Simulation of Disruptions in EAST Tokamak

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1. Introduction

Fusion devices and tokamaks in particular are subject to the so-called disruptions, i.e. rapid loss of plasma (magnetic and thermal) energy content [1]. These events are a concern especially for future devices, like ITER and DEMO, because they can cause significant heat and electromagnetic loads on structures surrounding the plasma. The main phases of a disruption are the Thermal Quench (TQ), in which the thermal energy content of the plasma is lost on a very short time scale (milliseconds or fractions in present-day devices), and the Current Quench (CQ), during which the plasma current and magnetic energy decay on a longer time scale (tens of milliseconds in present-day devices). During a disruption, a Vertical Displacement Event (VDE) may take place, causing the plasma to hit the wall, with subsequent direct injection of halo currents from the plasma to the structures.

Very detailed codes such as JOREK [2] are used to provide a comprehensive understanding of plasma modelling during a disruption. Other models focus instead on the description of the electromagnetic interaction between the plasma and the structures, with the aim of predicting the time behavior of the main plasma parameters. This is the approach used for instance by DINA [3], which has been extensively applied to build up a database for ITER, and is also used in this paper with the CarMa0NL code [4], to simulate the Experimental Advanced Superconducting Tokamak (EAST) [5].

2. The EAST tokamak and the CarMa0NL code

The EAST tokamak [5] is equipped with fully superconducting poloidal field (PF) and toroidal field (TF) coils and is designed and built to explore the physical and engineering issues under steady state and long pulse operation for the support of future fusion reactors. Since the toroidally continuous conducting structures in EAST are relatively far from the plasma [6], the 3D features of the plasma facing components (PFCs) must be necessarily taken into account in disruption simulations,. The CarMa0NL code [4] has the unique feature of taking into account

3D geometry of conductors, and hence is particularly well suited to simulate disruptions in EAST. It assumes that the plasma mass can be neglected, so that the plasma evolves through equilibrium states; it is based on the coupling surface approach, so that different mathematical models and numerical formulations are used in the plasma region and in the surrounding conductors. In the plasma domain, thanks to the equilibrium constraint, before halo current arise, the toroidal current density is a function of the normalized poloidal flux $\bar{\psi} = \frac{\psi - \psi_A}{\psi_L - \psi_A}$, where ψ_A is the value of ψ at the plasma magnetic axis and ψ_L is the value of ψ at the plasma boundary, defined as the last closed magnetic surface. When a halo current flows in the plasma after plasma-wall contact, the current density is extended outside the plasma limiter point ψ_L, up to a value ψ_B. The halo region width is imposed as a function of time.

3. Results

We consider a disruptive shot (#43887) that has been already analyzed in [6]. Starting from a lower single null configuration, the plasma is intentionally left free to evolve by switching off the vertical position feedback controller at t=5 s. Consequently, a VDE takes place, with the plasma centroid vertical position displacement evolving downwards exponentially in time with a growth rate γ, experimentally estimated as 200 s^{-1}. The initial random excitation of the unstable mode has been reproduced in the simulations as a fictitious glitch in the current of the in-vessel coils (IC) dedicated to vertical position control. This leads in the evolution of the plasma configuration described in Fig. 1, where also the identification coming out of EFIT code is reported until available.

After the plasma-wall contact, the plasma experiences the Thermal Quench, simulated as a poloidal beta drop occurring linearly in 4 ms. Immediately after, the Current Quench occurs and halo currents develop. In the simulations, the total (halo + core) toroidal plasma current is imposed coherently with the experimental measure of Rogowski coils (Fig. 2), which is simulated using 201 simulated magnetic measurements along the actual path of the Rogowski coil to provide a numerical estimate of the integral of the magnetic field along the path. In Fig.2 also an estimate of the current flowing in the vessel is positively compared with the experiment. Simulations have been carried out for different choices of halo region width, giving rise to different evolutions of the plasma configuration. Fig. 3 shows a snapshot at a given time instant and the time evolution of the toroidal current under different assumptions.

