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# ITER safety studies: The effect of two simultaneous perturbations during a loss of plasma control transient

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## HIGHLIGHTS

- We have re-examined the methodology employed in the analysis of the "Loss of plasma transients in ITER" safety reference events.
- We show the possible transient effects of a combined malfunction in external heating system and change in plasma confinement.
- We show the possible transient effects of a combined malfunction in fuelling system and change in plasma confinement.
- We have shown that new steady-states can be achieved that are potentially dangerous for the wall integrity.

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## ABSTRACT

The loss of plasma control events in ITER are safety cases investigated to give an upper bound of the worse effects foreseeable from a total failure of the plasma control function. Conservative analyses based on simple 0D models for plasma balance equations and 1D models for wall heat transfer are used to determine the effects of such transients on wall integrity from a thermal point of view.

In this contribution, progress in a "two simultaneous perturbations over plasma" approach to the analysis of the loss of plasma control transients in ITER is presented. The effect of variation in confinement time is now considered, and the consequences of this variation are shown over a $n$–$T$ diagram. The study has been done with the aid of AINA 3.0 code. This code implements the same 0D plasma-1D wall scheme used in previous LOPC studies.

The rationale of this study is that, once the occurrence of a loss of plasma transient has been assumed, and due to the uncertainties in plasma physics, it does not seem so unlikely to assume the possibility of finding a new confinement mode during the transient.

The cases selected are intended to answer to the question "what would happen if an unexpected change in plasma confinement conditions takes place during a loss of plasma control transient due to a simultaneous malfunction of heating, or fuelling systems?"

Even taking into account the simple models used and the uncertainties in plasma physics and design data, the results obtained show that the methodology used in previous analyses could probably be improved from the point of view of safety.



## 1. Introduction

ITER plasma transients have been studied in the past for safety purposes [1]. The most severe plasma transients expected are disruptions, VDEs and runaway electrons. From the safety studies of these events was concluded that the only consequences affecting the wall integrity would imply the erosion and partial melting of the plasma facing components and, very infrequently, in vessel leaks due to a local perforation of the first wall by runaway electrons.

Together with them, the Loss of Plasma Control transients (LOPC), which postulate a total failure of the three tiered plasma control system, were also included in the Design Basis Accident Study [2].

These postulated transients are the following:

- sudden increase in fuelling rate (2×),
- sudden termination of fuelling or auxiliary heating,
- sudden improvement of confinement time (3×),
- sudden increase of auxiliary heating into steady-state plasma (up to 500 MW of heating power).

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In the past, the loss of plasma control transients have been studied with a coupling of a 0D plasma model and a 1D radial thermal profile for several positions in the poloidal section [3].

The results showed that the plasma discharge would passively shut down once the limits of the operating window are reached, thus bounding the maximum power achievable over the wall.

If the transients last only a few seconds, they will not damage the wall significantly. Otherwise, a long lasting overpower transient would damage the plasma facing components or even achieve to melt the copper heat sink. These cases are therefore used to show the inherent safety of the tokamak operation.

The classical procedure for the analysis of these transients is as follows:

First, equilibrium points are selected over curves of constant fusion power (usually 400, 500 and 700 MW) inside the operating window of the plasma, which is defined by the operation limits of H-mode plasma (Greenwald, beta, H–L transition).

Then, a parametric scan is done over these equilibrium points, for the considered plasma perturbations, to determine the perturbation values leading to the most severe plasma transients from the point of view of the wall thermal integrity.

This is done by propagating in time the perturbed plasma from its initial equilibrium state and analyzing the eventual increase in plasma thermal fluxes over the wall.

In this way, the most interesting transients can be detected, from the point of view of duration and of maximum and medium values of power deposition over blanket and divertor.

Finally, for the selected transients, a complete analysis of the wall thermal equilibrium is done, to detect possible risks for the wall integrity (melting) during the transient.

In a previous work [4], the synergistic effect of simultaneous overfuelling and overheating perturbations over ITER 500 MW inductive reference scenario during a loss of plasma control transient was shown, together with a way to assess the characteristics of the transients over a $n$,$T$ diagram.

This contribution continues that work and presents new results on this topic, now exploring different combinations of perturbations and detecting the critical transients attending to the severity of their effects.

