U.S. Spherical Tokamak Program Initiatives for the Next Decade

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Overview of U.S. ST Missions

During the next decade, the U.S. Spherical Torus/Tokamak (ST) research program aims to make major contributions to narrowing or closing key gaps in the development of fusion energy. The U.S. ST program missions are to: (1) establish the physics basis for the spherical tokamak (ST) as a candidate for a Fusion Nuclear Science Facility (FNSF), (2) understand and develop novel solutions to the plasma-material interface (PMI) challenge, and (3) advance the understanding of toroidal confinement physics for ITER and beyond. In support of ST mission 1, NSTX-U and Pegasus will establish non-solenoidal start-up and ramp-up for FNSF, and NSTX-U will establish the physics, scenario, and control basis for ST-FNSF and DEMO. In support of ST mission 2, NSTX-U will establish the physics basis for integrating high-flux-expansion snowflake/X divertors with partial detachment and/or vapor shielding from liquid metals, and LTX and NSTX-U will explore liquid lithium as a means of achieving very high confinement to reduce the size and/or auxiliary power of an ST-based FNSF. For ST mission 3, NSTX-U and Pegasus will explore high-β regimes at low aspect ratio to validate models for transport and MHD stability including edge localized mode (ELM) stability, and NSTX-U will provide world-leading studies of non-linear Alfvénic instabilities and electromagnetic turbulence for ITER burning plasmas and high-β configurations.

Overview of U.S. ST Initiatives

The proposed NSTX-U initiatives for the coming decade are split roughly into two 5 year periods (see Figure 2). The first 5 year period is largely dedicated to establishing the ST physics and operational scenarios for ST-FNSF to inform the choice of FNSF aspect ratio and divertor configuration. The second 5 year period is largely dedicated to converting from graphite to all high-Z PFCs/substrates, testing flowing liquid lithium in the divertor, pulse-length extension, and a comparative assessment of high-Z and high-Z plus liquid lithium PFCs to inform the choice of FNSF/DEMO plasma facing materials. On Pegasus, an initiative is planned to exploit the unique aspects of operating at near-unity aspect ratio (A ~ 1.1 – 1.3) in a spherical tokamak to: 1) advance the predictive understanding of non-solendoidal startup of ST plasmas using localized helicity injection (LHI); and 2) tests theories of plasma edge stability in the H-mode regime, particularly nonlinear ELM dynamics. Development of LHI physics and technology will support deployment of a MA-class non-solendoidal startup capability on NSTX-U and eventually FNSF. For the Lithium Tokamak Experiment (LTX), an initiative is planned to develop low recycling, very high confinement regimes in the ST, with liquid lithium walls. The near-term effort would require neutral beam fueling and heating as an upgrade to LTX (LTX-U). Pending successful results from LTX, a next-step in NSTX-U could be to implement liquid lithium PFCs and a flowing liquid lithium divertor. The overall goal of the LTX initiative would be to support development of very high confinement for a very compact ST-FNSF. More detailed descriptions of the ST initiatives are provided below, and a brief summary of ST contributions to ReNeW Themes is also provided.

NSTX-U: Initiatives to Accelerate Fusion Development

Underlying NSTX-U support of the three ST missions stated above is access to a unique plasma physics parameter regime of high normalized pressure combined with reduced inter-particle collision frequency (collisionality) to address fundamental questions in plasma stability and turbulent transport and to greatly extend understanding of toroidal plasma science. The key enabling capability of the NSTX-U research program is the upgraded NSTX-U facility which consists of two main device enhancements including:

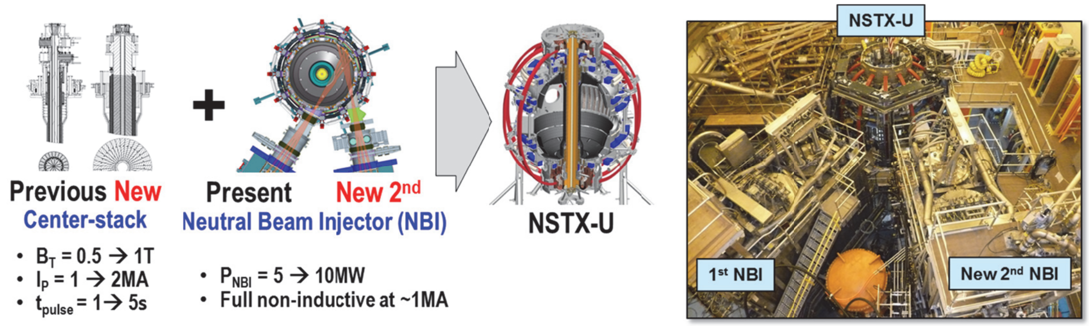


Figure 1 *– NSTX Upgrade components and parameters and an aerial view of the NSTX-U test cell*

(1) a new larger center-stack to double the magnetic field and plasma current and quintuple the pulse length, and (2) a new more tangential second neutral beam injector to double the heating power and non-inductive current drive and access full non-inductive sustainment at the ~1MA level. These components and an aerial view of the NSTX-U test-cell are shown in Figure 1. The Upgrade Project is 90% complete and first plasma is scheduled for early calendar year 2015. Upon completion, NSTX-U will be the most scientifically capable and highest performance ST in the world fusion program.

As noted in previous reports, the Spherical Torus/Tokamak (ST) is potentially attractive as a Fusion Nuclear Science Facility (FNSF) since it is projected to access high neutron wall loading (W_n ~ 1-2 MW/m^2) at moderate size (R_0 ~ 0.8-1.8m) and fusion power (P_{fusion} ~ 50-200MW) with a modular configuration and simplified access using a vertical maintenance scheme. Further, recent calculations show that the Tritium Breeding Ratio (TBR) near 1 is achievable at sufficiently large major radius and with careful design. While the ST has the potential to be utilized as an FNSF, several challenges and gaps need to be addressed as identified in previous FESAC reports (Toroidal Alternates Panel – 2008, Research Needs Workshop – 2009). These gaps are summarized as follows:

1. *Non-inductive start-up, ramp-up, and sustainment - since a low-aspect ratio FNSF has minimal inboard shielding and little or no space for a central solenoid*
2. *Confinement scaling and optimization - especially electron thermal confinement*
3. *Stability and steady-state control - in a high beta and low collisionality regime*
4. *Divertor solutions for high heat flux - including integration of a high-performance core with heat flux mitigation and high-Z PFCs and/or substrates*
5. *Radiation-tolerant magnets - including device design and magnet implications*

