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Fusion Development Facility

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Abstract— **A Fusion Development Facility (FDF) is proposed to fill the technological gaps between ITER and a Fusion DEMO. FDF is a steady state copper, water-cooled coil machine with a primary goal to test multiple blanket configurations in its lifetime. For efficient remote exchange of the breeding blankets, it is planned to exchange them through vertical motions in large segments. Two principal remote handling approaches are under study, one exchanging large rings and the second poloidal wedge sections. For either scenario, a joint is required in the toroidal coils to allow for the vertical blanket exchange. Two approaches to making the joint are under study along with the respective structural models for reacting the magnetic loads. Initial stress analysis is reported showing the structural feasibility of either concept. Coolant flow is important in such a steady state device and manifold and piping layout concepts are developed as part of the remote handling study. The base machine design parameters are reported including size and cooling requirements. The different machine configuration options are presented which consider the design aspects for the machine including alignment of the first wall and divertor, coolant access, and exchange of the blankets.**

Keywords- Fusion Magnets, Component Development Facility, High Fluence, High Heat Flux, Demountable TF Coils

I. INTRODUCTION

To support the successful development towards a DEMO plant of the ARIES-AT type, a Fusion Development Facility (FDF) is needed. The proposed facility addresses many of the gaps identified by the recent FESAC Planning Panel between ITER, IFMIF, current superconducting tokamaks and DEMO. One of the primary missions for FDF is to close the fusion fuel cycle by producing its own tritium and building a supply to start up DEMO. This requires a full breeding blanket with a tritium-breeding ratio greater than one. The first full blanket installed will utilize the simplest technology possible that produces net tritium with the plan to exchange the blanket for more advanced types in later phases. The design of the tokamak produces a neutron flux of 2 MW/m^2 at the outer midplane or about four times the ITER goal at the midplane. At a duty factor of 0.3, fluences of 3–6 MW-yr/m^2 are produced in ten years of operation as compared with ITER lifetime fluences of 0.3 MW-yr/m^2. The neutron fluence for FDF is on complete blanket structures and includes smaller material volumes (up to 1 m^3) in port sites to enable material irradiation qualification for the initial operation of DEMO.

The design of FDF is based on conservative Advanced Tokamak (AT) physics which will be demonstrated in steady-state burn producing 100–250 MW of fusion power with modest energy gain of Q < 5 in a reasonable sized FDF device (Fig. 1). The targeted continuous run time for FDF is two weeks. The size of FDF is between DIII-D and JET. The D-T operating regime of FDF will provide burning plasma physics understanding to complement the contributions from ITER. More on the FDF mission and physics plans can be found in [1].

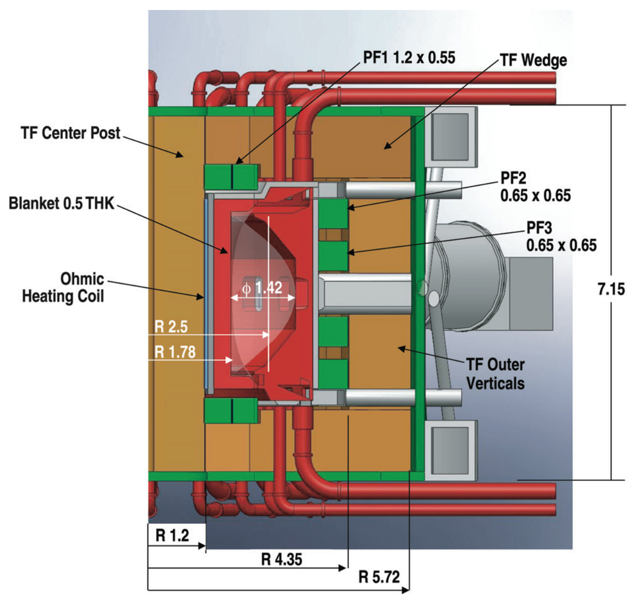

Figure 1. Cross section of FDF with dimensions in meters.

