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DIII-D: RECENT PHYSICS RESULTS, IMPLEMENTED AND PLANNED HARDWARE UPGRADES

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DIII-D: Recent Physics Results, Implemented and Planned Hardware Upgrades*

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Abstract—During the last two years, the DIII-D tokamak has been operated for a total of 34 run weeks, during which significant advances have been achieved in many areas of plasma physics. This progress was only possible because of the improvements in the tools available to the DIII-D program to control and manipulate the plasma core and edge conditions. The important systems in this effort include the electron cyclotron (EC) system, the fast wave system (restarted after being sidelined for four years), 12 new internal coils, an upgraded plasma control system and a comprehensive set of turbulence diagnostics.

The EC system's versatility was demonstrated by the various roles it played in the physics research program. It was used as a probe to demonstrate the "hybrid" plasmas are regulated by m/n = 3/2 tearing modes, it was used to suppress the m/n = 2/1 Neoclassical Tearing Mode, which allowed the plasma pressure to be raised to new heights, and in an active feedback mode EC power was used to control the q-profile using real-time equilibrium reconstructions based upon motional Stark effect data. The fast wave system was used in conjunction with the EC system for current profile experiments. The internal control coil system was used to investigate suppression of the resistive wall modes and reduction and/or elimination of ELMs.

During this year, the DIII-D facility will implement major changes and upgrades to expand the frontiers in several of the areas of tokamak plasma physics research. One of the four neutral beams will be rotated from the co- to the counter-injection mode so that heated plasmas with little or no rotation can be studied. The present lower divertor will be removed and a new extended shelf divertor will be installed to provide the capability of pumping high triangular double null plasmas. Three new long pulse one MW class gyrotron systems will be brought on line, which will double the long pulse capability of the EC system. Two of the three aging cooling towers will be replaced with two new high efficiency towers that can handle the higher heat loads expected in the future from 10 s pulse operation. These and other improvements to the facility will be discussed and presented.

Keywords–tokamak, physics results, hardware upgrades

I. INTRODUCTION

The goals of the DIII-D research include basic plasma physics understanding and questions specific to the need of ITER. The basic physics areas that DIII-D is pursuing include plasma stability, transport, heating and non-inductive current drive, and divertor and edge physics. The ITER directed research includes extending high performance operation, stabilization of neoclassical tearing modes and resistive wall modes, control of edge localized modes (ELMs), and control and mitigation of plasma disruptions. There is of course no sharp boundary between the two goals.

In support of this research DIII-D has an impressive number of diagnostics and an advanced digital control system that enables us to get a better understanding of the plasma behavior and use the acquired knowledge to control the plasma. More and more diagnostic signals are being used in real time by the plasma control system, which does not just control the plasma shape and profiles, but also is used to control the instabilities mentioned above.

In order to pursue the research it is important to recognize the symbiotic relationships between theory, experiments and hardware. As the understanding of the plasma is improved new hardware capabilities are needed to test the new theories. While gradual upgrades to the DIII-D tokamak have been made over the years, further progress in understanding advanced tokamak physics motivated several major upgrades to the DIII-D tokamak. This is being done during a twelve month down period from April 2005 – March 2006, referred to as the Long Torus Opening.

The paper is ordered as follows; first some experimental results from the advanced tokamak and hybrid scenarios that have been obtained during the last two years will be presented (Section II). Then the advances in the control of the resistive wall and neoclassical tearing mode instabilities will be discussed in Section III. The use of the internal I-coils to suppress ELMs will be discussed in Section IV. New understanding of the effect of using a massive gas puff to mitigate the effect of disruptions is the subject of Section V. Upgrades to the hardware and software made to the DIII-D plasma control system is the subject of Section VI. Diagnostic upgrades are discussed in Section VII, and Section VIII deals with other hardware upgrades. Section IX describes the upgrades that are currently being made to DIII-D as part of the Long Torus Opening Activities (LTOA). A conclusion is given in Section X.

