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IMPACT OF A POLOIDAL DIVERTOR IN IGNITION TOKAMAK DESIGN*

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ABSTRACT

System design studies were performed to assess the effect of assuming a poloidal divertor instead of a limiter as a means of impurity control for ignition tokamak configurations. Results show that for the nominal Tokamak Fusion Core Experiment (TFCX) device with superconducting TF coils, a feasible poloidal divertor configuration can be obtained without increasing the major radius. In the TFCX nominal copper TF coil device, however, field limits at the PF coils are exceeded when the effects of asymmetry associated with a poloidal divertor are included. It was found that a 12% increase in the major radius of this device is necessary to simultaneously satisfy the plasma-shaping requirements of a poloidal divertor and the magnetics constraints at the superconducting PF coils.

INTRODUCTION

Studies were performed to assess the effect of assuming a single null poloidal divertor instead of a limiter for impurity control for ignition tokamak configurations. The assumptions used in the preconceptual design of the Tokamak Fusion Core Experiment[1] (TFCX) were followed, including constraints imposed by magnetohydrodynamic (MHD) equilibrium considerations and poloidal magnetics requirements. Modifications in the vertical build are necessary to include the structure associated with a divertor channel. The redistribution of coil currents due to poloidal asymmetry and a magnetic separatrix may imply an increase in machine major radius.

Preconceptual design studies have produced four candidates for the TFCX, two assuming superconducting (SC) TF coils, and two copper TF coil designs. In each case, (copper and SC) a relatively nominal, and higher performance (leading to smaller size) set of TF coil assumptions were considered. In all but the smallest copper TF coil device, superconducting poloidal field (PF) coil systems are employed. Given that these four devices are based on a certain set of engineering and physics assumptions, including a limiter for impurity control, it is of interest to perform sensitivity studies about these design points. One such study is to examine the impact of a poloidal divertor on a machine configuration.

The design configuration for a given TFCX option is the result of an iterative process involving the Fusion Engineering Design Center (FEDC) tokamak systems code[2], the FEDC MHD equilibrium code[3], and the Electromagnetic Field, Force and Inductance (EFFI) magnetics code[4] from Lawrence Livermore Laboratory. The tokamak systems code is used to establish a design point, or a set of plasma geometry and performance parameters consistent with TFCX assumptions. From this, a configuration drawing can be made to establish access and maintenance constraints on PF coil locations. A PF coil system is integrated into this configuration and the equilibrium code is used to compute coil currents and determine if the system is feasible with respect to volt-seconds and plasma shape requirements. The magnetics code determines feasibility with respect to field limits at the superconducting PF coils. The objective of this analysis is to obtain a device of minimum size that simultaneously satisfies these criteria. This minimum is typically defined by whether the solenoid is at an adequate radius to produce the necessary volt-seconds, or by the fields at critical points on the Ohmic heating (OH) solenoid or inboard shaping field (SF) coils. This process is described in the form of a systems flowchart in Fig. 1.

A single null poloidal divertor with given channel length (from separatrix to neutralizer plate) impacts the result of this

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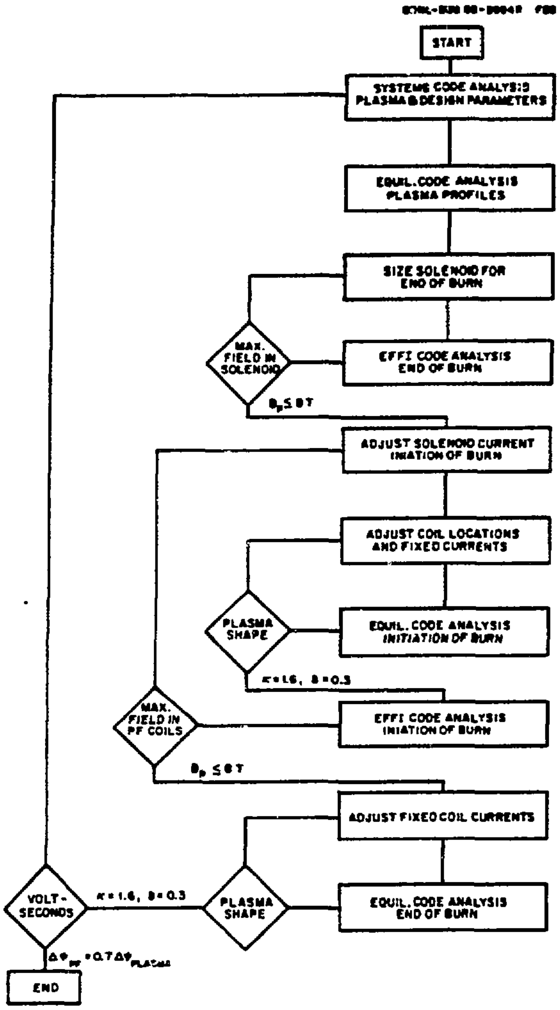

