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Poloidal Field Control for the HT-7U Superconducting Tokamak*

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Abstract Controlling the poloidal field (PF) in the HT-7U superconducting tokamak is critical to the realization of the mission of advanced tokamak research. Plasma start-up, plasma position, shape, current control and plasma shape reconstruction have been performed as a part of its design process. The PF coils have been designed to produce a wide range of plasmas. Plasma start-up can be achieved for multiple conditions. Fast controlling coils for plasma position inside the vacuum vessel are used for controlling the plasma vertical position on a short timescale. The PF coils control the plasma current and shape on a slower timescale. VXI (VME bus extensions for Instrumentation) Bus system and DSP (Digital Signal Processor is a basic unit of the feedback control system), the response time of which is about (2~4) ms. The basic unit of this system, the shape-controlling algorithms of a few critical points on plasma boundary and real-time equilibrium fitting (RTEFIT) will be described in this paper.

Keywords: plasma control, VXI Bus, equilibrium, start-up,DSP

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1 Introduction

The HT-7U Superconducting Tokamak, as a national project, is an advanced steady-state plasma experimental device which is being build in the ASIPP (Institute of Plasma Physics, Chinese Academy of Sciences). The mission of the HT-7U project is to develop scientific issues on the sustainment of a nonburning plasma scenario for the steady-state operation of next-generation advanced tokamak device, and to achieve engineering issues on establishing a technology basis of superconducting tokamak to support future reactors. The key physics issue of HT-7U is to operate at a high beta with improved plasma confinement and advanced power/particle handling in steady-state plasmas [1]. HT-7U is designed to have a long-pulse (~ 1000 s) operation, a flexible PF system, an auxiliary heating system and current-driving systems, and will be able to accommodate heat loads of divertor that make it an attractive test for the development of advanced tokamak operation mode. To obtain simultaneously these features will require a significant control capability. As a part of the larger plasma control strategy on HT-7U, the control of the poloidal magnetic field is central to these goals. This paper reports the progress achieved for HT-7U in this area, including plasma equilibrium and operation space: plasma start-up, dynamic simulations of plasma position, shape and current control, and reconstruction of the plasma boundary from magnetic measurement. Extensive calculations of plasma equilibrium have been carried out for a wide range of plasma parameters and operation constraints to guarantee that the poloidal field

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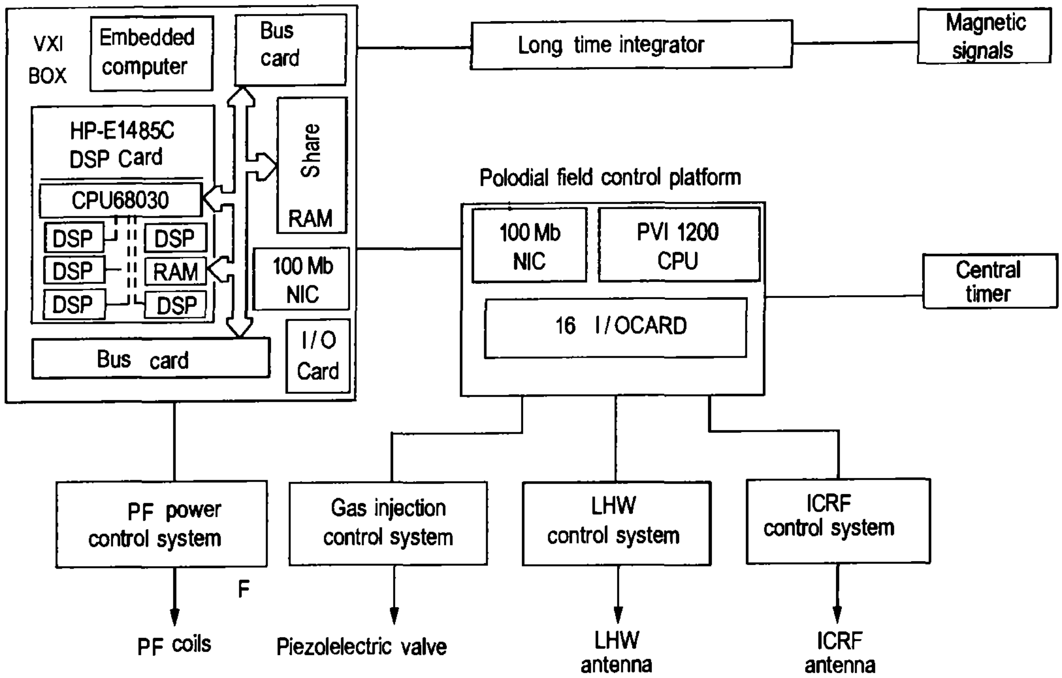

Fig.1 HT-7U pólodial field control system.

