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ELECTROMAGNETIC COMPUTATIONS FOR FUSION DEVICES*

Larry R. Turner
Fusion Power Program
Argonne National Laboratory
Argonne, IL 60439 USA

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Larry R. Turner
Fusion Power Program
Argonne National Laboratory
9700 S. Cass Ave.
Argonne, IL 60439 USA

Abstract

Among the difficulties in making nuclear fusion a useful energy source, two important ones are producing the magnetic fields needed to drive and confine the plasma, and controlling the eddy currents induced in electrically conducting components by changing fields. All over the world, researchers are developing electromagnetic codes and employing them to compute electromagnetic effects. Ferromagnetic components of a fusion reactor introduce field distortions. Eddy currents are induced in the vacuum vessel, blanket, and other torus components of a tokamak when the plasma current disrupts. These eddy currents lead to large forces, and 3-D codes are being developed to study the currents and forces.

Introduction

Considered as an energy source, nuclear fusion has tremendous promise: virtually unlimited fuel supply; availability day and night, summer and winter; freedom from atmospheric polution; and, compared to nuclear fission, no danger of meltdown or fuel diversion and greatly reduced radioactive waste. Achieving fusion power, however, requires overcoming tremendous technological problems.

The most widely studied method of employing fusion energy is the tokamak. The International Thermonuclear Experimental Reactor (ITER)[1] is shown in Fig. 1 as an example of a tokamak. In a tokamak, magnetic fields heat a plasma of heavy isotopes of hydrogen to a high temperature so that fusion can occur. Other magnetic fields confine the plasma and keep it in an equilibrium shape and location. These magnetic fields and those from the plasma current interact with other electrically conducting components of the tokamak, such as the vacuum vessel in today's tokamaks, and the first wall, blanket, and shield in future tokamak reactors.

The design of a fusion device or reactor requires the computation of electromagnetic effects between the magnetic fields and conducting components. This paper discusses some of those computations.

The magnetohydrodynamic (MHD) behavior of the plasma is basically electromagnetic, but such computations are a separate, specialized activity and will not be discussed here. Instead I will first discuss magnetostatic effects, particularly those involving magnetic materials or temperature dependent electrical conductivity. Then I will discuss eddy current effects resulting from a plasma disruption, and finally eddy current computations for the vacuum vessel of a reversed-field pinch device, an alternative concept to the tokamak.

Magnetostatics

In general tokamaks and other fusion devices do not incorporate steel or other magnetic materials, and so the magnetostatic computations are not difficult. The codes EFFI[2] and GFUN[3] have often been used for such computations. The D-shaped toroidal-field coils are often modeled by some combination of circular arcs.

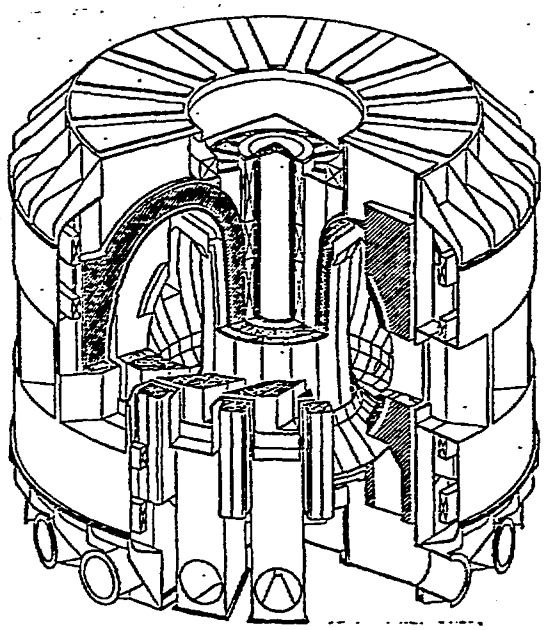

Figure 1. A design for the ITER fusion reactor, as proposed by the U.S. ITER Team. Courtesy of Royce Sayer, Fusion Engineering Design Center, Oak Ridge National Laboratory.

Three areas where ferromagnetic alloys with field dependent permeabilities may be of use have been studied; analysis of these three requires a magnetostatics code with non-linear capability. First, some tokamaks, the Joint European Tokamak (JET) for instance, use an iron core for the coils which drive the plasma current.

