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by

M.G. Bell for the NSTX Research Team

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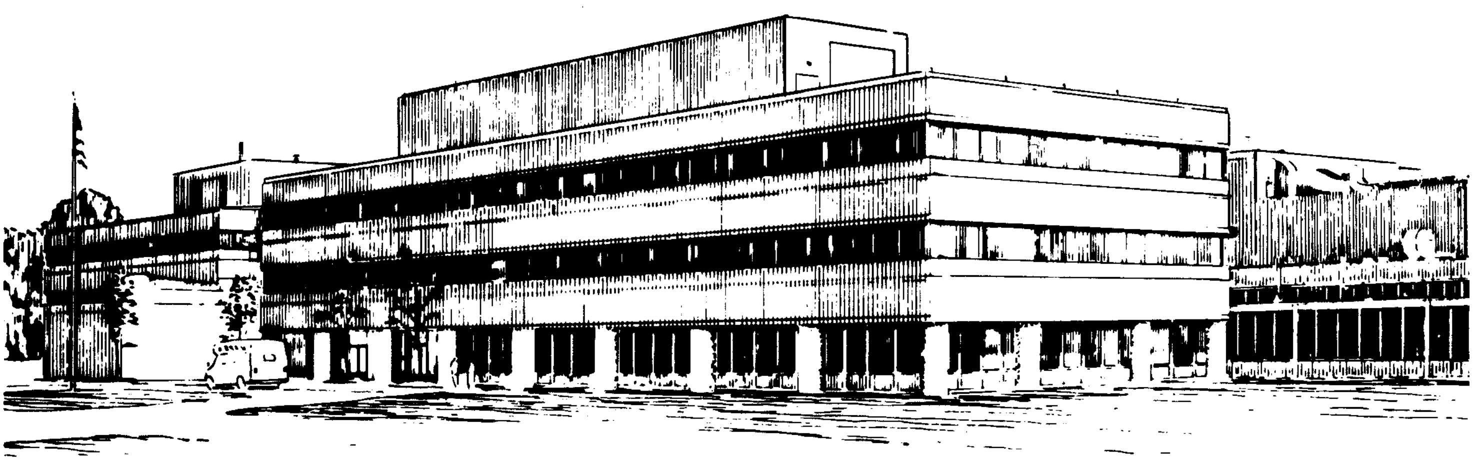

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Physics Results from the National Spherical Torus Experiment

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The National Spherical Torus Experiment (NSTX) at the Princeton Plasma Physics Laboratory is designed for studying toroidal plasma confinement at very low aspect-ratio, A = R/a = 0.85m/0.68m ~ 1.25, with cross-section elongation up to 2.2 and triangularity up to 0.5, for plasma currents up to 1 MA and vacuum toroidal magnetic fields up to 0.6 T on axis. Conducting plates are installed close to the plasma on the outboard side to stabilize kink modes. This should permit operation with toroidal-β approaching 40% [1]. The plasmas will be heated by up to 6 MW High-Harmonic Fast Waves (HHFW) at a frequency 30 MHz and by 5 MW of 80 keV deuterium Neutral Beam Injection. Inductive plasma startup can be supplemented by the process of Coaxial Helicity Injection (CHI).

The first operational phase from September 1999 to January 2000 concentrated on commissioning the control, heating and current-drive systems, developing the range of plasma configurations possible, measuring the poloidal flux consumption and characterizing the operational space. The first wall was prepared for plasma operation by a combination of bakeout (300°C on the center stack and ~200°C on the stabilizer plate tiles) and glow-discharge conditioning, initially in deuterium (to remove residual hydrogen) and then in helium [2]. As shown in Fig. 1 , low aspect-ratio plasmas with inductively driven currents up to 1 MA have been achieved at a toroidal magnetic field of 0.3 T with quasi-steady elongations 1.5 - 2.0 and triangularity up to 0.4. Higher elongation, up to 2.2, and triangularity, up to 0.6, have been reached transiently. The plasma current is initiated in deuterium at a pressure of 8 – 12 mPa by inducing a toroidal loop voltage of 3 - 4 V. Breakdown may be assisted by applying a 10 kW, 20 ms pulse of 18 GHz RF power, resonant with the electron cyclotron frequency at a radius of ~0.4 m.

