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TITLE THE TITAN MAGNET CONFIGURATION

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THE TITAN MAGNET CONFIGURATION†

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Abstract: The TITAN study uses copper-alloy ohmic-heating coils (OHC) to startup inductively a reversed-field-pinch (RFP) fusion reactor. The plasma equilibrium is maintained with a pair of superconducting equilibrium-field coils (EFCs). A second pair of copper EFCs provides the necessary trimming of the equilibrium field during plasma transients. A compact toroidal-field-coil (TFC) set is provided by an integrated blanket/coil (IBC). The IBC concept also is applied to the toroidal-field divertor coils. Steady-state operation is achieved with oscillating-field current drive, which oscillates at low amplitude and frequency the OHCs, EFCs, the TFCs, and divertor coils about their steady-state currents. An integrated magnet design, which uses low-field, low technology coils, and the related design basis is given.

1. INTRODUCTION

The TITAN fusion reactor study[1,2] is exploring the potential of high-power-density operation based on the reversed-field pinch (RFP). The high-power-density goal forces the magnet configuration to be compact while minimizing the recirculating power. Furthermore, steady-state plasma operation has been mandated, resulting in the adoption of oscillating-field current drive (OFCD)[3] as the means to sustain the 18-MA toroidal plasma current, I_ϕ.

Two high-power-density designs were considered: (a) a Li/Li/V (breeder/coolant/structure) poloidal loop configuration (TITAN-I); and (b) a $LiNO_3/H_2O$/HT-9 configuration immersed in a water pool (TITAN-II). The first option uses the integrated-blanket-coil (IBC) concept,[4] wherein currents are driven in the Li breeder/coolant to produce the toroidal magnetic field. The second option uses an aqueous-loop blanket with normal-conducting Cu toroidal-field coils (TFCs) encasing the blanket. The focus herein is the TITAN-I magnet configuration.

The magnet configuration consists of a poloidal-field-coil (PFC) set, a toroidal-field-coil (TFC) set, and a divertor-coil set. No separate coil set is used for OFCD; the PFC, TFC, and divertor coils are oscillated about their steady-state currents to achieve OFCD.[3] The PFC set performs an equilibrium and an ohmic-heating (start-up) function. The equilibrium function requires that a vertical field of appropriate magnitude and index corresponding to the plasma current and beta[-7] be imposed over the plasma

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cross section. The ohmic-heating function provides the poloidal-flux swing required to establish the steady-state plasma current, which is then subsequently sustained by OFCD. Since the ohmic-heating function is required only during start-up and the equilibrium function is required continuously, the PFC set is naturally, but not necessarily, split into two coil sets: an equilibrium-field coil (EFC) set and an ohmic-heating coil (OHC) set.

The TFC system should generate a uniform toroidal field and ideally would be a continuous toroidal shell. The IBC TFC design of TITAN-I, with minor but necessary gaps for first-wall coolant channels, approaches such an ideal shell. The proximity of these gaps to the plasma raises concerns about magnetic-island formation caused by field ripple. Consequently, toroidal-field ripple, which is a design issue for discretized TFCs as in TITAN-II, requires examination. In addition the displacement of a portion of the IBC TFCs by the divertor couples the TFC and divertor designs.

A magnetic divertor is used in conjunction with a highly radiating plasma to manage sputtering at reasonable levels. The divertor nulls the minority toroidal field to minimize its effect on the plasma and the divertor coil currents. The divertor coils are designed to minimize toroidal-field ripple.

2. RESULTS

2.1. Equilibrium-Field Coils (EFCs)

Continuous EFC operation suggests superconducting EFCs. Superconducting EFCs require thicker shields than normal-conducting EFCs; hence, more current is needed to produce the same field resulting in a more massive and expensive coil set. However, superconducting EFCs were found to be slightly less expensive than normal-conducting EFCs[8] and were adopted. An additional constraint imposes the use of a single pair of superconducting EFCs which do not interfere with vertical or horizontal movement of the torus assembly during maintenance procedures.

The steady-state EFC currents are determined by equating the on-axis EFC vacuum field to the vertical field required for toroidal equilibrium.[5,6] The position of the EFCs is determined by constraining the decay-index, n, to the range $0 < n < 0.65$, which is required for a circular plasma.[6,7] The resulting EFC design is shown is Fig. 1 and the associated parameters are given in Table I.

