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# NSTX: Facility/Research Highlights and Near Term Facility Plans

M. Ono for the NSTX Research Team

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# NSTX: Facility/Research Highlights and Near Term Facility Plans*

M. Ono for the NSTX Research Team

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## Abstract

The National Spherical Torus Experiment (NSTX) is a collaborative mega-ampere-class spherical torus research facility with high power heating and current drive systems and the state-of-the-art comprehensive diagnostics. For the 2008 experimental campaign, the high harmonic fast wave (HHFW) heating efficiency in deuterium improved significantly with lithium evaporation and produced a record central $T_e$ of 5 keV. The HHFW heating of NBI-heated discharges was also demonstrated for the first time with lithium application. The EBW emission in H-mode was also improved dramatically with lithium which was shown to be attributable to reduced edge collisional absorption. Newly installed FIDA energetic particle diagnostic measured significant transport of energetic ions associated with TAE avalanche as well as n=1 kink activities. A full 75 channel poloidal CHERS system is now operational yielding tantalizing initial results. In the near term, major upgrade activities include a liquid-lithium divertor target to achieve lower collisionality regime, the HHFW antenna upgrades to double its power handling capability in H-mode, and a beam-emission spectroscopy diagnostic to extend the localized turbulence measurements toward the ion gyro-radius scale from the present concentration on the electron gyro-radius scale. For the longer term, a new center stack to significantly expand the plasma operating parameters is planned along with a second NBI system to double the NBI heating and CD power and provide current profile control. These upgrades will enable NSTX to explore fully non-inductive operations over a much expanded plasma parameter space in terms of higher plasma temperature and lower collisionality, thereby significantly reducing the physics parameter gap between the present NSTX and the projected next-step ST experiments.

## 1. Introduction

The National Spherical Torus Experiment (NSTX) is a collaborative spherical tokamak (ST) facility with participating researchers from over 20 US and 15 overseas institutions [1]. The NSTX can access an exceptionally wide operating plasma parameter space (e.g. $\beta_T$ up to 40%, $\beta_N$ up to 7 %.m.T/MA, $\kappa$ up to 3, $V_{fast}/V_{Alfvén}$ ~ 5, and $V_{flow}/V_{Alfvén}$ ~ 1), a high degree of facility flexibility, and comprehensive, state-of-the-art diagnostic systems [2]. The NSTX mission elements are to: 1) Determine the physics principles of the ST, utilizing its low aspect-ratio (A ~ 1.5) and very high ratio of plasma pressure to magnetic pressure; 2) Establish attractive next-step ST operating scenarios and configurations for plasma-material-integration with high performance, an ST-based fusion neutron science facility, and an ST Demonstration

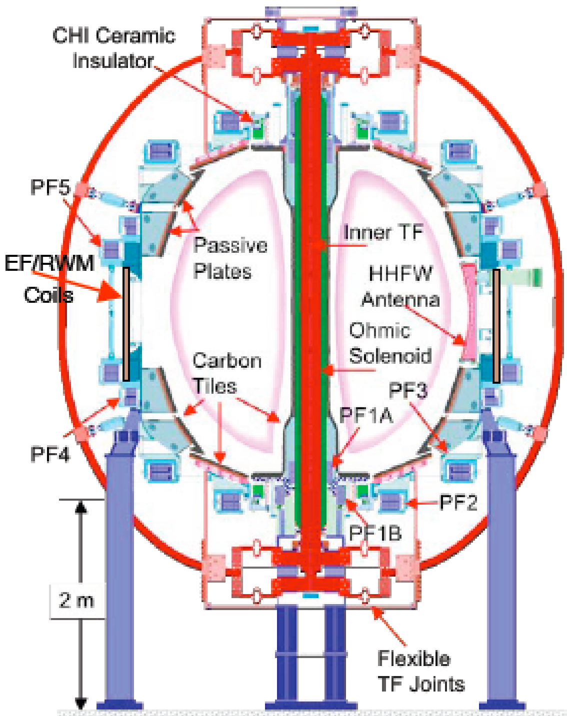


**Fig. 1. NSTX Device Cross-sectional View.**

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Power Plant; 3) Support final design activities and preparation for burning plasma research in ITER through the International Tokamak Physics Activities (ITPA), and benefit from it; 4) Complement and extend conventional higher aspect-ratio (A ~ 3) and lower-$\beta$ experiments in addressing key scientific issues of toroidal fusion plasmas [3].

