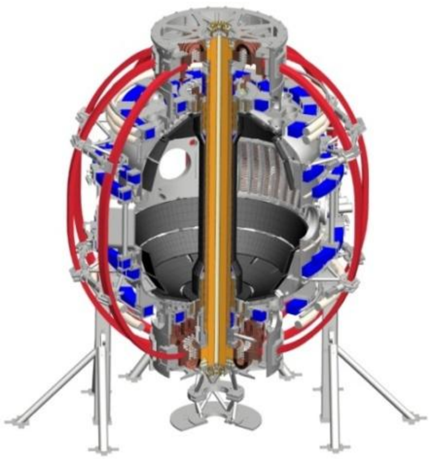
Table of Contents for Chapter 1

Chapter 1



Overview of the NSTX Upgrade Five Year Plan for 2014-2018

1.1 Introduction

Research utilizing the low-aspect-ratio "spherical" tokamak (ST) [1] makes important contributions to the world magnetic fusion program on two fronts. First, the ST configuration is characterized by strong intrinsic plasma shaping and enhanced stabilizing magnetic field line curvature. These unique ST characteristics enable the achievement of a high plasma pressure relative to the applied magnetic field and provide access to an expanded range of plasma parameters and operating regimes relative to the standard aspect ratio tokamak. For example, NSTX [2,3] has accessed a very wide range of dimensionless plasma parameter space with toroidal beta up to 40%, normalized beta up to 7, plasma elongation up to 3, normalized fast-ion speed $v_{fast}/v_{Alfvén}$ up to 5, Alfvén Mach number $M_A = v_{rotation}/v_{Alfvén}$ up to 0.5, and trapped-particle fraction up to 90% at the plasma edge. All of these parameters are well beyond what is accessible in conventional tokamaks, and these parameters approach those achievable in other high-beta alternative concepts. Further, it is also possible and common to overlap with conventional aspect ratio tokamak physics parameters. These characteristics therefore allow ST research to complement and extend standard aspect-ratio tokamak science while providing low-collisionality, long pulse-duration, and well-diagnosed plasmas to address fundamental plasma science issues – including burning plasma physics in ITER [4].

The fundamental scientific goal of ITER is to generate plasmas dominated by alpha particle heating and to understand the dynamics of the thermal and energetic plasma particles under such conditions. The dynamics is potentially non-linear, since a relatively large population of energetic ions originates from fusion reactions (alpha particles), Neutral Beam (NB) injection and injected RF waves may drive Alfvénic instabilities. Of particular concern for ITER are fast-ion-related ‘bursting’ phenomena that occur on fast time scales, since few control tools are envisioned (possibly dominant ECH heating) for their mitigation or suppression. Thus, energetic particle (EP) research is of high importance to understand the coupled (and predominantly non-linear) dynamics of fast ions and Alfvénic instabilities and eliminate or minimize their potential harm to a reliable exploitation of fusion energy.

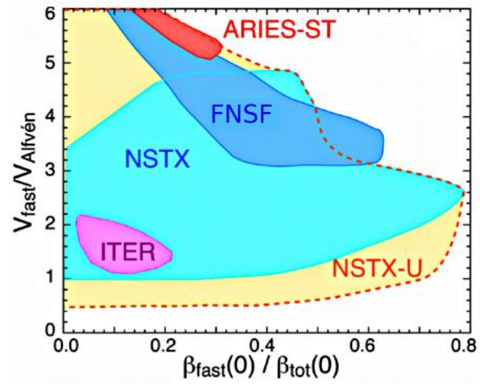


Figure 1.1.1: Ratio of fast ion to Alfvén velocity vs. ratio of fast ion to total pressure for energetic particle research for existing US tokamaks in comparison with expected values for ST-based reactors and for ITER.

The challenge for present tokamaks is to provide the required physics basis to develop and validate theoretical and numerical modeling tools. The parameter regimes accessible in NSTX will be significantly broadened in NSTX-Upgrade/NSTX-U [5] as shown for example in Figure 1.1.1 (and discussed extensively in Chapter 6), with significant overlap with expected energetic particle parameters of future devices such as ITER and ST-FNSF [6]. The capability of spanning a much broader range of parameters for EP physics than conventional tokamaks represents an important advancement for extrapolations from today’s experiments to burning plasma regimes.

Second, the reduced aspect ratio and smaller major radius of the ST can provide access to high divertor heat flux which can be exploited to develop plasma-material-interface (PMI) solutions for heat-flux mitigation and particle control [7,8]. Further, the high neutron wall loading potentially accessible in an ST-based Fusion Nuclear Science/Component Test Facility (FNSF/CTF) [6,9,10] could prove attractive for materials and component development for next-step devices. The modular configuration of the ST [9] could also potentially improve accessibility and maintainability in support of high duty factor operation for PMI and FNSF missions. Longer term, the resistive dissipation in the toroidal field of a normally conducting ST power plant [11] or Pilot Plant [12] is generally a disadvantage for electricity production. However, the ST ($A < 2$), combined with $A \approx 3$ conventional aspect ratio tokamaks can strongly inform the physics performance of reduced aspect ratio ($A=2$-2.6) super-conducting Demo concepts studied in Japan that are projected to minimize device mass, cost, and radioactive waste [13-18]. A range of low-A configurations is shown in Figure 1.1.2. NSTX Upgrade parameters are highlighted in yellow to show that NSTX-U will provide the next factor two in plasma current, magnetic field, and heating power toward an ST-FNSF or Pilot Plant.

Parameter	NSTX	NSTX Upgrade	Fusion Nuclear Science Facility	Pilot Plant
Major Radius R_0 [m]	0.86	**0.94**	1.3	1.6 – 2.2
Aspect Ratio R_0 / a	≥ 1.3	**≥ 1.5**	≥ 1.5	≥ 1.7
Plasma Current [MA]	1	**2**	4 – 10	11 – 18
Toroidal Field [T]	0.5	**1**	2 – 3	2.4 – 3
Auxiliary Power [MW]	≤ 8	**≤ 19***	22 – 45	50 – 85
P/R [MW/m]	10	**20**	30 – 60	70 – 90
P/S [MW/m^2]	0.2	**0.4**	0.6 – 1.2	0.7 – 0.9
Fusion Gain Q			1 – 2	2 – 10

Includes 4MW of high-harmonic fast-wave (HHFW) heating power

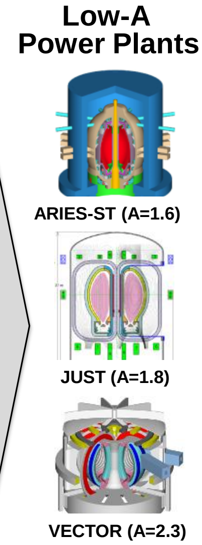
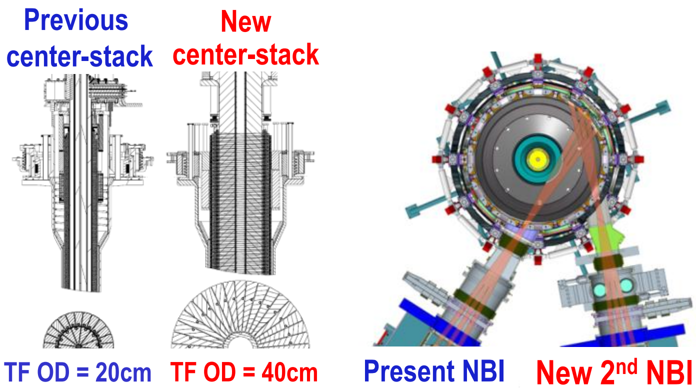


Figure 1.1.2: Example low-A tokamak configurations and parameters ranging from NSTX to Power Plants.

The ST configuration also faces many challenges that must be overcome before the ST could be utilized for PMI or FNSF applications [19]. First, both PMI and FNSF applications require very long pulse-lengths (10^3-10^6s), which for STs with small or no solenoid flux implies steady-state operation. To-date, STs (including NSTX) have only sustained non-inductive current-drive fractions of ~65-75% in high-performance H-mode plasmas. Fully non-inductive current drive and sustainment must be demonstrated and the integrated plasma performance assessed before any long-pulse next-step ST can be reliably designed. For the FNSF mission, neutron damage to the centerstack precludes usage of a conventional ohmic solenoid, so non-inductive current start-up and ramp-up are beneficial or required in addition to the requirement for non-inductive sustainment. Further, for the FNSF mission, compact ST-based devices with acceptable auxiliary power are only achievable if the energy confinement time is above the ITER ELMy H-mode scaling. While the ion thermal confinement in ST plasmas is routinely near neoclassical levels in beam-heated H-mode plasmas, thermal electron transport is the dominant power loss channel and remains relatively poorly understood. Long-pulse divertor solutions are also required for an FNSF of any aspect ratio and should be developed in present devices and in either a next-step PMI facility and/or the first phase of FNSF operation. Lastly, the development of radiation-tolerant magnets for normal-conducting TF coils in an ST-FNSF would also be required.

A major goal of the next 5 year plan period (2014-2018) is to address the key physics challenges of the ST identified above while advancing predictive capability for toroidal magnetic confinement more broadly. The primary enabling capability for the next 5 year period of NSTX operation is the major upgrade of NSTX presently scheduled to be completed in 2014.

Figure 1.1.3: (left) Comparison of previous vs. new center-stack of NSTX-U, (right) comparison of present and new 2nd NBI.



The upgraded NSTX facility (i.e. NSTX Upgrade/NSTX-U) consists of three new major components: (1) a new center-stack capable of doubling the toroidal field, tripling the solenoid flux, and increasing the flat-top duration up to a factor of 5, (2) a second more tangentially injected neutral beam to double the plasma heating and external current drive while also increasing the current drive efficiency and controllability, and (3) structural enhancements to withstand up to a factor of 4 increase in electromagnetic loads enabling a doubling of the plasma current. The new centerstack and 2nd NBI of NSTX-U is shown pictorially in Figure 1.1.3.

The doubling of the toroidal field, plasma current, and NBI heating will provide data critical to the determination of the scaling of thermal confinement as a function of field and current while maintaining access to high β. Expressed in terms of dimensionless parameters, a key physics goal of NSTX-U is to access 3-5 times lower collisionality ν* at similar β and shaping (κ, δ) as NSTX to assess transport, stability, and non-inductive start-up and sustainment at dimensionless parameters much closer to those expected in an ST-based FNSF as shown in Figure 1.1.4. In particular, it is noteworthy that NSTX and MAST have measured a strong (nearly inverse) scaling of normalized confinement with ν*, and if this trend holds at low ν*, high fusion neutron fluences could be achievable in very compact ST-FNSF devices.

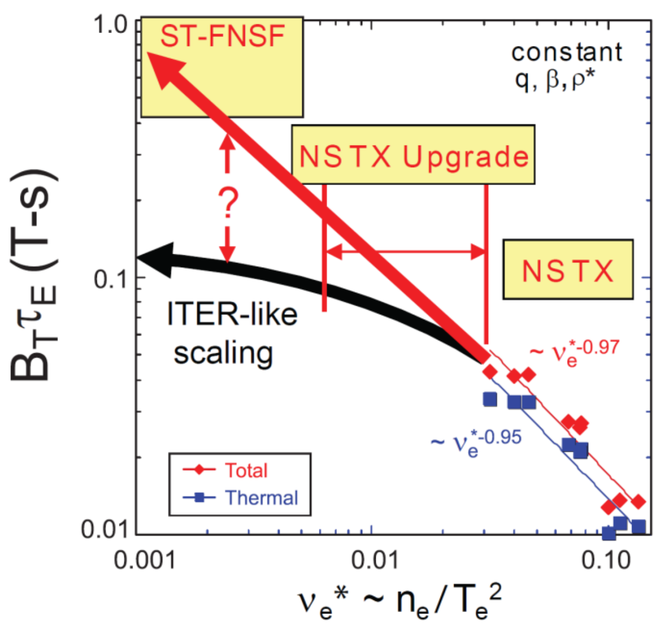


Figure 1.1.4: Product of toroidal field (B_T) and energy confinement time (τ_E) versus ν^*_e for NSTX and projections for NSTX-U and ST-FNSF for ITER H-mode and ST confinement scalings.

Achieving high $\beta_N \sim$ 5-6 at increased aspect ratio is important not only for accessing the unique physics regime of high β at reduced ν*, but also for assessing the ability to access the required normalized performance of next-step STs as shown in Figure 1.1.5. Beyond extending toroidal plasma predictive capability and establishing the physics basis for possible ST applications, NSTX-U also has a 5 year plan goal of accessing performance levels prototypical of next-steps and approaching Pilot-Plant regimes.

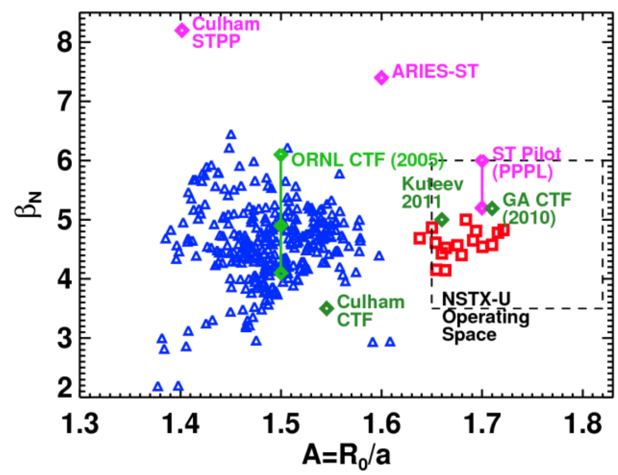


Figure 1.1.5: Achieved β_N vs. aspect ratio A for NSTX (triangles, squares) and values for next-step STs.

Figure 1.1.6 shows that the bootstrap current fractions projected to be accessible in NSTX-U overlap with PMI and FNSF applications and can access the lower-bound levels needed for a Pilot Plant. Similarly, the highest normalized confinement levels (based on extending NSTX values) overlap with the lower ranges needed for PMI, FNSF, and Pilot regimes. The expected high $\beta_N \sim$ 5-6 is sufficient for PMI and FNSF applications and could also support Pilot Plant requirements. Lastly, the highest levels of normalized heat flux P/S accessible in NSTX-U touch on the lower-bounds expected in PMI, FNSF, and Pilot applications, and are within a factor of 2 of providing prototypical heat fluxes.

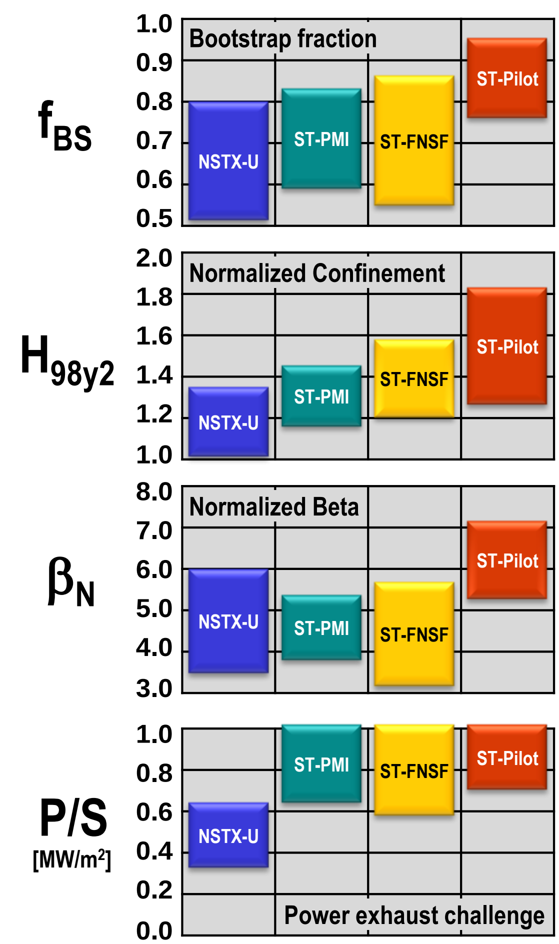


Figure 1.1.6: Normalized parameters projected for NSTX-U compared to possible next-step STs.

Thus, in summary, NSTX-U is expected to inform the achievable performance of all next-step ST options while also contributing strongly to conventional aspect ratio tokamak predictive capability and high-performance scenarios.

1.2 Long-Range Goals and Plans for NSTX Upgrade

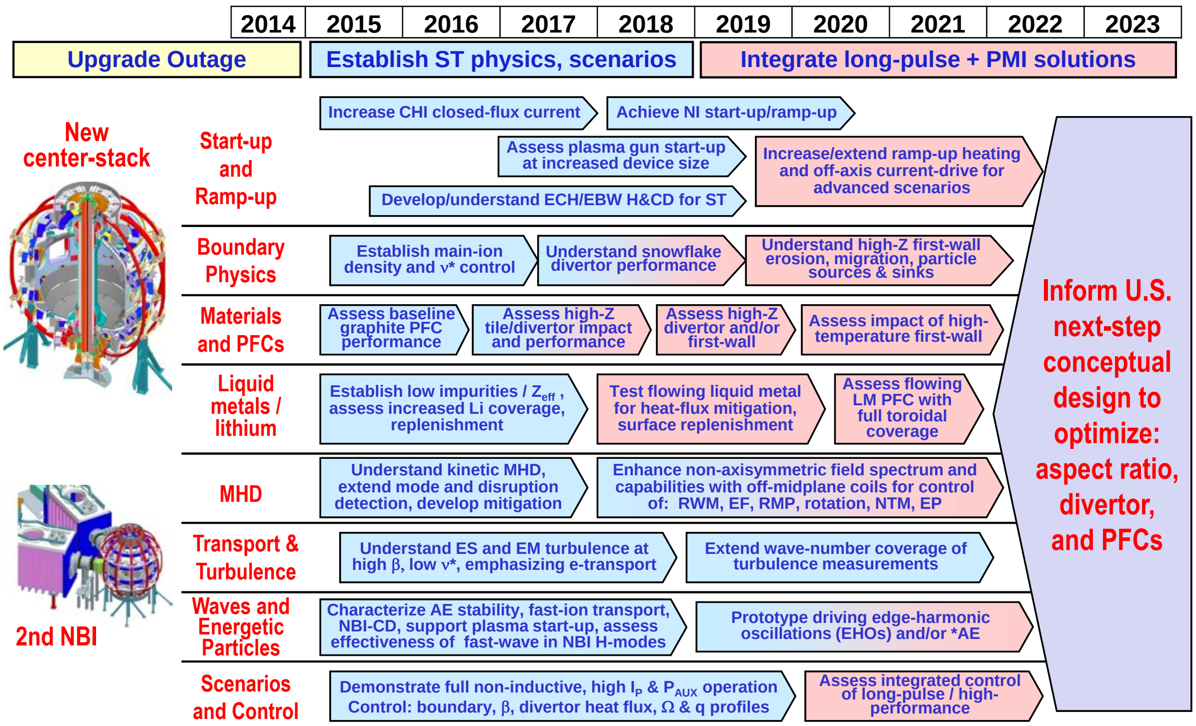


Figure 1.2.1.1: Long-range goals and plans for NSTX-U research in support of PMI research, FNSF, ITER, and DEMO assuming incremental funding of 10% above base funding.

1.2.1 Overview

The NSTX-U research team has developed a comprehensive long-range (10 year) set of goals and plans in support of advancing the ST configuration as a next-step PMI or FNSF facility and to address important scientific and operational issues for ITER and a DEMO. As shown in Figure 1.2.1.1, this long-range plan is divided into 2 phases. The first phase extends from 2014-2018 and emphasizes extending the ST physics basis and demonstrating high-performance operational scenarios including advanced divertor operation. At the end of this first 5 year period, the potential of the ST as a viable next-step candidate can be evaluated by considering NSTX-U (and MAST-U) progress toward achieving non-inductive start-up, ramp-up, and sustainment, reducing high divertor power loads, understanding electron thermal transport, and achieving high plasma [19]. The second phase (2019-2023) consists of further increasing the pulse-length and assessing the viability of integrating high-performance plasma scenarios (100% non-inductive, high β, high confinement) with next-step-relevant long-pulse PMI solutions such as high-Z solid and low-to-mid-Z liquid metal plasma facing components. This second phase of research would strongly contribute to next-step facility design and development independent of whether such a facility was low aspect ratio or conventional aspect ratio.

