U.S. Department of Energy

Field Work Proposal

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1. Work Proposal Title: NSTX

2. Proposal Short Name: NSTX

3. Fiscal Year: 2008

4. Proposal Purpose: Budget Call

5. Proposal Reason: Ongoing Work

6. Internal Lab Number: 11$$ (1101/1135/1136/1150/1151)

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8. Estimated Begin Date: 10/01/2007

9. Estimated End Date: 9/30/2008

10. HQ Program Manager Organization: SC-24.2 - OFES Research Division

11. Contractor Org Code and Name: Princeton University/Princeton Plasma Physics Laboratory

12. Budget and Reporting Code: AT5505/AT5015010

13. Project Objective Description:

The National Spherical Torus Experiment (NSTX) is a major U.S. facility designed to study the physics of fusion plasmas magnetically confined in a very low aspect-ratio Spherical Torus (ST) configuration. This unique NSTX characteristic, which allows the achievement of a high plasma pressure relative to the applied magnetic field, complements the tokamak in addressing the overarching issues in magnetic fusion energy science, encompassing macroscopic MHD stability, turbulence and transport, wave-particle interactions, solenoid-free current generation and sustainment of magnetic flux, and the plasma interface with its surrounding environment. The unique operating regimes of NSTX provide high leverage to address several important issues in the physics of burning plasmas to optimize the performance of ITER. The NSTX Program further aims to determine the attractiveness of the compact ST in cost-effective development of fusion power using a Component Test Facility, as indicated in the present DOE Office of Science Strategic Plan.

During FY 2006-2008, the NSTX Team proposes to make major progress in these directions, by performing research to reach a comprehensive set of research milestones described herein, and participating in collaborative joint research with other fusion programs, including the International Tokamak Physics Activities (ITPA), on topics of broad interest in fusion plasma physics. Research on NSTX has hitherto been made more successful by strong collaboration with numerous research institutions in the U.S. and in other countries.

To provide measurements needed to carry out this scientific research, the NSTX Team proposes to install and commission during FY 2006-2008 upgraded plasma diagnostic systems. These include a number of diagnostics provided by NSTX Team members from U.S. laboratories outside PPPL.

The NSTX Facility proposes to operate, maintain, and enhance the NSTX device, auxiliary systems, and site infrastructure during FY 2006-2008 to support the planned research efforts. Onsite support for equipment provided by offsite Team members will also be provided. Preparation for future upgrades, as driven by the longer-term NSTX goals, will also be carried out.

14. Abstract:

The purpose of NSTX is to address issues important for Fusion Energy Sciences (FES) using the special ST properties. For this purpose, the NSTX National Team aims to:

1) Determine the physics principles of ST, utilizing its low aspect-ratio (A ~ 1.5) and very high ratios β of plasma pressure to magnetic pressure (β as high as unity);
2) Support preparation of burning plasma research in ITER through the International Tokamak Physics Activities (ITPA) and benefit from it;
3) Complement and extend A ~ 3 and lower-β experiments in addressing key scientific issues of toroidal fusion plasmas;
4) Complement ITER by establishing attractive configurations for a Component Test Facility (CTF) and Demonstration Power Plant.

The approach is to utilize the intrinsic scientific capabilities of NSTX to support ST-specific research as well as research to optimize the burning plasma performance in ITER. The NSTX research plan is therefore organized to maximize its contributions to both these goals. While all of the research on NSTX is naturally relevant to ST, about three-quarters of it is relevant to advancing the scientific basis for the tokamak, and about one-third of it aims to address the 2006 goals of the ITPA joint experiments in support of ITER optimization.

With this approach, the NSTX Program aims to deliver, as its objectives for FY2006-2008, the following anticipated benefits to the national FES Program:

1) Address and resolve key issues defined by the 2005 FESAC Priorities Panel, which can be organized into the following topical areas.

Transport and Turbulence: to clarify the physical processes that govern heat, particle and momentum confinement.

Macroscopic Stability: to determine the role of magnetic structure on plasma pressure and bootstrap current.

Wave-Particle Interaction: to elucidate the mechanisms of electromagnetic waves and modes in sustaining and controlling hot plasmas.

Boundary Physics: to understand the interface between fusion plasmas and normal-temperature surroundings.

Start-up Ramp-up and Sustainment: as uniquely important to ST, to determine the physical processes of magnetic flux generation and reconnection.

Physics Integration: to understand the synergistic effects of external control and self-organization in sustaining the low A and high beta ST plasmas.

2) The NSTX research will benefit the ITER burning plasma program through the above contributions. In particular, NSTX plans to contribute to 22 of the 2006 ITPA goals in all key topical areas aimed at resolving the remaining key physics issues of ITER, namely:

Confinement Data Base and Modeling;

Transport Physics;

Pedestal and Edge;

MHD, Disruptions and Control;

Steady-State Operation;

Diagnostics.

3) The NSTX research will establish key elements of the physics basis for CTF and define the additional elements that will require data from a Performance Extension ST experiment (~ 3 MA in current and ~ 1 T in magnetic field). This will contribute to a unique and important capability of the U.S. FES Program to lead the world in an upcoming CTF program to test and develop fusion nuclear components needed for practical fusion power.

The NSTX tasks and deliverables for FY2006-2008 are contained under the Section 18 below (Milestones).

15. Category of Research: Applied

16. Category of Work: Research

17. Research Area: Plasma Physics and Fusion Technology

18. Milestones

a. FY 2006

Projected Milestone Date: September 2006

Milestone Description: R(06-1) Perform highly localized measurement of magnitude of high-k turbulence.

The role of short-wavelength turbulence, i.e. on the scale of a few radii of the electron gyration around the magnetic field, in governing electron thermal transport is an unresolved issue that is important for future fusion-grade plasmas, such as those in ITER, which will be heated by alpha particles. Experiments on NSTX have indicated that electron thermal conduction can vary widely and often can dominate over the ion thermal conduction in high-confinement and high-beta plasmas. Turbulence calculations indicate that short-wavelength microinstabilities may play a strong role in such cases. Advanced microwave scattering techniques have been implemented on NSTX to enable highly localized measurements of these short-wavelength fluctuations.

Projected Milestone Date: September 2006

Milestone Description: R(06-2) Characterize effectiveness of closed-loop Error Field control with ITER-like EF/RWM coils.

As the pressure in strongly shaped NSTX plasmas is raised toward and sustained near the "wall-stabilized" ideal limit for durations longer than the time scales of eddy-current decay in the nearby wall, Resistive Wall Modes (RWMs) can grow and limit plasma stability. In strongly rotating plasmas, theory predicts that by controlling the error fields the onset of the RWM can be delayed or suppressed. Active feedback of the amplitude of the resonant error fields will be studied to characterize the effectiveness of this approach in FY06. Maintaining stability near the "wall-stabilized" pressure limit is of interest to performance improvements in NSTX, burning plasma experiments such as ITER, and future fusion energy producing devices.

Projected Milestone Date: September 2006

Milestone Description: R(06-3) Assess closed-flux generation using transient Coaxial Helicity Injection (CHI).

Elimination of the central solenoid is an important consideration for future toroidal confinement devices, which will then require alternative methods for initiating the plasma current. Experiments were performed during FY05 applying the new technique called "transient CHI," which was successfully developed recently on the HIT-II Concept Exploration experiment. In this method, the plasma current is rapidly generated by discharging a capacitor bank between coaxial electrodes in the presence of a toroidal magnetic field and an applied poloidal field configuration that propels the plasma rapidly into the chamber. When the injected current is abruptly terminated, magnetic reconnection occurs near the injection electrodes to form closed flux surfaces. On NSTX in FY05 this method achieved persistent toroidal plasma currents up to 60 kA with camera images indicating closure of the magnetic flux surface. These interesting experimental results will be analyzed and submitted for publication.

Projected Milestone Date: September 2006

Milestone Description: R(06-4) Characterize effects of lithium wall coating on recycling.

In 2005, the NSTX lithium pellet injector was used to coat the contact areas of the plasma facing components with small amounts of lithium. This produced a transient reduction in the particle recycling for both limited and divertor discharges until the coating was depleted. In 2006, a lithium evaporator will be installed to coat a substantial fraction of the plasma facing surfaces in NSTX. Experiments will be performed to characterize the effectiveness of using lithium coating to control particle recycling.

Projected Milestone Date: September 2006

Milestone Description: F(06-1) Operate NSTX Facility for 11 Experimental Run Weeks.

Projected Milestone Date: June 2006

Milestone Description: F(06-2) Install and commission a lithium evaporator on NSTX.

A lithium evaporator will be designed, fabricated, installed, and commissioned to enable application of lithium coating on a substantial fraction of the plasma facing components in NSTX.

Projected Milestone Date: September 2006

Milestone Description: D(06-1) Complete the shop fabrication of port flanges, shutters, and divertor plates for an interim poloidal rotation diagnostic using charge-exchange recombination emission.

The shop fabrication of new port flanges, shutters, and divertor plates in support of the FY 2007 outage for an interim poloidal charge-exchange spectroscopy (P-CHERS) diagnostic will be completed. This diagnostic will be installed in FY 2007 to measure the spatial profile of the poloidal plasma flow in the outer half of the minor radius in FY 2007. These will enable measurement in FY 2007-2008 of the poloidal flow, which contributes to the shearing rate and plays a role in the suppression of large-scale turbulence that is believed to be responsible for the good ion confinement in NSTX. The rotation measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

b. FY 2007

Projected Milestone Date: September 2007

Milestone Description: R(07-1) Study variation of local high-k turbulence with plasma conditions.

The results of Milestones R(06-1) are expected to establish the basis for performing in FY2007 a detailed investigation of the variation of local high-k turbulence with core plasma conditions. The high-k scattering system on NSTX, capable of measuring core density fluctuations with radial wavenumbers 2 – 22 cm^{-1} with excellent spatial resolution, will enable detailed studies to be made of electron-scale-length fluctuations; data from this diagnostic will be augmented by correlation and quadrature reflectometry and far-infrared tangential polarimetry. Systematic scans will be performed of parameters affecting electron thermal transport. The correlations amongst these measurements, the experimentally inferred transport and simulations of fluctuations with a suite of turbulence codes will be determined over a range of plasma conditions. Progress in this area will be of high interest to the goals of developing a predictive understanding of transport and turbulence, and to the optimization of the burning plasma in ITER and the confinement in CTF.

Projected Milestone Date: September 2007

Milestone Description: R(07-2) Characterize effectiveness of closed-loop RWM control & dependence on rotation using ITER-like control coils.

Research on Milestone R2(06-2) will have characterized the effectiveness of feedback control of the resonant field errors on stabilizing the wall-coupled, pressure-limiting RWMs over a range of plasma conditions. As an important extension of this work, active feedback on the amplitude of the RWMs is expected to enhance plasma stability beyond feedback on the resonant error fields alone, particularly accounting for the effects of plasma rotation. Externally applied field errors from a set of six feedback coils will be used to reduce plasma rotation or to compensate the device intrinsic error fields to maintain high rotation. The physics of plasma damping by resonant field errors, via interactions with electromagnetic and sound-wave perturbations, will be clarified. These results, together with further improvements in the speed of the control circuits and the feedback algorithms, will be used as a basis for testing active feedback control of the RWM under conditions where such control is needed to maintain stability. Stabilization of RWMs as the plasma pressure approaches the "wall-stabilized" limit by direct feedback and by plasma rotation is of interest to performance improvements in NSTX and burning plasma experiments such as ITER and CTF.

Projected Milestone Date: September 2007

Milestone Description: R(07-3) Measure, identify & characterize modes driven by super-Alfvénic ions.

NSTX will in FY 2007 measure, identify, and characterize magnetosonic modes that are driven by energetic ions traveling faster than the Alfvén speed. Such super-Alfvénic ions are expected in ITER and CTF. A suite of tools will be applied to this research. Fast soft x-ray cameras, magnetic sensors, reflectometers, and interferometers will measure the mode properties. Energetic neutral particle analyzers, fast-ion loss probes and neutron detectors will diagnose the energy distribution of these super-Alfvénic ions to reveal the effects of the modes. Diagnostics, including Thomson scattering, charge-exchange recombination spectrometry, and motional Stark effect polarimetry will measure the plasma profiles. The combined data will be compared with extensive modeling, simulation, and transport analysis to establish an understanding of modes driven by the super-Alfvénic ions. The results will shed light on how the modes can affect the burning plasma performance in ITER and CTF.

Projected Milestone Date: September 2007

Milestone Description: F(07-1) Operate NSTX Facility for 12 Experimental Run Weeks.

Projected Milestone Date: September 2007

Milestone Description: D(07-1) Install and commission an interim poloidal rotation diagnostic using charge-exchange recombination emission spectroscopy.

The interim configuration of this diagnostic, with reduced spatial coverage and resolution, will be installed in FY 2007 to measure the spatial profile of the poloidal plasma flow in the outer half of the minor radius. This poloidal flow contributes to the total plasma "shearing rate" and thus plays a role in the suppression of large-scale turbulence that is believed to be responsible for the good ion confinement in NSTX. The rotation measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

c. FY 2008

Projected Milestone Date: September 2008

Milestone Description: R(08-1) Measure poloidal rotation at low A to constrain theory.

With the anticipated progress during FY2006-2007 in determining the variation of local high-k turbulence with plasma conditions, measurements of poloidal flow velocity will provide key data needed to assess the role of flow shear in controlling plasma turbulence and transport in NSTX plasmas. The research will utilize the new poloidal rotation diagnostic, based on charge-exchange recombination emission spectroscopy, to be installed and commissioned during FY2007, and build on the broad progress already made and planned on NSTX in transport and turbulence. Progress in this direction contributes to the scientific basis needed to optimize the burning plasmas in ITER and CTF.

Projected Milestone Date: September 2008

Milestone Description: R(08-2) Study edge/divertor at low plasma collisionality with ITER-level heat fluxes.

Results from Research Milestones in FY2006-2007 are expected to provide the information needed to produce long-pulse, high-performance plasmas in NSTX over ranges in the edge shaping, temperature, density, H-mode pedestal conditions, and the types of Edge Localized Modes (ELMs). In particular, research using lithium wall coating to reduce recycling will be aimed at obtaining low density, and hence low collisionality, in the plasma edge. This will establish a basis for assessing the plasma conditions and operational requirements for handling edge heat fluxes up to the level of 10 MW/m^2 anticipated in ITER. Progress in understanding the plasma boundary for high levels of the edge power relative to major radius is important to achieving reproducible, sustained burning plasmas in ITER. Heat and particle fluxes from H-mode plasmas in NSTX will be characterized over a range of ELM conditions, using a suite of diagnostics for the plasma edge, the scrape-off layer (SOL), and the plasma facing components. The effects of lithium coating on the edge fluxes will be measured, together with its potential benefits on core plasma properties. The effects of the varying ELM conditions will be documented to determine their role in the power flow and the balance between cross-field and parallel transport in the SOL. Operating regimes that show improved plasma core conditions with control of the ELMs and edge fluxes will be characterized and documented. Simulation codes for the edge heat and particle transport, instabilities and turbulence will be applied to interpret the data, to clarify the mechanisms that drive these fluxes and to achieve the conditions of interest. The results will contribute to a predictive understanding of plasmas fluxes on NSTX and the burning plasmas in ITER and CTF.

Projected Milestone Date: September 2008

Milestone Description: R(08-3) Perform long-pulse plasmas in conditions relevant to CTF.

Conditions will be studied on NSTX in which the toroidal plasma current is maintained for durations beyond the plasma current redistribution time with minimal magnetic flux induction from the central solenoid. Strong neutral-beam injection (NBI) will be applied in low density plasmas, obtained using such techniques as lithium wall coating to be developed during FY2006-2007. This will increase the fraction of current driven by the NBI towards that anticipated in CTF. Such plasmas would be developed from discharges already produced in NSTX in which the solenoid-induced loop voltage has been reduced to the range 0.1 - 0.2 V for durations much greater than the current redistribution time. This was achieved by a combination of optimizing the current ramp-up, an early transition to the H-mode and strong plasma shaping to increase plasma stability and to minimize the impact of ELMs. Possible synergistic effects between the current drive mechanisms will be investigated to determine the optimal plasma scenarios. Simulation codes, which will have been benchmarked through comparison with recent NSTX data, will be used to suggest combinations of techniques to produce long-pulse plasmas in conditions relevant to CTF.

Projected Milestone Date: September 2008

Milestone Description: F(08-1) Operate NSTX Facility for 10 Experimental Run Weeks.

