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14. WORK PROPOSAL 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 toroidal 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 scientific issues in magnetic fusion energy science of 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 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 Spherical Torus (ST) in cost-effective development of fusion power, as indicated in the present DOE Office of Science Strategic Plan.

During FY2005-2007, 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 on topics of broad interest in fusion plasma physics. Research on NSTX has hitherto been successful due to 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 2005-2007 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 2005-2007 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 longer-term NSTX goals, will also be carried out.

15. Signatures:

signature on hard copy Contractor Work Proposal Manager	signature on hard copy . Contractor Work Proposal Manager

16. Human/Animal Subjects: No

OVERVIEW-PURPOSE

Missions: The scientific mission of the National Spherical Torus Experiment (NSTX) is to advance fusion plasma science by determining and understanding the physics principles of the Spherical Torus (ST), which is characterized by strong magnetic field curvature and high β_T (the ratio of the average plasma pressure to the applied toroidal magnetic field pressure) due to its low aspect ratio. These unique properties complement the normal aspect ratio tokamak in addressing the overarching scientific issues in magnetic fusion energy science, covering turbulence and transport, macroscopic MHD stability, wave-particle interaction, solenoid-free generation and sustainment of magnetic flux, and plasma interface with the surrounding environment. The programmatic mission of NSTX is to contribute to resolving important burning plasma physics issues anticipated in ITER and to determine the attractiveness of the ST for reducing cost, time and risk of development of practical fusion energy, through these scientific investigations.

Science Priorities and Relevance: The investigation of the ST plasmas is of interest to plasma science because the unique magnetic field line shape in an ST, depicted in Fig. 1, has properties desirable for confining fusion plasmas. It has strong shaping of the plasma cross section, combining low aspect ratio, high vertical elongation κ ($\leq$2.6), high edge safety factor q_{edge} (~10), and strong pitch of the field line B_p/B_T (~1) in the plasma outboard. The NSTX plasma is characterized by:

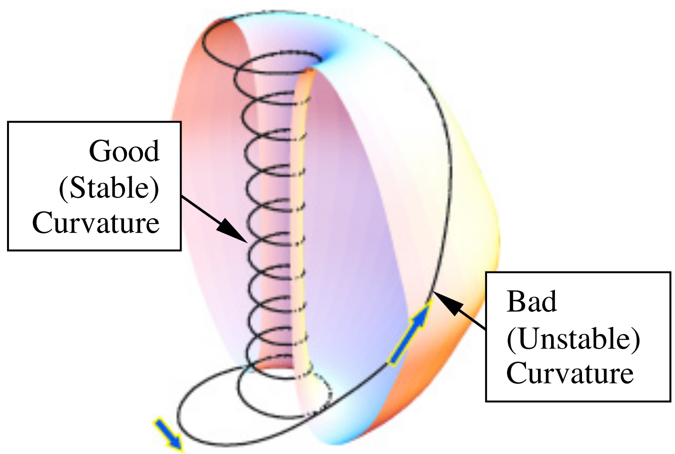

Figure 1. Strongly curved magnetic field line and surface near the ST plasma edge, with increased length of field line of "good (stable) curvature" over that of "bad (unstable) curvature," resulting in a strengthened hold on high-temperature, high-pressure plasmas.

- Very high toroidal average beta β_T ($\leq$0.4) and central local beta β_0 (~1) with an attendant high self-driven "bootstrap" current fraction f_{BS} ($\geq$0.5). This introduces strong electromagnetic features in plasma turbulence, and an opportunity to advance the science of electron scale turbulence and transport and β scaling in confinement. These remain unresolved issues of burning plasma optimization for ITER.
- Relatively small plasma size divided by the radius of ion gyration, ρ_{ci}, around the magnetic field: $a/\rho_{ci} = 1/\rho^*$ (~ 30-50), suggesting the possibility of moderate $1/\rho^*$ values in ST devices for energy applications. This introduces large flow gradients in the scale of

the ion orbit size to challenge and refine theories of ion scale turbulence and transport to reduce the uncertainties in ITER plasma performance projections that hinge strongly on this dimensionless parameter.

- Large plasma flow velocities relative to the intrinsic plasma velocities of magnetic perturbations (the so-called Alfvén velocity), $V_{flow}/V_{Alfvén} = M_A$ (≤ 0.4) and sound perturbations, $V_{flow}/V_{sound} = M$ (≤ 1), and flow shearing rate ($\sim 10^6$ /s). The presence of strong plasma flow relative to the these velocities introduces leverage with which to study and understand macroscopic stability, and energy, particle, and momentum transport. Progress in these directions will assist in developing a robust stable operating regime for driven steady state burning plasmas such as envisioned for the "hybrid" mode of ITER, and the conditions of a compact component test facility that could substantially reduce the cost, time, and risk of fusion energy development.
- Large velocities of the energetic ions relative to the Alfvén velocity, $V_{fast}/V_{Alfvén}$ (~ 2-5), i.e., the supra-Alfvénic ions introduced by the injection of neutral deuterium beams. This condition, which overlaps with that of ITER, will provide a crucial test of the physics of fast-ion-driven Alfvén Eigenmodes and their effects on the burning plasma.
- Strongly refractive "over-dense" plasmas of large dielectric constant, ε ($= \omega_{pe}^2/\omega_{ce}^2 \sim$ 10-100) affecting the conversion, propagation, absorption, and emission of radiofrequency "plasma" waves in the plasma. This new regime in the physics of wave-particle interactions will advance the science of plasma initiation and sustainment, which is of high importance to fusion power plants in the future.
- Large outboard edge and Scrape-Off Layer (SOL) magnetic mirror ratios, which can be ~2 for double-null divertor plasmas, ~4 for inboard limited plasmas, and ~4 for the plasma edge region inside of the SOL. This introduces large in-out asymmetries in the plasma edge and scrape-off-layer. Characterizing the interface between the plasma and its normal temperature surroundings in these conditions will contribute to establishing a predictive understanding of out-flowing heat fluxes from burning plasmas anticipated in ITER and other more compact options.
- Much reduced magnitudes of the internal magnetic flux and helicity, which is the product of the intertwined toroidal and poloidal magnetic fluxes. This minimizes inductive requirements of operation, facilitates plasma startup techniques without reliance on a central solenoid, and ensures the operability of highly compact component test facilities.

Office of Science Strategic Plan for Fusion Energy Sciences: The NSTX National Team, with participation by research collaborators from other countries, proposes to achieve during FY2005-2007 important progress in the above scientific and programmatic missions through vigorous investigation of these physics topics, taking advantage of the unique ST plasma properties and the reliable NSTX facility operation. The proposed achievements are consistent with the Strategic Goal 3.2 of the DOE Fusion Energy Sciences Plan (February 2004), which is to "*Develop a fundamental understanding of plasma behavior sufficient to provide a reliable predictive capability for fusion energy systems*." The high scientific leverage of research on NSTX, recognized by the International Tokamak Physics Activities (ITPA) Working Groups, enables key contributions to the Strategic Goal 3.1, which is to "*Demonstrate with burning plasmas the scientific and technological feasibility of fusion energy.*" The NSTX Program remains a key component of the Fusion Energy Sciences Program to address the Strategic Goal 3.3, which is to "*Determine the most promising approaches and configurations to confining hot plasmas for practical fusion energy systems*." A natural consequence of this programmatic logic will further establish a strong scientific basis for a highly compact configuration for application as a component test facility, supporting over the longer term the Strategic Goal 3.4, which is to "*Develop the new materials, components, and technologies necessary to make fusion energy a reality*."

The above considerations have guided the selection of the FY2005-2007 NSTX research milestones, which are described in FUTURE ACCOMPLISHMENTS and RELATIONSHIP TO OTHER PROGRAMS AND INTERNATIONAL COOPERATION.

APPROACH

A comprehensive approach covering research, diagnostics, and facility operations and upgrades has been developed and updated to achieve the NSTX mission and goals described above.

National Research Team: A broadly based NSTX National Research Team was brought together in November 1998 to carry out the NSTX research program and diagnostic upgrades. DOE selected initial members of the team, which have since then been broadened to include researchers from the following fusion research institutions: Columbia University, Comp-X, General Atomics, Idaho National Laboratory, Johns Hopkins University, Lawrence Livermore National Laboratory, Lodestar, Inc., Los Alamos National Laboratory, Lodestar, Massachusetts Institute of Technology, New York University, Nova Photonics, Oak Ridge National Laboratory, Princeton Plasma Physics Laboratory, Princeton Scientific Instruments, Sandia National Laboratory, University of California at Davis, University of California at Irvine, University of California at Los Angeles, University of California at San Diego, University of Maryland, University of Rochester, University of Washington, University of William and Mary, and University of Wisconsin. NSTX research achievements by the National Team during FY2004 are described in ACCOMPLISHMENTS.

National and International Cooperation: The present National Research Team is further strengthened by cooperation with experts separately funded by Tokamak Research, Advanced Innovative Diagnostic, Enabling Technology, and Theory programs of OFES, for the benefit of both NSTX and these programs. International cooperation has also become an important part of the research program, involving scientists from European Union (EURATOM/UKAEA Fusion Association, ENEA Frascati of Italy, IPP Jülich and IPP Garching of Germany), Japan (Hiroshima University, University of Hyogo, Kyushu Tokai University, Kyushu University, Niigata University, Tsukuba University, University of Kyoto, University of Tokyo), Russia (Ioffe Institute, TRINITI-Kurchatov Institute), and Korea (Korean Basic Science Institute, Korean Advanced Institute of Science and Technology). These cooperative research activities, briefly described in RELATIONSHIP TO OTHER PROGRAMS AND INTERNATIONAL COOPERATION, are encouraged and growing.

Annual NSTX Research Forum: Scientists of the National Research Team and other researchers interested in NSTX research are invited to participate in an annual NSTX Research Forum, usually held in September – October each year. The participants present and discuss ideas for research in the up-coming fiscal year. The results of the forum form the basis for the experimental run plan for the up-coming fiscal year. The Experimental Task (ET) forces, drawn from the National Research Team members, are formed annually to develop the Experimental

Proposals (XP's) for review and approval, and to carry out the approved XP's under the leadership of the NSTX Head of Research and Run Coordinator. A Team-wide operational assessment and a results review are conducted each year following the experimental campaign to take stock of key advances over the year to guide the preparation of the research, diagnostics, and facility plans for the subsequent fiscal year.

NSTX Five-Year Plan of 2003: The National Team developed a Five-Year NSTX Research Plan for FY2004-2008, for review by DOE in June 2003. The plan received strong endorsement by the DOE international review panel, and is available at http://nstx.pppl.gov/DragNDrop/NSTX_Five_Year_Plan/5Yr_Plan_Final/. The aggressive goals articulated in this plan directly support the NSTX mission, and can be achieved in the 5-year timeframe given adequate resources and funding. The proposed NSTX research and facility plan for FY 2005-2007, whilst aiming toward the goals of the Five-Year Plan, will do so according to a much stretched schedule due to strong limitations in available funding. However, the process to develop this plan sharpened the research team's understanding of the visions, the importance, the research program, and the diagnostic and facility requirements of the NSTX mission.

NSTX Program Advisory Committee (PAC): The NSTX PAC, consisting of the scientific leaders from the fusion community, usually meets twice annually (typically September, January) to advise the PPPL Director on the NSTX research program and facility operation plans and issues. The PPPL Director provides guidance to the research program and facility operations plans in response to the PAC's recommendations. The NSTX PAC has been instrumental to the success of the NSTX Program. The charter of the PAC and the PAC reports since FY1997 are available at: http://nstx.pppl.gov/.

NSTX Facility Operation: PPPL has primary responsibility for NSTX Facility operations and upgrades in support of the research program. This activity includes all NSTX engineering operations and facility maintenance and repair, diagnostics interface support, and a number of diagnostics upgrades in addition to the diagnostics provided through collaboration research team members.

The NSTX Device: The focus of NSTX Facility operation and upgrade is the NSTX device, which is depicted in Figure 2. Recent diagnostics and device operation achievements are summarized in ACCOMPLISHMENTS. A new toroidal field coil joint system was successfully built, installed, and commissioned during 2003 and successfully operated during 2004. Detail of the diagnostics and facility operation and upgrade plans is provided in FUTURE ACCOMPLISHMENTS.

Figure 2. Layout of NSTX, with magnets external to the vacuum vessel, close fitting stabilizing conducting plates, nearby HHFW launcher, large ports for NBI, ceramic insulators to enable CHI operation, and graphite tiles covering entire inboard, divertors, and the conducting plates. The fully demountable center stack assembly is a common feature of ST devices.

TECHNICAL PROGRESS (FY2004 – FY2005)

Research Accomplishments in FY 2004

The NSTX made scientific contributions to many areas that are critical to ITER, that contribute broadly to plasma science, and that help establish the physics basis for advanced, long-pulse ST operation. Much of this research took place through growing joint experimental efforts within the International Tokamak Physics Activity (ITPA).

Research was organized largely along topical physics lines (transport, MHD, wave-particle interactions, boundary physics), with additional emphases in plasma startup (helicity injection and poloidal field induction) and integration of key elements in plasma control to extend the high-beta plasma pulse length, the non-inductive current fraction, and the stability properties. Aspects of this research for 2004 are described in what follows.

1. Extending the NSTX Operating Space

Highlights - The NSTX operating space was expanded in 2004 through advances in control, including improved shaping. Plasma stability was also improved through the development of early access to the enhanced confinement H-mode regime. Aspects of this research were performed as a joint ITPA research activity.

The FY04 NSTX experimental run campaign began in January with a major milestone of 18 weeks of operation, and a programmatic goal to complete 20 weeks of experiments. By the end of the run in mid-August, NSTX had completed 844 hours of high power operation, or 21 effective run weeks, encompassing 2701 plasma attempts resulting in 2460 plasmas. Of these, 2166 discharges achieved currents greater than 0.2 MA, 1310 of them with neutral beam injection (NBI) only, 260 with High Harmonic Fast Wave (HHFW) heating only, and 214 with combined NBI and HHFW heating. A total of five Machine Proposals (XMPs) and 42 Experimental Proposals (XPs) received run time. Plasma operation during this campaign was reliable, with many extended run days and few unplanned stoppages, and it was interrupted only for a three week opening in March in order to replace a neutral beam calorimeter bellows and install optical dumps for the Charge-Exchange Recombination Spectroscopy (CHERS) diagnostic.

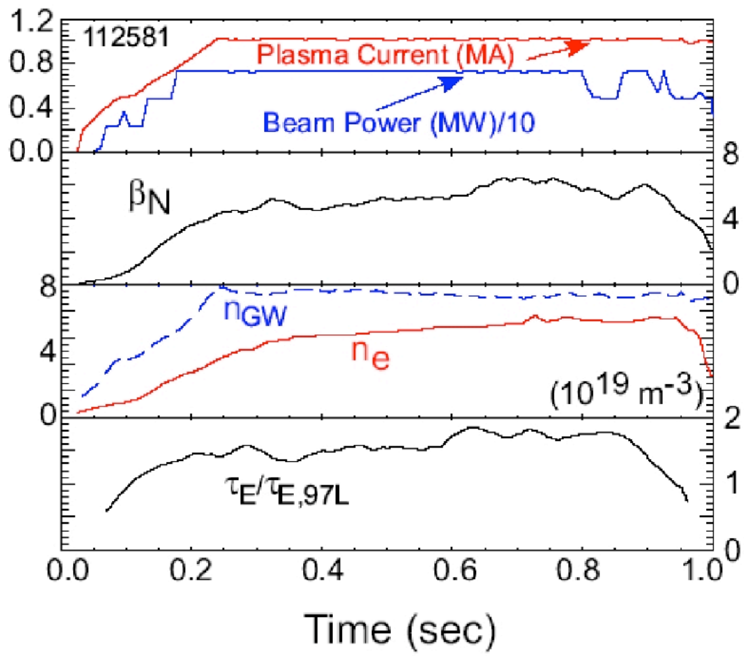

Figure 1. Integrated high-performance discharge with significant non-inductive current

Considerable progress was made towards accomplishing the integrated performance goal for the year, which was to produce high performance, long-duration plasma discharges sustained by significant amounts of non-inductive current drive. A specific example of such an integrated high performance discharge is shown in Figure 1. This 1 MA discharge was heated by 7 MW of NBI, and had a current flattop time of 0.8 s, which is approximately four current relaxation times. According to model calculations, the current profile remained approximately constant for the last 300 ms of the discharge (> 1 current relaxation time). The stored energy of the plasma reached 280 to 300 kJ with toroidal beta, β_T, at > 20 % for approximately 0.5 s, which is over ten energy confinement times. The normalized beta, $\beta_N = \beta_T/(I_p/aB_T)$, exceeded 5 % m·T/MA and the energy confinement time was 70% above the predicted L-mode value for the same duration. The line-averaged density exhibited only a modest increase after t=0.3 s, and was then held constant at 80% of the Greenwald limit by ELM activity, with no confinement degradation at these high densities.

In this and similar discharges, the loop voltage remained low (< 0.5 V) through the duration of the current and energy flattop, indicative of a significant amount of non-inductive driven current. Approximately 60% of the total current was driven non-inductively by NBI (10%) and bootstrap current (50%), as calculated by TRANSP. In order to develop this type of plasma discharge, progress and understanding was required in topical areas of research, as described below.

Macroscopic Plasma Behavior

Highlights - MHD studies benefited directly from the improved shaping capability, which allowed routine access to higher stability limits. The influence of plasma rotation on both internal and external mode stability was a focal point of experimental and theoretical research in FY '04. The deployment of the first elements of a new active coil set for error field control, ultimately aimed at resistive wall mode control, enabled a new class of MHD stability and control studies to begin. These studies have already yielded reductions in the density below which the onset of deleterious locked modes occurs. Improved diagnostics enabled the identification of RWM mode structure that provides the first validation of RWM theory at low aspect ratio. This research is the basis for ITPA joint experiments, and serves near-term needs for the design of the ITER RWM control system. The study of coaxial helicity injection and poloidal field induction took advantage of new facility capabilities in 2004 to advance research in solenoid-free operation.

The reduced response time of the plasma control system improved the feedback control for vertical stability, leading to routine operation at higher elongation, κ, triangularity, δ, and longer pulse lengths than achieved in previous years. Plasmas were developed with κ exceeding 2.6 at low internal inductance l_i (~0.5) and δ up to 0.8. The increase in κ represented an approximately ~20% increase over the previous maximum elongation, and a 30 to 40% increase over previous pulse lengths was obtained.

The higher elongation benefits most operational scenarios in NSTX. In particular, the stronger shaping allowed for higher plasma current at otherwise fixed conditions, leading to higher values of normalized current, I_p/aB_T, and hence higher values of β_T. Shown in Figure 2 is a plot of peak toroidal beta, β_T , the ratio of plasma pressure to the magnetic pressure of the toroidal field, as determined from the EFIT magnetic reconstruction code. The benefit of being able to achieve higher κ and thus higher I_p/aB_T is

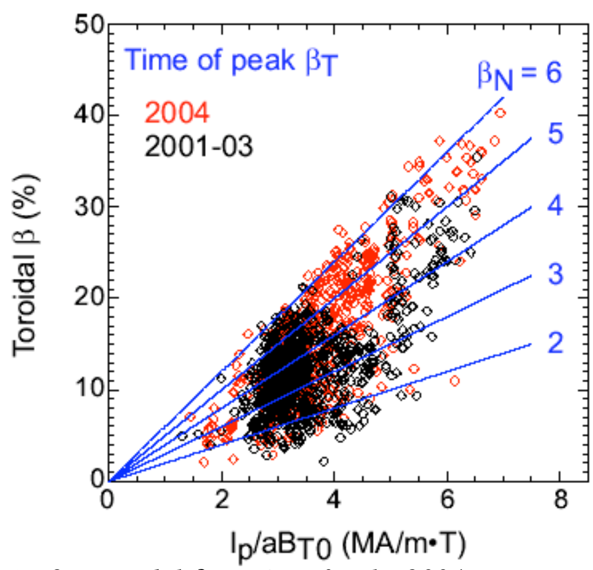

Fig. 2 Toroidal β vs I_p/aB_T for the 2004 experiments (red) and earlier years (black)

evidenced by significantly more high-β_T (>30%) shots during the 2004 experimental campaign than in previous years.

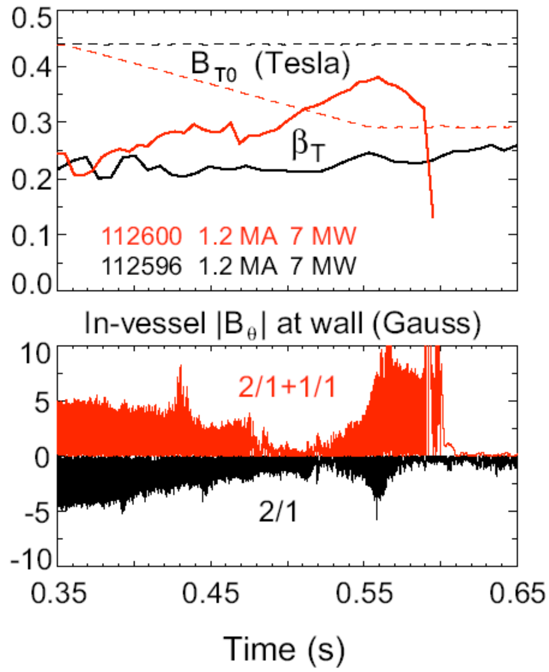

Fig. 3 Comparison of MHD activity for two NSTX discharges

The high-β_T discharges were limited by the growth of low-n internal modes. Aspects of this are shown in Figure 3. The red traces in the figure show the evolution of the toroidal field, β_T and the magnetic fluctuation amplitude in the high-β_T discharge. The black traces are taken from a similar discharge in which the toroidal field was held steady and, therefore, which had lower β_T (~20%). The magnetic fluctuations are an indication of a long-duration 2/1 MHD tearing mode in the mid-radius region in both discharges. An additional 1/1 mode became unstable in the high-β_T discharge at about 560 ms, as reflected by the increase in the amplitude of the magnetic fluctuations. As a result, the β_T started to decrease as the modes coupled and the rotation decreased (Fig. 4, top panel), The fall in plasma pressure was gradual at first, but this initial stage was followed by a total collapse as the rotation frequency decreased through 2 kHz. The plasma rotation remained high in the discharge with no TF ramp-down (Fig. 4, bottom panel) [1].

