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Plasma-Material Interface Development for Future Spherical Tokamak based Devices in NSTX*

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Abstract

The divertor plasma-material interface (PMI) must be able to withstand steady-state heat fluxes up to 10 MW/m^2 (a limit imposed by the present day divertor material and engineering constraints) with minimal material erosion, as well as to provide impurity control and ion density pumping capabilities. In spherical tokamaks (STs), the compact divertor geometry and the requirement of low core electron collisionality n_e^* at n_e < 0.5-0.7 n_G (where n_G is the Greenwald density) for increased neutral beam current drive efficiency impose much greater demands on divertor and first-wall particle and heat flux mitigation solutions. In NSTX, divertor heat flux mitigation and impurity control with an innovative “snowflake” divertor configuration [1] and ion density pumping by evaporated lithium wall and divertor coatings [2] are studied. Lithium coatings have enabled ion density reduction up to 50 % in NSTX through the reduction of wall and divertor recycling rates. The “snowflake” divertor configuration was obtained in NSTX in 0.8-1 MA 4-6 MW NBI-heated H-mode lithium-assisted discharges using three divertor coils. The snowflake divertor formation was always accompanied by a partial detachment of the outer strike point with an up to 50 % increase in divertor radiation from intrinsic carbon, the peak divertor heat flux reduction from 3-6 MW/m^2 to 0.5-1 MW/m^2, and a significant increase in divertor volume recombination [3]. High core confinement was maintained with the snowflake divertor, evidenced by the t_E, W_{MHD} and the H98(y,2) factors similar to those of the standard divertor discharges. Core carbon concentration and radiated power were reduced by 30-70 %, apparently as a result of reduced divertor physical and chemical sputtering in the snowflake divertor and ELMs. In the SFD discharges, the MHD stability of the H-mode pedestal region was altered leading to the re-appearance of medium size (DW/W=5-10 %), Type I, ELMs otherwise suppressed due to lithium conditioning [4]. Fast divertor measurements showed that impulsive particle and heat fluxes due to the ELMs were significantly dissipated in the high magnetic flux expansion region of the snowflake divertor. The snowflake divertor configuration is being combined in experiments with extrinsic deuterium or impurity gas puffing for increased dissipative divertor power losses, additional upper divertor nulls for increased power sharing between the upper and the lower divertors, and lithium coated plasma facing components for large area ion pumping. These efforts are aimed at the development of an integrated PMI for future ST-based devices for fusion development applications. Supported by the U.S. DOE under Contract DE-AC52-07NA27344, DE-AC02-09CH11466, DE-AC05-00OR22725, DE-FG02-08ER54989.

References

[1] D. D. Ryutov, Phys. Plasmas 14, 064502 (2007)

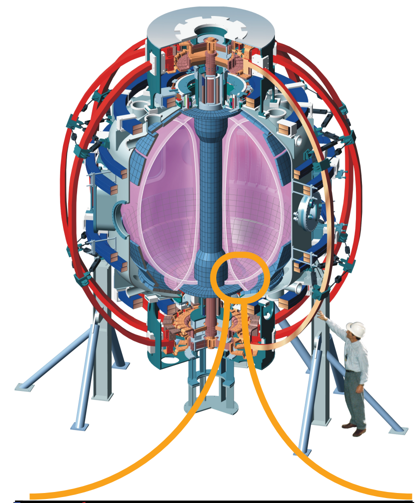
[2] H. W. Kugel et. al, Phys. Plasma **15** (2008) 056118

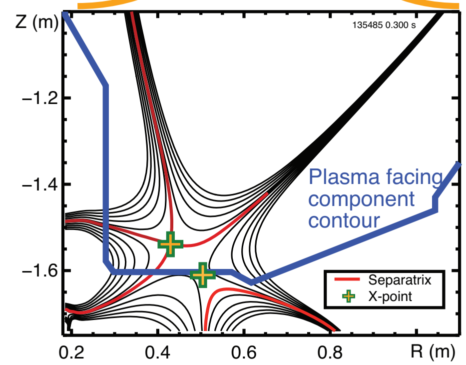
[3] V.A. Soukhanovskii et. al, Nucl. Fusion 51 (2011) 012001

