NSTX Upgrade Program Letter for Research Collaboration by National Laboratories for FY 2013-2016

Introduction

This NSTX Upgrade (NSTX-U)[1] Program Letter provides updated information about NSTX-U topical research priorities and collaboration opportunities during the upcoming four years (FY2013-2016). This information is useful for the preparation of proposals in response to the Office of Fusion Energy Sciences Notice: *Collaborative Research in Magnetic Fusion Energy Sciences on the National Spherical Torus Experiment Upgrade* issued July 19, 2012. New and continuing development and implementation proposals from U.S. national laboratories are the primary emphasis of this Program Letter. This letter suggests specific collaboration opportunities, as well as broader areas of research, in order to encourage proposals that address the research goals of NSTX-U. These research areas are described in the NSTX Five-Year Plan for 2009-13[2], recent NSTX-U Program Advisory Committee (PAC) presentations[3], and the FESAC Facilities Report[4]. The NSTX-U PAC reviewed this Program Letter on August 1, 2012 and PAC recommendations were incorporated in this final version.[5]

Mission of NSTX-U

The *programmatic* mission of NSTX-U is to evaluate the attractiveness of the compact Spherical Torus (ST) configuration for reducing cost, risk, and development time for practical fusion energy. The ST appears particularly attractive for: integrating Plasma Material Interface (PMI) solutions with high plasma performance (a major goal of NSTX, the NSTX Upgrade Project[6], and the National High-power advanced-Torus eXperiment - NHTX[7]) and for an ST-based Fusion Nuclear Science Facility (ST-FNSF)[8] with goals of progressively exploring and understanding the integrated fusion nuclear environment and ultimately accessing regimes with high neutron flux and fluence at high duty-factor in an ST-based Component Test Facility (ST-CTF)[9]. NSTX-U also contributes to the physics basis for an ST-based Pilot Plant[10] and DEMO[11] devices and accesses unique plasma regimes for resolving key burning plasma physics issues anticipated in ITER. The NSTX-U programmatic mission thus addresses two of the long-term goals of the Office of Fusion Energy Sciences: configuration optimization, and developing a predictive capability for burning plasmas. Both ITER participation and also FNS/CTF development are included in the DOE 20-year strategic plan for the Fusion Energy Sciences Program.[12]

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[1] http://nstx-u.pppl.gov
[2] http://nstx.pppl.gov/DragNDrop/Five_Year_Plans/2009_2013/NSTX_Research_Plan_2009-2013.pdf.
[3] http://nstx-u.pppl.gov/program/program-advisory-committee/pac-31
[4] http://www.ofes.fusion.doe.gov/more_html/FESAC/FacilitiesVol1.doc and FacilitiesVolume2_v3.pdf.
[5] http://nstx.pppl.gov/DragNDrop/Program_PAC/Program_Letters/ (available August 8, 2012).
[6] http://nstx-u.pppl.gov/nstx-upgrade
[7] http://nstx.pppl.gov/DragNDrop/NHTX_Information
[8] Y-K M Peng et al, Fusion Science and Technology Vol. 56, August 2009 page 957
[9] Y-K M Peng et al, Plas. Phys. Cont. Fus. **47** (2005) B263 (also http://nstx.pppl.gov/DragNDrop/CTF_Information)
[10] http://nstx.pppl.gov/DragNDrop/Publications_Presentations/Publications/2011%20Papers/Menard_NF.pdf
[11] http://www.sciencedirect.com/science/journal/09203796/65/2
[12] http://www.sc.doe.gov/bes/archives/plans/SCSP_12FEB04.pdf.

In support of the above programmatic mission, the *scientific* mission of NSTX-U is to advance fusion plasma science by understanding the special physics properties of the Spherical Torus (ST). Due to its low aspect ratio, the ST is characterized by strong magnetic field curvature and by high β_T (the ratio of the average plasma pressure to the applied toroidal magnetic field pressure). The ST, with its unique properties, thus extends and complements the higher aspect ratio, lower β_T tokamak in addressing the overarching scientific issues in magnetic fusion. Proposals that support collaborative research leveraging unique ST parameter regimes relative to other configurations and devices are strongly encouraged. For example, the NSTX-U research team carries out a substantial number of the ITPA joint experiments and/or joint data analysis activities each year. Potential NSTX-U collaborators are strongly encouraged to propose and participate in these cross-device activities. For more information on past participation and future collaboration opportunities in ITPA activities involving NSTX-U, please contact the ITPA coordinator for NSTX-U: Stan Kaye (kaye@pppl.gov).

NSTX-U Research Priorities and Key Collaboration Opportunities

This section lists the high priority topics in the NSTX Upgrade research program during FY 2013-2016 and highlights key collaboration opportunities for research proposals. Collaboration proposals should aim to support the NSTX Upgrade research program by utilizing and/or upgrading existing high priority capabilities and/or implementing new capabilities and carrying out new experiments and analysis.

