The DIII-D Upgrade to Prepare the Scientific Basis for Burning Plasma Operation in ITER and Steady State Fusion in Future Reactors

R.J. Buttery, H.Y. Guo, D.N. Hill*, A.G. Kellman, T.S. Taylor, M.R. Wade
and the DIII-D Team

General Atomics, PO Box 85608, San Diego, CA 92186-5608, USA
**Lawrence Livermore National Laboratory, 7000 East Ave, Livermore, CA 94550*

Abstract. The next generation of fusion devices poses key challenges for present facilities in order to ensure full performance is rapidly realized and resolve critical design choices. For ITER, it is vital to anticipate the different behavior expected in the burning plasma regime, so we have the tools to interpret the physics and regimes can be re-optimized for these conditions. For longer pulse reactors, such as DEMO or a nuclear science facility, it is critical to establish the physics basis for projecting to a self-consistent self-sustaining regime, and develop a long pulse boundary solution to handle the hot plasma exhaust. A major upgrade to DIII-D is proposed to address these challenges, emphasizing electron heating, off-axis current drive, and a new divertor with reactor relevant temperatures and materials.

I. Preparing for Burning Plasma Conditions with Torque-free Electron Heating

Present tokamaks generally utilize neutral particle beams to heat plasmas. These drive strong rotation, fuel the core, and heat through the ions. However in a burning plasma, heat will come primarily from fusion α's, augmented by radio-frequency heating or much higher energy particle beams, which all deposit heat principally on the electrons, drive relatively little torque, and provide little fuel. These changes in energy, particle and momentum throughput are expected to substantially alter the nature of the transport in the plasmas, and decrease stability. Calculations with TGLF (Fig. 1) show that with high torque neutral beam ion heating on DIII-D, electron turbulence is dominated by fine scale electron temperature gradient (ETG) modes. However the low torque electron heating in ITER leads to a much greater prevalence of lower order ion temperature gradient (ITG)/trapped electron modes (TEMs). Such changes have been observed to lead to increases in thermal and particle transport on DIII-D as the ratio of ion to electron heating and torque are reduced [1]. For ITER and future burning plasma devices, it is vital to understand how to re-optimize performance in this regime, and to have the scientific understanding to interpret behavior. Access to this physics in present devices requires increases in torque-free power and electron heating. Thus, on DIII-D, a tripling of electron cyclotron heating (ECH) power

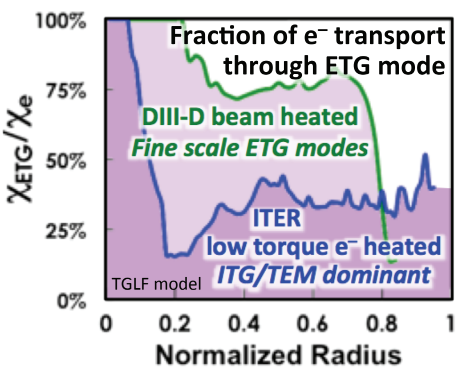

Fig. 1. Predicted change in transport from neutral beam injection (NBI) heated to burning plasma.

from 3.5 to 10.5 MW is planned) utilizing new 1.5 MW long pulse gyrotrons, together with increased energy and re-orientation of neutral beams for full power balanced torque operation using rotatable beam lines. This will enable DIII-D to access dominantly electron-heated regimes (Fig. 2) over a wide range in β, that approach ITER collisionalities, and also to vary profiles and the degree of ion-electron coupling to explore trends and underlying transport physics. The precise deposition control with ECH will enable perturbative studies, such as gradient modulation to measure stiffness and probe transport more deeply

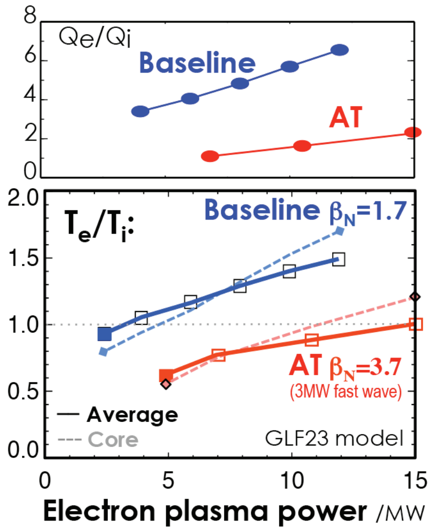

Fig. 2. Increase to dominant electron heating with increased ECH.

