Study of ITER plasma position control during disruptions with formation of runaway electrons

V.E. Lukash[1], A.A. Kavin[2], Y. Gribov[3], R.R. Khayrutdinov[1], A. Loarte[3]

[1]*Kurchatov Institute, Moscow, Russia,* [2]*NIIEFA, St. Petersburg, Russia*

[3]*ITER Organization, Route de Vinon sur Verdon, 13115 St Paul Lez Durance, France*

Introduction

Generation of runaway electrons (RE) in disruptions may lead to damage of the first wall. The damage can be significantly reduced, if the poloidal field (PF) system allows control of the RE beam position avoiding the beam interaction with the first wall [1, 2]. Due to up/down asymmetry of the ITER conducting structures relative to the plasma current centroid, fast reduction of the plasma current during generation of RE leads to the plasma rapid vertical jump, if the current centroid is not in the “neutral point” [3]. This rapid and significant vertical jump of vertically unstable plasma has a time scale much lower than decay time of the vacuum vessel eddy currents and therefore vertical displacement of the RE beam cannot be stopped by the PF coils located outside the vacuum vessel. In this case only the in-vessel coils (VS coils) can be used for RE vertical stabilization. Results of the study of capability of ITER PF system to control position of RE in the absence of disruption precursor have shown that this capability is very limited [4]. If the rapid reduction of the plasma current in the initial phase of the current quench is larger than 1 MA (i.e. maximum current of RE, I_{re}^{max}, is less than 14 MA in a 15 MA ITER plasma), the RE vertical stability is lost due to the limitation on current in the VS coils (60 kA). The control of RE could be improved if the disruption precursor can be diagnosed about 1 s before the disruption (e.g. when auxiliary heating is lost) by moving the plasma closer to the neutral point before the disruption.This paper presents a special control scheme developed for RE position control that implement such strategy. The key element of the new control scheme is the plasma fast vertical shift by the VS coils to a more vertically stable optimum position (by ≈ -12 cm) before the disruption. This control scheme was validated with the DINA code [5] in simulations of RE position control during disruptions caused by “unexpected” loss of the auxiliary heating.

Plasma model

Before the disruption, the transport model includes the diffusion equations for the electron and ion temperatures. After the thermal quench, the temperatures were adjusted for getting a desired maximum value of RE current during the current quench. The electron density is prescribed. The generation of the RE current density, j_{RE}, on a magnetic surface with minor radius r, is evaluated by the avalanche model [6] with the initial runaway source S_{RE} from Dreicer acceleration [7]:

$$\frac{\partial j_{RE}}{\partial t} = \frac{j_{RE}}{\tau \ln \Lambda}\sqrt{\frac{\pi\gamma}{3(Z_{eff}+5)}}F\left(\frac{E_{||}}{E_c},\gamma\right) + S_{RE} - \frac{j_{RE}}{\tau_{loss}}. \quad (1)$$

Here τ_{loss} controls the decay of the RE current, $\tau = mc/eE_c$, $ln\Lambda$ is the Coulomb logarithm, γ is the function specified in [6] which depends on $\varepsilon = r/R$, $E_{||}$ is the toroidal component of the electric field and E_c is the critical value of $E_{||}$, below which formation of the RE current is not possible. It should be noted, that the decay of the RE current due to the term with τ_{loss} contributes to the electric field $E_{||}$, which, due to the first term in equation (1), results in the RE current decay time being significantly longer than the parameter τ_{loss}.

Scenario before disruption

In this study we simulated disruptions caused by the loss of auxiliary heating ($P_{aux} = 0$ at $t_0 = 317$ s) during the burn in 15 MA DT scenario. After t_0 the total heating power $P_{tot} = P_\alpha + P_{aux} + P_{Ohm}$ decreases. When P_{tot} becomes less than P_{HL}, where P_{HL} is the power threshold for the H to L mode transition, the H to L transition takes place ($t_1 = 317.7$ s) and the electron density decreases: $< n_e > = 10^{20} \cdot \exp[-(t - t_1)/4.5s]\ m^{-3}$. The disruption (thermal quench) starts at $t_{tq} = 319.012$ s, when the power conductive flow through the separatrix $P_{sep} = P_\alpha + P_{aux} + P_{Ohm} - P_{rad} - \frac{dW_p}{dt}$ becomes less than 50 MW, which is the expected level of divertor radiation. Here P_{rad} is the radiated power in the core plasma and W_p is the plasma thermal energy. At $t \geq t_{tq}$ the value of $< n_e >$ is assumed fixed. Fig. 1 shows time evolution of the main plasma parameters from 1 s before the loss of auxiliary heating until the start of disruption (thermal quench).

