

Plasma Control in Tokamaks. Part 3.1. Plasma Magnetic Control Systems in ITER

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Abstract: The plasma magnetic control systems for ITER (International Thermonuclear Experimental Reactor) are presented. The systems comprise original engineering solutions for plasma position, current and shape control for the two versions of ITER: ITER-1 and ITER-2, including those designed in V. A. Trapeznikov Institute of Control Sciences of the RAS. It is noted that in ITER-1 the plasma position and shape were controlled by all PF-coils and robust H_∞ -controllers, while to decrease peaks of control power for suppressing minor disruptions the additional nonlinear circuit was used without significant changes in displacements of the gaps between the plasma separatrix and the first wall. In ITER-2 the special circuit with a fast voltage rectifier was used for plasma vertical speed stabilization about zero, while for plasma current and shape control the special cascade control systems were designed with and without the control channels decoupling, with robust H_∞ -controllers and predictive model, and with adaptive stabilization of the plasma vertical position as well. To increase the plasma controllability region in the vertical direction the additional horizontal field coils were introduced into the ITER-2 vacuum vessel and the capabilities of the system with the new circuits to control the plasma vertical position at the noise presence were investigated.

Keywords: tokamak, plasma, plasma magnetic control, ITER

INTRODUCTION

Currently, the flagship in solving the problem of controlled thermonuclear fusion is ITER – International Thermonuclear Experimental Reactor [1-3]. Tokamak reactor ITER is being built in France (Cadarache) by an international consortium consisting of the European Union, the USA, Japan, Russia, China, India, and South Korea. All the most advanced tokamaks work in support of ITER to provide a basic physical understanding of its reliable operation capabilities, including through magnetic and kinetic plasma control systems.

On the Russian side, a number of magnetic control systems for the plasma position, current, and shape have been developed for ITER, which have been modelled on plasma-physical codes DINA (TRINITI, Troitsk, Russia) and PET (D.V. Efremov Research Institute of Electrophysical Equipment, St. Petersburg, Russia). Due to the peculiarity of the ITER poloidal system associated with superconductivity of the poloidal field coils, in V.A. Trapeznikov Institute of Control Sciences of RAS (Moscow, Russia) the robust magnetic

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plasma control systems were developed. These systems are capable of working in conditions of uncertainty caused by the unmodeled dynamics of the plasma as a complex not fully studied plant. The developed systems work in the general hierarchical structural scheme of the plasma control system in ITER, taking into account the structural scheme of controlling the position, current and gaps between the plasma separatrix and the first wall of the vacuum vessel (Fig. I.1) [4].

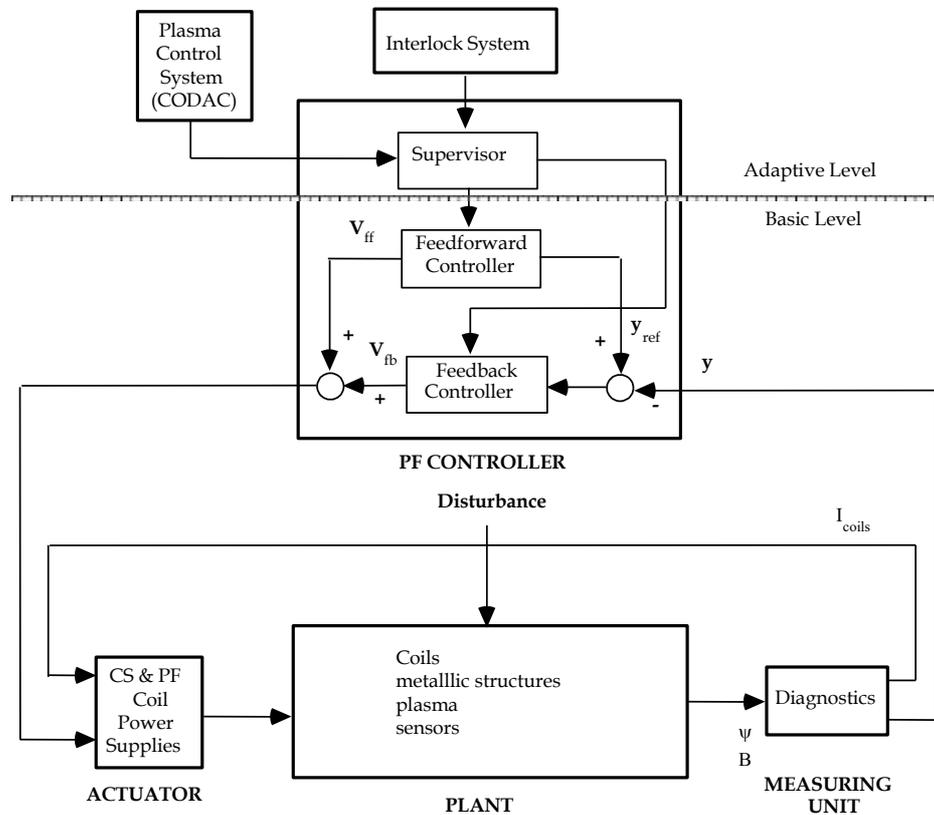


Fig. I.1. Block diagram of the hierarchical system of magnetic plasma control in ITER

In the block diagram of magnetic plasma control in ITER (Fig. I.1) the poloidal field control device (PF-controller, PF – Poloidal Fields) will presumably consist of three separate units, namely: a supervisor located at the upper (adaptive) level of control, a feedforward controller and a feedback controller of the lower (basic) level. The basic level (control loop) of the poloidal field control system includes a controlled plant consisting of a plasma, coils of a poloidal magnetic field, passive metal structures, sensors, a measuring (diagnostic) unit, an actuator device representing power sources for the central solenoid and poloidal field coils; a feedback controller and a feedforward controller. The supervisor (Fig. I.1), interacting with the plasma control system and the interlock system, controls the operation of the basic level. It generates program signals at the rate of observation and performs adaptive correction of the feedback controller about 10 times slower than the basic circuit.

At present, magnetic plasma control systems are also being developed for the first thermonuclear power plant DEMO (DEMONstration Power Plant) [5]. It is important to develop a poloidal system, which would not be forced to introduce additional plasma position control coils inside the vacuum vessel, as was done in ITER-2 which reduces the reliability of the thermonuclear reactor in stationary mode. Coils for effective control of the plasma position, providing a sufficient controllability region in vertical direction with a given limit on the supply voltage at the vertical instability of the plasma, can be installed between the vacuum vessel and the coil of the toroidal magnetic field, as is done in the ASDEX Upgrade tokamak and the T-15MD tokamak project [2]. In the Japanese variant of DEMO

[5] such magnetic coils are not provided, which requires further development and research of the DEMO poloidal system.

1. THE ITER PROJECT VARIANTS AND SCENARIOS

In the ITER project the ITER-1 version with self-sustaining thermonuclear reaction and $Q = \infty$ initially was developed but since 1998 the ITER-2 version with $Q = 10$ was begun (Fig. 1.1, Table. 1.1) [4], where $Q = P_{out}/P_{in}$ is the ratio of the output power of thermonuclear fusion P_{out} to the input power P_{in} .

