Spherical Torus Community Input on Priorities, Gaps, and Opportunities of the U.S. ST Fusion Program for the ITER Era*

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1. Introduction

The Spherical Torus (ST) concept is a low-aspect-ratio tokamak magnetic configuration characterized by strong intrinsic plasma shaping and enhanced stabilizing magnetic field line curvature. These 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. ST devices can access a very range wide dimensionless plasma parameter space with toroidal beta β_t up to 40% (local $\beta \sim 1$), normalized beta β_N up to 7, plasma elongation κ up to 3, normalized fast-ion speed $V_{fast}/V_{Alfvén}$ up to 5, and Alfven Mach number $M_A = V_{rotation}/V_{Alfvén}$ up to 0.5. All of these parameters are well beyond that accessible in conventional tokamaks, and these parameters either approach or overlap with those achievable in other alternative concepts including the Reversed Field Pinch (RFP) and Compact Torus (CT). These characteristics therefore allow ST research to complement and extend standard aspect-ratio tokamak science while providing the prospect of low-collisionality, long pulse-duration, and well-diagnosed plasmas to address fundamental plasma science of interest to alternative concepts. Conversely, STs can be optimized by methods developed in the higher aspect ratio tokamaks. In this manner, the physics basis of the effects of finite aspect ratio (e.g. larger fraction of trapped particles) can be fully tested and confirmed. The ST Program goal for the ITER era (Section 2), informs the scientific and technical challenges of the ST (Section 3), and motivates key scientific and technical questions, which in turn help define the research needed to address them (Section 4). Present and upgraded ST experimental capabilities, including possible new facilities, aimed at achieving the

U.S. ST Program goal, would strongly complement ITER's burning plasma physics mission, and accelerate the development paths of all fusion concepts toward a practical fusion energy source.

2. ST Program Goal for the ITER Era

Three broad themes of scientific and technical questions that must be resolved to proceed to Demo were identified in the 2007 FESAC report entitled "Priorities, Gaps, and Opportunities: Towards a Long-Range Strategic Plan for Magnetic Fusion Energy." These themes are:
Theme A: Creating predictable high-performance steady-state burning plasmas.
Theme B: Taming the Plasma Material Interface.
Theme C: Harnessing Fusion Power.

The charge to FESAC on magnetic alternates requires that a goal for the ITER era is defined for the spherical torus (ST). To that end, the ST community has chosen a mission for the U.S. ST Program: **To develop compact, high-beta burning plasma capability for use-inspired R&D.** This mission is motivated by the questions defined by the above themes, and can be approached by the realization in the ITER era of the following goal for the ST Program: **To produce a sustained plasma fusion environment of high heat flux and high neutron fluence to enable the R&D that establishes the knowledge base for an attractive fusion energy source.**

This report presents the features of the ST that would allow this goal to be realized, identifies the key scientific and technical questions, defines the research gaps, and outlines the necessary research and new capabilities needed to resolve these questions and fill these gaps.

The ST has intrinsic strengths and engineering properties that enable it to address these three themes with a potential of lower physics risk and reduced cost compared to normal-A tokamaks. The shared physics basis between the ST and the tokamak provides an increased level of confidence when designing future ST experiments. This common physics basis further indicates that the ST has a significant advantage in achieving this goal, due to increased margins at high beta to known plasma stability limits, and increased confidence for obtaining driven-burn at moderate Q. Significant engineering and cost advantages are obtained through simplified, demountable and potentially cheaper copper magnets that in turn permit compact designs of reduced fusion power and tritium consumption while permitting remote handling of all activated components.

The proposed fusion environment would effectively address for the ITER era the fusion scientific and technical questions in the following manner:
Theme A: The ST is part of an extended tokamak physics basis. Leveraging this common physics will allow the realization of sustained, high-performance ST plasmas. Establishing the knowledge base required for the single-turn copper center rod TF magnet will advance ST Demo-relevant magnet engineering, materials science, and remote handling capabilities.
Theme B: The ST uses a compact design to provide a cost-effective capability for Demo-relevant heat flux, ultimately in the presence of very high neutron fluence.

Theme C: The ST uses a compact, modular, remote-handled, high-β design to provide a cost-effective capability for Demo-relevant neutron flux and fluence.

Results from existing ST facilities and achievable upgrades, together with predictive capabilities upgraded to encompass the low and normal aspect ratio physics including moderate Q operation, should establish the scientific knowledge base needed for this ST goal. Accomplishing this ST goal in parallel with success on ITER will allow the U.S and world fusion programs to proceed to practical fusion energy sources in a timely manner.

3. Scientific and Technical Challenges for the ST Program

The ST concept offers an attractive path to address the scientific and technological gaps identified by the Greenwald panel that are not being addressed by other concepts. The ST can address these gaps by achieving the goal of developing a compact, high-beta burning plasma capability for use-inspired R&D. This goal can be achieved by utilizing the intrinsic strengths of the ST to produce a sustained plasma fusion environment of high heat flux and neutron fluence to enable the R&D that establishes the knowledge base for an attractive fusion energy source (including ST power plant).

The commonality between the ST and the tokamak gives the ST a large physics database to draw on in designing future ST devices with a high confidence of success. The increased margin to known stability boundaries and compact nature of the ST could give it the ability to achieve a stable, driven-burn plasma state at lower-risk than the tokamak. Achieving the ST goal in parallel with success on ITER will accelerate the path to fusion energy by addressing all of the themes identified by the Greenwald panel in a timely manner.

There are some gaps in current knowledge that must be filled to achieve this goal, and the ST program is currently addressing these. These gaps by science topical area and the means to address them can be summarized as follows:

1) Current start-up and current drive: An adequate knowledge base to initiate, ramp-up and sustain the plasma current with limited or no central induction is required. A variety of initiation techniques have been tested and the physics understanding of these is being developed, but further progress is needed and being addressed with present experiments and planned upgrades. Techniques for non-inductive ramp-up and sustainment have been identified and partially tested, but modeling of current drive techniques with reactor-relevant conditions and demonstration of fully non-inductive ramp-up to full current and sustainment in an ST is required. Increased understanding of the effects of fast-ion modes on current drive efficiency is needed.

2) Macroscopic stability: An overarching research gap to fill is the physics understanding of instabilities in low aspect ratio, high beta plasmas at significantly reduced collisionality. This regime will accentuate trapped particle effects, and potential negative impact on plasma rotation, pressure, and discharge sustainment must be evaluated for confident extrapolation to ST fusion plasma devices. Control and/or avoidance of instabilities by direct external means or through the control of key plasma profiles appropriate for unique aspects the ST must be demonstrated to maintain steady-state profile conditions in the low collisionality regime. Understanding of these processes and their limitations is critical to determine the need and implementation for such systems in ST fusion plasma devices.

3) Turbulence and transport: While there is significant understanding of the ion transport and progress is being made in electron transport in current experiments, more work is needed on current facilities to understand how these will scale to larger devices. Impurity transport appears neoclassical in L and H mode. Momentum transport is anomalous, and further study is required to allow scaling to larger devices. New or upgraded facilities are needed to study all classes of transport at higher field and reduced collisionality.

