Adv Syst Sci Appl 2020; **03**; 136-152
Published online at https://ijassa.ipu.ru.

Plasma Control in Tokamaks. Part 3.2. Simulation and Realization of Plasma Control Systems in ITER and Constructions of DEMO

Y. V. Mitrishkin [1,2*], N. M. Kartsev [2], A. E. Konkov [1,2], M. I. Patrov [3]

[1)] *M.V. Lomonosov Moscow State University, Faculty of Physics, Moscow, Russia*

[2)] *V.A. Trapeznikov Institute of Control Sciences of Russian Academy of Sciences, Moscow, Russia*

[3)] *A.F. Ioffe Institute of Russian Academy of Sciences, St. Petersburg, Russia*

E-mail: y_mitrishkin@hotmail.com

E-mail: n.kartsev@yandex.ru

E-mail: konkov@phys.msu.ru

E-mail: michael.patrov@mail.ioffe.ru

Abstract: Experimental testing of ITER scenarios on the DIII-D (US) and WEST (France) tokamaks, approaches for simulation and realization of ITER plasma control systems, the preparation of ITER plasma control systems for starting and exploitation are presented. The roadmaps that are known in Europe for the development and creation of the first fusion power plant DEMO (the next step after ITER) are reported. These maps show two directions of DEMO development: (i) based on conventional tokamaks with relatively large aspect ratio and (ii) based on spherical tokamaks of module type giving a chance to noticeably reduce the time needed to create the DEMO and to get cheap competitive electrical energy. The basic trends in the design of DEMO poloidal systems, as well as the initial version of the DEMO plasma vertical position control system, are given.

Keywords: tokamak, plasma, plasma magnetic control, ITER, DEMO constructions

INTRODUCTION

In the first part of the survey [1], tokamaks with plasmas in their magnetic fields together with diagnostics and actuators as controlled plants are considered. The second part [2] presents systems for the magnetic control of the position, current, and shape of the plasma, as well as resistive wall-modes in operating tokamaks. Part 3.1 [3] shows the magnetic plasma control systems for ITER (International Thermonuclear Experimental Reactor). The systems include original engineering solutions for plasma position, current and shape control systems for the two versions of ITER (ITER-1 and ITER-2) including those proposed and made in the Institute of Control Sciences of RAS. It was noted that in ITER-1 the position and shape of the plasma were controlled by all PF-coils and robust H_∞-controllers, and an additional non-linear control circuit was designed to reduce the peaks of control power during the suppression of minor disruptions. Due to the reduction of the large installation radius from 8.1 m to 6.2 m, a special control circuit with a relatively fast actuator connected to the PF2-PF5 coils was used for vertical plasma stabilization in ITER-2. But at the same time, the vertical control area of the plasma turned out to be a catastrophically small scale of 3-4 cm with a minor radius of ITER-2 equal to 2 m. In order to increase the controllable area of the ITER-2 project, additional

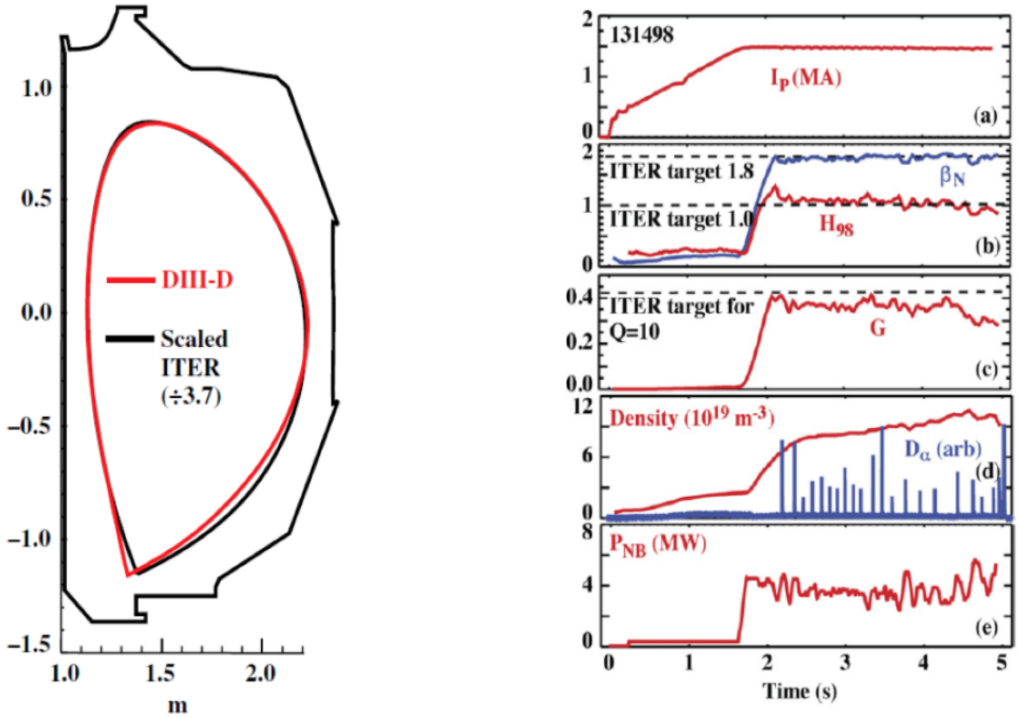
[*] Corresponding author: y_mitrishkin@hotmail.com

horizontal field coils were introduced into the vacuum vessel of the tokamak, which made it possible to increase the controllable area by about one order of magnitude. Modeling has shown that the best result in suppressing vertical instability of the plasma is achieved by combining the external and internal coils of the poloidal field.

In Section 1 of this part of the survey, the experimental development of scenarios for ITER-2 on two operating tokamaks, DIII-D and WEST, is considered. All 4 planned ITER-2 scenarios with a scale factor of 3.7 were tested on DIII-D, and a tungsten diverter plate for ITER-2 with a discharge duration of 1000 s will be studied on the WEST tokamak. In Section 2 there is information about the algorithm of the development of the plasma control systems in ITER-2, about the software-computing platform, about the real-time testbed for the implementation of the algorithms of plasma control, about the information-control system CODAC (Control, Data Access and Communication) for ITER-2, about the scheme of the interfaces of the plasma control systems with CODAC, about the simulator of the plasma control systems, and about the developed software package IMAS: The ITER Integrated Modelling & Analysis Suite. Section 3 lists the measures taken to prepare the plasma control system in ITER-2 for start-up and operation. The first roadmap shows the way to create a DEMO for traditional tokamaks with a relatively large aspect ratio of 3-4, and the second roadmap shows the prospect of faster creation of the DEMO on modular spherical tokamaks with an aspect ratio of 1.5-1.7 and cheaper electricity. This part concludes with a brief overview (Section 5) of the planned DEMO designs and with information on the DEMO vertical plasma position control system for one of the constructions listed above.

1. EXPERIMENTAL DEVELOPMENT OF SCENARIOS FOR ITER-2

The ITER International Consortium attaches special importance to the development of scenarios for ITER not only based on plasma-physical codes by numerical simulation [1, *issues 4.2.1* and *4.2.2*] but also on experiments on operating tokamaks even before starting ITER and bringing it to the nominal operating modes. Four ITER operating scenarios were tested on the DIII-D tokamak in a scaled form [4]. The unique properties of this work are that the plasma parameters reflect the essential properties of the ITER scenarios and the expected performance characteristics, such as the plasma vertical section and aspect ratio in the DIII-D discharges, correspond to the ITER project with a reduction factor of 3.7 (Fig. 1.1). Key aspects of all four scenarios, such as β_N and H_{98}, have been successfully imitated on DIII-D, providing an improved and unified physical basis for transport and stable modeling as well as for extrapolation to the ITER operating modes. (The value $\beta_N = \beta a B_T / I_p$ is a normalized β value, where a is the minor radius, B_T is the toroidal field induction, I_p is the plasma current, $\beta = \langle p \rangle / (B^2 / 2\mu_0)$ is the ratio of the plasma gas kinetic pressure to the external magnetic field pressure, $\langle p \rangle$ is the average plasma pressure, B is the average value of the full field, $H = \tau_E / \tau_E^L$ is the confinement factor, where τ_E is the confinement time, τ_E^L is the confinement time in L-mode, $\tau_E = \frac{3}{2P} \int n(T_i + T_e) d^3x$ where P is the full input power, n is the plasma density, while T_i and T_e are the ion and electronic temperatures). An example of such a discharge for the ITER base case scenario is shown in Fig. 1.1, b. In all four scenarios, the normalized quality of control coincided or was close to achieving the required one to realize the physical and technological goals of ITER, and the DIII-D discharge displays were coordinated with ITER, reaching its goal of generating at least 400 MW of thermonuclear power and $Q \geq 10$, where $Q = P_{out}/P_{in}$ is the ratio of the output power of thermonuclear fusion P_{out} to the input power of P_{in}. These studies are also related to many key physical issues concerning the ITER project, including the L-H power threshold crossing, the level of edge localized modes, the scaling of parameters at the flat discharge phase, the effect of tearing modes on confinement and disruptions, the limitation on β and the required capabilities of the plasma control system. An example of the direct impact of this work on the ITER project is the modification of the physical requirements for the set of poloidal field coils at the 15 MA plasma current, based on observations that the internal inductance in the basic scenario changes to a level that lies outside the original ITER specification.



