Princeton Plasma Physics Laboratory

PPPL-4011

PPPL-4011

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October 2004

PPPL
PRINCETON PLASMA PHYSICS LABORATORY

Prepared for the U.S. Department of Energy under Contract DE-AC02-76CH03073.

FIRE, A TEST BED FOR ARIES-RS/AT ADVANCED PHYSICS AND PLASMA TECHNOLOGY

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ABSTRACT *The overall vision for FIRE is to develop and test the fusion plasma physics and plasma technologies needed to realize capabilities of the ARIES-RS/AT power plant designs. The mission of FIRE is to attain, explore, understand and optimize a fusion dominated plasma which would be satisfied by producing DT fusion plasmas with nominal fusion gains ~10, self-driven currents of ≈ 80%, fusion power ~ 150 - 300 MW and pulse lengths up to 40 s. Achieving these goals will require the deployment of several key fusion technologies under conditions approaching those of ARIES-RS/AT. The FIRE plasma configuration with strong plasma shaping, a double null pumped divertor and all metal plasma facing components is a 40% scale model of the ARIES-RS/AT plasma configuration. "Steady-state" advanced tokamak modes in FIRE with high β, high bootstrap fraction and 100% non-inductive current drive are suitable for testing the physics of the ARIES-RS/AT operating modes. The development of techniques to handle power plant relevant exhaust power while maintaining low tritium inventory is a major objective for a burning plasma experiment. The FIRE H-modes and AT-modes result in fusion power densities from 3 - 10 MWm^{-3} and neutron wall loading from 2 - 4 MW m^{-2} which are at the levels expected from the ARIES–RS /AT design studies.*

I. INTRODUCTION

The Advanced Reactor Innovation Evaluation Studies (ARIES) have defined the plasma requirements for an economically attractive tokamak power plant [1, 2]. These include: high fusion power gain (Q ≈ 25 – 50), high fusion power density (≈ 5 MWm^{-3}), steady-state operation with ≈ 90% self-driven current, and plasma exhaust at high power density. The overall vision for FIRE is to develop and test the fusion plasma physics and plasma technologies needed to realize capabilities of the ARIES-RS/AT power plant designs while reducing technological risks and minimizing costs of implementing the next step in fusion development.

The ARIES power plant studies had fusion power densities of ~5 MWm^{-3} , which will require βB^2 sufficient to produce volume average plasma pressures, $\langle p \rangle$, of ~10 atmospheres (atm). Progress toward achieving the plasma pressure needed for a magnetic fusion power plant is shown in Fig. 1.

Plasma pressures of just over 1 atm. have been achieved in several tokamaks including; Alcator C, C-Mod, TFTR, JT-60U, JET and DIII-D. The maximum pressure of 1.6 atm. was achieved by Alcator C in 1983 using only ohmic heating. Very high and very low aspect ratio tokamak plasmas like PBX-M and NSTX have achieved plasma pressures of 0.3 atm. and 0.25 atm. respectively. The maximum plasma pressure is limited by

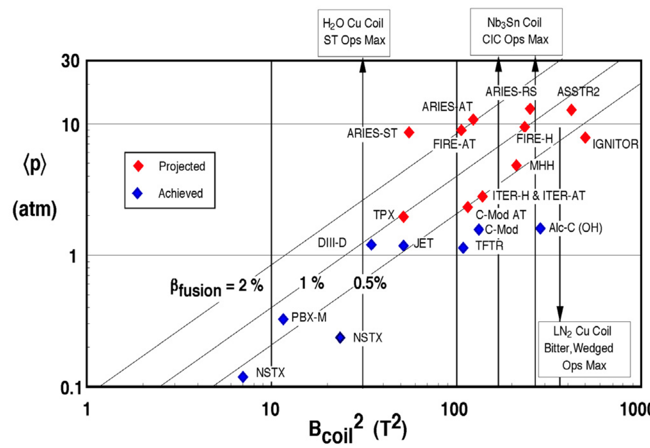

Fig.1. Progress toward the plasma pressure required by a magnetic fusion power plant.