Projected Milestone Date: September 2008

Milestone Description: D(08-1) Upgrade the edge poloidal rotation diagnostic to increase its spatial resolution and coverage.

The interim poloidal charge-exchange spectroscopy (P-CHERS) diagnostic will be upgraded by purchasing and commissioning additional spectrometers and detectors to increase its spatial resolution and coverage toward that available with the 51-channel toroidal CHERS system now operating on NSTX. As shown by measurements made in TFTR, for example, extremely localized structures can develop in the poloidal rotation profile prior to the development of internal transport barriers. Such structures may be related to the suppression of ion-scale turbulence by sheared plasma flow. The measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

NSTX

OVERVIEW-PURPOSE

The purpose of NSTX is to address issues important for Fusion Energy Sciences (FES) using the special ST properties. For this purpose, the NSTX National Team aim to:

1) Determine the physics principles of ST, utilizing its low aspect-ratio (A ~ 1.5) and very high ratios of plasma pressure to magnetic pressure (beta as high as unity).

2) Support preparation of burning plasma research in ITER through the International Tokamak Physics Activities (ITPA) and benefit from it.

3) Complement and extend A ~ 3 and lower beta experiments in addressing key scientific issues of toroidal fusion plasmas.

4) Complement ITER by establishing attractive configurations for a Component Test Facility (CTF) and Demonstration Power Plant.

APPROACH

The NSTX research approach is to utilize the intrinsic scientific capabilities of NSTX to support the ST-specific as well as the needs to optimize the burning plasmas performance in ITER. The NSTX research plan will therefore be organized in such a way as to maximize relevance to these goals; whereas 100% is relevant to understanding and optimizing the performance of ST plasmas, about three-quarters of it is relevant to advancing the scientific basis for the tokamak plasmas, and about one-third of it is to address the 2006 goals of the ITPA joint experiments in support of ITER optimization.

With this approach, the NSTX Program aims to deliver, as its objectives for FY2006-2008, the following anticipated benefits to the national FES Program:

1) Address and resolve key issues defined by the 2005 FESAC Priorities Panel, which can be organized into topical areas of:

- Transport and Turbulence: to clarify the physical processes that govern heat, particle and momentum confinement.
- Macroscopic Stability: to determine the role of magnetic structure on plasma pressure and bootstrap current.
- Wave-Particle Interaction: to elucidate the mechanisms of electromagnetic waves and modes in sustaining and controlling hot plasmas.
- Boundary Physics: to understand the interface between fusion plasmas and normal-temperature surroundings.
- As uniquely important to ST, Start-up Ramp-up and Sustainment: to determine the physical processes of magnetic flux generation and reconnection; and Physics

Integration: to understand the synergistic effects of external control and self-organization in sustaining the low A and high beta ST plasmas.

2) The NSTX research will benefit the ITER burning plasma program through the above contributions. In particular, NSTX plans to contribute to 22 of the 2006 ITPA goals in all key topical areas aimed at resolving the remaining key physics issues of ITER:

- Confinement Data Base and Modeling
- Transport Physics
- Pedestal and Edge
- MHD, Disruptions and Control
- Steady-State Operations
- Diagnostics

3) The NSTX research will establish key elements of the physics basis for CTF and define the additional elements that will require data from a Performance Extension ST experiment (~ 3 MA in current and ~ 1 T in magnetic field). This will contribute to a unique and important capability of the U.S. FES Program to lead the world in an upcoming CTF program to test and develop fusion nuclear components needed for practical fusion power.

NSTX Contributions to FESAC Priorities

The FESAC Priority Panel identified important ten year goals for the US fusion science community. In the following concluding table, we summarize the unique and important contributions NSTX will make to the Priority Panel ten year goals. As shown below, NSTX will make major unique contributions to the ten year goals.

FESAC Priorities Report 10-Year Goals	*Unique and World-Leading NSTX Contributions*
Macroscopic Plasma Physics	
1. Understand the coupled dependencies of plasma shape, edge topology, and size on confinement in a range of plasma confinement configurations.	1.The most powerful and most fully diagnosed low aspect ratio magnetic configuration in the world, with the most flexible shaping capability.
2. Identify the mechanisms whereby internal magnetic structure controls plasma confinement.	2.Unique capability to measure shear and shear reversal at low A, unique magnetic well, trapped particle fraction.
3. Identify the effects and consequences on confinement of large self-generated plasma current.	3.Highest bootstrap fraction at low A, T_i and V_ϕ measurements at ρ_i scale for study of NTM's.
4. Learn how to control the long scale-length instabilities that limit plasma pressure.	4.Close-fitting conducting shell, ITER-like EF/RWM coils, rotation up to 0.5 V_A, decoupling of V_s and V_A, high resolution T_i and V_ϕ measurements for understanding dissipation physics, outstanding magnetic diagnostics.
5. Understand and control intermediate to short wavelength modes responsible for limiting the plasma pressure, particularly at the edge, and extrapolate their effects to the burning plasma regime.	5.Unique access to high edge shear regimes with Gas Puff Imaging, very fast visible-light camera, edge multi-color Ultra-Soft X-rays, and MSE measurements. Low B allows scaling studies and slows dynamics for imaging.
6. Understand the equilibrium pressure limits in a range of magnetic configurations, including the effects of islands, stochastic magnetic fields, and helical states.	6.Highest β operation of any major toroidal facility, high resolution V_ϕ measurements very sensitive to island structures, tangential imaging x-ray camera, USX system, detailed magnetic diagnostics for internal structures.
7. Understand and demonstrate the use of self-generated currents and mass flows to achieve steady-state high-pressure confined plasmas and improve fusion energy performance.	7.Combination of high power, strong rotation, conducting shell, RWM stabilization, EBW current profile control, strong shaping form the basis for NSTX plan to access 100% non-inductive operation near with-wall limit.
8. Understand how external control can lead to improved stability and confinement in sustained plasmas in a range of magnetic configurations.	8.Most sophisticated external shaping, current profile control, RWM control suite available to test stability and confinement at low A.
9. Understand the pressure limits and confinement properties in configurations where magnetic turbulence controls the distribution of the equilibrium magnetic field and for similar configurations with reduced turbulence. Assess their prospects for study in more collisionless plasma regimes for possible extrapolation to practical sustained burning plasmas.	9.Only major toroidal facility world-wide with in-out insulation to allow Coaxial Helicity Injected plasmas. Only major facility world-wide able to stabilize CHI plasmas with toroidal field. High available heating power will allow access to collisionless regime.

FESAC Priorities Report 10-Year Goals	*Unique and World-Leading NSTX Contributions*
Multi-Scale Transport Physics 1. Develop predictive capability for ion thermal transport using simulations validated by comparison with fluctuation measurements. 2. Identify the dominant particle transport mechanisms, including the conditions under which pinch/convective processes compete with diffusive processes. 3. Identify the dominant mechanisms for momentum transport and their relationship to thermal transport. 4. Understand generation of flow shear, regulation of turbulence, and self-consistent profile dynamics and local steepening, and to identify conditions and thresholds for edge and core barrier formation. 5. Identify the dominant electron thermal transport mechanisms, including the role of electromagnetic fluctuations, short-scale versus long-scale turbulence, and spectral anisotropy. 6. Identify the dominant driving and damping mechanisms for large-scale and zonal flows, including turbulent stresses and cascades. 7. Identify the dominant mechanisms by which turbulence generates and sustains large-scale magnetic fields in high-temperature plasma. 8. Identify the mechanisms and structure of magnetic reconnection, including the role of turbulent and laminar processes, energy flow, and the production of energetic particles. 9. Identify the conditions for onset of island growth and the factors controlling saturation and coupling with transport.	1. Correlation reflectometer to be upgraded to unique high-resolution 2-D microwave imaging. Access to regimes with neoclassical as well as strongly anomalous ion thermal transport. 2. Unique USX imaging array allows tracking of impurity transport fully across plasma column. Charge exchange spectroscopy allows highly resolved carbon profile. 3. Unique capability to study rotation at speeds approaching Alfvénic, unique capability to study interaction with magnetic perturbations on the ρ_i scale. 4. Unique capability for ρ_i scale measurements of flow shear and T_i profiles, leading capability for high radial resolution measurements of both ion and electron turbulence. 5. Unique capability for high resolution measurements of electron and ion turbulence, including spectral anisotropy, over widest range in β world-wide, allowing access to regimes predicted to have strong e-m effects. 6. Leading capability to drive strong large-scale flows, highly time and space resolved turbulence diagnostics to resolve zonal flow shear effects, as well as interplay between ion and electron turbulence. 7. Unique capability for Coaxial Helicity Injection in a major facility with advanced diagnostics for magnetic structures and thermal profiles, as well as strong heating. 8. Unique combination of MSE and tangential neutral particle analysis as well as fast tangential x-ray imaging to study both tearing mode and disruption physics. 9. Leading capabilities to determine the role of poloidal mode coupling on tearing mode seeding and saturation, Glasser effect on growth, and island effects on transport.

FESAC Priorities Report 10-Year Goals	Unique and World-Leading NSTX Contributions
Plasma Boundary Interfaces 1. Predict the expected magnetohydrodynamic stability and plasma parameters for the ITER H-mode edge pedestal with high confidence. This is a time-sensitive issue relevant to the success of ITER 2. Identify the underlying driving mechanisms for mass flow and cross-field transport in the scrape-off-layer plasma, in H-mode attached and detached plasmas. 3. Resolve the key boundary-physics processes governing selection of plasma-facing components for ITER. This is a time-sensitive issue relevant to the success of ITER. 4. Complete the evaluation of candidate plasma-facing materials and technologies for high-power, long-pulse fusion experiments. This is a time-sensitive issue relevant to the success of ITER.	1. Leading capabilities to test stability physics of plasma edge due to wide range of accessible shear, lower B for physics scaling and for raising scale size and slowing dynamics, allowing precise diagnostic measurements. 2. Leading access, space, and time resolution for high-quality gas-puff-imaging measurements, allowing new insights into the physics of SOL transport, L *vs.* H mode edges, and physics of large and small ELMs. 3. Leading divertor heat flux of over 10 MW/m^2 for ITER-like studies of attachment/detachment, erosion physics. Unique real-time dust and surface deposition measurements. 4. Unique plan for staged development of lithium PFCs; pellet injection and evaporative coating leading to liquid lithium divertor target, based on LTX high heat-flux results showing very effective heat spreading.
Waves and Energetic Particles 1. Develop the capability to design high-power electromagnetic wave launching systems that couple efficiently and according to predictions for a wide range of edge conditions. 2. Produce, diagnose in detail, and model with nonlinear, closed-loop simulations the macroscopic plasma responses produced by wave-particle interactions, including localized current generation, plasma flows, and heating, in both axisymmetric and non-axisymmetric configurations. *3.* Develop long-pulse radio-frequency wave scenarios for optimizing plasma confinement and stability and to benchmark against models that integrate wave coupling, propagation, and absorption physics with transport codes (including microturbulence and barrier dynamics) and with	1. Unique capabilities in both High Harmonic Fast Wave heating, with 12-strap antenna, and Electron Bernstein Wave Ohkawa current drive. Unique diagnostic results on parametric decay of HHFW waves and edge ion heating and rotation. EBW OKCD allows highly efficient current drive near the plasma edge, as needed for the most advanced scenarios. 2. Unique capability to diagnose current drive in overdense plasmas. Strong capability to measure fast ion tails with toroidally and poloidally scanning Neutral Particle Analysis. Microwave imaging for HHFW propagation. 3. Most complete ST capabilities for long-pulse, non-inductive, very high beta and bootstrap operation, based on transport code simulations of wave dynamics and coupling with free-boundary plasma evolution. 4. Widest range of V_{fast}/V_A and β_{fast}/β_{tot}, overlapping and extending beyond ITER, allowed discovery of new CAE, GAE and

FESAC Priorities Report 10-Year Goals	*Unique and World-Leading NSTX Contributions*
magnetohydrodynamic stability models. 4. Improve analysis and models to match the experimental measurements and scale the understanding to predict the dynamics of energetic particle-excited modes in advanced regimes of operation with high pressure, inverted magnetic shear, and strong flow. 5. Identify the character of Alfvén turbulence and the evolution of the energetic particle distribution in a nonlinear system, which can be used to predict alpha-particle transport in a burning tokamak experiment; and to evaluate and extrapolate energetic particle behavior in present-day confinement systems to reactor parameters.	bounce-fishbones. High β_{fast}/β_{tot} and high β_{fast} allow widest range of instabilities in high β regimes with high flow, normal and reversed shear. 5. Unique capability for MSE in plasmas with high V_{fast}/V_A and β_{fast}/β_{tot} allows quantitative evaluation of theoretical stability predictions and direct measurement of impact on fast-ion current drive. Scanning Neutral Particle Analysis provides radial and energy resolution of resonant interactions. Tangential interferometer array, imaging reflectometer, and fast tangential and radial x-ray imaging provide analysis of nonlinear mode structure.
Fusion Engineering Science 1. Deliver to ITER the blanket test modules required to understand the behavior of materials and blankets in the integrated fusion environment. 2. Determine the "phase space" of plasma, nuclear, material, and technological conditions in which tritium self-sufficiency and power extraction can be attained. 3. Develop the knowledge base to determine performance limits and identify innovative solutions for the plasma chamber system and materials. 4. Develop the plasma technologies required to support U.S. contributions to ITER. 5. Develop the plasma technologies to support the research program.	1. and 2. While NSTX will not develop blanket test modules, it will provide the U.S. a unique path to lead in the development of fusion nuclear technology for Demo, by developing the physics basis for a compact, cost- and tritium-efficient Component Test Facility. 3. Leading access to divertor heat flux at ITER levels and beyond, allowing the test of innovative plasma chamber systems and materials. 4. Unique plan to test the implications of recent favorable LTX results on the effectiveness of thermal convection in liquid lithium, using a liquid lithium divertor target. Unique real-time dust and surface deposition measurements. 5. X-ray Crystal Spectroscopy diagnostic specifically developed for ITER. Unique test-bed for the current-drive and heating technologies required in the overdense plasmas anticipated in high-beta fusion systems.

TECHNICAL PROGRESS (FY 2005 – FY 2006)

Facility and Diagnostic Accomplishments in 2005

The 2005 fiscal year began during the outage period following the experimental campaign in 2004. We were able to make use of this outage to install or modify selected facility systems and diagnostics which then enhanced the research program in 2005. After operation of NSTX resumed in April, 18 weeks of experimental operation were completed in FY 2005, thereby meeting the NSTX Facility DOE Joule Milestone F(05-1): "Operate the NSTX Facility for 17 Experimental Run Weeks".

The toroidal field coil, which had undergone inspection and refurbishment of its bolted joints at the conclusion of the 2004 experiments, operated for about 2500 pulses. At the end of the operating period, the coil was run up to 95% of its design rating of 0.6 T, although the majority of experiments were conducted with fields up to 0.45 T. Throughout the 2005 operation, the measured resistances of the 72 joints remained well within specifications and did not show evidence of significant deterioration.

Towards the end of the experimental campaign in September 2005, a small leak developed in a cooling water connector at the lower end of the OH solenoid. Since this coil is highly stressed both mechanically and electrically, it was decided to devote the last week, after having met the milestone for run weeks in 2005, to CHI experiments, which do not require the solenoid.

The solenoid and the TF central conductor bundle were withdrawn from the center stack at the conclusion of the run for inspection and repair of the water fitting. The surface insulation around the water connector was excavated and the leak was found to be caused by failure of a soldered joint between the connector block and the pipe connecting it to the cooling channel in the main copper conductor. An alternate method for attaching the connector block to the pipe was developed and implemented. The repair was hydrostatically pressure tested at its maximum operating temperature before the insulation was reapplied. During the removal of the TF center bundle, it was found that three of the flexible current-carrying links and the L-connectors which join them to the TF "flags" at the top of the TF center bundle had become distorted during the run. Analysis showed that the connectors were marginal in their ability to withstand their electromagnetic loads so a stiffening gusset was added to all these connectors to prevent a recurrence of the problem.

Progress in Facility Capabilities

New PF1A divertor coils for producing plasmas with high elongation and triangularity

During the outage prior to the 2005 experimental campaign, the two innermost poloidal field coils at the upper and lower ends of the central solenoid, known as the PF1A coils, were replaced by an axially shorter pair further from the midplane to increase the plasma shaping, in particular the capability to produce simultaneously high elongation, κ, and triangularity, δ, of the plasma cross-section. The changes in the coils and representative plasma shapes are shown in Fig. 2.1. These coils were used throughout the 2005 experiments and were instrumental in achieving several of the advances in plasma performance described in the subsequent section on Research Accomplishments. This fulfilled the NSTX Facility Milestone F(05-3): "Commission a new pair of PF1A poloidal-field coils to produce high-triangularity, high-elongation plasma equilibria".

