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RESEARCH PROGRESS AND HARDWARE SYSTEMS AT DIII–D

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P.I. PETERSEN and the DIII–D TEAM

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Research Progress and Hardware Systems at DIII–D

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Abstract. During the last two years significant progress has been made in the scientific understanding of DIII–D plasmas. Much of this progress has been enabled by the addition of new hardware systems. The electron cyclotron (EC) system has been upgraded from 3 MW to 6 MW, by adding three 1 MW gyrotrons with diamond windows and three steerable launchers (PPPL). The new gyrotrons have been tested to 1.0 MW for 5 s. The system has been used to control the 3/2 and 2/1 neoclassical tearing modes and to locally heat the plasma and thereby indirectly control the current density. Electron cyclotron current drive ECCD has been used to directly affect the current density. A Li-beam diagnostic has been brought on-line for measuring the edge current density using Zeeman splitting. A set of 12 coils (I-coils), consisting of six picture frame coils each above and below the midplane, with a capability of 7 kA for 10 s has been installed inside the DIII–D vessel. These coils, along with the existing six C-coils, are used to apply non-axisymmetric fields to the plasma for both exciting and controlling plasma instabilities. The DIII–D digital plasma control system is now used to not just control the shape and location of the plasma but also the electron temperature, density, the NTMs, RWMs, plasma beta and disruption mitigation. Plasma disruption experiments are extended to mitigation of real time detected disruptions on DIII-D.

1. INTRODUCTION

The main emphasis of the DIII–D research program is the advanced tokamak thrust, which strives to improve the tokamak performance by minimizing the size of a reactor through active control of the plasma cross section and the plasma profiles. Several transient advanced tokamak modes have been established in DIII–D. However, the main thrust has been the negative central shear mode, which has a broad current profile, which requires stabilization of the resistive wall modes and the neoclassical tearing modes. These are the modes that currently end the high performance discharges in DIII–D. Edge localized modes (ELMs) play an important role in the design of reactors, since these modes can cause high heat loads and particle fluxes to the plasma facing components at levels that cause unacceptable erosion. Plasma disruptions, if not avoided or mitigated could be detrimental or at least shorten the lifetime of the PFCs.

New hardware has become available on the DIII–D tokamak that will help modify the profiles and stabilize the RWMs and NTMs. Twelve internal coils have been installed inside the DIII–D tokamak. These coils will be used to stabilize the RWMs and produce a magnetic ergodized edge, which can modify the ELM characteristic. The electron cyclotron (EC) system has been upgraded from 3 MW to 6 MW. With the additional power the 2/1 NTMs can be stabilized in addition to the 3/2 NTMs, which are stabilized with less power. The EC has also been used to modify the current and temperature profiles.

In Section II the advanced tokamak modes will be discussed. In Section III the RWMs and NTMs, which currently limit the high performance discharges will be discussed. In Section IV the internal coil system will be described. The EC upgrade and plans will be outlined in Section V. As the advanced tokamak features start to be merged in the same discharge additional demands are put on the digital plasma control system, which is described in Section VI. Disruption mitigation is described in Section VII, and diagnostic upgrades in Section VIII. A summary of the paper is given in Section VIIII.

II. ADVANCED TOKAMAK MODES

Advanced tokamak modes [1] have been the hallmark of the DIII–D research for many years. Four different advanced tokamak modes have been identified: negative central shear (NCS), high ℓ_i, radiative improved mode, and the quiescent H–mode (QDB, quiescent double barrier). However, only the two most commonly used modes, NCS and QDB will be discussed in this paper. The common goal of these four modes is that they lead to a smaller steady state operating reactor. The modes have to work steady-state and achieve high fusion power density in a reactor, which means that they have to operate at high normalized beta, $\beta_N = \beta_T/(I/aB)$. The improved performance includes a pressure limit above the conventional tokamak, a confinement better than the standard H–mode, and a high fraction of self-generated (“bootstrap”) current.

