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Characterization of fueling NSTX H-mode plasmas diverted to a liquid lithium divertor

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ABSTRACT

Deuterium fueling experiments were conducted with the NSTX Liquid Lithium Divertor (LLD). Lithium evaporation recoated the LLD surface to approximate flowing liquid Li to sustain D retention. In the first experiment with the diverted outer strike point on the LLD, the difference between the applied D gas input and the plasma D content reached very high values without disrupting the plasma, as would normally occur in the absence of Li pumping, and there was also little change in plasma D content. In the second experiment, constant fueling was applied, as the LLD temperature was varied to change the surface from solid to liquid. The D retention was relatively constant, and about the same as that for solid Li coatings on graphite, or twice that achieved without Li PFC coatings. Contamination of the LLD surface was also possible due to compound formation and erosion and redeposition from carbon PFCs.

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1. Introduction

Reduction in edge recycling in spherical torus divertor plasmas has been observed with solid lithium PFCs [1]. This occurs through the formation of lithium compounds or complexes that bind deuterium [2]. Solid lithium coatings eventually saturate, however, and thus provide only short-pulse capability [3]. Flowing liquid lithium contained in a suitable medium has a much higher capacity for absorbing deuterium, and in addition has the potential to provide self-healing walls for high power reactors. Improvements in plasma performance in the National Spherical Torus Experiment (NSTX), using solid lithium coatings on a graphite divertor surface [3], motivated the installation of a fully-toroidal Liquid Lithium Divertor (LLD) [4–6].

In 2010, NSTX experiments were conducted using the LLD, which was installed on the large major radius side of the lower divertor.

Fig. 1 shows the LLD as it appeared prior to the start of plasma operations. The purpose of the LLD was to test the effectiveness of maintaining the D retention properties of a continuously replenished liquid lithium surface. For the LLD, this was effected by lithium evaporation onto its surface between discharges instead of flowing liquid lithium.

The LLD consisted of four 22 cm wide plates, each having a toroidal extent of 82.5°. The quadrants are separated toroidally by graphite tiles, which contained diagnostics and electrodes for edge plasma biasing experiments [7]. The LLD plates consisted of 2.2 cm thick copper with a 0.25 mm stainless steel liner vacuum brazed to its surface. This stainless steel surface was coated with a 0.165 mm thick layer of plasma-sprayed molybdenum, with a porosity of 45%. The stainless steel provided a barrier between the lithium in the porous molybdenum plasma-facing surface and the copper baseplate which would otherwise be eroded by the lithium. The porosity of the molybdenum facilitated the wetting and subsequent spreading of liquid lithium over the LLD. It also insured that surface tension forces are large relative to electromagnetic forces to retain the liquid lithium during plasma operations [8].

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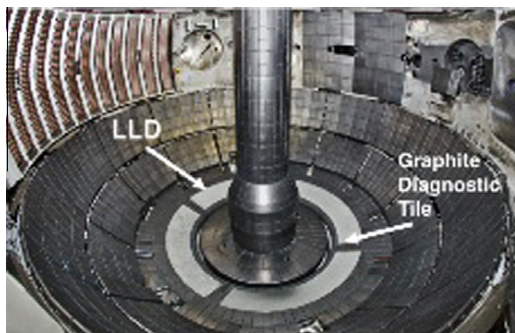


Fig. 1. Photo of the interior of NSTX before the start of the 2010 experimental campaign. The Liquid Lithium Divertor (LLD) plates form the light-colored toroidal ring in the lower divertor region. One of four inter-quadrant graphite tiles containing diagnostic sensors is also indicated.

The lithium capacity of the porous LLD surface was 37 g. The 2010 experiments were performed with the LLD filled with increasing amounts of lithium, from a few percent to more than 160% of capacity, over the duration of the campaign. Each of the four LLD plates contained embedded electrical resistive heaters and thermocouples to monitor the heating. Two lithium evaporators in the upper dome of the NSTX vacuum vessel, used in previous experiments to coat the graphite PFCs [3], were used to evaporate lithium over the entire LLD surface. Lithium was deposited, over 10-min intervals, in selected amounts varied between 100 and 700 mg in about 90% of NSTX discharges. In addition, special experiments were performed in which 7–20 g of lithium was evaporated on to the LLD prior to a discharge sequence.