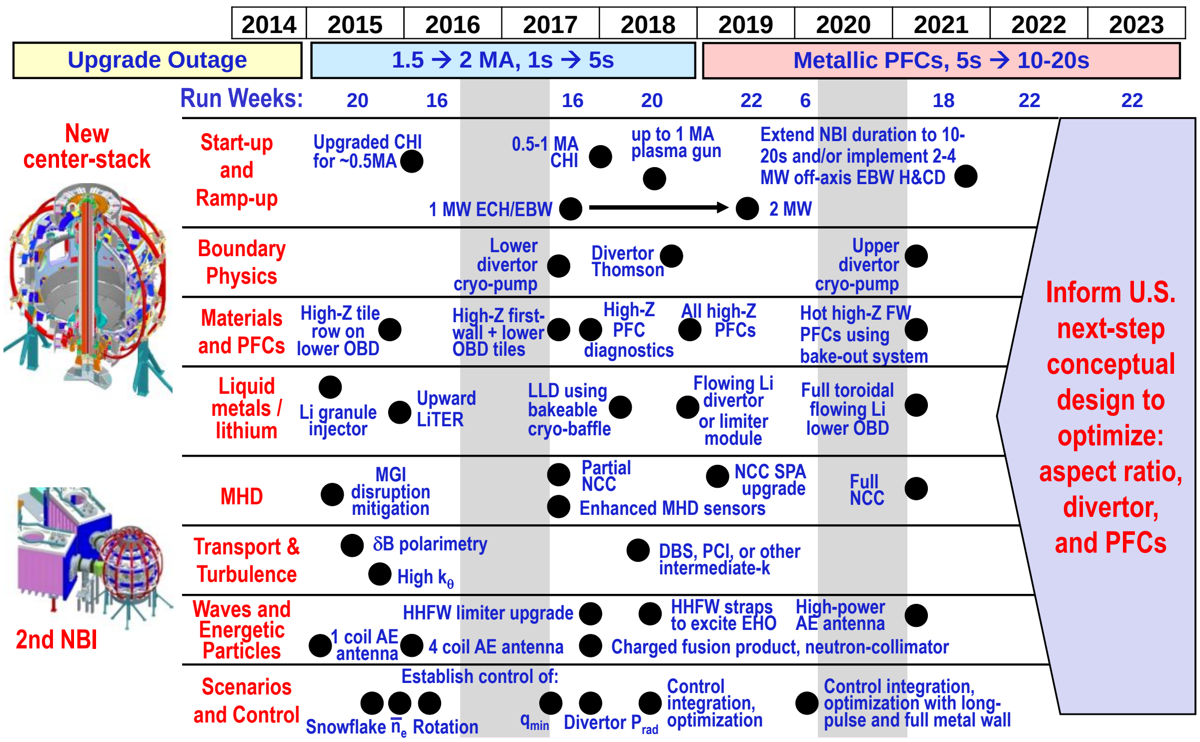


Figure 1.2.1.2: Proposed tools supporting long-range goals and plans from Figure 1.2.1.1 and assuming incremental funding 10% above the base 5 year plan guidance funding level. Vertical grey bars represent ~1 year outage periods utilized to implement major in-vessel upgrades.

Figure 1.2.1.2 shows the tools proposed to support the long-range goals and plans. The plans and tools shown in Figures 1.2.1.1 and 1.2.1.2 would provide a major contribution to the U.S. fusion program in support of developing predictive capability for magnetic fusion and for design of a next-step PMI or FNSF facility. The program and upgrade plan as shown assumes an incremental funding level 10% above base NSTX-U 5 year plan guidance funding. This phased research approach is also feasible at lower funding levels but would require additional time to implement. The plan above also proposes operation for an average of 18 run weeks per operating year which is at or near the highest level achieved over the operating history of NSTX. Additional run-weeks (up to an approximate maximum of 24) could be achieved by reducing both facility upgrades and the duration of associated outage periods.

Plan overviews assuming incremental and lower funding levels are provided in the sections that follow. It should be noted that these plans emphasize new capabilities beyond those already implemented on NSTX, which are expected to also be available on NSTX-U (see Chapter 10 for more details).

1.2.2 High-level Goals for the NSTX-U 5 Year Plan Period

To guide the prioritization of research and facility upgrades for the near-term, five high-level goals have been established for the NSTX-U five year plan period of 2014-2018:

1. Demonstrate stationary 100% non-inductive operation at performance that extrapolates to $\geq$ 1MW/m^2 neutron wall loading in FNSF
2. Access reduced ν* and high-β combined with the ability to vary q and rotation to dramatically extend ST plasma understanding
3. Develop and understand non-inductive start-up and ramp-up to project to ST-FNSF operation with small or no central solenoid
4. Develop and utilize the high-flux-expansion "snowflake" divertor combined with radiative detachment to mitigate very high heat fluxes
5. Begin to assess high-Z PFCs plus liquid lithium to develop high-duty-factor integrated PMI solutions for SS-PMI and/or FNSF facilities, and beyond

In support of these high-level goals, and expressed in terms of the key scientific issues to be resolved, a cross-cutting set of tools has been identified. These tools are given highest priority in the plan and are included in both the incremental and base 5 year plan budget scenarios. These tools and the issues they help resolve are shown pictorially in Figure 1.2.2.1. These and other tools available to achieve the high-level goals are described for 3 different funding scenarios in the following sections: 1.2.3: Incremental, 1.2.4: base, and 1.2.5: FY2013 Administration levels.

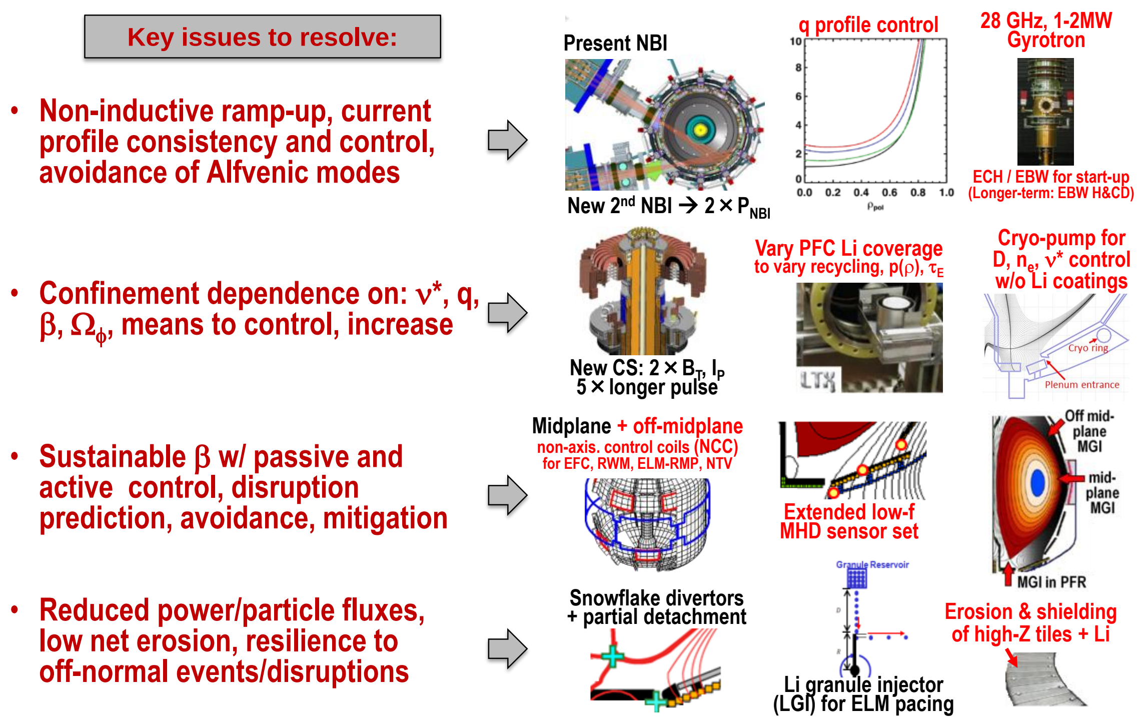


Figure 1.2.2.1: Key scientific issues to be resolved and proposed high-priority research tools in support of the 5 high-level goals. Existing/early tools are labeled in black, new/additional tools are labeled in red.

1.2.3 Goals, Plans, and Tools with Incremental 5 Year Plan Funding

First, to recapitulate, the incremental 5 year funding plan is shown above in Figure 1.2.1.2 in Section 1.2.1. In strong support of the development of non-inductive current formation and ramp-up in the ST, the CHI start-up current will be progressively increased to the 0.5-1 MA range exploiting the higher toroidal field of NSTX-U, plasma guns would be implemented and tested to high current (pending successful development on Pegasus), and 28GHz ECH heating of 1MW (later up to 2MW) implemented to heat CHI and/or high-current (~1MA) plasma gun target plasmas from the 10's of eV range to the 100's of eV range. Increased electron heating of the helicity injection plasma is needed to provide higher electron beta for effective high-harmonic fast-wave (HHFW) absorption (see Chapter 7), and to extend the helicity-injection-formed plasma current decay time from ~1-10ms to ~10-100ms (see Chapter 8) to enable sufficient time to establish gap control for efficient HHFW coupling. Without heating from either ECH or HHFW, NBI heating and current drive will likely not be effective either due to shine-through in the low-density CHI target, and/or because the current drive efficiency is likely too low to provide enough current drive to counteract the rapid decay of the helicity injection plasma (see Chapters 8 and 9). Later in the research program, the same gyrotrons would also be used for high-power EBW coupling, heating, and current drive studies in over-dense plasmas during start-up and in high-performance NBI-heated H-modes to begin to assess EBW as a possible long-term complement to (or eventual alternative to) NBI heating and current-drive (see Chapter 7).

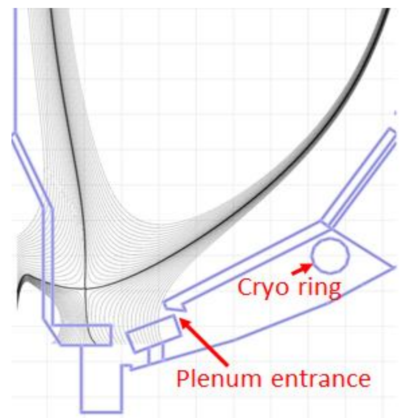


Figure 1.2.3.1: Proposed lower divertor cryo-pump

In boundary physics research, a divertor cyro-pump (see Chapter 4) as shown in Figure 1.2.3.1 would be implemented to provide controllable deuterium pumping without relying on lithium coatings, to compare to lithium-conditioned walls, and to support access to reduced collisionality. Achievement of long-pulse density control to support long-pulse scenario development on NSTX-U is a major motivating factor for the divertor cryo-pump. Further, divertor Thomson scattering would be implemented to measure electron density and temperature near the divertor target plate to support detailed assessments of the validity of leading divertor models for conventional and snowflake divertors for attached and detached conditions. The snowflake divertor and detachment will be systematically developed and utilized to mitigate the high heat fluxes expected for the higher power and current in NSTX-U.

Also in boundary physics research and contingent upon the development of sufficient divertor heat-flux mitigation strategies, the impact and performance of high-Z divertor PFCs (either Mo/TZM or W) would be assessed starting with a single row of high-Z tiles on the lower outboard divertor combined with nearly complete high-Z first-wall coverage of low heat-flux regions using high-Z coated graphite tiles all to support studies of low-hydrogenic-retention PFCs for next-step facility options. The single rows of high-Z tiles in the outboard divertor regions would provide initial data on the survivability of PFC designs in the high-heat-flux and potentially high-disruption-load regions of the chamber while also providing erosion and melting data (see Chapter 5). The inboard divertor PFCs would remain graphite (until outboard divertor high-Z PFC designs have been proven) to reduce risk of PFC damage in the high-triangularity plasma operation that is expected to be mostly routinely used for NSTX-U (see Chapter 9).

In lithium/liquid metal research, the lithium granule injector (LGI) originally developed for NSTX and successfully tested on EAST for triggering ELMs would be tested on NSTX-U as a means of triggering high-frequency (up to several hundred Hz) ELMs for controlling edge impurities (C and/or Mo/W) for long pulses and for replenishing lithium near the divertor strike-point during a discharge. Further, an upward-pointing lithium evaporation (LiTER) system would be implemented to increase lithium coating coverage and to reduce the exposed graphite PFC surface area to reduce C impurity accumulation in no/small ELM regimes that typically occur with strong lithium wall conditioning.

An important technical detail of the baffle plate of the cryo-pump is that it would be protected using primarily graphite PFCs, and would therefore require bake-out capability to remove water and other impurities following major vents. This bake-out capability also enables controlled (initial) temperature variation of the outboard divertor PFCs and variation of the lithium phase. This control capability will greatly expand studies of vapor shielding of high heat fluxes by lithium (see Chapter 5). Lastly, in support of the long-term goal of developing the understanding and technology of flowing lithium systems capable of providing a replenishable PFC surface either to mitigate erosion and/or high heat flux (through vapor shielding and/or radiation), a prototype flowing lithium module will be developed in the laboratory and implemented and tested in NSTX-U (if technically ready) near the end of the 5 year plan (see Chapter 5).

In MHD research, disruption mitigation tests using massive gas injection (MGI) planned for the NSTX 2011-2012 run campaign will be carried out to assess the fueling/assimilation dependence on poloidal location of injection in support of NSTX-U operation, ITER, and FNSF. The RWM state-space controller developed and utilized for NSTX will be tested and optimized to support long-pulse NSTX-U operation above the no-wall stability limit. Enhanced MHD sensors (divertor RWM δB sensors and possibly shunt tiles for disruption/SOL halo currents) would be implemented to improve disruption and/or ELM evolution understanding and/or enhance non-axisymmetric mode detection and control. A partial set (i.e. 6-12 coils vs. 24 for a full set) of off-midplane non-axisymmetric control coils (NCC) would be installed on the primary passive plates to dramatically expand the spectral flexibility of the 3D field system on NSTX-U in support of neoclassical toroidal viscosity (NTV) damping research, error field and RWM control, rotation profile control, and RMP ELM control.

In turbulence and transport research, the high-k system used to investigate ETG turbulence on NSTX will be re-configured to better sample the wave-number range of the projected strongest ETG fluctuations and will include high-k_θ measurements. Polarimetry will be implemented to attempt to measure electromagnetic fluctuations from micro-tearing instabilities that non-linear gyrokinetic simulations indicate strongly contribute to electron thermal transport at high β and collisionality. Lastly, Doppler Backscattering (DBS) will be implemented to measure zonal flows and geodesic acoustic modes (GAMs) generated through non-linear turbulence interactions and thought to regulate low-k turbulence levels and which likely influence edge transport barrier (i.e. H-mode) formation.

In Wave Heating and Current Drive research, the high-harmonic fast-wave (HHFW) system will be utilized for heating low-current start-up plasmas from induction and/or helicity injection to support increasing the plasma temperature and duration to provide a target for NBI plus bootstrap current over-drive for non-inductive ramp-up. Further, HHFW acceleration of fast-ions and associated power absorption will be studied at higher toroidal field and correspondingly lower cyclotron harmonic number. The effectiveness of HHFW heating and current drive in high-performance NBI H-modes will be assessed, and if shown to be effective for ~0.5-1s timescales, the antenna limiter will be upgraded. Depending on the HHFW coupling efficiency as a function of antenna-plasma gap, and the fast-ion heating of the limiter from NBI injection, a more robust and/or actively-cooled outboard limiter may be needed to support simultaneous high-power NBI and HHFW for long pulses. Pending assessments of the operational impact of using a reduced number of HHFW straps, several HHFW antenna straps may be utilized for other purposes potentially including excitation/driving of edge harmonic oscillations (EHOs) (observed previously on NSTX) as a possible means of increasing edge particle transport.

In Energetic Particle research, the existing NSTX fast-ion diagnostic suite will be utilized to investigate the fast-ion density and current-drive profiles from the new 2nd NBI and to compare the measurements to theory (i.e. TRANSP/NUBEAM). The prototype antenna planned to be used for measuring Alfvén Eigenmode (AE) damping rates will be utilized on NSTX-U and quickly extended to 4 coil operation. TRANSP and FIDASIM simulations (see Chapter 6) indicate that NBI diagnostic signals could be significantly attenuated at the highest plasma densities anticipated in NSTX-U due to beam ionization over shorter injection distances. If attenuation is problematic and causes low signal-to-noise on fast-ion profile diagnostics, the fast-ion diagnostic set would be extended to include fusion source profile arrays and/or a neutron-collimator. Alfvénic instability excitation and fast-ion redistribution and loss will also be measured as a function of injection radius using the new 2nd NBI and compared to theory.

In advanced scenarios and control research, scenarios with non-inductive current fraction near 100% will be systematically explored, and high-current, partial inductive operation aiming for the facility maximum performance goal of 5 second pulses at I_P=2.0 MA and B_T=1.0T will also be explored as a key means of accessing low collisionality in NSTX-U. Emphasis will be placed on axisymmetric control development including control of the plasma boundary, vertical position, beta, and divertor heat flux control including snowflake divertor and detachment control. Density feedback control will also be developed in support of physics studies and scenario optimization and is expected to be especially important once the divertor cryo-pumping system is operational. Profile control development will be pursued emphasizing core plasma safety factor and rotation control using the 2nd NBI and 3D fields. Disruption detection and slow and fast ramp-down methods will be developed, and MGI tested and utilized as appropriate. Lastly, advanced scenario research will assess the achievable integrated performance of NSTX-U scenarios and then project to next-step ST device design.

1.2.4 Goals, Plans, and Tools with Base 5 year Plan Funding

The long-range goals of the NSTX-U research program assuming base funding are similar to those with incremental funding, but several goals and tools are necessarily delayed or deferred in the base funding case as shown in Figure 1.2.4.1. Highest priority is placed on establishing the ST physics and operational basis for next-step applications, and long-pulse PMI development and several of the more resource-intensive tools are deferred.

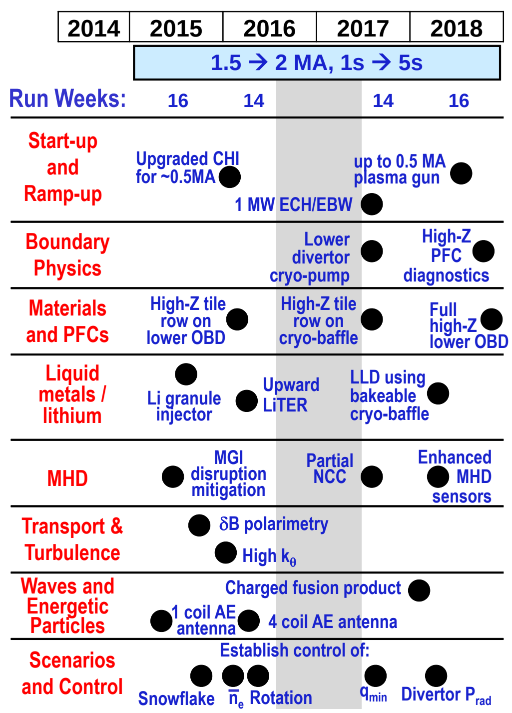


Figure 1.2.4.1: Proposed tools with base funding. Vertical grey bar represents ~1 year outage utilized to implement major in-vessel upgrades.

The development of non-inductive start-up and ramp-up is a unique ST challenge and remains strongly supported in the base plan to assess the viability of the ST as an FNSF. Capabilities for accessing up to 0.5MA of transient CHI are supported, as is up to 1MW of ECH/EBW for ECH heating and pulse-length extension of start-up plasmas for tests of HHFW/NBI heating and current over-drive ramp-up. However, in the base funding scenario, high-current (~1MA) CHI or plasma-gun research cannot be supported. Increased ECH/EBW power (from 1MW to 2MW) would be deferred several additional years.

