FY2013 Office of Fusion Energy Sciences 3 Facility Joint Research Milestone:

Responsible TSGs: Advanced Scenarios and Control, Boundary Physics

Conduct experiments on major fusion facilities, to evaluate stationary enhanced confinement regimes without large Edge Localized Modes (ELMs), and to improve understanding of the underlying physical mechanisms that allow increased edge particle transport while maintaining a strong thermal transport barrier. Mechanisms to be investigated can include intrinsic continuous edge plasma modes and externally applied 3D fields. Candidate regimes and techniques have been pioneered by each of the three major US facilities (C-Mod, D3D and NSTX). Coordinated experiments, measurements, and analysis will be carried out to assess and understand the operational space for the regimes. Exploiting the complementary parameters and tools of the devices, joint teams will aim to more closely approach key dimensionless parameters of ITER, and to identify correlations between edge fluctuations and transport. The role of rotation will be investigated. The research will strengthen the basis for extrapolation of stationary high confinement regimes to ITER and other future fusion facilities, for which avoidance of large ELMs is a critical issue.

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: TBD

NSTX FY2013 Research Milestones:

R(13-1): Perform integrated physics and optical design of new high-k_θ FIR system

Responsible TSGs: Transport and Turbulence

Previous high-k scattering measurements in NSTX have identified ETG turbulence as one candidate for the anomalous electron thermal transport for both H and L-mode plasmas. However, a definitive connection between ETG turbulence and electron thermal transport could not be established since the previous high-k_r microwave scattering system was not able to measure the predicted peak power of the wavenumber spectrum of ETG turbulence. In collaboration with UC-Davis, a new high-k_θ FIR scattering system will be designed to make this measurement. Detailed physics and optical design of this scattering system will be performed. In particular, the spatial and spectral resolution and coverage of the scattering system will be optimized by integrating ray tracing, quasi-optical analysis and the launching and receiving optics design, based on predicted NSTX-U equilibrium profiles. The FIR laser for the scattering system will also be designed. Alignment and calibration schemes for both launching and receiving optics will be investigated. The above activities will lay a solid foundation for the implementation of this high-k_θ FIR scattering system on NSTX-U.

R(13-2): Investigate the relationship between lithium-conditioned surface composition and plasma behavior.

Responsible TSGs: Lithium research, Boundary Physics

The plasma facing surfaces in a tokamak have long been known to have a profound influence on plasma behavior. The development of a predictive understanding of this relationship has been impeded by the lack of diagnostics of the morphology and composition of the plasma facing surfaces. Recently, a probe has been used to expose samples to NSTX plasmas and subsequent post-run analysis has linked surface chemistry to deuterium retention. However, with very chemically active elements such as lithium, more prompt surface analysis is likely required to characterize the lithiated surface conditions during a plasma discharge. In support of prompt surface analysis, an in-situ materials analysis particle probe (MAPP) will be used to investigate sample exposure under NSTX-U relevant vacuum conditions. The MAPP will enable the exposure of various samples to plasma followed by ex-vessel but in-vacuo surface analysis within minutes of plasma exposure using state of the art tools. The reactions between evaporated lithium and plasma facing materials and residual gases will be studied. The MAPP will be installed on LTX and the intershot analysis capability will be demonstrated. These inter-shot/time-dependent measurements will provide unique data for benchmarking codes for modeling particle control in NSTX-Upgrade.

R(13-3): Perform physics design of ECH and EBW system for plasma start-up and current drive in advanced scenarios

Responsible TSGs: Waves and Energetic Particles, Solenoid-Free Start-up, Advanced Scenarios and Control

