Testing Advanced Divertors on NSTX-U

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The problem of heat and particle exhaust in fusion grade plasmas is, currently, recognized as serious enough that testing innovative approaches can make important contributions in our quest for fusion power. The NSTX-U, with great flexibility in the magnetic geometry of the SOL, allows various advanced divertors, such as X-Divertors (XD), to be created with the NSTX-U PF coils set; the extensive NSTX-U diagnostics provide just the infrastructure needed to study and analyze these geometries in detail. Recent theoretical and simulation advances, in the understanding of magnetic field structure in the SOL provide a suitable backdrop for the analysis. In particular, new simple-to-measure, physics-based metrics, such as the index DI, can be exploited to quantify edge and core plasma performance gains enabled by such divertors. Thus, the NSTX-U-IFS collaboration to quantitatively test advanced divertors is naturally indicated.

Background and Goals:
The PF coil set on NSTX-U can vary three basic metrics, measured at points along the divertor plate (especially at the location of the maximum heat flux):

1) F = flux expansion at the strike point can be varied by a factor of 15,
2) L = magnetic line length, in terms of plasma Coulomb collision mean free paths (strongly related to detachment in 1D models), can be varied by a factor of 1.3, and
3) DI = the amount of flaring near the divertor plate (see appendix 1), can be varied from around 1 to 10.

The core plasma shape can be held nearly constant while (F, L, DI) are varied.

Measuring the changes in the following quantities will provide important information for increasing advanced divertor predictive capabilities:

1) Heat flux profile on the divertor plate,
2) Onset and motion of detachment fronts from the plates,
3) Radiation profile near the divertor plate, and its evolution,
4) Upstream density and temperature in various stages of detachment,
5) Changes in radiation location (e.g., MARFE-like radiation)
6) Changes in core confinement (H factor) with detachment.

One key goal will be to find the effect of DI changes on the ability to reach stable, full detachment, i.e., find if larger (F, L, DI) allow detachment at lower upstream density, and if larger DI makes it easier to hold the detachment front steady.

In addition to the abovementioned 3 parameters, considering only the magnetic geometry of the SOL, there are multiple dimensionless parameters that could be important; values of these need to be considered that distinguish this experiment from others, and that may be particularly pertinent to future devices:

1) The “divertor aspect ratio”: DAR: the ratio of the divertor throat distance to the wetted distance on the plate
2) The “SOL normalized distance between the X-points”: the X-point distance divided by the width of the SOL around the core X-pt. If the ratio of these distances is less than 1, it corresponds to the “proximity condition” espoused by the Livermore group

Since experimental time is very expensive and limited, it is important to select experiments carefully to explore all the relevant physics. We wish to pursue experiments where there are larger values of both the “divertor aspect ratio” and the “SOL normalized distance between the X-points” is significantly larger than one. This regime is the opposite of the regime espoused by the snowflake group. Yet by putting the X-pint near the divertor plate, it is still possible to have large flux expansion and long line lengths, together with a flared geometry (DI >1).

This regime is also of great practical interest. For FNSF/reactor parameters, the SOL is much narrowed than the minor radius compared to today’s experiments. In such cases, it can require much less PF coil current to create a second X-pt near the divertor plate (XD), than to induce a new X-pt very near the core (snowflake).

Experiment outline, preparation, and diagnostic tasks:
In order to successfully perform these experiments, the collaboration will need to:
1) Generate NSTX-U magnetic equilibria in which the core plasma shape is held nearly fixed, while F, L, and DI at the expected strike point (highest heat flux) are varied. This can be done by creating a second X-point beyond the divertor plate, but near the strike point, and varying its distance from the strike point. The closer it gets to the strike point, the higher F, L, and D get. One can also vary F, L, and DI by changing the angular location of the second X-point.
2) For these configurations, the PF coil currents must be within limits – a task that seems quite feasible. Using these currents as targets, the PF coils will need to be programmed to produce these configurations.
3) The choice of the strike point should be dictated by the available diagnostics. Key diagnostics: IR cameras, divertor Langmuir Probes, divertor spectroscopy; divetor bolometry may not happen till later. The precise plasma shape will depend on what we find during the run, but likely an uuter Strike Point between 65 and 80 cm would work. The lower delta would probably be 0.4-0.5, and kappa probably between 1.8 and 2.2.
4) SOLPS simulations should be done on some of these configurations to estimate plasma operational and detachment parameters. SOLPS can be used as a guide for experimental design and then used as an interpretive tool to understand the experimental results.
5) Based on all these, a run plan will be decided. NSTX had a slow, somewhat uncontrolled density ramp in most discharges. We can modestly change that

ramp with fueling and lithium. NSTX-U might be the same in early experiments - we'll have to see. So a density ramp should be incorporated in the experiments, with a request to change the ramp rate and also obtain quasi-steady periods (say 0.2-0.3 sec) at particular density levels. This should not be a requirement, merely desirable.

