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Spherical torus plasma interactions with large-area liquid lithium surfaces in CDX-U

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Abstract

The current drive experiment-upgrade (CDX-U) device at the Princeton Plasma Physics Laboratory (PPPL) is a spherical torus (ST) dedicated to the exploration of liquid lithium as a potential solution to reactor first-wall problems such as heat load and erosion, neutron damage and activation, and tritium inventory and breeding. Initial lithium limiter experiments were conducted with a toroidally-local liquid lithium rail limiter (L3) from the University of California at San Diego (UCSD). Spectroscopic measurements showed a clear reduction of impurities in plasmas with the L3, compared to discharges with a boron carbide limiter. The evidence for a reduction in recycling was less apparent, however. This may be attributable to the relatively small area in contact with the plasma, and the presence of high-recycling surfaces elsewhere in the vacuum chamber. This conclusion was tested in subsequent experiments with a fully toroidal lithium limiter that was installed above the floor of the vacuum vessel. The new limiter covered over ten times the area of the L3 facing the plasma. Experiments with the toroidal lithium limiter have recently begun. This paper describes the conditioning required to prepare the lithium surface for plasma operations, and effect of the toroidal liquid lithium limiter on discharge performance.

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1. Introduction

Reactor studies [1] have identified liquid walls as a potentially ‘revolutionary’ solution of generic magnetic fusion energy (MFE) first-wall problems,

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such as heat load and erosion limits of dry walls, neutron damage and activation, reliability/maintainability of first walls, and tritium inventory and breeding. To address the practicality of this concept, key liquid lithium-plasma interaction questions are being addressed in the current drive experiment-upgrade (CDX-U) device [2] at the Princeton Plasma Physics Laboratory (PPPL). Initial investigations were performed with a liquid lithium limiter (L3) probe supplied by the University of California at San Diego (UCSD). These experiments were followed by discharges utilizing a large area liquid lithium pool as the target.

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). Lithium limiter experiments have also been performed on the T-11M device [4], where a capillary porous system (CPS) was used to form a ‘self-restoring’ liquid lithium limiter surface [5]. However, introduction of large area lithium limiter targets and walls into existing tokamak facilities has not yet taken place. The primary goal of the CDX-U experiments will thus be to study auxiliary-heated discharge whose surface contact is primarily with a large-area liquid lithium limiters.

Surface conditioning was identified as a critical issue in both the TFTR and T-11M experiments. The DOLLOP process was a technique where the graphite plasma facing components in TFTR were coated with a lithium surface film. This was effective in improving plasma performance, however, only after the graphite surface was subject to extensive discharge conditioning prior to the deposition of the lithium. Particle recycling in T-11M was reportedly reduced only after plasma operations were conducted long enough to clean the CPS limiter surface. The experience on CDX-U was consistent with these observations, where long periods of (GDC) discharge cleaning preceded lithium experiments.

This paper provides a brief description of CDX-U (Section 2). This is followed by a discussion of experiments with the L3 in Section 3. Results with the toroidal lithium tray limiter are described in

Section 4, and the conclusions are summarized in Section 5. A clear change in the particle recycling was observed with liquid lithium in the tray limiter, and some of the highest plasma currents ever obtained in CDX-U were achieved.

2. Description of CDX-U and the L3 probe

The CDX-U facility (Fig. 1) is capable of spherical plasma operations in toroidal fields up to 2.3 kG with a ‘flattop’ of 100 ms. Power supplies for the vertical and shaping fields permit discharges with plasma current up to 150 kA for greater than 25 ms. Capacitor banks provide the Ohmic heating (OH) current. The achievable current ramp rate and pulse duration of about 20 ms with the present OH supply, however, limit the plasma current to about 90 kA. With the exception of the capacitor banks for the OH system and the field null formation coils, the power supplies are preprogrammed and controlled by digital to analog waveform generators. The basic discharge parameters summarized in Table 1.

The Ohmic heating system is capable of providing 0.2 MW to CDX-U, and the facility also has a radio frequency (RF) heating system[2] that is rated at 0.3 MW. The resulting parallel and normal heat fluxes will be 8–10 MW/m² and 2–3 MW/m², respectively, because of the compact ST geometry. At present, there is no feedback control on the plasma current. For this reason, the plasma current achieved is a measure of plasma confinement in CDX-U. In all of the experiments described in this paper, deuterium was the working gas.

