Error Field Impact on Plasma Boundary in ITER Scenarios

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Discrepancies in magnetic field maps produced by confinement coils in thermonuclear fusion reactors may drive plasma to loss of stability, and must be carefully controlled during the whole time evolution of each shot using suitable correction coils. Anyway, even when kept below safety thresholds, error fields may alter the geometry of magnetic flux lines, defining the plasma geometry. The present paper, using high accuracy 3D magnetic field computations for confinement coils, addresses the issue of evaluating the effect of error fields on the plasma boundary shape during the shot, modeled as a sequence of equilibrium configurations. In particular, a procedure able to compute the shape perturbations due to given deformations of the coils has been set up and used to carry out an analysis of relationship between the error field and shape perturbations during the time evolution of ITER programmed scenario.

***Index Terms*—Thermonuclear Fusion, Error Fields, Numerical Methods.**

I. Introduction

In thermonuclear fusion magnetic confinement devices (e.g. Tokamaks), the performance of the machine is quite sensitive to discrepancies between the nominal magnetic field and the actual one. These differences are called "Error Fields" (EF), and result from several causes, mainly mechanical inaccuracies in the manufacturing and assembly of magnets, although also other non axi-symmetric sources (e.g. current leads for superconducting magnets or iron parts) do contribute [1], [2]. The error fields are by their intrinsic 3D nature quite complex to describe and compute; an effective tool for their quantitative analysis is a Fourier expansion in toroidal and poloidal harmonics of the normal magnetic field component on flux surfaces, which should vanish in nominal configuration. This is the standard approach adopted in the existing machines, and also in the design of ITER, the next generation Tokamak, capable of demonstrating feasibility of fusion as an energy source, presently under construction at Cadarache (Fr). EF may cause loss of stability of the plasma column, e.g. stop plasma rotation with a subsequent growth of a so-called locked mode, thus forcing a premature end of the experiment. To avoid such extreme consequences, ITER will be equipped with suitable correction coils, driving back error fields under acceptable threshold.

EF have also an impact on the axisymmetric shape of the plasma column, which fits its designed geometry only under nominal confinement and shaping field. A statistical analysis of the relationship among EF amplitudes and their effect on plasma boundary is presented in [3]. As a matter of fact, the correction coils are fed by currents optimized to reduce EF at the Start of Flat top instant during plasma discharge, and are not varied. Currents in shaping coils, on the other hand, do evolve during discharge, thus EF may not be canceled with the same efficiency in other time instants.

In this work, a new numerical tool aimed at studying the impact of EF on plasma boundary during the full time evolution of plasma discharge, is proposed.

In this short version, due to space shortage, a brief overview of the approach, and some preliminary results, will be presented. In the full paper, a more throughout description of the method, including numerical aspects, will be discussed. A careful analysis of the EF impact on plasma boundary will be presented for ITER class Tokamaks.

II. Overview of the Analysis Method

The procedure used to evaluate the effect of coil deformations on ITER axisymmetric equilibria is the following. A deformed coils configuration is randomly generated, assuming tolerances in the expected ranges. EF corresponding to actual coils are then computed, at each time instant of the plasma discharge, using a high accuracy 3D field computation tool, the MISTIC code [4]. Then, the CarMa0NL code [5], able to solve nonlinear Grad-Shafranov equations in presence of 3D conductors, is used to evaluate the subsequent perturbation of the plasma axisymmetric equilibrium configuration starting from field maps obtained in the first step.

A. MISTIC

The tolerance ranges for ITER magnets are in the order of few parts per thousand, and related field variations are expected to be in the same ranges. Assuming that iron parts do not vary their magnetization state with respect to nominal values, to compute field variations MISTIC adopts a decomposition of coils into a high number of filamentary conductors, possessing closed form expressions for magnetic field. It is possible to discretize conductors up to the level of strands inside superconducting cables, altought in this study one single current line is used for each conductor in the Winding Pack (WP) of each coil. Each deformed conductor is then described by an interpolating curve, typically a spline, defined using a limited number of parameters, such as the coordinates of a few control points (this is the case of toroidal field coils) or the center and semi-axes of a tilted ellipsis (this is the case of poloidal field coils). The interpolating curve is sampled in a number of break points, defining the tips of the segments representing the curve. In order to fulfill the required accuracy in the field computation, the number of sticks is suitably selected according to the local curvature of the interpolating curve and to the distance from field points.

