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Plasma Control in Tokamaks. Part. 2. Plasma Magnetic Control Systems

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Abstract: Various systems of magnetic control of plasma position, current, and shape are considered in vertically elongated tokamaks including spherical tokamaks being in operation. The systems of Resistive Wall Modes suppression are described. Tokamaks constructions and cross-sections, structural schemes of plasma control systems are given; heed is paid to operating principles of plasma control systems; experimental results of plasma control in tokamaks are given. Various engineering realizations of plasma magnetic control systems in tokamaks are presented.

Keywords: tokamak, plasma, plasma magnetic control, plasma position control, plasma current and shape control, resistive wall modes suppression, plasma control systems realization.

INTRODUCTION

The first part [1] of the survey was devoted to the general problem of controllable thermonuclear fusion. It covers the key features of tokamaks and components of plasma control systems, describes the constructions of tokamaks. In particular, the diagnostics system of the Globus-M tokamak is considered. The experimental data obtained from this tokamak were used in M.V. Lomonosov Moscow State University and V.A. Trapeznikov Institute of Control Sciences to develop original plasma position, current, and shape control systems.

The second part of the survey considers plasma magnetic control systems (plasma is a completely ionized gas) in tokamaks, which are distributed dynamic plants with a complex structure and uncertainties subjected to uncontrollable disturbances. Plasma in a magnetic field is not in thermodynamic equilibrium, and, as a consequence, is liable to various instabilities, which are the main cause of a relatively slow approaching of plasma parameters to the Lawson criterion.

The scientific direction related to research, development, and improvement of tokamaks has been significantly advanced in former Soviet Union under the leadership of academician L. A. Artsimovich [2] and then has spread across the world [3, 4]. The first tokamaks had the circular cross section and were meant to wide-ranging high temperature plasma physics research. A general trend toward enlargement of such tokamaks was observed. These tokamaks included devices which were located in the I.V. Kurchatov Institute of Atomic Energy (Moscow, Russia): T-3, T-4, T-7, TO-1, T-10 и T-15; in the A.F. Ioffe Institute (St. Petersburg, Russia): TUMAN-3 (toroidal device with plasma magnetic adiabatic heating); and also

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foreign devices: PLT, TFTR (*Test Fusion Tokamak Reactor*) (USA), TORE-SUPRA (France), TEXTOR-94 (Germany), FT-U (Italy), etc.

A specific feature of the next generation of tokamaks consists in the vertical elongation of the cross section. This feature enables to increase the plasma pressure and to raise the heating by the own plasma current [5]. A "price to pay" for these advantages is vertical instability of the plasma caused by the magnetic fields required for elongation. Nevertheless, the vertically elongated tokamaks are now the main experimental base for researches on the problem of controlled thermonuclear fusion. These are JET (Joint European Torus), United Kingdom, JT-60U, Japan, ASDEX UPGRADE, Germany (Max Planck Institute for Plasma Physics), DIII-D, C-MOD, USA, TCV, Switzerland, COMPASS (COMPact ASSembly), Czech Republic; EAST (Experimental Advanced Superconducting Tokamak), China, KSTAR (Korean Superconducting Tokamak Reactor), South Korea.

The spherical tokamaks with a small aspect ratio have also appeared: MAST (Mega-Amp Spherical Tokamak), United Kingdom; NSTX (National Spherical Torus Experiment), USA; Globus-M/M2, Ioffe Institute, Russia. These tokamaks allow to even further increase in the gas kinetic plasma pressure at a specified magnetic field and can lead to an additional reduction in cost of a reactor as a result.

Plasma as an automatically controlled plant has the following typical characteristics, which create difficulties of both fundamental and technical nature:

- plasma is a distributed system with an infinite number of degrees of freedom;

- imperfection of the theoretical models and the fact that processes proceeding in a plasma have not been sufficiently explored lead to considerable uncertainties in the structure and parameters of plasma models;

- plasma is a non-stationary plant which means that plasma parameters may change by several orders of magnitude for a short period of time when plasma is produced and heated at one operational cycle or experiment;

- plasma can be non-minimal-phase plant, since the transfer functions on certain control channels, under the assumption that the parameters are replaced by the fixed values, may have both poles and zeros with a positive real part;

- plasma is also subjected to uncontrollable disturbances, which in some cases can be estimated in re al time by observing the inputs and outputs of the plant;

- plasma is a source of broadband noises which are not sufficiently explored. This leads to difficulties in plasma parameters identification;

- plasma is a nonlinear dynamic plant by its nature;

- large values of natural frequencies of plasma oscillations require high speed of response and significant power of control systems;

- actuators, which are generating the control signals sent to the plasma, may contain energy converters with nonlinear characteristics (often discontinuous), dead-zones, and transport delays. This makes it significantly difficult to design and analyze closed loop plasma control systems.

- the complexity of dynamics and nonlinearity of the actuators serve as an additional source of uncertainties when building a controlled plant (plasma in a tokamak) model;

- diagnostic units in thermonuclear devices, in many cases, have uncertainties when identifying plasma, which also provide contribution to the total uncertainty of plasma models.

Despite the presence of peculiarities of plasma, which characterize it as one of the most complex controlled plants in nature, the automatic feedback control systems started being used in the 1960s for plasma confinement in the magnetic traps, and then started to play the significant role in controlled thermonuclear fusion. The investigations along this direction were begun in 1967-1968 by Doctors in physics and mathematics V.V. Arsenin and V.A. Chuyanov in experiments on the trap with magnetic mirrors Ogra-2 in the I.V. Kurchatov Institute of Atomic Energy. In the experimental device Ogra-2 the flute and ion-cyclotron (kinetic) plasma instabilities have been suppressed and then the control systems have obtained a spread to suppress other instabilities: drift, ionization, helical instabilities in tokamaks, to stabilize θ-pinches etc. In [6] a detailed overview of the results of researches and experiments on suppressing plasma instabilities is presented.

Initially, the main control problem in tokamaks was to stabilize the position of a plasma ring along the major radius by poloidal (lying in a meridional plane) magnetic field. The first experiments aimed to solve this task were carried out on the TO-1 tokamak using an impedance controller in 1971 jointly by the staff of the I.V. Kurchatov Institute of Atomic Energy (L.N. Artemenkov, I.N. Golovin, et al.) and the V.M. Glushkov Institute of Cybernetics of Academy of Sciences of the Ukraine Soviet Socialist Republic (Yu.I. Samoilenko, V.F. Gubarev et al.) [7]. Combined plasma equilibrium control is used in modern tokamaks, namely: pre-programmed reference control provides a scenario and correction of the plasma location is performed by a feedback control system. These systems became standard components of tokamaks and found application for the joint stabilization of orthogonally decoupled stable horizontal and unstable vertical plasma position in tokamaks. Then this approach has started to be used for plasma shape control by means of poloidal field coils and in fact the controlled plant started to belong to a class of multivariable systems, these include the plasma in ITER [8, 9].

The use of automatic control methods to ensure the stability and equilibrium of plasmas in thermonuclear devises with magnetic confinement became a generally recognized necessity.

Experiments on tokamaks have shown that the main parameters of plasma, which directly provide creation of the conditions for initiation of thermonuclear reaction, are solely sensitive even to slight displacements of the external magnetic surface of plasma column in relation to the vacuum vessel or diaphragm limiting the column. Therefore, the accuracy of the plasma equilibrium control enables to reduce the rate of harmful impurities inflow into the plasma as well as loss of particles that makes it possible to increase attainable values of plasma density, temperature, and energetic lifetime [2, 7-9]. The position of the plasma boundary is stabilized as close as possible to the first wall in order to ensure the efficient use of the internal space of the vacuum vessel and also to reduce the increments of unstable displacements of the vertically elongated plasma. The smallest failures in the plasma control system can cause the melting of the vacuum vessel. Uncontrollable contacts of the plasma with the vacuum vessel lead to the plasma powerful energy release outward, and also result in excessive mechanical loads and damage of the thermonuclear device. These failures are totally unacceptable for the thermonuclear reactor.

Apart from the plasma magnetic control systems [8, 9], plasma kinetic control systems are also being developed [10, 11] that enables to control the profiles of plasma parameters: plasma current, safety factor, temperature, density, and pressure, and also the burning power during thermonuclear reaction. These systems are necessary for obtaining the most profitable (optimal) operation modes of future thermonuclear reactors.

The research work aimed at developing and implementing of plasma control systems in the Soviet Union was commenced in the laboratory of Professor, Doctor of Engineering Sciences Lev N. Fitsner in V.A. Trapeznikov Institute of Control Sciences since 1973. The introduction of developed plasma control systems to physical experiments and their numerical investigation on the distributed parameter models of the plasma were done in cooperation with collaborators of the I.V. Kurchatov Institute of Atomic Energy (Moscow), Troitsk Institute for Innovation & Fusion Research (TRINITI) (Troitsk, Moscow Region), A.F. Ioffe Physical-Technical Institute (St. Petersburg), the D.V. Efremov Institute of Electro-Physical Apparatus (St. Petersburg).

General requirements for plasma control systems in tokamaks and general design methodologies of new plasma control systems have not been developed so far. Each scientific group who works on operating tokamak designs plasma control systems depending on device functionalities and the type of tasks to be solved. Since tokamaks have different configurations of poloidal systems and energy resources of power supplies, different plasma control systems are obtained. Presently, at the International Conference on Decision and Control (CDC) the sections devoted to plasma control systems have been organized on the initiative of American specialists. There are also the papers on that subject presented at the IFAC World Congress in 2014 (references are in the survey). The papers have been published by the V.A. Trapeznikov Institute of Control Sciences of the Russian Academy of Sciences at the CDCs [12–18] and IFAC Congresses [19–21].

1.CONTROL SYSTEMS OF PLASMA POSITION

In all operating tokamaks there are control systems of a plasma position, namely, stabilization systems of the plasma magnetic axis or the center of the plasma current in various implementations. Several Russian and Foreign developments will be showed in greater depth.

1.1. The T-14, Tuman-3 and TVD tokamaks (Russia)

In the early national projects a number of plasma position control systems were developed: T-14 (tokamak with a strong field, TRINITI, Troitsk), Tuman-3 (toroidal installation with magnetic adiabatic heating, Ioffe Institute, St. Petersburg), TVD (elongated tokamak with a divertor, Kurchatov Institute, Moscow).

Control systems were developed for the T-14 and Tuman-3 devices, modeled and implemented in practice of physical experiments on Tuman-3: control system that evaluates and compensates the external disturbances during stabilization of a major radius of the plasma column [22-25], as well as an adaptive self-oscillating system for stabilization of the horizontal plasma position, that minimizes the amplitude of self-oscillations at each quasi-period under variable parameters of the controlled plant [26-30]. The additive perturbation and two variable parameters of the model were estimated on line by the adaptive Kalman filter [31]. A two-loop orthogonally decoupled self-oscillating system for stabilization of the horizontal and vertical plasma positions with thyristor voltage inverters as actuators was developed and applied in experiments [32-35].

1.2. The Globus-M tokamak (Russia)

The block diagram of the stabilization system for the plasma vertical position of the Globus-M tokamak is shown in Fig. 1 [36]. The control system includes an actuator based on a current inverter [37], loaded on the control coil of the horizontal field. The control coil generates a radial magnetic field that is proportional to the current passing through it, and has an impact on the plasma column (controlled plant) and shifts it in the vertical direction *Z*. The feedback loop of the system is closed through a controller that implements a proportionally-differentiate control algorithm (see Fig. 1 for designations) $U_{Contr} = -\alpha(\varepsilon + T_d\dot{\varepsilon})$, where $\varepsilon(t)=Z(t)-Z_{ref}(t)$ is the error, $Z_{ref}(t)$ is the vertical displacement of the plasma. The error ε is formed as $\varepsilon = ZI_p - Z_{ref}I_p = K\Psi_R(t) - Z_{ref}I_p$, where $\Psi_R(t)$ is the radial flux, I_p is the plasma current, K is the proportional gain.

The results of the work of this version of the stabilization system of the plasma vertical position are shown in Fig. 2 (the case of plasma stabilization in the equatorial plane of the tokamak at $Zref = 0$). It can be seen from the oscillogram $\Psi_R(t)$ that plasma vertical stabilization system holds the plasma in the equatorial plane ($\Psi_R=0$) with good accuracy until the end of the discharge pulse. The calculated elongation *k* of the plasma, that is determined by the EFIT code, at the end of the discharge pulse equals to two: k = 2.

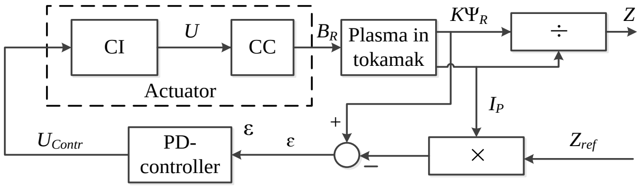


Fig. 1. The block diagram of the stabilization system for the plasma vertical position of the Globus-M tokamak**:** CC – control coil; CI – current inverter; Ψ_R – radial flux [36]; B_R – induction of a radial magnetic field

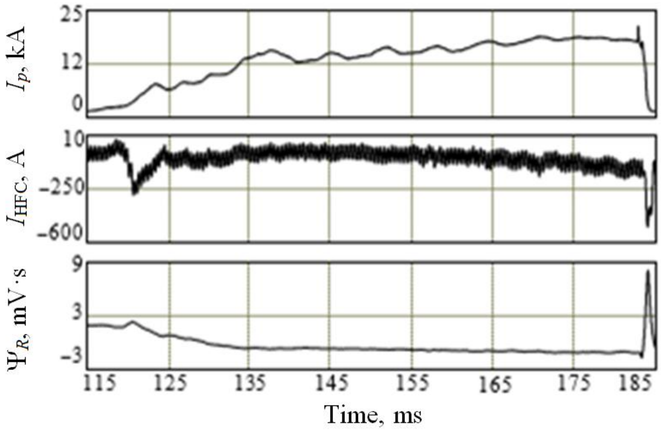


Fig. 2. Oscillograms $I_p(t)$, $I(t)$, $\Psi_R(t)$: **discharge pulse № 10446 [36]**

1.3. The JET tokamak (United Kindom)

The JET (Joint European Torus) tokamak [38] is one of the largest operating machines in the world. Its magnetic configuration is close to the architecture of the ITER tokamak project. In the JET tokamak magnetic control system there is a plasma adaptive vertical stabilization system, which stabilizes the vertical plasma velocity about zero. It consists of three main subsystems: the vertical stabilization controller, the controller of the actuator current and the adaptive controller. Four poloidal coils connected to a fast FRFA amplifier, that are capable of switching between nine output voltage modes during time of the order of 200 μs according to the non-linear law with hysteresis zones, are used as an actuator. The measuring system (Speed Observer) calculates the plasma vertical speed on the basis of the Ampere's law and the approximation of the vertical moment of the total current as a weighted sum of measurements of magnetic field outside the plasma [38–40].

The controller of vertical stabilization is a proportional unit. However, the behavior of the system closed by such controller will be similar to the case of a relay controller due to the presence of dead zones and hysteresis in the model of the actuator: as soon as the plasma velocity becomes greater than the threshold value, a voltage appears at the output of the actuator, a current creating a force acting on the plasma in the opposite direction increases rapidly in the coil. When the plasma velocity passes through zero, the voltage at the output of the actuator disappears. Further, when the opposite threshold value is reached, the reverse voltage appears. Thus, an oscillating process arises.

