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PROGRESS TOWARD AN ADVANCED TOKAMAK AT DIII–D

by
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ABSTRACT

Recent additions to the DIII–D tokamak have permitted the many issues critical to advanced tokamak operation to be addressed. These include an expanded divertor pumping and baffle system, improved energy handling of the plasma facing components, higher power and longer pulse electron cyclotron heating and current drive systems, an active instability control system, an upgraded plasma control system and upgraded diagnostic systems. Initial results of the plasma pumping, heating and stabilization experiments are promising.

1. INTRODUCTION

Design of conventional tokamaks is focussed around moderate values of plasma confinement, H < 2 ($H=\tau_E/\tau_{ITER89P}$), and plasma stability, $\beta_N < 2.5$ [$\beta_N = \beta/(I/aB)$] with high power rf and/or neutral beam heating proposed to achieve steady state operation. To enhance the commercial attractiveness of the tokamak relative to a conventional design, the DIII–D program is focussing on developing the scientific basis for advanced modes of tokamak operation. This advanced tokamak (AT) is envisioned as a more compact, highly shaped plasma operating at higher β_N (approaching 5), higher confinement (H approaching 3), with no inductive current drive, and low recirculating power. The high stability and confinement require the use of both current and pressure profile control, while the latter two conditions require a steady-state current drive and a high bootstrap current fraction to reduce the power requirements for the current drive system. Recent additions to the DIII–D tokamak permit many of the issues to be addressed. These new systems and upgrades include: an active instability control system, higher power and longer pulse rf heating and current drive systems, an expanded divertor pumping system to provide more effective density and impurity control in AT discharges, improved energy handling of the plasma facing components, an upgraded plasma shape control system, and upgraded diagnostics systems.

2. RECENT PROGRESS AND UPGRADES

A figure of merit often used as a shorthand to capture progress in advanced tokamak research is the product, $\beta_N H$. Significant progress has been made on DIII–D in achieving and sustaining a high value of $\beta_N H$ in discharges with an internal transport barrier and an optimized (weakly negative) central shear profile. While conventional tokamaks are typically characterized by $\beta_N H < 5$, DIII–D has sustained a value of $\beta_N H > 9$ for 2 s with approximately 50% bootstrap and a non-inductive current fraction of 75% (Fig. 1). This value is close to the design value of $\beta_N H \sim 11$ for the ARIES-RS design. The duration of 2 s is more than 16 energy confinement times and is comparable to the current relaxation time.

Further improvement in AT performance is often limited by an n=1 ideal external kink mode which, in the presence of a resistive wall, is converted to a "resistive wall mode" (RWM). This mode causes a significant loss of plasma thermal energy and often leads to a disruption. A system designed to actively control the RWM has been installed on DIII–D [1]. The system consists of a set of six external sensor coils (three pairs of saddle loops on the vessel midplane wired in an n=1 configuration) and three corresponding pairs of independently powered external saddle coils (C-coil) used as the excitation coil. The C-coil had previously been used to reduce the inherent error field of DIII–D, however, in these experiments, the C-coil pairs are driven by a set of three new,

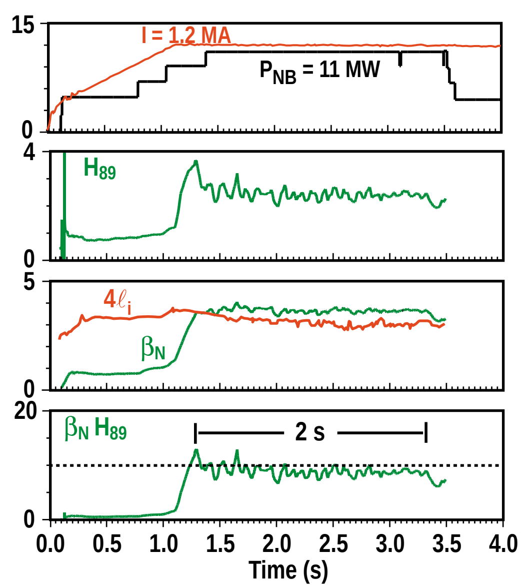

Fig. 1. $\beta_N H > 9$ sustained for 2 s with ~50% bootstrap current fraction. Neutral beam heating during current ramp produces weak negative shear in inner region.

bipolar current amplifiers with a bandwidth of 800 Hz and the coil provides both error correction and active mode stabilization. The sensor signals are processed in real time in our digital plasma control system (PCS). Using an algorithm that nulls out the radial field of the mode at the wall, proof of principle experiments have succeeded in extending the duration of high β above the no wall limit by stabilizing the RWM [2] (Fig. 2). The mode eventually goes unstable when its growth rate exceeds the bandwidth of the feedback system. We expect to install a new set of internal sensors this year and possibly expand the set of external active coils. Modeling indicates that the closer proximity of the internal sensors to the plasma, elimination of the phase shift introduced by the vessel wall, and the extended control coil set should significantly extend the stable operation above the no wall limit.

