Five-Year Spherical Torus Research Priorities in Support of High-Gain Burning Plasma, Fusion Nuclear Science, and Plasma Material Interface Missions

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Table of Contents

1. The assignment and the approach
2. Missions of thrusts and the enabling ST applications
   2.1. High-Gain Burning Plasma (BP)
   2.2. Fusion Nuclear Science (FNS)
   2.3. Plasma Material Interface (PMI)
3. Attractive features of ST applications
4. High Priority Research Needs and Tools
   4.1. Plasma current formation and ramp-up
   4.2. Innovative magnetic geometries and first wall solutions
   4.3. Confinement, stability at fusion-relevant parameters
   4.4. Active and passive control to enable long-pulse disruption-free operation at low li
   4.5. Tools to maintain the current and control the plasma profiles
   4.6. Develop normally conducting radiation-tolerant magnetic field coils for low-A applications
   4.7. Extend ST performance to near-burning plasma
5. Summary and discussion

References
Appendices

Five-Year Spherical Torus Research Priorities in Support of High-Gain Burning Plasma, Fusion Nuclear Science, and Plasma Material Interface Missions

1. The Assignment and the Approach

The Spherical Torus (ST) is a Tokamak with the inboard hole of the plasma torus reduced to the practical limits. So far the ST experiments (22 operating worldwide at present) and designs have plasma aspect ratios (A=R/a) in the range of 1.1 – 2, with A~1.5 being the most common. Here R is the major and a the minor radii of the vertical plasma cross section. The properties of the ST, its plasma physics relationship with the Tokamak, and its prospect for making major contributions in fusion energy sciences during the next 20 years are summarized in a recent Fusion Energy Sciences Advisory Committee (FESAC) Toroidal Alternates Panel report [1].

In this assignment the STCC was requested to produce a concise white paper that addresses the high priority ST research needs during the next 5 years. These research needs are directed toward supporting three major thrusts identified in the ReNeW report [2]. The strengths of the world ST programs are taken into account in identifying the tools that can be used to address the needs effectively. High priority research needs and tools for the succeeding 5 years (during 6-10 years from now) will be addressed in a separate future assessment.

The magnetic fusion community ReNeW process focused on the research required to enable design of a DEMO, but did not define specific research facilities required to address this research. In the interest of enhancing the contributions of the ST programs to Fusion Energy Sciences, the STCC chose to consider the ST embodiments for three ReNeW thrusts: high-gain Burning Plasma (BP), Fusion Nuclear Science (FNS), and Plasma Material Interface (PMI), arranged in alphabetical order in this white paper.

These embodiments, as concepts of ST facilities, have been characterized in recent publications and discussions (see details in Appendices F-H). The plasma conditions in these ST concepts required to carry out the research thrusts in turn determine the high priority ST research needs. In this white paper we further focus on those research needs that can be addressed during next 5 years.

The STCC chose to use the key research elements identified in the ReNeW Report Thrust 16 ("Developing the spherical torus to advance fusion nuclear science") to organize the description of high priority research needs presented in this white paper.

2. Missions of Thrusts and the ST Embodiments

The missions of these thrusts and the essential features envisioned for the ST embodiments of BP, FNS, and PMI are provided below:

2.1. High-gain BP

The BP mission aims to explore the high fusion gain plasma conditions prototypical of a tokamak and ST magnetic fusion DEMO core in which the fusion alpha power dominates the external plasma heating power, nearly all of the plasma current is self-generated by the neoclassical bootstrap effect, and the strongly coupled nonlinear physics of plasma

equilibrium, transport, and macroscopic stability must therefore be controlled and sustained in steady-state.

The ST plasma configuration for this mission is characterized by compact size (major radius ~ 1.5m), high fusion gain (Q up to 20) and reduced fusion power (250–350MW), high fusion neutron flux (2 – 4MW/m^2 peak outboard) but low fluence due to pulse lengths below and up to hundreds of seconds, which are set by current profile equilibration and control time-scales. An ohmic induction solenoid for current ramp-up is therefore included. This configuration is further characterized by high plasma current (9–17MA) in combination with high thermal energy confinement enhancement factor (H_{98} = 1.5 – 1.8) to provide the confinement needed for high fusion gain and high bootstrap current fraction (70-95%), and advanced stability control to support high β operation (β_N = 5-7, β_T = 25-50%). (See, Appendix F)

2.2. FNS

This mission includes a facility that provides a fully integrated fusion nuclear environment to elucidate and resolve the synergistic effects in the science of fusion plasma material interactions, tritium fuel cycle, and power extraction under very high fusion neutron fluence – simultaneously encountering four phases of matter across the nuclear, atomic, nano, meso, and macroscopic scales, for continuous plasma durations from an hour to a week.

The ST application for this thrust can be characterized by compact size (major radius ~ 1.2m), mid-range plasma current (4–10MA), conservative stability conditions, moderate fusion gain (Q = 0.8–3.5) and power (20–150MW), substantial fusion neutron flux (0.3–2MW/m^2) and total neutron fluence (1 MW-yr/m^2), using extensive remote handling of fully modularized components for efficient component replacements and flexible staging of the research program. (See, Appendix G)

2.3. PMI

This mission aims to qualify candidate Plasma Facing Components (PFCs) including divertors in a long-pulse, DD facility with edge conditions, power loads, and other non-nuclear operating factors approaching those expected in a fusion nuclear device (e.g. a fusion nuclear science facility or DEMO), in order to reduce design uncertainties.

Within a 5-year time frame, the upgraded NSTX, including minor extensions of this upgrade, would make substantial progress toward the materials requirements for an ST embodiment to address this mission. (See, Appendix H)

3. Attractive Features of ST Embodiments

The ST configuration has several attractive features for the chosen ST embodiments:

- Macroscopic stability is verified at very high plasma pressure relative to the applied magnetic field pressure (β) with passive or active control, reducing the size and cost of the magnets and/or plasma size and current for a given fusion power.
- Very good confinement of plasma thermal ion energy content has been shown in present ST experiments under the above conditions, reducing the required auxiliary heating power for a given fusion power.
- The above combine to allow high fusion neutron and plasma heat fluxes on internal

components for given fusion and auxiliary heating powers in plasma pressures ranging from 1 to 3× the level regularly operated on the JET Tokamak.

- With high plasma energy confinement and normalized plasma pressure, high fusion gain (Q~20 or more) is possible with the above features.
- Alternatively, operation far within the known stability limits increases resilience to plasma disruptions, while delivering the required performance for the FNS and PMI mission.
- The preceding two features also reduce or minimize the required fusion power and hence tritium inventory and consumption.
- Compact size with reduced magnetic field allows extensive modularization of device components, which in turn permits extensive remote handling of these components.
- These allow efficient replacement of all activated components and enable flexible staging of the research program, which in turn minimize the negative impacts of unavoidable risks of uncertainties on the research outcomes and maximize the positive advantages of research discoveries.
- These all combined to minimize the cost of fusion research devices and facilities that address the BP, FNS and PMI research missions.

Scientifically, the very high β, strong magnetic curvature, and the reduced magnetic field extend the plasma physics regimes and understanding to the benefit of all toroidal fusion plasma configurations. These ST embodiments would accrue similar scientific benefits to the level of the JET (D-D only and Q~0.7) for PMI and FNS, 2×JET (Q~1.5–2) for FNS, and eventually beyond ITER (Q up to 20) for BP.

4. High Priority Research Needs and Tools during the Upcoming 5 Years

The information base for this section is provided in the appendices: A-E provides opportunities for ST research contributions from active and proposed programs in LTX, NSTX, Pegasus, and non-US experiments to the ReNeW thrusts; F-G to cover working parameters and research issues for ST applications to high gain BP and FNS; and H to cover PMI/PFC research needs.

This section lists specific research needs and tools available from the US-funded program activities to address these needs. Included are only the research needs of high priority for at least one of the three missions. We note that the foreign programs (described in more detail in the Appendices), will address additional critical issues in ST and general tokamak physics. The subsections here are organized according to the elements of ReNew Thrust 16. Priorities are indicated by H (high), M (medium), and L (low).

4.1. *Exploit and understand magnetic turbulence, electromagnetic waves, and energetic particles for megampere plasma current formation and ramp-up.*

Research Needed and Priorities for	BP	FNS	PMI
4.1.1: *Develop helicity-injection-based and/or RF-assisted startup to provide a target for radiofrequency (rf) or neutral beam (NB) ramp-up and/or sustainment*	M	H	L
4.1.2: *Integrated model of startup and ramp-up systems, using helicity injection, neutral beam, and/or radiofrequency waves*	M	H	L

Research Tools	~5-Year Goals
Pegasus: Localized helicity injection + P_{EBW} ~ 1MW (UW, ORNL) + P_{HHFW} ~ 0.8 MW	• Develop and validate predictive model for point-source helicity injection to ~0.3MA • Investigate role of ion dynamics in strongly reconnecting plasma • Core EBW heating and current drive with poloidal induction to start plasmas to ~ 0.2 MA. • Test EBW coupling physics with angular variability • Explore synergy of helicity startup with RF-heating • Growth via EBW current drive after startup
NSTX: P_{HHFW} ~ 6 MW + coaxial helicity injection + P_{NBI} ~ 6 MW	• Demonstrate coaxial helicity injection to ~0.3-0.5MA • Demonstrate handoff from non-solenoidal startup to fast wave, bootstrap, and NBI-drive for growth and sustainment
DIII-D: (GA, ORNL, PPPL) ECH + ECCD with only poloidal induction.	• Validate and model of poloidal field induction to high current • Validate handoff to NB-driven current growth
MAST: (UK, US, J) P_{EBW} ~350 kW @ 28 GHz + merging-compression + P_{NBCD} ~ 5 MW	• Demonstrate startup with EBW heating and current drive to high current • Demonstrate current ramp-up via NBCD • Test and model synergy with merging-compression

4.2. ***Develop innovative magnetic geometries and first-wall solutions such as liquid metals to accommodate multi-megawatt per square meter heat loads.***

Research Needed and Priorities for	BP	FNS	PMI
4.2.1: *Verification of mitigation of high heat and particle fluxes in eXtended or eXpanded-SOL Divertors (XXD)*	H	H	H
4.2.2: *Assess the particle pumping, heat-flux handling, importance of MHD effects, and the plasma confinement response to liquid metal plasma facing components in the presence of high heat and particle fluxes.*	H	*	H
4.2.3: *Understanding the integrated coupling of plasma material interface under hot walls (600-700 C) and high heat flux conditions, including innovative solid materials designs based on the results of single-effect evaluations of solids and liquid surface materials.*	M	H	H

