Princeton Plasma Physics Laboratory

PPPL- 4830

PPPL- 4830

Recent Progress in the NSTX/NSTX-U Lithium Program and Prospects for Reactor-Relevant Liquid-Lithium Based Divertor Development

M. Ono, et. al.

October, 2012

Prepared for the U.S. Department of Energy under Contract DE-AC02-09CH11466.

Recent Progress in the NSTX/NSTX-U Lithium Program and Prospects for Reactor-Relevant Liquid-Lithium Based Divertor Development*

M. Ono[1], M.A. Jaworski[1], R. Kaita[1], H.W. Kugel[1], J-W. Ahn[2], J.P. Allain[3], M.G. Bell[1], R.E. Bell[1], J.M. Canik[2], S. Ding[4], S. Gerhardt[1], T.K. Gray[2], W. Guttenfelder[1], Y. Hirooka[5], J. Kallman[1,6], S. Kaye[1], B.P. LeBlanc[1], R. Maingi[2], D.K. Mansfield[1], A. McLean[6], J. Menard[1], D. Mueller[1], R. Nygren[7], S. Paul[1], M. Podesta[1], R. Raman[8], Y. Ren[1], S. Sabbagh[9], F. Scotti[1], C. H. Skinner[1], V. Soukhanovskii[6], V. Surla[10], C.N. Taylor[3], J. Timberlake[1], L. E. Zakharov[1], and the NSTX Research Team

[1] Princeton Plasma Physics Laboratory, PO Box 451, Princeton, NJ 08543, USA.

[2] Oak Ridge National Laboratory, PO Box 2008, Oak Ridge, TN 37831, USA.

[3] Purdue University, West Lafayette, IN 47907, USA.

[4] Academy of Science Institute of Plasma Physics, Hefei, China.

[5] National Institute for Fusion Science, 322-6 Oroshi, Toki, Gifu 509-5292, Japan.

[6] Lawrence Livermore National Laboratory, Livermore, CA 94551, USA.

[7] Sandia National Laboratory, Albuquerque, NM 87185, USA.

[8] University of Washington at Seattle, Seattle, WA 98195, USA.

[9] Columbia University, New York, NY 10027, USA.

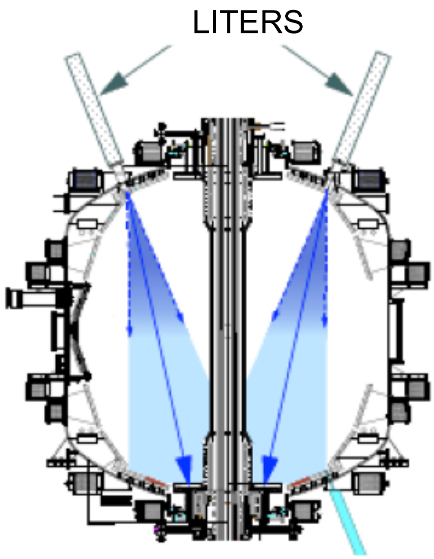
[10] Center for Plasma-Materials Interactions, University of Illinois, Urbana-Champaign, IL, USA.

