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TECHNOLOGIES TO OPTIMIZE ADVANCED TOKAMAK PERFORMANCE

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ABSTRACT

Commercial fusion power systems must operate near the limits of the engineering systems and plasma parameters. Achieving these objectives will require real time feedback control of the plasma. This paper describes plasma control systems being used in the national DIII-D advanced tokamak research program.

INTRODUCTION

Development of a broad range of technologies will be required to produce a commercially attractive fusion power system. These technologies include materials to withstand the harsh fusion environment, remote maintainance methods, fusion blanket and tritium breeding systems, improved magnetic conductors, and advances in active plasma control technologies. Plasma control technologies that optimize the performance of advanced tokamak systems are the subject of this paper.

Optimizing advanced tokamak performance requires simultaneous control of a multitude of coupled plasma characteristics. Figure 1 illustrates the interaction between transport control, magnetohydrodymics (MHD) stability control, and current profile control. Control of one element impacts others. Such a multiply interactive system requires detailed understanding as well as the ability to control plasma properties and profiles. We describe here control technologies being developed and employed by the DIII-D advanced tokamak research program. These systems include sensors [plasma diagnostics (1)] as well as activators [plasma control systems (2)] that are integrated in a multiple-input multiple-output digital plasma control system (3).

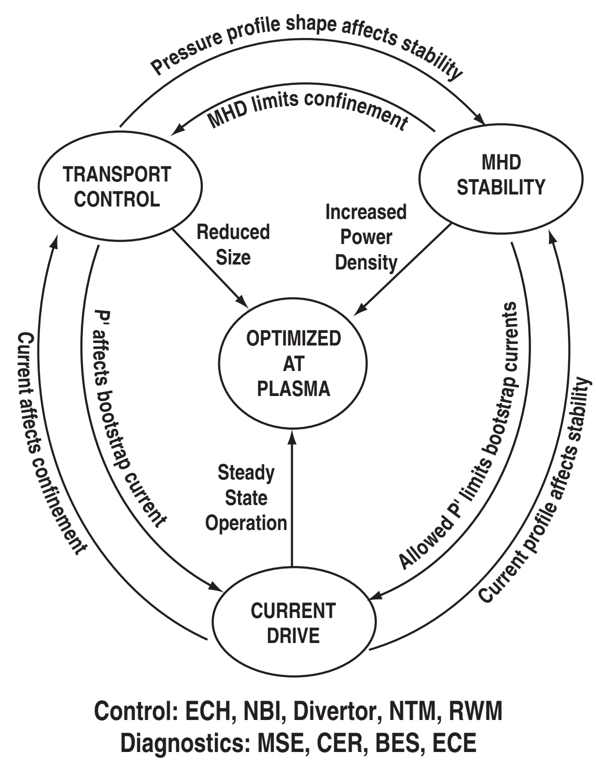

Fig. 1. Illustration of the coupling between plasma transport, MHD stability and current drive that requires technologies to optimize advanced tokamak performance.

Conceptual fusion power plant system design studies indicate a number of possibilities for an advanced tokamak power system. Two designs (4,5) are compared with ITER in Table I. Both designs are similar in size and plasma current to ITER. However the output power is significantly higher because of higher field magnets (A-SSTR2) or because of higher plasma pressure as indicated by the normalized beta (ARIES-AT). Both power plant designs, and especially ARIES-AT, will require plasma control technologies well beyond those in common usage today. To operate near the highest power will require a firm understanding of the upper limits of plasma parameters as well as the characteristic responses of the plasma control systems. This understanding can be incorporated in models, as illustrated in Fig. 2, which can be benchmarked by existing experiments and applied to future experiments such as ITER and eventually to power plants. In order to develop robust models a wide range of world experiments is needed for validation.

Table I
Commercial Tokamak Power Plant Designs are ITER Size But Require Higher Magnetic Field or Higher Plasma Pressure

	ITER	A-SSTR2* Japan High B	AIRES-AT† U.S. High β_N
Major radius, R (m)	6.2	6.2	6.0
Plasma current, I_p (MA)	10	12	13
Fusion power, P (GW)	0.4	4.0	1.8
Magnetic field, B (T)	5.3	11.0	5.9
Normalized beta, β_N	3.5	4.0	5.4

*Reference 4.
†Reference 5

The DIII-D integrated plasma control system that enables advanced tokamak operation is illustrated in Fig 3. Actuators and sensors that are operative and those under development are indicated. For example, to optimize the plasma beta, but remain below the beta limit, the auxiliary heating power is controlled. The beta sensor is an array of magnetic diagnostics that provide inputs to a real time magnetic analysis code (RTEFIT). The code output is used to regulate the heating power. In the event that the MHD stability limit is about to be exceeded a gas jet system quenches the plasma to prevent a disruption. Real time motional Stark effect (MSE) current profile data is under development. This will increase the precision of the determination of the plasma beta stability limit. In the future the electron cyclotron heating (ECH) system will optimize in real-time the plasma current profile to allow the plasma beta to be increased up to its maximum stable value.

