analysis such as ICP-MS and ammonia extraction method. Then, the impurity was adjusted by the addition of Li_2O, Li_3N [5] and carbon to investigate the influence on the compatibility. The test material is the reduced activation ferritic martensitic steel, JLF-1 (Fe-9Cr-2W-0.1C), and the oxide dispersion strengthened (ODS) steel (Fe-9Cr-2W-0.14C-0.23Ti-0.29Y-0.16O). The corrosion tests were performed at a static condition and the flowing condition, which was made by an impeller induced flow in the mixing pot [6]. The compatibility was investigated by the chemical analysis of the liquid metals and the metallurgical analysis for the tested specimens.

The results showed the compatibility was affected by the non-metal impurities and the flow in the Li. It was newly found that the influence of the oxygen dissolved in Li was large as the same as that of nitrogen. This was possibly because the corrosion products formed in the Li with high oxygen concentration was not stable and dissolved in the Li as the Li_2O dissolved in the Li. On the contrary, the addition of carbon to Li prevented the phase transformation [7] from martensite to ferrite of the steel since the carbon in the steels did not dissolve into the Li with high concentration of carbon. The occurrence of the erosion-corrosion in the flowing condition was detected. The mechanism was explained by the peeling off of the corroded surface. The corrosion characteristic of the Er_2O_3 coated specimen in the Li was investigated and it was found that the coating possessed the corrosion resistant though there were some cracks, which were made by the difference of the thermal expansion ratio between solidified Li and the coating. The corrosion data of JLF-1 and ODS steels exposed to Pb-17Li up to 3000 hours was obtained. The solubility of the metal elements in Pb-17Li was summarized with the reported data and that obtained by the immersion of the pure metals of Fe, Cr, W and Mo. The corrosion resistant of the specimen coated by Er_2O_3 and Al_2O_3 was investigated.

These compatibility data was summarized based on the modeling as diffusion and mass transfer. The necessary condition for the improvement of the compatibility was summarized as conclusion.

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Cluster/Aerosol Formation and Hydrogen Co-deposition by Colliding Ablation Plasma Plumes of Lithium and Lead

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In high-repetition inertial confinement fusion (ICF) reactors, the interior of target chamber is exposed repeatedly to intense pulses of fusion neutrons, X-rays, unburned DT-fuel particles, He-ash and pellet debris in the form of C_xH_y, the total deposited energy of which could amount to a few tens of Joules/cm^2/implosion. As a result, wall materials are subject to ablation, emitting particles in the state of plasma. Ablated plasma particles will either be re-condensed elsewhere on the wall or collide with each other in the center-of-symmetry region, if any, of the target chamber. Colliding ablation plasma particles may possibly form clusters which can grow into aerosol, floating thereafter, in a yet-to-be explored manner. Subsequent laser beams may be scattered and/or deflected to affect pellet implosion performance. Despite its critical importance, the chamber clearing issue has not widely been recognized in the ICF research community.

In our previous studies [1,2,3], the dynamics and re-condensation behavior of colliding plasma plums of selected materials for solid wall ICF reactors, including W and C, was investigated using a unique experimental setup referred to as LEAF-CAP [1] (for the Laboratory Experiments on Aerosol Formation by Colliding Ablation Plumes). The present work focuses on Li and Pb, materials envisaged for reactors with a liquid first wall. As shown in Fig. 1, in the LEAF-CAP setup two arc-shaped targets are irradiated in vacuum by 6ns pulses of 3ω–YAG laser at 10Hz, the deposited energy of which ranges from 1 to 10 Joules/cm^2/pulse. Ablation plasma plumes thus generated are to collide with each other in the center-of-arc region which is diagnosed by a CCD/ICCD camera, qudrapole mass analyzer, Langmuir probe, visible spectrometer, etc.

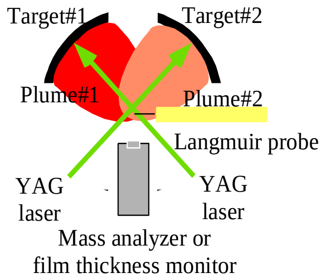

Fig. 1 The LEAF-CAP setup [1]

From ICCD camera observations shown in Fig. 2, Li/Li plumes collide to merge with each other, traveling to slow down in the compound velocity direction, suggestive of an inelastic process. Consistently, cluster ions of Li_2^+ have been identified in mass spectra, as shown in Fig. 3. These findings are similar to those on colliding C/C plumes, forming C_n clusters and nano-scale aerosol [2]. In contrast, colliding Pb/Pb and Li/Pb plumes appear to penetrate each other, similarly to W/W plumes [3].

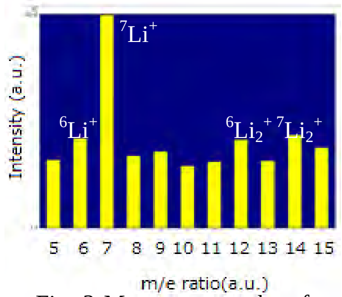

Fig. 3 Mass spectra taken for colliding Li/Li plumes.

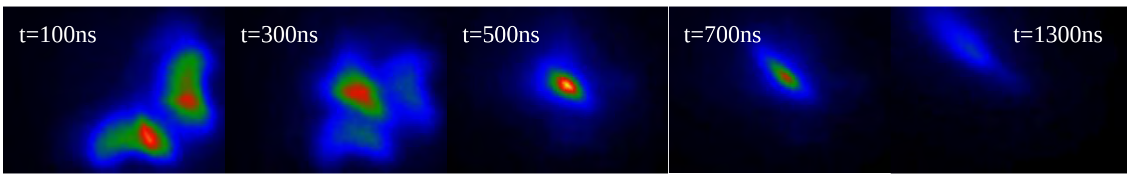

Fig. 2 ICCD camera observations of colliding Li/Li plumes generated at 10 J/cm^2/pulse (laser irradiation at t=0).

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Hydrogen transports at interface between gas bubbling and liquid breeders

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The recovery of tritium from liquid breeders, such as Li, Pb-17Li and Flibe, is one of the critical issues for the self-sustaining D-T fueling of fusion reactors. The recovery systems have been designed based on tritium transports through gas/liquid interface, and also the experimental studies have been carried out [1-3]. Authors have investigated the availability of hydrogen sensors made of a proton conducting solid electrolyte as tritium monitoring systems [4-5]. In the previous work, hydrogen concentration in molten salt Flinak was controlled by sweep gas according to the gas- liquid equilibrium [6]. It is known that the bubbling has beneficial influence on the hydrogen transports by the convection of the fluids and the generation of the fresh interface between the bubble surface and the fluids [7]. In the present study, the characteristics of hydrogen transports at the interface between gas bubbling and liquid breeders were experimentally studied. The experimental results were analyzed by model evaluation based on mass transfer.