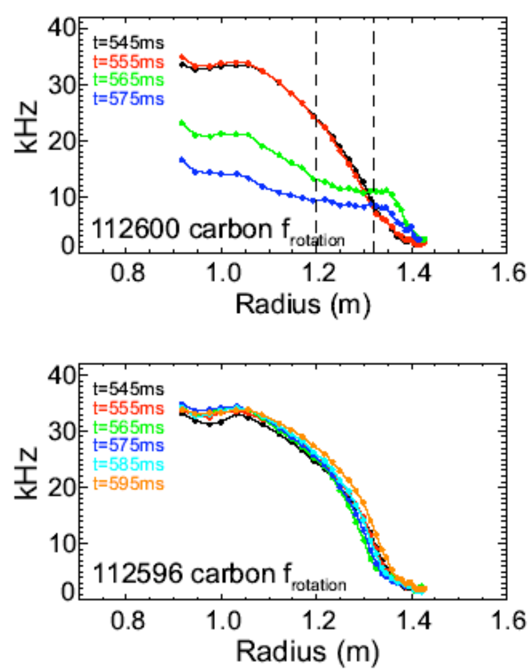

Fig. 4 Evolution of rotation profiles for the two discharges shown in Fig. 3

At low edge safety factor, q_{95}, Resistive Wall Modes (RWM) were prevalent and often were the β_T-limiting mechanism [2]. Magnetic sensors mounted close to the plasma on the passive stabilizers show nearly simultaneous growth of n=1 to 3 modes, consistent with the DCON stability code result which shows unstable n=1 to 3 RWM components. Visible light emission from the plasma during an RWM is shown in the left panel of Figure 5, and it is compared to the

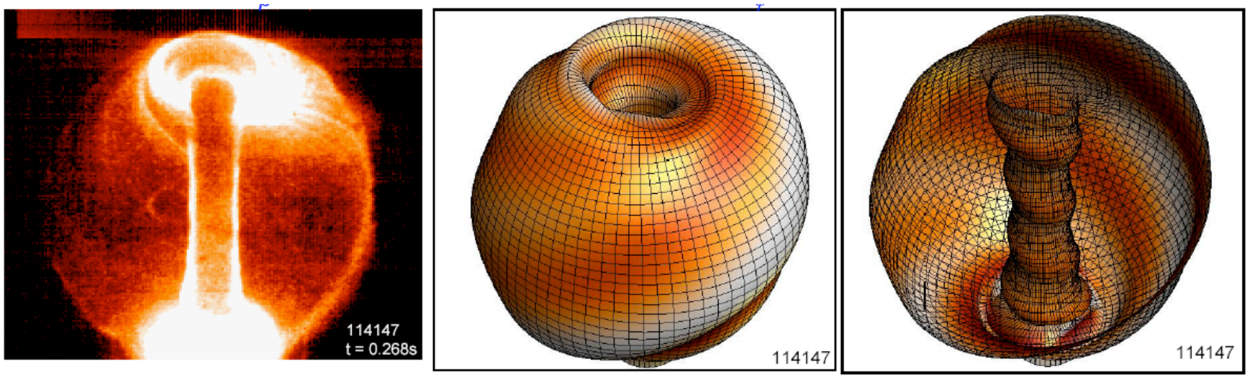

Figure 5. Fast camera image of NSTX plasma during an RWM (left) along with computed perturbed surfaces observed from outside (middle) and inside (right) the plasma

DCON computed perturbed magnetic field normal to the surface exterior and interior to the plasma in the middle and right panels of Figure 5. The computation uses an EFIT experimental equilibrium reconstruction and the illustration includes the sum of the n=1 to 3 components. The fast camera images (left panel) confirm the toroidal asymmetry and macroscopic scale of the mode.

In order to expand the NSTX operating space and allow further increases in β_T, it was essential to explore means by which performance-limiting MHD modes could be stabilized. This was done using the first of three pairs of error field (EF) compensation/RWM control coils. The other two pairs will be commissioned for the 2005 experimental campaign. The first active control coil pair was used to eliminate locked modes, which prevent achieving high performance, as well as to understand the effect of the applied radial magnetic fields on modes at higher density and β_T. With these coils, the low density threshold for avoiding locked modes was reduced from ~1.2 to ~0.6 $\times 10^{19}$ m^{-3}. The suppression of the locked mode at low density by use of this coil aids HHFW operation at low density. The error correction coils also allow the study of possible performance limiting modes at higher density and β_T. Experiments during the next campaign will focus on using the EF/RWM coil to suppress the low density locked modes and the RWM simultaneously.

Non-inductive operation will be essential for future STs because of space and neutron loading limitations. Several techniques of non-solenoidal plasma startup were explored on NSTX in 2005. In initial experiments to demonstrate startup using only coils outside the center column, plasmas were pre-ionized using HHFW and ECH in the outside region near the RF antenna. The outer PF coil currents were initially adjusted to establish a field null over a substantial portion of the plasma, and then they were ramped to produce a toroidal loop voltage of 5 to 15 Volts near the antenna. Currents up to 20 kA were produced, but the plasmas terminated on the center stack. The goal for future work using this technique is to control the radial position of the nascent plasma to confine it to the region where the loop voltage is high, and thus achieve higher current.

Another technique that was tested is transient Coaxial Helicity Injection (CHI) [3], in which a pulse of voltage lasting for only a few milliseconds is applied between the inner and outer vessel segments, causing plasma breakdown and generating a toroidal current which is propelled into the main chamber. The transient CHI technique has the benefit of reduced energy on the injection electrodes, since the injection current flows for only a short time. With this technique, plasma currents up to 140 kA with amplification factors (I_p/I_{CHI}) of up to 40 were achieved. This amplification is a factor of two greater than that obtained previously with longer duration CHI application. Ion and electron temperatures of up to 25 eV were measured, suggesting the possibility of closed flux surfaces. Future experiments will focus on maintaining plasma current beyond the duration of the injector current in order to couple the seed current to other current drive sources, both inductive and non-inductive.

Transport

Summary - Transport research focused on the characterization of H-mode confinement and scalings, detailed measurements of fluctuation characteristics in L-mode plasmas, and the study of internal transport barriers, particularly electron thermal barriers. The H-mode studies revealed evidence for a toroidal field dependence of confinement. Much of this research was carried out within the ITPA, in partnership with MAST. The turbulence studies revealed a scaling of turbulent eddy size that is consistent with ion-scale dynamics. In plasmas where a central reversal of the magnetic shear was indicated by measurements of MHD activity, significant reductions in electron thermal transport were observed. The study of electron thermal transport is of critical importance to ITER and will be a focus of NSTX research through FY '07.

H-mode operation in NSTX resulted in the highest performance plasmas, with stored energies reaching 400 kJ in 1 MA plasmas with ~7 MW of NB heating power. An experiment to study the L-H threshold power was conducted as part of an NSTX/MAST identity experiment. The threshold in NSTX was found to be low, P_{NBI}~350 kW, in balanced Double Null Divertor (DND) plasmas at 0.5 MA and 0.45 T, with the threshold increasing to between 1 and 2 MW in Lower Single Null Divertor (LSND) plasmas (with the ∇B drift towards the X-point), consistent with

MAST results for similar configurations and parameters [5]. Ohmic H-modes were often observed, most reproducibly at B_T>0.4 T, and these exhibited a gradual decrease in the edge rotational shear and E_r, as measured by the edge spectrometer, starting up to 30 ms before the drop in D_α that signified the L-H transition.

The confinement trends in NSTX were similar to those at conventional aspect ratio in some respects, but differed in others. Systematic scans of LSND H-mode plasmas at fixed power and B_T indicated a linear increase in both the global and thermal confinement time, τ_E, as a function of plasma current (0.6 to 1.2 MA). Figure 6 shows the results of this scan; the linear increase of total stored energy with plasma current is also seen in the electron stored energy, W_e, as measured by the Thomson scattering diagnostic. The electron density was seen to vary by approximately 30% over the range of currents, but the electron temperature remained almost constant. The "ears" on the density profile reflect the buildup of carbon at the edge during the early and mid H-mode phases.

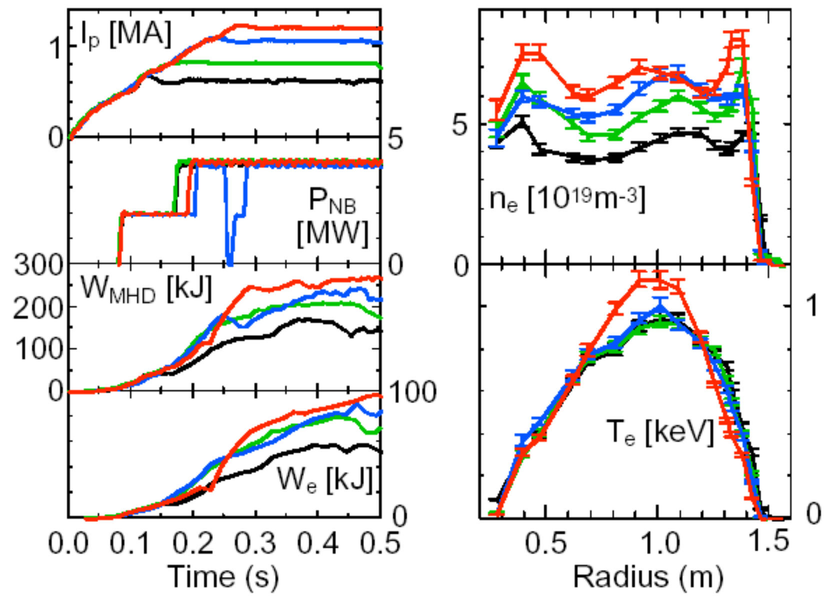

Fig. 6 Plasma and electron stored energy evolutions for discharges from a systematic current scaling experiment at fixed neutral beam power. Also shown are the electron temperature and density profiles for these discharges

Results from these systematic scans, as well as from other discharges with similar operating parameters, indicate that at fixed current and toroidal field, the global and thermal τ_E have a slightly weaker power degradation than at higher aspect ratio. Contrary to conventional aspect ratio, however, a dependence on the toroidal magnetic field was observed. This trend in the global and thermal confinement times is shown in Figure 7. The left panel shows the global τ_E normalized to the ITER97 L-mode scaling [4], and the right panel shows the thermal τ_E normalized to the H-mode ITER98pby,2 thermal τ_E scaling [5]. The figures show that the global τ_E values are enhanced over the L-mode value, with enhancement factors of close to 2.8 at the highest B_T for both L- and H-mode plasmas. The thermal confinement enhancement factors are more modest, reaching 1.4 at the highest B_T for H-mode plasmas. A reduction in confinement enhancement at the lowest B_T (<0.3 T) is seen for both the global and thermal values. The potential impact of increased MHD activity at the lower fields degrading confinement is being investigated.

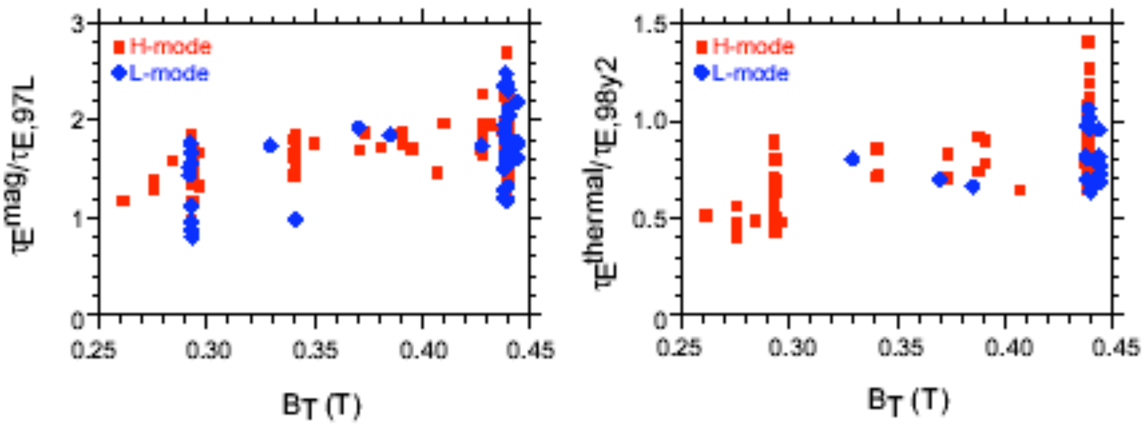

Fig. 7 Variations with toroidal field of the global energy confinement time normalized to the 97L L-mode scaling (left) and thermal confinement time normalized to the 98pby,2 scaling (right).

There is evidence that at least some of this confinement dependence on B_T may be due to changes in core turbulence properties. Insight in this was gained through turbulence measurements using fixed-frequency (30, 42 and 49 GHz) quadrature and swept-frequency (26 to 40 GHz) homodyne correlation reflectometry systems. For the first time in an ST, quantitative long-wavelength turbulence measurements have been made in the core (r/a=0.2 to 0.7) of beam-heated L-mode plasma discharges. Correlation reflectometry data indicate radial correlation lengths (L_c) ranging from 2 to 25 cm with significantly smaller values observed in the outer plasma (r/a~0.65). The correlation lengths measured in the outer plasma at r/a=0.7 (where

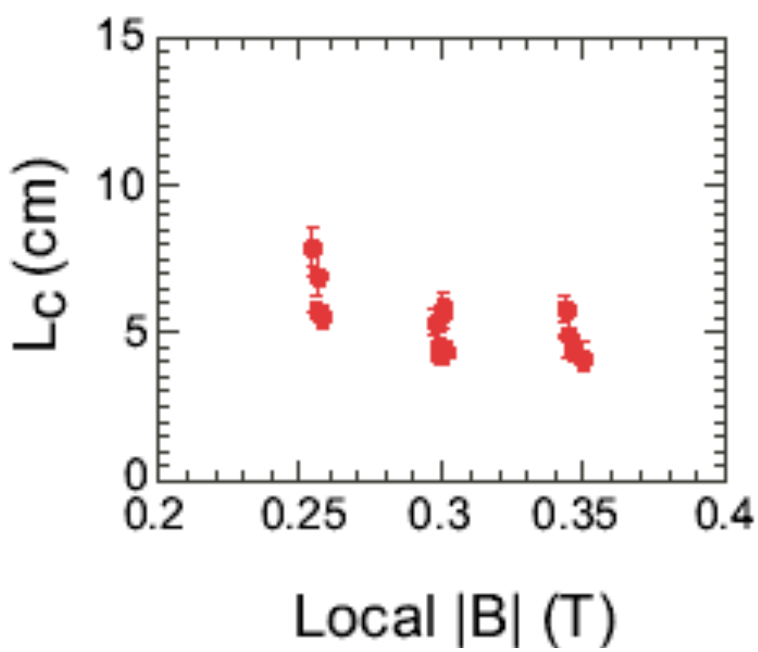

Fig. 8 Reflectometer fluctuation radial correlation length as a function of local magnetic field

Ion Temperature Gradient turbulence is predicted to exist in these cases) are illustrated in the top panel of Figure 8 as a function of local |B| during a fixed edge q scan. Correlation lengths are observed to increase with decreasing field, reaching values of approximately 8 cm at the lowest field. Reflectometer measurements taken for a similar set of discharges show a reduction in the measured reflectometer phase fluctuation level, and associated reduction in the density fluctuation levels, as the magnetic field is increased.

The determination of the transport properties of NSTX plasmas by TRANSP has benefited greatly from the increased number of spatial points of the CHERS diagnostic. The calculations indicate that the electron channel dominates the transport loss in most H-modes ($\chi_e \sim 10 - 20$ m^2/s), with the ion thermal diffusivity near or above the NCLASS [6] neoclassical value in many cases (χ_i~1 to 5 m^2/s). However, NCLASS neoclassical values do not take into account possible enhancements to the neoclassical diffusivity by up to a factor of 2 due to finite orbit effects [7]. In the L-mode, $\chi_i \sim \chi_e$ (1 to 10 m^2/s) for line-averaged densities <4 × 10^{19} m^{-3}, but $\chi_I < \chi_e$ for higher densities.

The local transport properties of NSTX plasmas appeared to be sensitive to variations in magnetic shear, as is seen in Figure 9 which compares electron and ion temperature profiles between two low density ($n_{e0} \sim 2 \times 10^{19}$ m^{-3}) L-mode discharges with different q-profiles [8], produced by changing the current ramp rate and the NBI timing. In the discharge with a fast I_p ramp and early NBI, the T_i and T_e exhibited much stronger gradients than in a discharge with a slower I_p ramp and later NBI, signifying the formation of an internal transport barrier. The q-

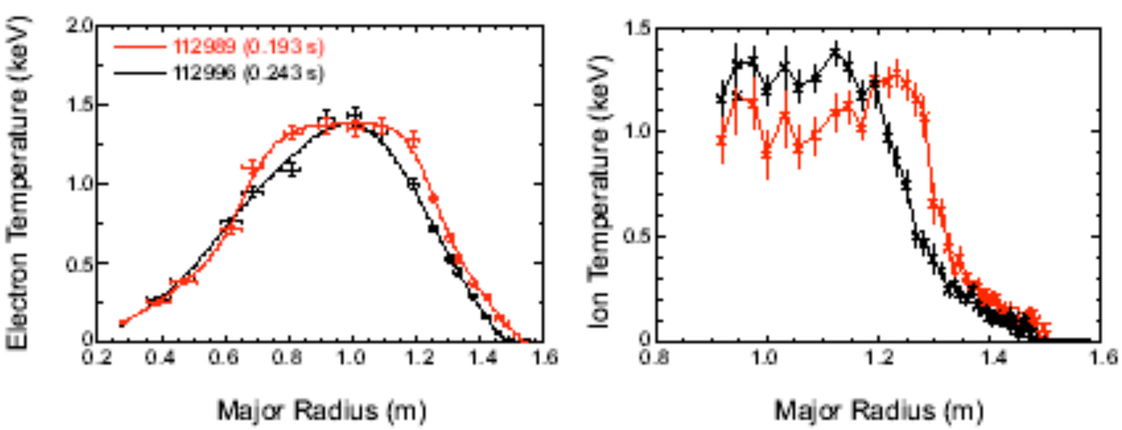

Fig. 9 Electron and ion temperature profiles for two comparison discharges at times of comparable density and rotation

profiles for these two discharges, as calculated by TRANSP are shown in the top panel of Figure

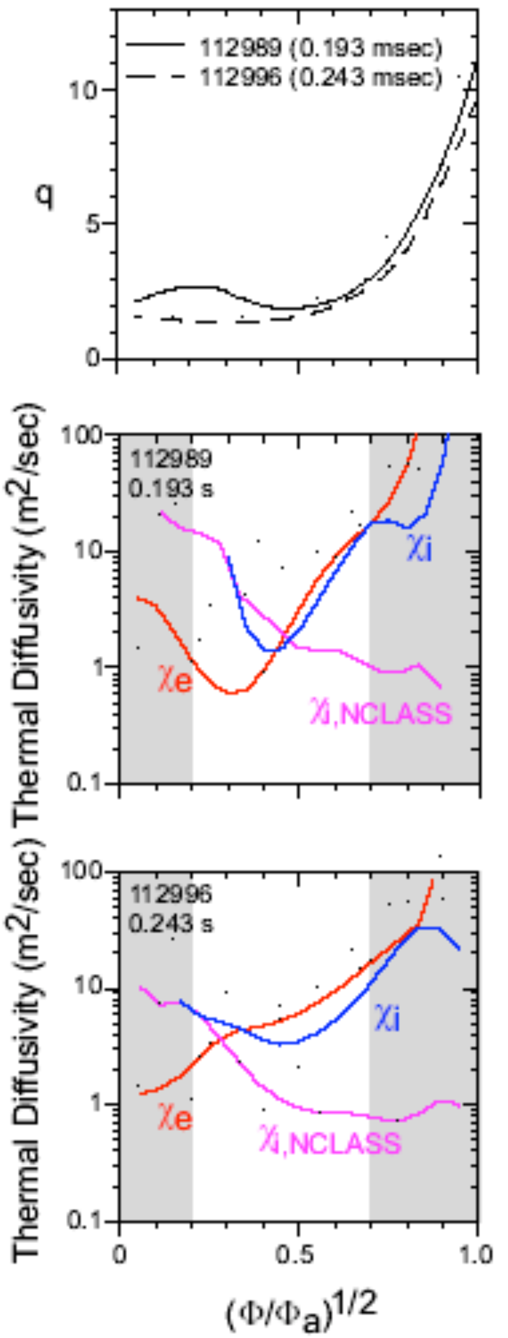

Fig. 10 q-profiles and thermal diffusivities calculated by TRANSP for the discharges of Fig. 9. The thermal diffusivities have the least uncertainty in the unshaded regions.

10. The modeling for the slow ramp/late NBI discharge (112996) shows a monotonic q-profile, while that for the fast ramp/early NBI discharge (112989) exhibits a magnetic shear reversal from r/a=0.2 to 0.5. The effects of the possible reversed shear are seen in the bottom panels of Figure 10, which show a reduction by a factor of 3 to 7 in the thermal diffusivities of both the electrons and ions in the region of reversed shear. Outside this region, the ion and electron thermal diffusivities are comparable. Because of uncertainties in T_e, T_i and their gradients, the χs are highly uncertain in the shaded region, r/a < 0.2 and > 0.7.

Reflectometer measurements indicated both longer turbulence correlation lengths and higher density fluctuation levels in the discharge with monotonic shear than in the one with reversed shear. GS2 gyrokinetic calculations indicate linear growth rates for microinstabilities near r/a=0.45 which are significantly higher in the monotonic than in the reversed shear case. Non-linear gyrokinetic calculations are underway to confirm these results and to study the stabilizing effect of sheared rotation.

Waves and Energetic Particles

Highlights - Studies of High Harmonic Fast Waves (HHFW) revealed a strong dependence of power absorption on wavenumber. A candidate for some of this absorption is parametric decay of the fast wave into an ion Bernstein wave (IBW) near the plasma edge. Evidence for this was found in measurements of the edge impurity temperatures and flow velocities, as well as RF probe measurements. Electron Bernstein Wave (EBW) emission studies confirmed wave propagation theory of EBWs. These results, together with continuing collaborative research with MAST, form the basis for the development of an off-axis current drive method for NSTX and the ST in general. Fast-ion MHD studies, developed jointly with DIII-D under the ITPA, were also performed and serve as benchmarks for codes being used to predict fast-ion MHD properties on ITER.

The 30 MHz High Harmonic Fast Wave system provides the potential for heating electrons selectively to reduce ohmic flux consumption and for providing non-inductive current drive directly. The twelve-strap HHFW antenna has the capability to launch waves over a range of toroidal wavenumbers ($k_{||}$ = 3 to 14 m^{-1}) in both the co- and counter- directions. Significant electron heating has been observed in low density deuterium and helium plasmas. However, the actual power absorption of the electrons was found to depend sensitively on the spectrum of launched waves, with greater absorption at higher $k_{||}$. Electron heating profiles are consistent with model calculations which predict broader heating profiles for higher $k_{||}$, but the increment in electron stored energy is less than what would be expected for pure electron heating.

Heating of the edge thermal ions during HHFW was measured by the Edge Rotation Diagnostic. This heating is a possible explanation for the apparent deficit in electron heating. This edge measurement indicates that the edge ions can be described as a two-temperature component plasma, with a significant hot component whose temperature scaled with the HHFW power, and which could reach 0.6 keV. The edge ion heating was associated with parametric decay of the launched HHFW wave as measured by an RF probe. A frequency spectrum of the probe signal is shown in Figure 11; the fundamental wave at 30 MHz is seen along with sidebands separated by $f_{c,D}$, indicative of a decay into an Ion Bernstein Wave (IBW) wave. More IBW sidebands are observed with increasing P_{HHFW}. This conversion of the HHFW to the IBW may be part of the explanation for the reduced HHFW power absorption for the fastest wave, $k_{||}$=3 m^{-1}, where the driven current is predicted by theory to be maximal. It is also noted that a significant amount of

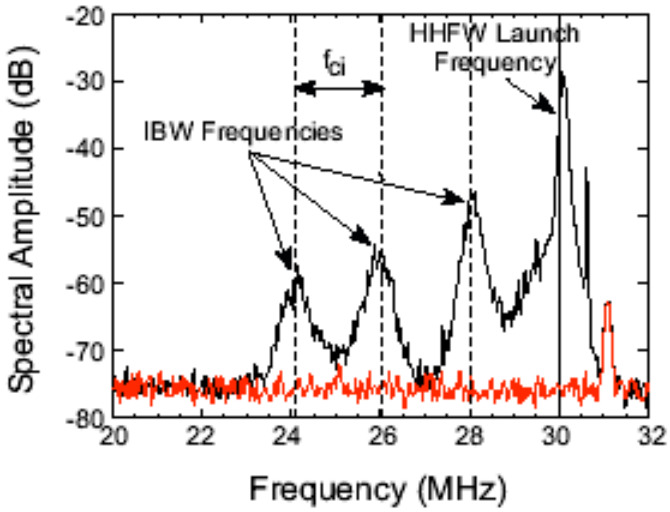

Fig. 11 Evidence for parametric decay of HHFW into IBW waves as measured by an RF probe

HHFW power could be absorbed by fast ions in experiments combining HHFW and NBI heating.