Fig. 1. The systems design process applied in the preconceptual design analysis of the TFCX.

systems analysis in several ways. An obvious increase in the vertical build of the machine below the midplane is necessary to include the divertor structure. This further removes SF coils from the plasma boundary and increases their currents. Poloidal asymmetry and the need to position a separatrix at a given location again greatly increase these SF coil currents. If the minimum size of a limiter device is set by field strength considerations at these SF coils, then these current increases will force the corresponding divertor device to a larger major radius. In this paper, machine parameters are defined for poloidal divertor devices with TFCX nominal superconducting and copper TF coil assumptions.

SYSTEMS ANALYSIS

The essential mission of the TFCX is to achieve ignition and long pulse equilibrium burn. For a given major radius R, the tokamak systems code arrives at a set of plasma parameters based on this objective and the physics assumptions that have been made to achieve this objective. To enable a comparison of different TF coil options, the devices have been designed with common ignition, pulse length, heating, and rf current drive physics assumptions. These are implemented in a zero dimensional physics module in the systems code. Some of the assumed TFCX parameters are summarized in Table 1.

Table 1. TFCX device design criteria.

Parameter	Value
Elongation, κ	1.6
Triangularity, δ	0.3
Safety Factor, q	2.4
Ignition Parameter, C_{ig}	1.5
Max. TF Ripple (%)	1.5
Plasma Temperature (keV)	10.0
Max. Field at PF Coils (T)	8.0
PF Coil Current Density (kA/cm^2)	2.0

The physics model in the systems code analysis of the TFCX includes an ignition parameter defined by

$$C_{ig} = 0.295\ \beta\ \tau\ B^2 \quad , \qquad (1)$$

where β is the plasma volume averaged beta, τ the energy confinement time in seconds, and B the field at the major radius in tesla. Plasma beta is given by an assumed ideal MHD stability limit:

$$\beta = \frac{0.2}{Aq}\left(\frac{1 + \kappa^2}{2}\right) , \qquad (2)$$

where A is the aspect ratio and q the safety factor at the plasma edge. For TFCX limiter studies, the plasma shape has been fixed at elongation $\kappa = 1.6$ and triangularity $\delta = 0.3$, and the safety factor prescribed to be $q = 2.4$. These constraints have been used in the systems analysis of divertor plasmas also, but they are necessarily relaxed in the equilibrium analysis of the divertor cases. A Mirnov confinement scaling is used, given by

$$\tau = 0.39\ a\ I_p \quad , \qquad (3)$$

where a is the minor radius in meters and I_p the plasma current in mega-amperes.

For a fixed major radius, a 10 T field at the TF coil inboard leg and MHD equilibrium considerations imply that B, I_p, and β depend on the minor radius a, which is then fixed by the TFCX assumption $C_{ig} = 1.5$.

The operating scenario calls for rf current ramp-up at low density, followed by an inductive burn phase where the volt-seconds provided by the PF system are equal to 70% of the plasma internal flux during burn.

CONFIGURATION ANALYSIS

For a set of plasma global parameters from the systems code, a configuration drawing defines the maximum radial location of the OH solenoid, and places spatial constraints on PF coil locations.