Research Tools	5-Year Goals
NSTX: Low heat flux Liquid Lithium Divertor (LLD) with outboard and inboard "Snowflake" divertor; higher heat flux LLD	• Test and validation of models of edge/SOL transport, divertor heat flux, and volumetric momentum and power exhaust techniques for the available divertor configurations • Evaluate liquid-metal based low heat flux

	divertor solutions as a means of particle control and heat-flux mitigation for NSTX/NSTX-U and future ST facilities. • Develop and deploy actively cooled LLD for increased power exhaust handing capability.
LTX: Full liquid metal Li walls; hot (≥ 600C) walls; advanced fueling techniques and suite of recycling, impurity influx, and PMI diagnostics	• Develop advanced targets and liquid-metal coolants proposals for use in X-type divertors • Explore global recycling coefficient with liquid lithium, optimum recycling for an ST, and measure impurity influx as a function of wall temperature, particle pumping rate, thermal transfer characteristics • Characterize PMI on hot, high recycling liquid metals, with and without active lithium lower walls.
MAST: Super-X divertor with cryo-pumping	• Carry out proof of principle tests of the Super-X divertor concept
Enhanced PMI/divertor diagnostics in NSTX, LTX; test stands to provide input to theory and modeling	• Characterize and understand plasma – liquid and solid material interactions • Validation and verification of PMI modeling

* Unknown due to unknown impact on the viability of an FNS embodiment, resulting from an absence of test data and design basis for a fast flowing, high temperature, liquid lithium surface divertor in time varying magnetic field that handles high heat and particle fluxes continuously in a full fusion nuclear environment (see, Appendix G).

4.3. ***Utilize upgraded facilities to increase plasma temperature and magnetic field to test the understanding of ST confinement and stability at fusion-relevant parameters.***

Research Needed and Priorities for	BP	FNS	PMI
4.3.1: *Determine confinement trends with adequate understanding of transport in the ST to project with confidence the confinement and heating requirements of chosen ST missions*	H	M	M
4.3.2: *Measure and develop models for energetic particle transport and effects on background plasma in the presence of multiple interacting modes and copious super-Alfvenic ions*	H	M	L

Research Tools	5-Year Goals
NSTX: Plasma performance at multiple-keV temperatures and high β; strong shaping; liquid lithium divertor; P_{NBI} ~ 5-7 MW; P_{HHFW} ~ 6 MW; full diagnostic complement	• Determine confinement trends over wide range of collisionality • Measure low and high-k turbulence spectra and comparison to nonlinear turbulence theory • Ascertain aspect-ratio dependence of confinement • Measure Alfvenic instabilities and compare with theory • Determine effect of Alfvenic instabilities on fast

	ion current drive
LTX: Liquid metal Li walls; hot (≥ 600C) walls with metal wall	• Measure effects of global recycling variation (from very low to high recycling) on confinement and plasma properties and compare with theory
Pegasus: Non-solenoidal startup via helicity injection and rf heating; high-power density EBW heating	• Measure and model helicity dissipation in driven plasmas • Measure and model confinement and losses in rf-assisted startup plasmas • Explore EBW heating in plasma core

4.4. ***Implement and understand active and passive control techniques to enable long-pulse disruption-free operation in plasmas with very broad current profiles.***

Research Needed and Priorities for	BP	FNS	PMI
4.4.1: *Determine passive and active control requirements for achieving low-disruptivity operation at the mission-targeted β_N in low li, low collisionality ST plasmas.*	H	H	M
4.4.2: *Develop and validate models of the plasma response to intrinsic and mode-induced 3D field perturbations for ST plasma conditions.*	H	H	M

Research Tools	5-Year Goals
NSTX: Plasma performance at high β with multiple-keV temperatures; passive stabilizing plates; variable plasma rotation; active 3D field error control; active n=1 mode control; operation at low l_i with strong edge pumping (Li coating and LLD)	• Test RWM stabilization and resonant and non-resonant braking at low collisionality • Test kink/ballooning and RWM stability at low internal inductance • Measure rotation profile impact on plasma stability • Systematically study active feedback control to assess the requirements for very low disruptivity
LTX: Full, close-fitting conducting wall; low-recycling	• Measure effects of wall-localized shear flow layer • Study edge stability at low l_i and high edge T_e • Measure β_N limits with very close fitting (r~1.02a), conducting wall stabilization
Pegasus: Near-unity aspect ratio; strong edge current drive	• Determine stability boundaries at high-I_N, high β_t, low l_i as A approaches unity • Validate peeling-ballooning mode properties via high j_{edge}/B operation

4.5. ***Employ energetic particle beams, plasma waves, particle control, and core fueling techniques to maintain the current and control the plasma profiles.***

Research Needed and Priorities for	BP	FNS	PMI
4.5.1: *Development of tools and understanding to control plasma density, pressure, and rotation profiles at needed plasma performance-levels.*	H	M	M

4.5.2: *Time-dependent integrated modeling and simulation of plasma start-up, ramp-up, and sustainment for the appropriate plasma scenarios*	H	H	M
Research Tools	**5-Year Goals**		
NSTX and upgrades: Liquid Li divertor and strong pumping; 3D field control of rotation; heating with low momentum input (HHFW)	• Use increased particle pumping and fast wave heating to access lower-collisionality • Confirm NBI and fast-wave current drive physics accounting for energetic particle physics • Begin to optimize core and edge transport to improve bootstrap current profiles		
LTX and upgrades: high efficiency gas fueling, NBI central fueling	• Test pressure profile control with core fueling, full-wall pumping • Measure toroidal rotation and damping without edge neutrals.		
Pegasus: High-power EBW heating & CD; Edge-current plasma guns	• Test startup j(r) control via helicity injection, EBW current drive, and RF heating • Test sustained core and off-core j(r) drive via high-power-density EBW		
MAST upgrades	• Test on/off center and co/ctr NBCD • Test EBW-CD		
Large scale integrated simulation tools: SWIM, TSC, TRANSP, ONE-TWO, etc., and eventually FSP.	• Extend transport, stability, and wave-particle interaction models developed for conventional-A tokamaks to present ST conditions.		

4.6. ***Develop normally conducting radiation-tolerant magnets for low-A applications.***
No specific goals are planned in this area.

4.7. ***Extend ST performance to near-burning plasma***

Research Needed and Priorities for	**BP**	**FNS**	**PMI**
4.7.1: *Develop and extend operational scenarios and tokamak tools to simulate future ST plasma regimes relevant to the ST mission.*	H	M	M
Research Tools	**5-Year Goals**		
NSTX and proposed upgrades	• Demonstrate and understand scenarios with high integrated performance, namely: high beta, high confinement, high non-inductive current fraction, low l_i, low collisionality, and low disruptivity and minimal stored energy fluctuation • Utilize control of density, heating source, 3D fields, and plasma rotation control to optimize profiles for high core and edge confinement and MHD mode stability (including ELMs)		
Comprehensive modeling and simulation tools	• Identify and develop favorable scenarios for near-BP plasma access • Compile and apply tokamak and ST integrated		

	modeling and simulation tools to compare with data and project to the anticipated steady state near-burning plasmas

5. Conclusion and Discussion

The STCC has identified high priority ST research needs from among the research elements organized according to the ReNeW Thrust 16 Report [2]. Information for the ST embodiments for the high-gain BP, FNS, and PMI thrusts points to a smaller number of high priority research needs than those listed in this reference.

Appendices A-E further contain information for a very broad list of research opportunities and proposals driven by a broader set of scientific needs than required by the present assignment, which defines the scope of this white paper. As a result, this white paper avoids the temptation of repeating existing research proposals already known to DOE and equivalent funding agencies in E.U, and Japan.

References:

[1] FESAC Toroidal Alternates Panel Report, http://fusion.gat.com/tap/files/FESAC%20TAP%20Final%20Report.pdf.
[2] Magnetic Fusion ReNeW Report, http://burningplasma.org/web/ReNeW/ReNeW.report.press1.pdf.

Appendix A. Opportunities for NSTX ST Research Contributions to ReNew Goals

(Jon Menard)

NSTX Mission Elements:

1. **Understand the unique physics properties of the ST**, including low aspect ratio, high beta, and high fast ion speed relative to the Alfven speed.
2. **Complement tokamak physics and support ITER**, by exploiting unique ST features to improve tokamak understanding, while also benefiting from tokamak R&D.
3. **Utilize advantages of the ST configuration to address key gaps** between ITER levels of performance and that needed for Fusion Nuclear Science (FNS) missions and DEMO, and establish attractive ST operating conditions for fusion applications.

NSTX Capabilities: Base program as of FY2010

Device Parameters:

Major Radius 0.85 m
Minor Radius 0.68 m
Elongation 1.8 - 3.0
Triangularity 0.2 - 0.8
Core Plasma
- $T_e, T_i \leq$ 6keV, 3keV
- $\beta_T \leq 40\%$
- $\beta_N \leq 7.2$
- $l_i \geq 0.3$

Plasma Current
- 1 MA (1.5 MA peak)

Toroidal Field
- 0.35 - 0.55 T

Heating and CD
- 7 MW NBI (2 sec)
- 5 MW NBI (5 sec)
- 6 MW HHFW (5 sec)
- 0.2 MA CHI

