Innovative Divertor Concept Development on DIII-D and EAST*

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***Abstract*—A critical issue facing the design and operation of next-step high-power steady-state fusion devices is the control of heat fluxes and erosion at the plasma-facing components, in particular, the divertor target plates. A new initiative has been launched on DIII-D to develop and demonstrate innovative boundary plasma-materials interface solutions. The central purposes of this new initiative are to advance scientific understanding in this critical area and develop an advanced divertor concept for application to next-step fusion devices. DIII-D will leverage strong collaborative efforts on the EAST superconducting tokamak for extending integrated high performance advanced divertor solutions to true steady-state.**

Keywords—Plasma-Materials Interaction; Divertor; Tokamak

I. Introduction

The path towards next-step fusion development requires increased emphasis on the boundary/plasma-material interface (PMI). One of the major issues facing the design and operation of next-step high-power steady-state fusion devices is the control of heat fluxes and erosion of the plasma-facing components (PFCs). Thus, it is essential to find plasma solutions that control heat fluxes to keep them within the heat exhaust limitations of the PFCs, i.e, below 10 MW/m^2 (including both graphite and tungsten), with divertor plasma temperature below 5 eV. This poses a significant challenge for the next-step fusion devices, such as a Fusion Nuclear Science Facility (FNSF) [1] with the envisioned Advanced Tokamak (AT) or Spherical Tokamak (ST) scenarios and the China Fusion Engineering Test Reactor (CFETR) [2], because of relatively low density ($n/n_G \sim 0.5$ for both cases with n_G being the Greenwald density) and stringent needs to simultaneously control divertor and core parameters.

To respond to this challenge, a new initiative has been launched to develop and demonstrate innovative boundary/PMI solutions on DIII-D, aimed at making major contributions in this critical area for ITER and beyond [3]. The DIII-D boundary/PMI initiative adopts the following approaches: (1) Developing and testing an advanced divertor concept compatible with the core plasma high performance operational scenarios of future fusion devices. This entails validation of predictive models for heat flux dissipation, development and testing of advanced divertor configurations on DIII-D to reduce the density threshold for detachment; (2) validating Advanced Materials (AM) solutions for reactor plasma facing components at reactor-relevant temperatures in DIII-D high-performance Advanced Tokamak (AT) plasmas, in collaboration with the broad material research/development community; (3) integrating validated AMs with ATs to provide integrated boundary/PMI solutions for next-step high-power steady-state fusion devices.

DIII-D will leverage strong collaborative efforts on the EAST superconducting tokamak for developing integrated core/boundary solutions. EAST has recently upgraded its upper divertor to full W/Cu-PFCs with an ITER-like W monoblock structure [4]. The EAST heating and current drive system has also been significantly upgraded with total heating power exceeding 20 MW. This will allow EAST to address major issues facing long-pulse AT operation. This presentation discusses the present progress and near-term boundary/PMI research plan on DIII-D, as well as joint research work on EAST.

II. Developing An Advanced Divertor on DIII-D

DIII-D features well-established ITER-like scenarios and high-performance discharges with robust plasma control and extensive boundary/PMI diagnostics, providing a unique fusion environment suitable for conducting reactor-relevant divertor research. DIII-D is making upgrades to allow more complete exploration of reactor-relevant conditions compatible with high-performance ATs. We will take an integrated approach toward developing advanced divertor concepts on DIII-D to establish validated boundary/PMI solutions for next-step devices. This embraces the following closely coupled aspects:

- Advancing scientific understanding and validating models of divertor plasma detachment;
- Optimizing the divertor to maximize divertor dissipation for facilitating detachment; and
- Identifying suitable AM solutions for a DIII-D divertor upgrade in early 2020s.

A. *Advancing Plasma Dynamics Approaching Detachment*

It is of central importance to understand the plasma physics controlling detachment under conditions and geometries relevant to next-step tokamaks to minimize divertor target heat flux and erosion. DIII-D has established a comprehensiveset of divertor diagnostics (Fig. 1), including Divertor Thomson Scattering (DTS) capable of measuring electron density and temperature, n_e and T_e, from the target to the X-point [5], and 2D divertor coherence imaging system for measurements of plasma flow, V_i, and ion temperature, T_i, thus providing unique opportunities to advance understanding and develop predictive capabilities for ITER and beyond.

