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Control system and the controllability of CPD and QUEST

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Abstract

Superconducting tokamak TRIAM-1M has just ceased its operation in December 2005. However, in order to develop the systematic study on plasma–wall interaction in long duration discharges in the TRIAM-1M tokamak, spherical tokamak QUEST is under construction. To fulfill the mission of QUEST under a small group of the university in collaboration with other group, control and interlock system must be designed appropriately. Hence, a spherical tokamak CPD was constructed and the experiment is carried out for preparation of QUEST. The control and interlock system for QUEST is tested and developed during the experiment in the CPD.

Keywords: CPD; QUEST; ST; Feedback control; Passive coil; Eigenvalue; MATLAB

1. Introduction

Superconducting tokamak TRIAM-1M (Modified 1st Tokamak of Research Institute for Applied Mechanics) ceased its operation in December 2005. The discharge duration of 5 h and 16 min was achieved in November 2003 [1]. However, for a systematic study of PWI (Plasma–Wall Interaction) in long duration discharges in the TRIAM-1M tokamak, spherical tokamak QUEST (Q-shu University Exp. on Steady-State Spherical Tokamak) is currently under construction. Keywords of the mission are “spherical tokamak”, “steady-state”, “divertor” and “high-temperature wall”. To fulfill the mission of QUEST under university of a small group in collaboration with other group, control and interlock system must be designed appropriately.

For the steady-state operation, steady plasma control, steady impurity reduction, steady particle control, steady heat removal, etc., are necessary. In the plasma control, pulse trigger timing system, fast feedback control system and sequence control may cooperate by adopting FPGA (Field Programmable Gate Array). For the impurity reduction, divertor configuration must be controlled precisely and stably. Moreover, for steady particle control, high-temperature first-wall of 300–500 °C is planned. For heat removal, water and gas are used creatively according to the heat characteristics.

In order to achieve the QUEST mission by a small group, the control systems must be automatized and the interlock system must be complete. Furthermore, to promote long-time operation in collaboration with other group in other locations, real-time and remote data acquisition and browsing are indispensable. Moreover, for real-time view and control of the steady-state plasma, reproduction of spherical plasma by CCS (Cauchy Condition Surface) method will be introduced [2].

CPD (Compact PWI experimental Device) was constructed and an experiment was carried out to prepare QUEST. The control and interlock system for QUEST was tested and developed during the experiment in the spherical tokamak CPD.

2. Schedule and existing infrastructures

Superconducting tokamak TRIAM-1M finished its operation after 20 years. Spherical tokamak QUEST is under construction (from 2005 to 2007) to develop a systematic study on PWI in long duration discharges on TRIAM-1M. The control system has been tested and developed on CPD.

Main parameters of CPD and QUEST are listed in Table 1. Ports for evacuation, heating and diagnostics are designed with the same configuration so that they are compatible with existing devices.

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Table 1
Main parameters of CPD and QUEST

	CPD	QUEST
Plasma major radius	0.3 m	0.64 m
Plasma minor radius	0.2 m	0.36 m
Aspect ratio	1.5	1.78
Toroidal field	0.3 T at $R = 0.25$ m	0.25 T at $R = 0.6$ m
Operation period	1.0 s for Bt = 0.3 T	
Operation cycle	5 min	
Plasma current	150 kA	20 kA (I) 100 kA (II steady) 300 kA (II pulse)
PFC		W (>300 °C)

Keywords of the mission of QUEST are as follows:

(1) Spherical tokamak.
(2) Steady-state operation.
(3) Divertor configuration.
(4) High-temperature wall (>300 °C).

The mission of QUEST will be achieved using the existing infrastructure, and the university (a small group and large collaborations).

At the Chikushi Campus, there are 3 power lines: 4000-kVA line, 3000-kVA line, and 4000-kVA line. The first 4000-kVA line is for heating device and for the power supply for the pulsive PF (Poloidal Field) coil including MG (Motor Generator). The second 3000-kVA line is for the power supply for the TF (Toroidal Field) coil. The third 4000-kVA line is for the buildings, where 2000 kVA is for cooling, pumping, steady power supply, diagnostics, etc.

