# Addressing the Challenges of Plasma–Surface Interactions in NSTX-U

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***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 (TFTR), experiments on devices with different aspect ratios and magnetic geometries like the National Spherical Torus Experiment (NSTX) continue to show the relationship between the lithium PFCs and good confinement and stability. While such results are promising, their empirical nature do not reflect the detailed relationship between the 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, or the National Spherical Torus Experiment-Upgrade (NSTX-U) that is nearing completion. New measurement capabilities include the materials analysis and particle probe for *in situ* surface analysis of samples exposed to tokamak plasmas. The onion-skin modeling for edge analysis (OEDGE) suite of codes, for example, will be used 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.**

***Index Terms*— Lithium, magnetic confinement, materials science and technology, plasma confinement.**

## 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 differing PFCs.

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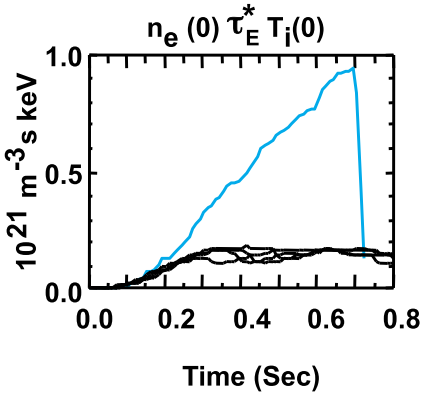


Fig. 1. Comparison of TFTR discharges with (top curve) and without lithium PFC conditioning (bottom curves). Reprinted with permission from [1].

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 × confinement time × temperature) was obtained (Fig. 1), with the highest stored energy ever achieved in TFTR [1].

The lithium PFC conditioning experiments were among the last conducted on TFTR, which was the largest fusion device in the United States 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, heating plasmas in 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 (SS) 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) scaling by a factor of two to three, and represented the largest increase in energy confinement ever observed for an ohmic tokamak plasma (Fig. 2) [2]. The CDX-U confinement results are compared with

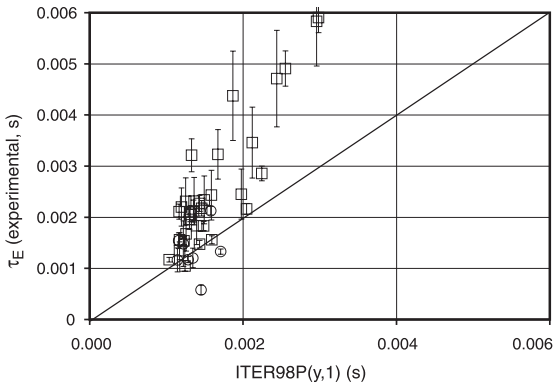


Fig. 2. Experimental energy confinement times from CDX-U compared with values expected from ITER98P($y$, 1) confinement scaling. Plasmas with passivated lithium PFCs (circles) have lower confinement times than discharges with active lithium PFCs (squares). Reprinted with permission from [2].

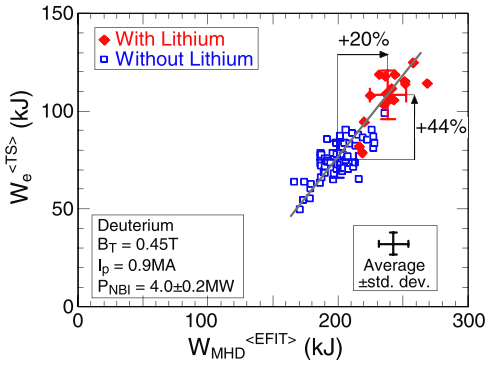


Fig. 3. Plot showing increase in total stored energy for similar discharges with a lower divertor coated by lithium evaporation. Reprinted with permission from [3].

ITER98P($y$, 1) scaling because it is the first to include data from Small Tight Aspect Ratio Tokamak (START), a low aspect ratio tokamak similar in size to CDX-U [2].

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, however, it has a comparable minor radius (0.65), and a similar plasma cross-sectional area if the elongation (1.7–2.7) 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 locating the LITERs approximately 180° apart, full toroidal coverage of the lower divertor is possible.

