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

D. M. Meade[1], N. R. Sauthoff[1], C. E. Kessel[1], R.V. Budny[1], N. Gorelenkov[1], G.A. Navratil[2], J. Bialek[2], M. A. Ulrickson[3], T. Rognlein[4], J. Mandrekas[5], S. C. Jardin[1] 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

e-mail contact of main author: dmeade@pppl.gov

Abstract. 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. Both ITER and FIRE are being designed with the objective to address these issues by exploring and understanding burning plasma physics both in the conventional H-mode regime, and in advanced tokamak regimes with $\beta_N \sim 3$ - 4, and $f_{bs} \sim 50$ - 80%. ITER has employed conservative scenarios, as appropriate for its nuclear technology mission, while FIRE has employed more aggressive assumptions aimed at 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), the US ARIES-RS Power Plant Study and the Japanese Advanced Steady-State Tokamak Reactor (ASSTR). The goal of the present work is to develop advanced tokamak scenarios that would fully exploit the capability of ITER and FIRE. This paper will summarize the status of the work and indicate critical areas where further R&D is needed.

1. Introduction

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 [1, 2] for time scales long compared to internal plasma time-scales. Two activities (ITER and FIRE) have been undertaken to develop designs of experiments capable of addressing 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 pulses ($\sim 10\ \tau_{cr}$) while the FIRE advanced regimes are focused on high power densities sustained for moderate pulse lengths ($3 - 5\ \tau_{cr}$) sufficient to address burning plasma issues. 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 fraction, (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 advanced plasma control.

2. General Characteristics of ITER and FIRE Compared to Fusion Power Plants

The plasma parameters determined in fusion power plants design studies carried out in the US (ARIES-RS [3]), Japan (ASSTR [4]) and Europe (PPCS-C [5]) are compared with the ITER and FIRE steady-state AT scenarios in Table I. The power-plant designs invoke advanced tokamak operating regimes with high β to produce power densities of 2-6 MWm^{-3} and high bootstrap fractions of 70-88 % to facilitate efficient steady-state operation. These power plant scenarios also invoke magnetic fields of 6-11 T, somewhat higher than existing experiments. The steady-state power-handling requirements for the divertor (P_{loss}/R ~ 70-100 MW/m) and heat-flux loads on the first wall approaching 1-2 MWm^{-2} are a significant step beyond today's capabilities. A second goal of a next-step burning-plasma experiment is to develop plasma

Table I. Advanced Tokamak Parameters

	ITER-AT	**PPCS-C**	**FIRE-AT**	**ARIES-RS**	**ASSTR-2**
R (m), a (m)	6.35,1.85	7.5,2.5	2.14, 0.595	5.52, 1.38	6, 1.5
κ_x , κ_a , κ_{95}	-, 1.85,1.82	2.1, 1.9,	2.0, 1.85,1.82	1.9, - ,1.70	-, -, 1.8
δ_x , δ_{95}	0.55, 0.40	0.7, 0.47,	0.7, 0.55	0.77, 0.5	-, 0.4
Div. Config., material	SN, C(W)	SN, W	DN, W	DN, W	SN,W
$(P_{loss})/R$ (MW/m)	15	~70	16	80	~100
$B_t(R_o)$ (T), I_p (MA)	5.1, 9	6, 20	6.5, 4.5	8, 11.3	11, 12
$q(0)$,q_{min}, q_{95}	3.5, 2.2, 5.3		4, 2.7, 4.0	2.8, 2.49, 3.5	-, -, 4.8
β_t(%), β_N , β_p	2.8, 3.1, 1.5	5, 4 ,	4, 4.1, 2.15	5, 4.8, 2.29	-, 3.7,-
f_{bs} (%)	48	63	77	88	80
Current Drive Tech	NIBI, ECF	NBI	ICFW, LHF	ICFW, LHF	NINB
Non Inductive CD. %	100	100	100	100	100
$n(0)/\langle n\rangle_{vol}$, $T(0)/\langle T\rangle_{vol}$	1.3, 2.4	1.5,2.5	1.5, 3.0	1.5, 1.7	1.5, -
n/n_G, $\langle n\rangle_{vol}$ (10^{20} m^{-3})	0.8	,1.2	0.85, 2.4	1.7, 2.1	1.2, 2.1
$T_i(0)$, $T_e(0)$	27, 25	25	14, 16	27, 28	~35
Z_{eff}	2.1	2.2	2.3	1.7	1.6
H98(y,2)	1.6	1.3	1.7	1.4	
τ_E, (s)	2.7		0.7	1.5	
Burn Duration/τ_{cr} , s	10, 3000	Steady-state	4, 32	Steady-state	Steady-state
$Q = P_{fusion}/(P_{aux} + P_{OH})$	6	30	4.8	25	58
Fusion Power (MW)	360	3400	140	2160	3530
P_{fus}/Vol (MWm^{-3})	0.45	1.9	5.5	6.2	7.3
Γ neutron (MWm^{-2})	0.5	2.2	1.7	4	4.7