In a spherical tokamak, the toroidal field is low and a normal conductor can be adopted. However, a large electric power system is necessary for steady-state operation. Thus, the power supply for the toroidal field coil is fed directly from a 6.6 kV line, and the consumed power is high to be 3000 kW. The display for a 30 min demand is shared and supervised also in a control room through the Internet.

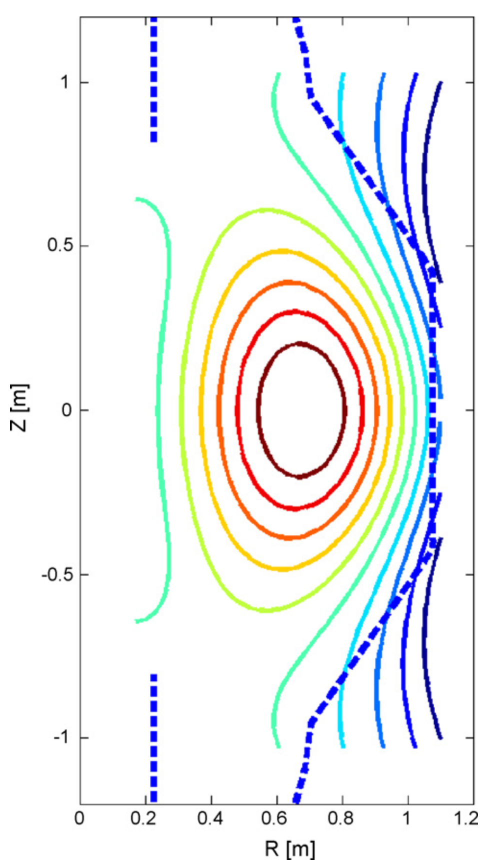

Fig. 1. Passive model of the liner for vertical stability analysis.

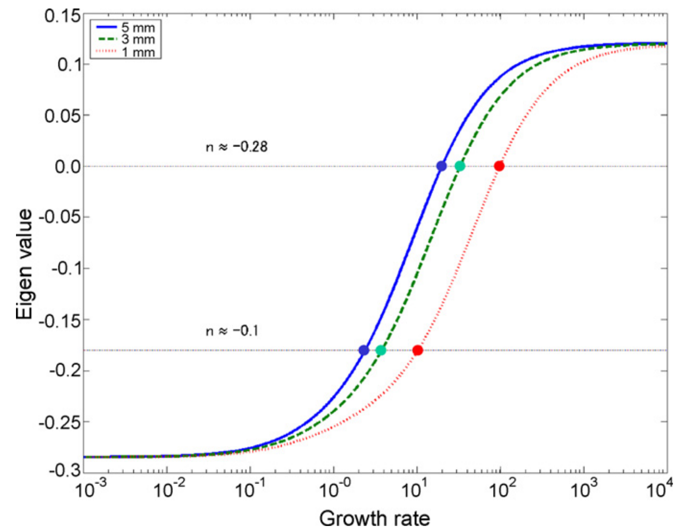

Fig. 2. Stabilizing effect of the liner on the vertical position instability for $\kappa \sim 2$ ($n \sim -0.28$).

