ITER Plasma Vertical Stabilization

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Abstract
In this paper we summarize the results of the modelling, analyses and assessment of the ITER Vertical Stabilization (VS) system design carried out in the last two years by the ITER IO and Domestic Agencies as well as its possible improvements: these include the use of an additional VS system connecting some of the Central Solenoid coils, a power supply upgrade from 6 kV to 9kV, the use of passive stabilizers and in-vessel active coils. The assessment is performed on the basis of key figures of merit that are, in most cases, dimensionless and feedback control independent thus allowing ITER to be compared to present day experiments. On the basis of these indicators, different design solutions for possible VS system upgrades are compared. Experimental guidance is of key importance to understand the level to which present-day machines are able to push their VS systems without increased frequency of VDEs and experimental data are presented. Conclusions and recommendations are given for an integrated use of in-vessel active coils for the control of axi-symmetric and higher order modes.

1 Introduction and Overview

In present and future tokamak devices featuring elongated plasmas cross-sections, vertical stabilization must be reliably maintained over a wide range of operating conditions including off-normal event such as emergency shutdown. Any failure or poor reliability of the plasma Vertical Stabilization (VS) system will lead inevitably lead to a loss of machine performance or to an increase in Vertical Displacement Events (VDEs) rate and associated thermal and mechanical loads on several key machine components (e.g. blanket modules).

In tokamaks plasma vertical stabilization is achieved by combining passive stabilization by the image currents flowing in the surrounding metallic structures with active (feedback) control obtained by powering external coils to produce an n=0 horizontal field. In ITER – whose full bore plasma has a typical elongation $\kappa \sim 1.85$ (Fig. 1) - the VS system is designed on the basis of some, most unstable plasma equilibria to be controlled, likely plasma disturbances affecting such equilibria and an assumed level of noise entering in the feedback loop [1]. On this basis, a solution was initially found by adopting a plasma “tight-fitting” vessel design to provide passive stabilization and by connecting the four outermost Poloidal Field (PF) coils in anti-series to generate the required radial field for plasma active stabilization [1] (Fig. 1).

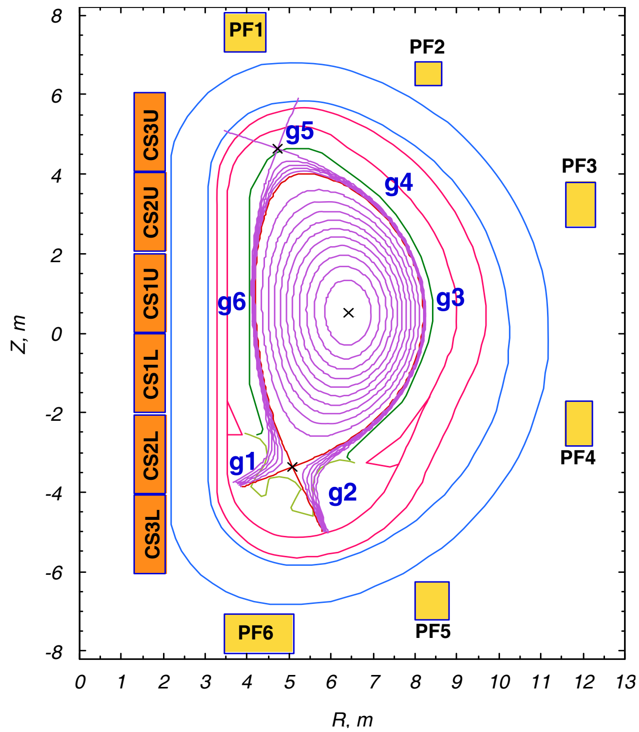

FIG. 1. ITER cross-section view: passive stabilization is provided by the double-wall vessel, active control by connecting in anti-series and powering PF2-PF3 to PF4-PF5

