REQUIREMENTS FOR DIAGNOSTICS IN CONTROLLING ADVANCED TOKAMAK MODES

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INTRODUCTION

Advanced tokamak modes are tokamak operation modes having a possibility to establish a concept of an attractive tokamak fusion reactor with a small size, high fusion power density and a capability of steady-state operation. Present tokamaks have realized some advanced tokamak modes such as a high β_p ELMy H-mode [1] and a negative magnetic shear mode [2]. These advanced modes are considered to be candidates for an operation mode in realizing the non-inductive steady-state operation in the International Thermonuclear Experimental Reactor (ITER) [3]. Advanced tokamak modes are accompanied with an internal and/or edge transport barrier forming a steep gradient on the density and the temperature profile. It is commonly understood that simultaneous optimization of the plasma pressure profile and the plasma current density profile, keeping the MHD stability, is required for obtaining and sustaining a high-performance plasma. Moreover, stabilization of MHD modes destabilized in a regime of high plasma pressure and high bootstrap current fraction is important.

Exploration of advanced tokamak modes in present tokamaks has shown that measurements of the plasma current density profile (safety factor profile), the pressure profile (density and temperature profiles of electrons and ions) and fluctuation measurement of MHD modes are indispensable for controlling the advanced tokamak modes. Measurement of plasma rotation profile is also important because transport barrier formation and occurrence of MHD modes relates closely to it. In obtaining and sustaining the advanced tokamak modes, control of the transport barrier is essential. Therefore, measurements must be done with enough spatial resolution to identify the location of the transport barrier and to control the plasma pressure gradient. Other important aspect of the diagnostics for controlling advanced tokamak modes is whether measurements can be utilized in real-time for the feedback control of the plasma.

In this paper, role and issues of the plasma control towards advanced tokamak modes,

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key diagnostics and requirements for diagnostics, plasma controls towards advanced tokamak modes in JT-60U, issues in developing diagnostics and feedback control schemes for controlling advanced tokamak modes will be presented.

ROLE AND ISSUES OF PLASMA CONTROLS TOWARDS ADVANCED TOKAMAK MODES

An "Advanced Tokamak Mode" would be defined as a tokamak operation mode, which has a possibility to establish a concept of an attractive tokamak fusion reactor with a small size, high fusion power density and a capability of a steady-state operation. These features are based on high energy confinement, high MHD stability, high bootstrap current fraction and high capability of heat and particle exhaust. A concept of a steady-state tokamak reactor has been developed. For example, following key physics elements are required to be realized simultaneously for the Advanced Steady-State Tokamak Reactor (A-SSTR); H-factor to L-mode ~2.7, a ratio of averaged electron density to the Greenwald density n_e/n_{eGW} ~1.2, normalized bate β_N ~ 3.7, bootstrap current fraction ~ 0.8, non-inductive current fraction ~1, fuel purity ~ 0.85, radiation power to heating power ~0.95 [4]. Role of plasma controls towards advanced tokamak modes is to establish a stable and well-reproducible route to a reactor-grade plasma and sustain it at the highest performance in a steady state.

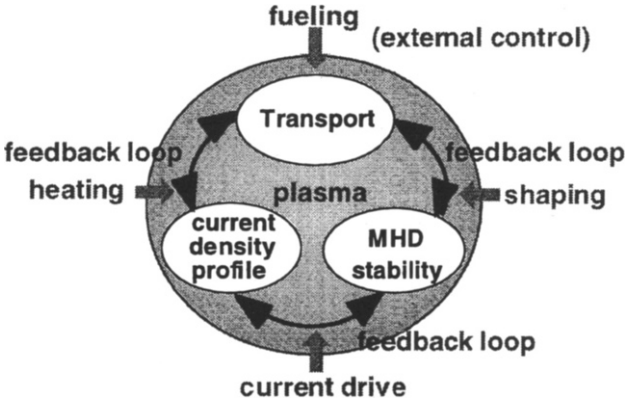

Figure 1. A schematic diagram of mutual interactions between the confinement, the current density profile and the MHD stability in a tokamak plasma and plasma control using methods of heating, current drive, shaping, fueling.