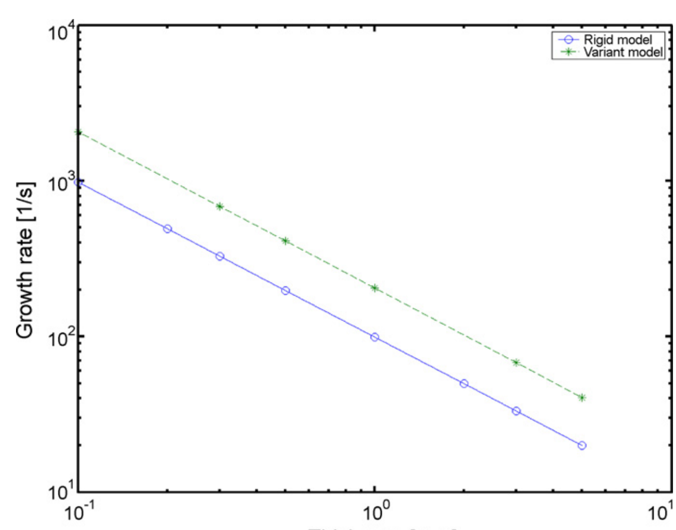

Fig. 3. Liner thickness dependence of the growth rate in the unstable mode ($n \sim -0.28$).

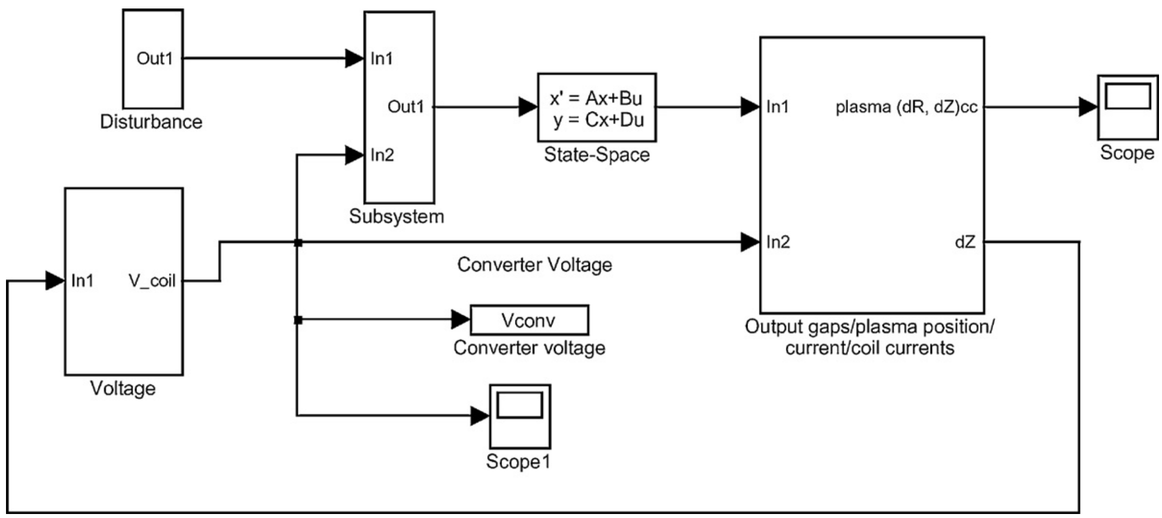

Fig. 4. Simulation of vertical position control by MATLAB.

3. Control system

3.1. Plasma control system

The control system is composed of sequential control with a timing system, and a plasma control system. In the former system, the guiding principles are unification of the base clock, interlock in the cyclic operation and individual interlock in the distributed system. Currently sequence controller HISEC04 and digital delay generator DG-200 for TRIAM-1M are also used for CPD. However, QUEST will unify these into FPGA.