II. ADVANCED TOKAMAK AND HYBRID SCENARIOS

The mission for DIII-D is "to establish the scientific basis for the optimization of the tokamak approach to fusion energy production". Significant advances have been made toward this goal, and since the US has rejoined ITER and the site selection is completed, there is new emphasis on exploring advanced operating modes for ITER that will allow it to exceed its original

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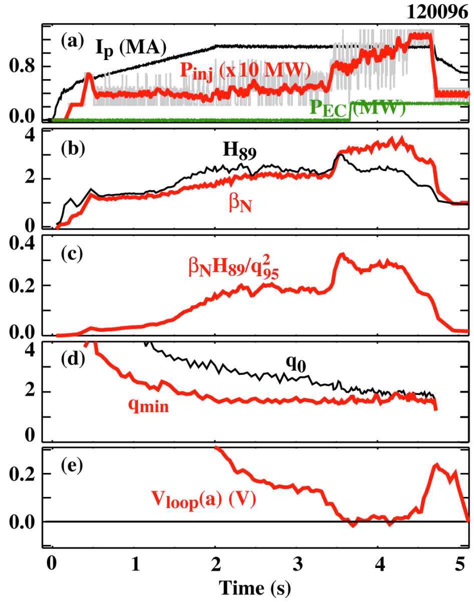

Figure 1. The time evolution of a DIII-D AT discharge with 100% of the plasma current driven noninductively.

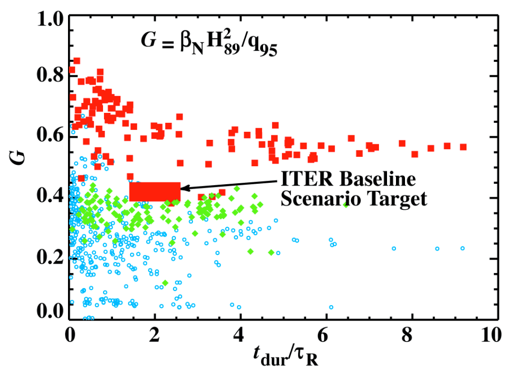

Figure 2. Normalized fusion performance $G = \beta_N H_{89} / q_{95}^2$ versus duration normalized to current relaxation time for DIII-D discharges. Filled squares are ITER baseline scenario discharges, filled diamonds are hybrid scenarios and open circles are other types.

goals. Advanced tokamak modes [1,2] are highly shaped plasmas, normally with high elongation and triangularity, operating above the no-wall β_N limit and with 100% non-inductive current drive. [$\beta_N = \beta_T / (I/aB)$ and the no-wall limit is the limit at which a plasma with no surrounding walls will be unstable to a "kink" instability.] Recent efforts have been to extend the modes to periods longer than the equilibration time of the current profile τ_R. Two discharges will be discussed; one is an advanced tokamak discharge, with 100% non-inductive current drive where the q-profile is controlled and the high performance phase is extended. The second discharge is a "hybrid", so called because it does not have 100% non-inductive current drive but would extend the pulse length in ITER by more efficient use of the OH transformer flux.

The approached taken to establish an AT discharge in DIII-D is to create the desired q-profile during the plasma current ramp-up and the early flattop phases and sustain it during the subsequent high beta phase using off-axis ECCD combined with bootstrap current and neutral beam drive (Fig. 1) [3]. To maintain relatively high $q_{\min}$, an H-mode transition is induced at 430 ms and to improve reproducibility feedback control of β_N was used beginning at 500 ms. The high beta phase begins at 3400 ms when $q_{\min} \sim 1.7$ ($q_{95} \sim 5$). In discharges similar to this, with up to 2.5 MW of off-axis electron cyclotron current drive (ECCD) and up to 15 MW neutral beam injection, ~100% of the plasma current has been sustained non-inductively for 1 s at high beta ($\beta = 3.6\%$, $\beta_N = 3.4$) above the no-wall limit.

In the hybrid scenario DIII-D has demonstrated [4] normalized fusion performance which meets or exceeds the ITER goals under stationary conditions (Fig. 2). $G = \beta_N H_{89} / q_{95}^2$ is a term for fusion gain used to compare the performance of current tokamaks with ITER. The baseline design for ITER [5] indicates $G = 0.42$ is needed to achieve $Q_{fus} = 10$ for ~2 τ_R, where Q_{fus} is the ratio of fusion power produced to input power supplied.