a combination of plasma physics, coil geometry and engineering as shown by the three terms in Eq. 1 below.

$$\langle p \rangle = \beta_{to} B_{to}^2 = \beta_{to} (B_{to}/B_{coil})^2 B_{coil}^2 \quad (1)$$

$$= \beta_{fusion} B_{coil}^2 \quad (2)$$

For superconducting coils, B_{coil} is subject to the limits of electromagnetic stress and the superconducting transition field while normal conducting coils are subject to the limits of electromagnetic and thermal stress as well as coil power dissipation. The operational limits for various coil systems are also shown in Fig. 1. Therefore, the appropriate "Figure of Merit" for utilization of magnetic field in a magnetic fusion reactor is $\beta_{fusion} = \langle p \rangle / B_{coil}^2$ where B_{coil} is the maximum field achievable at the toroidal field coil as shown in Eq. 2 above. As shown in Fig 1, the achieved β_{fusion} ranges from 0.2% to 1%. A variety of power plant design studies and burning plasma experiments proposals are also shown on Fig. 1 with power plant designs in the range of β_{fusion}= 1 – 2% that would result in plasma pressures of 10 atm and DT power densities of 5 MW^{-3}.

The ongoing tokamak program and a next step burning plasma experiment have the goals to understand the physics, and to determine the requirements for attaining, controlling and sustaining high-β steady-state advanced tokamak regimes for time scales long compared to internal plasma time scales.

II. REQUIREMENTS FOR FIRE

FIRE physics objectives are to attain, explore, understand and optimize burning plasma physics in the conventional inductively driven H-Mode with Q ~10 as the target but higher Q is not precluded [3-5]. This would provide plasmas that are strongly self-heated with f_α = P_α/P_{heat}~ 66% as target and up to 83% self-heating at Q = 25. In addition, the plasma should be stable to Toroidal Alfven Eigen (TAE) modes driven by energetic alpha particles at the nominal operating point with regions of instability accessible for physics studies.

A unique feature of the FIRE program is the emphasis on ARIES-like steady-state advanced tokamak (AT) modes [6,7] with 100% non-inductively driven plasma current (≈80% bootstrap) and high beta (β_N ~ 4.0). This will require wall stabilization of slowing rotating plasmas using resistive wall mode feedback coils similar to that envisioned in the ARIES-RS/AT designs. Attaining this regime at Q = 5 and β = 5% will require the advances presently being achieved individually in confinement and beta to be made simultaneously in a burning plasma.

The pulse duration must be sufficiently long in physical time scales to study the physics and technology issues of interest. The important phenomena and durations for FIRE are:

- Pressure profile evolution and burn control~ 20 to 40 τ_E
- Alpha ash control and removal ~ 4 to 8 τ_{He}
- Plasma current profile redistribution ~ 2 to 5 τ_{CR}
- Divertor pumping and heat removal ~15 to 30 $\tau_{divertor}$
- First wall heat removal > 1 $\tau_{first-wall}$

where τ_E is the plasma energy confinement time, τ_{He} is the helium ash confinement time (typically 5), τ_{CR} is the plasma current profile redistribution time, $\tau_{divertor}$ is the thermal time constant of the divertor target and $\tau_{first-wall}$ is the thermal time constant of the first wall tiles. This capability will allow investigation of "quasi-steady-state" behavior of the key physics phenomena, and all plasma related technology areas except steady-state operation of the first wall.

