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OVERVIEW OF INNOVATIVE PMI RESEARCH ON NSTX-U AND ASSOCIATED PMI FACILITES AT PPPL*

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Developing a reactor compatible divertor and managing the associated plasma material interaction (PMI) has been identified as a high priority research area for magnetic confinement fusion. Accordingly on NSTX-U, the PMI research has received a strong emphasis. With ~ 15 MW of auxiliary heating power, NSTX-U will be able to test the PMI physics with the peak divertor plasma facing component (PFC) heat loads of up to 40-60 MW/m^2. To support the PMI research, a comprehensive set of PMI diagnostic tools are being implemented. The snow-flake configuration can produce exceptionally high divertor flux expansion of up to ~ 50. Combined with the radiative divertor concept, the snow-flake configuration has reduced the divertor heat flux by an order of magnitude in NSTX. Another area of active PMI investigation is the effect of divertor lithium coating (both in solid and liquid phases). The overall NSTX lithium PFC coating results suggest exciting opportunities for future magnetic confinement research including significant electron energy confinement improvements, H-mode power threshold reduction, the control of Edge Localized Modes (ELMs), and high heat flux handling. To support the NSTX-U/PPPL PMI research, there are also a number of associated PMI facilities implemented at PPPL/Princeton University including the Liquid Lithium R&D facility, Lithium Tokamak Experiment, and Laboratories for Materials Characterization and Surface Chemistry.

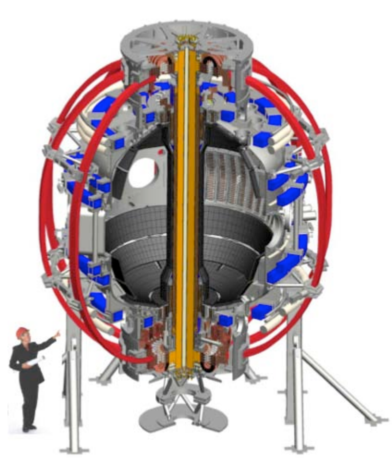

Fig. 1. A schematic view of NSTX-U.

I. INTRODUCTION

Plasma material interaction (PMI) research has been identified as a particularly important scientific problem for magnetic confinement fusion development. ITER's (Ref. 1) tungsten-based divertor tiles, for example, are designed to withstand the required 5 – 10 MW/m^2 heat flux. While tungsten-based material 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) (Ref. 2). In addition, for tokamaks, large ELMs cause significantly higher transient divertor heat loads which can further exacerbate surface deterioration. Looking beyond ITER, the anticipated peak heat flux could increase by another factor of 6 (up to 60 MW/m^2) for a 1 GW-electric-class tokamak power plant. No satisfactory divertor heat flux solution has been thus far developed. In this paper, we give an overview of the NSTX-U PMI research that is focused on developing innovative divertor heat flux solutions for future devices.

II. NSTX-U PMI RESEARCH

The National Spherical Torus Experiment (NSTX) facility has been operational since 2000 (Ref. 3). The mission elements of NSTX are 1. to advance the spherical tokamak (ST) as a candidate concept for the Fusion Nuclear Science Facility (FNSF) (Ref. 4), 2. to develop solutions for the plasma-material interface, 3. to advance toroidal confinement physics for ITER and beyond, and 4. to develop the ST as fusion energy system. During FY2012-2014, major upgrades (NSTX-U) including a new center-stack and a second more tangentially injecting neutral beam injector will be carried out (Ref. 5). A schematic of NSTX-U is shown in Fig. 1. The new center-stack will double the available magnetic field and plasma current while increasing the plasma pulse length from NSTX ~ 1 s at 0.5 T to 5 s at 1 T providing the highest performance ST facility in the world fusion program. The second more tangential neutral beam injector will double the NBI heating power to enable high beta at higher field with improved NBI current drive efficiency and the current profile control needed for achieving fully noninductive operation required for next-step applications. The NSTX-U capabilities will increase the range of variation of the plasma collisionality by up to an order of magnitude with minimum values approaching those expected for the next step STs. These upgrades would therefore provide the database needed to establish confidence in the design of such facilities.

