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Abstract. The Advanced Reactor Innovation Evaluation Studies (ARIES) have identified the key physics and technical issues that must be resolved before attractive fusion reactors can be designed and built. The Fusion Ignition Research Experiment (FIRE) design study has been undertaken to define the lowest cost facility to address the key burning plasma and advanced tokamak physics issues identified in the ARIES studies. The configuration chosen for FIRE is similar to that of ARIES-AT, a steady-state advanced tokamak reactor based on a high-β and high-bootstrap-current operating regime. The key advanced tokamak features of FIRE are: strong plasma shaping, double-null pumping divertors, low toroidal field ripple (< 0.3%), internal control coils and space for wall stabilization capabilities. The initial burning plasma experimental phase will utilize the Elmy H-Mode regime with $Q \approx 10$ sustained under quasi-stationary conditions for ~ 2 plasma current redistribution times (τ_{cr}). A longer term goal of FIRE is to explore "steady-state" high-β advanced tokamak regimes with high bootstrap fractions (f_{BS}) ≈ 75% at $\beta_N \approx 4$ and moderate fusion gain ($Q \approx 5$ to 10) under quasi-steady-state conditions for ≈ 3 τ_{cr}. FIRE activities have focused on the physics and engineering assessment of a compact, high-field, cryogenic-copper-coil tokamak with: $R_o = 2.14$ m, $a = 0.595$ m, $B_t (R_o) = 6$ to 10T, $I_p = 4.5$ to 7.7 MA with a flat top time of 40 to 20 s for 150 MW of fusion power. FIRE will utilize only metal plasma facing components; Be coated tiles for the first wall and W brush divertors to reduce tritium retention as required for fusion reactors. FIRE will be able to test divertor and plasma facing components under reactor relevant power densities since the fusion power density of 6 MWm^{-3} and neutron wall loading of 2.3 MWm^{-2} approach those expected in a reactor.

1. Introduction

Magnetic fusion is technically ready to proceed to the next stage of fusion research, the study of burning plasmas dominated by the fusion process. A team consisting of scientists and engineers from more than 15 institutions in the U.S. fusion community is designing an advanced tokamak device, known as the Fusion Ignition Research Experiment (FIRE), whose primary mission is to attain, explore, understand and optimize magnetically confined fusion-dominated plasmas [1,2,3]. The FIRE experiment is envisioned as part of a diversified international portfolio that consists of burning plasma experiment(s), very long-pulse non-burning experiments in advanced configurations (advanced tokamak, advanced stellarator, etc,), a strong fusion plasma simulation capability and fusion technology development facilities including a high fluence fusion materials irradiation facility and component test facility. These individual facilities could be in the ~$1B range, and could be located at sites around the world with activities led by the host party.

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

The first goal of FIRE is to address confinement, MHD stability, fast alpha physics and alpha heating and edge plasma issues expected in fusion reactor scale burning plasmas. This would be accomplished using quasi-stationary H-Mode plasmas with an alpha heating fraction of $f_\alpha = P_{alpha}/P_{heating} \geq 2/3$ or $Q = P_{fusion}/P_{ext\text{-}heating} \geq 10$. Ignition would not be 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. In this phase, FIRE would explore near steady-state high-β high-bootstrap-current advanced tokamak (reversed shear) plasmas with Q = 5 to 10.

The duration of controlled burn is a very important requirement 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 goal for duration of controlled burn in FIRE 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 in the H-Mode (AT-mode) operating regimes.