2. Experimental results

During the 2010 experimental campaign, NSTX operated over 19.6 weeks and performed 3780 discharges. During this campaign, lithium was evaporated over the entire lower divertor region (graphite tiles and LLD) for experiments employing plasma shapes ranging from high-elongation discharges with strike points inboard of the LLD to low-elongation discharges with their outer strike points directly on the LLD (Fig. 2). More than 50 experiment types were performed involving diverted, neutral beam injection (NBI) and radio frequency (RF) heated plasmas. Most of these experiments involved NBI heated, high-elongation plasmas with

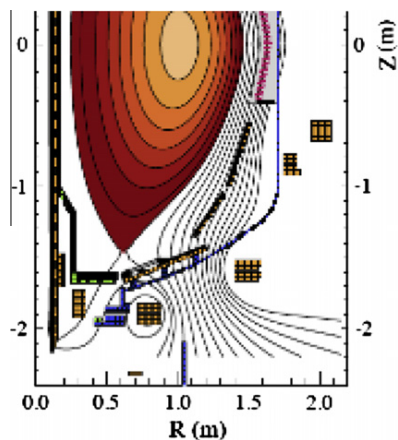


Fig. 2. Equilibrium reconstruction for plasma shape used in LLD experiments. The low-elongation discharges have the outer strike point incident on the LLD and the inner strike point incident on lithiated graphite inboard of the LLD.

the diverted strike points incident on the inner graphite divertor inboard of the LLD. There were also more than 110 H-mode discharges with outer strike-points directly incident on the LLD. The discharge pulse lengths were 1–1.5 s. H-mode characteristics for discharges on the LLD are given elsewhere [8].

Two methods were used to characterize the physics effects of the LLD, and compared with previous measurements [3] on graphite PFCs. The first involved measuring the effects of the different lithium related edge response on the discharge characteristics, e.g., plasma edge temperature and density, plasma edge MHD stability (ELM-free H-modes), volume averaged density, stored energy, and confinement time, etc. The second focused on determining the amount of deuterium fuel gas needed to maintain similar reference discharges on graphite and the LLD, as lithium deposition was increased.

During the 2010 NSTX campaign, many different experiments were performed with the diverted outer strike point on the LLD as shown in Fig. 2. In this paper, we report on two experiments, in particular, that characterize the fueling of NSTX H-mode plasmas diverted to the LLD. The first experiment (Fig. 3), was conducted to investigate the effect of plasma heating of the surface when a series of discharges with 4 MW of NBI heating was run with the outer strike point centered on the LLD. At this stage of the experimental campaign, the porous LLD surface was estimated to be about 20% filled.

Lithium continued to be evaporated between the discharges. During this discharge sequence, the lithium surface temperature was raised from below melting, through the melting point, and up to surface temperatures of 300 °C. Fig. 3 shows the total number of deuterons added as gas and the resultant D and C⁶⁺ core particle content at 0.5 s in each discharge. Also shown is the LLD surface temperature at 0.5 s, which increased above the lithium melting point during the series. The difference between the deuterium gas input and the plasma deuterium content reached very high values in this experiment without disrupting the plasma, as would normally occur in the absence of lithium pumping, but there was little change the plasma deuterium content. This suggests that the added deuterium was not required to maintain stable operating. Rather, the deuterium became ionized in the scrape-off layer and flowed to the divertor, and was absorbed by the liquid lithium rather than recycling and eventually increasing the plasma density.

The second experiment was performed at the end of the 2010 experimental campaign, as the accumulated lithium on the LLD surface exceeded more than 160% of its surface porosity capacity. A discharge sequence was performed under a constant fueling scenario, i.e., the gas programming was maintained unchanged as the LLD temperature was varied. The LLD average temperatures at the start of these discharges transitioned through the range 57–102 °C. The peak LLD surface temperatures were measured during the discharge with a 2-color IR camera [10]. The measured LLD surface temperatures in the interval 500–600 ms reached 160–300 °C so the plasma at the outer strike point was incident on what was expected to be liquid lithium in many discharges.

Fig. 4 shows that as the LLD surface temperature transitioned through the lithium melting point, the core plasma electron density and deuterium particle content remained relatively constant. This indicated that deuterium absorption at the solid and liquid lithium temperatures was the same, and comparable to what was reported previously for fresh lithium coatings on graphite [3]. During this sequence, however, the plasma C⁶⁺ content decreased as the LLD transitioned through the lithium melting point (Fig. 5). Additional work is needed to understand the origin of this behavior, but it was coincident with increasing ELMs. Performing the reverse experiment by proceeding from liquid to solid may facilitate an understanding of this behavior.

