Design Innovations of the Next-Step Spherical Torus Experiment and Spherical Torus Development Path

M. Ono 1), M. Peng 2), C. Kessel 1), C. Neumeyer 1), J. Schmidt 1), J. Chrzanowski 1), D. Darrow 1), L. Grisham 1), P. Heitzenroeder 1), T. Jarboe 3), C. Jun 1), S. Kaye 1), J. Menard 1), R. Raman 3), T. Stevenson 1), M. Viola 1), J. Wilson 1), R. Woolley 1), I Zatz 1)

1) Princeton Plasma Physics Laboratory, Princeton, NJ, USA
2) Oak Ridge National Laboratory, Oak Ridge, TN, USA,
3) University of Washington, Seattle, WA, USA

e-mail contact of main author: mono@pppl.gov

Abstract. The spherical torus (ST) fusion energy development path is complementary to the tokamak burning plasma experiment such as ITER as it focuses toward the compact Component Test Facility (CTF) and higher toroidal beta regimes to improve the design of DEMO and a Power Plant. To support the ST development path, one option of a Next Step Spherical Torus (NSST) device is examined. NSST is a "performance extension" (PE) stage ST with a plasma current of 5 - 10 MA, R = 1.5, $B_T \leq 2.7$ T with flexible physics capability to 1) Provide a sufficient physics basis for the design of the CTF, 2) Explore advanced operating scenarios with high bootstrap current fraction / high performance regimes, which can then be utilized by CTF, DEMO, and Power Plants, 3) Contribute to the general plasma / fusion science of high β toroidal plasmas. The NSST facility is designed to utilize the TFTR site to minimize the cost and time required for the construction.

1. Spherical Torus Fusion Energy Development Path

The potential of the Spherical Torus (ST) configuration to enable attractive fusion energy was discussed in a number of recent papers.[1-3] The engineering feasibility of a single-turn center leg for the toroidal field coils was identified as a key element for attractive ST reactors [1], which in turn pointed to the importance of solenoid-free startup and sustainment of the ST plasmas. The use of single-turn center leg also provides the possibility of compact driven steady state burning ST plasma. These plasmas would have R ~ 1+ m [2,3] assuming standard ST physics performance (such as $\beta_T \sim 25\%$) to produce substantial neutron wall loading ($W_L \sim 1$ MW/m^2 or higher) at a modest total fusion power (~ 50 MW or higher). Such an approach can be a strong candidate for the Component Test Facility (CTF) (also called the volume neutron source or VNS) needed in the test and development of reliable fusion nuclear components [4] for a DEMO power plant.

The importance of CTF in the accelerated development of fusion energy was recently recognized more broadly in the U.S. fusion research community [5]. The higher end of the CTF performance can in principle include the potential for producing net electrical power under high performance plasma conditions, if high performance fusion nuclear components can also be developed (such as for $W_L \sim 5$ MW/m^2 or higher). One can envision that the CTF facility would begin the operation at the level of $W_L \sim 1$- 2 MW/m^2, and progressively upgrade the test components to handle higher W_L while improving ST plasma performance. The CTF device is therefore expected to satisfy stringent operational requirements [3], such as complete modularity of all fusion core components (including the single-turn center leg) to permit rapid change out for replacement under full remote conditions. The CTF device should also achieve the high neutron fluence (~6 MW-yr/m^2 or higher) required in the testing program.

The expected high tritium consumption required for component testing puts a premium on highly compact devices that maximize W_L while minimizing the total fusion power. Due to the limited supply of tritium anticipated for the next few decades, tritium breeding for tritium self-sufficiency becomes a critical requirement. For this purpose, the fusion neutrons lost to the center leg needs to be minimized while maximizing the tritium capture and breeding ratios of the outboard blanket components. This in turn requires the minimization of the aspect ratio of CTF [1]. Such a compact CTF with high fusion and external drive powers is expected to lead to very high plasma heat and particle fluxes on the plasma facing components. Plasma facing component testing under fusion nuclear conditions will be a key part of the CTF program.