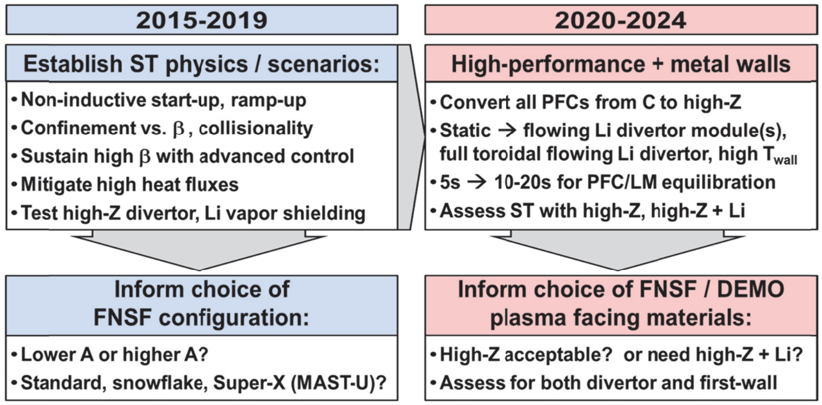


Figure 2 – *Research elements of NSTX-U initiatives for the next decade.*

These gaps are systematically addressed with 5 initiatives described below and the approximate time-line for carrying out these initiatives is shown in Figure 2. As shown in the figure, the first 5 year period is largely dedicated to establishing the ST physics and operational scenarios for ST-FNSF to inform the choice of FNSF aspect ratio and divertor configuration. The second 5 year period will be largely dedicated to converting NSTX-U to all high-Z PFCs/substrates, testing flowing liquid lithium in the divertor, pulse-length extension, and a comparative assessment of high-Z and high-Z plus liquid lithium PFCs to inform the choice of FNSF/DEMO plasma facing materials. Physics and programmatic details of the 5 initiatives and cost estimates for the first 4 NSTX-U specific initiatives are provided below.

NSTX-U Initiative 1: Form plasmas with helicity injection and ramp and sustain with NBI

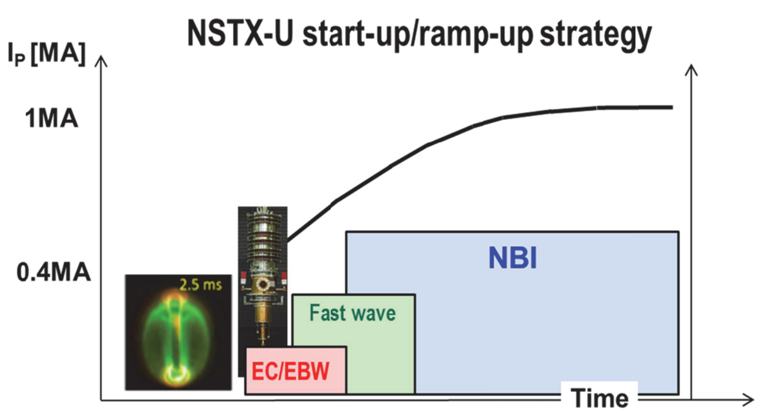
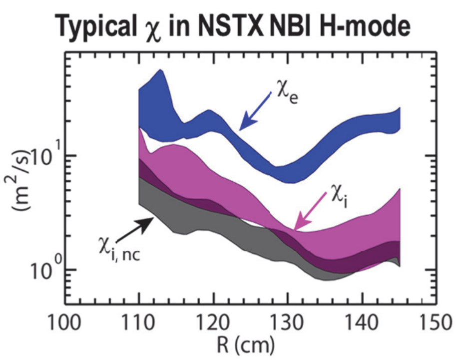


Figure 3 – *Planned NSTX-U non-inductive start-up and ramp-up progression: CHI or LHI current formation, ECH heating and/or EBW heating/CD, fast wave heating (if ECH/EBW insufficient), NBI plus bootstrap ramp-up to the flat-top current value.*

Helicity injection (HI) start-up has achieved 150-200kA of solenoid-free plasma current initiation on both NSTX (coaxial helicity injection - CHI) and Pegasus (localized helicity injection - LHI). CHI/LHI projects to ~0.4-0.6MA on NSTX-U and 1.5-2MA on ST-FNSF. The new 2nd NBI injector (combined with bootstrap current) on NSTX-U is projected to enable non-inductive ramp-up from ~0.4MA to 1MA to demonstrate ramp-up for ST-FNSF. This improved ramp-up capability is a result of more tangential injection which increases CD efficiency and reduces bad-orbit losses at low current.

Testing plasma current ramp-up using NBI and bootstrap is a major goal of the NSTX-U research program. However, a major challenge for solenoid-free current start-up and ramp-up is that HI plasmas are generally too cold (few 10's of eV) and/or too low density for either fast-wave or NBI to be efficiently absorbed. Thus, a major element of this initiative for the first 5 year period is to procure, install, and utilize a new 1MW, 28GHz gyrotron system to heat HI target plasmas for subsequent heating and/or densification with fast-wave and/or NBI to enable complete solenoid-free ramp-up to the ~1MA range as shown in Figure 3. Increasing the gyrotron power to 3MW over a 10 year period would provide world-leading start-up/ramp-up capabilities for ST-FNSF research. This system would also provide means to develop and understand an efficient off-axis EBW current drive technique (and current profile optimization) for over-dense ST, RFP, and AT plasmas. *Additional details can be found in the whitepaper by R. Raman.*

NSTX-U Initiative 2: Understand and optimize high-β / low-ν ST energy confinement



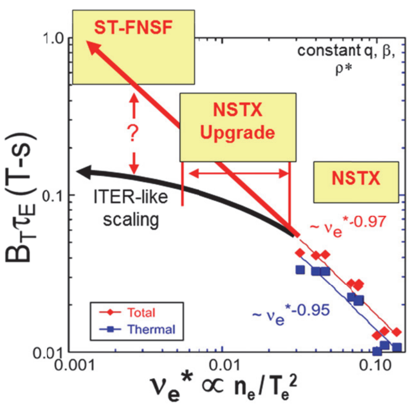


***Figure 4** – Left: Typical thermal diffusivity profiles for NSTX NBI H-mode. Right: normalized confinement $B\tau_E$ versus collisionality for NSTX and possible extrapolations to NSTX-U and ST-FNSF.*