These upgrades will also enable DIII-D to confront the stability challenges of burning plasma operation, where low rotation is expected to decrease tearing mode stability and raise susceptibility to non-axisymmetric 'error' fields [2], that arise in tokamak construction or with 3D field control tools (e.g. for ELM suppression). Here the precise heating and current drive possible with the ECH will be a powerful tool in both optimizing underlying stability by varying the profiles, and directly stabilizing modes through localized current drive. It is proposed to augment this with improved 3D field flexibility to explore how to optimize 3D fields to achieve the required control. This would increase the toroidal distribution of field coils to twelve (Fig. 3) to provide toroidal mode number, n, up to 6 and rotatable n=3 and n=4 perturbations with increased poloidal harmonic flexibility. These improvements will enable development of robust integrated stability and mode control methodologies to maintain high performance and avoid disruptions in burning plasma regimes.

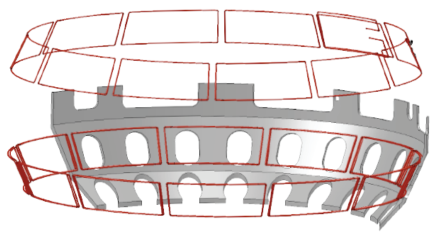

Fig. 3. Proposed new 3D field coils.

II. Establishing The Physics Basis for Sustained Fusion with Off-Axis Current Drive

The attractiveness of fusion energy is critically dependent on the ability to achieve high fusion power density over long durations with high duty cycle. Simultaneous achievement of high toroidal β and high poloidal β in the steady-state Advanced Tokamak (AT) enables both high fusion power and a high self-driven current to sustain high performance continuous operation (Fig. 4). In this regime, the transport, current distribution, stability and energetic particles become tightly coupled, leading to nonlinear responses and behavior that must be well understood to determine an optimized solution. Research is vital to project requirements to future devices such as DEMO or a nuclear science facility, or indeed the steady state mission of ITER.

A steady-state regime is likely to require considerably broader current and pressure distributions than present devices access, in order to stably reach the high β_N required. Thus major developments are proposed to raise DIII-D's off-axis current drive capabilities and provide the flexibility to study the physics basis of a self-consistent self-sustaining solution for a reactor (Fig 4). A second pair of beams will be re-oriented to provide large-scale off-axis current drive. The ECH upgrade will provide additional off-axis current drive capability, with precise deposition control to understand the optimization. Finally, ultra high harmonic "Helicon" current drive is being developed, which offers the potential for a far more efficient reactor current drive solution, as well as additional off-axis current drive in DIII-D. Coupled with increases in beam energy and a doubling of beam pulse length, these improvements will provide DIII-D with the flexibility to match and optimize power plant equivalent levels of performance (Fig. 5), with solutions converged over two current redistribution times. Candidate solutions can then be tested in superconducting AT facilities to assess longer timescale issues, such as wall evolution and control.

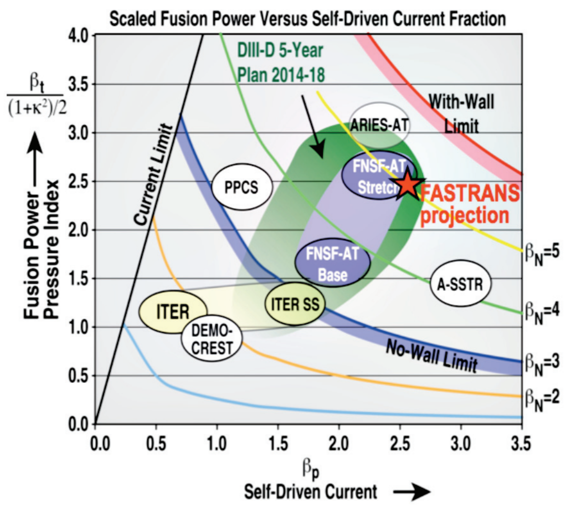

Fig. 4. Additional off-axis current drive and heating will enable optimization of solution for steady-state fusion (TGLF model).

