Liquid Lithium Wall Experiments in CDX-U

R. Kaita,[a] R. Majeski,[a] S. Luckhardt,[b] R. Doerner,[b] M. Finkenthal,[c] H. Ji,[a] H. Kugel,[a] D. Mansfield,[a] D. Stutman,[c] R. Woolley,[a] L. Zakharov,[a] and S. Zweben[a]

[a]Princeton Plasma Physics Laboratory, Princeton University, Princeton NJ 08543-0451 USA
[b]University of California at San Diego, San Diego CA, 92093-0417 USA
[c]Johns Hopkins University, Baltimore MD, 21218 USA

ABSTRACT

The concept of a flowing lithium first wall for a fusion reactor may lead to a significant advance in reactor design, since it could virtually eliminate the concerns with power density and erosion, tritium retention, and cooling associated with solid walls. Sputtering and erosion tests are currently underway in the PISCES device at the University of California at San Diego (UCSD). To complement this effort, plasma interaction questions in a toroidal plasma geometry will be addressed by a proposed new ground breaking experiment in the Current Drive eXperiment – Upgrade (CDX-U) spherical torus (ST). The CDX-U plasma is intensely heated and well diagnosed, and an extensive liquid lithium plasma-facing surface will be used for the first time with a toroidal plasma. Since CDX-U is a small ST, only ≈1 liter or less of lithium is required to produce a toroidal liquid lithium limiter target, leading to a quick and cost-effective experiment.

I. INTRODUCTION

Key liquid lithium-plasma interaction questions will be addressed by a new experiment in PPPL's existing CDX-U device.[1] The primary goal will be to measure the interactions between the plasma and the lithium in an auxiliary-heated discharge whose surface contact is solely with a large-area liquid lithium limiter. The objectives of these investigations are:

- Demonstrate operation of a toroidal plasma with liquid lithium as the sole plasma-wall contact. Initial operation will be with the liquid lithium sample probe supplied by UCSD, followed by discharges utilizing a large area liquid lithium pool as the target.

- Investigate the effects of the toroidal plasma on the lithium, including thermal and magnetohydrodynamic (MHD) interactions.

A first study of lithium transport will be conducted in CDX-U in FY99. Lithium will be introduced into CDX-U using the "spark plug" technique used in previous impurity injection experiments.[2] This study will use the existing set of CDX-U profile diagnostics, including a new multipoint Thomson scattering system,[3] and an ultrasoft X-ray array from the Johns Hopkins University with multilayer mirror detectors selected for lithium line radiation.[4] These can be used to determine the lithium influx into CDX-U discharges and the lithium content and profile in the core of the plasma.

There is a small but growing experimental database on transport of liquid lithium and the behavior of lithium in contact with plasmas. Sputtering and erosion tests are currently underway at divertor simulation facilities such as PISCES at the University of California - San Diego (UCSD).[5] Lithium limiter experiments have also been performed on the T-11M device,[6] where a Capillary Porous System was used to form a "self-restoring" liquid lithium limiter surface.[7] However, introduction of large area lithium limiter targets and walls into existing tokamak facilities has not yet taken place. This proposal describes the first experiments of this type, which will be done in collaboration with UCSD.