a b

Fig. 1.1. a) Comparison of scaled (reduced by a factor of 3.7) ITER plasma cross-section (black) and experimental DIII-D plasma cross-section (red/grey). b) Time evolution of key plasma parameters of a baseline scenario demonstration discharge, operating at a normalized plasma current equivalent to 15 MA on ITER (131498). Illustrated are (a) plasma current I_p, (b) normalized beta β_N and confinement factor H_{98} with ITER target values, (c) fusion performance factor $G = \beta_N H_{98} / q_{95}$ with target value for Q = 10 operation on ITER indicated, (d) line average electron density and divertor D_α emission D_α, indicating ELM timing, (e) neutron injection beam power P_{NB}

The closest scenarios for ITER in the experiment are supposed to be achieved on the WEST tokamak (W for tungsten Environment in Steady-state Tokamak), which is an upgrade of the Tora Supra tokamak (France, Cadarache), which has a round cross-section. In the WEST tokamak, inside the circular vacuum vessel, there are the coils of the poloidal magnetic field, allowing the achievement of a diverter configuration of the plasma (Fig. 1.2). The main purpose of the WEST tokamak is to investigate the tungsten diverter plate for ITER on long plasma discharges in ITER scenarios, which will be created due to the superconducting coils of the WEST tokamak. The WEST tokamak parameters are as follows: plasma current I_p = 1 MA, toroidal field BT = 3.7 T, major radius R = 2.5 m, minor radius a = 0.5 m, aspect ratio A = 5-6, elongation k = 1.3-1.8, triangularity δ= 0.5-0.6, and the discharge time at a plasma current of 0.8 MA is t_{flattop} = 1000 s.

In the WEST tokamak, the first plasma was obtained in December 2016 with the new control system based on the ASDEX Upgrade real-time discharge control system and adapted to the specific needs of the WEST tokamak. Currently, the development phase of the WEST project is coming to an end, and the first operational version is constantly being used in experiments [6, 7].

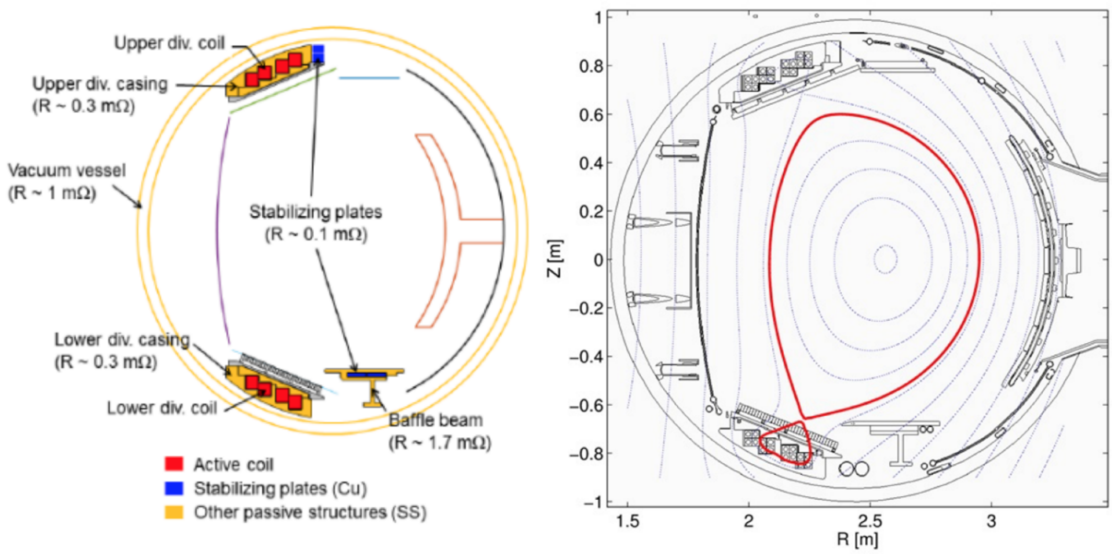


Fig. 1.2. Cross-section of the WEST tokamak: (*a*) design [7], (*b*) magnetic configuration of the plasma [8]

2. APPROACHES IN MODELING AND IMPLEMENTING PLASMA CONTROL SYSTEMS IN ITER

Development of algorithms for plasma control systems. Fig. 2.1 shows the scheme of the development of plasma control system algorithms which has been used at the V.A. Trapeznikov Institute of Control Sciences of the Russian Academy of Sciences for the development of magnetic plasma control systems in ITER. The main model of the controlled plant was a non-linear DINA code [9], in which the plasma model in ITER was implemented and plasma discharge scenarios were simulated. The necessary plasma parameters were loaded into the code with the ARM. With the help of the code, linear plasma models were obtained by means of its linearization [10, 11] or identification. Then linear models were used for synthesis of a plasma vertical instability suppression controller as well as a plasma current and shape controller. The obtained controllers were used for mathematical modeling of the control processes for the plasma position, current and shape in a closed-loop control system on linear plasma models and DINA code.

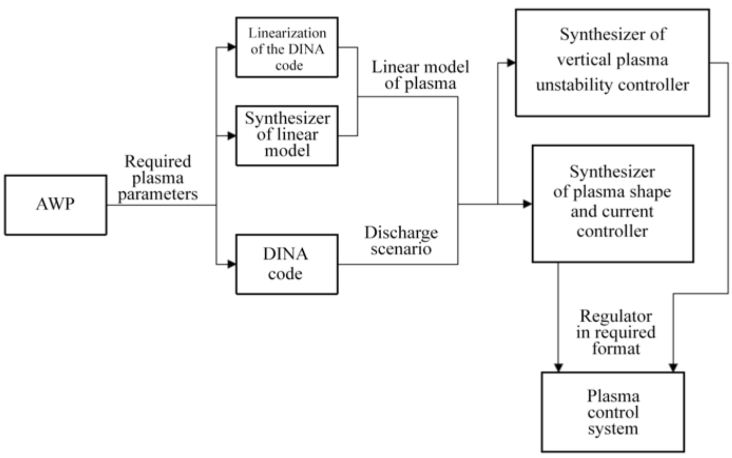


Fig. 2.1. Development of plasma control systems for ITER: AWP is an automated workplace

Software and computing platform. For the mathematical modeling of the systems of magnetic plasma control at the V.A. Trapeznikov Institute of Control Sciences of the Russian

Academy of Sciences, there was created a software and computing platform based on the DINA plasma code and MATLAB/Simulink environment [12]. The purpose of the platform is to organize a convenient software tool for numerical experiments on linear and nonlinear plasma models, to implement the possibility of transition from one model to another without changing the control system, and the subsequent analysis of the results of the modeling. The developed software and computing platform allows

- loading into the MATLAB workspace all the variables necessary for the simulation (more than two hundred);
- switching the operation of the control system from a linear model to a non-linear model and back by a special switch;
- carrying out modeling in batch mode (sequentially with different parameter values), for example, to search for the best controller settings;
- in the process of modeling, saving all registered signals in the variables of the MATLAB workspace;
- saving the simulation results in a database, viewing them in the form of graphs, and comparing the results of various experiments by superimposing them on each other.

