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OVERVIEW OF THE DIII–D PROGRAM AND CONSTRUCTION PLANS

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GENERAL ATOMICS

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ABSTRACT

The DIII–D tokamak is a mid size tokamak operating at reactor relevant parameters. Because of its size it is relatively easy to modify the machine as required to test new ideas or theories. During the last few years several new hardware items have been added to the DIII–D tokamak and improvements have been made to others. The main addition in the last two years is the installation of the I-coil system and upgrades to the electron cyclotron heating (ECH) system. In addition the Fast Wave system is being brought back into operation after having been idle for three years. The I-coil system, which consists of 12 coils installed inside the DIII–D vessel, is used to stabilize the resistive wall modes and to produce a stochastic edge, which has suppressed Edge localized modes. (ELMs) can be detrimental to ITER, since they can erode the plasma facing surfaces. The I-coils are powered by three switching power-amplifying units, which together with a flexible patch panel allow the I-coils to be operated in many different configurations. The ECH system has been upgraded to six gyrotrons, which have been used to heat the plasma, modify the current profile and stabilize the neoclassical tearing 3/2 and 2/1 modes. Three ECH launchers built by Princeton Plasma Physics Laboratory are installed on the DIII–D tokamak and have the capability of changing the beam direction in both toroidal and poloidal direction.

Three additional long pulse gyrotrons have been ordered for the DIII–D program. They are required for current profile control and stabilization of the NTMs. The gyrotrons are scheduled to be installed during a 10–12 month facility enhancement period, which spans 2005–2006. At the same time a modification is scheduled to be made to the lower divertor to make it pump double null high triangularity plasmas, which are important for studying advanced tokamak plasmas. One of the four neutral beam lines will be rotated for counter injection, which will allow study of the quiescent double barrier mode with central co-rotation of the plasma and of the resistive wall mode with low rotation.

1. INTRODUCTION

A description of the DIII–D tokamak and the major upgrades that have been made to the device is discussed in Ref. [1]. During the last two years some major and minor upgrades have been made to the DIII–D tokamak. The upgrade of the electron cyclotron system was upgraded to a six gyrotron system, which in addition to adding gyrotrons required new power supplies and control systems. The electron cyclotron heating (ECH) systems and the physics improvement obtained by using the system will be described in Section 2. Another upgrade was the installation of the 12 internal coils (I-coils), also with power supplies, patch panels and control system. The I-coil system, which will be described in Section 3, is used to stabilize the resistive wall mode, minimize the magnetic error fields in the plasma and produce a stochastic plasma edge. There continues to be significant upgrades to the digital plasma control system as more and faster CPUs are used and the software is upgraded to include more diagnostics, which in turn allows better and additional plasma parameters to be controlled. The plasma control upgrades for the last two years will be described in Section 4.

During the last two years a number of diagnostics upgrades have been made to the world class diagnostics already installed on DIII–D. These diagnostic upgrades will be discussed in Section 5. The use of massive gas puff has been shown to mitigate the effect of a disruption. The massive gas puff system used on DIII–D will be described in Section 6. The understanding and control of plasmas in DIII–D has reached a point where the different components can be put together to obtain high performance plasmas, which have been controlled for as long (~6 s) as it is possible to run the DIII–D tokamak.

The upgrades (Section 7) that are planned for DIII–D during the next two years include upgrade of the ECH system to 6 MW long pulse capability. During that period the lower divertor in DIII–D will be upgraded to allow high triangularity double-null divertor plasma to be fully pumped to extend the performance of the high performance double-null divertor plasmas. At the same time one of the four beam lines, which all currently inject the eight beams in the same direction, will be rotated to allow the study of plasma with balanced beam injection to among other things study the stabilization of the resistive wall mode without plasma rotation.