B. CarMa0NL

The CarMa0NL code [5] is able to solve non-linear axisymmetric plasma MHD equilibrium equations, self-consistently coupled with the surrounding 3D conducting structures. These features make the CarMa0NL code particularly suitable to evaluate the effect of EF on equilibrium configurations.

The non-linear Grad-Shafranov equation is solved in the plasma region Ω:

$$\begin{aligned} \mathcal{L}\psi &= J_\varphi(\psi)\ in\ \Omega \\ \psi|_{\partial\Omega} &= \hat{\psi} \end{aligned} \tag{1}$$

where $\mathcal{L}$ is the Grad-Shafranov operator, ψ is the poloidal magnetic flux per radian, J_φ is the toroidal plasma current density. The boundary value $\hat{\psi}$ is the sum of two contributions, one related to the plasma current and the other one to the external sources. In the 3D conducting region surrounding the plasma, the eddy current equations are solved according to the integral formulation presented in [6]:

$$\underline{\underline{L}}\frac{d\underline{I}}{dt} + \underline{\underline{R}}\,\underline{I} + \frac{d\underline{U}}{dt} = \underline{V} \tag{2}$$

where $\underline{\underline{L}}$ and $\underline{\underline{R}}$ are, respectively, the inductance and resistance matrices of elementary 3D current loops related to the active edges of the hexahedral mesh used to discretize the conducting structures. The term $\underline{V}$ describes voltage sources applied to the electrodes and $\underline{I}$ is the vector of the degrees of freedom describing the 3D currents. The model takes into account the effect of the plasma current by means of the term $\underline{U}$.

The two mathematical models are coupled by means of a suitable surface $\partial\Omega$ placed in between the plasma and the 3D conductors, where boundary conditions to (1) are applied. The final model describing the plasma evolution is obtained discretizing (1) on a 2nd order triangular mesh and using the Galerkin method. The resulting nonlinear system of equations is solved by means of a Newton-Raphson algorithm.

III. Example of Application

In order to show the capabilities of the method, the reference 15 MA inductive scenario in ITER programmed discharges has been used [7]. The geometry of nominal plasma boundary at Start of Flat Top is reported in Fig. 1 as an exemplification. In Fig. 1 also the "gaps" are reported, defined as the distance between plasma boundary and vacuum vessel in a number of points. Such gaps are used as input parameters to plasma geometry control process during the experiment, and then represent relevant plasma shape descriptors.

Using a randomly generated set of coils deformations, the effect of EF are illustrated in Fig. 2, where two typical gap variations during the scenario are reported.

IV. Conclusions

In this work, we have quantified the effect of error fields on the equilibrium configurations of ITER during the whole plasma discharge. In the full paper a more complete description of the method will be presented, and a broader set of simulations presented.

Acknowledgments

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V. References

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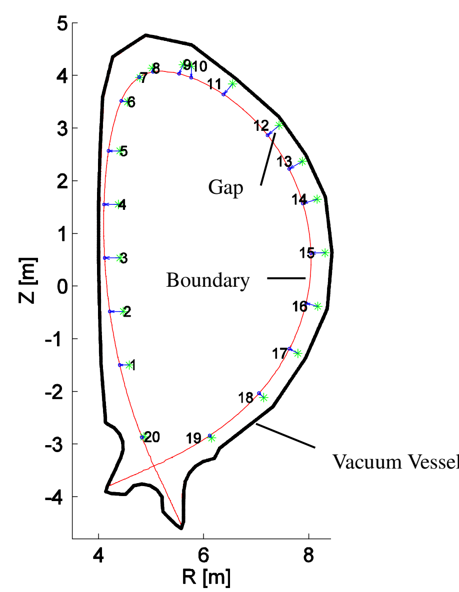

Fig. 1. Plasma boundary and gaps at Start of Flat Top in the considered scenario.

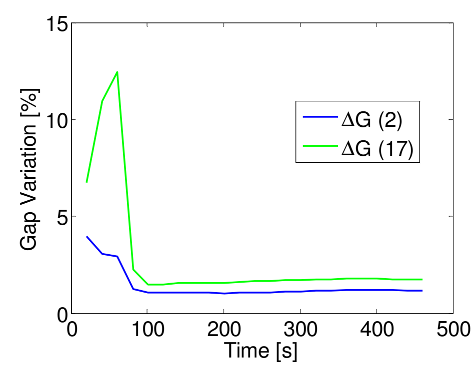

Fig. 2. Two typical gap variations during the considered scenario.