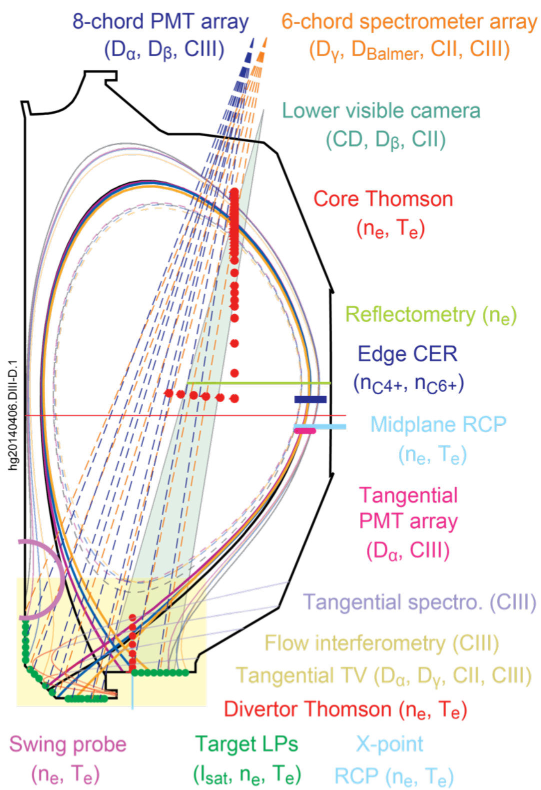

Fig. 1. Schetch of boundary/PMI diagnostics on DIII-D.

Recently, we have confirmed a radiation shortfall in the 2D fluid codes, including both UEDGE and SOLPS; radiative models fail to capture observed dependencies. To illustrate this, Fig. 2 plots the electron temperature at the outer strike point, T_{ed}^{OST}, measured by DTS, as a function of the upstream separatrix density, n_{es}^{U}, along with the predictions from SOLPS modeling [6]. As can be seen, SOLPS underestimates the radiative losses in the divertor as the plasma detaches, as manifested by a pronounced drop in T_{ed}^{OST}, and does not accurately capture observed detachment dynamics, *i.e.*, the sharp transition to complete detachment with $T_{ed}^{OST} \leq 1\ eV$.

It is thus necessary to understand the underlying atomic/molecular physics processes that control volumetric power and momentum losses, quantify parallel and perpendicular transport especially near detachment, and examine compatibility with the pedestal, in particular, the relationship between the pedestal and scrape-off-layer (SOL)/divertor, to provide a basis for reliable extrapolation of edge plasma/PMI to enable divertor/PFC design optimization. This requires validation and development of predictive boundary/PMI models with improved diagnostic measurements to isolate specific aspects of underlying physics, and further augmentation of heating and drive capabilities. Upgrades at DIII-D will enable research of more reactor-relevant conditions. In addition, existing and future high-power and long-pulse devices, including ITER, will provide new opportunities for model validation at more extreme conditions closer to reactor levels.

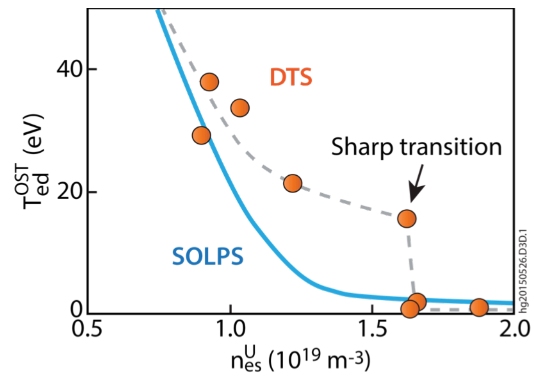

Fig. 2. Comparison between DTS measurements and SOLPS modeling: electron temperature at the outer strike point, T_{ed}^{OST}, as a function of upstream separatrix density, T_{es}^{U}.

