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DISCLAIMER

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The gyrotron complex on the DIII-D tokamak was used for 34 weeks of operation during the last experimental period. The entire complex is now shut down for a year while the DIII-D tokamak and the ECH systems undergo major modifications. The principal projects underway during this period are: rotation of one of the neutral beam lines to inject counter to the plasma current; installation of a new divertor; acquisition and installation of three new 110 GHz gyrotrons with the present 1.0 MW, 10 s performance specification; and installation and testing of a single stage 110 GHz depressed collector gyrotron having >1.0 MW long pulse capability. Following the modifications, the ECH system will comprise six production gyrotrons, plus the depressed collector prototype and a short pulse gyrotron held in ready reserve. Of these, six gyrotrons will be available for experiments at one time using the six transmission lines and three dual launchers. At least one of the launchers is expected to be capable of real time fast spatial scans at the resumption of operations.

Introduction

A multi-faceted experimental campaign has concluded on the DIII-D tokamak. In the area of evaluation of electron cyclotron current drive (ECCD) efficiency, a new technique was developed using EC injection modulated at 10 Hz to create a periodic response in the motional Stark effect polarimeter leading to a direct measurement of the spatial profile of the ECCD. Experiments on high performance operation were performed in which off-axis ECCD was used to adjust the current density profile. Noninductive discharges with about 60% bootstrap fraction were created in this way. By controlling the m/n = 3/2 island width, stationary discharges were produced having q(0) slightly >1.0, which scaled to inductive ITER operation for 1 h at Q = 5-10. Preemptive injection of EC power at the $q = 2$ surface was successful at preventing initiation of the m/n = 2/1 neoclassical tearing mode (NTM) and studies were performed on the control of the ECCD profile width required for stabilization of both 2/1 and 3/2 neoclassical tearing modes once the instabilities had developed. A series of measurements of the efficiencies of the separate elements of the electron cyclotron heating (ECH) transmission system and thermal performance of the articulating launchers was performed.

The DIII-D ECRH Complex

During the 2004-2005 campaign, the DIII-D electron cyclotron resonance heating (ECRH) complex [1] comprised up to six gyrotrons operating simultaneously, although the typical number of operating tubes was four, owing to several failures, which will be discussed below. The generated power was about 3 MW, of which about 2.2 MW was injected into the tokamak after passage through the transmission lines. The system includes two groups of gyrotrons in the MW class, with maximum pulse lengths at full power of 2.0 s and 10 s for the two types. Typical DIII-D maximum pulse lengths are 6-7 s and nearly all the experiments were performed with rf pulse lengths of 2-4 s. The present status of the gyrotrons that were operated during the most recent campaign is shown in Table 1.

The oldest gyrotron in the complex, G1 (Katya), has been operating for about 8 years and has exhibited excellent reliability. This gyrotron has had a strong 100 MHz parasitic emission from the time of its first operation. The generated power from this tube has slowly decreased due to a slow decrease in cathode performance over time, but the generated power is still about 600 kW for 2 s pulse length.

Table 1. Summary status of the gyrotrons which have seen extensive service in the DIII-D system (τ_p is the maximum pulse length.)

Gyrotron	Manufacturer	Date	P_{GEN}/τ_p (MW/s)	Operational Status/History
G1 (Katya)	Gycom*	1996	0.6/2.0	Normal operation/parasite; cathode efficiency is slowly decreasing, but operation is reliable with $P_{GEN} \sim 600$ kW.
G2 (Boris)	Gycom	2000	0.65/2.0	Removed from service in good condition for eventual use at PPPL on the National Compact Stellarator Experiment. $P_{GEN} \sim 750$ kW.
G3 (Natasha)	Gycom	2000	0.5/2.0	Retired/vacuum failure. Will be sent to PPPL for use on NCSX if repaired.
P1 (Scarecrow)	CPI†	2000	0.85/5.0	Normal operation. The gyrotron had its window replaced after failure of the aluminum seal and later had its collector replaced after a large water leak developed.
P2 (Tinman)	CPI	2000	0.65/2.0	Has a small leak in the bottom of the collector, which limits performance. The tube has been removed from service and sent to CPI for repair.
P3 (Lion)	CPI	2002	0.85/5.0	Gyrotron had a collector failure with substantial overheating, slight buckling of the structure and extensive stress cracking on the hot sides of the cooling tubes. Being repaired.

