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Recent and future upgrades to the DIII-D tokamak

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ABSTRACT

Research on the DIII-D tokamak focuses on support for next-generation devices such as ITER by providing physics solutions to key issues and advancing the fundamental understanding of fusion plasmas. To support this goal, the DIII-D facility is planning a number of upgrades that will allow improved plasma heating, control, and diagnostic measurement capabilities. The neutral beam system has recently added an eighth ion source and one of the beamlines is currently being rebuilt to allow injection of 5 MW of off-axis power at an angle of up to 16.5° from the horizontal. The electron cyclotron heating (ECH) system is adding two additional gyrotrons and is using new launchers that can be aimed poloidally in real-time by an improved plasma control system. The fast wave heating system is being upgraded to allow two of the three launchers to inject up to 2 MW each in future experiments. Several diagnostics are being added or upgraded to more thoroughly study fluctuations, fast ions, heat flux to the walls, plasma flows, rotation, and details of the plasma density and temperature profiles.

1. Introduction

The DIII-D tokamak research program concentrates largely on investigations of fundamental plasma physics issues important to the designers of next-generation devices such as ITER. To facilitate these studies, several tokamak systems are being upgraded, including the auxiliary heating systems, plasma control system, and many of the plasma diagnostics.

To study higher energy plasmas, the power available from all auxiliary heating systems is being increased, including the neutral beam injectors, electron cyclotron heating (ECH) system and the fast wave system for heating ions. One of the four neutral beamlines is currently being rebuilt to allow off-axis injection to significantly affect the beam-driven current profile. The ECH system has been routinely operating six 1 MW class gyrotrons simultaneously, and is building the power supply and waveguide system to add a seventh gyrotron, with plans for an eighth. The fast wave system has recently doubled the input stage power on two of the three systems to provide more ion heating capability.

Better plasma control is available with the upgrade of the digital real-time plasma feedback control system. Real-time poloidal steering of the ECH launch mirrors has just been brought on line, for example. Also, several diagnostic systems have been added and existing systems upgraded to improve the large suite of plasma diagnostics on DIII-D.

In Section 2, the upgrades to the auxiliary heating systems of the tokamak will be discussed. Section 3 briefly describes the improvements to the plasma control system (PCS). Section 4 summarizes some of the upgrades to the plasma diagnostics and conclusions are presented in Section 5.

2. Auxiliary heating systems

2.1. Neutral beams

Since 1996, seven ion sources have been available in the neutral beam system [1], each capable of injecting a nominal 2.5 MW for plasma heating. Prior to the most recent physics campaign, an additional source was made available when a new power supply was built, bringing the total power routinely available for physics experiments up to 20 MW. Six of the ion sources inject in the direction of the plasma current, I_p, and two are available for counter-injection (opposite to the direction of I_p) [2]. Until now, all sources injected in the midplane. A major modification currently underway, however, will allow *off-axis* injection of 5 MW of power from one beamline (two sources). Large hydraulic actuators will be used to tilt the beamline from the usual midplane injection angle of 0° up to an angle of 16.5° for off-axis injection up to 40 cm below the plasma centroid (Fig. 1) [3].

The ITER device plans to use off-axis current drive to control the current profile. On DIII-D, off-axis beam heating will enable a significant modification of the plasma current profile, thus affecting one of the most critical parameters in a high performance plasma. To date, off-axis current drive has been available with modest levels of electron cyclotron current drive (ECCD), but in high β plasmas

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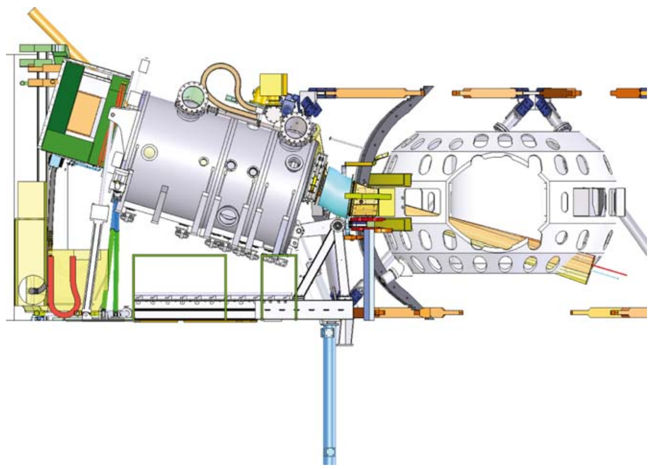

Fig. 1. Injection of up to 5 MW of off-axis power from a tilted beamline will be possible in 2011. Shown above is beam injection (into cutaway vessel) at maximum 16.5° off-axis, depositing power up to 40 cm below the plasma major radius.