E-mail contact of main author: mono@pppl.gov

Abstract. Developing a reactor compatible divertor has been identified as a particularly challenging technology problem for magnetic confinement fusion. While tungsten has been identified as the most attractive solid divertor material, the NSTX/NSTX-U lithium (Li) program is investigating the viability of liquid lithium (LL) as a potential reactor compatible divertor plasma facing component (PFC). In the near term, operation in NSTX-U is projected to provide reactor-like divertor heat loads ≤ 40 MW/m^2 for 5 s. During the most recent NSTX campaign, ~ 0.85 kg of Li was evaporated onto the NSTX PFCs where a ~50% reduction in heat load on the Liquid Lithium Divertor (LLD) was observed, attributable to enhanced divertor bolometric radiation. This reduced divertor heat flux through radiation observed in the NSTX LLD experiment is consistent with the results from other lithium experiments and calculations. These results motivate an LL-based closed radiative divertor concept proposed here for NSTX-U and fusion reactors. With an LL coating, the Li is evaporated from the divertor strike point surface due to the intense heat. The evaporated Li is readily ionized by the plasma due to its low ionization energies, and the ionized Li ions can radiate strongly, resulting in a significant reduction in the divertor heat flux. Due to the rapid plasma transport in divertor plasma, the radiation values can be significantly enhanced up to ~ 11 MJ/cc of LL. This radiative process has the desired function of spreading the focused divertor heat load to the entire divertor chamber facilitating the divertor heat removal. The LL divertor surface can also provide a "sacrificial" surface to protect the substrate solid material from transient high heat flux such as the ones caused by the ELMs. The closed radiative LLD concept has the advantages of providing some degree of partition in terms of plasma disruption forces on the LL, Li particle divertor retention, and strong divertor pumping action from the Li-coated divertor chamber wall. By operating at a lower temperature than the first wall, the LLD can serve to purify the entire reactor chamber, as impurities generally migrate toward lower temperature Li-condensed surfaces. To maintain the LL purity, a closed LL loop system with a modest capacity (e.g., ~ 1 Liter/sec for ~ 1% level "impurities") is envisioned for a steady-state 1 GW-electric class fusion power plant.

1. Introduction

Lithium wall coating techniques have been experimentally explored on the National Spherical Torus Experiment (NSTX) [1] for improving tokamak plasma performance [2]. This research also has the long-term goal of developing a reactor compatible divertor, which has been identified as a particularly challenging scientific problem for magnetic confinement fusion development [3]. While tungsten has been identified as the most attractive solid divertor material, many challenges including surface cracking and deleterious modification of the surfaces by the plasma must be overcome to develop robust plasma facing components (PFCs) [4]. In addition, for tokamaks, large ELMs cause significantly higher transient divertor heat loads which can further exacerbate surface deterioration [5]. It is therefore prudent to develop an alternative divertor solution for future devices. In the near term, operation in the NSTX Upgrade (NSTX-U) [6] is projected to produce very high divertor heat loads ≤ 40 MW/m^2 for 5 s in certain configurations. For future devices such as the Fusion Nuclear Science Facility (FNSF) [7], such a high heat flux must be handled in steady-state in a fusion nuclear environment which is very challenging indeed. The unique feature of the NSTX lithium research program has been that it can investigate the effects of Li coatings (both in solid and liquid phases) on the divertor in H-mode plasmas. Based on the experimental results from NSTX with its Liquid Lithium divertor (LLD) and experimental results from other devices and test stands, promising divertor solutions based on LL are emerging.