Figure 1 shows a schematic diagram of mutual interactions between the confinement, the current density profile and the MHD stability in a tokamak plasma [5]. A change in the transport characteristics modifies the pressure gradient of the plasma and it affects the bootstrap current. The changed current density profile affects the MHD stability, and the pressure profile modified due to the change in the transport affects the MHD stability directly. In a real plasma, many internal feedback loops act mutually and they determine

plasma behaviors. In a case, the internal feedback act in one direction and it leads to a plasma disruption. Therefore, simultaneous optimization of transport, MHD stability and current density profile is required using external controls of heating, current drive, shaping and fueling. Compatibility of a high performance plasma with divertor is also required in a high fusion power plasma.

KEY DIAGNOSTICS AND REQUIREMENTS FOR CONTROL DIAGNOSTICS

A large fraction of the bootstrap current induced by the pressure gradient of the plasma is necessary to establish a concept of an attractive steady-state tokamak reactor. Candidates of the advanced tokamak modes are high β_p ELMy H-mode [1] and negative magnetic shear mode [2] etc. having internal transport barrier and/or edge transport barrier (ITB/ETB). Figure 2 shows profiles of the electron density, the electron temperature, the ion temperature and the safety factor in the negative shear plasma on JT-60U [6]. In a shadowed region density and temperature profiles have a steep gradient due to the ITB formation, which is closely connected to the MHD stability. Therefore, control of ITB/ETB under constraint of the MHD stability is essential in controlling advanced tokamak modes and it requires key diagnostics. The safety factor profile is the most important for producing advanced tokamak modes because it affects significantly the optimization of the plasma confinement, the current drive capability and the MHD stability. The motional Stark effect polarimeter [7] used widely in present tokamaks is a first candidate of the measurement. The Thomson scattering measurement, interferometer, electron cyclotron emission (ECE) measurement and charge-exchange recombination spectroscopy (CXRS) are necessary to measure the electron density and temperature profiles and ion temperature profile, on which a result of the plasma control appears directly. In sustaining the high performance plasma with a large fraction of the bootstrap current (high poroidal beta β_p) and a high toroidal beta β_t, internal structure measurement of MHD modes such as the neoclassical tearing mode (NTM) [8] and the resistive wall mode (RWM) [9] is required to suppress these modes.

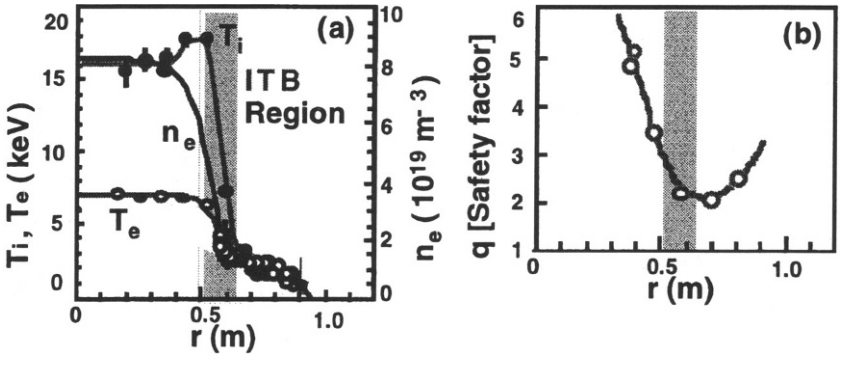

Figure 2. Profiles of (a) the electron density, the electron temperature, the ion temperature and (b) the safety factor in the negative shear plasma on JT-60U.

For control diagnostics, following additional conditions are required, compared to those used for physics understanding because a plasma disrupts at a large stored energy in the worst case if the plasma control is missed.

(1) High reliability and accuracy: Control diagnostics must be used routinely with high accuracy and without being affected by plasma phenomena.

(2) Sufficient time and spatial resolution: Simple scalars and complete spatial profiles must be measured with sufficient time and spatial resolution for physics phenomena to be controlled. For example, a growth rate $\gamma \sim (10\,\mathrm{ms})^{-1}$ for the RWM and an ITB width ~a/10 is an important time and spatial scale to determine the time and the spatial resolution of the diagnostics.

