

## **NSTX Lithium Technologies and Their Impact on Boundary Control, Core Plasma Performance, and Operations**

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**Abstract.** Replenishable liquid lithium plasma facing components have been proposed for density and impurity control, tritium and dust removal, and long-lifetime walls for DT reactors by providing a low-Z, absorbing, and self-healing plasma facing surface. Motivated by this potential, lithium pellet injection, evaporated lithium, and injected lithium powder have been used in succession to apply lithium coatings to graphite plasma facing components in National Spherical Torus Experiment (NSTX) high-power divertor plasma experiments. In 2005, following wall conditioning and LPI, discharges exhibited edge density reduction and performance improvements. Since 2006, first one, and now two lithium evaporators have been used routinely to evaporate lithium on to the lower divertor region at total rates of 10-70 mg/min for periods 5-10 min between discharges. Since 2008, prior to each discharge, the evaporators are withdrawn behind shutters. Significant improvements in the performance of NBI heated divertor discharges resulting from these lithium depositions were observed. In 2010, these evaporators were used for almost all NSTX experiments. The improvements in NBI-heated divertor discharges resulting from these lithium depositions include: reduced edge recycling; increased energy confinement; suppression of edge localized modes (ELMs); and decreases in the inductive flux consumption resulting in longer pulse lengths. Initial work with injecting fine lithium powder into the edge of NBI heated deuterium discharges using a recently developed piezoelectric resonant acoustic injector, yielded comparable changes in performance. The next step in this work now in progress is testing of a Liquid Lithium Divertor (LLD) surface that was installed recently on the outer part of the lower divertor. The LLD consists of a toroidal array of 20 cm wide plates with a 165 micron layer molybdenum with 45% porosity, plasma sprayed on a protective barrier of 0.25 mm stainless steel, bonded to a 2.2 cm thick Cu. Several technology issues encountered with lithium wall conditions are discussed.

### **1. Introduction**

Replenishable liquid-lithium plasma facing components (PFCs) show promise for density and impurity control, tritium and dust removal, and long-lifetime walls for DT fusion reactors by providing a low-Z, low recycling, pumping, and self-healing plasma facing surface [1-3]. They have been proposed as enabling an alternative DT fusion regime [4]. Lithium has the potential for control of density due to its ability to absorb the atomic and ionic deuterium efflux through formation of lithium deuteride [5] and other compounds [6], which sequester deuterium, reducing its tendency to recycle from reactor walls where it is usually weakly bound. Due to the range of deuterium in lithium, and the immobility of the lithium deuteride formed in solid lithium, the absorption can saturate in the near surface layer, limiting the deuterium pumping capability of solid lithium. Subsequent recoating can replenish the surface with fresh lithium. Liquid lithium on the other hand has a much higher capacity for sequestering deuterium because lithium-deuteride is more mobile in liquid lithium [5]. These lithium properties can be exploited in a circulating liquid lithium system to control the tritium inventory in a reactor and also to remove impurities. Motivated by this potential, high-power divertor plasma experiments in the National Spherical Torus

Experiment (NSTX) have successively used injected lithium pellets, evaporated lithium, injected lithium powder, and a liquid lithium divertor to characterize lithium PFCs [7-11] in a staged approach. This has allowed the control systems, diagnostics, and the research program to be adapted to the evolving lithium wall conditions.

## 2. Lithium Pellet Injection

The first step of the NSTX lithium program used a lithium pellet injector (LPI) to produce lithium coatings on the plasma contact areas of the PFCs [7]. A low velocity pellet injector was developed which allowed the injection of lithium and other low-Z pellets of varying mass into NSTX edge plasmas at velocities in the range 10 to 200 m/s. The injector employed a pellet-carrying sabot propelled through a barrel by either deuterium or helium gas. The sabot was restrained by a stop and gas seal at the end of the barrel while the pellet continued along its inward radial trajectory. This gave controlled reproducible velocities for lithium pellets of 1 to 5 mg mass, and a reduced gas load to the torus. Vespel polyimide material was used for the sabot to prevent lithium reactions from causing adherence of the lithium pellet to the cartridge walls. The injector assembly consisted of four hundred barrels contained in a rotating magazine with eight barrels available for injection into each plasma discharge [7].