4) Boundary physics and plasma material interface: The divertor heat flux has been adequately controlled in current ST experiments with high recycling/collisionality, and can be tested at low collisionality with improved particle control. Edge pedestal and stability physics at low-A are being studied in existing facilities, and the effects of low collisionality can be examined with upgrades. The experimental scaling of the threshold power for L-H transition is understood, but better diagnosis and expanded operating space of current experiments is required to understand the underlying physics for reliable H-mode access.

5) Integrated operation: Particle control is needed to test/develop high performance integrated operating scenarios at low collisionality, and this can be achieved in existing facilities with upgrades. Once adequate particle control is achieved, there is much remaining work to be done to understand the conditions which will allow high confinement, core and edge stability, non-inductive sustainment, and high divertor heat flux to be realized simultaneously. Near-term tests are required in which the optimized density, pressure and bootstrap current profiles are sustained at near steady-state values without the use of NBI or gas puffing for the bulk of core fuelling. Pellet and CT injection systems should be tested to assess their abilities for fuelling discharges with high levels of bootstrap current.

6) Fusion nuclear operation: The technical knowledge base for remote handling of activated materials is not sufficient, and will need to be advanced several fold in precision metrology to sub-cm scales of complex heavy payloads using existing facilities. Material science knowledge up to ~5 dpa for low activation ferric steel, and mechanical performance of TF conductor beyond ~5 dpa are available for designs that enable the ST goal. This is separate from the dose required in the test articles. The basic knowledge base exists to design the type of magnets (central TF conductor with MIC solenoid) required, but this knowledge base must be extended to account for the high neutron fluence environment.

These gaps motivate a series of scientific and technical questions that are described in detail in the next section along with the current state of knowledge and how they are being addressed in current and planned facilities. These questions can be answered largely with research on existing and upgraded facilities over the next decade. Establishing this knowledge base will allow fulfillment of the ST goal, which in turn will address the broader goal of achieving the necessary knowledge base needed to develop an attractive fusion energy source

4. Knowledge Base, Gaps, and Opportunities for Achieving the ST Goal

4.1. Current Start-up and Current Drive

4.1-i) Can the plasma current be initiated and ramped up to full values using limited or no central induction?
In order to realize compact ST-devices with burning plasma capability, a reliable means to initiate and ramp up the plasma current to full values without a central solenoid is highly desirable. There are options for using a small reactor compatible central iron core or a mineral insulated (MI) solenoid to provide limited induction during a DD startup phase.

Status of Knowledge Base: Co-axial Helicity Injection (CHI) has been used to initiate approximately 100 kA of non-inductive current, which was subsequently ramped to 600 kA in H-mode via central induction. Point-source (plasma gun) DC helicity injection has been used to initiate approximately 50 kA of non-inductive current and has also been coupled to central induction. Toroidal current has been generated using induction from PF coils located on the outboard side of the plasma in devices with internal PF coils. A combination of EBW/ECH heating with a small transient central induction succeeded in starting and sustaining > 50 kA current and Te~500 eV at an efficiency of over 0.5 A/W. The physics knowledge required to apply limited central induction to create substantial plasma targets is well understood.

Remaining Scientific and Technical Gaps: Increasing the CHI current level in present machines appears to be a hardware limit. Spheromaks operating with metal walls have demonstrated 500eV electron temperature plasmas using CHI. The importance of metal electrodes for CHI in STs and auxiliary heating to boost the initial temperature of CHI and plasma gun discharges is not yet tested. Theory indicates that CHI scaling for machines with higher toroidal field is favorable, but experimental data is needed to validate this result. The physics of plasma gun startup is less developed and being investigated to allow scaling to larger devices and coupling to other current drive schemes. The required density, bias voltage and current, fueling, and magnetic field evolution for plasma gun startup are being studied both experimentally and with numerical simulations, but the predictive capability needed to design plasma gun startup systems for future devices needs to be developed, The effect of conducting walls on Outer PF startup is lacking experimental data. The magnitude of RF power required for solenoid-free startup under condition where a large field null is not used needs experimental data in an ST geometry. The synergisms and tradeoffs between different startup methods, including limited OH induction, have not been studied.

R&D Opportunities: The underlying physics of helicity injection and EBW/ECH assisted startup is under active study in both STs and tokamaks. Demonstrations are required of: ramp-up to full current using some combination of CHI and RF techniques, plasma gun startup, outboard PF startup using ex-vessel PF coils, and limited central induction. All of these issues are either being addressed on existing facilities and with numerical modeling, or can be addressed with upgrades to existing facilities such as additional ECH/EBW power and all metallic divertor plates for CHI. A conceptual design study for reactor applications of those techniques including an assessment and or development of reactor compatible insulators (Re: Section 4.6) is needed Low induction field startup using induction from the CS is also being conducted on tokamaks. Further design studies for OH solenoid systems which can survive a high neutron environment are needed.

4.1-ii) Can RF and NBI power adequately heat plasma and drive current? To maintain the steady-state operating conditions required of future STs, non-inductive heating and current drive is required as in tokamaks. Pressure driven currents will account for a significant fraction of the required current, but RF techniques and NBI must be able to provide the balance of the required steady-state operating current.

Status of Knowledge Base

Since EC harmonics may not be accessible in future ST devices, EBW physics is being studied as an alternative means of current drive. Emission studies show up to 70% EBW mode conversion efficiency. HHFW coupling to ST plasmas is also under study, and up to 85% non-inductive current has been obtained in short pulse HHFW discharges. NBI current drive in the presence of fast ion transport needs to be understood to design systems for future devices. Up to ~ 70 % of non-inductive current fraction has been demonstrated on STs with a combination of NBI, RF and pressure driven currents.

Remaining Scientific and Technical Gaps

Full non-inductive sustainment with NBI and or RF has not yet been demonstrated on an ST and existing ST facilities lack sufficient fast ion confinement to test NBI ramp-up. The effect of energetic particle instabilities on NBI current drive efficiency is not adequately understood. RF-edge interactions are still an issue for fast wave heating and CD (including HHFW). The interaction of HHFW and NBI needs further study in the ST environment. EBW current drive will require further studies in STs.

R&D Opportunities: HHFW coupling and current drive needs to be optimized in H-mode plasmas. High power EBW heating and current drive must be investigated. A higher TF ST facility, or TF upgrades to existing facilities, would allow the study of the effect of AE activity on NB CD, and possibly access to more conventional RF techniques, such as ECCD, LHCD, or lower harmonic FWCD. Design studies of future devices are needed to determine if these devices can use LHCD or ECCD.

4.2. Macroscopic Stability

4.2-i) Can instabilities and their negative impact on plasma rotation, pressure, and discharge sustainment be sufficiently understood and reliably avoided or controlled?
4.2-ii) Can deleterious effects of energetic particle instabilities be mitigated?

The spherical torus (ST) magnetic fusion concept uses strong geometric alteration of the magnetic field to produce stable plasmas at relatively low toroidal field, and hence high toroidal beta, β_t. The primary goal of the ST macrostability research is the stabilization, avoidance, or control of modes that significantly reduce fusion production efficiency in magnetically confined plasmas. Deleterious effects range from reduced confinement to plasma disruption. Combined experimental observation and theoretical understanding of these modes, including energetic particle modes, leads to the highest confidence in reliability and extrapolability of techniques to reduce the negative impact of these modes on power reactor performance.

