NSTX-U

NSTX-U is sponsored by the
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Radiative divertor optimization for NSTX Upgrade based on open geometry standard divertor experiments in NSTX

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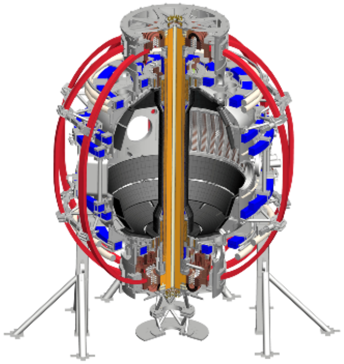
PPPL



Abstract

Recent analyses of NSTX divertor experiments suggest a way to optimize the standard open geometry divertor configuration for partial detachment with deuterium puffing and intrinsic carbon radiation. Results from the NSTX experiments and the divertor transport and radiation model obtained with the multi-fluid code UEDGE are used to show that detachment onset and properties are sensitive to 1) placing the neutral gas source in the vicinity of the strike point, 2) directing the recycling neutrals toward the separatrix by adjusting the poloidal separatrix angle, and 3) entrapping neutrals by plasma plugging via the high poloidal magnetic flux expansion configuration. These findings will be tested in NSTX Upgrade, where H-mode scenarios with 2 MA, 1 T, 10 MW NBI-heated discharges and 5 s flattop are predicted to produce unmitigated peak divertor heat fluxes above 10 MW/m2, necessitating the scrape-off layer power sharing between upper and lower divertors and inducing dissipative losses.

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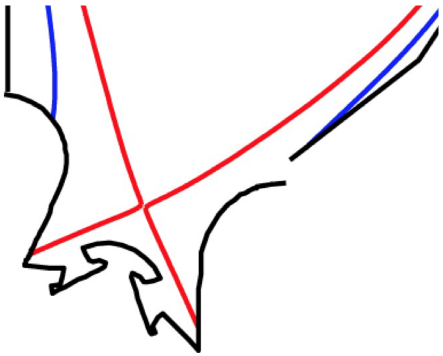
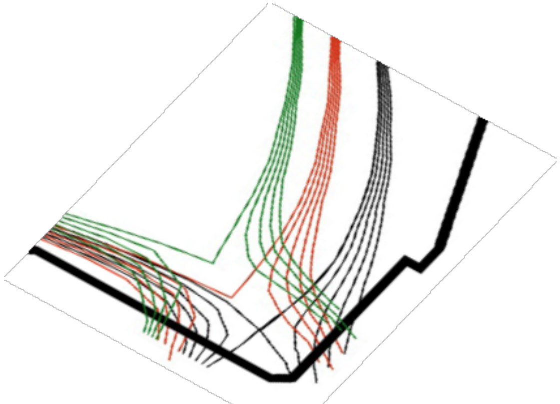
Summary

- Compact ST divertor geometry in combination with high plasma current and high input power present significant difficulties for radiative detachment regimes
- NSTX experiments suggest that optimization (for detachment) of an open unpumped divertor may be possible via standard magnetic divertor configuration
- ST plasma configurations with high-triangularity naturally produce high-flux expansion divertor with "vertical-plate divertor" characteristics (despite being an open horizontal plate divertor)



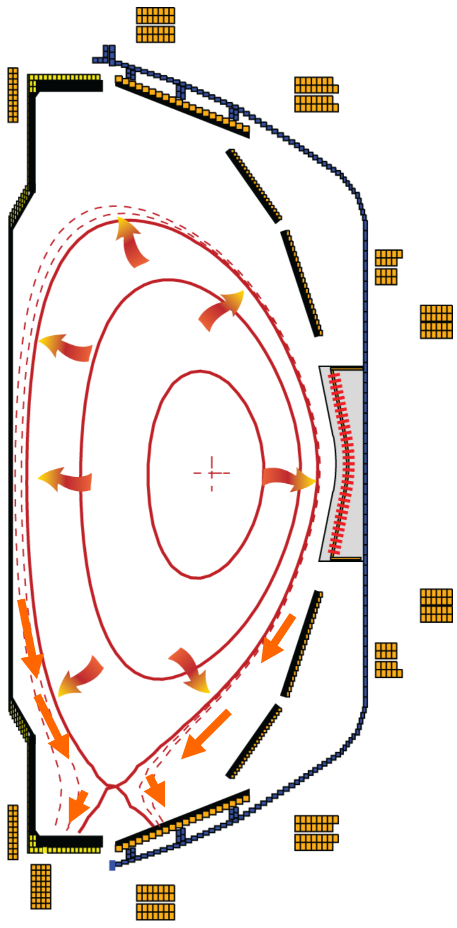
ITER vertical target baffled divertor



NSTX open horizontal target divertor (rotated 45 degrees)

Significant gaps exist between present divertor solutions and future device requirements

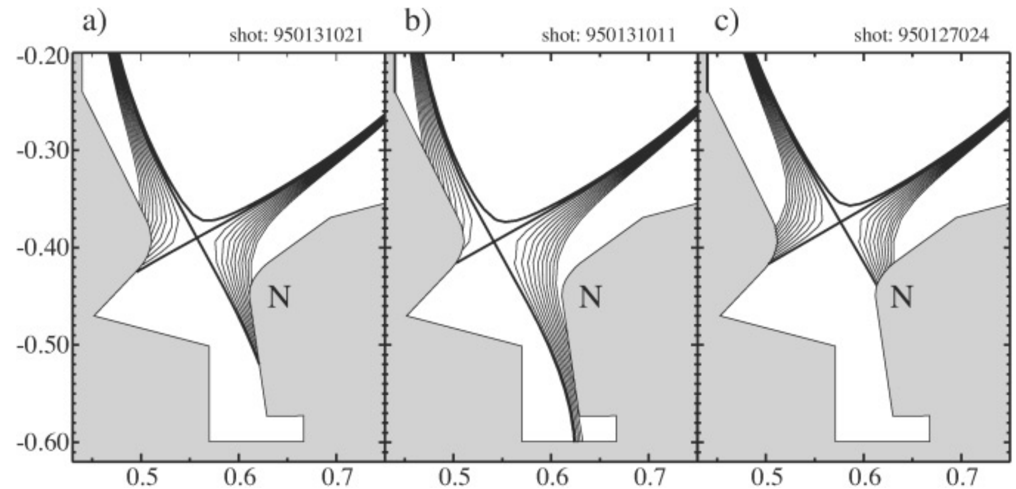
- **Critical divertor tasks**
  - Power exhaust
  - D/T and He pumping
  - Impurity source reduction
  - Impurity screening
- **Outstanding issues**
  - Steady-state heat flux
    - Technological limit $q_{peak} \leq$ 5-15 MW/m^2
    - ITER: $q_{peak} \leq$ 10 MW/m^2 (Mitigated)
    - DEMO: $q_{peak} \leq$ 150 MW/m^2 (Unmitigated)
  - ELM energy, target peak temperature
    - Melting limit 0.1-0.5 MJ/m^2
    - DEMO: Unmitigated, ≥ 10 MJ/m^2
  - Impurity erosion
    - Divertor target T_e < 5-10 eV

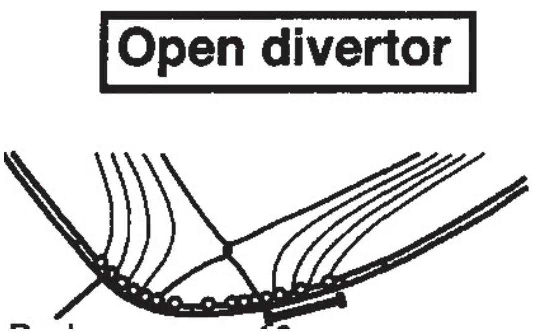


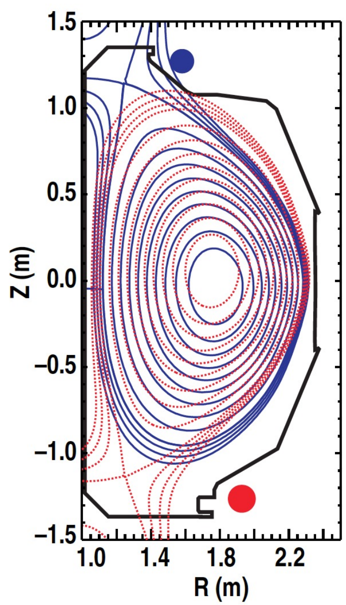
NSTX

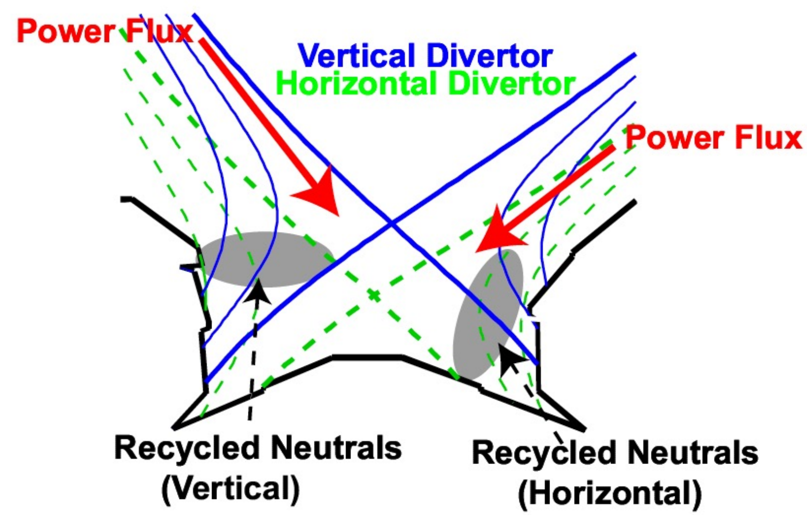
Divertor plasma facing component and hardware geometry plays critical role in detachment

- Open divertor
- Baffled divertor
- Vertical target divertor









A. Loarte, Plasma Phys. Control. Fusion 43 (2001) R183–R224

B. Lipschultz, Fus. Sci. Tech. 51 (2007) 379

Magnetic geometry impacts heat fluxes q_{peak} (divertor target), $q_{||}$ (divertor SOL) and detachment

- In high-recycling conditions

$$q_{\mathrm{peak}} \sim \frac{P_{\mathrm{div}}}{A_{\mathrm{wet}}} = \frac{P_{\mathrm{heat}}(1-f_{\mathrm{rad}})f_{\mathrm{out/tot}}f_{\mathrm{down/tot}}(1-f_{\mathrm{pfr}})\sin\alpha}{N_{\mathrm{div}}2\pi R_{\mathrm{SP}}f_{\mathrm{exp}}\lambda_{q_{\|}}}$$

- Detachment criteria

$$\frac{14}{3}c_Z L_Z n_u^2 l_{\|} > q_u,$$

Divertor physics is inherently 2D or even 3D

- Parallel / cross-field transport and turbulence
- Radiation front (detachment) stability
- Neutral pressure / density distribution

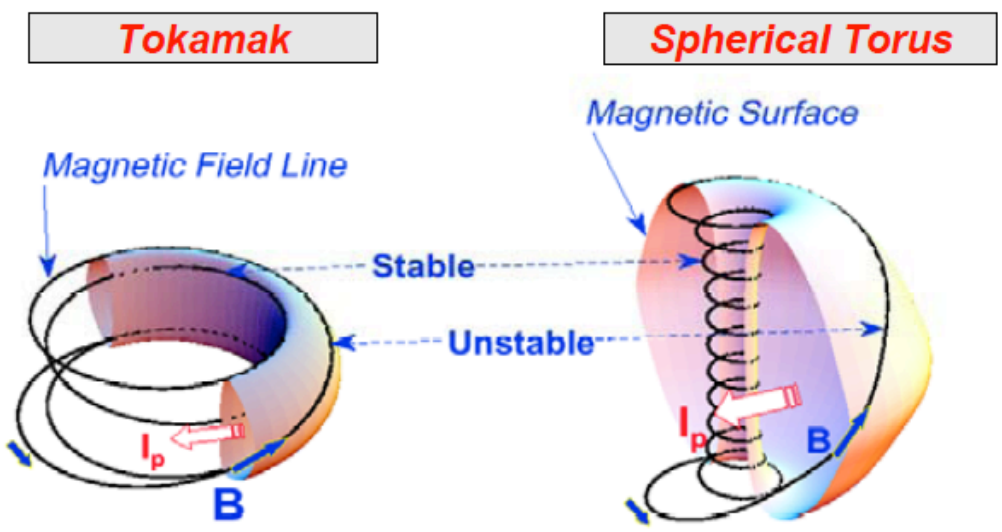
$$T_e^{div} \sim \frac{q_{\|}^{10/7}(1-f_{loss})^2}{l_{\|}^{4/7}n_u^2} \sim \frac{P_{SOL}^{10/7}(1-P_{rad}^{div}/P_{SOL})^2}{l_{\|}^{4/7}\lambda_q^{10//7}n_u^2}$$

