Princeton Plasma Physics Laboratory

PPPL- 5059

PPPL-5059

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AUGUST, 2014

PRINCETON PLASMA PHYSICS LABORATORY

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

Addressing the Challenges of Plasma-Surface Interactions in NSTX-U*

R. Kaita, T. Abrams, M. Jaworski, M. Lucia,
J. Nichols, C. H. Skinner, D. Stotler,
and the NSTX-U Team
Princeton Plasma Physics Laboratory
Princeton, NJ, USA

J-P. Alain and F. Bedoya
Department of Nuclear, Plasma, and Radiological Engineering
University of Illinois at Urbana-Champaign
Urbana, IL, USA

***Abstract*— The importance of conditioning plasma-facing components (PFCs) has long been recognized as a critical element in obtaining high-performance plasmas in magnetic confinement devices. Lithium coatings, for example, have been used for decades for conditioning PFCs. Since the initial studies on the Tokamak Fusion Test Reactor, experiments on devices with different aspect ratios and magnetic geometries like the National Spherical Torus Experiment (NSTX) continue to show the relationship between lithium PFCs and good confinement and stability. While such results are promising, their empirical nature do not reflect the detailed relationship between PFCs and the dynamic conditions that occur in the tokamak environment. A first step developing an understanding such complexity will be taken in the upgrade to NSTX (NSTX-U) that is nearing completion. New measurement capabilities include the Materials Analysis and Particle Probe (MAPP) for *in situ* surface analysis of samples exposed to tokamak plasmas. The OEDGE suite of codes, for example, will provide a new way to model the underlying mechanisms for such material migration in NSTX-U. This will lead to a better understanding of how plasma-facing surfaces evolve during a shot, and how the composition of the plasma facing surface influences the discharge performance we observe. This paper will provide an overview of these capabilities, and highlight their importance for NSTX-U plans to transition from carbon to high-Z PFCs.**

Keywords—lithium, plasma confinement, magnetic confinement, materials science and technology

I. Introduction

The conditioning of plasma-facing components (PFCs) has long been recognized as key to achieving high-performance plasmas in magnetic confinement devices. The efficacy of lithium coatings, for example, has been well established for PFC conditioning. It has been demonstrated not only across devices of different sizes, but also PFCs.

Among the earliest results with PFC conditioning using lithium were from the Tokamak Fusion Test Reactor (TFTR). A combination of techniques were used, including the injection of lithium pellets, ablation of lithium from a crucible inside the TFTR vacuum vessel with a high-power laser, and "spreading" lithium on the PFCs by operating successively larger plasmas ("painting"). An large enhancement of the fusion "triple product" (density x confinement time x temperature) was obtained (Fig. 1), with the highest stored energy ever achieved in TFTR.[1]

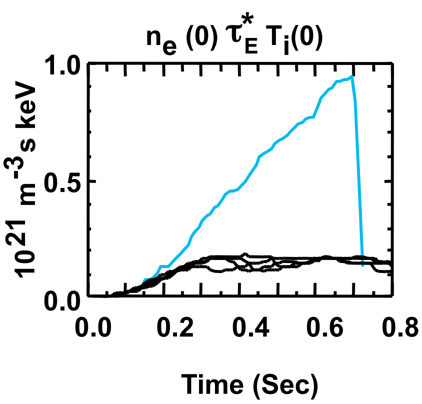

Fig. 1 Comparison of TFTR discharges with (upper curve) and without lithium PFC conditioning (lower curves)

The lithium PFC conditioning experiments were among the last conducted on TFTR, which was the largest fusion device in the US at that time. It had a major radius of 2.52 m and a minor radius of 0.87 m. It also had a maximum beam power of 39.5 MW, injecting into a vacuum vessel with PFC that were entirely carbon.

In contrast, the Current Drive Experiment-Upgrade (CDX-U) was a much more modest device, with a major radius of 0.34 m and a minor radius of 0.22 m. It also only had Ohmic heating for plasmas surrounded by stainless steel PFCs. The lithium in this device was in liquid form in a fully-toroidal "tray" limiter, centered at the 0.34 m major radius and having a width of 0.1 m. When heated to 350 C, the lithium from the tray also evaporated to coat about 50% of the plasma-contacting area. Under these conditions, the measured confinement times exceeded expectations from ITER98P(y,1) by a factor of 2 two to 3, and represented the largest increase in energy confinement ever observed for an Ohmic tokamak plasma (Fig. 2).[2]