The NSTX plasma is surrounded by closely-fitting conducting plates (shown in Fig. 1), which passively stabilize pressure-driven modes, provided the plasma toroidal rotation exceeds a critical frequency. Together with non-axisymmetric active feedback control coils and fast real time plasma control system, NSTX routinely produces highly shaped plasmas with total pressure above the "no-wall beta limit"[4]. The NSTX plasmas are heated by up to 6 MW High-Harmonic Fast Waves (HHFW) and up to 7 MW of deuterium Neutral Beam Injection (NBI). 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 high 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.

## 2. High Harmonic Fast Wave Heating and Electron Bernstein Wave Emission Experiments:

The NSTX HHFW system is designed to heat electrons and drive plasma currents for non-inductive current ramp-up and sustainment [5]. 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 is driven by six power amplifiers operating at 30 MHz with delivered power of up to 6 MW [6]. High priority research topic this year is a demonstration of efficient electron heating in deuterium plasmas. Previously, the HHFW heating in deuterium plasmas has been significantly less efficient than in helium. The utilization of lithium evaporator enabled efficient core electron heating in deuterium as in helium. As shown in Fig. 2, the HHFW has heated the core electrons from the few hundred eV in the ohmic phase to as high as 5 keV in deuterium plasmas. With lithium application, the HHFW core electron heating was also observed in NBI heated deuterium plasmas. Lithium coating of the plasma facing components, which decreases the edge density [6] and recycling, has improved 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. This physics current drive efficiency is now being investigated using the MSE diagnostic to measure the effects on the local current density and advanced HHFW model calculations [7].

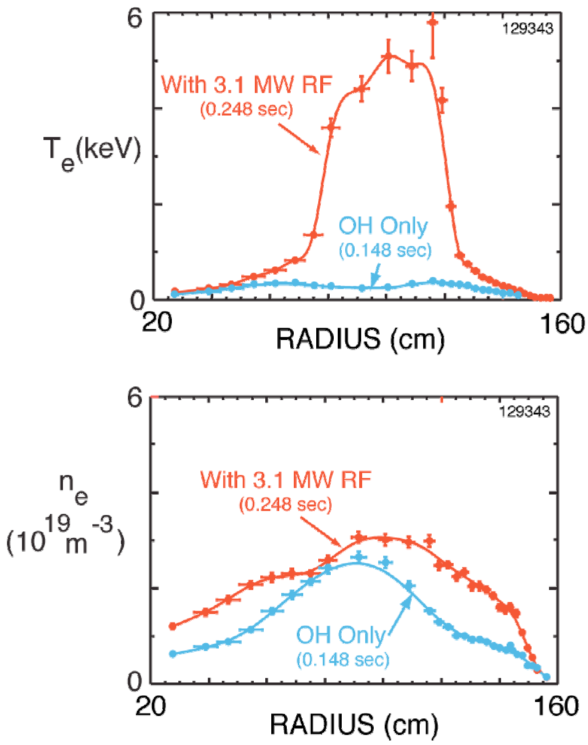


**Fig 2. HHFW electron heating in deuterium plasma**

Electron Bernstein Waves (EBW) are a promising tool to drive localized off-axis current needed for sustained operation in advanced ST modes. 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 in part due to the reduced edge collisionality due to lithium [8].