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†Communications and Power Industries, 831 Hansen Way, Palo Alto, California 94304, USA.

In 2000, the DIII-D project acquired two similar gyrotrons from the closed Tokamak de Varennes. Both of these tubes performed well during the 2004-2005 and previous campaigns until the G3 (Natasha) gyrotron developed a vacuum leak and was removed from service. The DIII-D program plan calls for replacement of all the short pulse gyrotrons, which are limited in pulse length by heating of their boron nitride output windows, so G3 (Natasha) will not be repaired. At the end of the campaign, G2 (Boris) also was removed from service and is expected to be available to the National Compact Stellarator Experiment [2] at Princeton in about three years. Because the G1 (Katya) gyrotron is located in an area much closer to the tokamak than the other gyrotron sites, this tube is being kept in ready reserve, with the possibility of connecting to any of the six transmission lines midway between the main group of gyrotrons and the tokamak. The configuration planned for startup in late spring 2006 is shown in Fig. 1.

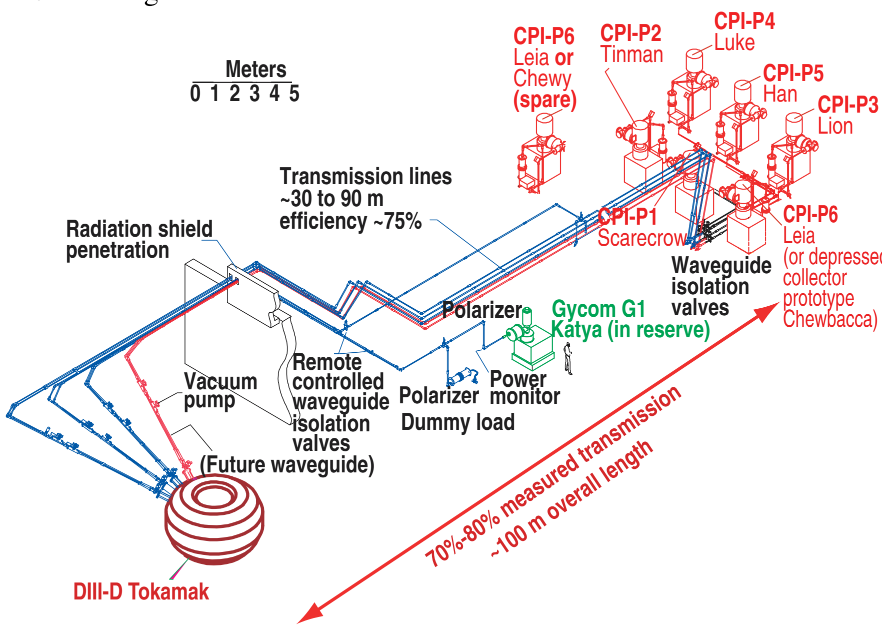

Fig. 1. Schematic diagram of the DIII-D gyrotron complex as it will exist for plasma operations in 2006. A total of six 1.0 MW, 10 s tubes will be available in addition to a depressed collector prototype operating at 1.2 MW, 10 s and a single short pulse tube, generating about 600 kW, 2.0 s in ready reserve. The system will have six transmission lines and two rapid scan launchers.

The DIII-D tokamak is presently in the middle of an extensive project to modify and upgrade the facility [3]. To provide for the possibility of balanced injection and co- versus counter-neutral beam injection experiments, one of the four NBI tanks is being rotated about 39 degrees to inject in the counter-current drive direction. At the same time, a new lower divertor geometry is being installed, which will permit higher triangularity discharges to be run. The final major element in the project will be upgrades to the gyrotron complex.

