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FIRE, Exploring the Frontiers of Burning Plasma Science

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Abstract

The burning plasma regime will exhibit a number of complex phenomena that must be studied and understood if magnetic fusion is to be successful. The FIRE design study of a burning plasma experiment has the goal of developing a concept for an experimental facility to explore and understand the strong non-linear coupling among confinement, self-heating, MHD stability, edge physics and wave-particle interactions that is fundamental to fusion plasma behavior. This will require plasmas dominated by alpha heating ($Q \approx 10$) that are sustained for duration longer than characteristic plasma time scales. The FIRE pre-conceptual design activities, carried out by an U. S. national team, have been undertaken with the objective of finding the minimum size (cost) device to achieve the essential burning plasma science goals while building on the physics understanding of today. FIRE is envisioned as an extension of the existing advanced tokamak program leading to an attractive magnetic fusion reactor (e.g., ARIES-RS). FIRE activities have focused on the physics and engineering assessment of a compact, high-field tokamak with the capability of achieving $Q \approx 10$ in the Elmy H-mode for a duration of ~ 1.5 plasma current redistribution times (skin times) during an initial burning plasma science phase, and the flexibility to add advanced tokamak hardware (e.g., lower hybrid current drive and feedback stabilization) later. The configuration chosen for FIRE is similar to that of ARIES-RS, namely a highly shaped plasma ($\delta_x = 0.7$, $\kappa_x = 2$), with double-null divertor and aspect ratio ≈ 4. The key "advanced tokamak" features are: strong plasma shaping, double null poloidal divertors, low toroidal field ripple (< 0.3%), internal control coils and space for wall stabilization capabilities. Only metallic plasma facing components (tungsten divertor targets and Be coated first wall tiles) are used to reduce the in-vessel tritium inventory. The reference design point is $R_0 = 2.14$ m, $a = 0.595$ m, $B_t(R_0) = 10$ T, $I_p = 7.7$ MA with a flat top time of 20 s for 150 MW of fusion power.

Keywords:

fusion, burning plasma experiment, ignition, tokamak, alpha particles

1. Introduction

Magnetic fusion is technically ready to proceed to the next stage of fusion research: the study of burning plasmas dominated by self-heating due to the fusion process, the optimization of the magnetic configuration and the development of low-activation, neutron-resistant materials. A national team consisting of scientists and engineers from more than 15 institutions in the U.S. fusion community is designing an advanced tokamak

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device known the Fusion Ignition Research Experiment (FIRE). The primary mission of FIRE is: to attain, explore, understand and optimize magnetically-confined fusion-dominated plasmas [1,2]. This will require plasmas dominated by alpha heating ($Q \approx 10$) that are sustained for a duration longer than characteristic plasma time scales such as the energy confinement time (τ_E), the alpha ash confinement time (τ_{He}), and the current redistribution or skin time (τ_{skin}). The FIRE pre-conceptual design activities have been undertaken with the objective of finding the minimum size (cost) device to achieve the essential burning plasma science goals. The work is managed through the Virtual Laboratory for Technology and is funded by the U. S. Department of Energy. The FIRE experiment is envisioned as part of an international multi-machine strategy that consists of: burning plasma experiment(s), very long-pulse non-burning experiments in advanced configurations (advanced tokamak, advanced stellarator, etc,) and a high fluence fusion materials irradiation test facility [3,4]. These individual facilities could be in the ~$1B scale or less, and could be led by parties sited around the world. This distributed international program has greater flexibility, less technical and management risk than a single large facility such as ITER which is estimated to cost $5B.

2. General Physics Requirements for a Burning Plasma Experiment, FIRE

The first goal of FIRE is to carry out burning plasma experiments to address confinement, MHD stability, fast alpha physics and alpha heating and edge plasma issues expected in fusion reactor scale plasmas. For these experiments alpha heating must dominate the plasma dynamics, therefore f_α, the fraction of plasma heating due to alpha particles, must be greater than 50%. This in turn requires that the minimum $Q = P_{fusion}/P_{ext.\ heat} \geq 5$. The goal for the design is to achieve $Q \approx 10$, with ignition not precluded under slightly more optimistic physics.

