

Liquid Lithium Limiter Effects on Tokamak Plasmas and Plasma-Liquid Surface Interactions*

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Abstract. We present results from the first experiments with a large area liquid lithium limiter in a magnetic fusion device, and its effect on improving plasma performance by reducing particle recycling. Using large area liquid metal surfaces in any major fusion device is unlikely before a test on a smaller scale. This has motivated its demonstration in the CDX-U spherical torus with a unique, fully toroidal lithium limiter. The highest current discharges were obtained with a liquid lithium limiter. There was a reduction in recycling, as indicated by a significant decrease in the deuterium-alpha emission and oxygen radiation. How these results might extrapolate to reactors is suggested in recycling/retention experiments with liquid lithium surfaces under high-flux deuterium and helium plasma bombardment in PISCES-B. Data on deuterium atoms retained in liquid lithium indicate retention of all incident ions until full volumetric conversion to lithium deuteride. The PISCES-B results also show a material loss mechanism that lowers the maximum operating temperature compared to that for the liquid surface equilibrium vapor pressure. This may restrict the lithium temperature in reactors.

1. Introduction

The use of liquid lithium as a plasma-facing component (PFC) in a fusion reactor offers many potential benefits over conventional solid PFC's.[1] They include high heat removal capability, resilience against radiation damage, and possibilities for enhanced magnetohydrodynamic (MHD) stability if fast flowing liquid metals are used. Among the most significant properties of lithium is the ability to pump hydrogenic species, so a tokamak with a low recycling lithium PFC has the potential of achieving flat electron temperature profiles with high edge temperatures.[2]. However, large area liquid metal surfaces as a PFC in any of the present major fusion research devices is unlikely unless it is first tested on a smaller scale, and this has motivated its demonstration in the Current Drive eXperiment-Upgrade (CDX-U). The results from the CDX-U experiments are the first of their kind with a large area liquid lithium limiter in a magnetic fusion device, and we report on its clear effect on improving plasma performance through the reduction of particle recycling.

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2. Toroidal Liquid Lithium Limiter Experiments in CDX-U

The CDX-U device is a compact spherical torus (ST), with parameters $R = 34$ cm, $a = 22$ cm, $B_{\text{toroidal}} \sim 2$ kG, $I_p \sim 100$ kA, $T_e(0) \sim 100$ eV, and $n_e(0) \sim 5 \times 10^{13}$ cm $^{-3}$. Its research program is dedicated to the study of core and edge plasma modifications associated with very low recycling edges, and the effect of forced disruptions and the associated halo currents on liquid metals. To perform these investigations, a unique, fully toroidal lithium limiter was installed. The lithium was contained in a tray with an inner diameter of 58 cm and an outer diameter of 78 cm, for an area of about 2000 cm 2 . The tray is mounted on insulators to provide thermal and electrical separation between the limiter target and the vacuum vessel, and it has heaters to control the lithium temperature.

The tray heating capability was used to compare plasma behavior with solid and liquid lithium limiters. Figure 1 is a plot of the peak plasma current obtained as a function of line-average density for discharges with an unfilled stainless steel tray, solid lithium in the tray, and liquid lithium in the tray. Each data set was obtained from a single day of operation. Since CDX-U does not have plasma feedback control, higher plasma currents mean hotter, cleaner discharges. It is clear that the highest current discharges were obtained with liquid lithium in the tray.

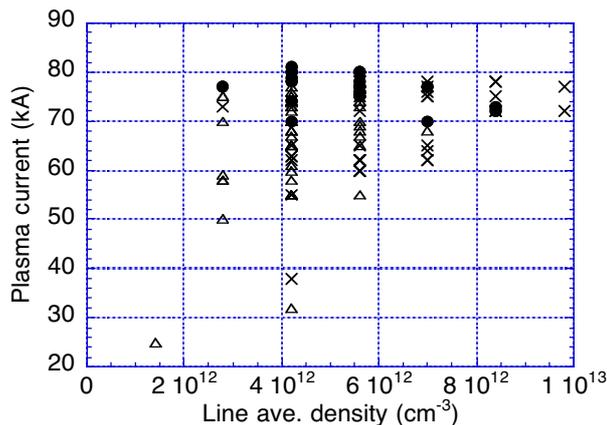


FIG. 1 - Plasma current – density operating space for the tray experiments. The crosses are from plasmas with no lithium in the tray, the triangles correspond to operation with solid lithium ($\sim 23^\circ\text{C}$) in the tray, and the solid circles represent data from liquid lithium (250°C) in the tray.