Ion thermal transport in the ST is typically near neoclassical levels in NBI-heated H-mode plasmas with co-injection and significant equilibrium ExB rotation shear. As shown in the left-hand-side of Figure 4, electron thermal transport typically dominates ion thermal transport ($\chi_e >> \chi_i$). Because of the higher β typical of higher performance ST scenarios, electromagnetic effects in plasma turbulence become more prominent, and electromagnetic instabilities including micro-tearing, compressional or global Alfvén eigenmodes, and kinetic ballooning modes have all been implicated in causing some degree of anomalous energy transport in high-β ST plasmas. ST global thermal confinement has thus-far been observed to scale as $B\tau_E \sim \nu^{-0.9} \beta^{-0.2}$ (see right hand side of Figure 4) which differs considerably from the ITER 98y,2 ELMy H-mode scaling $B\tau_E \sim \nu^0 \beta^{-0.9}$. Thus, a major research element of the transport initiative is to determine the β and ν scaling of ST transport accessing the unique high-β and low-ν parameter regime in support of FNSF and DEMO – both ST and AT. The initiative involves utilizing advanced diagnostics including beam emission spectroscopy, high-k_r and k_θ scattering, polarimetry, Doppler back-scattering, and cross-polarization scattering to measure density and magnetic field fluctuations over a range of gyro-scale-lengths. Routine control of density and collisionality ν will be achieved with lithium wall coatings, ELM pacing with lithium granules and externally applied 3D magnetic fields, and a lower outboard divertor cryo-pump. *Additional details can be found in the whitepaper by W. Guttenfelder.*

NSTX-U Initiative 3: Achieve non-inductive sustainment and stability at high β/low ν

NSTX achieved 65-70% non-inductive current drive using NBI and bootstrap current drive sources and at an FNSF-relevant-level $\beta_T \sim 15$-20%. As shown in the left-hand side of Figure 5, NSTX-U was specifically designed to access 100% non-inductive current drive from a combination of bootstrap current and increased NBI current drive efficiency and power using the new 2nd more tangential NBI source. As shown in the Figure, NSTX-U is projected to be able to access full non-inductive operation over a range of density values and with confinement near ITER ELMy H-mode scaling levels, and this flexibility is unique in the world program for STs. However, it was observed in NSTX that TAE modes could cause redistribution and/or loss of NBI fast ions and associated current drive – especially when TAE "avalanches" (non-linear/multi-mode fast-ion transport events) occurred. In NSTX-U, access to more TAE-quiescent regimes is predicted to be accessible utilizing higher toroidal field to lower the minimum v_{fast} / v_{Alfven} from 2 → 1 in NSTX → NSTX-U. Further, more off-axis NBI should also help mitigate TAE effects by broadening the fast-ion pressure profile and operating closer to AE marginal stability. *Additional details can be found in the whitepaper by M. Podestá.*

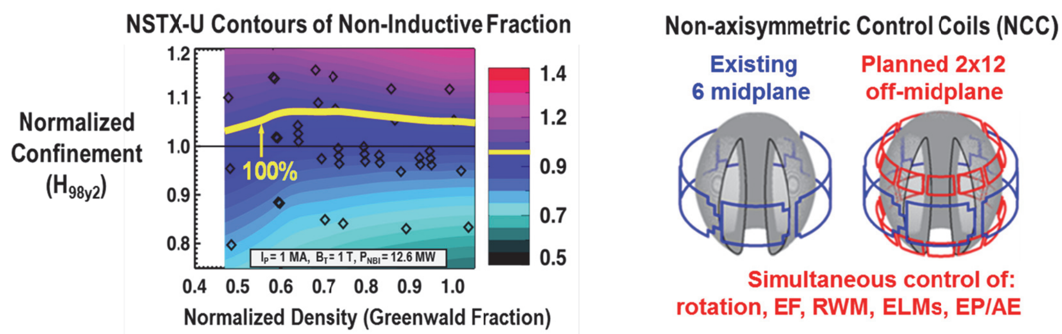


***Figure 5** – Left: contours of non-inductive current drive projected for NSTX-U versus normalized density and confinement. Right: planned NSTX-U off-midplane non-axisymmetric control coils (NCC).*

NSTX sustained normalized beta $\beta_N \sim 6$ for several current redistribution times utilizing a combination of passive RWM stabilization and active error-field/RWM feedback. The integration of high-β_N operation with full non-inductive current drive and disruption avoidance is a major research initiative for NSTX-U. A critical facility enhancement for NSTX-U during the next 10 year period includes new off-midplane non-axisymmetric magnetic field control coils (NCC – see right-hand side of Figure 5) to greatly aid control of rotation, error fields, RWMs, ELMs, and *AE while also enabling leading tests of disruption avoidance for ITER and FNSF. The NCC capability is particularly important to ST research since NSTX-U will be the only ST in the world program routinely operating above the no-wall limit and accessing very high beta MHD stability physics. NSTX-U will also test novel disruption warning and mitigation techniques including fast massive gas injection (MGI) located at different poloidal locations and a (proposed) electromagnetic mass injector to increase the speed and mass of particle delivery during disruption mitigation. *Additional details can be found in the whitepaper by S. Sabbagh.*

NSTX-U Initiative 4: Divertor heat flux solutions, core-edge integration with metallic PFCs

NSTX made substantial progress in reducing peak divertor heat flux (factor of 2-3 reduction) utilizing a high flux-expansion snowflake/X-divertor. The addition of partial detachment (i.e. divertor radiation) further reduced the peak heat flux and led to an overall reduction up to a factor of 5 as shown in Figure 6. In NSTX-U it is anticipated the peak divertor heat flux will be at least a factor of 4 higher due to doubling of the heating power and narrowing (by at least a factor of 2) of the scrape-off-layer (SOL) heat flux width. The highest unmitigated heat fluxes in NSTX-U could be in the range of 30-50MW/m^2 which is far beyond the steady-state heat-flux-removal limits (5-15MW/m^2) of solid PFCs. A major goal of NSTX-U is to test the controllability of up/down symmetric double-snowflakes (i.e. simultaneous control of 4 X-points) combined with divertor radiation control for mitigation of very high heat fluxes with application to FNSF.