Many numerical studies [2] have shown that there are three major approaches to increase the beta limit: (1) plasma shape, (2) internal profiles, and (3) wall stabilization. Fig. 1 shows that β_N increases with broader pressure profile and strong shaping, increased elongation and triangularity. Fig. 2 shows that β_N can be increased by a factor of 2 above the “no-wall” limit with wall stabilization. The DIII–D tokamak was built to investigate plasma with different shapes. This capability is made possible by the shape of its vacuum vessel and the 18 close fitting field shaping coils. Negative central shear discharges [1] are obtained by applying auxiliary heating during the plasma ramp-up phase. The early heating slows the current diffusion toward the center and drives the toroidal electrical field on axis to near zero. The resulting current density profile during the current ramp is peaked off-axis. The bootstrap current, with a maximum off-axis, can contribute and help extend the duration of the hollow current profile. These discharges have a broad profile and can be stabilized against the resistive wall mode with plasma rotation which gives the effect of a close wall. Such a discharge is shown in

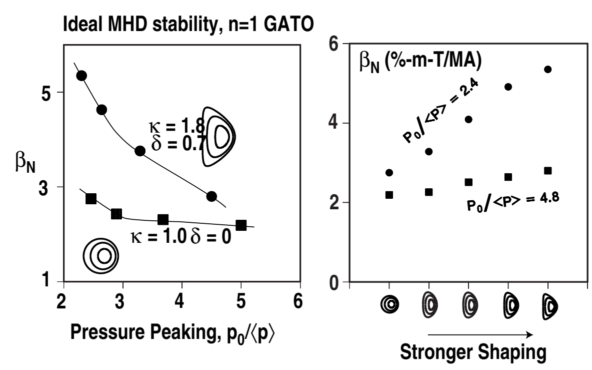

Fig. 1. Calculation with the code GATO shows that higher β_N can be obtained with higher triangularity and broader pressure profiles.

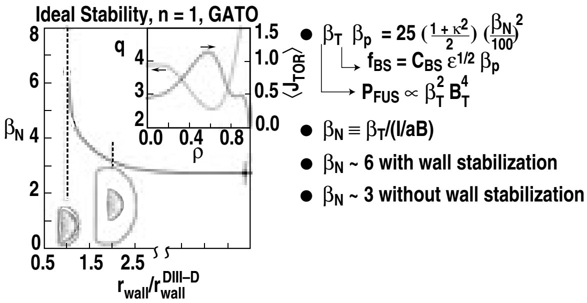

Fig. 2. Calculation with the GATO shows that $\beta_N = 6$ can be uptained with a tight fitting wall, whereas the limit is much lower with the walls far from the plasma. r_{wall}^{DIII-D} is the actual location of the DIII–D vacuum vessel, and represents the distance from the plasma center to the vessel wall. In the calculation this wall is expanded out so that the distance r_{wall} from the plasma center to the wall is increased by the same factor everywhere.

Fig. 3. This discharge achieved $\beta_N H_{89P}$ >12 for five energy containment times. The discharges are normally terminated by a NTM due to the current density evolution. To extend discharges with a hollow current profile an external drive mechanism such as ECCD is required to drive current off-axis. The ECH might also be required to stabilize the NTMs depending on the operating regime.

The QDB-mode [3] was discovered two years ago on DIII–D. The mode has an internal transport barrier (ITB) and a quiescent H–mode edge transport barrier (no ELMs). The mode is obtained so far only with counter neutral beams. It is ELM free and is associated with the presence of an edge harmonic oscillation (EHO) (<10 kHz for n=1), which shows up in the magnetic, electron density and temperature diagnostics. The absence of ELMs in QDB is advantageous for obtaining ITBs and eliminating large pulsed heat flux to the first wall. The quiescent H–mode edge is thought to be a consequence of the deep electric field well created by the counter-beam. A mixture of co and counter neutral beam injection might provide the deep well, and changing one of the DIII–D four neutral beam lines around will allow this idea to be tested. Currently the QDB duration is limited by machine considerations, particularly the neutral beam pulse lengths.

The most recent results in advanced tokamak research on DIII–D, which has used ECCD, are described in Section V, since the electron cyclotron system upgrade has been very valuable in extending the AT modes.