The FIRE plasma configuration with strong plasma shaping, a double null pumped divertor, low toroidal field ripple (< 0.3%), internal control coils, space for wall stabilization capabilities and all metal plasma facing components is a 40% scale model of the ARIES-RS/AT plasma configuration. The divertor targets and dome baffle are actively cooled tungsten and the first wall consists of beryllium coated copper tiles that are cooled between pulses. This close fitting conducting structure is similar to ARIES and provides passive stabilization. Only ARIES-like current drive technologies of fast wave current drive (FWCD) and lower hybrid current drive (LHCD) are employed so there is no external toroidal momentum input. The tokamak size and project cost are significantly reduced by using by LN cooled BeCu/OFHC bitter plate coils to produce a toroidal magnetic field at levels comparable to ARIES [8,9]. Recent design improvements have added additional cooling tubes to the TF conductor to triple the repletion rate. The divertor target plates and dome baffle have been integrated into a single module to simplify cooling paths and simplify remote handling. The power and site needs are comparable to previous DT tokamaks (TFTR/JET). The tritium required/pulse ≤ 0.3g-T in FIRE is comparable to TFTR tritium needs, and does not strain tritium supplies, shipping or waste requirements. FIRE parameters are compared to ARIES in Table 1.

Table 1.	**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

III. PLASMA OPERATING REGIMES IN FIRE

FIRE would be capable of operating in a number of physics regimes including: conventional H-Mode, optimized shear (flat q profile in the core), hybrid, and reversed shear advanced tokamak modes. The physics databases for the H-Mode, the hybrid mode and the reversed shear AT have been significantly improved since the 2002 Snowmass Fusion Summer Study [10]. The physics benefits of the FIRE choice for double null pumped divertor have been expanded by recent results including:

- confinement time increased at high density by increased triangularity and double null
- beta limits increased as triangularity is increased
- edge localized modes (ELMs) reduced as triangularity is increase toward DN

These results are being documented by the International Tokamak Physics activity (ITPA).

IV. H-MODE OPERATING SPACE

The first critical issue to be addressed by a burning plasma experiment is to attain burning plasma conditions that are dominated by fusion phenomena. The 2002 Snowmass Fusion Summer Study concluded that there is confidence that FIRE will achieve burning plasma performance in the H–mode based on an extensive experimental database [11,12]. The FIRE standard ELMing H–mode scenario is a reasonable extrapolation from the existing database as shown in Fig. 2. FIRE would require an increase of 2.5 in the dimensionless confinement time, $B\tau_E$, to achieve Q = 10 while ITER requires an increase of 4 to achieve Q =10 and ARIES would require an increase of 3 to achieve Q = 25. Note, the dimensionless confinement of spherical tori (ST) would have to be increase by two orders of magnitude from present NSTX/MAST results to reach ST burning plasma conditions at $B\tau_E$ = 2 T-s.

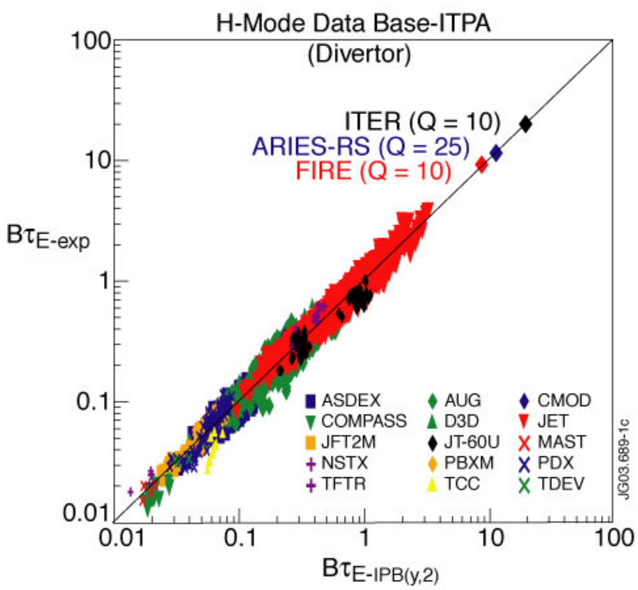

Fig. 2. Dimensionless Confinement time database for the H-Mode and projections to FIRE, ITER and ARIES.

