RECENT RESULTS FROM THE DIII–D TOKAMAK*

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The DIII–D national fusion research program focuses on establishing the scientific basis for optimization of the tokamak approach to fusion energy production. The symbiotic development of research, theory, and hardware continues to fuel the success of the DIII–D program. During the last year, a radiative divertor and a second cryopump were installed in the DIII–D vacuum vessel, an array of central and boundary diagnostics were added, and more sophisticated computer models were developed. These new tools have led to substantial progress in the understanding of the plasma. We now have a better understanding of the divertor as a means to manage the heat, particle, and impurity transport. Pumping of the plasma edge using the *in situ* divertor cryopumps effectively control the plasma density. The evolution of diagnostics that probe the interior of the plasma, particularly the motional Stark effect diagnostic, has led to a better understanding of the core of the plasma. This understanding together with tools to control the profiles, including electron cyclotron waves, pellet injection, and neutral beam injection, has allowed us to progress, in making plasma configurations that give rise to both low energy transport and improved stability. Most significant here is the use of transport barriers to improve ion confinement to neoclassical values. Commissioning of the first high power (890 kW) 110 GHz gyrotron validates an important tool for managing the plasma current profile, key to maintaining the transport barriers. An upgraded plasma control system, “isoflux control,” which exploits real time MHD equilibrium calculations to determine magnetic flux at specified locations within the tokamak vessel and provides the means for precisely controlling the plasma shape and internal profiles. Disruptions are a serious issue for ITER or any other future large tokamak. New disruption diagnostics and computer models have lead to better understanding of disruptions, and tools such as impurity pellet injection have effectively reduced disruptive EM forcer and heat loads. Erosion from disruptions is found to contribute significantly to the erosion of tiles in the divertor region. The remote operation of DIII–D from other sites via the ESNET is providing better access for collaborators and helping develop such capabilities for ITER. Remote manipulation techniques have been used to repair the currently unused portion of the ohmic heating coil solenoid. This repair is scheduled to be completed this year, and will increase the available ohmic flux swing by 2.5 Vs. Vanadium is being developed as a low activation material that can be used for the structure of future fusion devices. The DIII–D program plans to continue to develop the experimental understanding and hardware systems to optimize the tokamak configuration both by improving the core performance and by developing the divertor configuration to manage the plasma edge.

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