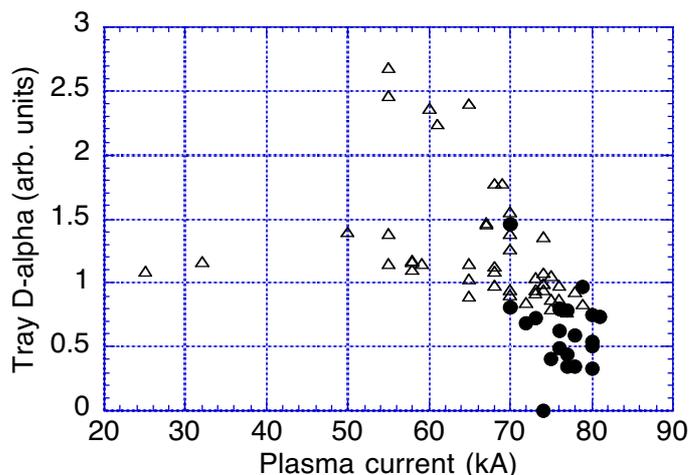


FIG. 2 – D-alpha light emitted at the tray surface as a function of peak plasma current. The triangles are from plasmas with of solid lithium ($\sim 23^\circ\text{C}$) in the tray, and the solid circles represent data obtained with liquid lithium (250°C) in the tray.

All of the data for plasmas with a liquid lithium limiter (solid circles in each of the figures) are clustered between 70 and 80 kA, including the highest point at a line-averaged density of about 4×10^{12} cm $^{-3}$. The reason for this improvement in CDX-U plasmas is related to the reduction in recycling expected from clean lithium PFC's.

This is supported by spectroscopic measurements from a direct view of the limiter, as shown in Fig. 2. They indicated that the deuterium-alpha emission was an order of magnitude lower when the lithium was a liquid, at a temperature of 250 degrees C, than when it was a solid. This is also consistent with spectroscopic evidence for significantly reduced oxygen emission with a liquid lithium limiter. As seen in Fig. 3, the oxygen level fell by about a factor of ten from its peak value with no lithium in the tray.

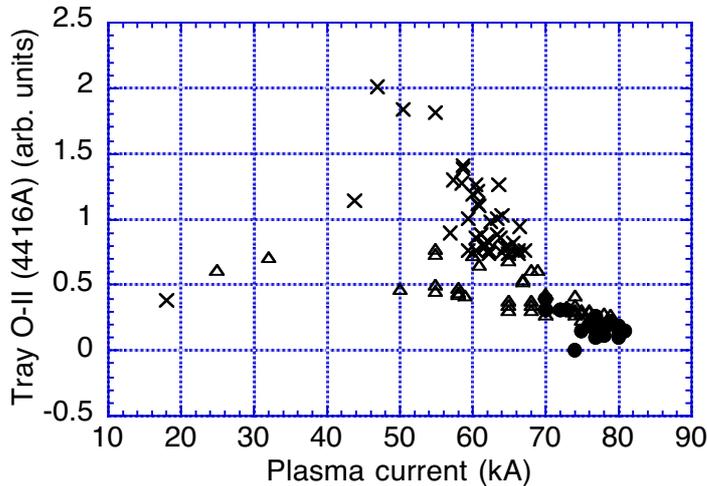


FIG. 3 - OII (4416 Å) emission measured with a detector viewing the limiter tray as a function of plasma current. The crosses correspond to no lithium in the tray, the triangles are for the case of solid lithium (~23°C) in the tray, and the solid circles represent data obtained with liquid lithium (250°C) in the tray.

