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FEBRUARY 1994

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Abstract

This paper describes recent DIII-D high power heating and current drive experiments. Described are experiments with improved wall conditioning, divertor particle pumping, radiative divertor experiments, studies of plasma shape and high poloidal beta.

1. Introduction

High power heating experiments in the DIII-D tokamak are providing new physics understanding in the areas of heat and particle removal, improved confinement, increased stability limits, and current drive. These experiments are now being carried out with recently installed new capabilities which include: additional graphite coverage of the vacuum vessel walls to reduce plasma impurities, a cryogenic divertor pump for plasma density control, and a fast real-time digital control system for plasma control.

Additional graphite wall coverage with non-hydrogenic (helium) discharge cleaning has resulted in lower H-mode power threshold and achieved VH-mode without boronization. The DIII-D plasma performance has been increased to $n_D(0)\tau_E T_i(0) := 5 \times 10^{20} \text{ m}^{-3} \text{ s keV}$. The high confinement (H-mode) confinement time is typically twice low confinement (L-mode). Very high confinement (VH-mode) is typically twice that of H-mode.

A divertor cryopump provides deuterium pumping for control of the H-mode plasma density and increasing the plasma temperature. This capability to vary the plasma density has allowed us to confirm that the H-mode confinement time depends primarily on plasma current, rather than plasma density. In other divertor experiments we have investigated dispersal of the peak divertor power flux by radiation.

High beta confinement experiments with plasma shaping have been carried out to investigate plasma performance for various plasma elongation and triangularity. High triangularity shaped double-null discharges produce the highest plasma performance as measured by stability and confinement.

Current profile control experiments employing long pulse neutral beam current drive experiments show improved confinement with the absence of MHD instabilities when the central safety factor $q(0)$ relaxes to a state with $q(0) > 2$.

2. Improved DIII-D Wall Conditioning

The graphite wall tile coverage of the DIII-D vacuum vessel was increased [1] from 40% to 90%; full coverage other than the vacuum vessel ports. During the machine opening, all the previously installed graphite tiles were removed from DIII-D and grit-blasted to remove 20 to 40 microns from the surface. After grit blasting, the tiles were vacuum backed to 1000°C for 4 hours and cooled to 300°C in vacuum. Then argon was vented in during the cooldown to 80°C before exposure to air at temperatures below 80°C. A photograph of the new graphite tile installation is shown in Fig. 1.

The tiles were installed with a compressible graphite material to increase the thermal conductivity to the water-cooled Inconel vacuum vessel wall and reduce local stress concentrations. Previously a copper foam had been used which resulted in occasional bursts of copper during tokamak discharges. This source of impurities has been eliminated.

After installation, the inside graphite surfaces were wiped with alcohol to remove loose carbon dust. Before plasma discharges were initiated, we carried out 47 hours of baking above 300°C, 3.5 hours of baking and helium Taylor discharge cleaning, and 3 hours of baking and helium glow discharge conditioning. This wall conditioning was done exclusively with helium, rather than hydrogen and deuterium, because of the concern of loading the wall with hydrogen that would recycle during plasma operation. The only use of deuterium was to fuel the tokamak discharges.

The above mentioned procedures resulted in a rapid vent recovery and high plasma performance. H-mode transitions were observed on the fourteenth discharge, full toroidal field ohmic H-mode was observed for the first time in DIII-D, and

FIG. 1. Photograph of interior of DIII-D vacuum vessel. Shown on the left is the four element fast wave current drive antenna and below is the divertor ring and baffle chamber.

VH-mode discharges were obtained with an unboronized wall during the first experimental day with high triangularity double-null divertor operation. These observations are summarized in Table I. The radiated power was reduced by a factor two to three with the all-graphite wall. This is a result of the reduction in the metal impurities as shown in Fig. 2.