II. DESIGN

A design code was used to optimize the initial design parameters of FDF while meeting the physics requirements of the device. Variables considered included power consumption, coil stresses, wall loading and divertor heat flux. The resulting AT device has a high triangularity, double null divertor plasma with a major radius of 2.5 m and a minor radius of 0.7 m producing 100-250 MW of fusion power. Other design values can be seen in Table 1. The facility power required is targeted to be less than 500 MW with the largest fraction of the power used in the copper magnets. While power consumption is large, the copper magnets allow easier remote maintenance access to the nuclear components inside the coils.

Each of the three magnet systems are water-cooled copper conductors. The TF magnet provides a 6 T field on axis. The central solenoid (CS) is wound around the TF to maximize the volt-seconds of the device. It is the coil closest to the blankets and thus receives the highest dose of all the magnet systems. It is planned that the CS is not a lifetime component and can be replaced. Still, inorganic insulators might be required. This is still being considered and optimized with the shield thickness of the inboard wall. The CS is divided into six separate modules to provide field-shaping capability and to enable plasma start-up. The flux swing of the coil currents during startup was optimized to minimize the power and thus steady state cooling requirements during plasma flattop. There are 25 V-seconds in the solenoid with a flat top power consumption of 90 MW.

TABLE I. FDF MACHINE PARAMETERS

Major radius (m)	2.49
Minor radius (m)	0.71
Plasma elongation	2.31
Aspect ratio	3.5
Plasma current (MA)	6.7
Triangularity	0.71
Fusion power (MW)	246
Wall loading (MW/m^2)	2
Field axis (T)	6
Power to run plant (MW)	500

The toroidal field coil set consists of copper plates bundled into 12 return legs. During the device optimization, the stress in the central column was a critical design limit, keeping it below 275 MPa. The field at the conductor is 12.5 T. A joint in the TF coils enables complete disassembly of the machine for remote replacement of the fusion core much like DIII-D and Alcator C-Mod. Two structural options for the device are under evaluation. In the first option, the large cross section of copper in the TF coils is used in carrying the coil loads with the joint transmitting the forces through a series of sawteeth as proposed by P.H. Rebut [2] (Fig. 2). A large ring outside the TF coils is used to apply a radial preload keeping the joint in compression. The vertical forces on the horizontal legs of the TF coil are transmitted through the joints to the central column and outer TF legs. The second option has sliding joints and a large external structure is used to carry the coil loads (Fig. 3). With the sliding joints, none of the vertical loads are carried by the copper but are transmitted to large caps on the machine and then to side plates. Analysis of the two concepts indicates the feasibility of each and has guided the sizing of the main elements. More detail on the structural concepts and coil joints can be found in [3].

The joints in the TF coils allow the eight poloidal field coils to be inside of the TF coils providing efficient plasma shaping capability. Each PF coil is water-cooled and designed to minimize the steady state current during plasma flattop. The outer four coils are stacked in a vertical column to allow for remote removal of the vessel components without moving the coils. At both the top and bottom of the machine are two coils for divertor control of the double null plasma. The total power in the PF coils is 90 MW. The insulation scheme for the coils is planned to be standard epoxy resin with cyanate ester resins considered if the dose requires higher radiation resistance.

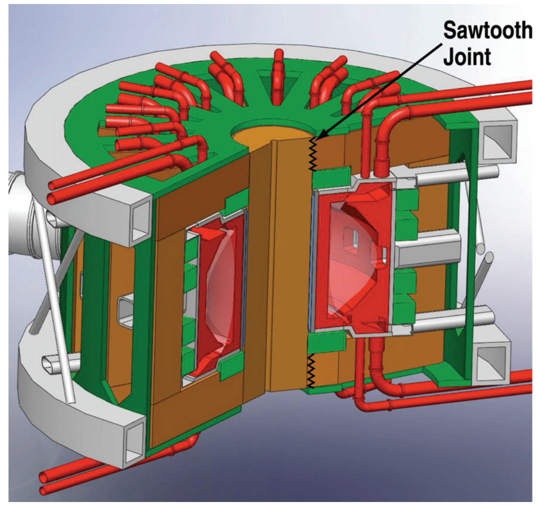

Figure 2. FDF concept with loads transmitted across TF joints.