(PF) coils can produce all plasma of interest.

Based on the design scheme of HT-7U poloidal-field control system, an integrative design way will be used. Because the coils are of superconductivity, the voltage on the poloidal coils cannot be too high and the frequency cannot be changed too fast. All these factors must be considered in the control system design. In the system model, the current in the poloidal coils is made up of preset program current and relatively small feedback current, so as to determine the relationship of plasma shape parameters with time. As a key part of the control system in the HT-7U superconducting tokamak, it determines directly whether the experiment will be successfully implemented. But in order to meet the need for a strict real time (the response time less than 2 ms) and a much large matrix operation, some special methods must be adopted. This paper will give a detailed introduction.

2 The components of the HT-7U poloidal field control system

The poloidal field control system consists of plasma setup, long time digital / analog composite integrator, plasma shape and current control etc. Fig. 1 shows the frame of the system.

2.1 Plasma setup

Ideally, the initial conditions of a tokamak prior to plasma breakdown consist of a vessel filled with neutral gas and a purely toroidal magnetic field. The breakdown is initiated by applying a toroidal electric field. Because of natural background radiation and cosmic rays, a few free electrons are present and will be accelerated by the electric field along the toroidal magnetic field lines. These electrons will be collided with neutral atoms to ionize these neutrals. A chain reaction takes place until all neutrals are ionized. In the real life there are two main reasons that will slow down the chain reaction. First of all, the radiation of ionized plasma usually causes an energy loss. In order to continue the ionization, the ohmic power dissipation proportional to the square of the applied electric field must exceed the radiation loss proportional to the effective charge Z_{eff}. Thus both the increased applied electric field and decreased Z_{eff} are of advantage to the breakdown. Electrons loss from the plasma is the second reason for slowing down the chain reaction. The ionized length of an electron, i.e. the typical length of the trajectory between two ionizing collisions, decays exponentially with the increased electric field. The reconnected length is the typical length of the trajectory of an electron between its formation (ionization) and its loss from the plasma.

It appears that both factors resulting in the breakdown can be overcome simply by increasing the toroidal electric field strength. However, there are two effects to hinder such an increase, they are:

• a large electric field implying a large consumption of the flux capacity of the transformer core. Since this capacity is limited, the consumption in the breakdown phase will influence the maximum duration of the ohmic plasma current;

• a large electric field giving rise to the production of so-called runaway electrons, which is generally considered to be unwanted.

The other options to improve the conditions of breakdown are the decreased Z_{eff} and B_V. There are quite a number of techniques to decrease Z_{eff} in HT-7U. Baking the vessel, glow discharge cleaning, ICRF (Ion Cyclotron Range of Frequency) cleaning discharge and pulse discharge cleaning are the techniques that can free absorbed and absorbed gases from the vacuum vessel. And additional techniques such as boronization, siliconization and carbonization are directed at the bonding of adsorbed atoms to the vessel. The bonded atoms will never be released again. even under an extreme condition of a plasma discharge, which is especially successful in deoxidization. In Reference [2] descriptions of these techniques were given.

B_V is the effect of misalignment of the toroidal-magnetic-field coils with the eddy currents in both the vacuum vessel and the supporting structure. The criterion of B_V is severe: for HT-7U the allowed B_V of 7 mT corresponds to less than 0.05 percent of the toroidal field. The B_V can be decreased by an accurate adjustment of the position of the magnetic field coils and by a precise choice of the winding configurations for all magnetic field coils.

The set-up course of tokamak plasma is in fact a course of collided-ionization snow-slide. When the quantity of ionized electrons is larger than that of the lost electrons, the snow-slide will occur and the plasma is set up. The course of collided-ionization was well described by the TOWNSOND snow-slide model, because there are very little free electron on the condition of weak pre-ionization [3]. So for electron plasma $\nu = 9\times10^5 E_r \sim 1.7\times10^6 E_r$. The total electron loss is mainly composed of diffision loss, diffusion-induced excursion loss, and excursion loss caused by device-varied field. The formulas for the ratio of these losses is given below respectively:

• Diffusion loss ratio: $\eta_D = 3\times10^4 E_r/a^2B^2$,

• Excursion loss ratio: $\eta_s = 7\times(\frac{E_r}{P}+400)\frac{E_r}{P}\frac{1}{aRB}$,

• Excursion loss ratio caused by device-varied field:

$$\eta_z = 3.5\times10^2\frac{E_r}{P}\frac{B_z}{B}\frac{1}{a},$$

where $E_r(V/\text{cm}) = V_\lambda/2\pi R$ is the induced electric field, a(cm), R(cm) is the plasma minor radius and major radius, respectively, V_λ is the loop voltage, $B(T)$ is the intensity of vertical magnet, $B_z(T)$ is the intensity of device-varied field, P(Pa) is the pressure of charge. By introducing the HT-7U design parameters (a = 50 cm, R = 190 cm, B = 2 T) into the above formulas, it is clear that the error field will affect the plasma breakdown, however, if $E_r = 0.005$ V/cm, and $B_z \le 7\times10^{-3}$ T , the breakdown will occur successfully in accordance with HT-7U poloidal

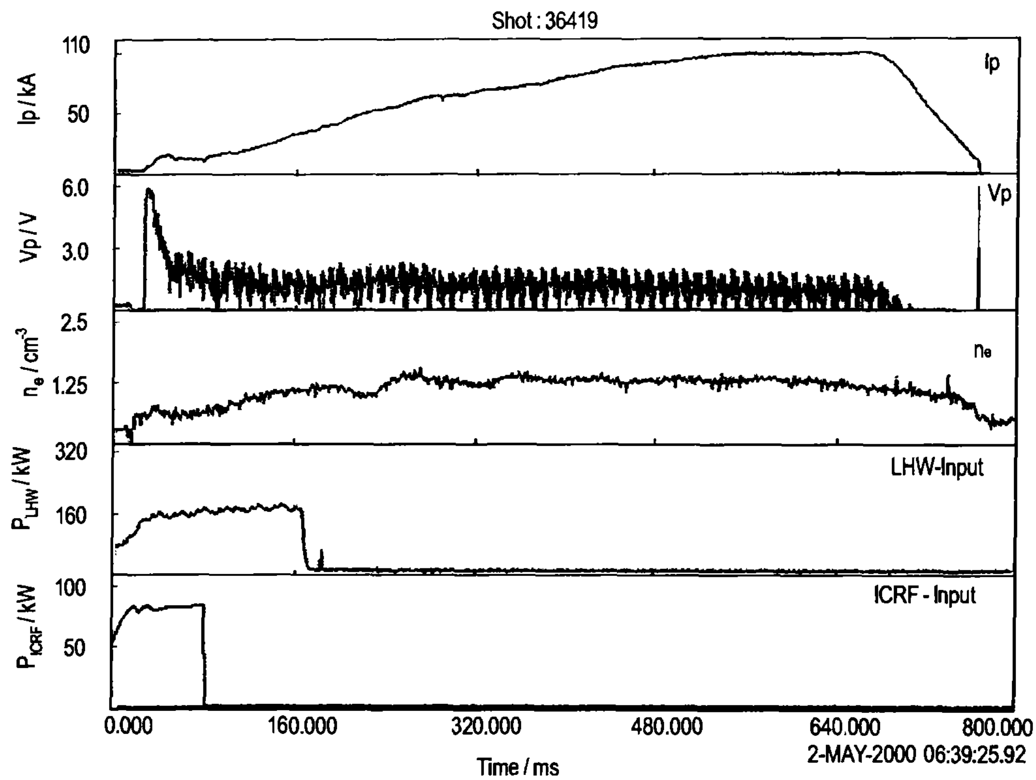

Fig.2 Typical lowloop voltage tokamak startup wave ICRF and LHW assistance.

field design, because the error field in the plasma region is less than 5×10^{-3} T.

In fact, there are impurity particles affecting the course of plasma breakdown, the ionization ratio is directly proportional to Z_{eff} and inversely proportional to E_r. The plasma quality will be poor if there are too many impurity particles in the vacuum vessel. In the PF coil design, with an inductive electric field at startup stage limited to a value of $E_r \approx$ 0.005 V/cm, the low loop voltage tokamak startup experiment has been tested in HT-7 superconducting tokamak [4]. With the assistance of ICRF and LHW (Lower Hybrid Waves) the low ramp-up rate of the plasma current, typically $\mathrm{d}I_p/\mathrm{d}t \sim 0.15$ MA/s was obtained in the experiments. After adjusting the pre-fill and error field which are necessary for these experiments, a successful ICRF (150 kW)- and LHW (200 kW)-assisted startup with $E \approx 0.005$ V/cm was achieved in the HT-7 superconducting tokamak. Fig. 2 shows the waveform of a typical shot with low-loop-voltage startup with the ICRF and LHW assistance. The constant preset current ramp rate had been maintained ($E \approx 0.005$ V/cm) until the final flattop value of $I_p \sim 0.1$MA was reached at a time of $t \sim 700$ ms. This technology will guarantee the safety of PF coils during the HT-7U operation period.