A global systems code was used to determine the FIRE H-mode operating range [13]. 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: $5 \le Q \le 30$, $5 \le P_{aux}$ (MW) ≤ 30, $1.05 \le n(0)/\langle n\rangle \le 1.25$, $0.3 \le n/n_{Gr} \le 1$,and $1.5 \le \beta_N \le 3$. 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. 3. Optimizing the distribution of exhaust power on the first wall, the

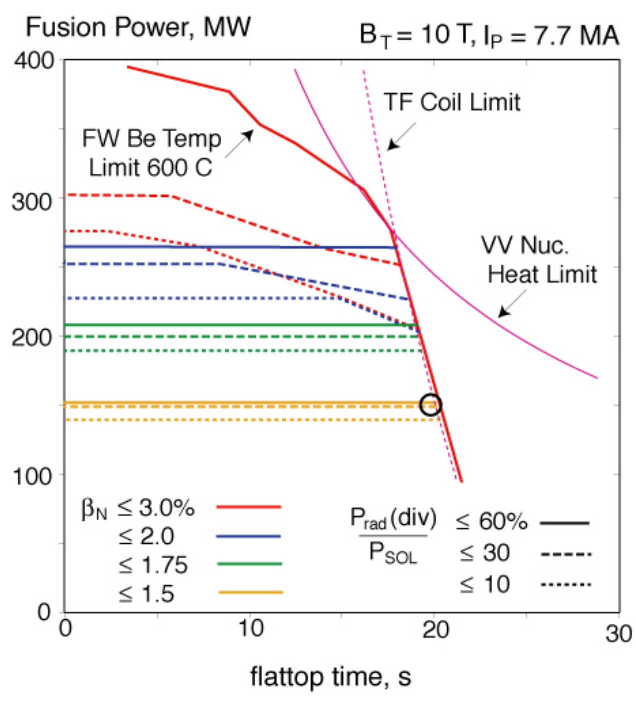

Fig. 3. H-Mode Operating Space in FIRE

divertor chamber walls and the divertor targets can significantly expand 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 10s 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_{Gr} (0.7 – 1.0) and H98 (1.03 - 1.1).

The area of greatest progress on FIRE has been the development of a "steady-state" high−β ARIES-like AT configuration [13]. The AT configurations in FIRE rely on ICRF Fast Wave (FW) on-axis current drive and lower hybrid (LH) off-axis current drive like the current drive systems in ARIES-RS/AT. The ICRF system can provide 200 kA of current by injecting 20 MW of power with the existing two-strap antenna design. Typical AT plasmas require less than 200 kA of on-axis current drive. Upgrading to four strap antennas would improve the CD efficiency. Off-axis current drive in FIRE is critical for establishing and controlling the safety factor profile, and is accomplished using up to 30 MW of LHCD at 5 GHz. The experience developed on Alcator C-Mod advanced tokamak experiment with lower hybrid current drive will strengthen the basis for FIRE projections.

Bootstrap and external current drive consistent equilibrium and stability analysis show that the high-n ballooning limit for typical plasmas is $\beta_N > 4.7$. With no wall the ideal MHD β_N limits for n=1, 2 and 3 are 2.7, 3.6, and 4.0, respectively. With a wall located at b/a = 1.35 on the outboard side only, the ideal MHD β_N limits for n=1, 2 and 3 are 6.1, 5.3, and 5.1, respectively. Calculations show that feedback coils, located near the front face of the shield plug in every other mid-plane port, could stabilize the n=1 resistive wall mode (RWM) up to 80-90% of the with-wall limit. The plasma configurations targeted have safety factor values above 2.0 everywhere, so that the (5,2) and (3,1) are the lowest order neoclassical tearing modes (NTMs) of interest. Stabilization of the NTMs using electron cyclotron current drive (ECCD) from the low field side at the toroidal field of 6.5 T would require frequencies of 140-170 GHz, which is close to the range of achieved values in the high-power long-pulse gyrotron R&D program. The LHCD system could also be used to launch two spectra, one for bulk CD and the other for NTM suppression.

