LAWRENCE
LIVERMORE
NATIONAL
LABORATORY

LLNL-JRNL-434142

NSTX Plasma Response to Lithium Coated Divertor

H. W. Kugel et al.

June 7, 2010

Journal of Nuclear Materials

Disclaimer

NSTX Plasma Response to Lithium Coated Divertor

H.W. Kugel[1], M.G. Bell[1], J.P. Allain[2], R.E. Bell[1], S. Ding[3], S.P. Gerhardt[1], M.A. Jaworski[1], R. Kaita[1], J. Kallman[1], S.M. Kaye[1], B.P. LeBlanc[1], R. Maingi[4], R. Majeski[1], R. Maqueda[1], D.K. Mansfield[1], D. Mueller[1], R. Nygren[5], S.F. Paul[1], R. Raman[6], A.L. Roquemore[1], S.A. Sabbagh[7], H. Schneider[1], C.H. Skinner[1], V.A. Soukhanovskii[8], C.N. Taylor[2], J.R. Timberlake[1], W.R. Wampler[4], L.E. Zakharov[1], S.J. Zweben[1], and the NSTX Research Team

[1] Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA
[2] Purdue University, West Lafayette, IN 47907, USA
[3] Academy of Science Institute of Plasma Physics, Hefei, China
[4] Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA
[5] Sandia National Laboratories, Albuquerque, NM 87185, USA
[6] University of Washington, Seattle, WA 98195, USA
[7] Columbia University, New York, NY 10027, USA
[8] Lawrence Livermore National Laboratory, Livermore, CA 94551, USA

Abstract

NSTX experiments have explored lithium evaporated on a graphite divertor and other plasma facing components in both L- and H- mode confinement regimes heated by high-power neutral beams. Improvements in plasma performance have followed these lithium depositions, including a reduction and eventual elimination of the HeGDC time between discharges, reduced edge neutral density, reduced plasma density, particularly in the edge and the SOL, increased pedestal electron and ion temperature, improved energy confinement and the suppression of ELMs in the H-mode. However, with improvements in confinement and suppression of ELMs, there was a significant secular increase in the effective ion charge Z_{eff} and the radiated power in H-mode plasmas as a result of increases in the carbon and medium-Z metallic impurities. Lithium itself remained at a very low level in the plasma core, <0.1%. Initial results are reported from operation with a Liquid Lithium Divertor (LLD) recently installed.

PSI19 Keywords: NSTX, Lithium, Recycling, Coating, Divertor material

JNM Keywords: C0600 Coatings; P0500 Plasma-materials interaction; S1300 Surface effects; I0100 Impurities.

PAC Numbers: 52.25.Vy; 52.40.-w; 52.40.Hf; 52.55.Fa.

* *Presenting and Corresponding author:*

H. W. Kugel, PPPL, P. O. Box 451, Princeton NJ 08543, USA. hkugel@pppl.gov.

1. Introduction

The National Spherical Torus Experiment (NSTX) [1, 2] has been investigating lithium (Li) evaporated on its plasma-facing components (PFCs) for density and impurity control [2-5]. This research with lithium applied to graphite surfaces with temperatures in the range 30-50°C (i.e., below lithium melting temperature of 180°C) has been a prelude to using liquid lithium PFCs to control density, edge collisionality, impurity influxes.

Lithium evaporated on graphite has the potential for control of density due to its ability to absorb the atomic and ionic deuterium efflux through formation of complex chemical bonds which sequester deuterium, making it unavailable for recycling [6, 7]. Due to the range of deuterium in lithium (<250 nm [8]), the intercalation of lithium into NSTX graphite [9], and the immobility of the lithium compounds formed in graphite, the absorption can saturate in the near surface layer [8, 9], limiting the deuterium pumping capability of lithium evaporated on graphite. Due to these effects, these lithium depositions can provide only short pulse capability, and subsequent recoating between NSTX discharges is required to replenish the surface with fresh lithium. Liquid lithium on the other hand has a much higher capacity for sequestering deuterium [10] because lithium-deuteride is more mobile in liquid lithium. In addition, it also has potential for handling high power fluxes, and liquid surfaces are self-healing after high-power transients [11]. NSTX will investigate if liquid lithium can produce and integrate four important potential benefits for fusion: a) divertor pumping over large surface area compatible with high flux expansion solutions for power exhaust and low collisionality, b) improved confinement, c) ELM reduction and elimination, and d) high-heat flux handling, e.g., via capillary flow.

