Advanced Tokamak Regimes in the Fusion Ignition Research Experiment (FIRE)

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Introduction The ARIES fusion power plant studies [1,2] have determined that an economically competitive tokamak fusion power plant must have high fusion gain (Q > 25), high fusion power density (~ 4 MWm^{-3}) and steady-state plasma operation. These plasma requirements could be satisfied using an advanced tokamak scenario ($\beta_N \sim 5$ and $f_{bs} \approx 90\%$) based on a reversed shear configuration [3,4]. A U.S. national team is designing a burning plasma experimental device, known as the Fusion Ignition Research Experiment (FIRE), whose primary mission is to attain, explore, understand and optimize magnetically confined fusion-dominated plasmas [5,6]. The theme of the FIRE project is to emphasize testing the physics of an advanced tokamak fusion reactor, while invoking only those technology issues needed to accomplish the burning plasma physics mission. The strategy is to use low cost compact cryogenically cooled Cu coils at reactor-relevant magnetic fields to achieve quasi-stationary reactor-relevant plasma conditions.

General Requirements for a Burning Plasma Experiment, FIRE. In its first phase of operation FIRE would operate using quasi-stationary H-Mode plasmas with an alpha heating fraction of $f_\alpha = P_{alpha}/P_{heating} \geq 2/3$ or $Q = P_{fusion}/P_{ext\text{-}heating} \geq 10$. FIRE is also being designed to explore high-β high-bootstrap-current ARIES-like advanced tokamak regimes near steady-state with Q = 5 to 10 in a later phase. The duration of controlled burn is a very important parameter for burning plasma experiments, and should be specified in terms of the natural plasma time scales such as: τ_E the energy confinement time, τ_{He} the He ash confinement time and τ_{CR} the plasma current redistribution time. The duration of controlled burn in FIRE H-Mode (AT-mode) is: > 20(40) τ_E for pressure profile evolution, > 4(8) τ_{He} for alpha ash transport and burn control, and ≈ 2(5) τ_{CR} for plasma current profile evolution. These durations are sufficient to explore the physics of a burning plasma.

FIRE Configuration and General Parameters The FIRE configuration and aspect ratio is similar to that of other advanced tokamak designs such as TPX [7], KSTAR [8] and JT60-SC[9] and the advanced power plant designs ARIES-AT [1] and ASSTR [10]. The key advanced tokamak features of FIRE are: segmented central solenoid for flexibility and strong plasma shaping, double-null pumped divertors, low toroidal field ripple (< 0.3%), internal control coils and space for resistive wall stabilization capabilities. The reference design point for the FIRE-AT mode is compared with ARIES-RS in Table I. Auxiliary heating would be provided by 20 MW of ion cyclotron (ICRF) in the frequency range of 75-115 MHz utilizing 4 ports. The primary heating modes will be He^3 minority and second harmonic tritium for H-mode operation near 10T, or H-minority, second harmonic deuterium and direct electron heating for operation near 6.5 T. Lower hybrid current drive (LHCD) of 20 - 30 MW is being proposed as an upgrade for off-axis current drive and plasma heating in the advanced tokamak (AT) mode, and NTM control in the H-mode. ECCD is being evaluated for NTM control in the AT mode.

Table I. Advanced Tokamak Parameters

	FIRE-AT	**ARIES-RS**
R (m), a (m)	2.14, 0.595	5.52, 1.38
κ_x , κ_a , κ_{95}	2.0, 1.85,1.82	1.9, - ,1.70
δ_x , δ_{95}	0.7, 0.55	0.77, 0.5
Div. Config., material	DN, W	DN, W
(P_{loss})/R (MW/m)	20	80
$B_t(R_o)$ (T), I_p (MA)	6.5, 4.5	8, 11.3
q(0),q_{min}, q_{95}	4, 2.7, 4.0	2.8, 2.49, 3.5
β_t(%), β_N , β_p	4, 4.1, 2.15	5, 4.8, 2.29
f_{bs} (%)	77	88
n(0)/ $\langle n\rangle_{vol}$, T(0)/ $\langle T\rangle_{vol}$	1.5, 3.0	1.5, 1.7
n/n_{GW}, $\langle n\rangle_{vol}$ (10^{20} m^{-3})	0.85, 2.4	1.7, 2.1
$T_i(0)$, $T_e(0)$	14, 16	27, 28
Z_{eff}	2.3	1.7
H98(y,2)	1.7	1.4
τ_E, (s)	0.7	1.5
Burn Duration/τ_{cr} , s	3.2, 40	Steady-state
Q = $P_{fusion}/(P_{aux} + P_{OH})$	4.8	25
Fusion Power (MW)	140	2160
P_{fus}/Vol (MWm^{-3})	5.5	6.2
Γ neutron (MWm^{-2})	1.7	4