A DIII-D discharge [4] that obtains high performance with magnetic geometry similar to the ITER baseline scenario is shown in Fig. 3. The plasma parameters are: $I_p = 1.2$ MA, B_T=1.7 T and $n_e = 5 \times 10^{19}\,\mathrm{m}^{-3}$. Moderate neutral beam injection power is applied during the current

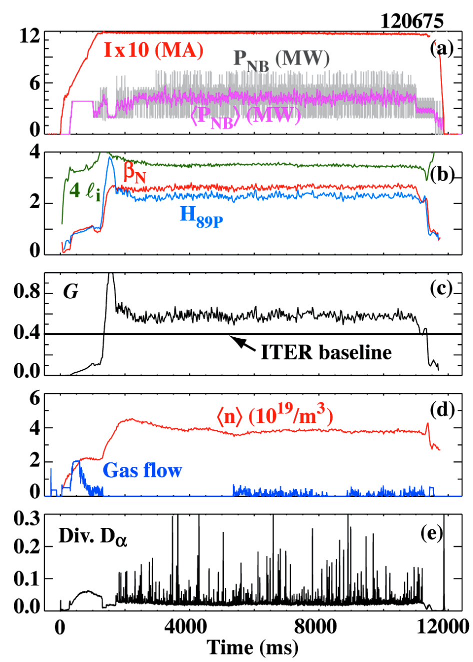

Figure 3. Time histories of plasma parameters for an ITER baseline scenario discharge from DIII-D (B = 1.26 T). (a) Plasma current (I)x10 (MA) (red), NBI power (P_{NB}) (MW) (gray), and time-averaged NBI power ($\langle P_{NB} \rangle$) (MW), (b) normalized pressure (β_N) (red), normalized energy confinement (H_{89}) (blue), internal inductance $\langle \ell_i \rangle$ x 4 (green), (c) normalized fusion performance (G), (d) line-averaged density ($\langle n \rangle$) ($10^{19}\,\mathrm{m}^{-3}$) (red), D_2 gas flow (a.u.) (blue), (e) upper divertor D_α (a.u.).

ramp to obtain nearly flat q-profile at the end of the current ramp. Then the normalized pressure $\beta_N = 2.7$ is maintained by feedback control of the neutral beam injection power. The average neutral beam power is 4.3 MW. The normalized fusion performance $G = 0.58$ at $q_{95} = 3.2$ is sustained for 9.2 τ_R. Shortly after the high β phase begins, a small 3/2 neo-classical tearing mode is triggered followed by small periodic 1/1 sawteeth. The proximity to the $n = 1$ ideal no-wall pressure limit is shown by comparison of β_N to 4 ℓ_i, which is a good estimate of that limit in these discharges. The density is controlled by continuous gas puffing and pumping by cryopumps in the divertor. Detailed analysis of the plasma profiles indicates that the plasma pressure and current profiles are truly stationary. The termination of the discharge is due to the limit of neutral beam energy throughput.

Whereas discharges like the one just described, called hybrid discharges, give confidence that ITER will reach it goal, they do not push the ultimate capability of advanced tokamak modes. The hybrid discharges seems to indicate that the presence of 3/2 neo-classical tearing modes are effective in eliminating sawteeth when q_{95} is greater than 4 or reduced them when q_{95} is close to 4.

III. RWM AND NTM INSTABILITIES

DIII-D can operate with higher triangularity (up to 0.9) than is possible for ITER. The highest performance is reached in these discharges, which however are normally terminated by either resistive wall modes or neoclassical tearing modes. In the absence of a wall surrounding the plasma, kink instabilities are excited when β_N is raised above a certain limit $\beta_{N-no\ wall}$, which is about 4 ℓ_i. Resistive wall modes are kink modes in the presence of a nonperfect conducting wall. In DIII-D the wall is several centimeters away from the plasma boundary. In DIII-D these modes can be stabilized by plasma rotation or non-axisymmetric feedback coils, the C-coils or I-coils (Fig. 4), and the β_N can be increased to close to the ideal, perfectly conducting wall limit β_N^{Ideal} [6]. The stabilization of RWMs in ITER [7] has been studied and a proposal has been made to have control coils outside the vacuum vessel. The results from DIII-D have shown that both coils outside the vacuum vessel, the C-coils, and coils inside the vacuum vessel, the I-coils, can stabilize the RWMs. The I-coils have been shown to be more effective in stabilizing the modes. The best results have been obtained by using high bandwidth audio amplifiers with the I-coil and using the C-coils with slow switching power amplifiers.