In ITER, it was proposed to control the plasma boundary in the divertor configuration (lower X-point) by controlling the distances (called *gaps*) between the separatrix and the first wall at six different points, including the intersection points of the separatrix with the divertor plates. The plasma cross section, the location of the nine poloidal field coils (PF1 – PF9) in ITER-1 and six PF coils (PF1 – PF6) in ITER-2, the solid central solenoid (CS – Central Solenoid) in ITER-1 and the six-section CS in ITER-2, and the locations of the controlled gaps between the plasma boundary and the surface of the surrounding elements are shown in Fig. 1.1.

Table 1.1. The plasma parameters in two versions of ITER

Plasma Parameters	ITER-1	ITER-2
Plasma Current, MA	21	15
Major Radius, m	8.1	6.2
Minor Radius, m	2.8	2.0
Elongation	1.6	1.85
Thermonuclear fusion power, GW	1.5	0.5
Q	∞	>10
Burning time, c	1000	400
Vertical Instability Time Constant, sec	1.1	0.1

Separatrix deviations from the given location must be within ± 10 cm, which is less than 5% of the plasma major radius. The gap locations shown in the figures are chosen to provide a reasonably good overall performance of plasma shape control and at the same time for precise control at key points such as divertor strike points (g_1, g_2) and the plasma boundary in front of the ion-cyclotron radio frequency heating antenna (g_3). All poloidal field coils are superconducting with a relatively small cross-sectional area. An AC/DC converter controls each coil.

In ITER-1, all the PF coils and the central solenoid control the unstable vertical position of the plasma.

In ITER-2, a special circuit is provided to control the plasma vertical position or speed, and in the subsequent version for this purpose, additional horizontal field coils are introduced into the vacuum vessel.

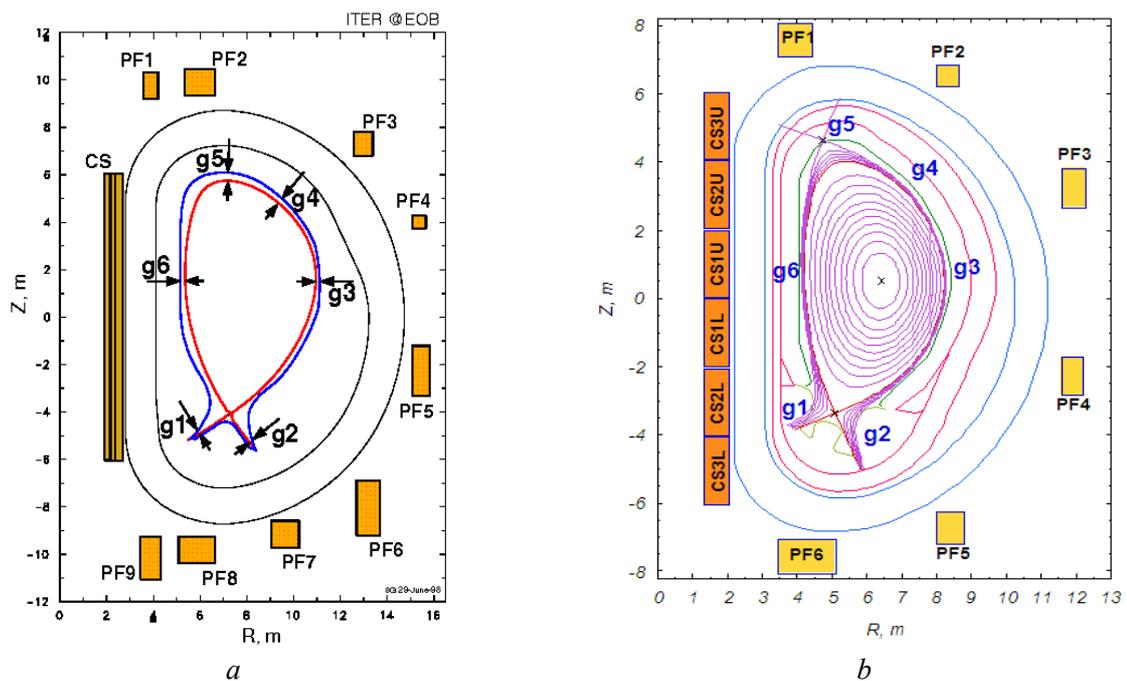


Fig. 1.1. Poloidal systems and arrangements of controlled gaps $g_1 - g_6$ between the separatrix and the first wall: ITER-1 (a) and ITER-2 (b)

The ITER-2 scenario includes the following sequence of phases [3]: plasma start and growth of the plasma current, divertor configuration (X-point) formation, heating to ignition, controlled continuous burning, controlled reduction of fusion power and plasma current, termination of the X-point, plasma shutdown (the end of a discharge). The required evolution of the plasma configuration is created by changing of the poloidal currents (feedforward components) and adding of the feedback control. Two main phases that can be identified are the most different in terms of control requirements:

- limiter phase characterized by the plasma restricted by a limiter with a relatively low thermal and magnetic energy content: the control requirements are not very strict during this phase, and it is necessary basically only to suppress the vertical instability of the plasma;
- divertor phase, during which the high amount of thermal and magnetic energy is a potential danger to the preservation of the outer layer of the plasma and its surrounding elements; this phase requires precise control of the plasma shape and current.

2. SYNTHESIS AND MODELING OF PLASMA CONTROL SYSTEMS IN ITER-1

The PF-controller (Fig. 1.1) should be able to control with given performance the gaps between the plasma and the first wall under the action of specified disturbances of the type of Minor Disruption (MD). In doing so, the controller must guarantee to stabilize the vertical unstable position of the plasma in ITER-1 or to stabilize the vertical velocity of the plasma around zero value in ITER-2.

The H_∞ robust plasma current, position and shape control system was developed in ITER-1 at $Q = \infty$ [4, 6, 7] with an NCF-controller in the feedback (NCF – Normalized Coprime Factorization). The robust system is weakly sensitive to errors and uncertainties of the tokamak plasma model. It remains stable at a large set of ITER magnetic plasma configurations in comparison to classical systems, for example linear-quadratic control. For the first time by the method of mathematical modelling on the plasma-physical code DINA it was shown that a larger robust stability margin of the multivariable control system provides a greater number of magnetic configurations from the ITER database. For these configurations the control system stabilized the position of the separatrix under the action of MD-perturbations on the divertor phase of the discharge [6, 7]. The study of the effect of perturbations of the ELM type (Edge Located Modes) showed a slight increase in the

amplitude of the currents in the poloidal field coils compared to their natural amplitude without feedback.

In addition to the linear H_∞ NCF robust controller in the feedback system, a nonlinear correction unit of the total control power (Power Management System – PMS) was developed, which allowed to reduce power peaks by 30-40% in the simulation of the system in the presence of MD-type perturbations without significant changes in the behavior of gaps [4, 6, 7]. Fig. 2.1 shows the block diagram of closed-loop control system with the linear H_∞ NCF controller in the feedback and the nonlinear correction unit.

