NEOCLASSICAL ISLANDS, β - LIMITS, ERROR FIELDS, AND ELMS IN REACTOR-SCALE TOKAMAKS

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Abstract

An assessment is presented of the impact of recent magnetohydrodynamic research results on performance projections for reactor-scale tokamaks as exemplified by the ITER Final Design Report facility. For nominal ELMy H-mode operation, the presence and amplitude of neoclassical tearing modes governs the achievable β-value. Recent work finds that the scaling of β at which such modes onset agrees well with a polarization drift model, with the consequence that, with reasonable assumptions regarding seed-island width, the mode onset β will be lower in reactor-scale tokamaks than in contemporary devices. Confinement degradation by such modes, on the other hand, depends on relative saturated island size which is governed by principally by β and secondarily by ν^*-effects on bootstrap current density. Relative saturated island size should be comparable in present and reactor devices. DT ITER Demonstration Discharges in JET exhibited no confinement degradation at the planned ITER operating value of β_N=2.2. Theory indicates that Electron Cyclotron Current Drive can either stabilize these modes or appreciably reduce saturated island size. Turning to operation in candidate steady-state, reverse-shear, high-bootstrap-fraction configurations, wall stabilization of external kink modes is effective while the plasma is rotating but (so far) rotation has not been maintained. Recent error field observations in JET imply an error-field size scaling that leads to a projection that ITER/FDR will be somewhat more tolerant to error fields than thought previously. ICRF experiments on JET and Alcator C-Mod indicate that plasmas heated by central energetic particles have benign ELMs compared to the usual type 1 ELM of NBI-heated discharges.

1. INTRODUCTION

Magnetohydrodynamics governs many areas of plasma performance for a reactor-scale experimental device as exemplified by the ITER Final Design Report facility [1]. Since fusion power increases as β^2B^4, there is considerable incentive to assure that $\beta_N = (\beta_{\%})aB/I > 2.2$ to accomplish the nominal mission of the ITER/FDR device and to investigate the prospects for reliable operation at $\beta_N > 3.0$, thus providing an experimental database to support design of a demonstration fusion power plant. This paper assesses the implications of current research on projections for reactor-scale plasma performance in four areas: 1) neoclassical tearing modes and their control by Electron Cyclotron Current Drive (ECCD), 2) evolution of discharges that require a nearby conducting wall for stability against ideal n=1 kink modes, 3) the threshold error fields above which locked modes will grow and degrade plasma performance, and 4) experimental observations of benign ELMs with ICRF heating.

2. NEOCLASSICAL TEARING MODES

During the ITER/EDA, it has become recognized that neoclassical tearing modes govern the β-values that long-duration discharges can attain [2]. The general experimental observation is that usually (m,n)=(3,2) modes arise first and cause a 10-20% degradation in confinement, which corroborates well with calculations of the saturated island sizes. At higher β-values, (2,1) modes can onset and lead to plasma disruptions. Normally, only a single mode is present. Theory indicates and experiments support the concept that neoclassical tearing modes are a metastable phenomena [3], requiring a finite β-value and finite seed island size to exhibit growth. Once growth is initiated, the island size will grow to a saturated island size that is well-supported by experimental observations [4]. This island size is directly proportional to the bootstrap current density in the vicinity of the mode rational surface. Recent work has focused on the onset β-value and island size needed to initiate neoclassical island growth. The theoretical model introduces a stabilizing polarization drift term into the island evolution equation. This term is found to better replicate mode-onset data than existing models for finite island thermal conductivity physics. For a reactor scale device, the seed island size necessary to initiate mode growth scales with the ion gyroradius and therefore is relatively smaller compared to the plasma size in a reactor scale device. The onset β-value also has a normalized gyroradius scaling. Theoretical estimates for the scaling of sawtooth-induced seed-island widths are not yet well-established so definitive projections of mode occurrence can not be made. The principal interim conclusions are that sawtooth-induced neoclassical tearing modes will arise on reactor scale devices and that the saturated island sizes (and hence confinement degradation) will depend on β and ν^*, which are similar in contemporary devices and reactors. Thus ITER Demonstration Discharges should be representative of the limitation imposed by neoclassical tearing modes. Figure 1 portrays a DT ITER Demonstration Discharge on JET, which supports the conclusion that neoclassical tearing modes may not occur or significantly degrade confinement at β_N = 2.2.

