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A tool to support the construction of reliable disruption databases

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A B S T R A C T

An algorithm for detection and automatic calculation of disruption main quantities has been proposed and tested on the discharges of recent campaigns in both JET and ASDEX Upgrade. The purpose of this paper is to describe a tool to support the construction of a reliable database, which is theoretically applicable to a wide variety of tokamaks and can support the operators in a very time consuming activity, reducing significantly the possibility of human errors. The algorithm performs its calculations on the basis of common and well defined criteria discussed with the Plasma and Control Operation Groups of the considered devices. Moreover, being the algorithm fully parameterized, it can be easily customized to other tokamaks and/or used for statistical purposes, according to criteria adopted in the framework of each study.

1. Introduction

Disruptions, especially in the last years, have been the subject of several dedicated experiments aiming to support modelling efforts and to provide solid basis for empirical scaling laws [1–3].

Unfortunately, physical models able to reliably recognize and predict these events still elude the scientific community, posing serious concerns in the design and operation of next-step fusion devices, such as ITER. A considerable effort has been devoted in numerical modelling of disruptions. To this purpose, several codes have been implemented and are currently being tested and benchmarked on several devices to model different aspects concerning disruptions, as pre-thermal quench phase, gas propagation of MGI triggered disruptions [8,9], the physics of current quench phase, VDE, halo current diffusion and rotation, toroidal asymmetries of forces and radiation, etc. [10–13]. Nevertheless, a full model of the disruptive process has not been achieved yet, and, despite the well-known operational limits and boundaries, it is still rather difficult to describe exhaustively the underlying conditions that trigger a disruption.

In this framework, in the last years, the exploitation of data-driven approaches to predict disruptions has been continuously developed [14,15,16,17] being applied to always larger databases in order to be able to make confident extrapolations to ITER. Moreover, a multi-device database of disruptions is under development within the International Tokamak Physics Activity of the MHD topical group, with the main purpose of building a common base for modelling, allowing on the one hand to further improve the knowledge of the physics underlying disruptions and, on the other hand, to extrapolate from existent devices to larger scale machines such as ITER. A large amount of data is, therefore, available from several devices, even if often with different accuracies. This is mainly due to the fact that each device has its own diagnostic systems and, consequently, also calculation and elaboration techniques might be different for similar quantities. In general, in order to allow a more consistent cross-device comparison, the aspect of standardization is not negligible.

Analogously, it would be important to define common and shared criteria also to compare different analysis and methodologies. Disruption prediction and avoidance, for instance, require the definition of reference times, as the time of the thermal quench, which available tools should refer to in order to evaluate performance. Therefore, the application of shared and unique criteria should start from the calculation of such reference times.

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[1] See the Appendix of F. Romanelli et al., Proceedings of the 25th IAEA Fusion Energy Conference 2014, Saint Petersburg, Russia.
[2] See http://www.euro-fusionscipub.org/mst1.

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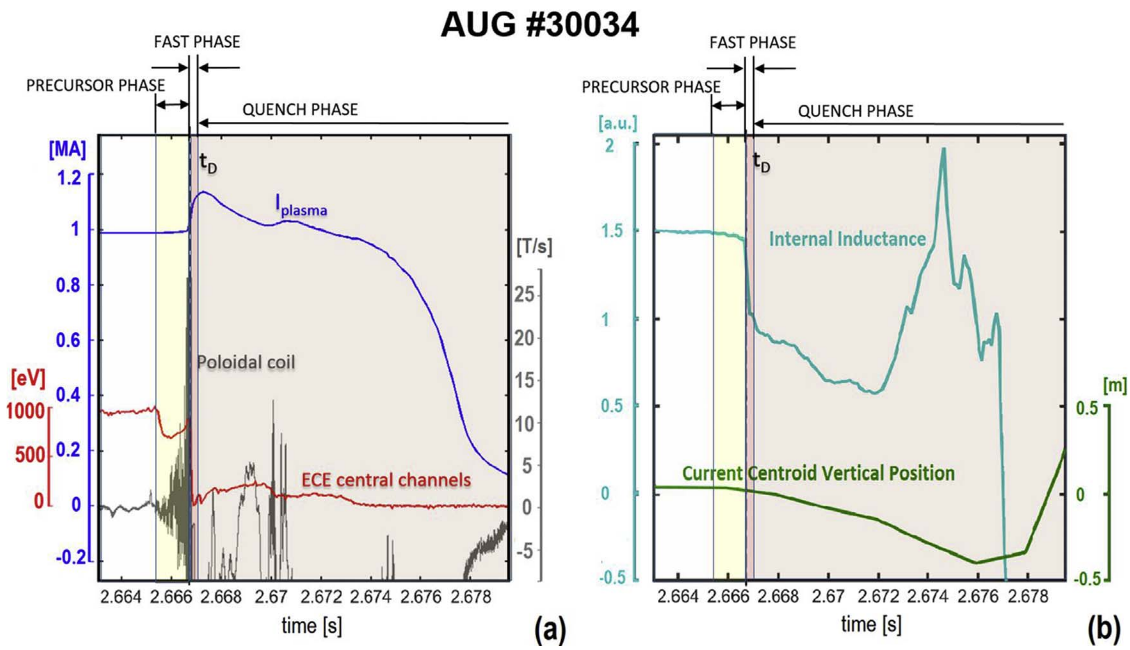