3. Optimization of a Burning Plasma Experiment in Conventional H-Mode Regime

A systems study was undertaken to find the minimum size burning plasma needed to satisfy the physics requirements discussed above. This study was specialized for inductively-driven tokamaks with unlinked TF and PF coils that are pre-cooled to LN_2 temperature, and then heated adiabatically during the pulse. The Burning Plasma Systems Code [4] includes constraints for stress, resistive and nuclear heating of the coils and volt-sec requirements. 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 was taken to be H-mode with ITER98(y,2) scaling [5]. For these studies, the plasma density profile had peaking of $n(0)/\langle n\rangle = 1 + \alpha_n = 1.2$, $n/n_{GW} \leq 0.75$, and temperature profile peaking of $T(0)/\langle T\rangle = 1 + \alpha_T = 2.0$ where α_n and α_T are the exponents of parabolic squared profile functions for density and temperature. The plasma impurities consist of helium ash with the $\tau_{He} = 5\ \tau_E$ and 3% Be impurities. 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 Q = 10 and 20 s burn time. For these constraints the smallest size device to achieve the burning plasma requirements has a shallow minimum around A ≈ 3.6, B ≈ 10 T and R ≈ 2.1 m as shown in Fig. 1. The normalized burn time measured in plasma current redistribution times, $\tau_J = \tau_{burn}/\tau_{CR}$, increases significantly as the aspect ratio is increased as shown in Fig. 1. The minimum aspect ratio that satisfies the physics requirement of $2\tau_J$ is A ≥ 3.4. The advanced tokamak feature of significant bootstrap current is also enhanced at higher aspect ratios. Fusion power plant design studies based on advanced tokamak scenarios, such as ARIES-AT [6] and ASSTR [7], have chosen A = 4. These considerations have led to the choice of A = 3.6 for FIRE.

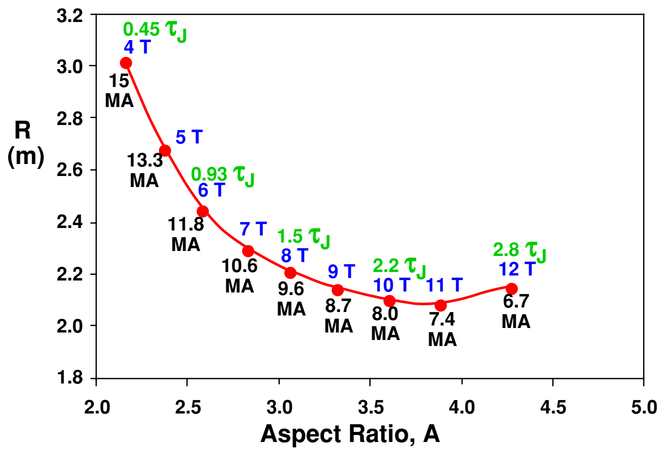

FIG. 1 Optimization of H-Mode Performance

4. FIRE Configuration and General Parameters

The FIRE configuration and aspect ratio is similar to that of other advanced tokamak designs such as TPX [8], KSTAR [9] and JT60-SC[10]. The key advanced tokamak features 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 wall stabilization capabilities. The reference design point is R_o = 2.14 m, a = 0.595 m, $B_t(R_o)$ = 6 - 10 T, I_p = 4.5 - 7.7 MA with a flat top time of 40 - 20 s for 150 MW of fusion power with the cross-section shown in Fig. 2. Plasma heating for the FIRE reference operating mode would be provided by 20 MW of ion cyclotron (ICRF) in the frequency range of 80-120 MHz utilizing 4 ports. The primary heating modes will be He3 minority and second harmonic tritium for 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 of is being considered as an upgrade for advanced tokamak (AT) operation and NTM control.

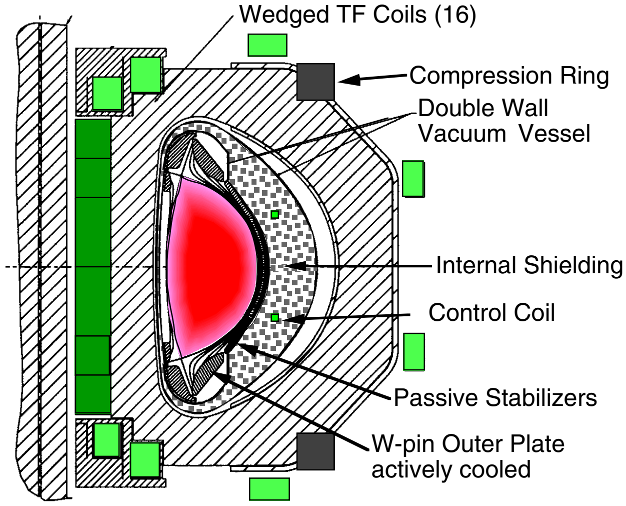

FIG. 2 FIRE Cross-section. The major radius is 2.14 m.

