Chapter 7 - The NSTX National Fusion Facility Status and Upgrades

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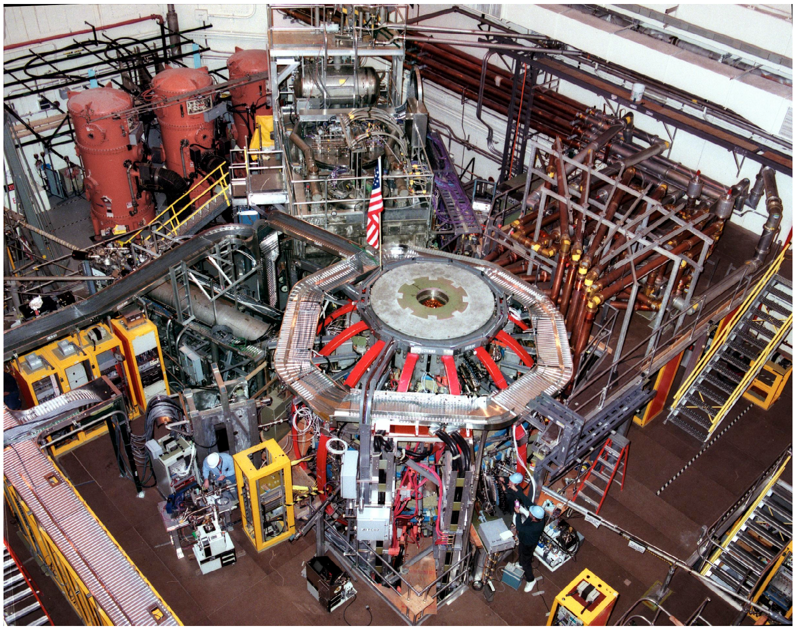


Chapter 7

The NSTX National Fusion Facility - Status and Upgrades

The National Spherical Torus Experiment (NSTX) is the world's leading spherical tokamak, combining an exceptionally wide plasma parameter space, a high degree of facility flexibility, and state-of-the-art diagnostic systems. Presently, the NSTX plasmas are heated by up to 7 MW of deuterium Neutral Beam Injection (NBI) and up to 6 MW High-Harmonic Fast Waves (HHFW). In addition to advanced plasma shaping capabilities, NSTX is equipped with a set of non-axisymmetric control coils to enhance plasma stability and is exploring and developing the use of lithium as a plasma facing material. Attracted by its unique capabilities, NSTX hosts a large number of national and international researchers (over 200 annually) with 55 institutions participating in the research program. Nationally, the team members are from 29 universities, national laboratories, and industries.

Major upgrades for 2009 include a liquid lithium divertor target to access lower collisionality, a modification to the HHFW antenna to double its power capability, and a Beam Emission Spectroscopy diagnostic to extend the localized measurements of plasma turbulence toward the ion gyro-radius scale. For the longer term, a new center stack is planned to enable operation at a magnetic field up to 1 T, plasma current 2 MA with a 5 s pulse length, along with a second neutral beam system for current profile control and a divertor upgrade to handle the expected heat-flux for the full pulse. These upgrades are aimed at achieving fully non-inductively sustained, long-pulse high-performance operation and exploring an expanded plasma parameter space in terms of higher plasma temperature and lower collisionality. The new physics regimes made accessible by these upgrades will significantly reduce the gap, and thus the uncertainty in extrapolating, from the present NSTX to projected next-step ST experiments.

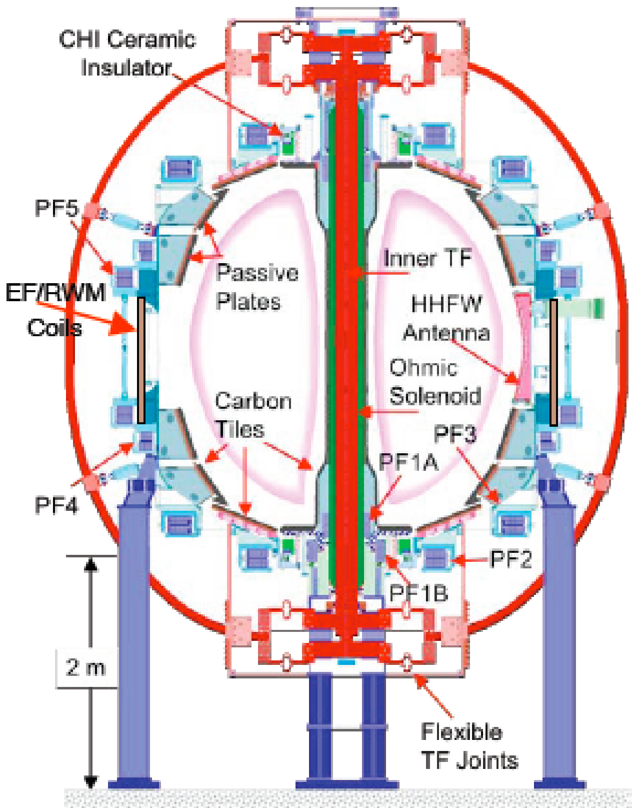


Fig. 7.1 NSTX Device Schematic

Present Machine Capabilities
Major Radius 0.85 m
Minor Radius 0.68 m
Elongation 1.8 - 3.0
Triangularity 0.2 - 0.8
Plasma Current
1 MA (1.4 MA peak)
Toroidal Field
0.35 - 0.55 T
Heating and CD
7 MW NBI (2 sec)
5 MWNBI (5 sec)
6 MW HHFW (5 sec)
0.2 MA CHI
Pulse Length
~ 1 sec at 0.55 T
~ 2 sec at 0.38 T

Table 7.1 Facility Parameters

7.1 Facility Overview

NSTX is a major component of the restructured U.S. Fusion Energy Sciences Program, which encompasses both the investigation of innovative confinement concepts and the advancement of the underlying physics to strengthen the scientific basis for attractive fusion power. Operational since 1999 [7.1.1-3], the NSTX has been steadily building up its facility and diagnostic capabilities. To accomplish this mission, the NSTX facility, shown schematically in Fig. 7.1 and summarized in terms of major device parameters in Table 7.1, provides the following capabilities:

- Low aspect ratio, $R/a \geq 1.27$, and a strongly shaped plasma cross-section with nominal elongation $\kappa \leq 3.0$ and triangularity $\delta \leq 0.8$;
- Close-fitting conducting plates to stabilize pressure-driven instabilities at high plasma β;
- Toroidal field (TF) coil operating routinely at up to 0.55 T;
- Plasma current $I_p \sim 0.75$ MA for ~ 1.8 s duration, limited by the TF coil pulse length, and up to 1.4 MA transiently, exceeding the toroidal magnet current;

- Neutral Beam Injection (7 MW) and High-Harmonic Fast-Wave system (6 MW) for heating, current drive, and current profile control;
- Transient Coaxial Helicity Injection (CHI) for solenoid-free startup current;
- Dual lithium evaporator system for ELM suppression, improved electron confinement, and improved HHFW and electron Bernstein wave coupling;
- Error Field Correction and Resistive Wall Mode feedback coils to stabilize MHD modes occurring at high plasma beta.

The NSTX national team members, many supported by competitive, peer-reviewed research grants, bring to NSTX highly valuable expertise, major plasma diagnostics, and computational tools, including data analysis packages and modeling codes. A number of research collaborators are directly supported by theory programs and the Virtual Laboratory for Technology (VLT). Internationally, there are collaborating researchers from 26 institutions in Japan, Korea, UK, Italy, France, Germany, Israel, the Czech Republic, Canada, Ukraine, and Russia. Most are researchers working on the ST concept in their respective countries but there area number from tokamak and stellarator laboratories, such as LHD, KSTAR, and JT-60U. The collaboration with Japan is the largest, involving nine universities and two national laboratories because, while there are a number of small-scale ST research facilities in Japan, none is comparable with NSTX.

The NSTX engineering and technical support team brings valuable skills, knowledge, and experience in magnetic fusion engineering, experimental research, and plasmas diagnostics. NSTX taps resources from rest of the Princeton Plasma Physics Laboratory including engineering designers, and versatile machine and electronic shops.

This NSTX Five Year Plan proposes significant facility upgrades, which together with the planned innovative diagnostic systems will provide the data needed to achieve the NSTX Five Year Plan Mission and in particular strengthen the database needed for the design of next-step ST facilities. Overview plans for these facility and diagnostic upgrades are summarized in Sec. 7.2 and 7.3. The proposed new center-stack(7.2.7) and the second NBI (7.2.8) are significant upgrades which will support all science topical areas of interest in NSTX. Accordingly they are discussed in greater detail in this plan. It should be noted that the schedule for their implementation is determined by the availability of the required funding.

7.2 NSTX Facility Status and Plans

The NSTX device is located in a well-shielded test cell at PPPL and utilizes many of the former TFTR facilities, notably the power systems which provide reliable operation of the magnets and auxiliary heating and current drive systems. The NSTX facility has commissioned all the major subsystems originally planned, and achieved or exceeded its original design capabilities. The innovative center-stack design (Fig. 7.1), together with a state-of-the-art plasma control system has yielded plasma elongations up to 3.0 transiently and 2.7 sustained, and the highest shaping factor $q_{95}(I_p/aB_T) \approx 42$ MA/m·T of any toroidal device. This extended operating space has produced a world record toroidal beta of ~40% in a high temperature plasma. The NSTX plasma current has reached 1.5 MA, well above the original design value of 1 MA. The plasma current has exceeded the total toroidal field coil current, demonstrating the efficient magnetic field utilization of the ST configuration. The NBI system has delivered up to 7 MW total power by increasing the acceleration voltage of some sources to 100 kV. The energy confinement time in NSTX has exceeded 100 ms, and the plasma stored energy has reached 400 kJ.

Since its first plasma in 1999, NSTX has achieved its facility and operational milestones, except in 2003 when a joint in the toroidal field (TF) coil failed, causing a loss of 8 run weeks. A new center conductor bundle for the TF coil incorporating improvements to the joints was installed before the 2004 run. [7.2.1] The TF coil joints have since operated reliably and within design specifications for over 9000 pulses, routinely accessing its upper B_T range of 5.5 kG.

The NSTX plasma is surrounded by closely-fitting conducting plates (shown in Fig. 7.1), which passively stabilize pressure-driven modes, provided the plasma toroidal rotation exceeds a critical frequency. As a result, NSTX routinely produces plasmas with total pressure above that which cause MHD instability in the absence of the conducting wall, the so-called “no-wall beta limit”. To explore plasma operation with beta above the no-wall limit and approaching the limit for stabilization even by an ideal conducting wall, the “ideal-wall limit”, a fast feedback coil system, known as the Error Field and Resistive Wall Mode (EF/RWM) system, was commissioned at the start of the 2005 experimental run. The NSTX EF/RWM coil system has been highly effective in exploring and extending the stable high beta regimes. Furthermore it is contributing to the design of the system for ITER, since the NSTX coil locations are similar to those proposed for ITER and, in ITER, the tritium blanket modules on the outboard side will slow down the instability growth similarly to the NSTX passive plates. NSTX EF/RWM research benefits from a comprehensive array of magnetic sensors, the data from which can be processed in real time to generate time-varying corrections of both intrinsic and plasma-generated field perturbations.

The NSTX plasma is highly accessible due to the large number and size of diagnostic access ports. A comprehensive suite of diagnostics is now operational, many of them unique in their capabilities, providing physics information with excellent accuracy and resolution. Because the large mid-plane ports are close to the plasma and because of the compact outer TF coils, NSTX provides tangential access for a wide range of diagnostics. The strong toroidicity and excellent diagnostics make NSTX an excellent test bed for toroidal plasma theory and modeling to increase our confidence in the predicted performance of devices such as ITER and a Component Test Facility.

In the following subsections, we will describe the status and upgrade plans for several critical subsystems which provide support for the whole research program or are linked to specific research topics within that program. A timeline for the implementation of several important upgrades to the facility is provided in Fig. 7.2.

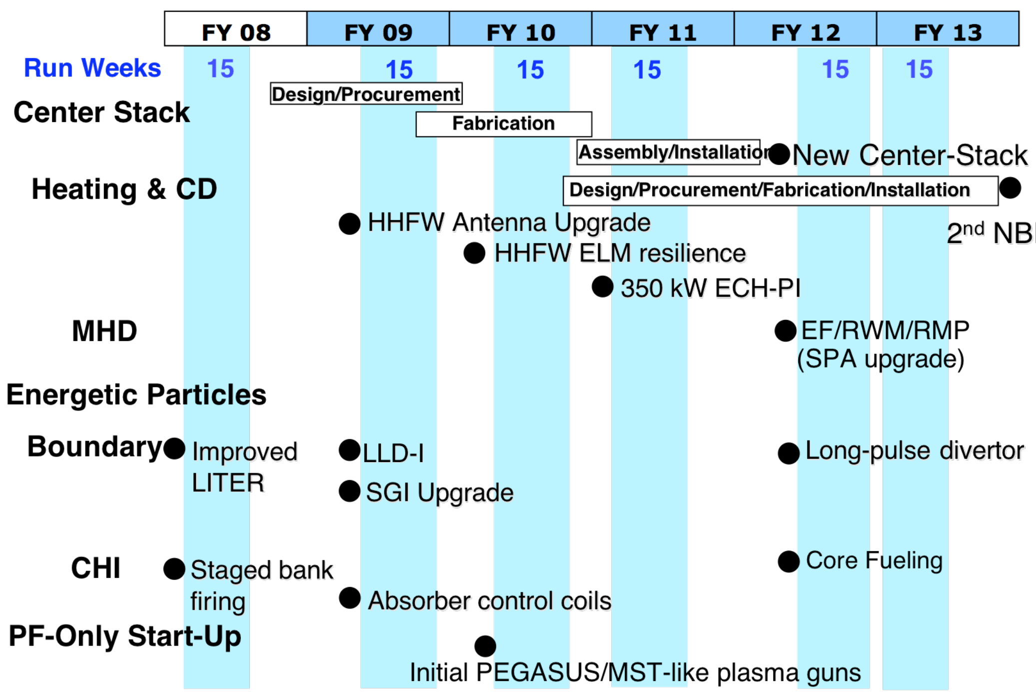


Fig. 7.2 Facility 5 Year Base Upgrade Plan. The shaded box indicates a range of implementation period determined by programmatic priorities.