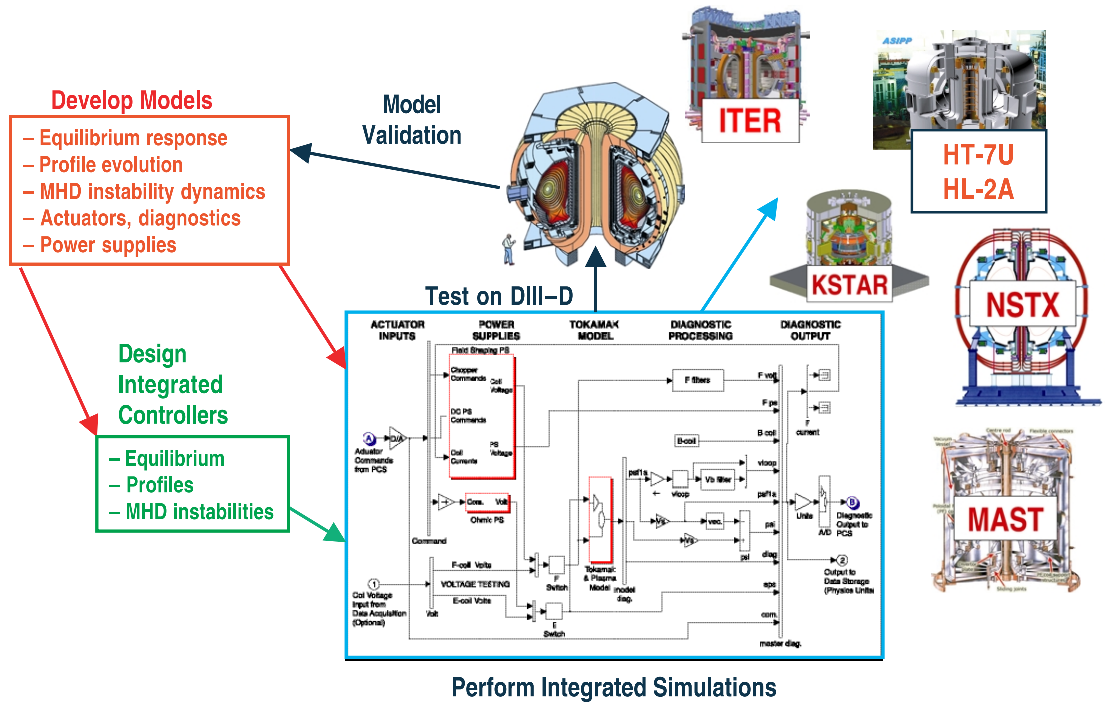

Fig. 2. Model validation on existing experiments will enable development of plasma controllers for future experiments.