2.2 Design of a long-time digital analog composite integrator

Long-time integrator is one of the key devices of the PF control system. The measured magnetic signals of the tokamak must transfer from differential

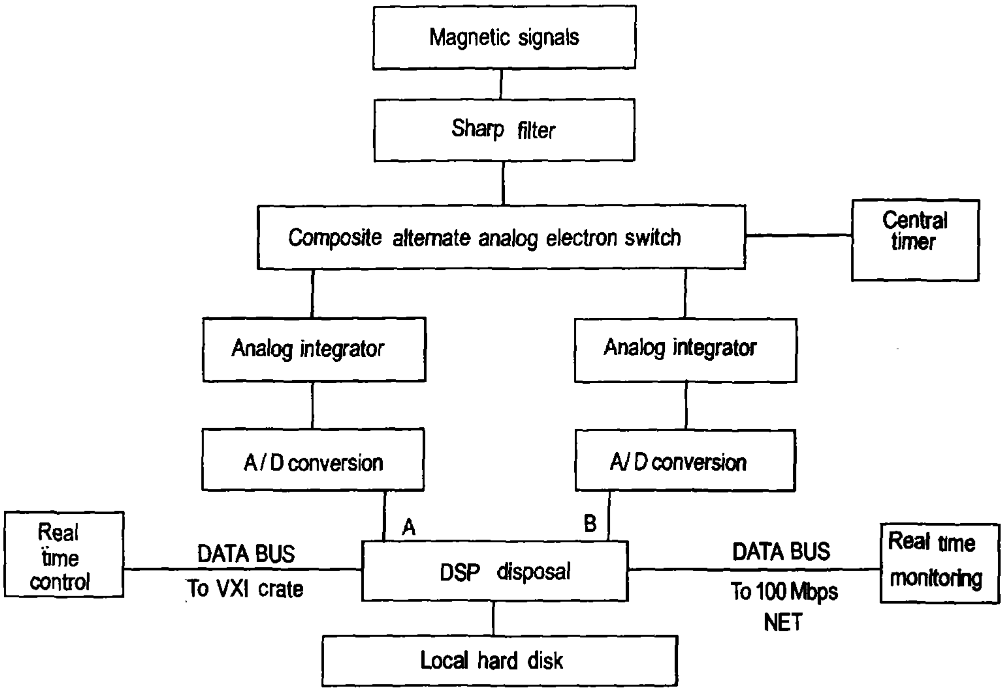

Fig.3 Long-time D/A composite integrator hardware frame.

coefficient to integral via the integrator. In the case of HT-7U, the analog integrator cannot be used, because of temperature-induced drift and unbalanced voltage. The full digital integrator is not only expensive but also poor in anti-jamming, and it is difficult to be applied in a steady-state tokamak. An analog and digital-combined method is designed for the composite integrator, which can provide a long-time integral with a lower (almost zero) drift. The basic idea of the method is that the signal is sent to two analog integrators with same parameters, in which both of them can measure signal plus drift in an alternate mode at a regular time interval, that is to say, within a given time interval, one works to measure signal plux drift, while simultaneously the other works to measure drift only. And next time, by turns, they will make exchange of the role they have played between themselves. And then all signals sent to A/D conversion board are managed by DSP, by which a specific program is required to obtain the calculated result without drift. Then the data are fed into two paths: one is led to a real-time display monitor, the other is to a storage so that they can be written into file to be kept in the archives and simultaneously sent to the power supply control system for the feedback control in real time. Fig. 3 shows the basic hardware frame of the facility.