The NSTX LPI experiments started by first thoroughly depleting (degassing) the graphite PFCs of the plasma wetted areas of the center stack and divertor of absorbed deuterium and surface impurities. In NSTX, this was found to be essential for reducing the chemical interaction of very thin (*e.g.*, <500 nm) lithium coatings on the PFCs with absorbed deuterium and surface impurities, and their saturation, *i.e.*, unavailability for combining with incident deuterium from the plasma. This gas depletion of the PFCs of the center stack and divertor strike-point region was done using ohmic helium conditioning discharges. As an indicator of the lithium deposition process, the Li I (671 nm) luminosity was measured as the total lithium injection increased to 30 mg. These helium degassing discharges were followed by deuterium reference discharges with NBI heating. The first subsequent deuterium NBI-heated discharge exhibited a decrease in volume-average density by about 50% compared to the preceding reference discharge before the lithium application. The reduction in density was less on the second and absent on the third shot, consistent with exhausting the capacity of the deposited lithium to absorb deuterium. Approximately 3.5 mg of D<sub>2</sub> gas fueling was injected into each discharge. Pumping by the lithium coating on the conditioned graphite PFCs exhibited almost a 29% decrease in density relative to the comparison discharge. As noted above, NSTX, degassing of the PFCs using helium discharge cleaning was found to be essential for reducing the chemical interaction of very thin (*e.g.*, 0.3-500 nm) lithium coatings with absorbed deuterium and surface impurities, and their subsequent unavailability for combining with incident deuterium [7].

## 3. Lithium Evaporators

In 2006, a Lithium Evaporator (LITER) was installed to test the effect of thicker lithium coatings on plasma performance [8]. The evaporator was mounted at the top of the vacuum vessel and its exit duct was aimed toward the base of the lower center stack and divertor region. This was used to apply 100-400 mg over an area of about 2m<sup>2</sup> between discharges. The results obtained with this evaporator motivated the installation of two LITERs to evaporate lithium on to the lower inner divertor region, during the 10 min intervals between discharges in typical operation of NSTX [9]. These units are now used for the majority of NSTX plasma discharges [10]. Typical LITER deposition rates for

maintaining stable NSTX operating conditions for 2-6 MW NB heated, 1 MA plasmas are about 20 mg/min. Fig.1a shows a schematic diagram of one of two identical LITER units, used to apply these coatings. Each oven has a liquid lithium capacity of about 90 g. The evaporation rate is controlled by the oven temperature (Fig.1b). The LITER ovens are each inclined  $22^\circ$  from the vertical to allow passage through the NSTX upper divertor ports. The oven exit duct is inclined an additional  $10^\circ$  downward from the probe drive axis to aim at the inner lower divertor region. Each oven is attached to a bellows motion drive which transports it about 56 cm from its storage position outside the vessel chamber to its operating position at the edge of the graphite PFCs (Fig.1c). The oven (Fig.1a) is stainless steel (304-SS) and is surrounded by electric radiant heaters. The exit duct has a separate heater so it can be operated at a higher temperature than the main oven to prevent lithium condensate build up. The heaters are clad with alumina-based, ceramic cloth covered by an outer stainless steel