Pitcher C S and Stangeby P C 1997 Plasma Phys. Control. Fusion 39 779

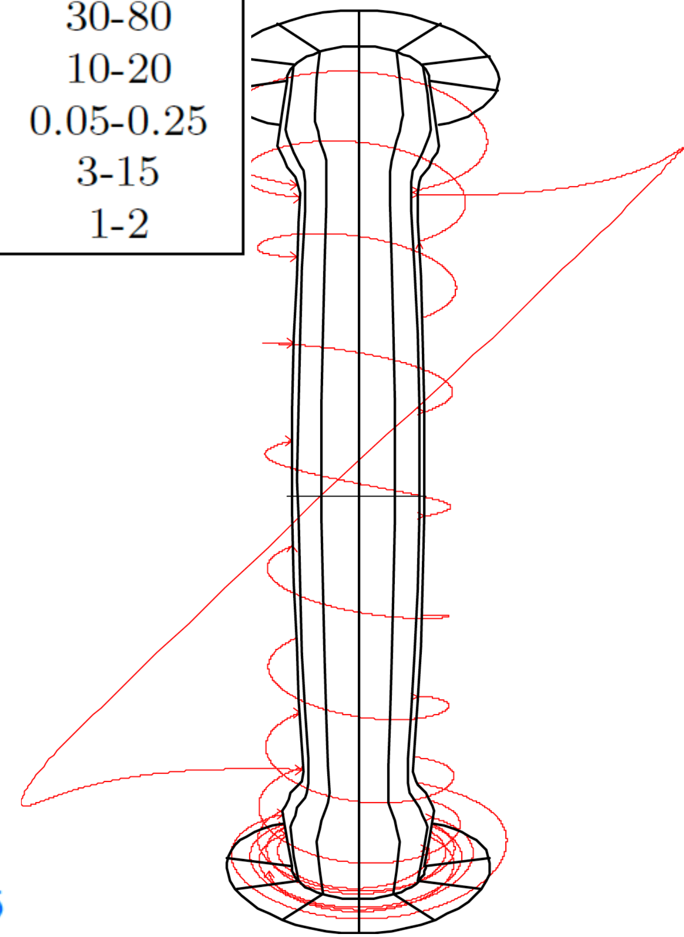
B. Lipschultz, Fus. Sci. Tech. 51 (2007) 379

SOL / divertor geometric properties are different in spherical tori and large aspect ratio tokamaks

Quantity	NSTX	Tokamak
Aspect ratio	1.4-1.5	2.7
In-out plasma boundary area ratio	1:3	2:3
Midplane to target connection length L_c (m)	8-10	30-80
X-point to target parallel length L_x (m)	5-7	10-20
X-point to target poloidal length L_p (m)	0.05-0.15	0.05-0.25
Poloidal magnetic flux expansion f_m at OSP	16-24	3-15
Magnetic field angle at target (degree)	2-5	1-2

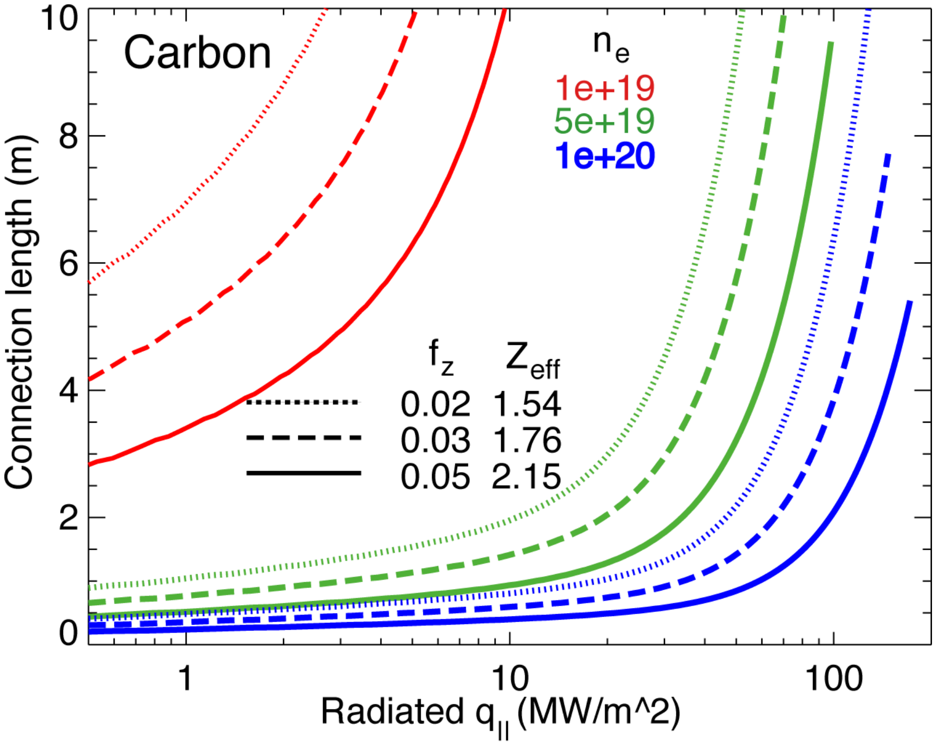


R/a=2.5-4, κ=1.5-2, q_e=3-5   R/a=1.1-1.6, κ=2-3, q_e=6-15



High f_{rad} can be achieved with carbon in NSTX divertor at high n_e and n_z

- Hulse-Post non-coronal radiative cooling curves for low Z impurities for n_0/n_e, n_e-τ_{recy}
- Calculate max $q_{||}$ that can be radiated
- Express max $q_{||}$ as function of distance from heat source for range of f_z
  (Post JNM 220-222, 1014 (1995))
- Power losses due to deuterium P_{rad} and ionization not considered
- For NSTX, use n_0 = 0.1 % and n_e-τ_{recy} = n_e x 1e-3 s

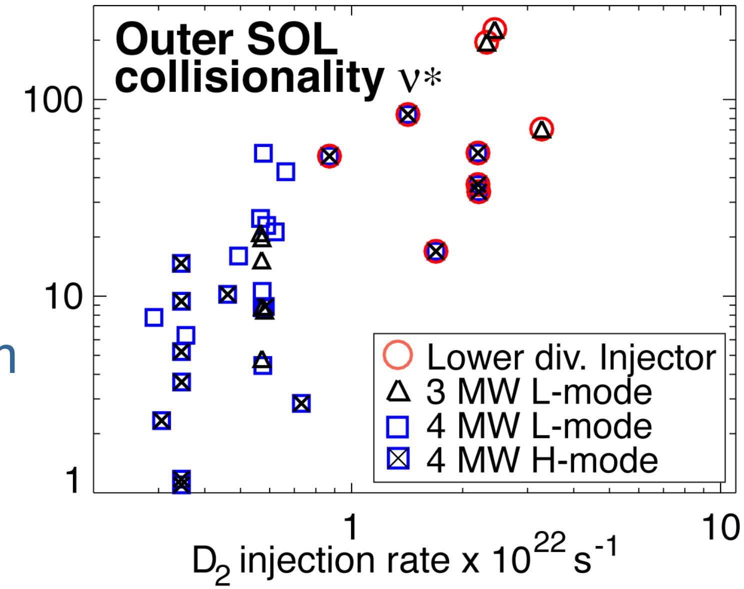


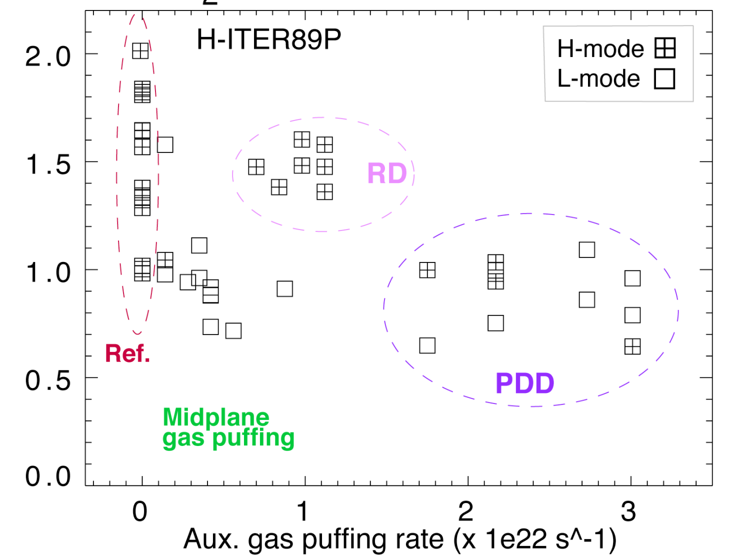
$$q_{||} = -\kappa_0 T_e^{5/2} \frac{\partial T_e}{\partial x}$$

$$\frac{\partial q_{||}}{\partial x} = -n_e n_z L_Z(T_e)$$

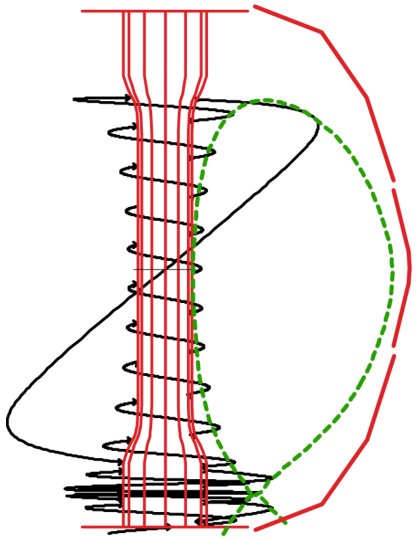
Divertor detachment sensitive to gas injection location and divertor configuration in NSTX

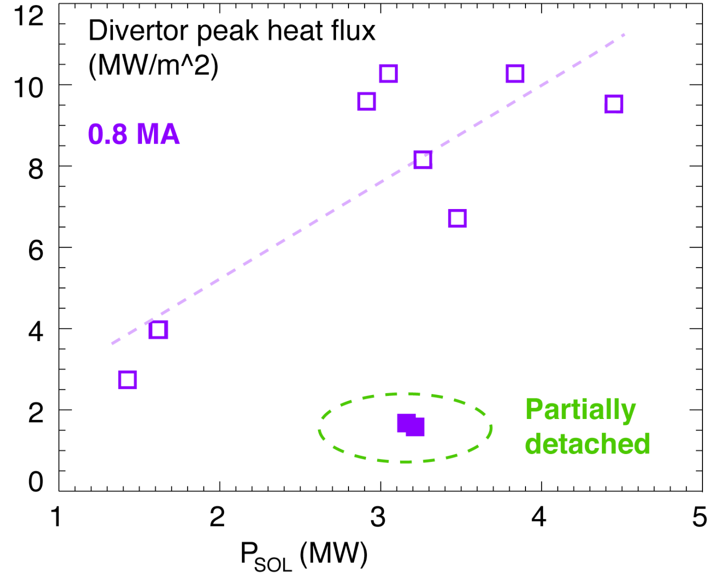
- Gas injection location
  - Only divertor gas injection led to partial OSP detachment compatible with core confinement
  - $q_{||} \leq 150$ MW/m^2 and q_{pk}=6-12 MW/m^2
  - Large radiated power and momentum losses needed
- High flux expansion divertor
  - High flux expansion divertor may be less transparent to neutrals, better recycling neutrals retention, “plasma plugging”
- Divertor separatrix angle with PFCs
  - Lower angle leads to more recycling neutrals directed toward the separatrix (cf. angle > 90 deg. - neutrals away from separatrix)
  - Similar to “vertical plate” divertor ?

