Exploration of spherical torus physics in the NSTX device*

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Abstract. The National Spherical Torus Experiment (NSTX) is being built at Princeton Plasma Physics Laboratory to test the fusion physics principles for the spherical torus concept at the MA level. The NSTX nominal plasma parameters are $R_0 = 85$ cm, $a = 67$ cm, $R/a \geq 1.26$, $B_t = 3$ kG, $I_p = 1$ MA, $q_{95} = 14$, elongation $\kappa \leq 2.2$, triangularity $\delta \leq 0.5$ and a plasma pulse length of up to 5 s. The plasma heating/current drive tools are high harmonic fast wave (6 MW, 5 s), neutral beam injection (5 MW, 80 keV, 5 s) and coaxial helicity injection. Theoretical calculations predict that NSTX should provide exciting possibilities for exploring a number of important new physics regimes, including very high plasma β, naturally high plasma elongation, high bootstrap current fraction, absolute magnetic well and high pressure driven sheared flow. In addition, the NSTX programme plans to explore fully non-inductive plasma startup as well as a dispersive scrape-off layer for heat and particle flux handling.

1. Motivation for building NSTX

A broad range of encouraging advances have been made in the exploration of the spherical torus (ST) concept [1]. Such advances include promising experimental data from pioneering experiments, theoretical predictions, near term fusion energy development projections such as the Volume Neutron Source [2] and future applications, including power plants [3]. Recently the START device has achieved a very high toroidal β, $\beta_t \approx 40\%$, regime with $\beta_N \approx 5.0$ at low $q_{95} \approx 3$ [4, 5]. The National Spherical Torus Experiment (NSTX) is being built at Princeton Plasma Physics Laboratory (PPPL) to test the fusion physics principles for the ST concept at the MA level [6]. The NSTX device/plasma configuration allows the plasma shaping factor $I_p q_{95}/aB$ to reach as high as 80, an order of magnitude greater than that achieved in conventional high aspect ratio tokamaks. The key physics objective of NSTX is to attain an advanced ST regime, i.e. simultaneously attaining ultra-high β, high confinement and high bootstrap current fraction (f_{bs}) [7]. This regime is considered essential for the development of an economical ST power plant because it minimizes the recirculating

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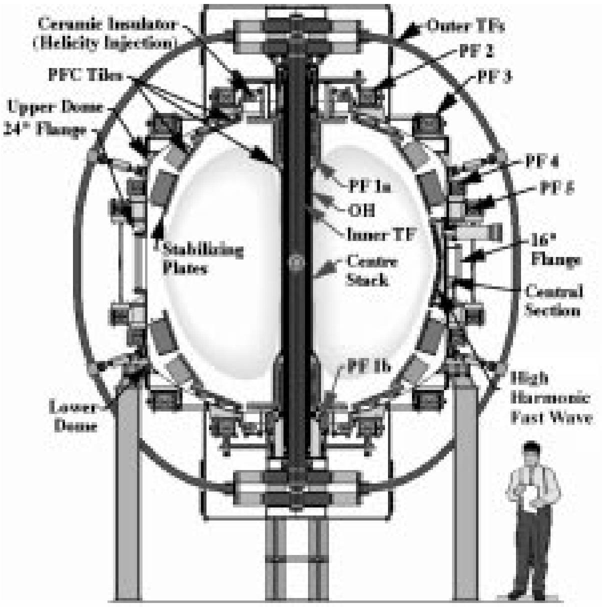

Figure 1. NSTX device cross-section. 1 inch (") = 2.54 cm.

power and power plant core size. Other NSTX mission elements crucial for ST power plant development are the demonstration at the MA level of fully non-inductive operation and the development of acceptable power and particle handling concepts.

2. NSTX facility design capability and technology challenges

The NSTX facility is designed to achieve the NSTX mission with the following capabilities:

- $I_p = 1$ MA for low collisionality at relevant densities;
- $R/a \geq 1.26$, including OH solenoid and coaxial helicity injection (CHI) [8] for startup;
- High harmonic fast wave [9] (HHFW — 6 MW, 5 s), CHI, neutral beam injection (NBI — 5 MW, 5 s) for heating and $j(r)$ control;
- Close fitting conducting shell for maximum β and β_N;
- Pulse length 5 s $\geq$ skin time $\approx L/R$ time.