Important: The NSTX Upgrade involves an extended outage period which began in October 2011, and operation is expected to resume in April 2014 (resources permitting). In view of this, the normal three year collaboration grant cycle has been extended to four years for this solicitation in order to provide collaborators an opportunity to acquire new data in the first years of NSTX Upgrade operation. With respect to plasma and device modifications, NSTX Upgrade is anticipated to provide access to plasma regimes of reduced collisionality, longer plasma duration, and increased overall plasma performance. To achieve these goals, the toroidal magnetic field, plasma current, and neutral beam injection (NBI) heating power are planned to be increased by up to a factor of two, and the pulse duration increased by up to a factor of five. The diameter of the center-stack will be increased and the minimum plasma aspect ratio will be increased from 1.3 to 1.5. Further, the NBI power will be increased through the addition of a 2nd more tangential NBI. Several diagnostic ports will be modified, and port access may also be impacted by additional structural enhancements.

The staging of NSTX-U capabilities during the first few years of operation is presently planned as follows: we expect NSTX-U to achieve field and current values 50% above NSTX values (i.e. up to 0.75T and 1.5MA) by the end of the first full year of operation of operation (2015), and values up to 100% higher than NSTX (i.e. up to 1T and 2MA) by the end of the second full year of operation (2016). Pulse durations of up to 3-5s are also anticipated during the first 2 years of operation, but achievable pulse-lengths will depend on the effectiveness of particle control techniques and on the performance of the upgraded magnets and structural supports of NSTX-U. During the first years of NSTX-U operation (2014-2016), the new 2nd NBI will be available for the entire period, a new second switching power amplifier will commissioned for independent control of all 6 mid-plane 3D field coils by the end of the first full run year, the divertor and first-wall will utilize graphite PFCs (although a molybdenum upper divertor is under consideration for 2016), and lithium coating coverage will be extended to the upper divertor and first-wall by 2016. **The impact of the above schedule, performance, and device changes should be assessed and incorporated into all collaboration plans.**

The NSTX-U research priorities and key collaboration opportunities are organized according to the six categories used in the FESAC Priorities Plan. For each of the six scientific categories described below, the highest priority research topics are provided in approximate priority order. For each category, a key person is listed who can be contacted for further information and to assist in identifying a research contact for the "Record of Discussion" (RoD) form that must be included with each submitted proposal following review and signature by the NSTX-U program and project directors. The RoD documents the proposed research goals of the collaboration, collaborator off-site research tasks, on-site research support tasks and estimated effort required, on-site engineering support tasks and estimated effort required, and estimated hardware costs. Successful proposals requiring implementation of significant hardware on NSTX-U will further require a "Record of Agreement" (RoA) form to document hardware and interface implementation tasks and milestones. The URLs for the RoD and RoA forms are:

RoD: http://nstx.pppl.gov/DragNDrop/Program_PAC/Program_Letters/NSTX_Record_of_Discussion_FY2013.doc
RoA: http://nstx.pppl.gov/DragNDrop/Program_PAC/Collaborations/NSTX_record_of_agreement_April2009.doc

The development of stationary, high-performance, long-pulse plasmas is a high priority programmatic objective of NSTX Upgrade. A summary overview of high priority capabilities and opportunities supporting this programmatic goal is as follows: For Macroscopic Stability research, collaboration aimed at understanding the impact of varied rotation, rotation shear, and fast-ion density profile on resistive wall modes, neoclassical tearing modes, and mode locking. Understanding plasma response to 3D fields is encouraged, and understanding disruption thermal quench and halo current evolution and the development of disruption mitigation techniques are also encouraged. For Multi-Scale Transport Physics research, collaboration supporting improved understanding of electron and ion thermal, momentum, and impurity and particle transport at higher field and current are encouraged. For Plasma Boundary Interfaces research, collaboration on scrape-off-layer transport and turbulence, cryo-pump design, H-mode pedestal structure and control, high-flux-expansion divertors and divertor detachment, and lithium-based plasma facing components are encouraged. For research in Waves and Energetic Particles, collaboration on optimizing fast-wave heating performance at increased magnetic field and improved understanding of Alfvén Eigenmode stability and fast-ion transport with varied instability drive are encouraged. For Plasma Start-up and Ramp-up research, analysis and experiments of HHFW heating and current drive, simulation and design of ECH heating systems, and simulations of Coaxial Helicity Injection (CHI) current formation and ramp-up are encouraged. For Advanced Scenario Development, real-time diagnostics and algorithms for controlling divertor PFC temperature and heat flux profiles, and for disruption onset detection are especially encouraged.

Collaboration proposals integrating the comparison of experimental measurements with theory and simulation are especially encouraged to maximize the development of new predictive capability.

The collaboration opportunities highlighted below were determined on the basis of what the NSTX-U program considers necessary to support the research priorities while also complementing ongoing contributions from university and industry and PPPL researchers. Proposals for innovative collaboration activities beyond the ones listed in this Program Letter are also welcomed. All proposals will be considered by means of the normal DOE peer review process, according to the criteria described in the solicitation announcement.

I. **Macroscopic Stability** – the role of magnetic structure in plasma confinement and the limits to plasma pressure in sustained magnetic configurations.