The use of a single closed stabilization loop of the plasma vertical velocity around zero cannot guarantee the stability of the plasma in the vertical direction [40]. The process will oscillate around zero, however, a drift of the average values of the plasma vertical position and current in the control coils is possible, since the control is carried out by speed, and not directly by position. The problem of vertical stabilization is solved jointly with the plasma shape control system, which gives a guaranteed stable closed control system.

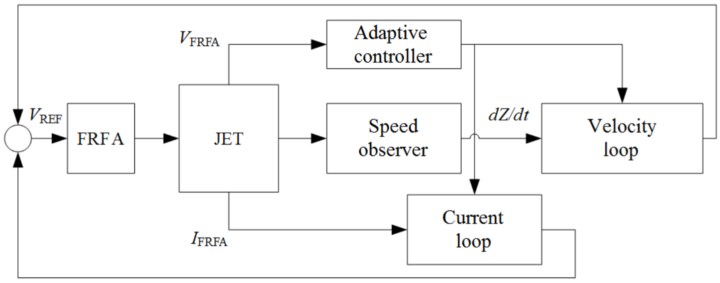


Fig. 3. Stabilization system of the vertical plasma velocity around zero on the JET tokamak: FRFA is Fast Radial Field Amplifier [38]

It is also necessary to avoid exceeding the current limit value in the control coil and the output of the FRFA amplifier in a saturation mode. A slow PI controller, which stabilizes the current in the control coil around zero, is used to solve this problem. Due to the non-minimal phase nature of the vertical plasma motion model [40] the gain in this loop is negative. The risk of overheating of the FRFA amplifier and the release of heat on it depend on the switching frequency, these determine the limit of the permissible gain in the stabilization loop of speed around zero. This limitation is further enhanced by the fact that the vertical speed is calculated from the magnetic measurements, i.e. the sensors are connected to the field of the control coil current, which creates additional oscillations.

The model of the vertical plasma velocity varies substantially during the discharge. In addition, a significant uncertainty of the model arises when linear approximation of the actuator, working on complex nonlinear principles, is done. Thus, it is impossible to use one set of stationary parameters of the controllers throughout the whole discharge, so the adaptation of the parameters of the controller of vertical stabilization is used on the JET tokamak.

There is an approximation of the dependence of the FRFA switching frequency f_{sw} on the gain k_{vs} in the stabilization loop of the vertical velocity and on the module γ of a single unstable pole of its linear model [38-40]: $f_{sw}=\gamma f(k_{vs})$, where f is a monotone function. An adaptive controller operates on the basis of this ratio, leading away the FRFA mode from saturation. The switching frequency of the FRFA in the JET tokamak is about 500 Hz, which ensures that there is no overheating and provides small amplitude of the plasma oscillations over the entire range of the module of the unstable pole γ of the linear vertical plasma motion model.

1.4. The EAST tokamak (China)

Stabilization of the vertical position and plasma velocity around zero on the EAST tokamak (Experimental Advanced Superconducting Tokamak) is given serious attention, since vertical instability is a risk for the operation of tokamaks. In the EAST, an algorithm called RZIP [41] is used to control the current and position of the plasma. The block diagram of the plasma position and current control system on the limiter phase is presented in Fig. 4.

The value of the plasma current is measured by the Rogowski loop, and the estimations of the vertical and horizontal positions of the magnetic axis are reconstructed from the signals of the magnetic diagnostics outside the plasma. The feedback is closed via PID controllers and matrices M_matrix(R_p, Z_p) and M_matrix(I_p) [41]. The control is carried out by means of currents in poloidal field coils, for which the target values consist of the sum of the scenario signals and signals of the closed-loop control system. The coils of the central solenoid are used to control the plasma current, the vertical and horizontal positions of the plasma are controlled by the coils of the poloidal field.

Additional coils connected to high-speed power supplies and located inside the vacuum vessel are used to suppress the vertical plasma instability, as well as in the ITER project.

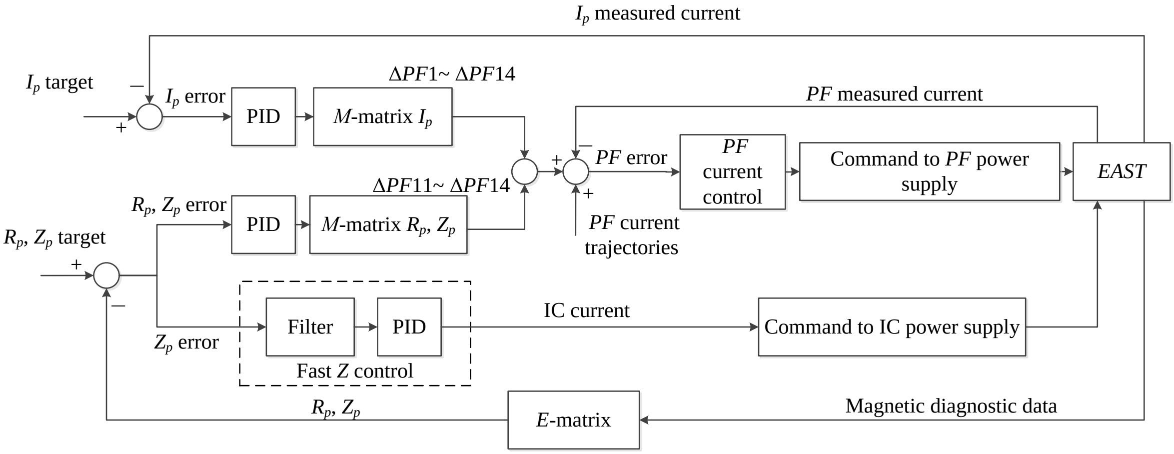


Fig. 4. The block diagram of the plasma current and position control system on the EAST tokamak on the limiter phase of the discharge [41] (IC is an active feedback coil inside the vacuum vessel)

The results of tracking the vertical and horizontal position of the plasma in the EAST for the target in the form of triangular pulses in the real experiment, discharges 10112 and 10113 respectively, are shown in Fig. 5. The control error for the vertical position does not exceed 1 mm. The control contour of the horizontal position tracks the target noticeably more slowly due to the effect of the field penetration into the conducting structures of the vessel during horizontal displacement. The control error does not exceed 2 mm after the end of the transient process.

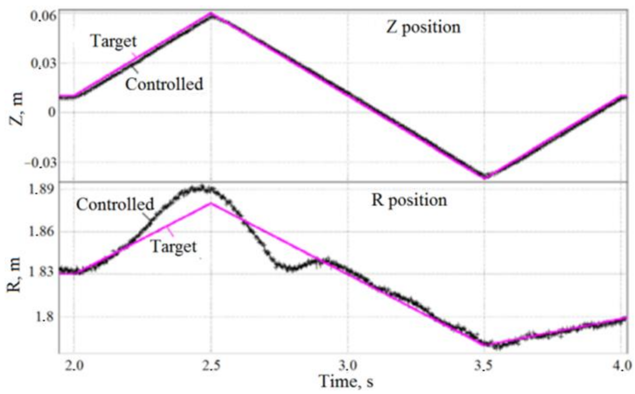


Fig. 5. Tracking the vertical and horizontal position of the plasma on the EAST tokamak [41]

The Italian CREATE group and Chinese specialists during the experimental campaign of 2016 on EAST developed and applied the stabilization system of the plasma vertical velocity around zero, that was frequency decoupled with the plasma current and shape control system [42]. The control directly of the plasma position was conducted using the plasma shape control system. The controller scheme of this system is shown in Fig. 6, it is similar to the plasma control system used on the JET tokamak [38-40].

The vertical plasma position control system in the EAST tokamak is modeled on the TSC (Tokamak Simulation Code, USA) plasma-physical code. It is claimed in [43] that the results of modeling on the main parameters, such as plasma current, plasma shape and position, flux contours and magnetic measurements, coincided well with the experimental data.

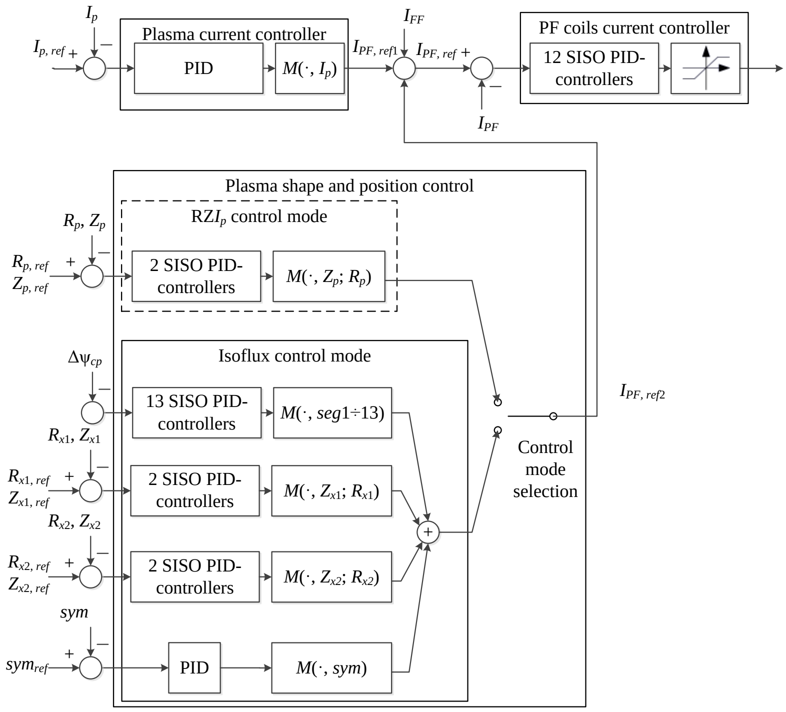


Fig. 6. The simplified scheme of the position, current and plasma shape controller on the EAST tokamak: *M* matrices are used to distribute signals between 12 current control loops in poloidal field coils; the target currents are monitored by the current controller in PFC coils [42]

2.PLASMA POSITION, CURRENT, AND SHAPE CONTROL SYSTEMS

All tokamaks given in Table 2 of the first part of this survey [1] have a common similarity consisting in the vertical elongation, but all of them differ from each other by the poloidal systems, that are the magnetic coils systems creating poloidal fields. The JET tokamak has an iron core that other tokamaks do not have. All the rest of tokamaks have an air central solenoid. The iron core makes additional problems associated with its nonlinear magnetization curve. Tokamaks DIII-D, NSTX, JT-60U, and TCV have the poloidal field coils inside the toroidal field coil; other tokamaks on the contrary have the poloidal field coils outside the toroidal field coil. The EAST and ITER tokamaks have all superconductive coils. These tokamaks have the horizontal field coils inside the vacuum vessel in order to stabilize the unstable plasma vertical position, which significantly expands the area of controllability and stability on the vertical coordinate with limited power supply sources of these coils. In all tokamaks, the coils of the poloidal field are located differently in the space around the vacuum vessel. Such difference of the poloidal systems of known tokamaks leads to the different configurations of the plasma position, current, and shape control systems.

2.1. DIII-D tokamak (USA)

A general view of the DIII-D tokamak is shown in Fig 7, *a* and the magnetic configuration of the plasma in it is shown in Fig. 7, *b* [44]. On the DIII-D tokamak the magnetic flux is controlled at 13 points on the separatrix (Fig. 7, *b*). The block diagram of the magnetic plasma control in the DIII-D tokamak is shown in Fig. 8, *a*, Fig. 8, *b* shows the result of tracking the upper and lower gaps between the separatrix and the first wall, as well as the X-point coordinates.

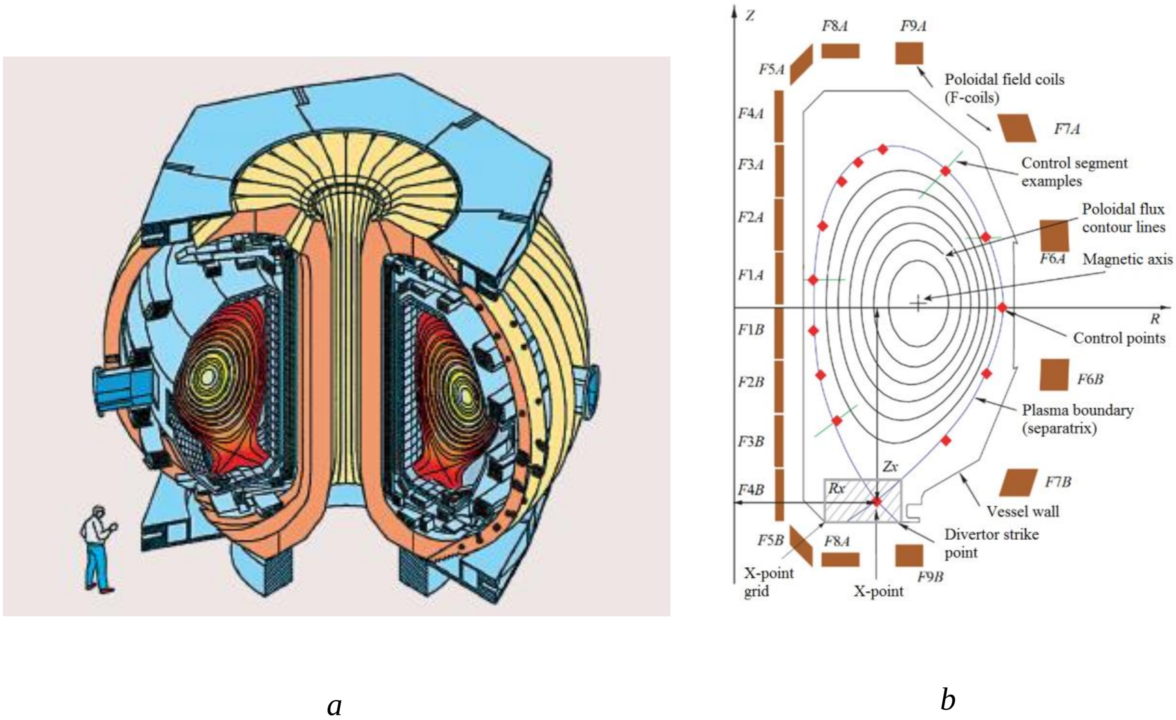


Fig. 7. DIII-D tokamak: *a* – general view (internet resource), *б* – poloidal flux contour lines (configuration with lower *X*-point) [43]

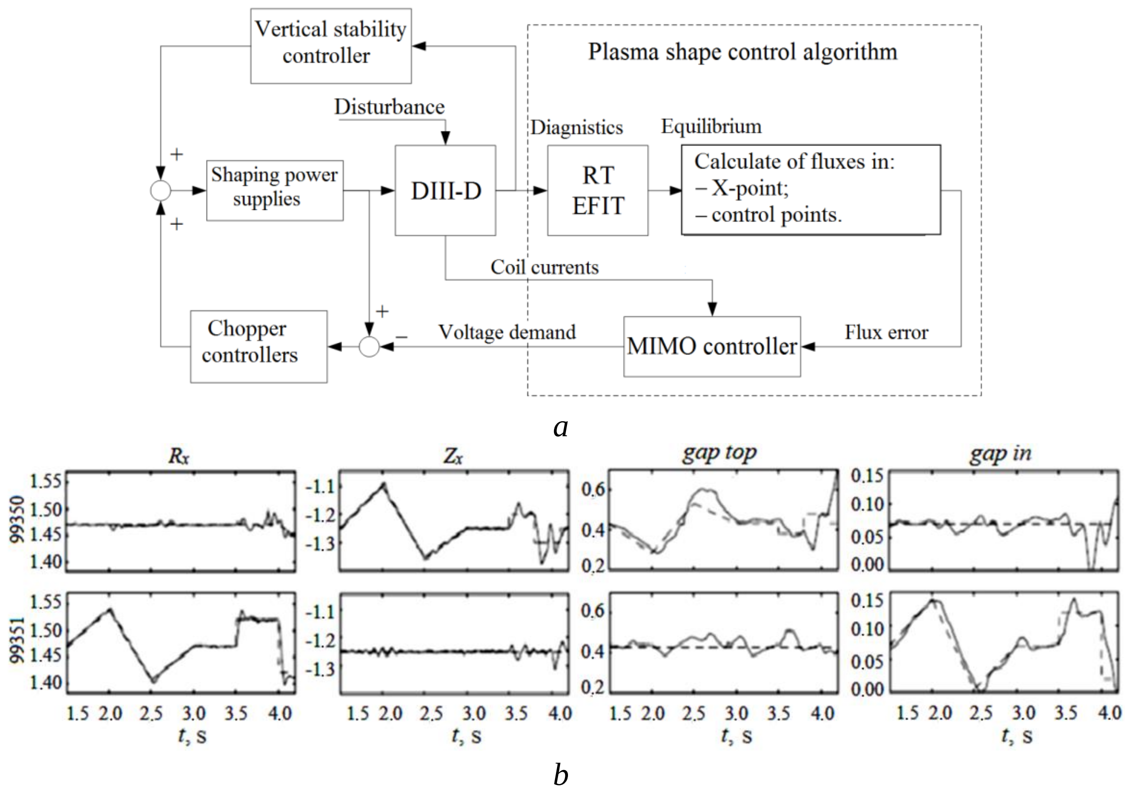


Fig. 8. Plasma control on the DIII-D tokamak [45]**:** *a* – block diagram of the isoflux-control at the 13 separatrix points (see Fig. 7, *b*); RT – real time; *b* – system response to control of the upper and left gaps between the separatrix and the first wall and X-point position coordinates.