3. Experiments with the L3 probe

The L3 probe was developed at UCSD for the first experiments involving the use of solid and liquid lithium as a plasma limiter in CDX-U. The head of the L3 probe is a lithium covered rail 5 cm in diameter and 20 cm long. The rail can be inserted or removed via a double gate valve airlock system to prevent exposure of the lithium to air. When the limiter is fully inserted, it forms the

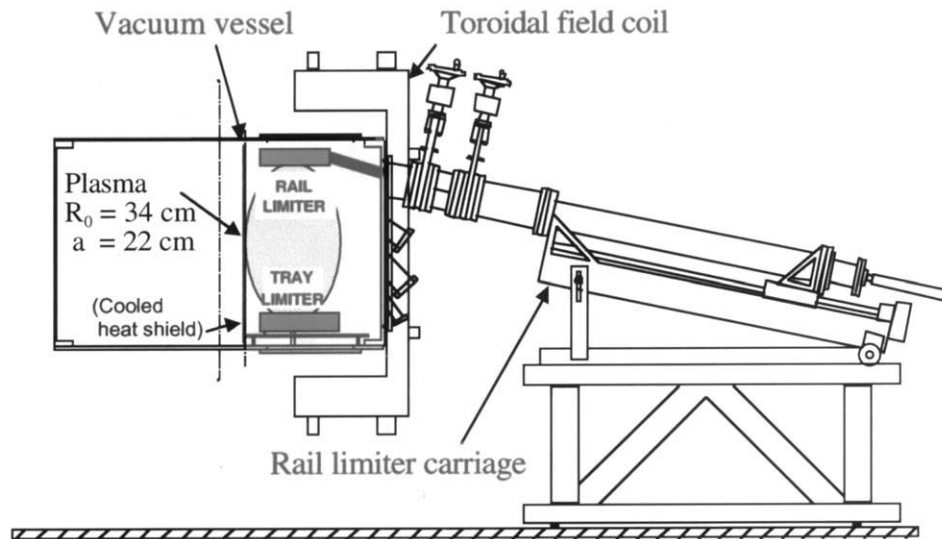


Fig. 1. Elevation of CDX-U showing the vacuum vessel, field coil configuration, and L3 limiter assembly. The plasma contact points with the L3 limiter head and the fully-toroidal liquid lithium limiter are also indicated.

Table 1
CDX-U parameters

Major radius (R_0)	34 cm
Minor radius (a)	22 cm
Aspect ratio (R_0/a)	≥ 1.5
Elongation (κ)	≤ 1.6
Triangularity (δ)	> 0.2
Toroidal field (B_t)	0.23 Tesla
Ohmic current (I_p)	≤ 90 kA
$P_{\text{auxiliary}}$ (radio frequency heating)	≤ 0.3 MW

upper limiting surface for the discharge and is intended to define the last closed flux surface for the discharge (Fig. 1). If the limiter is retracted, ceramic boron carbide rods form the upper limiting surface for the discharge. The limiter has an internal heater and has been operated in contact with the plasma over a temperature range of 20–300 °C.

The CDX-U plasma interacted strongly with the L3 head. Fig. 2 shows a sequence of images taken with a fast framing camera through a LiI (6708 Å) interference filter. The L3 head formed the upper rail limiter (Fig. 1), and the camera viewed the underside surface that faced the plasma. The frames in Fig. 2 are 1 ms apart. The location of

the bright atomic lithium emission varies as the plasma moves radially with time, and its contact points change. Occasionally, a lithium droplet is ejected, as seen in the fourth and fifth frames, and its rapid motion is governed by the electromagnetic ($\mathbf{j} \times \mathbf{B}$) forces [6].

Such events had little effect on the plasma. The lower panels in Fig. 2 show the time evolution of the OII (4416 Å), LiI (6708 Å), and D_α (6561 Å) emission, and the plasma current. The vertical lines correspond to the times the camera images were taken. While bursts were observed in the spectroscopic measurements, correlations with the ejection of the lithium were not always clear. As seen in the plasma current trace, the lithium droplets also did not cause major phenomena like large internal reconnection events [7] or disruptions.

The L3 experiments were useful in addressing the practical issues of lithium handling and the compatibility of liquid lithium with spherical torus plasmas. The general absence of any significant effect related to the L3 on plasma performance, however, is believed to be due to the relatively modest lithium surface area it presented. This issue was addressed with the fully-toroidal lithium tray