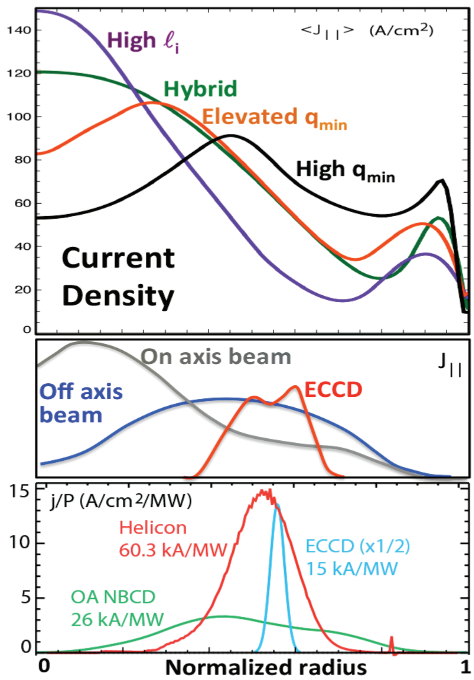

Fig. 5. Beams and ECH provide flexibility to study a range of candidate solutions (upper two panels), while Helicon (lower panel) is predicted to be strongly absorbed in β_e>2% scenarios.

III. A Divertor Research Facility to Resolve the Boundary Solution for Future Reactors

A fusion power plant or nuclear science facility will go beyond ITER in terms of heat and particle flux and fluence, placing additional demands on the boundary solution. As well as mitigating heat fluxes, erosion must be virtually eliminated to enable high duty cycle operation. This requires a cold detached plasma at the divertor target plates. Thus, DIII-D is launching an effort to design, install, and operate a new Divertor Research Facility in DIII-D to develop the physics basis for an optimized boundary solution, compatible with a high performance steady-state core plasma. The concept is to improve isolation between the divertor and core, facilitating robust detachment with increased

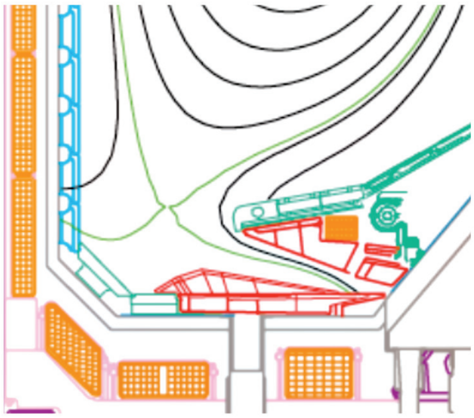

Fig. 6. A Divertor Research Facility with hot walls and relevant material will be implemented to develop the physics basis for a reactor boundary solution.

operating space. Initial work is exploring elements of this optimization with the present facility configuration, such as the role of connection length or divertor leg geometry through concepts such as snowflake divertor [3] and elements of “X” and “super-X” divertor physics [4]. These studies will inform the design of the new divertor, which is intended to be a flexible tool to develop and optimize an improved solution. Here, an internal PF coil (Fig. 6) will help provide configuration flexibility to aid physics studies and decouple the divertor leg from X point geometry. This will be accompanied by heated wall and divertor tiles, in order to study the dynamic in a relevant recycling regime. Baffling would be progressively introduced to control flows and isolate the boundary solution from the core to arrive at an optimized geometry. Reactor relevant materials would also be introduced to assess compatibility with the divertor solution, first with a toroidal row before progressive implementation throughout the vessel. Thus studies will combine with research around the world to enable resolution of a self-consistent solution, optimizing between plasma exhaust mitigation, relevant materials and a high performance steady-state core.

Conclusion

A major upgrade is proposed for the DIII-D tokamak to make unique and vital contributions for the preparation of ITER and future steady state reactors. Dominant torque-free electron heating will enable development of the understanding and optimization of burning plasma transport and stability. Increased off-axis current drive will allow resolution of the path to a steady state burning plasma. A divertor research facility in DIII-D will provide the basis for a reactor relevant boundary solution for quasi-continuous operation. These developments will ensure that in the next decade, the world program is optimally prepared for ITER, and lay the foundations for decisions on a fusion power plant and nuclear science facilities.

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