The principle of operation of the correction unit is as follows. The total control power signal, which is the sum of the products of the currents I_n and the voltages of the control coils u_n

$$p = \sum_{n=1}^{10} I_n u_n, \quad I_n = I_{eqn} + \delta I_n$$

where I_{eqn} is the current in the poloidal field coil, which specifies the plasma equilibrium (see Fig. 2.1). The vector of corrective (additional) voltages is created as a product of the power module and voltages on the control coils in accordance with $\Delta u = -|p|u$. This vector is introduced into the feedback to reduce the input voltages at a strong increase in power, which reduces the power peaks under the action of perturbations.

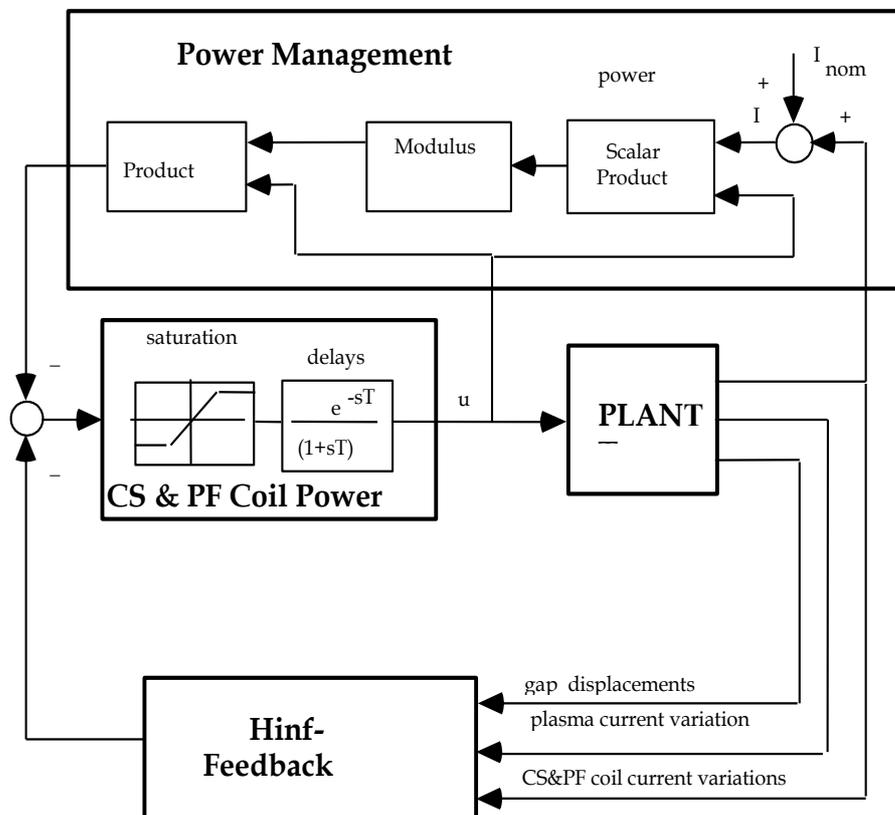


Fig. 2.1. Control system with the correction unit of the full control power

The developed controllers were modeled in closed-loop control systems on linear (CREATE-L, PET-L, TSPS-L, CORSICA-L) and non-linear (TSC, DINA, CORSICA, PET, MAXFEA, TSPS) models of the plant under the action of specified MD-type and ELM-type disturbances. Modeling has shown that systems with developed controllers met technical requirements. As an example in Fig. 2.2 for the system with the H_∞ NCF controller at the point of the SOB (Start of Burn) scenario on the PET-L linear model, graphs of transients after the MD are given for variations of gaps, voltages, poloidal coil currents, total control power, its derivative, variations of plasma current.

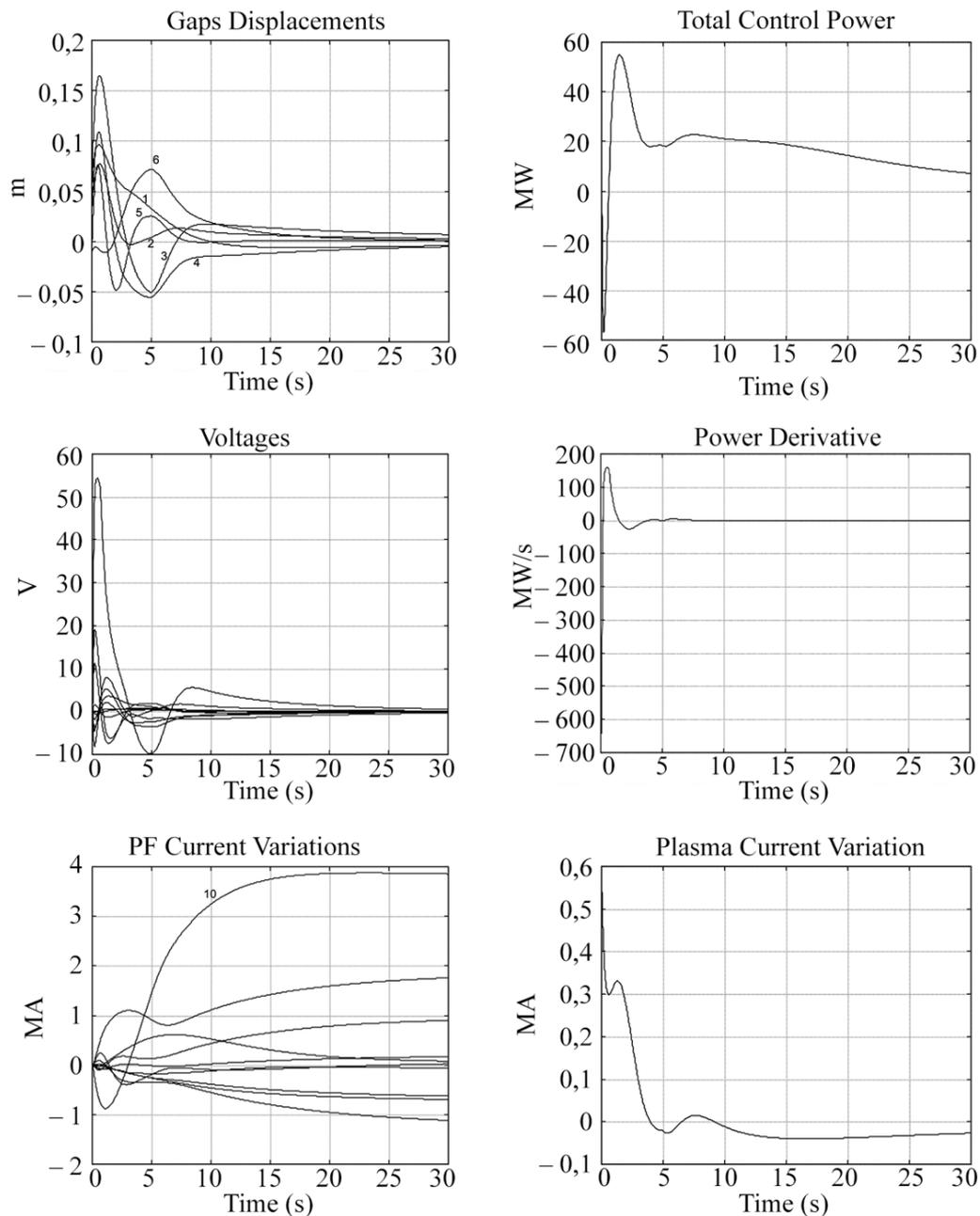


Fig. 2.2. Transient responses with H_∞ NCF controller in case of MD-type disturbance (PET linear model, SOB)

3. SYNTHESIS AND MODELING OF PLASMA CONTROL SYSTEMS IN ITER-2

The new principles and plasma control systems for ITER-2, which were developed in solving of problems of modeling magnetic plasma control systems in ITER were proposed and applied. The focus of the development of the systems is associated with an increase in their reliability, survivability and performance of control (speed and accuracy).