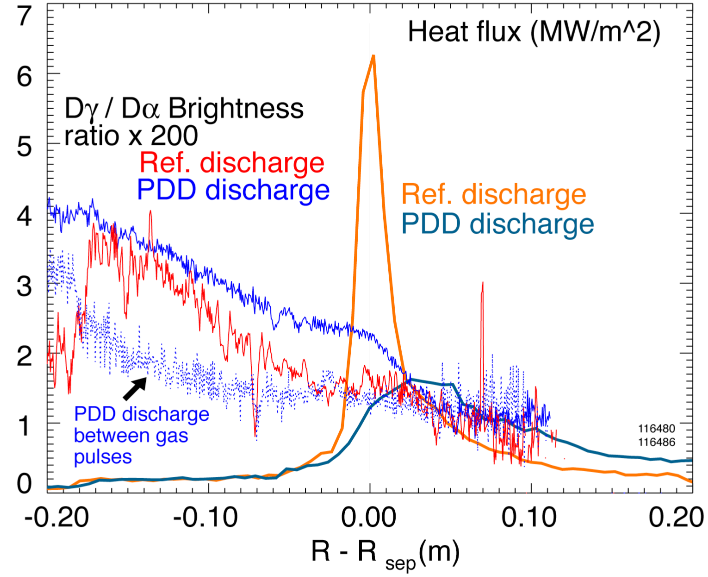




In low κ, δ configuration with rad. divertor, q_{peak} reduced albeit with confinement degradation

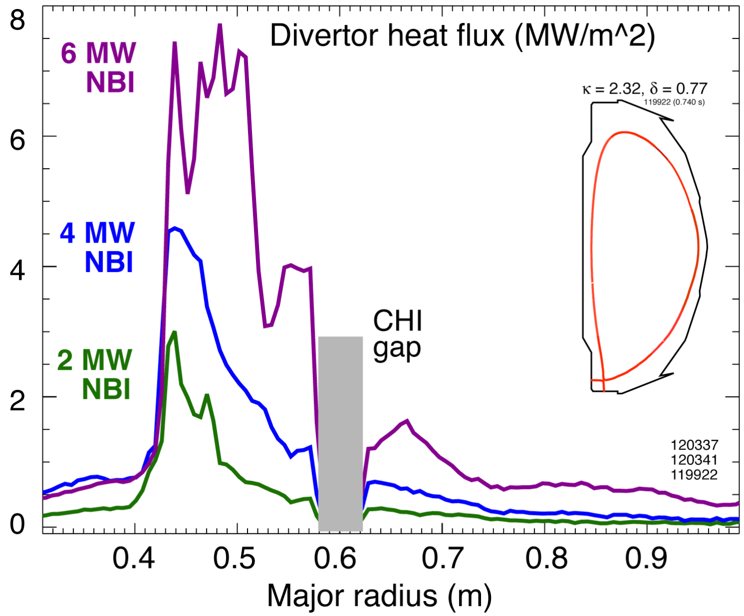


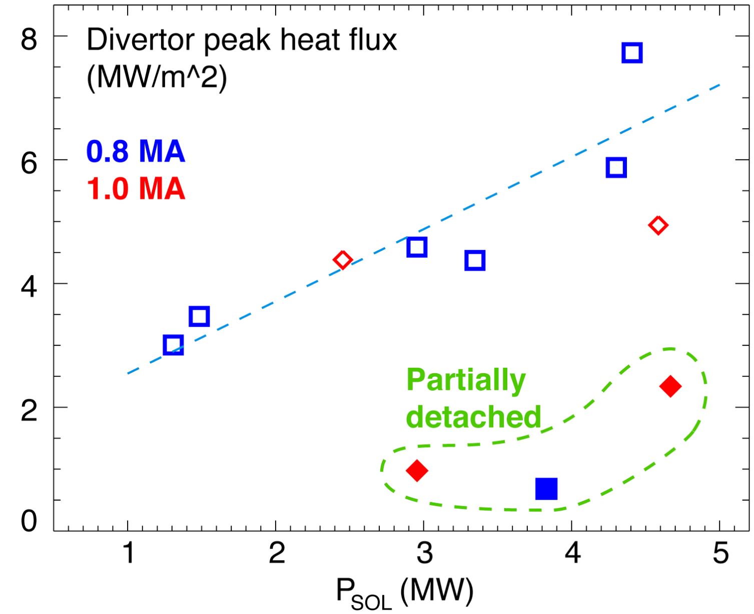




- Peak heat flux in outer divertor (Maingi JNM 363-365, 196 (2007)):
  - ITER-level q_{out}< 10 MW/m^2
  - Scaling of q_{peak}: linear with P_{sol} (P_{NBI}), linear-monotonic with I_p
  - Large q_{peak} asymmetry - 2-10; inner divertor always detached
- Experiments using D_2 injection (Soukhanovskii IAEA 2006):
  - q_{peak} reduced by up to 60 % in transient PDD regime
  - X-point MARFE degraded confinement within 2-3 x τ_E

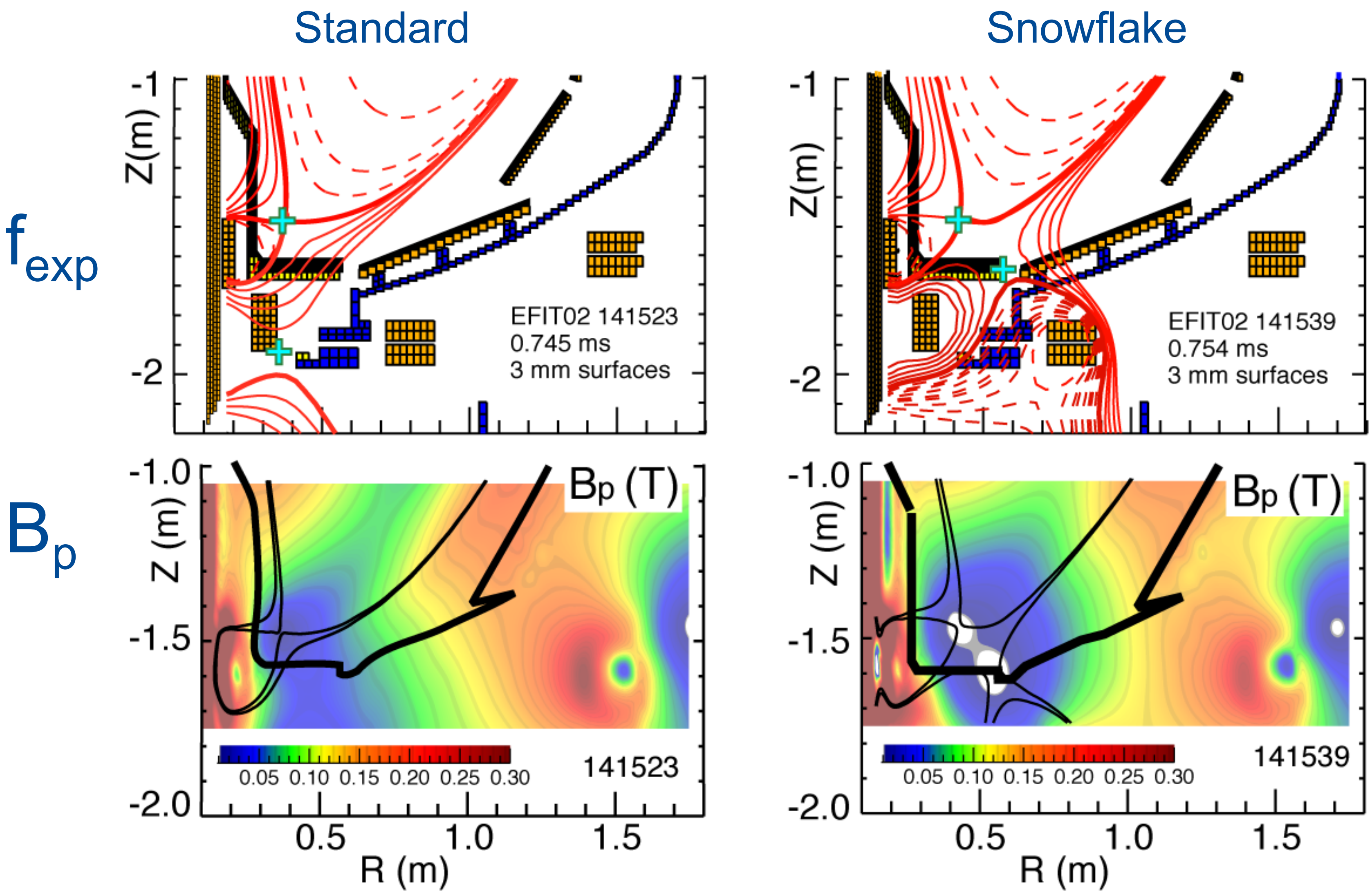
Good core plasma performance and significant q_{peak} reduction with PDD obtained at high κ, δ





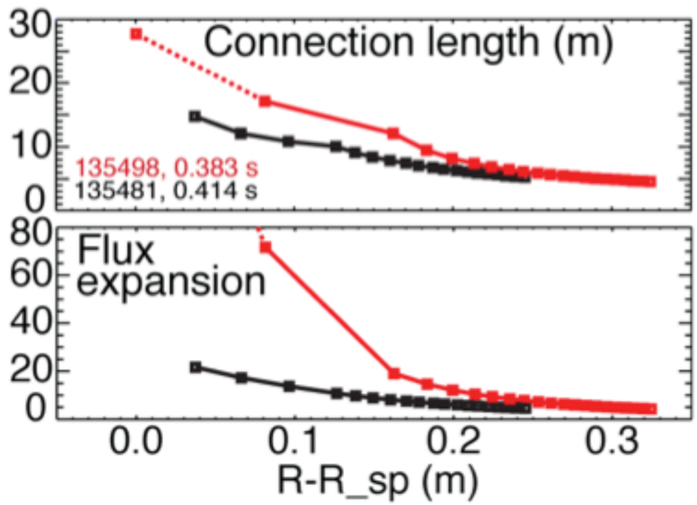
- Experiments conducted in 0.8-1.0 MA 4-6 MW NBI discharges with κ=2.2-2.3, δ=0.6-0.75 (Soukhanovskii APS 2007)
- Obtained partially detached divertor (PDD) outer strike point using divertor D_2 injection, however, P_{rad} due to intrinsic carbon and helium
- q_{peak} reduced by 60 - 80 % in PDD phase with min. confinement degradation

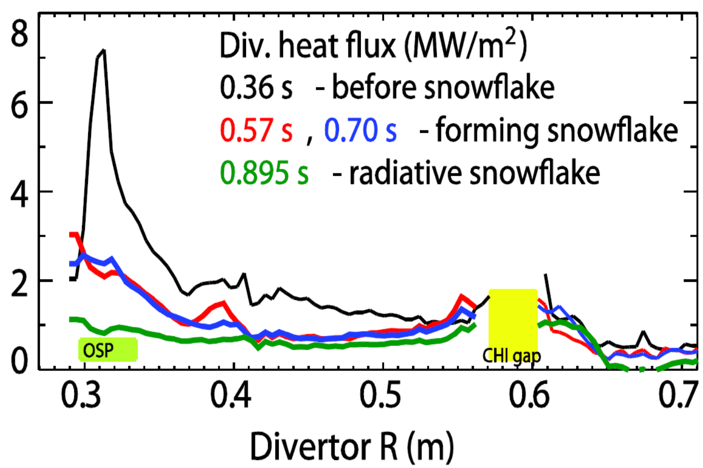
Snowflake divertor configurations obtained in NSTX show very high f_{exp} and $l_{||}$

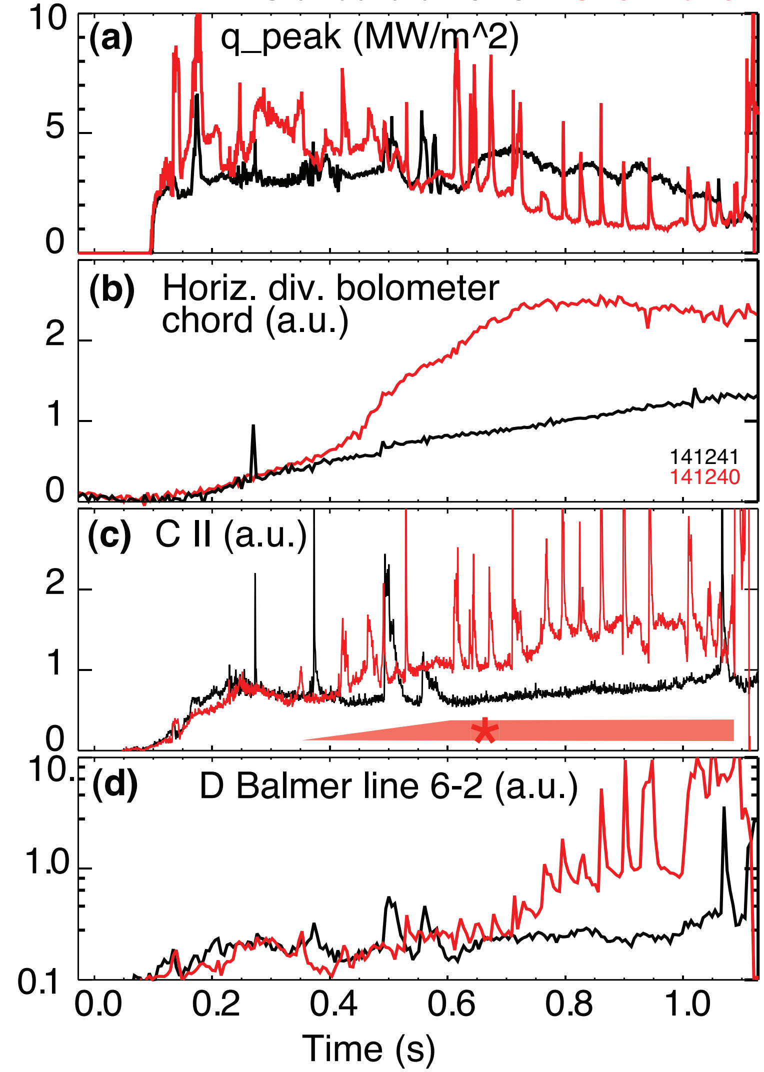


Natural partial detachment in NSTX snowflake otherwise inaccessible with standard divertor

