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Recent liquid lithium limiter experiments in CDX-U

R. Majeski[1], S. Jardin[1], R. Kaita[1], T. Gray[1], P. Marfuta[1], J. Spaleta[1], J. Timberlake[1], L. Zakharov[1], G. Antar[2], R. Doerner[2], S. Luckhardt[2], R. Seraydarian[2], V. Soukhanovskii[3], R. Maingi[4], M. Finkenthal[5], D. Stutman[5], D. Rodgers[6] and S. Angelini[7]

[1] Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA
[2] University of California at San Diego, San Diego, CA 92101, USA
[3] Lawrence Livermore National Laboratory, Livermore, CA 94550, USA
[4] Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA
[5] Johns Hopkins University, Baltimore, MD 21218, USA
[6] University of Delaware, Wilmington, DE, USA
[7] Columbia University, New York, NY 10027, USA

E-mail: rmajeski@pppl.gov

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Abstract
Recent experiments in the Current Drive Experiment-Upgrade (CDX-U) provide a first-ever test of large area liquid lithium surfaces as a tokamak first wall to gain engineering experience with a liquid metal first wall and to investigate whether very low recycling plasma regimes can be accessed with lithium walls. The CDX-U is a compact ($R = 34$ cm, $a = 22$ cm, $B_{\text{toroidal}} = 2$ kG, $I_P = 100$ kA, $T_e(0) \sim 100$ eV, $n_e(0) \sim 5 \times 10^{19}$ m^{-3}) spherical torus at the Princeton Plasma Physics Laboratory. A toroidal liquid lithium pool limiter with an area of 2000 cm^2 (half the total plasma limiting surface) has been installed in CDX-U. Tokamak discharges which used the liquid lithium pool limiter required a fourfold lower loop voltage to sustain the plasma current, and a factor of 5–8 increase in gas fuelling to achieve a comparable density, indicating that recycling is strongly reduced. Modelling of the discharges demonstrated that the lithium limited discharges are consistent with $Z_{\text{effective}} < 1.2$ (compared with 2.4 for the pre-lithium discharges), a broadened current channel and a 25% increase in the core electron temperature. Spectroscopic measurements indicate that edge oxygen and carbon radiation are strongly reduced.

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1. Introduction

Liquid lithium walls have been identified as a potential solution to many of the engineering problems associated with the first wall of a fusion reactor [1]. In addition, a nonrecycling liquid lithium boundary is predicted to allow access to fundamentally different tokamak equilibria [2]. Experiments in the Current Drive Experiment-Upgrade (CDX-U) have provided valuable insight into the practical engineering aspects of handling and stabilizing liquid lithium in a tokamak environment, as well as a confirmation that liquid lithium walls do indeed produce fundamental changes in a tokamak discharge.

The benefits of a surface that has low or no recycling conditions have been demonstrated during the 'Deposition of Lithium by Laser Outside of Plasma' (DOLLOP) lithium wall conditioning experiments [3], for example, in the Tokamak Fusion Test Reactor (TFTR). Since TFTR had carbon walls, intercalation of the lithium into the graphite is a complicating factor in those experiments. Lithium limiter experiments have also been performed on the T-11M device [4], where a capillary porous rail limiter system was used to form a 'self-restoring' liquid lithium surface [5]. The T-11M limiter is relatively small, and evaporated lithium wall coatings are thought to be a factor in the experiments [4]. In this paper, we focus on experiments in which a substantial fraction of the plasma-facing surface is liquid lithium.