Fig. 1. Time evolution of a) plasma current (in blue), Electron Cyclotron Emission (ECE) central channels (in red), poloidal coil signal (in gray), b) internal inductance (in cyan) and current centroid vertical position (in green) for the AUG disruptive discharge # 30034. The typical phases of the disruptive process are highlighted with different background colors; the time of disruption is labeled with t_D. (For interpretation of the references to colour in this figure legend, the reader is referred to the web version of this article.)

The aim of the paper is to present a tool (named DIS_tool) that supports a user in the construction of reliable disruption databases. Note that, in the paper, JET and ASDEX Upgrade has been taken as reference machines but the tool can be customized to any other tokamak. The tool, starting from shared definitions and criteria and from a basic set of diagnostics, provides the reference times that are non-ambiguously defined with a time resolution suitable for fast transient events detection and without human intervention. Concerning the choice of the disruption time (t_D), it is well known that there is not a really shared and accepted definition of it. Hence, the user, depending on the purpose of his own analysis (prediction, control, etc.), can decide which time to consider, independently on the label "disruption time". Note that, the human intervention is an added value of the tool, in the sense that the tool is not a black box but it can be used interactively to validate certain hypothesis or to use different definitions.

In order to assess the reliability of the produced data, two test sets of 100 disruptions have been considered for JET and ASDEX Upgrade and the deviation of the automatically calculated values with respect to those manually evaluated have been analyzed. The test sets have been built upon disruptions belonging to databases already analyzed in previous works by the same authors [4,5,7,6].

This paper will, in Section 2, summarize basic physics of the disruptive process providing main definitions on the basis of which the tool will calculate characteristic times that will be discussed in the subsequent sections. Sections 3 and 4 will discuss the main requirements for the construction of reliable databases, with particular reference to the basic set of diagnostics, available and routinely in operation in most of the devices, required to perform the calculations implemented in the tool. In Section 5 the tool for disruption analysis will be outlined in its main parts, describing in particular the processing algorithms for thermal quench and current quench calculations (respectively in subsections 5.1 and 5.2). Subsection 5.3 will deal with the detection of pure "hot plasmas" VDEs together with the determination of their disruption times. Finally, section 6, will summarize the conclusions discussing further developments and applications to other devices, highlighting generalization and standardization of the implemented calculation methods aiming to provide a common base for multi-machine database construction and analysis.

2. Disruptive process

A disruption is a sudden and uncontrolled loss of the plasma current in a tokamak: the thermal energy is lost within a time span of few milliseconds, exposing the plasma facing components to severe thermal loads and the conductive structures surrounding the plasma to huge electromagnetic forces. Disruptive instabilities typically present a precursor phase, characterized by a change in the plasma state during the discharge, which often evolves with the development of MHD instabilities. This phase is usually characterized by large tearing modes, among which the (2,1) mode is generally the predominant one, or in specific conditions as high-β plasmas, by an ideal kink mode. The precursor phase is followed by a fast phase, referred to as thermal quench (TQ), where the central temperature collapses in few milliseconds, accompanied by a corresponding increase of the plasma resistivity and a consequent redistribution of the current density. The locking of tearing modes is, very often, the final precursor prior to the disruption. Because of the flattening of the current profile, which is a consequence of the flattening of the temperature profile, and the corresponding reduction of the internal inductance, flux conservation usually gives rise to a characteristic spike in the plasma current with a transient of opposite sign in the loop voltage. The thermal quench precedes the final phase that is known as quench phase or current quench (CQ), being characterized by the loss of the whole plasma current. Even when the disruption itself is not induced by a Vertical Displacement Event (VDE), it is not uncommon that the final loss of plasma current is associated to the loss of vertical stability. The disruption is characterized by a sudden and large change of plasma parameters which, among the other things, affects the vertical feedback control. In fact, it is well known that nowadays plasmas, being elongated for confinement and stability reasons, are normally stabilized in position by a feedback control system.

In Fig. 1, a sketch of the final phases of the disruptive discharge # 30034 of ASDEX Upgrade (AUG) is reported, showing the time evolution of several plasma parameters, which reflect the main phenomenology described above. As clearly shown in Fig. 1(a) by the time evolution of the poloidal coil signal (grey line), a MHD instability takes place in conjunction with the first stage of the core temperature drop