The platform is a technical tool to improve the efficiency of scientific work and allows more productive research into various approaches to plasma control in the ITER reactor tokamak, as well as comparing the results obtained to make a decision on the best plasma control methods. The platform in the MATLAB environment enables the implementation of developed control systems in the CODAC control complex of the ITER reactor tokamak. The platform includes:

- a set of MATLAB *m-files*,
- Simulink models, which are built on a block principle and include subsystems implementing plasma models in the tokamak, actuators, diagnostics, and controllers,
- DINA code files,
- as well as files that store parameters from the ITER database, discharge scenario, and other data.

Real time testbed. Nowadays it is generally recognized that experimental real-time testbeds are necessary for the application of control methods and systems on real dynamic plants. These include computer testbeds capable of operating in a Hardware In the Loop Simulation (HIL) mode [13]. Such a testbed consists of a plant model and controller, each of which is implemented on its own industrial computer (Fig. 2.2, *a*). These computers form a feedback system, the signals in which are circulating in real time. Data from the external computer of the automated workplace.} are loaded to the industrial computers in real time. Such a testbed, in particular, is used on the DIII-D tokamak [14].

A real-time testbed for the plasma position, current and shape control system in ITER has been developed, created and tested at the V.A. Trapeznikov Institute [15]. Fig. 2.2, *b* shows the real time operation of the plasma model in ITER realized on the computer, which is controlled by another computer in real time under the SimulinkRT operating system from the MathWorks company.

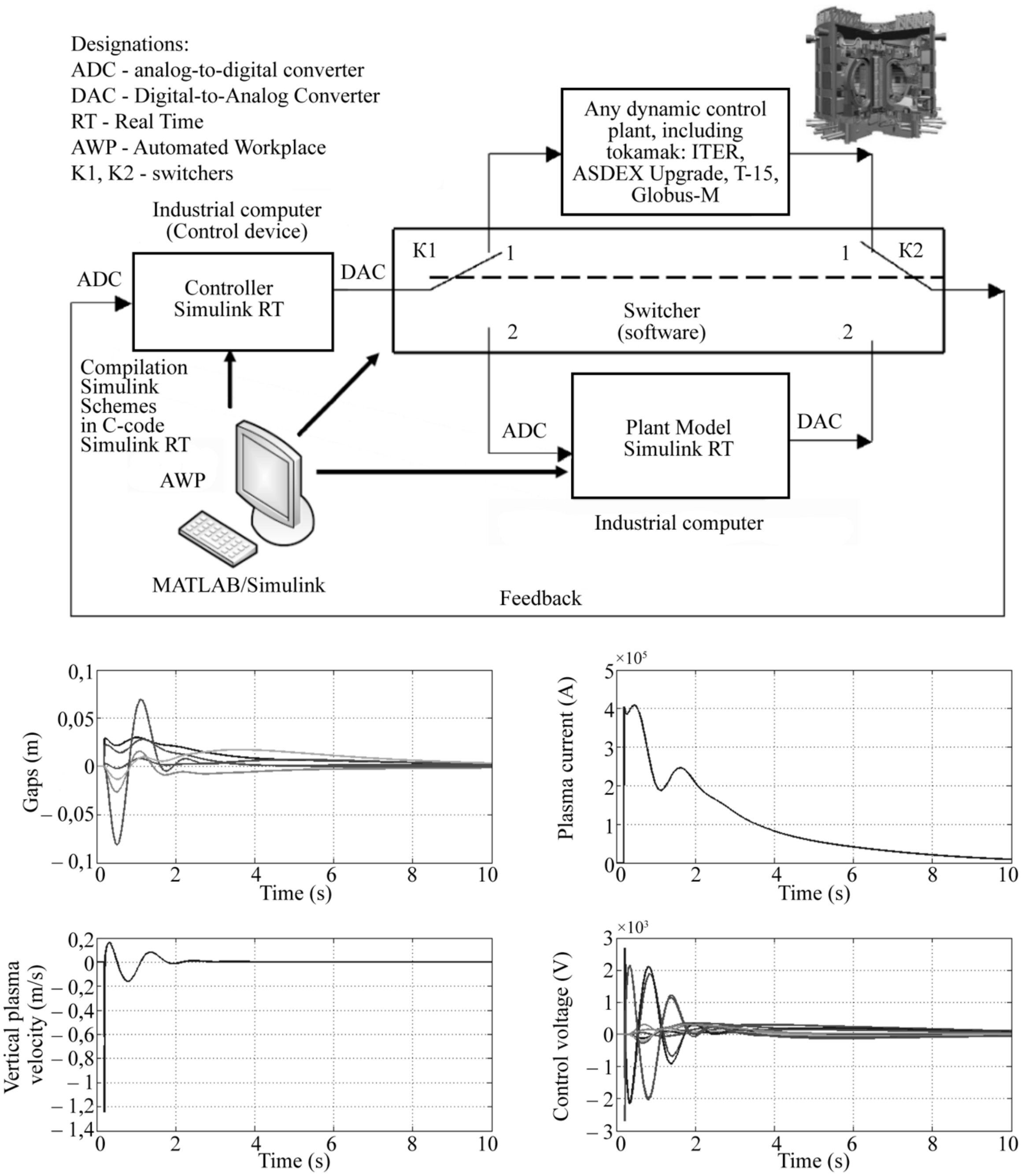


Fig. 2.2. Real-time testbed: (*a*) testbed structure; (*b*) real-time results of the plasma model in ITER when controlling the model by the real-time controller at minor disruption (from right to left and from top to bottom): gaps, plasma current, vertical position of the plasma, control voltages

The ITER information-control system. All information and control systems in ITER, including the plasma control systems, are integrated into a common system called ITER CODAC (Control, Data Access and Communication) [16, 17]. The purpose of the common system is to integrate more than 30 subsystems, based on the open software platform EPICS (Experimental Physics and Industrial Control System) [18], which provides flexible interfaces for components and the principle of their description SDD (Self-Description Data). The ITER CODAC system is designed to process about *one million signals* in an united system.

Different CODAC subsystems, which provide the operation of the installation, contain specific local devices through which the logic is formed and the control and data acquisition is performed. Such devices include programmable logic controllers PLC (Programmable Logic Controller), in the ITER project the models of the Siemens SIMATIC S7 line are used [19], as well as "fast controllers" with input/output through the PCIe bus, which provide fast processing of signals at the control rate; in the project, the devices of National Instruments company are accepted [20]. Each subsystem also includes a PSH (Plant System Host), providing the standard functions inherent in all elements of CODAC.

The following functions for integration of subsystems are implemented in the system:

- monitoring the status of the PSH host and the state of the "fast controllers";
- a time synchronization service, which is a subnet for the transmission of accurate time signals, namely, TCN (Time Communication Network) according to the standard IEEE-1588 Precision Time Protocol (PTP), using devices NI PXI-6683H;
- a synchronous data service that is a subnet for real-time data transmission, specifically, SDN (Synchronous Databus Network) to ensure data exchange between all elements of the system;
- data archiving: a high-performance data transmission subnet for DAN (Data Archiving Network) archiving and visualization;
- a standard state machine for COS (Common Operating State) subsystems provides control over system elements by reading the states and requesting transitions between them.

The subsystems are connected via a PON (Plant Operation Network) control subnet to the CODAC servers that provide the monitoring, control, and processing of the upper level data, as well as to the human-machine interface of the locking and safety systems.

