# NSTX Upgrade Program Letter for Research Collaboration by Universities and Industry for FY 2014-2017

## Introduction

This NSTX Upgrade (NSTX-U)[1] Program Letter provides updated information about NSTX-U research priorities and collaboration opportunities during the upcoming four years (FY2014-2017). This information is useful for the preparation of proposals in response to the Office of Fusion Energy Sciences Notice: *National Spherical Torus Experiment: Collaborative Research on Configuration Optimization* issued August 2013. New and continuing collaboration proposals from U.S. universities and industry are the primary emphasis of this 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-U Five-Year Plan for 2014-18[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 July 30, 2013 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 strategic goals of the Office of Fusion Energy Sciences[12]: developing predictive capability and designing/deploying materials needed to support a burning plasma environment. In support of these goals, the NSTX-U facility has been judged as "absolutely central" as described in the recent FESAC report on the "Prioritization of Proposed Scientific User Facilities" for FES for 2014-2024.[13]

[1] http://nstx-u.pppl.gov
[2] http://nstx.pppl.gov/DragNDrop/Five_Year_Plans/2014_2018/chapter_text/full_text/NSTXU_5YearPlan_text.pdf.
[3] http://nstx-u.pppl.gov/program/program-advisory-committee/pac-33
[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 9, 2013).
[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://science.energy.gov/fes/
[13] http://science.energy.gov/~/media/fes/fesac/pdf/2013/FESAC_Facilities_Report_Final.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 $\beta_T$ (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 $\beta_T$ 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, and the NSTX-U research team carries out a substantial number of the ITPA joint experiments and/or joint data analysis activities each year. Example ITPA joint experiment / activity topics that particularly benefit from access to the unique ST parameter regime include: studies of the beta dependence of confinement, the shape and/or aspect ratio dependence of the L-H threshold and H-mode pedestal transport and stability, the relative importance of thermal and energetic ion kinetic effects in resistive wall mode stability, aspect ratio and rotation shear effects in neoclassical tearing stability, increased/varied drive for fast-ion instabilities, and potential aspect ratio or beta effects on disruption precursors, detection/warning, and dynamics. Potential NSTX-U collaborators are strongly encouraged to propose and participate in these and other 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 2014-2017 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.

**Upgrade Summary and Near Term Schedule:** The NSTX Upgrade involves an extended outage period which began in October 2011, and operation is expected to resume in late fall of 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 pulse 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 is being 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 $2^{nd}$ more tangential NBI. Several diagnostic ports have been modified, and port access may also be impacted by additional structural enhancements.

NSTX-U capabilities during the first few years of operation are presently planned as follows:

- It is expected NSTX-U will 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 (2015).

- Field and current values up to 100% higher than NSTX (i.e. up to 1T and 2MA) are expected 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 operational years of NSTX-U relevant to this letter (2015-2017), the new 2$^{nd}$ NBI should be available for the entire period.
- A new second switching power amplifier for independent control of all 6 mid-plane 3D field coils will be commissioned by the end of the first full run year.
- The divertor and first-wall will utilize graphite PFCs (although a row of molybdenum tiles on the lower outboard divertor is under consideration for 2016).
- Lithium coating coverage will be extended to the upper divertor and first-wall by 2016.

Additional major facility enhancements have been proposed as part of the NSTX-U five year plan which can only be implemented if sufficient resources are available. These enhancements include:

- A 1MW, 28GHz gyrotron for ECH/EBW heating and current drive has been proposed to be available by the end of 2017.
- A lower outboard divertor cryo-pump for improved density control has been proposed, and would be available by the end of 2017.
- A partial set (6 to 12 coils) of off-midplane in-vessel non-axisymmetric control coils (NCC) has been proposed to be available by the end of 2017.

Note that the cryo-pump and NCC would nominally be installed during an extended outage that would be completed in FY2017.

**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. It should be noted that proposals will be evaluated based on overall scientific quality at least as much as their relevance to the priorities indicated in this Letter. 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-U_Record_of_Discussion_FY2014.doc
**RoA:** http://nstx.pppl.gov/DragNDrop/Program_PAC/Collaborations/NSTX-U_record_of_agreement_Sept2012.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 (including new in-vessel coils) is encouraged, as are understanding disruption thermal quench and halo current evolution, and the development of disruption avoidance and mitigation techniques. For Multi-Scale Transport Physics research, collaboration supporting improved understanding of confinement scaling, electron and ion thermal, momentum, and impurity and particle transport at higher field and current are encouraged. For Plasma Boundary Interfaces research, collaboration on pedestal structure analysis and control, scrape-off-layer heat flux and particle control utilizing advanced divertors, lithium vapor shielding, and material migration are encouraged. For research in Waves and Energetic Particles, collaboration on optimizing fast-wave (and possibly ECH/EBW) 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, experiments and simulations supporting Helicity Injection (HI) current formation and studies of current overdrive using wave and NBI heating are encouraged. For Advanced Scenario Development, scenario optimization, real-time diagnostics and algorithms for controlling internal profiles, 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 U.S. national laboratory (including 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 and advance passive and active feedback control to sustain internal and external macroscopic stability.*