Recent evaluations of the ITER inductive scenario [2, 3] and experimental simulations [4], has highlighted that, under various conditions, the present VS system capability may need more margin to accomplish the ITER mission. Particular concern relates to the plasma current decay, when the plasma internal inductance l_i tends towards values above the design range of 0.7÷1. In addition, plasmas at reduced current, which will be required for the development of I_p~15 MA Q~10 scenarios, will tend towards high l_i values. These concerns are compounded by uncertainties in the level of magnetic noise entering the feedback loop, plasma-generated fluctuations affecting the magnetic sensors, which measure the speed of the plasma centre, and plasma displacements amplitude due to ELMs, etc. On this basis, steps have been taken in the last two years to improve the ITER VS system design by (1) increasing the stabilization voltage of the present PF system, (2) maximizing the passive stability by the metallic structures surrounding the plasma and (3) exploring the use of in-vessel copper coils. An effort was also made to better characterize the perturbations to the vertical stabilization circuits in existing tokamaks and to extrapolate these findings to ITER.

The aim of this paper is to summarize the assessment of the ITER VS system design made in the last two years by the ITER IO in collaboration with some Domestic Agencies and to describe possible design upgrades that may bring significant improvements. We report in Section 2 the formulation of the main criteria applied to the VS system design, in Section 3 the main results of the performance assessment and provide some conclusions and outlook in Section 4.

2 Vertical Stabilization System Design Criteria

2.1 Plasma Stabilization Indices

The results reported in the following sections are obtained by linearizing the plasma response to external voltages applied to the PF coils and to changes in the plasma profile parameters l_i and β_p about MHD equilibria. The resulting plasma response is consistent with the *n=0* MHD equilibrium constrain but it does not account for high frequency modes associated to plasma inertia [5]. The resulting linear models have been validated on experiments [6-8] and widely used in previous ITER studies [1]. The basic dynamic system reads as [9]:

$$\begin{aligned} \mathbf{L}^{*}\dot{\mathbf{x}}+\mathbf{R}\mathbf{x} &= \mathbf{v}+\mathbf{E}^{*}\dot{\mathbf{w}} \\ z_p &= \mathbf{C}\mathbf{x}+\mathbf{F}\mathbf{w} \end{aligned} \qquad (1)$$

In (1) above **x** is the vector of (first order perturbed) passive and active currents, **L*** is the modified inductance matrix [9] and **R** the resistance matrix of the coils, **v** are the voltages applied (whose entries are zero for the passive currents equation), the output matrix **C** maps the currents **x** in the displacements of the plasma current centre z_p. The matrices **E*** and **F** map a current profile variation (parametrized by $\mathbf{w} = [l_i(3), \beta_p]$) in the corresponding variation of **x** and z_p. The current-averaged poloidal flux is conserved by the perturbation described by (1). We stress that (1) is valid also in the case of 3D models of the metallic structures provided that the ($\mathbf{L}^*$, **R**) matrices and the currents **x** are defined accordingly [9].

The instability growth rate γ (growth time τ_g=1/γ) is the largest positive eigenvalue of the dynamic matrix $\mathbf{A} \equiv -\mathbf{L}^{*-1}\mathbf{R}$. The stability margin m_s is the largest positive eigenvalue of the stability matrix $\mathbf{M} \equiv -\mathbf{L}^{-1}\mathbf{L}^*$, where **L** is the inductance matrix of the coils without plasma [10].

In the following we will also refer to the maximum initial displacement z_{p0} stabilizable by the available control voltage $\mathbf{v}_0$ [4, 10], to the maximum tolerable time delay T_{max} in the control loop and to the maximum stabilizable plasma sudden variation of $\beta_{p0,max}$ and/or $l_{i0,max}$. The last two indices essentially relate to the excitation of the unstable mode that occurs in an up-down asymmetric system (such as ITER) by plasma current profile changes.