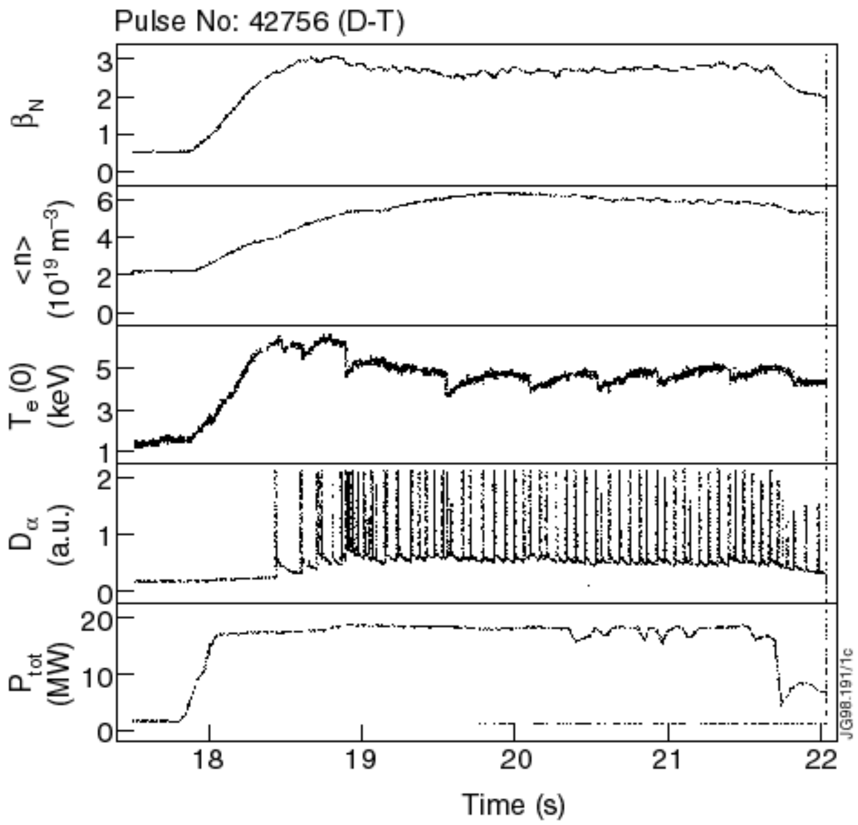

FIG. 1. JET D-T ELMy H-mode ITER Demonstration Discharge. Normalised β, line average electron density (10^{19} m^{-3}), central electron temperature Te(0), D_α and total power (MW) versus time for JET pulse 42756.

3. ACTIVE CONTROL OF NEOCLASSICAL MODES

Control and suppression of neoclassical tearing modes will assure operation of the ITER/FDR device at $\beta_N \approx 2.2$ and open the possibility of reliable operation of a demonstration reactor at $\beta_N \geq 3.0$. Since neoclassical tearing mode growth depends on resistive reconnection near the mode rational surface, the growth time can be long for a large, hot plasma as anticipated for the ITER/FDR device. Representative growth times are 20-50 s, so that active control and/or stabilization are distinct and attractive possibilities [5,6,7].

Two possibilities based on ECCD are examined: 1) Mode stabilization by modulated ECCD and 2) saturated island reduction by continuous ECCD. The first approach is based on replacing by ECCD the bootstrap current within the island which has become lost as a result of the flat pressure and temperature profiles within the island. Theoretical calculations [5] show that if the ECCD is modulated 50% *on*-50% *off* then a driven current density exceeding $1.5 \cdot \mathbf{j}_{\text{bootstrap}}$ in the neighborhood of the rational surface will stabilize arbitrarily small islands. Stabilization is insensitive to the width of the driven current profile and thus to the total driven current. The phase of modulation must be such as the drive forward current at the island O-points. The second approach rests on two observations: 1) The saturated island width is inversely proportional to the quantity ($-\Delta'$) which customarily depends on the global current density profile, and 2) thin current drive layers just outside the mode rational surface can make large changes in ($-\Delta'$). In representative calculations, the value of ($-\Delta'$) can increase an order of magnitude leading to an inverse decrease in saturated island size [8], provided again that the driven current density exceeds the bootstrap density. The two processes can be combined and lead to either stabilization or saturated island widths comparable to the driven current width which can be quite small. Theoretical ray-tracing calculations based on 100 MW ECCD power show that current densities exceeding 0.3 MA/m^2 in an 0.2 m layer can be driven for normalized radius ρ in the range $0.2 < \rho < 0.8$, while a representative bootstrap current density is $\mathbf{j}_{\text{bootstrap}} \approx 0.1$ MA/m^2 [2]. However, the driven current density profiles are sensitive to gyrotron frequency as well as poloidal and toroidal launch angles, indicating that experimental current drive studies will be necessary to learn how to place thin current drive layers just outside mode rational surfaces as theory requires. Current layers further from the rational surface will destabilize regular tearing modes. Overall, theoretical prospects for control appear quite favorable. Preliminary experimental results are just beginning to arrive [9].