An effective and accelerated development of fusion energy using key contributions from ST can therefore be envisioned as shown in Fig. 1. The ST development path starts with the on-going Proof of Principle (PoP) level ST experiments (e.g., NSTX, MAST) to establish the physics principles of the ST concept at the Ip ≈ 1 MA level. The next Performance Extension (PE) level ST experiment (NSST) with Ip = 5- 10 MA is needed to provide the physics database for the fusion plasma parameters, including burning plasmas with potentially high Q values (≥ 2). Early in its operations, NSST will test and develop non-inductive start-up and sustainment scenarios needed for an ST-based CTF. Once the CTF physics feasibility demonstration is achieved on NSST, the CTF engineering design and construction can proceed at a separate nuclear site. The NSST facility can then continue to explore more advanced ST regimes to raise the plasma performance in CTF and future power plants. The NSST facility can also continue to produce reactor relevant data to help optimize the design of DEMO. The initial CTF fusion blanket and other core components can utilize the ITER core technology and component designs, and benefit from the materials developed by IFMIF. Basic parameters of representative ST facilities are shown in Table 1.

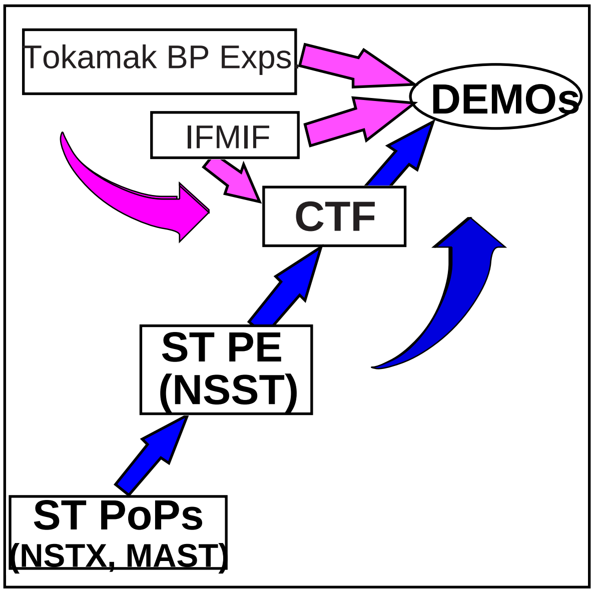

Fig. 1. ST development path

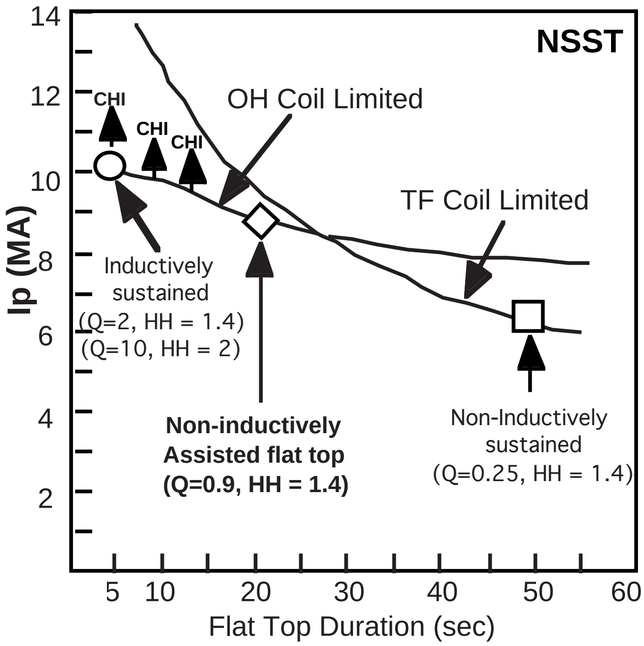

Fig. 2. NSST Operating Design Points

	NSTX	NSST	CTF
R(m), a(m)	0.85, 0.65	1.5, 0.94	1.5 - 2.0, 1 - 1.4
B_T (T), t (sec)	0.3 - 0.6, 5 - 1	1.1 - 2.6, 50 - 5	1.1 - 2.6, steady state
Ip (MA)	≤ 1.5	5 - 10	≥ 10
κ, δ	2, 0.8	2.7, 0.6	~ 3, ~ 0.5

