Advanced Magnetic Divertor Control

Egemen Kolemen, Princeton University, ekolemen@princeton.edu

Introduction: The present vision of the tokamak plasma-material interface is an axisymmetric magnetic X-point divertor. The X-point divertor is simple to make and operate, but this divertor configuration introduces a peaked heat flux near the strike point. One way to reduce the high heat exhaust per unit area on the plasma-facing components (PFCs) is to use alternative magnetic configurations. Examples of these advanced divertors are the snowflake divertor (SFD) [1], X-divertor [2], super X-divertor [3], and X-point target divertor [4]. Extra flexibility in the manipulation of the magnetic geometry allows the optimization of the divertor in order to achieve reduced particle and heat flux at the PFCs.

The advantages of advanced magnetic divertor configurations have been recognized by the tokamak community. NSTX-U is designed to operate SFD configurations at the upper and lower divertors in order to reduce the heat loading on the divertors at high plasma current and heating power. Extra coils were added around the divertor region in order to enable this. In the 5-year plan [5], the lower-only SFD will start in 2015 and the fully operational upper+lower SFD will be in use by 2018.

At DIII-D, in the 2014-2018 time frame, the main thrust of the Plasma Material Interface (PMI) group will be to conduct research to inform decisions on a divertor upgrade implementation in subsequent years [6]. With this aim, SFD and the open vessel Super-X magnetic configurations have begun to be tested. Further study of these configurations and the X-divertor will be conducted in the current 5-year plan [7].

Advanced divertors for FNSF and fusion power plants are currently under investigation [8]. These studies are based on static modeling, which assume that the advanced divertor configuration of interest will be achieved and held stable. However, the advanced divertor topologies are generally less stable then the standard divertor and require accurate diagnostics to identify the complex magnetic configuration and advanced control algorithms to stabilize it. In order to take advantage of these configurations, I propose to develop these algorithms and study their applicability to fusion power plant, in currently operational test reactor: NSTX and DIII-D.

Background: All the control research in this field is new and conducted since the ReNeW. We implemented the world's first real-time SDF detection and control system on DIII-D in order to stabilize and manipulate this configuration [9]. Compared to the standard divertor, SFD geometry, which uses a second-order poloidal field null created by bringing together two first-order poloidal field null points (X-points), greatly reduces peak heat flux through its high poloidal flux expansion, a large plasma-wetted area, and extra strike points. The new system uses a fast real-time snowflake identification algorithm that is based on local expansion of the Grad-Shafranov (G-S) equation in toroidal coordinates up to the third-order terms. The X-point locations are obtained by using this expansion and the real-time equilibrium reconstruction (rt-EFIT) [10] in a one-step algorithm (no iteration necessary) in <<1 ms. Then, for precise control, the effect of the change in PF coil currents on the X-point locations is calculated. This is achieved by applying the chain rule on the snowflake parameters through the manipulation of the G-S expansion and by using the Green's Function of the G-S problem. Finally, the control needed to achieve the requested snowflake configuration is obtained by taking the pseudo-inverse of this equation and multiplying it by a weighting function.

This control enabled SFD minus, SFD plus, and exact SFD formations with varying σ, the distance between the X-points normalized to the minor radius, ranging from 0.08 to 0.5 in various scenarios. SFD was successfully integrated to an advanced tokamak scenario with $\beta_N = 3.0$ and H-factor of $H_{98(y,2)} \cong 1.35$. We achieved a 2.5 times increase in the flux expansion and a 2.5 reduction in peak heat flux for many energy confinement times (2–3 s) without any adverse effects to the core plasma, such as confinement.

Proposal: In order to take advantage of the advanced divertors, I propose to develop these algorithms and study their applicability to fusion power plant in currently operational test reactor: NSTX and DIII-D.

Current SFD can only achieve lower SFD. At NSTX-U, we will need a lower+upper SFD. This has never been accomplished before. It introduces four nulls and will require special algorithms to find the correct X-points at any time, avoiding the control oscillation due to discontinuity in X-point calculations. Another important issue is that, as an ST, NSTX-U does not have control coils on the inner wall, and the inner gap distance is not directly controlled by any coil. This leads to the coupling of the SFD parameters with the plasma inner gap distance. In addition, SFD control will interact with the vertical stability control due to the high elongation. Thus, a highly sophisticated coupled shape control system must be developed to address the aforementioned issues.

At DIII-D, the final divertor design is not settled. Currently, control was only used for the SFD option and a small range of conditions. In order to optimize a possible SFD for DIII-D, a more detailed parameter scan is necessary. Other divertors are under consideration: the X-divertor, which has larger distance between the two X-point, and the Super-X divertor, which has a very long separation between the X-point and the strike point. We propose to expand the SFD control to these scenarios and study the stability of this configuration and the realistic heat flux variation under different control scenarios.

References:

[1] D. Ryutov, "Geometrical properties of a "snowflake" divertor", Physics of Plasmas 14, 064502, 2007

[2] M. Kotschenreuthe et al., "The super X divertor (SXD) and a compact fusion neutron source (CFNS)", Physics of Plasmas 14, 072502, 2007

[3] P. M. Valanju et al., "Super-X divertors and high power density fusion devices" Physics of Plasmas 16, 056110, 2009

[4] B. LaBombard et al., "X-point target divertor concept and the Alcator DX high power divertor test facility", Bulletin of the American Physical Society, Vol. 58, No 16, 2013

[5] Project Staff, "NSTX Upgrade five year plan for FY2014-2018", accessed on November 18th 2014

[6] R.J. Buttery and the DIII-D Team, "DIII-D Research to Address Key Challenges for ITER and Fusion Energy", Proceedings of IAEA 2014, OV1-4

[7] Project Staff, "The DIII-D National fusion program five-year plan 2014–2018", https://fusion.gat.com/images/4/4b/A27526-5YP-final.pdf, accessed on November 18th 2014

[8] J.M. Canik et al., "Feasibility of Power and Particle Handling in an ST-FNSF and the Effects of Divertor Geometry," IEEE Transactions on Plasma Science, Vol. 42, No.3, pp.573, 579, March 2014

[9] E. Kolemen et al., "Burning Plasma Relevant Control Development: Advanced Magnetic Divertor Configurations, Divertor Detachment and Burn Control", Nuclear Fusion, under review, 2014.

[10] J.R. Ferron et al., "Real time equilibrium reconstruction for tokamak discharge control", Nuclear Fusion 38, 1055–66, 1998