2.2. Ohmic-Heating Coils (OHCs)

A number of OHC configurations were examined ranging from the "close-fitting" OHC configuration shown in Fig. 1, which maximizes the coupling between the OHCs and the plasma, to a "vertical stack" configuration with one stack positioned inboard of the torus and the other outboard which maximizes the unobstructed vertical access. The OHC configurations were optimized for either inductive coupling or vertical access using the code CCOIL.[9,10]

The locations of the OHCs are determined in CCOIL by first specifying an arc upon which the coils are to be arrayed. The Fournier coefficients for a series representation of the current distribution on the arc that excludes flux from the entire plasma cross section are then determined. Assuming equal-current coils to facilitate series electrical connection of the coils, the current distribution is integrated along the arc to yield the OHC current-center location.

The single-turn back-bias and forward-bias OHC currents, I_{OH}^{-} and I_{OH}^{+}, are determined by imposing inductive flux conservation and ignoring the plasma resistive losses. An additional constraint of a bipolar current swing is imposed to minimize the energy-storage and power-handling requirements. The mutual inductances used for flux conservation are estimated using an expression for two coaxial hoops.[11] The individual coils are simulated with 100 hoops. The plasma, however, is simulated with a single-hoop current, which is positioned in the equatorial plane at a major radius that includes a Shafranov shift[5] from the plasma centroid. The calculation of the single-turn mutual inductance involves a summation over each hoop. The single-turn self-inductances also are determined by application of the mutual-inductance formula. The resulting singularity is replaced with the self-inductance of a wire of finite minor radius[11] equal to the separation between the filaments. The plasma self-inductance is expressed as a sum of an external inductance, $L_{p,ex}$, and an internal inductances, L_p, (i.e., $L_p = L_{p,in} + L_{p,ex}$). The external inductance is taken to be that for a wire[11] with the same dimensions as the plasma. The internal inductance is derived from results of a one-dimensional MHD equilibrium calculation.[8]

Two additional constraints are placed on the OHC design due to plasma breakdown. The first constraint is on the maximum level of the stray vertical field during breakdown. Assuming a 100-mT toroidal field prior to plasma breakdown and field-line confinement for a minimum of one toroidal revolution to establish a toroidal plasma current, a maximum value of 2.45 mT for the stray vertical field results. A second constraint is that the OHC set exhibit a field null within the plasma chamber to provide a current-formation channel.

The above algorithm was used to analyze the "close-fitting" configuration shown in Fig. 1, the previously described "vertical stack" configuration, and an intermediate "pill box" configuration consisting of a vertical stack positioned inboard of the torus and coils above and below the plasma as shown in Fig. 3. Only the "close-fitting" and pill box configurations meet the breakdown constraints as is demonstrated in Fig. 2 for the "close-fitting" configuration. The "close-fitting" configuration achieves better coupling with the plasma than the pill box-configuration, as is evidenced by the OHC current swings of 48 MA and 82 MA, respectively. Based on the coupling efficiency, the close-fitting configuration of Fig. 1 and Table I was adopted for the PFC design.[1]

Cooling of the OHCs has led to a re-examination of the OHC configuration in Fig. 3. Helium was chosen as the OHC coolant medium over water to minimize the consequences of the accidental mixing of the OHC coolant with the blanket coolant, lithium. The helium coolant requires large headering to the coils that is best accommodated by the "pill box" configuration of Fig. 3. Both of the configurations in Figs. 1 and 3 meet the design constraints and illustrate the tradeoff of minimum OHC current swing and minimum coil mass that must be moved for vertical maintenance.

The second pair of copper-alloy EFCs also is shown in Fig. 3. The equilibrium field during plasma transients need only be trimmed a few percent so that the position of the secondary EFCs is dictated by maintenance rather than magnetics considerations.