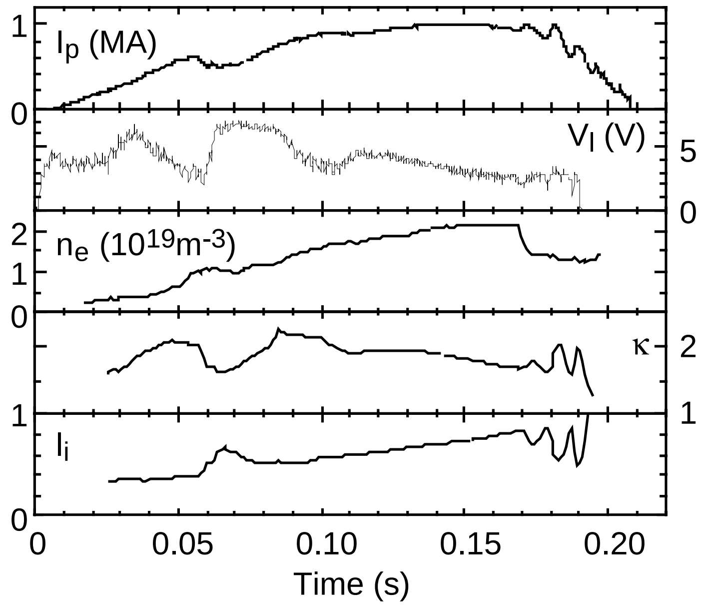

Fig. 1 Evolution of plasma current (I_p), loop voltage (V_l), line-average electron density (n_e), elongation (κ) and internal inductance (l_i) for a 1 MA inductive discharge with aspect ratio 1.28.

In this first phase, the gap between the plasma boundary and the first wall at the outboard midplane, the plasma vertical position and the plasma current were feedback controlled using digital signal processing to control the currents in two pairs of poloidal field coils and

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the central solenoid. The shape of the outer boundary was controlled by pre-programming the currents in other poloidal field coils. The equilibrium configuration is analyzed after each shot with the EFIT code [3] using the measured coil currents and data from up to 75 sensors which measure the poloidal flux and magnetic field components outside the plasma. Plasmas limited on the inboard side by carbon tiles, single-null (asymmetric) and double-null (symmetric) divertor plasma configurations have been produced as shown in Fig. 2. In addition to providing the shape of the boundary, the EFIT analysis yields the internal inductance parameter, l_i, and energy content. In future experiments, real-time analysis of the equilibrium and feedback control of many more discharge parameters will be implemented [4].

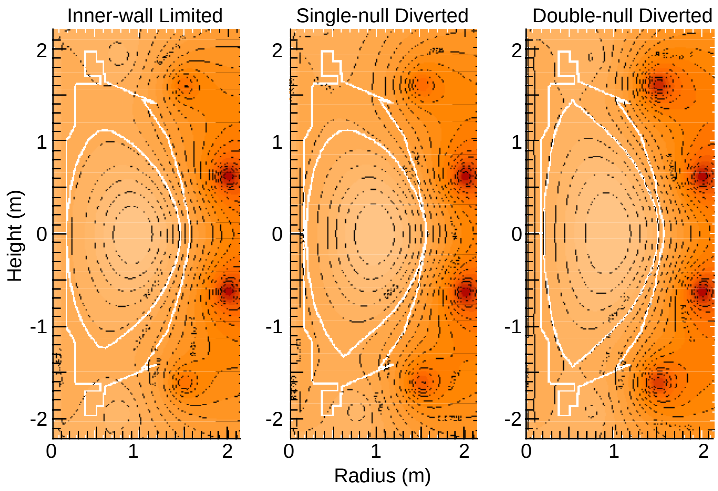

Fig. 2 Cross-sections from EFIT analysis of plasmas run in the three main equilibrium configurations in NSTX: a) inner wall limited; b) lower single-null divertor; c) double-null divertor

The poloidal flux consumption during inductive startup was studied as a function of the plasma current and its rate of rise. For peak currents in the range 0.6 – 0.8 MA, the flux consumption was minimized by programming a rate of rise of 5 – 7 MA/s. The Ejima coefficient, defined as the normalized resistive flux consumption to reach a given current I_p, $C_e = \int V_{res} dt/\mu_0 R_p I_p$, where V_{res} is the resistive part of the loop voltage at the plasma surface and R_p is the major radius of the magnetic axis, was as low as 0.35, similar to values measured for conventional aspect ratio tokamaks [5].