Second, because of their enhanced resistance to radiation damage, ferritic stainless steels such as HT-9 have been suggested for the first wall tokamak reactors. H. Attaya[4,5] used the code GFUN to study ferritic steel blankets for fusion reactors. The forces on some components, such as protruding pipes, could be very large.[4] Also he found a troubling assymmetry in the induced field due to a temperature gradient around the blanket (inlet temperature 330°C, outlet temperature 500°C), and the temperature dependence of the saturation magnetization. In the case he examined[5], field perturbations with or without the temperature variation were less than 0.4% and were judged to be insignificant. Ohira et al.[6] developed a thin-shell variant of GFUN and applied it to a ferromagnetic blanket. They found the toroidal field to increase the force on the first wall by a factor of 2.5, but the stresses were still acceptable.

Third, because the toroidal field coils are localized and do not form a closed shell, the field they produce varies toroidally, being stronger inboard from the coils and weaker inboard from the gaps between the coils. This toroidal field ripple depends on the number of TF coils and on how far the outer legs of the coils are from the plasma. Reducing the ripple to an acceptable limit can require more coils (resulting in access problems) or larger coils (resulting in added costs). But computations with GFUN showed[7,8] that ferritic steel located inboard of the TF coils would reduce the field there and increase it between coils, thus reducing the ripple by a factor of four or more.

Molfino et al.[9] studied the coupled magnetic and thermal problem of the redistribution of current in copper coils (in this case, those of the IGNITOR tokamak) using the coupled thermo-electromagnetic code COMPELL. The Joule heating of the copper conductor raises its resistance in a transient way.

Eddy Current Analysis of a Plasma Disruption

Disruptions of the plasma in a tokamak fusion device or reactor can induce large currents in the electrically conducting components of the tokamak, resulting in large electromagnetic forces. These forces have been observed to cause severe damage in tokamaks. In tokamak reactors, the plasma current is expected to decay at rates approaching 1 MA/ms.

Because of differences between the structure of existing tokamaks and the structure of tokamak reactors now being considered, methods and codes previously used to study the eddy currents and forces resulting from a disruption are no longer adequate.

Present Generation of Tokamaks

The vacuum vessels of most existing fusion experiments, TFTR, JT-60, and JET in particular, have bellows or other segments to provide continuous but high resistance current paths. The high resistance bellows sections decrease the L/R time constant of the first wall, facilitate magnetic flux penetration, and tend to limit the induced current; but to first order they do not change or constrict the current path. Induced currents flow toroidally and do not interact with the toroidal field. Other tokamaks, such as PLT, have had a dielectric break in the vacuum vessel to prevent circulating currents. Opposite currents flow on the inside and outside surfaces of the vessel, and the currents decay even more quickly, but their analysis is still fairly straightforward.

I would date the modeling of a realistic geometry from Kameari and Suzuki's 1977 eddy current analysis[10] of the vacuum vessel and support plates of JT-60. The analysis treated the vacuum vessel as a thin shell with the two components of current density expressed as the curl of a current potential V. The equations were formulated from the energy and solved by an eigenvalue approach. The code EDDYTORUS treated a segment of the odd-shaped vacuum vessel including holes and resistive regions representing the bellows, as shown in Fig. 2. Others[11,12] also treated eddy current effects at about that time.

Weissenburger[13] applied the SPARK code[14,15] to the eddy current analysis of the TFTR vacuum vessel. SPARK is a network code with loop currents as variables and is now widely used for fusion applications. The TFTR vacuum vessel was modelled as a shell with high-resistance bellows sections and with the many ports and lumpy port structures shown in Fig. 3. Results were presented as a movie.

Future Tokamaks

By contrast, future fusion reactors such as the Next European Torus (NET), the Japanese Fusion Experimental Reactor (FER), and the International Thermonuclear Experimental Reactor (ITER) will have segmented blankets and other components to permit remote maintenance and convenient replacement. Eddy current paths will not be strictly toroidal; there will be poloidal and radial current components as well. Whereas toroidal currents interact only with the poloidal field, the radial and poloidal currents will interact with the toroidal field, typically ten times stronger, and produce a severe and complex distribution of forces and torques.