B. *Optimizing the divertor to maximize dissipation*

The overall goal of divertor optimization is to develop and validate design concepts to improve the divertor performance in next-step burning plasma tokamaks. The primary aim of divertor design is to reduce target plate heat flux and erosion to tolerable levels while providing control of fuel and core impurity content. To achieve these goals, dissipative/detached divertor operation with cold dense radiative plasma will be required. The SOL and divertor plasma and neutral density must also be consistent with the core plasma operation scenario. It is expected that such core-edge compatibility will require a lower midplane separatrix and SOL density than would be achieved for detached divertor operation in a standard configuration. Therefore, the DIII-D divertor optimization program aims at developing an advanced divertor concept that can achieve detached/dissipative divertor conditions across the target plate at lower midplane density.

Increased efforts have been dedicated to the development of advanced divertors over the last decade, with a number of innovative divertor concepts having emerged, including snowflake divertor (SFD) [7], X-divertor (XD) [8], super X-divertor (SXD), X-point target divertor [9] and isolated divertor [10]. Developing advanced divertor solutions requires innovative physics and engineering. This is shown by the following formula, which indicates some of the requisite elements for target heat-flux mitigation:

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$$q_{\text{target}} = \frac{(1-f_{\text{rad}})P_{\text{SOL}}\sin(\theta_{\text{div}})}{4\pi R_{\text{target}}\lambda_q f_{\text{exp}}},$$

where f_{rad} is the radiative power fraction in the SOL and divertor; P_{SOL} is the power flow into the SOL, allowing for radiative loss from the confined plasma, $P_{\text{rad-core}}$; θ_{div} is the angle between the poloidal flux surface and target plate; λ_q is the radial decay length of the SOL heat flux; and f_{exp} is the poloidal flux expansion factor. Thus, divertor optimization requires:

- Optimizing divertor geometry, i.e, through target plate orientation (θ_{div}) and baffle shaping to improve heat exhaust and enhance the divertor retention for neutrals and impurities, thus improving particle exhaust and reducing impurity contamination.
- Optimizing magnetic configuration to maximize poloidal and toroidal flux expansion (f_{exp}, R_{target}) and increase field-line length, effectively increasing the divertor volume.
- Active radiation and particle control by injecting highly radiative impurities in the divertor (f_{rad}) to enhance radiation, and in the core plasma to reduce the power flow into the divertor (P_{SOL}), without compromising the core plasma performance, thus promoting divertor detachment.

DIII-D features two divertors with different structures: an open divertor at the bottom and a relatively-closed divertor at the top, thus affording two research opportunities. Experiments can examine the effect of divertor closure using the upper divertor to maximize neutral entrapping and minimize core performance degradation, while exploring advanced magnetic configurations with the lower divertor, including the SFD and XD configurations (Fig. 3), taking advantage of DIII-D's flexible poloidal field coils and robust control system, to provide insight and guidance for the development of a fully optimized divertor concept in DIII-D.

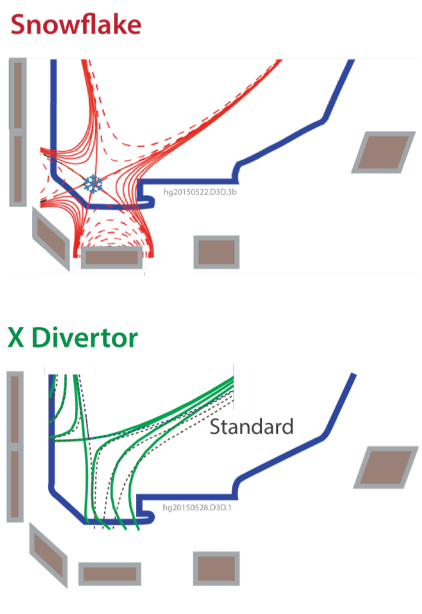

Fig. 3. Advanced divertor configurations on DIII-D.

Promising progress has been made on DIII-D in developing the SFD divertor configuration, demonstrating that SFD significantly enhances target heat spreading and reduces heat fluxes in both attached and radiative divertor regimes, between and during edge localized modes (ELMs), while maintaining good H-mode confinement [11]. The world's first real-time SFD detection and control system has also been developed on DIII-D to stabilize and manipulate this configuration [12]. In addition, we have recently started to explore the XD configuration, taking advantage of the open structure of the bottom divertor and the flexible plasma control system. Initial XD experiments on DIII-D are promising, both for target heat flux reduction and detachment facilitation [13]. Consequently, this enables plasma to enter detachment at significantly higher pedestal pressure, thus mitigating degradation of H-mode performance.