2.2 Plasma Equilibrium Configurations

The ITER VS system is designed and assessed on the basis of a broad range of plasma equilibria that are representative of the various operation scenarios (e.g. ELMy H-mode, hybrid, etc.) and that are most demanding for vertical control. These conditions are, typically, associated to MHD equilibria with low kinetic pressure (e.g. L-mode plasmas), high separatrix elongation and peaked current profiles since these plasmas maximize the external destabilizing force (high κ_x, high l_i) and minimize the coupling to the metallic structures that provide passive stabilization (low β_p).

Experimental simulations of ITER scenarios on JET, ASDEX-U and DIII-D [2, 3] confirm that vertical instabilities are most dangerous during the current ramp-up/down phases of non-heated tokamak discharges. In particular [2], current peaking as high as $l_i(3)$~1.2 can be reached during ohmic ramp-ups and $l_i(3)$~1.8 during ramp-downs. These values are consistent with the scaling $l_i \sim [0.5+\ln(q_{95}/q_0)]2\kappa_x/(1+\kappa_x^2)$ [11] once the constraints q_{95}~3 and q_0~1 are included and κ_x ~1.8.

To mitigate the detrimental effects of such large increases in l_i at ramp-down, the ITER reference operating scenarios feature a prompt reduction of κ_x once I_p is reduced and the use of additional heating. To study the effect of l_i and κ_x on Ohmic plasmas, in Fig. 2 are shown a full aperture plasma with minor radius a~2 m, κ_x ~1.84 and β_p ~0.1 and the reduced elongation sequence adopted in the ITER I_p ramp-down scenario that still satisfies the criterion q_{95}>3. In Fig. 3 are reported the corresponding growth times. As shown, with κ_x ~ constant the reduction of τ_g saturates at $l_i(3)$ ~1.6 and such plasmas may reach instability growth times as low as τ_g~50 ms.

It is worth noticing that unlike the passive stability indices (e.g. τ_g), the active control indices (e.g. voltage and current needed to stabilize a given plasma displacement) depend upon I_p, higher current plasmas being more demanding for control. In particular, the minimum voltage necessary to stabilize a given offset z_{p0} scales as $v_0 \sim z_{p0}\, I_p\, \gamma$. Therefore, for the sake of the design of the VS system, in the following we will focus on low beta Ohmic (β_p~0.1), high l_i (~1.2), high current (I_p~15 MA) plasmas as they are most demanding for vertical stabilization.