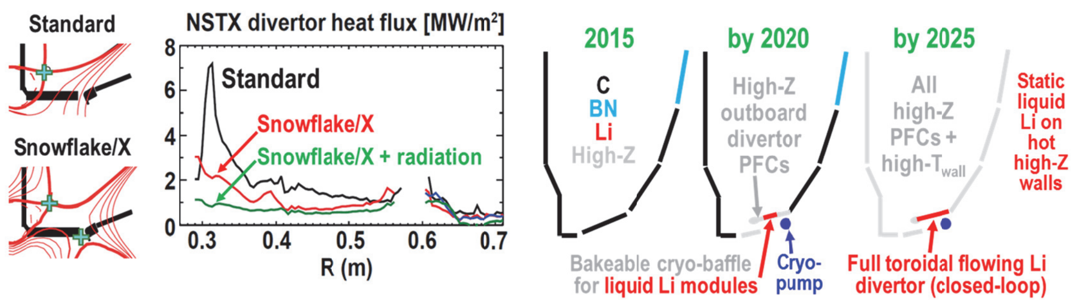


Figure 6 *– Left: Divertor heat flux reduction in NSTX using a snowflake/X divertor in combination with partial detachment and divertor radiation. Right: Planned progression of conversion to high-Z PFCs, cryo-pump installation, liquid Li testing, and high-temperature high-Z walls.*

It is important to note that steady-state FNSF-relevant-performance has not yet been demonstrated in any device – tokamak or stellarator – with high-Z walls. A major initiative for the NSTX-U research program during the 2nd 5 year period is to convert from all graphite to metallic (W and/or TZM) PFCs to assess the feasibility of integration of full non-inductive operation with high confinement, divertor heat flux mitigation, and acceptable accumulation of high-Z impurities. It is expected such integration will be very challenging, and a potential means of overcoming this integration challenge is to provide a thin and replenishable surface in the form of a liquid metal. Lithium is particularly attractive in that it could provide a low-Z surface and could substantially reduce radiative collapse/disruption issues associated with high-Z PFCs. Further, flowing liquid metals could reduce the erosion/re-deposition challenge of solids (albeit with new challenges of collecting the liquid metal material). NBI pulse-length extensions at full power would also be pursued to increase the full-power pulse-length from 5s to 10-20s to provide quasi-stationary conditions for flowing liquid metals and for equilibrated PFC temperatures (which may additionally require active cooling of some PFCs). Finally, the vapor pressure of evaporating lithium (i.e. vapor shielding) may provide additional means of heat flux mitigation for both steady-state and off-normal events and will be explored. The planned progression of associated in-vessel modifications is shown in Figure 6. *Additional details can be found in whitepapers by R. Maingi, M. Jaworski, J.P. Allain.*

ST/AT-FNSF Initiative 5: Radiation-tolerant magnets - including device and magnet design

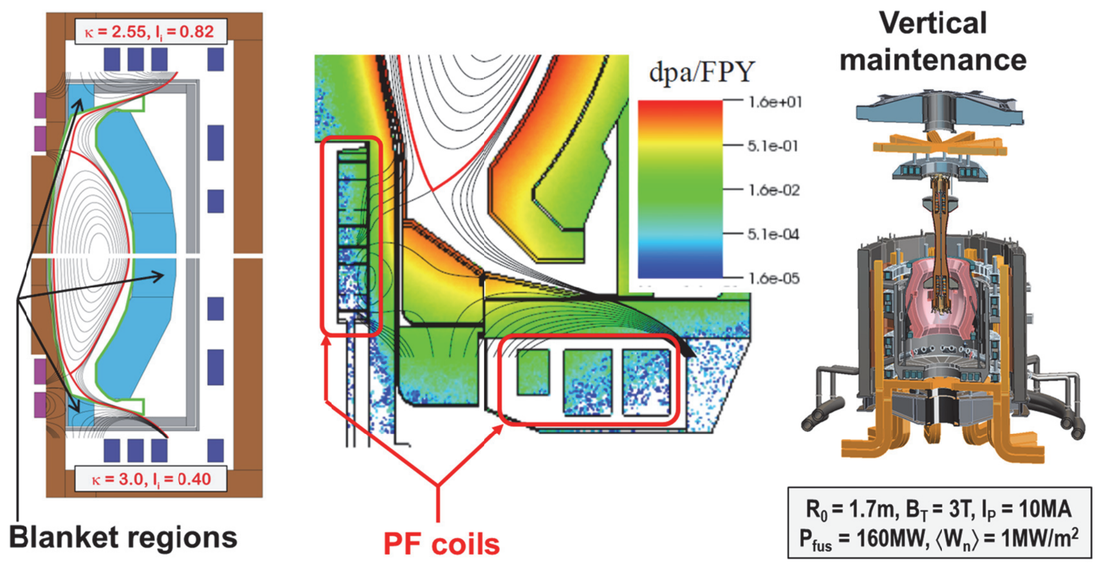


***Figure** 7 – Left: cross-section of ST-FNSF concept (PPPL version) showing blanket modules, vacuum vessel, PF and TF coil locations. Middle: computed nuclear irradiation of blanket modules, divertor PFCs, and divertor PF coils. Right: vertical maintenance scheme, and some device parameters.*

NSTX-U is a plasma physics research device, and obviously cannot directly contribute to fusion power production or to nuclear materials testing or component development. However, NSTX-U and the U.S. ST research community will continue to play a crucial role in determining the physics basis and the integrated design of low aspect ratio fusion power systems including an ST-based FNSF, Pilot Plant, or DEMO. As shown in Figure 7, ST-FNSF configurations have recently been identified which can simultaneously incorporate: (1) a blanket capable of tritium breeding ratio TBR ~ 1 with ports provided for test modules and heating and current drive, (2) a poloidal field (PF) coil set supporting high elongation $\kappa \sim 2.8$ and triangularity $\delta \sim 0.55$ for a range of internal inductance l_i and normalized beta β_N values consistent with NSTX/NSTX-U operation and with sufficient shielding for MgO insulated coils, (3) a long-legged / Super-X divertor analogous to the planned MAST-U divertor which substantially reduces projected peak divertor heat-flux and has all outboard PF coils outside the vacuum chamber and as superconducting to reduce power consumption, and (4) a vertical maintenance scheme in which blanket structures and the centerstack (CS) can be removed independently. We note that $\tau_E > 1.5\times$ ITER H-mode confinement is needed for a compact tokamak-based FNSF (either ST or AT), and this motivates the development of enhanced confinement operation including the use of Li walls to reduce recycling and improve confinement as is being aggressively pursued on LTX as described below.

Conventional/Cu magnet ST configuration optimization will continue to be pursued during the 10 year period. Recent studies also indicate that high-temperature superconductors (HTS) are potentially attractive for more electrically efficient (yet still compact) ST devices. HTS operation at higher temperature may allow for increased nuclear heating of the magnets and/or higher magnetic field which could potentially improve ST confinement and stability. Possible applications include a steady-state toroidal PMI facility, an ST-FNSF or Pilot Plant (Q_{eng} ~ 1), or ST DEMO. Key research needs (outside the scope of NSTX-U initiatives) include HTS radiation limits and manufacturability and reliability. Finally, we recommend enhancing ST and AT FNSF pre-conceptual design funding from the present ~$1M / year to $4M / year to better understand the interplay between physics and engineering design constraints (including magnet technology) across a range of leading FNSF configurations.