Kameari[16] applied the code EDDYCUFF to the complex blanket of the Fusion Experimental Reactor (FER). The mesh shown in Fig. 4 consists of multiple shells and plates in several orientations.

More recently Charissecourte et al.[17] have modelled the segmented ITER blanket with the 3-D eddy current code TRIFOU.[18] The plasma was taken to have a specified homogeneous current and a specified decay profile in time. The 3-D code CARIDDI,[19] like TRIFOU, uses edge elements. Its use by the NET team is described below.

Crutzen et al.[20] at the Joint Research Center, Ispra, have modeled the effects of plasma disruptions in NET and ITER using the 2-D codes UNISH, SCILLA, and CORFOU and the 3-D codes CARIDDI and TRIFOU. Figure 5 shows the first wall as modelled with the different codes. We may all benefit from this opportunity to model the same situations with the different codes, and compare their strengths and their limits of applicability.

Coupled Analysis of the Torus-Plasma Interaction

To analyze the interaction between the plasma and the conducting torus (vacuum vessel, blanket, etc.), both the torus and the plasma must be modeled. In the past, either of two approaches was taken. In the first, the torus was modeled by a state-of-the-art eddy current code, and the plasma by a circular loop with fixed position and specified time behavior of the plasma current (usually a linear decay). This approach, with some variations, was used especially to study the effects of disruptions. In the second, the plasma was described by a detailed MHD calculation, and the surroundings treated as a perfectly conducting axisymmetric shell. This approach was used to study the stability of the plasma.

Recently, there have been attempts to treat the plasma and torus in comparable detail. Three approaches are described here.

DSTAR

The DSTAR code[21] treats the plasma with the Tokamak Simulation Code (TSC)[22], and models the vacuum vessel and other torus components with many resistive loops inductively coupled to each other and to the plasma. TSC solves the coupled MHD and Maxwell's equations for a resistive plasma.

Segmentation of the vacuum vessel and blanket is modelled in DSTAR by requiring a zero net circulating current in the loops representing the torus. That is a reasonable first approximation for segmentation effects, but cannot predict the poloidal and radial components of current that are responsible for the largest electromagnetic forces on the torus.

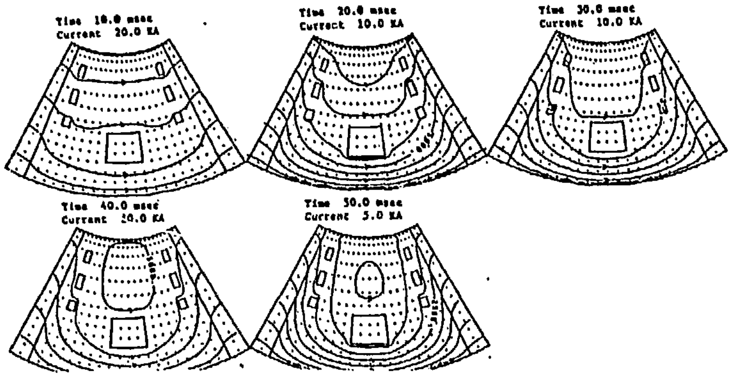

Figure 2. Eddy current distribution in the JT-60 vacuum vessel with holes. Courtesy of Akihisa Kameari, Controlled Thermonuclear Fusion Team, Mitsubishi Atomic Power Industries, Inc.

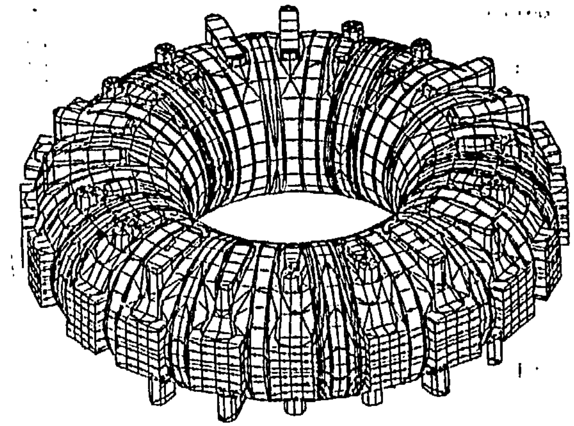

Figure 3. Eddy current paths in the TFTR vacuum vessel with structure. Courtesy of Don Weissenburger, Princeton Plasma Physics Laboratory.