Pulse Length
- ~1 sec at 0.55 T
- ~2 sec at 0.38 T

Control
- Passive conducting shell
- Active shape, β, 3D field control

Available NSTX Diagnostics

*(**bold** indicates new capability for 2010)*

Beam Emission Spectroscopy (BES)	Mirnov coils - poloidal array
Bolometer - divertor array	Mirnov coils - three-axis
Bolometer - midplane array	Mirnov coils - toroidal array
CHERS - poloidal	MSE polarimetry
CHERS - toroidal	Neutron detectors
Edge bias & Langmuir probes - inter-LLD	NPA - E ‖ B scanning
Edge deposition monitors	NPA - solid state
Edge pressure gauges	Plasma TV
Edge Rotation Diagnostic (ERD)	Reflectometer - 65 GHz backscattering
Fast cameras - divertor & LLD	Reflectometer - correlation
Fast Ion D-Alpha (FIDA)	Reflectometer - fixed freq.
Fast lost ion probes	Reflectometer - FM/CW
Filterscopes, 1-D D-alpha cameras (EIES)	RF Probe - lower dome
FIReTIP - multi-chord interferometry	Sample probe
Gas puff imaging - divertor	**Spectroscopy - Divertor UV/visible**
Gas puff imaging - midplane	Spectroscopy - LOWEUS EUV
High-k scattering	**Spectroscopy - Lyman-alpha array**
Infrared cameras	Spectroscopy - SPRED VUV
Interferometry/forward scattering - 1 mm	Spectroscopy - SWIFT 2D flow
Langmuir probes (high density) - inter-LLD	Spectroscopy - VIPS visible
Langmuir probes (baffled) - outboard	Spectroscopy - XEUS EUV
Langmuir probes - PFC tiles	Spectroscopy - X-ray crystal - horizontal
Magnetics - diamagnetism	Spectroscopy - X-ray crystal - vertical
Magnetics - flux loops	Thomson scattering - MPTS
Magnetics - locked modes	Visible bremsstrahlung
Magnetics - outboard divertor halo current	X-rays - bremsstrahlung spectrum
Magnetics - Rogowski coils	X-rays - "Optical" array
Magnetics - RWM sensors	X-rays - Tangential TG spectrometer
Mirnov coils - high frequency	X-rays - ultrafast pinhole camera
	X-rays - Ultrasoft arrays

Additional upcoming capabilities from ARRA funding: FY09-11

- Additional Thomson scattering channels at edge to improve pedestal structure diagnosis
- Real-time rotation measurement for rotation profile control using magnetic braking
- 2nd Switching Power Amplifier for 3D coils to improve ELM, RWM, and rotation control
- Enhanced diagnostics and replenishment technology for liquid lithium divertor (LLD)
- MSE-LIF system completion for q-profile diagnosis w/o heating beam + |B| measurement

Proposed major upgrades of NSTX: FY12 and beyond

- New center-stack to support B_T = 1 Tesla, τ_{pulse} = 5 second, I_P = 2 MA plasma operation

- 2nd more tangential NBI for increased heating, current drive, and J(r) profile control

NSTX Contributions to Addressing ReNew Thrust elements

1 – Exploit and understand magnetic turbulence, electromagnetic waves, and energetic particles for mega-Ampere plasma current formation and ramp-up.

- Develop coaxial helicity injection startup to a current level of 0.3-0.5MA (0.2MA achieved)
- Heat start-up plasma to Te ~ 1keV with high-harmonic fast wave, ramp-up and sustain current fully non-inductively at ~0.4MA with bootstrap + fast-wave + NBI
- Collaborate on ECH start-up experiments on DIII-D tokamak device at the 0.2-0.4MA level.
- Develop physics-based, self-consistent, integrated models of startup, with sufficient predictive capability to permit an evaluation of the most promising startup approaches.

2 – Develop innovative magnetic geometries and first wall solutions such as liquid metals to accommodate multi-megawatt per square meter heat loads.

2.1 – Develop and understand innovative magnetic geometries and particle control

- Test and develop the "snowflake" divertor and volumetric momentum and power exhaust, and explore innovative plasma-facing components, e.g. liquid lithium (see below)
- Implement and utilize a liquid lithium divertor for hydrogenic and low-Z impurity control with a goal of density control at levels of 50-70% of the empirical (Greenwald) density limit.
- Implement and utilize edge and SOL turbulence diagnostics to provide data for validation of theoretical and numerical models of edge transport and turbulence in the ST.

2.2 – Develop and understand liquid metal plasma facing components

- Characterize particle pumping performance of, and plasma response to, Liquid Lithium Divertor version I (LLD-1). Depending on LLD-1 results, possibly install additional LLD plates on inboard divertor, improve Li replenishment technology, and/or develop and deploy an actively cooled LLD for increased power exhaust handing capability.
- Based on LTX results, assess whether a full lithium wall is useful and/or required, or if partial walls, or a divertor alone, are sufficient.
- Utilize the new higher spatial resolution Thomson scattering diagnostic capability to obtain first measurements of the detailed pedestal structure in the presence of Li (from evaporated Li and from the LLD).
- Diagnose edge plasma effects and plasma-material interactions with a liquid lithium divertor. Use this data to validate models for the plasma edge, and core-edge coupling, over the available range of global recycling coefficients. Understand the effect of low recycling on the edge and core plasma to project the results to larger, hotter devices.

3 – Utilize upgraded facilities to increase plasma temperature and magnetic field to understand ST confinement and stability at fusion-relevant parameters.

Thermal transport:

- Determine confinement trends over an extended range of collisionality utilizing particle pumping from the liquid lithium divertor and enhanced high-harmonic fast electron heating.
- Utilize high-k scattering diagnostic to measure ETG and AE turbulence, and relate measured

turbulence to measured electron transport using advanced simulations.

- Utilize the Beam Emission Spectroscopy (BES) diagnostic to measure long wavelength turbulence and its relation to ion energy, momentum, and particle transport. Use BES data and advanced simulation to project ion confinement to next-step ST devices/missions.

Energetic particle (EP) transport:

NSTX will continue to engage in theoretical investigation of the interaction between multiple Alfven modes and the background plasma, and measure the Alfven mode structure and the effects of these modes on thermal ion heating and electron transport.

- Utilize the BES diagnostic to measure the radial profile of the eigenstructure of Alfvenic instabilities and perpendicular and tangential fast-ion D_α to measure moments of the fast-ion density profiles. Utilize this data to establish the database of EP loss and transport, for both present day and next step STs, and compare this data to simulation to develop a predictive capability for fast-ion transport.
- Utilize new MSE-LIF diagnostic to measure q profiles during AE experiments for much wider (i.e. arbitrary) range of NBI injection energies and heating source combinations.

4 – Implement and understand active and passive stability control techniques to enable long-pulse disruption-free operation in plasmas with very broad current profiles.

- Reduce the plasma collisionality (utilizing the liquid lithium divertor for improved particle control and/or HHFW for core electron heating) to test theoretical RWM stabilization and the expected increase of non-resonant magnetic braking.
- Exploit electron temperature profile broadening from Li operation to reduce the plasma internal inductance to test kink/ballooning and resistive wall mode (RWM) stability near the current-driven kink limit.
- Utilize upgraded 3D field control with 2nd switch power amplifier to improve variation/control of plasma rotation and assess impact of varied rotation profiles on RWM, EF, and RMP physics.

5 – Employ energetic particle beams, plasma waves, particle control, and core fueling techniques to maintain the current, and control the plasma profiles.

- Implement particle pumping to achieve factor of 2 to 4 reduction in normalized density to support access to reduced collisionality values for investigation of NBI and fast-wave current drive efficiency physics and scaling.
- Assess and optimize the impact of reduced collisionality on core and edge transport – in particular the impact of transport on the bootstrap current density profiles.

6 – Develop normally-conducting radiation-tolerant magnets for low-A applications

There are presently no NSTX plans to design and construct candidate magnet systems for future ST facilities. At the present time, the NSTX project is focused on design and implementation of a new center-stack supporting proposed 1 Tesla, 5s operation.

Appendix B. Possible LTX contributions to the ReNew Thrust 16 elements (Richard Majeski)

LTX capabilities base program + ARRA-funded (as of FY12)

I_p < 400 kA, $\tau_{discharge}$< 0.25 sec., B_{TF} < 3.5 kG, R_0 = 40 cm, a=26 cm (A=1.5), κ = 1.6, configuration: wall limited, with conformal heated SS-faced copper shell, continuous temperature limit 550 °C, replacement molybdenum-faced shell, fast internal gas-cooled PF coil pair. P_{aux} = 100 – 200 kW (5 A, 20 – 40 keV), 1 sec. H NBI, primarily for diagnostics.

Primary diagnostic capabilities, base

16 radial channels core Thomson scattering, 5 channel edge Thomson, (both at single time point), 140 channels magnetics, 2mm moveable + 1 mm fixed interferometers, Lyman-alpha arrays, scanning VUV spectrometer, digital holography (time-resolved high resolution density perturbations), materials sample exposure probe.

Proposed additional LTX capabilities, FY12

Li-CHERS for impurity, T_i measurements, BES. Tungsten spraying of present SS faced shell to compare full hot W vs. Mo first walls in the same device.

FY13 and beyond (proposed upgrade)

Full core NB fueling with 40A, 15 keV, 0.5 sec source.

Contributions to Addressing ReNew Thrust elements

1 – Exploit and understand magnetic turbulence, electromagnetic waves, and energetic particles for megampere plasma current formation and ramp-up.

A. Possible minor contribution to outer coil PF rampup (somewhat off-topic).

2 - Develop innovative magnetic geometries and first-wall solutions such as liquid metals to accommodate multi-megawatt per square meter heat loads.

a. Develop and understand innovative magnetic geometries and particle control.
b. Develop and understand liquid metal plasma-facing components.

A. Proof-of-principle operation of a spherical tokamak with a full liquid metal wall.
B. First tokamak operation with a full low-recycling liquid lithium wall for particle control.
C. First tokamak operation with a full 500 °C high-Z wall.
D. Material probe operation to test solids, nonlithium liquids (tin, gallium) as PFCs.

3 - Utilize upgraded facilities to increase plasma temperature and magnetic field to understand ST confinement and stability at fusion-relevant parameters.

A. Effect of global recycling coefficient on confinement, edge and core temperature.

B. Confinement scaling (I_p, B_{TF}, n_e, T) with very low recycling boundary.
C. Examine global electron confinement with a very low recycling boundary.

4 - Implement and understand active and passive stability control techniques to enable long-pulse disruption-free operation in plasmas with very broad current profiles.

A. Examine effect of broad, flat T_e profile on current (q) profile.
B. Examine effect of wall-localized E×B generated shear flow layer.
C. Disruptions with a full, close fitting shell.
D. (+FY13 NB fueling) MHD stability, β_N limits with very close-fitting conducting wall.
E. (+FY13 NB fueling) Plasma rotation and stability with very high momentum input (40 A NBI into 0.8 m^3 plasma).

5 - Employ energetic particle beams, plasma waves, particle control, and core fueling techniques to maintain the current, and control the plasma profiles.

A. Test control of density, pressure profiles with fueling profile, reduced edge particle source.
B. (+FY12 beam-based diagnostics) Validate, verify models for density-gradient turbulence.
C. (+FY13 NB fueling) Test pressure profile control with full beam fueling, negligible wall/edge particle source.
D. Test novel fueling techniques such as molecular cluster injection.

6 - Develop normally conducting radiation-tolerant magnets for low-A applications.
No significant contribution.