technologies needed for a fusion power plant, particularly in the plasma-facing component (PFC) area since these are strongly coupled to the performance of a burning plasma. The FIRE AT scenarios would produce fusion power densities close to those in ARIES-RS and ASSTR for plasma durations sufficient to address the strong non linear coupling of alpha heating and advanced-tokamak current profile but FIRE would be limited in its ability to exploit long-pulse technology issues. ITER would be able to address long-pulse issues but would be limited in its ability to address high-power-density physics and technology issues. The purpose of this study is to identify the physics and plasma technology requirements that would allow higher power density regimes (~ 1 MW/m^{-3}) in ITER, and longer duration regimes in FIRE.

3. H-Mode Operating Space

The physics issues, operating regimes and physics-design guidelines for projecting burning plasma performance in FIRE [6] are similar to those for ITER [7]. The H-mode operating space for ITER has been extensively documented [8]. A global systems code was used to determine the FIRE H-mode operating range. The analysis used for operating point calculations incorporated plasma power and particle balance and engineering constraints on power handling. ITER98(y,2) scaling is assumed for the global energy confinement time. The plasmas considered spanned the ranges: $5 \leq Q \leq 30$, $5 \leq P_{aux}$ (MW) ≤ 30, $1.05 \leq n(0)/\langle n\rangle \leq 1.25$, $0.3 \leq n/n_G \leq 1$,and $1.5 \leq \beta_N \leq 3$. In addition, the impurity concentrations in the plasma core were varied over 1 to 3% for Be and 0.0 to 0.3% for Ar, allowing higher radiated power fractions to more optimally distribute the exhaust power. Viable solutions must be within the engineering limits set by the heating of the cryogenically cooled toroidal field coils, stresses due to nuclear heating of the vacuum vessel, a temperature limit of 600 °C for the first wall Be tiles, particle power to the outboard divertor (<28 MW), and the radiated power load on the divertor and baffle (<6 - 8 MWm^{-2}). The duration of the nominal operating point in FIRE of 150 MW (5 MWm^{-3}) for 20 s (2 τ_{CR}) is limited by the heating of the toroidal-field coil as indicated in FIG. 1. Optimizing the distribution of exhaust power on the first wall, the divertor chamber walls and the divertor targets significantly expands the operating range of the conventional H-mode. If higher $\beta_N \approx 3$ (the no-wall stabilization limit) can be achieved, then fusion powers up to 300 MW (10 MWm^{-3}) could be accommodated for a 17s pulse length limited by the nuclear- and radiation-heating of the inertial first wall. High Q (15 - 30) operation could be attained for cases with low impurity content (1-2% Be), modest density peaking $n(0)/\langle n\rangle = 1.25$, $n/n_G = 0.7 - 1.0$ and H98(y,2) =1.03 - 1.1.