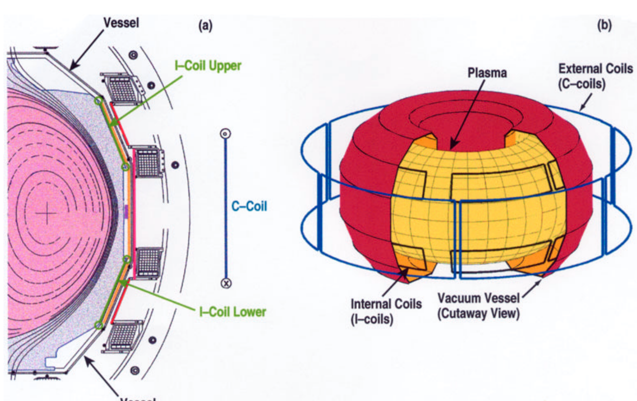

Figure 4. A schematic of the C-coils and I-coils installed on DIII-D. The coils are used to correct error fields, stabilize the resistive wall modes and produce edge stochastic fields to control ELMs.

In DIII-D the plasma is normally rotating, since all the beams are injected in the same direction and thus transfer a net momentum to the plasma. The critical rotation speed for stabilization of the RWMs was measured in DIII-D [8] for low ℓ_i and moderate ℓ_i, by raising β_N above $\beta_{N-no\ wall}$ by increasing the neutral beam injection. An incomplete correction of the magnetic error fields leads to a decrease of the plasma rotation, Fig. 5. Once the rotation is no longer sufficient to stabilize the mode, the RWM grows, leading to a much faster decrease of the rotation and a β-collapse. The onset of an exponentially growing $n = 1$ RWM marks Ω_{crit}. For low ℓ_i, Ω_{crit}, measured at the $q = 2$ surface, was found to be independent of β with $\Omega_{crit}\tau_A = 0.02$. τ_A is the inverse of the Alfvén time. For moderate ℓ_i, Ω_{crit} is found to be lower by a factor 2. Experiments have been performed, where the plasma rotation has been slowed down below the critical rotation speed. In these experiments the I-coils were able to stabilize the RWMs and β_N was increased above the $\beta_{N-no\ wall}$ limit, suggesting that RWM stabilization can be done with coils for non or low rotating plasma. The final test will however be to do the experiment in a plasma with no rotation or no net angular momentum input, which will be possible in DIII-D in 2006, when one of the neutral beams is turned around for counter-injection.

A set of 12 audio amplifiers have been purchased and installed at DIII-D. The audio amplifiers can be connected to the I-coils and used for fast response to stabilize the resistive wall modes, while the normal switching power amplifiers (SPAs) were used on the C-coils for slow response and correction of the inherent error fields. The first results with the audio amplifiers show that they do respond faster than the SPAs.

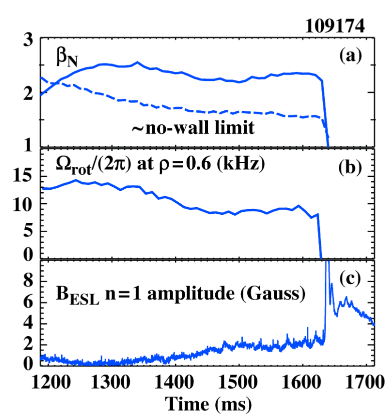

Figure 5. (a) When β_N exceeds the no-wall limit, (b) the toroidal plasma rotation decreases until the onset of an $n = 1$ RWM (c) seen with external saddle loops (ESL) in the mid-plane.

Neoclassical tearing modes are other instabilities that can limit high performance plasmas. Both the 3/2 and 2/1 modes have been stabilized in DIII-D by applying electron cyclotron current drive (ECCD) to the island [9,10]. During the last two year the control system has been upgraded to include the real time equilibrium reconstruction algorithm, which calculates the real time q-surfaces, and the active tracking system, which follows the movement of the q=3/2 or 2/1 surfaces, depending on which mode is the target for stabilization. With the planned upgrades of the ECH system and the ECH launchers (see Section IX) it should be possible to stabilize both modes simultaneously. The active tracking system has also been used to preemptively avoid the 3/2 or 2/1 modes by aligning the corresponding q-surface with the ECCD location. The 3/2 and 2/1 modes can also be avoided by operating at higher q values, so that there are no q=3/2 or q=2/1 surfaces in the plasma.