2. Results Using Lithium Evaporated on Graphite Divertor Tiles

NSTX has been evaporating lithium on graphite tiles of the lower divertor region since 2007 [2-5]. Typically the graphite tile bulk temperatures are in the range 30-50 °C. Figure 1 shows a schematic diagram of the poloidal cross section of NSTX, and the locations of two LIThium EvaporatoRs (LITERs) at toroidal angles 165° and 315°, and the LITER central axes aimed at the lower divertor. The shaded regions indicate the measured half-angle of the roughly Gaussian measured angular distribution at the 1/e intensity of the vapor plume emerging from the exit duct. In 2009, the LITERs provided evaporated lithium depositions immediately before the majority of discharges. Shown in Fig. 2 are typical waveforms for relatively oxygen free pre-lithium wall conditions (blue) obtained using the LITER system. After 260 mg lithium deposition (red), the deuterium gas puffing required to achieve a similar average density was increased although the deuterium Balmer-α line emission (D_α) was lower, indicating reduced deuterium recycling, ELMs were suppressed, and confinement improved, as indicated by the increases in total and electron stored energy. Note, however, that with improved confinement and without ELMs, impurity accumulation increased radiated power and Z_{eff}. Quantitative measurements of C^{6+}, Li^{3+} with charge-exchange recombination spectroscopy indicate that the carbon to lithium density ratio is 30 – 100 in the core of the plasma [2]. It is possible to eliminate this impurity accumulation by introducing controlled ELMs via application of pulses non-axisymmetric radial field perturbations [12].

Shown in Fig. 3 are the radial profiles before (blue) and after 260 mg lithium deposition (red) at the time reference 0.72 s for standard pre-lithium wall condition. The radial density and temperature profiles indicate that the transition to a lithium-coated wall produced a higher temperature, lower density plasma edge. Noteworthy is the reduction in D_α luminosity, an indicator of recycling reduction, across the lower divertor surface as the lithium deposition increased from zero initially (Fig. 4).

The increase in plasma stored energy (Fig. 2) is mostly in the electron channel. The improvement in electron confinement with lithium edge conditions arises from a broadening of the temperature profile. Figure 5 compares the central electron temperature and volume averaged electron temperature between ensembles of otherwise similar discharges for pre-lithium and lithium conditions. TRANSP analysis results (Fig. 6) confirm that electron thermal transport is progressively reduced as lithium deposition increases [13]. In addition, the fast-ion contribution to total energy increased. The thermal ion confinement remains close to the neoclassical level both with and without lithium.

Analysis of the data suggests that modification of the plasma density profile by lithium edge pumping changes ELM stability, as previously predicted [14]. The lithium edge conditions reduce recycling and edge fueling. Consequently, the edge density decreases, and although the edge temperature increases, the peak edge pressure gradient is shifted inwards in minor radius where the magnetic shear is lower. As test of the applicability of the PEST and ELITE codes to the observed ELM behavior, these codes were applied to the analysis of the normalized edge current and normalized edge pressure gradient. The results indicate that NSTX pre-lithium edge profiles are close to the kink/peeling instability threshold [15], whereas the post-lithium cases are stable to these modes [15]. The PEST results which indicate instability of low-n modes are consistent with our previous observations of low n=1-5 pre-cursor oscillations before the ELM crash in NSTX [15].