Steady-state control of the plasma profiles is key to achievement of enhanced tokamak performance. Electron cyclotron heating and current drive have evolved as essential tools for profile control. Seven 1 MW class 110 GHz gyrotrons have been commissioned [3]. Three tubes built by Communication Power Industries (CPI) have 10 s pulse capabilities due to the use of a single disc chemical vapor deposition (CVD) diamond window. The new long pulse CPI tubes are of diode design and have demonstrated a rapid commissioning with a larger window of stable operation in both current and voltage than the earlier triode designs. The gaussian output beam on these gyrotrons allows a single focussing mirror system to couple to the waveguide and launcher system and this has resulted in a lower power loss (<10%) in the coupling system than the previous two mirror system (15% loss). Two additional short pulse (2 s) tubes manufactured by GYCOM and purchased from TdeV have also recently been installed. Two other short pulse tubes, one from CPI and one from GYCOM are in

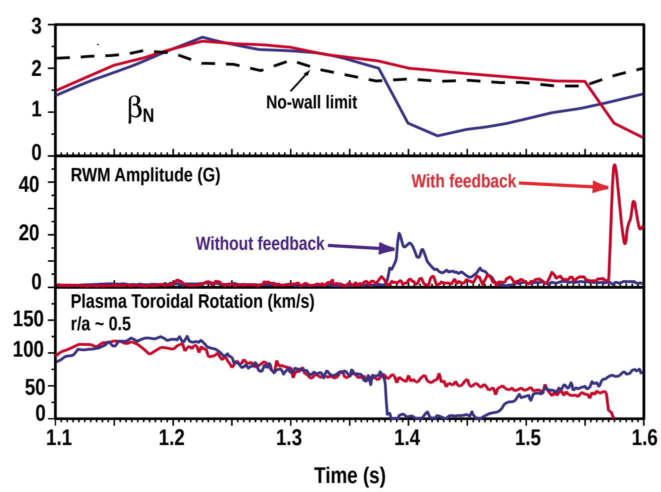

Fig. 2. Active feedback stabilization of RWM extends high β duration above the no-wall limit.

reserve status. We presently have two power supplies capable of running four gyrotrons and expect to have a third supply in Fall 2000. When combined with the six launcher assemblies installed on DIII–D, the system will have a capacity of 5.1 MW of rf generation with between 75%–80% of the power delivered to the plasma. All the launchers have poloidal sweeping capability but a new launcher designed by PPPL has both poloidal and toroidal sweeping with both co- and counter current drive capability. Although initial operation of the systems has been impacted by leaks in the CVD diamond window braze, experimental results have been very encouraging and we expect that a newer braze design using a gold braze should be more robust. Localized current drive ($\delta(r/a) < 0.1$) has been demonstrated both on and off axis, and very strong electron heating has been observed (Fig. 3). In addition, the formation of an electron internal transport barrier has been observed following the application of early ECH heating [4]. Although the mirrors in the launchers presently do not have long pulse capability, we are

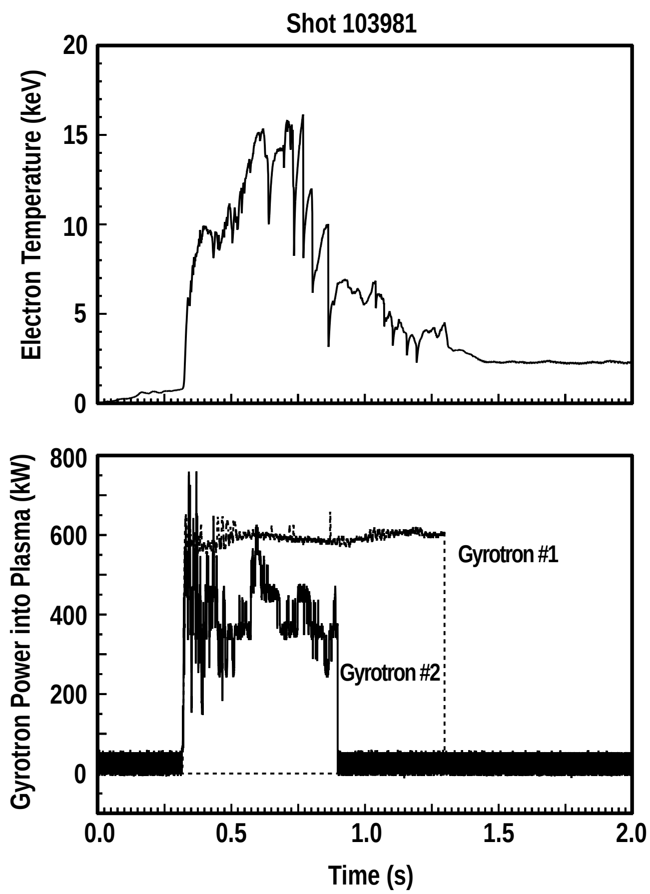

Fig. 3. Strong electron heating observed with newly installed 110 GHz gyrotrons.

presently pursuing two designs to extend the duration of all launchers to 10 s and the addition of another launcher with toroidal sweep capability next year. A fourth 10 s CPI gyrotron will be installed late in calendar year 2000.