II. OVERVIEW OF LIQUID LITHIUM WALL EXPERIMENT

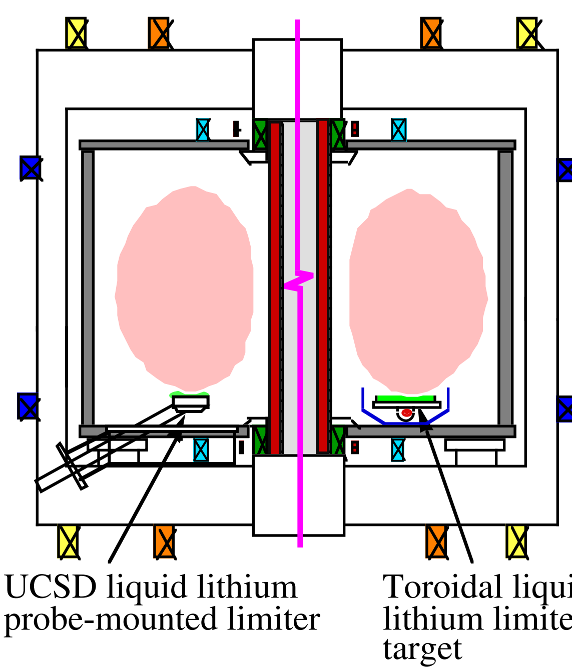

Fig.1. Schemes for liquid wall studies on CDX-U.

The first investigations on the interaction of a spherical torus (ST) plasma with liquid lithium will be performed using a liquid lithium sample probe. This probe consists of a heated sample manipulator which can be inserted into the CDX-U plasma chamber using an existing drive mechanism, as shown in Figure 1. The PISCES group at UCSD will design and construct the manipulator, and no modifications to the vacuum vessel are necessary because of the relatively small amounts of lithium (≈100 gm) used.

These sample exposures in CDX-U will permit the evaluation of effects that could not be investigated in the PISCES device, such as the dependence of lithium sputtering on ion angle of incidence, the importance of the magnetic sheath on redeposition, and the role of the ion energy distribution function on the loss rate of lithium from the sample.[7] This initial experience with liquid lithium will couple into the design and installation of the toroidal liquid lithium limiter target.

In early CY00, we will reconfigure the CDX-U to accommodate a large area liquid lithium limiter target as shown in Figure 1. The primary goal of these investigations will be to produce an ST discharge in which the plasma-wall interaction is dominated by a liquid lithium surface. The effects of operating with lithium walls will be quantified for the first time, greatly reducing the uncertainties of incorporating liquid lithium walls in larger toroidal devices such as Alcator C-Mod or NSTX.

In summary, the steps in this effort will be as follows.

- Investigate the effects of high power density (8-10 MW/m^2) but short pulse plasma interactions with liquid lithium, first with the liquid lithium sample probe and then a toroidal limiter target.

- Study thermal and MHD effects on the lithium under standard and “off-normal” conditions such as disruptions.

The liquid lithium limiter on the bottom of the plasma chamber can also be compared directly with a solid limiter on top by changing the vertical plasma position, so that the discharge is limited on the upper or lower surface of the vacuum vessel.

The CDX-U facility has recently undergone an extensive program of upgrades which has resulted in an increase of the toroidal field to 2.3 kG with a “flattop” of 100 msec. The new power supplies for the vertical and shaping fields permit discharges with plasma current up to 150 kA for greater than 25 msec. All power supplies (with the exception of the two capacitor banks) are preprogrammed and controlled by digital to analog waveform generators. The plasma geometry remains substantially unchanged, with 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[8] that is rated at 0.3 MW. The resulting parallel and normal heat fluxes will be 8-10 MW/m^2 and 2-3 MW/m^2, respectively, over 25 to 50 msec because of the compact ST geometry (see Section VI for toroidal limiter dimensions).

Major Radius (R_0)	34 cm
Minor Radius (a)	22 cm
Aspect Ratio (R_0/a)	1.5
Elongation (κ)	≤ 2
Triangularity (δ)	> 0.2
Toroidal Field (B_t) - pulsed	0.5 Tesla (≈0.5 sec)
Toroidal Field (B_t) - CW	0.1 Tesla (CW)
Ohmic Current (I_p)	≤ 150 kA
Discharge “Flat-top”	≈20-50 msec
$P_{auxiliary}$ (radio frequency heating)	≤ 600 kW

Table 1. Parameters of existing CDX-U facility.

Diagnosis of the effects of a low-recycling limiter target on the CDX-U plasma will utilize the extensive set of CDX-U diagnostics (spectroscopy, tangential bolometer array, soft x-ray diode arrays, multipoint Thomson scattering, interferometers, and edge probes).

