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Plasma Magnetic Control Systems in D-shaped Tokamaks and Imitation Digital Computer Platform in Real Time for Controlling Plasma Current and Shape

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Abstract: The direction of developing plasma magnetic control systems in D-shaped tokamaks with its actuality and scientific novelty is presented. Shortly the poloidal systems of ITER and DEMO are characterized. The new imitation computer digital platform for modeling plasma magnetic control systems in real time for application in physical experiments is described. The preliminary results of testing the plasma magnetic control system for the Globus-M2 tokamak (Ioffe Institute, S-Petersburg, Russia) on the target computer of the platform are given. The digital twin of the imitation platform is characterized. The methodology of risk minimization is presented.

Keywords: tokamak, plasma control, imitation digital platform, real time, Globus-M2, ITER, DEMO

1. INTRODUCTION

The world challenge of the controlled thermonuclear fusion (CTF), researches on which have been begun in the beginning of 50th years of the last century, is one of central problems in science and engineering. The solution of this problem will open a new, safe, almost inexhaustible source of energy from the synthesis of nuclei of light elements, as well as eliminate carbon dioxide emissions into the atmosphere from the combustion of natural resources and stop climate change on the Earth in an unfavorable and dangerous direction for mankind.

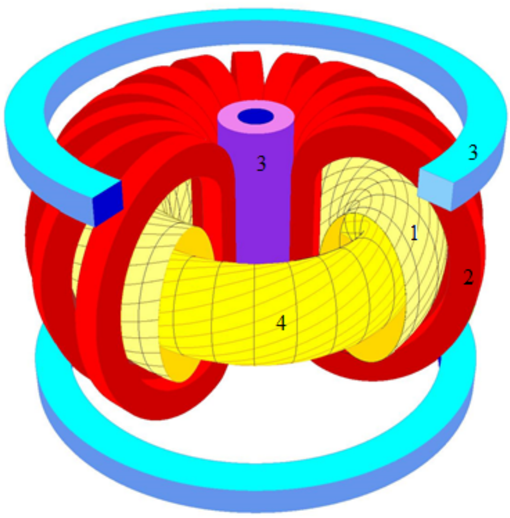

Fig. 1. Vertically elongated tokamak without an iron core: 1 is the vacuum vessel; 2 is the toroidal field coil; 3 is the poloidal field inner and outer coils; 4 is plasma and helical magnetic lines (© ITER Project Center, Russia)

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The most perspective direction of the CTF is tokamaks with D-shaped cross-section (Fig. 1), on the basis of which it is planned to create thermonuclear power plants. Confinement and heating a plasma in tokamaks to provide a self-sustaining thermonuclear reaction are ensured by *magnetic and kinetic plasma control systems with feedback*. The plasma in the magnetic field of a tokamak vacuum vessel represents extremely difficult for diagnostics and control a nonstationary nonlinear dynamical plant with distributed parameters, with structural and parametric uncertainties, subject to the influence of uncontrolled disturbances in the presence of broadband noises. That's why one of the main tasks of the CTF has not been solved yet specifically prolonged confinement of the high-temperature plasma in tokamaks with set parameters and prevention of plasma major disruptions.

The paper is structured as follows. Section 2 highlights the significance of plasma magnetic control systems in tokamaks. In Section 3 the scientific novelty of the direction presented in the paper is shown and concentrated on the imitation digital computer platform in real time for controlling plasma current and shape in tokamaks. Section 4 is devoted to some important features of tokamaks in whole, ITER, and DEMO. In Section 5 the short introduction into practice of physical experiments of the Globus-M2 tokamak of the digital plasma control system is given. Section 6 presents the basic new idea of the paper specifically the imitation digital computer platform structure in real time having new design for application to plasma control in tokamaks. In Section 7 the digital twin of the plasma magnetic control system is shown in the digital imitation platform. The results of the preliminary testing of one of the plasma current and shape control systems for the Globus-M2 tokamak on the target computer is given in Section 8. Section 9 describes the methodology of risk minimization of application of the direction suggested. The conclusion summarizes the main new features of the direction of the plasma control systems development.

2. SIGNIFICANCE OF PLASMA MAGNETIC CONTROL SYSTEMS IN TOKAMAKS

In this situation the urgency of continuing the development and research of magnetic plasma control systems in tokamaks only increases [2, 3]. It is explained first of all by the fact that the systems of magnetic plasma control have not yet reached a proper level of reliability and survivability for their application in thermonuclear reactors, where increased reliability of systems operation in conditions of presence of the thermonuclear reaction in the plasma of reactors is required. It is especially important for round-the-clock operation of thermonuclear power plants. Moreover, modern experiments on tokamaks require more and more advanced plasma control systems, which would allow deeper and more accurate study of plasma processes during generated discharges, could guarantee rejection of minor disruptions and also prevent major disruptions, so that they do not lead to emergency situations and destruction of fusion power plants. In doing so, at present nobody knows models of the plasma with the thermonuclear reaction and so nobody can guarantee the operability of the plasma magnetic control systems in thermonuclear reactors like ITER (International Thermonuclear Experimental Reactor) and DEMO (Demonstration Fusion Power Plant). Design of plasma magnetic control systems in tokamaks should have the feature of automatic adaptation to fusion plasma new models.

