NSTX Program Letter for Research Collaboration by Universities and Industry for FY 2011-2013

Introduction

This NSTX[1] Program Letter provides updated information about NSTX topical research priorities and collaboration opportunities during the upcoming three years (FY2011-2013). This information is useful for the preparation of proposals in response to the Office of Science Notice *National Spherical Torus Experiment: Collaborative Research on Configuration Optimization* issued in August 2010. New and continuing collaboration proposals from U.S. universities and industry are the primary emphasis of this Program Letter. This Program Letter suggests specific collaboration opportunities, as well as broader areas of research, in order to encourage proposals that address the research goals of NSTX. These research areas are described in the NSTX Five-Year Plan for 2009-13[2] and the FESAC Facilities Report[3]. In the Appendix to this Letter, these areas are cross-referenced to the ten-year goals in the FESAC Priorities Report[4] as background information. The NSTX Program Advisory Committee (PAC) reviewed this Program Letter on July 20, 2010 and the PAC recommendations were incorporated in this final version.[5]

Mission of NSTX

The *programmatic* mission of NSTX 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, NSTX Upgrade[6], and the mission of 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 also contributes to the physics basis for an ST-based DEMO device and accesses unique plasma regimes for resolving key burning plasma physics issues anticipated in ITER. The NSTX 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.[10]

[1] http://nstx.pppl.gov/index.html
[2] http://nstx.pppl.gov/DragNDrop/Five_Year_Plans/2009_2013/NSTX_Research_Plan_2009-2013.pdf.
[3] http://www.ofes.fusion.doe.gov/more_html/FESAC/FacilitiesVol1.doc and FacilitiesVolume2_v3.pdf.
[4] http://www.ofes.fusion.doe.gov/more_html/FESAC/pp_Rpt_Apr05R.pdf
[5] http://nstx.pppl.gov/nstx/NSTX_Program_Letter (available July 29, 2010).
[6] http://nstx.pppl.gov/upgrade_overview.html
[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://www.sc.doe.gov/bes/archives/plans/SCSP_12FEB04.pdf.

In support of the above programmatic mission, the *scientific* mission of NSTX 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.

NSTX Research Priorities and Key Collaboration Opportunities

This section lists the research topics of high priority in the NSTX Program during FY 2011-2013 and then highlights key collaboration opportunities for which research proposals are solicited. The NSTX research priorities and key collaboration opportunities are organized according to the six categories used in the FESAC Priorities Plan. **For each category, a key person is listed who should be contacted for further information prior to submitting collaboration proposals**.

The list of research priorities is guided by the NSTX programmatic and scientific missions and by its schedule of milestones, subject to anticipated funding during FY 2011-2013. Recently completed upgrades include an improved high-harmonic fast-wave (HHFW) system for higher heating power compatible with H-mode, and a Liquid Lithium Divertor (LLD) for enhanced particle pumping and Lithium wall conditioning. A Beam Emission Spectroscopy (BES) diagnostic for ion-gyro-scale turbulence and Alfvén eigenmode displacement profile measurements has also recently been commissioned. Additional upgrades will include higher spatial resolution measurements of the electron pressure profile in the H-mode pedestal, an LIF-based MSE system for measuring magnetic field pitch angle and |B| without the primary heating beam (thereby enhancing fast-wave and energetic particle research and scenario optimization), and a 2nd switching-power amplifier for improved control of resonant and non-resonant magnetic perturbations, rotation damping, resistive wall modes, and error-fields.

For each of the six scientific categories described below, the numbered research priorities are listed in priority order. To fully exploit the new research tools described above, proposals addressing high priority research areas including understanding the impact and operation of the LLD, understanding H-mode pedestal structure, developing a predictive capability for ion-gyro-scale turbulence and associated transport, and the utilization of the upgraded HHFW system and CHI for scenario optimization and current ramp-up are strongly encouraged. Emphasis on the utilization of simulation in the analysis, interpretation, and planning of experiments is encouraged, especially in the areas of Multi-scale Physics and Plasma Boundary Interfaces. Research on advanced divertor configurations for improved power-handling, density control, reduced impurity production for NSTX and especially NSTX Upgrade is also encouraged. The collaboration opportunities highlighted below were determined on the basis of what the NSTX program considers necessary to support the research priorities while also complementing ongoing contributions from diagnostic support collaborations, national laboratories, and PPPL researchers. Proposals for innovative collaboration activities beyond the ones listed in this Program Letter are also welcomed; such proposals might be motivated by the list in Appendix A. 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 Plasma Physics* – the role of magnetic structure in plasma confinement and the limits to plasma pressure in sustained magnetic configurations.