III. RESISTIVE WALL MODES AND NEOCLASSICAL TEARING MODES

The RWMs seen in DIII–D originate from an n=1 ideal external kink mode, which in the presence of a resistive wall, are converted to a slowly growing RWM. The characteristic time scale for the growth of the mode is the skin time of the vessel, which for DIII–D is a few milliseconds. There are two ways to stabilize the RWM: either with a combination of neutral beam induced plasma rotation and reduction of the resonant error fields which slow the rotation or through active feedback. Experiments in DIII–D have demonstrated that passive stabilization of the RWM by plasma rotation past a conducting wall is possible, even at β_N values significantly above the no-wall limit [4]. In these discharges, feedback is used to improve the magnetic field correction by sensing and

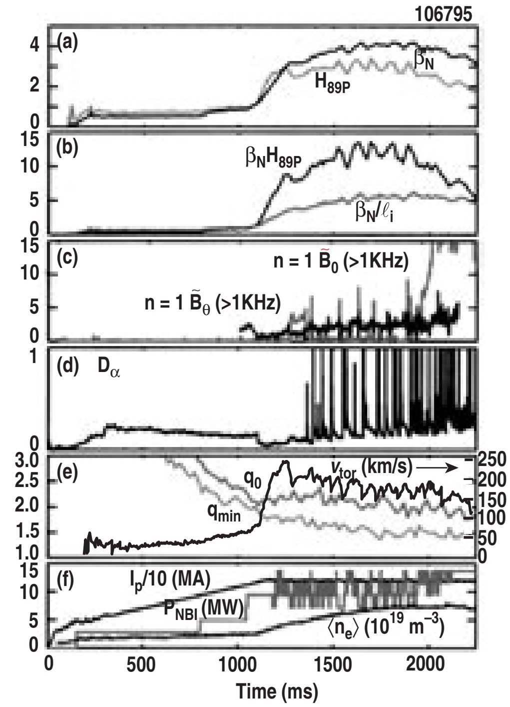

Fig. 3. A fully wall stabilized discharge, with an ITB and weekly NCS has been achieved with $\beta_N H > 12$ for 5 τ_E. (a) β_N ~ 4 and H_{89P} > 3; (b) $\beta_N H_{89P}$ ~ 12 and β_N/l_i ~6 about 50% above the no-wall limit; (c) rotation stabilizes the low frequency RWM, but a 2/1 trearing mode grows at ~1900 ms; (d) D_α showing ELMy H–mode during high performance phase; (e) the toroidal rotational velocity of the plasma is 150–250 kms^{-1} during the time of high $\beta_N H_{89P}$; minimum safety factor, q_{min}, is maintained above the 1.5 throughiout the high performance phase with the central q near 2; (f) plasma current I_p, injection neutral beam power, P_{NBI}, and line averaged density, $<n_e>$.

opposing the resonant response of the stable RWM to the uncorrected field asymmetry. The initial experiments were done with 6 external coils. However, modeling showed a significant benefit of using 12 internal coils and a set of poloidal magnetic probes. Thus, a set of 12 internal coils [5] and poloidal probes were installed inside the DIII–D vessel in the fall of 2002. The coils are discussed in Section IV.

Neoclassical tearing modes are the topological rearrangement of field lines through reconnection which form islands [6]. A method to stabilize the NTM once the mode is formed is to use ECCD to replace the missing bootstrap current in the O-point of the island. This has been demonstrated in DIII–D where the saturated m=3, n=2 NTM was fully suppressed using 2.3 MW of ECCD [7]. As additional power became available in the spring of 2003 the m=2, n=1 NTM was for the first time [8] completely suppressed. Fig. 4 shows an example of such a discharge. The toroidal field B_T is ramped down from 3.2 s to 4.1 s close to the optimal value for mode suppression of the mode with ECCD. Soon afterwards the m=2, n=1 mode starts to grow and 2.7 MW of power is injected into the plasma near the island location, driving 40 kA of current. The plasma control system (PCS) is put in a "search and suppress" mode to make small changes in B_T to find and lock onto the optimum position for complete island stabilization.

