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P.I. PETERSEN
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APRIL 2002

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Long Pulse Advanced Tokamak Discharges in the DIII–D Tokamak

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Abstract. One of the main goals for the DIII–D research program is to establish an advanced tokamak plasma with high bootstrap current fraction that can be sustained in-principle steady-state. Substantial progress has been made in several areas during the last year. The resistive wall mode stabilization has been done with spinning plasmas in which the plasma pressure has been extended well above the no-wall beta limit. The 3/2 neoclassical tearing mode has been stabilized by the injection of ECH into the magnetic islands, which drives current to substitute the missing bootstrap current. In these experiments either the plasma was moved or the toroidal field was changed to overlap the ECCD resonance with the location of the NTMs. Effective disruption mitigation has been obtained by massive noble gas injection into shots where disruptions were deliberately triggered. The massive gas puff causes a fast and clean current quench with essentially all the plasma energy radiated fairly uniformly to the vessel walls. The run-away electrons that are normally seen accompanying disruptions are suppressed by the large density of electrons still bound on the impurity nuclei. Major elements required to establish integrated, long-pulse, advanced tokamak operations have been achieved in DIII–D: β_T = 4.2%, β_P = 2, f_{BS} = 65%, and $\beta_N H_{89}$ =10 for 600 ms (~ $4\tau_E$). The next challenge is to integrate the different elements, which will be the goal for the next five years when additional control will be available. Twelve resistive wall mode coils are scheduled to be installed in DIII–D during the summer of 2003. The future plans include upgrading the tokamak pulse length capability and increasing the ECH power, to control the current profile evolution.

I. INTRODUCTION

The goal of the worldwide fusion research is to build an energy-producing reactor that is environmentally attractive and economically competitive with other energy technologies. To meet this goal a fusion reactor will have to run steady state and have a low fraction of re-circulating power to run auxiliary equipment for heating the plasma, drive plasma current and control the plasma profiles.

The tokamak is currently the device that is closest to meeting these goals. Fusion reactor studies [1] have incorporated advanced tokamak [2] features in order to obtain high confinement τ_E for ignition margin and compact size; high plasma pressure or β, $\beta_T = 2\mu_0 \langle P \rangle / B_T^2$ for high power density; and high bootstrap fraction f_{BS} for low re-circulating power and steady state operation.

The DIII-D tokamak is a mid-size tokamak, which is operating at reactor relevant temperature and has as one of its main goals to study advanced tokamak scenarios. Four advanced scenarios are studied in the DIII-D research program: radiative improved mode, high ℓ_i mode, the negative central shear modes (NCS), and the quiescent double barrier mode (QDB). The last two mode are currently used in long pulse experiments and will be described in Sections II and III. The high ℓ_i modes will be studied when more ECH and fast wave power becomes available.

Two MHD instabilities that limit the duration of advanced tokamak plasma in DIII-D have been identified to be resistive wall modes [3] and neoclassical tearing modes [4]. These two modes will be described in Sections IV and V together with the mechanisms for stabilizing them. The resistive wall modes can be stabilized by either plasma rotation or feedback control. Since plasma rotation is expected to decrease with the size of the device, feedback stabilization would probably be required for reactors. Neoclassical tearing modes NTMs can be stabilized by electron cyclotron current drive, which replaces the bootstrap current missing in the NTM islands.

When the plasma in a tokamak is pushed to its MHD limit it becomes unstable and might disrupt. In disruptions the energy stored in the plasma will be lost in 10–100 μs and the plasma current will disappear in a few milliseconds setting up a large induced current in the vessel wall and plasma facing surfaces. The resulting force can be large enough to severely damage the vacuum vessel or plasma facing surfaces if they are not designed for the disruption forces. In addition to plasma instabilities, flakes falling into the plasma can trigger disruptions. One technique for mitigating plasma disruptions is to inject a massive gas puff into the vessel at the time of a disruption. This technique and the results from DIII–D will be discussed in Section VI.