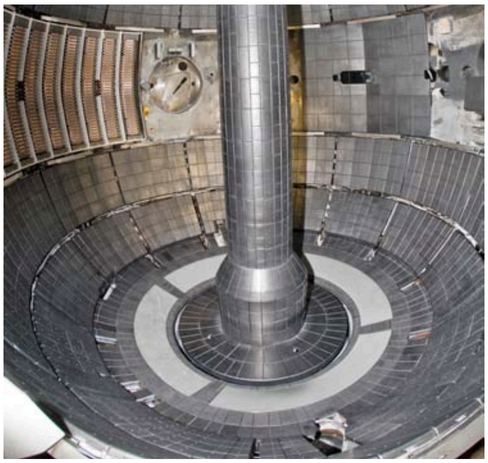

Fig. 2. Interior view of NSTX vacuum vessel. Four - segment LLD plates are shown.

II.A. NSTX-U AS AN INTEGRATED PMI FACILITY

In view of the critical need to develop acceptable divertor solutions for the future facilities, PMI research is pursued on NSTX-U with an exceptionally high priority. The facility will have 10 MW of neutral beam injection (NBI) for 5 sec and 15 MW for 2 sec. In addition ~ 4 MW of high harmonic fast wave (HHFW) for 5 sec is available. The facility also plans to implement up to 4 MW of electron cyclotron heating (ECH) for start-up and current profile control. The interior of the NSTX vacuum vessel is shown in Fig. 2 that also shows four - segment liquid lithium divertor (LLD) plates (Ref. 6). As can be seen from the figure, the low-aspect-ratio configuration allows a direct view of most of the NSTX plasma and surrounding PFCs. This open access for example allows full toroidal views of divertor area for spectroscopic diagnostics as discussed in Sec. II.B. It also allows toroidally uniform coating of divertor tiles by lithium evaporators as discussed in Sec. II.D. The strong auxiliary heating capability together with its relatively small major radius ~ 0.94 m makes the NSTX-U facility a powerful test bed for innovative divertor concept research. With the recent divertor heat load scaling projection, the NSTX-U facility could provide reactor relevant divertor heat flux of up to 60 MW / m^2. NSTX-U will have a flexible divertor control system that enables significant flux expansion of up to ~ 50 as was recently obtained by a snowflake divertor configuration (Ref. 7). Divertor gas injection also enables radiative divertor ("detached") operation. The NSTX-U PFCs are graphite tiles that are bakeable at 350°C. The divertor PFCs can be coated with boron and lithium. The NSTX-U facility plans to implement metal PFCs such as tungsten and TZM (titanium-zirconium-molybdenum alloy) to replace graphite tiles. Because of the small major radius (e.g., the divertor radius is ~ 0.5 m for high triangular discharges), the divertor configuration is relatively compact which facilitates ready replacement of divertor tiles.

II.B. NSTX-U PMI DIAGNOSTIC SYSTEMS

The NSTX/NSTX-U facility has been investing in PMI related diagnostics in the past several years. Some of the recently implemented PMI diagnostics are shown in Fig. 3. There are over 20 PMI and edge plasma diagnostic systems on NSTX-U and additional ones are being readied. They includes Gas-puff Imaging (500kHz), High –density Langmuir probe array, Edge Rotation Diagnostics (T_i, V_ϕ, V_{pol}), 1-D CCD H_α cameras (divertor, midplane), 2-D fast visible cameras for divertor and overall plasma imaging, Divertor bolometer, IR cameras (30Hz), Fast IR camera (two color), Tile temperature thermocouple array, Divertor fast eroding thermocouple, Dust detector, Quartz Microbalance Deposition Monitors, Scrape-off layer reflectometer, Edge neutral pressure gauges, Material Analysis and Particle Probe, Divertor Imaging Spectrometer, Lyman Alpha (Ly_a) Diode Array, Visible bremsstrahlung radiometer, Visible and UV survey spectrometers, VUV transmission grating spectrometer, Visible filterscopes (hydrogen & impurity lines), and Wall coupon analysis. We shall now describe some of the recently added diagnostics.

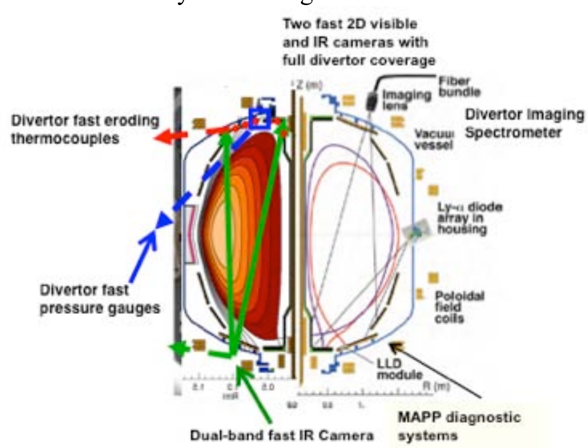

Fig. 3. Select PMI diagnostics on NSTX-U.