Appendix C. Opportunities for Pegasus ST Research Contributions w.r.t. ReNew Goals (Ray Fonck)

A. Pegasus Capabilities: Base program + ARRA-funded (as of FY12)

I - Existing and planned with proposed minimal resource increase
$I_p \leq 0.3$ MA; A ~ 1.15 – 1.3; R_o ~ 0.25 – 0.45m; $B_t \leq 0.3$T; κ ~ 1.5 – 2.5; $\tau_{discharge}$ < 0.05 sec; l_i ~ 0.2 – 0.5; I_N > 10; $P_{HHFW} \leq 0.8$ MW; double-null divertors; Plasma-guns for point-source helicity injection; active I_p, R_o, Z control; high-field (> 10 T)) central solenoid; variable $B_{TF}(t)$; independently programmable poloidal field coil sets; edge plasma biasing.

Existing and developing diagnostics capabilities
120 channels of magnetics, including wall-current loops; 32-ch SXR array; 32-ch bolometer; 1 mm interferometer; multipoint Thomson scattering @ single time; passive ion spectroscopy; equilibrium j(r,t) via Hall probe array; scanning Langmuir probes; High-speed imaging; scanning VUV spectroscopy; visible bremsstrahlung; edge probe arrays for helicity injection; EBW radiometry; SXR PHA.

B. Proposed additional capabilities: FY12

$P_{EBW} \leq 1$ MW @ 2.45 GHz

C. Proposed additional capabilities: FY13

DNB and spectroscopy for $T_i(r,t)$ and flows diagnostics,.

Contributions to Addressing ReNew Thrust elements, keyed to capabilities

1 – Exploit and understand magnetic turbulence, electromagnetic waves, and energetic particles for megampere plasma current formation and ramp-up.

A. Scalable non-solenoidal formation via helicity injection
   - Develop understanding of point-source helicity injection (HI) for projection of non-solenoidal startup of ST to ~MA scale in NSTX-class experiment
   - I_p ~ 0.3 MA demonstration with resolution of critical elements: a) Verification of scaling model for I_p limits; and b) Determination of physical processes which set the edge current scale length, source impedance, and helicity dissipation rates (i.e., confinement)

B. Non-solenoidal startup using EBW, HI, HHFW, and synergistic combinations thereof
   - Verify predictive models for integrated startup in larger experiments.
   - Use of high-power EBW to grow and sustain ST plasma

C. Ion dynamics in presence of strong reconnection processes during HI + EBW, HHFW
   - Ion channel contributions to HI-plasma helicity and power balance
   - Flows and flow shears resulting from strong edge biasing.

2 - Develop innovative magnetic geometries and first-wall solutions such as liquid metals to accommodate multi-megawatt per square meter heat loads.

No significant contributions.

3 - Utilize upgraded facilities to increase plasma temperature and magnetic field to understand ST confinement and stability at fusion-relevant parameters.

A. Confinement and transport in nonsolenoidal startup plasmas
- Helicity dissipation and losses in reconnecting, HI-driven plasmas
- Confinement and losses in RF-assisted startup and growth
- Impurity properties of HI & RF startup and growth plasmas
- Minor contributions to general confinement
    - Complement database at extremely low A and B_t.
    - Possibly compare to LTX if Li-vapor deposition added

B. Confinement properties of combined HI and EBW-heated plasmas

C. Quantitative assessment of ion dynamics with strong magnetic reconnection
- Added $T_i(r,t)$ and $v_\phi(r,t)$ measurements for ion channel energy and flows

4 - Implement and understand active and passive stability control techniques to enable long-pulse disruption-free operation in plasmas with very broad current profiles.

A. MHD and disruption boundaries at high I_N, high β_t, low l_i as A approaches unity
- Determination of internal stability and disruption boundaries with no wall

A. Peeling-ballooning studies exploiting high $j_{edge}(r)/B$ intrinsic to operation at A ~1.
- Validation of peeling-ballooning model for edge stability and ELM precursors,
    - j(r) and $P_e(r)$ profiles in edge plasma region
    - Effects of local shear and separatrix operation
    - Potential modification of j_{edge} via current injection
    - Possible extension to H-mode operating regime

C. Validation of peeling-ballooning models with critical flow-field measurements
- Modification of edge flows via biasing

5 - Employ energetic particle beams, plasma waves, particle control, and core fueling techniques to maintain the current, and control the plasma profiles.

A. Current channel control and evolution during formation
- Tests of understanding of formation and stability at $l_i \geq 0.2$, including j(r) manipulation via HI and OH formation schemes, and auxiliary heating
- Extend low-l_i plasmas to longer pulse length

B. Test EBW local heating and CD to manipulate j(r) for formation and maintenance

6 - Develop normally conducting radiation-tolerant magnets for low-A applications.

Generally minimal contributions; experience with high stress systems at A ~ 1.

Appendix D. MAST programme strengths & near-term plans in the context of the ReNew research elements

Program goals

- Explore the long-term potential of the ST as a CTF and/or power plant
- Advance key tokamak physics issues for optimal exploitation of ITER/DEMO

Plasma current formation & ramp-up

- EBW-assisted start-up at 28GHz (EU-US-JA collaboration)
- Merging-compression (using internal coils) – incl. study of reconnection process in collaboration with JA (where plasma merging is being studied using external PF coils).
- Design studies on use of iron core, retractable solenoid
- Ramp-up by NBCD (+ higher power proposed) and EBW-CD (18 – 20GHz proposed)

Divertor & PFCs

- Radiative detachment
- Divertor/first wall heat loads
- Mitigation of transient events (ELMs, disruptions) – see below
- Divertor Science Facility (cf. DIMES, DIII-D) for studies of erosion, deposition, and material injection/migration
- Super-X divertor with cryo-pumping (proposed).
- Increased TF and solenoid flux (proposed) will allow longer pulse, higher current operation*

** Also relevant to other research elements*

Confinement & stability

- Confinement scaling & transport understanding exploiting excellent suite of high resolution diagnostics (TS, CXRS, MSE…..)
- Turbulence studies using BES (+ 2D imaging system under construction), PCI (proposed)
- Edge turbulence (probes, upgraded Doppler spectroscopy system)
- L-H transition physics and pedestal stability/transport
- Impact of rotation on stability, effect of instabilities on rotation, NTV studies
- NTM island physics exploiting very high temporal/spatial resolution of upgraded TS system
- Impact of fast ions on beta limit
- Fast particle instabilities – fast ion interaction exploiting NPA, collimated neutron detector (under construction), FIDA (planned), high frequency Mirnov coils. Influence of fast particle instabilities on transport, NBCD.
- TAE damping exploiting active antenna array.

Stability control

- ELM control using RMPs (installation of additional coils in 2010 will allow n = 6 perturbations)
- Disruption mitigation by Massive Gas Injection
- Off-axis NBCD for q(r) control.

Maintenance of current and control of profiles

- NBCD. Higher power, more adaptable NBI system with on/off-axis and co/cntr capability (proposed)
- EBW-CD (proposed)
- Pellet injection fuelling; ablation physics and particle transport incl. interaction with RMPs. Impact of drifts & microturbulence on pellet deposition. New pellet injector (proposed)
- Divertor cryopump (proposed)

Brian Lloyd
Draft, 30 November 2009

Appendix E. Japanese ST program strengths & near-term plans in the context of the ReNew research elements

Program strategy – to broaden future ST options by innovative research in

- Particle control and plasma-wall interaction in steady state
- Compact ST plasmas at ultra-high beta

Plasma current formation & ramp-up

- EBW-assisted start-up (2.45GHz, 5GHz, 8.2GHz on TST-2, LATE, QUEST)
- Start-up assist with plasma gun (UTST, TST-2)
- Double-null merging start-up (using external coils, UTST)
- Reconnection heating (TS-4, UTST)
- CHI (HIST)
- Ramp-up by LHCD (TST-2, planned) and EBWCD (LATE, QUEST)

Divertor & PFCs

- All-metal PFC in steady state with open divertor (QUEST)
- High-temperature tungsten wall with closed divertor (QUEST, planned)

Confinement & stability

- Fluctuation measurements with probes and reflectometer (TST-2)
- Core fluctuation measurements with heavy ion beam probe (TST-2, proposed)

Stability control

- Stability control with CHI (HIST)

Maintenance of current and control of profiles

- NBCD. (UTST)
- LHCD (TST-2) and EBWCD (LATE, QUEST)
- CHI (HIST)
- Divertor cryopump (QUEST, proposed)

Yuichi Takase
Draft, 28 November 2009

Appendix F. High-priority research needs for ST high-gain burning plasma (Jon Menard)

The burning plasma ST (BPST) mission aims to explore the high fusion gain plasma conditions relevant to a tokamak/ST magnetic fusion DEMO in which the fusion alpha particle power dominates the external heating power, nearly all of the plasma current is self-generated by the neoclassical bootstrap effect, and the strongly coupled nonlinear physics of the plasma equilibrium, transport, and macroscopic stability are controlled in steady-state. The ST configuration is potentially attractive for this mission, as it could provide a reduced size, complexity, and cost path to a burning plasma. Preliminary conceptual design studies for a BPST find that a very compact device with major radius R_0 = 1.5m can access fusion gain Q up to 20 with high bootstrap fraction 70-95%, high β_N = 5-7 and β_T = 25-50%, and I_P = 9–17MA (100% non-inductive: BS+NNBI) assuming high confinement H_{98} = 1.5-1.8. High priority BPST research is outlined below organized by the elements of ReNeW Thrust 16, and a BPST overview presentation is included in subsequent pages to provide supporting information.

1. Plasma current formation and ramp-up

The shorter pulse duration and reduced neutron damage to the center post of BPST should enable use of solenoid for inductive ramp-up to 8-10MA. High Q~10 requires I_P=12-17MA, so achieving flux savings from non-solenoidal start-up is high priority.

2. Innovative magnetic geometries and first wall solutions

Mitigation requirements for high heat, particle, and neutron flux are comparable to those expected in fusion nuclear science facilities, but at reduced pulse duration (few 10^2s). For divertor heat flux reduction, magnetic flux expanded/extended divertors are high priority.