Another promising technique for generating off-axis current in NSTX is the launching of the electron Bernstein wave (EBW). For this, an O-mode wave is launched into the plasma, and becomes converted to an EBW, which then heats the electrons locally at the cyclotron layer in the perpendicular direction [9]. The key to making this a viable technique is to have a >80% conversion efficiency from O-mode to the EBW. Assessments of EBW emission and estimates of mode conversion efficiency in NSTX support this requirement, and plans for developing a high power EBW system are underway.

NSTX provides a unique test bed for fast ion MHD theory, and understanding the underlying physics of this is particularly relevant to ITER. The super-Alfvénic 80 kV neutral beam ions have similar dimensionless parameters to 3.5 MeV alpha particles from D–T fusion reactions in ITER, as seen in Fig. 12. In NSTX, neutral beam heated plasmas typically exhibit a broad spectrum of instabilities excited through a resonant interaction with fast ions, from compressional and global Alfvén waves (CAE and GAE) at frequencies $0.3 < \omega/\omega_{ci} < 1$, to toroidal Alfvén eigenmodes (TAE) at frequencies ~100 kHz. While there was no observed degradation in performance correlated with the appearance of CAE activity, enhanced fast ion losses were correlated with both the TAE-like and fishbone-like modes; this could be relevant to ITER in the super-Alfvénic regime [10].

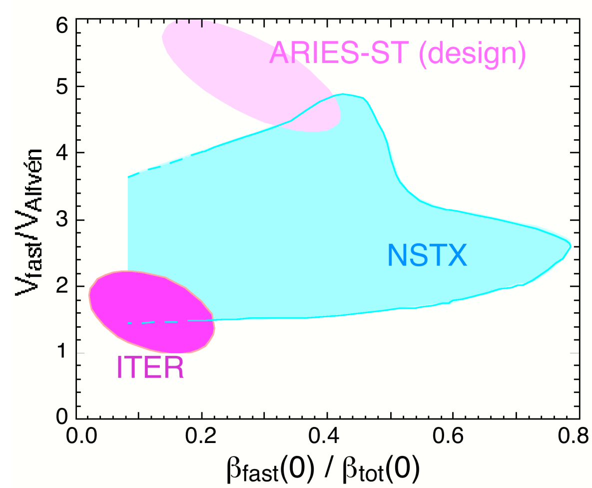

Figure 12. The wide operating space of NSTX with respect to two parameters of critical importance to ITER and any burning plasma, the fast ion pressure and the fast ion velocity compared to the Alfvén velocity.

Plasma-Boundary Interface

Highlights - The compact ST geometry and neutral beam heating combine to yield heat fluxes on the divertor of NSTX that rival those expected for ITER. Edge stability and ELM characteristics are of critical importance to ITER, as the edge pressure properties will be a determining factor in core confinement and fusion performance. Also, the ELMs must be controlled on ITER to reduce potentially damaging impulses of heat on the plasma facing components. Measurements of plasma heat fluxes and their profiles in the lower divertor of NSTX have yielded good power accountability. New fast imaging capability has provided detailed measurements of the structure of Edge Localized Modes (ELMs) that allow detailed comparison with edge stability theory. The first stage of ITPA joint experiments involving NSTX, DIII-D, and MAST were performed, aimed at validating edge stability models developed for moderate aspect ratio experiments. Studies of the edge turbulence continued to mature in 2004 with fast camera imaging of the turbulence structure in the scrape-off layer. This work continues to play a leading role in joint turbulence imaging research sponsored by the ITPA.

The exploration of improved particle control and plasma fueling benefited from the implementation of several new techniques and capabilities. Boronization during 350 °C bakeout, the deposition of 1 to 2 g of boron prior to a run day, as needed, and interspersing plasma and helium conditioning discharges all helped to maintain good wall conditions and led to better density control. Initial experiments were successfully performed using a Li pellet injector and a supersonic gas injector for localized and efficient fueling. The use of these capabilities and techniques will be expanded in future operation.

Power accountability in both LSN and DND plasmas was found to be good, with up to 70% and 90% of the power accounted for in the two configurations respectively[11]. The largest fraction of the power loss (35%) was deposited on the divertor plates, with an out-in ratio of up to 5:1. Because of the compact nature of the ST, it is important to reduce the power to the material surfaces. Inner divertor detachment was found to reduce the power loading of the inner divertor plates to below 1 MW/m^2 [12]. Inner divertor detachment was observed in both L- and H-mode NBI-heated plasmas at densities $>2 \times 10^{19}$ m^{-3}. The outer divertor in all experiments remained attached, with heat fluxes up to 10 MW/m^2.

A variety of ELMS, which can cause increased divertor power loading, was observed in H-mode plasmas. An apparent new type of ELM, Type V, was identified [13]. This ELM is small in amplitude with minimal energy loss and minimal resulting power loading. It occurs when the normalized electron collisionality frequency, $\nu_e^*>1$, where ν_e^* is evaluated at the top of the pedestal[14]. At lower ν_e^*, this small ELM was interspersed between large Type I ELMs. Type I ELMs were found to decrease in size with increasing density[15], and they often exhibited low-n and intermediate-n external kink-like structures on the fast camera images, while structures associated with Type V ELMs were more localized, lown, flux-tube type perturbations (Fig. 13). The effect of ELMs on the plasma stored energy was found to depend sensitively on plasma elongation.

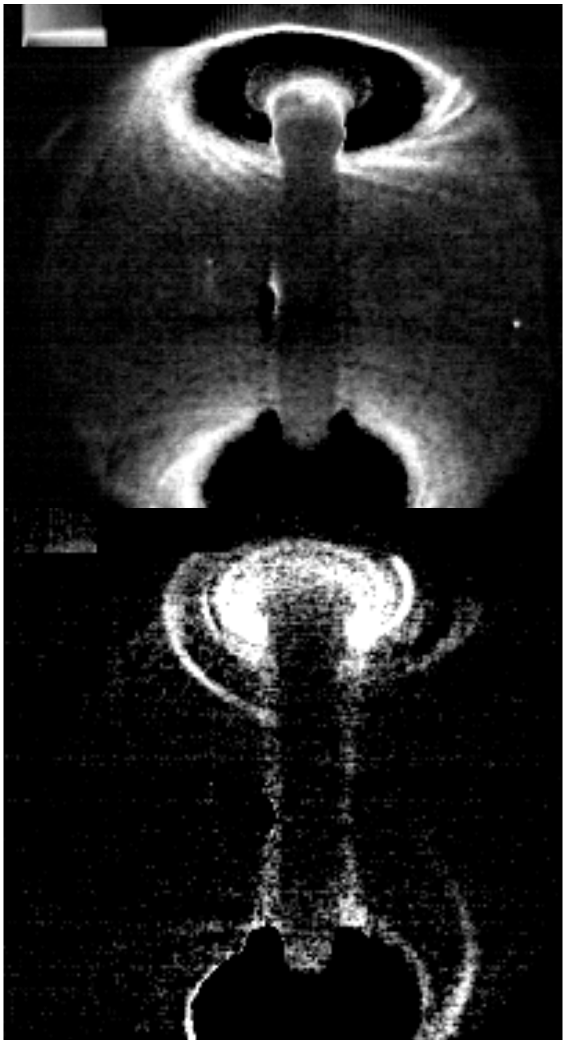

Fig. 13 Contrast-enhanced, fisheye lens images of unfiltered light during a large, Type I ELM (top), and a small, Type V ELM (bottom)

The 2-D structure of edge plasma turbulence was measured by viewing with an ultra-high speed CCD camera the emission of D_α or helium spectral lines locally enhanced by gas puffing [16]. Transitions from L-mode to H-mode could appear as a continuous evolution from a turbulent "blob-like" or intermittent state to a quiescent state over 0.1 ms, apparently without any new spatial features or flows. Transitions from H-mode to L-mode appeared as high-n poloidal perturbations which evolved into radially moving blobs. ELMs normally were associated with an increase in blob-like activity, although sometimes ELM-free H-modes had intermittent blob-like turbulence.

REFERENCES

[1] MENARD, J., et al., 20th IAEA Conference, Vilamoura, Portugal, 2004, Paper EX/P2-26.

[2] SABBAGH, S.A., et al., 20th IAEA Conference, Vilamoura, Portugal, 2004, Paper EX/3-2.

[3] RAMAN, R., et al., Phys. Plasmas **11** (2004) 2565.

[4] KAYE, S.M., et al., Nucl. Fusion **37** (1997) 1303.

[5] THE ITER TEAM, Nucl. Fusion **39** (1999) 2137.

[6] HOULBERG, W. et al., Phys. Plasmas **4** (1997) 3230.

[7] GATES, D.A. et al., Phys. Plasmas **11** (2004) L45.

[8] STUTMAN, D., et al. 20th IAEA Conference, Vilamoura, Portugal, 2004, Paper EX/P2-8.

[9] TAYLOR, G., et al., Phys. Plasmas **11** (2004) 4733.

[10] FREDRICKSON, E., et al, 20th IAEA Conference, Vilamoura, Portugal, 2004, Paper EX/5-3.

[11] Paul, S.. et. al., J. Nucl. Mater. at press 2/2005.

[12] SOUKHANOVSKII, V., et al., J. Nucl. Mater. At press2/2005.

[13] Maingi, R. et al., "Observation of a High Performance with Small ELMs in NSTX", Nucl. Fusion at press, 2/05.

[14] MAINGI, R. et al., 20th IAEA Conference, Vilamoura, Portugal, 2004, Paper EX/2-2.

[15] Maingi, R., et. al., J. Nucl. Mater. at press 2/2005.

[16] ZWEBEN, S.J., et al., Nucl. Fusion **44** (2004) 1134.

Facility and Diagnostic Upgrade Accomplishments in FY 2004

The 2004 fiscal year began during the latter half of the outage period to repair the NSTX toroidal field coil. We were able to make use of this outage to accelerate the installation or modification of some facility systems and diagnostics which then enhanced the research performed in the 2004 experiments. After operation of NSTX resumed in January, 21.1 weeks of experimental operation were completed in August 2004, thereby exceeding the NSTX Facility Milestone F(04-1) of 18 weeks of operation and the programmatic goal of 20 weeks. A total of 52 NSTX Experimental Proposals and Machine Proposals received run time in FY 2004.

During FY 2004, 2809 shots were run with current in the toroidal field coil, including the daily test shots. Of these, 2164 were shots which reached a plasma current above 0.2 MA. About 370 plasma shots were run in experiments aimed towards the NSTX Research Milestone R(04-4) to investigate alternative, non-solenoid plasma startup techniques, including coaxial helicity injection; these shots were performed at lower plasma currents.

Both the auxiliary heating systems for NSTX, namely the neutral beam injection (NBI) system and the 30 MHz RF High-Harmonic Fast-Wave (HHFW) system, maintained reliable operation in FY 2004. The NBI system delivered power in 1524 shots, and the HHFW system in 474 shots; both systems injected in 214 shots. The peak NBI power was 7.4 MW (with all sources operating at acceleration voltages of 100 kV) and the peak HHFW power was 4.2 MW (with the antenna elements operating at RF voltages of 11 – 13 kV). Real-time control of the phasing of the HHFW antenna was used in experiments on RF current drive. Experiments with high-power HHFW heating also benefited from improvements made during the run to reduce spurious RF pickup on diagnostic systems, particularly the magnetic sensors used in real-time plasma control.

The new center bundle of the NSTX toroidal field (TF) coil was commissioned at the start of the FY 2004 run for operation at a nominal toroidal field of 0.45 T (at R = 0.85 m). The resistance of the all the joints between the conductors in the center bundle and their respective “flags” were monitored on every shot throughout the run. All the joint resistances remained below the goal established during the redesign of the TF coil and the associated internal and external reviews. However, during the first four months of operation, when about 75% of all shots were run at a nominal field of 0.45 T, a trend became apparent in the measured joint resistances suggesting that some flexing and “lift-off” of the contact surface of the joints was occurring at the highest operating level. As a results, decisions were taken, first, to restrict the TF to a maximum nominal field of 0.3 T for the last month of operation, and then, to disassemble the TF coil during the subsequent outage to investigate and remediate the cause of the unexpected behavior of the joints.

At the end of the FY 2004 experiments, NSTX began an outage which is expected to last through March 2005. Tasks to be undertaken during this outage include the removal and repair of the TF coil, mentioned above, the installation of new PF1A poloidal field coils in the center-stack, the replacement of the bellows in the neutral beam injection duct, the installation of new diagnostics, and modifications to the systems to inject gas and to provide microwave preionization power for CHI experiments.

The replacement of the upper and lower PF1A coils, which was recommended in the NSTX Five-Year Plan developed in 2003, is designed to provide greater flexibility in shaping the cross-section of the plasma boundary, particularly in achieving equilibria with both high elongation, $\kappa \approx 2.3$, and high triangularity, $\delta \approx 0.8$, which have been shown theoretically to combine good stability and a high fraction of the bootstrap current. This upgrade was originally planned for FY 2005 but was advanced by one year to take advantage of the removal of the TF center bundle for repair. The decision to accelerate the PF1A coil upgrade has, however, forced the postponement by a year of the installation of the charge-exchange recombination spectroscopy diagnostic for measuring the plasma poloidal rotation velocity, the PCHERS system, and its associated preparatory Diagnostic Milestone D(04-3).

The major diagnostic to be installed for initial operation in the FY 2005 experiments is the high-k microwave scattering system. This diagnostic is designed to measure in the plasma gradient region, normalized radius $\rho = 0.4 – 0.6$, the spectrum in frequency and radial wavenumber of the turbulent density fluctuations with wavenumbers in the range $k_r = 5 – 20\ cm^{-1}$ where fluctuations associated with electron temperature gradient (ETG) instabilities are predicted to occur. Preparations for installing this diagnostic, including commissioning and testing of the 1 mm microwave source and design of the launching and receiving microwave optics, were made in FY 2004, thereby meeting the NSTX Diagnostic Milestone D(04-4).

Progress in Facility Capabilities and Operation

Resonant Field Correction Coils

A new capability introduced during FY 2004 was the first of three pairs of coils to produce controllable radial magnetic field perturbations in NSTX. The individual coils are nearly rectangular “picture frame” groups of two turns each, centered on the mid-plane and mounted just outside the vacuum vessel wall conforming to its locally cylindrical shape. The pairs are mounted diametrically opposite each other. The first pair installed was chosen to be in the orientation that would allow correction of the intrinsic radial error field in NSTX, inferred from measurements of the “locking” position of MHD perturbations occurring during the current ramp-up. The two coils were connected in series to produce a radial field perturbation with a predominant toroidal mode-number $n = 1$ and powered by a single phase-controlled rectifier supply with a current waveform programmed in the plasma control system prior to a shot. When the coil current was programmed appropriately in magnitude and sign, the tendency to develop locked MHD modes during the current ramp-up at low density was significantly reduced. During the subsequent flattop in the plasma current, however, it appeared that the intrinsic error field changed and, when the plasma pressure was sufficiently high, another phenomenon was observed, the amplification by a factor 2 – 3 of the component of the applied radial field resonant with a non-rotating n = 1 unstable mode in the plasma. This demonstration of the effect of the initial coils, together with the preparations made to complete the installation of the full system during the subsequent outage, met the NSTX Facility Milestone F(04-2) and contributed to achieving the NSTX Research Milestone FY(04-1).

Commissioning the Capacitor Bank for CHI Experiments

The NSTX Facility Milestone F(04-3) was achieved in July when a capacitor bank was successfully used in experiments on Coaxial Helicity Injection (CHI) in NSTX. CHI involves injecting a poloidal plasma current between the inner and outer divertor lower plates, which, under the influence of applied toroidal and poloidal magnetic fields, creates a toroidal plasma current. Although this toroidal current initially flows on open field lines connecting the divertor plates, it can become transferred onto closed field lines forming flux surfaces by a process of magnetic reconnection. In experiments on the HIT-II device at the University of Washington, it had been demonstrated that by injecting a very brief (a few milliseconds) current pulse, an axisymmetric magnetic reconnection could be obtained which produced a high-quality plasma carrying significant toroidal current. This technique is known as transient CHI. However, when this process was first tested in NSTX in 2003, it became apparent that the external inductance of the rectifier power supply used was too large to permit transient CHI. Since the external inductance was a necessary part of the circuit to guard against excessive current and energy dissipation from the power supply in the event of a fault, it was concluded that a small capacitor bank with limited energy storage was more appropriate for transient CHI experiments in NSTX. The bank which was constructed has a capacitance that can be varied from 5 to 50 mF (using 1 to 10 capacitors), a maximum voltage of 2 kV (energy storage capacity up to 100 kJ) and is designed to deliver current pulses up to about 50 kA through an ignitron switch accurately timed by the NSTX clock system.

In the first experiments using the CHI capacitor bank, from 1 to 7 capacitors were used at a maximum charging voltage of 1.0 kV. Over 100 CHI discharges were successfully produced with injected currents above 1 kA. The maximum injected current exceeded 20 kA and toroidal plasma currents up to 150 kA were produced. The multiplication factor (the ratio of the toroidal plasma current produced to the poloidal current injected) reached 46, much higher than previously achieved. Scans were performed of the gas pressure and the currents in the poloidal field coils that produce the initial flux connecting the CHI electrodes. Reliable breakdown of the prefill deuterium gas was obtained at lower pressures than previously. No problems were encountered with spurious external electrical breakdown outside the vacuum vessel which had occurred in some earlier CHI experiments. Some arcs did occur across the insulator inside the upper part of the vacuum vessel, the so-called absorber insulator, but unlike previous such occurrences, these arcs appeared to extinguish spontaneously and did not terminate the main CHI discharge. The CHI plasmas were sufficiently reliable that it was possible to obtain measurements of the electron temperature and density in the CHI plasma with the Thomson scattering diagnostic for the first time in NSTX. While these initial experiments were successful in many respects, an unambiguous demonstration of CHI-produced toroidal current flowing on closed flux surfaces by observing the persistence of the toroidal current beyond the end of the injected current pulse was not obtained. However, several promising methods for achieving this goal for CHI have become apparent from the analysis of the results and these will be pursued in future experiments.

Real-Time Data Acquisition and Control

A significant upgrade was made to the real-time data acquisition used by the NSTX Plasma Control System (PCS). The major thrust of this was to reduce the latency of the system in processing data from diagnostic sensors into commands for the power supplies which drive the coil systems for NSTX. Measurements made on the original system showed a propagation latency of about 3 ms average and 4.5 ms peak. Such delays were comparable to the growth times for the axial instability in plasmas with elongated cross-sections and thus limited the maximum elongation achievable. After optimization of the hardware configuration and settings and the PCS software, this latency was reduced to about 0.7ms average and 1.0ms peak, about a factor of four improvement. The improvements made to the real-time data processing system included the installation of a new parallelized data link to the power supply system designed and built at the laboratory.

This decrease in the propagation latency through the PCS did produce the desired result in FY2004. Plasmas with elongation up to 2.6 transiently and 2.5 sustained were produced at high plasma current (above 0.9 MA) and for low values of the internal inductance. Such elongated plasmas were a focus of the experiments because theory and, indeed, previous NSTX research, had shown that higher elongation could allow higher values of the normalized plasma current for a fixed value of the MHD safety factor q at the plasma edge, and that this, in turn, would permit higher toroidal beta, a main goal of the NSTX research program. The highest elongation plasmas were obtained at low values of the plasma internal inductance, which is indicative of a broad profile of the plasma current, also desirable for the ST plasma to make optimum use of the intrinsic self-generated bootstrap current.

In addition to the enhanced capability of the PCS to produce plasmas with high elongation, the control of the plasma equilibrium using the real-time equilibrium analysis code rtEFIT was improved during FY 2004. Standard plasmas have now been produced in symmetric double-null, lower single-null and upper single null divertor configurations, as well as inner-wall limited plasmas. Experiments with High Harmonic Fast Wave (HHFW) heating benefited from the good control provided by the rtEFIT algorithm for the gap between the plasma boundary and the RF antenna. The capability of the rtEFIT algorithm to control explicitly the separation between the magnetic separatrix surfaces connected to the upper and lower divertor X-points, was used in another experiment investigating the effect of the magnetic geometry on the plasma flow in the outer region and the scrape-off region.

The PCS was also upgraded during FY2004 to include control of the PF4 poloidal-field coils. This capability was then used in experiments performed in August 2004 to investigate techniques for starting up the NSTX plasma without relying on the central solenoid, an important element of the NSTX research program. These and other experiments on non-solenoid startup benefited from the capability also introduced in FY2004 to use the high-power 30 MHz RF system to produce a preionized plasma in front of the RF antenna in the region where the conditions for initiating the discharge were optimal.

Improving the Process of Boronization

Wall conditioning, particularly “boronization”, that is coating the plasma facing surfaces with a boron-rich film, has played an important role in achieving good plasma performance in NSTX. The coating, which is produced by running a glow discharge from anodes mounted off the vessel walls while the vessel is filled with a flowing mixture of deuterated trimethyl boron (TMB) [chemical formula $B(CD_3)_3$] and helium, has previously been applied with the whole vacuum vessel at room temperature. In the preparation for plasma operation following a brief vent of the vacuum vessel in March, boronization was applied for the first time to the hot plasma facing surfaces during the vacuum system bakeout[H.W. Kugel, et. al., J. Nucl. Mater. At press, 2/2005]. This hot-wall boronization process appeared to be very successful in rapidly restoring good plasma operation: the second ohmically heated plasma exhibited an H-mode transition and a plasma current above 1 MA was achieved within 45 plasma discharges. Additional boronizations with the walls at ambient temperature were conducted as needed during the run, at intervals from 2 days to 2 weeks, when spectroscopic measurements and other aspects of plasma performance indicated that the boron film was losing its effectiveness in controlling impurity influx into the plasma.

The standard boronization process involved the introduction of about 10 g of trimethyl boron and required several hours to complete, including the subsequent period of glow discharge cleaning in pure helium (HeGDC) to remove adsorbed deuterium. Later in the run, a process which became known as “morning mini-boronization” was developed which reduced the amount of TMB applied to 1 – 2 g and also shortened the subsequent HeGDC, thereby reducing the total time for the process to about an hour. When applied every few days, the “morning mini-boronization” was equally effective as the full-fledged boronization and became the preferred method of boronization in the last two months of the 2004 run. The effect of applying an even smaller amount of boron between successive plasma shots was also investigated, but it was found to be no more effective than the morning mini-boronization and the time required reduced the overall number of plasma shots which could be produced in a day of operation.

Lithium Pellet Injector and Advanced Fuelling Techniques

During the outage in 2003, a new solid-pellet injector had been designed, built and tested on the bench. This injector was mounted on NSTX during the 2004 run and after off-line testing was used in experiments for the first time in July. In this injector, the pellets are launched from re-usable “sabots” accelerated to velocities variable from 10 to 200 m/s by a high-pressure deuterium gas pulse. The injector has a remotely controlled rotatable turret with 400 barrels and is capable of injecting up to 8 pellets per discharge, each with a mass of up to about 5 mg, of any room temperature solid.

In the initial experiments, 17 lithium pellets, each with a mass of 2 mg, were injected at a velocity of about 100 m/s into a sequence of 16 discharges in different divertor configurations, with and without neutral beam injection (NBI) heating. Images were obtained with a fast TV camera of the neutral lithium (Li-I) line emission from the ablated cloud surrounding pellet on its way through the plasma. In ohmically heated discharges, the pellets were able to pass through the plasma with the remnants ablating in the scrape-off layer along the center column. When the pellets were injected during NBI, the pellets were ablated in the outer scrape-off layer. By varying the time between the end of a NBI pulse and the pellet injection, the depth of penetration of the pellet into the plasma could be controlled. Poloidally localized plumes of Li-I emission were observed, apparently following field lines from the pellet towards the divertor plates. In plasmas connected to the lower magnetic X-point, the Li-I emission appeared preferentially towards the upper divertor. The emission occurred in both divertor regions when the plasma was in an up-down symmetric double-null divertor configuration. Some Li-I line emission was observed in subsequent plasmas without pellet injection. There was a reduction in the emission from intrinsic oxygen impurities in plasmas when the pellets fully penetrated the plasma and deposited some lithium on the surface of the center stack.