Access to, and investigation of, reduced collisionality is a major research goal of NSTX-U and impacts nearly all topical science groups. Electron density control has historically been difficult in NSTX with only boron and/or lithium conditioned graphite plasma facing components. In the case of boronized walls, deuterium density pumping is difficult with wall pumping alone but carbon accumulation is acceptable (Z_{eff} ~ 2-3). With strong lithium evaporation, low deuterium density has been sustained, but the lack of ELMs can lead to carbon ion accumulation and Z_{eff} ~ 3-4. ELM triggering by 3D fields has been utilized in NSTX to help control carbon accumulation and achieve stationary electron density, but only near Greenwald fractions in the range of 0.8-1. Improved control of the Li evaporation rate and increased total Li deposition to control both the deuterium pumping and the impurity inventory (via ELM-induced flushing of impurities) may be able to provide sustained and controllable density control in NSTX-U. However, such techniques have not yet been developed for NSTX-U physics studies.

As a result, at the present time, for all funding scenarios, it appears prudent to plan for a divertor cryo-pump to provide a reliable and controllable means of deuterium inventory control with a configuration like that shown in Figure 1.2.3.1. The quantitative rationale for this choice is as follows: Using a definition of normalized collisionality ν^* in which the electron-ion collision frequency is normalized to the electron bounce frequency and assuming fixed geometry, it can be shown that $\nu^* \propto f_{GW}^3 / B_T \beta_N^2$. Here f_{GW} is the line-average electron density normalized to the Greenwald density limit, B_T is the toroidal magnetic field, and β_N is the normalized toroidal beta. Increased toroidal field and maintenance of high β_N are clearly important for accessing reduced collisionality. However, reducing and controlling electron density is especially important since $\nu^* \propto f_{GW}^3$. Thus, assuming naturally occurring or triggered ELMs can provide impurity density control, control of the deuterium via cryo-pumping to control the Greenwald fraction is projected to be an effective means of accessing low ν^*.

Due to resource limitations in the base plan, divertor Thomson scattering would be deferred and the conversion to high-Z PFCs would also be significantly delayed with only the conversion of the lower outboard divertor to high-Z completed by the end of the 5 year plan. The flowing lithium module would not be tested during the 5 year plan period, but usage of the bakeable cryo-baffle as a liquid lithium divertor would be retained in the base plan.

In MHD research, disruption mitigation tests would be carried out and the installation of a partial NCC retained, but improved MHD/RWM sensors would be implemented at a later date and upgrades to the NCC switching power amplifiers (SPA) to provide improved spectral control of the 3D fields would be delayed past the end of the 5 year plan period.

In turbulence and transport research, diagnosis of ETG (high-k) and micro-tearing turbulence (polarimetry) potentially responsible for electron transport are considered vital to establishing a predictive capability for ST transport and are supported in the base plan. However, DBS and other possible intermediate-k diagnostics (such as PCI) would be deferred.

In wave and energetic particle research, there are insufficient resources to implement a long-pulse HHFW antenna limiter and an EHO excitation antenna in the 5 year plan period. The decision to proceed with either will be based on short-pulse assessments of HHFW heating in H-mode plasmas and on additional data analysis of EHO modes. Energetic particle measurements of the fusion source profile using a charged fusion product diagnostic would be retained in the base plan to support diagnosis of plasmas at high current and density.

Lastly, for advanced scenario development and control, the upgraded center-stack and the 2nd NBI of NSTX-U provide nearly all of the necessary actuator capability for the proposed research, and additional diagnostics and algorithms are either planned to be provided by collaborators and/or are expected to be sufficiently funded in the base program. Thus, the base and incremental plans for advanced scenario and control development are largely the same.

In terms of run-time, the base-plan supports an average of 15 run weeks per operating year which is roughly typical of NSTX operation. Additional run-weeks could be achieved by reducing facility upgrades and associated outage periods.

1.2.5 Goals, Plans, and Tools with Reduced Funding Levels

The near-term deliverables of the NSTX-U program assuming a funding level consistent with the FY2013 Administration Request budget would be significantly delayed and the scope substantially reduced relative to the 5 year plan base budget guidance as shown in Figure 1.2.5.1. Highest priority remains placed on establishing the ST physics and operational basis for next-step applications, but long-pulse PMI development and nearly all major facility upgrades are deferred. In this funding scenario, only 3 years of the 5 year plan have any significant operational time for NSTX-U, and during those 3 years there are only 12 run weeks each year.

As stated previously, the development of non-inductive start-up and ramp-up is a unique ST challenge and must remain supported at some level in any funding scenario. Capabilities for accessing up to 0.5MA of transient CHI are supported, but both the 1MW of ECH/EBW for ECH heating and CHI pulse-length extension are deferred, as is testing of a plasma-gun (at any current level) for non-inductive start-up.

Despite the attractiveness of divertor cryo-pumping for density control, the cost and time needed for design and implementation of cryo-pumps would delay implementation for 1-1.5 run-years relative to the base and incremental funding scenarios. Given the necessary delay in implementing cryo-pumping, means of possibly reducing impurity accumulation (via triggered

ELMs using lithium granule injection) and extending the duration/coverage of lithium pumping of deuterium (via upward evaporation) are the highest priority lithium research activities. High-Z PFC coverage in the divertor region would be reduced to a single row of high-Z tiles.

In MHD research, disruption mitigation tests would be carried out and enhanced MHD sensors would be implemented, but mitigation studies are delayed ~1 year and no NCC coils would be installed during the 5 year plan period.

In turbulence and transport research, diagnosis of ETG and micro-tearing turbulence potentially responsible for electron transport are considered vital to establishing a predictive capability for ST transport and remain supported in all funding scenarios due to modest resource requirements.

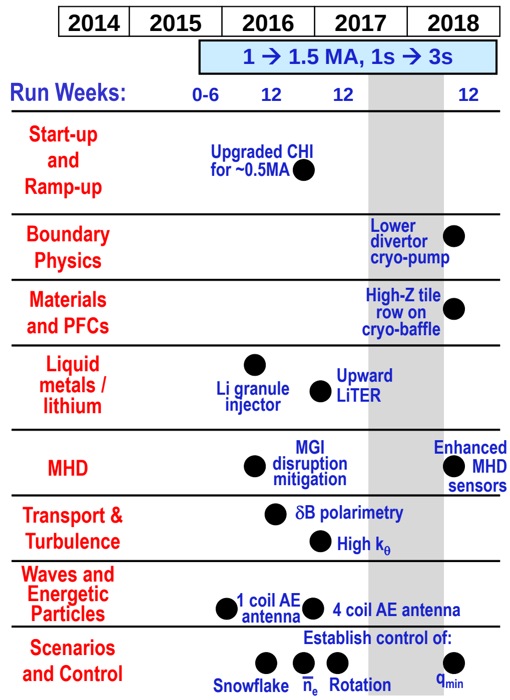


Figure 1.2.5.1: Proposed tools with FY2013 budget. Vertical grey bar represents ~1 year outage period utilized to implement major in-vessel upgrades.

In fast-wave research, there are insufficient resources to implement a long-pulse HHFW antenna limiter or an EHO excitation antenna. Thus, no major upgrades are implemented for HHFW (or ECH/EBW) during the five year period. In energetic particle research the AE antennae are retained since they are relatively inexpensive, but enhanced fast-ion profile diagnostics are deferred which will reduce data fidelity at high plasma density.

Lastly, for advanced scenario development and control, the control algorithm development would be delayed ~1 year and divertor radiation control would not be implemented during the 5 year period, but control development for snowflake, density, rotation, and q profile is retained.

Even with funding for the NSTX-U research program at the FY2013 Presidential budget level, a scientifically compelling and highly productive research program can be achieved. This opportunity is enabled by the substantial investment and capability implicit in the new centerstack and 2nd NBI of the NSTX Upgrade project. Nevertheless, full utilization and exploitation of NSTX-U within the 2014-18 five year plan period is only possible with funding at the 5 year plan base or incremental levels.

1.3 Organization of the Five Year Plan and Team

The NSTX Upgrade research program is organized by topical science area into "Topical Science Groups" (TSGs). Each TSG represents an important area of tokamak/ST research of sufficient importance and specialization that a dedicated group is required to lead research in this area. The TSG leadership structure has one TSG leader, one deputy leader (typically both experimentalists), and in almost all cases a third leader for theory and modeling support of the experiments. The inclusion of theory and modeling in the leadership structure is designed to enhance the scientific coupling between theory and experiment in NSTX-U research. TSG leadership responsibilities/actions include:

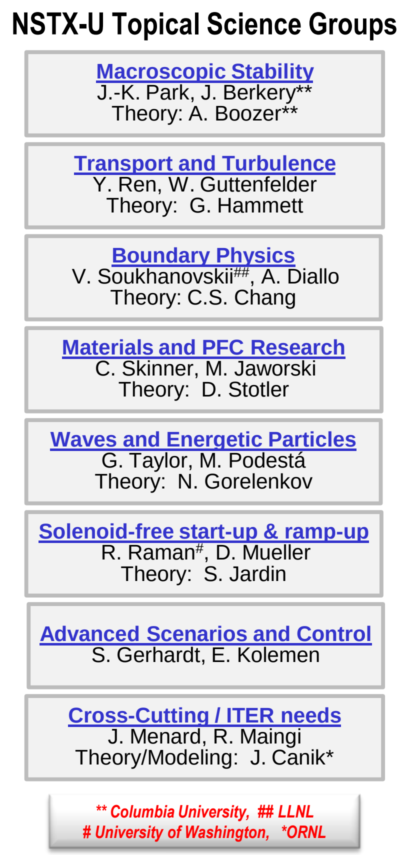


Figure 1.3.1: Topical Science Groups of the NSTX-U research program.

1. Organize, propose, and execute experiments to achieve milestones and address priorities and thrusts
2. Determine and address the highest priority scientific issues within a topical area through group discussion and including the NSTX-U team in TSG meetings
3. Organize the NSTX-U Research Forum guided by (but not limited to) these priorities
4. Define draft scientific and performance milestones utilizing expertise of the TSG
5. Define facility and theory resources required to achieve research goals
6. Aid dissemination of results (help Physics Analysis & Simulation Division)
7. Provide brief summaries of TSG scientific progress at team meetings and other venues
8. Assist and report to the NSTX-U Program and Project directors

The topical science group structure (as of January 2013) is shown in Figure 1.3.1. The NSTX-U five year plan chapters of this document are organized with a very similar structure and were coordinated and/or written in large part by members of the TSG leadership. The only significant differences between the TSG structure and 5 year plan chapter structure are: 1) the Waves and Energetic Particles TSG research is divided into two chapters (Chapter 6: Energetic Particles and Chapter 7: Wave Heating and Current Drive), and 2) cross-cutting activities (such as cryo-pump physics design) and ITER needs and support are incorporated throughout the plan. Lastly, as described further in Chapter 11, NSTX-U collaborators play an essential role in defining and leading the NSTX-U program and strongly contributing to the 5 year plan.

1.4 Summaries of Five Year Plan Research Thrusts

1.4.1 Overview

The NSTX-U 5 year plan is organized into 11 chapters. There are 8 dedicated topical science plan chapters based on the TSG group structure described above. The chapters and corresponding topics are: (1) Overview, (2) Macroscopic Stability, (3) Transport and Turbulence, (4) Boundary Physics, (5) Materials and Plasma-Facing Components, (6) Energetic Particles, (7) Wave Heating and Current Drive, (8) Plasma Formation and Current Ramp-up, (9) Plasma Sustainment: Advanced Scenarios and Control, (10) Facility Capabilities and Upgrades, and (11) Documentation of Collaborator Plans. Each of the 8 topical science areas (Chapters 2 through 9) have identified 2 to 4 research "thrusts" that will be carried out to achieve the high priority research goals identified in Section 1.2.2. Brief summaries of the research thrusts are described below to provide an overview of the research goals of the NSTX-U 5 year plan. More detail on thrust motivation and plans can be found in corresponding chapters of the 5 year plan.

1.4.2 Macroscopic Stability Research Thrusts

1.4.2.1 Sustainment of macroscopic stability

Global MHD instabilities (e.g. kink/ballooning mode, resistive wall mode (RWM), neoclassical tearing mode) are critically important to avoid or control as they lead to plasma disruption, terminating the discharge and leading to large, potentially damaging electromagnetic forces and heat loads on the structure of fusion producing devices. While many targeted performance parameters have been reached in world tokamaks, such plasmas will need to be sustained for far longer pulse lengths in machines such as FNSF, ITER, and DEMO than have been produced to date. Research has therefore changed focus to examine sustained global mode stability over long pulses and to examine profile evolution for routine long pulse operation at high beta and at high non-inductive current fraction. Common to the following studies is the unique physics understanding and control ramifications that come from such operation, and the understanding and prediction of the effect of excursions from this condition due to transient behavior. It is especially important to realize that plasma operation under marginal stability points (set, for example by plasma beta, internal inductance, rotation) is insufficient to ensure disruption-free, continuous operation in either ITER inductive or advanced scenarios due to these transients in plasma profiles. Such transients can rapidly change a stable operational point to an unstable plasma state. Therefore, understanding plasma stability gradients vs. key profiles affecting stability is essential for all operational states in ITER. NSTX-U will provide key capabilities for critical physics understanding based on present research in plasma operation regimes applicable to ITER and future magnetic fusion devices. NSTX-U will be unique in its operation in non-inductively driven plasmas. This major operational regime for NSTX-U, which may require greater control, provides a unique laboratory to test advanced stability physics. Additionally, NSTX-U will operate in the unique high beta ST operational space, which will allow performance of advanced stability control in an operating space where disruptivity is *not* most probable at the highest β_N, or β_N/l_i. These considerations motivate the following thrust:

Thrust MS-1: Understand and advance passive and active feedback control to sustain macroscopic stability

> *With significantly expanded profile control capabilities (e.g. q, plasma rotation), NSTX-U will allow greater ability to vary these important profiles for investigations of how to prevent plasma disruptions excited by profile excursions. Additionally, profiles theoretically expected to improve stability will be attempted to be sustained at high performance, including fully non-inductive operation. Real-time models to determine favorable profiles for stability will be developed from experimental testing of full theoretical models. Research will focus on the effect of reduced plasma collisionality and energetic particles on the kinetic stabilization of disruptive instabilities, and sustaining stability. Shape and aspect ratio changes, the effect of advanced divertor configurations, and 3D multi-mode spectrum will be assessed. ITER-relevant low plasma rotation regimes will be evaluated. Disruption prevention research will be conducted as a combination of advanced active mode control techniques and active profile control. For example, instability onset leading to global modes growing on a relatively fast timescales (milliseconds) will be actively controlled by an expanded model-based RWM state space controller, while concurrent control of q and plasma rotation profile working on slower timescales (~100ms) will steer the plasma toward stability. Instabilities occurring before plasmas attain full current, and the ability to control largely internal modes will be addressed.*

1.4.2.2 Utilization of non-axisymmetric (3D) magnetic fields

A small non-axisymmetric (3D) field almost always exists in tokamaks, due to imperfect primary magnets and surrounding conductors and machine components. Tokamaks are highly sensitive to 3D fields, which can cause unnecessary transport and instability and even lead to a disruption if not properly compensated. On the other hand, 3D fields can be greatly beneficial if properly controlled, by timely inducing new neoclassical process with non-ambipolar transport and by consequently modifying equilibrium profiles and macroscopic stability, as well known by edge localized mode (ELM) control using resonant magnetic fields and resistive wall mode (RWM) and tearing mode (TM) control using non-resonant magnetic fields. Therefore, it will be critical to achieve the controllability as well as the predictability of these 3D field applications, in order to improve plasma stability and performance in the next-step devices such as FNSF, ST Pilot, and ITER.

There has been substantial progress in understanding of 3D field effects. Research on the n=1 resonant error field correction in NSTX using the ideal perturbed equilibrium code (IPEC) highlighted the importance of plasma response and the developed method has been actively applied to ITER along with the paradigm change in the error field correction. Also, the n=3 non-resonant error field correction was routinely used to maximize the toroidal angular momentum based upon successful validation across experiment and neoclassical toroidal viscosity (NTV) theory. In general, the study of magnetic braking in NSTX using 3D fields demonstrated the importance of NTV physics in tokamaks. Magnetic braking will be an essential tool to control the toroidal rotation and thereby suppress various instabilities in next-step devices. The NTV physics behind it is highly complex depending on regimes and needs further and deeper studies, but nonetheless a number of fundamentals in NTV physics have been theoretically understood, numerically verified, and validated from experiments in various devices including NSTX, DIII-D, and KSTAR. Another well-known effect of 3D fields is the local modification of transport and stability, either by non-ambipolar transport or by stochastic transport with magnetic islands. Although this is demonstrated by RMP ELM suppression in DIII-D, NSTX n=3 applications showed another possible 3D field effect by triggering ELMs rather than mitigating ELMs. All of this evidence points to the utility of 3D fields, but also to complex interactions between 3D fields and tokamak plasmas, and thus substantial research is still required.

NSTX-U will provide a highly relevant environment for 3D field research. Collisionality is one of the most important parameters to govern 3D field physics, and the low collisionality regime relevant for the next-step devices should be carefully and deeply explored. NSTX-U aims for higher temperature and low density operation, which will result in much lower collisionality. Also, the 6 independent switching power amplifiers (SPAs) will give more flexibility in field spectrum. Furthermore, the proposed non-axisymmetric control coil (NCC) will give unprecedented flexibility in 3D field studies, with two rows of coils that can produce static fields up to n=6 and rotating fields up to n=4. The poloidal field spectrum will also be as rich as the ITER RMP coils, if the NCC is combined with the present RWM/EF coil at the midplane and supplied by 6 SPAs. Initial studies show that a 2x6 coil set on the primary passive plates with a toroidal distribution of odd up-down parity is favorable for a range of applications. A possible progression of NCC implementation from a partial set (12 coils) to the full set (24 coils) in NSTX-U is shown in Figure 1.4.2.2-1. These considerations motivate the following thrust:

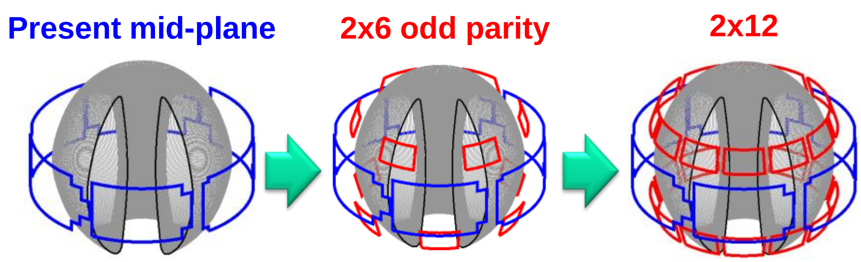


Figure 1.4.2.2-1: Possible progression of NCC coil implementation in NSTX-U.