3. SCIENTIFIC NOVELTY OF DIRECTION OF CONTROL SYSTEMS DESIGN

Further development of the plasma magnetic control systems in the stated direction in the continuation of the achievements in the grant of the Russian Science Foundation № 17-19-01022 (2017-2019) “Hierarchical automatic control systems of plasma position, current, and shape in toroidal axial-symmetric magnetic configurations in a wide range of aspect ratios” will be carried out in real time with their high scientific novelty by the team of [4].

The *novelty* will be provided primarily by the originality of the imitation digital computer platform created in real time. Installation of new plasma control algorithms obtained on the digital twin of the control system is planned to be practiced over the Internet. An application for a patent of the Russian Federation is being made for this digital imitation platform.

The platform will allow setting up plasma control systems with an algorithm for plasma equilibrium reconstruction in real time. The Russian Federation patent for invention No. 2702137 was received in 2019 for the method of modeling such setting [5].

The industrial computer will allow replacing all analog controllers in the control loops of plasma position and currents in the coils of the central solenoid and poloidal field with digital controllers (8 controllers) on the Globus-M2 tokamak itself, and will also allow applying the plasma shape control system with feedback in experiments. Digitalization of the whole system of magnetic plasma control on the tokamak through the third platform computer will allow to set and solve a *new* task of optimization of joint coordinated operation of all plasma control circuits in real time. As it follows from the surveys on the systems of magnetic plasma control in tokamaks [6, 7], such a task was not set and was not solved by anyone. The solution of this problem is necessary for an optimal integration of the plasma magnetic control system with the kinetic control system of plasma parameters profiles (current density, safety factor q, density, temperature, pressure) in the long term.

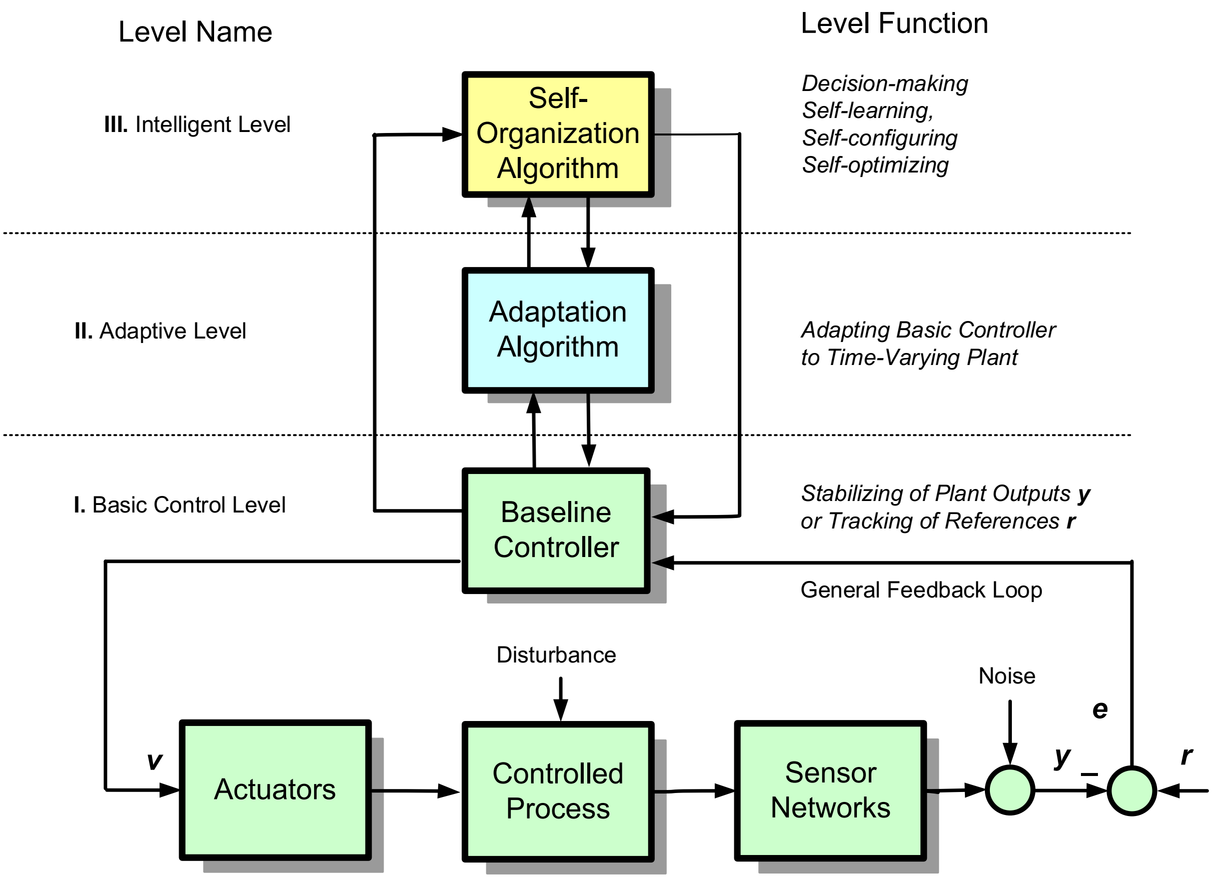


Fig. 2. General configuration of three levels hierarchical control system [8]

Novelty of the plasma control systems will be provided by the fact that the control systems will be developed in the class of digital *hierarchical* control systems (Fig. 2) for multivariable dynamic plants with time varying parameters [8]. This scientific direction allows to obtain various practically unlimited new combinations of different control and identification algorithms at different hierarchical levels.