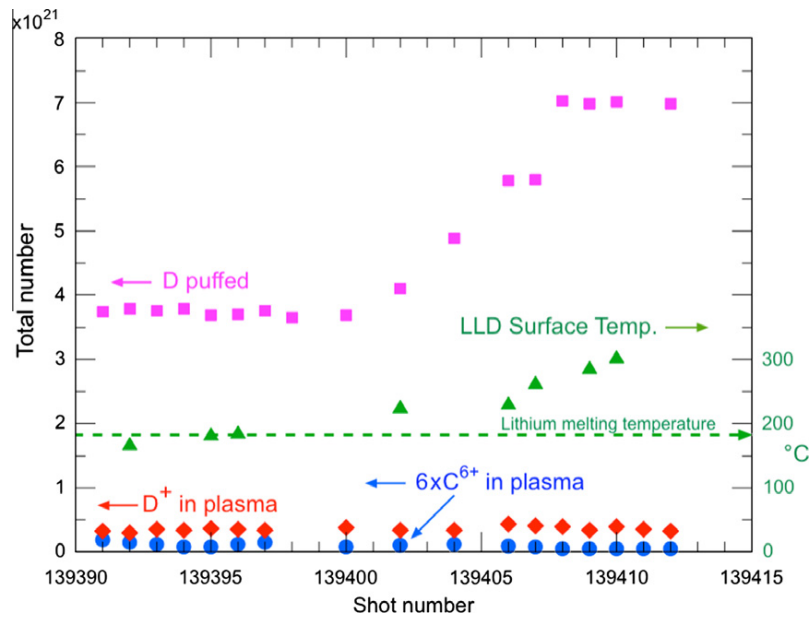


Fig. 3. Fueling scan data at 0.5 s for discharges with outer strike point on LLD. Shown is total number of deuterium particles puffed (fueling), resultant deuterium plasma particle content, and C^{6+} core particle content as a function of LLD surface temperature.

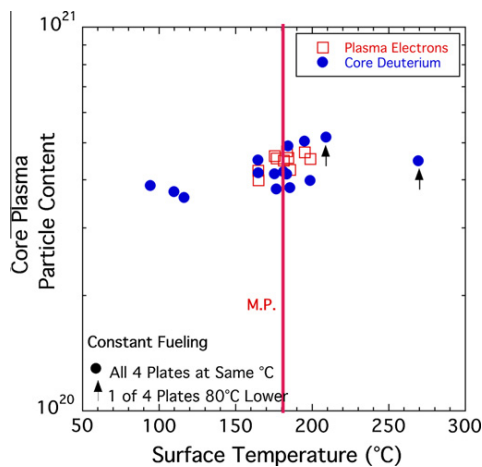


Fig. 4. Core D particle content determined with Charge Exchange Recombination Spectroscopy (CHERS) and the volume average plasma electron content measured using Multi-Point Thompson Scattering (MPTS) as a function of LLD surface temperature. The heaters on one of the four plates malfunctioned and that plate was 80 °C lower in temperature than the temperatures of the other three plates as indicated on the abscissa.

In these fueling experiments, additional features reminiscent of plasma behavior for active lithiated graphite [3] included increase in confinement [8], ELM suppression [8], and volume-average plasma lithium concentrations of less than 0.1% [9]. In addition, two-color measurements with an IR camera showed no hot spots, indicating that the thermal response of the LLD during discharges was determined by the effective heat distribution by the copper substrate [10].

There was no spectroscopic indication of lithium ejection from the plasma [11]. The absence of lithium ejection is also supported by a post campaign inspection of the LLD, which found no indication that either the molybdenum PFC or stainless steel liner for the LLD was exposed to the plasma. There was little spectroscopic evidence for radiation from metallic impurities during normal opera-

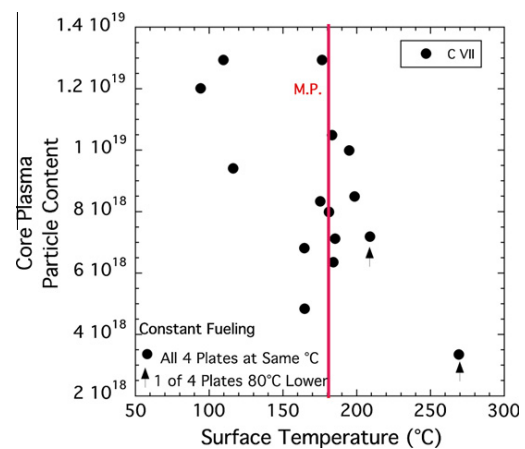


Fig. 5. Plasma C^{6+} content as function of LLD surface temperature. The carbon level decreased as the LLD transitioned from solid to liquid lithium temperatures. The heaters on one of the four plates malfunctioned and that plate was 80 °C lower in temperature than the temperatures of the other three plates as indicated on the abscissa.

tions, and examination of the plasma-sprayed molybdenum or the underlying liner after the run campaign did not reveal any macroscopic damage. However, there were spectroscopic indications of increasing deuterium desorption during discharges with the diverted outer strike point on the LLD lithium surface [6]. Evidence for carbon and oxygen impurity contamination was also observed [12].