As further evidence of the strong effect of the lithium on recycling, a persistent 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. In the near term, experiments with a high velocity gas jet will be attempted.

3. PISCES Studies of Liquid Lithium Recycling and Retention Behavior

The CDX-U results provide evidence for the strong effect of the lithium on recycling in an ST. How they might extrapolate to reactors has been suggested by experiments on the recycling/retention behavior of liquid lithium surfaces exposed to high-flux deuterium and helium plasma bombardment in the PISCES-B device.

Figure 4 shows the number of deuterium atoms retained in a liquid lithium sample as a function of the incident deuterium ion fluence.[3] The dotted lines bound a region defined by the uncertainty in Langmuir probe ion flux measurements, and the results indicate the full retention of all incident ions. This uptake continues until the sample is volumetrically converted to LiD, the dashed line indicating the number of lithium atoms in the sample. Once this occurs, high recycling is recovered. In contrast to the deuterium results, no detectable helium retention is found for liquid lithium samples exposed to helium plasmas up to an ion fluence of 5×10^{25} ions/m².

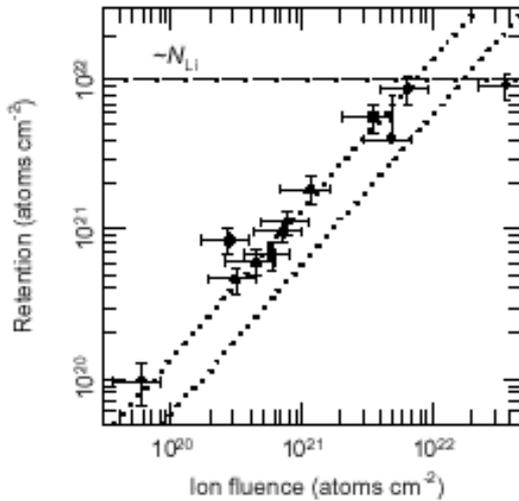


FIG. 4 – Plot of deuterium atom retention against plasma ion fluence. Low deuterium recycling occurs until the lithium sample is volumetrically converted to LiD.

The operational limit of a liquid plasma-facing surface is typically based on the amount of impurity atoms the core plasma can tolerate. This calculation involves setting a maximum operating temperature for the liquid, derived directly from the equilibrium vapor pressure. Recent measurements obtained in the PISCES-B device indicate that there is a material loss mechanism that further restricts the maximum operating temperature. This effect is observed during either deuterium [4] or helium plasma bombardment of lithium as well as in other liquid metals such as gallium.[5] What is observed is an enhanced loss of surface material in which the loss rate follows the behavior of evaporative loss, but at temperatures several hundred degrees below where the standard flux of evaporating material can explain the result. We are led to conclude that this observation is fundamental to plasma-liquid surface interactions.

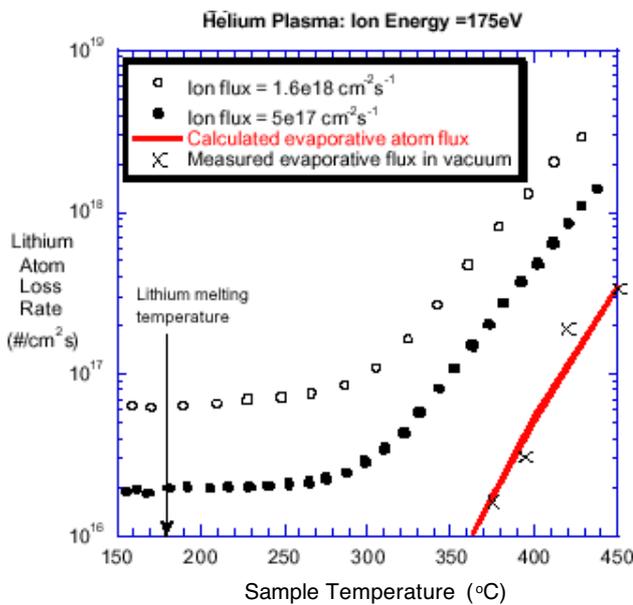


FIG. 5 – Lithium erosion rate increases exponentially at lower temperature than predicted by the vapor pressure. The erosion rate depends on the incident ion flux at high temperature.