2.3. Toroidal-Field Coils (TFCs)

The major goal for the TFC design of RFPs is the achievement of a minimum toroidal-field ripple. Toroidal-field ripple produces magnetic islands within the edge-plasma region. Particles and energy flow freely within this island structure. Plasma confinement then is degraded according to the island size. To ensure that confinement is not adversely affected by the ripple, the radial extent of the islands is required to be smaller than the radial distance between the reversal surface and the plasma surface; this region may be primarily responsible for confinement in an RFP.[12]

An estimate of the magnetic-island size produced by toroidal-field ripple is given by the following formula for the radial thickness of an island:[13]

$$\Delta r = r\left[\frac{r\Delta B_R}{n B_\theta (dq/dr)}\right]^{\frac{1}{2}} ,$$

where r is the minor radius of the resonant surface, ΔB_R is the amplitude of the radial magnetic-field perturbation, n is the toroidal mode number of the resonant surface, B_θ is the poloidal field at the resonance, and the derivative of the safety factor, dq/dr, is evaluated at the resonant surface. The toroidal mode number of the resonance is the number of TFCs, N_{TF}. Conventional TFC designs for RFP reactors[9] strive for island widths $\Delta r \lesssim 0.1\ r_p$ which are achieved with ripples, ΔB_R, of a few mT produced by $N_{TF} = 25$

TFCs. The IBC TFC design shown in Fig. 4 and described in Table II is analogous in this context to a conventional design with $N_{TF} \sim 10^3$.

The ripple for the IBC TFC is calculated to be a few μT based on two-dimensional field-line tracings at the plasma surface using the three-dimensional vacuum-field magnetics code, TORSIDO,[14] with only the TFCs simulated. Consequently, the island width in the IBC design is two orders of magnitude smaller than the island constraint requires. Similarly, the field errors produced by the gap to permit the ingress/egress of the first-wall coolant channel and by the leads are a few μT and does not appear to present a problem.

2.4. Divertor Coils

The divertor design approach adopted here builds on the results of Ref. 15. The TORSIDO code is also used to compute the magnetic topology, but only in two-dimensions confined to the equatorial plane. The more economical two-dimensional field-line tracings were found to reproduce the three-dimensional field-line tracings in front of and at the side of the nulling coil, which is the region of interest for an open divertor. The open divertor was judged preferable to a closed divertor,[16] because the closed divertor concentrates heat flux at the divertor neutralizer plate whereas the open divertor diffuses heat flux at the plate and the closed divertor cannot entrain impurities to radiate any of the power flowing into the divertor. The diffusing of the heat flux in the open divertor is the result of the expansion of the distance between two field lines in the vicinity of the null. The expansion/contraction of field lines relative to their spacing at the divertor midplane is measured by a flux-surface expansion factor. An accurate calculation of the flux-surface expansion factor assumes that the flux in the equatorial plane is constant along a field line; the flux-surface expansion factor then is the ratio of the local field (toroidal and radial) to the field at the divertor midplane. The calculation of connection lengths for input into edge plasma models[17] from the divertor midplane to either the null, L_N or neutralizer plate, L_P, accounts for three-dimensional effects with the additional simulation of the plasma current and the EFCs.

Using the two-dimensional field-line tracings, the divertor configuration shown in Fig. 4 and described in Table III was obtained using the IBC concept. The nulling coil conducts a current sufficient to null the toroidal field and is located close to the plasma on the inboard side to minimize the divertor-coil current and obtain acceptable flux-surface expansion factors as determined by radial sensitivity studies. The outboard locations of all divertor coils are determined by requiring an inboard/outboard symmetry of the reversal surface to minimize magnetic island width.[15] The divertor coils also are constrained to remain inside the IBC TFC envelope so as not to displace shielding. The flanking coils carry the same total current as the nulling coil to localize the effect of the nulling coil and are positioned radially in the middle of the TFC envelope to minimize ripple. A pair of trim coils is needed to conduct a current equal to that in the portion of the IBC TFC tube bank displaced by the divertor. The trim coils are located radially as far as possible from the nulling coil within the TFC envelope to minimize the divertor currents. The trim coils are positioned toroidally to minimize ripple. The toroidal extent of the divertor is nearly the same as that of the design in Ref. 15 and should produce the same acceptable magnetic island width in a full three-dimensional simulation. The number of divertors (three) was determined by the calculated heat flux on the first wall and neutralizer plate.[18]

5. CONCLUSIONS

Designs for each coil set (EFCs, OHCs, TFCs, and divertor coils) for a compact RFP reactor have been presented. These coil designs would operate with low fields ($\lesssim$ 8T for copper-alloy OHCs and $<$ 6T for superconducting EFCs), stresses ($\lesssim$ 200 MPa), and ripple ($\Delta B_R/B_\theta \le 10^{-6}$, giving magnetic island widths $\Delta r < 0.01$ m). The design constraints imposed generally were met by a wide margin in a range of design options, with the exception of the vertical stack OHC configuration. This design margin facilitates the integration of the coil sets into the overall reactor torus design. Additionally, non-magnetic considerations such as safety (e.g., He-cooled OHC), economics (e.g., IBC TFCs which recover Ohmic losses), and maintenance (e.g., non-interferring EFC locations) were included as major elements/constraints of the overall design.

