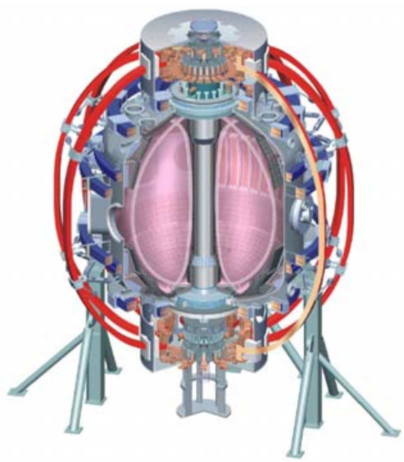
Chapter 1 - Overview of the NSTX Research Program Plan for 2009-2013

This page intentionally left blank

Chapter 1



Overview of the NSTX Research Program Plan for 2009-2013

1.1 Introduction

The Spherical Torus (ST) concept is a low-aspect-ratio tokamak magnetic configuration characterized by strong intrinsic plasma shaping and enhanced stabilizing magnetic field line curvature. These characteristics are shown pictorially in Figure 1.1.1. 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. NSTX has demonstrated that ST's can access a very wide range of dimensionless plasma parameter space with toroidal beta β_t up to 40% (local $\beta \sim 1$), normalized beta β_N up to 7, plasma elongation κ up to 3, normalized fast-ion speed

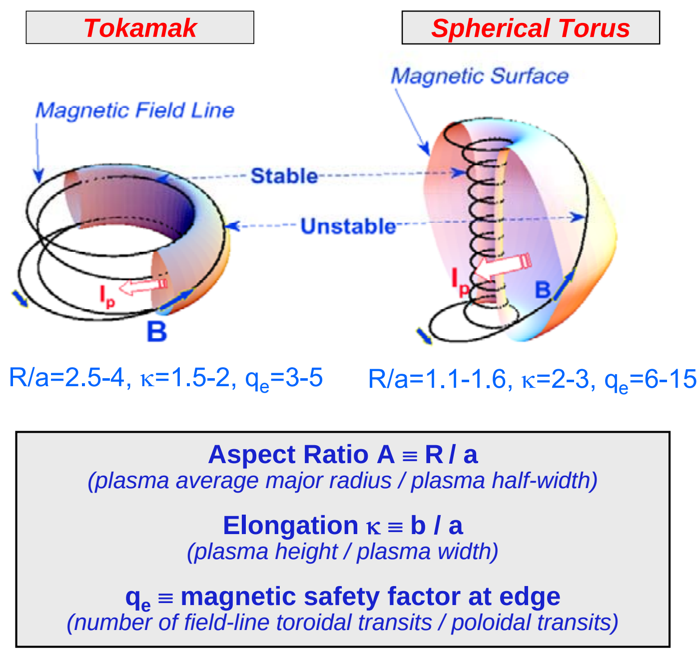


Figure 1.1.1: Comparison of magnetic field-line trajectories, aspect ratio, elongation, and safety factor parameters for a tokamak (left) and spherical torus (right).

$V_{fast}/V_{Alfvén}$ 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 that accessible in conventional tokamaks, and these parameters approach those achievable in other high-β alternative concepts. 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. The ST addresses fundamental issues in magnetic fusion energy science in the areas of: macroscopic stability, turbulence and transport, wave-particle interactions, boundary physics, and solenoid-free current formation and sustainment. For fusion applications, the high β, compact geometry, accessibility, modularity, and simplified magnets of the ST are potential advantages for plasma material interaction studies at high heat flux, nuclear component testing, and for a fusion power reactor.

NSTX is the world's highest performance ST research facility and is the centerpiece of the U.S. ST national research program. As illustrated below in Figure 1.1.2, NSTX is an essential element in the program to advance the understanding and development of the ST concept while also complementing and accelerating the development of all DEMO concepts.

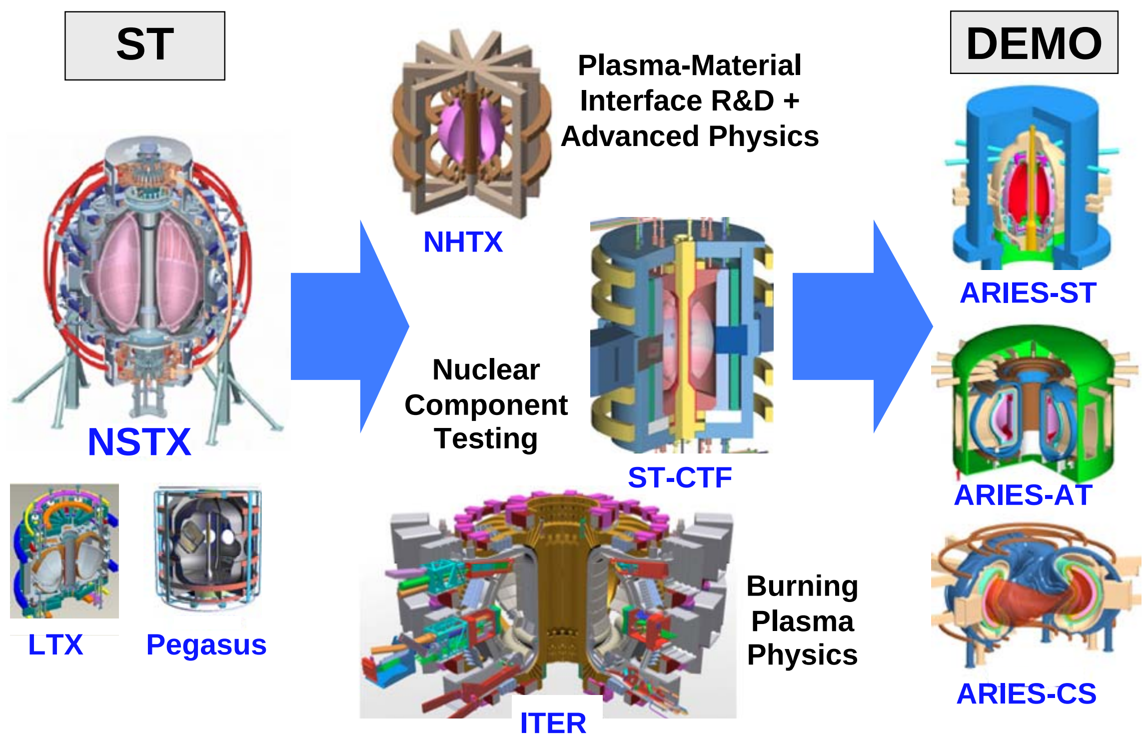


Figure 1.1.2: (left) U.S. ST research facilities, (middle) next-step STs to complement tokamaks and ITER burning plasma research by addressing key PMI and neutron fluence gaps between ITER and DEMO, and (right) spherical torus (ST), advanced tokamak (AT), and compact stellarator (CS) DEMO/reactor concepts developed by the ARIES reactor studies group (http://www-ferp.ucsd.edu/aries/).

The three overarching mission elements of the NSTX research program for 2009-2013 are:

(1) Determine the physics properties of the ST - utilizing its low aspect-ratio (A ~ 1.5) and very high ratio of plasma pressure to magnetic pressure (up to order-unity beta) - to advance toroidal plasma science.

(2) Provide unique ST data to the tokamak knowledge-base in support of ITER final design activities and preparation for burning plasma research in ITER through participation in the International Tokamak Physics Activity (ITPA) and U.S. Burning Plasma Organization (USBPO), while also benefiting from tokamak and ITER R & D.

(3) Develop the ST knowledge-base and attractive ST operating scenarios and configurations to enable the ST to narrow key gaps between expected ITER performance and the fusion environment expected in a demonstration fusion power plant (DEMO) – including an ST-DEMO. In particular, NSTX research aims to enable the ST to be utilized to:

a) Integrate DEMO-relevant Plasma Material Interface (PMI) solutions with high plasma performance (mission of the National High-power advanced-Torus eXperiment - NHTX).

b) Produce DEMO-relevant neutron flux and fluence at high duty-factor in an ST-based Component Test Facility (ST-CTF).

Mission **(3)** above ("Develop the ST knowledge-base and attractive ST operating scenarios…") is the highest priority mission of the NSTX research program for 2009-2013, and mission **(1)** provides the scientific underpinning for missions **(2)** and **(3)**. Developing the knowledge-base and performance for addressing missions **3a)** and **3b)** above would also greatly enhance the prospects for developing an ST-based DEMO.

The remainder of this chapter first describes the unique parameter regimes accessed in NSTX (Section 1.2) to provide context for understanding NSTX contributions to tokamak physics and ITER (Section 1.3) and, most importantly, for motivating fusion energy science applications of the ST (Section 1.4) and identifying scientific gaps (Section 1.5) and opportunities (Section 1.6). Section 1.7 briefly summarizes the NSTX 10 year scientific research objectives, and Section 1.8 describes the scientific organizational structure of the NSTX national research team.

1.2 Unique Parameter Regimes Accessed by NSTX

1.2.1 Macroscopic Stability

The fundamental fusion advantage and scientific opportunity enabled by low aspect ratio $A \equiv R/a$ (see Figure 1.1.1) is stable access to high plasma beta. The most commonly used definition of beta for tokamaks is the toroidal beta = $\beta_T \equiv 2\mu_0\langle p\rangle / B_T^2$ where $\langle p\rangle$ is the volume-averaged pressure and B_T is the vacuum toroidal field at the plasma geometric center. Since the fusion power density of a tokamak scales as $\beta_T^2 B_T^4$, achieving high β_T is clearly advantageous. The reduced aspect ratio of NSTX increases the plasma current I_P accessible for a given plasma minor radius *a*, toroidal field B_T, and magnetic safety factor. As shown in Figure 1.2.1.1, the maximum normalized current $I_N \equiv I_P/aB_T$ achievable in NSTX is a factor of 2 to 3 higher than in a standard aspect ratio tokamak. Experimental results largely confirm ideal MHD theoretical predictions that the maximum achievable β_T is proportional to the normalized current. The proportionality coefficient relating β_T to I_N is the normalized $\beta = \beta_N \equiv \beta_T(\%) / I_N$ (MA/mT). As shown in Figure 1.2.1.1, the maximum β_N achieved in NSTX is also significantly higher (up to a factor of 1.7 at high I_N) than is achievable at higher A.

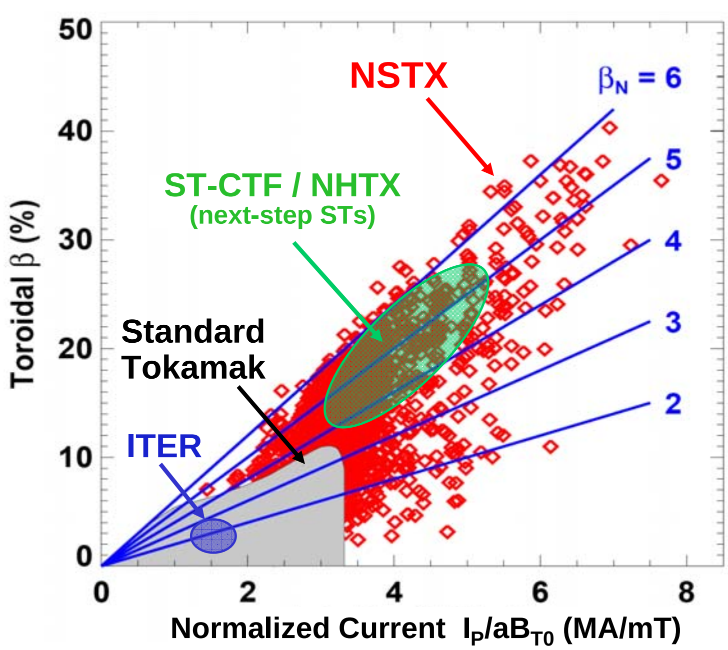


Figure 1.2.1.1: Troyon diagram showing toroidal beta achieved/expected versus normalized current I_P/aB_T for ITER, present standard aspect ratio (A~3) tokamaks, NSTX, and next-step STs NHTX and ST-CTF.

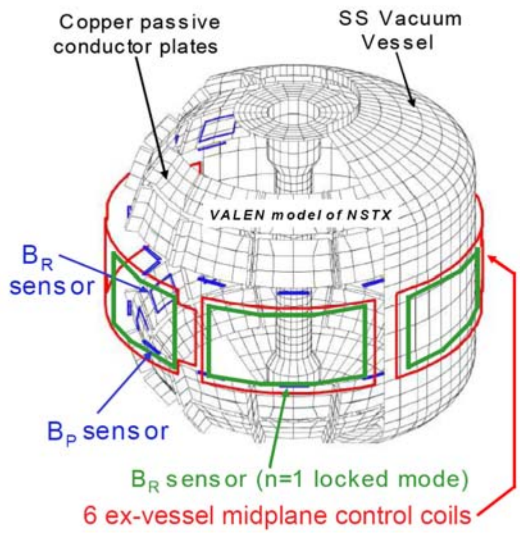


Figure 1.2.1.2: Vacuum vessel, passive conducting structures, non-axisymmetric field sensors, and ex-vessel mid-plane MHD mode control coils of NSTX.