The Divertor Index DI:

Consider two positions *a* and *b*, where b is the downstream terminus of an SOL field line on the divertor plate, and *a* is the position on that same field line that is closest to the core X-point (see Fig.1).

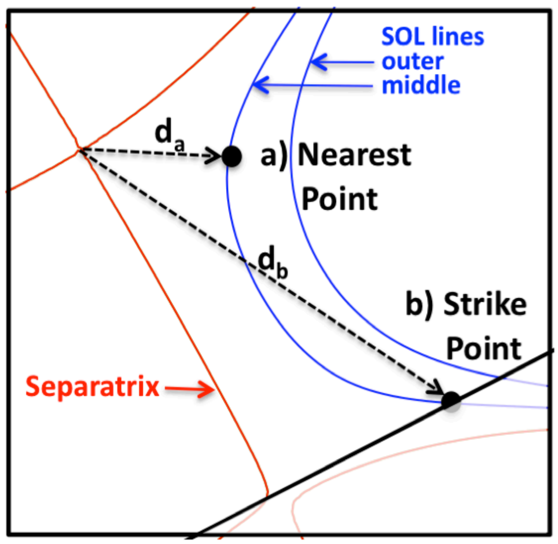


Figure 1

The ratio of the flux expansion at *b* to its value at *a* is B_a/B_b. (Throughout this paper we will use the symbol *B* to denote the poloidal magnetic field). The standard divertor magnetic fields B_{SD} varies *linearly* with distance *d* from the core X-point. Hence, the convergence of flux surfaces *relative to a standard divertor*, is given by the SOL Divertor Index (DI_{SOL}):

$$DI_{SOL} \equiv \frac{d_b / B_b}{d_a / B_a} = \frac{B_a}{B_b}\frac{d_b}{d_a} \qquad (1)$$

If $DI_{SOL} > 1$, the flux surfaces are more flared than a standard divertor near the target, and if $DI_{SOL} < 1$, it is more contracting than a standard divertor near the target. An easier-to-measure index DI can be defined by taking the limit when the point a becomes the main X-point.

Expected relationship between detachment dynamics and DI:

A detachment front is a transition layer between upstream SOL plasma and downstream neutrals. Its dynamics depends on plasma energy loss (from ionization, charge exchange, enhanced atomic radiation, etc.) in the "interaction area" of the front. A strong detachment front (initially formed near the plate) tends to move towards the main X-point bringing the cold plasma (sometimes termed an X-point MARFE) to the boundary of the main plasma. A larger (smaller) interaction area *with neutrals* will result in larger (smaller) energy losses from the plasma. Although the plasma behavior is determined by its parallel dynamics along a magnetic field, the neutral dynamics is not. In an axisymmetric configuration, the cross sectional area of interaction between the plasma and the neutral buffer depends on the shape of the plasma in the poloidal plane. The major mechanisms that affect the losses, and are controlled by the shape of the SOL near the divertor platre (DI) are:

1) Variation in the contact area between the plasma and the neutral buffer
2) Variation in the upstream plasma pressure due to parallel thermal conduction

As shown if Fig. 2a, In a standard divertor (SD, DI=1) the "interaction area" increases away from the divertor plate, thus pushing the front towards the main X-point. In an XD (DI>1), the opposite happens, so XD can stabilize the front.

