Topic: EX-S

Physics and Control of ELMing H–mode Negative Central Shear Advanced Tokamak Scenario for ITER Based on Experimental Profiles*

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The goal of magnetic fusion research is to develop fusion energy as an economical and viable energy source. Two of the major research elements are the development of advanced tokamak (AT) configurations with good confinement and improved stability at high β and steady-state, and the study of these configurations under burning conditions. An international fusion energy advanced tokamak burning plasma machine, ITER-FEAT [1], has recently been proposed to study inductive and steady state burning plasmas, which offers an opportunity to continue development of the AT path.

In this paper, the potential for ITER-FEAT to operate in an ELMy H-mode AT scenario is evaluated. Using the ITER-FEAT inductive and steady state advanced scenarios [2] as starting points, key DIII–D AT experimental and modeling results are applied to assess the requirements for ITER-FEAT to operate in an ELMing H–mode negative central shear (NCS) AT scenario. Both physics and control issues are examined. These include rotation and feedback stabilization of resistive wall modes (RWM) for high β operation, constraints on the edge pedestal for high fusion performance due to drift-wave based core transport and edge localized modes (ELM), disruption mitigation, and divertor heat load. The effects of a finite edge pressure pedestal and current density are self-consistently included. This is crucial for evaluation of the effects of ELMs on edge pedestal, divertor heat load, and hence fusion performance. For these AT configurations, stability and transport analyses indicate that a modest amount, ~35 MW, of 1 MeV negative NBI can provide sufficient rotational drive for stabilization against the n=1 RWM and allow operation at attractive β_N values in the range 3.0–3.5. Consideration of edge stability and core transport suggests that a pedestal width in the range of ~5% ψ_N, is likely to be sufficient for the projected fusion performance. The pedestal width requirement can be reduced somewhat by operating at lower edge density. This will involve tradeoffs with divertor heat flux and may impact the overall fusion performance.

The equilibria used in this study are based on current and pressure profiles taken from recent DIII–D long-pulse high-performance AT discharges with realistic edge density and temperature pedestals. They are computed using the equilibrium package of the CalTrans code. The plasma shape [Fig. 1(a)] and other global plasma parameters for the base case are similar to the ITER-FEAT steady-state AT configurations [2], although β_N is slightly higher. These are summarized in Table 1. Stability analysis indicates that at β_N ~ 3.4 these AT configurations are unstable to the ideal n=1–3 modes without a conducting wall. They are stable with a wall at 1.2 times the actual ITER-FEAT wall. With increased triangularity, the n=2 and 3 modes become stable even without a wall. For operation at these attractive β values, it is important to maintain the stability against these pressure-driven kink modes.

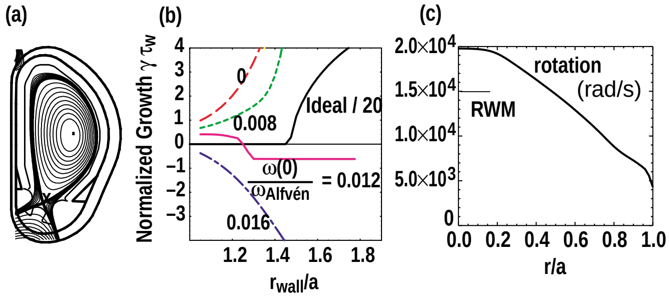

Fig. 1. (a) The ITER-FEAT plasma shape used in this study. (b) The growth rate variation of the n=1 resistive wall modes normalized to the resistive wall time with the distance to the wall at various central plasma toroidal rotation velocities normalized to Alfvén velocity. (c) The toroidal rotation profile produced by ~35 MW of 1 MeV negative NBI. Also shown is the rotation needed for stabilization against the n=1 RWM $\cong$ 1% Alfvén frequency.

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Table 1: Comparison of ITER-FEAT Major Plasma Parameters for 3 different scenarios. Also shown are the papameters for a JT-60U discharge.