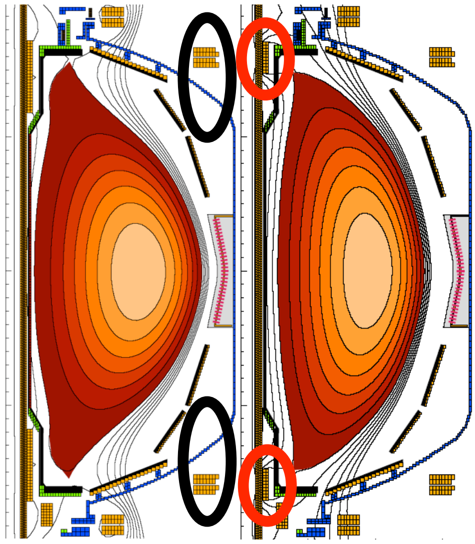


Fig. 2.1 Cross-section through NSTX showing the PF1A coils (circled) and typical plasma configurations available in 2004 (left) and 2005 (right).

Error Field / Resistive Wall Mode (EF/RWM) coils powered by Switching Power Amplifiers (SPAs) for plasma stability control

During FY 2004, preliminary experiments had been conducted in NSTX with the first of three pairs of external magnetic field coils which are designed to apply non-axisymmetric radial field perturbations to the plasma in order to correct intrinsic error fields in the device and to control the development of the "resistive-wall mode", a form of MHD instability which can develop when a high-pressure plasma is being stabilized by the presence of surrounding conducting surfaces made of material with finite electrical resistivity. Since NSTX has such conducting plates inside its vacuum vessel and routinely operates at plasma presssures where their stabilizing effect is important, control of resistive wall modes is an area of active research interest. In the experiments in 2004, the first pair of these error-field correction and resistive-wall mode (EF/RWM) coils had been powered by one thyristor-rectifier power supply of the type used for the other poloidal field coils.

Installation of the remaining two pairs of EF/RWM coils began at the end of the 2004 experiments and was completed early in FY 2005. The air-cooled "picture-frame" coils, each with two turns, are mounted on the midplane just outside the vacuum vessel. To avoid existing equipment, each coil has a slightly different shape, but they are arranged as closely as possible to a uniform toroidal distribution and to have about the same area of 1.6 m^2.

To power the coils, three fast switching power amplifiers (SPAs) were purchased and commissioned in FY 2005. These SPAs convert the pulsed DC power from the single thyristor-rectifier supply used in the 2004 experiments into three programmable AC currents; each SPA can switch up to ±3.3 kA, ±1 kV, at frequencies up to several kiloHertz. With these SPA supplies, the EF/RWM coils can produce radial field perturbations in the plasma region with toroidal mode components n = 1 to 3 and amplitudes up to several milliTesla. Control signals for the SPAs are produced by the Plasma Control System (PCS), which can be programmed to produce static or rotating fields or, in the future, to provide closed-loop feedback on the measured behavior of the plasma itself.

The EF/RWM coils and the SPAs were commissioned in June 2005 and worked well through the remainder of the 2005 campaign, producing data for several experiments as will be described in the Research Accomplishments section. This fulfilled the NSTX Facility Milestone F(05-2): "Implement a Resonant Field Correction system".

New techniques for conditioning plasma facing surfaces

In NSTX, boronization has been carried out by running a DC glow discharge in mixture of 90% helium and 10% deuterated-trimethyl boron (TMB). The same discharge anodes, which are insulated from but close to the vessel wall at the outer midplane, have been used both for boronization and for the helium glow-discharge cleaning (HeGDC) applied routinely between shots. Measurements of the material deposited on coupons mounted on the vacuum vessel walls and also the quartz microbalance probe had shown that the boronization film was deposited preferentially near these anodes. To promote greater uniformity of the boronization and also of the conditioning effect of the HeGDC, a glow-discharge anode mounted on a movable probe carriage (the "movable glow probe" - MGP) was installed during 2005. The MGP is capable of reaching roughly to the center of the poloidal cross-section defined by the plasma facing surfaces and can be inserted and retracted between shots. It was commissioned in July 2005. With this anode, it was possible to run a stable glow discharge at significantly lower helium pressure than with the wall-mounted anodes, which is desirable for increasing the energy of the ions impinging on the walls. However, in these lower pressure conditions, significant heating of the anode also occurred, so it was necessary to add forced-air cooling of the anode assembly to allow for extended operation. The probe was then used routinely for HeGDC between shots. However, no boronization was performed during the remainder of the run after the probe became available to test the uniformity of the deposition.

The NSTX "lithium pellet injector" (LPI), introduced in 2004, was used in 2005 to investigate the effect on the edge recycling of a thin coating of lithium applied to the plasma contact area of the carbon tiles on the plasma facing surfaces. The surface layers of the plasma contact area were first depleted of deuterium by a series (~10) of low-density, ohmically heated, helium discharges. One or two lithium pellets with masses 1.7 – 5 mg were then injected into each of a series (10 – 20) of similar helium discharges, to introduce a total of 24 – 30 mg of lithium. A dramatic reduction in the density was then observed in a subsequent NBI-heated plasma fueled with deuterium gas, indicating that the recycling of deuterium from the plasma contact area had been reduced. The technique was effective in both plasmas limited on the central column and lower single-null divertor plasmas. This success has motivated the development of a lithium evaporator for depositing thicker films of lithium on the plasma contact areas. This evaporator will be available in 2006.

Real-time data acquisition and control

The Plasma Control System (PCS) for NSTX is fully digital and uses high-speed fiber-optic communication to bring data from several physically separated diagnostic systems to its central computer where processing is performed to produce control signals for the power supplies and the gas injectors. During 2005, real-time data acquisition was added for the diagnostic signals from the SPAs, several new poloidal flux loops and the 48 magnetic field sensors mounted in and adjacent to the passive stabilizer plates inside the vacuum vessel to detect locked-modes and resistive-wall modes (RWMs). This brought the total number of channels acquired to 352 from 192 previously. Although this increased the total signal propagation time through the PCS from about 0.7 ms to 1.2 ms, it did not degrade the plasma control performance significantly.

For analyzing the magnetic data in real time, the rtEFIT model was modified to match that used in the offline EFIT analysis, taking into account the additional diagnostic data, the new PF1A coils and changes to the vessel structure made since the original model was developed. With the new data and model, the rtEFIT calculation of the plasma boundary matched the offline calculation within a few millimeters. The inclusion of data from flux loops near the divertor plates improved the accuracy of determining the location of the discharge X-points. Use of rtEFIT for feedback control of the plasma equilibrium now allows precise control of the magnetic balance between the upper and lower X-points. This capability was used in experiments to determine the H-mode power threshold in NSTX and to investigate the nature and structure of ELMs since small changes in the plasma boundary reproducibly lead to large differences in ELM behavior.

The PCS was used to control the SPAs to produce static and rotating n = 1 and n = 3 radial field perturbations with the EF/RWM coils. This capability was used in experiments to investigate the interaction of MHD modes with applied resonant perturbations. Real-time control software was also developed for the PCS which produced coil currents from the SPAs proportional to a linear combination of PF currents in order to counteract the intrinsic field perturbations caused by small residual coil displacements and misalignments.

Improved capabilities for CHI experiments

Coaxial Helicity Injection (CHI) involves injecting a poloidal plasma current between the inner and outer divertor lower plates to create a toroidal plasma current. In 2004, promising results for CHI in NSTX had been obtained using a capacitor bank (10 – 50 mF, 2 kV rated) to inject the poloidal current. The capacitor bank was upgraded in 2005 with a fast ignitron "crowbar" switch to reduce the injected poloidal current rapidly once the discharge had filled the vacuum chamber and a substantial toroidal current was flowing. In addition, changes were made to inject both the gas needed to form the plasma and the microwave power which ionizes that gas into the chamber immediately below the gap between the injector electrodes. As described in the Research Accomplishments section, these changes produced for the first time in NSTX, a clear demonstration of toroidal plasma current which persisted on closed magnetic surfaces beyond the end of the injector current pulse.

Progress in Diagnostic Capabilities

Multi-Point Thomson Scattering diagnostic upgraded to 30 spatial points

The NSTX multi-point Thomson scattering (MPTS) diagnostic routinely provides spatial profiles of the electron temperature and density at a rate of 60 Hz throughout NSTX discharges which are the foundation for the analysis transport. However, the need had been identified for additional channels to resolve fine-scale spatial structures which had been observed in the NSTX data, including, for example, the "ears" which develop in the electron density profile immediately following the H-mode transition. During FY 2005 an additional

10 polychromators and detectors were procured and installed. Optical fibers bringing the scattered light from many additional spatial locations to the polychromator room were already available. The locations selected for the new channels, which were chosen following discussions within the NSTX research team, are concentrated in the edge pedestal region of typical H-mode plasmas in NSTX. The new analyzers were commissioned and gathered data during most of the 2005 experiments. This fulfilled the NSTX Diagnostic Milestone D(05-1): "Install an additional 10 channels for the multi-pulse Thomson scattering system". At the conclusion of the experiments, an extensive calibration of the complete MPTS system was launched to enable analysis of the data from the 10 new channels and the refinement of the analysis for the previous 20 channels.

Diagnostic system to measure short-wavelength plasma turbulence

Because of its low magnetic field and the exceptional tangential access afforded by its low aspect ratio, NSTX offers a unique opportunity to measure through scattering of microwave radiation, turbulent density fluctuations with a scale length comparable to the typical gyro-radius of thermal electrons. During FY 2005, such a system was installed on NSTX. A backward-wave oscillator generates microwaves with a frequency of 280 GHz, free-space wavelength ~1 mm, which are transmitted by low-loss corrugated waveguides to a launcher on the vessel midplane at Bay H which can launch through the plasma collimated probe beams with a tangency radius of either 1.07 m or 1.41 m. Radiation scattered from these beams is collected and focused by a large mirror at the Bay K port through 5 vacuum windows into waveguides which convey the radiation to sensitive superheterodyne receivers. The vacuum ports at Bays H and K had been specially modified to accommodate this system during the preceding outage. The system is designed to measure the amplitude of density fluctuations, in a localized region between the magnetic axis and the outboard midplane edge in typical plasmas, with a radial component of their wavevector in the range $k_r = 4 – 22\ cm^{-1}$, corresponding to $k_r\rho_e = 0.1 – 0.8$ in typical conditions, where ρ_e is the thermal electron gyro-radius. The estimated detection limit is a fluctuation level $\delta n_e/n_e \sim 3 \times 10^{-5}$. The first probe beams were launched and the first signals detected in September 2005. This fulfilled the NSTX Diagnostic Milestone D(05-2): "Install a diagnostic system to measure short-wavelength plasma turbulence by scattering from the plasma density fluctuations".

Additional channels for the FIReTIP diagnostic

During the 2005 experiments, the far infra-red tangential interferometer and polarimeter (FIReTIP), supplied to NSTX by collaborators from the University of California at Davis, made measurements on four channels with tangency radii R_{tan} = 0.32m (Ch 1), 0.57m (Ch 2), 0.85m (Ch 4), and 1.50m (Ch 7). In addition to making routine measurements of the line integrated density, the instrument was used to measure the density perturbations associated with high-frequency MHD modes, such as Alfvén eigenmodes, energetic particle modes and fishbone oscillations, to measure the edge density perturbation with high time resolution at the H-mode transition, to follow the toroidal propagation of the filamentary structures observed by the fast visible cameras accompanying the Type 5 ELMs which occur in some H-mode plasmas in NSTX.

Routine operation of the Motional Stark Effect diagnostic

The Motional Stark Effect (MSE) diagnostic operated routinely throughout the 2005 experiments with eight spatial channels fully instrumented. This diagnostic measures the radial profiles of the pitch of the local magnetic field based on polarimetry of the collisionally-induced fluorescence (CIF) from the atoms injected by the heating neutral beams. The MSE-CIF system was calibrated by observing the fluorescence when the neutral beams were injected into the vacuum vessel filled with deuterium gas to a pressure of about 0.02 mTorr (~3 mPa) while specific magnetic field patterns were applied by the NSTX toroidal and poloidal field coils. The MSE data were used to constrain the analysis of the NSTX equilibrium with EFIT and other analysis codes.

Between-shots analysis for the Charge-Exchange Recombination Spectroscopy system

The analysis for the 51-channel charge-exchange recombination spectroscopy (CHERS) diagnostic was automated to produce profiles as functions of time for the temperature, density and toroidal rotation velocity of the carbon impurity ions in plasmas heated by the neutral beams. The analysis, which includes the full corrections for atomic physics effects, was completed and the results were stored and made available in the time interval between shots. These data, together with the MPTS data, provide an unprecedented amount of information to the members of the NSTX team while conducting experiments.

NSTX Research Accomplishments in 2005

During the 18 weeks of operation in FY 2005, 38 separate experiments were performed. The experiments were aimed at the following five Research Milestones, which were achieved:

R(05-1) on Turbulence and Transport: “Characterize the effects of variations in the magnetic shear and gradients in electron temperature on electron transport in low-aspect ratio plasmas”;

R(05-2) on Macroscopic Stability: “Produce and characterize strongly shaped, rotating, low aspect ratio plasmas close to the ‘wall-stabilized’ pressure limits with error field correction”;

R(05-3) on Wave-Particle Interactions: “Assess the effects of supra-Alfvénic fast ion driven instabilities on driven current in the plasma core”;

R(05-4) on Boundary Physics: “Characterize the plasma edge pedestals and scrape-off layer of low-aspect ratio, high confinement, high P/R plasmas”;

R(05-5) on Physics Integration: “Characterize strongly shaped low-aspect ratio plasmas with high fractions of self-driven current and low toroidal induction voltage for durations that allow internal currents to redistribute”.

Improvements in plasma shaping capability and the effects on plasma stability

As a result of the replacement of the PF1A coils with the axially shorter pair further from the midplane, plasmas with $\kappa = 2.7$ and $\delta_{av} = 0.8$ (where δ_{av} is the average of the upper and lower triangularity) were produced at an aspect ratio A = 1.5. Comparisons of the values of κ and δ_{av} achieved in 2004 and 2005 are shown in Fig. 2.2. The highest value of the “shaping factor” $q_{95}I_p/aB_T$ (where q_{95} is the safety factor at the 95% normalized flux surface, I_p the

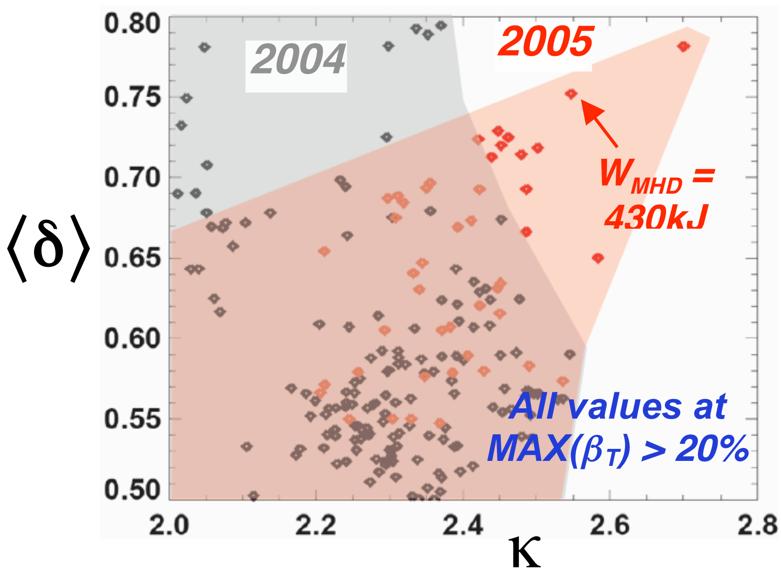


Fig. 2.2 Values of the plasma cross-section elongation κ and the triangularity averaged between the upper and lower X-points at the time of maximum β_T for discharges in which exceeded 20%. Data from 2005 with the modified inner PF coils and from previous years are distinguished.

plasma current, a the mid-plane half-width of the cross-section and B_T the vacuum toroidal magnetic field at the plasma geometric center) reached 37 MA/m·T at a slightly lower aspect ratio A = 1.35 with κ = 2.3, δ = 0.6. Plasma currents I_p up to 1.5 MA have now been achieved. At an applied toroidal field of 0.45 T, the plasma stored energy reached 430 kJ, a record for NSTX, for a NBI heating power of 7.3 MW. At lower field, 0.34 T, the toroidal beta, β_T, reached 35% in 2005 operation, although maximizing β_T was not a major focus of the experiments in 2005. By ramping down the plasma current during NBI, poloidal-beta, β_P, up to 2.1 and Troyon-normalized beta $\beta_N = \beta_T/(I_p/aB_T)$ up to 7.2 %·m·T/MA were produced, which significantly exceed the ideal stability limit calculated without wall stabilization.