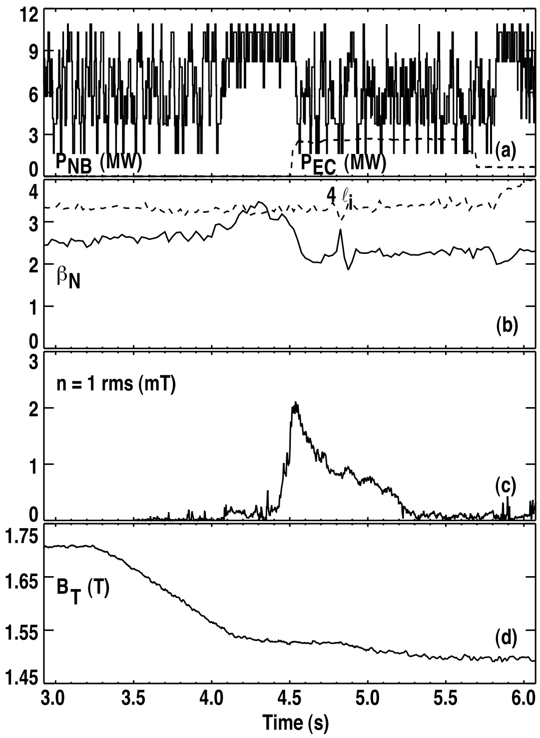

Fig. 4. Time history of discharge #111366 showing (a) NBI (solid) and ECCD (dashed) powers, (b) normalized beta (solid) and ideal no-wall stability limit (dashed), (c) rms amplitude of n=1 tearing mode measured at the wall, and (d) toroidal magnetic field strength.

IV. INTERNAL COILS

The external C-coils that were installed on DIII–D to correct inherent magnetic field asymmetries have also been used to stabilize the resistive wall mode. Model calculations showed that internal coils (I-coils) and poloidal sensor probes would be much more effective in stabilizing the resistive wall modes for two reasons. Firstly the I-coils are inside the conducting wall and have a lower induction and can therefore respond faster. Secondly the coil configuration allows a better match to the poloidal mode number. A set of 12 I-coils [5] were therefore installed inside the DIII–D vacuum vessel in the fall of 2002. Six coils are installed above and six below the midplane of the vessel. Each coil has a picture frame shape (0.5 m x 2 m) and spans about 60° in toroidal direction. They consist of a single turn water-cooled copper conductor, which is housed inside a stainless steel vacuum shield. The copper conductor and shield are isolated with a high temperature polyamide (Vespel) and Kapton® sheets and have been tested to 4 kV. The steady-state current capability of the coils is 7 kA. The coil leads are coaxial in order to minimize any error field and installed behind the carbon tiles to protect the coils from physical interaction with the plasma. A patch panel was installed, which allows the four available power supplies, which were provided by PPPL, to be connected to the twelve coils in many different configurations as required by the research program.

After extensive testing the coils were made available for physics experiments. The first set of experiments [9] showed that the new coils provided effective correction of the magnetic error fields, which slow down the plasma rotation. In Fig. 5 two discharges are compared, one with the control coils on, the other with the control coils turned off during the discharge. In both cases significant plasma rotation is seen when the control coils are on and β_N is above the normal no-wall limit. Shortly after the coil current is turned off, the plasma rotation stops and the β_N drops below the conventional limit.

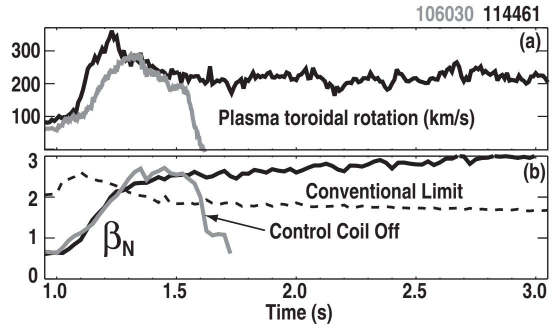

Fig. 5. (a) Application of the I-coil provides effective correction of the magnetic field irregularities which permits sustained high plasma rotation. (b) The high rotation permits plasma pressure above the conventional limit to be maintained. When correction current is turned off, rotation drops, instability grows and pressure drops.

A second set of experiments [10] allowed a detailed comparison between the experiment and a simple theoretical model that included rotation. This is shown in Fig. 6 where both the theory (solid curve) and experiment (diamonds) indicate that the plasma response to a magnetic field applied with the I-coils is maximized when the magnetic field rotates at the same rate as the plasma instability. The response also increases with β_N as predicted by theory.