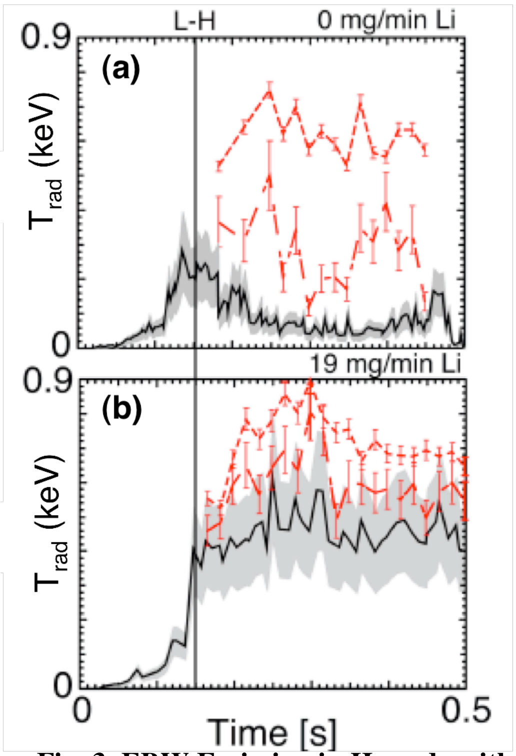


**Fig. 3. EBW Emission in H-mode without (a) and with (b) lithium. Solid curves - measured, dashed curves - simulation no collision, and dotted curves - with collision**

## 3. Energetic Particle Physics

Fast-ion confinement in STs is an important issue, due to the large ratio of their gyro-radius to plasma minor radius. NSTX is also uniquely positioned to investigate ITER-relevant energetic-particle instabilities and their consequences. In the 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.

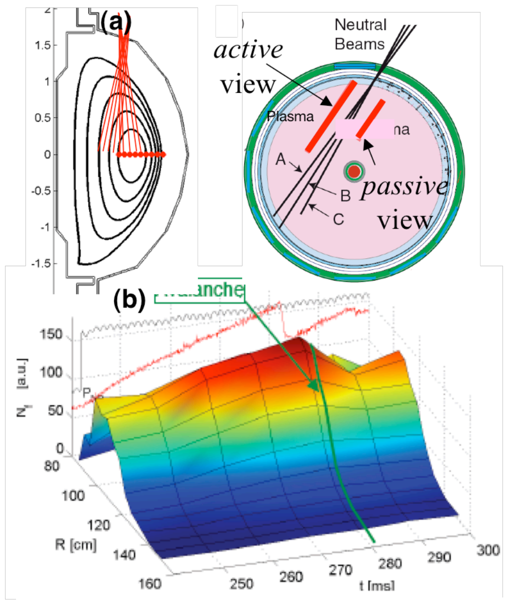


**Fig. 4. FIDA energetic particle profile diagnostic system. (a) FEDA schematic. (b) Energetic particle profile evolution during TAE avalanche.**

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) [9]. 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 by the sFLIP. The newest addition to the complement of fast ion diagnostics is the fast ion D-alpha (FIDA) line emission camera, illustrated in Fig. 3(a), which was successfully implemented in 2008 to measure light emitted by fast ions in the plasma which undergo charge exchange with injected beam neutrals with an initial data with the TAE avalanche is shown in Fig. 3(b) [10, 11]. For studying the MHD activity driven by energetic particles, high frequency magnetic sensors with up to 5 MHz response are operational on NSTX [11]. 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. The planned 32 channel BES system in 2009-2010 time frame will be a powerful addition to the energetic particle mode diagnostic.

## 4. New Diagnostic Capabilities

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. The 75 channel Poloidal Charge-Exchange Recombination Spectroscopy (P-CHERS), Beam Emission Spectroscopy (BES), MES-LIF, and Divertor Bolometer are becoming available in the near term.

The PCHERS, shown schematically in Fig. 7.13, was commissioned in FY 2008 [12]. 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 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 P-CHERS system is now producing tantalizing ion gyro-scale poloidal flow profile measurements on NSTX.