TABLE 1

2. Physics Basis and Mission of NSST

The ST research has recently made substantial progress in a broad range of physics topical areas [6]. Plasma energy confinement times significantly greater than the ITER-98-pby2 H-mode scaling (HH ≈ 1.4) have been obtained on NSTX and MAST using neutral beam injection (NBI) heating, in H-mode and L-mode, diverted as well as inboard-limited plasmas.[7]. The NSTX made significant progress on β front, achieved $<\beta_T> \approx 35\%$, $\beta_N \equiv \beta_T\, a\, B_T / I_p \approx 6$, and $\beta_{pol} \approx 1.4$ sustained over τ–skin [8], and β values exceeding by 30% of the no-wall β limits [9]. H-mode operation is becoming routine, and the observed power threshold is coming closer to the ITER-based H-mode scaling [10]. Based on these encouraging results, a next-step spherical torus (NSST) device design is being developed. NSST is envisioned as a "performance extension" (PE) stage ST with Ip A ≈ 8 – 16 MA, which is similar in Ip A to PE tokamaks such as JET, JT-60, and TFTR. The primary mission elements of NSST are to conduct spherical torus research at fusion plasma parameters to 1) Provide sufficient physics basis for the design of a compact CTF, 2) Explore advanced physics and operating scenarios with high bootstrap current fraction / high performance sustained advanced ST regimes, which can then be utilized by CTF, DEMO, and/or Power Plants, 3) Contribute to the general plasma / fusion science of high β toroidal plasmas.

3. NSST Device Design

3.1 -Basic Device Design Parameters: To guide in the selection of a design point, which can meet the requirements of the NSST mission, a systems code was developed and a parametric study was performed [11]. Many promising design points have emerged. The Tokamak Simulation (TSC) Code was used to validate the systems code findings. In Fig. 2, the targeted NSST parameter space is shown. The current sustainment regime at B_T = 1.7 T (τ-pulse = 20 sec ≈ 2 τ-skin) is ideal for investigating the CTF-like regimes at moderate Q. Here, a half-swing of the ohmic heating (OH) coil (from initial pre-charge current to zero) can start up the plasma. Another important research aim of NSST is to demonstrate multi-MA non-inductive start-up. To allow sufficient pulse time investigate such non-inductive scenarios, NSST can operate for 50 sec at 6 MA with B_T = 1.1 T. Finally, to explore wider ST plasma parameter space, the NSST device can operate in a purely inductive mode up to 10 MA with B_T = 2.7 T with a 5 sec flat top using the full OH swing, where Q = 2 can be achieved with HH = 1.4. This operating mode will enable an exploration of α-particle related physics in high β plasmas for τ-pulse ≈ 5 τ_E.

3.2. NSST Device Design Overview

To achieve the NSST mission, a flexible NSST device design was developed [12]. An isometric view of NSST device and a device cross sectional view are shown in Figs. 3 & 4. The magnets are liquid nitrogen cooled to allow long pulse as well as high performance operation. To facilitate timely progress for the NSST research program, an innovative ohmic solenoid is designed into the baseline center stack design to deliver sufficient flux for 10 MA operation with full swing and 6 MA operation with half swing. As shown in Fig. 4, with the in-board PF-1 coils, a strong shaping capability (κ = 2.7, δ = 0.6) is incorporated in the design. Together with the tightly fitted stabilizing plates, NSST is designed to access the high β_N (high bootstrap current fraction) regimes relevant for attractive power plants. The outboard PF coils are placed sufficiently far from the plasma to reduce local shape distortions. The device is designed with a removable center stack to facilitate remote maintenance and allow for the possibility of future upgrades. The present NSST design utilizes the TFTR site with the peak electrical power of 800 MW and energy per pulse of 4.5 GJ and long pulse auxiliary heating and current drive systems (30 MW of NBI and 10 MW of RF). The existing tritium related facilities and experience can be also utilized for alpha-physics research.