The gyrotron complex which operated for the 2004-2005 campaign will be improved for the 2006 campaign by replacement of the short pulse tubes with new tubes having CVD diamond output windows [4] and nominal 1.0 MW, 10 s output pulses [5]. These tubes are nearly identical to the first group of three long pulse tubes already installed. The major difference has been the addition of two 8 liter/s vacion pumps in the region of the output window in addition to the two 75 liter/s pumps in the collector. The testing to half power at 10 s pulse length for the first of the new gyrotrons was done more rapidly than had been the previous experience. About one month of around the clock operation was required, possibly because of the presence of these additional small pumps in a critical region. The first of the three new gyrotrons in this set was received at DIII-D in September and is expected to be ready for plasma operations in November. The second is expected to be finished with initial testing at CPI in November and will then begin testing at DIII-D to full parameters early in 2006. The third tube in the series will follow about 3 months later.

In parallel with the production of these three gyrotrons, CPI and the U.S. Gyrotron Development Program have built a single stage depressed collector gyrotron [6], which was tested for short pulses at 1.2 MW and measured efficiency of 44%. This gyrotron will be tested to full parameters at DIII-D during testing of the other three new tubes. Depending on the results of these tests, this tube either will remain as part of the DIII-D complex, or will be returned to CPI for modifications. There should be seven long pulse high power gyrotrons and one short pulse tube at DIII-D during the 2006 campaign, of which six can be used at any one time until additional transmission lines and launchers are completed. At least one of the dual launchers will have real time fast scan capability under control of the plasma control system. The steering mirrors will be capable of scanning at 10 deg/s for ±20 deg travel and at a rate of 100 deg/s for 2 deg movements. The positional accuracy for the rf beam will be about 1 cm at the tokamak center.

Gyrotron Failures on the DIII-D System

A number of failures have occurred on the DIII-D gyrotron system during the past year. These were all related to loss of gyrotron vacuum. The Gycom gyrotron G3 (Natasha), which had been in service at DIII-D since 2000, developed a vacuum leak during normal operation. The leak was moderate, but prevented further operation. Because of plans to replace the short pulse gyrotrons in the DIII-D system with long pulse tubes, it was decided to retire, rather than repair, this gyrotron. No specific cause of the failure was identified, although this gyrotron had been the most difficult of the short pulse tubes to operate for some time and also produced the lowest power for reliable operation.

Vacuum leaks localized to the collectors were experienced on all three of the CPI long pulse gyrotrons in the DIII-D complex. The common thread in the failures was operation during initial testing with collector power loading approaching 1 kW/cm^2. This limit had been established earlier in the gyrotron development program when initial testing at half power, 80 kV, 25 A, was performed without sweeping of the electron beams in the collectors. The cause of the first failure was cracking of the copper collector, but the cause was attributed to material problems with the copper. Following the second failure, which was concluded to have been caused by thermal stress in the copper collector, the stress calculations were revisited using more sophisticated models, with the result that a more conservative limit of 600 W/cm^2 was established. Collector sweeping during all gyrotron operation was required and stronger sweeping was implemented in order to meet the new limit. In addition, four quadrant power supplies capable of ±100 V, ±20 A were tested for use with the next group of three gyrotrons for the DIII-D system. These power supplies can produce an arbitrary time dependence of the sweep coil current. A test of a sawtooth waveform at 7 Hz gave a scalloped sweep magnetic field in the collector with minimum dwell at the low point in the collector sweep when the electron beam footprint is smallest. After limited operation under the new limit, no further failures have occurred.

One additional vacuum failure occurred in a long pulse gyrotron. In this case, the leak was in a poorly brazed joint in the cavity assembly, which could have been stressed by freezing of the stagnant cavity cooling water during a planned warm-up of the magnet over a holiday period. Although it was impossible to demonstrate bore freezing during subsequent tests, and no distortion of the cavity assembly, which could have been the result of freezing, was seen, new procedures for warming the cryogenic magnets were implemented, bore heaters were installed and a low temperature alarm was developed to forestall any similar problems. The three new gyrotrons in the current upgrade project will be equipped with cryogen-free magnets using Joule-Thomson coolers. These will not have sufficient thermal inertia to freeze the gyrotrons under any scenarios related to failure of the magnet cryostat vacuum, quench or warm-up.