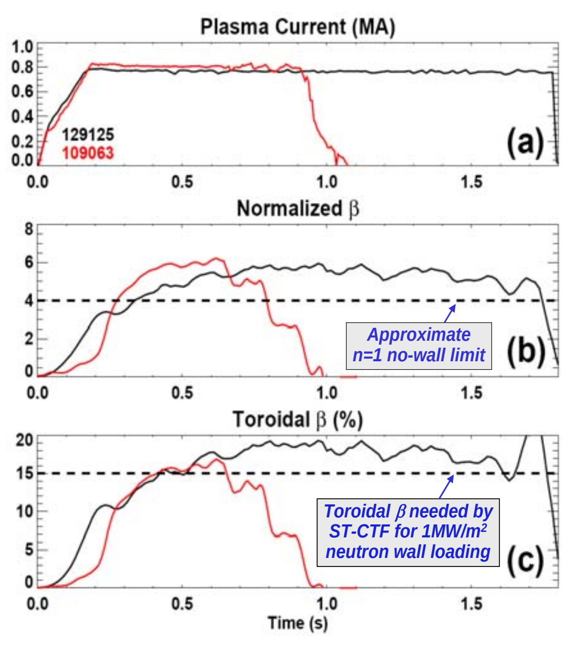


Figure 1.2.1.2: Since 2002 NSTX has (a) doubled the flat-top duration, (b) increased the duration of operation above the n=1 no-wall limit by a factor of 2.5, and (c) increased the duration of β_T > 15% (needed for ST-CTF) by a factor of 4. For reference, the energy confinement time $\tau_E \approx$ 30-50ms and current redistribution time $\tau_{CR} \approx$ 0.3-0.4s in these plasmas.

Rotational stabilization of the resistive wall mode (RWM) made possible by rapid rotation generated by strong tangential NBI heating is an essential element in achieving these high beta values far above the n=1 no-wall kink instability limit ($\beta_N \approx 4$ in NSTX). Access to very high β values made possible by low aspect ratio and wall stabilization from rapid toroidal rotation is a unique capability in the world program that impacts all science areas in NSTX research and improves understanding of tokamak MHD equilibrium and stability physics.

NSTX has progressed far beyond passive stabilization of the resistive wall mode and presently has the most advanced mode control system of any ST in the world program. As shown in Figure 1.2.1.2 and discussed in Chapter 2, NSTX has an extensive array of non-axisymmetric field sensors and 6 external mid-plane active control coils. These sensors and active control coils have been utilized to correct intrinsic error fields and maintain high plasma rotation to passively stabilize the resistive wall mode, to actively stabilize resistive wall modes when the plasma rotation is insufficient to passively stabilize the RWM, and to controllably modify the toroidal rotation profile by damping toroidal flow with 3D fields. As shown in Figure 1.2.1.3, the duration of sustained high-β has increased significantly during the last 5 years. This has been achieved through a combination of increased plasma shaping, control of error fields, resistive wall mode control, and conditioning the plasma facing components with lithium. The equilibrium and stability control tools of NSTX (described in Chapter 2) have played a major role making NSTX the highest performance ST in the world program, and the planned major upgrades of NSTX will further advance the understanding and achievement of sustained high β operation for STs and tokamaks.

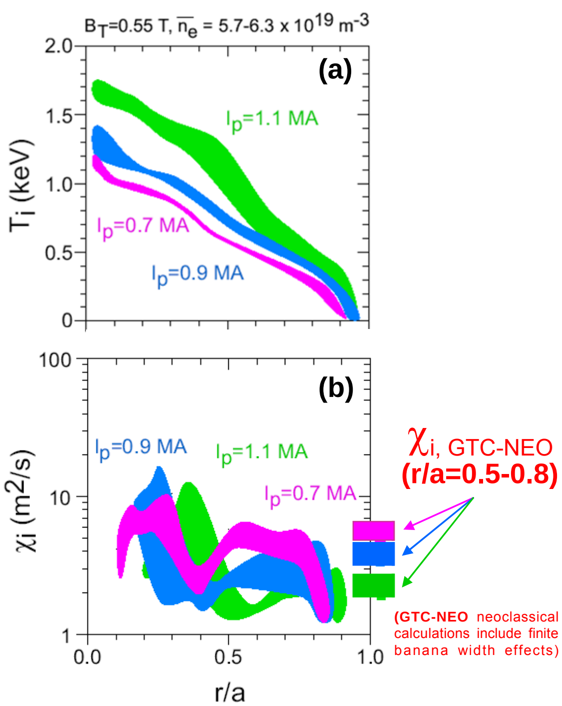


Figure 1.2.2.1: (a) Illustration of nearly linear dependence of ion temperature on I_P, and (b) consistency between measured ion thermal diffusivities and neoclassical predictions of the GTC-Neo code.

1.2.2 Transport and Turbulence

Just as the enhanced toroidicity (low aspect ratio) and increased natural shaping of NSTX plasmas can suppress macroscopic instabilities, these ST characteristics are also expected to reduce microturbulence levels, and thus reduce the transport associated with the microturbulence. Enhanced toroidicity also results in higher trapped particle fractions, which can influence Trapped Electron Mode (TEM) turbulence. Furthermore, the low toroidal field and near sonic toroidal flow yield large values of the E×B shearing rates expected to be important for the suppression of long wavelength micro-turbulence and its associated transport. Consistent with such suppression,

Figure 1.2.2.1 shows examples of H-mode plasmas with high flow velocities (and E×B shearing rates) which exhibit neoclassical thermal transport over most of the outer plasma minor radius for a range of scanned plasma current values at fixed toroidal field and density. As described in Chapter 3, this unique regime of apparently suppressed ion turbulence enables studies of electron transport with reduced influence of long-wavelength turbulence, thereby potentially enabling isolation and determination of the modes responsible for anomalous electron transport.

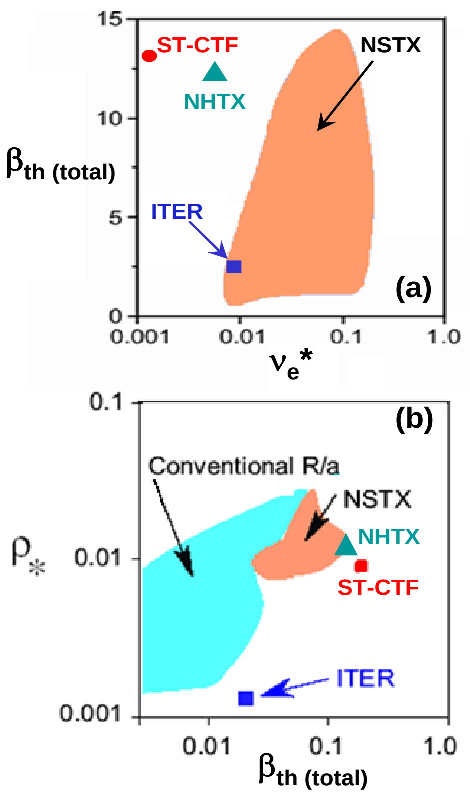
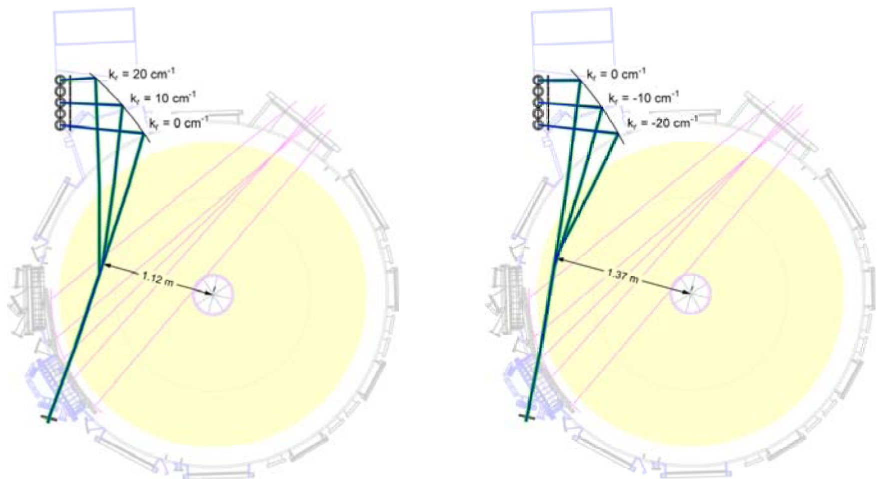


Figure 1.2.2.2: (a) Range of thermal β accessible in NSTX versus electron collisionality with comparison to ITER, NHTX, and ST-CTF, and (b) ρ^* vs. β_{th}.

Further, as shown in Figure 1.2.2.2, NSTX can operate in regimes with plasma collisionality similar to that of conventional aspect ratio tokamaks including ITER, but in which both the thermal plasma beta β_{th} and normalized ion gyro-radius $\rho^* \equiv \rho_i / a$ respectively, can be up to a factor of 10 greater than those at higher toroidal field and aspect ratio. The high β (up to 40%) enables NSTX to explore electromagnetic and stochastic magnetic effects that may influence transport, and the large electron gyro-radius (0.1 mm) has already enabled direct and unique measurements of spatially-resolved electron-scale turbulence using the tangential microwave scattering diagnostic shown in Figure 1.2.2.3. High-k fluctuations consistent with Electron Temperature Gradient (ETG) turbulence have been measured with this diagnostic, and the possible link between ETG turbulence and anomalous electron transport is actively being investigated as described in Chapter 3. Such research is important for understanding magnetic confinement generally, since the underlying cause(s) of electron transport remain elusive, and electron confinement could determine the burning plasma performance of ITER and the performance of future STs.



- k_r=2 (upper ITG/TEM) to ~24 (ETG) cm^{-1} (ρ_e~0.01 cm)
- Δr ~ 6 cm, Δk ~ 1 cm^{-1}
- Can vary scattering volume location (axis to near edge)

Figure 1.2.2.3: High-k microwave scattering diagnostic beam trajectories and measurement capabilities.

1.2.3 Waves and Energetic Particles

Spherical torus (ST) plasmas, such as those in NSTX, provide a unique opportunity for studying wave-particle interactions in plasma at high β and high dielectric constant $\varepsilon \equiv \omega_{pe}^2 / \Omega_{ce}^2$ ~ 10-100. The over-dense plasma conditions of the ST make plasmas inaccessible to lower hybrid (LH) and electron cyclotron (EC) waves. As a result, waves that can propagate in over-dense plasma conditions are required for the ST, and the two leading candidates are the high harmonic fast wave (HHFW) with ω/Ω_D=8-16 and electron Bernstein wave (EBW) with ω/Ω_e = 1-3. These waves are being developed and explored in NSTX to assist non-inductive plasma startup, plasma current ramp-up, and sustained plasma operation at high β. The most challenging aspect of exploiting these waves for fusion applications is efficient coupling of these waves from the edge to the hot plasma core.

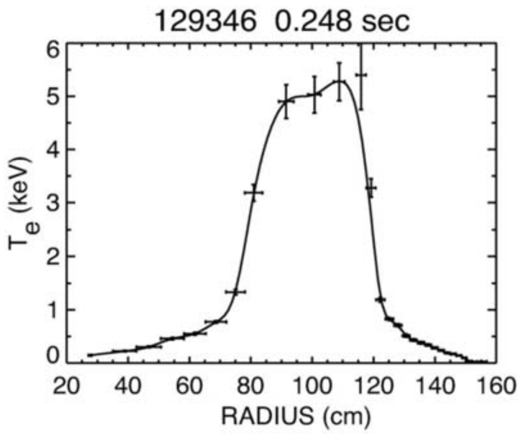


Figure 1.2.3.1: The highest electron temperature $T_e \approx$ 5keV produced in an ST has been achieved with 3MW of HHFW heating in NSTX in a target plasma with I_P = 600kA, $B_T(0)$ = 5.1kG, $n_e(0)=1.3\times10^{19}m^{-3}$, and $\beta_e(0) \approx 10\%$.

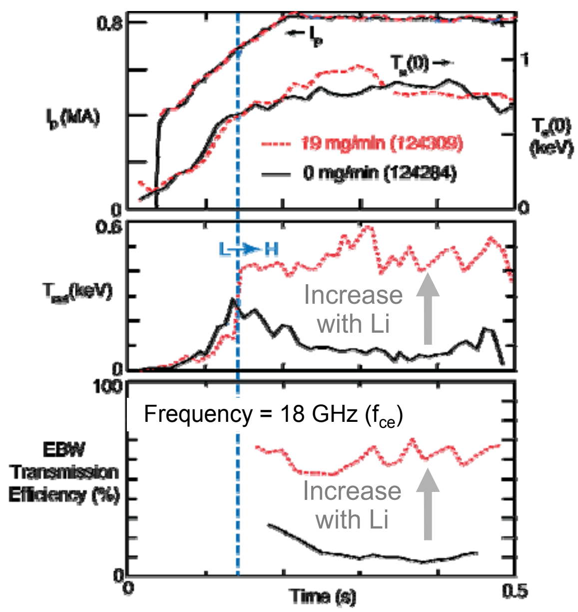


Figure 1.2.3.2: EBW radiation temperature (middle) and transmission efficiency (bottom) as a function of lithium evaporation rate which influences T_e at the mode conversion layer in the scrape-off-layer.

For the HHFW, research in NSTX has shown that edge effects including the excitation of parametric decay instabilities and surface waves can significantly damp wave power at the plasma edge reducing coupling efficiency to the plasma core. As described in Chapter 4, increased TF, reduced plasma density near the RF antenna, and lower wave phase velocities all reduce HHFW edge losses. Up to 90% HHFW coupling efficiency has now been demonstrated, and as shown in Figure 1.2.3.1, the highest electron temperatures attained in NSTX have been achieved with HHFW. With this improved understanding of coupling and heating, the HHFW has become a powerful tool for probing core electron transport physics and is poised to become a reliable auxiliary heating and current drive tool for the ST. Further, as described in Chapter 4, the coupling issues encountered for HHFW on NSTX are also relevant to ICRF heating schemes planned for ITER.

While the HHFW is launched as a radially evanescent wave that must tunnel through a cutoff determined by the plasma density profile in front of the antenna, the EBW is typically launched as a radially propagating electromagnetic wave in vacuum that must undergo multiple mode conversion process (O-mode to X-mode to Bernstein → O-X-B) to transition into an electrostatic wave that can propagate freely in an over-dense plasma and be absorbed at an electron cyclotron resonance. Thus, the EBW is especially sensitive to plasma conditions at the mode-conversion layer(s). For present ST experimental conditions the EBW mode conversion layer commonly resides near or in the edge scrape-off-layer (SOL). A key finding from NSTX experiments is that collisional damping of the EBW at the mode conversion layer strongly influences the EBW coupling efficiency.