2. ECH SYSTEM

The DIII–D electron cyclotron heating system [2,3] consists of six gyrotrons with all the support equipment; gyrotron, superconducting magnets, gyrotron/magnet supporting tanks, power supplies, waveguide transmission lines, launchers and associated controls. Three of the gyrotrons were manufactured by Gycom and have a nominal output of 800 kW for 2 s. The other three gyrotrons were manufactured by Communication and Power Industries (CPI) and have a nominal rating of 1 MW for 10 s. The Gycom gyrotrons have boron nitride windows, which absorb about 4% of the transmitted power, which limits the gyrotron pulse length. The CPI gyrotrons have artificial diamond (CVD) windows [4], which only absorb about 0.2% of the transmitted power. Because of this low absorption and the high heat conduction of diamond (2000 W/mK), about four times that of copper, these edge water-cooled windows do not limit the gyrotron pulse length. The diamond windows reach thermal equilibrium after about 3 s. However worldwide, several CVD windows have failed either due to brazing problems or surface contamination. The brazing problem has been solved by using a Au/Cu braze, and by paying strict attention to the vacuum conditions during the bake process of the gyrotrons the surface contamination can be avoided.

The transmission lines are made of 2 m long sections of circular corrugated waveguides 31.75 mm in diameter. The waveguides are joined by Helicoflex seals and the lines are evacuated to a pressure $<1\times10^{-6}$ Torr. The transmission efficiency has been measured, and the losses in the 100 meter long lines are about 20%, with most of the losses being in the about eleven 90 degree miter bends. The six installed waveguide lines end in three launchers each having two mirrors per line. The first mirror focuses the beam, whereas the second mirror is steerable (±20 deg) in both the poloidal and toroidal directions. The poloidal steering allows precise control of the location of the plasma heating and current drive, whereas the toroidal steering allows the injected power to drive current in either the co- or counter direction or just heat the plasma.

The ECH system has already shown that it is very effective in heating the plasma, driving plasma current and stabilizing plasma instabilities [5]. Significant plasma heating has been demonstrated in DIII–D, with central temperatures of 6 keV obtained in less than 100 ms [6] and peak electron temperature of 23 keV. The small size of the ECH beam makes it well suited for measurement of the local electron energy transport. Current profiles that peak off axis are characteristic of high performance advanced tokamak discharges. These discharges are normally initiated with neutral beam injection during the plasma current ramp up. Usually the profile would relax to one that is peaked on-axis. However, by using off-axis electron cyclotron current drive (ECCD) the current profile can be maintained and the high plasma performance can be sustained. In an experiment [7] in DIII–D 130±40 kA was driven off axis with ECCD, which is in agreement with computer models. ECCD has also been used to stabilize the 3/2 and 2/1 neoclassical tearing modes [8,9], which often terminate high performance discharges in DIII–D. In these experiments the ECCD is injected into the islands created by the instability. However, since the rational surfaces on which these islands are created move, the plasma or the toroidal field was adjusted by feedback so that the resonance layer would coincide with the rational surfaces. About 2.3 MW of ECH power is required to stabilize the 3/2 mode, and about 3 MW of power is required to stabilize the 2/1 mode. In order to stabilize both modes and simultaneously drive off-axis plasma current, 9 MW of power would be required. There is a proposal to upgrade the ECH system on DIII–D to six 1 MW and two 1.5 MW gyrotrons. The first step is to upgrade the system to six 1 MW long pulse gyrotrons. The upgrade is discussed in Section 7.

3. I-COILS

In 2002 twelve internal one-turn picture frame coils were installed and brought into operation in early 2003. The coils were installed after experiments using the C-coils, which are six coils installed outside the vacuum vessel at the midplane of the tokamak, to stabilize the resistive-wall mode instabilities were successful, however model calculations showed that internal coils and poloidal sensor probes would be much more effective in stabilizing the resistive wall modes. The twelve internal coils are installed in two bands inside the DIII–D vacuum vessel. Six coils are located above the midplane and six coils below the midplane. Each coil is 0.5 m $\times$ 2 m and spans 60 deg in toroidal direction. The current capability of the coils is 7 kA. The coil leads are coaxial in order to minimize the error field and the coils are installed behind the protecting carbon tiles. A patch panel was installed, which allows the power supplies to be connected in many different configurations to the coils. Already the first set of experiments showed that the new coils provided effective correction of the error fields and were more effective than the C-coil in stabilizing the RWMs [10]. Because of the flexibility that the patch panel and the 12-coil net give for producing different field structures, they have been used to produce a stochastic magnetic boundary for the plasma. The stochastic field suppressed the large ELMs. The confinement of the H–mode was not affected, which indicates that this approach to controlling the ELMs is attractive for the-next-step burning tokamaks [11].