For a reactor-relevant ST operation it is critical to develop discharge initiation, plasma current ramp-up, and plasma sustainment techniques that do not require a central solenoid. Earlier ECH modeling of NSTX CHI startup plasmas with GENRAY and CQL3D predicted 25-30% first pass absorption. In addition, EBW startup experiments on MAST in 2009 showed good electron heating when the discharge became overdense. Several hundred kilowatts of coupled ECH/EBWH power in NSTX-U should heat a solenoid-free startup discharge sufficiently to allow coupling of 30 MHz high harmonic fast wave power, that will in turn generate non-inductive plasma current ramp-up. While pressure gradient-driven bootstrap current can provide a large fraction of the plasma current required to non-inductively sustain an ST plasma, an externally driven off-axis current may still be required to provide magnetohydrodynamic stability during the plasma current flat top. Electron Bernstein Wave current drive (EBWCD) can provide this non-inductive current and thus may play a critical role in enabling high beta, sustained operation of ST plasmas. A 28 GHz ECH and EBWH system is being proposed for NSTX-U. Initially the system will use short, 10-50 ms, 0.5-1 MW pulses to support development of non-inductive startup scenarios. Later the pulse length may be extended to 0.2-0.5 s and the power increased to provide EBWH and EBWCD during the plasma current flat top. EBW startup experiments are being planned on MAST for 2013 to extend the 2009 experiments to higher EBW power. Results from those experiments will support the design for the EBW startup system for NSTX-U. In 2013-2014 GENRAY and CQL3D ECH and EBWH modeling will be performed for NSTX-U plasma startup scenarios and for EBWH and EBWCD during the plasma current flat top for advanced NSTX-U plasma scenarios to support the physics design of the NSTX-U ECH/EBWH system.

R(13-4): Identify disruption precursors and disruption mitigation & avoidance techniques for NSTX-U and ITER

Responsible TSGs: Macro Stability, ITER Needs

In order for the tokamak/ST concept to reach its full potential, disruptions must be infrequent, detectable in advance, and amenable to intervention in order to eliminate their consequences. High current disruptions in NSTX-U, for instance, could decondition lithium coated plasma facing components (PFCs) or other lithium conditioning systems, while unmitigated disruptions in ITER have the potential for severe damage to the vessel and PFCs. Indicators of proximity to or the crossing of global, disruptive stability boundaries in NSTX discharges will be developed; these could include MHD signals like resistive wall modes (RWMs), locked modes, rotating MHD modes and/or resonant field amplification, scrape-off-layer current (SOLC), confinement indicators such as the flux consumption and neutron rate, real-time comparison to RWM state-space observer computation, or equilibrium properties such as the pressure peaking and edge safety factor. Strategies for processing and combining the various precursors will be developed, as will requirements for real-time measurements in NSTX-U. A real-time architecture for response to these and other off-normal events will be developed. Potential responses include rapid plasma ramp down, or discharge termination via massive gas injection (MGI). An engineering optimization of the MGI system will be made for NSTX-U, and MGI modeling may be pursued in support of both NSTX-U and ITER. This research will facilitate disruption free operation of present and next-step STs and tokamaks, including ITER.

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.

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 2nd 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 numerical multi-fluid two-dimensional 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.

IR(15-2): Establish Coaxial Helicity Injection start-up and perform initial tests of Neutral Beam Injection current ramp-up in inductively generated plasmas

Responsible TSGs: Solenoid-Free Start-up, Advanced Scenarios and Control

The start-up and ramp-up of a spherical torus burning plasma such as FNSF-ST utilizing little or no induction from a central solenoid is a major challenge for magnetic fusion. On NSTX, Coaxial Helicity Injection (CHI) was successfully employed to non-inductively start-up a discharge that reached a closed-flux plasma current of about 200 kA in less than 10 ms, and in NSTX-U the peak plasma current reached in CHI-only discharges is expected to be at least 400 kA. However the plasma current in these CHI-only discharges typically decays in about 30 ms due to low electron temperature. RF heating can be used to mitigate this plasma current decay by rapidly heating the plasma. The baseline 5-year research plan for NSTX-U includes the installation of a 1 MW 28 GHz electron cyclotron (EC) heating system to provide electron heating for CHI plasmas. An important goal of NSTX-U research is to ramp the current generated from ECH-heated CHI discharges using NBI and HHFW to the full non-inductive current sustainment levels near ~1MA. In preparation for this activity, the plasma start-up and ramp-up will be optimized separately and then linked together after ECH and/or HHFW heating of the CHI target is established. For start-up optimization, CHI plasma formation will be re-established and the impact of possible enhancements to CHI in NSTX-U including increased injector flux, higher CHI voltage, higher toroidal field, and full lithium coverage will be assessed. On NSTX-U, the combination of higher neutral beam injection (NBI) power, including three additional sources that are more tangential and therefore better suited to driving current, and the higher maximum toroidal field, should enable NBI plasma current ramp-up in discharges at lower plasma current. For ramp-up optimization, 300-600kA inductive plasma targets will be generated and the ability to ramp the plasma current to higher values non-inductively using NBI and HHFW will be assessed.