EBW emission (B-X-O) measurements in NSTX indicate that the electron temperature at the mode conversion layer is a critical parameter, and strong damping is observed for $T_e < 20$eV. As discussed in Chapter 4 and shown in Figure 1.2.3.2, evaporated lithium has been shown to increase T_e from 10 to 30eV at the mode conversion layer and increase the transmission efficiency from 10% to 65% at 18 GHz (f_{ce}) and from 10% to 50% at 28 GHz ($2f_{ce}$). These are the highest EBW transmission efficiencies observed to-date in ST H-mode plasmas, and further optimization is possible to increase the EBW coupling efficiency for wave heating and current-drive applications in NSTX, future STs, and alternative concepts with over-dense plasmas.

Beyond waves that are excited by external means, fast ions produced during neutral beam injection in NSTX have velocities that exceed the Alfvén speed and can consequently resonate strongly with a wide variety of Alfvén waves and destabilize them. Further, the fraction of fast ion beta relative to the total beta accessible in NSTX can be large and is also important in determining the instability drive. As shown in Figure 1.2.3.3 the ratio of NBI fast-ion speed normalized to the Alfvén speed can approach 5 in NSTX, and the fast-ion to total central beta ratio can reach up to 80%. As seen in the figure, NSTX can access the

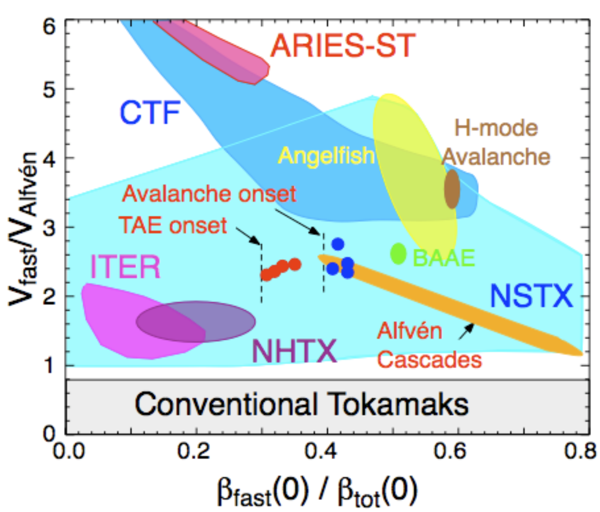


Figure 1.2.3.3: Normalized fast-ion velocity and β accessible in NSTX with NBI heating compared to ITER (with only α's included) and future ST devices: NHTX (NBI), ST-CTF (NBI+α's), and ARIES-ST reactor (α's).

fast-ion instability drive regime expected in ITER (shown for α-particles only - not including NBI fast-ions) and NHTX (NBI fast-ions only), has significant overlap with the fast ion regime expected in an ST-CTF (NBI+α-particles), and can approach the parameter regime of an ST reactor (ARIES-ST).

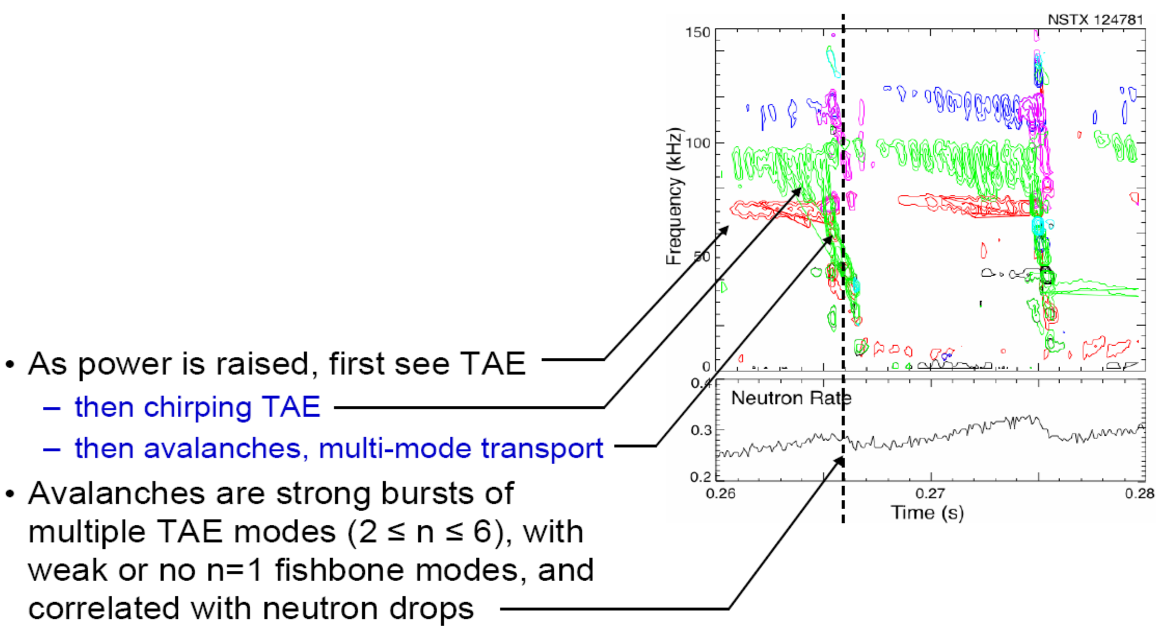


Figure 1.2.3.4: Time evolution of mode activity (top) and neutron rate (lower) for TAE avalanche events.

Importantly, as described in Chapter 4, the high normalized fast ion velocity and pressure accessible in NSTX can excite Toroidal Alfvén Eigenmode (TAE) avalanches in which multiple interacting TAE modes are excited simultaneously potentially leading to significant fast-ion transport and/or loss. ITER with α-particles and NBI is expected to be unstable to such TAE instabilities and may exhibit avalanches, and ST-CTF is likely strongly in the avalanche regime. The time evolution of mode activity associated with an avalanche event is shown in Figure 1.2.3.4 which shows how multiple modes interact and lead a significant (up to 15%) decrease in the neutron rate indicating significant fast ion transport and/or loss from the plasma core. As described in Chapter 4, NSTX has unique access to and diagnosis of this TAE avalanche regime including MSE q profile measurements at very low B_T = 0.35-0.55T and fast ion distribution function and fast-ion loss diagnostics. NSTX research has yielded additional research results including the discovery of CAE/GAE (Compressional/Global AE), Alfvén-acoustic modes, and a new understanding of chirping modes. These experimental results have helped advance theoretical modeling and have significant implications for future burning plasma devices.

1.2.4 Boundary Physics:

The plasma boundary is especially influenced by the unique magnetic geometry and compactness of the ST. As described in Chapter 5, the high global magnetic shear, large fraction of trapped particles, and reduced connection length impact edge transport and stability in both the closed-field-line edge confinement region and on the open magnetic field lines in the scrape-off layer (SOL). For example, as shown in Figure 1.2.4.1, the compactness of the ST can result in very high heat fluxes up to 10MW/m^2 at the divertor in lower-single-null diverted plasmas at reduced triangularity. Such heat fluxes are comparable to those expected in ITER and enable research on heat-flux mitigation strategies such as enhanced divertor radiation and poloidal flux expansion. As seen in the figure, operating in a double-null configuration can reduce the peak heat flux by nearly ½, and increasing the triangularity and exhausting at smaller major radius actually reduces the peak heat flux through a combination of flux expansion and increased radiation. Access to, diagnosis of, and mitigation

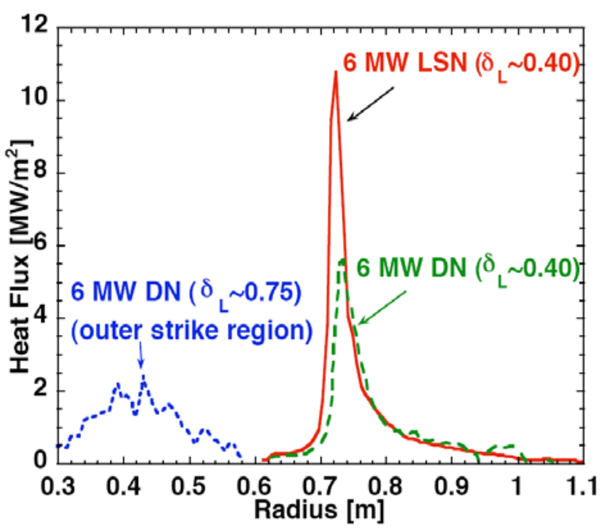


Figure 1.2.4.1: Divertor heat flux profiles for low triangularity lower-single null (LSN) (red), double-null (DN) (green), and high triangularity DN plasmas at fixed NBI heating power = 6MW.

of these very high heat fluxes is a unique capability of NSTX and opens the possibility of using the ST as a test-bed for developing heat flux solutions not only for the ST but for all DEMO configurations.

NSTX is also unique in the world program for investigating the impact of lithium plasma facing components (PFCs) on diverted H-mode plasmas. As discussed in Chapter 5, lithium has been shown to offer several potential benefits in NSTX plasmas including particle pumping, enhanced thermal confinement, and the suppression of edge localized modes (ELMs). ELM reduction and/or suppression is a vital research area in the world program, since large ELMs must be avoided in next-step devices such as ITER to avoid ablating and/or melting divertor PFCs. As shown in Figure 1.2.4.2, unlike some tokamak experiments which have demonstrated ELM suppression (DIII-D) and/or reduction (JET) using resonant magnetic perturbations (RMPs) at the plasma edge, NSTX experiments have thus far exhibited ELM *destabilization* and/or ELM enlargement/enhancement with the application of RMP fields. Understanding of the causes of these differences is required to confidently extrapolate RMP effects to ITER. ELM destabilization by RMP is presently being exploited in NSTX to reduce impurity accumulation in otherwise ELM-free plasmas (made ELM-free using lithium wall coatings) by controllably applying RMP pulses to induce ELMs and expel impurities. As shown in Figure 1.2.4.2, preliminary tests have already demonstrated a factor of two reduction in radiated power with this pulsed-RMP technique,

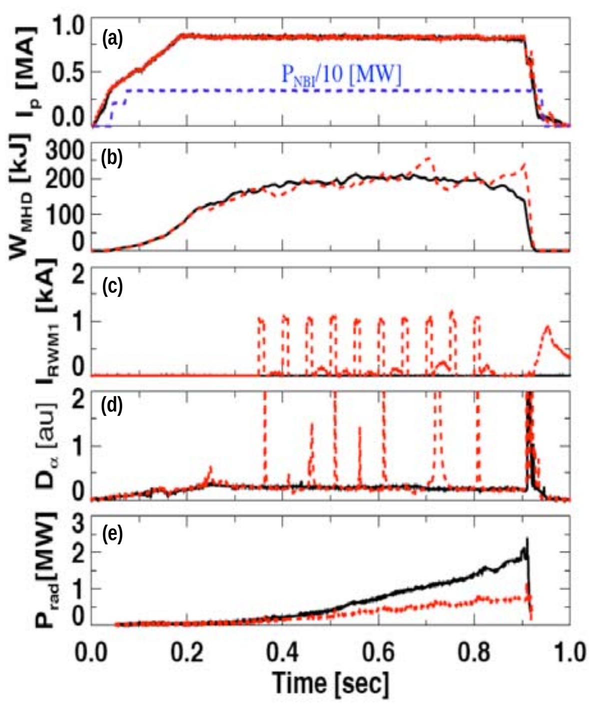


Figure 1.2.4.2: (a) Plasma current and NBI power, (b) plasma stored energy, (c) RMP coil current, (d) divertor D_α light, and (e) radiated power in an ELM-free discharge (black) and with Type-I ELMs triggered by RMP fields (red).

and further optimization of ELM size and frequency is possible. The unique plasma boundary regimes accessible by NSTX are enabling improved understanding of the complex interplay between wall conditions and materials, pedestal stability and transport, 3D edge magnetic fields, and power and particle exhaust to the divertor.

1.2.5 Plasma Formation and Sustainment:

In present ST devices, the compact geometry of the ST leaves little room for the central solenoid normally used for inductive current drive in a tokamak. In future nuclear-capable ST devices, in order to maintain low aspect ratio, the shielding of the central column must be minimized. However, with this level of shielding, the insulation of multi-turn coils is likely damaged by fusion neutrons, and future ST-based nuclear facilities are therefore envisioned to operate with little or no central solenoid. As a result, techniques for plasma current start-up, ramp-up, and sustainment must be developed for next-step STs and would also greatly benefit tokamak reactors. Because these techniques are ultimately essential to the ST concept, ST research takes the leading role in the development of non-inductive startup and ramp-up techniques for STs and tokamaks.

On NSTX, the Coaxial Helicity Injection (CHI) plasma start-up technique has generated up to 20% of the nominal high-performance operating current, and CHI has recently been coupled to inductive operation in NSTX. As shown in Figure 1.2.5.1, up to 160kA of closed-flux current has been produced in NSTX using CHI, and as discussed in Chapter 6, a record high current multiplication factor of 60 was achieved in these plasmas. NSTX is presently the only device in the world program studying CHI, and these experiments clearly indicate that high closed-flux plasma currents can be produced. However, additional effort is needed to further heat these plasmas to achieve impurity burn-through and to raise the electron temperature so that the HHFW can be absorbed to enable plasma current ramp-up using current over-drive.