Standard **Snowflake**

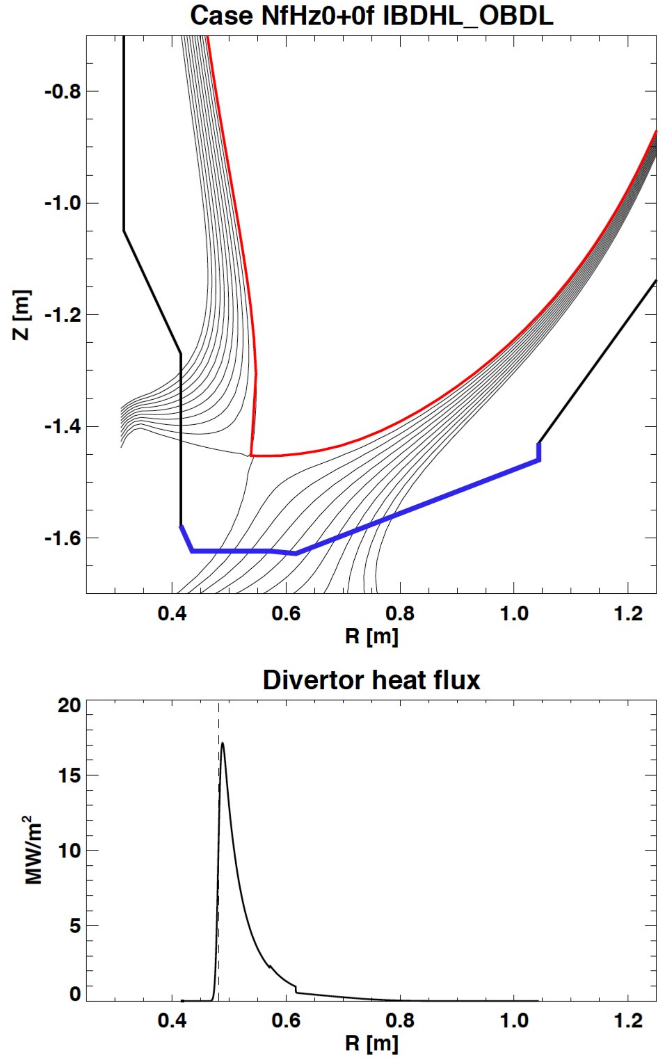






Open divertor with graphite PFCs will be used in initial years in NSTX-U

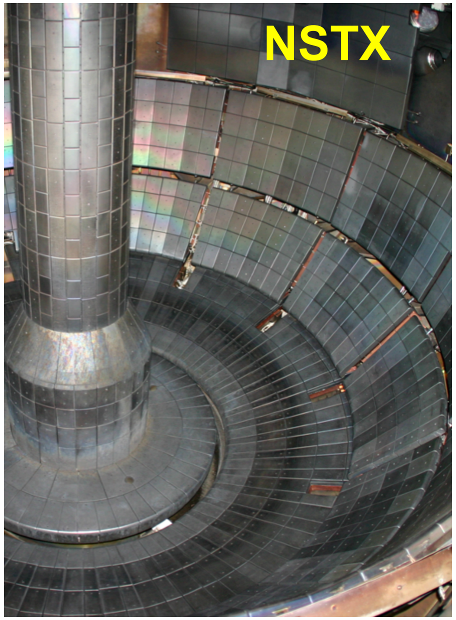
- Open divertor geometry
  - Low neutral and impurity compression
- No active divertor pumping
  - Lithium coatings for reduced recycling
  - Planned cryopump in future years
- Graphite plasma facing components
  - Physical and chemical erosion, dust
  - Max P_{rad} fraction limited by carbon radiation efficiency
- Unbearable divertor heat fluxes in 2 MA, 10 MW, 5 s H-mode discharges are predicted – see J. E. Menard, Poster PP11 43



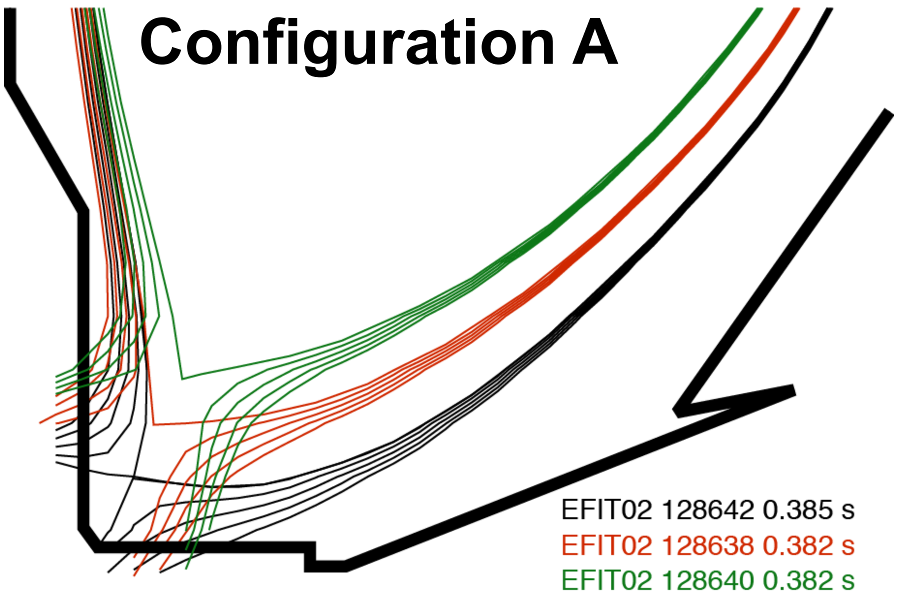
NSTX-U divertor will present additional challenges

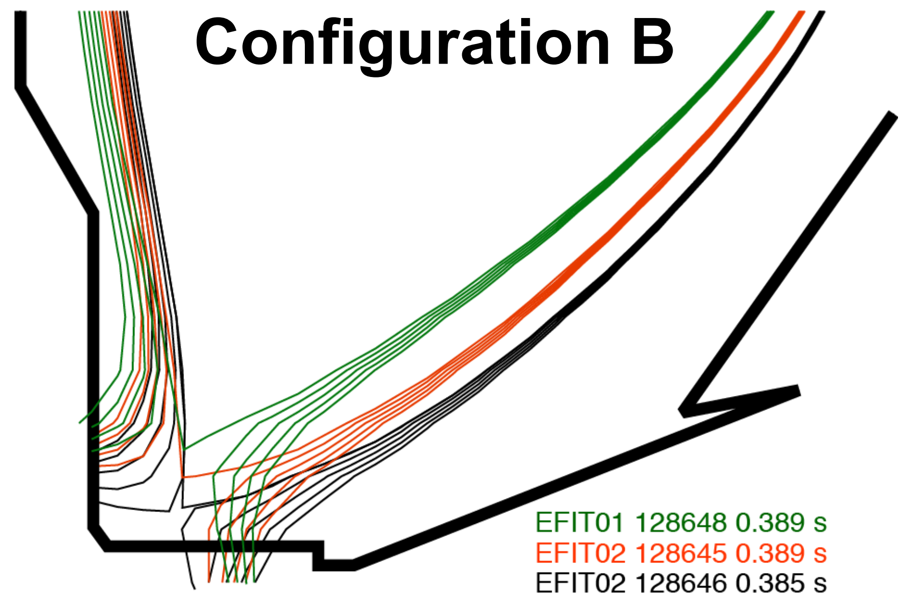
- Main divertor differences between NSTX and NSTX-U for particles
  - Geometry changed
    - Vacuum vessel volume decreased
    - Inner wall (CS) area increased
    - Pumping duct volume decreased
    - Divertor area decreased
  - Divertor and main wall power density to increase
  - Pulse length to increase
  - SOL power and particle channel width (decreased?)
  - Two NBI boxes and two NBI cryopumps

- These changes likely to lead to modified operating space w.r.t. recycling, impurity flux, pedestal stability and ELMs, etc



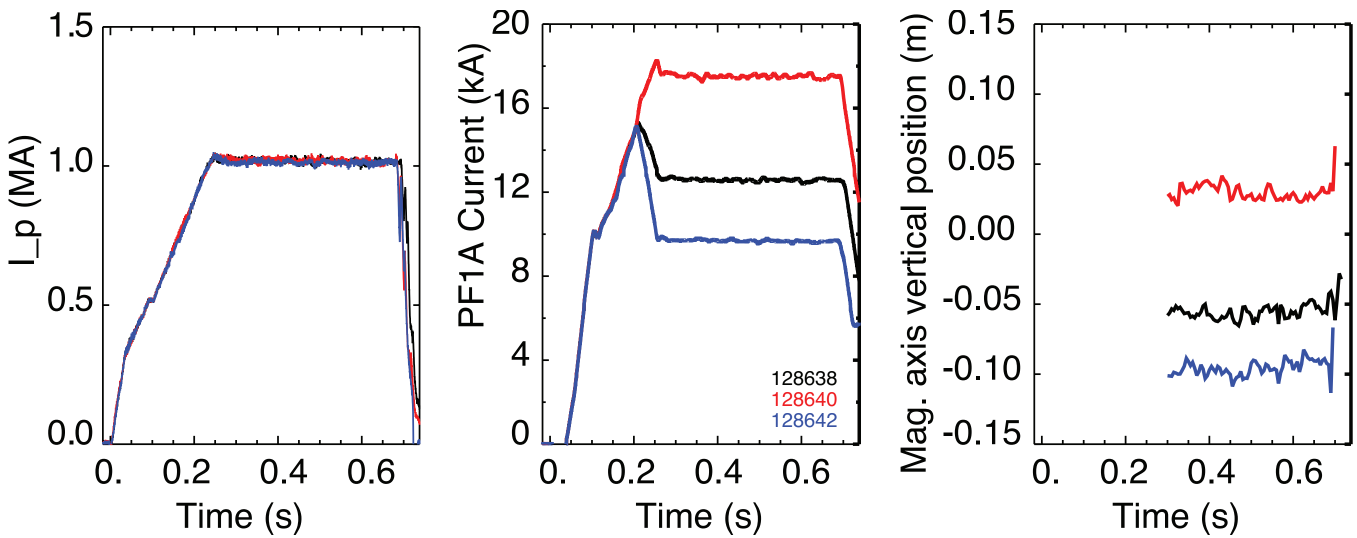
Two divertor configurations with different magnetic geometries were obtained in NSTX experiment





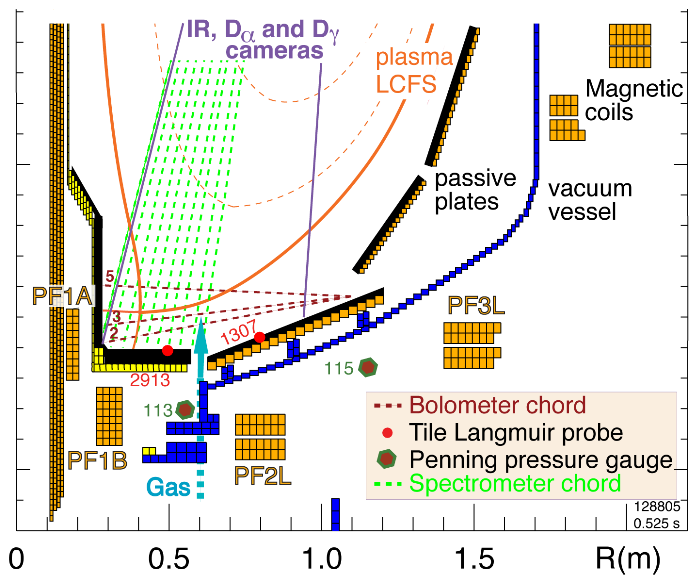
Configuration	Coils	X-pt height	OSP major radius
A ■	PF1AL	6-23 cm	0.30-0.39 m
B ●	PF1AL & PF1B	4-11 cm	0.42-0.45 m