V. AT MODE OPERATING SPACE

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 as shown in Fig. 4. An expression for the bootstrap current fraction is included and the current drive power is given by Pcd = [nRIp(1-fbs)]/ηcd [14]. 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 ηcd = 0.2 and 0.16 $A/W\text{-}m^2$ 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 $B_t = 6.5T$, P_{LH} (MW) ≤ 30, P_{ICRF} (MW) ≤ 30, $1.05 \le n(0)/\langle n \rangle \le 2$, $2 \le T(0)/\langle T \rangle \le 3$, $0.3 \le n/n_{Gr} \le 1$, $3.25 \le q_{95} \le 5$, and $3 \le \beta_N \le 4.5$. Attainment of $\beta_N \ge 3$ 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 $1.5 \le Z_{eff} \le 2.3$. The fraction of power radiated in the divertor (P_{rad}(div)) to power exhausted into the scrape-off layer (P_{SOL}) was allowed to vary from 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

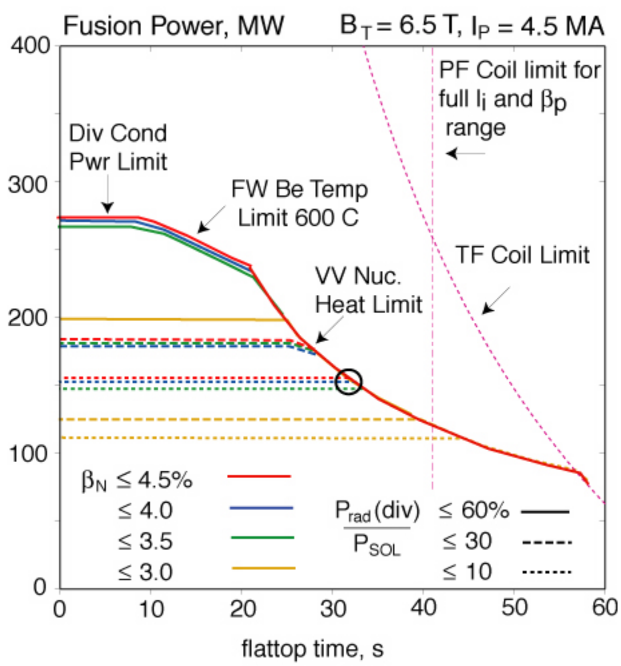

Fig. 4. AT Mode Operating Space in FIRE

are limited primarily by the nuclear heating in the vacuum vessel rather than TF coil heating as shown in Fig. 4. 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}.

VI. SIMULATION OF BURNING PLASMAS IN FIRE

Simulations of steady-state high-beta Advanced Tokamak plasma discharges with 100% non-inductive current composed of fast-wave, lower-hybrid, and bootstrap currents have been done for FIRE using the Tokamak Simulation Code [15]. This is accomplished by programming the heating and current-drive sources so that the inductive contribution to the plasma current is reduced to zero by the end of the ramp up and the current profile is that desired for AT operation. Although inductive and non-inductive current drive are used to ramp the plasma current, the flattop plasma has a "steady-state" 100% non-inductive current provided by the combination of bootstrap, lower hybrid, and fast wave current, and the current profile is held constant for ≈4 τ_{CR}.