5. Plasma Performance Projections for Elmy H-Mode Operation

The physics issues, operating modes and physics design guidelines for projecting burning plasma performance in FIRE are similar to those for ITER-FEAT [5]. 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[3, 11]. Nominal FIRE parameters are shown in Table I. As a result of the Snowmass Assessment of FIRE, some of the standard parameters were modified. The plasma elongation and triangularity are slightly higher due to self-consistent treatment of the bootstrap current produced by the edge pressure pedestal. The density and temperature peaking were decreased and increased respectively to be more consistent with ITER-FEAT assumptions. The Be impurity level was decreased due the high plasma density and an Ar impurity was added to reduce the plasma heat loads on the divertor targets. A POPCON operating diagram shows a significant operating region (Fig. 3).

Table I. Nominal FIRE Parameters

	H-Mode	**Rev Shear AT**
R (m), a (m)	2.14, 0.595	2.14, 0.595
κ_x , κ_a , κ_{95}	2.0, 1.85,1.82	2.0, 1.85,1.82
δ_x , δ_{95}	0.7, 0.47	0.7, 0.55
q(0),q_{min}, q_{95} , li(3)	1.0,1.0,3.15, 0.7	4, 2.2, 3.5, 0.5
$B_t(R_o)$ (T), I_p (MA)	10, 7.7	6.5, 4.5
$n(0)/\langle n\rangle$, $T(0)/\langle T\rangle$	1.1, 2.5	1.5, 3.0
n/n_{GW}, $\langle n\rangle_{vol}$ (10^{20} m^{-3})	0.7, 4.5	0.73, 2.4
$T_i(0)$, $T_e(0)$	12.3, 12	18,16
Z_{eff}	1.5	2
H98(y,2), H(two term)	1.03, 0.87	1.7, 1.5
P_{loss}/P_{LH}	1.3	3
τ_E, (s)	0.95	0.6
Burn Duration/τ_{cr}	2.0	3.2
$Q = P_{fusion}/(P_{aux} + P_{OH})$	10	4.8
Fusion Power (MW)	150	140
β(%), β_N, β_p	2.2, 1.7, 0.7	4, 4.2, 2.15
f_{bs} (%)	25	74

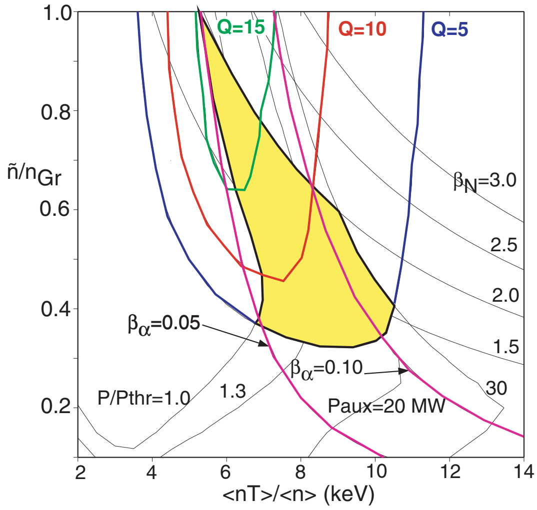

Fig. 3. H-Mode Operating Space in FIRE H98(y,2) = 1.1, n(0)/⟨n⟩ = 1.25, T(0)/⟨T⟩ = 2.5, τ_{He}/τ_E = 5, 3% Be

Recent 0-D analyses [12] of the H-mode confinement data base have developed a two term scaling relation that accounts for different scaling of the core and edge plasma. These results are somewhat more optimistic for FIRE, allowing the attainment of Q = 10 with H ≈ 0.9. Physics based models using marginal stability transport such as GLF23 [13] have also been used to predict burning plasma performance in FIRE. These models depend sensitively on the value of the H-mode temperature pedestal which is projected to be higher for plasmas with strong shaping (triangularity), and pedestal density low relative to the Greenwald density. Application of GLF23 to FIRE by Kinsey [13] has shown that it is possible to attain Q ≈ 10 with edge temperature pedestals in the range of 3.5 to 4 keV. 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 simulation of the Elmy H-mode regime using the 1 1/2 - D Tokamak Simulation Code (TSC) [14], with transport modeled by matching H(y,2) = 1.1 and n(0)/ $\langle n \rangle_V$ = 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} [2,3]. 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 testing of techniques for burn control.