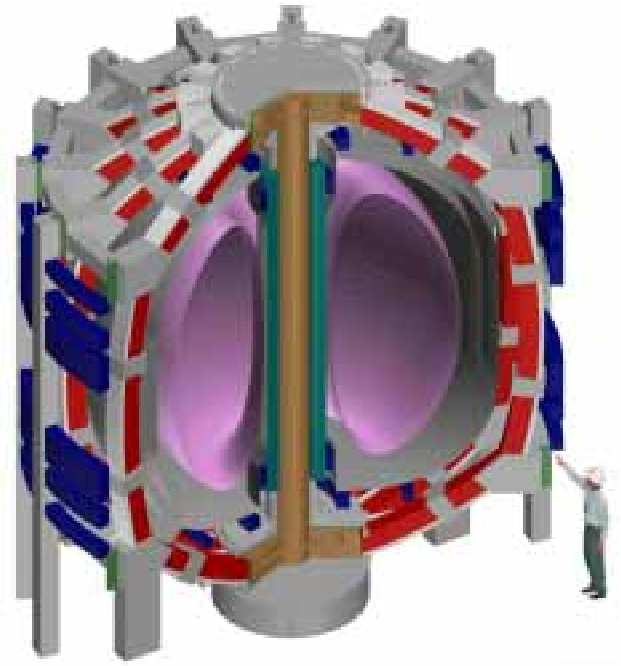

Fig. 3. Isometric View of NSST

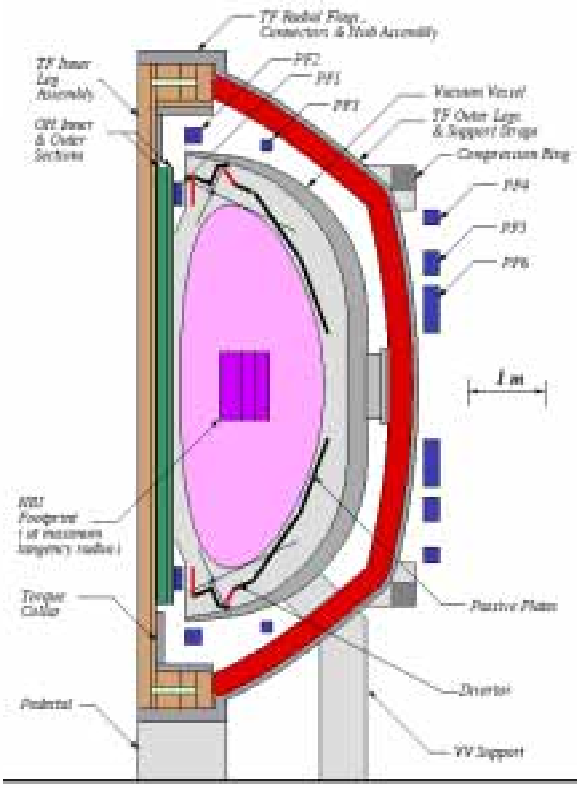

Fig. 4. NSST Device Cross Section

3.3 Toroidal Field (TF) Coils - Like NSTX, NSST features a demountable TF coil design, which permits the "center stack" of the device (i.e. the TF Inner Legs, OH Coil Sections, PF1a coils, and the inner section of the vacuum vessel) to be removed separately as an integrated assembly. The TF inner legs consisting of 96 standard OFHC copper (Cu) turns, of wedged shaped conductors, arranged in two layers, are cooled by liquid nitrogen (LN_2) via passages extruded in the conductors. Turn-to-turn transitions in the two layers proceed in opposite directions so as to cancel the net toroidal current. The assembly is fabricated in a fashion similar to NSTX, except a high temperature, high shear stress cyanate ester resin insulator is used. Torsional loads arising from the OH radial field crossing the TF current are reacted through the outer TF coil legs and structures via torque collars clamped around the inner leg assembly at the ends of the OH coil. The torque collars attach to the hub which, in turn, is attached to the outer TF coil legs and structures. The current density in the outer legs is relatively low and the temperature rise is less than 10^{o}C per pulse. They are cooled by the exit flow of nitrogen (gas initially; liquid at full cool-down) routed though extruded passages in the outer leg conductor. Radial flags and connectors are used to make the joints between the inner legs and the outer legs. The radial flags are wedged into a hub assembly to form a monolithic structure. The connectors are slightly flexible in the radial direction to avoid the development of a large radial force on the flags, and to allow the outer legs to rest against their support structure. As shown in Fig. 4, the shape of the outer legs is chosen such that the outward magnetic pressure due to the TF current crossing with the TF field results in a constant tension in the support strap, with minimal vertical tension imposed on the inner legs. Compression rings are used to adjust the constant tension shape to suit the desired height of the TF coil assembly. With the constant tension, moment-free shape, the outer legs and associated support structure can be made relatively flexible in the axial direction, thereby allowing the thermal expansion and contraction of the inner leg assembly without generating large stresses. The outer leg out-of-plane forces due to the radial component of TF current crossing with the vertical field of the PF coils are transmitted to the strap assembly via the compression panels and straps. The intrinsic torsional rigidity of the strap/compression ring structure is supplemented by mechanical keys which transmit torsional loads to the "cage" surrounding the machine, which is formed by PF coil support columns and the compression rings. Shear panels between the PF support columns will be added if further analyses indicates the need for additional torsional stiffness.