2.3 The plasma shape and current control on HT-7U

The discharge shape of HT-7U tokamak is regulated by controlling the currents in the 12 poloidal field coils. The current in each coil is controlled by a separate power supply, so the power supply commands must be generated in each computation cycle of feedback system. About 100 magnetic diagnostic signals are digitized to provide input data. The control algorithm uses a predetermined linear mapping, implemented with a matrix multiplication between the input data and the discharge shape to determine the difference between the actual shape and the desired shape. In each cycle of the control system, one hundred of input are used to compute errors in the 12 aspects of the discharge shape that are to be controlled, and the values of the poloidal beta and internal inductance are used to choose the appropriate linear mapping. The errors are then used to compute the commands for the poloidal field power supplies. The control current in the coils will not be high between the error of the plasma shape and the defined range, so we can restrict the shape and the current of the plasma to our anticipated state in each control cycle. The optimal value of the PID controller parameter is determined by the simulation. The main goal of this system is to realize the control matrix. In other words, by using the error control method, the control matrix could be got by different parameter control arithmetic. According to the design model of HT-7U plasma shape and current, the control result and the respond time completely depend on the precision and speed of solving the control arithmetic, which will be realized by the use of advanced hardware at present such as high-speed VXI bus machine, embeded computer and DSP technology.

2.3.1 The principle of choosing the hardware with the PF control system

Tokamak is just like a simplified transformer, in which the primary side is the poloidal field coils with various currents and the other side is a cylindrical loop of plasma current with a distribution profile. The aim of the PF control is to control the plasma current profile in accordance with the requirement of physics experiments.

But up to now, there have been no reliable measuring methods to measure the distribution of the plasma current profile, the only way is to use the experimental data from measured magnetic signals to reconstruct the flux surface. According to the plasma equilibrium analysis theory, feedback controlling arithmetic is given to get stable numeric results based on the linearity repeat according to the physical principle. Every linearity repeat needs 10 seconds on HP-235 Unix workstation to perform a $(30\sim100)\times10^3$ matrix operation depending on the precision required. Such an operating speed cannot satisfy the need of real-time control and we must design hardware carefully, because the respond time must be less than 2 ms according to the experimental requirement. Whether the plasma shape and current feedback system can achieve a high speed in real time is based on solving such key problems as: (1) system control bus mode; (2) fast data transfer; (3) high-speed data disposal; (4) fast data exchange. In order to satisfy such requirement, we should consider advanced system frame, flexible compatibility of hardware related operating systems and related application programs.

a. The data transfer speed varies with bus mode. The VXI bus is excellent not only in capability but also in stability, and can guarantee the PF control system always at the advanced level of the plasma control.

b. In order to improve data transfer speed, the cable connection from the integrator to VXI bus control case will assure digital signals transfer within $1\mu s$ between the two devices. The reason for choosing such an original method is determined by the system response time, so that it can leave much time for data analysis of the system.

c. There are huge matrix operations on the error calculation in the control cycle. For comparing the calculated result with the preset result, the error data will be sent through the PID controller under the condition that the error is within a given scope. Due to huge repeated calculation, the DSP processor with a floating-point operation will be applied. Good result cannot be achieved just by one DSP processor, so we adopt four DSP processors to calculate separately and pass through Motorola 68030 to the RAM and will be managed by embedded computer VXIPC-870/450.

d. In order to assure system response time for HT-7U plasma feedback control system, a fast data