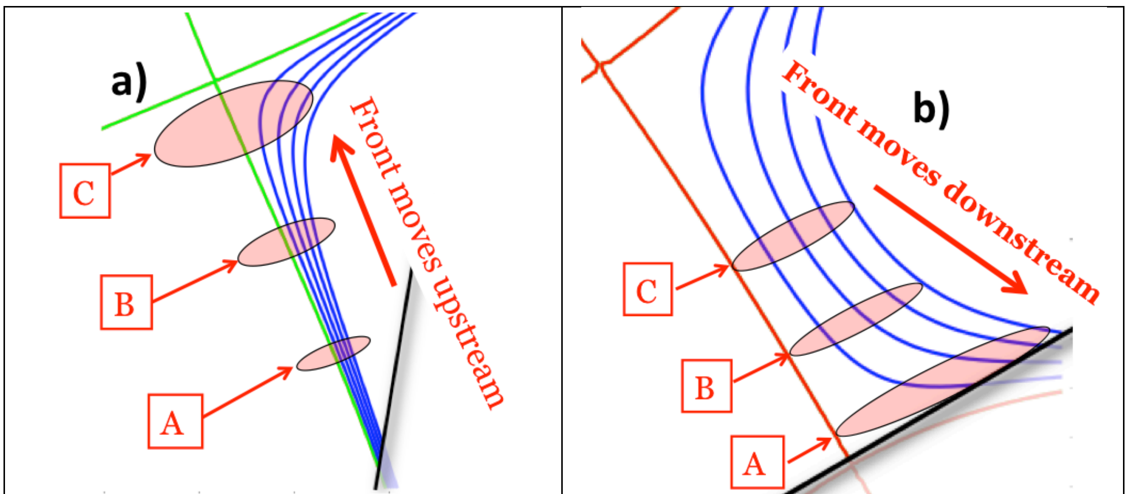
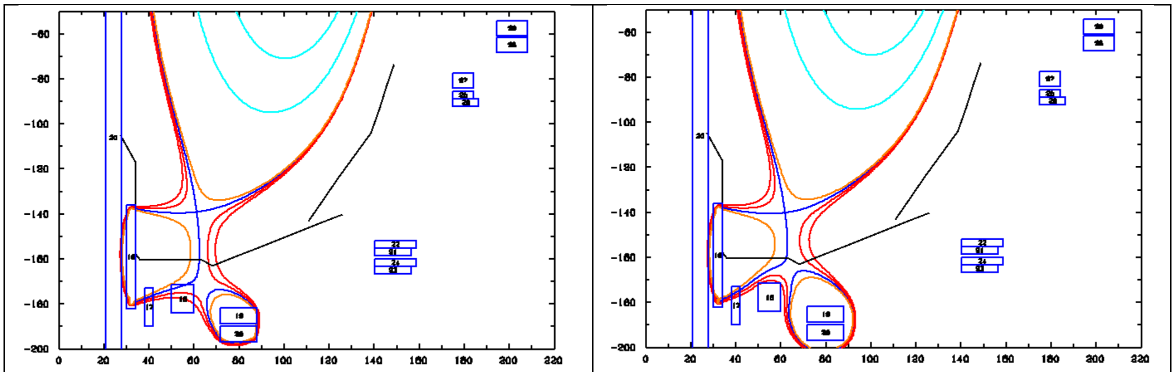


Figure 2: The plasma-neutral interaction area of a) a standard divertor increases as the detachment front moves toward the main X-point. Thus, energy losses increase, leading to an unstable feedback, so that the front moves toward the core X-point. An XD geometry b) with flared field lines near the plate reverses this feedback so the front could be arrested near the divertor plate.

Examples of CORSICA equilibria for NSTX-U:

Some examples of CORSICA equilibria for NSTX-U are shown in Fig. 3. These will have to be refined for the actual experiment. However, they show the range of equilibria that are possible while staying below the NSTX-U PF coil current limits.



a) Flux exp = 18, DI = 2.7

b) Flux exp = 52, DI = 6.5

Figure 3. CORSICA equilibria for NSTX-U coils and typical core plasma values. These show same core plasma with XDs with varying flux expansion and divertor index. These variations can be made continuously – starting from SD.

Examples of SOLPS simulations for advanced divertors:

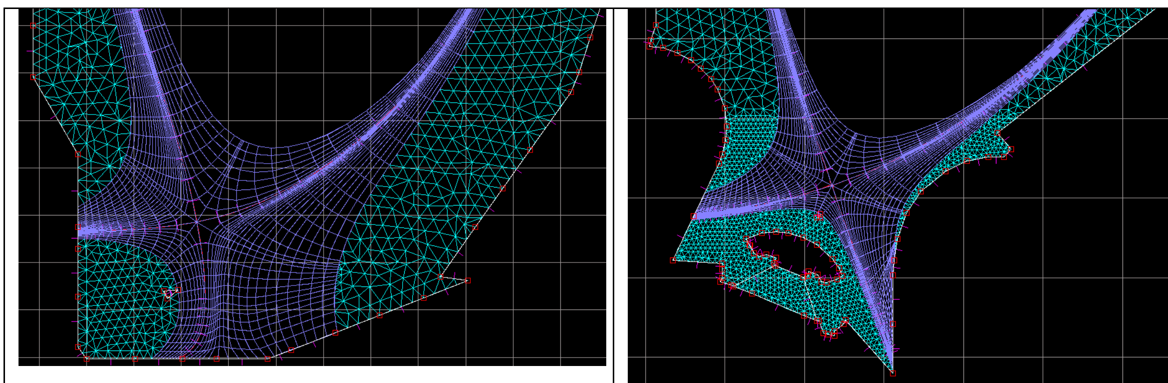


Figure 4. a) NSTX-U XD in SOLPS 5.1, using both the B2.5 mesh (purple) for plasma modeling, and the Eirene mesh (blue-green) for neutral tracking.

b) ITER XD in SOLPS 5.1 with Eirene mesh refinement in the divertor region for better neutral physics resolution.