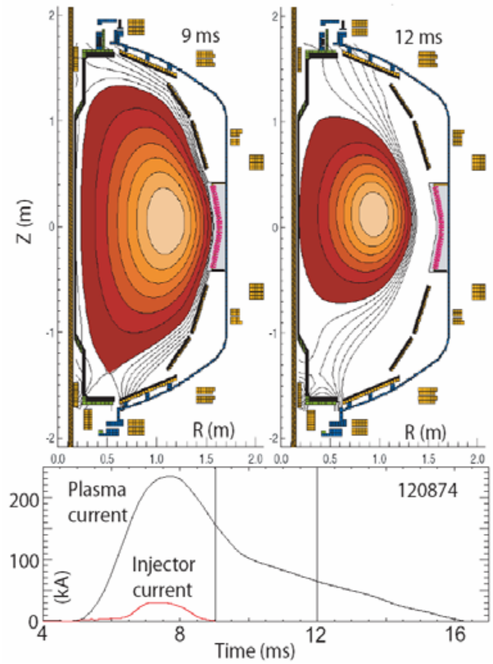


Figure 1.2.5.1: (Top) Equilibrium reconstructions show the shape evolution of the CHI-produced plasma in response to decaying current. (Bottom) Up to 160kA of closed-flux plasma current has been produced by CHI.

With regard to sustained high-performance scenarios, NSTX has sustained up to 70% of the plasma current at high plasma performance with a combination of pressure-gradient-driven "bootstrap" current and neutral beam injection current drive (NBI-CD). As discussed in Chapter 6, this high fraction of non-inductive current drive is achieved at high beta (β_T = 15-20% and β_N = 5.5-6) and simultaneously has high pressure-gradient-driven current fraction ($f_{\nabla p}$ = 55%) comparable to that assumed for ST-CTF. Thus, based on NSTX results, fully non-inductive current drive appears within reach for the ST. The largest gap in achieving completely solenoid-free ST operation is arguably the plasma-current ramp-up phase between plasma start-up and plasma sustainment. As discussed in Chapter 6, HHFW heating has driven up to 85% of the plasma current at low plasma current with bootstrap (BS) current in high-confinement-mode (H-mode) plasmas in NSTX – the highest bootstrap fraction achieved in any ST to-date. However, additional current-drive from either BS or other means (such as HHFW-CD or NBI-CD) is needed to achieve plasma current ramp-up to high plasma current.

Contributions to tokamak physics and ITER

1.3.1 - ITPA: The unique parameter regimes accessible by NSTX described in Section 1.2 provide new insight into the underlying plasma physics relevant to tokamaks of all aspect ratios and for future burning plasma experiments. As a result, NSTX data is increasingly being sought out to clarify aspect ratio, beta, fast-ion, and other dependences in tokamak physics. Since 2005, NSTX has become increasingly engaged in the **I**nternational **T**okamak **P**hysics **A**ctivity (ITPA) and has participated in many joint experiments involving participation by NSTX researchers and/or contributions of experimental data to international databases – for example databases for H-mode confinement (Chapter 3) and disruption current quench-rate (Chapter 2). For the H-mode confinement scaling example, NSTX confinement exhibits a relatively weak confinement degradation with β relative to the ITER 98(y,2) scaling, but is consistent with dedicated JET and DIII-D scans. NSTX experiments have shown that variability in the confinement dependence on β is likely related to differences in plasma shaping and/or ELM type. For the disruption current quench-rate example, the lower aspect ratio of the ST decreases the normalized external inductance and increases the current quench rate. However, when the aspect ratio dependence of the inductance is accounted for, the normalized quench rates of NSTX are comparable to those observed in tokamaks. This indicates that the plasma temperature during the quench is weakly dependent on aspect ratio and is consistent with processes in which impurity radiation dominates the dissipation of plasma inductive energy during the current-quench.

Participation in ITPA activities also benefits the ST by aiding in the understanding of the impact of higher beta and reduced aspect ratio on a variety of physics issues for next-step ST design. As of June 2008, NSTX was actively involved in 17 joint ITPA experiments and is contributing to a total of 24. As shown in Table 1.3.1, the contributions of NSTX to ITPA research extend across all topical science areas described in Section 1.2, and selected ITPA-related results are also discussed in Chapters 2-6.

Boundary Physics
- PEP-6 Pedestal structure and ELM stability in DN
- PEP-9 NSTX/MAST/DIII-D pedestal similarity
- PEP-16 C-MOD/NSTX/MAST small ELM regime comparison
- DSOL-15 Inter-machine comparison of blob characteristics
- DSOL-17 Cross-machine comparison of pulse-by-pulse deposition

Macroscopic stability
- MDC-2 Joint experiments on resistive wall mode physics
- MDC-3 Joint experiments on neoclassical tearing modes including error field effects
- MDC-12 Non-resonant magnetic braking
- MDC-13: NTM stability at low rotation

Transport and Turbulence
- CDB-2 Confinement scaling in ELMy H-modes: β degradation
- CDB-6 Improving the condition of global ELMy H-mode and pedestal databases: Low A
- CDB-9 Density profiles at low collisionality
- TP-6.3 NBI-driven momentum transport study
- TP-9 H-mode aspect ratio comparison

Wave Particle Interactions
- MDC-11 Fast ion losses and redistribution from localized Alfvén Eigenmodes

Advanced Scenarios and Control
- SSO-2.2 MHD in hybrid scenarios and effects on q-profile
- MDC-14: Vertical Stability Physics and Performance Limits in Tokamaks with Highly Elongated Plasmas

Table 1.4.1 - International Tokamak Physics Activity (ITPA) joint experiments involving participation by NSTX researchers and/or contributions of NSTX experimental data as of June 2008.

1.3.2 – ITER Design: During 2007, several ITER design issues were identified to be of critical importance because they potentially impact the ability of ITER to achieve its stated missions of achieving Q=10 at 500MW sustained for 400s and fully non-inductive operation at Q=5. As described in Section 1.2.4, a critical issue for all ITER scenarios is the mitigation of ELMs. Several dedicated RMP ELM control experiments were conducted in NSTX for ITER in FY2008, and a subset of these results are described in Chapter 5. A key issue in understanding RMP ELM mitigation is the plasma response to the applied field. As described in Chapter 2, the Ideal Perturbed Equilibrium Code (IPEC) has been developed to better model tokamak plasma response to 3D fields. This code was initially developed to better understand locked-mode threshold scalings and error-field effects on NSTX, and has since been applied to DIII-D and ITER as well. Of particular importance to ELM mitigation using RMP is the avoidance of the

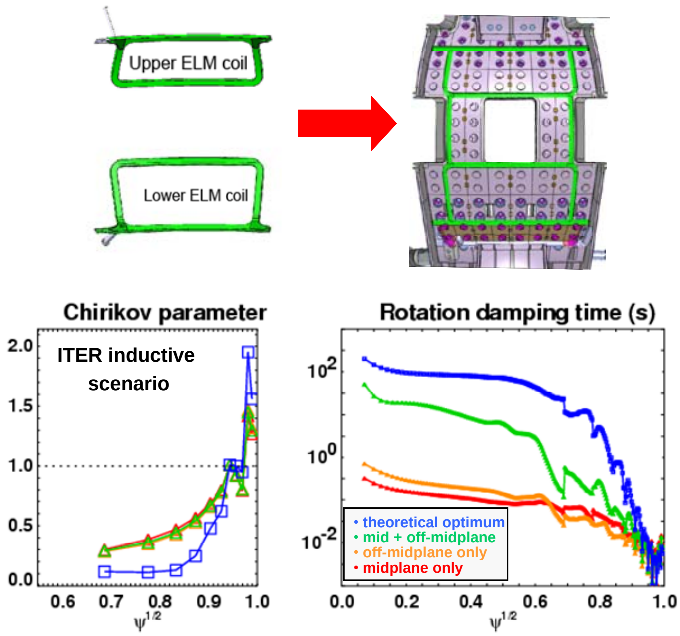


Figure 1.3.2.1: (Top) Non-axisymmetric control coils under consideration for ITER shown in green, (bottom left) Chirikov parameter vs. minor radius for the optimum coil current distribution computed by IPEC, and (bottom right) NTV rotation damping time computed by IPEC.

excitation of core resonant radial field perturbations (to minimize island excitation) and the minimization of both resonant and non-resonant rotation damping. By linking IPEC to Neoclassical Toroidal Viscosity (NTV) calculations (see Chapter 2), the optimal applied external field can computed to ergodize the edge (i.e. satisfy the Chirikov overlap parameter in the H-mode pedestal) while minimizing core island formation and NTV flow damping. As shown in Figure 1.3.2.1, the ELM control coils under consideration for ITER are computed to be capable of ergodizing the plasma edge while providing Chirikov << 1 in the plasma core. This figure also shows that the combination of midplane and off-midplane coils is up to 2 orders magnitudes more effective at minimizing rotation damping relative to mid-plane or off-mid-plane coils alone. The combination of the mid-plane and off-midplane coils approaches the theoretical optimum (shown in blue) for coil placement (i.e. normal field distribution on the plasma boundary) unconstrained by engineering considerations.

The coils shown in Figure 1.3.2.1 also appear attractive for fast vertical control and resistive wall mode control applications. A key issue for ITER is the maximum tolerable displacement of the plasma vertical position (typically caused by an MHD-induced perturbation to l_i and/or β_P) that is allowable before the vertical position control system fails to recover and a vertical displacement event (VDE) occurs leading a plasma current disruption. Dedicated experiments on NSTX with

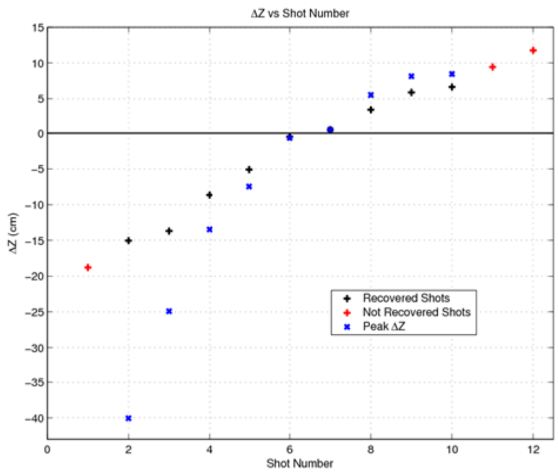


Figure 1.3.2.2: Recovered (black) and non-recovered (red) shots vs. vertical position displacement ΔZ from vertical control experiments in NSTX for ITER.

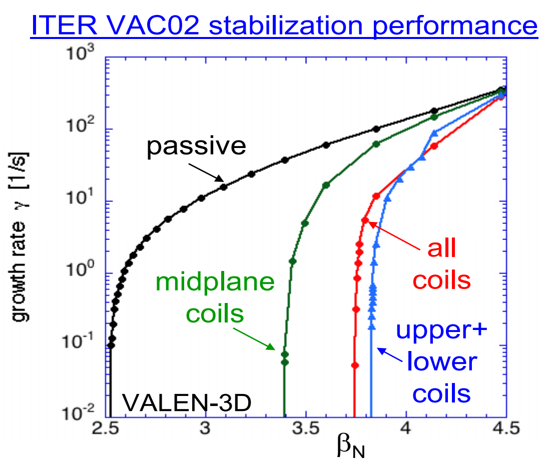


Figure 1.3.2.3: VALEN calculations of n=1 RWM control capabilities for different coil combinations for the proposed ITER 3D coils shown in Figure 1.3.2.1.

plasma elongation and internal inductance similar to ITER values have been performed and observed a maximum recoverable displacement $\Delta Z_{MAX}/a < 10\%$. This value is consistent with results at higher aspect ratio A ≈ 3 and further confirms the potential inadequacy of the original ITER vertical control system for providing robust control of large vertical excursions. This finding further motivates the usage of the coils shown in Figure 1.3.2.1 for more robust vertical control. A further potentially important finding of NSTX is shown in Figure 1.3.2.2 which indicates that the maximum recoverable displacement depends on the direction of displacement and is likely the result of up/down asymmetries in the vessel conducting structure of NSTX (up/down asymmetries in vessel conducting structure are also present in ITER).

Finally, in addition to sufficient ELM and vertical control, ITER will also require n=1 RWM control to operate stably in the proposed fully non-inductive scenario with Q=5 (ITER scenario 4). As shown in Figure 1.3.2.3, VALEN calculations (validated with NSTX RWM control data as discussed in Chapter 2) indicate that the internal 3D coils proposed for ITER are sufficient to stabilize the n=1 RWM up to β_N = 3.7. The operating $\beta_N \approx 3$ is therefore less than 50% of the separation between the no-wall and RWM control limits and should provide sufficient margin for robust RWM control in ITER advanced scenarios.

1.4 Fusion Energy Science Applications of the ST

1.4.1 Overview

Beyond the contributions of the NSTX to tokamak physics and ITER development, the ST is also attractive for a range of fusion energy science applications as shown previously in Figure 1.1.2. The ST offers high normalized pressure (i.e. high β), a very compact geometry resulting from low-A, and simplified normally conducting magnets with significantly reduced coil stress on the outboard low-field side. As shown in ARIES and CTF design studies, the ST also offers the possibility of a single-turn demountable central conducting rod for the toroidal field coil and ease of maintenance, accessibility, and change-out of components.