**Background**:
Magnetic feedback control with midplane 3D coils was used successfully and routinely on NSTX to control RWMs by detecting dual-field components of the dominant n=1 mode using poloidal and radial

sensors. This capability will be retained on NSTX-U, but advanced capability including multi-mode and state-space control based on plasma response and full eddy current models are desired to improve RWM stability and to maintain high $\beta_N$ and $\beta_N$ / $l_i$ above the no-wall stability limit for longer pulse-lengths. Previous NSTX research has established a new understanding of RWM kinetic stability by making quantitative correlations between experiment and theory/simulation. Rotation and rotation shear are key components to controlling and improving RWM kinetic stability, and both RWM control and rotation control will be more extensively tested utilizing independent control of each RWM coil (enabled by new switching power amplifiers – SPAs) and utilizing the second NBI system in various NSTX-U regimes including low $\nu^*$ and with different fast-ion populations. Rotation profile control can also modify internal mode stability including both internal kink and neoclassical tearing modes (NTMs). Tests of RWM theory with perturbative, self-consistent, and non-linear simulations will be important to achieve predictability for FNSF and ITER.

**Opportunities:**

- Perform experiments, analysis, and simulation to extend sustainable $\beta_N$ and $\beta_N$ / $l_i$ to long pulses via active RWM control. Poloidal and radial magnetic field sensors with toroidal mode resolution up to 3, and midplane coils with 6-indepdent SPAs will be available as actuators. Develop codes and RWM controllers based on self-consistent plasma response and/or full eddy current models.
- Perform experiments, analysis, and simulation to understand RWM kinetic stabilization, and internal kink and NTM stabilization.
    - Study and understand the effects of rotation/rotation shear on external and internal macroscopic stability physics in low $\nu^*$ regimes and with varied fast ion populations.
    - Test and optimize rotation feedback control to improve RWM, kink, and NTM stability.

I-2. *Understand 3D field effects on tearing layer physics and neoclassical transport, and provide the physics basis for 3D field utilization to improve macroscopic stability.*

**Background**:

NSTX research highlighted the importance of plasma response effects in the application of 3D fields to tokamaks. The understanding of resonant and non-resonant error field effects was improved using ideal perturbed equilibria for rotating plasmas, and error-field correction was successfully utilized using midplane coils. However, extrapolations to the next-step devices remain uncertain due to the complexity of parametric dependencies on collisionality and rotation. Locking physics understanding has improved in the core, but other layer dynamics (especially in the plasma edge) remains to be resolved to understand resonant 3D field effects. 3D neoclassical transport, such as neoclassical toroidal viscosity (NTV), is the key concept to understanding non-resonant 3D field effects and magnetic braking. NSTX research led to several NTV experimental and theoretical discoveries, and will be actively used and controlled to modify rotation and rotation shear to test and improve macroscopic stability. NTV physics itself, including kinetic resonance effects (such as superbanana-plateau behavior) should be more deeply investigated to optimize rotation feedback control and achieve predictive capability and optimized 3D field control in FSNF and ITER. Finally, expanded 3D field capability is essential to support better understanding and control of both resonant and non-resonant field components, and a new 3D coil set (non-axisymmetric control coils - NCC) will be designed and analyzed to assess new 3D physics capabilities.

**Opportunities:**

- Investigate resonant and non-resonant error field correction and locking and tearing responses in different $\nu^*$ and rotation. Study and understand layer dynamics by validating linear and non-linear modeling with more extensive SXR and magnetic diagnostics.
- Perform experiments, analysis, and simulations for 3D neoclassical transport and NTV.

  - Develop and validate advanced and more self-consistent NTV codes, and provide the physics basis for the control of rotation feedback and macroscopic stability. The new SPAs and NBIs will be used as actuators for rotation control.
- Assess recently proposed partial and full NCC options with application to error field, NTV, RWM, RMP for ELM control in NSTX-U, and/or study and propose new NCC configurations.

