FLIPEC: Free fLow Plasma equilibrium Code.

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Introduction.

Toroidal and poloidal flows are relatively common in some type of tokamaks and its consequences might have influence in several characteristics of the plasma. For instance, the shape of the plasma can be altered, including X-point location or the plasma boundary itself. In addition, some quantities and profiles may be modified, like density, pressure or even the toroidal field might be influenced by strong flows. All of these changes would help the interpretation of the diagnostic systems during the operation of a tokamak. Likewise, if the coil configuration is modified, these codes would be able to assess the variation in the new generated equilibrium.

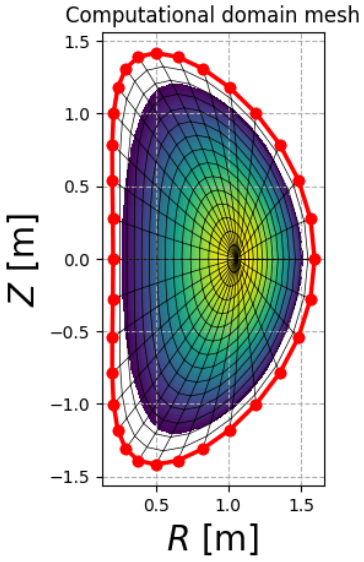

Figure 1: Illustrative example of a geometry adapted for the spherical tokamak NSTX. The red thick line stands for the boundary of the computational domain.

Therefore, the aim of FLIPEC is the calculation of *Ideal* MHD equilibrium with macroscopic plasma flows in the context of toroidal axisymmetric geometries. From the beggining, FLIPEC was designed to support free boundary calculations [1] by using an optimal method [2], that was already compatible with the 3D static equilibrium code SIESTA [3]. There are several codes, [4] [5], written to describe the equilibrium with flows, nevertheless , the free-boundary feature in conjunction with the capability of adapting the geometry is a characteristic that is usually not satisfied within the same code in the computation of MHD equilibrium. Thus, FLIPEC may open up many applications for this area.

In this proceedings, the reader will find a brief description of the code and its workout, followed by the display of the new implemented upgrades.

The free boundary code.

The early code manages a right handed Eulerian frame in toroidal coordinates, (r, θ, ξ), letting the plasma boundary to move freely within the computational domain. This selection let the code to exploit spectral methods in the poloidal angle for two reasons: This was a requirement for the free boundary scheme, but also this characteristic leads to an exponential decrease in the residual force in the simulation. On the other hand, radial variable is treated with second

order finite differences so that the convergence is limited to quadratic decrease. Although there is no need of locating the geometric center right in the magnetic axis, a proximity between them would benefit the convergence and it makes the code more compatible with any equilibrium calculated with VMEC. The nature of these kind of codes makes this last characteristic to become relevant, since having a closer initial solution makes the convergence much faster. Additionally, five free pseudo-quantities must be given to complete the source of the differential equation: pressure $P(\psi)$ and density $D(\psi)$; the description of the flows by their pseudo mach numbers, $M^{C_s}_{tor}(\psi)$ and $M^{C_s}_{pol}(\psi)$, and the pseudo-toroidal field, $B_0(\psi)$, that can be inferred from the poloidal current profile.

A Picard iteration between Grad-Shafranov and Bernoulli equations (1) (2), searches for the solution, which is described by the poloidal magnetic surfaces. The shape of these surfaces is characterized by the *stream function*, ψ. In addition to the main iteration, an update of the computational domain boundary, $\psi(r_b,\theta) = \psi_b(\theta)$ (red thick line in the fig. 1), is made by solving for the vector potential the equation $\nabla \times (\nabla \times A) = \mu_0 \mathbf{J}_{\mathscr{P}}$ and adding the contribution of the magnetic field created by any arbitrarily complex set of external coils.

Figure 2: Test of the elliptic general coordinates by calculating the same ITER equilibrium with a flat profile with $M_{\text{tor}} = 0.4$ as a scale factor for the maximum value, but setting two different ellipticities (left $\alpha = 0.8$, right $\alpha = 0.65$). The plots show the color map of the plasma density.

General coordinates.

Working with standard toroidal coordinates has a considerable drawback: the circular shape of the computational domain. Among others, the main reason is the singularity in the vacuum magnetic field, which is caused at the locations where the field is generated: the solenoid and/or the coils. Indeed, these structures, where the current is driven, cannot be included within the external boundary, what situated many tokamaks out of the scope of the early code. Therefore, implementing general coordinates in FLIPEC were a priority and now, the code is able to adapt the computational domain to any tokamak, including many spherical ones, where the solenoid is just a few centimeters from the plasma. Accordingly, NSTX geometry was selected to test the code. Besides, since an ITER baseline configuration was taken as main showcase of the early code test, it was also an adequate example for testing elliptic geometries with different ellipticities. Note that the final result must not depend on the geometry and, particularly, not on the computational boundary shape (see figure [2]). Comparing both equilibrium in different meshes is not an easy task due to the different nodes location and due to the increased number of nodes within the plasma in the cases with more tight geometries, however, some global quan-