For more information contact: Jon Menard (jmenard@pppl.gov)

Research Priorities:

I-1. *Determine the physics of RWM stabilization, by both passive and active means, and apply this understanding to reliably sustain high-beta, low-aspect-ratio, broad current profile plasmas.*

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. More advanced control algorithms and enhanced predictive capability (especially the effects of fast-ions on RWM stability) are sought to understand present RWM and EF results and allow extrapolation of high performance to ITER and future STs. In particular, the high-β, near-Alfvénic toroidal flow velocities from NBI, the substantial population of energetic particles, and the ability to control the toroidal flow velocity with magnetic braking, and the strong toroidicity of NSTX provide an excellent test-bed for validating models of RWM active and passive stabilization.

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

Background:
Neoclassical tearing 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. Overall, strong rotation and rotational shear, enhanced stabilization effects from curvature and large ρ^*, and enhanced poloidal mode coupling may all play a significant role in modifying tearing mode stability in NSTX.

I-3. *Characterize and mitigate the effects of disruptions in the ST.*

Background:
NSTX has contributed plasma current quench rate 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 research is 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. Disruption mitigation techniques are essentially unexplored in the ST. 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. Novel techniques for controllable, rapid, and deeply penetrating gas/plasma delivery are sought.

Key Collaboration Opportunities in Macroscopic Plasma Physics:

- *Exploit particle pumping from Li / LLD, fast-wave heating and NBI heating of high-beta H-mode plasmas, and magnetic braking to vary the plasma temperature and density, fast-particle content, and momentum input to improve understanding of RWM stability as a function of core electron/ion collisionality, fast-ion beta, and toroidal rotation and rotation shear.*

- *Aid in the analysis and simulation of neoclassical tearing mode stability emphasizing improved understanding of the physics of rotational shear stabilization and triggering of the NTM, and including the effects of externally applied 3D fields.*

- *Aid in the analysis and simulation of 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.*

- *Aid in the development of real-time control of the plasma rotation using NBI and magnetic braking to enable controlled variations of the plasma rotation for RWM and NTM stability, confinement research, and as a potential means to avoid disruptions.*

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

For more information contact: Stan Kaye (skaye@pppl.gov)

Research Priorities:

II-1. *Determine the role of low-k turbulence in causing anomalous 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 effective angular momentum diffusivity is routinely significantly smaller than ion energy diffusivity but larger than neoclassical predictions. These results are qualitatively different than those commonly obtained in higher aspect ratio tokamaks, and may be the result of increased ExB flow shearing rates achievable in rapidly spinning low aspect ratio, high β NSTX plasmas. Further, recent 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 and lithium conditioning have 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.

II-2. *Determine the modes (low-k, high-k, electrostatic, electromagnetic, Alfvénic) responsible for causing anomalous electron transport.*

Background:
NSTX results indicate that ion energy and particle transport levels are routinely at the neoclassical level, implying suppression of long-wavelength turbulence and associated anomalous transport. 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 the possible correlation between increased high-k fluctuation amplitude and increased electron thermal transport is being investigated. Micro-tearing and Global Alfvén Eigenmodes (GAE) have also been correlated with anomalous electron transport in NSTX, and Collisionless Trapped Electron Modes (CTEM) will also be studied.