Fig. 4 reports the comparison of the experimentally measured signals coming from the halo sensors (Rogowski coils embracing the supports) and the corresponding simulated signals, computed as the numerical surface integral of the current density across the cross-section of the various supports. The simulations are able to reproduce both qualitatively (i.e. positive and negative peaks) and quantitatively (i.e. amplitude of the peaks) the experimental time traces. The most noticeable feature of these results is that even the simulation with no halo currents injected in the structures provides a reasonable agreement with experimental results; the results of the simulations are however all very close, despite the significantly different halo currents injected in the structures. The first conclusion is that the currents measured by the Rogowski sensors placed around the supports are primarily not halo currents, but eddy currents induced by variations of plasma current and position, i.e. part of the "zig-zag" current flowing from the vessel to the plates through the support. Secondly, the halo current injected in the first wall covering the conducting plates flows mostly in the plates themselves, closing back the current loop directly in the plasma. Only a small amount of halo current (as compared to eddy currents) flows through the supports and to the vessel.

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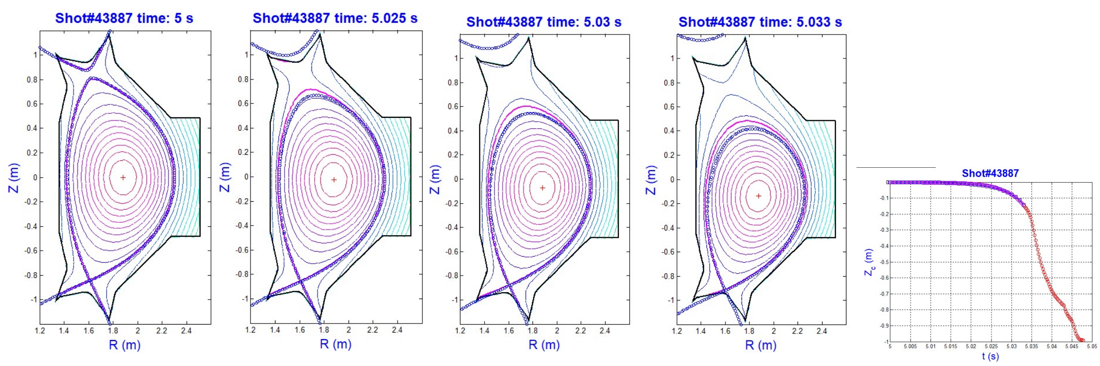

Fig. 1. Time evolution of the plasma during the disruption.

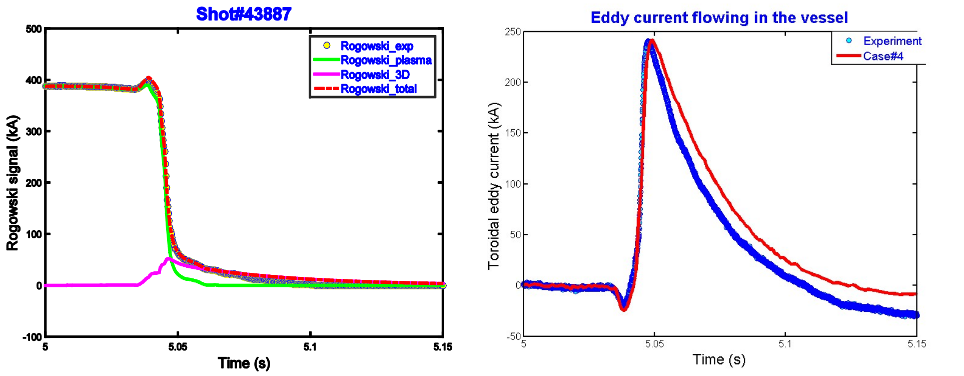

Fig. 2. Time evolution of the Rogowski coil measurement and of the eddy currents flowing in the vessel.