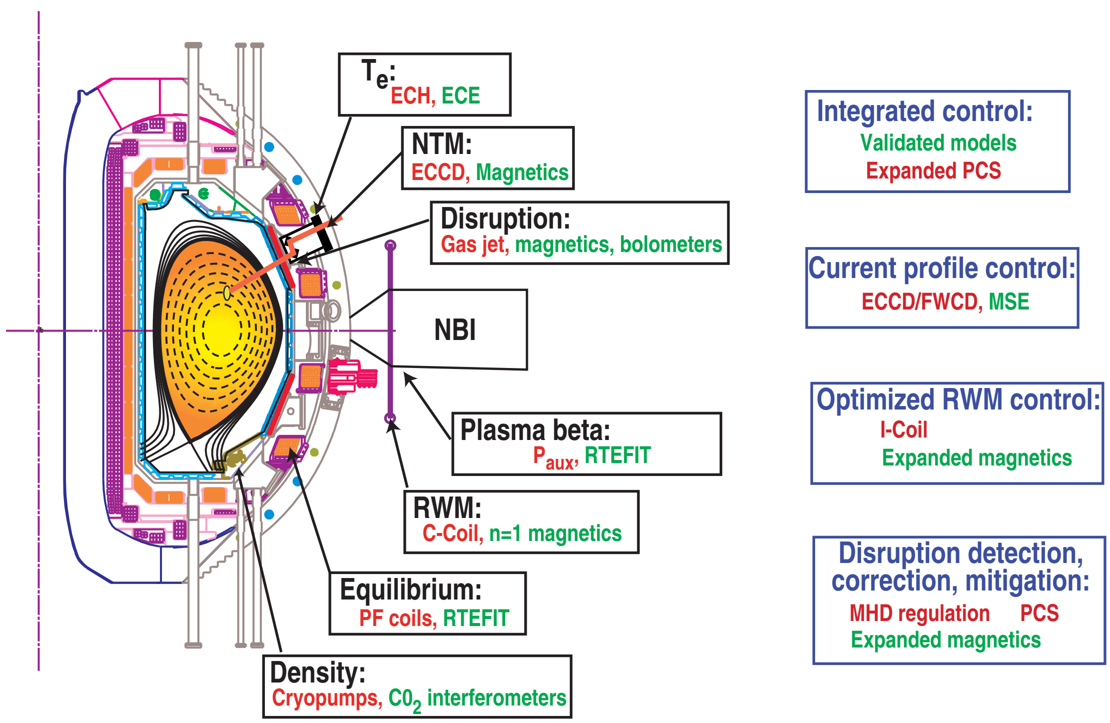

Fig. 3. Illustration of how integrated plasma control enables advanced tokamak operation in present DIII-D experiments along with future diagnostics and actuators currently under development.

PLASMA DIAGNOSTIC SENSORS

Examples of DIII-D plasma diagnostic sensors are described in this section. These include plasma temperature profiles, plasma rotation profiles, plasma current profiles, and plasma fluctuation profiles.

Electron temperature profiles are measured with three GA Thomson scattering systems (central, edge, and divertor). Radial profiles can be measured simultaneously with eight multipulse Yag lasers. Electron temperature profiles are also measured with a multichannel ECE system.

Ion temperature and rotation profiles are measured with a charge-exchange recombination (CER) system (6). In order to obtain fine resolution at the plasma edge and to measure toroidal and poloidal rotation profiles three neutral beams are used as shown in Fig. 4(a). This system measures a total of 40 radial positions with 0.5 to 2.0 cm spatial resolution and with 0.32 ms time resolution. Shown in Fig. 4(b) is a rack of four GA-developed spectrometers, that provide measurements from 16 radial channels.

The plasma current and electric field radial profiles are measured with a MSE system (7). A schematic of the Lawrence Livermore National Laboratory (LLNL)-developed system is shown in Fig. 5(a) together with an example of measured data. This 36-channel system views a neutral beam from three different ports to obtain wide radial coverage as well as to obtain differing viewing angles. The channel spacing is 4 to 5 cm in the core and 3.5 mm at the plasma edge. The time resolution is about 1 ms. A photo of a detector module with 8 channels is shown in Fig. 5(b).

The control of transport requires knowledge of plasma profiles as well as measurements of plasma turbulence. One fluctuation diagnostic is the University of Wisconsin beam emission spectroscopy (BES) system (8) illustrated schematically in Fig. 6. Images of density fluctuations are shown in Fig. 7. Presently such data is obtained only after detailed off-line analysis. In the future it may be possible to obtain such information in real time to effect control on plasma transport rates.