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

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. Recent NSTX results indicate that lithium evaporated onto the lower divertor surfaces of NSTX can act as an effective (albeit transient) pump of hydrogenic neutrals, and the liquid lithium divertor (LLD) is expected to significantly enhance this pumping. Impurity transport has been measured to be nearly neoclassical in NSTX and exhibits a particle pinch. 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, the dependence of particle diffusivity on global plasma parameters will be determined, and the possible relationship between anomalous 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 particle transport and measured and simulated turbulence will be pursued.

Key Collaboration Opportunities in Multi-Scale Plasma Physics:

- *Assist in the analysis and modeling of the ion-gyro-scale turbulence and transport to understand the relationship between the flow and flow-shear and turbulence in the core and H-mode pedestal region, improve understanding of the physics of the H-mode transition, flow-damping from 3D magnetic fields, and the possible transport of fast-ions by low-k turbulence.*

- *Utilize and optimize the upgraded high-harmonic fast-wave system for core electron heating for electron transport and turbulence studies in L-mode and H-mode, and as a means of producing low collisionality plasmas with reduced momentum and particle input for momentum and particle transport studies.*

- *Participate in experiments and analysis exploring impurity transport from the edge to the core, and linkages to inward momentum pinch physics. Utilize existing and/or any new main-ion and impurity edge particle sources for perturbative particle transport experiments.*

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

For more information contact: Charles Skinner (cskinner@pppl.gov)

Research Priorities:

III-1. *Analyze and understand the impact of a liquid lithium divertor (LLD) on particle control, energy confinement, and H-mode pedestal transport and stability. Analyze and simulate the surface characteristics of the LLD and the interactions between the LLD and the edge plasma including the transport of lithium from the edge to the core under both steady-state and transient edge conditions.*

Background:
NSTX has demonstrated that evaporated lithium deposited on the lower divertor can improve particle control, increase thermal energy confinement, and eliminate ELMs in diverted H-mode plasmas. To investigate the effects of liquid lithium in diverted H-mode plasmas, NSTX has implemented a liquid lithium divertor (LLD) module. Substantial additional analysis and simulation is needed to understand the effects of the LLD on core and edge plasma conditions and the impact of LLD-related plasma-material interactions during steady-state operation, ELMs, and disruptions.

III-2. *Assess the parallel and cross-field transport of heat and particles in the Scrape-Off-Layer (SOL), understand the linkage between SOL transport and turbulence and the peak heat flux to the divertor, and develop means for divertor heat-flux mitigation for NSTX and upgraded/next-step ST facilities.*

Background:
NSTX has made significant progress in developing a partially detached divertor regime for reducing peak heat flux at the divertor 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. However, 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. Further, additional understanding of the scalings and underlying causes of SOL transport (in particular cross-field transport) is also needed to develop a predictive capability for the peak divertor heat flux for the design of next-step devices.

III-3. *Understand the H-mode pedestal characteristics that provide access to small ELM and ELM-free regimes in the ST, and understand how boundary modifications including plasma shaping, 3D fields, and lithium impact pedestal transport and ELM 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. Recently, 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 threshold 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. Additional experiments, analysis, and simulation are needed to better understand the influence of Li, RMP, and boundary shaping on pedestal thermal, particle, and momentum transport and the resulting impact on ELM stability. Methods of reducing impurity accumulation in small-ELM and no-ELM scenarios including variations in magnetic balance and divertor gas puffing show promise, but substantial additional research in impurity control is needed.

Key Collaboration Opportunities in Plasma Boundary Interfaces:

- *Perform experiments, analysis, and simulation to understand the impact of Li coatings and the LLD 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. Also, perform scoping studies and initial testing of high efficiency core and edge fueling techniques for LLD operation. Particular emphasis will be placed on characterizing and understanding the particle pumping, impurity generation, thermal response, and power-handling of the liquid lithium divertor (LLD) module under both steady-state and transient conditions (such as ELMs and disruptions).*

- *Perform experiments and simulations to aid in the development of predictive capability for SOL and divertor thermal, particle, and heat-flux widths. Develop techniques for heat-flux mitigation compatible with the anticipated enhanced divertor particle pumping of the LLD. Participate in LLD operation, experiments, and data analysis, and simuluation contributing to the research, development, and laboratory tests of long-pulse, high-heat-flux, and high particle pumping efficiency divertor concepts applicable to NSTX, the proposed NSTX Upgrade, and future fusion devices.*

