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DISCLAIMER

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Control of MHD Stability in DIII-D Advanced Tokamak Discharges

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Advanced tokamak research in DIII-D seeks to optimize the tokamak approach for fusion energy production, leading to a compact, steady state power source. High power density implies operation at high toroidal beta, $\beta_T=\langle p\rangle 2\mu_0/B_T^2$, since fusion power density increases roughly as the square of the plasma pressure. Steady-state operation with low recirculating power for current drive implies operation at high poloidal beta, $\beta_P = \langle p\rangle 2\mu_0/\langle B_P\rangle^2$, in order to maximize the fraction of self-generated bootstrap current. Together, these lead to a requirement of operation at high normalized beta, $\beta_N = \beta_T(aB/I)$, since $\beta_P\beta_T \approx 25[(1+\kappa^2)/2]\ (\beta_N/100)^2$. Plasmas with high normalized beta are likely to operate near one or more stability limits, so control of MHD stability in such plasmas is crucial.

1. Kink Mode Stabilization With A Resistive Wall

In “advanced tokamak” scenarios, the broad current density profile associated with a large bootstrap current leads to a relatively low free-boundary kink mode beta limit, but also allows the possibility of stabilization by an ideally conducting wall. In the presence of a resistive wall, such as the DIII-D vacuum vessel, the kink mode is not completely stabilized but is converted to a slowly-growing resistive wall mode (RWM), which can be stabilized by feedback control or plasma rotation.

Sustained wall stabilization of the external kink mode has been demonstrated in an advanced tokamak discharge with high normalized beta and high bootstrap current [1]. As seen in Fig. 1, beta exceeds the calculated no-wall limit of $\beta_N \sim 4\ell_i$ for almost one second, and reaches a maximum near the estimated ideal-wall stability limit at $\beta_N \sim 6\ell_i$. Here plasma rotation plays a major role in the stabilization, with a set of six external, non-axisymmetric coils (C-coils) providing feedback-controlled correction of n=1 error fields that would otherwise slow the plasma rotation. Direct feedback control of the resistive wall mode has also been demonstrated in plasmas where the rotation had decayed to a value below the threshold for stabilization by rotation alone [2,3].

Recently, a new set of internal control coils (I-coils) has been installed in DIII-D, consisting of two sets of six coils, one set above and one below the midplane [4]. VALEN code calculations [5] predict that the I-coils can allow feedback-stabilized operation up to the ideal-wall limit even in the absence of plasma rotation. Preliminary results indicate that the I-coil provides feedback-driven error field correction at least as effectively as the C-coil, and stable operation above the estimated no-wall beta limit has been sustained for up to 2.5 s.

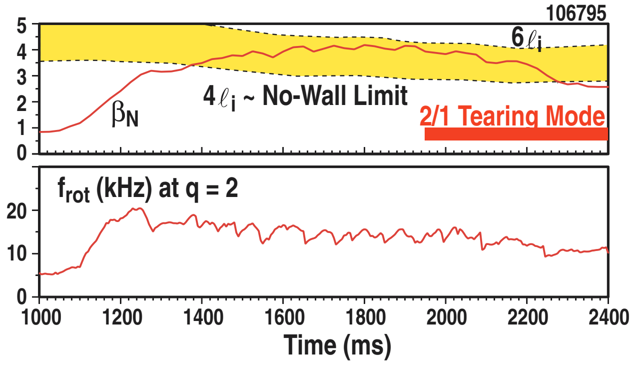

Fig. 1. (a) An advanced tokamak discharge (65% bootstrap current and 85% total noninductive current, $\beta_T \geq 4\%$) is stable above the ideal no-wall limit, with (b) rapid toroidal rotation made possible by feedback-controlled error field reduction.

Preliminary results are encouraging for the prospects of resistive wall mode control without plasma rotation. Figure 2 shows a case in which strong "magnetic braking" reduced the plasma rotation to a very low value, essentially zero in the outer half of the plasma. Feedback stabilization with the I-coil sustains the plasma at beta above the no-wall limit for about 100 ms after the outer plasma rotation reaches zero. A comparison discharge without feedback becomes unstable when the outer plasma rotation decays below a threshold value of about 6 kHz, even though beta is slightly smaller.