The capabilities for gas fueling the NSTX plasma were upgraded in FY 2004. First, an independent control was introduced for the “upper shoulder” injector, allowing a direct comparison to be made in successive discharges of the fueling efficiency and the effect on the H-mode transition for gas introduced from the center stack at the mid-plane and at the shoulder region about 1.1 m above the mid-plane. H-mode transitions occurred with both injectors but with the shoulder injector, more gas fueling was required particularly as the discharge was changed from a symmetric double-null to a lower single-null divertor configuration. This is believed to be because there is additional “leakage” of the gas injected nearer the open divertor channel to the outboard side. Camera images of the deuterium line emission revealed this flow of gas past the upper divertor. Interestingly, however, for similar H-mode plasmas with the same density profiles, a slightly higher central temperature was measured and there were fewer edge-localized modes (ELMs) with fueling from the shoulder injector.

Second, towards the end of the experimental campaign, an injector was installed which was designed to introduce a supersonic stream of gas into the outboard edge of the plasma. The Laval nozzle is capable of injecting a jet of deuterium at a flow-rate up to 50 Torr.l/s, with a divergence as low as 6°, at a velocity of 2.4 km/s (compared with a typical thermal molecular velocity of about 1 km/s). It was developed in collaboration with the Department of Mechanical and Aerospace Engineering at Princeton University. The nozzle and its associated control valve inside a graphite thermal shield were inserted radially on a movable shaft about 0.2 m above the midplane and to about 0.1 m from the outer boundary of the plasma. High-speed camera images of the region confirmed that a collimated stream of deuterium entered the plasma boundary from the nozzle, producing local cooling of the plasma which is expected to improve the gas penetration further. Preliminary indications are that the fueling efficiency of the nozzle was 0.3 – 0.4, which is much higher than achieved with conventional gas injection in NSTX.

Progress in Diagnostic Capabilities

Several new diagnostic systems were commissioned in FY 2004 and some other existing diagnostics were upgraded or reached their fully operational state.

Motional Stark Effect Diagnostic

A major advance in NSTX diagnostic capability this year was the commissioning of the first eight channels of the Motional Stark Effect (MSE) diagnostic which 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. This met the NSTX Diagnostic Milestone D(04-1) (the original milestone had specified commissioning ten channels, but this proved impractical to accomplish after delays in deliveries of components by commercial suppliers). The MSE-CIF system required commissioning and dedicated calibration time in which 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. MSE data were taken in several experiments in the latter phase of the run and have been used to constrain the analysis of the NSTX equilibrium for the first time. This demonstration of the feasibility of performing measurements in a low-field device such as NSTX represents a significant step for the MSE technique. The MSE data contributed to meeting the NSTX Research Milestone R(04-3)

Fast Cameras for Visualizing Plasma Instabilities and Fluctuations

In February, the toroidally viewing ultra-fast soft x-ray pinhole camera became operational, fulfilling the diagnostic milestone D(04-2). This diagnostic records 2-dimensional images of the x-ray continuum from most of the plasma cross-section in the energy range 1 – 5 keV with selectable energy filters available. Up to 300 64 × 64 pixel images can be recorded by the CCD camera at a rate up to 500,000 frames per second to resolve rapidly evolving phenomena such as MHD instabilities. This PSI5 CCD camera was developed in a DOE-sponsored SBIR collaboration between Princeton Scientific Instruments and PPPL. The images from this camera complement the data from the four arrays of x-ray detectors which view across poloidal planes through the plasma. These data are being compared with full 3-dimensional MHD theoretical simulations of the NSTX plasma.

A similar PSI5 CCD was also installed in the Gas Puff Imaging (GPI) diagnostic which records images of visible line emission from neutral hydrogen and helium injected as gas into the plasma edge region. In previous operation with a slower camera capable of recording only 30 frames per shot, this diagnostic had revealed fluctuations in the density in the plasma edge associated with the turbulence believed to be responsible for anomalous heat and particle transport there. Using the new camera, it was possible to track the development and subsequent motion of “blobs” of plasma as they detached from the boundary of the confined plasma. These “movie” segments were also recorded during transitions from the “low” (L-) to the “high” (H-) mode of confinement showing the plasma boundary evolving from a highly turbulent to a more quiescent state. Interestingly, no evidence was found in the images for the development of a region of strong shear in the poloidal velocity of the turbulent structures. The spontaneous development of such shear has been postulated as the cause of the H-mode transition.

Another fast camera for visible light was brought to NSTX by a collaborator from the University of Hiroshima, Japan, and used to observe the divertor region through a re-entrant port in the lower divertor chamber. This camera revealed very interesting structures in the plasma edge which accompany a new type of edge-localized mode (ELM), the so-called Type V ELM, which has been identified in NSTX H-mode plasmas.

Diagnostics for Thermal and Non-Thermal Ions

Fast ions produced in the NSTX plasma by ionization of the heating neutral beams initially have orbits which are similar in normalized parameters to those of the energetic alpha-particles produced by deuterium-tritium fusion reactions in a future reactor based on the ST. Thus measuring the characteristics of the NBI-ions in NSTX will reveal much about the confinement of fusion alpha particles. The losses of energetic NBI-ions from NSTX have been detected by a new scintillator-based detector probe (SFLIP) mounted close to the vacuum vessel wall. This detector resolves both the energy and the pitch angle of the ions incident on its entrance aperture. Prompt losses have been detected of full-energy NBI-ions which are born onto orbits that intersect the detector. The dominant pitch angle of the losses changes in response to the evolution of the plasma density and thus the region of the plasma where the beams are ionized. Losses have also been detected of both full and partially thermalized ions which were originally confined but which have been scattered onto unconfined orbits by MHD instabilities of the plasma.

The NSTX charge-exchange recombination spectroscopy (CHERS) diagnostic was brought into full operation this year. Radial profiles of the impurity carbon ion temperature, toroidal flow velocity and density became available on a routine basis.

Radiometer for Mode-Converted Electron Bernstein Wave Emission

The first measurements were made during this year with an obliquely viewing 16 – 18 GHz radiometer which is designed to detect electromagnetic radiation mode-converted from the intrinsic thermal level of electron Bernstein waves (EBW) which exist in the plasma but cannot propagate directly to an external receiver. The apparent thermal temperature of the detected radiation and its polarization characteristics were generally consistent with the theory for the mode-conversion process and indicate that the reverse process, that is the mode conversion of powerful externally launched electromagnetic waves into EBW and their absorption in the plasma, is likely to be feasible for providing heating and current drive. These measurements directly addressed the NSTX Research Milestone R(04-5).

Turbulence Diagnostics

In addition to studying turbulence in the edge and scrape-off region, experiments were performed in FY 2004 to address the NSTX Research Milestone R(04-2) to measure long wavelength turbulence in spherical torus plasmas. In pursuit of this goal, a homodyne correlation microwave-reflectometer, developed by collaborators at the University of California at Los Angeles, made the first direct measurements of long-wavelength turbulence in the core of NSTX (normalized minor radius r/a in the range 0.45 – 0.75). The system has a frequency coverage of 26 – 40 GHz, corresponding to a density at the reflection layer of $(0.84 - 2.0) \times 10^{13}\ cm^{-3}$ for O-mode propagation. In NBI-heated, L-mode discharges, scans were performed to determine the scaling of the turbulence measured by the reflectometer and the confinement properties: the toroidal magnetic field and the plasma current were varied both together, to maintain constant q, and independently. The measured radial correlation lengths L_{cr} of the turbulent fluctuations were in the range 2 – 12 cm and decreased inversely with increasing local magnetic field strength so that at a fixed normalized minor radius, the ratio L_{cr}/ρ_s, where $\rho_s = c_s/\omega_{ci}$ is the characteristic ion gyro-radius, was approximately constant.

The fast reciprocating probe diagnostic, provided to NSTX by collaborators from the University of California at San Diego, was upgraded with a new measurement head and electronics. This system was used to measure both the profiles of the electron density and temperature and the turbulent fluctuations through the scrape-off layer into the edge pedestal region, up to 3 cm inside the last closed flux-surface, bridging the gap between the turbulence measurements in the core with the correlation reflectometer, and in the scrape-off region with the gas-puff imaging. The probe showed the occurrence of large, isolated bursts of plasma density, known as intermittency, near the plasma boundary. The amplitude of the bursts was only slightly higher in L-mode than in H-mode plasmas, but the frequency of the bursts was much higher in the L-mode.

Diagnostic Coils to Detect Slowly Rotating Magnetic Perturbations and Resistive Wall Modes

Prior to the FY 2003 experiments, a set of magnetic pickup coils had been installed on the passive stabilizer plates in NSTX to measure the structure of the magnetic perturbations from slowly evolving MHD instabilities such as locked-modes and resistive wall modes (RWM). There was not sufficient time in the abbreviated FY 2003 run to calibrate and characterize the response of these coils, however. This system was brought into operation during FY 2004 and used in several investigations of RWM physics, including the experiments using the first pair of external coils to apply time-varying radial-field perturbations to influence the plasma rotation and the growth of RWMs. The superiority of these coils in detecting such modes over the previous set of external pickup coils was evident in terms of both improved signal-to-noise in the inferred amplitudes of the mode components and reduced latency in detecting the mode onset. This bodes well for eventual use of the coils in real-time control to stabilize such modes in high-β plasmas.

Development of 'Multi-Color' Soft X-ray Diagnostics

A new diagnostic called a 'multi-color optical soft X-ray' (OSXR) array is being developed for NSTX by collaborators from the Plasma Spectroscopy Group of Johns Hopkins University. This diagnostic consists of an array of many discrete convertors (pixels) for soft x-ray emission from the plasma into visible light, coupled via high-throughput fiber-optic conduits to photomultipliers, whose output passes through charge integrators or fast amplifiers to a data acquisition system. 'Multi-color' refers to the possibility of stacking multiple arrays with different cut-off energies to provide spectrally resolved measurements. The advantages of the OSXR arrays compared to conventional SXR diode arrays are improved signal-to-noise ratio and, since only the convertor elements are inside the vacuum system, better plasma access. In addition, the lower cost per channel will allow the deployment of large pixel-count imaging systems. A 'single color' prototype array was successfully tested on NSTX during the FY 2004 run, showing performance similar to, or better than the conventional SXR diode arrays. A 'three-color', 48-channel, tangentially-viewing OSXR array is being built for the upcoming run, for fast ($\leq$ 100 μs time-resolution) and continuous measurements of changes in the electron temperature profile.

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 04	**FY 05**	**FY 06**	**FY 07**
Run weeks planned / achieved	20 /21	17	0	12
Hours of operation planned / achieved	800 / 840	680	0	480

Participating Research Personnel

	PPPL	non-PPPL
Researchers	53	65**
Post Doc.	3	3
Grad. Students	2	8
Undergrad. Students	2	1

** In addition there are over 25 overseas collaborating researchers from countries including Canada, France, Italy, Japan, Korea, Russia, UK, and Ukraine during FY 2004-2005.

Table 1: Facility Utilization

FUTURE ACCOMPLISHMENTS (FY2005-FY2007)

Future Research Accomplishments

The NSTX research accomplishments anticipated for FY2005-2007 are organized by research milestones that address the following overarching scientific issues for fusion plasmas:

A. Transport and Turbulence – Physical process that govern heat, particle and momentum confinement;
B. Macroscopic MHD Stability – Role of magnetic structure on plasma pressure and bootstrap current;
C. Wave-Particle Interactions – Role of electromagnetic waves and modes in sustaining and controlling hot plasmas;
D. Startup, Ramp-up and Sustainment – Physical processes of magnetic flux generation;
E. Boundary Physics – Interface between fusion plasmas and normal temperature surroundings;
F. Physics Integration – Physics synergy of external control and self-organization.

The major goals of the FY2005-2007 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 to reduce the cost, time and risk of fusion energy development. The following research milestones are proposed to achieve these goals. These milestones are organized according to the following three progressive levels of assumptions concerning the budget and the experimental run-time:

CASE-1: No run-weeks in FY2006, 12 run-weeks in FY2007,

CASE-2: 12 run-weeks in FY2006, 12 run-weeks in FY2007, and

CASE-3: 17 run-weeks in FY2006, 17 run-weeks in FY2007.

For clarity, we present a plan for each of these levels in the succeeding subsections: **CASE-1, CASE-2, and CASE-3**. The research milestones associated with these three cases will be designated with R1, R2 and R3, respectively. The cost estimates for facility operations in support of these levels of research is provided in EXPLANATION OF BUDGET, where more detailed tradeoffs in experimental run-time for FY2007, covering 0, 6, 12, 16, 20, and 25 experimental run-weeks, will also be delineated.

CASE-1

The organization and logic of the NSTX CASE-1 (baseline) research milestones for FY2005-2007 are summarized in Fig. 1. In this plan, the research team during FY2006 plans to take advantage of the anticipated research data from FY2005 to benchmark key theoretical models and simulation codes in Macroscopic Stability, Wave-Particle Interaction, and Physics Integration. These milestones for FY2006 are chosen to support effectively the research milestones in FY2007. During FY2006, the research team plans to analyze, publish, and present the extensive results from the FY2005 experimental campaign, which is scheduled to complete in September 2005. The research team proposes to enhance research collaboration on C-Mod, DIII-D, MAST and JET, emphasizing ITPA-relevant experiments and complementary capabilities on MAST, to participate in ITER Physics Tasks utilizing the team's expertise, and to prepare for the FY2007 experimental campaign. NSTX research participation in ITPA and ITER Physics Tasks, and collaboration with MAST are described in RELATIONSHIP TO OTHER PROGRAMS AND INTERNATIONAL COOPERATION.

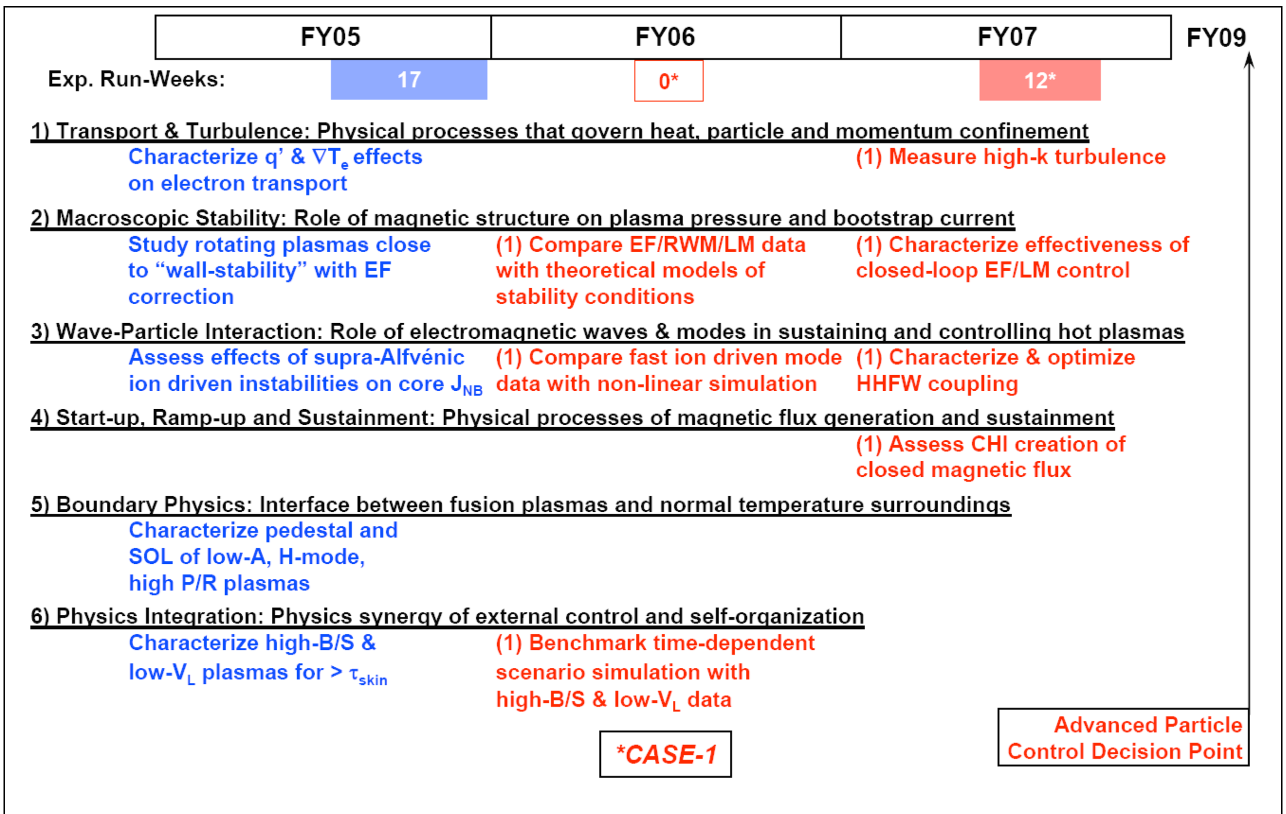

Figure 1. Research milestones for CASE-1 (baseline) funding according to DOE guidance.

A. Transport and Turbulence – Physical process that govern heat, particle and momentum confinement

Milestone R(05-1) on Turbulence and Transport: *Characterize the effects of variations in the magnetic shear and gradients in T_e on electron transport in low-aspect ratio plasmas. (September 2005)*

Description: The gradients in the safety factor q (the average magnetic field line rotation around the torus), otherwise known as magnetic shear, have been observed to strongly influence he electron temperature gradients and heat diffusivity in toroidal devices at both low and conventional aspect ratios. The microinstabilities driven by the electron temperature gradients are candidates for causing much of the electron heat transport. The ability to vary both the magnetic shear and the electron temperature gradients in high beta plasmas at low aspect ratio offers a unique possibility to elucidate the roles of these properties in altering the electron heat transport as possible beta-induced electromagnetic effects emerge. NSTX is in a strong position to research this topic since electron thermal conduction is the dominant energy transport channel in many high performance plasmas. Understanding the electron-scale turbulence and transport is one of the highest priority topics of interest to fusion energy sciences.

Technical Approach: Preliminary experiments to study the effect of magnetic shear on electron transport have been carried out in NSTX. These reveal reductions in electron and ion transport levels and the development of electron and ion Internal Transport Barriers in regions of reversed and low magnetic shear. The magnetic shear in these experiments was varied by changing the plasma current ramp-up rate and the timing of neutral beam injection. The q-profile, and thus the magnetic shear, in these experiments were estimated using model calculations that solved the magnetic field diffusion equation and from multi-chord measurements of soft x-ray emissions. This research will first characterize the variations in the measured electron energy transport with the q-profiles using the newly available Motional Stark Effect (MSE) diagnostic to provide a reliable determination of the q-profile. Higher resolution electron temperature profiles will be available with additional channels of the laser Thomson Scattering diagnostic, facilitating the study of the effects of changing magnetic shear on the electron temperature gradient, and, in turn, the effect of the electron temperature gradient on transport. To this latter end, the electron temperature gradient will be changed more rapidly than the magnetic shear, via turning-off of HHFW heating, injecting low-Z impurity pellets, and cold pulse propagation from Edge Localized Modes. Transient transport analysis will be carried out to determine electron heat diffusion and study critical temperature gradient physics. A new multi-color soft X-ray diagnostic will provide the fast electron temperature profile change measurements necessary for determining the perturbative transport coefficients. Low-k correlation reflectometry will provide information on core density fluctuation levels, while MSE will provide information on coherent magnetic fluctuations (up to ~100 kHz) in the finite-beta NSTX plasma core. Linear and non-linear gyrokinetic calculations will help in our understanding of the sources of electron transport, while accounting for the effects of ion transport.

<u>Milestone R1(07-1) on Transport and Turbulence</u>: *Measure short wavelength turbulence in the plasma core in a range of plasma conditions. (September 2007)*

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 toroidal plasmas at fusion-grade temperatures. 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. Gyrokinetic analysis indicates that short wavelength microinstabilities may play a strong role in such cases. Advanced microwave techniques will be available on NSTX to measure the short-wavelength turbulence properties over a wide range of conditions. These measurements will be used to test predictions from state-of-the-art theory and computation, especially as they pertain to the heat loss by the electrons due to short wavelength turbulence.

Technical approach – Experiments during FY2005 will have characterized the effects of radial gradients in safety factor q and T_e in governing electron thermal fluxes, and theoretical analysis will have revealed the classes of turbulence expected in high beta plasmas with large flow shear. In FY2006 under incremental funding, measurements of the density turbulence radial wavenumber spectra between electron scales and extending towards ion scales (2 - 20 cm^{-1}) will be carried out over a wide range of these gradients. These plasmas will likely have flow shearing rates up to a megaHertz and toroidal betas up to ~30%. To diagnose this short-wavelength turbulence, far-infrared microwaves will be launched from the plasma edge. The scattering of these microwaves off of the density fluctuations of the turbulence will yield key information about the short-wavelength turbulence characteristics of the high beta plasma. The compact ST geometry and strong field line pitch should yield measurement localization within a few centimeters. Diagnostics to measure the plasma density, ion temperature, electron temperature, ion flow velocity, magnetic field line pitch, and radial electric field will be used as input to theory codes. These codes will be used to predict turbulence characteristics and the ensuing electron transport for comparison with the experimental measurements.

B. Macroscopic MHD Stability – Role of magnetic structure on plasma pressure and bootstrap current

Milestone 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. (September 2005)*

Description – The effect of correcting magnetic field errors on increasing the pressure limits in strongly shaped rotating plasmas will be characterized. Large-scale, pressure driven plasma instabilities normally seen at very high pressures are restrained by plasma rotation, amplified by the asymmetries in the magnetic field, and can be mitigated by counteracting perturbations produced by the newly installed field-correction coils. The interactions among these properties will be studied as very high-beta plasmas approaching the "wall-stabilized" limits are produced and maintained for periods longer than the timescale for the stabilizing eddy currents in the walls to decay naturally. This research is expected to extend the understanding of achievable plasma pressure in the absence of active control, leading to a relatively robust plasma conditions in practical fusion energy systems.

Technical Approach – The beneficial effects of a conducting wall surrounding the plasma in stabilizing pressure-driven MHD instabilities are well known and have been confirmed in NSTX. The resistive wall mode (RWM) that has been observed in this situation can be stabilized if sufficient plasma rotation can be maintained. Large toroidal plasma rotation has been routinely produced in NSTX by neutral beam injection. As the plasma beta rises above the point when wall stabilization occurs, the interaction of the plasma rotation with the RWM, other MHD modes, and asymmetries in the magnetic fields can act to decrease the stabilizing rotation. These destabilizing mechanisms can be tamed by correcting the resonant component of the field asymmetries acting on the plasma. Correction field coils and power supplies have been designed and installed in NSTX with the aim of maintaining higher beta for longer timescales beyond the levels achieved during FY2004. Neutral beam injection heating will be applied to create strongly rotating plasmas that approach the ideal wall-stabilized stability limit. Detailed measurements of internal magnetic field pitch angle will be made to strengthen present theoretical calculations of these modes for comparison with the observed mode behavior. The data from poloidal and toroidal arrays of magnetic field sensors and plasma profiles will be used in these analyses to determine the dependence of the critical rotation frequency for stabilization of the RWM on plasma parameters, and the nature of the resonant field response by the plasma to the asymmetries in the external magnetic fields. Under the conditions that this field response shows simple and reproducible structure, it may be possible to counteract it by applying a preprogrammed correction field. When the modes rotate or grow rapidly, active feedback control of the error field and/or the RWM will most likely be required. Results of this research will provide crucial information to compare with modern theories of MHD instabilities and prepare for testing of such active feedback in the succeeding years. Sustainment of plasma pressure up to the wall-stabilized limit will enable a systematic investigation of the properties of high-performance plasmas and in turn determine the attractiveness of the ST configuration, and contribute to the optimization of the burning plasma in ITER.

