FY2014 Office of Fusion Energy Sciences 3 Facility Joint Research Milestone:

Responsible TSGs: Macroscopic Stability, Boundary Physics, Transport and Turbulence

Conduct experiments and analysis to investigate and quantify the plasma response to non-axisymmetric (3D) magnetic fields in tokamaks. The effects of 3D fields can be both beneficial and detrimental and the research will aim to validate theoretical models in order to predict plasma response to varying levels and types of externally imposed 3D fields. The dependence of the response to multiple plasma parameters will be explored in order to gain confidence in the predictive capability of the models.

FY2015 Office of Fusion Energy Sciences 3 Facility Joint Research Milestone:

Responsible TSGs: Energetic Particles, Advanced Scenarios, Macroscopic Stability, T&T

Conduct experiments and analysis to quantify the impact of broadened current and pressure profiles on tokamak plasma confinement and stability. Broadened pressure profiles generally improve global stability but can also affect transport and confinement, while broadened current profiles can have both beneficial and adverse impacts on confinement and stability. This research will examine a variety of heating and current drive techniques in order to validate theoretical models of both the actuator performance and the transport and global stability response to varied heating and current drive deposition.

FY2016 Office of Fusion Energy Sciences 3 Facility Joint Research Milestone:

Disruption Avoidance, Detection, and Mitigation – 3 facility description text TBD.

Responsible TSGs: Macroscopic Stability, Advanced Scenarios

NSTX FY2014 Research Milestones:

R(14-1): Assess access to reduced density and collisionality in high-performance scenarios

Responsible TSGs: Macro-Stability, Advanced Scenarios and Control, Boundary Physics

The high performance scenarios targeted in NSTX-U and next-step ST devices are based on operating at lower Greenwald density fraction and/or lower collisionality than routinely accessed in NSTX. Collisionality plays a key role in ST energy confinement, non-inductive current drive, pedestal stability, resistive wall mode (RWM) stability, neoclassical toroidal viscosity that affects plasma torque balance, and plasma response and transport with 3D fields. Lower density and/or higher temperature are required to access lower ν^*. Potential means identified in NSTX to access lower ν^* included high harmonic fast wave heating, reduced fueling and/or Li pumping. However, while D pumping from lithium has been observed, additional gas fueling was typically required to avoid plasma disruption during the current ramp and/or in the high β phase of the highest performance plasmas of NSTX. The goal of this milestone is to identify the stability boundaries, characterize the underlying instabilities responsible for disruption at reduced density, and develop means to avoid these disruptions in NSTX-U. In support of this goal, tearing mode, RWM, neoclassical toroidal viscosity transport, disruption physics, and scrape-off-layer current (SOLC) in low density and collisionality will be investigated through analysis of NSTX data. This analysis will be used to project to NSTX-U scenarios and will include analysis of the potential impact of proposed/new non-axisymmetric control coils (NCC), and related research will also be carried out in other devices such as DIII-D, KSTAR, and MAST. These physics studies will be utilized to prepare for high-performance scenarios using methods such as current ramp-rate (li and q(r) evolution), H-mode transition timing, shape evolution, heating/beta evolution and control, optimized tearing mode and RWM control, rotation control, error field correction, fueling control (SGI, shoulder injector), and optimized Li pumping. This milestone will also aid development of MISK, VALEN, IPEC, and 3D transport models, as well as TRANSP and TSC integrated predictive models for NSTX-U and next-step STs.