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. Encouraging results on the complete stabilization of NTMs using lower hybrid current drive have been obtained on COMPASS-D [15]. Calculations with a LHCD model in the TSC code have shown that a 12 - 15 MW 4.6 GHz system with 50/50 on/off modulation should be capable of suppressing the m/n = 3/2 mode up to β_N ≈ 2.0 on FIRE. The possibility of using electron cyclotron current drive (ECCD) in the range of 170 GHz for advanced tokamak modes at ~6.5T is being investigated.

6. Advanced Tokamak Operating Modes in FIRE

The previous studies of AT modes on FIRE [16] have been extended. Systems analysis with zero-dimensional physics models and engineering constraints shows that FIRE can access a large advanced tokamak operating space assuming an internal transport barrier can be achieved with reversed shear. 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

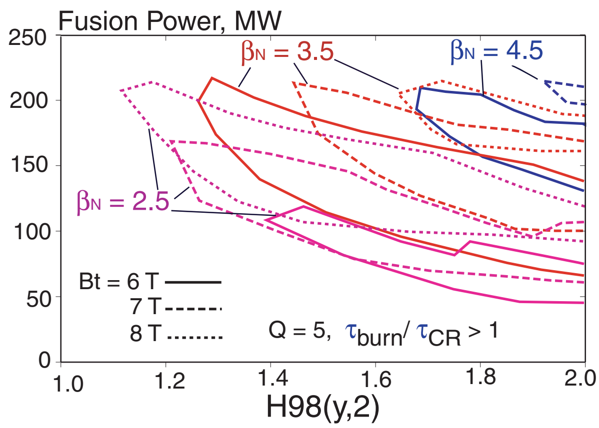

FIG. 4a Operating Space for FIRE AT modes

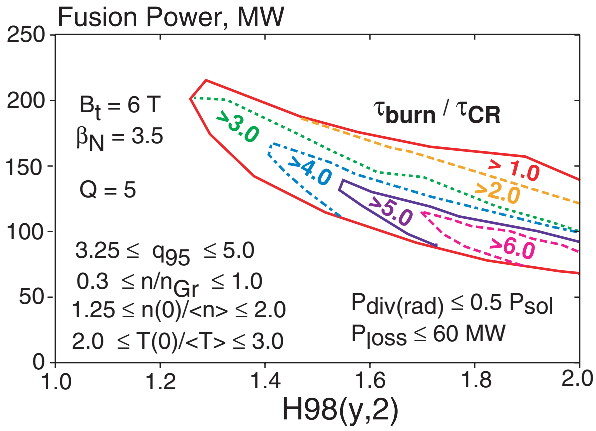

Fig 4b Duration of high-β AT modes

higher radiated power fractions. Viable solutions must be within the engineering limits set by the first wall heat flux including a peaking factor of 2 (<1 MWm^{-2}), particle power to the outboard divertor (<28 MW), and the radiated power load in the divertor and baffle (<6-8 MWm^{-2}). Increasing the radiated power in the divertor reduces the particle heat load and expands the operating space significantly. The cases shown in Fig. 4 have a fusion gain of 5. The toroidal field is varied to optimize the size of the operating space, combining low β_N with high B_t, and visa versa (Fig. 4a). 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 lower H98 factors by increasing Ar in the plasma to radiate more power in the divertor and on the first wall resulting in $1.5 \le Z_{eff} \le 2.3$. 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} as shown in Fig. 4b.

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 [17] 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 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 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, launching requirements, and whether heating alone may be sufficient.

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 [16], and in a stand-alone equilibrium using ACCOME [19]. Reverse power and trapped particle effects reduce the current drive efficiency giving 1.7 MA of current at 8.5 T and 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 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. 5). 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.