The experiments were performed with liquid metals Li and Pb-17Li and molten salt Flinak as the same condition. The temperature of the melts was increased to 600°C under Ar atmosphere. Then, hydrogen gas (1 atm) was injected to the melts using I shape nozzle. The inner diameter was 0.6mm. The flow rate of hydrogen gas was 12cc/min. After the saturation of the hydrogen in the melts, the injection of hydrogen gas was stopped and Ar gas was injected to the melts to recovery the hydrogen. The flow rate was 27cc/min. These processes were repeatedly carried out. The theoretical diameter of the bubble in the melts was determined as 3.2~6 mm according to the balance between buoyancy force and surface tension [8]. The rising velocity of the bubble in the melts was evaluated according to the balance between the drag force and the buoyancy force. The Re number for bubble was used to evaluate the drag force coefficient of the bubble. The hydrogen concentration of exhaust gas was measured by the solid electrolyte hydrogen sensor. Then, the hydrogen transport to the melts was evaluated by mass transfer model.

The results indicated that the hydrogen concentration in the melts was controlled by the injection of the gases. The results for Li, Pb-Li and molten salt Flinak indicated that the transient of the hydrogen concentration in the melts were influenced by the fluid characteristics, such as the solubility of hydrogen and the dissolution ratio.

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The IFMIF Target Facility engineering design and the validation of key issues within the IFMIF-EVEDA Project

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The Engineering Validation and Engineering Design Activities (EVEDA) of the International Fusion Materials Irradiation Facility (IFMIF) Project is part of the Broader Approach Agreement between Japan and Euratom signed in 2007. The project is coordinated by a team established in Rokkasho, whereas the technical contributions are provided by local research institutes via the Implementing Agencies (JAEA and Fusion for Energy) on the basis of Procurement Arrangements (PA). Six PAs have been established to cover the design and validation of the IFMIF target facility. Main contributors in Japan are JAEA and several Japanese Universities, whereas in Europe ENEA provides contributions specific to their experience.

With the IFMIF Comprehensive Design Report (January 2004) as baseline, the Engineering design task shall provide all information necessary to decide on the construction of the facility. Key challenges are

- The design of the target assembly to assure a stable high velocity lithium flow of defined thickness to generate the forward peaked neutron flux and safely evacuate the beam power. Two concepts of differing in complexity and waste generation will be realized.
- The proper selection of the trapping materials and the design of the lithium purification system to limit nitrogen, carbon and corrosion products content important for corrosion/erosion mitigation and the radiological impact due to tritium and beryllium.

These two issues are investigated experimentally in the current phase to demonstrate the feasibility of the concepts and to optimize their design. The experimental facilities have been constructed and their commissioning is under way. The engineering design is still in the definition phase, in which the general layout is discussed, different concept are evaluated and fundamental issues are studied.

The presentation will describe the evolution of the target and loop design from the start of IFMIF considerations until the current concepts, address the implications of purification and impurity monitoring on operations and safety and describe the validation approaches selected for these issues.

Status of the activities for the development of the remote handling techniques for the maintenance of IFMIF target assembly system

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The International Fusion Materials Irradiation Facility is a facility where fusion candidate materials will be tested up to a damage rate of about 150-200 dpa. Materials are tested by using a high-energy neutron flux produced by a stripping reaction of two D^+ beams impinging on a free surface liquid lithium jet flowing in a concave backplate on the Target Assembly. The Target Assembly is located in the most severe region of neutron irradiation (50 dpa/fpy), and it must be designed to be exchanged remotely. Two design options of the target system are under development: the so called integral target, in Japan, and the one based on the replaceable backplate bayonet concept in Europe. The first target concept foresees the removal of the entire target assembly from the test cell and its transportation to a hot cell where the maintenance is performed. For the refurbishment of the second target concept two potential approaches are under investigation: the first relies on the possibility to perform the entire refurbishment of the target assembly, including inspection and testing, inside of IFMIF test cell cavern while the second one foresees to perform its refurbishment off-line in a dedicated hot cell. The refurbishment process of the target assembly is a rather complex activity which requires sophisticate remote handling technologies and tools. It consists of a number of remote handling tasks and, among these, the backplate replacement, the cleaning of surfaces from lithium solid deposition, the inspection and repair of component inside of the target body itself and the diagnostics substitution are considered critical. In fact to fulfil the stringent requirement of IFMIF plant availability (70%) all these refurbishment operations have to be performed during the annual shutdown of the facility within a one week period. In the paper, the status of the ongoing remote handling activities are discussed together with the outcomes of the preliminary tests carried out and with the design solutions adopted to optimize the entire refurbishment process of the Target Assembly system.

IFMIF LITHIUM TARGET

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IFMIF is an accelerator-driven irradiation facility that is being developed in the frame of the Europe-Japan Broader Approach agreement, with the goal of providing a high intensity, fusion-like (14 MeV-peaked) neutron source for testing candidate materials under thermal and irradiation conditions similar to those expected in future fusion power plants. IFMIF neutron source is based on nuclear stripping reactions occurring within a free surface jet of liquid lithium flowing on a windowless 10 MW power target exposed to 2x125 mA - 40 MeV deuteron beams impinging on a 20x5 cm^2 footprint area. To remove such a high power density, the lithium in the target must flow at a typical velocity of 15 m/s, while maintaining a stable thickness of the jet (25 ± 1 mm) in order to assure the requested neutron efficiency and avoid the damage of the structural material. A hydraulic channel with a suitable curved-shape profile is foreseen to guide the lithium jet on the exposed surface (the so-called "back-plate") of the target and to generate the centrifugal forces aimed at increasing the pressure in the liquid metal in order to prevent its boiling. Two different concepts are conceived for the IFMIF target system: the integral concept developed in Japan by JAEA and the removable bayonet back-plate concept developed in Europe by ENEA. Although technically more complex, the bayonet concept has the advantage to limit the nuclear wastes to be disposed and to make the remote handling replacement operations easier and faster. It consists of a replaceable element (the back-plate) that couples to the permanent structure (interface frame) of the target assembly by means of two lateral guides each equipped with a roller skate mechanism. A high performance gasket is placed on the back-plate to assure the required sealing between it and the frame. The skates have the double function to guide the back-plate on the frame and, once in the right position, to provide the necessary closure force on the gasket. The closure force on the other two sides (upper and bottom sides) of the back-plate is applied by means of bolts. The connection of the target system with the lithium loop and the accelerator beam duct is a very delicate aspect as it has to assure a fast, safe, easy and reliable way to operate under remote control. The current candidate solution is based on the Garlock® Quick Disconnecting System (QDS®) which permits to connect and disconnect the target by simply working on a few bolts. Another key feature of the target design is the shape of the lithium channel profile. In order to avoid hydraulic instabilities that might result from a sudden change of pressure in correspondence of geometrical discontinuities of the profile, special attention must be paid to the design of the channel geometry. Concerning this point, a variable-curvature profile for the lithium channel has been calculated by ENEA using an analytic approach based on the simplified Navier-Stokes equations obtained by imposing a gradual pressure change to the free surface flow. The engineering design work of the IFMIF lithium target represents a quite involved task that covers different and often correlated activities including thermohydraulics and thermomechanical calculations, neutronic analysis, safety studies, technological choices, lifetime assessment. In particular, the latter task is a very crucial issue since the lifetime of the back-plate has a strong impact on different aspects of IFMIF design and operating strategies, as well as on maintenance plans. An estimation of the expected lifetime can be made by identifying the principal modes of failure of the back-plate including corrosion-erosion phenomena, neutron-induced swelling and creep, irradiation embrittlement, thermal fatigue due to flow instability. In this contribution, a brief overview of the engineering design aspects outlined above for the IFMIF lithium target system will be presented with particular reference to the bayonet concept developed by ENEA.