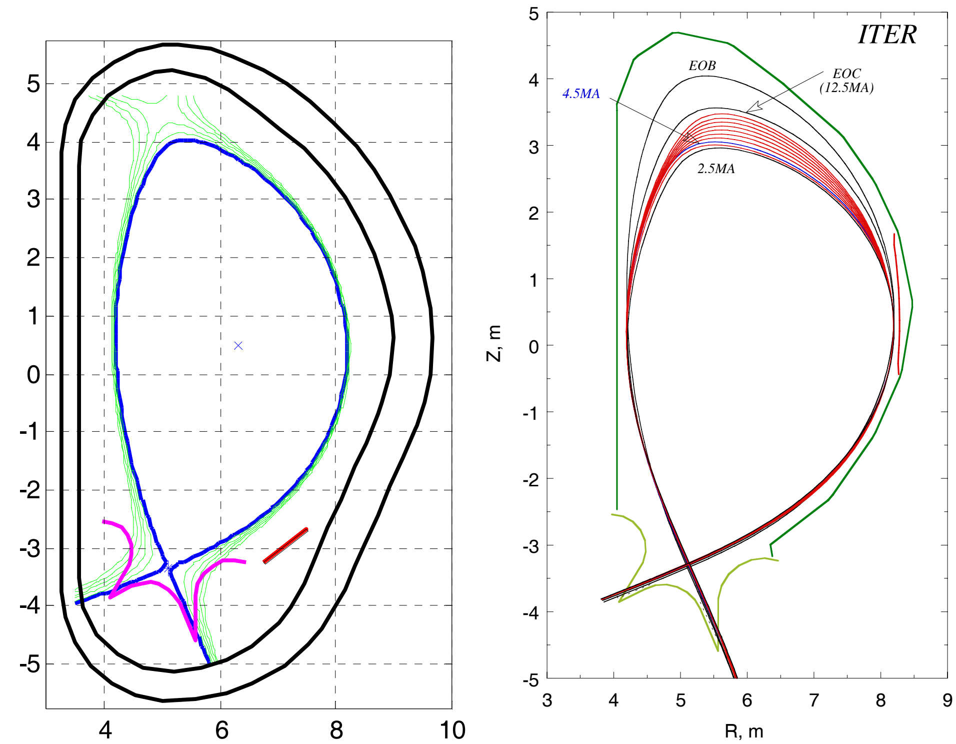

FIG. 2. Left: Plasma equilibrium configuration with κ_x ~ 1.84, l_i = 1.2, β_p = 0.1. Right: plasma elongation reduction adopted to reduce the vertical instability during current ramp-down.

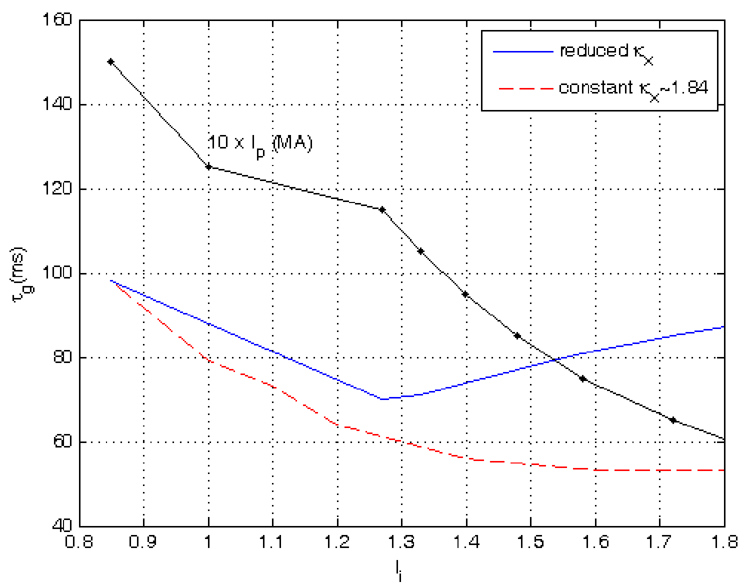

FIG. 3. Growth time dependence on l_i for different plasma shapes evolution in Ohmic (β_p~0.1) ramp-down. Additional heating may be used to limit li excursions [2]

2.3 Plasma Disturbances and Noise

A key element of any VS design is the estimation of the spectrum of disturbances to be controlled by the feedback control system, and in this study we focus on the excitation of the unstable mode by external events (e.g. failure in the control system).

Following the experimental guide-lines [4], we design the VS system to stabilize initial offsets in the range z_{p0}/a~5%÷10%. As shown in [12], H→L transitions excite the unstable mode to a much lesser extent due to the proximity of ITER plasmas to the equilibrium neutral point [13] and it will not be analysed in detail.

In addition to plasma drifts and profile changes – that trigger *n=0* plasma motions - the VS system must be robust to external as well as plasma generated magnetic noise affecting the *estimation* of the plasma vertical velocity $\dot{z}_p$ (since this is the variable fed-back) by the diagnostic system [14]. Scaling from JET [12] suggests that in ITER noise with "white" spectrum up to ~1 kHz and with standard deviation σ~0.6 m/s is likely to affect $\dot{z}_p$.