I-3. *Provide stability physics basis for disruption prediction and avoidance, develop techniques for disruption mitigation, and understand disruption dynamics in high-performance ST plasmas.*

**Background**:
Multiple techniques for disruption detection and avoidance were explored in NSTX, and this research will continue on NSTX-U. Particular research success on NSTX was found in the areas RWM control, RFA analysis, and disruption prediction algorithms. In the future, global and kinetic stability physics and control models, with expanded sensor input, will be applied for disruption prediction; real-time implementation of these physical models and measurements will be attempted to improve disruption avoidance. The plasma dynamics resulting from the interplay of different stability controllers will be measured and simulated. In the area of disruption mitigation, it is critical to develop novel techniques of rapidly injecting large quantities of radiating material into the plasma, as this is the process being developed for mitigation in ITER. Examples of novel technologies may include massive gas injection (MGI) from multiple poloidal locations (divertor, off-midplane, midplane) or other means of achieving mass injection. Simulation capability that can produce validated models of these mitigation technologies shall be pursued where possible. NSTX also made significant progress in understanding halo current dynamics, including measurements of halo current rotation in the outer divertor. However, measurement and understanding of the global current structure and dynamics remains to be developed. Measurements of thermal loading during disruptions were not pursued on NSTX, but will be important in NSTX-U for projecting the transient thermal loading during disruptions in next-step STs. The knowledge gained in this research will improve knowledge of the physics and technologies required for disruption prediction, avoidance, mitigation (PAM) system in STs and tokamaks including ITER.

**Opportunities:**
- Investigate and understand stability physics (including simplified/reduced and real-time models of stability physics and boundaries) in order to detect and avoid disruptions. Develop and apply various MHD spectroscopy, observer techniques, and real-time profile controllers.
- Use existing and new mass delivery technology and 3D diagnostics for disruption mitigation. Perform simulations and validate models of the mitigation process using these capabilities.
- Study and understand key transport and stability physics during the disruption process.
  - Study and understand thermal quench physics and transient disruption heat loads.
  - Utilize and validate models for halo currents in NSTX-U.

## 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* (mailto:yren@pppl.gov)

Research Priorities

II-1. *Characterize H-mode global energy confinement scaling in the lower collisionality regime of NSTX-U.*

**Background:**

A wide range of collisionality was obtained in NSTX using two different wall conditioning techniques, one with boronization and between-shot helium glow discharge conditioning (HeGDC+B), and one with lithium evaporation (Li EVAP). Previous studies of HeGDC+B plasmas indicated a strong increase of normalized confinement with decreasing collisionality. Discharges with lithium conditioning further reduced collisionality by 50%. The confinement dependences on dimensional/engineering variables of the HeGDC+B and Li EVAP datasets differed. Collisionality, however, was found to unify the trends, with the lower collisionality Li EVAP discharges also showing increasing normalized confinement time with decreasing collisionality when other dimensionless variables were held nearly fixed. The plan for confinement trend studies in NSTX-U will closely follow the planned engineering capabilities, e.g. using increased $B_T$, Ip, beam power and the addition of cryopumping to progressively increase field and current and decrease the plasma collisionality. NSTX studies have shown that the global energy confinement dependence on collisionality is independent of how collisionality is varied, and this conclusion will be further tested in NSTX-U by employing various techniques to vary collisionality. The resulting global energy confinement scaling will be used to project the confinement performance of ST-FNSF.

**Opportunities:**

- Perform experiments and analysis to determine NSTX-U H-mode plasma energy confinement scaling as a function of engineering parameters (i.e. $B_T$, $I_P$ and beam power) and dimensionless parameters ($\nu_e^*$, $\beta$, Mach number, and $\rho^*$) and compare to NSTX results for a range of operating scenarios and wall conditions.
- Utilize a range of turbulence diagnostics (Beam Emission Spectroscopy (BES), high-k scattering, polarimetry and reflectometry) to identify turbulence most correlated with thermal transport, and compare these measurements to linear and nonlinear gyrokinetic simulations

II-2: *Identify regimes of validity for instabilities responsible for anomalous electron thermal, momentum, and particle/impurity transport in NSTX-U*

**Background:**

Considerable progress has been made in identifying possible mechanisms responsible for the anomalous transport observed in NSTX. In particular, multiple instabilities, both electrostatic (ITG/TEM/ETG) and magnetic (micro-tearing, compressional/global Alfven eignmode), have been identified as potential candidates responsible for anomalous electron thermal transport, which may ultimately limit the confinement performance of future devices. Furthermore, experiment and modeling show that low-k turbulence is likely responsible for observed anomalous momentum and impurity transport whose understanding is important for calculating flow, bootstrap current, and the density profile. Having identified several candidates, the aim of this research priority is to identify the regime of validity for these instabilities. This involves theoretically identifying isolated regimes for the instabilities, experimentally measuring turbulence and transport in these regimes, and then comparing the measured transport levels and turbulence characteristics with first principles simulations or other theoretical model predictions. Experimental parametric dependences can be used for further distinguishing different instabilities. For example, the dependence of microtearing and ETG modes on s/q and $Z_{eff}$ are opposite to each other. The enhanced capabilities of NSTX-U, i.e. increased range of collisionality, doubled heating power from the 2nd NBI and active flow and current profile modification using the 2nd NBI and external and proposed in-vessel NCC coils, provides a versatile set of experimental tools for modifying transport and turbulence which will be valuable for performing this research. The dependence of transport produced by the above instabilities on collisionality, plasma beta and the toroidal angular velocity will be emphasized since these dependences are most relevant for the projection to future devices. Note that the validation of first-principles simulations and turbulence theory against experiments is essential to the achievement of the next goal, II-3.