(1) **Synthesis of H_∞ control system based on the scheme of external disturbance rejection (Fig. 3.1–3.3)** [4, 8, 9]. The block diagram of the control system is shown in Fig. 3.1a [3]. It consists of two control loops: the first scalar loop serves to stabilize the vertical plasma velocity relative to zero, and the second MIMO loop is designed to control the plasma current and shape: the six gaps between the plasma separatrix and the first wall (Fig. 1.1, b). To stabilize the vertical plasma velocity, a special scheme is applied (see Fig. 3.1, b), consisting of parallel connection of PF2 - PF5 coils with appropriate directivity and slow

rectifiers, to which a fast rectifier VS (Vertical Stability) is connected, directly acting on vertical plasma velocity.

First, the loop shaping of the open-loop system [10] and the mixed sensitivity [11] were used to synthesize robust controllers and by means of μ -analysis the controllers were compared with each other, as well as with LQG-, Lead-Lag- and P-controllers. The loop-shaping controller showed the smallest peak of μ , that is, the greatest robust stability margin [8].

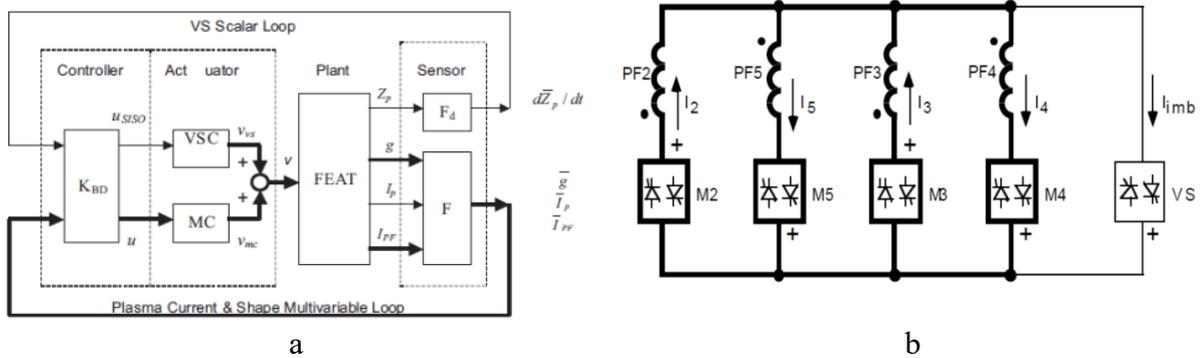


Fig. 3.1. ITER double-loop magnetic control system: a) VSC is a Vertical Stabilization Converter, MC is a Main Converter, KBD is a block-diagonal controller, F_d is a differentiating filter, F is a filter; b) connection diagram of a fast voltage converter for suppressing the vertical plasma velocity, VS is a vertical plasma stabilization thyristor rectifier, M2 – M5 are main rectifiers for PF2 – PF5 coils

Then two block-diagonal controllers were synthesized to control the speed of plasma vertical movement, current, and shape with a cascade of currents in the coils of the poloidal field (Fig. 3.2, a) and without this cascade (Fig. 3.2, b) [4, 9]. The systems were tested at various points of the ITER scenario on linear models obtained from a plasma-physical PET code under the action of the MD perturbation. A typical transient with such testing is shown in Fig. 3.3 for a controller without a cascade with PF currents, when, after a splash of gaps, they enter the specified tube, and the plasma position in the vertical and horizontal reach certain levels consistent with the plasma shape, when the vertical plasma velocity is stabilized about zero. The currents in the central solenoid and in the coils of the poloidal field come to some new level, corresponding to the compensation of disturbances. This controller showed the best robust properties when comparing it with other controllers for which the LQG approach and decoupling of control channels were used [4, 9].

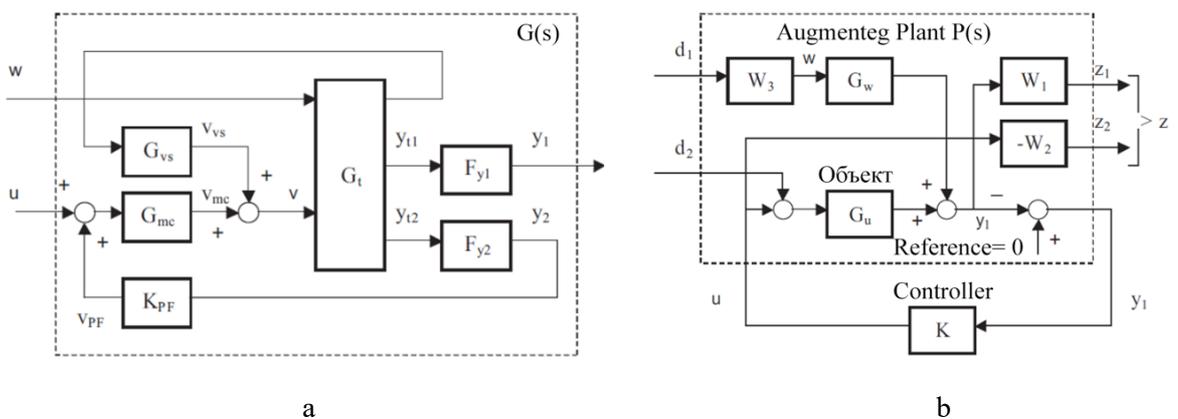


Fig. 3.2. Block diagrams of controller synthesis: a) currents in CS & PF coils, b) plasma current and shape

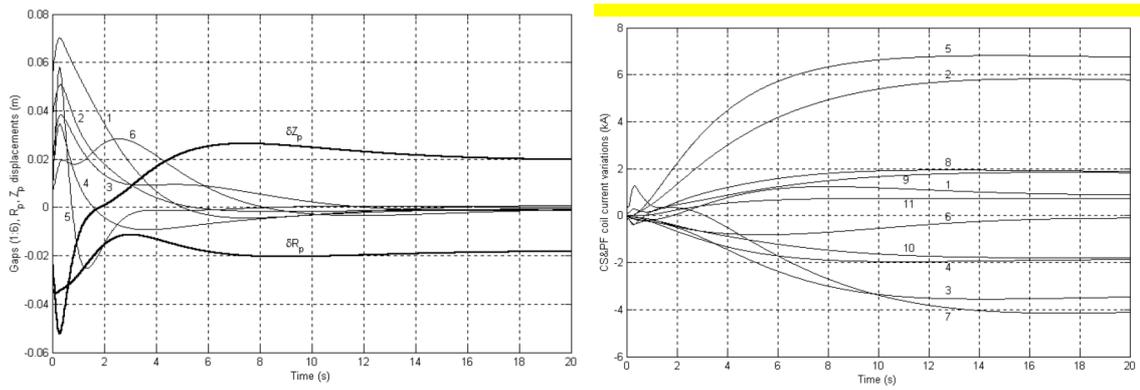


Fig. 3.3. Variations of the output signals of the control system in case of MD-type disturbance at the point of the SOF (Start Of Flattop) scenario: a - gaps, vertical and horizontal plasma positions; b - CS & PF currents