IV. ELM SUPPRESSION WITH EDGE RESONANT MAGNETIC FIELDS

Large edge localized modes (ELMs) produce rapid loss of energy from the pedestal region and can cause excessive erosion of the divertor in burning plasma tokamaks including ITER. In DIII-D, the I-coils have been used to eliminate all but a few isolated Type-I ELMs [11]. In the configuration shown in Fig. 6, the coils produce a vacuum field which resonates with the plasma flux surfaces across most of the pedestal region, when q_{95}=3.7±0.2 creating small remnant magnetic islands surrounded by weakly stochastic fields lines. Whereas a variety of ELM suppression and modification results have been obtained during experiments in DIII-D the ELM suppression results found in this configuration are interesting, since they leave the time average electron profile essentially unaltered, along with the global energy confinement time, the radial electrical field and the poloidal rotation profile. Good ELM suppression is characterized by a global change in the dynamics of the D_α recycling light at various toroidal and poloidal positions. An example of such a change is seen in the divertor D_α ELM dynamics, when the I-coils are applied to a discharge in the reference shape shown in Fig. 7. The fixed Langmuir probes in the lower divertor, measuring ion saturation current, also see a reduction in the particle flux. Small 130 Hz oscillations are observed between intermittent Type-I ELM like events. This coherent oscillation appears to result from a process that produces about the same time averaged transport through the pedestal as is seen during the Type-I ELMing phase. During the ELMing phase as much as 30 kJ of the pedestal energy is transported into the scrape-off layer during one ELM period, about 200 μs or less, whereas during the ELM suppressed period it is less than 15 kJ in about 500 μs, thus significantly reducing the peak power to the divertor. ELM suppression has also been seen in isolated windows with q_{95} higher than 3.7. However the ones at 3.7 have consistently produced good suppression results. The range and position of the window is expected to change with the shape of the discharge [11] and the pedestal parameters. These ELM suppression results have also been achieved at pedestal collionalities close to those desired for ITER.

V. 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. A thermal collapse of the plasma occurs first in a few hundred micro-seconds, causing high heat load to the plasma facing components with most of the energy going to the divertor eroding its surface at an unacceptable rate. Then on a few millisecond time scale follows a quench of the plasma current and it is associated with eddy currents in the surrounding conductors and with halo current flowing from the plasma into the plasma facing components. The forces from the halo currents can be so large that components in the vessel are ripped off. Finally in many cases run-away electrons can be created. The run away-electrons have energies (~10 MeV) and can cause localized damage and component damage, if the repeated run-away strikes should hit in the same area [12,13].

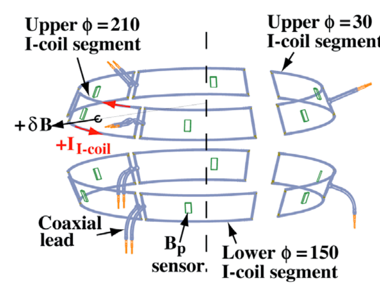

Figure 6. The I-coil is comprised of six segments above the equatorial plane (upper) and six segments below (lower).

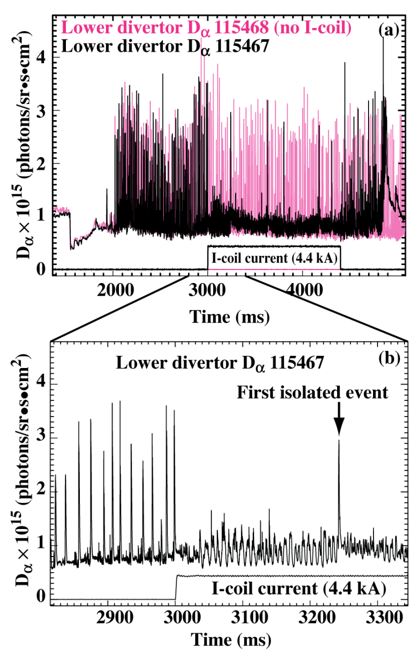

Figure 7. (a) Lower divertor D_α signal near the outer strike point showing Type-I ELM suppression in discharge 115467 (black) during a 4 kA I-coil pulse compared to an identical discharge, 115468 (magenta), without an I-coil pulse. (b) A shorter time window showing a change in the dynamics across the I-coil turn on time for the Type-I ELM suppression discharge only.