Milestone R1(06-1) on Macroscopic Stability: *Compare 2005 data on error field, resistive wall mode (RWM), and locked mode properties with theoretical models of error field and MHD mode stability conditions that apply to low and high aspect ratios. (September 2006)*

Description – Data from Milestone R(05-1) on macroscopic stability is expected to cover, for the low aspect ratio, a wide range of parameters in plasma shaping, locations of resonant field surfaces, pressure relative to the applied magnetic field, non-axisymmetric fields including applied and induced field errors, plasma rotation relative to the plasma Alfvén and sound velocities, and the amplitude and rotation speed of the global modes. These will used in a detailed comparison with the modern theoretical models of RWM and locked modes, using such codes as MARS, VALEN and DCON. The resulting improvements will contribute to a predictive understanding of the macroscopic stability requirements of high-performance plasmas in NSTX, ITER, and other tokamaks.

Technical Approach – Experiments in FY2005 on the plasma conditions close to the "wall-stabilized" pressure limit and the macroscopic mode characteristics will enable in FY2006 a detailed comparison of data with analytic theory, numerical models and simulation codes, such as MARS-F, VALEN, PEST-III, M3D, and DCON. This work will include comparisons with the results of normal aspect ratio through collaborative experiments to ensure direct relevance of results to plasma macroscopic stability at all aspect ratios. The following three scientific questions will be addressed in this research. How does the plasma rotation frequency required for the global mode stabilization depend on key plasma properties such as density, temperature, rotation, safety factor q, field errors, and the aspect ratio? How do these dependencies reveal the underlying mechanisms of dissipation of the global modes and plasma rotation, and of penetration of external error fields into the rotating plasma? What are the likely sources of error fields in addition to those applied through external coils, and if identified, how might they be corrected? The results will include possible improvements in theory, modeling and simulation that will lead to a predictive understanding of the stability conditions required by high-performance plasmas in low and higher aspect ratio experiments.

Milestone R1(07-2) on Macroscopic Stability: *Characterize the effectiveness of active feedback control of resonant error fields and, with sufficient resources, wall-coupled, pressure-limiting global modes, using closed-loop control of currents in ex-vessel correction coils. (September 2007)*

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, the stabilizing effects on the Resistive Wall Modes (RWMs) from resonant error field correction and strong plasma rotation are expected to change. Active feedback on the amplitude of the resonant error fields and the RWMs is expected to be required to maintain stability of such plasmas. Stabilization of RWMs 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.

Technical Approach – Six ex-vessel field correction coils, driven by Switching Power Amplifiers (SPA), will have been fully commissioned and tested in FY2005. The effectiveness of using feedback control of the resonant field errors, and when appropriate, the wall-coupled, pressure-limiting RWMs, will be characterized over a range of plasma conditions, profiles, shapes, rotation frequencies, and feedback control parameters (current amplitude, toroidal rotation frequency, and phase lag) for the lowest toroidal mode numbers. Of particular interest will be the interplay among plasma rotation, error fields, and RWM as the "wall-stabilized" pressure limit is approached. Extensive data will be compared with a suite of theoretical models (slab models, VALEN, MARS, M3D, etc.) to understand the conditions in which control of the resonant field errors alone can remain effective in maintaining large plasma rotation and suppressing the RWM. The contributions of eddy currents in nearby asymmetric conductors, induced by currents in the field correction coils, will also be clarified. With incremental funding, direct, active feedback control of the RWM will be tested under the conditions where such control is needed to maintain stability, extending beyond the regimes achieved via error field-rotation control. The results will improve the predictive understanding of rotation, error field, and control requirements for future experiments on NSTX and optimization of the ITER burning plasma.

C. Wave-Particle Interactions – Role of electromagnetic waves and modes in sustaining and controlling hot plasmas

Milestone R(05-3) on Wave-Particle Interactions: *Assess the effects of supra-Alfvénic fast ion driven instabilities on driven current in the plasma core. (September 2005)*

Description – Instabilities driven by supra-Alfvénic ions and the resulting fast-ion loss and diffusion may play a strong role in the NB heating and current drive efficiencies in the ST as well as ITER. These processes may affect the sustainable current density profiles in high-performance NBI-heated ST plasmas and the long-pulse "hybrid" plasmas projected in ITER. Enhanced losses of fast ions have been observed in the presence of a broad range of fast-ion-driven MHD activity on NSTX. This research therefore aims to assess the effects of these interactions on the core plasma current stemming from the fast ions. The results will enhance the basis for predictive understanding of long-pulse high performance plasmas on NSTX, and to strengthen the scientific basis for the ITER hybrid mode operation.

Technical Approach – Experiments will be carried out where the neutral beam driven current is maximized, such as in moderate density, high electron temperature plasmas. Toroidal Alfvén eigenmodes (TAE) and energetic particle modes (EPM) are expected to be excited with a relatively sharp threshold in the beam energy, and hence the degree to which the fast ions become supra-Alfvénic. By operating just above and below this threshold, the current profile evolution can be compared in otherwise very similar discharges, correlated with the presence and absence of these modes. The Motional Stark Effect (MSE) diagnostic will be used to measure current profile evolution. Recent analysis of the onset of the sawtooth-like core reconnection instability when q(0) reaches unity provided independent validation of this measurement. Mirnov coils, soft x-ray cameras, heterodyne reflectometers, and far-infrared tangential interferometers will be used to characterize these modes. The scanning and solid-state neutral particle analyzer (NPA) diagnostics will be used to monitor fast ion transport, assisted by the fast lost ion probes (FLIP) and neutron diagnostics. Initial analysis will be carried out using TRANSP, which calculates the fast-ion distribution and beam-driven currents in the absence of the fast-ion-driven modes. The current density profile will be compared with that from EFIT, which ensure MHD equilibrium consistency with measured plasma density, temperature, rotation profiles and plasma shape. These results will be used in FY2006 to benchmark the Alfvén eigenmode mode theory of mode structure and evolution that can affect on the fast-ion distribution and the core plasma current densities in NSTX as well as in future burning plasmas such as ITER.

Milestone R1(06-2) on Wave-Particle Interactions: *Compare data on supra-Alfvénic fast ion driven modes with non-linear simulations of these modes. (September 2006)*

Description – Simulation of the non-linear instabilities driven by supra-Alfvénic ions, and their impact on fast ion diffusion, loss, and driven current is necessary to develop a predictive understanding of the ITER hybrid-mode burning plasma operation. Verification of such simulation will also advance the study of long-pulse high-performance plasmas on NSTX.

Simulations of these modes have begun with the non-linear kinetic M3D and particle-following HYM codes, prompted by recent experimental data indicating a wide frequency range and a strong non-linear nature of these instabilities. This research therefore aims to compare the experimentally measured characteristics of such instabilities during 2005, including the changes in the fast ion driven current and plasma heating, with the anticipated results from these simulations. The outcome will be important to predictions on future plasmas driven by supra-Alfvénic fast ions in NSTX and anticipated in ITER.

Technical Approach - Experimental data collected in the 2005 campaign from high performance regimes with supra-Alfvénic-ion driven instabilities will be used to benchmark M3D and HYM simulations and guide the code development. Data from the Fast Ion Loss Probe (FLIP) on the energy and pitch angle of the lost fast ions, and from the scanning Neutral Particle Analyzer (NPA) on the distribution of the fast ions in energy, pitch-angle, and position, will be used to correlate with the responsible modes. Data on mode structure from the Mirnov magnetic coil arrays, fast ultra-soft x-ray cameras, the heterodyne reflectometers and tangential interferometers will be compared with calculations using the HYM and M3D extended MHD codes. These codes account for the kinetic effects of the fast ions, and the non-linear evolution of the interactions between the fast ion-driven instabilities and the fast ions. High frequency (>0.5 MHz) Compressional Alfvén Eigenmode (CAE) and Global Alfvén Mode (GAE) instabilities will be calculated with HYM, which is designed to study high toroidal mode numbers ($n \geq 10$ to 30). Modes with lower toroidal mode numbers ($n<5$) in the Alfvén range of frequencies (50 to 150 kHz) will be calculated with M3D. The simulations will account for the nonlinear modification of beam ion distribution functions on the mode-growth time-scale. Mode excitation and saturation mechanisms will be determined to guide future experiments, such as Milestone FY07-3-II to quantify diffusion and loss of these fast ions.

Milestone R1(07-3) on Wave-Particle Interactions: *Characterize the interaction between the edge plasma region and the launched High Harmonic Fast Waves (HHFW), and determine plasma conditions that permit efficient heating and current drive via HHFW. (September 2007)*

Description – Strong interactions between the edge region plasma and the impressed RF fields from the launchers during HHFW heating will be investigated and characterized in detail using a collection of measurements including probes, reflectometry and electromagnetic sensors. In addition, the edge plasma and antenna characteristics will be varied to maximize the efficiency of HHFW heating and current drive.

Technical Approach – Initial results in FY2004 indicated the existence of strong ion heating in the plasma edge region by the RF fields launched from the HHFW antenna. These investigations will continue using edge probes and edge rotation diagnostics (ERD) to look for likely edge RF heating mechanisms due to decay waves resulting from nonlinear coupling in the presence of large plasma gradients. In addition reflectometry will be used to monitor wave penetration deeper than the plasma periphery. Electromagnetic sensors will be used to look for the presence of far-field sheaths which can be driven by surface waves propagated away from the launchers. Variations in the edge plasma parameters and antenna geometry will be explored to determine approaches to maximize the overall effectiveness of the HHFW heating and current drive. The results will be important to improve of the HHFW theory and computation models to develop a predictive understanding of the wave interactions of the over-dense high beta plasmas at the high ion cyclotron harmonics.

D. Startup, Ramp-up and Sustainment – Physical processes of magnetic flux generation and sustainment

Milestone R1(07-4) on Start-up, ramp-up and Sustainment: *Assess the conditions in which a substantial amount of closed poloidal magnetic flux is created via Coaxial Helicity Injection. (September 2007)*

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 will be performed in NSTX to apply an improved method of non-inductive startup, referred to as transient coaxial helicity injection (CHI), which has been successfully developed on the HIT-II Concept Exploration experiment. The plasma and operational conditions on NSTX for effective initiation of plasma current using this method will be assessed and identified.

Technical Approach – 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 rapidly stopped, magnetic reconnection occurs near the injection electrodes to form closed flux surfaces. In the HIT-II experiments, closed-flux toroidal plasma currents up to 100kA were achieved, and when induction was applied to increase the plasma current, a substantial saving of the inductive flux was realized. On NSTX, an initial test of this method was conducted during 2004. Toroidal plasma currents up to 140 kA were produced for injector currents of only 4.4 kA, representing a record current multiplication factor over 30. However, an unambiguous demonstration of closed flux beyond the end of the injection pulse was not achieved because the electron temperature, measured by Thomson scattering to be about 16 eV peak, was too low for the L/R decay time of the toroidal plasma current to exceed the RC decay time of the injector current. Three areas for improvement have been identified: (1) doubling the injector voltage to 2 kV and improving the gas preionization to allow breakdown at lower gas pressure, thereby increasing the overall energy input per particle, (2) reducing the separation of the injector flux footprints on the electrodes to promote reconnection and detachment of the plasma, and (3) improving equilibrium control of the evolving discharge. In the forthcoming experiments on NSTX, preionization will be improved by injecting both neutral gas and 10 kW of 18 GHz ECH power into the chamber below the lower divertor plates which are used as the injector electrodes. After demonstration of initial current persistence, the capacitor bank voltage, toroidal field and injector flux will be optimized to produce the highest possible start-up currents. The CHI-produced plasma will be coupled to induction from the central solenoid to demonstrate compatibility of this new method with the conventional inductive method. In subsequent experiments, instead of using the central solenoid, the CHI produced plasma will be ramped up using induction from the outer PF coils to a level where neutral beams and HHFW could be applied to sustain the current. In support of this work we will perform modeling and conduct experiments to develop an understanding of the capability of the outer PF system to initiate closed flux currents and that of the HHFW system to ramp-up the plasma current from an initial level of about 200kA.

E. Boundary Physics – Interface between fusion plasmas and normal temperature surroundings

Milestone 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. (September 2005)*

Description - High performance toroidal plasmas in the spherical torus and tokamak are expected to set stringent requirements on the plasma edge conditions and introduce high heat fluxes that require increased control. Plasma edge parameters and configurations in NSTX will be varied to obtain the optimal dispersal of edge heat flux in conditions that have favorable performance characteristics in the core. The effects of transient phenomena, such as the Edge Localized Modes (ELMs) and intermittent bursts of ejected plasma filaments, on the edge pedestal and plasma facing components will also be assessed. Comparison of these results with those from higher aspect-ratio tokamaks will improve the capability to predict the edge plasma properties in future devices.

Technical Approach - The plasma edge efflux will be characterized under various conditions to explore the basic driving parameters for the dominant transport phenomena, namely, ELMs, intermittent transport and broadband turbulent transport. Variations in experimental parameters, such as the plasma shape, density, magnetic configuration, plasma heating, fueling, and wall control methods, will be use to vary fundamental plasma parameters such as edge bootstrap current, magnetic shear, edge collisionality and X-point location/resistivity in order to understand the mechanisms leading to the edge particle and heat dispersal. The edge pedestal height, width, and gradient to the plasma edge fluxes, parameters, and configurations will be studied with improved spatial resolution in measurements using laser Thomson scattering, fixed probes, scanning probes, edge impurity spectroscopy, and reflectometry of microwave pulses. The impact of edge-localized modes and reconnection events on the edge and divertor and its distribution with conditions in the edge and scrape-off layer (SOL) will also be investigated. In addition, the balance between diffusive and convective cross-field transport will be studied with improved edge turbulence diagnostics. Also, filtered cameras, an array of fixed divertor Langmuir probes, a divertor bolometer, and a fast time response infrared camera will be implemented for more detailed measurements in the divertor region. Results will be compared with numerical calculations of the edge pedestal and SOL regions based on a collection of MHD stability (e.g., ELITE), and fluid steady-state (e.g., UEDGE), and time-dependent (e.g., BOUT) models.

F. Physics Integration – Physics synergy of external control and self-organization

Milestone 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. (September 2005)*

Description – A physics basis for non-inductive sustainment of strongly shaped high-confinement and high-beta plasmas, driven primarily by high-power energetic neutral particle injection on NSTX, will be characterized in this study. MHD equilibrium, transport, and current evolution models for such plasmas will be computed accounting for a full suite of measured plasma profiles. Stability analysis for these plasmas will be carried out to compare theory with measurements of MHD induced effects. The results, together with those from Milestone R(05-4) to characterize the plasma edge, will be crucial to the subsequent investigations of long-pulse high-performance plasmas. The results will establish the baseline requirements for sustaining plasmas with zero surface loop voltage, which is of high interest to future burning plasmas in ITER and possible component test facilities.

Technical Approach - High-confinement and high-beta plasmas are sustained for increased duration by avoiding deleterious MHD and providing more of the total plasma current with non-inductive sources (bootstrap, HHFW, and NBCD), thereby reducing the inductive current fraction. In the 2005 run period the plasma elongation will be increased using the modified PF1A coil to broaden the current profile and increase the bootstrap current fraction. The early heating and H-mode transition developed in the 2004 run period will be used to elevate the safety factor, keeping q > 1 for long periods, and allow the tailoring of the current profile by varying the timing of the transition. Routine use of the Motional Stark Effect diagnostic in the 2005 run period to measure the safety factor will enable correlating its evolution with MHD and transport features. Broad pressure profiles provide higher ideal beta-limits, and optimization of the pedestal and ELM regime will be attempted to access high confinement and high beta, which in turn enhance the bootstrap and diamagnetic current sources. Efforts will begin to examine the impact of lower toroidal fields, which is necessary to reach the very high beta, long pulse integrated scenario target. Lower toroidal fields increase beta, which will challenge the regime of strong n>0 external kink modes, RWMs, and their stabilization.

Milestone R1(06-3) on Physics Integration: *Benchmark and improve physics models and the time-dependent simulation codes with the FY2005 data from high-performance plasmas characterized by large self-driven current, high pressure relative to the applied toroidal field, and low toroidal induction voltage. (September 2006)*

Description – Plasma modeling and integrated simulation, using such codes as TRANSP and TSC, are important to understanding experimental data and predicting future high-performance plasmas. Physics models for heating, stability, transport, and driven current will be benchmarked and improved by comparison with the FY2005 experimental data to strengthen the scientific basis of the integrated scenario simulation. The results will be important to the preparation of future experiments on high-performance plasmas with nearly zero toroidal induction voltage in NSTX and for the ITER "hybrid" operation scenario.

Technical Approach – This research will couple the free-boundary plasma evolution simulation of the TSC code to state-of-the-art heat, particle and momentum deposition modeling of the TRANSP code. In parallel, the gyrokinetic theory-based models for micro-turbulence and transport, such as the non-linear GS2 and GYRO codes, will be benchmarked against the experimental data from Milestone R(05-5) on turbulence and transport. Data of high-performance plasmas, from Milestone R(05-2) on physics integration, will be used to benchmark and update the time-dependent scenario simulation from the TSC. Data of the plasma edge properties, from Milestone R(05-4) on boundary physics, will be used to determine appropriate boundary conditions for the time-dependent scenario simulation. Finally, an improved numerical plasma structure model will be implemented to improve the realism of time-dependent MHD stability simulations. The updated TRANSP and TSC codes will be applied to making improved predictions of future high-performance plasmas in NSTX and ITER.

CASE-2

The organization and logic of the NSTX CASE-2 research milestones for FY2006-2007 are summarized in Fig. 2. In this plan, the FY2007 baseline milestones can be advanced to FY2006. Important milestones can be accomplished in FY2007, enabling more effective scientific progress towards the NSTX programmatic goals identified above. As indicated in Fig.2, one CASE-1 milestone, R1(06-3), is retained in this case. New CASE-2 milestones are described below.

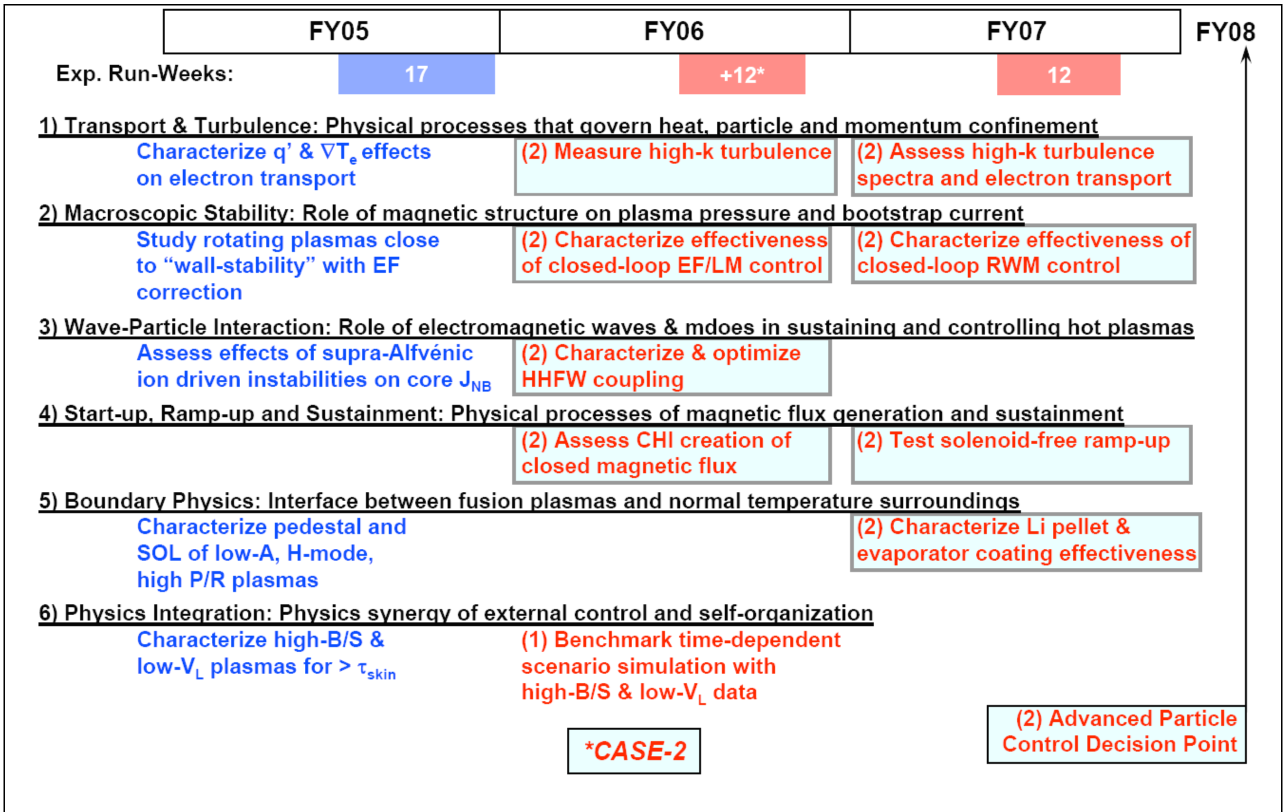

Figure 2. Research milestones assuming CASE-2 funding for FY2006-2007.

A. Transport and Turbulence – Physical process that govern heat, particle and momentum confinement

Milestone R2(06-1) on Transport and Turbulence: *Measure short wavelength turbulence in the plasma core in a range of plasma conditions. (September 2006)*

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 toroidal plasmas at fusion-grade temperatures. 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. Gyrokinetic analysis indicates that short wavelength microinstabilities may play a strong role in such cases. Advanced microwave techniques will be available on NSTX to measure the short-wavelength turbulence properties over a wide range of conditions. These measurements will be used to test predictions from state-of-the-art theory and computation, especially as they pertain to the heat loss by the electrons due to short wavelength turbulence.

Technical approach – Experiments during FY2005 will have characterized the effects of radial gradients in safety factor q and T_e in governing electron thermal fluxes, and theoretical analysis will have revealed the classes of turbulence expected in high beta plasmas with large flow shear. In FY2006 under incremental funding, measurements of the density turbulence radial wavenumber spectra between electron scales and extending towards ion scales (2 - 20 cm^{-1}) will be carried out over a wide range of these gradients. These plasmas will likely have flow shearing rates up to a megaHertz and toroidal betas up to ~30%. To diagnose this short-wavelength turbulence, far-infrared microwaves will be launched from the plasma edge. The scattering of these microwaves off of the density fluctuations of the turbulence will yield key information about the short-wavelength turbulence characteristics of the high beta plasma. The compact ST geometry and strong field line pitch should yield measurement localization within a few centimeters. Diagnostics to measure the plasma density, ion temperature, electron temperature, ion flow velocity, magnetic field line pitch, and radial electric field will be used as input to theory codes. These codes will be used to predict turbulence characteristics and the ensuing electron transport for comparison with the experimental measurements.

Milestone R2(07-1) on Turbulence and Transport: *Assess the correlation between measured and calculated high-k turbulence spectra, and the measured electron thermal conductivity. (September 2007)*

Description – The results of Milestones R(05-1) and R2(06-1) are expected to establish a physics basis with which to begin in FY2007 a detailed investigation of the parametric dependence of the turbulence-induced local electron thermal conductivity in the NSTX plasma core. Physics of interest will include the presence or absence, in strongly shaped, collisionless, high-beta, low-aspect-ratio plasmas, of the electron temperature gradient (ETG) driven turbulence, its possible nonlinear consequence in the form of “streamers” in the radial direction, and the micro-tearing fluctuations. The correlation of these simulated fluctuations with the electron heat conduction inferred from transport analysis of a suite of plasma measurements will be determined over a range of plasma conditions of interest. Progress in this area will be of high interest to the goal of predictive understanding of transport and turbulence.