Evaporated lithium depositions consistently and promptly restore good operational conditions in NSTX without boronization, including following vacuum vents to atmosphere. With a typical base pressure in the range of $3\text{-}5\times10^{-8}$ torr, the deposited lithium reacts only slowly with the residual vacuum constituents (H_2O, CO, CO_2) to form lithium compounds containing oxygen, predominantly LiOH, and to a lesser extent Li_2O and Li_2CO_3 [6]. After a typical annual 12-18 week experimental campaign, NSTX is vented to atmosphere, and purged with humidified air for at least 1 week prior to the personnel entry. This venting procedure converts any residual active lithium and LiOH to mostly inert Li_2CO_3. At the beginning of operation in 2009, the walls remained coated with some residual Li_2CO_3 after the preceding vent for installation of diagnostics and maintenance. Following bakeout and boronization (with deuterated TMB), plasmas exhibited high oxygen content which was believed to be due to this contamination. After 6 weeks of discharge conditioning, and argon, neon, and helium glow-discharge cleaning, plasmas still exhibited high oxygen content. However, after a 2.3 g lithium deposition, good operational conditions were restored in less than 10 discharges (Fig. 7). In 2010, prior to evacuation, all the plasma-facing graphite surfaces were cleaned with an abrasive pad and all exposed surfaces, both graphite and metallic, were washed with a 5% solution of acetic acid (in common vinegar) to convert Li_2CO_3 to water-soluble lithium acetate ($LiC_2H_3O_2$) so it could be removed on damp lint-free cloths. Finally, all surfaces were washed with deionized water and ethanol. After evacuation, a 3 week vessel vacuum bake was performed, and following this, a 12 g evaporated lithium deposition was applied (without boronization). Within a few hours of the first discharge, extended H-mode discharges were produced with neutral beam heating.

3. Initial Results Using Liquid Lithium Divertor

Motivated by the potential benefits of liquid lithium as a plasma facing surface, NSTX has started to test a Liquid Lithium Divertor (LLD) for integrating high plasma performance with solutions to the challenges of plasma surface interactions. Shown in Fig. 8 is a photo of the LLD installed prior to the start of operation in 2010. The LLD consists of four heated metal plates, each spanning 80° toroidally, which replace a 20 cm wide annulus of the graphite tiles on the conical outer divertor [16, 17]. The plasma facing surface is a 0.165 mm thick layer of molybdenum plasma-sprayed with a 45% porosity onto a 0.25 mm thick stainless steel barrier brazed to a 22.2 mm thick copper baseplate. The porosity is intended to make the lithium surface tension forces large relative to electromagnetic forces in the liquid layer. The thin stainless steel serves as a barrier to prevent liquid lithium from reacting with the copper substrate. Each plate is supported at its corners by the divertor baseplate with fasteners providing structural support, electrical isolation, and allowing thermal expansion. Each is electrically grounded to vessel at one mid-segment location to reduce eddy currents and to measure via a Rogowski coil "halo currents" entering the plate from the plasma. Electrical heaters and gas cooling lines are used to maintain a surface temperature in the range 20 - 400 °C. Between the sectors toroidally, graphite tiles are mounted containing diagnostics and bias electrodes. For the experiments in 2010, the LLD and the lower divertor graphite PFCs are being coated with lithium using the LITER units. Later LLD embodiments may involve a flowing lithium fill system.

The LLD is expected to provide pumping, not only when the outer strike point is directly on the LLD, but also when it is inboard of the LLD. This is due to the high flux expansion factor (15 - 20) in NSTX between the SOL at the outboard midplane and at the divertor surface. The capability to apply this range of diverted discharge configurations will test the potential benefits of liquid lithium divertor for integrating high plasma and PSI performance. In particular, it will test the capability of broad-area pumping to reduce plasma density and thereby increase neutral beam current drive to achieve advanced discharge scenarios in NSTX.

The initial lithium depositions on the LLD with the LITER system were performed with 3 of the 4 LLD plates varied in temperature from ambient to 320°C (the melting point of lithium is 180°C). During these depositions, the fourth plate remained at room temperature. Evaporation at a total rate of 20 – 40 mg/min was used. The initial depositions were applied before plasma operation started. Visible camera images of the plates during this initial loading indicated that lithium was absorbed into the porous molybdenum front face of the LLD. The unheated plate exhibited higher reflectivity than the heated plates as lithium soaked into the surface of the heated plates. It is estimated that the porous molybdenum layer on the LLD was filled to about 5% of its total capacity over several weeks.

