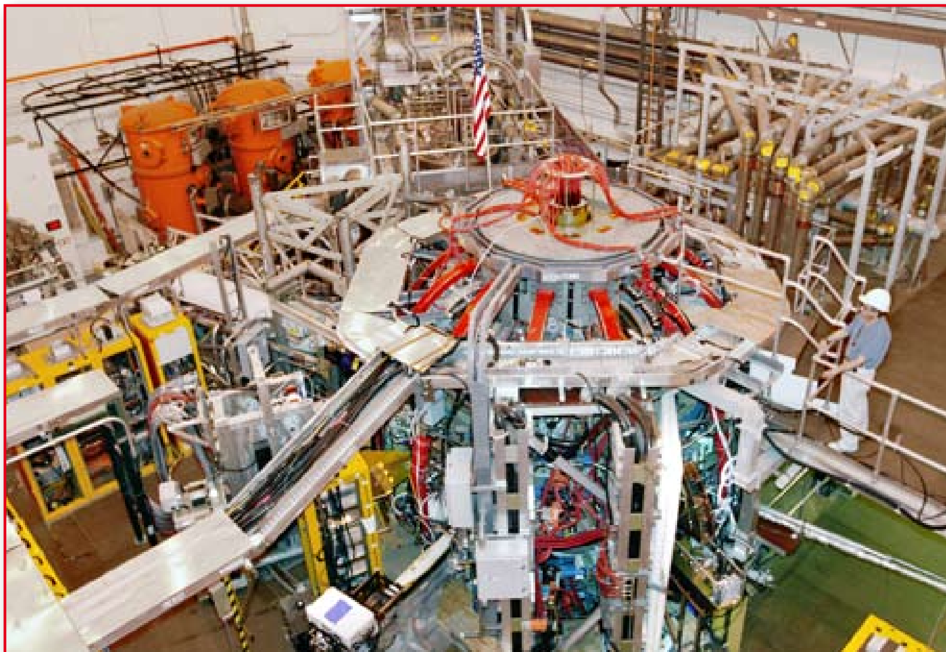
INFORMATION BULLETIN

PPPL
PRINCETON PLASMA PHYSICS LABORATORY

U.S. DEPARTMENT OF ENERGY'S
PRINCETON PLASMA PHYSICS LABORATORY
JAMES FORRESTAL CAMPUS, P.O. BOX 451
PRINCETON, NEW JERSEY 08543

NSTX

The National Spherical Torus Experiment



The National Spherical Torus Experiment

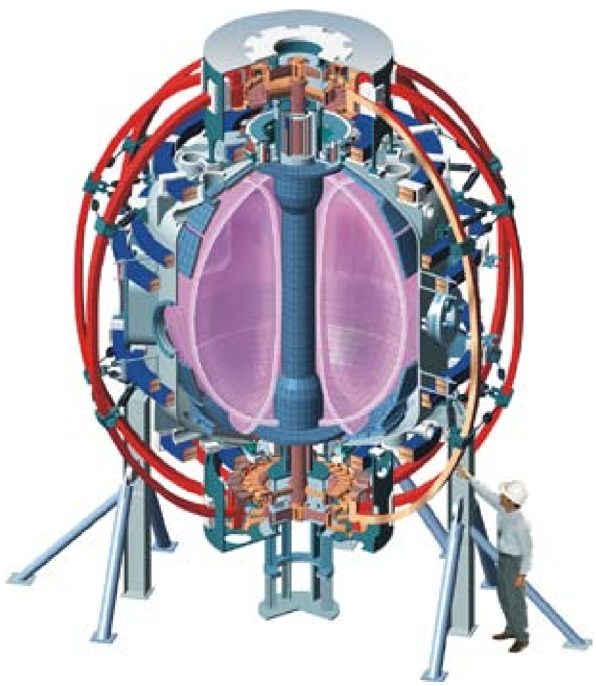
The National Spherical Torus Experiment (NSTX) at the U. S. Department of Energy's Princeton Plasma Physics Laboratory (PPPL) is yielding data that may provide an attractive path for the development of fusion energy as an abundant, safe, and environmentally sound means of generating electricity for the long term. The NSTX device is being used to explore a novel magnetic fusion concept that may lead to practical fusion energy at reduced cost. A national team comprised of 30 U.S. fusion research institutions is performing the experiments. Scientists from the U.K., Japan, Russia, Korea, France, Germany, Israel, Italy, and Czech Republic also participate.

In NSTX a magnetic field is used to confine an ionized gas (plasma) shaped like a cored apple (See page 2). Theory indicates that plasma pressure in a spherical torus (ST) can be maintained with a lower magnetic field strength. Fusion power production is roughly proportional to the square of the plasma pressure, and the cost of a fusion power plant will rise with the size and strength of the magnetic field coils. Consequently, successful experiments on NSTX may lead to a smaller, more economical fusion component test facility (CTF) and fusion power plants.