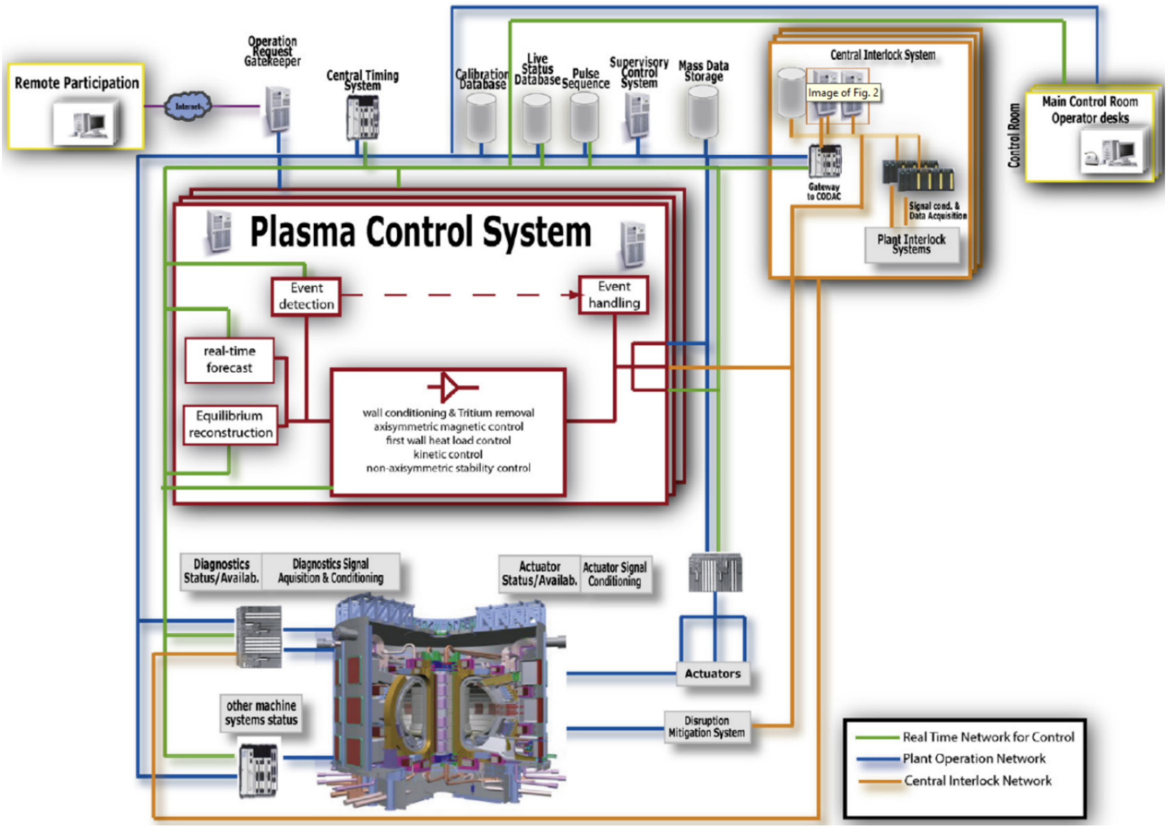


Fig. 2.3. Scheme of interfaces between plasma control systems and CODAC [21]

Plasma control systems are included as components in ITER CODAC, but can be implemented on specialized computing facilities, which allows the use of virtually any modern information processing technology, such as calculations on a GPU (Graphics Processing Unit), on multi-core multiprocessor systems, and programmable logic integrated circuits.

In contrast to diagnostic subsystems and actuators, plasma control systems use a wide range of CODAC system services: time synchronization, real-time data access, data archiving and lockout and security systems. A scheme of the interfaces for the interaction of the plasma control systems with CODAC is shown in Fig. 2.3 [21].

The huge number of requirements and tasks in the project have created the need to develop modeling tools. A joint team of specialists from General Atomics, IPP Garching and the CREATE group developed the PCSSP (Plasma Control System Simulation Platform) within the CODAC system, which provides:

- development of plasma control systems,
- creation and validation of discharge scenarios,
- analysis and troubleshooting,
- support for development and modification of the system as a whole.

A functional diagram of the plasma control system simulator in PCSSP is shown in Fig. 2.4 [22, 23].

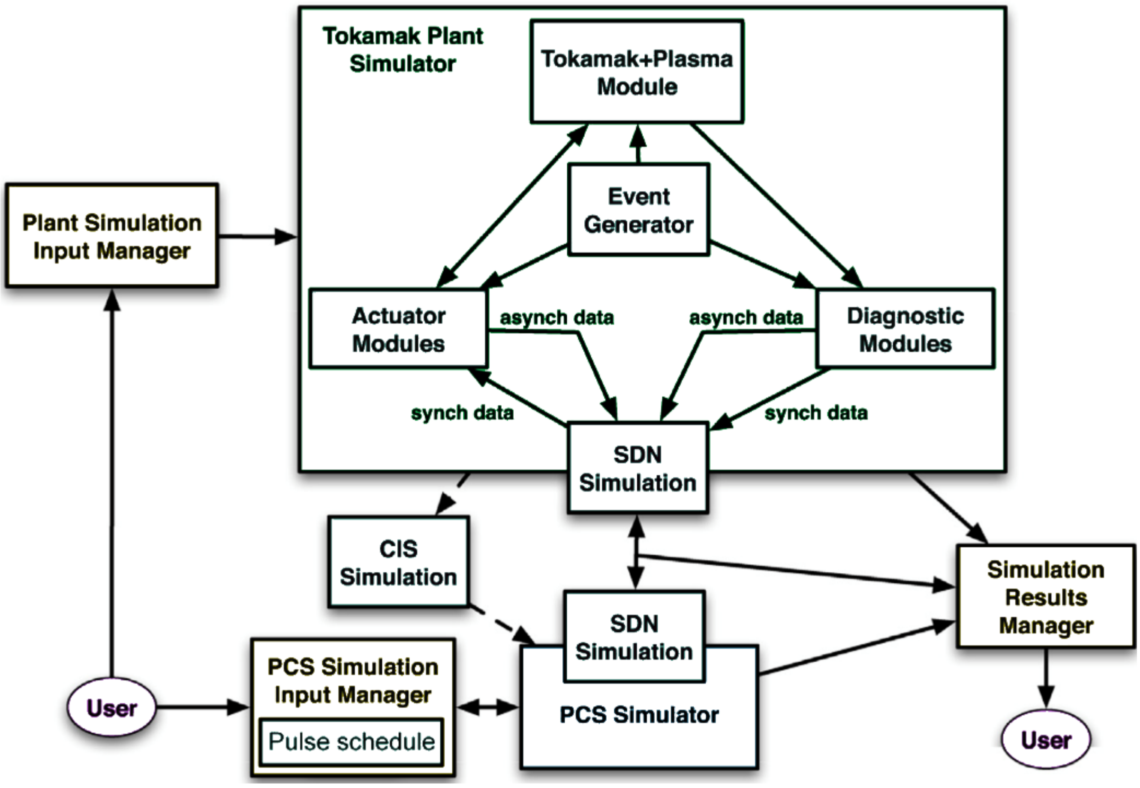


Fig. 2.4. Functional diagram of the plasma control system simulator [22, 23]

The work on the plasma simulation software and the PCS (Plasma Control System) continues. One of the approaches to solving this task is the developing IMAS software package: The ITER Integrated Modelling & Analysis Suite [24]. It will be available to all participants of the ITER project as a key tool for the scientific operation of ITER. The IMAS package will enable the collective development of integrated modeling tools through the exchange of data, code components and sequences of technological operations based on the different code components that are interconnected. The development of IMAS started in 2011 and the first prototype of the IMAS infrastructure has been implemented in the ITER organization. This package is also used in the WEST tokamak.

