GA–A26450

RECENT RESULTS AND PLANNED UPGRADES FOR THE DIII-D TOKAMAK

by
J.T. SCOVILLE for the DIII-D TEAM

JUNE 2009

GENERAL ATOMICS

DISCLAIMER

GA–A26450

RECENT RESULTS AND PLANNED UPGRADES FOR THE DIII-D TOKAMAK

by
J.T. SCOVILLE for the DIII-D TEAM

This is a preprint of a paper to be presented at the 36th Symposium on Fusion Engineering, May 31 through June 5, 2009 in San Diego, California and to be published in the Proceedings.

Work supported by
the U.S. Department of Energy
under DE-FC02-04ER54698

GENERAL ATOMICS PROJECT 30200
JUNE 2009

GENERAL ATOMICS

Recent Results and Planned Upgrades for the DIII-D Tokamak*

J.T. Scoville for the DIII-D Team
General Atomics
P.O. Box 85608
San Diego, California 92186-5608 USA
scoville@fusion.gat.com

Abstract—A major focus of the experimental program on the DIII-D tokamak is the need to better define the optimal design requirements and operating scenarios of future devices such as ITER. Using the flexible DIII-D plasma control system, several of the proposed ITER operating scenarios have been investigated and experiments have been carried out to demonstrate control of a variety of plasma instabilities and mitigate the effects of plasma disruptions. Several major upgrades to the DIII-D device are being planned for the near future to allow research to continue to address critical issues that arise as tokamak research advances into the regime of longer pulse lengths, fully non-inductive current drive, and higher performance operation. The auxiliary heating systems of the tokamak will be upgraded by increasing the ECH power to 12 MW and extending the neutral beam power pulse length to 10 s. Important plasma current profile effects will be addressed by modifying two of the neutral beam injectors to allow up to 10 MW of off-axis heating. Also, a flexible array of feedback control coils will be installed inside the vacuum vessel to continue the investigation of the control of ELMs and other MHD phenomena. A description of these upgrades and a discussion of some of the important recent results will be presented.

Keywords: DIII-D, tokamak, recent results, upgrades

I. Introduction

Research on the DIII-D tokamak is helping lead the way to the development of a fusion power reactor. Experiments are designed to contribute to the general fusion knowledge base, work toward an advanced steady state machine, and provide guidance to the designers of the ITER device [1]. Approximately half of the experiments on DIII-D are dedicated to studying issues that directly impact the ITER design. Fundamental issues are being addressed, such as the formation of the plasma discharge and the physics parameter regimes in which the plasma is maintained. The so-called ITER baseline, advanced inductive, hybrid, and steady state scenarios have been demonstrated and performance metrics have been documented and scaled to the ITER device.

Plasma control issues critical to ITER are being addressed as well. Experiments exploring control of instabilities have focused on edge localized modes (ELMs), neoclassical tearing modes (NTMs), resistive wall modes (RWMs), and disruptions, each of which can threaten the integrity of the ITER vacuum vessel and first wall. The importance of the details of plasma profiles and how they influence the high-energy confinement H-mode have also been investigated employing feedback control of neutral beam and electron cyclotron heating (ECH) systems. Simultaneous stabilization of deleterious NTMs and RWMs has been demonstrated. DIII-D experiments have investigated the use of disruption mitigation using massive gas puffs and pellet injection in order to control the power flux to the walls and avoid the generation of runaway electrons during disruptions. The advanced plasma control system on DIII-D is being expanded routinely to allow more sophisticated shape control, feedback control of instabilities and many other plasma parameters relevant to ITER.

Upgrades to the DIII-D machine are planned that will allow improved study of emerging physics issues in plasmas that can be extrapolated to the ITER device. Auxiliary heating systems are being increased in both power and pulse duration and off-axis current drive will be possible with not only ECH, but modified neutral beam injectors as well. A new internal coil system will be installed on the vessel centerpost wall that will allow simultaneous control of multiple plasma instabilities.

Section II presents a brief summary of a few of the results obtained during recent DIII-D experiments and Section III describes some of the planned upgrades. Conclusions are presented in Section IV.

