SOFE05-082

LITHIUM WALL EXPERIMENTS IN CDX-U*

R. Majeski, R. Kaita, T. Gray, H. Kugel, J. Spaleta, J. Timberlake, L. Zakharov
Princeton Plasma Physics Laboratory, Princeton, NJ 08543
R. Doerner, R. Seraydarian
University of California at San Diego, San Diego, CA
V. Soukhanovskii,
Lawrence Livermore National Laboratory, Livermore, CA
R. Maingi
Oak Ridge National Laboratory, Oak Ridge, TN
rmajeski@pppl.gov

A series of experiments in the Current Drive eXperiment-Upgrade (CDX-U) have tested the use of liquid lithium as a tokamak first wall. The CDX-U is a compact ($R = 34$ cm, $a = 22$ cm, $B_{toroidal} = 2$ 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. Tokamak discharges in the CDX-U utilize a toroidal liquid lithium pool limiter 0.5 cm deep with an area of 2000 cm^2 (half the total plasma limiting surface). The use of a liquid lithium limiter is found to greatly enhance plasma performance. Tokamak discharges which used the liquid lithium pool limiter required only a quarter of the loop voltage to sustain the plasma current, compared to nonlithium limiter discharges, and a factor of 5–8 increase in gas fueling to achieve a comparable density, indicating that recycling is strongly reduced. The limiter design was demonstrated to eliminate ejection of lithium by plasma-induced $\mathbf{J}\times\mathbf{B}$ forces.

Experiments with an electron beam evaporation system have also been performed, in order to test the effect of evaporative lithium wall coatings on the discharge. These experiments also simulate conditions of high localized power deposition, as would be present in a divertor, on static liquid lithium. A 1.2 kW beam with a spot size of 6 mm (power density up to 40 MW/m^2), guided by a modest (100 G) magnetic field, is used. Surprisingly, convective transport of heat away from the beam spot is so effective that several *hundred* seconds of beam heating are required to heat the lithium to the point of evaporation ($T > 500$–600 °C). These results suggest that very high power divertor targets consisting of thin layers of static, MHD stable liquid lithium on a thermally managed substrate may be an option for future fusion devices.

*Work supported by U.S. DOE Contract No. DE-AC02-76CH03073.