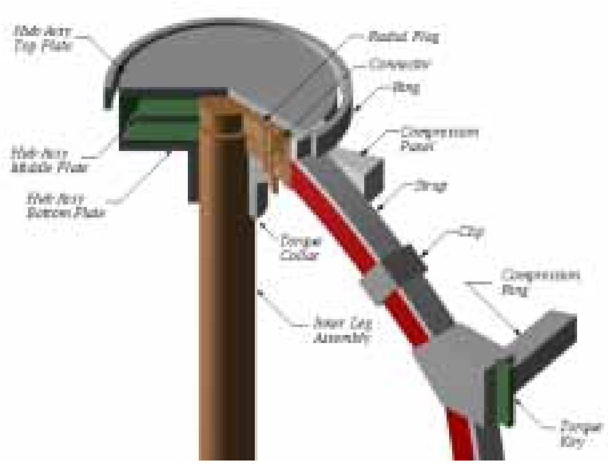

Fig. 5. NSST TF Coil schematics

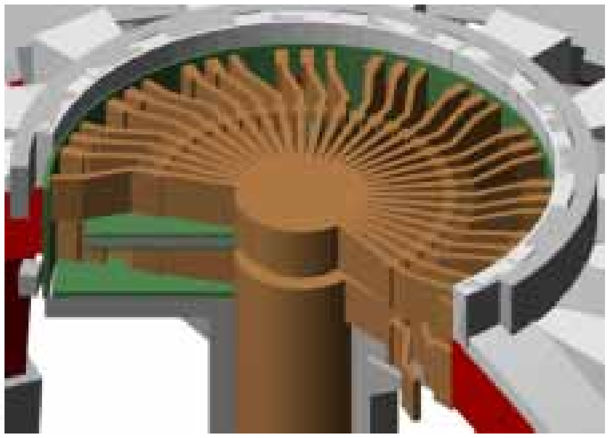

Fig. 6. TF Coil Joint Configuration

3.4 Ohmic Heating Solenoid and Poloidal Field Coils - A two-part OH coil is used, consisting of two concentric sections, with different current density in each section to increase the total available flux. The sections are connected in series and carry the same current per turn. The outer section uses standard OFHC copper (Cu) conductor, which is operated to its thermal limit. The inner radius of the outer section is chosen such that the hoop stress is at the allowable limit for copper. The inner section uses a beryllium copper (BeCu) alloy operating to its thermal limit. Due to the higher strength and lower conductivity of the BeCu, the inner layer is thermally limited before it reaches its allowable stress. The OH coil sections are cooled by LN_2 flowing through the annular regions between the OH and TF coils and between the OH sections. The bipolar swing of OH current is asymmetric about zero to exploit the higher strength of the conductors at cold temperatures during the first swing, with the ratio of the first swing of current to the second swing equal to 1.8. The PF Coil System consists of 6 coil pairs symmetric about the device midplane. Current per turn is 24kA in all circuits, based on the rating of the existing TFTR power supplies.

3.5 Vacuum Vessel and PFCs - A double walled vacuum vessel with integral shielding is used on NSST. The vessel is fabricated of 316SS. The inner wall is 3/4” thick and the lower stressed outer wall is 5/8” thick. Welded ribs are provided between the inner and outer walls to stiffen the structure. The inner space between walls is filled with 60% 316SS balls and 40% water. Ports are based on 16-fold symmetry. Four (4) midplane ports are included to accommodate the TFTR NBI. Eight pairs of 6” diameter ports are included to accept feed-throughs for an 8 strap RF antenna subtending $8*7.5^{\circ} = 60^{\circ}$. The remaining nine rectangular midplane ports are 24” wide x 36” tall. Sixteen 12” diameter ports are provided on upper and lower domes, total 32. An ample plasma access will be provided for plasma profile diagnostics to facilitate NSST research. The inner wall of the vacuum vessel (i.e., the center-stack casing) is formed by the 0.1875” thick Inconel “center stack casing”. Bellows assemblies and flanges are provided to allow for differential thermal expansion with respect to the outer vacuum vessel. The power and particle handling is a challenging issue for NSST and it is anticipated that the PFCs are actively cooled for heat removal. The same cooling path will be also used to heat the tiles for the bakeout. The initial PFC material will be graphite-based tiles due to the operational experiences, however more advanced PFCs can be considered for the DT operations.