- *Perform experiments, data analysis, and simulations to develop a predictive capability for the H-mode pedestal structure and stability. Optimize H-mode performance for high thermal confinement, small/no ELMs, density control, and acceptable impurity accumulation. Utilize evaporated and liquid lithium, externally applied 3D fields, boundary shaping, and other techniques to develop edge localized mode (ELM) suppression, triggering, and control. Understanding how 3D fields impact pedestal transport is particularly important, could benefit from cross-comparisons between NSTX and other experiments utilizing 3D fields, and is potentially important for ITER ELM control.*

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. *Study and optimize high-harmonic fast-wave (HHFW) heating and current drive in deuterium H-mode plasmas, with emphasis on understanding and minimizing parasitic loss mechanisms including: interactions between the HHFW and Neutral Beam Injection (NBI) fast ions, surface wave excitation, sheaths, and other RF-induced changes to the plasma edge.*

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 for the first time. 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. NSTX is now prepared to increase the HHFW heating power in high-performance NBI-heated discharges and for assessing plasma current ramp-up using HHFW-heated bootstrap current overdrive. 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. Other RF-induced changes in the edge plasma such as RF-driven sheaths and parametric decay instabilities may also impact HHFW heating efficiency in high-performance H-mode plasmas.

IV-2. *Study the range of observed energetic-particle-driven instabilities (for example Toroidal Alfvén Eigenmode (TAE) avalanches) and their role in redistribution of neutral-beam-driven current. Study interactions between electrostatic turbulence and fast-ion transport.*

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 research 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. Recently, 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.

Key Collaboration Opportunities in Waves and Energetic Particles:

- *Model HHFW heating and current-drive efficiency and RF-induced changes in the plasma core and edge and compare to experiment. Perform experiments, data analysis, and simulation to understand the interactions between fast-ions from NBI and the HHFW. Perform experiments and simulation to better understand the effects of edge transients on HHFW coupling – in particular the effects of edge-localized-mode (ELM) activity associated with H-mode operation.*

- *Perform research to develop a predictive capability for fast-ion transport by fast-ion-driven instabilities for FNSF and ITER. Utilize NBI fast-ion injection 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. Perform assessments of possible fast-ion transport by electrostatic turbulence utilizing the available operational tools and diagnostics.*

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 and characterize efficient plasma current start-up utilizing techniques such as coaxial helicity injection, fast wave heating, poloidal-field ramp-up and plasma gun start-up, incorporating the impact of increased divertor pumping, divertor impurity reduction, and improved equilibrium control.*

Background:
Coaxial helicity injection (CHI) has demonstrated transformer flux savings equivalent to 180kA 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. The Pegasus toroidal experiment has recently generated similar plasma current magnitudes (170kA) using plasma gun start-up. These favorable results motivate development and testing of means to efficiently heat (such as high-harmonic fast wave) and densify ST start-up plasmas to conditions compatible with non-inductive plasma current ramp-up to high plasma current.

V-2. *Assess non-inductively-driven plasma current ramp-up utilizing high-harmonic fast-wave heating and current-drive with increased RF power and with improved resilience to variations in plasma edge density.*

Background:
Plasma current ramp-up to conditions compatible with sustained high-performance - without a solenoid - is an important and challenging research objective for the ST. Current overdrive (from bootstrap and RF and NBI sources) is being pursued to provide current ramp-up to high performance. HHFW heating of low current H-mode discharges has achieved high poloidal beta and bootstrap current fractions up to 85%. Higher heating power and resilience to ELMs will be utilized to assess the possibility of current ramp-up using bootstrap current and RF current overdrive.