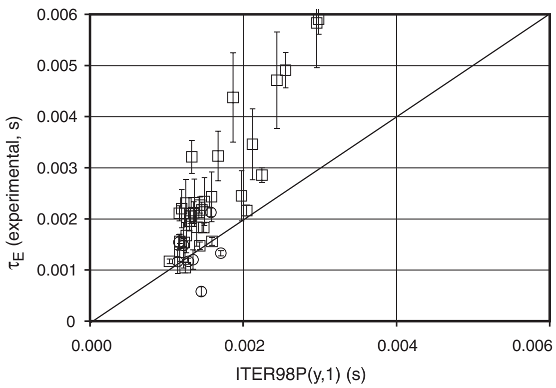

Fig. 2 Experimental energy confinement times compared to values expected from ITER98P(y,1) confinement scaling. Plasmas with passivated lithium PFCs (circles) have lower confinement times than discharges with active lithium PFCs (squares.)

Improvement in plasma performance was also obtained in the National Spherical Torus Experiment (NSTX) with lithium surface conditioning. Like TFTR, NSTX PFCs are almost entirely carbon. It has a much smaller major radius (0.85 m). As a spherical torus or tokamak (ST), however, it has a comparable minor radius (0.65), and a similar plasma cross sectional area if the elongation (1.7 to 2.7) is considered. It has neutral beam heating as well, with a maximum injected power of over 7 MW. Unlike either TFTR or CDX-U, however, NSTX is a divertor tokamak.

The main method of lithium PFC conditioning on NSTX involved the evaporation from two LIThium EvaporatoRs (LITERs). These were mounted at two locations on the upper dome of NSTX, and aimed toward the lower divertor region. By located the LITERs approximately 180 degrees apart, full toroidal coverage of the lower divertor is possible.

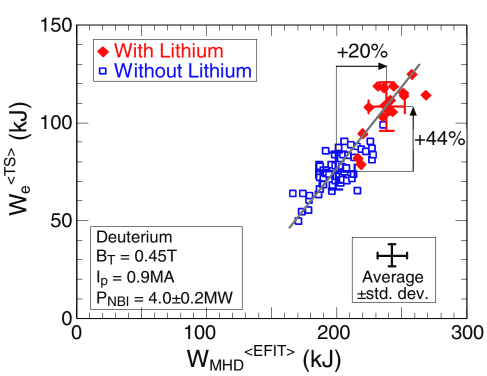

Fig. 3 Plot showing increase in total stored energy for similar discharges in the presence of a lower divertor after lithium evaporation.

The salient result with lithium evaporation on NSTX was the increase in stored energy, as shown in Fig. 3. It compares the total stored energy from MHD equilibrium (EFIT) analysis with the electron stored energy from volume integration of measurements of electron density and temperature (from the Thomson scattering diagnostic) for similar discharges with and without lithium evaporation onto the lower divertor.[3] The increase in the electron stored energy is particular significant in its implications for lithium coatings as a means to reduce anomalous electron transport.

Lithium PFC conditioning thus appears to be effective in improving discharge performance across a broad range of plasma devices. The three representative machines described above include a large, conventional aspect ratio tokamak with carbon PFCs, and two ST that span size, PFC type, and magnetic configuration. This leads to the expectation that lithium conditioning will be effective with high-Z PFCs in NSTX-U, and the challenges related to their implementation are discussed in Section II. The development of novel diagnostics and new modeling capabilities that are needed to go beyond the empirical observations of the relationship between lithium PFC conditioning and plasma performance are described as well.

Experiments with lithium PFCs are also consistent with the conclusion that the chemical reactivity of the lithium is more critical to its effectiveness than the particular substrate on which they are placed. Techniques like evaporation are not suitable for replenishing lithium surfaces during long discharges, and this motivates exploring the feasibility flowing liquid lithium PFCs. This is part of the long-term NSTX-U PFC program, and efforts in prototyping concepts are discussed in Section III.

II. Challenges for High-Z PFCs in NSTX-U

Present plans for high-Z PFCs on NSTX-U are to use the molybdenum alloy TZM in the divertor. Its constituents are titanium (0.50%), zirconium (0.07-0.08%) and carbon (0.02-0.05%), with the remainder consisting of molybdenum. For PFC applications, the attractive properties of TZM include good thermal conductivity, low vapor pressure and ease of machining. A toroidal row of tiles with a TZM surface was originally installed in the divertor of NSTX.[4] A maximum power flux of 3.63 MW/m^2 was assumed, based on NSTX-U design point for a “double-null divertor” plasma. Under these conditions, the goal was to keep the peak TZM temperature below 1000 C to avoid embrittlement from recrystallization, and the cyclical stress below 300 MPa to avoid low-cycle fatigue.

