Report of the 2005 FESAC Facilities Panel, Vol. 2:

Requested Information Provided by the Three Major United States Toroidal Magnetic Fusion Facilities

Submitted to the Fusion Energy Sciences Advisory Committee

July 11, 2005

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On April 28, 2005, the FESAC Facilities Panel requested input from each of the three major U.S. toroidal magnetic fusion facilities. The request included an invitation to each facility program director to provide a document (30 pages maximum length) that addressed in detail the panel charge as provided in Appendix 2 of this report Volume 1. The three '30-page' documents that were received by the Facilities Panel in response to this request are here reproduced as background to report Volume 1.

The Facilities Panel wishes to express its sincere gratitude to the program directors of the three major facilities, and to their staffs, for the thoughtful and detailed input provided during the panel activity.

A. Executive Summary

The C-Mod program explores a broad range of scientific and technological issues of importance to fusion science, and is positioned to answer many of the critical questions on the path to ITER and ultimately to a fusion reactor. Research on C-Mod is guided by the facility's unique capabilities, access to exclusive parameter regimes, observations of unique or unusual phenomena, and the opportunity to provide high leverage contributions to coordinated, comparative research with other devices.

C-Mod is the only diverted tokamak in the world which exclusively employs high-Z metallic plasma facing components. ITER is faced with a critical materials choice for PFCs. Carbon can lead to unacceptable tritium retention. C-Mod studies the issues in a unique parameter range, overlapping those on ITER for divertor particle and power densities. Studies of hydrogenic retention, disruption survivability, particle control, power handling, wall conditioning and impurity dynamics, under these ITER prototypical conditions, can be carried out only on C-Mod.

C-Mod produces the highest pressure tokamak plasma in the world, close to those expected in ITER. The most promising method for disruption mitigation uses massive gas puffs. To understand the physics of the gas penetration, and to qualify the technique for application to ITER, experimental results at high absolute pressure will be critical. C-Mod is about to begin these experiments, and definitive results can be expected over the course of the next two to three years.

C-Mod has revolutionized divertor design with the invention and implementation of the vertical plate geometry. Adopted by the ITER design, the vertical plate geometry affords significant advantages in power handling. Geometry influences many other aspects of divertor behavior, including radiation trapping, neutral transport, impurity compression and detachment. Research into all of these is currently being aggressively pursued on C-Mod. An unanticipated consequence of changing the inner-divertor geometry in C-Mod was a decrease of disruption-induced halo currents, another significant issue for ITER which must be explored and understood.

C-Mod is currently the only tokamak in the world able to measure the effects of divertor radiation trapping. Trapping, principally of Ly-α radiation, strongly influences divertor detachment and can even prevent it in some cases. The ITER divertor design relies on detachment to handle the parallel power flux coming to the plates. C-Mod and JET are the only tokamaks capable of accessing the regime where trapping becomes important (C-Mod at high density, JET because of large size), but only C-Mod presently has the diagnostics needed for the measurements.

C-Mod leads the world in research on cross-field particle transport on open field lines in the Scrape-Off-Layer (SOL). The unexpected results from C-Mod show that strong turbulence-driven perpendicular particle transport can compete with parallel transport in the SOL, forcing ITER (and world) attention on this issue which can have important implications including risk of damage to first wall structures. State-of-the-art diagnostics on C-Mod are enabling detailed physics investigations of the intermittent turbulent structures that drive this transport.

Alone among high performance diverted tokamaks, C-Mod uses RF waves exclusively for all auxiliary heating and current drive. This decouples these drives from particle and momentum sources, precisely the expected situation in reactors. ICRF will be a main auxiliary heating technique on ITER, operating in the minority regime investigated on C-Mod. Lower Hybrid is likely the best tool for off-axis current drive and the C-Mod results will inform the ITER decision whether to proceed with a major LHCD upgrade for Advanced Tokamak operation.

C-Mod has pioneered the study of intrinsic plasma rotation in the absence of externally imposed torques. ICRF is everywhere torque-free (distinct from co+counter NBI, which is only balanced globally). The discovery of strong core plasma rotation at high pressure/pressure gradient has opened a new line of research, directly relevant to stability in ITER, on the effects of intrinsic flow, momentum transport, and the boundary conditions imposed by edge/SOL flows. This work has already produced intriguing results on the relation between divertor topology and the H-mode threshold, and must be better understood to evaluate flows in ITER.

C-Mod is the only divertor tokamak in the world which routinely operates with equilibrated electrons and ions. Large B/R naturally allows C-Mod to operate at relatively high plasma density, enforcing $\tau_{e-i} << \tau_E$. The vast majority of high performance discharges in the ITER H-Mode data base are neutral beam heated, with T_i substantially above T_e, which is strongly stabilizing for ion energy transport mechanisms, especially ITG turbulence. Enforcing temperature equality through equilibration, as must be the case in ITER and reactors, rather than through additional heating aimed at electrons, ensures that the equalization is uniform through the profiles, and makes C-Mod uniquely suited to studies of transport, burn control and stability.

C-Mod routinely operates with pulse-length exceeding the natural current decay time, and will have the unique capability to operate pulses for up to ten resistive skin times. With full implementation of LHCD tools in the coming few years, C-Mod will study fully non-inductive high performance AT plasma regimes, at the no-wall β limit, with completely relaxed current density profiles. This regime is exactly where the currently configured ITER could, with the addition of LHCD, operate.

Investigations on C-Mod are well coupled to international tokamak research, primarily through strong participation in the ITPA. C-Mod personnel are actively engaged on all of the ITPA committees. Currently twenty ITPA joint research topics include C-Mod participation. Additional research is carried out through bilateral international efforts.

C-Mod is addressing critical ITER R&D. Most of the areas of C-Mod emphasis utilize the unique facility capabilities. They include transport with equilibrated species, small/no ELM H-Mode pedestal regimes, hydrogenic species retention and wall conditioning with molybdenum and tungsten PFCs, disruption mitigation of high pressure plasma, rotation in torque-free plasmas, size and field scaling of error field effects and locked mode thresholds, RF stabilization of NTMs and elimination of the sawtooth seed, ICRF minority heating, mode-conversion current and flow drive, and AT physics toward steady-state with far off-axis LHCD. Targeted technology development for ITER includes RF antenna technology and tungsten high heat flux divertor components.

C-Mod is, in many respects, complementary to DIII-D and NSTX, as well as to all other tokamak facilities world-wide. Through coordinated joint experiments involving two or sometimes all three facilities (and often with additional international participation), it becomes possible to answer questions that could not be addressed by one facility alone. Important examples include effects of size, field, pressure, plasma density, power density, rotation, aspect ratio, and heating and current drive techniques. In many cases these experiments involve matching the dimensionless plasma parameters (ρ^*, ν^* and β) while varying dimensional parameters. These studies can be particularly important when non-plasma effects, such as neutrals, radiation, or neutral driven torques (NBI) are expected to play important roles; examples include disruption mitigation, and most pedestal and divertor physics. C-Mod accesses high leverage extremes for many of the dimensional parameters, not available on any other device.

C-Mod has mature infrastructure, including power supplies, control systems, data acquisition and computation facilities, and RF systems, which enable efficient, reliable research operation. Major recent upgrades include the addition of a lower hybrid current drive system, and a modified divertor which has enabled 2 MA, 8T operation as well as increased flexibility in shape. We are poised to exploit these hardware investments for increased plasma performance and for exploration of advanced scenarios.

C-Mod has state-of-the-art diagnostic systems for plasma measurements, and these continue to be upgraded. The core diagnostic set is strong and improving; the edge/pedestal/SOL diagnostic set is second to none in the world.

C-Mod has a highly experienced, highly productive, innovative research team, with an excellent research track record. One of the most important assets of the project is the strong integrated team of scientists, engineers and technicians, which includes many outside collaborators. The inclusion of graduate and undergraduate students enhances the vitality of the group, yielding a constant influx of fresh ideas. The experimental team is tightly coupled to theory and modeling, both with the theory team at MIT, and through broad national and international collaborations.

C-Mod is a premier facility for training graduate students, the next generation of fusion scientists. At any one time, typically 25 to 30 graduate students are performing their PhD research on C-Mod, far more than on any of the other major facilities. ITER is projected to start operating in about 10 years, and during the subsequent 10 to 20 years of research, the US will reap the benefits from ITER only through the participation of US scientists. Their training must begin now.

B. What are the unique and complementary characteristics of Alcator C-Mod?

B.1 Uniqueness of Alcator C-Mod (both nationally and internationally)

B.1.1 Introduction: Alcator C-Mod is the only high-field, high-density divertor tokamak in the world fusion program. While compact in physical dimensions (about 0.1 ITER scale, as shown in Figure 1), C-Mod produces plasmas which overlap, in dimensionless parameters and absolute performance, with those produced in much larger devices. C-Mod is able to achieve this in a very cost-effective manner because fusion performance scales as $\sim B^3a^2$, while costs scale as $\sim B^2a^3$. Auxiliary heating and current drive systems for C-Mod are exclusively RF (Ion-Cyclotron and Lower Hybrid), leading to ITER and reactor relevant decoupling of heating, fueling and momentum sources. Routine operation at high absolute density, enabled by large B/R, allows C-Mod to explore ITER and reactor-relevant regimes with fully equilibrated electrons and ions. Compactness also yields very high power densities and particle fluxes, expanding the available parameter space for scrape-off-layer and divertor physics and technology studies. C-Mod has always and exclusively used high Z metallic plasma facing components for all high heat flux regions, also unique among divertor tokamaks, and highly relevant to ITER. C-Mod produces the highest absolute pressure plasmas (at the ITER magnetic field and β), providing an extremely important opportunity to extend the study of disruption mitigation by massive gas puff, whose dynamics are expected to depend not only on dimensionless plasma parameters, but explicitly on absolute pressure, as well.

B.1.2 Burning Plasma Support Thrust: Alcator C-Mod is uniquely positioned to provide physics data and understanding of several critical issues facing tokamaks on the path to a viable fusion reactor. C-Mod is the only divertor tokamak in the world operating at (and above) the ITER design field of 5.3 T; the maximum C-Mod field of 8 T is prototypical of future advanced reactor designs. Therefore C-Mod is able not only to operate at the same non-dimensional plasma parameters as the burning plasma (except for gyro-size), but also at the same absolute plasma pressure and power density. The C-Mod integrated research program is therefore able to address many issues, such as plasma-wall interactions, disruption mitigation techniques, fueling, radiation transfer and opacity, and momentum transport via neutrals, which are not well-modeled by the equations of plasma physics alone, while also maintaining similarity with respect to the relevant plasma parameters. This ability, a consequence of C-Mod's high-field and compact design, is unmatched in the world program.

As a consequence of its high magnetic field and compact size, C-Mod operates with higher plasma density than larger, lower field devices. A direct result of this fact is that over most of its operating space, the product ν_{ei} x τ_E is large, such that electrons and ions are always closely coupled and equilibrated. Such close coupling, resulting in $T_e \sim T_i$, is characteristic of a reactor, or a burning plasma experiment such as ITER. While other tokamaks can achieve near-equality of temperatures by careful adjustment of the relative ion and electron heating, C-Mod achieves this condition naturally, as would a reactor. The T_e/T_i ratio is of fundamental importance to the transport physics, and C-Mod operates in the most relevant regime with respect to this parameter. Moreover, the fact that the temperature equality is imposed by equilibration rather than externally forced makes C-Mod uniquely well-suited to studies of burn control and stability in support of the burning plasma mission.

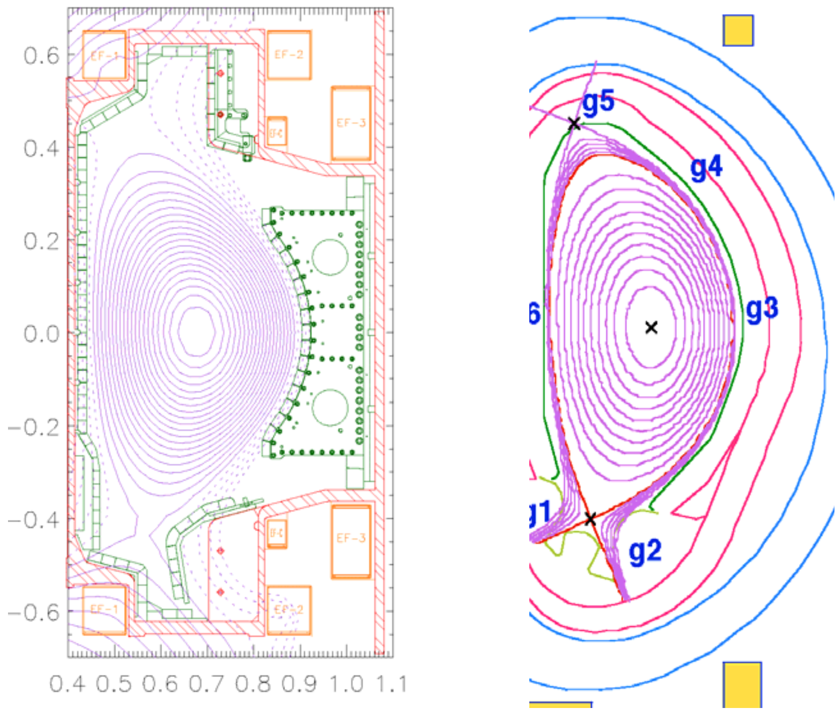


Figure 1 C-Mod equilibrium (left) can match precisely the shape of the ITER design (right, shown scaled by 1/9 in linear dimension). The parameters for this actual C-Mod discharge were B_T = 5.3 T, I_P = 1.6 MA. The corresponding parameters for ITER are B_T = 5.3 T, I_P = 15 MA.

C-Mod utilizes RF (ICRF and LH) exclusively as auxiliary heating and current drive sources. This choice separates the heating and current drive from the particle and momentum sources, and distinguishes C-Mod from almost all other tokamaks which use neutral beams as the principal heating source. In this sense, the heating in C-Mod is prototypical of the alpha particle heating which will dominate in a burning plasma. This decoupling of source terms facilitates design of unique transport experiments. In particular, the lack of momentum sources has allowed the discovery and investigation of intrinsic rotation, which has important implications for ITER, which will have little externally imposed core momentum input.

The high power and particle density of the C-Mod divertor and scrape-off layer exceed those achievable in all other tokamaks by factors of two to five and provide access to conditions projected for ITER. C-Mod studies of divertor detachment are performed at ITER-relevant parameters, providing key data on reduction of power flux to material surfaces and the effects of volume recombination for reducing particle fluxes. Both plasma and neutral densities in the C-Mod divertor are similar to those predicted for ITER, providing a unique capability to study issues of radiation transport and trapping, and ionization balance under divertor conditions (n_0L) approaching those of ITER. The similarity in C-Mod and ITER of normalized neutral mean free path (SOL transparency) is important for fueling, plasma-wall interactions, impurity screening and transport.

B.1.3 Advanced Tokamak Thrust: The Advanced Tokamak program on Alcator C-Mod is an integrating thrust which combines aspects of transport physics, wave-plasma interactions, stability and plasma boundary physics. It focuses on the development of scenarios which feature control of current as well as kinetic profiles, leading to largely non-inductive current drive with high bootstrap fractions. Typically, this requires reversed shear current profiles which can be achieved on C-Mod (and in ITER) by the simultaneous application of efficient off-axis LHCD, perhaps combined with modest central mode conversion or fast wave current drive, and intense ICRF heating to generate significant bootstrap current (~70%). Such scenarios could lead to more attractive fusion reactors, as shown in the Aries RS and Aries AT studies, and will be an important research topic on ITER. While advanced scenarios, as well as intermediate 'hybrid' scenarios, are being studied on several tokamaks worldwide, the physical parameters of C-Mod, combined with its set of control tools make the program unique in several respects, and particularly relevant for applicability to ITER.

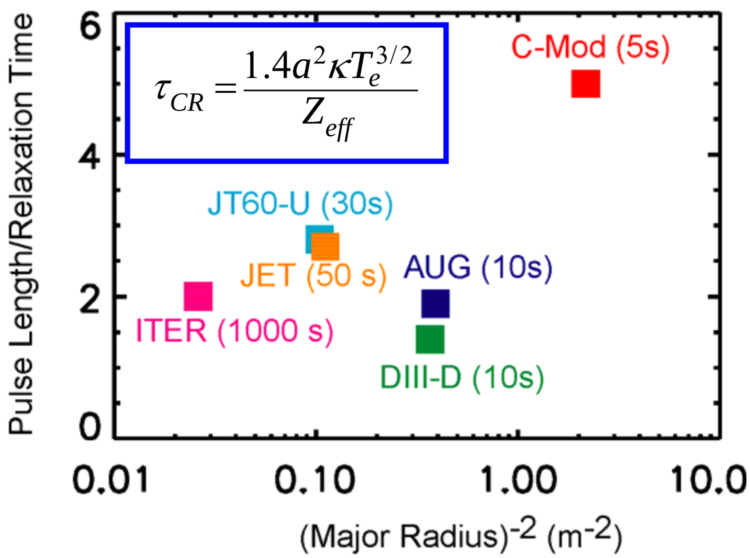


Figure 2 Pulse length, normalized to current relaxation time (τ_{CR}) for high performance divertor tokamaks. For the operating devices, Te = 6 kev and Z_{eff} = 2 are used to calculate τ_{CR}. For ITER, 19 keV is assumed. Even inductively-driven shots in C-Mod routinely have pulse lengths greater than τ_{CR}.

C-Mod operates without the external sources of particles and momentum which have been a key part of most "advanced" scenarios elsewhere. The tight coupling of electron and ion heat channels makes the formation and control of transport barriers, particularly in the core, more challenging: reducing transport in only one channel is not sufficient to guarantee the strong gradients required for large bootstrap current generation. Internal Transport Barriers (ITBs) have been produced in C-Mod plasmas using off-axis ICRF heating. These ITBs, with high pressures (up to 4 bar peak) and pressure gradients, are unique in that they are formed with no particle or momentum sources, have coupled ions and electrons, and can occur with normal magnetic shear. The barrier foot location can be controlled by changing the magnetic field and ICRF frequency, while the peak pressure (as well as impurity accumulation) can be controlled with additional on-axis ICRF heating power. In most other experiments, core particle sources from NBI, as well as NBI momentum input to modify the local velocity shear, are thought to play important roles in the formation of core barriers.

The current relaxation time (τ_{CR}) is short in C-Mod, due to its small minor radius ($\tau_{CR} \propto a^2$). C-Mod is perhaps the only divertor tokamak which *routinely* operates for pulse lengths exceeding the current relation time, which is the relevant time scale for maintaining "quasi-steady" current profiles. This situation prevails in C-Mod even at full heating power and with low Z_{eff}. Figure 2 shows a comparison of pulse length/ τ_{CR} for different tokamaks. Once significant fractions of the current are non-inductively driven on C-Mod, pulse lengths of up to 5 seconds will be possible.

C-Mod is unique in the U.S. program in its heating and current drive tools. In particular, it is the only experiment using lower hybrid current drive for control and sustainment of the current profile. This technique is the most efficient available, particularly for deposition far off-axis as required to optimize advanced scenarios. C-Mod is the only diverted tokamak in the world testing LHCD at the density and field range of ITER, which is key to the RF physics including accessibility and current drive efficiency. The other high-field experiment, FTU in Italy, is a limiter device which is not able to access edge transport barriers that pose a particular challenge due to their high edge density. Since many open questions remain about the applicability of LHCD in the regimes of interest, it has not yet been decided whether a LH system will be used on ITER. The experience on C-Mod will be essential to answer these key questions and timely results are needed to impact the ITER decision.

C-Mod features extremely high parallel heat flux in the SOL, equal to that expected in ITER. Power handling will be a particular challenge in long-pulse, advanced regimes since efficient current drive is favored by lower edge densities which may preclude radiative detachment. This may well prove one of the most difficult aspects of applying such scenarios on ITER or in a reactor. Near term experience on C-Mod, both with optimizing physical parameters and in testing advanced divertor materials, will be particularly important in establishing feasibility of long pulse advanced scenarios elsewhere.

B.1.4 Transport: C-Mod operates in a unique dimensional parameter range but can be run so as to overlap in dimensionless parameters with larger low-field tokamaks (Figure 3). This combination allows us to 1) dramatically extend the range and reduce covariance of absolute

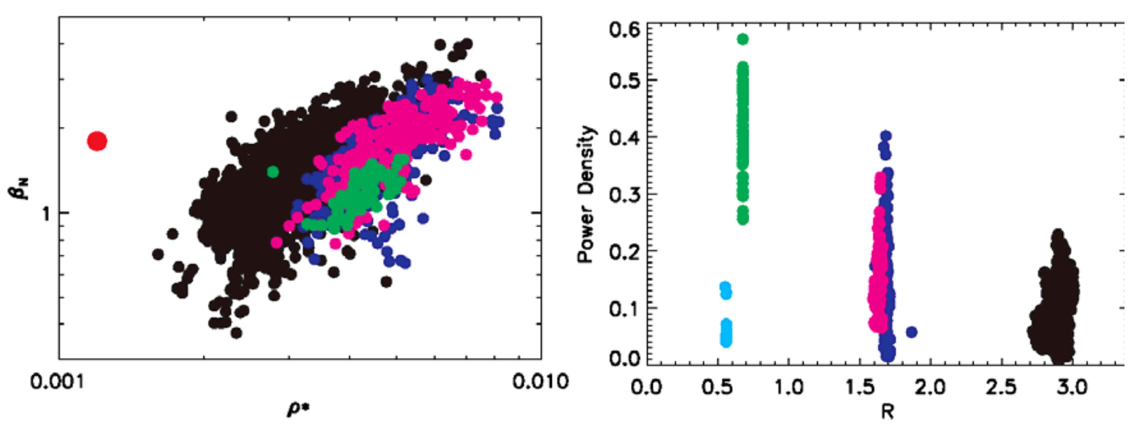


Figure 3 Data from the international multi-machine ITER H-Mode database. The plot on the left shows the range in two of the important dimensionless parameters, β_N and ρ^*. The plot on the right shows the range in the dimensional space power/surface area (MW/m^2) vs. major radius (m); ITER is designed for R=6.2m, P/A=.2 MW/m^2. The points are color keyed: C-Mod – green; ASDEX-U – magenta; DIII-D – blue; JET – black; Compass-D – cyan; ITER – red.

parameters in global scaling studies; 2) perform dedicated dimensionless identity experiments where all dimensionless parameters are held fixed between two or more machines; 3) perform dimensionless scaling experiments, where one dimensionless parameter is scanned over a wide range between two or more machines with other dimensionless parameters held fixed; and 4) extend physics studies to unique regions of dimensionless parameter space.

C-Mod provides a high-leverage laboratory for testing the standard scaling laws. For example, while C-Mod's L-mode performance matched the original L-mode scaling laws quite well, the confinement times in H-mode were about a factor of two above the then existing (1993) H-mode scaling laws. This difference was significant, about three standard deviations. C-Mod data improved the range of the H-mode database as well; unconstrained fits performed after the inclusion of C-Mod data matched the Kadomtsev constraints, which was not the case before inclusion.

Another example of a high-leverage C-Mod contribution is in testing the role of atomic physics in determining the pedestal structure and L to H-mode threshold. Atomic processes (mainly radiation and neutral penetration) will not match tokamak to tokamak under the conditions of plasma physics identity. Combining C-Mod data in joint dimensionless scaling experiments with ASDEX-Upgrade, DIII-D and JET allows for the widest possible range of dimensionless parameters.

In general, C-Mod generally runs at higher collisionality and lower β than low-field tokamaks, while operating at similar values of ρ^*. This characteristic is useful in validating particular aspects of transport physics. Theoretical interest in the role of collisionality arises in several ways. Zonal flow damping mechanisms are of great importance, since these flows are believed to be the principal saturation mechanism for ITG turbulence. The only linear damping mechanism for zonal flows involves ion-ion collisions, though nonlinear mechanisms via other turbulent processes have also been proposed. Experimental scans over a wide range in collisionality will be important in resolving this issue. As an example, consider predictions of the ion temperature profile based on ITG turbulence. Early physics models (IFS-PPPL, GLF23) had been tested successfully on several machines, but failed for C-Mod, where the ion temperature was significantly under-predicted. The problem was resolved when the full electron dynamics and the effects of collisions were included in a nonlinear model. This process showed that the earlier agreement was in some cases fortuitous. Collisionality can also be important for transport via TEM turbulence since these modes can be stabilized by collisions. In the edge regions, higher collisionality/resistivity is predicted to cause resistive ballooning modes to become unstable. The Quasi-Coherent (QC) mode in C-Mod, which is believed to be the source of particle transport in Enhanced D-Alpha (EDA) H-modes, has been tentatively identified as resistive ballooning.

It is worth noting that the neoclassical ν^* is not a unique formulation to define normalized collisionality. For turbulence, a more appropriate normalization factor might be the diamagnetic frequency. Depending on the k values of interest, this parameter will differ from ν^* by about one power of ρ^*. Another relevant collisionality can be defined by normalizing the collision frequency by the inverse energy confinement time. This parameter differs from ν^* by approximately two powers of ρ^*. C-Mod attains the reactor-relevant value of this collisionality and, like a reactor, has $\nu_{ei} < \tau_E$ and therefore has $T_i \sim T_e$. The equilibration of T_i and T_e has important implications for turbulent transport; the temperature ratio has a significant effect on the critical gradients predicted for ITG and ETG modes. Further, in the strongly coupled regime, the transport channels cannot be considered in isolation.

High fields allow C-Mod to run in enhanced confinement regimes with high power but with β well below MHD limits. The effect of β and the proximity to MHD limits is of considerable interest. A strong dependence of confinement on β is found in multi-machine scalings but not in dedicated dimensionless scans on individual tokamaks. Most theoretical studies predict a strong dependence only at very high β, perhaps only near the limit.

The combination of ICRF heating and LH current drive also offers unique opportunities for transport studies. ICRF, unlike neutral beams, decouples the heating profile from density. Further, it decouples heat, particle, momentum and current drive sources. ICRF is everywhere torque-free, unlike co+counter NBI which is only balanced globally. In this respect, experiments on C-Mod are more reactor-like than those in beam heated devices. LH current drive provides the ability to vary the current profile in a controlled manner, and investigate the effect of magnetic shear on turbulent transport.

C-Mod experiments unambiguously demonstrate that large self-generated flows arise in the core. The results have stimulated intense theoretical interest in attempting to explain the unexpected aspects of momentum transport. Experiments suggest that the SOL provides a crucial boundary condition for core flows and have led to a novel explanation for the effect of topology on the L-H threshold. The ability to run H-Mode plasmas with no core particle source and with no toroidal electric field will allow us to conduct a series of experiments probing turbulent particle transport. The role of turbulence in establishing the core density profile will be crucial for machines like ITER.

C-Mod has a strong diagnostic set for transport studies. Of particular note is an array of *high-resolution x-ray spectrometers* which allows measurement of velocity profiles in beam-free (torque-free) conditions. The *phase-contrast imaging system* (PCI) density turbulence diagnostic has a wide bandwidth and extremely large dynamic range. Recent modifications should allow the measurements of anisotropies in the k-spectrum of short wavelength fluctuations and may help resolve the current dispute over the existence of extended radial structures (streamers) in ETG turbulence. A novel set of *fast-scanning Langmuir-Mach probes* has been developed to study edge plasma profiles and fluctuations with high spatial resolution, including a unique probe on the high-field side scrape-off layer. These have revealed remarkably fast (near-sonic) plasma flows along field lines in the edge, connected to ballooning-like cross-field transport physics. Specially designed *fast-scanning probes with embedded poloidal magnetic field pick-up coils* have also been developed. These allow high poloidal wave-number magnetic fluctuations, such as that associated with the quasi-coherent mode seen near the separatrix in EDA H-mode discharges, to be studied in detail. C-mod uses a unique *fast camera*, capable of taking 300 movie frames at up to 250,000 frames/s, to yield high resolution 2D images of the turbulence structures and dynamics. Studies using the camera have determined the size-scale and wave-number spectra of the turbulence and have correlated the observed turbulence with the anomalously large radial particle transport in the scrape-off-layer

B.1.5 Macroscopic Stability: Alcator C-Mod's high magnetic field, high current density, high plasma pressure (energy density), and compact size allow us to address several important MHD research topics, including disruptions and dimensional scalings, while also alleviating many problematic issues related to very high-β operation, such as resistive wall modes. Disruptions on C-Mod are characterized by much higher halo currents and $J \times B$ forces than on other tokamaks, due to its unique high-field, high-current-density operational parameters as well as its all-metal first wall, and mitigation is correspondingly more difficult. The C-Mod vacuum vessel and inboard and outboard divertor modules are instrumented with extensive halo current and eddy current diagnostics, which not only measure total disruption-induced currents, but also provide data on toroidal asymmetry. This has resulted in C-Mod being a major

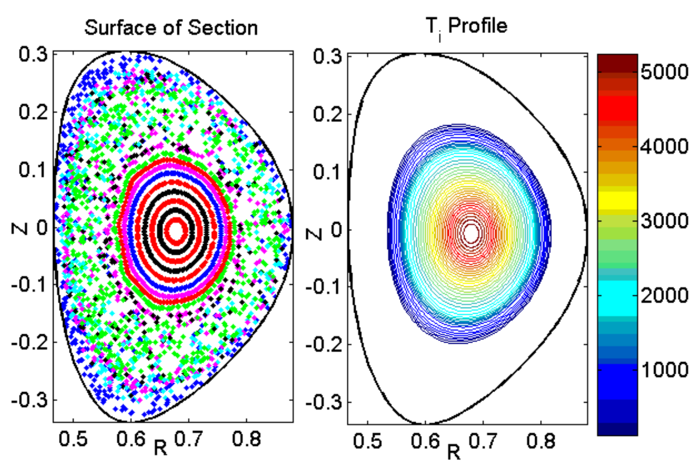


Figure 4 NIMROD 3-D MHD simulation of gas-puff disruption dynamics in C-Mod. Edge cooling from the gas injection is imposed as an initial condition. Instabilities grow rapidly resulting in the inward propagation of B-field stochasticity which is faster than the gas velocity. The plot on the left shows flux surface breakup 11 μs after the initial cooling. The plot on the right shows the dramatic cooling that results.

contributor to the ITPA disruption database. Also unique to C-Mod have been the studies of halo currents with two different inboard divertor configurations. Changing the shape of the C-Mod divertor has been shown to affect the magnitude of halo current flowing in the vessel structure, and thus the disruption *JxB* forces.

C-Mod's short disruption timescale (~1 ms), coupled with its ITER-like high plasma energy density and pressure, poses a unique challenge for proposed mitigation schemes, such as high-speed gas jet injection. Experiments to determine the feasibility and effectiveness of this technique are being carried out with a highly optimized gas jet system and fast imaging of the gas penetration. In parallel we are using NIMROD modeling of the three-dimensional MHD physics involved in the disruption quench and halo current generation (see Figure 4). These mitigation experiments are expected to provide the most stringent test available of whether or not such a scheme will be effective in ITER.

The active MHD antennas installed in Alcator C-Mod are also unique in the US program. They are used primarily for studying Alfvén modes and cascades, which can provide information on q-profile evolution, but also be applied to studying the stability properties of global MHD.

Non-axisymmetric field effects are an important research area on Alcator C-Mod. With its A-coils, C-Mod has resolved an important question in respect of mode locking for ITER. Prior scalings of the error field penetration threshold predicted in ITER a possibly extreme sensitivity to field errors. We find, in contradiction to the results of Compass, that compact machines do not intrinsically have less sensitivity to non-axisymmetric fields than e.g. DIII-D and JET. C-Mod's results, at ITER's value of toroidal field, safety factor, and Greenwald number, show that the extrapolation to ITER is much more favorable, and is tractable for correction. Future work in this area will address the field magnitude scaling, utilizing C-Mod's ability to operate over a very large range of fields. Moreover, the question of mode penetration has important application to active and wall stabilization of MHD instabilities. Therefore understanding the basic physics of these processes may have much wider significance beyond locking phenomena in ohmic discharges. C-Mod's capabilities for static non-axisymmetric field control are now excellent, and the installation of close-coupled internal coils is under consideration for the future.

B.1.6 Wave-Plasma Interaction: C-Mod exclusively uses RF power for auxiliary heating (ICRF – 8MW, 40-80 MHz) and current drive (LHRF – 3 MW, 4.6 GHz). IC-based mode conversion current drive (MCCD), ion cyclotron current drive (ICCD), and fast wave current drive (FWCD) are also within our capability for special applications. Our ICRF frequency range is appropriate for H, ^{3}He minority heating in D plasmas for the C-Mod magnetic field in the range of 3-8 Tesla. This reliance on RF for auxiliary heating and current drive has focused our research attention on solving the many physics and engineering challenges of reliable RF deployment on a tokamak. These challenges must be solved for ITER. ITER currently plans to use ICRF as one of the main auxiliary heating systems, particularly to trigger H-Modes. The U.S. may provide half of the ITER ICRF system and C-Mod is the only US facility to focus on minority heating physics and related technology issues. For Advanced Tokamak quasi-steady-state operation, ITER will probably require Lower Hybrid current drive system for substantial off-axis current drive, and C-Mod results should have a strong influence on this decision. C-Mod possesses a unique combination of flexible RF systems, sophisticated wave diagnostics, and first principle physics simulation models for validating the RF physics models and computational algorithms. Furthermore, accurate RF deposition and driven current simulation is vitally important for C-Mod and ITER for transport and stability analysis of experimental discharges.

For ICRF, C-Mod has stimulated and supported a major effort to develop a first principles RF simulation code (TORIC) for wave propagation, absorption and current drive and the capability to benchmark it against experiment. A novel diagnostic, phase contrast imaging (PCI), measures the RF waves, and high-resolution power deposition profiles measure the plasma response. The PCI measures the wave-number and spatial structure of the wave's density fluctuations simultaneously and this unique capability has allowed the first identification of the Ion Cyclotron mode converted wave. The combined PCI wave and high resolution power deposition profiles (see Figure 5) present the most stringent test for the RF simulation to date.

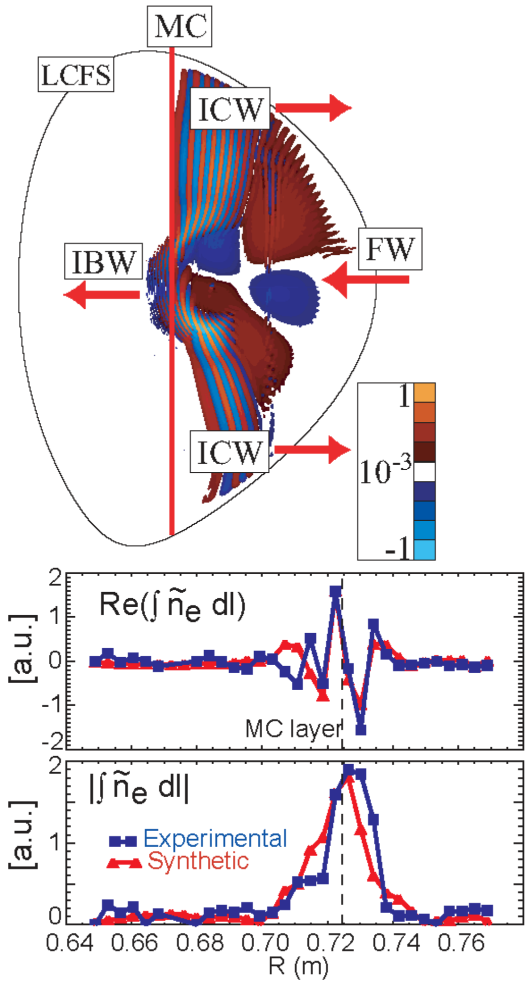


Figure 5 Comparison between TORIC simulations and experimental data measured with the Phase Contrast Interferometer (PCI) diagnostic. The top plot shows the contours of fluctuating electric field from the incoming fast wave, and the mode-converted (Bernstein and Ion-Cyclotron) waves in the core of the plasma. The code includes a synthetic simulation (red curve) of the PCI measurements (blue points).

With the application of high power (>4 MW) ICRF in C-Mod, an emerging issue is to self consistently account for the fast ion population, an important issue for ITER, that can perhaps modify ICRF absorption and play a significant role in modifying MHD stability (e.g. monster sawteeth, wave excitation). With MHD spectroscopy (Alfvén wave spectrum) and compact neutral particle analyzer measurements, the validation of the physics kernel and computational methods used in an advanced finite banana-width Fokker-Planck code with self consistent RF fields can be pursued.

An outstanding issue for ICRF is predicting antenna performance. Through an iterative process, we have developed a compact 4-strap ICRF antenna that has obtained the highest power density of any 4-strap antenna and worked to minimize and understand impurity production with metallic and insulating limiters. Compared with other ICRF experiments, the C-Mod plasma loading is high and the loading variations are a result of plasma density profile effects, rather than variations in wave evanescent length. In ITER, the loading and loading variations will likely be a combination of changes in evanescent length and density profile effects. Predictive modeling of antenna loading and performance is limited and we have begun a collaborative effort with U. Torino and RF SciDAC to experimentally benchmark a 3-D antenna code (TOPICA coupled with TORIC) in an effort to develop a code capable of simulating an ICRF antenna in a tokamak plasma.

For radiofrequency current drive, mode conversion current drive (MCCD) can be a valuable adjunct to LHCD because of its predicted localized nature. In initial experiments, sawtooth period variation was shown to be dependent on antenna phase and deposition location, suggesting a means to control the sawtooth period. We plan to couple TORIC to a Fokker Planck solver allowing proper wave characteristics in the calculation of driven current. Finally, FWCD is most effective in driving current near the plasma center, and we will investigate its effectiveness near, or below the majority ion cyclotron frequency as is expected to be used in ITER.

Another more speculative mode conversion application, with the important potential to trigger and control transport barriers, involves RF driven flows. Although theoretical calculations have proven to be difficult, experiments may provide insight into which of the many forces are important. For example flows can be driven by ponderomotive forces or Reynolds stress. In the former case, damping on electrons is dominant and damping on electrons will be small in the latter. With present diagnostics, the poloidal rotation, RF power deposition, and RF density fluctuation profiles can be simultaneously measured and the magnitude of the poloidal flow, its profile, and its relation to the RF power, its absorption and wave propagation characteristics. Depending on its success, flow shear can be investigated to determine RF powers for triggering and maintaining internal transport barriers.

A new LHCD experiment is underway to provide non-inductive current drive for current profile control and extended discharge duration. Pressure profile control will be provided by the ICRF heating. For off-axis current drive, the most efficient RF technique employs LH waves to accelerate passing electrons and is a primary candidate for off-axis current drive on ITER. The Alcator C-Mod LH experiment is the only one in the world that operates at the field, density and shape of ITER, and the modeled current profiles are similar to those envisioned for ITER's weak negative shear regime. An additional unique feature of the LH experiment is the ability to control dynamically the $n_{||}$ spectrum with ~1 ms response time. The latter determines the spatial deposition of the LH power and driven current and this capability will be exploited to effect dynamic control of the plasma current profile. Initial LH coupling experiments at low power (<300 kW) have shown good coupling efficiency with reflected power ratio as low as 10%.