On the DIII-D tokamak, the following ideology of developing plasma position, current, and shape control system is adopted [44]. The vertical plasma instability is suppressed through a separate circuit, which stabilizes the plasma vertical velocity around zero (see Fig. 8, *a*). A set of coils is used to control the plasma current. The 18 poloidal field F-coils are used to control the plasma shape. The isoflux-control method is applied: magnetic flux at 13 points on the plasma separatrix, as well as vertical and horizontal position of the X-point are calculated in real time using signals of magnetic diagnostics of currents, fluxes and fields by the EFIT algorithm. The error signals between the specified (reference) values of magnetic flux on the separatrix and the X-point coordinates are fed to the multivariable isoflux controller (MIMO controller). The isoflux controller also receives signals of currents in the control coils. The output signals of the controller come to the inputs of the actuators, thereby closing the multivariable feedback loop. The multivariable plasma shape controller itself is designed on the basis of linear models, which are generated by a special software package named TokSys. The set of points is given, which determines the desired location of the plasma separatrix and in which control will be carried out. Currents in the poloidal field coils are tuned so as to keep equal magnetic poloidal flux at the X-point and at all the other boundary points. Let the difference between the flux at the controlled points and the given flux references at these points be defined as $\delta\Psi$. The relation between the change of currents in the control coils δI and $\delta\Psi$ can be represented as $\delta I = M^{-1}\delta\Psi$. Here M^{-1} is a control matrix, which is the inverse matrix to the matrix M composed of the values of the Green's function for the Grad–Shafranov equation. The elements of the matrix M are poloidal flux values at each control point while control currents equal to unit.

2.2.ASDEX Upgrade tokamak (Germany)

A general view of the ASDEX Upgrade tokamak (modernized Axially Symmetric Divertor Experiment) is shown in Fig. 9, *a*, and its cross section is shown in Fig. 9, b. The control of the vertical and horizontal position of the plasma, as well as the coordinates of the divertor strike-points is carried out on this tokamak (Fig. 10, 11).

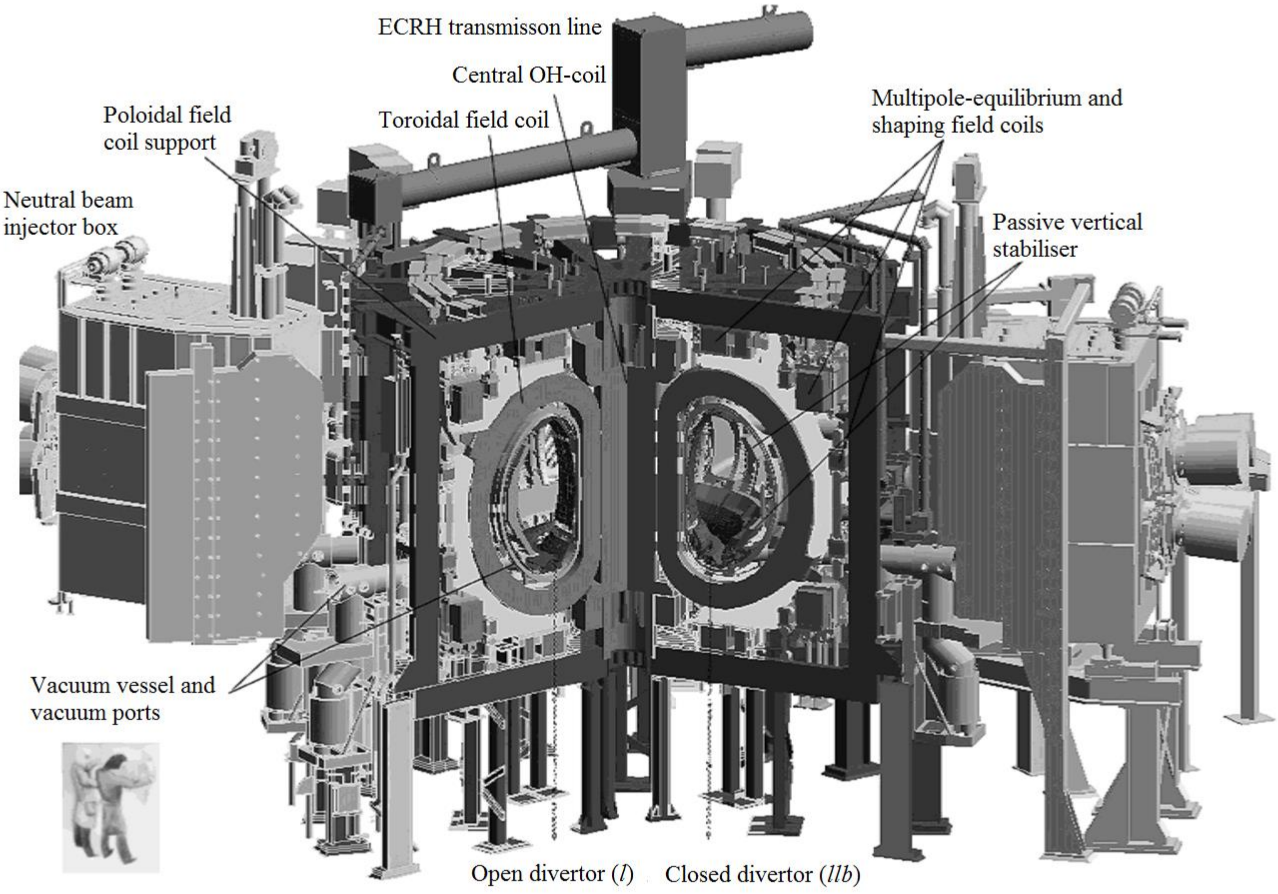
2.3.JET (UK) tokamak

On the JET tokamak (Fig. 12) various control modes are used in conjunction with the vertical stabilization of the plasma, they are the horizontal position of the plasma, the currents in the poloidal fields coils, the plasma current, the coordinates of the divertor strike-points control modes, and the control mode of the plasma shape by the gaps between the first wall and the separatrix [38].

The structure of the controller in JET is founded on physical principles and based on the Kirchhoff’s vector equation

$$V_{PF} = M\,\frac{dI_{PF}}{dt} + RI_{PF}\,,$$

where V_{PF} and I_{PF} are voltage vectors on the coils and the measured currents in them, M and R are matrices of mutual inductance and resistance, respectively, control of currents in the poloidal field coils is carried out in absolute values. Not all coils are used for different control modes, but the most efficient individual coils or sets of coils. Control modes can be combined: for example, most discharges can be performed with simultaneous use of modes of the currents in the poloidal field coils control, the plasma current control, the divertor strike-points coordinates control, and plasma shape control by values of the gaps. The control system has a cascade structure [49] shown in Fig. 13, where there are controllers for plasma shape, current, and position of divertor strike-points in the block "Shape and plasma current controller".



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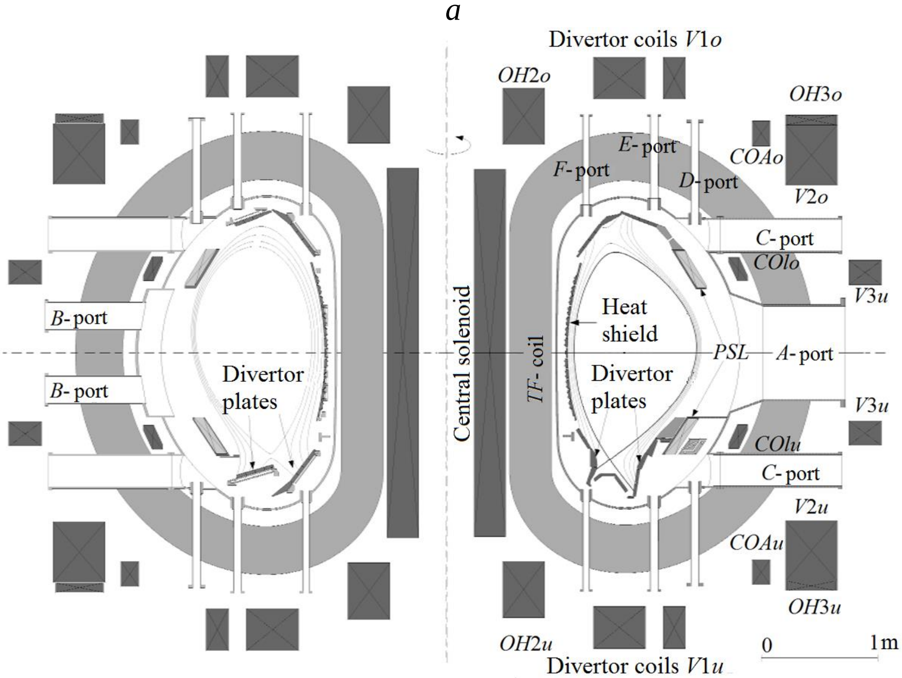


b

Fig. 9. Tokamak ASDEX Upgrade: a – general view [46]; b –cross section with poloidal field coils [47]; the control coil for the vertical position of the plasma is placed between the vacuum vessel and the toroidal winding

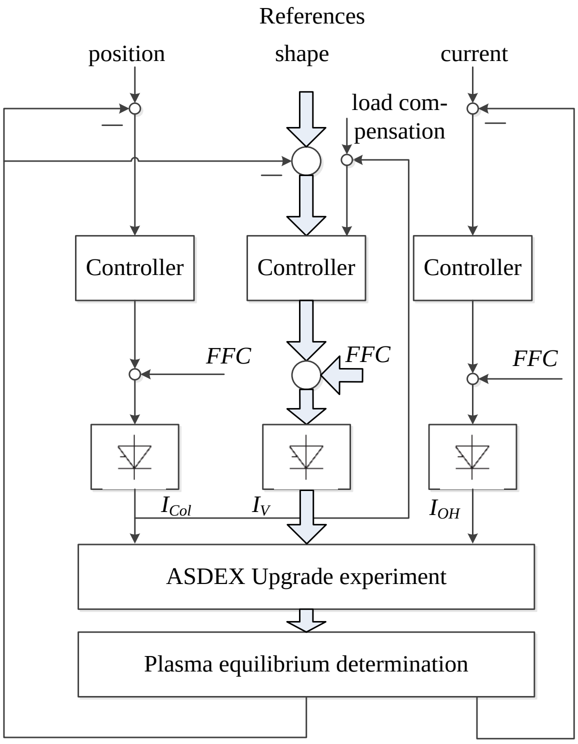


Fig. 10. Block diagram of the magnetic plasma control system of the ASDEX Upgrade tokamak [48]

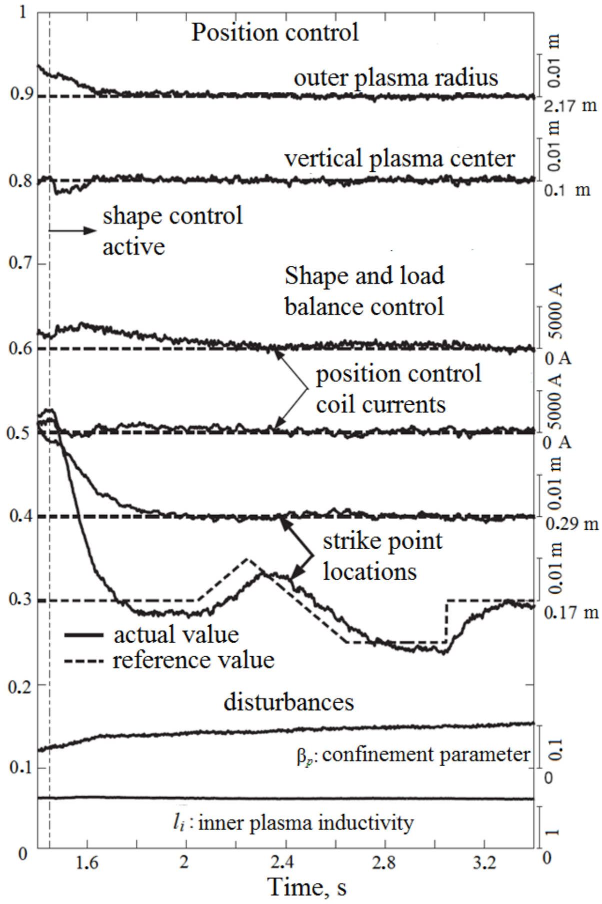
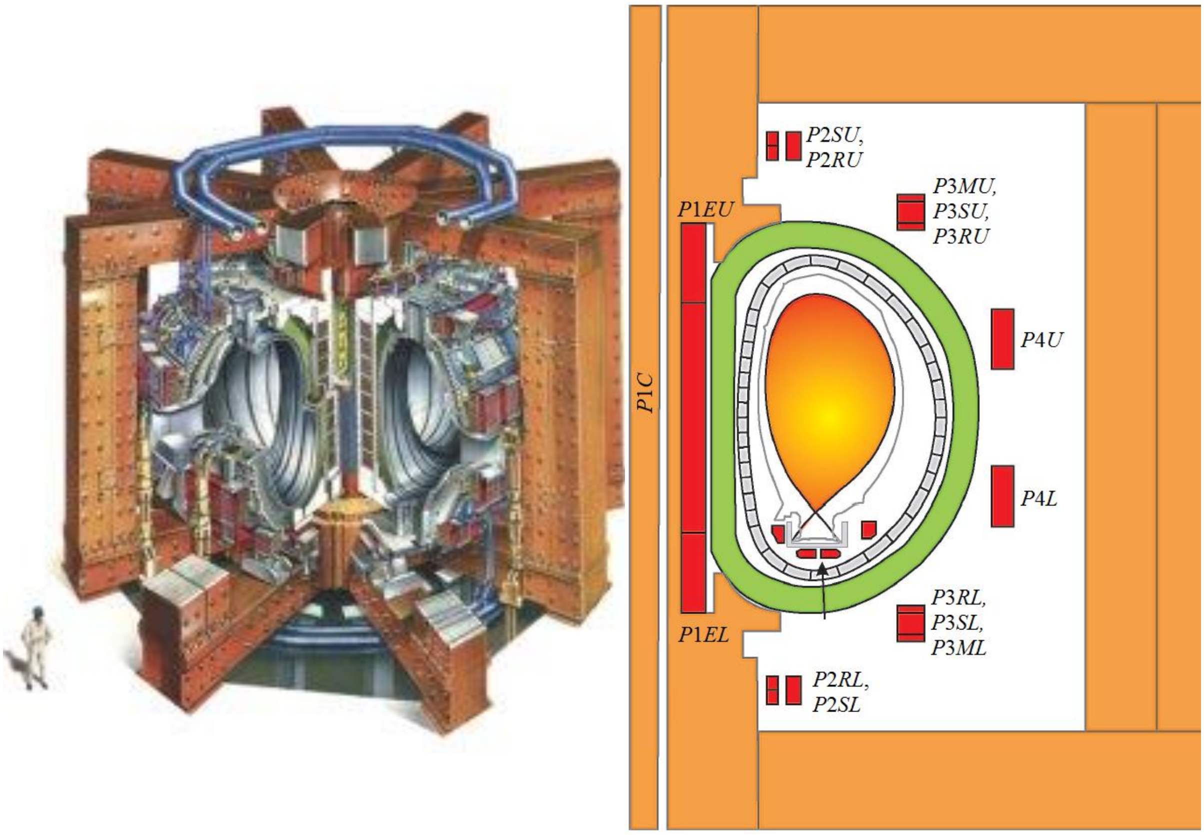