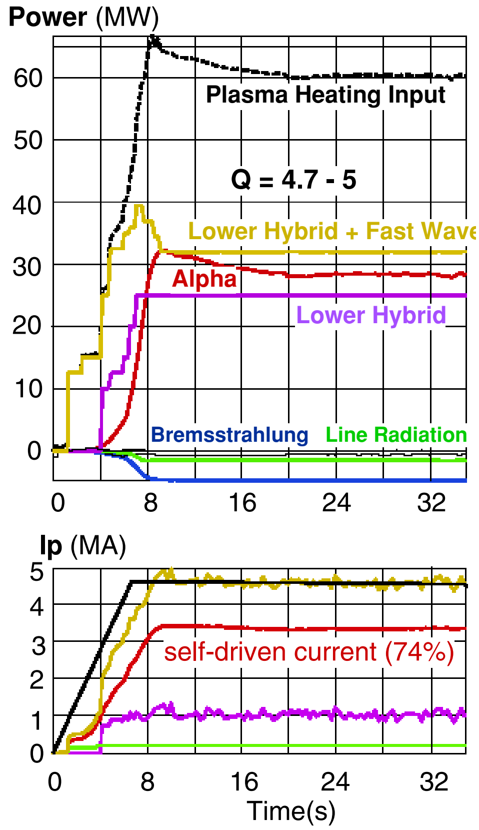

FIG. 5. "Steady-State" high-β AT in FIRE sustained for > 3 τ_{CR}.

7. Energetic Particle Effects in FIRE

The ripple loss of alpha particles establishes the lower limit on plasma current for AT operation. Calculations show that fast particle losses range from 2 - 8 % over the range of FIRE AT modes. Instabilities driven by the gradient of the energetic particle pressure, such as fishbones and toroidicity induced Alfven Eigen-modes (TAEs) are potential threats to fusion alpha particle confinement in a fusion reactor. A detailed analysis [20], of Alfven Eigen-modes including TAEs, Ellipticity (EAE) and Triangularity (NAE) induced Alfven modes, has been done with the use of analytical methods and numerical codes including the local fully kinetic non-perturbative code HINST and global perturbative code NOVA. HINST is a local stability analysis, which does not include stabilizing effects coming from the global mode structure. It shows that TAEs are weakly unstable for the nominal Q = 10 FIRE Elmy H-mode plasma with central plasma temperature T_0 = 12 keV. The most unstable toroidal mode numbers are around n = 8 within the minor radius 0.52 < r/a < 0.65. A higher temperature case at T_0 = 21 keV shows strong instability over a broader minor radius domain 0.45 < r/a < 0.75. The global analysis with the kinetic NOVA code of the FIRE nominal case shows that all AEs are stable for the low temperature case. However for the high temperature case TAE's and EAE's are marginally stable, while NAE's show instability. These modes are localized closer to the core.

8. Diagnostics

FIRE will need a comprehensive set of plasma and machine control diagnostics to carry out its mission. A preliminary set of diagnostics has been identified and a tentative assignment of ports has been made [21]. At this point in the design process, it has only been possible to consider conceptual aspects of the integration of diagnostics with the tokamak, its internal hardware and the necessary radiation shielding. There are many diagnostics requiring optical sight-lines to the core plasma and to the divertor, which will require labyrinthine paths through thick shielding plugs in the access ducts. Magnetic diagnostics, for measuring parameters such as the plasma current and position and high-frequency instabilities, will necessarily be mounted immediately behind first-wall tiles and must be integrated with the structures planned for these areas. A diagnostic neutral beam is being considered to enable measurement of the core ion temperature, plasma rotation, current density profile and alpha-ash buildup. Significant design integration with the divertor and first wall components is required for access to the divertors, and to gain sight-lines for the x-points and separatrix legs into the divertors and their contact points. A draft R&D plan for diagnostics has been prepared and will be used to guide diagnostic development.

9. 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. 3-D finite-element stress analyses of the TF coils including electromagnetic, and thermal stress due to ohmic and nuclear heating have shown that this design has a margin of 30% beyond the usual allowable engineering stress requirements. The present innermost central solenoid coils are near the design stress allowable but have considerable thermal margin. Consequently, consideration is being given to changing the design from OFHC to Cu-Cr-Zr copper alloy. FIRE has very little neutron shielding between the plasma and the toroidal field coil, therefore the nuclear heating must be taken into account in the calculation of pulse duration. 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. Neutron damage to the TF insulator limits the total fusion energy production to 5.5 TJ the same as BPX. An insulator R&D program is proposed that would allow the fusion energy to be increased. The repetition time at full field and full pulse length will be ~ 3 hr, with significantly shorter times at reduced field or pulse length. The addition of a second cooling tube to reduce this repetition rate time by a factor of four is being investigated. 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. Remote maintenance inside the vacuum vessel would be accomplished using a cantilevered articulated boom inserted through the large mid-plane port. The engineering systems are described in more detail by Thome et al. [22, 23].

