**PREPARED FOR THE U.S. DEPARTMENT OF ENERGY,
UNDER CONTRACT DE-AC02-76CH03073**

**PPPL-3509**
UC-70

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November 2000

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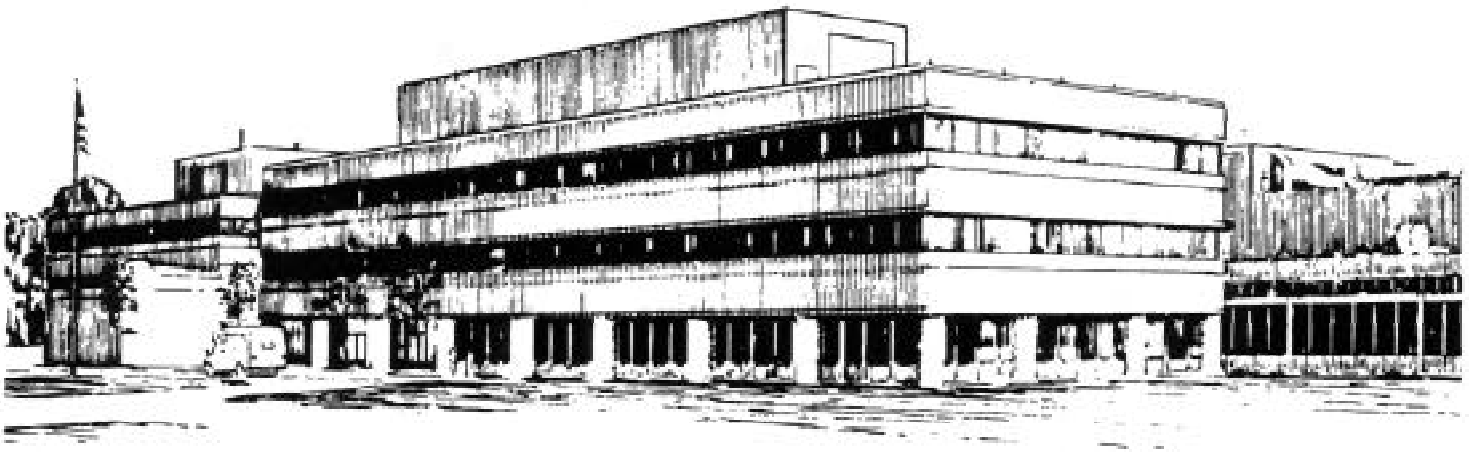

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# Liquid Lithium Experiments In CDX-U

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**Abstract.** The initial results of experiments involving the use of liquid lithium as a plasma facing component in the Current Drive Experiment - Upgrade (CDX-U) are reported. Studies of the interaction of a steady-state plasma with liquid lithium in the Plasma Interaction with Surface and Components Experimental Simulator (PISCES-B) are also summarized. In CDX-U a solid or liquid lithium covered rail limiter was introduced as the primary limiting surface for spherical torus discharges. Deuterium recycling was observed to be reduced, but so far not eliminated, for glow discharge-cleaned lithium surfaces. Some lithium influx was observed during tokamak operation. The PISCES-B results indicate that the rates of plasma erosion of lithium can exceed predictions by an order of magnitude at elevated temperatures. Plans to extend the CDX-U experiments to large area liquid lithium toroidal belt limiters are also described.

## 1. Introduction

Flowing liquid metal surfaces have been proposed as a revolutionary solution to the 30 year old problem of finding suitable plasma-facing components for a fusion reactor. A flowing liquid metal could in principle provide heat removal, elimination of erosion concerns via constant renewal of the wall, and possible stabilization of MHD modes by substituting a moving conducting wall for plasma rotation. Among the liquid metals, lithium and tin-lithium appear to be the best prospects. In addition, lithium coatings have previously been shown to enhance fusion performance in large tokamaks; experiments on the Tokamak Fusion Test Reactor (TFTR) with lithium wall coatings resulted in the third highest fusion power TFTR discharge.[1] However, there are significant physics and engineering problems which must be addressed before liquid metals can be considered a viable solution to the plasma wall problem. As a step toward this goal, plasma - lithium interactions are being studied in the PISCES-B linear plasma simulator at the University of California - San Diego (UCSD) and liquid lithium limiters are being tested in the CDX-U spherical torus at the Princeton Plasma Physics Laboratory. In addition to research on plasma-surface interactions, investigations into MHD effects in a free-surface flowing conductor in a magnetic field are beginning.[2]

Pioneering work in the area of plasma - lithium surface interactions has been previously done in the T-11M device.[3] Here we present the first research into the use of lithium as a limiting surface in a spherical torus (ST), CDX-U. In the near future, these studies will be extended to larger area liquid lithium limiters (belt limiters), liquid lithium divertor targets, and possibly flowing liquid lithium systems.