Thermal Measurements and Analyses

Temperature measurements of gyrotron cooling water and launcher mirrors are used on the DIII-D ECH system to provide calibrated injected rf power on a shot by shot basis and provide a diagnostic of the performance of the launchers. These measurements will be described in this section.

A total of approximately 2 MW is dissipated in the gyrotron cooling water during pulses at full power operation. This water is circulated through several circuits and the temperature rise and measured flow can be used to determine the power loading on the various gyrotron components calorimetrically and to calculate the generated rf power. The maximum water pressure is about 10 bar. Coupled with measurements of the transmission line efficiencies, these thermal measurements permit the rf power injected into the tokamak to be inferred. After passing through the gyrotron cooling circuits, the water flows to a cooling tower heat exchanger and then returns to the gyrotrons in a closed circuit loop. Unfortunately, in the DIII-D system the mixing of the water is extremely poor, therefore packets of heated water produce ghost ΔT signals with similar time dependence to the real time signal that can, after the firing of several rf pulses, interfere with the accuracy of the real time measurement. Circuits which have long cooling times, such as the dummy loads (D/L) and the matching optics units (MOU), are especially susceptible to errors from ghost heat pulses, while fast responding circuits such as the output window and cavity, have very small ΔT signals with attendant signal/noise measurement difficulties. The characteristics of the calorimetry measurements on the DIII-D systems are presented in Table 2.

The long time constant calorimetry data were analyzed by developing an analytical fitting function which takes into account the heating portion and then the cooling portion of the measurement cycle, the specific heats of the various components and coolant, heat transfer coefficients and suitable boundary conditions. The problem and the quality of the fit to the calorimetry response using this function are illustrated in Fig. 2, in which it is clear that the function provides an excellent fit to the calorimetry characteristic and results in a decrease in the scatter in the data for pulses of various lengths from ±15% to about half that value.

Table 2. Calorimetry circuit characteristics for a 1 s rf pulse at ~700 kW generated rf power. The power for each circuit is given as a percentage of the generated rf power.

Calorimetry Circuit	Flow (l/min)	Time Response (s)	Power (%)	ΔT (ºC)
Collector	1200	20	220	4.0
Cavity	80	10	2.3	1.0
Diamond window	20	5	0.2	0.3
BN window	28	400	4	0.2
MOU	12	200	5–18	1.0
Waveguide D/L	240	20	60	10
Backstop D/L	30	300	40	2.0

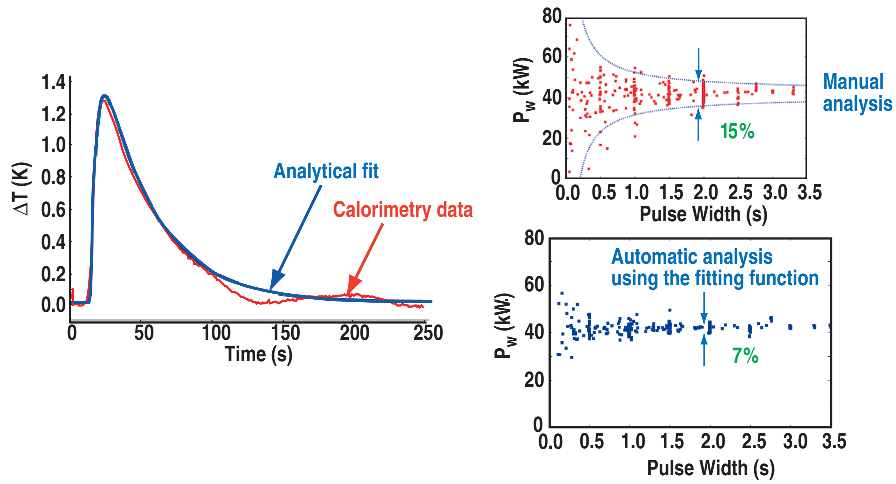

Fig. 2. On the left a typical calorimetry trace is shown, with ghost ΔT signals registered as variations from an exponential characteristic after the end of the heating pulse. The analytical fitting function is seen to provide a good representation of the data and in the summaries of a number of pulses on the right, is seen to reduce the scatter in the measurement by about a factor of 2, with statistical accuracy approaching ±5% for the longer pulses. For the gyrotron cavity, P_w, the power absorbed by the cooling water, is about 40 kW.