Thrust MS-2: Understand 3D field effects and provide the physics basis for optimizing stability through equilibrium profile control by 3D fields

> *Expanded 3D field application capabilities will be utilized to study 3D field effects on resonant error field correction, and to develop the physics basis for control of toroidal rotation through non-resonant magnetic braking while simultaneously applying RWM control and dynamic error field correction. Non-ideal and non-linear island dynamics including neoclassical tearing modes, and the interplay with resonant and non-resonant error fields, will be studied together with strong shaping, high-β, and low collisionality. The NTV physics yielding magnetic alteration of the rotation profile will be systematically studied as a function of plasma collisionality as density control tools are improved during the 5 year plan period. This physics research will be greatly enhanced by the proposed NCC coils. The toroidal rotation profile will also be more widely varied using the 2nd off-axis NBI system. The combination of the new NBI, upgraded independent control of the present midplane coils, and proposed NCC will be utilized to vary and understand toroidal momentum transport for a range of magnetic field spectra and plasma collisionality conditions to develop the physics basis for magnetic rotation profile control to improve and optimize plasma performance.*

1.4.2.3 Disruption physics, avoidance, detection, and mitigation

A key issue for ITER and the tokamak/ST line of fusion devices in general is the avoidance and mitigation of disruptions. As a general goal, disruptions must be avoided. The research program in Thrusts MS-1 and MS-2 above as well as in Thrusts ASC-2 and 3 develop many of the necessary tools for disruption avoidance. These tools by themselves, however, are not sufficient to meet the stringent requirements for reliable tokamak/ST operation. For example, disruption "mitigation" techniques is of great importance to ITER, where methods to terminate a discharge without excessive thermal or mechanical loading of the plant are required. These considerations motivate the following thrust:

Thrust MS-3: Study and develop techniques for disruption prediction, avoidance, and mitigation, and understand disruption dynamics and in high-performance ST plasmas

> *The physics understanding gained in the balance of the macroscopic stability research will be applied to improve disruption prediction and avoidance. Kinetic stability physics models, low frequency resonant field amplification, RWM state-space control models, and expanded sensor input will be used to improve disruption prediction. Real-time implementation of these physical models and measurements will be used for disruption avoidance. The plasma dynamics resulting from the interplay of different stability controllers will be measured and simulated. For instance, resonant field amplification methods will be developed to assess global stability, and may be coupled to the β_N or rotation control algorithms to ensure that the plasma remains in a stable regime. The characteristics and dynamics of NSTX-U disruptions will be measured and modeled – in particular heat loading, the thermal quench, and the generation of halo currents – all to improve the understanding of the impact of disruptions in STs and tokamaks. Techniques for disruption mitigation will be developed and explored. Massive gas injection (MGI) will be explored: the poloidal angle dependence of mitigation efficacy will be examined, and modeling of the gas penetration will be done. A novel electromagnetic particle injector (EPI) will be developed and tested. This technology has the potential to very rapidly deliver large amounts of the material to the plasma.*

1.4.3 Transport and Turbulence Research Thrusts

1.4.3.1 Characterize global energy confinement

Future ST devices such as ST-FNSF are expected to operate at significanly higher toroidal field B_T, plasma current I_P, and heating power P_{NBI} than either NSTX or NSTX-U. To help establish the physics basis for future STs, which are generally expected to operate in lower collisionality regimes, it is important to characterize energy confinement over an expanded range of plasma parameters. Determining energy confinement scalings are especially important since compact (i.e. small-major-radius) ST FNSF devices will only be possible if sufficiently high confinement can be obtained. H-mode confinement studies in NSTX have shown that the global energy confinement exhibits a more favorable scaling with collisionality ($B\tau_E \sim 1/\nu^*_e$ – see NSTX data points in Figure 1.1.4) than that from ITER ELMy H-mode ITER98y,2 scaling ($B\tau_E \sim$ independent of ν^*_e – see black "ITER-like scaling" curve in figure 1.1.4). This strong ν^*_e scaling observed in STs unifies disparate engineering scalings between boronization ($\tau_E \sim I_P^{0.4} B_T^{1.0}$) and lithiumization ($\tau_E \sim I_P^{0.8} B_T^{-0.15}$) wall-conditioned plasmas. With a doubling of B_T, I_P and P_{NBI} and beams at different tangency radii, NSTX-U provides an excellent opportunity to assess the core confinement characteristics in regimes more relevant to future STs and to explore the accessibility to lower collisionality. These considerations motivate the following research thrust:

Thrust TT-1: Characterize and validate NSTX H-mode global energy confinement scaling in the lower collisionality regime of NSTX-U

> *NSTX-U researchers will re-establish and extend energy confinement studies and scaling to lower collisionality with higher B_T, I_P and NBI power ($B_T/I_P \leq 0.8T$ / 1.6MA in FY15 and 1.0T / 2MA in FY16 and beyond) for modest discharge durations (~1-3 seconds). Comparisons of the current and field scaling to the different scalings found on NSTX, which depended on wall conditions (lithiated or not) and different PFC materials, e.g. carbon vs high-Z will be carried out. Confinement will also be characterized in longer duration discharges that approach the magnet engineering limits (~5s flat-top) of NSTX-U and in advanced scenarios (i.e. fully non-inductive discharges) as they are developed. The range of collisionality for assessing global confinement will be extended to lower ν* by reducing the plasma density using a divertor cryo-pump. These experiments will be coupled with low-k and high-k turbulence diagnosis and gyrokinetic simulations and will provide comprehensive datasets for understanding the mechanisms underlying the observed confinement scalings and also pedestal structure. Parameterizations of the NSTX-U confinement data will be extrapolated to future STs (e.g. FNSF) to assess the implications for ST-FNSF device size and required auxiliary heating power.*

1.4.3.2 Identify instabilities responsible for anomalous ST transport

NSTX has made considerable progress in identifying instabilities underlying anomalous transports observed in NSTX. In particular, multiple instabilities have been identified as potential candidates responsible for anomalous electron thermal transport which ultimately limits the confinement performance of future devices and thus is of critical importance. Furthermore, experiment and modeling show that low-k turbulence is likely responsible for observed anomalous momentum and the non-neoclassical contribution to impurity transport whose understanding is important for calculating flow, bootstrap current and density profile etc. in future devices and thus is crucial for predicting and optimizing plasma stability and fusion gain and for achieving scenario sustainment. For example, to achieve the hot-ion scenario, low-k instabilities have to be controlled by ExB shear with optimized flow profile. From both theories and numerical modeling, NSTX has identified a range of the relevant instabilities, which include low-k modes (i.e. ion temperature gradient - ITG, trapped electron mode - TEM, kinetic ballooning mode - KBM, micro-tearing - MT), high-k modes (i.e. electron temperature gradient - ETG), and compressional and global Alfvén eigenmodes (AE), i.e. CAE/GAE. Example micro-instabilities already identified and studied for select NSTX equilibria (and to be extended to lower ν in NSTX-U) are indicated by the colored symbols in Figure 1.4.3.2.1 for different combinations of electron beta and collisionality. The approximate regions of possible mode instability and identity are indicated by the corresponding colored semi-transparent ellipses. Experimental evidence indicates that low-k and high-k modes and AE are all potentially important for electron thermal transport in different scenarios and in different radial regions. Some low-k modes, i.e. ITG/TEM/KBM, are also found to be relevant to momentum and particle/impurity transport. The identification of these candidate NSTX-U instabilities potentially responsible for anomalous transport motivates the following research thrust:

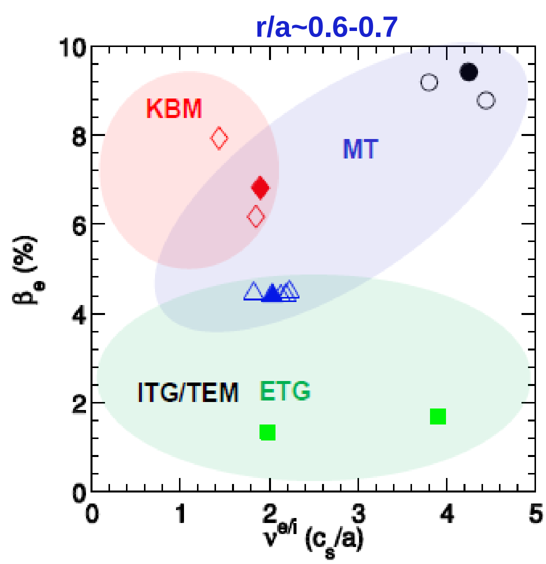


Figure 1.4.3.2.1: Ranges of accessible electron beta and electron-ion collisionality (normalized to sound-wave frequency) in NSTX/NSTX-U, and the micro-instabilities typically found to be unstable.

Thrust TT-2: Identify regime of validity for instabilities responsible for anomalous electron thermal, momentum, and particle/impurity transport in NSTX-U

NSTX researchers will identify isolated regimes for micro-instabilities using theories and reduced/first principle models, experiments for measuring turbulence and transport in these regimes and comparisons between measured transport levels and turbulence characteristics with theoretical and numerical predictions. Experimental parametric dependence will be used for further distinguishing different instabilities. For example, the dependence of micro-tearing and ETG modes on s/q and Z_{eff} are opposite to each other, and this trend can be utilized to identify each mode. The enhanced capabilities of NSTX-U, in particular the increased range of collisionality, doubled heating power from the 2nd NBI, and active flow and current profile modification using the 2nd NBI and 3D coils, will provide a versatile set of experimental control tools for modifying transport and turbulence to achieve the goals of this thrust.

1.4.3.3 Reduced transport models

It would be highly valuable to ITER, FNSF, and other next-steps to develop a physics-based 1D predictive capability of transport for use in integrated modeling that can predict plasma profiles which determine confinement scaling, MHD stability, and current profiles (from bootstrap and NBI/RF driven current) important for studying advanced operating regimes such as steady-state fully non-inductive scenarios. Even without 1D predictive capability, it is useful to have predictions for global (0D) energy confinement times that are used to estimate the required machine size, magnetic field, plasma current and density to achieve prescribed design criteria for future devices (e.g. ST-FNSF or Pilot). The highest priority transport channel to predict in STs is the electron temperature profile since the ion temperature is often expected to be described by neoclassical theory. The next highest priority transport channel to predict are density profiles for electrons, main ions and impurity ions. Perhaps the most challenging prediction is of toroidal flow profiles (momentum transport), which is ultimately critical to have a self-consistent prediction as the corresponding E×B shear strongly influences the drift wave turbulence. These considerations motivate the following research thrust:

Thrust TT-3: Establish and validate reduced transport models (0D and 1D)

NSTX-U researchers will develop and attempt to validate reduced transport models for a range of transport channels (thermal, particle, momentum) that can be used for profile predictions. For validation with experimental core measurements, boundary conditions will first be taken directly from measured profiles, such as near the pedestal top in H-mode plasma. A more complete predictive capability will also be pursued for projecting both 0D and 1D performance and will require a predicted pedestal height and location of each plasma species (and flow for flow profiles). These pedestal values will taken either from empirical pedestal scalings or validated physics-based pedestal models developed in the Boundary Physics TSG.

1.4.4 Boundary Physics Research Thrusts

1.4.4.1 H-mode pedestal structure and stability

For future fusion devices to achieve maximum fusion performance, it is vital to understand, control, and optimize the structure and dynamics of the H-mode edge transport barrier (aka the H-mode pedestal) to maximize the pedestal pressure and to avoid large damaging heat fluxes from edge-localized modes (ELMs). The expanded operating range of NSTX-U with respect to field, current, power, collisionality, and shaping will enable new H-mode transport and stability studies relevant to both spherical and conventional aspect ratio tokamaks. For example, large variations in triangularity and the ability to operate with up-down symmetric conventional or snowflake divertors will enable strong tests of leading models for the H-mode transition such as changes in the ion-orbit loss near the X-point leading to changes in the edge E×B shear and turbulence and transport. As is true for core transport and turbulence, assessing the dependence of edge transport and turbulence on collisionality will be high priority for NSTX-U. As shown in Figure 1.4.4.1.1, NSTX-U is expected to bridge the pedestal collisionality gap between NSTX and FNSF/Pilot Plant regimes. These considerations motivate the following boundary physics research thrust:

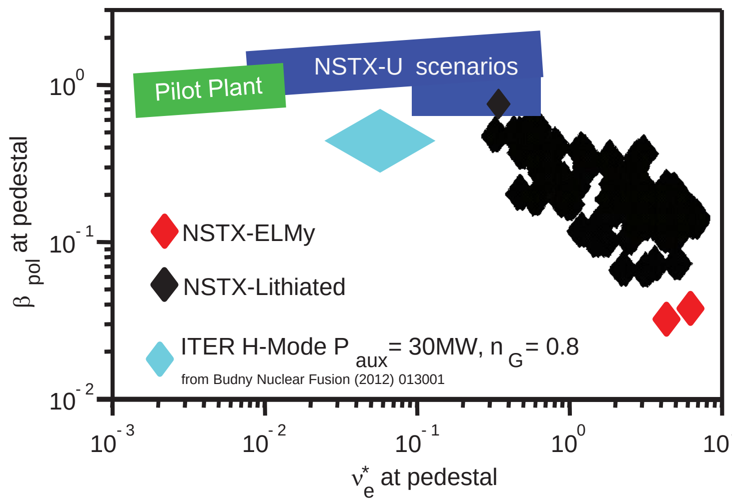


Figure 1.4.4.1.1: Pedestal poloidal beta and collisionality expected to be accessible in NSTX-U.

Thrust BP-1: Assess, optimize, and control pedestal structure, edge transport and stability

> *NSTX-U researchers will characterize the L-H mode transition thresholds utilizing the extended field, current, and power range of NSTX-U and by measuring turbulence and zonal flow dynamics with beam emission spectroscopy, Doppler backscattering, and gas-puff imaging diagnostics. The maximum achievable pedestal height and variation in pedestal structure will be assessed as a function of increased field, current, and power and also shaping and measured with enhanced spatial and temporal resolution Thomson Scattering. Increased control of pedestal transport and stability will be attempted using such techniques as edge density profile modification with improved fueling control, extended lithium coating coverage, and cryo-pumping. ELM triggering and suppression with 3D fields from mid-plane (existing) and off-midplane NCC (planned) coil-sets and*

also triggering with lithium granule injection will be assessed. Enhanced pedestal H-modes (EP H-modes) will be further explored. As high-Z PFCs are introduced, their compatibility with good H-mode pedestal performance will be assessed.

1.4.4.2 Divertor performance

Divertor research on NSTX demonstrated that the ST divertor can present both an opportunity and a challenge for the development of the divertor plasma-material interface (PMI). In future ST-based devices, the inherently compact ST divertor combined with low-collisionality SOL enable a rigorous test of the PMI concepts. In NSTX-U, long-pulse high-power-density plasmas will enable divertor research to address the gaps identified for the ST PMI, namely, high heat and particle flux control at low normalized density, while also addressing all other PMI challenges shared with conventional aspect ratio tokamaks. As shown in Figure 1.4.4.2.1, and circled in blue, NSTX-U will be able access P/R values near ITER and FDF (at ~1MW/m^2 neutron wall loading) values and approximately half the P/R value of ST-FNSF with P/S values comparable to the highest values projected to be achievable in other devices. Recent assessments of the divertor heat flux scaling in NSTX project to peak divertor heat fluxes ≥ 20MW/m^2 in the NSTX-U for conventional divertor configurations with flux expansion ~20. Very high flux

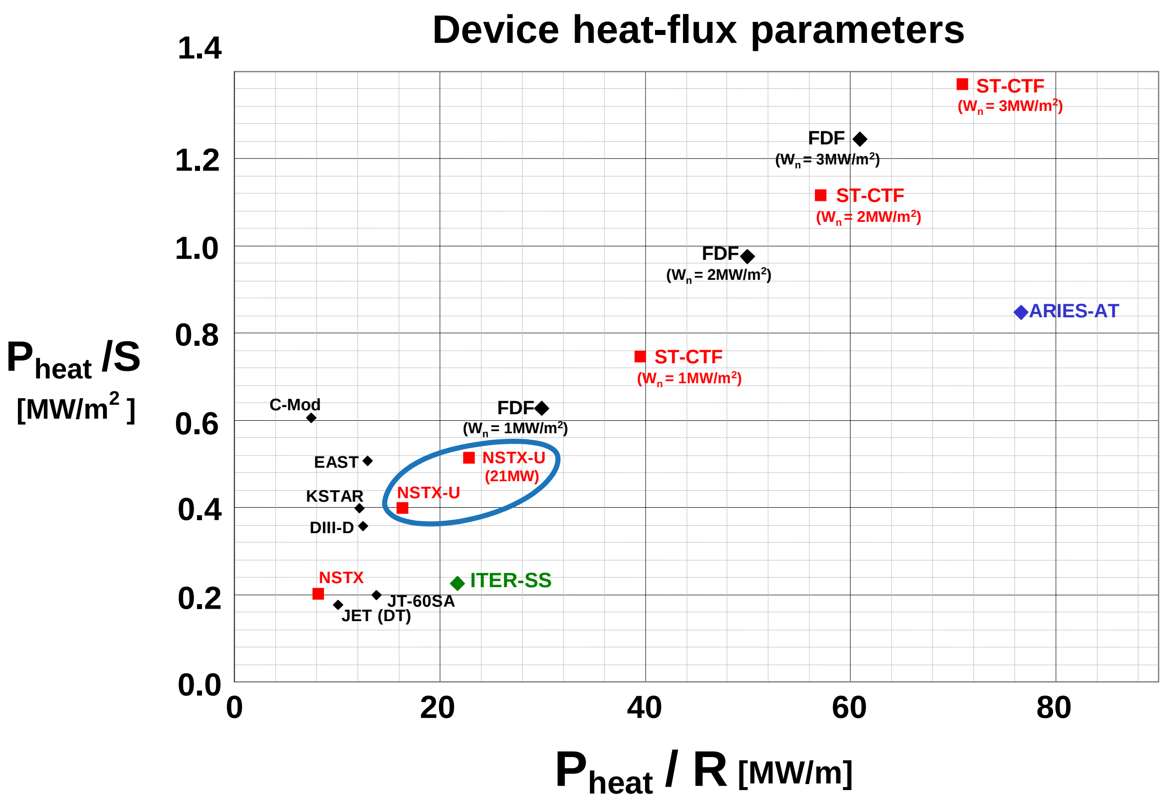


Figure 1.4.4.2.1: Heating power normalized to plasma surface area (P/S) and major radius (P/R) for present and proposed tokamaks.

expansions of ~40-60 have recently been shown to successfully reduce peak heat flux in NSTX, and additional divertor poloidal field coils have been incorporated into the Upgrade design to support high flux expansion "snowflake" divertors and strike point control for high heat flux mitigation. Enhanced divertor radiation using both intrinsic and externally applied divertor impurities have also been shown to achieve partial detachment and factor of 2-3 reduction in peak divertor heat flux in NSTX. The snowflake divertor, partial detachment, and combinations of these heat-flux mitigation techniques are high-priority research areas for NSTX-U.

A key unknown in projecting divertor performance to larger and/or higher-current next-step devices including ITER is an understanding of scrape-off-layer (SOL) transport and the associated heat and particle flux widths which determine the peak fluxes. NSTX-U can shed substantial light on these scalings by extending/increasing the range of field, current, and power for SOL-width studies by up to a factor of two. Divertor cryo-pumps are projected (see Chapter 4, Section 2.3.1) to provide access to deuterium density control for a wide range of normalized density fractions, and characterizing and optimizing the performance of divertor cryo-pumping is a critical goal for NSTX-U research. Lastly, high-Z PFC components will be introduced into the lower divertor surfaces in the base budget plan (with much more extensive coverage in the incremental budget plan), and the impact of plasma-material interactions with these surfaces on boundary physics processes will begin to be assessed. These considerations motivate the following research thrust:

Thrust BP-2: Assess and control divertor heat fluxes

> *NSTX-U researchers will study the investigate SOL heat and particle transport and turbulence and associated flux-widths extending the existing NSTX database to lower ν^*, higher I_P, and P_{SOL}. Measurements will be compared to multi-fluid turbulence and gyro-kinetic models. Novel divertor geometries such as the snowflake divertor will be systematically investigated for power and particle control and magnetic control of divertor configurations will be developed to support standard and snowflake divertor configuration studies. Steady-state and transient heat and particle transport and divertor PFC loads in these configurations configurations will be studied as a function of magnetic balance, magnetic configuration parameters, feedback-controlled impurity seeding rate over a range of SOL powers and widths. Highly-radiating boundary solutions with feedback control will be developed, and divertor detachment operating window and access parameters will be studied. As high-Z PFCs are introduced, their compatibility with the divertor heat exhaust solutions will be assessed.*

1.4.4.3 Particle Exhaust Sustainability

Integrated scenarios on NSTX-U are designed for steady density at 0.5-1 times the Greenwald density limit. To achieve this, a central element of the boundary physics program is the installation of a divertor cryo-pump - a proven technology to control both main ion and impurity density. A key component of the research is to compare lithium pumping and cryo-pumping for density control, while considering the possible synergy of the two methods to contribute to power and particle exhaust solution for NSTX-U and future devices. An integral element to both of these studies is the assessment of impurity sources and transport. These considerations motivate the following research thrust:

Thrust BP-3: Compare the sustainability of particle exhaust via lithium pumping and cryo-pumping, for density, impurity, and Z_{eff} control consistent with integrated scenarios

> *Experiments will be conducted to validate cryo-pump physics design activities, perform initial density control studies, and assess compatibility with H-mode pedestal and core performance and divertor power exhaust scenarios. Comparisons will be made between lithium conditioning and cryo-pumping for density and impurity control, as well as changes to the edge density profile, which are central to ELM elimination with lithium conditioning. Local recycling coefficients in the upper and lower divertor regions, as well as the wall will be measured. Impurity sources and SOL transport, and the role of ELMs in impurity control will be assessed. As high-Z PFCs are introduced to NSTX-U, their impact on core impurity transport, boron and lithium coatings, and divertor retention and particle control will be assessed.*