Technical Approach – The high-k scattering system on NSTX, capable of measuring fluctuations with radial wavenumbers 2 – 20 cm^{-1} with excellent spatial resolution, will serve as the experimental focal point of these studies. Systematic scans of parameters that are expected theoretically to affect electron thermal transport, such as the magnetic shear, plasma beta, temperature gradient, and the ratio of electron and ion temperatures, will be performed. The measured spectra will be compared to global and flux-tube, linear and nonlinear gyrofluid and gyrokinetic particle simulations and of turbulence amplitudes driven on the electron gyroradius scales, using codes such as GS2, GYRO, and GTC presently under development in the community. To assess the possible role of electromagnetic effects on turbulence in high beta plasmas, the simulations will be carried out under various assumptions regarding the electron dynamics to sort out the relevant assumptions. For these same plasmas, the simulated turbulence-induced electron thermal heat fluxes will be compared to those inferred from power balance analyses using TRANSP.

B. Macroscopic MHD Stability – Role of magnetic structure on plasma pressure and bootstrap current

Milestone R2(06-2) on Macroscopic Stability: *Characterize the effectiveness of active feedback control of resonant error fields using closed-loop control of currents in ex-vessel correction coils. (September 2006)*

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, the stabilizing effects on the Resistive Wall Modes (RWMs) from resonant error field correction and strong plasma rotation are expected to change. Active feedback on the amplitude of the resonant error fields may be effective in maintaining stability of such plasmas. 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.

Technical Approach – Six ex-vessel field correction coils, driven by Switching Power Amplifiers (SPA), will have been fully commissioned and tested in FY2005. The effectiveness of using feedback control of the resonant field errors for the lowest toroidal mode numbers will be characterized over a range of plasma conditions, profiles, shapes, rotation frequencies, and feedback control parameters (current amplitude, toroidal rotation frequency, and phase lag). Of particular interest will be the interplay among plasma rotation, error fields, and RWM as the "wall-stabilized" pressure limit is approached. Extensive data will be compared with a suite of theoretical models (slab models, VALEN, MARS, M3D, etc.) to understand the conditions in which control of the resonant field errors alone can be effective in maintaining large plasma rotation and suppressing the RWM. The contributions of eddy currents in nearby asymmetric conductors, induced by currents in the field correction coils, will also be clarified. These results will improve understanding of rotation, error field, and their control requirements for future experiments on NSTX and optimization of the ITER burning plasma.

Milestone R2(07-2) on Macroscopic Stability: *Characterize the effectiveness of active feedback control of wall-coupled, pressure-limiting global modes, using closed-loop control of currents in ex-vessel correction coils. (September 2007)*

Description – As an important extension of Milestone R2(06-2), active feedback on the amplitude of the RWMs is expected to enhance the stability of such plasmas beyond active feedback on the resonant error fields alone. Direct stabilization of RWMs 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.

Technical Approach – 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, profiles, shapes, rotation frequencies, and feedback control parameters (current amplitude, toroidal rotation frequency, and phase lag). These results, together with further improvements in the speed and algorithms of the control circuits, will be used as a basis for testing direct, active feedback control of the RWM under the conditions where such control is needed to maintain stability. The results will improve understanding of RWM control requirements for future experiments on NSTX and optimization of the ITER burning plasma.

C. Wave-Particle Interactions – Role of electromagnetic waves and modes in sustaining and controlling hot plasmas

Milestone R2(06-3) on Wave-Particle Interactions: *Characterize the interaction between the edge plasma region and the launched High Harmonic Fast Waves (HHFW), and determine plasma conditions that permit efficient heating and current drive via HHFW. (September 2006)*

Description – Strong interactions between the edge region plasma and the impressed RF fields from the launchers during HHFW heating will be investigated and characterized in detail using a collection of measurements including probes, reflectometry and electromagnetic sensors. In addition, the edge plasma and antenna characteristics will be varied to maximize the efficiency of HHFW heating and current drive.

Technical Approach – Initial results in FY2004 indicated the existence of strong ion heating in the plasma edge region by the RF fields launched from the HHFW antenna. These investigations will continue using edge probes and edge rotation diagnostics (ERD) to look for likely edge RF heating mechanisms due to decay waves resulting from nonlinear coupling in the presence of large plasma gradients. In addition reflectometry will be used to monitor wave penetration deeper than the plasma periphery. Electromagnetic sensors will be used to look for the presence of far-field sheaths which can be driven by surface waves propagated away from the launchers. Variations in the edge plasma parameters and antenna geometry will be explored to determine approaches to maximize the overall effectiveness of the HHFW heating and current drive. The results will be important to improve of the HHFW theory and computation models to develop a predictive understanding of the wave interactions of the over-dense high beta plasmas at the high ion cyclotron harmonics.

D. Startup, Ramp-up and Sustainment – Physical processes of magnetic flux generation and sustainment

Milestone R2(06-4) on Start-up, ramp-up and Sustainment: *Assess the conditions in which a substantial amount of closed poloidal magnetic flux is created via Coaxial Helicity Injection. (September 2006)*

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 will be performed in NSTX to apply an improved method of non-inductive startup, referred to as transient coaxial helicity injection (CHI), which has been successfully developed on the HIT-II Concept Exploration experiment. The plasma and operational conditions on NSTX for effective initiation of plasma current using this method will be assessed and identified.

Technical Approach – 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 rapidly stopped, magnetic reconnection occurs near the injection electrodes to form closed flux surfaces. In the HIT-II experiments, closed-flux toroidal plasma currents up to 100kA were achieved, and when induction was applied to increase the plasma current, a substantial saving of the inductive flux was realized. On NSTX, an initial test of this method was conducted during 2004. Toroidal plasma currents up to 140 kA were produced for injector currents of only 4.4 kA, representing a record current multiplication factor over 30. However, an unambiguous demonstration of closed flux beyond the end of the injection pulse was not achieved because the electron temperature, measured by Thomson scattering to be about 16 eV peak, was too low for the L/R decay time of the toroidal plasma current to exceed the RC decay time of the injector current. Three areas for improvement have been identified: (1) doubling the injector voltage to 2 kV and improving the gas preionization to allow breakdown at lower gas pressure, thereby increasing the overall energy input per particle, (2) reducing the separation of the injector flux footprints on the electrodes to promote reconnection and detachment of the plasma, and (3) improving equilibrium control of the evolving discharge. In the forthcoming experiments on NSTX, preionization will be improved by injecting both neutral gas and 10 kW of 18 GHz ECH power into the chamber below the lower divertor plates which are used as the injector electrodes. After demonstration of initial current persistence, the capacitor bank voltage, toroidal field and injector flux will be optimized to produce the highest possible start-up currents. The CHI-produced plasma will be coupled to induction from the central solenoid to demonstrate compatibility of this new method with the conventional inductive method. In subsequent experiments, instead of using the central solenoid, the CHI produced plasma will be ramped up using induction from the outer PF coils to a level where neutral beams and HHFW could be applied to sustain the current. In support of this work we will perform modeling and conduct experiments to develop an understanding of the capability of the outer PF system to initiate closed flux currents and that of the HHFW system to ramp-up the plasma current from an initial level of about 200kA.

Milestone R2(07-4) on Solenoid-free Startup, Ramp-up and Sustainment: *Test conditions for solenoid-free ramp-up of plasma to substantial plasma current. (September 2007)*

Description – Externally driven current and internally driven current via plasma pressure gradient will be combined with the intrinsic electromagnetic induction produced by the external equilibrium control coils, to increase the plasma current from ~100 kA up to the level of 300–400 kA. High Harmonic Fast Wave (HHFW), Neutral Beam Injection (NBI), and Coaxial Helicity Injection (CHI) will be used for this purpose. The conditions of the plasma as the current is ramped up without adding inductive flux from the central solenoid will be characterized.

Technical Approach – As plasma current is increased to substantial levels, the HHFW launched into the plasma can impart directed momentum to the plasma electrons and drive current. The NSTX HHFW system is capable of launching up to 6 MW of RF power in a directed wave. By varying the spectrum of the launched waves the location of the driven current can be varied to match the current profiles needed by the particular experiment. The pressure gradient driven bootstrap current, which has been demonstrated in the low-aspect-ratio NSTX and the higher aspect-ratio tokamaks, is calculated to be higher in magnitude in collisionless, high-beta plasmas in NSTX. It may be an efficient mechanism for current ramp-up, and is likely to add current away from the plasma core to help match the profile of the total current needed for MHD stability. Given adequate plasma current, energetic neutral beams injected (NBI) tangentially to heat the plasma can also drive significant current in the plasma core. CHI current drive may be used to modify the edge current profile and the edge plasma flows to assist the current ramp-up. Furthermore, as the plasma pressure increases as a result of the heating, the external equilibrium coils couple substantial inductive flux, thereby creating additional toroidal plasma current. Different combinations of these techniques will be studied to test effective current ramp-up from an initially low level to substantially higher plasma currents in NSTX.

E. Boundary Physics – Interface between fusion plasmas and normal temperature surroundings

Milestone R2(07-5) on Boundary Physics: *Characterize the effectiveness of lithium pellet injection and tile coating in controlling fuel recycling from the plasma facing components. (September 2007)*

Description – Experiments and theory of toroidal plasmas have suggested that a controlled reduction of fueling recycled from the plasma facing components can lead to large improvements of the plasma confinement not only near the edge, but also in the core. Reducing and controlling the plasma density is also an important requirement for efficient solenoid-free plasma current startup and sustainment. Lithium coating, already shown on TFTR and CDX-U to produce beneficial effects, has the potential provide an effective control of wall recycling in NSTX. Lithium coating via both pellet injection into helium plasmas and evaporation will be applied before normal plasma pulses to determine the effectiveness of this technique. The results will clarify long pulse heat and particle control requirements on NSTX and contribute to future decisions on advanced particle control.

Technical Approach – The techniques of lithium coating, which allowed high-performance plasmas to be routinely achieved on TFTR, will be tested and developed on NSTX via a sequence of lithium pellet injection and evaporation coating experiments. Starting with plasma-facing surfaces well-conditioned via helium discharges, lithium pellets will be injected into ohmically heated plasmas systematically varied from inboard-limited to fully diverted configurations, to coat the plasma-interacting graphite tiles in NSTX. The effects of activated lithium coating on subsequent deuterium-fueled plasmas in varying configurations will then be documented and characterized. Of key interest in this research are detailed comparisons of fueling rates, deuterium, lithium, and carbon radiation, recycling, diffusion, and migration between discharges with and without lithium coating. A suite of spectroscopic, infrared and soft x-ray diagnostics will be applied for this purpose. The concomitant changes in the plasma core and edge properties will also be measured and documented, for ohmically and neutral beam heated L-mode and H-mode plasmas in inboard-limited and diverted plasma configurations. The results will clarify the location, surface area, and thickness of lithium coating needed to derive substantial benefits of lithium coating on graphite tiles, and guide the operation of a subsequent lithium evaporator aimed at achieving similar benefits more effectively. Successful lithium techniques for plasma wall conditioning and recycling control will benefit plasma performance on NSTX and other major fusion experiments.

CASE-3

The organization and logic of the NSTX CASE-3 research milestones for FY2006-2007 are summarized in Fig. 3. In this plan, the research team would be able to achieve major progress in resolving important scientific issues in all topical areas, supporting the effort to enhance burning plasma performance in ITER, and determining how the ST can reduce the cost, time and risk in the development of practical fusion power. As indicated in Fig.3, CASE-2 milestones, R2(06-1), R2(06-3), R2(06-4), R2(07-1), R2(07-4) are retained for this case. CASE-2 milestones, R2(06-2) and R2(07-2), are combined into R3(06-2). New CASE-3 milestones are described below.

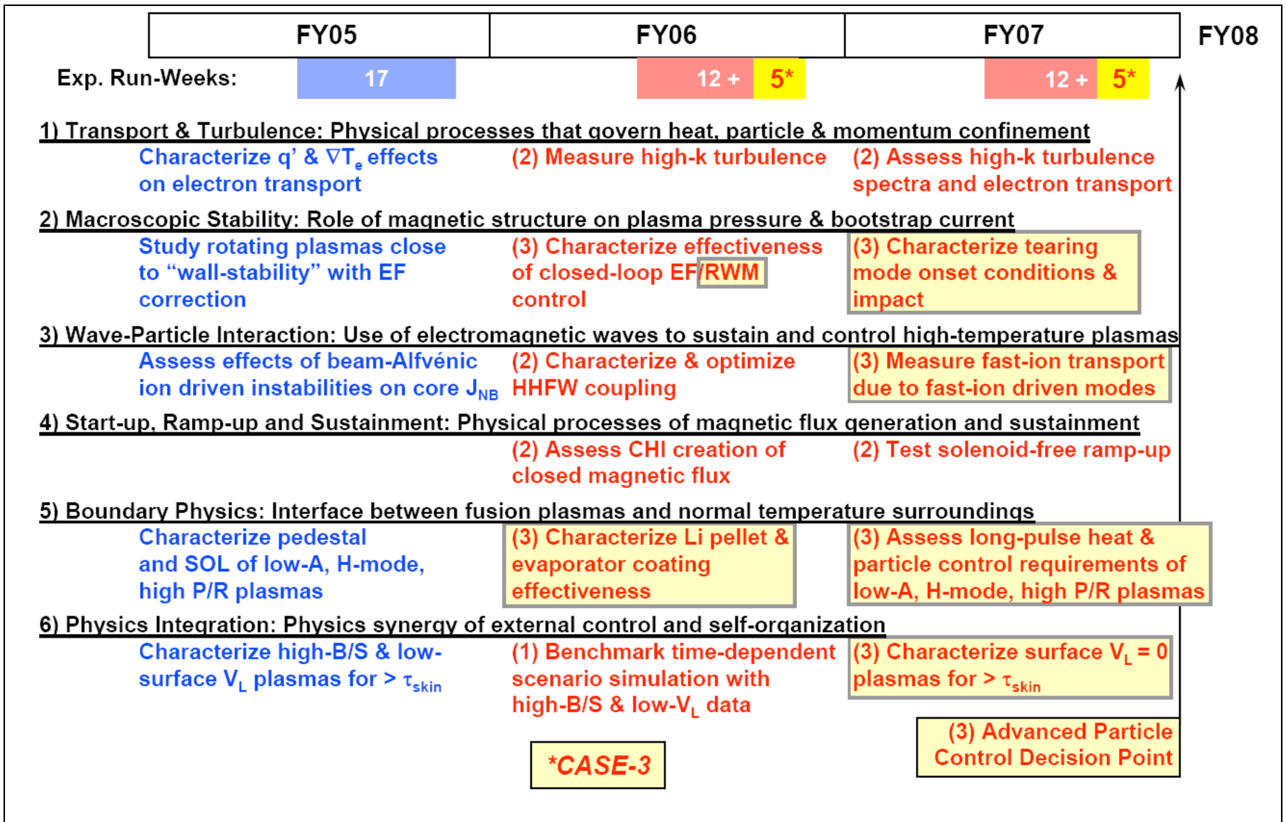

Figure 3. Research milestones for CASE-3 funding for FY2006 and 2007.

B. Macroscopic MHD Stability – Role of magnetic structure on plasma pressure and bootstrap current

Milestone R3(06-2) on Macroscopic Stability: *Characterize the effectiveness of active feedback control of resonant error fields and, with sufficient resources, wall-coupled, pressure-limiting global modes, using closed-loop control of currents in ex-vessel correction coils. (September 2006)*

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, the stabilizing effects on the Resistive Wall Modes (RWMs) from resonant error field correction and strong plasma rotation are expected to change. Active feedback on the amplitude of the resonant error fields and the RWMs is expected to be required to maintain stability of such plasmas. Stabilization of RWMs 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.

Technical Approach – Six ex-vessel field correction coils, driven by Switching Power Amplifiers (SPA), will have been fully commissioned and tested in FY2005. The effectiveness of using feedback control of the resonant field errors, and when appropriate, the wall-coupled, pressure-limiting RWMs, will be characterized over a range of plasma conditions, profiles, shapes, rotation frequencies, and feedback control parameters (current amplitude, toroidal rotation frequency, and phase lag) for the lowest toroidal mode numbers. Of particular interest will be the interplay among plasma rotation, error fields, and RWM as the "wall-stabilized" pressure limit is approached. Extensive data will be compared with a suite of theoretical models (slab models, VALEN, MARS, M3D, etc.) to understand the conditions in which control of the resonant field errors alone can remain effective in maintaining large plasma rotation and suppressing the RWM. The contributions of eddy currents in nearby asymmetric conductors, induced by currents in the field correction coils, will also be clarified. With incremental funding, direct, active feedback control of the RWM will be tested under the conditions where such control is needed to maintain stability, extending beyond the regimes achieved via error field-rotation control. The results will improve the predictive understanding of rotation, error field, and control requirements for future experiments on NSTX and optimization of the ITER burning plasma.

Milestone R3(07-2) on Macroscopic Stability: *Identify modes that tear magnetic field surfaces and limit plasma pressure and energy confinement as the plasma pressure increases toward the "wall-stabilized" limit. (September 2007)*

Description – Progress under incremental funding during FY2005-06 in the study of plasmas approaching the "wall-stabilized" pressure limit will introduce a broad range of plasma beta values in which to investigate MHD modes that tear the magnetic field lines and limit the plasma pressure and energy confinement. The conditions for producing such magnetic tearing activity, including neoclassical tearing modes in the presence of large pressure-gradient driven current, will be studied in high beta, auxiliary-heated plasmas with broad current and pressure profiles. A suite of internal measurements will be augmented by theory and computational modeling to characterize the tearing mode location, frequency, structure, and associated modifications to the current density. This research will establish a basis for tearing mode stabilization requirements using the Electron Bernstein Wave current drive and contribute to predictive understanding of impact of the tearing modes in future burning plasmas such as in ITER.

Technical Approach – Experiments will be carried out in rotating high-performance plasmas well above the "no-wall" pressure limit. External magnetic coils will be used to measure the toroidal and poloidal mode numbers. The amplitude of edge magnetic fluctuations will provides a constraint on the mode amplitude/island width. Measurements of the internal mode structure will discriminate between tearing and kink modes and indicate possible coupling between them. These internal diagnostics include multi-chord multi-time Thomson scattering, a fast scanning EBW radiometer system (a DOE-supported advanced diagnostic development), and soft x-ray cameras. The electron temperature profile evolution and fluctuations so measured will provide a direct indication of the radial mode structure for comparison with theory and simulation for interpretation. For H-mode plasmas the density profile is expected to be relatively flat, likely limiting the spatial resolution of local soft x-ray emission measurements. These data, coupled with the plasma profiles and EFIT equilibrium reconstruction that accounts for large rotation and q-profiles measured via the MSE, will be used to benchmark the predictive capability of theory and simulation codes. These will include a modified Rutherford/delta-prime model, the M3D, NIMROD and PIES simulation codes to assess the non-linear effects of toroidal mode coupling. The results will contribute to a predictive understanding of tearing mode effects on future burning plasmas.

C. Wave-Particle Interactions – Role of electromagnetic waves and modes in sustaining and controlling hot plasmas

Milestone R3(07-3) on Wave-Particle Interactions: *Characterize the diffusion and loss of supra-Alfvénic fast ions due to fast-ion driven oscillations in low-aspect ratio, high-beta plasmas. (September 2007).*

Description – Diffusion and loss of supra-Alfvénic fast ions, substantially enhanced by the so-called "Alfvén Eigenmodes" driven by the fast ions themselves, may limit the performance of burning plasmas in such devices as ITER. Supra-Alfvénic fast ions are introduced in NSTX via neutral beam injection, and therefore allow the key elements of this new physics to be exhibited, measured, and studied directly. NSTX research planned for FY2005-2006 is expected to have identified the plasma conditions, established extensive measurements, benchmarked key theory and simulation codes, and commissioned new collimated neutron detectors to enable this research in FY2007. The results will be compared with larger aspect ratio plasmas to identify and understand possible enhanced diffusion and loss of supra-Alfvénic neutral beam ions and 14-MeV fusion alphas anticipated in ITER. These results are also indispensable in studying NSTX high-performance long pulse plasmas to determine attractive ST applications in advancing the fusion energy sciences.

Technical Approach – Experiments will be performed in high performance regimes where supra-Alfvénic-ion driven instabilities are predicted to enhance fast ion diffusion and loss, as well as in regimes chosen to optimize the effectiveness of the necessary diagnostics. A new collimated neutron detector will be used to measure the profile of neutron emission from which the fast-ion birth profile can be inferred. The Neutral Particle Analyzers (NPA and SSNPA) will be used to measure the distribution of the fast-ion population that remains confined in the plasma, from which the changes in fast-ion distribution caused by the instabilities can be inferred and correlated with the presence of fast-ion driven instabilities. The energy and pitch angle of fast ions lost from the plasma will be measured directly with the Fast Lost Ion Probe (FLIP). The fast-ion driven instabilities will continue to be characterized with diagnostics which include the fast soft x-ray cameras, the heterodyne reflectometers and the multi-channel far infrared tangential interferometer polarimeter (FIRe-TIP). Non-linear simulations, benchmarked during FY2006, will be used to corroborate measurements of the non-linear instabilities driven by supra-Alfvénic fast ions and their impact on fast ion dispersion and loss, over a wide range of plasma operational parameters. The results of this research establishes a strong scientific basis for predictive understanding of the supra-Alfvénic ion properties anticipated in the ITER burning plasmas at normal aspect ratio and in the NSTX high-performance plasmas at low aspect ratio.

E. Boundary Physics – Interface between fusion plasmas and normal temperature surroundings

Milestone R3(06-5) on Boundary Physics: *Characterize the effectiveness of lithium pellet injection and tile coating in controlling fuel recycling from the plasma facing components. (September 2006)*

Description – Experiments and theory of toroidal plasmas have suggested that a controlled reduction of fueling recycled from the plasma facing components can lead to large improvements of the plasma confinement not only near the edge, but also in the core. Reducing and controlling the plasma density is also an important requirement for efficient solenoid-free plasma current startup and sustainment. Lithium coating, already shown on TFTR and CDX-U to produce beneficial effects, has the potential provide an effective control of wall recycling in NSTX. Lithium coating via both pellet injection into helium plasmas and evaporation will be applied before normal plasma pulses to determine the effectiveness of this technique. The results will clarify long pulse heat and particle control requirements on NSTX and contribute to future decisions on advanced particle control.

Technical Approach – The techniques of lithium coating, which allowed high-performance plasmas to be routinely achieved on TFTR, will be tested and developed on NSTX via a sequence of lithium pellet injection and evaporation coating experiments. Starting with plasma-facing surfaces well-conditioned via helium discharges, lithium pellets will be injected into ohmically heated plasmas systematically varied from inboard-limited to fully diverted configurations, to coat the plasma-interacting graphite tiles in NSTX. The effects of activated lithium coating on subsequent deuterium-fueled plasmas in varying configurations will then be documented and characterized. Of key interest in this research are detailed comparisons of fueling rates, deuterium, lithium, and carbon radiation, recycling, diffusion, and migration between discharges with and without lithium coating. A suite of spectroscopic, infrared and soft x-ray diagnostics will be applied for this purpose. The concomitant changes in the plasma core and edge properties will also be measured and documented, for ohmically and neutral beam heated L-mode and H-mode plasmas in inboard-limited and diverted plasma configurations. The results will clarify the location, surface area, and thickness of lithium coating needed to derive substantial benefits of lithium coating on graphite tiles, and guide the operation of a subsequent lithium evaporator aimed at achieving similar benefits more effectively. Successful lithium techniques for plasma wall conditioning and recycling control will benefit plasma performance on NSTX and other major fusion experiments.

