High-β Steady-State Advanced Tokamak Regimes for ITER and FIRE

D. M. Meade[1], N. R. Sauthoff[1], C. E. Kessel[1], S. C. Jardin[1], G.A. Navratil[2], J. Bialek[2], M. A. Ulrickson[3], T. Rognlein[4], J. Mandrekas[5], C. K. Skinner[1], G. Hammett[1], R.V. Budny[1], G. Kramer[1], N. Gorelenkov[1], T K. Mau[6], P. H. Rutherford[1], K. M. Young[1], D. W. Swain[7], P Bonoli[8], J. Decker[8] and J. A. Schmidt[1],

[1]Princeton Plasma Physics Laboratory, Princeton, NJ 08543, USA
[2]Columbia University, New York, NY 10027, USA
[3]Sandia National Laboratory, Albuquerque, NM 87185, USA
[4]Lawrence Livermore National Laboratory, Livermore, CA 94551, USA
[5]Georgia Institute of Technology, Atlanta, GA 30332, USA
[6]University of California at Los Angeles, Los Angeles, CA
[7]Oak Ridge National Laboratory, Oak Ridge, TN 37831, USA
[8]Massachusetts Institute of Technology, Cambridge, MA 02139, USA

An attractive tokamak-based fusion power plant will require the development of high-β steady-state advanced tokamak regimes to produce a high gain burning plasma with a large fraction of self-driven current and high fusion power density. The ongoing tokamak program and a next-step burning plasma experiment have the goal to understand the physics and to determine the requirements for attaining, controlling and sustaining high-β steady-state advanced tokamak regimes for time scales long compared to internal plasma time-scales. Two activities (ITER and FIRE) are underway to develop the experimental capability to address these requirements in a burning plasma. The present ITER regimes are focused on the physics and plasma technology of moderate power density plasmas sustained for very long pulse ($\sim$10 τ_{cr}) while the FIRE regimes are focused on high power densities sustained for moderate pulse lengths (3 – 5 τ_{cr}). The physics and plasma technology issues of ITER and FIRE are very similar, and technical solutions for one will likely be applicable to the other. The major common issues are: (1) refinement of predictive capability and optimization of confinement modes, (2) improved understanding of edge plasma behavior leading to reduction of edge plasma power loss during ELMs and disruptions, (3) extension of advanced tokamak scenarios toward higher β and bootstrap current, (4) analysis of instabilities driven by energetic particles in fusion plasmas,(5) development of plasma-facing components to handle high power densities while maintaining a low tritium inventory, (6) development of practical plasma control techniques (profile control and feedback systems) and (7) development of diagnostics suitable for burning plasma physics and plasma control.

Both ITER and FIRE are being designed with the objective to address these issues by exploring and understanding burning plasma physics in the conventional H-mode regime, and in the advanced tokamak ($\beta_N \sim 3$ - 4, f_{bs} ~50 - 80%) regime envisioned for an attractive steady-state high power density fusion power plant. The goal of the present work is to develop AT modes that would fully exploit the capabilities of ITER and FIRE. ITER has emphasized conservative scenarios, as appropriate for their nuclear technology mission, while FIRE has emphasized more aggressive assumptions aimed at expanding present tokamak AT experiments and exploring the scenarios envisioned in the ARIES power plant studies. The main characteristics of the advanced scenarios presently under study for ITER and FIRE are compared with advanced tokamak regimes envisioned for the European Power Plant Conceptual Study (PPCS-C) and the US ARIES-RS Power Plant Study in Table 1.