7.2.1 NSTX Facility Subsystems

Magnet Systems: ***Status*** - The NSTX magnet systems are composed of the coils contained in the center-stack assembly, namely the inner TF, ohmic heating (OH) solenoid, top and bottom PF1A coils and the bottom PF1B coil, and those outside, namely the outer TF bundles, and top and bottom outer PF coils 2, 3, 4, & 5 as shown in Fig. 7.1. [7.1.3, 7.2.2] The inner coils are encased in, and thermally insulated from, an inconel vacuum casing on which are mounted the graphite tiles which form the inner plasma-facing surface. The entire center-stack assembly is fully demountable by removing the TF, OH, and PF1A connections. The demountable design provides opportunities for maintenance, repair, and upgrade as shown in Fig. 7.3. The magnets are powered by phase-controlled rectifier supplies; a motor generator provides the peak feed power (27 kA, 6 kV, 5 s) needed for operation. The TF magnet is operated routinely up to 0.55 T (nominal design limit 0.6 T) at the nominal vessel center R = 0.86 m.

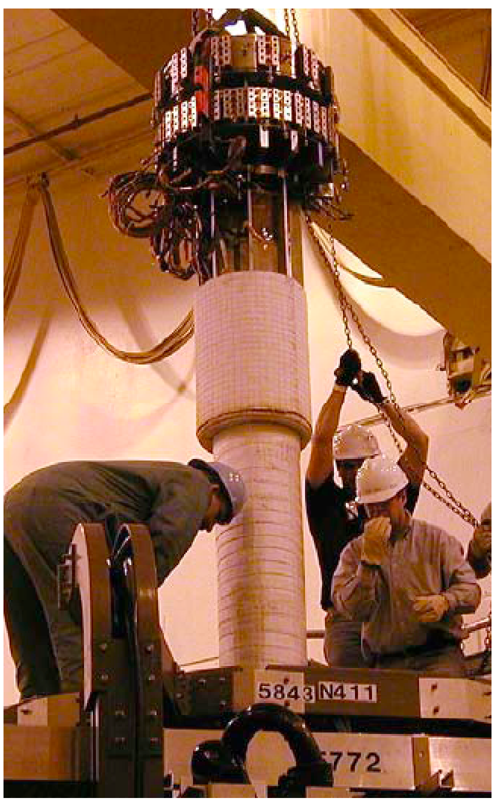


Fig. 7.3 Center-stack being installed

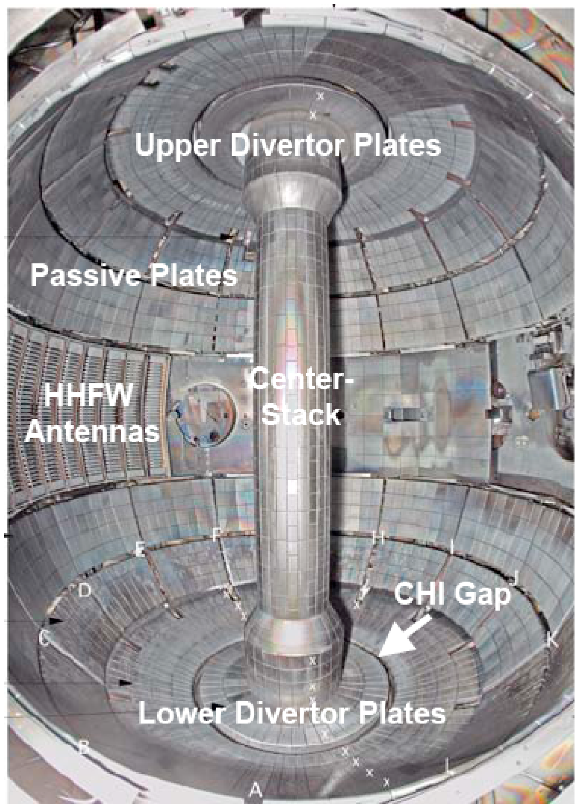


Fig. 7.4 Vacuum vessel internal components

Plans - A new center stack with 1 T, 5 s capability is planned. The new center stack is expected to include top and bottom PF1A and 1B coils for improved divertor and plasma shaping control. The power supply for the TF magnet and support structures will be upgraded for the 1T operation (see Sec. 7.2.7 for the new center stack details).

Vacuum Systems – *Status* -The NSTX vacuum vessel is roughly a spherical shape, 3 m tall and 3.5 m in diameter, with volume of about 30 m^3. A view of vacuum vessel interior is shown in Fig. 7.4. The plasma facing surfaces are covered by graphite tiles. The volume available to the plasma is ~15 m^3. The vacuum chamber provides very good plasma access with 12 major ports at the mid-plane between 12 outer TF coil sets. There are two NBI compatible rectangular ports 0.84 m (height) x 0.74 m (width); one is currently used for the existing NBI. The other, which is presently occupied by diagnostics, will be used for the proposed second NBI. Three port sections are occupied by the 12 element HHFW antenna systems and one port is dedicated for the vacuum pump duct which presently has two 1600 l/s turbo-molecular pumps. The remaining six sections each with a 0.60 m diameter circular port are mainly utilized for diagnostics. In addition, there are eleven medium size vertical-viewing ports and six smaller ports (horizontal-viewing toward the divertor regions) located on each of the upper and lower dome sections. There are gas feeds installed at three outboard mid-plane locations, one in-board at the mid-plane, one in-board at the "shoulder" where the center-stack case expands to accommodate the PF1A coils, and one in the CHI injector chamber.

Plans - The installation of the liquid lithium divertor in FY 2009 (see Sec. 7.2.5) will involve modifications of some plasma facing components on the lower divertor and their supports. A relocation and rearrangement of some diagnostics will accompany the planned installation of the second neutral beam injector system in FY 2013. Apart from these, no significant changes are planned to the main vacuum vessel.

Plasma Control System: Plasma control is an important aspect of achieving the improved performance associated with advanced plasma regimes. In particular, performance optimization involves approaching as near as possible to operating boundaries while avoiding deleterious plasma instabilities that can degrade or even terminate the plasma discharge. The approach taken to achieve the goals of plasma control is to use high speed computing combined with state of the art digital data acquisition and transmission to calculate accurate representations of the plasma state and to use this information in feedback loops to control plasma parameters

Status - The NSTX plasma control system (PCS) has been entirely digital from the outset. [7.2.4] It uses a dedicated fiber-optic communication network to bring real-time digitized data from diagnostics and other sensors to a central computer which analyzes important characteristics of the plasma equilibrium, compares them to desired values and generates feedback commands to the power supplies for the coils and other plasma actuators, such as the heating power and the gas fueling. The NSTX PCS was developed in collaboration with General Atomics, and uses the real-time equilibrium reconstruction code rtEFIT to control the plasma boundary. In particular, the PCS uses the real-time implementation of the EFIT (rtEFIT) solution to the Grad-Shafranov equation to control the position (radial and axial) and the cross-sectional shape of the axisymmetric MHD equilibrium. [7.2.5] The plasma control system has been continuously expanded and improved during the lifetime of NSTX. To facilitate creation of these highly shaped plasmas, two poloidal field coils, the PF1A upper and lower coils, were replaced by shorter bundles before the 2005 run. The rtEFIT/isoflux control capability of the NSTX plasma control system was extended to strongly shaped plasmas by the inclusion of additional magnetic measurements in the upper and lower divertor regions. These modifications led to the simultaneous achievement of a world record plasma elongation $\kappa \sim 3$ at very high triangularity. The use of rtEFIT/isoflux control has led to excellent reproducibility in the strongly shaped regime and is now the predominant control scheme on NSTX.

In addition to axisymmetric shape control, NSTX has been developing non-axisymmetric control for error field correction and resistive wall mode (RWM) feedback. This involves the control of the six EF/RWM coils powered by 3 independent high speed power supplies. Data is taken using an array of large area magnetic sensors measuring both the radial and vertical magnetic field at 12 toroidal and 2 poloidal locations for a total of 48 measurements. These measurements have been used to determine the optimal error field correction as well as for direct RWM feedback. Use of these control techniques has led to a significant increase in the achievable pulse length.

In 2007 the NSTX plasma control system (PCS) was upgraded and now uses a high speed server with 8 2.2GHz processors using a version of the Linux operating system that has been modified to support the needs of real-time operation. Data is acquired over a modular high-speed (1Gb/s) data network based on the Front Panel Data Port (FPDP) standard. The PCS currently acquires 432 channels of data at a 5kHz rate. The data acquired includes coil currents, magnetic field measurements, gas valve status and plenum pressures, RF loading, and neutral beam power. The PCS currently controls all the NSTX magnetic field supplies and the gas valves used for fuelling. For 2008 the hardware capability for controlling the neutral beam source timing and measuring the neutral beam power has been added. Initially the control will mimic the preprogrammed control that is currently in use. This new capability will be demonstrated at the end of the 2008 run.

Plans - The NSTX PCS is a flexible system which will be upgrades and improved with the availability of new plasma real time diagnostic information to control the actuators (such as NBI, HHFW, fueling. feedback coils). In 2009 the capability to control the neutral beam power will be used in conjunction with rtEFIT to control the plasma pressure. rtEFIT will be used to predict the plasma pressure using magnetics measurements. An algorithm will be developed and employed that modulates the neutral beam source power to maintain a specified time history for the plasma β. This capability should allow the pressure to be held closer to the operational limits without going over them. In addition, the existing capability to measure the RF loading will be used to see if feedback on the loading can help maintain RF coupling during H-mode transitions. Experiments will be performed to see if signals suitable for real time control of the plasma position can be developed based on the loading measurements. This capability could help the RF systems to avoid trips caused by sudden changes in loading observed during H-mode transitions. In 2010, the capability to measure the real-time rotation will be added. Development of a real-time signal that is a reasonable representation of the existing charge exchange recombination spectroscopy (CHERS) rotation measurement will feed into the capability to control plasma rotation profiles in the 2011 run. Significant diagnostic development is required in order to make the CHERS measurement compatible with real-time. In 2011 the newly added capability to measure rotation in real-time will be combined with the existing capability to control the neutral beam power and the ability to apply n=3 non-axisymmetric fields to control the rotation profiles. Algorithms will be developed that will generate control commands for the power supplies that control the n=3 fields such that the combination of these fields and the torque from the neutral beams maintains the rotation at a specified location in the plasma at a specified value throughout the discharge. This capability could add significantly to the ability to do experiments that investigate the correlation between the plasma rotation speed and both plasma stability and plasma confinement. Starting at the end of the 2010 run, hardware modifications to the PCS will be made to support the implementation of real-time measurements of the motional-Stark effect polarimetry measurement of the magnetic field pitch angle. This capability would be available by the end of the run in 2011. This will support implementation of real-time equilibrium reconstructions with q-profiles constrained by internal measurements for the 2012 run. This capability would be directly support control of the current profile or with neutral beam current drive in future run years.

7.2.2 Heating and Current Drive Systems

The two main heating and current drive systems on NSTX are Neutral Beam Injection (NBI) and High-Harmonic Fast-Wave (HHFW) RF power. A heating and current system based on the electron Bernstein wave (EBW) is also being considered.

Neutral Beam Injection System: *Status -* The NSTX NBI system utilizes parts of the unsurpassed TFTR system. [7.2.5] The NSTX NBI system has delivered up to 7 MW at an acceleration voltage up to 100 kV. The beam system is the workhorse of the NSTX high-β plasma research program, providing not only reliable plasma heating but also other important functions: imparting toroidal momentum to induce toroidal plasma rotation of up to ~ 300 km/s; driving plasma current (in addition to the bootstrap current) of up to ~ 200 kA; providing an energetic ion population for simulating alpha-particles (Ch. 4); and enabling plasma profile diagnostics such as the charge-exchange recombination spectroscopy (CHERS) and motional Start-effect (MSE) systems (Sec. 7.3.)

Plans - With the increased toroidal field and current available with the proposed center stack upgrade, a second NBI system injecting more tangentially is planned to provide higher heating and current drive power as well as an improved NBI current profile control capability. Section 7.2.8 will present more detail on the second NBI upgrade.

High Harmonic Fast Wave System:

***Status* –** The NSTX HHFW system heats electrons and drives plasma currents for non-inductive current ramp-up and sustainment. It has also been used for pre-ionization during experiments to start the plasma using only the outer PF coils. A twelve-element-antenna system, shown in Fig. 7.5(a), is driven by six power amplifiers operating at 30 MHz with delivered power of up to 6 MW. [7.2.6] The antenna array utilizes an RF matching network developed by ORNL to vary the phase of the RF current in the six pairs of current straps in real-time to control the parallel wavenumber $k_{||}$ of the plasma waves for driving current and optimizing the wave absorption as the discharge evolves. This capability is required to track the plasma evolution from a few 100eV in OH plasmas at the start-up to a few keV in the heated phase. The HHFW has heated the core electrons from the few hundred eV typical in the ohmic phase to typically 2 – 3 keV and as high as 5 keV. The HHFW heating efficiency has been found to have a strong positive dependence on the toroidal magnetic field and an inverse dependence on the edge plasma density. Lithium coating of the plasma facing components, which decreases the edge density and recycling, has been shown to improve the HHFW heating performance. Experiments to study HHFW current drive have shown a significant change in the loop voltage as the direction of the driven current was varied, consistent with the theoretical expectations. However, the experiments have also suggested that parasitic absorption of the HHFW by thermal and fast ions may limit the current drive efficiency of HHFW. This physics is now being investigated using the MSE diagnostic to measure the effects on the local current density.