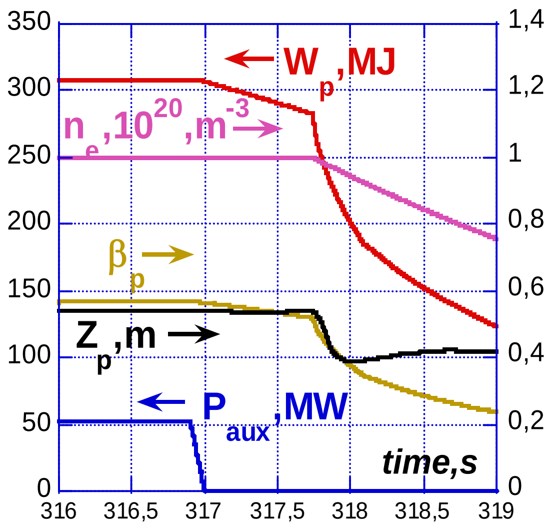

Fig.1. Time evolution of Z_p, β_p, W_p, P_{aux} and plasma electron density before t_{tq}

Magnetic control before thermal quench

Before the H to L mode transition (before the disruption precursor) the basic controller for divertor magnetic configurations ("DINA 2010") is used. The plasma vertical stabilization is performed by the VS3 feedback circuit (in-vessel VS coils), using as input vertical velocity of the plasma current centroid, dZ_p/dt [8]. After the

H to L mode transition, until t_{tq}, the stabilization of plasma vertical displacements is switched to the control of plasma vertical position, using in the feedback loop the signal Z_p. In this case, before the current quench, the plasma can be moved vertically by a target value, ΔZ_{ref} (Fig. 1), which was optimised in preliminary simulations with the goal of stabilization of the plasma vertical position within the engineering limits on the VS3 current (60 kA) and voltage (65 kV) during the thermal quench and the ensuing rapid drop of the plasma current at the initial phase of the current quench. In the scenarios considered, the optimum value of ΔZ_{ref} is about -10 cm, resulting in identical positive and negative peak values of the VS3 current.

Magnetic control during current quench

After the beginning of thermal quench, the divertor controller "DINA 2010" is switched in a few seconds (≈ 3 s) to a limiter one controlling the plasma elongation and maximum radius of its boundary (R_{max}). At this phase, the plasma touches the inboard region of the first wall due to the fast drop of the plasma current. Time traces of the VS3 current and voltage, together with the coordinates of plasma current centroid just after thermal quench, are shown in Fig. 2 for the a case with $I_{re}^{\max} = 9.7$ MA and $\Delta Z_{ref} = -0.1$ m. The plasma position is recovered with VS currents and voltages within engineering limits. Analysis of the plasma position control was carried out for different values of $I_{re}^{\max}$. If $I_{re}^{\max} < 9.7$ MA, the plasma vertical control is lost due to the limits on the VS coils currents and voltages. If $I_{re}^{\max} \geq 9.7$ MA, the plasma position control is possible if the parameter τ_{loss} is not too low. Results of simulations of four scenarios with $I_{re}^{\max} = 9.7$ MA and different values of τ_{loss} (300, 400, 500 and 1000 ms) are shown in Fig. 3. One can see that for $\tau_{loss} > 350$ ms, which corresponds to $|dI_p/dt| < \approx 0.5$ MAs^{-1} (i.e. an effective loss timescale for runaway current of ~19.5 s once avalanche generation is taken into account), the plasma position control during disruption is possible down to a plasma current ≈ 2 MA (Fig. 3a). During decay of the plasma current, the plasma is kept at $Z_p \approx 40$ cm (Fig. 3b) and $R_p \approx 560$ - 580 cm (Fig. 3c). Minimum values of τ_{loss}, which still allow the control of the plasma position within the VS coil current and

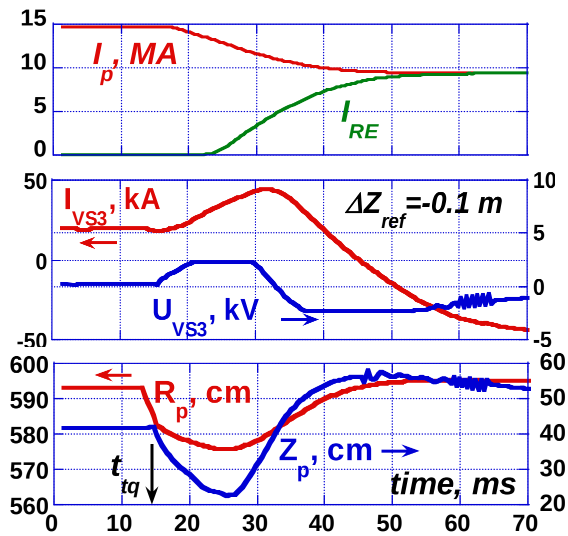

Fig. 2. Time traces of I_p, I_{RE}, I_{VS3}, U_{VS3} and plasma position in case of $I_{RE}^{\max} = 9.7$ MA just after thermal quench

voltage limits, are shown in Fig. 4a as a function of $I_{re}^{\max}$. The corresponding maximum values of dI_p/dt are shown in Fig. 4b.