The NSTX device design is shown in Fig. 1. The device centre stack is designed and fabricated to allow for very low aspect ratio ($R/a \geq 1.26$) operation. It has a sufficient ohmic drive to create 1 MA ohmically heated discharges. The centre stack is connected to the outer vessel via ceramic insulators and bellows which provide an electrical isolation for CHI. This also provides mechanical isolation to allow for the relative growth of the centre stack with respect to the outer vacuum vessel during bakeout and operation. The device is designed with close fitting 1.2 cm thick copper passive stabilizing plates for MHD mode stabilization. CHI and HHFW will be used for the initial plasma startup studies while ECH + HHFW is being considered for RF only startup as an upgrade. The NBI heating and current drive system is also expected to provide plasma rotation for mode stabilization as well as central plasma fuelling. The NBI system will also be used for the NBI based plasma profile diagnostics such as charge exchange recombination spectroscopy (CHERS) (both toroidal and poloidal) and motional Stark effect (MSE) diagnostics. After two and a half years of design and construction activities, the NSTX first plasma was achieved on 15 Feb. 1999, ten weeks ahead of the original schedule, and with the Total Project Cost tasks completed within the budget. The NSTX plasma performance thus far has been excellent, with a number of important research objectives having already been achieved, including ohmic plasma discharges with current level approaching the full design value of 1 MA. The real time plasma control system is now producing plasmas with various configurations (inner wall limited, double null diverted and single null diverted) and elongations ($\kappa = 1.5$–2.3). The 12-element HHFW antennas were installed on NSTX and the initial experimental operation has started with an injected power of up to 400 kW. The CHI experiment has also started creating plasmas with up to 25 kA of discharge current. The NBI heating system and associated NBI based diagnostics such as the CHERS will be commissioned in the fall of 2000.

3. Access to advanced ST regimes

The ultimate goal of the NSTX research programme is to access the advanced ST regime simultaneously with a high β, high bootstrap fraction and high confinement regime in a non-transient fashion. The ST configuration, owing to its short outboard connection length combined with a strong global magnetic shear and the naturally high κ and δ, has the potential for achieving a high performance regime with high plasma β and f_{bs} approaching unity. The plasma equilibrium was generated using the JSOLVER code, and the stability of equilibria was determined utilizing the PEST-II code for low n kink stability and the BALLOON code for high n stability. The predicted ideal MHD stability limit against low n kinks and high n ballooning modes is very high: $\beta_t \to 60\%$, $\beta_N \to 8$ with $f_{bs} \approx 100\%$ for $\kappa \approx 3.4$. In this regime, a close fitting conducting shell with $r_{wall}/a \leq 1.2$ is needed for suppressing the low n kink modes. For $\kappa \approx 2$, as planned for NSTX in the initial configuration, an ideal MHD stable regime with $\beta_t \approx 40\%$, $\beta_N \approx 8$ with $f_{bs} \approx 75\%$ is predicted. In Fig. 2 the corresponding plasma current profiles are shown. The j_{bs} is indeed relatively well aligned with the j_{total}, and an outer region current drive contribution of ~20% is required. NSTX has sufficient heating power (~11 MW) to reach the desired β value (~40%) with a relatively modest confinement assumption using an H factor of ~2 over ITER97L scaling. A plasma pulse length of 5 s is sufficient to allow the current profile $j(r)$ to fully relax. The low n kinks are predicted to be stabilized by a close fitting conducting wall together with plasma toroidal rotation induced by NBI. It should be noted that the plasma toroidal rotation requirement for NSTX is an order of magnitude lower than for conventional tokamaks owing to a low Alfvén velocity, which is typically on the order of the ion thermal velocity for NSTX. It is expected that the rotation velocity requirement for effective wall stabilization is a fraction of the Alfvén velocity. For $j(r)$ control, the combination of NBI, HHFW and CHI systems will be used to augment the bootstrap current. The TRANSP calculations show that the NBI is capable of driving 100–200 kA of current in the central region, which should be sufficient to provide the small central seed current (a few kiloamperes) required. For off-axis current drive, a 12 element real time phased HHFW antenna array will be used for driving up to 300 kA of off-axis current to supplement the bootstrap current. Theoretical analyses and modelling calculations show that the HHFW power absorption is one to two orders of magnitude larger in the NSTX parameters than is found in conventional aspect ratio tokamaks [9]. The strong single pass absorption together with the real time antenna phasing capability allows efficient off-axis current drive by HHFW. As for the edge current drive, CHI is the most promising tool. The expected edge current for CHI in the well formed ST may be estimated as $I_{inj} \times q_{95}$, where I_{inj} is the current injected into the plasma by CHI and q_{95} is the expected toroidal current amplification by the geometric factor. For NSTX, up to 350 kA of edge current may be driven by CHI with an injection of ~25 kA for the expected $q_{95} \approx 14$. It should be noted that the ideal ballooning mode prediction may turn out to be conservative for NSTX, because of the strong finite Larmor radius and trapped particle effects in the ST configuration. In addition, the present ideal MHD stability calculations assume the plasma edge pressure to be zero (no pressure edge pedestal). This may be a conservative assumption in that significant edge pedestal formations during H mode, for example, have been observed in many tokamak experiments. The START NBI heated high β discharges also show a sharp edge pressure pedestal [4, 5]. If proven, the higher edge pressure gradient in NSTX will further reduce the edge current drive requirement, though it is not favourable for kink mode stability. If the edge current drive turns out not to be practical, another obvious path is to move towards a higher elongation regime ($\kappa \approx 3$) where the bootstrap current fraction can be near 100% with very high β. In this regime there is no current drive requirement for plasma sustainment except for a very small central seed current. It should be mentioned that NSTX ideal MHD calculations indicate that the details of the plasma boundary shape in the outer boundary (large major radius) region are quite important for the ideal ballooning stability. The NSTX device poloidal field coils are therefore configured to give sufficient outer boundary shape control capability.