R(14-2): Develop models for *AE mode-induced fast-ion transport

Responsible TSGs: Wave-Particle Interactions

Good confinement of fast ions from neutral beam injection and fusion reactions is essential for the successful operation of ST-CTF, ITER, and future reactors. Significant progress has been made in characterizing the Alfvénic modes (AEs) driven unstable by fast ions and the associated fast ion transport. However, models that can consistently reproduce fast ion transport for actual experiments, or provide predictions for new scenarios and devices, have not yet been validated against a sufficiently broad range of experiments. In order to develop a physics-based parametric fast ion transport model that can be integrated in general simulation codes such as TRANSP, results obtained from NSTX and during collaborations with other facilities (MAST, DIII-D) will be analyzed. Information on the mode properties (amplitude, frequency, radial structure) and on the fast ion response to AEs will be deduced from Beam Emission Spectroscopy, Reflectometers, Fast-Ion D-alpha (FIDA) systems, Neutral Particle Analyzers, Fast Ion Loss Probes and neutron rate measurements. The fast ion transport mechanisms and their parametric dependence on the mode properties will be assessed through comparison of experimental results with theory using both linear (e.g., NOVA-K) and non-linear (e.g., M3D-K, HYM) codes, complemented by gyro-orbit (ORBIT) and full-orbit (SPIRAL) particle-following codes. Based on the general parametric model, the implementation of *reduced* models in TRANSP will then be assessed. For instance, the existing Anomalous Fast Ion Diffusion (AFID) and radial fast ion convection models in TRANSP could be improved by implementing methods to calculate those transport coefficients consistently with the measured (or simulated) mode properties. Further improvements will also be considered, for instance to include a stochastic transport term or quasi-linear models.

R(14-3): Develop advanced axisymmetric control in sustained high performance plasmas

Responsible TSGs: Advanced Scenarios and Control, Boundary physics, Macro Stability

Next step tokamaks and STs will need high-fidelity axisymmetric control. For instance, magnetic control of the plasma boundary and divertor impact the global stability, power handling, and particle control from poloidally localized pumps. Control of the current and rotation profiles will be critical for avoiding resistive wall modes and tearing modes, thus maximizing the achievable β. The 2nd neutral beamline for NSTX-U will provide considerable flexibility in the neutral beam driven current profile, while additional divertor coils will allow a wide range of divertor geometries; it is thus an appropriate facility for the development of these critical control techniques. As part of this milestone, realtime control algorithms for the snowflake divertor will be designed; these will likely use methods for rapid tracking of multiple X-points, and additions will be made to the ISOFLUX boundary control algorithm to target specific divertor quantities for control. These divertor control algorithm will be prepared for use in NSTX-U, and may be tested in DIII-D. For profile control, a real-time Motional Stark Effect diagnostic will be developed for NSTX, and the data provided to the NSTX-U implementation of rtEFIT for constrained reconstruction of the current profile; the feasibility of realtime rotation measurements in NSTX-U will be determined and that system implemented as appropriate. Real-time control algorithms will be developed for the current profile using the various neutral beams as actuators; integrated modeling of the current profile evolution with codes such as PTRANSP and TSC will be used for system identification. Similarly, algorithms for control of the rotation profile will be developed, using the neutral beams and magnetic braking as actuators. This profile control development may be based on existing DIII-D control algorithms, but with NSTX-U specific constraints. The ability of the proposed non-axisymmetric control (NCC) coils to provide improved actuator capability for rotation control compared to the existing mid-plane coils will be addressed using NTV calculations. The feasibility of simultaneous rotation, current, and β control will be assessed. This research will provide a considerable head start developing the required control algorithms for NSTX-U, as well as provide valuable guidance on the axisymmetric control designs for next-step tokamaks and STs, including ITER.

NSTX FY2015 Research Milestones:

R(15-1): Assess H-mode energy confinement, pedestal, and scrape off layer characteristics with higher B_T, I_P and NBI heating power

Responsible TSGs: Transport and Turbulence, Boundary Physics, Advanced Scenarios

Future ST devices such as ST-FNSF will operate at higher toroidal field, plasma current and heating power than NSTX. To establish the physics basis for future STs, which are generally expected to operate in lower collisionality regimes, it is important to characterize confinement, pedestal and scrape off layer trends over an expanded range of engineering parameters. H-mode studies in NSTX have shown that the global energy confinement exhibits a more favorable scaling with collisionality ($B\tau_E \sim 1/\nu^*_e$) than that from ITER98y,2. This strong ν^*_e scaling unifies disparate engineering scalings with boronization ($\tau_E \sim I_p^{0.4} B_T^{1.0}$) and lithiumization ($\tau_E \sim I_P^{0.8} B_T^{-0.15}$). In addition, the H-mode pedestal pressure increases with $\sim I_P^2$, while the divertor heat flux footprint width decreases faster than linearly with I_P. With double B_T, double I_P and double NBI power with beams at different tangency radii, NSTX-U provides an excellent opportunity to assess the core and boundary characteristics in regimes more relevant to future STs and to explore the accessibility to lower collisionality. Specifically, the relation between H-mode energy confinement and pedestal structure with increasing I_P, B_T and P_{NBI} will be determined and compared with previous NSTX results, including emphasis on the collisionality dependence of confinement and beta dependence of pedestal width. Coupled with low-k turbulence diagnostics and gyrokinetic simulations, the experiments will provide further evidence for the mechanisms underlying the observed confinement scaling and pedestal structure. The scaling of the divertor heat flux profile with higher I_P and P_{NBI} will also be measured to characterize the peak heat fluxes and scrape off layer widths, and this will provide the basis for eventual testing of heat flux mitigation techniques. Scrape-off layer density and temperature profile data will also be obtained for several divertor configurations, flux expansion values, and strike-point locations to validate the assumptions used in the FY2012-13 physics design of the cryopump to inform the cryo-pump engineering design to be carried out during FY2015.