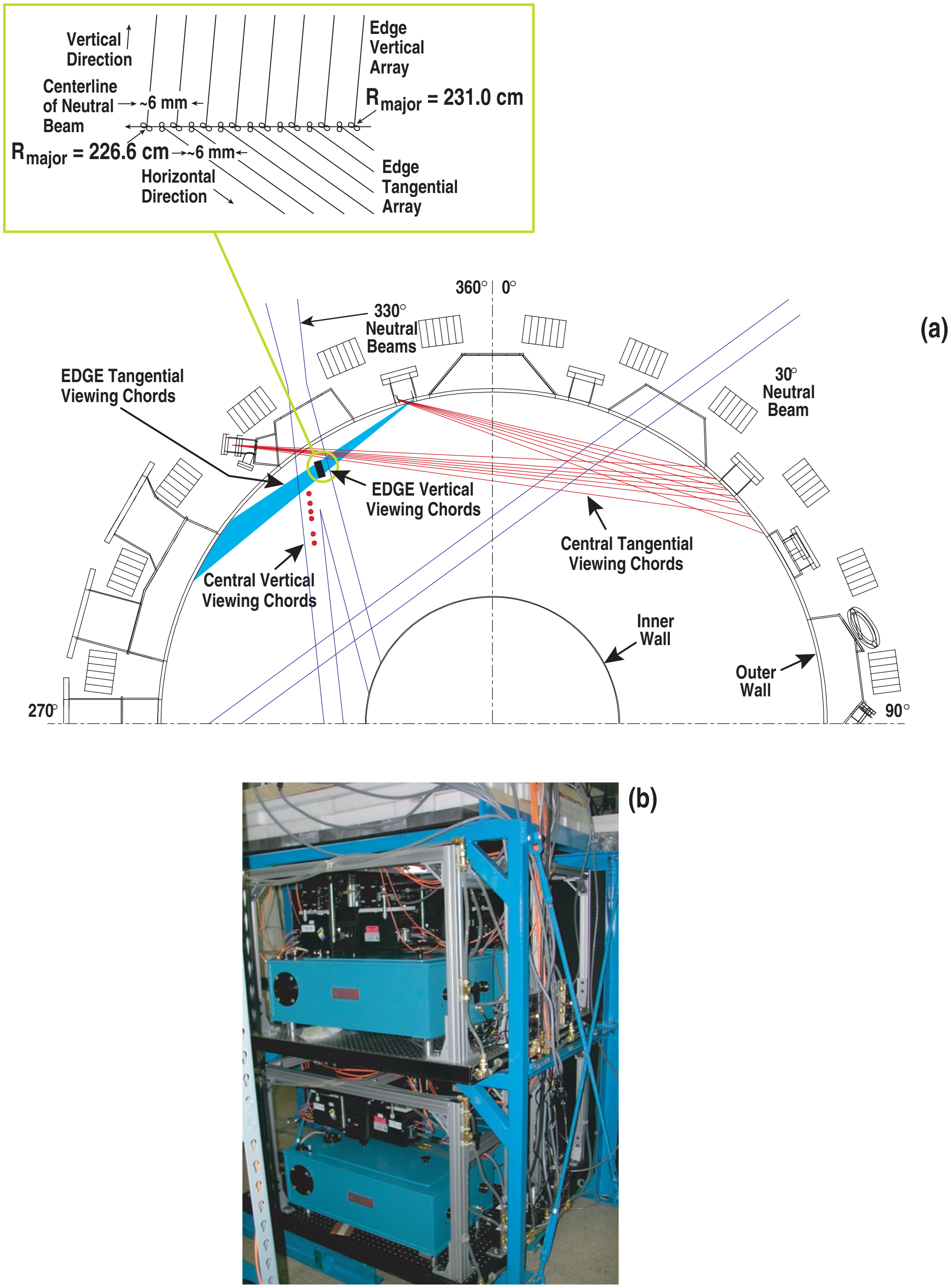

Fig. 4. Charge exchange recombination diagnostic measures ion temperature, ion rotation, impurity ion density and temperature, and plasma turbulence: (a) neutral beam geometry, (b) photo of four spectrometers, which provide 16 radial channels total.