Effects of the plasma on the lithium will be diagnosed with spectroscopy (monitoring the neutral lithium line emission at 670.8 nm), an infrared camera measuring the temperature distribution of the lithium surface, and a 10,000 frame per second fast visible camera viewing waves and turbulence on the liquid lithium surface. The possibility of measuring 2-D profiles of the lithium density in the limiter region, using a laser-induced fluorescence technique separately funded and under development by Fusion Physics and Technology, is also being explored.

III. PHYSICS ISSUES TO BE INVESTIGATED

The CDX-U research will involve the introduction of a large-area, toroidal liquid lithium limiter target (Section 4VI). A liquid lithium wall should have a very low recycling coefficient, and a direct comparison of a solid limiter versus liquid lithium target will be made by running the plasma on a solid limiter on the upper surface or a liquid limiter on the lower surface of the vacuum vessel. In addition, the effect of liquid lithium walls can be compared to boron pellet conditioning. The Boron Low Velocity Edge Micropellet Injector that was developed on CDX-U can be used for this purpose.[9]

Auxiliary RF heating will permit investigation of the effects of high power density plasma interactions with liquid lithium, using existing edge and core diagnostics. Discharge start-up and plasma current penetration are key issues that need to be resolved in the formation of plasmas in the presence of a lithium limiter target. Deuterium fueling efficiency and lithium impurity accumulation will also be investigated with the spectroscopic diagnostics on CDX-U. Possible surface coatings such as lithium hydride might be formed which could necessitate plasma conditioning, but due to their high solubility, it may not be an issue for liquid lithium targets.

Among the effects of the plasma on the liquid lithium surface to be investigated are jxB forces that result from MHD activity and disruptions. They can cause toroidal currents to flow within the toroidally-continuous lithium limiter target. The degree to which the liquid lithium serves as a conducting "shell" that affects plasma current formation, position control, and MHD stability will also be examined.

Disruptions can be studied by inducing vertical displacement events (VDE's) toward the liquid lithium limiter target. Halo currents induced by VDE's might cause the lithium to "splash," but its large surface tension and adhesion may prevent this in practice.

Surface impurities may interfere with lithium's normal 181°C melting, as has been found in some previous lithium experiments.[10-12] The lithium temperature can be varied by heaters on the toroidal limiter container up to 500°C to address this issue, and the influx of lithium as it evaporates will be monitored spectroscopically as a function of target temperature.

Experiments will also be done to discharge clean the lithium surface. A 2.45 GHz radio frequency source is available, and it can operate CW at about the 5 kW level. The resonance can be located radially in the vicinity of the limiter target and swept over its surface by oscillating the toroidal field.

IV. DEVELOPMENT OF TECHNOLOGY FOR LIQUID LITHIUM TARGETS

The first experiments will be performed with a liquid lithium sample probe inserted into the CDX-U plasma chamber, and no modifications will be made to the vacuum vessel because of the relatively small amounts of lithium (<1 gm) used. In early CY00, the CDX-U chamber and pumping system will be modified to accommodate the larger quantities of lithium required by the toroidal liquid lithium limiter. It is sufficient for this purpose to replace a few aluminum port covers with stainless steel, and to install a cold trap on the turbopump.

The design for a toroidal liquid lithium target is intended to be inexpensive and to simplify lithium handling. Lithium will be introduced to the assembled limiter target inside CDX-U as a solid, and then melted in place. Resolidified lithium will be removed from CDX-U via limiter target sector removal.

The annular toroidal limiter target will extend radially between its inner and outer sidewalls, which are located at R=29 cm (=R_0-a/4) and R=39 cm (=R_0+a/4). The limiter target will be constructed of stainless steel in four 90 degree sectors. "Knife-edge" straight interfaces between the limiter target's sectors are kept tightly pressed against each other to prevent lithium leakage, with the pressure maintained by pairs of clamps adjacent to each interface which force the sectors together. The limiter target sectors wil be mounted on insulators that provide thermal and electrical separation between the limiter target and the vacuum vessel.