Module of Lithium Divertor for KTM Tokamak

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Activity on projects of ITER and DEMO reactors have shown that solution of problems of divertor target plates and other plasma facing elements (PFE) based on the solid palasma facing materials cause serious difficulties. Problems of PFE degradation, tritium accumulation and plasma pollution can be overcome by the use of liquid lithium - metal with low Z. Application of lithium will allow to create a self-renewal and MHD stable liquid metal surface of the in-vessel devices possessing practically unlimited service life; to reduce power flux due to intensive re-irradiation on lithium atoms in plasma periphery that will essentially facilitate a problem of heat removal from PFE; to reduce Z_{eff} of plasma to minimally possible level close to 1; to exclude tritium accumulation, that is provided with absence of dust products and an opportunity of the active control of the tritium contents in liquid lithium. Realization of these advantages is based on use of so-called lithium capillary-porous system (CPS) - new material in which liquid lithium fill a solid matrix from porous material.

The progress in development of lithium technology and also activity in lithium experiments in the tokamaks TFTR, T-11M, T-10, FTU, NSTX, CDX-U, LTX, CPD, HT-7 and stellarator TJ II permits of solving the problems in development of steady-state operating lithium divertor module project for Kazakhstan tokamak KTM (R/a = 0.9/0.45 m, B_T = 1 T, $J_p \approx 0.75$ MA, τ=4-5 s). At present the lithium divertor module for KTM tokamak is under development and manufacturing. Initial heating up to 200°C and lithium surface temperature stabilization during plasma interaction up to 600°C will be provided by external system for thermal stabilization due to circulation of the Na-K liquid. Lithium filled tungsten felt is offered as the base plasma facing material of divertor.

Development, manufacturing and experimental research of lithium divertor module for KTM will allow to solve existing problems and to fulfill the basic approaches to designing of lithium divertor and in-vessel elements of new fusion devices generation - fusion neutron source for atomic energy aplication and fusion reactor of DEMO type, to investigate plasma physics aspects of lithium influence, to develop technology of work with lithium in tokamak conditions.

Results of designing, calculations and manufacturing of lithium divertor module are presented.

"Fast flowing liquid lithium divertor concept for NSTX"

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Innovative concepts using fast flowing thin films of liquid metals (like lithium) have been proposed for the protection of the divertor surface in magnetic fusion devices. However, concerns exist about the possibility of establishing the required flow of liquid metal thin films because of the presence of strong magnetic fields which can cause flow disrupting MHD effects. A study was performed under the ALPS program in the US to evaluate liquid lithium based divertor protection concepts for NSTX. Of these, a promising concept is the use of modularized fast flowing liquid lithium film zones, as the divertor (called the NSTX liquid surface module concept or NSTX LSM). The dynamic response of the liquid metal film flow in a spatially varying magnetic field configuration is still unknown and it is suspected that some unpredicted effects might be lurking.

The results being reported here provide qualitative and quantitative information on the liquid metal film flow dynamics under spatially varying magnetic field conditions, typical of the divertor region of NSTX. The liquid metal film flow dynamics have been studied through a synergic experimental and numerical modeling effort. The Magneto Thermo-fluid Omnibus Research (MTOR) facility at UCLA has been used to design several experiments to study the MHD interaction of liquid gallium films (which has been used as a surrogate for lithium) under a scaled NSTX outboard divertor magnetic field environment.

A 3D multi-material, free surface MHD modeling capability is under development in collaboration with HyPerComp Inc., an SBIR vendor. This numerical code called HIMAG provides a unique capability to model the equations of incompressible MHD with a free surface. HIMAG has been used to study the MHD interaction of fast flowing liquid metal films under NSTX divertor relevant magnetic field configurations through numerical modeling exercises.

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Lithium-based surfaces controlling fusion plasma behavior at the plasma-material interface

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The plasma-material interface and its impact on performance of magnetically-confined thermonuclear fusion plasma is considered to be one of the key scientific gaps in the realization of nuclear fusion power. At this interface high particle and heat flux from the fusion plasma can limit the material's lifetime and reliability and therefore hinder operation of the fusion device. The power dissipated over the plasma-wetted surfaces can lead to substantial degradation of materials. In particular, when operating in burning plasma, long-pulse regimes the plasma-facing components are exposed to large fluxes of helium compromising candidate materials, which include tungsten, graphite and beryllium. Both a neutron flux and particle ion flux (e.g. He, D, T) penetrate the plasma-facing solids at various spatial scales that modify the intrinsic properties of the materials. The plasma-material interface is a key region in the device since material can be emitted both atomistically (evaporation, sputtering, etc…) and/or macroscopically (i.e. during disruptions or edge localized modes).

This talk highlights the current limitations of existing materials and the methods to study them including: 1) in-situ tokamak PMI diagnosis, 2) PMI plasma test stands and 3) single-effect particle-beam test stands (e.s. surface science facilities). The talk will focus on current applicaton of lithium-based surfaces and their ability to manipulate core plasma performance. In particular this talk will compare the various strategies at using lithium as a plasma-facing surface. A discussion on the key mechanisms at the plasma-material interface will include the role of: hydrogen retention and recycling, recombination, sputtering, surface diffusion, surface morphology and surface chemistry.

The talk will also briefly outline the challenges that face future lithium-based plasma-facing components for the DEMO fusion device era and summarizes possible strategies at connecting single-effect test stand facilities, computational materials science and in-situ PMI diagnostics to elucidate on lithium-based materials development

Deciphering energetic deuterium ion interactions with lithiated ATJ graphite *

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As lithium has been used in fusion devices such as TFTR, CDX-U, FTU, T-11M, TJ-II and NSTX with great success, it is of interest to understand the fundamental mechanisms that are responsible for improved plasma performance and in particular hydrogen retention. Plasma performance improvements include a suppression of edge localized modes (ELM), reduced lower divertor C and O luminosity, and reduced deuterium recycling [1,2]. Previous laboratory work has shown that deuterium interacts with lithium-oxygen and lithium-carbon complexes predominantly compared to LiD bonding [3]. X-ray photoelectron spectroscopy (XPS) and thermal desorption spectroscopy (TDS) analyses indicate that deuterium binds via two states in lithiated graphite: 1) weak bonding due to dipole interactions induced by the high electropositive nature of lithium in carbon and oxygen and 2) covalent bonds to C and O. Computational atomistic simulations using self-consistent charge density-functional tight-binding (SCC-DFTB) method corroborate these observations indicating that the probability for D to bind to Li in C is three times higher than D bonding to C only [4].