To examine the effect of divertor closure on DIII-D, a comparison will be made between the open lower divertor with the more closed upper divertor in DIII-D with extensive diagnostic coverage in the next experimental campaign. Further enhancement of the degree of divertor closure by modifying the shape of the upper divertor tiles is planned for 2017. Initial modeling of increased closure of the DIII-D upper divertor has identified a promising narrow divertor option, i.e. a "shaped-slot" divertor (Fig. 4). The shaped-slot is predicted to strongly reduce the divertor T_e and target heat flux, $q_\perp$, by redirecting and confining recycling neutrals and impurities near the corner (Fig. 5). This will enable detachment onset at a lower upstream density than for a rectangular slot.

C. *Validating reactor-relevant AM solutions*

Three key scientific issues for PFCs (in particular, the divertor target plates) include (1) erosion, redeposition, and material migration under both steady and transient plasma conditions; (2) the impact of PFC materials selection on the core plasma; and (3) the effect of high bulk temperature PFCs. Presently, it is widely expected that tungsten will be the structural material of the first wall and potentially the plasma-facing materials (PFM) of next-step fusion devices, although there are concerns regarding its susceptibility to melting/cracking under high heat flux and the production of surface nano-tube 'fuzz' possibly leading to arcing and dust production. Future high duty cycle devices may experience significant material migration leading to local buildup of deposits which could interfere with operation. Managing such buildup appears more tractable using low-Z materials than high-Z metals.

Validation of reactor-relevant AM solutions will involve integrated research on linear material facilities and long-pulse devices, such as EAST, KSTAR and JT-60SA. DIII-D will make important contributions by providing a unique, well-diagnosed world-class fusion plasma environment for materials evaluation and testing, through integration of AMs with high performance ATs. This will be coupled to the excellent existing platform for boundary/PMI studies based on the

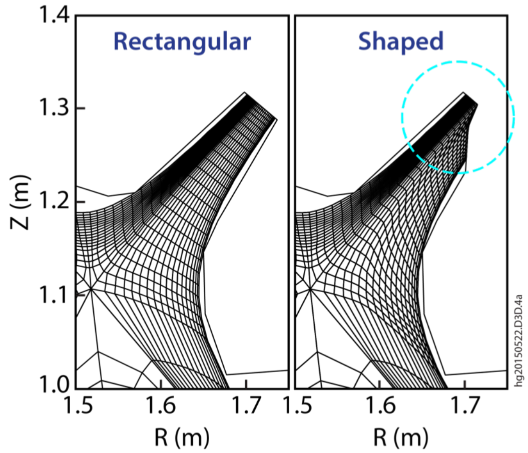

Fig. 4. Potential narrow slot divertor options for test on DIII-D.

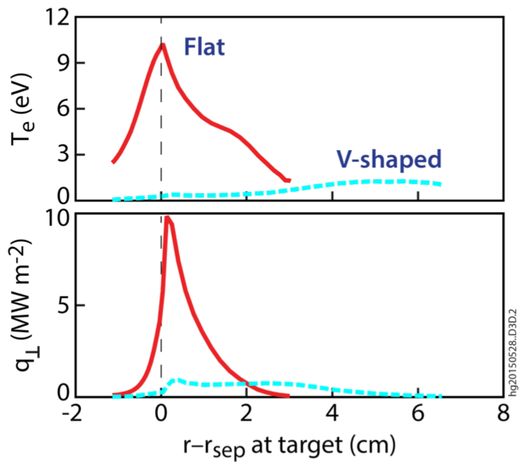

Fig. 5. Divertor target profiles of electron temperature, T_e, and heat flux, $q_\perp$, predicted by SOLPS for the flat target and "V-shaped" target, respectively, with power flow across the separatrix, P_{SOL} = 3 MW, and the upstream separatrix density, $n_{sep}^{U} = 5\times10^{19}$ m^{-3}.

DiMES divertor materials evaluation system and close collaboration with SciDAC PMI modeling efforts. DiMES provides a quantitative and comprehensive capability for research on critical PFM issues of erosion, redeposition, and migration of low-Z and high-Z materials. In addition, with its carbon walls, DIII-D affords a unique test environment because high-Z materials can be studied as true trace elements greatly facilitating measurements of their migration.

