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MHD EQUILIBRIUM METHODS FOR ITER PF COIL DESIGN AND SYSTEMS ANALYSIS

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Fusion Energy Division

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CONTENTS

ABSTRACT	v
1. INTRODUCTION	1
2. THE EQUILIBRIUM PACKAGE HEQ	1
2.1 THE EQUILIBRIUM CALCULATION	1
2.2 THE COIL CURRENT CALCULATION	2
2.3 PROFILE FUNCTIONS	4
2.4 EXAMPLE. HIGHLY ELONGATED ITER SOLUTION	5
3. APPLICATION TO SYSTEMS ANALYSIS OF ELONGATION DEPENDENCE	5
3.1 SYSTEMS CODE POLOIDAL MAGNETICS MODEL	5
3.2 REFERENCE EQUILIBRIA FOR ELONGATION STUDY	7
3.3 IMPACT OF ELONGATION ON MINIMUM-COST DESIGNS	10
4. OPTIMIZATION OF COIL CURRENT DISTRIBUTION WITH VEQ	12
5. SUMMARY	15
REFERENCES	16

ABSTRACT

Two versions of the Fusion Engineering Design Center (FEDC) free-boundary equilibrium code designed to compute the poloidal field (PF) coil current distribution of elongated, magnetically limited tokamak plasmas are demonstrated and applied to the systems analysis of the impact of plasma elongation on the design point of the International Thermonuclear Experimental Reactor (ITER). These notes were presented at the ITER Specialists' Meeting on the PF Coil System and Operational Scenario, held at the Max Planck Institute for Plasma Physics in Garching, Federal Republic of Germany, May 24-27, 1988.

1. INTRODUCTION

Several poloidal field (PF) coil design issues are identified when assessing the benefits of high plasma elongation in an ITER device. Among these are (1) the sensitivity of PF ampere-turns and coil energy to specific constraints on the plasma shape and profiles and (2) the optimal placement of PF coils subject to assumed operating scenarios.

Two versions of the FEDC free-boundary equilibrium code now under development at Oak Ridge National Laboratory (ORNL), HEQ and VEQ, are used to solve several PF design problems related to these issues. Recent efforts have been directed toward problems in the analysis of highly elongated, magnetically limited plasmas, such as X-point and divertor strike-point placement and control.^{1,2} In this report, we describe the capabilities of HEQ and discuss the major options available for computing PF coil currents. Applications of HEQ and VEQ in the area of plasma elongation trade-off studies are presented.

2. THE EQUILIBRIUM PACKAGE HEQ

2.1 THE EQUILIBRIUM CALCULATION

In HEQ, a free-boundary equilibrium code is called from the software package HYBRD1,³ which is designed to solve N nonlinear equations in N variables. In this application, HYBRD1 is used to determine coil currents consistent with a set of equilibrium shape and volt-second constraints. The equilibrium code uses relatively standard methods to solve the Grad-Shafranov equation for the poloidal flux function in cylindrical coordinates on a rectangular mesh. Similar methods have been treated in detail elsewhere (e.g., ref. 4) and will only be outlined here:

1. Outer Loop: For appropriate initial estimates, an outer loop scales the solution to fix the plasma current and computes boundary values of the flux function based on a knowledge of the plasma and coil current distributions.
2. Inner Loop: Given the boundary conditions, an inner loop solves the nonlinear equilibrium equation by iteratively computing the limiting flux values at the magnetic axis and plasma edge, evaluating the plasma current density from

assumed plasma pressure and toroidal magnetic flux functions, and inverting the elliptic operator using a direct method.⁵

2.2 THE COIL CURRENT CALCULATION

HEQ has several options for the specification or calculation of coil currents. Subsets of coil currents may be:

1. Designated to have fixed current.
2. Related linearly to the external poloidal flux at prescribed plasma boundary points. These currents are computed by solving a least-squares approximation problem with a regularization term. This calculation is part of the inner loop in the solution of the equilibrium equation.
3. Related through a set of nonlinear equations to constraints on plasma X-point coordinates, divertor strike-points, PF volt-seconds, or coil energy. With this option, the currents are treated as fixed in the equilibrium solution, and HYBRD1 calls the equilibrium subprogram iteratively to solve the equations for the designated coil currents.