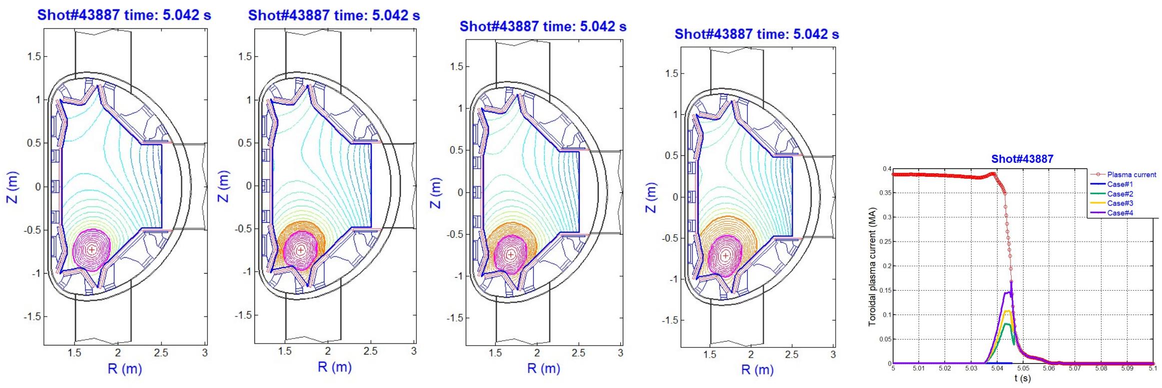

Fig. 3. Equilibria calculated referring to cases#1-4.

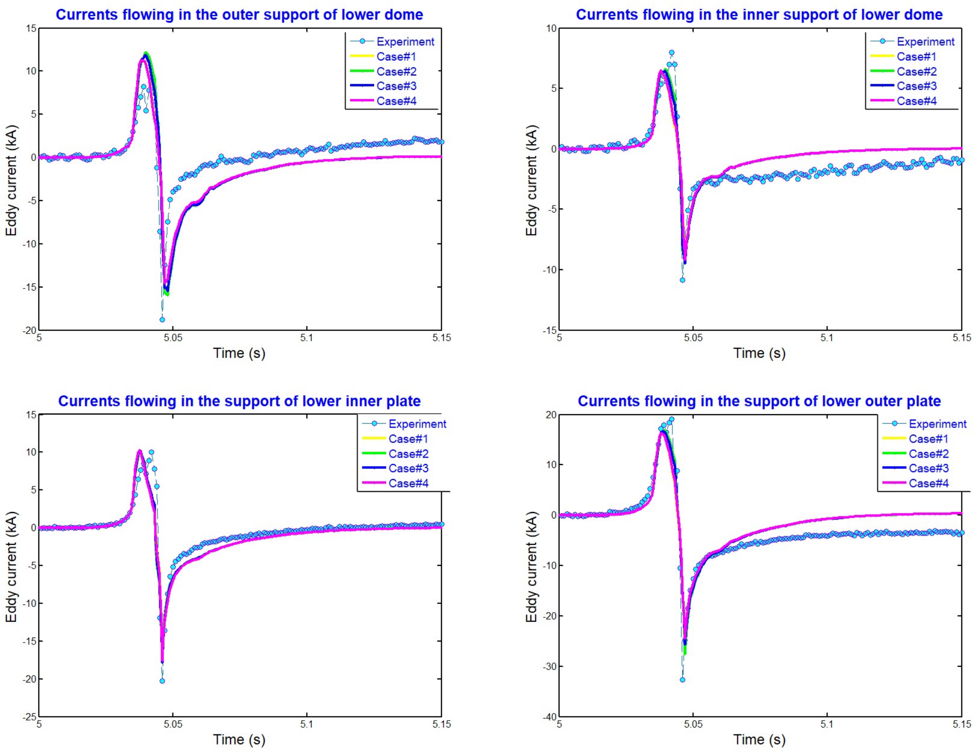

Fig. 4. The time behaviour of currents flowing in the supports of PFCs in the lower-half plane.