The data, including the time dependent flow from the vortex flow monitors, are acquired during and after an rf pulse and then are fit once the data acquisition has been completed. The data are archived, analyzed for accuracy, corrected for the measured transmission line efficiencies and then are added to the DIII-D database for that shot. Conditioning pulses in between tokamak discharges invalidate use of the MOU calorimetry for power determination, but in such cases the system disregards the MOU data. In general, calorimetry on the cavity cooling has provided the best accuracy, although this relies on a prior determination of the relationship between cavity heat load and generated rf power. The system provides a power measurement with better than ±7% accuracy on each tokamak shot. Efforts are underway to develop a suitable rf pickoff near the DIII-D machine for providing real time power measurements, but thus far the calorimetric measurement has provided the best accuracy.

Performance of the launcher mirrors in the DIII-D vacuum vessel [7] is measured using resistance temperature devices (RTDs) attached to the back surfaces of the mirrors and by relating the time dependent measurements of these temperatures to the mirror surface temperatures using experimentally verified models. Each launcher has a pair of mirrors to focus and direct the rf beam. The fixed focusing mirrors are Glidcop and are securely mounted to the launcher structure. The steering mirrors have flat copper reflecting surfaces which are bonded to a laminated structure of copper and stainless steel designed to reduce eddy current induced loads. The mirrors are radiatively cooled to the launcher shroud and have performed well. Temperature measurements from the RTDs are shown in Fig. 3 for a series of rf pulses into the DIII-D vessel during plasma operations.

During the course of a day's tokamak operations using about 600 kW injected power at 2 s pulse length, the temperatures measured by the RTDs on the back surfaces of the mirrors ratchet up to about 100ºC during the periods between rf pulses. The maximum increases are about 20ºC during the pulses. As can be seen in the upper portion of Fig. 3, which presents a comparison of the measurements for plasma operation with and without ECH, plasma radiation accounts for about half of the heating of the mirrors.

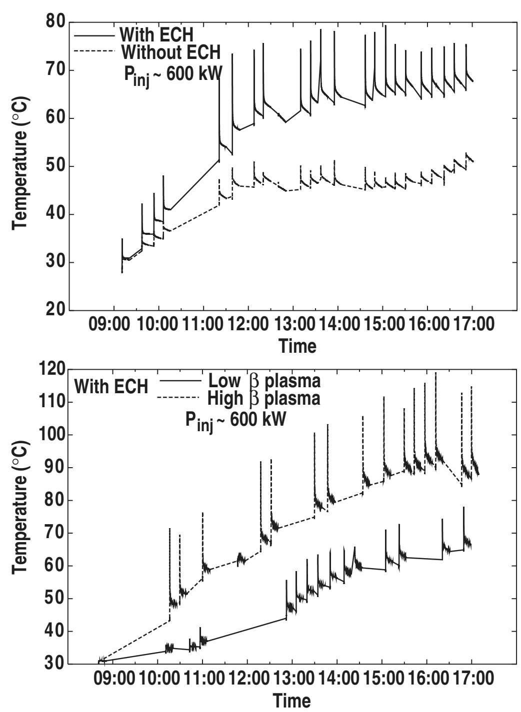

Fig. 3. The daily record of RTD measurements comparing equivalent ECH and no ECH shots at about 600 kW injected power for 2 s pulses in the upper figure and ECH shots at high and low beta in the lower figure. For the worst case, ECH at high beta, the peak RTD measurement was 120ºC late in the day. In the comparison between ECH and no ECH shots, it is seen that about half of the power loading was from plasma radiation, and half from heating by the rf reflected from the steering mirror.

The worst case is documented in the lower part of the figure, which shows the difference between high and low beta operation with rf injection. The maximum measured temperature on the back of the mirror approaches 120ºC during the pulses late in the day when the baseline temperature has increased to about 90ºC. Calibration of the thermal response of the system using heat sources or ice applied to the mirror surfaces permits the peak surface temperature on the mirror to be inferred from the RTD measurements on the back surface. Such a calculation is summarized in Fig. 4, where the model was used to calculate the peak mirror surface temperature for a 1.2 MW 5 s rf pulse with Gaussian profile and the worst case polarization with the electric vector in the plane of the reflection. The peak surface temperature for this case is calculated to be about 500ºC, which is acceptable for Glidcop. After two years of experimental operation, there was evidence of isolated tracking on the mirror surfaces, but the overall condition was excellent.