For more information contact: Jong-Kyu Park (jpark@pppl.gov)

Research Priorities:

I-1. *Understand the role of kinetic effects in RWM stability and toroidal rotation damping to optimize RWM stability and control in ITER and future facilities.*

Background:

Significant progress has recently been made in NSTX in identifying and understanding the role of kinetic resonances in resistive wall mode stability. Substantial progress has also been achieved in the detection and active control of error fields (EF) and resistive wall modes (RWMs), and in understanding and controlling toroidal flow damping from non-axisymmetric fields. Advanced control algorithms and enhanced predictive capability (especially the effects of fast-ions on RWM stability) are under development to understand present RWM and EF results and allow extrapolation of high performance to ITER and future STs. In NSTX Upgrade, the reduced collisionality, modifications to safety factor and rotation and rotation shear profiles, and changes in the fast-ion distribution function from more tangential injection are anticipated to modify RWM stability and control.

I-2. *Study the impact of low aspect ratio, high beta, large ion gyro-radius, magnetic shear, and flow shear on classical and neoclassical tearing mode stability.*

Background:

Neoclassical tearing mode (NTM) instabilities have been observed to limit the plasma performance in some operational scenarios of NSTX. NTMs can be triggered by energetic particle modes, edge localized modes, error fields, and can also arise from natural resistive tearing instability drive. Low-frequency (f=1-30kHz) MHD activity (including tearing activity) has also been observed to redistribute the fast ions from neutral beam injection, and this physics may be relevant to proposed "hybrid" operating scenarios for ITER. For the 2/1 NTM, the magnitude of local NTM instability drive from the bootstrap current correlates most strongly with the rotation shear rather than the absolute rotation, suggesting the importance of rotation shear on delta-prime. However, the threshold for NTM triggering by error fields has also been shown to be a function of the local rotation magnitude, potentially providing new insight into the mode coupling physics for NTM triggering. In NSTX Upgrade, the reduced collisionality, modifications to the safety factor and rotation and rotation shear profiles, and changes in the stabilizing curvature and poloidal mode coupling from increased aspect ratio may all modify the tearing mode stability.

I-3. *Assess neoclassical toroidal viscosity, plasma equilibrium and stability response to 3D fields, and the physics and control of toroidal rotation at reduced collisionality.*

Background:

Neoclassical toroidal viscosity (NTV) models coupled to ideal 3D perturbed equilibrium solutions have shown reasonable agreement with measured plasma rotation damping in the core of NSTX plasmas and in other devices. However, the effects of rotation, rotation shear, kinetic damping, and magnetic islands on the plasma response and thus NTV remains an active area of research of importance to ITER and future ST devices – especially in the area of ELM control using 3D fields. NSTX Upgrade will substantially extend this physics through access to reduced collisionality and through modifications of the rotation and rotation shear profiles using both the 2^{nd} NBI and through magnetic braking.

I-4. *Characterize the dynamics of disruptions at low aspect ratio and high beta by measuring halo currents and thermal and current quench characteristics.*

Background:

NSTX has contributed plasma current quench rate and halo current data to the ITPA international database on disruptions with application to both ITER and future ST-FNSF facilities. NSTX has recently extended these studies with improved measurements of halo current distributions, fractions, and halo-peaking factors. Additional measurements are needed to understand thermal quench characteristics of the ST, and data analysis and modeling of disruptions is desired to develop a predictive capability for disruptions applicable to NSTX Upgrade and next-step STs. The uniquely low internal inductance, high elongation, high β, strongly wall-coupled plasmas of the ST could change the instability dynamics (and hence the impurity penetration and radiation evolution) of conventional mitigation techniques such as massive gas injection (MGI). The dependence of plasma density increase on poloidal injection location (midplane vs. divertor) is planned to be assessed for the first time during the first few run years of NSTX-U operation.

Key Collaboration Opportunities in Macroscopic Plasma Physics:

During the first few years of NSTX-U operation, emphasis will be placed on re-establishing optimized error-field correction and RWM control, development of scenarios that avoid NTMs, the understanding and development of rotation control, and first tests of the off-midplane gas injection for disruption mitigation. Key collaboration opportunities include:

- *Advance the understanding of RWM passive stability and advanced state-space control as a function of collisionality, fast-ion content, and rotation.*
    - *Exploit the off-axis heating and momentum deposition from the new more tangential 2nd NBI combined with central HHFW electron heating, particle pumping from lithium coatings, and more flexible magnetic braking from 6 independently controlled 3D field coils to vary the plasma temperature and density, fast-particle content, and rotation profiles.*
- *Analyze and simulate neoclassical tearing mode stability emphasizing improved understanding of the physics of rotational shear stabilization and triggering of the NTM including the effects of varied rotation profile and varied toroidal mode number of externally applied 3D field.*
    - *Utilize the new tools described above for RWM studies for NTM studies.*

- *Contribute to the development of diagnostics, models, and algorithms for real-time control of the plasma rotation using NBI and magnetic braking to enable controlled variations of the plasma rotation.*
    - *Exploit this new capability for NTV studies, for RWM and NTM stability and error-field penetration research, as a potential means of avoiding disruptions, and for transport and turbulence research.*
- *Analyze and simulate disruption precursor onset and disruption evolution – especially the thermal quench and halo-current dynamics.*
    - *Assess and develop novel techniques for disruption mitigation via very rapid density build-up from efficient edge and/or core fueling.*

II. ***Multi-Scale Transport Physics*** – the physical processes that govern the confinement of heat, momentum, and particles in plasmas.