The main directions in development of hierarchical plasma control systems will be as follows:

- identification (building models based on experimental data of observation of inputs and outputs of the controlled plant),
- robust control,
- adaptation,
- elements of artificial intelligence.

In the control directions it is supposed to be developed:

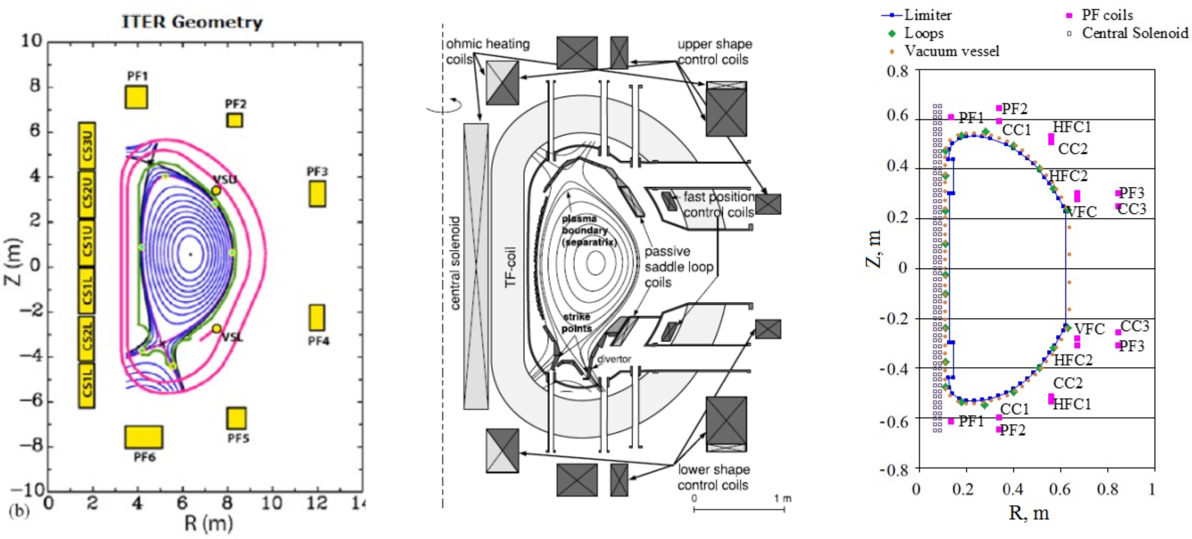
- control systems with a predictive model with adaptation of parameters to changing parameters of the controlled plant [9];
- robust systems with multivariable PID controllers adjusted by means of Linear Matrix Inequalities [10];
- decoupling of control channels with subsequent adjustment of control channels by means of the method of Quantitative Feedback Theory [11];
- decoupling of control channels by means of Relative Gain Array approach and H_∞-optimization theory [12];
- robust and adaptive systems with coordination of control loops of position and shape of the plasma [13];
- methods of Picard iterations, moving filaments, and neural networks for reconstruction of plasma equilibrium on the basis of the magnetic measurement outside the plasma [14-16].

At the same time, the development and application of new linear and non-linear plasma models will be continued. Linear models are designed for synthesis of linear magnetic plasma control systems without taking into account slow plasma processes related to the transfer of matter and energy. Non-linear models take into account the transfer processes and are aimed at future development and implementation of kinetic plasma control systems. Novelty of plasma control systems development directions of our team is confirmed by three reports at the IFAC World Congress in 2020 (IFAC WC 2020), which was held remotely in Berlin (Germany) [9, 10, 16].

4. TOKAMAKS, ITER, AND DEMO

Europe is a leader in solving the CTF problem. At the European tokamaks, in particular, JET (major radius R=3 m, England), ASDEX Upgrade (R=1.65 m, Germany), TCV (R=0.88 m, Switzerland), the experimental researches of the high-temperature plasma are intensively conducted, new plasma control systems are developed, applied, and investigated. In Europe, the first ITER thermonuclear reactor with R=6.2 m is under construction in Cadarache (France) [17, 18], where the assembly of the reactor itself has already been started.

There are two roadmaps for the creation of DEMO on tokamaks: (a) with large aspects ratio [19] and (b) spherical tokamaks [20, 21]. The first roadmap leads to the giant DEMO having the very large major radius of about 9-10 m. The second roadmap leads to the modular DEMO on spherical tokamaks as modules with the lower aspect ratio and relatively small major radius specifically less than 2 m. The cost of electricity from giant tokamaks should be very high but the cost of electricity of the modular DEMO is very competitive and is of about 0.06 $/kWh [21].



a b c

Fig. 3. Vertical cross-sections of (a) ITER [22], (b) ASDEX Upgrade [23], (c) Globus-M2 [24]

There are a set of poloidal systems (poloidal field coils around the vacuum vessel) for the giant DEMO as copies of the poloidal system of ITER with the rough mistake in the design of this system [22]. This original system led to the nonsense: at the major radius of ITER of 6.2 m the controllability region of the plasma unstable vertical position was about 3-4 cm. Because of that the internal extra coils to control plasma unstable vertical position have been installed into the vacuum vessel of the ITER tokamak (Fig. 3a) [22].