DIII–D AT scenarios propose using ECCD for steady state operation and current profile control. Since the efficiency of ECCD is given by $\eta \sim T_e / n_e^2\ (5 + Z_{eff})$, effective density and impurity control is essential to achieving this AT operation. To achieve the low density required, a third liquid helium cryopump has been installed in DIII–D (Fig. 4). The new pump, located at the upper inner corner of the DIII–D vessel, together with the private flux baffle and pre-existing upper, outer cryopump and baffle, permits pumping of both the inner and outer divertor legs of the highly triangular AT plasmas. The new, inner pump has a pumping speed of 20,000 l/s compared to 37,000 l/s for the outer pump. The combination of the two pumps with the reduction of neutral recycling from the private flux and outer baffle have allowed densities as low as 0.3 of the Greenwald density limit to be obtained in high confinement H–mode plasmas [5], consistent with the density required for the steady state AT scenario.

Installed with the cryopump was an extensive addition to the gas puff system. Eleven new puff locations provide up to 400 Torr-l/s of nearly axisymmetric puffing from either the lower or upper outer baffle or from the private flux dome. These valves permit studies of simultaneous impurity injection during D_2 puffing and pumping experiments to study impurity and heat flux control.

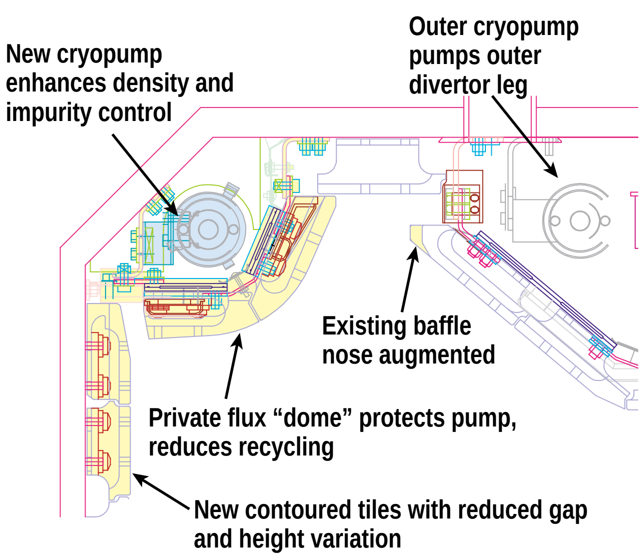

Fig. 4. A cross section view of the AT divertor.

As part of the divertor installation, the heat handling ability of the target region graphite armor tiles was significantly increased by replacement of flat tiles with tiles contoured to the cylindrical shape of the inner wall, improved edge to edge alignment of the tiles (<0.1 mm), reduction of tile spacing from 2.5 mm to 0.62 mm, and the use of high conductivity graphite in certain key locations. 2-D thermal analysis shows that the time for the tile edges to reach sublimation is increased by a factor of ten for the new tiles and this is consistent with significant reductions in measurements of tile edge heating using IR camera measurements. In addition, a comparison of similar discharges before and after the tile modifications show a reduction in the carbon content of the plasma by approximately 50%.

A critical element to the success of these experiments and to the future progress of the DIII–D AT program is the continued advancement of the plasma control system. Presently, the system provides real-time control of plasma shape, position, current, density, error fields, and RWM feedback. The longer term goal is to fully integrate these controls with current and the pressure profile control. Recent advances in the control algorithms included the successful control of the shape and position of a single-null divertor using a model-based multiple input, multiple output (MIMO) controller developed using closed loop simulation tools [6]. To provide the computing speed required for the fully integrated control and the implementation of the more advanced MIMO controller, a hardware upgrade is now in progress. The present system, which uses six CSPI i860 VME format processors will be replaced with a PCI based Alpha processor running a customized version of a Linux kernel. The new system is anticipated to provide a factor of twenty increase in performance for the plasma shape control [7].

3. SUMMARY

Significant progress has been made in obtaining enhanced performance, AT discharges in DIII–D. Much of this progress has been enabled by recent additions and upgrades to the DIII–D hardware systems. Combined advancements in both physics understanding and engineering systems are required for the successful achievement of an advanced tokamak on DIII–D.

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