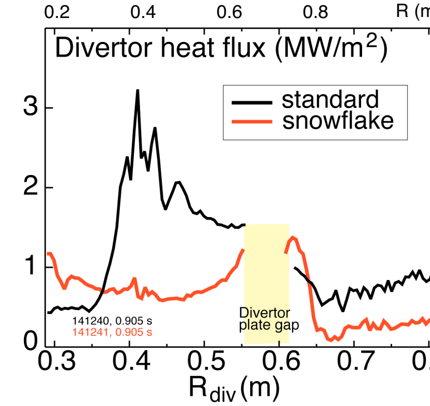
[4] R. Maingi et. al, Phys. Rev. Lett. 103 (2009) 075001

Summary: High flux expansion (snowflake) area-pumping (lithium) divertor is studied in NSTX

- **Evaporative lithium coatings on carbon PFCs**
  - Surface pumping reduced ion inventory (density) by up to 50 %
  - Recycling reduced by up to 50 % in both divertors and wall
  - Local recycling coefficients reduced on inner wall and far SOL, remained similar in the outer strike point region
  - Parallel heat transport regime in the SOL changes from conduction-limited to sheath-limited (low-recycling)
  - Core lithium density low, uncorrelated with divertor source
- **"Snowflake" divertor configuration**
  - Obtained with three existing divertor coils for up to 600 ms
  - H-mode confinement maintained with significant reduction in core impurities
  - Significant reduction in peak heat flux (and outer strike point partial detachment)
  - Higher divertor plasma-wetted area A_{wet} and divertor volume
  - Consistent with theoretical predictions and 2D fluid model
  - Candidate divertor solution for NSTX-Upgrade

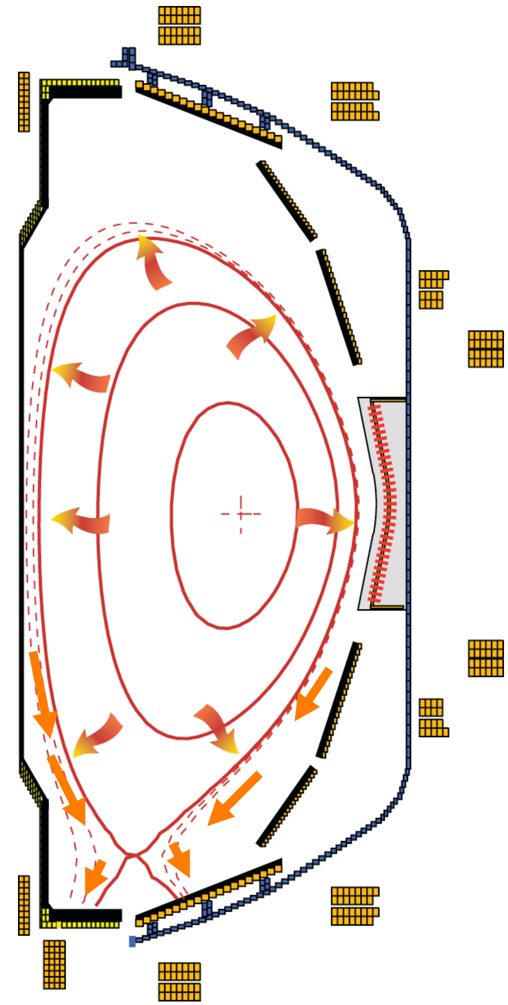






Solid lithium coatings are studied in NSTX for impurity and density control applications

- Access to reduced core collisionality
  - n_e ~ 0.6-0.9 n_G for transport, stability, start-up, high non-inductive current fraction scenario studies for future STs (e.g., NSTX-Upgrade)
  - n_e ~ 0.3-0.7 n_G for adequate NBI current drive efficiency in scenarios relevant to fusion and nuclear science ST-based devices
- Spherical tokamak: compact divertor for power and particle exhaust
- NSTX (Aspect ratio A=1.4-1.5)
  - $I_p \leq 1.4$ MA, $P_{in} \leq 7.4$ MW (NBI), P / R ~ 10
  - $q_{peak} \leq 15$ MW/m^2, $q_{||} \leq 200$ MW/m^2
  - ATJ and CFC graphite tiles as PFCs
  - Typical divertor strike point region $T \leq 500$ C ($q_{peak} \leq 10$ MW/m^2) in 1 s discharges