Table I. Advanced Tokamak Parameters

	ITER-AT	PPCS-C	FIRE-AT	ARIES-RS
R (m), a (m)	6.35,1.85	7.5,2.5	2.14, 0.595	5.52, 1.38
κ_x , κ_a , κ_{95}	, 1.85 ,	2.1, 1.9,	2.0, 1.85,1.82	1.9, - ,1.70
δ_x , δ_{95}	, 0.40,	0.7, 0.47,	0.7, 0.55	0.77, 0.5
Div. Config., material	SN, C(W)	SN, W	DN, W	DN, W
(P_{loss})/R (MW/m)	15	~70	16	80
$B_t(R_o)$ (T), I_p (MA)	5.1, 9	6, 20	6.5, 4.5	8, 11.3
q(0),q_{min}, q_{95}	3.5, 2.2, 5.3		4, 2.7, 4.0	2.8, 2.49, 3.5
β_t(%), β_N , β_p	2.8, 3.1, 1.5	5, 4 ,	4, 4.1, 2.15	5, 4.8, 2.29
f_{bs} (%)	48	69	77	88
Non Inductive CD. %	100	100	100	100
n(0)/ $\langle n\rangle_{vol}$, T(0)/ $\langle T\rangle_{vol}$	1.5,	1.5,2.5	1.5, 3.0	1.5, 1.7
n/n_{GW}, $\langle n\rangle_{vol}$ (10^{20} m^{-3})	0.8	1.5	0.85, 2.4	1.7, 2.1
$T_i(0)$, $T_e(0)$	31	40	14, 16	27, 28
Z_{eff}	2.1	2.2	2.3	1.7
H98(y,2)	1.6	1.3	1.7	1.4
τ_E, (s)	3.1		0.7	1.5
Burn Duration/τ_{cr} , s	10, 3000	Steady-state	3.2, 40	Steady-state
Q = $P_{fusion}/(P_{aux} + P_{OH})$	6	30	4.8	25
Fusion Power (MW)	360	3400	140	2160
P_{fus}/Vol (MWm^{-3})	0.45	1.9	5.5	6.2
Γ neutron (MWm^{-2})	0.5	2.2	1.7	4

The AT modes for FIRE have been developed using the Tokamak Simulation Code (TSC) for integrated scenario analysis, JSOLVER/BALMSC/PEST/DCON for MHD stability analyses, LSC/AORSA/ACCOME/CURRAY for CD, and VALEN for resistive wall mode feedback stabilization studies. The FIRE AT mode has FWCD for an axis current drive and lower hybrid (~5 GHz) for off-axis current drive. Key features are the strong plasma shaping and the use of four n=1 pairs RWM stabilization coils mounted just behind the first wall tiles in each of the 8 open mid-plane ports. Although inductive and non-inductive current drive are used to ramp the plasma current up, the flattop plasma has 100% non-inductive current provided by the combination of bootstrap, lower hybrid, and fast wave current, and the current profile is held constant for 3.2 τ_{CR}. The FIRE-AT $\beta \approx 4\%$ would result in a fusion power density of 5.5 MW/m^{-3}, which is close to ARIES-RS. Studies using VALEN are underway to investigate the benefits of mounting RWM coils inside the vacuum vessel of ITER. Initial results for coils mounted just behind the shield modules indicate that β_N could be raised to 3.7 while maintaining n= 1 stability; these studies are ongoing. The excitation of TAE-like modes driven by energetic alpha particles in a background of fast particles from NNBI and ICRF heating is an important area for AT scenarios. Work is underway using TRANSP to calculate energetic particle distribution functions to use in the NOVA-K and HINST codes to determine TAE mode stability of the AT modes for ITER and FIRE. The operating space in power and pulse length is limited mainly by the power handling capability of the first wall and divertors. The UEDGE code was used to calculate the expected edge plasma conditions in FIRE. The tungsten brush/copper backing plate divertor targets are actively cooled and are capable of steady-state operation. The operating space of FIRE was extended up to ~5 τ_{CR} by optimizing distribution of power radiated in the divertor relative to the first wall. Work is ongoing on an e-beam test stand to develop all metal divertor and first wall materials suitable for high power densities and low tritium retention in ITER and FIRE.

Work supported by DOE Contract # DE-AC02-76CHO3073.