A variety of MHD phenomena has been observed in the ohmically heated plasmas, including sawtooth oscillations, internal and global reconnections, kinks and disruptions. These are evident in the signals from magnetic pickup coils outside the plasma, from collimated soft x-ray detectors and from fast optical TV images of the plasma. During the current ramp, MHD instabilities frequently occurred and affected the plasma evolution, as seen for example in Fig. 1 at 0.045s. The increase in the internal inductance indicates rapid redistribution of the plasma current. Magnetic pickup coils showed the presence of rapidly growing perturbations which are believed to be double-tearing modes [6] occurring when the q-profile becomes double valued as a result of the rapid rise of the current initially. Higher current ramp rates increased the MHD activity, causing greater flux consumption. After the period of tearing instability, sawtooth relaxations appeared on the soft x-ray emission along central chords and the plasmas became otherwise quiescent unless limits in density or the safety factor, q, were encountered.

The operational density limit with deuterium gas puffing was investigated as a function of plasma current and toroidal field. The approach to the limit was manifested by the appearance of slowly growing, rotating MHD instabilities which eventually locked, followed by a series of global reconnection events and termination of the plasma current. The highest line-average density was $2.7 \times 10^{19} m^{-3}$. Over the current range 0.3 - 0.6 MA, the density reached with gas fueling was about 60% of the Greenwald limit [7]. During this first phase of operation, the copper stabilizer plates close to the plasma boundary were only partially covered by graphite tiles. As a result, copper was a significant impurity and this may have reduced the density limit. The copper surfaces will be entirely covered in future operation.

The operational q-limit was studied by decreasing the toroidal field during the approach of the plasma current to its maximum. As the q-limit was approached, large distortions of the plasma surface became apparent on the TV image and the soft x-ray data showed the growth of large perturbations with poloidal and toroidal mode numbers m/n = 2/1 respectively. This occurred as the radius of the q = 2 surface from the EFIT analysis approached within about 0.1m of the plasma boundary. The growth of the mode was followed by termination of the plasma current. The highest average rate of decay of the plasma current observed so far is 0.12 MA/s, less than in comparable higher aspect-ratio tokamaks. Plasmas with q_{95}, the value of q at the surface with 95% of the poloidal flux at the boundary, of 2.6 have been produced; the equivalent cylindrical q, $q_{cyl} = 5a^2B_T(1 + \kappa^2)/2RI_p$, was as low as 1.3. The toroidal plasma current slightly exceeded the total poloidal (or threading) current of the toroidal field coil transiently in some plasmas with the toroidal field ramp-down.

The energy confinement time in the ohmically heated plasmas has been estimated from the EFIT magnetic analysis to be in the range 15 – 25 ms, although this could not be independently confirmed because measurements of the density and temperature profiles were not available. In general, the confinement time increased with plasma density and current. The values were consistent with estimates based on tokamak scalings [8] for ohmic confinement. The plasma energy reached 48 ± 10 kJ, corresponding to a toroidal beta β_T of 8.6 ± 1.8 %, defined with respect to the vacuum field at the plasma geometric center.

