Liquid Lithium Divertor Characteristics and Plasma-Material Interactions in NSTX High-Performance Plasmas*

M.A. Jaworski[1], T. Abrams[1], J.P. Allain[2], M.G. Bell[1], R.E. Bell[1], A. Diallo[1], T.K. Gray[3], S.P. Gerhardt[1], B. Heim[2], R. Kaita[1], J. Kallman[1,4], H.W. Kugel[1], B.P. LeBlanc[1], R. Maingi[3], A. G. McLean[4], J. Menard[1], R. Nygren[5], M. Ono[1], S.F. Paul[1], M. Podesta[1], A.L. Roquemore[1], S.A. Sabbagh[6], F. Scotti[1], C.H. Skinner[1], V.A. Soukhanovskii[4], D.P. Stotler[1], C.N. Taylor[2], and the NSTX Team

[1] Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA
[2] Purdue University, West Lafayette, IN 47907, USA
[3] Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA
[4] Lawrence Livermore National Laboratory, Livermore, CA 94551, USA
[5] Sandia National Laboratories, Albuquerque, NM 87185, USA
[6] Columbia University, New York, NY 10027, USA

ITER and future fusion experiments are hampered by erosion and degredation of plasma-facing components forcing regular replacement. One novel approach to solving the erosion issue is the usage of liquid metal plasma facing components. The National Spherical Torus Experiments (NSTX) is the only US confinement device operating a liquid metal divertor target to examine the technological and scientific aspects of this innovative approach.

Experiments have been conducted using a toroidal Liquid Lithium Divertor (LLD) module in the outer region of the lower divertor[1]. In addition to testing the stability of the liquid metal surface, liquid lithium in particular is expected to provide a low-Z target material and potentially absorb and retain incident particle flux through chemical bonding. The LLD surface in NSTX was replenished between discharges through evaporation and consisted of a porous molybdenum substrate on a stainless steel liner and copper heat-sink materal.

The LLD surface temperature ranged below and above the lithium melt temperature and could be heated via plasma bombardment and/or embedded heaters. No significant influx of substrate material was observed during LLD experiments evidencing maintenance of a lithium coating throughout plasma discharges. No ejection of lithium from the LLD into the plasma was observed despite operation with the outer strike-point on the LLD.

A high-density Langmuir probe (HDLP) array was implemented to better understand the demonstrated capabilities of the LLD as operated in the NSTX divertor[2,3]. A measurement of the parallel particle flux in the SOL is shown in figure 1 indicating a typical flux at the strike-point. Near-SOL particle flux is found to scale as $I_p^{-1/2}$ with the cumulative fraction within the secondary separatrix reaching 90% at currents above 1MA (weaker than the power flux scaling[4]). Heat fluxes ~5MW/m^2 were regularly applied to the LLD surfaces as well. The HDLP is also used to estimate impact energy which is required to estimate the erosion and implantation by incident particle fluxes.

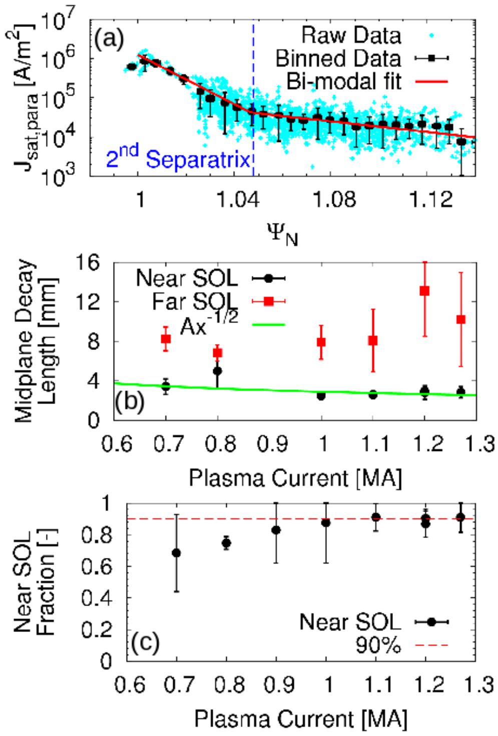

Figure 1: Particle exhaust profile and scaling with plasma current in NSTX high-triangularity discharges. (a) Particle exhaust profile, (b) radial decay length mapped to the mid-plane and (c) Near-SOL fraction of measured exhaust particles.

Kinetic interpretation of the Langmuir probes allows the determination of the entire distribution function of the measured plasma as well as the plasma potential[5,6]. Typical estimates of impact energy assume Maxwellian electron distributions. On the contrary, bi-modal distributions are obtained from the non-local interpretation[6]. The tail populations (as opposed to the cooler bulk) are found to be correlated with the biasing of the plasma above floating potential. Figure 2 shows impact energy determined from the derived plasma potential with $T_i \sim T_{e,bulk}$ assumed compared to a common assumption of $\sim 5T_e$ (*n.b.*: V_{plasma} is a minimum impact energy). The non-Maxwellian distribution temperatures can be reproduced with non-local electron transport models[7] that indicate the short electron thermal gradients give rise to tail populations. Such non-Maxwellian effects were predicted to occur in detached plasmas[8] leading to concern that simple reduction of a bulk-population temperature (e.g. by gas puffing and/or impurity radiation near the target) may not be sufficient for reducing the actual ion impact energy.

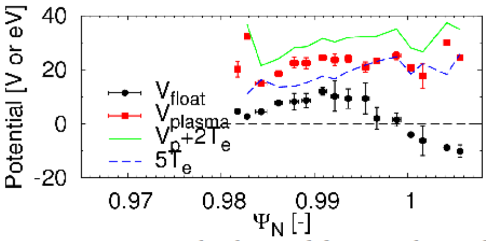

Figure 2: Potentials obtained from analysis of Langmuir probes and estimated impact energies along the outer divertor target. Impact energy calculated using plasma potential and temperature derived from non-local interpretation assuming $T_i = T_e$ and also estimated from simple $5T_e$ estimate.

Using the impact energy estimate and published sputter yields[9] gross erosion rates suggest ~350nm/s of lithium is eroded each discharge in the vicinity of the strike-point highlighting the importance of active replenishment (*c.f.* typical deposition of 150-300nm between discharges and ~3 shot degredation rate on Li-ATJ[10]). Evidence exists, however, that the surface of the LLD consisted of a *mixed-material* due to lithium gettering of background gases. Reference SRIM runs indicate impurity layers reduce the implantation depth of incident ions and could hinder retention of incident deuterium as observed in ion beam experiments[11]. On the other hand, complex surface chemistry is known to exist when Li-C-O compounds are found together indicating non-zero retention is still possible[12]. In these experiments carbon is still the majority PFC in NSTX and discharges with the LLD were found to obtain similar performance improvements and require similar fueling as previous campaigns with lithium wall-conditioning.

New design directions for next-step divertor modules are suggested by the successful implementation of this liquid metal divertor target and analysis of discharges. In particular, flowing systems can supply fresh lithium to areas underneath the strike-point in real time while removing gettered materials and purifying the liquid metal elsewhere. These technologies are under active development within the NSTX program and by the Princeton Plasma Physics Laboratory.

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