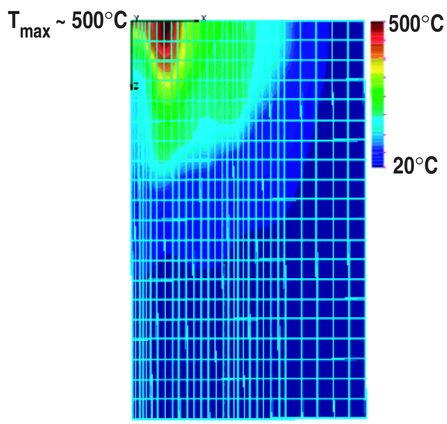

Fig. 4. Based on calibrated model calculations and the data shown in Fig. 3, the maximum mirror surface temperature for a 1.2 MW, 5 s rf pulse is predicted to be about 500ºC, which is an acceptable result for these radiatively cooled mirrors. The model invokes symmetry and shows the temperature on one quarter of the mirror surface.

Summary of Experimental Results

The DIII-D facility was active for a total of 34 experimental weeks during the 2004-2005 campaign. This was the longest period of continuous operation in the history of the facility and was done to create a time window to accommodate the major modifications discussed above. The typical schedule was two or three weeks of plasma operations followed by one or two weeks of maintenance activity. Many of the experiments made use of the ECH system and for some the system was a major element. Some of the ECH-related experiments will be described briefly.

A major line of investigation continued to be suppression of NTMs [8] using ECCD to restore a stable current density profile and eliminate islands that decrease confinement and can lead to disruptions. Attention continued to focus on the *m*/*n* = 2/1 and 3/2 modes. In Fig. 5, a shot is shown which is particularly complete in its illustration of the effects. Beta increased as the neutral beam power was increased. When β_N, a normalized indicator of beta, reached about $4 \times \ell_i$, the 2/1 NTM began to grow, decreasing β_N. The NTM was immediately suppressed by

2 MW of ECCD applied at the $q = 2$ surface, after which it was possible to restore the previously unstable β_N by ramping up the neutral beam injection (NBI) power. When one gyrotron dropped out, it was impossible to maintain the suppression and the NTM returned.

A significant development in the NTM suppression experiments was preemptive NTM suppression, which is shown in Fig. 6. The previously described shot was followed by a shot on which ECCD at the $q = 2$ surface was applied prior to the development of the 2/1 NTM. Increasing the neutral beam power then increased β_N to the threshold in the previous discharge for driving the NTM unstable, but it did not grow until after the ECCD terminated. This result is significant for large machines like ITER, because it means that the possibility exists to increase the operating window for the machine without getting perilously close to conditions for disruption.

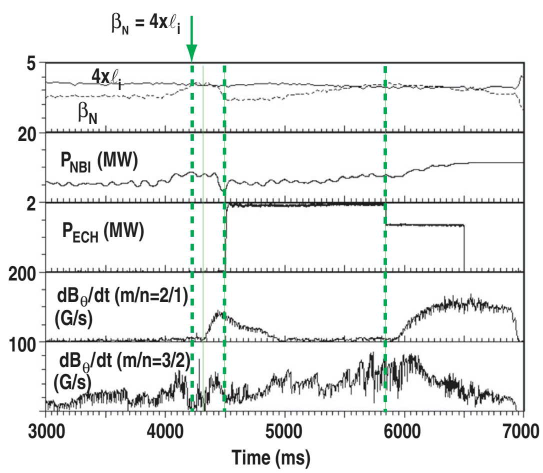

Fig. 5. Evolution of a discharge in which ECCD applied at the $q = 2$ surface suppressed the $m/n = 2/1$ NTM. The mode returned when one gyrotron dropped out and the rf power decreased.