## 2. Experiments in CDX-U

CDX-U is a compact (R=34cm, a=22 cm, $B_{toroidal}$ = 2 kG, $I_P$ =100 kA, $T_e(0)$~100 eV, $n_e(0)$~ 5 × $10^{19}$ $m^{-3}$) ST. The typical discharge duration is ~25 msec at present. Diagnostics available on CDX-U for lithium studies include a 10-channel multi-layer mirror (MLM) array, which will monitor the 135 Å LiIII line for core lithium concentrations, a filterscope system to monitor deuterium recycling, lithium influx, and edge carbon and oxygen impurities, a 10,000 frame per second fast visible light camera, an infrared camera, and removable silicon samples for surface analysis. Other diagnostics include a 10 channel tangential bolometer array, multipoint Thomson scattering, and a large number of other spectroscopic systems in the visible to soft x-ray spectral range.

The first experiments involving the use of solid and liquid lithium as a plasma limiter in CDX-U have recently begun, and utilize a lithium covered rail 5 cm in diameter, 20 cm long which was developed at UCSD. The lithium limiter 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 upper limiting surface for the discharge and is intended to define the last closed flux surface for the discharge. 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 the temperature range of 20 - 300° C. The results of the first operation of CDX-U with a solid lithium limiter are shown in Fig. 1. Here we compare the D-α emission at the limiter surface with a lithium coating which has not been previously exposed to plasma, to the emission from a lithium coating which has been deuterated by exposure to plasma and gas puffing. Note that although recycling is markedly reduced for the case of initial operation with a solid lithium limiter, it is not eliminated. We have not yet observed any condition, for liquid or solid lithium over the ~20 - 300° C temperature range, for which recycling is completely eliminated. At this point it is unclear whether surface impurities may be responsible for the residual recycling.

The deuterium prefill required to obtain breakdown was 60% higher in the case of a "fresh" lithium surface than for either the deuterated case, or for normal operation with high recycling

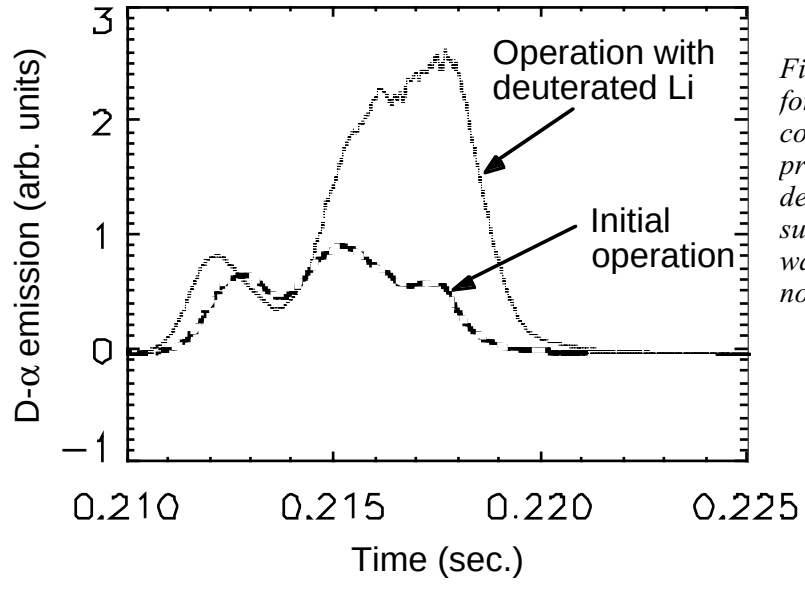


*Figure 1. Recycling comparison for a "fresh" lithium limiter coating (exposed to a base pressure of 3 x $10^{-7}$ T but not deuterated) and a deuterated surface. Discharge line density was identical to 5%; traces are normalized to the plasma current.*