CDX-U is a small spherical torus, with a major radius $R_0 = 34$ cm, minor radius $a = 22$ cm, aspect ratio $= 1.5$, elongation $\kappa = 1.6$, toroidal field $B_T = 2.1$ kG and ohmic current $I_p \leqslant 90$ kA. With the exception of the capacitor banks for the OH system and the field null formation coils, the power supplies are pre-programmed and controlled by digital to analogue waveform generators. At present, there is no feedback control on the plasma current; therefore, the applied loop voltage magnitude and time history are approximately the same for every discharge. For this reason, the plasma current

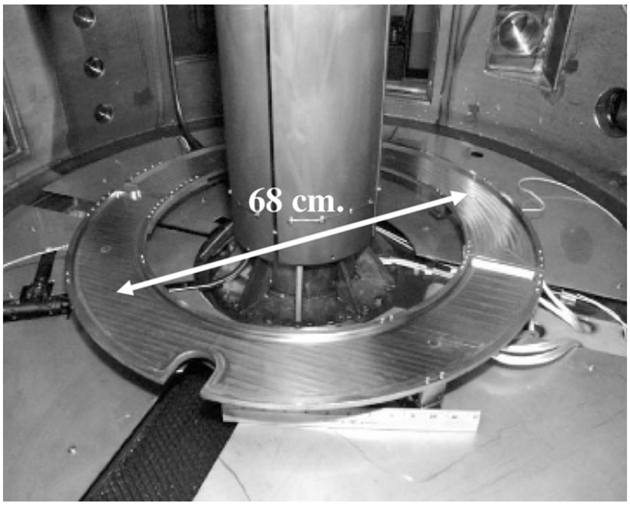

Figure 1. Interior of CDX-U showing the toroidal bottom tray limiter. Not visible are the heating elements, mounted on the bottom of the tray. The semicircular cutout in the tray at lower left permits interferometer access to central chords. Also visible are the heat shields installed to protect the lower vacuum vessel and centrestack, electrical connections to the heaters and tray halves (right), and tray thermocouples.

achieved is a good measure of plasma performance in CDX-U. Deuterium was the working gas for all experiments.

The first experiments with lithium limiters in CDX-U employed a small area rail limiter [6]. Following the rail limiter experiments, a shallow, heated, stainless steel tray was installed at the bottom of the CDX-U vacuum vessel. The tray has an inner radius of 24 cm, is 10 cm wide and 0.5 cm deep and exposes 2000 cm^2 of lithium pool to the plasma. It is constructed in two halves, with a single electrical break to prevent induction of large currents in the tray due to the ohmic transformer. The tray ends on either side of the electrical break are connected to vacuum electrical feedthroughs. One end of the tray is then externally connected to ground through a current transformer. The other end is not connected, which eliminates inductively driven tray currents due to ohmic transformer action. Currents drawn by the tray from the plasma either as a result of normal operations (limiter currents) or due to a disruption are therefore forced to run in the toroidal direction, parallel to the toroidal magnetic field. This construction is designed to eliminate the largest component of possible $\boldsymbol{J} \times \boldsymbol{B}$ forces on the liquid lithium. A photograph of the tray installed in CDX-U is shown in figure 1.

For the first experiments with the tray limiter, it was loaded under vacuum or dry argon with approximately 200 cm^3 of solid lithium in the form of rods, which were subsequently melted. This approach produced a partial (∼50% coverage), uneven layer of lithium in the tray. Oxide and hydroxide surface coatings on the lithium were visually evident and were only partially removed by glow discharge cleaning. Nevertheless, global improvements in impurity content and plasma performance were observed [7].

For the experiments described here, a new fill system was developed by the University of California at San Diego PISCES group. This system injects liquid lithium onto the pre-heated (500˚C) tray, under an atmosphere of argon, in order to obtain a uniform fill of the tray. Prior to the lithium fill, tokamak discharges were run for several months, using the empty stainless steel tray as a limiter. Afterwards, when sufficient baseline data had been obtained with a high recycling limiter, the tray was filled with approximately 500 cm^3 of liquid lithium. Subsequent cycles of reheating the tray, combined with 4–8 h cycles of argon glow discharge cleaning, produced 100% coverage of the tray. In this case, argon glow discharge cleaning at tray temperatures of 300˚C was effective in removing coatings of oxides and hydroxides which accumulate on the surface of the lithium at the normal base pressure of CDX-U ((1–2) × 10^{-7} Torr) during periods when the tokamak is not operating, producing a highly reflective metallic surface. Typically a 'lithium pool' discharge denotes one in which the tray temperature is maintained at 300˚C or above, well above the melting point of lithium (186˚C). It should also be noted that at normal operating temperatures the evaporation rate of the lithium is significant; this leads to lithium coatings on the windows (which is undesirable) as well as on the titanium carbide-coated, stainless steel, centrestack, which is a primary plasma limiter.

