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Abstract. Experiments are needed to test and extend present understanding of confinement, macroscopic stability, alpha-driven instabilities, and particle/power exhaust in plasmas dominated by alpha heating. A key issue is to what extent pressure profile evolution driven by strong alpha heating will act to self-organize advanced configurations with large bootstrap current fractions and internal transport barriers. A design study of a Fusion Ignition Research Experiment (FIRE) is underway to assess near term opportunities for advancing the scientific understanding of self-heated fusion plasmas. The emphasis is on understanding the behavior of fusion plasmas dominated by alpha heating ($Q \geq 5$) that are sustained for durations comparable to the characteristic plasma time scales ($\geq 20\ \tau_E$ and $\sim \tau_{skin}$, where τ_{skin} is the time for the plasma current profile to redistribute at fixed current). The programmatic mission of FIRE is to attain, explore, understand and optimize alpha-dominated plasmas to provide knowledge for the design of attractive magnetic fusion energy systems. The programmatic strategy is to access the alpha-heating-dominated regime with confidence using the present advanced tokamak data base (e.g., Elmy-H-mode, ≤ 0.75 Greenwald density) while maintaining the flexibility for accessing and exploring other advanced tokamak modes (e. g., reversed shear, pellet enhanced performance) at lower magnetic fields and fusion power for longer durations in later stages of the experimental program. A major goal is to develop a design concept that could meet these physics objectives with a construction cost in the range of $1B.

1. Introduction

FIRE is envisioned as an extension of the existing advanced tokamak program leading to an attractive magnetic fusion reactor. The configuration chosen for FIRE is similar to that of ARIES-RS [1], namely a highly shaped plasma, with double-null divertor and aspect ratio ≈ 4. The FIRE design activities have focused on the physics and engineering evaluation of a compact, high-field tokamak with the parameters shown in Table I. The key "advanced tokamak" features are: strong plasma shaping, double null poloidal divertors, low toroidal field ripple (< 0.34%), internal control coils and space for wall stabilization capabilities. The magnets and structure are also

TABLE I. DESIGN GOALS FOR FIRE

R (m), a (m)	2.0, 0.525
κ_{95} , δ_{95}	≈1.8, ≈ 0.4
q_{95}	> 3
$B_t(R_o)$ (T)	10(12)*
Wmag TF (GJ)	3.7
I_p (MA)	6.44(7.7)*
flattop time (s)	~20(12)*
alpha heating fraction	>0.5
τ_E, τ_{skin} (s)	~ 0.6, ~ 13
Z_{eff} (3% Be + He (5 τ_E))	1.4
Fusion Power (MW)	100 - 200
ICRF Power (MW)	30
Tokamak Cost ($B)	~ 0.3
Project Cost ($B)	~1

* Higher Field Mode

capable of operation at $B_t(R_o)$ = 12T and I_p = 7.7 MA with a flat top time of 12 s at 200 MW of fusion power. Recently, an improved physics performance design point, FIRE*, has been identified with slightly lower aspect ratio (3.6 vs. 3.81), slightly increased triangularity (0.5 vs 0.4), R_o = 2m, a = 0.565m, $B_t(R_o)$ = 10T, and I_p = 7.7 MA with a flat top time ~ 15 to 20s at 125 MW of fusion power. The engineering feasibility of FIRE* is being evaluated.

2. Physics Performance Projections for FIRE

The physics design guidelines for FIRE are similar to those developed for ITER-FEAT [2] from analysis of the H-Mode confinement database DB03v5(9) [3];

- Confinement is assumed to scale as ITER98(y,2) with an H-factor determined by matching JET H-Mode data for FIRE-like conditions
- Operating density from $0.3 < n/n_{GW} < 0.85$ where n_{GW} is the Greenwald density
- H-mode power threshold is given by $P_{th} = 2.84\, n_{20}^{0.58} B^{0.82} R a^{0.81} M^{-1}$
- $\beta_N = \beta(\%) / (I_p/aB) < 2.5$
- Helium ash confinement $\tau_{He}^* = 5\ \tau_E$, with 3% Be impurities in the plasma core.