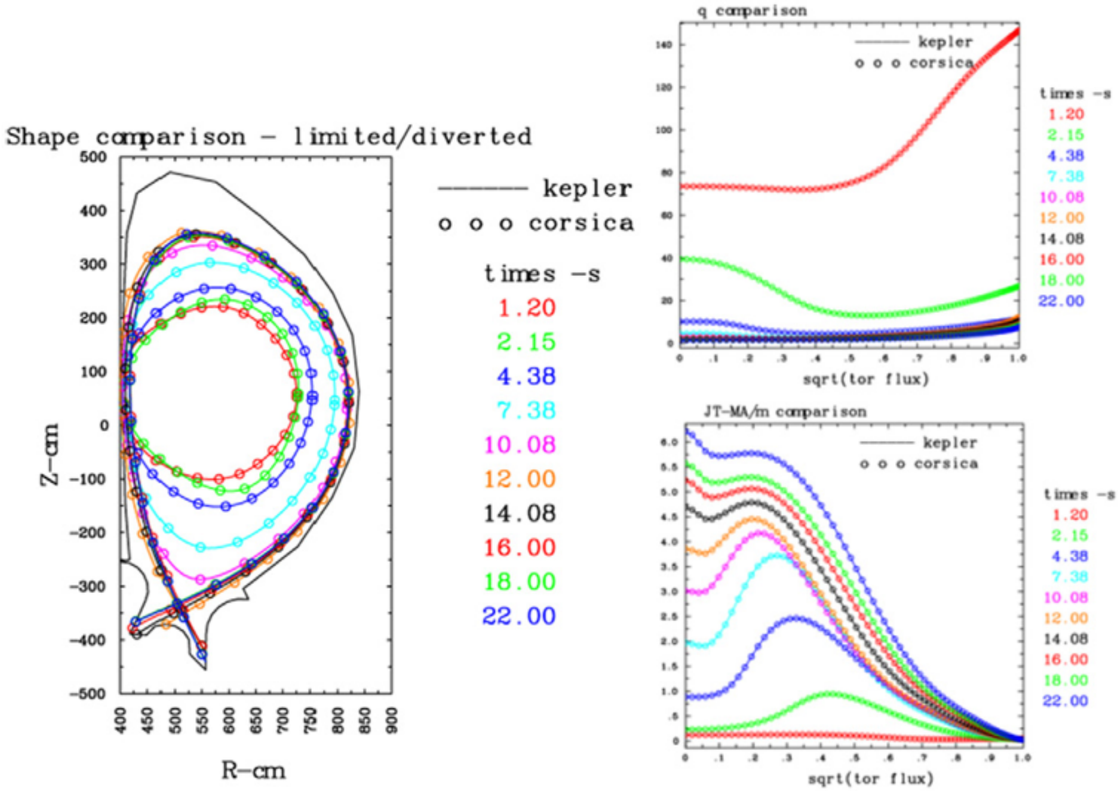


Fig. 2.5. Verification of the CORSICA-IMAS composition (‘Kepler’, solid lines) with respect to the stand-alone CORSICA code (‘Corsica’, circles): evolution of the plasma boundary (left), safety factor *q* profiles, and plasma current density (right) during plasma current rise in ITER. Both schemes provided exactly the same results: the curves completely coincided

The Kepler [25] graphics tool has been implemented in IMAS, which allows accessing the CORSICA code embedded in IMAS. Fig. 2.5 shows the result of the verification of the integrated CORSICA code and its stand-alone version.

3. PREPARING THE PLASMA CONTROL SYSTEM IN ITER FOR START-UP AND OPERATION

The ITER tokamak reactor is an extremely complex controlled plant, so it is important to determine the necessary elements and algorithms for further operation of the PCS together with the rest of the plant systems [26-29]. The list of basic control functions required for development when obtaining the first plasma in ITER has been compiled in [27]:

- control of currents in the poloidal field (PF) and central solenoid (CS) coils;
- magnetic zero feedback;
- initial control of plasma current;
- initial control of plasma position;
- initial control of plasma boundary;
- gas start-up control;
- initial density control;
- control of electron-cyclotron heating (ECH);
- prevention of scattered radiation at ECH.

The list of models (plasma, actuators, and sensors) corresponding to the final PCS development for the first plasma in ITER is also presented in [27]:

- PF and CS models;
- power supply models;
- magnetic sensor models;
- passive component models (e.g. eddy current model on the vessel);
- models of gas supply;

- model of the gas valve;
- model of pressure in the vessel;
- model of plasma fuel;
- model of the vacuum gauge (Penning sensor);
- model of the density interferometer;
- model of ECH;
- model of gas breakdown;
- model of the plasma position.

The paper [27] shows a structural scheme of the design, implementation, and operational testing of PCS, which consists of three steps.

- ✓ Step 1 confirms that the implementation of the system meets the design goals and technical requirements,
- ✓ Step 2 checks that the components of the system technically function correctly in accordance with the design goals,
- ✓ Step 3 checks that the complete system functions correctly in accordance with the design goals in all operating modes.

To facilitate the implementation, all the controllers and models are presented as modules of the PCSSP system platform [21-23], which is capable of automatically generating implementation code by means of Simulink®.

In the same paper [27] the scheme of designing and evaluation of the control is presented: Step 0 and the three basic commissioning steps in connection with the modeling of the controller, plant and on-line testing. Step 2 will require PCS capability testing in the loop testing of real equipment (HIL testing). Commissioning of the PCS architecture is an important feature of this stage, which ends with the transfer of the system for operational verification.

The ITER development and research plan, including PCS, defines a detailed development program [27]. Once the program is adopted, based on predictive modeling and risk analysis, the prepared stages can be executed in a workflow. Each planned stage should be checked before and after its execution, the measurements should be verified, and the results should be compared with the ones that were predicted by the modeling. Depending on the result, the planned stage will be configured within the approved range to perform the next test. The models employed in the simulation and the preparation of the work and, moreover, those needed to evaluate the PCS design, will be tested and can be refined. The latter may cause changes in the future work program, in the redesign of PCS or may affect the research plan of the next stage of the work in ITER.

As an example, Fig. 3.1 shows the modeling of the initial phase of the discharge in ITER at operation of the RT-EFIT equilibrium reconstruction code for the ITER scenario with a plasma current of 15 MA at t = 0.5 s, 4 s, 6 s, and 28 s, together with the deviations of the internal and external gaps between the plasma boundary and the first wall [26]. There is another code for the reconstruction of plasma equilibrium in real time, namely, LIUQE, which works well on TCV and includes a detailed model of the eddy currents. This code is also approved for ITER and is being tested [26].

In addition to the magnetic position, current, and plasma shape control system, in ITER there are also algorithms developed to improve the reliability of the plasma, so as to ensure the survivability of the entire plant. The limited number of actuators, in particular the additional plasma heating systems, necessitates the introduction of heterogeneous control functions during a plasma discharge, either sequentially or simultaneously with failures in the plasma or in the controlled plant systems. A partial failure in the actuator, e.g. in a gyrotron providing electronic cyclotron resonance heating, or in a gas valve, must be compensated for by automatic switching to another gyrotron or gas valve. To overcome failures and malfunctions, an advanced level of control of the actuators and processing of exceptional (emergency) situations in the plasma control system has been developed [30, 31].

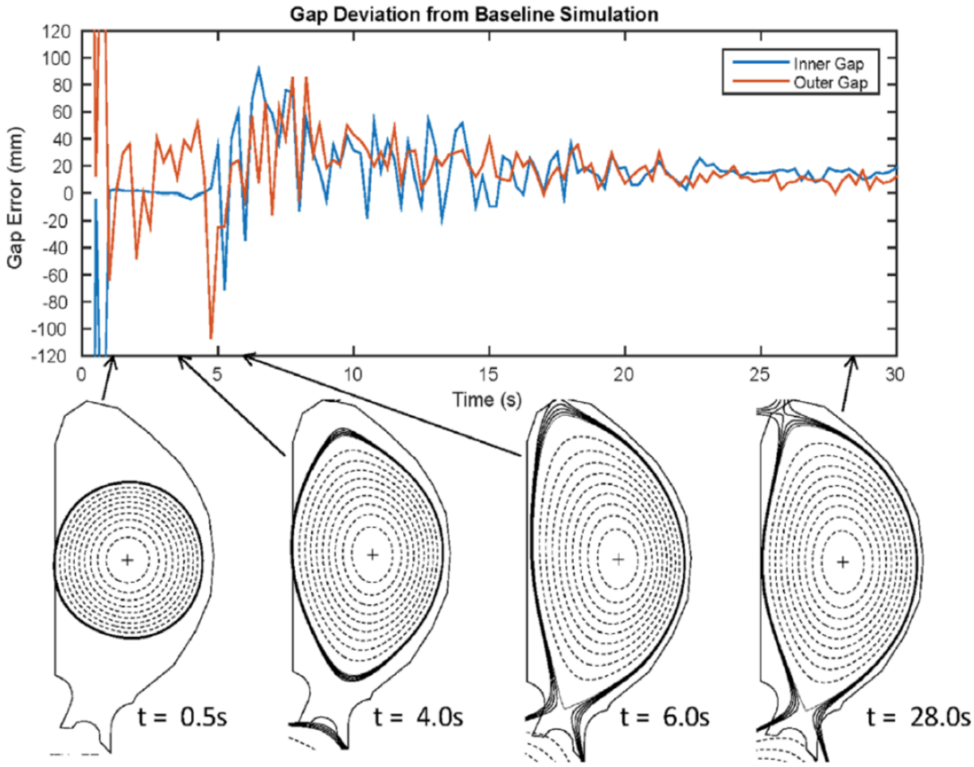


Fig. 3.1. Restoring plasma equilibrium using RT-EFIT code

Minor disruptions in ITER will be counteracted by PCS. The very high level of thermal and magnetic energy in ITER plasma requires an in-depth approach to protect the tokamak components from potential damage due to thermal loads, electromagnetic forces, and increasing electron flux in the event of major disruptions. The PCS is the first tokamak protection line that will try to avoid disruptions by confining the tokamak plasma within the specified limits and according to the specified conditions. The PCS will be equipped with advanced prediction algorithms of disruptions to ensure achieving the stability levels based on multiple real-time stability measurements and calculations. There are two approaches to predicting disruptions: one, based on physical measurements extrapolated into ITER [32], and the other, on machine learning techniques to analyse large data sets [33]. If a major disruption is unavoidable, the PCS sends a request to the Central Interlock System (CIS) to initiate a Disruption Mitigation System (DMS) so that the DMS injects large amounts of Z impurities to reduce the energy of the plasma, control the rate of the current decrease, and attenuate any thermal and electromagnetic loads due to the disruption. The DMS is designed with 12 independent granule (pellet) injectors to attenuate the thermal load and 15 independent injectors to attenuate the increase in the electron flow. In [26] the scheme of the interaction of PCS, CIS, and DMS is shown. The PCS decides on the sequence of injector initiations in the DMS to ensure the required amount and mixture of gas for the required time of the emergency situation, the PCS determines or predicts the inevitable cooling, current decrease, loss of vertical control, the presence of a rising flow of electrons or critical system failure in the plant.