Steady-state H-mode discharges with various X-point heights were used for conf. A and B

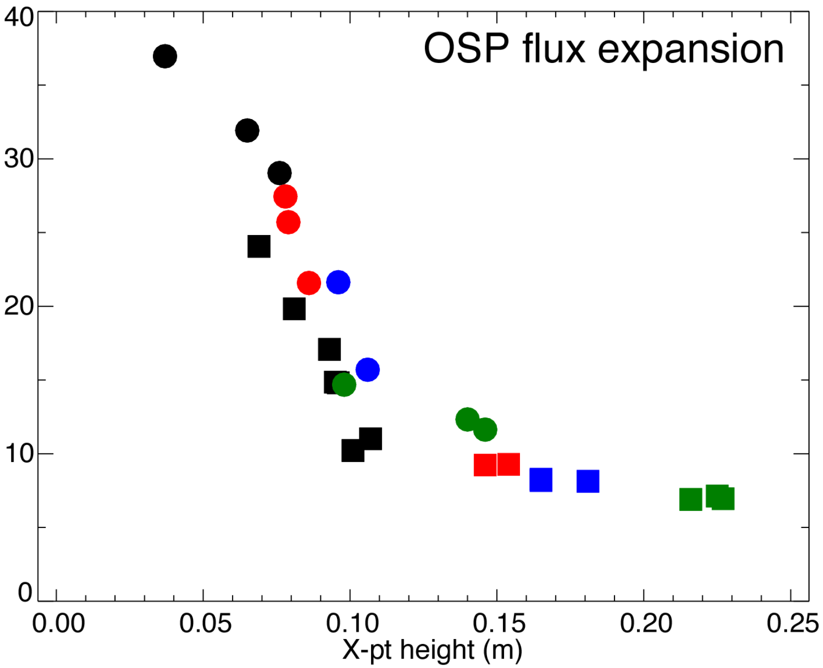


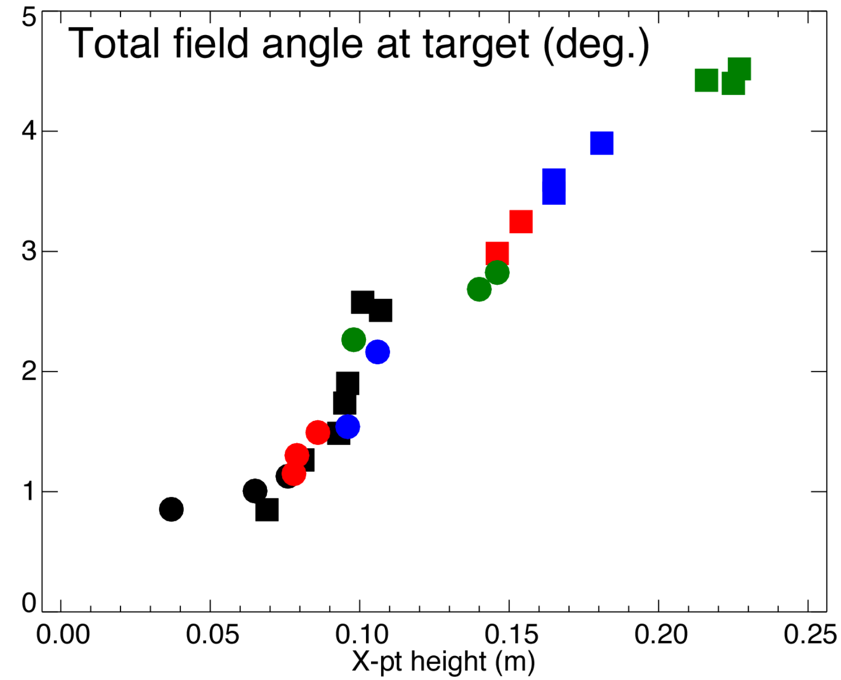
Color coded X-point height:
Low (6-7 cm)
Medium (9-12 cm)
High (20-22 cm)

- 1 MA, 6 MW NBI-heated H-mode
- Small ELM regime
- Similar plasma shape parameters
  - Elongation 2.20-2.40, triangularity 0.7-0.85, drsep ~ 5 – 10 mm
- X-point height was changed by shifting the plasma up and down by several cm using PF1A divertor coil



Flux expansion and total magnetic field angle scale with X-point height similarly in conf. A and B

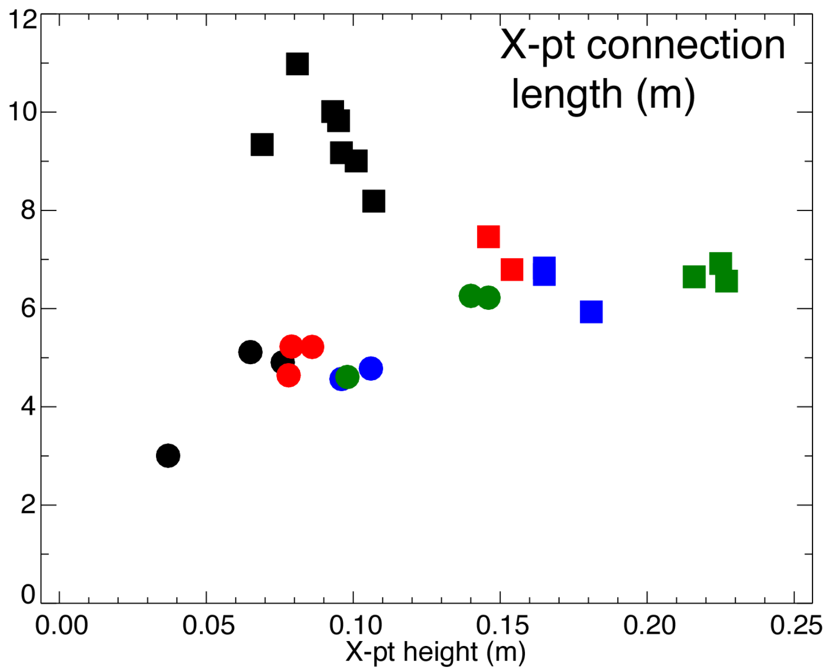


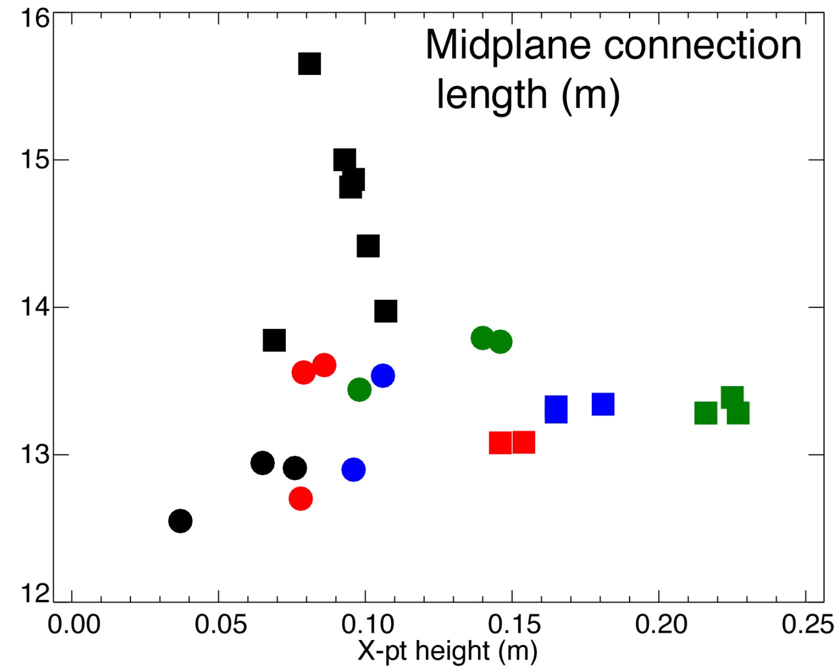


$$f_{exp} = \frac{(B_p/B_{tot})_{MP}}{(B_p/B_{tot})_{OSP}}$$

Configuration	
A	■
B	●

Midplane and X-point connection lengths different in configurations A and B



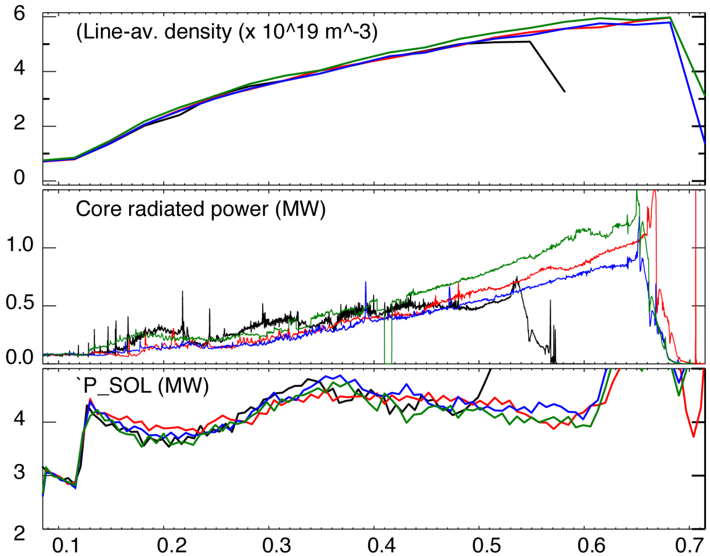


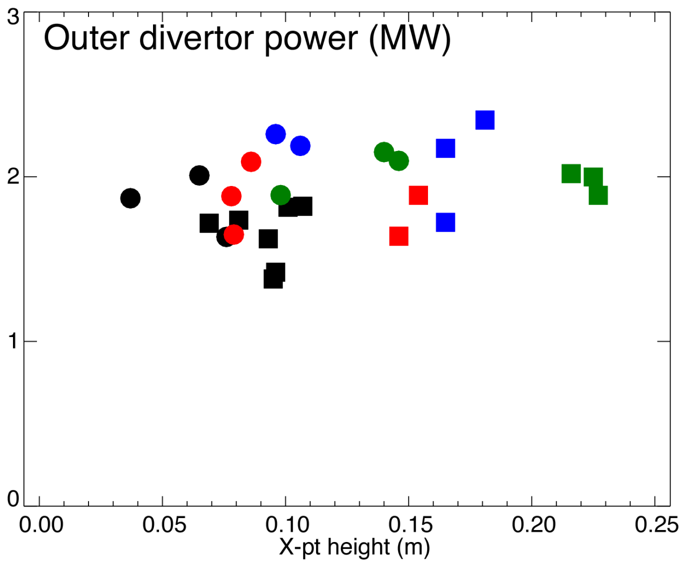
- In config. B, $l_{||}$ is proportional to X-point height
- In config. A, $l_{||}$ is inversely proportional to X-point (unintuitively)

$$l_{\|} = \int_{\mathrm{SP}}^{X\,(\mathrm{MP})} \frac{\sqrt{B_t^2 + B_p^2}}{B_p} \mathrm{d}l.$$