Disruption and mitigation of disruptions have been studied in DIII-D for many years [14,15]. The DIII-D machine is robust to disruptions and does not have problems recovering after disruptions. The normal 8 minute He glow discharge between shots is sufficient for the next shot to be unaffected by a previous disruption. The DIII-D machine is therefore ideal for disruption studies. Disruption mitigation studies have been done with impurity pellets and massive gas puff. High-pressure gas injection has been shown to provide fast shutdown with a significant reduction of halo currents and thermal loads, but without the generation of significant run-aways, which is normally seen when cryogenic argon or neon pellets were used for mitigation. Understanding the dynamics of the radiating impurities is crucial for evaluating pellet and high-pressure gas injection as potential mitigation techniques for ITER. Recent measurements in DIII-D [15] indicate that the impurity transport during high-pressure gas injection in DIII-D is complex and can occur in several stages. The configuration is shown in Fig. 8. The jet neutrals typically appear to stop soon after hitting the plasma edge. Simulations indicate that the neutral cloud is opaque to plasma electrons [16]. Impurity ions and the associated cold front then begin to diffuse radially inward. The penetration of the neutral gas is observed with a high speed imaging camera. When this cold front reaches sufficiently far into the plasma core, typically around q=2, an explosive growth of MHD instabilities occurs. Most of the plasma thermal energy is radiated away, although a complete mixing of the impurity ions and hot plasma does appear to occur. Despite the large MHD, divertor and main chamber heat loads appear smaller than in normal disruptions. These results suggest that ideal, uniform deposition of neutrals may not be required for disruption mitigation in future tokamaks.

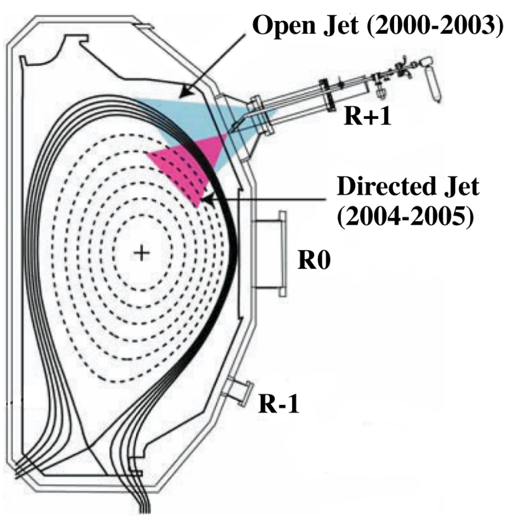

Figure 8. Configurations used for disruption studies. The two configurations give the same mitigation results, whereas the “edge fill” is less than 1 ms in the open jet, whereas it is several ms for the direct jet. The reservoir pressures are between 20 – 80 atm.

VI. PLASMA CONTROL SYSTEM

Many upgrades in both hardware and software have been made to the DIII-D real-time digital plasma control system [17] during the last two years and more are scheduled to be made during the LTOA period. The total number of processors has been increased from 9 to 13 and the number of real time displays in the control room has been increased from 4 to 7. CPUs were dedicated to the real time EFIT [18], which comes in two versions, one which include the current density profile obtained from the motional Stark effect (MSE) data and one without MSE for speed. Additional CPUs are dedicated for collecting data from the following diagnostics, Thomson (32 ch), MSE (32 ch) and ECE (32 ch). A CPU has been dedicated to the resistive wall mode control. The software upgrades include improvement to the EFIT equilibrium reconstruction with E_r correction and spline parameterization. The equilibrium grid is 32x32 and reproduced every 4-8 ms and includes the 3/2 and 2/1 q-surfaces for NTM control. The NTM suppression algorithm includes an option that constrains a chosen q-surface (2/1, 3/2) to coincide with a fixed ECCD deposition location using rigid radial shifts of the plasma or B_T variation. β_N feedback control has been improved with the real time EFIT improvements, a better PID gain selection, and improvements in the individual feedback beam sequencing software. Real time Thomson electron temperature and density profiles are constructed, as are ion temperature and plasma toroidal rotation velocity profiles. Algorithm improvements include Kalman filters, which are used to discriminate against ELMS for RWM control. The RWM algorithm now also includes the control of the new audio amplifiers that are used to drive the I-coils for RWM stabilization. Algorithms have been developed for two point electron temperature feedback control and work has begun on developing feedback control of the q-profile using neutral beams, ECCD, and Fast Waves.