FIRE would provide reactor relevant experience for divertor and first wall power handling since the anticipated thermal power densities on the divertor plates of ~6 MWm^{-2} for detached operation, ~12 MW/m^2 with modest amounts (few %) of neon added to the divertor, and a maximum of ~25 MWm^{-2} for attached operation, approach those anticipated for ARIES-RS [24, 25]. These estimates, based on modeling using the UEDGE Code with edge transport parameters expected for FIRE conditions, result in a power e-folding width of 2.3 cm [24]. The D-T experiments on TFTR and JET observed tritium retention fractions of ≈ 15 to 30% with carbon limiters and divertor plates [26] which 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. Therefore, FIRE has chosen to use only metallic materials for plasma facing components (PFC). The divertor PFCs 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 come into steady-state equilibrium during the pulse [24]. UEDGE modeling shows that the plasma temperature at the divertor target is sufficiently low as to prevent significant sputtering of the tungsten targets [24]. The effect of Type I ELMs on the FIRE tungsten divertor plate is a major concern and has been analyzed assuming the following characteristics: 1) between 2 and 5% of the total stored energy is lost in an ELM; 2) the energy lost in an ELM is deposited over approximately the same area as normal operation (spreading of up to a factor of 3 was

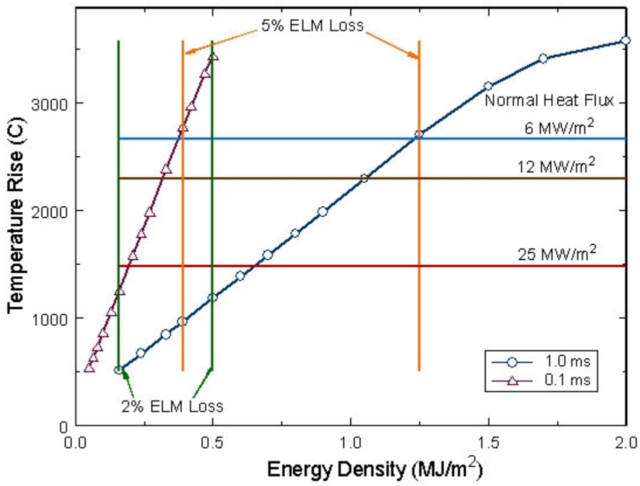

FIG. 6. Impact of Type I Elms on the FIRE tungsten divertor target

examined); 3) the duration of an ELM is between 0.1 and 1.0 ms; 4) the observed ELM frequency is a few Hz. Fig. 6 shows the calculated temperature rise of the tungsten surface due to an ELM and horizontal lines show the temperature rise allowed before surface melting occurs. The calculated heat flux to the divertor is about 12 MW/m^2 for partially detached operation. For the 2% per ELM loss case, very little spreading of the energy deposition is needed to prevent melting for the 0.1 ms duration while no spreading is needed for 1.0 ms. For the 5% per ELM loss case, the 0.1 ms duration will always cause melting while very little spreading is needed for the 1.0 ms cases to be acceptable. We conclude that Type I ELMs would be a life limiting process for the FIRE divertor. The existing experience [27] on ELMs also suggests that FIRE's double-null operation, high triangularity, and high edge density help reduce the size of ELMs with a transition to Type II ELMs. Continued R&D from operating tokamaks is needed to develop ELM mitigation techniques for FIRE.

The first wall is comprised of Be (5mm) plasma-sprayed onto copper tiles [24]. The neutron wall loading at the outboard wall in FIRE is ~ 2.3 MWm^{-2} and produces significant nuclear heating of the first wall and vacuum vessel during the 20 – 40s pulse. The inner divertor targets and first wall are cooled by mechanical attachment to water-cooled copper plates inside the vacuum vessel. 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 beyond previous US fusion program experience in tritium shipping and handling.

The construction cost of the tokamak (magnets, divertor, plasma facing components and mechanical structure) has been estimated to be ≈ $350M (FY02US) including $75 M of contingency. Another ≈ $850 M including $170 M of contingency would be required for auxiliary heating, startup diagnostics, power supplies and buildings.

10. Acknowledgments

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