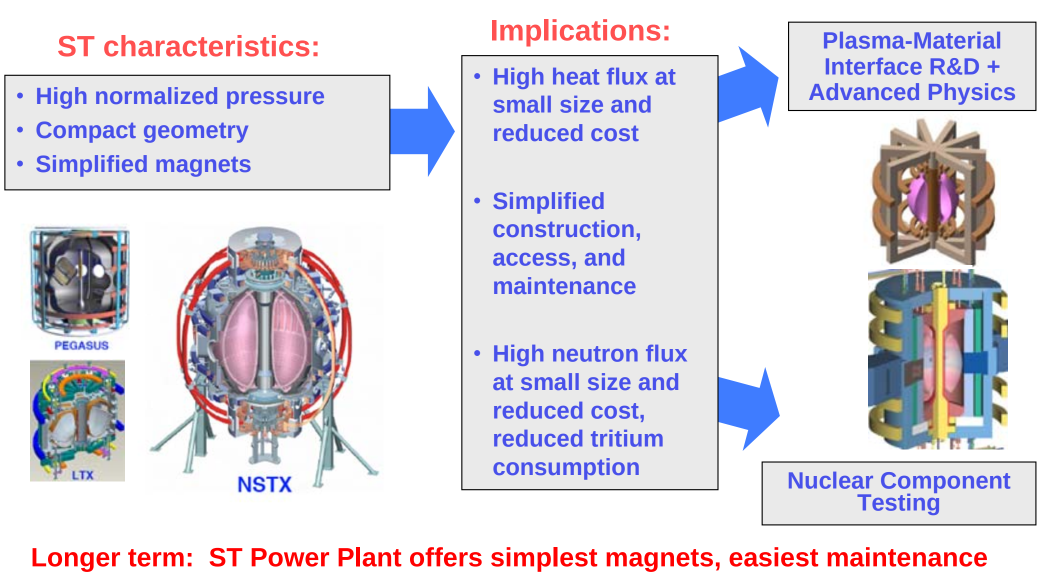


Figure 1.4.1.1: The characteristics of the ST relevant to fusion energy, their implications, and resulting near-term and long-term applications of the ST.

As shown graphically in Figure 1.4.1.1, the intrinsic ST characteristics described above have several implications. First, the compactness of the ST results in very high heat fluxes to the Plasma Material Interfaces (PMI) – namely the divertor. High plasma exhaust heat flux is common to all fusion DEMO concepts based on the ST, tokamak, or compact stellarator, but the issue is most severe for the ST. Empirically, the measured power scrape-off width (projected from the divertor back to the outboard midplane) does not vary strongly with machine size, the divertor heat-flux challenge can be approximately parameterized by the ratio P_{heat}/R where R is the average plasma major radius. The ratio P_{heat}/S (where S is the wall surface area) is also

Device	R (m)	a (m)	P_{heat} (MW)	P_{heat}/R (MW/m)	P_{heat}/S (MW/m^2)	Pulse (sec)	I_p (MA)	Species	Comments
Planned Long-Pulse Experiments									
EAST	1.70	0.40	24	14	0.55	1000	1.0	H (D)	Upgrade capability
JT-60SA	3.01	1.14	41	14	0.21	100	3.0	D	JA-EU Collaboration
KSTAR	1.80	0.50	29	16	0.52	300	2.0	H (D)	Upgrade Capability
LHD	3.90	0.60	10	3	0.11	10,000	–	H	Upgrade capability
SST-1	1.10	0.20	3	3	0.23	1000	0.2	H (D)	Initial heating
W7-X	5.50	0.53	10	2	0.09	1800	–	H	30MW for 10sec
NHTX	**1.00**	**0.55**	**50**	**50**	**1.13**	**1000**	**3.5**	**D (DT)**	**Initial heating**
ITER	6.20	2.00	150	24	0.21	400-3000	15.0	DT	Not for divertor testing
Component Test Facility Designs									
CTF (A=1.5)	**1.20**	**0.80**	**58**	**48**	**0.64**	**Weeks**	**12.3**	**DT**	**2 MW/m^2 neutron flux**
FDF (A=3.5)	2.49	0.71	108	43	1.61	Weeks	7.0	DT	2 MW/m^2 neutron flux
Demonstration Power Plant Designs									
ARIES-RS	5.52	1.38	514	93	1.23	Months	11.3	DT	US Advanced Tokamak
ARIES-AT	5.20	1.30	387	74	0.85	Months	12.8	DT	US Advanced Technology
ARIES-ST	**3.20**	**2.00**	**624**	**195**	**0.99**	**Months**	**29.0**	**DT**	**US Spherical Torus**
ARIES-CS	7.75	1.70	471	61	0.91	Months	3.2	DT	US Compact Stellarator
ITER-like	6.20	2.00	600	97	0.84	Months	15.0	DT	ITER @ higher power, Q
EU A	9.55	3.18	1246	130	0.74	Months	30.0	DT	EU "modest extrapolation"
EU B	8.60	2.87	990	115	0.73	Months	28.0	DT	EU
EU C	7.50	2.50	794	106	0.71	Months	20.1	DT	EU
EU D	6.10	2.03	577	95	0.78	Months	14.1	DT	EU Advanced
SlimCS	5.50	2.12	650	118	0.90	Months	16.7	DT	JA

Table 1.4.1.1: Comparison of the accessible normalized heat-flux (highlighted in yellow) and other parameters of planned long-pulse experiments, Component Test Facility (CTF) design points, and demonstration power plant (DEMO) designs. ST concepts: NHTX, ST-CTF, and ARIES-ST are highlighted in green.

important for parameterizing the expected wall heat loading challenge. As shown in Table 1.4.1.1, NHTX could access normalized heat fluxes twice those of ITER with values comparable to those of CTF designs and with a P/S value similar to all DEMO design concepts. With magnetic flux compression, NHTX should also be able to access heat flux values of tokamak-based reactor designs, but accessing and mitigating ST reactor-level heat fluxes would be more challenging. The heat flux challenge must be resolved to achieve a successful ST-DEMO, and PMI solutions developed in ST research benefit not only the ST path but also all other DEMO concepts shown in Table 1.4.1.1.

The compactness of the ST can also potentially create very high neutron wall loading (1-2MW/m^2) in a device not much larger (20-50% in major radius) than present mega-ampere-class STs. Using the ST as a volume neutron source (VNS) to test nuclear components in a high fusion neutron irradiation environment with neutron fluences 10× those of ITER and approaching DEMO-level fluences could reduce the usage of tritium and also the risks of proceeding directly from ITER to DEMO. For both PMI and VNS uses of the ST, the simplified construction, access, and maintenance are significant potential advantages for rapidly testing new materials, components, and techniques, in an environment of high heat, particle, and neutron flux sustained for long durations.

The 2007 FESAC panel report entitled "Priorities, Gaps and Opportunities: Towards a Long Range Strategic Plan for Magnetic Fusion Energy" identified plasma facing components and materials (i.e. the PMI) as the most challenging gaps to close between ITER and DEMO. The ST appears particularly well suited to address the related highest priority FESAC panel "theme" of "Taming the Plasma Material Interface" (Theme B) which encompasses: plasma-wall interactions, plasma facing components, RF antennae, launching structures and other internal components. The ST also appears particularly well suited to address the FESAC panel theme of "Harnessing Fusion Power" (Theme C) which encompasses: fusion fuel cycle, power extraction, materials science in the fusion environment, safety, reliability, availability, maintainability, and inspectability. As described in section 1.4.2 below, the proposed NHTX device could address Theme B in a non-nuclear environment (operating with D-D and very limited D-T capability) including prototyping of PMI solutions for ST-CTF and testing the PMI applicability of neutron-irradiated materials from the International Fusion Materials Irradiation Facility (IFMIF). ST-CTF could strongly address the neutron irradiation implications of Theme B issues on a much larger scale while also strongly addressing Theme C.

A third theme of "Creating predictable high-performance steady-state plasmas" (Theme A) was identified by FESAC encompassing: measurement, performance integration, validated theory and predictive modeling, control, disruptions, control auxiliary systems, and magnets. Significant research and development is expected to be performed in ITER steady-state scenarios and in the long-pulse superconducting tokamak program utilizing the EAST, KSTAR, and JT-60SA devices. Since the normalized plasma stability requirements of an ST-DEMO are similar to those of an AT-DEMO, the ST can leverage expected future progress in advanced tokamak physics research. However, the confinement (and other) properties of the ST may be sufficiently different from the tokamak that high-performance steady-state plasmas must also be demonstrated in the ST prior to proceeding to an ST-DEMO. Normally conducting magnets in a nuclear environment are the most important and unique technology aspect of the ST with respect to Theme A, and the ST community has identified critical issues for the ST-CTF center post:

1) Materials interface and design to ensure uniform current distribution over large area high current possibly sliding joints and feeders
2) Replaceable vacuum seal located near the TF sliding joints
3) Requirements for remote handled assembly and replacement of such center posts
4) Manufacturability of the TF center post using radiation tolerant normal conductors
5) Center post water cooling capability and management of activated corrosion products

These design issues are also directly relevant to an ST-DEMO, and ST-CTF would therefore contribute strongly to developing the magnet technology for ST-DEMO. More detailed information on the goals and operating parameters for NHTX, ST-CTF, and an ST reactor are provided below for reference.

1.4.2. Mission, design concept, and operating parameters for NHTX

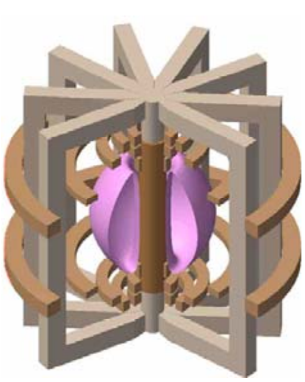
As described in Section 1.4.1, the 2007 FESAC panel identified “Taming the Plasma Material Interface” as the highest priority gap that must be addressed to progress from ITER burning plasmas to an electricity producing DEMO. PMI solutions must also be compatible with sustained high plasma performance to be relevant to DEMO, and this challenge motivates the proposed scientific mission of the National High-power advanced Torus eXperiment (NHTX).

<u>NHTX mission:</u> "Integration of a fusion-relevant plasma-material interface with stable, sustained, high-performance plasma operation"

The NHTX mission aims to integrate the following PMI and plasma performance goals:

1. Access and diagnose DEMO-relevant heat-fluxes
2. Perform rapid testing of new PMI concepts, including comparing solid vs. liquid PFCs and testing novel divertor concepts (X, Super-X, snow-flake divertor, etc.)
3. Perform PMI research at DEMO-relevant $T_{wall} \sim 600°C$
4. Access pulse durations sufficient to achieve plasma-wall equilibration: $\tau_{pulse} \sim 200\text{-}1000s$
5. Develop methods to avoid tritium retention
6. Demonstrate the compatibility of PMI solutions with high plasma performance:
   a. High confinement will tolerable or no ELMs
   b. High beta without disruptions
   c. Steady-state with full non-inductive current drive
7. Test advanced performance regimes (high β_N and f_{BS}) for ST-DEMO and ST-CTF, test I_P start-up/ramp-up techniques for ST-CTF at similar size, toroidal field, and P_{AUX}.

The NHTX device geometry, parameters, and possible plasma and divertor configurations to be studied are shown in Figure 1.4.2.1, and NSTX and future ST divertor and plasma exhaust issues are discussed in much more detail in Chapter 5.



National High-power advanced Torus eXperiment (NHTX)

P_{heat}	50MW
R_0	1m
A	1.8-2
κ	≤ 3
B_T	2T
I_P	3-3.5MA
β_N	4.5
β_T	14%
n_e/n_{GW}	0.4-0.5
f_{BS}	$\approx$ 70%
f_{NICD}	100%
$H_{98Y,2}$	≤ 1.3
E_{NB}	110keV
P/R	50MW/m
Solenoid	½ swing to full I_P

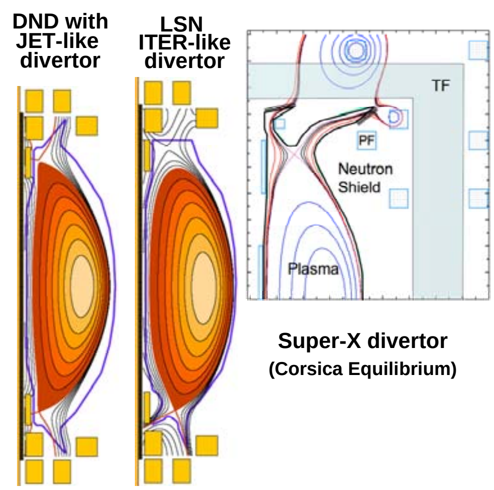


Figure 1.4.2.1: (Left) 3D cut-away of NHTX coils and plasma, (middle) device parameters for the base-line operating scenario, and (right) examples of plasma shapes and divertor configurations that could be accessed in NHTX including a JET-like, ITER-like, and "Super-X" divertor.

1.4.3. Mission, design concept, and operating parameters for ST-CTF

As described in Section 1.4.1, the ST is potentially very well suited to address FESAC panel Theme C, "Harnessing Fusion Power" – a critical gap that must be addressed to progress from ITER burning plasmas to an electricity producing DEMO. The goal of performing component testing with nuclear conditions relevant to DEMO with potentially significantly reduced risk, cost, and tritium consumption as compared to performing the same R&D in DEMO itself motivates the mission of the (ST-based) CTF program, namely:

> ***"Create a lowered-risk, reduced-cost approach to a fusion environment beyond the ITER level, and utilize it to test, discover, innovate, and develop the remaining needed engineering science and technology knowledge base for Demo."***
>
> *(From "Bridging Engineering Science and Technology Gaps From ITER to DEMO" – M. Peng et al., APS-DPP meeting Nov. 12-16, 2007)*

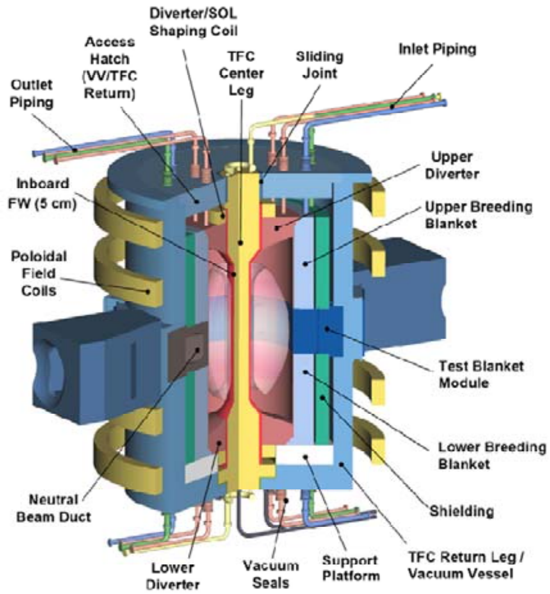
This mission aims to exploit the following potential advantages of the ST for component testing:

1. Compact device and high-β operation
2. Reduced device cost
3. Reduced operating cost ($P_{electric}$)
4. Reduced T consumption
5. Simplified vessel and magnets
6. Fully modularized core components
7. Fully remote assembly and disassembly

The ST-CTF design concept geometry, parameters, and an example of how device modularity could facilitate assembly/disassembly and maintainability are shown in Figure 1.4.3.1.