(2) Minimization of the H_∞ -norm of the mixed sensitivity function for the linearized plasma model while stabilizing the vertical plasma velocity about zero and simultaneously controlling the plasma shape and current for synthesizing robust scalar and multivariable controllers (see Fig. 3.1) [12]. At the next stage, a SISO and MIMO controllers were synthesized while minimizing the H_∞ -norm of mixed sensitivity

$$\left\| \begin{matrix} W_1(s) S(s) \\ W_2(s) K(s) S(s) \end{matrix} \right\|_{\infty} \xrightarrow{K(s)} \min, \text{ where } S(s) \text{ is the sensitivity transfer function of the closed-}$$

loop system, $K(s)$ is the controller. The block-diagonal controller was tested both on a linear model obtained by linearizing a plasma-physical DINA code that implemented partial differential plasma equations, and on the nonlinear code itself, which distinguishes this work from the previous results. The plots of the processes obtained are similar to those shown in Fig. 3.3.

(3) Cascade control (Fig. 3.4 –3.6) [13–15]. Further progress is related to the transition to cascade control and the solution of the problem of tracking the scenario values of gaps.

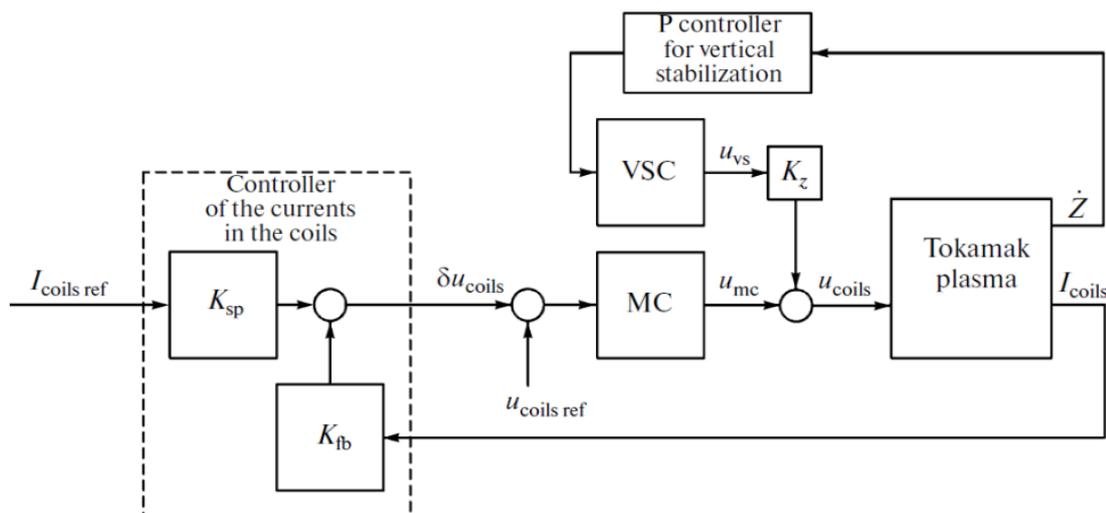


Fig. 3.4. Structural scheme of the internal cascade (the lower level) for CS/PF coil current control

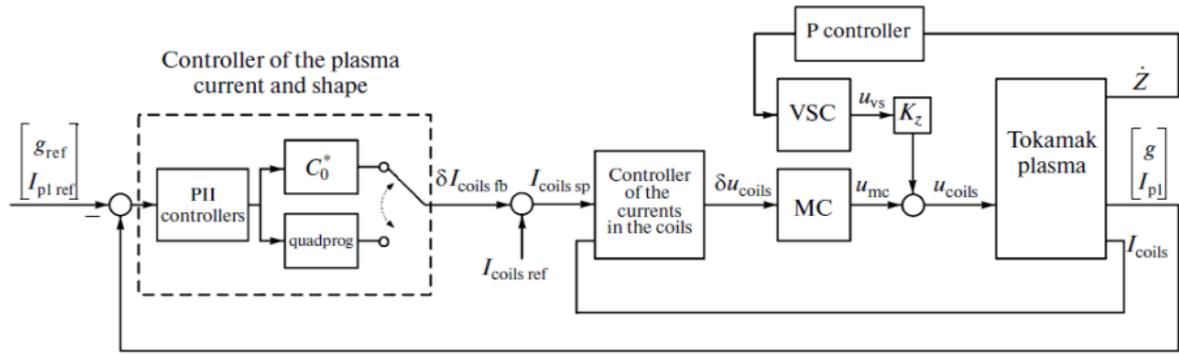


Fig. 3.5. Structural scheme of the external cascade (the upper level) for plasma shape and current control

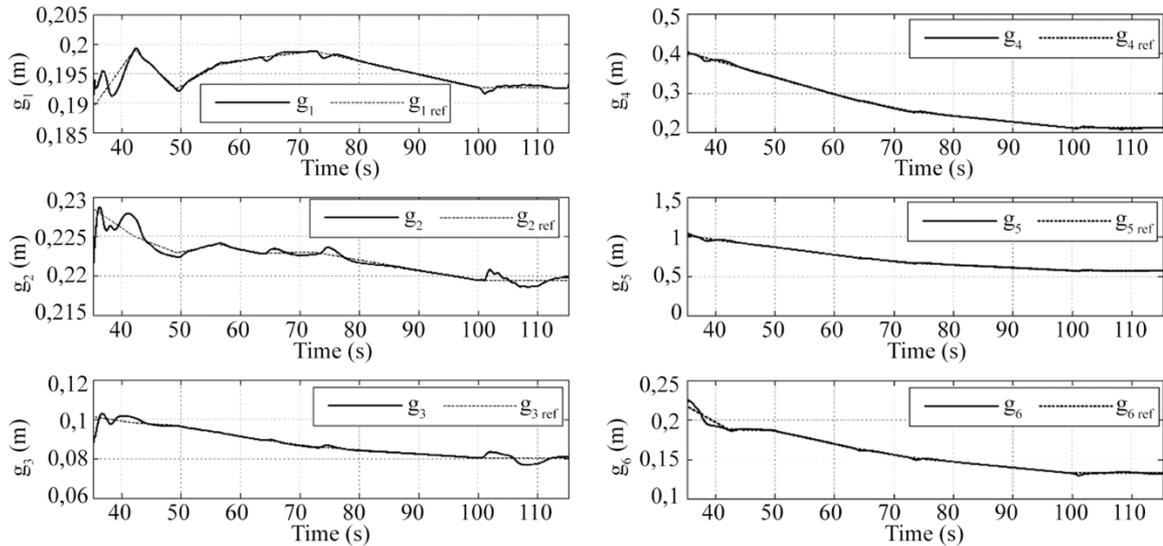


Fig. 3.6. Tracking of gaps g_1 – g_6

Here, in the inner loop, the original decoupling of control currents is carried out (see Fig. 3.4). In the outer loop, the Moore-Penrose pseudo-inverse matrix establishes a connection between the displacements of the gaps between the first wall and the separatrix and the variation of the plasma current with control currents through the diagonal PII controller. At the same time, in order to avoid saturation of control currents, a nonlinear quadratic programming problem was solved (increasing system survivability) (see Fig. 3.5). Fig. 3.6 shows the operation of the cascade control system for the position, current and shape of the plasma with a decoupling of channels in the tracking mode for gaps on the DINA code with a sufficiently high accuracy.