II. NEGATIVE CENTRAL SHEAR MODES

The negative central shear discharge is obtained by applying auxiliary heating during the plasma current ramp-up phase. The early heating slows the current diffusion toward the center and drives the toroidal electric field on axis to near zero. The resulting current density profile during the current ramp is peaked off axis. The bootstrap current, with a maximum off-axis, can contribute to and help extend the duration of the hollow current profile. Eventually an external drive mechanism such as ECCD is needed to sustain the hollow current profile. A negative central shear discharge with rotational stabilization of the resistive wall mode has produced a high performance AT plasma with a value of $\beta_N H > 12$ for 5 τ_E with approximately 65% bootstrap fraction (Fig. 1).

III. QUIESCENT DOUBLE BARRIER MODES

The QDB mode [5] has both internal and edge transport barriers (Fig 2). It is obtained with counter-injection of neutral beams, which gives a broader internal transport barrier and an edge transport barrier. The combined barriers raise the plasma temperature everywhere, increasing the fusion reactivity. Cryopumping is used to reduce the density. A key feature of the QDB is the absence of edge-localized mode (ELM) and thus no pulsed heat load to the divertor. The mode is often associated with an edge harmonic oscillation, which shows up in the magnetic, electron density and temperature diagnostics. The absence of ELMs in the QDB is advantageous for obtaining internal transport barriers and eliminating large pulsive heat flux to the first wall. ELMs will normally degrade internal transport barriers and cause large short bursts of significant energy to the divertor and the first wall. The best QDB plasmas in DIII–D have achieved $\beta_N H = 7$ for up to 5 τ_E. In other QDB discharges it has been demonstrated that the internal transport barrier can be produced and maintained inside the edge H–mode barrier for long time (> 3.5 s or 25 τ_E). The duration in the present experiments is limited by the choice of plasma current flattop and the choice of neutral beam pulse length.

IV. RESISTIVE WALL MODES

The resistive wall modes (RWM) seen in DIII–D originate from an n=1 ideal external kink mode, which in the presence of a resistive wall, is converted to a slowly growing RWM. The RWM is driven by the phase difference between the plasma surface perturbation and the dissipation of the inductively coupled currents in the vessel wall. The characteristic time for the growth of the mode is the skin time of the vessel wall, which in DIII–D is a few milliseconds. This is enough for a feedback system to interact with the mode and stabilize it. Without any feedback the beta value that can be obtained is called the no-wall beta limit. With an ideal wall (superconducting) calculations with the computer code GATO show that the beta value that can be obtained is typically a factor of two higher (Fig. 3). Since the fusion output power is proportional to β^2, this would mean a factor of four in output power for a reactor with a beta

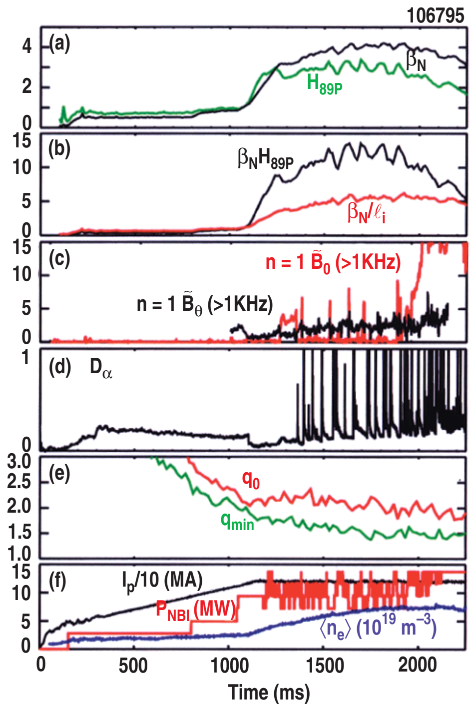

Fig. 1. A fully wall stabilized discharge, with an internal transport barrier and weekly negative central shear has been achieved with $\beta_N H > 12$ for 5 τ_E. (a) β_N~4 and H_{89P}>3; (b) β_N H_{89P}~12 and $\beta_N \ell_i$ ~ 6 about 50% above the no-wall limit; (c) rotation stabilizes the low frequency RWM, but a 2/1 tearing mode grows at ~1900 ms; (d) D_α showing ELMy H–mode during high performance phase; (e) minimum safety factor, q_{min}, is maintained above 1.5 throughout the high performance phase with the central q near 2; (f) plasma current I_P, injection neutral beam power, P_{NBI}, and line averaged density, $\langle n_e \rangle$.

limit close to the ideal limit. Figure 3 also shows that with a broader profile higher normalized beta values can be obtained than with more peaked profiles.