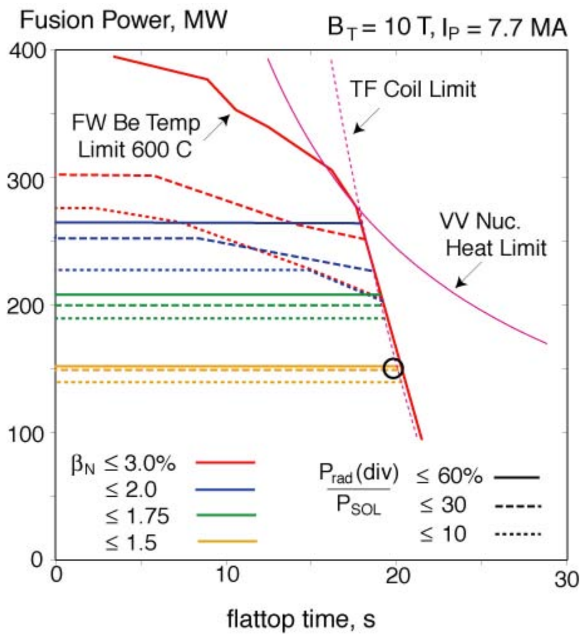

FIG. 1. H-Mode Operating Space in FIRE

Recent 0-D analyses [9] of the H-mode confinement data base by the ITPA CDBM group have refined the multi-machine ITER98(y,2) scaling [8] used for earlier projections of performance of ITER and FIRE. A two-term scaling relation incorporating the effects of plasma triangularity with separate scaling of the core and edge plasma [10] has improved confinement relative to ITER98(y,2). A new scaling with reduced beta degradation (eq. 10 in [9]) predicts somewhat lower confinement than ITER98(y,2) at nominal H-Mode operating

Table 2. Extensions of 0-D Confinement Scaling

τ_E(s)	B(T), I_p(MA)	ITER98(y,2), Q	Two Term[10]	No β scaling [11]
ITER-H	5.3, 15	3.66, ≈ 10 (H98=1)	3.86	3.24 ($\beta_N = 1.8$)
FIRE -H	10, 7.7	0.94, ≈ 10 (H98=1.03)	1.14	0.87 ($\beta_N = 1.8$)
ITER-AT	5.3, 9	2.02, ≈ 5 (H98 = 1.57)	1.8	2.04($\beta_N = 2.8$)
FIRE-AT	6.5, 5	0.36, ≈ 5 (H98= 1.7)	0.58	0.42 ($\beta_N = 4.2$)

points but improved confinement for higher β_N cases as shown in Table 2. The hybrid operating scenario [11] offers the possibility of improved performance (H98(y,2) = 1.2-1.5, $\beta_N = 2.7$, $f_{bs} \approx 50\%$ and $q_{95} \approx 3.2$). This regime would lead to Q values well above 10 in FIRE at full I_p, or could be used in ITER to run longer pulses at lower I_p with the same Q.

4. AT Mode Operating Space for FIRE

A global analysis, similar to that used for H-modes, has been used to determine the operating space for 100% non-inductive advanced-tokamak modes in FIRE [12]. An expression for the bootstrap-current fraction is included and the current-drive power is given by $P_{cd} = [nRIp(1-f_{bs})]/\eta_{cd}$. The on-axis current drive is fixed at 200 kA from ICRF/FW, so that LHCD must make up any current not driven by the bootstrap effect. The current drive efficiency used in these scans is η_{cd} = 0.2 and 0.16 A/W-m^2 for ICRF/FW and LH, respectively, and is based on detailed LH and ICRF/FW analysis for FIRE. The operating space was scanned for cases with Q = 5, at B_t = 6.5 T, P_{LH} (MW) ≤ 30, P_{ICRF} (MW) ≤ 30, 1.05 ≤ $n(0)/\langle n\rangle$ ≤ 2, 2 ≤ $T(0)/\langle T\rangle$ ≤ 3, 0.3 ≤ n/n_G ≤ 1, 3.25 ≤ q_{95} ≤ 5, and 3 ≤ β_N ≤ 4.5. Attainment of β_N ≥ 3 will require feedback stabilization of the resistive wall modes (RWM). In addition, the impurity concentrations are varied over 1 to 3% for Be and 0.0 to 0.3% for Ar, allowing higher radiated-power fractions. The operating space can be expanded by increasing Ar in the plasma to radiate more power in the divertor and on the first wall resulting in 1.5 ≤ Z_{eff} ≤ 2.3. The fraction of power radiated in the divertor (P_{rad}(div)) to power exhausted into the scrape-off layer (P_{SOL}) was scanned at 10%, 30% and 60%. The same power-handling limits were imposed as for the H-mode analysis. The nominal operating point has 150 MW of fusion power for 32 s flattop. The flattop burn times for these AT plasmas are limited primarily by the nuclear heating in the vacuum vessel rather than TF coil heating as shown in FIG. 2. Imposing these constraints, the system study found that FIRE could attain high-β high-bootstrap AT plasmas with near steady-state conditions for up to 5 τ_{CR}. If the vacuum vessel/ shield design was modified to withstand the nuclear-heating-induced stresses, the reference AT pulse length could be extended to ≈ 50 s (5 - 6 τ_{CR}). These Q = 5 plasmas require confinement corresponding to H98(y,2) ranging from 1.4 – 1.8 similar to those required in ITER. At the higher ranges of confinement, H98(y,2) = 1.6 – 2.0, Q = 10 plasmas are produced that have a reduced duration of 1-3 τ_{CR}.