1.4.5 Materials and PFC Research Thrusts

1.4.5.1 Lithium surface-science

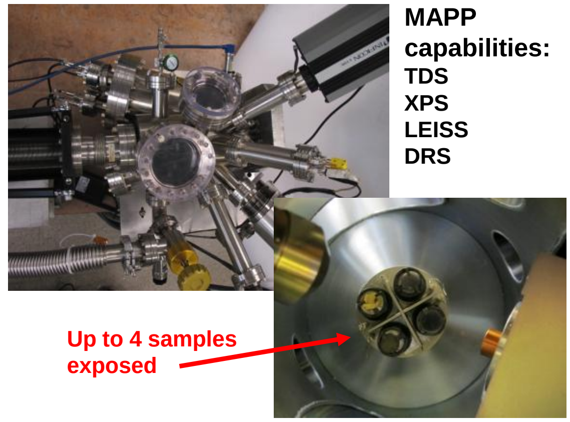


Figure 1.4.5.1.1: MAPP diagnostic capabilities

NSTX has a long history of performance improvement with the usage of lithium wall conditioning. Results from 2008 and 2010 indicate a potential paradox in which confinement improves with small amounts of additional lithium (~1g), but did not improve during the liquid lithium divertor (LLD) campaign in which much larger amounts (~1kg) were used. Further, there are indications that oxygen gettered from vacuum and possibly "scavenged" from the bulk carbon is interacting with the Li and H/D in complex ways. The interactions between lithium, the impurities present in tokamak plasmas, and high-Z PFCs are also relatively unexplored. A key diagnostic capability to be utilized for this research is the Materials Analysis Particle Probe (MAPP) shown in Figure 1.4.5.1.1. New experiments and the utilization of advanced diagnostics are needed to address the aforementioned issues in a systematic way and motivate the following thrust:

Thrust MP-1: Understand lithium surface-science during extended PFC operation

> *Experiments will be conducted to improve understanding of the role of more complete coverage of the PFCs by evaporated lithium using upward-facing evaporators and/or diffusive evaporation as well as the impact of boronization. The Materials Analysis Particle Probe (MAPP) will be utilized to identify in-situ between-shot chemical states and compositions of the coatings. A boronization campaign in year one will be followed by a controlled Li introduction. The identical total deposited lithium as supplied in 2008 will be applied on both upper and lower divertors. The changes in plasma performance will be compared to the 2008 database. In the second year, a boronization campaign will be eliminated and an identical introduction of lithium as in the previous year will be applied to bare graphite surfaces. These studies will provide a comparison between C+B substrate vs. a C substrate alone in terms of plasma performance. The MAPP diagnostic will be utilized during this year also, and will enable comparisons to surface-science laboratory results including Li-coatings on high-Z metal substrates such as TZM. The uptake of hydrogenic species and the impact of varying levels of impurities (e.g. C, O) on that uptake will be examined in the Li-metal system, and in years 3-5 the surface interactions of Li on high-Z tiles in at least one NSTX-U divertor region will be studied.*

1.4.5.2 Material migration and evolution

A key PFC challenge facing long-pulse, high-power devices is the net-reshaping of the PFCs as a result of erosion, migration and redeposition. Charge-exchange to the first-wall scales with the plasma exhaust power normalized the surface area (P/S ratio) of the machine. The value of this parameter is expected to be quite high in NSTX-U with values up to P/S = 0.5MW/m^2 which is approximately half the value of an FNSF or DEMO, albeit with far shorter pulse-length (~5-10s in NSTX-U versus ~10^3-10^7s in FNSF/DEMO) and much lower duty factor (~10^4 shot seconds per year in NSTX-U versus ~10^6 to 2-3×10^7 s/year in FNSF/DEMO). The power exhausted to the divertor also increases particle fluxes in that region, and this is also expected to be a large number in the highest-power discharges projected for NSTX-U. These attributes, combined with longer pulse-lengths expect to be achievable in NSTX-U, can be exploited in experiments to measure the net erosion and re-deposition inside NSTX-U in order to begin to provide a basis for projecting to FNSF. These considerations motivate the following thrust:

Thrust MP-2: Unravel the physics of tokamak-induced material migration and evolution

A combination of quartz-crystal microbalances (QCMs) and marker-tiles will be utilized to provide measurements of shot-to-shot erosion and redeposition in NSTX-U. An upgraded MAPP system with QCM will enable shot-to-shot analysis of not just the quantity of deposited material, but also the composition of that material. The scaling of wall-erosion for typical NSTX-U divertor conditions over a range of P/S values will be used to construct a data-base to estimate/project total wall erosion expected for an FNSF or DEMO. The material transport variations as a function of divertor configuration (e.g. snowflake, detached, vapor-shielded) will also be measured and compared to simulation.

1.4.5.3 Continuous vapor-shielding

Lithium is known to evaporate at significant rates at modest temperatures (T ~ 450C). In experiments on plasma-gun and electron-beam devices (e.g. QSPA or SPRUT), transient vapor-shields have been produced and little damage to the substrate material was observed. Usage of a vapor-layer to protect a solid substrate is analogous to detached divertor operation but utilizes the PFC material itself as the vapor-layer. Such operation has not been demonstrated in steady-state or in the presence of a high-performance tokamak plasma. However, the operation of Li PFCs in the

$$p_u = p_t(1 + M_t^2) + p_{Li}$$
$$q_t = \gamma \Gamma^+_{sat} T_e = \gamma n_{es} c_s T_e = \gamma c_s p_t$$

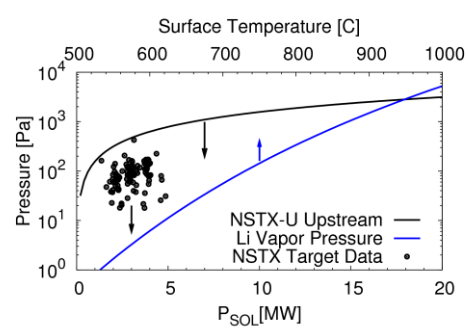


Figure 1.4.5.3.1: (Top) SOL pressure balance and heat flux equations, (bottom) NSTX data and NSTX-U projections indicating possibility of Li vapor pressure providing SOL operation analogous to detachment with application to protecting PFC substrates.

experiments above and results from NSTX in which divertor radiation was observed to increase and divertor peak heat flux decrease with increased lithium indicate vapor shielding may be favorable for reducing divertor heat fluxes. Figure 1.4.5.3.1 shows that NSTX divertor target pressure values (black circles) and projected NSTX-U up-stream SOL plasma pressures (black curve) can be comparable to the projected Li vapor pressure (blue curve) for sufficiently high PFC surface temperature (top abscissa). These projections motivate the following thrust:

Thrust MP-3: Establish the science of continuous vapor-shielding

> *NSTX-U researchers will begin to establish the scientific basis and experimental demonstration of a continuously vapor-shielded surface in the tokamak environment with application to heat-flux mitigation. Extensive research on the long-pulse linear plasma device Magnum-PSI will be carried out to provide an extrapolable experimental demonstration of such vapor-shielding. These experiments will then be extended to NSTX-U using a lithium-coated high-Z substrate (for transient operation) or a flowing liquid lithium system (for long-pulse operation) with a high-power-density strike-point impinging on the PFC to raise the front-face temperature above the lithium evaporation temperature. MAPP, QCMs, Langmuir probes, and divertor spectroscopy, bolometry, and two-color IR diagnostics will be utilized to provide data for interpreting the results.*

1.4.6 Energetic Particle Research Thrusts

1.4.6.1 Develop predictive tools for fast-ion transport

Heating and some non-inductive current drive methods for next step magnetic fusion devices utilize non-thermalized (e.g. fast) ion populations whether as rf-tail ions, energetic neutral beam ions or fusion alphas. Instabilities driven by these fast ions, in particular the Toroidal Alfvén Eigenmodes (TAE), can modify the expected heating or current drive profiles. Projecting which operational regimes will minimize these instabilities, or if necessary, predicting the effect of these modes on the fast ion population, is valuable for designing new devices and planning their operation. These predictions are made through the extrapolation of results obtained in present research devices to larger scale, fusion-grade reactors. The development and benchmarking of numerical codes based on experimental observations from existing experiments is therefore one of the most critical issues in fusion research.

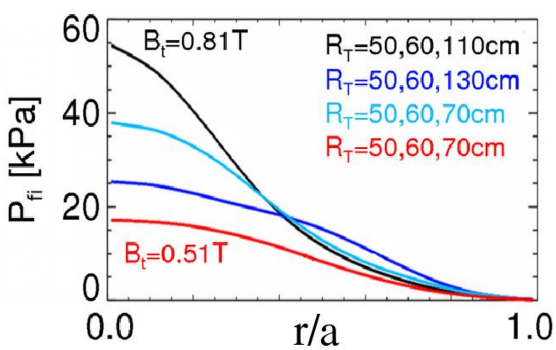


Figure 1.4.6.1.1: Fast-ion pressure as a function of toroidal field strength and NBI tangency radii for various combinations of NBI sources in NSTX-U.

A primary goal for Energetic Particle research on NSTX-U is to develop capabilities that enable reliable and quantitative predictions on properties of, and fast ion response to, unstable modes in future devices such as ITER and FNSF. Variation of $v_{fast}/v_{Alfvén}$ by varying density and field, variation of the fast-ion density and pressure profiles, and variation of the rotation profiles are key NSTX-U capabilities to elucidate fast ion physics. For example, Figure 1.4.6.1.1 shows the wide range of fast-ion pressure peaking (assuming classical slowing down) projected to be accessible using combinations of the original NBI and 2nd more tangential NBI of NSTX-U. Based on results from NSTX and other devices, modes that can induce substantial redistribution and loss of fast ions will be targeted first. These include TAE, RSAE and Energetic-Particle modes (EPMs), as well as higher frequency *AEs (GAE/CAEs) that are responsible for deviations of the neutral beam fast-ion distribution function (F_{nb}) evolution from classical behavior. These considerations motivate the following research thrust:

Thrust EP-1: Develop predictive tools for projections of *AE-induced fast ion transport in FNSF and ITER

> *NSTX-U researchers will: (i) study the modes' properties (radial structure, frequency and wavenumber spectrum, stability) and (ii) characterize fast ion transport associated with specific classes of *AEs. In parallel with the experimental research on NSTX-U, the development of numerical and theoretical tools will be pursued to lay the basis for predictive capability. Experimental results will be utilized for extensive code verification and validation, based on the detailed characterization of instabilities and fast-ion distribution available on NSTX-U from the upgraded set of fast ion diagnostics.*

1.4.6.2 Fast-ion Phase-Space Engineering (PSE)

Phase-Space Engineering (PSE) consists of controlled, externally-induced modifications of the fast-ion distribution in order to achieve specific goals such as performance improvement or enhanced stability. Possible applications of PSE include *AE control, stochastic ion heating, alpha-channeling, and stabilization/regulation of sawteeth and other MHD modes (e.g. NTMs). The longer-term PSE objective that requires dedicated research on NSTX-U is the development of schemes for direct *AE control. Experiments will aim at developing tools to control AE activity in NSTX-U and exploit instabilities as an additional actuator to modify, in a controlled way, the evolution of fast ion radial profile and energy spectrum. A parallel line of research is the regulation of electron thermal transport through high-frequency AEs. If successful, PSE will help to control the TAE and other Alfvénic instabilities in a reactor by bypassing the thresholds imposed by thermonuclear instabilities, which is an important (and often under-estimated) issue for devices such as ITER, FNSF and future Pilot plants.

PSE research requires a detailed knowledge of the stability properties of AEs. TAE linear damping rate has been measured on the C-Mod and JET tokamak experiments with the goal of benchmarking codes used to project TAE stability to ITER relevant regimes. Linear damping rates are inferred by exciting stable modes with an antenna and by computing the resonance width from the complex impedance modeling the driven plasma response. Next step Spherical Tokamaks will operate in a different fast ion parameter regime, thus it is necessary to extend the previous benchmarking activity to encompass a broader range of scenarios and different classes of *AE instabilities. For example, the much stronger magnetic shear at the plasma edge (found to be stabilizing in triangularity scans on JET) and the relatively stronger rotational shear, resulting in enhanced continuum interactions, are predicted to affect TAE stability in STs. An example of changes of the TAE radial gap width caused by rotation in NSTX is shown in Figure 1.4.6.2.1

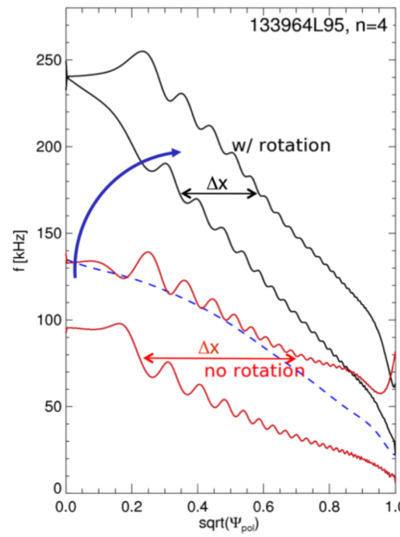


Figure 1.4.6.2.1: NSTX TAE gap modification due to rotation.

Looking beyond the lower frequency TAE modes, higher frequency Alfvénic instabilities are routinely observed in START, MAST, and NSTX NB-heated plasmas. These instabilities clearly have the potential to affect the fast ion distribution, although experimental evidence of the extent of the perturbations is weak at present. Reliance on fast ions to destabilize the natural eigenmodes of Compressional and Global Alfvén waves (CAE and GAE) reveals only a limited subset of the eigenmode spectrum. These considerations motivate the following research thrust:

Thrust EP-2: Assess requirements for fast-ion phase-space engineering techniques

> *Active spectroscopy experiments on NSTX-U, similar to those carried out on JET and C-Mod, will be performed to provide a similar data set of linear TAE damping rates for further benchmarking stability codes at low aspect ratio. This will also help to validate ITER projections by challenging the fundamental understanding of the physics underlying the drive and stability mechanisms of these modes. With incremental funding, the bandwidth of external antennae will be extended to study high frequency modes. The higher frequency of these modes, together with the possibility of cyclotron resonances, may provide an attractive route towards controlled modifications of the non-thermal ion distribution function and possibly lowering the overall fast ion pressure.*

The outcome from PSE research on NSTX-U will be evaluated, possibly in conjunction with results from other devices such as DIII-D, MAST and JET, to assess the feasibility and potential of PSE schemes on future devices in terms of required power and spectrum, temporal response, time-scales for diagnostics and control. Extrapolations to future devices, such as a ST-based FNSF/Pilot and ITER, will provide information on achievable performance improvement, for instance in the capability of reliable access to the so-called "hot-ion mode" and efficiency of alpha-channeling schemes.

1.4.7 Wave Heating and Current Drive Research Thrusts

1.4.7.1 Fast-wave heating and current-drive for non-inductive start-up and ramp-up

A major goal of the NSTX-U FY2014-18 research program is the development of fully non-inductive discharges. This goal is important for the design and development of an ST-FNSF device and an advanced technology (AT) fusion reactor. It is challenging both scientifically and operationally. The approach on NSTX-U will therefore be to initially develop non-inductive start-up, ramp-up and plasma sustainment scenarios separately, and then later combine these non-inductive scenarios. On NSTX, as shown in Figure 1.4.7.1.1, HHFW heating was successfully used to generate a $T_e(0)$ =2-3keV H-mode plasma with 70% non-inductive current fraction with a total plasma current of ~300 kA. This scenario used an ohmically-initiated L-mode discharge as the target plasma for the HHFW heating and 1.4 MW of HHFW power. These considerations motivate the following research thrust:

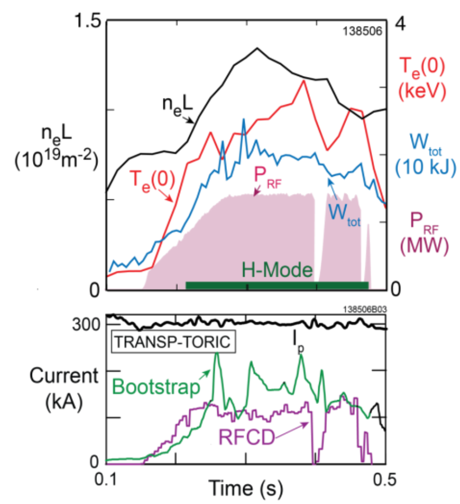


Figure 1.4.7.1.1: HHFW heating and current drive in a low-I_P target plasma.

Thrust RF-1: Develop RF heating and current drive for fully non-inductive plasma current start-up and ramp-up

> *Experiments in NSTX-U will extend NSTX HHFW H-mode studies by using higher HHFW powers to demonstrate HHFW-driven fully non-inductive plasma current ramp-up to plasma currents significantly higher than 300 kA. Further, high-power HHFW and EC heating (assuming a ~1MW 28GHz gyrotron) will be used to significantly increase the core electron temperature of CHI and plasma-gun-initiated discharges in order to increase the plasma current and extend the non-inductive discharge duration. EBW heating will also be tested as a wave-based means of generating plasma start-up current in NSTX-U using a technique similar to that developed on MAST in which a grooved mirror-polarizer incorporated in a graphite tile on the center stack provides mode conversion from O-mode to X-mode for efficient EBW absorption.*

1.4.7.2 Validation and Application of Advanced RF Codes

Advanced RF numerical codes, such as the AORSA-3D full-wave solver with SOL modeling and a realistic antenna model, and the full finite-orbit-width CQL3D Fokker-Planck neoclassical simulation code have previously shown promising capabilities in modeling edge power loss mechanisms and fast-wave interactions with energetic ions in NSTX plasmas. With access to increased toroidal field and reduced cyclotron harmonic number, NSTX-U plasmas will be able to operate significantly closer to the RF parameters of next-step ST plasmas and for ITER. Thus, with sufficient validation, AORSA-3D and CQL3D would be valuable tools for predicting the behavior of the wave fields in the SOL, the plasma edge and the interaction with fast-ions in the bulk plasma in future fusion machines such as ITER and FNSF.

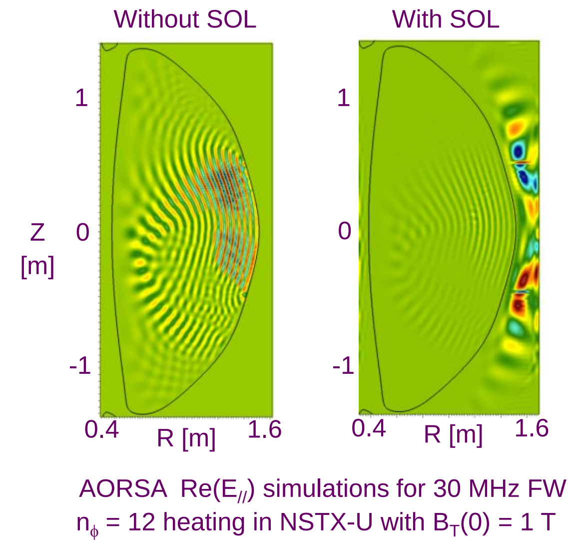


Figure 1.4.7.2.1: AORSA simulations of NSTX-U plasma heating with and without inclusion of SOL density profile

For example, a potentially critical issue for ITER is fast-wave power coupling through the SOL into the core plasma. As shown in Figure 1.4.7.2.1, inclusion of the experimentally-relevant exponentially-decaying SOL density profile projected for NSTX-U can strongly influence fast-wave propagation and surface wave excitation in the plasma edge region. Thus, NSTX-U can play an important role in validating leading models that can be used for modeling ITER ICRF coupling and heating. Considerations such as these motivate the following research thrust:

Thrust RF-2: Validate advanced RF codes for NSTX-U to aid prediction of RF performance in FNSF-ST and ITER

> *Building on measurement and simulation results from NSTX, detailed measurements of the SOL density and temperature profiles, edge fluctuations, RF power flows to the divertor regions, the RF power deposition profile, the RF-driven current profile, and the core fast-ion distribution function will be obtained. These measurements and associated analysis will be systematically compared to modeling results from AORSA-3D and CQL3D. After these advanced RF simulation models have been validated and verified during the first three years of the NSTX-U 5 year plan, they will subsequently be used to predict RF performance in ST-FNSF and for ITER.*

1.4.8 Plasma Current Start-up and Ramp-up Research Thrusts

As described in Section 1.1, a major challenge for an ST-based FNSF is the necessity of plasma formation with either small or no central solenoid flux combined with the need for non-inductive current ramp-up using a combination of bootstrap current and NBI and/or RF current overdrive. The goal for NSTX-U non-inductive plasma current formation and ramp-up research for the next 5 years is to first demonstrate current formation and ramp-up independently and then to integrate and optimize start-up and ramp-up to the ~1MA level.