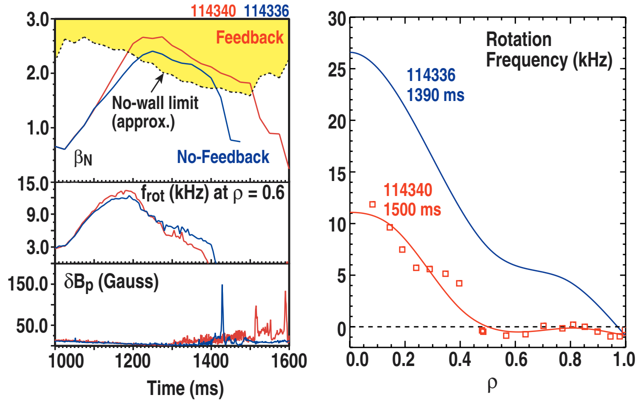

Fig. 2. (a) A discharge with a low no-wall stability limit (red curves) is stabilized by feedback after (b) the rotation at $\rho \geq 0.5$ has decreased to zero. A lower-beta discharge without feedback (blue curves) becomes unstable (c) as the rotation at $\rho = 0.6$ decays below about 6 kHz. (d) Comparison of rotation profiles just before the onset of the resistive wall mode.

2. Neoclassical Tearing Mode Stabilization

High beta plasmas are subject to the neoclassical tearing mode (NTM), where the helical perturbation to the bootstrap current caused by a magnetic island further destabilizes the island and causes it to grow. The wall-stabilized discharge shown in Fig. 1 is ultimately ended by an NTM. Stabilization of neoclassical tearing modes can be achieved through replacement of the missing bootstrap current with electron cyclotron current drive. DIII-D experiments have demonstrated suppression of the m/n=3/2 and 2/1 modes, using precise feedback control of the current drive position to within 1 cm by variation of the plasma position or toroidal field [6]. As shown in Fig. 3, once the NTM is stabilized, beta can be raised to a value higher than at the initial onset of the NTM. In this example, the feedback system is driven by minimizing the amplitude of the detected NTM and cannot follow the location when the mode amplitude vanishes. More recently, feedback control has been upgraded to provide real-time tracking of the rational surface associated with the NTM, even in the absence of an unstable mode.

3. Profile Control

Stable, steady-state operation with high fusion performance may depend on maintaining profiles that differ from those that would naturally evolve in the plasma, so local control of the pressure, current density, and rotation profiles is an essential element of advanced tokamak plasmas. Recent DIII-D experiments have demonstrated feedback control of the local electron temperature by electron cyclotron heating, and modification of the central current density profile evolution with electron cyclotron current drive [7]. Real-time equilibrium reconstructions using motional Stark effect data are being implemented, and will ultimately allow real-time control of the current density profile.

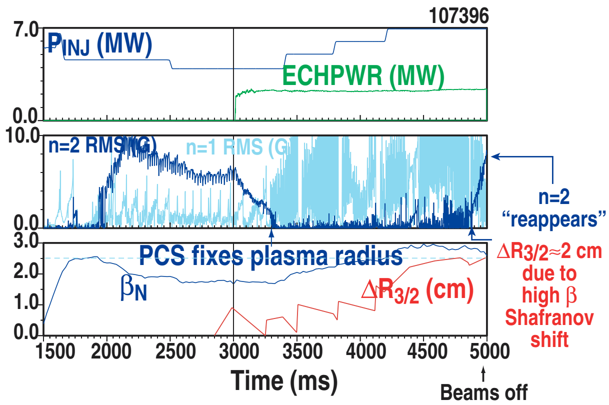

Fig. 3. Suppression of an m/n=3/2 neoclassical tearing mode. (a) When electron cyclotron current drive is turned on, (b) the n=2 mode amplitude is reduced to zero. (c) Beta can then increase above the level where the NTM first appeared. Eventually the stabilization is lost as the increased Shafranov shift moves the q=3/2 surface away from the ECCD resonance.