(4) Model Predictive Control (MPC). This control is attractive because the synthesis of the controller takes into account the restrictions on the input and output signals, which eliminates the development of specialized approaches, in particular, to prevent the saturation of plasma control currents. Therefore, the proposed approach has no separation on two control loops of the plasma vertical velocity and plasma current and shape (Fig. 3.7) [16]. This configuration takes into account limitations on the control currents. Numerical modeling of the system with a predictive model on the non-linear DINA code showed that, due to the natural consideration of constraints and the solution of the optimal problem at each control step, the system with a predictive model gives smaller deviations in the gaps when a MD disturbance occurs [16].

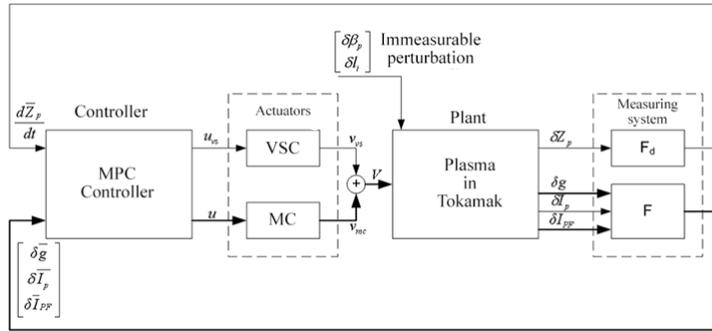


Fig. 3.7. Block diagram of a magnetic plasma control system in ITER with a predictive model

The predictive model approach for plasma control in tokamaks continues to evolve. Thus, a scalar system for stabilizing the vertical position of the plasma in ITER is developed and simulated relative to zero with prediction [17]. The method with a predictive model involving the singular decomposition of matrices [18] is used to control the current and shape of the plasma in ITER. Prediction is used to control the profile of safety factor q in ITER with variable constraints on the RAPTOR code [19], as well as to control the vertical position of the plasma on the T-15MD tokamak model with a variable parameter [20].

(5) A cascade control system with decoupling of channels in the inner cascade for controlling currents in the poloidal field coils and robust controller in the outer cascade (Fig. 3.8) [21]. Cascade control of plasma shape and current leads to a trend towards increasing margins of robust system stability, which is associated with an increase in system reliability. To achieve this goal, the outer cascade was synthesized as a robust when solving the minimization problem of the H_∞ -norm of mixed sensitivity. Then, on a linear model, the synthesized system was compared with the system of issue (3) according to the criteria of margins of robust stability and robust performance. The result showed that the robust system surpasses the system from issue (3) by about five times according to robust criteria. Simulation of the robust system on the DINA code in the tracking mode for gaps showed the result on tracking accuracy close to the system of issue (3) (see Fig. 3.6).

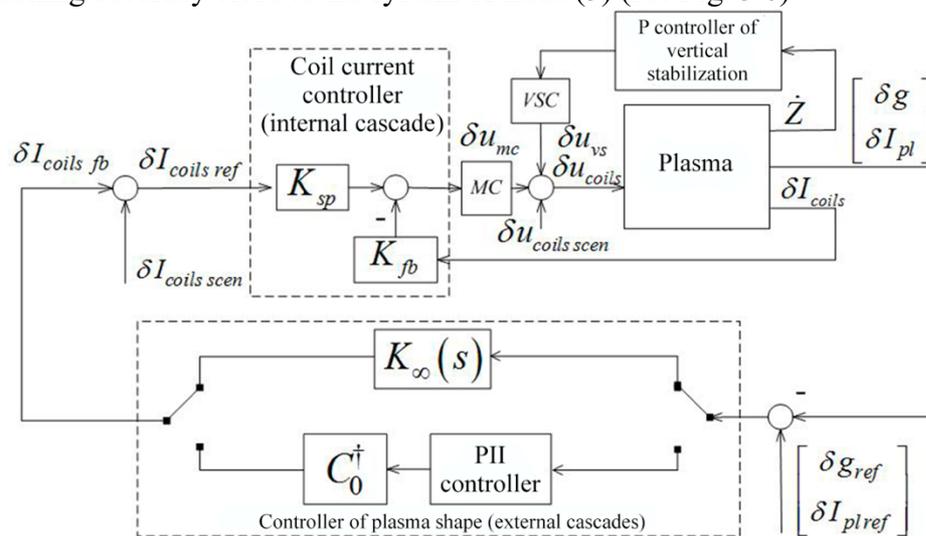


Fig. 3.8. Block diagram of cascade control system of plasma shape and current in tokamak: the keys provide the ability to switch controllers in the outer cascade

(6) Hierarchical plasma control with MIMO robust control loop for plasma current and shape, synthesized by MacFarley-Glover method, and adaptive control circuit for the vertical position of the plasma with predictive model (Fig. 3.9) [22, 23]. In the

previous approaches, the speed of the plasma vertical position was stabilized around zero, and not the position itself. This approach has two drawbacks. One of them is connected with the fact that this approach does not allow the plasma to maintain its position in a limited region of vertically controllability, since the limitation of this region arises due to the instability of the plasma and the restriction on the control voltage. Another disadvantage is associated with the fact that with such control the system is not strictly stable, since the speed and not the position of the plasma in the vertical direction is stabilized. For these reasons, a system was developed to control the position, current, and shape of the plasma, in which the vertical unstable position of the plasma is stabilized. To avoid contradiction between the plasma shape and its position, an adaptive contour with a controller based on a predictive model was added, which adapts the position of the plasma to its shape (see Fig. 3.9).

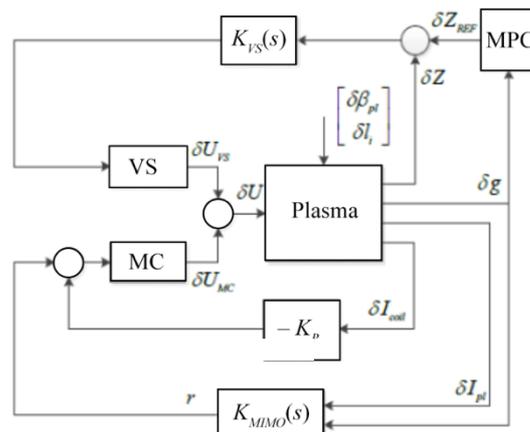


Рис. 3.9. Block diagram of hierarchical system of magnetic plasma control with adaptation in ITER: MPC - controller with a predictive model

A brief comparison of the applied approaches. ITER used various principles and structures of control systems for magnetic control of plasma (1) - (6). This is because the plasma in ITER is a complex plant with distributed and time-varying parameters, having parametric and structural uncertainties, which makes it necessary to search for the most effective methods of multivariable hierarchical plasma control.

