Study of strike-point and flux-expansion control in diverted NSTX plasmas

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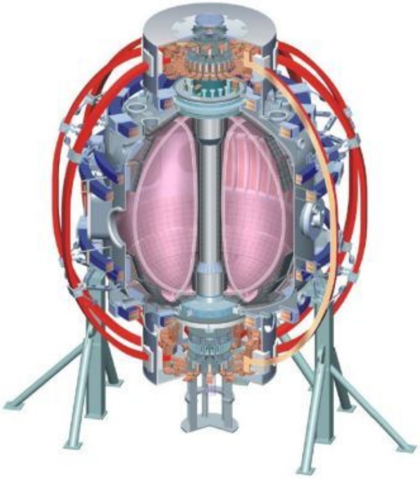

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Abstract

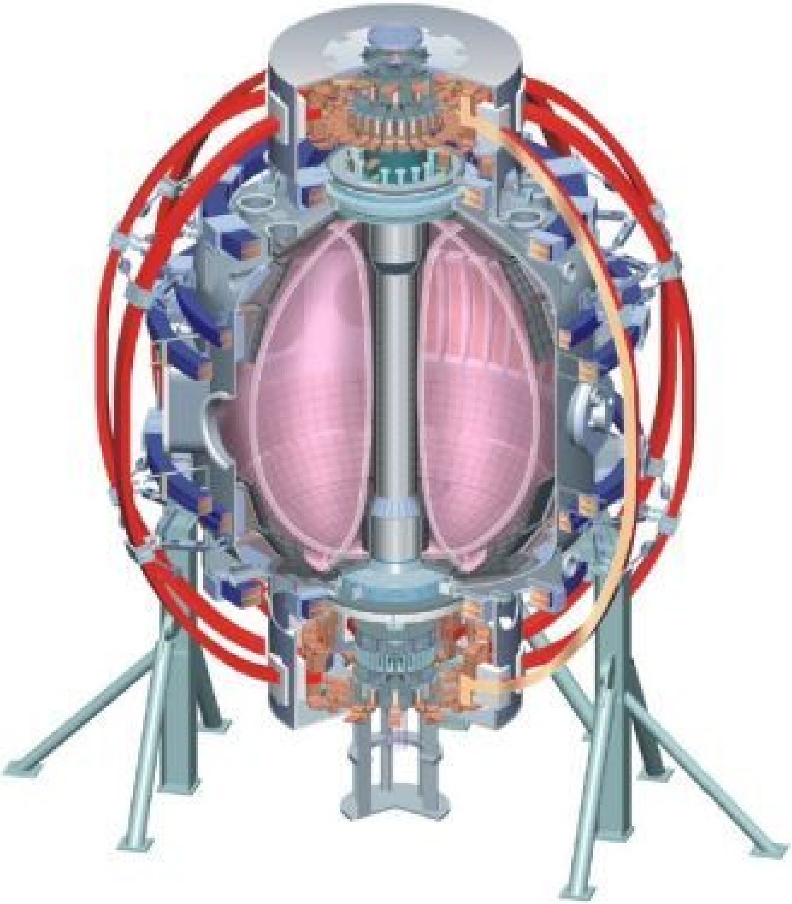
Tokamaks commonly use a poloidal divertor to tailor the trajectory and length of open magnetic field lines outside the main plasma volume. Optimization of the magnetic field configuration at the divertor is desirable to control the heat and particle flux to the material surfaces. Such optimization is important in NSTX, in the design of new divertor configuration for the proposed upgrade of NSTX, and for future Spherical Torus (ST) devices. In particular, simultaneous control of the strike point location and the flux expansion at the strike point is highly desirable to control the location and magnitude of peak heat and particle flux at the divertor target. The free-boundary equilibrium code ISOLVER is utilized and modified to assess the boundary shape implications of controlling the strike point location. The viability of simultaneous strike point and flux expansion control is also assessed.

Outline

- Overview of NSTX
- How to mitigate high heat flux in ST, NSTX Upgrade
- Role of divertor geometry in reducing high heat flux
- ISOLVER capabilities and simulation results
- Summary
- Future work

Overview of NSTX

- The National Spherical Torus Experiment (NSTX) is a medium sized, low aspect ratio spherical torus (ST) - a compact magnetic confinement device.
- In the ST, less magnetic field is required to maintain a given plasma pressure (high β) → potentially more efficient confinement of plasma energy for fusion applications