Ideally, real-time profile measurements should be compared to predicted stability limits in order to avoid instabilities, but reliable prediction of those limits in real time may be a significant challenge. "MHD spectroscopy," based on the plasma's resonant response to small-amplitude magnetic perturbations near a stability limit [8], provides a direct, real-time measurement of plasma stability that may prove useful as a control input.

4. Disruption Mitigation

In the event that avoidance of instabilities by profile control and suppression of instabilities by direct feedback control both fail, a disruption may follow. Disruption mitigation by injection of a high-pressure impurity gas jet leads to a radiative thermal quench and rapid current decay, reducing runaway electrons, thermal loads, and electromagnetic forces on plasma facing components [9]. In recent DIII-D experiments, real-time detection of off-normal conditions has been successfully used to trigger such a controlled termination. In Fig. 4, the vertical position control is deliberately switched off to create a disruption. The control system detects the excursion in vertical position and triggers a high-pressure neon jet. More sensitive detection and earlier triggering increase the radiated power and reduce the halo currents and associated electromagnetic forces. Similar detection schemes have been developed for density limit and locked-mode disruptions.

5. Conclusions

Experiments in DIII-D have demonstrated many of the elements needed for control of MHD stability in advanced tokamak plasmas. Stabilization of the external kink by a resistive wall with rapid plasma rotation is an effective technique, made possible by feedback-controlled reduction of error fields. The feasibility of stabilizing resistive wall modes and neoclassical tearing modes by direct suppression of the instabilities has been demonstrated. The tools for real-time profile control are at hand, including profile measurements and means of localized heating and current drive, while MHD spectroscopy provides a direct measurement of proximity to a stability limit. Gas jet injection can terminate the discharge safely if a stability limit is exceeded.

A number of scientific and technical issues remain for implementing these techniques in a burning plasma. A better understanding of the physics of plasma rotation and rotational stabilization is needed for extrapolation to a burning plasma, which is likely to have little or no torque from neutral beam heating. Models of RWM feedback stabilization in a rotating plasma are being developed [10], and validation is needed. Nonmagnetic methods such as reflectometry may allow resistive wall mode detection at a threshold of 1 part in 10^3 or 10^4 of the ambient magnetic field in long pulses where the accuracy of inductive sensors is a challenge. In general, active control of profiles and of the instabilities themselves will require flexible systems of actuators and complete measurements in real time of the internal state of the plasma. An integrated system of real-time measurement and control remains to be demonstrated. Finally, although the gas jet mitigation technique appears to scale favorably to the burning-plasma regime, this must be shown through comparative experiments on existing tokamaks.

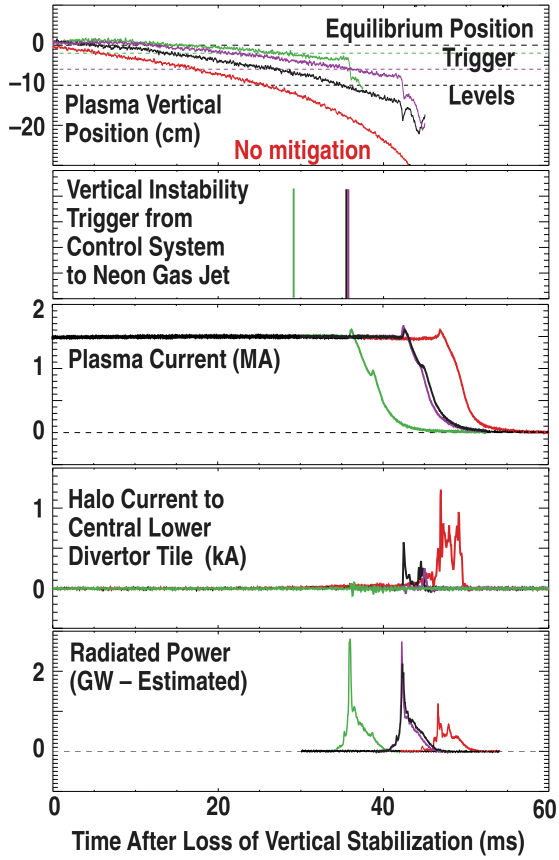

Fig. 4. Real-time triggering of a high-pressure gas jet during a vertical displacement event increases the radiative dissipation and reduces the halo currents during the disruption.

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