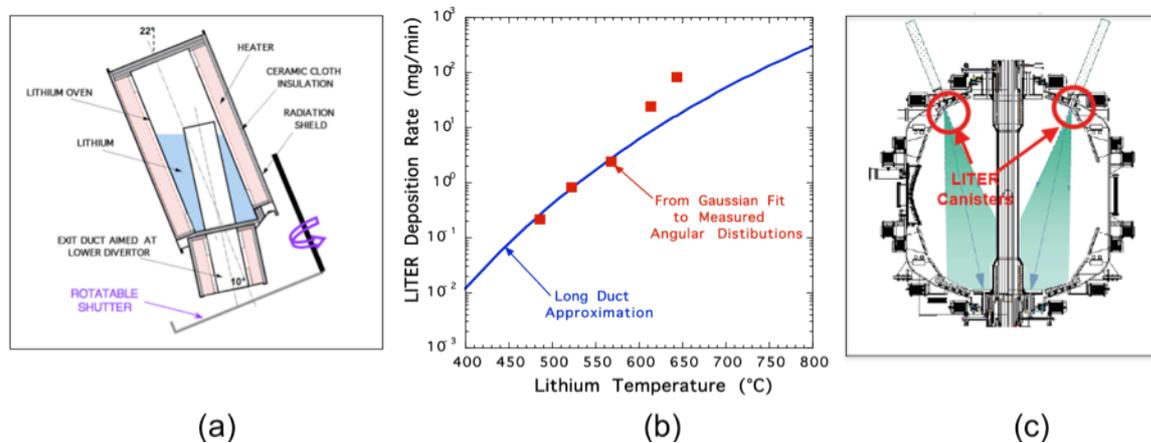


Fig.1 (a) Diagram of LITHIUM Evaporator (LITER). (b) The measured deposition rate. Above  $600^\circ\text{C}$ , LITER evaporation rate increase may be due to non molecular flow. (c) A schematic poloidal cross section of NSTX showing the locations of two LITERS at toroidal angles  $165^\circ$  and  $315^\circ$ , and the LITER central axes aimed at the lower divertor. The shaded regions indicate the measured gaussian half-angle at the  $1/e$  intensity of the measured evaporated lithium angular distributions.

radiation shield. The LITER oven temperatures are monitored and controlled using type-K thermocouples. The ovens are operated in the temperature range  $550\text{--}640^\circ\text{C}$  producing lithium evaporation rates of 5-70 mg/min. After fabrication, each oven was vacuum baked to  $900^\circ\text{C}$  to remove surface oxides to prevent their floating to the liquid lithium surface, and possibly reducing the evaporation rate. A rotatable lithium shutter is positioned in front of each LITER output aperture to stop the lithium vapor stream when the diagnostic window shutters are open during a discharge [9]. This capability also allows lithium deposition to be stopped for a sequence of discharges. This is done to establish "pre-lithium wall conditions", for reference discharges by allowing the plasma deuterium efflux to be absorbed by previously deposited lithium coatings. The plasma efflux typically saturates the lithium coating within about 2 discharges for fresh coatings of 100-200 mg between discharges and renders it unable to pump.

The initial procedure for lithium filling of LITER units employed solid, lithium pellets ( $\sim 0.6$  cm diameter). The lithium pellets were prepared by hand, and were then injected through a tube into the LITER duct using argon propellant. The injected pellets were then distributed uniformly with a tool. The method and apparatus for this technique are illustrated in Fig.2a. A full loading of LITER required the insertion of about 300 solid pellets and required about 8 loaded tube insertions. Due to the spherical packing fraction, the measured LITER loading



coating apparently due to improved confinement of higher Z impurities resulting from the elimination of ELMs [9, 10].

#### 4. Lithium Powder Injection

Experiments have been performed in which lithium was introduced in the form of a powder using a piezoelectric injector [11]. The powder consists of lithium spheres with a mean diameter of 44  $\mu\text{m}$  which are stabilized against rapid oxidation in air by a thin layer of either lithium carbonate or a paraffin wax, constituting less than 1% of the mass of lithium. Shown in Fig.4a is a schematic diagram of the injector. Particles are delivered through a small central aperture in a resonating piezoelectric disk on which the powder is supported and are allowed to fall downwards under gravity to the plasma column. Fig.4b shows the injector