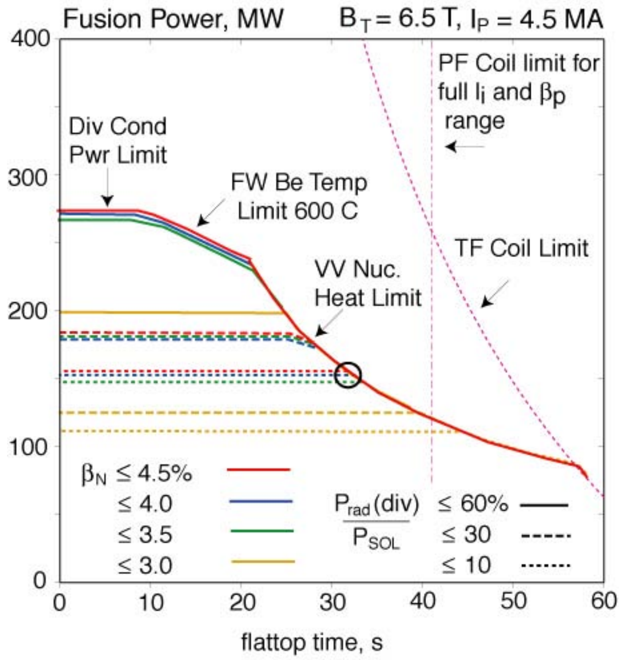

FIG. 2. AT Mode Operating Space for FIRE. The nominal operating point is at 150 MW with a duration of 4 τ_{CR}.

5. MHD Stability including Resistive wall Modes (RWM)

Stability analyses of bootstrap- and external-current-drive consistent equilibrium for FIRE show that plasmas stable to high-n ballooning and n = 1, 2, 3 with a wall, can reach β_N = 5.0, and with no wall the ideal MHD β_N limits for n=1, 2 and 3 are 2.7, 3.5, and 4.2, respectively. Calculations of RWM stability for FIRE with VALEN [13] show that feedback coils, located near the front face of the shield plug in every other mid-plane port, could stabilize the n=1 RWM mode up to 80-90% of the n=1 with wall limit. The influence of the n=2 mode on the achievable β_N is being investigated. The analysis of the RWM stabilization is benefiting from the experimental progress on DIII-D [14]. The FIRE AT configurations have safety-factor values above 2.0 everywhere, so that the (5,2) and (3,1) are the lowest order NTM's of interest. NTM stabilization using ECCD from the LFS at toroidal fields of 6-8 T would require frequencies of 147-197 GHz, which is close to the range of achieved values. Examination of this is continuing to determine if trapped electrons will degrade the CD efficiency excessively on the LFS, and whether heating alone may be sufficient.