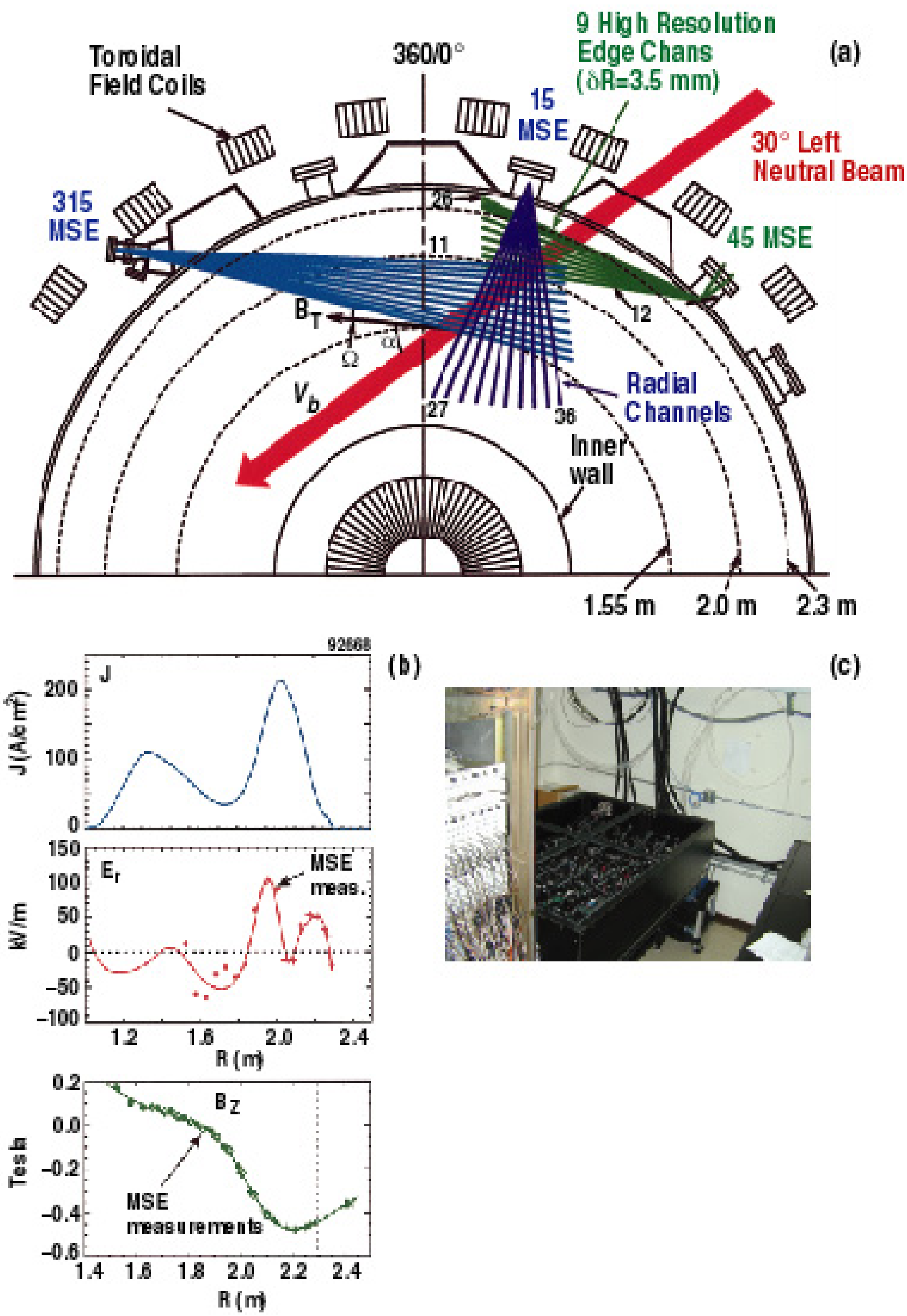

Fig. 5. Motional Stark effect diagnostic: (a) viewing geometry and typical data, (b) photo of MSE detector system.

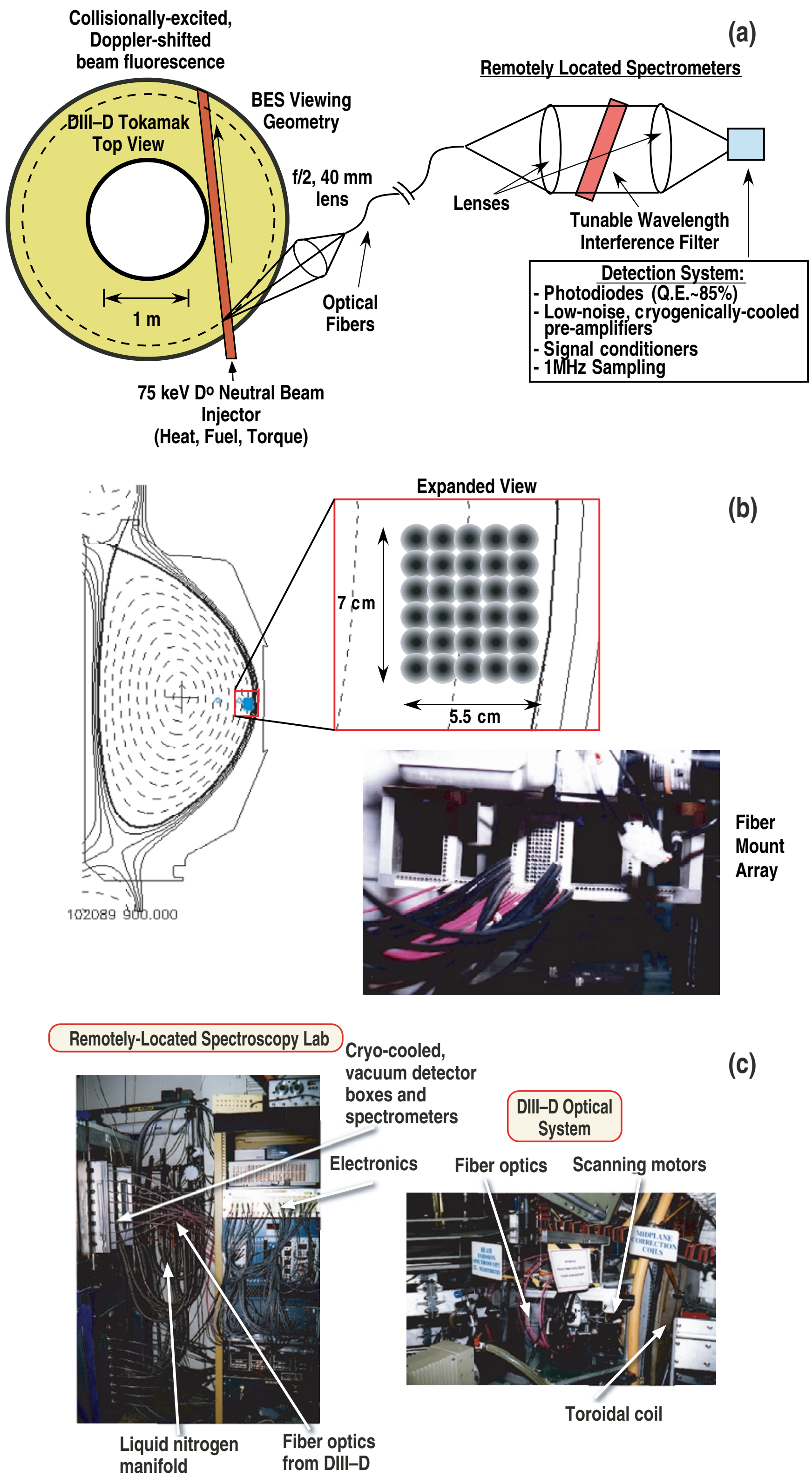

Fig. 6. Beam emission spectroscopy measurement of density fluctuations: (a) viewing geometry, (b) imaging geometry, and (c) photo of BES instrumentation.