Because of cost and schedule constraints, the rapid fabrication of TZM PFC tiles by first removing 10 mm from the plasma-facing side of existing carbon tiles. A TZM plate of equivalent thickness was then attached to form the PFC. This approach allowed the design goal to be met, as long as the maximum pulse length was kept under 2 s and there was a minimum of ten minutes between shots. No active cooling of the carbon tiles was assumed in the analysis. The resulting geometry is illustrated in Fig. 4.

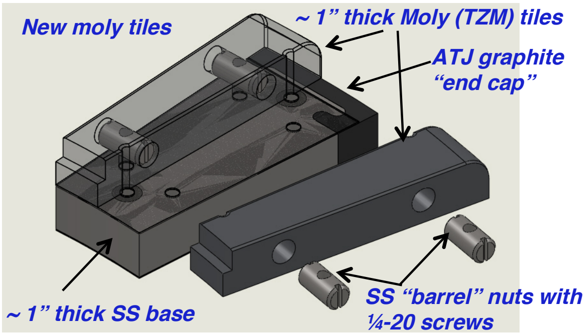

Fig. 4 Scheme for PFC using TZM plate attached to stainless steel (SS) base. ATJ graphite "end cap" provides shield for SS base in CHI gap.

A photograph of the row of tiles as installed on the NSTX center stack is shown in Fig. 5. The NSTX divertor region is separated into inboard (IBD) and outboard (OBD) sections by the gap required for coaxial helicity injection (CHI). The TZM tiles are at the edge of the inboard divertor, and retain the "bullnose" feature of the original graphite tiles to protect the sides of the CHI gap.

Fig. 5 Photograph showing row of TZM PFC tiles on inboard lower divertor in NSTX. Gap for CHI can be seen to the left of the tiles.

Machining existing graphite tiles and attaching TZM plates is an economical way of converting from carbon to high-Z PFCs, and a similar scheme based on coatings has been used in ASDEX-U. This is not suitable in the long term, however, as the NSTX-U design point includes power fluxes approaching 7 MW/m^2 for single-null plasmas, and pulse lengths in the 7 to 10 s range. An alternative to PFCs where large areas are exposed directly to the plasmas is an approach that uses a castellated surface. In Alcator C-Mod, for example, each tile in the region of the divertor strike point was made up of eight small tungsten plates or lamellae, each 4 mm thick.[5] To prototype tiles for the ITER divertor, the JET version has lamellae that are 6 mm thick, for power fluxes up to 7 MW/m^2 for 10 s.[6]

While existing lamellae designs thus satisfy the NSTX-U divertor power flux handling requirements, challenges remain that are related to the phased implementation of high-Z PFCs. Present plans call for a single row, or at most a few rows, of TZM tiles to be installed initially in the OBD region of NSTX-U. The choice of location is conservative, in that the strike points of the highest performance plasmas will be in the IBD region. This means, however, that not only the IBD tiles, but most of the NSTX-U PFCs will remain carbon. Erosion and redeposition of carbon has been an issue in NSTX, and are also expected in NSTX-U.[7] This would result in mixed materials at the location of the TZM tiles, and will make the assessment of high-Z PFCs difficult.

The problem of mixed materials might be recognized, but the details of how plasma-surface interactions distributed them around the PFCs in NSTX were not known. This issue will be addressed in NSTX-U with improved modeling of material migration. The OEDGE suite of codes, which couples a 1D ("onion skin model") plasma fluid code with simulations of neutrals and impurities, will be the basis of interpretive modeling with input from additional diagnostics.[8] Key among them is the Materials Analysis and Particle Probe (MAPP).[9]

The MAPP is a system that enables *in situ* characterization of tokamak PFCs. It allows the insertion of up to four samples into the plasma chamber. After exposure to a discharge, it is possible to withdraw the samples into an analysis chamber without breaking vacuum. Fig. 6 shows the chamber on LTX, where MAPP is being tested prior to installation on NSTX-U.