Fusion power plants will employ deuterium-tritium (D-T) plasmas in which the fuel and energy containment must be sufficient to sustain the D-T fusion reactions at high temperatures. The spherical shape may overcome plasma turbulence and other instabilities that cause energy to leak from the plasma, stopping the fusion process prematurely.

NSTX Mission

The mission of the NSTX program is to establish the scientific potential of the ST configuration as a means of achieving practical fusion energy. Recent assessments indicate that the advantages of the ST may lead to cost-effective steps to develop practical fusion energy. The Next Step Spherical Torus (NSST) may take advantage of the former Tokamak Fusion Test Reactor (TFTR) facility at PPPL to test the ability of the ST to confine D-T fusion-producing plasmas. Further down the line, there is the possibility of an ST-based compact Component Test Facility (CTF) to develop and test fusion power plant components. The research results from NSTX in the coming years will be instrumental in establishing the attractiveness of these future steps for fusion development.



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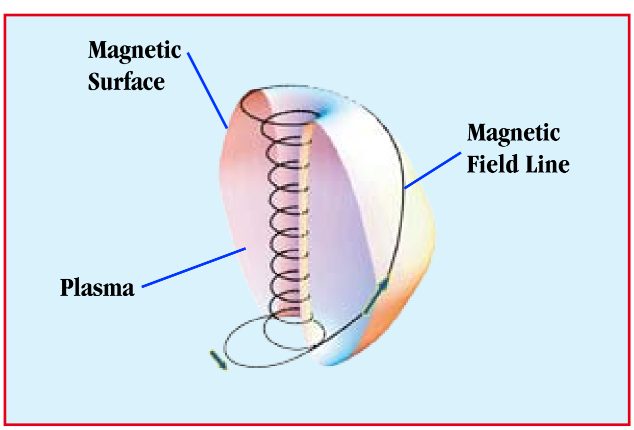
Recent NSTX Accomplishments

NSTX began operation in September 1999. Through 2006, researchers have made rapid progress in exploring scientific issues relating to the effectiveness of the ST configuration for practical fusion power production and also those scientific issues relevant for ITER and future experiments. Key examples are highlighted below. In the process, the NSTX team has implemented numerous improvements in measurement and operational capabilities.

High Plasma Pressure in Moderate Magnetic Field

Toroidal beta is the ratio of the average plasma pressure to the pressure of the main applied magnetic field. NSTX has attained a toroidal beta approaching 40%, which is its target value. This is a result of improvements in plasma shaping control and operating conditions during intense plasma heating by neutral-beam injection (NBI) with power levels up to 7 MW.

Two useful regimes of plasma stability have been identified in NSTX. One with high plasma current and a toroidal beta in the range of 30 to 40% is appropriate for NSST. Another, with nearly sustained plasma current with ~ 50% bootstrap current fraction and toroidal beta of 20-25%, is suitable for CTF.



Typical magnetic surface and field line in a spherical torus.

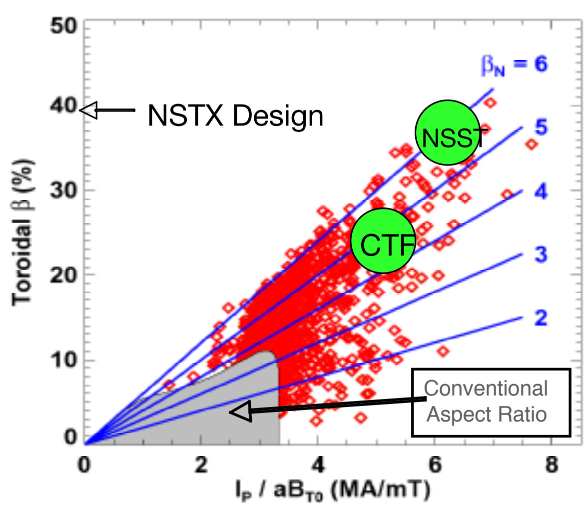
Efficient Confinement of Plasma Energy

Neutral-beam injection involves the introduction of high-energy neutral atoms into the plasma. The atoms are immediately ionized and trapped by the magnetic field. The high-energy ions then transfer part of their energy to the plasma particles in repeated collisions, thus increasing the plasma temperature. NBI may also be used to drive the plasma current.