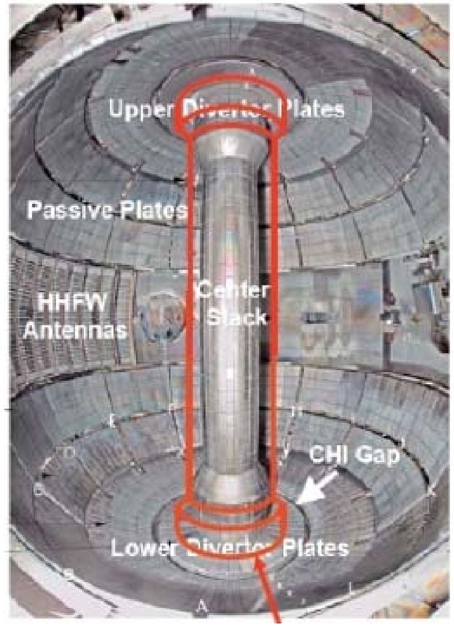


A major upgrade of NSTX is proposed in order to advance ST performance and understanding

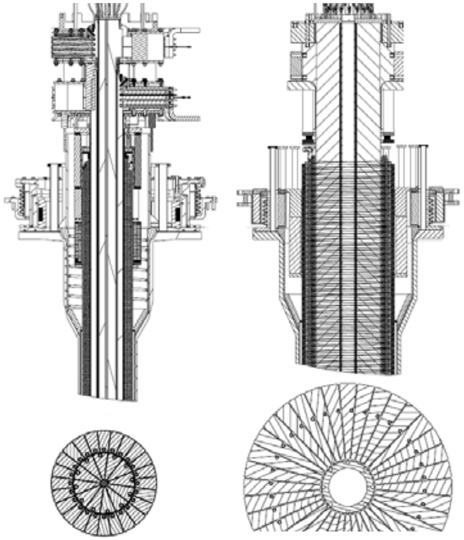
NSTX Upgrade → next factor of 2 increase in current, field, and NBI power

	NSTX	**NSTX Upgrade**	**Plasma-Material Interface Facility**	**Fusion Nuclear Science Facility**
Aspect Ratio = R_0 / a	≥ 1.3	≥ 1.5	≥ 1.7	≥ 1.5
Plasma Current (MA)	1	2	3.5	10
Toroidal Field (T)	0.5	1	2	2.5
P/R, P/S (MW/m,m^2)	**10, 0.2***	**20, 0.4***	**40, 0.7**	**40-60, 0.8-1.2**

***** *Includes 4MW of high-harmonic fast-wave (HHFW) heating power*



Outline of new center-stack (CS)

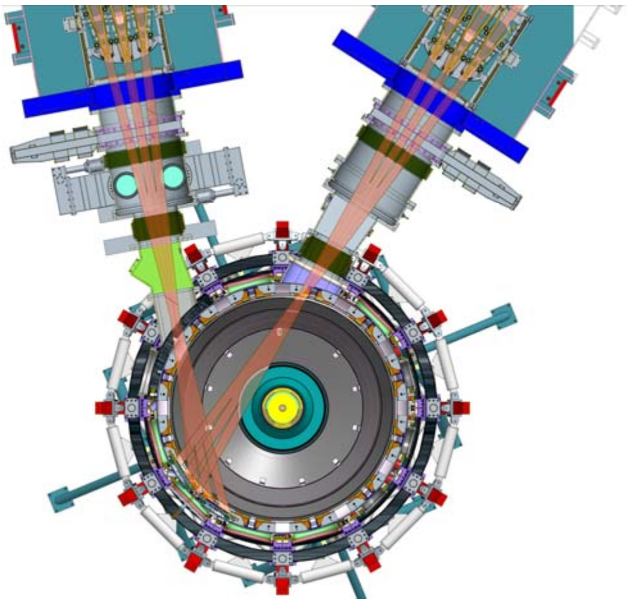


Present CS

New CS

TF OD = 20cm

TF OD = 40cm



New 2nd NBI
(R_{TAN}=110, 120, 130cm)