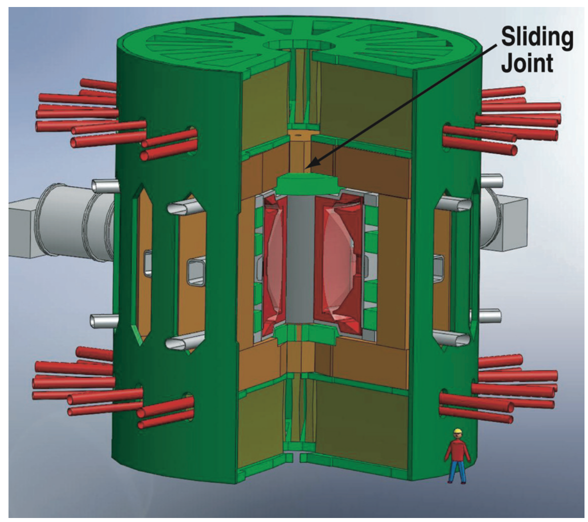

Figure 3. FDF concept with sliding TF joints.

The breeding blankets and shield are the key structural elements of the fusion core. At least 0.5 m thick blanket/shields are required to protect the coils. The machine concept utilizes the inherent strength of the blankets to carry the loads from operation and off-normal events. The vacuum vessel outside of the blanket structure will be primarily a vacuum barrier and not a key structural element of the machine. Dead weight loads are transmitted through the vacuum vessel to the outside supporting structure.

The divertor design is a combination of outer strike point target plates angled at 10 degrees in the poloidal plane coupled with a chamber connected to pumping ducts for particle and ash removal. Materials have not yet been chosen for the divertor.

As development of fusion nuclear science and technology is at the forefront of the FDF mission, breeding blankets are instrumental to the program. It is planned that the first breeding blanket installed will be made of the simplest technology that would provide a tritium breeding ratio greater than 1.0. After the first 10 years of operation, a second blanket type would be installed using a different technology. After its operational period concludes, a third final optimized blanket would be installed and operated to provide data for DEMO. In parallel, utilizing the equatorial ports, advanced blanket concepts or technologies would be installed and tested including electric power and hydrogen production modules. Material samples can be irradiated in test port modules. Two test ports with approximately two cubic meters of material samples could be exposed for 10 years to accumulate fluences of 3–6 MW-yr/m^2.

The coolant requirements for the in-vessel components for FDF are large and routing of the feedlines is considered in the design. The different blanket concepts such as Dual Coolant Lead Lithium (DCLL) or helium cooled ceramic breeders have similar requirements for the number of inlets and outlets and their sizes. The inlets and outlets will come from the top and bottom of the machine, through the TF sections. The coolants will have to be routed through the blankets around the divertor pumping plenums. The coolant pipes can be seen in Figs. 2 and 3.

As currently designed, 60 MW of auxiliary heating power is required. While not finalized, it is planned that a mix of neutral beams, electron cyclotron heating and current drive, and lower hybrid current drive will likely be used. Large equatorial ports are planned for the heating systems.

III. REMOTE HANDLING CONCEPTS

With the complete exchange of the breeding blankets planned twice in the life of the device, remote handling and maintenance concepts can have a large impact on the design. One of the early choices made was to drive toward the jointed TF coil to allow removal of relatively large internal components rather than exchange a large number of smaller components through equatorial ports. With a joint in the TF coils similar to DIII-D or C-Mod, vertical removal of components from the top of the machine is considered, simplifying the remote handling tools and saving space around the machine. With vertical access, two blanket concepts are considered.