TABLE I. PFC Parameters for TITAN-I

Parameter	Value
EFC current (MA)[a]	17.8
EFC volume (m^3)	39.7
EFC mass (tonne)	292.1
EFC joule losses (MW)[a]	(378.4 NC)(0.0 SC)
EFC peak field (T)[a]	5.9
EFC current density (MA/m^2) [a]	18.3
Vertical field index	0.16
OHC current (MA)	
· back bias	-32.9
· forward bias	15.1
OHC volume	40.9
OHC mass (tonne)	301.2
OHC joule losses (MW)	(68.1[c])/321.6[b]
OHC von Mises stress (MPa)[b]	215.6
OHC peak field (T)[b]	8.3
OHC current density (MA/m^2)[b]	(12.2-24.9)
OHC stray vertical field (mT)[b]	1.25($<$ 2.45[d])
PFC transparency (%)	67.2

(a) Steady-state values.

(b) Back-bias values.

(c) Forward-bias values.

(d) Stray vertical field constraint.

TABLE II. TOROIDAL-FIELD-COIL PARAMETERS

Parameter	Value
Current per trisector (MA)	2.08
Toroidal field, $B_\phi(T)$	0.36
Number of tubes per trisector	975
Average current per tube (kA)	2.13
Tube inner area ($10^{-3}m^2$)	1.40
Average current density	1.52

Tube data by row:

r_c (m)	Number per trisector	I_c (kA)	$\langle j_c \rangle$ (MA/m^2)
0.706	162	2.22	1.54
0.752	163	2.19	1.54
0.797	162	2.15	1.53
0.843	163	2.11	1.52
0.888	162	2.08	1.52
0.934	163	2.04	1.51

Resistivity, $\eta(\mu\Omega m)$	0.353
Total power, $P(MW)$	24.0
Blanket coverage	0.887

TABLE III. DIVERTOR COIL PARAMETERS

	Nulling	Flanking	Trim
Number per trisector	1	2	2
Toroidal angle (°)	0	5.72	2.94
Major radius (m)	3.95	3.94	3.90
Minor radius (m)	0.855	0.860	0.900
Current per coil (kA)	164.	82.0	131.
Average current density[b] (MA/m^2)	27.5	29.1	20.8
Resistivity, $\eta(\mu\Omega m)$	0.353	0.353	0.353
Power per coil(MW)	11.8	6.24	7.38
Lithium volume per coil ($10^{-2}m^3$)	3.20	1.52	3.56
Total volume per coil ($10^{-2}m^3$)	3.62	1.82	4.01

Total average current density (MA/m^2)	24.4
Conductor filling fraction, λ	0.874
Total power (MW)	117.
Midplane-to-null distance, $L_N(m)$	72.7
Midplane-to-plate distance, $L_P(m)$	74.6
Peak flux expansion factor (in/outboard)	2.27/4.23

FIGURE CAPTIONS

Fig. 1. A cross-sectional view of the "close fitting" poloidal-field coil set for the TITAN-I design. The locations of the IBC TFCs,

the first wall, reflector, shield, and the plasma are shown in addition to the EFCs and the OHCs.

Fig. 2. The stray-vertical-field profile in the equatorial plane for the close-fitting OHC configuration shown in Fig. 1. Also shown are the bands for the allowed vertical field when the field null is on axis (clear) and when the field null is off axis (cross-hatched) as is the case for the close-fitting configuration.

Fig. 3. A cross-sectional view of a "pill box" OHC configuration for the TITAN-I design. The locations of the IBC TFCs, the first wall, reflector, shield, and the plasma are shown in addition to the superconducting EFCs and the OHCs. Also shown are the secondary EFCs.

Fig. 4. Equatorial-plane view of the IBC TFC tube bank and the divertor coils. Also shown are field-line tracings at the reversal surface and at several locations spanning the SOL.

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