One result of operating with higher κ and δ simultaneously was the re-emergence of small, high-frequency ELMs in H-mode plasmas. Previously, operating at κ > 2.2 with lower triangularity produced large ELMs which caused significant drops in the plasma energy. The small ELMs do not perturb the plasma energy significantly but they do slow the rate of density rise. This regime has provided a significant extension in the pulse length achievable at moderate plasma currents, 0.7 – 1.0 MA. Figure 2.3 shows basic waveforms for a lower single-null divertor (LSND) discharge which extended to 1.5 s at 0.7 MA; the current was constant for about 4 current-relaxation times. Calculations have been made with the TRANSP code of the non-inductively driven current in this plasma assuming classical thermalization of the energetic ions introduced by NBI; this assumption provides a good match between the measured and predicted DD fusion neutron rates during MHD-quiescent periods. As seen in Fig. 2.4, the calculation shows that the non-inductive components, including the neoclassical bootstrap current, other ∇p terms and the beam-driven current, provide up to 70% of the total current at peak β. Despite the beneficial effect of the ELMs in slowing the density rise, this discharge reached the nominal density limit predicted by Greenwald scaling. This roughly coincided with a drop in the central rotation of the plasma, the development of a persistent

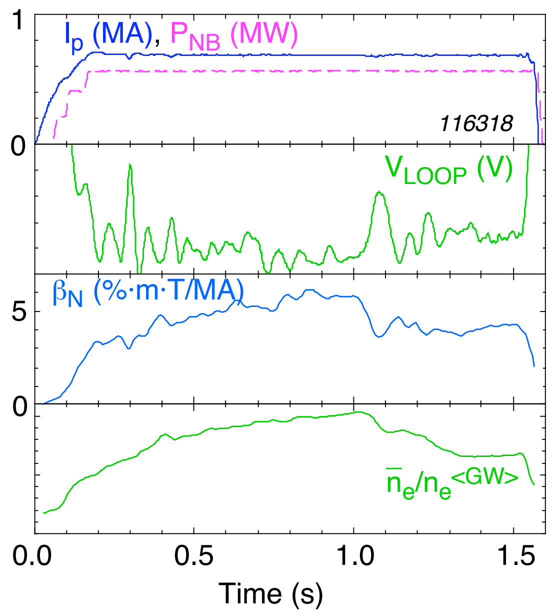


Fig. 2.3 Waveforms of discharge parameters for the longest duration 0.7MA plasma

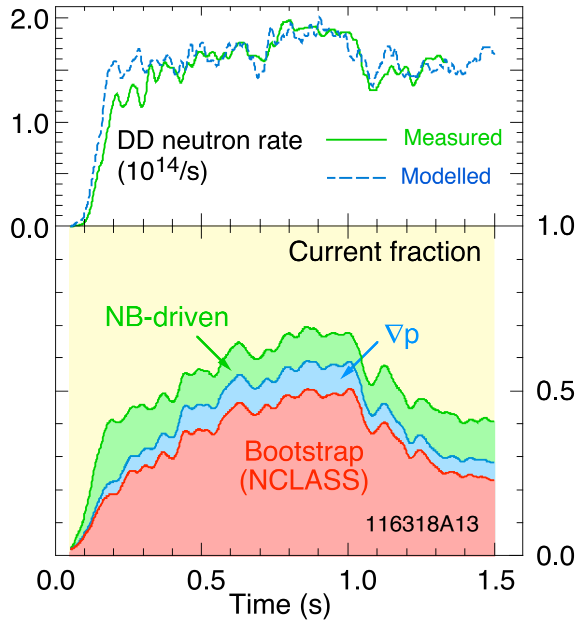


Fig. 2.4 Results of TRANSP analysis for the plasma in Fig. 2.3 showing the time evolution of the components of the total current and the measured and modelled neutron rate

saturated n=1 mode and a drop in β. While this coincidence does not show causality, it does suggest that density control will be an issue for the development of even longer H-mode discharges in NSTX. During the period of MHD activity, it was necessary to introduce an anomalous diffusivity of $5 m^2/s$ for energetic ions in the TRANSP model to match the measured neutron rate. This discharge did not utilize the full flux swing available from the transformer but suffered a final MHD-related collapse at the end of the NBI pulse when the central safety factor q(0) had reached 1. A small extension of the pulse length over previous results was also obtained at a higher plasma current, 1.0 MA, in a double-null divertor discharge with $\kappa = 2.3$, $\delta_{av} = 0.6$.

Generation of persistent toroidal current by Coaxial Helicity Injection

Coaxial Helicity Injection has the potential to initiate toroidal plasma current in the ST by creating a discharge and injecting poloidal current from electrodes coaxial with the major axis in the presence of applied toroidal and appropriate poloidal magnetic fields. Recent experiments in NSTX have aimed to exploit the technique of "transient CHI" originally developed in the HIT-II device at the University of Washington. The NSTX experiments in 2005 benefitted from the capability to inject both the gas and the electron cyclotron resonant (18 GHz) microwave power for initiating the discharge directly into the chamber below the CHI electrodes. This reduced the amount of gas needed to create the discharge, thereby increasing the energy input per plasma particle and thus the possible temperature of the CHI plasma in its high-current phase. When the injected current had reached its peak and the discharge had expanded to fill the region available for plasma inside the vacuum vessel, the new fast "crowbar" switch rapidly reduced the injected current. With these changes, a clear demonstration was obtained of toroidal plasma current which persisted on closed magnetic surfaces beyond the end of the injector current pulse. Fig. 2.5 shows examples of the discharge waveforms for a shot obtained with a 15 mF capacitor bank charged to 1.5 kV in which a peak toroidal plasma current of 120 kA was generated for an injected current of 1.9 kA, representing a current multiplication factor greater than 60. When the injector current had decayed to zero ($t \approx 11$ ms), approximately 6.5 kJ of electrical energy had been dissipated in the plasma circuit. At this time, a toroidal plasma current of 50 kA was still flowing, which subsequently decayed on a timescale of about 7 ms. Images of the visible emission during this phase showed the formation

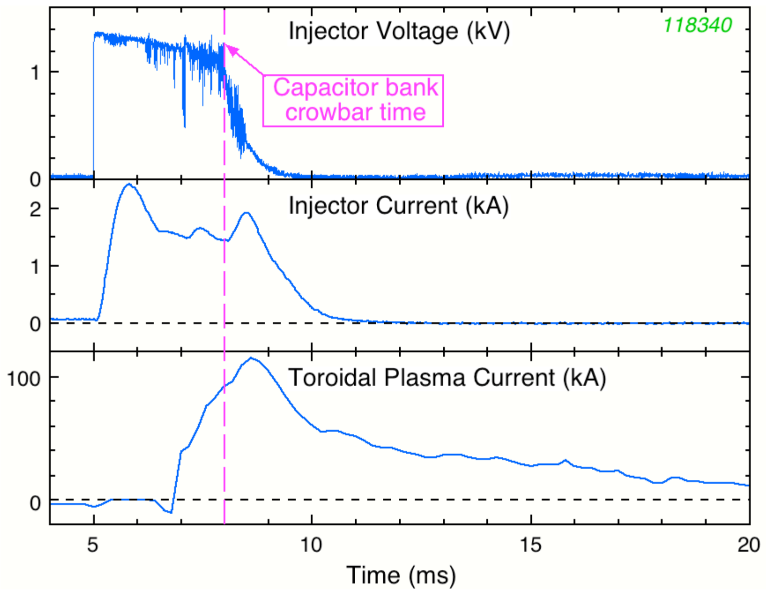


Fig. 2.5 Waveforms for a CHI discharge which produced a toroidal current of about 50kA as the injector current returned to zero. The toroidal current persisted for a further 10ms.

of a plasma ring detached from the injector and close to the center column. Future experiments will attempt to maximize the CHI-initiated current and to couple these discharges both to inductive and non-inductive current drive.

Effects of applied radial field perturbations on plasma stability and rotation

NSTX routinely operates with normalized-beta above the stability limit calculated without the stabilizing effect of the conducting wall, so plasmas are susceptible to the growth of resistive wall modes (RWM) unless sufficient plasma rotation can be maintained. Non-axisymmetric field perturbations, both intrinsic and induced by the modes themselves, can act to slow the rotation induced by the NBI heating in NSTX and thereby contribute to mode growth. To allow extended operation near the ideal-wall limits, three pairs of error-field correction and resistive-wall mode (EF/RWM) coils powered by three switching power amplifiers (SPAs) have been installed and commissioned, as described in the Facility and Diagnostic Accomplishments section.

The effect of DC perturbations generated by the EFC/RWM coils with toroidal mode number n = 1 on the development of locked modes was investigated first; the results are summarized in Fig. 2.6. In a series of otherwise similar ohmically heated, low-density, deuterium, LSND plasmas, the amplitude of the applied perturbation needed to trigger a locked mode as a function of its direction, traced out a circle, suggesting that there is an intrinsic radial error field perturbation corresponding to the vector from the center of the circle to the origin, *i.e.* about 1.3 Gauss in the conditions of these discharges.

The response to stationary perturbations with toroidal mode numbers n = 1 or 3 has also been investigated in initially rapidly rotating plasmas heated by 6 MW of NBI to $\beta_N \approx 5$. It was found that in these conditions, the apparent n = 1 error field was in the opposite direction to that observed in the low-β OH discharges. Analysis indicates that this error field is proportional to the product of the OH solenoid and TF coil currents, suggesting that some motion of the TF coil inner bundle is involved. As seen in Fig. 2.7, when a small n = 1 perturbation was applied in the direction to augment the intrinsic error field, the plasma toroidal rotation (measured by CHERS) collapsed, starting near the edge but then extending across the profile. A locked mode then developed and the discharge terminated earlier than a reference shot with no perturbation. Conversely, when the applied perturbation counteracted the intrinsic error field, the rotation collapse was avoided and the high-β_N phase was extended. When a

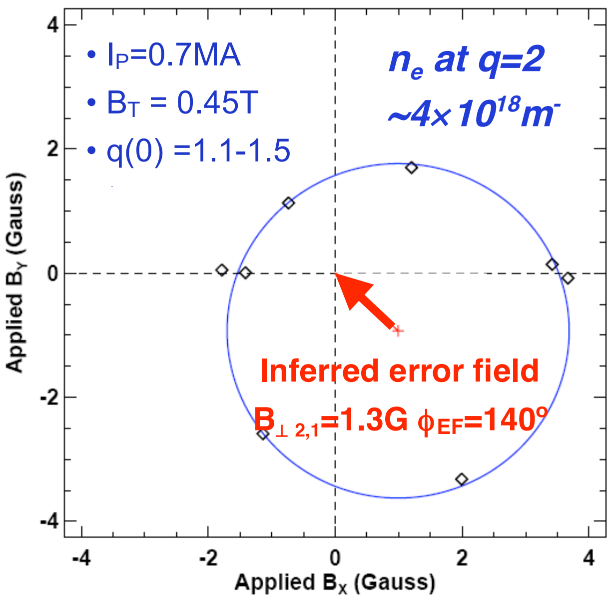


Fig. 2.6 Measurements of the threshold in the applied n=1 error field to generate a locked mode in otherwise similar plasmas as the direction of the applied field was varied.

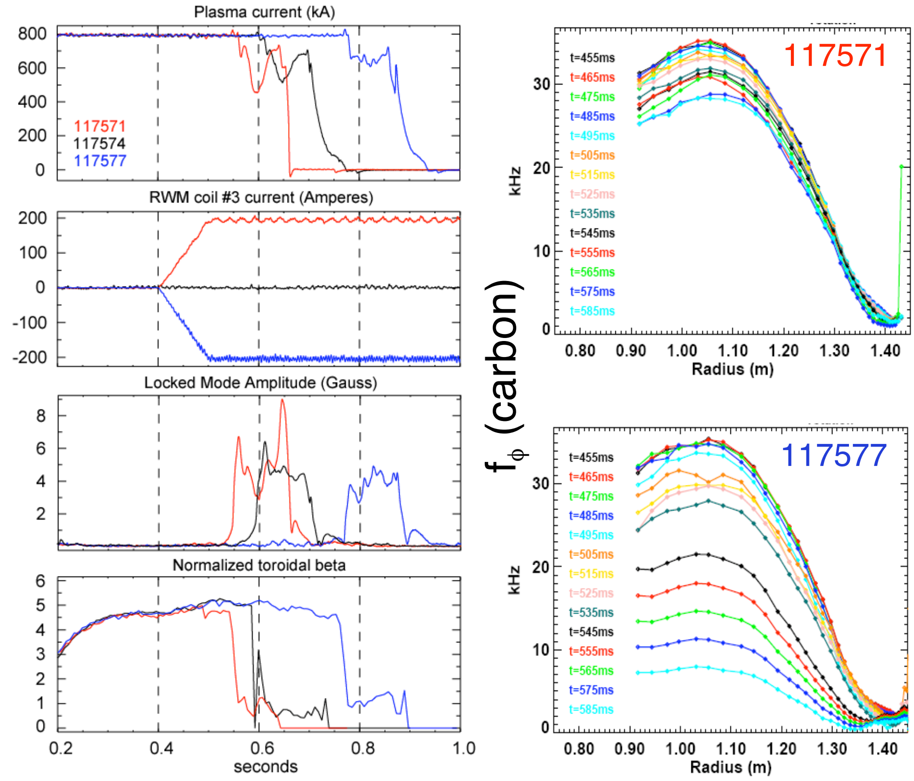


Fig. 2.7 a) Waveforms for three discharges with an n=1 error field applied in the canceling and augmenting directions with respect to the intrinsic field and without any applied correction; b,c) Profiles of the toroidal rotation frequency (=$v_\phi/2\pi R$ measured by CHERS) at various times as the n=1 error field is applied in the augmenting (b) and canceling (c) directions.

50 ms long, n = 3 perturbation pulse was applied to a similar NBI-heated discharge, the plasma rotation at the edge stopped temporarily but, provided that a locked mode not develop while the plasma edge was stationary, resumed after the perturbation was removed. In this case, the perturbation of toroidal rotation in the plasma center, which did not contain q-surfaces resonant with the applied n = 3 perturbation, appeared to be damped by neoclassical toroidal viscosity.

The EFC/RWM coils were also used to probe the response of the system to a rotating n = 1 field perturbation in plasmas where a slowly rotating RWM had already developed. The rotating perturbation was generated by programming the currents in the three pairs of coils with sinusoidal waveforms phase shifted by 120°. Depending on the frequency of rotation of the applied perturbation, the plasma was found to increase or decrease the applied field, a phenomenon referred to as resonant field amplification (RFA). In the conditions of this experiment, the greatest RFA occurred when the applied n = 1 perturbation was rotating at a frequency of about 30 Hz in the same direction as the intrinsic plasma toroidal flow.

Effect of modifying the magnetic shear on electron thermal transport

With NBI heating, the confinement of both the thermal and unthermalized ions is extremely good in NSTX and, in most operating regimes, the dominant thermal loss is through the electron channel. However, in plasmas with a fast initial current ramp which develop a region of reversed magnetic shear in the core, improved electron confinement has also been observed. The creation of reversed shear in these discharges was previously inferred from the behavior of perturbations on the soft x-ray emission profiles, supported by TRANSP modeling of the current diffusion assuming neoclassical plasma resistivity. The MSE measurements of the q-profile made this year have confirmed the inference of reversed shear and also guided the development of a scenario for obtaining reversed magnetic shear reliably.

Figure 2.8a shows the waveforms for two successive discharges with a flattop current of 1MA but slightly different initial current ramps and timing of the NBI, while Fig. 2.8b shows the resultant q-profiles at t = 0.3s as determined by the LRDFIT code using the MSE pitch-angle data as a constraint on the fit. The production of reversed shear through a fast current ramp is very sensitive to the MHD mode activity in the ramp-up phase of the discharge: quite small bursts of activity detected by the Mirnov coils can cause a rapid drop in the central q and result in a profile with positive or near-zero shear in the center. However, once established, the reversed-shear profile can be maintained for up to 0.2s.