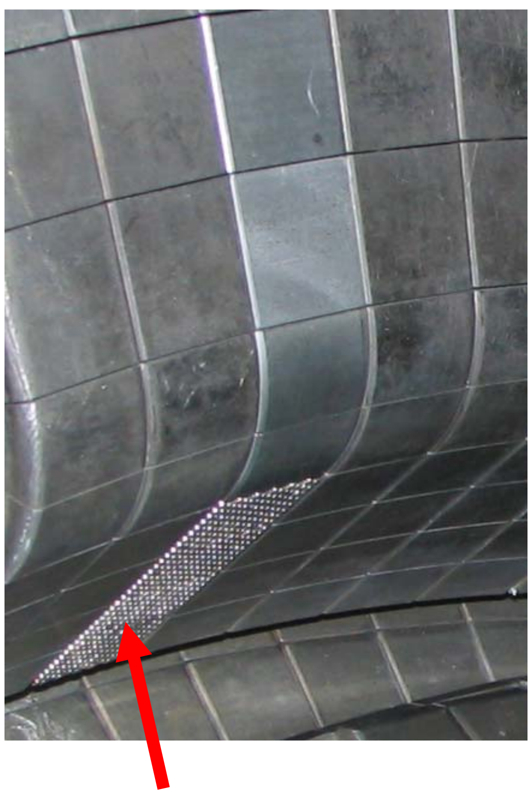


Figure 6 Tungsten brush tiles have been installed in selected regions of the divertor since the beginning of plasma operations in 2005. So far, they have performed extremely well.

This experiment also occupies the same velocity parameter space as ITER, where the quasi-linear plateau (distorted part of the parallel electron distribution function deformed by the LH spectrum) is relatively narrow. In this unique parameter space, an accurate accounting of 2-D velocity space effects must be included in the RF simulation. Further new full-wave electric field simulations elucidate the importance of focusing and diffraction on LH wave propagation and damping not easily included in the geometrical optics framework. The results suggest that the spectral gap is bridged by the $k_{||}$ upshift at caustics and reflections. To validate this model, a detailed comparison with experiments, including spatial profiles and energy distributions of fast electrons is required and will be a principal focus of our experiments.

B.1.7 Plasma Boundary: The SOL/divertor program takes advantage of two unique and important attributes of the Alcator C-Mod facility: (1) high-Z plasma-facing components (presently molybdenum) and (2) high power and particle density plasmas, prototypical of reactor regimes. The former provides a means for expanding the knowledge base of tokamak operation with high-Z plasma facing components (PFCs); these are recognized as the most relevant for power-producing reactor designs. More immediately, it is essential to understand the operational differences between high and low-Z PFCs (i.e. carbon) in order to provide guidance for ITER. Of particular concern is the allowed tritium inventory, which may necessitate an early change (already being considered) to high-Z PFCs for all internal components. Because it already has all high Z PFCs, Alcator C-Mod is uniquely positioned to provide key information on issues to contribute to this decision: the resultant drop in overall tritium retention; viable methods for tritium recovery; metal-wall conditioning techniques including boronization; and the risks associated with tungsten surface melting, including core cleanliness and operational continuity.

In an effort to provide specific operational experience directly applicable to ITER, tungsten test tiles are already in use in C-Mod (see Figure 6), and near term plans call for converting the outer divertor target plates to tungsten. The design of the tungsten tiles is being developed to mimic the geometries envisioned for ITER; castellated or lamellae-like structures.

The vertical-plate divertor geometry in C-Mod, a concept developed at MIT and now planned for ITER, is unique within the US program. This has been an important tool for understanding the effect of divertor geometry on plasma detachment phenomena. It continues to be essential for understanding how geometry impacts other aspects of the divertor (e.g. radiation trapping, neutral transport, and impurity compression).

The high power and particle densities produced in C-Mod provide unique experimental access to plasma physics regimes that are prototypical of reactor operation. Densities, temperatures, and parallel power fluxes in the divertor match those projected for ITER (1-2x10^{21} m^{-3}, 1-20 eV, ~500 MW/m^2), all of which are two to five times higher than those presently available in other tokamaks, and C-Mod routinely operates with $Z_{eff} < 1.8$. This has led to important insights into transport processes in the SOL and divertor plasmas. A notable example is in the physics of divertor detachment at ITER-relevant parameters, exploring the efficacies of radiation for reducing power flows and volume recombination for reducing particle fluxes. The dual combination of high power flux and high-Z PFCs necessitates the use of extrinsic impurities as radiating species rather than relying on intrinsic C from the PFCs. Continued experience in this area, particularly in regimes with ITER-like parallel power fluxes, provides information essential for assessing the prospects for detachment in ITER, required to maximize divertor lifetime.

The high plasma densities in C-Mod result in record-high neutral densities in the divertor, similar to those predicted for ITER. Exclusive C-Mod access to this diffusive regime provides additional opportunities for important science contributions. For example, state-of-the-art neutral transport codes have had difficulty in reproducing the observed neutral pressures in the C-Mod divertor. These studies have provided valuable insights into the underlying physical processes (e.g. the role of neutral-neutral collisions and viscosity) as well as the geometric details that must be properly modeled in ITER-like regimes. The unique mixture of high neutral densities plus plasma recombination physics also provide the opportunity to benchmark models of the radiation trapping in the divertor (dependent on n_0L, where L is the divertor size), which strongly affects ionization balance and thus the capability to reach detachment.

C-Mod's scrape-off layer (SOL) also has uniquely high density plasmas. This has enabled a number of key insights into SOL transport physics – an active area of present research. Since nR ~ B, the SOL of lower field tokamaks is more transparent to neutral penetration. However, since ITER will operate at magnetic fields similar to C-Mod, it will operate with similar *nR*, and therefore transparency to neutrals should be essentially the same as in C-Mod. As has been shown recently, this physics result has important implications for fueling, plasma-wall interactions (impurity sources and wall lifetime), and transport. C-Mod provides unique and essential access to this regime. C-Mod SOL characteristics will allow it to continue to make important contributions to the scaling studies of the pedestal height and width, and the role of atomic physics and transport in determining these parameters.

B.1.8 Alcator C-Mod Facility Capabilities: The C-Mod facility, including ancillary power and RF systems, diagnostics, computer hardware and software, buildings and other systems, has an estimated current replacement value of 180 million dollars. Table 1 lists some of the most important C-Mod engineering parameters. The C-Mod facility consists of the tokamak, switch gear and power converters to supply the magnets, ICRF transmitters and transmission lines, lower hybrid transmitters and waveguides, data acquisition and control systems, and a large and ever improving set of plasma diagnostics. The tokamak is located in a 2500 ft^2 experimental cell which abuts the 7,700 ft^2 power room that houses the magnet power supplies and ICRF transmitters. There is an additional 18,000 ft^2 of space dedicated to the project, including the control room, diagnostic labs, and shop space.

The C-Mod facility uses MIT's alternator, which together with a 72 ton flywheel, is capable of supplying approximately 400 MW of power to the magnets from a stored energy supply of 2 GJ. A total of 24 MVA of power is available from Cambridge Electric via a 13.8 kV line. Much of this power is used to power directly the RF systems, some magnet power supplies, and the new long pulse diagnostic neutral beam. Approximately 2 MVA of this is used to drive the alternator between shots. Four RF transmitters provide up to 8 MW of tunable (40 to 80 MHz) source power which is used to drive three ICRF antennas. These transmitters are nearly identical to the expected ITER system and a primary issue for these transmitters, as with most transmitters, is the final tube life. Recently, modifications to operations procedures have allowed us to double the tube life and an investigation into further extension is ongoing. Lower hybrid power at 4.6 GHz is supplied by twelve 250 kW CW klystrons reused from the Alcator C LH experiment. These tubes are powered by a 208 A, 50 kV, IGBT controlled power supply that is also capable of supplying power for an additional 4 klystrons, planned for future installation. Together with the power supplies, crowbars, and control instrumentation, this system represents one of the world's major RF facilities. C-Mod is unusual among major facilities in relying entirely on RF for heating and current drive, and considerable effort has been devoted to making these systems highly reliable.

The Plasma Science and Fusion Center has over 700 network attached devices interconnected by a Gbit Ethernet backbone and connected to ESnet via a 45 Mbps (T3) link. The C-Mod experimental project is supported by a large data acquisition system, 65 workstations, 150 personal computers and over 6 TB of raided disks for data storage. Current data acquisition hardware consisting of both CPCI and CAMAC allow over 2 Gbytes of data per shot to be archived. The project has dedicated staff for system support and software development.

Plans for the next two years include installation of an upper divertor cryopump, installation of a new 4-strap ICRF antenna, advanced divertor prototype installations, addition of a second lower hybrid launcher and an increase in lower hybrid source power from 3 to 4 MW. Installation of new diagnostics, including an FIR Faraday rotation system to measure current density profile, is also planned.

Table 1 Alcator C-Mod Facility Parameters

Physical Dimensions, plasma shaping	R=.68m, a=.22m, κ<1.9, δ<.85
Max Toroidal Field	8.1 tesla

Max Plasma Current	2.01 MA
Max Discharge length	5 seconds, $>>\tau_{CR}$
Ohmic heating power	up to 2.7 MW
ICRF source power	8 MW, 50 to 80 MHZ
Lower hybrid source power	3 MW, 4.6 GHz
Alternator peak power and stored energy	400 MW, 2 GJ
Vessel/Plasma volume	4/1 m^3

B.2 Complementarity with respect to DIII-D and NSTX

Table 2 summarizes a number of factors that address the complementarity of the US national fusion facilities. These complementarities enable the program to undertake both a wider range of experiments, and also critical tests of theory and understanding by coordinated experiments.

C-Mod and DIII-D are tokamaks having conventional **aspect ratios**, with C-Mod more compact in size by approximately a factor of 3. NSTX is a spherical torus, with aspect ratio about 1.3, and minor radius the same as DIII-D. All three have strong shaping capabilities in both elongation (shown) and triangularity, which are critical factors for macroscopic stability, and probably also for confinement and ELM suppression.

C-Mod's high **magnetic field capability** implies high magnetic pressure, and correspondingly the ability to operate at high **plasma pressure**. These differences are reflected also in the plasma beta, which is highest for NSTX and lowest for C-Mod (though still substantial: β_N=1.8, demonstrated). NSTX and DIII-D tend to run near the beta limit in high-performance regimes, with C-Mod rather further from ultimate limits in conventional H-mode scenarios.

Table 2 Complementary Aspects of the National Facilities.

Parameter	***C-Mod***	***DIII-D***	***NSTX***
Size: R/a (m), elongation	0.67/0.21, to 1.9	1.67/0.67, to 2.3	0.85/0.67, to 2.6
Magnetic Pressure (bar)	250	16	1
<Plasma Pressure> (bar)	2	1	0.3
Heating Schemes	ICRF (8MW) LH (3MW)	NBI (20 MW) ECH (6 MW)	NBI (7MW) HHFW (6MW)
Current/Flow Drive	LH, ICRF(MC)	NBI,ECCD,(FW)	NBI,(CHI,EBW)
P/S (MW/m^2)	0.7	0.2	0.2
First Wall Material	Molybdenum	Carbon	Carbon
Pulse length, $\tau_{L/R}$ (s)	3, 2	5, 20	0.6, 20
Typical $1/\rho^*$	220	180	70
ν^* range	0.06 – 1	0.007 – 0.2	0.1 – 2

Parameter	***C-Mod***	***DIII-D***	***NSTX***
Wall Stabilization	Distant	Close	Close

The facilities accordingly have different **programmatic emphases**. C-Mod has to date placed lower priority on β limit investigations than DIII-D and NSTX. However, modeling shows that by optimizing shaping and profiles, in its near-term AT program, C-Mod can achieve β_N up to 3. That thrust will focus on demonstrating such scenarios, with high densities ($>10^{20}/m^3$) and strongly coupled electrons and ions, and without externally driven rotation, conditions which are prototypical of those envisioned on ITER using its baseline hardware capabilities. Active or passive stabilization techniques to enable even higher β_N are being considered in the longer term, depending on results on C-Mod, DIII-D and elsewhere. In contrast, for example, the future NSTX program will explore non-inductive operation at very high β_T (< 40%) and low density ($\sim 3x10^{19}/m^3$).

C-Mod can operate with the same non-dimensional plasma parameters (β, ν^*, ρ^*, q) as DIII-D, and with the exception of aspect ratio, NSTX, but at very different absolute parameters (size, field, density). This ability is exploited in numerous "non-dimensional identity" experiments which explore key issues in transport, MHD stability, etc. on the one hand, and identify processes which are not fully described by plasma physics considerations on the other.

An example of the sort of fundamental plasma science topic that this diversity enables is a planned experiment to test neoclassical tearing mode (NTM) limits. The limit is generally expressed as a critical β_N, whose value depends on parameters such as safety factor, shaping, collisionality, etc. The theory of NTMs still has significant uncertainty in these dependencies. Therefore a comparison of NTM threshold between C-Mod and DIII-D is being planned which will serve to explore, (1) the extent to which the physics indeed follows dimensionless similarity between machines of substantially different dimensional parameters, and (2) what are the threshold dependencies on the other dimensionless parameters such as collisionality and gyro-size.

The **heating and current drive tools** deployed on the facilities are strongly complementary. Both DIII-D and NSTX rely upon strong NBI as their workhorse heating scheme, which also provides significant momentum and particle sources in the plasma interior. C-Mod relies entirely on RF heating. The wave schemes used on C-Mod provide zero particle and insignificant momentum sources, as is likely to be the situation on ITER. This distinction means that in C-Mod rotation is unambiguously self-generated while in DIII-D and NSTX, it is strongly driven. The resulting levels of rotation are quite different; in C-Mod the Mach number is up to 0.3. It can approach 1 in DIII-D and comfortably exceed 1 in NSTX, where rotation velocities are on the order of the Alfvén velocity. One result is that we can compare transport and transport barriers with different levels of ExB stabilization. On C-Mod, decoupling the heating profile from the plasma density source complements beam-heated experiments on DIII-D and NSTX where they are linked. So for example, when in C-Mod a peaked density arises in ITB plasmas, it is certain that this is a transport, not a fueling effect.

The types of RF heating are in themselves very different, in part because of the machine differences and in part by programmatic choice. In the ICRF range, C-Mod uses primarily minority heating and mode conversion heating and current drive schemes. These are the ICRF types most critical for use on ITER. DIII-D explores the fast-wave regime, where the wave has more than twice the majority ion cyclotron frequency. NSTX uses high harmonic ($\sim 10\ \omega_{ci}$) fast wave schemes, necessitated by its parameter regime. The launching antennas and related RF technology for these three schemes, which are a very important part of the overall experimental investigations, have similarities and differences. C-Mod ICRF antennas utilize compact structures and metallic (and for comparison insulating) limiters. DIII-D and NSTX have utilized antennas with graphite limiters. The C-Mod antennas operate at the highest power densities, because of the machine's compact dimensions. Areas where each group contributes significantly are power and voltage handling, plasma RF-edge interactions, and parasitic absorption. The phenomena are numerous and different processes can be dominant for different regimes. For example in DIII-D, an ad hoc 3% absorption per pass for fast wave heating has been thought to be associated with far field sheath effects. In NSTX, significant edge ion absorption is associated with parametric decay; in C-Mod, parametric decay has been observed but is energetically negligible.

For their second heating sources, also vital for current drive, C-Mod uses Lower Hybrid (LH) waves, an option for ITER, DIII-D uses ECH, which is in the ITER design, and NSTX is developing the more speculative Electron Bernstein Wave scheme. These are of course totally different from one another.

For current drive, both DIII-D and NSTX derive substantial current drive from their neutral beams. This mechanism is rarely used for current profile control because of the difficulty of drive localization. For current profile control, DIII-D uses ECCD. It has local absorption, which makes it excellent for precision localization. However, its current drive efficiency is substantially less than that of C-Mod's choice: LHCD. The LHCD can be localized in radius, notably farther off-axis (r/a~0.8) than ECCD, by parallel wavelength selection. This makes it ideal for generating the reversed- and low-magnetic-shear profiles that are characteristic of Advanced Tokamak operation. Both ECCD and LHCD have important application also to the stabilization of NTMs. For this purpose, ECCD experiments are somewhat more advanced, but again the issue of current drive efficiency, as well as localization, is crucial, which is one reason why ITER retains LHCD as an option. NSTX, in contrast, is exploring the use of the ST-specific helicity injection scheme for current drive as well as high harmonic fast waves.

Current drive is particularly important on C-Mod because it otherwise tends to run with an equilibrated inductive current profile, q_0 near 1 and sawteeth in most regimes. This characteristic can be considered to arise from the transport regime in which it operates; but a simple figure of merit in this regard may be taken as the ratio of the pulse duration to the current relaxation time. In C-Mod the magnets permit the pulse duration to exceed $\tau_{L/R}$. Thus C-Mod Advanced Tokamak operation will be completely unambiguous in establishing the control of the current by the RF, without synergistic inductive current relaxation effects. NSTX and DIII-D in contrast tend to run with non-equilibrium current profiles, $q_0 > 1$ and no sawteeth, which they have used to different advantage in producing AT scenarios by controlled current and shape transients.

The characteristics of the C-Mod **SOL and divertor plasmas** dovetail well with the capabilities and research at other US tokamaks, providing important tests of physics models. C-Mod has the highest **power density** in the SOL, because of its compact overall size, and because the SOL thickness is reduced by the lower transport coefficients caused by high magnetic field. Where C-Mod is exploring the operational characteristics of high-Z **plasma facing components**, DIII-D and NSTX are concentrating on low-Z (carbon). Where C-Mod's plasma characteristics lead to high neutral opacity of the SOL and divertor, the other US facilities provide tests in a complementary regime -- more transparent to neutrals. C-Mod's distinct divertor geometry allows experiments within the US to explore its effect on operation and divertor physics.

The research emphasis of C-Mod in the divertor and edge plasma region also complements the other US tokamaks. C-Mod brought to world attention, and continues to pioneer, techniques for understanding radial transport in the SOL and the resultant effects on PFC surfaces outside the divertor. It shares its SOL imaging capability with NSTX, as well as several other SOL diagnostics. C-Mod has gained great insights recently from a new inboard reciprocating probe; DIII-D has an excellent divertor Thomson scattering diagnostic.

For **transport studies** many of the important complementarities are described by **dimensionless variables**. Among the machines, C-Mod and NSTX run at intermediate neoclassical collisionality while DIII-D typically operates at very low levels. On the other hand C-Mod and DIII-D operate at similar, intermediate values of ρ^* while NSTX values are larger. C-Mod's higher density and compact size leads to strongly coupled ions and electrons in most regimes, typical of ITER, whereas the species are weakly coupled in most regimes for the other devices.

One of the most important issues for a burning plasma experiment such as ITER is control of the pedestal height in H-mode and the relaxation mechanism for the pedestal. Type I ELMs, which are the dominant relaxation mechanism observed in most lower field tokamaks, are considered problematic for ITER because of divertor erosion and survivability. Even at very high power densities and relatively low collisionalities, C-Mod does not typically experience type I ELMs, in contrast to DIII-D which most often runs H-modes in the type I ELMy regime. Alcator C-Mod has documented a more benign operational regime, the EDA H-mode, which is characterized by the QC mode which serves to remove particles and impurities and regulate the pedestal pressure without the large transients associated with giant ELMs. DIII-D has, in low density discharges with counter NBI, observed a similarly benign relaxation mechanism, but with different instability signatures referred to as the edge harmonic oscillation (EHO). NSTX observes yet another small ELM regime, referred to as type V ELMs. The regimes and conditions under which these different phenomena occur, and their potential applicability to ITER, are topics of intense investigation involving coordinated experiments among the facilities (and other tokamaks around the world).

A topical area of notable complementarity is in the interaction of the plasma with external **non-axisymmetric** currents. Here DIII-D and NSTX have a compelling interest in wall stabilization at high β, the associated resistive wall modes, and their suppression and control. C-Mod has focused on locked modes, installing asymmetric magnetic field control coils for their suppression and exploring the scaling of their thresholds in cooperation with JET and DIII-D.

Another important complementarity exists in the study of **energetic particle modes** such as Alfvén eigenmodes. In C-Mod these arise from ICRF minority tails, which have higher perpendicular energy, while in DIII-D and NSTX predominantly NBI ions, with parallel or more isotropic velocity distributions. NSTX's very low field means that the ions are readily super-Alfvénic in velocity.

C. How do the characteristics of each of the three U.S. fusion facilities make the U.S. toroidal research program unique as a whole in the international program?

C.1 Coordinated Experiments with DIII-D and NSTX: A substantial strength of the US national fusion research program is the close interaction among the three major toroidal research programs. Cooperation, collaboration, and coordination take place on several levels, from programmatic to individual experiments. Each facility provides a unique set of accessible plasma parameters, configurations, heat, current and particle sources and control methods, diagnostics, etc. C-Mod is actively involved in collaborative experiments with both DIII-D and NSTX.

Coordinated experiments between C-Mod and NSTX are planned to investigate the relationship between the small ELM regimes observed on these two devices. On C-Mod, steady-state H-modes without large ELMs are observed in the so-called EDA regime. In this case, a broadband QC mode is observed which apparently provides a regulation of the pedestal and reduces particle confinement without seriously degrading energy confinement. At higher pedestal pressures and lower collisionality, a regime of very small irregular ELMs predominates. NSTX observes a regime in which small "Type V" ELMs serve a similar function. The coordinated experiments will assess the parameter ranges (in non-dimensional terms) in which each phenomenon occurs and attempt to determine the underlying physics. In conjunction with similar coordinated experiments carried out between C-Mod and JFT2-M (aspect ratio~5), the dependence of small ELM relaxation mechanisms on aspect ratio will be addressed.

C-Mod and NSTX are also actively involved in studies of edge turbulence and transport involving visualization using fast framing cameras. These experiments actually share the same camera hardware, prototype units developed under SBIR funding from DoE. The cameras are transported between Princeton and MIT according to the requirements of the experimental schedule of the two facilities, and personnel from each lab are involved in the experiments. The turbulence imaging includes studies of propagation of "blobs" near the separatrix and through the SOL, as well as studies of pedestal relaxation phenomena.

C-Mod and DIII-D each have observed examples of strong core rotation in discharges with no applied external torque. In C-Mod this rotation is observed in both Ohmic and ICRF-heated discharges, while in DIII-D rotation is observed in Ohmic H-modes and with electron cyclotron heating (ECH). The origin of this "spontaneous rotation" is unknown. A burning plasma experiment such as ITER is not expected to have significant external momentum sources, so predicting the rotation properties of such a device depends on understanding the origin of the intrinsic "spontaneous" rotation. Projecting the rotation is important both because it influences the transport properties and also because of the impact of rotation on stability, particularly in advanced scenarios which might be subject to resistive wall modes. A series of coordinated experiments on C-Mod and DIII-D are planned to address the physics of the observed rotation. These experiments would draw on the ability of these devices to operate at identical non-dimensional plasma physics parameters to try to establish whether non-plasma effects, such as charge exchange momentum loss, may be playing a role in the phenomena. Parameter scans will then be carried out to identify the scalings in the two devices. Additional experiments will address the influence of the auxiliary heating technique and power deposition profiles.

C-Mod and DIII-D have an ongoing collaboration in SOL physics aimed at understanding radial particle transport. This has involved coordinated experiments across these two machines where dimensionlessly similar SOL parameter scans are run. It was found that, for L-mode discharges, the radial transport across the SOL was essentially the same (as measured by an effective convective velocity or diffusion coefficient. The collaborative work has expanded now to H-mode plasmas where the intention is to infer the radial particle transport coefficients in the SOL as a function of time during ELMs.

D. How well do we cooperate with the international community in coordinating research on our major facilities and how have we exploited the special features of U.S. facilities in contributing to international fusion research, in general, and to the ITER design specifically?

C-Mod research is well-integrated with international fusion research, both in direct support of ITER and in the continuing overall scientific program in the areas of transport, divertor and SOL physics, plasma-wall interactions, wave-particle physics, and MHD stability. Coordinated research is carried out through collaborative experiments, contributions to international databases, and participation in task forces. C-Mod's unique parameter range and capabilities provide an invaluable resource for the international community in addressing key scientific issues, as well as providing a test-bed for technological development.

C-Mod is an active participant in all of the ITPA Committees (and the subsidiary ITPA/IEA process), which constitute the principal forum for coordination of international collaboration and joint experiments in support of ITER. C-Mod scientists are active members of all of the ITPA Groups, and one, the Divertor/SOL group, is co-chaired by a C-Mod staff member. In addition, C-Mod participates strongly in the U.S. and European Transport Task Forces, and carries out international collaborations under the auspices of the IEA agreements on Large Tokamak Facilities and Tokamaks with Poloidal Field Divertors. Collaborations with other international institutions are also carried out on a bilateral basis.

C-Mod has made significant contributions to the H-mode threshold, L- and H-mode confinement, profile, pedestal and disruption databases. The unique characteristics of C-Mod proved critical for extending the range and conditioning of these databases.

In 2005, C-Mod is engaged in the planning or execution of at least 20 ITPA Joint Experiments, in essentially all of the ITPA topical areas. Many of the ITPA Joint Experiments take advantage of the high leverage provided by C-Mod's compact size and high field. In particular, experiments involving both C-Mod and JET provide the largest possible range of physical dimension among existing tokamaks, while allowing overlap in non-dimensional plasma parameters. Participation by C-Mod in this class of experiments is of high value in benchmarking models to be used for predicting the physics of ITER.

An example of such research is the ITPA PEP-7 experiment, a non-dimensional pedestal identity study among C-Mod, JET, ASDEX-U and DIII-D. These experiments aim at establishing the physics which determines the scaling of the height and width of the H-mode pedestal, which has dramatic implications for transport and performance of a burning plasma. An example of one of the results from comparisons with DIII-D is shown in Figure 7. A key aspect of these experiments is to distinguish between the role of atomic physics, through the ionization mean-free-path, and plasma physics effects in setting the scale of the width of the density pedestal.

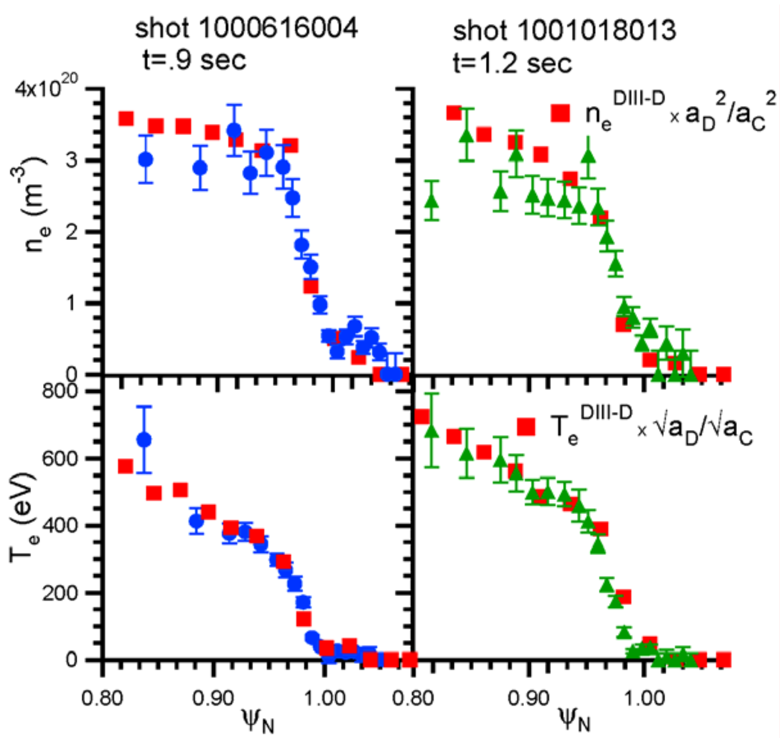


Figure 7 Detailed dimensionless comparison of H-Mode pedestal structures on C-Mod and DIII-D show excellent agreement across the entire profile. The discharges are tuned so that they match at the top of the pedestal. The DIII-D data are scaled using the Kadomtsev constraints.

The closely related topic of pedestal relaxation physics is addressed in experiments on small-ELM regimes, including ITPA experiments PEP-12 (EDA/HRS comparison with JFT2-M) and PEP-16 (C-Mod/NSTX/MAST small ELM comparison). Both of these joint experiments address the C-Mod EDA H-mode regime, which features good energy confinement and reduced particle confinement, with no large impulsive heat loads to the divertor. The dominant relaxation mechanism is governed by a quasi-coherent mode in the 100 kHz range; at the highest pressures obtained in this regime, the QC mode is supplemented or replaced by small ELMs, which maintain the steady-state character. Similar, but not identical effects are observed at similar values of collisionality and q_{95} in the high aspect ratio (A~5) tokamak JFT2-M. C-Mod is undertaking an experiment in which the shape (but not the aspect ratio) of JFT2-M is used, and a detailed comparison of the pedestal parameters and edge relaxation is studied. NSTX, has reported similar small ELM behavior, referred to as "Type V" ELMs, but has not observed the analog of the QC mode. Coordinated experiments between C-Mod and the low-aspect ratio STs NSTX and MAST are proposed under Joint Experiment PEP-16 to elucidate further the physics of the pedestal relaxation mechanisms involved. The importance of this topic to the burning plasma experiment is high, because the survivability of the ITER divertor under the impulsive heat loads of giant "Type I" ELMs is questionable.

Additional examples of ITPA Joint Experiments which depend on the substantial leverage in scale size between C-Mod and JET are CDB-4 (ν^* vs n_G scaling experiment) and CDB-8 (ρ^* scan). The first of these aims to resolve the issue of whether the apparent dependence on collisionality observed in database scalings of global confinement is in fact governed by collisionality or by proximity to the density limit. Collisionality, usually expressed in terms of the neoclassical parameter ν^*, is a true non-dimensional plasma physics parameter. On the other hand, the ratio n/n_G where n_G is the Greenwald density limit, is not such a parameter; assuming the Greenwald expression is accurate, the implication is that atomic physics effects influence the density limit, which is not unreasonable since the limit seems to be imposed largely by effects near the plasma edge. However, the difference between the two parameters is very small, scaling with the 1/4 power of the size of the device. An experiment to distinguish between the two models therefore requires the largest possible range in device size, which is afforded by the pairing of C-Mod and JET, providing a scale factor of about 4.5, or a range in $a^{1/4}$ of about 1.5. The difference is important to ITER, which will operate at low values of ν^* but at rather large values of n/n_G. The ITPA Joint Experiment CDB-8 aims to refine the prediction of the scaling of confinement with the inverse gyro-size ρ^*. Present-day tokamaks, including C-Mod, can operate with the same beta and collisionality as ITER, but have smaller gyro-size (larger ρ^*) by significant factors. The scalings derived from the present database imply an uncertainty on the ρ^* scaling of the non-dimensional confinement $B\ \tau \sim \rho^{*\ -3}$; since the extrapolation in ρ^* from present devices to ITER is at least a factor of 3, the resulting uncertainty corresponds to a confinement uncertainty of 40%. The aim of the joint experiment, which was in part proposed by DIII-D, is to carry out overlapping ρ^* scaling experiments in JET and C-Mod, nearly doubling the range of ρ^* that could be covered in any single device. In carrying out such an experiment, holding ν^* and β fixed, the temperature and density are varied along with the toroidal field for each device in such a way that ρ^* varies with $B^{-2/3}\ a^{-5/6}$. The range of ρ^* accessible within a single device is operationally restricted. In the plan of CDB-8 a non-dimensional identity would be established between JET and C-Mod at one point (5.3T for C-Mod, 0.9T for JET), and C-Mod would proceed to scan to larger ρ^* (lower field) and JET to smaller ρ^* (higher field), spanning a total range in ρ^* of more than a factor of 3. Additional experiments tentatively planned for DIII-D and ASDEX-U would provide checks on intermediate points, but would not further extend the range. The results should lead to improved predictive capability for the confinement in ITER.

Departures from non-axisymmetry in the tokamak magnetic field structure produced by the as-built coil set can destabilize non-rotating tearing modes (locked modes) which can significantly impact plasma operation. Such error-field-induced locked modes can lead to disruptions, as well as degradation of confinement. The impact of these modes on the operation of future burning plasma experiments such as ITER has been a matter of concern; key issues are the prediction of the error-field sensitivity (threshold perturbation) in such devices and the requirements for corrective measures. Joint experiments (ITPA experiment MDC-6) making use of non-axisymmetric coils to apply known field perturbations are being undertaken among C-Mod, DIII-D, and JET in order to provide a basis for extrapolative prediction (in field and size) of the symmetry requirements for ITER. These experiments are conducted with matching shape and normalized plasma parameters in the three devices, spanning a range of size of nearly a factor of five. Preliminary results comparing C-Mod and JET indicate that the required symmetry will be in the range of a part in 10^4, which is within the capability of the planned ITER correction coil system. Earlier projections had indicated a more stringent requirement, in the range of a part in 10^5. Most of the above experiments were carried out with the *a priori* expectation that plasma physics considerations would provide the dominant effect, or are intended to test that hypothesis.

Major disruptions pose a significant challenge for ITER. Experiments, first carried out on DIII-D, using disruption mitigation with massive gas jet injection have shown significant promise. These studies must be expanded to explore higher absolute pressure plasmas (C-Mod) and larger size plasmas (JET). Joint studies among C-Mod, JET and DIII-D are coordinated through ITPA activity MDC-1.

Dimensionless scaling comparisons are also part of the collaborations under the auspices of the ITPA Divertor/SOL group. Dimensionlessly similar SOL discharges at C-Mod, JET and DIII-D have been used to study the scaling of radial ion transport (DSOL-3). This study showed transport to be a very weak function of ν^*, ρ^*, and β. Additionally, it gave rise to the result that neutral physics could be playing an important role in setting density profile shapes in the SOL. Given that ITER will have SOL parameters very close to those of C-Mod, this has important implications for the wall and second divertor fluxes as well as choice of tile material in ITER. The DSOL-3 work is being continued with DIII-D (transport during ELMs), ASDEX-Upgrade and MAST.

A rather different set of considerations informs the ITPA activity DSOL-5 (Role of Lyman absorption in the divertor). A burning plasma such as ITER or a reactor is predicted to operate under conditions for which the divertor plasma is opaque to Ly-α radiation. This situation has importance for the ionization balance and detachment physics in the divertor, and therefore can impact the performance (ability to achieve detachment) and survivability of the ITER divertor. To lowest order the radiation trapping phenomenon depends on the product n_0L, where n_0 is the divertor neutral density and L the scale size of the divertor. The only existing tokamaks which even approach the n_0L product of ITER, and they only to within a factor of 3 or so, are C-Mod (large $n_0{\sim}10^{21}$) and JET (large L). ITER is projected to have similar n_0 to C-Mod and size larger than JET. Benchmarking of codes, which would be used to extrapolate to ITER, depends on data from these two current extremes of tokamak experience in this parameter. C-Mod is currently the only tokamak of the two that can provide the data needed.

In the area of hydrogen (D or T) retention C-Mod's unique high-Z surfaces provide an important contribution to the DSOL-13 collaboration – D co-deposition on the sides of tiles. Tiles of various geometries and gaps are installed on a number of tokamaks. Some, including those from C-Mod, are already being analyzed. The striking result is that the D co-deposition on the sides of tiles is substantial, rivaling that on front surfaces. The implication for ITER is serious as it could greatly complicate T removal.

Because of C-Mod's leading role in investigating radial edge ion transport, we have just initiated a collaboration (DSOL-15) with NSTX and TJ-II to compare turbulence characteristics with a shared high-speed camera (PSI). Such 'blob' characteristics are the basis for comparison with models and central to determining the underlying physics.

The physics of MHD modes driven by energetic particles (Alfvén eigenmodes) is crucial to a burning plasma. By comparing experimental measurements of Alfvén eigenmode damping and drive in large size high temperature regimes (JET) and smaller size regimes at ITER densities and toroidal fields with equilibrated ion and electron temperatures (C-Mod) we expect to learn how to extrapolate the results from present devices to ITER. Through continuous improvement of the already extensive energetic particle-wave diagnostics as well as combined ICRH and LHCD on both machines, direct comparisons with theory of both active and passive energetic particle driven modes can be made in these complementary regimes. These are considered high priority joint experiments by the ITPA because of their importance to the assessment of operational regimes for ITER.

Results from a pair of moderate toroidal mode number active MHD antennas on Alcator C-Mod have shown for the first time that moderate-*n* AEs can be driven and detected. These results indicate that the low-*n* stability results previously obtained on JET may not scale in the same way at higher mode number. This is particularly relevant to ITER because the α-particle driven TAEs are expected to have moderate toroidal mode numbers. Expected results from the new moderate-*n* AEs antennas on JET will help to answer the questions raised by these recent C-Mod results, and joint experiments between the two machines should help improve our understanding of moderate-*n* Alfvén eigenmode stability for ITER.

Experiments on the ITER "Hybrid Scenario" regime are also a high priority ITPA activity. This regime is intermediate between the standard ELMy H-mode scenario and fully non-inductive AT operation, and is characterized by long pulse operation with high neutron fluence. C-Mod experiments in support of these scenarios, proposed by A. Sips (IPP-Garching) would employ lower hybrid current drive to maintain the desired current profile with q(0) just above 1 for pulse lengths longer than the current relaxation time. Key features of the C-Mod experiments, in addition to the long normalized pulse length, the use of LHCD and the lack of core particle and momentum sources, would be the evaluation of divertor power handling and impurity radiation with high-Z plasma facing components.

C-Mod is also involved in bilateral collaborations involving development and testing of state of the art modeling software. Examples include collaboration with IPP-Garching on full wave modeling of ICRF and LH (TORIC) scenarios and CEA-Cadarache on 2-D Fokker Planck modeling of bremsstrahlung emission from LH generated energetic electrons. Experiments are performed to validate the physics kernel and computational methods of these codes. Benchmarking of the TOPICA antenna modeling code is described in the section on wave-particle interactions.

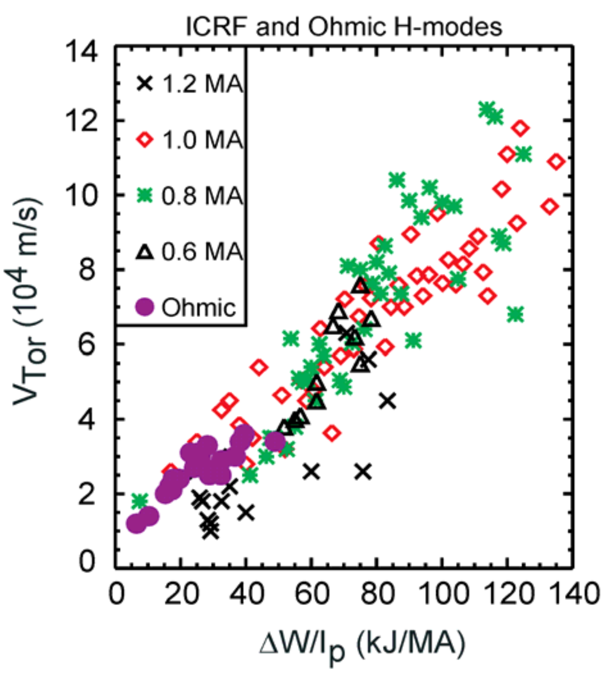


Figure 8 Strong core rotation seen in high pressure C-Mod discharges. In most other tokamak experiments, transport-driven flows are masked by the presence of dominant externally applied torque from neutral beam injection.