R(15-2): Assess the effects of neutral beam injection parameters on the fast ion distribution function and neutral beam driven current profile

Responsible TSGs: Energetic Particles, Transport and Turbulence

Accurate knowledge of neutral beam (NB) ion properties is of paramount importance for many areas of tokamak physics. NB ions modify the power balance, provide torque to drive plasma rotation and affect the behavior of MHD instabilities. Moreover, they determine the non-inductive NB driven current, which is crucial for future devices such as ITER, FNSF and STs with no central solenoid. On NSTX-U, three more tangentially-aimed NB sources have been added to the existing, more perpendicular ones. With this addition, NSTX-U is uniquely equipped to characterize a broad parameter space of fast ion distribution, F_{nb}, and NB-driven current properties, with significant overlap with conventional aspect ratio tokamaks. The two main goals of the proposed Research Milestone on NSTX-U are (i) to characterize the NB ion behavior and compare it with classical predictions, and (ii) to document the operating space of NB-driven current profile. F_{nb} will be characterized through the upgraded set of NSTX-U fast ion diagnostics (e.g. fast-ion D-alpha: FIDA, solid-state neutral particle analyzer: ssNPA, scintillator-based fast-lost-ion probe: sFLIP, and neutron counters) as a function of NB injection parameters (tangency radius, beam voltage) and magnetic field. Well controlled, single-source scenarios at low NB power will be initially used to compare fast ion behavior with classical models (e.g. the NUBEAM module of TRANSP) in the absence of fast ion driven instabilities. Diagnostics data will be interpreted through the "beam blip" analysis technique and other dedicated codes such as FIDASIM. Then, the NB-driven current profile will be documented for the attainable NB parameter space by comparing NUBEAM/TRANSP predictions to measurements from Motional Stark Effect, complemented by the vertical/tangential FIDA systems and ssNPA to assess modifications of the classically expected F_{nb}. As operational experience builds up during the first year of NSTX-U experiments, additions to the initial F_{nb} assessment will be considered for scenarios where deviations of F_{nb} from classical predictions can be expected. The latter may include scenarios with MHD instabilities, externally imposed non-axisymmetric 3D fields, and additional High-Harmonic Fast Wave (HHFW) heating.

R(15-3): Develop the physics and operational tools for obtaining high-performance discharges in NSTX-U

Responsible TSGs: Advanced Scenarios, Macro-Stability, Boundary Physics, Materials and PFCs

Steady-state, high-beta conditions are required in future ST devices, such as a FNSF/CTF facility, for increasing the neutron wall loading while minimizing the recirculating power. NSTX-U is designed to provide the physics knowledge for the achievement of such conditions by demonstrating stationary, long pulse, high non-inductive fraction operation. The ultimate toroidal field (1.0 T) and plasma current (2.0MA) capability of NSTX-U is twice that in NSTX. NSTX-U has a capability for >5 second discharges, and it has an additional beamline which doubles the available heating power and provides much greater flexibility in the beam current drive profile. The aim for studies during the first year of operation of NSTX-U is to lay the foundation for the above operational scenario goals by developing needed physics and operational tools, using toroidal fields up to ~0.8 T, plasma currents up to ~1.6 MA, improved applied 3D field capabilities from additional power supplies, a variety of plasma facing component (PFC) conditioning methods, and advanced fuelling techniques. As an example of the latter, supersonic gas injection provides higher fuelling efficiency, and will be used to develop reliable discharge formation with minimal gas loading. Differing PFC conditioning techniques, including boronization and lithium coatings, will be assessed to determine which are most favorable for longer pulse scenarios. Impurity control techniques, an example of which is ELM pacing, will be developed for the reduction of impurity accumulation in otherwise ELM-free lithium-conditioned H-modes. The higher aspect ratio, high elongation ($2.8 < \kappa < 3.0$) plasma shapes anticipated to result in high non-inductive fraction in NSTX-U will be developed, and the vertical stability of these targets will be assessed, with mitigating actions taken if problems arise. An initial assessment of low-n error fields will be made, along with expanding the RWM control and dynamic error field correction strategies using both proportional and state-space $n \geq 1$ feedback schemes, taking advantage of the spectrum flexibility provided by the 2^{nd} SPA power supply. Resonant field amplification measurements, ideal MHD stability codes, and kinetic stability analysis will be used to evaluate the no-wall and disruptive stability limits in these higher aspect ratio and elongation scenarios. These physics and operational tools will be combined to make an initial assessment of the non-inductive current drive fraction across a range of toroidal field, plasma density, boundary shaping, and neutral beam parameters.