Cost estimates for NSTX-U Initiatives 1 through 4:

Cost estimates for achieving the 4 NSTX-U initiatives described above are provided in Table 1. The capital equipment and improvement portion of the NSTX-U budget would be utilized to fund the facility enhancements associated with these initiatives.

NSTX-U 5YP base / +10% sustained for 10 years → $65M / $95M for enhancements

10 year total:	Initiative	Tool / initiative	2015-19	2020-24	10 yr total ($M)
~$19M	**Initiative 1:** Form plasmas with HI and ramp and sustain with NBI	1MW, 28GHz ECH/EBW	7		7
		Additional 1-2MW (2-3MW total) ECH/EBW		10	10
		Steerable mirror		2	2
		Gap sub-total:	7	12	19
~$8M	**Initiative 2:** Understand and optimize high-β / low-ν ST energy confinement	Full polarimetry	1.5		1.5
		Full 2D high-k	1.5		1.5
		Doppler back-scattering	1		1
		Cross-polarization scattering	1	3	4
		Gap sub-total:	5	3	8
~$23M	**Initiative 3:** Achieve non-inductive sustainment, stability at high β / low ν	Lower divertor cryo-pump	7		7
		Partial non-axisymmetric control coils (NCC)	5		5
		Electromagnetic particle injector	1		1
		Full non-axisymmetric control coils (NCC)		3	3
		Enhance MHD sensors: RWM + halo current	2		2
		Additional NCC power supplies (6 → 12 chan)		5	5
		Gap sub-total:	15	8	23
~$45M	**Initiative 4:** Divertor heat flux solutions, core-edge integration with metallic PFCs	High-Z lower outboard divertor	3		3
		Divertor Thomson scattering		5	5
		Flowing liquid Li test module	3		3
		Partial → all high-Z PFCs		4	4
		Hot high-Z PFCs (bakeout system for 350C)		5	5
		Full toroidal coverage flowing Li divertor		10	10
		10-20s NBI (PFC thermal/LM flow equilibration)		15	15
		Gap sub-total:	6	39	45

$65M

$95M

Table 1 *– Cost estimates for the NSTX-U initiatives to addresses key gaps to FNSF and DEMO.*

As is evident from the Table, the first three and part of the 4^{th} initiative can be funded if the NSTX-U 5 year plan (FY2014-2018) base funding level is sustained for 10 years. In this funding scenario, the assessment/establishment of the ST physics basis and operational scenarios would be completed and the full set of high-Z PFCs and a flowing liquid lithium divertor test module would be implemented. If the 5 year plan base + 10% incremental funding was sustained for 10 years, the first through 4^{th} initiative could be completed. In this funding scenario, the PMI/PFC program would be enhanced to provide for hot high-Z walls for high-Z and high-Z + liquid Li tests, a full flowing lithium divertor would be implemented, and the NBI pulse-length extended at full power to provide sufficient pulse-duration for PFC thermal and liquid-metal-flow equilibration.

Pegasus-U: Exploiting Near-Unity Aspect Ratio for ST and AT Physics

An initiative is planned to exploit the unique aspects of operating at near-unity aspect ratio (A ~ 1.1 – 1.3) in a spherical tokamak to: 1) advance the predictive understanding of non-solenoidal startup of ST plasmas using localized helicity injection (LHI); and 2) tests theories of plasma edge stability in the H-mode regime, particularly nonlinear ELM dynamics. Development of LHI physics and technology will support deployment of a MA-class non-solenoidal startup capability on NSTX-U and eventually FNSF.

Framework: The Pegasus experiment is an intermediate-scale ST with R_p = 0.3–0.45 m, a = 0.25–0.40 m; I_p ~ 0.2 MA, B_{TF} ~ 0.1–0.2 T, and Δt_{pulse} ~ 25 msec. A 14-kA, 20 MW edge current injector array provides DC helicity injection for plasma startup to I_p ~ 0.2 MA and strong perturbations of the edge current and/or flow shear profiles. Limited or diverted (single or double null) plasmas are produced.

LHI uses high intensity electron current sources in the plasma edge region to inject helicity into the bulk plasma and thereby initiate and drive toroidal current. The maximum I_p achieved with LHI startup is set by Taylor relaxation constraints as the unstable injected current streams relax to a tokamak-like magnetic configuration. When the helicity input rate is insufficient to attain the I_{p_TL} limit, helicity conservation and power balance governs the $I_p(t)$ evolution as shown in Figure 8.

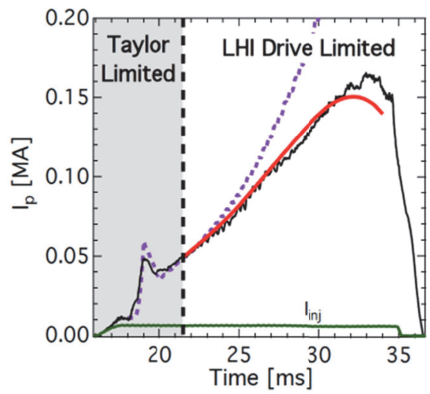


Figure 8 – Measured I_P (black) compared to 0-D power balance model

Nonlinear MHD simulations from the NIMROD code suggest repetitive reconnection interactions inject axisymmetric current-carrying rings from the edge into the plasma core region to drive the core plasma current. These simulations show intermittent bursts of MHD activity similar to that seen in experiments.

Operation at very low A allows stable high I_p at low B_{TF}. This lowers the threshold power (P_{LH}) for transition to the H-mode confinement regime, which is readily achieved on Pegasus at A ≤ 1.2 with Ohmic heating and high-field-side gas fueling. As shown in Figure 9, P_{LH} is found to depend strongly on A, rising to > 10 times greater than predicted by high-A scalings for A ≤ 1.2 which suggests missing physics in those scalings.

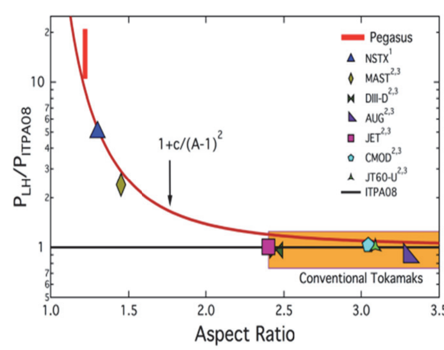


Figure 9 – L-H power thresholds compared to high-A scaling.