(3) Real-time processing of acquired data: Acquired data must be processed in real-time to establish real-time and active feedback control loops.

PLASMA CONTROL TOWARDS ADVANCED TOKAMAK MODES IN JT-60U

Real-time feedback control schemes in JT-60U

Real-time feedback control schemes of physics quantities presently used or under development for creating and sustaining advanced tokamak modes in JT-60U is listed in Table 1. Used quantities, objectives and applications are summarized in it [10]. In addition to scalars and a single point data of a profile, utilization of a part of the electron temperature profile and the current density profile are being explored.

Quantity	Objectives	Applications
•Averaged electron density (FIR interferometer)	•Optimize target and operation density	•Versatile in a variety of exp.
•Neutron emission rate (fission chamber)	•Control stability, fusion burn	•Establish route toward high Q_{DT}^* in R/S plasmas
•Plasma stored energy (diamagnetic loop)	•Control stability, avoid disruption	•Sustain plasma performance in a steady-state
•Central electron temp. (ECE measurement)	•Optimize target temperature	•Study non-inductive CD, High Te / Ti confinement
•Temperature gradient (ECE measurement)	•Control internal transport barrier	•Sustain bootstrap current for a steady-state operation
•Temperature fluctuations (ECE measurement)	•Suppress neoclassical tearing mode	•Sustain high performance in a steady-state
•Current density profile (MSE polarimeter)	•Sustain minimum q above 2	•Sustain a R/S plasma stably
•Main plasma radiation (bolometers)	•Improve confinement by enhanced radiation	•Explore Radiative Improved mode
•Outer gap distance (magnetics & equilibrium)	•Keep stable RF coupling, investigate wall effect on stability	•Sustain R/S by LHRF heating, improve stability in R/S plasmas
•Divertor pressure (ionization gauge)	•Control MARFE/detachment	•Sustain radiative divertor
•Divertor radiation (bolometers)	•Reduce divertor heat load	•Explore radiative divertor

Table 1. Real-time feedback control schemes of physics quantities in JT-60U.

Feedback control of ∇T_e in a negative shear plasma

Figure 3 shows a feedback control of ∇T_e in a negative shear plasma on JT-60U [11]. This technique has been developed to sustain the ITB by controlling a gradient of the

electron temperature at the ITB, aiming at sustaining the bootstrap current constant towards a steady-state operation. The plasma stored energy W_{dia} is controlled to produce internal transport barriers in a early phase. Increases in ∇T_e measured with ECE Fourier transform spectrometers shows the formation of the ITB in the electron system. After the formation of the ITB, a control logic is changed form the W_{dia} -control to the ∇T_e –contorl by the central heating using neutral beams. When the ∇T_e becomes lower than the reference value, the neutral beam power is increased. The ∇T_e is successfully kept almost constant with a similar profile to that at 6.5 s. The ECE Fourier transform spectrometers have been replaced recently by the ECE radiometers in the control loop. It is because electron temperature measurements with the Fourier transform spectrometer is affected by edge localized modes (ELMs) during a scan of the mirror in the spectrometer.

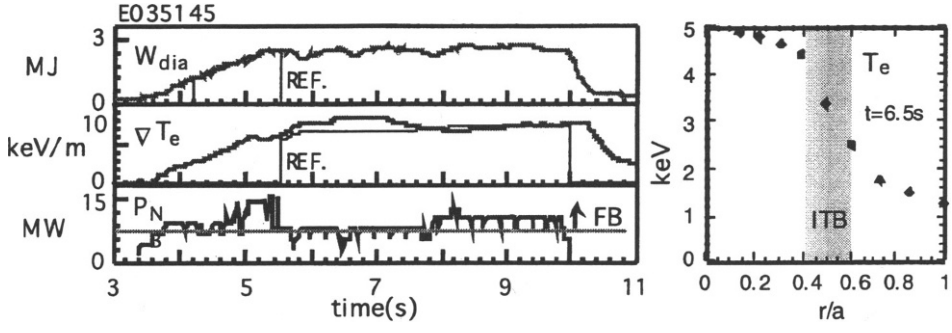

Figure 3. (a) Waveforms of ∇T_e feedback control applied to the ITB control in the negative shear plasma on JT-60U and (b) electron temperature profile at 6.5 s.

