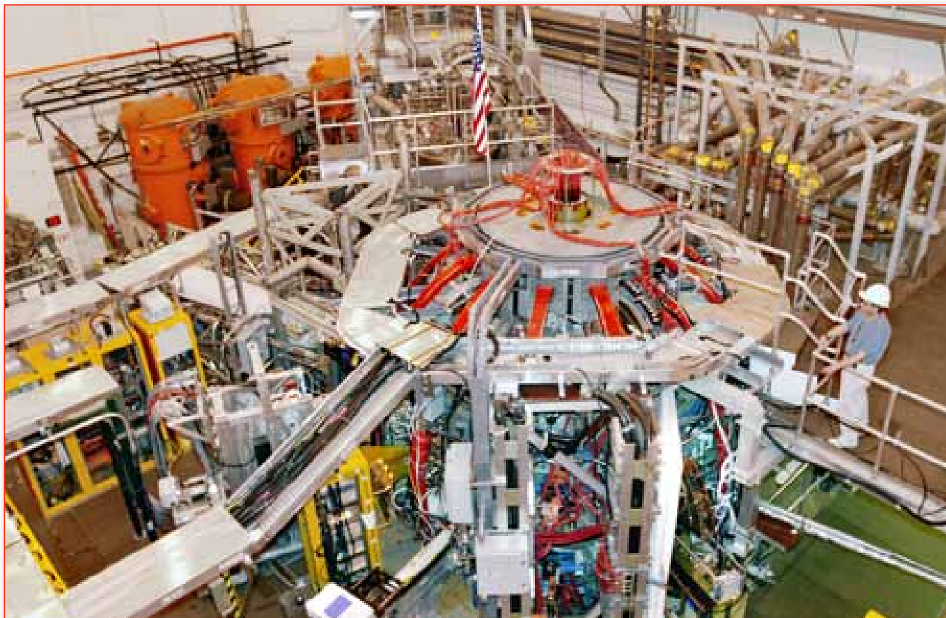
INFORMATION BULLETIN

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

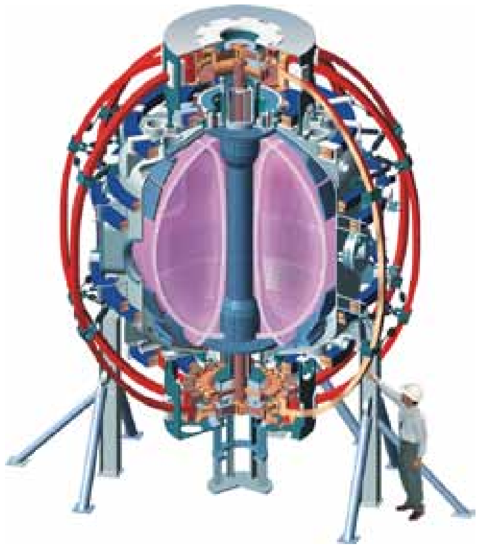


The National Spherical Torus Experiment

The National Spherical Torus Experiment (NSTX) at the U.S. Department of Energy's Princeton Plasma Physics Laboratory (PPPL) is yielding research results that may open an attractive path towards developing fusion energy as an abundant, safe, affordable and environmentally sound means of generating electricity. The NSTX device is exploring a novel structure for the magnetic field used to contain the hot ionized gas, called "plasma", the fuel for the production of fusion energy.

Future fusion power plants will contain plasmas consisting of a mixture of the hydrogen isotopes deuterium and tritium, which can undergo fusion reactions to produce helium, accompanied by a large release of energy. For this to occur, sufficient temperature and pressure must be maintained in the plasma using the insulation provided by a suitably shaped magnetic field. As shown in Figure 1, the magnetic field in NSTX forms a plasma that is a torus since there is a hole through the center, but where the outer boundary of the plasma is almost spherical in shape, hence the name "spherical torus" or "ST". The theory of magneto-hydrodynamics (MHD) describing the interaction of a plasma and a magnetic field shows that the plasma pressure needed to produce self-sustaining fusion in a ST can be maintained with a lower magnetic field strength. Since the cost of a fusion power plant will increase with the strength of its magnetic field, successful development of the ST approach to plasma confinement may lead to economical fusion power plants.