The present study examines the surface chemistry revealed through XPS, as well as ultraviolet photoelectron spectroscopy (UPS) and low energy ion scattering spectroscopy (LEISS). With a probing depth of <10 nm, XPS is used to identify Li-O-D interactions in the O 1s energy range at 533.0 ± 0.5 eV, and Li-C-D interactions manifest in the C 1s energy range at 291.2 ± 0.5 eV. UPS is used to discriminate additional details regarding deuterium bonding represented by valence shell behavior. For pure lithium UPS reveals distinct peak structures characteristic of lithium carbonate (with distinctive peaks at 8.5, 11.5, and 15.2 eV), as well as lithium hydroxide (peaks at ~7 and 10 eV). Systematic experiments are performed using UPS for lithiated graphite, with spectra collection for virgin ATJ graphite, after lithium deposition, and following deuterium irradiation.

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Investigation of LLD Test Sample Performance Under High Heat Loads

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Future tokamaks currently under construction such as NSTX-Upgrade and ITER will experience peak divertor heat fluxes of 20-25 MW/m^2[1,2]. Adequate handling of these heat loads remains one of the most crucial unsolved issues for the next generation of toroidal confinement devices. Therefore, investigating novel possibilities for first-wall materials has become one of the primary thrusts of fusion technology research during the last several years. Recent research has demonstrated the capability of solid and liquid Li to withstand heat fluxes on the order of 10 MW/m^2 in the divertor region in NSTX[3]. These initial results motivate a deeper inquiry into the physical and chemical mechanisms governing the power handling capabilities of solid and liquid lithium coatings on a porous metal substrate.

A small prototype sample of the NSTX Liquid Lithium Divertor (LLD) was exposed to a diagnostic neutral beam (DNB) original developed for the Motional Stark Effect Laser-Induced Fluorescence System. The DNB bombarded the sample with neutral hydrogen at a power of ~1.5 MW/m^2 for 1-5 seconds. Like the LLD itself, the sample consists of a 22.2 mm think copper block with a 0.25 mm stainless steel liner brazed to its surface. The liner has a plasma sprayed molybdenum coating with a thickness of 0.165 mm and a porosity of 45%. Calibrated IR measurements of front face temperature and thermocouple measurements of bulk sample temperature were obtained. Exposures were performed on a bare porous Mo surface and subsequently on a surface coated with a 150 μm thick Li layer. Li-coated sample exposures occurred in the solid and liquid Li temperature regimes. Predictions of temperature evolution were derived from a simple 1D heat flux model and compared with experimental data. These results, together with detailed examination of the sample surface, demonstrated the effective power handling of a thin stainless steel liner with Li-filled porous Mo coating on a copper heat sink, suggesting usefulness as NSTX-Upgrade PFCs.

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Lithium technologies for edge plasma control

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Lithium technologies are increasingly more considered and already investigated in magnetically confined experiments for controlling the tokamak discharge. They are used for wall conditioning by pellets, heat and particle control by liquid and capillary-pore limiters and divertor plates as well as evaporation prior shot and dust injection at plasma edge during shot. In this report we summarize our activity on lithium operations and compare them with other approaches.

The technology for covering the lithium pellets of about 1 mm in size by a thin surface layer had been developed and used in T-10 plasma conditioning experiments [1]. Modifications of the Li liquid jet injector (~70-100 microns in diameter and ~5-20 m/s jet velocity) and their features preventing its reliable operation are briefly described. Results of T-10 experiments with the Li liquid jet injection [2] are presented.

For controllable injection of lithium dust that is industrially produced [3] a new rotary batcher has been designed, tested and commissioned at T-10 tokamak. This dust injector provides a flow of lithium dust (~40 microns in diameter) with about 5 m/s velocities and a rate of 10^{20-21} atoms/s. The injector operation in this winter experimental cycle of T-10 is foreseen and we intend to present the first results.

Integrated modelling of tokamak plasmas is being developed in the frames of ITER/ITPA and EFDA activities. It is aimed at joint operation of several sophisticated codes, which separately describe MHD equilibrium and stability, transport, heating and current drives, fueling, SOL and divertor plasmas in tokamaks. This approach is a long-term strategic task for fusion plasma simulators. However, development of many tokamak subsystems (e.g. fueling and pumping, wall conditioning, heating and current drive, divertor etc.) and preliminary analysis of experimental data require simpler approaches for simulations of coupled core and edge plasmas.

In this report we present an improved version of the semi-analytical model [4] which couples core and SOL regions of multi-species plasma and allows simulating of steady state tokamak regimes with a broad range of plasma actuators. Those include gas puffing and pumping, pellet and dust injection, auxiliary and fusion heating and recycling. This approach is helpful for analysis of plasma behavior when impact of different actuators on a tokamak plasma is varied. The influence of Li dust injection in ITER&DEMO conditions had been evaluated by the model [2]. Recent model improvements explore the “2 point onion skin” approach of Ref. [5]. They allow us to evaluate radial distributions of heat fluxes at divertor plates. The model was tested and verified on the experimental results of ASDEX Upgrade and NSTX. Application of the model for analysis of Li dust experiments on T-10 is planned. Model simulations of the divertor operation of the compact fusion neutron source based on spherical tokamak [6] are presented.

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Electrical characteristics of lithium surfaces exposed to a plasma

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Lithium is becoming a material of high potential for Plasma Facing Components in a Fusion Reactor. The reasons for that are its low atomic number, high capability of particle and power handling, in particular in its liquid form, and its low melting point, thus opening the possibility of developing liquid PFC concepts at moderate temperatures. To date, a direct relation between the enhanced performances of Li based plasma devices and the associated low recycling of cold Li surfaces (T<400ºC) has been postulated. However, tritium inventory control in a reactor calls for a high recycling wall. It is expected that D and T recycling in liquid lithium could become unity at high enough temperatures (450 ºC), so that a compromise between high recycling and low vapour pressure in the range 400-500 ºC must be achieved. However, it is unknown whether the positive effects on plasma confinement will be lost under high recycling conditions.

In the present work, a search for other positive effects of lithium surfaces, other than low recycling, has been addressed. The I-V characteristics of lithium covered metallic electrodes have been recorded in a He plasma. It has been found that al low negative potentials of the electrode, an apparent excess of ion current is driven, not corresponding to a pure secondary electron emission from the lithium surface. This anomalous current has been interpreted as an effect of the strong secondary ion emission through sputtering and the concomitant re-trapping of these low energy ions by the biased electrode. Details about these phenomena, its dependence on lithium surface conditions and the impact on reported observations in Li-based plasma devices will be presented.

Lithium particle detector for fusion applications.

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Lithium can be continually supplied to a plasma by delivery systems that utilize a stream of lithium particles injected into the scrape off-layer[1]. This is particularly useful to replenish lithium on plasma facing surfaces during long pulse discharges. Some of the injected lithium particles accumulate on interior vessel surfaces and have been detected by an electrostatic dust detector in NSTX[2]. We report on the calibration of the electrostatic dust detector with lithium particles. Detection of beryllium dust is part of the ITER dust strategy[3] and an absolute detection accuracy of 50% has been specified[4] but experiments with berylium dust are hazardous because of its toxicity. Lithium particles may be useful as a proxy for beryllium until beryllium measurements become available.