Fig. 11. Plasma position and divertor strike-points location control response on the ASDEX Upgrade tokamak [48]

The general control law for controllers [38] is $V_{PF} = R_{est}\, I_{PF} + K\,(Y_{ref} - Y)$, where R_{est} is the estimation of the resistance matrix, K is the matrix gain, Y_{ref} and Y are the vectors of the reference inputs and the measured outputs. In the control law structure there is the compensation element $R_{est}\, I_{PF}$ of the voltage on the resistance R_{est} that makes the behavior of the closed-loop control system of currents in JET coils close to the devices with superconducting coils and eliminates the need to introduce integral components for increasing the degree of astatism and achieve the required tracking error 1-2% [38]. The matrix gain is synthesized in the form of $K = H^*(M_{est}\, T^{-1}C^{-1})$, where H is the matrix of inputs/outputs choice, M_{est} is the estimation of the mutual inductance matrix used for decoupling of the currents in the poloidal field coils, T is the static gain matrix linking the variations of the currents in the coils with variations of output variables: $\delta Y = T\, \delta I_{PF}$, and C is the matrix of the desired values of time constants in the closed-loop system.

The plasma model in the vessel is taken into account as a coil with a distributed current. In doing so, the plasma resistance and mutual inductance of the plasma with all coils [50], except the coil $P1$, are neglected, that is justified by the slow dynamics of the plasma current. In practice, only during disruption the plasma can induce a significant voltage on the poloidal field coils.

The values of magnetic fluxes at the points of the equatorial plane on the outer and inner circumference are used for the plasma horizontal position control mode. Since the plasma separatrix is a line of the equal flux, it is enough to make the difference of fluxes in these points equal to zero by the coil $P4$ in order to stabilize the horizontal position (Fig. 12, *b*, 14, *a*).



a *b*

Fig. 12. JET tokamak: *a* – general view (internet resource), *b* – cross-sectional view with control coils and central solenoid [38]

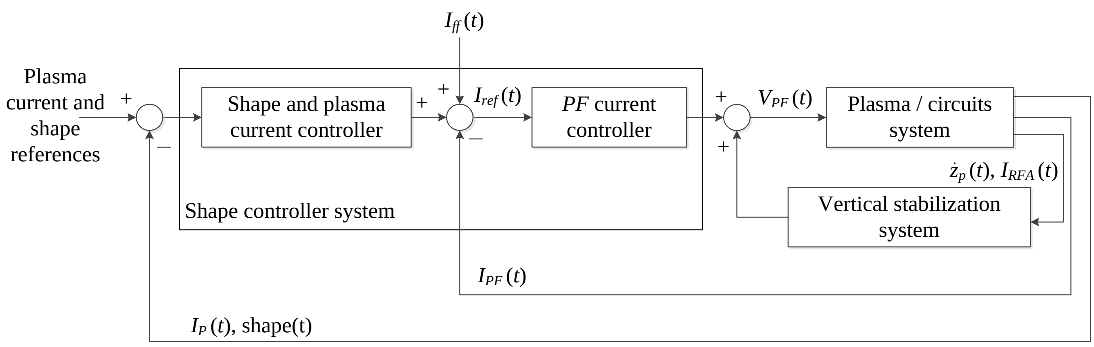
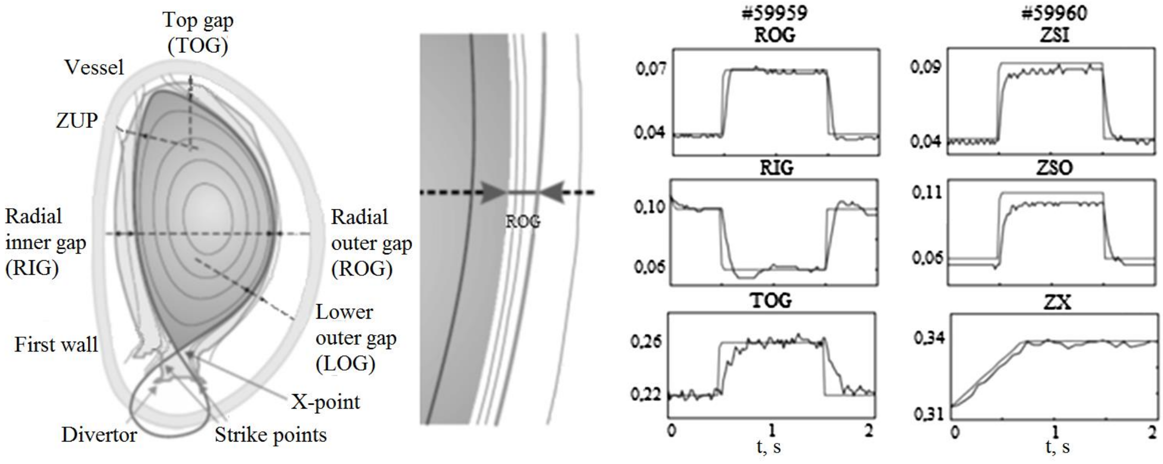


Fig. 13. Block diagram of the plasma position, shape, and current on JET **[49]**

The XLOC algorithm [51] is used to calculate the gaps values between the vessel first wall and the plasma separatrix in the mode of plasma shape control. The gaps response to the test signals in the closed-loop control system on the JET tokamak is shown in Fig 14, *b* when a discharge duration of at least 2 seconds.

The poloidal field coils D2 and D3 are used to control the locations of the divertor strike-points (see Fig. 12, b). The signals of location of the intersection points of the separatrix with two vertical or horizontal axes (the specific option is set by the operator) are used as feedback signals. An oscillating mode of control for the location of the divertor strike-points is used for energy scattering, which allows to distribute the energy over a larger area of the divertor. It is implemented by introducing the reference saw-edged signal at frequency of 4 Hz while the control loop bandwidth has a value of 10 Hz.

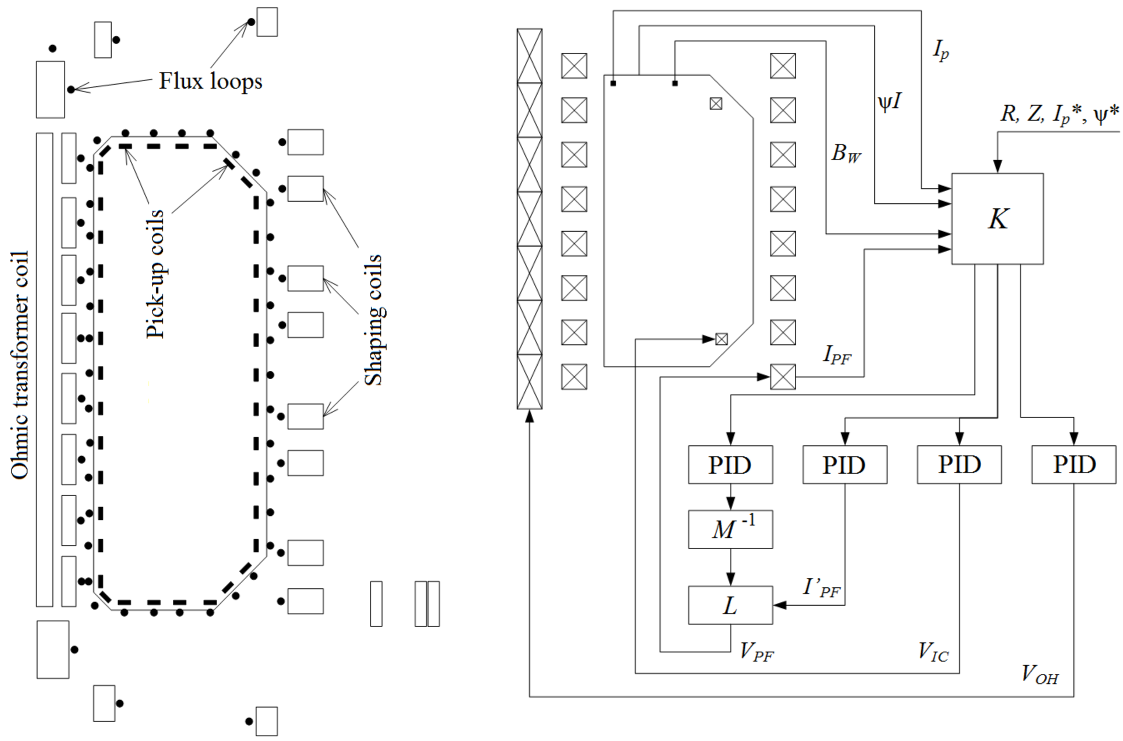


a *b*

Fig. 14. Tokamak JET: *a* – vertical cross-section, *b* – gaps response to the test signals [38]

2.4.TCV tokamak (Switzerland)

The control method of the plasma shape in the TCV tokamak (Variable Configuration Tokamak) [52] by the values of the magnetic flux and field at certain points on the plasma separatrix is illustrated in Fig. 15 [53]. The method is based on the fact that the separatrix is a line of a constant magnetic flux, and the X-point in the divertor configuration is the point of zero field. Thus, having chosen the plasma shape scenario in advance, it is necessary to align the flux values at the selected points, reducing the field to zero at the expected X-point.

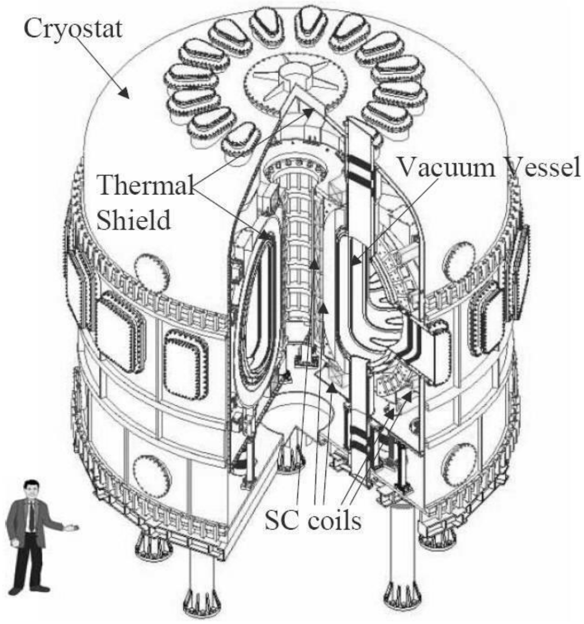


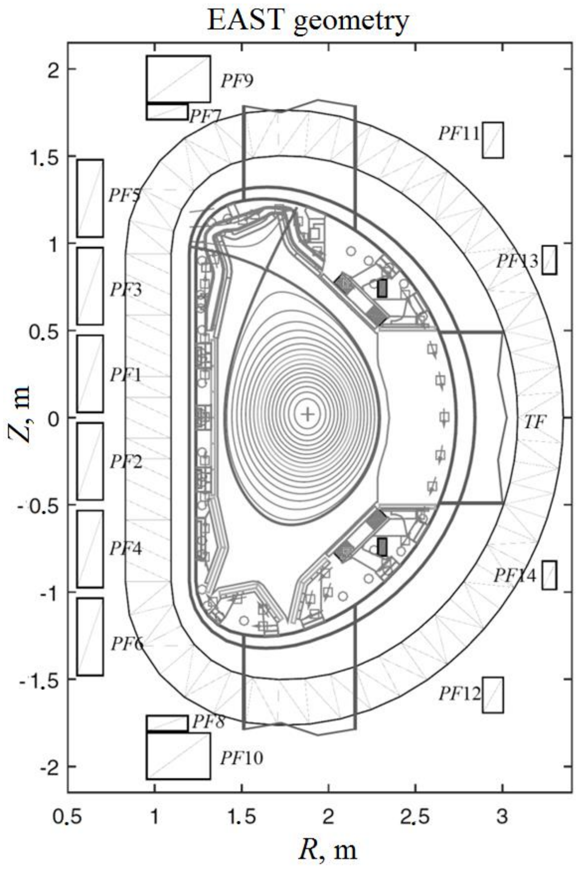
a *b*

Fig. 15. Tokamak TCV: *a* – cross-section; *b* – block diagram of the plasma shape control by flux and field values on the separatrix of the TCV tokamak [53]

The control scheme (Fig. 15, *b*) consists of four main elements. One of them is the element *K*, it serves to calculate the deviations of the magnetic flux and field on the separatrix from the scenario values as a linear combination of the measurements of the sensors outside the plasma. Another element is a block of PID controllers, one for each deviation signal, which reduces the values of controlled signals to zero. The next element M^{-1} calculates the derivatives of currents in the poloidal field coils, in which the current deviations will be reduced to zero in the finite time. This calculation is carried out for the model of magnetic configuration without plasma and currents in the passive structures of the vessel. Finally, the element *L* calculates on the basis of the Kirchhoff's equations the required values of the voltages on the poloidal field coils as a function of the currents and their derivatives, as well as the voltage on the plasma ohmic heating coil. The given algorithm does not take into account the model of the plasma in the vessel of the tokamak and, therefore, can be used throughout the discharge; however, it does not allow achieving a high degree of decoupling of control channels.

This approach was applied to the KSTAR tokamak [54], and its modified versions are being used in DIII-D, EAST and Alcator C-Mod [55].





a *b*

Fig. 16. Tokamak EAST: *a* – construction; *b* – cross-section [42]

2.5.EAST tokamak (China)

The complete reduced copy of the ITER is the Chinese EAST tokamak with superconducting coils (SC) (Fig. 16), in which there are control coils inside the tokamak vessel. The system of plasma shape magnetic control for the EAST, that was developed jointly by the Chinese and American specialists, is one of the most advanced.

