Question 3a: How would the ST program address and resolve the most crucial scientific questions ahead of it for this goal – startup/sustainment, transport and boundary physics?

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Start-up/sustainment:

The baseline approach for startup is limited induction to a current level ≥ 0.5 MA. Two candidate radiation-tolerant systems exist: a mineral-insulated cable (MIC) solenoid, and a resistively-shimmed iron core. Both require additional qualification, but the use of MIC for signal cables, and other uses, has been studied for ITER, and is probably more mature, although considerable test and engineering effort is still required. Inductive startup is well understood, and the ST has low external inductance, which minimizes flux consumption compared to a larger AT. However, if noninductive startup techniques were available, the complete elimination of the central solenoid would provide up to ~25% in additional cross sectional area for the centerpost TF conductor. The design flexibility afforded by eliminating the solenoid could be used to reduce the recirculating TF power, or increase the TF for fixed geometry.

Therefore concept-enhancing candidate techniques for initiation and rampup of the discharge, possibly to multi-MA current levels, which would eliminate the OH solenoid, are under active investigation. These include:

1. Coaxial Helicity Injection: Up to 160 kA of closed flux current has been produced, and 100 kA coupled to inductive H-mode discharges. A demonstration of higher current discharges in present machines requires a replacement of the present low-Z divertor targets with high Z, lower sputtering material, and auxiliary heating to increase the electron temperature of the CHI-produced plasma.
2. Plasma gun startup: Up to 60 kA of toroidal current has been generated and coupled to induction. Research over the next 1-2 years will continue to examine the underlying physics of plasma gun startup. Additional work is required to develop plasma gun hardware compatible with future needs, which could be performed in parallel with current research.
3. Outer PF only startup, was demonstrated in START and MAST using internal coils. With ex-vessel coils, the effect of conductive structures between the external PF coils and the plasma needs to be understood. NSTX and the UTST device in Japan are investigating outer PF startup with external PF coils.

Synergism between these methods may further increase the startup current and should be studied, since increases in startup current will ease rampup needs. Following initiation, non-inductive rampup to the full current using NBCD and/or RFCD is required. Rampup with NBCD has not been demonstrated in an ST; here fast ion confinement at low current is an issue. NBCD will be extensively investigated with future upgrades to NSTX. Although ECH/EBW has been used to initiate self-bootstrapped discharges on MAST, LATE, and CDX, the only RF technique which has been used to conclusively demonstrate noninductive rampup (in an AT) remains LHCD, which may be feasible at the higher TF in next-generation devices.

The baseline approach for sustainment is NBCD + bootstrap, possibly supplemented by EBW/ECCD and or fast wave current drive. This approach is substantially the same as the AT, although fast ion redistribution during NBCD in the ST is less well understood. Sustainment for times on the order of a few current relaxation times will be performed in upgrades to NSTX and MAST. The study of sustainment and stability over many current relaxation times would benefit greatly from the availability of long, near-steady-state ST discharges, with discharge durations of many current relaxation times.

Transport:

In present spherical torus experiments, the energy confinement time in neutral-beam heated H-modes has different parametric dependences than it does at higher aspect ratio, scaling linearly with B_T but more weakly with plasma current ($I_p^{0.5}$). The ion transport is commonly measured to be near neoclassical levels, and the ion turbulence is apparently suppressed by large shear in the plasma flow. In contrast, electron energy transport is observed to be anomalous, and commonly dominates the overall energy loss.

At least four micro-instabilities are thought to play a role in anomalous electron transport in tokamaks and STs: micro-tearing modes (electromagnetic), collisionless trapped electron modes – CTEM (electrostatic), electron temperature gradient modes – ETG (electrostatic), and Global Alfven Eigenmodes – GAE (Alfvenic). Micro-tearing, ETG, and GAE modes have been correlated in a preliminary way with anomalous electron transport. Multiple instabilities may be present simultaneously, and isolating the effects of individual instabilities is difficult. The baseline approach for developing an understanding of the source of anomalous electron transport in the ST is to increase the operating toroidal field and plasma current, and reduce the collisionality. Higher magnetic field and current provide access to much lower collisionality, allowing suppression of micro-tearing modes. Higher magnetic field would provide access to reduced fast-ion instability drive and enable the reduction (possibly suppression) of GAE modes. Thus, access to higher magnetic field would provide control of the onset of electromagnetic and Alfvenic modes, and separate the impact of these modes from electrostatic modes.