The electrostatic dust detector is based on a grid of interlocking circuit traces biased to 50 V, and has been developed for the detection of dust on remote surfaces in air and vacuum environments. It has been absolutely calibrated for lithium and carbon particles. The sensitivity in vacuum for carbon particles with a count median diameter of 2.14 µm was found to be 0.15 ng/cm^2/count for a 51 mm detector with cover mesh in vacuum conditions, and for lithium particles of average diameter 44 µm is 14.5 ng/cm^2/count for a 13 mm detector without cover mesh in vacuum. We will describe the special conditions for the measurements of lithium particles.

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Laboratory Investigation of an Effect of Lithium on ICRF Antenna in DEVeX

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Abstract

Impurity production at Ion Cyclotron Resonance Frequency (ICRF) antenna has been studied for a long time for the associated problem of radiation and power-handling limit. [1] In order to study erosion of ICRF antenna and material-plasma interaction for fusion plasma condition, the Divertor Erosion and Vapor Shielding eXperiment (DEVeX) facility was built at University of Illinois at Urbana-Champaign. [2] The DEVeX consists of a theta-pinch coil driven by pulse discharge to produce hot and dense plasmas. Previous experiments have shown that plasma has n_e in the order of $10^{21} \sim 10^{22}/m^3$ and T_e of 10 - 100 eV for 80 μs. [3]

To investigate an effect of lithium on an ICRF antenna, a small size ICRF antenna, connected to 13.56 MHz RF power supply, is manufactured and inserted in DEVeX chamber. A lithium magnetron sputtering system is used to deposit lithium on ICRF antenna. A triple Langmuir probe diagnostic is carried out at various positions near the ICRF antenna to understand reaction of plasma and ICRF and effect of lithium. Plasma characterization, antenna erosion at various plasma energy, and effect of Lithium coating for erosion will be presented.

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Capillary Wicking of Lithium on Laser-Textured Surfaces

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Three 1”x3.5” (0.125” in thickness) SS-316 laser-textured samples were recently tested to evaluate their suitability for lithium wicking in different orientations. These samples were prepared by making 20 passes per line in the vertical direction with spacing of 50 µm per line. The sample was placed in a sealed quartz cylinder and heated to approximately 1100 °F by five radiant heaters with a power of 750 W. Sample temperature was monitored by four thermocouples instrumented into the back of each sample at locations of 1”, 1.75”, 2.5” and 3.25” from the bottom. The lithium bath was heated to approximately 1050-1075 °F by a 300 W ring heater and the five radiant heaters. Lithium bath temperature was monitored using two thermocouples coming up into the cup. The sample was then dipped into the lithium bath to observe lithium wicking. Each sample was tested at a different orientation: 90 degree, 45 degree texture up, and 45 degree texture down. The 45 degree texture up scenario showed the most vigorous and fully-wetted wicking out of the three samples.

The sample thermocouples were designed to monitor the lithium wicking front. A temperature drop in the sample thermocouples would be expected as the cooler lithium metal wicked up the sample. However, in the 45 degree texture up test the two lower thermocouples registered a slight drop in temperature (a few degrees) followed by a spike upward in temperature (5-10 degrees). This behavior is consistent with other lithium wicking tests conducted at ARL Penn State. In order to determine the cause of this behavior, a SS-316 sample with no laser-texturing was tested in the 90 degree orientation and in similar thermal conditions to those of the 45 degree texture up test. When this bare sample was dipped into the lithium, all thermocouples started to register a drop in temperature (15-40 degrees) with no subsequent spikes, with the most significant change occurring closest to the lithium. This temperature drop corresponded to heat conduction from the steel sample to the cooler lithium bath. These results explain the behavior of the two lower thermocouples in the 45 degree texture up test. The small dip in temperature corresponds to heat loss through conduction to the lithium bath. The sharp rise in temperature corresponds to the arrival of the lithium wicking front, possibly led by lithium vapor then followed the capillary pumped liquid. The capillary pumped liquid lithium is heated rapidly due to the radiant heaters and transfers the heat to the sample because of its high thermal conductivity.

Other sample textures as well as a titanium-zirconium-molybdenum alloy (TZM) will be tested and will be discussed.

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NSTX Liquid Lithium in Vacuo Delivery System

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Lithium evaporator ovens have been used to deposit lithium coatings on the NSTX lower divertor and other plasma facing surfaces. The use of this approach in extensive experimental campaigns is limited by slow evaporation rates, mild focusing, and small capacity requiring frequent reloading. In some experiments, relatively fast deposition of several grams of lithium at a specified location would be desirable. To accomplish this, a system has been devised which can deliver ~0.5 g/s of molten lithium at ~320°C into a vacuum to a selected location. This MAnaged INjection FIller for Liquid Lithium (MAIN FILL) system has been constructed from a valve bellows with a precision threaded rod and nut attached for controlled delivery. The liquid lithium passes through several different fittings, a high temperature valve, a 90° bend, and travels for 110.5 cm. The system is described and photographs show the results of liquid lithium delivery on a cold stainless steel surface, on cold lithium, on hot molten lithium, and on a hot porous molybdenum surface. A configuration with additional applications on NSTX is presented.

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Dynamics of deuterium retention and sputtering of Li-C-O surfaces

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The main challenge for computer simulations of the dynamics and chemistry in materials mixed with lithium is the high polarizability of lithium when interacting with most other materials, inducing position and dynamically dependent redistribution of charges across all atoms in the system. The objectives of our research are two-fold: 1) To develop realistic methods for computational simulation of the dynamics of the Li-C-O bombarded by slow D, validated by experiments and by advanced methods of computational chemistry; 2) To explain the specifics of the chemistry of deuterium bonding in lithiated/oxidated carbon.

The mutual charging of the atoms of lithiated materials necessitates the treatment of long-range and nonbonding interactions (Coulomb, Lenard-Jones) in addition to the short range covalent and metallic bonds, which introduces conceptual and numerical difficulties for both building potentials and application of classical molecular dynamics. Upon reviewing outcomes of the classical approach, we employ the Self-Consistent-Charge Density-Functional Tight-Binding (SCC-DFTB) method [1], an approximation to the fully-quantal DFT method, in which only valence orbitals are considered and the most difficult integrals are parameterized in advance, enabling computational efficiency about 1,000 times faster than *ab initio* quantal methods. For the dynamics, electronic motion is treated quantum-mechanically, thus producing self-consistently forces and charges for classical nuclear motion at each time step.

• The experiments of the Purdue group in situ and with the PMI probe in NSTX indicated that lithium will always bind with oxygen (when present) and carbon, and impacting deuterium will interact preferably with existing Li-O and Li-C structures rather than with C [2]. Application of the SCC-DFTB method to simulate the dynamics in this experiment has required several hundreds of atoms in a simulation cell, as well as use the supercomputing facilities of the National Center for Computational Sciences in ORNL. The results explained the experimental observations, assigning them to the interplay of the charging between Li, O, C and D. The simulations also provided predictions for the chemical sputtering and reflection data by deuterium impact of Li-O-C-D surface.