ST-based Component Test Facility (ST-CTF)

Performance metrics	ITER	Required Conditions	Demo Goals
Continuous operation	~hour	weeks	~months
14-MeV neutron flux on module (MW/m^2)	~0.8	1.0-2.0	~3
Total neutron fluence goal (MW-yr/m^2)	~0.3	6	~6-15
Duty factor goal	~1%	30%	~80%
Tritium self-sufficiency goal (%)	~0	~100	≥100



W_L [MW/m^2]	0.1	1.0	2.0
R0 [m]		1.20	
A		1.50	
kappa		3.07	
qcyl	4.6	3.7	3.0
Bt [T]	1.13	2.18	
Ip [MA]	3.4	8.2	10.1
Beta_N	3.8		5.9
Beta_T	0.14	0.18	0.28
n_e [10^{20}/m^3]	0.43	1.05	1.28
f_{BS}	0.58	0.49	0.50
T_{avgi} [keV]	5.4	10.3	13.3
T_{avge} [keV]	3.1	6.8	8.1
HH98		1.5	
Q	0.50	2.5	3.5
$P_{aux\text{-}CD}$ [MW]	15	31	43
E_{NB} [keV]	100	239	294
P_{Fusion} [MW]	7.5	75	150
T M height [m]		1.64	
T M area [m^2]		14	
Blanket A [m^2]		66	
$F_{n\text{-capture}}$		0.76	
P/R [MW/m]	**14**	**38**	**61**
Solenoid	Small iron core or mineral insulated transformer for startup		

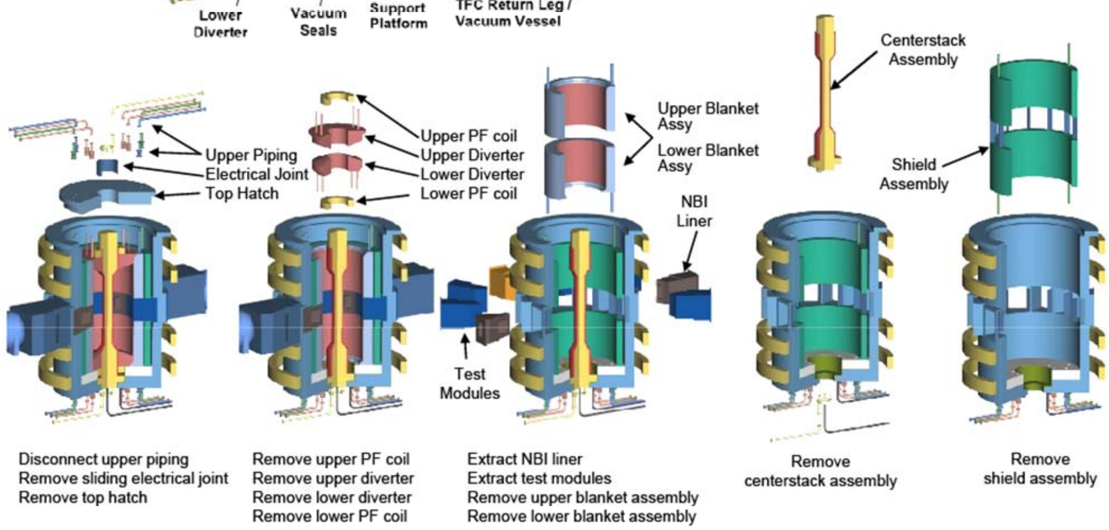
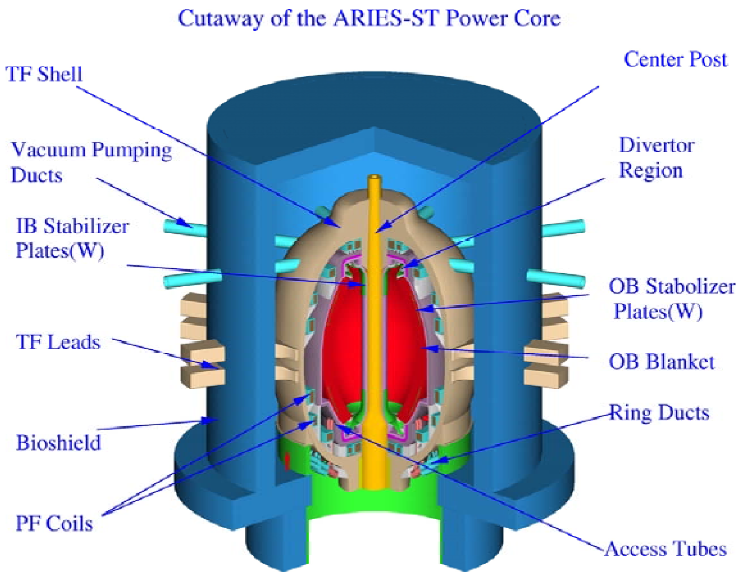


Figure 1.4.3.1: (Top left) Required performance and nuclear conditions for a CTF mission and for DEMO goals, (middle left) ST-CTF design concept, (right top) table of parameters for ST-CTF, and (bottom) example of how an ST CTF device with modular design could facilitate assembly/disassembly and maintenance.

1.4.4. Design concept and operating parameters for an ST reactor



A	1.6
R_0 [m]	3.2
a [m]	2
B_{T0} [T]	2.1
I_P [MA]	28
κ_X	3.7
δ_X	0.67
β_T [%]	50
β_N	7.5
f_{BS} [%]	99
$H_{98Y,2}$	1.33
n_e / n_{GW}	0.8
P/R [MW/m]	200
W_L [MW/m^2]	4.1
P_{fusion} [MW]	2980
P_{recirc} [MW]	520
P_{TF} [MW]	325
$P_{electric}$ [MW]	1000

Figure 1.4.4.1: (Left) ARIES-ST design concept and (right) table ARIES-ST parameters.

The design requirements to optimize a tokamak, ST, or stellarator-based power reactor have been studied extensively by the ARIES reactor studies group (http://www-ferp.ucsd.edu/aries/). The design concept and plasma and fusion parameters for an ST-based reactor (ARIES-ST) are shown in Figure 1.4.4.1. As for ST-CTF, there is a single-turn central TF bundle and the vacuum vessel provides the current return for the TF current. Similar to advanced tokamak reactor concepts, ARIES-ST relies heavily on operation far above the no-wall stability limit (i.e. β_N = 7.5 = 90% of ideal-wall β_N limit = 8.3), nearly all the plasma current is provided by the bootstrap current, and the elongation is maximized (κ_X up to 3.7 in ARIES-ST) to maximize the toroidal beta at high bootstrap fraction. One issue for the ST-based reactor is the relatively high resistive power dissipation (≈ 300MW) in the TF coil which reduces the overall plant efficiency. However, the cost of electricity (COE) is driven primarily by the plant capital cost. The COE of ARIES-ST (78 mill per kW_e h in 1992 dollars) is very close to that computed for superconducting tokamaks such as ARIES-RS (76 mill/kW_eh) using design assumptions from the mid-to-late 1990's. For the more recent ARIES-AT studies, improvements in technology including higher thermal conversion efficiency blankets (59% thermal conversion efficiency vs. 45% assumed for ARIES-ST) and other cost reductions account for a significant reduction of COE to 50 mill/kW$_e$h. The low COE of ARIES-AT is attractive because it is competitive with present electricity costs adjusted for inflation. Preliminary estimates indicate that if ARIES-ST could utilize higher efficiency blankets and operate near ~1.5GW$_e$, COE values much closer to those of ARIES-AT values are achievable. However, more detailed design studies are needed to assess if such design modifications and COE reductions are possible for ST-based reactors.

1.5 Gaps Between Present and Future STs

All of the future ST concepts described in Section 1.4 require advances in plasma performance and understanding to achieve their stated missions. Present MA-class ST experiments (NSTX and MAST) have progressed significantly in the last decade – particularly in sustaining high normalized performance as described in Chapter 6. However, as shown quantitatively in Table 1.5.1, several gaps between present and future ST performance remain.

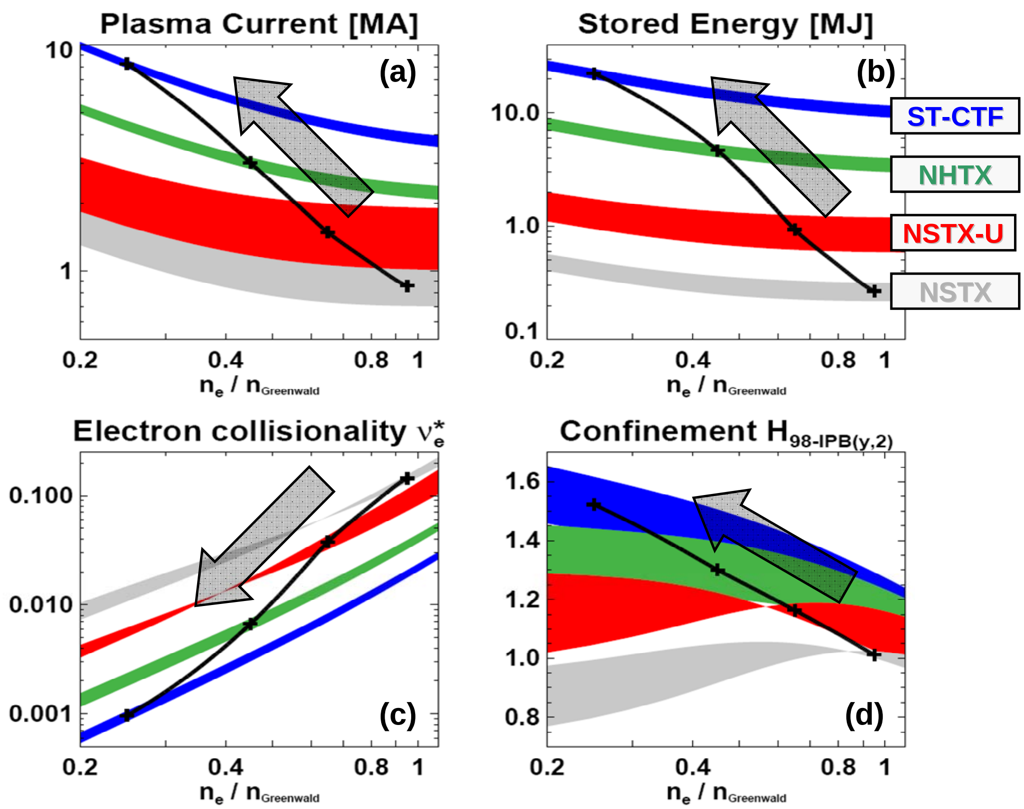
For the NHTX and ST-CTF device concepts, NSTX already achieves bootstrap fraction and elongation values comparable those assumed, and achieves β_N values above those assumed. However, significant increases in NBI current drive are needed to achieve fully non-inductive current drive. Much of this increased current drive can be achieved via reduced electron collisionality as a result of increased electron temperature and/or reduced normalized density. In order to maximize NBI current drive, ST-CTF is projected to operate at $n_e/n_{Greenwald}$ = 0.25-0.3 which is approximately 1/3 of the normalized density presently achieved in H-mode (without active pumping). ST-CTF is also projected to require up to 50% higher H-factor $H_{98\text{-}IPB(y,2)}$ than presently achieved in order to minimize device size and power requirements. While NHTX is designed to include a solenoid for ramp-up to I_P flat-top, ST-CTF is designed to have minimal (or no) central solenoid, so plasma current start-up and ramp-up to high plasma current (up to

Present high β_N & f_{NICD}	NSTX	NSTX-U	NHTX	ST-CTF	ARIES-ST
A	1.53	1.65	1.8	1.5	1.6
κ	2.6-2.7	2.6-2.8	2.8	3.1	3.7
β_T [%]	14	10-16	12-16	18-28	50
β_N [%-mT/MA]	5.7	5.1-6.2	4.5-5	4-6	7.5
f_{NICD}	0.65	1.0	1.0	1.0	1.0
$f_{BS+PS+Diam}$	0.54	0.6-0.8	0.65-0.75	0.45-0.5	0.99
f_{NBI-CD}	0.11	0.2-0.4	0.25-0.35	0.5-0.55	0.01
$f_{Greenwald}$	0.8-1.0	0.6-0.8	0.4-0.5	0.25-0.3	0.8
H_{98y2}	1.1	1.15-1.25	1.3	1.5	1.3
Dimensional/Device Parameters:					
Solenoid Capability	Ramp+flat-top	Ramp+flat-top	Ramp to full I_P	No/partial	No
I_P [MA]	0.72	1.0	3-3.5	8-10	28
B_T [T]	0.52	0.75-1.0	2.0	2.5	2.1
R_0 [m]	0.86	0.92	1.0	1.2	3.2
a [m]	0.56	0.56	0.55	0.8	2.0
I_P / aB_{T0} [MA/mT]	2.5	1.8-2.4	2.7-3.2	4-5	6.7

Table 1.5.1: Comparison of key dimensionless and dimensional parameters of present high non-inductive current fraction scenarios in NSTX, and those proposed for NSTX-Upgrade, NHTX, ST-CTF, ARIES-ST.

8MA) are required. Thus, high priority NSTX research goals for 2009-13 include access to reduced density and collisionality, improved thermal confinement and improved understanding of thermal confinement to reliably project to future ST devices, and a demonstration of non-inductive plasma current ramp-up from low (~200kA) to intermediate (~400kA) current values.