The first option is to design the blankets as toroidally continuous rings, segmented in the poloidal direction. Fig. 4 shows a series of blanket segments being removed as continuous rings. To be able to remove the blanket modules, once the primary structure is disassembled, the upper portion of the TF coils is removed, followed by the lifting of the divertor coils. The top of the vacuum vessel along with the coolant pipes is cut and removed leaving access to the top divertor and blanket modules. The module size was considered with the maximum weight of a module being 60 tonnes. The poloidal segmentation could be modified as the design progresses. For cooling of the modules, pipes for the upper rings would come from the top of the machine and the lower modules from the bottom of the machine. The continuous ring structure has several distinct advantages. First and foremost is the alignment of the divertor and first wall components in the toroidal direction. Misalignments and steps can be minimized or even eliminated with the continuous rings. While large currents can be induced in these structures, the rings are inherently strong as hoops and can carry large internal loads. It also minimizes voltage potentials inside the vessel that upon breakdown create unplanned for currents and loads. This design philosophy has been used successfully in DIII-D in both the Advanced and Radiative Divertor projects. The disadvantages of this concept relate primarily to access for minor repairs which could require major machine disassembly if repair was not possible through more traditional remote maintenance access.

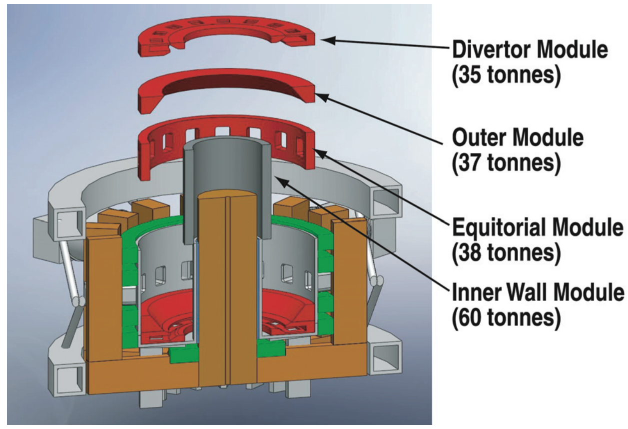

Figure 4. Toroidally continuous ring version of machine. TF coil upper sections are already removed.

The second remote maintenance option for the blanket design is to have poloidal sections that can be removed from the top of the machine (Fig. 5). In this concept, wedges forming 40–45 degree segments are being considered resulting in lifts of approximately 30 tonnes. To remove components, the divertor coils (shifted to outside of the TF coils) are removed and just the TF segments above the module to be replaced are removed. The local portion of the vacuum vessel top is removed and access to the blanket module is gained. The primary benefit in this concept is a smaller portion of the machine is disturbed for a single module failure or maintenance

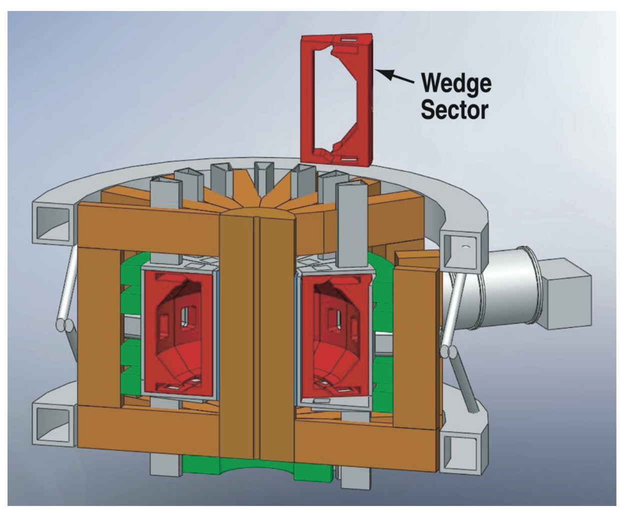

Figure 5. Sector maintence version of FDF with one blanket section shown in the removal process.

event. Also different blanket types could be tested at different toroidal positions under the same poloidal flux. Alignment in this concept is unlikely to reach the high level obtained in the ring concept. Analysis has not yet been completed on the impact of moving the divertor coil outside of the TF turns.

IV. CONCLUSION

A design concept for a new tokamak to fill the gaps towards the development of DEMO has been developed. The primary mission is to close the fusion fuel cycle. Operating for up to 2 weeks, the device, using water-cooled copper magnets allows for multiple exchanges of the breeding blankets. Joints in the TF coils allow for vertical remote maintenance of large internal components and early analysis confirms feasibility of two different structural concepts. Work continues on the development of the concepts.

ACKNOWLEDGMENT

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