The operating regime for FIRE is well matched to the existing H-mode database from $0.3 < n/n_{GW} < 1.0$ as shown in Fig. 1. The performance of FIRE was projected by selecting JET data from DB03v5(9) [4] with parameters similar to FIRE, namely $\beta_N \geq 1.7$, $Z_{eff} < 2.0$, $\kappa > 1.7$ and $2.7 < q_{95} < 3.5$. The average density profile peaking, $n(0)/\langle n \rangle_V$ for these FIRE-like JET points was found to be 1.2. Modeling of standard pellet injection scenarios indicates peaking factors of 1.2. Guided high field side pellet launch will be incorporated with the goal of providing additional density profile peaking. The impurity level for these analyses was chosen to be 3% Be, where Be is the plasma facing material of the first wall, giving a Z_{eff} (imp) = 1.36. This assumption is consistent with the trend toward lower Z_{eff} as the density is increased in Alcator C-Mod [5]. The tungsten concentration in the plasma core must be below ~10^{-5} to avoid significant radiation loss. It is assumed that tungsten from the divertor plate and any other impurities present in the divertor do not migrate to the plasma core. The use of tungsten coated tiles in the baffle region of the divertor of ASDEX Upgrade [6] is encouraging with regard to maintaining acceptably low levels of tungsten in the plasma core.

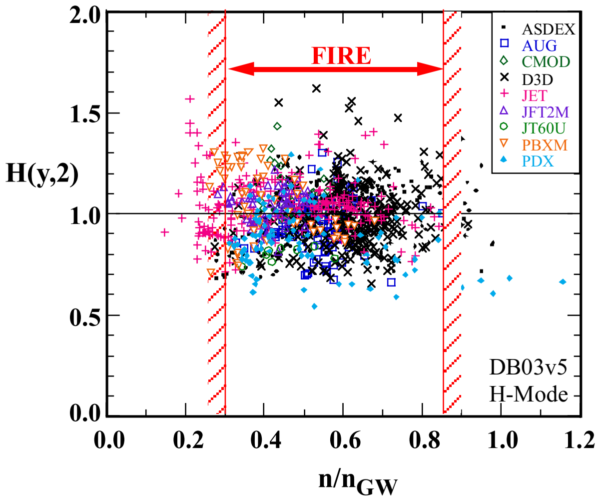

FIG. 1. H-Mode Database and FIRE

A 0-D power balance code was used to calculate the H factor required to achieve a given Q-value as shown in Fig. 2. The density profile was assumed to have $n(0)/\langle n \rangle_V = 1.2$ with 3% Be and self-consistent alpha ash accumulation. The required H-factor increases at high density where the plasma temperature decreases thereby reducing fusion reactivity. At low density the H-factor increases because τ_E must increase to maintain $n\tau_E$ approximately constant.

The uncertainty in the projections of FIRE performance can be expressed as a fraction of the