The mission of the NSTX is to establish the scientific potential of the ST configuration as a means of achieving practical fusion energy. If successful, the NSTX could be followed by a larger experiment to explore the issues needed for eventually harnessing fusion power continuously from a reactor. Farther 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 NSTX research program for the next few years is designed both to explore the feasibility of these future steps for fusion development and to contribute to the success of other magnetic fusion experiments, such as the major international experiment ITER, by establishing a firm physics and technology foundation for their design and operation. The experiments on NSTX are being conducted by a collaborative research team of physicists and engineers from 30 U.S. laboratories and universities and 28 international institutions from 11 countries.



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NSTX Accomplishments

NSTX began operation in September 1999. As described below, through 2008, the NSTX research team has made excellent progress, both in exploring the characteristics and effectiveness of the ST configuration and in resolving scientific issues relevant for ITER and future experiments. In the process, the team has implemented numerous improvements in measurement and operational capabilities thereby opening the door to future progress in ST research.

High Plasma Pressure Contained in a Moderate Magnetic Field

The quantity called "toroidal beta" is the ratio of the average plasma pressure to the pressure of the main applied magnetic field and is a measure of the efficiency with which the magnetic field is utilized in containing the plasma. NSTX has attained its original target, a toroidal beta approaching 40%, which is higher by a factor of about 4 than the best values achieved in conventional tokamaks. The high plasma pressure in NSTX was produced by heating the plasma with intense neutral-beam injection (NBI) at power levels up to 7 MW and by carefully controlling the shape of the plasma to avoid violent instabilities that can destroy its confinement. This NSTX achievement provided an important confirmation of the MHD theory on which its design was based. Recently researchers have developed methods to sustain high beta by employing a set of small magnetic field coils controlled by feedback to counteract the growth of other types of MHD instability that develop on longer timescales at high plasma pressure.

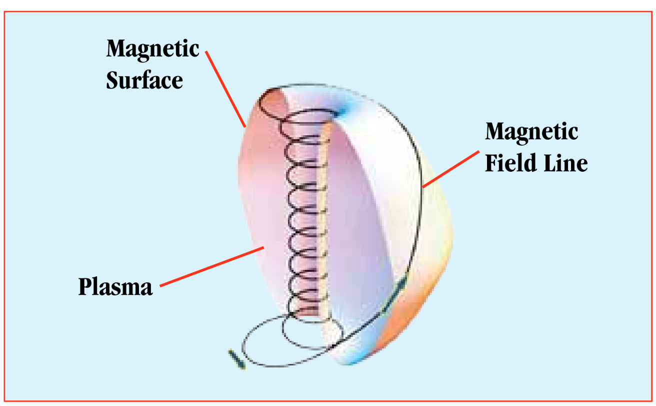

Fig. 1 Schematic view of shape of plasma in a "spherical torus" showing the path traced by a typical magnetic field line.

Efficient Confinement of Plasma Energy

Another important quantity for assessing progress towards fusion goals is the energy confinement time which is a measure of how long the plasma can contain the energy used to heat it. Instruments, referred to as diagnostics, to measure the energy confinement time and other properties of the plasma in NSTX have been developed and operate routinely during experiments. With NBI heating, the energy confinement time in NSTX has been consistently 1.5 to 2.5 times that expected from the results accumulated from conventional tokamaks run in their basic (unenhanced) mode of operation. This result is very favorable for the potential of the ST in fusion energy development.

Recently diagnostics have been installed for measuring the turbulence that can develop when multiple small-scale instabilities grow in a hot plasma. This turbulence may cause losses of energy and particles across the confining magnetic field. The NSTX diagnostics have revealed possible roles for two classes of instabilities. The first, the electron temperature gradient (ETG) instability, is uniquely amenable to measurement in the ST because of the low magnetic field. Figure 2 shows how a state-of-the-art diagnostic, measuring the scattering of a beam of microwaves by small nonuniformities in the plasma density, has revealed the growth of turbulent plasma fluctuations when the plasma is heated so that the electron temperature gradient exceeds the critical level at which the ETG instability starts to grow. The second, the global Alfvén eigenmode (GAE) is potentially important for future experiments such as ITER where fusion reactions will create a large population of energetic alpha particles in the plasma.