We compare results of the system evolution at characteristic times with full DFT calculations. Though numerically too intensive to describe dynamics, the DFT framework provides very powerful tools in analyzing the fundamental chemical processes governing the reactivity of graphitic surfaces [3]. We present results on plane-waves DFT calculations on lithium adsorption first on a pristine graphite surface then on a damaged surface (single vacancy). These two models are then oxidized and the consequences on lithium are detailed. Hydrogen (deuterium) atoms are adsorbed on all these model surfaces, and compared with the DFTB calculations, providing additional validation to our results. Finally, we discuss challenges for parameterization and dynamics of lithium with refractory metals, in particular with molybdenum and tungsten.

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Abstract 12/07/10

"Modeling of plasma/lithium-surface interactions in NSTX: status and key issues"[1]

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We review the status and key issues for plasma/lithium-surface interaction analysis for the liquid lithium divertor (LLD), and proposed inner molybdenum divertor subject to D, C, and Li impingement, in the National Spherical Torus Experiment (NSTX). Modeling of NSTX is actually more complicated than future devices such as DEMO due to short pulses, transient heating, high surface/volume ratio, possibly different sheath geometry, and use of multiple surface materials (Li, C, Mo). We have analyzed LLD liquid lithium sputtering, evaporation, transport, and D/C/Li surface effects, for selected planned high heating power plasma conditions, and this work continues. We are studying erosion/redeposition of a Mo inner divertor including effects of D, Li, C, Mo impingement on material mixing and subsequent surface response. Various coupled codes are used for analysis. The REDEP/WBC impurity transport erosion/redeposition package is the overall integration tool, with sputter yield and velocity distributions from ITMC and TRIM-SP binary-collision, mixed-material codes, and inputs of background D plasma edge/SOL solutions from external data-calibrated plasma fluid codes. A sheath code is used to compute sheath potential and structure for the relatively low magnetic field (~0.5 T) and less-oblique incidence (~5°) NSTX geometry at the divertors. Results to date are generally favorable, showing non-runaway LLD lithium self-sputtering, and acceptable core plasma contamination, for the plasma regime studied. This is also shown for an inner Mo divertor, for an edge/SOL plasma consistent with high recycling (low D trapping), but analysis is needed on Mo compatibility with low-recycle conditions obtained due to outer-divertor D/Li trapping. We note that all conclusions are subject to uncertainties in material mixing and subsequent surface response. We outline code validation and R&D needs.

[1] Work supported by the US Department of Energy, Office of Fusion Energy.

Turbulent Transport in Lithium Doped Fusion Plasmas

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A detailed analysis of the effects of Lithium dopant on turbulent transport in a deuterium-electron fusion plasma has been carried out with the nonlinear electromagnetic gyrokinetic code GKW [1]. The plasma parameters for the analysis have been chosen at mid radius of an FTU ohmic plasma with Liquid Lithium Limiter (LLL). Peaked electron density profiles, absence of Lithium accumulation in the plasma core and improved energy confinement have been observed in this type of discharges [2]. This paper presents results on local stability analysis of electromagnetic modes with wavelengths ranging from multiple deuterium Larmor radii down to the electron Larmor radius. A lithium concentration as in the early phase of FTU discharges is found to have an impact on the linear growth rate and flux driven by long wavelength ITG - TEM modes. The linear particle fluxes associated to the unstable modes are found to be directed inward for electrons and deuterium and outward for the lithium species. The heat flux due to the electron channel is strongly reduced by the presence of the lithium dopant due to stabilization of the electron temperature driven modes.

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Kinetic simulation of Lithium transport and its effect on diverted NSTX plasma*

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The kinetic particle code XGC0 is used to study neoclassical lithium transport and its effect on the plasma in diverted NSTX geometry. In addition to the deuteron and electron main plasma particles, lithium and carbon impurity particles are simulated together on Lagrangian equation of motion with a mass-momentum-energy conserving Monte Carlo Coulomb collision operator. Main and impurity ion recycling coefficients are controlled separately. Radial electric field profile is calculated self-consistently with the radial transport of multiple plasma ion species. Assuming that the radial electric field shear is related to turbulence suppression and H-mode transition, the relation of Li concentration to H-mode power threshold is investigated. It is found that the radial profiles of various ion species self-organize together. C^{+6} ions tend to concentrate toward the plasma center, consistently with the Rutherford's picture, at the expense of the main deuteron ions and Li^{+3} ions. This provides radial shielding of Li^{+3} species penetration from the edge. Concentration of C^{+6} ions toward the plasma center leads to improved electron confinement. It is also found that the radial E_r shear strength is sensitive to carbon fraction in the pedestal area. Validation of the simulation against NSTX experimental results will also be presented, together with the possible effect of Li on plasma edge turbulence.

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+Work performed in collaboration with K.H. Kim, G. Park and S. Ku

Modeling of lithium dust injection and wall conditioning effects on edge plasmas with DUSTT/UEDGE code

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Recently, it has been demonstrated experimentally that lithium conditioning of divertor plates can improve the core plasma performance by reducing the plasma temperature gradients, increasing the energy confinement time and the fusion power density [1]. However, due to eventual saturation with hydrogen of lithium layer deposited on the plates continuous recovery of lithium is required to achieve sustained effect. To facilitate lithium layer recovery on large surface areas during plasma discharge, continuous lithium dust injection has been proposed and experimentally demonstrated on NSTX. However, as the lithium dust partially ablates in the edge plasmas, the dust transport and impact of lithium impurities ablated from the dust on the plasmas need to be considered simultaneously with the lithium wall conditioning effects.

In the present work, we numerically investigate the lithium dust transport and its impact on edge plasma performance due to both the reduced plasma recycling and the lithium impurities using the coupled DUSTT/UEDGE code. The code validation is performed using 3D reconstructed lithium dust trajectories measured during dust injection experiments on NSTX. The simulated parameters of the injected lithium dust resemble those in the experiments using the calibrated dust dropper. First, the properties of the low-recycling edge plasmas regimes (e.g. onset of sheath-limited plasma conditions) have been simulated without dust. The results show that impurities originated from divertor plates are well retained in the divertor volume. We also show that the inverse transition from low to high recycling regime is possible due to excessive evaporation of lithium at high surface temperatures (~900K). Then, the dust injection with different rates up to several of 10mg/s are modeled demonstrating that dust originated impurities can have significant effect on the edge plasma profiles, transport, and stability in the low-recycling regimes. In particular, it is shown that dust can lead to substantial increase of the radiation power losses and decrease of the radial pressure gradients and of the radial plasma fluxes in the edge. Significant reduction of the power load to the divertor plates is also demonstrated for the simulated dust injection rates.