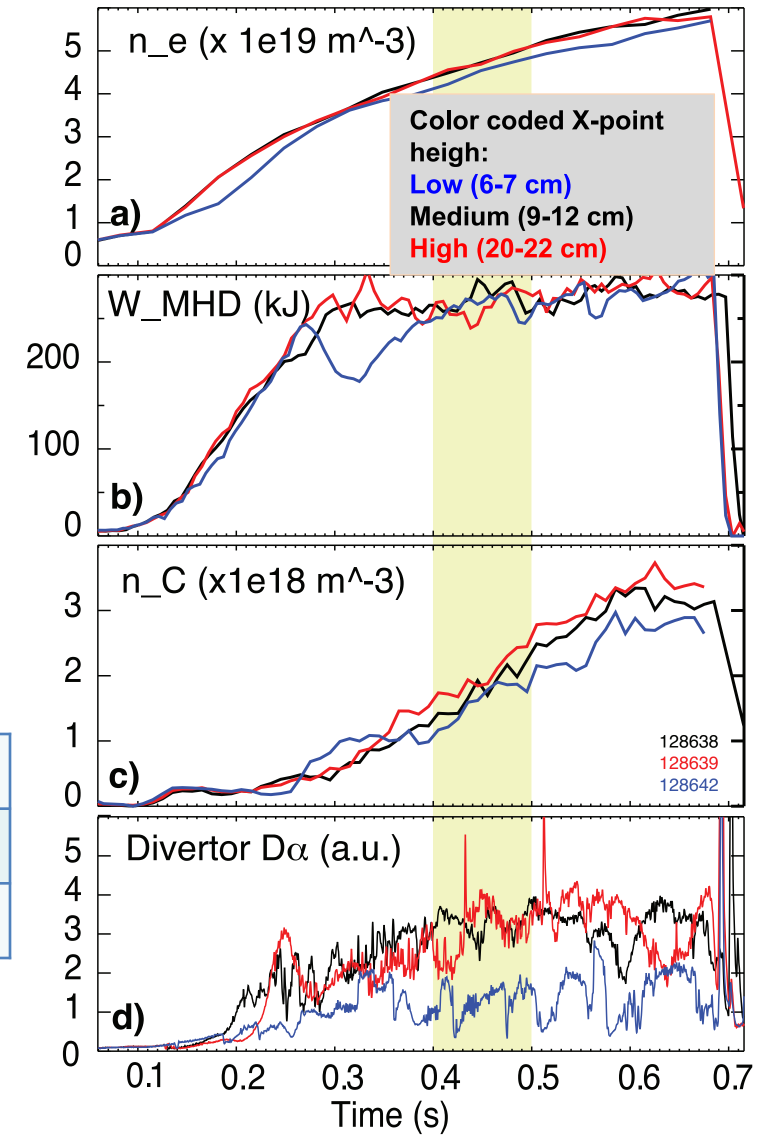
Configuration	
A	■
B	●

Steady-state H-mode discharges with various X-point heights were used for conf. A and B

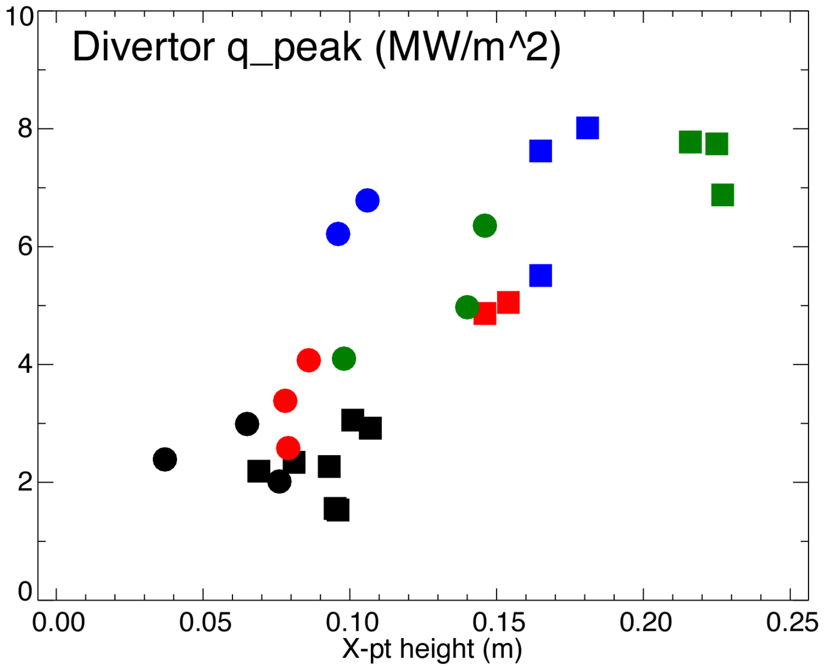


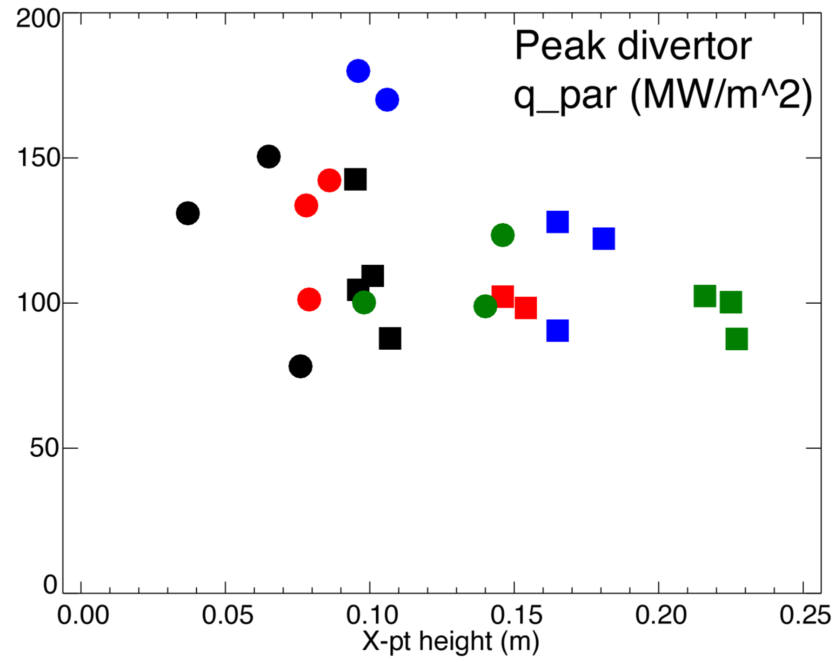


Configuration	
A	■
B	●



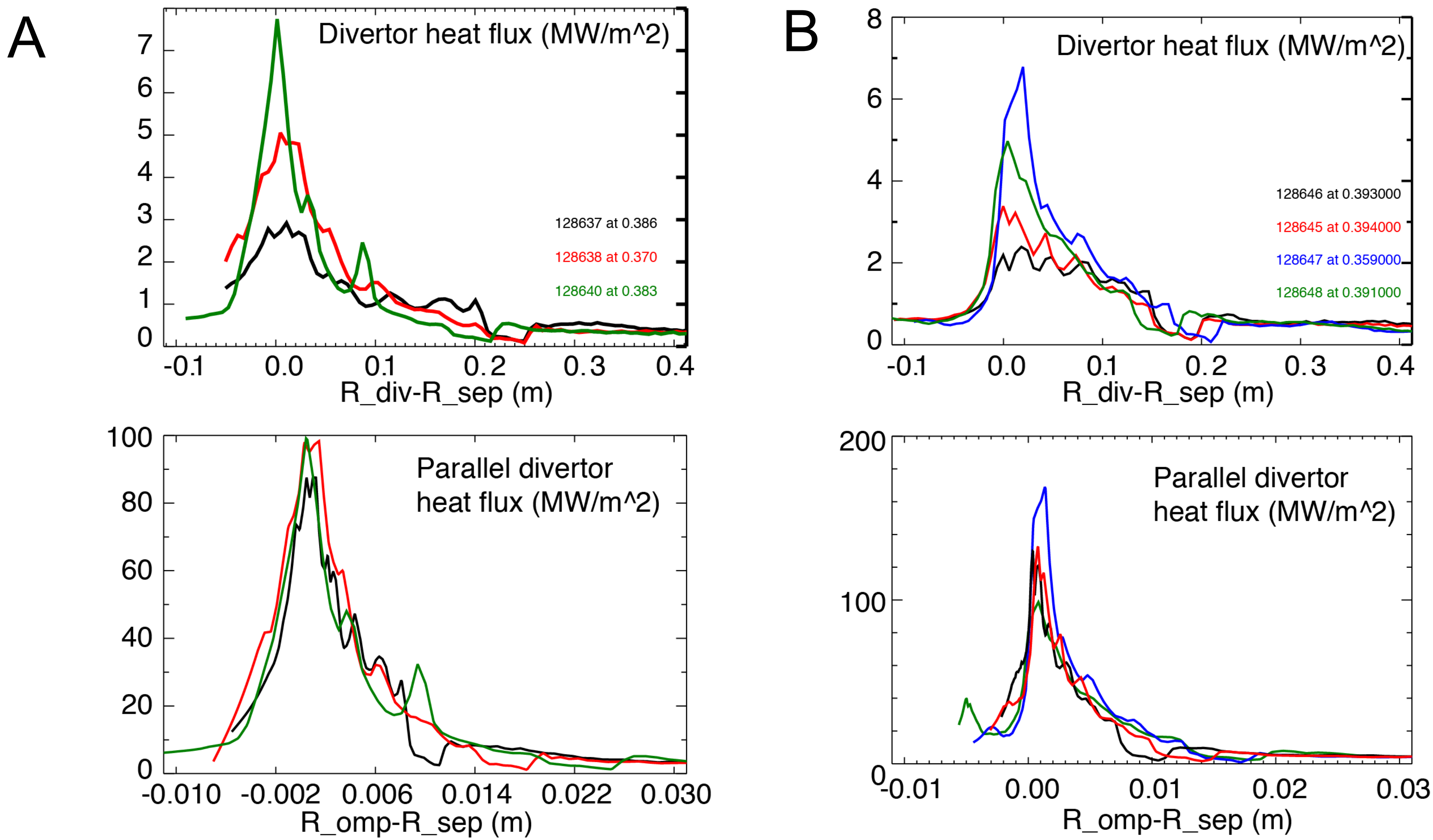
Parallel heat fluxes at OSP are higher in configuration B