3 Results

3.1 Comparative Performance Analysis

Several design solutions have been proposed to improve the VS system performances with respect to the reference baseline solution described in Section 1, that is use of a plasma tight fitting vessel and PF2-4 ex-vessel coils with 6 kV voltage limit. The solutions have been assessed with 2D axi-symmetric models of the vessel and for the plasma equilibrium shown in Fig. 2 by comparing τ_g, m_s and z_{0max}. Four alternatives have been analyzed, namely:

(1) **Ex-vessel coils**. Increase by 50% the active control voltage in PF2-5 (i.e. from 6kV to 9 kV) and use of an additional 6 kV converter connected in anti-series to CS2U and CS2L (Fig. 1). This solution has the main disadvantage of imposing higher operation and test voltage to these coils but it minimizes the modifications to the present design;
(2) **Copper cladding**. Passive stabilization capability can be improved by applying a thin (~ 1.2 mm) layer of copper on the inner vessel shell on plasma-facing side (Fig. 4a). This solution offers a low impact on PF coils with moderate impact on vessel engineering, but limited improvement in vertical stability performance.
(3) **Toroidal straps**. Alternatively, improved passive stabilization is achieved by connecting electrically through toroidal straps several shield blanket modules rows (Fig. 4b). This solution has significant engineering and cost impact on the blanket modules and vessel, but offers equally significant improvement in controllability performance.
(4) **In-vessel coils.** A pair of toroidal copper coils of 40 cm^2 cross section is clamped to the vessel inner shell on the plasma facing side (Fig. 4c). These coils are modeled as a pair of single-turn, toroidal ring coils connected in anti-series [12, 15]. This solution represents the highest cost and impact on vessel design, but also the highest performance improvement.

The results obtained by the CREATE [12] and PET [15] codes are reported in Table I for the solution with ex-vessel coils (1-3) and in Table II for the in-vessel coils (4). As shown, the ex-vessel coils can stabilize only small initial offsets of these highly unstable plasmas, well below the z_0/a~0.1m/2 m ~ 5% criterion [4]. In fact, as V_{max} scales linearly with z_0, to achieve z_0~ 100 mm it would be necessary to reach V_{max} ~ 30 kV.

If toroidal straps are used, despite the improvement brought to passive stabilization, the magnetic measurements of the plasma position are strongly perturbed [12] and most of the performance gain is lost once the detrimental effects on magnetic diagnostics (i.e. sensors screening by the passive stabilizers currents) are accounted for. This solution can be viable only if full information regarding the currents flowing in the passive stabilizers is available.

The in-vessel coils, on the other hand, meet the all requirements, the performance limits being now set by the peak current I_{max} (related, for example, to the feeder cross section) and, more generally, by several design integration issues (e.g. routing, blanket manifolds interference, etc.). Since control is faster in the absence of vessel shielding, operational robustness against VDEs is greatly improved, position overshooting reduced (likewise any potential plasma-wall contact) and the VS system has now intrinsic margin against feedback control delays and model uncertainties.