A central deuterium ion temperature of $T_i(0) = 21$ keV was achieved in hot ion VH-mode discharges with high fusion triple product $n_D(0) \tau_E T_i(0) = 5 \times 10^{20}$ keV · s/m³ at a high normalized beta of $\beta_N = \beta/(I/aB) = 3.5$. Here τ_E is the global energy confinement time and $n_D(0)$ is the central deuterium ion density.

TABLE I. Rapid 1993 Vent Recovery After Adding Graphite Armor

	Recovery Vent 1987 Vent	Recovery After 1992 Vent
Graphite coverage increase	9% to 40%	45% to ~100%
Baking	130 hr	47 hr
Baking and Taylor discharge cleaning	>70 hr (D ₂)	3.5 hr (He)
He glow and baking	–	3 hr
Helium glow wall conditioning before every discharge	No	Yes
Achieved VH-mode	No	Yes

FIG. 2. Metal impurity line radiation is reduced with the all-graphite wall; compare solid line data with dashed line data. Shown as a function of time is radiation from C III, Ni XVII, Ni XXVII, and Cu, XIX as well as the effective Z , Z_{eff} . The carbon content did not increase with the increased graphite wall since the oxygen content was reduced, so chemical sputtering was reduced.

3. Active Density Control with Divertor Pumping

A toroidally continuous cryogenic pump [2] has recently been installed near the outside divertor strikepoint, as shown in Fig. 3. The DIII-D advanced divertor system³ was designed and built with the primary goal of particle control in H-mode plasmas. The system consists of an integrated divertor bias electrode, a baffle, and an in-vessel cryopump. The combination of the bias ring electrode, the baffle, and the vessel wall form the cryopump plenum (Fig. 3). A toroidal gap separating the bias ring from the divertor floor defines a short duct to the pumping plenum. Biasing applications include enhancement and regulation of particle flux into the pumping plenum.

Cryopumping is a well established technology and it has been successfully used in tokamak subsystems such as neutral beam heating drift ducts and as appendage vessel pumps. However, the DIII-D divertor cryopump is the first application of this technique inside a tokamak plasma chamber. Because of the adverse environment of the DIII-D plasma chamber, including disruptions, heat load, and baking cycles to 450°C, the design of the cryopump posed a challenging engineering problem.

Details of the system design, construction and testing can be found in earlier publications cited in Ref. 2. The pumping speed of the installed system was measured during an applied pulsed heat load of 300 W in the absence of a plasma. At an ambient pressure of 5 mTorr, the pumping speed is approximately 40,000 ℓ/s .

The efficacy of divertor pumping was tested by observing the rate of density decay after moving the divertor strike point from an initial position, several scrape-off layer (SOL) widths away from the opening to the pumping plenum, to its optimum pumping position. In general, divertor pumping is found to be more effective in L-mode than

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CRYOPUMP UNDER DIVERTOR BAFFLE

FIG. 3. A cross sectional view of the DIII-D advanced divertor configuration, showing the bias ring and the cryopump. The gap between the bias ring and the vessel floor armor defines the opening to the pumping plenum.

in H-mode because of the comparatively broader and denser SOL of L-mode plasmas. Typically in L-mode plasmas the density decay time constant is $\lesssim 0.1$ s, and densities as low as $3 \times 10^{18} \text{ m}^{-3}$ have been obtained. In contrast, in H-mode plasmas with edge localized modes (ELMs), the initial density decay time constant is typically 0.5 s.

We used a combination of gas puffing and divertor pumping to control the plasma density. The pumping rate, as shown in Fig. 4, was controlled by magnetically adjusting the position of the divertor strike point relative to the pumping plenum aperture. To assure stability, the feedback logic is programmed such that, normally, simultaneous gas puffing and divertor pumping are not allowed. With this feedback system during H-mode shots, the pump exhaust rate is typically a few tens of Torrℓ/s; when the divertor strike point is placed at its optimum pumping position with the density feedback active, exhaust rates as high as 200 Torrℓ/s are observed.