E. How does Alcator C-Mod contribute to Fusion Science and the Vitality of the U.S. Fusion Program?

E.1 Introduction: The Alcator C-Mod program addresses and contributes to a broad range of scientific and technical issues of importance to fusion science. We emphasize areas where C-Mod has unique capabilities, is in unique parameter regimes, observes unique or unusual phenomena, or can make important comparisons with other devices. However, we do not artificially limit the scope of our scientific investigations. Comparisons with theory and modeling form an important part of the program with theory playing a critical role in motivating experiments and in determining their design. The C-Mod program is very well aligned with scientific issues identified by the recent FESAC Priorities Sub-panel. Their report was organized around 15 topical questions to which C-Mod contributes strongly to 9 (T1-T5, T10-T13) and makes some contributions to 3 others (T6, T14, T15). Of the 14 activities called out as "high priority" and worthy of additional funding, 8 are already major foci of the C-Mod program. These are: 1) carry out additional science and technology activities supporting ITER; 2) predict the formation, structure, and transient evolution of edge transport barriers; 3) mount a focused enhanced effort to understand electron transport; 4) pursue an integrated understanding of plasma self-organization and external control, enabling high-pressure sustained plasmas; 6) extend understanding and capability to control and manipulate plasmas with external waves; 10) pursue optimization of magnetic confinement configurations; 11) resolve the key plasma-material interactions, which govern material selection and tritium retention for high-power fusion experiments; and 14) expand the effort to understand the transport of particles and momentum.

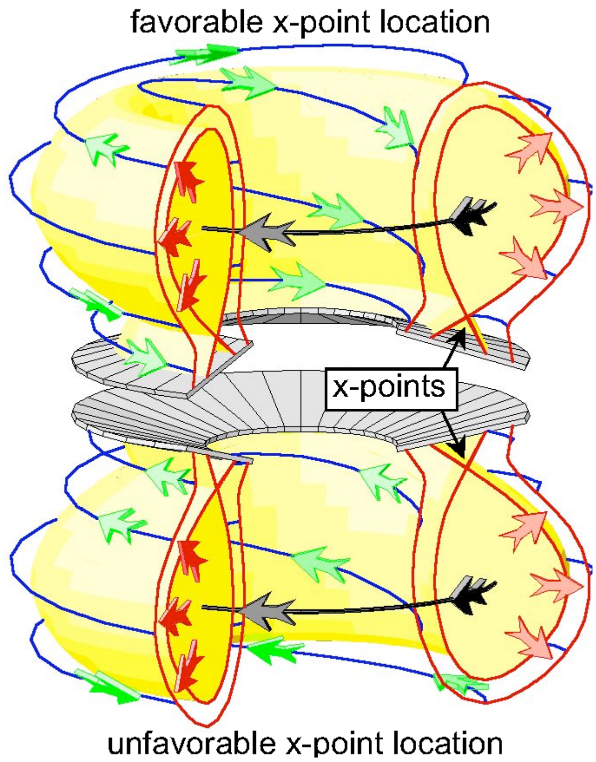


Figure 9 Strong SOL flows are driven by the dominance of particle losses across the separatrix on the low-field (large R) edge of the plasma. This leads to a density deficit on the high field side; the resulting parallel gradients drive particle flows, which have net momentum. For single-null topology, the flow direction with respect to I_p is co or counter, depending on the position of the x-point relative to that of the B×∇B ion drift. Core rotation always increases in the co-current direction when the plasma pressure and pressure gradient are increased, but the initial (L-Mode) core rotation is more in the counter direction when the drift is in the unfavorable direction. These results point to the explanation for the long-standing puzzle of H-Mode threshold dependence on ion drift direction.

E.2 Transport: Our ability to measure rotation dynamics in torque-free discharges led to the unexpected observation of strong, self-generated toroidal flows in C-Mod (see Figure 8) provided new insights into momentum sources and transport, and generated significant interest within the theory community. These studies, combined with measurements of SOL flows and transport have led to a novel explanation for the field direction effect on the L-H power threshold, perhaps explaining a long-standing mystery (see Figure 9). With enhancements to the phase contrast imaging diagnostic, C-Mod will be addressing a key issue for electron transport, namely the existence of anisotropic fluctuations. These structures or streamers, with $k_r \ll k_\theta$ are at the heart of the dispute over the role of ETG turbulence in driving anomalous electron transport. C-Mod has conducted some groundbreaking work on the role of critical gradient length and marginal stability, lending quantitative support to the ITG theories for ion transport and leading directly to their refinement.

This approach also provides a paradigm in which to understand the creation and control of ITBs produced in C-Mod apparently without ExB stabilization. Future work with LHCD will enable quantitative tests for the role of magnetic shear in determining the critical gradient length and allow for investigations of anomalous particle transport in discharges with no core source and no neoclassical pinch. Work on the H-mode has included studies of local threshold conditions and comparisons with theory; demonstration of regimes with good confinement but no large, potentially destructive ELMs; and elucidation of the relative role neutrals and plasma physics play in determining the pedestal structure.

E.3 Plasma Boundary: Studies of C-Mod divertor plasmas have shown that vertical plate geometry leads to a reduction in the detachment threshold; other tokamaks (including ITER) have since adopted this geometry. Our high densities enabled the first experimental demonstration of divertor recombination and radiation trapping of Ly_{α}, and their roles in detachment; we are currently providing the only tests of the models in this area. Experiments with ITER-like parallel heat fluxes demonstrated that extrinsic impurities are required for detachment with high-Z plasma facing components. Dimensionless plasma comparisons of time-averaged transport across machines have shown remarkably similar convective velocities for tokamaks with a factor of five difference in size. The screening of impurities generated by main-chamber wall interaction has been found to vary poloidally, with the outer mid-plane, the location of the strongest radial fluxes and measured impurity source, having the poorest impurity screening. The C-Mod group has made critical discoveries concerning anomalous radial transport, clarifying the separate roles and nature of the "near" SOL (with steep gradients, lower levels of turbulence and Gaussian PDFs) vs. the "far" SOL (with flat profiles, high levels of turbulence and highly intermittent, bursty transport). The scaling of the near SOL transport with normalized collisionality has made a direct connection to the physics of electromagnetic fluid drift turbulence and to a theoretical basis for the tokamak density limit. The limit may now be understood as a transport phenomenon where the critical edge cooling occurs as the regime of "blobby" fluctuations crosses the separatrix and intrudes into the edge plasma. High-speed imaging techniques and fast-scanning probes have elucidated the nature of bursty transport. The high levels of radial transport in the far SOL led to a paradigm of main chamber recycling with clear implications for erosion, impurity sources and wall lifetime. Unique inner wall probe measurements have found SOL flows close to Mach 1, with implications for the transport of impurities as well as transport of momentum between the core and edge and the L-H threshold. These measurements together with the inboard-SOL spectroscopic studies have established that there are far lower transport and associated turbulence on the inboard (favorable curvature) side than the outboard. This discovery transforms the field of SOL modeling.

E.4 Macrostability: Disruption research receives strong emphasis on C-Mod due to its operational parameters. Extensive instrumentation has yielded a wealth of information on halo current magnitudes, scalings, and toroidal structure, providing the first evidence for non-axisymmetric currents and forces. This information, incorporated into the ITPA disruption database, provides crucial guidance for the engineering specifications on ITER in-vessel hardware. Tests of mitigation techniques using massive pellet injection and impurity pellet injection have been carried out and experiments employing massive impurity gas puffs are in process. These will be an important part of the C-Mod disruption program and are particularly relevant to ITER and future reactor plasmas due to the similarity of plasma pressure, energy density, and current density. Locked mode research on C-Mod enables us to push performance to higher currents, and provides an important benchmark for the scalings of size dependence of the threshold error field, particularly since it operates at the same field as ITER. This work has confirmed size scalings which lead to tractable error-field requirements in ITER. C-Mod also has an active MHD antenna system which is used primarily for studying Alfvén modes and grand cascades. Emphasis has been on studying Alfvén eigenmode drive and damping in reactor-relevant regimes.

E.5 Wave-plasma Interactions: C-Mod's combination of novel RF diagnostics and advanced full-wave simulations has allowed us to understand the physics of ICRF mode conversion, an important power absorption mechanism in multi-ion species plasmas. The RF wave measurements combined with high-resolution power deposition profiles (plasma response) constitute the most stringent test for ICRF simulation to date, accurately resolving wavelengths ranging from 0.5 – 10 cm. In the near future, experiments will investigate how to optimize the localized strong absorption of the mode converted waves for local current profile control and/or shear flow control. As a result of the high density, variations in antenna loading are dominated by profile affects rather than simply the distance to cutoff as on other experiments (ITER loading is likely to be a combination). This indicates that new integrated 3-D antenna models, where the plasma is treated as a dielectric, will be insufficient to model accurately antenna loading. In antenna design, we have clearly demonstrated the positive effect of orienting the RF electric field perpendicular to the tokamak magnetic field. We have also detected the existence of two-surface multipactor discharges in a vacuum coaxial transmission line perhaps demonstrating a mechanism for providing seed electrons for initiating high voltage breakdown. In the LH area, measurements made by a special purpose energy-resolving X-ray camera will permit detailed comparison with modeling results concerning the damping of LH waves and the generation and confinement of LH-driven fast electrons.

E.6 Advanced Tokamak Thrust: The advanced tokamak thrust involves *integrating* the physics issues described above, as expressed by the FESAC high priority activity "Pursue an integrated understanding of plasma self-organization and external control, enabling high-pressure sustained plasmas." Many non-linear interactions of phenomena with a wide range of time and spatial scales occur in plasmas operating close to performance limits. For example, C-Mod has discovered that precise tailoring of ICRF power profiles influences, and can be used to control, the energy and particle transport. The associated reduction in turbulence in turn leads to localized gradients of temperature and density, and thus of self-generated bootstrap current. Modeling has shown that external current drive by LHCD will also be strongly influenced by the n and T barriers. Conversely, the resulting non-inductive current modifies magnetic shear which will further change the transport and kinetic profiles, as well as MHD stability limits. Strong shaping and broad pressure profiles have been shown to increase these limits. All of these interactions are highly non-linear and require an experimental campaign coordinating issues usually considered in isolation, closely linked with integrated modeling, to determine the self-consistent, steady-state solutions as well as the dynamic evolution. Understanding of pedestal and ELM physics arising from the transport program has for example recently been used to demonstrate H-mode density and temperature profiles optimized for efficient LH current drive. C-Mod plays a unique role in this challenging effort due to its record-breaking pressures, and quasi-steady state operation. C-Mod scientists are also involved in advancing modeling capabilities.

E.7 Collaborations with Theory and Modeling: The C-Mod project has fostered fruitful and active collaborations with numerous members of the theory and computation community in the US and abroad. This interaction has led to seminal advances in the understanding of ICRF mode conversion processes, including the first ever detection of mode converted ion cyclotron waves in a tokamak. Other work has featured 3D ICRF modeling and experiments used for validation of antenna design. Observations of self-generated rotation in C-Mod has stimulated significant theoretical activity and led to the off-axis ICRF heating experiments which resulted in a unique internal transport barrier regime. Collaboration with gyro-kinetic experts on this topic has led to a possible understanding of the production and control of ITBs. The C-Mod group also actively collaborates with the modeling community on the use of integrated computer models for transport analysis and for predictive advanced tokamak scenario development. Disruption mitigation research on C-Mod has leveraged heavily with large scale extended MHD simulations and active external collaborations with experts in nonlinear MHD evolution have also led to an understanding of TAE mode and Alfvén cascade observations in C-Mod. The project also sustains active collaborations with theorists working in the areas of divertor, edge and pedestal physics where we interact with most of the important modelers working on integrated edge modeling, nonlinear edge turbulence and the L-H threshold.

E.8 Student Education and Training: As a university laboratory with 60-70 graduate students enrolled at any one time, MIT's Plasma Science and Fusion Center has an important role in training the next generation of fusion scientists and engineers. The C-Mod program is unique, world-wide, among high-performance magnetic confinement experiments in being sited on the main campus of a university and being fully integrated into an academic environment. C-Mod takes as a major element of its mission the education of this new generation of plasma physicists and engineers. On the experiment, typically 25-30 graduate students (see Figure 10) and 5-10 undergraduates work side by side with PhD scientists on problems central to the machine's goals. Students form an integral and important part of the research team and are given significant resources and run time on the experiment. Since the start of the C-Mod project, more than 50 of our students have graduated and found work in academia, industry and at the national labs. Given the demographic challenge that the fusion program faces, including the start of ITER operations in about 10 years, this steady stream of trained scientists is one of our most valuable contributions.

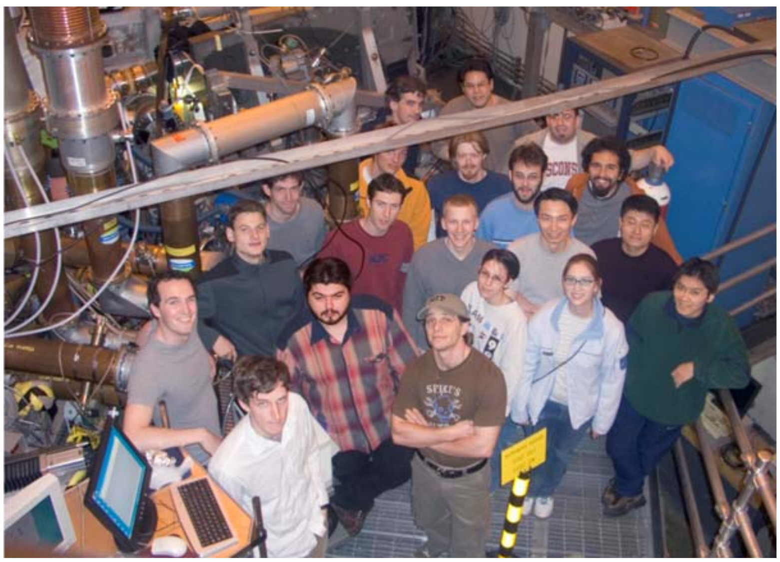


Figure 10 Some of the current graduate students doing their research on Alcator C-Mod. Students are drawn from four departments at MIT, as well as from collaborating institutions.

E.9 MDSplus/Remote Participation: The C-Mod team has made a number of important contributions in the areas of data acquisition, data management and remote participation. The MDSplus data system, a joint development with LANL and IGI, led by MIT, is now used at over 30 sites around the world, including all of the major fusion facilities in the US. By providing common data structures and interfaces, it has enabled experimental collaborations around the world. (For example, it now serves as the remote data access standard for the JET collaboration and is applied to the ITPA databases.) We have proposed extending this work for use by ITER. In the area of remote participation, C-Mod was the first magnetic confinement experiment to be run from an off site location (from LLNL in 1995). Using lessons from that experience, we have built up tools to provide an off-site environment that could be nearly as productive as the on-site one. One measure of success is that in the current campaign, three collaborators have chosen to lead experiments from their own sites rather than travel to MIT. This experience should be applicable to ITER and strengthen US participation. We are currently exploring new technologies which will exploit the convergence of interpersonal communications media including voice, video, email, messaging and telephony.

F. What opportunities would be lost if Alcator C-Mod does not operate?

A large number of very significant research opportunities would be lost if Alcator C-Mod were to cease operations during the coming five years. Many of these rely on facility capabilities that are unique in the world fusion program, and can not be accomplished anywhere else. Others are tied to joint experiments with other facilities (both domestic and international), where C-Mod provides unique high leverage contributions because of size, field and power density.

ITER is contemplating using all tungsten Plasma Facing Components (PFCs), based on concerns of T retention, due to carbon, limiting operation. C-Mod is the only diverted tokamak in the world which has exclusively employed high-Z metallic plasma facing components in all high heat flux regions from the very beginning of operations. Over the next 2 years, we will convert from molybdenum to the ITER specific choice of tungsten for all high heat flux outer divertor PFCs. C-Mod also operates in a unique parameter range with regard to divertor particle and power densities, both similar to or above the values expected in ITER. ***Over the next 3 to 5 years, critical studies of hydrogenic inventory, disruption survivability, particle control, power handling, wall conditioning and impurity dynamics, with ITER-prototypical materials, and heat and particle flux conditions, can be carried out only on C-Mod.***

Disruption mitigation, another potential make or break ITER issue, ***must be studied under conditions of high absolute plasma pressure,*** where gas penetration may be the most difficult. C-Mod produces the highest pressure tokamak plasmas in the world ($<P>\sim 2$ bar), close to that expected in ITER (and at the same field and therefore β). Massive gas puff disruption mitigation experiments will begin later this year, and important results can be expected over the next 3 years.

It has been recognized that the high neutral densities in ITER will lead to radiation trapping (principally Ly-α) and strongly affect the capability to achieve divertor detachment, an operational regime required to handle the very high parallel power flux densities coming to the plates. Models are just beginning to include this effect but only C-Mod (high density, n_0) and JET (large divertor size, L) approach the conditions needed to provide a test of the models near the high n_0L of ITER. ***C-Mod is currently the only facility supplying the data required for this important test.*** The JET diagnostic complement is currently not configured for this type of study.

C-Mod has recently brought to the attention of the international community the result that perpendicular transport can compete with parallel transport in the SOL. This has forced ITER (and world) attention on this issue because of the obvious danger for damage of the first-wall surfaces and secondary divertor which are not designed for significant plasma fluxes or heat load. C-Mod has led the world research in this area both on a microscopic level (e.g. turbulence) and in a time-averaged sense. The ***opacity of the SOL to neutral penetration*** may be the determining factor in wall interaction and the ***C-Mod SOL is uniquely like that of ITER*** in that parameter thus providing another important area of contribution.

Minority ICRF heating is one of the key ITER heating schemes. It has been proposed and tentatively agreed that the US will contribute approximately half of the ICRF system for ITER. Since ***Alcator C-Mod is the only US tokamak that is pursuing minority ICRH***, it would seriously undermine our international credibility, as well as removing important opportunities for research and prototyping in support of this task, if C-Mod does not operate. Moreover, if the US is to benefit by exerting leadership in the physics aspects of ICRF, which are just as important as the technology, ***when ITER becomes operational, we need expert tokamak ICRF physicists. C-Mod is the place where they will be trained and gain experience***.

Alcator C-Mod's focus on ICRF, state-of-the art wave diagnostics, and strong modeling capabilities, provide unique opportunities to study innovative wave-particle physics such as mode-conversion current drive and flow drive. Since ***ICRF decouples the major heating and current drive from particle and momentum sources***, ***C-Mod is a facility without peer for investigation of the source-free particle transport and natural rotation*** that is going to be characteristic of a reactor and to a large degree of ITER. The C-Mod discoveries of spontaneous rotation are already changing the "standard view" of this topic. Rotation has a strong influence on both transport and MHD stability. Understanding the prospects in ITER for shear flow transport suppression and wall stabilization of MHD modes is increasingly urgent.

To access quasi-steady state Advanced Tokamak regimes, ***ITER will need efficient far-off axis current drive. Lower Hybrid is the most likely (perhaps the only) tool that can fulfill this requirement.*** ITER has reserved port space for LHRF, but considers it an upgrade, rather than a part of the baseline design. The recent implementation of Lower Hybrid RF systems on C-Mod enables experiments which will provide unique information at the ITER field and density (and thus dielectric constant). Combining this with the exclusive ***C-Mod*** capability to run long pulse lengths (relative to resistive skin times) at high temperature and low Z_{eff}, ***will provide essential guidance to ITER regarding the possible implementation of LH tools.***

Large (type I) ELMs, which regulate the pedestal and prevent impurity accumulation in the vast majority of H-modes studied world-wide, ***pose a significant danger to ITER.*** The resulting divertor erosion is predicted to cause significant damage in as few as 10 high-power discharges, which would be completely unacceptable. There is great interest in H-Mode regimes which exhibit high pedestal pressure and good energy confinement, without impurity accumulation and without large, discrete ELMs. Many approaches are being explored. ***C-Mod brings a unique perspective to aspects of this problem, because of the high absolute densities (plasma and neutral) in the pedestal,*** and the discovery of the enhanced D-alpha (EDA) H-mode regime. EDA H-mode satisfies the ITER requirements for performance and impurity control, but extrapolation to ITER is currently uncertain. In particular, the relative importance of collisionality and density has yet to be unfolded (ITER will operate in a regime of high density and low collisionality, which cannot be accessed on any current device). At the highest input powers and β_N, the plasma makes a transition from EDA to a regime with small ELMs, also of interest for ITER. Because ITER will operate with simultaneous high density and low collisionality, not accessible on any current device, it will be necessary to combine dedicated C-Mod experiments with coordinated experiments between C-Mod and other tokamaks to improve the accuracy of extrapolation to ITER.

C-Mod is the only divertor tokamak that routinely operates in the regime with equilibrated electrons and ions. This situation will prevail in both ITER and ignited reactor plasmas and it influences a broad range of plasma properties, particularly turbulent transport. Studies of energy, particle and momentum transport under these conditions provide insights that are difficult or impossible to obtain elsewhere and extend the validation of transport models into unique, reactor relevant parameter ranges.

C-Mod is poised to answer a critical question in transport physics, namely the role and nature of ETG driven turbulence. The electron transport channel is poorly understood, but cannot be neglected in future machines like ITER, where electrons and ions are tightly coupled. Historically, short wavelength fluctuations, like ETG, were thought to be incapable of driving significant transport because the turbulent step size was very small (on the order of a few electron gyro-radii.) Recent work with nonlinear gyro-kinetic continuum codes (GS2, GENE, GYRO) found that ETG turbulence developed extended radial structures which could increase the calculated transport to important levels. However, results from another major code, GTC, came to contrary conclusions. Ongoing enhancements to the PCI diagnostic should enable independent measurement of k_r and k_θ on C-Mod, with a reasonable degree of spatial localization. A crucial test of the theory would be to find $k_r << k_\theta$.

C-Mod will continue its pioneering work on the role of turbulent transport in defining the tokamak density limit. Optimal performance of reactor scale machines, like ITER, depends on operation at very high density. Without knowledge and validation of a physics-based mechanism, extrapolation is untrustworthy. Past work has demonstrated important modifications of edge and SOL turbulence as the density limit is approached. This has been connected, quantitatively, to theories of electromagnetic fluid drift turbulence which predict important dependences on the normalized collisionality and normalized pressure gradient. Future work will concentrate on the manner in which these mechanisms interact with the overall transport and MHD equilibrium to create the density limit.

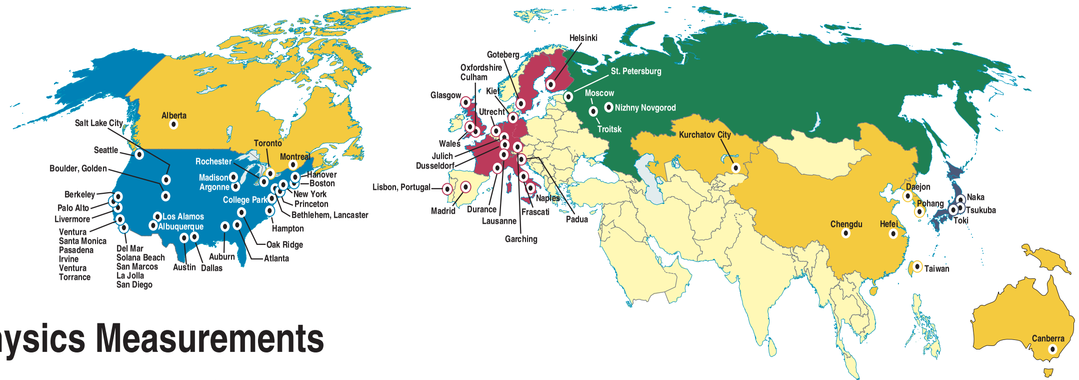
C-Mod provides unique high leverage points for dimensionless and dimensional scalings. Three important demonstrated examples are H-Mode confinement, disruption halo currents, and error-field induced mode-locking. In the world portfolio of currently operating high-performance conventional aspect-ratio divertor tokamaks, two devices are mid-sized and two are large. C-Mod is the only high-performance compact tokamak. If C-Mod were to stop operations, the highest leverage points for future scaling studies would be lost.

C-Mod is the premier US tokamak for training students. It has by far the most graduate students of the major US fusion facilities. Including students from MIT and collaborating institutions, there are currently 28 graduate students doing their research on C-Mod. In addition, typically about 5 to 10 undergraduates are involved in the C-Mod program, both from MIT’s Undergraduate Research Opportunities Program, and through the DoE Undergraduate Summer Fellowship Program. We currently expect that ITER will begin operations in about 10 years. In the subsequent 10 to 20 year period, there will be a critical need for highly qualified and experienced US scientists and engineers. Without C-Mod's major contribution to educating these experts, between now and the start of ITER operations, the US will be in a far weaker position to take advantage of the exciting ITER opportunities.

C-Mod is unique in the world. There is no other high field, compact high performance divertor tokamak. In the coming five years, vitally important research, including many critical ITER R&D tasks, can and will only be accomplished on Alcator C-Mod.

This work is supported by the US Department of Energy, Office of Fusion Energy Sciences.

DIII–D National Fusion Program
Submitted to FESAC Facilities Panel May 28, 2005



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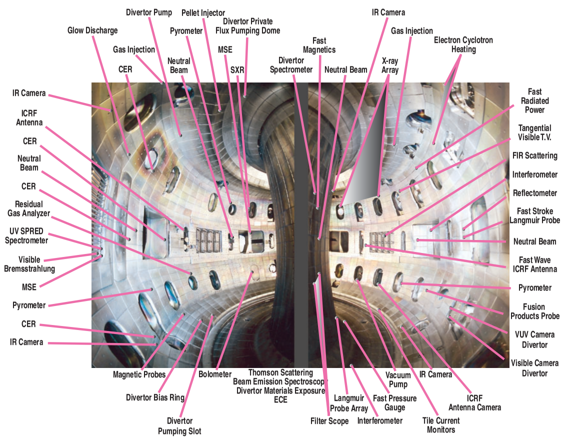

International Research Team

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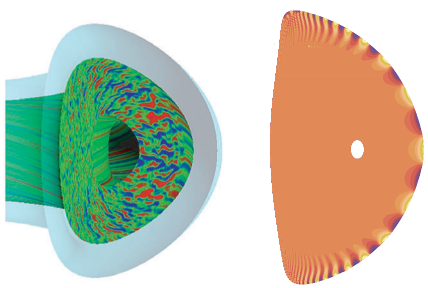

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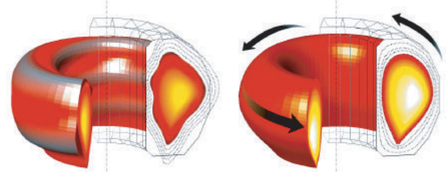
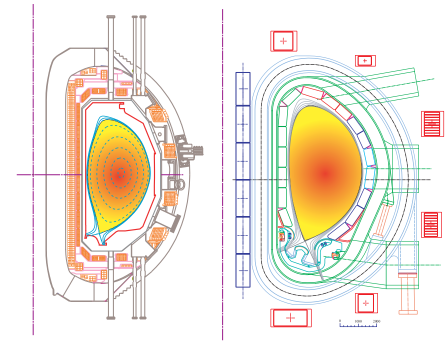
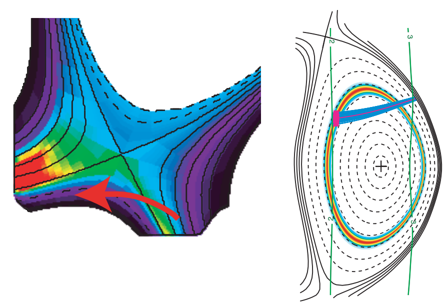


Scientific Flexibility
DIII–D is 1/4 Scale ITER



Plasma Control



DIII-D NATIONAL FUSION FACILITY

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PREFACE

It is my pleasure to submit this document on behalf of the research team of the DIII-D National Fusion Program to the Fusion Energy Sciences Advisory Committee Panel on Facilities.

We have written this document in essentially two parts. This first part presents a concise description of the components of the DIII-D National Fusion Program: its excellent international research team; its unique and broad research program; and the DIII-D facility, the superb scientific instrument that enables DIII-D's research productivity. In the second part, we give explicit answers to the four questions posed to the panel by Dr. Raymond Orbach in his charge letter of April 5, 2005. The text is supplemented by several detailed tables which show the breadth of the Program and provide the Panel with information in depth.

Additional information requested by the Panel is included in the appendices.

Appendix A contains the information requested on the staff of the DIII-D Program. An introductory page guides the reader to nine documents describing the international team, the 515 users of the facility, the record and current status of the DIII-D Program in education, recent foreign collaborations, and DIII-D's large role in the International Tokamak Physics Activity.

Appendix B contains the requested publication list for 1999-2004.

Appendix C contains the requested citation study.

Appendix D is a summary of the FESAC Priorities Panel key questions and critical activities. It is provided as a convenient reference.

If the Panel requires any further information, we will be pleased to provide it

Ronald D. Stambaugh

Ronald D. Stambaugh
Director, DIII–D National Fusion Program

May 27, 2005
Date

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DIII–D NATIONAL FUSION PROGRAM

1. EXECUTIVE SUMMARY

The DIII-D Program is the world's premier fusion science program and the DIII-D facility is the world's best fusion plasma physics scientific instrument. The DIII-D Program is a large international program, with 90 participating institutions and research team of 515 users. DIII-D research has been recognized a record four times with the American Physical Society Excellence in Plasma Physics Prize. The research team contains 11 winners of the Excellence Prize and 71 Fellows of the American Physical Society. The quality of the broad and basic scientific research contribution from DIII-D was a principal reason the recent National Research Council Assessment of the Department of Energy's Office of Fusion Energy Sciences Program had as its main conclusion: "the quality of the science funded by the United States fusion research program in pursuit of a practical energy source of power from fusion (the fusion energy goal) is easily on a par with the quality in other leading areas of contemporary physical science."

DIII-D Makes Unique Research Contributions in the International Fusion Program

The DIII-D Program is the world leader in Advanced Tokamak (AT) research, which aims to discover the physical limits to stably containing plasma. DIII-D pioneered the AT research line, which has now taken root worldwide; almost every tokamak now pursues those aspects of AT physics its hardware capabilities allow. The focus of the AT line is to develop the scientific basis for **high gain, steady-state operating modes for ITER** and future fusion systems. The building blocks for this goal have all been separately demonstrated in DIII-D and an integrated scientific basis for high gain, steady-state should be achievable in the next five years.

The DIII-D Program affords the world fusion program a unique opportunity to finally identify the basic physics mechanisms of heat transport from turbulence owing to its unique set of turbulence diagnostics, the new fast computer cluster that runs the most comprehensive gyrokinetic code in the world, and the excellence of its research staff in the transport area. DIII-D established that sheared E×B flows in plasmas suppress turbulence to form transport barriers. Its balanced neutral beam injection and rf heating systems enable research to understand further the physics of transport barriers and flows in plasmas.

DIII-D is a world leader in macroscopic plasma stability research, with its unique resistive-wall-mode and world-class neoclassical-tearing-mode, disruption-mitigation, and pedestal research.

DIII-D's Unique Facility Capabilities Enable Unique Research

DIII-D's unique **resistive wall mode (RWM) feedback system** enables sustained operation above the no-wall beta limit both with and without plasma rotation. Its heating systems have sufficient **power to reach the beta limit** anywhere in the DIII-D operating range. DIII-D's high normalized beta allows **high bootstrap fraction**, for which it shares the world record (80%) with JT-60U. The remaining current drive for **100% noninductive operation** in DIII-D will be made up of neutral beam current drive, fast wave current drive, and electron cyclotron current drive (ECCD). The **most powerful, long pulse EC system** in the world will provide the key off-axis current drive. DIII-D is also a world leader in stabilization of **neoclassical tearing modes** (NTMs) by highly localized ECCD. The DIII-D **real-time plasma control system** is the most advanced in the world, recently demonstrating the complete closed loop NTM suppression using a real-time q profile measurement from the motional Stark effect (MSE) diagnostic to guide the EC waves to the NTM island location. This control system has been chosen for all the most recently constructed experiments: NSTX (PPPL), MAST (Culham), KSTAR (Korea), and EAST (China). The **unique flexibility** of DIII-D's plasma shape control system enables investigation of issues that depend on details of plasma shape and makes it the partner of choice in joint **dimensionless parameter matching** experiments worldwide.

DIII-D has two methods of edge localized mode (ELM) suppression: **edge stochastic fields** (unique in the world) and **quiescent H-mode** operation (discovered on DIII-D and made possible by one counter neutral beamline). DIII-D has already **demonstrated 100%-noninductive operation** and **ITER hybrid mode plasmas** have been operated for over nine current diffusion times**.** With its pumped divertor, DIII-D is the only tokamak in the world that can control the **density in H-mode** and will be capable of 10-second pulses with controlled density in **high triangularity and double-null** plasma shapes needed for high beta.

The DIII-D diagnostic set is the best in fusion research. No other tokamak has a **divertor Thomson scattering** system to make 2-D maps of divertor electron temperature and density, a suite of **turbulence diagnostics** (all from universities) that provide unique coverage of the **whole wavelength range for turbulence** in the electron and ion channels and enable unprecedented views of **Alfvén instabilities, and a lithium beam** system to measure the **edge current peak** crucial to pedestal stability. For over a decade, only DIII-D has had a **radial electric field measurement** (DIII-D has two techniques) crucial to transport studies. DIII-D was the first to develop a **real-time equilibrium calculation** inside its control system (now also in use on NSTX) and to couple that calculation and control to a **real-time q profile measurement**.

<u>The DIII-D National Fusion Program Will Continue to Play a Key Role in the US Scientific Contribution to ITER and in Enabling the US to Benefit From Its Participation in ITER</u>

DIII-D had a profound impact on the redesign of ITER**.** DIII-D is developing the physics basis for key ITER issues and **advanced ITER operation** (hybrid and steady-state modes). Continued research on DIII-D will significantly advance the **research program on ITER**, reducing technical risk and saving expensive operating time. Pedestal physics will determine ITER's baseline performance. Disruption mitigation and edge instability physics are crucial to the lifetime of in-vessel components. DIII-D's carbon walls and unique diagnostics of the edge region equip DIII-D well to contribute to the best choice of first wall material for ITER. Its heating systems and extensive energetic ion diagnostics make it a world class test bed to evaluate fast ion instabilities. DIII-D will be a principal **training ground** for the US research team for ITER.

<u>Vital Fusion Research Opportunities Would be Lost Without DIII-D</u>

ITER will be the focus and flagship of the world fusion program for the next 2-3 decades, and DIII-D is the premier US facility supporting ITER. DIII-D results will be critical for ITER in-vessel and auxiliary-component design, for success in ITER's baseline performance, for enabling ITER's performance beyond its baseline, and for increasing ITER's scientific benefit generally. It will be 15-20 years before any other tokamak could be configured as DIII-D is <u>presently</u> configured to provide the scientific basis for high fusion gain, steady-state operation. No other tokamak presently operating or planned is likely to be able to answer the fundamental questions about transport from turbulence that DIII-D can answer in the next five years. It is imperative to seize these research opportunities now. With DIII-D heading its portfolio of facilities, the US fusion program will retain its position as the world leader in fundamental plasma science, and it will continue to be the recognized world leader for innovative research that is improving the tokamak concept for ITER and beyond.

2. THE DIII-D RESEARCH TEAM

The DIII-D Program is a large, international program, on the scale of the JET and JT-60U programs. Presently 90 institutions participate (Appendix A). General Atomics operates DIII-D for the Department of Energy as a true user facility. The user list (Appendix A) shows 515 professional users of DIII-D, 119 from General Atomics and 396 from other institutions. Over 50% of the scientific staff (full time equivalents) are from collaborating institutions. There were 317 scientific authors of papers in 2004 and 577 authors cumulative since the start of the program. In 2000-2004, there were 1082 visits to the GA MFE program by foreign and domestic collaborators. The DIII-D Program accounts for about 25% of the US papers at the biannual IAEA Fusion Energy Conference.

This outstanding research team has been recognized for its excellence. The staff contains 71 Fellows of the American Physical Society. DIII-D research has been recognized in the American Physical Society Division of Plasma Physics highest research award, the Excellence Prize, (Table 1) a record four times.

Table 1. Winners of the Excellence in Plasma Physics Prize

Year	Citation	Winners
2004	*For the Theoretical Discovery and Experimental Identification of Toroidicity Induced Alfvén Eigenmodes*	E.J. Strait (GA-DIII-D), Prof. W.W. Heidbrink (UCI-DIII-D). C.Z. Cheng (PPPL), K-L. Wong (PPPL-TFTR), Prof. L. Chen (UCI)
2001	*For experiments that show that sheared ExB flows can suppress turbulence and transport in tokamak plasmas and that such flows can spontaneously arise at the edge and in the core of tokamak plasmas.*	K.H. Burrell (GA-DIII-D), E.J. Doyle (UCLA-DIII-D), R.J. Groebner (GA-DIII-D), and E.J. Synakowski (PPPL-DIII-D, TFTR)
1999	*For his implementation, development, and exploitation of beam emission spectroscopy for measuring fluctuations and their relations to anomalous transport in hot, fusion-relevant plasmas*	Professor R. Fonck-Wisconsin, partially for work on DIII-D, mainly for work on TFTR.
1994	*For the experimental validation of theoretically predicted beta stability limit, and for the demonstration of extremely high beta operation of a tokamak*	T.S. Taylor, E.J. Strait, R.D. Stambaugh, and L.L. Lao (GA-DIII-D)

Research on DIII-D is open to proposals from all countries with which the DOE has a cooperative agreement. The high level of interest in doing experiments on DIII-D is shown in the table in Appendix A, which shows the country or domestic institution of origin of the 451 research proposals received for the 2004-2005 research campaign. Funding constrained run time means about 100 research proposals can get time in any year; hence the research backlog is generally about 4 years. DIII-D has an open, national system of governance.

3. THE DIII-D RESEARCH PROGRAM

The DIII-D Program mission goal is

To establish the scientific basis for the optimization of the tokamak approach to fusion energy.

Working with our international colleagues, the DIII-D Program pursues the above mission through three research goals.

1. The DIII-D Program's primary focus is the Advanced Tokamak (AT) Thrust that seeks to find the ultimate potential of the tokamak as a magnetic confinement system.
2. The DIII-D Program will undertake the resolution of key issues for ITER and other burning plasma initiatives.
3. The DIII-D Program will advance the science of magnetic confinement on a broad front, utilizing its extensive facility and national team research capability.

DIII-D has a superb record of scientific contributions to the world fusion program. Table 2 contains a summary of those achievements and their impact.

DIII-D supports a broad program of scientific advance relevant to 12 of the 15 key questions (excepting questions T7-T9 specific to inertial fusion) posed by the recent Fusion Energy

LIST OF FIGURES (Continued)

Sciences Advisory Committee Priorities Panel Report. See Table 3 for a detailed list of these future research areas, their impact, the unique capabilities of DIII-D that enable their study, and their mapping to the Priorities Panel key questions and prioritized activities.

4. ADVANCED TOKAMAK (*AT*) RESEARCH AND ITER

The DIII-D Program can establish an integrated scientific basis for high gain, steady-state fusion systems in the next five years. DIII-D is uniquely capable of providing the integrated basis for high bootstrap fraction, high beta, full noninductive operation. **DIII-D research will enrich ITER's physics program and advance the benefit of ITER by developing advanced operating modes for ITER.** Projections from advanced operating modes (advanced H-mode or hybrid and steady-state *AT*) already achieved on DIII-D (Fig. 1) indicate the potential of ITER operation with full output power (Fig. 2): (1) in steady-state with energy gain Q=5, (2) for longer than one hour with Q=9, and (3) for shorter pulses with very high fusion gain, Q $\gtrsim$ 40. DIII-D has been at the forefront in the development of these advanced scenarios and continued participation by DIII-D in these key areas is essential for maintaining the US leadership role.