Status of Knowledge Base

Stability advantages of the ST have been demonstrated in plasmas at the megampere level, and an extensive database exists with high toroidal beta, $\beta_t \equiv 2\mu_0 \langle p \rangle / B_0^2$ up to 39%, high $\beta_N \equiv 10^8 \langle \beta_t \rangle a B_0 / I_p$ up to 7.2, high energy confinement (τ_E / $\tau_{E\ ITER\text{-}89P}$ > 2.5), and β_N to internal inductance ratio, β_N/l_i exceeding 11 in plasma with relatively broad H-mode pressure profiles, favorable for high beta stability. The highest β values are attained in plasmas passively stabilized by a dissipation mechanism related to plasma rotation, V_ϕ. Ratios of β_N to the ideal no-wall stability limit exceed 50%. It is significant that this high ratio has been produced at the highest β_N values. Active control of global n = 1 MHD modes has been initially demonstrated to stabilize high beta plasmas when passive stabilization is insufficient. Recent research is studying the active control physics, and identifying practical improvements to increase the reliability of active feedback. Multiple modes can enter the dynamics, and interaction of modes with each other and with feedback is being better understood. For example, n = 1 feedback at RWM-level frequencies while controlling the RWM can lead to kink modes that rotate, and saturate or damp. However these kink modes can be intermediaries for NTMs, which can saturate, reducing beta and stabilizing V_ϕ. This mode evolution suggest ways in which feedback control – of mode amplitude, beta, plasma rotation, and pressure and q profile – can be specifically used to reduce deleterious effects with high reliability. Significant NTM activity (e.g. 2/1 mode) causes beta saturation, loss of V_ϕ., and locked mode-induced disruption, so must be avoided or mitigated. High plasma shaping, with record shaping factors and elongation up to 3 has been demonstrated at low internal inductance, l_i. Low l_i operation is the most favorable for both passive and active mode stabilization, including n = 0 (vertical stability). Highly shaped, high β plasmas can largely avoid Type I ELM activity, however understanding and either stabilizing or mitigating ELMs when they appear is vitally important for future STs. ELM mitigation with high edge confinement and V_ϕ through lithium deposition is a recent favorable result. The application of non-axisymmetric fields has shown that ELM dynamics can be altered and the mode destabilized, indicating a potential method of control. Beta-induced resonant field amplification (RFA) by stable RWMs, saturated NTMs, ELMs, and error fields at low β cause drag torques on the plasma that need to be sufficiently reduced to maintain stabilizing V_ϕ, even in present ST experiments with large momentum input from NB heating, drag torques can stop plasma rotation and cause disruptions. ST experiments have found good quantitative agreement between experiment and the theory of non-resonant torques due to neoclassical toroidal viscosity. The $\delta B^2 \varepsilon^{1.5}/\nu_i$ scaling of this torque emphasizes its importance in ST geometry, and also points out the key importance of testing this theory at reduced collisionality more suitable for comparison with reactor plasmas. New diagnostics are providing the data needed to understand energetic-particle instabilities: accurate measurements of the mode eigenfunction, amplitude, and polarization, measurements of the fast ions including the spatial profile of confined fast ions, the velocity-space dependence of their distribution function and their losses, and background plasma parameters (such as the q profile) that impact stability. Recent data indicate that effects beyond the ideal MHD model are important and numerical models that include relevant thermal-plasma kinetic effects are under development. Much effort is also devoted to modeling of the nonlinear dynamics of the instabilities.

Remaining Scientific and Technical Gaps:

High beta ST plasmas have been created and sustained for times equal or greater than plasma resistive timescales. Reliable stabilization of these plasmas at sufficiently high beta, and elimination of the negative impact of stable/saturated modes on V_ϕ, pressure, and current profiles, needs to be demonstrated with sustained density and V_ϕ, and at plasma collisionality that is substantially closer to levels envisioned for ST burning plasmas devices. This unexplored and unique operating regime at low aspect ratio, high beta, and low collisionality amplifies neoclassical trapped particle physics, which can be favorable or detrimental to stability.

Remaining critical questions regarding passive mode stabilization center around understanding the mode stabilization physics. RFA by stable resistive wall modes, and/or the appearance of NTMs create torques on the plasma which reduce V_ϕ, typically ending in local rational surface locking or global mode locking and subsequent disruption. Also, plasmas with relatively high V_ϕ can also suffer unstable RWMs with significant probability. This new realization challenges past tokamak experience and simpler RWM stability theories that predict a simple critical rotation value above which the RWM is stable. Present research using MHD models with kinetic effects indicates the possible need for control of certain profiles, including V_ϕ and its shear. NTM marginal stability conditions (vs. rotation, kinetic parameters, beta) and the triggering physics (ELMs, ideal Δ' destabilization, RWM-Kink-tearing mode evolution) needs to be understood for both for 2/1 and 3/2 modes. Control of the current profile is highly desirable as in tokamaks but with different challenges: accessibility of RF to the plasma and a q profile that has less core shear and greater edge shear. Active control of RFA or RWMs with high reliability is needed to maintain steady-state plasmas and profiles. The occasionally observed and theoretically expected poloidal deformation of modes in high beta ST plasmas may defeat stabilization. The relative importance of multiple mode dynamics and stronger mode coupling at lower aspect ratio must be understood to optimize sensor, actuator, and feedback algorithm design. The thermal and current dynamics (including halo currents) of high beta disruptions need to be understood based on mode stability physics at low aspect ratio, to confidently extrapolate thermal and electro-mechanical disruption loads to future devices. Highly shaped plasmas ($\kappa \sim 3$) show great promise for improved fusion performance with consistent bootstrap current profiles. At sufficiently low l_i, the purely current driven kink (unstable at all β) will limit plasma operation, but this limit (far lower than at higher aspect ratio) is not fully investigated. Variation of the edge pressure, V_ϕ, and current density is highly desirable to test or control ELM stability at low collisionality. If large ELMs are stable in this operating space, the generation of small ELMs may be desirable to regulate confinement. If they are unstable, ELM mitigation may be possible by using simple, multi-purpose non-axisymmetric coil systems used for RFA suppression and global mode control. The unfavorable scaling of non-resonant field torques on the plasma with decreased ν_i is theoretically expected to saturate as high E_r is generated by these torques. The uncertainly in the scaling of mode stability and torque balance for steady-state V_ϕ as a function of ν_i show the critical importance of producing lower plasma collisionality as a next step in advancing macroscopic stability research for ST fusion plasmas. Currently, numerical simulations of energetic particle modes often under-predict the observed fast-ion transport. Agreement between code predictions and observed transport levels is an urgent task in code validation. It is likely that many unstable modes will be excited by energetic particles such as NBI and alpha particles in next-step STs. Measurements already show that transport changes when many modes are excited but these conditions are more difficult to model theoretically. Conditions with broad spectra of fast-ion modes need further study. Experiments show that the linear stability or nonlinear dynamics of some instabilities are altered by RF waves. Further research is needed to determine if waves can efficiently control fast-ion instabilities.