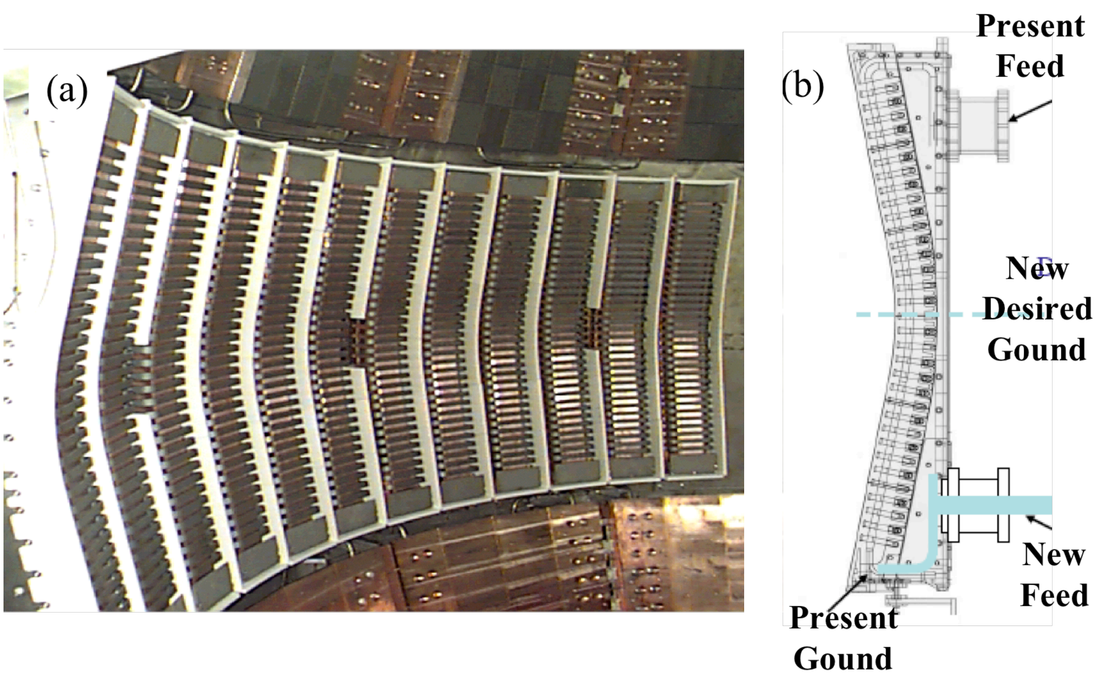


Fig. 7.5 HHFW Antenna Upgrade. (a) HHFW Antenna View (b) Antenna upgrade schematic

Plans - The HHFW power coupled to H-mode plasmas needs to be improved for planned experiments. The observed break-down voltage across the insulators in the RF feed-throughs is considerably lower in NSTX than TFTR, probably caused by increased cross-field particle diffusion into the feed-through assembly as a result of the lower toroidal field of NSTX. A solution is to upgrade the antenna to a symmetric-feed configuration which will increase the coupled power level by a factor of two for a given feed-through voltage. A schematic view of the modification which will be implemented in FY 2009, is shown in Fig. 7.5(b). This HHFW antenna upgrade should double its power handling capability and optimize the antenna radiation pattern by locating the virtual antenna ground to the antenna mid-plane. In 2010, to reduce the antenna impedance changes during the H-mode ELM activity, an HHFW ELM resilience capability will be implemented for H-mode plasmas in collaboration with ORNL.

Electron Bernstein Wave System: ***Status -*** Electron Bernstein Waves (EBW) are a promising tool to drive localized off-axis current needed for sustained operation in advanced ST modes. The current is driven via the Ohkawa effect, in which a part of the electron thermal distribution which resonates with the wave is accelerated from co-passing to trapped orbits, thereby suppressing their counter-directed parallel electron current. Modeling predicts high current drive efficiency for this scheme, two to three times greater than conventional Fisch-Boozer electron-cyclotron current drive. However, the EBW current-drive scheme depends on converting externally launched electromagnetic waves into the EBW which can only propagate inside the plasma. Measurements in NSTX of the inverse process, that is mode conversion of thermally excited EBW inside the plasma to electromagnetic emission from the plasma show the possibility of achieving adequate coupling efficiencies for the viability of this current drive scheme. With lithium evaporation onto the NSTX divertor, good coupling to H-mode plasmas has been recently demonstrated. However, the physics of the EBW coupling is still not yet fully understood so research will continue in this area.

Plans - In view of recent encouraging results, research on the fundamental physics of EBW coupling and absorption will continue on NSTX. A medium power ECH/EBW system (350 – 700 kW at 28 GHz) is being considered for NSTX. [7.2.7] This system, which is being developed as a collaboration between PPPL and ORNL, would become available in FY 2011. It could also benefit plasma start-up research on NSTX by providing conventional ECH heating for the low-density plasmas created by CHI or the outer-PF coil startup scheme, which will be discussed in Sec. 7.2.6. The ECH heating would not only reduce the plasma resistivity but also could heat the plasma to higher electron temperature ~ 100 eV where HHFW heating becomes effective. An active collaboration on EBW physics will be pursued with the MAST experiment at the UKAEA Culham Laboratory where a medium EBW power system will become available in the near term.

7.2.3 Macro-stability Tools

The possibility of achieving high plasma pressure at modest magnetic field, *i.e.* high plasma β, is a primary motivation for investigating the ST concept. NSTX is exploring two main avenues to maximizing the sustainable β: optimization of the axisymmetric equilibrium configuration and suppression or control of the growth of non-axisymmetric pressure-driven instabilities.

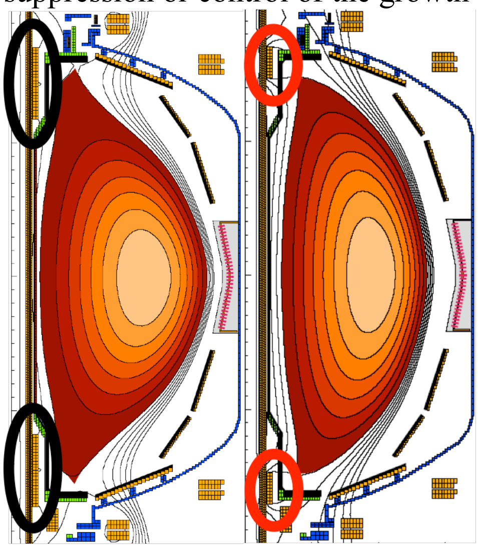


Fig. 7.6Cross-section of NSTX showing the PF1A coils (circled) and typical plasma configurations available in 2004 (left) and 2005 (right) and the Error Field and Resistive Wall Mode (EF/RWM) coils.

In experiments conducted on NSTX in 2004 with the original PF coils, encouraging results, in terms of higher achievable plasma current and thus lower normalized plasma beta $\beta_N = \beta_T a B_T/I_p$, were obtained with high elongation plasmas. However, MHD stability calculations suggested that further improvements in the stable β_N could be made if the plasma cross-section conformed more closely to the passively stabilizing conducting plates on the large major radius side. Because the NSTX center-stack is removable, ready access is available to any part of the device. This enabled the installation of a new set of upper and lower inner PF1A coils in early 2005 to achieve high-elongation, high-triangularity. In recent experiments, the new shaping coils produced plasmas with high elongation, κ = 3.0 transiently and 2.7 sustained, and triangularity, δ ≈ 0.8. Examples of the plasma shapes obtained with the old and new coil sets are shown in Fig. 7.6.

As the plasma β increases above the level at which a surrounding conducting wall is necessary for MHD stability to pressure-driven modes, the so-called "no-wall β-limit", it is important to have methods to control mode growth on the timescale for resistive decay of currents induced in the wall. Two approaches are being tested on NSTX: maintaining sufficient toroidal rotation and applying non-axisymmetric magnetic field perturbations to sustain plasma rotation and counteract mode growth.

A nearby conducting wall can stabilize the ideal pressure-driven MHD kink/ballooning mode, provided that sufficient plasma rotation is maintained. However, below a critical rotation rate, the resistive wall mode (RWM), a kink mode modified by the presence of a conducting wall, can become unstable and grow, leading to rapid rotation damping and, ultimately, β collapse on the timescale of the decay of wall eddy currents. Such unstable modes appear to grow preferentially from perturbations caused by the intrinsic error fields in the plasma region which act as the "seeds". In 2002, the maximum normalized plasma beta β_N was improved significantly by reducing the intrinsic error fields by realigning the PF5 coils. With sufficient plasma rotation maintained by the momentum imparted by the NBI, the β limit of NSTX plasmas has significantly exceeded the no-wall limit for many resistive wall times, demonstrating the effectiveness of wall stabilization. The second approach of applying non-axisymmetric magnetic field perturbations is discussed in the following section.

Error Field/Resistive Wall Mode Coils – ***Status -*** To explore plasma operation with beta above the no-wall limit and approaching the limit for stabilization by an ideal, perfectly conducting wall, the "ideal-wall limit", three pairs of non-axisymmetric coils have been installed on NSTX to produce controllable radial magnetic field perturbations as shown in Fig. 7.7. [7.2.8] The individual coils are nearly rectangular "picture frames" of two turns each, centered on the mid-plane and mounted just outside the vacuum vessel wall. The three pairs of diametrically opposite coils are powered by a three-channel Switching Power Amplifier (SPA) with the capability for driving multi-kiloamp currents at frequencies up to 1 kHz. The individual coil polarities can be changed at a patch panel to provide controllable perturbations with dominantly odd (n = 1, 3) or even (n = 2) toroidal harmonics. The complete system was commissioned at the start of the 2005

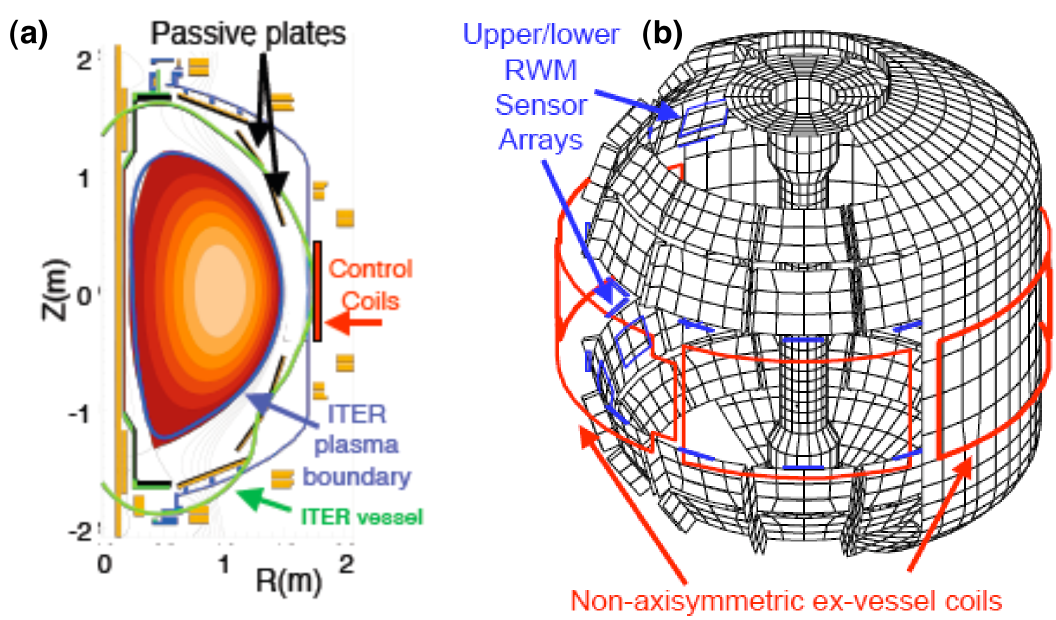


Fig. 7.7 EF/RWM/RMP Coil Schematics (a) Cross-sectional view. (b) 3-D view

experimental run. Experience with the NSTX EF/RWM coil system will contribute to the system proposed for ITER, since the NSTX coil locations are similar to those proposed for ITER and, in ITER, the tritium blanket modules on the outboard side will slow down the instability growth similarly to the NSTX passive plates as illustrated in Fig. 7.7 (a). NSTX EF/RWM research benefits from a comprehensive array of magnetic sensors, the data from which can be processed in real time to generate time-varying corrections of both intrinsic and plasma-generated field perturbations as shown in Fig. 7.7 (b). [7.2.9] Use of the EF/RWM coils has enabled sustained high beta operation in NSTX, even at low plasma rotation rates relevant to those expected in ITER and ST-Demo where the momentum input from NBI heating will be much lower. More recently the feedback coil system was used to test ELM stabilization and controllable destabilization using the Resonant Magnetic Perturbation (RMP) concept in support of the ITER design activity.

Plans – The present SPA provides three independent controllable currents to the EF/RWM coil set. These coils are generally configured through hardwired connections at a patch panel to provide either the odd (n = 1, 3) or even (n = 2) perturbations, although other “exotic” configurations which include higher field harmonics are possible, such as simulating the “missing” control coil of one set of such coils proposed for ITER. However, maximum flexibility would be achieved by installing an additional 3-channel SPA so that each coil would be separately powered, allowing simultaneous control of the full range of field harmonics. An assessment is being made to determine the need for this upgrade. Design studies are also being performed to assess the desirability of installing a second set of non-axisymmetric control coils inside the vacuum vessel and located symmetrically above and below the midplane. These coils would provide faster reaction to the evolution of MHD perturbations and better capability to counteract mode growth due to the very high pitch of the field lines at the outboard side of high-β ST plasmas. Depending on the results of these assessments and design studies which are based on both theory and the results of ongoing experiments, one or both of these upgrades would be undertaken in FY 2011 to be available in FY 2012.

7.2.4 Boundary Physics Facility Tools

The achievement of good vacuum and surface conditions on the plasma-facing components (PFCs) in NSTX has been crucial to the progress in its plasma performance. NSTX employs state-of-the-art wall conditioning capabilities, including 350°C bake out of PFCs, periodic boronization, and between-shots helium glow discharge cleaning (HeGDC), to control the impurity influxes and the hydrogenic loading of the PFCs. This combination of methods both suppresses oxygen impurities and reduces the ratio of the hydrogen to deuterium concentrations to below 0.05 in plasmas with deuterium fueling.