Many of the key physics elements and key control tools for true steady-state are in hand, including high beta operation, high confinement, RWM stabilization, NTM suppression and avoidance, disruption mitigation, on- and off-axis current profile control, density control, and integrated digital plasma control. A key to sustaining high beta, fully noninductive discharges is off-axis current profile control to maintain stable profiles (Fig. 3). DIII-D has a very versatile set of proven current drive control tools (ECCD, FWCD, NBCD), facilitating generation and evaluation of scenarios with different current (q) profiles. DIII-D has a unique ability to pursue *AT* research at high beta (T2), made possible by strong shaping and active MHD feedback control (RWM and NTM). Because the projected fusion power scales approximately as β_N^4 in steady-state scenarios, high beta operation is extremely high leverage. DIII-D also has the unique capability to carry out *AT* research in both SND and DND, over a wide range of triangularities, and to quantify the advantages of strong shaping (T1). Density control with cryopumps in both SND and DND shapes improves the effectiveness of the current drive tools and allows operation in ITER relevant collisionality regimes. *AT* research on DIII-D is critical to advance the physics basis for steady-state operation for ITER and other future fusion devices (CTF, DEMO), will improve the probability of the success of ITER, and will increase the scientific knowledge and technological benefit gained.

Integrated, self-consistent optimization leads to new and exciting science (T3,A10). Challenging limits in the different physics areas (stability, transport, current sustainment, boundary) and operating in new regimes leads to improved fundamental physics understanding. The integration leads to new and rich scientific challenges as a consequence of complex nonlinear coupling among different elements. Finally developing self-consistent scenarios lead to a partnership between experiment and integrated modeling. Integrated modeling is required to interpret the experiment, to develop improved stable operating trajectories, and to predict steady-state high performance for ITER and beyond.

5. ITER SUPPORTING RESEARCH

DIII-D is in a unique position to provide key enabling physics solutions to ITER design issues, to advance the understanding of ITER relevant fusion science, and to develop predictive code capability to guide the experiments on ITER (A1). The DIII-D plasmas are approximately 1/4 the size of ITER plasmas and DIII-D can match all the non-dimensional parameters of ITER except the ratio of gyroradius to machine size. Besides the key issues listed below, see Table 4 for a detailed list of issues, DIII-D's unique capabilities, and the impact on ITER.

1. Development of advanced operating modes for ITER (T3,A4).
2. ELM mitigation, active ELM control, and ELM free operation (T10,A2).
3. Understanding and reducing the retention of hydrogenic species (tritium in ITER) (T10,A10).
4. Disruption mitigation, and operational methodology (T2,T6).
5. Stabilization of NTMs, including quantifying requirements for modulated ECCD on ITER (T6,A6).
6. Transport understanding and prediction, to provide basis for modeling projections and guidance for operation on ITER (T4,T5,A2,A3).
7. Development of radiative divertor solutions and validation of codes to project to ITER operation (T10).
8. Validation of codes for Alfven eigenmodes in ITER relevant operating regime (T12,A8).
9. Quantifying the benefit of internal vs external coils for RWM stabilization (T2).

6. BROAD FUSION ENERGY SCIENCE

Transport Science

Over the next five years, the DIII-D facility and program affords the world fusion program a unique opportunity to identify the basic physics mechanisms of heat transport from the plasma, owing to its turbulence diagnostics, excellent profile diagnostics, wide range of operating regimes and scenarios, excellence of a broad national transport research group, and the new fast computer cluster to facilitate experiment and simulation comparison (T4,T5,A3,A14). This facility will provide world leadership in detailed comparison of experiment and transport models. DIII-D research established the principle of turbulence suppression by sheared E×B flows and will clarify the roles of E×B shear and Shafranov shift in core transport barriers. DIII-D is the world leader in non-dimensional scaling studies (Table 2), and is the partner of choice in many ongoing similarity experiments owing to its unique shape flexibility, wide range of operating capabilities, comprehensive set of profile and fluctuation diagnostics (Fig. 4), and the ability to match all ITER non-dimensional parameters except size. DIII-D will greatly advance the understanding of rotation in plasmas and momentum transport, making use of rotational profile measurements, variable momentum input with co plus counter NBI, and a range of momentum drag with controlled non-axisymmetric perturbations. Transport models developed and validated on DIII-D will form the cornerstone of integrated modeling that will greatly increase the scientific benefit of ITER discharges.

Stability Science

The DIII-D Program is the recognized world leader in MHD stability research (T2,T6). The seminal work on the dependence of beta on the normalized current and plasma shape, the current profile and pressure profile; and the ongoing research on active MHD stabilization is unparalleled. World leading RWM stability control research on DIII-D is possible as a consequence of unique internal non-axisymmetric feedback coils, high bandwidth actuators, and variable rotation control with co plus counter NBI (Fig. 5). DIII-D is the recognized world leader in active control of NTMs with ECCD and will quantify requirements for ITER. This work will take advantage of the modulation capability of the EC gyrotrons, variable width of driven current afforded by multiple gyrotrons, steerable antennas, and rotation control provided by the co plus counter NBI. Unique high spatial and time resolution measurements of the both ion and electron temperature, wide variability in plasma shape, control of the current profile, and control and measurement of the fast particle content make DIII-D the best tokamak in the world to advance the basic physics understanding of the sawtooth instability. DIII-D is an established world leader in disruption mitigation research, owing to the high tolerance to disruptions, massive gas injection capability, and comprehensive diagnostics.

Heating and Current Drive

The DIII-D Program is the world leader in localized measurements of driven current and the validation of current drive models for ECCD and FWCD, as a consequence of excellent EC and FW current drive tools and unique reliable, and high resolution MSE system (Tables 2 and 5) (T11,A6). DIII-D will validate ECCD requirements for NTM stabilization in ITER, FW absorption on fast ions, and FWCD models. Co and counter beams, rf heating, and detailed measurement of internal magnetic field structure provide unique capability to validate bootstrap current models over a wide range of plasma conditions and gradient lengths.

Pedestal Research

The DIII-D facility is ideally suited for pedestal research because of the extensive set of edge profile and turbulence diagnostics, the broad range of plasma shapes available, and the broad range of edge parameters (T10,A2). Research on DIII-D established why the H-mode pedestal forms: turbulence suppression by sheared E×B flows. The trigger mechanism for the L-H transition is being pursued using DIII-D's extensive set of edge turbulence and profile diagnostics, especially important here are unique measurements of the radial electric field and edge current density. The DIII-D facility has played the leadership role in developing the peeling – ballooning model as the limit to the pedestal pressure in H-mode plasmas. The DIII-D facility provides a unique opportunity to develop ELM control with resonant magnetic perturbations. Steady-state ELM-free operation has been demonstrated using both resonant perturbations and using QH-mode (counter-injection); such operation can significantly reduce the erosion and increase the lifetime of ITER divertor components (Fig. 6).

Boundary Research

The DIII-D facility provides a unique opportunity to investigate the use of carbon divertor material in ITER (the baseline design), by measuring the source of carbon, the flow in the boundary region, the location of co-deposited hydrogenic species; and by developing carbon removal techniques (T10,A11). The development of boundary codes validated using DIII-D experiments will provide the basis for a highly radiative divertor solution for ITER and beyond. Detailed comparison of boundary codes with experiment is made possible by the extensive set of divertor and boundary diagnostics, especially the divertor Thomson scattering system (Fig. 7), which is unique in the world. Unique DiMES and MiMES systems will advance the understanding of erosion and sources of impurities.

Energetic Particle Research

DIII-D is uniquely qualified to evaluate fast ion instabilities and validate models for key ITER relevant Alfvén eigenmode physics (T12,A8). Worldclass energetic particle research on DIII-D is made possible by an extensive set of fluctuation diagnostics to measure the frequency spectrum and internal structure of the instabilities (Fig. 8); excellent profile measurements — especially the current density profile, and the new unique capability to measure the fast ion profile.

The study of a wide range of such instabilities is facilitated by the extreme shape flexibility to produce gaps in the continuum due to toroidicity, elongation, and triangularity, capability to produce and control a wide range of current profiles, an impressive range of control over the fast particle distribution with co plus counter NBI and FW heating, an extensive range of $V_{fast}/V_{Alfvén}$ made possible by the flexible operation over a wide range of B_T, and flexible plasma heating and density control to vary β_{fast}.

7. FACILITY

The DIII-D facility provides unmatched capabilities both in the US and the world to carry out front line research in basic fusion science, advanced steady-state operation, and key ITER-specific issues. World leading research is enabled by the best and most comprehensive diagnostic set among world tokamaks, the most flexible tokamak and field shaping system, a highly versatile heating and current drive system, unique density control capability, and a world leading digital plasma control system. See Table 5 for details.

Diagnostics

DIII-D has the best set of physics measurements of any tokamak, consisting of more than 50 scientific instruments, built and operated by staff from many different institutions. See Table 6 for details. The DIII-D diagnostics set covers all areas of fusion research and includes extensive divertor and edge measurement capability, plasma core profile measurements of density, temperature, rotation and internal magnetic field, particles and impurities, magnetic structure and an unparalleled suite of fluctuation diagnostics. The spatial and temporal resolution of many of the diagnostic systems are unparalleled, and a number are unique to DIII-D (lithium beam polarimetry, radial electric field from MSE and CER, BES, divertor Thomson, material sample probes in the divertor and midplane, fast ion profile from D_α spectroscopy, etc.). The accuracy and quality of the measurements on DIII-D has opened a new era of precise real time control of plasma parameters.

The excellence of the diagnostics suite is the key enabler for the outstanding research on DIII-D and attracts fusion experts around the world to participate in DIII-D experiments. The excellence of this diagnostic system results from a long-standing national and international collaborative effort between universities, national laboratories, and industries. Universities have the primary responsibility for all the fluctuation diagnostics, and this diagnostic set covering turbulence from $0.1 < k < 40\ \mathrm{cm}^{-1}$ provides DIII-D with a unique opportunity in transport research over the next 5 to 10 years. The set of boundary/divertor region diagnostics is the best in the world. The world class capability to measure the internal magnetic field structure (MSE) with high resolution and reliability provides unique opportunities in current drive research and *AT* steady state research.

Tokamak Device and Coil Sets

The hallmark of the DIII-D device is the experimental breadth provided by plasma shaping system flexibility. Eighteen independently controllable poloidal field coils located inside the TF coils and close to the plasma and vessel, an OH system that is magnetically decoupled from the shaping coils, and full graphite tile coverage inside, permits almost any shape to be produced for experiments. Systematic scans of the full set of shape parameters (elongation, triangularity, squareness, aspect ratio, plasma-wall gaps, X-point radial and vertical position, minor radius, indentation) are easily realized (Fig. 9). Double-null and single-null divertors are both routinely used. This flexible shaping capability provides high leverage in almost all areas of scientific investigation and enables DIII-D to precisely match the shape of almost any tokamak, making DIII-D the partner of choice in comparison experiments with other devices. Tables 7 and 8 show the ranges of key machine and physics parameters.

DIII-D is a world leader in the use of non-axisymmetric coils on the tokamak to open up critical new research areas in locked modes and error fields, active MHD feedback stabilization, plasma rotation, and ELM stabilization using resonant magnetic perturbations (RMP). The two coils sets are a six-element external correction coil set at the midplane and a 12 element internal coil set (6 above and 6 below the midplane). The internal coil set, coupled with a high bandwidth feedback system, and a flexible patch panel enables active RWM stabilization in low rotation plasmas, and complete ELM suppression with RMPs, both unique capabilities of the DIII-D facility.

DIII-D vessel conditioning allows very reliable, reproducible plasma operation, resilient to plasma disruptions. The first wall in DIII-D is all graphite. The vessel is bakeable to 350ºC, boronized periodically, and He glow conditioning is run between discharges. Routine low Z_{eff} discharges are reliably produced over a range of toroidal field, $0.5 < B_T < 2.2$. Following disruptions, the next discharge runs with only a little extra He glow between discharges.

Heating, Current Drive, and Fueling Systems

The highly versatile DIII-D heating and current drive systems provide unique capabilities to meet the DIII-D research objectives. The systems consist of seven NBI ion sources (co plus counter), six (1 MW) 10 s gyrotrons for ECH and ECCD, and 3 MW FW system (being upgraded to 6 MW). The EC system is the key off-axis current drive tool essential in the *AT* research and active stabilization of NTMs. The FW system provides axial q control for *AT* research and electron heating for transport and improved current drive. The unique co plus counter beam combination facilitate important new physics studies including a range of momentum input for transport, RWM stability, and QH-mode investigations. The magnitude of the auxiliary heating power is sufficient to permit reaching the beta limit in almost all regimes. Its unique in-vessel cryopumps make DIII-D the only tokamak that can control the density in H-mode plasmas in both DND and SND configurations. Both gas injection fueling and fueling by pellets (low field side, top, high field side) are used.

Advanced Integrated Plasma Control

The advanced, high speed digital plasma control system is a state of the art system that provides highly accurate control of the plasma shape and 0-D parameters, such as n_e, I_p, and beta and performs feedback control of non-axisymmetric phenomena (RWM and NTM). Profile control [e.g. q profile, $T_e(r)$] is becoming increasingly routine. Real time equilibrium reconstruction, including the internal field pitch measurements, allows the precise shape control and q-profile control. This highly flexible, high performance system has now been exported to other devices as their chosen control system (NSTX, KSTAR, EAST, MAST). The DIII-D PCS is fully integrated with a comprehensive set of design and simulation tools that enable it to provide control levels simulations of the plasma using actual hardware and software. The key strength is the predictive capability of this fully integrated system so that it can be used to develop the control capability for future machines and ITER.

8. QUESTION 1: What are the unique and complementary characteristics of each of the major US fusion facilities?

The DIII-D facility is characterized by unparalleled flexibility of operation, a wide range of plasma control tools, some of which are unique, and set of diagnostics that is unrivaled. These combine to make DIII-D a facility that is uniquely suited to develop the knowledge base and control tools both for the ITER baseline scenario and for advanced operating scenarios in ITER and other future burning plasmas, and to advance the underlying fusion science applicable to any magnetic fusion device. In many areas, the unique scientific capabilities of DIII-D are enhanced by the complementary nature of the other US fusion facilities, leading to scientific results beyond what could be achieved in any single facility. See Tables 3, 4, 5, and 6 for more specifics on the uniqueness of the DIII-D Program.

Opportunity for High Fusion Gain, Steady-State Physics Basis. Many of the key physics elements and key control tools for true steady-state are in hand, including high beta operation, high confinement, RWM stabilization, NTM suppression and avoidance, disruption mitigation, on- and off-axis current profile control, density control, and integrated digital plasma control. Projections from advanced operating modes (advanced H-mode or hybrid and steady-state *AT*) already achieved on DIII-D indicate the potential of ITER operation with full output power: (1) in steady-state with energy gain Q=5, (2) for longer than one hour with Q=10, and (3) for shorter pulses with very high fusion gain, $Q \gtrsim 40$. DIII-D has been at the forefront in the development of these advanced scenarios and continued participation by DIII-D in these key areas is essential for maintaining the US leadership role.

Turbulent Transport Science. DIII-D's capability to carry out detailed scientific studies of turbulent transport processes is unmatched in the world. The complete set of temperature, density, rotation, and current density profile diagnostics with high resolution from the core through the edge pedestal, and the comprehensive turbulence diagnostics (covering wave-numbers from 0.1 to 40 cm^{-1}), are critical for such studies, and the combined diagnostic set is a unique feature of DIII-D. In addition, DIII-D's operational flexibility over a wide range of available plasma parameters, operating regimes, and heating schemes allows transport models to be tested in widely varying conditions.

These capabilities are complemented by C-Mod and NSTX, which further extend the parameter space of toroidal field, density, aspect ratio, and trapped particle fraction over which transport can be studied.

High Beta Operation. Strong discharge shaping, a close-fitting wall, and the high power density afforded by DIII-D's heating systems, combine to create a unique opportunity for study of transport, stability, and current drive in plasmas up to the beta limit in almost all regimes. DIII-D's achievement of 12.5% toroidal beta remains a record for conventional aspect ratio tokamaks. Advanced-scenario experiments are routinely carried out in plasmas with ITER-like aspect ratio, toroidal beta of 4% to 6%, and normalized beta of 4 or greater, values that lie in the recently developed operational regime above the free-boundary stability limit. These experiments will lead the way to high fusion power, steady-state operation in ITER and fusion devices beyond ITER.

DIII-D's high beta capability is complemented by that of NSTX, allowing validation of MHD stability theory in plasmas with varying aspect ratio, and study of the beta-dependence of transport over an even wider range of beta.

Current Drive and Current Profile Control. DIII-D's capability for noninductive current drive is unique within the U.S. fusion program. Electron cyclotron current drive is used routinely for current profile control in advanced tokamak experiments and for neoclassical tearing mode stabilization; feedback control of the q-profile using ECCD has been demonstrated. These developments are directly applicable to ITER, where electron cyclotron current drive is also planned. DIII-D's fast wave system provides additional on-axis current drive or electron heating to enhance the ECCD. Mixed co/counter neutral beam current drive provides additional flexibility. These capabilities will continue to be key tools in development of steady-state advanced tokamak scenarios.

DIII-D's current drive capability will be complemented by the use of lower hybrid current drive in C-Mod and high-harmonic fast wave in NSTX, allowing tests of current drive physics and technology with a wide range of methods.

MHD Stability Control. DIII-D's set of tools for control of MHD instabilities is unique. These include external non-axisymmetric coils to avoid error field-driven locked modes and to help maintain plasma rotation for resistive wall mode stabilization, internal non-axisymmetric coils with fast power supplies for direct feedback control of RWMs, and non-axisymmetric coil configurations that create stochastic fields for suppression of edge-driven instabilities, all pioneered by DIII-D. The tools also include ECCD for control of neoclassical tearing modes, and a digital control system capable of the precise control needed for NTM stabilization and the fast response required for RWM stabilization.

The external non-axisymmetric coils, in particular, are well complemented by similar sets in NSTX and C-Mod, allowing cross-machine studies of the scaling of error field effects with plasma size, magnetic field, and aspect ratio. Cross-machine experiments are also in progress with NSTX on rotational stabilization of RWMs and resonant field amplification at high beta.

Plasma Rotation Control. DIII-D's set of tools for control of plasma rotation is unique. Mixed co- and counter neutral beam injection allows study of the effects of plasma rotation and rotational shear on transport and MHD stability, in high beta plasmas with plasma rotation rates that vary continuously from zero to well above the critical value for stabilization of resistive wall modes. Mixed beam injection also allows control of the radial electric field shear over a wide range, for study of turbulence suppression and the physics of internal transport barriers. Non-axisymmetric coils provide the option of selective braking of rotation by the application of resonant or non-resonant magnetic perturbations.

These capabilities are complemented by C-Mod (rf heating with no external momentum source) and NSTX (unidirectional beam injection with very strong rotation), and allow DIII-D experiments to match both conditions. DIII-D can also investigate the spontaneous generation of rotation in low-torque rf-heated plasmas that has been observed in C-Mod, and compare low-torque rf heating with low-torque neutral beam heating.

Fast Ion Distribution Control and Measurement. With co- and counter beam injection and fast wave heating, fast ion populations can be generated that are primarily co-going, counter-going, or perpendicular, or a mixture that is more nearly isotropic. This capability represents a unique opportunity to investigate the effects of fast ion velocity distribution in driving Alfvén eigenmodes and other instabilities; these studies can be done with sub-Alfvénic or super-Alfvénic ions through DIII-D's flexibility to vary the toroidal field. The new D_α spectroscopy diagnostic provides a unique capability to measure the fast ion distribution, and the effects of instabilities on fast ion transport.

These capabilities are complemented by C-Mod (rf heating with perpendicular ion velocity) and NSTX (beam injection with strongly super-Alfvénic, co-going ions), broadening the range of available fast ion distributions. The diagnostics are also complementary, with the three machines bringing a host of fluctuation diagnostics to bear on measurements of the frequency and mode structure of fast ion-driven modes, including C-Mod's active MHD spectroscopy in the Alfvén mode frequency range.

Pedestal and Boundary Plasma Control. Pumping of both upper and lower divertors in DIII-D, with DIII-D's shaping flexibility, creates a unique opportunity for density control and boundary plasma science in strongly shaped, high performance H-mode plasmas. Density control is crucial for advanced tokamak experiments with non-inductive current drive, and DIII-D can exercise this control in high-triangularity double-null or single-null plasmas. A wide range of scrape-off layer flow experiments are enabled by the flexibility to pump one or both divertors and to operate balanced double-null plasmas or configurations where one X-point or the other is dominant. Density control also allows study of edge transport, stability, and pedestal width in both collisional and collisionless regimes. Edge-driven instabilities are sensitive to subtle changes in the triangularity or squareness of the plasma shape, and the DIII-D shape control system has a precision of a few mm. Magnetic perturbations applied with the non-axisymmetric coils are another tool for control of the plasma boundary and pedestal.

These capabilities are complemented by C-Mod, where high triangularity double-null plasmas have power handling and particle pumping at opposite divertors, and where density control without core particle fuelling can be compared to DIII-D's case with neutral beam fueling.

Flexibility in Shape and Operating Parameters. The DIII-D coil set allows an extremely wide variation of discharge shape, while the heating and current drive systems and pumped divertors give access to a wide range of temperature, density, beta, and collisionality. The digital control system makes it possible to achieve the various shapes and operating parameters accurately and reproducibly. These capabilities combine to enable studies of transport and stability in widely varying plasma conditions.

DIII-D's flexibility of operation is also uniquely suited for comparative studies, and thus complements the other U.S. facilities. DIII-D can match the shape and other key dimensionless parameters of virtually any other tokamak, enabling scaling studies of transport coefficients, stability limits, or edge pedestal parameters over a wider range of plasma size, aspect ratio, and magnetic field than could be achieved in any one machine.

Wide Range of Core and Pedestal Operating Regimes. DIII-D's unique flexibility enables physics studies in a wide range of conventional and enhanced operating regimes: core conditions include conventional sawtoothing plasmas, weak central shear, strongly negative central shear, high internal inductance, and internal transport barriers, while edge pedestal conditions include L-mode, H-mode, VH-mode, QH-mode, and RI-mode. Many of these regimes were first discovered in DIII-D, and subsequently "exported" to other tokamaks for study under different operating conditions. DIII-D's operational flexibility also allows it to readily "import" new operating regimes or other phenomena observed elsewhere, for investigation with a comprehensive set of diagnostic measurements.

Best Diagnostics System in the World. Scientific studies are only as good as the measurements, and DIII-D's measurement capability is unmatched. A complete set of profile diagnostics provides the density, electron temperature, ion temperature, and impurity density profiles that are the foundation of any study of transport, stability, or current drive. DIII-D's ability to measure the current density profile with high resolution from the plasma core to the separatrix, using motional Stark effect and lithium beam polarimetry diagnostics, is unique; this measurement is essential to understanding the stability of the core and the edge pedestal. An unrivaled set of fluctuation diagnostics covers the full range of transport-relevant wavenumbers (~0 to 40 cm^{-1}), suitable to investigate ion temperature gradient modes, trapped electron modes, and electron temperature gradient modes. The divertor diagnostics set is the best in the world, featuring a unique Thomson scattering system that yields two-dimensional electron temperature and density data.

DIII-D's set of comprehensive, high-resolution diagnostics is a key element of the U.S. fusion program. With it, the broader cross-machine comparisons are complemented by in-depth studies in a single machine. Both are essential to maximizing the scientific output of the program.

9. QUESTION 2: How do the characteristics of each of the three US fusion facilities make the US toroidal research program unique as a whole in the international program?

DIII-D's capability for a complete, integrated approach to fusion science and improvement of the tokamak concept makes this facility unique within the international program. The comprehensive plasma control tools and diagnostics available on DIII-D allow experiments to investigate the full range of plasma science, including turbulent transport, plasma stability, the physics of heating and current drive, and plasma-wall interactions. They also make it possible for DIII-D to address issues of scenario development and optimization for ITER in a truly integrated way, combining the necessary elements of core and edge plasma control within a single discharge. While some of the individual features are available at other facilities, the combination of capabilities represented in the DIII-D facility is unmatched.

Complete Plasma Diagnostics. No other facility equals DIII-D's capability for complete, high-resolution diagnostic measurements (Table 6): temperature and density profiles through the entire plasma radius, scrapeoff layer, and divertor; rotation, radial electric field, and current density profiles throughout the core and edge pedestal; and plasma turbulence across the entire plasma radius, in the full range of wavenumbers relevant to ion and electron heat transport. These combined capabilities have established DIII-D's position as a world leader in plasma science, particularly in the areas of core transport, current drive, MHD stability, and edge pedestal and boundary physics.

Complete Plasma Control Tools. DIII-D's range of plasma control tools is unmatched in the world. Multiple complementary tools (Table 5) are available for control of plasma pressure, non-inductive current sustainment and current density control, control of plasma rotation, indirect and direct control of plasma stability, control of the edge pedestal pressure, and control of the plasma boundary and plasma-wall interactions. These combined capabilities have established DIII-D's position as a world leader in development of high performance, steady state operating regimes for ITER and beyond.

Integrated Plasma Control System. DIII-D's integrated approach to plasma control is unsurpassed, and can provide solutions for control of advanced scenarios in ITER. The diagnostics and plasma control tools are unified in a fast, flexible digital plasma control system, based on real-time solution of the Grad-Shafranov equilibrium equation using motional Stark effect data. Reliable control algorithms are developed through extensive simulations with realistic models of the plasma, tokamak, and control system. The system is routinely used in DIII-D experiments for control of discharge shape, beta, and temperature profile, precise positioning of divertor strike points at the pumps, precise positioning of electron cyclotron current drive using real-time q-profile reconstruction, high-speed feedback control of resistive wall modes, and real-time control of the q-profile. DIII-D pioneered the use of digital plasma control, and the DIII-D control system has been chosen by most new tokamaks worldwide, including NSTX, MAST, KSTAR, and EAST.

Integration of Advanced Scenarios. The DIII-D facility is unique in allowing the integration of all elements of optimized tokamak scenarios. Such integration includes control of q-profiles that are consistent with good energy confinement and high stability limits, pressure profiles that are consistent with the desired bootstrap current profile, a plasma boundary and divertor solution that is compatible with high-performance, steady state operation, and external stability control as required for these conditions. Present high-performance experiments routinely exploit the complete diagnostic coverage, and include elements of integrated control and operation such as:

– feedback control of electron temperature to regulate the current profile evolution
– feedback control of beta to allow operation close to stability limits
– feedback-controlled error field correction and/or direct feedback stabilization of resistive wall modes to allow sustained operation above the free-boundary stability limit
– q-profile modification with electron cyclotron current drive to regulate internal transport barrier formation and obtain quasi-stationary, high beta discharges with little or no inductive current
– electron cyclotron current drive for suppression of neoclassical tearing modes
– control of edge density through divertor pumping to allow penetration of electron cyclotron waves, and to regulate the density during quasi-stationary discharges
– ELM suppression using resonant magnetic perturbations or counter neutral beam injection.

The success of these experiments and the high scientific value of the results is ensured by self-consistent modeling of heating and current drive, transport of particles, energy, and magnetic flux, MHD stability, and the plasma boundary and divertor, which can be validated by comparison with the extensive diagnostic measurements. DIII-D's unique capability to integrate the various plasma control elements in high-beta, high-confinement discharges, along with a full range of diagnostic measurements, is crucial for developing the scientific basis for optimized tokamak plasmas, and for ensuring the maximum scientific output from ITER.

New Scientific Challenge: Self-Organized Plasmas. DIII-D's integrated diagnostics and plasma control tools are a unique opportunity to address the new scientific challenge of self-driven plasmas. Steady-state burning plasmas will be dominated by self-heating and self-generated bootstrap current, and therefore will be less susceptible to external control. Furthermore, the strong coupling of the pressure and current density profiles through the bootstrap current and turbulent transport creates a complex, nonlinear, self-organized system; its understanding will require integration of transport and stability science with plasma control. DIII-D experiments in bootstrap current-dominated plasmas, in conjunction with integrated modeling, can begin to address the challenging issues of predicting and controlling the self-consistent solutions of such a system, crucial for the long-range development of optimized steady-state fusion scenarios.

10. QUESTION 3: *How well do we cooperate with the international community in coordinating research on our major facilities and how have we exploited the special features of US facilities in contributing to international fusion research, in general, and to the ITER design specifically?*

DIII-D's innovative research has had a profound impact on the worldwide fusion program, and on the design of ITER and its planned operating scenarios. International cooperation has long been a hallmark of the DIII-D program, and DIII-D is closely connected to the international fusion research community through numerous active collaborations. Joint experiments have been carried out between DIII-D and the major international tokamaks in topics including basic transport physics, control of MHD instabilities, pedestal physics, ELM control, error fields, and development of high performance operating scenarios, with a scientific value far beyond what can be achieved by any single facility (Appendix A). See Table 4 for more specifics on DIII-D support of ITER.

International Cooperation. The DIII-D facility has a strong, long-standing, two-way involvement with the international fusion community. Ongoing collaborative agreements exist with all of the major international tokamak facilities, including most recently KSTAR and EAST. The versatility of the DIII-D facility attracts collaborators and the program welcomes them. In 2004-2005, over 10% of the experimental proposals originated outside the US, and more than 30 foreign scientists were active participants in DIII-D experiments – some using new internet video technology for remote participation. Currently many collaborative experiments are coordinated through the International Tokamak Physics Activity (ITPA), where the DIII-D team is represented by 35 members, including 3 chairs of international working groups and 8 leaders of US working groups (Appendix A).

Collaborations and Joint Experiments. The scientific value of DIII-D experiments, and the impact on the international fusion program, are enhanced by international participation and joint experiments. Recently, for example, joint experiments in well- matched plasmas with other tokamaks including JET, JT-60U, and ASDEX Upgrade have shown that the H-mode pedestal width scales with plasma size, the neoclassical tearing mode threshold scales inversely with plasma size, that resonant field amplification in high beta plasmas is independent of the plasma size, and that energy transport is independent of beta. In addition, the development of improved operating scenarios such as the "hybrid" mode, quiescent H-mode, and ELM suppression by resonant magnetic perturbations have been areas of intense international interest and collaboration, for their potential benefit to other present tokamaks as well as to ITER. Earlier, the "optimized shear" regime of enhanced performance was developed at DIII-D and then exported to other facilities including JET.

ITER Design. DIII-D research results have had a major impact on the design of ITER. In the recent re-design of ITER, the cross-sectional shape was heavily influenced by DIII-D research showing the importance of triangularity on both pedestal height and overall plasma performance. DIII-D experiments demonstrating the ability to suppress m=3/n=2 and m=2/n=1 neoclassical tearing modes via electron cyclotron current drive influenced the decision to increase the EC power on ITER. DIII-D experiments demonstrating feedback control of resistive wall modes have motivated serious consideration of the addition of internal non-axisymmetric control coils in ITER. The recent development of ELM suppression in DIII-D using resonant magnetic perturbations has been an area of intense international interest and collaboration, and is a promising approach for ELM control in ITER with the addition of the proper coils.

ITER Operating Scenarios. DIII-D research results have also had a major impact on planned operating scenarios for ITER, resulting in part from their close match in discharge shape, aspect ratio, beta, collisionality, and other dimensionless parameters, as well as the use of electron cyclotron current drive, and in part from the innovative research that DIII-D has produced. DIII-D has been the pioneer and worldwide leader in modeling and experimental development of high beta, high bootstrap fraction, steady state scenarios, which have become the leading candidate for advanced operation in ITER. In addition, DIII-D has recently taken the lead in the development of so-called hybrid operation for high performance, stationary operation, which has the potential to replace the conventional, ELMing H-mode regime as the ITER baseline scenario. Projections from advanced operating modes (advanced H-mode or hybrid and steady-state *AT*) already achieved on DIII-D indicate the potential of ITER operation with full output power: (1) in steady-state with energy gain Q=5, (2) for longer than one hour with Q=10, and (3) for shorter pulses with very high fusion gain, $Q \gtrsim 40$. The capability to test new operating regimes for ITER in a well-matched smaller facility such as DIII-D is a crucial element of an efficient and dynamic international research program.

11. QUESTION 4a: *How do these three facilities contribute to fusion science and the vitality of the US fusion program?*

DIII-D is the premier fusion facility in the world for plasma science and fusion energy science research. DIII-D's unique capabilities, and its adaptability to pursue new ideas and high priority scientific issues as they arise, make it a powerful and versatile instrument for fusion science research. The advances in fundamental plasma science and the innovations in high-performance plasma operation and plasma control that emerge from this facility have had and will continue to have a major impact on worldwide fusion research. The quality and the breadth of scientific research that is possible in DIII-D have attracted a large team of national and international users, and place DIII-D as a leading fusion facility within the world fusion program.

The DIII-D facility is essential to a vital US fusion program, and establishes the US as a leading member of the international fusion community. Its unmatched flexibility of operation, comprehensive set of tools for plasma control, superb diagnostic set for detailed scientific investigations, and well-established national and international working teams offers a unique opportunity for contributions to plasma and fusion energy science in a broad range of topics:

Advanced Scenarios. The DIII-D team pioneered the "advanced tokamak" concept, and the facility is unique in the world for high bootstrap fraction, high beta discharges above the free boundary stability limit, with precise profile measurements and current profile control, strong double-null shaping, and active stability control. This research in DIII-D is critical for steady-state operation for ITER, and development of the basis for a compact, high fluence Component Test Facility. In addition, DIII-D's contribution to "hybrid scenario" research is critical to the world effort, and will enhance the scientific benefit from ITER's baseline operation.

Confinement and Transport. Understanding of the physics of turbulent transport and the mechanisms that regulate it, such as zonal flows, is a major achievement in fundamental science that is within our grasp. DIII-D's complete sets of profile and turbulence diagnostics, uniquely flexible heating tools, and wide range of operating regimes will be critical in testing transport models against experiment. These tools, with the new capability for rotation control with mixed co/counter beam injection, will also allow tests of models for momentum transport and internal transport barrier formation. Validated models are needed for prediction of ITER performance.

MHD Stability. Non-ideal effects on stability, including resistivity, bootstrap current, and plasma rotation, and active control of MHD stability, represent the present frontier of stability science. DIII-D's control tools, including non-axisymmetric control coils and variable momentum input with mixed co/counter beam injection, plus the comprehensive diagnostic set, represent a unique opportunity to advance the understanding of MHD stability physics and to develop control of NTMs and RWMs for conventional and advanced scenarios in ITER.

Current Drive. DIII-D's range of current drive sources, including electron cyclotron, fast wave, and neutral beam current drive (co and counter) allow validation of current drive theories in a broad range of operating conditions, including the synergistic effects of the different sources. The unique capabilities of six steerable ECCD launchers to vary the current drive width, and controlled plasma rotation at low, ITER-relevant rates, will allow crucial tests to accurately define the benefits and requirements for modulated NTM stabilization in ITER.

H-mode Pedestal. DIII-D has a unique capability to measure detailed profiles of the edge radial electric field, current density, electron and ion temperature, density, and turbulence, for investigation of the physics that sets the pedestal width and height. Validation of predictive models for the pedestal is crucial for projection of fusion performance in ITER, since the pedestal temperature is the largest source of uncertainty. DIII-D also has unique capabilities to investigate the physics of pedestal stability, including variable co/counter beam injection in ELM-free QH-mode discharges, and resonant magnetic perturbations for ELM suppression. Scientific understanding of pedestal stability is crucial to solving the divertor lifetime limit due to erosion by ELMs.

Boundary and Divertor. DIII-D has unique capabilities for detailed boundary and divertor measurements in both open and closed divertors with axisymmetric gas sources, which facilitate progress in modeling and understanding of radiative divertors. This understanding could yield a solution to divertor erosion in ITER. DIII-D also has the unique capability to measure the detailed flow and deposition of ^{13}C in a range of operating regimes, allowing validation of models for tritium deposition and retention. These measurements are crucial information for choosing divertor materials for ITER.

Energetic Particles. DIII-D's comprehensive turbulence diagnostics and operational flexibility allow detailed study of the full range of fast ion-driven Alfvén eigenmodes and energetic particle modes, including core-localized Alfvén cascades, a new focus of scientific study for ITER, as well as TAE, EAE, BAE, and CAE modes. The new D_α diagnostic is a unique opportunity for direct measurements to validate models of the driving term for these instabilities, and of the effects of these and other modes on fast ion transport.

Versatile Research Tool. The DIII-D facility (Table 5) is a highly flexible research tool, capable of addressing a wide range of physics issues (Tables 2, and 3). It is readily modified to address new scientific issues as they arise: examples are the current addition of double-null divertor pumping and counter-beam injection, and the past addition of internal non-axisymmetric coils, divertor cryopumping, and change of first wall material. DIII-D has the most complete set of diagnostics in the world (Table 6), promoting excellent science over a wide range of topics. The flexible heating and current drive systems produce localized and modulated heat sources, widely varying momentum input, and a large range of current profiles. The range of discharge shapes available through a uniquely flexible shaping system makes DIII-D a key element of dimensionally similar experiments worldwide.

Fusion-Relevant Parameters. Plasmas in DIII-D are in the relevant range for fusion energy science research (Table 8). Fusion relevant temperatures ($T_i > 25$ keV, $T_e > 15$ keV) are attained, and DIII-D readily reproduces ITER's non-dimensional parameters (ν_*, β, shape, q, aspect ratio) except for ρ_*: the DIII-D plasma is a 1/4 scale ITER plasma. Balanced beam injection allows low, ITER-relevant rotation. A wide range of critical fusion science issues can thus be addressed with ITER-like parameters, in operating regimes that include L-mode, H-mode, VH-mode, QH-mode, and stochastic edge, with or without internal transport barriers. Divertor cryopumping, gas fueling and pellet fueling allow a wide range of edge pedestal collisionality to evaluate critical issues of the H-mode transport barrier and ELMs.

Breadth of Scientific Research. Research in DIII-D embraces a wide range of fusion science, including fundamental physics issues of turbulent transport, resistive MHD stability, current drive, transport barriers, and plasma-wall interaction, as well as the science underlying advanced plasma control and operational regimes of improved performance. The research covers almost the entire range of key questions and priorities indicated in the recent FESAC priorities panel (Table 3). The breadth of topics addressed is illustrated in Tables 2, 3 and 4. At the yearly Research Opportunities Forum, the number of good proposals submitted in each research area exceeds the number of experiments the program has run time to carry out by up to a factor of 5.

Theory-Experiment Interaction. A close coupling between theory and experiment has been a hallmark of the DIII-D program, and is crucial for advancing fundamental plasma science. An outstanding diagnostic set with unique physics measurements (e.g., edge current density), precise plasma control, innovative experiments, and strong multi-institutional teams have facilitated the detailed comparison of DIII-D results with theory, and placed DIII-D as a world leader in basic physics understanding. Examples include turbulence suppression by sheared E×B flow, consistent with gyrokinetic calculations, and the recent identification of ELMs as moderate-n peeling/ballooning modes, in good agreement with the ELITE and BOUT code models.

Innovation. DIII-D has pioneered many improvements of the tokamak concept. Two present areas of innovation are the stabilization of resistive wall modes and the use of resonant magnetic perturbations to control ELMs. In addition, the flexible facility continues to lead to discovery of new physics and new operating regimes, such as the QH and QDB modes of stationary ELM-free operation. Previous examples include strong shaping to increase beta, pedestal and ELM control with detailed shaping, open divertors, divertor biasing for radial electric field control, feedback-controlled error field correction, barriers with ion neoclassical transport, and VH-mode. The current facility modifications for mixed co/counter beam injection and pumping of double-null divertors will open new areas of scientific study. Such innovations are a crucial element of a vital, exciting research program.

Impact on World Fusion Program. Many innovative experiments on DIII-D have shaped the direction of the world fusion program. These include error field correction with non-axisymmetric coils, the impact of triangularity on the pedestal, negative central shear operation and transport barrier formation, resistive wall mode stabilization, and quiescent, ELM-free operation. Experiments on DIII-D have shifted the ITER design toward stronger shaping, and there is a growing interest in non-axisymmetric coils for RWM stabilization and ELM control. The strong national and international participation in DIII-D experiments (Appendix A) highlights the relevance and impact of the research. The selection of 13 papers on DIII-D experiments for oral presentation at the 2004 IAEA conference, comparable to JET and JT-60U, indicates its international recognition as a world leader in fusion research.