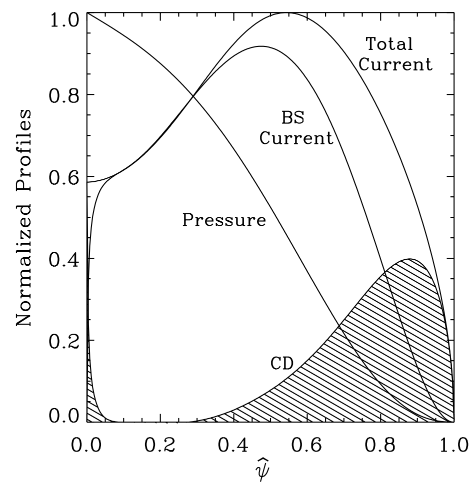

Figure 2. NSTX advanced ST regime: $\beta_t = 40\%$, $\beta_N = 8.5$, $q_{95} = 14$, $f_b = 0.77$, $\kappa = 2$, $\delta = 0.45$.

4. Prospect of high confinement

The ST configuration can potentially achieve very high confinement in the advanced ST regime. The predicted linear growth rates of the kinetic electrostatic mode such as dissipated trapped electron–η_i

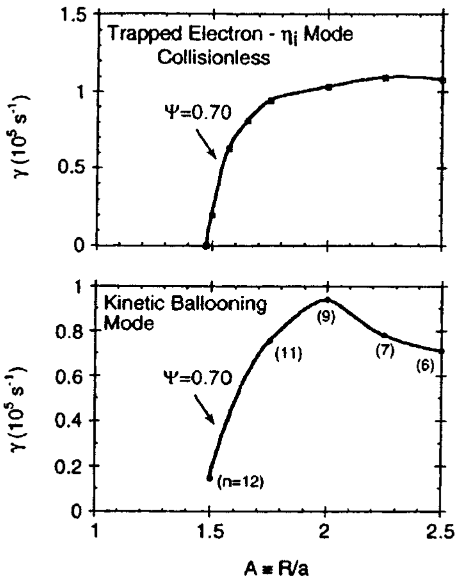

Figure 3. Electrostatic and electromagnetic instability suppression for low R/a.

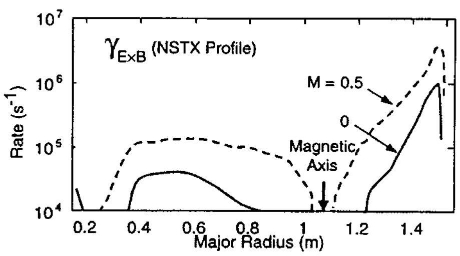

Figure 4. Expected sheared flow rate profiles in NSTX.