The engineering solution of including the PF-coils inside the vacuum vessel of DEMO is not the best idea from the following points of view:

- The thermonuclear plasma may burn these coils,
- The space for the plasma became less inside the vacuum vessel and the plasma is located farther from the first wall in comparison with tokamaks which do not have the PF-coils inside the vacuum vessel. This essentially decreases the effectiveness of acting on the plasma of the eddy currents induced in the first wall.

It is much effective and reliable to locate the PF-coils for the plasma vertical position control between the vacuum vessel and the toroidal field coil like for instance in the ASDEX Upgrade tokamak (Fig. 3b) [23] or design the whole poloidal system like in the spherical Globus-M2 tokamak (Fig. 3c) [24].

5. INTRODUCTION INTO PRACTICE OF PHYSICAL EXPERIMENTS OF THE GLOBUS-M2 TOKAMAK OF THE DIGITAL PLASMA CONTROL SYSTEM

The digital simulation platform will provide an opportunity for the first time to implement the developed new plasma magnetic control systems in real time in the physical experiments of the operating Globus-M2 spherical tokamak. Solution of off-line identification tasks will be carried out on the existing Globus-M2 tokamak equipment, which is planned in the direction for successful guaranteed application of new plasma control systems in the experiments.

The multivariable plasma magnetic control system which is operating at present on the spherical Globus-M2 tokamak in experiments is shown in Fig. 4a [13]. It contains 2 closed control loops for plasma position, 1 control loop for the current in the Central Solenoid (CS), and 5 control loops for the currents in the Poloidal Field (PF) coils. To generate plasma discharge scenarios the program reference signals are fed to these loops.

A set of plasma magnetic control systems for the Globus-M2 tokamak was developed and modelled in line with the new methodology presented in [25] for plasma current and shape control with feedback in the MATLAB/Simulink environment.

6. DESCRIPTION OF THE IMITATION DIGITAL PLATFORM

There is the basic approach to model plasma control systems in real time by means of test bed containing two target computers connected by feedback [26]. For control of plasma current and shape with feedback in real time on the Globus-M2 tokamak it is proposed to create and apply a *new* real-time imitation digital computer platform, which has a feedback bunch on two high-tech computers with high efficiency. These are target computers such as " plant simulator–controller", where the switching in the “controller” of the control algorithm from the internal model to the “plant simulator” and back takes place to configure the control system with subsequent transfer of a control algorithm through remote access to the target computer with its internal model connected to the tokamak (Fig. 4b).

The real-time platform realizes real-time plasma simulation in the tokamak with diagnostics, actuators, control loops for the horizontal and vertical position of the plasma and currents in the PF-coils as well as plasma current and shape loops on three Industrial Computers (IC) with communication devices with the controlled plant, which is the plasma in the tokamak.

The IC1 simulates only the plasma in the tokamak with acting control loops of plasma position and CS/PF-currents (Fig. 4a), while the other IC2 and IC3 simulate the plasma in the tokamak with acting control loops, and the plasma current and shape controller connected to each other by internal feedback with the plasma equilibrium reconstruction code at the controller input.

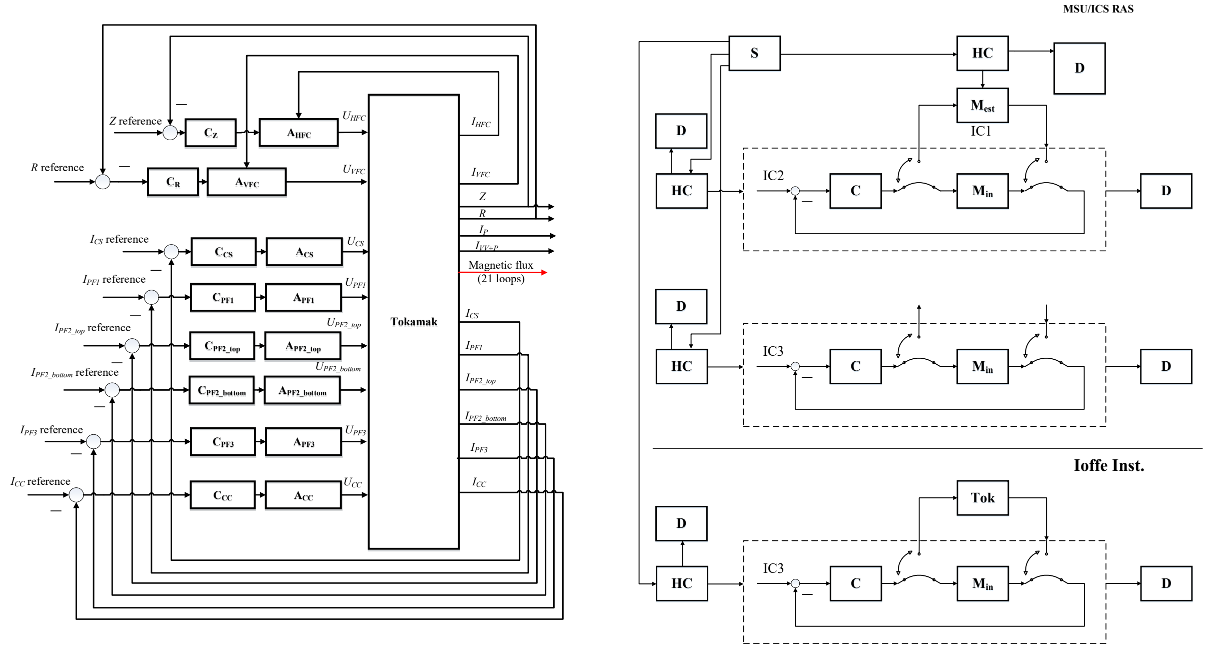
The IC2 and IC3 use two multivariate switches which switch the C controller of the IC2 from the plasma internal model (M_{in}) to the plasma external model (M_{ext}) or the tokamak and back to adjust the control system on the M_{in} and then apply it out on the M_{ext} or the tokamak. That allows to simulate by the C controller of the IC2 connected to the M_{ext} real physical experiments on the tokamak.