There are two ways to wall stabilize the RWM, either with plasma rotation obtained with high power neutral beams or through active feedback control. The plasma rotation significantly reduces the magnetic field errors that cause a magnetic drag on the plasma and reduce the plasma rotation. Experiments in DIII–D have shown a plasma above the no-wall limit without rotation magnifies the inherent magnetic field errors and thereby reduces the plasma rotation further. When the plasma rotation falls below a critical limit, the RWM becomes unstable. The best shot with stabilization of the RWM is shown in Fig. 1, where both rotation and feedback were used to stabilize the mode. The plasma pressure is stably maintained up to the ideal wall limit with stabilization of the RWM (see Fig. 4).

Initially a set of external sensor coils were used together with the magnetic field error correction coil was used to detect the RWM and through a feedback system suppress it by superimposing the mode stabilization current on top of the error field correction current. The field picked up by the external sensors was of the order of a few gauss. However, calculations done with the VALEN3D codes showed that sensors coils installed inside the vessel would significantly increase the obtainable beta (Fig 5). Calculations have also shown that an additional twelve coils installed inside the

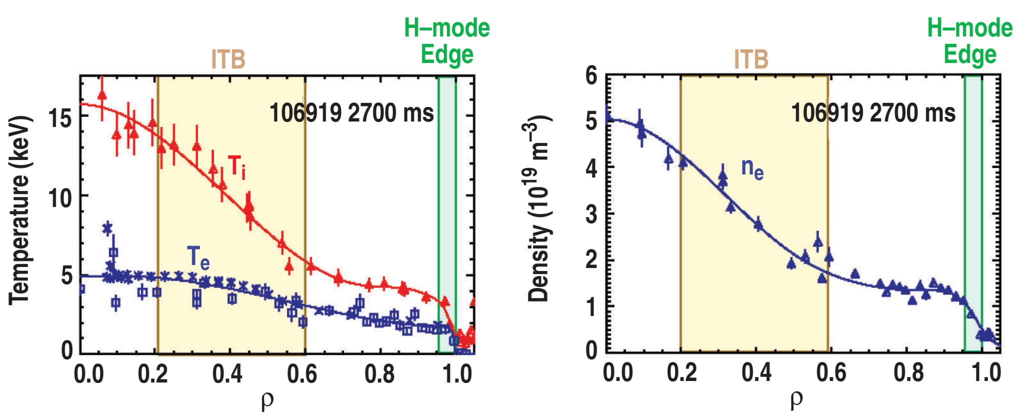

Fig. 2. Clear double barriers are seen in the ion temperature and the electron density, but not in the electron temperature in this shot.