National Spherical Torus Experiment

Various techniques considered for SOL / divertor $q_{||}$ and q_{pk} control

- **Divertor heat flux mitigation solutions:**

$$q_{pk} \simeq \frac{P_{heat}\,(1 - f_{rad}) f_{out/tot} f_{down/tot} (1 - f_{pfr}) \sin\alpha}{2\pi R_{SP} f_{exp} \lambda_{q_{||}}}$$

  - ✓ Divertor geometry (poloidal flux expansion)
  - ✓ Strike point sweeping
  - ✓ Radiative divertor (or radiative mantle)
  - ✓ Divertor plate tilt and divertor magnetic balance

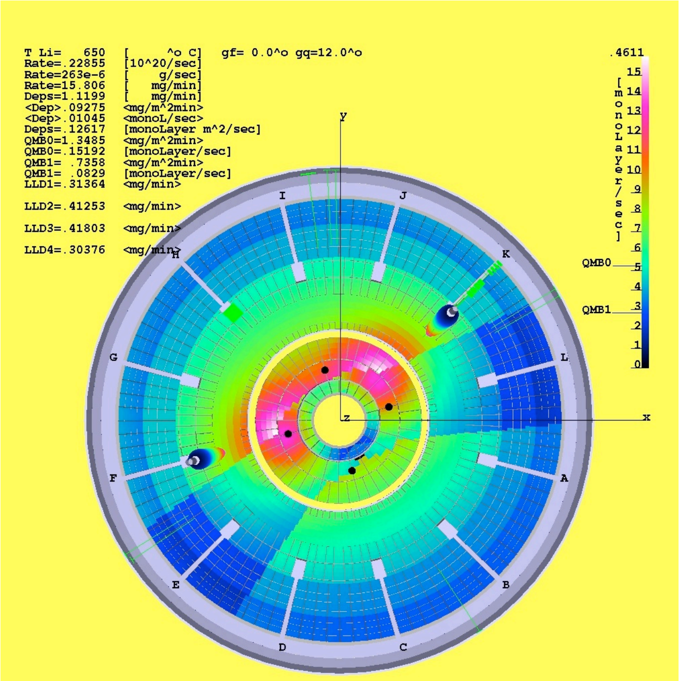
$$f_{exp} = \frac{(B_p/B_{tot})_{MP}}{(B_p/B_{tot})_{OSP}}$$

$$A_{wet} = 2\pi R\, f_{exp}\, \lambda_{q_{||}}$$

- **Candidate solutions must**
  - **be compatible with good core plasma performance (H-mode confinement, MHD, ELM regime, density)**
  - **be compatible with particle control (e.g., cryopump, lithium)**
  - **scale to very high q_{peak} (15 - 80 MW/m^2) for future devices**

Plasma-surface interactions with solid lithium coatings on graphite plasma-facing components

- Solid lithium coatings in NSTX
  - deposited by two lithium ovens (LITERs)
    - oven T= 600-680°C
    - Evaporation rate: 1 mg/min – 80 mg/min
  - divertor coating thickness up to 200-400 nm
  - up to 50 % variation in toroidal thickness