In this contribution, progress in the “two simultaneous perturbations over plasma” approach to the analysis of the loss of plasma control transients in ITER is presented. The effect of variation in confinement time is now considered, and the consequences of this variation are shown over a $n$–$T$ diagram. The study has been done with the aid of AINA 3.0 code. This code implements the same 0D plasma-1D wall scheme used in previous LOPC studies.

The rationale of this study is that, once the occurrence of a loss of plasma transient has been assumed, and due to the uncertainties in plasma physics, it does not seem so unlikely to assume the possibility of finding a new confinement mode during the transient.

Therefore, two cases have been selected for this study, the two of them corresponding to the ITER 500 MW inductive reference scenario. They are intended to answer to the question “what would happen if an unexpected change in plasma confinement conditions takes place during a loss of plasma control transient due to a simultaneous malfunction of heating, or fuelling systems?”

In Section 2, a brief outline of Aina 3.0 code is summarized. In Section 3, numerical results from this study are presented. In Section 4, results are discussed. In Section 5, conclusions are presented.

## 2. The AINA 3.0 code

AINA code is based on SAFALY code, an ITER safety code developed by Honda et al. in the early 90s [5]. SAFALY code is a fortran code which reads input from and writes output to text files. By contrast, the AINA 1.0 version developed at FEEL-UPC included a graphic interface and complete code models documentation [6]. It can be regarded as an improved SAFALY code.

The AINA 2.0 version was a new code, since its physical and numerical models were redefined and then coded in C++. It also included a self-consistent erosion model which allows to calculate the varying erosion rate during plasma transients [7].

The AINA 3.0 version implements several improvements. First, now the terms participating in the 0D balance equations are fully integrated over the radial profile, without using mean values. This has a noticeable influence over the values of power losses, for example. For the integration, parabolic profiles are used.

Another improvement, the finite volume blanket discretisation model implemented in AINA 2.0 version was changed, in order to control the convergence and stability of the solution, by introducing varying meshes and stability and convergence checks.

The wall configuration parameters were also changed, to allow for flexibility in defining the number and position of the poloidal calculation regions, with the purpose of providing the users of the code with the ability to configure new scenarios, and to bring the wall design definition closer to the real ITER design.

The calculation scheme remains the same used in previous LOPC analyses: a 0D plasma model coupled with a 1D thermal equilibrium model applied over several calculation regions in which the poloidal section is divided.

The 0D plasma model comprises two energy balance equations, for electrons and ions, and different mass balance equations for each species. For the calculation of the steady state equilibrium, several options are available. If the objective is to match the equilibrium state of an ITER reference scenario [8], the fusion power and the $Q$ value are taken as inputs.

The plasma transient is calculated as an initial value problem for the 0D mass and energy balance equations, where the initial value is the steady state equilibrium.

The impurity fractions are defined as input parameters for the calculation of the steady state equilibrium, so the electronic density can be calculated directly by using the charge balance, assuming complete ionization for the different ions. The electronic temperature is also calculated from the electron energy equilibrium. The equilibrium is solved by using a numerical root-finding method. IPB-98(y,2) scaling for confinement time is assumed.

A 1D wall model solves the heat transfer equilibrium equation in several calculation regions in which the code divides an ITER poloidal section. The equation is solved considering a Neumann boundary condition for the surface heat flux coming from the plasma and a Dirichlet boundary condition at the back side of the blanket. For the steady-state, an elliptic problem results, making zero the time derivative, whereas for the transient calculation, a parabolic problem is solved.

The cooling system is modelled with the simplifying assumption of considering coolant tubes arranged at several radial positions, in the toroidal direction. For these radial positions, the coolant tubes are supposed to occupy a fraction of the transversal area, whereas the rest corresponds to the structure. The transversal conduction effects are neglected.

The volumetric neutron heat source is modelled with an exponential function, whose coefficients have been adjusted to fit a MCNPX simulation of the neutronic heat deposition on blanket and divertor during an ITER reference scenario.

The heat flux from the plasma splits in two sources, the surface flux from EM radiation and particle fluxes, and the volumetric neutron flux from the fusion reaction. Both surface and neutronic heat fluxes in the first wall are estimated from plasma losses calculation, by applying a safety coefficient over the peaking factors of the poloidal distribution. The divertor surface heat flux is calculated with the algorithm pointed out by Honda [5]. The AINA models