**Opportunities:**

- Perform electron thermal transport and turbulence studies in L-mode and H-mode to understand the source of anomalous electron thermal transport
  - Utilize polarimetry diagnostics to measure magnetic field fluctuations and determine the relative role of electrostatic and electromagnetic turbulence by comparing the measurements with density fluctuations from BES, high-k scattering system, and reflectometry
  - Perform perturbative experiments, e.g. with cold pulses from gas injection or laser blow-off coupled with Multi-Energy Soft X-ray diagnostics
- 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, improve understanding of momentum transport by turbulence and 3D magnetic fields
  - Utilize higher spatial resolution BES, study intrinsic rotation
  - Carry out cross-device comparisons of momentum and impurity transport
- Participate in experiments and analysis exploring particle/impurity transport from the edge to the core. Assess possible linkages to inward momentum pinch physics utilizing the higher field and current and varied momentum deposition of NSTX-U.
  - Utilize a laser blow-off system to study impurity transport

### *II-3: Establish and validate reduced transport models (0D and 1D)*

**Background:**

It is important for both spherical and conventional tokamaks to develop reduced transport modeling capability to predict plasma profiles, which determine confinement scaling, MHD stability, and current profiles (from bootstrap and NBI/RF driven current). The validated transport models can be used to predict confinement scaling (to compare with results of II-1) and can also be used in self-consistent integrated modeling scenarios to study and develop advanced operating regimes, such as steady-state fully non-inductive scenarios. The highest priority focus is on electron temperature profile as it is potentially influenced by a large number of mechanisms. Ultimately, it is also desired to predict density and toroidal flow profiles self-consistently, as both profiles influence the underlying transport mechanisms. In addition, increasing emphasis will be placed on studying and modeling global, or non-local, effects due to the relatively larger ratio of gyroradius to machine size typical of spherical tori.

**Opportunities:**

- Develop and validate physics-based models of electron transport in the core region of the plasma to develop a predictive capability for NSTX-U and future ST devices.
  - Develop and test reduced drift wave transport models with first principles gyrokinetic turbulence simulations, validating linear thresholds, saturated transport and their parametric dependencies, with an increased emphasis on high beta conditions where microtearing and kinetic ballooning modes are unstable, and also on high Mach number regime
  - Develop and test models for electron thermal transport due to energetic particle modes (GAEs and CAEs), including thresholds, transport, and their sensitivities to the fast-ion velocity space distribution function.
  - Validate transport models with NSTX and NSTX-U data and test the accuracy of transport solvers with priority on predicting electron temperature profiles.
  - Assess the ability to predict density and flow profiles as model fidelity improves.
  - As global turbulence simulations become more readily available, assess the influence of non-local effects.

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

*For more information contact:* Rajesh Maingi (rmaingi@pppl.gov)

Research Priorities:

### III-1. *Characterize, control, and optimize H-mode pedestal transport and stability.*

**Background**:
H-mode pedestal and ELM control includes characterization of the L-H transition, pedestal structure, turbulence, and edge stability in ELMy and ELM-free discharges, extending studies done in NSTX to lower $\nu^*$, while making use of new capabilities in NSTX-U. The lower $\nu^*$ is expected to affect the pedestal in several ways, e.g. 1) the bootstrap current will be increased; 2) the intermediate collisionality enhancement of bootstrap current relative to the Sauter formulation will decrease; 3) possible re-ordering of which micro-instabilities (micro-tearing vs. hybrid Trapped Electron Mode (TEM)/Kinetic Ballooning Mode (KBM) vs. Electron Temperature Gradient (ETG)) dominate pedestal transport; 4) possible re-ordering of the importance of 2-D particle transport vs. 3-D expected particle transport with application of magnetic perturbations, which could lead to density pump-out in NSTX-U, which was never observed in NSTX H-mode discharges. Pedestal control will be pursued using ELM pace-making via either pulsed 3-D fields or injected granules or pellets, as well as access to advanced/alternate edge stability regimes.

**Opportunities:**

- Perform experiments, analysis, and pedestal transport and stability simulations to assess the maximum achievable pedestal height and variation in pedestal structure as a function of increased $I_P$, $B_T$, and $P_{heat}$ available in NSTX-U, and measured with enhanced spatial and temporal resolution profile and turbulence diagnostics, toward the development of an ST pedestal structure model for projection to FNSF and ITER.
- Assess ELM triggering and suppression with 3D fields from mid-plane (existing) and off-midplane Non-axisymmetric Control Coils (staged implementation planned), and also triggering with lithium granule injection. Understand regimes devoid of large ELMs, e.g. Enhanced Pedestal H-modes, I-modes, and Quiescent H-modes.
- As high-Z PFCs are introduced, assess their impact on H-mode pedestal performance.