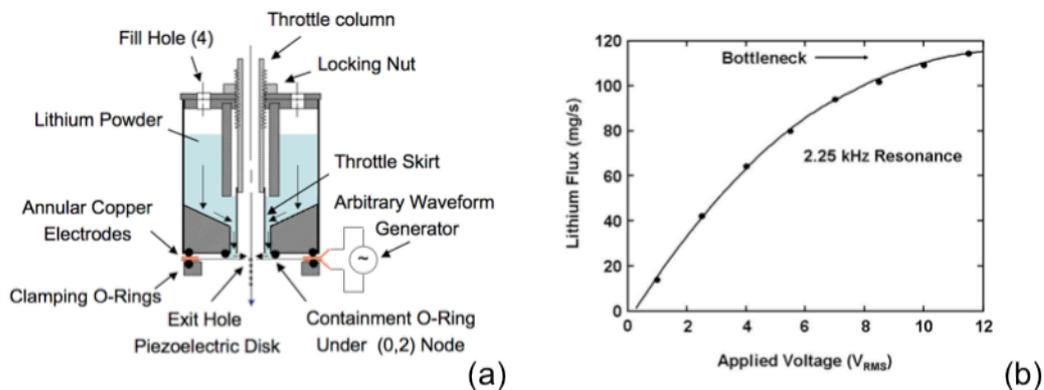


Fig.4 (a) Schematic of lithium powder piezoelectric acoustic injector. Fig.4 (b) lithium powder dropping rate (mg/s) and applied voltage. Each point represents the mass dropped during a 1s constant resonant sinusoid until the aperture becomes plugged with powder indicated by the term "bottleneck".

dropping rate (mg/s) versus voltage applied to the piezoelectric resonator. Two such devices were installed on NSTX and used in experiments to inject lithium powder into the scrape-off layer during NSTX discharges. The NSTX H-Mode plasmas have proved to be remarkably tolerant to this method of lithium injection: fluxes as high as 35 mg/s have been injected without deleterious effects, a rate stoichiometrically equivalent to amount of deuterium fueling rate. The lithium powder has also been injected prior to plasma breakdown to provide a reservoir of lithium powder on the lower divertor region. Clear improvements to plasma performance have been demonstrated with lithium powder injection [11].

#### 5. Liquid Lithium Divertor

The next step in the NSTX lithium research plan was to test a Liquid Lithium Divertor (LLD) surface (Fig.5a) that was installed recently on the outer part of the lower divertor [12,13]. The LLD consists of four plates, 22 cm wide and each spanning 80° toroidally. The plasma-facing surface has a 165  $\mu\text{m}$  layer of molybdenum with 45% porosity plasma sprayed on a protective barrier of 0.25 mm stainless steel, bonded to a 2.2 cm thick copper substrate. The molybdenum 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 of the 4 LLD plates is supported at its corners by the divertor base-plate with fasteners providing structural support, electrical isolation, and allowing thermal expansion. Each plate

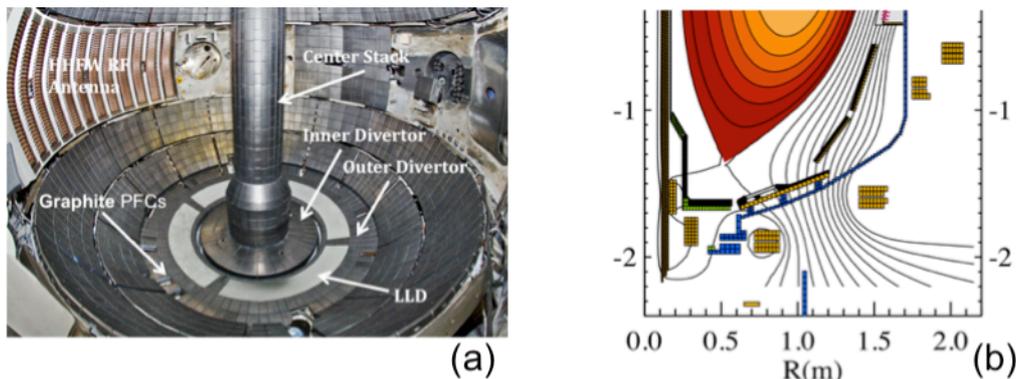


Fig.5 (a) Photo showing NSTX midplane and lower divertor regions with graphite and porous Mo LLD plasma facing components. (b) Equilibrium flux plot of shot 139587 showing LSN, 0.9MA, 3 MW NBI heated, ELM free H-mode plasma with outer strike point on LLD at  $R=68\text{cm}$ .