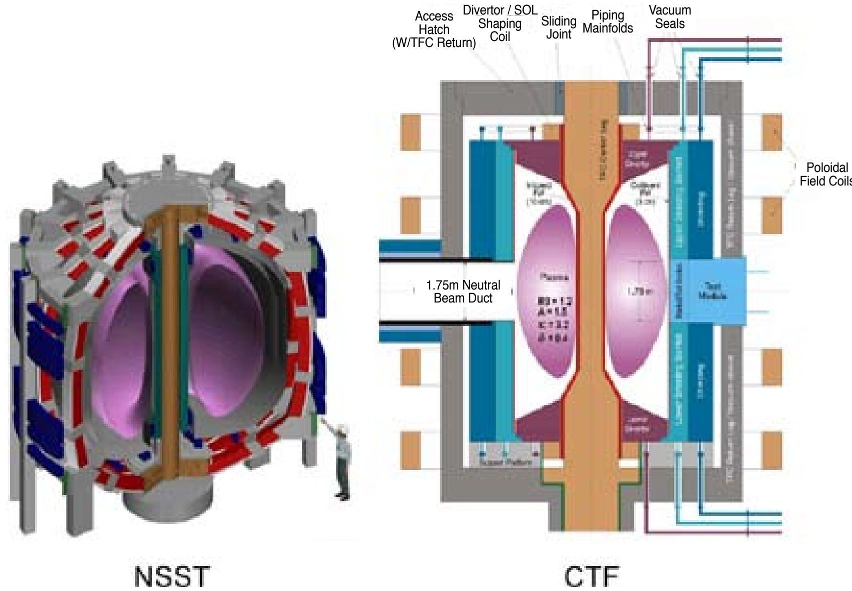
With NBI heating, NSTX's containment of plasma energy has been consistently 1.5 to 2.5 times that obtained from the standard operating mode of tokamaks. Diagnostics that enable the details of the plasma's confinement properties to be measured are in place or are being built and developed. These include systems for measuring the turbulence in the plasma that may be responsible for most of the transport of energy and particles across the confining magnetic field. This will permit establishing a physics foundation for predicting future ST performance.

Approaching Sustained Operation

A distinguishing feature of the tokamak and spherical torus magnetic confinement configurations is the large plasma current required to confine the hot plasma particles. This current, which runs in the long direction around the torus, is commonly induced and sustained in today's experiments for a limited duration by reversing the current in a solenoid placed at the center of the device. This technique is intrinsically limited by the achievable



The efficiency of NSTX field and size utilization has approached design target and the physics basis needed by future ST steps: NSST and CTF.



Design concepts of a Next Step Spherical Torus (NSST) with R=1.5 m and a Component Test Facility (CTF) with R=1.2 m.

solenoid current, so to sustain the plasma for long durations, alternative current sources are required. NSTX is investigating and developing the physics of several alternatives including NBI (explained above), Coaxial Helicity Injection (CHI), High Harmonic Fast Wave (HHFW), Electron Bernstein Wave (EBW), and the plasma pressure driven "bootstrap" current.

High Harmonic Fast Wave and Electron Bernstein Wave

Both HHFW and EBW involve the introduction of radiofrequency (RF) waves into the plasma. HHFW — an RF wave at many times the frequency with which ions gyrate around the magnetic field — can heat electrons and, in theory, drive plasma current. Strong electron heating using HHFW was observed for the first time in NSTX experiments. In NSTX, electron temperature was increased from about 5 million degrees to above 40 million degrees Kelvin using 3 MW of HHFW power. The ability of HHFW to drive current is under investigation.

In theory, EBW — a RF wave at one to three times the frequency with which electrons gyrate around the magnetic field — can heat electrons and drive plasma current in precise locations. Research is underway on how best to implement megawatt-level EBW systems to enable this research.

Bootstrap Current

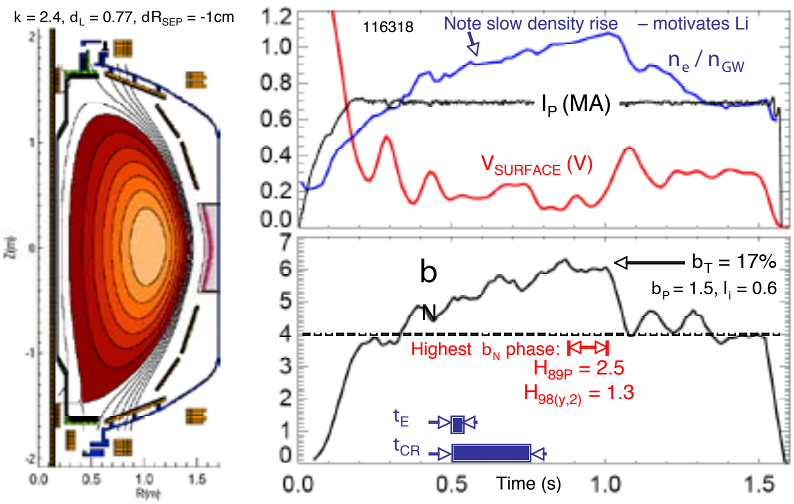
In the mid-1980s, experiments on the Tokamak Fusion Test Reactor at PPPL verified the existence of the theoretically predicted bootstrap current, which can sustain itself when the plasma pressure is high. The bootstrap plasma current in conjunction with RF waves or NBI may be used to allow steady-state operation of a fusion power plant. Bootstrap current may in time account for as much as 70% of the total plasma current flowing in NSTX.