Developing Methods for Sustained Operation

To confine plasma particles in both the conventional tokamak and the ST, a large electric current must flow in the plasma itself to enclose the central hole in the torus. This current is usually generated in today's experiments by the electromagnetic induction produced by changing the current in a multi-turn coil or solenoid through the central hole. However, to pave the way towards compact ST fu-

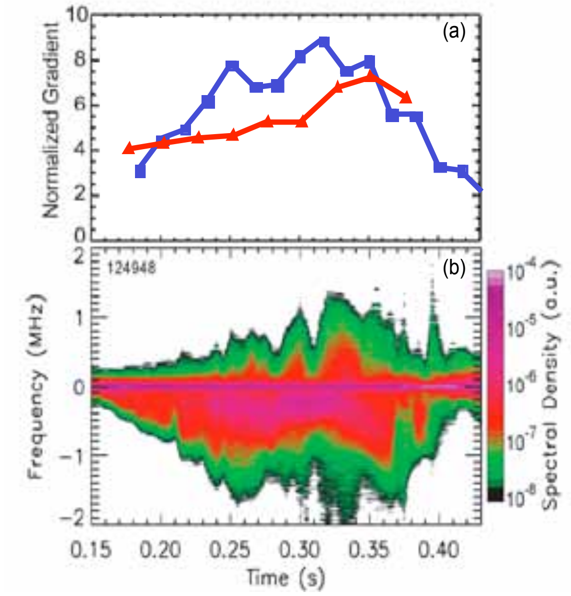

Fig. 2 (a) Evolution of measured (blue) normalized electron temperature gradient and the calculated threshold gradient for onset of the ETG instability (red); (b) a color contour plot of the measured plasma fluctuation level shows that turbulence characteristic of the ETG instability develops over a wide frequency range when the instability threshold is exceeded.

sion plants, in which there will be little room for a central solenoid, NSTX is investigating and developing the physics of several alternative methods for initiating and sustaining the plasma current.

In NSTX successful current initiation without induction from the central solenoid has been demonstrated using the technique of coaxial helicity injection (CHI), originally developed in experiments at the University of Washington. A record current of up to 160 kiloAmpères has been obtained in NSTX by this method. This initial current has now been ramped up to 700 kiloAmpères by subsequent induction, demonstrating the compatibility of CHI with conventional operating scenarios. Future experiments will combine CHI initiation with currents driven by the plasma heating systems available on NSTX.

NBI-driven Current and Bootstrap Current

The NBI heating applied to NSTX plasmas creates a population of energetic ions which can circulate many times around the torus before they slow down by collisions with the background plasma. In doing so, they carry a current which can collectively amount to several hundreds of kilo-Ampères even though the ion current of the input beam is only of order 50 Ampères.

In the 1970s, it was predicted theoretically that in a toroidal plasma, the difference in pressure between the hot plasma in the central core and the cooler edge region would actually drive a toroidal plasma current due to the friction force between the plasma particles. This current, called "bootstrap current" since it is self-generated within the plasma, was experimentally confirmed in tokamaks in the 1980s. Because of its shape and its high beta, the bootstrap current is very effective in the ST, particularly in discharges in the H-mode of plasma confinement where the pressure gradient creating the bootstrap current occurs close to the plasma edge. In optimized NSTX H-mode discharges heated by NBI, the combination of NBI-driven and bootstrap current has now been increased to about 60% of the total plasma current, thereby relaxing the requirement for induction to sustain the current.

High-Harmonic Fast-Wave and Electron Bernstein Wave Current Drive

Radio-frequency (RF) electromagnetic waves launched into the plasma at many (10 to 20) times the frequency with which ions gyrate around the magnetic field can heat electrons and, if appropriately directed, also drive plasma current. In NSTX, these high-harmonic fast-waves (HHFW) are launched into the plasma by a multi-element antenna outside the plasma. With 3 MW of HHFW power, the electron temperature has been increased from about 5 million to above 50 million degrees Kelvin. In addition, an advanced spectroscopic diagnostic utilizing the motional Stark effect has revealed changes in the magnetic field within the plasma consistent with current being driven by directed HHFW power. The data shown in Figure 2 were obtained from plasmas heated by HHFW in NSTX.

Another type of higher frequency plasma wave, the electron Bernstein wave (EBW), launched into the plasma at one or two times the frequency with which electrons gyrate around the magnetic field can also heat the electrons and drive plasma current. In this case, the power is transferred to the electrons through a resonant process determined by the magnetic field so the effects can be produced at a

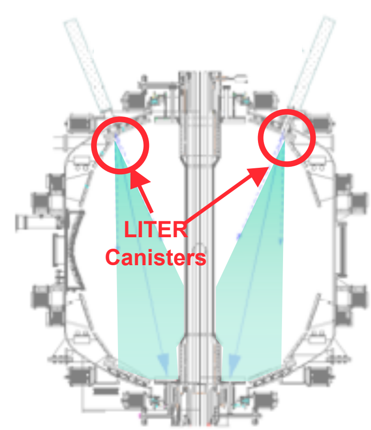

Fig. 3 Cross-section of NSTX showing location of two lithium evaporators (LITERs) mounted on retractable probes. Shading represents the plumes of lithium vapor which condense on the PFCs in the lower part of the plasma chamber.