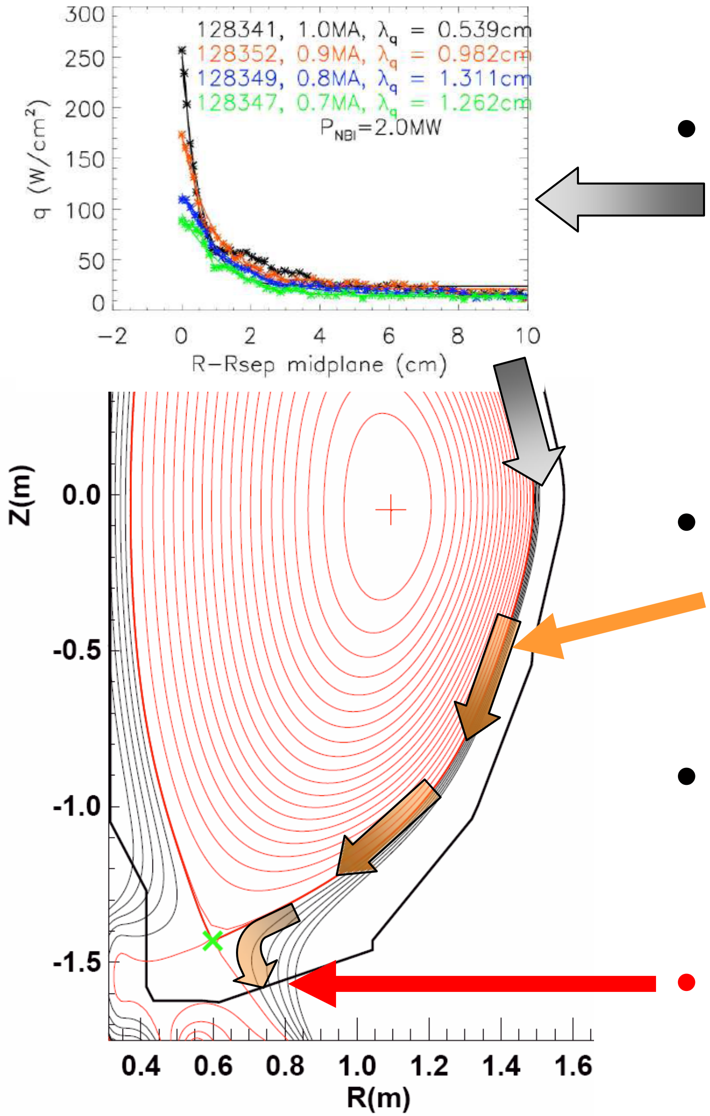
Present NBI
(R_{TAN}= 50, 60, 70cm)

How to mitigate high heat flux in NSTX-U?

- Enhance radiation from the plasma boundary to reduce the power conducted to PFCs.
- Use evaporated lithium coatings on PFCs to improve confinement, reduce required heating power
- Increase wetted-area of target plate to spread heat.
  - **THE FOCUS OF THIS POSTER**

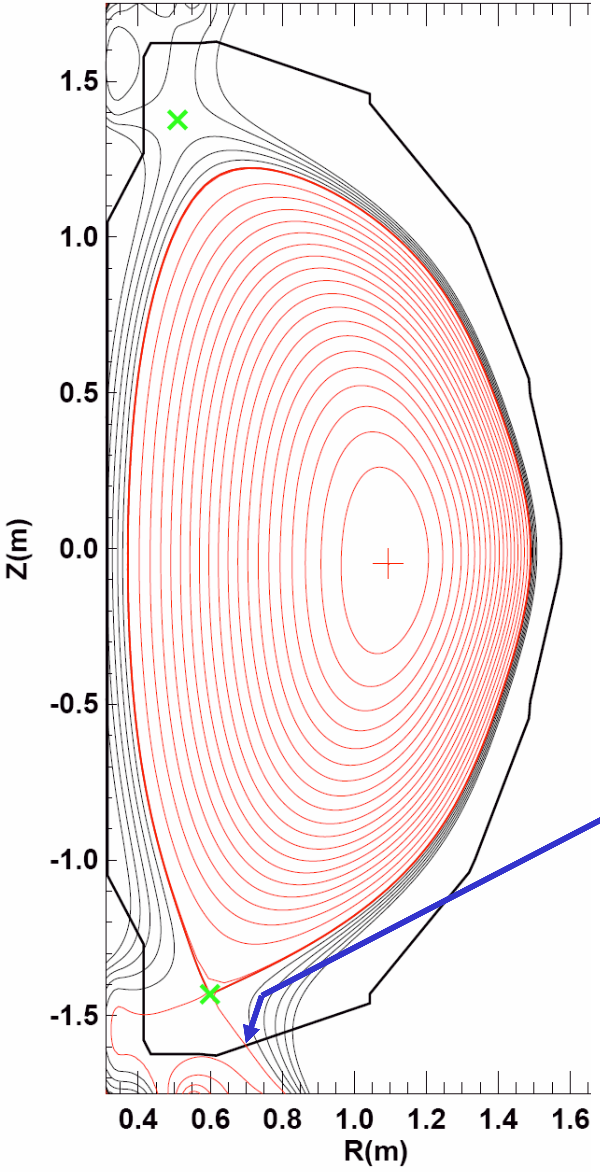
The heat flux in the scrape-off-layer (SOL) largely determines the peak divertor heat flux

***NSTX data** (Ahn, Maingi)*



- Mid-plane heat flux in scrape-off layer (SOL) varies as $q=q_{sep}\exp(-(R-R_{sep})/\lambda_q)$ where λ_q is mid-plane exponential heat-flux width near separatrix
  - A. Loarte et al., Journal of Nuclear Materials 266-269 (1999) 587-592
- Heat flows rapidly along **B** from plasma/SOL region to divertor region
- $q_{div} \equiv$ peak heat flux at the divertor - can be calculated from λ_q+ other parameters
- **Engineering limits q_{div} to < 10MW/m²**