One coil developed a leak through the stainless steel barrier that separates the nitrogen blanketed insulated conductor from the DIII–D vacuum. The leak was found to be due to low cyclic fatigue due to operation at 100 Hz the natural frequency of the coaxial lead. The leak was repaired [12], and all the coils were made stiffer and the first natural frequency of the coaxial transition was significantly increased. The coils have since been used for experiments, but so far operating near 80 Hz has been avoided.

4. PLASMA CONTROL SYSTEM

Just a decade ago the control systems for most tokamaks were analogue systems, which made it hard to rapidly change the configuration and to restore a previous configuration. With the advent of fairly low cost and fast computers it became natural to shift to a digital system. The original DIII–D digital system, which consisted of six Intel i860 processors, has been replaced with PCI based Pentium 4 and Alpha computers and a 2 Gb/s Myrinet network for interprocessor communication [13]. The plasma control system has transitioned from handling discrete parameters into an integrated control system [14], which can simultaneously handle the highly coupled plasma parameters in advanced tokamak discharges. Physics understanding of the plasma has produced models of system responses, which are used in the design of control algorithms. The multi-CPU architecture allows simultaneous execution of many computationally intensive control algorithms. In addition to controlling the plasma shape and profiles the system has been use to control plasma instabilities, which would normally cause a termination of the discharge. Two such instabilities are the resistive wall modes and the neoclassical tearing modes. The plasma control system is controlling the resistive wall modes by adjusting the current level and phase in the different I-coils. The control of the neoclassical tearing mode is done by aligning the 3/2 or 2/1 q surface with the ECH resonance absorption layer. During the initial experiments the plasma was moved or the toroidal field was changed to obtain this overlay. The control includes a Search and Suppress routine, which systematically searches for the mode location and then suppresses it. When the mode has been suppressed, another algorithm, Active Tracking, takes over to track the q surface of interest by linear and nonlinear predictor calculations, while the mode is not present. Lately a real time plasma equilibrium reconstruction [15] has been added to the plasma control system, which contains information from the MSE diagnostic on the location of the rational q surfaces. The planned upgrades of the system include control of launcher mirrors to point the EC beam to the right position instead of moving the plasma or changing the radial location of the plasma.

5. DIAGNOSTICS

While DIII–D remains one of the best diagnosed magnetic fusion experiments, additional new diagnostics and/or upgrades have been implemented recently.

Many of the efforts have concentrated on improving our measurements and therefore understanding of the edge/pedestal area. One critical need was to measure adequately the edge current profile. We recently completed the installation and testing of a new edge current diagnostic based upon Zeeman spectroscopy of an injected 30 keV lithium beam [16]. Polarization analysis of the collisionally induced fluorescence at multiple spatial locations allows us to determine the poloidal field profile, and hence the underlying current distribution. The 32-channel design spans a range of normalized minor radius between 0.7 and 1.0 (edge), with a spatial resolution of approximately 5 mm. The first results, as shown in Fig. 1, indicated the presence of additional current (bootstrap) generated by the pedestal sharp gradient, which in turn influences the stability in regards to ELMs.