4. DEMO DEVELOPMENT ROADMAPS

The development trend of conventional tokamaks with an aspect ratio of about 3-4 [1-3] has led to the design of ITER and next-generation projects of DEMO (thermonuclear power plants on tokamak reactors) with relatively large dimensions: the major radius R_0 of ITER is 6.2 m, and R_0 of DEMO is about 9 m. In doing so, spherical tokamaks with a relatively small aspect ratio of 1.4-1.6 were developed parallel to the conventional tokamaks. The plasma physics of these tokamaks differs from the physics of conventional tokamaks. Hence, it is much cheaper and faster to use them as the basis for commercial thermonuclear power stations than it would be to use conventional tokamaks. In particular, it has been shown that on the basis of one of the versions of the modular DEMO one can get an output of thermonuclear electricity costing less than 6 cents per 1 kWh [34].

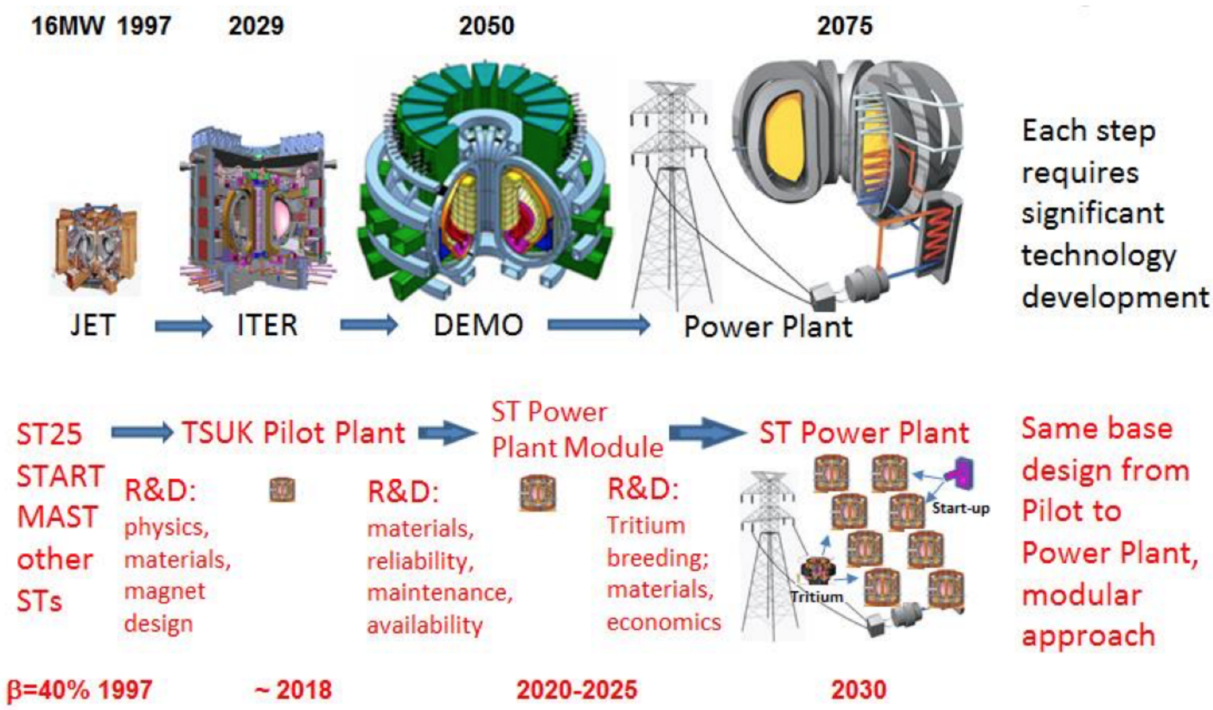


Fig. 4.1. Roadmaps for the development and construction of the first commercial tokamak reactor power plants

In this regard, two roadmaps for the development and construction of commercial power plants are currently available in Europe (Fig. 4.1) [35, 36]. It is estimated that commercial thermonuclear power plants of the conventional type may appear not earlier than 2075. At the same time the estimates for commercial thermonuclear power plants of spherical type have a more favorable forecast: a modular thermonuclear power plant can be created by 2030. Creating such a power plant is much easier, because first of all only one module is created, and then a power plant of full capacity is assembled from a number of modules. A special company has been established to develop such a power plant in the UK: Tokamak Energy Ltd (URL: https://www.tokamakenergy.co.uk/).

5. DEMO STRUCTURES AND POLOIDAL SYSTEMS

Offered designs of poloidal systems in DEMO on conventional tokamaks essentially repeat similar designs in the ITER-2 tokamak. One of the first conceptual designs of the DEMO tokamak was published in 2000 [37].

Fig. 5.1 shows the designs of DEMO poloidal systems presented by groups of researchers from different countries. The papers [38, 39] demonstrate schemes of Chinese and Korean-American DEMO tokamaks: the HCSB-DEMO project and the K-DEMO project. It should be noted that the presented list is not exhaustive: only the most different variants are given.

In Fig. 5.1, the project of one spherical tokamak module [34] is presented, the other options are conventional and non-modular. In the module, the poloidal field coils are located inside the toroidal field coil, and they are concentrated in two groups of four coils at the top and bottom of the vacuum vessel.

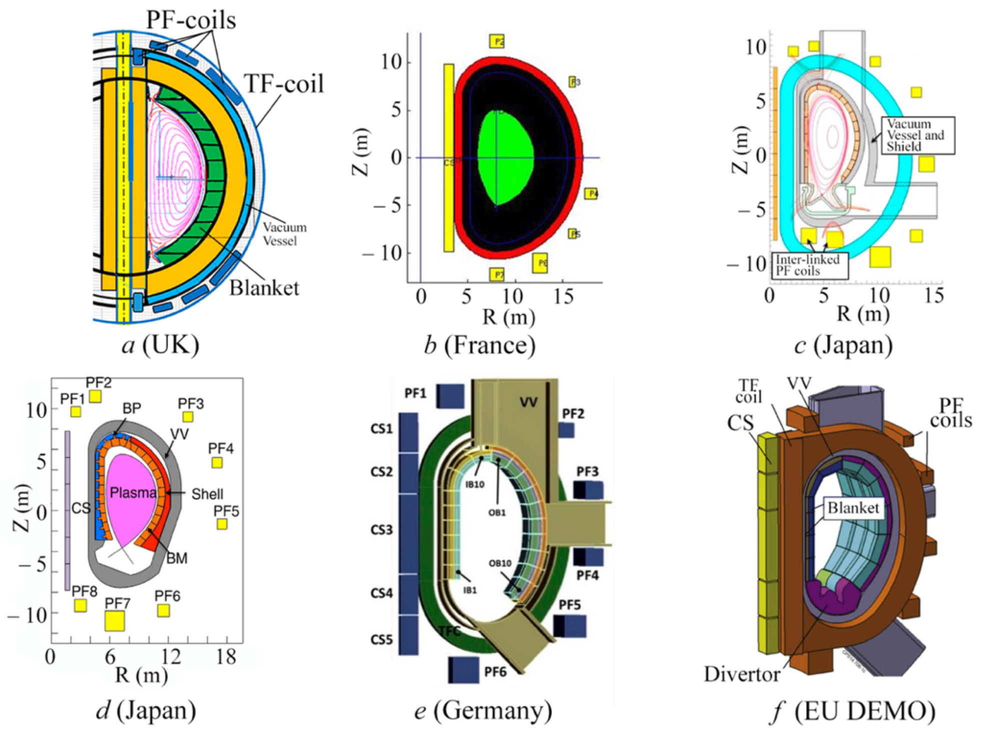


Fig. 5.1. Vertical cross sections of DEMO tokamaks with various poloidal system options

In [40] (Fig. 5.1, *c*) it is proposed to use 9 PF-coils with two lower PF coils placed inside the toroidal coil, in [34] (Fig. 5.1, *a*) and [44] (Fig. 5.1, *d)* there are 8 PF-coils, and in [42] (Fig. 5.1, *b)*, [41] (Fig. 5.1, *e)*, and [38] (Fig. 5.1, *f*) there are 6 PF-coils.