1.4.8.1 Re-establish solenoid-free current start-up, test non-inductive ramp-up

Coaxial helicity injection (CHI) – first demonstrated on the HIT-II ST at the University of Washington and subsequently demonstrated on NSTX – is a promising candidate for non-inductive current initiation. In addition, CHI has the potential to drive edge current during the sustained phase of a discharge for the purpose of controlling the edge current profile to improve plasma stability limits and to optimize the bootstrap current fraction. Experiments on the HIT-II experiment at the University of Washington demonstrated that the method of *transient* CHI (TCHI) could generate high-quality plasma equilibrium in a ST that could be coupled to inductive drive. Since then the transient-CHI method has been successfully applied to NSTX for solenoid-free plasma start-up followed by inductive ramp-up. These coupled discharges have now achieved toroidal currents of 1 MA using significantly less inductive flux than standard inductive discharges in NSTX. These results must be re-established on NSTX-U using new divertor poloidal field coils and other configurational changes, and then extended to higher current (from 200kA to 400kA) to provide plasma current levels compatible with confining NBI fast-ions for subsequent current ramp-up. For current ramp-up, TRANSP simulations indicate that the new 2nd NBI of NSTX-U is predicted to be much better confined at low I_P due to the less perpendicular injection and associated reduction in prompt/bad-orbit loss (see Chapter 8). Thus, substantial experimental tests need to be carried out to characterize and optimize non-inductive current drive in lower-current target plasmas using the 2nd NBI. These considerations motivate the following research thrust to be carried out during first few years of the five year plan period:

Thrust PSR-1: Establish and extend solenoid-free plasma start-up and test NBI ramp-up

> *NSTX-U researchers will re-establish transient CHI discharges utilizing graphite lower divertor tiles, the increased toroidal field capability of NSTX-U, and full Li coating of the lower divertor tiles followed by subsequent lithium conditioning of the upper divertor. The maximum toroidal currents that can be generated with CHI will be assessed by varying and increasing the amount of injector flux, the size of the capacitor bank, and the CHI voltage (up to 2 kV). The upper divertor buffer coils will be used to suppress absorber arcs, and studies of the coupling of the CHI generated plasma to inductive drive will be performed. Researchers will also generate 300-400kA flat-top current inductive*

plasmas and inject the new more tangential beams to assess NBI coupling and current drive efficiency and compare to TSC/TRANSP simulations. NBI coupling to CHI targets will also be assessed and compared to simulation. Combinations of NBI and HHFW heating and current drive will be utilized to heat inductive plasmas and attempt to non-inductively ramp-up the plasma current to the ~0.8-1MA range.

1.4.8.2 Couple CHI plasmas to non-inductive current ramp-up

Previous CHI experiments in HIT-II and NSTX have shown that minimizing impurity radiation is critical to maximizing transient CHI plasma temperature and current decay times and that metallic divertor surfaces can reduce low-Z impurity levels. One and possibly both NSTX-U divertor surfaces are planned to be converted to high-Z PFCs (likely TZM or a W alloy) in order to investigate the performance impact of such PFCs on integrated high-performance plasmas, and such a conversion could also benefit CHI operation. ECH power absorption simulations combined with TSC time-dependent modeling indicate that ~0.5MW of 28GHz ECH heating could significantly increase CHI plasma electron temperatures (from ~10-50eV to 100-500eV) and greatly lengthen the CHI plasma current decay time from tens to a few 100ms. This increase in plasma current duration should be sufficiently large for NBI fast-ions to be confined and slow-down to heat the CHI plasma (see Chapter 8). Increased CHI injector voltage to 3kV (if technically feasible) combined with the higher toroidal field of ~1T of NSTX-U is projected (based on TSC simulations) to generate CHI closed-flux currents of 0.4-0.6MA which would further improve coupling to NBI heating and current ramp-up. As shown in Figure 1.4.8.2.1, TRANSP/TSC simulations indicate that early HHFW heating followed by NBI heating and current drive combined with bootstrap current can in principle increase the plasma current non-inductively from the 0.4-0.5MA level to the 0.8-1MA range. However, these simulations utilized scaled profiles from higher current NSTX experimental plasmas and also ignored possible fast-ion redistribution and/or loss from Alfvénic or other MHD instabilities. Validation of both 2D and 3D simulations of CHI start-up is necessary to provide a reliable basis for extrapolation to an ST-based FNSF. Lastly, the potential for point helicity injection using plasma guns as being developed on the Pegasus toroidal facility (see Chapter 8) makes this technique particularly attractive for an ST-FNSF, and important data on the size and field scaling of gun start-up is needed from NSTX-U to reliably extrapolate to an ST-based FNSF. These considerations motivate the following research thrust:

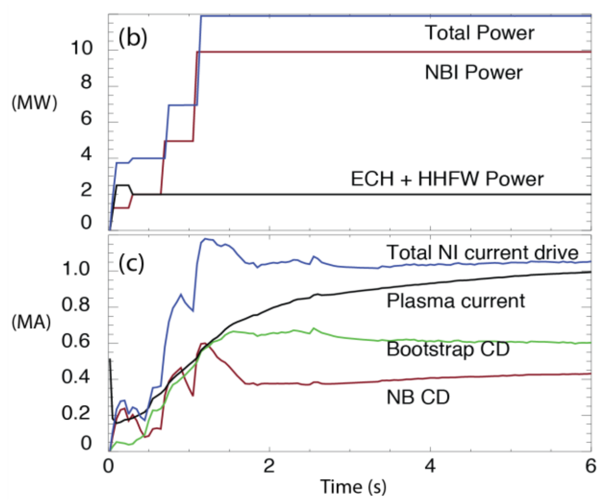


Figure 1.4.8.2.1: TSC simulations of non-inductive ramp-up of an NSTX-U plasma to 0.8-1MA.

Thrust PSR-2: Ramp-up CHI Plasma discharges using NBI and HHFW and Test Plasma Gun Start-up

NSTX-U researchers will maximize the levels of CHI-produced plasma currents using new operational capabilities including 1) metallic divertor plates, 2) 1 MW 28GHz ECH, and 3) 2.5-3 kV CHI capability. All these should allow more injector flux to be injected into the vessel at reduced levels of low-Z impurities. Initial tests of the effectiveness of NBI coupling to a CHI-generated target will be carried out using the best available CHI targets including the expected increase in CHI plasma duration achieved with ECH electron heating, and the NBI coupling and ramp-up of CHI current will be systematically investigated. Detailed comparisons of CHI current drive results to 2D TSC and 3D NIMROD simulations will be carried out to develop a TSC/NIMROD model of CHI for FNSF design studies. If technically ready, plasma gun hardware will be commissioned on NSTX-U and point-source helicity injection (plasma gun) plasma formation will be initially tested and compared to results from Pegasus.

1.4.9 Plasma Sustainment: Scenarios and Control Research Thrusts

1.4.9.1 Scenario development

A critical element of NSTX-U operation will involve developing high performance and/or long-pulse scenarios in support of the NSTX-U scientific research program and for ST development generally. The demonstration of sustainment of high non-inductive fraction (up to 100%) is essential for FNSF applications of the ST. Example projections for 100% non-inductive operation in NSTX-U as a function of normalized density and H-mode confinement multiplier are shown in Figure 1.4.9.1.1. In both figures, the diamond symbols indicate individual TRANSP runs used to generate the contour plots, the red line in the upper plot indicates non-inductive fraction = 1, and the red line in the lower plot indicates q_{min} = 1. Beyond high non-inductive fraction, access to high-current partial-inductive scenarios is necessary to explore the current and collisionality scaling dependence of confinement and stability. These operational goals motivate the following research thrust:

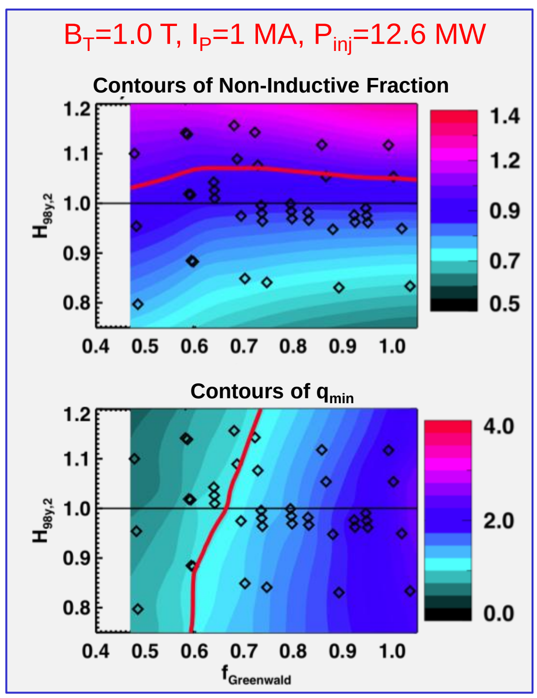


Figure 1.4.9.1.1: TRANSP simulations of non-inductive current fraction and q_{min} vs. normalized confinement and density for NSTX-U

Thrust ASC-1: Development of high non-inductive fraction and high current scenarios

Scenarios with non-inductive current fraction near 100% will be systematically explored. Key questions to be examined include: the impact of the confinement level and profile shapes on the non-inductive current level, the global stability of scenarios with large neutral beam current drive and central fast ion pressures, the optimal density for non-inductive sustainment, and consistency of the non-inductive operating state with divertor integration. Further, NSTX-U researchers will develop high-current, partial inductive operation aiming for the facility maximum performance goal of 5 second pulses at I_P=2.0 MA and B_T=1.0T. These scenarios are the key means of accessing low collisionality in NSTX-U, and the development of lower-density operations is thus critical. It is likely that these high current scenarios will result in the highest divertor peak heat fluxes, so the development of integrated heat flux management solutions is a requirement.

1.4.9.2 Axisymmetric control development

The development of advanced control strategies for achieving and maintaining optimal ST performance is important for NSTX-U operation and for advancing the ST for FNSF applications. Boundary and vertical position control is potentially more challenging for the ST since inboard coils for maintaining the inner gap may not be available and very high elongations are desired to increase the stable plasma beta and bootstrap fraction. Divertor heat flux control is also important – especially for high-current NSTX-U scenarios. NSTX results and collaborative experiments on DIII-D have demonstrated stable single snowflake divertor operation with significant heat flux reduction (see Chapter 4), but control advances will be required in order to control up to four x-points in up/down symmetric snowflake configurations (see Chapter 10). An example snowflake configuration for NSTX-U exhibiting 2 nearby X-points in the lower divertor is shown in Figure 1.4.9.2.1. Developing both core plasma safety factor and rotation control is important for optimizing plasma stability and confinement, and density control will also be a very important tool for varying and optimizing plasma collisionality and NBI current drive. These considerations motivate the following research thrust:

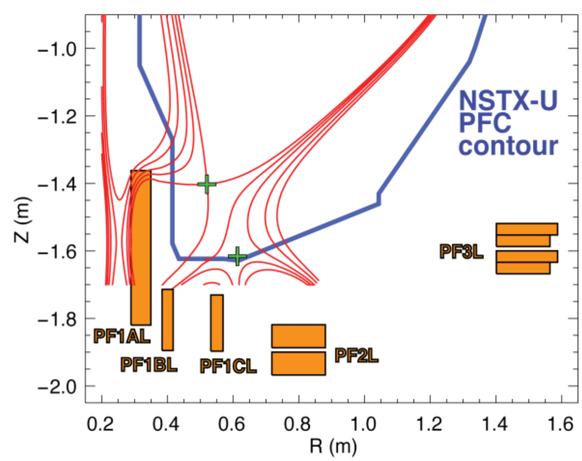


Figure 1.4.9.2.1: Simulated "Snowflake" divertor in NSTX-U showing presence of 2 nearby x-points (green plus signs) in the lower divertor.

Thrust ASC-2: Development of axisymmetric control

> *NSTX-U researchers will optimize multi-input multi-output boundary shape controllers and improved vertical stability algorithms. This research will also support improved control of the divertor heat flux using the snowflake divertor which uses two or three divertor coils to pull nearly overlapping X-points and has been shown to lead to a significant reduction in the divertor heat flux in NSTX and DIII-D. In particular, NSTX-U researchers will work to develop real-time tracking and closed-loop control of multiple X-points and also divertor radiation using feedback control of impurity gas injection building on earlier success with open-loop detatched divertor experiments in NSTX. Experiments will be conducted to verify TRANSP calculations that show by varying the neutral beam source mix and/or plasma density, the minimum safety factor (qmin) can be controlled. These results will be used to develop simultaneous β_N and q_{min} controllers in NSTX-U. Similarly, the variation of neutral beam torques from the different sources and n=2 and 3 magnetic braking from the RWM/EFC coils will be used to control β_N and the values of toroidal rotation at selected points across the profile. Experiments will also be attempted to examine the feasibility of combined beta, rotation, and q control. Finally, real-time density measurements will be brought to PCS, and improved fuelling actuators will be developed in support of density feedback control.*

1.4.9.3 Controlled plasma shut-down

While NSTX-U is designed to withstand the electromagnetic and thermal loads from high current and stored energy disruptions, it is advantageous to minimize the frequency of disruptions associated with the end-of-pulse to minimize any possible risk of machine damage, maximize shot-to-shot reproducibility, and to prototype safe/controlled plasma shut-down techniques for FNSF and for ITER. Example potential complicating factors include rapidly time-varying beta and/or density control needed to avoid operational limits during the ramp-down phase, loss of H-mode due to decreased heating power and/or increased core radiation, and increased internal inductance due to loss of H-mode and/or negative surface voltage possibly leading to vertical instability. These considerations motivate the following research thrust:

Thrust ASC-3: Disruption Avoidance By Controlled Discharge Shutdown

> *NSTX-U researchers will optimize disruption detection with sufficient time to make a useful intervention during the discharge progression. Real-time inspection of quantities such as coil heating and the solenoid flux evolution will be used to determine when slow ramp-downs are required to avoid exceeding operational limits which might otherwise trigger fault conditions. Multiple real-time diagnostic signals will also be synthesized to form efficient disruption detectors, requiring more rapid ramp-downs. This information*

will be used trigger automated rapid ramp-down sequences. It is envisioned that multiple types of ramp-down sequences will be developed, pending the different sources of alarms. A massive gas injection (MGI) sequence will also be included, to take advantage of the MGI system being developed in the MS TSG Thrust MS-3.

1.4.9.4 Scenario optimization for next step devices

A major goal for NSTX-U advanced scenario research is to investigate the achievable integrated performance of NSTX-U scenarios and then assess the potential impact of the observed performance capabilities on next-step ST device design. For example, the plasma shaping and profiles that optimize fully non-inductive operation in NSTX-U will be investigated, and the required shaping and actuator capabilities for a next-step ST will be addressed. These considerations motivate the following research thrust:

Thrust ASC-4: Scenario optimization for next step devices

NSTX-U researchers will study aspects of scenario optimization physics relevant to next-step devices, in ways that may not produce optimized scenarios for NSTX-U. For instance, the simultaneous current and rotation profiles providing optimal performance will be examined. The conditions for classical beam current drive will be explored. Finally, integrated modeling of the thermal energy, toroidal rotation, and current will be pursued, first for validation against NSTX-U results, and then for projection to next-step ST scenarios.

It is noted that while the planned research described above is divided into four separate thrusts for the purpose of organization, there are clearly several interconnections. Examples include:

- The development of high-current long pulse scenarios is captured in thrust #1. However, progress in heat flux mitigation described in Thrust ASC-2 is clearly required to realize these scenarios. This provides a natural linkage between Thrusts ASC-1 and ASC-2.

- The utility of variations in the neutral beam current drive to modify the minimum/central safety factor is a common theme through Thrusts ASC-1, 2, and 4.

- A common analysis tool through the Thrusts ASC-1, 2, and 4 of this research program is the TRANSP code. This tool will, on the one hand, continue to be used to develop experimental plans. On the other hand, experiments will be designed to validate the predictions of the code, revealing areas where its physics treatment is insufficient.

1.4.10 Research Thrust Support of High-Priority 5 Year Plan Goals

The research thrusts described in Sections 1.4.2 through 1.4.9 are supportive of the 5 high priority research goals listed in Section 1.2.2, and the mapping between the high-priority goals and TSG/chapter thrusts is shown in Figure 1.4.10.1. As shown in the figure, all of the thrusts contribute to the achievement of the 5 high-level goals, but the macro-stability (MS), transport and turbulence (TT), boundary physics (BP), and advanced scenarios and control (ASC) thrusts contribute most broadly to the high-priority research goals of the NSTX-U 5 year plan.

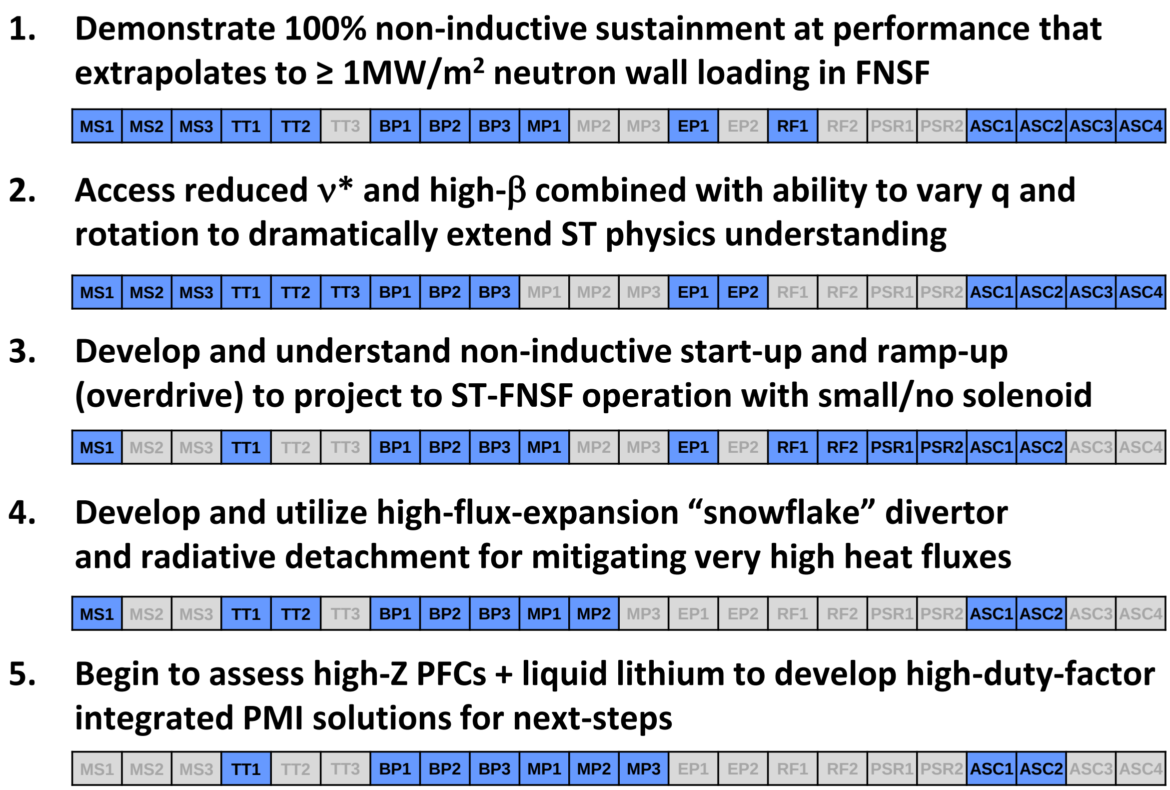


Figure 1.4.10.1: Mapping between the 5 year plan high-priority research goals and the research thrusts.