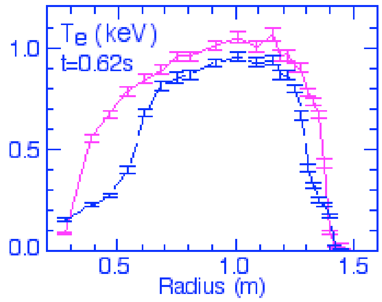

Fig.4 Measured profiles of the electron temperature (Te) in two discharges with the same conditions of plasma current and NBI heating power before (blue) and after (red) the application of 260mg of lithium with the evaporators. The lithium coating on the PFCs produces a broadening of the profile indicating an improvement in plasma confinement.

precise location in the plasma. Research is underway on NSTX investigate the physics underlying this process to enable a megawatt-level EBW system to be installed on NSTX in the future.

Managing the Plasma Heat Flux

The high power levels anticipated in both conventional tokamak fusion reactors such as ITER and compact STs such as a CTF will create large heat loads on the plasma facing components (PFCs). It is therefore important to develop and test both advanced PFCs capable of withstanding high heat fluxes and methods to spread the heat over the PFCs. In NSTX, experiments are underway and being planned to investigate these critical issues employing advanced diagnostics and applying a variety of operational techniques, such as enhancing radiation from the plasma boundary to reduce the power conducted to the PFCs. As illustrated in Figures 3 and 4, innovative approaches using evaporated lithium coatings on PFCs to suppress the return of particles which escape from the plasma have been tested and shown to be effective in improving the plasma confinement and in preventing the instabilities called Edge-Localized Modes

The NSTX Team.

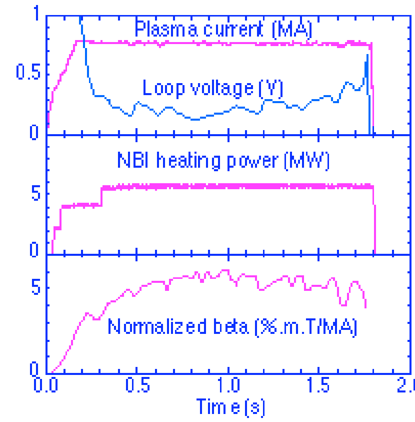

Fig. 5 Waveforms from the discharge with the record pulse length in NSTX. The plasma current is sustained at 0.75 MA for 1.5 s by a very small voltage, averaging 0.25 V, as a result of the current driven by the 6 MW of NBI heating and the large bootstrap current accompanying the high normalized beta. This discharge benefited from both feedback control of instabilities and lithium coating of PFCs.

(ELMs) that frequently occur in tokamak and ST plasmas in their high-confinement mode (H-mode) of operation.

The combined benefits of optimizing the NBI-driven and bootstrap currents, applying feedback control of MHD instabilities and improving the confinement with lithium coating of the PFCs has recently allowed the plasma pulse length in NSTX to be extended to about 1.8 sec, as seen in Figure 5.

Future Research Plan

In June 2008, the Department of Energy convened an international panel to review a new NSTX Five-Year Research Plan for the years 2009 to 2013. This plan aims to understand, control, and optimize ST plasmas, and extend the knowledge base of plasma science to benefit both the ST approach to fusion and conventional tokamaks as embodied in ITER. Several upgrades to the capabilities of NSTX are proposed in the plan. Following the success of the experiments with lithium-coated PFCs, the NSTX team is now preparing to deploy a liquid-lithium surface module in NSTX in 2010. A new central conductor bundle for the toroidal field coil is proposed to provide higher magnetic field and longer pulses and it is planned to install a second neutral beam injector to drive current near the outer edge of the plasma for improved MHD stability in extended plasma pulses. The plan received enthusiastic endorsement by the review panel. The rapid progress already made by the NSTX team has positioned NSTX to embark on this exciting research plan.

The PRINCETON PLASMA PHYSICS LABORATORY is operated by Princeton University under contract to the United States 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: www.pppl.gov .