3. Discussion

The spectroscopic indications of increasing deuterium desorption during discharges with the diverted outer strike point on the LLD lithium surface [6], together with increasing carbon and oxygen impurity contamination [12], suggests that while the LLD was able to retain bulk lithium during high-power divertor discharges, the surface behavior of its static lithium coatings was

slowly modified by incident plasma flux and interactions with the vessel vacuum impurities.

The similarity of results obtained with the LLD temperature controlled through embedded heaters and solid lithiated graphite may be due in each case, to the behavior of the PFC being determined by the purity of the lithium coating on a chemically-inert substrate. Although the temperature of the LLD exceeded the melting point of lithium, it never rose about the temperature of the lithium compound with the lowest melting point, i.e., lithium hydroxide at 462 °C. A lithium compound layer can thus be present on the LLD surface, and this possibility has been corroborated with laboratory tests where an LLD sample was exposed to a high-power neutral beam [13]. Power densities approaching levels found in NSTX were reached, but the lithium compound layer persisted on the surface facing the beam.

The arguments for the existence and persistence of a lithium compound layer when embedded heaters were used also apply for LLD heating with plasmas, where the maximum surface temperature was in the vicinity of 300 °C. To estimate the range of incident D particles in the lithium compound layer, data from a 99-probe High-Density Langmuir Probe (HDLP) array were used. The HDLP array was installed between the LLD plates (Fig. 1). Assuming that the ion temperature was equal to the electron temperature, an incident D energy of about 50 eV was deduced at the outer strike point. This result is consistent with midplane Charge Exchange Recombination Spectroscopy (CHERS) measurements.

Fig. 6 shows the results of a calculation with the TRIM code [14]. They indicate a stopping range of about 5 nm in lithium for NSTX 50 eV deuterium ions, and a stopping range of about 3 nm in typical lithium compounds formed with common residual gases in the vacuum environment. V_p is the typical floating potential of a Langmuir probe at the divertor strike point, and T_i the typical ion temperature in the plasma scrape-off layer (SOL). In addition to assuming that the ion temperature is equal to the electron temperature, the incident ion energy is taken as the sum of the measured plasma potential (V_p) and an assumed ion temperature contribution to the incident energy of twice the ion temperature (T_i). Also indicated is a deuterium ion energy estimate based on Kocan et al. [15], who report that T_i in various fusion devices could range from 1 to 10 times the electron temperature in the SOL.

The TRIM results also indicate that even if the ion temperature was 10 times greater than the electron temperature [16], the stopping range is only about 6 nm. Impurity layers of this thickness were easily accrued during the 2010 experimental campaign

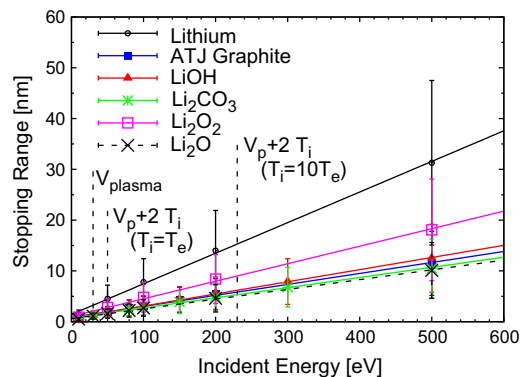


Fig. 6. Results from TRIM code for various PFC materials and lithium compounds. The calculations indicate a stopping range of about 5 nm in lithium for 50 eV deuterium ions, and a stopping range of about 3 nm in typical lithium compounds formed with common residual gases in the vacuum environment. V_p is the typical floating potential of a Langmuir probe at the divertor strike point and T_i the typical ion temperature in the plasma scrape-off region.

[17]. Static liquid lithium on the LLD getters the NSTX residual vacuum impurity gases H_2O , CO, and CO_2 . X-ray photoelectron spectroscopy (XPS) measurements at 5×10^{-8} Torr find that O_2 and H_2O at 10^{-6} Torr could oxidize 20 monolayers of lithium (5 nm) in 20 s. Under both LLD heating scenarios, this would result in a continuous codeposition of such species during the evaporation of lithium between shots.