1.5 Planned NSTX-U Research in Support of ITER

NSTX-U research can support ITER in several different ways. The unique parameter regimes accessible by NSTX/NSTX-U will provide new insight into underlying tokamak physics through direct experiments and associated analysis as well as through theory and code validation. Specifically NSTX-U will be able to explore fundamental toroidal physics issues, use its high toroidicity, shaping and expanded operating space as leverage for theory validation, and develop operational/control hardware, approaches and capabilities. This will help provide the basis for addressing key physics and technology issues for tokamaks at all aspect ratio, including future burning plasma experiments such as ITER.

NSTX-U will contribute to specific High Priority ITER Research and Development needs in all topical science areas. In most cases, these contributions will address longer-term physics and operational scenario development, although in some cases there will be the opportunity for direct contributions to near-term design issues. In the following, we will focus on and detail the more urgent ITER R&D issues where NSTX-U can make the greatest impact. A discussion of additional work and ITPA involvement will follow. These discussions will focus on the high-level aspects of the work; more details are given in the individual chapters of each NSTX-U Topical Science Group.

In the MHD area, NSTX-U is engaged in studies of both Error Field (EF) and Resistive Wall Mode (RWM) physics. These are two critical pieces of the physics that will aid in the development of an integrated framework to provide stable plasma operation, including a physics-based disruption “prediction-avoidance-mitigation” system. Experiments and associated model development are presently focused on developing a self-consistent treatment of both the non-ideal plasma response to the EF coupled to a δf-treatment to determine the associated Neoclassical Toroidal Viscosity through the use of the IPEC and POCA codes. This will provide a general approach that is valid at all aspect ratio for predicting the EF threshold for rotation locking. This work up to now has already revealed the importance of non-resonant components of the EF. It has also identified rotation as a key piece of physics in determining the EF threshold for locked mode growth. IPEC modeling has been used to develop a scaling for this EF threshold which depends more strongly than linear on the plasma rotation, giving better agreement with results from NSTX experiments. Direct contributions to ITER based on this work have already been made; this scaling has been applied to ITER operational scenarios to determine the EF threshold for locked mode growth. In particular, levels of various ITER coil currents have been determined that provide low enough EF for avoiding the growth of locked modes.

The importance of multiple parameters in controlling plasma stability has also been identified in the study of Resistive Wall Modes. As one example, it was found that collisionality, plasma rotation and kinetic effects of both the thermal and fast ions all factor into determining the plasma stability to RWMs. These effects couple to produce a much more complicated picture of RWM stability than was previously thought, when rotation was the only effect being considered. For instance, the plasma can enter a more stable configuration if its rotation is resonant with either the precession drift or banana motions of the thermal particles. At these resonances, the plasma stability increases with decreasing collisionality. Off resonance, there is no collisionality dependence of the plasma stability. This result, for which experiment and theory agree, is promising for the lower collisionality ITER and ST-FNSF scenarios, and the results framework has been used to map out sample stability space as a function of the β of the alpha-particles and rotation for ITER scenarios.

Both the EF and RWM studies will benefit by the enhanced capabilities in NSTX-U, including the off-axis Neutral Beam (NB) for modification of the fast ion distribution and the current and rotation profile, the state-space controller for active control of low-n modes, and eventual real-time rotation and current profile control. These studies will be aided further by the proposed Non-axisymmetric Control Coil (NCC) system, which can provide an expanded mode spectrum for applied 3-D magnetic fields. RWM research will benefit by the expanded parameter regime in NSTX-U, allowing for a more detailed assessment of the coupled role of collisionality and rotation in determining RWM stability.

A critical element of any disruption mitigation system is the knowledge of when to trigger it in order to limit the deleterious effects of both the thermal and the current quench. To this end, a methodology to predict imminent disruptions was developed. This disruption warning algorithm development proceeded in two steps; first, by using individual sensor- and physics-based signals (neutron emission, loop voltage, plasma motion, RWM growth rates, locked mode growth, plasma inductance, pressure peaking, etc) to predict a disruption, and second by combining these single-signal tests using a simple algorithm. For the latter, a “point score” was assigned to each test, and the total points from 17 such tests were summed at each time slice to determine a aggregate warning level. Different values of this aggregate warning level were chosen, trading off false positives and late warnings. For a carefully chosen warning level threshold, only ~2% of disruptions could not be detected within 10 ms, and only 4% of discharges were false positives (Figure 1.5.1). This methodology will be developed further on NSTX-U and tested on other devices, at higher aspect ratio, as part of an ITPA Joint Experiment/Activity.

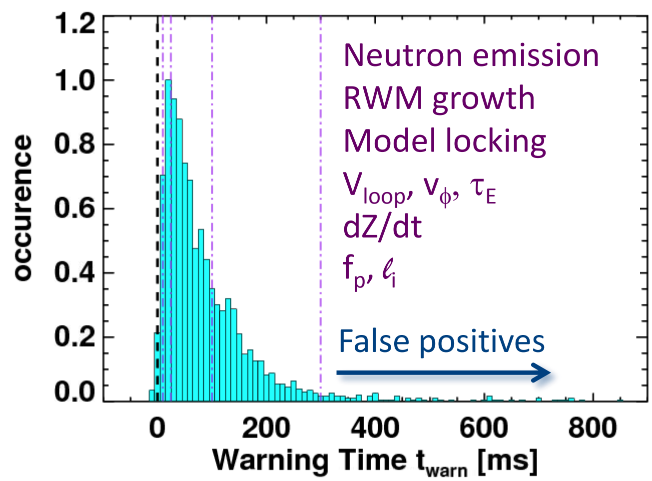


Figure 1.5.1: Results from disruption warning system showing time delay between warning and actual disruption for a chosen-combination of variables and trigger levels. Shown are only some of the variables that were used for this case.

If a disruption is imminent, a Massive Gas Injection system will be used to mitigate the effects of the thermal and current quenches. The system will be implemented in NSTX-U from the start of operation. An important feature of the system will be the ability to inject gas at different poloidal locations (midplane, off-midplane, lower divertor region and private flux regions) to assess the SOL gas penetration and to optimize the amount and type of gas injected. In particular, it is believed that due to the low energy of the plasma in the private flux region, this particular location may give the best penetration. The effect of this localized injection on the radiation profile and related localized heat deposition/melting will also be addressed. These studies will be aided by analysis of the gas penetration physics by such codes as DEGAS-2, and this can lead ultimately to validated predictions for ITER.

An alternative disruption mitigation approach is use of an electromagnetic particle injector (EPI), which uses a rail gun technique for rapid injection of a large amount of particles. This hardware will be proposed by an NSTX-U collaborator during 2013.

The lessons learned from the mode control and disruption studies will be integrated in the longer-term into a framework for physics-based disruption prediction, avoidance and mitigation. This framework is outlined in Fig. 1.5.2. It starts with a set of predictors (measurements) that consist of various plasma equilibrium, profile, and response characteristics. This information is then fed into a set of control algorithms that, using avoidance actuators, steer the plasma towards stable operation in the event that a disruption can be avoided. However, the same set of information is fed into a disruption warning system similar to the one discussed above. If it were deemed from this and other control information that a disruption is unavoidable and imminent, mitigation through early shutdown, MGI or EPI would be initiated. While many of the control algorithms, disruption warning triggers, etc., may be specific to NSTX-U, the general framework and algorithms would be applicable to ITER and ST-FNSF.

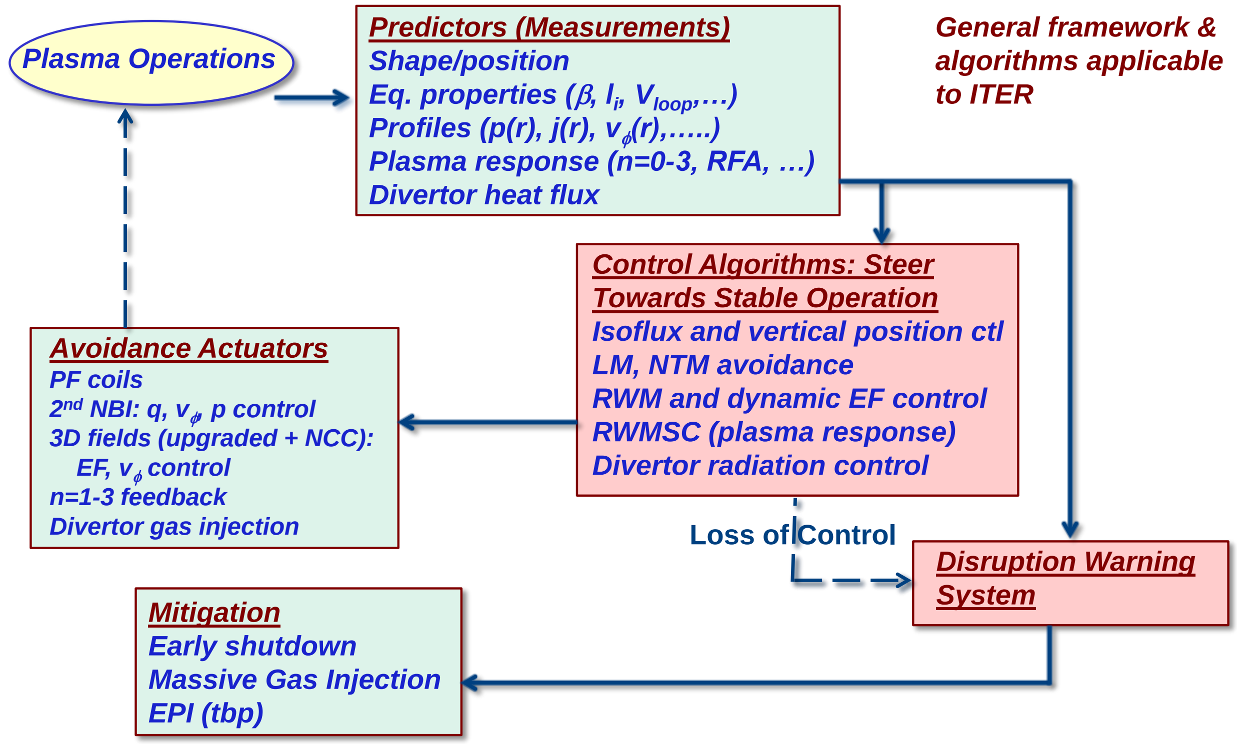


Figure 1.5.2: Framework for integrated, physics-based disruption prediction-avoidance-mitigation system.

ELM-control is one of the highest priority issues for ITER, as well as for present and other future devices such as ST-FNSF, for both limiting high power transient heat loads on the plasma facing components and for mitigating the effects of impurity accumulation. ELM control studies using applied, pulsed 3D fields in NSTX resulted in ELM-pacing, not suppression. By controlling the frequency of the 3D pulses, impurity accumulation and resulting radiated power levels could be controlled, with lower radiation using higher frequency pulses. NSTX-U will be able to use its expanded capability to study further ELM-mitigation with applied 3D fields using the upgraded 3D system plus the NCC. For one, this expanded capability will provide greater range of poloidal mode spectra that can be produced by the applied fields, which may lead to ELM suppression in certain configurations. Second, the applied 3D field NTV profiles can be controlled to some extent, leading to more edge localized perturbations and less core rotation damping. Lower core rotation is known to lead to compromised plasma stability. Finally, higher frequency pulsing of the applied 3D fields will be possible, which, if the trend observed on NSTX continues, can lead to further reductions in impurity accumulation and radiated power from the resulting ELM-pacing. In addition, NSTX-U will explore vertical kicks in its lower collisionality operating space as well as small-ELM and ELM-free regimes (such as Enhanced Pedestal H-mode).

NSTX-U will also be employing a lithium granule injector for developing ELM-pacing scenarios. The injector, which can control speed, size and thus penetration depth of the granules, has been developed and tested on the EAST tokamak. Each injected granule was found to trigger an ELM with almost 100% success rate. The injector will be installed on NSTX-U for further testing on that platform. This research and development can feed directly into that for a closely-related Be injector, which is of possible interest in both JET and ITER for ELM-pacing.

Key to optimizing ELM control is to understand the transport and stability of the pedestal region. To this end, NSTX research has utilized lithium conditioning to suppress ELMs as a basis for studying the stability and the key microinstabilities that govern the pedestal structure. It was found that it was primarily the change in the density gradient with Li conditioning that changed the microstability properties in the pedestal region and thus determined the stability to ELMs. Micro-tearing, hybrid TEM/KBM and ETG modes were all found to be important in different regions of the pedestal, with micro-tearing important at the top of the pedestal, the hybrid TEM/KBM dominant in the strong gradient region, and the ETG was believed to clamp the temperature gradient near the bottom of the pedestal. As the density profile changed with Li conditioning, so did the absolute location of the modes, although their relative locations (e.g, top, middle, bottom of pedestal) remained the same. A variety of tools on NSTX-U will be used to study the profile and microinstability changes. These include lithium conditioning with both bottom and top facing evaporators, cryo-pump to control the density especially at the edge, and the expanded operating space of NSTX-U, allowing for lower collisionality. A polarimeter to measure magnetic field fluctuations will be implemented early after the beginning of operations, and this will allow for a direct measurement to assess the importance of micro-tearing modes, which may be important for ITER, near the pedestal top [20]. Almost full coverage of the k-spectrum of density fluctuations, from ion scale to electron scale turbulence, will be available from the Beam Emission Spectroscopy and Microwave Scattering diagnostics. This research on NSTX-U will contribute to the determination of whether fueling and/or conditioning techniques will need to be developed on ITER for the purpose of modifying the edge density profile to control ELMs.

NSTX-U will study and mitigate high divertor heat fluxes in long-pulse discharges, with projected high heat flux values of P/R~20 MW/m and P/S~0.4 MW/m^2, which are comparable to the heat fluxes of 25 MW/m and 0.2 MW/m^2 projected for ITER. NSTX has already performed studies and contributed to a multi-machine database examining midplane SOL heat flux widths as a function of poloidal magnetic field. While the NSTX data helped define the strong trend by residing in the transition region between large and small heat flux widths (Fig. 1.5.3), the range of poloidal field was limited. NSTX-U will expand the operating space in both toroidal and poloidal field to validate the SOL width scalings, especially with its operation at lower collisionality, which is more ITER relevant.

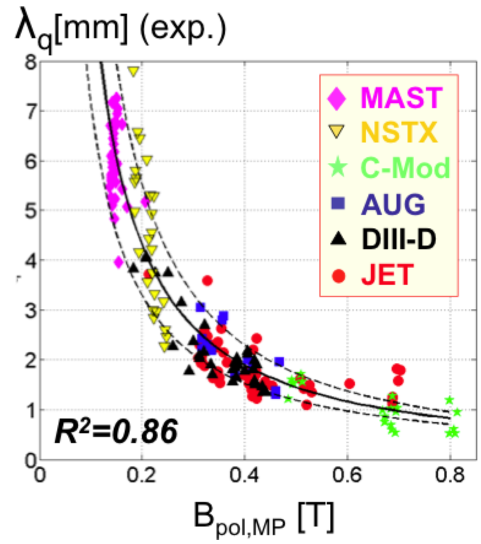


Fig. 1.5.3 Multi-machine study of SOL heat flux widths as a function of poloidal magnetic field, both in the outer midplane.

Divertor gas puffing was found to reduce the heat flux in the NSTX divertor by causing the divertor plasma to detach and become more highly radiating. In NSTX-U, this technique, which is one of the primary approaches proposed to mitigate heat fluxes in ITER, will be explored further with an aim towards developing closed loop control for producing and sustaining a radiative divertor compatible with high performance, steady-state operation. NSTX-U detachment studies and data can also contribute to model development necessary to improve detachment threshold predictions for existing experiments and also for ITER. The SOL heat flux width scalings discussed above will be studied in this partially detached regime, as will the effects of applied 3D fields. In the longer-term, once rows of high-Z plasma facing components are installed in NSTX-U, the resulting study of mixed material issues will address the ITER high priority issues of metal PFC heat load handling, migration and dust.

ITER's interest in studying and understanding impurity transport has clearly heightened over the past several years. This is understandable given that impurity seeding is required for achieving and maintaining good confinement in the metal and ITER-like wall machines ASDEX-U and JET, respectively. Is impurity transport neoclassical, especially near the edge? Will impurities accumulate in the ITER core plasma? These are critical questions for which the answers will help define the requirements for ELM-pacing to control the impurity content. NSTX has engaged in impurity transport studies of both carbon and lithium. MIST and STRAHL modeling results indicate that in plasmas with no lithium conditioning, the carbon density profile structure (i.e., peaking locations) agrees with predictions from neoclassical theory. With lithium conditioning, however, there are large departures from neoclassical, especially near the plasma edge. NSTX-U will be able to study the transport of impurities at lower collisionality. In particular, the near-full k-range of the density turbulence diagnostics will enable an assessment of the roles of neoclassical versus turbulent driven transport. The relative strengths of these two particle transport mechanisms will be assessed and manipulated by control of rotation shear (for suppressing the turbulence) using the second NB and the flexible 3D coil system. Mixed impurity effects will be studied at first with carbon and lithium, and later on high-Z transport will be studied once the rows of high-Z tiles are installed.

An area in which the NSTX-U operational regime will contact directly that of ITER is in Energetic Particles, where the unique NSTX-U unique parameter regime overlaps that of ITER with respect to $v_{fast}/v_{Alfvén}$ and β_{fast}/β_{tot} (Fig. 1.1.1). This contact/overlap will enable linking of the Alfvén Eigenmode (AE) activity and its associated effect on the fast ion population observed in NSTX-U to those expected for ITER with the caveat that the NSTX-U and ITER fast-ion ρ^* will differ. The observed AE activity in NSTX (and that projected for NSTX-U) was strongly non-linear, and such non-linearity may also be present in ITER hybrid and reversed-shear plasmas. This non-linear behavior was reflected by strong avalanches and the associated loss or redistribution of the fast ion distribution resulting from multiple mode overlap, and coupling of AE modes up to ion cyclotron frequencies to lower frequency kink and RWM modes in a predator-prey like relationship. The NSTX-U research on avalanches and non-linear physics is essential for the code validation needed for projecting to ITER. NSTX-U will be uniquely positioned to address this not only because of its operational regime but especially also by its ability to vary the q-profile, known to have a strong effect on non-linear behavior, with the 2nd NB. Furthermore, NSTX-U can study mode stability using its expanded range of toroidal field and the flexibility in the NB system to vary v_{fast} and β_{fast}. An AE antenna will be implemented to study mode stability and excitation, which will be measured by a suite of fast ion diagnostics. Finally, a particular strength of the NSTX-U EP research program will be the code Verification and Validation for both linear and non-linear calculations.

An additional high priority EP issue for ITER is to assess how applied 3D fields affect AE stability and the fast ion distribution. NSTX has laid the groundwork for this with initial studies of applied 3D fields. It was found that the both the mode characteristics and fast ion distribution changed after application (and removal) of these 3D fields on time scales comparable to the fast ion slowing down time. In particular, the changes in the fast ion distribution affected particles close to the higher frequency AE resonances. NSTX-U will bring added capabilities to study this issue, with its flexible neutral beam system to test a variety of injected fast ion distributions and to modify the q-profile, and with the flexible 3D applied field system (including the NCC) to change the mode spectrum of the applied fields and coupling to the fast ion distribution (see Chapter 6).

NSTX-U will be developing and validating heating and current drive scenarios with RF and NB heating, addressing issues of importance to both the ITER ICRF program and the Integrated Operating Scenarios task group. HHFW research includes studying the dependence of coupling on geometry and edge profiles using capabilities such as cryo-pumping and conditioning to change the plasma edge profiles and determine the amount of power that propagates into the core. Related to this is the generation of surface waves, which inhibit the deposition in the plasma core; the ability to vary the toroidal field and density is critical to this study. Wave power deposited in the SOL near the plasma edge was found to propagate to the divertor region along open field lines, causing higher heat loading of the divertor PFCs. The deposition and transport of wave power will be studied with advanced RF codes, such as TORIC, AORSA and GENRAY, Langmuir probe arrays and IR cameras. Preliminary calculations against NSTX data have already shown that the codes can qualitatively reproduce the topology of the RF power losses through the SOL into the divertor region. These studies will aid in the development of models that can be used to guide operational scenarios in ITER that can optimize the ICRF wave heating and current drive. Related to this development will also be studies of neutral beam and bootstrap current drive in partially and fully inductive NSTX-U discharges, and developing capabilities for controlling the current profile with NB, RF and other discharge evolution techniques. This control capability will be extended to developing robust algorithms for rotation control using, e.g., the off-axis NB and applied 3D fields, this development being important for tailoring target scenarios in both ITER and other future devices.