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 fitting (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 standard deviation is at the center of the cluster of points for discharges with and without lithium-coated PFCS. The uncertainties they represent fall within the range of the data under each condition, so the measurements have sufficient accuracy to show the trend of increased stored energy with lithium coatings. The increase in the electron stored energy is particularly 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 the 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 STs 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 the limitations of such approaches motivate 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 titanium-zirconium-molybdenum (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 an 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, a method for the rapid fabrication of TZM PFC tiles was chosen. This began 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 10 min between the shots. No active cooling of the carbon tiles was assumed in the analysis. The resulting geometry is shown in Fig. 4.

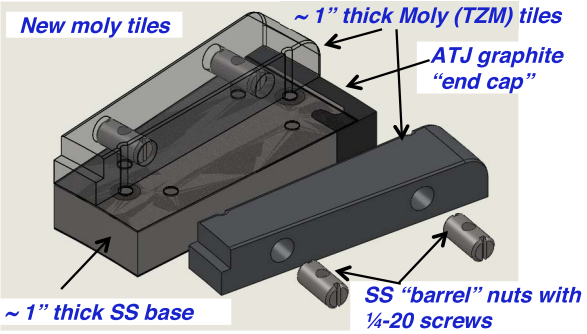


Fig. 4. Scheme for PFC using TZM plate attached to SS base. ATJ graphite end cap provides shield for SS base in CHI gap.

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

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 IBD divertor, and retain the bullnose feature of the original graphite tiles to protect the sides of the CHI gap.

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 the Axisymmetric Divertor Experiment-Upgrade (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 7–10-s range. An alternative to PFCs where large areas are exposed directly to the plasmas is an approach that uses a castellated surface. In the Alto Campo Torus C-Modification (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 International Thermonuclear Experimental Reactor (ITER) divertor, the Joint European Torus (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.

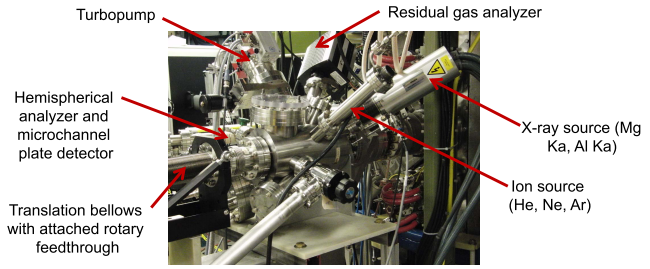


Fig. 6. MAPP analysis chamber installed on LTX.

The extent to which plasma–surface interactions distribute materials around the interior of NSTX has been inferred from analysis of PFCs at the conclusion of an operational period. Knowledge of the mechanism behind material migration, however, is limited. This issue will be addressed in NSTX-U with improved interpretive modeling using the OEDGE suite of codes. They couple a 1-D (onion skin model) plasma fluid code with simulations of neutrals and impurities [8]. Input from diagnostics for the simulations will include data from 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 the Lithium Tokamak Experiment (LTX), where MAPP is being tested prior to installation on NSTX-U.

The surface properties of the samples can be determined with a variety of techniques, including X-ray photoelectron spectroscopy, low-energy ion scattering spectroscopy, direct recoil spectroscopy, and thermal desorption spectroscopy (TDS). Because the samples have separate 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 deuterated 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 to 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 could reduce 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

Liquid lithium PFCs are 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 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 a chemical vapor deposited (CVD) 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 dynamically (explosively) bonding an SS 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 SS liner. No lithium ejection 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.

Experiments on NSTX appeared to indicate that the surface chemistry of static lithium PFCs determined plasma performance, rather than whether the substrate was graphite or plasma-sprayed molybdenum [21]. Detailed analysis of graphite NSTX PFC surfaces after exposure to lithium revealed chemical complexes involving lithium, oxygen, and deuterium [22]. Furthermore, examination of metallic PFCs exposed to lithium demonstrated lithium oxide formation [11]. Quantum-classical molecular dynamics modeling has shown that oxygen on PFC surfaces dominates the bonding of incident deuterium [23]. The application of lithium is believed to improve plasma performance by reducing wall recycling. This is due to the retention of deuterium in PFCs, and these results suggest that the underlying mechanism may be the same for static lithium PFC layers regardless of substrate.