Case	Inductive [2]	Steady State [2]	DIII-D AT-Like	JT-60U [3]
I_P (MA)	15	10	10	1.5
B_T (T)	5.3	5.3	5.3	3.6
R (m)	6.20	6.35	6.35	3.30
a (m)	2.0	1.85	1.86	0.82
κ_X	1.85	1.95	1.94	1.48
δ_X	0.49	0.56	0.52	0.36
q_0	1.0	2.4	2.2	1.90
q_{min}	1.0	2.0	1.6	1.75
q_{95}	3.0	4.4	4.7	4.1
β_N	1.8	3.2	3.4	2.8
β_P	0.65	1.80	1.90	1.97
β_T(%)	2.5	3.3	3.6	1.43
ℓ_i	0.85	0.67	0.56	0.78

Resistive wall mode stability analysis using the MARS code and a sound wave damping model indicates that a central rotation rate ~1% of the Alfvén rotation frequency is sufficient to stabilize the n=1 resistive wall modes [Fig. 1(b)]. The required neutral beam injection power is estimated using the ONETWO transport branch of the CalTrans code. The results indicate that ~35 MW of 1 MeV negative NBI is needed to provide sufficient rotational drive for stabilization against the n=1 RWM at a density ~n_{GW}. This is shown in Fig. 1(c).

Vertical stability and the resulting halo currents are analyzed using the stability package of the CalTrans code and an analytic halo current model. The post-thermal quench and halo plasma resistivities are estimated based on a set of DIII–D AT disruption data. The results indicate that reliable disruption mitigation is crucial but disruption mitigation using an impurity gas such as Ar can significantly reduce the peaked halo current to an acceptable level.

Transport analysis using the recently re-normalized drift-wave based transport model GLF23 indicates that the temperature at the top of the edge pedestal plays a very important role in determining the overall fusion performance, as expected. To produce 400 MW of fusion power (Q~10) at n_e ~ n_{GW} requires an edge pedestal temperature of ~4–5 keV. This requirement is reduced with higher plasma density. This is illustrated in Fig. 2(a).

The constraints on the edge pedestal height and the divertor heat load due to ELMs are evaluated by analyzing the edge stability of these configurations against the ideal intermediate n=10–30 peeling-ballooning modes using the ELITE code. The results indicate that a pedestal width in the range of ~5% ψ_N, as expected from empirical scaling, is likely sufficient for the projected fusion performance. The stability limit on the pedestal temperature can be increased by reducing the edge pedestal density [Fig. 2(b)]. However, this involves tradeoffs with divertor heat flux and may impact the overall fusion performance. As in the low n case, increasing triangularity is found to stabilize these modes. At high upper triangularity, the flux surfaces in the far scrape-off layer may peel off and intersect the wall at a non-divertor location.

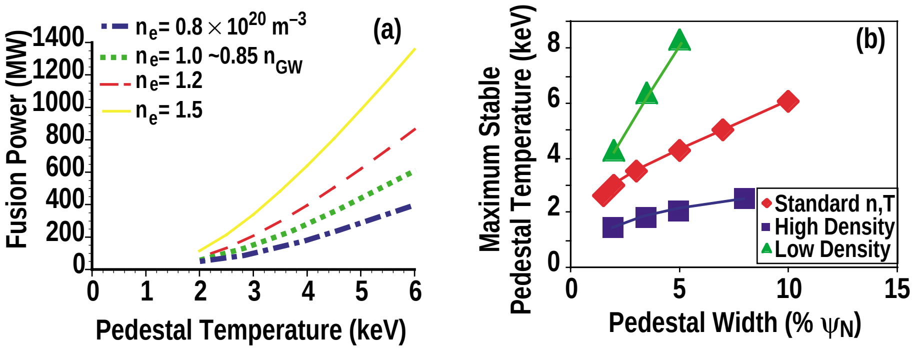

Fig. 2. (a) Variation of fusion power with edge pedestal temperature at various densities from drift-wave based transport. (b) Variation of maximum stable edge pedestal temperature with pedestal width at various densities due to intermediate n ~ 10–30 edge peeling-ballooning modes.

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[3] Y. Kamada, *et al.*, Fusion Energy 1996 (Proc. 16th Int. Conf., Montreal, 1996) Vol. 1, Vienna (1997) 227.