2. Plasma characteristics during lithium operations

A comparison of pre- and post-lithium discharges in deuterium is shown in figure 2. The most obvious differences in the two discharges are in the fuelling requirements and the loop voltage evolution. In the case of the discharge operated against the liquid lithium, a factor of 5 or more increase in fuelling is required. This corresponds to the maximum flow rate of the piezoelectric valve used to fuel CDX-U and is still not sufficient for attaining a plasma density comparable with the pre-lithium discharge. In the pre-lithium discharge, only a pre-fill is required to fuel the entire discharge. Recycling alone is sufficient to build and maintain density during the discharge. The density of the post-lithium discharges also pumps out promptly when gas puffing is terminated at 0.222 s, with an e-folding time of 1 ms, which is approximately the energy confinement time for a CDX-U discharge. A quantitative determination of the global recycling coefficient is not available since the fuelling efficiency and particle confinement time are not known experimentally for these discharges. However, the observed particle pumpout is strongly suggestive of a very low recycling coefficient. Since the lithium tray limiter itself represents less than 50% of the total surface area wetted by the plasma, this result suggests that evaporation of the lithium in the tray and continual coating of the centrestack surface with fresh lithium may play a significant role in the discharge modifications seen with lithium.

Figure 3 is a summary plot of the fuelling requirements, plotted as a function of the peak discharge plasma current, for pre- and post-lithium discharges in CDX-U. Note that although the fuelling of the lithium shots utilized the full gas throughput of the available valve (up to 60 Torr l s^{-1}) the maximum attainable density during lithium operations was approximately 75% of the pre-lithium discharges, which utilized only a deuterium pre-fill.

The differences in fuelling are expected from previous experiments which indicate that liquid lithium has very low recycling properties [8]. Another indication of very low recycling in the post-lithium discharges is the reduction in