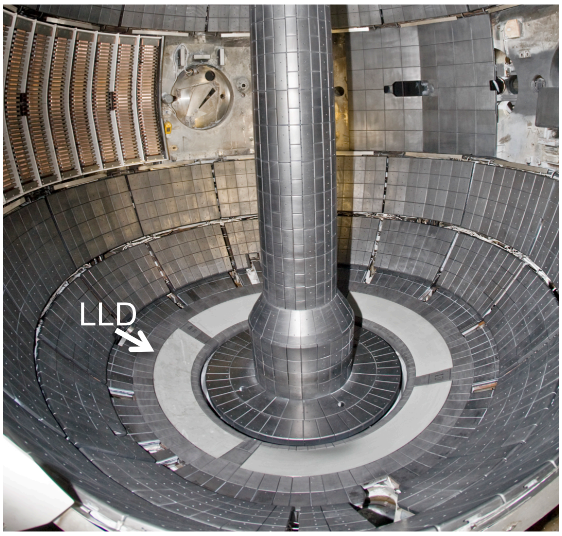


FIG. 1. Schematic of Li Evaporator Set-up. *FIG. 2. Lquid Li Divertor in NSTX*

2. NSTX Lithium PMI Experiments

Li wall coating techniques have been experimentally explored on NSTX since 2006 [8]. A unique feature of the NSTX Li research is the ability to investigate the effects of Li in H-mode divertor plasmas. The Li experimentation on NSTX started with a few milligrams of Li injected into the plasma as pellets and it has evolved to a dual Li evaporation system which can evaporate up to ~160 g of Li onto the lower divertor plates between re-loadings as shown in Fig. 1. In 2010, the NSTX Li research has focused on the effect of LLD surface on the divertor (shown in Fig. 2) including the divertor heat load, deuterium pumping, impurity control, electron thermal confinement, H-mode pedestal physics, and enhanced plasma performance [1]. To fill the LLD with Li, 1300g of Li was evaporated into the NSTX vacuum vessel during the 2010 operations. This Li evaporation system has produced many intriguing and potentially important results. The application of Li coating on NSTX has yielded a significant improvement in the electron confinement with Li coating of carbon tiles in H-mode plasmas. Importantly, the Li evaporation resulted in a broadening of H-mode electron temperature profiles compared to plasmas without Li applied [9]. The broadening of the electron temperature broadens the pressure profile, which helps to improve the plasma

MHD stability at high beta as needed for advanced plasma operations. Analysis with the TRANSP code indicates that the electron thermal diffusivity in outer region is progressively reduced with increasing Li evaporation [10]. The improving electron energy confinement with Li is consistent with the trend of improved confinement with reduced collisionality generally observed in NSTX [11]. Thus far, the electron energy confinement continues to improve with the amount of Li evaporated without reaching an apparent saturation which suggests that further improvements maybe possible. Additionally, Li was shown to reduce the H-mode power threshold [12] as well as to eliminate ELMs [13]. The NSTX Li experiments have also produced an enhanced pedestal H-mode with improved energy confinement with $H_{ITER-98}$ up to 1.7 [14]. This degree of H-mode confinement improvement should enable a compact fusion system such as the FNSF. It is also noted that even with significant applications of Li on PFCs, very little contamination (< 0.1%) of Li fraction in main fusion plasma core was observed even during high confinement modes [15]. Importantly, measurements showed a ~ 50% reduction in heat load on the LLD, indicated by enhanced bolometric radiation signals, by Li above the divertor surface [16]. The reduced divertor heat flux through radiation observed in the NSTX LLD experiment is consistent with the results from theoretical model calculations and other Li experiments [17, 18]. The Li therefore appears to be highly desirable for use as a plasma PFC material from the magnetic fusion plasma performance and operational point of view. Finally, it should be noted that the routine use of Li in NSTX has significantly improved (by up to 50% over the pre-Li plasma operations), the plasma shot availability, resulting in a record number of plasma shots in any given year [19].

3. Closed Radiative Liquid Lithium Divertor Concept

In this section we propose a radiative liquid lithium divertor (RLLD) concept in an attempt to solve the fusion reactor divertor issues where the expected divertor heat load for 1 GW-electric class fusion power plan is likely to be an order of magnitude larger (~ 50 MW/m^2) than the design limit of tungsten-based divertor PFCs (~ 5 MW/m^2). Simplified schematics of the RLLD concept is shown in Figs. 3 and 4. With an LL coating, the Li is evaporated from the divertor strike point surface due to the intense heat. By placing the liquid lithium (LL) surface in the path of the main divertor heat flux (divertor strike point), the Li is evaporated from the surface. The evaporated Li is quickly ionized by the plasma and the ionized Li ions can provide a strongly radiative layer of plasma ("radiative mantle"), thus could significantly reduce the heat flux to the divertor strike point surfaces, protecting the divertor surface. The ionized Li migrates to the surrounding closed divertor wall surfaces as shown in Fig. 4. Due to the cooler diverter wall, the Li is condensed on the wall forming LL film surfaces. The LL surfaces should provide strong divertor pumping for both hydrogenic and impurity species. The LL is eventually collected at the bottom of RLLD and purified through a close-loop system as shown in Fig. 5.