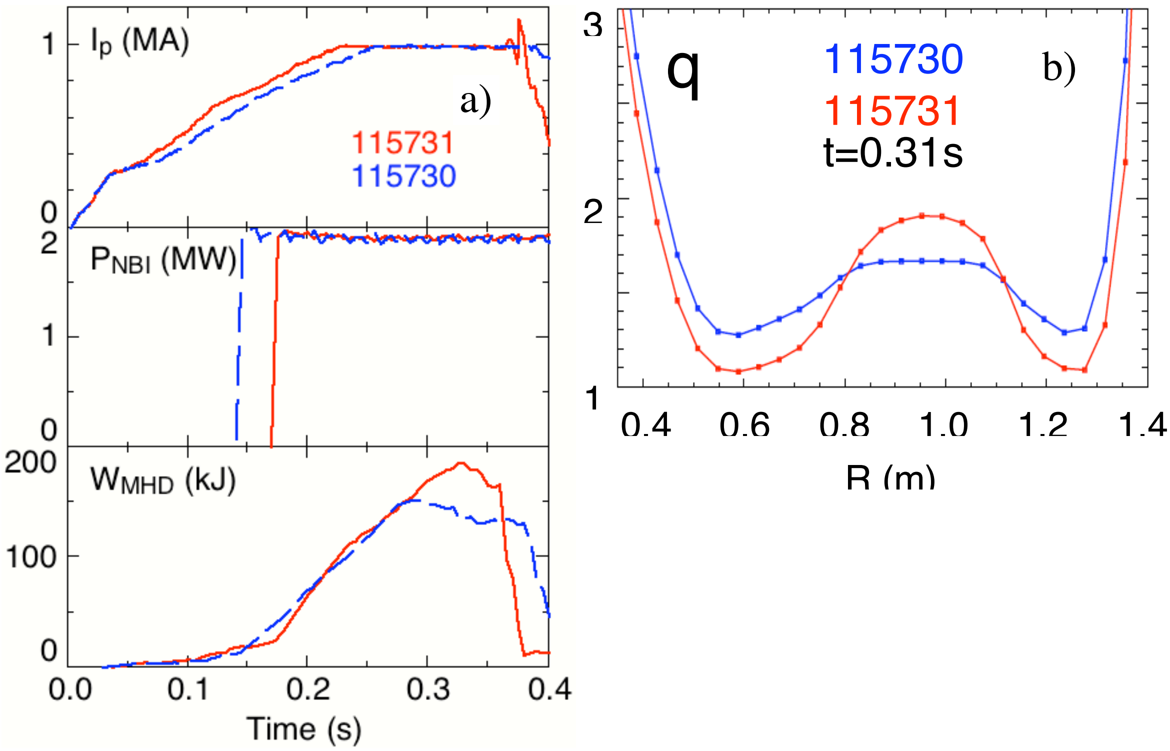


Fig. 2.8 a) Comparison of discharge waveforms for successive plasmas, the first developing a region of weakly, the second of strongly reversed shear; b) q-profiles for the two discharges at t=0.31s from analysis of the MSE data with the LRDFIT code.

A comparison of the profiles of the plasma temperatures and density is shown in Fig. 2.9a. Although the ion temperature is very similar in the two discharges, the electron temperature is significantly higher in the reversed-shear case, suggesting a reduction in electron thermal transport. This has been confirmed by TRANSP analysis, based on classical thermalization of the fast ions from NBI including an anomalous fast-ion diffusivity of 0.5 – 1.5 m^2/s to bring the measured and calculated DD neutron rates into agreement. The electron thermal diffusivities calculated by TRANSP for the two discharges are shown in Fig. 2.9b; a reduction of χ_e by a factor 2 – 4 over the inner 70% of the minor radius is evident.

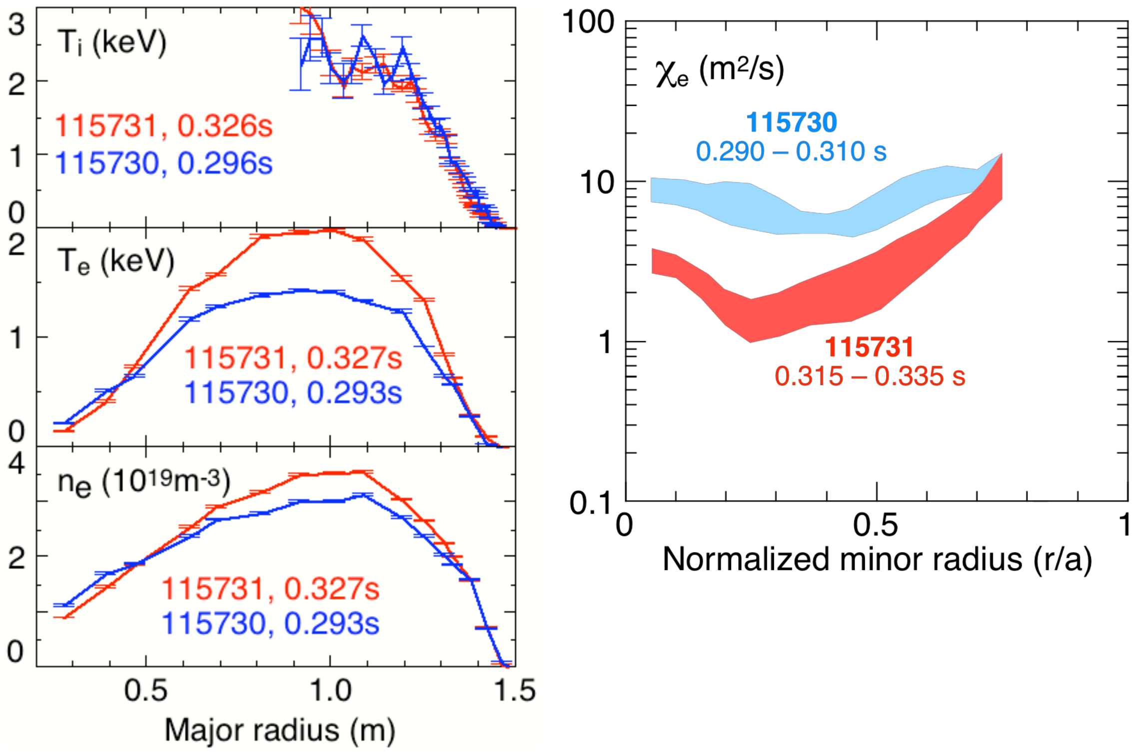


Fig. 2.9 a) Profiles of the plasma temperatures and density for the two discharges in Fig. 2.8 at the times of peak plasma energy; b) Profiles of the electron thermal diffusivity calculated by TRANSP (the bands indicate the variability of the diffusivity over the time interval indicated).

Adding a modest amount of high-harmonic fast-wave RF heating power, ~0.7 MW, launched with balanced $k_{||} \approx \pm 7\ m^{-1}$ produced electron heating in the central region of a reversed-shear plasma established with 2 MW of NBI. The region where the heating was observed roughly coincided with the region of shear reversal inferred from the equilibrium analysis.

Use of lithium coating to control recycling from plasma facing surfaces

The Lithium Pellet Injector, first introduced in 2004, was used to produce changes in the recycling of hydrogenic species from the plasma contact surfaces, which contributes to the secular density rise observed in most NBI-heated plasmas in NSTX. The experiments involved both plasmas limited on the central column and lower single-null divertor plasmas; both these contact areas are covered by carbon tiles. For each configuration, the surface layers of the plasma contact area were first depleted of deuterium by a series (~10) of low-density, ohmically heated, helium discharges. These were followed by a reference deuterium discharge with 2 MW of NBI heating. One or two lithium pellets with masses 1.7 – 5 mg were then injected into each of a series (10 – 20) of helium discharges, to introduce a total of 24 – 30 mg of lithium. Spectroscopic data indicated that the injected lithium was deposited primarily on the surfaces surrounding the plasma contact area. In both the limiter and divertor configurations, the first subsequent deuterium NBI-heated plasma showed a reduction in the volume-average plasma density during the NBI heating by a factor of about 2 compared to the respective reference discharge before the lithium deposition. This reduction in density was less on the next shot and was not evident on the third shot. This is illustrated for the divertor configuration plasmas in Fig. 2.10. The saturation of the apparent wall pumping can be understood if the effect occurs through the formation of lithium deuteride on the surface: the amount of lithium introduced could react with about 6 – 9 mg of deuterium and about 3.5 mg of deuterium was injected on each discharge. The lithium deposition was repeated for the plasmas limited on the central column, without any preceding helium-only sequence, and a similar reduction in density was observed on the first subsequent NB-heated shot. The results for the limiter plasmas are similar to the experience with lithium coating in TFTR and with a liquid-lithium limiter in CDX-U, but these NSTX experiments extend the potential benefits of lithium surface coating for plasma density control to divertor plasmas.

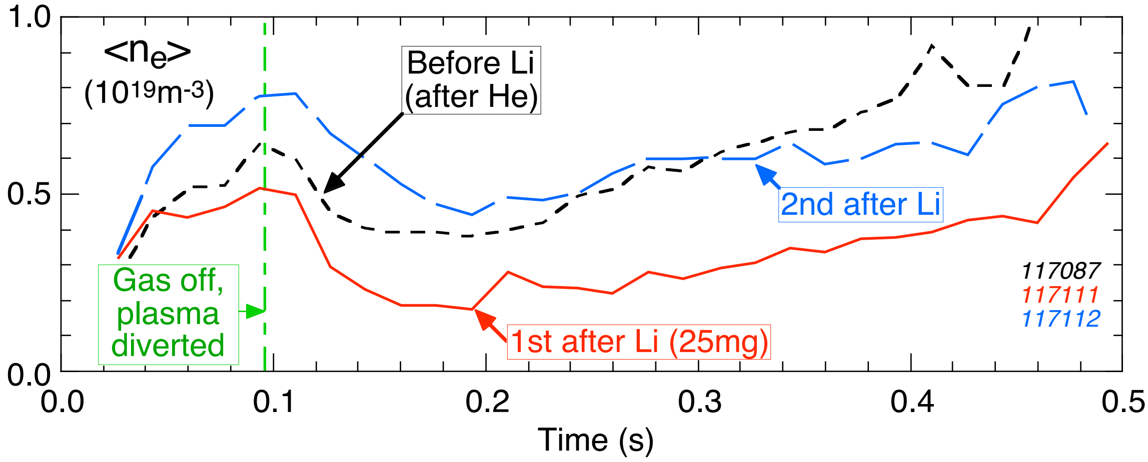


Fig. 2.10 Waveforms of the plasma volume average density (calculated from the Thomson scattering profiles) for a reference NBI-heated deuterium discharge before deposition of lithium on the plasma contact surfaces, and for the first two similar discharges after depositing 25mg of lithium in a series of helium ohmically heated plasmas.

H-mode pedestal, edge-localized modes (ELMs) and boundary physics

The dependence of the power threshold for the H-mode transition on the plasma configuration was investigated in an experiment conducted jointly with the MAST group at the UKAEA Culham Laboratory and the Alcator C-Mod group. Similar to MAST experiments with NBI, it was found that in NSTX the minimum threshold power with either NBI or HHFW heating occurred for a symmetric double-null divertor configuration, as defined by $\delta r_{sep} < \rho_I$, where δr_{sep} is the radial separation of the inner and outer separatrices at the outer midplane, as calculated by EFIT, and ρ_i is the local ion gyro-radius, typically 0.5 cm. The NSTX experiment, however, produced the first example of an H-mode in an ST for the upper single-null divertor configuration with the ion ∇B-drift away from the divertor X-point; the threshold power in this case was between two and four times higher than for the complementary lower single-null configuration. This experiment also demonstrated that the L-H transition power was comparable with NBI and RF heating for the double-null configurations with comparable density in the L-mode phase.

An experiment was performed to measure the profiles in H-mode plasmas just before and after ELMs of different types to assess the MHD stability of the plasma edge. The conditions included Type I ELMs in double-null (DN) divertor plasmas with triangularity $\delta \approx 0.8$, Type I with transition to Type III ELMs in DN with $\delta \approx 0.4$, the NSTX Type V ELMs, again in DN (EFIT parameter $\delta r_{sep} < 0.3$ cm) with $\delta \approx 0.8$, and, finally, in a mixed Type I and Type V ELM regime in lower single-null divertor (LSN) plasmas with $\delta \approx 0.4$. In addition to measuring profiles of the electron density and temperature with MPTS and the ion temperature and flow velocity with CHERS, data were taken with the FIReTIP diagnostic and a new filterscope array, and images of the ELM perturbation were recorded with a high-speed visible camera viewing the lower divertor. Filamentary structures were observed propagating counter to plasma current. The filaments associated with the Type V ELMs were found to be ribbon-like and aligned with the edge magnetic field, with a typical midplane vertical thickness five to ten times larger than the radial thickness of a few centimeters. Larger ELMs involved more filaments than the typical one or two for Type V ELMs. Based on these data, calculations are underway of the stability of the plasma edge to various MHD instabilities.

The Fast Reciprocating Edge Probe (FREP) measured phenomena accompanying ELMs with high time resolution (1 M-sample/s), including the electric fields, velocity and particle flux. These data showed that an ELM comprises several short (~10 μs) bursts in rapid succession (20 – 50 μs separation). The radial velocity of the ELM perturbation can be high, ~500 m/s, near the last closed flux-surface (LCFS), slowing to ~200 m/s in the scrape-off layer. The ELM burst decayed exponentially away from the LCFS. In L-mode plasmas and between ELMs in the H-mode, intermittent plasma objects, commonly referred to as plasma “blobs” were also observed by the FREP. The velocity of these objects was typically ~400 m/s at the LCFS, decaying to ~100 m/s ~6 cm outside the LCFS.

Imaging and tomographic reconstruction techniques have been applied to data from the multi-chord soft x-ray (SXR) detector system to follow the propagation of ELM perturbations from the plasma edge into the core. By using different filters to select the energy range, it was possible to change the region of observation within the plasma. Type I ELMs cause a large electron temperature crash at the edge which then propagates to the center on a timescale of 1 – 2 ms. Although the density at the edge is also reduced by the ELM, there is no significant perturbation of the central density. The central temperature perturbation appears to be the result of rapid thermal transport, as no large MHD modes are observed on the SXR data.

An experiment was also conducted jointly with DIII-D and MAST to assess the effect of aspect ratio and wall proximity on the height, width and gradients in the pedestal of ELMy H-mode plasmas. This required a dedicated experiment to create in NSTX the low and high squareness shapes run in the other devices. Good data were obtained at an edge collisionality parameter $\nu^*_{e,ped} \approx 1$ and normalized ion-gyroradius $\rho^*_{i,ped} \approx 0.01$ which matched well with the data from the other experiments. Assessment of the pedestal widths and edge stability is in progress.

Finally, a dedicated experiment showed the onset of partial detachment of the plasma from the divertor strike plate on the outer leg of the divertor when deuterium gas was puffed into the private flux region of H-mode plasmas in a lower single-null configuration. The divertor detachment was accompanied by volumetric recombination in the outer-leg scrape-off layer resulting in a 75% decrease in the peak heat flux to the outer strike plate.

Wave-particle physics studies

An experiment was performed in 2005 using the sightline scanning capability of the NSTX charge-exchange neutral-particle analyzer (NPA) to make spatially resolved measurements of the energy distribution of the energetic ions introduced by NBI heating. For H-mode discharges, the measured NPA spectra exhibit depletion of the energetic ion population primarily for $E > E_{inj}/3$; this depletion is not seen in the preceding L-mode phase however. The measured depletion of the population is greatest for a sightline tangency radius $R_{tan} \sim 0.50 \pm 0.10$ m, vanishing at larger R_{tan}. Charge-exchange emissivity effects can account for part, but not all, of the observed behavior. Modeling of the fast-ion orbits and classical loss processes with the TRANSP code reproduces some features of the measurements, but the predicted energy dependence of the depletion, which is largest at the injection energy, is different from the measurement, which peaks at intermediate energies.