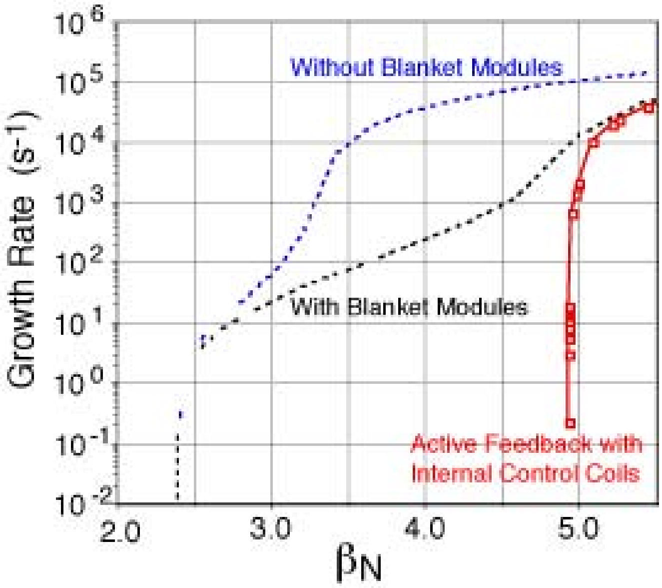

FIG. 3. VALEN computation of β-limits for passive and active stabilization of resistive wall modes in ITER.

Active control coils inside the vacuum vessel were found to have much better coupling to the unstable RWM and lead to improved feedback performance [15]. Motivated by the improved performance seen in FIRE using internal control coils, a similar improved RWM control coil configuration for ITER was analyzed. ITER AT modes similar to those described in the next section were found to have a no-wall β_N limit for n=1 about 2.5. These ITER AT modes aim at steady-state, non-inductive operation at $\beta_N \sim 3.3$, hence they will require active control of the RWM. The passive stabilization and active control of the RWM in these ITER AT plasmas was analyzed using the VALEN code and these results are summarized in FIG. 3 plotting the computed RWM growth rate versus β_N. We see in FIG. 3 that including only the passive stabilizing effects of the ITER double-wall vacuum vessel implied an ideal wall limit of $\beta_N \sim 2.4$, while inclusion of the stabilizing effect of the close fitting ITER blanket modules results in a substantial increase of the ideal wall limit to $\beta_N \sim 4.9$. Two feedback control configurations were modeled: (i) the baseline external error field correction coil set, and (ii) an improved feedback control configuration using six internal control coils driven as three n=1 coil pairs located in six of the eighteen ITER mid-plane ports, similar to the FIRE RWM control design. Optimization of RWM feedback using the ITER baseline error-field-control coils was limited to a relatively low maximum stabilized $\beta_N \sim 2.4$. Use of the improved configuration with six mid-plane-port-mounted control coils was found to be stable at $\beta_N >$ 4.9 approaching the ideal wall limit, which is more than adequate to permit operation of ITER in the proposed steady-state AT modes.

6. Integrated Modeling of Steady-State ITER and FIRE AT Modes.

AT modes for ITER and FIRE have been developed using the Tokamak Simulation Code (TSC) for integrated scenario analysis [16]. The JSOLVER/BALMSC/PEST/DCON codes are used for ideal MHD stability analyses, LSC/AORSA/ACCOME/CURRAY for RFCD, and VALEN for resistive wall mode feedback stabilization studies. The TRANSP code is used in conjunction with TSC to incorporate neutral beam heating for ITER and to provide more detailed information on energetic particle distribution functions for stability analysis of TAE-like modes driven by energetic particles using NOVA-K.

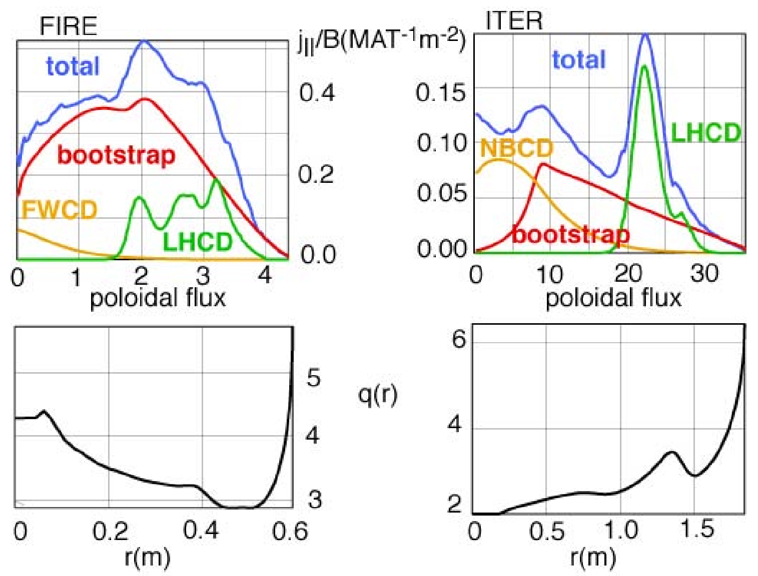

FIG. 4. The current- and q-profiles for the FIRE and ITER advanced scenarios developed using TSC and TRANSP.