Using the new I-coils, it has also been possible to compare the growth rate of the RWM instability with the predictions of theory. The comparison is shown Fig. 7, which shows that there is good agreement between theory and experiment.

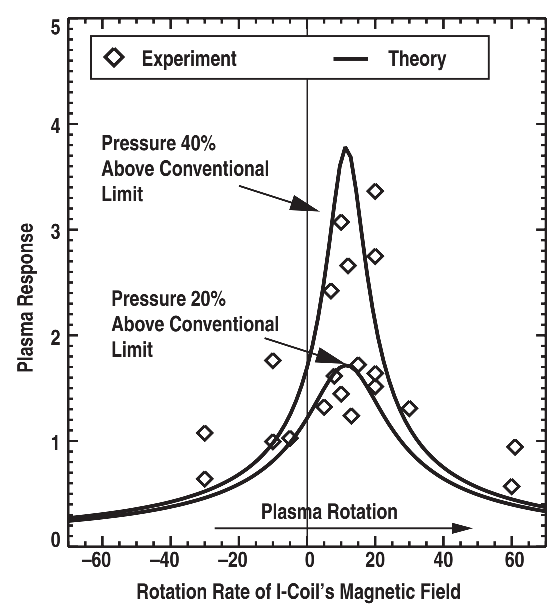

Fig. 6. Resonant field amplification mesured for various applied external freqquencies at two values of average plasma pressure. The experimental data (diamonds) is in good agreement with the predicted frequency dependence of a semi-empirical single mode model (solid line).

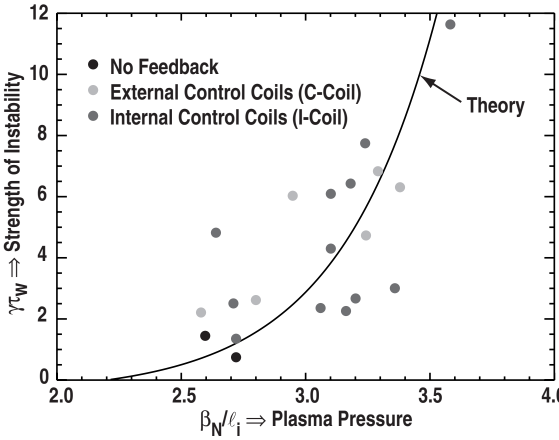

Fig. 7. Comparison of experimentally measured growth rate of plasma instability with theory (solid curve) shows good agreement. The I-coil permits comparison with theory at higher plasma pressure and growth rate than earlier experiments.

V. ELECTRON CYCLOTRON SYSTEM UPGRADE

The EC system [11] installed on DIII–D includes six gyrotrons operating at 110 GHz. Three of these gyrotrons are made by Gycom in Russia, and three are made by CPI in California. The Russian gyrotrons have boron nitride edge-cooled windows, which limits the operation to 2.0 s at power levels approaching 0.8 MW. The CPI gyrotrons have edge-cooled diamond windows, which allow the gyrotron to operate for 10 s at 1 MW. To date the CPI gyrotrons have only been tested for 5 s at 1 MW, since that is the current need for the DIII–D experiments. However, since the window temperature reaches thermal equilibrium after 3 s, there should be no problem extending the gyrotrons to 10 s, which is required when DIII–D hopefully in the near future is made capable of 10 s flattop discharges. The layout of the EC system is shown in Fig. 8. The transmission system consists of evacuated waveguide with a 31.75 mm i.d., which propagates the $HE_{1,1}$ hybrid mode, transporting the rf power from the gyrotrons to the DIII–D tokamak.

Eventually the EC system needs to be upgraded to 9 MW, 10 s in order to support the planned integrated advanced tokamak experiments in the future. Three ports on the DIII–D tokamak house three launcher structures. Each structure has two launchers, each capable of launching 1 MW of EC power. Each launcher is steerable ±20° in both the toroidal and poloidal direction, which allows the location of the heating and the direction of the current drive to be set according to the experimental needs. Four of the six launchers have mirrors rated for 10 s pulses, whereas two are rated for 5 s.