R&D Opportunities

Confident understanding for the modeling and design of an ST device capable of producing a driven burning fusion plasma is best served by a new, or upgraded machine with the capability of producing high beta plasmas with steady-state profiles at reduced collisionality. In addition, the ability to vary and control key profiles that affect stability (rotation, q, and their shear, and pressure, including edge gradients) is critical to theoretically study and experimentally verify the robustness of the established equilibria. A next-step device should have an integrated active control system to maintain steady-state profile control and produce such profile variations to the largest possible degree, in addition to global quantities such as plasma beta, particularly to determine what control systems will be needed for a power reactor. Diagnostics that provide sensors for these control systems are available using present technologies, but some may need to be demonstrated in closed loop feedback. Passive stabilization physics and active control of finite toroidal mode number RFA and global modes is being addressed in present devices, and could productively continue for a few years before significant device upgrades would be required for the most efficient experimental progress. Associated theoretical research is making progress in understanding the complex nature of the stability limits, including the specific role stabilizing dissipative resonances that will determine the correct V_ϕ profiles for stability. Upgrades for active control include neutral beam injection designed to maximize q and p profile control, greater non-axisymmetric field spectrum and flexibility for RFA control, RWMs including multi-mode control, ELM control and/or mitigation, and V_ϕ. control. NTM control is presently envisioned to be provided by raising and maintaining the minimum q value. Sensor coverage and control algorithms yielding high system reliability are important upgrade considerations. The physics of the control schemes in almost all areas are profound and not entirely understood, so theoretical research must accompany experiments. Improved local fluctuation diagnostics for measurement of the wave fields such as BES are needed for energetic particle modes. Carefully designed experiments on existing devices are needed for code validation. Incorporation of new physics into codes (especially kinetic effects) is in progress but needs to be benchmarked against experiment. Once validation in existing devices is complete, the parametric dependencies under conditions that are closer to those of next-step STs should be investigated. If necessary to achieve acceptable fast-ion confinement, control tools that alter the linear stability or nonlinear dynamics of the most dangerous modes will need to be tested.

4.3. Turbulence and Transport

4.3-i) Can electron turbulence be understood and controlled so that electron energy confinement is sufficient to achieve the goal?
4.3-ii) Can high rotation and ExB shear in ST plasmas be adequately understood to control and suppress ion turbulence and transport?
The status of electron and ion turbulence and transport studies in the ST are in very different states. The improvement of electron confinement requires both understanding of the causes of electron turbulence in present STs, and the development of tools for control in future devices. High rotation rates and E×B shear have already been used to reduce ion turbulence and transport. Here the challenge is additional understanding, which would allow reliable projections for future devices to be made

Status of Knowledge Base

In present-day STs, ion transport is observed to be reduced to neoclassical levels over most of the plasma minor radius in NBI heated H-mode discharges. This observation is in accordance with expectations for ion turbulence reduction by the strong **E×B** shearing rate observed in the ST. However, in strongly NBI-heated H-mode discharges with ion transport near neoclassical levels, electron thermal diffusivity can be a factor of 2 to 10 greater than the ion thermal diffusivity, and the electron channel dominates the total energy loss. In contrast to H-mode discharges, low density, NBI heated reverse-shear L-mode discharges exhibit strong electron internal transport barriers with core electron temperatures reaching 2 keV. The electron transport is responsible for the strong B_T-scaling observed in the ST ($\tau_E \sim B_T$), while the ion transport controls the I_p scaling ($\tau_E \sim I_p^{1/2}$). The transport in both species scales differently than in standard aspect ratio tokamaks. High-k scattering measurements at varied field, current, and magnetic shear in strongly-beam-heated monotonic q-shear H-modes show a strong correlation between reduced electron temperature gradient scale lengths and increased electron gyroscale fluctuations. It is also found that enhanced high-k turbulence is seen in discharges at lower toroidal field, consistent with the scaling of electron transport with B_T. However, it would be premature to conclude that there is a direct causal link between ETG turbulence and electron transport, and several other modes and mechanisms that can cause electron transport, such as microtearing and TEM modes, are under active investigation.. Similarly, experiments in low density L-mode discharges heated by HHFW heating, which is primarily deposited on the electrons and is localized to the plasma core, indicate that high k turbulence increases as the radial gradient scale length of T_e decreases. There is some experimental evidence for a reduction in electron gyro-scale turbulence with reversed magnetic shear, over a narrow range in wave number. Non-linear gyrokinetic simulations of high-k turbulence in the ETG range result in predicted electron heat fluxes that are consistent with those inferred from transport analysis of discharges in the electron temperature gradient region. There are also theory calculations, however, suggesting that microtearing may also cause electron transport in the core regions of monotonic-q NBI discharges.

In STs, it is found that momentum transport is anomalous, with the inferred momentum diffusivity much greater than the neoclassical momentum diffusivity, which is near zero due to the large fraction of trapped particles at low aspect ratio. This relation holds even in H-mode discharges where the ion thermal diffusivity is near the neoclassical value, which is much greater than that for momentum diffusivity. Inferred momentum pinch velocities are consistent with those predicted from low-k turbulence theory, indicating that, because of the high neoclassical ion thermal diffusivity, momentum transport can be a better probe of ion-scale turbulence than ion thermal transport in STs..

Additionally, impurity injection has been used to probe particle transport in both L-mode, and high power H-mode plasmas. In both regimes, the measured particle transport was near neoclassical values across the plasma radius, which is consistent with the measured ion thermal transport also near neoclassical levels.

Remaining Scientific and Technical Gaps

It is unclear if suppression of the ITG mode due to strong **E×B** shear will extend to next-step devices, since the toroidal field will be higher (though still lower than a conventional tokamak). At this time, gyrokinetic codes have not been fully benchmarked, and they are not routinely applied to experimental data. It is also unclear if differences in the scaling of electron and ion transport is an aspect ratio effect, a high-beta effect, an effect of collisionality or a low toroidal field (~ 0.5T) effect. Scaling to higher field, current, temperature (>5 keV) and to low collisionality comparable to values in next-step STs will require an upgrade to the existing facilities. In the plasma core, where T_e gradients are weak, high electron thermal diffusivities are observed, and do not have a theoretical explanation within the standard ETG framework. The roles of micro-instabilities other than ETG have not been fully explored; micro-tearing and intermediate wave-length TEM are two examples. Although there is some evidence for reduced electron-scale turbulence in reversed-shear discharges, the spectral range over which the turbulence was reduced is narrow, and this is not understood theoretically. Experiments to study the impact of recycling on electron confinement in experiments with lithium PFCs or evaporators have just started.