Surface deposition is also expected to occur as plasmas interact with PFCs, including redeposition of eroded graphite PFCs in NSTX that would also be codeposited on the lithium coatings. The dissimilarities in the D efflux during the two LLD heating experiments could lead to differences in the LLD surfaces during discharges, and hence affect how the D is retained. This is very much an open question, however, and laboratory work is in progress measure the relative retentions of D in lithium impurity complexes under a variety of conditions that simulate the tokamak environment [17].

4. Conclusions

The NSTX LLD was implemented to provide a fully-toroidal lithium PFC for high power divertor plasmas. The LLD was loaded with lithium evaporators, and its surface was recoated with lithium between shots. With the diverted outer strike point on the LLD, its molybdenum plasma-sprayed surface retained the lithium, and the thin stainless steel liner under the molybdenum was an effective lithium barrier for the copper heat sink.

Plasmas were studied with the outer strike point on the LLD. As the applied gas input increased, the plasma D content remained relatively unchanged. This result is consistent with the absorption of the deuterium in the lithium, rather than recycling and eventually increasing the plasma density. Experiments were also performed at constant gas fueling, but with increasing LLD temperature. The D retention relatively constant as the temperature varied from below to above the lithium melting point. The results were comparable to what was observed for solid Li coatings on graphite, or approximately twice that obtained with uncoated graphite.

These observations suggest that the LLD behavior is determined by a lithium coating evaporated on an inert layer formed by lithium compounds on the LLD surface. The persistence of such a layer is possible because the LLD temperature never exceeded the melting point of any lithium compound. Modeling suggests that the layers formed by typical lithium compounds will not be penetrated by D ions at energies expected in the SOL, which supports the conclusion that the D retention is due to a solid coating from Li evaporation onto the LLD surface.

On the other hand, earlier studies have indicated that D retention efficiencies in liquid and solid lithium may be comparable for clean surfaces, but are lower for lithium with surface contamination [18]. This suggests that the LLD experiments may not have been able to distinguish between the behavior of a liquid lithium surface and a solid lithium coating evaporated onto the lithium compound layer that forms on static liquid lithium. A flowing lithium LLD is thus needed to demonstrate more clearly the efficacy of a lithium PFC specifically in its liquid state. This approach is required to avoid the issues introduced by complications such as lithium compound layer formation and lithium loading through evaporation. Work has begun at PPPL to address the technological challenges that must be overcome for the implementation of a flowing LLD on the NSTX-Upgrade. The experience in developing a static lithium LLD has been a key element in this process.

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References

- [1] H.W. Kugel et al., *Phys. Plasmas* 15 (2008) 056118.
- [2] C.N. Taylor et al., *J. Appl. Phys.* 109 (2011) 053306.
- [3] H.W. Kugel et al., *J. Nucl. Mater.* 390–391 (2009) 1000.
- [4] H.W. Kugel et al., *Fusion Eng. Des.* 84 (2009) 1125.
- [5] R.E. Nygren et al., *Fusion Eng. Des.* 84 (2009) 1438.
- [6] H.W. Kugel et al., *Fusion Eng. Des.* 84 (2012) 1724.
- [7] S. Zweben et al., *J. Nucl. Mater.* 390–391 (2009) 417.
- [8] M. Jaworski et al., in: *Proceedings of the 24th IAEA Fusion Energy Conference*, San Diego, CA, USA, October 8–11, 2012.
- [9] M. Podesta et al., *Nucl. Fusion* 52 (2012) 033008.
- [10] T. Gray, et al., in: *Proceedings of the 24th IAEA Fusion Energy Conference*, San Diego, CA, USA, October 8–11, 2012.
- [11] V. Soukhanovskii et al., *Rev. Sci. Instrum.* 81 (2010) 10D723.
- [12] F. Scotti, et al., in: *Proceedings of the 24th IAEA Fusion Energy Conference*, San Diego, CA, USA, October 8–11, 2012.
- [13] T. Abrams et al., *J. Nucl. Mat.* (2013), <http://dx.doi.org/10.1016/j.jnucmat.2013.01.057>.
- [14] R. Bastaz, J.A. Whaley, *Fusion Eng. Des.* 72 (2004) 111.
- [15] M. Kocan et al., *J. Nucl. Mater.* 415 (2011) S1133.
- [16] J.F. Ziegler, J.M. Manoyan, *Nucl. Instrum. Methods B35* (1989) 215.
- [17] C.H. Skinner et al., *J. Nucl. Mat.* (2013), <http://dx.doi.org/10.1016/j.jnucmat.2013.01.136>.
- [18] S.K. Erents, G.M. McCracken, P. Goldsmith, *J. Phys. D: Appl. Phys.* 4 (1971) 672.