Using initially only 8 of the 12 coupler elements, up to 1 MW of HHFW heating power was coupled to the plasma for 50 ms during the current flattop phase, and up to 2 MW was coupled briefly [9]. Based on density profiles near the edge of the plasma measured with a frequency-scanning reflectometer, the coupling resistance of the RF waves has been calculated and found to be in approximate agreement with the measurements. The successive coupler elements were operated in either a 0-π-0-π or a 0-π-π-0 phasing, corresponding to toroidal wave-numbers $k_{tor} \approx 13\ m^{-1}$ (lower phase velocity) and $k_{tor} \approx 9\ m^{-1}$ (higher phase velocity), respectively. The RF frequency of 30 MHz is typically 10 – 15 times the fundamental cyclotron frequency in the core of NSTX plasmas. Under these conditions, the RF waves are expected to couple mainly to the electrons. The magnetic analysis showed small increases in the stored energy during the RF pulse for the higher k_{tor} phasing. Corresponding indications of electron heating in the plasma core have been observed on filtered x-ray detectors. Analysis of this data is complicated by metal line emission, but modeling suggests an electron temperature rise from 0.8 to 1.2 keV early in the RF pulse. The heating was not observed for the lower k_{tor} phasing.

Coaxial Helicity Injection (CHI) involves creating a discharge between the inner and outer lower divertor plates which are insulated from each other by ceramic breaks in the vacuum vessel. The discharge is produced in deuterium at a pressure typically 0.25 – 1 Pa by a pulsed supply capable of 1 kV and 25 kA. The polarity of the electrodes is chosen so that, in the presence of a toroidal field, and an appropriate poloidal field, the $\mathbf{J} \times \mathbf{B}$ force is directed into the plasma chamber. To reach an approximately force-free equilibrium on the open field lines, a strong toroidal current develops in the discharge. Experiments [10] have shown that by a process of magnetic reconnection, some of this toroidal current can be transferred onto closed magnetic surfaces. The experiments in NSTX have produced 100 ms pulses with up to 130 kA of toroidal plasma current without magnetic induction [11], as shown in Fig. 3. The ratio of the toroidal current to the injected poloidal current was as high as 10 and scaled appropriately with the applied fields. With the diagnostics currently available, it has not yet been possible to determine from the external field and flux measurements whether any of the toroidal current is flowing on closed surfaces. However, during the interval 0.06 - 0.10 s, MHD fluctuations similar to those expected to accompany the reconnection process were observed on the Mirnov coils.

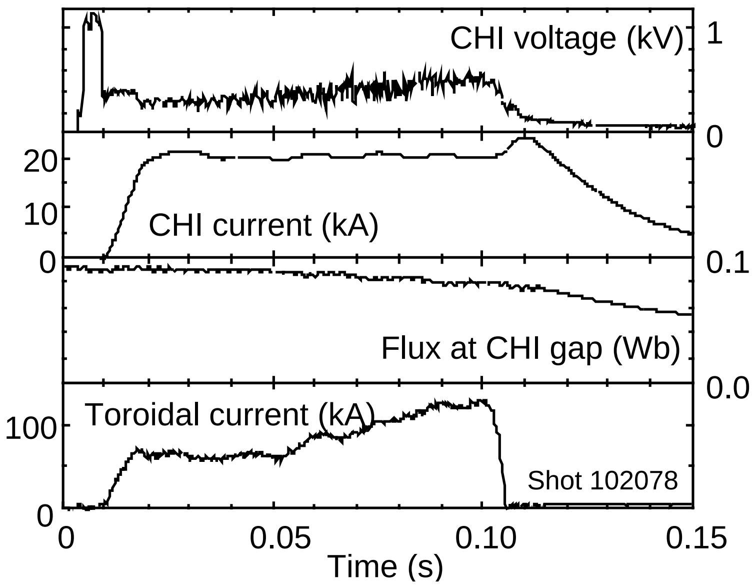

Fig. 4 Waveforms during a CHI discharge. The deuterium fill pressure was 0.4 Pa.

NSTX Status

The NSTX facility is currently being prepared for operation following a four-month shutdown for installation of the NBI system, final plasma-facing components, additional shielding and baffles around the CHI electrodes and new diagnostics. These include a multi-pulse, multi-point Thomson scattering system and additional quantitative spectroscopy for both intrinsic and NBI-excited emission. Plasma operation is scheduled to resume in July and NBI experiments to begin in October 2000.

Acknowledgements

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