As mentioned in Section III, the ECCD has been valuable in being able to stabilize the NTMs by replacing the missing bootstrap current inside the magnetic island created by the NTMs and thereby stabilize the mode. The ECH is also used to modify the temperature profile and at the same time indirectly modify the current profile. The ECCD can directly modify the current profile and is therefore essential for extending the advanced tokamak mode and eventually making them steady-state.

An example of a high performance plasma discharge in which ECCD [12] is used to modify and extend the pulse length is show in Fig. 9. This discharge has $q_{min}>2$ at the end of the current ramp. This is obtained by inducing a L-H transition during the current ramp which broadens the temperature profile and slows down the penetration of the current density. Just after the end of the current ramp neutral beam feedback is used to keep the $\beta_N \sim 2.8$ for the remainder of the shot. At 1.5 s 2.5 MW of EC waves are steered toroidally and vertically to generate current parallel with the plasma current at $\rho = 0.4$ on the inboard side of the magnetic axis. A significant increase in negative magnetic shear is produced within 500 ms of the turn on of the ECCD and maintained for the duration of the ECCD pulse (2 s). By

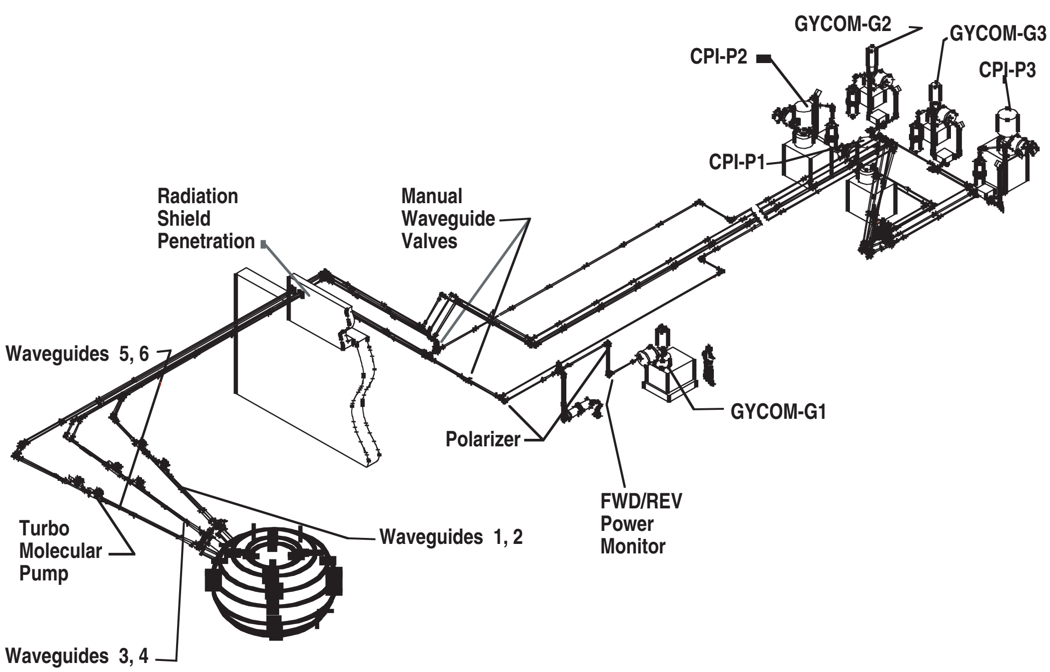

Fig. 8. Layout of the electron cyclotron system.