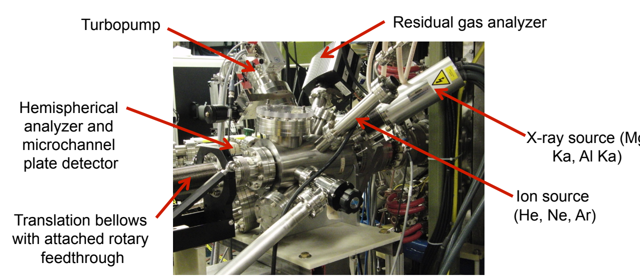

Fig. 6 MAPP analysis chamber installed on LTX

The surface properties of the samples can be determined with a variety of techniques, including x-ray photoelectron spectroscopy (XPS), low-energy ion scattering spectroscopy (LEISS), direct recoil spectroscopy (DRS), and thermal desorption spectroscopy (TDS). Because the samples have separated heaters, they can be analyzed individually with TDS.[10,11] On NSTX-U, the MAPP samples will be inserted through a gap in the PFC tiles in the OBD region. Because the samples are close in major radius to the row of TZM tiles initially planned for NSTX-U, they are expected to provide data on how the tile surfaces evolve as a function of time. The information on erosion, redeposition, and material migration obtained with MAPP should provide useful input for modeling the characteristics of future TZM PFCs as NSTX-U proceeds with their implementation.

Conditioning techniques also introduce complexities. Carbon continues to be a common PFC material in present-day tokamaks. The most common approaches to reduce impurities prior to plasma operations include high-temperature PFC bakeout and glow discharge cleaning (GDC). As observed in other tokamaks, GDC with a mixture of helium and boron (in the form of duterated trimethyl boron), or "boronization," was effective in reducing oxygen in NSTX.[12] During the last years of NSTX operations, evaporating lithium on PFCs has been demonstrated as an effective surface conditioning technique.[13] Direct evaporation of lithium ("lithiumization") was much more efficient in creating a PFC film than boronization, and substantially reduced the time needed lower impurities to levels that allowed plasma operations.

The challenge lithiumization poses for lamellae PFCs is its potential for filling the spaces between the plates. This has the potential of reducing the surface area advantage of the lamellae concept for power handling. On the other hand, the lithium itself might be used to mitigate the effects of high power densities. Sputtering and evaporation could create a lithium vapor cloud in the scrapeoff layer (SOL), and provide radiative cooling ("vapor shielding"). There may be evidence that the conditions for this to occur were already achieved in the SOL in the vicinity of the NSTX Liquid Lithium Divertor (LLD), and they are expected to exist in NSTX-U. A lamellae approach for handing even higher power densities might actually require introducing more lithium into the structure, where liquid lithium would be drawn toward the plasma-facing surface by capillary action from a reservoir at its base.[14]

III. Challenges for Flowing Liquid Lithium System in NSTX-U

The ability to feed PFC structures with liquid lithium, as mentioned in the previous section, is not only promising for mitigating the effects of high power loads. The value of lithium as a low recycling PFC has also been demonstrated as a means of improving confinement in a variety of fusion devices. For long-pulse applications, however, an efficient means of maintaining the chemical reactivity ("active surface") required of the lithium remains a challenge.

A variety of approaches have already been developed for creating liquid lithium PFCs. Liquid lithium has been introduced from a reservoir into a porous "mesh" that served as a toroidally-local limiter surface.[15,16] In such "capillary porous system" (CPS) concepts, capillary action replenishes the lithium that ablates from the PFC surface. A reservoir external to the vacuum vessel was used to fill a fully-toroidal "tray" with liquid lithium to form a limiter for CDX-U plasmas.[2,17]

The goal of the NSTX Liquid Lithium Divertor (LLD) was to extend the applicability of a liquid lithium PFC for lowering recycling to a divertor configuration. As with the CDX-U lithium tray limiter, the LLD was fully-toroidal. The original concept was not to create a large liquid lithium free surface as on CDX-U, but fill a structure created by chemical vapor deposition (CVD) of a refractory metal on substrate mesh (CVD mesh). This would restrain the lithium against MHD-induced body forces arising from the currents flowing through the LLD.

When time constraints prevented the development of the CVD mesh and a suitable liquid lithium filling method, an alternative similar to the close-fitting conducting shell in LTX was chosen.[18,19] As with the LTX shell, the bulk of the material was copper. Instead of a dynamically ("explosively") bonding a stainless steel liner to protect the copper from the lithium in LTX, a much thinner (0.25 mm) liner was brazed to a 2.2 cm copper substrate. This insured that the mass of the copper determined the thermal response of the LLD, and this was demonstrated in tests of LLD samples under high heat loads.[20] To retain the liquid lithium on the LLD surface, a porous molybdenum layer approximately 0.15 mm thick was plasma-sprayed onto the stainless steel liner. No lithium ejection during was observed during NSTX plasma operations, consistent with MHD stability analysis for the pore size of the LLD surface.[14]

In each of these systems, however, the lithium was not circulated in any form of "closed loop." An active surface is maintained by capillary action in a CPS, but the evaporated lithium accumulates on the PFCs and is not recoverable. Evaporation of lithium onto the LLD was required between discharges to create an active surface, and this process puts a limit on how long such a surface can be maintained. Again, the lithium accumulates on the LLD surface, and is not recoverable.