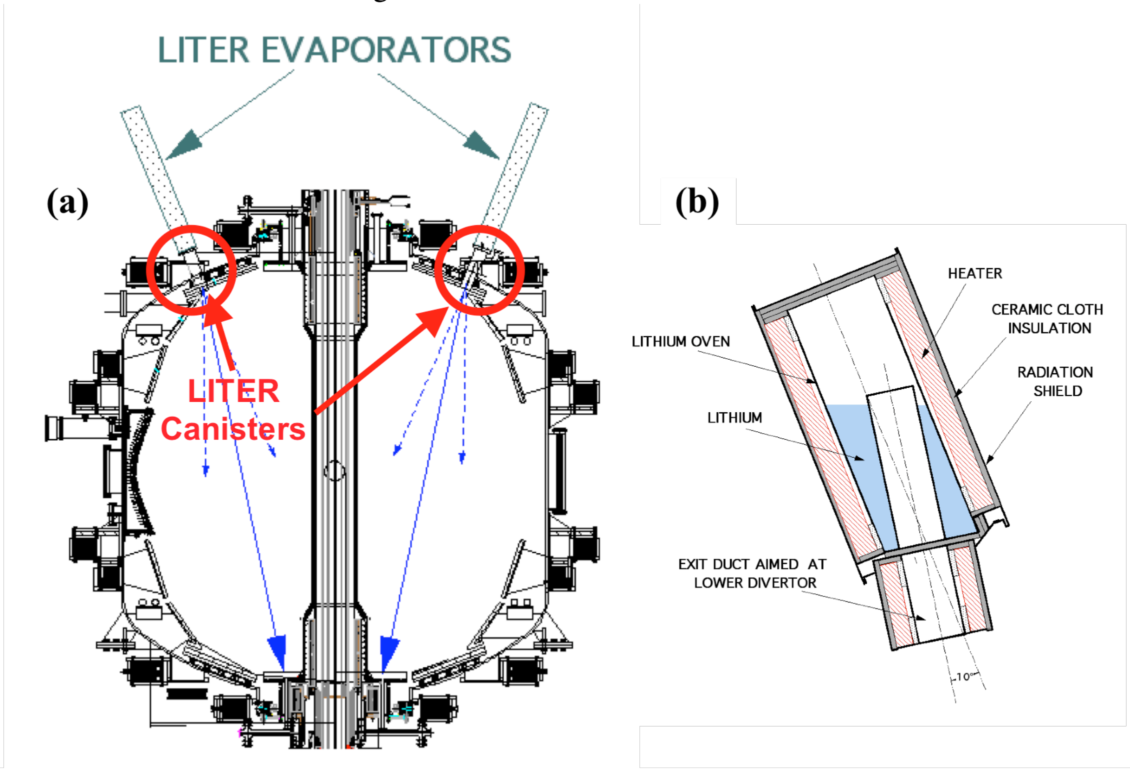


Fig. 7.8 NSTX "LITER" lithium evaporators (a) positioning, (b) canister schematic

Lithium Evaporator for Coating Divertor Surfaces: ***Status*** - A lithium evaporator, dubbed "LITER", had been developed, installed on NSTX since 2006 and used in a variety of experiments which had shown several benefits for plasma operation. [7.2.10] In 2006, when a total of 10 g of lithium was evaporated during the run period, some promising indications of reduced recycling and impurity levels were measured. For the FY 2007 experiments, the LITER was upgraded by installing more robust radiant heaters, increasing the capacity of the lithium reservoir, enlarging the diameter of the exit duct and angling it to direct the lithium vapor stream towards the lower divertor plate. With these changes, the maximum evaporation rate was increased by a factor of 3 to about 60 mg/min and the peak local deposition rate of lithium on the lower divertor was increased by a factor 3. A total of over 90 g of lithium was evaporated in the entire campaign with LITER in FY 2007 producing clear benefits in H-mode plasmas, including improved confinement, particularly of the electrons, and ELM-free periods. For FY 2008 experiments, a second LITER was installed at another upper port, separated 150° toroidally from the first, to complete coverage of the lower divertor target. The location of the LITER evaporators in 2008 and a schematic cross-section showing features of the evaporators are shown in Fig. 7.8. Shutters were installed in front of both LITERS to prevent lithium entering the plasma region both during the plasma pulse (to prevent lithium deposition on diagnostic windows while their protective shutters were open) and during any subsequent period of helium glow-discharge cleaning (to prevent the entrapment of helium in the deposited lithium layer). The LITER system can be refilled with about 100 g of lithium in about a half day. The effects of the lithium coating on the plasmas will be described in the section on Research Accomplishments to follow. The dual LITER system has contributed strongly to the NSTX research program in 2008 in numerous ways, including improving the plasma confinement, stabilizing ELMs, improving the HHFW heating efficiency, particularly in deuterium plasmas, and improving the EBW coupling in H-mode plasmas.

Plans - The LITERs now provide a mature, reliable lithium delivery system. In 2009, after the installation of the liquid lithium divertor target (described below), the LITERs will used as its primary lithium delivery method, although they can also continue be used in their present mode of operation.

Liquid Lithium Divertor Target:

Status - Following the success of the lithium evaporator development and experiments, the NSTX team is focusing its effort on a liquid lithium divertor (LLD) target. [7.2.11] This initiative is a collaboration between PPPL and a team at Sandia National Laboratory (SNL). As shown in Fig. 7.9, the LLD will be located at the bottom of the vessel in the outer divertor strike point region. It will consist of a set of temperature-controlled, molybdenum-coated stainless-steel plates forming an almost continuous conical annular ring in the lower outboard divertor. When coated with lithium and heated above its melting point the plates will provide about 7000 cm^2 of active pumping surface area in contact with the outboard scrape-off layer of the plasma. The dual LITER system presently installed on NSTX will be used to deposit and resupply the layer of lithium onto the heated LLD surface. The liquid lithium limiter experiment on CDX-U was quite encouraging in terms of drastically reducing the oxygen level and the particle recycling while handling high heat flux, but this experience was limited to ohmically heated L-mode plasmas. The NSTX system will be unique in the world to test the effectiveness of liquid lithium as the plasma contact surface for H-mode plasmas with significant auxiliary heating power. A liquid lithium divertor target has the potential to be a much more attractive option than a cryo-pump system because it can manage the intense heat flux as well as controlling particle recycling in the steady-state conditions expected in future devices such as CTF and Demo.

Plans – The design of the LLD system is complete and the components are currently being procured and fabricated. Ongoing R&D activities at PPPL and SNL are assessing and developing coating techniques. The LLD system is scheduled to be installed on NSTX in the fall of 2008 and will be available for experiment in FY 2009. Based on results of continuing laboratory tests and the initial LLD results, the LLD will be improved in FY 2010 (LLD-II) to increase its pumping and power handling capability sufficient to meet the demands of high-performance discharges extended to the limits of the NSTX magnets. One possibility being explored to increase the capacity is a mesh-based system, which can retain a larger volume of liquid lithium for long-pulse operation. A prototype mesh system is being developed and tested in collaboration with SNL. Another possibility for LLD-II is a second plate system located on the inboard side of the lower divertor to augment the pumping capability for high-triangularity discharges. An assessment will be made of the LLD performance after the FY 2009 run to decide on the long-pulse high-power divertor design including a possible transition to a cryo-based system in FY 2011.

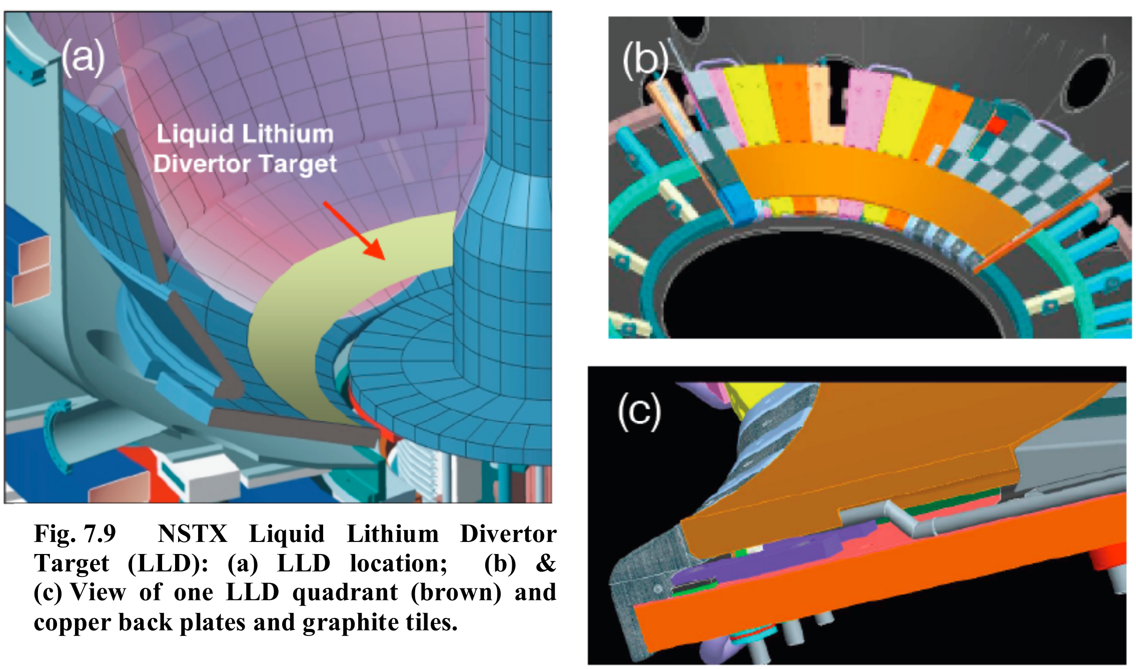


Fig. 7.9 NSTX Liquid Lithium Divertor Target (LLD): (a) LLD location; (b) & (c) View of one LLD quadrant (brown) and copper back plates and graphite tiles.

Fueling Tools: ***Status -*** The capabilities for gas fueling the NSTX plasma have been steadily enhanced with gas injectors in several locations, including the outer mid-plane, inboard mid-plane, "upper shoulder" and divertor regions. [7.2.12] Experiments have shown that puffing deuterium into the private flux region or close to the outer strike point of the lower divertor can produce partial detachment of the outer divertor and a substantial reduction of the heat flux to the divertor surface without degrading the H-mode energy confinement. A supersonic gas injector (SGI) has been installed which introduces deuterium gas at a velocity up to 2.4 km/s into the plasma near the mid-plane on the outside. [7.2.13] High-speed camera images have confirmed that a collimated stream of deuterium enters the plasma from the nozzle. The fueling efficiency of the SGI is a factor 2 – 5 higher than conventional deuterium gas injectors and the H-mode threshold power is lower with SGI fueling.

Plans - The NSTX plan for core fueling includes both a frozen deuterium pellet injector and a compact toroid (CT) injector. [7.2.14] Interestingly, high beta plasmas on NSTX present a minimum-B target for outside-launch pellet injection, with the gradient of the total magnetic field pointing radially outwards a few centimeters inside the plasma edge, thereby presenting the possibility of inward transport of the pellet fuel even with the technically simpler outside launch. The NSTX field structure is also well suited to test controlled fuel deposition by the CT injector. A CT injector developed for a tested on the Tokamak de Varennes in the 1990s is available to NSTX although some refurbishment would be necessary and, if initial indications are favorable, an upgrade to the capacitor banks and their associated power supplies would be needed to provide the capability for multiple CT injections during a single NSTX discharge. The ITPA steady-state team has expressed interest in an investigation in NSTX of CT injection, which has reactor-relevant potential for particle fueling and radial deposition control.

7.2.5 Start-up and Ramp-up

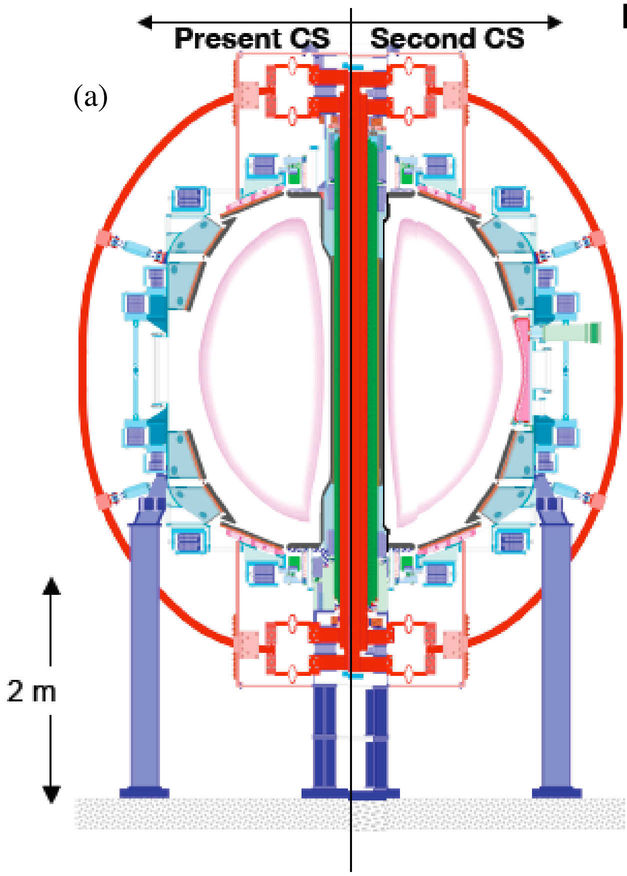
In order to develop attractive ST power plants, a practical technique must be developed to start up the toroidal plasma current needed for confinement without reliance on induction from a central solenoid. Designs for ST reactors such as CTF and ARIES-ST assume no ohmic solenoid and even advanced tokamak reactors such as ARIES-AT also dispense with an ohmic solenoid. Solenoid-free start-up is therefore an important component of NSTX research. The main approach investigated to date in NSTX is coaxial helicity injection (CHI). This technique has been investigated previously in smaller devices including HIT/HIT-II (U. Washington). NSTX has also investigated techniques to start the plasma using only the outer PF coils; initial experiments have shown promise.

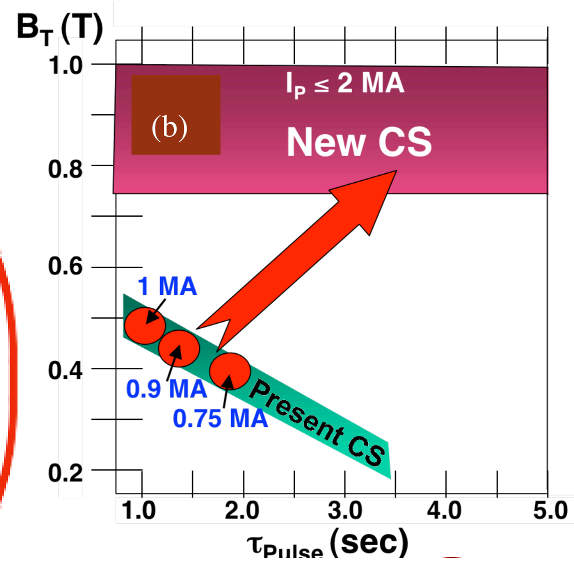
Coaxial Helicity Injection: ***Status*** **–** Coaxial Helicity Injection (CHI) involves generating a poloidal current between electrodes on the inner and outer parts of the vacuum vessel which are electrically isolated. [7.2.15] In the early NSTX experiments which used a rectifier supply to produce the discharge, CHI generated up to 400 kA of toroidal current and a current multiplication factor (the ratio of the toroidal plasma current to the injected current) up to 14 was obtained. However, it was difficult to assess how much of the toroidal current was actually flowing on closed toroidal flux surfaces. Recently on HIT-II, a new technique called a “transient” CHI was developed which produced substantial toroidal current flowing on closed flux surfaces after the injector current had been terminated. [7.2.16] By utilizing relatively short injection pulses, it was found to be possible to “detach” the plasma current ring from the injector to create flux closure. By initiating the plasma with the transient CHI technique, the performance of the plasma in HIT-II during a subsequent inductive phase was significantly improved, reducing the inductive volt-seconds needed and producing significantly higher final plasma current. On NSTX, a capacitor bank was commissioned in 2004 to test this method. Initial experimental results were encouraging, producing about 150 kA of toroidal plasma current with only a few kA of injected current, yielding current multiplication factors of as high as 40. For the 2005 experiments, direct gas-feed into the injector region was implemented to allow breakdown with reduced gas input to the main chamber. In 2008, staged firing of the CHI capacitor was implemented to provide control of the CHI current during the critical expansion of the discharge into the main plasma region. A successful transition from CHI initiation to inductive ramp-up of the plasma current has now been demonstrated.