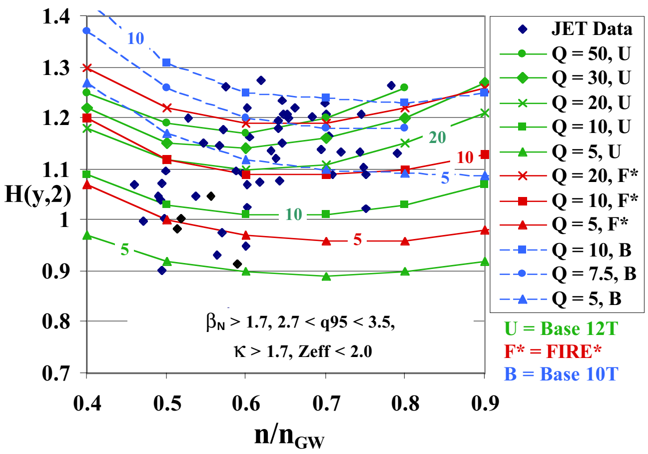

FIG. 2. FIRE Performance Projections

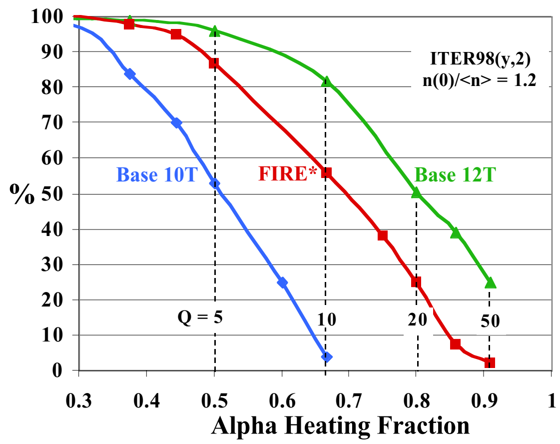

FIG. 3. Per Cent of Projected Points achieving a specific alpha heating fraction

FIRE-like JET data points that would achieve a specific alpha-heating fraction $f_\alpha = P_{alpha}/P_{heat} = Q/(Q + 5)$ as shown in Fig. 3. The baseline configuration is projected to achieve the FIRE Mission requirement of $f_\alpha \geq 0.5$ ($Q > 5$) for 53% of the FIRE-like JET points and would achieve $f_\alpha > 0.67$ ($Q \geq 10$) for 4% of the data set. Operation of FIRE at 12T/7.7 MA or FIRE* the candidate design point (7.7 MA) have significantly higher performance with > 90% of the FIRE-like JET data projected to achieve $Q \geq 5$, > 50% projected to achieve $Q \geq 10$ and > 25% projected to achieve $Q \geq 20$.

The operating space for FIRE can also be described using the PopCon plot as shown in Fig. 4 for cases similar to those described above with H98(y,2) = 1.1 and $n(0)/\langle n\rangle_V = 1.2$. The

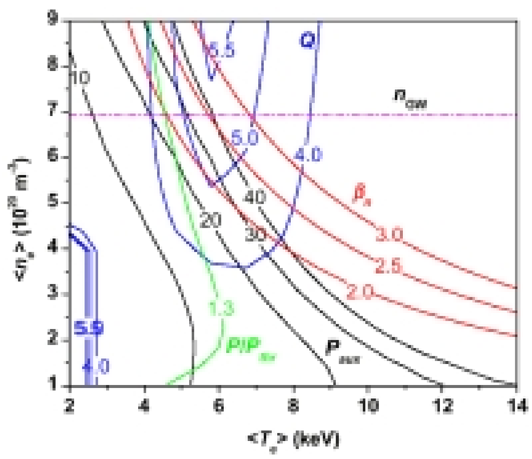

FIG. 4a Operating Space for FIRE (10T)

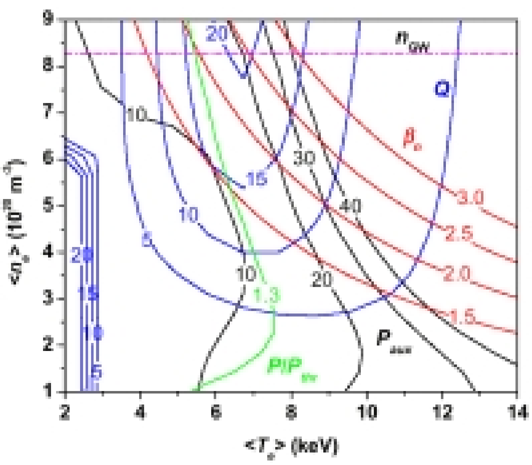

FIG. 4b. Operating Space for FIRE (12T)

transported power exceeds the H-mode threshold by a factor of at least 1.3 to the right of the $P/P_{thr} = 1.3$ curve. The baseline FIRE (10T) attains alpha-dominated conditions but only over a limited range. The operating range for FIRE* and FIRE (12T) are similar, and achieve alpha-dominated conditions, $Q > 5$, over a large range and $Q > 10$ over a significant range.