IR(15-1): Develop and assess the snowflake divertor configuration and edge properties in NSTX-U

Responsible TSGs: Boundary Physics, Advanced Scenarios and Control

The high flux expansion snowflake divertor configuration is a leading candidate for mitigation of high power exhaust in NSTX-Upgrade, where projected peak heat fluxes up to 20 MW/m^2 are anticipated in 12 MW NBI-heated discharges. In NSTX-U, an upgraded up-down symmetric set of three divertor coils will be used to develop a variety of snowflake configurations. Experiments in FY2015 will be focused on 1) development of magnetic configuration control; 2) initial studies of edge and divertor properties. In the area of magnetic control, fast numerical algorithms will be implemented and tested with the Plasma Control System to develop feedback control of inter-X-point distance, X-point orientation, and flux expansion. Divertor heat flux handling and power accountability and impurity production trends with engineering parameters that will become accessible in NSTX-U, such as I_P = 1-2 MA, P_{NBI} = 6-12 MW, will be assessed. H-mode pedestal stability will also be assessed to determine if the snowflake configurations can be used for ELM control. Measurements will be compared to multi-fluid and kinetic model predictions. Results from initial NSTX-U snowflake divertor experiments will also be compared with the experiments that were performed in NSTX and DIII-D. This research will provide a significant step in the snowflake divertor concept development for both the ST and tokamak.

NSTX FY2016 Research Milestones:

R(16-1): Assess scaling and mitigation of steady-state and transient heat-fluxes with advanced divertor operation at high power density

Responsible TSGs: Boundary Physics, Advanced Scenarios and Control

Handling plasma power exhaust in the divertor region is a critical issue for ITER and FNSF. Of particular concern is the projected narrow scrape-off-layer (SOL) heat-flux width (observed to scale inversely with plasma current I_P) projected for next-step tokamak devices that will operate at higher I_P. By operating at twice the I_P, B_t, and heating power relative to NSTX, NSTX-U will access narrow SOL widths and high parallel heat fluxes and will provide important contributions to understanding SOL cross-field transport scalings and to the development of techniques for mitigating high heat flux. Heat flux mitigation techniques to be tested both individually and in combination include: (1) magnetic balance (i.e. double-null operation), (2) radiative and partially detached divertor operation, (3) high flux expansion configurations e.g. "snowflake", and (4) ELM pacing. Experiments will be performed to compare the efficacy of these techniques for reducing steady and transient heat fluxes, impurity production, and deleterious effects on the core plasma performance. To increase dissipative losses in the radiative divertor, D_2 or impurity seeding will be used, enabled by an improved gas injection system. Detachment operating window for the highest SOL power and I_P will be studied as a function of density and divertor seeding species (D_2, CD_4, Ne, Ar) in the standard and snowflake configurations. ELM-control/triggering techniques such as 3D fields and injected granules will be used to assess and reduce ELM-induced transient heat fluxes. Plasma control will be enhanced for long-pulse magnetic control of the double-null and the snowflake divertor configurations. Edge diagnostic data, i.e. infrared thermography, Langmuir probes, cameras, impurity spectroscopy, and bolometry will be compared to state-of-the-art numerical model predictions. These results will enable further development of high performance ST plasma scenarios with acceptable divertor power exhaust solutions, and aid in the validation of divertor power and particle exhaust models for ITER and FNSF.