R&D Opportunities

Deployment of BES on STs for measurement of ion-scale turbulence ($k_\theta\rho_i < 1$) to characterize directly the effects of flow shear is needed. New facilities which extend these measurements to toroidal fields and plasma currents typical of next-generation devices are also needed. Higher plasma current will reduce the intrinsic level of neoclassical ion transport relative to the level of turbulent transport. Variable beam torque would allow an assessment of the flow shear stabilization of ion and electron-scale turbulence at toroidal fields and plasma currents relevant to next-generation STs. Additional measurements of short wavelength turbulence with high-k scattering are needed to understand the polarization of turbulence (k_r vs. k_θ), in order to better compare to ETG theory and assess the impact of radial streamers. Also, measurements of intermediate wavelength turbulence ($k_\theta\rho_i \sim 1$) in STs with BES and high-k scattering are needed to understand the possible role of the TEM in electron transport. Diagnosis of δB fluctuations (from microtearing and/or *AE modes) are needed to understand the role that magnetic fluctuations play in electron transport, especially in the plasma core. New and/or upgraded ST facilities to study transport at higher field and current, and lower collisionality, to bring parameters closer to next-step STs, are needed. Studies of transport and turbulence at $T_e > T_i$, with ECRH heating in tokamaks or HHFW and/or EBW heating in present generation STs are needed. Further experiments in low recycling regimes, with either cryopumped divertors or lithium PFCs are needed to confirm and quantify the changes in electron transport with reduced recycling. Momentum transport studies over a wide range of operating parameters will allow probing the effect of low-k turbulence.

4.4. Boundary Physics & Plasma Material Interface

4.4-i) Can the divertor system handle continuous and pulsed high plasma heat fluxes and control particles adequately in H-mode plasmas?

Spherical tori are compact systems by design, which can lead to concentration of power and particle fluxes at the divertor targets. Correspondingly power and particle handling, both steady and transient, represents a substantial challenge in predictability of future experiments. In general the modification of the edge magnetic topology is more pronounced in STs than higher aspect ratio devices; thus geometry (e.g. 1/R effects, $R_{out}^{mid}/R_{in}^{mid}$, etc.) plays a major role in various aspects of boundary physics and plasma-wall interactions.

Status of Knowledge Base

Spherical tori are compact systems by design, which can lead to concentration of power and particle fluxes at the divertor targets. High heat fluxes, in excess of 10 MW/m^2, have been measured in STs in certain conditions, and these have been mitigated with either high poloidal flux expansion (which has been varied from 5 to 25) and/or extra gas puffing to induce partial detachment (reducing peak heat flux by 60%), each without a significant reduction in energy confinement. The particle fluxes and recycling have been partly controlled in STs with standard wall conditioning techniques and with evaporated lithium coatings, resulting in modest density control (15-20%) in high confinement regimes.

An important facet of STs is that the safety factor profile at the plasma edge differs from high aspect ratio tokamaks, owing to geometric effects. Edge stability calculations (ideal MHD) have shown that these ST profiles should improve access to second stability in many shapes, leading to the prediction of higher pressure gradients and pressure pedestal heights at low R/a by factors of 2 or more. This prediction is still being tested in dedicated cross-machine comparisons. Specifically H-mode pedestal parameters comparable to higher R/a devices have been measured at intermediate and high collisionality ($\nu_e^* > 0.5$) in coordinated experiments. The pressure at the top of the pedestal is indeed limited by edge localized modes (ELMs) in these cases, and most of these ELMs have common features with those identified in higher aspect ratio devices, e.g. Type I and Type III ELMs are both observed. The recent success of using lithium evaporation to reduce edge collisionality ($\nu_e^* < 0.2$) has also lead to ELM-free operation.

Finally, most of the scenarios envisioned for future STs require a reproducible access to H-mode profiles and confinement. While easy access and a wide operational window has been demonstrated in present day machines, some of the dependencies of the power threshold (P_{LH})

differ from higher aspect ratio devices. Specifically P_{LH} appears to increase (decrease) with I_p (triangularity), as distinct from international multi-machine scalings.

Remaining Scientific and Technical Gaps

While acceptable heat flux control has been demonstrated in present day short-pulse machines, all of these techniques have employed a high recycling/collisionality scenario. The compatibility of these solutions with low edge plasma collisionality ($\nu_e^* < 0.1$) is untested, as is the resilience of the control to significant transient events, such as large ELMs (with up to 10-20% fractional stored energy loss per ELM) and internal reconnections. Moreover the physics of the heat flux width at the outer midplane is not well understood. In this regard, STs tend to have up to a 300% larger SOL width than predicted by multi-machine scalings, but the origin of this width and its relation to turbulent transport processes is undetermined. It is noted that the measured midplane SOL power flux widths are comparable in magnitude to higher aspect ratio devices, and that the projected unmitigated steady-state peak divertor heat flux of next-step STs with even these widths can be very high (10-40MW/m^2), requiring further work.

Low collisionality regimes are predicted to enhance pedestal performance and increase the edge stability limit for ELMs. However, the detailed impact of Lithium evaporative coatings (which has resulted in ELM suppression) on the edge collisionality profile has not yet been documented; in particular, Z_{eff} actually increases in certain regions of the edge plasma, which might offset the decrease in collisionality from increasing T_e and reduced n_e.

Multi-machine scalings have shown that P_{LH} increases linearly with n_e and B_t. The observations in STs of P_{LH} seem to correlate more strongly with $|B|$ than with B_t, which is consistent with the observed I_p dependence. In addition, the value of B_t or $|B|$ in the vicinity of the dominant X-point and the proximity to a pure double-null configuration strongly affects P_{LH}; however the physics needs to be isolated for proper extrapolation of P_{LH} in future STs, which in particular will have higher I_p and B_t.

R&D Opportunites

Existing STs can improve access to reduced collisionality by enhancing particle control capabilities, either by expanding wall coating programs (including the use of liquid Lithium), and/or by adding cryopumps. To fully uncover the resulting physics changes, additional diagnostics with sufficient spatial (< 1cm) and time resolution to study the H-mode pedestal structure and the divertor characteristics could be implemented in the same devices. In the area of SOL power flux widths, upgrades to existing facilities (e.g. B_t, I_p auxiliary heating power, pulse length) would extend the available data range to help identify the important cross-field transport physics that sets the width and project to next step STs with more confidence. The present projected unmitigated steady-state peak divertor heat flux of next-step STs can be very high (up to 40MW/m^2), i.e. comparable to the expected Demo-level heat flux. This would present R&D opportunities to develop Demo-relevant innovative divertor solutions.

A new set of diagnostics in existing STs is needed to isolate the importance of X-point phenomena in L-H transition physics. These could include divertor Thomson scattering and reciprocating probes. Finally, the candidate first wall materials for next step devices operating at low collisionality can be studied in both existing integrated facilities and single-purpose dedicated facilities.

4.5. Integrated Operation

4.5-i) Can high beta and high confinement plasmas be non-inductively sustained with high divertor heat fluxes?

The primary challenge of integrated operation is the simultaneous achievement and sustainment of multiple fusion-relevant plasma and device performance parameters. For tokamak and ST-based devices, "integrated operation" implies the control and sustainment of high plasma beta, high confinement of thermal and fast-ion species, full non-inductive sustainment of the plasma current for steady-state operation, and plasma facing components and pumping and fueling systems capable of handling and controlling high heat, particle, and ultimately neutron flux in steady-state. For the near term missions of ST-based devices, strong reliance on neutral beam heating, fueling, and current drive is assumed. This external drive reduces the required bootstrap current fraction compared to that of a power reactor.