II. Recent Results

The success of the ITER tokamak is important to the world-wide fusion research community. The DIII-D tokamak is positioned strategically within the research program to greatly contribute to that success. Experiments covering a variety of ITER-related research topics are being carried out at DIII-D; one of the most fundamental is an investigation of the basic physics parameter ranges of the future machine. Four distinct operating scenarios have been investigated in detail on the DIII-D tokamak, including the conventional H-mode baseline scenario, the steady state scenario, the “hybrid” scenario, and the advanced inductive scenario (see Fig. 1). The goal of these experiments is to match the ITER plasma cross-section and aspect ratio, but with scaled down plasma size, and then measure the optimum performance of the discharge. In each of the scenarios, DIII-D was able to produce plasmas with high performance, scaling to ITER’s goals for a power factor $Q \equiv P_{fus}/P_{input}$ of $Q = 10$ ($Q = 5$ for steady state). Performance is measured by the fusion gain, $G \equiv \beta_N H_{89}/q_{95}^2$, where $\beta_N = \beta(aB/I)$ is the normalized beta and H_{89} is energy confinement time normalized to the ITER89P L-mode scaling [2]. The ability to create such ITER-relevant discharges provides a useful platform for projections of fusion performance and tests of plasma control.

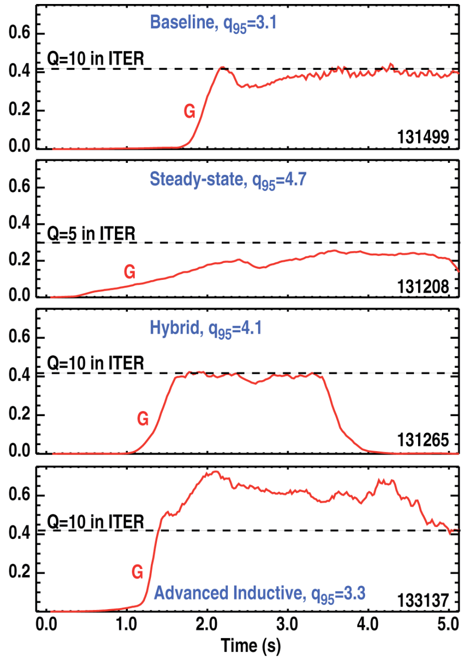

Figure 1. ITER demonstration discharges for the baseline scenario with $q_{95} \approx 3.1$ and $\beta_N \approx 2$, steady-state scenario with $q_{95} \approx 4.7$ and $\beta_N \approx 2.8-3.0$, hybrid scenario with $q_{95} \approx 4.1$ and $\beta_N \approx 2.8$, and advanced inductive scenario with $q_{95} \approx 3.3$ and $\beta_N \approx 2.8$.

Robust control of a number of MHD instabilities is a prerequisite for the safe operation of a high-powered tokamak such as ITER. Modes such as the RWM (kink mode) and NTMs (tearing modes) can both be excited in high performance plasmas and threaten plasma stability or limit the performance of the discharge. DIII-D is equipped with a variety of tools with which to investigate the control of these instabilities using feedback control of plasma rotation, stored energy, resonant magnetic perturbations, and the current profile [3]. An example is shown in Fig. 2, where the RWM and NTM were stabilized simultaneously for long duration (> 2 s). After the ECCD was terminated, plasma rotation slowly decreased until a non-rotating tearing mode was excited at nearly zero plasma rotation. In this discharge, the plasma stored energy (measured by β_N) was regulated using feedback control of the injected neutral beam power and plasma rotation was controlled using feedback on the mixture of co- and counter-injected neutral beams [4]. Electron cyclotron current drive (ECCD) was applied at a specific location to suppress the NTM for the duration of the ECH pulse and external magnetic perturbation coils were used (with plasma rotation) to aid in the suppression of the RWM even though β_N was maintained well above the no-wall limit. The high β_N value and relatively low rotation in this discharge would typically result in the presence of deleterious tearing modes and/or RWMs without the feedback-controlled stabilization systems energized. Success in the prevention of multiple instabilities simultaneously in these DIII-D plasmas helps to delineate a path for the design of similar systems for the ITER device.

Another concern for future high power tokamaks such as ITER is the safe termination of the discharge. Disruptions might damage the device due to excessive thermal loads, vertical forces, and runaway electrons, but these effects can be reduced or controlled using an appropriate disruption mitigation system [5]. Recent experiments in DIII-D have employed a massive gas puff system and a cryogenic pellet injector to rapidly introduce various gas species for studying the radiative dissipation of the plasma thermal energy. It is important to try to mix the impurity gas as much as possible into the plasma core for optimum mitigation. Fig. 3 shows some fast bolometry pictures of the radiation plume from a cryogenic shell pellet injection experiment. The hydrocarbon shell of the pellet ablates in the outer region of the plasma while the inner argon core is able to penetrate approximately halfway into the center of the plasma before ablating. Delivering the argon to the core of the plasma in this manner promises to be a useful disruption mitigation tool.