At the second Lithium Symposium in 2011 [20], a panel discussion was held addressing the previously identified questions, "Is a Li PFC viable in magnetic fusion reactors such as ITER?" The following specific technical issues for Li reactor applications were identified in the first Lithium Symposium [21]: 1. Handling high divertor heat flux, 2. Removal of deuterium, tritium, and impurities from LL, 3. Removal of high steady-state heat flux from divertor, 4. Flowing of LL in magnetic fields, 5. Longer term corrosion of internal components by LL, 6. Safety of flowing LL, and 7. Compatibility with LL with a hot reactor first wall. In this section, we shall attempt to address those questions for the RLLD concept being proposed in this paper as shown in Figs 3 and 4. The divertor heat flux handing prospect of RLLD (issues # 1 and 3) is addressed in Sec. 3.1, and the compatibility of RLLD with a hot reactor first wall (issue # 7) is covered in Sec. 3.2. The remaining issues (issues #

2, 4, 5, and 6) are relatively well defined engineering and technology problems related to a circulating LL loop system as shown in Fig. 5 which can be resolved, in principle, through targeted R&D efforts. It should be also noted that those issues are actively addressed in the engineering R&D for the IFMIF and fusion blanket module development as discussed in Sec. 3.3. In Sec 3.4, we shall comment on the advantages of closed RLLD system.

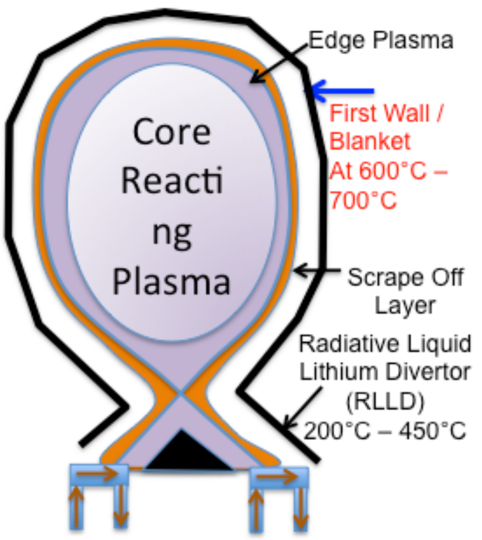


FIG. 3. Schematic of closed RLLDs in a reactor.

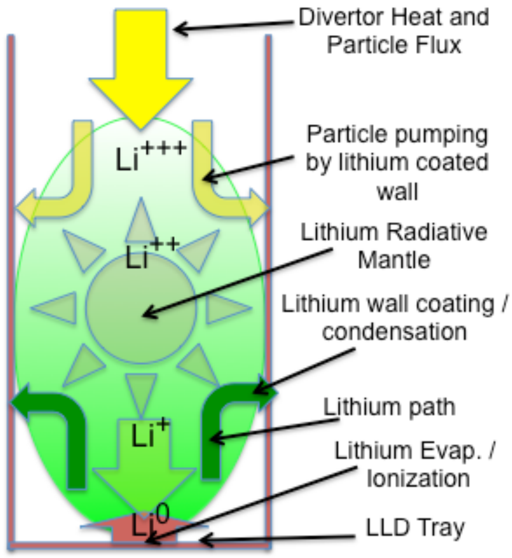


FIG. 4. Schematic of a closed RLLD concept.