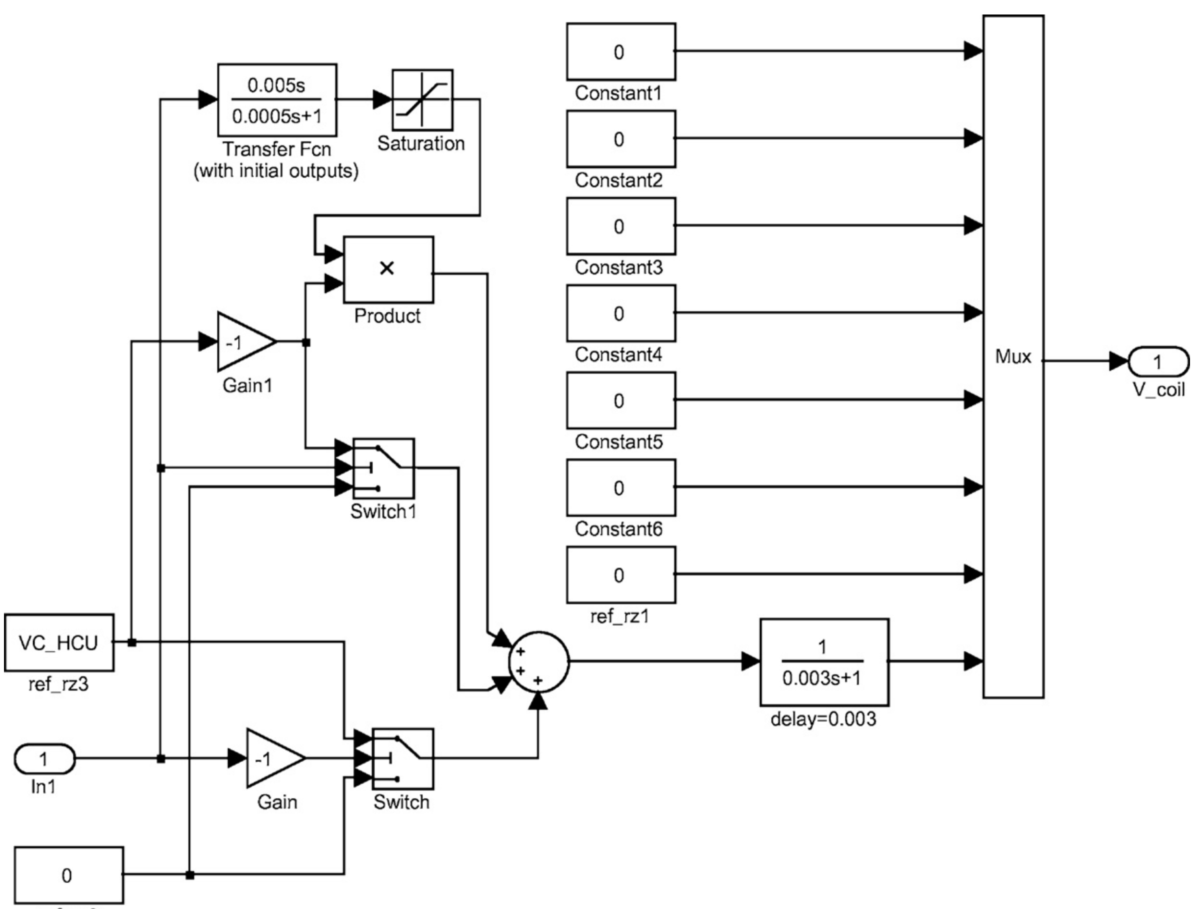

Fig. 5. Simulation of vertical position control by MATLAB (voltage block in detail).

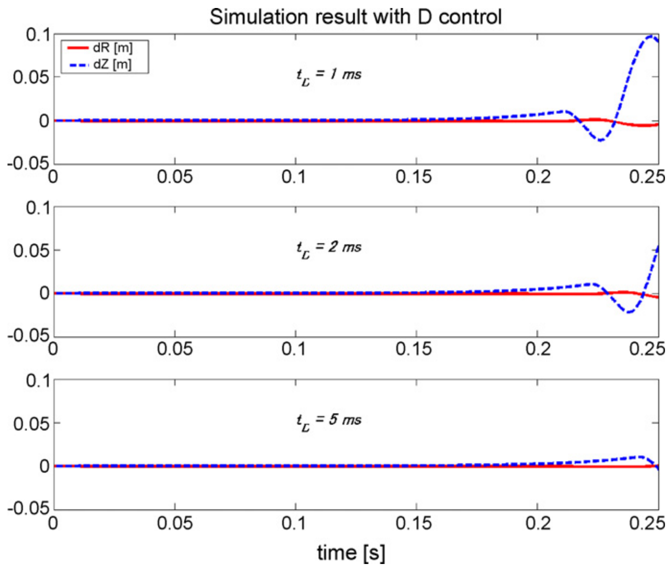

Fig. 6. Simulation of vertical position control by MATLAB (thickness 5 mm).