3. Test the understanding of ST confinement, stability at fusion-relevant parameters

$Q_{DT} > 10$ requires high $H_{98} \geq 1.5$. The underlying modes/scalings for electron transport are not well understood, and could make high H_{98} difficult to achieve, so e-transport is a high priority research area. Means of increasing confinement far above H_{98}~1 is required to achieve a compact device size with minimized auxiliary power. A low recycling wall is a leading candidate tool for confinement improvement and is high priority. Drive for Alfvénic modes may be strong since W_α / W_{TOT} = 5-20% and $v_\alpha / v_{Alfvén}$ = 5-8, so fast-ion transport predictive capability is high priority.

4. Active and passive control to enable long-pulse disruption-free operation at low l_i

Reduced gain (Q ~ 2-10) scenarios operate with $\beta_N \sim 4$, $l_i \leq 0.6$ near the no-wall limit, and higher gain scenarios (Q ~ 5 to 20 and above) approach the ideal-wall limit $\beta_N \sim 5$-7, $l_i \leq 0.4$. Thus, active and passive control of resonant field amplification and RWM instability is high priority.

5. Tools to maintain the current and control the plasma profiles

For low-medium density, NBI is likely sufficient to control profiles while sustaining I_P, but for high density/f_{BS} scenarios, NBI will not be sufficient for profile control, and core fueling is a leading candidate tool for density/pressure profile control and is high priority.

6. Develop normally-conducting radiation-tolerant magnets

Determination of the TF and OH coil shielding requirements and materials is high priority.

7. Extend the ST to near-burning plasma conditions

The BP goal is to extend the ST beyond "near-burning plasma conditions" to high fusion gain.

Overview of a Burning Plasma ST (BPST)

Jon Menard (PPPL)

For the US STCC Five-Year ST Research Priorities Whitepaper

December 2009

Mission overview

- High-gain ST burning plasma mission statement:

  "To explore the high fusion gain plasma conditions relevant to a tokamak and ST magnetic fusion DEMO in which the fusion alpha particle power dominates the external heating power, nearly all of the plasma current is self-generated by the neoclassical bootstrap effect, and the strongly coupled nonlinear physics of the plasma equilibrium, transport, and macroscopic stability must be controlled in steady-state"

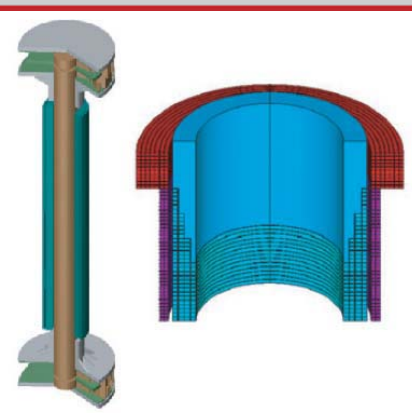
- Attractiveness of ST for burning plasma mission:
  - Assuming adequate thermal + energetic particle confinement, the ST could provide a reduced size, complexity, and cost path to a burning plasma. Preliminary scoping studies find:
    - A compact ST with R_0 = 1.5m can achieve high fusion gain Q up to 20
- The study presented here assumes pulse duration & neutron fluence can be limited to enable usage of conventional coils:
  - Shielded multi-turn insulated Cu TF coil to provide B_T = 2-2.5T
  - Shielded multi-turn insulated Cu OH coil to provide 10MA ramp-up I_P

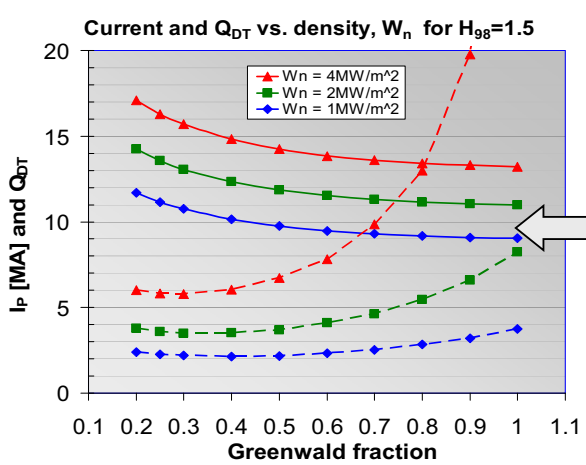
BPST parameters

• Aspect ratio:	A = 1.7
• Major radius:	R_0 = 1.5 (≈ ½ size of ARIES-ST R_0=3.2m)
• Elongation:	κ = 3.3
• Volume	V = 60m^3
• Toroidal field:	B_T = 2.4T
• Plasma current:	I_P = 9-17MA
• Normalized pressure:	β_N = 3.5-7, β_T = 15-50%
• Operating density:	Greenwald fraction = 0.2 to 1
• Cylindrical q	$q^* \geq 2$
• Sustainment:	100% non-inductive, f_{BS} = 50-95%
• Aux. heating and CD:	NNBI: E_{NBI}= 0.5MeV, up to 80MW
• Ramp-up:	½ swing OH solenoid: I_P = 9-10MA
• Pulse duration:	Few 10^2 seconds
• Confinement:	ITER H-mode scaling: H_{98} = 1.3-1.8
• Neutron wall loading:	W_n = 1-4 MW/m^2 (peak outboard)
• Heat flux:	P/S = 0.4-1.2MW/m^2, P/R=30-90MW/m

Inclusion of shielded OH transformer would provide ramp-up plasma current to access burning plasma regime

- NSST studies developed 2-layer OH design:
  - OFHC for outer, BeCu for inner + LN_2 cooling
  - Estimated OH field limit ~ 7 Tesla
- NSTX has demonstrated the benefits of optimizing the partially-inductively-driven current ramp-up:
  - H-mode + NBI heating/CD ***during ramp-up*** results in low effective Ejima coefficient (OH component) ~ 0.3
- Additional design effort is required to determine allowable dpa and shielding requirements

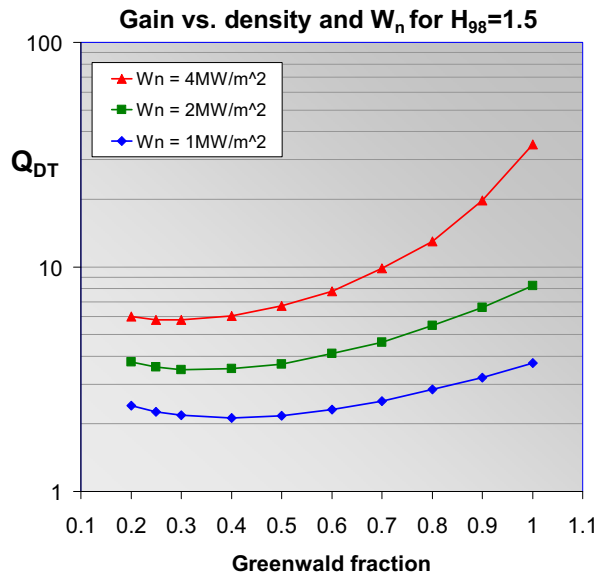


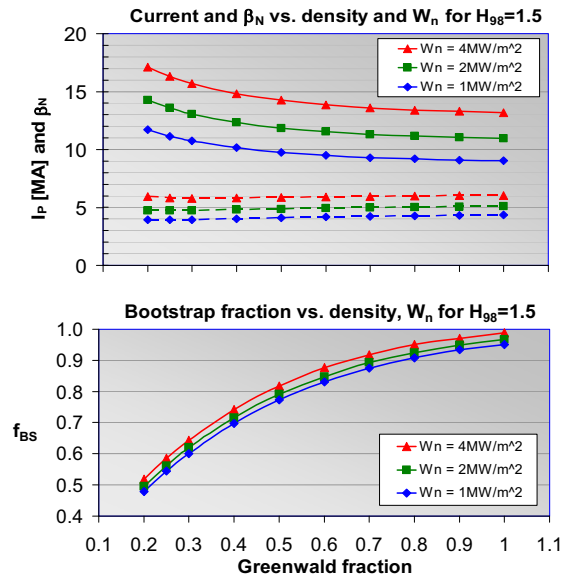


- BPST OH could provide single-swing flux of 5-6Wb to provide I_P = 9-10MA ramp-up current to reliably access Q_{DT}= 2-4 regime at β_N ~ 4
  - Higher power and β_N enable access to higher I_P and fusion gain through the bootstrap current

BPST would access Q_{DT} = 2-5 over a wide operating range, and Q > 10 at high current, density, β_N, and f_{BS}

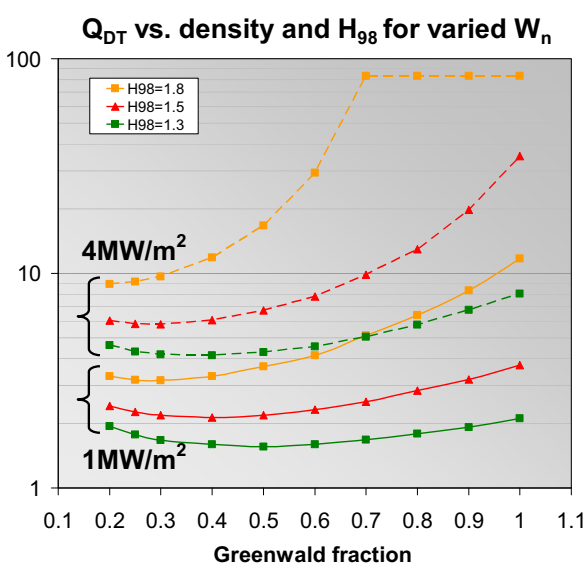
- Assuming ITER H-mode scaling and H_{98} = 1.5 → $Q_{DT} \geq 10$ requires:
  - Greenwald fraction ≥ 70% → ≥ 90% bootstrap fraction, I_P = 13-15MA
  - Operation above no-wall limit, near ideal-wall limit: $\beta_N \sim 6$
  - Implication: $Q_{DT} \geq 10$ requires advanced control of nearly completely self-driven plasma state operating near multiple stability limits (i.e. the mission)

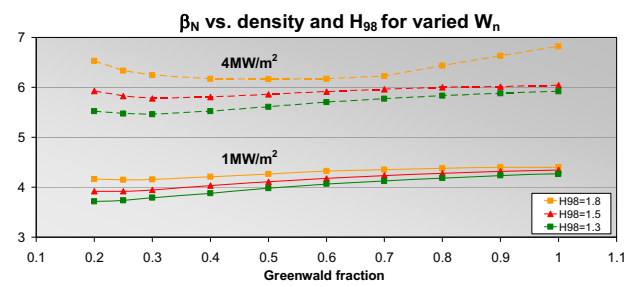


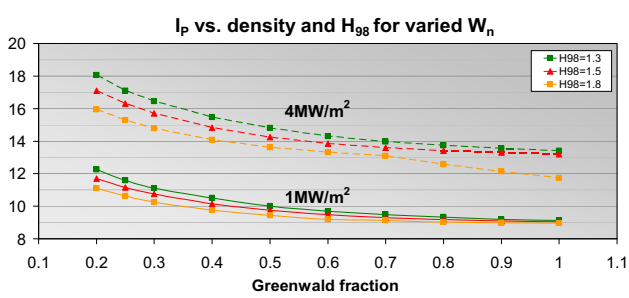


BPST could access Q_{DT} >> 10 with very high confinement H_{98}~1.8 and stable operation above no-wall stability limit