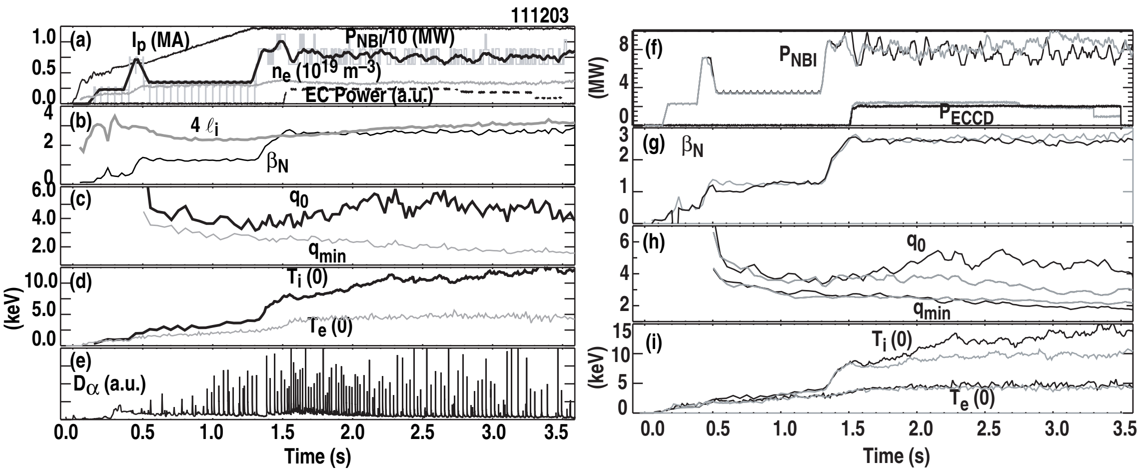

Fig. 9. (left side) Plasma parameters versus time for a discharge (111203) in which ECCD is used to modify the current profile: (a) plasma current (MA), neutral beam injected power (10 MW), line-averaged density (10^{20} m^3) and ECCD power (a.u.), (b) β_N (darker trace), 4 ℓ_i (lighter trace), (c) q_0 (upper trace), q_{min} (lower trace), (d) central ion and electron temperature and (e) divertor D_α (a.u.). (right side) Comparison of the temporal evolution of plasma parameters for similarly prepared discharges with off-axis ECCD (black) and ECH (grey) applied during the high performance phase: (f) neutral beam injected power and EC power (MW); (g) β_N; (h) q_0 (upper traces) qmin (lower traces); and (i) central ion (upper traces) and electron temperature (lower traces).

comparing to a similar shot in which ECH is used instead of ECCD it is clearly seen that the current change is produced by the ECCD. In. Fig. 10, the toroidal current density profiles inferred from the motional Stark effect diagnostic [13] are compared for the two discharges at the 1.5 s and 2.7 s. The ECCD contribution calculated by the CQL3D [14] code is also shown. The comparison shows that 120 kA of current is driven by the ECCD, which gives a normalized current drive efficiency η=0.26x10^{20} A/Wm^2 keV.

VI. DIGITAL PLASMA CONTROL SYSTEM

The DIII–D plasma control system (PCS) [15] consists of multiple high speed realtime processors communicating through a Myrinet high speed packet switched network. The PCS programs take a complex set of measurements and direct the control tools (magnetic coils, neutral beams, gas injectors, ECH, ICH, etc) to affect the plasma as desired. The controls are based on validated models, which are essential to control the advanced tokamak modes, for which the plasma parameters are interdependent in a highly nonlinear fashion. New algorithms are continuously developed to increase the control of the plasma and increase the reliability of the operation. Examples of such controls are the plasma shape control by controlling the current in the field shaping coils, operation at constant β_N by modulating the neutral beams, adjusting the current in the C-coils and I-coils to minimize the error fields and stabilize the RWMs, and the search and suppress mode used to stabilize the NTMs by either moving the plasma or changing the toroidal field to ensure that the ECCD is injected at the right location. Eventually the EC launchers will be controlled by the PCS so the ECCD can be directed to the right location without moving the plasma or changing the toroidal field. Disruption precursor detection has been developed for the vertical instability and will be used to trigger a mitigation system.

VII. DISRUPTION MITIGATION

A fusion reactor has to be able to handle or mitigate disruptions, which are a sudden loss of control of the plasma. The effect of plasma disruptions can be divided into electromagnetic loads, thermal loads, and run-away electrons. When a plasma disrupts, the energy stored in the plasma is rapidly lost, causing high heat loads on the plasma facing surfaces. These heat loads are non-uniform with most of the energy going to the divertor eroding its surfaces at an unacceptable rate. The electromagnetic force related to the sudden current quench can be so large that components in the vessel can be ripped off. The run-away electrons have high energies (~10 MeV) and can cause localized damage and component failures, if repeated run-away strikes should hit in the same area [16,17].