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

- *Perform experiments, data analysis, and simulation of CHI plasmas to improve start-up plasma energy and particle content to provide higher temperature and beta target plasmas for non-inductive ramp-up techniques such as HHFW and NBI. Optimize the equilibrium evolution of CHI to minimize impurity content and radiated power in the presence of a metallic outboard divertor (LLD) and lithium. Assess the viability of heating CHI plasmas to high temperature with HHFW. Perform initial assessments (via simulation and/or experiment) of plasma gun start-up in NSTX and for NSTX Upgrade.*

- *Perform experiments and time-dependent modeling of HHFW coupling and heating efficiency during the plasma current ramp-up, measure the production/acceleration of fast ions by HHFW, and measure/infer the sources of non-inductive current drive during the RF + bootstrap-current driven current ramp-up.*

VI. ***Physics Integration*** – 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. *Achieve and maintain high-performance plasmas with reduced density and collisionality. Produce discharges with increased non-inductive current fraction – in particular via changes in the neutral beam current drive efficiency, core and pedestal confinement and stability, edge bootstrap current, and impurity content.*

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%. Next-step ST devices sustained by non-inductive current drive are projected to operate with higher beam current fractions of 30-50% by operating at lower normalized density and collisionality. Reduced density and increased T_e with Li/LLD and HHFW are being assessed as possible means of increasing the NBI current fraction. Modifications to plasma core and edge confinement, stability, and impurity content will be assessed in plasmas with reduced density and high non-inductive fraction, and increasingly sophisticated plasma control techniques will be developed to sustain high plasma performance while avoiding disruptions. Achieving reduced impurity accumulation in ELM-free/small-ELM scenarios is also a major objective for high-performance integrated scenarios.

Key Collaboration Opportunities for Physics Integration:

- *Perform experiments, analysis, and simulation to increase and control non-inductive current drive utilizing Li/LLD and NBI current-drive redistribution from MHD and Alfvénic instabilities. Characterize the energy, momentum, and particle confinement properties and the full spectrum of plasma turbulence of discharges optimized for high non-inductive current drive fraction and long pulse duration.*

- *Contribute to the operational goal of reducing and controlling the core plasma collisionality by integrating HHFW heating with high-performance long-pulse NBI-heated H-mode scenarios with normalized parameters prototypical of an ST-FNSF. Test the compatibility and efficiency of HHFW core electron heating with NBI-driven H-mode operation, and the impact of RF heating on core impurity accumulation and control. Perform experiments to measure fast-ion acceleration by the HHFW, and develop and simulate means to modify and minimize the interaction between the HHFW and NBI fast-ions.*

- *Develop and simulate real-time control tools for improving the sustainability of integrated high performance plasmas. Examples include improved control of the boundary shape and divertor power and particle exhaust, control of the toroidal rotation profile, current profile, and control of operational proximity to stability limits to avoid and/or mitigate disruptions.*