R(16-2): Assess high-Z divertor PFC performance and impact on operating scenarios

Responsible TSGs: Materials and PFCs, Boundary Physics

The present leading candidate divertor plasma facing component (PFC) is tungsten due to favorable material properties including: high melting point and thermal conductivity and low activation, thermal expansion, and sputtering yield. Challenges of tungsten include high hardness and brittleness which make the fabrication of tungsten components difficult, and the potential accumulation of high-Z impurities in the plasma core which can lead to poor plasma confinement and/or radiative collapse. Further, all solid PFCs are subject to surface erosion and re-deposition which could substantially degrade their thermo-mechanical properties. Flowing liquid-metal PFCs may offer a potential long-term solution to many of these challenges. To study high-Z PFC performance in a high-power-density ST environment, and to provide a substrate more suitable for liquid metal studies, NSTX-U is planning to operate with high-Z PFCs in a portion of the outboard divertor in FY2016. The performance of the high-Z PFCs and the impact on high-performance operating scenarios will be evaluated in order to assess the potential impact of further expanded future high-Z coverage. The capability to handle steady and transient heat fluxes will be assessed, to determine future requirements on mitigation schemes. Operation on the high-Z tiles in conjunction with heat-flux mitigation schemes (e.g. radiative gas puffing, strike-point sweeping) will be initiated for a range of incident heat and particle fluxes. Impurity production consisting of eroded low-Z coatings (such as boron and lithium) and the high-Z substrate material will be evaluated during these experiments. Migration of the impurities from their source locations throughout the rest of the machine will be diagnosed with a range of material and plasma diagnostics, and analyzed with interpretive codes such as OEDGE and WALLDYN. Influx of impurities into the NSTX-U core and evolution of the H-mode pedestal will also be assessed during operation on the high-Z PFCs. Infrared thermography will be utilized to infer divertor heat flux profiles and to identify any local heating and melting resulting from PFC leading edges and/or misalignments.

R(16-3): Assess fast-wave SOL losses and core thermal and fast ion interactions at increased field and current

Responsible TSGs: Waves and Energetic Particles

Use of high-harmonic fast wave (HHFW) injection for heating and current drive in NSTX-U is important to develop non-inductive start-up/ramp-up schemes, to achieve higher plasma performance, and to validate RF codes in support of the ITER FW program. The successful exploitation of HHFW depends on how much power can be coupled to the core plasma through the scrape-off layer (SOL) and how the coupled power is partitioned between different species in the core. For NSTX-U H-mode plasmas, the latter typically include thermal electrons/ions and fast ions from Neutral Beam (NB) injection. Mechanisms responsible for RF power losses in the SOL will first be addressed to optimize FW coupling to the core as a function of excited wave spectrum, toroidal field and plasma current (B_T and I_P). For instance, full-wave simulations with the AORSA code suggest a local enhancement of RF fields for propagating (rather than evanescent) fast waves in the SOL, which results in increased losses through damping. Losses are predicted to decrease at lower SOL density and/or at higher B_T. AORSA predictions will be tested against improved measurements of RF fields through dedicated probe arrays. The amount of lost power will be quantified through infrared cameras. The improved understanding of SOL loss mechanisms will provide more accurate estimations of the fraction of injected HHFW power reaching the core. The synergy between NB and FW injection in H-mode plasmas will be evaluated for the higher B_T and I_P of NSTX-U. In particular, the efficiency of thermal ions heating at higher B_T, predicted to increase by full-wave simulations, will be investigated. Similarly, modifications of the fast ion distribution, previously observed on NSTX, will be assessed with the enhanced fast ion diagnostics set of NSTX-U as a function of plasma parameters, FW spectrum and NB injection geometry, including the new (more tangential) NB lines. The characterization of scenarios with combined NB and FW injection will provide a valuable dataset to benchmark RF codes (e.g. AORSA, TORIC and CQL3D-FOW), which have been considerably improved in the past years. NSTX-U will provide the required data for extensive validation before their use for scenario predictions in future devices such as ITER and FNSF.