For more information contact: Yang Ren (yren@pppl.gov)

Research Priorities:

II-1. *Determine the modes (low-k, high-k, electrostatic, electromagnetic, Alfvénic) responsible for causing anomalous electron energy transport.*

Background:

NSTX results indicate that ion energy and impurity transport levels are routinely at the neoclassical level, implying suppression of long-wavelength turbulence and associated anomalous transport in those channels. Such suppression is likely related to NBI-induced toroidal rotation leading to ExB shearing rates exceeding low-k turbulence growth rates. This plasma state with (controllable) suppression of long-wavelength turbulence provides an excellent environment to study short wavelength turbulence and its relationship to electron heat transport. Using a high-k scattering diagnostic, high-k fluctuations with the characteristics of Electron Temperature Gradient modes (ETGs) have been measured when the ETG critical temperature gradient is exceeded, and correlations between increased high-k fluctuation amplitude and increased electron thermal transport have been observed. Recent non-linear gyrokinetic simulations of micro-tearing modes find a nearly linear dependence of the electron thermal diffusivity on collisionality – seemingly consistent with ST global confinement trends. Global Alfvén Eigenmodes (GAE) have also been previously been correlated with anomalous electron transport in the core NSTX NBI heated plasmas. NSTX Upgrade plasmas are anticipated to have lower collisionality and different GAE instability drive, which could in turn change which modes dominate electron turbulent transport.

II-2. *Determine the role of low-k turbulence in causing anomalous ion energy and momentum transport, and understand the influence of plasma rotation on low-k and high-k turbulence.*

Background:

Recent NSTX results indicate that the angular momentum diffusivity as determined from steady-state momentum balance is routinely significantly smaller than ion energy diffusivity but larger than neoclassical predictions. This is true also for the momentum diffusivity as inferred from perturbative experiments, where the momentum pinch was also inferred. These results are qualitatively different than those commonly obtained in higher aspect ratio tokamaks. The experiments using perturbative momentum transport techniques indicate the existence of an inward momentum pinch consistent with values predicted by gyro-kinetic simulations of low-k turbulence. Externally applied 3D magnetic field perturbations lead to decreased rotation and rotation shear, and this is associated with increased inferred transport levels in the ion energy channel. Lithium conditioning has also recently been shown to substantially modify the near-edge rotation profile and the threshold power for access to the high-confinement mode (H-mode). Finally, some tokamak results suggest low to intermediate-k turbulence could lead to anomalous transport of energetic ions from neutral beam injection – a finding which could be tested utilizing the high and controllable ExB shearing rate conditions of NSTX plasmas. Higher field and current in NSTX Upgrade will reduce the neoclassical transport rates and could modify the relative dominance of neoclassical ion thermal transport over turbulent ion thermal transport, with implications for auxiliary power and size requirements of future ST devices.

II-3. *Determine the relationship between the measured particle and impurity transport and simulated micro-turbulence and neoclassical transport*.

Background:

Improved density control is needed to access reduced density scenarios predicted to maximize beam-driven current in next-step ST devices. However, particle confinement is poorly understood relative to energy and momentum transport in NSTX. Recent NSTX results indicate that lithium evaporated onto the lower divertor surfaces of NSTX can act as an effective pump of hydrogenic species. Impurity transport has been measured to be nearly neoclassical in NSTX and exhibits a particle pinch generally consistent with neoclassical transport, but with some differences with respect to predicted profiles. ELM-free scenarios from Li surface coatings can lead to deleterious accumulation of impurities and further motivates additional research into impurity transport. Used in combination with dominant core fueling from NBI, and using perturbative approaches, the dependence of impurity and particle diffusivity on global plasma parameters, including rotation will be determined, and the possible relationship between impurity/particle and thermal diffusivity will be assessed. The possible relationship between the particle pinch and the momentum pinch will be investigated, and comparisons between anomalous impurity/particle transport and measured and simulated turbulence will be pursued. The higher field and current of NSTX Upgrade plasmas will reduce neoclassical particle diffusivity, which may impact the main ion and impurity evolution of all NSTX Upgrade operating scenarios.

