National Spherical Torus Experiment (NSTX)*

Masayuki Ono and the NSTX Team**
Princeton Plasma Physics Laboratory
Princeton, NJ 08550, USA

Abstract - The main aim of National Spherical Torus Experiment (NSTX) is to establish the fusion physics principles of the innovative spherical torus (ST) concept. Physics outcome of the NSTX research program is relevant to near-term applications such as the Volume Neutron Source (VNS) and burning plasmas, and future applications such as the pilot and power plants. The NSTX device began the plasma operations in February 1999 and the plasma current was successfully ramped up to the design value of 1 million amperes (MA) on Dec. 14, 1999. The CHI (Coaxial Helicity Injection) and HHFW (High Harmonic Fast Wave) experiments have also started. Stable CHI discharges of up to 133 kA and 130 msec duration have been produced using 20 kA of injected current. Using eight antennas connected to two transmitters, up to 2 MW of HHFW power was successfully coupled to the plasma. The NBI heating system and associated NBI based diagnostics such as the CHERS will be operational in Oct. 2000.

NSTX Mission - The National Spherical Torus Experiment (NSTX) is a new US Department of Energy National Fusion Energy Science facility [1,2] located at Princeton Plasma Physics Laboratory whose main purpose is to establish the fusion physics principles of the innovative spherical torus (ST) concept [3]. The mission of the National Spherical Torus Experiment (NSTX) is to investigate the physics principles of:

- Non-inductive start-up, current sustainment and profile control,
- Confinement and transport,
- Pressure limits and self-driven currents,
- Stability and disruption resilience, and
- Scrape-off layers and divertors;

in a low-aspect-ratio (spherical) torus as a plasma confinement innovation. These principles are to be investigated in scientifically interesting regimes characterized by:

- High average toroidal beta β_T (up to 40 %),
- High pressure gradient driven current fraction f_{bs} (up to 70 %),
- Fully relaxed, non-inductively sustained current profile,
- Collisionless plasmas with high temperature and densities, and
- Low aspect ratio as low as 1.26 and plasma elongation as high as 2.0.

The physics outcome of the NSTX research program is relevant to near-term applications such as the Volume Neutron Source (VNS) and burning plasmas, and future applications such as the pilot and power plants.

NSTX Facility Overview - The NSTX facility came on line in Feb. 1999 utilizing much of the existing TFTR (Tokamak Fusion Test Reactor) site infrastructure. The NSTX facility is managed by PPPL and experiments are carried out by the NSTX Research Team composed of researchers from over 15 institutions. The NSTX nominal device and plasma parameters are $R_0 = 0.85$ m, $a = 0.67$ m, $R/a \geq 1.26$, $B_T = 0.3$ T, $I_p = 1$ MA, $q_{95} = 14$, elongation $\kappa \leq 2.2$, triangularity $\delta \leq 0.5$, and plasma pulse length of up to 5 sec. The plasma heating / current drive (CD) tools are High Harmonic Fast Wave (HHFW) (6

MW, 5 sec)[4], Neutral Beam Injection (NBI) (5 MW, 80 keV, 5 sec), and Coaxial Helicity Injection (CHI) (I-injection = 50 kA, V-bias = 1 kV) [5].

The cross section of the NSTX device is shown in Fig. 1. The device mid-plane is about 3.5 m from the floor. The device center-stack is designed and fabricated to allow for the very low aspect ratio $R/a \geq 1.26$ operation [6]. It has a sufficient ohmic drive to create 1 MA ohmically heated discharges. The center stack is connected to the outer vessel via ceramic insulators and bellows to provide an electrical isolation for CHI and a mechanical isolation to allow for the relative growth of center-stack with respect to the outer vacuum vessel during bakeout and operation. The center stack can be removed or replaced relatively quickly. The device is designed with close-fitted 12 mm thick copper passive stabilizing plates for MHD mode stabilization. The CHI will be used for the initial plasma start-up studies while ECH (Electron Cyclotron Heating) / EBW (Electron Bernstein Wave) + HHFW is considered for RF only start-up as an upgrade. The NBI heating and current drive system is also expected to provide plasma rotation for mode stabilization and central plasma fueling. The NBI system will also be used for NBI based plasma profile diagnostics such as CHERS (Charge Exchange Recombination Spectroscopy) for the ion temperature and plasma rotation velocity profiles and MSE (Motional Stark Effect) for the plasma current profile measurements.