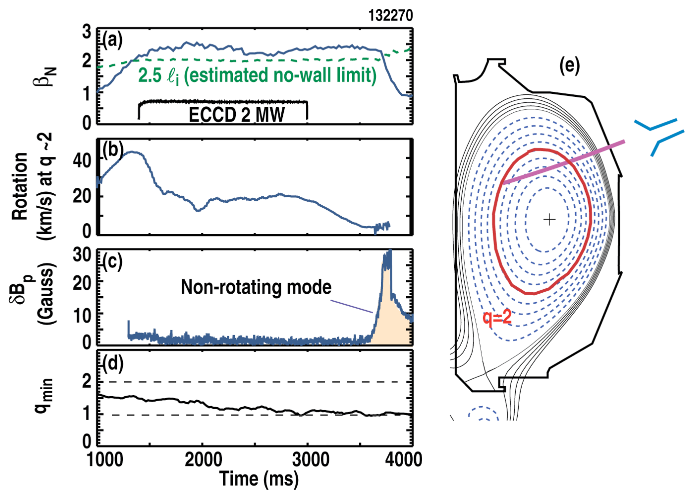

Figure 2. Long duration RWM/NTM free operation at high β_N (a) β_N and the estimated no-wall limit $\beta_{N,no-wall} \cong 2.5\ell_i$, (b) the plasma rotation at $q \sim 2$, (c) the $n = 1\, \delta B_p$ amplitude, (d) $q_{\min}$ versus time, (e) the plasma configuration and ECCD launching path.

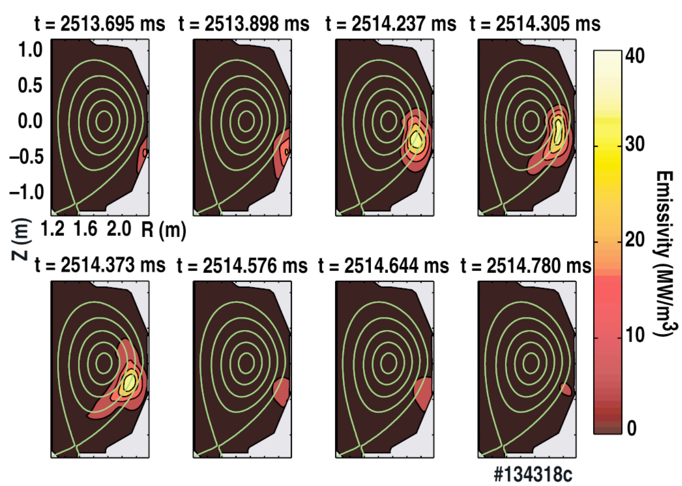

Figure 3. Fast bolometry pictures showing the burn-up of a shell pellet during a disruption mitigation experiment. The pellet penetrates to approximately $r/a \approx 0.5$ before burning up.

In some cases, a disrupting plasma can accelerate a channel of highly energetic electrons (runaways) that could threaten the vessel walls. The electron density required for collisional suppression of these runaways is quite high, and has not been achieved in DIII-D experiments. An alternative approach for controlling them is to use the magnetic perturbation coils on DIII-D to produce a stochastic edge region in the plasma to break up the high-energy electron channel. Results from an experiment that applied an $n = 3$ resonant magnetic perturbation (RMP) field are shown in Fig. 4. In this experiment, a combination of the external C-coil and in-vessel I-coil was used to apply a perturbation of strength $\delta B / B_T \approx 7 \times 10^{-4}$, where B_T is the toroidal field. The reduction of scintillator counts when applying the RMP is a measure of the reduction in high-energy radiation generated when the electrons impact the vessel walls. Although still under study, it is possible that the stochasticity induced by the RMP may be dispersing the electrons before they can be accelerated to very large energies, therefore reducing the impact on the vacuum vessel walls. A similar experiment using an $n = 1$ perturbation on DIII-D was not able to produce the same results; the effect *was* observed, however, in the TEXTOR device for both $n = 1$ and $n = 2$ [6].

III. Future Upgrades

One of the most critical parameters in high performance steady state tokamak plasmas is the current profile. The ITER device is planning for the ability to control the current profile through the use of off-axis current drive. On DIII-D, off-axis current has been driven with modest levels of ECCD, but the effectiveness is limited in high performance plasmas due to the dominance of central current resulting from the high neutral beam power needed to obtain the high β plasma (see Fig. 5). Off-axis current drive from neutral beams (NBCD) is as efficient as on-axis current drive, and appears to be an expedient solution. Compared to ECCD, similar levels of input power induce comparable levels of current density. Detailed transport modeling predicts that up to 200 kA of off-axis current can be driven using a vertically tilted beam [7], allowing ECCD to be used to tailor the current profile. In order to support the ITER design and study off-axis heating, DIII-D has developed plans and is scheduled to modify the first of two beamlines during the tokamak shutdown starting in April 2010. The beamline will be able to tilt from horizontal (0°) up to 16.5° and inject up to 5 MW of power that will drive current 40 cm below the plasma centroid [8]. A design illustration of the tilted beamline is shown in Fig. 6.