The "steady-state" high-β AT configurations for FIRE rely on ICRF/FW on-axis current drive and LH (5 GHz) off-axis current drive. The ICRF/FW, 70-115 MHz, can provide about 150-200 kA of on-axis current by injecting 20 MW of power with the existing two-strap ion heating system. Upgrades to four strap antennas or an expanded antenna design would improve the CD efficiency 50-100%. Typical AT plasmas require only about 100-150 kA. The LH analysis was done in the Tokamak Simulation Code with LH ray-tracing package LSC [17], and in a stand-alone equilibrium using ACCOME [19]. Reverse power and trapped particle effects reduce the current-drive efficiency by about 25-30%. This leads to CD efficiencies of 0.16 A/W-m^2 at 6.5 T and 0.25 A/W-m^2 at 8.5 T. The typical parallel index spectrum is centered between 2.0-2.15 with a FWHM of 0.25, typical of the proposed C-Mod LH launcher. Off-axis LHCD in FIRE is critical for establishing and controlling the safety factor profile and the Lower Hybrid current drive experiments on C-Mod will be essential for assessing the feasibility of this approach.

Simulations of "steady-state" high-β AT discharges on FIRE with 100% non-inductive current composed of FW, LH, and bootstrap currents that are sustained for ≈ 4 τ_{CR} have been done with TSC (FIG. 4, 5). The other parameters are indicated in Table I. This is accomplished by programming the heating/CD sources so that the inductive contribution to the plasma current is reduced to zero by the end of the ramp up, so the safety factor profile has no significant change during the flattop burn phase. This scenario approaches the ARIES-RS power plant scenario described in Table I.

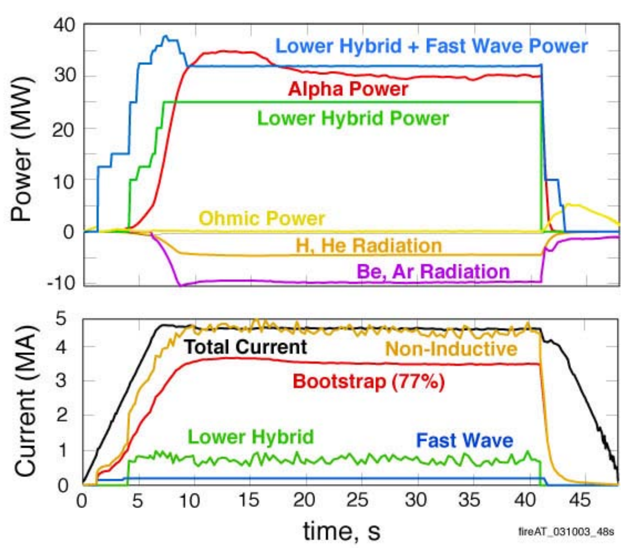

FIG. 5. "Steady-State" high-β AT in FIRE sustained for ≈ 4 τ_{CR}.

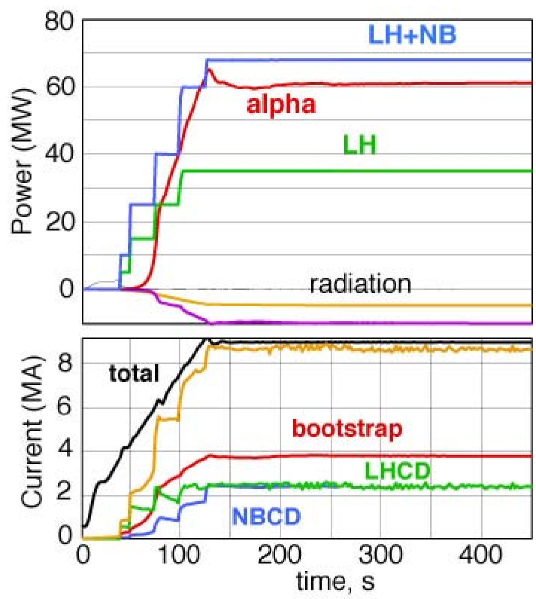

FIG. 6. Evolution of steady-state ITER advanced scenario discharge using TSC.