A given HEQ calculation, therefore, may have coil currents that are fixed, computed within the equilibrium loops, or determined by HYBRD1.

For option 2, if $\mathbf{x}_i^p = (R_i^p, Z_i^p)$, $i = 1, \dots, N^p$ is a set of points describing a desired plasma boundary shape and including a limiter point $\mathbf{x}_L = (R_L, Z_L)$ and if $\mathbf{x}_j^c = (R_j^c, Z_j^c)$, $j = 1, \dots, N^c$ is a set of coil locations, we solve the approximation problem

$$\sum_i \left[\sum_j G(\mathbf{x}_i^p, \mathbf{x}_j^c) I_j - \psi^e(\mathbf{x}_i^p) \right]^2 + \alpha \sum_j I_j^2 = \text{minimum} \quad (1)$$

for the variable coil currents I_j . G relates the flux at plasma boundary point i to a unit current in coil j . The desired external flux is determined at every inner equilibrium iteration by

$$\psi^e(\mathbf{x}_i^p) = \psi_L + \psi^p(\mathbf{x}_i^p), \quad (2)$$

where $\psi_L = \psi(\mathbf{x}_L)$ is the limiter flux, and the flux due to the plasma current distribution ψ^p is determined by interpolation in the two-dimensional (2D) mesh

of known values. Thus, for this option, the coil current I_j depends largely on coil location, plasma boundary points, and the regularization parameter α , which gives some weight to minimizing total ampere-turns.

Option 3 may be used, for example, to constrain the separatrix flux surface of a symmetric, divertor plasma boundary to pass through two points on the midplane, $(R_0 - a, 0)$ and $(R_0 + a, 0)$, and to intersect prescribed inner and outer divertor strike points, (R_I, Z_I) and (R_O, Z_O) , as shown in Fig. 1. Here the equilibrium problem is solved—assuming that the plasma is limited by a poloidal separatrix—and the current in one pair of outboard equilibrium field coils is adjusted in the inner loop of the equilibrium algorithm [using Eq. (1) with $\alpha = 0$] to make the separatrix flux

Fig. 1. HEQ equilibrium with fixed plasma position $(R_0 - a, 0)$, $(R_0 + a, 0)$ and divertor strike-points (R_I, Z_I) , (R_O, Z_O) .

surface pass through $(R_0 + a, 0)$. HYBRD1 is used to determine the remaining free coil currents as roots of the equation

$$\mathbf{F}(\mathbf{I}) = 0, \quad (3)$$

where

$$\mathbf{F} = \begin{bmatrix} (\psi_0 - \psi_x)/\Delta\psi_p \\ (\psi_I - \psi_x)/\Delta\psi_p \\ (a - a_0)/a_0 \\ (\psi_{PF} - \psi_{PF,0})/\Delta\psi_p \end{bmatrix}, \quad \mathbf{I} = \begin{pmatrix} I_1 \\ I_2 \\ I_3 \\ I_4 \end{pmatrix}, \quad (4)$$

a_0 is a given minor radius, ψ_I and ψ_0 are the values of the poloidal flux at the inboard and outboard strike-points, respectively, and $\psi_{PF} = \sum M_{i,p} I_i$ is the PF flux linkage with the plasma. Here $\Delta\psi_p = \psi_x - \psi_0$, where ψ_0 and ψ_x are flux values at the magnetic axis and separatrix, respectively, and $\psi_{PF,0}$ is a prescribed PF flux linkage with the plasma. For given plasma profile functions, HYBRD1 calls the equilibrium subprogram to obtain values of the function $\mathbf{F}$ and solves for the coil current distribution $\mathbf{I}$. Typically, with a good initial estimate of the solution vector, 10 to 20 equilibrium calculations are needed for convergence.