Centerpiece for National and International Research. The DIII-D scientific team is one the greatest strengths of the program, and is a corollary of DIII-D's major role in the world fusion program. The engagement of so many excellent scientists is largely a consequence of the outstanding facility: the versatile tokamak systems and detailed physics measurements allow excellent experiments on a wide range of topics. In addition, the program management is committed to maintaining a world-class research facility and a program that welcomes the involvement of the best scientists nationally and worldwide. Of the 71 APS invited plus IAEA oral presentations from the DIII-D program in the last five years, 35 were delivered by presenters from institutions other than GA.

Training Ground for Young Scientists. Training of fusion scientists for the future is an important role of the DIII-D facility. The broad scientific program and "hands-on" nature of the facility provide an opportunity for young scientists to carry out exciting and relevant work: at the 2004 IAEA Fusion Energy Conference, four of the DIII-D papers represented work by post-docs. The DIII-D program could easily support 20 more graduate students, provided funding were available. In 2005, nine students supported by National Undergraduate Fellowships will participate in research at DIII-D (Appendix A).

12. QUESTION 4b: ***What research opportunities would be lost by shutting down one of the major facilities?***

Loss of the DIII-D facility would result in loss of the US position of leadership in important areas of the world fusion program, most notably in the investigation of the fundamental scientific question of turbulent transport in magnetized plasmas, and in development of high beta, steady state fusion plasmas. With its unmatched flexibility of operation, comprehensive set of tools for plasma control, superlative diagnostic set for detailed scientific investigations, and well-established national and international working teams, DIII-D offers a unique opportunity to advance plasma and fusion energy science, resolve key issues for ITER, and develop steady-state high performance scenarios for ITER and beyond. Its loss would be a major setback for both the US and the world fusion programs.

Loss of U.S. Leadership in a Broad Range of Fusion Science Topics, among them the fundamental understanding of turbulent transport, the scientific basis for control of plasma instabilities, and the understanding of "self-organized" plasmas where most of the plasma current is self-generated.

Loss of the Capability to Develop High Beta, High Bootstrap Fraction Steady-State Scenarios. DIII-D is unique in the world for high bootstrap fraction, high beta discharges above the free boundary stability limit, with precise profile measurements and current profile control, and strong double-null discharge shaping. This loss would delay progress in developing steady state operation for ITER, and in developing the basis for a compact, high fluence Component Test Facility.

Loss of the Opportunity to Develop a Solid Scientific Basis for Hybrid or Advanced H-mode Operation for ITER. DIII-D's capabilities for a wide range of operating regimes with strong shaping, and control of MHD stability, current profile, edge pedestal, and boundary plasma, are unique. DIII-D's contribution in this area is critical to the world effort, and its loss would result in a loss of scientific benefit from ITER's baseline operation.

Loss of the Opportunity to Understand Turbulent Transport. DIII-D has the most complete sets of profile diagnostics and turbulence diagnostics, both critical in testing transport models against experiment. In addition, DIII-D's flexible set of heating tools and wide range of operating regimes are conducive to rigorous tests of transport models. Such a loss would delay development of transport models suitable for prediction of ITER performance, and would give up a major achievement in fundamental science that is within our grasp.

Loss of the Capability to Accurately Define Benefits of Modulated Current Drive for NTM Stabilization in ITER. DIII-D has the unique capabilities of six steerable ECCD launchers to vary the current drive width, and plasma rotation control to allow modulation at a relatively low rate. Lack of this crucial test will compromise the ability to predict requirements for stabilization of neoclassical tearing modes in ITER's baseline operation.

Loss of the Opportunity to Develop the Physics Basis for Resistive Wall Mode Stabilization. DIII-D is unique in the world in its ability to investigate both rotational stabilization and active feedback stabilization of high beta plasmas, using a wide range of rotation with mixed co/counter beam injection, and internal feedback coils with high bandwidth actuators. This loss would delay development of high beta, high bootstrap fraction steady-state scenarios for ITER.

Loss of the Opportunity to Understand Plasma Rotation and Momentum Transport. In addition to its superior measurement capability of both toroidal and poloidal rotation, DIII-D has the unique capability to vary the momentum input over a wide range, apply variable magnetic braking to the plasma, and evaluate spontaneous rotation in rf-heated discharges. Momentum transport is poorly understood, and this would be a major loss to fundamental plasma science.

Loss of the Opportunity to Develop a First-Principles Understanding of the H-mode Pedestal. DIII-D has a unique capability to measure detailed profiles of the edge radial electric field, current density, electron and ion temperature, density, and turbulence, for investigation of the physics that sets the pedestal width and height. This loss would limit our capability for performance projections in ITER and beyond, since the pedestal temperature is the largest source of uncertainty.

Loss of Capability to Develop Steady-State ELM-Free Operating Regimes for ITER. The QH-mode, discovered in DIII-D, is the only steady-state ELM-free operating regime reproduced on a range of tokamaks. DIII-D has the unique capability to investigate it with variable co/counter beam injection, over a wide range of collisionality and shape. This loss would delay progress on a potential solution to the divertor lifetime limit due to erosion by ELMs.

Loss of Capability to Develop and Understand ELM Solutions for ITER. DIII-D has demonstrated that ELMs can be suppressed with application of resonant magnetic perturbations, and DIII-D has the only set of coils suitable for this investigation worldwide. This loss would halt progress on a potential solution to the divertor lifetime limit due to erosion by ELMs.

Loss of Capability to Develop Dissipative Radiative Divertors for ITER and Beyond. DIII-D has unique capabilities for detailed boundary and divertor measurements in both open and closed divertors with axisymmetric gas sources, to facilitate progress in modeling and understanding of radiative divertors. This loss would delay progress on a potential solution to divertor erosion.

Loss of Opportunity to Understand Tritium Co-Deposition and Validate the Use of Carbon for ITER's Divertor. DIII-D has the unique capability to measure the detailed flow and deposition of ^{13}C in a range of operating regimes, allowing validation of models for tritium deposition and retention. This loss would significantly increase the uncertainty over choice of divertor materials for ITER.

Loss of Opportunity to Develop Physics Models of Fast Ion Instabilities. DIII-D's wide range of turbulence diagnostics and operational flexibility allow detailed study of the full range of Alfvén eigenmodes and energetic particle modes, including TAE, BAE, EAE, CAE, and core-localized Alfvén cascades. The last are a new focus of scientific study for ITER. This loss would delay the understanding of these modes and their potential impact on alpha particle transport.

13. CLOSING STATEMENT

The DIII-D facility offers a unique opportunity to advance plasma and fusion energy science, resolve key issues for ITER, and develop steady-state high performance scenarios for ITER and beyond. Its loss would result in loss of the US position of leadership in important areas of the world fusion program, most notably in the investigation of the fundamental scientific question of turbulent transport in magnetized plasmas, and in development of high beta, steady state fusion plasmas. With DIII-D heading its portfolio of facilities, the US fusion program will retain its position as the world leader in fundamental plasma science, and will continue to be the recognized world leader for innovative research that is improving the tokamak concept for ITER and devices beyond ITER.

Table 2. DIII-D Program Research Accomplishments

RESEARCH ACCOMPLISHMENTS	IMPACT	DIII-D UNIQUE CAPABILITY
ADVANCED TOKAMAK		
Steady State Advanced Tokamak: Uniquely achieved 100% noninductive current at high toroidal β and high confinement, and bootstrap current fractions up to 90% at higher q_{95} (lower toroidal beta)	Proved feasibility of steady state operation of burning plasmas with values of toroidal beta, confinement and bootstrap current well beyond standard H-mode. Providing basis for steady-state operation of ITER	High beta plasma shape; ECCD; cryopumps; external and internal coils; PCS
Hybrid Scenarios: Discovered, optimized, and extended range in q_{95} and density up to no-wall β limit	Hybrid will become new ITER baseline scenario (higher beta and confinement), displacing standard H-mode. Can run ITER one hour at full power	Cryopumps; high heat-flux divertor; high beta plasma shape; external coils; PCS
High Internal Inductance Mode: Proved no-wall β limit increases for high ℓ_i allowing simultaneous high β and high bootstrap current	Alternative advanced tokamak scenario consistent with efficient central current drive. DIII-D□s results motivated the consideration of internal control coils for ITER	Dynamic shape flexibility; FWCD
STABILITY		
Resistive Wall Modes (RWMs): First suppression of RWM's using internal coils and/or plasma rotation; discovered resonant field amplification of error fields; feedback control.	Improved tokamak concept by dramatically increasing accessible plasma beta above no wall limit. DIII-D□s results motivated the consideration of internal control coils for ITER	Internal and external coils; strong plasma rotation; high beta plasma shape
Neoclassical Tearing Modes (NTMs): First suppression of 2/1 NTM using ECCD (also 3/2 NTM); raised β during ECCD suppression using q location tracking; NTM threshold physics	Validated 2/1 NTM disruption avoidance for ITER using ECCD. Demonstrated plasma beta could be raised significantly above NTM threshold with good confinement during ECCD suppression	ECCD; PCS; MSE
High Toroidal β: Pioneered non-circular plasma shapes; achieved record values of β for tokamaks; proved operational limits are due to ideal kink mode stability	Elucidated role of ideal kink mode stability in disruptions. Discovered halo currents that set structural limits in ITER	High beta plasma shape; current profile diagnostics (MSE, Li beam)
Disruption Mitigation: Demonstrated efficacy of using impurity gas injection in mitigating disruptions	Eliminates catastrophic risk to ITER associated with jxB forces and runaway electrons	Pellets; massive gas puff; PCS; extensity array of diagnostics
Error Fields: Showed effect on locked mode threshold	Defined tolerance limit for error fields in ITER design	External coils
Triangularity: First to demonstrate benefit of high δ	Motivated increased triangularity of new ITER design	Flexible plasma shape; cryopumps
TRANSPORT		
E×B Velocity Shear Stabilization: Proved E×B shear is unifying cause of both edge and core transport barriers; discovered VH-mode; demonstrated ion transport close to neoclassical level	Robust method for suppressing turbulent transport. First example of improved confinement beyond H-mode [achieved Q_{DT} (equiv)=0.3 in VH-mode]. Proved ion transport could reach theoretical minimum value	Unique radial E field measurement. Comprehensive set of transport and fluctuation diagnostics; boronization; high triangularity plasma shape
Nondimensional Transport Scaling: Proved H-mode transport is independent of β and gyroBohm-like; developed comprehensive transport scaling with ν_*, q, k, T_e/T_i	Proved dimensionless plasma physics parameters govern turbulent transport. Demonstrated path to compact burning plasma device at high beta. Confirmed picture of drift wave turbulence	Flexible plasma shape; cryopumps; MSE; ECH and fast wave electron heating
Helium Ash Removal: First to show helium particle transport in H-mode is sufficiently fast for helium ash removal	Relieved concerns about Helium ash buildup quenching the D-T burn phase in ITER H-mode plasmas	Argon frosting of cryopump for active helium pumping
Internal Transport Barriers: Demonstrated ($\approx 30\ \tau_E$) enhanced core confinement duration limited only by hardware	ITB's are not inherently transient. Core transport barriers are possible in burning plasmas	External coils; cryopumps; long pulse operation
Heat Pinch: First demonstration of heat flowing up T_e gradient during off-axis ECH	Heat transport not purely diffusive phenomenon	ECH; extensive set of transport diagnostics
Rotation Without Externally Applied Torque: Found that ohmic H-mode rotated in opposite direction as ECH H-mode	Expect rotation and E×B velocity shear stabilization in burning plasmas	ECH; non-perturbative diagnostic beams
Turbulent Structures: Unique measurements of turbulent eddies, turbulent velocity fields, and short wavelength turbulence; direct measurements of turbulent particle transport	Experimentally confirmed leading candidates for turbulent transport (ion temperature gradient modes and zonal flows)	BES; FIR scattering; microwave backscattering; PCI

Table 2. DIII-D Program Research Accomplishments (Continued)

RESEARCH ACCOMPLISHMENTS	IMPACT	DIII-D UNIQUE CAPABILITY
PEDESTAL		
ELM Suppression: Uniquely created resonant magnetic perturbations that suppressed ELMs in H-mode regimes	Solves vexing problem of high divertor erosion rates in ITER by eliminating ELMs	Internal coils; cryopump
ELM Physics: Proved ELMs are helical phenomenon associated with high-n peeling-ballooning modes	Improved predictive modeling of ELM size and frequency for ITER	ELITE code and unique lithium beam diagnostic
Quiescent H-mode: Discovered stationary H-mode regime without ELMs; showed ITB can be combined with this quiescent edge (QDB mode)	Possible solution to ITER issue of rapid divertor erosion by ELMs	Counter NBI; cryopumps; extensive set of transport and fluctuation diagnostics
H-mode Pedestal Width: Characterized H-mode pedestal width with machine size	Favorable scaling for ITER H-mode pedestal width	Edge Thomson scattering; X-mode density reflectometer; Lithium beam
BOUNDARY		
H-mode Density Control: Unique factor-of-two density control in H-mode plasmas using active divertor exhaust	Increased the available operating space in ITER. Can match ITER collisionality in DIII-D and increase current drive effectiveness	Cryopumps; between shot helium glow
Detached Radiative Divertor: Characterized plasma profiles in two dimensions in detached divertor plasmas	Solves problem of steady-state high heat flux for ITER's divertor. Develops predictive model of ITER's divertor	Extensive divertor diagnostics, including unique divertor Thomson scattering; flexible plasma shape
Divertor Impurity Enrichment: First demonstration of ability to entrain impurities in plasma boundary through induced SOL flow	ITER baseline scenario now utilizes gas flow for improved impurity entrainment. Validated entrainment model in UEDGE/OEDGE	Active divertor exhaust (cryopumps and baffling); toroidally symmetric gas injection
First Wall Materials: Pioneered measurements of divertor erosion/redeposition; validated erosion/redeposition codes	Improved predictive modeling for ITER divertor design	Divertor material exposure system; micro-balance detectors
SOL Transport/Tritium Retention: Elucidate SOL flow giving rise to co-deposition in divertor	Improved predictive modeling of tritium retention in ITER's divertor	Toroidally symmetric gas injection, pumped divertor
E×B Flows Around X-Point: Elucidated 2-D plasma flows and improved UEDGE model	Possible explanation for geometry dependence of power threshold for L-H transition	Gated midplane camera
Blob Transport: Demonstrated transport of large structures near the boundary becomes increases with increasing density	Extends C-Mod results to lower collisionality regime. Test predictions of BOUT and kinetic BOUT codes	Plunging probe; BES
Density Limit: Demonstrated H-mode operation 50% above Greenwald limit with good confinement	Operation above Greenwald density limit may be possible in ITER	Pellet injection; strong divertor exhaust
HEATING AND CURRENT DRIVE		
Electron Cyclotron Current Drive: Demonstrated highly localized current drive, first comprehensive theory validation	Developed ECCD for current profile control and MHD stabilization in present and future machines	Gyrotrons; steerable launchers; MSE current drive measurement
Fast Wave Current Drive: First proved centrally peaked current drive; validated theory in detail; discovered high harmonic ion cyclotron absorption on beam ions	New reactor-relevant method of noninductive current drive and q(0) control. New method of robust direct electron heating in high β plasmas	Phased antenna arrays; MSE (local current drive) and fast D_α diagnostic (energetic particle pressure profile)
Ion Bernstein Waves: Discovered nonlinear physics of launching IBW's; showed parametric decay instability can dissipate IBW power	IBW's are not directly launched by antenna owing to pondermotive force. New edge loss mechanism for waves in plasmas	Toroidally phased IBW antenna
Neutral Beam Current Drive: First direct measurement of current drive profile	Important validation of NBCD physics, critical to present day AT studies on large tokamaks	Accurate current drive measurements (MSE); D_α fast ion diagnostic
ENERGETIC PARTICLES		
Alfvén Eigenmodes: Discovered TAE modes and lower frequency energetic particle modes, effect on fast ion transport	Established role of super-thermal ions in plasma confinement. Solid scientific basis for projection of fast ion behavior in ITER	Low ripple losses; flexible plasma shape; MSE

Table 3. Future Research on DIII-D

RESEARCH SUBJECTS	IMPACT	DIII-D UNIQUE CAPABILITY	DIII-D ROLE	F/P
ADVANCED TOKAMAK				
Steady-state, high gain: Integrated demonstration of physics basis for steady-state, high gain fusion	Demonstrate feasibility of operation in ITER and compact, steady-state power plants with pressure above conventional limits, improved energy confinement, and 100% noninductive current drive	Versatile system of control actuators and extensive profile diagnostics, both coupled to advanced digital plasma control with real-time equilibrium analysis. See specific scenarios below. ECH, ECCD, FW, co/counter NBI for heating, current drive, and torque: control of pressure, current density, electric field, rotation profiles, double-null pumping for density control	Unique, World Leader	T1,T2,T3, T5,T10, T11,T12, T15/A1, A4,A6, A10
Hybrid scenario: Stationary advanced scenarios with large noninductive current fraction	Probable improved baseline scenario for operation of ITER	Flexible heating and current drive systems (ECH, ECCD, FW, co/counter NBI) to control pressure, and current density profiles; cryopumping for high triangularity double-null configurations	World leader	All above
High bootstrap current fraction plasmas with weak magnetic shear	Leading candidate for advanced scenario in ITER: improved confinement and high stability limits	ECCD, FW, co/counter NBI to control current density profiles; cryopumping for density control in high triangularity double null configuration; non-axisymmetric coils maintain stability at high β	Unique	All above
Current hole: limit of strong negative central shear	Strong ITB- potential high confinement, high β, and high bootstrap current	Flexible heating and current drive systems (ECH, ECCD, FW, co/counter NBI) to control pressure and current density profiles;	World class	All above
VH-mode: neoclassical-level confinement throughout	Potentially highest confinement mode in a tokamak if edge can be kept stable.	Co/counter NBI for edge radial electric field control, cryopumping for edge density control, non-axisymmetric coils for stochastic edge	Unique	All above
Quiescent Double Barrier (QDB)	High-confinement core and edge without ELMs. Best plasma edge for a tokamak	ECH, ECCD, FW, co/counter NBI for transport barrier control	World leader	All above
High ℓ_i: Centrally peaked current density profile	High confinement and very high stability limits without wall stabilization	High current drive power (ECCD, FWCD, NBCD) to maintain and control central current density	World class	All above
CONFINEMENT				
Basic physics of transport from turbulence: test gyrokinetic codes against turbulence measurements	Identify basic mechanisms of transport from turbulence. Code validation leading to predictive capability	Extensive turbulence measurements: Turbulent density field ($0.1<k_\perp \rho<10$). turbulent velocity field ($0.1<k_\perp \rho<0.3$). 2D imaging of turbulent fluctuations using BES. Local turbulent particle transport measurements using BES. Co plus counter NBI allows direct changes of E×B shear at constant power and particle input	Unique	T4/ A3, A14
Short wavelength turbulence in electron thermal transport	Possible basic physical mechanism of electron transport	Short wavelength turbulence measurements with FIR, PCI and μW backscatter	World leader	T4/ A14
Zonal flows regulate turbulence?	Verify new element of modern theory	BES system to directly determine turbulent velocity field	Unique	T5/A14
Separate effect of E×B shear and Shafranov shift stabilization	Transport control for both electron and ion channels using E×B shear for ions, Shafranov shift for electrons	Co plus counter NBI to change E×B, ECH power to change j(r) and Shafranov shift	Unique	T4/ A3,A14
Transport models: test with profiles and modulated sources	Code validation leading to predictive capability of key dependences	Excellent set of core, edge and SOL profile measurements, spatially localized ECH modulates T_e, T_i and v_ϕ	World class	T4/ A14
Poloidal rotation: test neoclassical theory	Crucial for applying E×B shear stabilization in burning plasmas	56 chord CER system (26 poloidal), including views of counter beam	Unique	T5/ A14
Rotation physics: basic understanding	Understanding rotation crucial for E×B shear stabilization in burning plasmas	Co plus counter NBI	World leader	T3,T4/ A14
Rotation without angular momentum input	Rotation important for RWM stability in ITER; joint work with C-Mod	ECH and fast wave power	World class	T4,T11/ A14
Nondimensional transport studies at constant Mach number, varying aspect ratio	Separate effect of rotation and E×B shear from other nondimensional variables and aspect ratio	Co plus counter NBI for independent rotation control; DIII-D can match ITER aspect ratio, shape, normalized β and collisionality. Combination with NSTX and MAST to get aspect ratio range	World leader	T4/ A3,A14

Table 3. Future Research on DIII-D (Continued)

RESEARCH SUBJECTS	IMPACT	DIII-D UNIQUE CAPABILITY	DIII-D ROLE	F/P
STABILITY				
RWM stabilization by rotation	Opens new regime of high β operation, improved performance of ITER	Balanced injection for rotation control; internal non-axisymmetric. coils for MHD spectroscopy	World leader	T2,T3,T6 A4.A12
RWM stabilization by feedback control	Enables RWM control in ITER if rotation in ITER is not sufficient	Internal control coils and fast amplifiers; balanced injection simulates low torque in burning plasma	Unique	T2,T3,T6 A4,A12
NTM physics: threshold, Glasser term, polarization current, bootstrap current, seeding, rotational shear	Project NTM onset and effects in ITER	High-resolution and high-speed profile diagnostics for stability conditions and island structure, incl. magnetic structure (MSE); balanced beams for rotation control; non-axisymmetric coils for active MHD spectroscopy and controlled seeding of NTM	World class	T2,T6/ A12
NTM stabilization by ECCD, test benefits of modulated ECCD	Demonstration of sustained high β without tearing modes, reduce ECCD requirements for ITER	Flexible EC system including real-time steerable mirrors; plasma control system precisely positions current drive, rotation control with co/counter NBI	World leader	T2,T3,T6 T11,A4, A12
Sawtooth stability and control	Quantitative validation of models □ ITER-demonstration of sawtooth control by current profile modification	High-resolution, high-speed diagnostics for m=1 mode structure; flexible EC and FW current drive systems; balanced beams to vary rotational shear	World class	T3,T6/ A1,A2
Disruption mitigation	Validated models for penetration of cold gas jet and thermal quench, means to handle disruptions safely in ITER	Gas jet with variable nozzle geometry; comprehensive diagnostics with high time and space resolution, including fast framing cameras; wall conditioning tolerates repeated disruption tests	World leader	T2,T3,T6 A1,A2, A4
Alfvén eigenmode stability	Quantitative validation of models for effect of Alfvén modes on fast ion transport in ITER	Unique fast ion profile diagnostic and balanced beams to vary fast-ion velocity distribution; comprehensive fluctuation diagnostics for short-wavelength Alfvén modes	Some unique capabilities	T2,T12/ A1,A2, A4
Ideal MHD stability	Validated models for the role of rotation in ideal MHD	Wide flexibility in control of discharge shape, pressure profile, current density profile; balanced beams for rotation control; real-time control with fast equilibrium reconstruction	World leader	T2,T3, T6/ A4,A12
Extended MHD	Validation of nonlinear 3D numerical models (NIMROD, M3D); non-ideal effects (resistivity, viscosity, fast ions□)	Fluctuation and profile diagnostics for detailed 2-D and 3-D measurements of mode structure and evolution	World class	T2,T3, T6/ A4,A12
CURRENT DRIVE				
Bootstrap current: Investigate bootstrap dominated discharges	Steady-state reactor plasmas require >70% bootstrap	Significant ECH power; reduced NBCD due to co plus counter NBI; FWEH to raise T_e	World class	T3,T11, T12/A6
ECCD	Validate ECCD electron trapping model; optimize off-axis ECCD for AT program	High power EC system. MSE measurement on co plus counter beams. Lithium beam system for edge j(r)	Unique	T11/ A6
FWCD	Validate FWCD models to include fast ion damping; FWCD for q(0) control.	FW system with phased antenna arrays, ECH to raise T_e and increase efficiency, MSE measurement on co plus counter beams	Unique	T11/ A6.A8
NBCD: Direct determination	Validated models required for predictive understanding	Co plus counter NBCD allows direct determination of current drive. High T_e with rf. MSE measurement on co plus counter beams allows separation of E_r and j(r) effects.	Unique	T12/ A6,A8
PEDESTAL				
Stabilize ELMs by resonant magnetic perturbation	Eliminate impulsive loading and excessive erosion of ITER divertor	Only high performance diverted tokamak with edge stochastic field coil capability	Unique	T2,T3, T6,T10/ A2,A4
Avoid ELMs: QH-mode	Understand physics to eliminate impulsive loading and excessive erosion of ITER divertor	Li beam edge j(r), co+counter beam injection with detailed radial electric field from CER	Unique	T3,T4, T6,T10/ A2,A4

Table 3. Future Research on DIII-D (Continued)

RESEARCH SUBJECTS	IMPACT	DIII-D UNIQUE CAPABILITY	DIII-D ROLE	F/P
ELM onset: Validate peeling/ ballooning theory	ELITE edge stability code validation and prediction of stability margins in ITER	Li beam edge j(r), comprehensive high resolution edge diagnostics, plasma shaping flexibility	Unique	T2,T6, T10/A2
Nonlinear ELM evolution: validate theories/codes	Predictive capability of ELM energy and particle loss for ITER	Comprehensive high time resolution edge diagnostics, fast gated framing cameras, CER, reflectometry, probes, BES, interferometry	World leader	T2,T6, T10/A2
ITPA edge profile database	Extensive, multimachine database needed to aid in understanding pedestal	Comprehensive set of high spatial and time resolution edge diagnostics and edge fluctuation diagnostics	World class/ joint	T10/ A1,A2
Edge fueling profiles: effect on the pedestal	Fueling depth can determine depth of density pedestal	Comprehensive suite of edge and divertor diagnostics and fueling capabilities	World leader	T10,T15/ A1,A2, A14
Control pedestal structure and stability	Maintain high performance in ITER with tolerable ELMs	Active control combining gas, pellet, and beam fueling, stochastic magnetic field structure, and edge rotation	World class/ joint	T3,T10/ A2,A4
BOUNDARY				
Density control enhances AT performance	Enables pressure and current profile control for optimizing AT steady-state	Baffled, strongly pumped upper and lower divertors, versatile fueling, gas, beams, pellets, rf heating to control particle transport	Unique, World Lead	T10,T15/ A2,A4
Plasma flow in the SOL and divertor	Understanding flows crucial to control the migration and redeposition locations of eroded material	High degree of toroidal symmetry, good poloidal coverage of edge diagnostics, plunging probes with fast T_e and Mach measurements UEDGE, OEDGE, BOUT, SOL-PS DEGAS2 codes	World class	T5,T10/ A2
Carbon transport: Provide 2D picture, validate codes	Qualify graphite, most durable of known plasma facing materials, for BP/ITER. REDEP, WBC, UEDGE, OEDGE codes	Toroidally symmetric impurity gas injection; materials test stations (divertor-DiMES, midplane-MiMES divertor Thomson scattering, all graphite wall	World class	T10,T14/ A2, A11
Radiative divertor: Reduce surface heat flux	Realistic divertor design for long-pulse, ITER and AT reactors	Baffled, strongly pumped upper and lower divertors; wide range of edge collisionalities, UEDGE, SOL-PS codes	World class	T10,T15/ A2

Table 4. Future Research Supporting ITER

FUTURE ACCOMPLISHMENTS	UNIQUE DIII-D CONTRIBUTION	IMPACT ON ITER DESIGN/PROGRAM
ELM suppression: Develop physics understanding of ELM suppression via application of asymmetric fields	Unique capability to apply asymmetric field perturbations over a wide range of plasma collisionalities and plasma shapes	A solution to issue of ELM erosion of divertor plates for ITER
Pedestal physics: Develop self-consistent picture of transport, stability, current drive, and fueling of edge pedestal region	High resolution (both spatial and temporal) measurements of kinetic profiles; unique capability to measure the current density profile; edge fluctuation diagnostics; unique capability to vary plasma collisionality and shape over a wide range.	Improved performance in ITER through improved understanding of mechanisms responsible for pedestal formation and sustainment
Electron transport: Improve the understanding of electron heat transport through detailed comparison with theory	Unique capability to measure short-wavelength fluctuations; ability to probe electron transport dynamics via perturbative heating; new computer cluster allows throughput of intensive gryokinetic turbulence codes	Significant improvement in capability to predict performance in ITER
Tritium retention: Characterize processes leading to migration of carbon and co-deposition of hydrogenic species	Unique capability (DiMES, microbalance detectors) to measure redeposition and erosion; ability to two-dimensionally characterize the divertor and SOL plasma; toroidally symmetric gas injection	Improved predictive modeling of tritium retention and divertor, possibly leading to a solution for ITER
Hybrid scenarios: Demonstrate and document stationary, improved performance plasmas with low rotation and $T_e = T_i$ at ITER-like collisionalities	World leading ability to obtain regime over a wide operating space; balanced NBI, sufficient electron heating, ability to operate over a wide range in plasma collisionality	Improved baseline scenario with higher β and improved confinement, run ITER one hour at full power
Steady-state operation: Develop physics basis of fully noninductive, high β discharges	Only device in the world to have established the existence proof of this regime; full set of current drive and heating tools as well as diagnostics to allow detailed assessment	Provides the basis for steady-state operation in ITER
Current profile control: Develop methods for feedback control of the current profile in high β plasmas	Best in world plasma control system capable of real-time equilibrium analysis with MSE measurement included and real-time control of full set of current drive actuators (ECCD, FWCD, NBCD)	Improved ability to control the current profile in advanced performance regimes in ITER
High β operation: Develop the physics basis for operation well above the no-wall pressure limit with or without the benefit of rotation using RWM feedback	Unique internal coils and external coils; ability to routinely operate at high β; co-, counter-, or balanced NBI to assess the impact of rotation on RWM feedback	Provide a mechanism to dramatically increase performance in ITER
Fast ion physics: Validate/extend the theory of Alfvén eigenmodes and other fast ion instabilities	World class ability to access a wide range of plasma current profiles; full array of fluctuation diagnostics; co-, counter-, or balanced NBI with low ripple losses; MSE; D_α fast ion distribution diagnostics	Improved understanding of fast-ion driven instabilities and fast-ion transport
NTM stabilization: Develop the physics basis for NTM stabilization (both post-onset and preemptive) with modulated ECCD	Unique capability to reduce NTM rotation frequency through variation in NBI torque; sufficient ECCD; PCS lock-in control	Significantly reduced ECCD requirements in ITER
Disruption mitigation: Determine the important mechanisms for mitigating the thermal quench and runaway production when using massive impurity gas injection	World class capability to apply massive gas puff and impurity pellets along with an extensive array of disruption diagnostics	Possible solution to disruption erosion limiting lifetime of surfaces in ITER

Table 5. DIII-D Facility Capabilities

FACILITY CAPABILITY	UNIQUE DIII-D CAPABILITY	RESEARCH PROGRAM IMPACT
Flexible PF coil system — 18 independently controllable PF coils; decoupled Shaping and OH coil set.	Unparalleled shaping capability; ability to match shape of all major toroidal devices; upper and lower single-null divertor and high triangularity double-null capability	Provides unique ability to study interaction between MHD, divertor and boundary physics and plasma shape; matching ability critical to similarity expts and extended parameter scans across devices
Non-axisymmetric coil systems — 6 external coils and 12 internal coils	Unique and highly flexible set of internal coils (n=1,2,3 and m-1-5) coupled with high bandwidth feedback system; unique capability to create ergodic edge in divertor configuration	External coils enable error field studies, high plasma rotation for RWM stabilization; internal coils enable world leading research on active RWM stabilization and ELM stabilization using ergodic edge
Advanced, digital plasma control system — 13 high speed parallel CPUs; tightly coupled with magnetic and profile diagnostic	Unique real time Grad Shafranov equilibrium with q profile using MSE diagnostic; advanced control algorithms fully integrated with detailed control simulation capability e.g. NTM stabilization and prevention algorithm	Enables world leading equilibrium control; world leading research into NTM and RWM stabilization at high β; q and T_e profile control; unique simulation capability provides predictive control design for ITER.
Heating and current drive systems – Electron cyclotron, fast wave, and neutral beam injection	Highest power density; mix of ECCD (off-axis), FWCD (on-axis), NBCD, and high bootstrap current at high β	Sufficient power to achieve high β in almost all experimental conditions; high power mix of current drive and bootstrap current permits study of fully noninductive plasmas at high β
— **ECH/ECCD**—6 MW, 10 s gyrotrons; fully steerable launchers	World class ECH and ECCD system; unique capability in US program	ECCD stabilization of NTMs; q(r), T_e(r) control for steady-state AT discharges
— **Neutral beams** —14 MW, 5 s or 7 MW, 10 s. Co and counter injection	Mix of co/counter/balanced neutral beam injection unique in US program	Mix of co/counter beams enables studies of rotation physics and rotation effects on MHD stability and turbulence; high power NBI enables fast ion studies
Flexible pumping/fueling systems — 2 upper and 1 lower in-vessel LHe cryopumps — 11 valves,19 inlet locations, 5 gases; fast valves for massive gas puff; pellet injector with inner/outer/upper injection locations	—Unique density control in H-mode; pumping of both inner and outer strikepoints in SND; pumping of both upper/lower strikepoints in high triangularity DND —Highest throughput massive gas injection (MGI) system coupled with disruption triggers	—Density control in AT discharges permits efficient ECCD; ITER relevant —World leadership in disruption mitigation expts using MGI and pellet injector; extensive gas injection system permits wide range of expts including puff and pump, impurity flow, radiative divertor
First wall — Full graphite coverage	350°C bake capability	High temperature bake and graphite tile coverage provides high degree of disruption tolerance and enables wide range of disruption and stability expts; rapid recovery from vents; all graphite wall permits ITER relevant carbon erosion and redeposition experiments to be performed.

Table 6. DIII-D has the World's Best Diagnostics Set

PROFILE MEASUREMENTS	CHARACTERISTICS	DIII-D UNIQUE CONTRIBUTION
PROFILE MEASUREMENTS		**BEST SET IN THE WORLD**
Multiple Thomson scattering	8 lasers, 44 points, 20 Hz rep rate/laser	Unique divertor system. Edge system: detailed studies of pedestal physics
Charge exchange recombination spectroscopy	23 vertical channels; 33 horizontal channels, time resolution 270 μs	Best in the world. Unique radial electric field measurement. Edge system: Unique detailed studies of pedestal characteristics, ELM effects on ions and impurities. Core: best studies of role of rotation, ExB shear, ITG; co and ctr beam views
Fast ion characteristics (D_α)	4 chords (prototype)	First measurement of fast ion population using D_α lines
ECE Michelson interferometer	Horizontal midplane profiles	
ECE radiometer	Midplane, 40 channels, 2 μs time resolution	Successfully used for electron temperature feedback through PCS
Infrared interferometers	3 vertical chords, 1 radial chord	Vibration compensated
Microwave reflectometer	Midplane edge profiles, 5 mm, 10 μs res.	
Microwave interferometer	1 chord	
Fast wave reflectometer)	4 antennas	Unique. First measurement of hydrogen isotopic composition, ITER relevant
FLUCTUATIONS, WAVES, TURBULENCE		**BEST SET IN THE WORLD**
Beam emission spectroscopy	2-D, 32 channels (48 planned)	Best in the world. State-of-the-art system allowing localized imaging of density fluctuations in the plasma, plus localization of MHD activity
Microwave reflectometers	2 radial systems	World Class. Comprehensive set of reflectometers (LFS) providing density profile information and fluctuations
Far infrared scattering	Radial view, large range of k numbers (from 0 to 40 cm^{-1})	Best in the world. A unique set of scattering systems capable of measuring turbulence from expected ITG to ETG and TEM activity.
Microwave backscattering	2 views, range of k numbers (from 0 to 40 cm^{-1}),	Best in the world. A unique set of scattering systems capable of measuring turbulence from expected ITG to ETG and TEM activity.
Phase contrast imaging	Vertical view, 16 channels	World class. Measurements of short scale turbulence with capability of localization with k range $\lesssim$50 cm^{-1}
Mirnov coils	70 coils	World class
Multichannel vibration compensated (infrared) interferometer	3 vertical chords, 1 radial chord	Recent upgrade gives unprecedented sensitivity and bandwidth in density fluctuations measurements across profile
X-ray imaging system	100 channels, 5 arrays	World class.
RF probes	5 plasma-facing antennae, 6 recessed loops	World class. Very high bandwidth system capable of measuring electromagnetic phenomenon ($\lesssim$100 MHz) throughout plasma
Scanning probe	Temperature, potential, particle flux, lower divertor and outer midplane	World class. Localized measurements of density, potential and temperature fluctuations in scrape-off-layer in both divertor and midplane locations
MAGNETIC STRUCTURE		**BEST SET IN THE WORLD**
Motional Stark effect polarimetry	45 channels, 4 half-energy channels, 10 (30 planned) for counter beam	Best MSE system in the world. Provides full current and E_r profiles, supporting a wide array of experiments, especially AT. Unique system with both co and counter current views. Used successfully in control system (PCS)
Lithium beam Zeeman polarimetry (edge current profile)	Radial beam with 32 vertical viewing channels (5°mm resolution)	Unique. Unique system confirmed edge current profile peak crucial to understanding pedestal stability and ELMs, close to neoclassical theory
Plasma current Rogowski loops Diamagnetic loops Voltage loops	3 toroidal locations 9 toroidal locations 41 poloidal locations and 30 saddle loops	Comprehensive set of magnetics diagnostics allowing the study of RWM, NTM, CAEs/TAEs and providing accurate basic equilibrium data for EFIT(including real-time).
B_P loops External B_r loops Internal B_r loops	44 poloidal locations, 3 toroidal arrays, 20 loops 3 toroidal arrays, 30 loops; 2 poloidal arrays, 22 loops 3 toroidal arrays, 18 loops	Comprehensive set of magnetics diagnostics allowing the study of RWM, NTM, CAEs/TAEs and providing accurate basic equilibrium data for EFIT(including real-time).