Advanced ST Regimes - The ultimate goal of the NSTX research program is to access the power-plant-relevant advanced ST regime with simultaneous high beta, well aligned high bootstrap current fraction [7], and high confinement in non-transient fashion. The ST configuration, due to the short outboard connection length combined with strong global magnetic shear, and the naturally high κ and δ, has the potential of achieving a high-performance regime with high plasma β and f_{bs} approaching unity. The predicted ideal MHD stability limit against low n-kinks and high-n ballooning modes is very high: $\beta_T \rightarrow$ 60%, β_N(normalized beta) $\rightarrow 8$ 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 the initial NSTX configuration, an ideal MHD stable regime with $\beta_T \approx 40\%$, $\beta_N \equiv \beta_T / (I_p/aB_T) \approx 8$ with $f_{bs} \approx 75$ % is predicted. The passive stabilizing plates (a close fitting conducting shell) as installed in NSTX are shown in Fig. 1. NSTX has a sufficient heating power ($\approx$ 11 MW) to reach the desired β value ($\approx$ 40 %) with a relatively modest confinement assumption of H-factor of $\approx$ 2 over the tokamak L-mode (ITER96P) scaling [2]. A plasma pulse length of 5 sec 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 in the presence of plasma toroidal rotation induced by NBI. For the j(r) control, the combination of NBI, HHFW, and CHI systems will be used to augment the bootstrap current. The calculations show that NBI is capable of driving 100-200 kA of current in the central region which should be sufficient to provide the central seed current ($\approx$ a few kA) required for bootstrap current generation. For off-axis current drive, a twelve-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 modeling calculations show that, due to high plasma dielectric of 30 – 100 and high plasma beta, the HHFW power absorption is one to two orders of magnitude larger in the NSTX parameters than in conventional aspect ratio tokamaks where the plasma dielectric is order of 1 [4]. The strong single-pass absorption with the real-time antenna phasing capability allows efficient off-axis current drive by HHFW. As for the edge current drive, the CHI is a possible 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 injection of $\approx$ 25 kA for $q_{95} \approx 14$.

NSTX First Plasma - After two and half years of design and construction activities, the NSTX first plasma discharge of 300 kA (using about 1/3 of designed OH flux) was achieved on Feb. 16, 1999, 10 weeks ahead of schedule. The first plasma operation provided important confirmation of the NSTX device operational readiness as well as valuable experimental data of initial ohmically heated plasmas. It should be noted that the newly formed NSTX National Research Team played a crucial role from the start. The Los Alamos (LANL) Team brought the fast visible camera to capture the plasma evolution, which was an essential tool in bringing the plasma current to 300 kA in just two days of plasma operations [8]. The EFIT reconstruction [9] of the first plasma was also successfully carried out by the Columbia University team using the magnetic data. The EPICS from Argonne (for engineering systems control) and MDS-PLUS from MIT (for data acquisition) software performed extremely well.

Attainment of 1 MA Plasma Current Discharges - After the First Plasma Operations, the NSTX construction team went on to install remaining in-vessel hardware including the passive/outer divertor plates, HHFW (High Harmonic Fast Wave) antennas, and CHI ceramic insulators, and about 2500 graphite tiles. In order to facilitate the in-vessel hardware installation, the center stack was pulled out of the device. With the center stack out of way, it was much easier to perform the in-vessel installation tasks. The NSTX plasma operations restarted on Sept. 1, 1999. With the double swing 6 kV OH power supply, the plasma current was successfully ramped up to the design value of 1 MA on Dec. 14, 1999 about 9 months ahead of schedule as shown in Fig. 2. The observed ohmic current drive efficiency is quite good, yielding Ejima coefficients in the 0.4 – 0.5 range [10]. The TSC (Tokamak Simulation Code) [11] was able to reproduce many of the global features of the NSTX OH discharges with some deviations possibly resulting from the MHD activity. IREs (Internal Reconnection Events) occur particularly in the shutdown phase of the discharges. An ultra-soft x-ray diagnostic array by Johns Hopkins University [12], X-Ray Pulse Height Analyzer (PHA), and X-Ray Crystal Spectrometer became operational. The x-ray diagnostics measureed central plasma temperature of typically 0.5 – 1.0 keV range, consistent with the stored energy obtained from equilibrium reconstructions (EFIT) and TSC. The ohmic plasma density limit observed thus far is consistent with the Murakami-Huggil Limit. All of the planned plasma shaping parameters were achieved. The plasmas with the elongation of κ = 1.6 - 2.2 (up to 2.6 transiently) and the triangularity δ = 0.2 – 0.4 (up to 0.6 transiently) have been obtained. The plasma shaping factor (defined as $I_p\, q_{95} / a\, B$) of 30 has been thus far achieved at 1 MA compared to ≤ 6 to conventional aspect ratio tokamaks. Plasmas with volumes in excess of 12 m^3 has been routinely produced. The ohmic heated plasmas with the total stored energy of up to 30 kJ, average plasma toroidal beta of up to 6.5%, and the global confinement time of up to 20 msec have been achieved during the 1 MA current discharge.