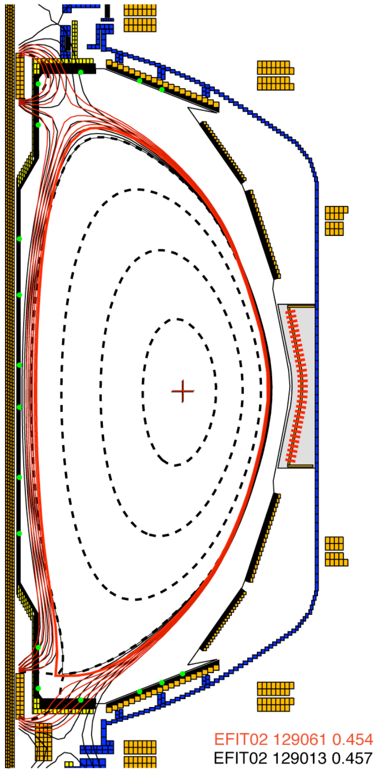


- Interaction of solid lithium coatings with plasma
  - Physical sputtering of lithium atoms
    - by D ions - 2/3 lithium sputtered as Li^+
    - by lithium (self-sputtering) and carbon
  - Re-deposition
  - Melting (T = 180^o C) and evaporation (significant rate at T > 300^o C)
  - Reaction with D^0 atoms leads to pumping of hydrogenic plasma
    - Coating can bind D with all Li inventory up to a full thickness
    - After saturation - high recycling, low pumping rate
  - Reaction with H_2O, C and O to form various compounds

Impact of lithium conditioning was investigated in NBI-heated H-mode discharges

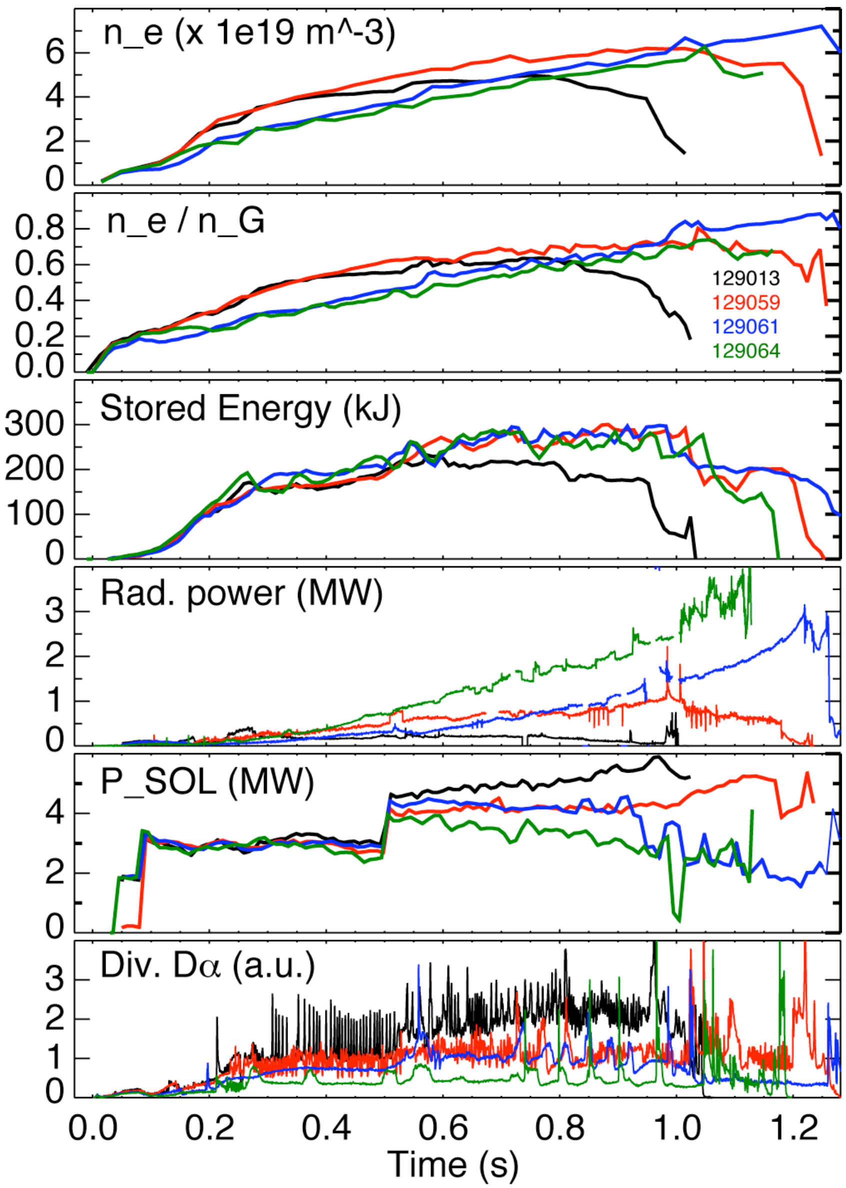
- Pumping and recycling on PFCs
- Lithium influx from PFCs and core lithium density
- Control of divertor carbon influx