FIRE is also being designed to study burning plasmas in advanced configurations in a later phase as an extension of the existing advanced tokamak program. For these experiments, it will be desirable to study regimes that are bootstrap current dominated, $f_{bs} \geq 50\%$ ($\beta_N \geq 2.6$) with the possibility of exploring f_{bs} up to 75% ($\beta_N \sim 3.6$). These regimes will require strong plasma shaping and stabilization of the $n = 1$ kink by a conducting first wall or feedback.

The pulse duration is a very important requirement for these experiments and should be specified in terms of the natural plasma time scales such as $\tau_E \approx 1$ s, $\tau_{He} \approx 5$ s and $\tau_{skin} \sim 10 - 15$ s. The goal for FIRE pulse duration of 20 s at 10 T and 30 s at 8.5 T is sufficient for >10 τ_E for pressure profile evolution, > 4 τ_{He} for alpha ash transport and burn control, and ~ 2 τ_{skin} for plasma current profile evolution in advanced regimes.

3. Optimization of a Burning Plasma Experiment

A systems study was undertaken to find the minimum size burning plasma to access the physics requirements discussed above. This study was specialized for inductively-driven tokamaks with toroidal field (TF) and poloidal field (PF) coils that are pre-cooled to LN_2 temperature and then heated adiabatically during the pulse. The systems code includes constraints for stress, resistive and nuclear heating of the coils and volt-sec requirements. The geometry can be chosen to have TF and PF coils unlinked as in FIRE or linked as in low aspect ratio or spherical tokamaks. The code optimizes the allocation of the space in the inner coil stack between the free standing ohmic solenoid and the wedged TF coil. The confinement is taken to be H-mode with ITER98 (y,2) scaling. For these studies, the plasma density had a small density peaking of $n(0)/\langle n\rangle = 1.2$, $n/n_{GW} \leq 0.75$ and 3% Be impurities due to sputtering of first wall. The systems code varied the major radius, R, and aspect ratio, A, with H(y,2) = 1.1, $\kappa_{95} = 1.8$, $q_{cyl} = 3.1$ and $P_{fusion} = 150$ MW to obtain plasmas with a $Q \sim 10$ and a 20 s burn time. For these constraints, the smallest size device to achieve the burning plasma requirements for a cryogenically-cooled inductively driven tokamak with unlinked TF/PF coils has a shallow minimum around $A \approx 3.6$, $B \approx 10$ T and $R \approx 2.1$ m as shown in Fig. 1. The normalized burn time measured in plasma current redistribution times, $\tau_J = \tau_{burn}/\tau_{skin}$, shown in Fig. 1 increases significantly as the aspect ratio is increased. The minimum aspect ratio that satisfies the physics requirement of 2 τ_J is $A \geq 3.4$. The advanced tokamak features of significant bootstrap current are also enhanced at higher aspect ratios. Indeed, the fusion power plant design studies based on advanced tokamak scenarios have all chosen $A = 4$ and relatively high magnetic field such as 8.2 T for ARIES-RS [5] and 11 T for ASSTR2 [6]. These considerations have lead to the choice of $A = 3.6$ for FIRE that is higher than the aspect ratios chosen for other burning plasma experiments (ITER-FEAT: $A = 3.1$ and IGNITOR: $A = 2.8$).

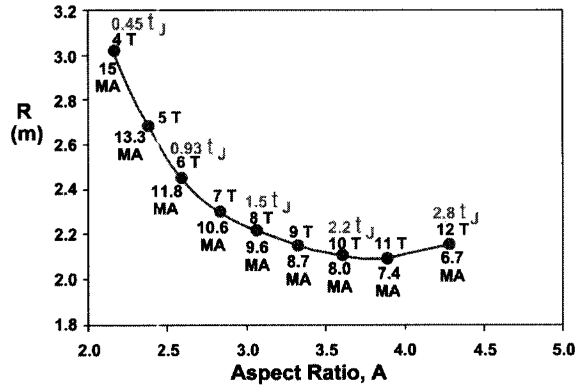

Fig. 1 Optimization of an inductively-driven copper coil burning plasma experiment.