Plans **–** The coupling of a discharge initiated by CHI to induction will be further optimized to minimize the requirements for inductive flux to reach the full operating current. Such a demonstration is needed to validate the designs for the ST-CTF and ST-Demo where only limited inductive flux can be provided.

PF-Only Start-Up: ***Status -*** In parallel to the research on CHI, the NSTX team is pursuing another approach utilizing the outer PF coils. There are several variants of the outer-PF-only plasma start-up approach. On the JT-60U and TST-2 devices, significant plasma currents of up to 150 kA were generated by the simple vertical field swing in the presence of strong ECH heating. In addition to the TST-2/JT-60U technique, a merging plasma ring approach developed on TS3/4 has been successfully demonstrated on the MAST device recently. A poloidal flux / field-null optimized scenario is also being developed for STs. On NSTX, a scheme for outer-PF-only plasma start-up was tested in 2004. Using HHFW as a pre-ionization source, a successful start-up was demonstrated producing ~20 kA of toroidal current, but only when a very good field-null condition was satisfied. A high power ECH preionization and pre-heating source could relax the field-null requirements, as was demonstrated in TST-2 and JT-60U.

Plans - A medium power ECH system with power level of 350 -700 kW is planned in collaboration with ORNL which should improve the initial breakdown condition. Other alternatives, including a plasma gun and CT injection, are being considered to provide a suitable target plasma for the start-up. The goal is similar to that for CHI start-up, that is to reach full plasma current with minimum use of induction.





Parameters	Units	Present CS	2nd CS
Minimum A		1.28	1.5
R_0	cm	85	91
A	cm	67	61
R_0-a	cm	18.5	30.5
I-TF	kA	71	119
BT_0	T	0.55	1
τ-TF-flat top	sec	0.8	5.6
OH-flux	V-S	0.75	2.59
Ip	MA	1	2
Flat-top VS	VS	0.29	1.59

Fig. 7.10(a) The cross section of the present and second center-stack.
(b) Accessible B, I_p, and τ-pulse space for the present and second center-stack.
Table 7.2 The base parameters of the present and second center-stack

7.2.6 New Center-Stack Upgrade

Status - This upgrade involves a replacement of the slender central column of NSTX, which holds some of the magnet conductors, with a slightly wider column, capable of 2× higher magnetic fields for 5× longer pulses. This would raise the magnetic fields in NSTX half-way to those of the next-step devices and expand the accessible magnetic field range by a factor of 3.

The new center-stack is a logical step for the NSTX facility which was designed originally to be able to accommodate a replacement. The center-stack can be removed relatively easily by removing the 72 flexible joints (36 top and 36 bottom) of the TF coil as shown in Fig. 7.10 (a).

The removable/replaceable center-stack design facilitates the maintenance and repair of the center-stack which is the most stressed and complex component on NSTX. During the eight years of operation to date, the center-stack has been removed three times for maintenance, repair and improvements. The new center-stack will have the same vacuum sealing surfaces to the flexible elements of the vacuum vessel boundary so that it can be installed during a normal annual outage period (typically a period of few months.) A comparison of the present and the second center-stack cross sections and accessible parameter space are shown in Fig. 7.10 (a) and (b). Some of the basic parameter is listed in Table 7.2. The two designs are similar except that the new center-stack is about 12 cm larger in major radius as shown in the Table 7.2. In fact, it would be possible to use the two assemblies interchangeably if it made good programmatic sense. In addition to the TF and OH coils, the new center-stack will also include top and bottom PF 1A and 1B coils to retain divertor and plasma shaping flexibility. The present center-stack has symmetric top and bottom PF 1A coils but only a bottom PF 1B without a corresponding top PF 1B coil. Due to larger conductor sizes and larger radial build, the stresses in the upgraded TF and OH magnets are similar to those in the present center-stack. Similarly, the TF joint current density and the magnetic field at the joint are similar. So the overall design and the manufacturing processes for the TF, OH magnets and the TF joints should remain similar to those used for the present center-stack with some improvements. The overall forces (e.g., vertical lifting force at the TF joints and overturning force) will be larger due to the higher toroidal and poloidal fields so appropriate support structure enhancements will be implemented. The outer PF coils (PF 2, 3, 4, 5) as well as the outer TF coils are sufficient to support the higher field operation. The power system for the new center-stack will require doubling of the toroidal field coil current. This will be done by reconfiguring the available TFTR power supplies. The existing OH and PF power supplies are adequate for the projected plasma current of 2 MA, for a 5 s pulse. The new center-stack will cause relatively minor difficulties for most diagnostic systems although the laser path for the multi-point Thomson scattering system will need to be shifted outward by about 10 cm.

Plans - The design of the center-stack will be finalized in FY 2009 while starting the procurement of the long-lead items. In FY 2010 - 2011, component fabrication and installation will be completed to be ready to support the FY 2012 run.

7.2.7 Second Neutral Beam Injection System Upgrade

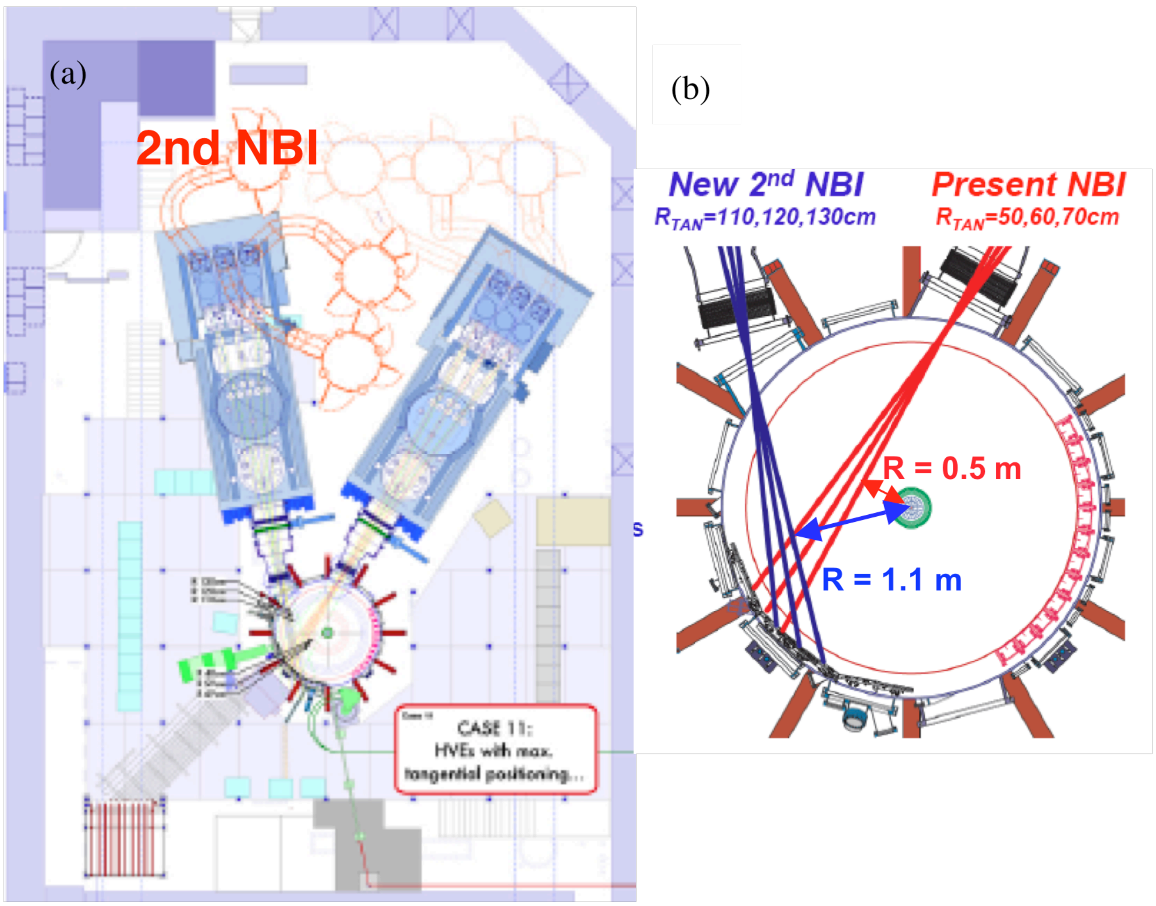


Fig. 7.11 Second NBI Upgrade (a) NBI systems in the NSTX Test Cell. (b) NBI beam trajectories showing more tangential injection for the second NBI (in blue).

Plans - A second TFTR Neutral Beam Injection system with three sources will be refurbished and moved to NSTX, thereby doubling its neutral beam power. Provision for a second NBI system was made in the original design of the device which includes a second specialized NBI port (at Bay K). Space is available in the NSTX test cell for the second beamline and its high-voltage power supplies. The second beamline is shown in the NSTX Test Cell in Fig. 7.11. The second NBI system will have more tangential aiming, with tangency radii R_{TAN} = 1.10, 1.20, 1.30 m, compared to the present NBI radii R_{TAN} = 0.50, 0.60, 0.70 m. The work to accomplish this upgrade should be similar to that for installation of the first NBI, although the second beam box from TFTR remains tritium contaminated and therefore requires decontamination process which may involve the replacement of some internal components. NSTX already uses ion sources from TFTR which have undergone a well-established decontamination process. There will be a relocation of the diagnostics and the pumping port.

The second NBI installation not expected until FY 2014 due to the resources required. It is planned to utilize a modest effort to perform some tests on the beam box to assess the decontamination work scope and the need for replacement of the internal components. It is also desirable to perform the engineering design up front to tighten the cost and schedule. If the resources were to become available, the second NBI could be accelerated by one year for the FY 2013 run in the incremental plan.

7.3 NSTX Diagnostic System Status and Plans

In addition to providing the capabilities to produce and control high plasmas, it is crucial to have appropriate plasma diagnostics in order to develop our understanding of the physical processes governing plasmas through the comparison of data with theory and modeling. NSTX has been continuously implementing modern plasma diagnostic systems in key research areas. In this section, we will describe briefly some of the recent diagnostic additions and planned upgrades. A complete list of major diagnostics currently installed and operating on NSTX is shown in Table 7.3. The implementation of diagnostics on NSTX has benefited greatly from collaborations with many national and international institutions: more than half of the diagnostics listed (highlighted in italicized red in Table 7.3) are either collaborator-led or have a strong collaboration component. In the Appendix, those collaborative NSTX diagnostics and upgrades plans are described as an important part of the collaboration activities and plans.

• Collaboration contributions

MHD/Magnetics/Reconstruction
- Magnetics for *equilibrium reconstruction*
- Diamagnetic flux measurement
- Halo current detectors
- High-n and high-frequency Mirnov arrays
- Locked-mode detectors
- RWM sensors (n = 1, 2, and 3)

Profile Diagnostics
- Multi-pulse Thomson scattering (30 ch, 60 Hz)
- T-CHERS: $T_i(R)$ and $V_f(r)$ (51 ch)
- P-CHERS: $V_q(r)$ (51 ch)
- *MSE-CIF (15 ch)*
- *FIReTIP interferometer (119mm, 6 ch)*
- Midplane tangential bolometer array (16 ch)

Turbulence/Modes Diagnostics
- *Tangential microwave high-k scattering*
- *Microwave reflectometers*
- *Ultra-soft x-ray arrays – tomography (4 arrays)*
- *Fast X-ray tangential camera (2ms)*

Energetic Particle Diagnostics
- Neutal particle analyzer (2D scanning)
- SSNPA
- Fast lost-ion probe (energy/pitch angle resolving)
- Neutron measurements
- *Fast Ion D_a profile measurement*

Edge Divertor Physics
- *Reciprocating Edge Probe*
- *Gas-puff Imaging (2ms)*
- *Fixed Langmuir probes (24)*
- Edge Rotation Diagnostics (T_i, V_f, V_{pol})
- *1-D CCD H_a cameras (divertor, midplane)*
- *2-D divertor fast visible camera*
- Divertor bolometer (12 ch)
- *IR cameras (30Hz) (3)*
- Tile temperature thermocouple array
- Dust detector
- *Scrape-off layer reflectometer*
- *Edge neutral pressure gauges*

Plasma Monitoring
- *Fast visible cameras*
- Visible bremsstrahlung radiometer
- Visible survey spectrometer
- UV survey spectrometer
- *VUV transmission grating spectrometer*
- *Visible filterscopes*
- *Wall coupon analysis*
- *X-ray crystal spectrometer (astrophysics)*

Table 7.3. Major Diagnostic Systems Installed on NSTX

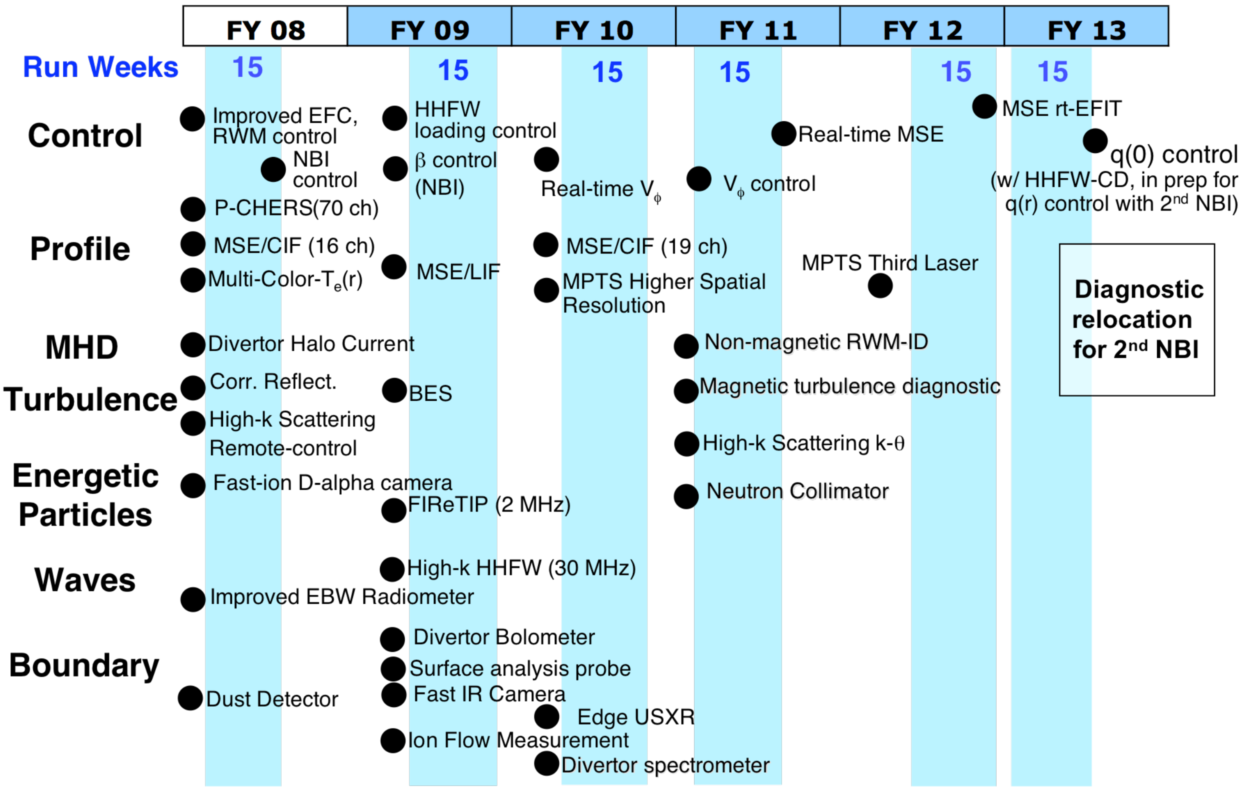


Fig. 7.11 NSTX 5 Year Diagnostic Upgrade Plan

Fig. 7.11 shows an overview of the diagnostic upgrade plan to support the NSTX Five Year Research Plan. The implementation of specific diagnostics depends on the available budget and programmatic priorities. This diagnostic upgrade plan is based on the projected available budget profile.