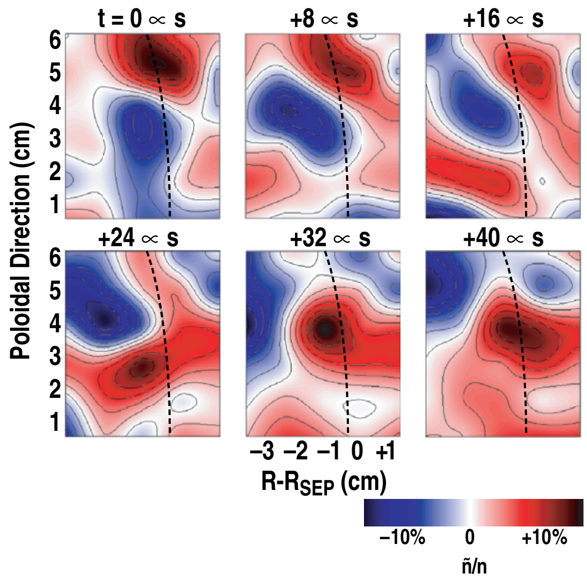

Fig. 7. Two-dimensional BES images of density fluctuations show complex, nonlinear interactions of turbulent eddies.

DIII-D PLASMA CONTROLLERS/ACTUATORS

DIII-D plasma control is facilitated using divertor pumping, magnetic feedback, neutral beam injection, ECH and electron cyclotron current drive (ECCD), and ion cyclotron heating (ICH).

The divertor provides control of a several plasma parameters. These include control of particles by varying the amount of pumping, control of the characteristics of the power flow by varying the divertor separatrix location, and control of the edge localized mode (ELM) characteristics by varying the plasma triangularity. The DIII-D divertor geometry (2) is illustrated in Fig. 8. Divertor research is aimed at determining the optimum configuration for advanced tokamak performance as well as to increase understanding of divertor physics and materials studies. To this end the upper divertor has higher triangularity to facilitate advanced tokamak performance while the lower divertor has lower triangularity allowing it to be heavily diagnosed.

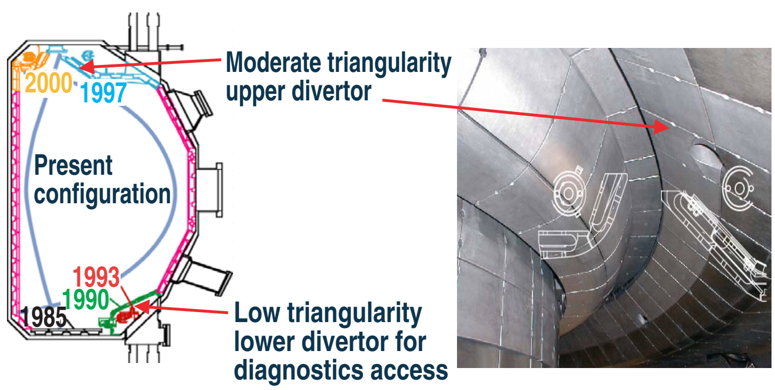

Fig. 8. Schematic and photo of the DIII-D divertor, which optimized plasma shape flexibility as well as power and particle control.

DIII-D magnetic control consists of feedback control of plasma position, plasma shape, separatrix location, divertor power loading and particle strike point location. This control is facilitated by a segmented central solenoid that produces no stray field in the plasma region and poloidal field coils located within the toroidal field coils. While not suitable for a fusion power plant, the large number of poloidal field coils enables simulating a wide range of coil configurations to evaluate potential future tokamak designs. A more recent addition to DIII-D magnetic control are resistive wall mode control coils (9) as a collaboration between Columbia University, Princeton Plasma Physics Laboratory (PPPL), and General Atonics (GA). As shown in Fig 9 there are six external saddle coils at the midplane and twelve internal saddle coils; six above and six below the midplane. The coils are powered with 5 kA of current at frequencies up to 1 kHz. These toroidally and poloidally localized coils counter act magnetic perturbations generated by resistive wall modes in high beta plasmas. Feedback control of these coils has enabled operation up to the ideal MHD beta limit. These coils also are used to correct small magnetic errors, which impede plasma rotation.