A Tokamak Simulation Code (TSC) simulation of this regime with H(y,2) = 1.03 and $n(0)/\langle n\rangle_V = 1.2$ indicates that FIRE can access the H-Mode and sustain alpha-dominated plasmas for $\approx 30\ \tau_E$, $> 6\ \tau_{He}$ and $\sim 1.5\ \tau_{skin}$ as shown in Fig. 5. In addition, sufficient time is provided for plasma startup and a controlled shutdown to avoid plasma disruptions. The burn phase can study plasma profile evolution, alpha ash accumulation and techniques for burn control and begin studies of plasma current evolution due to alpha heating.

A longer term goal of FIRE is to explore advanced tokamak regimes using pellet injection and current ramps to create reversed shear plasmas (e.g., PEP modes) for durations of 1 to 3 current redistribution times. This AT capability is expected to produce modestly enhanced confinement and beta as observed in present large tokamak experiments, and would provide a continuous transition from H-mode operation to advanced tokamak operation. An important feature of the FIRE cryogenic copper alloy magnets is that the pulse length increases rapidly as the field is reduced with flattops of ~ 40 s at 8 T and ~90 s at 6 T. If confinement and β are increased by 20 and 40% respectively, then the fields could be reduced by 20% and FIRE would have the capability to explore fusion-dominated plasmas for 40 s (~ $3\tau_{skin}$). Physics scenarios and engineering solutions for power handling are not yet developed for the longer pulse (~ 40 s) scenarios.

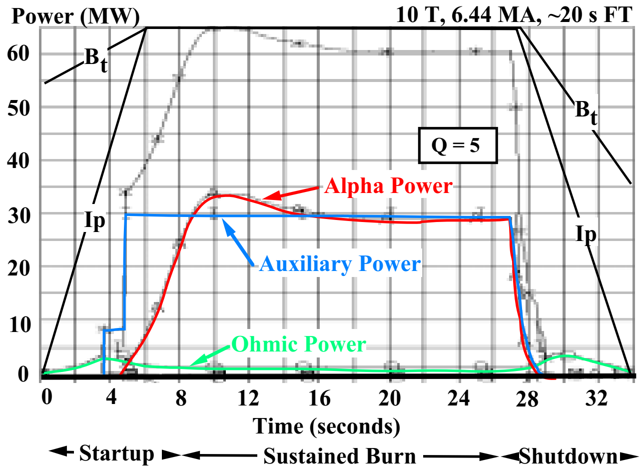

FIG. 5. Evolution of a fusion-dominated plasma.

3. Technology Considerations for FIRE

The baseline magnetic fields and pulse lengths can be provided with BeCu/OFHC toroidal field (TF) coils and OFHC poloidal field (PF) coils that are pre-cooled to 77 °K prior to the pulse and allowed to warm up to 373 °K at the end of the pulse [7]. The cross-section of FIRE is shown schematically in Fig. 6. The 16 TF coil system is wedged with a compression ring to resist de-wedging at the top and bottom of the inner TF leg. Shielding is added between the walls of a double wall vacuum vessel to reduce nuclear heating of the coils, limit insulation dose and allow hands-on maintenance outside the envelope of the TF coils within a few hours after a full power D-T shot. Large (1.3 m by 0.7 m) midplane ports provide access for heating, diagnostics and remote manipulators, while 32 angled ports provide access to the divertor regions for utilities and diagnostics. FIRE is being designed mechanically to accommodate 3,000 full field, full power pulses and 30,000 pulses at 2/3 field. The repetition time at full field and full pulse length will be < 3 hr, with shorter times at reduced field or pulse length. The fusion energy production of 5 TJ (similar to BPX) produces a lifetime neutron dose to the TF insulating material at the inboard midplane of ≈ 1.5 x 10^{10} Rads which is consistent with the polyimide insulation being considered.