Higher field and current would also reduce neoclassical ion transport by up to an order of magnitude (due to smaller orbit sizes and reduced collisionality) thereby enhancing the relative importance of anomalous ion transport in the overall ion transport. This capability would also provide new insight into the underlying causes of anomalous momentum transport (most likely ITG and/or CTEM) and the flow-shear suppression of ion turbulence in the ST. Reduced/suppressed ion turbulence is especially important for achieving a "hot-ion" H-mode regime for high fusion gain in next-step ST-based CTF devices.

Overall, the planned major upgrades of NSTX and MAST, which include replacing the center stacks in both experiments and the installation of a Liquid Lithium Divertor in NSTX, would greatly enhance the ability to isolate the roles of different micro-instabilities in anomalous electron and ion transport by doubling the achievable toroidal magnetic field and thereby increasing the range of magnetic field variation to over a factor of three and expanding the range of collisionality by up to an order of magnitude. Importantly, the minimum collisionality made accessible by the major upgrades would approach (to within a factor of two) the collisionality values expected of next-step STs assuming comparable Greenwald density fractions. Existing high-k diagnostics and planned low-k diagnostics (BES) will provide the ability to distinguish between the electrostatic CTEM and ETG modes, and BES and high-k may also be capable of measuring GAE fluctuations. In parallel, and coupled to the fluctuation and transport measurements, theory model validation will permit extrapolation to next-generation devices.

Regarding edge transport, there is limited data on particle and energy transport in the pedestal region. As in high aspect ratio tokamaks, the pressure gradient is thought to be limited by ideal MHD stability in STs. Globally particle confinement is good in STs, often leading to ramping density in H-mode. Coupled to these two observations, the pedestal top temperatures in STs are usually limited to a few hundred eV. With careful recycling and/or fueling control, however, higher pedestal temperatures have been observed. The change in transport under these conditions needs to be quantified with 2-D edge plasma calculations. Access to higher B_T and auxiliary heating power, and improved density control techniques, would clarify whether present pedestal parameters are limited by dimensional/dimensionless parameters, or aspect ratio itself.

The SOL transport has not yet been thoroughly assessed with 2-D models. Generally speaking, the required cross-field transport rates needed to match data are higher than at high aspect ration, but dedicated studies would be required for a more detailed comparison.

<u>Boundary physics</u>:

An ST invariably has a higher ratio of volume to surface area, a smaller major radius, and a shorter connection length from midplane to the divertor target (in the case of a conventional divertor), than an AT. Therefore, mitigation of what would otherwise be an unacceptably high divertor heat flux is required. The baseline approach to solution of this problem is to employ high flux expansion, which has been demonstrated on present STs for outer divertor heat flux control. The successful extrapolation of these techniques to higher auxiliary power and higher field STs must be demonstrated. A second solution is partial divertor detachment, which, in a conventional divertor, may be incompatible with maintaining high energy confinement and low density operating scenarios for efficient NBCD.

A concept-enhancing "inclusive" approach is the use of novel divertor geometries: the ones based on flux expansion and a moderate increase in field line connection length (the X-Divertor (XD) and the Snowflake, for instance), and the others (Super-X divertor (SXD)) that, in addition, locate the divertor target plates at larger major radius (allowing considerably larger connection lengths) to reduce the temperature and power density at the target. These approaches - in particular, the SXD - may allow partial detachment at low core density.

Another concept-enhancing approach is the use of liquid metals, including lithium. Liquid PFCs do not suffer structural damage from plasma exposure, and can facilitate heat removal. Thin layers of liquid lithium have exhibited very high power handling capability (~60 MW/m^2 for 300 sec) in preliminary focused electron beam experiments. Liquid lithium PFCs also have the potential for near-complete removal of the in-vessel tritium inventory.