Using the above procedure, we were able to control the density and current of H-mode plasmas independently. The H-mode plasma density was varied at constant I_p over a range of a factor of two, and conversely, I_p was varied at constant density over a range of a factor of two. Preliminary analysis suggests that H-mode confinement is only a weak function of n_e and increases almost linearly with I_p .

Divertor pumping modifies the core plasma density, temperature, and current profiles. Typically in ELMing H-mode plasmas, the profiles evolve for the first two seconds after the start of pumping before reaching a quasi steady state. As the plasma density decreases the electron and ion temperatures increase such that the plasma energy confinement time remains roughly unchanged. However, in most cases the increase in the central ion temperature is much higher than that of the electron temperature. This difference is explained by the lower energy exchange rate between the plasma ions and electrons at lower density and the fact that, in the parameter range for these plasmas, most of the neutral beam heating power is delivered to the plasma ions.

Figure 5 shows a comparison of the electron density and temperature profiles of pumped and unpumped shots measured at $t = 4.1$ s, 2.5 s after the start of pumping.

FIG. 4. Magnetic control of pumping. The divertor pumping rate is controlled by adjusting the position of the divertor strike point relative to the pumping plenum aperture.

FIG. 5. Electron density (n_e) and temperature (T_e) profiles of two otherwise similar H-mode shots with different densities controlled by divertor pumping.

Divertor pumping reduced the electron density gradient and increased the electron temperature gradient just inside the separatrix. Reduced recycling during divertor pumping can explain the change in the density gradient in this region, although a concurrent change in particle transport can not be ruled out. In the core plasma, with $\rho \leq 0.6$, the density gradient is nearly a factor of two higher in the higher density plasma. Since in this region the recycling source is insignificant while the neutral beam particle deposition profile is much broader at the higher density, it follows that particle transport timescale in this region increases nonlinearly with density. We should note that a similar peaking of the density profile was not observed in low current plasmas.

4. Radiative Divertor Experiments

A major effort is underway to reduce the power density striking the divertor. Two distinct approaches [4] are being investigated at DIII-D. First, the enhanced recycling divertor where the radiation is predominantly from the divertor/X-point region during deuterium injection. Second, the impurity radiative mantle, where the radiation is predominantly from inside the separatrix during neon injection. While both approaches are effective at reducing the target plate power density the consequences of the core plasma differ, as indicated in Table II. Figure 6 shows an example of divertor heat flux reduction with deuterium is injected and the divertor power density is reduced. The plasma energy confinement time is not decreased during the remaining 1.5 seconds of the discharge. Bolometric tomography shows that the strong increase in radiated power is in a high plasma density region near the X-point. Experiments are being carried out to maintain low peak heat flux by feedback control of gas puffing. Future radiative divertor experiments will be carried out with closer baffling installed around the divertor legs.

TABLE II. Comparison Summary of Two Radiative Divertor Operating Modes

Approach	Enhanced Recycling	Radiating Mantle
Gas	D ₂	Neon
Significant reduction in Q_{div}	Yes	Yes
Location of radiation increase	Mostly in the divertor/X-point region	Mostly inside the separatrix/more even distribution of plasma power
Dilution of fuel ions	No	Yes (~20%)
Density of main plasma	~10%–15% increase	Constant
H-mode degradation in energy confinement	<5%	10%–15%
Effect on edge T_e and T_i -edge profile	Modest	Significant
ELM-control	No	Yes
Low- q operation	Yes	Yes

FIG. 6. Divertor heat flux reduction in deuterium puffing experiments is shown as a function of time.

5. Studies of Plasma Shape

An experiment [5] was carried out this year to explore the effect of plasma shape on confinement. The basis of the experiment lies in a previous observation that there is a systematic difference between single-null and double-null diverted plasmas in plasma performance, as measured by the product $\beta \times \tau$ [6]. In that data we could not separate the effects of plasma shape and number of nulls. The goals of this experiment were to further explore the underlying cause of this difference and to explore the performance characteristics of particular plasma shapes of interest for the DIII-D radiative divertor. Of major importance is finding a configuration which is consistent with the goals of the DIII-D Advanced Tokamak and Radiative Divertor Programs.