### III-2. *Control divertor heat and particle fluxes, and recycling with a combination of innovative and proven techniques.*

**Background**:
Heat flux scaling experiments in NSTX and other devices have identified a strong inverse dependence of the extrapolated midplane heat flux width, $\lambda_q^{mid}$, on $I_P$, with no dependence on $B_T$ or input power, with projections indicating that heat flux mitigation would be needed for achievement of full $I_P$, full $B_T$, long pulse NSTX-U plasmas. Fortunately, the divertor heat flux magnitude was shown to decrease approximately inversely with flux expansion, including for snowflake/X-divertor configurations, as well as conventional radiative divertor operation. With respect to particle control, integrated scenarios on NSTX-U are designed for steady density at 0.5-1 times Greenwald density limit scaling. To achieve this, a central element of the boundary program is installation of a divertor cryo-pump, representing a proven technology to control both main ion and impurity density. A key component of the research is to compare lithium pumping and cryo-pumping for density control, while considering the possible synergy of these two techniques with high flux expansion and radiative divertor operation to contribute to the required power and particle exhaust solution for future devices and for NSTX-U itself.

**Opportunities**:

- Perform experiments, analysis, and edge transport and turbulence simulations to investigate the SOL heat and particle transport and turbulence and associated widths, extending the existing NSTX database to lower $\nu^*$ and higher $I_P$ and $P_{SOL}$ toward projections for FNSF. These studies should include both the inter-ELM and ELM phases of the ELM cycle. Comparisons of snowflake and other advanced divertor configurations with standard divertors are also encouraged.
- Perform experiments, analysis, and edge fluid and/or kinetic plasma/neutral transport simulations to validate cryo-pump physics design activities, perform initial density control studies, and assess compatibility with H-mode pedestal and core performance and divertor power exhaust scenarios.

III-3. *Develop lithium surface science for long pulse PFCs, including continuous vapor shielding.*

**Background**:

Lithium wall conditioning has led to increased discharge performance in NSTX. However, key questions on the plasma – lithium interaction remain, particularly regarding the role of impurities (e.g. B, O, C) on the performance of lithium conditioned plasmas. It is expected that the plasma facing components of NSTX-U will remain predominantly graphite during the next 5 year period. However, a subset of the divertor tiles may be converted to high-Z (W, Mo, or TZM), so there is interest in determining how both plasma and PFC performance might change when lithium is applied to a high-Z substrate. Tokamak experimental studies and surface-science laboratory studies will continue to address these issues. It is further noted that lithium has a significant evaporation rate at modest temperatures, leading to questions about maximum operating temperatures in long-pulse scenarios. Interestingly, experiments in FTU at elevated temperatures ~550˚C showed a strong evaporation into the local plasma, simultaneously with reductions in limiter heat fluxes. This suggests a potentially attractive operating regime may exist where strong local evaporation leads to a continuously vapor-shielded PFC. An experimental determination of the ultimate temperature limits for a liquid lithium PFC in the diverted configuration with good core performance is envisioned.

**Opportunities**:

- Perform experiments, analysis, and simulations to advance lithium surface science to assess NSTX/NSTX-U results and the extrapolability to future devices
    - Perform simulations (for example quantum-classical molecular dynamics (QCMD) modeling) to understand the interactions between graphite and/or high-Z substrates and lithium surface coatings including mixed material effects.
    - Utilize surface analysis diagnostics to identify in-situ between-shot chemical states and compositions of the coatings, and connect these results to stand-alone surface-science studies and to possible changes in plasma performance – for example changes in wall recycling and/or plasma confinement and stability.
    - Improve understanding of the role of more complete coverage of the PFCs by evaporated lithium using upward-facing evaporators, and/or diffusive evaporation, as well as the impact of boronization.
- Establish the scientific basis and experimental demonstration of a continuously vapor-shielded surface in the tokamak environment with application to heat-flux mitigation.
    - Assist in the design and deployment of a lithium-coated high-Z substrate (for transient operation) or a flowing liquid lithium system (for long-pulse operation) with a high-power-density strike-point impinging on the PFC to raise the front-face temperature above the lithium evaporation temperature

III-4. *Unravel the physics of tokamak-induced material migration and evolution.*

**Background**:
The edge plasma of fusion devices continuously interacts with incoming neutral atoms resulting in a charge-exchange flux of high-energy neutrals that will impinge and sputter the first-wall, limiting its lifetime. Wall erosion is projected to result (for example) in thousands of kg per year of circulating material in a fusion power reactor. The eventual fate of the eroded material is presently unknown, and requires further study. An extensive set of spatially- and time-resolved photometrically calibrated spectroscopic measurements will be available on NSTX-U for evaluation of impurity fluxes. To interpret the measurements, knowledge of accurate atomic physics factors, such as ionizations per photon and photon emission coefficients, is needed for common NSTX-U impurities originating from plasma-facing component materials, surface contamination, or those seeded externally. The high power densities and particle fluxes achievable in NSTX-U should provide readily measurable erosion, and this is important for both understanding NSTX-U impurity generation and for projecting to future devices.