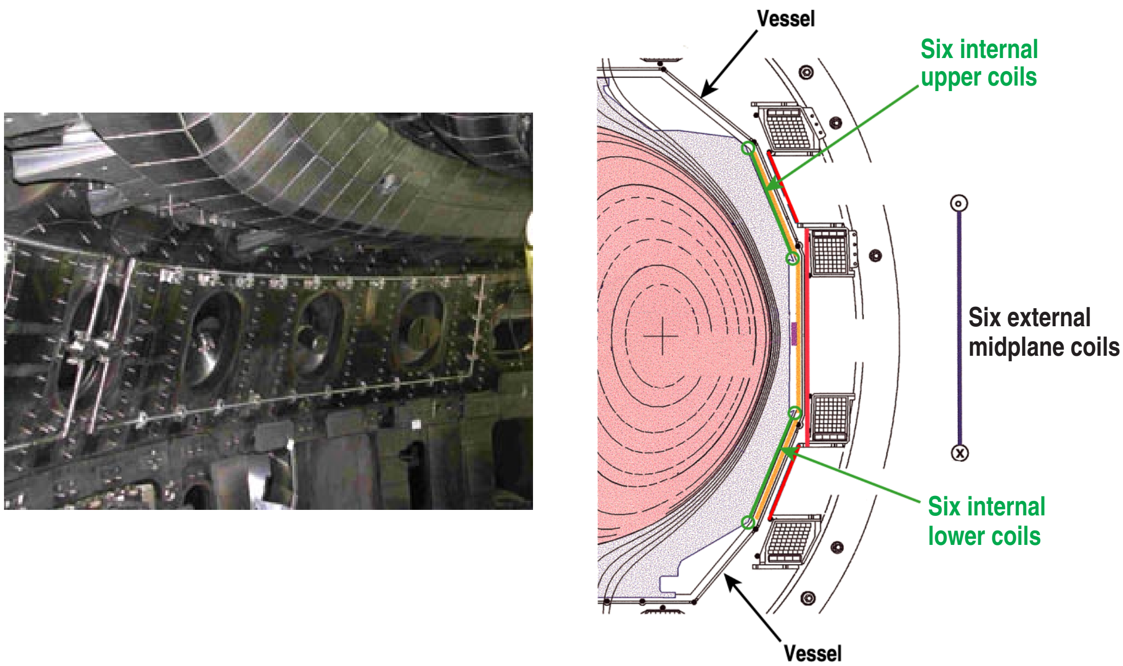

Fig. 9. DIII-D resistive wall mode control coil array.

Neutral beams are used extensively for feedback control in advanced tokamak experiments. The beam power is regulated to stay within MHD stability limits, the beam timing is controlled to best establish advanced tokamak operation, beams provide a source of rotational momentum, and as described above, beams enable a wide range of diagnostics that are used in plasma control. The DIII-D neutral beam system (10) consists of four beam lines, all oriented for tangential injection in roughly the same direction. Each beam line contains two 80 keV injectors injecting 2.5 MW each. Figure 10(a) illustrates the key components of each beam line.

DIII-D utilizes ECH for precise spatial and temporal heating and current profile control. ECH power absorption can drive localized current. Changing the current profile can optimize MHD stability and modify local transport rates. The DIII-D system (11) consists of six 110 GHz ECH gyrotrons rated at 1 MW maximum power. Three gyrotrons are 2 s units from Gycom and three are long pulse units from CPI. The gyrotron system is located away from the tokamak and connected via high efficiency GA-developed transmission lines. The system layout is illustrated in Fig. 11(a,b). The microwave power is directed toward the plasma with launchers, which have poloidal and toroidal steering capabilities

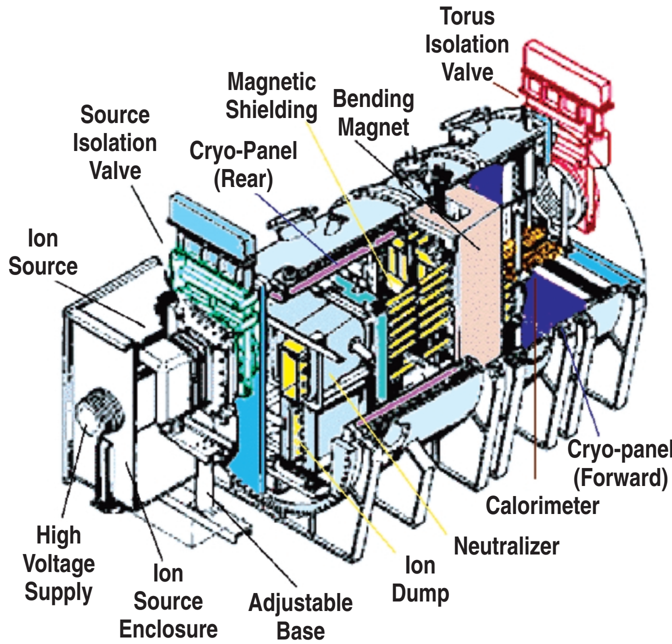

Fig. 10. DIII-D 80 keV neutral beam system.