Status of Knowledge Base

Substantial integration progress has been made in 1MA-class ST devices in the last decade. In these larger, hotter ST devices, the collisionality has been reduced to values sufficiently low to enable substantial bootstrap current drive. On present ST devices, integrated high-performance scenarios with non-inductive current drive fraction approaching 70% have been achieved. In these scenarios, up to 55% of the current is provided by the bootstrap effect, and the remaining 15% is provided by beam current drive. At lower density/collisionality, the beam current drive fraction can be increased, but the bootstrap current has thus far been observed to be reduced under such conditions, resulting in somewhat lower total non-inductive current fraction. The above 65-70% non-inductive current drive condition has now been sustained for over 3 current redistribution times with toroidal beta ~ 15-20%, normalized beta ~ 5-5.5, H-mode confinement enhancement (relative to ITER H-mode scaling) $H_{98\text{-}IPBy2}$ ~ 1-1.1, and with tolerable ELMs. Wall conditioning with evaporated Lithium has also recently led to high-performance discharges that are completely ELM-free. The bootstrap fraction, toroidal beta, and normalized beta achieved are near or at the values expected to be needed for the ST goals proposed for the ITER era. Passively cooled divertors are capable of handling the peak divertor heat fluxes of ≤ $10 MW/m^2$ in present short-pulse (< 2s) ST experiments. Taking advantage of knowledge and systems developed for tokamaks, present spherical tori are utilizing increasingly advanced plasma controllers for various processes including: fueling, beam and RF heating, boundary shape control, vertical control, error-field correction, and resistive wall mode control.

Remaining Scientific and Technical Gaps

The integrated high performance results described above are typically achieved in a high density condition in which the density evolves to a Greenwald fraction of 0.7 to 1. Present ST devices have not yet achieved density control through active pumping. Next-step ST devices are projected to require significantly lower Greenwald density fraction n_e/n_{GW} = 0.25-0.5 to increase the fraction of NBI current drive to enable fully non-inductive operation. Due in part to high density operation, present ST devices are operating at normalized collisionality (ratio of collision time to trapped particle bounce time = ν^*) 1-2 orders of magnitude higher than anticipated in future ST devices. Significantly reduced collisionality could impact several elements of future integrated operation. First, the impact of reduced ν^* on core and pedestal transport in the ST is largely unstudied. Further, while pedestal stability limits may increase with decreased collisionality, ELM size also likely increases, making small ELM regimes and/or ELM mitigation more difficult to achieve. Off-midplane non-axisymmetric field coils will be tested for resonant magnetic perturbation (RMP) control of ELMs in the near term, but ELM mitigation via RMPs has not yet been demonstrated in any ST. While Li can produce high performance ELM-free discharges, such discharges suffer from impurity accumulation, and wall coatings are eventually depleted and/or passivated. In addition to the challenge of heat pulses from ELMs, the unmitigated steady-state peak divertor heat flux of next-step STs is expected to be very high (10-40MW/m^2). Reduced ν_i may adversely impact locked mode and RWM stability and error field torques. Low density/collisionality operation will also increase the fraction of fast-ion beta, while the higher toroidal field of next-step STs will lower the ratio of NBI v_{fast} / $v_{Alfvén}$. Further, the more tangential injection of beams in next-steps (to increase NBICD efficiency and provide off-axis CD to maintain q_{min} > 2 to avoid low-n NTMs) could increase the fast-ion pressure anisotropy. All of these modifications to the fast-ion distribution function will impact fast-ion instability drive and associated fast-ion transport in ways that are not presently predictable. As for the tokamak, additional control tools are needed – for example for control of the current profile, divertor exhaust, and ELM stabilization. Integrated models of relevant physics responses, including many complex actuators and the role of secondary plasma responses such as bootstrap current are needed.

R&D Opportunities

Particle pumping for density control and reduced collisionality will be tested on existing devices using a liquid lithium divertor and lithium walls, and upgraded/future ST devices could implement divertor cryo-pumping. Increased toroidal field, current, and heating power in upgraded ST devices would also enable access to higher plasma temperature and decreased collisionality to study transport, MHD, boundary, and energetic particle research in parameter regimes much closer to next-step devices. Off-midplane non-axisymmetric coil systems could be tested and optimized for ST ELM control at reduced collisionality, and for enhanced error field and resistive wall mode control. Present ST results indicate that RMP fields can destabilize ELMs, potentially providing a tool for controlled purging of accumulated impurities. Actively cooled solid and/or liquid metal divertors for steady-state particle and power exhaust could be tested in upgraded or new ST devices. Tokamaks with actively pumped divertors can contribute substantially to design and operational control solutions to regulate density in STs. Additional and/or re-aimed neutral beam injectors could be implemented on existing devices to optimize the beam-driven current profile and to enable tests of control of the current and rotation profile needed for long-pulse operation of an ST. Increased physics understanding specifically for the elements of these control problems, coupled with development of accurate and validated models which are simple enough to permit timely control design and enable real-time execution are needed for the ST. AT research on sustained and controlled high performance can also be utilized to help extrapolate to next-step ST perfor

4.5-ii) Can the density, pressure and bootstrap current profiles be adequately controlled?
To reduce external power input and simultaneously increase fusion power output, it is necessary to operate future devices at optimized values of the density, pressure and bootstrap current profiles. Plasma control systems during this phase cannot introduce strong perturbations that alter the optimized profiles. An important goal for integrated operation is to know the extent to which these profiles can be maintained at optimized values during steady-state operation.

Status of Knowledge Base

At present there are three fuelling techniques that have the potential for core fuelling. These are neutral beams, pellet injection and compact toroid injection. Neutral beams are routinely used in present machines for core fuelling high-performance discharges. NBI is applicable to near-term machines but their role for fuelling and momentum injection will significantly diminish as alpha heating and bootstrap current drive begin to dominate future plasmas. Pellet injection is used on most large machines and is being developed for ITER. It has been used to fuel H-mode discharges but not yet used to fuel quasi steady-state AT type discharges. CT fuelling has the potential for arbitrarily varying the fuel deposition location inside a device with simultaneous toroidal momentum injection. Recent work has shown a strong inward pinch effect on the density at low collisionality. However, devices that operate at values of density close to the Greenwald density still show flat profiles. The suggested scaling for the predicted density profile has a $-0.117\log(\nu_{EFF})$ dependence, but a stronger $-4.03*\beta_{Toroidal}$ dependence. This strong beta scaling is especially relevant to STs.

Remaining Scientific and Technical Gaps

Because gas puffing will no longer be a viable fuelling tool for next-step ST and beyond, and because of the diminished role of NBI in the post next-step ST phase, there is an important research gap to fill in developing or testing a flexible fuelling system. Depending on the particle transport and the extent of density pinching achieved in these devices, the balance of fuelling needs to provided by a combination of pellets and or CTs. Here additional data is needed on the predicted density peaking as the collisionality is reduced to <0.1 and at high toroidal beta, and at values of the density $\sim 0.5n_{Greenwald}$. Data is also needed on fuelling quasi steady-state AT discharges in tokamaks and high bootstrap current fraction discharges in STs. For CTs, data is needed on its capacity for density profile control and the extent of toroidal momentum it could inject.