A key to the achievement of high current plasma discharges was the implementation of the real time plasma control system in collaboration with General Atomics. The Skybolt I computer system was able to feedback control on the plasma radial and vertical positions as well as the plasma current [13]. The high plasma current (i.e. Ip > 0.6 MA) was, in fact, possible only with current ramp control using the real time plasma control system. The control system was also able to create single null and double null diverted. Another important factor in producing high quality ST discharges is the vacuum quality. Using the bakeout system, the device center stack tiles have been baked to 300 °C while the vacuum vessel was baked to 150 °C. The bakeout capability will be improved toward getting all the graphite tile temperatures to >300°C. To further improve the vacuum condition, boronization and other wall preparation techniques will be also implemented in the future.

Coaxial Helicity Injection (CHI) for Plasma Start-up -In order to eventually eliminate the OH solenoid for ST, it is important to develop efficient start-up tools which does not rely on the OH solenoid. The relatively modest magnetic flux and helicity per plasma current for ST tend to ease noninductive startup requirements. The main non-inductive (without OH) plasma start-up tool for NSTX is the Coaxial-Helicity-Injection (CHI). CHI delivers poloidal flux to the plasma edge through the use of biased electrodes (for the NSTX case, the center stack and outer vessel are biased with respect to each other), and this flux (toroidal current) is believed to be transported throughout the plasma via global MHD fluctuations. The CHI experiments on NSTX successfully started in Nov. 1999 [14]. Plasma currents of up to 133 kA were produced using about 20 kA of injected current. Stable CHI discharges of up to 130 msec have been produced. The LANL fast camera clearly showed a CHI plasma column extended well into the NSTX chamber. With Electron Cyclotron preionization [15], the initial fill pressure for CHI initiation was reduced to as low as 1 mTorr thus far. This is important in order to make CHI compatible with the OH operation. For the longer range, the injector current will be increased toward 50 kA level in order to produce up to 500 kA of CHI discharges.

High Harmonic Fast Wave Heating – The High Harmonic Fast Wave (HHFW) heating experiment started in Nov. 1999. The HHFW system construction is a joint project between PPPL and ORNL. Using eight antennas connected to two transmitters, up to 2 MA of rf power was successfully coupled to the plasma with dielectric of order of 30. The ORNL edge reflectometry has successfully measured the edge density profiles in front of the antenna [16]. The ultra-soft x-ray diagnostic shows some indication of core electron heating with a slow wave velocity antenna phasing. The system is designed to eventually deliver 6 MW using 12 antennas and 6 transmitters.

NSTX Research Plan - The NSTX Research Program for the next four years is shown in Fig. 3. The NSTX device at present (Jan. – June, 2000) is undergoing the installation of the NBI system and Upgrade Diagnostics. The NBI heating system and associated NBI based diagnostics such as the CHERS (Charge-Exchange Recombination Spectroscopy) will be operational in the fall of year 2000. With HHFW, NBI, and CHI tools in place, the high beta regimes consistent with the no-wall beta limit of about 25 %will be investigated. The bootstrap current fraction is relatively modest 40%. In the longer term, the passive stabilizing plate jumpers may be reconfigured electrically for plasma kink stabilization. This wall stabilization of the kinks is essential for the attainment of the high beta (40%)/high bootstrap fraction (70%) discharges. A possible upgrade item to be implemented in 2004- 2005 time frame is a new center stack to increase the device/plasma performance and to investigate ARIES ST-like higher elongation plasmas [7] which has higher beta (50%) and higher bootstrap current fraction (90%). If the NSTX ST physics is successful, the plan is to construct a 10 MA class performance extension ST device in the TFTR Test Cell to test the ST concept in the reactor grade plasmas with DT fuel capability.