2.3 PROFILE FUNCTIONS

Typical plasma pressure (P) and toroidal magnetic flux ($F = RB_t$) profile functions used in HEQ in the evaluation of the plasma current density distribution,

$$J_t = R \, dP/d\psi + F(dF/d\psi)/(\mu R), \quad (5)$$

are given in terms of their derivatives:

$$dP/dx = P_0[\exp(-Ax) - \exp(-A)]/[\exp(-A) - 1], \quad (6)$$

$$\begin{aligned} dF^2/dx &= 2\mu R_0^2 P_0(1/\beta_J - 1) \\ &\times [\exp(-Bx) - \exp(-B)]/[\exp(-B) - 1], \end{aligned} \quad (7)$$

where $x = (\psi - \psi_0)/(\psi_L - \psi_0)$. The parameter P_0 is scaled in the outer loop of the equilibrium algorithm so that the total plasma current,

$$I_p = \int \int J_t dR dZ , \quad (8)$$

is fixed at an input value. In general, β_J controls the plasma beta and determines whether the plasma is paramagnetic ($\beta_J < 1$) or diamagnetic ($\beta_J > 1$), and A and B control the width of the current profile.

2.4 EXAMPLE: HIGHLY ELONGATED ITER SOLUTION

Using the methods described above, an ITER-size equilibrium solution with elongation (to the X-point) $\kappa_x = 3.0$ is computed by prescribing major radius, minor radius, and X-point coordinates (R_x, Z_x). In this example, five sets of independent PF coils are used (Fig. 2). Currents I_1 – I_4 are determined with HYBRD1 (option 3) to fix the minor radius a , R_x , Z_x , and the PF flux linkage (80 V·s). The outboard equilibrium field coil current I_5 is computed within the equilibrium solution (option 2 with $\alpha = 0$) to fix $R_0 + a$.

3. APPLICATION TO SYSTEMS ANALYSIS OF ELONGATION DEPENDENCE

3.1 SYSTEMS CODE POLOIDAL MAGNETICS MODEL

The ITER systems analysis and optimization code TETRA⁶ requires as input a reference PF system and coil current distribution consistent with an MHD equilibrium. For given plasma elongation and triangularity, the reference equilibrium is characterized by plasma geometry, plasma current, poloidal beta, and plasma internal inductance [i.e., $\bar{\mathbf{P}} = (\bar{R}_0, \bar{a}, \bar{I}_p, \bar{\beta}_p, \bar{\ell}_i)$]. The poloidal magnetics module in TETRA computes the external field $\bar{B}_s(\bar{R})$ of this equilibrium on the plasma midplane $\bar{R}_0 - \bar{a} \leq \bar{R} \leq \bar{R}_0 + \bar{a}$ from the reference coil coordinates $\bar{\mathbf{x}}_j = (\bar{R}_j, \bar{Z}_j)$ and currents $\bar{I}_j$. As TETRA searches for an optimal device/plasma design, this external field is scaled to the updated design point $\mathbf{P} = (R_0, a, I_p, \beta_p, \ell_i)$ by

$$B_s(R) = \{ \bar{R} f(\mathbf{P}) / [R f(\bar{\mathbf{P}})] \} \bar{B}_s(\bar{R}) , \quad (9)$$

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Fig. 2. Highly elongated ($\kappa_x = 3.0$) ITER-size equilibrium.

where

$$f(\mathbf{P}) = I_p [\ln(8R_0/a) + \beta_p + \ell_i/2 - 3/2] \quad (10)$$

and

$$R = (\bar{R} - \bar{R}_0 - \bar{a})(a/\bar{a}) + R_0 + a \quad (11)$$

Also, coils are relocated to the current design point in the TETRA PF module, and coil currents are determined by approximating the scaled external field using a method similar to that of Sect. 2.2.