When plasma operation started, reproducible, ELM-free, H-mode discharges were obtained with the OSP at major radii ranging from the inner divertor out to the LLD. These ELM-free, H-mode discharges with neutral beam heating of 2 – 6 MW exhibited reproducibly higher energy confinement times and reduced flux consumption early in these discharges relative to pre-lithium conditions. However, in these initial experiments, little pumping difference was measured compared to previous evaporated lithium depositions over the same region prior to installation of LLD. The LLD diagnostics are functioning, including thermocouples embedded in the LLD plates, the halo current sensors, other magnetic sensors, and an array of 99 Langmuir probes in one of the graphite tiles between the LLD plates designed to measure temperature, density, and potential for future strike point studies.

4. Discussion

NSTX experiments with lithium evaporated on the graphite divertor tiles have demonstrated an increased plasma current pulse length relative to the pre-lithium reference discharges, earlier H-mode transitions, significant density reduction in the early part of discharges, requiring more fueling early to avoid deleterious MHD activity, increased electron temperature, electron stored energy and confinement time, and reduced OV/CIII impurity ratios [5]. As the evaporated lithium deposition thickness increased, high elongation H-mode discharges became ELM-free. Eventually the HeGDC previously routinely employed between plasma shots, was eliminated, allowing an increased duty cycle.

Initial experiments have been performed with a recently installed liquid lithium divertor module. Edge pumping by the combination of the LLD, about 5% filled, and the evaporated lithium deposition on the graphite tiles exhibited little difference compared to previous evaporated lithium depositions over the same region prior to installation of LLD. This can be understood, if the initial efflux incident on active evaporated lithium deposition on the graphite tiles and on the liquid lithium in the LLD results in comparable sequestration of the incident deuterium. Post-discharge heating of the LLD plates show an increase in deuterium partial pressure consistent with thermal desorption studies of hydrated lithium [10]. Since, in NSTX lithium evaporated on the graphite divertor tiles saturates in about 1-2 discharges, its pumping effect can in effect be turned-off by ceasing to replenishment with evaporation between discharges. Hence, the next step in this work is to fill the LLD to about 50% of its capacity, then turn-off LITER to let the evaporated lithium deposition saturate. Under these conditions the resulting effects should be due predominantly to the liquid LLD.

More than a decade of research has pointed to the merits and potential reactor relevance of replenishable liquid lithium walls for providing a pumping, impurity flushing, low-Z, self-healing plasma facing surface [11, 18, 19]. NSTX lithium experiments are the first extension of this work to high beta, high power, long pulse, NBI conditions in a spherical torus.

Acknowledgments

This work is supported by United States Department of Energy (US DOE) Contracts DE-AC02-09CH11466 (PPPL), DE-AC05-00OR22725 (ORNL), and DE-AC52-07NA27344 (LLNL), and that of Sandia, a multi-program laboratory operated by Lockheed Martin Company, for the United States Department of Energy's National Nuclear Security Administration, under contract DE-AC04-94AL85000.

References

[1] M. Ono, *et al.*, Nucl. Fusion **40,** (2000) 557.

[2] M.G. Bell, *et al.*, Plasma Phys. Control. Fus. **51**, (2009) 124056.

[3] H.W. Kugel, *et al*., J. Nucl. Mater. **363-365** (2007) 791.

[4] H.W. Kugel, *et al.,* Physics of Plasmas **15** (2008) 056118.

[5] H.W. Kugel, *et al*., J. Nucl. Mater., **390-391** (2009) 1000.

[6] J.P. Allain, et al., J. Nuc. Mater. **390-391**, (2009) 942.

[7] C. N. Taylor et al., in this Conference.

[8] J.N. Brooks, *et al*., J. Nucl. Mater., **337-339** (2005) 1053.

[9] W.R.Wampler, *et al*., J. Nucl. Mater. **390-391** (2009) 1009.

[10] M. J. Baldwin, et *al*., Nucl. Fusion, **42**, 1318 (2002).

[11] V.A. Evtikhin *et al*., Plasma Phys. Control Fusion **44** (2002) 955, and references therein.

[12] J.M. Canik, et. al., Phys. Rev. Lett. **104**, at press (2009).

[13] S. Ding, et al., Plasma phys. Control. Fusion **52** (2010) 015001.

[14] L.E. Zakharov et. Al., J. Nuc. Mater., 363-365 ((2007) 453

[15] R. Maingi, et. al., Phys. Rev. Lett. **103**, 075001 (2009).