Table 6. DIII-D has the World's Best Diagnostics Set (Continued)

PROFILE MEASUREMENTS	CHARACTERISTICS	DIII-D UNIQUE CONTRIBUTION
Internal B_T loops Magnetic fluctuation probes (dB/dt)	4 toroidal locations 44 poloidal locations, 2 toroidal arrays, 22 probes	Comprehensive set of magnetics diagnostics allowing the study of RWM, NTM, CAEs/TAEs and providing accurate basic equilibrium data for EFIT(including real-time).
Tile current monitors	34 lower divertor, 6 upper	
BOUNDARY/DIVERTOR DIAGNOSTICS		**BEST IN THE WORLD**. Most comprehensive and flexible set of boundary/divertor diagnostics in any tokamak. Allows the study of particle, energy transport on open field lines and quantifying plasma wall interactions
Divertor Thomson scattering	2 lasers, 20 Hz, 8 spatial locations	Unique capability of 2-D mapping of n_e and T_e in lower divertor
Visible spectrometer	12 channels, upper and lower divertor	
Tangential TV (visible)	2-D image of lower divertor in visible	
Tangential TV (VUV)	2-D image of lower divertor in VUV	
Tangential fast framing camera	2-D image fast events (1MHz rate)	World class, diagnose disruptions, ELMs, turbulence
Quartz microbalance	2 balances in lower divertor	Assess in real-time the redeposition of material on divertor surface
Visible filter scopes	24 locations, 5 wavelengths each	
Moveable Langmuir probe	2 probes, lower divertor and outer midplane	
Infrared cameras	4 cameras	
DiMES	2 locations (divertor and midplane)	Versatile system allowing materials and diagnostic testing. Installations can be changed on a daily basis. Over 60 samples and instruments fielded.
Fast neutral pressure gauges	6 locations, 5 in divertors, 1 main chamber	
Penning gauges	Under divertor baffle (upper and lower)	
Baratron gauge	Under divertor baffle	
Langmuir probes	26 probes lower divertor, 28 probes upper divertor and centerpost	
Plasma TV	4 cameras, views rf antennae, main chamber	
IR camera	Inner wall and ceiling views, floor	
Thermocouples	40 channels	
PARTICLE, IMPURITY MEASUREMENTS AND RADIATED POWER		**WORLD CLASS SET**
VUV survey spectrometer (SPRED)	Radial midplane view (dual range)	
Visible Bremsstrahlung array	Radial profile at midplane, 16 channels	
Bolometer arrays	2 poloidal arrays, 48 channels each	Support many plasma scenarios including the study of detached divertor operation in high performance discharges, included in control system
Fast bolometers	Poloidal array (30 channels), 2 arrays to be added	Unique capability to study very fast transient phenomenon during disruptions and associated mitigation
Neutral particle analyzer	3 vertical views	
Natural diamond detector	1 vertical chord	
Lost fast ion detector	2 toroidal locations	
Fast neutron scintillation counters	2 radial channels	
Neutron detectors	4 toroidal locations	
MISCELLANEOUS		
Hard x-ray monitors	4 toroidal locations	
Synchrontron (IR) radiation detector	2 tangential chords on midplane	
Torus pressure gauges		
Residual gas analyzer		

Table 7. Comparison of DIII-D System Parameters (maximum capability when no range is given) to ITER

System Parameters	DIII-D	ITER
Major radius (m)	1.49 to 1.88	6.2
Minor radius (m)	0.42 to 0.67	2.0
Aspect ratio	2.5 to 4.5	3.1
Elongation	0.9 to 2.6	1.85
Triangularity	-0.1 to 1.0	0.49
Plasma volume (m^3)	7 to 24	837
Toroidal field (T)	2.2	5.3
Plasma current (MA)	3.0	15
Neutral beam power (MW)	20	33
Gyrotron power (MW)	6 (9)	20
Radio frequency power (MW)	3 (6)	20
Current flattop (s)	10	400
Baking temperature (°C)	400	240
Wall material	Boron coated carbon	Beryllium/tungsten/carbon

Table 8. DIII-D covers the ITER-Relevant Regime, as Seen From This Comparison of the Maximum (except where range is noted) DIII-D Plasma Parameters to the Q=10 ITER Operating Point

Plasma Parameters	DIII-D	ITER
Normalized current, I/aB (MA/m/T)	3.3	1.4
Toroidal beta, $\langle\beta\rangle$ (%)	12.5	2.5
Normalized beta, β_N	6.0	1.8
Poloidal beta, β_p	5.2	0.65
Fast particle beta, $\langle\beta_f\rangle$ (%)	3.4	0.2
Normalized fast ion velocity, V_f/V_A	0.5 to 1.5	1.5
Relative gyroradius, $\langle\rho^*\rangle$	1×10^{-4} to 1×10^{-5}	5×10^{-5}
Collisionality, $\langle\nu^*\rangle$	0.001 to 1	0.002
Pedestal collisionality, ν_e^*	0.02 to 20	0.06
Electron temperature, T_e (keV)	16	23
Ion temperature, T_i (keV)	27	18
Electron density, n_e ($\times10^{20}$ m^{-3} s)	3.0	1.0
Normalized density, n_e/N_G	0.15 to 1.7	0.85
Confinement time, τ_E (s)	0.48	3.67
Normalized confinement, $H_{98(y,2)}$	2.2	1.0
Helium confinement, τ_{He}^*/τ_E	8	5
Lawson value, $n_e(0)\,\tau_E$ ($\times10^{20}$ m^{-3} keV s)	0.4	3.9
Fusion product, $n_i(o)T_i\tau_E$ ($\times10^{21}$ m^{-3} keV s)	0.7	6
Fusion power, P_{fus}	28 kW (DD)	500 MW (DT)
Divertor heat flux, q_{max} (MW/m^2)	6	10
Radiative divertor, P_{rad}/P_{div} (%)	60	65
Argon enrichment, η_{Ar}	17	??
(2,1) NTM suppression, J_{EC}/J_{BS}	3	0.9
Bootstrap current fraction, I_{BS}/I_p (%)	80	20

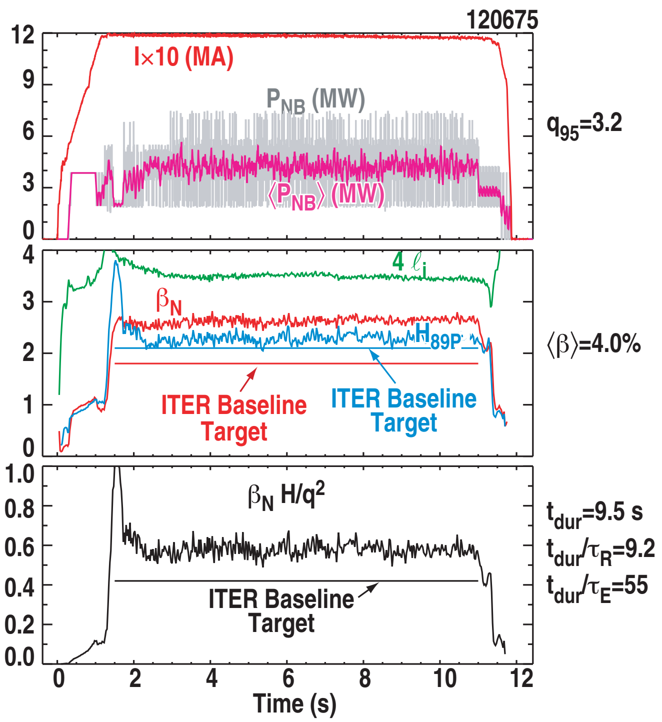


Fig. 1. Stationary high performance, at or above ITER baseline targets, in DIII-D advanced inductive scenario.

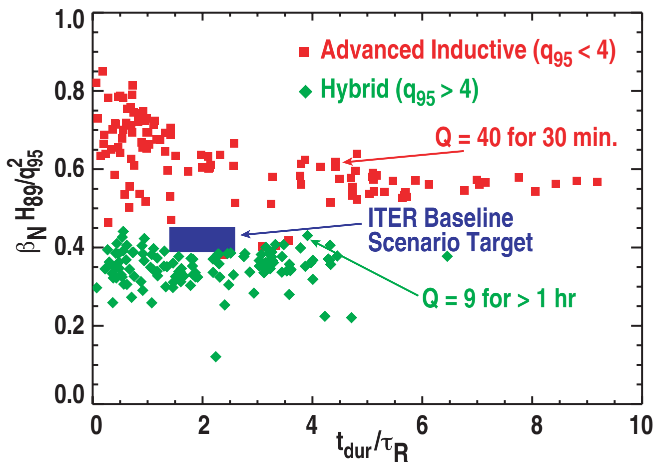


Fig. 2. Advanced inductive and hybrid scenario discharges in DIII-D project favorably for ITER.

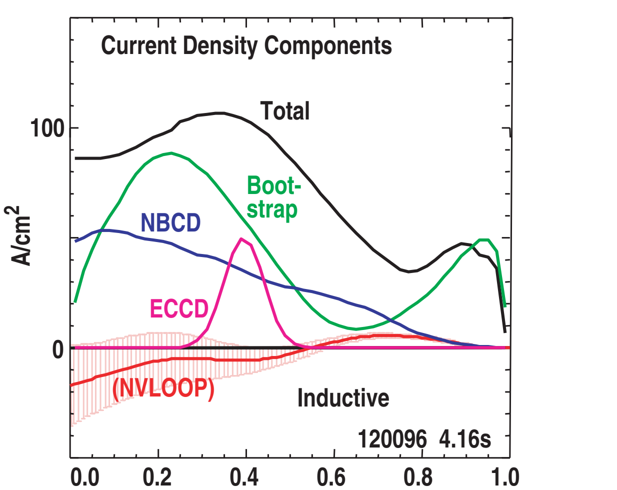


Fig. 3. 100% noninductive current achieved with good current profile alignment (loop voltage ≈0 everywhere).

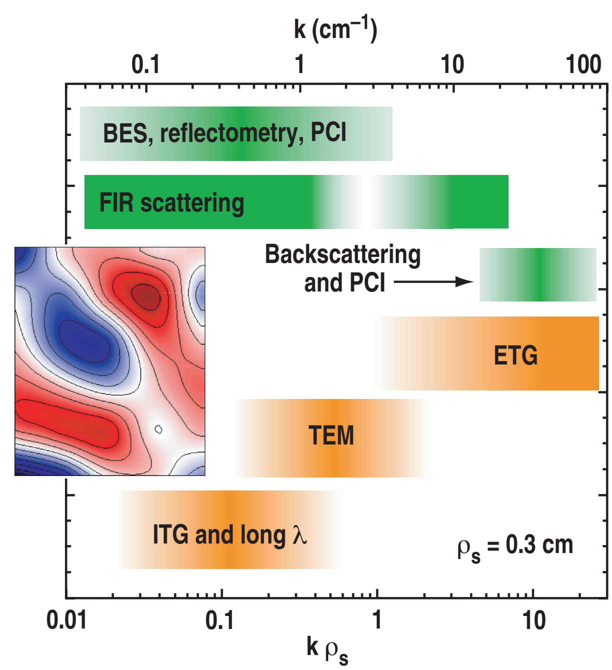


Fig. 4. DIII-D's comprehensive turbulence diagnostics cover all relevant spatial scales for transport physics. Turbulence cells from BES measurement.

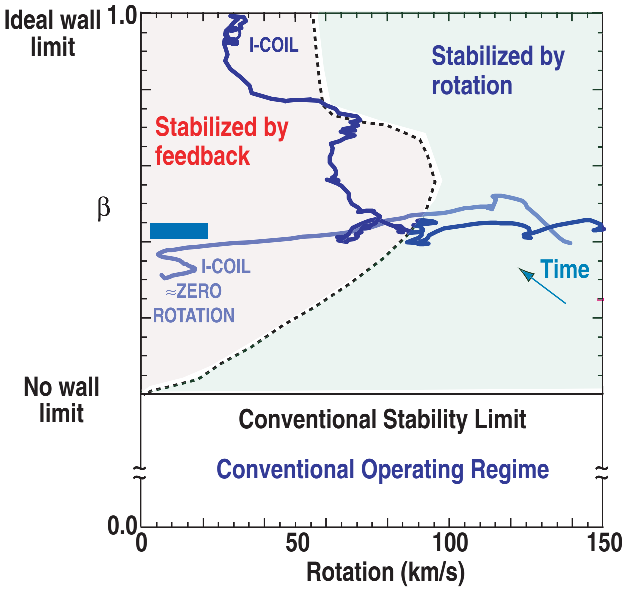


Fig. 5. RWM feedback control gives DIII-D unique access to the ITER-relevant regime of high beta and low rotation.

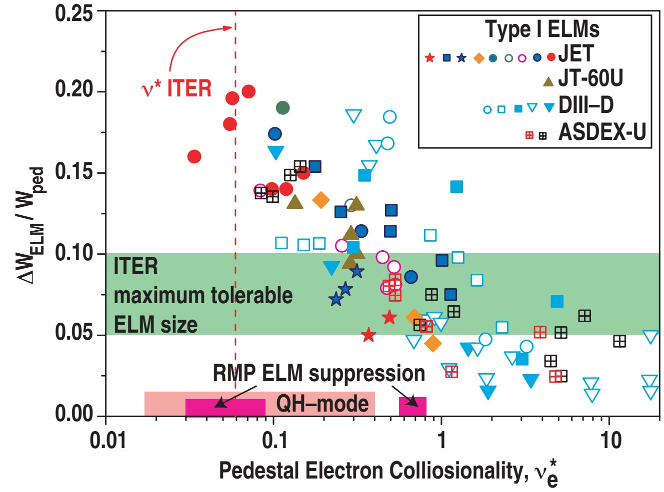


Fig. 6. DIII-D has two means of ELM control at ITER-relevant regimes of high beta and low rotation.

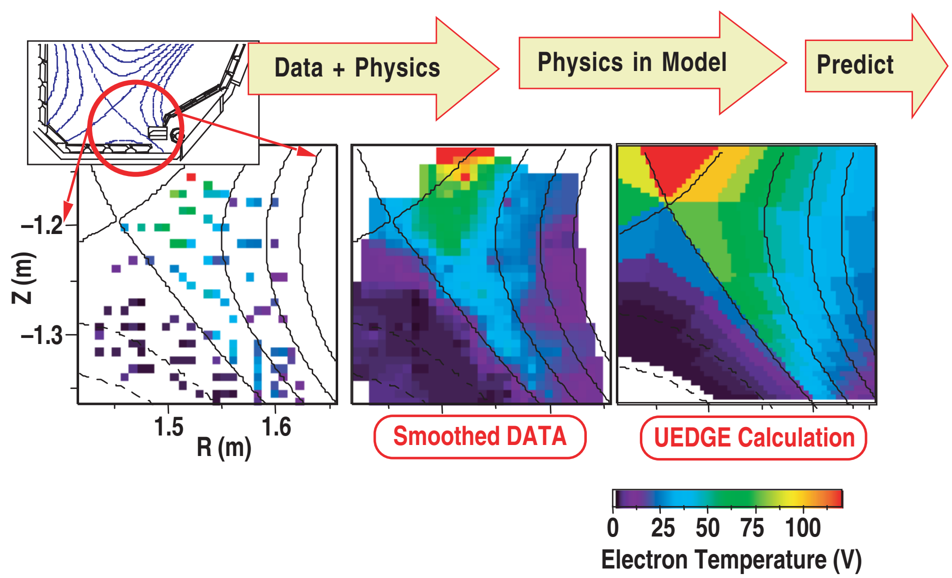


Fig. 7. Divertor Thomson scattering yields 2-D map of T_e, in good agreement with modeling.

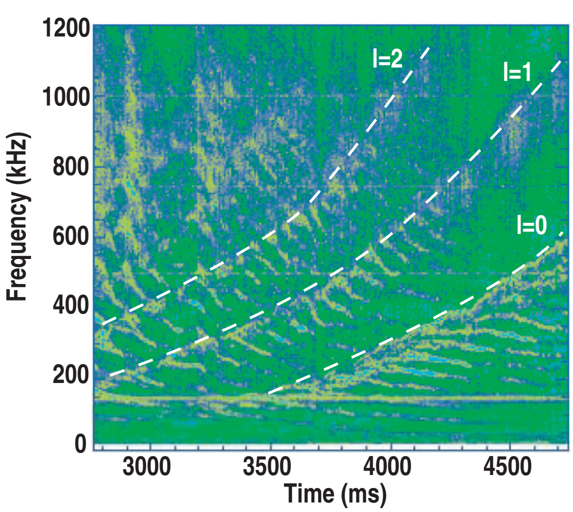


Fig. 8. FIR scattering shows a "sea of Alfvén eigenmodes" not visible on external magnetic diagnostics.

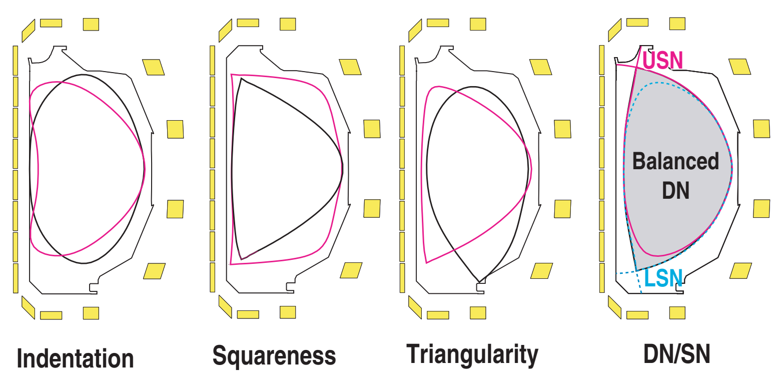


Fig. 9. DIII-D's unique shape control capability yields a wide spectrum of experimentally achieved discharge shapes.

DIII-D is a Large, International Program

Active Collaborations 2004

US Labs

ANL (Argonne, IL)
LANL (Los Alamos, NM)
LBNL (Berkeley, CA)
LLNL (Livermore, CA)
ORNL (Oak Ridge, TN)
PPPL (Princeton, NJ)
SNL (Sandia, NM)

Industries

Calabasas Creek (CA)
CompX (Del Mar, CA)
CPI (Palo Alto, CA)
Digital Finetec (Ventura, CA)
DRS (Dallas, TX)
DTI (Bedford, MA)
FAR Tech (San Diego, CA)
IOS (Torrance, CA)
Lodestar (Boulder, CO)
SAIC (La Jolla, CA)
Spinner (Germany)
Tech-X (Boulder, CO)
Thermacore (Lancaster, PA)
Tomlab (Willow Creek, CA)
TSI Research (Solana Beach, CA)

US Universities

Auburn (Auburn, Alabama)
Colorado School of Mines (Golden, CO)
Columbia (New York, NY)
Georgia Tech (Atlanta, GA)
Hampton (Hampton, VA)
Lehigh (Bethlehem, PA)
Maryland (College Park, MD)
Mesa College (San Diego, CA)
MIT (Boston, MA)
Palomar (San Marcos, CA)
New York U. (New York, NY)
SDSU (San Diego, CA)
Texas (Austin, TX)
UCB (Berkeley, CA)
UCI (Irvine, CA)
UCLA (Los Angeles, CA)
UCSD (San Diego, CA)
U. New Mexico (Albuquerque, NM)
U. Rochester (NY)
U. Utah (Salt Lake City, UT)
Washington (Seattle, WA)
Wisconsin (Madison, WI)

Russia

Ioffe (St. Petersburg)
Keldysh (Udmurtia, Moscow)
Kurchatov (Moscow)
Moscow State (Moscow)
St. Petersburg State Poly (St. Petersburg)
Triniti (Troitsk)
Inst. of Applied Physics (Nizhny Novgorod)

European Community

Cadarache (St. Paul-lez, Durance, France)
Chalmers U. (Goteberg, Sweden)
CFN-IST (Lisbon, Portugal)
CIEMAT (Madrid, Spain)
Consorzia RFX (Padua, Italy)
Culham (Culham, Oxfordshire, England)
EFDA-NET (Garching, Germany)
Frascati (Frascati, Lazio, Italy)
FOM (Utrecht, The Netherlands)
Helsinki U. (Helsinki, Finland)
IFP-CNdR (Italy)
IPP (Garching, Greifswald, Germany)
ITER (Garching, Germany)
JET-EFDA (Oxfordshire, England)
KFA (Julich, Germany)
Kharkov IPT, (Ukraine)
Lausanne (Lausanne, Switzerland)
IPP (Greifswald, Germany)
RFX (Padova, Italy)
U. Dusseldorf (Germany)
U. Naples (Italy)
U. Padova (Italy)
U. Strathclyde (Glasgow, Scotland)

Japan

JAERI (Naka, Ibaraki-ken, Japan)
JT-60U
JFT-2M
Tsukuba University (Tsukuba, Japan)
NIFS (Toki, Gifu-ken, Japan)
LHD
Hiroshima University (Japan)

Other International

Australia National U. (Canberra, AU)
ASIPP (Hefei, China)
Dong Hau U. (Taiwan)
KBSI (Daegon, S. Korea)
KAERI (Daegon, S. Korea)
Nat. Nucl. Ctr. (Kurchatov City, Kazakhstan)
Pohang U. (S. Korea)
Seoul Nat. U. (S. Korea)
SWIP (Chengdu, China)
U. Alberta (Alberta, Canada)
U. of Kiel (Kiel, Germany)
U. Toronto (Toronto, Canada)

- 90 institutions participate
- 515 active users
  - 119 GA
  - 396 others
- 317 scientific authors (2004)
  - 577 cumulative
- 1082 visits to GA (2000–2004)
- Students and faculty have been from
  - 65 universities
  - 28 states

STRONG INTERNATIONAL INTEREST IS SHOWN IN THE 451 RESEARCH PROPOSALS FOR 2004

FOREIGN

Belgium 1	Switzerland 3
France 4	Russia 3
Germany 8	Japan 2
JET 3	Australia 1
Portugal 3	England 7
Spain 2	Canada 4
Italy 1	

Total: 42

DOMESTIC

Columbia 29	SNL 9
FarTech 2	UCI 2
GA 183	UCLA 17
Lehigh 2	UCSD 18
LLNL 23	U. Texas 4
NRL 1	U. Wisconsin 6
MIT 3	Unaffiliated 1
ORNL 43	
PPPL 66	

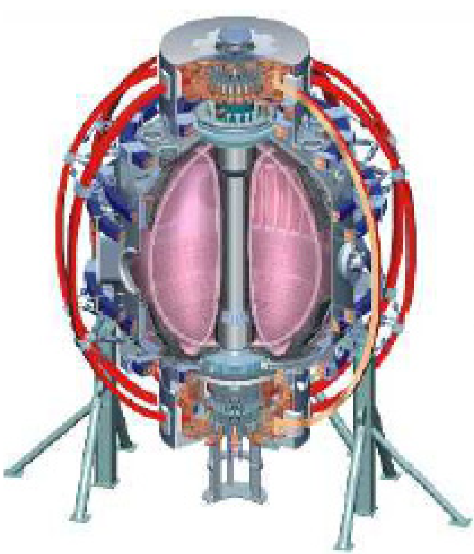
Total: 409

NSTX Research and Its Role in Advancing Fusion Energy Science

A Report Prepared for the FESAC Facilities Review

May 28, 2005

<u>Contents</u>



I. Overview

The NSTX research program takes advantage of unique and complementary plasma properties and facility capabilities, as well as broad collaboration to address issues of central importance to ITER, to provide the physics basis required for optimizing future burning plasma configurations and to advance the understanding of toroidal magnetic confinement of plasmas generally. Enabling these broad goals is an exceptionally wide operating space, a high degree of facility flexibility, unique diagnostic opportunities afforded by the compact ST geometry, and vibrant collaborations with other experimental research programs and the theory community worldwide. In addition to the unique science revealed by research on NSTX itself, a unique perspective on many issues emerges from its contact with and extensions beyond the operating space of moderate aspect ratio magnetic confinement devices around the world, including C-MOD, DIII-D, ASDEX-U, JET, JT-60U, MAST, and TEXTOR.

NSTX research is well aligned with the major research campaigns outlined by the FESAC Priorities Panel. The subsequent chapters of this document will describe NSTX research following the structure of the FESAC Campaigns, with specific responses to the charge questions provided to FESAC for this review. In this Overview section, summary responses to the questions are provided. At the end of this section, the integration of this research, which yields a vision for high beta, solenoid-free steady-state operation is presented.

1. What are the unique and complementary characteristics of the NSTX facility? - NSTX researchers study plasmas at the lowest aspect ratio, and over the widest range of beta, Alfvén Mach number, ratio of the fast-ion to Alfvén velocity (V_f/V_A), and plasma shape factor (Iq_{95}/aB) of any major toroidal confinement device in the world. NSTX accesses plasma regimes with key dimensionless parameters relevant to burning plasma conditions on ITER, as well as to scientifically powerful extensions beyond them. NSTX research therefore contributes directly to the creation of the sound physics basis needed for ITER, and to the optimization of steps beyond ITER, including a Component Test Facility.

Stability and Startup - NSTX is unique in its unusually wide range of beta (approaching unity in the plasma core), wide range of rotation speed compared to the Alfvén and sound speeds, strong shaping, and low aspect ratio. These qualities allow important tests of ideal and non-ideal MHD stability theory under conditions important for ITER and for steps beyond. Its position and shape flexibility, its closely-fitting wall, its set of non-axisymmetric coils for error correction and feedback control of modes, and its profile diagnostics, including MSE measurement of the current profile, allow one-of-a-kind studies of the impact of conducting structures, rotation, and its shear, on MHD stability, saturation, and mode damping. The error correction and feedback control coils, similar to those proposed for ITER, permit leading research in active mode stabilization over an exceptionally wide range of V_ϕ/V_A, allowing validation of theory at an opportune time for ITER. The toroidal rotation diagnostic which resolves structures on the scale of the ion gyroradius is particularly important in providing physical insight into the coupling of plasma rotation to relevant modes. Low aspect ratio creates the opportunity for unique tests of the understanding of tearing mode stability important to ITER, both in terms of the mode-coupling which triggers the instability and Glasser terms which provide stabilization. NSTX also has the capability of studying the physics of reconnection associated with internal modes using very fast soft x-ray imaging diagnostics. Reconnection is central to magnetic helicity injection, a technique for solenoid-free startup uniquely possible in NSTX where the inner and outer vacuum vessel components are electrically isolated.

Transport and turbulence - Electron thermal transport is an outstanding issue for fusion and is critical to the prediction of the performance of ITER plasmas dominated by electron heating from alpha particles. The relative roles of electron *vs.* ion scale turbulence is an area of vigorous discourse in the field. Plasmas spanning the widest range of beta in the world will enable turbulence predicted to have a strong electromagnetic component to be studied for the first time in hot, collisionless plasmas. On NSTX, this physics will be studied with reflectometry, which measures fluctuations at ion scales, including a new technique for imaging of these turbulent structures. NSTX tangential scattering measurements spanning from ion to electron scales will be unmatched in spatial resolution at the higher wavenumbers. This capability can only be implemented on NSTX, because its slender TF coils allow excellent tangential access. The role of rotational shear in reducing ion transport to the neoclassical level will be explored by rotation variation imposed through the active control coils. A unique high-precision measurement of poloidal rotation in the plasma core will enable a more thorough understanding of the role of internal flows and flow shear which are expected to be strongly influenced by trapped particle and magnetic pumping effects in NSTX.

Energetic particles - The rich array of fast ion MHD activity, driven by a super-Alfvénic fast ion population produced by neutral beam injection on NSTX, provides data essential to ITER and burning plasmas. The NSTX data in this regime are unique because q-profile measurements are available for the analysis of mode stability. Furthermore, the tangential interferometer array and a spatially scanning neutral particle analyzer can measure the strucure of modes, and their localized impact on the fast-ion velocity distribution. The array of interacting fast-ion instabilities and the predominant heating of electrons by fast ions on NSTX yields a unique overlap with a burning plasma in ITER. For example, Compressional Alfvén Eigenmodes, which were first discovered on NSTX, are now also anticipated on ITER. The theory of these modes is now being validated using data from NSTX.

The high fraction of trapped particles intrinsic to low aspect ratio enables a unique opportunity to use Electron Bernstein Waves (EBW) to explore the potential of the Ohkawa effect for current drive. This method is predicted to provide efficient far-off-axis current drive, something which is not possible with conventional Fisch-Boozer electron cyclotron current drive, but important for accessing the most advanced operating modes. The overdense plasmas of NSTX also enable exploration of the physics of High-Harmonic Fast Waves (HHFW) using the unique NSTX 12-strap antenna which provides control of wave phase velocity in real time. Together, HHFW and EBW capabilities will form the foundation of current drive research in the U.S. with great potential benefit to STs and other high β toroidal confinement concepts.

Boundary physics and technology - Understanding the transport associated with edge instabilities and turbulence is of critical importance to managing the heat and particle fluxes in ITER. The large turbulence scales and relatively slow dynamics in the edge of NSTX provide a unique opportunity for imaging turbulent structures. Combined with the shaping capability in NSTX, and MSE measurements of the current profile, this permits strong tests of the effects of edge magnetic geometry on turbulence and ELM activity. Measured heat fluxes on the divertor plates on NSTX match and exceed those expected for ITER, due to the compact NSTX size and high auxiliary heating power. Flexible shaping and high B_p/B_T enable NSTX to explore strong divertor flux expansion for heat management, which can improve designs for burning plasma devices. Unique real-time measurements are being made in NSTX of the deposition of films and dust, issues of critical importance to tritium retention and window life in burning plasmas. For recycling control and power handling, novel approaches for coating the plasma-facing surfaces with lithium in NSTX are a step towards deploying a liquid lithium divertor target, based on remarkably favorable results in LTX. Successful demonstration of the liquid lithium approach could revolutionize particle and heat flux management for any toroidal confinement concept.

2. How do the characteristics of NSTX, when combined with those of C-Mod and DIII-D, make the U.S. toroidal research program unique as a whole in the international program? The wide range of plasma parameters and the flexibility of NSTX are complementary to those of C-Mod and DIII-D, providing the U.S. assets unmatched in the world program and the capability to address issues critical to toroidal confinement science and burning plasmas.

Stability and Startup - The shaping, wide range of beta, and active mode control tools on NSTX are key elements of an unrivaled U.S. capability to study MHD and its control. The active control capability in NSTX, unique among STs, involves control coils that are similar in their plasma coupling to those proposed for ITER. NSTX complements the mode control capability of DIII-D which operates at lower Alfvén Mach number and uses both internal and more weakly coupled external coils. Together, the NSTX and DIII-D programs are working to clarify the role of the sound speed and Alfvén speed in mode damping and dissipation physics. The high resolution measurements of toroidal flow on NSTX are contributing strongly to understanding these phenomena.

The wide range of aspect ratio between the three devices provides leverage to understand the physics of tearing modes, including destabilization through mode coupling and differences in the importance of the Glasser term. NSTX, C-Mod, and DIII-D provide a complete set of tools for investigating avoidance of locked modes and the role of magnetic field strength. NSTX can overlap the beta and shape parameters of C-Mod and DIII-D and also extends well beyond them, testing the physics basis for both ITER and innovative concepts. Coaxial helicity injection research in NSTX combines with reconnection research in spheromaks and astrophysics to yield a unique approach to plasma initiation and current ramp-up that may benefit STs as well as advanced tokamak reactor designs.

Transport and turbulence - The suite of turbulence diagnostics on NSTX, C-Mod and DIII-D yields a world-leading capability for ion and electron scale turbulence research that complements the cutting-edge theory and simulation in which the U.S. is a leader. International focus has increasingly moved towards electron thermal transport, critical for understanding burning plasmas, and the role of electron-scale turbulence. NSTX is deploying a major, novel scattering diagnostic that takes advantage of the ST geometry, the excellent tangential access and the relatively large electron gyro-radius to system size, to give the U.S. high radial resolution measurements of such turbulence and of an important class of instabilities driven by ion-scale dynamics. The wide range of β in NSTX gives the U.S. unmatched capability to ascertain the beta dependence of turbulence and confinement. Joint experiments in NSTX and DIII-D are beginning to reveal the dependence of confinement on aspect ratio, important for future burning-plasma devices.

Energetic particles and waves - U.S. research in fast-ion physics has a leadership role in the world, and NSTX provides the capability to develop the physics basis for ITER and beyond. Due to the high β in NSTX, its plasmas contain the highest fraction of super-Alfvénic ions, with a wide range of $V_{fast}/V_{Alfvén}$ that include and extend beyond that expected for ITER. These features make NSTX the only facility in the world where the stability properties of plasmas with large populations of super-Alfvénic ions can be studied with a direct measurement of the q profile, both to confirm theoretical predictions of mode structure and to measure the impact of the instabilities on beam-driven current. The wide array of interacting fast-ion instabilities on NSTX is similar to that expected on ITER. The wide range of parameters in which fast-ion modes are excited overlaps with very-low field operation of DIII-D. This enables tests of fast ion physics important not only to ITER, but to steps beyond it. Joint studies in NSTX and DIII-D have already revealed the roles of aspect ratio and $V_{fast}/V_{Alfvén}$ in this physics. This complementary research enables the linear physics of fast-ion-MHD interactions to be confirmed and extended to nonlinear regimes often present on NSTX and likely to be present in ITER.

The major RF current drive methods being pursued on NSTX (HHFW and EBW), C-Mod (LHCD), and DIII-D (ECCD and FW) together represent the broadest approach worldwide, permitting the development of a comprehensive understanding of RF current drive for ITER and beyond. The complementary characteristics of the NSTX HHFW and DIII-D FW systems will deepen understanding of fast-wave physics in plasmas containing energetic particles. The NSTX EBW and DIII-D ECCD systems will elucidate the physics of trapped electrons in current drive.

Boundary physics and technology - Edge turbulence imaging and flow measurements at high β and relatively low magnetic field complement Alcator C-Mod at lower β and higher field, giving the U.S. an unmatched capability in this realm. The similar shapes that can be created on the three major U.S. devices combine with the smaller aspect ratio of NSTX and its wide range of shape factor to allow investigations of the role of magnetic geometry, including shear, in the stability of the edge pedestal. The study of the potential of lithium coatings, and the planned liquid lithium divertor, for managing edge heat and particle fluxes will make NSTX unique in the world. Combining NSTX with C-Mod and DIII-D, the U.S. is unique in its breadth, evaluating carbon, high-Z materials, and lithium as options for managing these fluxes.

3. How well does the NSTX research program cooperate with the international community in coordinating its research, and how have we exploited its special features to contribute to fusion research internationally, in general, and to the ITER design specifically? - The NSTX research team involves a wide range of national and international research partners. This team participates in many joint research efforts that take advantage of key similarities and differences between experiments to gain a broader perspective on critical issues in toroidal confinement physics. Notably, with a keen interest in the physics that can learned from NSTX, but without a major small-aspect-ratio device of their own, Japanese researchers are very strongly involved in the NSTX program. Overall, cooperation and coordination among STs worldwide is strong.

Stability and Startup - Joint research with DIII-D on resistive wall mode physics and control is fostered by participation within each other's program planning activities and through the ITPA. Further, Columbia University researchers play a strong role at both NSTX and DIII-D, while GA researchers collaborate with NSTX and PPPL scientist with DIII-D. This program capitalizes on key device differences to reveal the impact of magnetic field, its shear, and rotation on ideal-wall stability limits. The research is designed specifically to take advantage of similar cross-sectional shapes and sizes between the two devices, but different plasma velocities when normalized to the Alfvén speed and different aspect ratios and consequently magnetic shear profiles.

Substantial progress in research in coaxial helicity injection has grown from a close working relationship with the University of Washington. The development of plasma startup techniques using induction from the outer poloidal field coils only is being pursued under a collaborative agreement with the University of Tokyo, KAIST, and Kyushu Tokai University.

Transport and turbulence - Joint experiments with MAST on momentum transport are underway and others are planned for 2006 within the ITPA on the roles of diamagnetic and driven flows in transport barrier dynamics. Understanding the role of aspect ratio in plasma confinement is important, not only for predicting confinement in ITER, but for steps beyond ITER, including a Component Test Facility (see Section VI). Developing this understanding is the subject of a three-way ITPA collaboration between NSTX, DIII-D, and MAST. With plasmas spanning the widest range of beta anywhere, NSTX is making crucial contributions to determining the beta dependence of confinement, a critical issue being pursued vigorously within the ITPA.

Energetic particles - Current drive is key to long pulse, solenoid-free plasma performance on NSTX and will be critical for advanced ST operation. The unique capabilities of NSTX contribute to understanding the requirements for ITER "hybrid" operation, also being studied on tokamaks worldwide. NSTX is investigating the impact of fast-ion instabilities on beam-driven current taking advantage of ITER-like population with $V_{fast} > V_A$ where its ability to diagnose the q profile is particularly important. Joint research in support of the ITPA has been undertaken by NSTX and DIII-D to assess the aspect-ratio dependence of fast-ion MHD modes and, thereby, test theories being applied to ITER. Compressional Alfvén wave studies on DIII-D have grown from the first observations of these modes on NSTX. Fast-ion transport and loss studies are performed in conjunction with MAST. Research into Electron Bernstein Waves to drive current on NSTX is conducted in collaboration with and builds upon research in MAST.

Boundary physics and technology - Taking advantage of similarities and differences between C-Mod and NSTX, including the wide range of toroidal field, and thus gyroradius and ExB drift speed, joint studies are being performed within the ITPA on the dynamics of turbulence in the edge. H-mode pedestal characteristics are critical for ITER performance, and the possibility that low aspect ratio and the resulting high edge shear may allow access to second stability in the pedestal is being explored in studies involving NSTX, DIII-D and MAST. Joint experiments in NSTX and MAST will exploit their differences in wall proximity to study the role of neutral influxes on H-mode transition physics. NSTX is advancing ELM research through images captured with a state-of-the-art fast camera provided by collaboration with Hiroshima University.

4. How does NSTX contribute to fusion science and the vitality of the U.S. fusion program? What research opportunities would be lost by stopping research on NSTX?

Stability and Startup - A comprehensive, validated understanding of MHD stability is needed for ITER and for concept optimization, while a method for starting a plasma without reliance on a central solenoid is needed for further steps beyond ITER (*e.g.*, ARIES-AT). The low aspect ratio, close-fitting wall and the wide parameter ranges available on NSTX, particularly in beta and Alfvén Mach number, are key to developing our understanding of MHD stability. Taking advantage of these and its unique diagnostic capability, studies on NSTX can establish the physics of mode damping and dissipation in wall mode stabilization. This physics, needed to design active mode control for ITER with confidence and to optimize a next-step, would be severely weakened without NSTX. U.S. leadership in the development of plasma startup techniques using CHI, EBW, outer PF-coil induction, and HHFW would be lost without NSTX

Transport and turbulence - With its unique opportunities for measuring turbulence on both electron and ion scales and its wide range of β, the U.S. program has an unmatched opportunity in NSTX to study ion- and electron- scale turbulence and their interactions. This includes crucial contributions to understanding electron thermal transport, the core transport issue that yields the greatest uncertainty in ITER confinement prediction. Cessation of the NSTX program will waste these opportunities. NSTX is the only device that can explore the continuum of electrostatic and electromagnetic effects on turbulence as beta varies from nearly zero to approaching unity. Such tests are important for validating simulations for ITER. An understanding of the impact of aspect ratio, and thus magnetic geometry, on transport will spur concept innovation. Furthermore, because of the high β, the impact of NSTX extends beyond fusion into the astrophysical realm. All of this fundamental physics will be lost without NSTX.

Energetic particles - NSTX plasmas exhibit the widest range of dimensionless parameters relevant to fast-ion MHD. It is the only device capable of studying an ITER-like super-Alfvénic fast-ion population with the benefit of a measured q profile. The loss of NSTX would undermine U.S. leadership in this field and scuttle the opportunity for stringent tests of theory for ITER and all burning plasmas. The opportunity to study the interplay between low and high frequency modes is unique to NSTX. Progress on forms of current drive, HHFW and EBW, needed for overdense plasmas in the ST and other high β devices, would cease without NSTX.

Boundary physics and technology - An understanding of SOL turbulence, afforded by the large turbulent structures that are readily imaged in NSTX, is crucial for understanding the heat and particle fluxes to the first wall in ITER. Development of this understanding would be seriously undermined without NSTX. Critical tests of edge stability made possible by the low aspect ratio and shaping would be lost, denying the program information critical for extrapolation to ITER and for concept optimization. Unique studies at the highest divertor heat loads would also be lost. An assessment of lithium wall coatings and liquid lithium divertors, the latter offering great potential benefits to all toroidal confinement concepts, including potentially ITER, would be unavailable if the NSTX program were curtailed and U.S. leadership in this area would be lost.

Integration of the topical research elements - The NSTX program integrates many scientific elements that together advance NSTX performance with respect to plasma stability, confinement, waves and energetic particles, and boundary physics. For example, research aimed at demonstrating solenoid-free startup and sustainment will be combined with advanced control strategies to generate long-pulse, high-confinement, and high-beta plasmas. Advanced diagnosis will be coupled with theoretical tools to further broaden and deepen the basis for physics understanding of these high beta, high confinement regimes.