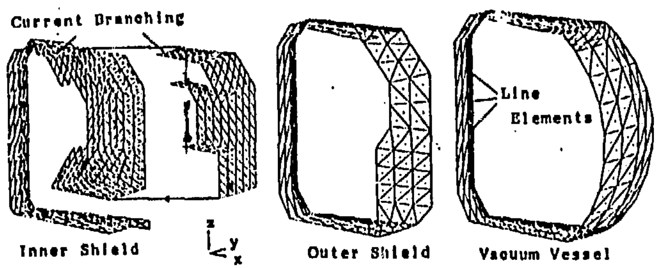

Figure 4. Eddy currents in the vacuum vessel and shield of FER. Courtesy of Akihisa Kameari, Controlled Thermonuclear Fusion Team, Mitsubishi Atomic Power Industries, Inc.

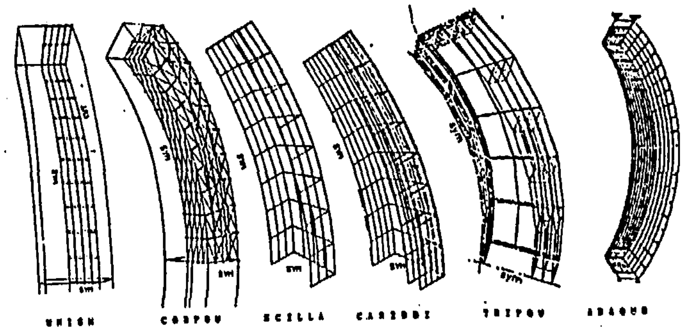

Figure 5. The first wall of NET as modelled with six eddy-current codes. Courtesy of Yves Crutzen, Joint Research Centre, Institute for Systems Engineering, Ispra.

PROTEUS/CARIDDI

The NET Team have undertaken a fuller treatment of electromagnetic effects on the vacuum vessel and blanket from plasma descriptions using the two codes PROTEUS[23] and CARIDDI[19]. In particular they have looked at vertical instabilities in the plasma of ITER[24]; see Fig. 6.

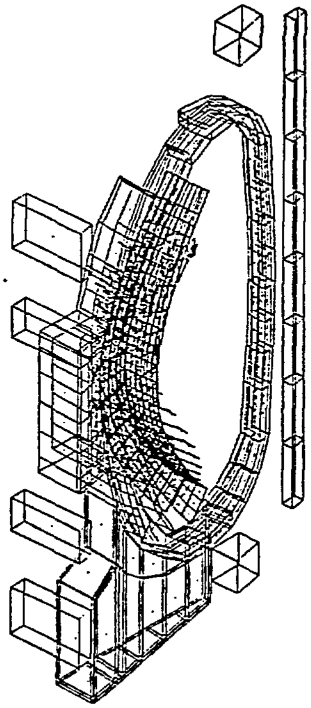

Figure 6. Eddy current distribution in the vacuum vessel and first wall of ITER. Courtesy of Raffaele Albanese, NET Team and University of Salerno.

PROTEUS is an axisymmetric finite element code to study plasma behavior. It is used to model plasma evolution, instabilities, and disruptions. It can include external structure (passive stabilization) and circuits (e.g., active stabilization). The code can treat non-linear iron; hence it can be calibrated to JET results.

CARIDDI is a true 3-D integral code for the current density J (actually for a unique current vector potential T). The plasma current is treated as a single filament.

There is little direct 3-D effect on the plasma from the 48 segment first wall of NET. Poloidal current on opposite sides of each gap are equal and opposite, and so there is little perturbing field. Major effects are axisymmetric, namely:

1. Added resistance due to the poloidal current path,
2. Added inductance due to the field in the gap,
3. No net circulating current in the first wall.

The axisymmetric plasma code PROTEUS permits coupling to external circuits. Shorting elements together poloidally yields no net current, and other circuits can incorporate the resistance and inductances.