Advanced shape control has been demonstrated using MIMO control. This control has three basic features: it takes into account every coil’s effect on each control point and x-point; it includes a pre-computed, model based shaping coil current trajectory for each coil; and machine hardware limitations are incorporated to closely match the shape given those limitations. An example of such a limitation is the field shaping coils’ current limits.

During the LTOA it is the plan to add more MSE and Charge Exchange Recombination (CER) channels to the PCS. The number of processors will be increased from 13 to 24, and the EFIT equilibrium grid will be increased to 65x65 with a cycle time of 4-8 ms. The RWM control cycle time will be reduced to 25 µs and possibly 10 µs. Real time sweeping will be added to 1 of 3 of the gyrotron launchers. With this capability the plasma will not have to be moved or the toroidal field change to align the 2/1 or 3/2 q-surfaces with the ECCD resonance location for NTM stabilization.

VII. DIAGNOSTIC UPGRADES

Several of the DIII-D diagnostics have been upgraded during the last two years. Of particular interest are the upgrades and additions to the turbulence diagnostics. The FIR (far-infrared) scattering has been augmented by two systems one, in the microwave band in a back-scattering configuration, one at the equatorial level (sensitive to density fluctuations k vector range near 10 cm^{-1}), and one at an upper port (with k vectors near 40 cm^{-1}). The BES (beam emission spectroscopy) system was upgraded with the installation of 16 high-sensitivity, low noise chords, which enabled a dramatic increase of the observed signal to noise ratio by a factor of 30, enabling routine measurements all the way near the core of the discharge. The PCI (phase contrast imaging) diagnostic was relocated and upgraded, changing its line of sight close to the center of the plasma, and increasing its k-range to nearly 50 cm^{-1}. The new capabilities provided by these fluctuation diagnostics have led to the observation of a “sea of Alfven Eigenmodes” in the core of the DIII-D plasmas, closely matching some recent code simulations. In the boundary science area, the DiMES probe was used to insert molybdenum mirror surfaces about 2 cm below the floor tiles. This experiment, which is crucial for the determination of life expectancy of mirrors in an ITER environment, showed deposits on unheated mirrors, whereas there was no deposit on mirrors heated to 80 – 140°C. Also, two fast framing cameras were installed during the last campaign for imaging of rapidly evolving plasma events (disruption, gas jet mitigation, ELMs, etc.). Several other diagnostics upgrades are schedule to be made during the LTOA period, including the addition of new MSE and CER systems viewing the newly rotated counter-beam.

VIII. HARDWARE UPGRADES

Modest, but significant, upgrades have been made to DIII-D during the past two years. The most significant hardware addition was a new neutral beam ion source built by GA [19]. This is the first time in 19 years that a new grid rail for the DIII-D ion sources has been built. The ion source was used during the last four weeks of operation in 2005.

The I-coils were strengthened with braces near the feeds inside the vessel as well as outside to minimize vibrations that caused metal fatigue and eventually caused a leak. In addition motion detectors were install to all the external leads and port displacement monitors were installed to detected additional motion of the external leads and shut the system down if the motions become too excessive. The problem only occurs, when the coils are operated in a $n = 3$ configuration with currents varying at frequencies of 80–100 Hz. The coils were not designed for operation in the $n = 3$ mode, but this configuration is now being used for ELM suppression experiments.

A new C power supply (300 V, 6 kA) was installed to improve the operation for the RWM experiments. Six new audio amplifiers were installed for the same reason. Two amplifiers in parallel per quartet of I-coils can provide 800A, when used with coupling transformers.