7.3.1 Profile Diagnostics:

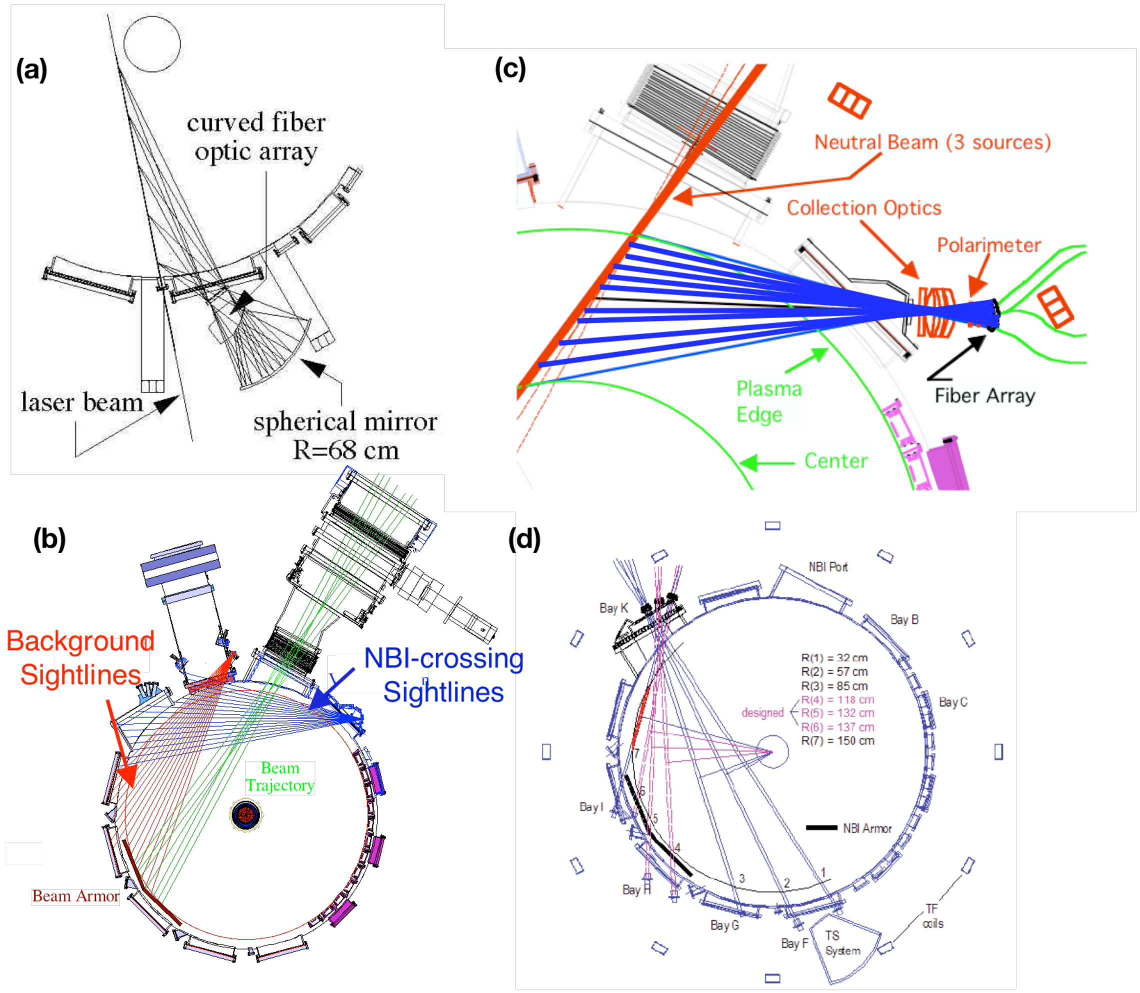


Fig. 7.12 Present NSTX profile diagnostics taking advantage of excellent tangential access. (a) 30 channel, 60 Hz Multi-Pulse Thomson Scattering System. (b) 51 Channel Toroidal CHERS system with both NBI and background views. (c) 16 channel Motional Stark Effect using Collisionally-Induced Florescence current profile diagnostic system. (d) 6 channel Far-infrared Tangential Interferometer.

For transport studies, NSTX has developed an excellent set of plasma profile diagnostics which measure profiles near the plasma mid-plane to facilitate rapid data analyses. Through a combination of state-of-the-art profile and turbulence diagnostics with advanced theory and modeling, NSTX is contributing to the fundamental understanding of plasma transport and stability which is needed to develop predictive capability for ITER and future STs.

Multi-pulse Thomson Scattering System: ***Status -*** The large vacuum window and light collection mirror allow the multi-pulse Thomson scattering (MPTS) system to measure plasma density and temperature with excellent accuracy at a rate of 60 profiles per second and at 30 spatial points to resolve the structure of the H-mode edge pedestal and internal transport barriers. [7.3.1, 2] A schematic of the MPTS system is shown in Fig. 7.12 (a). ***Plans -*** The MPTS system is designed to be upgraded to 45 spatial channels and 3 lasers for 90 Hz operation. The MPTS system will be upgraded as the resources become available and according to programmatic needs.

Toroidal Charge Exchange Recombination Spectroscopy (TCHERS): ***Status -*** This system provides detailed ion temperature, plasma toroidal rotation and carbon density profiles as a function of time at 51 spatial locations across the outer half of the plasma region. [7.3.3] As shown in Fig. 7.12 (b), the NSTX TCHERS system involves viewing chord fans both crossing and outside the NBI path to achieve separation of the charge-exchange light from carbon impurities (C-VI line at 529 nm wavelength) in the center from the intrinsic (thermally excited) emission near the edge of the plasma. The emission from each sightline is transported along optical fibers to high resolution spectrometers. Charge-coupled device (CCD) cameras record the spectra in the image plane of the spectrometers with a time resolution of 10 ms. In the H-mode pedestal region, the spatial accuracy is comparable to the local ion gyro radius which is a unique NSTX capability. With a lithium filter and different diffraction gratings, the TCHERS system can be also used to measure the lithium density profiles. ***Plans -*** The present TCHERS system is state-of-the-art and further upgrades are not currently planned for the next 5 years. However, a real-time capability for its data acquisition and analysis within the plasma control system will be installed in 2011. This will enable real-time control of the rotation profile using the NBI and non-resonant magnetic braking as the actuators.

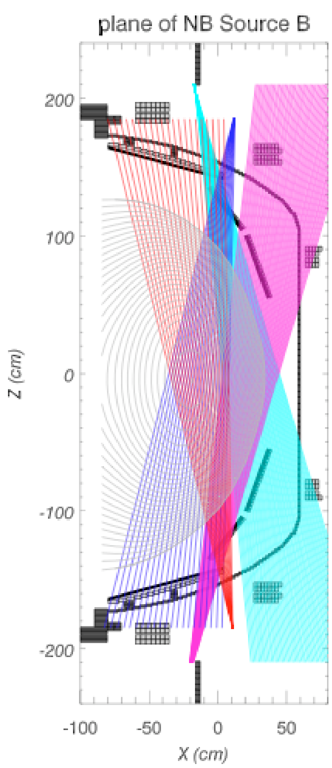


Fig. 7.13 Poloidal CHERS system schematic.

Poloidal Charge-Exchange Recombination Spectroscopy (PCHERS): ***Status -*** An interim version of the PCHERS system was installed and commissioned in FY 2007 and the full 75-channel PCHERS system, shown schematically in Fig. 7.13, was commissioned in FY 2008. This NSTX PCHERS system is a follow-on system from the TFTR PCHERS system. [7.3.4] This diagnostic, which also analyzes C-VI emission, measures the spatial profile of the poloidal plasma flow across the entire outboard minor radius with ion gyro-radius spatial resolution. In the PCHERS system, 75 pairs of complementary sightlines view the plasma from above and below where the heating neutral beam passes through the plasma. There are additional top and bottom views of the plasma which do not intersect the beam; the emission on these sightlines is used to remove the background emission from the edge of the plasma where the carbon is not fully ionized. Because the neutral beams in NSTX are relatively tall compared to the plasma cross section, the centrally viewing chords collect light from an extended region inside the plasma, so an Abel inversion of the data from all the sightlines is needed to produce the local poloidal flow velocity. This poloidal flow contributes to the total plasma "shearing rate" and thus plays a role in the suppression of large-scale turbulence. This intrinsic suppression in the ST configuration is believed to be responsible for the good ion confinement in NSTX. The poloidal flow measurements will also be used to refine the interpretation of the MSE data for accurate determination of the q-profile and MHD stability of plasmas. ***Plans -*** The NSTX PCHERS system is fully implemented and there is no plan for further upgrade.

Far-infrared Tangential Interferometer and Polarimeter: ***Status -*** Taking advantage of the excellent tangential access, the tangential far-infrared laser system (FIReTIP) simultaneously measures density and polarimetry profiles with microsecond time resolution, a unique capability as shown in Fig. 7.12 (d). [7.3.5] The FIReTIP system has also yielded important information on the movement of ELMs around the torus. ***Plans -*** The FIReTIP system will be upgraded to detect higher frequency few MHz range to measure the high frequency energetic particle modes such as CAEs and GAEs.

Motional Stark Effect using Collisionally-Induced Florescence (MSF-CIF): ***Status –*** This diagnostic was successfully introduced to NSTX in 2004 to measure the plasma current profile which is crucial for advanced ST operation. [7.3.6] A schematic of the present 16 channel MSE system is shown in Fig. 7.12 (c). The sensitivity to the field-line pitch angle is excellent even in the relatively low field NSTX plasmas. Indeed, the NSTX MSE-CIF system has achieved a factor of 12 improvement in optical throughput over earlier high field systems through the development of an advanced spectroscopic filter and geometric optimization. In addition, higher quantum efficiency detectors provide a further factor of 3.3 improvement in sensitivity, resulting in an overall factor of ~40 improvement relative to the system in TFTR, for example. The MSE data are now routinely incorporated into the equilibrium analysis for NSTX. The measured q profiles have, for example, revealed deeply reversed-shear profiles in some conditions, confirming earlier inferences based on the ultra-soft x-ray and external magnetic data. Some of the MSE channels have now also been equipped with higher sensitivity detectors to improve their time resolution to detect magnetic fluctuations. ***Plans -*** The MSE system will be upgraded toward the full 19 channel system originally planned in the near future. The more sensitive detectors for magnetic fluctuation measurements will be also installed on additional channels as needed.

Motional Stark Emission Diagnostic Using Laser-Induced Fluorescence (MSE-LIF): ***Status* –** This diagnostic has been under development by Nova Photonics Inc. under an OFES Advanced Diagnostic Initiative Grant. [7.3.7] The MSE-LIF diagnostic will view the plasma on a midplane radial sightline. The combination its data with that from the existing MSE-CIF system will determine the profiles of both the magnetic field line pitch and the radial electric field, which are both needed for plasma transport research. The MSE-LIF system will also measure the radial profile of the total magnetic field inside the plasma, providing for the first time a direct measurement of the total plasma pressure profile. By subtracting the thermal pressure profiles measured by the diagnostics discussed above, the fast-ion pressure profile can be inferred. These profiles will be compared to the predictions of classical thermalization processes to determine the influence of Alfvén Eigenmodes and other MHD activity on fast-ion confinement. ***Plans* -** The MSE-LIF system is now funded by an OFES Advanced Diagnostic Initiative Grant to be implemented on NSTX in 2009-2010. Following a successful implementation of MSE-LIF (together with its other profile diagnostic), NSTX will be in a unique position to obtain a complete plasma profile information needed for rapid data analyses and comprehensive plasma transport modeling code validation.