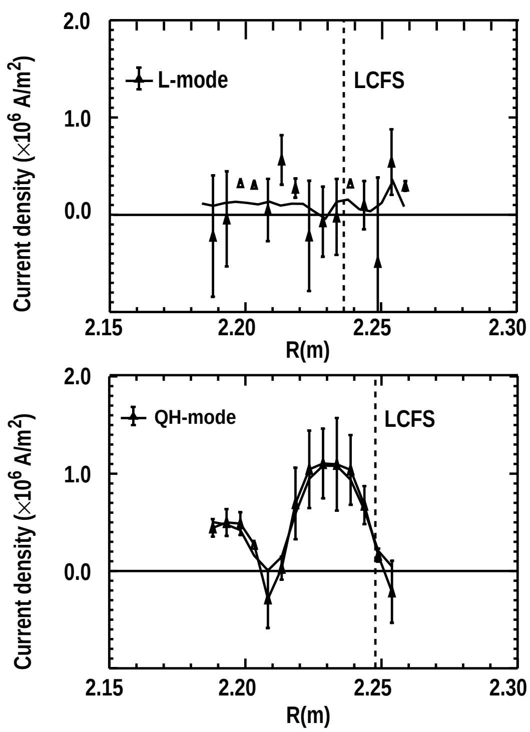

Fig. 1. Calculation of edge toroidal current density for the L–mode and QH–mode phases of DIII–D shot 115099, using the measured B_{VIEW} from the lithium beam diagnostic. the last closed flux surface (LCFS) as calculated by EFIT is shown by the dotted line. (Adapted from D.M. Thomas et al. [15]).

The charge-exchange recombination (CER) spectroscopy (core and edge) system has also been upgraded [17] with new cameras and spectrometers, which now allows the study, for example, of fast events (such as ELMs), and how they impact the ions dynamics. At the same time, the reflectometer system is now capable of fast frequency sweeps, yielding a full reconstruction of the edge electron density structure from 0 to $6\times19\ \mathrm{m}^{-3}$. These three new capabilities taken together are generating a deeper understanding of the dynamics of the pedestal.

Much progress has been observed in the understanding of the ion heat transport channel, especially near the core of the plasma, in part due to the new measurement capabilities brought forward in the last decade. However, we are still facing significant mysteries in regards to the electron heat channel. One candidate for the observed diffusion lies on the role of the short wavelength (high k vector) turbulence. Theories predict the presence of these structures with k vector of ~20–50 cm^{-1}. However, these structures are very difficult to observe and new diagnostics are needed in order to identify and characterize their effects. One technique presently being upgraded, is based on microwave scattering, at far-infrared, or at ~100 GHz frequencies. Presently three systems [18] are installed and a 4th one being commissioned, in collaboration with the University of California at Los Angeles (UCLA) and the University of New Mexico (UNM). They are presently covering a range of k vectors between 1 and 40 cm^{-1}. In parallel, the phase contrast imaging system [19] fielded by MIT, observes the density fluctuations along the laser beam path, and its view has been upgraded to study and localize these fluctuations.

Although interferometry has been a basic diagnostic technique for many years, we are continuing to improve the system, for reliability and accuracy. We are now integrating a new digital electronic system that can translate fringes into line-integrated density. The system uses two lasers at different wavelengths (HeNe and $C0_2$), along four chords through the plasma. The new digital system allows very fast time resolution (better than 1 μs) and improved accuracy.

Near future plans include the addition of two views for the MSE diagnostic. These two views will cover the edge and core part of the planned counter-injection neutral beam. These two systems in combination of the existing two views of the co-beam will allow the unambiguous resolution of the current and radial electric field across the full profile.

6. MITIGATION OF PLASMA DISRUPTIONS

Experiments [20] with massive gas jet injection have shown that it is possible to mitigate the three serious effects of disruptions: thermal heat loads to the plasma facing components, forces from the poloidal induced currents and run-away electrons. Disruptions can be avoided by operating away from instability limits, but off normal events can occur and cause disruptions, which will have to be mitigated.