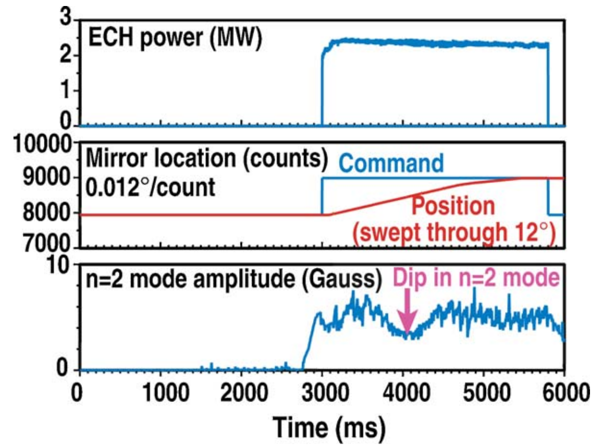

Fig. 3. ECCD deposition location is swept by programmed mirror movement, affecting the amplitude of the $n = 2$ mode.

with significant beam power injected on axis, the current profile is dominated by the large level of central current driven by beam heating (NBCD). Fig. 2 illustrates this phenomenon and shows that NBCD is as efficient as central heating for driving current, and is comparable to ECCD. The availability of off-axis NBCD starting in 2011 will allow carrying out valuable experiments to investigate control of the current profile.

2.2. ECH

During the 2009–2010 physics campaign, six gyrotrons were routinely used for injection of electron cyclotron microwaves for heating the plasma. With a peak generated power of 4.5 MW at 110 GHz, the maximum injected power was approximately 3.5 MW. A pulse length limit of 5 s resulted in a maximum injected energy of 16.6 MJ.

Planned ECH upgrades call for the addition of two more gyrotrons to the system in the future. The output power of the first of these new "depressed collector" gyrotrons will be 1.2 MW. A prototype of this gyrotron was tested several years ago and the new system is currently being built. The gyrotron vault is being expanded to accommodate the new tube and the power supply and waveguide are under construction. Factory tests of the first new gyrotron are scheduled for completion by July 2011. The second new gyrotron will operate at 117.5 GHz at a power level of 1.5 MW and the design of this system has been initiated. Long-range plans call for a 15 MW ECH system based on this gyrotron.

The ECH system is making additional improvements by rebuilding the waveguide runs from the gyrotron vault to the vacuum vessel, eliminating several miter bends to increase the delivered power to the plasma. Also, new electric motors have been implemented recently on the steerable mirrors in the launcher, allowing the ECH launch angle into the plasma to be controlled by the PCS during the discharge (Section 3).

2.3. Fast wave

The DIII-D fast wave system consists of three 4-strap antennas, each connected to a high power amplifier for launching radio frequency waves into the plasma. Two antennas are powered by 90 MHz transmitters that have recently been upgraded to 2 MW. Operation of these two systems for short pulses has already been achieved. The other transmitter operates at 1.5 MW with a frequency of 60 MHz. The power coupled to the plasma from fast wave was increased recently with the use of localized antenna gas puff systems. The discreet gas puff systems increase the density at the plasma edge near the antenna, thereby improving the power coupling.

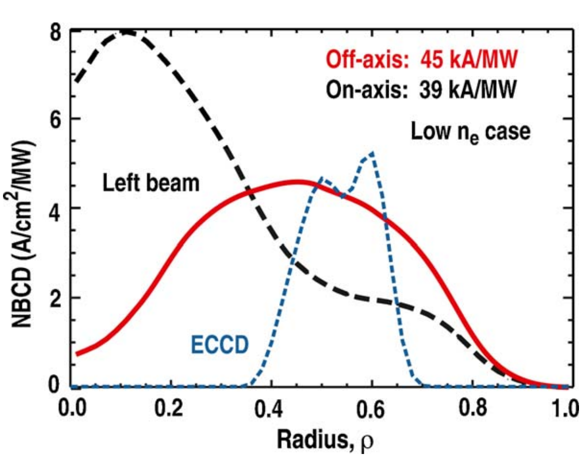

Fig. 2. Transport modeling predicts on- and off-axis NBCD and ECCD current drive efficiencies to be comparable.

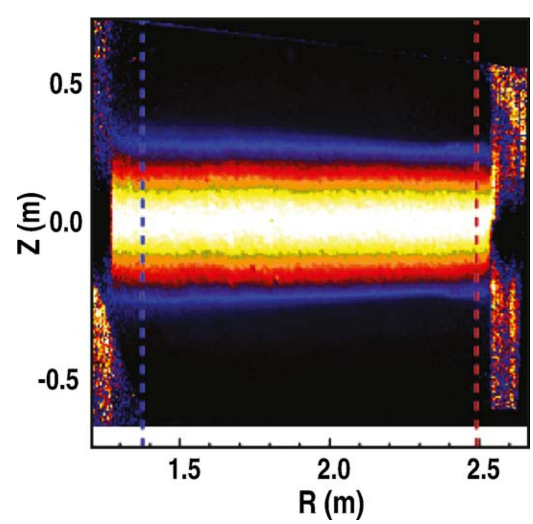

Fig. 4. BES diagnostic uses data from beam injection into helium gas to measure vertical profile of neutrals. Variation with radius yields beam divergence.