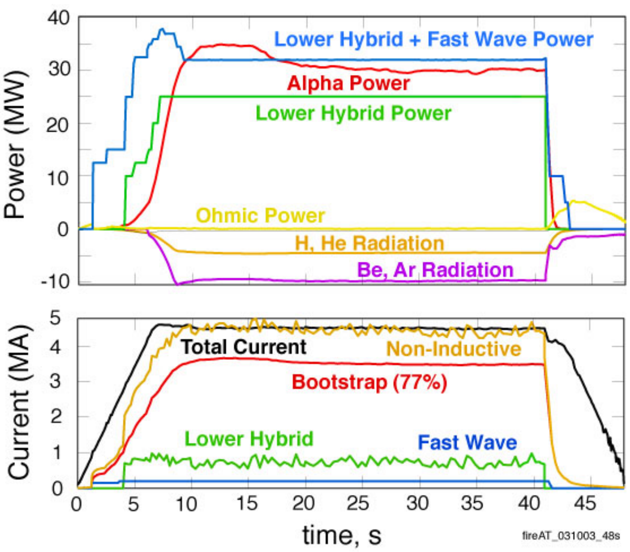

Fig. 5. Integrated simulation of the "steady-state" high–β ARIES-like AT mode in FIRE.

VI. FIRE TECHNOLOGICAL CONTRIBUTIONS

VI. A. General Considerations

FIRE would focus on developing power plant relevant technologies, such as stabilizing first wall and closely coupled feedback coils, plasma facing components, plasma current drive, and fueling, that are closely coupled to the burning plasma. FIRE would deemphasize power plant technologies that are not coupled to the burning plasma such as superconducting coils, high fluence nuclear testing and high inventory tritium systems to reduce the technical risk and cost of a next step burning plasma experiment. In a FIRE-based multi-machine development plan, the R&D to address issues not directly coupled to the burning plasma (superconducting coils, high neutron fluence materials development, and high inventory tritium handling) would be carried out in separate optimized facilities.

VI. B. RWM Coils Integrated with the First Wall.

Both FIRE and ARIES-RS/AT would rely on passive stabilization from close fitting conducting structures and Resistive Wall Mode (RWM) stabilization from coils integrated with the first wall structure. FIRE proposes to place the RWM coils just behind the Be coated first wall tiles in the first wall of removable shielding port plugs. The integrated port plug shielding assembly could also include diagnostics and RF wave launchers. This configuration allows both close coupling and possibility of RWM coil removal for maintenance and modification. Calculations with VALEN [16] 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 β_N = 4.2. The influence of the n=2 mode on the achievable β_N is being investigated. The analysis of RWM stabilization is benefiting from the experimental progress using external RWM coils on DIII-D [17] and the recently installed internal RWM coil system [18]. The RWM coils could also be utilized for ultra fast vertical position control, which in combination with the neutral stability point of the DN divertor could lead to avoidance of vertical displacement events (VDE). The feasibility of this approach will be investigated as part of the Next Step Option evaluation ARIES-like AT modes on ITER. The key technical issues include: maintaining adequate electrical insulation in an intense neutron flux while providing a high frequency magnetic response and withstanding the electromagnetic loads due to disruptions. The development of closely coupled RWM coils for an environment with neutron fluxes similar to ARIES-RS/AT would be an important contribution of the FIRE program.

VI. C. All Metal Plasma Facing Components Compatible with Tritium Inventory Control

Maintaining a low tritium inventory within the plasma chamber is essential to realize the potential safety advantages of a fusion reactor. A power plant like ARIES would need an effective retention coefficient of < 0.04% to meet it's in-vessel tritium limit for a 9 month operating period. FIRE has the additional requirement of maintaining a tritium site inventory below 30g-T, so that

FIRE could classified as a Low Hazard Category 3 nuclear facility. This would ease regulatory restrictions and increase the number of sites where FIRE could be located. The tritium experiments on TFTR and JET demonstrated clearly that carbon PFCs had effective tritium retention coefficients greater than 10%, and were therefore incompatible with the needs of a tokamak power plant [19]. Refractory materials like Mo and W have been used successfully in C-Mod and ASDEX-U. The tritium retention measured by Wampler(Sandia) in C-Mod with a Mo first wall and divertor was < 0.2% after extended DD operation. The use of tungsten coated tiles in the baffle region of the divertor of ASDEX Upgrade is encouraging with regard to maintaining acceptably low levels of tungsten in the plasma core [20].