Edge current density measurements with high spatial and temporal resolution are available to measure ELM dynamics during H-mode. A nonlinear, multimodal collapse of $J_{edge}(R,t)$ is observed. At low time resolution, the pedestal collapse and filament ejection are quite similar to that seen in the JOREK code [*Plasma Phys. Cont. Fus.* ***53,*** 054014 2011], but higher time resolution measurements show more complex behavior during the collapse as indicated in Figure 10. Type I and type III ELMs show dominant intermediate n ~ 5-15 and low n ~1-3 mode structures, respectively. Such low-A results (from both Pegasus and NSTX) are opposite that seen at high-A, and likely reflect the increased peeling drive obtained at low A.

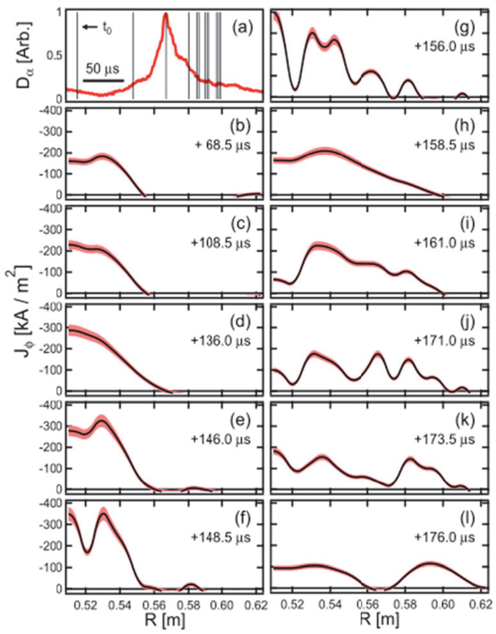


Figure 10 – Measured $J_{edge}(r,t)$ thru ELM cycle.

Pegasus-U Initiative: Extensions to the experimental capabilities of the Pegasus facility are proposed for critical tests of the understanding and scalability of the LHI technique to NSTX-U and FNSF scales. These expansions of capability will support tests of the nonlinear dynamics of ELMs and related neoclassical phenomena in plasmas that operate in transport equilibrium. The Pegasus-U initiative includes: a new centerstack with a new solenoid and TF rod assembly; improved divertor coils; an upgraded TF power supply; and expanded helicity injector arrays. This will increase the pulse length to 100 msec, increase B_{TF} by a factor of 5, and increase the available Volt-sec by 6 times.

The new Ohmic solenoid is being fabricated for Pegasus-U by PPPL. Hardware for a new long pulse power supply for the increased B_{TF} is already available, as is a new multipoint Thomson scattering diagnostic system. A diagnostic neutral beam for ion spectroscopy is being provided by a turbulence diagnostic development grant. A new center-rod assembly will be fabricated by industry. Support for equipment and operation will require a total project budget of ~$1.5M per year, compared to the present level of $0.94M per year.

Importance: Startup of an ST without an Ohmic solenoid benefits designs for a FNSF based on the ST, and can be valuable to tokamak reactors in general. An attractive feature of LHI startup is the possibility that the local current injectors can be retracted from the plasma region after startup, before a facility initiates its nuclear phase.

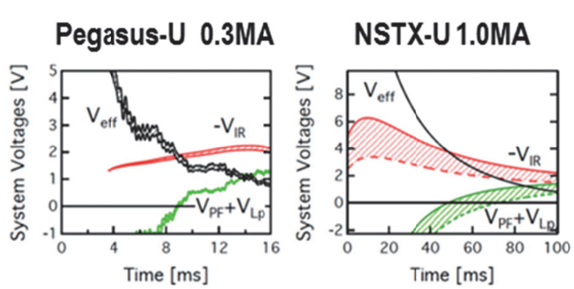


Figure 11 – Drive voltage contribution predictions for high current startup. At end, V_{eff} from LHI ≈ inductive voltages $V_{PF} + V_{LP}$.

A central issue for LHI scaling is the behavior of the plasma electron confinement and the resulting helicity dissipation rate through the average resistivity <η>. The Pegasus-U initiative will allow the measurement of the electron confinement properties during LHI in the reactor-relevant regime where, in contrast to experiments to date, the helicity injection drive is comparable to the poloidal and geometric induction drives (see Figure 11). The resulting plasma confinement data will provide a model for a Tokamak Simulation Code (TSC) 1-D model of plasma startup to ~ 1 MA in NSTX-U.

In addition, longer pulse and high B_{TF} are required to test LHI injector technology appropriate for NSTX-U and FNSF. In contrast to present designs, this technology will need to evolve to cooled, multiple small-aperture arrays as the B_{TF} increases.

Longer pulse and controlled separatrix operation will facilitate detailed nonlinear ELM studies in H-mode plasmas that reach transport equilibrium. Measurements with high time and space resolution during repetitive ELM cycles will allow comparisons to model calculations from relevant codes such as NIMROD, JOREK, BOUT++, and EPED. These measurements will be integrated with existing experimental investigations of pedestal $n_e(r,t)$ measurements using BES on DIII-D and NSTX-U, and will help the interpretation of that necessarily more limited data set.

Finally, new non-solenoidal startup techniques (*e.g.* iron core, partial stellarator windings, etc.) and tests of ELM mitigation (*e.g.* direct tests of ELM pacing via C pellet injection, etc.) will be enabled under this initiative and are under consideration for the latter part of the next decade. Access to high I_p will test the limits of the tokamak configuration at high toroidicity and high field utilization, where $I_p \gg I_{TF}$.

World leading: The Pegasus experiment addresses ST and tokamak physics at the lowest available aspect ratio in the world. It has a unique, high-current localized helicity injection system capable of routinely producing plasmas with $I_p \sim 0.15$ MA and Pegasus–U will expand that capability to the scientifically significant 0.2–0.3 MA regime. These powerful edge current injectors can also support experiments in $J_{edge}(R,t)$ and $v_{edge}(R,t)$ modifications. The capability for detailed measurements of plasma edge equilibrium and stability are unique in the world fusion program and can directly support the development of ELM understanding for ITER and beyond.

LTX: Exploring the advantages of liquid lithium walls

An initiative is planned to develop low recycling, very high confinement regimes in the ST, with liquid lithium walls. The near-term effort would require neutral beam fueling and heating as an upgrade to LTX (LTX-U). Pending successful results on LTX, this initiative supports a next-step in NSTX-U to implement liquid lithium coated high-Z PFCs and a flowing liquid lithium divertor. The overall goal of the initiative would be to support development of a high confinement and very compact ST-FNSF for fusion nuclear material and component development.