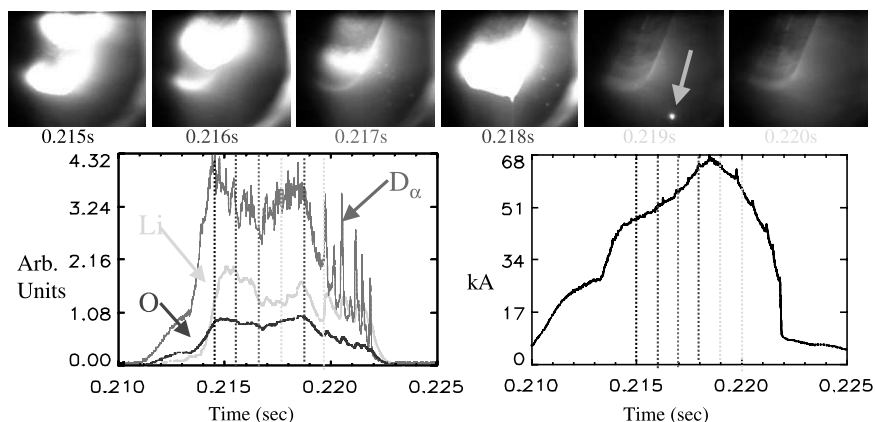


Fig. 2. Data from a CDX-U plasma with the L3 as a limiter. The upper row of panels shows fast camera images of the L3 head through a LiI (6708 Å) interference filter. The lower panels show the time evolution of the D_{α} (6561 Å), LiI (6708 Å), and OII (4416 Å) emission (arbitrary units) from detectors viewing the L3 head, and the plasma current (kA) as a function of time.

limiter (Fig. 1). Experiments with the limiter are described in the next section.

4. Results with the toroidal lithium limiter

The tray for the toroidal lithium limiter has an inner diameter of 58 cm and an outer diameter of 78 cm. Its area of about 2000 cm² thus represents an increase of more than a factor of ten over the L3 lithium surface facing the plasma. The tray is mounted on insulators to provide thermal and electrical separation between the limiter target and the vacuum vessel. Like the L3, the tray has heaters for controlling the lithium temperature. The tray isolation also permits GDC in a manner similar to the technique developed for the L3 head.

The first challenge was to fill the tray with lithium. Because of its size, this had to be done with the tray already placed in the CDX-U vacuum vessel. The chamber was filled with argon, and four quantities of solid lithium were placed on the tray with a manipulator inserted through ports in the lid. Each of the four loads consisted of four small lithium rods that were 1 cm in diameter and 5 cm long. This was intended to provide a layer of lithium 0.5 mm thick layer of liquid lithium if the tray was uniformly coated.

Melting the lithium to provide such a coating proved to be difficult. The tray heaters were able

to raise its temperature well above the lithium melting point of 181 °C. Even at 350 °C, however, the lithium rods only deformed to create oblate lumps roughly 5 cm in diameter. It appeared that a lithium hydroxide coating formed on the rods, and this prevented any liquid lithium from flowing. The base pressure in the stainless steel CDX-U vacuum vessel is $1\text{--}3 \times 10^{-7}$ T, with residual gas analyzer (RGA) measurements showing that the primary contribution was due to water. A coating could thus be readily created as the water reacts with the lithium surface.

Argon AC GDC was performed for durations approaching 30 h, both with the tray unheated and heated to 250 °C. In the latter case, the base pressure rose to about 10^{-6} T, and RGA spectra indicated that the main contribution was from hydrogen. While this was presumably liberated from the lithium hydroxide, the coating on the lithium lumps persisted.

A manipulator with an articulated stainless steel tip was inserted through the same ports used for loading the lithium. It was able to break through the coating, and spread the lithium to its maximum reach of a few centimeters along the surface of the tray. While this process did not disperse the lumps, lithium was seen to flow well beyond them. From visual inspection of the tray surface, over half of the surface area appeared to be covered with lithium.

Discharges were obtained with this degree of lithium coverage, and there were clear differences in plasma behavior with solid and liquid lithium limiters. Spectroscopic measurements from a direct view of the limiter indicated that the deuterium-alpha emission was consistently lower when the lithium was a liquid, at a temperature of 250 °C, than when it was a solid. This is shown in Fig. 3, where the measurements from liquid limiter plasmas are all clustered below the values from solid limiter discharges. There was also spectroscopic evidence for significantly reduced oxygen emission (Fig. 4).

One possible explanation for the observations with liquid lithium may be due to the fact it reacts more readily with water to form lithium hydroxide and water. Lithium deuteride could also be forming. Since this compound has a higher density than the liquid lithium, however, it may not result in a surface coating. This scenario could account for the increase in the hydrogen (mass 2), decrease in the water (mass 18), and a lack of deuterium (mass 4) in the RGA spectrum.

During one attempt at melting the lithium in the tray, a local region of strong heating developed. This may have been due to inhomogeneities in the lithium distribution resulting from the tray loading process. The resulting vaporization created a thin metal layer of lithium, which appeared to cover a large area of the vacuum vessel that had direct ('line-of-sight') exposure to the tray. The machine base pressure dropped to about 6×10^{-8} T, or

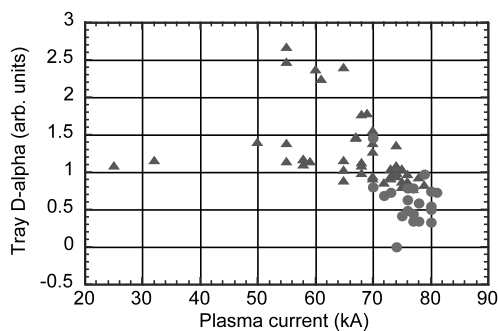


Fig. 3. D_α (6561 Å) emission measured with a detector viewing the limiter tray as a function of plasma current. The triangles are for the case of solid lithium (23 °C) in the tray, and the circles represent data obtained with liquid lithium (250 °C) in the tray.

