# A MAGNETIC SHIELDING SYSTEM FOR THE TOKAMAK EXPERIMENTAL POWER REACTOR*

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## Summary

A poloidal field system is proposed for superconducting tokamaks which magnetically shields the toroidal field coil windings from pulsed fields and reduces the plasma inductive requirements. The principal elements of the system are a primary air core winding and a shield-vertical-field winding. The properties of the system are investigated using computer codes for MHD equilibrium, inductances, and fields.

## Introduction

In the design of a poloidal magnetic system for a tokamak with superconducting toroidal field (TF) coils, desirable design features include: shielding the TF coils from pulsed fields to prevent quenching; minimizing plasma-stored magnetic energy and flux requirements; minimizing stored vertical field (VF) energy; and driving the vertical field by induction from the plasma. The system proposed here is shown in Figure 1 and can fulfill these requirements. It consists of: an air core primary (A) for driving the plasma current which creates a flux path external to the TF coils; a combined shield and vertical field coil (S) which cancels the plasma flux outside it and produces an equilibrium vertical field; decoupling windings (D) which create nearly the same flux pattern as (A) and have the same number of turns as (S), so that when connected in series with (S), there is no net induced voltage from (A); a trimming vertical field winding (T) placed outside the TF coils, which creates an auxillary vertical field to supplement (S); and the plasma (P). The concept of shielding and the techniques of the optimization of the coil locations of these windings is given in reference 1. The design presented is specific to an EPR-sized device.[2]

The central feature of this design is the S winding. This winding is closely coupled to the plasma and has a distribution of series connected windings that are driven to carry a current in amp-turns equal and opposite to that of the plasma and to produce the vertical field necessary for plasma equilibrium. The current distribution is similar to that of a closed conducting shell, which both contains all flux inside it and provides an equilibrium field. The wires are wound in series to ease construction and control requirements and generate one value of an optimally chosen vertical field at each current level. The choice of vertical field decay index depends on requirements of shielding, access, and plasma control.

## Description of Operation

The A winding is charged to full reverse current, and the current in the other windings is zero. This creates a maximum field in the central solenoid, and its flux is returned outside the TF coils. The current in A is rapidly changed from full reverse to some fraction of full forward current. This changing flux induces a voltage in the S, D, and P windings. The T windings are decoupled from this voltage. Since the S and D windings have the same mutual inductance with the A windings, this voltage does not induce current in them; and the electric field penetrates to the plasma, thereby inducing current. The plasma has a much higher mutual inductance to the S winding than to the D winding, so the plasma induces nearly an equal and opposite total current in the S winding and the D winding. A circuit analysis shows that if the S self-inductance equals the S-P mutual inductance, their currents are equal in the absence of resistive losses. If not, a feedback controlled power supply in series with S and D drives their current to be equal and opposite to the plasma current and supplies the resistive losses as well. The S current is also providing the vertical field and effectively canceling external flux. In case of a rapid change in plasma current due to instabilities, the inductively driven S coil will shield the TF coils from the rapid field change. The T coils may be used to adjust the vertical field to the required value. In quasi-steady state, the A winding current is slowly changed to provide the plasma resistive losses.

Instead of the use of the D coil, a tap could be made from the S winding into the A winding, as is done in the Doublet designs.[3]

## Detailed Implementation

The plasma major and minor radius in the Oak Ridge Experimental Power Reactor are R = 6.75 meters, a = 2.25 meters. The S coils are imbedded in the neutron shield with a minor radius of 3.15 m, and the nearest TF winding is at a $r_c$ of 3.80 m. The plasma current maximum is $I_p = 7.2 \times 10^6$ Amp at q = 2.5 in a toroidal field $B_T = 4.8$ Tesla.

An analytic expression based upon an aspect ratio expansion for the required equilibrium vertical field is[4] given in tesla as

$$B_v = \frac{I_p}{10^7 R} \left[ \ln \frac{8R}{a} - \frac{3}{2} + \beta_p + \frac{\ell_i}{2} \right] \qquad (1)$$

where $\beta_p = \Sigma\, nkT/(B_p^2/8\pi)$ is the ratio of plasma pressure to poloidal field pressure, and $\ell_i$ is the internal inductance per unit length in EMU. The internal inductance is calculated by integrating flux linkages using the total flux due to the plasma and to the external coils. For circuit analysis purposes, the self inductance is based only upon the plasma flux, leading to high values for peaked current profiles.

The $\ell_i$ and $\beta_p$ will depend upon the plasma parameters and equilibrium. In order to obtain near ignition conditions, a plasma $\beta_p$ of order 1 to 1.7 is required for the total ion, electron, and alpha pressure.