In [34] (Fig. 5.1, *a*) and [42] (Fig. 5.1, *b*) it is proposed to apply the whole CS, in other projects there is a sectioned CS, and in [41] (Fig. 5.1, *f*) and in [43] (Fig. 5.1, *e*) the CS consists of 5 sections, where the height of the central section is twice as high as all the others, in [44] (Fig. 5.1, *d*) the CS contains 6 sections, and in [40] (Fig. 5.1, *c*) the CS consists of 8 sections of the same height.

Table 5.1. Key features of DEMO projects

Project	Major radius, m	Minor radius, m	Aspect ratio	Maximum toroidal field, T	Plasma current, MA
UK, spherical DEMO module, Fig. 5.1, *a*	1.6	0.94	1.7	3.17	6.7
France, Fig. 5.1, *b*	9	2.65	3.4	10.5	17
Japan, Fig. 5.1, *c*	8.2	2.6	3.2	17	14.6
Japan, Fig. 5.1, *d*	8.2	2.57	3.19	17	14.6
Germany, Fig. 5.1, *e*	9	2.25	4	7.1	20
European Union (EU DEMO), Fig. 5.1, *f*	9	2.9	3.1	13.3	20
China (HCSB-DEMO)	7.2	2.1	3.4	6.86	14.8
Korea – USA (K-DEMO)	6.8	2.1	3.2	16	12
DEMO-S (Russia)	7.8	1.5	5.2	8.75	12.8

All presented projects are executed according to the conventional scheme [1], according to which the PF-coils are located outside the toroidal field coil, except for the spherical module [34] (see Fig. 5.1, *a*).

Until recently, no comparisons of the different magnetic configurations of DEMO had been conducted, except for [45], who partly conducted simulations for different sizes of the vacuum vessel and configurations of the PF-coils. It follows from the presentation in that article that the development of DEMO is decentralized, unlike the ITER project, in which the development is conducted by an international consortium of researchers. Table 5.1 shows the main characteristics of the above DEMO projects.

The project of the plasma position control system in DEMO, in fact, was presented only in [44], the results on current and plasma shape control systems for DEMO have not yet been given, the areas of the vertically controllable plasma and the areas of achievability of the separatrix have not been investigated, which can be explained by the need to check the corresponding algorithms in real work on ITER.

In [44], the assumptions made are as follows: the plasma is considered to be "rigid", taking into account the plasma current density profile; the components inside the vessel and the vacuum vessel itself are approximated as a three-dimensional and "thin" structure; the evolution of the eddy currents and coil currents is estimated using closed equations. The numerical simulation code for the plasma position control analysis consists of three modules:

- TOSCA plasma equilibrium code [46] for creation of a plasma current density profile and a magnetic surface by solving the Grad-Shafranov equation;
- EDDYCAL eddy current code [47] to evaluate the passive effect of stabilization of the vacuum vessel and components inside it by the finite element method;
- plasma position control code for calculation of the time evolution of the vertical and radial plasma positions, eddy currents and coil currents of the feedback with a PID controller taking into account the results of calculating the equilibrium coefficients in the plasma and eddy currents [48].

Fig. 5.1, *d* shows a cross-section including the plasma, protective modules (BM), conductive shell (Shell), rear plates (BP), vacuum vessel (VV) and the system of poloidal coils (PF).

The sequence of calculations in the structural diagram of the plasma position control system is as follows: after calculating, using the equilibrium reconstruction code, the plasma current density profile (J_p), the eddy current time constant (τ_S) and the mutual inductances between the plasma and coils are calculated using the eddy current code EDDYCAL. Then, taking into account the data from the equilibrium and eddy current analysis, the evolution of the plasma motion (in vertical and radial directions), eddy currents and control coil currents are calculated. The developed numerical simulation code can work out exact models of the vacuum vessel and the components inside it [44].

6. CONCLUSION

In addition to modeling and implementing plasma control systems in ITER, this part of the survey has provided information on the development of the first DEMO thermonuclear power plant, including the design, poloidal systems, and plasma position control system. Two DEMO development roadmaps have been presented for conventional tokamaks with relatively high aspect ratios and spherical tokamaks of modular type with a low aspect ratio. According to current estimates, the first commercial thermonuclear power plant with relatively cheap electricity can be developed much faster when based on the latter.

ITER experiments in conditions of thermonuclear reaction will allow formulating more precise requirements for the plasma control systems for DEMO on conventional tokamaks. It will then be possible to proceed with the development of the appropriate systems for DEMO conventional types with some certainty. For DEMO modular type on spherical tokamaks it is advisable to work out in detail plasma control systems on existing spherical tokamaks such as ST40 (UK), Globus-M2 (Russia), NSTX (USA).

Research is being carried out on the integration of the magnetic and kinetic plasma control systems, where an important direction is the control of the profile of the plasma parameters (current density, safety factor q, density, temperature). This direction, in particular, should lead to an understanding of which plasma parameter profiles must be provided so that no major disruptions occur. The methods and systems of kinetic plasma control will be reviewed in the fourth part.

ACKNOWLEDGEMENTS

The work was supported by the Russian Foundation of Basic Research (grant no. 17-18-00293) and the Russian Science Foundation (grant no. 17-19-01022).

REFERENCES

1. Mitrishkin, Yu. V., Korenev, P. C., Prokhorov, A. A., et al. (2018). Plasma control in tokamaks. Part 1. Controlled fusion problem. Tokamaks. Control system components // *Control Sciences*, 1, 2–20 (in Russian). [English translation in *Advances in Systems Science and Applications,* 18(2), 26–52].
2. Mitrishkin Yu. V., Korenev P. S., Prokhorov A. A., et al. (2018). Plasma control in tokamaks. Part 2. Plasma magnetic control systems, *Control Sciences,* 2, 2–30 (in Russian) [English translation in *Advances in Systems Science and Applications,* 18(3), 39–78].
3. Mitrishkin, Y. V., Kartsev, N. M., Konkov, A. E., & Patrov, M. I. (2019). Plasma control in turntables. Part 3.1. Plasma magnetic control systems in ITER, *Control Sciences,* 3, 3–15 (in Russian).
4. Doyle, E. J., DeBoo, J. C., Ferron, J. R., et al. (2010). Demonstration of ITER operational scenarios on DIII-D, *Nuclear Fusion,* 50(7). DOI: 10.1088/0029-5515/50/7/075005
5. Wesson, J. (2004) *Tokamaks*. 3rd ed. Oxford: Clarendon Press.
6. Colnel, J., Ravenel, N., Nouailletas, R., et al. (2018). Adapting DCS real time framework for WEST plasma control, *Fusion Engineering and Design*, 129, 24–28.
7. Nardon, E., Heumann, H., Artaud, J. F., et al. (2018). Magnetic configuration and plasma start-up in the WEST tokamak, in: *45th EPS Conference on Plasma Physics, 2018, Prague, Czech Republic*, URL: https://hal.archives-ouvertes.fr/hal-01959689/file/O3.110.pdf.
8. Bourdelle, C., Artaud, J. F., Basiuk, V., et al. (2015). WEST Physics Basis, *Nuclear Fusion*, 55(6) 063017. DOI: 10.1088/0029-5515/55/6/063017.
9. Khayrutdinov, R. R., & Lukash, V. E. (1993). Studies of Plasma Equilibrium and Transport in a Tokamak Fusion Device with the Inverse-Variable Technique, *Journal Comp. Physics*, 109(2), 193–201.
10. Dokuka, V. N., Kadurin, A. V., Mitrishkin, Y. V., & Khayrutdinov, R. R. (2007). Synthesis and modeling of the magnetic plasma control system in the reactor tokamak, *Automatics and Telemechanics*, 8, 126–145 (in Russian). [*English translation:* Synthesis and modeling of the H∞-system of magnetic control of the plasma in the tokamak, *Automation and Remote Control*, 68(8), 1410–1428].
11. Mitrishkin, Y. V., Dokuka, V. N., & Khayrutdinov, R. R. (2005). Linearization of ITER plasma equilibrium model on DINA code, in: *Proc. of 32nd EPS Plasma Physics Conference. Tarragona, Spain, 2005,* article ID P5.080.
12. Mitrishkin, Yu. V., Korostelev, A. Ya., Dokuka, V. N., & Khayrutdinov, R. R. (2011). Synthesis and simulation of the two-level system of the magnetic control of the reactor tokamak plasma, *Plasma Physics,* 37(4), 307–349 (in Russian) [English translation in *Plasma Physics Reports*, 37(4), 279-320.]
13. Applied Dynamics International. Solutions in Real Time. URL: http://www.adi.com [Accessed on 7 July 2020].
14. Walker, M. L., Humphreys, D. A., Leuer, J. A., et al. (2003). Practical Control Issues on