DIII-D is ready to test advanced material solutions to address the three key PMI issues aiming at defining suitable PFC solutions for divertor and main chamber walls, in collaboration with the materials development community. In the case of (1), the existing DiMES will be used to test advanced material PFCs using structural carbon and tungsten substrates with low-Z coatings. To assess (2), we will install strategically positioned rings of tiles, each with a different high-Z (or different W isotope) metal coating; this will provide critical data on sputtering sources and erosion due to steady-state and transient loads and their effect on high performance plasmas. To study (3), we will take a staged approach, starting with heated DiMES samples, followed with small heated divertor and main chamber elements, potentially leading to installation and operation with large area divertor and main chamber heated surfaces.

The ultimate goal of the DIII-D advanced divertor development program is to combine optimizations in divertor geometry and magnetic configuration to develop and test advanced divertors with high temperature operation capability and to establish compatibility with high-performance scenarios in DIII-D in the early 2020s. Advanced materials for reactor PFCs at reactor-relevant temperatures in DIII-D high-performance plasmas will be validated in collaboration with the broad material research/development community.

III. Extending Advanced Divertor Solutions to Steady-State on EAST

EAST is a fully superconducting tokamak with technologies similar to ITER. EAST has unique capabilities to address many of the critical physics and technology issues that ITER will encounter. EAST is capable of long-pulse operations with high power electron heating (as in ITER) to challenge power and particle handling at levels (10 MW/m^2) comparable to ITER, and is poised to make contributions in nearly all of the high priority research topics to address risks to ITER.

Significant progress has been made on EAST towards high-power, long-pulse operations on both technology and physics fronts, achieving fully steady-state long-pulse L-mode plasmas over 400s, and reproducible H-mode plasmas over 30 s [4, 14, 15], chiefly enabled by (a) advanced lithium (Li) wall conditioning, and (b) Lower Hybrid wave Current Drive (LHCD). To accomplish this, great efforts have been made to control edge recycling and divertor heat load. Li wall conditioning has proven to be the most effective method to control impurity influxes and hydrogenic recycling, which also promotes H-mode access and reduces the H/D ratio, thus improving ion cyclotron resonant heating. This, along with improved plasma facing components with active water-cooling and periodically changes to divertor configurations to minimize divertor heat load, greatly facilitates long pulse operation. LHCD provides an intrinsic boundary control for ELMs, leading to a dramatic reduction in the transient power load on the vessel wall, compared to the standard Type I ELMs. LHCD also induces edge plasma ergodization, broadening heat deposition footprints; the heat transport caused by ergodization can be actively controlled by regulating edge plasma conditions, thus providing a new means for stationary heat flux control.

EAST has undertaken an extensive upgrade with significantly enhanced current drive and heating capabilities, with a total power exceeding 20 MW, and ITER-like W monoblock target structure to allow for high heat load on divertor targets, up to 10 MW/m^2. These new capabilities advance EAST to the forefront of international magnetic fusion facilities and will allow EAST to make significant contributions to ITER and next generation fusion facilities, such as CFETR currently under conceptual design in China.

EAST also aims to explore steady state operation with advanced tokamak characteristics that would be of great value to ITER but also could lead to an attractive fusion power plant. Towards this end, a steady-state advanced tokamak operation scenario has recently been developed for EAST in collaboration with General Atomics. The initial test on DIII-D has demonstrated that fully non-inductive H-mode plasma operation is possible with plasma parameters and plasma formation schemes consistent with EAST capabilities [16]. Joint efforts will be made on EAST to develop and test steady-state AT scenarios with ITER-like W mono-block divertor to address critical PMI issues facing high-power, long-pulse plasma operations in the near future.

IV. Summary

DIII-D is embarking on an ambitious program of upgrades and research to advance scientific understanding and validate models of boundary/PMI physics that can be applied to reactor-relevant conditions, and to develop an advanced divertor concept on DIII-D to address key issues in this critical area for ITER and beyond. Joint efforts on DIII-D and EAST will extend integrated high performance AD-AT scenarios towards steady-state on EAST. This initiative will establish a compelling bridge for US research on long-pulse facilities, and leverage shared resources to accelerate progress toward next-step fusion development.

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