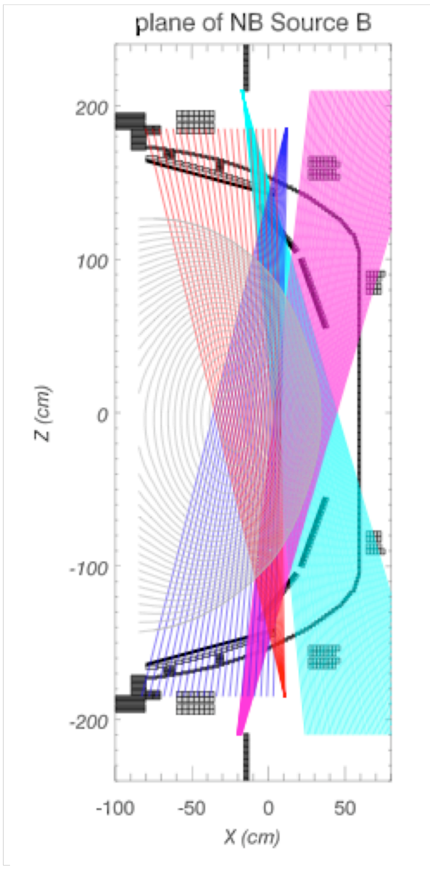


**Fig. 5. 75 ch P-CHERS system schematic with top and bottom views**

The 32-channel BES system is being designed and installed on NSTX in collaboration with University of Wisconsin 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 [13]. The BES diagnostic together with the existing microwave tangential scattering diagnostic (which measures medium to short wavelength turbulence) will provide a comprehensive turbulence diagnostic set to develop predictive capability for the performance of next-step STs. 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. The system is expected to achieve its full 32 channel capability by 2010.

Motional Stark Emission Diagnostic Using Laser-Induced Fluorescence (MSE-LIF) by Nova Photonics Inc. will be implemented under the innovative diagnostic initiatives [14]. 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

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 by ORNL will be also implemented in 2008-2009 to measure the large heat fluxes due to transient events and ELMs.

## 5. NSTX Research Plans and Associated Facility Upgrades:

The NSTX has recently gone through a DOE review of its Five Year Plan (2009 - 2013) [3]. In the near term, the main goals for NSTX research plan are to: 1) understand and increase beam-driven current at lower $n_e$, $\nu^*$; 2) understand and improve H-mode confinement at low $\nu^*$; 3) demonstrate and optimize non-inductive start-up and ramp-up; and 4) sustain high $\beta_N$ and understand MHD near and above the no-wall limit.

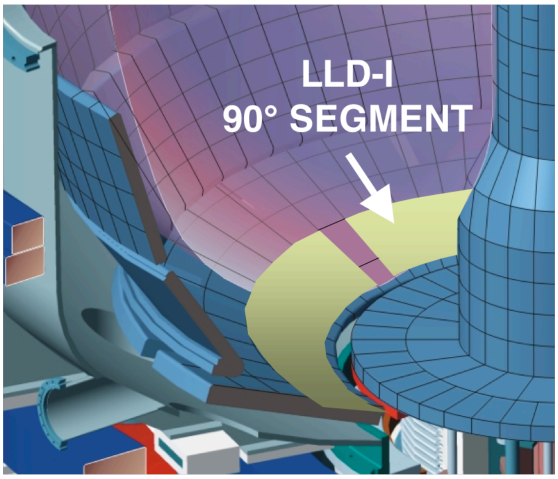


**Fig. 6. Lquid Lithium Divertor target schematic**

To support these research goals, major upgrade activities for the 2009 run are a liquid-lithium divertor target to achieve lower collisionality regime and to upgrade the HHFW antenna to double its power handling capability and ELM resilence to improve the electron heating capability of H-mode plasmas. Following the success of the lithium evaporator development and experiments, the NSTX team is focusing its effort on a liquid lithium divertor (LLD) target. This initiative is a collaboration between PPPL and a team at Sandia National Laboratory (SNL). 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 NSTX HHFW antenna system is being upgraded to a symmetric-feed configuration. A schematic view of the modification which will be implemented in FY 2009, is shown in Fig. 7. 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. NSTX is also planning to utilize the 27 GHz 350 kW sources from ORNL to utilize it on the start-up and EBW research.