R&D Opportunities

Near-term tests are required in which the optimized density, pressure and bootstrap current profiles are sustained at near steady-state values without the use of NBI for the bulk of core fuelling. Pellet fuelling needs to be tested in plasmas relevant to steady-state operation of next-step STs. This involves injecting pellets into quasi-steady-state discharges with high levels of bootstrap current, an experiment that could be conducted in an existing large ST. Tests are also needed in conditions of next-step ST relevant pedestal electron temperature to demonstrate adequate pellet penetration and sustainment of transport barriers. Such an effort is currently underway in support of ITER. In a parallel effort new fuelling technologies, such as CT injection, should be developed and tested in present facilities to investigate their potential for profile sustainment and to investigate their potential for toroidal momentum injection. An understanding of the role of collisionality on density profile is needed at high values of the toroidal beta and at densities >0.5 $n_{Greenwlad}$. For a more complete understanding, such a test is also needed under simultaneous high density operation at values close to the Greenwald density limit, a test which may be possible in an existing tokamak.

4.6 Fusion Nuclear Operation

4.6-i) Can remote handling of activated components be adequately implemented to ensure high operational duty factor?

The ST goal presents exceptional remote handling challenges that are beyond the present state-of-art from fission, nuclear science and fusion experiences. High duty factor operation required for the ST goal is expected to be an order of magnitude in weight, complexity, metrology precision, and radiation beyond the anticipated capability of ITER. These challenges stem from simultaneous conflicting requirements, such as long reach, heavy and complex payload, and precise positioning and alignment of components in high radiation constrained spaces.

Status of Knowledge Base: Remote handling equipment and operation on JET and SNS provide a technical knowledge base directly relevant to the design and construction of similar systems needed by the ST goal. Remote handling R&D for the design and/or construction of ITER, GNEP and accelerator-target facilities (IFMIF, FRIB, ILC) is expected to advance this knowledge substantially toward the level required. Computational simulation of remote handling designs and operations, and advanced modeling of radiation effects (for pre and post plasma operation) on special materials used in remote handling equipment are available for application to the remote handling systems for the ST goal.

Remaining Scientific and Technical Gaps: Two areas of gap in knowledge base are identified for the goal relevant remote handling capabilities: 1) Precision metrology (sub-cm in multi-meter scales) for high load-bearing (tens of metric tons or more) and dexterous manipulators (translation and rotation), and 2) Radiation tolerant, tactile sensitive, feedback controlled manipulation capabilities of heavy and complex payloads under constrained access.

R&D Opportunities: Existing and planned new nuclear-capable facilities will add to the knowledge needed by the ST goal. Dedicated R&D capabilities are needed to test and establish the knowledge for 1) linear motion solutions to remote handling to reduce maintenance costs and schedule, 2) fusion system modularization that is time and cost efficient, and 3) radiation tolerant remote handling materials and designs.

4.6-ii) Are fusion nuclear radiation tolerant materials available to enable adequate device performance and lifetime?
The ST goal will require an order of magnitude increase beyond the ITER design in the neutron dose encountered by all the components that face the fusion plasma core. While the ST goal aims to address the questions in Taming the Plasma Material Interface and Harnessing Fusion Power, radiation tolerant materials will be needed for the enabling components to perform adequately between scheduled maintenance shutdowns (such as for 6 months each year). These enabling components include the largely unshielded narrow center post that is expected to contain the TF center leg and a modest MIC-based solenoid largely without nuclear shielding, and nuclear shields and load-bearing structures behind the chamber components such as divertors, blankets, neutral particle injectors, RF launchers, and diagnostics.

Status of Knowledge Base: extensive data and knowledge for fission materials have provided strong basis for developing fusion material knowledge in decades. Recent materials research has arrived at a Low Activation Ferric Steel (LAFS) that is expected to be adequate for use as structure material up to a dose of ~5dpa. Advanced alloy and dispersion-strengthened copper is available with acceptable structural stability for a similar dose. Radiation tolerant insulation (high purity MgO) in Mineral Insulated Conductor (MIC) has been tested successfully in fission environment up to 10dpa for application during startup of the deuterium-only plasma before D-T burn. Extensive and growing capability in computational simulation of material response to irradiation can be extended to the ST goal.

Remaining Scientific and Technical Gaps: While a dose of ~5dpa on LAFS is expected to be adequate for the load bearing structure to enable the ST goal, radiation stability of the LAFS and influence of helium production (due to the more energetic fusion neutrons) on the microstructure evolution can only be adequately understood if studied and modeled up to ~30dpa, which is an order of magnitude beyond the anticipated surface dose of the ITER design. Mechanical performance of the central TF conductor and the MIC solenoid systems needs to be studied for doses greater than ~5dpa to increase the operational replacement for the ST goal from 2 toward 10 full duty factor years (~25 dpa). These additional knowledge, which may be provided by IFMIF, will be directly relevant to the design and manufacturing of the test articles.

R&D Opportunities: Irradiation studies of LAFS in fission environment at increased dose should be carried out under loading conditions relevant to the ST goal, together with the large scale computational simulation of material response, to enable predictions to ~30dpa. R&D should be carried out to establish the knowledge needed to design radiation and fatigue tolerant materials and material combinations for the center post conductor and the MIC solenoid, with adequate toughness in hardened state under the expected thermal and mechanical loads.

4.6-iii) Can the compact central toroidal field and possible solenoid magnets be operated reliably? The anticipated use by the ST goal of all-metal single-turn TF center post wrapped in a modest MIC central solenoid, in the absence of substantial nuclear shielding, presents new questions beyond the present state of art of jointed normal conducting magnetics that are common in magnetic fusion research.

Status of Knowledge Base: Broad knowledge base for jointed normal conducting magnets is available for design and construction of the magnets anticipated in the ST goal, including on-going ST experiments (NSTX, MAST, LTX, Pegasus, TST-2, etc.). Adequate knowledge exists among the industrial vendors for all metal, single-turn, demountable, water cooled, steady state center post. Commercial 3D CAD capabilities are adequate for the application to the ST goal.

Remaining Scientific and Technical Gaps: The following gaps in knowledge needed to design and build the compact central TF and MIC solenoid are identified: 1) Materials interface and designs to ensure uniform current distribution over large area high current possibly sliding joints and feeders. 2) Replaceable vacuum seal located near the TF sliding joints. 3) Requirements for remote handled assembly and replacement of such center posts. 4) Manufacturability of the TF center post using radiation tolerant normal conductors. 5) Center post water cooling capability and management of activated corrosion products. 6) Manufacturability of helium-cooled MIC central solenoid using LAFS and radiation tolerant normal conductors. 7) Instrumentation for the center post assembly for monitoring and fault detection.

R&D Opportunities: The following R&D capabilities are identified to fill the gaps: 1) High DC current tests integrating high current, low voltage power supplies, current feed and distribution, demountable sliding joints, etc. 2) Corrosion loops, in fission environment with prototypical materials, fluence and coolant. 3) Industry based study of fabrication knowledge needed for the center post manufacturing and assembly. 4) Remote handling tests for the center post payload of the order of 100 metric tons.