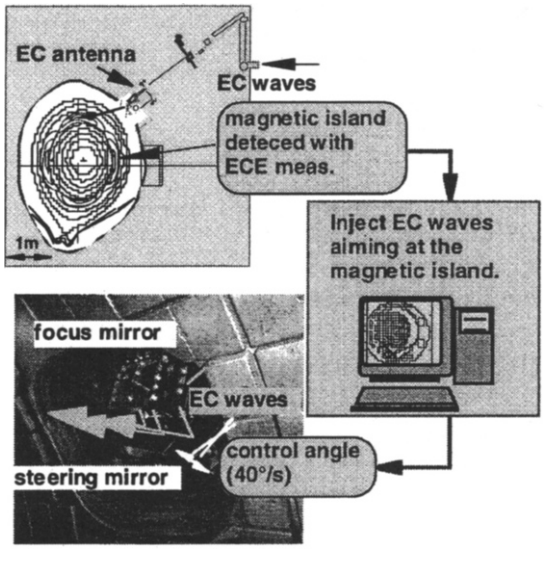

Figure 4. Schematic diagram of a complete loop of the NTM suppression by detection with ECE radiometers and EC wave injection.

Suppression of the NTM with ECE measurements and EC wave injection

Suppression of the NTM is one of important issues in a plasma with high beta and high bootstrap current and has been succeeded with the electron cyclotron (EC) wave injection in ASDEX [12], DIII-D [13] and JT-60U [14]. A complete loop, in which the location of magnetic islands of the NTM is identified by ECE radiometers and EC waves are injected aiming at the magnetic island by controlling the steering mirrors of the EC antenna, is under development in JT-60U. Figure 4 presents the control loop schematically.

Time traces of magnetic fluctuations of the NTM and the peak perturbation of electron temperature are presented on the left and profiles of the electron temperature perturbation are shown on the right in Figure 5. The electron temperature profile is measured with ECE radiometers at every 20 μs. Acquired data of eight channels around the magnetic island are processed in real-time at every 2 ms to obtain a perturbation profile of the electron temperature. When the NTM grows, a clear "notch" indicating the center of the magnetic island is detected in profiles of perturbation level of electron temperature (4-th channel is excluded in the figure because of a large detector noise) [15]. The injection of EC waves was successfully triggered when the peak perturbation exceeded the reference level of 2%. The complete loop will be tested with high power EC wave injection for the current drive to suppress NTMs in a high β_p ELMy H-mode plasma.

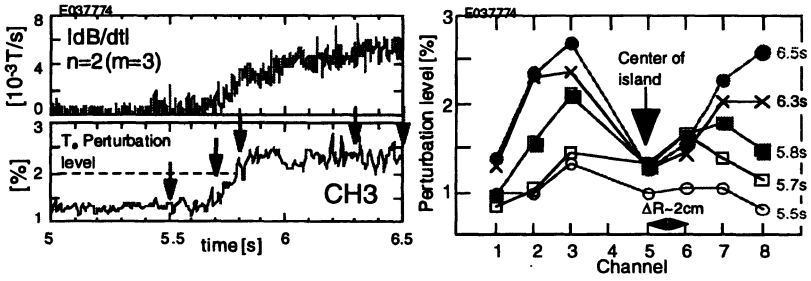

Figure 5. Time traces of magnetic fluctuations of the NTM and peak perturbation of the electron temperature around the magnetic island and profiles of the electron temperature perturbation.