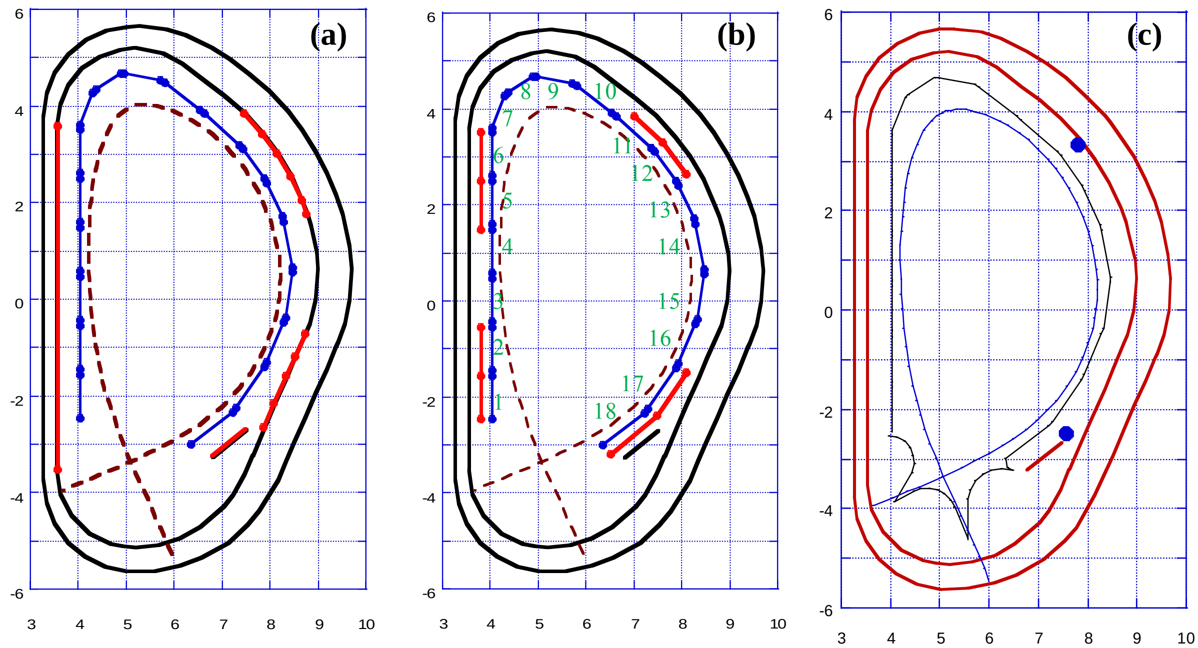

FIG.4. Location (red) of (a) copper cladding surfaces, (b) toroidal straps, (c) in-vessel coils (blue dots).

Other (not reported) variants have also been analyzed (e.g. thicker copper cladding, partitioning of PF2-PF5 coils to reduce the VS circuit inductance) but they do not bring any substantial improvement upon the results quoted Table I below.

All these analysis have been performed by 2D models of the vacuum vessel. The effects of the presence of the vessel ports have been evaluated with an extension of the CarMa code [16] to the n=0 case. It is confirmed that, as preliminary reported in [12], the presence of the vessel ports increases the growth rate γ by ~ 10 %.

TABLE I: EX-VESSEL COILS PERFORMANCE ANALYSIS FOR
I_p=15 MA, β_p=0.1, $l_i(3)$=1.2 PLASMA. CREATE (**bold**) AND PET (*italics*) CODES

	τ_g (ms)	$\tau_g / \tau_{L/R}$	m_s	z_{p0max} (mm) VS1 6 kV	z_{p0max} (mm) VS1 9 kV	$z_{p0,max}$ (mm) VS1 6 kV VS2 6 kV	z_{p0max} (mm) VS1 9 kV VS2 6 kV
Ex-vessel coil	**64** *64*	**0.28** *n.a.*	**0.37** *0.37*	**22** *20*	**33** *n.a.*	**35** *32*	**46** *n.a.*
Copper cladding	**88** *89*	**0.32** *n.a.*	**0.42** *0.41*	**34** *25*	**51** *n.a.*	**51** *40*	**68** *n.a.*
Toroidal straps	**158** *155*	**0.45** *n.a.*	**0.62** *0.61*	**67** *53*	**101** *n.a.*	**101** *75*	**135** *n.a.*

TABLE II: IN-VESSEL COILS PERFORMANCE ANALYSIS FOR I_p=15 MA, β_p=0.1, $l_i(3)$=1.2 PLASMA. CREATE (**bold**) AND PET (*italics*) CODES

z_{p0} (mm)	*100*	**100**	*100*
V_{max} (kV)	*0.125*	**0.3**	*0.5*
I_{max} (kA)	*130*	**105**	*110*

3.2 In-vessel Coils Analysis

From the comparison above it is clear that the in-vessel coils design is the only solution able to meet the necessary VS performance requirements (Table II). In particular, the faster response of these coils result in a superior close loop performance (e.g. smaller position overshoot, higher phase margin, overall system robustness) that cannot be matched by any ex-vessel coil design.

The 1-turn coils design of Table II requires high currents (100÷200 kA) and high associated costs that can be reduced by a multi-turn design and by raising the terminal voltage. For example, a 3-turn coils pair with V_{max}~1.5 kV, I_{max}~75 kA can stabilize z_{0max}=100÷200 mm. Fig. 5 shows the present in-vessel coils that are strongly coupled –mechanically and magnetically- to the Resonant Magnetic Perturbation (RMP) coils, recently introduced in the ITER design to suppress ELMs.