The geometry of the divertor strongly influences the peak divertor heat flux



- Ability to remove heat from material surfaces limits **q_{div} to < $10 MW/m^2$** to avoid PFC melting/sublimation/damage

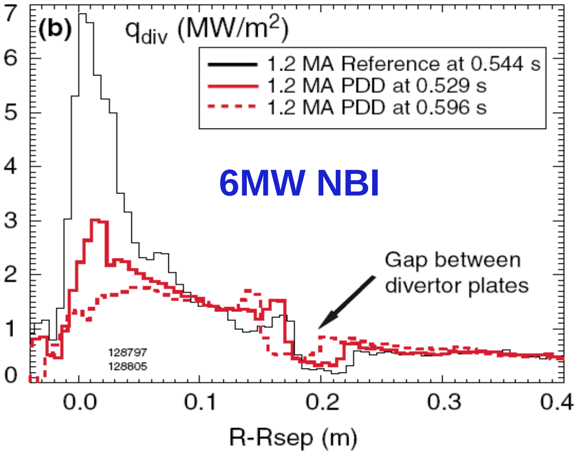
$$q_{div\text{-}out} = \frac{P_{heat}\,(1-f_{rad})\,f_{OD}\,\sin(\theta_P)}{2\pi R_S\, f_{exp}\, \lambda_q\, N_D}$$

- P_{heat} = Plasma core heating power (no radiation)
- f_{rad} = core radiated power fraction = P_{rad} / P_{heat}
- f_{OD} = fraction of power to outer (vs. inner) divertor
- θ_P = poloidal tilt angle between field line and plate
- R_S = radius of field-line strike-point at outer divertor
- N_D = number of divertors - N_D=2 for double-null
- f_{exp} = poloidal flux expansion at strike-point
- λ_q = exponential heat-flux width at outboard midplane

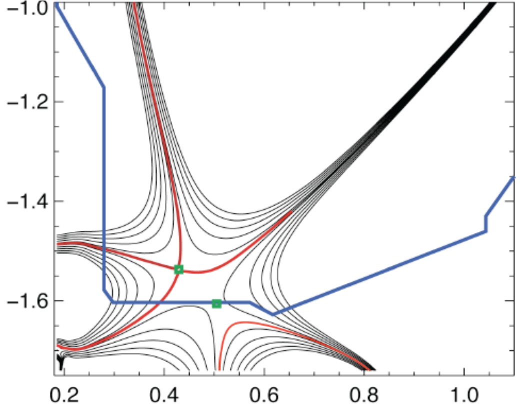
Increased R_S, f_{exp}, reduced θ_P → reduced q_{div}

A combination of plasma-material interface (PMI) solutions will likely be required to manage NSTX Upgrade heat flux

- **NSTX: High divertor heat flux was reduced through increased gas puffing and divertor radiation → partially detached divertor (PDD)**



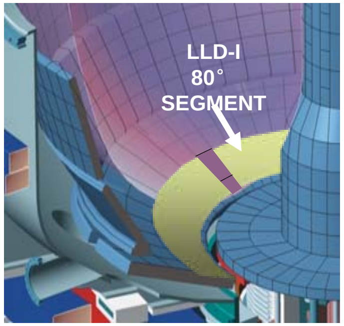
- **NSTX: has demonstrated the formation of high flux-expansion “snow-flake” divertor**



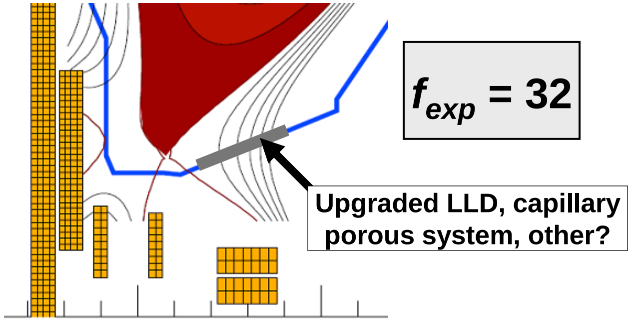
- **The PDD operating regime and other PMI solutions will be challenged in NSTX-U due to:**
  - 2-3⑨ higher input power
  - 1.5-2⑨ lower Greenwald fraction
  - 3-5⑨ longer pulse duration, leading to substantial increase in $T_{divertor}$

- **NSTX and NSTX-U will test the compatibility of high flux expansion, PDD, and a liquid lithium divertor (LLD) at 2⑨ higher power and 10⑨ higher energy**

- **NSTX LLD**



- **NSTX-U “snow-flake”:**



f_{exp} = 32

Upgraded LLD, capillary porous system, other?