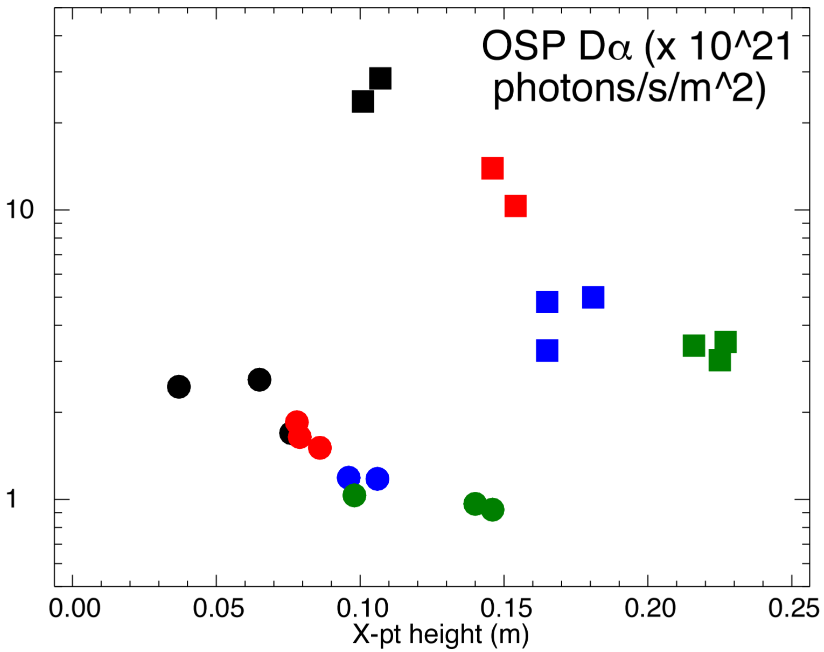


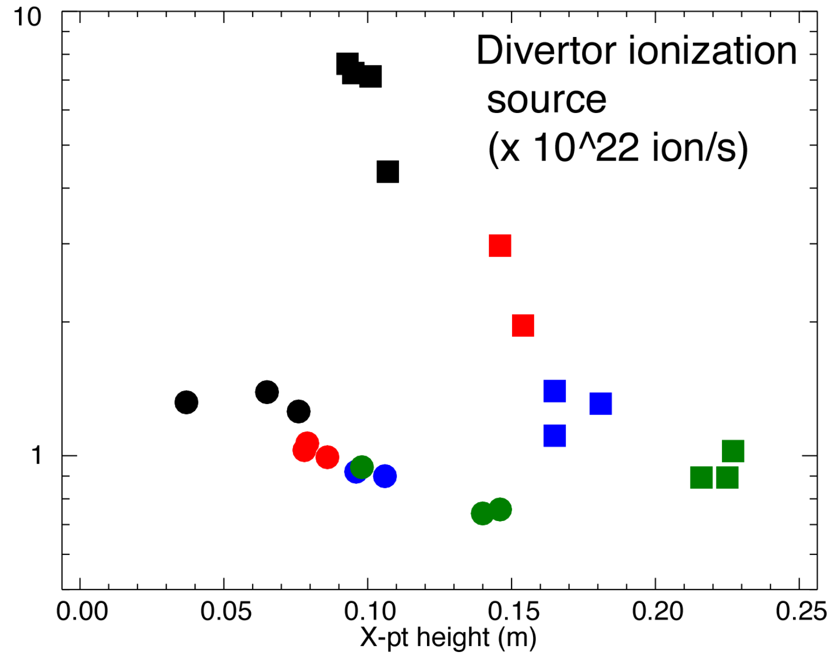
Configuration	
A	■
B	●

Parallel heat fluxes higher in configuration B



Divertor D-alpha and ionization source higher in configuration A

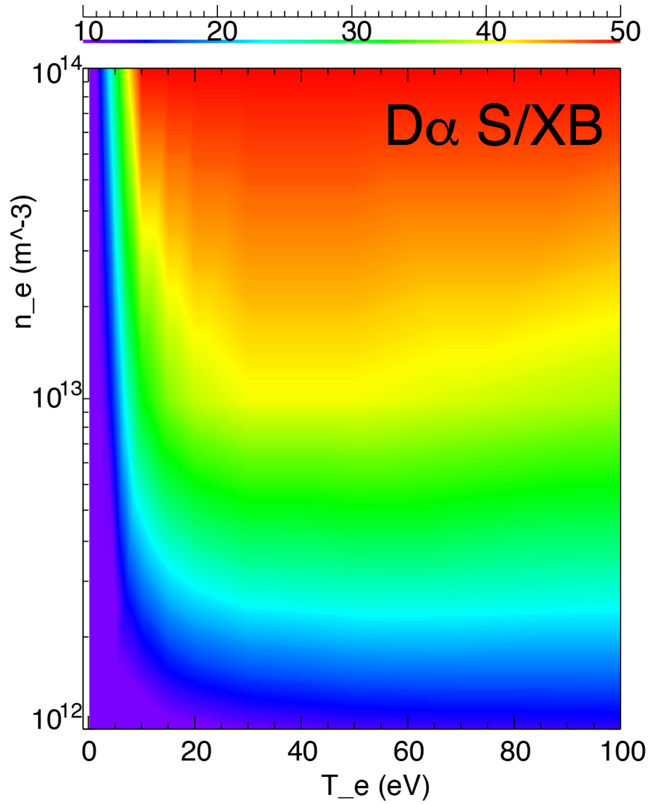


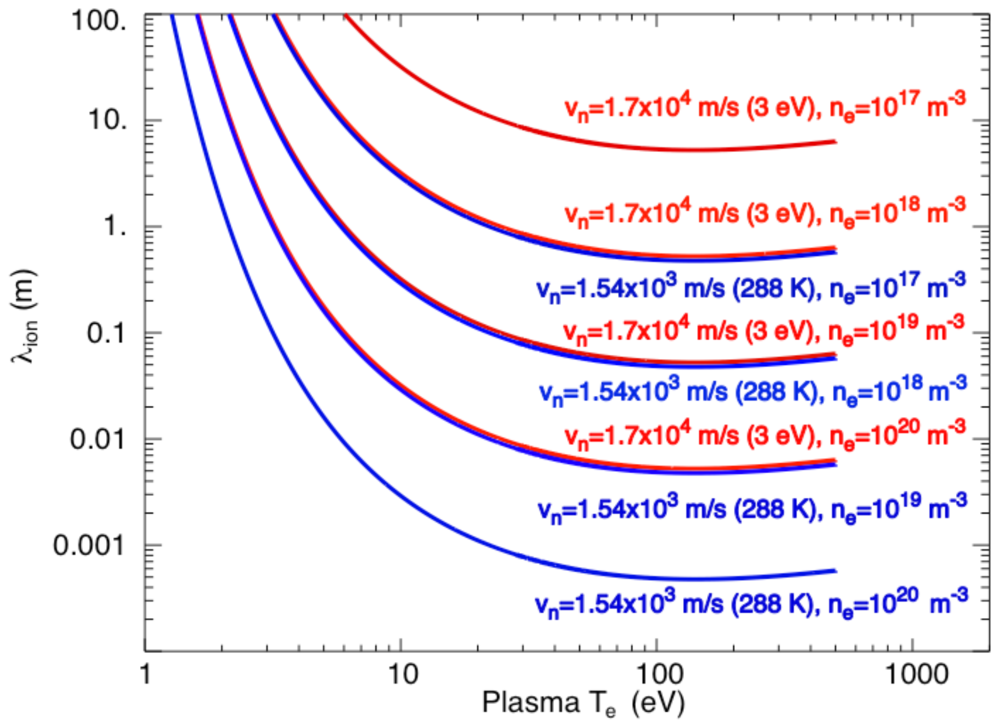


- Divertor Dα profiles measured by photometrically calibrated camera
- Ionization source inferred using S/XB=30 from ADAS by spatially integrating divertor Dα brightness profiles over one λ_q

Configuration	
A	■
B	●

Neutral ionization length lower than divertor leg size (X-point height) in configurations A and B





- ADAS S/XB for D n=3-2 transition (Dα)

Shown D atomic ionization length only - atomic CX length scales similarly

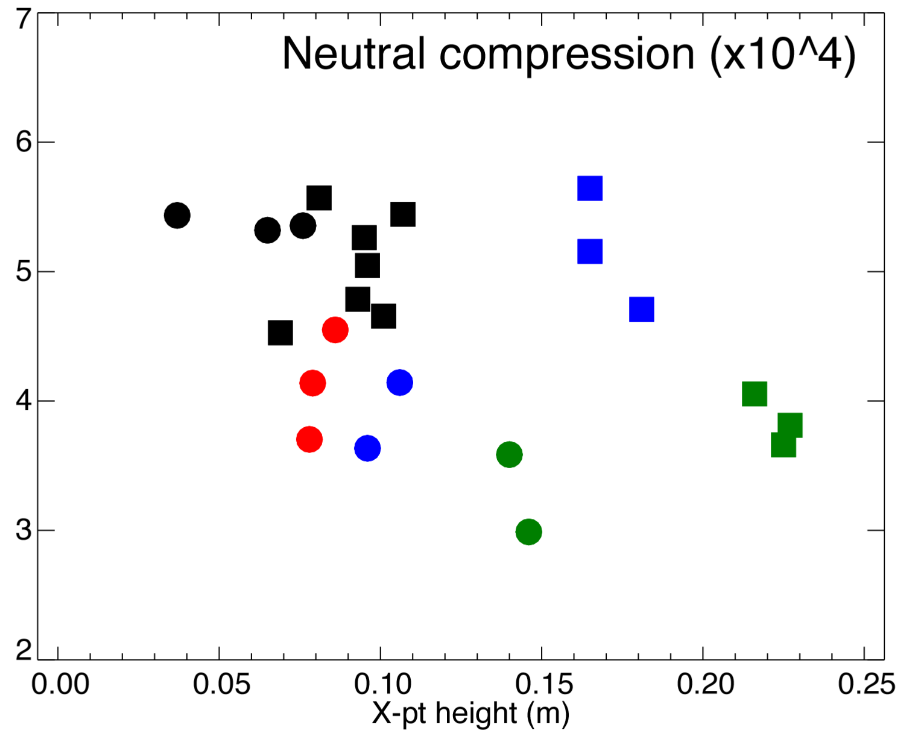
$$\lambda_{ion} = \frac{v_n}{n_e S_{ion}} = \frac{v_n}{n_e \langle \sigma_{ion} v_e \rangle}$$

Neutral compression higher in conf. A

- Neutral compression inferred from midplane microion and divertor Penning pressure gauge measurements

- P_div/P_mid

Configuration	
A	■
B	●

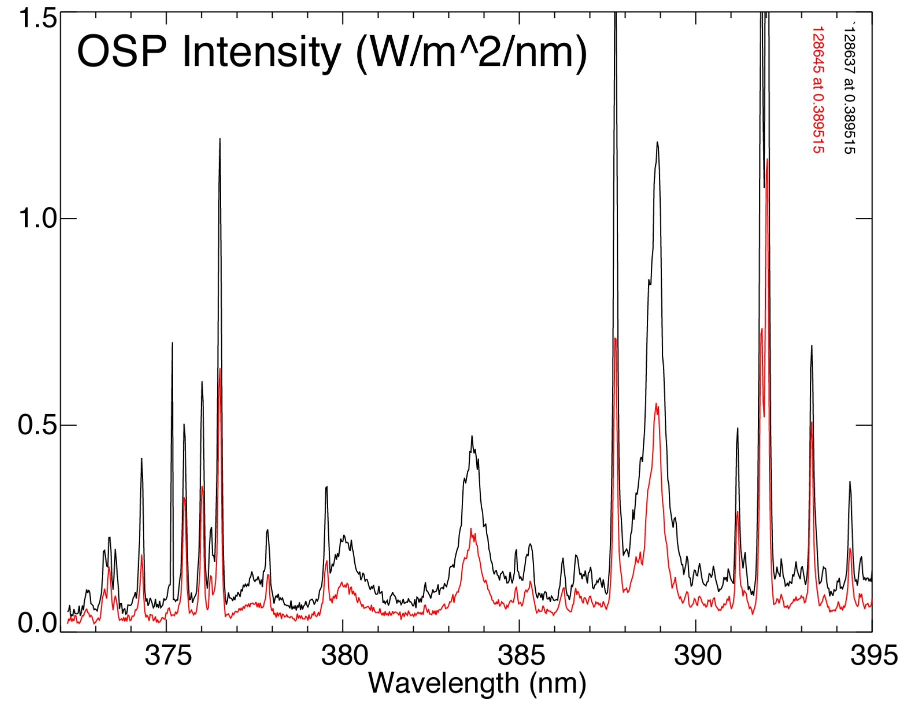


- Neutral pressure of 2-3 mTorr is needed for detachment to explain plasma pressure drop of dp/dx = 9-10 Pa/m

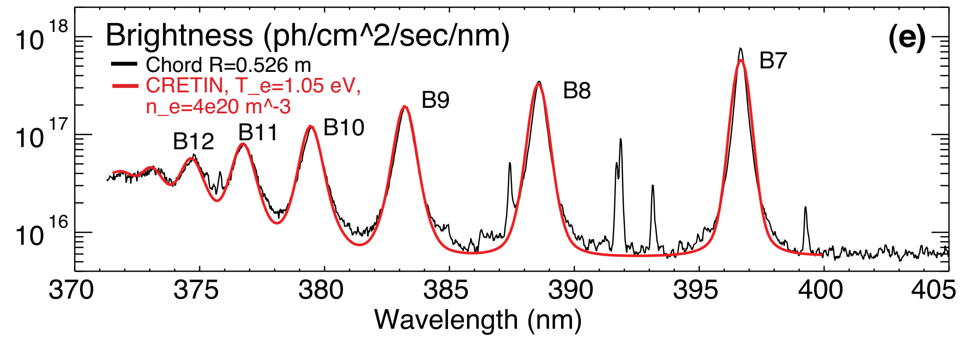
$$\frac{dp}{dx} = m\Gamma_i n_n \langle \sigma v \rangle_{cx+el} + m n^2 \langle \sigma v \rangle_{rec}$$

Recombination rate is inferred from high-n Balmer line divertor intensities

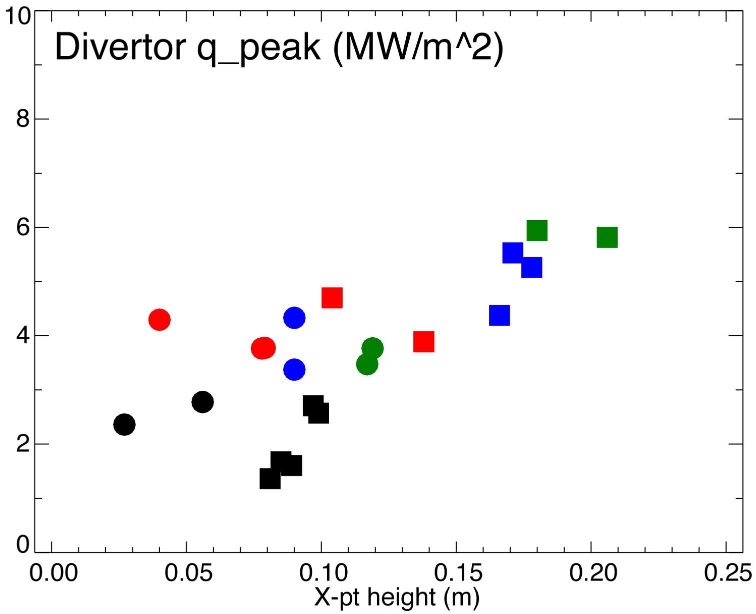
- Intensities of B9-B11 are low
- At T_e > 10 eV, likely to be due to excitations, not recombinations
- Electron-ion recombination rate depends on divertor ion residence time
  - Ion recombination time: τ_{ion}~ 1−10 ms at T_e =1.3 eV
  - Ion residence time: $\tau_{ion} \leq$ 1 ms