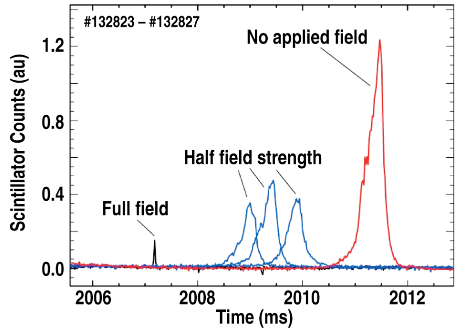

Figure 4. Gamma scintillator counts measure a dramatic reduction in runaways striking the vessel walls for cryogenic neon pellet shutdown experiments with an $n = 3$ I-coil RMP on five consecutive discharges.

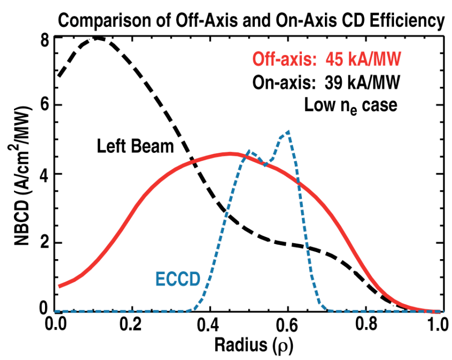

Figure 5. On- and off-axis beam current drive efficiencies are comparable, but high beam power injected on-axis produces a dominant central current drive component in advanced tokamak plasmas.

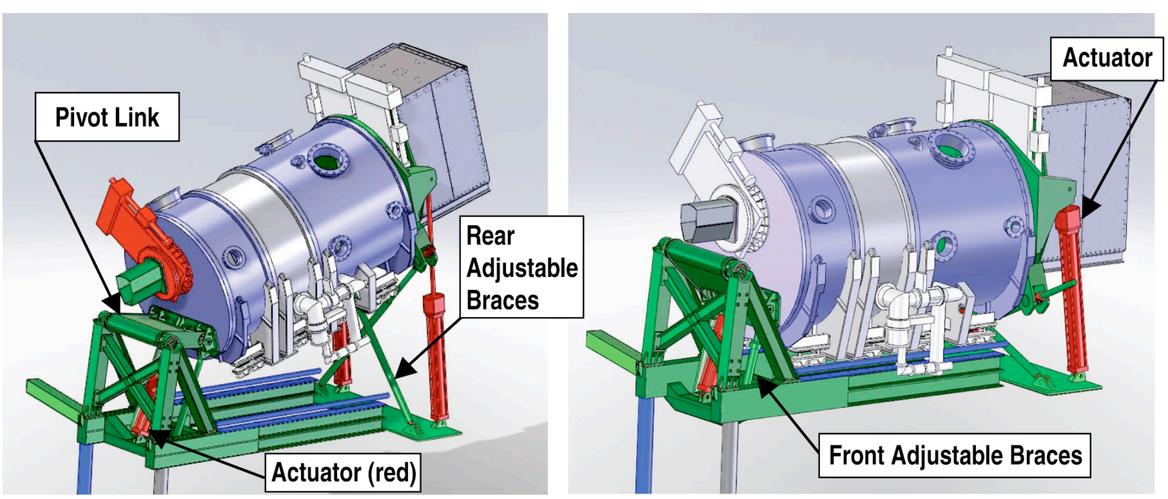

Figure 6. Design pictures of the off-axis neutral beamline showing tilt for 15° injection angle (left) and horizontal for 0° injection angle (right).

DIII-D has long been a leader in the investigation of error fields and plasma instabilities, employing three different perturbation coils throughout the years. Starting with the $n = 1$ coil in 1990 [9], experimental results led to the C-coil and then the internal I-coils [10]. Results accumulated from experiments using these coils [11–13] have driven the design of a fourth coil set to be mounted on the vessel centerpost wall in 2010 (dubbed "CP-coils"). Shown in Fig. 7, the CP-coils, combined with the existing C-coils and I-coils, will significantly increase DIII-D's ability to explore the effects of resonant magnetic perturbations (RMPs) on the stability of various MHD modes such as ELMs, RWMs, NTMs, and locked modes. Fundamental physics issues will be addressed, investigating the importance of the radial localization of the RMP and the effects on peeling-ballooning mode stability and turbulent transport physics. The new coils will lend themselves to experiments that can test theoretical and numerical models such as the ideal-resistive MHD and fluid-kinetic transport models to see if they accurately describe RMP effects over a wide range of coil geometries and spectra.