The strike point location and flux expansion play important roles in reducing high heat flux

- What is the strike point?
  - The strike point is the point where magnetic field line connected with x-point strikes the target plate.

- What is the flux-expansion $\equiv f_{exp}$
  - f_{exp} is the ratio of $|\nabla\psi_{\mathrm{poloidal}}|$ at the outboard mid-plane to that at the strike point. The larger the flux-expansion, the more efficient is the spreading of heat over target plate.

- Control of strike-point and flux-expansion in NSTX becomes more important for NSTX Upgrade →
  - Need improved models → extend ISOLVER code

ISOLVER capabilities and simulation results

- What is ISOLVER
- Main algorithm
- Present boundary control algorithm
- Improved boundary control algorithm
- Results and analysis

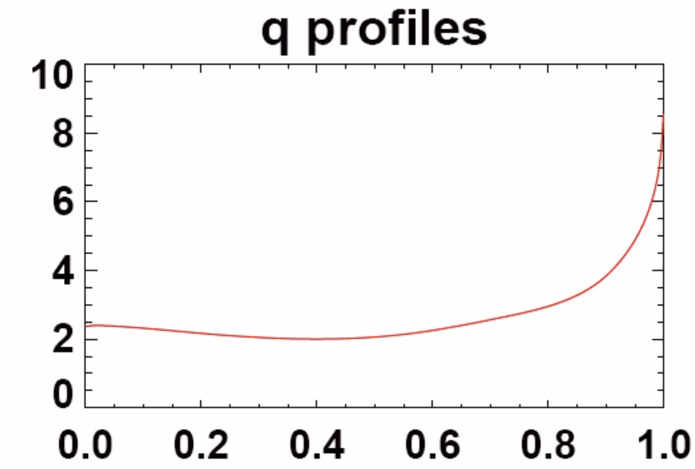
What is ISOLVER

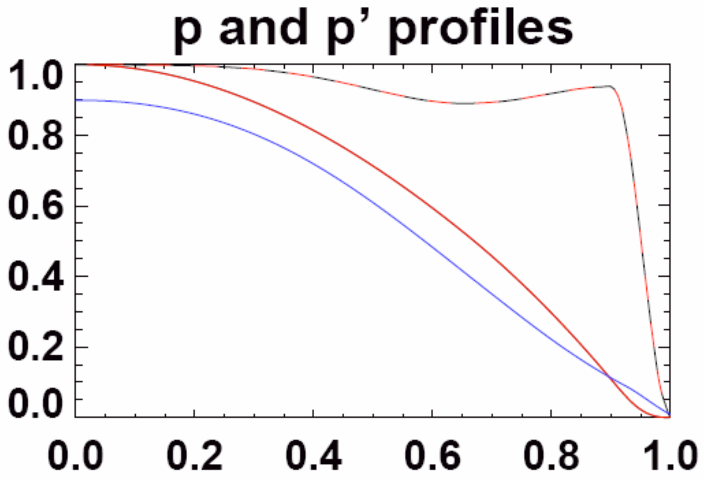
- A free-boundary auto-convergent axisymmetric equilibrium solver for Grad-Shafranov equation.

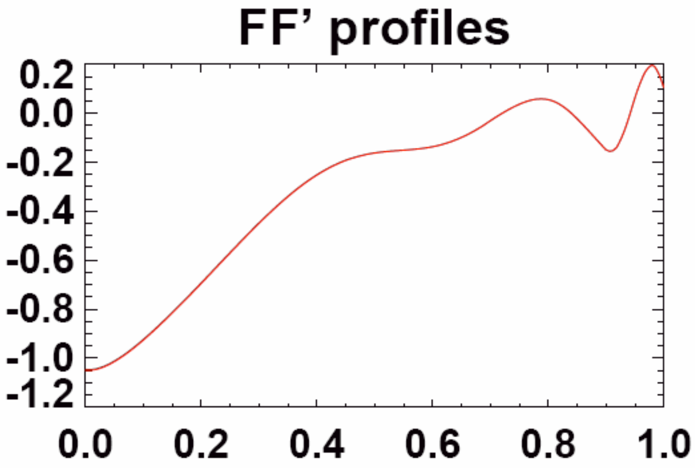
- It takes normalized pressure and current profile and boundary shape as input, matches a specified plasma current and beta, and computes coil currents as output.