- Q_{DT} > 10 achievable with $H_{98} \geq 1.5$ for range of β_N = 4.5 – 7, I_P = 9-15MA
  - Most readily achieved at higher density
  - Q_{DT} > 10 ***not*** achievable with H_{98} = 1.3 over entire range of W_n = 1-4MW/m^2
- W_n = 1MW/m^2 peak outboard neutron loading → β_N = 3.7 – 4.5, I_P = 9 –12MA
- W_n = 4MW/m^2 peak outboard neutron loading → β_N = 5.5 – 6.8, I_P = 12 –18MA

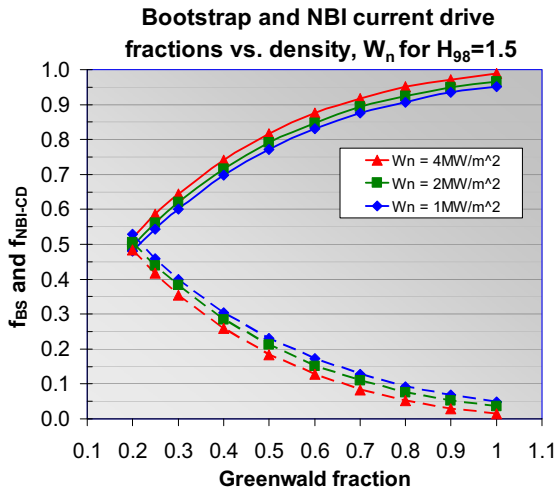




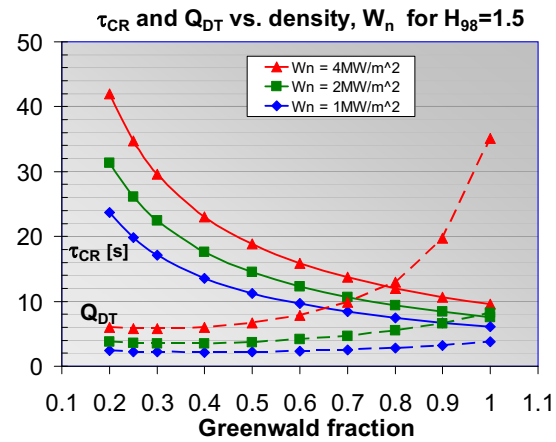


BPST pulse duration of few 100s is required to achieve current profile relaxation, test disruption avoidance

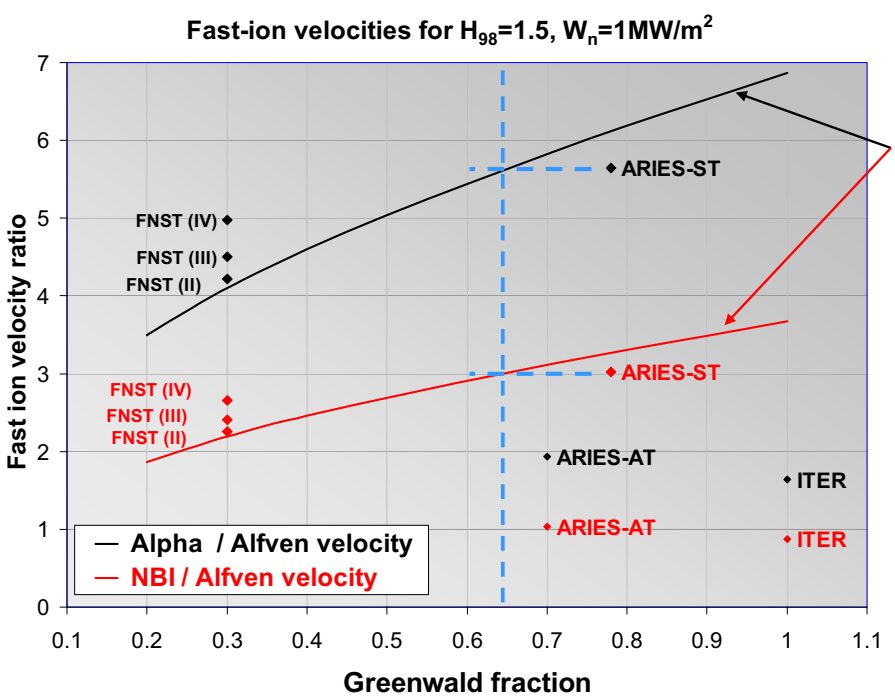
- OH only provides ramp-up current
  - All scenarios shown are sustained with 100% non-inductive CD
- BS current ranges from 50% at low density to 95% at high density
  - NNBI-CD provides remaining current
    (See slide 12 for NBI model)



- Characteristic times of interest:
  - J profile equilibration time ~ 3 τ_{CR}
  - Profile controllability time for demonstration of disruption avoidance ~ 10 τ_{CR} (lower bound)
  - 3-10 τ_{CR} → τ_{pulse} up to 400s at low n_e
- High Q_{DT} = 10-30 with f_{BS}=0.9-0.95 has reduced τ_{CR} ~ 10s → min. τ_{pulse} ~ 100s
  - Scenario controllability likely challenging



ST FNS and reactor (ARIES-ST) operate in fast-ion regime with v_{fast} / v_{Alfven} = 2.5-3× higher than conventional aspect ratio

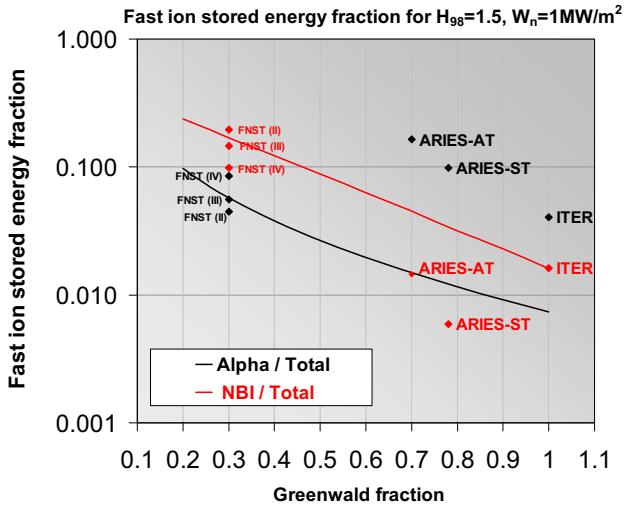


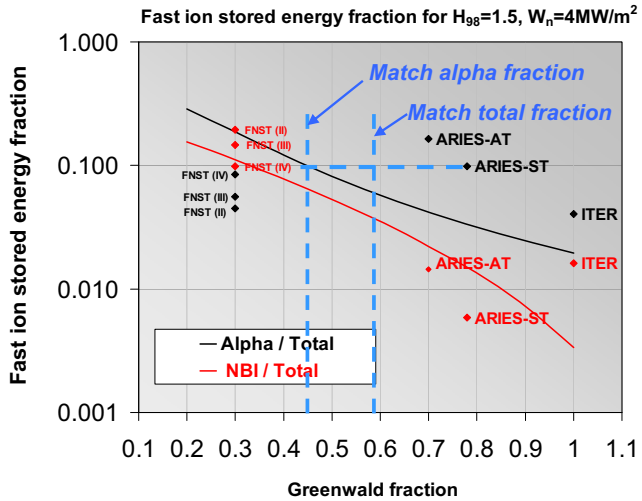
- BPST would access range of fast-ion velocities relevant to ST through density variation
  - Matches ARIES-ST ratios at n_e / n_{GW} = 0.65
- Key questions for BPST:
  - How does high fast-ion velocity impact fast-ion confinement in high-Q operating regime?
  - Is energetic particle physics of ST regime extrapolable to higher-A tokamak reactors?

BPST can access fast-ion content fractions approaching ST reactor by operating at reduced density, high $W_n \geq 4MW/m^2$

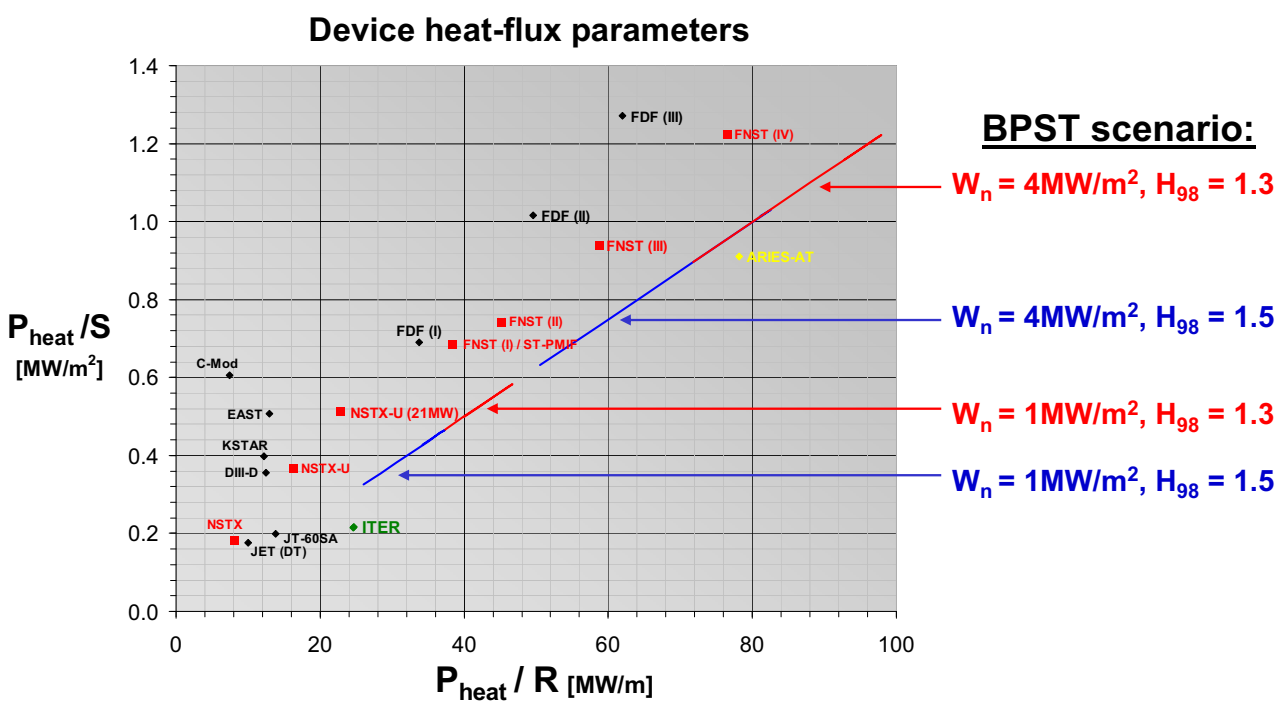
- Fast ion pressure is dominated by NBI for W_n=1MW/m^2 scenarios
  - This differs from reactor which is dominated by alpha population
- Fast ion pressure is dominated by alphas in $W_n \geq 4MW/m^2$ scenarios
  - BPST at n_e/n_{GW} ~ 0.5-0.6 has similar total fast-ion content as ARIES-ST

