ENGINEERING FEATURES OF THE
FUSION IGNITION RESEARCH EXPERIMENT (FIRE)

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The FIRE tokamak is an option for the next step in the US magnetic fusion energy program. It has a major radius of 2 m, and a minor radius of 0.525 m. The general requirements for FIRE are B_T= 10T; I_P= 6.5MA; minimum flat top time=10 s, and maximum number of full-power, full-field pulses = 3000 with 30,000 pulses at 2/3 full field. The baseline design is able to meet or exceed these objectives. All magnets are inertially cooled with liquid nitrogen and have the capability for an 18s flat top at 10 T. The use of BeCu in the inner legs of the toroidal field coil (TF) also allows for a field of 12T with a pulse length of 12s. Extended pulse lengths at lower fields (eg-214 s at 4T and 2 MA) will allow FIRE to explore advanced tokamak modes [1].

1. Features of the FIRE Design

The design effort on FIRE was initiated in 1999 [2]. During 2000, more detailed analyses have been completed to support the design development [3]. The configuration features a vertically elongated plasma with a double null divertor. Radiation shielding is integrated into the double walled vacuum vessel to minimize the dose to magnet insulators and facilitate hands on maintenance of components outside the tokamak. The wedged TF is constructed of liquid nitrogen cooled copper and beryllium copper. Added wedging pressure at the top and bottom of the inner legs of the TF is provided by two large compression rings. The divertor modules are actively cooled where required with tungsten plasma facing surfaces. Ion cyclotron heating will be launched with antenna through the large ports. The major features are shown in Figure 1.

-TF Coils and Global Structure- The TF coils are wedged and use C17510 beryllium copper for the inner legs and C10200 OFHC copper for the balance of the coils. Two

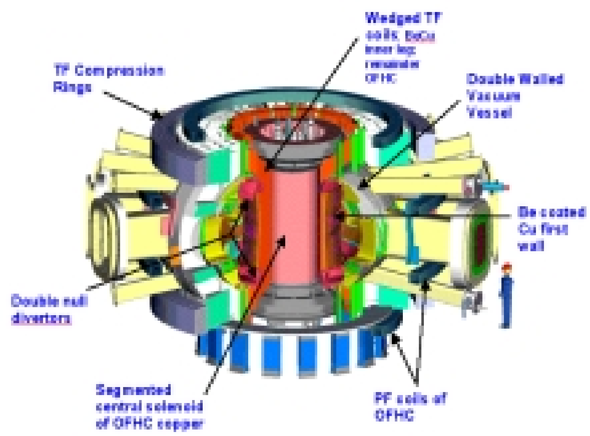

Figure 1-Section through FIRE

large steel compression rings preload the TF coils radially and augment the wedge compression at the top and bottom of the inboard legs, thus aiding the support of the overturning moments during a pulse. The TF coil peak conductor membrane plus bending stresses are 469 and 689 MPa for operation at 10 T and 12 T, respectively. This is within the static 724 MPa allowable for C17510 beryllium copper with a conductivity of 68% IACS. The pulse length is determined by a peak allowable temperature of 373 K. A summary of selected scenarios for FIRE is given in Table 1.

-*Central Solenoid (CS) and Poloidal Field Coils (PF)*-The poloidal field coil system consists of a free standing 5 segment central solenoid and eight ring coils. All are made of C10200 copper and are inertially cooled with LN_2.

-*Vacuum Vessel (VV)*- A double walled VV with steel/water shielding located between the walls decreases the dose rate on the TF coils and allows hands on maintenance outside the TF coils. Each vessel quadrant has a large mid plane port, angled ports above and below the mid-plane, and vertical ports top and bottom. Port plug shielding, passive stabilization plates and active control coils have been incorporated into the VV. Carbon will not be used inside the vessel; hence, high temperature bakeout is not needed. The vessel will operate at 100C. The vessel is fabricated in octants from Type 316 LN stainless steel. When all the octants are in place within the TF coils, they are welded together from the plasma side of the torus. The field joint for the double wall structure uses splice plates to accommodate assembly tolerances, and for accessing the coil-side, face-sheet from the plasma side of the torus. This type of joint has undergone significant, full scale testing using remote welding equipment as part of the ITER R&D program.

-*Divertor and Plasma Facing Components (PFC)*-

The outer divertor plate and baffle are actively cooled; the inner divertor plate and first wall armor are conduction cooled to the copper clad vessel walls. As shown in Table 1, these cooling provisions are adequate for all operating modes. The outer divertor module design configuration builds on fabrication technologies developed for the ITER divertor and consists of 24, modular, copper-alloy "finger" plates with tungsten brush armor. The plates are mechanically attached to a stainless-steel support structure that spans the toroidal width of the module.

-*Thermal Shield* - The thermal shield provides the insulating environment for the liquid nitrogen cooled coils. It consists of a stainless steel structure with a thin shell of stainless steel covered by insulating panels and sprayed-on insulation. Penetrations will be sealed with rubber or fabric bellows that accommodate the relative motion between the VV and thermal shield.

-*Ion Cyclotron Heating* - Plasma transport calculations indicate the need for 30 MW of ICRH. The design calls for a four port system with two antennas per port. With a 6 cm gap to the plasma, the 30 MW can be delivered at 150 MHz with 35 kV peak voltage. The design value for the gap is 3-4 cm and calculations indicate that 30 MW can be delivered at 100 MHz with a 3.5 cm gap.

-*Fueling and Pumping* - Three sets of injectors aimed towards the plasma center fuel FIRE. Injection will be from the outside mid-plane, vertically, and from the inside lower quadrant. Tritium-rich pellets will be used for core fueling; deuterium-rich pellets will be used for edge fueling.