Host Computers (HC) for the development and modeling of plasma control systems in computer time with a display D for visualization of modeling of plasma control processes are connected to each of IC1, IC2, and IC3. These HCs are used to carry out loading from them the developed controllers and plasma models into IC1, IC2, and IC3. As well as displays D are connected to IC1, IC2, and IC3 to visualize internal control processes in real time.

A database Server (S) is connected to each of the three HCs, through which the data are downloaded for all HCs to develop plasma control systems in computer time and further to be downloaded to IC1, IC2, and IC3 for the purpose of modelling plasma control systems in real time in line with the simulation method with plasma reconstruction code in the feedback [5] and future plasma control in real time.

The IC3 with HC and D is installed and connected directly to the tokamak, the internal controller C with the reconstruction code at its input is set up on the M_{in} in real time, and then is switched to the tokamak. As a result, the controller C previously adjusted on the M_{in} controls the plasma current and shape in real time during plasma discharges.

For the progress of plasma control systems, the internal controller C in the IC3 may be changed by means of remote access to a new internal controller. Then this new controller is adjusted on the M_{in} and is switched to plasma control in the tokamak in real time by means of two multivariate switches.



a b

Fig. 4. (a) Block diagram of the plasma control system operating on the Globus-M2 tokamak: C_Z, C_R, C_{CS}, C_{PF1}, C_{PF2_top}, C_{PF2_bottom}, C_{PF3}, and C_{CC} are analogue controllers, A_{HFC}, A_{VFC}, A_{CS}, A_{PF1}, A_{PF2_top}, A_{PF2_bottom}, A_{PF3}, and A_{CC} are actuators [13]. (b) Structural scheme of the imitation computer platform: C is a controller, M_{in}, M_{out} are internal and external plant models, D is a display, HC is a host computer, IC is an industrial computer, S is a server, Tok is a tokamak

7. DIGITAL TWIN OF REAL CONTROL SYTEM IN IMITATION PLATFORM

Digital twins of control systems are being developed in industry when designing control systems because control processes can be simulated in real time on the digital twin, they can be optimized or new control algorithms can be synthesized and the optimal system results found are then applied to the real plant under control. This is much cheaper than solving the same problems at once on the real plant. At the same time, digital twins are also used to help to control complex dynamic plants, for example, in experimental physics, in order to find optimal control solutions for these plants independently of the physical experiments.

In the imitation platform (Fig. 4b) two industrial computers namely IC1 and IC2 constitute the digital twin and the third industrial computer IC3 is planned to be connected to the real plant specifically the Globus-M2 tokamak to organize the real plasma digital control system

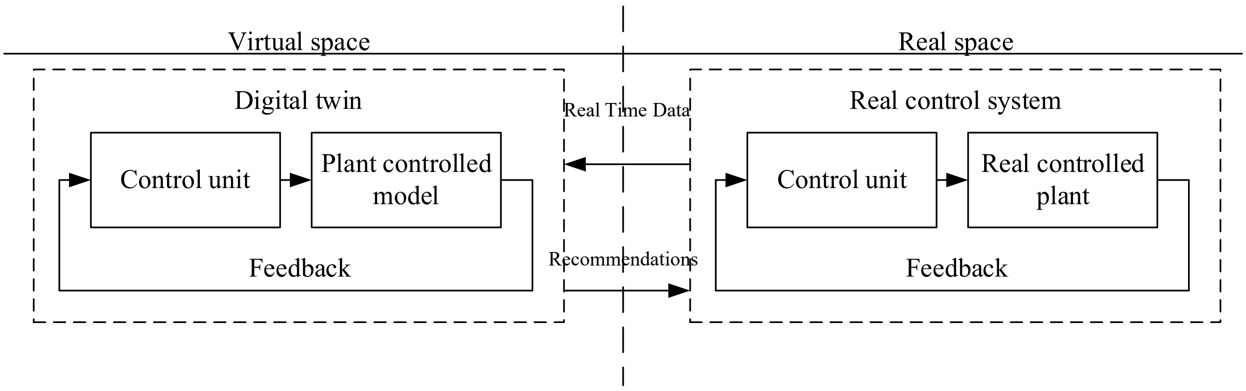


Fig. 5. Digital twin of feedback control system of real dynamical plant [IFAC 2020 WC]

8. PLASMA CONTROL SYSTEM TESTS IN REAL TIME

The preliminary tests of the control system of plasma current and shape as "isoflux control" (Fig. 6) [15, 27] on the target IC of Speedgoat type (www.speedgoat.com) with SimulinkRT operation system were carried out in real time, which showed sufficient speed of response for physical experiments on the Globus-M2 tokamak.