Framework: The Lithium Tokamak eXperiment (LTX) is an ST with R=0.4 m, a=0.26 m, and κ=1.5. Operational limits are $B_{toroidal} < 3.4$ kG, $I_P < 200$ kA, and a discharge duration < 100 msec, although LTX operates at reduced parameters at present. LTX features a conformal 1 cm thick copper shell. The plasma-facing surface of the shell is clad with stainless steel, and can be coated with up to 100 nm of lithium. The shell covers

85% of the plasma surface area, and is heated by electrical cable heaters, up to 350 °C (lithium melts at 180.5 °C). LTX was designed to investigate the modifications to tokamak equilibrium with low recycling walls, and to test the feasibility of liquid metal plasma-facing components (PFCs) for near-term tokamak and longer-term reactor applications. At present, LTX is the only tokamak in the world program which operates with a hot metallic wall, with or without lithium coatings.

Clean lithium surfaces provide a low recycling plasma-facing surface because low-Z lithium has a low reflection coefficient for H, D, or T plasmas. For a high-Z wall such as tungsten, 90% of the incident plasma ions are promptly reflected into the edge plasma, for typical edge plasma parameters. For lithium, only 10% of the incident plasma ions are neutralized and reflected; the remaining 90% remain in the lithium. Hydrogen is also highly soluble in lithium, so that it is retained in solution, in the liquid lithium PFC. Liquid lithium also permits rapid diffusion of hydrogen in solution, so that the surface does not become saturated with hydrogen.

A low recycling wall, with core fueling and heating of a tokamak discharge, results in a high plasma edge temperature (analogous to a high pedestal temperature in divertor H-mode). With very low recycling, the edge temperature can approach the core temperature, reducing the free energy associated with the electron temperature gradient, and therefore reducing the growth rate for instabilities driving turbulent thermal diffusion. Experimentally, reduced recycling has almost universally been associated with improved performance in tokamaks. However, no tokamak has ever demonstrated more than a 10 – 20% reduction in recycling. The research goal for the LTX project is to reduce global recycling for the discharge to < 50% with liquid lithium wall coatings.

LTX recently tested a new lithium coating system utilizing a 1.5 kW electron beam evaporation system. The electron beam system stirs and heats a lithium inventory of up to 150 g in toroidal pools, in the bottom of the shells. A few minutes of beam operation heats the lithium to 450 °C, and coats the shells with liquid lithium. With this system, high performance discharges have recently been obtained in LTX with molten lithium PFCs. Molten lithium PFCs cover 2 m^2 of the wall in LTX, or half the total shell area. The molten lithium area is determined by the areal coverage of the e-beam evaporation system. For comparison, the total plasma surface area at the last closed flux surface is < 5 m^2. This was the first successful operation of a tokamak plasma with a large-area molten liquid metal wall, and demonstrates the feasibility of liquid metal PFCs. This will increase the liquid lithium plasma-facing surface area to 4 m^2. All plasma-contacting surfaces will be coated with liquid lithium.

The energy confinement times for discharges in LTX with partial liquid lithium walls exceed ITER98P(y,2) ELMy H-mode scaling law by up to a factor of 4-5. Energy confinement times in LTX with both liquid and solid e-beam generated lithium wall coatings are shown in Figure 12, and are compared with older results from the CDX-U device, and with earlier results from LTX using a less robust and much slower lithium coating approach (helium dispersed evaporation from a resistively heated crucible).

Complete coating of the shells with liquid lithium requires a second electron beam system, which is presently being installed on LTX.

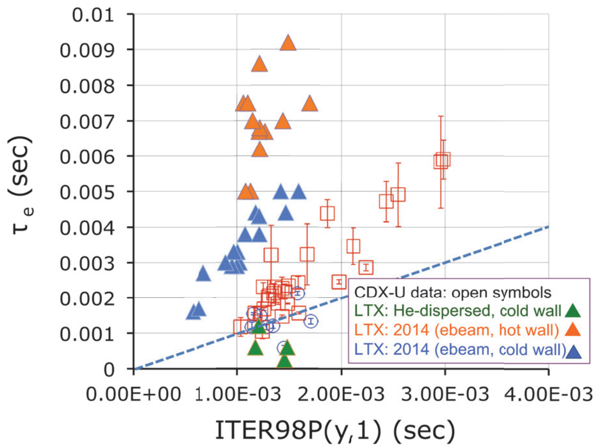


Figure 12 – Measured confinement time compared to ITER98P(y,1) scaling. Open symbols denote data from experiments in CDX-U (Majeski 2006), with 600 cm² liquid lithium and 1000 cm² solid lithium, open symbols, compared to LTX data using helium dispersed solid lithium coatings (coated area = 4 m²), e-beam generated solid coatings on a cold wall (coated area = 4 m²), and e-beam generated liquid lithium coatings on a hot, 300 C wall (2 m² liquid lithium coatings). The confinement data with helium-dispersed solid coatings was found to agree with kinetic measurements within ~20%.

LTX Initiative:

Although operation with a full 4 m² liquid lithium wall can reduce overall recycling to < 50%, the gas jet fueling systems developed for LTX are at most 30 – 40% efficient. This compares favorably to the 5-10% fueling efficiency available with simple gas puffing or recycling, but introduces significant cooling neutral gas into the edge. The initiative proposed here would substitute low energy neutral beam fueling. A suitable compact neutral beam which operates at 18 – 23 keV, at up to 35A current, has been offered at no cost to the LTX group by Tri-Alpha Energy. Since the volume of the LTX plasma is modest (0.8 m³), the fueling provided by this neutral beam source should significantly reduce or possibly eliminate the need for gas jet fueling. NBI also extends results to a regime where the ions are strongly heated relative to the electrons. Note that the Ohmic power input in LTX is <100 kW, since the loop voltage during lithium wall operation is less than 0.5V. A 700 kW beam provides strong auxiliary heating.

A neutral beam will also enable active CHERs measurements of the core Li concentration in LTX, using the system installed and operated on LTX by ORNL. NSTX operation with solid lithium-coated walls resulted in very low core lithium concentrations; it is important to also measure core lithium concentration with liquid walls. A beam provides toroidal momentum input as well as the opportunity to employ active CHERs for plasma rotation measurements and momentum transport estimates; such measurements have not been previously performed in the absence of edge neutral drag.