The energetic ions from NBI heating in NSTX can excite toroidal Alfvén eigenmodes (TAE), which are fundamental modes of the background plasma, and energetic-particle modes (EPM), which are modes in which the energetic particles themselves are involved in the instability. A study was conducted in 2005 of the stability of TAE and EPM as a function of the central q, q(0), and the magnetic shear, $S = (r/q)\partial q/\partial r$. In general, EPM were present with low q(0) and a monotonic q-profile with $S > 0$, whereas TAE were predominant with elevated q(0) and q-profiles with reversed shear, $S < 0$, in the center. The microwave reflectometer showed that in NSTX, rather than a few modes being present, many modes occur simultaneously, similar to the "sea of TAE" activity predicted for ITER where the large population of energetic fusion alpha-particles will replace the NBI ions. The bursts of multiple-mode TAE/EPM activity in NSTX can cause significant transport of the energetic beam-ions, as evidenced by modulations in the signals from the NPA, which show reductions in the neutral flux predominantly well below the beam injection energy, and rapid drops of order 10% in the DD fusion neutron emission, which suggest that the high energy ions are also lost. These phenomena are shown in Fig. 2.11. The broad range of energy interaction is consistent with the loss being caused by bounce-resonances of the particles and waves.

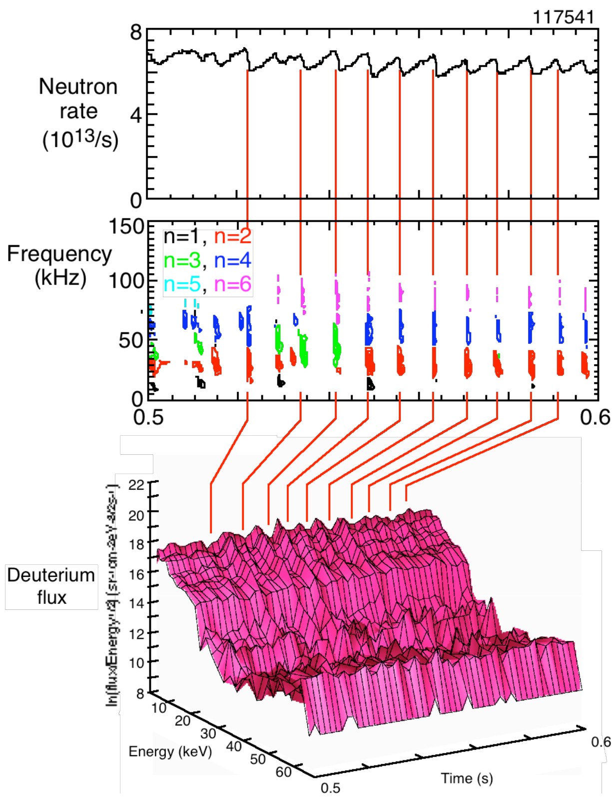


Fig. 2.11 a) DD neutron rate showing sawtooth modulations in phase with high-frequency bursts of MHD activity seen in b) spectrogram of a Mirnov coil signal. These bursts cause repetitive depletion of c) the charge-exchange neutral flux from the plasma in the range between the thermal ion component and the half-energy component of the beam-injected neutrals.

The absorption of and heating by HHFW power was studied as a function of the phasing of the antenna straps, which determines the parallel wavenumber $k_{||}$, or equivalently the phase velocity, of the launched waves. The time response of the plasma energy to the HHFW power pulse was used to determine the effective energy confinement time and the absorbed power. The 180° phasing, which produces a symmetric spectrum with $k_{||} \approx 14\ m^{-1}$, achieved higher absorption, 40 – 60 %, than the ±90° phasings, which produce directed spectra with $k_{||} \approx \pm 7\ m^{-1}$ for counter- and co- current drive, respectively. However, the apparent confinement time with the $k_{||} \approx \pm 14\ m^{-1}$ spectrum was lower, probably because the power was deposited further out in minor radius in this case. Changing the plasma current from 0.3 to 0.6 MA, to vary the pitch of field lines with respect the antenna straps, did not affect the absorption. Measurements with the edge RF probe indicate that the signature of the parametric decay instability was strong for the ±90° phasing but undetectable for 180°.

The first attempt was made to measure with the MSE diagnostic the current driven by HHFW power. Since the MSE diagnostic relies on NBI excitation, it was necessary to operate at low current in helium plasmas to minimize the fast-ion population in the plasma, so that parasitic absorption of the RF power would not prevent the RF current drive. The antenna phasing was varied to produce the symmetric $k_{||} = \pm 14\ m^{-1}$ and the co, counter and balanced $k_{||} = 7\ m^{-1}$ spectra, with up to 3 MW injected power. The target OH plasmas were quiescent but MHD activity occurred during the deuterium NBI which may have affected the driven current; there was little change in stored energy when the RF power was added during NBI or vice versa. Indications of current drive in the direction expected were seen on q-profiles analyzed with the MSE data and averaged over the RF pulse, but the effects were small with $\Delta q(0) < 0.1$.

The first measurements were made in NSTX with an obliquely-viewing 20 – 40 GHz microwave radiometer to detect the 2nd harmonic thermal electron Bernstein waves (EBW) inside the plasma. These plasma waves undergo mode conversion to detectable O-mode radiation at the upper-hybrid resonant (UHR) layer near the edge of H-mode plasmas where there is a steep density gradient. The radiometer measurements indicated an apparent coupling efficiency for the 2nd harmonic EBW of ~20%, compared with ~80% coupling of fundamental EBW previously measured. While this was less than originally expected, subsequent modeling of the wave propagation ad mode-conversion process has indicated that the low electron temperature and the relatively high $Z_{eff} \sim 3$ at the UHR layer in the plasmas used caused significant collisional loss of the power coupled from the EBW.

NSTX Facility Utilization

The NSTX facility utilization is summarized in Table 1. As we move forward, the number of run weeks is mainly determined by the budget. The utilization of the NSTX facility by researchers, post-doctoral researchers in FY 2005 as well as students is also shown in Table 1.

Facility Plasma Operations Availability

	FY 05	**FY 06**	**FY 07**	**FY 08**
Run weeks planned / achieved	17 /18	11	12	10
Hours of operation planned / achieved	680 / 720	440	480	400

Participating Research Personnel

	PPPL	***non-PPPL***
Researchers	***52***	***133*****
Post Doc.	***3***	***10***
Grad. Students	***6***	***10***
Undergrad. Students	***0***	***6***

** There are over 41 overseas collaborating researchers from countries including Canada, Czech Republic, France, Germany, Israel, Italy, Japan, Korea, Russia, and UK, during FY 2005-2006.

Table 1: Facility Utilization

FUTURE ACCOMPLISHMENTS (FY 2006 – FY 2008)

Future Research Accomplishments

The NSTX research accomplishments anticipated for FY2006-FY2008 are organized by research milestones that address the following overarching scientific issues for fusion plasmas:

1. Transport and Turbulence – Physical processes that govern heat, particle and momentum confinement;
2. Macroscopic Stability – Role of magnetic structure on plasma pressure and bootstrap current;
3. Wave-Particle Interaction – Role of electromagnetic waves & modes in sustaining and controlling hot plasmas;
4. Start-up, Ramp-up and Sustainment – Physical processes of magnetic field generation and reconnection;
5. Boundary Physics – Interface between fusion plasmas and normal temperature surroundings;
6. Physics Integration – Synergistic effects of external control and self-organization.

The major goals of the FY2006-FY2008 research on NSTX are to make important and frequently unique contributions toward advancing fusion plasma science, optimizing burning plasmas anticipated in ITER, and making the ST an attractive approach, such as the Component Test Facility (CTF), to reduce the cost, time and risk of fusion energy development. The following research milestones are proposed to achieve these goals.

1. Transport and Turbulence: Physical processes that govern heat, particle and momentum confinement:

Research Milestone R(06-1) Perform highly localized measurement of magnitude of high-k turbulence.

Description: The role of short-wavelength turbulence, i.e. on the scale of a few radii of the electron gyration around the magnetic field, in governing electron thermal transport is an unresolved issue that is important for future fusion-grade plasmas, which are heated by alpha particles such as in ITER. Experiments on NSTX have indicated that electron thermal conduction can vary widely and often can dominate over the ion thermal conduction in high-confinement and high-beta plasmas. Turbulence calculations indicate that short-wavelength microinstabilities may play a strong role in such cases. Advanced microwave scattering techniques have been implemented on NSTX to enable highly localized measurements of these short-wavelength fluctuations.

Research Milestone R(07-1) Study variation of local high-k turbulence with plasma conditions.

Description: The results of Milestones R(06-1) are expected to establish the basis for performing in FY2007 a detailed investigation of the variation of local high-k turbulence with core plasma conditions. The high-k scattering system on NSTX, capable of measuring core density fluctuations with radial wavenumbers 2 – 22 cm^{-1} with excellent spatial resolution, will enable detailed studies to be made of electron-scale-length fluctuations; data from this diagnostic will be augmented by correlation and quadrature reflectometry and far-infrared tangential polarimetry. Systematic scans will be performed of parameters affecting electron thermal transport. The correlations amongst these measurements, the experimentally inferred transport and simulations of fluctuations with a suite of turbulence codes will be determined over a range of plasma conditions. Progress in this area will be of high interest to the goals of developing a predictive understanding of transport and turbulence, and to the optimization of the burning plasma in ITER and the confinement in CTF.

Research Milestone R(08-1) Measure poloidal rotation at low A to constrain theory.

Description: With the anticipated progress during FY2006-2007 in determining the variation of local high-k turbulence with plasma conditions, measurements of poloidal flow velocity will provide key data needed to assess the role of flow shear in controlling plasma turbulence and transport in NSTX plasmas. The research will utilize the new poloidal rotation diagnostic, based on charge-exchange recombination emission spectroscopy, to be installed and commissioned during FY2007, and build on the broad progress already made and planned on NSTX in transport and turbulence. Progress in this direction contributes to the scientific basis needed to optimize the burning plasmas in ITER and CTF.

2. Macroscopic Stability: Role of magnetic structure on plasma pressure and bootstrap current:

Research Milestone R(06-2) Characterize effectiveness of closed-loop Error Field control with ITER-like EF/RWM coils.

Description: As the pressure in strongly shaped NSTX plasmas is raised toward and sustained near the “wall-stabilized” ideal limit for durations longer than the time scales of eddy-current decay in the nearby wall, Resistive Wall Modes (RWMs) can grow and limit plasma stability. In strongly rotating plasmas, theory predicts that by controlling the error fields the onset of the RWM can be delayed or suppressed. Active feedback of the amplitude of the resonant error fields will be studied to characterize the effectiveness of this approach in FY06. Maintaining stability near the “wall-stabilized” pressure limit is of interest to performance improvements in NSTX, burning plasma experiments such as ITER, and future fusion energy producing devices.

Research Milestone R(07-2) Characterize effectiveness of closed-loop RWM control & dependence on rotation using ITER-like control coils.

Description: Research on Milestone R2(06-2) will have characterized the effectiveness of feedback control of the resonant field errors on stabilizing the wall-coupled, pressure-limiting RWMs over a range of plasma conditions. As an important extension of this work, active feedback on the amplitude of the RWMs is expected to enhance plasma stability beyond feedback on the resonant error fields alone, particularly accounting for the effects of plasma rotation. Externally applied field errors from a set of six feedback coils will be used to reduce plasma rotation or to compensate the device intrinsic error fields to maintain high rotation. The physics of plasma damping by resonant field errors, via interactions with electromagnetic and sound-wave perturbations, will be clarified. These results, together with further improvements in the speed of the control circuits and the feedback algorithms, will be used as a basis for testing active feedback control of the RWM under conditions where such control is needed to maintain stability. Stabilization of RWMs as the plasma pressure approaches the "wall-stabilized" limit by direct feedback and by plasma rotation is of interest to performance improvements in NSTX and burning plasma experiments such as ITER and CTF.

3. Wave-Particle Interaction: Role of electromagnetic waves & modes in sustaining and controlling hot plasmas:

Research Milestone R(07-3) Measure, identify & characterize modes driven by super-Alfvenic ions.

Description: NSTX will in FY 2007 measure, identify, and characterize magnetosonic modes that are driven by energetic ions traveling faster than the Alfvén speed. Such super-Alfvénic ions are expected in ITER and CTF. A suite of tools will be applied to this research. Fast soft x-ray cameras, magnetic sensors, reflectometers, and interferometers will measure the mode properties. Energetic neutral particle analyzers, fast-ion loss probes and neutron detectors will diagnose the energy distribution of these super-Alfvénic ions to reveal the effects of the modes. Diagnostics, including Thomson scattering, charge-exchange recombination spectrometry, and motional Stark effect polarimetry will measure the plasma profiles. The combined data will be compared with extensive modeling, simulation, and transport analysis to establish an understanding of modes driven by the super-Alfvénic ions. The results will shed light on how the modes can affect the burning plasma performance in ITER and CTF.

4. Start-up, Ramp-up and Sustainment: Physical processes of magnetic flux generation and reconnection:

Research Milestone R(06-3) Assess closed-flux generation using transient Coaxial Helicity Injection (CHI).

Description: Elimination of the central solenoid is an important consideration for future toroidal confinement devices, which will then require alternative methods for initiating the plasma current. Experiments were performed during FY05 applying the new technique called "transient CHI," which was successfully developed recently on the HIT-II Concept Exploration experiment. In this method, the plasma current is rapidly generated by discharging a capacitor bank between coaxial electrodes in the presence of a toroidal magnetic field and an applied poloidal field configuration that propels the plasma rapidly into the chamber. When the injected current is abruptly terminated, magnetic reconnection occurs near the injection electrodes to form closed flux surfaces. On NSTX in FY05 this method achieved persistent toroidal plasma currents up to 60 kA with camera images indicating closure of the magnetic flux surface. These interesting experimental results will be analyzed and submitted for publication.

5. Boundary Physics: Interface between fusion plasmas and normal temperature surroundings:

Research Milestone R(06-4) Characterize effects of lithium wall coating on recycling.

Description: In 2005, the NSTX lithium pellet injector was used to coat the contact areas of the plasma facing components with small amounts of lithium. This produced a transient reduction in the particle recycling for both limited and divertor discharges until the coating was depleted. In 2006, a lithium evaporator will be installed to coat a substantial fraction of the plasma facing surfaces in NSTX. Characterize the effectiveness of using lithium evaporation in controlling particle recycling.

Research Milestone R(08-2) Study edge/divertor at low plasma collisionality with ITER-level heat fluxes.

Description: Results from Research Milestones in FY2006-2007 are expected to provide the information needed to produce long-pulse, high-performance plasmas in NSTX over ranges in the edge shaping, temperature, density, H-mode pedestal conditions, and the types of Edge Localized Modes (ELMs). In particular, research using lithium wall coating to reduce recycling will be aimed at obtaining low density, and hence low collisionality, in the plasma edge. This will establish a basis for assessing the plasma conditions and operational requirements for handling edge heat fluxes up to the level of 10 MW/m^2 anticipated in ITER. Progress in understanding the plasma boundary for high levels of the edge power relative to major radius is important to achieving reproducible, sustained burning plasmas in ITER. Heat and particle fluxes from H-mode plasmas in NSTX will be characterized over a range of ELM conditions, using a suite of diagnostics for the plasma edge, the scrape-off layer (SOL), and the plasma facing components. The effects of lithium coating on the edge fluxes will be measured, together with its potential benefits on core plasma properties. The effects of the varying ELM conditions will be documented to determine their role in the power flow and the balance between cross-field and parallel transport in the SOL. Operating regimes that show improved plasma core conditions with control of the ELMs and edge fluxes will be characterized and documented. Simulation codes for the edge heat and particle transport, instabilities and turbulence will be applied to interpret the data, to clarify the mechanisms that drive these fluxes and to achieve the conditions of interest. The results will contribute to a predictive understanding of plasmas fluxes on NSTX and the burning plasmas in ITER and CTF.

6. Physics Integration: Synergistic effects of external control and self-organization:

Research Milestone R(08-3) Perform long-pulse plasmas in conditions relevant to CTF.

Conditions will be studied on NSTX in which the toroidal plasma current is maintained for durations beyond the plasma current redistribution time with minimal magnetic flux induction from the central solenoid. Strong neutral-beam injection (NBI) will be applied in low density plasmas, obtained using such techniques as lithium wall coating to be developed during FY2006-2007. This will increase the fraction of current driven by the NBI towards that anticipated in CTF. Such plasmas would be developed from discharges already produced in NSTX in which the solenoid-induced loop voltage has been reduced to the range 0.1 - 0.2 V for durations much greater than the current redistribution time. This was achieved by a combination of optimizing the current ramp-up, an early transition to the H-mode and strong plasma shaping to increase plasma stability and to minimize the impact of ELMs. Possible synergistic effects between the current drive mechanisms will be investigated to determine the optimal plasma scenarios. Simulation codes, which will have been benchmarked through comparison with recent NSTX data, will be used to suggest combinations of techniques to produce long-pulse plasmas in conditions relevant to CTF.