(BPST matches ARIES-ST NBI and α fractions at W_n=6MW/m^2, n_e/n_{GW}~0.4-0.5, Q_{DT}=50, β_N=7)





BPST will require mitigation of power exhaust comparable to that expected in proposed ST and AT FNS devices



- High confinement reduces heat-flux mitigation requirement by reducing total exhaust power at fixed neutron wall loading

Summary

- With adequate confinement and stability, the ST could provide a simplified and cost effective means of studying a high performance burning plasma with a reactor-like highly-coupled non-linear core plasma state. This would be achieved by integrating and exploring:
  - Large fraction of self-generated current and heating power
  - High normalized pressure above no-wall limit, approach with-wall limit
  - Potentially strong Alfvénic instabilities and fast-ion transport driven primarily by the fusion alpha population

- A successful BPST mission could reduce risks and increase fusion performance in a fusion nuclear science facility or fusion power reactor

0D (XL spreadsheet) model of operating points developed

(Similar to C. Neumeyer version developed for ST-CTF/NHTX, but simpler)

- **Special attention given to NBI-CD and fast-ion (NBI + alpha) pressure contribution**

- **NBI CD efficiency estimated including all trapping and slowing-down effects**
  - D.F.H. Start et al., Plasma physics, Vol. 22, pp. 303 to 316

Normalized current drive efficiency

$$\frac{I_T}{P} = \frac{\tau_s v_0 K_1 Z_b}{2\pi R(1+\alpha^3)\varepsilon_0}(1 - Z_b/Z_{\text{eff}} + 1.46\varepsilon^{1/2} A Z_b/Z_{\text{eff}}) \times [1 + (3 - 2\alpha^3\beta_1)(1+\gamma)\delta/(1+\alpha^3)^2] \times \int_0^1 x^{3+\beta_1}\left(\frac{1+\alpha^3}{x^3+\alpha^3}\right)^{1+\beta_1/3} dx$$

$$\alpha^3 = 0.75\pi^{1/2} m_e v_e^3 \left(\sum n_i Z_i^2/n_e m_i\right) \Big/ v_0^3$$

$$\beta_n = m_i Z_{\text{eff}} n(n+1)/2m_b,\ \delta = T_e/2\varepsilon_0,\ \rho = n_b/n_e$$

$$\tau_s = 3 m_e v_e^3 m_b/(16\pi^{1/2} e^4 Z_b^2 n_e \ln\Lambda)$$

- **NBI and alpha pressure derived from energy moment of slowing-down f(E)**
  - T.H. Stix, Plasma Physics, Vol. 14, pp. 367 to 384

Energy loss rate *i-collisions* *e-collisions*

$$\left\langle \frac{dW}{dx} \right\rangle = -\frac{\alpha}{W} - \beta W^{1/2},$$

$$\alpha \equiv 1{\cdot}30 \times 10^{-13} A Z^2 \ln\Lambda \sum_j \frac{n_j Z_j^2}{A_j}$$

$$\beta \equiv 2{\cdot}28 \times 10^{-15} \frac{Z^2}{A^{1/2}} \frac{n_e \ln\Lambda}{(kT_e)^{3/2}},$$

$$\left\langle \frac{dW}{dt} \right\rangle = \left\langle \frac{v\, dW}{dx} \right\rangle = 1{\cdot}39 \times 10^6 \left(\frac{W}{A}\right)^{1/2} \left\langle \frac{dW}{dx} \right\rangle$$

$$\tau = -\int_0^W \frac{dW}{dW/d_t} = \frac{t_s}{3} \ln\left[1 + \left(\frac{W}{W_{\text{crit}}}\right)^{3/2}\right]$$

Appendix G. Working Assumptions and Parameters of ST embodiment of FNS mission, research issues, and information supporting the research priorities (Martin Peng)

The key physics characteristics of an ST embodiment for the FNS mission are obtained from recent publications and discussions [1,2,3,4,5].

Stages*	**I**	**II**	**III**
Fuel	D-D → D-T	D-T	D-T
Pressure, $\beta_T B^2$ (%T^2)	22 → 24	86	133
Outboard fusion neutron W_L (MW/m^2)	0.01 → 0.25	1.0	2.0
Plasma current I_p (MA)	3.4 → 4.2	8.2	10.1
Safety factor q_{Cyl}	8.5 → 6.7	3.7	3.0
Toroidal beta β_T (%)	5	18	28
Normal beta β_N (MA/Tm)	2.0 → 2.7	3.8	5.9
Avg. T_i (keV)	5.4 → 5.6	10.3	13.3
Avg. T_e (keV)	3.1 → 3.6	6.8	8.1
Research questions organized by ReNeW Thrust-16 research elements			
1: Startup and ramp-up			
1a: Startup (formation)	- Can CHI, EBW, CHI+EBW startup large toroidal current with high Te?		
1b: Ramp-up	- Can CHI, EBW, NBI, CHI+EBW+NBI ramp up to full current with high Te and density?		
2: Divertor and PFC			
2a: Configuration	- Can extended or expanded SOL divertor be made to reduce peak heat flux to levels that permit very long pulse operations, even with uncertain SOL thickness?		
2b: Liquid metal surface	- Can high impurity influx be prevented? - What research will be required to provide long pulse high heat flux data?		
3: Confinement stability			
3a: Confinement	- Will $\tau_{Ei} \sim 0.7\ \tau_{Neo,i}$; $\tau_{Ee} \sim 0.7\ \tau_{ITER\text{-}H}$ remain sufficiently correct? - Can HIHM be maintained, even if τ_{Ee} improves as $\nu^* \rightarrow 10^{-3}$?		
3b: Stability (energetic particles)	Will sub-Alfvenic beam and some super-Alfvenic α cause unacceptable effects on fast ion confinement and J_{NB} profile?		
4: Stability control			
4a: Active	Will $\beta_N << \beta_{nwl}$ require active control?	Will $\beta_N < \beta_{nwl}$ require active control?	Will $\beta_N \sim 1.3\beta_{nwl}$ require active control?
4b: Passive	- Does disruption-free plasma operation require only passive control?		
4c: Resonant field error B_{error}/B_T	- Can B_{error} be made sufficiently small to avoid the need for active stability control for Stage I, II, or III?		
5: Maintain current and profiles			

5a: Energetic particle beam (co-E_{NBI}, kV)	Can 100-kV PINB be made continuous?	Can 240-kV NINB be made continuous?	Can 300-kV NINB be made continuous?
5b: Plasma wave	- Can EBW be applied to maintain q_{min} > 2 or 3 and avoid NTM?		
5c: Particle control	- Can extended or expanded SOL divertor + cryo-pump be adequate?		
5d: Core fueling	- Can high-field side pellet provide adequate fueling?		
5e: Continuous burn time	- Can the plasma be maintained continuously for 10^3 s, and in steps progressively for 10^6 s?		

*using A_{95}=1.5 case used in 2008 publications with R_0 = 1.2 m, B_T = 2.1-2.2T, f_{BS} ~ 0.5; A_{edge}=1.35 (A_{95}=1.5) designs, including D-T operation with the JET level plasma pressure, are also calculated during 2009 showing similar range of possibilities, based on the same set of systems analysis models and assumptions [3,4]. **$\beta_{nwl} = \beta_{no\text{-}wall\ limit}$.

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The relative priorities of the research questions identified in Appendix B are assessed here. The scientific Uncertainties (U) in the research outcome, and the potential positive Benefits/negative Impacts (B/I) of the U to the design and performance of the ST application to the FNS thrust are indicated approximately by H=3, M=2, and L=1. The Priority (P) with regard a research question is roughly proportional to the product of U and B/I (P = U*(B/I)/2).