A toroidally-continuous shroud will also be mounted below the limiter target and on the center stack. The purpose of the shroud is to protect vacuum vessel structures in the vicinity of the limiter target when it is heated up to 500 degrees C. This structure will be cooled with silicone diffusion pump oil, because of its low viscosity and compatibility with high vacuum and the presence of lithium if any leaks should occur.

The amount of lithium to be used in the proposed experiments can be estimated from the dimensions of the limiter target. If its depth is 0.5 cm, the quantity will be ≈1,000 cubic centimeters. Because this is comparable to the amount used previously on TFTR,[10-12] this experience is relevant to the lithium quantities, handling, and safety analysis required for the work on CDX-U.

The facilities at PPPL that were used to prepare the lithium samples for the TFTR lithium experiments are available for the proposed work on CDX-U. They include a glove box, vacuum chamber, and heater for handling lithium samples and testing various liquid metal container concepts prior to installation in CDX-U.

The transfer procedures will be similar to those used for TFTR. The UCSD lithium sample will be mounted on a probe drive, and will be brought to CDX-U in a transfer container filled with argon. The probe assembly will then be connected to a valve on the CDX-U plasma chamber and pumped out before insertion into the vacuum vessel.

The toroidal container for the lithium target will be constructed in sections as described above, and installed while the CDX-U plasma chamber is vented. Pieces of lithium will be transferred to CDX-U in an argon-filled container. The plasma chamber will then be filled with argon, after which the lithium pieces will be distributed evenly around the target container. The chamber will then be pumped down, and the toroidal container will be heated to melt the lithium.

V. CONCLUSIONS

In summary, the tasks to be undertaken are:

- Investigate the effects of plasma on a lithium target, first with a liquid lithium sample probe, then with a large area liquid lithium toroidal limiter target.

- Monitor lithium influx spectroscopically and investigate the effect of a very low recycling limiter on the plasma. Measure the dependence of the lithium content in the plasma core on the temperature (up to 500°C) of the liquid lithium limiter.

- Investigate the electromagnetic (MHD) interactions between the plasma and the lithium, including the results of a forced disruption on the lithium, and the effects of lithium on current penetration and discharge control.

The CDX-U investigations will constitute the first experiments in a toroidal plasma device where the dominant plasma-wall interaction will be with a liquid lithium surface. The research will begin with a lithium sample probe (in collaboration with UCSD), and will be followed by the introduction of a large-area, toroidal liquid lithium limiter.

Auxiliary heating in both stages will be provided by 0.3 MW of RF heating, with power deposition in the electron channel. The local power densities are expected to be in the range of 8-10 MW/m^2. The CDX-U facility has an extensive set of diagnostics already in place that are capable of evaluating lithium-plasma interactions. Measurements with these systems should be able to provide benchmarking for modeling future liquid lithium experiments.

VI. FUTURE PLANS

Experiments with divertor plasmas can be done with small modifications to the CDX-U facility. Installation of additional poloidal field coils to produce a single null configuration could be performed after the completion of the limiter experiments with the toroidal liquid lithium limiter target. The coils will be relatively simple, external to the vessel, and utilize existing power supplies.

The Tokamak Simulation Code (TSC) has been used to determine if it would be possible to create a single-null discharge in CDX-U with a modest modification to the existing coil system. The plan would be to add a pair of 180 kA-turn coils which are simple to wind and install above and below the vacuum vessel for future divertor operation. The result of a TSC calculation with the addition of these coils is shown in Figure 2. By choosing an upper or lower null, divertors made of molybdenum (a) and liquid lithium (b) can be compared directly in a fashion similar to limiter experiments. This allows experiments with H-mode plasmas, and conditions involving the edge plasma and wall that are closer to those on NSTX or Alcator C-Mod.