IX. LONG TORUS OPENING ACTIVITIES

During the period, May 2005 and April 2006, the DIII-D tokamak will not be operating in order to upgrade the facility to provide new scientific capability and to maintain the required infrastructure. There are seven specific activities scheduled for the Long Torus Opening period: (1) the purchase of three long pulse replacement gyrotrons, (2) refurbishment of the subsystems of the three short pulse gyrotrons to be able to operate the long pulse replacement gyrotrons, (3) conditioning the three long pulse replacement gyrotrons so that they are ready for operation in April 2006, (4) rotating the 210 neutral beam line for counter-injection, (5) installing a high triangularity lower divertor, (6) replacing two old cooling towers with two new ones, and (7) upgrading the toroidal return beltbus to 10 s full field (2.2 T) capability. All the tasks including the known maintenance and refurbishment tasks have been scheduled and are tracked in minute details using Microsoft Project 2003.

The purchase and installation of three long-pulse gyrotrons are motivated by the many exciting results that have been obtained with ECH already. The ECH in contrast to neutral beams, only heats electrons, heats in a very localized volume, and does not add particles to the plasma. ECH is used for heating the electrons, controlling the current profile, sustaining high performance plasma through current profile control, and stabilization of the neoclassical tearing modes. It is valuable for studies of transport barrier modulated transport and critical gradients.

The rotation of the 210 neutral beamline is motivated by the possibility of studying Quiescent H-modes (QH-modes) with central co-rotation and understanding the physics of rotation. With a balanced set of co- and counter-beams it should be possible in the future in DIII-D to have high performance, low or non-rotating plasmas, which up to now has not been possible, since all the beams in DIII-D have been injecting in the same direction. Low or non-rotating plasmas are interesting since it is believed that ITER with its high energy neutral beams requirement will have low or non-rotating plasmas. For DIII-D the new configuration will allow studies of resistive wall modes in low rotation plasmas and neoclassical tearing mode stabilization with modulated rf. The plasmas in DIII-D have been rotating too fast to try modulated rf stabilization of NTMs.. The new configuration will also open the possibility for getting a better understanding of the physics of neutral beam current drive, fast ions and transport barrier control. The new configuration will also increase the accuracy of the MSE and CER measurements with additional viewing cords for the two diagnostics.

The main emphasis of the DIII-D research program is the study of high performance advanced tokamak modes. The highest performance was obtained in high triangularity (δ =0.9 double null plasmas. However, the lower divertor that has been in DIII-D was not able to pump high triangularity plasmas. Modeling has shown that the two upper cryo pumps with their current location and one lower cryo pump with the pumping aperture further radially inward are sufficient for pumping high triangularity plasmas. An inner lower pump would not add much additional density control. A new lower divertor [20] has therefore been designed and is being fabricated in collaboration ASIPP in China. The new divertor will give density control for high performance single and double null advanced tokamak plasmas and for QH-modes. It will allow studies of pedestal physics with a range of pedestal density (n_e). The divertor is built such that low and medium triangularity single null plasma shapes including the ITER shape can be accommodated by having the outer strike point on top of the divertor plate. Only private flux or no pumping will be possible in this configuration. This will also allow optical access to the inner divertor leg without looking through the outer leg. The three lower rows of carbon tiles on the center post will be contoured in order to reduce the carbon flux into the plasma.

Upgrade of the water cooling system is required because of the additional ECH power and because two of the existing towers are have reached the end of their service life. The upgrade of the toroidal return beltbus, which currently is limited to 5 s at full field, is the need to eventually produce 10 s plasmas in DIII-D. The toroidal coil itself is capable of 10 s.

X. CONCLUSION

The paper has described the progress that has been made at DIII-D in the advanced tokamak and the hybrid modes. The hybrid modes projection to ITER shows a path whereby ITER might exceed its design values. Progress has also been made in the understanding and stabilization of the RWMs and NTMs. The I-coils in DIII-D have been used to produce a stochastic field at the plasma edge, which eliminates nearly all ELMS. The understanding of the massive gas puff mitigation of plasma disruptions has been increased with high speed images showing that the process is complicated and that the neutral gas penetration stops close to the plasma edge, but nevertheless, the thermal energy is radiated away. Upgrades to the plasma control system continues with adding more diagnostic data and control of more plasma parameters, e.g., q. The recent hardware upgrade and the seven major upgrades currently underway at the DIII-D facility should provide significant new capabilities for the DIII-D research program.

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