Future Facility and Diagnostic Accomplishments

Facility and Diagnostics Plan for FY 2006

Facility Milestone F(06-1): Operate NSTX Facility for 11 Experimental Run Weeks (September 2006)

Facility Milestone F(06-2): Install and commission a lithium evaporator on NSTX (June 2006)

A lithium evaporator will be designed, fabricated, installed, and commissioned to enable application of lithium coating on a substantial fraction of the plasma facing components in NSTX.

Diagnostic Milestone D(06-1): Complete the shop fabrication of new port flanges, shutters, and divertor plates for an interim poloidal rotation diagnostic using charge-exchange recombination emission (September 2006)

The shop fabrication of new port flanges, shutters, and divertor plates in support of the FY 2007 outage for an interim poloidal charge-exchange spectroscopy (P-CHERS) diagnostic will be completed. This diagnostic will be installed in FY 2007 to measure the spatial profile of the poloidal plasma flow in the outer half of the minor radius in FY 2007. These will enable measurement in FY 2007-2008 of the poloidal flow, which contributes to the shearing rate and plays a role in the suppression of large-scale turbulence that is believed to be responsible for the good ion confinement in NSTX. The rotation measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

Facility and Diagnostics Plan for FY 2007 – FY 2008

As described in the FY 2007 and FY 2008 Research Plan, for the purpose of this planning exercise, we shall consider the following two cases as requested: Base Case - 12 run weeks in FY 2007 and 10 run weeks in FY2008 with only a modest upgrading of facility and diagnostic capabilities; Incremental Case: 20 run weeks for both FY 2007 and FY 2008 which significantly increases the facility utilization and allows a proper level upgrading of facility and diagnostic capabilities including the 1 MW EBW system consistent with the pace of progress envisioned in the NSTX Five Year Plan.

<u>Base Case: 12 run weeks in FY2007 and 10 run weeks in FY 2008</u>

Facility Milestones for FY 2007

Facility Milestone F2(07-1): Operate NSTX Facility for 12 Experimental Run Weeks (September 2007)

Diagnostic Milestones for FY 2007

Diagnostic Milestone D(07-1): Install and commission an interim poloidal rotation diagnostic using charge-exchange recombination emission. (September 2007)

The interim diagnostic will be installed in FY 2007 to measure the spatial profile of the poloidal plasma flow in the outer half of the minor radius in FY 2007. These will enable measurement in FY 2007-2008 of the poloidal flow, which contributes to the shearing rate and plays a role in the suppression of large-scale turbulence that is believed to be responsible for the good ion confinement in NSTX. The rotation measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

Facility Milestones for FY 2008

Facility Milestone F(08-1): Operate NSTX Facility for 10 Experimental Run Weeks (September 2008)

Diagnostic Milestones for FY 2008

Diagnostic Milestone D(08-2): Upgrade the edge poloidal rotation diagnostic to increase its spatial resolution and coverage. (September 2008)

The interim poloidal charge-exchange spectroscopy (P-CHERS) diagnostic will be upgraded by purchasing and commissioning additional spectrometers and detectors to increase its spatial resolution and coverage toward that was achieved by the present 51 channel toroidal CHERS system on NSTX. Data from TFTR, for example, have shown that extremely localized structures can develop in the poloidal rotation profile prior to the development of internal transport barriers. Such structures may be related to the suppression of ion-scale turbulence by sheared plasma flow. The measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

<u>Incremental Funding Case: 20 run weeks in FY 2007 and FY 2008</u>

This incremental request case restores both the facility utilization facility/diagnostic upgrades toward the NSTX 5 year plan. In this case, the majority of the incremental facility funds will go toward the construction of the 1 MW EBW system to be completed and available for operations in FY 2009 and support 20 run weeks.

Incremental Facility Milestones for FY 2007

Incremental Facility Milestone IF(07-1): Increase the NSTX Experimental Run Weeks from 12 to 20 (September 2007)

Incremental Facility Milestone IF(07-2): Complete preliminary design of 1 MW EBW heating and current-drive system, begin site preparation, launcher design and modification of the power supply. (September 2007)

This is the first milestone related to a multi-year project to install and commission a 1 MW EBW system on NSTX to provide localized heating and current drive for advanced high-beta, long-pulse scenarios, as described in the NSTX Five-Year Plan developed in 2003. During FY 2007 the conceptual design for the system, including the launcher, the transmission system and the ancillary services, will be completed and site preparations will begin for installing two 0.5 MW gyrotron sources. The design of the launcher for the 1 MW EBW system will progress from the conceptual to the preliminary design stage. Work will also begin to modify and upgrade an existing power supply (that was originally part of the TFTR NBI system) to provide power to the gyrotron sources.

Incremental Facility Milestone IF(07-3): Design and fabricate the components for the symmetric end-feed HHFW antenna system. (September 2007)

The NSTX HHFW system has heated electrons to up to ~ 4 keV and shown promising indications of driving plasma current. The power modulation experiment showed good heating efficiency of 70 – 80% with the antenna elements operating in the heating phasing. However, reliable operation of the HHFW system has been limited to power levels of 3 – 4 MW by arcing in the antenna RF feed-through area. The power handling capability of the antenna could be dramatically improved by implementing a double-end-fed, center-ground design (symmetric antenna) which can couple twice as much power for a given feed-through voltage. Suitable RF feed-throughs are already available which will reduce the cost of this upgrade. This milestone calls for the design and fabrication of the symmetric antenna components.

Incremental Diagnostic Milestones for FY 2007

Incremental Diagnostic Milestone ID(07-1): Accelerate the upgrade of the edge poloidal rotation diagnostic to increase its spatial resolution and coverage by one year from FY 2008 to FY 2007. (September 2007)

In place of the interim poloidal charge-exchange spectroscopy (P-CHERS) diagnostic, the incremental funding will enable a full set of spectrometers and detectors to achieve the spatial resolution and coverage comparable to the present 51 channel toroidal CHERS system on NSTX in FY 2007 on year ahead of the base plan. Data from TFTR, for example, have shown that extremely localized structures can develop in the poloidal rotation profile prior to the development of internal transport barriers. Such structures may be related to the suppression of ion-scale turbulence by sheared plasma flow. The measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and the MHD stability of plasmas.

Incremental Facility Milestones for FY 2008

Incremental Facility Milestone IF(08-1): Increase the NSTX Facility Experimental Run Weeks from 10 to 20. (September 2008)

Incremental Facility Milestone IF(08-2): Complete 1 MW EBW system site preparation and power supply modifications, and begin the antenna system fabrication. (September 2008)

This is the second year of the project to install and commission a 1 MW EBW system for localized heating and current drive. In FY 2008, with incremental funding, the power supply for the gyrotron sources and related infrastructure will be completed and fabrication of other components including the microwave transmission lines and the coupler will be initiated.

Incremental Facility Milestone IF(08-3): Install the HHFW antenna system with symmetric end feed. (February 2008)

The HHFW symmetric end-feed components fabricated in FY 2007 (milestone IF(07-3)) will be installed and commissioned on NSTX in preparation for the FY 2008 run.

Incremental Facility Milestone IF(08-4): Complete conceptual design for advanced power and particle handling (September 2008)

In order to provide the power and particle handling required for non-inductively-sustained, long-pulse discharges, NSTX is pursuing two options: one utilizing lithium and the other cryo-pumping. The lithium pellet injector and, under the Requested Budget scenario for FY 2006, the lithium evaporator, will have been used to investigate the effectiveness of lithium wall coating on the power and particle control. If the lithium approach looks promising, NSTX will consider adopting the lithium based power and particle handling system. If the lithium is found not to be suitable, then NSTX will evaluate the alternative of a divertor cryo-pump, an approach that has been extensively tested on DIII-D and other tokamaks for particle control. The adaptation of a cryo-pump to the NSTX geometry is considered to be relatively straightforward. The decision on which method to implement will be based on the design and modeling studies and the experimental results from NSTX and other relevant experiments.

Incremental Diagnostic Milestones for FY 2008

Incremental Diagnostic Milestone ID(08-1): Commission an additional laser on the multi-pulse Thomson scattering system (September 2008)

The multi-pulse Thomson scattering system presently has two Nd-YAG lasers, each operating at 30 Hz. These are usually interleaved with equal spacing although syncopated operation can be used for studying reproducible transient events with higher time resolution. However, better resolution of random events, such as instabilities and the formation of transport barriers could be obtained by installing third laser to achieve 90 Hz coverage of the profile evolution. This will be particularly useful for diagnosing the phenomena accompanying pellet injection or other advanced fueling techniques.

RELATIONSHIP TO OTHER PROGRAMS AND INTERNATIONAL COOPERATION

Contribution to Burning Plasma Enhancement and Cooperation with Tokamak Research

International Tokamak Physics Activity (ITPA) contributions: The large toroidal curvature and high beta of the low aspect ratio introduce high scientific leverage in resolving several remaining physics issues of importance to enhancing the burning plasma performance of ITER. The NSTX research team was recently encouraged by the Topical Groups (TGs) to increase participation in the ITPA joint experiments beyond the 2005 goals. For 2006, discussions with the TGs and the ITPA Coordinating Committee (CC) identified 22 (see Table below) high-priority joint experiments in which NSTX will participate.

Topical Group	ID #	Joint Experiment	Participating Programs
Confinement Database & Modeling	CDB-2	β degradation in confinement scaling of ELMy H-modes	NSTX, MAST, AUG, DIII-D, JET, JT-60U, Tore-Supra
	CDB-6	Improving the condition of Global ELMy H-mode and pedestal database: low A	NSTX, MAST, DIII-D
	CDB-8	ρ* scaling along an ITER relevant path at both high and low beta	NSTX, JET, DIII-D, C-Mod, AUG
	CDB-9	Density profiles at low collisionality	NSTX, JET, DIII-D, C-Mod, AUG, JT-60U, TCV, Tore-Supra, MAST, FTU, T-10
Transport Physics	TP-6.3	NBI-driven momentum transport study	NSTX, DIII-D, JT-60U, MAST, JET
	TP-8.1	ITB similarity experiments	NSTX, MAST
	TP-9	H-mode aspect ratio comparison	NSTX, MAST, DIII-D
Pedestal and Edge Physics	PEP-9	Pedestal similarity experiments	NSTX, MAST, DIII-D
	PEP-10	Radial efflux at the mid-plane and the structure of ELMs	NSTX, AUG, MAST, C-Mod
	PEP-16	Small ELM regime comparison	NSTX, MAST, C-Mod
Divertor, Scrape-Off	DSOL-15	Comparison of edge "blob" characteristics	NSTX, C-Mod, TJ-II, Tore-Supra, JT-60U

Layer	DSOL-18	Impurity migration and deposition study	NSTX, AUG, JET
MHD, Disruption Control	MDC-2	Resistive wall mode physics	DIII-D, NSTX, JET, TEXTOR, JT-60U, AUG
	MDC-4	Neoclassical tearing mode physics – aspect ratio comparison	NSTX, AUG, MAST, DIII-D
	MDC-5	Comparison of sawtooth control methods for neoclassical tearing mode suppression	NSTX, AUG, DIII-D, JET, TCV, HL2A, C-Mod, FTU
	MDC-6	Error field comparison	NSTX,C-Mod, MAST, TEXTOR, DIII-D, JET
	MDC-9	Fast ion redistribution by fast ion driven Alfvén modes and Alfvén Cascades threshold	NSTX, JT-60U, JET, AUG, DIII-D
Steady State Operation	SSO-2.1	Complete mapping of hybrid scenario	NSTX, JET, JT-60U, DIII-D, AUG
	SSO-2.2	MHD effects on q-profile and confinement for hybrid scenarios	NSTX, AUG, JET, DIII-D, JT-60U, C-Mod
	SSO-2.3	ρ* dependence on confinement, transport, and stability in hybrid scenarios	NSTX, AUG, JET, DIII-D, JT-60U, C-Mod
Diagnostics	DIAG-1	Assessment of the effect of noise on vertical velocity measurements	NSTX, JET, JT-60U, TCV, AUG, C-Mod
	DIAG-2	Environmental tests on diagnostic first mirrors (FMs)	T-10, TEXTOR, Tore-Supra, JET, DIII-D, TCV, AUG, LHD, FTU, C-Mod, JT-60U

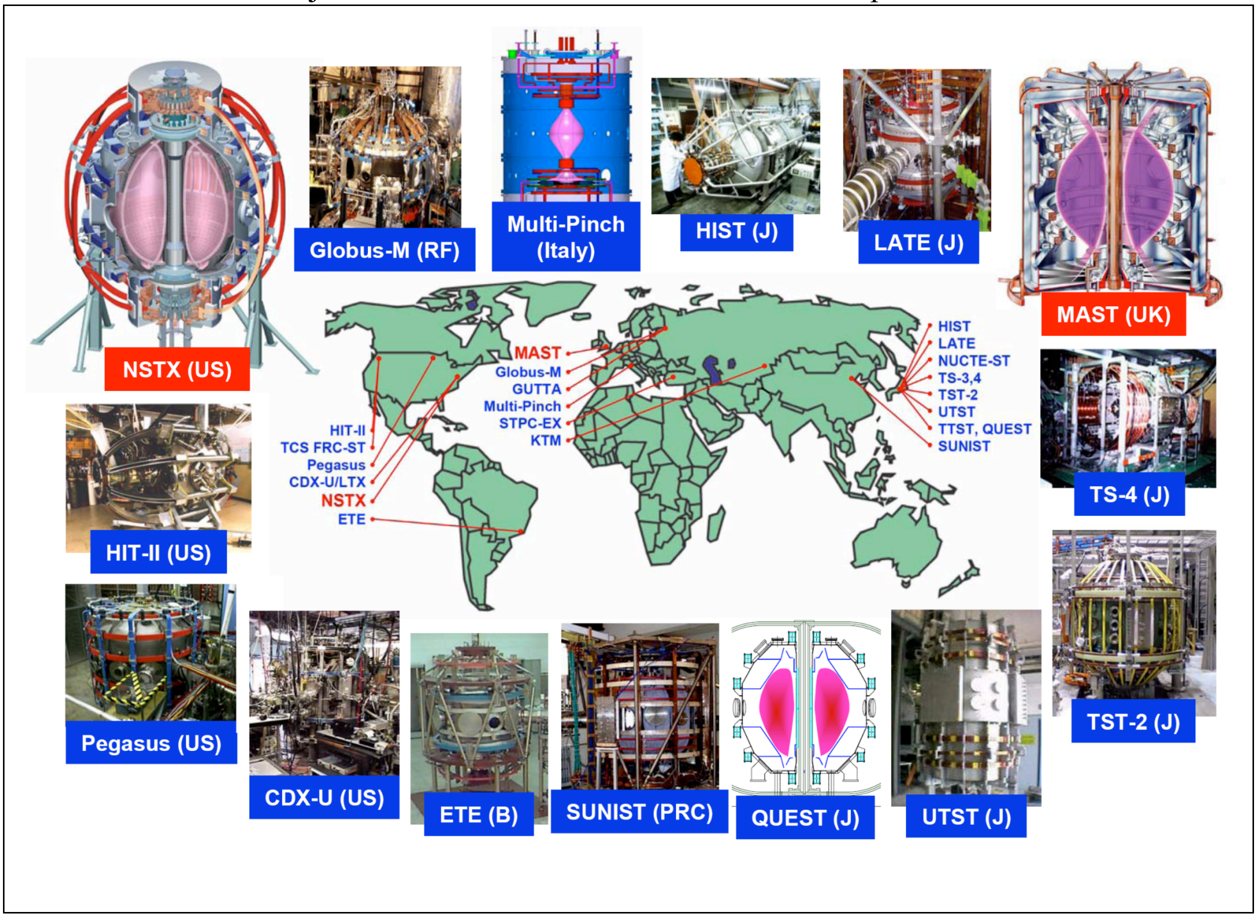
DIII-D, C-Mod, HBT-ET coordination and co-operation: A substantial level of collaboration is planned with the U.S. tokamak experiments recently, to jointly address issues that have bearing on enhancing the performance of future burning plasma experiments (see table above). DIII-D at General Atomics and C-Mod of MIT are actively involved in this cooperation. Additional joint research with C-Mod is in progress on X-ray crystal spectroscopy to measure the core T_i profiles.