In the TFCX analysis, a PF system consists of an OH solenoid, four shaping field (SF) coils, and two outboard equilibrium field (EF) coils. The SF and EF coils are superconducting as is the OH solenoid in the nominal TF coil cases considered here. All PF coils are positioned external to the TF coils and associated structure, and the outboard EF coils are constrained in elevation by a maintenance concept that assumes horizontal access to the torus. This PF coil concept is shown in Fig. 2. The fields of the three coil sets are coupled, a concept that reduces total ampere-turns and allows the magnetic energy stored in the system to assist in current buildup while providing for MHD equilibrium and plasma shaping.

EQUILIBRIUM ANALYSIS

For fixed OH solenoid currents at the start and end of a burn pulse, fixed plasma parameters, and possibly some subset of the SF coil currents fixed, MHD equilibrium calculations relate the remaining SF and EF coil currents to the prescribed plasma shape and provide an estimate of the available volt-seconds. The FEDC equilibrium code computes a free-boundary solution of the Grad-Shafranov equation, including a regularization method to estimate coil currents that best reproduce a given plasma boundary shape, as described in Ref. 3. The plasma current density used in the equilibrium calculations is given in terms of the plasma pressure profile (P) and the toroidal field function (F):

$$J_\phi = r \frac{dP}{d\psi} + \frac{F}{\mu r} \frac{dF}{d\psi} , \tag{4}$$

where r is the radius and ψ the poloidal magnetic flux function. This current density is scaled during a calculation to fix the plasma current I_p. The profile functions used in this study are

$$\frac{dP}{dx} = P_o \frac{e^{-Ax} - e^{-A}}{e^{-A} - 1} \tag{5}$$

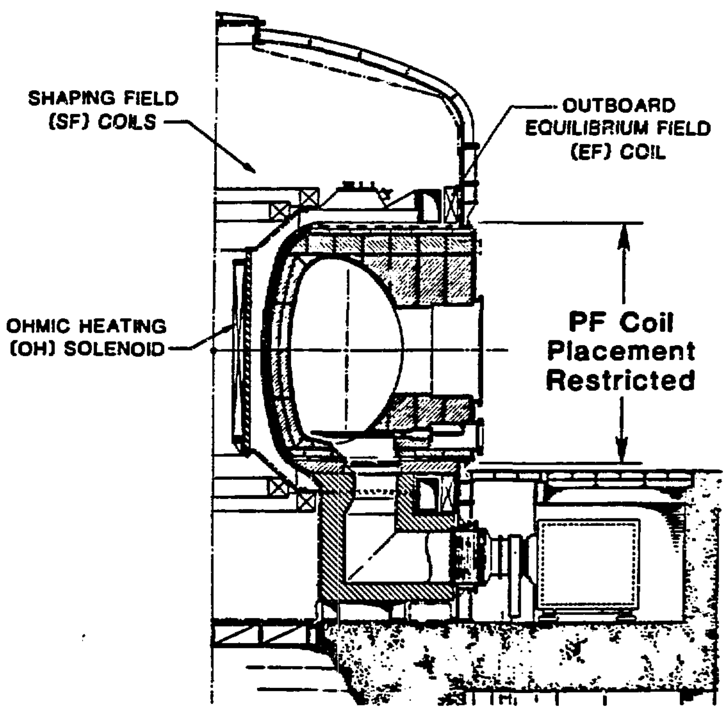

Fig. 2. The TFCX nominal superconducting configuration.

and

$$\frac{dF^2}{dx} = 2\pi R_o^2 P_o \left(\frac{1}{\beta_J} - 1\right)\frac{e^{-Bx} - e^{-B}}{e^{-B} - 1} \quad , \quad (6)$$

where $x = (\psi - \psi_o)/(\psi_L - \psi_o)$ and ψ_o, ψ_L are the axis and limiter values of the poloidal flux, respectively. The parameters A, B, and β_J, together with the current I_p, are typically adjusted in a sequence of equilibrium calculations to produce profiles giving the correct values of q(axis), q(edge), and β. For TFCX values of β and q, $\beta_J \sim 0.5$–0.6, and A and B are in the range -3.0 to -4.0.