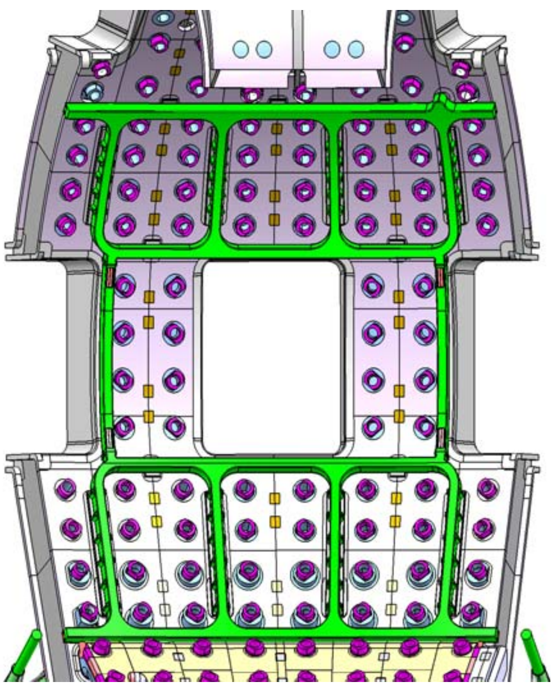

FIG. 5. Vessel sector (1 of 9) with the upper, middle and lower RMP coils and upper and lower VS coils

The 3D effects associated with the magnetic coupling between the RMP-VS coils have been quantified in [12], which show that this coupling is important as it increases the peak control current I_{max} by 30-40%. Due to the large difference in time response between in-vessel and ex-vessel coils, PF2-5 coils do not reduce the peak current I_{max} in the in-vessel coils [12].

One drawback of such copper coils is their temperature increase due to active current fluctuations resulting from magnetic noise pick-up and amplification. Magnetic noise with σ~0.6 m/s results in an rms current – averaged over a 1 s window – I_{rms} <10 kA, I_{max}<25 kA, V_{max}<150 V and plasma integrated oscillation < 1 cm. Accurate filtering of the $\dot{z}_p$ signal may reduce I_{rms} and, with them, coils heating and power consumption.

Major disruptions and VDE's can lead to peak terminal-to-terminal voltages up to V_{max}~1.3 kV (open-circuit) and turn currents up to ~ 70 kA (short-circuit). Provided that the coil supports are properly dimensioned and the power supply protected, no major problems are envisaged during these off-normal conditions as they do not differ substantially from the normal operation ones. Further analysis is being carried out to quantify the consequences of different failure modes.

4 Conclusions

In order to provide the ITER VS system with the margins that, from present experimental practice, are recognized as essential to avoid high plasma disruption and VDE frequency in operation, the most effective solution of those analysed is the use of in-vessel coils. This may in fact be the only one to reliably control higher l_i plasmas at full elongation.

Two sets of 3-turn coils located above and below the machine mid-plane and connected in anti-series to a single power supply have been proposed in addition to the present VS1 system that should be maintained for redundancy at a 9 kV on-load voltage rating capability. It is also recommended to make all necessary provisions to enable the installation of the VS2 circuit at a later stage, if necessary.

The in-vessel coils power supply should be rated for current of ~75 kA and terminal voltage of 1.5 kV to guarantee reliable operation (z_{p0}/a~10%) of ITER plasmas during the normal operation phases envisaged and in presence of magnetic noise and large disturbances.

In off-normal conditions (e.g. major disruptions and VDEs) these coils are not expected to be over-loaded beyond their normal operational requirements.

Given their key function for reliable operation, the newly proposed coil set should become a fully integrated, maintainable component of the new ITER baseline design and, in case of failure, it should be efficiently replaced to avoid reduce performance operation and long shut-down times.

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