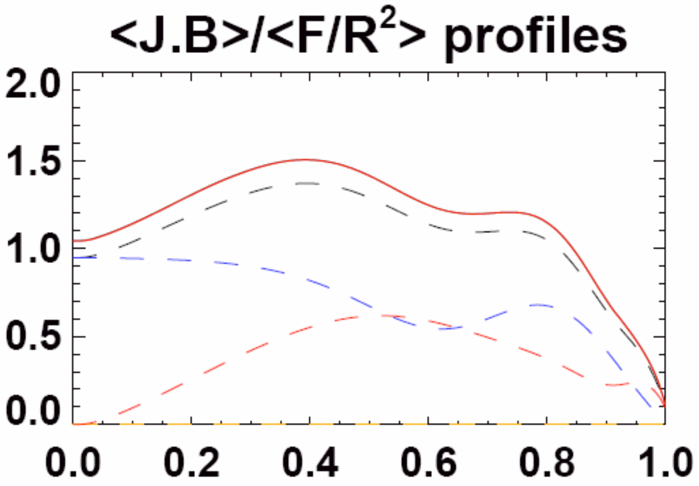
- The core components of ISOLVER include a modular, iterative algorithm, coupled with a fast, direct elliptic solver to the Grad-Shafranov equation.

Profiles used for 2MA, 1T NSTX Upgrade target plasma


q profiles


p and p’ profiles


FF’ profiles


<J.B>/<F/R^2> profiles

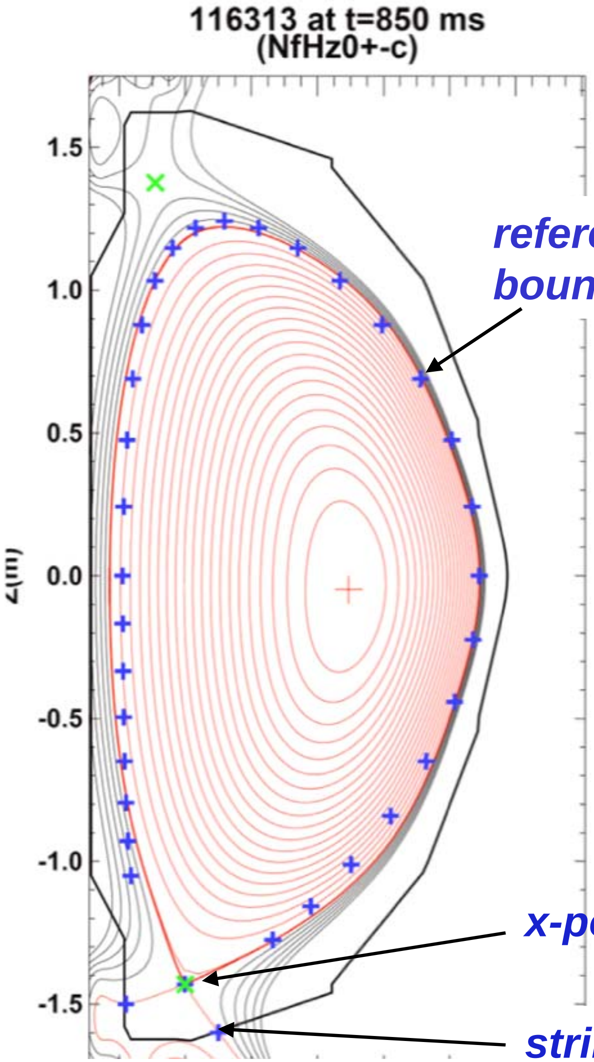
ISOLVER computes the coil currents which best-fits the last-closed flux surface to the shape of the reference surface

- **Previous version**

Use the experimental data to define the reference surface, but strike points are not included as reference points

- **Improved version**

Use the experimental data, combined with the user-specified parameters to define the reference surface and **include <u>strike point</u> specification to better control the magnetic field configuration**



reference boundary point

x-point

strike-point

Basic logic underlying the determination of coil currents

- How do we fit the reference surface to the flux surface?

  To make a surface a flux surface, that means every point on the surface has the same flux. The total flux at each point is consisted of flux from plasma and coils: $\Psi = \Psi_p + \Psi_c$ .On the surface, the flux is constant which means $\Delta\Psi_p + \Delta\Psi_c = 0$, where the comparison is conducted between reference surface and reference point.

- How do we match the x-point and strike points?

  To make a point become a x-point on the flux surface, two conditions must be satisfied: 1) $\Psi =$ reference flux; 2) $\nabla\Psi = 0$.

  To match the strike point on the flux surface, to make the flux on strike point equal to reference flux would be enough.