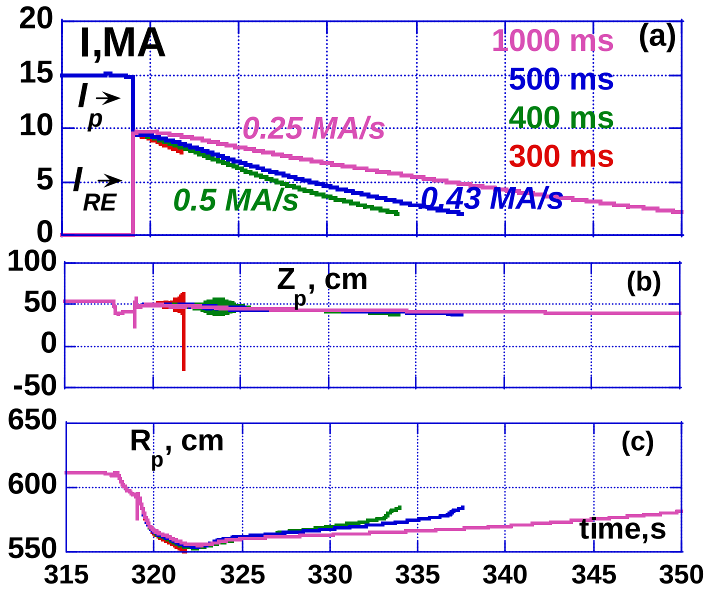

Fig. 3.Time traces of I_p, I_{RE}, Z_p and R_p during current quench in case of $I_{RE}^{\max}$ = 9.7 MA and different τ_{loss}

Conclusions

A special control scheme has been demonstrated for the stabilisation of a RE beam for disruptions in which precursors can be diagnosed about 1 s before the disruption in ITER. The key element of the control scheme is a fast vertical shift of the plasma by the VS in-vessel coils to the optimum position from the vertical stability point of view before the disruption. In the disruption scenarios considered, the optimum position of the plasma current centroid is located about 12 cm below its position before the disruption. The application of this new scheme to 15 MA disruptions in ITER allows the control of the runaway plasma position control down to a runaway current of ≈ 2 MA, provided that $I_{p0} - I_{re}^{\max} < \approx 5$ MA and $\tau_{loss} \geq 350$ ms, (which is equivalent to a maximum $|dI_p/dt| < \approx 0.5$ MAs^{-1}).

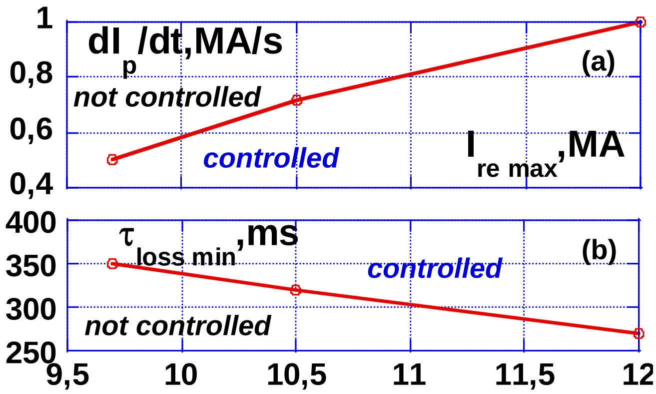

Fig. 4. Limits to of plasma position control of runaway plasmas in ITER in terms of dI_p/dt (a) *and* τ_{loss} (b) *versus runaway current.*

Disclaimer: The views and opinions expressed herein do not necessarily reflect those of the ITER Organization.

References

[1] F.Saint-Laurent et al. 36th EPS Conf. on PPCF, Sofia, Bulgaria, June 29 – July 2, 2009, P4.205
[2] N.Commaux et al. Nucl. Fusion **51** (2011) 103001
[3] Y Nakamura et al, Plasma Phys. Control. Fusion **44** (2002) 1471
[4] 19th Meeting of the ITPA MHD Topical Group, March 5-9, 2012, Toki, Japan
[5] R.R. Khayrutdinov and V.E. Lukash. Journal of Comp. Physics, **107** (2), 106 (1993).
[6] Rosenbluth, M.N., Putvinski, S.V., Nuclear Fusion, v. 37 (1997) 1355
[7] Parail, V.V., Pogutse, O.P., in “Reviews of Plasma Physics”, ed. by B.B. Kadomtsev, Consultans Bureau, New Yirk (1986), v. 11, p.1
[8] V.E. Lukash, et al. 39th EPS Conf. on PPCF, Stockholm, Sweden, 2 – 6 July, 2012, P2.058