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. Embedded electrical heaters and gas cooling lines are used to maintain a surface temperature in the range 20 - 400 °C. The temperature distribution of each LLD plates was measured by an array of 32 type-K embedded thermocouples for bulk temperature measurements and heater control. Each LLD section is separated by a row of graphite diagnostic tiles containing magnetic sensors, thermocouples, Langmuir probes and bias electrodes. The LITER system was used to deposit lithium over the entire lower divertor region (spanning the major radius from 0.35m to 1.1m) which includes the graphite tiles and all 4 plates of the LLD (spanning the major radius from 0.65m to 0.85m). Evaporation at a total rate of 20 mg/min to 40 mg/min was used. The total lithium capacity of the porous molybdenum is 37g. The initial testing was performed using lithium depositions that filled the porous molybdenum layer to the 5-35% level (*i.e.*, 2-13g) and were applied before plasma operation commenced. 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 LLD and associated diagnostic systems have performed successfully, *i.e.*, no apparent damage to LLD or the LLD inter-quadrant diagnostic tiles was observed, and no molybdenum influxes or halo current effects observed. Fig.5b shows an equilibrium flux plot for a low triangularity, diverted discharge with an outer strike point on the LLD.

For an LLD experimental sequence, using a reference diverted discharge with its outer strike point on the LLD, and with the LLD average bulk temperatures at the start of each discharge in the range from 57°C to 102°C, 2-color IR camera temperature measurements found, that as a result of plasma auto-heating, the LLD surface temperatures during discharges reached 160-300°C (Fig.6a). This indicated that the measured LLD surface temperatures exceeded the 180°C melting point of lithium by an amount depending on the starting temperature and plasma conditions, and that the D-efflux at the outer strike point was incident on liquid lithium. Shown in Fig.6b is the ratio of electrons in the plasma to electrons input as  $\text{D}_2$  gas, *versus* the LLD bulk temperature at the start of these discharges. The plasma density decreased by more than a factor of 2, as the lithium surface of the LLD became liquid, and D retention increased. The possible presence of liquid lithium surface impurities and the complex surface morphology (roughness  $\sim 100\text{-}200\ \mu\text{m}$ ) could reduce the formation of LiD and decrease pumping relative to that of clean lithium. Laboratory work is in progress to determine the effects of surface impurities and surface morphology on solid and liquid

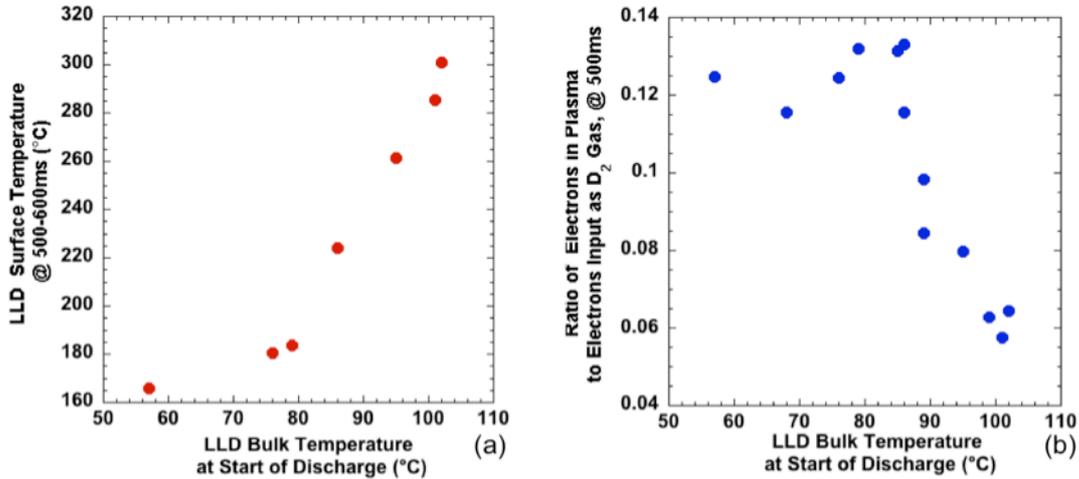


Fig. 6. (a) LLD surface temperature measured with a 2-color IR camera averaged between 500-600 ms versus the LLD plate bulk temperatures measured using thermocouples at start of discharge, and (b) the ratio of the electrons in the plasma to electrons input as D<sub>2</sub> gas at 500 ms.