4. Physics Capabilities and Opportunities

4.1 The NBI Induced Plasma Rotation - One of the long term goals of NSST research is to access the advanced ST regime. If successfully demonstrated, it could impact the operational scenarios of the CTF by supporting higher Q pilot-plant-like (even net-electric) operations. In addition, it would support the design of DEMO and power plants. To foster advanced ST research, we have chosen tangential Neutral Beam Injection (NBI) as a reliable means to impart toroidal momentum to the plasma together with tightly fitted stabilizing plates. In Fig. 7(a), the NBI top view is shown. In Fig. 7(b), the resulting toroidal rotation velocity calculated by TRANSP is shown as a function of the major radius using four co-injected beam lines. Here the toroidal angular momentum diffusivity is assumed to equal the neoclassical ion thermal diffusivity. As shown in the figure, the toroidal rotational speed could reach 600 km/sec or 35% of the local Alfven velocity. Since the rotational speed needed for the RWM stabilization is typically a few % of the Alfven velocity, we can consider placing one of the four NB injectors in the counter-direction to introduce additional physics flexibility. The remaining three NBI injectors will be installed in the co-direction at various injection angels and tangency radii. This will potentially allow finer control of the rotational velocity per given injected NBI power as well as control of the heating and current drive profiles and velocity sheared layer location for the ITB formation to control the pressure profile.

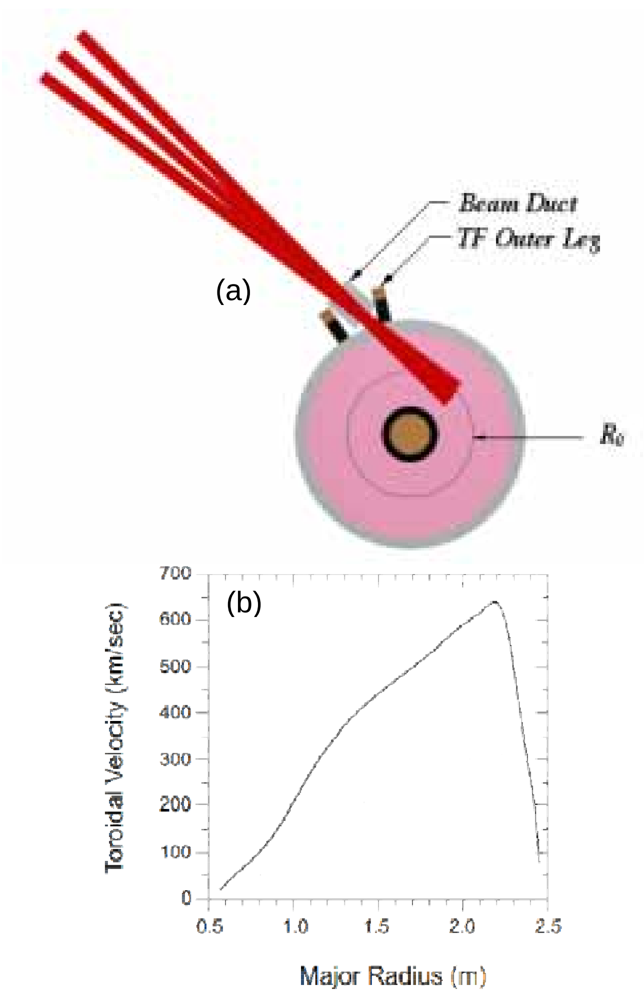

Fig. 7 NBI (a) NBI beam configuration . (b) Plasma rotation toroidal profile.