The flux provided by the PF system, $\Delta\psi_{PF}$, is computed from two equilibrium calculations (at the start and end of burn) by

$$\Delta\psi_{PF} = \sum_i \Delta\,(M_{ip} I_i) \quad , \quad (7)$$

where M_{ip} is the mutual inductance between the i^{th} coil and the plasma and is computed in the FEDC equilibrium code as

$$M_{ip} = \sum_j M_{ip_j} J_{\phi_j} / \sum_j J_{\phi_j} \quad , \quad (8)$$

where J_{ϕ_j} is the current density at the j^{th} node in the discrete distribution J_ϕ, and M_{ip_j} is the mutual inductance between axisymmetric current filaments at the center of coil element i and at node j. The model therefore accounts for changes in inductance due to a shift in the current profile.

TFCX NOMINAL SUPERCONDUCTING DEVICE

For the nominal superconducting device, the peak field at the TF coil is 10 T in a force flow conductor, resulting in a vacuum toroidal field of $B_t = 3.7$ T at the major radius $R = 4.08$ m. In the equilibrium analysis of the divertor option, the TFCX plasma shape requirements were modified to reflect asymmetry and to position the null point relative to the divertor channel. This was accomplished with an elongation $\kappa = 1.7$ and a triangularity $\delta = 0.4$ below the midplane. Instead of obtaining a prescribed flux surface averaged value of q at the plasma boundary, as in the TFCX limiter devices, the plasma current, pressure profile, and toroidal field function were set to be consistent with the corresponding symmetric equilibrium at the same design point.

The maximum current in the outboard SF coil (occurring at the start of burn) below the midplane increased from 3.6 MA in the limiter device to 13.0 MA in the divertor device, but the coil was at a sufficiently large radius to satisfy the field condition $B \leq 8$T. Volt-second requirements in the divertor device were met with a small adjustment in the solenoid current, and field limits at the solenoid were not exceeded. As a result, a feasible divertor configuration was obtained at the same major radius as the limiter device. Table 2 lists the design parameters for the nominal superconducting limiter and divertor options. The total mega-amper-turn-meters, a cost parameter for superconducting PF systems, is also given in Table 2.

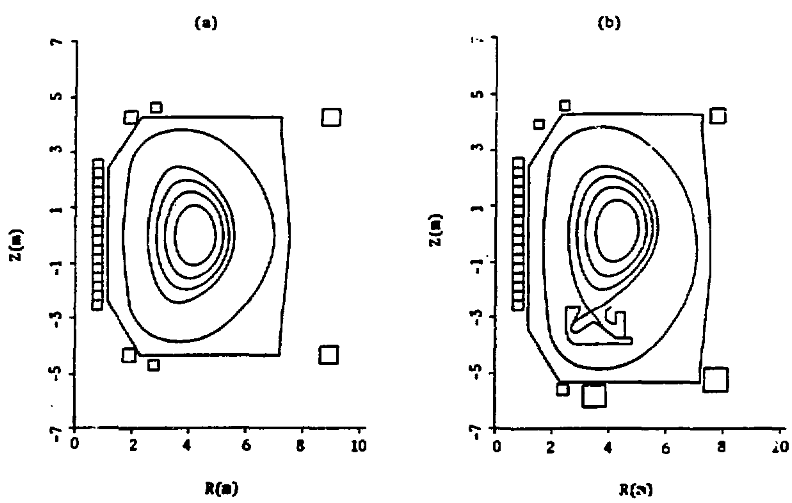

Fig. 3. The TFCX nominal superconducting a) limiter and b) divertor coil configurations.

Figure 3 is an elevation view of the nominal superconducting coil systems where the size of the PF coils are proportional to their current at the end of burn.

Table 2. TFCX nominal superconducting limiter and divertor device parameters.