Pellet injection scenarios with high-field-side launch capability will reduce tritium throughput, and enhance fusion performance by providing D/T isotope control and possible density profile peaking. The in-device tritium inventory will be determined primarily by the cycle time of the divertor cryo-pumps, and can range from < 2 g-T for regeneration overnight to ~10 g-T 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 U.S. fusion program experience in tritium shipping and handling.

The removal of plasma exhaust power and particles is a major challenge for a magnetic fusion power plant, and the development of techniques to handle power plant relevant exhaust power densities is a major objective for a burning plasma experiment. This is an area where FIRE can make major contributions to a critical area. The FIRE-AT $\beta \approx 4\%$ would result in fusion power densities from 3 to 10 MWm^{-3} and neutron wall loading from 2 to 4 MW m^{-2} which are at the levels expected from the ARIES–RS/AT design studies. The divertor and first wall thermal loads would also be in the range expected for ARIES. The tungsten brush divertor, developed as part of the U.S. R&D effort for ITER, has the capability of handling steady-state thermal loads approaching those of ARIES while maintaining low tritium inventory [15]. FIRE would be able to provide a good test of the feasibility of a W/Be divertor first wall design for ARIES-RS/AT.

The first wall design of FIRE features copper tiles that have 5 mm of Be plasma sprayed on the plasma facing side that are capable of absorbing 1 MWm^{-2} for ~40s before the Be surface reaches 600 °C. The tiles are cooled by contact Cu cladding. Water flowing through channels in the copper cladding cools the tile to ambient temperature for the next pulse. This configuration would provide a good test of whether Be would be suitable as a first wall material in ARIES-RS/AT. Future work in this

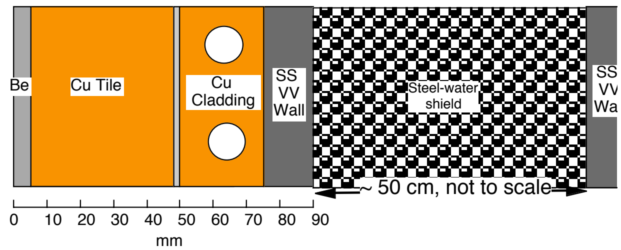

Fig. 4. FIRE first wall cross-section at outer midplane

area would look at the possibility of enhancing the thermal transfer properties of the tile–cladding interface and increasing the water cooling to extend the first wall capability to near steady-state.

VI. D. Remote Maintenance

Remote maintenance requirements have been incorporated into the preconceptual design of the FIRE tokamak and the FIRE facility. The in-vessel remote handling system for FIRE utilizes articulated boom remote manipulators. The divertor is comprised of 16 modules top and bottom aligned with the midplane ports between the 16 TF coils. Each module consisting of the inner divertor target, the dome baffle and outer divertor target would be removed a single unit by the boom manipulator inserted through a horizontal midplane port and withdrawn into a shielded cask similar to the ITER concept.

VII. CONCLUDING COMMENTS

The pre-conceptual design for FIRE has been completed, and resources have been redirected to support the advancement of ITER as recommended by the DOE Fusion Energy Sciences Review Committee (FESAC). If the ITER project does not go forward, FIRE is positioned to begin Conceptual Design Activities as recommended by FESAC. If constructed, FIRE would provide essential contributions to physics and technology R&D needed for a fusion power plant of the ARIES class.

The FIRE design study is a U. S. national activity managed through the Virtual Laboratory for Technology. The FIRE activities are carried out by participants at Advanced Energy Systems, Argonne National Laboratory, Boeing Company, General Atomics, Georgia Institute of Technology, Idaho National Environmental and Engineering Laboratory, Lawrence Livermore National Laboratory, Los Alamos National Laboratory, Massachusetts Institute for Technology, Oak Ridge National Laboratory, Princeton Plasma Physics Laboratory, Sandia National Laboratory, University of Illinois, and University of Wisconsin. The PPPL work was supported by DOE Contract # DE-AC02-76CHO3073.

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07/07/03

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