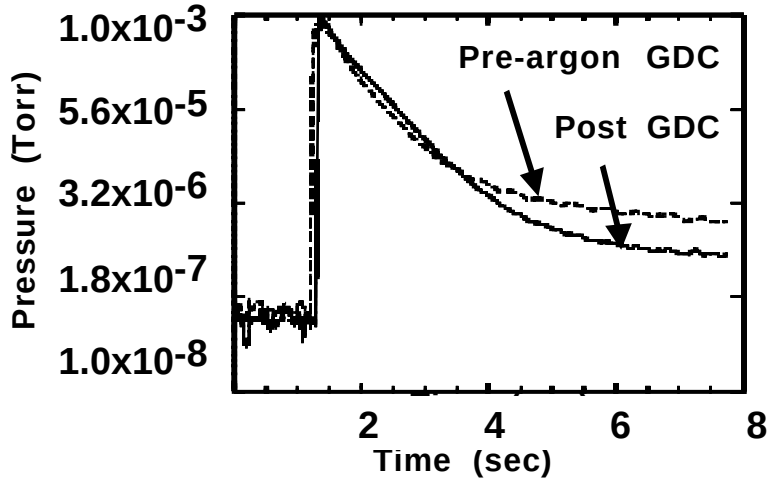


*Figure 2. Deuterium pumpout following a discharge for a deuterated lithium limiter and a freshly discharge cleaned surface.*

boron carbide limiters.

Surface cleanliness has been a significant issue. Discharge cleaning in an argon glow with the lithium rail limiter serving as the cathode has been found to be reasonably effective at removing visible surface coatings. Ion bombardment is sufficient to liquefy the lithium and heat the surface to 200 - 300$^{o}$ C. Glow discharge cleaning appears to be most effective if the lithium is liquefied.

Deuterium pumping by the limiter is significantly enhanced following an argon glow. Pumpout rates following a tokamak discharge are compared for the cases of a freshly discharge cleaned and a deuterated lithium limiter in Figure 2.

Finally, the lithium influx observed to date with either a solid or liquid lithium limiter has been modest. Spectroscopic examination of the lithium light from the surface of the limiter has shown only transient spikes of lithium emission during the discharge.

**3. Measurements of Plasma-Liquid Surface Interactions in PISCES**

Experiments are also carried out in the PISCES linear plasma simulator investigating the behavior of liquid surfaces in contact with a steady-state plasma environment. A technique has been developed in PISCES that is used to clean the surface of lithium samples prior to experiments to ensure the primary plasma-surface interaction is with a pure lithium surface. This technique has been described previously.[4] During plasma exposure, line emission from neutral lithium atoms (670.8 nm) in the plasma can be used to provide information on the erosion behavior of the sample. By viewing normal to the magnetic field (i.e. across the plasma column) the axial profile of the emission from the plasma is used to obtain the ionization rate of lithium atoms. These measurements are in agreement with rate calculations based on the Belfast atomic physics database.[5] Also, by altering the viewing geometry to be parallel to the magnetic field (i.e. along the plasma column) the Doppler shift of the neutral lithium line emission can be used to provide a direct measurement of the ejection velocity of eroded material. For low-temperature measurements, where the lithium sample is solid, the ejection velocity agrees with that calculated from an analytical Thompson model [6] using a surface binding energy of 1.7 eV. However, at higher temperatures when the lithium surface liquefies during the plasma exposure, the average ejection energy of lithium atoms leaving the surface decreases. Figure 3 is a plot of the average ejection velocity of lithium

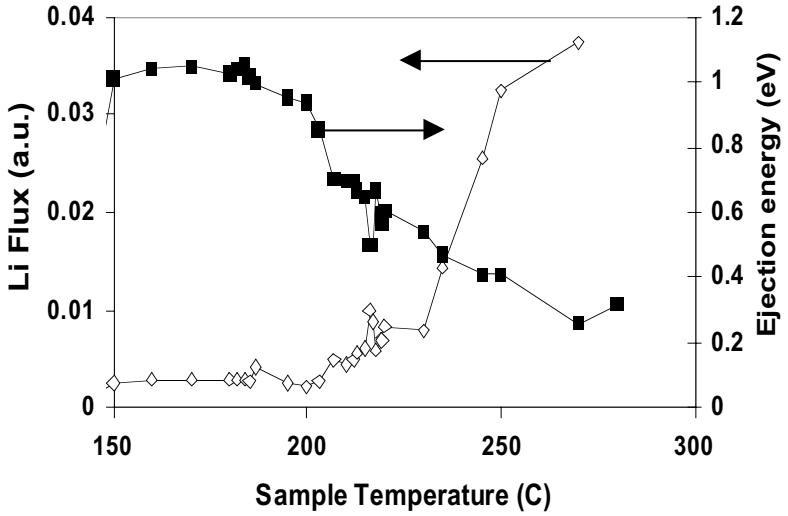


*Figure 3. Spectroscopic measurements showing the temperature dependence of the eroded lithium atomic flux from a sample, and the ejection energy of the eroded lithium atoms.*