The limitations of existing liquid lithium PFCs motivate the development of a flowing liquid lithium system (FLLS), and the long-term plans for NSTX-U include the implementation of a flowing liquid lithium divertor. Such concepts, however, present new and difficult challenges. Liquid lithium propulsion in active systems has tended to involve mechanical devices like impellers. Their location within the flow path makes them subject to corrosion and are difficult to maintain. The options are limited, however, if heat transfer through conduction is the main focus and the high pressures associated with fast flows are needed.[21]

More recently, slow flow alternatives are being investigated. For example, the creation of a lithium vapor cloud in the SOL means that the power handling is primarily through the vapor shielding it provides. In that case, the required lithium flow rate drops by an order of magnitude, as it would just need to be sufficient to maintain an active surface and replace any evaporated lithium. This enables the use of electromagnetic induction pumps (EMPs), where a set of rotating magnets drives the lithium within a coiled tube surrounding it. There are no mechanical parts inside the fluid path, making them readily serviceable.[22]

Operation the FLLS at lower pressures potentially reduces safety risks due to the consequences liquid lithium leaks. The effects of leaks in any flowing liquid lithium system need to be mitigated, however, and ways to accomplish this are being developed in a prototype liquid lithium loop (LLL) at PPPL. This is shown schematically in Fig. 7. It includes an EMP to circulate liquid lithium, and is nearing completion. The locations most prone to leaks are the joints in the tubes that make up the flow path. A copper "clamshell" surrounds each joint, and the resistance between the each clamshell and the tube is monitored. A leak would cause this resistance to drop rapidly, and power to the EMP and heaters that keep the liquefied would shut off.[23]

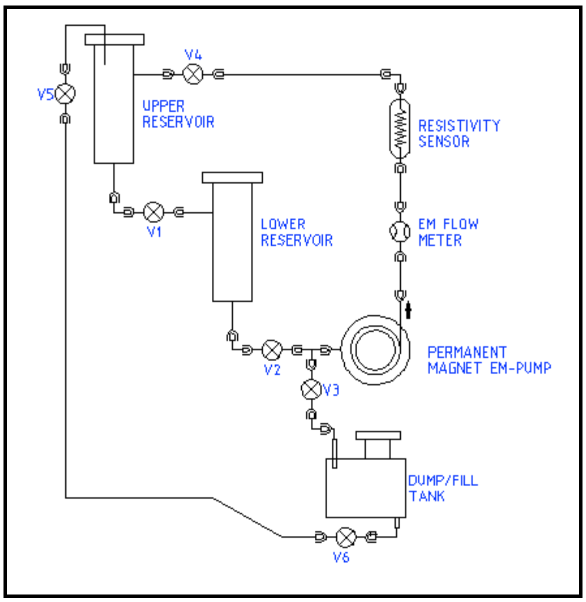

Fig. 7 Schematic of prototype liquid lithium loop with permanent magnet electromagnetic induction pump

Each LLL is intended to drive the liquid lithium flow in a toroidal divertor segment of a future FLLS. This approach simplifies fabrication and maintenance, and allows for phased implementation in NSTX-U. Because the divertor is modular, different designs can also be tested simultaneously by installing a variety of segment types.

IV. Summary

The NSTX-U PFC research program builds on the extensive experience in the fusion community with surface conditioning techniques, particularly with lithium. It also draws heavily on work related to high-Z PFCs, motivated by their use in ITER and future fusion devices instead of the carbon PFCs still common at present.

Among the main challenges will be in understanding the behavior of high-Z PFCs with lithium conditioning. To this end, data from unique diagnostics such as MAPP for *in situ* PFC sample analysis will be combined with the application of new interpretive modeling tools.

Promising approaches for a flowing liquid lithium system are under development for eventual used in the NSTX-U divertor. The main challenges include safety and reliability, and prototype liquid lithium loop to address them is nearing completion.

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*This work supported in part by USDOE contracts DE-AC02-09CH11466 and DE-SC0010717.

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(DE-AC02-09CH11466)

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