The difficulty arises when PFC temperatures exceed 400 K, which is well above room temperature but below the melting point of lithium. At such temperatures, a PFC surface layer of lithium oxide releases deuterium [24], so lithium deuteride formation in the bulk liquid lithium becomes important in binding deuterium [25]. This then becomes a significant motivation for the development of a flowing liquid lithium system (FLLS), to insure that the properties of a static lithium oxide layer do not govern deuterium retention. The long-term plans for NSTX-U include the implementation of a flowing liquid lithium divertor, but such concepts 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 [26].

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. Lithium films on metallic substrates, for example, were subject to high-flux deuterium plasma bombardment on the Magnum-PSI (plasma-surface interactions) linear plasma device to create a deuterium–lithium mixed material surface. High lithium redeposition fractions were observed, as manifest in a vapor cloud that persisted long after models based on lithium sputtering alone would predict the film to erode away [27].

With vapor shielding, the required lithium flow rate drops by an order of magnitude, as it would just need to be sufficient

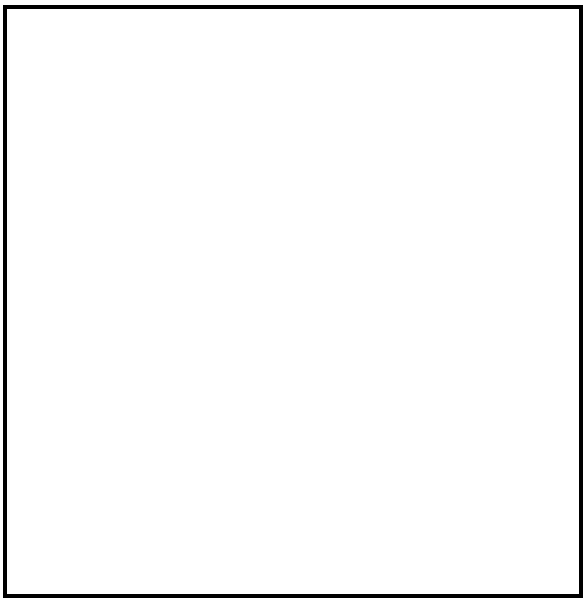

Fig. 7. Schematic of prototype LLL with permanent magnet EMP.

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 [28].

Lower flow rates mean that the pressure at which the FLLS needs to operate can be reduced. This improves safety by lowering the fluid outflow should any liquid lithium leaks occur. The effects of any leaks in a 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 Princeton Plasma Physics Laboratory (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. This would trigger the fault detection circuit which would shut off power to the EMP and the heaters that keep the lithium liquefied [29].

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. Conclusion

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 an FLLS are under development for eventual use in the NSTX-U divertor. Unlike the NSTX LLD, the FLLS does not require the PFC surface to be loaded by lithium evaporation. The reduction in erosion provided by vapor shielding also limits material migration. The FLLS can thus potentially minimize the contamination of non-PFC surfaces with lithium. The main challenges include safety and reliability, and a prototype LLL to address them is nearing completion.

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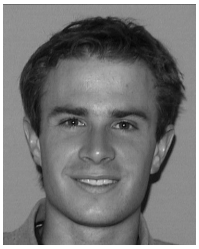
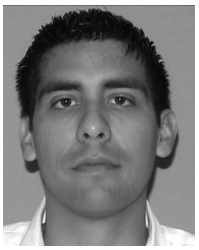
**Robert Kaita** has done extensive work on the development of a wide variety of diagnostics for high-temperature plasmas. He is currently the Head of Boundary Physics Operations on the National Spherical Torus Experiment-Upgrade and a Co-Principal Investigator on the Lithium Tokamak Experiment with the Princeton Plasma Physics Laboratory, Princeton, NJ, USA. His focus is also on plasma-surface interactions and the use of liquid metals as first-wall materials for fusion reactors.

Dr. Kaita is a fellow of the American Physics Society. He was a recipient of the Kaul Prize for Excellence in Plasma Physics Research and Technology Development for pioneering work in the use of liquid lithium metal as a renewable wall for fusion devices.



**Tyler Abrams** received the Ph.D. degree from Princeton University, Princeton, NJ, USA.

His dissertation research at the Princeton Plasma Physics Laboratory focused on characterizing the erosion and redeposition rates of graphite and high-Z materials coated with low-Z materials such as lithium and boron. These materials were studied under high-plasma fluxes and at high-surface temperatures. He is currently a Post-Doctoral Research Associate with DIII-D, Boundary Plasma Material Interactions Center at General Atomics in San Diego.