**Opportunities:**
- Perform experiments, analysis, and simulations to assess shot-to-shot erosion and re-deposition including (for example) interpretation of microbalance diagnostics. Evaluate the impact of high-Z PFCs on erosion and migration as such PFCs are introduced into NSTX-U.
  - Utilize the extended ranges of P/R and P/S in NSTX-U for model comparison, as a step toward projection of total wall erosion for an FNSF or DEMO.
  - Evaluate the material transport variations as a function of divertor configuration (e.g. high flux expansion, detached, vapor-shielded), and compare with simulations.

Lastly, it is noted that improved atomic physics models and calculations for atomic structure and rate coefficients would be beneficial for supporting research priorities III-2, III-3, and III-4 listed above.

**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. *Develop capability to heat high-power neutral beam heated H-mode plasmas with fast waves and assess fast wave interaction with fast-ions.*

**Background:**
High-harmonic fast wave (HHFW) coupling and heating studies on NSTX have identified a major loss of RF power that occurs along open field lines that pass in front of the antenna over the width of the scrape-off layer (SOL), and this RF power loss mechanism can significantly reduce the effective core heating efficiency. The fast wave heating efficiency was found to depend on the location of the critical electron density for the onset of perpendicular fast wave propagation ($n_{ec}$). 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 allowed fast waves to reliably heat electrons in deuterium H-mode plasmas in NSTX. These performance improvements motivated an upgrade to the 12-strap HHFW antenna system to a configuration with an RF feed at the top and bottom of each strap and a ground in the middle of the strap, instead of one RF feed at the top and a ground at the bottom. This upgrade was implemented to reduce the voltage on the antenna needed for a given coupled power. Record ST central electron temperatures (above 6

keV) have been achieved using the upgraded fast wave antenna, but the performance of the antenna upgrade has never been evaluated with little or no lithium coating present and following a boronization. Fast wave acceleration of neutral beam injection (NBI) fast-ions has previously been observed in NSTX plasmas, and additional research is needed to understand and minimize interactions between fast waves and NBI ions in conditions with reduced fast wave power losses in the SOL. In NSTX-U the increased magnetic field strength will enable greater exploration of the effects of the location of $n_{ec}$, while more detailed diagnostic measurements will confirm whether or not strong RF fields are present in the SOL. Further, SOL density control tools such as lithium coatings and divertor cryo-pumping will influence the location of $n_{ec}$. In NSTX-U the higher toroidal field is expected to further reduce RF power losses in the SOL 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 that could decrease fast wave coupling efficiency and increase RF power loss in the SOL.

**Opportunities:**

- Perform experiments, analysis, and full-wave and/or ray-tracing simulations to assess the performance of the fast wave antenna after boronization and with optimized lithium coatings needed to improve coupling over a wide range of antenna phase, $I_p$, $B_t$ and $n_e$ utilizing the suite of improved and upgraded diagnostics (RF probes, IR cameras, and fast-ion diagnostics).
    - Evaluate, study, attempt to mitigate RF power flows in the SOL and to the divertors in the H-mode regime. Test RF code predictions for RF power flows in the SOL using data from the upgraded diagnostics that will measure RF fields and power flows near the antenna, in the SOL and on the divertor plates.
    - Measure and simulate fast wave interactions with NBI fast-ions at higher field and plasma current
    - Simulate reduced-strap (12 → 8 straps) fast wave antenna performance to assess the potential impact of providing wall space for other *AE antennae and/or edge-harmonic-oscillation (EHO) excitation systems.

## IV-2. *Develop fast wave, electron cyclotron, and electron Bernstein wave heating for non-inductive plasma current start-up, ramp-up and sustainment.*

**Background:**

Fast wave experiments in NSTX have demonstrated that as little as 1.4 MW of fast wave power can generate and sustain an H-mode discharge with a plasma current of 300 kA and a non-inductive plasma current fraction of 0.7 – 1. Fast wave heating experiments planned for NSTX-U will use much higher RF power and are predicted to demonstrate fully non-inductive plasma current ramp-up via bootstrap current overdrive. A high-priority facility enhancement on NSTX-U (contingent on funding) is the installation of a megawatt-level 28 GHz heating system for non-inductive plasma start-up research. This heating system would initially be used for electron cyclotron (EC) heating of plasmas started by coaxial helicity injection (CHI), and longer-term for electron Bernstein wave (EBW) plasma startup and off-axis current-drive. To support the optimal utilization of high-power ECH/EBW, an advanced microwave EBW imaging diagnostic will be deployed on NSTX-U to assess the EBW coupling efficiency in various NSTX-U plasma regimes. These emission measurements will provide important data that will guide the design of a multi-megawatt off-axis EBW heating and current drive system. Earlier EBW emission experiments on NSTX clearly demonstrated that the EBW mode conversion efficiency can be significantly improved in the H-mode regime by using lithium wall coatings to mitigate RF power losses in the SOL that occur as a result of collisions near the EBW mode conversion layer. At the higher toroidal field in NSTX-U it is expected that the collisional EBW damping will be lower than on NSTX and the EBW coupling efficiency correspondingly higher. Longer-term, the effects of cryo-pumping on the SOL density and EBW emission, heating, and current drive will also be assessed.