The two algorithms for the plasma shape control on the isoflux-control principle (control of the magnetic flux and field values at certain points on the plasma separatrix) are used. These algorithms are ISO-elong and ISO-dnull to control the elongated limiter configuration and the divertor configuration with two X-points, respectively. Fig. 17 shows the basic block diagram of the control system for these algorithms.

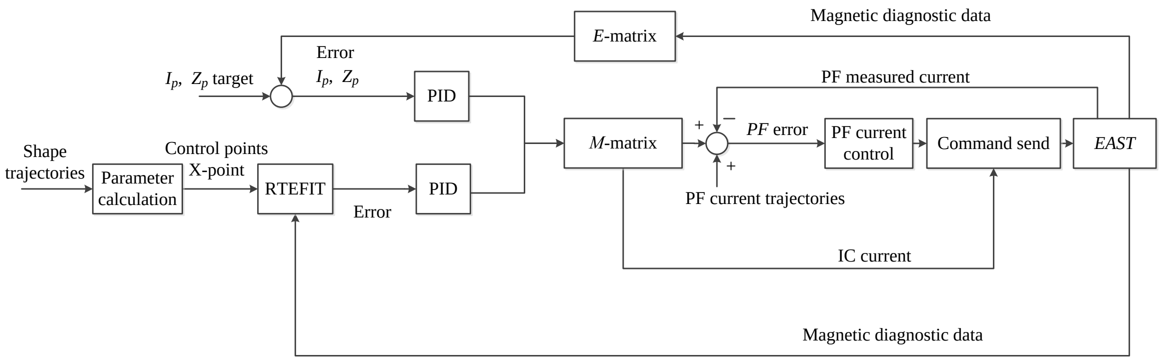


Fig. 17. Block diagram of the plasma current, shape, and vertical position control systems on the EAST tokamak for the divertor discharge phase (isoflux control): **RTEFIT (*Real Time Equilibrium Fitting*) [41]**

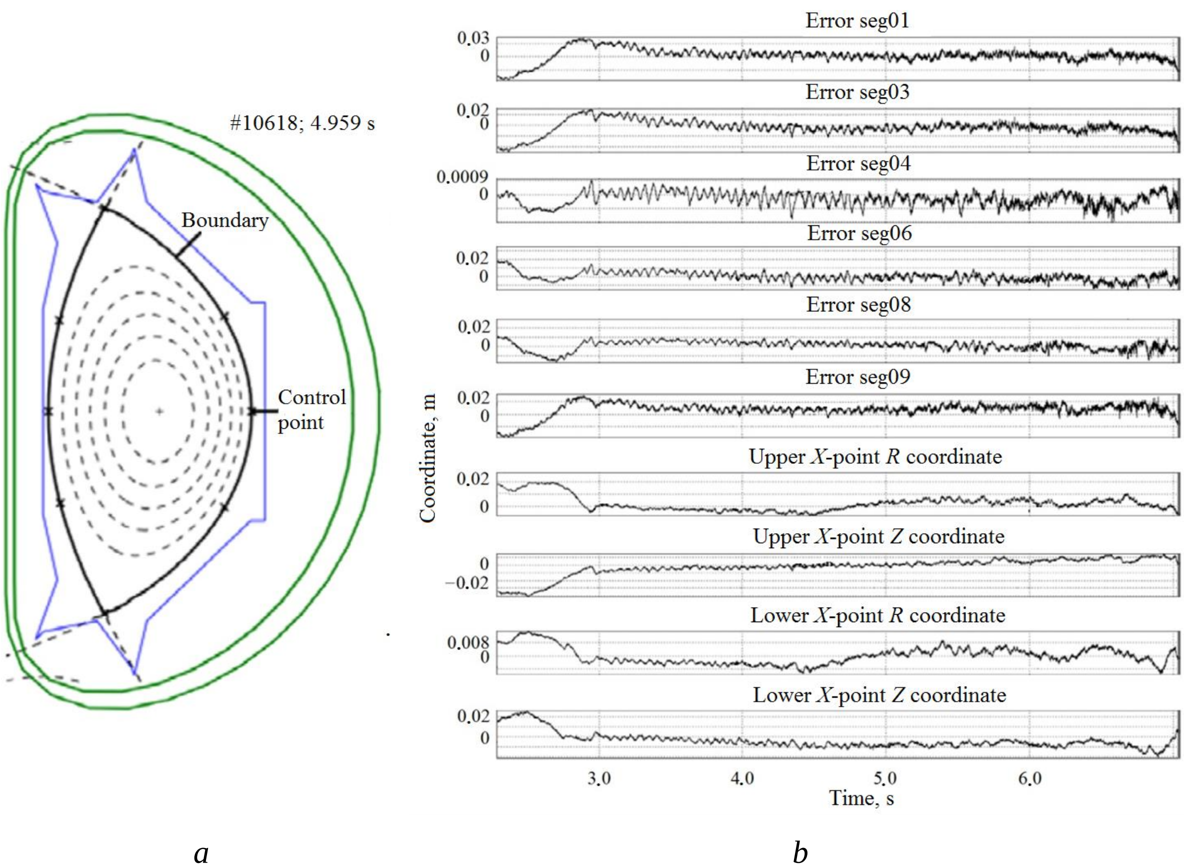


Fig. 18. Plasma shape and position stabilization on the EAST tokamak for discharge 10618: *a* – magnetic configuration of EAST at the moment of 4.958 s; *b* – tracking errors at the reference points and the X-points [41]

The positions of characteristic points on the separatrix, such as X-points or points of contact with the limiter, are calculated during the plasma shape reconstruction. The maximum value of the magnetic flux from the characteristic points is selected as the reference for all points. Then, the differences of the magnetic flux values are used to close the feedback through the PID controllers (see Fig. 17). One or two nearest poloidal field coils control the value at each reference point, the plasma current is controlled by the central solenoid coils. The reference currents in the coils are calculated using the matrices of the control channel decoupling, scenario signals are added to them, and they are fed to the multivariable control loop of the currents in the poloidal field coils.

The typical value of the unstable pole module on the EAST tokamak is 200-300 s^{-1}; it can exceed 1000 s^{-1} for configurations with greater elongation or non-standard plasma parameters. It is necessary to quickly calculate the vertical position of the plasma and use high-speed power supplies as actuators in order to suppress vertical instability with such dynamics. An analogue of the RZIP algorithm is used in the control system to calculate the estimation of the plasma vertical position, which is controlled by individual coils located inside the vessel.

Fig. 18, *b* shows the signals for discharge 10618 [41] with an elongation of 2.0 and a plasma current of up to 250 kA. Algorithm ISO-dnull starts working at the moment of 2.7 s and by 3.0 s the differences of values of the magnetic flux are less than 0.001 V·s/rad, and the error of X-points position control is less than 1 cm.

3.SPHERICAL TOKAMAKS

3.1.Advantages of spherical tokamaks

For successful operation of a thermonuclear reactor the plasma parameters must satisfy the numerous constraints imposed by the magnetohydrodynamics theory. One of the most important constraints is the maximum achievable β which characterizes the efficiency of plasma confinement and is defined as the ratio of the plasma pressure to the magnetic field pressure. Numerical calculations [56] show that as the aspect ratio *A* decreases, the maximum acceptable β increases as $1/A$ for conventional tokamaks ($A \approx 3$) and faster for spherical tokamaks ($A \approx 1.5$). Thus, the best plasma confinement can be achieved on spherical tokamaks.

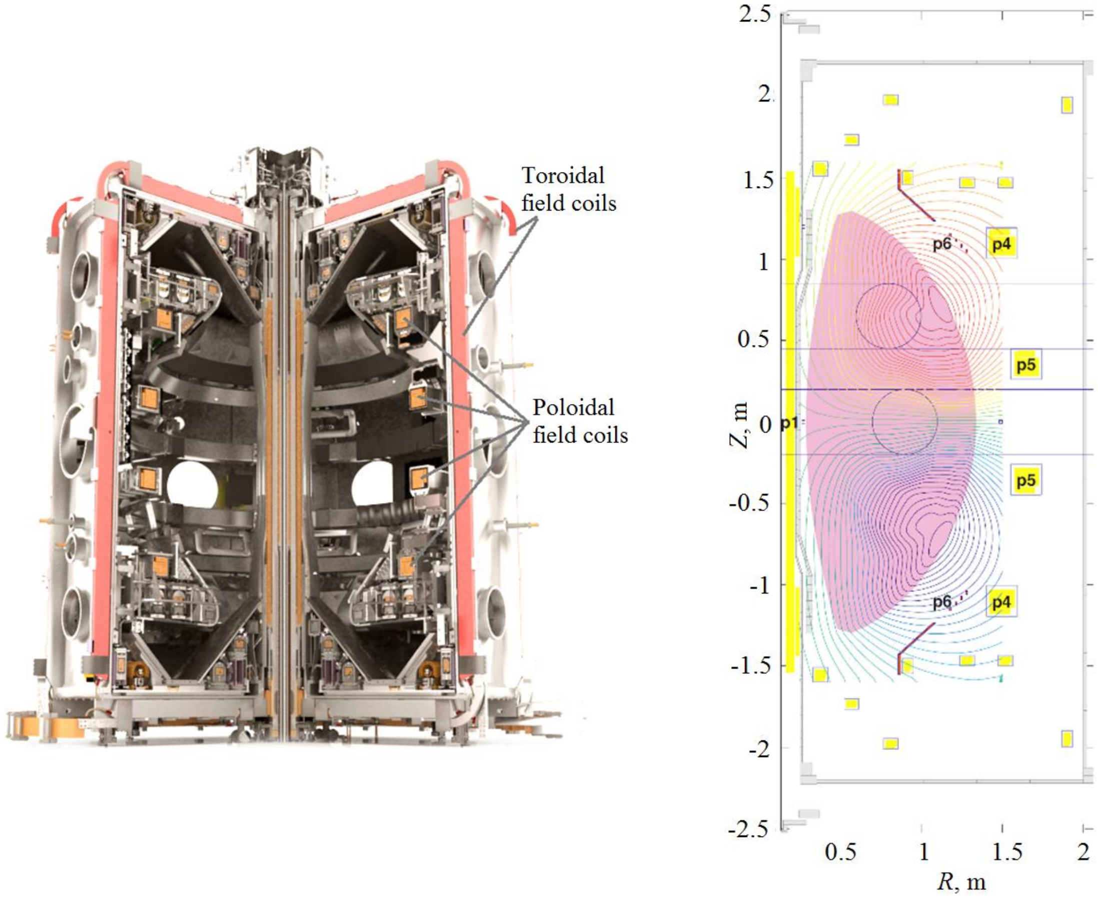
Another advantage of spherical tokamaks is the high value of the safety factor *q* at the plasma boundary. The safety factor increases as $1/A(1-A^{-2})^{3/2}$ with decreasing aspect ratio, allowing to suppress the kink instability and enabling higher plasma current than on conventional tokamaks with the same magnetic field strength and small radius of the plasma.

Finally, relatively small size of the spherical tokamaks makes their creation and operation less costly, and allows for higher values of magnetic and electric fields to be attained with the same currents as in conventional tokamaks.

These advantages mark the spherical tokamaks as promising candidates to be the future commercial thermonuclear power plants, which gives more relevance and significance to the plasma control problems on spherical tokamaks. In this context, the results of plasma control for the operating spherical tokamaks MAST, NSTX, and Globus-M are given below.

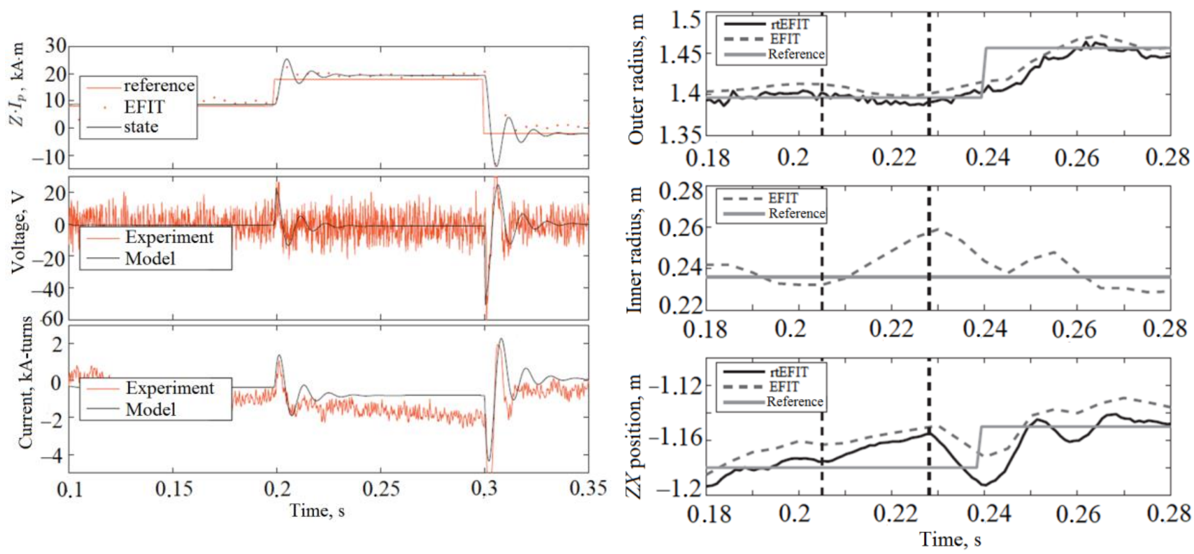
3.2.MAST-U tokamak (UK)

The cross-section and structure of the MAST-U tokamak (*Mega Ampere Spherical Tokamak*) are shown in Fig. 19. The vertical position control on MAST-U is implemented via the PD controller [57], while not the plasma vertical coordinate is taken as controlled variable but the product of plasma vertical coordinate and its current (Fig. 20, a). Like on EAST and DIII-D tokamaks, the *isoflux control* method is applied for the shape control on the MAST-U tokamak, with the real-time RTEFIT plasma equilibrium reconstruction code calculating flux values at control points [59].



a *b*

Fig. 19. MAST-U tokamak: *a* – tokamak structure, *b* – cross-section [57]



a *b*

Fig. 20. Plasma shape and position stabilization on MAST-U: *a* – vertical position control [56]; *b* – inner and outer plasma radii control, *X*-point position control [59]

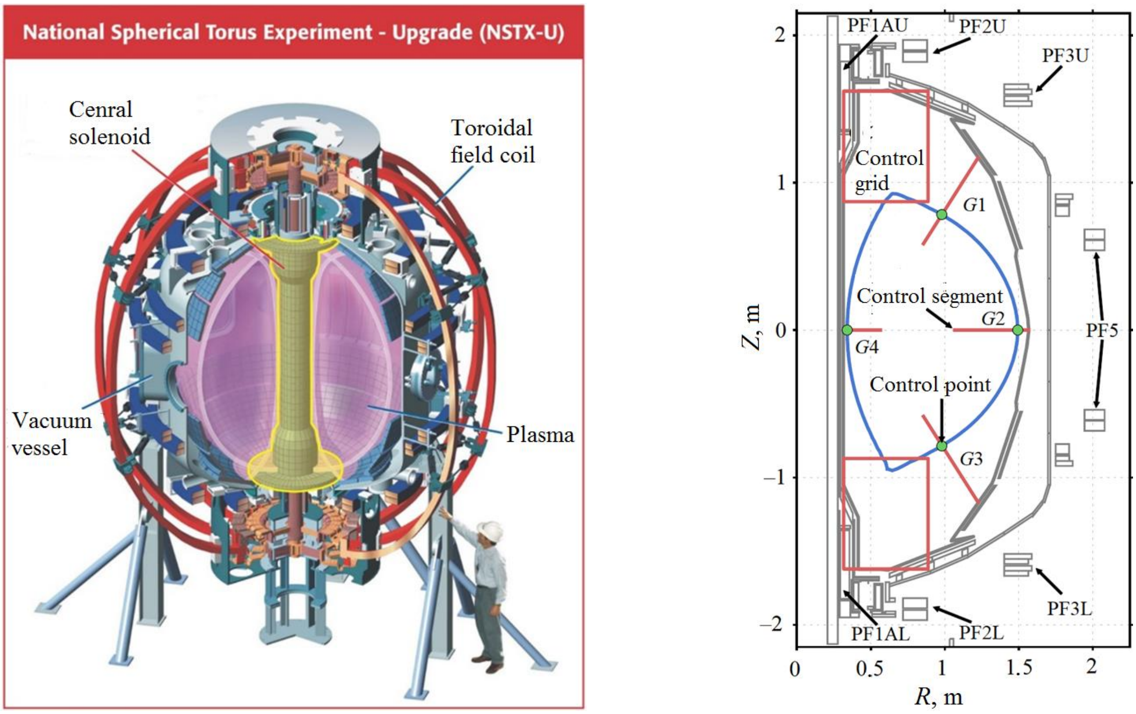


Fig. 21. NSTX-U tokamak: *a* – structure; *b* – cross-section [58]

3.3.NSTX-U tokamak (USA)

The structure and cross-section of the NSTX-U tokamak (National Spherical Torus Experiment) are shown in Fig. 21. The isoflux control method [58] is used to control the plasma shape on the NSTX-U tokamak. Magnetic fluxes in the set of control points are calculated in real time using the RTEFIT code and via PID controllers are equated to a given flux that is determined by the flux at the X-point or the flux at the contact point of the plasma with the vessel.