From the MHD theory of tokamak plasma configurations it is known that the magnetic field at the X-point for any discharge is zero and the plasma boundary is an isolevel line of the poloidal flux. Thus, by minimizing $|B_r|$, $|B_z|$ the X-point is placed at the desired location and by minimizing $|\delta\psi_1|, |\delta\psi_2|$ the plasma separatrix is placed at the points 1 and 2 (Fig. 7).

To test the operation of the control system in real time, the entire control system was transferred to discrete time and tested on the Performance real-time computer of Speedgoat (www.speedgoat.ch) with a processor Intel(R) Core(TM) i7-3770K CPU 3.50GHz. The main time step was 50 μs for all subsystems, except for the equilibrium reconstruction algorithm: for it the time step was 500 μs. As a result, a 50 μs task is performed on average in 24 μs and a 500 μs task is processed on average for 347 μs for 5 filaments. This shows that the system, taking into account computer power, is quite applicable in real experiments. The results of the simulation are shown in Figs. 8 and 9.

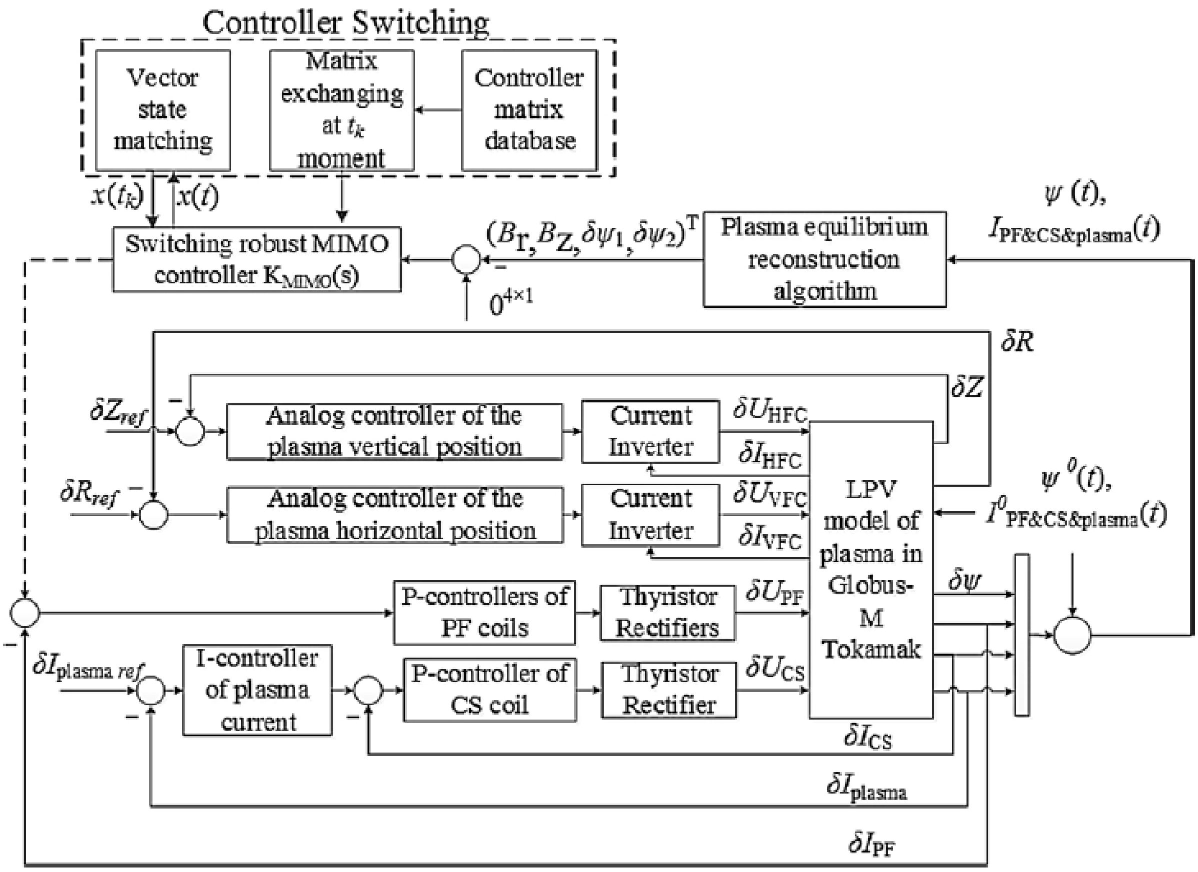


Fig. 6. Block diagram of multiloop and multivariable hierarchical plasma magnetic control system of the Globus-M2 tokamak [15]