ISOLVER algorithm for computing coil currents

- Definitions:
  - Total poloidal flux $\psi \equiv \psi_{coil} + \psi_{plasma}$
  - **Coil flux = ψ_{coil} at location R_i, Z_i = $G_{ij} \bullet I_j$** (“j” is the coil current index)
  - Reference flux $\psi_{ref} \equiv$ poloidal flux at outboard midplane at $R=R_{max}$
- For boundary flux control, request ψ = constant on desired plasma boundary ➔ $\Delta\psi \equiv \psi - \psi_{ref} = 0$
  - Coil currents for flux control determined from $\Delta\psi_{coil} = -\Delta\psi_{plasma}$
- For X-point control, request $\nabla\psi = 0$ at desired X-point location
  - Coil currents for strike control determined from $\nabla\psi_{coil} = -\nabla\psi_{plasma}$
- For current control, request I_{coil} = user specified current
  - Default is to request $I_{coil} = 0$ with small weighting to minimize coil current and magnetic energy

ISOLVER flowchart of the solving for coil currents

Compute flux and flux gradient on reference surface from plasma and coils, respectively

Compute flux at reference point from plasma and coils, separately

Compute flux difference between reference surface and reference point from plasma and coils, separately

Compute weighting function for boundary flux and flux gradient

Construct matrices using the calculation results above

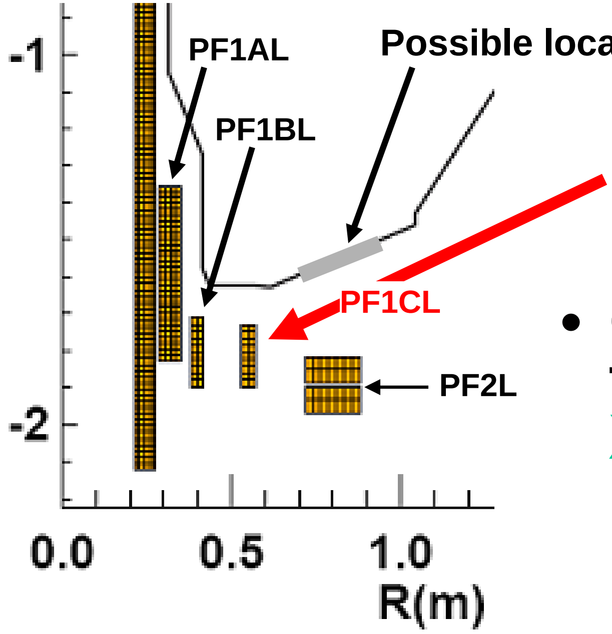
Use least square fitting method to solve

ISOLVER matrix equation for computing coil current array I_j

$$\begin{pmatrix}\Delta G_{ij}\\ \partial G_{ij} / \partial R\\ \partial G_{ij} / \partial Z\\ \underset{\uparrow \textit{Kronecker delta}}{\delta_{kj}}\end{pmatrix}\begin{bmatrix} \\ I_j \\ \\ \end{bmatrix} = \begin{pmatrix}-\Delta\psi_{\text{plasma-i}}\\ -\partial\psi_{\text{plasma-i}} / \partial R\\ -\partial\psi_{\text{plasma-i}} / \partial Z\\ I_{\text{request-k}}\end{pmatrix}\begin{array}{l}\leftarrow \textit{Shape}\\ \left.\begin{array}{l}\leftarrow\\ \leftarrow\end{array}\right\rangle \textit{X-point}\\ \leftarrow \textit{Coil currents}\end{array}$$

- User specifies relative weighting for:
  - Shape moments: elongation, triangularity, squareness, …
  - X-point at $Z=Z_{min,max}$ boundary location
  - Requested coil currents
- Compute weighted least-squares solution for coil current I_j

New ISOLVER algorithm + additional PF1CL coil enables variation of flux expansion with fixed strike-point location