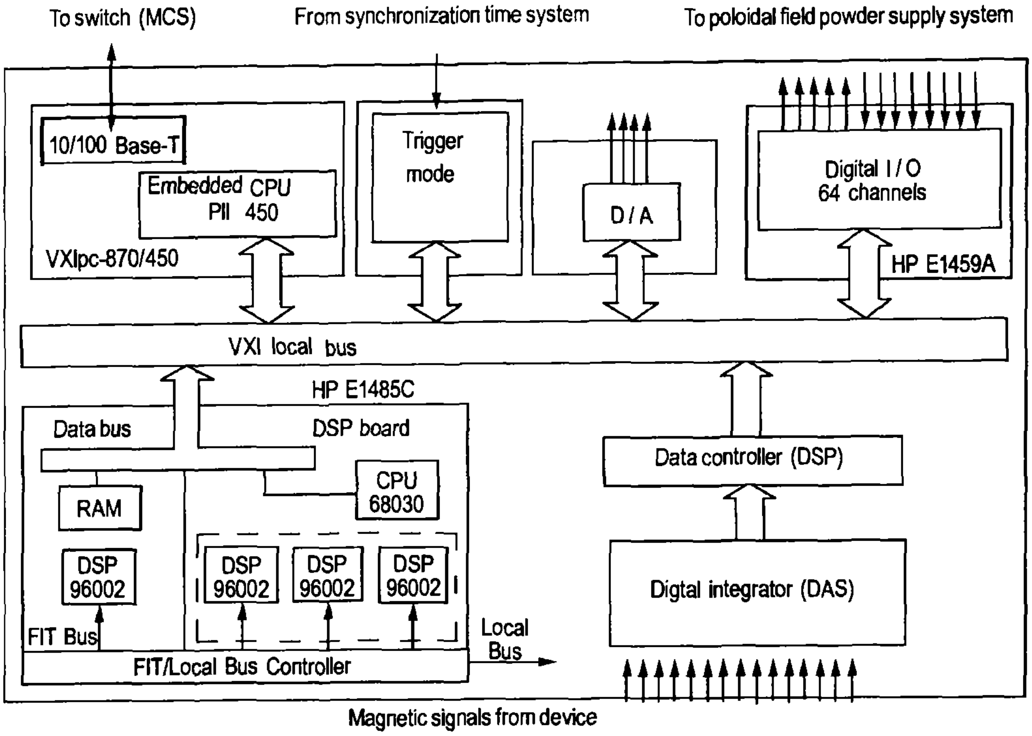

Fig.4 HT-7U plasma shape and current control system hardware frame.

exchange is needed. The data exchange between the DSP processors and host computer is achieved by the local bus, with a data transfer ratio of 100 MB/s reached.

2.3.2 PF control system hardware frame

In Fig. 4 is shown the hardware frame, in which the system consists of an embedded computer VXIPC-870/450 (as host computer), a DSP board HP E1485C (including central CPU Motorola 68030) a 256 M memory, a DSP unit (Motorola 96002), and an optic-isolated 64-channel digital I/O board HP E1459A. There is a central processor on the DSP board, with which the data exchange speed can be improved and the multi-DSP task assignment and management performance improved. Because there is a CPU on the HP E1485C board, the host computer is not in charge of the data exchange. In such a way the host computer will have enough time to analyze the error data and send the command to the power supply system.

2.3.3 The work flow chart for the PF control system

The flow process of plasma shape and current control system includes three parts: data readout, data transfer and data management. Data acquisition will adopt 500 K A/D boards, and there are eight analog input channels with a sampling precision of 12 bit, and a FIFO for each board. A time trigger is used to start a synchronized 100 channel data acquisition, and to integrate by the integrator in parallel, and then to send them to the VXI case. During the data transfer, a status signal is sent through the VXI bus. The central processor on the DSP board will detect the control line of the VXI bus, which will drive local bus as soon as the status signal is got from the VXI bus and read the data from FIFO to the board memory, at the same time it will start up a thread to detect another VXI status bus. The main function of this thread is to quarantee the data sending into local bus until the host computer completes one control cycle, and the thread will be closed automatically after completing one control cycle. Meanwhile, the data will be sent from data bus (20 ~ 40) MB/s to each DSP unit memory (4 MB) in parallel. The DSPs will operate in terms of prescribed arithmetic (it will be discussed in the third part of this paper), and the Motorola 68030 of the DSP will be in charge of the results. Then the correct results will be sent to the embedded computer VXIPC-870/450 to be compared with the preset value. The error within the defined scope will be sent to poloidal field power supply through the local bus, otherwise it will do the cycle again.

In this system, the main function of host computer is in charge of data acquisition, data storage and data transfer to data server via Ethernet, the central processor on the DSP board is in charge of data exchange and startup, management of DSP processor. The single DSP processor is in charge of data disposal. Every DSP operates with a parallel and independent arithmetic. As there is no communication between DSPs, the advantage of the hardware have not been fully utilized. Therefore, a new arithmetic is under development. The arithmetic is based on two basic hypothesis: (1) The plasma equilibrium parameters cannot change too fast. (2) The last control error can be used in the following several cycles. This arithmetic can be divided into two loops: one is the fast loop (< 2 ms) and the other is the slow loop (about 60 ms). The plasma can be controlled with the fast loop by feedback controlling and finely adjusted with the slow loop. The successful development of the arithmetic will improve the system response time, decrease the need for hardware and improve the capability/price ratio of the system.

3 Arithmetic of equilibrium feedback control in real time

Optimum performance of tokamak discharge requires an accurate feedback control of many discharge parameters, the value of which must be accurately measured. The values of many discharge parameters, such as the shape and safety factor profile, are not directly measured but can be evaluated from available diagnostic data, such as magnetic field and flux measurements. The most complete evaluation comes from a least squares fit of the diagnostic data according to the Grad-Shafranov model describing the force balance of the tokamak equilibrium, which is dependent on a distributed current source. This fully reconstructed equilibrium has been normally performed off the line using a fitting-sensitive code such as EFIT [5].