modes and electromagnetic modes such as kinetic ballooning microturbulence are found to be greatly reduced owing to favourable drift orbits as R/a falls below 1.5, as shown in Fig. 3 [10]. The growth rates are plotted as functions of aspect ratio for both modes at the $\psi = 0.7$ flux surface. As can be seen, the growth rates start to decrease sharply at $R/a \approx 1.5$, with complete stabilization of these modes on this flux surface at $R/a \approx 1.3$–1.5. The reason for stabilization of these modes with decreasing aspect ratio can be linked to the strong toroidicity of these plasmas, which leads to a reduction in the orbit averaged bad curvature (the instability drive). These properties are expected to improve plasma confinement in the outer region of NSTX. In recent work on transport barrier formation, it was shown that if the plasma flow shearing rate (and the associated radial electric field shear rate) is sufficiently strong compared with the plasma turbulence growth rate, then turbulence suppression leading to reduced transport and improved plasma confinement will occur. This mechanism is believed to be responsible for the formation of internal transport barriers. In NSTX a very strong plasma sheared flow rate, $\gamma_{\boldsymbol{E}\times\boldsymbol{B}}$ (s^{-1}), over a significant portion of the minor radius is predicted — particularly with NBI toroidal drive (toroidal Mach number $M = 0.5$), as shown in Fig. 4. This very high level of sheared flow, which is nearly an order of magnitude greater than that obtained in conventional tokamaks, is due to the combination of high β (high diamagnetic drive), relatively low field and small major radius of NSTX. As can be seen from Figs 3 and 4, the expected shearing rate could readily exceed the linear growth rates of expected turbulence in the ST regime [11]. In addition, owing to the high β as well as the high poloidal field nature of the ST configuration, a significant absolute magnetic well ($\leq$30%) is expected to form in the core region, which tends to stabilize a class of plasma instabilities. This absolute minimum B formation and its effect on plasma confinement make an interesting area of research for NSTX. These physics mechanisms could so greatly reduce the level of turbulence driven energy and particle transport that the NSTX confinement could approach the ion neoclassical confinement values. If proven, this favourable confinement in high β and bootstrap current fraction could lead to an economical compact fusion power reactor.

5. Non-inductive startup and power and particle handling

In order to eliminate the OH solenoid, it is important to develop an efficient non-inductive startup tool for the ST. The relatively modest magnetic flux and helicity per unit of plasma current for the ST tend to ease non-inductive startup requirements. The main tool for NSTX is CHI. CHI delivers poloidal flux to the plasma edge through the use of biased electrodes, and this flux (toroidal current) is transported throughout the plasma via global MHD fluctuations. Approximately 0.2 MA of plasma current has thus far been driven in the Helicity Injected Tokamak (HIT) experiment with CHI [8]. The NSTX device is designed with the ceramic electrical insulators located at both ends of the centre stack, which permits DC biasing (up to 2 kV) between the centre stack and the outer vacuum chamber. This NSTX bias configuration is similar to the one used in the HIT experiment. The NSTX CHI power supply is a 2 kV, 50 kA rectifier, which provides greatly improved bias control compared with the previous capacitor bank based CHI experiments. A CHI modelling code prediction shows that it should be possible to produce 500 kA of non-inductive CHI discharges from zero current in NSTX. From a CHI physics perspective, NSTX therefore provides an excellent 1 MA test bed for CHI at low aspect ratio. The HHFW absorption is also expected to be sufficiently strong, even for low electron temperature plasmas with $T_e \approx 300$ eV, that it could heat and drive current in the startup phase of the plasma. A Tokamak Simulation Code (TSC) simulation shows that the HHFW heating and current drive along with bootstrap current (and a small amount of OH assist) can ramp up the current to the full value of 1 MA. In addition, a startup experimental plan based only on RF using the ECH startup [12] together with HHFW current ramp-up is also under study.

Owing to the inherently high power density of compact ST reactors, it is essential to develop an acceptable power and particle handling concept. The NSTX plasma facing components are designed to handle the anticipated power heat load in different divertor and limiter configurations, including single and double nulls. One of the promising configurations from a power and particle handling viewpoint is a 'natural' inner wall limited configuration. ST plasmas permit inboard limited plasmas with mostly diverted SOL, large magnetic variations along the SOL field line (e.g. strong magnetic mirror ratios of up to 4 to 1 and field line curvatures), potentially large flux tube expansion (order of 10) and strong interchange instabilities due to high edge pressure gradients. These edge properties may lead to an effective power and particle handling concept for STs.

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