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Recent progress of NSTX lithium research and opportunities for magnetic fusion research *

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Various lithium wall coating techniques have been experimentally explored in NSTX over the last few years. In 2010, a liquid lithium divertor plate system was tested with about 1300 g of lithium evaporated into the NSTX vacuum vessel.[1] The overall NSTX lithium evaporator results suggest attractive opportunities for future magnetic confinement research including H-mode power threshold reduction, the control of Edge Localized Modes (ELM), electron energy confinement improvements, impurity control during non-inductive plasma start-up, and improving radio frequency wave heating and current drive efficiencies. These lithium results on NSTX combined with the recent H-mode results on EAST[2] suggest that the lithium application could directly benefit ITER particularly in the early phase of its operation where access to the H-mode plasma with relatively modest heating power is required. Another intriguing result from NSTX is that the surprisingly low levels of lithium ions observed in the NSTX H-mode plasma core (well below 1% dilution) compared to higher Z impurities (such as carbon) even when the plasma is essentially surrounded by the lithium coated walls. This observation bodes well for lithium based plasma facing component and divertor applications. However, core accumulation of carbon impurity in lithium induced ELM-free plasma is an unresolved issue for NSTX. In order to utilize lithium in future experiments such as for ITER, we need to develop fundamental understanding of the on-going lithium induced processes. A kinetic particle code XGC0 has been utilized to begin study of neoclassical lithium transport and its effect on the divertor.[3] In addition to the kinetic particle effects, there are complex lithium plasma-material chemical processes going on which clearly require deeper understanding of lithium chemistry. A dramatic improvement of plasma operation with relatively low level lithium evaporation for example is not well understood. Fundamental lithium plasma-material chemistry work has started in laboratory test stands[4] and molecular level modeling[5]. Eventually, it is highly important to develop an integrated model which combines MHD, transport, boundary, and plasma-wall interactions including plasma chemistry for developing a predictive capability lithium performance in the future magnetic fusion experiments such as ITER.

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Study of Processes of Hydrogen Isotope Interaction with Lithium CPS

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The issues of degradation and destruction of the plasma-facing materials, tritium accumulation and plasma contamination can be solved by using the liquid metals with low Z number. Lithium is the best candidate as a plasma-facing material for the receiving divertor tiles and other intra-chamber devices.

Lithium advantages can be realized in the fusion reactors by using a so-called lithium capillary-porous system (CPS). It is a new material, where liquid lithium is sealed in matrix of porous solid material. Potentials of the capillary-porous liquid metal units as power divertor receivers are proposed to study at the tokamak KTM.

In order to justify the use of lithium divertor module the experiments were carried out to study sorption characteristics of lithium CPS against hydrogen isotopes.

The samples under study were lithium CPS based on net-matrix of stainless steel 12Cr18Ni10Ti of 100 μm thickness; cell was about 100 μm. Temperature range was similar to the working temperatures of lithium divertor module and was 200 – 250^{o}C.

The gas absorption technique was used to study hydrogen/helium isotope interaction with the samples of lithium CPS. During the experiments there were determined the constants of sorption rates of hydrogen isotopes from the lithium CPS in the range of the working temperatures of lithium divertor and sorption capacity of lithium CPS against hydrogen isotopes and other gases typical for residual spectrum of high vacuum chambers.

Lithium / Molybdenum Infused Trenches (LiMIT): A heat removal concept for the NSTX inner divertor

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Abstract

With NSTX starting to design a new inner divertor system to be able to handle the power load of the next upgrade, replacement of the inner divertor "shelf" ATJ graphite tiles with molybdenum tiles has been suggested. In the past, CDX-U has shown that lithium can handle powers up to 60 MWm^{-2} without any adverse effects to operations [1], which is highly desirable since removal of 20 MWm^{-2} is needed in NSTX. A solution is Lithium/Molybdenum Infused Trenches (LiMIT) whereby the lithium creates a *self-driven* divertor plate structure utilizing the Thermo-Electric MHD force.

The basic concept of how a TEMHD-driven [2] Li divertor could work to remove significant heat fluxes is fairly straight forward. The divertor heat stripe hits LiMIT causing the top of the region with the lithium flow to get hotter than the regions below it. This causes a thermo-electric current in the lithium to go in the vertical direction. Since the magnetic field is in the toroidal direction there will be a force on the lithium in the radial direction along the trenches. Even without the cooling channels, the temperature gradient will still be established during the shot due to thermal inertia. What makes this approach attractive is that it is self limiting. The hotter it gets, the faster the lithium flows, which in turn cools the structure. With no heat flux the lithium is stationary.

In NSTX-upgrade a peak-power load of 20 MW/m^2 is expected on the divertor strike points. To see if a flowing surface can remove that much power with a lower temperature rise, we can use the flowing equation [3]:

$$Q=kLw\Delta T_{\perp}h+\rho C_P h w \Delta T_{\parallel} u$$

Here the first term is the heat conduction term and the second is the heat convection which transfers the heat along the trench. $\Delta T_{\perp}$ is the temperature difference across the depth of Li and $\Delta T_{\parallel}$ is the temperature difference between the inlet and outlet of the Li trench. The maximum flow velocity in the lithium that the TEMHD force can provide is determined by the thermoelectric power which is a function of the Seebeck Coefficient [4, 5].

In this talk we will present the first experiments and measurements of such flowing lithium in an infused trench at the University of Illinois in the SLiDE device and how it would be used in NSTX-Upgrade.

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"Design guidance for the flowing lithium systems in tokamaks"

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The flowing liquid lithium (LiLi) system in necessary for long lasting plasma discharges for pumping the plasma particles. At the same time it may address the issues of lithium surface contamination by impurities from the tokamak wall.

In designing the system, it is necessary to take into account that the LiLi flow in tokamak is significantly affected by the plasma and toroidal magnetic field. Also, the currents from the scrape off layer affect the flow.

In the talk we give the assessment of different effects affecting the flow: its direction (poloidal or toroidal), gravity, viscosity, Hartmann effect due to side walls, centrifugal force, thermo-electric currents and Marangoni effect, disruptions and external currents, etc.

Although different sophisticated mechanisms could be proposed for driving the LiLi flow, the simplest poloidal flow of 0.1 mm thick LiLi under gravity satisfies all known requirements and seems to be the most attractive.

Pacing Small ELMs at High Frequency using Spherical Lithium Granules and a Dropper / Impeller Injection Technology

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The deleterious effects of large ELMs on the future performance of ITER H-Mode discharges are known to be a serious cause for concern. Among the technologies being developed for ELM mitigation on ITER are pacing by repetitive injection of deuterium pellets.

This work explores the alternative possibility of pacing ELMs by high-frequency (~100 Hz) repetitive injection of small (~1 mm) spherical lithium granules at modest velocities (~100 m/s) using a dropper / impeller technology. In this concept, lithium ablated at the plasma edge would automatically coat PFCs in real-time and thus aid in the suppression of large low-frequency ELMs. The small high-frequency "quasi-grassy" induced ELMs would continuously purge the plasma core of impurities while not threatening PFC integrity.

If this concept can be shown to be viable it has the added benefit of replacing large, expensive and complicated technology of repetitive cryogenic pellet injectors with the compact and inexpensive dropper / impeller technology described in this work.