Key Collaboration Opportunities in Multi-Scale Plasma Physics:

During the first few years of NSTX-U operation, emphasis will be placed on measuring instabilities responsible for anomalous thermal, momentum, and particle/impurity transport focusing on low-k and preliminary high-k measurements, and differentiating/controlling turbulence utilizing access to higher field and current and the 2nd NBI and 3D field coils to vary the rotation and rotation shear. Key collaboration opportunities include:

- *Perform electron thermal transport and turbulence studies in L-mode and H-mode to understand the source of anomalous electron thermal transport.*

  - *Utilize the increased magnetic field, plasma current, and power of the Upgrade combined with high-harmonic fast-wave core electron heating for producing low collisionality plasmas.*

- *Analyze and model ion-gyro-scale turbulence and transport to understand the relationship between the flow and flow-shear and turbulence and transport in the core and pedestal, to improve understanding of the H-mode transition, and to improve understanding of momentum transport by turbulence and 3D magnetic fields.*

  - *Exploit upgraded Beam Emission Spectroscopy (BES) diagnostic capabilities and lead or assist in the development and implementation of high time and spatial resolution rotation diagnostics for the near-edge region.*

- *Participate in experiments and analysis exploring particle/impurity transport from the edge to the core, and linkages to inward momentum pinch physics.*

  - *Utilize the higher field and current and varied momentum deposition of NSTX-U to vary the neoclassical transport levels, and utilize existing and/or any new main-ion and impurity edge particle sources for perturbative particle/impurity transport experiments*

The comparison of turbulence measurements with theory and simulation using a suite of micro-turbulence codes coupled with synthetic diagnostics is especially encouraged for the above opportunities in Multi-Scale Plasma Physics.

III. ***Plasma Boundary Interfaces*** – the interface between fusion plasma and its lower temperature plasma-facing material surroundings.

For more information contact: Bob Kaita (rkaita@pppl.gov)

Research Priorities:

III-1. *Investigate energy and particle transport and turbulence in the Scrape-Off-Layer (SOL), and understand the linkage between SOL parameters and heat and particle flux to the divertor*.

Background:

The transport of energy and particles in the tokamak plasma edge has important implications for the ability of the magnetic divertor to handle plasma exhaust. The SOL heat flux width parameter is particularly important, as the peak divertor heat flux scales inversely with this width. Multi-machine (including NSTX) studies of the SOL heat flux width have recently measured a strong inverse dependence of the heat flux width on plasma current and a weak dependence on magnetic field and heating power. NSTX Upgrade plasmas at high current and power are projected to have high peak heat flux values that challenge divertor PFCs. The underlying causes of the SOL heat flux scalings are poorly understood, and edge heat flux and transport and turbulence measurements remain vital to improving this understanding. In addition, a divertor cryo-pump is under consideration for installation in NSTX Upgrade during the next 5 year planning period, and divertor heat and particle flux profile characteristics and modeling are needed for the physics design of cryo-pumping systems.

III-2. *Understand boundary plasma response to applied 3D magnetic field perturbations and other perturbations designed to control edge plasma transport and stability.*

Background:

NSTX has made significant progress in characterizing pedestal stability and developing small ELM regimes, and is actively comparing measured ELM stability thresholds to peeling-ballooning theory. With the application of evaporated lithium onto the lower divertor, long-pulse ELM-free discharges have been obtained. In these discharges, the pedestal temperature increased as did the computed edge bootstrap current density, and these edge profile changes have modified stability thresholds consistent with peeling-ballooning theory. Further, unlike the results of some higher aspect ratio tokamaks, edge 3D resonant magnetic perturbation (RMP) fields are found to be destabilizing rather than stabilizing in NSTX for reasons that are not yet understood. RMP fields have been used to purposely trigger ELMs to expel impurities from ELM-free H-mode, and the plasma boundary shape influences size of the triggered ELMs. NSTX Upgrade will have higher pedestal temperature and pressure and lower pedestal collisionality, and these differences could impact the edge bootstrap current density profile and the plasma response to 3D fields including the transient particle expulsion driven by ELMs. Off-midplane non-axisymmetric control coils (NCC) are under consideration for installation in NSTX Upgrade during the 5 year plan period, and such coils could also improve control of pedestal transport and stability. In addition, lithium granule injection has recently been successfully tested on EAST for ELM triggering, and will be further tested on NSTX-U.

III-3. *Study the synergy of high flux-expansion divertor configurations with the radiative divertor and assess applicability to a Fusion Nuclear Science Facility.*

Background:

To mitigate high heat fluxes, NSTX has made significant progress in developing a partially detached divertor regime for reducing peak divertor heat flux consistent with good H-mode confinement and acceptable density control (at high density). Further, a high flux expansion divertor – namely the “snow-flake” divertor – has demonstrated substantial reductions in peak heat flux and impurity generation. In the last operational phase of NSTX, snowflake operation combined with detachment was utilized to achieve very large reductions (up to a factor of 7) in peak heat flux. Additional research on this configuration and other novel divertor configurations is needed to determine if such techniques extrapolate to the much higher heat-fluxes and lower normalized densities of a Fusion Nuclear Science Facility and DEMO.