Integration of the physics elements to obtain solenoid-free operation near 40% average toroidal beta enables important validation of the physics foundation for the models presently under development for ITER prediction. The highest leverage tests from NSTX of ITER-relevant models for stability, transport, wave-particle physics, and boundary physics are found by varying physical parameters across their ranges, including the high beta, low internal inductance, strongly shaped regimes projected for NSTX. As such, study of the integration of the science towards long pulse, and high beta plasmas will increase confidence in the projections these models make for ITER itself while it also has a direct bearing on assessing the technical requirements and ultimate operating range for an ST-based Component Test Facility (CTF).

Integrated modeling will assume a large role in the research planning on ITER, owing to the fact that each plasma will be a highly prized resource. NSTX integration and its modeling therefore takes on an important role for ITER as the successful prediction of NSTX operation helps validate the models that ITER will use. NSTX is unique among existing programs in its use of full free-boundary simulations of the plasma growth

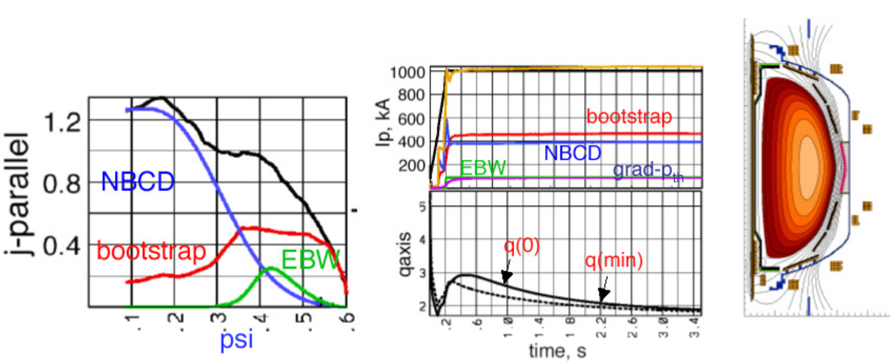


Fig. 1 Simulation of a high β integrated scenario being developed for NSTX.

and subsequent dynamics during the current flattop. This approach, an example of which is shown in Fig. 1 below, was brought to the ITER team by NSTX researchers participating with ITER and has now been adopted by them.

Validation of the models at various extremes of operating space of toroidal confinement systems also helps establish the physics basis needed for concept innovation. Joint studies between moderate and low aspect ratio devices are being used to challenge and validate theories of physics that will provide the scientific answers needed to decide what will make for the most attractive toroidal confinement concept.

II. MHD Science and Solenoid-Free Startup

The promise of the spherical torus concept is based on theoretical predictions of the MHD stability advantages afforded by low aspect ratio. NSTX plasmas routinely operating near the wall-stabilized limit at beta values that far exceed values obtained in any other major experiment. Because the predicted enhancement of stability has been born out experimentally, the ST represents a unique and powerful tool for building the physics basis for successful ITER operation and for the optimization of toroidal confinement concepts.

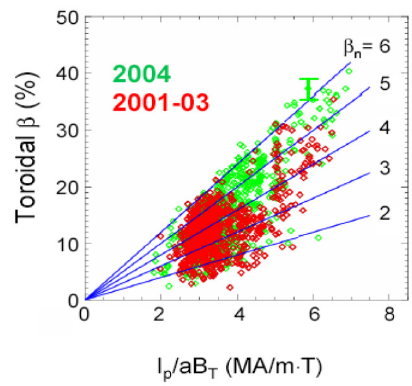


Fig. 2.1 Toroidal beta vs. I/aB_T from 2001 - 2004. High beta operation has become routine with the development of advanced shaping, control, and wall preparation. Also shown are lines of constant normalized beta.

1. What are the unique and complementary characteristics of the NSTX facility that enable important advances in MHD and solenoid-free startup research?

• The low aspect ratio, combined with the natural shaping and the nearby conducting wall, enhance MHD stability and allow the highest toroidal beta values on any major toroidal confinement facility. Routine access to high beta regimes, illustrated in Fig. 2.1, allows validation of MHD theory and extends the operating space covered by major toroidal confinement systems.

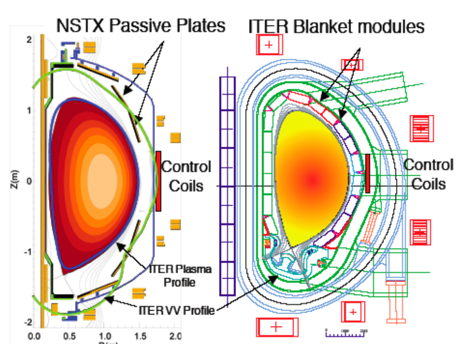


Fig. 2.2 Active mode control coils and passive plates for NSTX (left) and coils and (conducting) blanket modules proposed for ITER.

• NSTX is the only spherical torus with advanced mode stabilization tools. Its set of external control coils is now being used for error field correction and active mode stabilization and has already proved valuable by reducing the low-density threshold for locked modes. Complementing the control coils are extensive magnetic measurement capabilities for n = 1 – 3, with which both locked and slowly rotating RWM's have been observed. Research on mode stabilization at low aspect ratio and low field challenge theory in ways that are unique and complementary to moderate aspect ratio research. The mode control coils on NSTX are similar those being considered for ITER in that they are external mid-plane coils, and are near the (blanket-like) passive stabilizer plates. This is shown in Fig. 2.2.

• NSTX spans the widest range of Alfvén Mach number, up to 0.5, of any toroidal confinement device. This extends those high enough to impact mode growth and saturation, enabling one-of-a-kind tests of nonlinear theory of MHD and its interaction with flows.

• NSTX is the only spherical torus facility with current profile measurements, the result of a major research and development effort that advanced the state of the art of MSE measurements to low magnetic field. The MSE data are combined with kinetic profile measurements in equilibrium reconstruction which includes the effects of plasma flow, enabling detailed analyses of MHD stability that are unique in ST research.

• NSTX has the unique capability to measure the profiles of ion temperature and toroidal flow velocity with resolution at the ion-gyroradius scale, providing insight into the physics of rotation dissipation and coupling to resistive wall modes and tearing modes.

• The large ratio of poloidal to toroidal magnetic field is predicted to enhance rotation damping from internal modes, enabling important tests of mode damping theory.

• The low aspect ratio enables important tests of tearing mode physics theory that are important to ITER and burning plasmas. This is through studies of the impact of the fundamental role of ρ^*, magnetic field curvature and aspect ratio in the conditions for mode onset and evolution. Essential for these studies is a measure of the q profile, and NSTX is the only ST in the world that can measure this.

• X-ray imaging diagnostics measure internal mode structure with both tangential imaging and poloidal tomographic arrays, including fast and slow reconnection events. Combined, these measurements are unsurpassed in their ability to probe mode structure and MHD dynamics. They also enable tests of nonlinear, 2-fluid MHD theory.

• NSTX is the only major device in the world with the capability of coaxial helicity injection. Building on the success of research in the HIT-II device, results on NSTX show promise for CHI as the basis for solenoid-free startup. NSTX experiments on inductive startup using the outer PF coils are unique in the U.S. and complement those performed at MAST, the University of Tokyo and JT-60U. Success in this arena will benefit the spherical torus concept and will validate a key element of advanced tokamak reactor concepts.

2. How do the characteristics of NSTX, when combined with those of C-Mod and DIII-D and other devices in the community, make the U.S. MHD and solenoid-free startup research program unique as a whole in the international program?

• The U.S. capability in resistive wall mode (RWM) physics leads the world, and its contributions are central to developing the physics basis of control for ITER. For example, NSTX and DIII-D plasmas both have similar sound speeds but Alfvén speeds that differ by nearly an order of magnitude. This motivates joint RWM experiments to distinguish between the scaling of the dissipation of the RWM with the sound speed or the Alfvén speed. The MAST device has neither a conducting wall nor an MSE diagnostic for q profile measurements.

• Studies to assess if the stabilizing effects of RWM dissipation are localized to resonant surfaces, or are more global, are underway with joint experiments between NSTX and DIII-D, making use of the excellent resolution of NSTX toroidal rotation measurements.

• The theoretical aspect-ratio dependence of the RWM spectrum is being tested by comparing NSTX and DIII-D. The role of trapped particle effects on ion Landau damping of the RWM is also being investigated in joint studies.

• The active control coils now available on NSTX, C-Mod and DIII-D make it possible to assess the physics and scaling of mode locking. NSTX and C-Mod provide particularly relevant data with regard to the magnetic field scaling of mode locking.

The NSTX capability and research emphasis on solenoid-free startup gives the U.S. a strong, leading position in solenoid-free startup research. NSTX is partnering successfully with researchers at several institutions on this topic. Coaxial helicity injection involves MHD reconnection physics, with its broad applicability both in fusion science and many different astrophysical contexts. Another approach is startup through induction by outboard poloidal field coils. Success in either approach and development of understanding would benefit not only the ST, but also would improve the advanced tokamak as a reactor concept.

3. How well does the NSTX research program cooperate with the international community in coordinating its MHD and solenoid-free startup research, and how have we exploited its special features to contribute to fusion research internationally, in general, and to the ITER design specifically?

• Joint experiments between NSTX and DIII-D are uncovering aspects of resistive wall mode physics on the relative roles of sound speed *vs.* Alfvén speed on mode dynamics. This research, which is only possible by virtue of the complementary characteristics of the devices, will provide direct benefit to ITER design efforts.

• Experiments to use aspect ratio variation to study neoclassical tearing mode seeding and mode coupling are being developed within the ITPA. Understanding NTM physics is of high importance to prediction of ITER performance and the optimization of its operating scenarios.

• The NSTX program participates in an ITPA activity assessing the role of sawtooth control for NTM suppression.

• Low-beta mode locking experiments are being pursued within the ITPA, and NSTX contributes so as to assess the impact of low toroidal field and low aspect ratio on this physics.

• Research in NSTX, with its nearby conducting wall, complements that in MAST, where the wall is much farther away and has a negligible effect on stability. Joint experiments to compare stability limits are being prepared.

• NSTX research on solenoid-free startup using outer poloidal field coil induction is performed in collaboration with the University of Tokyo, which also contributes to such research on the JT-60U tokamak in Japan.

4. How does NSTX contribute to MHD and solenoid-free startup research and the vitality of the U.S. fusion program? What research opportunities would be lost in these areas by stopping research on NSTX?

The unique plasma properties of low aspect ratio, high beta plasmas, and NSTX features such as active mode control coils and strong shaping, make the U.S. MHD research program vibrant and unique in the world. The impact of NSTX is far-reaching. Many opportunities critical for the understanding of ITER plasmas, their control, and the optimization of toroidal confinement concepts generally will be lost if NSTX research is stopped. Viewing critical MHD issues from a perspective that is different from that afforded by higher aspect ratio and lower beta is essential for validating stability theories needed to optimize designs for burning plasmas. NSTX is the only major device in the U.S. pursuing the topic of solenoid-free startup, and its capabilities in CHI are unique in the world program. It is critical not only for the ST, but its success is assumed in the advanced tokamak vision of a burning plasma. Several approaches are being investigated. This research places the U.S. in a world-leading position, which would be lost without NSTX.

Specific opportunities central to maintaining U.S. leadership in MHD and startup research, and which will be lost without NSTX, include the following:

• The low aspect ratio of NSTX, combined with the resultant natural shaping and nearby conducting wall, enhance MHD stability and allow access to the highest beta values on any major toroidal confinement facility. Such high beta regimes allow validation of MHD theory.

• Linear and nonlinear MHD theory is being tested by NSTX, which provides the widest range of Alfvén Mach numbers and the highest beta on any major fusion facility, and measurements of the q profile that are unique for STs.

• Unique, world-leading tests of strong diamagnetic effects on internal mode saturation will not be possible without NSTX research.

• The opportunity will be lost to establish the physics basis of MHD mode stabilization by taking advantage of key aspects of this research that are complementary to those on DIII-D (e.g. similar sound speed, but different Alfvén Mach number) as well as routine excitation of $n > 1$ modes, as is expected for ITER.

• The low aspect ratio enables important tests of the theory tearing modes that is central to the design of ITER and future burning plasmas.

• Unique tangential x-ray imaging measurements of internal mode structure would be lost.

• NSTX enables the U.S. to maintain a leadership role in solenoid-free startup research, which is essential both for the progress of the ST concept and for the advanced tokamak as a reactor.

III. Core Transport and Turbulence

The unique properties of NSTX plasmas make them scientifically powerful both for the study of issues of importance to burning plasmas and for developing a fundamental understanding of turbulent transport in toroidal confinement systems. NSTX started operation with promising predictions regarding its transport, namely, that ion thermal transport might approach the collisional transport limit in many circumstances as a result of two effects on ion scale turbulence. First, theory indicated that the increased region of good curvature of the ST geometry compared to that of higher aspect ratio devices would be intrinsically stabilizing. Second, it was predicted that ExB flow shear in the ST would be large compared to instability growth rates. In fact, ion thermal transport at or even below levels predicted by neoclassical theory is often inferred from power balance analyses of neutral beam heated plasmas with strong rotation in NSTX. The fact that theory predicted this general characteristic of NSTX plasmas is an indication of the productive and exciting interplay between NSTX research and leading theoretical efforts.

1. What are the unique and complementary characteristics of the NSTX facility that enable important advances in core transport and turbulence research? The plasma and device properties enabling unique and powerful studies of electron, ion, particle, and momentum transport include the following:

• The wide range of beta tests confinement predictions in new regimes. It provides powerful leverage on the subject of the beta dependence of confinement, an issue of critical importance to ITER and burning plasmas in general.

• Electron thermal transport is readily diagnosed on NSTX due to the unique, dominant beam-heating of electrons, similar to alpha heating in a burning plasma. This transport, which is of critical importance to predicting performance in ITER, is highlighted in the FESAC Priorities Report and is cited by the international transport community as needing qualitative advances in measurement and theory understanding. Generally, electron thermal transport is dominant on NSTX. Theoretical models suggest that long wavelength turbulence suppression may be responsible for the ion transport being small, leading to this situation. The role of radial streamers in electron transport is the subject of much discussion in the community. The high toroidicity of NSTX is predicted to increase the importance of nonlinear processes that may generate them, and NSTX is in an excellent position to diagnose them. High beta is predicted to make electromagnetic effects particularly important in many NSTX plasmas. These may be important in electron thermal transport as well. Thus NSTX is an ideal facility for addressing issues currently at the cutting edge of discussion and debate in the community.

• NSTX is the only ST in the world with a measurement of the q profile. A major research and development effort was undertaken to enable magnetic pitch angle measurements to be made at low field. Transport studies on NSTX are now taking advantage of knowledge of the q profile, which is both predicted and observed to be of critical importance in the transport properties of toroidal plasmas. Being developed is a novel MSE diagnostic that eliminates the complication of radial electric fields on the measurement through the use of laser-induced fluorescence (LIF). This diagnostic will provide local measurements of the magnitude of the magnetic field as well.

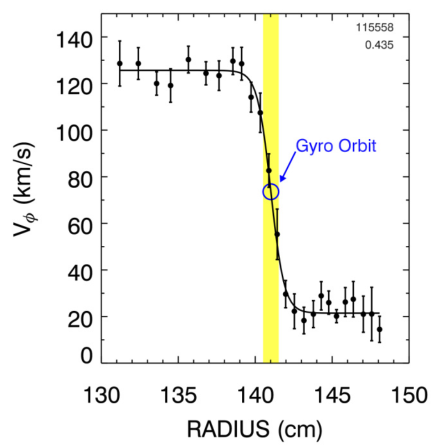


Fig. 3.1 Profile of V_ϕ, measured with CHERS, showing a shear layer with a scale length comparable to the ion gyro orbit size. The shear layer is formed between two magnetic islands.

• Measurements of the ion toroidal flow velocity profile in NSTX can exhibit gradient scale lengths that are on the order of the thermal ion gyro-orbits, as shown in Fig. 3.1. No other toroidal confinement device can study ion transport properties on the scale of the motion that drives the turbulence. This capability is enabled by the routine access to high beta, the moderate toroidal field, and the design of the charge-exchange spectroscopy system that achieves a spot size smaller than the gyroradius.

• NSTX presents an unprecedented opportunity for the study of ion- and electron- scale turbulence dynamics, opening the possibility of comparing fluctuation measurements to advanced turbulence simulations that include the interactions between ion- and electron- scale turbulence, an issue that is at the leading edge of theory development. The nature of electron-scale turbulence and its role in electron thermal transport is an issue of vital importance to ITER, in which alpha-heating will be primarily on electrons. The tangential access in NSTX, combined with the high beta, makes possible measurements of electron scale turbulence with excellent spatial resolution and sensitivity. A scattering diagnostic that takes advantage of these characteristics is being installed on NSTX this year. The generous access in NSTX will also allow new advances in the imaging of core ion-scale turbulence to be implemented in the next five years. A microwave imaging will allow reconstruction of the critical surface position as a

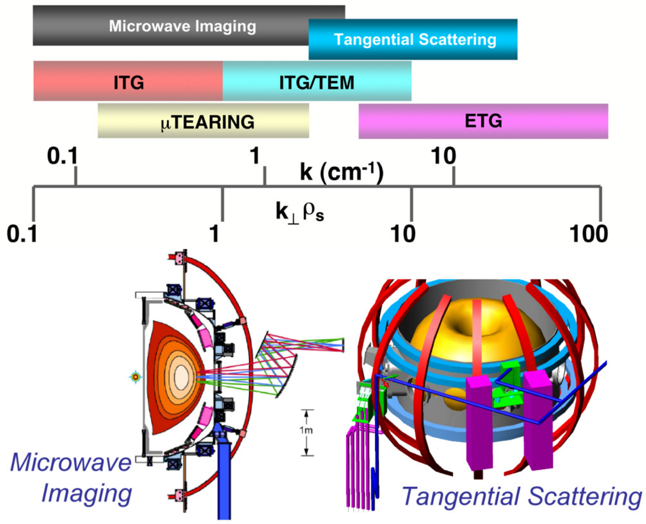


Fig. 3.2 The ranges of radial wavenumber that will be covered by the microwave imaging and tangential scattering diagnostics. Both systems (below) take advantage of the large port access and slender toroidal field coils.in NSTX

function of frequency to assemble a fully two-dimensional image of turbulent structures. Together with the scattering measurements and existing single-point correllation reflectometry, these systems, shown schematically in Fig. 3.2, will provide a diagnostic suite capable of simultaneous, localized measurements of ion- and electron- scale turbulence that is unrivaled in measurement accuracy.

• The role of electromagnetic effects on turbulence dynamics is a leading question in turbulence simulation. At low beta, purely electrostatic microinstabilities are predicted by theory but as the local beta increased, the instabilities are expected to develop a strongly electromagnetic character. This permits unique research using NSTX to validate important aspects of turbulence simulations and theory that are relevant to ITER and all toroidal confinement systems.

2. How do the characteristics of NSTX, when combined with those of C-Mod and DIII-D, combined to make the U.S. core transport and turbulence research program unique as a whole in the international program?

• The three major U.S. devices have and are developing a suite of diagnostics for core turbulence unrivaled in the world. In addition to the tangential scattering being deployed on NSTX, which will provide excellent spatial resolution, collective scattering measurements for electron-scale turbulence are under development on DIII-D and C-Mod. Ion-scale turbulence measurements are made with FIR scattering, beam emission spectroscopy, and single-point reflectometry on DIII-D. Correlation reflectometry is also used on NSTX and will be expanded to ion-scale turbulence imaging. Measurements of ion-scale turbulence with beam emission spectroscopy are underway on C-Mod. These diagnostics are used on plasmas that span wide ranges in both dimensional parameters (*e.g.*, the magnetic field ranges from 0.3 to 8T), and dimensionless parameters that govern the underlying physics. The combination of facilities and capabilities places the U.S. in a leadership position in turbulence measurement and theory comparison.

• The similarity in size and shape of the DIII-D and NSTX poloidal cross-sections provides the opportunity to test directly the role of toroidicity in confinement. Such comparative studies offer the potential to understand the impact of key dimensionless differences on plasma transport phenomena. This understanding is important for ITER and for concept innovation.

• The three devices differ in complementary ways with respect to flows and flow shear generation. DIII-D and NSTX have neutral beam injection. DIII-D is implementing balanced and counter neutral beam injection, which will allow a change of sign of the rotation velocity and its shear, similar to earlier studies on TFTR. Unidirectional beam injection on NSTX, combined with the braking by the correction coils and/or strong electron heating with HHFW, can vary the

Mach number from nearly zero to unity. C-Mod has no direct source of momentum, yet observes flows in the core. Together, the three facilities provide the capability to understand momentum transport and the flows needed for burning plasma optimization.

• A novel core poloidal rotation diagnostic on NSTX will complement that developed on DIII-D, and test key assumptions in the interpretation of this data by providing opposing vertical views of the same volume elements. This will supplement the NSTX edge poloidal rotation measurements that have already revealed the presence of RF-driven flows and absorption of fast waves by thermal ions. The core diagnostic will build on the experience developed with these measurements and modeling on TFTR to provide a poloidal rotation measurement that is unparalleled in spatial resolution and precision. Differences between DIII-D and NSTX are expected in the poloidal flow damping of neoclassically driven and turbulence-driven flows. These expectations should be testable with joint experiments.

3. How well does the NSTX research program cooperate with the international community in coordinating its core transport and turbulence research, and how have we exploited its special features to contribute to fusion research internationally, in general, and to the ITER design specifically? - Collaborative research is developed through the ITPA and in planning forums conducted regularly by all three facilities within the U.S. as well as overseas. The experiments are chosen to take maximum advantage of the complementary characteristics of the devices.

• The role of aspect ratio in governing transport processes in toroidal plasmas is being addressed in joint studies developed through the ITPA. Understanding of the underlying physics is important for understanding ITER plasma confinement and for concept innovation.

• A joint study between NSTX, DIII-D, and MAST is underway to elucidate the local effect of aspect ratio on confinement while matching other dimensionless parameters. An NSTX H-mode plasma serves as the starting point for this study. Comparison plasmas were recently generated on DIII-D and additional experiments on both NSTX and MAST are planned for this year.

• Recent results on the beta scaling of confinement have yielded valuable additions to an international confinement database organized by the ITPA. NSTX research suggests a favorable scaling with beta. This is contrary to results obtained over the more limited range of beta available on moderate aspect ratio devices; the difference may be a consequence of the large leverage that the wider range of beta accessible on NSTX brings to this research.

• The physics of ion thermal transport barrier formation and dynamics in tokamaks is not fully understood. ST's are expected to allow better discrimination between the contributions of diamagnetic flow and driven flow, which will help resolve this uncertainty. Comparison experiments with DIII-D, AUG, JET, and JT-60U are proposed for 2006 within the ITPA.

• The ITPA has identified momentum transport as a key topic needing research in order to have confidence in projections of the rotation of ITER plasmas. A joint experiment involving NSTX and MAST is underway in support of this research.

4. How does NSTX contribute to core transport and turbulence research and the vitality of the U.S. fusion program? What research opportunities would be lost in these areas by stopping research on NSTX?

• NSTX provides a unique opportunity to study electron thermal transport, the core transport issue that yields the greatest uncertainty in predicting ITER confinement. Fast ions primarily heat electrons on NSTX, as alphas will on ITER, while the low ion transport makes the electron channel dominant and, therefore, more readily measurable. Cessation of the NSTX program would eliminate this opportunity to make progress on this important topic.

• The interplay between ion and electron scales in governing turbulence dynamics is at the leading edge of transport theory and simulation. The excellent diagnostic access on NSTX, made possible by the ST geometry, as well as the comparatively large mode amplitude and spatial extent, resulting from high beta and low magnetic field, yield the opportunity for measuring turbulence on ion and electron scales with excellent spatial and wavenumber resolution. The ability to control the flow shear through the mode control coils and combinations of neutral beam and HHFW heating will make these studies particularly important. This opportunity to validate theory and simulation in this critical arena of multiscale dynamics would be lost with the departure of NSTX.

• The wide range of beta on NSTX is now providing important contributions on the beta scaling of transport. Resolution and clarification of this issue is needed for ITER, and the highest-leverage contributor on this subject will be lost should research on NSTX cease.

• NSTX is the only device that can explore the continuum of electrostatic and electromagnetic effects on turbulence as beta is varied from nearly zero to unity. Admixtures of these effects may be important at lower beta values typical of ITER, and the parameter range of NSTX allows for the isolation of one from the other. Such tests are of critical importance to validate modern simulations, and extend in applicability beyond fusion into the astrophysical realm.

• An understanding of the role of aspect ratio in confinement, both global and local, is critical for optimizing a CTF or a burning plasma step beyond ITER. Efforts to understand the fundamental role of aspect ratio would be severely compromised without the contributions afforded by NSTX.

IV. Energetic Particles and Waves

The super-Alfvénic population of fast ions routinely produced on NSTX, and the strong interplay of MHD modes excited by this population make NSTX a powerful and unique test bed for studies that are highly relevant to ITER and burning plasma physics generally. Two classes of high-frequency fast-ion-induced waves, Compressional Alfvén Eigenmodes and Global Alfvén Eigenmodes were discovered on NSTX and are now predicted to occur in ITER plasmas. An important modification of the fishbone instability, the bounce-resonant fishbone, was also discovered on NSTX. With respect to externally launched waves, NSTX offers a unique opportunity to study electron Bernstein waves (EBW) for driving current via the Ohkawa effect that takes advantage of the high particle trapping intrinsic to the ST. High Harmonic Fast Waves are also being employed for current drive and heating on NSTX, complementing the approaches being pursued elsewhere in the U.S and making the U.S. a world leader in current drive research. The heating and current drive approaches being developed on NSTX will be of benefit in establishing the physics of current drive and wave heating for any device that operates with plasmas that are overdense, *i.e.,* having $\omega_{pe} > \omega_{ce}$.

1. What are the unique and complementary characteristics of the NSTX facility that enable important advances in the physics of energetic particles and waves?

• NSTX is unique in containing a strongly super-Alfvénic ($V_{fast}/V_{Alfvén} = 1 - 4$; Fig. 4.1) fast-ion population, in nearly every neutral beam heated plasma, a feature in common with ITER, where $V_{alpha}/V_{Alfvén} > 1$, and with burning plasmas in general. NSTX is also unique in having a measurement of the q profile by MSE in plasmas with a substantial population of super-Alfvénic ions. The q-profile is critical for understanding fast-ion MHD properties, both in terms of the predicted modes and of their impact.

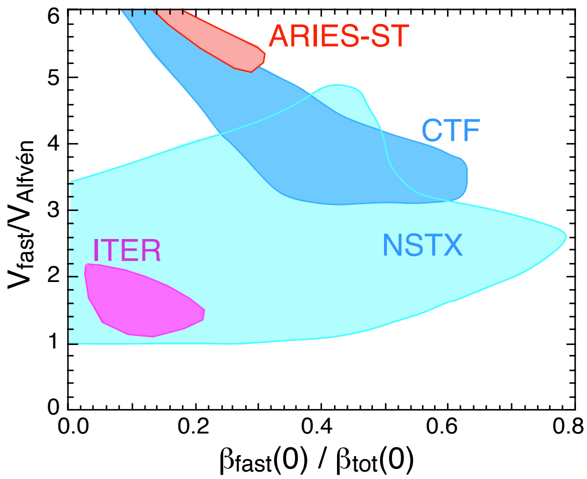


Fig. 4.1 Operating space for fast ions

• The NSTX fast-ion operating space overlaps with and extends well beyond that of ITER with respect to the ratio of fast ion velocity to Alfvén speed as well as the fast ion beta normalized to the total beta, as seen in Fig. 4.1.

• The large fast-ion beta achievable in NSTX, combined with the number of Alfvén eigenmodes available for excitation, yield important similarities with ITER with respect to fast ion MHD drive and nonlinear mode overlap. These combine to enable powerful tests of the theory of alpha-particle driven instabilities and associated alpha-particle transport.

• A wide array of diagnostics on NSTX enables excellent characterization of fast-ion MHD. These include Mirnov coils with fast time response and far-infrared tangential interferometry/polarimetry (FIReTIP) for local mode structure. The latter is supplemented by reflectometry, which will be upgraded to an imaging system, as previously discussed.

• NSTX is the only ST that can investigate the interplay between the current profile and the fast-ion MHD. This is enabled by the combination of its current profile measurements with the capability to measure the local fast-ion energy and spatial distribution using a vertically and horizontally scanning neutral particle analyzer. The tangential infrared interferometer array also provides spectral and spatial information on radial mode structures. The effect of fast-ion MHD on the current profile is of concern to ITER, in particular, its hybrid mode, especially in light of results recently reported from ASDEX-U.

• Fast-ion driven instabilities in NSTX span a wide range of frequency. Low frequency modes, well below the ion cyclotron frequency, may play an important role in the radial transport of energetic particles. This is relevant to alpha-particle transport in ITER and burning plasmas in general. Higher frequency modes may be important in the phase space transport of fast ions on ITER as well.

• Nonlinear interactions between low and high frequency fast-ion-driven modes are routinely observed on NSTX. It is of critical importance to understand these interactions to extrapolate to ITER, where they may play a significant role. In NSTX and ITER, there is also common physics in the overlap of phase space resonances. The large β_{fast} in NSTX contributes to this overlap.

• New classes of high frequency Alfvén eigenmodes, Compressional Alfvén (CAE) and Global Alfvén (GAE) Eigenmodes, were discovered on NSTX. They occur below the ion cyclotron frequency, and are routinely observed. Both may have practical import for ITER performance and the optimization of burning plasmas beyond ITER.

• CAE mode excitation, energy transfer, and control present the possibility of coupling of alpha-particle energy to the fusion fuel in burning plasmas. In addition to providing an excellent test bed for the passive study of CAE physics, NSTX offers the possibility of launching RF waves in the CAE range of frequencies to test the physics of direct transfer of fast-ion energy to the thermal particles with modifications to the HHFW system.

• Validation of CAE physics on NSTX will advance the understanding of solar coronal heating of thermal ions; this mechanism is central to new theories of energy transfer.

• The high particle trapping fraction intrinsic to low aspect ratio provides a unique opportunity for driving current with electron Bernstein waves (EBW) through the Ohkawa effect. This has the potential for generating efficient off-axis current drive crucial to advanced ST operation.

• The coupling, propagation, and deposition of both high harmonic fast waves (HHFWs) and EBWs in overdense plasmas can be studied in NSTX. Measurements are now being made of the coupling of the intrinsic thermal EBW in the NSTX plasma to propagating electromagnetic waves which can be detected outside the plasma, as shown in Fig. 4.2. This research is essential for advancing STs and other lower field toroidal confinement concepts that demand electron heating and current drive tools in overdense plasmas. It also extends the fundamental understanding of wave physics in plasmas in general.

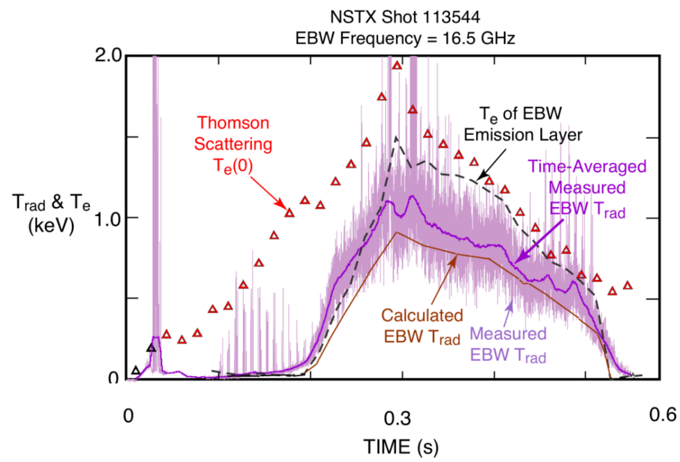


Fig. 4.2 Apparent radiation temperature from mode-converted EBW emission, and theoretical values accounting for the finite acceptance angle of the antenna. Also shown is the central value from Thomson scattering.

2. How do the characteristics of NSTX, together with those of C-Mod and DIII-D, make the U.S. waves and energetic particles research program unique as a whole in the international program? - NSTX, DIII-D, and C-Mod form a powerful combination that is unrivaled with respect to research on energetic particle driven instabilities.

• NSTX spans wide ranges of dimensionless parameters that govern fast-ion instabilities. Its parameter space over which its full diagnostic set is applicable, including MSE, connects with that of DIII-D and includes the full region expected for ITER and beyond to a Component Test Facility. Together, NSTX and DIII-D can explore the ITER operation space and provide an unmatched opportunity for combining their attributes to establish the physics basis for optimization of toroidal confinement concepts. For example, joint experiments have already verified the predicted differences in mode number excited in the two devices as a consequence of the dependence of mode number on q, at common ρ_{fast}/a.

• The capability of exciting fast ion MHD with neutral beam ions and RF on NSTX and DIII-D is complemented by the ability to excite stable Alfvén eigenmodes and measure their damping on Alcator C-Mod. This complements work being done on JET.

• The physics of wave-driven currents is being studied for Lower Hybrid waves (LHCD) on C-Mod, electron cyclotron waves (ECCD) and fast-waves (FWCD) on DIII-D, and high-harmonic fast-waves (HHFW) and electron Bernstein waves (EBW) on NSTX. Together, this represents a comprehensive approach to wave-driven current drive that spans low and high density, higher field plasmas with modest dielectric constant, and lower field, high-dielectric-constant plasmas characteristic of the ST.

3. How well does the NSTX research program cooperate with the international community in coordinating waves and energetic particles research, and how have we exploited its special features to contribute to fusion research internationally, in general, and to the ITER design specifically?

• The NSTX program has been involved in experiments on the MAST device, using a neutral particle analyzer loaned from NSTX to the MAST group. This system enables measurements of the energy distribution of fast ions, including the impact of core MHD on this distribution.

• The EBW research program on NSTX is performed in close collaboration with EBW research being performed on the MAST device. This joint effort encompasses both experiments and modeling and is key both for establishing the physics basis of this current drive method for designing the system to be employed on NSTX.

• NSTX researchers collaborate with the JET program on theory development and experimental tests of the physics of energetic particle modes such as high frequency fishbones. Kinetic theory of these instabilities developed to address NSTX observations is now being applied to JET.

4. How does NSTX contribute to waves and energetic particles research and the vitality of the U.S. fusion program? What research opportunities would be lost in these areas by stopping research on NSTX? NSTX plasmas provide an unmatched laboratory for research on energetic particles in burning plasmas in general and ITER in particular. Its research on current drive is required for establishing the physics basis for concept optimization. Cessation of NSTX research would imperil U.S. leadership in both areas.

• NSTX research on energetic particles places the U.S. program in a leadership position for many of the central research questions pertaining to burning plasmas. It does so by providing the widest parameter space with respect to fast-ion MHD of any facility in the world, its unique diagnostic capabilities, including the only q profile measurements in plasmas with substantial super-Alfvénic populations, and its strong interactions with the theory community. NSTX can study fast-ion physics in regimes that both overlap with ITER and complement it. This would be lost if research on NSTX were stopped.

• NSTX and DIII-D, together with ASDEX-U, are complementary for fast-ion research by virtue of their similar shapes and neutral beam energies but differing aspect ratios and fields. The scientific leverage afforded by this will be severely weakened without NSTX.

• Through its focus on EBW and HHFW, NSTX research over the next five years will establish the physics basis for current drive and electron heating in overdense plasmas. NSTX is ideally suited to exploring the promise of Ohkawa current drive through its research with EBW. This places the U.S. in a leadership role worldwide which would be lost if the NSTX program were stopped.

V. Boundary physics and technology

1. What are the unique and complementary characteristics of the NSTX facility that enable important advances in boundary physics and technology research?

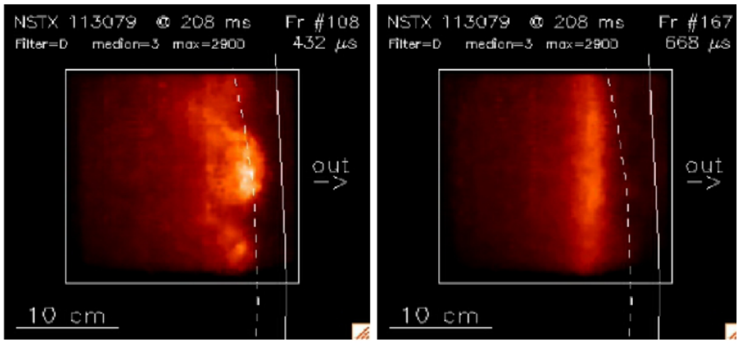


Fig. 5.1 Images of turbulent structures near the NSTX scrapeoff layer, before (left) and after an L-H transition

• The state-of-the-art imaging of turbulence in the scrape-off layer in NSTX, illustrated in Fig. 5.1, takes advantage of the slower evolution of turbulent structures compared to higher field devices, enabling detailed evolution of intermittent structures ("blobs") to be better followed in time and compared to simulations. In addition to the fundamental physics probed by these measurement, the edge transport community regards characterizing edge turbulence as being important to several issues for ITER, including tritium inventory, particle management in general, and heat flux management.

• Understanding and controlling the H-mode edge pedestal on ITER, and optimizing it for steps beyond, is a high priority research element in toroidal confinement research. NSTX is uniquely well positioned to study edge pedestal stability aimed at testing second-stability access and its predicted dependence on magnetic geometry. This research will provide challenging tests for stability codes such as ELITE that are presently used to assess ITER pedestal stability.

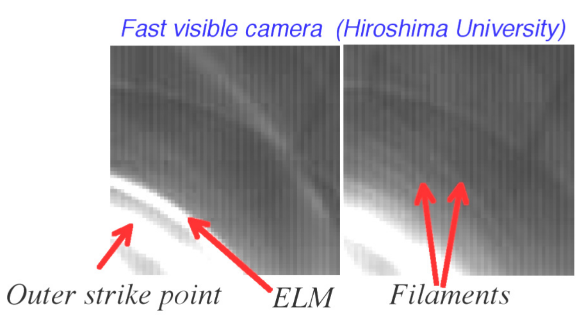


Fig. 5.2 Fast camera images reveal the filamentary structure of ELMs in the lower divertor from the midplane during small Type V ELMs..

• The nature and control of ELMs in the H-mode is a major concern to ITER. The fast cameras and ultra-soft x-ray arrays on NSTX have already revealed new and important aspects of ELM structure, including their filamentary nature (Fig. 5.2) and the strong suggestion that the physics of large ELMs is connected to the turbulent intermittency observed between ELM events and in L modes. The fast cameras also highlight the differences between the large Type I and small Type V ELMs quite clearly. As edge stability is strongly affected by edge magnetic shear, the NSTX shaping capability offers high leverage in assessing ELM physics.

• Heat and particle flux management is essential for the success of ITER and burning plasmas in general. Due to its compact size and high heating power, the divertor heat flux in NSTX can exceed what is expected in ITER, making NSTX a good test bed for heat flux mitigation techniques. In particular, the low aspect ratio, high poloidal field relative to toroidal field, and strong shaping capability on NSTX can be used to test very strong scrape-off layer flux expansion to reduce the heat flux to plasma facing components.

• NSTX measurements of dust formation in the boundary and the deposition of solid films on plasma-facing surfaces are unique. These issues have far-reaching implications, including for ITER, ranging from tritium inventory management to protection of diagnostic windows.

• Plans on NSTX for advanced fueling include a supersonic gas injector which is coming on line at this time, and a multiple-pellet injection system. Note that high-beta plasmas already achieved in NSTX provide the unique opportunity to inject pellets into a minimum |B| configuration, where grad B points outwards in major radius starting just a few cm in from the plasma surface. Finally, NSTX is maintaining the option to test CT injection, as requested by the ITPA.