4. FIRE Configuration and General Parameters

The FIRE engineering activities have focused on the physics and engineering assessment of a compact, copper-alloy conductor, high-field tokamak with the capability of achieving $Q \approx 10$ in the Elmy H-mode for a duration of ~ 1.5 plasma current redistribution times (skin times) during an initial burning plasma science phase, and the flexibility to add advanced tokamak hardware (e.g., lower hybrid current drive) later. The configuration chosen for FIRE is similar to that of advanced tokamak designs such as TPX [7], KSTAR [8] and ARIES-RS [5], namely a highly shaped plasma, with double-null divertor and aspect ratio ≈ 4. The key "advanced tokamak" features are: segmented central solenoid for flexibility and strong plasma shaping, double null poloidal divertors, low toroidal field ripple (< 0.3%), internal control coils and space for wall stabilization capabilities. The reference design point is $R_o = 2.14$ m, $a = 0.595$ m, $B_t(R_o) = 10$ T, $I_p = 7.7$ MA with a flat top time of 20 s for 150 MW of fusion power with the parameters and cross-section shown in Table 1 and Fig. 2.

5. Plasma Performance Projections for Elmy H-Mode Operation

The physics issues and physics design guidelines for operating modes and projecting burning plasma performance in FIRE are similar to those for ITER-FEAT. However, the natural operating regime for FIRE is better matched to the existing H-mode database, and FIRE can access the density range from $0.3 < n/n_{GW} < 1.0$ through a combination of pellet fueling and divertor pumping [1,2]. This flexibility is important for investigating the onset of alpha-driven modes at the lower densities and to optimize the edge plasma for confinement studies and optimal divertor operation. The performance of FIRE can be projected by selecting JET data 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 H(y, 2) and density profile peaking, $n(0)/\langle n \rangle_V$ for these data was found to be 1.1 and 1.2, respectively. This is consistent with the analysis of JET H-mode data presented by Cordey et al [9] that shows that the H factor is higher at higher plasma cross-section triangularity, and has a maximum at $n/n_{GW} \approx 0.6$. A 0-D power balance calculation was used to determine the Q-value in FIRE as a function of H-factor as shown in Fig. 3. The density profile was assumed to have $n(0)/\langle n \rangle_V =$

Table 1 FIRE, Parameters at $Q = 10$

R (m), a (m)	2.14, 0.595
κ_x, κ_{95}	2.0, 1.77
δ_x, δ_{95}	0.7, ≈ 0.4–0.55
q_{95}	> 3
$B_t(R_o)$ (T), I_p(MA)	10, 7.7
$Q = P_{fusion}/(P_{aux} + P_{OH})$	10
H98(y,2)	1.1
τ_E, (s)	1.04
β_N	1.81
P_{loss}/P_{LH}, †	1.3
Z_{eff} (3% Be + He (5 τ_E))*	1.4
$R\nabla\beta_\alpha$ (%)	3.8

† Margin for H-Mode Power Threshold
* He ash accumulated with $\tau_{He} = 5\ \tau_E$