[16] H.W. Kugel, *et al.,* Fusion Eng. Des. **84** (2009) 1125.

[17] R.E. Nygren *et al*.. Fusion Eng. Des. **84** (2009) 1438.

[18] M. Abdou et al., Fus. Eng. Design **72** (2004) 1, and references therein.

[19] S.V. Mirnov, *et al*., **390-391** (2009) 876.

Figure Captions

Fig.1. Schematic diagram of the poloidal cross section of NSTX, showing the locations of the two LIThium EvaporatoRs (LITERs) at toroidal angles 165° and 315°, and the LITER central axes aimed at the lower divertor. The shaded regions indicate the measured half-angle of the roughly Gaussian measured angular distribution of the lithium vapor plume at the 1/e intensity.

Fig.2. Reference waveforms for standard pre-lithium wall conditions (discharge 129239, blue). After 260mg lithium deposition (discharge 129245, red), deuterium recycling was reduced, ELMs were suppressed, and confinement improved. The verical line at 0.72 s is the time reference for Fig.3.

Fig. 3. Measured radial profiles of the plasma density and temperatures for discharges shown in Fig.2 before (blue) and after 260 mg lithium deposition (red) at discharge time 0.72 s (vertical line in Fig.2) for standard pre-lithium wall conditions.

Fig.4. Radial profiles and reduction in Dα luminosity, an indicator of recycling reduction, as the lower divertor plasma facing surface was progressively covered with lithium. Shown are profiles at (a) 0.45 s and (b) 0.8 s.

Fig.5. Central electron temperature versus volume averaged electron temperature for an ensemble of plasmas without lithium and with evaporated lithium deposition on the PFCs under otherwise similar conditions.

Fig.6 TRANSP analysis for the electron thermal diffusivity as lithium deposition was increased with other discharges held constant.

Fig. 7 Lithiumization promptly restored good operational conditions without boronization. The application of 2.3 g of lithium following plasmas with high oxygen content resulted in significantly (a) extending pulse duration, and (b) reducing edge recycling as indicated by the deuterium Balmer-α line emission.

Fig. 8. Photo of the NSTX Liquid lithium Divertor (LLD-1) installed in 2010 near the inner edge of the outer divertor. It consists of four 80° toroidal sections, each 20 cm wide in the radial direction. Each section is separated by a row of graphite diagnostic tiles containing magnetic sensors, thermocouples, Langmuir probes and bias electrodes.

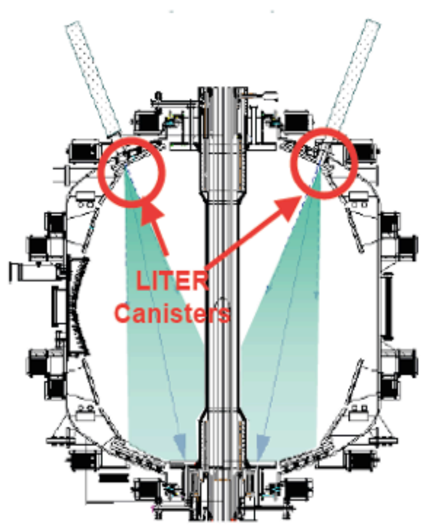

Fig.1. Schematic diagram of the poloidal cross section of NSTX, showing the locations of the two LIThium EvaporatoRs (LITERs) at toroidal angles 165° and 315°, and the LITER central axes aimed at the lower divertor. The shaded regions indicate the measured half-angle of the roughly Gaussian measured angular distribution of the lithium vapor plume at the 1/e intensity.