Present Status

Nearly sustained plasma currents are routinely produced in NSTX using the NBI heating. For a significant duration of ~ 1.5 sec, the induction loop-voltage was reduced to about one-tenth of the normal value. Analysis of the data indicates that a large fraction (~60%) of the plasma current in NSTX can be driven by a combination of NBI and bootstrap current. To remove the induction requirement in NSTX completely, analysis indicates that radiofrequency power is needed to heat and drive current in a region toward the periphery of the plasma. The efficacy of adding HHFW power at 30 MHz in frequency is being vigorously investigated.

Initiating Plasma Current without Inductive Solenoid

A compact ST-based fusion Component Test Facility and power plant will have no room for a central solenoid. It is therefore critically important to develop and test solenoid-free techniques that initiate and ramp-up the plasma current to substantial levels (~10% of the full current). In NSTX a successful transient demonstration of such a current was obtained using CHI, which was developed at the University of Washington. Tests to capture this initial current for subsequent ramp up by other current-drive methods are underway. A record well-defined current of up to 160 kA was recently obtained by this method without use of the central solenoid. Investigations of additional innovative methods that combine radiofrequency power (e.g., HHFW, EBW) with outer magnetic field coil currents are in preparation and early testing.



NSTX made progress in sustained higher plasma current, neutral beam injection heating power, shaping (elongation of plasma cross section), and field utilization (toroidal beta). Record discharge pulse lengths have been achieved by operating with high bootstrap current fraction.

Dispersing Plasma Heat Flux

The high power densities anticipated in compact ST fusion energy devices will lead to large heat loads on the plasma facing components. It is therefore important to develop and test ST edge plasma properties aiming to enhance dispersal of the heat over wide areas on these components. Measurements of plasma fluxes, light radiation, and fluctuations are underway or being planned to investigate the details of this complex scientific topic. Innovative techniques of particle control, such as lithium coatings using lithium pellets and lithium evaporator, are being tested to establish a basis for deploying a liquid-lithium surface module on NSTX in the future. (See LTX Information Bulletin, June 2006) In 2006, a choice between the lithium approach and the more conventional use of cryogenic pumps will be made based on these results.

Near-Term Research Plan

In June 2003, DOE convened an international panel to review the NSTX Five-Year Research Plan, which subsequently was strongly endorsed by the group. This plan aims to understand, control, and optimize ST plasmas, and extend the knowledge base of plasma science, by establishing the tools for and carrying out sustained high-pressure plasma operations. The present phase of research in 2006 is testing advanced control of sustained high-pressure plasmas and would develop scenarios for initiating them without the benefit of a central solenoid. The succeeding phase of research in 2007-2008 would integrate the new physics results to optimize the sustained high-pressure plasmas. The rapid progress already made by the NSTX Research and Operations Teams has positioned NSTX well in embarking on this exciting research plan.

Research Goals	Determine passive limits and control tool needs	Develop advanced control • Sustained high pressure plasmas • Solenoid-free startup	Integrate & Optimize • Sustained highest pressure plasmas
Operational Capabilities			
• Toroidal Beta, β_T	• → 40 % transiently	• ~ 20% steady, no solenoid	• ~ 40% steady,no solenoid
• Bootstrap Current	• → 50%	• → 60%	• → 70%
• Total Current	• → 1 MA	• ~ 1 MA	• ~ 1 MA
• Elongation	• → 2.5	• → 2.7	• ~ 2.7
• Pulse length	• → 1+ s	• → 1.5 s	• → 3.5 s
• HHFW Power	• → 6 MW	• ~ 6 MW	• ~ 6 MW
• NBI Power	• → 7.5 MW	• ~ 7.5 MW	• ~ 7.5 MW
• EBW Power	• ~ 30 kW	• → 1 MW	• → 4 MW
• CHI Startup	• → 0.4 MA	• → 0.5 MA	• ~ 0.5 MA
• Control	• Ohmic current, shape	• Density, current drive, MHD	• Integrate long pulse & high beta
• Measure	• Profiles of temperature, density, current, & flows	• Turbulence, edge and core	• Turbulence spectra & core structures

Goals and operation capabilities of the NSTX 5-Year Research Plan.

The Princeton Plasma Physics Laboratory is operated by Princeton University under contract to the U.S. Department of Energy. For additional information, please contact: Information Services, Princeton Plasma Physics Laboratory, P.O. Box 451, Princeton, NJ 08543; Tel. (609) 243-2750; e-mail: pppl_info@pppl.gov or visit our web site at: http://www.pppl.gov .