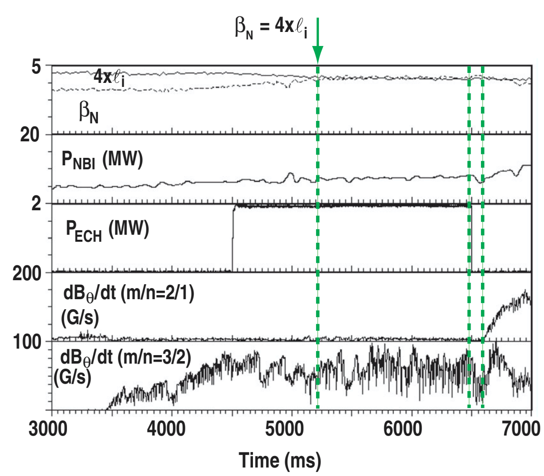

Fig. 6. Preemptive suppression of the 2/1 NTM is demonstrated. In this discharge, co-ECCD at $q = 2$ was applied prior to growth of the 2/1 NTM and the mode onset was delayed until the termination of the ECCD.

Although operation of the ITER device is still many years away, its operational scenarios are being investigated on a number of tokamaks worldwide. At DIII-D, plasma shaping studies coupled with active control of the *q(r)* profile with nearly fully noninductive current drive are leading to operational scenarios for ITER which push the ITER extrapolation to higher beta and better overall performance. Discharges in this Advanced Tokamak (AT) scenario [9], Fig. 7, have strongly inverted *q(r)* profiles, with $q_{\mathrm{min}} > 2$,which is controlled by relatively low EC driven current at large values of normalized radius, $\rho \sim 0.55$.

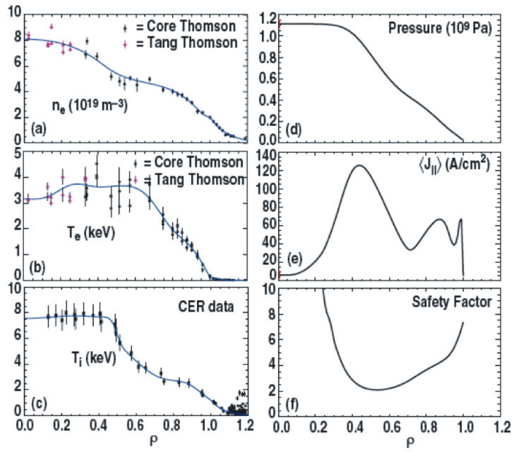

Fig. 7. The DIII-D AT scenario has a strongly inverted q profile, with hollow current density profile and $q_{\mathrm{min}} \sim 2$. The mode is not stationary on DIII-D with the non-inductive power available at present.

These discharges are formed by ramping down the toroidal field while ramping up the plasma current and have $\beta_{\mathrm{T}} \sim 6\%$, $\beta_{\mathrm{N}} \sim 6 \times \ell_{\mathrm{i}}$. Although the discharges are developed using non-stationary techniques as seen in Fig. 8, the high performance phase is quasi-stationary or over 2 s with a large noninductive current fraction and probably can be sustained with higher noninductive current drive. Very small ECCD compared with other noninductively driven currents can have a disproportionate effect on the performance of the AT discharges, as in the case where only about 5% of the total current is due to ECCD at $\rho \sim 0.55$.

It is probable that very high performance discharges can be achieved in ITER using the AT scenario, but the machine can also be successful with more modest extrapolations from present device performance when plasma parameters can be controlled, possibly using innovative techniques. In DIII-D, a specific mode called the “Hybrid Mode” [10] has been studied, which makes use of the fact that the width of the 3/2 NTM island can be controlled by ECCD and this in turn can affect $q(0)$. In suppressing the NTM, co-ECCD is applied at the q surface corresponding to the mode and the island width shrinks. When counter-ECCD is applied at the same q surface, the island grows. To some extent the central temperature is affected by the presence of the island, as can be seen in Fig. 9, where ECE T_{e} data shows that periodic changes in the electron temperature at the island location periodically affect the central temperature with a phase shift. This can be used to control the central q value, through coupling between $T_e(0)$ and the current density, at a value slightly greater than 1.0 for long periods of time as seen in Fig. 10.