A major NSTX-U PMI diagnostic addition is the Material Analysis Particle Probe (MAPP) as shown in Fig. 4 (Ref. 8). MAPP is an in-vacuo inter-shot diagnostic capable of correlating surface chemistry evolution with plasma response to PMI conditioning. MAPP utilizes multiple surface-science measurement techniques to characterize a sample material exposed to NSTX-U conditions and assess plasma-surface interactions near the divertor strike point. MAPP includes a manipulator probe to insert a probe head with four samples that can be exposed to plasmas and withdrawn between plasma shots to characterize their surfaces. Techniques for surface analysis currently include: Thermal Desorption Spectroscopy (TDS), X-ray Photoelectron Spectroscopy (XPS), Low energy Ion Secondary Scattering (LEISS), and Direct Recoil Spectroscopy (DRS). The novelty with this design is the ability to apply three separate surface-sensitive characterization techniques promptly under the NSTX-U tokamak and without exposure to air.

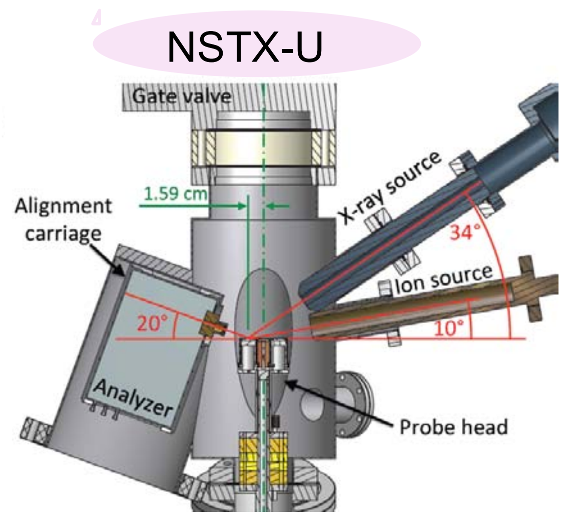

Fig. 4. A schematic of MAPP diagnostics.

In the area of divertor spectroscopy, two new spectroscopic diagnostics will be installed to study the plasma surface interactions on the liquid lithium divertor (Ref. 9). A 20-element absolute extreme ultraviolet (AXUV) diode array with a 6 nm bandpass filter centered at 121.6 nm (the Lyman-α transition) provides spatially resolved divertor recycling rate measurements in the highly reflective LLD environment, and an ultraviolet-visible-near infrared R = 0.67 m imaging Czerny–Turner spectrometer enables spatially resolved divertor D I, Li I-II, C I-IV, Mo I, D_2, LiD, CD emission and ion temperature on and around the LLD module. The use of photometrically calibrated measurements together with atomic physics factors enables studies of recycling and impurity particle fluxes as functions of LLD temperature, ion flux, and divertor geometry. Another important divertor diagnostic recently developed is a two-color or dual-band device developed for application to high-speed IR thermography (Ref. 10). Temperature measurement with two-band infrared imaging has the advantage of being mostly independent of surface emissivity, which may vary significantly for LLD as compared to that of an all-carbon first wall. In order to take advantage of the high-speed capability of the existing IR camera at NSTX (1.6-6.2 kHz frame rate), a commercial visible-range optical splitter was extensively modified to operate in the medium wavelength (MWIR) and long wavelength IR (LWIR). This two-band IR adapter utilizes a dichroic beamsplitter which reflects 4-6 micron wavelengths and transmits 7-10 micron wavelength radiation, each with > 95% efficiency and projects each IR channel image side-by-side on the camera's detector.

Taking advantage of open access of the divertor view in NSTX, a pair of two dimensional fast cameras with a wide angle view (allowing a full radial and toroidal coverage of the lower divertor) was installed in order to monitor non-axisymmetric effects (Ref. 11). A custom polar remapping procedure and an absolute photometric calibration enabled the easier visualization and quantitative analysis of non-axisymmetric plasma material interactions (e.g., strike point splitting due to application of 3D fields and effects of toroidally asymmetric plasma facing components). In Fig. 5, a full polar plot of divertor Li I brightness is shown, obtained by combining two camera views after a Type I edge localized mode (ELM). The two regions in which the lithium evaporators (discussed in Sec. II.D) are shadowed by the center stack result in a lower lithium influx and closely match the simulated distribution.