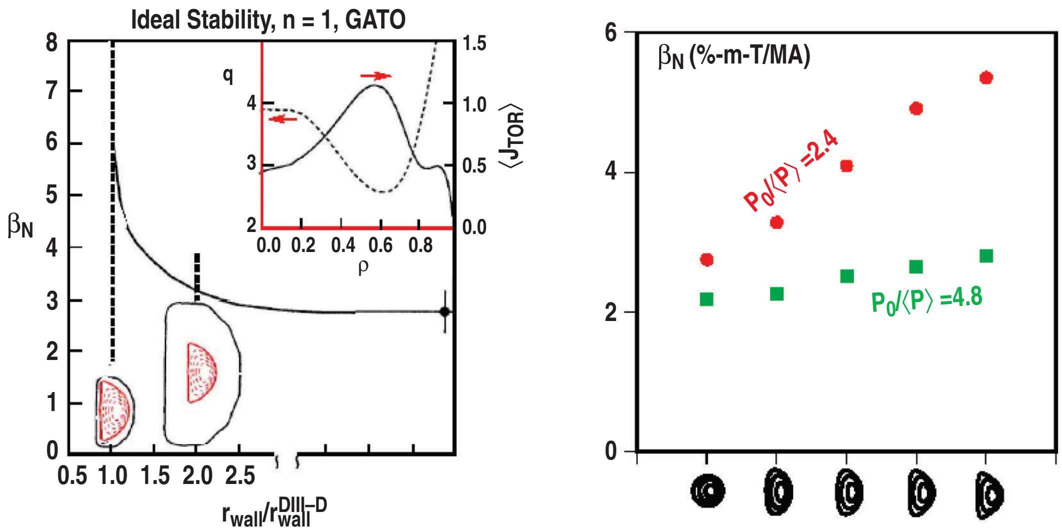

Fig. 3. Calculation with the code GATO shows that β_N = 6 can be obtained with tight fitting wall where as the limit is much lower with the walls far from the plasma. Higher β_N can be obtained with higher triangularity and broader pressure profiles.

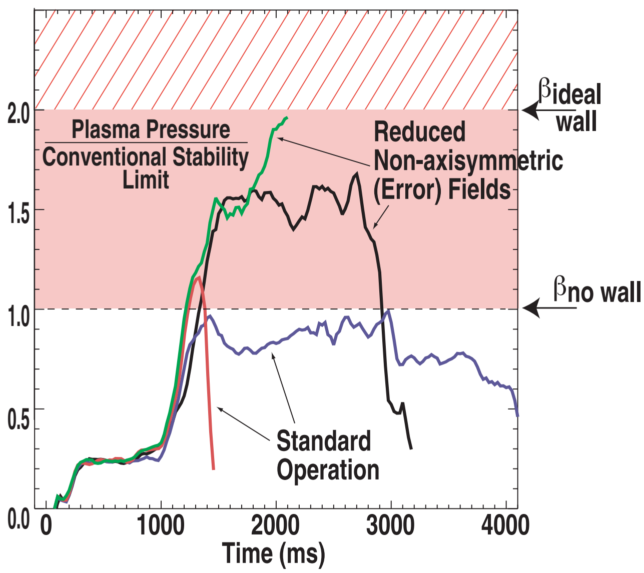

Fig. 4. Plasma presure is stably maintained up to the ideal wall limit, which is a factor two above the conventional pressure limit, with stabilization of the resistive wall mode. The nonaxisymmetric error fields are reduced by continued plasma rotation.

DIII–D vessel, with six installed above and six below the midplane will allow discharges with beta very close to the ideal limit. Two prototype coils have been installed inside the DIII–D vessel for engineering testing and the full set of coils will be installed during the summer/fall vent of 2002. These coils have a picture frame shape (0.5m × 2m), each covering about 60 deg in the toroidal direction. Each coil consists of a single turn water-cooled copper conductor, which is housed inside a stainless steel vacuum shield. The copper conductor and shield are isolated with a high temperature polyimid, Vespel and Kapton® sheets and have been tested to 4 kV. The coil leads are coaxial to minimize any error field. The coils are designed for 7 kA, which produces ~ 20% higher field than the external four-turn coils with 20 kA-turns. The coils have lower inductance and have significant higher bandwidth (dI/dt ~ 5-10 higher than the external coils with the same power supply). At 1 kHz the field produced at the plasma edge will be ~3 times higher for the internal coils compared to the external coils due to the reduced shielding effect of the wall.

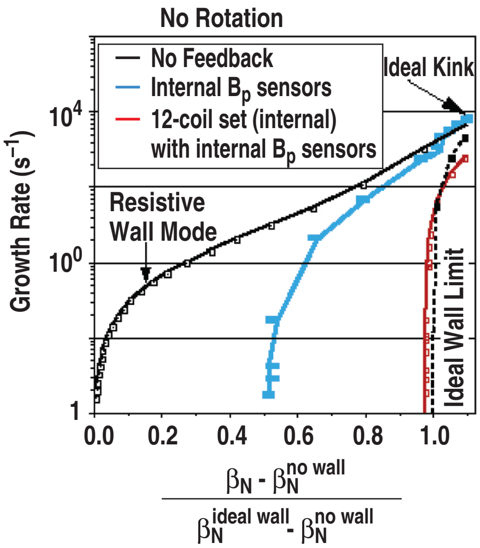

Fig. 5. With internal sensor the obtainable β_N is halfway between the no-wall limit and ideal wall limit. With twelve internal control coils the obtainable value is very close to the ideal wall limit.