Figure 5 shows the measured loss rate of neutral lithium atoms from a sample exposed to two different helium plasma fluxes. During the solid phase of the measurement, the erosion rate is constant in temperature, and varies with the incident ion flux as expected. However, as the temperature of the liquid sample increases, the erosion rate begins to increase exponentially at a temperature much lower than that expected from the temperature dependence of the equilibrium vapor pressure. In addition, at high temperature, the material loss rate from the surface is still a function of the incident ion flux. This fact indicates that the mechanism involved is due to an increase in the erosion yield associated with each individual incident ion striking the surface, and not simply to the equilibrium evaporation rate. To better understand this effect, we have also measured the evaporative flux of lithium in the absence of plasma. These data points are also shown in the figure, and agree well with values calculated from the equilibrium vapor pressure.

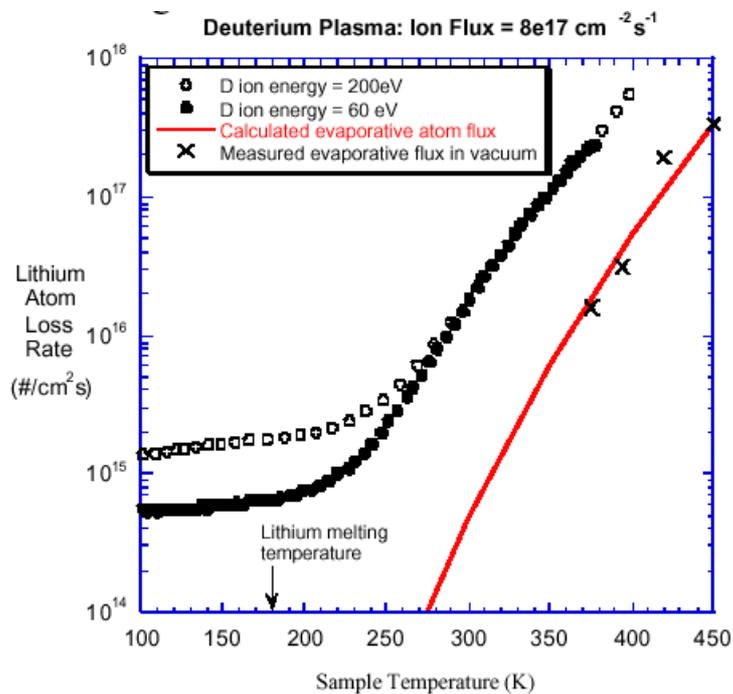


FIG. 6 – Enhanced erosion is observed to be independent of the incident ion energy.

Finally, Fig. 6 shows the enhancement of erosion as the incident energy of the ions is varied, while maintaining a constant incident flux. Obtaining a low-recycling boundary in present-day confinement devices with fairly moderate edge power fluxes may be possible using a flowing plasma-facing component system, given the results reported here. However, the extrapolation of these lithium techniques to a reactor environment appears less feasible. This is a consequence of the more restrictive operational temperature regime imposed by the unexpectedly enhanced surface erosion mechanisms observed during plasma bombardment in these experiments.

References

- [1] ABDOU, M., *et al.*, *Fus. Eng. Des.* **54** (2001) 181.
- [2] ROGNLIEN, T. D. and RENSINK, M. E., *J. Nucl. Mater.*, **290-293** (2001) 312.
- [3] BALDWIN, M. J., *et al.*, *Nucl. Fusion* (in press).
- [4] DOERNER, R. P., *et al.*, *J. Nucl. Mater.*, **290-293** (2001) 166.
- [5] CONN, R. W., *et al.*, *Nucl. Fusion* **42** (2002) 1060.