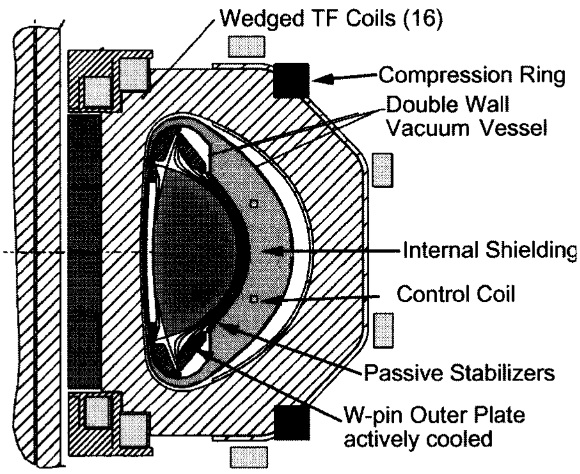

Fig. 2 FIRE Configuration

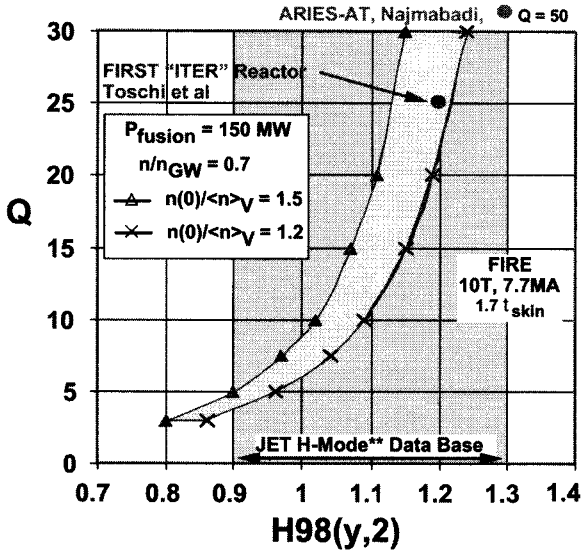

Fig. 3 Fusion Gain for FIRE

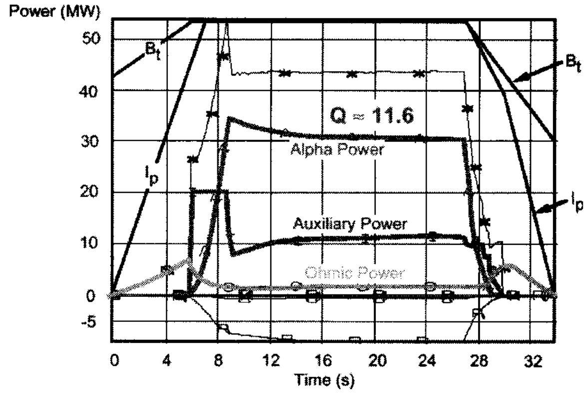

Fig. 4 Evolution of an H-mode plasma.

1.2 (× points) or 1.5 (Δ points) with 3% Be and self-consistent alpha ash accumulation. The slightly more peaked profiles are consistent with density profiles observed during neutral beam heating in DIII-D H-modes and more recently on JET. FIRE would aim to achieve this modestly peaked profiles using pellet fueling in ICRF heated discharges. On this basis, FIRE would be expected to achieve $Q \geq 10$ for JET-like H-modes. Typical plasma parameters for a $Q = 10$ plasma are given in Table 1. Physics based models using marginal stability transport models such as GLF23 also predict Q values around 10. These models depend sensitively on the value of the H-mode temperature pedestal which is projected to be higher for plasmas with high triangularity, and pedestal density low relative to the Greenwald density. A next step experiment, such as FIRE, would provide a strong test of these models and improve their capability for predicting reactor plasma performance. A 1 1/2 -D Tokamak Simulation Code (TSC) [10] simulation of this regime with H(y,2) = 1.1 and $n(0)/<n>_V = 1.2$ indicates that FIRE can access the H-Mode and sustain alpha-dominated plasmas for a duration > 20 τ_E, > 4 τ_{He} and ~ 2 τ_{skin} as shown in Fig. 4. In addition, time is provided for plasma startup and controlled shutdown to avoid plasma disruptions. The burn phase is sufficiently long to allow investigation of plasma profile evolution due to alpha heating, accumulation of alpha ash, testing of techniques for controlling fusion power output and the initial studies of plasma current evolution due to alpha heating.