First, to counter uncertainties of plasma models, robust control systems based on the H_∞ -optimization theory (1), (2), (5) and (6) were developed. Various system structures were proposed: (1) reflection of external disturbances; (2) minimization of the H_∞ -norm of the mixed sensitivity function; (5) decoupling of current control channels and H_∞ -controller for the shape of the plasma. All these schemes were used with the stabilization of the vertical plasma speed relative to zero, in order to eliminate contradiction between the position of the plasma and its shape. These schemes led to approximately the same performance at control of current and plasma shape, providing certain robust stability margins due to H_∞ -controllers. Systems with control of the vertical speed of the plasma, rather than its vertical position, are flawed: they are not strictly stable, and the vertical position can change uncontrollably in transients. For this reason, it was proposed to directly control the vertical position of the plasma in order to overcome this drawback in the vertical stabilization circuit (6). Therefore, an adaptation of the vertical position of the plasma to its shape (6) was developed.

At the same time, the systems with decoupling of control channels both in the inner cascade of currents control in the coils of the poloidal field and in the outer cascade of plasma current and shape (3) were developed. This is due to the fact that when decoupling the channels, it is easier to tune each control channel, while in comparison with the H_∞ -theory of optimization a multivariable controller is synthesized while minimizing one H_∞ -criterion, which makes it difficult to understand how the whole system works, although it gives large robust stability margins.

A current, position, and shape control system with a predictive model has been applied (4). This approach is promising in that with it it is possible to take into account the restrictions on the input and output values of the controlled plant, since these restrictions are laid into the algorithm with prediction at the input and output horizons. In addition, a method

with a predictive model can be applied to create a multivariable adaptive plasma current and shape control system, if there is an opportunity to identify a linear plasma model at each time instant.

4. SYNTHESIS AND MODELING OF PLASMA CONTROL SYSTEMS IN ITER-2 WITH INTERNAL COIL FOR CONTROLLING THE VERTICAL PLASMA POSITION

It has been established in [24] that the plasma controllability region on the vertical in ITER is limited to a relatively small value namely 4 cm, which is of 2% of the minor ITER radius. The idea of controllability region estimation is presented in Fig. 4.1. It consists in the fact that under the same conditions and the same saturation voltage supplied to the coils of the horizontal ITER field according to the scheme shown in Fig. 4.1, b, the plasma was released from different initial conditions vertically, and the voltage sign was chosen such that it counteracted the increase in vertical displacement. If the initial condition was within the controllability region, then the voltage after a certain time interval caused the vertical displacement to change in the opposite direction. This study was conducted, in particular, on the CORSICA code (USA).

Four proposals were put forward and investigated [25] on how to increase the region of vertical plasma controllability:

- increase in voltage VS of the power supply from 6 to 9 kV;
- insertion of a second VS-circuit using two CS sections: CS2L and CS2U;
- adding stabilizing rings inside the vacuum vessel;
- insertion of special control coils inside the vacuum vessel.

These proposals were studied and compared with each other in [25]. The most effective means turned out to be coils inside the vessel. For this reason, at the end of 2013, special coils were installed inside the vacuum vessel in the ITER project in order to bring them closer to the plasma as much as possible, thereby expanding the controllability region vertically and increasing the reliability of the plasma magnetic control system (see Fig. 4.1, b) [24].

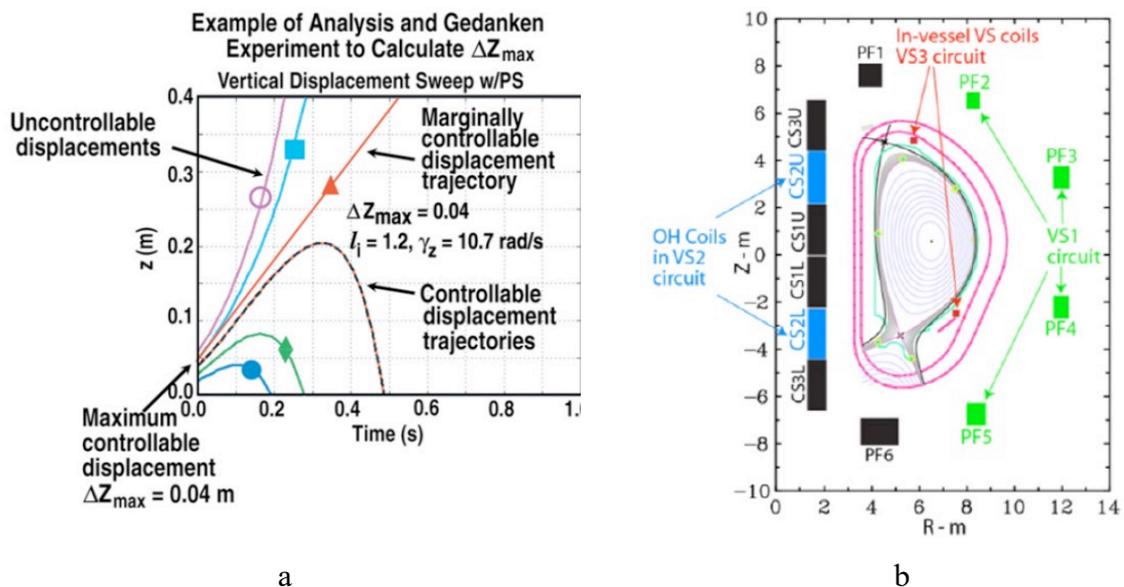


Fig. 4.1. Investigation of the vertical controllability region in ITER-2 and its expansion: a) experimental method of estimation of the controllability region ΔZ_{max} [24]; b) coils of the horizontal field which are located inside the vacuum vessel to increase the controllability region; the power supply VS3 is connected to these coils that is controlled by the controller in the feedback [25].

Two systems for stabilizing the plasma vertical speed with respect to zero with external and internal coils were investigated [26]. The authors used the JIN-TRAC (JET) transport

code and the equilibrium CREATE-NL code with the free boundary, which were connected together for modeling. In the first case, the control law was chosen as follows

$$VS_1(s) = -15000 \frac{1+s/18}{(1+s/60)^2} Z_{dot}(s)$$

where $VS_1(s)$ is the Laplace transform of the voltage function of the external coil, $Z_{dot}(s)$ is the Laplace transform of the function of the vertical speed movement of the plasma current center, and in the second case the control law in the Laplace transform domain was chosen as follows:

$$VS_3(s) = -8000 \frac{1+s/40}{1+s/6} \left[Z_{dot}(s) - 1,2 \times 10^{-5} \times I_{S3}(s) \right]$$

where $VS_3(s)$ is the Laplace transform of the voltage function on the inner coil, $I_{S3}(s)$ is the Laplace transform of the current function in the inner coil. The case of scenario 1 was chosen, L-mode, SOF (Start of Flattop: the beginning of the flat phase), $I_p = 15$ MA, $l_i = 0,88$, $\beta_{pol} = 0,06$. The simulation results are summarized in Table 4.1 where VDE_{max} (Vertical Displacement Event: the phenomenon of vertical displacement) is the maximum possible initial position of the plasma in the vertical direction, which can be stabilized (controllability region), m_ϕ is the phase stability margin, ω_t is the cut-off frequency, m_{GU} and m_{GL} are the upper and lower limits of the stability margin in amplitude. The results showed that the area of control VDE_{max} significantly more for the case of an inner coil, about 5-6 times.