V. NEOCLASSICAL TEARING MODES

Neoclassical tearing modes (NTM) are the topological rearrangement of field lines through reconnection to form islands [6]. A method to stabilize the NTM once the mode is formed is by local current drive with ECCD, where the ECCD replaces the missing bootstrap current in the O-point of the island. This has been demonstrated in DIII–D, where a saturated m=3, n=2 NTM was fully suppressed using 2.3 MW of ECCD for 1 s [8]. In initial experiments the plasma was moved on a shot-to-shot basis, and the rf power was directed off-axis to be coincident with the q=3/2 surface. In subsequent experiments the digital plasma control system was programmed to move the plasma rigidly in steps of 1 cm with dwell time at each position of 50–100 ms during a shot until the ECCD coincides with the island [7–9]. A second method that has been used is to vary the toroidal field in 0.01 T steps to move the resonant position. With new antennas built by Princeton Plasma Physics Laboratory (PPPL) and installed on the DIII–D tokamak it is possible to move the direction of the ECCD in the poloidal and toroidal direction, which would permit suppression without moving the plasma or changing the toroidal field. In experiments successful suppression of a 3/2 NTM was obtained using 2.3 MW of ECCD and the β_N was increased by 50% from 2 to 3 without reappearance of the mode. Attempts to stabilize the 2/1 NTM using 2.5 MW of ECCD resulted only in partial suppression of the mode. When more power becomes available this year the experiment will be repeated.

VI. PLASMA DISRUPTION MITIGATION

When a plasma is operated beyond its ideal MHD stability limit it will disrupt [10,11]. The growth rate of ideal MHD kink mode is of the order of 10-100 μs. In present experiments the growth of the kink mode leads to a rapid loss of the plasma thermal energy with a typical timescale of 100 μs and a subsequent rapid loss of the plasma current on a time scale of 1–10 ms. With finite conductivity walls or resistivity plasmas, the actual observed modes such as resisitive wall modes or neoclassical tearing modes grow on a slower time scale. The growth time for the non-ideal MHD modes are long enough that feedback stabilization of the modes can be employed as described above and if the feedback stabilization is not successful, there is ample time to terminate the discharge before the plasma disrupts. Thus disruptions due to MHD instabilities should be avoidable by operating some distance from the ideal MHD stability limits, or mitigated if they do occur.

An efficient reactor will have to operate close to the MHD limit in order to maximize the power output, and imperfections in the control system can therefore cause the plasma to be taken beyond the stability limit. In addition certain equipment malfunctions, flakes of material falling into the plasma from the walls and accidental air-ingress events can cause plasma disruption.

Thus a reactor will have to be able to handle the electromagnetic force loads from a worst-case disruption.

The effect of plasma disruptions can be divided into electromagnetic loads, thermal loads and run-away electrons. The electromagnetic forces arise from the toroidal currents induced in the conducting structure and are large (equivalent to about 10 atm pressure in a reactor tokamak), but manageable through proper design of the reactor components. Disruptions include vertical and radial motion of the plasma, which results in generation of halo-currents, when the plasma current completes the poloidal circuit by flowing through the vacuum vessel and plasma facing components. When a plasma disrupts the energy stored in the plasma is lost very rapidly causing a huge heat load to the plasma facing surface. These heat loads are non-uniform with most of the energy going into the divertor eroding its surfaces. Present estimates that include self-consistent consideration of a plasma-shielding layer suggest that the worst-case erosion magnitude is 3-30 μm per disruption for carbon material. However, for carbon, the surface erosion is estimated to be determined by the normal plasma operation, whereas for tungsten the normal plasma operation erosion is negligible. Run-away electrons current (~10 MA) is often created after the thermal quench of a disruption. The electrons have high energies (~10 MeV) and can cause localized damage and component failures (coolant leaks) if repeated runaway strikes should hit in the same area.