Other Profile Diagnostics - The NSTX density profiles are also measured by the microwave reflectometers in the edge and core region.[7.3.8, 9] To complement the TCHERS system, an Edge Rotation Diagnostic (ERD) was implemented on NSTX using impurity lines from the plasma edge region where the intrinsic emission is sufficiently strong. ERD can measure both poloidal and toroidal rotation velocities and temperatures.[7.3.10] A strong wave-particle interaction was observed which produces significant (~400 eV) perpendicular ion heating and rotation at the edge during application of high harmonic fast wave (HHFW) power. The radiated power loss profile (and total radiated power) from the plasma are monitored by the 16 channel mid-plane tangential core bolometer system. [7.3.11] The impurity profiles are being monitored by the newly commissioned VUV transmission grating spectrometer (TGS). [7.3.12]

7.3.2 MHD Diagnostics

Magnetics and between-shots EFIT equilibrium reconstruction: ***Status* -** Accurate reconstruction of the plasma MHD equilibrium is crucial for investigating and optimizing plasma stability. Utilizing the extensive external magnetic data together with the plasma internal profile information, the plasma reconstruction code "EFIT", extended and adapted to NSTX plasma conditions, provides detailed information on the basic plasma properties with time resolution typically down to 1 ms. A hierarchy of successively more sophisticated analysis steps has been developed with EFIT to provide experimenters first with immediate feedback on their experiments and then for detailed analysis of the results afterwards. [7.3.13] These analyses which use more extensive diagnostic data and an increasingly comprehensive model for the plasma have been added as the capabilities and breadth of the NSTX diagnostics have improved over the years, and are routinely applied as time increases after a shot and the input data is analyzed and becomes available. For example, the first level of EFIT analysis uses the measured coil currents and external magnetic measurements; this analysis is available for the entire shot duration 2 – 3 minutes after a shot. The next level includes the diamagnetic measurement and the electron pressure profile measured by the multi-point, multi-pulse Thomson scattering diagnostic to constrain the total pressure profile shape. Thereafter, on a timescale typically of hours, data from measurements of the ion rotation profile measured spectroscopically and the internal magnetic field structure measured by the motional-Stark-effect (MSE) diagnostic are included. The EFIT analysis incorporating MSE data is already yielding important and exciting data on magnetic shear reversal in NSTX discharges. The EFIT code has also been linked to the plasma MHD stability analysis code DCON for rapid between-shots evaluation of plasma MHD stability. ***Plans -*** The NSTX magnetics and EFIT analysis system will continue to be improved and optimized as the additional magnetic sensors and profile information and the analysis codes becomes available.

Soft X-ray MHD diagnostics:

***Status* -** The poloidal Ultra-soft X-ray (USXR) arrays have been serving as the primary diagnostic tool for the localization and imaging of internal MHD mode activity on NSTX. [7.3.14] A set of selectable filters allows for discrimination of the SXR contributions from the edge, bulk, and core of the plasma. With a mode frequency capability from DC to >100kHz, the arrays have been used to study a broad range of MHD behavior from the plasma edge to the core including $n = 1$ modes, NTMs, ELMs, and energetic particles modes. An identification of tearing mode activity provided, for example, the first indication that electron transport barriers were generated in reversed shear NSTX plasmas, later confirmed by MSE in 2004. The multi-energy or 'multicolor' SXR (ME-SXR) array simultaneously views the same plasma through a set of three SXR filters. The ratios of the filtered emission are used to provide fast T_e profiles on timescales of ≥ 0.1 ms, using normalization from intermittent MPTS measurements. [7.3.15] Continuous Te profiles have been obtained during MHD activity such the ELM and the RWM. The fast profiles have also been used to study core electron temperature perturbations, initiated both by ELMs (2007) and by pellet injection (2006). The perturbative studies confirmed rapid electron transport in the central plasma, possibly linked to shear Alfven MHD activity (2008). In 2004, a fast tangential soft x-ray camera for 2-D view of core MHD activity with a time resolution down to 2 ms was installed. [7.3.16] The core soft-x-ray (1–5 keV) signal is converted into visible light by a fast phosphor deposited on fiber-optic faceplate. The resulting image is captured by a CCD camera with 64 x 64 pixels with frame rates up to 500 kHz for 300 frames. Such a camera is a powerful tool to investigate the 2-D MHD mode dynamics including the sawtooth crash.

Plans - The Soft X-ray diagnostic upgrade plan includes (i) A *high resolution ME-USXR/ME-VUV tangential array* for fast measurements of the edge plasma profiles and (ii) *Two toroidally displaced, high resolution ME-SXR tangential arrays* for non-magnetic identification and measurement of low-frequency MHD modes (RWM, ELMs and disruptions). The edge multi-energy USXR/multi-energy VUV tangential array will cover the plasma from 0.8<r/a<1.1 and it will be a 'hybrid' device, incorporating together with a low-energy USXR filter set, also a VUV Transmission Grating (TG) dispersive and imaging element. The edge hybrid array will pave the path towards the implementation of a 2-D USXR/VUV tomographic array for divertor diagnostic in the 2011-2013 timeframe. Using a multi-energy technique that will include line emission in addition to the continuum, the device will aim to measure with high resolution (≥ 1 cm) and speed (≥ 1 ms) the edge T_e and $n_e x n_z$ profiles, together with their MHD perturbations. The two toroidally displaced ME-SXR tangential arrays will cover the plasma from 0<r/a<0.9 and will serve to identify and discriminate between toroidally symmetric profile perturbations, such as from the ELM, and between toroidally asymmetric perturbations, such as from the n=1 RWM. In addition, through comparison with predictions, the two arrays will provide unique information about the internal structure of low frequency MHD phenomena, such as the RWM or the disruption. These diagnostics also project into the research plans beyond 2011. The system of n=1 toroidally displaced ME-SXR arrays will be upgraded in the same period for the identification and diagnostic of low frequency MHD with n>1. In addition, this system can be used to test the possibility of active control of the RWM and the ELM using ME-SXR feedback, instead of magnetic feedback. Lastly, following verification of if a connection between shear Alfven activity and electron transport in NSTX is confirmed, an Extreme Ultraviolet BES (XUV-BES) system for the imaging of fast Alfven modes such as the GAEs, in the central NSTX plasma (r/a≤0.4) can be developed. The system will use focusing multilayer mirrors or Zone Plates to image with very high speed, few cm-sized perturbations in the heating or diagnostic beam XUV emission. It should be noted that the NSTX soft x-ray diagnostics plan is mainly carried out by the John Hopkins University collaboration. For more detail, their plan is described in the Appendix.

7.3.3 Turbulence Diagnostics

To investigate the sources of anomalous plasma transport, a set of core fluctuation diagnostics has been and continues to be developed in NSTX. The first element is a microwave reflectometer, operational since 2004, to measure longer wavelength fluctuations in the plasma core. Far-infrared interferometery with the FIReTIP diagnostic system is also being used to measure the edge density fluctuations in NSTX. A powerful addition to the NSTX turbulence diagnostics is a tangential microwave high-k scattering system installed in 2005 to measure the electron-transport-relevant short-wavelength modes including, possibly, Electron-Temperature-Gradient Modes (ETGs). This system has already yielded extensive electron transport relevant fluctuation data under a variety of plasma conditions including H-mode, reversed shear, and strongly electron heated HHFW plasmas. A proposed addition is Beam Emission Spectroscopy (BES) to complete the measurement of local turbulent fluctuations across the relevant range of wave-numbers to test turbulence transport models. A summary of the existing and planned fluctuation diagnostics for NSTX appears in Fig. 7.14.

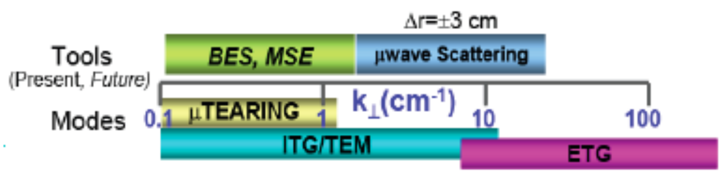


Fig. 7.14 NSTX turbulence diagnostics and expected wave number range of transport relevant fluctuations.

High-k Microwave Scattering System: ***Status -*** The NSTX "high-k" turbulence diagnostic is a coherent low-angle scattering system that can measure simultaneously turbulent fluctuations of the plasma density at five different wave-vectors perpendicular to the local magnetic field in the range $1 < k_\perp < 20\ cm^{-1}$. [7.3.17] Due to the relatively low magnetic field in NSTX, the fluctuations at the upper end of the k-range approach the spatial scale of the electron gyro-motion, i.e. $k_\perp\rho_e \sim 1$, where ρ_e is the typical local electron gyro-radius. The system, illustrated in Fig. 7.15, takes advantage of the large curvature of the magnetic field lines in the ST geometry to achieve a radial resolution $\Delta R \sim 6$ cm of the measurements. By moving the launching and collection mirrors, the scattering volume can be positioned throughout the out-board plasma radius. Wave vectors of these fluctuations are mainly perpendicular to magnetic surfaces, but also have small components along the plasma diamagnetic velocity and the toroidal plasma current that could be used for determining the phase velocity of fluctuations. Measuring plasma fluctuations on this scale can provide insight into the long-standing mystery of anomalous electron thermal transport. Indeed the high-k scattering has been highly productive in providing the data for the electron gyro-scale turbulence in the recent yeas. ***Plans -*** Due to its importance to the study of electron transport which is central to this Five Year Plan, upgrades to the high-k scattering system are underway and planned. The system is currently being modified for remote control of its launching and collection mirrors to provide between-shots spatial scanning of the scattering volume. Over the 2008 summer outage, the 1mm microwave source will be upgrade to a solid state diode. Next, the

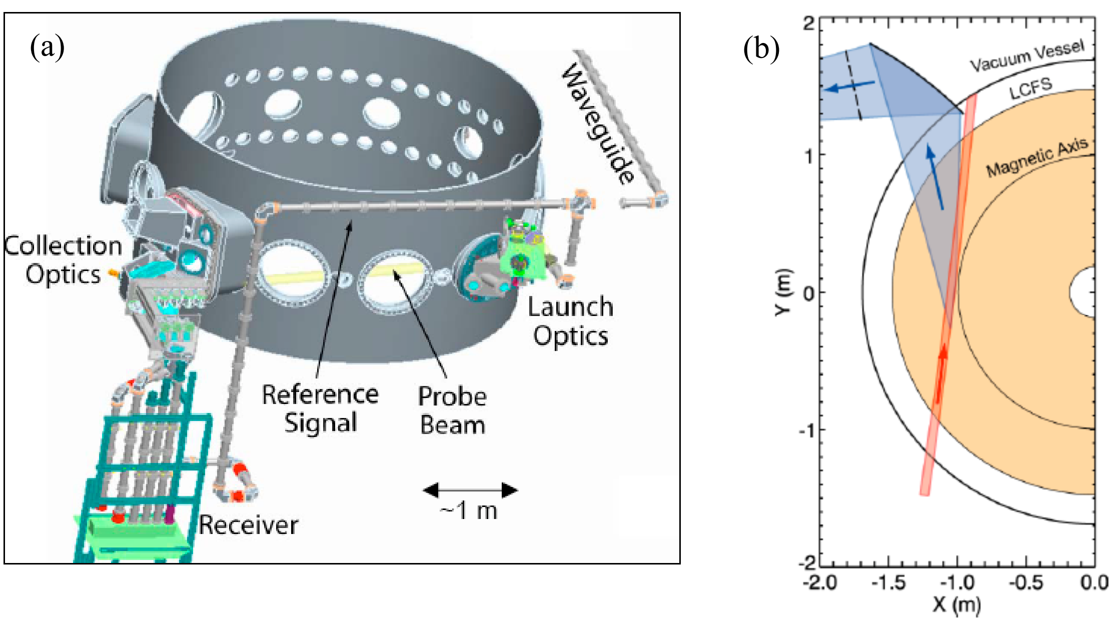


Fig. 7.15 NSTX High-k Scattering system. (a) Layout of the system; (b) Tangential scattering geometry.

system will be upgraded to measure the k_θ component of the turbulence in addition to k_r information currently obtained.

Beam Emission Spectroscopy (BES) System: ***Status -*** A BES diagnostic developed by the University of Wisconsin under an OFES Innovative Diagnostic Initiative has been successfully deployed on DIII-D. [7.3.18] The BES system proposed for NSTX will enable direct measurements of longer wavelength density fluctuations in the plasma core providing valuable insights into the suppression of ion turbulence and attainment of the near-neoclassical ion confinement observed on NSTX. The BES diagnostic together with the existing microwave tangential scattering diagnostic (which measures medium to short wavelength turbulence) will provide the most comprehensive turbulence diagnostic set of any ST in the world. The BES diagnostic should also enhance the determination of the spatial structure of fast-ion-driven instabilities such as the TAE and BAAE observed on NSTX. ***Plans -*** A 32-channel BES system for NSTX, illustrated in Fig. 7.16, is being designed with guidance from the University of Wisconsin group. In 2009, a prototype two-view BES system will be installed on NSTX with the first light expected during the 2009 plasma operation. The system is expected to achieve its full 32 channel capability by 2010.

7.3.4 Energetic Particle Diagnostics

Fast-ion confinement in STs is an important issue, due to the large ratio of their gyro-radius to

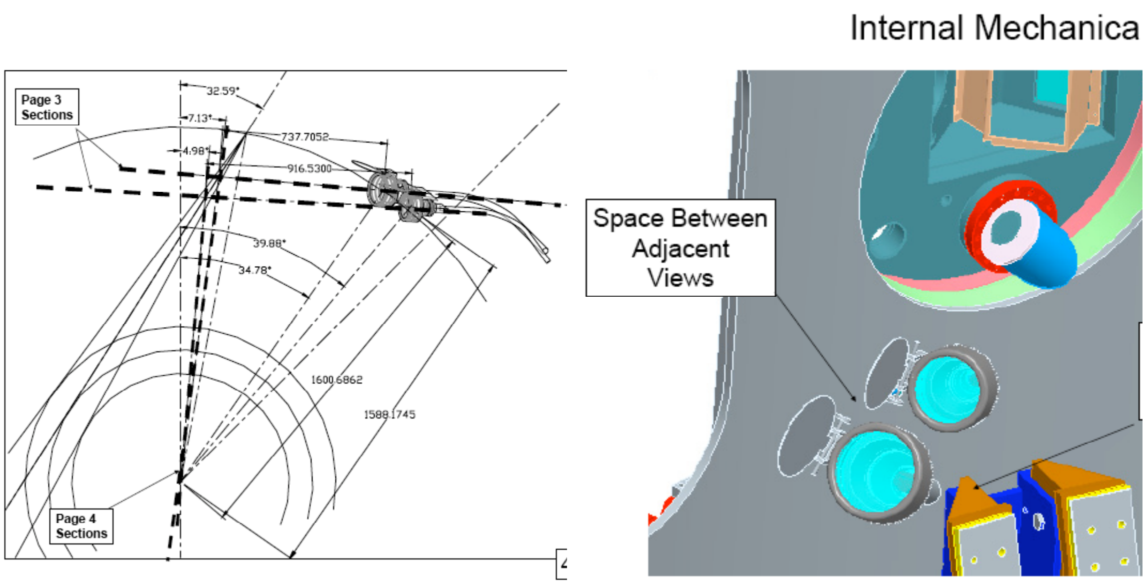


Fig. 7.16 BES diagnostic being developed for NSTX (a) viewing geometry; (b) Schematic of internal optical system providing two viewing cones

plasma minor radius. NSTX is also uniquely positioned to investigate ITER-relevant energetic-particle instabilities and their consequences. In NSTX high-β plasmas produced by neutral beam injection, there is a substantial population of energetic ions whose velocity exceeds the local Alfvén velocity, similar to the energetic alpha-particle population expected in ignited plasmas in ITER. Such fast ions can excite many types of instabilities which have been and will continue to be studied with a comprehensive set of plasma diagnostics, including several specialized diagnostics, described below, plus MSE for the j(r) profile, high-frequency magnetic pick-up coils, correlation reflectometers, the tangential FIR interferometer array, the ultra-fast x-ray camera, and soft x-ray tomography.