R(16-4) Develop high-non-inductive fraction NBI H-mode scenarios for sustainment and ramp-up

Responsible TSGs: Advanced Scenarios, Solenoid-Free Start-up/Ramp-up

Future spherical torus applications such as a Fusion Nuclear Science Facility (FNSF) will need 100% non-inductive sustainment, and very likely 100% non-inductive plasma current ramp-up as well. NSTX-Upgrade was designed to support high-bootstrap fraction operation through the increased toroidal field, and high beam driven current fraction via the additional neutral beams with large tangency radius. Research supporting this milestone will focus on maximizing the non-inductive current fraction in plasmas with inductive current ramps. Variations in the plasma current level, neutral beam power, and plasma shaping will be used to assess and maximize the non-inductive current fraction. Reductions of the initial I_P level will be used to begin to bridge the gap towards fully non-inductive ramp-up. New lithium evaporators and the lithium granule injector may be used to control the confinement and impurity levels. These results will provide data on the stability and confinement properties of high non-inductive fraction plasmas, in order to further optimize the ramp-up process and final flat-top operating point.

IR(16-1): Assess τ_E and local transport and turbulence at low ν^* with full confinement and diagnostic capabilities

Future ST devices such as ST-FNSF will operate at higher toroidal field, plasma current and heating power than NSTX. To establish the physics basis for future STs, which are generally expected to operate in lower collisionality regime, it is important to characterize confinement trends over an expanded range of engineering parameters. H-mode studies in NSTX have shown that the global energy confinement exhibits a more favorable scaling with collisionality ($B\tau_E \sim 1/\nu^*_e$) than that from ITER98y,2. This strong ν^*_e scaling unifies disparate engineering scalings with boronization ($\tau_E \sim I_P^{0.4}B_T^{1.0}$) and lithiumization ($\tau_E \sim$

$I_P^{0.8}B_T^{-0.15}$). Milestone R15-1 will provide initial data on how increasing plasma current, toroidal field and heating power will affect global energy confinement and access to lower collisionality in NSTX-U (compared to NSTX). During FY16 it will become feasible to assess confinement and local transport dependences over the widest range of accessible B_T (1T), I_P (2 MA), and P_{NBI} (15 MW). The addition of the upward facing LiTER will allow Lithium coating on the upper divertor, providing enhanced wall pumping and presumably allowing the greatest flexibility to access minimum ν^*. During this run time it is expected that the new high-k microwave scattering and polarimetry diagnostics will become operational (resources permitting), allowing acquisition of critical new turbulence data while operating over an extended range of parameter space. The new high-k scattering system will allow for an initial assessment of possible turbulence anisotropy and the polarimeter will provide initial data to assess the importance of magnetic microturbulence on confinement and transport. Coupled with gyrokinetic simulations and existing low-k turbulence measurements, these additional datasets will help distinguish between various theoretical transport mechanisms underlying the confinement scaling trends (e.g. micro-tearing vs. ETG).

NSTX-U Contribution to JRT-2016: Assess gas assimilation efficiencies for MGI, and utilize MGI and MHD control coupled to a disruption warning system

Responsible TSGs: Macroscopic Stability, Advanced Scenarios

Disruptions in a reactor-scale tokamak/ST can cause unacceptable damage to plasma facing components and thus must be avoided or detected in advance and properly mitigated. Disruption studies on NSTX-U will offer new insight for these critical subjects utilizing unique MGI equipment and pioneering disruption avoidance systems and warning algorithms developed during NSTX operation. Three identical MGI systems at different poloidal locations will enable the assessment of the gas assimilation efficiency as a function of poloidal injection location and edge plasma parameters. Consequent reductions to the divertor heat loads and halo currents by mitigation as a function of the gas assimilation fraction will also be measured. The gas penetration past the SOL will be studied during the thermal quench, which will be quantitatively compared with simulation such as from the DEGAS-2 code modeling. In parallel, NSTX-U will develop and test a disruption warning system, and the quality of individual warning sensors will be improved during the course of experimental applications. With the help of physics-based MHD modeling, magnetic sensor response will be compared with state-space observers in real-time to address the correlation with RWM stability. Total and/or various combinations of these warning sensors will be investigated and improved in conjunction with mitigation and avoidance actuators. In particular, MHD sensors, such as low frequency RWM sensors, state-space observers, and resonant field amplification will be tested along with the algorithm for real-time MGI triggering and also algorithms for real-time rotation and β control to avoid global instability and disruption. These studies from NSTX-U will be used to inform the disruption mitigation system (DMS) design for ITER and will also inform the development of advanced disruption warning and avoidance systems in next-step tokamaks/STs as well as ITER.