Studies of ITER steady-state advanced scenarios are under way using TSC and TRANSP. The first scenarios analyzed had ≈ 100% non-inductive drive with 33 MW of NINB for on-axis current drive and 35 MW of lower hybrid for off-axis current drive (FIG. 4, 6). This hybrid-like scenario with $\beta_N = 2.5$, $f_{bs} = 44\%$, Q = 5, $P_f = 350$ MW for H98(y, 2) = 1.6 is similar to the AT scenario 4 developed by the ITER group described in Table I. Additional optimization of the plasma startup and the mix of current drive among NINB, ICFW and LH is needed for this scenario to produce a q profile that is flatter or even slightly reversed.

7. Energetic Particle Effects in ITER and FIRE

Instabilities driven by the gradient of the energetic particle pressure, such as fishbones and toroidicity induced Alfvén Eigen-modes (TAEs) are potential threats to fusion alpha particle confinement in a fusion reactor. A detailed global analysis [18] of different branches of Alfvén-Eigen modes using the NOVA-K hybrid code including TAEs showed that they are stable in Q = 10 FIRE ELMy H-mode plasmas with central plasma temperature $T_0 \approx 12$ keV. A new study of FIRE AT plasmas shows weak multiple instabilities of TAE modes (FIG. 7). Several unstable modes have been found with n from 6 to 8. This is different from the previous study due to higher q-profile as the instability drive is proportional to q^2. The modes are global and mostly localized near the q_{min} minor radius. The reversed q-profile has strong negative shear within the q_{min} surface. This prevents the localization of the RSAE modes (Alfvén cascades). In the case of lower shear, we would expect even stronger instabilities of RSAE modes.

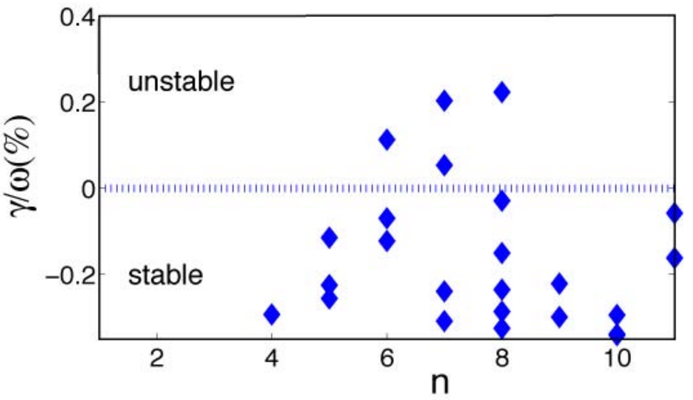

FIG. 7. Toroidal mode number dependence of the growth rates of alpha particle driven TAEs in FIRE AT plasma.

The injection of 1 MeV neutral beams into ITER introduces sufficient energetic fast ions to destabilize TAE modes [18]. Indeed, the ratios of the Alfvén velocity to injection velocity of beam ions and to the alpha particle birth velocity are very similar ~<2. The linear universal instability drive of the beam is comparable to the alpha-particle drive because the phase space density at the particle-wave resonance for a given β is larger for a beam distribution than for an isotropic distribution. The results of the NOVA-K stability analysis for the ITER Hybrid-like mode case are shown in FIG. 8. We see that by neglecting the beam drive the system