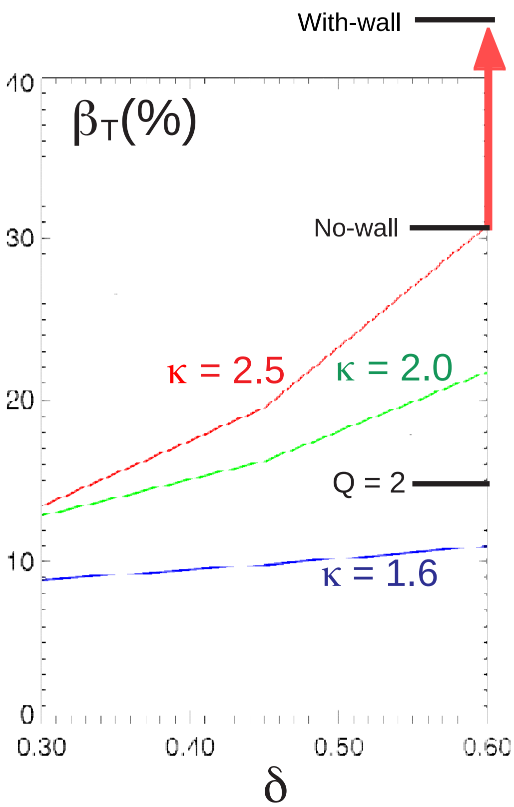

Fig. 8.Beta limit dependence on Plasma shaping

4.2 Dependence of Plasma Beta Limits on Plasma Shaping - The importance of high triangularity for plasma performance is shown in Fig. 8 where the no-wall beta limit for fixed bootstrap fraction ≈ 50% is plotted as a function of triangularity δ for various values of elongation κ for expected NSST parameters. As one can see from Fig. 4, the achievable no-wall β limit increases with κ and δ. It should be noted that the Q=2 high performance regime can be achieved with the plasma beta of only about 15% which is well below the calculated beta limits of up to 30 % with no wall. This shows that Q can be raised further (if confinement improves) without concern of hitting the β limit. If wall stabilization is possible on NSST, the toroidal beta limit increases substantially from 30% to 45%. The increased plasma beta is also desirable for increasing the bootstrap current fraction, which is an important element of the plasma sustainment in advanced ST scenarios.

4.3 Non-Inductive Start-Up - Non-inductive start-up research is a central topic for NSST, particularly, to demonstrate multi-MW level non-ohmically driven start-up current. At present, NSTX is investigating coaxial helicity injection (CHI) plasma current start-up, which has succeeded in driving about 400 kA of toroidal current with injection of 27 kA at about 700 V bias voltage [13]. Recently, on the HIT-II device, an experiment was conducted to show significant ohmic volt-second savings by the initial application of CHI [13]. While CHI is still under development, the trend from HIT-II to NSTX seems to show a favorable scaling of a factor of two improvement in current multiplication. The expected CHI requirement increases with the desired plasma start up current and the plasma size but decreases with the plasma temperature due to slower helicity dissipation. Using similar scaling, three times higher current multiplication can be expected on NSST compared to NSTX for the same injector flux and voltage. In addition, by going to higher voltage (2 – 3 kV), which is likely to be needed for higher toroidal field operations of NSST, it appears to be feasible to drive 5 MA of toroidal current with injection current of only about 30 kA. It is therefore important to understand this scaling through experiments as well as 3-D numerical simulations before finalizing the CHI design for NSST. The removable feature of the NSST centerstack should permit the incorporation of the required CHI insulator if the CHI technique can be shown to be extendable to multi-MA level plasma current. NSST with sufficiently long pulse length can also test other innovative non-inductive current drive techniques, such as bootstrap over-drive using rf based heating as invoked in the ARIES-ST /AT study. The recent JT-60U experiment on non-ohmic plasma start-up using small ohmic induction, rf and NBI current ramp-up + NBI heating induced bootstrap over-drive + vertical field ramp-up to obtain Ip = 700 kA is very encouraging experimental data.[14] The NSST could further develop this technique toward multi-MA regimes needed for the design and construction of the CTF facility. On NSST, we can utilize the ECH or EBW (electron Bernstein waves) [15] to initiate the plasma discharges and the 10 MW ICRF system in a HHFW (High harmonic fast wave) heating and current drive mode [16] to ramped up the plasma current. After the plasma current reaches an adequate level for the NBI confinement (> 1MA), the 30 MW NBI heating can be turned on to heat and densify the plasma to high-poloidal-beta / high bootstrap current fraction to reach multi-MA current. The optimization of the poloidal/vertical fields as done for the JT-60U is also a very important part of the study. The demonstration of nonohmic start-up technique is considered to be essential for an ST-based compact CTF as well as power plants.