3.1. Heat Flux Handling Prospects of RLLD

Previous model calculations have shown that due to the relatively rapid plasma transport in divertor plasmas, the radiation values can be significantly enhanced compared to the coronal equilibrium values, radiating up to ~ 1.2 keV per injected Li ion (equivalent to ~ 11 MJ/cc of LL) [17, 18]. This radiative process has the desirable function of spreading the narrowly focused divertor heat load to the entire divertor chamber, which facilitates the divertor heat removal. Since the projected Li radiation can be very high (i.e., ~100 MJ / mole of Li ions), it should be readily feasible to provide an adequate amount of LL in the high heat flux divertor region even for steady-state operation. For example, to radiate an estimated 200 MW of influx power into divertor for an 1 GW-electric class fusion power plant, it would only take two moles Li per sec to handle the divertor heat load. One can envision even higher evaporation rate if necessary. A 1-D model calculation of RLLD has been performed with a two-point model in a cylindrical geometry [22]. With Li radiation level of 10^{-26} W-cm^3, which assumes relatively poor confinement of $n_e\tau = 10^9 – 10^{10}$ in the divertor region, the radiative cooling can reduce the divertor heat flux by a factor of two as observed in NSTX [16] with only ~ 10^{17} Li particles in the divertor heat flux cylinder of R = 0.75 m and ΔR = 1cm. The same model predicts ~ 10^{18} Li particles would be needed for an ST-based FNSF with R= 1.5 m and ΔR = 2 cm. If the Li radiative mantle could reduce the direct divertor heat flux down to the vicinity of 5 MW /m^2, then the solid substrate material (perhaps made out of tungsten) could become viable as long as it is coated with Li. As for the actual substrate, the requirement is to ensure an adequate amount of LL available at the location of intense heat flux (i.e., near the divertor strike point). The LL surface can also provide a "sacrificial" surface to protect the substrate solid material even from transient high heat loads such as the ones caused by ELMs. If the transient heat flux is high, that much more Li would be evaporated and ionized, which would then increase the radiative cooling until an equilibrium condition is reached. For example, in an ITER scale tokamak reactor, with the enhanced radiative process, only a modest amount (~ 1cc) of LL is needed in principle to radiate the expected heat pulse of ~ 10 MJ for an exceptionally large ELM event which is a relatively modest amount.

3.2.Compatibility of RLLD with Reactor Hot First Wall

As shown Fig. 3, it is generally envisioned in magnetic fusion reactor studies that the reactor first wall temperature will be high (~ 600 - 700 °C) to keep the first wall surface relatively clean, particularly of tritium to keep the tritium inventory reasonably low, and to achieve high electrical power conversion efficiencies. On the other hand, the RLLD temperature is likely to be lower (~ 200 – 450 °C) to avoid excessive Li evaporation [17]. Because of this temperature discrepancy, LL is often thought to be not compatible with the reactor environment. However, the lower operating temperature of LL may in fact make it suitable for the reactor divertor operation. With the low melting temperature of ~ 180°C, it is quite practical to keep Li in a liquid state in a hot reactor environment, provided that the LL can be kept relatively clean as discussed in Sec. 3.3. If the RLLD operates below the first wall temperature as noted above, the Li and associated impurities should migrate toward the lower temperature RLLD chamber, and keep the higher temperature first wall relatively clean. There was an interesting experiment in T11-M where a lower operating temperature Capillary-Porous System (CPS) was able to actually collect Li from the higher temperature CPS via plasma interactions [17]. The purpose was to demonstrate how the particles and gas could migrate toward the lower temperature region in the vacuum-plasma system within a fusion reactor. The approach would be similar to a dehumidifier (which has colder condensing surfaces) collecting the water vapor within the room. In addition, the all plasma particles (except helium) tend to end up in the divertor chamber because of the net particle flow from the main chamber to the closed divertor chamber with frictional forces in the direction of the closed divertor chamber. Therefore, a RLLD operating below the first wall temperature, together with a purifying system as shown in Fig. 5, could serve as the gas pumping, tritium recovery, and impurity control system for the entire reactor plasma - vacuum system as discussed in Sec. 3.3. Also importantly, with lower operating temperature, one can envision utilizing steel-based materials as potential substrates and support structures of RLLD where the LL provides a low-Z protective layer over the high-Z steel. This material choice eliminates the need to transition from structural steels to tungsten where the operating temperature windows do not always overlap [23].