III-4. *Investigate lithium-based plasma facing components to better understand how lithium modifies recycling, confinement, edge localized mode (ELM) stability, and divertor radiation and power-handling.*

Background:

NSTX has demonstrated that evaporated lithium deposited on the lower divertor can pump deuterium, increase thermal energy confinement, and eliminate ELMs in diverted H-mode plasmas. However, ELM-free plasmas from lithiumization can also lead to impurity accumulation and increased Z_{eff} and edge collisionality. Access to improved particle control and reduced collisionality requires improved understanding and control of particle sources, pumping, and transport. NSTX also implemented a liquid lithium divertor (LLD) module in 2010 to investigate the effects of liquid lithium in diverted H-mode plasmas. The LLD results indicate that a critical issue for the performance of solid and liquid lithium as a pump of hydrogenic species is the Li chemical reactions with background vacuum gases and plasma impurities, and the rapidity of these reactions motivates consideration of flowing liquid lithium PFC systems. Another important observation during the LLD experiments was evidence of temperature clamping of the Mo substrate under the lithium surface as the Li was heated through and above the Li melting temperature. Further, and somewhat counter-intuitively, recent analysis also indicates that for the highest lithium evaporation rates used on the NSTX inboard divertor, increased lithium deposition correlated with increased divertor radiation and reduced (rather than increased) peak divertor heat flux. These results motivate additional research into the potential of lithium for handling high heat flux and protecting solid divertor PFCs. NSTX Upgrade will continue to utilize lithium coatings for deuterium inventory control during the initial phase of Upgrade operation, and the next generation of LLD will undergo preliminary design during the Upgrade outage.

Key Collaboration Opportunities in Plasma Boundary Interfaces:

During the first few years of NSTX-U operation, emphasis will be placed on SOL width studies at increased current and field, validation of cryo-pump designs, H-mode pedestal structure and ELM stability assessments at higher field, current, and power, ELM control experiments including design of off-midplane 3D field coils, divertor flux expansion and detachment studies, tests of lithium pumping persistence at longer pulse duration, and assessments of the impact of increased lithium coverage of the divertor and first-wall. Key collaboration opportunities include:

- *Perform experiments, measurements, and simulations to aid in the development of predictive capability and control of SOL energy and particle fluxes.*
    - *Utilize the increased current, power, and magnetic field capabilities of NSTX-U to extend the range of variation of key parameters thought to control the SOL heat-flux width (especially the plasma current), and assess the implications for full-performance operation of NSTX-U and for ITER and FNSF.*
    - *Perform simulations, experiments, and measurements in support of the physics and engineering design of divertor cryo-pumps for NSTX-U – for example characterizing far-SOL density and temperature profiles. For more information on the physics design of cryo-pumps, please refer to the presentation at this link.*
- *Perform experiments, data analysis, and simulations to develop a predictive capability for the H-mode pedestal structure, stability, and control.*
    - *Optimize H-mode performance for high thermal confinement, small/no ELMs, density control, and acceptable impurity accumulation.*
    - *Utilize evaporated Li, Li granule injection, externally applied 3D fields, boundary shaping, and other techniques to develop edge localized mode (ELM) suppression, triggering, and control.*
    - *Participate/lead in the physics design of off-midplane non-axisymmetric control coils (NCC) in support of edge stability optimization for NSTX-U and ITER ELM control predictive capability. For more information on the NCC coils, please refer to the presentations at this link.*

- *Implement diagnostics and perform experiments and simulations aimed at reducing divertor heat-flux to mitigate the anticipated increased peak divertor heat flux in NSTX-U resulting from operation at higher plasma current and heating power.*
    - *In particular, explore: divertor detachment, high flux expansion, possible synergistic effects, and real-time diagnostic and control development for divertor heat flux mitigation.*
- *Perform experiments, analysis, and simulation to understand the impact of Li coatings on divertor and wall pumping, retention, and impurity generation - including transport of lithium and other impurities from the plasma edge into the pedestal and core under both steady-state and transient conditions (such as ELMs and disruptions).*
    - *Participate in experiments and perform analysis related to increasing the coverage of carbon (and possibly high-Z metallic PFCs) with lithium coatings.*
    - *Provide thermo-mechanical analysis and other design support for high-Z solid and liquid lithium PFCs under consideration for possible usage in a next-generation liquid-lithium divertor (or mid-plane lithium limiter) on NSTX-U.*

IV. ***Waves and Energetic Particles*** – the use of waves and energetic particles to sustain and control high-temperature plasmas.

For more information contact: Gary Taylor (gtaylor@pppl.gov)

Research Priorities:

IV-1. *Measure and simulate interactions between high-harmonic fast-waves (HHFW) and neutral beam fast-ions with application to optimizing plasma heating and current-drive by the HHFW in NBI-heated H-mode plasmas.*

Background:

Extensive HHFW coupling and heating studies have identified surface wave excitation as a key parasitic absorption mechanism that can reduce the effective core heating efficiency, and similar physics could play a role in the coupling of ICRF to ITER plasmas. Higher magnetic field and reduced density in front of the antenna (achieved through the use of evaporated Lithium) has enabled HHFW to reliably heat electrons in deuterium H-mode plasmas. These performance improvements motivated the upgrade to the HHFW antenna system, and record ST central electron temperatures (above 6keV) have been achieved using the upgraded antenna. HHFW acceleration of NBI fast-ions has previously been observed in NSTX plasmas, and additional research is needed to understand and minimize HHFW-NBI ion interactions in conditions with reduced surface wave excitation. In NSTX Upgrade, the higher toroidal field is expected to further reduce parasitic edge losses and reduce the number of ion-cyclotron resonances present in the plasma. However, operation with increased NBI power and for longer pulses may require operation with a larger outboard gap which could decrease HHFW coupling efficiency and increase edge parasitic losses.