Neoclassical tearing modes (NTMs) pose a potential threat to the achievement of the required β_N values in tokamak burning-plasma experiments such as FIRE, since the polarization-current stabilization model predicts that the critical β_N for their onset scales like ρ_i^*. The value of ρ_i^* in FIRE is intermediate between that in present-day tokamaks such as JET and that in ITER-FEAT, and NTMs might arise in FIRE for the reference values of β_N (1.5 – 2.0). For this reason, NTM suppression by feedback-modulated LHCD is being evaluated. Calculations with a LHCD model in the TSC code have shown that a 10 MW 5.6 GHz system with 50/50 on/off modulation should be capable of suppressing the $m/n = 3/2$ mode up to $\beta_N \approx 2.0$.

6. Advanced Modes in FIRE

The standard regime in FIRE without wall stabilization is limited by kink instabilities to $\beta_N < 3$ and bootstrap factions, $f_{bs} \leq 50\%$. Exploitation of advanced tokamak regimes requires stabilization of the low n kinks as recently demonstrated on DIII-D [11,12]. If the $n = 1$ kink could be stabilized by a conducting wall or feedback in FIRE, then advanced tokamak regimes with $\beta_N \leq 3.6$ and $f_{bs} \leq 75\%$ are possible. TSC has been used to determine the current drive, plasma heating power and energy confinement required to dynamically access these advanced regimes in a burning plasma. The example shown in Fig. 5 has $B = 8.5$ T, $I_p = 5.5$ MA, which confines alphas very well, and the coils would allow burn times up to 35 s. LHCD is calculated self-consistently using LSC for density profiles with $n(0)/<n> \approx 1.5$. This quasi-steady reversed shear discharge attained $\beta_N = 3.0$, $f_{bs} = 64\%$ and $Q = 7.5$ for moderately enhanced confinement of H(y,2) = 1.6, and was 100% non-inductively driven after 11 s. Exploitation of these

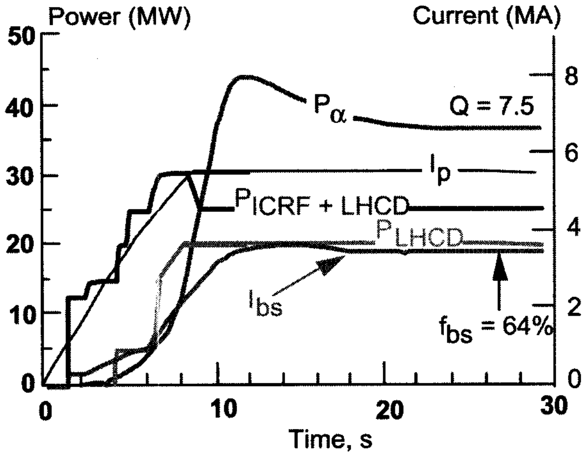

Fig. 5 Advanced Mode in FIRE

regimes will require stabilization of the n = 1 kink, either by feedback from coils mounted in the first wall of the FIRE vacuum vessel or by a method to rotate the FIRE plasma, and an improved long-pulse capability for the FIRE internal components. 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, ~ 90 s at 6 T and 240 s at 4 T. The primary limitation to exploiting this long pulse magnet capability is the generic magnetic fusion problem of handling the plasma exhaust power under reactor relevant conditions.

7. Engineering Description of FIRE

The baseline magnetic fields and pulse lengths can be provided by wedged 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 [13]. 3-D finite-element stress analyses including electromagnetic, and thermal stress due to ohmic and nuclear heating have shown that this design has an additional margin of 30% beyond the usual engineering allowable stress requirements. 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, like the previous BPX design, is being designed mechanically to accommodate 3,000 full field, full power pulses and 30,000 pulses at 2/3 field with a total fusion energy production of 5.5 TJ. The repetition time at full field and full pulse length will be < 3 hr, with much shorter times at reduced field or pulse length. The FIRE experimental program will utilize a variety of pulse lengths and field levels with the pulses per year (~2,000) and duty cycle (~0.001) similar to existing large experiments.