Stage	**I**	**II**	**III**	**Research needed to understand the physics basis and adequately project to FNS conditions**	**U**	**B/I**	**P = U*(B/I)/2**
Research question							
1. Plasma current formation and ramp-up							
1a: Startup (formation)	Can CHI, EBW, CHI+EBW startup large toroidal current with high Te (~1keV)?			• Form ~0.5MA high Te plasma	M	H	H
1b: Ramp-up	Can CHI, EBW, NBI, CHI+EBW+NBI ramp up to full current with high Te (~1keV) and density (~10^{13}/cm^3)?			• Ramp up to ~1MA at high Te and density	M	H	H
2. Innovative magnetic geometries and first wall solutions							
2a: Configuration	Can XXD* be made to reduce peak steady state and transient heat fluxes to levels that permit very long pulse operations, even with uncertain SOL thickness?			• Produce and test XXD at ~1MA in HIHM** • Determine common database and understanding with Tokamak	M	H	H
2b: Liquid metal surface	What research will be required to provide long pulse high heat flux data using liquid metal surfaces? Can high impurity influx be prevented?			• Develop high heat flux liquid metal surface research program to determine risk and benefit • Verify the capability of liquid metal surface to eliminate impurity influx	H	Unknown	Unknown
3. Test the understanding of ST confinement, stability at fusion-relevant parameters							

3a: Confinement	Will $\tau_{Ei} \sim 0.7\ \tau_{Neo,i}$; $\tau_{Ee} \sim 0.7\ \tau_{ITER\text{-}H}$ remain sufficiently correct for ensuring the required plasma performance for Stages I and II? Can HIHM be maintained, even if τ_{Ee} improves as $\nu^* \rightarrow 10^{-3}$?			• Review database on HIHM from Tokamak and ST to establish working τ_{Ee} scaling • Investigate electron turbulence, transport, confinement in ST for $\nu^* \rightarrow 10^{-3}$ in HIHM	M	M	M
3b: Stability (energetic particles)	Will sub-Alfvenic beam and some super-Alfvenic α cause unacceptable effects on fast ion confinement and J_{NB} profile?			• Update ITER energetic particle database with ST data for HIHM	M	M	M
4. Active and passive control to enable long-pulse disruption-free operation at low l_i							
4a: Active	Will $\beta_N << \beta_{nwl}$ require active control?	Will $\beta_N < \beta_{nwl}$ require active control?	Will $\beta_N \sim 1.3\beta_{nwl}$ require active control?	• Complete active stability control research on ST and Tokamak for $\beta_N << \beta_{nwl}$, $\beta_N < \beta_{nwl}$, and $\beta_N \sim 1.3\ \beta_{nwl}$ for HIHM plasmas	M	M	M
4b: Passive	Does disruption-free plasma operation require only passive control for $\beta_N << \beta_{nwl}$, $\beta_N < \beta_{nwl}$, and $\beta_N \sim 1.3\ \beta_{nwl}$?			• Update ITER database with ST data on low disruptivity conditions removed from known stability limits for HIHM plasmas	M	H	H
4c: Resonant field error B_{error}/B_T	Can B_{error} be made sufficiently small to avoid the need for active stability control for $\beta_N << \beta_{nwl}$, $\beta_N < \beta_{nwl}$, and $\beta_N \sim 1.3\ \beta_{nwl}$?			• Review Tokamak and ST database on plasma conditions in the presence of very low resonant error fields (possibly $<10^{-4\text{-}5}$) • Create and study plasma conditions in the presence of very low resonant error fields (possibly $<10^{-4\text{-}5}$)	M	H	H
5: Tools to maintain the current and control the plasma profiles							

5a: Energetic particle beam (co-E_{NBI})	Can 100-kV PINB be made continuous?	Can 240-kV NINB be made continuous ?	Can 300-kV NINB be made continuous ?	• Develop engineering science basis for steady state positive ion neutral beam for ~100 kV, similar to KSTAR, EAST NBI concepts. • Develop engineering science basis for steady state negative ion neutral beam for ~300 kV, scaled down from ITER prototype.	M	M	M
5b: Plasma wave	Can EBW be applied to maintain q_{min} > 2 or 3 and avoid NTM?			• Test EBW H&CD at ~ MW level in HIHM plasmas	M	M	M
5c: Particle control	Can extended or expanded SOL divertor + cryo-pump be adequate?			• Scale down ITER cryo-pump prototype	L	H	L-M
5d: Core fueling	Can high-field side pellet provide adequate fueling?			• Scale down ITER pellet injector prototype	L	H	L-M
5e: Continuous burn time (s)	Can the plasma be maintained continuously for 10^3 s, and in steps progressively for 10^6 s?			• Apply modern simulation capabilities such as SWIM, GTLF, TLF23, TSC, FSP, etc. to refine estimates of FNS ST plasma projections	M	H	H
6. Develop normally-conducting radiation-tolerant magnets (not considered)							
7. Extend the ST to near-burning plasma conditions (included in the FNS and PMI missions)							

*XXD = eXtended or eXpanded-SOL Divertor; **HIHM = Hot-Ion H-Mode

A working scientific basis, for the recommended levels of U, B/I, and P in the above, is provided below:

1. **Plasma current formation and ramp-up:** Preliminary progress in achieving T_e ~keV plasmas during EBW assisted startup at ~100 kW power reduces the U to M. Since minimal or no solenoid is strongly required to accrue the advantages of ST for FNS summarized in Section 3, the B/I is raised to H. Research in this area at the level of 0.5-1 MW EBW power to 0.5-1MA current is estimated to be required to prove the physics principles for reliable extrapolation to the 3-4 MA current higher EBW power conditions required for FNS Stage I operation.
2. **Innovative magnetic geometries and first wall solutions:** XXD configurations are anticipated to be required to ensure attractive ST FNS designs (BI=H), and extend from well established magnetic divertor models. However, experimental tests at the MA level current are proposed to happen in the next few years with a reasonable prospect for success at the proof of principle level (U=M). In the case of 2b: Liquid metal surface, we can assume that the potential B is H if a system could be

conceived to deliver a fast flowing, high temperature, liquid lithium surface divertor in time varying magnetic field that handles high heat and particle fluxes continuously in a full fusion nuclear toroidal environment. However, the basic test data and design basis for such a system is absent at the present time to allow an estimate of the design impacts (I), rendering the ratio B/I unknown, which in turn, renders P unknown.

3. **Test the understanding of ST confinement, stability at fusion-relevant parameters:** Electron energy transport has been shown to dominate energy confinement in HIHM ST and tokamak plasmas, and to be consistent so far with the ITER H-mode scaling expression (U=M). Initial results of e-transport experiments suggest an improvement of tau-Ee with decreased collisionality (B/I=M). At high central beta-energetic/beta-total (up to 0.5), Alfven instabilities are regularly observed without preventing high confinement and beta plasmas (U=M). Energetic particle physics across the low and normal aspect ratios is a well studied and substantially understood topic (U=M). Moderate dispersion beyond classical transport of these energetic particles (evidenced by a beneficial sustained small rise in q_0) is also observed (B/I=M).
4. **Active and passive control to enable long-pulse disruption-free operation at low l_i:** By choosing plasma operating conditions far below all known stability limits for Stages I and II of a ST FNS operation, active feedback mode control is expected to be avoided via long time scale profile control. For stage III, experimentally observed beta values (when field error was minimized) are assumed, introducing a possible need for active control, These combine to suggest U=M. This uncertainty is not expected to bring large BI, as there is sizable margin to known stability limits (B/I=M). Very long duration tokamak operations at high temperatures without active control and disruptions have been shown to exist (U=M). However, the B/I of even rare disruptions would be H. It is well established that MHD instabilities (locked mode, RWM, etc) become less virulent or are absent as the normalized resonant field errors are reduced toward 10^{-4} (U=M). However, the potential B/I should be H as essentially all large tokamak and ITER designs have normalized field errors substantially higher than 10^{-4}.
5. **Tools to maintain the current and control the plasma profiles:** Continuous plasma operation for long durations eventually up to ~week should be extendable in principle from the ITER-AT and Hybrid scenarios, which are being simulated at present and tested in the near future in large superconducting tokamaks (U=M). However, the B/I for this uncertainty is H as this is fundamentally required by the FNS thrust. The levels of U and B/I for the other research sub-elements (NBI, plasma wave edge particle control, and core fueling) are judged to lead to P=M or lower.

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Appendix H. ST-based PMI/PFC Research Needs (Richard Majeski, Jean Paul Allain)

The long-term vision for an ST-based PMI program is to establish a *staged* development path for first wall and divertor plasma-facing materials solutions suitable for use in a DEMO nuclear reactor. A nuclear reactor environment will feature very long pulses, hot walls, and high neutron wall loading, along with strong plasma-material interactions. There is a critical gap in our understanding of how current PMI candidate materials extrapolate to an environment with a factor of ~4 higher power delivered to plasma-facing materials and components, compared to ITER. This extrapolation raises a number of PSI issues that include: heat exhaust (an increase of ~10^4× in plasma burn times in DEMO, with wall P/S ~ 1 MW/m^2), wall temperatures which may exceed 600 C, a need for consistent SOL empirical scalings, lack of experience with steady-state plasma-material interactions (over > 10^7 sec), requirements for transient energy exhaust, neutron damage (dpa > 50), and gross material removal rates exceeding current systems by ~10^3×. *To address these critical needs in a ~10 year time frame, a promising approach is a staged, systematic and comprehensive ST-based PMI/PFC mission*. Extensive theoretical and computational studies are also necessary to identify the complex material response behavior, and its scaling with environmental parameters likely to be encountered in a burning plasma device.

Within 5-10 years, the ST-based PMI mission research needs are:

- **Development and integration of in-situ PMI and edge plasma diagnostics:** Employ in-situ diagnosis to probe coupling of edge plasma and its interface with wall and divertor material surfaces. For example, highly-spatially resolved SOL diagnosis of: n_e, T_e, ϕ, v_{par}, v_{perp}, T_i electron and ion distributions complementing surface-sensitive diagnosis in-situ or in-vacuo.
- **Controlled single-effect evaluation of plasma-surface interactions with solids and liquids under ST-relevant power loads:** Employ dedicated testing of samples in STs, and on test stands to understand solid and liquid surface coupling and evolution under plasma exposure. Test candidate systems in NSTX-Upgrade, employing divertor power loadings in the 10-20 MW/m^2 range, transient heat loads in the 1-5 MJ/s range, and particle fluxes in the range of 10^{22} m^{-2}s^{-1} up to 10^{24} m^{-2}s^{-1}, with strong divertor gas puffing.
- **Determine the operating temperature window of liquid-metal PFCs:** Evaluate both low recycling (lithium) and high recycling liquid metals as PFCs, in LTX and NSTX, for the purpose of selecting between the two options. Identify PMI attributes of both low and high recycling liquid metals including: particle (He, D) retention, MHD properties, sputtering, erosion, and redeposition, surface chemistry, and reactor-relevant operating temperature limits.
- **Investigate and mitigate MHD effects on flowing liquid-metal PFCs**: Employ magnetic test stands with ST-appropriate magnetic fields to develop flowing liquid metal wall solutions, at both high and low Hartmann number. Deploy the developed systems for testing in LTX and NSTX, first in sample exposure systems, then as full wall/divertor systems.
- **Coupling single-effect science to PMI with theoretical and computational modeling:** Validate and verify theoretical and computational models for solid and liquid materials, Use divertor and wall in-situ diagnosis in ST PMI environments to link test stand materials experiments with edge/materials simulation codes.
- **Develop and understand the fully integrated plasma-material interface:** Iteratively test, in collaboration with the PFC program, a spectrum of materials innovations (high-Z solids, innovative solids, high Z/lowZ alloys, high/low recycling liquids) in a fusion relevant PMI environment: 1) pulse length up to10's of seconds 2) hot walls, up to 600 C, and 3) wall heat exhaust levels of P/S ~ 0.5 MW/m^2, using NSTX-Upgrade.