Lithium granules may also offer other advantages as compared to deuterium pellets as an agent for inducing ELMs. (1) Because it has a higher melting point and larger heat capacity than deuterium, ablation model calculations suggest that lithium requires lower injection velocities to penetrate into an H-Mode pedestal – a necessary condition to trigger ELMs. (2) Once in the pedestal, lithium can deposit three times as many electrons as deuterium and so offers the possibility of inducing ELMs using smaller volume granules/pellets.

The dropper / impeller technology discussed in this work also allows for the possibility of injecting other candidate "pellet" materials such as LiD or even Beryllium.

Electrostatic Lithium Injector (ELI)

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Core impurity accumulation is one of the most important problems in NSTX relating to confinement, and to plasma material interactions. Lithium is a powerful getter of impurities and by injecting lithium into regions where it is needed most, one may getter these impurities. The current LITER system in NSTX is inefficient in controlling where the lithium goes so having the ability to control where lithium can be injected is of great value. Through the use of a well known concept called electro spray, charged lithium particles can be produced that could then be controlled by electric fields to target areas in most need of getting impurities. While several variants of droplet electro spray techniques are available, no research has been done to produce an electro spray with lithium droplets.

Preliminary measurements have shown a system capable to produce lithium droplets. The concept as well as the initial demonstration of lithium coming out of the orifice is shown in Fig.1. Some of the experiments at the University of Illinois to test the design and operation of ELI include: Quantifying the number of charged droplets, determining the spread or angular distribution of ELI, determine the power requirements needed to produce charged lithium particles, determine the number of particles reaching the biased substrate as opposed to other substrates at different potentials, test and optimize different geometries (nozzle/electrode) for an efficient lithium droplet production.

This talk will present the first results of the ELI system being developed at the University of Illinois.

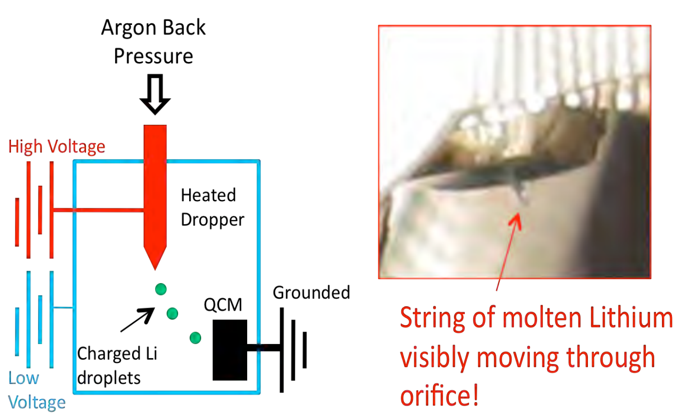

Figure 1: Left) Electrostatic Lithium Injector concept, and Right) Demonstration of lithium coming out of the heated injector

Draft Mission and Specifications for an Integrated PMI-PFC Test Stand

R Goldston, S Zweben, C Gentile, M Jaworski, H Ji, R Kaita, H Kugel, Y Raitses,
C Skinner, D Stotler, JR Wilson, I Zatz

We are examining options for a new integrated PMI-PFC test stand to help address the key issues of plasma-material interaction (PMI) physics and plasma-facing component (PFC) technology for magnetic fusion. As has been pointed out most recently in the 2009 Magnetic Fusion Research Needs Workshop Report, these issues are crucial for the development of magnetic fusion energy. They also have relevance for inertial fusion and other fields of science and technology.

The goal of the new integrated PMI-PFC test stand is to qualify *both* the plasma-material interactions and the technical operation of new plasma-facing components in as realistic an environment as practicable. This double (PMI + PFC) mission is particularly necessary in the case of actively cooled, flowing-liquid-metal plasma-facing surfaces, where magnetic fields and temperature gradients, as well as plasma impingement, both steady and intermittent, independently and interactively affect the flow of the liquid. At the same time the physics of deuterium absorption, and of liquid metal sputtering, evaporation, ionization, radiation and re-condensation needs to be understood, in order to predict effects on the bulk plasma and to predict the total flow pattern of the metal, including evaporated material, in a tokamak. Practical testing of an actively cooled, flowing-liquid-metal divertor surface will require a high magnetic field, a high heat flux, and a realistic plasma flux, since each of these can affect the flow of the liquid and its surface condition.

A staged approach appears promising, in which the capabilities of the PMI-PFC test stand are initially focused on providing support for near-term implementation of advanced PMI-PFC solutions in tokamaks, with emphasis on the upgraded NSTX. This effort would constitute a major step toward the solution of the longer-term needs for more powerful, longer-pulse devices culminating in Demo. The longer-term mission of supporting the implementation of PMI-PFC solutions in these devices would be addressed by staged upgrades to the proposed new integrated PMI-PFC test stand.

The capabilities to develop and exploit a facility of this character are available only through a national collaboration of scientists and engineers from national laboratories, universities and industry, and we propose that this new integrated PMI-PFC test stand be developed as a national collaboration. Since it is in its initial stage of conception, the concepts under consideration are presented here to facilitate wide discussion.

Concept Development and Engineering Considerations of a Steady-State Lithium-Coated Divertor

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The next step in the development of liquid metal divertor technology needs to be a steady-state solution incorporating both active cooling and also flowing lithium to continuously remove absorbed hydrogenic species and impurities. Presented here is a candidate design of a next-generation divertor with a self-healing lithium-filled mesh as plasma-facing surface. The mesh is open at the bottom to channels of continuously pumped lithium, and so lithium lost to evaporation is refilled via capillary action. These lithium channels share the region behind the plasma facing surface with a heat exchanger employing the primary PFC coolant. The performance of multiple coolants will be compared, and thermal and structural analyses of the design are presented along with expected temperatures, stresses and coolant and lithium flow rates.

"^{6}Li – An Enabling Material for Fusion"

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The Y-12 National Security Complex holds the US National supply of the ^{6}lithium isotope (Li6). Li6 is essential in the production of tritium for fusion fuel and breeder technology. Y-12 has more than 50 years experience in handling industrial quantities of Li6 and produces various Li6 compound for distribution to more than 35 foreign countries. Y-12 has inert processing capabilities and understands the kinetics of the reaction of many Li6 compound with moisture. Y-12 also understands material compatibility issues with Li6 compounds at room and elevated temperatures. Y-12 can meet challenging chemical purity requirements as well as demanding machining tolerances.

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Making turn toward fusion development

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It became evident from TFTR and JET tritium experiments in the 1990s as well as from the present ITER development that the conventional plasma regime is not suitable for the fusion energy source. The root problem is related to the anomalous electron energy losses, which drives the size of the machine, the magnetic field, the plasma current, the heating power together with funding up, while escalating the technology problems.

In fact, the alternative regime has been proposed based on utilization of pumping the plasma particles by a liquid lithium surface, while fueling is provided by the neutral beam injection (with the energy consistent with plasma temperature). This LiWall Fusion (LiWF) regime has completely different confinement when the energy is lost exclusively due to particle diffusion. In facts, this is a best possible confinement regime, suitable for fusion as the energy source.

In the talk, the step toward the LiWF regime is proposed for NSTX by using a lithium covered plate in the inner lower divertor. This approach is compared with the current plans of using Mo tiles and evaporators.