Table 4.2 shows the values of VDE_{max} in the presence of noise with a limited bandwidth of 1 kHz, zero mean, standard deviation of $\sigma = 210$ mm/s and $\sigma = 430$ mm/s. The machine configuration is rather insensitive to noise: at the noise level in question, there is a negligible degradation in performance in terms of VDE_{max} .

Table 4.1. Characteristics of the control system obtained in the analysis

	VDE_{max} , mm	m_ϕ	ω_t rad/s	m_{GU} , dB	m_{GL} , dB
Outer coil	34	22°	13	5.6	-4.0
Inner coil	> 200	65°	78	13	-18

Table 4.2. The best performance in the presence of noise obtained in the analysis

Configuration	VDE_{max} , mm	
	210	430
VS_1 (6 kV)	28	24
VS_3 (2,3 kV)	148	141

In ITER scenarios are continued to be developed and refined using plasma magnetic control systems on plasma-physical codes, taking into account only external coils for controlling the vertical position of the plasma [27], and taking into account the internal coil [26, 28].

Fig. 4.2 shows the block diagram of the plasma shape control system, which was used in [26] to simulate transients during the transition from mode L (15 MA) to mode H (15 MA) and vice versa on the linear model CREATE-L. The paper [26] presents the evolution of the internal g_6 and external g_3 gaps (see Fig. 1.1, b) for the cases: the direct link is updated at $T = 1$ s; direct link is updated at $T = 5$ s; the direct link is updated at $T = 1$ s exactly after the transition and then at $T = 9$ s; the direct link is updated at $T = 1$ s and a transport delay of 0.2 s; direct link is calculated as a function of β_{pol} .

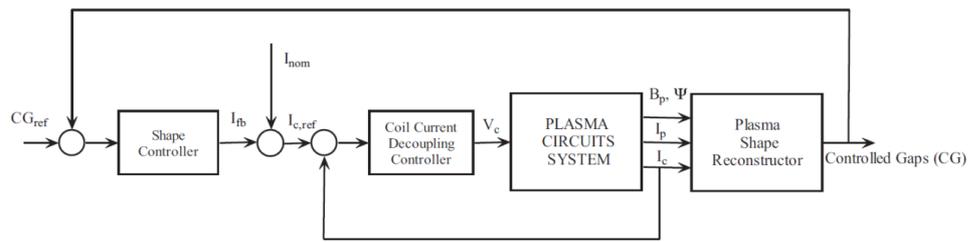


Fig. 4.2. Block diagram of the plasma shape control system, similar to the eXtreme Shape Controller scheme in the JET tokamak (UK)

Studies of the plasma vertical controllability region for ITER according to the method in [24] were continued in [27] on the PET code in more detail for various plasma variants. The results are summarized in Table. 4.3, which shows that when using only external coils and VS1 power supplies, the range of the controllability region is in the range of 1.5–13.7 cm. If both external VS1 coils and internal VS3 coils are involved, then the range increases noticeably up to 16–25.5 cm.

Table 4.3. Performance indicator max (Z_0) and instability time constant [26]

Plasma 15 MA		Plasma 1 $l_i(3) = 1.2, \beta_p = 0.1$	Plasma 2 $l_i(3) = 1.0, \beta_p = 0.1$	Plasma 3 $l_i(3) = 0.73, \beta_p = 0,6$
max Z_0 ,	VS1	15	30	137
	VS3+VS1	160	175	255
Instability time constant, ms		56	78	171

The systems of magnetic plasma control have been investigated taking into account noise and various combinations of coils for stabilizing the vertical position of the plasma in ITER on the DINA code [27]: external coils VS1; VS3 internal coils; simultaneous use of external and internal coils. The studies were based on limited data analysis from C-Mod, JET, and ASDEX-U tokamaks, which made it possible to specify for plasma activity in ITER a model of vertical plasma speed noise dZ/dt as a random signal evenly distributed over the frequency interval [0, 1] kHz with a standard deviation of 0.6 m/s. The simulation was performed with low-frequency noise injection into the diagnostic signal dZ/dt , which was used in the feedback of the vertical plasma stabilization. At the same time, the standard deviation of the uniformly distributed noise was increased until the vertical stabilization system lost stability or one of the engineering parameters reached its design limit.

The best result was given by the combination of the windings of the internal and external coils. In this case, two signals were applied to the input of the vertical stabilization controller: dZ/dt and I_{VS3} . The maximum allowed standard deviation for noise in this case was 3 m/s, the standard deviation of the I_{VS3} current was about 10 kA, the standard deviation of the vertical plasma displacement during the flat phase of the plasma current was about 44 mm. The plasma control system was synthesized using the standard LQG method [29].

CONCLUSION

In this part of the review, attention is paid to the development and modeling of plasma position, current, and shape control systems in ITER, and the contribution of V.A. Trapeznikov Institute of Control Sciences of Russian Academy of Sciences to this work is highlighted. The results show a trend in the development of these systems, associated with an increase in accuracy and speed when tracking scenario signals and rejecting disturbances such as minor disruption. The trend reflects increase in the robust stability margin, which in turn leads to increase in reliability and survivability of plasma control systems in ITER.

The introduction of horizontal field coils inside the ITER vacuum vessel expands the controllability region in the vertical direction of plasma movement, which, at the given the control coil voltage limits, specifications for minor disruption disturbances and vertical plasma instability, significantly moves away the closed-loop system of magnetic plasma control from the boundaries of stability loss.

Magnetic plasma control systems in operating tokamaks, as well as for ITER, continue to evolve in different ways. One of these areas is related to the development of high-precision methods for solving inverse plasma diagnostic problems, the data of which serve as input data for plasma control systems for internal plasma parameters, and approbation of control systems based on these data, rather than direct data, when all plasma parameters are known accurately enough [30, 31].

In Section 43.2, experimental processing of scenarios for ITER on DIII-D (USA) and WEST (France) tokamaks, approaches in modeling and implementing plasma control systems in ITER, preparation of the plasma control system for launch and operation in ITER will be presented. Road maps of the development and creation of the first DEMO fusion power station (following the ITER step) will be shown, which indicate two DEMO development directions: on traditional tokamaks with relatively large aspect ratios and modular-type spherical tokamaks that will significantly reduce the creation time of DEMO. It will be also shown the main trends in the development of DEMO poloidal systems, as well as the initial version of the magnetic plasma control system in DEMO.

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