lithium pumping [6]. Shown in Fig.7 are X-ray Photoelectron Spectra (XPS) of lithium evaporated on 1cm<sup>2</sup> LLD samples, followed by irradiation with a D<sub>2</sub><sup>+</sup> beam in controlled lab experiments for 5 cases of different bulk temperatures. These preliminary results indicate that D diffuses into the bulk lithium residing in the molybdenum-pores, and that the Li-O-D interactions indicative of D pumping on lithiated graphite [6] are not occurring in the laboratory LLD samples within 10 nm of the surface. This would suggest that the molybdenum LLD samples are absorbing the incident D, and that all D is being absorbed except for reflected D. Additional work is needed to understand the behavior of similar samples exposed to NSTX environment using an in-situ plasma-material interface probe.

The duration of lithium pumping, *i.e.*, ELM-free H-modes with lower edge density and higher edge temperature was measured to be  $\sim 2$ -3 discharges for a fresh thin lithium coating ( $\sim 500$ -1000 nm) for the outer strike point on either the graphite inner divertor or the LLD. However, for a 35% fill of the LLD porosity (*i.e.*, 13g), and with more than 260g of lithium on inner divertor graphite, and with the outer strike points of the reference discharge varied

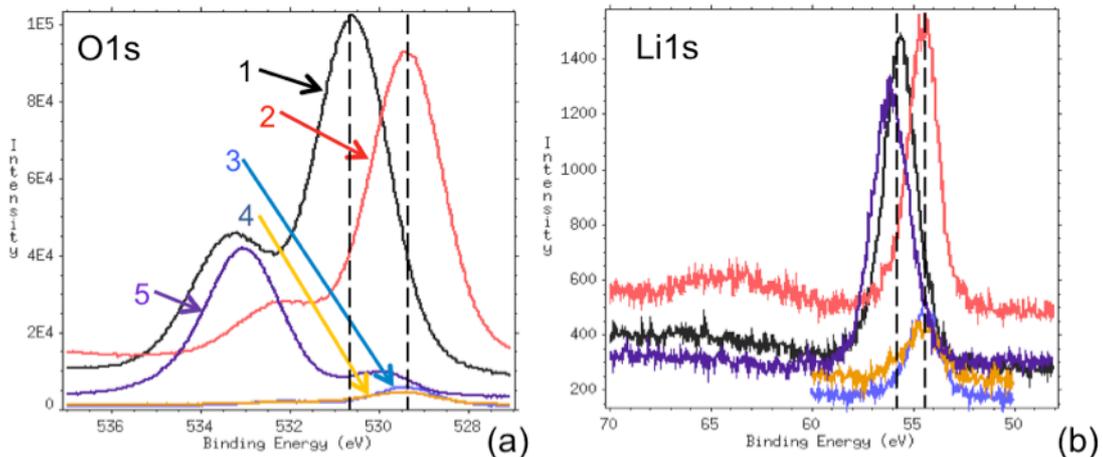


Fig 7a. XPS O1s spectra for the LLD samples, (1) 30 min D<sub>2</sub> irradiation on 2 $\mu$ m Li deposited on cold Mo, (2) 30 min D<sub>2</sub> irradiation on 2 $\mu$ m Li at 255 $^{\circ}$ C. Li deposited on cold Mo, (3) Mo202: 30 min D<sub>2</sub> irradiation on 3 $\mu$ m Li at 200 $^{\circ}$ C. Li deposited on Mo at 250 $^{\circ}$ C, (4) Carbon sputtered (via Ar) onto Li-coated Mo. 30 min D<sub>2</sub> irradiation at 200 $^{\circ}$ C, to simulate carbon impurities on LLD in NSTX (5) 30 min D<sub>2</sub> irradiation on ATJ graphite with 2 $\mu$ m Li dose. Fig.7b XPS Li1s spectra for the same samples.