Operation of FIRE in the Conventional H-Mode Regime: FIRE's combination of low density relative to the Greenwald density and high triangularity facilitates enhanced H-factors (≈ 1.1) relative to ITER98(y,2) scaling as described in previous papers[5, 6]. Simulations of the Elmy H-mode regime using the 1 1/2 - D Tokamak Simulation Code (TSC) [11], with the GLF23 transport model and n(0)/ $\langle n\rangle_{vol}$ = 1.2, indicates that FIRE can access the H-Mode and control high gain (Q ≈ 10) plasmas for a duration > 20 τ_E, > 4 τ_{He} and ≈ 2 τ_{CR} [5,6]. This quasi-stationary burn phase is sufficiently long to allow the study of plasma profile evolution due to alpha heating, accumulation of alpha ash, and test of techniques for burn control.

Advanced Tokamak Operating Modes in FIRE: Systems analysis with zero-dimensional physics models and engineering constraints shows that FIRE can access a large advanced tokamak operating space assuming enhanced confinement above ITER98(y,2) can be achieved with reversed shear[12]. A wide range of plasmas have been examined with 1.25 ≤ n(0)/⟨n⟩ ≤ 2, 6T ≤ B_t ≤ 8.5T, 3.3 ≤ q_{95} ≤ 5, 0.3 ≤ n/n_{Gr} ≤ 1, and 2 ≤ β_N ≤ 5. In addition, the impurity concentrations are varied over 1 to 3% for Be and 0.0 to 0.4% for Ar, allowing higher radiated power fractions for more optimum distribution of the exhaust power. Viable solutions must be within the limits set by the first wall heat flux (<1 MWm^{-2}) including a peaking factor of 2, particle power to the outboard divertor (<28 MW), and the radiated power load in the divertor and baffle (<8 MWm^{-2}). Increasing the radiated power in the divertor reduces the particle heat load and expands the operating space significantly. The toroidal field is varied to optimize the size of the operating space, combining low β_N with high B_t, and vice versa. The combination of high n/n_{Gr} and higher density peaking provides the lowest H98 factors required for power balance. The operating space can also be expanded to higher power and β_N 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 flattop burn times for these plasmas are also limited by the nuclear heating in the vacuum vessel or TF coil heating. 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 6 τ_{CR} [6]. These results show that β_N and bootstrap fractions approaching those needed to test ARIES-like AT modes can be accessed as shown in Fig 1.