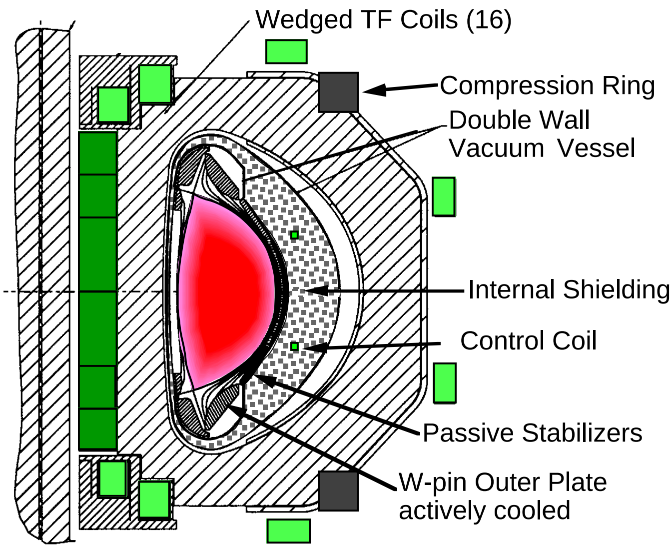

FIG. 6. FIRE Configuration

The power densities on the divertor plates are ~6 MWm^{-2} for detached operation and ~25 MWm^{-2} for attached operation [8]. Carbon is not allowed in the vessel due to tritium inventory build-up by co-deposition. The divertor plasma-facing components are tungsten "brush" targets mounted on copper backing plates, similar to a concept developed by the ITER R&D activity. The outer divertor plates and baffle are water-cooled, while the inner divertor targets and first wall are cooled by mechanical attachment to water-cooled copper plates inside the vacuum vessel. The first wall is comprised of Be plasma-sprayed onto copper tiles. The high neutron wall loading ($3 MWm^{-2}$) at full fusion power of 200 MW contributes significantly to the first wall and vacuum vessel heating. The water-cooled copper plates inside the vessel alleviate excess heating of the stainless steel vessel due to neutrons. Sixteen cryo-pumps – closely coupled to the divertor chambers, but behind sufficient neutron shielding – provide pumping (≥ 100 Pa m^3/s) for D-T and He ash during the pulse. Pellet injection scenarios will help minimize tritium throughput. The in-device tritium inventory will be determined primarily by the cycle time of the divertor cryo-pumps, and can range from < 2 g for regeneration overnight to ~20 g for monthly regeneration.

The confinement analysis based on ITER98(y,2) scaling indicates the importance of increasing the plasma current in FIRE to ~ 8 MA. This could be accomplished in the baseline design operated at 12 T for 12 s flattop. The design refinement of increasing the plasma shaping factor as in FIRE* is being evaluated along with the possibility of using only high conductivity (OFHC) copper in the TF coil. This lower strength material would require the addition of TF coil bucking on the central solenoid coils near the midplane. Initial results suggest that 11.5T could be produced with a flattop of ≈ 25 s using about 1/2 the electrical power required by the baseline BeCu TF coil. The limitation on burn time for both BeCu and OFHC TF coil designs is the power handling capability of plasma facing components and the vacuum vessel.

A number of other important physics and engineering issues remain to be addressed during the remainder of the design study including generic issues such as: mitigation and avoidance of disruptions and vertical displacement events, effects of neoclassical tearing modes, detached divertor operation with good confinement, and divertor/edge plasma modeling under high power conditions. The FIRE design study is a U. S. national activity managed through the Virtual Laboratory for Technology and is supported by DOE. PPPL work is supported by DOE Contract # DE-AC02-76CHO3073.

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