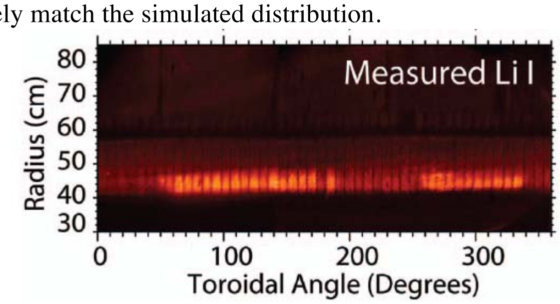

Fig. 5. 360° polar remapping of measured Li I brightness from the lower divertor outer strike point area.

Another novel PMI diagnostic is the first real-time detection of surface dust inside a tokamak that was made using an electrostatic dust detector (Ref . 12). As shown in Fig. 6(a), a fine grid with 25 μm spacing of interlocking circuit traces was installed in the NSTX vessel and biased to 50 V. Impinging dust particles created a temporary short circuit and the resulting current pulse was recorded by counting electronics. Various techniques was used to increase the detector sensitivity by a factor of 10,000 to match NSTX dust levels while suppressing electrical pickup, see Fig. 6(b). The results were validated by comparison to laboratory measurements, by the null signal from a covered detector that was only sensitive to pickup, and by the dramatic increase in signal when Li particles were introduced for wall conditioning purposes (Ref. 13). It should be noted that the real time dust measurement is necessary to safely manage the dust generated in ITER.

Dynamic retention of deuterium, lithium deposition, and the stability of thick deposited layers were measured by three quartz crystal microbalances (QMB) deployed in plasma shadowed areas at the upper and lower divertor and outboard midplane. Deposition of 185 µg/cm^2 over 3 months in 2007 was measured by a QMB at the lower divertor while a QMB on the upper divertor, that was shadowed from the evaporator, received an order of magnitude less deposition. Occasionally strong variations in the QMB frequency of thick lithium films were observed suggesting relaxation of mechanical stress and/or flaking or peeling of the deposited layers.

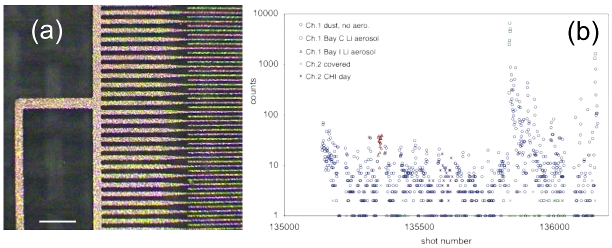

Fig. 6. (a) NSTX dust detection grid with 25 mm spacing. (b) Dust production vs. plasma events.

II.C. DIVERTOR HEAT FLUX AND SNOWFLAKE PMI RESEARCH

The NSTX-U has a new set of PF-1C poloidal magnetic field coils to provide flexible divertor control capabilities as shown in Fig. 7(a). On NSTX, even with a smaller number of PF-1 coils, a wide divertor configuration space has been accessed. The divertor peak heat flux was shown to be reduced by a factor of two by going to a double null configuration and reduced by another factor of 2.5 by increasing the triangularity δ from 0.4 to 0.75 (which increases the divertor flux expansion). In Fig. 7(b), the divertor peak heat flux is shown as a function of the divertor flux expansion in NSTX (Ref. 14). As shown in the figure, the peak heat flux goes down roughly inversely to the flux expansion. Another significant new development is the “snow-flake” divertor configuration which dramatically increased the effective divertor flux expansion to ~ 50 as was demonstrated on NSTX. The snow-flake divertor configuration resulted in a significant reduction in both parallel and deposited divertor heat flux and improved impurity screening, while maintaining H-mode confinement (Ref. 7). Further progress in reducing the peak heat flux in NSTX was demonstrated by utilizing the detached divertor (PDD) regime resulting in up to an additional factor of 2 peak heat flux reduction. For example, the deuterium gas puffing into the divertor area reduced the electron temperature to a few eV in front of the divertor plate, sufficient to facilitate radiative cooling. A potentially important trend recently observed is the inverse relation of the divertor heat flux width with the plasma current in NSTX (Ref.14). If this trend continues to hold for NSTX-U, it will reduce the divertor heat flux width by a factor of two (since the plasma current will be doubled), bringing the peak divertor heat flux to 40 – 60 MW /m^2 level since heating power is also being doubled. It is clearly important to investigate this current scaling on NSTX-U while further exploring the potential heat reduction benefit of the large divertor flux expansion provided by the snow-flake configuration and the divertor detachment.