4.4 a -Particle Physics - In terms of the α-particle related physics, a key dimensionless parameter is V_α/V_{Alf} which is about 4 - 5 for NSST but also the similar values for the future devices such as CTF and ARIES-ST reactor. This value is also comparable to the values attained on NSTX with NBI where $V_{NBI}/V_{Alf} \approx 3$. On NSTX, NBI heated discharges indeed yielded a variety of high frequency MHD modes including TAEs (Toroidal Alfven Eignmodes) at 100 kHz range and CAEs (Compressional Alfven Eigenmodes) at few MHz range (starting from near the half deuterium cyclotron frequency). In NSTX, CAEs are not observed to cause any NBI ion particle losses but there is an interesting theoretical prediction of CAEs stochastically heating bulk ions [17]. This prediction was stimulated by the apparent observation of unusually high ion temperature discharges on NSTX during NBI [7]. If proven to be true, this direct ion heating by α-particles may profoundly influence the high Q operational regimes in ST reactors. The NSST device and its diagnostic capability should therefore yield reactor relevant α-particle related physics data in high beta toroidal plasmas. In the 10 MA NSST discharges, the alpha particle orbits are estimated to be well confined [18]. It is interesting to note that a TSC simulation of high performance 10 MA discharge yielded Q = 2 with HH of only 1.2 which is considerably less than the HH = 1.4 needed in the systems code. This improved performance prediction is due to the profile effects. The alpha heating investigation in NSST will also yield valuable data for the core alpha-heating as well as the isotope confinement scaling in high beta plasmas.

5. Summary and Future Plans

The spherical torus concept can contribute effectively to fusion energy development path (i.e., PoP, PE, CTF, and DEMO). The ST development path is complementary to the tokamak based burning plasma experiments as it focuses toward a compact CTF facility and on exploring much higher toroidal beta regimes for DEMO and Power Plant. The CTF facility can provide a test bed for the development of blanket module and other fusion core components to high neutron wall loading and accumulated fluence. With advanced physics, the CTF facility could evolve toward tests of net electrical power out put. As an ST PE level experiment, the NSST facility can provide the necessary physics basis for CTF, while developing more advanced physics scenarios for CTF, DEMO and ST power plants. To support its mission, the NSST facility, with up to 10 MA of plasma current, is designed to include advanced physics features such as strong plasma shaping and wall mode stabilizing plates as well as physics flexibilities including the NBI system to drive sufficient toroidal rotation and rotational shear flows for stability and confinement, and with ample diagnostic access to facilitate physics research. Tritium operation will enable alpha-particle research at high beta.

ACKNOWLEDGEMENTS:

* Work supported by U.S. Department of Energy under contracts DE-AC05-00OR22725 and DE-AC02-76CH03073, and grants. Thanks are due to the NSTX Research Team for providing pertinent data for the NSST design, R. Goldston and R. Hawryluk for advice and encouragements, and S. Jardin for expert help in the analysis codes.

REFERENCES:

[1] M. Peng, J. B. Hicks, Fusion Technology 1990, **2**, 1287 (Elsevier Science Publishers B. V., 1991).
[2] M. Peng et al., Fusion Engineering, 1997, 17th IEEE/NPSS Symposium **2**, 733 (1998).
[3] I. V. Sviatoslavsky et al., Fusion Engineering and Design, **45**, 281(1999).
[4] M. A. Abdou et al., Fusion Technology, **29**, 1(1996).
[5] See for example the 2002 Snowmass Report.
[6] See for example, E. Synakowski et al., the paper OV/2-2 in this conference.
[7] B. LeBlanc et al., the paper EX/C5-2in this conference.
[8] J. Menard et al., the paper EX/S1-5 in this conference.
[9] S. Sabbagh et al., the paper EX/S2-2 in this conference.
[10] R. Maingi, et al., the paper EX/C2-5 in this conference.
[11] S. Jardin et al, PPPL-3666 (2002), to appear in Fusion Engineering Design.
[12] C. Neumeyer, et al., in proceedings of 19th SOFE Meeting, (2002).
[13] T. Jarboe, et al, the paper ICP-10 in this conference.
[14] Y. Takase, et al., the Journal of Plasma and Fusion Research,**78**, 719-721 (2002)
[15] P. Efthimion et al, the paper EXP/2-12in this conference.P
[16] P. Ryan et al, the paper EXP/2-13 in this conference.
[17] N. Gorelenkov et al, the paper EX/W-7a in this conference.
[18] D. Darraw et al, the paper EXP/2-1 in this conference.