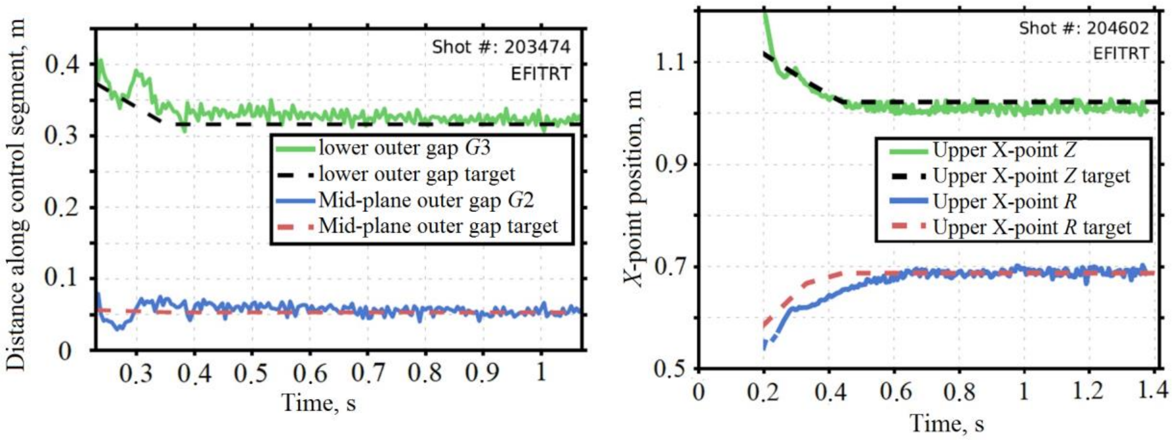


Fig. 22. Plasma shape stabilization on NSTX-U [58]: a – resulting gaps on the outer side of the plasma; b –X-point position control

PID controllers are also used to control the position of X-points and strike points. A control contour that minimizes the horizontal distance between the flux isolines of the upper and lower X-points is used for configurations with two X-points. Fig. 22a shows the resulting outputs of the control system for gaps between control points on the outer side of the plasma and the walls of the vessel, the result of the X-point position control is shown in Fig. 22, b.

3.4.Globus-M tokamak (Russia)

Spherical tokamak Globus-M (A.F. Ioffe Physical-Technical Institute of the Russian Academy of Sciences, St. Petersburg) [1] is the only tokamak with a vertically elongated cross-section in the Russian Federation, Fig. 23 [60, 61]. Feedback systems are applied on the Globus-M tokamak to control the horizontal and vertical position of the plasma with high-speed thyristor current inverters as actuators [37]. There is a set of poloidal field coils on the tokamak that is incorporated into the control loops of currents in these coils with multiphase thyristor rectifiers [72] and PD-controllers. So these loops enable to use program control to control plasma magnetic surfaces in each discharge. That has made it possible to collect a database of plasma discharges, which was used to develop and simulate hierarchical control systems for the position, current, and shape of the plasma with the equilibrium reconstruction codes in the feedback. Two codes for the plasma equilibrium reconstruction from magnetic measurements outside the plasma have been developed in the MATLAB environment: one using Picard iterations to solve the Grad-Shafranov equation by means of the Green's functions [18, 62] and one using non-iterative moving filaments method to approximate the plasma current distribution [21]. The plasma equilibria reconstructed from the experimental data were used to construct arrays of linear plasma models from which linear models with variable parameters were created by means of linear interpolation [18, 21].

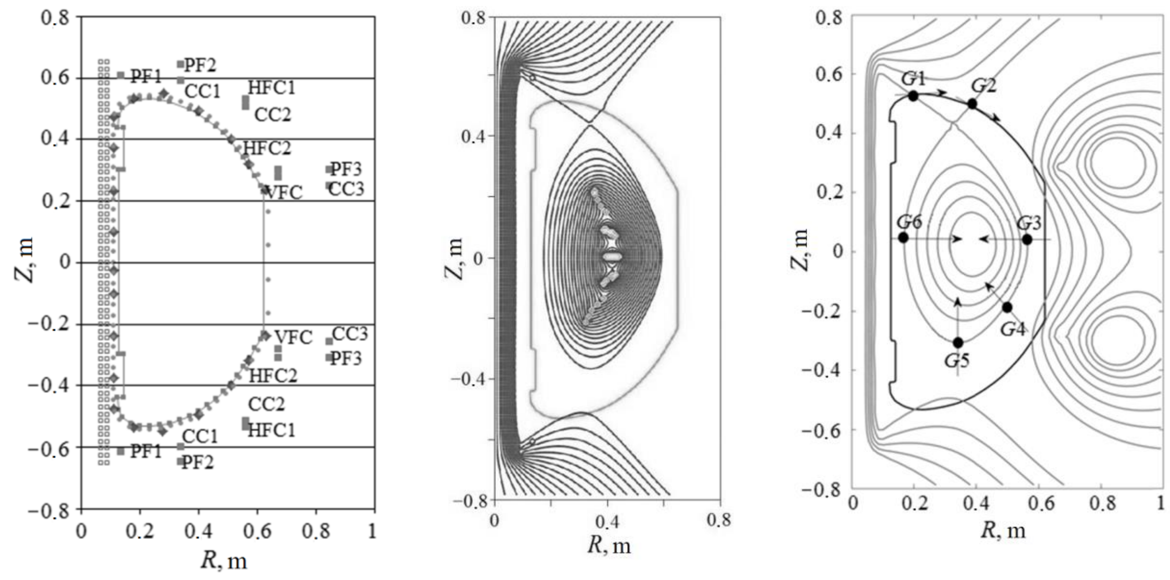
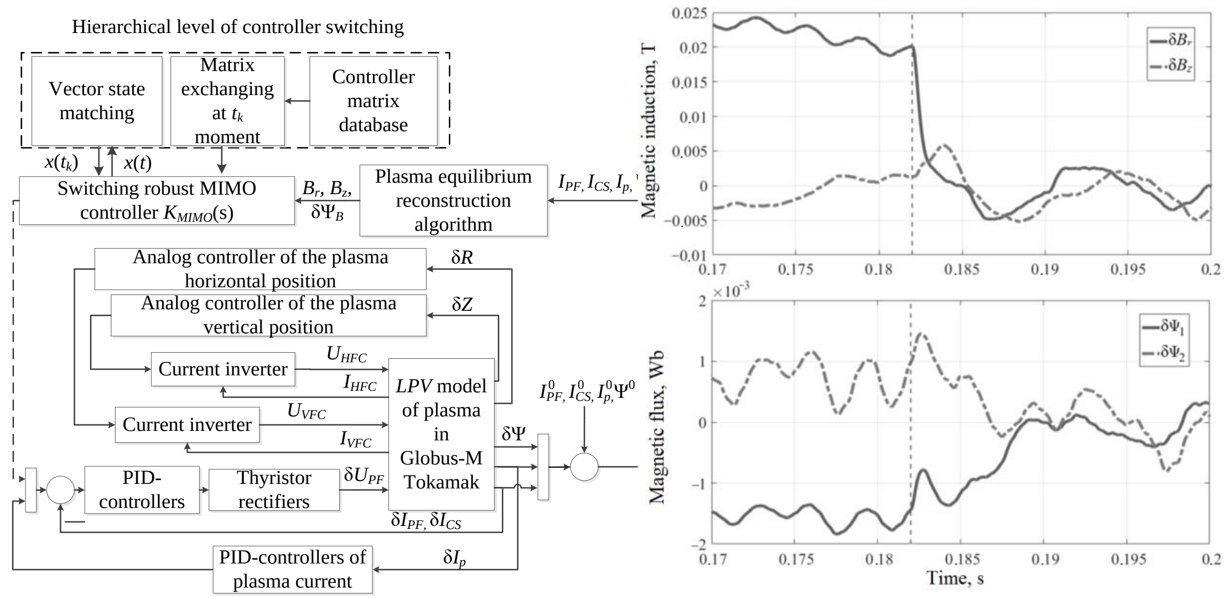


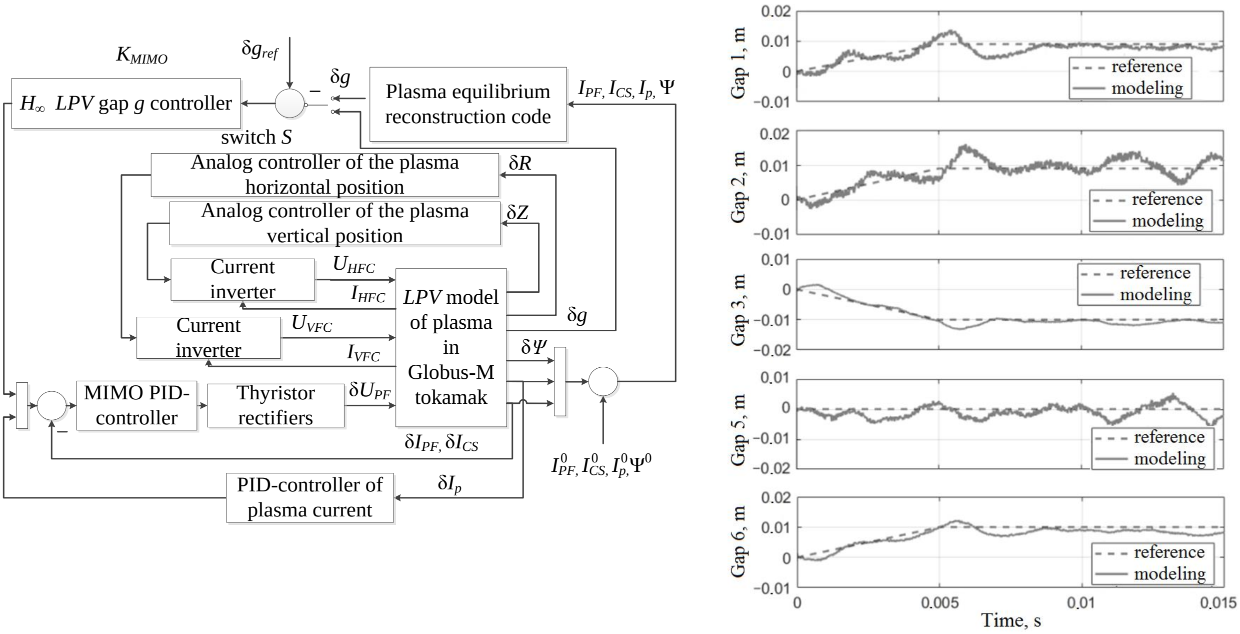
Fig. 23. Globus-M tokamak: *a* – cross-section: – limiter; – magnetic loops; – vacuum vessel; – *PF* coils; – central solenoid; *b* – magnetic configuration reconstructed by the moving filaments method [21]; *c* – magnetic configuration reconstructed by the Picard iterations method with the measuring displacements directions of gaps between the tokamak first wall and separatrix[18, 58]

Systems with time-varying robust H_∞ controllers with switching for isoflux control (Fig. 24) [21] and linear interpolation for controlling gaps between the first wall and separatrix (Fig. 25) [18] were designed for these time-varying plasma models and modeled, using a new technique with simultaneous application of the linear models, experimental scenario signals, and reconstruction codes in the feedback [18, 21, 63, 64]. The combination of the moving filaments reconstruction code with the flux control on the separatrix showed the highest computational speed of the system with the possibility of implementing it in real time on the Globus-M tokamak using the Speedgoat industrial computers (https://www.speedgoat.com) with the SimulinkRT operating system from MathWorks [21].



a *b*

Fig. 24. Time-varying system with controllers switching and moving filaments equilibrium reconstruction code for poloidal flux on separatrix and magnetic field in X-point control on the Globus-M tokamak: *a* – block-diagram with *LPV* (*Linear Parameter Varying*) model, *b* – deviations δB of magnetic field in X-point and deviations $\delta\psi$ of the poloidal flux difference between points on the separatrix during the shift from limiter to divertor phase of the discharge [21].



a *b*

Fig. 25. System with the time-varying controller based on linear interpolation of controllers array and Picard iterations equilibrium reconstruction code for gaps control on the Globus-M tokamak: *a* – block diagram, *b* – deviations of gaps between the plasma separatrix and vacuum vessel [18]

It is worth noting that the poloidal flux control in vacuum vessel without plasma was studied prior to the development of shape control system for the Globus-M tokamak [64].

Stabilization of the plasma vertical position with fast contour is considered in [16], where the method for the plasma vertical position adaptation to its shape is proposed. This method was applied to the Globus-M tokamak [65]. The block diagram of this application is shown in Fig. 26, and the operation of the plasma position, shape and current control system is shown in Fig. 27.