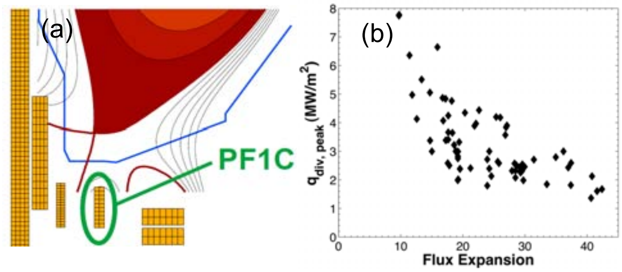

Fig. 7. (a) NSTX-U PF 1 Divertor Control Coils. (b) Divertor peak heat flux vs. flux expansion in NSTX.

II.D. LITHIUM PMI RESEARCH

Lithium wall coating techniques have been experimentally explored on NSTX since 2006 (Ref. 15). The lithium experimentation on NSTX started with a few milligrams of lithium injected into the plasma as pellets and it has evolved to a dual lithium evaporation system which can evaporate up to ~160 g of lithium onto the lower divertor plates between re-loadings. This lithium evaporation system has produced many intriguing and potentially important results. In 2010, the NSTX lithium research has focused on the effect of LLD surface on the divertor including the divertor heat load, deuterium pumping, impurity control, electron thermal confinement, H-mode pedestal physics, and enhanced plasma performance. To fill the LLD with lithium, 1300g of lithium was evaporated into the NSTX vacuum vessel during the 2010 operations. The routine use of lithium in 2010 has significantly improved the plasma shot availability resulting in a record number of plasma shots in any given year.

II.D.1. NSTX Lithium Evaporator

The NSTX lithium evaporator (LITER) set up is shown in Figure 8 (Ref. 6). The LITER system is essentially a temperature controlled stainless steel container filled with liquid lithium, with a nozzle to direct the lithium vapor for coating PFCs at desired locations. The nozzle is typically aimed toward the middle of the inner divertor to maximize the lithium deposition on the divertor plates. Two LITERS units were used for better toroidal PFC coverage of lithium on NSTX. The units each have a 90 g lithium capacity. LITER consists of a main reservoir oven and an output duct to allow insertion in a PFC gap in the upper divertor region. Two heaters were used on each LITER, one heater on the output duct and one heater on the main reservoir. The heater on the main reservoir was typically operated to maintain the liquid lithium temperatures of 600–650 ℃ which enables an adequate lithium evaporation rate, as this rate increases rapidly with temperature. The output duct was operated 50–100 ℃ hotter than the main reservoir to reduce lithium condensation on the output duct aperture. Typical evaporation rates have been in the range of 1 to 40 mg/min. The lithium evaporation typically takes place between plasma discharges to obtain the desired level of lithium coating on the PFCs, which could be in the range of 30 – 500 nm equivalent thickness. In NSTX, nearly 1,000 g of lithium was delivered onto the PFCs during an experimental campaign in 2010.

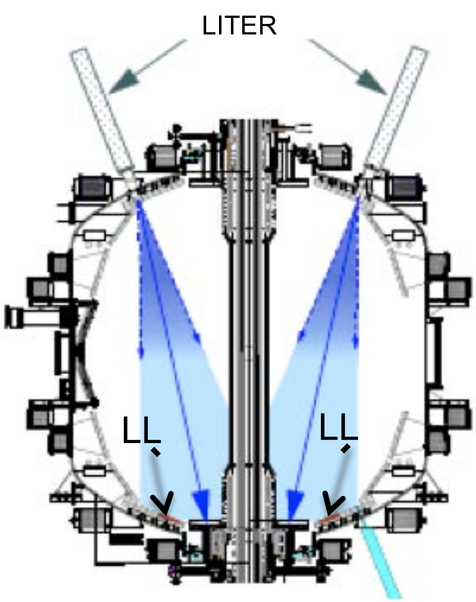

Fig. 8. A schematic of the lithium evaporators in NSTX injecting lithium vapor to coat the lower divertor surfaces including LLD .