Disruption mitigation experiments [11] have been performed on DIII–D with massive gas puffs (see Fig. 6). In the first series of experiments the disruption was deliberately triggered at a preset time. A fast-acting valve separating a high-pressure (~ 70 bar) reservoir is commanded to open at the time that the disruption is wanted. The gas triggers and mitigates the disruption. 3×10^{22} particles are injected in 7 ms. The penetration of the gas into the plasma center is consistent with simple time-of-flight calculations using the sonic speed of the gasses used (~ 250 m/s for Ar). The pressure of the gas jet ~ 15 kPa is higher than the pressure of the plasma electrons ($\langle P_e \rangle$~ 6 kPa). No difference in the penetration ability was observed for D2, He, and neon, which rules out radiation as playing a role in the penetration. The massive gas puff rapidly radiates the stored energy and produces a low effective charge state (Z_{eff}~1). This low charge state, which is confirmed by XUV spectroscopy, together with the extremely high density of high Z neutral impurity gas inhibits the generation of run-away electrons. This is contrary to disruption mitigation using pellets, which has a much lower density of impurities, where run-away electrons are not completely suppressed. The high pressure gas jet technique scales favorably for reactors. Non-mitigated disruptions caused by ideal mode leave a sufficiently "dirty" wall in DIII–D that adversely affects breakdown and current evolution of the following discharge. This is not the case, when massive gas puff is used to mitigate the disruption.

VII. UPGRADES TO DIII–D

Several upgrades are planned for the DIII–D tokamak over the next five years to extend the advanced tokamak high performance shots. A set of twelve internal resistive wall mode coils are currently being manufactured and will be installed inside the DIII–D vessel during the summer and fall of 2002. The ECH system is scheduled to be upgraded with two additional 1 MW gyrotrons to a total of eight IMW class gyrotrons. Toward the end of the five year plan the three short pulse gyrotrons are scheduled to be changed out with two 1.5 MW long pulse gyrotrons. The pulse length of the tokamak is planned to be extended from 5 to 10 s at full parameters, which is a few skin times for the vacuum vessel. This requires upgrade of the toroidal coil belt bus and diodes, and field shaping coil cables.

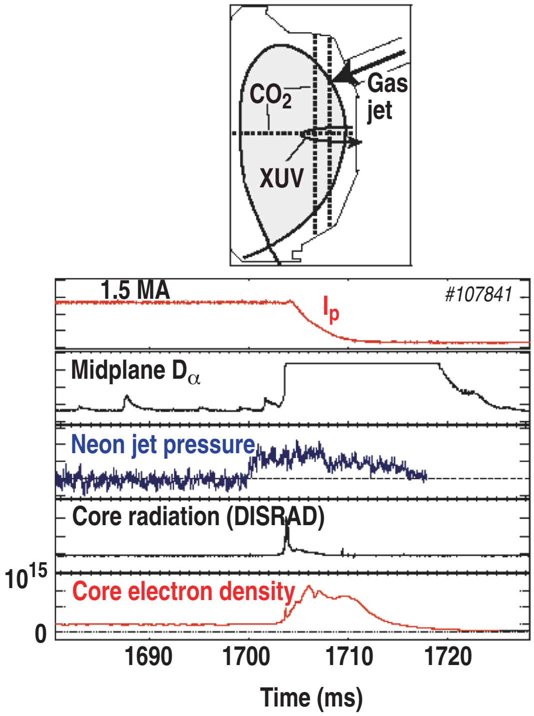

Fig. 6. Disruption mitigations in DIII–D. A high pressure gas jet is injected into the plasma. The plasma current has a smooth decay, and the gas is seen reaching the center of the plasma from the radiation and density measurements.

VIII. CONCLUSION

High performance advanced tokamak modes are routinely obtained in the DIII–D tokamak. Significant progress has been made in suppressing the resistive wall modes by using internal sensors and external feedback control coils, leading to high performance plasmas. Internal control coils are scheduled to be installed during 2002, which should allow full suppression of the RWMs and increase of β_N to within a few percent of the ideal wall limit. The 3/2 NTMs have been suppressed by using ECCD. With additional ECH power it should also be possible to suppress the 2/1 NTMs. Disruption mitigation has successfully been obtained using massive gas puffing, which uniformly radiates the stored energy to the plasma facing components, reduces the electromagnetic forces on the PFCs, and prevents the generation of runaway electrons.

ACKNOWLEDGMENT

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