Milestone R3(07-5) on Boundary Physics: *Assess the long-pulse plasma conditions and operational requirements of edge heat and particle control of low-aspect ratio, high-confinement, high P/R plasmas. (September 2007)*

Description – Results from Research Milestones in FY2005-06 are expected to acquire the information needed to produce long-pulse high-performance plasmas in NSTX over a range of conditions in the edge shaping, temperature, density, pedestal, and Edge Localized Modes (ELM). The results of this research will establish a basis for assessing the plasma conditions and operational requirements of high edge heat and particle flux control as the pulse length is increased to beyond 1 second in NSTX. The database and understanding so derived will be used in future decisions of an effective approach to control plasma edge heat and particle fluxes in NSTX. Progress in understanding of edge flux properties and drivers under high edge power relative to major radius is of high interest to safe operation of burning plasmas anticipated in ITER.

Technical Approach – Heat and particle fluxes from H-mode, high P/R (up to 8-14 MW/m on NSTX) plasmas will be characterized over a range of ELM conditions, using a suite of plasma edge, scrape-off layer, and plasma facing component measurements. The effects of lithium coating on the edge fluxes will be measured, together with the potential benefits on core plasma properties. The effects and compatibility of a high-density, radiative edge with long-pulse high-performance plasmas will be explored to assess its effectiveness in mitigating the high edge heat fluxes. The effects of the varying ELM conditions will be documented to determine its role in determining these fluxes and affecting the balance between cross-field and parallel transport in the SOL. Operating regimes that show improved plasma core conditions with adequate control of the ELMs and edge fluxes will be characterized and documented. A number of edge heat and particle transport, and instability and turbulence modeling and simulation codes will be applied to interpret the data in more detail and help clarify the mechanisms that drive these fluxes and the conditions of interest. The results will contribute to a predictive understanding of high-performance plasmas fluxes on NSTX and the burning plasmas anticipated in ITER.

F. Physics Integration – Physics synergy of external control and self-organization

Milestone R3(07-6) on Physics Integration: *Characterize strongly shaped low-aspect ratio plasmas with high fractions of bootstrap current and zero toroidal induction voltage (solenoid-free) for durations that allow internal currents to redistribute. (September 2007)*

Description – Conditions will be studied in which the toroidal plasma current necessary for confinement is maintained without magnetic flux induction from the central solenoid for durations beyond the plasma current redistribution time. In this study, strong shaping of the plasma cross section will be combined with neutral-beam and radiofrequency wave power to augment the current produced internally from the pressure gradient. The condition of zero loop voltage at the plasma edge will be studied to enable characterization of the remaining magnetic flux diffusion properties in the plasma core.

Technical Approach – The "bootstrap" current, *i.e.* the current driven internally by the radial pressure gradient in collisionless toroidal plasmas, has been identified to be particularly important in a spherical torus such as the NSTX. Such plasmas would be extended from the high-confinement discharges already produced in NSTX at ~800 kA in which up to 60% of the toroidal current has been sustained by a combination of the bootstrap effect and the current driven by the tangentially injected neutral beams (NBI). In addition, the solenoid-induced loop voltage was reduced to the range of 0.1-0.2 V for durations greater than the current redistribution times, suggesting that the plasma was sustained with little external inductive current. In

FY2006, simulation codes will have been benchmarked with data to suggest combinations of these current drive methods to sustain the plasma current while minimizing the need for electromagnetic induction from the central solenoid. Enhanced shaping of the plasma cross-section, improvements in the confinement, and optimization of the plasma profiles will be applied to match the driven current profile with that required for MHD stability at the high pressure required to enhance the bootstrap current. Possible synergistic effects between the current drive mechanisms will be investigated to determine the optimal plasma operation scenarios in this investigation.

Future Facility and Diagnostic Accomplishments

Facility and Diagnostics Plan for FY 2005

Facility Milestone F(05-1): Operate NSTX Facility for 17 Experimental Run Weeks (September 2005)

In FY 2005, the NSTX facility plans to operate for 17 run weeks. The FY 2005 experiments are scheduled to begin in April 2005.

Facility Milestone F(05-2): Implement Resonant Field Correction system (June 2005)

Six external correction-field coils and three fast switching power amplifiers (SPA) will be commissioned to apply time and spatially varying radial correction fields for experiments in FY 2005. In these initial experiments, correction for the dominant resonant field perturbations will be employed using coil current waveforms programmed according to the analysis of measurements made in preceding shots. At a later stage, feedback control based on real-time measurements of resonant field amplification will be implemented.

Facility Milestone F(05-3): Commission a new pair of PF1A poloidal-field coils to produce high-triangularity, high-elongation plasma equilibria (April 2005)

As described in the NSTX Five Year Plan developed in 2003, theoretical analysis and modeling of discharge scenarios in NSTX has shown that achieving the NSTX longer term goal of simultaneous high beta and high bootstrap current will be facilitated by producing plasmas with higher elongation, $\kappa \sim 2.5$, and triangularity, $\delta \sim 0.6 - 0.8$, of the poloidal cross-section. Calculations show that such strongly shaped plasmas require a new pair of PF1A upper and lower coils because the previous set of PF1A coils was optimized for operation with $\kappa \sim 2$ and a significant loss of plasma volume occurs for higher κ. In view of its critical importance to the NSTX research plan, a decision was made to accelerate the PF1A coil upgrade by one year. A new pair of PF1A coils is being fabricated and installed during the outage period following the FY 2004 experimental run and will be made available for the FY 2005 experimental run.

Diagnostic Milestone D(05-1): Install an additional 10 channels for the multi-pulse Thomson scattering system (September 2005)

The present 20 channel multi-pulse Thomson scattering diagnostic has revealed that there are structures in the electron temperature and density profiles which are not satisfactorily resolved. These structures are associated with the formation of transport barriers and occur regularly at the plasma edge during H-mode, but also in the plasma core region. These transport barriers play a crucial role in the plasma confinement as well as the MHD stability through the resulting pressure profiles. Improved spatial resolution is needed to develop a better understanding of these finer structure features and to connect them to theoretical models. Therefore, an additional 10 spatial channels (for a total of 30 channels) will be installed for the MPTS diagnostic. The collection optics and optical fibers to transmit the scattered light to the MPTS diagnostic room are already in place. Additional polychromators, detectors and data acquisition will be fabricated and installed to accomplish this upgrade.

Diagnostic Milestone D(05-2): Install a diagnostic system to measure short-wavelength plasma turbulence by scattering from the plasma density fluctuations (September 2005)

Following preparatory work in FY 2004, a diagnostic system will be installed to measure the spectrum of small-scale plasma density fluctuations with a radial wave number in the range 5 – 20 cm^{-1} at several radial locations near the mid-plane by scattering from beams of 1 mm microwave radiation launched through the plasma. On the basis of theory and code modeling, short-wavelength fluctuations characterized by $k_{\perp}\rho_e \sim 0.3$ (where $k_{\perp}$ is the wave-number perpendicular to the magnetic field and ρ_e is the electron gyro-radius), resulting from plasma instabilities such as the electron-temperature-gradient (ETG) mode, are predicted to play a dominant role in the electron transport in NSTX.

Facility and Diagnostics Plan for FY 2006 – FY 2007

As described in the FY 2006 and FY 2007 Research Plan, for the purpose of this planning exercise, we shall consider the following three cases: Case I - No run weeks in FY 2006 and 12 run weeks in FY 2007 which is our base case; Case 2: 12 run weeks in both FY 2006 and FY 2007; and Case 3: 17 run weeks in both FY 2006 and FY 2007. The facility and diagnostic milestones associated with these three cases are numbered as "F1, F2, F3" and "D1, D2, D3" respectively.

CASE I: No run weeks in FY2006 and 12 run weeks in FY 2007

In FY 2006, since the base budget does not provide for plasma operation, there is no runtime related Facility Milestone. The facility will be maintained and critical spares manufactured. The facility and diagnostics plans take advantage of the relatively long outage period and available engineering and technical manpower to perform some upgrades with high impact. Under this budget, we have given highest priority to those upgrades which would not require significant non-labor funds.

Case 1: Facility Milestones

Facility Milestone F1(06-1): Install lithium evaporator (September 2006)

This upgrade is directed toward controlling the particle recycling to permit sustained advanced ST operation. In FY 2005, experiments will have been conducted with the NSTX lithium pellet injector to assess the effect of lithium coating on edge recycling. If the indications are favorable, a lithium evaporator will be developed and installed in FY 2006 to provide distributed coating of the plasma facing components with lithium.

Facility Milestone F1(06-2): Design and fabricate the components for the symmetric end-feed HHFW antenna system. (May 2006)

The NSTX HHFW system has heated electrons to up to ~ 4 keV and shown promising indications of driving plasma current. The power modulation experiment conducted in 2004 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.

Facility Milestone F1(07-1): Operate NSTX Facility for 12 Experimental Run Weeks (September 2007)

Under the Case 1 budget, the NSTX facility will operate for 12 run weeks in FY 2007.

Facility Milestone F1(07-2): Install the HHFW antenna system with symmetric end feed. (December 2006)

The HHFW symmetric end-feed components fabricated in FY 2006 (milestone F1(06-2)) will be installed and commissioned on NSTX during the early phase of the FY 2007 run.

Case 1: Diagnostic Milestones

Diagnostic Milestone D1(06-1): Complete the in-vessel modification for the edge poloidal rotation diagnostic using charge-exchange recombination emission. (September 2006)

The in-vessel modification will be completed for implementing a poloidal charge-exchange spectroscopy (P-CHERS) diagnostic to measure the poloidal rotation in the outer half of the minor radius. This diagnostic will be made operational in an interim configuration when funding becomes available to purchase the initial spectrometer and detector. As additional funds become available, optical fibers, spectrometers and detectors will be added to complete the full P-CHERS diagnostic with improved spatial resolution.

Diagnostic Milestone D1(06-2): Fabricate and install a prototype neutron collimator. (September 2006)

The spatial profile of the DD fusion neutron emission is a sensitive diagnostic for the confinement of the unthermalized ions produced by deuterium NBI heating. It is proposed to install a prototype diagnostic to measure the flux of unscattered DD fusion neutrons on three collimated lines of sight in a poloidal plane through the NSTX plasma cross-section. This will require a large shield to reduce the background from the majority of the neutrons which suffer scattering before escaping from the vessel and coil structure. At a later stage of development, additional channels may be deployed to reveal finer structure. The data from this diagnostic will be compared with modeling of the beam-ion confinement based on classical processes and on the effects of instabilities both in the background thermal plasma and excited by the fast ions themselves.

Diagnostic Milestone D1(07-1): Commission an interim edge poloidal rotation diagnostic using charge-exchange recombination emission. (September 2007)

The poloidal charge-exchange spectroscopy (P-CHERS) diagnostic (milestone D1(06-1)) will be made operational in an interim configuration by purchasing and commissioning the initial spectrometers and detectors. The data from this diagnostic will be used to calculate the contribution of the poloidal flow to the shearing rate which plays a role in the suppression of large scale turbulence that is believed to be responsible for the good ion confinement in NSTX. 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.

Diagnostic Milestone D1(07-2): Prepare next-step fluctuation diagnostic system (September 2007)

NSTX is presently implementing the high-k microwave tangential scattering system, which is designed to measure electron-transport-relevant relatively short wavelength fluctuations, such as ETG modes. The system now being installed will have the capability to measure the spectrum of fluctuations as a function of radial wave number. After the initial assessment in FY 2005 and FY 2006, a decision will be made for the next stage of the fluctuation diagnostic upgrade. There are two main options. One is to extend the high k measurement to the poloidal direction to gain a more complete picture of high k fluctuations relevant to electron transport as theory suggests. Another option is to implement an imaging microwave reflectometer to measure the structure in the poloidal direction of the long-wavelength density fluctuations to assess their role in the plasma energy transport. Long-wavelength turbulent fluctuations, *i.e.*, those with a wavelength perpendicular to the magnetic field much larger than the ion gyroradius, are believed to cause anomalous ion transport in tokamaks. In the ST, these fluctuations may intrinsically be reduced by a combination of reduced instability drive and increased flow-shearing-rate at high beta.

CASE 2: 12 run weeks in both FY 2006 and FY 2007

Case 2: Facility Milestones

Facility Milestone F2(06-1): Operate NSTX Facility for 12 Experimental Run Weeks (September 2006)

Under the Case 2 budget, the NSTX facility will operate for 12 run weeks in FY 2006.

Facility Milestone F2(06-2): Install lithium evaporator (April 2006)

This milestone is described in F1(06-1) but under Case 2, the completion date is accelerated to benefit the experiments in FY 2006.

Facility Milestone F2(06-3): Design and fabricate the components for the symmetric end-feed HHFW antenna system. (September 2006)

This milestone is described in F1(06-2) but under Case 2, the completion date is delayed to accommodate the run time.

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

Under the Case 2 budget, the NSTX facility will operate for 12 run weeks in FY 2007.

Facility Milestone F2(07-2): Install the HHFW antenna system with symmetric end feed. (February 2007)

The HHFW symmetric end-feed components fabricated in FY 2006 (milestone F2(06-3)) will be installed and commissioned on NSTX in preparation for the FY 2007 run.

Case 2: Diagnostic Milestones

Diagnostic Milestone D2(06-1): Install and commission an interim edge poloidal rotation diagnostic using charge-exchange recombination emission. (September 2006)

In addition to completing the in-vessel modification for the poloidal charge-exchange spectroscopy (P-CHERS) diagnostic described in D1(06-2) this diagnostic will be made operational in an interim configuration with a limited number of channels by purchasing and commissioning the initial spectrometers and detectors. This milestone accelerates the commissioning described in D1(07-1).

Diagnostic Milestone D2(06-2): Install an infra-red camera to measure the temperature of plasma-facing surfaces with high time resolution (September 2006)

The present infra-red cameras installed on NSTX provide good spatial resolution of the power fluxes on the divertor tiles and other plasma-facing surfaces, but their time resolution of several milliseconds, is inadequate to resolve the surface temperature excursions during some transient phenomena, such as large ELMs or reconnection events. Imaging with fast visible cameras has revealed that phenomena frequently occur on a sub-millisecond timescale in the plasma edge region. Since such rapid events provide the most challenging conditions for materials, it is planned to install an infra-red camera and image transport system with greatly improved time resolution to be able to extrapolate measurements from NSTX to future devices with much larger power fluxes.

Diagnostic Milestone D2(06-3): Fabricate and install a prototype neutron collimator. (September 2006)

This diagnostic is described in D1(06-2).

Diagnostic Milestone D2(07-1): Prepare next-step fluctuation diagnostic system (September 2007)

This diagnostic is described in D1(07-2).

Diagnostic Milestone D2(07-2): Upgrade the edge poloidal rotation diagnostic to increase its spatial resolution. (September 2007)

The interim poloidal charge-exchange spectroscopy (P-CHERS) diagnostic (milestone D2(06-1)) will be upgraded by purchasing and commissioning additional spectrometers and detectors to increase its spatial resolution. 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.

CASE 3: 17 run weeks in both FY 2006 and FY 2007

This incremental request case aims to restore both facility utilization and 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 in FY2008 and available for operations in FY 2009.

Case 3: Facility Milestones

Facility Milestone F3(06-1): Operate NSTX Facility for 17 Experimental Run Weeks (September 2006)

Under the Case 3 budget, the NSTX facility will operate for 17 run weeks in FY 2006.

Facility Milestone F3(06-2): Complete preliminary design of 1 MW EBW heating and current-drive system, begin site preparation, launcher design and modification of the power supply. (September 2006)

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 2006 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.

Facility Milestone F3(06-3): Design and fabricate the components for the symmetric end-feed HHFW antenna system. (September 2006)

This milestone is retained from F2(06-3) and described in F1(06-2).

Facility Milestone F3(06-4): Install lithium evaporator with between-shots capability (April 2006)

Under the Case 3 budget in FY 2006, the lithium evaporator, described in milestone F1(06-1), will be installed with a dedicated, remotely-controllable carriage mechanism to allow between-shots application of lithium.

Facility Milestone F3(07-1): Operate NSTX Facility for 17 Experimental Run Weeks (September 2007)

Under the Case 3 budget, the NSTX facility will operate for 17 run weeks in FY 2007.

Facility Milestone F3(07-2): Complete 1 MW EBW system site preparation and power supply modifications, and begin the antenna system fabrication. (September 2007)

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 2007, under the Case 3 budget, 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.

Facility Milestone F3(07-3):): Install the HHFW antenna system with symmetric end feed. (February 2007)

This milestone is retained from and described in F2(07-2).

Facility Milestone F3(07-4): Complete conceptual design for advanced power and particle handling (September 2007)

In order to provide the power and particle handling required for non-inductively-sustained, long-pulse discharges, NSTX is pursuing two options: one utilizing liquid 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 liquid lithium surface module (LSM) being developed by the working group on the Applications of Liquid-Plasma Interactions Science and Technology (ALIST), supported by the Virtual Laboratory for Technology (VLT). 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.

Case 3: Diagnostic Milestones

Diagnostic Milestone D3(06-1): Install and commission the edge poloidal rotation diagnostic using charge-exchange recombination emission. (September 2006)

The poloidal charge-exchange spectroscopy (P-CHERS) diagnostic will be installed and commissioned in its full implementation. This diagnostic is described in D1(07-1).

Diagnostic Milestone D3(06-2): Install an infra-red camera to measure the temperature of plasma-facing surfaces with high time resolution (September 2006)

This milestone is retained from and described in D2(06-2).

Diagnostic Milestone D3(06-3): Fabricate and install a prototype neutron collimator. (September 2006)

This milestone is retained from and described in D1(06-2).

Diagnostic Milestone D3(07-1): Commission an additional laser on the multi-pulse Thomson scattering system (September 2007)

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.

Diagnostic Milestone D3(07-2): Prepare next-step fluctuation diagnostic system (September 2007)

This milestone is retained from and described in D1(07-2).

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. For 2005, discussions with the TGs and the ITPA Coordinating Committee (CC) identified 13 (see Table below) high-priority joint experiments in which NSTX will participate. These experiments are within the NSTX programmatic interest in achieving the NSTX goals.

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
Transport Physics	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-16	Small ELM regime comparison	NSTX, MAST, C-Mod
Divertor, Scrape-Off Layer	DSOL-9	Carbon migration and deposition	JET, DIII-D, TEXTOR, AUG, JT-60U, NSTX
	DSOL-15	Comparison of edge "blob" characteristics	NSTX, C-Mod, TJ-II, Tore-Supra, JT-60U
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-6	Error field comparison	C-Mod, NSTX, MAST, TEXTOR, DIII-D, JET
	MDC-9	Fast ion redistribution by fast ion driven Alfvén modes and Alfvén Cascades threshold	JT-60U, JET, AUG, DIII-D, NSTX
Steady State Operation	SSO-2.1	Complete mapping of hybrid scenario	JET, JT-60U, DIII-D, AUG, NSTX

DIII-D, C-Mod, HBT-ET coordination and co-operation: A substantial level of cooperation has begun 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.

An updated report on the coordinated research on NSTX, DIII-D, and C-Mod planned for FY 2005 is under preparation under the auspices of the Fusion Facilities Coordinating Committee (FFCC), and will be available by mid March 2005.

<u>ST Programs in U.S.:</u>

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-II (HIT-II), which is replaced by HIT-Steady Injection (SI) at the University of Washington, and Current Drive Experiment-Upgrade (CDX-U), which will be replaced by Lithium Tokamak Experiment (LTX) at PPPL, will support this goal by exploring the new boundaries in the physical and engineering sciences of the ST. If and 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.

HIT-II: The HIT-II Team at the University of Washington has been a key member of the NSTX Research Team, and made major contributions in the CHI startup research on NSTX. The HIT-II Team contributed directly to the design of the CHI "absorber" improvements, which were implemented in FY 2002, and plan to participate in the experiments to test solenoid-free initiation of substantial plasma current and ramp-up to higher currents at high poloidal beta values during FY2004-2006.

CDX-U: CDX-U (being upgraded to LTX in 2006) and NSTX are working together to establish the scientific basis for a decision on whether to proceed with the design and construction a Liquid Surface Module (LSM) for possible installation on NSTX in the future. The LSM is envisioned to be a joint effort with the Virtual Laboratory of Technology involving researchers from SNL, ANL and UCLA. CDX-U has also been successful in developing a scientific basis for the Electron Bernstein Wave (EBW) emission that preceded NSTX research on EBW emission and heating and current drive, which formed the basis for a future research on assessing the EBW heating and current drive requirements.

PEGASUS: The PEGASUS device at the University of Wisconsin aims to explore the ST plasma regime approaching the extreme low limit of R/a ~ 1.1. Plasma currents more than 100 kA and toroidal average betas more than 20% have already been obtained in PEGASUS with resistive heating alone, indicating the potential physics benefits of this regime. PEGASUS further plans to explore innovative means for rf heating via EBW, solenoid-free current initiation via electron beam injection, and connecting physics with the very low aspect ratio Spheromak plasmas.

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. Substantial cooperation with the ST programs in the former three countries has made major contributions to the research on these experiments and the NSTX.

MAST (U.K.): The Mega-Ampere Spherical Tokamak (MAST, U.K.) started operation in late 1999, and has design features and research capabilities highly complementary to those of NSTX. At present there is no conducting shell close to the plasma edge in MAST, making it particularly suited for testing solenoid-free current initiation using electron cyclotron preheating and vertical magnetic field swing. The combination of large vacuum chamber and internal PF coils further give MAST added flexibility in divertor configuration and boundary physics studies. MAST emphasizes initial operation with NBI heating and ECW-EBW heating and current drive. Large progress on MAST in a number of topical areas, notably the ready access to double-null H-mode plasmas sustained for the duration of NBI, access to H-mode in inboard-limited NBI heated, and double-null ohmically heated H-mode plasmas, has brought substantial benefit to the related research activities on NSTX. MAST is particularly strong in researching edge-divertor physics, considerably advancing the understanding of high heat flux issues in future high-power long-pulse ST experiments. These and other results have benefited the NSTX research program. A substantial number of collaborations and exchanges on a broad number of topics, including several ITPA joint experiments, are planned for FY2005-2007 to derive mutual benefits to the NSTX and MAST research programs (see **ITPA contributions** above).

ST Experiments in Japan and ST Collaborations: There are at present five university-level ST experimental programs in Japan: TST-2, TS-3, and TS-4 at the University of Tokyo, HIST at the Himeji Institute of Technology, and LATE at Kyoto University. Ten fusion researchers from universities in Japan (Himeji Institute of Technology, Hiroshima University, Kyushu-Tokai University, Niigata University, Tsukuba University, University of Tokyo, and Japan Atomic Energy Research Institute) participated in the NSTX research since during the Japan FY2004 (4/04 – 3/05). Topics of cooperative research covered divertor spectroscopy, ion Doppler spectroscopy of CHI plasmas, rf heating, electron Bernstein wave, plasma operation scenarios, 2-fluid equilibrium modeling of ST plasmas with large flows, and solenoid-free initiation of plasma current. Cooperation during 2005-2007 on ST is expected to expand by including new exchanges with researchers from Kyoto University, Kyushu University, and the National Institute for Fusion Studies (NIFS), likely in the form of new ST and other joint projects. These cooperative research activities contribute substantially to the success of research on NSTX as well as to enhancing the already substantial research interest in Japan in ST plasmas.

New ST Experiment in PRC: A new ST experiment, Sino United Spherical Tokamak (SUNIST) was recently brought into operation at the Chin-hwa University near Bei-jin. Its research focus is planned to be in radiofrequency plasma startup, heating and current drive. There are potential mutual benefits in establishing research collaborations between SUNIST and NSTX researchers in these topical areas.

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 two of these countries is likely. This agreement, if and 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's of VLT researchers on the topics of plasma facing material and particle deposition (SNL and ANL). In addition, the NSTX and the ALPS/APEX programs 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 a possible Liquid Surface Module (LSM) on NSTX to enable strong control of deuterium recycling from the wall.