IV-2. *Understand the transport of supra-Alfvénic fast ions due to Alfven eigenmode avalanches and other Alfvénic instabilities with particular emphasis on the possible redistribution of neutral beam current drive.*

Background:

The capability to excite and diagnose a broad range of fast-ion driven instabilities makes NSTX a powerful research tool for understanding energetic particle physics for ITER and future ST's. NSTX researchers have mapped and diagnosed the stability space of TAE modes - from mode onset to multi-mode avalanche threshold for a range of normalized fast-ion velocities and pressures. In addition, coupling between Alfvén Cascade modes and Geodesic Acoustic Modes has been characterized, and the eigenstructure of high-β Beta-induced Alfvén Acoustic Eigenmodes (BAAE) has been measured and successfully compared to theory. A major objective of NSTX research is to develop a predictive capability for determining the extent to which the Alfvénic MHD activity described above causes transport of energetic particles with application to ITER and an FNSF. In NSTX, successive TAE avalanche events have been measured to have an effect on the fast-ion confinement in the plasma core and to modify the NBI driven currents and resultant equilibrium q profile. Finally, the ability to largely suppress low-k turbulence in NSTX provides an excellent opportunity for comparison to previous observations (in several other fusion facilities) of fast-ion transport by electrostatic turbulence. In NSTX Upgrade, the radial transport of fast ions by *AE modes could significantly impact the beam current drive profile of the 2nd NBI and influence the ability to control current profile and support 100% non-inductive current drive.

Key Collaboration Opportunities in Waves and Energetic Particles:

During the first few years of NSTX-U operation, emphasis will be placed on assessing the performance of HHFW heating and current drive at higher toroidal field, investigating the compatibility of HHFW heating with NBI H-mode operation, comparing TRANSP classical slowing-down predictions with measurements of the fast-ion distribution function during usage of the 2nd NBI, and characterizing Alfvén eigenmode activity as a function of NBI tangency radius and $v_{fast}/v_{Alfvén}$. Key collaboration opportunities include:

- *Model HHFW heating and current-drive efficiency and RF-induced changes in the plasma core and edge.*
    - *Assess HHFW heating and current-drive performance at increased toroidal field, plasma current, heating power, and pulse duration in NSTX-U.*
    - *Perform experiments, data analysis, and simulation to understand the implications of interactions of NBI fast-ions with HHFW antenna limiters and study the NBI fast-ion acceleration by HHFW as a function of ion cyclotron harmonic number.*
- *Aid the development of predictive capability (especially reduced models) for fast-ion transport by fast-ion-driven instabilities for FNSF and ITER.*

- *Utilize varied field, current, and NBI tangency radius (to vary the fast-ion density profile) and HHFW fast-ion acceleration as tools to modify the fast-ion distribution and associated Alfvénic instabilities.*

- *Utilize several of the measured moments of the fast-ion distribution function to test linear and non-linear fast-ion instability models.*

V. ***Plasma Start-up and Ramp-up without a Solenoid*** – the physical processes of magnetic flux generation and sustainment.

For more information contact: Dennis Mueller (dmueller@pppl.gov)

Research Priorities:

V-1. *Develop electron cyclotron heating (ECH) of low current and low density helicity-injection start-up plasmas and electron Bernstein wave (EBW) heating and current drive for advanced scenarios.*

Background:

Coaxial helicity injection (CHI) has demonstrated transformer flux savings equivalent to 400kA of plasma current when CHI was added to an inductively-driven plasma current ramp. CHI coupled to induction has also been shown to be compatible with high performance H-mode operation. These favorable results motivate development and testing of means to efficiently heat and densify ST start-up plasmas to conditions compatible with non-inductive plasma current ramp-up to high plasma current. In NSTX Upgrade, helicity injection current drive is projected to scale favorably (linearly) with toroidal field strength. Thus, by doubling the toroidal field strength, the generation of 300-400kA of closed-flux plasma current should be achievable and is projected to provide a target plasma current suitable for NBI heating and current ramp-up. However, it is presently unclear if a CHI-produced target plasma will have sufficient energy input to achieve high electron temperature needed for efficient subsequent HHFW or NBI heating. Initial modeling indicates 1MW of 28GHz ECH heating is well suited for heating CHI target plasmas to 0.2-0.5keV sufficient for HHFW/NBI heating and current ramp-up. Further, such an ECH system could also be useful for ST electron heating using EBW and ultimately potentially for off-axis EBW current drive.