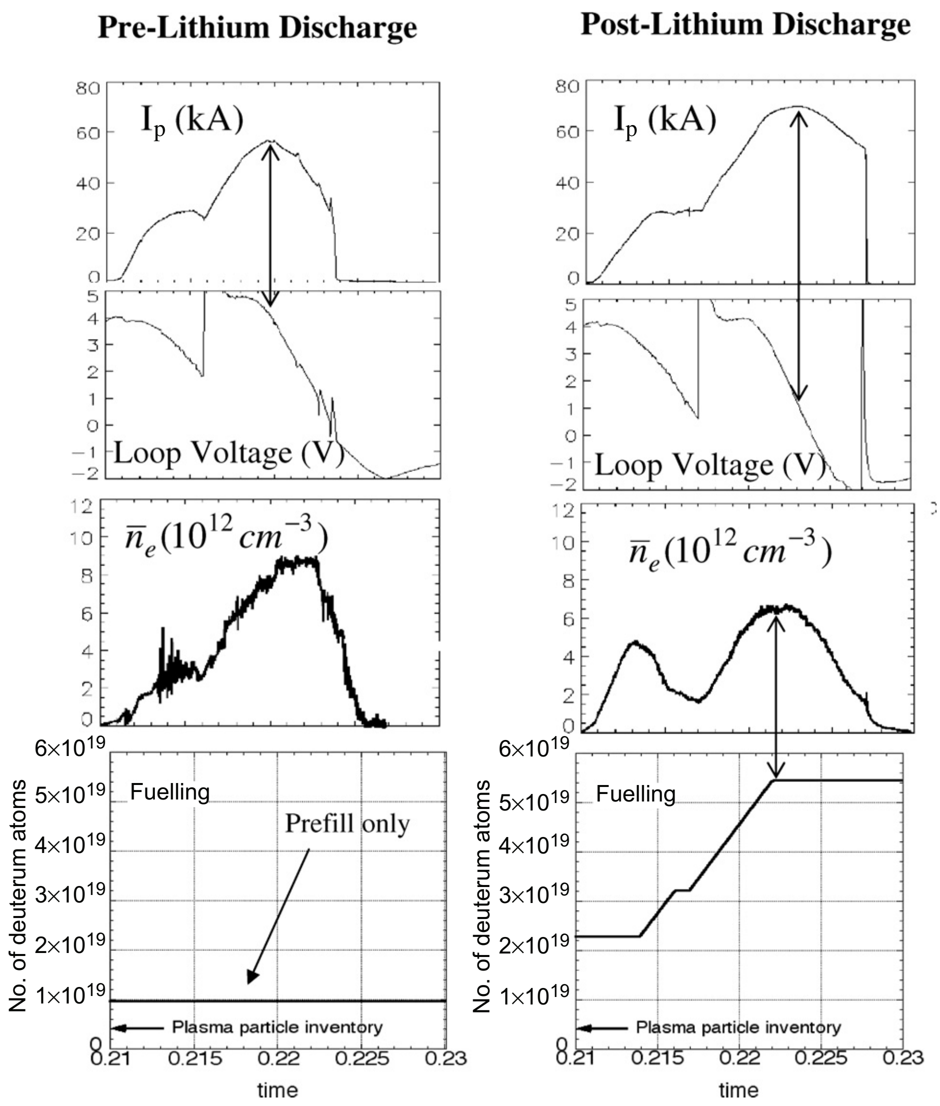

Figure 2. Comparison of plasma current, loop voltage, density and fuelling for a discharge limited by the toroidal stainless steel tray limiter, prior to filling with lithium, and for a discharge limited by liquid lithium. Note the large reduction in loop voltage required to sustain the plasma current, and the much higher fuelling rates required for the liquid lithium limiter plasmas.

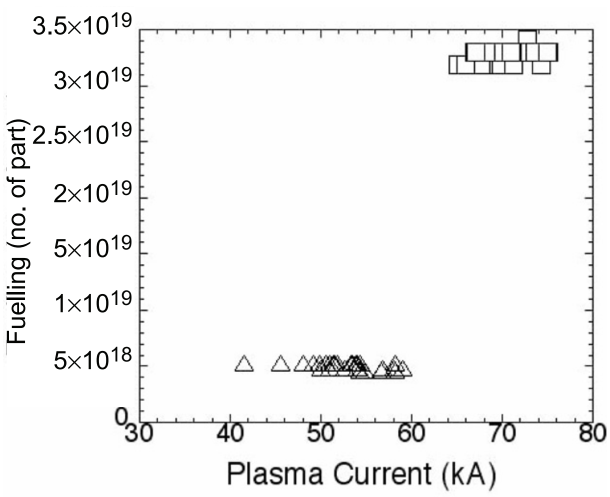

Figure 3. Summary plot of particle fuelling for discharges utilizing the bare stainless steel tray as a limiter (△) and for discharges limited by liquid lithium (□).

D_α emission from spectroscopy viewing the centrestack, which is a primary limiting surface for the discharge. A comparison of pre- and post-lithium D_α emission is shown in figure 4.

The loop voltage evolution is also strikingly different for pre- and post-lithium discharges. Far lower loop voltages are required to maintain the plasma current for the post-lithium discharges. Plasma termination does not occur until well after the loop voltage reverses. Whereas in a pre-lithium discharge 2 V is insufficient to sustain the plasma current, 0.5–0.8 V is sufficient to maintain the plasma current in a post-lithium discharge. During lithium operations it was determined that current ramps of 4 MA s^{-1} could be sustained with less than 1.5 V loop voltage. This represents exceptionally low resistive flux consumption for a small tokamak. A comparison of pre- and post-lithium discharge loop voltage and plasma current behaviour is shown in figure 5.