4. BETA LIMITS AND WALL STABILIZATION

The principal approach to steady-state operation of the ITER/FDR device is via reverse-shear, high-bootstrap-fraction discharges that achieve a high degree of congruence between bootstrap current density and the required current density. In the absence of a nearby conducting wall, such discharges are known to be unstable to the n=1 external kink mode at the β_N-values (β_N = 3.5-4.0) [10] needed to generate 1500 MW of fusion power in ITER/FDR. When a wall is present, these modes are no longer ideally unstable but can develop resistive wall modes which grow on a time scale characteristic of magnetic field penetration through the wall. Plasma rotation is essential to stabilizing resistive wall modes and a new sub-field of tokamak magnetohydrodynamics – the study of plasmas unstable in the absence of a wall – is just beginning [11,12]. This field is essential to the realization of steady-state tokamak reactors.

Initial [10,13] and more recent [14] experimental results from DIII-D indicate that wall stabilization based on plasma rotation works and discharges can be maintained for many MHD times and ~ 10 resistive wall times. But, in the discharges published to date, plasma rotation could not be maintained in conditions when kink modes were unstable in the absence of a conducting wall. The resulting gradual decrease of plasma rotation had the consequence that growth of modes on resistive wall times could not be avoided. Experimental values of the marginal rotation speed for instability to occur are lower than theoretical predictions [11,12]. A preliminary conclusion is that plasmas lose toroidal rotation when unstable in the absence of a conducting wall, but this is based on a handful of published discharges. If rotation cannot be maintained, then active n=1 magnetic feedback systems [15] will be needed.

5. ERROR FIELD SCALING

Small levels of non-axisymmeric error fields [16] can initiate locked mode instabilities which degrade confinement and lead to disruptions. Systematic experiments in JET and COMPASS-D [17] and in DIII-D [18] show that low n,m error fields with $B_{m,n}/B_o \leq 10^{-4}$ produce locked modes in comparable low-density plasmas in each tokamak. For ITER, the key issue is how these threshold observations (allowable $B_{m,n}/B_o$) will extrapolate in size and field. Extrapolation of the JET data to ITER using the Kadomstev quasineutrality constraint yields an $R^{0.4}$ size scaling and estimated ITER thresholds of about $B_{m,n}/B_o=10^{-4}$ [16]. Similar but more uncertain estimates result from B_o extrapolation of the DIII-D data, but COMPASS-D results show a different magnetic field scaling and lower (~ 2 x 10-5) allowable ITER error field [17]. Until this discrepancy is resolved, the requirement for ITER error field compensation of the rms-weighted sum of the 1,1, 2,1 and 3,1 modes to a level of 2 x10^{-5} [1] will be retained.

6. ICRF ELMs

Because of ITER's long pulse length, ELMs can cause significant erosion of plasma facing materials. This is especially true for type 1 ELMs which are present in most of the ELMy H-mode discharges used in the confinement database. It is therefore noteworthy that under ICRF heating conditions two tokamaks, JET and Alcator C-Mod, exhibit a different type of response. For JET, ELMS associated with ICRF heating are of a much more frequent and lower amplitude than the standard type 1 ELMs associated with NBI heating [19, 20]. In the case of Alcator C-Mod, ELMs are not observed at all in H-mode operation but instead an increased level of D_α radiation [21]. Future work should focus on the causes for the differences in behavior.

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