High-pressure gas injection of moderate Z noble gases can provide adequate mitigation of disruption caused damage. High-pressure gas injection experiments have been performed in DIII–D [18]. Early experiments using preemptive injection into a stable plasma have successfully been done. The jet species dissipate the plasma energy by radiation, providing thermal and halo current mitigation without creating run-away electrons. More recently, experiments using the PCS capability of detecting vertical instability have been done. However, a single reliable disruption precursor has not yet been found for all the different disruption types. It is important to note that detection of the final disruption event itself is not useful since the ideal instability growth rate is too fast (<1 ms) to allow for avoidance/mitigation. Therefore, we are planning on taking the following actions: 1) Install real-time plasma rotation measurement in the PCS as a precursor. Plasma rotation slowdown in the presence of MHD tends to indicate the onset of global destabilization well in advance (>10 ms) of the disruption. 2) Develop an algorithm to reliably account for radiated power fraction as a precursor for radiative limit disruptions. This entails developing a fast global radiated power measurement (from bolometry) and input of total heating power into the PCS (neutral beams, ohmic, ECH, etc.). 3) Install an MHD spectroscopy system to monitor the growth of Global Alfvén eigenmodes (GAE) in the 5–10 MHz frequency range. These modes may be playing a significant role in the onset of certain types of disruptions [19].

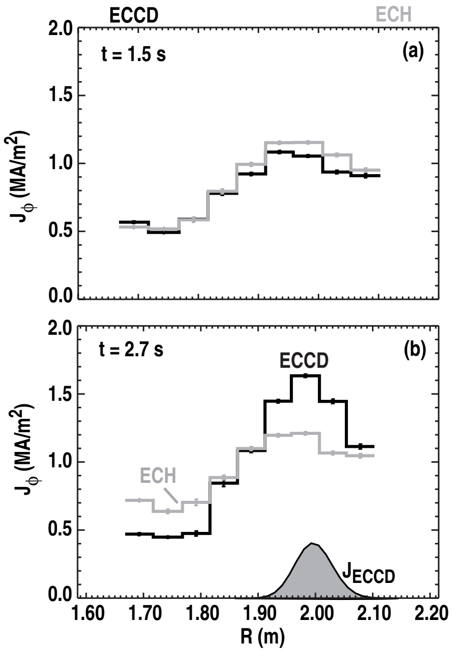

Fig. 10. (a)Comparison of the toroidal current density J_ϕ in the discharge shown in Fig. 9 (darker traces) to a case with ECCD replaced by ECH (lighter traces) at the beginning of the EC pulse (t = 1.5 s) and at t = 2.7 s. J_ϕ is inferred directly from the MSE measurement. Shaded region in (b) represents J_{ECCD} predicted by the CQL3D code.

VIII. NEW DIAGNOSTICS

The AT research which combines many aspects of plasma control in a highly non-linear fashion requires an extensive set of diagnostics, not just for probing the plasma but also for controlling the plasma. This requires high reliability of the diagnostics. The DIII–D tokamak is probably the best diagnosed tokamak in the world and new diagnostics are continuously added or old ones are upgraded. The new diagnostics are normally made by other institutions than GA. For the study of turbulence new density fluctuation diagnostics are being constructed: high k-scattering and microwave back scattering. The Phase Contrast Interferometer and beam emission spectroscopy are being upgraded.

The edge pedestal characteristics are very important for the plasma confinement and the heat and particle flux to the plasma facing components and divertors. A Li-beam diagnostic has been installed. To deduce the current density just inside the last closed flux surface, it is using the Zeeman splitting and the known polarization characteristics of the collisionally exited 670.8 nm Li resonance line to interpret local magnetic field components viewed using a closely packed array of 32 viewcords. The first data have been taken with the diagnostics.

VIIII. CONCLUSION

The DIII–D advanced tokamak research has reached the point where the different component of the AT mode now can start to be put together. Up to now the modes have normally been terminated by RWM or NTM. With the aid of the I-coils and the ECH, we have the ability to control these modes and we have been able to extend the AT mode. The capability of modifying the current density profiles has allowed us to start looking at near steady state discharges. The improvement to the plasma control system and addition and upgrades of diagnostics allow the required control of the many non-linear interdependent parameters. However, there is still a lot of work needed to demonstrated a fully integrated steady-state advanced tokamak discharge.

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