Impurities, especially oxygen, are reduced during lithium operations. Data taken with a residual gas analyser indicate that water levels in the chamber drop by an order of

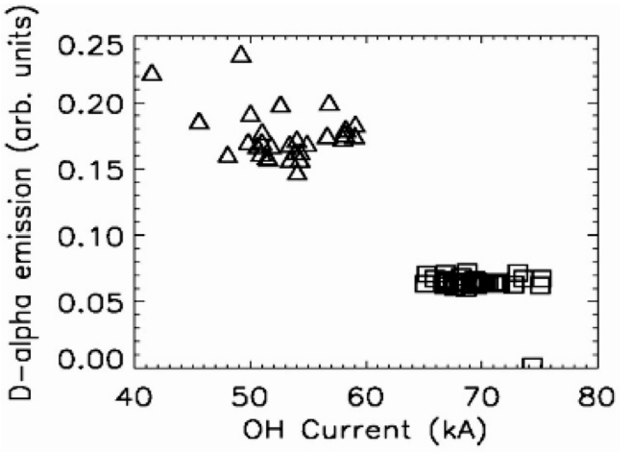

Figure 4. Comparison of edge D_α emission for pre-lithium (△) and post-lithium (□) discharges. The baseline evident in the lithium discharges is very consistent, and may be background D_α emission due to gas puffing.

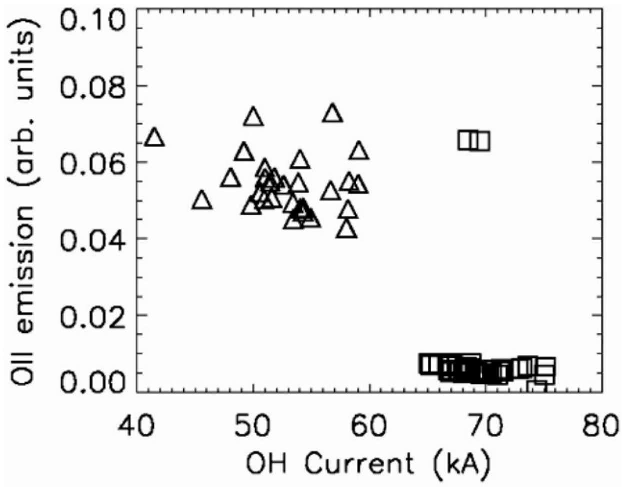

Figure 6. Oxygen II emission at the centrestack for plasmas limited by the stainless steel tray (△) and by liquid lithium (□).

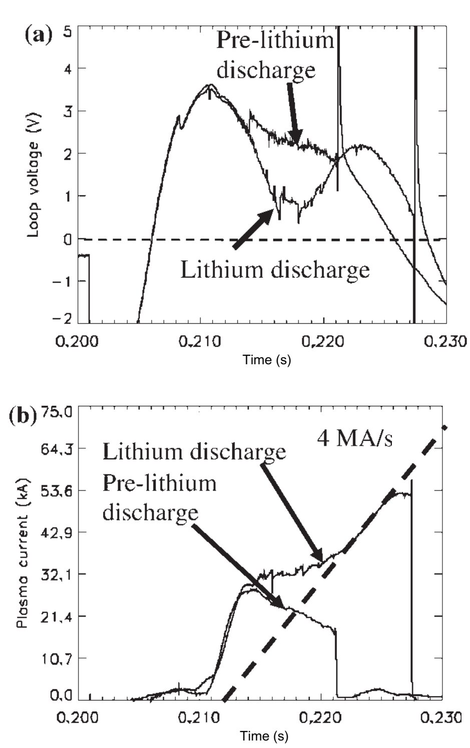

Figure 5. Loop voltage (*a*) and plasma current (*b*) comparison for pre- and post-lithium discharges. Note the zero in the loop voltage plot.