There are also synergisms between novel divertors and liquid metal concepts. Deleterious liquid metal MHD effects are minimized by the XD and snowflake divertors. These are further reduced in the SXD, which could also aid in liquid evaporation issues. Liquid metals and the SXD also offer a potential solution to vexing material issues for high heat flux components, which must simultaneously withstand large neutron fluences for fusion applications. Liquid metals and novel divertors could also reduce transient heat pulse damage.

Significant additional development is required for liquid metal solutions. Issues of evaporation and sputtering, with associated window and insulator coating, and MHD motion due to large edge currents, must be addressed. An integrated liquid lithium wall + divertor solution in an operating high-power ST must be studied to understand synergistic effects.

The baseline approach for density control in future machines remains cryopumping, although in a nuclear device the cryopanels must be located out of the neutron field – just as with the AT. Cryopumps will be implemented for particle control in ITER, and the ITER design should be applicable to the ST. The main advantage of cryopumping is that the technology and physics are well understood. One disadvantage is the sensitivity of the optimum particle exhaust to strike point geometry. Lithium offers a potential alternative to cryopumps with much greater flexibility, capacity, and pumping speed, but continuous active replenishment of lithium for long pulse discharges is still an issue.

In general, all materials for the first wall and divertor face substantially the same challenges as for an AT, except that the compact divertor size exaggerates the problem. The ST may require, or at least would benefit from, divertor designs which allow operation at power densities exceeding those in ITER. The development of these systems and materials would benefit greatly from the availability of very long-pulse ST discharges for testing, with discharge duration comparable to a wall equilibration time.

Gas puffing can likely be used to fuel an ST, but it is known that gas puffing is inefficient (~10% fueling efficiency), and if used to supply tritium, the resulting wall retention could result in an unacceptable in-vessel tritium inventory. For a next step, non-nuclear ST, gas fueling may be sufficient as it is in JET, but for a DT machine, the need for core fueling does not differ from a tokamak. Baseline approaches include NB fueling and pellets. In addition, fueling with CT injection may have additional benefits (e.g. momentum injection and density profile control). Pellets and neutral beams are used on STs, but density sustainment using core fueling techniques with strong active particle exhaust has not been demonstrated, and must be tested in upgrades to NSTX and next step STs.

<u>Summary of research needs:</u>

Prominent research needs in the startup and sustainment area include further nuclear qualification, design, engineering and testing of ohmic systems. RF current drive techniques have not been tested at CTF-relevant toroidal fields in an ST (only a few have been tested at all in the ST); higher fields will affect the choice of technique. Integrated tests, modeling, and systematic demonstrations, of candidate inductive and noninductive systems which explore startup, rampup, and long-term sustainment are required. System choices for the ST differ from the AT, therefore the integration requirements differ. The need for integrated testing imposes specific requirements on control system development, as well as MHD and stability boundary research. Wall and divertor conditions (high recycling, reduced or low recycling – and fueling) also strongly affect discharge confinement and evolution, and therefore impose additional integration requirements.

The entire area of transport qualifies as a theoretical and experimental “research need”.

Prominent research needs in the boundary area include experimental tests of flux expansion to mitigate divertor loading at higher fields, theoretical models, and further design, testing, and engineering of novel divertor geometries. For solid wall/divertors, nuclear and high heat flux testing, retention/retention mitigation studies, and evaluation for pulse lengths of multiple wall equilibration times at CTF-relevant wall temperatures are needed. For liquids, design and engineering studies, as well as much more extensive experimental tests, including high power density tests for pulse lengths of multiple fluid replacement times, MHD effects, hydrogenic removal, and testing of fluid recirculation approaches, are needed. Testing of divertor pumping with cryopumps and lithium systems for multiple wall times is needed, as well as integrated testing with fueling systems. The effect of wall and fueling choice on confinement and MHD stability must be evaluated.