The areas discussed above represent those in which NSTX-U can most strongly contribute to the ITER R&D Urgent and High Priority issues. NSTX-U can contribute also to other high priority items. In particular, the focus of future research will include measuring the SOL turbulence, including “blobs”, which is believed to control SOL widths. The inferred cross-field transport and parallel flows will be used to validate the theoretical models contained in such codes as BOUT++, SOLT and XGC1. NSTX-U will also be actively developing control technology for heat flux mitigation. Vapor shielding (lithium, in the case of NSTX-U) will be assessed as a possible divertor strategy for mitigating heat fluxes and assessing compatibility with high-performance, steady-state plasmas. Vapor shielding science developed by studying Li in NSTX-U may also be applicable to ITER. More specifically, off-normal events (large ELMs, disruptions) may cause melting of first-wall Be or divertor W, and this melted material may be further/later vaporized in the SOL near the strike-point region of the ITER divertor. Studies of the dependence of H-mode access on plasma species, applied 3-D fields, X-point location and plasma current will be expanded in NSTX-U using the enhanced capability to modify the rotation profile with the 2nd NB, and expanded mode spectrum of the proposed NCC. The focus will also be on assessing achieved confinement with $P{\sim}P_{LH}$, the role of ion physics, and ultimately on how metal walls affect both the L-H and H-L transitions. The studies will transcend simple global parametric dependence, and will focus on the changes in turbulence and flows leading up to both the L-H and H-L transitions and relating these changes to theory. Other confinement issues important for ITER that will be addressed in NSTX-U include plasma confinement with applied 3-D fields, and the sources and scaling of intrinsic rotation and momentum transport, the latter topic being key for control of both macro- and micro-instabilities. Finally, assessment and development of reduced transport models will be carried out and will be coupled to the continued development of (P)TRANSP as a state-of-the-art integrated predictive transport tool.

1.5.1 International Tokamak Physics Activity (ITPA)

NSTX-U physicists have been participating actively in ITPA activities. NSTX-U has representatives in every ITPA Topical Group, with leadership in many, including past and future Chairs of ITPA groups, leaders of special Working Groups, and spokespersons for many Joint Experiments and Activities. As of January 2013, NSTX-U physicists are participating actively in 24 ITPA Joint Experiments and Activities (Table 1.5.1), many of which address the highest priority ITER R&D needs across all topical science areas. The work NSTX-U physicists is and will be doing towards addressing the high priority ITER R&D needs is described below.

Boundary Physics and PMI

- DSOL-24 Disruption heat loads
- PEP-6 Pedestal Structure and ELM stability in DN
- PEP-19 Basic mechanisms of edge transport with resonant magnetic perturbations
- PEP-27 Pedestal profile evolution following L-H/H-L transition
- PEP-29 Vertical jolts/kicks for ELM triggering and control
- PEP-34 ELM energy losses and their dimensionless scaling

Waves and Energetic Particles

- EP-2 Fast ion losses and Redistribution from Localized AEs
- EP-6 Fast-Ion Losses and associated heat load from edge perturbations

Advanced Scenarios and Control

- IOS-3.2 Define access conditions to get to SS scenario
- IOS-4.1 Access conditions for advanced inductive scenario with ITER-relevant restrictions
- IOS-4.3 Collisionality scaling of confinement in advanced inductive plasmas
- IOS-5.2 Maintaining ICRH Coupling in expected ITER Regime

Macroscopic Stability

- MDC-2 Joint experiments on resistive wall mode physics
- MDC-8 Current drive prevention/stabilization of NTMs
- MDC-15 Disruption database development
- MDC-17 Active disruption avoidance
- MDC-18 Evaluation of Axisymmetric control aspects

Transport and Turbulence

- TC-9 Scaling of intrinsic plasma rotation with no external momentum input
- TC-10 Experimental identification of ITG, TEM and ETG turbulence and comparison w/ codes
- TC-11 He and impurity profiles and transport coefficients
- TC-12 H-mode transport and confinement at low aspect ratio
- TC-15 Dependence of momentum and particle pinch on collisionality
- TC-17 ρ^* scaling of the edge intrinsic torque
- TC-24 Impact of resonant magnetic perturbations on transport and confinement

Table 1.5.1: International Tokamak Physics Activity (ITPA) joint experiments and activities involving participation by NSTX-U researchers as of January 2013.

1.6 Example Contributions to Model Validation

In this short section, we will give several examples of how the NSTX-U program will advance model validation in an effort to develop predictive capabilities across a range of topical areas. The model validation/predictive model development activities focus on specific goals within each group. This work involves active collaborations among experimentalists, theorists and modelers both at PPPL and in the community at large.

1.6.1 Macrostability

Kinetic Resistive Wall Mode (RWM) theory has been a central theme for Verification and Validation (V&V) efforts for NSTX and future NSTX-U plasmas. In particular, including ion kinetic effects has markedly improved the agreement between calculated and measured RWM stability thresholds compared to models for which only rotation effects were taken into account. The kinetic effects include those from both thermal and energetic ions. This work, multi-institutional among PPPL, Univ. Rochester and Culham Laboratory has led to an active Verification exercise among the MISK, MARS-K and HAGIS codes, which are being benchmarked against one another under ITPA auspices. Within this theoretical framework, experiments measuring global RWM stability versus collisionality support the importance of including kinetic effects and provide guidance for NSTX-U operations. In particular, mode stability was measured in NSTX using MHD spectroscopy (active probing of the plasma response to applied MHD perturbations). Theoretical predictions indicated that if the plasma were rotating at frequencies near kinetic drift resonances, the RWM would be strongly stabilized with decreasing collisionality. Off-resonance, collisionality variations would have little effect. The measured plasma response showed precisely those trends, with the n=1 plasma response decreasing with decreasing collisionality for “on resonance” rotation levels, and showing no change “off resonance”. Further work will include improving the physics model by including the phase space anisotropy of the energetic ions. Further, the model is starting to be applied to NSTX-U scenarios, and thus regions of rotation/collisionality space in which the RWM is predicted to be stable are being mapped out. This result can be tested experimentally using the 2nd NBI and both present and future 3D magnetic field coils to modify plasma rotation, and by using lithium and the expanded operating space in NSTX-U to extend the range of accessible collisionality to lower values.

A number of codes are being benchmarked against each other for the calculation of the Neoclassical Toroidal Velocity (NTV) arising from applied magnetic field perturbations. The NTV can be sufficiently large to modify rotation profiles, which in turn can affect both the macro- and micro-stability of the plasma. Consequently, it is important to be able to determine the NTV accurately for predicting plasma performance. The verification study involves six different codes, IPEC-NTV, MARS-K, MISK, MARS-Q, POCA and FORTEC-3D, and is based on a case with an applied m=3/n=1 field. Kinetic effects are taken into account differently in the above codes, with a bounce-averaged treatment in the first four codes but exact drift orbit calculations in the last two. Because of this and other differences among the codes, their range of applicability has been illuminated by this verification activity. For instance, IPEC-NTV and MARS-Q have found to be good approximations at high aspect ratio, but not at low R/a. POCA and FORTEC-3D, with full drift orbit treatment, were found to be the most appropriate for the low aspect ratio, low collisionality and high rotation frequency regime in NSTX and expected in NSTX-U plasmas. Improvements to enhance the predictive usage of these codes will include incorporation of a self-consistent treatment of the field perturbations and plasma response. The results of the code calculations would then be incorporated into the solution of the momentum balance equation for a prediction of the torque and rotation profiles.

1.6.2 Transport and Turbulence

In the area of thermal plasma transport, NSTX/NSTX-U is addressing three areas of electron transport, which is the dominant energy loss channel in L- and H-mode discharges. For high-k modes, NSTX-U is validating the Electron Temperature Gradient (ETG) electron thermal transport model. ETG modes were seen in a variety of NSTX scenarios using a unique high-k_r scattering system. In particular, ETG modes were found to be important in NBI-heated H-modes near the pedestal, in RF-heated L-modes, and in electron Internal Transport Barrier discharges. Identification of the measured turbulence as ETG modes was supported by both linear and non-linear gyrokinetic calculations, which predicted ETG-driven transport levels that agreed with the experimentally inferred transport levels. Initial TGLF predictions of T_e due to ETG modes in NSTX ITB plasmas, however, showed an over-prediction of electron temperature near the edge of the plasma. In NSTX-U, a new FIR-based high-k_θ scattering system will substantially improve our understanding of the ETG. This new diagnostic will include a 2D k-spectrum measurement, which will allow for the identification of the radial streamers believed to lead to enhanced electron transport. Further validation efforts will focus on using TGLF and the Multi-Mode reduced models as well as further non-linear gyrokinetic calculations using GYRO and GTS to benchmark against each other as well as against experiment. These studies reflect a strong collaborative effort across multiple institutions (PPPL, UC-Davis, Lehigh Univ., Tech-X, GA).

NSTX-U researchers are validating the CAE/GAE-driven electron thermal transport hypothesis. Studies so far have used the ORBIT code to study the stochastic electron transport driven by these modes. Mode amplitudes, the locations of peak mode amplitudes, mode widths, frequencies, and mode numbers are taken from experimental measurements and dispersion equations of these high-frequency modes, and the resulting calculated transport based on these quantities agrees well with the experimentally inferred values in the core of the plasma, where the mode amplitudes are observed to be largest. Future work will move toward a more predictive calculation using ORBIT and non-linear HYM calculations of the CAE/GAE mode characteristics. This work is also a multi-institutional effort involving PPPL, Johns Hopkins Univ., Univ. of Wisconsin and UCLA.

Finally on the electron transport topic, NSTX-U researchers have performed seminal work in validating the role of micro-tearing modes in driving electron transport in the mid-radius region of NSTX(-U) plasmas. Non-linear gyrokinetic calculations using GYRO have shown that micro-tearing modes in high beta H-mode NSTX plasmas can drive substantial electron transport at levels that match those inferred from experiment, and further, can lead to the same scaling with collisionality that was evident in the NSTX discharges. Global calculations with GTS will be done as soon as electromagnetic effects are implemented in the code, which should be within one year. To further assess the importance of micro-tearing modes in NSTX-U, a polarimetry diagnostic that is capable of measuring magnetic fluctuations driven by micro-tearing modes will be implemented from the start of operation. This diagnostic was designed using predicted micro-tearing mode characteristics from non-linear gyrokinetic calculations that employed a synthetic polarimetry diagnostic. Efforts are also being made to assess the sensitivity of the BES diagnostic to micro-tearing-related density fluctuations to be able to distinguish these modes from electrostatic ITG/TEM turbulence. Again, this is being done in conjunction with the non-linear gyrokinetic calculations using synthetic diagnostics. For the development of reduced models, the theory-based TGLF will be validated and further developed against NSTX-U discharges, and analytic expressions that have been used to predict micro-tearing-driven transport will be improved with better $\delta B/B$ estimates.

1.6.3 Boundary Physics

Several V&V studies have been and are presently being undertaken in the Boundary Physics area. Pedestal widths and structures are being compared to those predicted from kinetic ballooning mode (KBM) calculations at low aspect ratio, with predicted neoclassical effects, and with predicted paleoclassical transport. NSTX data showed pedestal widths to scale more strongly with $\beta_{pol,ped}$ than was observed (and predicted from KBM calculations) for higher aspect ratio, at first-glance drawing this explanation into question for NSTX plasmas. However, KBM calculations specific to NSTX plasmas actually reproduce to a larger extent this stronger scaling, giving reasonable agreement with the NSTX data. XGC0 calculations, on the other hand, show that the predicted neoclassical pedestal width is significantly narrower than that observed experimentally. Paleoclassical transport semi-quantitatively agrees with NSTX pedestal gradients in plasmas both with and without lithium conditioning.

How the snowflake divertor configuration, where the magnetic shear inside the separatrix is higher than that in conventional divertor configurations, may change the pedestal stability due

KBM-peeling modes is presently being investigated. Results from TCV snowflake plasmas are consistent with improved pedestal stability, with increased frequency but reduced size of the Type I ELMs. NSTX plasmas indicate the opposite, however; during snowflake configurations, ELMs that were otherwise stabilized by lithium reappeared, and calculations indicated a reduced stable operating window for peeling-ballooning modes. Also, the overall pedestal stability of DIII-D snowflake discharges did not appear to change. These different experimental results clearly point to the need for future testing of the peeling-ballooning mode theory for these high magnetic shear plasmas. Further, analytic and 3D calculations using the BOUT++ code for snowflake configurations have suggested enhanced turbulence in the X-point region due to ballooning modes, electrostatic flute instabilities and resistive ballooning modes. Edge turbulence will be assessed in NSTX-U plasmas using the Gas Puff Imaging (GPI) diagnostic.

The last example in the Boundary Physics area is validation of the DEGAS-2 neutral transport code against NSTX GPI data. Implementation of a more sophisticated neutral transport calculation in codes such as TRANSP would be a major advance in the study of particle and impurity transport. For this validation exercise, the measured emission from the GPI diagnostic is compared to 3D simulations from DEGAS-2 using a GPI synthetic diagnostic. The DEGAS-2 calculations are constrained by electron density and temperature measurements near the edge as well as reconstructions of the magnetic equilibrium. The GPI data (emission and gas puff) are absolutely calibrated. Preliminary comparisons show good agreement between the two.

1.6.4 Materials and PFCs

New materials simulation techniques are providing an understanding of the behavior of deuterium in lithium-conditioned carbon PFCs. Many plasma codes utilize data from simple models that simulate the behavior of ions incident on materials but neglect the role of chemical interactions within the material (e.g. TRIM, MARLOWE). In contrast, quantum-classical molecular dynamics (QCMD) models simulate the interaction of these incident ions and apply quantum-mechanical calculations to determine the interaction potentials during the simulation. This simulation method has already yielded powerful results explaining the uptake of deuterium in lithium-conditioned, carbon PFCs and the important role of oxygen in this process [21]. In particular the simulations indicate that deuterium is more likely to be chemically bound to oxygen present in the PFC present from reactions with residual gas or other ubiquitous sources in the tokamak. These results have helped understand numerous experimental observations of the surface chemistry of these PFCs [22,23]. In particular, these experimental studies inferred the presence of deuterium by observing shifts in the energy spectra of oxygen atoms in PFC samples; a result that is consistent with those of the simulations, which show preferential oxygen bonding. Tabulated or parameterized results from the code for the uptake or reflection of incident deuterium ions will eventually be coupled to plasma simulation codes, to evolve the plasma alongside the material simulation.

1.6.5 Energetic Particles

One of the overarching aims of NSTX-U is to demonstrate stationary, 100% non-inductive performance that extrapolates to ≥ 1 MW/m^2 neutron wall loading in FNSF. A critical ingredient of this goal is to optimize the neutral beam-driven current profile and magnitude. Current profile studies in NSTX have sometimes exhibited discrepancies between reconstructions assuming classical fast ion physics and profiles based on MSE measurements, with much reduced core currents seen in the measurements. TAE avalanches are observed in these particular discharges, and the discrepancy can be reconciled by applying an impulsive anomalous fast ion diffusivity (AFID) in TRANSP that is both spatially and temporally localized to reflect the effect of the TAE modes have on the fast ion distribution. A proper choice of this fast ion diffusivity brings temporally evolving measured and calculated neutron rates, as well as current profiles, into agreement. The AFID, however, is clearly ad-hoc, and this underscores the need for predictive modeling of avalanche-induced fast ion transport.

Being able to predict this transport involves two steps. The first is the accurate prediction of the TAE mode stability and structure, and the second is the response of the fast ion distribution to these modes. For the first step, a strong theory-experiment collaboration is addressing the Verification and Validation of both linear and non-linear codes for predicting the TAE stability and mode structure. Unstable modes and their properties (mode spectrum, structure and temporal evolution) can be measured using Beam Emission Spectroscopy and reflectometry. On NSTX, these measured mode structures were consistent with those calculated from the linear NOVA-K code. The mode amplitudes calculated by NOVA-K were normalized to match the measured density perturbation, and these normalized modes were then input into the ORBIT guiding center tracking code. The particle losses in the presence of these TAE modes inferred by ORBIT agreed quite well with those inferred from drops in the measured neutron rate during the avalanches, confirming the validity of both the mode structure calculation and the particle tracking. Future work on NSTX-U will involve using SPIRAL, a full-orbit code, as well as ORBIT, and developing fully non-linear models with M3D-K that can predict mode amplitudes as well as structure.

The second phase of developing a predictive capability is the modification of the fast ion distribution in the presence of these modes. To this end, two different approaches are being pursued. The first treats the modifications to the fast ion distribution function and resulting transport that are induced by resonant interactions with AE modes. In this approach the resonant fast ion transport is modeled through use of a "probability function", based on adiabatic invariants, which reflects kicks in both energy and canonical angular momentum due to AE modes. While the probability function is calculated from models such as ORBIT, it is also based on experimental quantities such as neutron rate and mode amplitude. This reduced model is presently being tested against full ORBIT calculations, and resulting energy distributions show very good agreement for both co- and counter-phased particles using measured multi-AE mode amplitude spectra (at different n). Once developed, this model will be implemented in the TRANSP code.

The second approach to developing predictive capability for modification of the fast ion distribution in the presence of AE modes uses a quasi-linear relaxation model to determine the radial fast ion profile as a function of time in response to a given set of AE modes. This approach has been developed and tested for DIII-D, and will be extended into the NSTX-U parameter regime. The development approach for NSTX-U is to compute the spectrum of unstable TAE modes (using NOVA) for given NSTX scenarios. NOVA-K will be used to compute the growth and damping rate of these modes, and an unperturbed fast ion distribution function will be computed using TRANSP/NUBEAM. The quasi-linear model will then be applied to find the "relaxed" fast ion distribution function as a function of time, and comparisons to measured FIDA, NPA and neutron measurements will be made to validate this reduced model. Once validated, the model will be implemented in TRANSP.

1.6.6 Wave Heating and Current Drive

A major element of the NSTX-U RF program is the Verification and Validation of simulation codes that not only support the NSTX-U program, but which can also be used to predict heating and current drive in ITER and other fusion devices such as ST-FNSF. Details of this work are described in Sections 7.2 and 7.3, and only two elements will be summarized here.

One of the key issues in predicting the RF power deposition into the plasma is being able to predict direct absorption of RF power by the neutral beam fast ions. To this end, the 2D full wave AORSA code has been coupled to the Monte-Carlo ORBIT-RF codes to perform a self-consistent calculation of the interaction between resonant fast ions and injected HHFW power. Validation of this calculation is being carried out using the computed fast ion distribution into a synthetic diagnostic in the FIDASIM code, which then computes the expected fast ion profile that would be measured by the FIDA diagnostic. While these simulated profiles agreed in a quantitative manner reasonably well with the measured profiles, the computed profiles have a slightly outward shift relative to the measurement. Investigation of the source of this outward shift is underway.

GENRAY is a ray tracing code for HHFW that also contains modules for the determination of the RF current drive. GENRAY has been coupled to CQL3D, a bounce-averaged Fokker-Planck code, for taking into account the fast ion absorption of RF power and thus modifications to the current drive. The computed results, again processed with the synthetic diagnostic in the FIDASIM code, were compared to actual FIDA signals for NB+HHFW cases. Initial comparisons gave poor agreement, with the computed FIDA signals significantly shifted toward the magnetic axis relative to the measured signals. These calculations, however, tracked only the guiding centers and assumed zero orbit widths. Implementation of the finite orbit width model into the quasi-linear calculation shows a marked improvement in the agreement between measured and calculated profiles. Presently being worked on is a full-finite orbit width model that includes radial transport and a non-thermal bootstrap current contribution.

1.6.7 Plasma Formation and Ramp-up

The final example that will be presented, but which is discussed in much more detail in Chapter 8, is using both 2D and 3D simulation codes to understand non-inductive startup. Development of this capability will allow for projections to ST-FNSF operation with a small or no solenoid. Axisymmetric TSC simulations of CHI-initiated NSTX plasmas show good agreement, with closed flux surfaces forming due to the large electric fields generated by the CHI. However, these 2D calculations may not contain the full MHD physics that forms the basis for the creation of closed flux surfaces. To this end, 3D calculations using the NIMROD code have been initiated.

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