• Among the three major U.S. facilities, NSTX is unique in its study of lithium coatings and its planned exploration of liquid lithium for particle and heat flux management. NSTX is now utilizing its multi-barrel injector for pellets of room-temperature solids to inject lithium pellets for coating the plasma facing surfaces, similar to the technique that produced dramatic changes in confinement in TFTR. Liquid lithium research in the technology program and in LTX is aimed at unresolved issues pertaining to its use in a toroidal confinement device. If successful, this research will have profound implications for optimization of burning plasmas in any toroidal confinement configuration. Particularly encouraging is research performed in collaboration with the LTX that indicates that liquid lithium can endure very high heat fluxes without deleterious effects as a result of strong flows calculated to be induced by gradients in the surface tension, which rapidly spread the impinging heat. This heat spreading remains to be tested in the presence of strong magnetic fields, and NSTX can be a test-bed to determine if other issues, such as MHD effects from ELMs or disruptions result in problems for lithium. Success on NSTX could lead to consideration of liquid lithium as a back-up option, perhaps for a target at the bottom of the divertor Vee on ITER. The physics learned from these studies will also have direct relevance to the understanding of the behavior of melt layers on high-Z divertor targets.

2. How do the characteristics of NSTX, when combined with those of C-Mod and DIII-D, make the U.S. boundary physics and technology program unique as a whole in the international program?

• Together, the edge imaging available on NSTX and C-Mod spans a range of plasma conditions that is unrivaled. Comparisons of the edge turbulence at low *vs.* high fields are the subject of ongoing joint experiments on NSTX and C-Mod. This research sets the standard for edge turbulence studies.

• Theoretical development is motivating joint experiments in NSTX and DIII-D to test predictions of enhanced edge stability with increased edge shear and, in particular, the prediction that the strong shear in the ST enables access to second stability. These tests are facilitated by the ability to match in the two devices the poloidal cross-section size and shape, heating power and available edge pressures. The difference in major radius and, therefore, aspect ratio, affects the magnetic shear and particle trapping, especially near the plasma boundary.

• C-Mod, DIII-D, and NSTX in combination yield a program in plasma facing component research that leads the world in its breadth. The C-Mod and DIII-D programs participate in world-wide research on carbon (in common with JET and JT-60U) and high-Z (in common with ASDEX-U) plasma facing components, respectively. The NSTX investigation of lithium coatings and liquid lithium is one-of-a-kind and creates a breadth for the U.S. that is both necessary and prudent.

3. How well does the NSTX research program cooperate with the international community in boundary physics and technology, and how have we exploited its special features to contribute to fusion research internationally, in general, and to the ITER design specifically?

• The ITPA organization has sponsored the development of joint C-Mod/NSTX research on edge imaging of turbulence. This research has yielded several invited talks and contributions to international conferences drawing on research on both devices.

• The H-mode pedestal research on edge stability in DIII-D and NSTX also has an international partner, MAST. The differences in the edge particle sources between NSTX and MAST, including the large difference in proximity to the wall, is being used to clarify the role of edge fueling in determining pedestal stability characteristics and H mode access.

• NSTX research has extended the understanding of ELMs through images recorded by a fast camera provided through a collaboration with Hiroshima University. The images show that each ELM event has a transient, filamentary structure much like the intermittent SOL turbulence observed in L mode plasmas, with large ELMs comprised of multiple filaments and small ELMs comprised possibly of single filaments. The small ELM regime is of potential relevance to ITER, due to the projected problem of Type I ELM damage to PFCs. This ELM imaging collaboration was recently extended by using the Hiroshima University fast camera on DIII-D to enable comparison between small ELMS on NSTX and DIII-D.

• Specific approaches to edge fueling, including high-field-size fueling, have been developed on NSTX in collaboration with the MAST program. This has enabled better particle control and routine H mode access.

4. How does NSTX contribute to boundary physics and technology research and the vitality of the U.S. fusion program? What research opportunities would be lost in these areas by stopping research on NSTX?

• NSTX provides strong leadership as well as cooperation in the arena of edge imaging of turbulence. In addition, it is the leading ST in the study of SOL turbulence. Cessation of this program would eliminate major components supporting this research, including the magnetic field and shear dependence of this turbulence.

• Information essential to optimizing the edge conditions in H-mode plasmas is being provided by NSTX research. Importantly, the role of edge shear on governing edge stability properties that determine pedestal and ELM characteristics is being studied on NSTX. Loss of NSTX from this research would mean a weakening of the foundations for ITER and for concept innovation.

• NSTX is confronting the subject of edge flux management through its unique geometry and shaping capability. The roles of flux expansion is being tested in NSTX, as well as the effects of particle trapping in the SOL. The opportunity to gain information in this area, needed to make informed choices about the optimization of toroidal confinement concepts, would be lost in the absence of NSTX.

• NSTX is the only device poised to make real-time edge dust measurements. Combined with its surface deposition measurements, NSTX is confronting issues central to the management of tritium inventory, the survivability of plasma facing components, and the protection of diagnostics in ITER. These contributions would be lost if NSTX research is stopped.

• NSTX is investigating unique plasma fueling techniques including supersonic gas injection, pellet injection into minimum |B| geometry and possible CT injection, which would be lost from the world program without NSTX.

• NSTX is also confronting edge heat and particle flux management issues through its unique program on lithium coatings and its investigation of liquid lithium as solution for both particles and heat. This is unique research with the potential of offering a revolutionary solution to burning plasmas in any toroidal confinement configuration if the issues of concern can be understood and managed. The opportunity to pursue this path would be lost in the absence of NSTX research.

VI. A Strategic Option Made Possible by NSTX

In addition to its contributions to fusion science, ITER, and concept innovation for Demo, NSTX research opens a unique and probably vital strategic option for fusion development, that of a cost-effective Component Test Facility (CTF). Here we present a summary of the arguments for such a programmatic element in fusion development, noting that this key strategic option would be significantly delayed or even lost without NSTX research.

CTF Can Provide the Required Engineering and Technological Basis for Demo

The rationale for a CTF stems from experience in the development of fission energy where the construction and operation of low-power facilities enabled nuclear reliability testing of the materials and components that were required for U.S. Naval Nuclear Reactors and commercial fission power plants. International studies have determined that achieving practical fusion power will require a similar development step to be undertaken before embarking on a Demo fusion power plant. Such a step was included in the recent FESAC Fusion Development Path Report and is also included in the DOE Office of Science Strategic Plan for Fusion Energy Sciences.

A CTF must be designed to test the blanket modules that both convert fusion power to useful heat and breed tritium, the divertor modules, the auxiliary heating and current drive launchers, the diagnostic interfaces, and all other chamber components anticipated for Demo, together with the support systems which will ensure safe and reliable operation, including remote handling systems to replace and repair activated components. For these tests to be effective, the CTF performance capabilities presented in the following table are required. For comparison the capabilities of ITER and Demo are also given.

Studies indicate that ITER, because of its limited pulse length and accumulated fluence, can be expected to provide only very limited testing for tritium breeding blankets. To verify the blanket design and performance prior to deciding on a Demo blanket design concept, up to 3 MW-yr/m^2 in accumulated fluence is required. Then, to establish the engineering and reliability database needed before blanket implementation in Demo, up to an additional 3 MW-yr/m^2 in accumulated fluence is needed. A neutron flux of 2 MW/m^2 in CTF, maintained for weeks on test modules that have cross-section areas of a few m^2, can provide these needed testing conditions.

A compact CTF with much smaller fusion power than Demo, and utilizing moderate size components, would be highly desirable. The absence of such a step would burden the Demo with this testing mission, increasing its cost and imposing significant delay due to the need for repeated replacement of a very large blanket system. Furthermore, Demo operation with an early average of 80% effective tritium breeding ratio would consume 56 kg of tritium in achieving 6MW-yr/m^2 of fluence, while a CTF under similar assumptions would consume 3.2 kg. For reference, the world's supply of commercially available tritium, less that required for ITER, is estimated to be 10 – 15 kg.

Fusion Engineering & Technological Science Conditions	ITER	CTF	Demo
Expected fusion power, P_{DT} (MW)	~500	144	~2500
14-MeV neutron energy flux through chamber surface (MW/m^2)	~0.6	~2	~3
Total area of blankets (m^2)	~12	60	~670
Total accumulated 14-MeV neutron energy fluence (MW-yr/m^2)	~0.3	>6	6–20
Duration of sustained 14-MeV neutron interactions (s)	~10^3	>10^7	~10^8
Tritium self-sufficiency	–	>80%	>100%
Power / major radius (MW/m)	24	64	97

The ST is uniquely well suited to the CTF Mission

Conceptual design studies of the ST-based CTF have been performed recently ,based on the earlier concept of a Volume Neutron Source (VNS). Full advantage has been taken of the simplified ST configuration, which results from the slender single-turn center leg of the toroidal field coil operating at moderate fields, the absence of a solenoid, and the natural elongation of low-aspect-ratio plasmas (Figs. 6.1, 6.2). The elimination of an inboard blanket and shield leads to a highly compact device, in which only ~5% of fusion neutrons are lost to the center leg. This combination of consonant features leads to a compact low-power device, unique in comparison with conventional aspect ratio designs that require insulated conductors, shielding, and blanket modules on the low-major-radius side of the plasma. The design permits direct access via remote handling, for all activated components in the chamber.

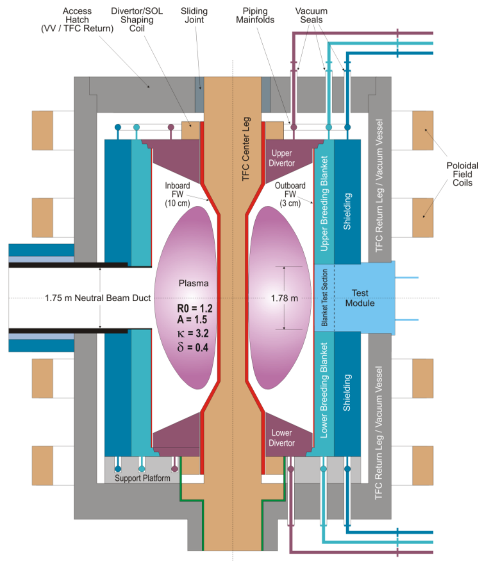


Fig. 6.1. Vertical cross section of CTF..

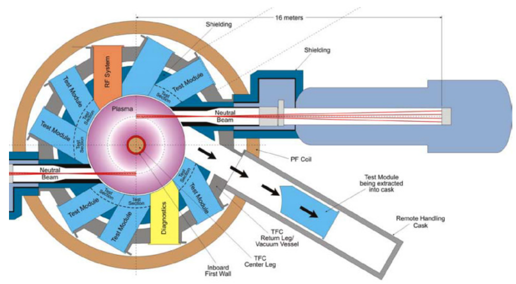


Fig. 6.2. Plan view of a CTF for rapid replacement of mid-plane test modules.

NSTX Is Critical to the CTF Option

Steady-state plasma parameters to satisfy the CTF mission are listed in the table to the right. Although the requirements for the CTF are not as demanding as for an ST power plant, there are very significant challenges. The plasma must be started up without a central solenoid; then the current must be ramped up to and sustained at the multi-MA level utilizing the bootstrap, RF, and beam-driven currents. Toroidal β of 24% must be sustained at a β_N of 3.9 in a rapidly rotating plasma. Current profile control, such as via EBW-CD, will be required in over-dense plasma conditions. Good confinement must be maintained in a plasma heated dominantly by neutral beam injection where $\langle T_i \rangle / \langle T_e \rangle$ ~ 2. Supra-Alfvénic beam ions and alpha particles must be well confined in the likely presence of Alfvén modes. Finally, the normalized divertor heat flux, P/R, while 30% less than required for Demo will be over twice that of ITER. Since each of these issues is being most directly addressed in the world program by NSTX, it is clear that without NSTX this important strategic option for the U.S. would be jeopardized.

Plasma conditions for CTF with R_0 = 1.2m, a = 0.8m, κ = 3.2, and B_{T0} = 2.5T

Normalized Parameters	CTF Conditions
I_p/aB_T (MA/m-T)	6.4
q_{cyl}	3.0
β_N (%-m-T/MA)	3.9
β_T (%)	24
n_{GW} (%)	17
H_{98}	1.5
f_{BS} (%)	43
I_p/I_{TF}	0.85
P/R (MW/m)	64
$\langle T_i \rangle / \langle T_e \rangle$	2
Q	3

VII. The NSTX Facility

7.1 Most Powerful and Best Diagnosed Spherical Torus - NSTX is a major component of the U.S. Fusion Energy Sciences Program, built to investigate the innovative ST concept in toroidal magnetic confinement and to strengthen, through elucidating the underlying plasma physics, the scientific basis for magnetic confinement fusion. Since its initial operation in 1999, 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 2.7 and the highest shaping factor $q_{95}(I_p/aB_T) \approx 40$ MA/m·T of any toroidal device. This greatly expanded operating space has produced a world record toroidal beta of ~ 40% in a high temperature fusion 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 field utilization of the ST configuration.

Because the NSTX center-stack is removable, ready access is available to any part of the device. With the encouraging high elongation results in 2004, this enabled the timely installation of a new set of upper and lower inner PF1A coils in early 2005. The NSTX plasma is also highly accessible due to the large number of diagnostic access ports. Because the large mid-plane ports are placed close to the plasma and because of the compact outer TF coils, NSTX provides unique direct tangential access for a wide range of diagnostics. A comprehensive suite of diagnostics is now operational, many of them unique in their capabilities providing physics information with unparalleled accuracy and resolution. 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 CTF. Together, NSTX, Alcator C-Mod and DIII-D can investigate advanced operation at plasma currents in the MA range in wide range of dimensionless plasma parameters.

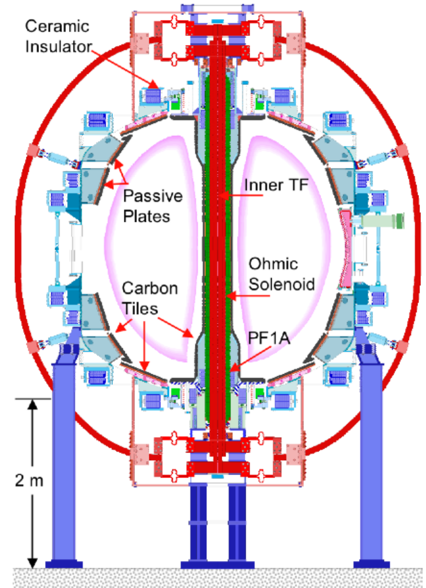


Fig.7.1 Cross section through NSTX

7.2 NSTX is supported by world-class infrastructure – The NSTX is located in a well-shielded test cell at PPPL utilizing many of the former TFTR facilities. The power systems provide reliable operational support for the magnets and auxiliary heating and current drive systems. A TFTR neutral beam injector with three sources was installed on NSTX in 2001 with a rated power of 5 MW at 80 kV. The NBI system has been the work horse of the high-beta experiments and has now delivered up to 7 MW by increasing the acceleration voltage of some sources to 100 kV. NSTX also utilizes the TFTR 30 MHz RF system for high-harmonic fast-wave (HHFW) heating and current drive; this system has delivered up to 6 MW to the plasma. The energy confinement time in NSTX has exceeded 100ms, and the stored energy has exceeded 400 kJ. The maximum toroidal b achieved is approximately 40%.

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.

7.3 NSTX is a vibrant collaborative research facility – Being a world-leading ST facility, NSTX attracts a large numver of national and international researchers with 55 institutions collaborating on the research program. Nationally, the team members are from 29 universities, national laboratories, and industries. Largely through competitive, peer-reviewed research grants, the national team members bring to NSTX highly valuable expertise, major plasma diagnostics, computational tools and software. In addition, there are a number of research collaborators who 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 who bring both expertise and unique diagnostics to NSTX. Most are researchers working on the ST concept in their respective countries but many are 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.

	PPPL	Non-PPPL
Research Staff	50	121
Post Doctoral Staff	0	7
Graduate Students	6	9
Undergraduates		5

Table 7.1

Since its first plasma in 1999, NSTX has achieved its facility and operational milestones, except in FY 03 when a TF joint failed, causing a loss of 8 run weeks. Improvements to the TF joints have since been implemented, and they are now performing to the design specification. After the initial shake down and ramp-up period, the NSTX team has been very productive in scientific/technical publications, averaging 86 publications per year for 2003-2004. According to data from the Web of Science, the number of publications from NSTX during the its first five years of operation compares favorably with those of other fusion research facilities of comparable size brought on line since 1980.

7.4 MHD Research – The NSTX plasma is surrounded by closely-fitting conducting plates, which can 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 the no-wall beta limit. In recent experiments, the new shaping coils produced plasmas with high elongation, $\kappa \sim 2.7$, and triangularity, $\delta \sim 0.8$ Examples of the plasma shapes obtained with the old and new coil sets are shown in Fig. 7.2.

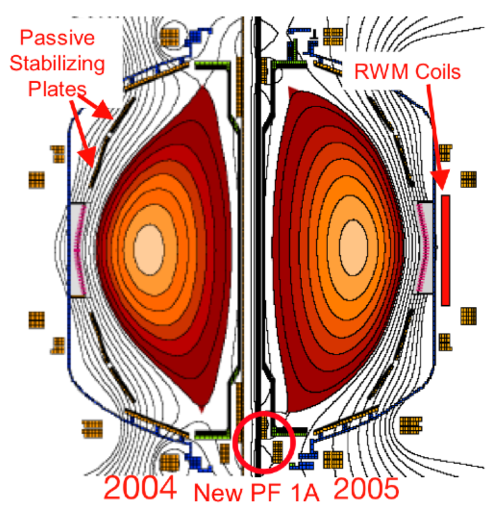


Fig. 7.2. New inner shaping coils and RWM Coils for the FY 05 operations

MHD equilibrium reconstruction is a central tool for toroidal plasma research. A major research effort, led by a Columbia University researcher, has implemented between-shots plasma reconstruction with capabilities beyond those available elsewhere. Utilizing extensive magnetic data together with the plasma thermal and rotational profile information along the mid-plane, the plasma reconstruction code “EFIT”, extensively extended and adapted to NSTX plasma conditions, provides detailed information on the basic plasma properties with time resolution typically down to 1 ms. The EFIT analysis runs very efficiently and is available in a few minutes after a shot. In 2004, plasma rotation measurements were incorporated into this analysis and this year the MSE field pitch angle measurements have been added into EFIT to constrain the q-profile. The EFIT analysis constrained by MSE data is already yielding very important and exciting and important 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 stability. The NSTX magnetic reconstruction technique, which takes into account the 2-D eddy currents in the walls and structure, is contributing to the ITER magnetics design.

To explore plasma operation near the ideal-wall limits, three pairs of coils to produce controllable radial magnetic field perturbations have now been installed on NSTX as shown in Fig. 7.2. 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, with the coils of each pair diametrically opposite each other wired with opposing phase. These coil pairs 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 complete system was commissioned at the start of the 2005 experimental run. Experience with the NSTX EF/RWM coil system will contribute to the proposed ITER design, since the NSTX coil locations are similar to those proposed for ITER and, in ITER, the blanket modules should slow down the instability growth similarly to the NSTX passive plates. NSTX EF/RWM research benefits from a sophisticated array of magnetic sensors, itself unique in the world. The 51-channel toroidal charge-exchange recombination spectroscopy (CHERS) system provides critical rotation data for understanding RWM stabilization with excellent spatial resolution. A very fast (600 kHz) two-color ultra-soft x-ray array routinely monitors MHD mode activity as well as impurity and electron thermal transport. A tangential soft-x-ray camera, which can capture 2-D MHD behavior with an unprecedented time resolution of 2 μsec, is capable of resolving, for example rapid internal MHD reconnection events.

In order to develop attractive ST power plants, a practical solenoid-free start-up technique must be developed. The main technique tested to date in NSTX is coaxial helicity injection (CHI), an NSTX capability unique among large-scale facilities because the machine is designed with in-out electrical insulation. This technique has been investigated previously in smaller devices including HIT/HIT-II (U. Washington). Experiments to develop CHI were performed in NSTX in 2004 using a large capacitor bank. These generated toroidal plasma currents of up to 150 kA, 40 times larger than the injected current. In parallel to this, NSTX is also investigating techniques to start the plasma using only the outer PF coils; initial experiments have shown promise. If, through these methods of operation, substantial solenoid flux savings can be realized, with appropriate insulation designs (*e.g,*. using silicon carbide as a dielectric), this technique can be applied to fusion power producing devices.

7.6 Transport & Turbulence Research – For transport studies, NSTX has developed excellent plasma profile diagnostics. The large vacuum window allows the multi-pulse Thomson scattering (MPTS) system to measure plasma density and temperature with unprecedented accuracy at a rate of 60 profiles per second (upgradable to 90 per second). An upgrade of the

MPTS to 30 spatial points (45 ultimately) currently underway will improve resolution of the structure of the H-mode and internal transport barriers. An edge reflectometer measures the edge density profiles to complement MPTS. The 51-channel toroidal charge-exchange recombination spectroscopy (CHERS) system measures the ion temperature and toroidal flow velocity with spatial resolution down to the typical deuteron gyroradius, as previously shown in Sec. III. An initial CHERS system to measure the poloidal flow will be implemented in 2006 upgradable to 50-channel capability. A major diagnostic addition this year is the Motional Stark Effect (MSE) current profile measurement with 8 channels (19 channels ultimately.) A MSE system using Laser Induced Fluorescence (LIF) is under development. This measurement, with its radial view will not be affected by electric fields within the plasma, and due to the high β in NSTX will provide a direct, local measurement of the diamagnetic effect. Taking advantage of the excellent tangential access, the tangential far-infrared laser system (FIReTIP) simultaneously measures density and polarimetery profiles with μsec time resolution, a unique capability. NSTX also has an excellent x-ray crystal spectroscopy system, which was adopted by the ITER diagnostic design team; this system also contributes uniquely to astrophysics research. For investigating plasma turbulence, the core correlation reflectometer is yielding important low-k turbulence information. This year, a tangential microwave scattering system is being installed to measure turbulence for a wide range of wave numbers encompassing both electron and some forms of ion turbulence with unparalleled spatial resolution at high k. Taking advantage of the large ports, an imaging reflectometer to reconstruct full images of the turbulence in the plasma core in the ITG wavelength range and above is planned. With these state-of-the-art profile and turbulence diagnostics, together with theory and modeling, NSTX is well positioned to contribute to the fundamental understanding of plasma transport needed to develop predictive capability for devices such as ITER and beyond.

7.7 Energetic particle research - NSTX is uniquely positioned to investigate the ITER relevant energetic-particle instabilities and their consequences. NSTX naturally covers a wide range of β_{fast}/β_{tot} and energetic-particle pressures, including the expected ITER operating space. This will be studied with the extensive NSTX plasma diagnostics, including MSE for the j(r) profile, available at the normal NSTX field strength at which $V_f > V_A$. In addition, the application of HHFW can modify the energetic particle velocity distribution. The toroidally and poloidally scanning neutral particle analyzer provides detailed velocity and spatial information about the energetic particles and their transport. The Scintillator Fast Lost Ion Probe on NSTX is unique in measuring the loss of fast ions with pitch angle and energy resolution, important for investigating a wide range of MHD-induced fast ion losses. The diagnostic capability of excited high frequency modes in the $V_{fast} > V_A$ regime is world leading with high frequency magnetic pick up coil array, correlation reflectometer, tangential FIR interferometer array to measure mode structure, an ultra-fast x-ray camera, and ultra-soft x-ray tomography. The NSTX facility therefore offers world-leading unique capability to investigate energetic particle physics critically important for ITER and concept innovation.

7.8 HHFW and EBW Heating and Current Drive – NSTX is investigating two types of RF waves for heating and current drive. Radio-frequency plasma waves at high harmonics of the ion-cyclotron frequency (HHFW) are expected to heat electrons in an ST to high temperatures and to contribute to sustaining the plasma current. A twelve-element-antenna system in NSTX is driven by six power amplifiers operating at 30 MHz with delivered power of up to 6 MW and source power of 10 MW. This is the world's most sophisticated high power ICRF antenna array, tilizing an RF matching network developed by ORNL, the phase of the RF current in the six pairs is variable in real-time to control $k_{||}$ for optimizing the wave absorption as the discharge evolves. The core electron temperature has been increased from 0.4 keV to 4 keV using 3 MW of power. 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 thermal and fast ion absorption may limit the current drive efficiency of HHFW. This physics will now be investigated more fully with MSE.

Electron Bernstein Waves (EBW) are a very promising tool to drive well-localized off-axis current needed for the advanced ST. Modeling predicts very high current drive efficiency two to three times greater than Fisch-Boozer ECCD, via the Ohkawa effect. EBW emission measurements show good coupling efficiency to the EBW. NSTX plans to install a 1 MW EBW system to demonstrate EBW Ohkawa current drive, expected to be ~35 kA, and then a 4 MW system to drive over 100 kA of localized current for sustained operation at simultaneously high $\beta_T \sim 40\%$ and high bootstrap current fraction ~60%. Modeling shows that this will provide sufficient margin for a wide range of physics studies.

7.9 Boundary Physics – The achievement of good vacuum and surface conditions on the plasma-facing components (PFCs) has been crucial to the progress in plasma performance in NSTX. NSTX employs state-of-the-art wall conditioning capabilities including 350°C bake out of PFCs, wall boronization, and between-shots helium glow to control the particle wall loading. This combination of methods both suppresses oxygen impurities and reduces the ratio of the hydrogen to deuterium concentration to below 0.05 in plasmas with deuterium fueling. In order to develop active particle control capability to sustain an advanced regime, NSTX is investigating lithium based techniques. In initial experiments, lithium pellets injected into NSTX plasmas further reduced oxygen impurities. An extensive experimental campaign utilizing the lithium pellet injector is planned for FY 05. A lithium evaporator to be tested on LTX will be installed on NSTX in FY 06 to test the effect of more extensive lithium wall coating. The recent results from LTX indicating that liquid lithium surfaces can handle very high heat flux, due to convective flows, gives impetus to the option for a liquid lithium divertor target. It suggests that liquid lithium technology may have application to very-high-power density systems, and should be investigated in NSTX as a back-up to the two options currently anticipated for ITER, carbon and tungsten, both of which present serious issues. Since NSTX can produce even higher local heat fluxes than anticipated for ITER normal operation, over 10 MW/m^2 at the divertor plate, this can provide an excellent proof-of-principle test for this concept.

The capabilities for gas fueling the NSTX plasma have been steadily enhanced with gas injectors in several locations, including four outer mid-plane, inboard mid-plane and divertor regions. A new addition is an outer mid-plane supersonic gas injector, which introduces gas at a velocity of 2.4 km/s. High-speed camera images confirmed that a collimated stream of deuterium entered the plasma from the nozzle. The NSTX plan for fueling includes both a frozen deuterium pellet and the option for a compact toroid (CT) injector. Interestingly high beta plasmas on NSTX present a minimum B target for outside-launch pellet injection, with grad B pointing outwards a few cm inside the plasma edge. The NSTX field structure is also well suited to test controlled fuel deposition by the CT injector. The ITPA steady-state team has expressed interest in an NSTX demonstration of the CT injector, which has reactor-relevant potential for particle fueling and radial deposition control.

Taking advantage of its large diagnostic access, NSTX is pioneering the investigation of edge turbulence and ELMs using high speed cameras viewing several locations including the divertor region. The relatively large size and slow motion of the edge "blobs" in NSTX together with excellent diagnostic access provide world leading capability for edge turbulence studies. A high speed IR camera is planned for 2007 to quantify the heat flow during rapid MHD events such as

ELMs. Other unique capabilities for NSTX are its dust detector and surface deposition monitor. By monitoring the dust generation and surface deposition in real time, NSTX can contribute to the critical ITER design issue of tritium retention due to dust and the surface coating of diagnostic windows.

7.10 NSTX in the Context of the World ST Program – In the worldwide ST research effort, NSTX and MAST are the only 1 MA–class ST facilities, and are complementary in many ways. With its powerful NBI, NSTX leads the world in high β research, where its passive stabilizing plates and the RWM coils permit investigating plasmas with β approaching the ideal-wall limit. The MAST PF coils are inside of its large vacuum vessel, which permits it to investigate alternative solenoid-free start-up approaches such as growing plasmas off of these coils and then merging them. It also permits a longer outer diverter leg than can be created in standard operation of NSTX. However the MAST configuration severely limits the bake-out temperature of the vacuum vessel and is inconsistent with resistive wall mode stabilization.

The world ST program is well coordinated. In addition to NSTX and MAST, there are 10 smaller scale ST facilities operating worldwide, focusing on various specialized ST-physics issues, including solenoid-free start-up, RF heating, lithium-coated walls, and ultra-low aspect-ratio regimes. Concepts developed on smaller STs, such as CHI, HHFW, and lithium coating, have been brought to NSTX. As near future additions, a modest-current but very long-pulse ST and an ST to study merging start-up are being initiated in Japan. The world ST program is therefore well balanced and well coordinated.

With world leading and unique capabilities, NSTX is well positioned in the world fusion and ST community to make major contributions to advance fusion energy science, ITER design and operations, and to support the development of future options for an ST based CTF and Demo.

VIII.NSTX Contributions to FESAC Priorities

The FESAC Priority Panel identified important ten year goals for the US fusion science community. In the following concluding table, we summarize the unique and important contributions NSTX will make to the Priority Panel ten year goals. As shown below, NSTX will make major unique contributions to the ten year goals. These opportunities would be lost if NSTX were shut down prematurely.

FESAC Priorities Report 10-Year Goals	***Unique and World-Leading NSTX Contributions***
Macroscopic Plasma Physics 1. Understand the coupled dependencies of plasma shape, edge topology, and size on confinement in a range of plasma confinement configurations. 2. Identify the mechanisms whereby internal magnetic structure controls plasma confinement. 3. Identify the effects and consequences on confinement of large self-generated plasma current. 4. Learn how to control the long scale-length instabilities that limit plasma pressure. 5. Understand and control intermediate to short wavelength modes responsible for limiting the plasma pressure, particularly at the edge, and extrapolate their effects to the burning plasma regime. 6. Understand the equilibrium pressure limits in a range of magnetic configurations, including the effects of islands, stochastic magnetic fields, and helical states. 7. Understand and demonstrate the use of self-generated currents and mass flows to achieve steady-state high-pressure confined plasmas and improve fusion energy performance. 8. Understand how external control can lead to improved stability and confinement in sustained plasmas in a range of magnetic configurations. 9. Understand the pressure limits and confinement properties in configurations where magnetic turbulence controls the distribution of the equilibrium magnetic field and for similar configurations with reduced turbulence. Assess their prospects for study in more collisionless plasma regimes for possible extrapolation to practical sustained burning plasmas.	1. The most powerful and most fully diagnosed low aspect ratio magnetic configuration in the world, with the most flexible shaping capability. 2. Unique capability to measure shear and shear reversal at low A, unique magnetic well, trapped particle fraction. 3. Highest bootstrap fraction at low A, T_i and V_ϕ measurements at ρ_i scale for study of NTM's. 4. Close-fitting conducting shell, ITER-like EF/RWM coils, rotation up to 0.5 V_A, decoupling of V_s and V_A, high resolution T_i and V_ϕ measurements for understanding dissipation physics, outstanding magnetic diagnostics. 5. Unique access to high edge shear regimes with Gas Puff Imaging, very fast visible-light camera, edge multi-color Ultra-Soft X-rays, and MSE measurements. Low B allows scaling studies and slows dynamics for imaging. 6. Highest β operation of any major toroidal facility, high resolution V_ϕ measurements very sensitive to island structures, tangential imaging x-ray camera, USX system, detailed magnetic diagnostics for internal structures. 7. Combination of high power, strong rotation, conducting shell, RWM stabilization, EBW current profile control, strong shaping form the basis for NSTX plan to access 100% non-inductive operation near with-wall limit. 8. Most sophisticated external shaping, current profile control, RWM control suite available to test stability and confinement at low A. 9. Only major toroidal facility world-wide with in-out insulation to allow Coaxial Helicity Injected plasmas. Only major facility world-wide able to stabilize CHI plasmas with toroidal field. High available heating power will allow access to collisionless regime.
Multi-Scale Transport Physics 1. Develop predictive capability for ion thermal transport using simulations validated by comparison with fluctuation measurements. 2. Identify the dominant particle transport mechanisms, including the conditions under which pinch/convective processes compete with diffusive processes. 3. Identify the dominant mechanisms for momentum transport and their relationship to thermal transport. 4. Understand generation of flow shear, regulation of turbulence, and self-consistent profile dynamics and local steepening, and to identify conditions and thresholds for edge and core barrier formation. 5. Identify the dominant electron thermal transport mechanisms, including the role of electromagnetic fluctuations, short-scale versus long-scale turbulence, and spectral anisotropy. 6. Identify the dominant driving and damping mechanisms for large-scale and zonal flows, including turbulent stresses and cascades. 7. Identify the dominant mechanisms by which turbulence generates and sustains large-scale magnetic fields in high-temperature plasma. 8. Identify the mechanisms and structure of magnetic reconnection, including the role of turbulent and laminar processes, energy flow, and the production of energetic particles. 9. Identify the conditions for onset of island growth and the factors controlling saturation and coupling with transport.	1. Correlation reflectometer to be upgraded to unique high-resolution 2-D microwave imaging. Access to regimes with neoclassical as well as strongly anomalous ion thermal transport. 2. Unique USX imaging array allows tracking of impurity transport fully across plasma column. Charge exchange spectroscopy allows highly resolved carbon profile. 3. Unique capability to study rotation at speeds approaching Alfvénic, unique capability to study interaction with magnetic perturbations on the ρ_i scale. 4. Unique capability for ρ_i scale measurements of flow shear and T_i profiles, leading capability for high radial resolution measurements of both ion and electron turbulence. 5. Unique capability for high resolution measurements of electron and ion turbulence, including spectral anisotropy, over widest range in β world-wide, allowing access to regimes predicted to have strong e-m effects. 6. Leading capability to drive strong large-scale flows, highly time and space resolved turbulence diagnostics to resolve zonal flow shear effects, as well as interplay between ion and electron turbulence. 7. Unique capability for Coaxial Helicity Injection in a major facility with advanced diagnostics for magnetic structures and thermal profiles, as well as strong heating. 8. Unique combination of MSE and tangential neutral particle analysis as well as fast tangential x-ray imaging to study both tearing mode and disruption physics. 9. Leading capabilities to determine the role of poloidal mode coupling on tearing mode seeding and saturation, Glasser effect on growth, and island effects on transport.

FESAC Priorities Report 10-Year Goals	*Unique and World-Leading NSTX Contributions*
Plasma Boundary Interfaces 1. Predict the expected magnetohydrodynamic stability and plasma parameters for the ITER H-mode edge pedestal with high confidence. This is a time-sensitive issue relevant to the success of ITER 2. Identify the underlying driving mechanisms for mass flow and cross-field transport in the scrape-off-layer plasma, in H-mode attached and detached plasmas. 3. Resolve the key boundary-physics processes governing selection of plasma-facing components for ITER. This is a time-sensitive issue relevant to the success of ITER. 4. Complete the evaluation of candidate plasma-facing materials and technologies for high-power, long-pulse fusion experiments. This is a time-sensitive issue relevant to the success of ITER.	1. Leading capabilities to test stability physics of plasma edge due to wide range of accessible shear, lower B for physics scaling and for raising scale size and slowing dynamics, allowing precise diagnostic measurements. 2. Leading access, space, and time resolution for high-quality gas-puff-imaging measurements, allowing new insights into the physics of SOL transport, L *vs*. H mode edges, and physics of large and small ELMs. 3. Leading divertor heat flux of over 10 MW/m^2 for ITER-like studies of attachment/detachment, erosion physics. Unique real-time dust and surface deposition measurements. 4. Unique plan for staged development of lithium PFCs; pellet injection and evaporative coating leading to liquid lithium divertor target, based on LTX high heat-flux results showing very effective heat spreading.
Waves and Energetic Particles 1. Develop the capability to design high-power electromagnetic wave launching systems that couple efficiently and according to predictions for a wide range of edge conditions. 2. Produce, diagnose in detail, and model with nonlinear, closed-loop simulations the macroscopic plasma responses produced by wave-particle interactions, including localized current generation, plasma flows, and heating, in both axisymmetric and non-axisymmetric configurations. 3. Develop long-pulse radio-frequency wave scenarios for optimizing plasma confinement and stability and to benchmark against models that integrate wave coupling, propagation, and absorption physics with transport codes (including microturbulence and barrier dynamics) and with magnetohydrodynamic stability models. 4. Improve analysis and models to match the experimental measurements and scale the understanding to predict the dynamics of energetic particle-excited modes in advanced regimes of operation with high pressure, inverted magnetic shear, and strong flow. 5. Identify the character of Alfvén turbulence and the evolution of the energetic particle distribution in a nonlinear system, which can be used to predict alpha-particle transport in a burning tokamak experiment; and to evaluate and extrapolate energetic particle behavior in present-day confinement systems to reactor parameters.	1. Unique capabilities in both High Harmonic Fast Wave heating, with 12-strap antenna, and Electron Bernstein Wave Ohkawa current drive. Unique diagnostic results on parametric decay of HHFW waves and edge ion heating and rotation. EBW OKCD allows highly efficient current drive near the plasma edge, as needed for the most advanced scenarios. 2. Unique capability to diagnose current drive in overdense plasmas. Strong capability to measure fast ion tails with toroidally and poloidally scanning Neutral Particle Analysis. Microwave imaging for HHFW propagation. 3. Most complete ST capabilities for long-pulse, non-inductive, very high beta and bootstrap operation, based on transport code simulations of wave dynamics and coupling with free-boundary plasma evolution. 4. Widest range of V_{fast}/V_A and β_{fast}/β_{tot}, overlapping and extending beyond ITER, allowed discovery of new CAE, GAE and bounce-fishbones. High β_{fast}/β_{tot} and high β_{fast} allow widest range of instabilities in high β regimes with high flow, normal and reversed shear. 5. Unique capability for MSE in plasmas with high V_{fast}/V_A and β_{fast}/β_{tot} allows quantitative evaluation of theoretical stability predictions and direct measurement of impact on fast-ion current drive. Scanning Neutral Particle Analysis provides radial and energy resolution of resonant interactions. Tangential interferometer array, imaging reflectometer, and fast tangential and radial x-ray imaging provide analysis of nonlinear mode structure.
Fusion Engineering Science 1. Deliver to ITER the blanket test modules required to understand the behavior of materials and blankets in the integrated fusion environment. 2. Determine the "phase space" of plasma, nuclear, material, and technological conditions in which tritium self-sufficiency and power extraction can be attained. 3. Develop the knowledge base to determine performance limits and identify innovative solutions for the plasma chamber system and materials. 4. Develop the plasma technologies required to support U.S. contributions to ITER. 5. Develop the plasma technologies to support the research program.	1. and 2. While NSTX will not develop blanket test modules, it will provide the U.S. a unique path to lead in the development of fusion nuclear technology for Demo, by developing the physics basis for a compact, cost- and tritium-efficient Component Test Facility. 3. Leading access to divertor heat flux at ITER levels and beyond, allowing the test of innovative plasma chamber systems and materials. 4. Unique plan to test the implications of recent favorable LTX results on the effectiveness of thermal convection in liquid lithium, using a liquid lithium divertor target. Unique real-time dust and surface deposition measurements. 5. X-ray Crystal Spectroscopy diagnostic specifically developed for ITER. Unique test-bed for the current-drive and heating technologies required in the overdense plasmas anticipated in high-beta fusion systems.