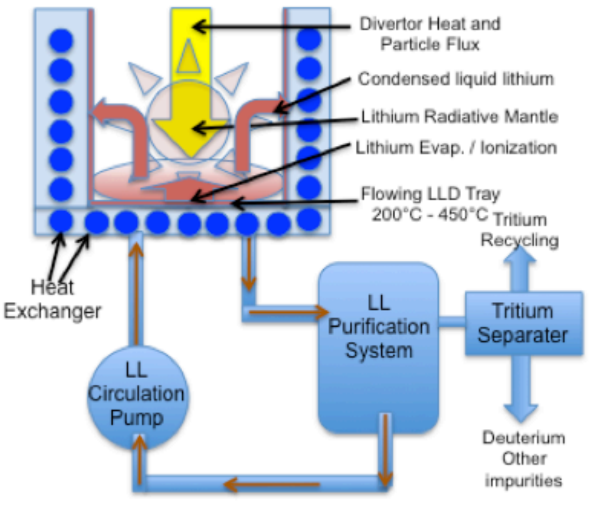


FIG. 5. Schematic of RLLD Li-loop with. purifying system

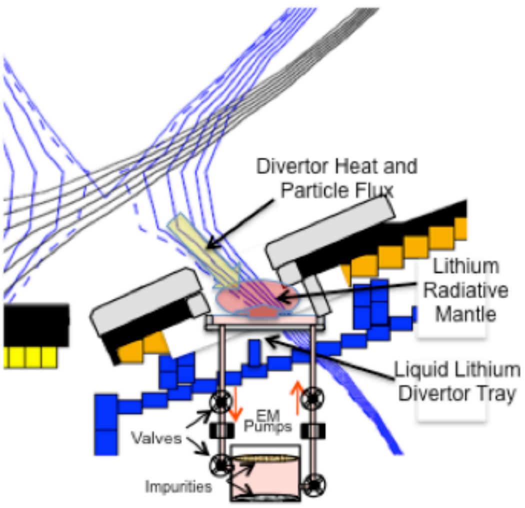


FIG. 6. Schematic of a possible closed LL RLLD in NSTX-U.

3.3.Lithium Loop Issues for RLLD

For RLLD to be viable for steady-state reactor operation, it is essential to continually purify the LL by removing D, T, and various impurities including the Li compounds (see Figure 5). Prompt removal of tritium is particularly important to keep the tritium inventory level low. Fortunately, the LL circulation requirements for impurity removal from LL are relatively modest. For a 1 GW-electric plant, the main "impurity species" would be the hydrogenic (D-T) fuel gas which maybe estimated to be order of one mole per second. If a LL loop were to transport ~ one mole of impurities per second for purification, assuming ~ 1 % impurity level, the circulating Li purification loop would be ~ 1 liter/sec. It should be noted that the IFMIF technology has been developing a LL purification loop [24]. It might be also possible to separate hydrogenic species from LL through thermal decomposition at temperatures well below the melting point of pure lithium-hydride (~960°K) [25].

Since LL is a liquid metal, it encounters electromagnetic forces if it tries to flow across the strong fusion reactor magnetic field. A simple estimate indicates that it would likely to take prohibitively large amount of power to move a sufficient amount of LL to carry out the divertor heat load to an outside heat exchanger. For RLLD however, since it is possible to spread the divertor heat load through radiation, the divertor heat load can be then removed through secondary divertor structures. Such structures can be cooled, for example, by circulating high-pressure helium gas system or super-critical CO_2 [26] which is considered a relatively safe approach for LL. Since the amount of circulating LL is relatively modest (~ 1 liter per sec) such a system should only require a negligible amount of power for the LL circulation.

The longer term Li corrosion issue is something we need to investigate in the future. This area has also been studied by IFMIF [27] and in fusion blanket R&D activities where a large amount of LL or its compounds are expected to be used. The LITER stainless steel containers operated for multiple years at high temperature ~ 600 °C without any corrosion or structural issues. The CPS systems also have operated over similar time scales without operational issues. Li is also known to be compatible with refractory metals such as molybdenum and tungsten. Li could chemically react with materials such as copper, aluminum, and ceramics. It is therefore important to design a system to segregate LL from materials that could be corroded by LL.

The safety issue of a large quantity of flowing LL is naturally being address by the IFMIF group, since the IFMIF facility handles large amount (~ a few tons) of flowing LL. An effective safety barrier using inert gas such as argon is being considered for IFMIF. While the amount of LL used in RLLD is likely to be much less, it would be logical to take advantage of the R&D performed by the IFMIF group on this topic.