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** The NSTX Team consists of researchers from Princeton Plasma Physics Laboratory, Oak Ridge National Laboratory, Columbia University, University of Washington, Los Alamos National Laboratory, University of California at Los Angeles, General Atomics, Johns Hopkins University, Sandia National Laboratory, University of California at San Diego, Massachusetts Institute of Technology, Lawrence Livermore National Laboratory, University of California at Davis, Fusion Physics &Technology, University of Tokyo, and University of Hiroshima.

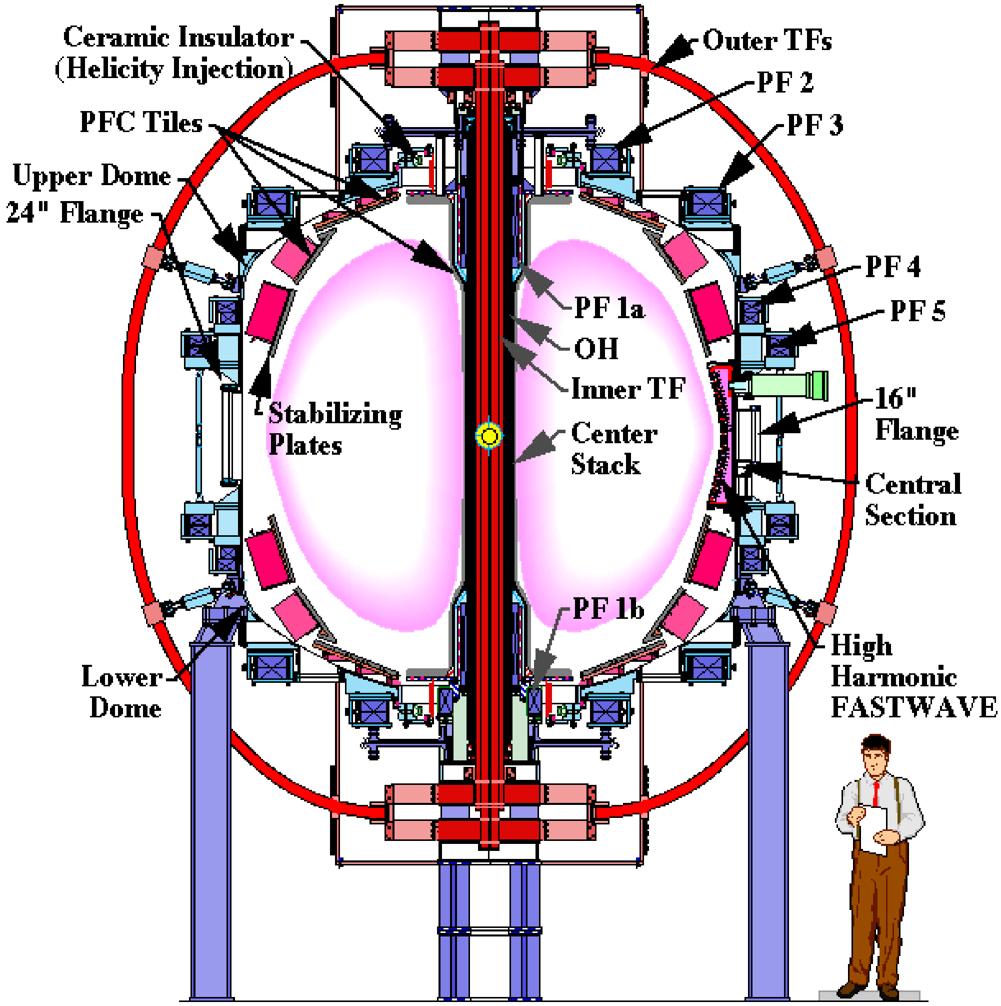

Fig. 1. A schematic of the NSTX device cross-section.