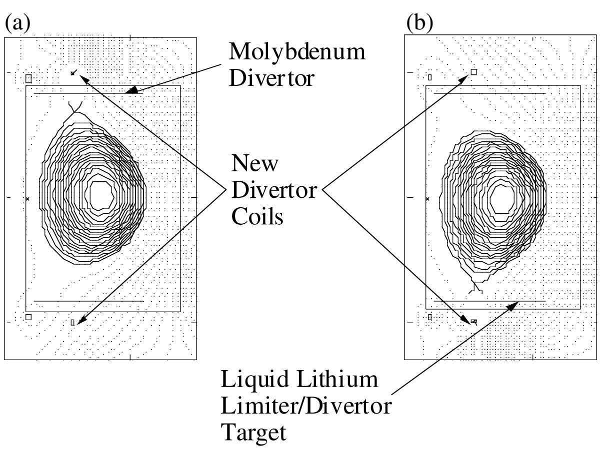

Fig. 2. TSC simulations of single-null divertor plasmas in CDX-U.

A liquid metal jet/droplet injector could be deployed after the static toroidal liquid lithium experiments. This will provide a test of the performance of flowing liquid metals in contact with the plasma boundary, an essential feature of ALPS technology.

The liquid metal jet/droplet injector is being developed by UCSD. Preliminary work on liquid metal droplet formation in the PISCES-A plasma device at UCSD has demonstrated what is required to control a high surface tension liquid metal, and many of the basic fluid properties have been investigated. A fast CCD imaging system has been used to study the trajectory and evolution of liquid metals as they pass through the discharge, and it has produced valuable insight into what should be expected as large quantities of liquid metal are introduced into the plasma-vacuum system.

The first stage would be to construct and operate a droplet injector for seeding the plasma with a liquid metal curtain in the PISCES-A machine at UCSD. This would be followed by its installation on CDX-U. The experience gained in this project and the static large area toroidal liquid lithium experiments can be used to investigate moving liquid lithium, including jxB flows, in CDX-U.

Acknowledgments

The authors acknowledge the assistance of N. Pomphrey, B. Jones, and T. Munsat in the equilibrium reconstructions for the proposed CDX-U divertor plasma configuration. This work was supported by USDOE Contract No. DE-AC02-76-CHO3073.

References

[1] R. Kaita et al., **Proceedings of the 17th IAEA Fusion Energy Conference,** Yokohama, Japan, October 19-24, 1998, IAEA-CN-69/CDP/12 (1998)
[2] F. M. Levinton and D. D. Meyerhofer, Rev. Sci. Instrum. **58**, 1393-1400 (1987)
[3] T. Munsat and B. LeBlanc, Rev. Sci. Instrum. **70**, 755-758 (1999)
[4] D. Stutman et al., Rev. Sci. Instrum. **70**, 572-576 (1999)
[5] Y. Hirooka et al., J. Vac. Sci. Technol. **A8**, 1790-1797 (1990)
[6] V. Lazarev et al., **26th EPS Conference on Controlled Fusion and Plasma Physics,** Maastricht, The Netherlands, June 14-18 1999, P2.076 (1999)
[7] N. V. Antonov et al., J. Nucl. Mater. **241-243**, 1190-1196 (1997)
[8] J. Menard et al., Physics of Plasmas **6**, 2002-2008 (1999)
[9] H. Kugel et al., Rev Sci Instrum. **70**, 493-497 (1999)
[10] H. W. Kugel et al., **Proceedings of the 17th IEEE/NPSS Symposium on Fusion Engineering,** San Diego, CA, October 6-10, 1997, 869-872 (1998)
[11] G. Labik et al., **Proceedings of the 17th IEEE/NPSS Symposium on Fusion Engineering,** San Diego, CA, October 6-10, 1997, 873-876 (1998)
[12] D. K. Mansfield et al., Phys. Plasmas **3**, 1892-1897 (1996)