3.4. Advantages of Closed RLLD Configuration

There are a number of concerns and questions regarding the usage of LL in tokamak reactors due to potential electro-magnetic disruptive force issues. Having a closed divertor configuration has the following additional attractive features for the RLLD concept:

1. Having been placed within the closed divertor chamber, the LL within the RLLD can avoid very fast disruptive electro-magnetic forces as the closed chamber could provide some degree of electro-magnetic shielding from fast disruptive events. Even if the LL is "splashed", it can be largely contained within the chamber. The relatively modest amount of LL used in RLLD should minimize excessive splashing of LL due to the strong surface tension of LL.

2 The closed RLLD chamber could provide some degree of particle partition from the main plasma chamber due to the strong divertor Li retention. The in-flowing plasma from the main chamber into the divertor chamber also tends to push back the Li outflow from the divertor chamber via the force of friction. While the main plasma is likely to tolerate a considerable amount of Li as indicated by the experiment [15], it would be advantageous to be able to control and limit the Li transport out of the divertor chamber from the particle inventory point of view particularly for tritium.

3 The closed RLLD chamber with a high population of Li ions within the chamber could be used as a radiative chamber to reduce the peak divertor heat load and spread the heat load throughout the divertor chamber, where the heat can be removed more readily by much larger volume/surface area heat exchanger structures (see Fig. 5).

4 The vaporized and ionized Li can be collected within the RLLD chamber, and the entire divertor chamber wall coated with LL can function as a strongly pumping chamber consistent with the attainment of the ultra-low recycling regime for advanced high performance plasma operations (see Fig. 4).

4. Discussions and Conclusions

The application of Li coating in NSTX has produced significant improvement in plasma confinement and performance improvements. Additionally, Li was shown to broaden the plasma pressure profile, which is advantageous for achieving high performance H-mode operation for tokamak reactors. It is also noted that even with significant application of Li on PFCs (e.g., up to 1000 grams of Li evaporated in NSTX), no adverse effects on plasma operations were evident. Indeed, very little contamination (< 0.1%) of Li in the main fusion plasma core was observed even in the H-mode plasmas. Another important observation in NSTX is that the application of Li coating on divertor surfaces resulted in 50% reduction in the divertor heat load. Li therefore appears to be a highly desirable PFC material for improving magnetic fusion plasma performance and operations.

In this paper, we propose a closed radiative mantled based liquid lithium divertor concept (RLLD) in an attempt to solve the highly challenging divertor heat load problem for fusion reactors. Because of the low melting temperature of ~ 180°C, Li can be readily kept in a liquid state in a fusion reactor environment. By placing LL surface in the path of main divertor heat flux (i.e., the divertor strike point), the LL will be evaporated from the surface. The evaporated Li is quickly ionized by the plasma and the ionized Li ions can radiate strongly, reducing the heat flux to the divertor strike point surfaces and protecting the substrate material. The Li condensed on the divertor chamber in liquid form could provide strong pumping and it is eventually collected and purified, and is then fed back into the divertor strike point region for divertor heat flux handling. In Figure 6, a conceptual radiative LLD prototype that could be tested in NSTX-U is shown [28]. NSTX-U can provide very high divertor heat flux (~ 40 - 60 MW/m^2), comparable to that is expected in future tokamak reactors. Finally, it should be emphasized that Li PFC applications are quite flexible and diverse. There are other divertor configurations that could greatly reduce the heat flux at the divertor strike point through expanding the divertor flux lines [29, 30]. Li application should be quite compatible with various divertor geometry and magnetic confinement configurations providing the same benefits of Li. Application of Li may also be considered for protecting the tungsten based solid PFC surfaces such as the ones for ITER, as long as a means to purity/refresh LL can be provided. In summary, a radiative mantle based LL divertor solution have the exciting prospect of providing a cost effect flexible means to improve the fusion reactor performance, while providing a practical solution to the highly challenging divertor heat handling issue confronting steady-state magnetic fusion reactors.

Acknowledgments: This work was supported by DoE Contract No. DE-AC02-09CH11466.