ST Programs in U.S.:

The ST Programs in U.S. form a strong component of the Innovative Confinement Concept (ICC) research, which addresses the FESAC-defined FES Strategic Goal #2 on Configuration Optimization: "*Resolve outstanding scientific issues and establish reduced-cost paths to more attractive fusion energy systems by investigating a broad range of innovative magnetic confinement configurations*." The research on PEGASUS at the University of Wisconsin, Helicity Injected Tokamak-SI (HIT-SI) at the University of Washington, and Lithium Tokamak Experiment (LTX) at PPPL, will support this goal by exploring the new boundaries in the physical and engineering sciences of the ST. When an International Energy Agency (IEA) Implementing Agreement (IA) on ST research cooperation is established, a U.S. Spherical Torus Coordinating Committee (STCC) would be formed by the research leaders of these programs and the NSTX to help formulate and advocate the U.S. ST Program goals and priorities internationally.

ST Programs in the World:

ST experimental programs have emerged during the past 10 years in U.K., Japan, R.F., Italy, Brazil, PRC, and Turkey, composing of 22 experiments operational of under construction (see figure below). Substantial cooperation with the ST programs in the former three countries has made major contributions to the research on these experiments and the NSTX.



IEA Implementing Agreement on ST Research Cooperation: Progress was made recently by DOE in obtaining agreement within the IEA to establish a new IA on ST R&D Cooperation, initially involving ST programs in U.K., Japan, and U.S. Approval from all of these countries is now likely. This agreement, when established, will provide a vehicle for effective coordination of cooperation among the already diverse world ST research programs mentioned above, ready application of the best available capabilities and expertise in the world to the challenging and exciting ST research, coordination of ST physics input of benefit to the International Tokamak Physics Activities (ITPA), and joint efforts toward future ST experiments, with potential high payoff to the U.S. Fusion Energy Sciences Program.

Research Cooperation with Other DOE Programs

OFES Innovations in Magnetic Fusion Energy Diagnostic Systems: NSTX has cooperated with the Advanced Diagnostic Program (AT 50 10 80 2) by providing access to the unique high beta high temperature plasmas and interface support for tests of innovative diagnostic techniques, which are of great interest to research on other high beta confinement concept such as the MST, SSPX, and FRC. The most recent examples include:

- Development of Motional Stark Effect (MSE) polarimetry based on laser-induced fluorescence (LIF) to enable the separation of the magnetic field and electric field effects on fast deuterium recombination spectrum, in low field high beta plasmas such as the ST.
- Innovative use of the Electron Bernstein Wave (EBW) emission from, and mode conversion at the edge of, over-dense high beta plasmas to measure turbulence fluctuations in the core electron temperatures and the edge density gradients.
- Innovative technique of X-ray crystal spectroscopy to map in 2-D the core electron and ion temperature profiles, which is expected to be applicable to ITER plasmas in the ITER nuclear environment.
- Progress being made by the Advanced Diagnostics Program on fluctuation imaging using 3-D microwave reflectometry to assess the long-wavelength turbulence in TEXTOR for application on NSTX.

OFES Enabling Technology of VLT: NSTX cooperates with the Enabling Technology program under the VLT. These include the collaboration on High Harmonic Fast Wave rf technology (ORNL), which has led to an extensive research plan on rf heating and current drive on NSTX. The NSTX program has further benefited from the expert participation of VLT researchers on the topics of plasma facing material and particle deposition (SNL and ANL). In addition, the NSTX and the ALPS program have cooperated in developing initial tests of lithium wall-plasma interactions on CDX-U and NSTX. These programs are working together to establish the scientific and technical requirements and database needed for possible future application of a liquid lithium surface plasma facing component on NSTX to enable strong control of deuterium recycling from the wall.

Theory Program: NSTX research has also benefited substantially from working with experts in the Theory program of OFES in areas of Boundary Plasma Stability and Turbulence, Transport, RF Heating, and energetic particle driven Compressional Alfvén Wave Instabilities. The new ST plasma regimes have introduced several new scientific opportunities of high importance to Fusion Energy Sciences Program that can benefit greatly from a fresh look by Theory experts in the field.

EXPLANATION of BUDGET

The NSTX Budget Summary for FY 2006 – 08 is shown in Table 2. The 12-run-week base budgets in FY 2007 have a minimal upgrades budget. The constant $ FY 08 base budget has an inflation impact of $1M and it reduces the operations to 10 run week and further reduces the facility and diagnostic upgrades. The proposed incremental budget will restore the facility operations to an optimum level of 20 run weeks for FY 2007 and FY 2008. The 20 run week plan with also has an adequate funding to provide for the 1 MW EBW system will enable the NSTX to make the progress as described in the NSTX Five Year Plan. The two cases therefore considered in the research, facility and diagnostic milestone section correspond to the following budget levels:

Base Case – 12 run weeks in FY 2007 and 10 run weeks in FY 2007

Incremental Case – 20 run weeks in FY 2007 and 20 run weeks in FY 2008

Table 2: NSTX Budget Summary ($M)

	FY 06	FY 07		FY 08	
Budget level	Actual	Base	Incremental	Base	Incremental
Run Weeks	11	12	20	10	20
Facility Operation	17.7	18.2	1.0	18.4	1.4
Facility Upgrades	0.5	0.5	1.8	0.4	2.1
Facility Total	**18.2**	**18.7**	**2.8**	**18.8**	**3.5**
PPPL Research	9.7	9.8	0.2	9.9	0.5
Diag Upgrades	0.6	0.8	0.5	0.6	0.5
Coll. Diag. Interf	0.5	0.6	0.1	0.6	0.1
Collaborations	5.0	5.2	0.3	5.2	0.3
Science Total	**15.8**	**16.4**	**1.1**	**16.3**	**1.4**
NSTX Total	**34.0**	**35.1**	**3.9**	**35.1**	**4.9**

The Base budget presented in Table 2 with 12 run weeks in FY 2007 and 10 weeks in FY 2008 gives highest priority to plasma operation in NSTX but allows only modest upgrades and facility maintenance. The Incremental budget in FY 2007 and FY 2008 allows 20 run weeks of operation nearly doubling the base run week plan. The Incremental plan also allocates a healthier budget to upgrades, amounting to about 10%, permitting exciting facility and diagnostic upgrades as envisioned in the NSTX Five Year Plan. In particular, it accelerates the 1 MW EBW upgrade to be built over 3 years (to be available in FY 2009).

Other important and timely upgrades possible with the Incremental budget include a deuterium-ice pellet injector for advanced plasma fueling, a bi-polar power supply for the solenoid-free start-up, and diagnostics for improved plasma profile and boundary physics.

FY 2006 Budget - In FY 2006, the base budget enables the NSTX facility to operate for 11 run weeks and to implement new facility and diagnostic upgrades including a lithium evaporator implementation and the shop fabrication for the new component needed for the poloidal-CHERS upgrade. The lithium evaporator will enable testing the effectiveness of the lithium coating for particle recycling control during the high performance long-pulse discharges. The planned 11 run weeks while relatively short will provide the NSTX Research Team the opportunity to execute an exciting experimental research program planned to explore new plasma regimes and to pursue new physics opportunities.

FY 2007-2008 Base Budget – The 12 run week budgets for FY 2006 and the 10 run weeks for FY 2007 are relatively restricted budgets under which the highest priority is facility operation and only modest upgrades and maintenance are undertaken. The upgrade budget is only about 4% in FY 2007 and 3% in FY 2008, low compared to the historical values and very low (only about on third) compared to the 5 Year Plan. Only upgrades which are of immediate benefit to the run will be implemented. A second lithium evaporator if needed will be installed with a remotely controllable carriage mechanism to allow application of lithium between shots. The fast IR camera will be implemented in FY 2007 since the rapid power loss during ELMs is an important topic for the ITPA as well as NSTX. The interim P-CHERS system will be commissioned in FY 2007 due to the importance of the poloidal flow measurements in determining the current profile with the MSE-CIF diagnostic and the P-CHERS system will be upgraded to full channels, comparable to the existing toroidal CHERS system in terms of spatial coverage and resolution in FY 2008.

Explanation of Incremental 20 Run Week Budgets for FY 2007 – FY 2008

The incremental funding proposed in FY 2007 and FY 2008 will be used to restore the NSTX run weeks toward full utilization of 20 weeks in both years and to implement key facility and diagnostic upgrades as described in the NSTX Five Year Plan. The 20 run weeks will enable the NSTX research program to advance the NSTX research mission more aggressively and expeditiously. Incremental funding will also enable preventive maintenance, the purchase of spare parts, and improvements in operations to ensure high facility availability and reliability.

Incremental funding in FY 2007 would accelerate research on minimizing the induction flux during current ramp-up in conditions relevant to CTF as described in the incremental milestone IR(07-1). This would strengthen the basis for achieving the Research Milestone R(08-3), which is to perform long pulse plasmas in conditions relevant to CTF. For FY 2008, the incremental funding would enable the NSTX Team to utilize the anticipated achievement of Research Milestone R(07-3), which is to Measure, identify & characterize modes driven by super-Alfvénic ions, to make the needed progress in an incremental milestone IR(08-1) aimed at evaluating the induced transport of super-Alfvénic ions anticipated in ITER .

The incremental funding requested would permit implementation of key facility upgrades. If an assessment at the end of FY 2006 for an EBW system on NSTX were favorable, the proposed incremental funding in FY 2007 and FY 2008 would permit the design and construction of 1 MW EBW system to be installed and commissioned in FY 2009. This system should produce a significant localized driven current of about 35 kA which would be measurable with the MSE diagnostic and verify the utility of EBWCD, which is highly relevant to increasing the performance of CTF to beyond 2 MW/m^2 in neutron wall loading at the test modules. This level of current induced locally can also contribute toward the investigation of NTM stabilization.

In the area of particle control, efficient core fueling and particle pumping are important to reduce the particle recycling and inventory. An assessment will made after the lithium evaporator experiments at the end of FY 2006 to identify the needs for advanced pumping and core fueling tools. In the area of solenoid-free start-up, an efficient pre-ionization source is identified to be critical for investigating an alternative start-up concept to CHI utilizing the outer poloidal field coils. The decision points for the pre-ionization source is schedule at the end of FY 2006. The incremental funding will permit the implementation of those required tools as identified at the decision points.

The incremental diagnostic funding in FY 2007 will enable the acceleration of the full poloidal CHERS system from FY 2008 to FY 2007 providing a comparable spatial coverage and resolutions to the 51 channel toroidal CHERS system. The incremental funding is requested for both in FY 2007 and FY 2008 to improve the NSTX diagnostics to better understand the particle and power flows in the boundary and divertor regions. These data are needed to assess the heat and particle handling requirements for the long-pulse high performance regimes envisaged in the CTF concept. The knowledge thus developed will also contribute to the design of plasma facing components (PFCs) for the CTF. The incremental funding in FY 2008 will also enable installation of a third laser for MPTS to increase the time resolution of the Thomson scattering system to improve the determination of the electron power flow in the plasma.

RESEARCH MILESTONES - Baseline

Milestone	Description	Baseline	Forecast	Actual
FY 2005				
R(05-1)	Characterize the effects of variations in the magnetic shear and gradients in T_e on electron transport in low-aspect ratio plasmas	Sep 05		Sep 05
R(05-2)	Produce and characterize strongly shaped, rotating, low aspect ratio plasmas close to the "wall-stabilized" pressure limits with error field correction	Sep 05		Sep 05
R(05-3)	Assess the effects of supra-Alfvénic fast ion driven instabilities on driven current in the plasma core	Sep 05		Sep 05
R(05-4)	Characterize the plasma edge pedestals and scrape-off layer of low-aspect ratio, high confinement, high P/R plasmas	Sep 05		Sep 05
R(05-5)	Characterize strongly shaped low-aspect ratio plasmas with high fractions of self-driven current and low toroidal induction voltage for durations that allow internal currents to redistribute	Sep 05		Sep 05
FY 2006				
R(06-1)	Perform highly localized measurement of magnitude of high-k turbulence.	Sep 06		
R(06-2)	Characterize effectiveness of closed-loop Error Field control with ITER-like EF/RWM coils	Sep 06		
R(06-3)	Assess closed-flux generation using transient Coaxial Helicity Injection (CHI).	Sep 06		
R(06-4)	Characterize effects of lithium wall	Sep 06		

coating on recycling

RESEARCH MILESTONES – Baseline (continued)

Milestone	Description	Baseline	Forecast	Actual
FY 2007				
R(07-1)	Study variation of local high-k turbulence with plasma conditions.	Sep 07		
R(07-2)	Characterize effectiveness of closed-loop RWM control & dependence on rotation using ITER-like control coils.	Sep 07		
R(07-3)	Measure, identify & characterize modes driven by super- Alfvénic ions.	Sep 07		
FY 2008				
R(08-1)	Measure poloidal rotation at low A to constrain theory.	Sep 08		
R(08-2)	Study edge/divertor at low plasma collisionality with ITER-level heat fluxes.	Sep 08		
R(08-3)	Perform long-pulse plasmas in conditions relevant to CTF.	Sep 08		

FACILITY MILESTONES - Baseline

Milestone	Description	Baseline	Forecast	Actual
FY 2005				
F(05-1)	Operate NSTX Facility for 17 Experimental Run Weeks	Sep 05		Aug 05
F(05-2)	Implement Resonant Field Correction system	Jun 05		Jun 05
F(05-3)	Commission a new pair of PF1A poloidal-field coils to produce high-triangularity, high-elongation plasma equilibria	Apr 05		Apr 05
FY 2006				
F(06-1)	Operate NSTX Facility for 11 Experimental Run Weeks.	Sep 06		
F(06-2)	Install and commission a lithium evaporator on NSTX.	Jun 06		
FY 2007				
F(07-1)	Operate NSTX Facility for 12 Experimental Run Weeks.	Sep 07		
FY 2008				
F(08-1)	Operate NSTX Facility for 10 Experimental Run Weeks.	Sep 08		

DIAGNOSTIC MILESTONES - Baseline

Milestone	Description	Baseline	Forecast	Actual
FY 2005				
D(05-1)	Install an additional 10 channels for the multi-pulse Thomson scattering system.	Sep 05		Jun 05
D(05-2)	Install a diagnostic system to measure short-wavelength plasma turbulence by scattering from the plasma density fluctuations.	Sep 05		Aug 05
FY 2006				
D(06-1)	Complete the shop fabrication of port flanges, shutters, and divertor plates for an interim edge poloidal rotation diagnostic using charge-exchange recombination emission.	Sep 06		
FY 2007				
D(07-1)	Install and commission an interim poloidal rotation diagnostic using charge-exchange recombination emission.	Sep 07		
FY 2008				
D(08-1)	Upgrade the edge poloidal rotation diagnostic to increase its spatial resolution and coverage.	Sep 08		

RESEARCH MILESTONES - Incremental

Milestone	Description	Baseline	Forecast	Actual
FY 2007				
IR(07-1)	Minimize induction flux during current ramp-up in conditions relevant to CTF.	Sep 07		
FY 2008				
IR(08-1)	Evaluate Alfvén and energetic particle mode induced transport of super-Alfvénic ions anticipated in ITER.	Sep 08		

FACILITY MILESTONES - Incremental

Milestone	Description	Baseline	Forecast	Actual
FY 2007				
IF(07-1)	Increase the NSTX Experimental Run Weeks from 12 to 20.	Sep 07		
IF(07-2)	Complete preliminary design of 1 MW EBW heating and current-drive system, begin site preparation, launcher design and modification of the power supply.	Sep 07		
IF(07-3)	Design and fabricate the components for the symmetric end-feed HHFW antenna system.	Sep 07		
FY 2008				
IF(08-1)	Increase the NSTX Facility Experimental Run Weeks from 10 to 20.	Sep 08		
IF(08-2)	Complete 1 MW EBW system site preparation and power supply modifications, and begin the antenna system fabrication.	Sep 08		
IF(08-3)	Install the HHFW antenna system with symmetric end feed.	Feb 08		
IF(08-4)	Complete conceptual design for advanced power and particle handling.	Sep 08		

DIAGNOSTIC MILESTONES - Incremental

Milestone	Description	Baseline	Forecast	Actual
FY 2007				
ID(07-1)	Accelerate the upgrade of the edge poloidal rotation diagnostic to increase its spatial resolution and coverage by one year from FY 2008 to FY 2007.	Sep 07		
FY 2008				
ID(08-1)	Commission an additional laser on the multi-pulse Thomson scattering system.	Sep 08		