Increases in the pulse length and auxiliary power level are planned on DIII-D over the next few years. An upgrade to 10 s beam pulses is being studied using some preliminary long pulse data acquired at lower beam voltages using a reduced aperture ion source. The combination of a narrower beam pulse (at only slightly reduced power) and upgraded internal beamline thermal capability may lead to the desired long pulse beam capability. The ECH system has made advances as well, recently conditioning up a sixth 1 MW gyrotron. Fig. 8 shows data from a discharge in which six 1 MW class gyrotrons were injected into plasma simultaneously. Future plans are to add two new 1.2 MW gyrotrons and phase out the existing six 1 MW units for new higher power gyrotrons, increasing the amount of ECH power available for plasma heating and current drive.

IV. Conclusions

The DIII-D tokamak continues to develop new capabilities in response to the demands of an evolving fusion energy research mission. Experiments investigating fundamental fusion science and many ITER-related issues have been carried out and are contributing to the design parameters of the future tokamak. Upgrades to DIII-D are planned that will allow it to continue investigating new physics regimes in higher performance plasmas with longer pulse length.

References

[1] Special issue on "Progress in the ITER physics basis," Nucl. Fusion **47**, S1 (2007).

[2] P. N. Yushmanov *et al*., Nucl. Fusion **30** (1990) 1999.

[3] M. Okabayashi *et al.*, submitted for publication in Nucl. Fusion.

[4] J. T. Scoville *et al.*, Fusion Eng. Design **82** (2007) 1045.

[5] R. S. Granetz *et al.,* Nucl. Fusion **47** (2007) 1086.

[6] M. Lehnen *et al.*, Phys. Rev. Lett. **100** (2008) 255003.

[7] M. Murakami *et al.,* to be published in Fusion Sci. Technol. (2008).

[8] P. M. Anderson and R.-M. Hong, "Development for extending the beam pulse length of the DIII-D neutral beam system," this conference.

[9] J. T. Scoville *et al.*, Nucl. Fusion **31** (1991) 875.

[10] M. Okabayashi *et al.*, Nucl. Fusion **45** (2005) 1715.

[11] J. T. Scoville and R. J. La Haye, Nucl. Fusion **43** (2003) 250.

[12] E. J. Strait *et al.*, Phys. Plasmas **11** (2004) 2505.

[13] T. E. Evans *et al.*, Nucl. Fusion **45** (2005) 595.

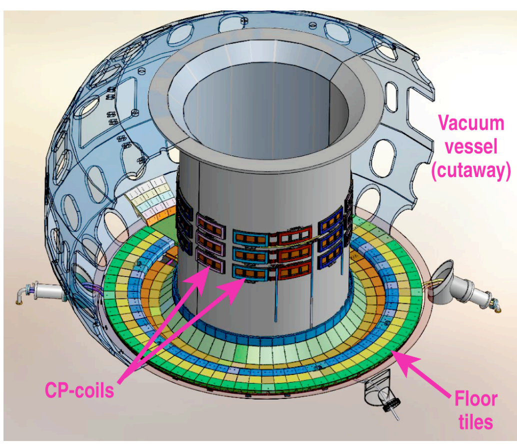

Figure 7. CP-coils installed on vacuum vessel centerpost.

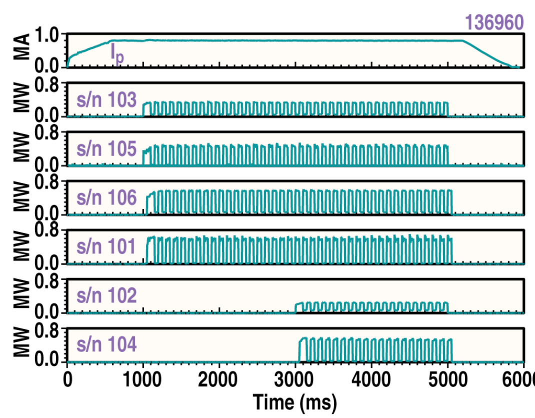

Figure 8. Six gyrotrons injecting 3 MW of modulated power into DIII-D plasma.