Theory Program: NSTX research has also benefited substantially from cooperation 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 2005 – 07 is shown in Table 1. In the NSTX Budget Summary in Table 1 , the maximum number of run weeks per year is 17 for all three years. This is below the 20 run weeks envisioned in the NSTX Five Year Plan and the 21 run weeks achieved in FY 2004, and represents less than full utilization of the facility, unfortunately. This tight budget also limits availability of non-labor funds, forcing postponement in procuring components for critical diagnostic and facility upgrades. The 17-run-week budgets for FY 2006 and FY 2007 assume constant staff level with an investment level of ~ 10% in facility and diagnostic upgrades. The 12-run-week budgets in FY 2006 and FY 2007 have constant effort and staff levels with minimal upgrades. The three cases considered in the research, facility and diagnostic milestone section correspond to the following budget level combinations:

Case 1 – no run weeks in FY 2006 (Base) and 12 run weeks in FY 2007 (Base);

Case 2 – 12 run weeks in FY 2006 (Request 1) and 12 run weeks in FY 2007 (Base);

Case 3 – 17 run weeks in FY 2006 (Request 2) and 17 run weeks in FY 2007 (Request).

Table 1: NSTX Budget Summary ($M)

	FY 05	FY 06			FY 07	
Budget level	Base	Case1 Base	Case 2 Request 1	Case 3 Request 2	Case1 & 2 Base	Case 3 Request
Run Weeks	17	0	12	17	12	17
Facility Operation	17.34	14.4	17.0	18.0	17.4	18.4
Facility Upgrades	1.10	0.8	0.9	2.0	1.0	2.1
Facility Total	**18.44**	**15.2**	**17.9**	**20.0**	**18.4**	**20.5**
PPPL Research	9.64	9.4	9.8	10.0	10.2	10.3
Diag Upgrades	0.73	0.6	0.8	1.2	0.8	1.1
Coll. Diag. Interf	0.65	0.5	0.6	0.6	0.6	0.7
Collaborations	5.13	5.0	5.1	5.5	5.4	5.5
Science Total	**16.15**	**15.5**	**16.3**	**17.2**	**17.0**	**17.6**
NSTX Total	**34.59**	**30.7**	**34.2**	**37.2**	**35.4**	**38.1**

Under the \$3.8 M cut proposed in the FY 2006 base budget, relative to FY 2005, NSTX will not be able to operate in FY 2006. Under this budget, NSTX will utilize its core engineering and technical staff to perform necessary maintenance, repairs, and some key upgrades to prepare for experiments in FY 2007. The budgets presented in Table 1 with 12 run weeks in both FY 2006 (Request 1 budget) and FY 2007 (Base budget) give highest priority to plasma operation in NSTX but allow only modest upgrades and facility maintenance. The budgets presented in Table 1 with 17 run weeks in FY 2006 (Request 2 budget) and FY 2007 (Request budget) do allocate 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, the 17-run-week plans would accelerate the 1 MW EBW upgrade to be built over 3 years (to be available in FY 2008) instead of 5 years (available in FY 2010) under the base plan. Other important and timely upgrades possible within the 17-run-week budgets 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. The FY 2006 base budget without plasma operation will utilize facility personnel to work on spare component fabrication (such as NBI ion source and OH solenoid). Also in Table 2 below, we show a top-down facility utilization emphasis estimate for various levels of run time in FY 2007 as requested for the Budget Planning Meeting presentation. This budget does not include major upgrades such as the high power EBW system.

FY 2005 Budget - In FY 2005, the base budget enables the NSTX facility to operate for 17 run weeks and to implement new facility and diagnostic upgrades including 1) the EF/RWM (Error Field/Resistive Wall Mode) coil system powered by Switching Power Amplifiers (SPA) for stable operation near the ideal MHD wall limit, 2) new PF1A coils for improving shape control at high elongation, 3) an additional ten channels (to 30 total) for the multi-pulse Thomson scattering system to provided detailed profile information in the H-mode pedestal and the ITB regions, and 4) the high-k tangential microwave scattering system to measure short wavelength plasma fluctuations, such as electron temperature gradient modes (ETGs), which could be causing the electron energy transport. Additionally, real-time deposition monitors will be implemented to measure in-situ surface deposition during plasma operation on NSTX; this is a critical physics and diagnostic issue for future devices such as ITER and NSST. The planned 17 run weeks 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. The FY 2005 NSTX run is scheduled to conclude in September 2005. The FY 2006 NSTX outage will start in October 2005.

FY 2006 Base Budget – Due to the significant budget reduction of \$3.8 M in the FY 2006 base budget, the NSTX facility will not operate in FY 2006. The \$3.8 M reduction from the FY 2005 budget is greater than the entire non-labor budget of about $3 M for the 12 run week budget in FY 2006. Together with the effect of inflation, equivalent to an additional expense of about $1 M, the budget reduction will therefore adversely impact the experimental research output of NSTX in FY 2006. The outage period will begin at the start of FY 2006 and the facility will be maintained and repaired as needed to be ready to operate in FY 2007. The critical core staff will perform necessary maintenance and repairs, and will implement select upgrades during the outage period. The researchers will perform post-run diagnostic calibrations, analyze data generated during the FY 2005 run, attend key meetings, publish in conference proceedings and refereed journals, participate in collaborative experiments related to the ITPA, and apply their expertise to critical ITER physics tasks. If additional funding were to become available in FY 2006, operation could be resumed earlier, or the FY 2005 run could be extended into FY 2006.

<u>FY 2006 Base Facility Upgrade Budget</u> - The NSTX team is planning to make significant progress in three key facility upgrade areas. 1) The 1 MW EBW system preliminary design will be completed and some select site preparation will begin. The 1 MW EBW system is designed to demonstrate the capability for off-axis current drive, via the “Ohkawa” current drive effect, which is needed to achieve the high beta, ~40%, high bootstrap fraction, ~60%, fully non-inductively sustained, advanced ST discharges which are a major goal of the NSTX Five–Year Plan. 2) The lithium evaporator, which is currently under development on CDX-U, will be installed on NSTX with the capability for daily application of lithium. This upgrade is directed toward testing the effectiveness of lithium wall coating for controlling the particle recycling needed for sustained advanced ST operation. 3) The design and component fabrication of the HHFW antenna for symmetric end feeds will be performed to provide more reliable operation at 6 MW for electron heating and current drive needed for current start-up, ramp-up and maintenance. This upgrade can be performed by the team with minimal non-labor funds since the most costly items needed for this upgrade, the 12 high power feed-thrus, are already available.

<u>FY 2006 Base Diagnostic Upgrade Budget</u> - In FY 2006, two modest diagnostic upgrades will be undertaken. 1) For the poloidal CHERS diagnostic, the in-vessel work, including the passive plate and divertor plate modifications will be performed, and the fiber bundles and related optical components will be installed. An interim P-CHERS system with a limited number of spatial channels will then become operational when the first spectrometer and detector can be obtained, either through a collaborative arrangement or procurement when the funds become available, as in the requested budgets. 2) To quantify the energetic particle population and to improve the power balance analyses, a prototype neutron collimator system will also be built and installed in FY 2006.

12 Run Week Budget for FY 2006 and FY 2007 - The twelve-run-week budgets for FY 2006 and FY 2007 are relatively restricted budgets under which the highest priority is facility operation and only modest upgrades and maintenance are undertaken. The 12 run week in FY 06 would insure steady experimental progress in the NST research and therefore clearly a preferred option compared to the no run weeks base plan. The upgrade budget is only about 4%, low even compared to the historical values. Only upgrades which are of immediate benefit to the run will be implemented. The HHFW symmetric antenna end-feed upgrade will be completed in time to support FY 2007 run. The lithium evaporator 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 2006 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 due to the importance of the poloidal flow measurements in determining the current profile with the MSE-CIF diagnostic.

Explanation of 17 Run Week Budgets for FY 2006 – FY 2007

The incremental funding proposed in FY 2006 and FY 2007 will be used to restore the NSTX run weeks toward full utilization of 17 weeks in both years and to implement key facility and diagnostic upgrades. The 17 run weeks will advance the NSTX research program at a relatively healthy pace. The enhancements to the FY 2005-2006 research milestones under these budgets are described in the FUTURE ACCOMPLISHMENTS section. The enhanced budget for upgrades will enable NSTX to make timely progress toward its established research goals. The incremental funding will also enable preventive maintenance, the purchase of spare parts, and improvements in operations to keep the facility availability and reliability at high levels.

The incremental funding requested would permit implementation of key facility upgrades, which will significantly increase the chance of success of the NSTX Five Year Research Plan. In addition to the upgrade activities described in the 12-run-week plan, if an assessment at the end of FY 2005 for an EBW system on NSTX is favorable, the proposed incremental 17-run-week budget in FY 2006 and FY 2007 would permit a timely ramp-up for installing the 1 MW EBW system to be ready for experiments in FY 2009. This system should produce a significant localized driven current of about 35 kA which would be measurable with the MSE diagnostic. This level of current induced locally can also contribute toward the investigation of NTM stabilization.

In the area of particle control, efficient core fueling is important to reduce the particle recycling and inventory. The incremental funding will permit the implementation of advanced fueling tools. In the area of solenoid-free start-up, incremental funding will permit investigating an alternative start-up concept to CHI utilizing the outer poloidal field coils.

Incremental funding is requested for both in FY 2006 and FY 2007 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 NSTX Five Year Plan. The knowledge thus developed will also contribute to the design of plasma facing components (PFCs) for future devices. The incremental fund will also enable installation of a third laser for MPTS in FY 2007 to increase the time resolution of the Thomson scattering system to improve the determination of the electron power flow in the plasma. To advance fluctuation studies, a decision will be made at the end of FY 2006 on the design of a next-step fluctuation diagnostic upgrade. This could be an improvement to the high-k scattering system (presently being installed) or a new novel microwave imaging diagnostic for low-k fluctuations. Under the enhanced budget, the chosen upgrade will be designed and fabricated in FY 2007 and be installed in FY 2008.

"Facility Utilization Emphasis" Budget Cases for FY 2007

As requested in the Budget Planning Meeting instructions Table 2 provides top-down budget estimates for the 0, 6, 12, 16, 20, and 25 run week cases for NSTX in FY 2007. These budgets emphasize facility utilization and do not include major facility upgrades such as the high power EBW system. The FY 2007 no run-week case assumes a 10% reduction in the over all budget from the 12 run week baseline case for FY 2007. Since the NSTX labor budget is about 90% of the total for the 12 run week budget, and also represents a significant fraction of the total laboratory labor, such a drastic cut will result in higher laboratory G&A rates to support the basic laboratory infrastructure needs. Therefore the NSTX labor in FY2007 will have to be reduced by approximately 13% to meet the 10% budget reduction target. The loss of experienced personnel from NSTX, and possibly from PPPL, could inflict significant damage on the NSTX and laboratory technical infrastructure. Not operating the NSTX research facility in FY 2007 will result in one year delay in the experimental research milestones. The 10% cut would have an equally severe impact for the collaborating groups.

Table 2: FY 2007 Budget for Various Run Weeks ($M)

Run Week	0	6	12	16	20	25
Facility	15.9	17.8	18.5	19.0	19.5	20.2
Science	16.0	16.8	16.9	17.1	17.4	17.8
NSTX Total	31.9	34.6	35.4	36.1	36.9	38.0

RESEARCH MILESTONES - Baseline/CASE-1

Milestone	Description	Baseline	Forecast	Actual
FY 2004				
R(04-1)	Assess confinement and stability in NSTX by characterizing high confinement regimes with edge barriers and by obtaining initial results on the avoidance or suppression of plasma pressure limiting modes in high-pressure plasmas.	Sep 04		Sep 04
R(04-2)	Measure long wavelength turbulence in spherical torus plasmas	Sep 04		Sep 04
R(04-3)	Measure plasma current profile modifications produced by radiofrequency, neutral beam injection, and pressure-gradient techniques	Sep 04		Sep 04
R(04-4)	Conduct initial tests combining available techniques to achieve solenoid-free initiation to substantial plasma currents.	Sep 04		Sep 04
R(04-5)	Measure Electron Bernstein Wave (EBW) emissions to assess heating and current drive requirements.	Sep 04		Sep 04
FY2005				
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		
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		
R(05-3)	Assess the effects of supra-Alfvénic fast ion driven instabilities on driven current in the plasma core	Sep 05		

RESEARCH MILESTONES – Baseline/CASE-1 (continued)

Milestone	Description	Baseline	Forecast	Actual
FY 2005				
R(05-4)	Characterize the plasma edge pedestals and scrape-off layer of low-aspect ratio, high confinement, high P/R plasmas	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		
FY 2006				
R1(06-1)	Compare 2005 data on error field, resistive wall mode (RWM), and locked mode properties with theoretical models of error field and MHD mode stability conditions that apply to low and high aspect ratios	Sep 06		
R1(06-2)	Compare data on supra-Alfvénic fast ion driven modes with non-linear simulations of these modes	Sep 06		
R1(06-3)	Benchmark and improve physics models and the time-dependent simulation codes with the FY2005 data from high-performance plasmas characterized by large self-driven current, high pressure relative to the applied toroidal field, and low toroidal induction voltage	Sep 06		

RESEARCH MILESTONES – Baseline/CASE-1 (continued)

Milestone	Description	Baseline	Forecast	Actual
FY 2007				
R1(07-1)	Measure short wavelength turbulence in the plasma core in a range of plasma conditions	Sep 07		
R1(07-2)	Characterize the effectiveness of active feedback control of resonant error fields and, with sufficient resources, wall-coupled, pressure-limiting global modes, using closed-loop control of currents in ex-vessel correction coils	Sep 07		
R1(07-3)	Characterize the interaction between the edge plasma region and the launched High Harmonic Fast Waves (HHFW), and determine plasma conditions that permit efficient heating and current drive via HHFW	Sep 07		
R1(07-4)	Assess the conditions in which a substantial amount of closed poloidal magnetic flux is created via Coaxial Helicity Injection	Sep 07		

FACILITY MILESTONES - Baseline/CASE-1

Milestone	Description	Baseline	Forecast	Actual
FY 2004				
F(04-1)	Operate NSTX Facility for 18 experimental run weeks	Sep 04		Aug 04
F(04-2)	Fabricate resonant field correction coil system	Sep 04		Sep 04
F(04-3)	Implement capacitor bank for transient-CHI Start-up	May 04		Jul 04
FY 2005				
F(05-1)	Operate NSTX Facility for 17 Experimental Run Weeks	Sep 05		
F(05-2)	Implement Resonant Field Correction system	Jun 05		
F(05-3)	Commission a new pair of PF1A poloidal-field coils to produce high-triangularity, high-elongation plasma equilibria	Apr 05		
FY 2006				
F1(06-1)	Install lithium evaporator	Sep 06		
F1(06-2)	Design and fabricate the components for the symmetric end-feed HHFW antenna system	May 06		
FY 2007				
F1(07-1)	Operate NSTX Facility for 12 Experimental Run Weeks	Sep 07		
F1(07-2)	Install the HHFW antenna system with symmetric end feed	Dec 06		

DIAGNOSTIC MILESTONES - Baseline/CASE-1

Milestone	Description	Baseline	Forecast	Actual
FY 2004				
D(04-1)	Install and operate a 10 channel Motional Stark Effect (MSE) diagnostic based on the collisionally induced fluorescence (CIF) from the heating neutral beams	Sep 04		Jul 04*
D(04-2)	Install a fast camera system to provide two dimensional images of the soft x-ray emission viewed along tangential sightlines.	Apr 04		Feb 04
D(04-3)	Prepare for installing the new poloidal charge-exchange spectroscopy (P-CHERS) diagnostic and for upgrading the time and spatial resolution of the edge rotation diagnostic by fabricating new passive stabilizer plates to accommodate the viewing slots and by procuring the fiber optics and the fast two-dimensional optical detector arrays.	Sep 04	**	
D(04-4)	Assemble and test microwave sources and other components for a diagnostic system to measure short-wavelength plasma turbulence by scattering from the plasma density fluctuations.	Sep 04		Sep 04

* *The NSTX MSE/CIF started taking data to contribute to the current profile science milestone with eight spatial channels using four available detectors.*

** *The milestone postponed due to a programmatic decision to accelerate the PF 1A coil. See P-CHERS related milestones in FY 2006 –FY 2007*

DIAGNOSTIC MILESTONES – Baseline/CASE-1 (continued)

Milestone	Description	Baseline	Forecast	Actual
FY 2005				
D(05-1)	Install an additional 10 channels for the multi-pulse Thomson scattering system	Sep 05		
D(05-2)	Install a diagnostic system to measure short-wavelength plasma turbulence by scattering from the plasma density fluctuations	Sep 05		
FY 2006				
D1(06-1)	Complete the in-vessel modification for the edge poloidal rotation diagnostic using charge-exchange recombination emission	Sep 06		
D1(06-2)	Fabricate and install a prototype neutron collimator	Sep 06		
FY 2007				
D1(07-1)	Commission an interim edge poloidal rotation diagnostic using charge-exchange recombination emission	Sep 07		
D1(07-2)	Prepare next-step fluctuation diagnostic system	Sep 07		

RESEARCH MILESTONES – CASE-2

Milestone	Description	Baseline	Forecast	Actual
FY2006				
R2(06-1)	Measure short wavelength turbulence in the plasma core in a range of plasma conditions	Sep 06		
R2(06-2)	Characterize the effectiveness of active feedback control of resonant error fields using closed-loop control of currents in ex-vessel correction coils	Sep 06		
R2(06-3)	Characterize the interaction between the edge plasma region and the launched High Harmonic Fast Waves (HHFW), and determine plasma conditions that permit efficient heating and current drive via HHFW	Sep 06		
R2(06-4)	Assess the conditions in which a substantial amount of closed poloidal magnetic flux is created via Coaxial Helicity Injection	Sep 06		
R2(06-5) *(retained from CASE-1* **R1(06-3)***)*	Benchmark and improve physics models and the time-dependent simulation codes with the FY2005 data from high-performance plasmas characterized by large self-driven current, high pressure relative to the applied toroidal field, and low toroidal induction voltage	Sep 06		

RESEARCH MILESTONES – CASE-2 (continued)

Milestone	Description	Baseline	Forecast	Actual
FY2007				
R2(07-1)	Assess the correlation between measured and calculated high-k turbulence spectra, and the measured electron thermal conductivity	Sep 07		
R2(07-2)	Characterize the effectiveness of active feedback control of wall-coupled, pressure-limiting global modes, using closed-loop control of currents in ex-vessel correction coils	Sep 07		
R2(07-3)	Intentionally left blank (reserved for R3(07-3)			
R2(07-4)	Test conditions for solenoid-free ramp-up of plasma to substantial plasma current	Sep 07		
R2(07-5)	Characterize the effectiveness of lithium pellet injection and tile coating in controlling fuel recycling from the plasma facing components	Sep 07		

FACILITY MILESTONES – CASE-2

Milestone	Description	Baseline	Forecast	Actual
FY2006				
F2(06-1)	Operate NSTX Facility for 12 Experimental Run Weeks	Sep 06		
F2(06-2)	Upgrade lithium evaporator	Apr 06		
F2(06-3)	Design and fabricate the components for the symmetric end-feed HHFW antenna system	Sep 06		
FY2007				
F2(07-1)	Operate NSTX Facility for 12 Experimental Run Weeks	Sep 07		
F2(07-2)	Install the HHFW antenna system with symmetric end feed	Feb 07		

DIAGNOSTIC MILESTONES – CASE-2

Milestone	Description	Baseline	Forecast	Actual
FY2006				
D2(06-1)	Install and commission interim edge poloidal rotation diagnostic using charge-exchange recombination emission	Sep 06		
D2(06-2)	Install an infra-red camera to measure the temperature of plasma-facing surfaces with high time resolution	Sep 06		
D2(06-3)	Fabricate and install a prototype neutron collimator	Sep 06		
FY2007				
D1(07-1)	Prepare next-step fluctuation diagnostic system	Sep 07		
D2(07-2)	Upgrade the edge poloidal rotation diagnostic to increase its spatial resolution	Sep 07		

RESEARCH MILESTONES – CASE-3

Milestone	Description	Baseline	Forecast	Actual
FY2006				
R3(06-1) *(retained from CASE-2* **R2(06-1)***)*	Measure short wavelength turbulence in the plasma core in a range of plasma conditions	Sep 06		
R3(06-2)	Characterize the effectiveness of active feedback control of resonant error fields and, with sufficient resources, wall-coupled, pressure-limiting global modes, using closed-loop control of currents in ex-vessel correction coils	Sep 06		
R3(06-3) *(retained from CASE-2* **R2(06-3)***)*	Characterize the interaction between the edge plasma region and the launched High Harmonic Fast Waves (HHFW), and determine plasma conditions that permit efficient heating and current drive via HHFW	Sep 06		
R3(06-4) *(retained from CASE-2* **R2(06-4)***)*	Assess the conditions in which a substantial amount of closed poloidal magnetic flux is created via Coaxial Helicity Injection	Sep 06		
R3(06-5)	Characterize the effectiveness of lithium pellet injection and tile coating in controlling fuel recycling from the plasma facing components	Sep 06		
R3(06-6) *(retained from CASE-1* **R1(06-3)***)*	Benchmark and improve physics models and the time-dependent simulation codes with the FY2005 data from high-performance plasmas characterized by large self-driven current, high pressure relative to the applied toroidal field, and low toroidal induction voltage	Sep 06		

RESEARCH MILESTONES – CASE-3 (continued)

Milestone	Description	Baseline	Forecast	Actual
FY2007				
R3(07-1) *(retained from CASE-2* **R2(07-1)***)*	Assess the correlation between measured and calculated high-k turbulence spectra, and the measured electron thermal conductivity	Sep 07		
R3(07-2)	Identify modes that tear magnetic field surfaces and limit plasma pressure and energy confinement as the plasma pressure increases toward the "wall-stabilized" limit	Sep 07		
R3(07-3)	Characterize the diffusion and loss of supra-Alfvénic fast ions due to fast-ion driven oscillations in low-aspect ratio, high-beta plasmas	Sep 07		
R3(07-4) *(retained from CASE-2* **R2(07-4)***)*	Test conditions for solenoid-free ramp-up of plasma to substantial plasma current	Sep 07		
R3(07-5)	Assess the long-pulse plasma conditions and operational requirements of edge heat and particle control of low-aspect ratio, high-confinement, high P/R plasmas	Sep 07		
R3(07-6)	Characterize strongly shaped low-aspect ratio plasmas with high fractions of bootstrap current and zero toroidal induction voltage (solenoid-free) for durations that allow internal currents to redistribute	Sep 07		

FACILITY MILESTONES – CASE-3

Milestone	Description	Baseline	Forecast	Actual
FY2006				
F3(06-1)	Operate NSTX Facility for 17 Experimental Run Weeks	Sep 06		
F3(06-2)	Complete preliminary design of 1MW EBW heating and current-drive system, begin site preparation, launcher design and modification of the power supply	Sep 06		
F3(06-3)	Design and fabricate the components for the symmetric end-feed HHFW antenna system	Sep 06		
F3(06-4)	Install lithium evaporator with between-shots capability	Apr 06		
FY2007				
F3(07-1)	Operate NSTX Facility for 17 Experimental Run Weeks	Sep 07		
F3(07-2)	Complete 1 MW EBW system site preparation and power supply modifications, and begin the antenna system fabrication	Sep 07		
F3(07-3)	Install the HHFW antenna system with symmetric end feed	Feb 07		
F3(07-4)	Complete conceptual design for advanced power and particle handling	Sep 07		

DIAGNOSTIC MILESTONES – CASE-3

Milestone	Description	Baseline	Forecast	Actual
FY2006				
D3(06-1)	Install and commission the edge poloidal rotation diagnostic using charge-exchange recombination emission	Sep 06		
D3(06-2)	Install an infra-red camera to measure the temperature of plasma-facing surfaces with high time resolution	Sep 06		
D3(06-3)	Fabricate and install a prototype neutron collimator	Sep 06		
FY2007				
D3(07-1)	Commission an additional laser on the multi-pulse Thomson scattering system	Sep 07		
D3(07-2)	Prepare next-step fluctuation diagnostic system	Sep 07		