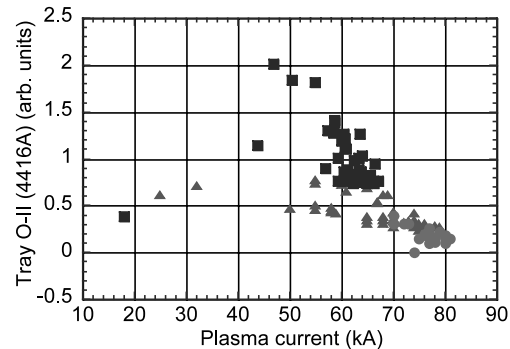


Fig. 4. OII (4416 Å) emission measured with a detector viewing the limiter tray as a function of plasma current. The squares correspond to no lithium in the tray, the triangles are for the case of solid lithium (23 °C) in the tray, and the circles represent data obtained with liquid lithium (250 °C) in the tray.

about a factor of two below typical levels for CDX-U. Discharges were first attempted without GDC, and 85 kA plasmas were achieved at very low densities of about $3\text{--}4 \times 10^{-12}$ cm⁻³. It was impossible to gas puff fuel the discharge to normal operating densities, which are in the $1\text{--}2 \times 10^{-13}$ cm⁻³ range for CDX-U.

Further experiments were performed to determine if conditioning the lithium surface on both the toroidal tray limiter and the vacuum vessel walls affected the behavior of CDX-U discharges. Argon AC GDC was conducted prior to plasma operations. Plasma currents in excess of 90 kA were observed. These were among the highest currents ever achieved on CDX-U, but the plasmas were in an even lower line-averaged density range of about $2\text{--}3 \times 10^{-12}$ cm⁻³. A sodium iodide detector outside of the stainless steel vacuum vessel usually does not detect hard X-rays, but their presence in these low density plasmas suggest the generation of runaway electrons.

Subsequent plasmas with identical discharge control programming had lower currents, higher densities, and higher impurity levels. This suggests that the GDC may have improved the surface conditions of the lithium, but that the effect did not persist. This is presumably because the walls that were coated were not heated, and the lithium

compounds subsequently formed on the surface remained there.

5. Summary and conclusions

Analysis of the data from the initial CDX-U experiments with a toroidal liquid lithium limiter has shown improvement in plasma performance. Experience to date has demonstrated that using large quantities of lithium as a plasma facing component can be done safely, but is technically challenging. Nevertheless, even incomplete coverage with clean lithium shows the benefits of a large lithium plasma facing surface.

The highest current discharges were obtained with lithium as a wall coating as well as a toroidal limiter. It thus appears that the effectiveness of the lithium on discharge performance depends on the area of the surface exposed to the plasma. The validity of this conclusion will be investigated further in future experiments with the toroidal lithium tray limiter. The present limiter will be replaced with an identical, empty tray. It will be loaded with liquid lithium, instead of solid lithium melted in place. This should reduce the inhomogeneities on the lithium surface, and increase its clean area beyond what was available during experiments with the original tray limiter.

A consistent feature of discharges with the highest plasma currents and lowest impurity emission was the difficulty in raising the density. Edge

gas puffing was the only fueling technique presently available on CDX-U, and increasing the gas fill did not lead to higher plasma densities. This suggests that the lithium has an effect on the particle recycling. In the near term, experiments with a high velocity gas jet will be attempted. The installation of a pellet injector is also being considered for the future.

Acknowledgements

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References

- [1] M.A. Abdou, et al., *Fus. Eng. Des.* 54 (2001) 181–247.
- [2] J. Menard, R. Majeski, R. Kaita, M. Ono, T. Munsat, D. Stutman, M. Finkenthal, W. Choe, Y.-S. Hwang, Y. Petrov, G. Gusev, T. Seki, *Phys. Plasmas* 6 (1999) 2002–2008.
- [3] D.K. Mansfield, et al., *Phys. Plasmas* 3 (1996) 1892–1897.
- [4] V. Lazarev et al., 26th EPS Conference on Controlled Fusion and Plasma Physics, Maastricht, The Netherlands, June 14–18 (1999) P2.076.
- [5] N.V. Antonov, et al., *J. Nucl. Mater.* 241–243 (1997) 1190–1196.
- [6] G. Antar, et al., *Fus. Eng. Des.* 60 (2002) 157–166.
- [7] M. Ono et al., 18th IAEA Fusion Energy Conference, (Montréal, Canada, October 1996) Paper IAEA-FI-CN-64/C2-2.