4.7 ST-relevant Innovations

There are Demo-relevant concepts in the high risk / high payoff category where the ST concept can serve naturally as a means to test and implement those concepts. Two examples mentioned here are the liquid lithium ST for a dramatic improvement in plasma confinement using liquid lithium to allow compact reactor system and the super X-divertor to handle Demo level heat flux. The liquid lithium ST is a natural extension of the liquid lithium programs being conducted on the on-going ST experiments where encouraging results are being produced. The super X-divertor concept maybe at least partially tested in the present ST facilities with some key upgrades and if proven promising, it could be naturally incorporated into future ST facilities including ST-Demo.

4.7.1. Liquid Lithium ST: The liquid lithium walled ST is designed to minimize wall recycling, and produce discharges with $T_e(a) \sim T_e(0)$. Small core electron temperature gradients are expected to eliminate the primary drive for anomalous electron transport. Recent theoretical analyses suggest that confinement times under these conditions would be characterized by ion neoclassical particle confinement times, or even longer timescales. While very long confinement times may not be advantageous in a conventional tokamak with modest β limits, in a high β ST such confinement could permit construction of a compact ($a \sim 0.5$m), very high gain, driven D-T

ST with fusion power in the 200 - 500 MW class, which relies only on modest NBI for heating and fueling. High gain would eliminate the need for alpha particle confinement, while the small size and high confinement would yield high burnup fractions, and require only a small tritium inventory. A liquid lithium wall would allow for continuous renewal of the PFCs, and removal of the tritium inventory. This research line points to an innovative paradigm for the implementation of fusion power, with small unit size fusion burn test units, and power sources. Research in both small and megamp class STs has produced enhanced electron confinement with lowered recycling. Discharges with very low global recycling have not yet been achieved. Research with more extensive liquid lithium walls and divertor targets is needed to explore this concept. Studies of particle transport in low recycling systems as a basis to predict helium ash removal is needed.

4.7.2. X-Divertor: The X-divertor employs reactor-compatible coils to generate an extra X-point downstream from the main X-point. As a result, the X-divertor greatly expands magnetic flux at the divertor plates, heat is distributed over a larger area, and the line length is greatly increased. The X-divertor has the potential to reduce the heat load at the divertor target plates in an ST by as much as a factor of 5-6. The high reduction factor is due to the high toroidal field gradient at the axis of an ST, compared to a conventional tokamak. This potential solution to the divertor heat load problem in STs would permit the construction of more compact D-T STs, and could in principle be combined with a low recycling divertor target. Design studies have been performed which show the potential of the concept, but the concept has not yet been implemented on an ST. Reactor compatibility of the additional coils must be established.

5. Summary and Conclusions

The ST program has identified a goal for the ITER era of developing a compact, high-beta burning plasma capability for use inspired R&D. Achieving this goal, in parallel with success on ITER, will provide a means to establish the knowledge base required for an attractive fusion energy source. The ST offers the unique capability of a low-risk option to achieve this goal through leveraging the extensive common physics basis with the tokamak combined with increased margins to known stability boundaries.

The present understanding of ST strengths and weaknesses leads to a series of scientific and technical questions to be answered for the ST program to proceed to this goal. These questions fall under the topics of current startup and current drive, macroscopic stability, turbulence and transport, boundary physics and plasma material interface, integrated operation, and fusion nuclear operation. Much of the current research in the ST program is focused on answering these questions, and the answers will have broad applicability that will benefit other fusion concepts.

The research gaps that must be filled to answer these questions have been identified, and the strategy to fill these gaps is being developed. While the detailed strategy to fill all of these gaps has yet to be finalized, a great deal of progress can be achieved using existing experimental capability, achievable upgrades, and updated theory & modeling. As these questions are answered, any new capabilities that are needed to address the remaining gaps will be identified.

Appendix A – Operating and Planned ST Facilities

Presently there are three operating ST facilities in the U.S. – NSTX, Pegasus, and LTX. The National Spherical Torus Experiment (NSTX) at PPPL is a MA-class world leading ST facility with strong auxiliary heating systems (7 MW NBI, 6 MW HHFW) and comprehensive state-of-the-art plasma diagnostic systems. The Pegasus device at the University of Wisconsin is a medium size ST facility with the lowest aspect ratio capability with $A \geq 1.1$. Presently, Pegasus is investigating an innovative plasma start-up technique utilizing a point plasma source (plasma gun). If successful, the plasma gun technique can be readily adopted to larger ST systems including NSTX and eventually next-step ST. The Lithium Tokamak Experiment (LTX) at PPPL (formerly CDX-U) will investigate the potential confinement benefits of liquid lithium-coated very-low-recycling walls. Due to a strong collaboration between NSTX and LTX/CDX-U, lithium applications are being actively implemented on NSTX including lithium evaporation and a liquid lithium divertor target.

ST experimental programs have also emerged during the past 10 years in the U.K., Japan, R.F., Italy, Brazil, PRC, and Turkey. There are now 22 experiments operational or under construction worldwide. The MAST device in the U.K. is a MA-class ST facility with capabilities and research programs that are in many ways complementary to NSTX. From the outset, NSTX has been actively collaborating with MAST in a number of research areas including Electron Bernstein Waves, energetic particles, and transport physics. There are also a broad range of the International Tokamak Physics Activity (ITPA) joint experiments being conducted on NSTX and MAST covering nearly all science areas of interest to ITER. The GLOBUS-M in Russia is a medium size ST with the emphasis on RF and NBI auxiliary heating. There are also several many smaller operating ST devices including TST-2, LATE, TS3/4, CPD, and HIST in Japan, SUNIST in China, and ETE in Brazil. A medium class long-pulse device QUEST and smaller high beta UTST are being constructed in Japan. Both MAST and NSTX are planning upgrades to increase the toroidal field strength toward 1 Tesla and plasma current toward 2 MA with substantial increase in the heating power and pulse length. These upgrades will enable NSTX and MAST to significantly reduce the physics engineering gaps between present and next-step ST facilities.

Appendix B – Support from Conventional Tokamaks

The DIII-D tokamak is pursuing a broad range of issues common to the ST. Macroscopic stability is under study in the areas of NTM control with ECCD (NTM physics under investigation with NSTX, and potentially controllable with EBW in STs), kink (RWM) control with rotation and active coil stabilization (in participation with NSTX), and in ELM control with resonant magnetic perturbations with RMP coils (with design of such a coil being examined for NSTX). Studies of energetic particles (with NSTX participation) are documenting a wide range of Alfven Eigenmodes and their effects. Integrated operation of sustained long pulse DIII-D discharges is a major effort dealing with all of these aspects. Alcator C-Mod is also investigating topics of interest to the ST program, in particular the compatibility of high plasma performance with high-Z metallic divertor and first walls. Further, C-Mod also plans to pursue hydrogenic retention research with hot plasma-facing components; mitigation of high heat flux and the minimization of particle retention are important issues for long-pulse next-step ST devices. In addition to support from US tokamaks, the US has extensive participation in the International Tokamak Physics Activity (ITPA) working groups; this allows drawing on expertise on an international level.

* The ST community paper was prepared by the ST community input working group: Chris Hegna, Rajesh Maingi (a. ed.), Masa Ono (ed.), Roger Raman and the STCC members: Bill Dorland, Don Hillis, Rob LaHaye, Fred Levinton, Dick Majeski, Jon Menard, Martin Peng (Chair), Steve Sabbagh, Aaron Sontag.