Appendix A: FESAC Priorities Panel 10-Year Goals (as additional background information)	**Relevant NSTX Research**
Macroscopic Plasma Physics	
1. Understand the coupled dependencies of plasma shape, edge topology, and size on confinement in a range of plasma confinement configurations.	1. I-1
2. Identify the mechanisms whereby internal magnetic structure controls plasma confinement.	2. I-2,3
3. Identify the effects and consequences on confinement of large self-generated plasma current.	3. V-1, VI-1
4. Learn how to control the long scale-length instabilities that limit plasma pressure.	4. I-1, 2
5. Understand and control intermediate to short wavelength modes responsible for limiting the plasma pressure, particularly at the edge, and extrapolate their effects to the burning plasma regime.	5. III-1,3
6. Understand the equilibrium pressure limits in a range of magnetic configurations, including the effects of islands, stochastic magnetic fields, and helical states.	6. I-2, III-3
7. Understand and demonstrate the use of self-generated currents and mass flows to achieve steady-state high-pressure confined plasmas and improve fusion energy performance.	7. II-1, VI-1
8. Understand how external control can lead to improved stability and confinement in sustained plasmas in a range of magnetic configurations.	8. I-1,2, III-3
9. Understand the pressure limits and confinement properties in configurations where magnetic turbulence controls the distribution of the equilibrium magnetic field and for similar configurations with reduced turbulence. Assess their prospects for study in more collisionless plasma regimes for possible extrapolation to practical sustained burning plasmas.	9. V-1
Multi-Scale Transport Physics	
1. Develop predictive capability for ion thermal transport using simulations validated by comparison with fluctuation measurements.	1. II-1
2. Identify the dominant particle transport mechanisms, including the conditions under which pinch/convective processes compete with diffusive processes.	2. II-3
3. Identify the dominant mechanisms for momentum transport and their relationship to thermal transport.	3. II-1
4. Understand generation of flow shear, regulation of turbulence, and self-consistent profile dynamics and local steepening, and to identify conditions and thresholds for edge and core barrier formation.	4. II-1, 2, 3
5. Identify the dominant electron thermal transport mechanisms, including the role of electromagnetic fluctuations, short-scale versus long-scale turbulence, and spectral anisotropy.	5. II-2
6. Identify the dominant driving and damping mechanisms for large-scale and zonal flows, including turbulent stresses and cascades.	6. II-1, 2
7. Identify the dominant mechanisms by which turbulence generates and sustains large-scale magnetic fields in high-temperature plasma.	7. V-1
8. Identify the mechanisms and structure of magnetic reconnection, including the role of turbulent and laminar processes, energy flow, and the production of energetic particles.	8. I-2, V-1,2
9. Identify the conditions for onset of island growth and the factors controlling saturation and coupling with transport.	9. I-2
Plasma Boundary Interfaces	
1. Predict the expected magnetohydrodynamic stability and plasma parameters for the ITER H-mode edge pedestal with high confidence. This is a time-sensitive issue	1. III-1,3

Appendix A: FESAC Priorities Panel 10-Year Goals (as additional background information)	**Relevant NSTX Research**
relevant to the success of ITER 2. Identify the underlying driving mechanisms for mass flow and cross-field transport in the scrape-off-layer plasma, in H-mode attached and detached plasmas. 3. Resolve the key boundary-physics processes governing selection of plasma-facing components for ITER. This is a time-sensitive issue relevant to the success of ITER. 4. Complete the evaluation of candidate plasma-facing materials and technologies for high-power, long-pulse fusion experiments. This is a time-sensitive issue relevant to the success of ITER.	2. II-1, III-2, 3 3. III-1,2,3 4. III-1,2,3
Waves and Energetic Particles 1. Develop the capability to design high-power electromagnetic wave launching systems that couple efficiently and according to predictions for a wide range of edge conditions. 2. Produce, diagnose in detail, and model with nonlinear, closed-loop simulations the macroscopic plasma responses produced by wave-particle interactions, including localized current generation, plasma flows, and heating, in both axisymmetric and non-axisymmetric configurations. *3.* Develop long-pulse radio-frequency wave scenarios for optimizing plasma confinement and stability and to benchmark against models that integrate wave coupling, propagation, and absorption physics with transport codes (including microturbulence and barrier dynamics) and with magnetohydrodynamic stability models. 4. Improve analysis and models to match the experimental measurements and scale the understanding to predict the dynamics of energetic particle-excited modes in advanced regimes of operation with high pressure, inverted magnetic shear, and strong flow. 5. Identify the character of Alfvén turbulence and the evolution of the energetic particle distribution in a nonlinear system, which can be used to predict alpha-particle transport in a burning tokamak experiment; and to evaluate and extrapolate energetic particle behavior in present-day confinement systems to reactor parameters.	1. IV-2 2. IV-1,2 3. IV-1, V-2, VI-1 4. IV-2 5. IV-2, II-1
Fusion Engineering Science 1. Deliver to ITER the blanket test modules required to understand the behavior of materials and blankets in the integrated fusion environment. 2. Determine the “phase space” of plasma, nuclear, material, and technological conditions in which tritium self-sufficiency and power extraction can be attained. 3. Develop the knowledge base to determine performance limits and identify innovative solutions for the plasma chamber system and materials. 4. Develop the plasma technologies required to support U.S. contributions to ITER. 5. Develop the plasma technologies to support the research program.	1. 2. 3. III-1, 2, 3 4. I-1, III-3, IV-1 5. I-1,3, III-1,3, IV-1, V-1,2