An approximate solution to the Grad-Shafranov equilibrium relation is found to be best fit for the diagnostic measurements so that an equilibrium solution consistent with the force balance, expressed in terms of the spatial distributions of the plasma current profile and poloidal flux, is available in real time for accurate evaluation of the discharge parameters. The algorithm is very similar to that of EFIT, and the equilibrium-reconstructing algorithm has been implemented on the digital plasma control system[6] for the DIII-D tokamak.

3.1 Isoflux control method

Shape control is implemented with an "isoflux" technique[7], with which a set of locations is specified to define the desired plasma boundary, and the poloidal field coil currents are adjusted to keep the poloidal flux equal at all of these locations.

The case of a single null divertor discharge shape is illustrated in Fig. 5, where the diamonds indicate the control points at the locations the value of the flux is controlled. Because the control is not based directly on errors of the shape, but rather on errors of the local poloidal flux at the control points, the requirements of the equilibrium-reconstructing algorithm are reduced since calculation of the plasma boundary is not necessary.

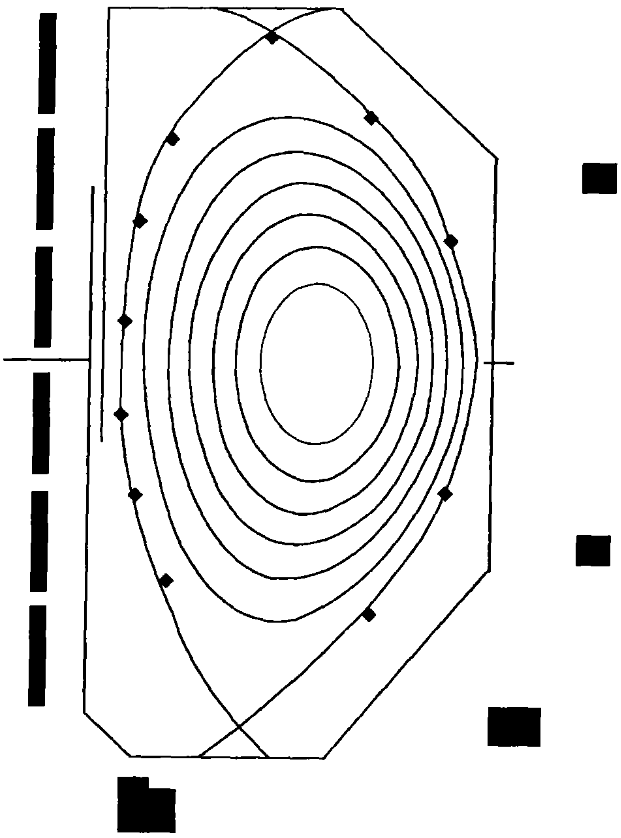

Fig.5 HT-7U shape "Isoflux" control sketch map.

In general, in the isoflux technique, some points will be set on the plasma boundary to adjust the poloidal coils current for keeping the poloidal flux equal to these setup values. In order to calculate the flux at the control points, the plasma current is taken as many current loops located on the rectangle grids. The total flux of the control point is produced by the following items: (1) the rectangle grids current (2) the current of the PF coils (3) vacuum-vessel current induced by the overall poloidal magnetic field. The response vector, multiple-response vector and loop current vector will be first calculated with the proportional coefficient (Green's Function) to get the value of the flux in real time. The required speed of shape control algorithm is mainly determined by the frequency response of the poloidal field power supply and the field-penetrating time. During this period, the loop will include:

- The diagnostic data acquired.
- The equilibrium reconstruction performed.
- The error value between the control-point flux and the reference flux.
- the calculated commands sent to coil power supply.