- I_p=0.9 MA, B_t=4.5 kG, P_{NBI}=4-6 MW, high κ~2.3, δ~0.6
  - **Discharge without lithium (129013)**
    - boronized carbon, no prior lithium use
  - **Discharge without lithium (129059)**
    - prior use of lithium (~ 20 discharges, ~ 8 g)
  - **Discharge with 190 mg lithium (~ 190 mg total, 129061)**
  - **Discharge with 600 mg lithium (2.2 g total, 129064)**



- Photometrically calibrated filtered cameras and spectrometers, tile-mounted Langmuir probes, neutral pressure gauges
- Γ_{ion} [ion/m^2/s] = 4 π I_λ [ph/m^2/s/sr] S/XB [ion/ph]
  - For deuterium, D_a and D_b ; for lithium, Li I λ=670 nm
  - Outer SOL region only

With lithium, reduced core density operation can be achieved



- $n_e/n_G \sim 0.2\text{-}0.7$
  - N_e and n_e increasing
- W_{MHD} increased
- P_{SOL} decreasing with lithium amount
  - Core P_{rad} increasing
  - $P_{SOL}=P_{OH}+P_{NBI} - P_{rad} - dW/dt - P_{fast\ ion\ loss}$
- ELMs suppressed
  - Pedestal MHD stability modified due to n_e (r) mod.

No lithium (129013)
No lithium (129059)
190 mg lithium (129061)
600 mg lithium (129064)

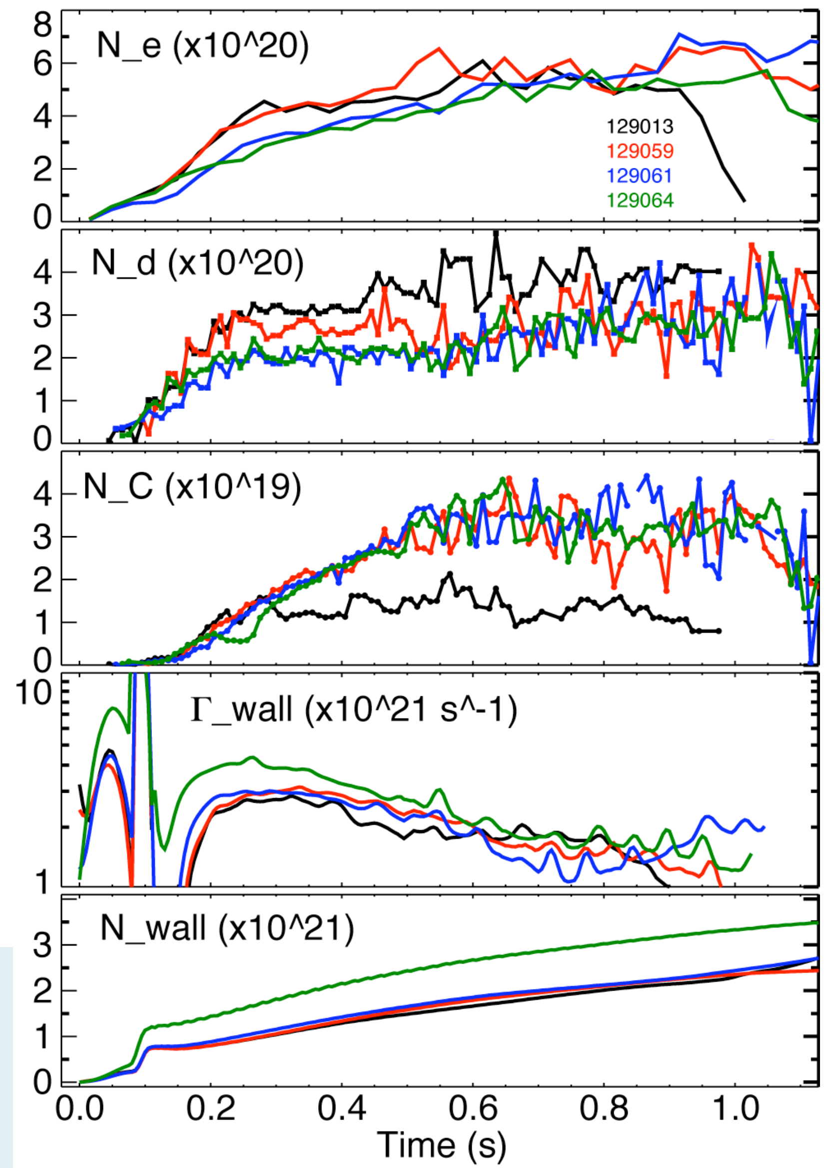
Ion density was reduced by up to 50 % by lithium conditioning in NSTX