There are a total of 16 cryopumps with 8 each on the top and bottom (at alternate divertor ports), close coupled to the torus in the pumping duct directly from the double null divertor.

-*Tritium*-The on-site tritium inventory has been set at 30 g to allow sufficient operational flexibility without introducing additional restrictions. The inventory can be reduced if a tritium reprocessing system is added to recycle the tritium daily.

-*Neutronics and Shielding*- Nuclear heating was calculated for the 200 MW fusion power DT pulses. During these pulses the average neutron wall loading is 3 MW/m^2 with values at the outboard (OB) midplane, inboard (IB) midplane, and divertor being

3.6 MW/m^2, 2.7 MW/m^2, and 1.8 MW/m^2, respectively.

The insulation dose is 1.5 x 10^{10} rads for 3000 full power DT pulses (fusion energy of 5 TJ) and 30, 000 DD pulses (fusion energy of 0.5 TJ). This is the peak, end of life, value and occurs at the magnet surface at the inboard mid-plane. The commonly accepted dose limit for epoxies is 10^9 rads. Polyimides and bismaleimides are more radiation resistant with experimental data showing only a small degradation in shear strength at dose levels in excess of 10^{10} rads. However, they are difficult to process due to their high viscosity and requirement for high temperatures to fully cure. Newly developed insulations, such as cyanate esters, should provide radiation resistance with easier processing requirements.

The vacuum vessel jacket/shield thickness has been sized so that it, in conjunction with the shielding provided by the TF coils and port plugs, will permit "hands on" ex-vessel maintenance.

-*Activation, Decay Heat and Radiation Exposure*- The dose rates outside the magnet and at the mid-plane are acceptable for hands on maintenance within a few hours after shutdown. The dose rate at the top of the machine drops to an acceptable level within one day after shutdown. The biological dose rates behind the vacuum vessel and the divertor remain high following shutdown.

-*Power Supplies*- A 10 T pulse will require about 13.7 GJ of energy with peak powers of 490 MW and 250 MW for the TF and PF systems, respectively. At 12T, the TF energy requirement is about the same because of temperature limits in the TF system, but the PF energy requirement increases, so the total needed is about 15.2 GJ. The peak powers for the 12 T pulse are 815 MW and 360 MW for the TF and PF systems, respectively.

-*Cryoplant*- Large liquid nitrogen storage tanks are planned with pipeline delivery from a new, on-site or near site, air liquefaction plant. The magnets are kept cold overnight and weekends, and only warmed up to room temperature during maintenance periods. This provides considerable flexibility for adjusting shot scenarios.

-*Facilities and Siting* - The test cell size is about 39mx39m and determined by space required to maneuver and dock remote handling casks at ports. Because of the length of the port inserts, remote handling casks are expected to be approximately 8 m in length and about 1.9 m in width. The hot cell concept is based on the expectation that some port mounted objects can be repaired and returned to the tokamak. The hot cell processes are expected to include divertor repair, tritium recovery from beryllium, size reduction by sawing or cutting, and encapsulation of radioactive material for subsequent shipment to a waste repository.

A conceptual layout and building design has been developed for a "green field" site. In the future, candidate sites will be identified and evaluated for their influence on the cost and schedule of the project since significant savings may be available in the form of "site credits".

-*Safety*- Examination of the potential safety impact of selected energy sources has not revealed any events that pose a serious challenge to the radiological confinement function. Preliminary analyses indicate:

a) The passive decay heat removal is sufficient, and oxidation of the activated PFC surfaces are not a concern, following a complete loss of coolant condition for the divertor and VV after a pulse.

b) The pressure rise is acceptable following a break in the divertor or VV cooling lines inside of the VV.

c) The hydrogen generated from Be-steam and W-steam interactions is insignificant. However the tritium on the cryopumps must be controlled. The deflagration limit of 30 g-moles translates into a deflagration limit of ~ 300 g DT. Regeneration will be scheduled frequently enough to stay well below this limit.

2. Concluding Remarks

The baseline design for FIRE meets or exceeds requirements. Recent estimates indicate that a change in design to a bucked and wedged configuration for the TF coils would allow them to be made entirely of OFHC copper. This change would reduce the TF power consumption by a factor of ~2 and would allow the TF flat top to be increased to ~ 40 s. This option will be considered next year and is expected to have a cost benefit as well.

Acknowledgement

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References

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Table 1 Selected FIRE Operating Modes

Operating Mode	***TF Coils : Materials and performance***	***CS Coils Materials and performance***	***PF Coils Materials and performance***	***PFCs Materials and performance***
I. Baseline : 10 T / 6.44 MA DT Fusion Power ~200 MW P_{ext}=20 MW	BeCu , 68% IACS for inner leg; OFHC elsewhere. 18s flat top w/D-T; 26 s w/D-D	OFHC (C10200); T_{max} =152 K	OFHC (C10200); T_{max} =173 K	Actively cooled divertor outer plate and baffle; Inner plate and FW cooled by conduction to the Cu clad vessel.
II. Higher Field Mode 12 T / 7.7 MA P_{fusion}=250 MW P_{ext}=25 MW	Same as (I) 12 s w/ D-T; 15 s w/ D-D	Same as (I) T_{max} = 161 K	Same as (I) T_{max} = 183 K	Same as (I)
III. TPX-like Mode 4T / 2 MA P_{fusion}=5 MW P_{ext}=15 MW	Same as (I) ~214 s pulse duration.	Same as (I) T_{max} = 144 K	Same as (I) T_{max} = 124 K	Same as (I)..
IV. AT/BP Mode 8T / 5 MA P_{fusion}=150 MW P_{ext}=15 MW	Same as (I) ~31 s w/ DT; ~46 s w/ DD.	Same as (I); $\mathbf{T_{max}}$ TBD	Same as (I) T_{max} TBD	Same as (I)