The principal result of the experiment is that plasma triangularity, δ , is critical to plasma performance, while plasma elongation, κ , plays a lesser role. Triangularity is the radial location of the peak plasma height expressed as a fraction of plasma minor radius. Whether the plasma is single- or double-null is not of importance, aside from the attendant limitations on δ . The shapes studied are shown in Fig. 7. The shape variations were done at 2 T magnetic field at three values of plasma current, corresponding to safety factors of $q_{95} = 3.3, 4.2$, and 5.1. The density varied in the usual fashion. The heating power was varied systematically over the range of 6 to 15 MW for all shapes at all currents. Typically 2/3 of the beam power goes to the ions. The difference in energy content of the low and high triangularity lies principally in the temperature profiles for normalized radius less than 0.8. The high triangularity plasmas exhibit a temperature gradient all the way to the magnetic axis, whereas the low triangularity plasmas have a large factor of two difference in axial values of ion and electron temperature. It is observed that q_0 rises above unity when the triangularity

is high, but this is not the case when the triangularity is low. A summary of key results is given in Table III. These experiments have influenced the selection of the high-triangularity double-null shape for the Tokamak Physics Experiment (TPX) [7].

FIG. 7. Cross sections for plasma discharges used for plasma shape studies. All plasmas have the same safety factor, $q = 4$.

TABLE III. Summary of Parameters Achieved in Plasma Shape Studies

The plasma triangularity is δ and the elongation is κ . The confinement quality is referenced to L-mode confinement, $H = \tau_E/\tau_L$.

$\kappa \approx 2.1$		$\kappa \approx 1.7$	
$\delta \approx 0.9$	$\beta \gtrsim 5\%, \beta_N \gtrsim 4$	$\beta \gtrsim 5\%, \beta_N \gtrsim 4$	
	$T_e(0) \approx 7.5 \text{ keV}, T_i(0) \gtrsim 20 \text{ keV}$	$T_e(0) \approx 7.5 \text{ keV}, T_i(0) \gtrsim 15 \text{ keV}$	
	$\tau_E^{th} \gtrsim 0.4 \text{ s}, H \gtrsim 3.3$	$\tau_E^{th} \gtrsim 0.4 \text{ s}, H \gtrsim 3$	
	$n_D(0)T_i(0)\tau_E^{th} \approx 4 \times 10^{20} \text{ keV} \cdot \text{s} \cdot \text{m}^{-3}$	$n_D(0)T_i(0)\tau_E^{th} \approx 3 \times 10^{20} \text{ keV} \cdot \text{s} \cdot \text{m}^{-3}$	
	$\beta\tau^{th} \approx 1.7\% \cdot \text{s}$	$\beta\tau^{th} \approx 1.3\% \cdot \text{s}$	
	Typical ELM-free period 0.65 s	Typical ELM-free period 0.22 s	
$\delta \approx 0.3$	$\beta \lesssim 3.7\%, \beta_n \lesssim 3.2$	$\beta \lesssim 2.8\%, \beta_n \lesssim 3.5$	
	$T_e(0) \lesssim 5 \text{ keV}$	$T_e(0) \lesssim 5 \text{ keV}$	
	$\tau_E^{th} \lesssim 0.3 \text{ s}, H \lesssim 2.2$	$\tau_E^{th} \lesssim 0.3 \text{ s}, H \lesssim 2.5$	
	$\beta\tau^{th} \lesssim 0.7\% \cdot \text{s}^*$	$\beta\tau^{th} \lesssim 0.5\% \cdot \text{s}^*$	
	Typical ELM-free period 0.10 s	Typical ELM-free period 0.15 s	

*Maximum occurs after ELMs begin.