References:

[1] ONO, M., KAYE, S. M., PENG, Y.-K. M., , et al., *Nuclear Fusion* **40**, 557 (2000).

[2] KUGEL, H.W., et al., Fusion Eng. Des. (2011), doi:10.1016/j.fusengdes.2011.07.010.

[3] Magnetic Fusion Energy Science Research Needs Workshop report, June 2009.

[4] NYGREN, et al., *J. Nucl. Materials* (2011), doi:10.1016/j.jnucmat.2010.12.289.

[5] ITER Physics Basis, Nucl. Fusion **39,** 2175 (1999).

[6] MENARD, J. E., GERHARDT, S., et al., *Nucl. Fusion* **52**, 083015 (2012).

[7] PENG, Y-K. M., A. J., et al., *Fusion Sci. Technol*. **56**, 957 (2009).

[8] KUGEL, H.W., BELL, M.G., J.-W. AHN, et al., *Physics of Plasmas* **15**, 056118 (2008).

[9] BELL, M.G., et al., *Plasma Phys. Control. Fus.* **51** 124054 (2009).

[10] MAINGI, R., BOYLE, D.P., CANIK, J.M., et. al., *Nucl. Fusion* **52**, 1823 (2011).

[11] S.M. KAYE, et al., *IAEA Fusion Energy Conference* EXD/7-1 (2012).

[12] KAYE, S.M., MAINGI, R., BATTAGLIA, D., et al., *Nucl. Fusion* **51**, 083001 (2012).

[13] MAINGI, R., OSBORNE, T.H., et al., *Phys. Rev. Lett*. **103**, 075001 (2009).

[14] MAINGI, R., BELL, R.E., CANIK, J.M., et al., *Phys. Rev. Lett*. 105, 135004 (2010).

[15] PODEST`A, M., BELL, R.E., et al., *Nuclear Fusion*, **52**, 033008 (2012).

[16] GRAY, T.K. MAINGI, R., et al., *IAEA Fusion Energy Conference* EXD/P5-27 (2012).

[17] ROGNLIEN, T.D., and RENSINK, M.E., Physics of Plasmas **9**, 2120 (2002).

[18] MIRNOV, S.V., et al., Plasma Phys. Control. Fusion **48** 821 (2006).

[19] ONO, M., Bell, M.G., Bell R.E., Kugel, H.W., et al., *Fusion Eng. Des*. 85, 882 (2010).

[20] ONO, M., BELL, M.G., HIROOKA, Y., et al., Nuclear Fusion **52**, 037001 (2012).

[21] HIROOKA, Y, MAZZITELLI, G., et al., *Nucl. Fusion* 50, 077001 (2010).

[22] STANGEBY, P.C., "The plasma boundary of magnetic fusion devices", 2002 (Bristol:IOP).

[23] S.J. ZINKLE and N.M. GHONIEM, *Fusion Eng. Des.* **51,** 55 (2000).

[24] KATEKARI, K., et al., *J. of Energy and Power Engineering*, **6**, 900 (2012)

[25] BALDWIN, M.J., DOERNER, et al., *J. Nucl. Materials* **306**, 15 (2012).

[26] DOSTAL, V., DRISCOL, M.J., and HEJZLAR, P., MIT-ANP-TR-100, March, 2004.

[27] http://en.wikipedia.org/wiki/International_Fusion_Materials_Irradiation_Facility.

[28] ONO, M., et al., *Fusion Eng. Des*. (2011), doi:10.1016/j.fusengdes.2011.10.011.

[29] SOUKHANOVSKII, V.A., J-W AHN, J-W., et al., *Nuclear Fusion* **51**, 012001 (2011).

[30] VALANJU, P.M., *Phys. of Plasmas* **16,** 056110 (2009).

The Princeton Plasma Physics Laboratory is operated
by Princeton University under contract
with the U.S. Department of Energy.

Information Services
Princeton Plasma Physics Laboratory
P.O. Box 451
Princeton, NJ 08543

Phone: 609-243-2245
Fax: 609-243-2751
e-mail: pppl_info@pppl.gov
Internet Address: http://www.pppl.gov