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It is possible to exceed $\beta_p = A$ and maintain equilibrium if reversed currents can be tolerated in the plasma.[5] The ORNL MHD 2-D Free Boundary Equilibrium Code was used to study the effect of $\beta_p$ and to calculate $\ell_i$. This code solves for the poloidal flux function $\psi$, in R, Z cylindrical co-ordinates, with a source function $j_T$, the toroidal current density.

Inputs include $\beta_J$, the ratio of plasma pressure to poloidal field pressure and details of the pressure profiles. Values of $\ell_i \approx 0.8$ to 1.0 were found for several values of $\beta_p$. In calculations based upon the plasma flux function alone, internal inductance values of approximately 2 were obtained. As discussed in reference 1, typical coil placement locations were obtained for $\ell_i = 1.0$, $\beta_p = 1.0$. For a decay index $N = 0.5$, the coil azimuthal locations are $\pm 8.2^o$, $24.5^o$, $41.1^o$, $58.0^o$, $75.4^o$, $93.8^o$, $114.5^o$, and $142.8^o$, for 16 coils. In a similar calculation for $N = 0.1$, the angles are $\pm 7.1^o$, $21.4^o$, $37.5^o$, $54.9^o$, $76.1^o$, $101.2^o$, $133.3^o$, $166.8^o$. These coils were used in the MHD code to generate the external field, and their total current was set equal to the plasma current times a scale factor VF, which is equal to the ratio of shield current to plasma current. Figure 2 shows an equilibrium in the R, Z plane for $\beta_J = 1.3$, $N = 0.5$. The $N = 0.5$ equilibrium is slightly flattened and has good stability against vertical displacements. The curve distortion in the upper corners is due to the presence of coils. For a similar $N = 0.1$ case, the plasma is nearly circular; and the external flux surfaces are more concentric.

The details of a $N = 0.5$ equilibrium are shown in Figure 3. The pressure profile and flux profiles are centered, while the current is shifted outwards in major radius and is peaked. The internal inductance has the moderate value of 0.9.

The sensitivity of the $N = 0.5$ equilibrium to $\beta_J$ is shown in Figure 4. The plasma envelope is shown for different values of $\beta_J$. In each case, the current in the shielding wires is equal to the plasma current. It is obvious that if the full plasma current is desired, only the design value of $\beta_J$ is acceptable, since the other discharges are too constricted and have significant stray fields at the inner part of the TF coil. Other $\beta_J$ plasma can occupy the full volume if the S current is varied to give the optimum vertical field, or if the T coils give a trimming field.

On the other hand, these equilibria provide the basis for a controlled plasma startup. If external heating is used to increase the total $\beta_J$ while the current is increasing, it should be possible to move steadily from one equilibrium to the next, adding current only to the outside of the plasma as it grows in size. Plasmas with $\beta_J$ greater than 1.3 do not have an equilibrium unless the vertical field is increased still further. The value of $\beta_J = 1.3$ is consistent with $\beta_p = 1$ in the analytic equilibrium formula. The other lower values of $\beta_J$ with their corresponding values of $\ell_i$, R, R/a, also satisfy the formula.

Given the uncertainties in plasma properties, a compromise choice for S coil optimization is $\beta_p = 1$, $\ell_i = 1$, which results in vertical field of 0.34 Tesla at $I_p = 7.2 \times 10^6$ Amps. At a decay index of $N = 0.5$, this is also a maximum for the successful superposition of the currents to form the S coil distribution. Values of $\beta_p = 1.7$ and $\ell_i = 1$ require a field of 0.41 T, which could be produced either from the S or the T windings. These T windings, for the circular case and $N = 0.5$, are shown at the same radius as the A coils in Figure 1.

## Magnetic Shielding of TF Coils

The magnetic shielding can be discussed both in absolute and relative terms. For a high $\beta_p$ plasma in equilibrium, the field due to the plasma alone is 0.73 T at the maximum field point of the TF coil, which occurs at the inner major radius portion of the coil. The shielding depends upon the plasma position, and the finite number of shielding coils, which generate fringe fields. Typically, the $N = 0.1$ shield reduces the field to 0.15 T, and the $N = 0.5$ shield reduces the field to .21 T. Fields elsewhere at the coils are lower still. A contour plot of field strength is shown in Figure 5 for $N = 0.5$ S coils. If the equilibrium field had instead been provided by vertical field coils at the S coil radius, the corresponding peak fields for $N = 0.1$ and 0.5 would have been 1.18 T and 1.32 T. If a comparison is to be made to external VF coils, the results depend on the local fringing fields. For the coils in reference 2, the maximum field is 0.60 T; and the field is 0.4 T over much of the coil. Thus a substantial field reduction can be achieved with shielding. Much improved shielding could be obtained if the ratio of shield coil radius to TF coil radius could be reduced. These results are for an outwardly shifted plasma.