Energetic Ion Diagnostics: ***Status -*** Fast-ion diagnostics on NSTX include a set of neutron detectors, a scanning energetic neutral-particle analyzer (NPA), a multi-sightline solid state detector NPA (SSNPA), and a scintillator-based fast lost-ion probe (sFLIP). [7.3.19] Since the DD fusion neutron rate in NSTX is dominated by beam-target reactions, the neutron measurements provide a global measure of beam ion confinement. The detectors include two absolutely-calibrated fission chambers and four scintillator detectors capable of resolving fluctuations in the neutron rate up to 100 kHz. The neutron response produced by short neutral beam pulses has shown classically-expected beam ion confinement in quiescent plasmas. The NPA can measure fast neutrals with energies up to 120 keV with an energy resolution of ~1 keV, and can be scanned horizontally and vertically on a shot-to-shot basis. [7.3.20] The NPA measures the fast ion population at the point of charge exchange and its ability to probe the fast ion population in the core of the plasma is greatly enhanced by the NBI providing the source of neutrals. The SSNPA has four fixed sightlines in the mid-plane with ~10 keV energy resolution and is designed to measure radial redistribution of neutral beam ions arising from MHD activity.[7.3.23] The sFLIP measures the energy and pitch angle of the escaping fast ions entering the detector, allowing determination of the orbits that are lost. Initial sFLIP measurements also appear consistent with classically-expected confinement in quiescent plasmas. However, MHD activity has been seen by the NPA to deplete the confined beam ion population and result in loss to the vessel wall of orbits with high pitch angle. The newest addition to the complement of fast ion diagnostics is the fast ion D-alpha (FIDA) line emission camera, illustrated in Fig. 7.17, which was successfully implemented in 2008 to measure light emitted by fast ions in the plasma which undergo charge exchange with injected beam neutrals. [7.3.21] ***Plans -*** A neutron collimator system is being designed to quantify the fast ion population in the plasma core region to complement the existing FIDA and NPA diagnostics.

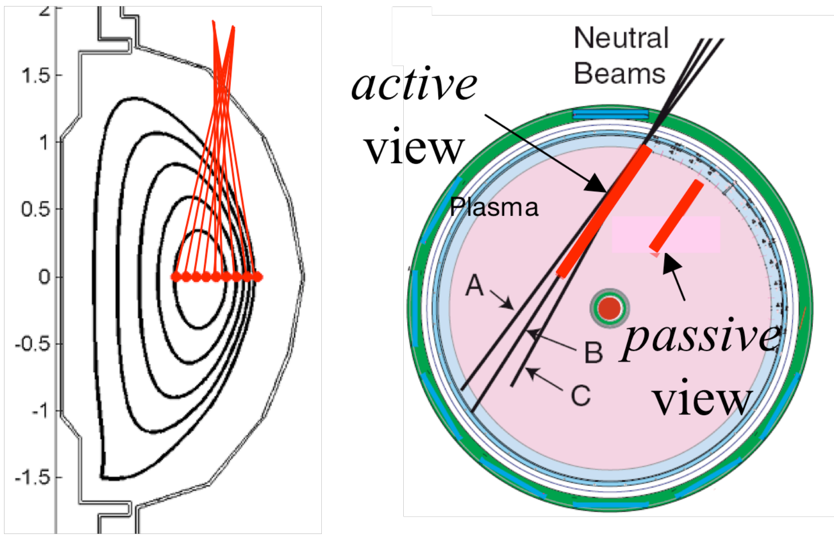


Fig. 7.17 FIDA diagnostic on NSTX (a) Poloidal cross sectional view; (b) Top view showing active and passive views.

Energetic-Particle-Driven Mode Diagnostics: ***Status -*** For studying the MHD activity driven by energetic particles, high frequency magnetic sensors with up to 5 MHz response are operational on NSTX. [7.3.22] The high-frequency MHD instabilities typically observed include Toroidal Alfvén Eigenmodes (TAEs) and Compressional Alfvén Eigenmodes (CAEs). These observed modes were reproduced by the kinetic high frequency MHD codes HYM, Nova-K and M3D-k. At present, the microwave reflectometer system and the poloidal Ultra-soft X-ray (USXR) arrays provide localized internal measurements of the mode amplitudes, with the FIReTIP and high-k systems providing complementary data extending further into the plasma core. ***Plans -*** The planned 32 channel BES system in 2009-2010 time frame will be a powerful addition to the energetic particle mode diagnostic.

7.3.5 Wave Diagnostics

HHFW-related diagnostics: ***Status -*** Several diagnostics are used to understand the physics of HHFW coupling. An edge reflectometer has been used to measure the electrostatic waves generated by the edge parametric decay instabilities. Three RF pickup probes are placed around the torus to measure the RF wave fields in the plasma edge region. The core reflectometer system has been used to measure the core HHFW density fluctuations. ***Plans -*** Additional diagnostic to measure directly the HHFW fluctuations in the core will be implemented to assess the wave propagation to the core. The existing high-k tangential scattering system will be upgraded to cover the 30 MHz range of the HHFW and FIReTIP will be also upgraded to measure the core HHFW density fluctuations.

EBW Radiometer system: ***Status -*** To assess the EBW conversion (coupling) efficiency, two microwave radiometers were installed and commissioned on NSTX in 2006. [7.3.23] These radiometers measure thermal EBW emission in the plasma mode converted from EBW through X-mode and finally to O-mode radiation in the bands 8 – 18 GHz and 18 – 40 GHz. Each radiometer views the plasma from the midplane obliquely at toroidal and poloidal angles of 20 – 40° approximately and measures radiation in two orthogonal polarizations using a quad-ridged antenna. ***Plans -*** The EBW radiometer will be utilized on NSTX to monitor the EBW emission under a variety of plasma condition. The system will be refurbished and improved as needed.

7.3.6 Boundary Physics Diagnostics

Status - The NSTX boundary physics diagnostic set has been developed to support and strengthen the on-going research in scrape-off layer transport and turbulence, experiments with lithium coatings, and ELM and pedestal physics. A good access to plasma through diagnostic ports and good port coverage of important plasma regions on NSTX enabled a number of unique and important measurements. A schematic summarizing NSTX boundary physics diagnostics is shown in Fig. 7.18. Scrape-off layer parameters in the inner and outer midplane locations are measured by the fast reciprocating probe [7.3.24], fast reflectometers [7.3.25], VUV and UV-visible spectrometers [7.3.26] and cameras [7.3.27-30] that are spectrally filtered for deuterium and impurity line emission. Unique turbulence measurements in the midplane and divertor are provided by several fast cameras in combination with the gas puff imaging system [7.3.31,32]. A comprehensive set of divertor diagnostics includes infrared [7.3.33] and visible cameras for heat flux, recycling and impurity emission measurements, tile Langmuir probes, a 4-channel divertor bolometer system [7.3.11], a UV-visible spectrometer, and neutral pressure gauges [7.3.34]. ELM and pedestal measurements are addressed by the Multi-point Thomson Scattering system, the CHERS system,

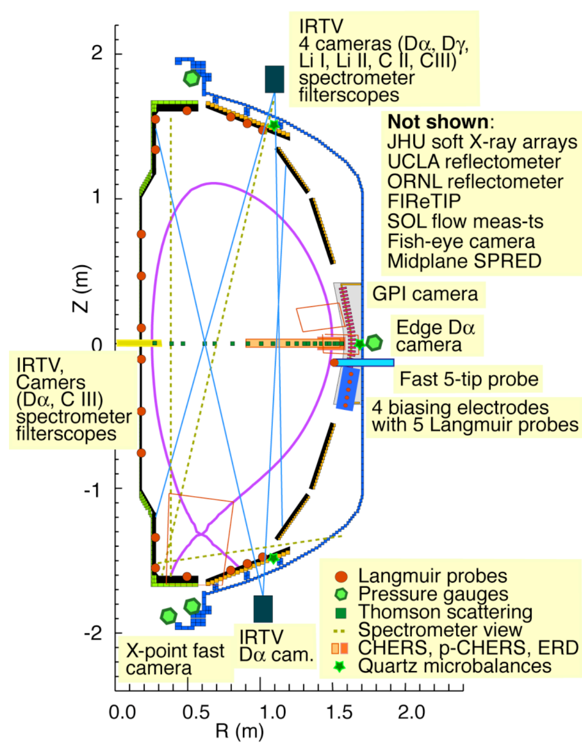


Figure 7.18 NSTX Boundary Physics diagnostics

the edge rotation diagnostic, and fast cameras. Measurements of plasma-wall interaction are achieved with unique surface deposition monitors [7.3.35], dust detectors [7.3.36] and dust-tracking system [7.3.37].

Plans - To support the Boundary Physics program and the planned facility upgrades in 2008-2013, a number of new diagnostics and diagnostic upgrades are planned. To understand the liquid lithium divertor performance and its effects on plasma, measurements of particle and heat fluxes in the challenging lithium environment will be developed. To this end, additional thermocouples and Langmuir probes embedded in "smart" diagnostic tiles, two-color IR measurements, Ly-α diode arrays, an imaging UV-visible divertor spectrometer, and a surface analysis probe will be implemented in 2008-2010. Additional MPTS channels will be commissioned to improve T_e, n_e profile resolution in the pedestal and scrape-off layer regions in 2010-2011. Further improvements are planned to the on-going developments of a unique scrape-of layer flow measurement system based on filtered camera imaging. To improve our understanding of the SOL and divertor power balance, a staged upgrade of the divertor bolometer system is planned in 2008-2009 to include a full coverage of the divertor. Together with the existing mid-plane bolometer system, the new 20-channel system should enable a tomographic reconstruction of 2D radiated power patterns in the divertor. A fast IR camera will be also implemented in 2008-2009 to measure the large heat fluxes due to transient events and ELMs. To measure ELM parameter evolution on the fast scale, new high-resolution soft X-ray arrays are planned in 2012-2013. Also in 2012 or sooner, if resources become available, a divertor Thomson scattering system and an X-point reciprocating probe are planned.

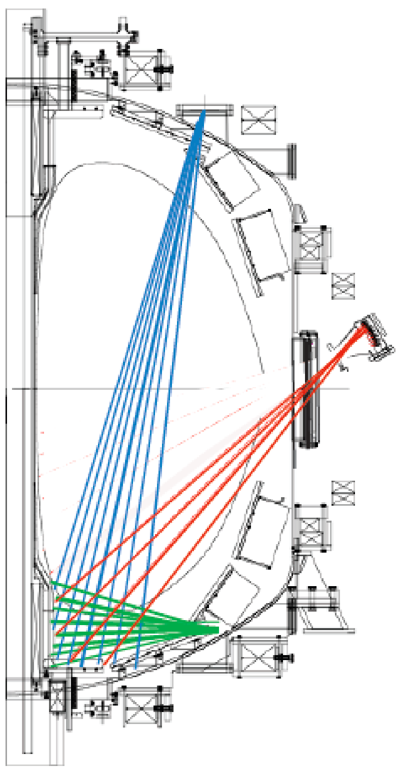


Fig. 7. 19 Three-view divertor bolometer system

7.4 NSTX Facility Utilization

Facility Operations Summary - The NSTX facility utilization for the preceding five year period (FY 2004- 2008) is summarized in Table 7.4. The NSTX facility has met or exceeded the planned run weeks for all five years. The facility operational efficiency has increased significantly over those years. One measure of operational efficiency is the number of plasma shots per run week (defined as 40 hours of operation). The total number of plasma shots is also directly related to the productivity of experimental research. As shown in the Table 7.4, the number of plasma shots per week has increased steadily by about 33% over the five year period from 117 plasma shots per week in FY 2004 to 155 plasma shots per week FY 2008. In fact, in 2008, the facility has produced a record number of plasma shots of 2571 in just 16.6. run week compared to 2460 plasma shots in 21 run weeks in 2004. This increased efficiency is attributable to many factors including the maturing of the facility, the increased experience of the operations team, the improvements in the real time plasma control system, and general improvements of plasma conditions resulting from the development and use of tools such as the lithium evaporator.

	FY 2004	**FY 2005**	**FY 2006**	**FY 2007**	**FY 2008**
Run weeks planned / achieved	20/21	17/18	11/12.7	12/12.6	15/16.6
Hours of operation planned / achieved	800/844	680/720	440/508	480/504	600/664
Total Plasma Shots	2460	2221	1617	1879	2571
Plasma Shots per Run Week	117	123	128	149	155

Table 7.4 FY 2004-2008 Facility Operations Summary

Facility Users Summary - Table 7.5 lists the research users from PPPL/Princeton University and non-PPPL institutions which include national as well as international collaborating researchers. Over 200 researchers have participated in the NSTX research in FY 2007. The number has been relatively steady over the years. However, it should be noted that participation by younger researchers (post-doc, graduate students) is rising, more than doubling from 18 in 2004 to 42 in 2007.

	PPPL	***non-PPPL***
Researchers	***57***	***145*****
Post Doc.	***1***	***10***
Grad. Students	***9***	***12***
Undergrad. Students	***1***	***10***

** There are over 41 overseas collaborating researchers from countries including Canada, China, Czech Republic, France, Germany, Israel, Italy, Japan, Korea, Russia, and UK, during FY 2006-2007.

Table 7.5 Participating Research Personnel Summary

Facility Operations Plan - The NSTX facility utilization plan for the next Five Year Period (FY 2009- 2013) is summarized in Table 7.6. The NSTX facility 20 weeks per year will enable the NSTX research team to make a satisfactory progress on the proposed NSTX Five Year Plan while implementing the necessary upgrade including the new Center Stack to be installed prior to the FY 2012 run. The reduced run weeks in FY 2013 is to allow the second NBI to be installed on NSTX.

Planned Facility Base Plasma Operations

	FY 2009	**FY 2010**	**FY 2011**	**FY 2012**	**FY 2013**
Run weeks planned	15	15	15	15	15
Hours of operation planned	600	600	600	600	600

Planned Facility Incremental Plasma Operations

	FY 2009	**FY 2010**	**FY 2011**	**FY 2012**	**FY 2013**
Run weeks planned	20	20	20	15	15
Hours of operation planned	800	800	800	600	600

Table 7.6 FY 2009-2013 Facility Utilization Plan

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