Coccorese[25] has described two approaches to combining plasma codes and eddy current codes in analyzing a disruption, a plasma start-up, or a vertical instability. In <u>open-loop integration</u>, the output of one code serves as input to the next. For example, PROTEUS can calculate a plasma equilibrium; NAPS-G, another code of the NET Team, then computes the motion of the plasma for an assumed disruption initiation and outputs a set of fixed filimentary loops with time varying currents modeling the moving plasma. CARIDDI then determines the eddy currents, induced fields, and forces in the vacuum vessel, for example, and sends these on to a stress analysis code.

In <u>closed-loop integration</u>, output from later codes (e.g., CARIDDI) are input into earlier codes (e.g., PROTEUS), and the process iterated until a consistent solution is achieved. However, the number of degrees of freedom of the CARIDDI solution far exceeds the capacity of PROTEUS. The technique called <u>model reduction</u>, commonly used in control theory, has been applied[26] to modeling the NET vacuum vessel. The CARIDDI solution, with 16 degrees of freedom, was transformed to a form ordered in decreasing contribution to ohmic power and magnetic energy. In that case, keeping one, two, or four degrees of freedom reproduced the maximum ohmic power and induced magnetic energy to within 4, 1, or 0.01 percent respectively. Applying the method to solutions with 100 or more degrees of freedom is underway.

TSC/EDDYNET

In America, a somewhat different approach is underway. Royce Sayer at Oak Ridge National Laboratory, Brad Merrill at Idaho National Engineering Laboratory, and the author are studying eddy current forces in the segmented ITER blanket with the plasma analysis code TSC and eddy current code EDDYNET.[27] We have analyzed a fixed-position midplane disruption,[28] and are now looking at a disruption scenario in which the plasma moves vertically to the divertor region in 80 ms before disrupting.

TSC models the axisymmetric poloidal field (PF) coils and vacuum vessel with coaxial rings on a mesh with 150 mm spacing radially and axially. Modeling the blanket the same way would neglect the all-important effects of segmentation.

Instead, an open loop approach is employed, in which the radial and vertical components of field are output from TSC at all times and at the radial and vertical coordinates r, z of the centers of all EDDYNET loops of the blanket. The field from the plasma, vacuum vessel, and PF coils can be examined separately, to find the effect of each. Changes in this applied field drive eddy currents. EDDYNET computes the currents, induced fields, and the forces of interaction between the eddy currents and the toroidal field (TF).

The overall goal of the computations is to define a blanket model, axisymmetric but with no net toroidal current. The resistances of the coaxial loops representing the blanket will be varied to get an eddy current pattern like that of the EDDYNET computation.

Reversed Field Pinch

A reversed field pinch (RFP) device has a thick stabilizing shell of high conductivity. In order that the applied fields can readily penetrate the shell, it normally has poloidal and toroidal gaps. These gaps perturb the eddy current paths and result in a net field that is not axisymmetric. Various approaches have been studied to minimize the asymmetry, e.g., compensated gaps and multiple shells with the gaps not aligned.

Many approaches have been taken to compute the eddy current effects Vogel and Preis[29] used a network code FEDIFF to study butt joints for the device ZT-40 M.

Turner[30] used the code EDDYNET to compare various gaps for ZT-P. The network codes use filament conductors and can give satisfactory results away from the walls of the shell but not near the walls. Fukuda et al.[31] compared the network code FEDIFF and an integral code using the current vector potential V. For the ZT-40M, they found that the two methods gave distinctly different results. The reasons for the differences are not determined. Vogel[32] modified the code FEDIFF to calculate fields from plates rather than filaments. The exaggerated waviness (sometimes an order of magnitude) of the filament effects disappeared. More analysis, as well as benchmarking of the codes, is needed.

Gnesotto et al.[33] studied the effect of gaps and holes in the RFP device RFX using the surface current density as variable. Uesaka et al[34] did a similar study of REPUTE-1 with the mesh code INCANET, as did Sugiura et al.[35].

Conclusions

In my mind, three recent developments are very encouraging for applying electromagnetic computation to fusion reactors:

1. Several groups doing coupled plasma/eddy current computation.

2. The ITER project, which requires European, Japanese, American, and Soviet groups to attack the same eddy-current computation.

3. The in-house comparison of TRIFOU, CARIDDI, and other codes at Ispra.

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