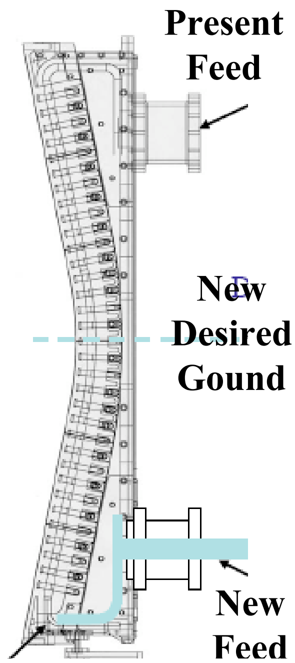


**Fig. 7. HHFW double antenna feeds upgrade schematic**

For the longer term, a new center stack is planned to greatly expand the operating parameter space along with a second NBI system to double the NBI heating and CD power and provide current profile control. The present NSTX collisionality is about a factor of 30 higher than the anticipated very low collisionalities in the next-step STs. The new center-stack is projected to reduce the collisionality gap by about one order of magnitude. Closing the collisionality gap will significantly increase the confidence level of the performance projection of the next-step STs. The new center-stack will contribute to all science areas including:

- Determine if the favorable ion and electron confinement scaling, and turbulence variation, observed in NSTX continues to higher magnetic fields and lower collisionality.

- Determine if the MHD stability of neoclassical tearing modes and resistive wall modes is favorable at lower collisionality and higher magnetic field.

- Using the higher magnetic field, which improves the efficiency of plasma heating and current drive both for RF and NBI, push further towards fully non-inductive start-up and current sustainment for long pulses.

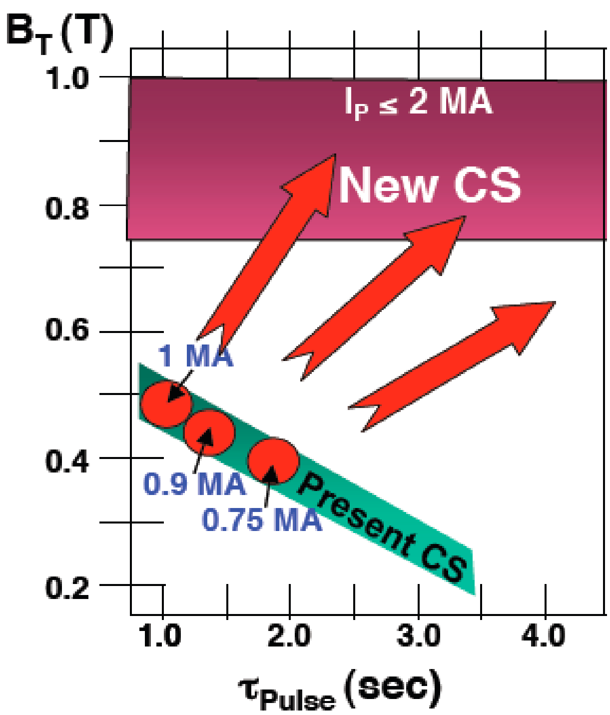


**Fig. 8. Target operating range of new center-stack.**

The new center-stack can be installed relatively easily by removing the 72 flexible joints (36 top and 36 bottom) of the TF coil as shown in Fig. 1 (a). In addition to the TF and OH coils, the new center-stack will also include top and bottom PF 1A, 1B, and 1C coils to retain divertor and plasma shaping flexibility. The new design will incorporate improved TF joint design to minimize lift off. (e.g. very high expansion for X-divertor). It will also incorporate additional capabilities (e.g. pellet guide tube for inside launch) to enhance physics capability.