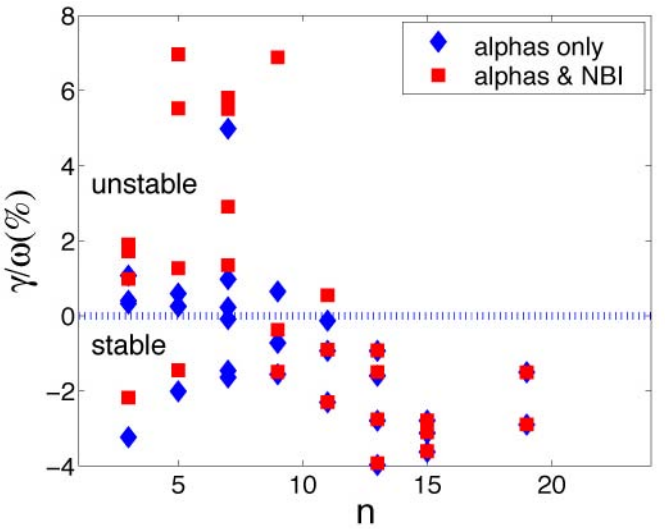

FIG. 8. Toroidal mode number dependence of the TAE growth rates for cases with the drive from alpha particles only (squares) and with drive from both the NBI ions and alpha particles (diamonds) in ITER Hybrid-like mode.

would be mildly unstable for the plasma temperature ~20 keV. With the inclusion of the beam drive in the NOVA-K analysis, we find that the system is strongly unstable. As expected the unstable mode numbers are shifted towards higher n's in ITER. In both ITER and FIRE cases, the number of unstable modes is expected to be similar in nominal regimes, so that fast-ion driven transport may be studied in a reactor relevant case of multiple mode instabilities.

8. Plasma Facing Components and First Wall Considerations for ITER and FIRE

All of the power plant designs in Table I have chosen refractory materials (W) for their divertor targets to maintain low in-vessel tritium inventory [19] while absorbing high power density exhaust. Tungsten bush targets developed during the U. S. ITER EDA, adopted by FIRE [20], have been tested successfully up to 25 MW/m^{-2} well beyond static operating conditions. Type I ELMs would be a life limiting process for the FIRE tungsten divertor [16]. The existing experience [21] on ELMs suggests that double-null operation, high triangularity, and high edge density would reduce the size of ELMs with a transition to Type II ELMs. Continued R&D is needed to develop ELM mitigation techniques for FIRE and ITER.

The first wall designs of ITER and FIRE are similar as shown in FIG. 9. Both have Be coated copper tiles as the plasma facing surface with water cooling before attachment to the stainless steel vacuum vessel. The FIRE copper tiles are capable of absorbing 1 MWm^{-2} for ~40s before the Be surface reaches 600 °C. The tiles are cooled by contact with a water cooled Cu cladding that cool the tiles to ambient temperature for the next pulse. The ITER design has the water cooling in closer thermal contact with the Be front surface and is capable of steady-state operation at 1 MWm^{-2}. These configurations would provide a good test of whether Be would be suitable as a first wall material in a power plant. Future work in this area would look at the possibility of enhancing the thermal transfer properties of the FIRE tile-cladding interface and increasing the water cooling to extend the first wall capability to near steady-state.

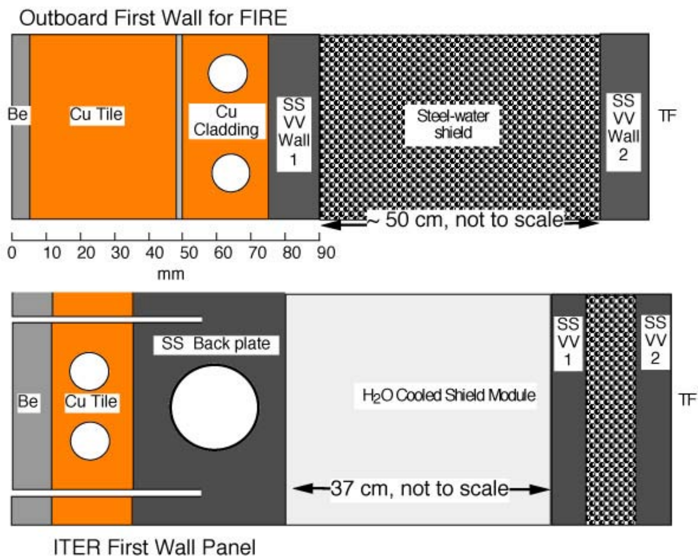

FIG. 9. Comparison of ITER and FIRE first wall Configurations.

9. Acknowledgments

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