3.2 REFERENCE EQUILIBRIA FOR ELONGATION SURVEY

To evaluate the impact of plasma elongation on overall ITER cost and performance, we have computed a sequence of TETRA reference equilibria with HEQ, making certain assumptions about PF coil locations to preserve the continuity of results. We have attempted to isolate the dependence of coil currents on the plasma elongation κ_x (measured at the X-point) by using HEQ in a fixed X-point mode to generate a family of equilibria with $R_0 = 4.087$ m, $a = 1.439$ m, $B_t = 4.76$ T, $I_p = -16.191$ MA, $\delta_x = 0.6$, $\beta_p = 0.5$, $\ell_i/2 = 0.3$, flux linkage $\psi_{PF} = 90$ V·s, and increasing values of the X-point elevation Z_x . The PF system (Fig. 3) consists of five pairs of coils, labeled PF₁ through PF₅, where PF₁ is a central solenoid. We have assumed $R_1 = R_2$, $Z_3 = Z_4$, and $R_4 = R_5$ so that coil centers lie on a

Fig. 3. PF coil system used for plasma elongation survey.

rectangular contour surrounding the plasma. This allows us to fix the distances between the plasma and the coils, defining these distances by $D_1 = R_0 - a - R_1$, $D_2 = R_5 - R_0 - a$, and $D_3 = Z_3 - Z_x$. The height of solenoid PF₁ and the elevation of PF₂ are scaled linearly with the X-point. No attempt has been made to optimize coil positions.

The sequence of EEQ reference equilibria is shown in Fig. 4. The dependence of coil currents and coil energy on plasma elongation, subject to the above assumptions, is shown in Table 1. Coil locations used for case A of Table 1 are given in Table 2.

Table 1. Dependence of PF coil currents and stored energy on plasma elongation

	A	B	C	D
Elongation (X-point)	2.02	2.30	2.57	2.72
Elongation (95%)	1.82	2.06	2.34	2.45
Triangularity (95%)	0.41	0.36	0.29	0.24
Coil currents (MA)				
PF ₁	23.36	26.04	28.58	29.70
PF ₂	4.25	0.33	-1.62	-3.55
PF ₃	-11.92	-4.97	-0.48	1.48
PF ₄	15.42	6.40	1.22	-0.36
PF ₅	0.30	3.93	5.65	6.14
PF coil energy (GJ)	10.79	4.66	3.48	3.59

Fig. 4. Reference equilibria used in systems analysis of device dependence on plasma elongation.

Table 2. Coordinates of filaments used to model the PF coils used for case A of Table 1^a

Coil	R (m)	Z (m)
PF ₁	1.330	0.264
		0.791
		1.319
		1.846
		2.374
PF ₂	1.330	4.040
PF ₃	3.008	5.370
PF ₄	8.040	5.370
PF ₅	8.040	1.600

^aCurrents in coils PF₁ through PF₄ are determined by HYBRD1, while the current in PF₅ is computed within the equilibrium loop.

3.3 IMPACT OF ELONGATION ON MINIMUM-COST DESIGNS

Systems analyses based on these reference equilibria were carried out with TETRA, which minimizes an objective function—in this case, device direct cost—subject to physics and engineering constraints on device subsystems. The coil currents of Table 1 were extrapolated (where necessary) to cover the interval $2.0 \leq \kappa_x \leq 3.0$. Studies were performed by using constraints applicable to pulsed-phase ignition operation, with several different energy confinement scalings, and for steady-state current drive phase operation. Figure 5 shows, for example, the sensitivity of cost to elongation when steady-state current drive constraints are applied. For these assumptions, a minimum cost occurs at about $\kappa_x = 2.3$, with a sharp increase at higher elongations. For pulsed-phase ignition operation, the minimum cost occurs between elongations of 2.3 and 2.9, depending on the assumed energy confinement scaling. Further details of this study are contained in ref. 7.

Fig. 5. Minimum-cost device plasma size parameters and cost for steady-state current drive phase operation. (a) Major radius vs elongation. (b) Aspect ratio vs elongation. (c) Direct cost vs elongation.