II.D.2. NSTX Lithium Experimental Results

In the NSTX spherical tokamak, lithium evaporation has produced many intriguing and potentially important results (Refs. 6, 15). In Figure 8, a schematic diagram of the poloidal cross-section of NSTX and the LITER arrangement for NSTX is shown. The locations of two LITERs are at toroidal angles 165° and 315°, and the LITER central axes are aimed at the lower divertor. The shaded regions indicate the measured half-angle of the roughly Gaussian angular distribution at the 1/e intensity. A unique feature of the NSTX lithium research is the ability to investigate the effects of lithium in H-mode divertor plasmas. This addresses the long standing question of the effectiveness of lithium in a diverted H-mode plasma compared, for example, to the improvement observed in the limited plasma TFTR supershots (Ref. 16). The application of LITER on NSTX has yielded a significant improvement in the electron confinement with lithium coating of carbon tiles in H-mode plasmas. Importantly, the lithium evaporation resulted in a broadening of H-mode electron temperature profiles compared to plasmas without lithium applied. 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 lithium evaporation (Ref. 17). The improving electron energy confinement with lithium is consistent with the trend of improved confinement with reduced collisionality generally observed in NSTX. Thus far, the electron energy confinement continues to improve with the amount of lithium evaporated without reaching an apparent saturation which suggests that further improvements maybe possible. Additionally, lithium was shown to reduce the H-mode power threshold (Ref. 18) as well as to eliminate ELMs (Ref. 19). It is also noted that even with significant applications (up to 1,000 grams in NSTX) of lithium on PFCs, very little contamination (< 0.1%) of lithium fraction in main fusion plasma core was observed even during high confinement modes (Ref. 20). The lithium therefore appears to be highly desirable to be used as a plasma PFC material from the magnetic fusion plasma performance and operational point of view.

II.D.3. NSTX-U Liquid Lithium Divertor

LLD was tested in 2010 in a high performance H-mode configuration with high divertor heat flux in NSTX (Ref. 21). A picture of the LLD plates in NSTX is shown in Fig. 2. The LLD consists of four plates, 22 cm wide and each spanning 80◦ toroidally. The plates are electrically isolated toroidally to prevent induction of large toroidal currents in LLD by the ohmic transformer. The quadrants were separated toroidally by graphite tiles containing diagnostics and electrodes for edge plasma biasing. The plasma-facing surface of the LLD has a 0.17 mm layer of Mo, plasma sprayed with 45% porosity onto a protective barrier of 0.25 mm stainless steel that is bonded to a copper substrate 2.2 cm thick. The Mo porosity is intended to facilitate wetting and subsequent spreading of liquid lithium over the LLD, and to make the lithium surface tension forces large relative to electromagnetic forces in the liquid layer. In the NSTX experiment, sufficient lithium was applied by the LITER system to the LLD surface to saturate the sprayed Mo layer. Even though the liquid lithium layer is very thin, there was no observation of Mo line radiation from the plasma (i.e., no indication of excessive Mo material erosion), and inspection of the LLD surface after the campaign yielded no visual evidence of power or cyclic thermal stress damage to the plasma sprayed porous Mo LLD surface. In NSTX, the in-situ measurement of the divertor heat flux with a "two color" fast infrared camera of lithium coated LLD plate was significantly less (~ x 2) than those surfaces with reduced lithium coating, again indicating the benefit of a lithium coating. The reduced head load was accompanied by the increased divertor bolometric radiation as expected (Ref. 22). It should be noted that the previous modeling calculations and subsequent experimentations indicate that the lithium radiative loss in divertor can be significantly enhanced over the coronal equilibrium value (Refs. 23-24). This is because of enhanced transport (or poor confinement, low $n_e\tau$) expected in the open field line divertor region. From the facility operational point of view, the routine use of lithium has significantly improved the plasma shot availability (~ 50%) resulting in a record number of plasma shots in any given year for NSTX (Ref. 25). These favorable experimental results bode well for future liquid lithium divertor applications for fusion plasmas. Taking advantage of the strong lithium radiation property, a conceptual radiative liquid lithium divertor for a power plant is shown in Fig. 9 (Ref. 26). A proto-type radiative divertor is considered for NSTX-U (Ref. 25) and the liquid lithium loop R&D is being initiated as descried in Sec. III.A. Overall, lithium as a PFC material has very exciting prospects in contributing to magnetic fusion research as a tool to control the plasma edge and as a potential solution for the very challenging magnetic fusion reactor divertor heat flux problem.