For the ST-DEMO device concept, a significant increase in bootstrap current (from $f_{\nabla p} \approx 55\% \rightarrow 100\%$) is required relative to presently achieved values at high β_T. As described in Chapter 6, HHFW-heated plasmas at low current have achieved up to 85% bootstrap fraction at $\beta_N \approx 4$, but with $\beta_T < 4\%$. Increased κ and β_N are required to access higher β_T at high f_{BS}. As described in Chapter 2, NSTX has stably achieved κ=3 and has sustained κ =2.7-2.8. At present, for q_{min} =1.2-1.5, the predicted and experimentally measured ideal-wall limit for the n=1 external kink is $\beta_N \approx 6$. Elevated $q_{min} > 2.2$-2.4 is projected to increase the n=1 ideal-wall limit to $\beta_N \approx 7$-8. Thus, a major NSTX performance goal for 2009-13 is to access high $\beta_N > 6$ plasmas at high κ up to 3 with elevated $q_{min} > 2$ to increase the bootstrap fraction toward ST reactor values.



NSTX	A=1.5	R_0=0.86m	B_{T0}=0.45T	β_N = 5.8	P_{HEAT} = 6MW	P_{NBI} = 6MW	f_{NICD} = 50-70%
NSTX-U	A=1.6	R_0=0.92m	B_{T0}=1.0T	β_N = 5.0	P_{HEAT} = 14MW	P_{NBI} = 10MW	f_{NICD} = 50-100%
NHTX	A=1.8	R_0=1.00m	B_{T0}=2.0T	β_N = 4.5-5	P_{HEAT} = 50MW	P_{NBI} = 30MW	f_{NICD} = 100%
ST-CTF	A=1.5	R_0=1.20m	B_{T0}=2.5T	β_N = 3.5-4	P_{HEAT} = 65MW	P_{NBI} = 30MW	f_{NICD} = 100%

Figure 1.5.1: Progression of (a) plasma current, (b) stored energy, (c) normalized electron collisionality, and (d) H-mode confinement enhancement vs. normalized density for proposed NSTX, NSTX-Upgrade and possible future ST devices. Note that NSTX and NSTX-U operate with a range of non-inductive fractions, whereas NHTX and ST-CTF curves assume fully non-inductive operation.

An ITER-era (20 year) goal under consideration by the U.S. ST community is to develop the ST knowledge-base to enable design and begin construction of an ST-CTF device. As shown in Figure 1.5.1, achieving ST-CTF conditions requires an extrapolation of 1 order of magnitude in current and 2 orders of magnitude in stored energy in a device only 40% larger in major radius than NSTX. In terms of dimensionless parameters, this entails a 2 order of magnitude decrease in collisionality and a seemingly modest H-mode confinement enhancement of 1.5. However, the strong power degradation of H-mode confinement implies that a roughly 3-fold increase in heating power would be required to compensate for an inability to increase H_{98} from 1.0 to 1.5.

1.6 Research Goals and Opportunities in NSTX

1.6.1 Overview

As discussed in Section 1.5, the dimensionless parameter most significantly altered along the ST-path from NSTX to ST-CTF is the normalized electron (and ion) collisionality. As shown in Figure 1.5.1, the first factor of 4 decrease in collisionality along this path can be accessed and diagnosed with upgrades to NSTX (i.e. NSTX-Upgrade = NSTX-U) – in particular a new center-stack to double the magnetic field and plasma current, a liquid lithium divertor (LLD) for density control, a range of advanced profile and turbulence diagnostics, and a 2nd more tangential NBI system for increased heating power, non-inductive current drive efficiency, and current drive flexibility. More detailed timelines for facility and diagnostic upgrades for the next 5 years are provided in Chapter 7. As described in Chapter 3, reductions in collisionality can be achieved by operating at lower density and reduced q, and NSTX-U can in principle achieve the electron collisionality of NHTX to within a factor of 2. As described in Chapters 2-6, reduced collisionality has the potential to impact a wide range of physics issues in ST plasmas including:

- Macroscopic Stability
  - RWM critical rotation and neoclassical viscous torques may increase at lower ν_i
- Transport & Turbulence
  - Underlying instabilities (micro-tearing, TEM, and ETG) scale differently versus ν^*. Micro-tearing modes are expected to be stabilized at reduced collisionality, and reduced collisionality can be stabilizing for dissipative-TEM or de-stabilizing for collisionless-TEM.
  - If $T_e(r)$ is set by a critical ∇T_e, H-mode confinement may be reduced at reduced n_e
- Boundary Physics
  - ELM ΔW_{ped} increases at lower ν_e* - could impact confinement, plasma purity, divertor
  - ELM stability may improve at lower ν_e* due to second-stability access

   - Detachment for heat flux reduction will be more challenging at reduced SOL density
- Wave-Particle Interaction
   - AE avalanches may be more easily triggered at reduced n_e due to increased fast-ion pressure fraction resulting in possible fast-ion redistribution and/or loss
- Current Start-up and Ramp-up
   - NBI-CD and RF-CD efficiency for ramp-up are increased at reduced n_e, increased T_e
- Current Sustainment
   - Steady-state scenarios rely on reduced n_e and increased T_e to increase NBI current drive efficiency to achieve 100% non-inductive current fraction.

While understanding the role of decreased collisionality is clearly important in future ST research, additional research questions and opportunities exist for all NSTX topical science areas. These questions and opportunities are described in much greater detail in Chapters 2-6, but highlights are provided below. In the descriptions below, key scientific questions for the ST (and tokamaks) are described and motivate the proposed major upgrades to NSTX.

1.6.2 Opportunities in Macroscopic Stability Research

As described in Chapter 2, advanced operating scenarios of the ST and tokamak require high normalized plasma pressures, and are therefore susceptible to dangerous large-scale instabilities. Such instabilities can lead to complete loss of confinement of the plasma and damage to the plasma facing components in next-step devices. Stabilizing these modes requires a nearby conducting wall and a combination of plasma rotation and/or active control coils. Experiments on NSTX have already provided data on stabilization physics in a conductor and coil geometry similar to that of ITER, but in a unique regime of rotation, collisionality, and toroidicity parameter space as described in Section 1.2.

NSTX has demonstrated passive stabilization of large-scale modes, and has also demonstrated active feedback control at ITER-relevant low plasma rotation. However, many questions remain regarding the physics of stabilization by plasma rotation and/or active control. In particular, the reduction of plasma collisionality by up to an order of magnitude will provide access to a new collsionality regime (ν-regime vs. present $1/\nu$ regime) in NSTX to fully test the theory of Neoclassical Toroidal Viscosity (NTV) rotation damping in conditions closer to those of next-step STs and ITER while also clarifying the important effects of tokamak plasma response to 3D fields. Access to reduced collisionality will enable the determination of the relative importance of collisional versus collisionless/kinetic dissipation in resistive wall mode (RWM) stabilization for STs and for ITER. Further, the ability to vary the magnetic field and current over a large factor will clarify the relative roles of Alfven and ion thermal effects in RWM rotational stabilization. By enabling up to a factor of three variation in magnetic field and current and factor of ten variation in collisionality, the major upgrades of NSTX will provide unique and definitive data on large-scale mode instability thresholds and the physics of rotation drag by 3D fields. This unique NSTX data coupled with advanced linear and non-linear stability and feedback control codes will greatly aid the development of a predictive capability for sustaining very high plasma pressures.

Further, there is the opportunity to assess the stability of advanced ST operating regimes in NSTX. As described in Chapter 6, NSTX-U (higher TF and 2nd NBI) is predicted to be capable of sustaining fully non-inductively-driven plasmas with $q_{min} > 2$. Such scenarios are predicted to have ideal-wall β_N limits of 7-8, and these high β_N states are particularly relevant to ST-DEMO. In such plasmas, multiple RWMs of different toroidal mode number may be unstable, and the proposed Non-axisymmetric Control Coils (NCC) may be required to stabilize very high β_N plasmas in NSTX in addition to their potential use for controlling ELMs. Demonstrating fully-non-inductive and sustained operation at high-β_N in NSTX-U and NHTX would also improve the prospects for increased neutron flux (W_L = 1MW/m^2 $\rightarrow$ 2MW/m^2) and fluence in ST-CTF.

1.6.3 Opportunities in Transport and Turbulence Research

As described in Chapter 3, in present spherical torus experiments, ion transport in neutral-beam heated H-modes is commonly measured to be near neoclassical levels – consistent with ion turbulence being suppressed by large shear in the plasma flow. In contrast, electron energy transport is observed to be anomalous, and commonly dominates the overall energy loss. In the ST, electron and ion transport scale differently with magnetic field and plasma current than they do in a conventional tokamak. This difference provides a unique opportunity to understand the underlying physics of electron transport. NSTX data exhibits a strong (nearly linear) scaling of electron confinement with magnetic field at larger minor radius in regions where ∇T_e is largest. This is also the region in which the measured high-k fluctuations have turbulence characteristics consistent with ETG and correlate most strongly with the measured electron thermal diffusivity.

In NSTX, at least four instabilities are potentially responsible for anomalous electron transport: micro-tearing modes (electromagnetic), collisionless trapped electron modes – CTEM (electrostatic), electron temperature gradient modes – ETG (electrostatic), and Global Alfven Eigenmodes – GAE (Alfvenic). Micro-tearing, ETG, and GAE modes have been correlated in a preliminary way with anomalous electron transport in NSTX. Multiple instabilities may be present simultaneously, and isolating the effects of individual instabilities can be difficult. Higher magnetic field and current will provide access to much lower collisionality allowing suppression of micro-tearing modes. Higher magnetic field will also provide access to reduced fast-ion instability drive and enable the reduction (possibly suppression) of GAE modes. Thus, access to higher magnetic field will enable the ability to control the onset of electromagnetic and Alfvenic modes and to separate the impact of these modes from electrostatic modes.

As described in Chapter 3, higher field and current will also reduce neoclassical ion transport by at least a factor of 3 (due to smaller orbit sizes and reduced collisionality) thereby enabling investigations of the extent to which shear-stabilization will suppress anomalous ion transport in future ST devices. This capability will also provide new insight into the underlying causes of anomalous momentum transport (most likely ITG and/or CTEM) and the flow-shear suppression of ion turbulence in the ST. Reduced/suppressed ion turbulence is especially important for achieving a "hot-ion" H-mode regime for high fusion gain in ST-CTF.

Overall, the planned major upgrades of NSTX will greatly enhance the ability to isolate the roles of different micro-instabilities in anomalous electron and ion transport by doubling the achievable toroidal magnetic field and thereby increasing the range of magnetic field variation from a factor of 1.5 to a factor of 3. Importantly, the minimum collisionality made accessible by the major upgrades will approach (to within a factor of two) the collisionality values expected of NHTX at comparable Greenwald density fraction. Existing high-k diagnostics and planned low-k diagnostics (BES) will provide the ability to distinguish between the electrostatic CTEM and ETG modes, and BES and high-k may also be capable of measuring GAE fluctuations. Understanding the physics of electron and ion thermal transport is key to reliably predicting the performance of an ST-CTF.

1.6.4 Opportunities in Energetic Particle Research

As described in Chapter 4, an area of major importance to STs and ITER is the confinement of both injected energetic particles and fusion-produced α-particles. At present, while the modes responsible for fast-ion transport have largely been identified, there is no quantitative predictive capability for the transport of fast ions by fast-ion-driven instabilities. Because of its low magnetic field and higher fraction of fast-particle pressure, an ST can be unstable to a variety of instabilities driven by energetic particles. These instabilities can lead to significant redistribution or loss of fast ions from the plasma. This results in degraded self-heating of the plasma and increased flux of fast particles to localized regions of the plasma facing components. Next-step ST devices are expected to rely heavily on auxiliary heating and current drive from injected fast ions, so understanding and minimizing (possibly even controlling) the redistribution and/or loss of fast ions by instabilities is a very important issue. Further, experiments in an ST can simulate and diagnose the complex multimode interactions expected in ITER and next-step devices more closely than can experiments in other configurations.

The fast-ion instabilities thought to influence fast-ion transport most strongly are high frequency modes existing at a fraction of the fundamental cyclotron frequency (sub-cyclotron), and lower frequency modes near the Alfven frequency. Higher magnetic field will decrease the ratio of fast ion speed to Alfven speed (v_{fast}/v_{Alfven}) and is expected to be stabilizing for both the Compressional Alfven Eigenmode (CAE) and Global Alfven Eigenmode (GAE) sub-cyclotron modes. In contrast, the lower frequency Toroidal Alfven Eigenmode (TAE) and Reverse Shear Alfven Eigenmode (RSAE) are expected to remain unstable at higher magnetic field, primarily because the velocity space instability drive increases as v_{fast}/v_{Alfven} decreases toward the most unstable value of ≈ 2. This destabilizing effect is expected to dominate the overall stability provided the fast-ion beta is not strongly reduced by increased magnetic field. Further, as the magnetic field is increased, the fast-ion orbit size is reduced by the same factor and moves closer to the regimes of next-step STs and ITER. Both the number of modes and the most unstable mode number are expected to increase linearly with magnetic field strength, and this will enable studies of avalanche physics in a regime where quasi-linear theory is expected to be more applicable, is more readily compared to non-linear kinetic MHD simulations, and is more similar to ITER conditions.

Overall, access to higher magnetic field in NSTX will enable control of the onset of high frequency Alfven Eigenmodes and thereby enable a determination of the relative importance of low versus high frequency modes on fast-ion transport, and test the quasi-linear treatment of fast-ion transport. The doubling of the magnetic field strength in an upgraded NSTX will greatly enhance the ability to isolate and understand the roles of different Alfvenic modes on the transport of energetic ions in fusion plasmas.