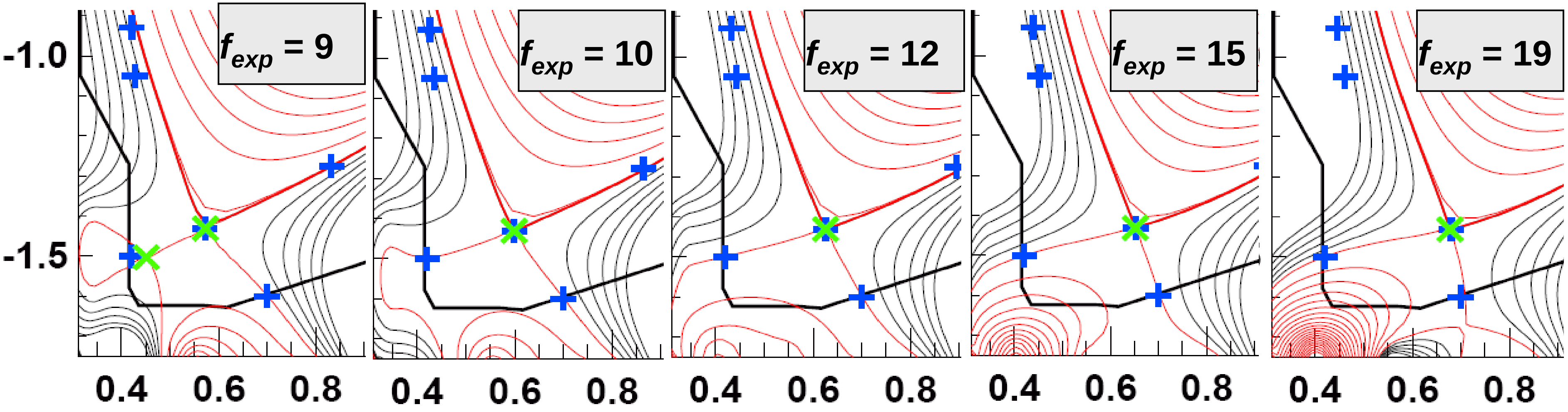


Possible location of liquid lithium divertor

Additional coil added in NSTX-U = PF1C

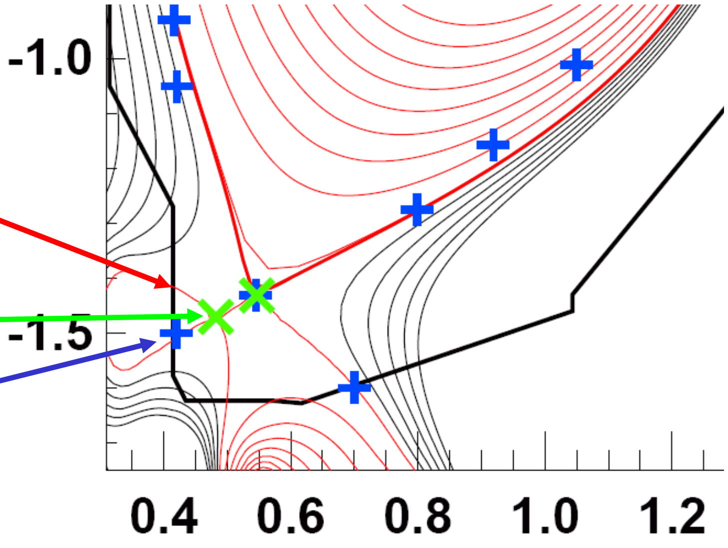
- Combination of PF1A,B,C + PF2 enables **flux expansion variation** with fixed X-point height and strike-point location:

Outboard poloidal flux expansion factor $f_{exp} \equiv |\nabla\psi|_{\text{mid-plane}} / |\nabla\psi|_{\text{strike-point}}$

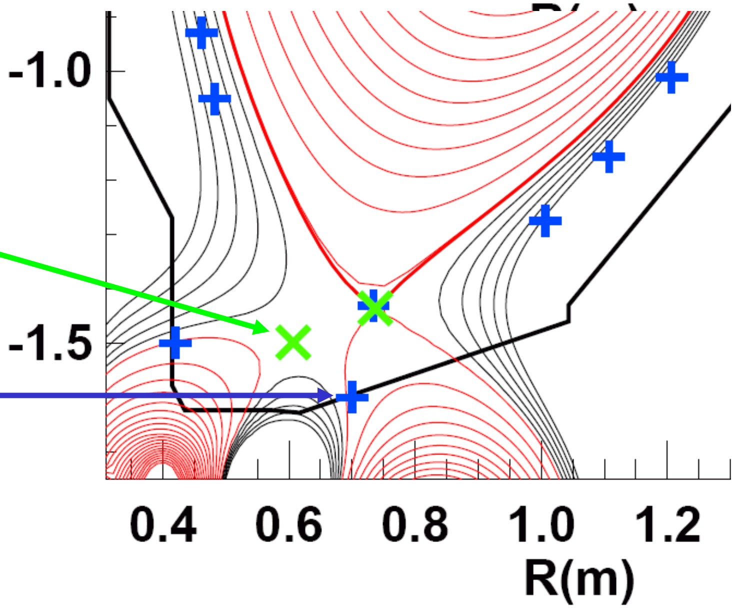


For small and large X-point radius, the strike-point control algorithm can fail due to the entrance of secondary X-points

- For this equilibrium, for small X-point radius, the actual inboard strike point location is modified because a 2nd X-point enters near the requested strike point



- For this equilibrium, for large X-point radius, a 2nd X-point enters near the target X-point, and connects the outboard strike-point to the private-flux region instead of the outboard SOL region



Summary

- Simultaneous control of strike point and flux expansion is highly desirable to controllably spread the heat exhaust on the edge of plasma, which is critically important for the upgrade of NSTX and future Spherical Torus

- Improved free boundary equilibrium code ISOLVER shows strong capability to control the equilibrium and better control of plasma boundary shape.

- Additional PF coils enhance heat exhaust flexibility in NSTX-U

Future work

- Improvement of the robustness of simultaneous control of strike point and flux expansion needs further investigation
- Facilitate the upgrade of NSTX PF coils using simulation

Sign-up