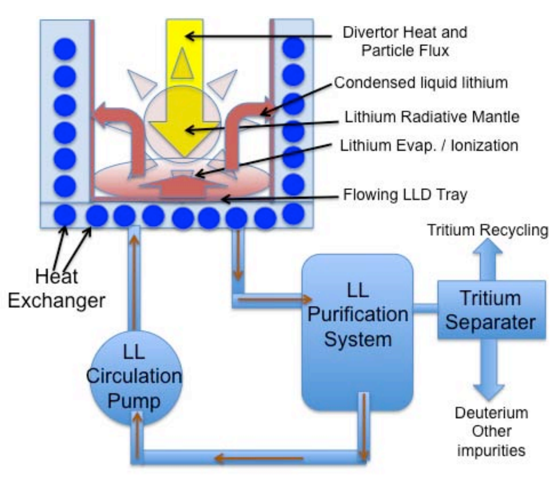

Fig. 9. A schematic view of closed radiative mantle based liquid lithium divertor with a closed loop liquid lithium purifying and tritium extraction system.

III. ASSOCIATED PMI FACILITIES AT PPPL

In order to develop sufficient PMI understanding and to support NSTX-U PMI research, there are a number of PMI facilities now operating at PPPL. We shall describe those activities in the following sections.

III.A. LIQUID LITHIUM R&D FACILITY

Development of flowing liquid lithium PFCs and related technologies is underway at PPPL. The present efforts focus on a capillary-restrained system that incorporates gaseous cooling. The conceptual diagram of the design is shown in Fig. 10. In this conceptual diagram, "T-tubes" (Ref. 27) are considered for the active cooling scheme with either helium or s-CO2 as the primary coolant. Cooling tube axes would lie within the poloidal plane of the machine, the approximate diameter of the cooling tube is ~1cm. The PFC includes a porous or textured front face, similar to that used on LLD (Ref. 21). Stability analysis indicated that for porous materials with pore sizes of order 0.01mm, very large current densities are required to destabilize and eject droplets (Ref. 28). This method for reducing droplet ejection via reduced pore size was experimentally observed on T11-M with the Red-Star CPS system (Ref. 29). In order avoid a long wicking path-length between a lithium source or reservoir and the PFC front-face, lithium flow channels are located parallel to the cooling channels with discrete ports to allow the liquid metal to wick to the front-face. The thin liquid layer results in large viscous forces in addition to MHD damping and the resulting flow velocities are expected to be small which enables the temperature transport to be described by thermal conduction (Ref. 30). This reliance on thermal conduction allows the significant work on gaseous cooling to be used in the liquid metal context and scaling studies with the US-based T-tube concept (Ref. 27) have been carried out. In addition to size scaling, an examination of the basic cooling fluid is also being carried out. S-CO2 has been identified by many in the fission power industry as having favorable properties in a power cycle over helium (Ref. 31). These include more efficient power-cycles overall for similar turbine inlet temperatures, more compact turbo-machinery leading to lower capital costs, lower leak rates overall and improved heat transfer efficacy at power-cycle-relevant pressures and temperatures. Calculations performed with ANSYS/CFX analysis code of the T-tube configuration with identical volumetric flow-rates of He and s-CO2 indicate that s-CO2 can reduce the peak surface temperature by over 250 °C of a solid-W target. While tungsten PFCs are the leading solid-PFC choice, lithium-coated PFCs are expected to operate at temperatures below the tungsten ductile-to-brittle transition temperature of 800 °C (Ref. 32). At lower temperatures, however, steel materials are potential substrates for use with liquid lithium where the lithium 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 (Ref. 32). It is likely, however, that the oxide-dispersion strengthened types of steels will be necessary for the elevated temperatures and coolant pressures in a divertor PFC.

Experimental demonstration of liquid metal PFC concepts is currently underway to complement the design studies described above. These begin with the development of a liquid lithium loop which will provide active pumping into and out of vacuum chambers. The facility will be utilized for testing of candidate PFC designs to show such things as (1) stable operation and flow in a tokamak-relevant vacuum environment (10^{-7} to 10^{-6} Torr pressures), (2) restart capability after periodic shut-down and gettering of residual gases, (3) maintainability and reliability in addition to safe operation. Future plans include active purification of lithium inventory and upgrades to include integrated tests with s-CO2 cooling systems.