DIII-D and Their Relevance for ITER, General Atomics, Engineering Physics Memo, EPM111803a.

15. Mitrishkin, Y. V., Efremov, A. A., & Zenkov, S. M. (2013). Experimental Test Bed for Real Time Simulations of Tokamak Plasma Control Systems, *Journal of Control Engineering and Technology*, 3(3), 121–130.
16. CODAC. Core System Overview (34SDZ5). (URL: http://static.iter.org/codac/pcdh7/Folder%201/3-CODAC_Core_System_Overview_34SDZ5_v6_1.pdf).
17. CODAC, Plant Control Design Handbook (URL: https://www.iter.org/mach/codac/PlantControlHandbook) [Accessed on 7 July 2020].
18. EPICS. Experimental Physics and Industrial Control System (URL: https://epics.anl.gov) [Accessed on 7 July 2020].
19. ITER Catalogue for I&C products. Slow controllers PLC (URL: http://static.iter.org/codac/pcdh7/Folder%202/7-ITER_catalogue_for_I%26C_products_-_Slow_c_333J63_v3_3.pdf) [Accessed on 7 July 2020].
20. ITER Catalog of I&C products. Fast Controllers (URL: http://static.iter.org/codac/pcdh7/Folder%202/9-ITER_Catalog_of_I%26C_products_-_Fast_Cont_345X28_v2_4.pdf) [Accessed on 7 July 2020].
21. Winter, A., Makijarvi, P., Simrock, S., et al. (2014). Towards a preliminary design of the ITER plasma control system architecture, *Fusion Engineering and Design*, 89(3), 143–272.
22. Winter, A., Ambrosino, G., Bauvir, B., et al. (2015). Implementation strategy for the ITER plasma control system, *Fusion Engineering and Design*, 96–97, 720–723.
23. Walker, L., Ambrosino, G., De Tommasi, G., et al. (2015). The ITER Plasma Control System Simulation Platform, *Fusion Engineering and Design*. 96–97, 716–719.
24. Imbeaux, F., Pinches, S. D., Lister, J. B., et al. (2015). Design and first applications of the ITER integrated modelling & analysis suite, *Nuclear Fusion*, 55(12) 123006. DOI: 10.1088/0029-5515/55/12/123006.
25. The Kepler Project (URL: http://kepler-project.org) [Accessed on 7 July 2020].
26. Snipes, J. A., Albanese, R., Ambrosino, G., et al. (2017). Overview of the preliminary design of the ITER plasma control system, *Nuclear Fusion*, 57(12) 125001.
27. de Vries, P. C., Snipes, J. A., Zabeo, L., et al. (2018). Preparing the Plasma Control System final design for ITER first plasma operations, *Fusion Engineering and Design*, 129, 334–340.
28. de Vries, P. C., and Gribov, Y. (2018). Size matters: ITER breakdown and plasma initiation revisited, in: *Proc. 45th European Physical Society Conference on Plasma Physics, Prague, Czech Republic, 2018*, O3.109.
29. Gribov, Y., Kavin, A. A., Lukash, V. E., et al. (2018). Progress in simulation of ITER First Plasma operation, in: *Proc. 45th European Physical Society Conference on Plasma Physics, Prague, Czech Republic, 2018*, P1.1075.
30. Raupp, G., Pautasso, G., Rapson, C., et al. (2017). Preliminary exception handling analysis for the ITER plasma control system, *Fusion Engineering and Design*, 123, 541–545.
31. Treutterer, W., Rapson, C. J., Raupp, G., et al. (2017). Towards a preliminary design of the ITER plasma control system architecture, *Fusion Engineering and Design*, 115, 33–38.
32. de Vries, P.C., Pautasso, G., Nardon, E., et al. (2006). Scaling of the MHD perturbation amplitude required to trigger a disruption and predictions for ITER, *Nuclear Fusion*, 56(2) 026007. DOI: 10.1088/0029-5515/56/2/026007.
33. Moreno, R, Vega, J., Murari, A., et al. (2014). Robustness and increased time resolution of JET Advanced Predictor of Disruptions, *Plasma Physics and Controlled Fusion*, 56(11) 114003. DOI: 10.1088/0741-3335/56/11/114003.
34. Chuyanov V., & Gryaznevich M. (2017). Modular fusion power plant, *Fusion Engineering and Design*, 122, 238–252.
35. Gryaznevich, M., Sykes, A., Costley, A.E., et al. (2013). The spherical tokamak path to fusion power, revisited, in: *Proc. of 40th EPS Conference on Plasma Physics, Espoo,*

Finland, 2013, 2.111.
36. Gryaznevich, M. P., Chuyanov, V. A., Kingham, D., et al. (2015). Advancing Fusion by Innovations: Smaller, Quicker, Cheaper, *Journal of Physics: Conference Series*, Vol. 591.
37. Shatalov, G., Kirillov, I., Sokolov, Y., et al. (2000). Russian DEMO-S reactor with continuous plasma burn, *Fusion Engineering and Design*, 51–52, 289–298.
38. Feng, K. M., Zhang, G. S., Zheng G. Y., et al. (2009). Conceptual design study of fusion DEMO plant at SWIP, *Fusion Engineering and Design*, 84(12), 2109–2113.
39. Kim, K., Oh, S., Park, J.S., et al. (2015). Conceptual design study of the K-DEMO magnet system, *Fusion Engineering and Design*, 96–97, 281–285.
40. Utoh, H., Asakura, N., Tobita, K., et al. (2014). Application of inter-linked superconducting coils for central solenoid and advanced divertor configuration of DEMO, *Fusion Engineering and Design*, 89(9), 2456–2460.
41. Maione, I., Vaccaro, A. (2014). Analysis of electromagnetic loads on EU-DEMO inboard and outboard blanket vertical segments, *Fusion Engineering and Design*, 89(7), 1314–1318.
42. Duchateau, J. L., Hertout, P., Saoutic, B., et al. (2014). Conceptual integrated approach for the magnet system of a tokamak reactor, *Fusion Engineering and Design*, 89(11), 2606–2620.
43. Bachmann, C., Aiello, G. Albanese, R., et al. (2015). Initial DEMO tokamak design configuration studies, *Fusion Engineering and Design*, 98–99, 1423–1426.
44. Takase H., Utoh H., Sakamoto Y., et al. (2016). Analysis of plasma position control for DEMO reactor, *Fusion Engineering and Design*, 109–111, 1386–1391.
45. Utoh, H., Tobita, K. Asakura, N., & Sakamoto, Y. (2014). Evaluation of remote maintenance schemes by plasma equilibrium analysis in Tokamak DEMO reactor *Fusion Engineering and Design*, 89(11), 2588–2593.
46. Shinya, K. (2000). Equilibrium analysis of tokamak plasma, *J. Plasma Fusion Res.*, 76, 479–488.
47. Kameari, A. (1981). Transient eddy current analysis on thin conductors with arbitrary connections and shapes, *J. Comput. Phys.*, 42, 124–140.
48. Kasai, M., Kameari, A., Matsuoka, F., et al. (1988). Simulation studies on plasma position control for the next generation tokamak machines with up-down asymmetry, *Fusion Engineering Design*, 5, 343–356.