In the latter plasma control system, the controlled parameters are ΔR, ΔZ, κ, δ, n, T, J, etc. Herein we discuss about ΔZ control. The ΔZ value is detected by a flux loop, eddy current loop, etc., using the CCS method [2,3]. The controllers are PID (Proportional–Integral–Derivative), optimum, NN (Neural Network), etc. Moreover, this paper will discuss the PID controller. Currently WS (Work Station) Power Hawk 640 with a real-time OS (Operating System) and VME (Versa Module Europe) bus for TRIAM-1M are used for CPD. However, a PXI (PCI [Peripheral Component Interconnect] extensions for Instrumentation) system will be adopted along with controller module PXI-8106, FPGA board PXI-7833R and PXI-6509 for QUEST.

The power supplies consist of a: 24-pulse mono-polar thyristor converter for the vertical coil, a bi-polar GTO (Gate Turn Off thyristor) inverter (3 kHz) for the shaping coil, a bi-polar transistor inverter (3 kHz) for the horizontal coil, and a bi-polar transistor dropper for the changing CS (Central Solenoid) coil current distribution.

Electric power to the above power supplies can be fed from a MG (Motor Generator) or SD (Step Down) transformer. In the former case, 6.6 kV is available for one minute, which makes tests and optimization possible by this high voltage power source. In the latter case, steady-state operation is possible, though the voltage is low to be 1.32 kV (1/5), 0.88 kV (1/7.5) or 0.66 kV (1/10).

3.2. Controllability of the vertical position instability

A spherical tokamak has a natural elongation due to the low aspect ratio. However, QUEST has a divertor configuration with a higher κ and a negative n-index. Therefore, the vertical position instability must be stabilized by conducting shell and feedback control.

For the conducting shell, we used a liner of copper (Fig. 1). We initially considered passive model of the liner for vertical stability analysis first. The liner was divided into segments, and each segment was approximated as the toroidally symmetric passive poloidal coils. The plasma was considered in two ways based on a rigid model and a variant shape model. In case of the rigid model, the linear growth rate of the vertical plasma position is calculated from eigenvalue analysis of the equation for vertical motion of the plasma center. The stabilizing effect of the liner to the vertical position instability using rigid shape model is shown in Fig. 2. Because the frequency of the inverter for the horizontal field coil is 3 kHz, vertical position instability may be controllable when the growth rate is less than 700 s^{-1}, and can be controlled even when it is less than 200 s^{-1}. The standard divertor configuration with a $\kappa \sim 1.6$ ($n > -0.1$) can be controllable even when the Cu liner is 0.1 mm thick. However, we must test whether the divertor configuration with a high $\kappa \sim 2$ ($n \sim -0.28$) is controllable. The liner thickness dependence of the growth rate for the unstable mode ($n = -0.28$) is shown in Fig. 3.

In contrast, the growth rate based on the variant shape model is about twice as large as the one from the rigid model [4]. The growth rate of $n \sim -0.1$ is about one tenth of $n \sim -0.28$, but becomes 200 s^{-1} for a thickness of 0.1 mm (copper) even in the variant shape model. Thus, it is determined that the vertical position instability can be controllable for a thickness of 4 mm (Stainless Steel).