atoms leaving the sample surface as a function of the sample temperature. (The melting temperature of lithium is 180°C.) An interesting feature of this data is that there does not appear to be an abrupt change in the ejection velocity immediately upon the change in state of the sample. Also, the ejection energy during the liquid-interaction phase of the exposure is larger than expected if a thermal release process (0.05 eV) was the dominant loss mechanism.

Since the spectroscopic system is absolutely calibrated, measurements of the erosion yield of lithium atoms from the sample can be performed. Typically, one expects the material loss rate to equal the sputtering yield at low temperature.[6] As the temperature of the sample increases the vapor pressure of the material will rise and eventually the material loss rate from thermal evaporation will become comparable to, and eventually dominate, the loss rate from sputtering. However, spectroscopic measurements of the lithium atom loss rate from plasma bombarded samples in PISCES-B[4] do not exhibit this behavior. Figure 4 shows the lithium atom flux leaving the surface of two liquid lithium samples exposed to either deuterium

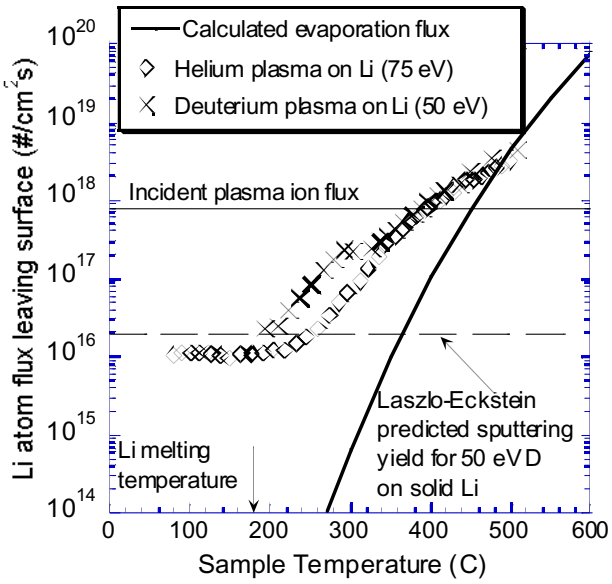


*Figure 4. Lithium efflux as a function of temperature in PISCES-B. At low temperature, near the phase transition, the loss rate coincides with that expected from physical sputtering. At high temperature (T>500C) the loss rate converges to the evaporation rate. The loss rate diverges from expectations between these two limits.*

or helium plasma bombardment as a function of the sample temperature. In the intermediate temperature regime the material loss rate can exceed, by a factor of ten or more, the predicted loss rate from the combination of physical sputtering and thermal evaporation.[7]

One possible explanation of these results could be the influence of chemical reactions between the incoming plasma ion flux and lithium atoms on the surface. In effect, a chemical erosion term that increases with temperature could be adding to physical sputtering to give enhanced loss rates. However, this is unlikely due to the fact that helium should not form stable molecules with lithium and, therefore, no chemical activity is expected in this case, but the enhanced material loss rate is still observed. Enhanced erosion at elevated temperature is also observed during helium plasma bombardment of liquid gallium surfaces in PISCES-B, indicating that this effect is not unique to lithium surfaces.

## 4. Conclusions and Future Plans

The PISCES-B results imply there may be a fundamental property of plasma interactions with liquid metal surfaces that is causing the enhanced material loss rate.[4] If this is true, it could impose a severe restriction on the use of flowing liquid metals as plasma-facing components in confinement devices. This enhanced loss will limit the operational-temperature window of the material in contact with power flux leaving the device.

In the case of CDX-U, it should be emphasized that these are initial results; operation of the device with the UCSD lithium rail limiter will be continued for several months. Following the rail limiter experiments, a full toroidal belt limiter, consisting of a 10 cm wide trough filled with liquid lithium, will be installed in CDX-U. This will increase the lithium surface area to 1600 $cm^2$ and the in-vessel volume to approximately 0.5 liter.

This work was supported by USDoE contract DE-AC02-76-CH0-3073.

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