**Opportunities:**

- Perform experiments, analysis, and wave damping and current-drive simulations to assess the viability of fast wave heating to support fully non-inductive plasma current ramp-up, and EC and EBW heating to support non-inductive plasma start-up. Also use EBW emission measurements and simulations of EBW heating and current drive to design an O-X-B heating system.
  - Simulate, generate, study fully non-inductive fast-wave-heated H-mode discharges.
  - Simulate 28 GHz EC and EBW heating for non-inductive start-up and design the 28 GHz heating system.
  - Conduct EBW emission measurements with an advanced synthetic aperture microwave imaging (SAMI) diagnostic to assess O-X-B coupling efficiency.
  - Simulate O-X-B heating and current drive in NSTX-U advanced H-mode discharges.

IV-3. *Improve the ability to predict fast ion driven instabilities and the resulting fast ion dynamics using the expanded parameter space of NSTX-U.*

**Background:**

A primary goal for Energetic Particle research on NSTX-U is to develop capabilities that enable reliable and *quantitative* predictions on properties of - and fast ion response to - unstable modes in future devices such as ITER and FNSF. For instance, although good progress has been made in understanding linear stability thresholds, self-consistent and reliable predictions of Alfvénic mode properties at saturation (e.g. stationary vs. bursting dynamics) still require extensive work from both experimentalists and theory/code developers. EP research on NSTX-U will benefit from the expanded range of plasma regimes enabled by the twofold increase in toroidal magnetic field and the addition of a second NBI system with more tangential injection. Knowledge of the fast ion distribution resulting from different NBI geometries (from more perpendicular to more tangential) is crucial to understand the device performance in terms of NB current drive and heating efficiency and the stability of the plasma to fast-ion-driven modes. When used in combination with NBI, RF injection can also result in large deviations of fast ion behavior from what is predicted by classical models. An initial characterization of the fast ion distribution performed in MHD-quiescent NSTX-U plasmas will be extended to higher performance scenarios, in which either RF injection or instabilities are expected to cause a departure of the fast ion behavior from classical. In this scenario, modes that - based on previous results from NSTX and other devices - can induce substantial redistribution and/or loss of fast ions will be targeted first. These include TAEs, Reverse-Shear AEs (RSAEs) and Energetic Particle modes (EPMs), as well as higher frequency Global/Compressional AEs. Experimental studies will benefit from the suite of fast ion diagnostics installed on NSTX-U. Two Fast Ion D-Alpha (FIDA) systems will provide time, space and energy resolved measurements of the fast-ion distribution; a vertical FIDA system more sensitive to trapped or barely co-going particles and a tangential FIDA system that measures co-passing fast-ions. FIDA measurements will be complemented by an upgraded solid-state Neutral Particle Analyzer array that measures trapped fast-ions. Additional diagnostics supporting studies of fast-ion dynamics include neutron rate counters and a scintillator-based Fast Lost Ion probe.

**Opportunities:**

- Support operation of, and analysis of data from, diagnostics for measurements of the fast ion distribution, and compare data to classical model predictions (NUBEAM/TRANSP)
- Characterize instabilities affecting fast ion confinement, with emphasis on Alfvénic mode frequency, mode structure and temporal dynamics.
  - Assess the impact of fast ion transport and redistribution by *AEs on NB current drive and assist in 100% non-inductive scenario development.
  - Perform experiments and data analysis to understand the saturation level of Alfvénic instabilities and underlying saturation mechanisms.

- Perform linear and non-linear *AE simulations for NSTX-U scenarios and compare with experiments.
  - Compare measured mode properties ($\omega$, k spectrum, radial structure) with predictions from both linear and non-linear codes such as NOVA, M3D-K, HYM.
  - Study transition from single-mode to multi-mode scenarios as a function of NBI and plasma parameters; compare with experiments.
  - Study the dependence of *AE saturation levels on specific mode regimes (e.g. for stationary vs. bursting/chirping modes).
  - Compare simulations of AE avalanches with predictions from numerical codes.