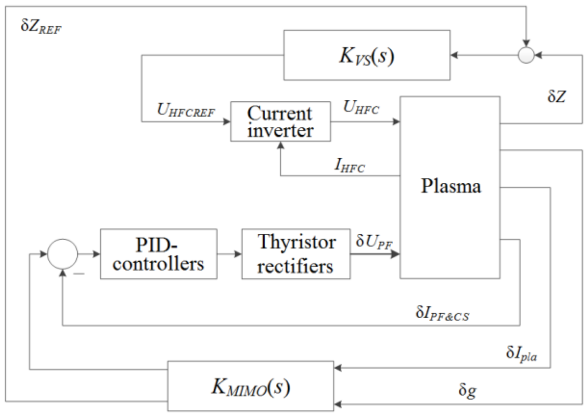


Fig. 26. Block diagram of the hierarchical control system for the plasma position, current and shape with the adaptation of the vertical position of the magnetic axis on the Globus-M tokamak [61]

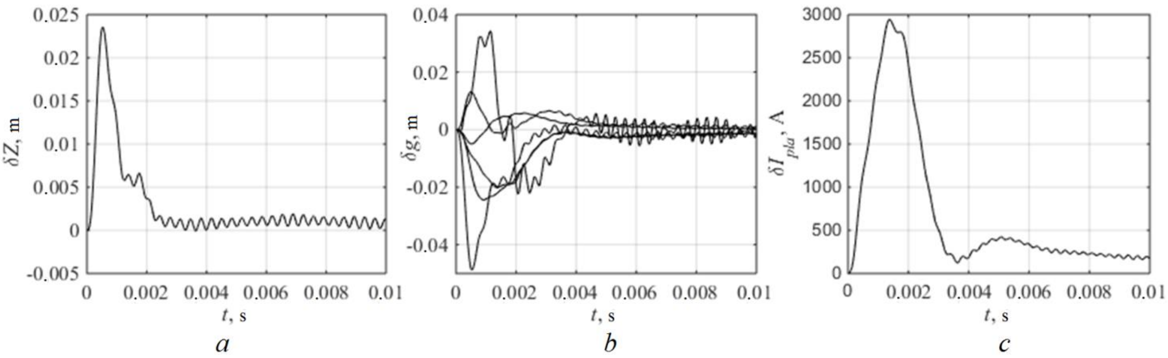


Fig. 27. Modeling results for plasma control system on Globus-M tokamak: *a* – plasma vertical position deviation; *b* – gaps deviation; *c* – plasma current variation [61]

3.5.T-15M tokamak (Russia)

During the developing of plasma control system for the T-15M tokamak (National Research Center "Kurchatov Institute") a part of the work was carried out at the V.A. Trapeznikov Institute of Control Sciences of the Russian Academy of Sciences [66]. An important result of the conducted investigations is the transfer of the horizontal magnetic field coil, intended for control of an unstable vertical position of the plasma, from the position outside of the toroidal field coil to the position between the vacuum vessel and the toroidal field coil [17]. This is due to the fact that the horizontal field coil in its initial position was shielded by the neighboring poloidal field coils needed for the plasma shape control. Because of that the control system acquires the internal instability property. This meant that the plasma position control system required unlimited increase of the current in the horizontal field coil to stabilize the vertical position of the plasma under the minor disruptions. This instability made the system, and, consequently, the entire plant, unworkable. This property of the system was discovered during its development and modeling [67, 68]. While the coil was transferred close to the vacuum vessel, but inside the toroidal field coil, the control system became able to achieve internal stability [69] and operability with the unstable plant under control.

The model of the vertical plasma motion in the T-15 tokamak for the poloidal system with the transferred horizontal field coil [71] was obtained by the identification of the plasma-physical code DINA (SRC RF TRINITI) [70]. After that the plasma vertical position control systems with different actuators have been developed: a multiphase thyristor rectifier and a transistor voltage inverter [72]. In the first case, a modal control system was synthesized with pole placement of the closed-loop system at single point in negative real part of the complex plane for maximum convenience of system tuning, in the second case the system was put into a sliding mode, and revealed its weaker robust properties.

A system with an adaptive predictive model for the variable parameter of the plasma model was developed [73], as well as the system with the placement of the poles of the system in the LMI (*Linear Matrix Inequalities*) -regions to reject the minor disruptions perturbations was designed. The latter system showed the advantages over the modal system in terms of rejection of the external disturbance and accuracy [74].

It has been shown that the plasma shape is controlled in the T-15 tokamak without the use of the horizontal field coil. With the multivariable controller with a state estimator [75] the shape control requires only poloidal field coils and the sections of the central solenoid, like in the ITER version of 1995-1997 [9]. For the T-15 tokamak, the plasma shape control system was modeled on the DINA code with stabilization of the vertical plasma velocity around zero and the LQG controller in the feedback [76].

4.MAGNETIC CONTROL SYSTEMS OF RESISTIVE WALL MODES

Let us consider the resistive wall modes (RWM) and methods of their suppression [77]. The general trend seems to be toward an increase of the plasma parameter $\beta = 2\mu_0\langle p\rangle / B_0^2$ ($\langle p\rangle$ is the average pressure, B_0 is the toroidal magnetic field, μ_0 is the vacuum permeability) and normalized parameter $\beta_N = a\beta / (B_0 I_p)$ (I_p is the plasma current, a is the minor radius) in modern tokamaks. The instability related to the RWM is one of the factors, which limits the growth of β and often leads to disruption of plasma discharge. Thus, the low toroidal modes n = 1, n is a toroidal wave number, which might arise with the growing pressure, are of particular interest from the viewpoint of control systems design (the modes of the form $\xi(r)e^{i(\omega t+m\varphi-n\theta)}$, m is the a poloidal wave number, are considered when examining the RWM, see § 5 of part 1 of this survey [1]). The tokamak conductive wall located close enough to the plasma could provide RWM modes suppression. However, it may only merely slow down the growth of the modes, since the wall with a finite conductivity could suppress the mode only for a certain time roughly the same as the time of magnetic field penetration into the wall. Therefore, these modes are called the Resistive Wall Modes.

The following system of equations describes the dynamics of RWM using a simple cylindrical model approximation and the assumption of a rigid mode structure and ignoring the effect of plasma rotation [79]:

$$L_{eff} I_p + M_{pw} I_w + M_{pc} I_c = 0,$$
$$M_{wp}\dot{I}_p + L_w \dot{I}_w + M_{wc}\dot{I}_c + R_w I_w = 0,$$
$$M_{cp}\dot{I}_p + M_{cw}\dot{I}_w + L_c\dot{I}_c + R_c I_c = V_c,$$

where I_p, I_w, I_c are the current on the plasma surface, the current in surrounding passive (wall) structures, and the control coil current, respectively, M_{ab} represents the mutual inductance between conductors a and b, R_a is the resistance in conductor a, L_a is the self inductance of conductor a, $a, b \in \{p, w, c\}$, L_{eff} is the effective self inductance, V_c is the voltage applied to the control coil. The model has been obtained for a tokamak where the control coils are inside the vacuum vessel (Fig. 28). The degree of interactions between currents on the plasma surface, in the surrounding wall, and control coil are characterized by mutual inductances M_{ab}.

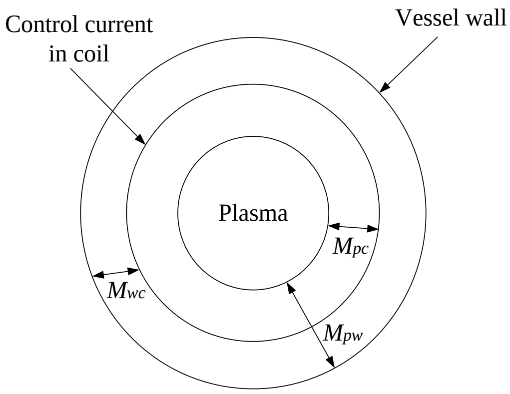


Fig. 28. Cross section of a cylindrical model of the RWM dynamics [78]

The interconnection of the plasma rotation and the appearance of RWM is not well studied, however, the model for describing this phenomenon was derived in [80] using an argument based on the exchange of energy between the plasma mode and external conductors:

$$\left(\delta W_{Iw} + i\Omega_{\phi} D\right) B_p = C_{pw} B_w,$$

where B_p, B_w are changes in the field at the plasma surface and at the vessel wall when the plasma is rotating; Ω_{ϕ} is the plasma toroidal rotation frequency; $C_{pw} = M_{pw}^{-1}$; δW_{Iw} is the value of the coupling of RWM energy transferred through the field B_p to the toroidally in phase component of the field B_w; $\Omega_{\varphi} D$ represents the energy coupled to the component B_w that is 90^{o} toroidally advanced, *D* represents dissipation mechanism.

There are two primary approaches to suppressing the RWM. *The first* one is that of applying feedback magnetic control systems in order to suppress the system's instabilities. The growth rate of oscillations decreases by an order of magnitude because of the tokamak vessel conductive wall, making it possible virtually to realize the feedback control by the control coils. *The second* approach is that of rotation stabilization of the plasma. In modern tokamaks a beam of neutral atoms is injected into the system, which transmits an angular momentum that is sufficient for keeping the plasma rotation, which in turn results in RWM suppression. The combination of these two approaches to control the RWM is considered as the most prospective for future thermonuclear reactors.

The RWM control systems [81] for the DIII-D tokamak (Fig. 29) [82] were created using different design methods of controllers. The feedback control underlies in most of the proposed methods. The structural scheme of the closed loop control system is presented in Fig. 30. The PD [83], LQG [83], $H_{\infty\mathrm{NCF}}$ [84], DK [84], etc. controllers are used as controllers in the feedback loop. The functional $J(u) = \int_0^\infty \left(x^T Q x + u^T R u\right) dt$, where $Q \geq 0,\ R > 0$ are weighting matrices, is minimized when designing an LQG optimal controller ($u = -Kx$ is the control law, x is the state). The Kalman filter (the observer) was applied to estimate the state [83]. DK-iteration also can be applied to synthesize the controller that is a combination of the H_∞-synthesis and μ-analysis. In this case the controller *K* meets the criterion $\min_K \left(\min_{D \in \wp} \left\| D N(K) D^{-1} \right\|_\infty \right)$, where

$N(K)$ is the transfer matrix of the closed loop system, $\wp$ is the set of matrices commuting with Δ ($\Delta = \delta I$, $\bar{\sigma}(\Delta) \leq 1$), $D\Delta = \Delta D$, where $\bar{\sigma}$ is the maximum singular value of the matrix. The results of numerical simulation when the RWM growth rate changes are presented in Fig. 31. It is shown that the LQG controller effectively suppresses the RWM and allows to increase the stability margin of the system [83, 84].

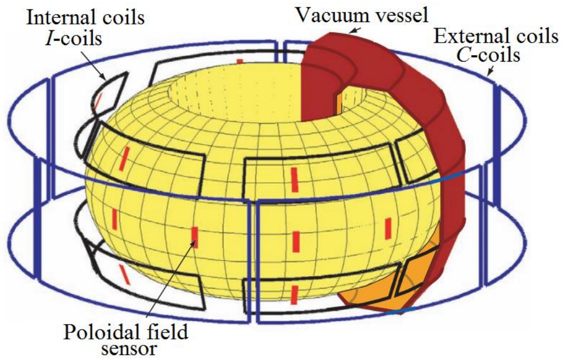


Fig. 29. DIII-D tokamak **[78]**

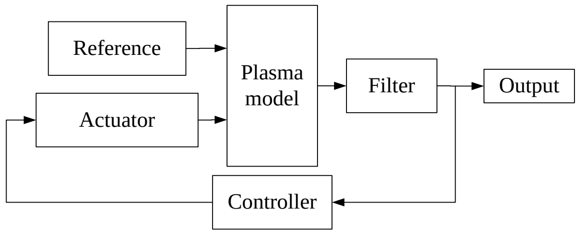


Fig. 30. Structural scheme of the RWM control system

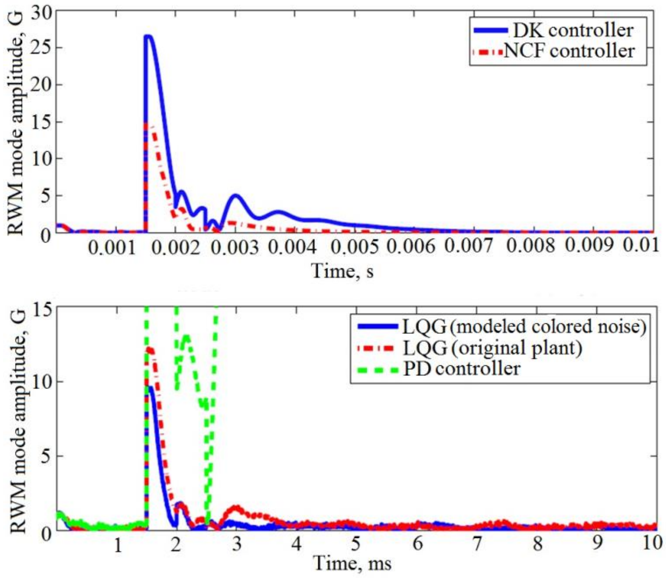


Fig. 31. The outputs of the RWM control system of the DIII-D tokamak when the RWM growth rate changes [83,84]

The tokamak DIII-D construction provides the neutral beam injectors (Fig. 32) which allow giving additional energy and momentum to the plasma in the direction opposite to its movement, or transmitting up to 10 MW of power without giving any additional momentum to the plasma [85]. The possibility of decoupling of the channels of transmission of power and momentum makes it possible to suppress the RWM and to slow down the plasma rotation considerably at $\beta_N > \beta_{N.no-wall}$, where $\beta_{N.no-wall}$ is the value of the β_N coefficient without the wall. The high values of β_N are also achieved by two series of control coils, located inside (*I*-coils) and outside (*C*-coils) of the vacuum vessel. Fig. 33, *a-e* shows that the RWM increase without the I-coils feedback (*I*-coils are switched off), which results in the plasma disruption. However, the RWM are successfully suppressed in case of the I-coils feedback (*I*-coils are switched on) (Fig. 33, *f-j*).

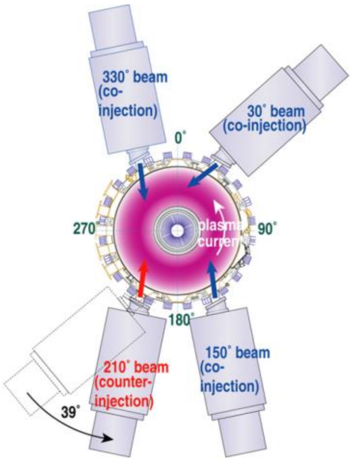


Fig. 32. The location of the neutral beam injectors in the DIII-D tokamak **[85]**

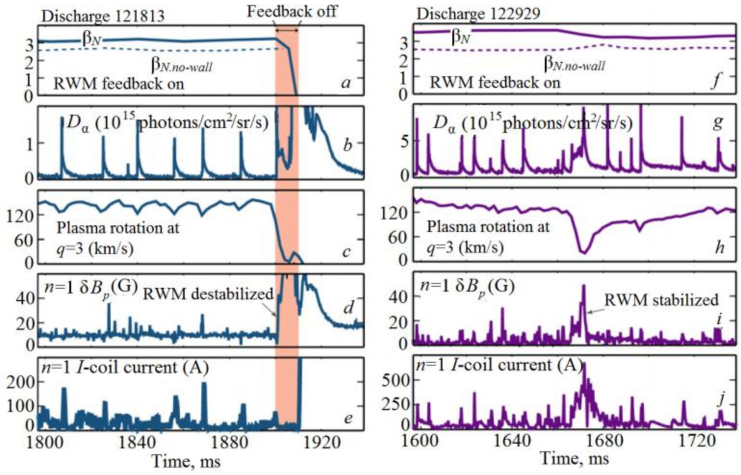


Fig. 33. A comparison of systems: a – e –without feedback control system (I-coils are switched off); f – j – with feedback control system (I-coils are switched on) [85]

On the other tokamaks, such as NSTX [82], ITER [83, 84] et al., similar principles are being applied to control the RWM taking into account their specific design features.