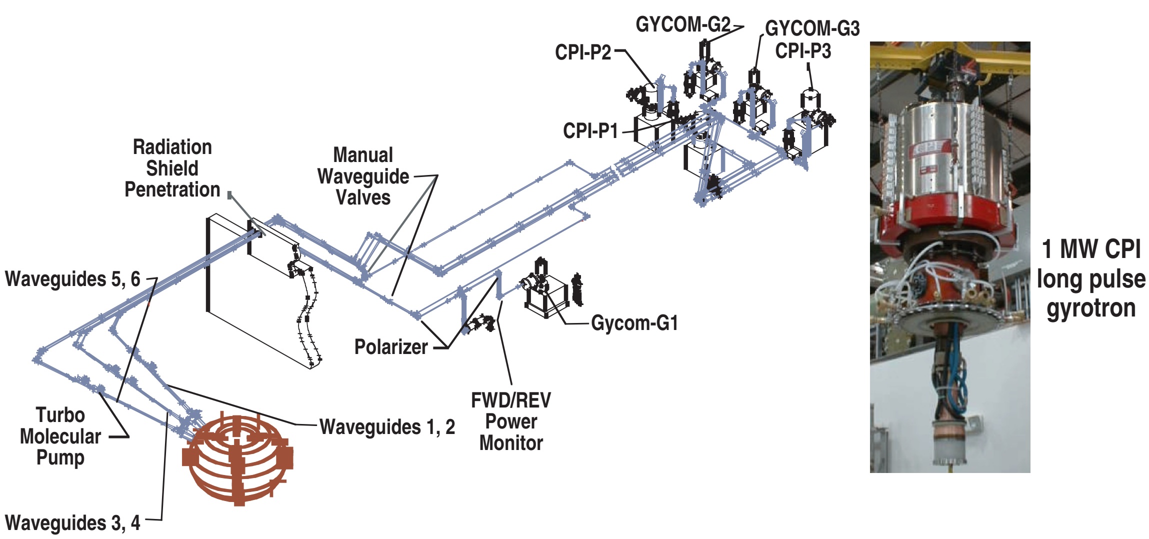

Fig. 11. DIII-D ECH system configuration and gyrotron.

ICH is employed for on-axis current drive and electron heating to increase the efficiency of ECH current drive. The DIII-D system (12) consists of three 2 MW transmitters tunable from 30 to 120 MHz. An Oak Ridge National Laboratory (ORNL)-developed four-element directional Faraday shielded antenna capable of driving current is shown in Fig. 12.

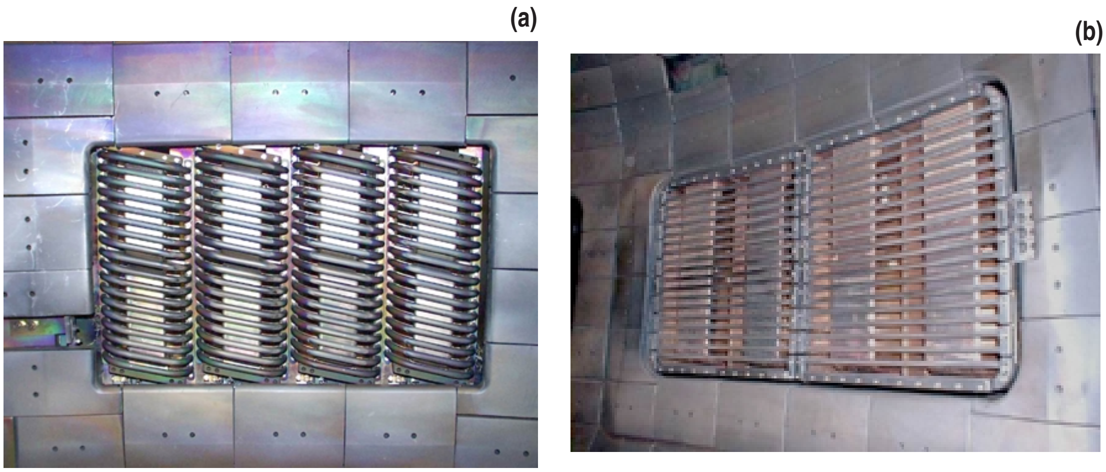

Fig. 12. Four element fast-wave current-drive antenna used on DIII-D.

SUMMARY

Optimizing advanced tokamak performance requires developing theoretical models, diagnostics, controllers, and real time plasma control systems beyond that currently being employed. This physics and technology challenge will require validation in a wide range of tokamak experiments and the successful application to new experiments, including burning plasma conditions for which further development of diagnostics and actuators will be needed.

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Additional publications can be found at “fusion.gat.com”.

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