In DIII–D a gas reservoir was installed close to the vacuum vessel only separated by a fast acting valve. The plasma control system was setup to detect vertical displacement events, and upon detection open the valve. Nitrogen or argon gas was then injected into the plasma at sonic velocity ~300–500 m/s. About 4×10^{22} particles were released in 2–5 ms. The high impurity level (35 times the electron inventory in the plasma) radiates the power before the plasma can move and this results in a decreased heat load to the divertor that is less than 10% of the stored energy. The plasma current rapidly decays and a high edge q value is maintained. This minimizes the conversion of the toroidal current to poloidal current and thus minimize the $j\times B$ forces on the plasma facing components. The runaway electrons that are normally seen when the plasma disrupts are quenched by the large drag from the numerous bound electrons in the injected gas.

Modeling of the massive gas injection mitigation of plasma disruptions has been validated by the experiments in DIII–D. The model shows that the technique scales well to burning plasma devices such as ITER and FIRE.

7. PLANNED UPGRADES TO THE DIII–D TOKAMAK

In the fiscal years FY05 and FY06 the DIII–D tokamak is scheduled to operate 14 weeks in each of the two years. The operation is scheduled for the first half of FY05 and the second half of FY06. This makes it possible to make upgrades to the DIII–D tokamak for a period of 12 months (Long Torus Opening). During this time several upgrades are scheduled: upgrade of the ECH system from three long pulse 1 MW tubes and three short pulse 0.75 MW tubes to six long pulse 1 MW tubes, modifying the lower divertor to allow it to pump high triangularity double null divertor plasmas, and to turn one of the four neutral beam lines around, so one beamline will inject in the opposite direction of the others. In addition upgrade to the cooling towers will be done and the tokamak pulse length will be extended to 10 s at full parameters.

Three 1 MW gyrotrons have been ordered and delivery is scheduled so that the gyrotrons can be installed during the Long Torus Opening period. In addition during the summer of 2004, a developmental 1.5 MW depressed collector gyrotron is scheduled to arrive at DIII–D for testing and conditioning to full parameters. The addition of the three long pulse gyrotrons will provide the required power to probe, modify, stabilize and actively control the high performance advanced tokamak discharges in DIII–D. Later it is planned to add two more long pulse gyrotrons in addition to a fourth launcher and two more transmission lines, a power supply and control. Modeling indicates that this additional power is required to stabilize the different instabilities and at the same time maintain the required advanced tokamak current profile.

The lower divertor in the DIII–D vessel was installed in 1991 and its elevated water-cooled ring formed an entrance aperture for the advanced toroidal cryopump. The pump has been used for plasma density control. The AT program at DIII–D has demonstrated high beta performance in highly triangular plasmas, which have their strike points at a relative small radius. For these plasmas the current divertor configuration is not efficient for pumping. To be able to pump to get to a density that is compatible with an efficient ECCD, the plan is to modify the lower divertor by extending the baffle and pumping aperture to a smaller radius so it can pump high triangularity plasmas.

Currently seven neutral beams housed in four neutral beam lines are available to heat the DIII–D plasmas. All the beams are injecting in the same direction. It is thus not currently possible to heat the plasma without at the same time injecting a net momentum to the plasma. The DIII–D program has been very successful in stabilizing plasmas against resistive wall modes with rotating plasma. However, future burning plasma devices might not have rotating plasmas, thus it is very desirably to study the stabilization of resistive wall modes in non-rotating plasmas. A beam injected in the opposite direction of the other beams will also allow a better measurement of the radial electrical field in plasmas with the motional Stark effect diagnostic. It is therefore planned to reverse one beamline, which will allow heating with four beams and a zero net momentum input. In these non-rotating plasmas the I-coils will be used to try to stabilize the resistive wall modes as models predict.

8. CONCLUSION

Hardware upgrades to the DIII–D tokamak have been important in the past and will continue to be important in the future to pursue the advanced tokamak research, which is one of the cornerstones of the DIII–D research program. In the past two years upgrades has been made to the ECH system, internal I-coils have been installed, and the plasma control system hardware and software have been upgraded. In FY05 and FY06 several major upgrades are planned for a 12 month period: Upgrade of the ECH system, modification to the lower divertor and rotation of one of the four neutral beamlines.

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