between the graphite inner divertor and the LLD, the duration of lithium pumping was increased to  $\sim 100$  discharges, *i.e.*, no fresh LITER evaporations between discharges were necessary in order to maintain ELM-free H-modes. Work is in progress to test a method for direct filling the LLD without evaporation on the surrounding graphite tile.

## 6. Technology Issues During Lithium Wall Conditions and After Venting

Several technology issues have been encountered with the extensive use of lithium-coated walls in NSTX. These include the need for increased core fueling, an increasing impurity confinement and core impurity radiation during the extended discharges as confinement improved, and the need for protecting diagnostic viewports from lithium coatings. In particular, special procedures are needed when the vacuum vessel is vented to allow safe access to the interior, and for PFC preparation prior to vessel evacuation for the resumption of experimental operations. The deposited lithium reacts slowly with the residual vacuum constituents ( $\text{H}_2\text{O}$ ,  $\text{CO}$ ,  $\text{CO}_2$ ) to form lithium compounds containing oxygen, predominantly  $\text{LiOH}$ , and to a lesser extent  $\text{Li}_2\text{O}$  and  $\text{Li}_2\text{CO}_3$  [14]. The venting procedure prior to allowing personnel access in to the vessel converts any residual active lithium and  $\text{LiOH}$  to mostly inert  $\text{Li}_2\text{CO}_3$ . In NSTX, 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  $\text{Li}_2\text{CO}_3$  to water-soluble lithium acetate ( $\text{LiC}_2\text{H}_3\text{O}_2$ ) so it could be removed on damp lint-free cloths. Finally, all surfaces were washed with deionized water and ethanol.

## 7. Summary and Conclusions

In addition to extending the capabilities of existing machines, experiments with solid lithium coatings and a liquid lithium divertor plate contribute toward developing replenishable liquid lithium walls for providing a pumping, impurity flushing, low-Z, self-healing plasma facing surface. Additional work is needed to investigate if liquid lithium can produce divertor pumping over large surface areas, and high-heat flux handling, during long pulse, next-step conditions.

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## References

- [1] ABDU, M.A., et al., *Fus. Eng. Design* **72** 1-326 (2004), and references therein.
- [2] EVTIKHIN, V.A., *et al.*, *Plasma Phys. Control Fusion* **44** (2002) 955.
- [3] MIRNOV, S. V., *et al.*, **390-391** (2009) 876.
- [4] ZAKHAROV, L. E., *Fus. Eng. and Design*, **72** (2004) 149.
- [5] BALDWIN, M. J., *et al.*, *Nucl. Fusion*, **42**, 1318 (2002).
- [6] TAYLOR, C. N., et al., Proc. 19th PSI, San Diego, CA, May 24-28, 2010.
- [7] KUGEL, H.W., *et al.*, *J. Nucl. Mater.* **363-365** (2007) 791.
- [8] KUGEL, H.W., *et al.*, *Physics of Plasmas* **15** (2008) 056118.
- [9] KUGEL, H. W., *et al.*, *J. Nucl. Mater.*, **390-391** (2009) 1000.
- [10] KUGEL, H. W., et al., Proc. 19th PSI, San Diego, CA, May 24-28, 2010.
- [11] MANSFIELD, D. K., et al., *Fus. Eng. and Design* (2010), (*in press*).
- [12] KUGEL, H.W., *et al.*, *Fusion Eng. Des.* **84** (2009) 1125.
- [13] NYGREN, R.E., *et al.*, *Fusion Eng. Des.* **84** (2009) 1438.
- [14] HARILAL, S., *et al.*, *Appl. Surf. Sci.* **255** (2009) 8539.