For the second Neutral Beam Injection (NBI) system, the TFTR NBI system with three sources will be refurbished and moved to NSTX, thereby doubling its beam power on NSTX. 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. as shown in Fig. 9(a). The second NBI will also increase the current drive efficiency and provide for much greater current profile control flexibility due to the more highly tangential injection angleas shown in Fig. 9 (b). This will provide access to fully non-inductive operations with higher beta at higher field and current, and very high heat flux divertor operation, for 5 second pulses contributing to the following important science topical areas:

- Extend Confinement scalings to higher power and plasma pressure, and lower collisionality, closer to next-step ST devices.
- Test Magneto-hydrodynamic stability at higher plasma pressure to explore the limits that could be encountered in future devices.
- Challenge the divertor in NSTX beyond the capabilities of any other device in the world (in terms of the scaling parameter, P/R).
- Test predictions that more tangential injection increases beam current drive efficiency and can sustain a broadened and more MHD stable current profile.
- Increased Alfvén eigenmode activity and fast-ion redistribution/loss could result from more tangential injection and increased fast-ion pressure.
- Test plasma ramp-up using beam injection and demonstrate fully non-inductive sustainment to simulate the scenarios planned for future ST devices.

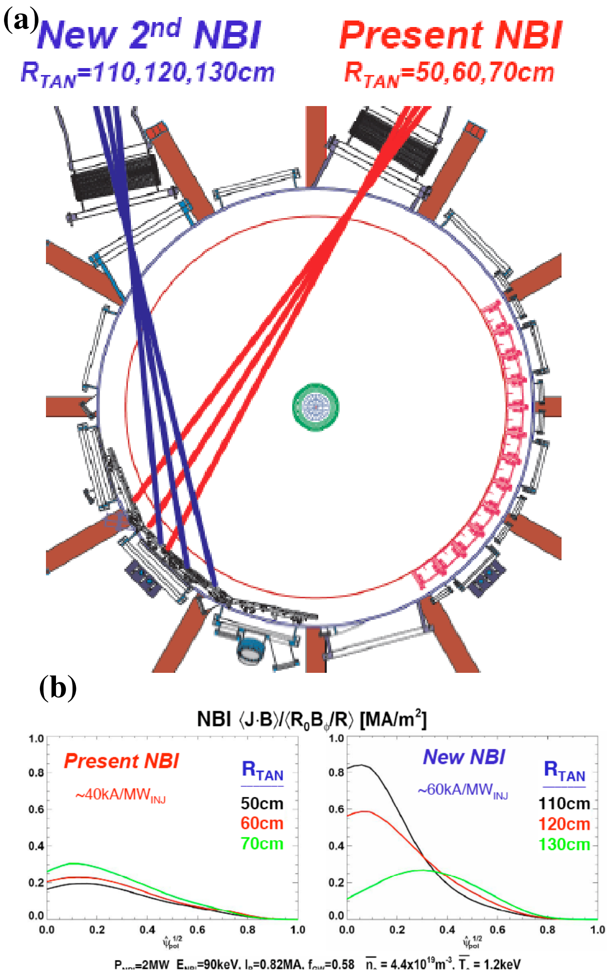


**Fig. 9. NSTX NBI injection geometry. (a) Top view of the NBI beam trajectories. (b) Respective current drive efficiencies.**

These upgrades will enable NSTX to explore fully non-inductive operation over a much expanded plasma parameter space in terms of higher plasma temperature and lower collisionality, thereby significantly reducing the physics parameter gap between the present NSTX and the projected next-step ST experiments.

## 6. Summary and Plans

NSTX has a very productive 2008 experimental campaign in the areas of HHFW, EBW and energetic particles reported here. Additional NSTX materials are presented by M. Bell on MHD, T&T and Boundary physics and CHI start-up by R. Raman in this workshop. In additions, extensive NSTX FY 08 results will be reported at the up-coming IAEA and APS meetings by the NSTX research team. The NSTX is preparing for the 2009 experimental campaign with new capabilities including the upgraded HHFW antennas for higher power operations, a full divertor bolometer system, and a surface analysis probe. The Liquid Lithium Divertor Target for enhanced pumping, 32 channel beam-emission spectroscopy to measure lower wave number plasma turbulence to complement the high-k scattering system, and MSE-LIF to complement MSE-CIF for radial electric field and total magnetic field measurements are being implemented in 2009-2010 time frame. As a part of the NSTX longer term program plan, a new center-stack to expand the accessible plasma parameters at lower collisionality and a second NBI system to demonstrate fully non-inductive operations are planned to support the design basis for next-step STs.

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