V-2. *Understand and optimize the non-inductive current ramp-up of low-current target plasmas driven by high-harmonic fast wave (HHFW) and/or neutral beam injection (NBI) heating and current drive.*

Background:

Plasma current ramp-up to conditions compatible with sustained high-performance – without reliance on a solenoid - is an important and challenging research objective for the ST. Current overdrive (from bootstrap and RF sources) is being pursued to provide non-inductive current ramp-up to current values compatible with efficient NBI heating and current drive. In particular, in NSTX Upgrade, the 2nd NBI is predicted to be absorbed by a low-current (~400kA) plasma target and to be capable of ramping-up the plasma current non-inductively to 0.8-1MA. HHFW heating of low current H-mode discharges has achieved high poloidal beta and bootstrap current fractions up to 85% at 250-300kA and central electron temperatures as high as 3keV Higher HHFW heating power and resilience to ELMs will be pursued to optimize heating of plasma targets suitable for NBI ramp-up.

Key Collaboration Opportunities for Start-up and Ramp-up:

During the first few years of NSTX-U operation, emphasis will be placed on re-establishing transient CHI start-up, assessing the effects of increase lithium coverage on CHI start-up, utilizing ECH heating of CHI target plasmas (resources permitting), time-dependent modeling of start-up and ramp-up, and developing RF-heated 100% non-inductive target plasmas for testing HHFW and NBI current ramp-up. Key collaboration opportunities include:

- *Perform simulations and contribute to design activities for optimizing CHI start-up*
    - *Simulate ECH heating of low-current/low-density CHI start-up plasmas, and for electron Bernstein wave (EBW) heating and current drive in advanced scenarios. For more information on physics design performed to date for ECH heating of CHI plasmas, please refer to the presentation at this link.*
    - *Perform 2D and 3D MHD and transport evolution simulations to interpret NSTX CHI plasma formation results, and project to NSTX-U and next-step STs including the effects of auxiliary heating – in particular from ECH and HHFW.*
    - *Diagnose and analyze CHI equilibrium evolution utilizing the new divertor poloidal field coil set of NSTX-U.*
- *Simulate, analyze, and perform experiments in support of optimizing HHFW heating and current drive in low-current/low-density target plasmas prototypical of plasmas formed by CHI start-up*
    - *Emphasize projected antenna performance, edge/SOL/divertor power losses, and outer-gap control requirements for efficient HHFW heating.*

VI. Advanced Operating Scenarios – the physics synergy of external control and self-organization of the plasma.

For more information contact: Stefan Gerhardt (sgerhardt@pppl.gov)

Research Priorities:

VI-1. *Develop techniques for advanced plasma control in support of advanced operating scenarios in NSTX Upgrade.*

Background:

High non-inductive current fraction (65-70%) has been sustained for up to 3 current redistribution times in NSTX by operating with high pressure-gradient-driven current fraction in the range of 35-55% and NBI current fraction of 10-30%. In NSTX Upgrade, the decreased collisionality and more tangential 2nd NBI are together projected to enable 100% non-inductive current drive at plasma currents of 0.8-1MA. Sustaining and optimizing these high-performance scenarios will require advanced plasma control including controlling some profiles (such as the rotation and current profiles) and measuring other profiles in real-time to improve real-time reconstructions of the plasma beta and in the longer term enabling real-time stability calculations. For high plasma current (~2MA) operation at high heating power (10-15MW), the projected high peak divertor heat flux on the inertially cooled divertor PFCs could limit pulse-durations to 1-2 seconds, and real-time control of divertor heat flux mitigation (flux expansion, strike-point sweeping, radiation, detachment) will become increasingly important.

VI-2. *Use existing diagnostics for the identification of disruption onset and/or MHD precursors for potential use in triggering controlled plasma shut-down and/or disruption mitigation techniques.*

Background:

NSTX has an extensive set of real-time in-vessel MHD diagnostics for the measurement and active feedback control of error fields, locked modes, and resistive wall modes. NSTX initiated and NSTX-U will utilize disruption mitigation using massive gas injection at several poloidal locations including the outboard mid-plane and the divertor region. Additional safe plasma shut-down algorithms are planned to be developed in NSTX Upgrade. Further, progress has been made in identifying combinations of offline measurements that can provide good disruption onset detection (e.g. 2% missed rate with 4% false positive rate). While NSTX Upgrade is designed to withstand high current and high stored energy disruptions, the potentially deleterious effects of disruptions are best avoided, and such disruption avoidance and machine protection techniques will be needed for ITER and future FNSF devices.

Key Collaboration Opportunities for Advanced Operating Scenarios:

During the first few years of NSTX-U operation, emphasis will be placed on re-establishing baseline operating scenarios, extending scenarios to higher field, current, and pulse length, demonstrating 100% non-inductive current drive for at least a current redistribution time, optimizing axisymmetric control (especially power exhaust and current and rotation profile control), and implementing disruption detection and safe ramp-down scenarios. Key collaboration opportunities include:

- *Participate in scenario development for NSTX-U – emphasizing long-pulse/high-non-inductive fraction operation, and also high current and power operation.*

- *Develop diagnostics and control algorithms and perform experiments and analysis supporting real-time control of divertor heat-flux and PFC temperature for high-power and high-current operation in NSTX-U.*

    - *Also contribute to the development of rotation and current profile control.*

- *Implement real-time versions of existing diagnostic signals and/or develop plasma control system algorithms to detect the onset of plasma disruptions for triggering safe-shut-down and/or disruption mitigation systems.*