In addition to the strong effects of magnetic field, the pitch angle distribution of the fast particles is an important determinant of low-frequency *AE stability. More tangential injection of fast ions is expected to be destabilizing to TAE modes, and this increased drive is expected to lead to stronger bursting/chirping mode behavior and to increased fast-ion transport. The combination of increased toroidal field and more tangential injection will dramatically widen the parameter space over which the transition from single-mode to multi-mode interactions responsible for increased fast-ion transport could be studied and understood. The planned NSTX upgrades will provide access to this widened parameter space, and in combination with advanced non-linear simulations will greatly aid the development of a predictive capability for the transport of fast ions by instabilities excited by fast ions in future STs and in ITER.

1.6.5 Opportunities in Boundary Physics Research

As described in Chapter 5, the performance of the plasma-material interface is a critical issue for future ST devices, for ITER, and especially DEMO, as all will operate with higher heat fluxes and longer pulse lengths than existing tokamak facilities. The situation at the plasma-material interface is exacerbated by an instability known as an Edge Localized Mode (ELM), which can lead to very high transient heat loads and can damage the plasma facing components. Predictive understanding and control of both the steady-state and transient heat fluxes are essential. As highlighted in Section 1.4.1, because of the compact geometry of the ST, future ST devices are expected to operate with heat fluxes beyond those of ITER and approach (or reach) DEMO-level edge conditions. Thus, NSTX and next-step ST devices provide excellent environments to understand these plasma-material interactions and to develop the heat-flux mitigation techniques needed for magnetic fusion reactors.

A fundamental issue in tokamak and ST boundary research related to power exhaust is the understanding of transport of heat from the main plasma into the scrape-off-layer (SOL) to the plasma material interface (PMI) – specifically the divertor. The SOL transport properties determine the peak heat flux at the divertor and have major implications for the design of next-step devices. While progress has been made at understanding parallel heat transport in the SOL, at present, even a rudimentary understanding of the scaling of the cross-field thermal transport (and corresponding peak divertor divertor heat-flux) remains elusive in both tokamaks and STs. However, the plasma collisionality and beta (or beta gradient) are thought to potentially play important roles in SOL cross-field transport. Increased magnetic field and power will enable tests of the collisionality and beta dependence of the SOL transport over a much broader parameter range, produce experimental SOL conditions and divertor heat-loads much closer to those of next-step ST devices, and provide unique data for comparison to edge turbulence models of SOL cross-field transport. Recent heat flux scaling studies in NSTX over a limited range of plasma current (700kA to 1MA) indicate a rapid narrowing of the divertor heat-flux width as the plasma current is increased. These scalings need to be developed over a much wider parameter range to improve their utility in projecting to future ST devices of much higher current (3-10MA). For example, using these scalings, very narrow midplane SOL profile widths (few mm) and high divertor heat-fluxes (10's of MW/m^2) beyond ITER values are projected to be possible at higher field and heating power in NSTX-U. The proposed major upgrades of NSTX enabling access to higher magnetic field, current, and power will greatly enhance the ability to understand SOL transport physics and to better project divertor performance to future devices.

Given the seriousness of the divertor heat flux problem, NSTX is actively investigating potential heat flux mitigation techniques, including large expansion of magnetic flux and enhanced divertor radiation. NSTX has shown that lithium coating of plasma facing components can stabilize ELMs and improve confinement. This illustrates the complex role of the edge in affecting core turbulence and confinement. A liquid lithium divertor (LLD) module - a unique capability in the world fusion program - will be tested as a means to extend these benefits to new regimes while also providing particle control. Enhanced pumping capability from the LLD will expand the range of accessible edge collisionality to lower values, which in turn could strongly influence ELM stability. Pumping is required to control the core plasma density to achieve high current drive efficiency to enable fully non-inductive current drive. A long-pulse divertor capability is also needed to support long-pulse plasma operation with stationary plasma conditions. While an LLD does not by itself reduce the peak divertor heat flux, an LLD may enable efficient pumping compatible with high flux expansion since the LLD is expected to provide pumping over its entire surface area. Further, active-cooling, flowing liquid lithium, and evaporative cooling could ultimately provide a means of steady-state heat removal from an LLD. As described in Chapters 5 and 6, an upgraded LLD combined with very high flux expansion (an X-divertor configuration) will be tested to assess the compatibility of an LLD with high heat flux, particle pumping, and sustained high-performance operation. A “Super-X” configuration for further peak divertor heat flux reduction is under consideration as a longer-term upgrade option. Such upgrades will provide the needed characterization and understanding of these potentially revolutionary divertor technologies for use in NHTX and ST-CTF.

1.6.6 Opportunities in Plasma Formation and Sustainment Research

As described in Section 1.2.5 and Chapter 6, techniques for plasma current start-up, ramp-up, and sustainment must be developed for next-step STs and could be beneficial for tokamak reactors. On NSTX, plasma start-up techniques such as Coaxial Helicity Injection (CHI) have generated up to 20% of the nominal high-performance operating current, and CHI has recently been coupled to inductive operation in NSTX. NSTX has also sustained up to 70% of the plasma current at high plasma performance with a combination of pressure-gradient-driven “bootstrap” current and neutral beam injection current drive (NBI-CD). Thus, the largest gap in achieving completely solenoid-free ST operation is arguably the plasma-current ramp-up phase between plasma start-up and plasma sustainment.

As described in Chapters 4 and 6, fast Alfven waves with frequencies at high multiples of the ion-cyclotron frequency (High-Harmonic Fast-Waves - HHFW) have driven up to 85% of the plasma current at low plasma current with bootstrap (BS) current in high-confinement-mode (H-mode) plasmas in NSTX. Increased plasma ramp-up current can be obtained with additional bootstrap current and/or RF current drive. Increased RF heating power could increase the plasma pressure and bootstrap current, and higher toroidal field would increase the magnetic safety factor and lower the normalized beta by a similar factor (at fixed poloidal beta), thereby improving the plasma stability of high bootstrap fraction over-driven plasmas. Higher toroidal field should also largely eliminate parasitic surface wave losses of the HHFW for high phase velocity current-drive phasing, thus enabling RFCD during the ramp-up phase. Quantitatively, the planned doubling of the toroidal field and doubling of the HHFW heating power (made possible by antenna upgrades) are projected to enable demonstration of non-inductive plasma current ramp-up from low (~200-250kA) to intermediate (~400-500kA) current values.

In next-step ST devices, neutral beam injection (NBI) will very likely provide the majority of auxiliary heating power and a significant fraction (30-50%) of the steady-state current drive. Thus, NBI heating and current drive will also very likely be the preferred method of plasma current ramp-up. However, in present ST devices, NBI current ramp-up is difficult to test because neutral-beam injected fast ions are poorly confined at currents only 30% (approximately) below the nominal operating current. Higher plasma current and NBI power are required to test NBI current ramp-up physics. In particular, the complex interplay between the beam-driven current and instabilities commonly triggered by the evolving current and pressure profiles (both thermal and fast-ion) requires much more extensive investigation. The major upgrades of NSTX to higher field, current, and NBI power with more tangential injection will enable assessments of NBI current drive efficiency (and possibly NBI-driven plasma current ramp-up) under conditions similar to the early ramp-up phase of an ST-CTF.

As described in Chapter 6, demonstrating and understanding full non-inductive sustainment is also essential for the ST concept. Efficient off-axis current drive is needed to sustain a fully non-inductive current profile with a minimum q ~ 1.2 to 1.5 to avoid deleterious core MHD activity. More tangential beam injection can increase the current drive efficiency relative to present NBI injection geometries while also driving current off-axis. Higher toroidal field in combination with more tangential injection should enable tests of fully non-inductive scenarios operating far from disruptive limits as envisioned for robust ST-CTF operation. Fully non-inductive scenarios with minimum q > 2 will also become accessible at high plasma confinement. These scenarios are much more likely to be inherently free of low-n neoclassical tearing modes (NTMs) and could largely eliminate the need for active NTM control in future ST devices. Accessing this physical regime and understanding its properties are very important for an ST-CTF, and would provide key data for understanding similar regimes that must be developed at higher aspect ratio for fully non-inductive operation of ITER. The proposed doubling of the toroidal field and more efficient off-axis current drive from a 2nd NBI in an upgraded NSTX will provide the means for achieving, sustaining, and understanding fully non-inductive operation in NSTX, next-step STs, and beyond.

1.7 NSTX Long-term Goals

As described in Section 1.1, the highest priority of the NSTX research program for 2009-2013 is to develop the ST knowledge-base and to develop attractive ST operating scenarios and configurations to enable the ST to be utilized as a next-step facility in support of DEMO. Major upgrades to NSTX are proposed within the next 5 year period to strongly address these objectives, and research utilizing these upgrades will extend into the next 5 year (2014-2018) period. Shown in Figure 1.7.1 are the major upgrades and scientific objectives of the NSTX 10 year research plan consistent with achieving the highest priority objectives. More detailed timelines for facility and diagnostic upgrades for the next 5 years are provided in Chapter 7, and detailed research plans are provided in Chapters 2-6.

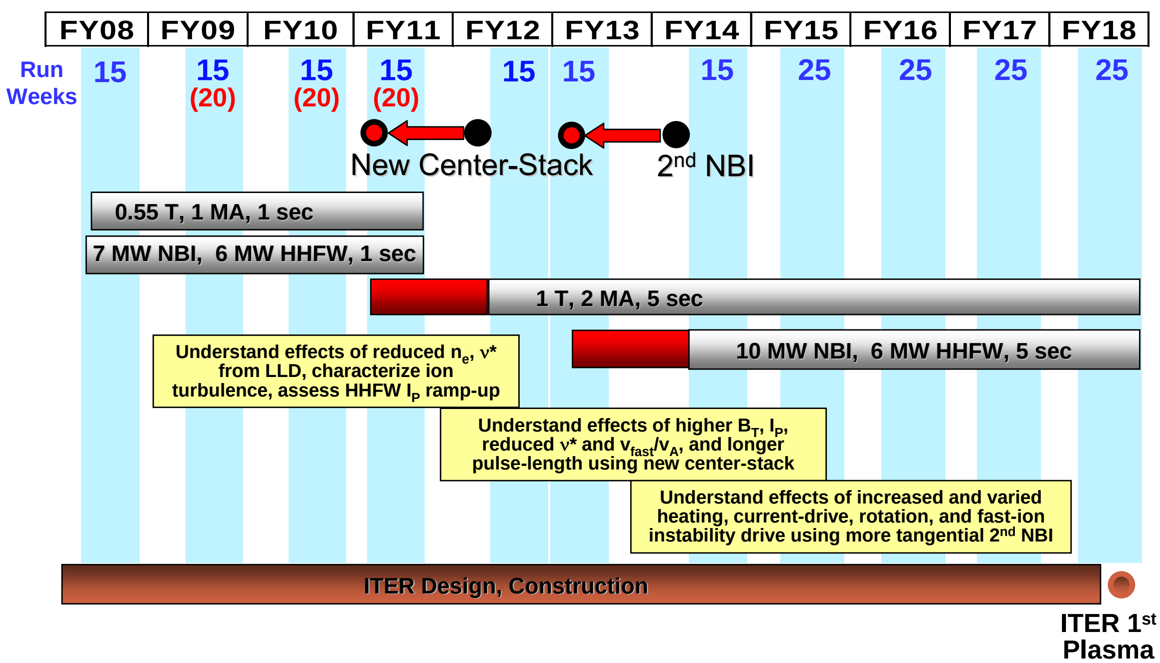


Figure 1.7.1: NSTX 10 year research timeline highlighting the timing of the proposed major upgrades and the roles of the major upgrades in research as summarized in the yellow boxes. The "base" research plan assumes a 15% increment over FY08 funding levels, and the "incremental" research plan ***shown in red*** *assumes a 25% increment over FY08 funding levels. The incremental plan enables acceleration of the major upgrades by 1 year, provides additional run-time, and enhances (not shown) divertor, diagnostic, and EBW capabilities.*

1.8 NSTX Scientific Organizational Structure

The NSTX scientific organizational structure is organized by topical science area into "Topical Science Groups" (TSGs). These groups have been formed to sustain scientific and programmatic focus beyond the annual milestones, and as such, tenure for the leadership team of a TSG is expected to be 2 years, but this leadership is both flexible and renewable. The TSG leaders are expected to share responsibilities among the team members and to try to achieve consensus on priorities from all members involved in a particular topical science area. The TSG leadership structure has one TSG leader, one deputy leader (typically both experimentalists), and 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 research. The TSG leadership responsibilities include:

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

The run coordination and topical science group structure (as of July 2008) is shown in Table 1.8.1 below. The NSTX five year plan chapters of this document are organized with a similar structure and were coordinated and/or written in large part by members of the TSG leadership. As described in the plan Appendix, NSTX collaborators play an essential role in leading the NSTX scientific program. As shown in Table 1.8.1, nearly half of the combined topical science group leadership, deputy leadership, and run coordination is provided by non-PPPL researchers.

NSTX Scientific Organization for FY2008 Run

	Coordinator	Deputy
Run coordination	Michael Bell	Roger Raman (Univ. Washington)

Topical Science Group	Leader	Deputy Leader	Theory / Modeling
Macroscopic Stability	Steve Sabbagh (Columbia Univ.)	Stefan Gerhardt	Josh Breslau
Transport and Turbulence	Stan Kaye	Kevin Tritz (Johns Hopkins Univ.)	Taik-Soo Hahm
Boundary Physics	Vlad Soukhanovskii (LLNL)	Rajesh Maingi (ORNL)	Daren Stotler
Wave-Particle Interactions	Gary Taylor	Eric Fredrickson	Nikolai Gorelenkov
Advanced Scenarios and Control	David Gates	Jon Menard	Charles Kessel
Solenoid-free Start-up and Ramp-up	Roger Raman (Univ. Washington)	Dennis Mueller	Steve Jardin

Table 1.8.1: NSTX scientific organizational structure for Run Coordination and Topical Science Group leadership as of July 2008.

This page intentionally left blank