5.IMPLEMENTATION OF PLASMA CONTROL SYSTEMS

5.1.Real time test beds for tokamaks

Currently, the real time test beds are applied in various fields of engineering, industry, and science that allow to model control systems in real time, enable them to be tuned and debugged and then to be switched to a real controlled plant [89]. Such an approach is also applied to the tokamaks as real controlled plants (Fig. 34) [90]. For instance, this approach was applied to the Tuman-3 tokamak and was based on the analog-digital controllers and analog plant model [9, p. 204, Fig. 5.16]. A controlling computer on the DIII-D [91] and EAST [92] tokamaks enables switching from the plant model to the tokamak or vice versa through a corresponding switch. This allows saving time during physical experiment and thoroughly tuning plasma control systems in real time on the tokamak plasma models.

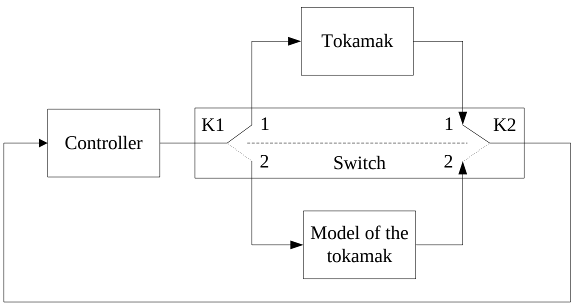


Fig. 34. The concept of the real time computer test bed: $K1$ и $K2$ are switches from the plant model to the tokamak and vice versa [9]

5.2.Software realization of plasma shape control system in the JET tokamak

The interface of the plasma shape control system in the JET tokamak [38] enables the user to divide the time of the experiment into segments, referred to as time windows, where it is possible to specify different combinations of the control regimes. For each time window the user may activate the specific controllers and set the programmed signals to track them. The user also can choose a Shape Control (SC) scenario containing the prepared set of controllers and programmed signals, the interface will adjust the signals shape for the specified initial and final values. The XSC scenarios became available when the Extreme Shape Controller [93] system had been implemented, allowing to specify the cross section of the plasma visually and interactively update the programmed currents and gaps signals. The difference between the defined values for adjacent windows may cause substantial spikes of signals during the transition. To exclude this effect, the interface automatically introduces the additional smoothing time windows for the smooth transition from measured values at the end of the first window to defined values at the beginning of the next window.

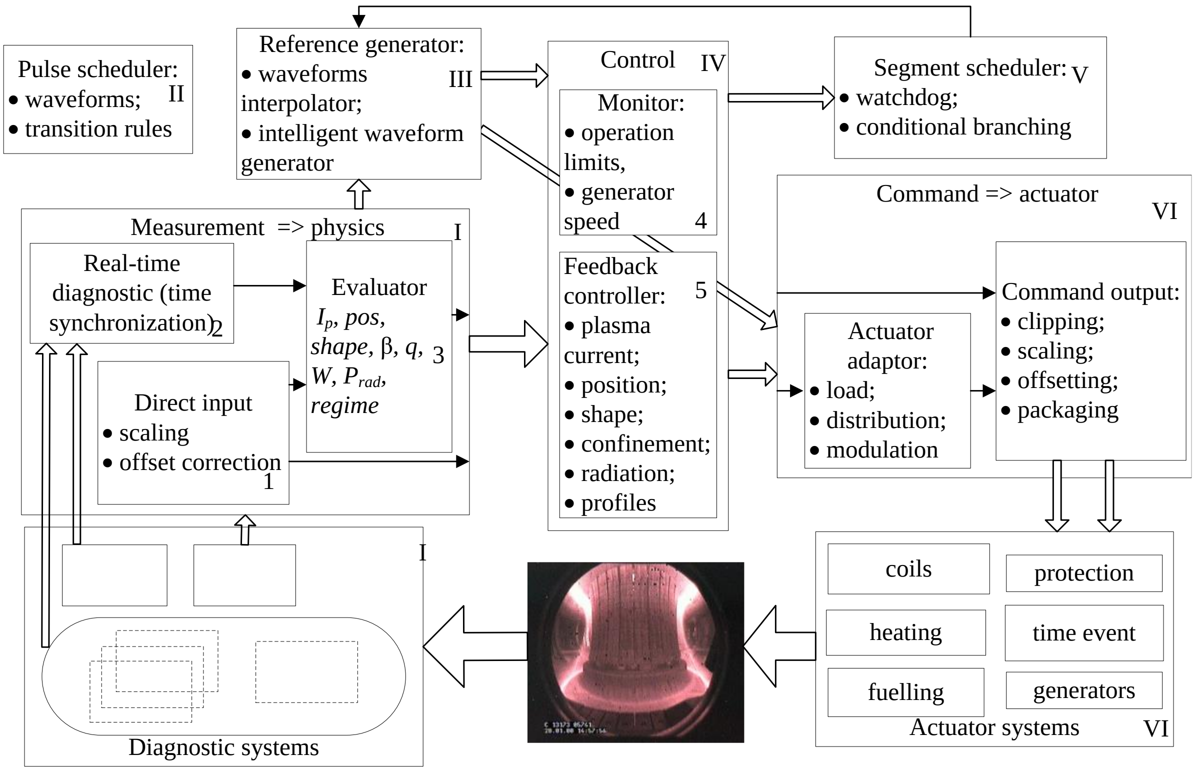


Fig. 35. General scheme of the digital plasma control system for the ASDEX Upgrade tokamak [94]

5.3.Plasma control system for the ASDEX Upgrade tokamak

The digital DCS (*Discharge Control System*) for the ASDEX Upgrade tokamak [94] (Fig. 35) comprises the following elements: diagnostic system *I*, which includes the system of direct input *1*, the real-time diagnostic *2* for time synchronization of the system, and evaluator for all measuring outputs *3*, the control unit *IV*, that contains monitor, checking the operation limits and generator speed *4*, and feedback control algorithms *5*, which allow controlling the outputs of the system through actuators *VI*, pulse scheduler *II*, reference generator *III*, and segment scheduler *V*, containing watchdog and conditional branching. The plasma current, position, and shape control systems, plasma magnetic confinement system, radiation and plasma profiles control systems are integrated into the DCS. The control actions are formed on the basis of the data of the diagnostic system. The control signals are fed to the actuators (control coils, additional heating system, fueling system, protection system, timing system of the events, generators), furthermore, a preliminary load, distribution and modulation and subsequently clipping, scaling, offsetting, and packaging of signals are carried out. The DCS provide the functional capabilities that allow one to guide the plasma discharges, coordinate measuring and actuating devices, and optimize the plasma parameters in the tokamak.

5.4.Plasma control system of the TCV tokamak

The closed-loop control system for the TCV (*Tokamak à Configuration Variable*) [95] tokamak is presented in Fig 36. The output signals from TCV tokamak (currents in coils, plasma density, magnetic characteristics, X-ray, etc.) come to the diagnostic system *1*, and then are archived in the TCV database, and come to the input of the control system. The control signals are formed in blocks SCD (*Système de Contrôle Distribué*) *2* and hybrid control system *3*, and are outputted to the programmable adder/switch *4*, and then through actuators *5* (*electron cyclotron* resonance heating (ECRH), *electron cyclotron* current drive (ECCD), toroidal (TF) and poloidal (PF) field coils, gas valve) are fed to the tokamak. The development of control systems for the TCV tokamak is carried out in a special interface of the SCD host computer, that has access to the TCV database, can receive and transmit signals to the control unit, and realizes cooperation among all blocks of the digital system through the TCVPC computer.

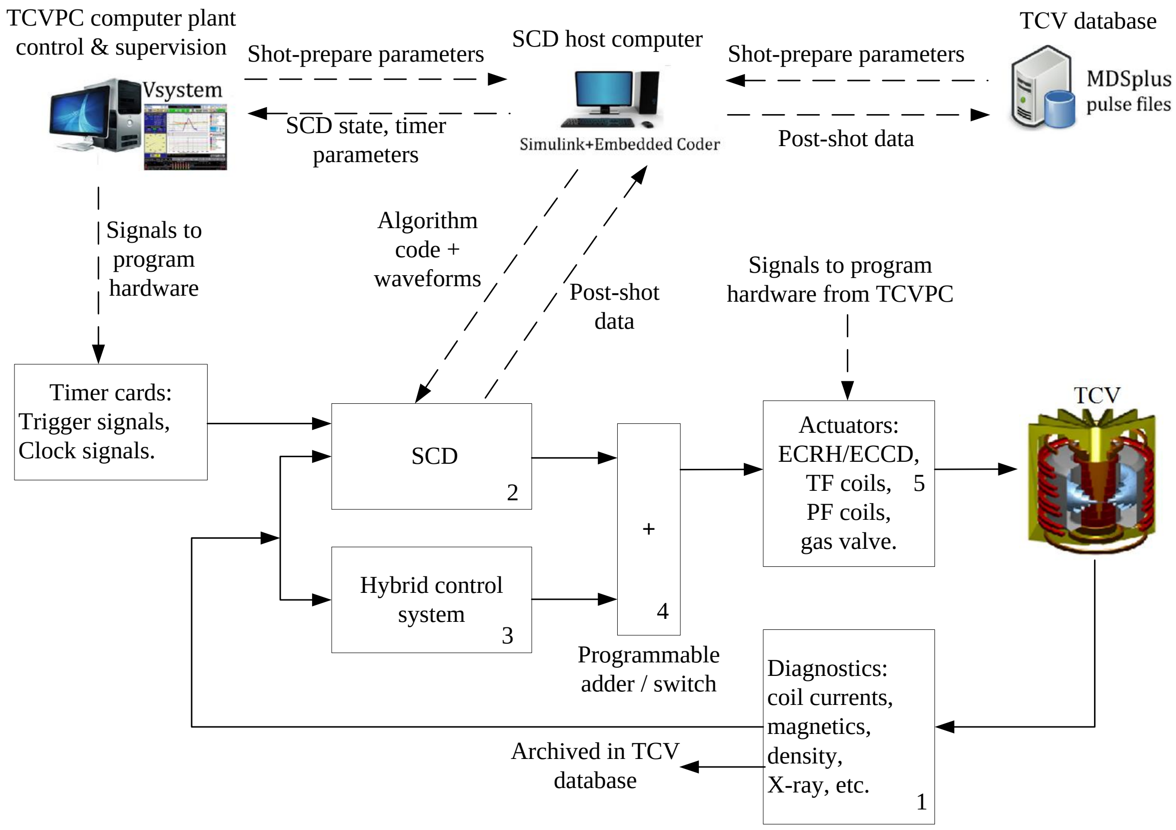


Fig 36. General scheme of the plasma control system for the TCV tokamak [95].

6.APPROACHES TO DEVELOPMENT OF MAGNETIC CONTROL SYSTEMS FOR PLASMA IN TOKAMAKS

The review of magnetic control systems that have been undertaken makes it possible to classify them into different groups depending on approaches to the plasma vertical position and shape control.

There are two approaches to plasma vertical movement control.

The stabilization of the plasma vertical speed around zero is the first one. This approach is being applied to many devices such as JET, DIII-D, EAST, KSTAR, NSTX, ITER project, where there are a wide range of plasma shape control systems. The stabilization of plasma vertical speed around zero makes it possible to avoid a contradiction with the plasma shape control problem, however, at the same time the system is not strictly stable and may have low stability margins. This approach also leads to the necessity of switching the controller in the stabilization loop of an unstable plant, since the plasma position stabilization system have to be applied before turning on the plasma shape control system (the stabilization of the plasma vertical speed around zero can be used only in combination with the plasma shape control system).

The stabilization of plasma vertical position is the second one. The plasma vertical position control is applied directly during the limiter phase of the discharge on the ASDEX Upgrade, JT-60SA, Globus-M, etc.

In some cases, there is a possibility to use control of the plasma shape during the divertor phase of the discharge without using the additional control loops for an unstable vertical position of plasma or plasma vertical speed around zero. Such technical solution was used in an earlier draft version of the ITER-1998 project [9] and during the analysis of such opportunity using mathematical modeling for the T-15 tokamak [75] that is under construction.

The values of controllable parameters in real time, which are impossible to obtain by direct measurement, are necessary to solve the plasma shape control problem. The parameters of plasma shape may be reconstructed from the magnetic measurements outside the plasma by indirect methods, which are, in fact, a complex computational task [1]. Today, there are a variety of approaches, using the values of different parameters of plasma, which are applied for plasma shape control in modern devices with a vertically elongated magnetic configuration.

The plasma shape control using the values of gaps between the separatrix and the first wall (gap control). This approach is being used in the JET, ASDEX Upgrade tokamaks, ITER project. The control of the gaps between the separatrix and the first wall (vacuum vessel or blanket) has direct physical significance since indeed the safety parameters, namely the distances between the plasma boundary and the first wall, are controlled. The disadvantages of this approach include the computational complexity of the algorithms of plasma equilibrium reconstruction and the difficulty of their implementation in real time.

The plasma shape control using the values of the magnetic flux in a set of points on the separatrix (isoflux control) is used in the DIII-D, TCV, EAST, KSTAR tokamaks. This approach lessens the computational complexity compared to the previous one since it applies the control on the basis of indirect parameters and the full plasma shape reconstruction is not required.

CONCLUSION

Control of plasma dynamics is one of the central fundamental problems of the theoretical and experimental study of thermonuclear fusion and the project of transition to thermonuclear energy. However, the control methods, especially with respect to the internal plasma parameters (see later the fourth part of this survey), remain insufficiently developed due to the complexity of the construction and the variety of tokamaks, the need to use complex mathematical models, and the solution of complicated ill-posed problems in plasma diagnostics (see the first part [1] of the review), the development of voluminous scienceintensive software and the use of high-performance computing. In practice, this leads to a long and costly work on the experimental selection of control system parameters and a large number of premature discharge disruptions during research and development campaigns. Therefore, the survey, systematization and classification of the real control systems for toroidal plasma are important and relevant.

The given overview of the magnetic control systems of the position, shape, and current of the plasma on modern vertically elongated tokamaks, also including spherical tokamaks, shows that the problem of selecting (developing) an efficient and reliable structure of the plasma magnetic control system has not been finally solved. In the world practice, there are competing approaches to vertical stabilization of the plasma and various approaches to controlling the shape of the plasma, which have their advantages and disadvantages that cause their use on specific devices. Hence, in particular, there is a lack of standards for the development of plasma control systems in tokamaks.

Nevertheless, it is possible to note the main trends both in the development of the tokamaks themselves and in the systems of magnetic control of the plasma in them. The vertically elongated tokamaks develop in the direction of decreasing the aspect ratio, which leads to spherical tokamaks, that have better physical and technical characteristics for the purposes of creation of future thermonuclear power plants based on tokamaks. In this regard, the work on the development, research, optimization, and simplification of the engineering implementation of plasma control systems is especially relevant for spherical tokamaks, in particular, primarily for the Russian spherical Globus-M2 tokamak, since it exceeds in the plasma parameters the known foreign analogs like MAST and NSTX (see subsection 3.4).

On the other hand, the analysis of plasma magnetic control systems done allows to mark out the general features of the configurations of such systems, namely:

- Multi Input Multi Output (MIMO);;
- multiple-loop;
- cascade control;
- hierarchy;
- robustness;
- adaptability.

A tendency has been outlined for the multivariable control of the plasma shape and current to decouple the control channels and to use the simplest PID controllers in them (see subsections 1.3, 1.4, 2.3-2.5). It should be noted that in addition to control systems for the plasma position, current, and shape, the development of control systems for the resistive wall modes takes place as well, which must be suppressed when the plasma density increases by means of special additional coils and feedback control. Similar properties and trends may appear in these systems.

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