3.2 Real-time equilibrium arithmetic

Although shape control algorithm has satisfied requirement of most physics experiment, modern tokamak experiments request a higher precision performance. The real-time EFIT feedback control is indispensable to the HT-7U superconducting tokamak. The task of the equilibrium reconstruction algorithm is to compute the distributions on the R, Z plane of the poloidal flux Ψ and the toroidal current density J_t, providing a least squares best fit for the diagnostic data, which simultaneously satisfies the model given by the Grad-Shafranov equation: $\Delta^* \Psi_p = -\mu_0 R J_t(R, \Psi)$, where the total poloidal flux is $\Psi = \Psi_p + \Psi_{coil}$, where Ψ_p is the poloidal flux resulting from the plasma current and Ψ_{coil} is the poloidal flux generated by the current external to the plasma. The diagnostic data consist of the measured flux and field outside the plasma, plasma current from a Rogowskii loop, and current in the poloidal field coils. The equilibrium solution consists of values of Ψ and J_t on a rectangular grid that covers the entire area of vacuum vessel. The current is modeled as being distributed among a set of rectangular elements, in which the centered one is taken as a grid point. There are typically a total number of 1000 grid points or more. The large number of grids allows the solution to provide a realistic distribution of the current profile, including a finite current density at the discharge edge. The current density is modeled, however, by only a small number of free parameters: $J_t(R, \Psi) = R\left[p'(\Psi) + \frac{\mu_0 F F'(\Psi)}{4\pi^2 R^2}\right]$, where P is the plasma pressure, FF' is related to the poloidal current, with the prime indicating a total derivative with respect to poloidal flux. Simple polynomial models are used for $FF'(\Psi)$ and $P'(\Psi)$: $P'(\Psi) = \sum_{n=0}^{n_p} \alpha_n \Psi_N^n, FF'(\Psi) = \sum_{n=0}^{n_f} \gamma_n \Psi_N^n$, where $\alpha_J = \lfloor \alpha_0, \alpha_1, \cdots, \alpha_{np} \rfloor$ and $\gamma_J = \lfloor \gamma_0, \gamma_1, \cdots \gamma_{nf} \rfloor$ are the free parameters of the iteration for the plasma current profile, and the $\Psi_N = (\Psi - \Psi_{\text{axix}})/(\Psi_{\text{bdy}} - \Psi_{\text{axis}})$ is the flux normalized to the flux difference from the center to the edge of the discharge, Ψ_{axis} is the flux on the magnetic axis, and Ψ_{bdy} is the flux on the last closed flux surface. The most general approach to the fitting problem is to treat all toroidal current sources as unknown values. Thus, in addition to the free parameters in the parameterization of J_t, the current in the external poloidal field coils can be free parameters and, potentially, the induced currents in the vacuum vessel and support structures can be treated in this way as well. The reference flux value Ψ_{ref} is also treated as a free parameter with the measured value weighted by its uncertainty being used as a constraint. Thus the total vector of unknowns for the fitting problem is: $\vec{U} = [\vec{I_c}, \vec{\alpha_i}, \Psi_{\text{ref}}]$, where $\vec{I_c}$ is the vector of external current sources(coils and induced currents) and $\vec{\alpha} = [\vec{\alpha_J}, \vec{\gamma_J}]$ is the vector of unknowns in the plasma current parameterization. Here it iterates the solutions for Ψ and J_t until the change in Ψ between the two successive iterations is small.

In the real-time version of the equilibrium-reconstructing algorithm, the basic idea of this algorithm believes that the equilibrium evolves too quickly in a tokamak, and the change since the previous solution can be accounted for a few iterations, so that the result is sufficiently accurate for discharge control. The time-consuming process of iterating the calculated solution for a fixed set of diagnostic data is eliminated. Instead, for each new reconstructed equilibrium a new set of diagnostic data is acquired, the most recent equilibrium solution is used as the starting point, and one iteration is performed. If the equilibrium is not evolving too quickly, the change since the previous solution can be accounted for one iteration so that the result is sufficiently accurate for discharge control. If the equilibrium evolves so slowly that the diagnostic data do not change from one sample to the next, then this procedure would be the same as that of the off-line algorithm.

4 Summary

HT-7U PF control system has several advantages over other tokamaks in the world such as acquisition speed, acquisition mode, system rationality, and advanced appliance technology, with following merits:

a. This system adopts a parallel acquisition, while JET uses only a delay acquisition, with which the data can't be acquired simultaneously. Although parallel acquisition is used by DIII-D, but its acquisition speed is too slow to control its precision.

b. System design is suitable. The frame, hardware and the methods of HT-7U tokamak PF control system are more reasonable after investigating various system frames of other tokamak in the world to avoid congenital shortcomings.

c. The advanced appliance technologies are used, such as VXI technology, being a high-lighted new technology during the last decade of the twenty century, leading to assurance of high-speed, stabilization, good compatibility in the measurement and control.

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