Design and modeling of a closed divertor with mid-leg pumping for core-edge integration in DIII-D

EX-D

R.S. Wilcox[1], J.H. Yu[2], A. Holm[3], M.W. Shafer[1], W. Choi[2], S.L. Allen[3], T. Elsey[2], M.E. Fenstermacher[3], F. Glass[2], M. Groth[4], J. Hicok[2], R. Hood[5], A.W. Leonard[2], J.D. Lore[1], R. Maurizio[2], A.G. McLean[3], A.L. Moser[2], C. Murphy[2], J. Ren[6], T.D. Rognlien[3], G. Ronchi[1], F. Scotti[3], C.K. Tsui[5]

[1]Oak Ridge National Laboratory, 1 Bethel Valley Road, Oak Ridge, Tennessee 37830, USA
[2]General Atomics, PO Box 85608, San Diego, California 92186-5608, USA
[3]Lawrence Livermore National Laboratory, 7000 East Avenue, Livermore, California 94550, USA
[4]Aalto University, PO Box 11000, Aalto, Finland
[5]Sandia National Laboratories, PO Box 5800, Albuquerque, New Mexico 87123, USA
[6]University of Tennessee, Knoxville, Tennessee 37996, USA
email: wilcoxrs@ornl.gov

DIII-D is installing a novel, well-diagnosed divertor structure to test the modeling prediction that mid-leg pumping in a closed, long-legged divertor can passively stabilize the detachment front position (the location of ionization / neutralization along the divertor leg, where $T_e \sim 10$ eV) and robustly maintain a cold divertor target with a hot plasma core. If experimental measurements confirm the predictions of the 2D boundary plasma modeling, this may lead to a new paradigm for divertor design in next-step tokamak devices approaching reactor-relevant divertor parameters with a smaller divertor volume, while potentially reducing the required impurity gas puffing and its associated core fuel dilution.

A new upper divertor structure is being installed in DIII-D to test the effectiveness of mid-leg divertor pumping as a tool for robust core-edge integration. As detailed in multiple machines, either in the absence of particle removal or using standard divertor pumping schemes where particle removal occurs near the divertor target, core energy confinement is typically degraded when the divertor target is detached [1]. When the plasma electron temperature at the material surface is reduced below the threshold for physical sputtering, $T_e < \sim 1$ eV, the detachment front encroaches on the core plasma, cooling the X-point and separatrix, reducing the plasma temperature on closed magnetic flux surfaces.

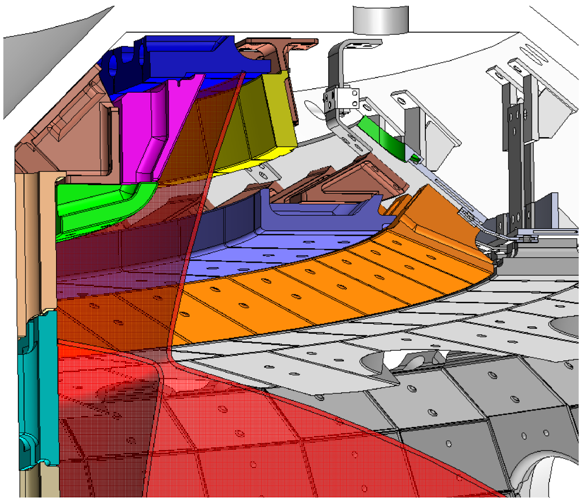

Fig. 1. CAD model of new divertor and separatrix

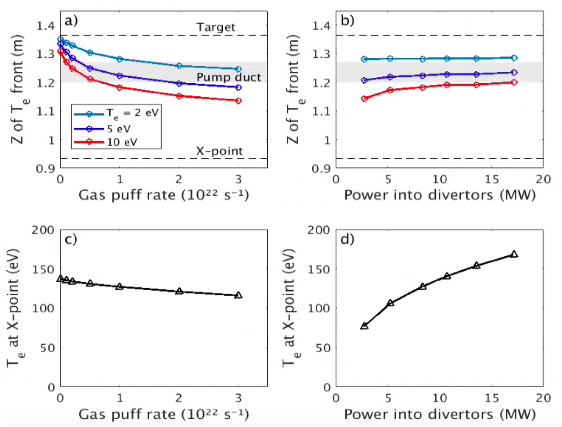

Fig. 2. Detachment front position and X-point Te predicted by SOLPS-ITER as a function of power and particle injection (from [3])

To passively stabilize the detachment front and thus simultaneously maintain a hot plasma core ($T_{e,Xpt} > \sim 60$ eV) with a cold divertor target ($T_{e,targ} < \sim 1$ eV), a divertor has been designed for DIII-D featuring a closed geometry to contain neutrals in the divertor slot, and a pump duct opening positioned away from the target, part way towards the X-point along the outer divertor leg. A colorized drawing of the engineering design and plasma separatrix is given in Fig. 1. The primary goals of this pump design are to create a region of high neutral gas density near the target, to dissipate plasma energy using cold recycling neutral flux and keep the electron temperature low near the plasma-material interface, while then removing neutral particles via the pump duct before the detachment front can propagate toward the main chamber and core plasma. Boundary plasma and neutral modeling with both SOLPS-ITER (including kinetic neutral physics) and UEDGE (including the physics of cross-field drifts) predicts that positioning the pump part way poloidally up the divertor leg of a closed divertor produces a high neutral gas pressure region near the target, maintaining a detached target and hot plasma core robustly across an experimentally relevant range of power and particle exhaust rates, as shown in Fig. 2 [2], [3].

Control analysis was performed to ensure the viability of the desired plasma shapes using the DIII-D coil set and power supplies. A suitable baseline equilibrium was selected with plasma current as high as I_P=1.8 MA, resulting in an edge safety factor q_{95}=3.3 and a projected scrape-off layer heat flux width λ_q=1.6 mm. The plasma shape is calculated to be controllable at this plasma current for experimental parameter scans of outer divertor leg length and radial flux acceptance into the divertor slot.

To enable the diagnosis and analysis of radiation, detachment and ionization properties, a comprehensive suite of diagnostics is being implemented in the upper divertor. Tiles will be instrumented with 44 in-situ Langmuir probes, 22 surface-eroding thermocouples, 6 ASDEX-style neutral pressure gauges, and 2 in-situ Penning gauges. In addition to an existing tangentially viewing camera, an array of in-vessel optical fibers will be installed viewing horizontally across the divertor slot to localize the impurity radiation along the outer divertor leg. A single Thomson scattering laser path will pass vertically through the outer divertor slot, with viewing optics enabling measurement of electron temperature both inside and outside of the divertor slot. This measurement will be particularly important to identify the electron temperature near the X-point in the case that it is above the threshold for impurity radiation that can be diagnosed spectroscopically.

With this innovative divertor design, DIII-D aims to explore a robust solution for solving the integrated tokamak exhaust and performance challenge.

This work was supported in part by the US Department of Energy under DE-FC02-04ER54698, DE-AC05-00OR22725, DE-AC52-07NA27344, DE-NA0003525, and DE-SC0023378.

[1] B. Lomanowski *et al.*, Nuclear Fusion **62**, 066030 (2022).
[2] A. Holm *et al.*, Nuclear Materials and Energy **41**, 101782 (2024).
[3] J. H. Yu *et al.*, Nuclear Materials and Energy **41**, 101826, (2024).