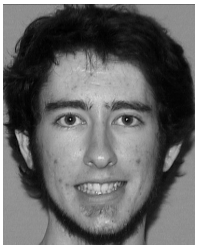


**Michael Jaworski** is currently the Leader of the Materials and Plasma-Facing Components Topical Science Group on the National Spherical Torus Experiment-Upgrade with the Princeton Plasma Physics Laboratory, Princeton, NJ, USA. He is involved in the development of high-Z plasma-facing components and liquid metal plasma-facing components for use in fusion devices. His previous experiments have led to the observation of thermoelectric magnetohydrodynamic effects in liquid lithium systems. His current research interests include physics and engineering aspects of high-temperature liquid lithium plasma-facing components for fusion applications.

**Matthew Lucia** received the B.S. degree in physics and mathematics from the University of Notre Dame, Notre Dame, IN, USA. He is currently pursuing the Ph.D. degree with the Program in Plasma Physics, Princeton University, Princeton, NJ, USA.

His research has focused on plasma-surface interactions and surface analysis in the Lithium Tokamak Experiment at the Princeton Plasma Physics Laboratory, Princeton.

Mr. Lucia is a National Science Foundation Graduate Research Fellow and was a National Defense Science and Engineering Graduate Fellow.



**Jacob H. Nichols** received the B.A. degree in physics from Cornell University, Ithaca, NY, USA, in 2010. He is currently pursuing the Ph.D. degree with the Program in Plasma Physics, Princeton University, Princeton, NJ, USA.

His current research interests include plasma-wall interactions and material migration in tokamaks.

Mr. Nichols received the U.S. DOE Fusion Energy Sciences Fellowship from 2010 to 2013.

**Charles H. Skinner** was involved in tritium retention issues on TFTR and invented and demonstrated a novel laser-based detritiation technique. He also invented and demonstrated a novel electrostatic dust detector in the National Spherical Torus Experiment-Upgrade Tokamak and a novel laser-based method for cleaning first mirrors for ITER. He is involved in application of scanning auger microscopy to investigate wetting by lithium at a nanoscale level. Besides the fundamental interest of understanding wetting on an atomistic scale, this work has practical relevance to the development of liquid metal plasma facing components for managing plasma wall interactions in the extreme environment of high heat and particle fluxes in a magnetic fusion device.

**Daren Stotler** was with Dr. Charles Karney to develop the DEGAS 2 neutral gas transport code that is currently used with the Princeton Plasma Physics Laboratory (PPPL), Princeton, NJ, USA, and other facilities to study how plasma-material interactions and the resulting particles affect current experiments and predict their impact on future devices. As part of PPPLs participation with the Center for Plasma Edge Simulation (CPES) from 2005 to 2011 and current Edge Physics Simulation projects, he has developed a comprehensive neutral particle transport routine based on DEGAS 2 and integrated it into the CPES kinetic plasma transport code. He is currently a Principal Research Physicist with the Department of Theory, PPPL.

**Jean Paul Allain** joined the Department of Nuclear, Plasma, and Radiological Engineering, University of Illinois at Urbana-Champaign, Champaign, IL, USA, as a faculty member, where he is also currently an affiliate faculty member with the Department of Bioengineering and the Micro and Nanotechnology Laboratory. He has authored over 90 papers in both experimental and computational modeling work in the area of particle-surface interactions. His current research interests include developing *in-situ* surface structure and composition evolution characterization of heterogeneous surfaces under low-energy irradiation promoting structure and function at the nanoscale.

Prof. Allain was a recipient of numerous awards, including the DOE Early Career Award in 2010 and the Research Excellence Award in 2011.

**Felipe Bedoya** is currently pursuing the Ph.D. degree with the Department of Nuclear, Plasma, and Radiological Engineering, University of Illinois at Urbana-Champaign, Champaign, IL, USA.

His current research interests include in online characterization of plasma facing components in tokamak devices. This includes the study of materials used in fusion environments using chemical and structural analysis techniques, e.g., X-ray photoelectron spectroscopy, ion scattering spectroscopy, and scanning electron microscopy.

Mr. Bedoya was a recipient of the Francisco Jose de Caldas Scholarship of the Department of Science, Technology and Innovation-Colciencias in Colombia.