Development of real-time detection method of q_{min} and r_{min} in negative shear plasma

A concept of the steady-state tokamak reactor requires the whole plasma current to be driven non-inductively with the bootstrap current of a large fraction (~80%), which forms a negative shear region in a core of the plasma. The safety factor profile must be compatible with the pressure profile having a large gradient. Therefore, in order to sustain such a plasma continuously, the information on the safety factor profile is indispensable in real-time although q profile can be obtained fairly accurately using the MSE data and magnetic data in a equilibrium code after the discharge is over. In JT-60U, a detection method of minimal value of the safety factor (q_{min}) and the location on minor radius (r_{min}) is under development for the real-time control of the safety factor profile. Figure 6 shows temporal behavior of q_{min} and normalized minor radius r_{min}/a. Filled circles show those evaluated with the equilibrium code using the data from the MSE polarimeter and the magnetic data. Solid waveforms present those obtained only from the MSE polarimeter by identifying the location of measurement points on one typical equilibrium of a similar

discharge. Temporal behaviors of values evaluated with two methods agree relatively well even though difference of ~1 for q_{min} and ~0.1 for r_{min}/a (~10 cm) can be seen between them. The difference is considered to be due to a discrepancy in the position and the shape of the real plasma from those referred to map the measurement position. Spikes in solid waveforms arise from the plasma light and the turn-off of the neutral beam used for the MSE polarimeter. It is important to improve the accuracy and the reliability by solving the problems in further developments.

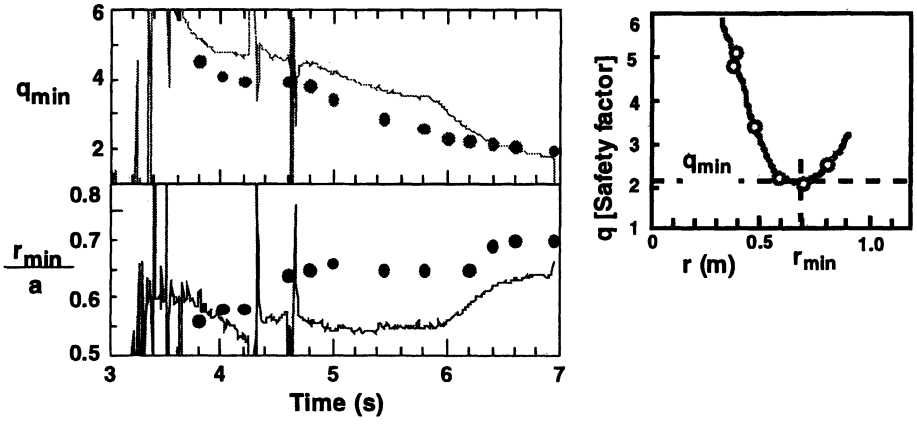

Figure 6. Time traces of q_{min} and r_{min}/a. in the negative shear plasma on JT-60U.

ISSUES IN DEVELOPING DIAGNOSTICS AND FEEDBACK CONTORL SCHEMES FOR CONTROLLING ADVANCED TOKAMAK MODES

Various feedback control schemes have been developed in present tokamaks and have contributed to create and sustain advanced tokamak modes. However, further developments are required in order to establish an operation mode, which can be utilized as a reactor plasma. Following items are issues in developing diagnostics and feedback control schemes for controlling advanced tokamak modes;

(1) Improvement of the reliability of diagnostics to be used as control diagnostics,
(2) Production of diagnostic data available in real-time to control systems,
(3) Need for profile data to be built into plasma control systems for the fine control of advanced tokamak modes. For example, $q(r)$, $n_e(r)$, $T_e(r)$, $v_\phi(r)$, etc,
(4) Establishment of various control algorithm based on physics basis of advanced tokamak modes.

SUMMARY

The most important in creating advanced tokamak modes is to control internal and edge transport barriers, optimizing the confinement, the MHD stability and the current density profile simultaneously. This requires key diagnostics. Measurements of plasma current density profile and MHD mode structure are the most important. High reliability

and accuracy, sufficient time and spatial resolution, real-time processing of acquired data are required for control diagnostics. Successful demonstrations of advanced tokamak modes have been shown in present tokamaks. Further improvements are required both in diagnostics and physics understanding for the fine control at the highest performance of advanced tokamak modes.

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