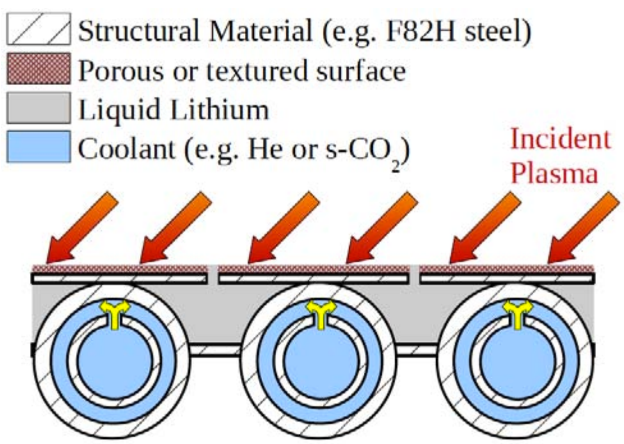

Fig. 10. Conceptual diagram for an actively-cooled, actively-wetted liquid lithium PFC.

III.B. LITHIUM TOKAMAK EXPERIMENT

The Lithium Tokamak Experiment (LTX) is being operated at PPPL to test effects of very aggressive lithium pumping to achieve very low recycling regime by essentially surrounding the entire plasma surface with lithium coated PFCs (Ref. 33). The PFCs can be heated to high temperatures to test the effects lithium PFC surfaces for both solid and liquid states. This was motivated by the previous CDX-U experiment (Ref. 34) and theoretical prediction of a very high confinement regime by eliminating the edge recycling (Ref. 35). The NSTX-U MAPP system is also being installed on LTX in preparation for the NSTX-U operations.

III.C. LABORATORIES FOR MATERIALS CHARACTERIZATION AND SURFACE CHEMISTRY

In order to fully understand plasma material interactions, it is essential to develop experimental capabilities with controlled environments to measure physical and chemical processes at a microscopic level. There are two new laboratories established at PPPL in collaboration with Princeton University dedicated for materials characterization and surface chemistry experiments. The Surface Science and Technology Laboratory is equipped with three surface analysis systems and an ultrahigh vacuum deposition chamber. Substrates for vapor deposition of metal films can be heated and cooled from 85 – 1500 K using liquid nitrogen cooling and resistive and electron-beam heating. These systems have a variety of surface diagnostics, including high resolution electron energy loss spectroscopy (HREELS), which is capable of probing both optical and vibrational excitations over a wide range of 0 - 100 eV with an electron energy resolution of 3 meV, alkali ion-scattering spectroscopy (ALISS), and angle-resolved X-ray photoelectron spectroscopy (XPS). Another instrument that has XPS, low energy ion scattering (LEIS), and reflection high-energy electron diffraction (RHEED) capability for thin film growth studies. In this laboratory, the time evolution of the chemical composition of lithium surfaces exposed to typical residual gases found in tokamaks was recently measured. Solid lithium samples and a TZM alloy substrate coated with lithium have been examined using XPS, temperature programmed desorption (TPD), and Auger electron spectroscopy (AES) both in ultrahigh vacuum conditions and after exposure to trace gases. Lithium surfaces near room temperature were oxidized after exposure to 1-2 Langmuirs ($1L=1x10^{-6}$ torr s) of oxygen or water vapor. The oxidation rate by carbon monoxide was four times less. An important result of the measurements is that lithiated PFC surfaces in tokamaks were found to be oxidized in about 100 s depending on the tokamak vacuum conditions which is much less than a typical time duration between the tokamak plasma shots (Ref. 36). A second laboratory, the Surface Imaging and Microanalysis Laboratory, contains a high-performance field emission Auger and multi-technique surface microanalysis instrument with a field emission electron source and lateral resolution of 30 nm for elemental analysis of surfaces of samples on the micro and nano scale.

IV. CONCLUSIONS

PMI is a high priority research area for magnetic fusion research as well as for NSTX-U and PPPL. The peak divertor heat flux in a 1 GW-electric class power plant could exceed the present solid-based divertor PFC power handling limit by an order of magnitude. It is therefore critical to develop an innovative divertor PMI solution for the realization of practical steady-state magnetic fusion reactors. Progress is being made on a variety of PMI fronts. An integrated PMI experimental research is being carried out at high priority on NSTX-U. The high power heat flux, flexibility divertor configuration control, open access to the divertor region are the strength of the NSTX-U facility. The state-of-the-art PMI diagnostic systems are being implemented to support the NSTX-U PMI research. The snow-flake divertor concept and liquid lithium PMI are two promising innovative NSTX-U PMI research topics for possible reactor divertor heat flux solutions. Specialized PMI test facilities are also available at PPPL / PU including the Liquid Lithium R&D facility, Lithium Tokamak Experiment, and Laboratories for Materials Chracterization and Surface Chemistry.

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