Unless confinement degrades strongly with neutral-beam injection, the addition of 700 kW of injected power will also allow an investigation of beta limits with a very close fitting lithium-coated wall, and the modifications to equilibrium induced by low recycling walls, e.g. broadened current profiles. Very preliminary studies of low-n stability with close fitting lithium coated walls have indicated that the β_{Normal} limit may be increased.

The incremental funding requirement for neutral beam operation of LTX is 1 – 1.2 M$ per year, in addition to the present FY14 budget of 1.3 M$. There is no cost for the neutral beam (except shipping), and LTX has in place an adequate diagnostic set, so the funding increment provides added manpower to fully staff the diagnostic set, operate the beam, implement modest plasma feedback systems, and perform deferred maintenance on the device. LTX is the only device in the world fusion program which operates, or can operate, with hot metallic walls, and thus is the only available venue for investigations into very low recycling liquid lithium walls.

Importance: The scale size of a Fusion Nuclear Science Facility (FNSF) is set by confinement requirements and beta limits. An ST-based very compact FNSF can likely be designed to operate at substantial beta, provided plasma confinement can be substantially increased. These recent results represent the largest relative increase in tokamak energy confinement ever observed, in any scale device, and present an opportunity to substantially rescale fusion power systems.

World leading: At present, LTX, although a modest scale experiment, is the only operating tokamak in the world with a high temperature metallic wall, suitable for experiments with liquid lithium PFCs. The addition of NBI will increase these unique capabilities and enable confinement assessments at higher ion temperature and plasma beta much closer to NSTX-U and FNSF operating regimes. Thus, the proposed LTX upgrades would strongly inform the value of implementing hot, high-Z, liquid lithium PFCs in NSTX-U and future devices.

ST contributions to ReNeW Themes:

In addition to assessing the ST for the FNSF application, and for developing solutions to the PMI challenge, NSTX-U, Pegasus-U, and LTX-U 10 year initiatives will contribute strongly to all ReNeW Themes as described below:

Theme 1: Burning Plasmas in ITER

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. The resulting fast ion pressure can destabilize Alfvén Eigenmodes (AEs) that, in turn, affect the fast ion distribution through enhanced transport in space and energy. The challenge for present tokamaks is to provide the required physics basis to enable the development, verification and validation of predictive theoretical and numerical tools. To address this challenge, NSTX-U will provide access to the widest range of fast-ion parameters spanning v_{fast} / v_{Alfven} from 1-2 (ITER range) to 3-5 (ST-FNSF range) with controllable neutral beam deposition to vary the spatial distribution of fast ions and explore the resulting AE stability response. Control of AE modes and associated transport will also pursued using varied NBI tangency radius of injection, fast-wave heating, and 3D magnetic fields. Further, a novel

Motional Stark Effect (MSE) – Laser Induced Florescence (LIF) diagnostic will be deployed for the first time as a means of measuring |B| and pitch angle (and thus the total pressure profile and q profile) without a heating beam and with optics/mirrors that could potentially better survive the ITER nuclear environment than conventional MSE. Finally, we note that a) understanding the physics underlying the L-H power threshold is critical for ITER to enable access to the H-mode and high fusion gain, b) understanding H-mode pedestal stability and edge localized mode (ELM) dynamics is also critically important for optimizing ITER and FNSF performance. ST data from Pegasus/Pegasus-U and NSTX-U will extend and improve leading models of this physics and inform ITER operating regimes.

Theme 2: Predictable, High-Performance, Steady-State Plasmas

A major goal of the NSTX-U program is the achievement of full non-inductive current drive at a high toroidal beta (~15-20%) sufficient to provide high neutron flux (≥ 1 MW/m^2) and fluence in an ST-based fusion nuclear science facility. The 2^{nd} more tangential NBI incorporated in NSTX-U was motivated in large part by the need for increased off-axis and higher efficiency NBI current drive to support non-inductive sustainment and ramp-up. NSTX-U will pursue a high-beta and high-bootstrap-fraction approach to ST-FNSF scenarios which will complement approaches planned for MAST-U which will rely more heavily on NBI current drive for non-inductive sustainment. All of this research in support of developing high-performance steady-state scenarios is directly relevant to an ST DEMO and aids AT-FNSF/DEMO development as well.

ST research will greatly extend predictive capability by exploiting the strong intrinsic plasma shaping and enhanced stabilizing magnetic field line curvature accessible at low-aspect-ratio. 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 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.

Theme 3: Taming the Plasma-Material Interface (PMI)

The increased heating power and compact geometry of NSTX-U will produce very high exhaust power flux prototypical of fusion reactors, requiring the development of solutions to handle these power levels at the Plasma-Material Interface (PMI). NSTX-U will explore novel solutions to the power exhaust challenge for FNSF and DEMO by testing extreme expansion of the magnetic field lines and by testing liquid metal plasma facing components (PFCs) to mitigate the erosion and melting problems associated with solid materials. Radiative detachment of the divertor will also be pursued, as will combinations of high flux expansion, detachment, and liquid metals – including the effects of liquid metal vapor shielding as a means of dissipating power. The ability to explore very high exhaust power density, high magnetic expansion, and the liquid metals in the same device is unique in the world fusion program. In the longer term, NSTX-U plans to convert from carbon to high-Z PFC substrates, and will test the viability of integrating a high-performance fusion core with both high-Z PFCs and liquid lithium on high-Z substrate materials to assess the advantages and disadvantages of high-Z solid versus liquid lithium PFCs. LTX/LTX-U will extend exploration of a novel liquid lithium wall regime with new ion heating and core fueling capabilities to access very low wall recycling and explore the upper limits of tokamak confinement. Pending success on LTX/LTX-U, such very low recycling regimes would also be tested on NSTX-U. Lastly, in support of improved boundary physics predictive capability, the low-A and high beta accessible in the pedestal and scrape-off layer of the ST can change the stability and collisional and turbulent transport mechanisms in the edge plasma (as described in Theme 1 above).

Theme 4: Harnessing Fusion Power

As described in ST/AT-FNSF Initiative 5 above, the NSTX-U and the U.S. ST research community will play a key role in determining the physics basis and the integrated design of low aspect ratio fusion power systems including an ST-FNSF, Pilot Plant, or DEMO.

Theme 5: Optimizing the Magnetic Configuration

The ST-specific thrust for Theme 5 is Thrust 16: “Develop the spherical torus to advance fusion nuclear science”. The overarching goal of this Thrust is to extend ST performance to regimes of lower collision frequency, approaching values needed for fusion nuclear science applications. The initiatives described above are directly aligned with the objectives of Thrust 16 to narrow or close gaps in plasma startup, exhaust power handling, controlled stability, and plasma sustainment for and ST-based FNSF.