6. High β_p Plasmas with Improved Stability and Confinement

Experiments [8] to explore the long-time evolution of noninductive, high poloidal beta (β_p) plasmas in the DIII-D tokamak, have identified a new, quiescent, high performance regime. The high β_p , high bootstrap fraction plasmas have a high triangularity, double-null separatrix shape. The toroidal magnetic field is 1.9–2.1 T, and the plasma current is 400–800 kA. Heating is with 3–10 MW of neutral beam injection (NBI). The lower limit of 400 kA is set by the increasing difficulty of confining fast ions as the current is reduced. At 400 kA there is a large (up to 50%) prompt loss of fast ions from the plasma.

The temporal evolution of a typical low current discharge proceeds through several well defined stages. Initially, we allow up to one second for the establishment and relaxation of the ohmic target plasma. With initiation of NBI, a transient very high β_p configuration, with a large fast ion population (30%–40%) and strong MHD activity is established. On a resistive time scale, the current profile broadens from the ohmic shape ($q_0 \approx 1$; $\ell_i \approx 1.8$) toward a profile characteristic of a combination of neutral beam current drive and bootstrap current ($q_0 \geq 2$; $\ell_i \approx 1.2$). During this evolution the MHD activity initially disappears when q_0 exceeds 2, but reappears when q_0 decays down to the neighborhood of 2. Following the disappearance of the $q = 2$ surface and the MHD activity, the central electron density rises, doubling in about one second. While the central density rises, the density profile in the outer half of the plasma changes very little from the characteristic ELMing H-mode shape seen earlier in the discharge. The improved central confinement appears similar to the reversed central magnetic shear seen in tokamak experiments[9], but in this case without external intervention

or transient excitation. At high current, a similar, but slower relaxation is seen. This regime is of particular interest for the development of steady-state tokamak operating scenarios, for TPX and following reactors.

The most interesting behavior occurs for the 400 kA discharges. At higher current similar behavior is seen, but on a slower time scale because of the higher electron temperature; consequently there is not enough time to observe the behavior after the transition to $q_0 > 2$. The time history of a 400 kA DIII-D discharge is shown in Fig. 8. The equilibrium at 4.36 s because of the high β_p (≈ 3.6) and peaked pressure ($p_0/\langle p \rangle \approx 8$), there is a large outward shift of the magnetic axis. There is a region of reversal in the local magnetic shear at the outer midplane.

The strongest evidence for the existence of the improved core confinement regime is the peaking of the density as indicated by the profile comparison (3.36 s and 4.36 s) shown in Fig. 9. The region of improved confinement corresponds to $\rho \leq 0.4$. There is also an increase in ion and electron temperature. The profiles outside this radius are characteristic of an ELMing H-mode, and do not change significantly during the discharge. The toroidal rotation profile is also peaked; the toroidal velocity at the axis is approximately 1.4×10^5 m/s during the first half of the discharge, rising to 2.2×10^5 m/s by 2.95 s and remaining constant thereafter (corresponding to $\Omega_\phi R/v_{ti} \approx 0.38$).

IMPROVED CONFINEMENT REGIME HIGH β_p :

Two distinct phases:

- ❖ resistive relaxation of current profile

at 1.2 sec: $q(0) \sim 1$ & $\ell_i \approx 1.8$

at 3 sec: $q(0) \sim 2$ & $\ell_i \approx 1.3$

- ❖ improved core confinement

$q(0) \rightarrow$ above 2.0

MHD fluctuations disappear

core confinement improves

$n_e(0) \times 2$

period with no MHD

fluctuations

& no first stability limit

FIG. 8. Time histories of q_0 , ℓ_i , Mirnov fluctuation amplitude, central n_e and T_e (from Thomson scattering), and β_p . The vertical line marks the transition between the resistive relaxation phase and the improved core confinement phase. NBI starts at $t = 1.0$ s.

FIG. 9. Electron density, electron temperature, and ion temperature profiles at 3.36 s and 4.36 s. Confinement improvement is inside $\rho \approx 0.4-0.6$.

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