## 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)

V-1. *Develop helicity injection start-up with Coaxial Helicity Injection (CHI) and prepare for the use of point-source helicity injection.*

**Background:**
Coaxial helicity injection (CHI) has demonstrated transformer flux savings equivalent to approximately 200kA 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 experiments and modeling with the goal of further increasing the helicity injection current. In NSTX Upgrade, helicity injection current drive is projected to scale favorably (linearly) with toroidal field and poloidal flux supplied by the injector coils. The toroidal field and injector flux will increase two-fold in NSTX-U. Thus, the generation of 400kA of closed-flux plasma current should be achievable and is projected to provide a plasma current suitable for NBI heating and current ramp-up. Results from point-source helicity injection on the Pegasus toroidal facility are also encouraging, and the use of such “plasma guns” that could be withdrawn after plasma initiation may be advantageous for an ST-FNSF.

**Opportunities:**
- Perform experiments, analysis, and simulation with the goal of increasing the closed-flux plasma current generated by helicity-injection current on open field lines.
  - Diagnose and analyze plasma equilibrium of CHI initiated discharges using higher toroidal field, the new injector coils, and higher injector voltage ($\leq$ 2-3 kV)
  - Provide simulations and experimental data (from other devices) in support of the design and installation of plasma guns for point helicity injection on NSTX-U.
  - Perform 2D and 3D simulations of the equilibrium evolution, MHD stability, and transport evolution for helicity injection plasma initiation to improve predictive capability for extrapolation of plasma-start-up, and to interpret experimental CHI plasma formation results.

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)*

**Background**:
Plasma current ramp-up to conditions compatible with sustained high-performance – without reliance on a solenoid – is a critical research objective for the ST. Current overdrive (from bootstrap and RF) is being pursued to provide non-inductive current ramp-up to current values compatible with efficient NBI

absorption, heating, and current drive. In particular, in NSTX Upgrade, TRANSP/TSC simulations indicate the new 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.

**Opportunities:**

- Perform experiments, analysis, and simulation for plasma current overdrive. In particular, time-dependent simulations of the impact of early auxiliary electron heating and the dynamics and stability of the plasma current ramp-up phase would be especially valuable. Such modeling could inform the choice of fueling timing and rate, heating rate, current-ramp rate, and choice of NBI source to optimize the current over-drive ramp-up phase. Comparisons between such modeling and experimental data are strongly encouraged.
- Experimentally investigate the effects of energetic particle modes, *AE modes, and other plasma MHD instabilities during current-overdrive experiments using HHFW and NBI, and support with data analysis and modeling.

V-3. *Develop electron heating of low-$I_P$ start-up targets using HHFW and ECH (if available) for both low-density and over-dense plasmas.*

**Background:**

Present modeling indicates that for efficient neutral beam current drive, the plasma electron temperature must be above about 1 keV, and above approximately 200 eV for efficient HHFW coupling and heating. It is 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 and current drive. 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. Furthermore, installing an ECH system (resources permitting) could also be useful for ST electron heating using EBW and ultimately potentially for off-axis EBW current drive.

**Opportunities:**

- Perform experiments, analysis, and wave heating and current drive modeling for heating low-$I_p$, low-$T_e$ plasmas to ~1keV
  - Simulate and experimentally vary the plasma and HHFW heating parameters to determine the ranges of plasma density and target $T_e$ that can be heated to 1 keV.
  - Model the heating of very low temperature plasmas (30 eV) by 28 GHz ECH and HHFW for both low-density and over-dense plasmas.
  - If an ECH system becomes available, perform ECH heating experiments and determine the heating efficiency as a function of initial plasma density and temperature.

# VI. **Advanced Operating Scenarios and Control** – 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, simulate, and implement advanced control algorithms in support of high-performance operating scenarios in NSTX-Upgrade.*

**Background:**
Control algorithm development in NSTX-U will both support the research program and improve the understanding of key physics for next-step ST devices. For example, the development of rotation and q profile algorithms are needed to access and sustain high plasma stability and confinement and to facilitate a wide variety of physics studies in energetic particle physics, transport, and global stability, as well as improve the general understanding of momentum transport and current drive. Further, given the higher aspect ratio and elongation of NSTX-U, more advanced algorithms for vertical position control will be required. Given the large heat fluxes at high current and heating power in NSTX-U, it is likely that advanced control of the power exhaust will be required to fully utilize the facility. This control could include magnetic control of multi-X-point divertors, or development of algorithms for controlling the divertor surface temperature and stabilizing the detachment front under impurity seeding. Algorithms will be required to both predict impending disruptions and automate the control response once a disruption is declared to be imminent. 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.

**Opportunities:**

- Contribute to the development of rotation and current profile control to access and sustain high confinement and high $\beta$ and to facilitate a range of physics studies in the areas of transport, energetic particle MHD, and global stability.

- Participate in scenario development for NSTX-U emphasizing control of high-elongation scenarios which may challenge vertical stability in NSTX-U. In addition, contribute to the development of control of strongly-shaped multiple x-point equilibria typical of snowflake and/or X-divertor configurations.

- Participate in implementation of real-time 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.

- 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.