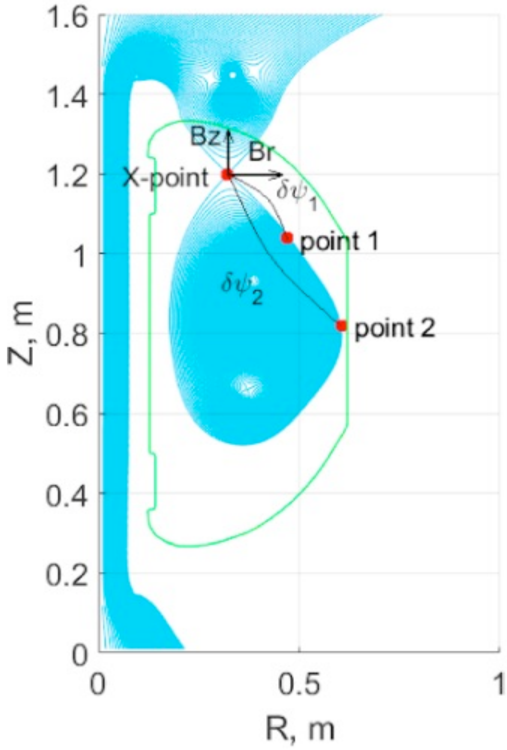


Fig. 7. The idea of the isoflux plasma shape control [15]

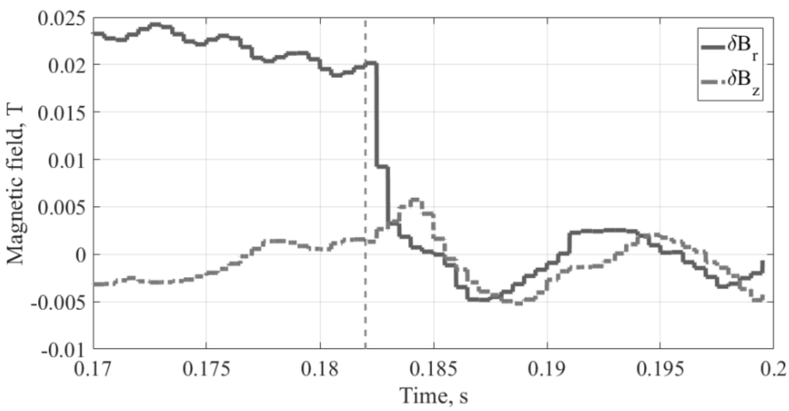


Fig. 8. Deviations of the magnetic field in the location of X-point for the shot № 31648 in discrete time [15]

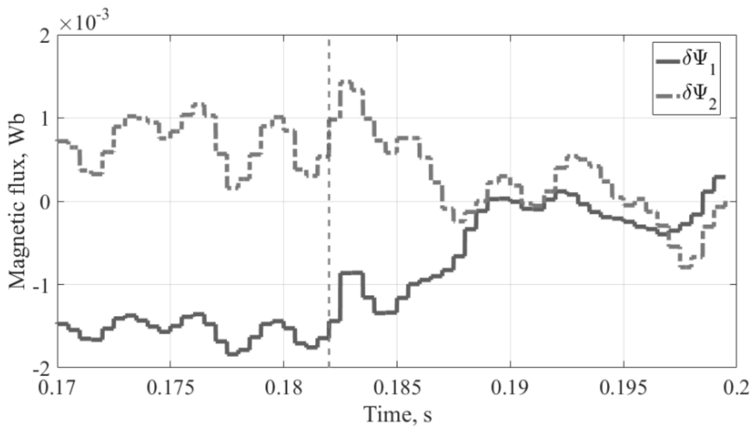


Fig. 9. Deviations of the flux in the desired location of the plasma separatrix (Fig. 4) for the shot № 31648 in discrete time [15]

9. METHODOLOGY OF RISK MINIMIZATION OF RESEARCH

Any new scientific work is subject to different kinds of risks, which may not lead to the desired result, especially as in this project, when the research is associated with a complex technique, a large field of research of a complex controlled plant, such as the plasma in tokamaks, the development of new methods and systems of plasma control, the introduction of real-time platforms in the computer network, organizational difficulties in solving problems with foreign partners, etc. Therefore, it is important to foresee in advance the possibility of appearance of different types of risks and methods of their minimization or complete elimination:

- ✓ When developing new tools, there may be the risk of their ineffectiveness or their application and refinement may take too long. This risk can be reduced by the creative abilities of team members. In this case, the most talented and persistent professionals get into the team who have abilities to minimize this risk.
- ✓ The risk that the purchased equipment in the form of Speedgoat Performance computers is not fast enough to solve the problems of magnetic plasma control in the Globus-M2 tokamak. In order to minimize this risk, it is necessary to continue the work on studying these computers. For this purpose, it is necessary to consider variants, when delays in communication devices with the plant can have even less delay, than 8 μs. It can be FPGA or use of the HDL Coder. This issue is part of the study for the applicability of plasma position control channels of the imitation platform. For plasma current and plasma shape control, these delays are quite small and do not interfere with the control process, as our already conducted studies show. In order to reduce the risk of applying real-time plasma control platform to control the plasma in the Globus-M2 tokamak, it is advisable to first purchase only two Speedgoat Performance computers, mount, configure, and test the platform on two "plant-controller" computers connected by the feedback, simulate on it the available plasma position, current, and shape control systems, to understand what needs to be changed in this platform (if necessary) in order to adapt it to solve plasma control tasks on the Globus-M2 tokamak. After that, one can purchase one Speedgoat Performance computer with corrected characteristics to be applied on the tokamak directly.
- ✓ The risk from withdrawal from the team of some of its members. Such risk exists, but it can be minimized by attracting the most talented young people to the team. For this purpose, the consortium has a leading higher education institution of Russia namely Lomonosov Moscow State University, which can solve this problem because it is the source of young talents in their specialty.
- ✓ The risk from unacceptable input-output controllability of the controlled plant [28]. It means that it is impossible to achieve the set control goals on this plant because of its properties, which were obtained when creating this plant. For example, such situation arose with ITER, when the poloidal system was already designed and the project development was in progress, it was found that at the major radius of tokamak of 6.2 meters, the vertical region of plasma controllability is only 3-4 cm. In this case it was necessary to change the controlled plant, which was done by introducing additional horizontal magnetic field coils inside the vacuum vessel. This led to expansion of the vertical controllability region by order of magnitude, which is enough for this machine. In the DEMO project such mistake should not be made: fusion electric power plant should work in twenty-four-hour mode and with increased reliability level, and presence of coils inside the vessel essentially reduces this reliability. Therefore, at the next stage in the world project it is proposed for DEMO, first of all, to develop a poloidal tokamak system, so that the controlled plant has an acceptable level of input-output controllability for the design of a sufficiently reliable control system of the plasma position, current, and shape.
- ✓ The risk from feedback controller synthesis, which does not provide sufficient stability margin and does not provide the required performance. In this case, to minimize this risk, the developers of the control system should first analyze the sources of poor operability of the controller. Such sources may be the following:

- insufficient accuracy of the plant model on which the control system is synthesized;
- insufficient accuracy of reduction of either the plant model or of the controller itself;
- failure to take into account all the properties of the plant during synthesis of the controller, which affect its operability, for example, failure to take into account all disturbances, which affect the controlled plant, plant varying parameters, nonlinearities, saturated input and output signals, etc.

In this case, the developers must identify the cause of the unsuccessful synthesis of the controller, redesign the controller, again simulate the entire system. Synthesis of the system should be iterative and cover not only modeling, but also application to the controlled plant. Fig. 10 shows the controller design cycle, which covers a number of stages [26].

In general, it should be noted that in thermonuclear tokamak reactors an explosion is impossible, because in case of emergency it is possible to close the valve of gas supply to the vacuum vessel and the process development will stop. Therefore, fusion electric power plants are safe sources of energy in comparison with nuclear power plants.

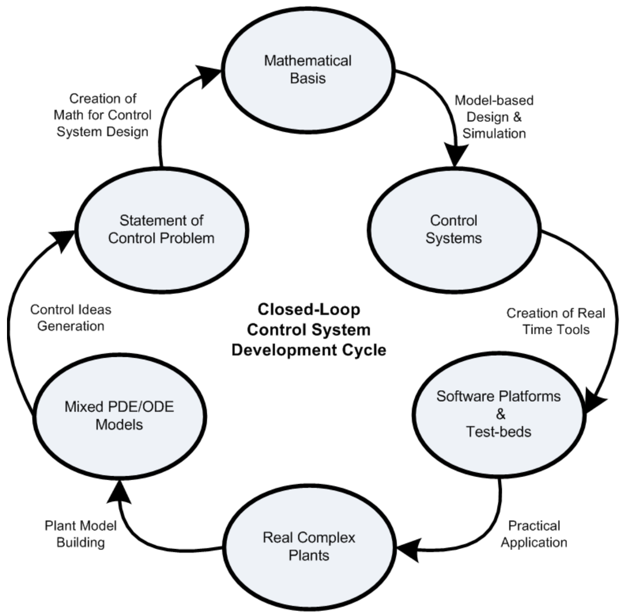


Fig. 10. Design cycle of dynamic plant control systems to achieve the required stability margins and performance of systems [26]

10. CONCLUSION

As follows from the indicated novelty of the direction and plenary reports at the IFAC 2020 World Congress, the magnetic plasma control systems in the Globus-M2 tokamak will be developed in accordance with the world trends in the development of automatic control systems:

- digitalization, automation, and elements of artificial intelligence lead to the autonomy of systems, which may give the human independent systems. The direction involves the use of artificial neural networks to reconstruct the plasma equilibrium, which are elements of artificial intelligence;
- creation of digital twins of control systems to be created on the digital simulation platform;
- the increasingly widespread use of machine learning, since the amount of information processed, is constantly increasing, for example, in ITER, the CODAC information management system will process about 1 million signals. In this direction, similar methods will be used to

configure neural networks both on evolutionary plasma models and according to the data from real experiments.

Nobody knows the models of the plasma with thermonuclear reaction and so nobody can guarantee the operability of any known plasma magnetic control system. So, one should design plasma magnetic control systems for ITER and DEMO having a feature of automatic adaptation to a fusion plasma new model. The design of such control systems is in line with our basic research direction which corresponds to the world development of control systems: combination of automation, digitalization, artificial intelligence giving autonomy.

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