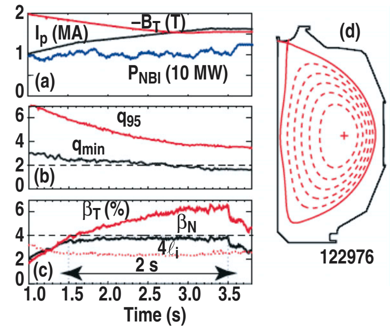

Fig. 8. At present, advanced tokamak discharges are produced on DIII-D by noninductive means, such as this example where the toroidal field was ramped down and the current ramped up to maintain a quasi-stationary configuration with high performance.

The potential for control is shown in Fig. 11, in which $q(0)$ is plotted as a function of island width for two discharges in which co- and counter-ECCD were used and a search algorithm was used to place the ECCD at the island location.

To summarize, the Hybrid Mode makes use of inductively driven main plasma current, and does not represent a radical departure from present tokamak operational scenarios. The operation is very robust and stable and $q(0)$ can be controlled in part by regulating the width of the 3/2 island with ECCD. The mode extrapolates from the DIII-D operation to one hour discharges in ITER with $Q \geq 10$.

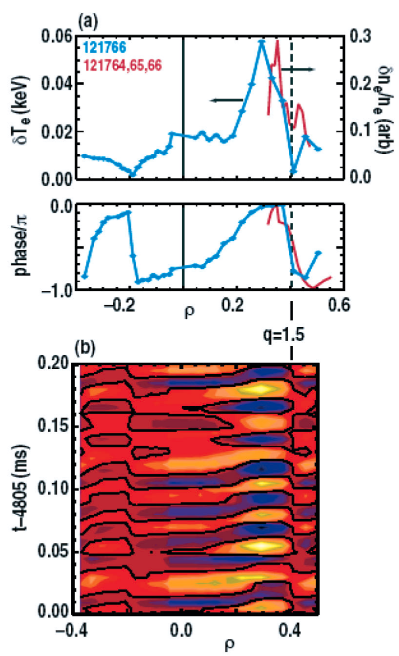

Fig. 9. The fluctuations in T_e at the 3/2 NTM island location affect the central temperature and hence the central current density.

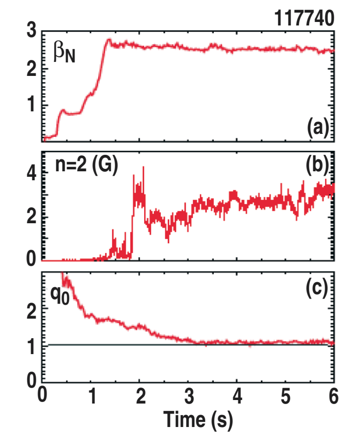

Fig. 10. By controlling the width of the NTM island with co- and counter-ECCD, $q(0)$ can be held slightly >1.0 for extended periods of time. This mode of operation is called the "Hybrid Mode".

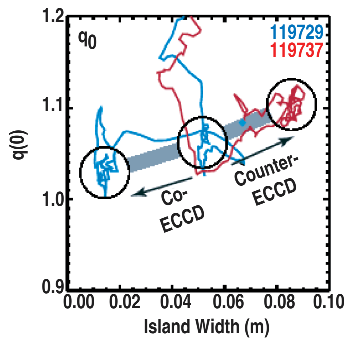

Fig. 11. Two discharges are shown in which co- and counter-ECCD was used to affect the island width for the 3/2 NTM. Co-ECCD shrinks the island and counter-ECCD increases the size, with concomitant effect on the q profile.

Summary

When the DIII-D facility resumes operations in mid 2006, the ECH complex will include six gyrotrons with 1.0 MW nominal output for 10 s pulse length. An additional gyrotron in this class will be available in reserve and one short pulse gyrotron will be available as a hot spare, able to be incorporated into the system on any of the waveguide lines. The next series of experiments will focus on applying this nearly factor of two increase in power and long pulse capability to extensions of a wide range of experiments in support of the ITER development.

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