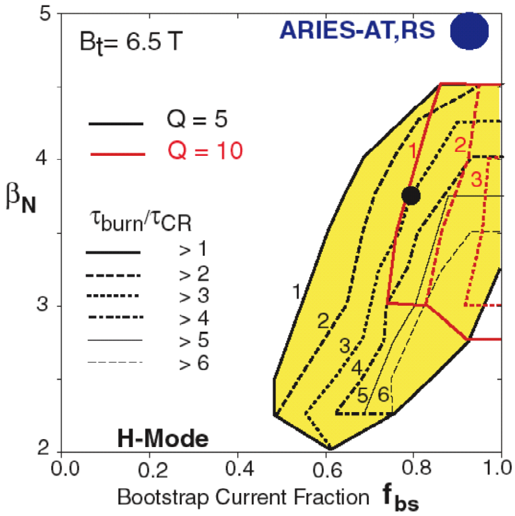

Fig. 1. FIRE AT mode Operating Space

Bootstrap consistent equilibrium and stability analysis show that the high-n ballooning limit for typical plasmas consistent with external current drive is $\beta_N < 4.7$, and with no wall the ideal MHD β_N limits for n=1, 2 and 3 are 2.7, 4.0, and 4.5, respectively. Calculations with VALEN [13] show that feedback coils, located near the front face the shield plug in every other mid-plane port, could stabilize the n=1 RWM mode up to $\beta_N = 4.2$. 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 [18]. The RWM coils may also be useful for ultra fast vertical position control, and in combination with the neutral stability point of the DN divertor lead to avoidance of vertical displacement events (VDE). The plasma configurations targeted have safety factor values above 2.0 everywhere, so that the (5,2) and (3,1) are the lowest order NTM's of interest. ECCD from the low field side (LFS) at toroidal fields of 6-8 T would require frequencies of 147-197 GHz, which is close to the range of achieved values targeted by ITER R&D. Examination of this is continuing to determine the powers required at high β and for higher order islands, the launching geometry and accessibility, and the use of Ohkawa current drive to achieve high CD efficiencies on the LFS.

The "steady-state" high-β AT configurations in FIRE rely on ICRF/FW on-axis current drive and LH off-axis current drive. The FW on axis can provide up to 400 kA of current by injecting 20 MW of power with the existing two strap ion heating system. Upgrades to four strap antennas would improve the CD efficiency. Typical AT plasmas require only half this current. The LH analysis was done in the Tokamak Simulation Code with LH ray tracing package LSC [12], and in a stand-alone equilibrium using ACCOME. Reverse power and trapped particle effects reduce the current drive efficiency giving 1.3 MA of current at 6.5 T for 30 MW of injected power and typical AT parameters (Table I). 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 CD in FIRE is critical for establishing and controlling the safety factor profile and the experience on C-Mod will strengthen the basis for

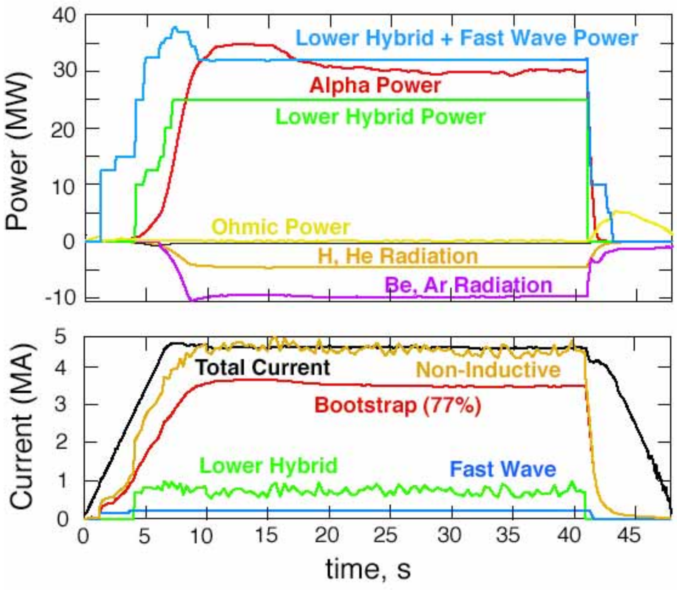

Fig. 2. "Steady-State" high-β AT in FIRE sustained for > 3 τ_{CR}. Parameters are given in Table I.

FIRE's projections.
Simulations of "steady-state" high-β AT discharges with 100% non-inductive current composed of FW, LH, and bootstrap currents that are sustained for > 3 τ_{CR} has been done with TSC (Fig. 2). 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. The safety factor profile has no significant change during the flattop burn phase of more than 3 τ_{CR}.

Issues needing R&D: The scaling of energy and particle confinement in these AT regimes is needed to predict the fusion gain and estimate the effects of alpha ash accumulation. Existing confinement models need to be benchmarked with regard to systematic scans of density and plasma triangularity to check burning plasma performance calculations. Experiments underway at DIII-D with internal RWM coils will refine our models for RWM stabilization. Work is needed to improve modeling of LHCD and ECCD including effects of particle trapping, reverse CD lobe interaction on edge bootstrap current and incorporation of Ohkawa CD. The development of a self consistent edge plasma and divertor model that can address both W and C divertor targets, and the incorporation of this model into the core plasma transport model. Determining the effect of high triangularity and double null on confinement, β-limits, Elms, and disruptions will help optimize the design of FIRE.

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