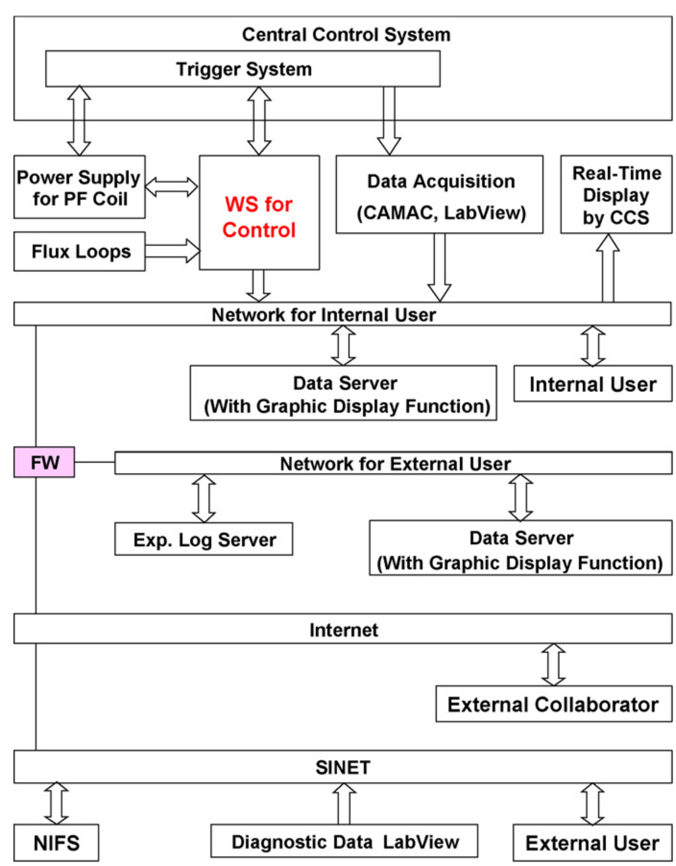

Fig. 7. Network system for control.

3.3. Control simulation by MATLAB

Next, we confirm the control system by a simulation with MATLAB (MATrix LABoratory). The simulation block diagram is shown in Fig. 4. The state space block includes plasma, PFC (Poloidal Field Coil), and a liner (passive coils). The voltage block includes a PID controller and power supply. The details are depicted in Fig. 5. When the vertical position becomes larger than 1 cm, the maximum voltage command is sent to the power supply for the vertical field coil. The derivative control is added to this bang–bang control. The simulation results are shown in Fig. 6. When t_D is 1 ms (derivative control is neglected), the amplitude of the vertical position increases exponentially (the vertical position has an unstable behavior). As the time is increased to 2 ms and 5 ms, the vertical position becomes more stable. Moreover, the vertical system is stabilized at 10 ms.

However, the high-frequency component of the vertical position must be measured by setting the eddy current loop. The eddy current distribution may be calculated from the flux loop signals after the delay time of the signal is considered.

In addition, the effect of the toroidal cut for the plasma current start-up and extension port must be taken into account. Furthermore, in QUEST, the temperature of the in-vessel component must be considered.

4. Network system for control

Network system for control is shown in Fig. 7. In the central control system, the base clock will be the same as that of trigger system in order to reduce jitters between them. External collaborators can access the data server of CPD and QUEST for external user only through a firewall. WEB services for mobile phones are being developed in order to access experimental information; to browse the electronic logbook and to monitor sequence status [5].

5. Summary

Spherical tokamak QUEST is currently under construction to develop a systematic study on PWI in long duration discharges on TRIAM-1M. The control system for QUEST has been tested and developed on CPD.

A standard divertor configuration with a $\kappa \sim 1.6$ ($n \sim -0.1$) may be controllable for QUEST. However, with regards to a divertor configuration with a high $\kappa \sim 2$ ($n \sim -0.28$): as the derivative time is increased to 2 ms and 5 ms, the vertical position becomes more stable, and is stabilized at 10 ms.

The high-frequency component of the vertical position must be measured by setting the eddy current loop. Fortunately, the eddy current distribution can be calculated from the flux loop signal.

Concerning network system for control: external collaborators can access the Data Server of CPD and QUEST via the Internet.

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