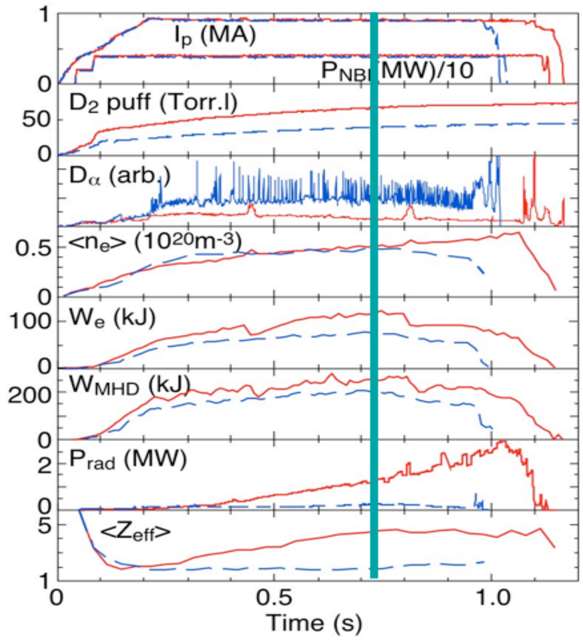

Fig.2. Reference waveforms for standard pre-lithium wall conditions (129239, blue). After 260mg lithium deposition (129245, red), deuterium recycling was reduced, ELMs were suppressed, and confinement improved. The verical line at 0.72 s is the time reference for Fig.3.

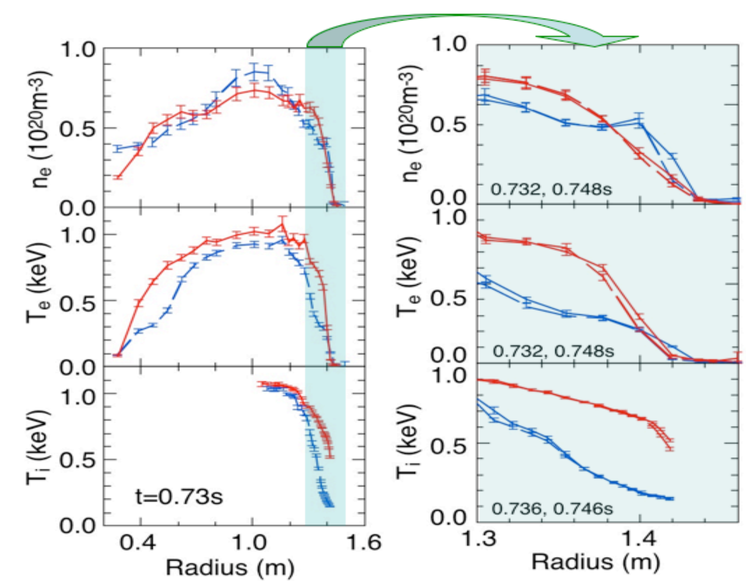

Fig. 3. Measured radial profiles of the plasma density and temperatures for discharges shown in Fig.2 before (blue) and after 260 mg lithium deposition (red) at discharge time 0.72 s (vertical line in Fig.2) for standard pre-lithium wall conditions.

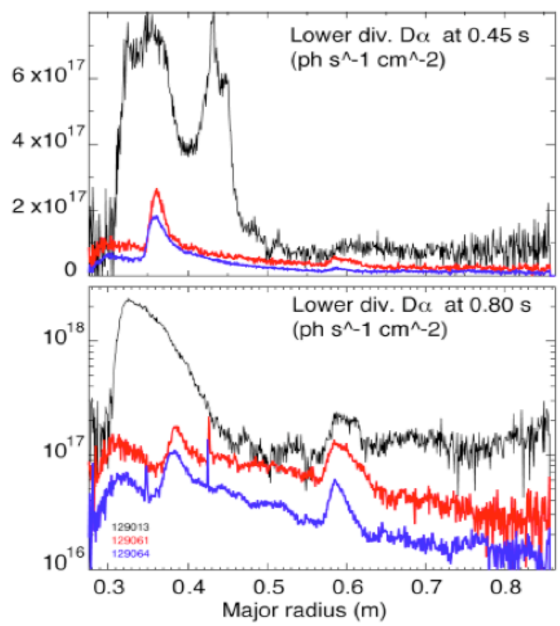

Fig.4. Radial profiles and reduction in Dα luminosity, an indicator of recycling reduction, as the lower divertor plasma facing surface was progressively covered with lithium. Shown are profiles at (a) 0.45 s and (b) 0.8 s.