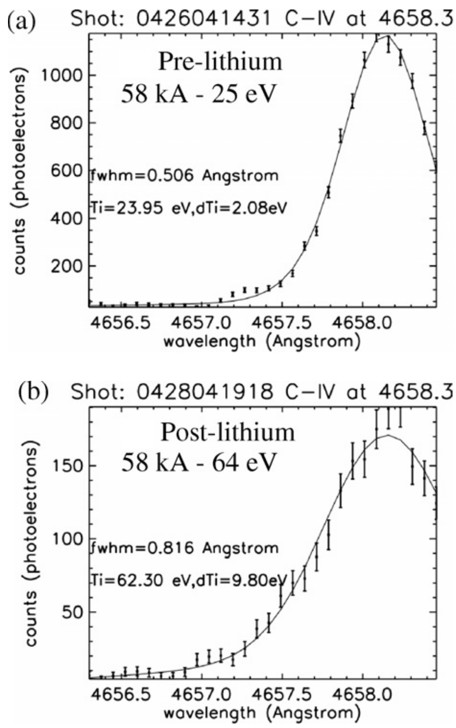

Figure 7. C IV line width measurements for (*a*) a pre-lithium discharge and (*b*) a post-lithium discharge. The peak plasma current for both discharges was 58 kA; the density of the lithium discharge was somewhat lower due to fuelling limitations.

magnitude during operations with the liquid lithium limiter. The resultant reduction in plasma oxygen radiation is shown in figure 6.

A direct measure of the core electron temperature was not available. However, a spectroscopic measurement of the Doppler broadened C IV line width indicates that the impurity ion temperature increases by over a factor of 2 for the lithium discharges. Spectroscopic measurements of the C IV line width for pre- and post-lithium discharges are shown in figure 7. Note also that the carbon line intensity drops by an approximate factor of 6 for the lithium discharges; the carbon impurity content of the discharge is also significantly reduced during lithium operation.

The liquid lithium contained in the toroidal tray was also observed to be mechanically stable during tokamak discharges. Significant currents—up to 500 A for 100 μs or 100 A for 10 ms—were drawn to the liquid lithium from the plasma as a result of vertical displacements or disruptions. The resultant current densities in the liquid lithium were in excess of 100 A cm^{-2}. However, there was no visual evidence of any lithium ejected from the tray after hundreds of discharges. Furthermore, fast (1000 frame s^{-1}) camera imaging of the liquid lithium surface detected no motion of the liquid surface during plasma operations. The stability of the lithium is likely due to the design of the tray, which forces all current conducted to ground to flow in the toroidal direction, parallel to the toroidal magnetic field, avoiding $\boldsymbol{J} \times \boldsymbol{B}$ forces on the lithium. Therefore, no splashing of lithium out of the tray occurred during plasma operations. The tray design, which represents a toroidal bottom limiter geometry, presents advantages compared with poloidal limiter concepts proposed, for example, for Tore Supra [9]. The design approach taken in CDX-U appears to have successfully inhibited undesirable displacement of the liquid lithium during plasma operations.

3. Modelling with the Tokamak Simulation Code

The evolution of the loop voltage and current for pre- and post-lithium discharges with similar plasma current and density values has been modelled with the Tokamak Simulation Code (TSC). Although the core electron temperature is not measured with Thomson scattering, soft x-ray measurements indicate that the peak electron temperature does not exceed ~150 eV for the lithium discharges; this is used as a constraint in the modelling. TSC indicates that the modelled internal inductance drops from 1.4 for the pre-lithium discharges to 0.65 for the post-lithium discharges. This drop in internal inductance is indicative of a significantly broadened current channel, in keeping with the analytic predictions for a very low recycling discharge [2]. Modelling also indicates that $Z_{\text{effective}}$ drops by a factor of 2 (from 2.4 to 1.16) for lithium operation, which is in qualitative agreement with the observed reduction in impurity radiation. Within the above-mentioned constraint on the electron temperature, TSC modelling also suggests that a modest increase in peak electron temperature, from 120 to 150 eV, occurs for the lithium discharges.

4. Summary and conclusions

The CDX-U experiments with a significant large area liquid lithium limiter have clearly demonstrated improvements, compared with former discharge results, in virtually every available measure of tokamak performance. The reduction in plasma resistivity as evidenced by the loop voltage characteristics is particularly remarkable for a small, ohmically driven tokamak. These improvements far exceed previously observed changes in CDX-U discharges which employed either boronization or titanium gettering. Note that neither of these surface conditioning techniques were utilized for any of the discharges described here. The effects of liquid lithium plasma-facing components will be further explored in the Lithium Tokamak Experiment (LTX), which is presently under construction at the Princeton Plasma Physics Laboratory.

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