Parameter	Limiter	Divertor
Major Radius (m)	4.08	4.08
Minor Radius (m)	1.52	1.52
Elongation	1.6	1.7
Upper Triangularity	0.3	0.3
Lower Triangularity	0.3	0.4
Field on Axis (T)	3.7	3.7
Plasma Current (MA)	11.2	11.2
Beta (%)	5.5	5.1
PF MA-m (start of burn)	191.3	241.8
PF MA-m (end of burn)	187.6	237.7
PF Volt-seconds (Wb)	23.9	23.9

TFCX NOMINAL COPPER DEVICE

The TF coil for the TFCX nominal copper option uses plate copper coils made of oxygen free, high-conductivity copper[1]. The vacuum toroidal field of the limiter device is 3.9 T at a major radius R = 3.55 m. In considering a copper divertor device, the TFCX equilibrium plasma shape and safety factor assumptions were modified in the same manner as in the superconducting divertor configuration.

In the nominal copper limiter device, the inner SF coils are at a smaller radius than in the nominal superconducting machine and require large currents at the start of burn (9.6 MA) in order to achieve triangularity $\delta = 0.3$. These coils and the upper and lower ends of the solenoid only marginally satisfy PF coil field limits at the start of burn. In the corresponding divertor machine, current in the lower SF coils shifts toward the outer SF coil to position the separatrix, but the upper inboard coil current is still rather large to maintain triangularity. Field limits at the SF coils were exceeded in a divertor configuration at the same major radius as the limiter device. Several iterations on the major radius produced a feasible nominal copper divertor configuration at R = 3.96 m. Table 3 lists the design parameters for the limiter and divertor options, together with the total mega-ampere-turn-meters in the PF system at the start and end of burn. Figure 4 is an elevation view of the nominal copper device coil systems.

Table 3. TFCX nominal copper limiter and divertor device parameters.

Parameter	Limiter	Divertor
Major Radius (m)	3.55	3.96
Minor Radius (m)	1.37	1.58
Elongation	1.6	1.8
Upper Triangularity	0.3	0.3
Lower Triangularity	0.3	0.3
Field on Axis (T)	3.9	3.4
Plasma Current (MA)	11.1	12.0
Beta (%)	5.5	5.7
PF MA-m (start of burn)	185.4	265.9
PF MA-m (end of burn)	169.3	309.1
PF Volt-seconds (Wb)	22.1	25.5

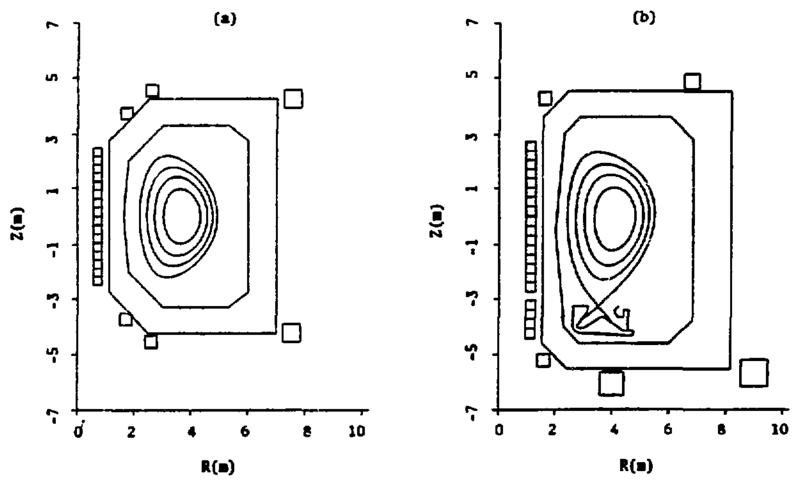

Fig. 4. The TFCX nominal copper a) limiter and b) divertor coil configurations.

CONCLUSIONS

Because of the larger plasma radius, and therefore larger SF coil radii, a divertor design for the nominal superconducting device was possible at the same major radius (R = 4.08 m) as the limiter device. Because of their smaller radius, fields at the solenoid and SF coils of the nominal copper device exceeded 8T when currents were increased to establish a separatrix, forcing the divertor machine design to increase in major radius from R = 3.55 m to R = 3.96 m. The modifications in the PF coil systems due to positioning a null point and the increased vertical build associated with the divertor resulted in a 25% increase in the mega-ampere-turn-meters in the nominal superconducting machine. The corresponding increase in mega-ampere-turn-meters was 40% at the start of burn to 80% at the end of burn for the nominal copper device, largely due to the increase in machine size.

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