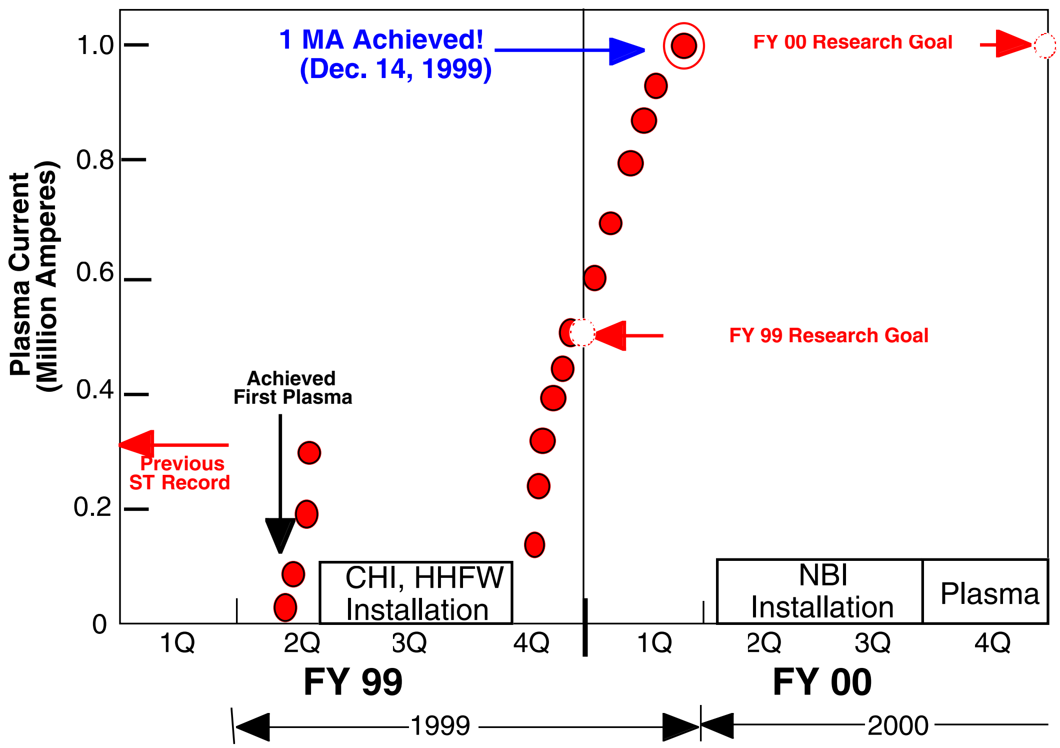

Fig. 2. Progress of the NSTX plasma operation

Baseline Scientific Milestones	(FY99)	(FY00)	(FY01)	(FY02)	(FY03)	(FY04)
		• Ohmic studies • Initial CHI • Initial HHFW	• Transport • Full HHFW	• Macroscopic stability • Full CHI	• Plasma-wall • Initial β-τ_E integration	
Run-wks:		(14)	(13)	(13)	(13)	

2/99 1st Plasma — *12/99* 1 MA, κ ~2 — ($\kappa \to 3$, $t >> \tau_s$)

Operation Capability	Inductive	Noninductive Assisted	Noninductive Sustained
• **Toroidal Beta, β_T**		• $\to$ 25%	• $\to$ 40%
• **Bootstrap Current**		• $\to$ 40%	• $\to$ 70%
• **Current**	• $\to$ 0.5 MA	• $\to$ 1 MA	• ~ 1 MA
• **Pulse**	• $\to$ 0.5 s	• $\to$ 1 s	• $\to$ 5 s
• **HHFW Power**	• $\to$ 4 MW	• ~ 6 MW	• ~ 6 MW
• **NBI Power**		• $\to$ 5 MW	• ~ 5 MW
• **EBW Power**	• ~ 30 kW	• $\to$ 0.4 MW (incremental)	• ~ 0.4 MW
• **CHI Startup**	• $\to$ 0.2 MA	• $\to$ 0.5 MA	• ~ 0.5 MA
• **Control**	• current, R, shape	• heating, density	• profiles, modes
• **Measure**	• $T_e(r)$, $n_e(r)$	• $j(r)$, $T_i(r)$, flow, edge	• turbulence

Fig. 3 NSTX National Research Program Plan

KEYWORDS:
Spherical Torus
Magnetic Fusion Research
Toroidal Plasma
High Beta Plasmas
Advanced ST Regimes
NSTX Spherical Torus