4. OPTIMIZATION OF COIL CURRENT DISTRIBUTION WITH VEQ

Whenever the number of variable coil currents is greater than the number of constraints, it is possible to minimize some function of the coil currents while still satisfying the given constraint equations. The equilibrium package VEQ has been developed for this application. In VEQ, the free-boundary equilibrium code described above is called from the numerical optimization package VMCON⁸ to find a solution, for example, of minimum PF coil energy $W_{PF} = 1/2 I^t M I$, while maintaining fixed major and minor radii, X-point (or strike-point) coordinates, and PF flux linkage. More generally, VMCON will minimize a nonlinear function of several variables subject to nonlinear equality or inequality constraints. In addition, simple bounds may be imposed on the independent variables.

In practice, we use HEQ to obtain an initial feasible point (i.e., a solution satisfying the constraint equations) for VEQ. As an example, we increase the number of coils in case A of Table 1 by dividing the central solenoid into two sections, PF_{1A} and PF_{1B}, and adding another shaping coil, PF_{3A}, between PF₃ and PF₄ (Table 3). With the solution of Table 1 as an initial estimate, VEQ is used to find a solution of minimum W_{PF} . The resulting VEQ coil current distribution is compared with the HEQ point in Table 4, and this gives a reduction in stored energy from 10.79 to 4.62 GJ.

Note that such an optimal solution may produce currents and fields that exceed physical limits in, for example, the central solenoid. In this case, the option of placing bounds on the solution may be implemented, or additional engineering constraints may be defined. Placing bounds of 18, 12, and 6 MA on the magnitude of currents in coils PF_{1A}, PF_{1B}, and PF₂, respectively, results in the second VEQ solution in Table 4. The two VEQ solutions are shown in Fig. 6. Other possibilities for VEQ optimization include minimizing $\sum R_i |I_i|$ and maximizing the flux linkage subject to bounds on solenoid currents.

Table 3. Coordinates of filaments used to model the modified PF coil system used in optimization of coil distribution^a

Coil	R (m)	Z (m)
PF _{1A}	1.330	6.54 0.791 1.319
PF _{1B}	1.330	1.846 2.374
PF ₂	1.330	4.040
PF ₃	3.008	5.370
PF _{3A}	5.500	5.370
PF ₄	8.040	5.370
PF ₅	8.040	1.600

^aCurrents in coils PF₁ through PF₄ are determined by VMCON, while the current in PF₅ is computed within the equilibrium calculation.

Table 4. Comparison of optimized VEQ coil current distributions with those of the initial estimate produced with HEQ

	HEQ	VEQ	Bounded-current VEQ ^a
Elongation (X-point)	2.02	2.02	2.02
Elongation (95%)	1.82	1.86	1.84
Triangularity (X-point)	0.60	0.60	0.60
Triangularity (95%)	0.41	0.33	0.39
Coil currents (MA)			
PF _{1A}	14.01	10.33	12.94
PF _{1B}	9.34	18.98	12.00
PF ₂	4.25	13.59	1.27
PF ₃	11.92	4.08	12.37
PF _{3A}	0.00	4.69	7.98
PF ₄	15.42	1.82	2.93
PF ₅	0.30	4.52	3.58
PF coil energy (GJ)	10.79	4.62	5.62

^aCurrents in coils PF_{1A}, PF_{1B}, and PF₂ are bounded by $|I_{1A}| < 18$ MA, $|I_{1B}| < 12$ MA, $|I_2| < 6$ MA. Note that I_{1B} reaches its bound value.

Fig. 6. Minimum coil energy solutions (a) without and (b) with bounds on solenoid currents.

5. SUMMARY

Free-boundary equilibrium codes with accurate control of plasma position, PF flux linkage, and X-point or divertor strike-points are needed to adequately survey the design space of next-generation tokamaks and to carry out the detailed design of an ITER PF system. The packages HEQ and VEQ can determine the coil currents of such equilibria and optimize the coil current distribution with respect to a specific objective function. These codes have been applied in a systems analysis of the impact of plasma elongation on ITER cost and performance.

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