FIRE will provide reactor relevant experience for divertor and first wall power handling since the anticipated thermal power densities on the divertor plates of ~5 MWm^{-2} for detached operation and ~20 MWm^{-2} for attached operation exceed present experiments and approach those anticipated for ARIES-RS. The D-T experiments on TFTR and JET observed tritium retention fractions of ≈ 15 to 30% with carbon limiters and divertor plates. This large retention fraction would have a significant impact on the operational schedule of a burning plasma experiment and reactor, and would require periodic shutdowns to remove the excess tritium inventory. FIRE would use only metallic materials for plasma facing components, and carbon would 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 and with a time constant of a few seconds come into steady-state equilibrium during the pulse [14]. The first wall is comprised of Be plasma-sprayed onto copper tiles. The neutron wall loading in FIRE is ~2 MWm^{-2} and produces significant nuclear heating of the first wall and vacuum vessel during the 20 s pulse. The inner divertor targets and first wall are cooled by mechanical attachment to water-cooled copper plates inside the vacuum vessel and have a thermal time constant of about 40 s. This is the primary limitation to exploring longer high power pulses on FIRE. 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 with high-field-side launch capability will reduce tritium throughput, and enhance fusion performance. 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 ~10 g for weekly regeneration. The tritium usage per shot and inventory is comparable to that of TFTR and therefore will not require a large step in complexity beyond previous US fusion program experience in regulatory approvals for tritium shipping and handling.

The construction cost of the tokamak subsystem (magnets, divertor, plasma facing components and mechanical structure) has been estimated by U. S. industry to be ≈ \$350 M (FY02US) including \$71 M of contingency. Another ≈ \$850 M would be required for auxiliary heating, startup diagnostics, power supplies and buildings to put the project at a new site. If an existing site is found with about \$200M of site credits then the overall project cost would be about \$1B (FY02US). The construction schedule for FIRE would be 6 to 8 years depending on the annual funding available. The first year of operation would utilize hydrogen plasmas, followed by two years of deuterium plasmas with tritium operation beginning in the fourth year. The total experimental program would last about 15 years.

8. Assessment of FIRE

FIRE is a natural extension of the existing state of the art tokamaks, and is based on the extensive international H-mode data base for projecting performance to the burning plasma regime. Due to the high magnetic field, the extrapolation required to attain $Q \approx 10$ is a modest factor of 3 in terms of the dimensionless confinement time ($\omega_c \tau_E \sim B\tau_E$). While this reduces the uncertainty in attaining a burning plasma, it does not extend some plasma parameters (e.g., ρ^*) to full reactor values. The MHD stability characteristics of FIRE, with $q_{95} \approx 3.1$ and $\beta_N \approx 1.8$ for initial burning plasma experiments, are similar to the standard MHD regimes in existing tokamaks and will explore the synergistic effects of energetic alphas and MHD modes such as sawteeth and TAE modes. Operation at $\beta_N \approx 3$ or higher in later phases would begin to explore the important areas of neoclassical tearing modes (NTM) and resistive wall modes (RWM). Lower hybrid current drive and feedback stabilization is being evaluated as an experimental tools to investigate the control of NTMs and RWMs. Divertor pumping and pellet fueling will allow FIRE to vary the density, hence the TAE driving terms $R\nabla\beta_\alpha$, by a factor of three providing a good test bed for exploring the instability boundary for TAE modes and determining the transport of energetic alpha particles due to multiple overlapping TAE modes.

The double null divertor configuration produces the strongest plasma shaping which is critical for resolving and exploiting a number of physics effects related to confinement and MHD stability. The double null divertor may also significantly reduce the frequency and intensity of vertical displacement disruptions which is a critical issue for the feasibility of a tokamak based reactor. The high power density in FIRE poses a significant challenge for the divertor and first wall designs, but this is a generic issue for magnetic fusion. The success of FIRE in this area would provide yield important benefits for technology development for future fusion devices.

A critical issue for all next step experiments is to supply auxiliary heating power at high power densities to a fusion plasma. FIRE proposes to use ICRF heating which has been demonstrated on existing experiments but the high power densities and neutron wall loading present in FIRE will require significant plasma technology R&D. This R&D will be needed if ICRF is to be used in a fusion application.

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