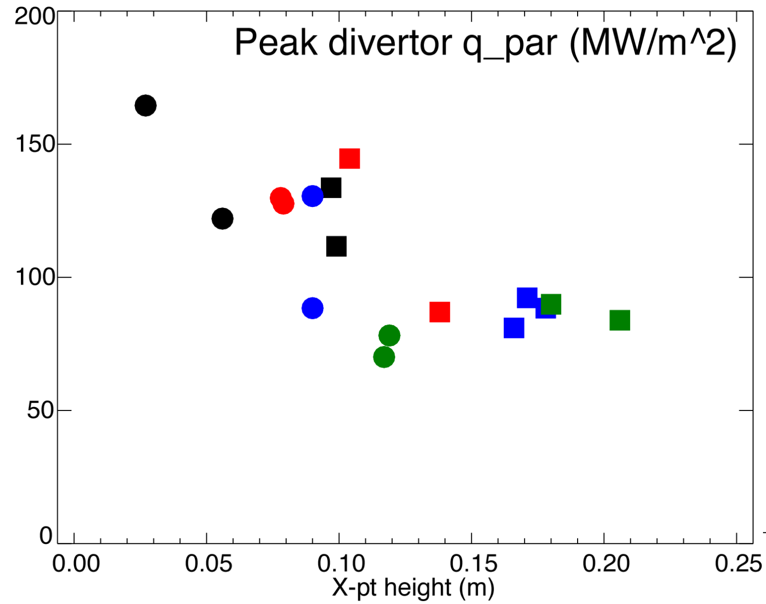


- During detachment, intensities are x100 higher
- High-*n* upper levels are populated by 3-body recombination



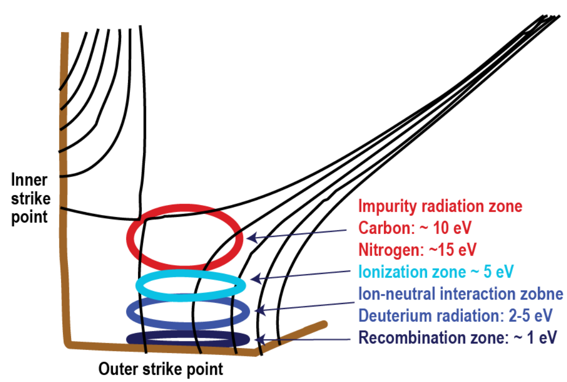
At higher upstream density, q_{peak} and $q_{||}$ are further decreased

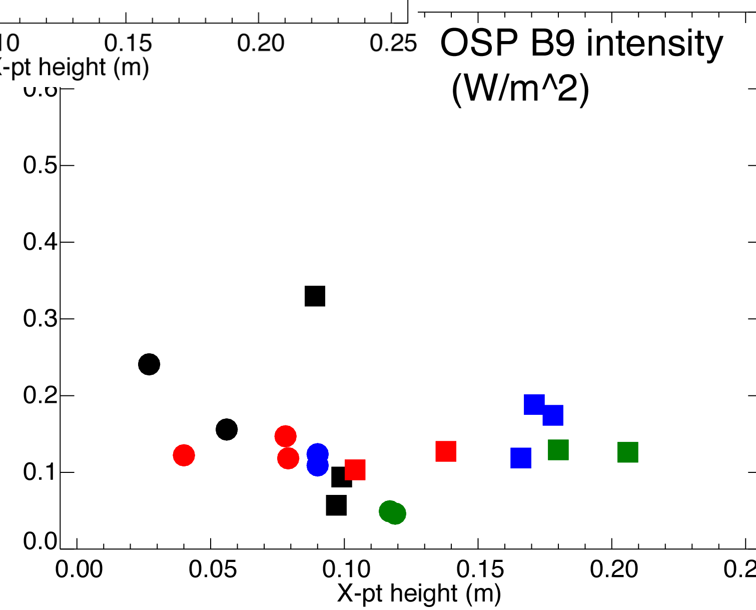




Configuration	
A	■
B	●

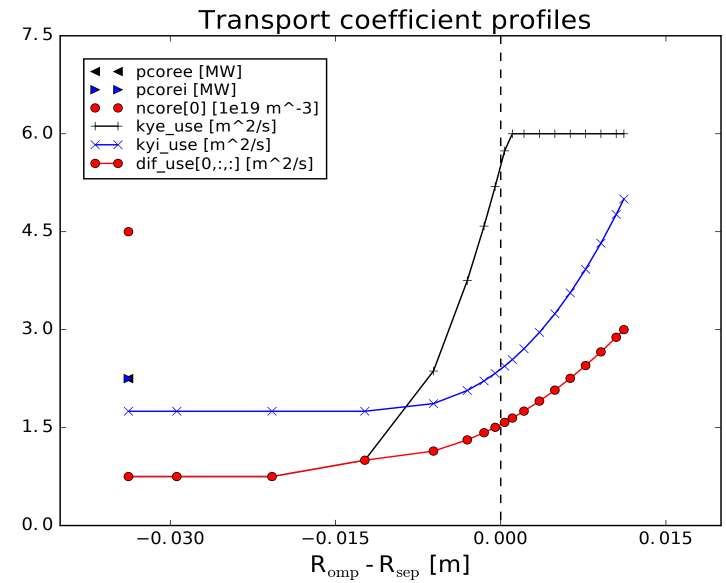
- However, no OSP detachment evident

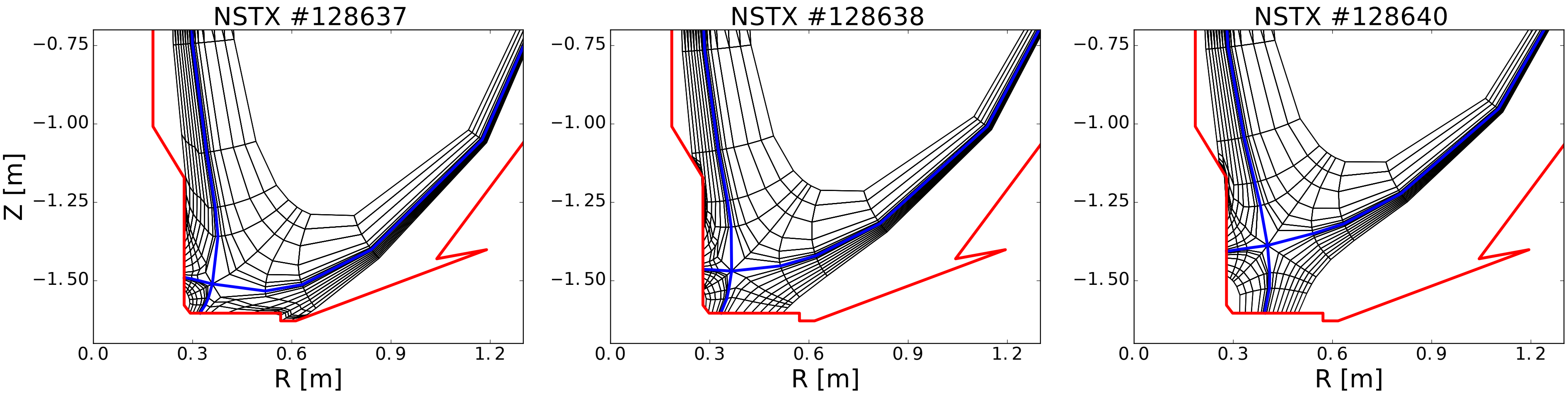




UEDGE is used to model divertor plasma conditions in conf. A

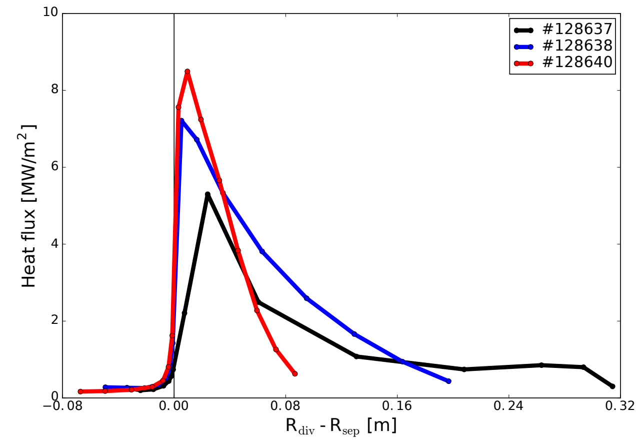
- Transport coefficients D (particles) and $\chi_{e,i}$ (energy) are adjusted to produce experiment-like upstream profiles
- Divertor plate recycling R=0.99
- Charge-state resolved impurity model with constant D, physical and chemical sputtering

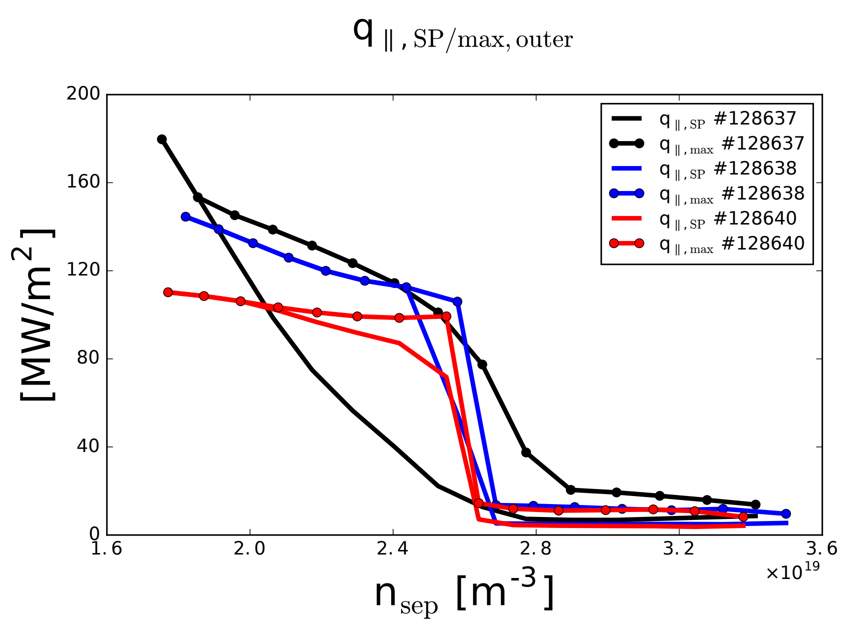


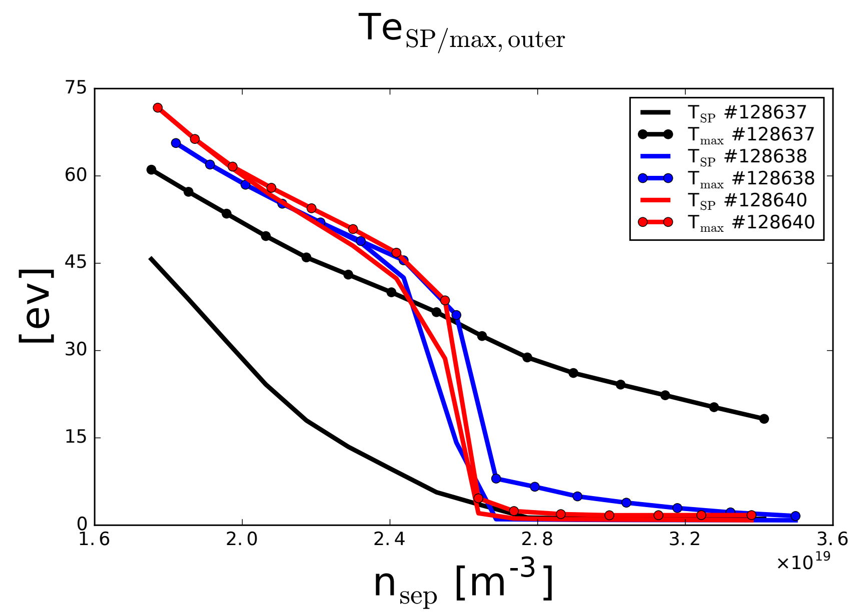


Lowest X-point height configuration A reaches detachment conditions faster

- Medium and high X-point height configurations reach detachment conditions abruptly (”cliff-like”)







NSTX SOL power width λ

- From Makowski, PoP 2015