## Inductance Calculations

The single turn self inductance L and mutual inductances are calculated for coils A, D, S, P, and T and shown in Table I. The inductances for the D coil are the same as for the A coil. A circuit analysis is performed for the A, D, S, and P coils and simplified by setting approximately equal inductances to the value $L_{AA}$. The S and P coils have resistance $R_s$ and $R_p$, and the A and S-D coils are driven by voltages $V_A$ and $V_s$. If currents are constrained so that $I_s = I_D = -I_p$,

$$\text{then} \qquad (V_s - I_s R_s) = (M_{SP} - L_{SS})\,\dot{I}_p$$
$$(V_A + I_p R_p) = -(L_{PP} - M_{SP})\,\dot{I}_p \qquad (2)$$

In the absence of resistive losses, the required voltage to drive the S windings approaches zero as $M_{SP}$ approaches $L_{SS}$, which would occur for perfect shielding. Similarly, the plasma inductance $L_{PP}$ is effectively reduced by the mutual inductance $M_{SP}$, which considerably lowers volt second and energy requirements. The detailed inductance calculations were obtained from computer program which integrates over the computer cross sections, of the coils as shown in Figure 1, with $N = 0.5$. Three cases are presented. In case I, the plasma has uniform current density corresponding to $\ell_i = 0.5$ and is centered at R = 6.75 m with radius a = 2.25 m. In case II, the plasma has R = 7.25 m,

a = 1.125 m, to simulate the peaking obtained at high plasma $\beta_p$. Case III gives the inductances for the non-circular coil configuration used in reference 2.

**Table I**

**Mutual and Self Inductances (μH)**

| Case | I | II | III |
|---|---|---|---|
| $L_{AA}$ | 1.67 | 1.67 | 1.05 |
| $M_{AS}$ | 1.60 | 1.60 | 1.08 |
| $M_{AP}$ | 1.59 | 1.59 | 1.04 |
| $L_{SS}$ | 12.25 | 12.25 | 12.25 |
| $M_{SP}$ | 9.10 | 10.01 | 9.10 |
| $L_{PP}$ | 12.04 | 19.55 | 12.04 |
| $L_{TT}$ | 16.45 | 16.45 | 16.0 |
| $M_{AT}$ | 0.02 | 0.02 | 0.18 |
| $M_{ST}$ | 7.63 | 7.63 | 6.5 |
| $M_{PT}$ | 5.58 | 6.38 | 4.6 |

For comparison, an analytic approximation of the plasma inductance is given by

$$L_{pp} = \mu_o R(\ln(8R/b) - 2 + \ln(b/a) + \ell_i/2) \qquad (3)$$

where the contributions inside and outside the S winding at radius b are explicitly written. As was discussed earlier, since $L_{pp}$ applies only to the flux generated by the plasma, and inductance $\ell_i$ of order 2 is applicable corresponding nearly to Case II. In that case the plasma stored magnetic flux is $(L_{pp} - M_{SP})\, I_p$, equal to 69 volt seconds. The $M_{SP}$ term more than halves flux and energy requirements. The ORNL-EPR design allows approximately twice this value in the A coil design in order to allow for startup and plasma driving losses.

## Conclusion

The poloidal magnetic system proposed here is capable of shielding the TF coils and of reducing the plasma volt-second and stored energy requirements. The effectiveness of the shielding depends upon obtaining a plasma with the optimum $\beta_p$.

## Acknowledgements

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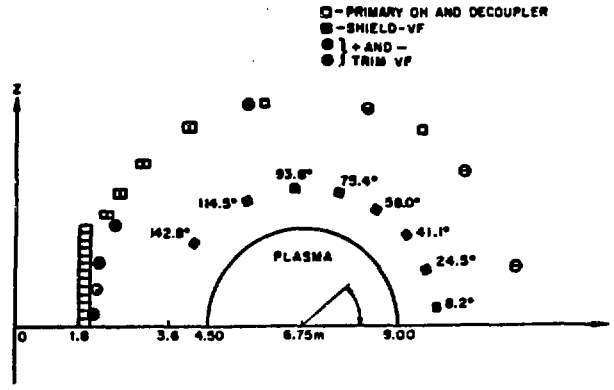