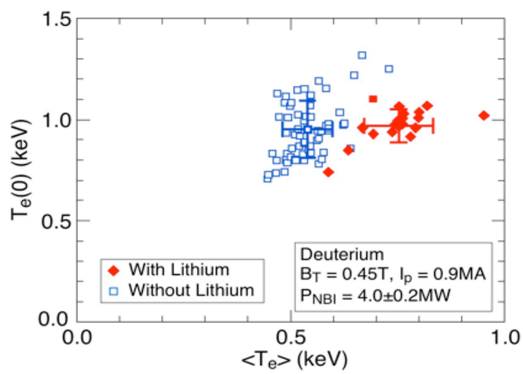

Fig.5. Central electron temperature versus volume averaged electron temperature for an ensemble of plasmas without lithium and with evaporated lithium deposition on the PFCs under otherwise similar conditions.

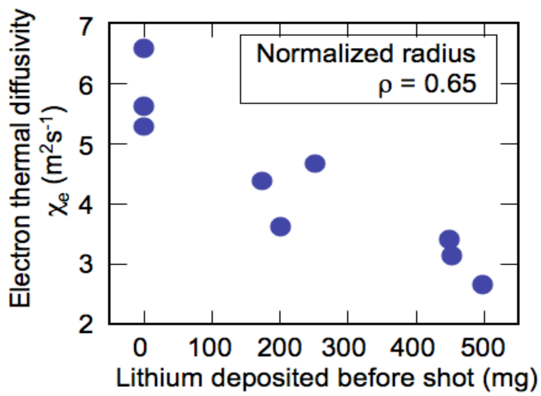

Fig.6 TRANSP analysis for the electron thermal diffusivity as lithium deposition was increased with other discharges held constant.

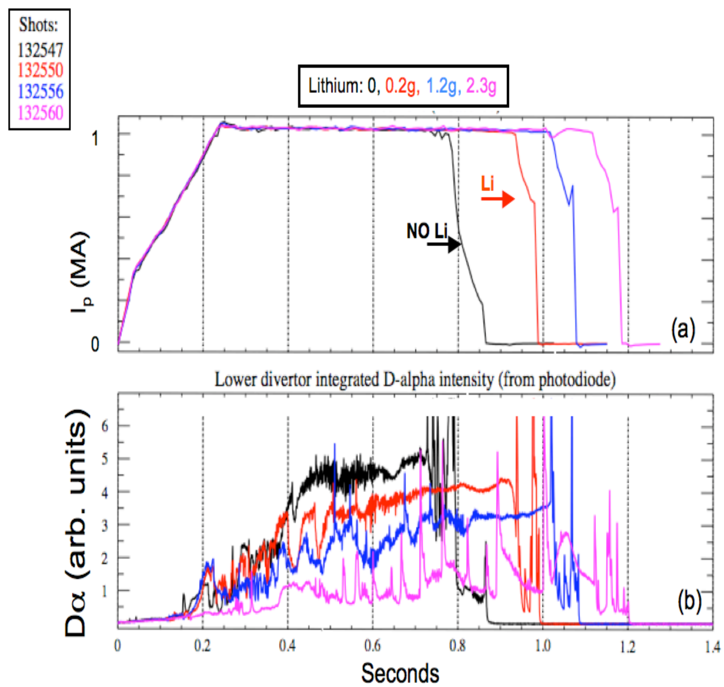

Fig. 7 Lithiumization promptly restored good operational conditions without boronization. The application of 2.3 g of lithium following plasmas with high oxygen content resulted in significantly (a) extending pulse duration, and (b) reducing edge recycling as indicated by the deuterium Balmer-α line emission.

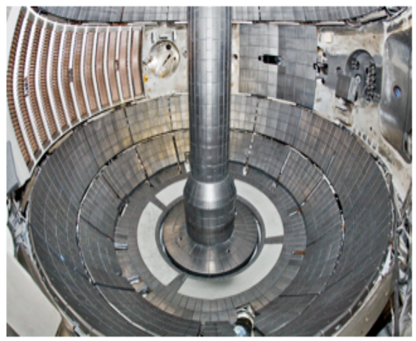

Fig. 8. Photo of the NSTX Liquid lithium Divertor (LLD-1) installed in 2010 near the inner edge of the outer divertor. It consists of four 80° toroidal sections, each 20 cm wide in the radial direction. Each section is separated by a row of graphite diagnostic tiles containing magnetic sensors, thermocouples, Langmuir probes and bias electrodes.