Figure 1. Magnetic Shielding System Location of Poloidal Field Coils

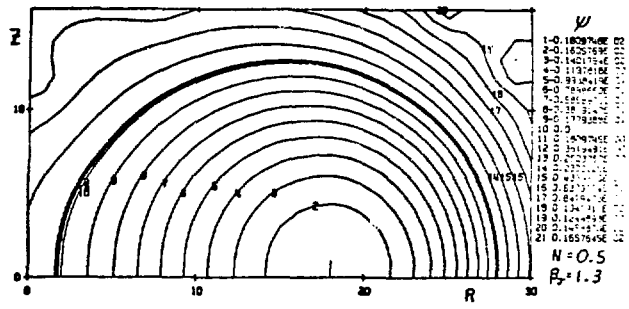


Figure 2. Flux Surfaces — MHD Equilibrium

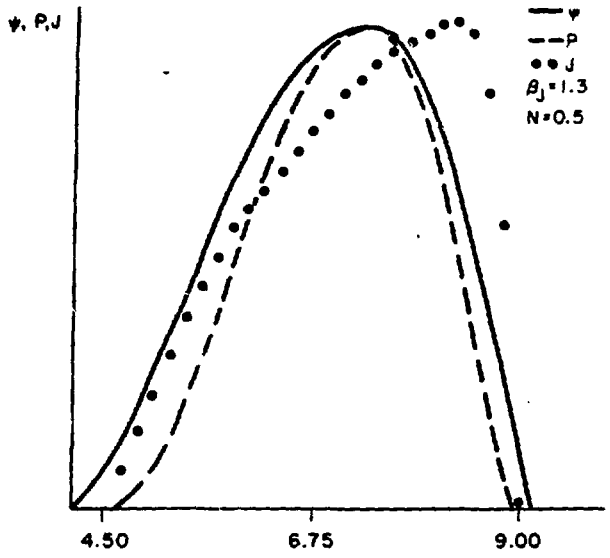


Figure 3. MHD Equilibrium
Values of Flux ψ, Pressure P,
Current Density J at Z = 0 Plane

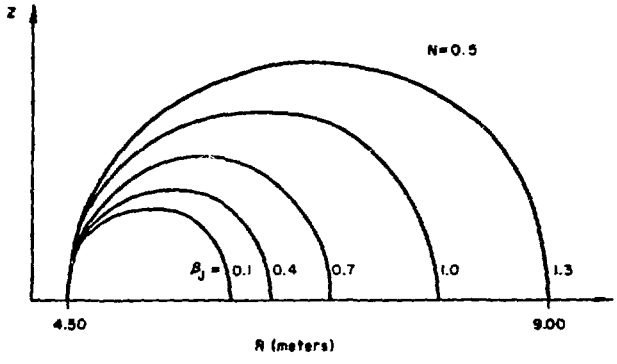


Figure 4. Outer Flux Surfaces For Varying
Poloidal Beta $[I_s = -I_p]$

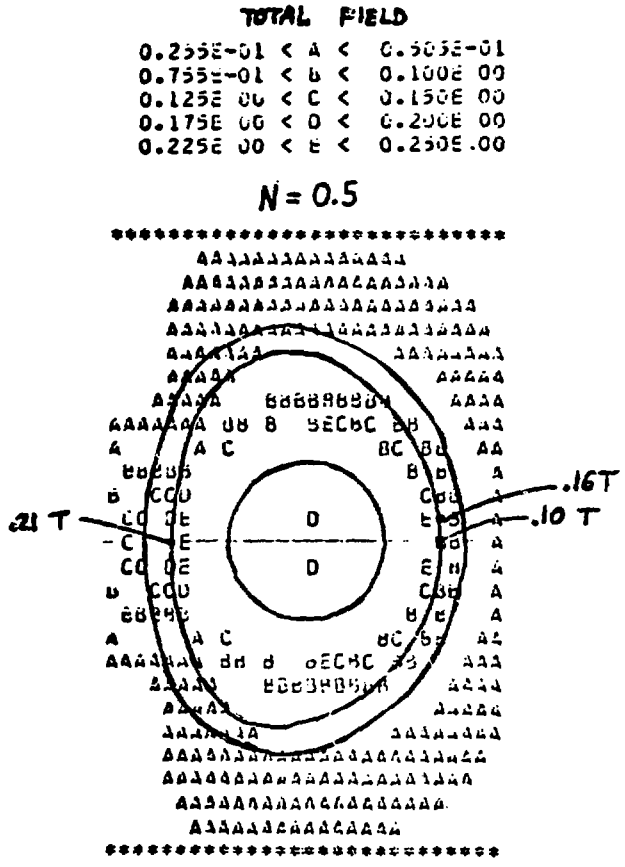


Figure 5. Field Strength At Toroidal Field Coils
Due to Plasma and Shield Coils