$$\frac{dN_p}{dt} = \Gamma_{gas} + \underbrace{\Gamma_{NBI} + \Gamma_{NBI\_cold}} + \Gamma_{NBI\_cryo} + \Gamma_{wall} + \Gamma_{pump} + \frac{dN_n}{dt}$$

Change in ion inventory	Gas feed rate	NBI fueling rate	NBI cryopump rate	Wall loading rate	Turbo. pump rate	Neutrals build-up rate

- Particle balance equation using measured quantities
- Particle inventory balance: $N_e = 6\ N_C + N_d$

- **Continuous pumping**
- **Cumulative coatings provide higher pumping rate**
- **Wall in pumping state far from saturation**



No lithium (129013)
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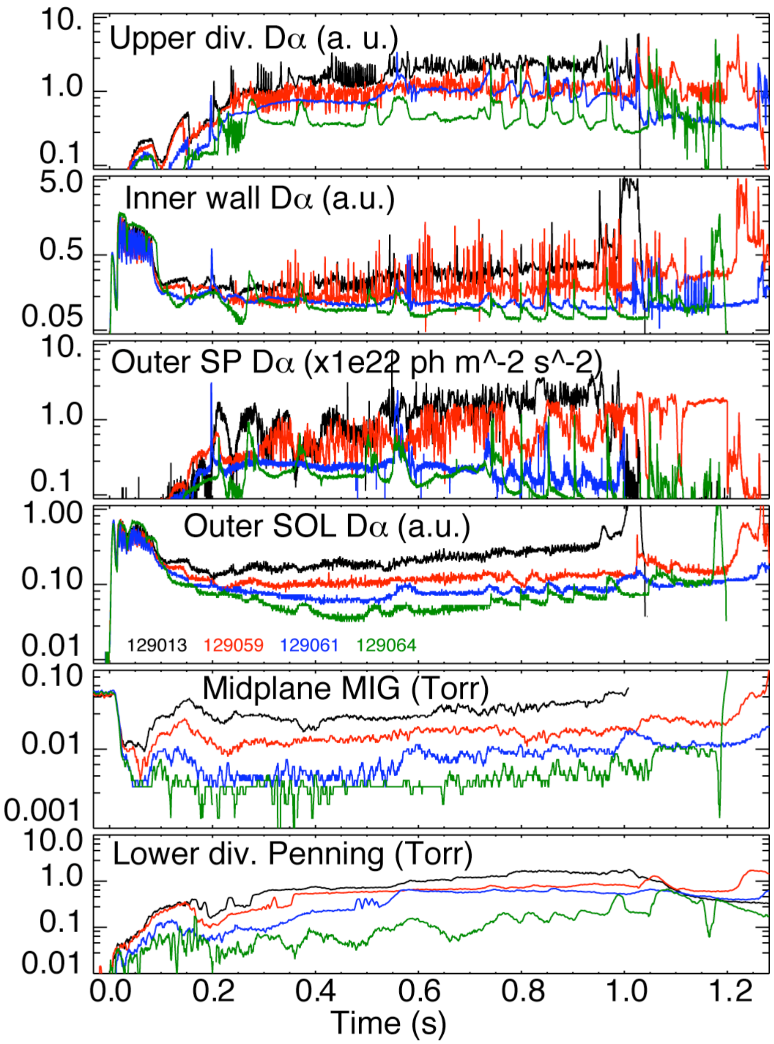
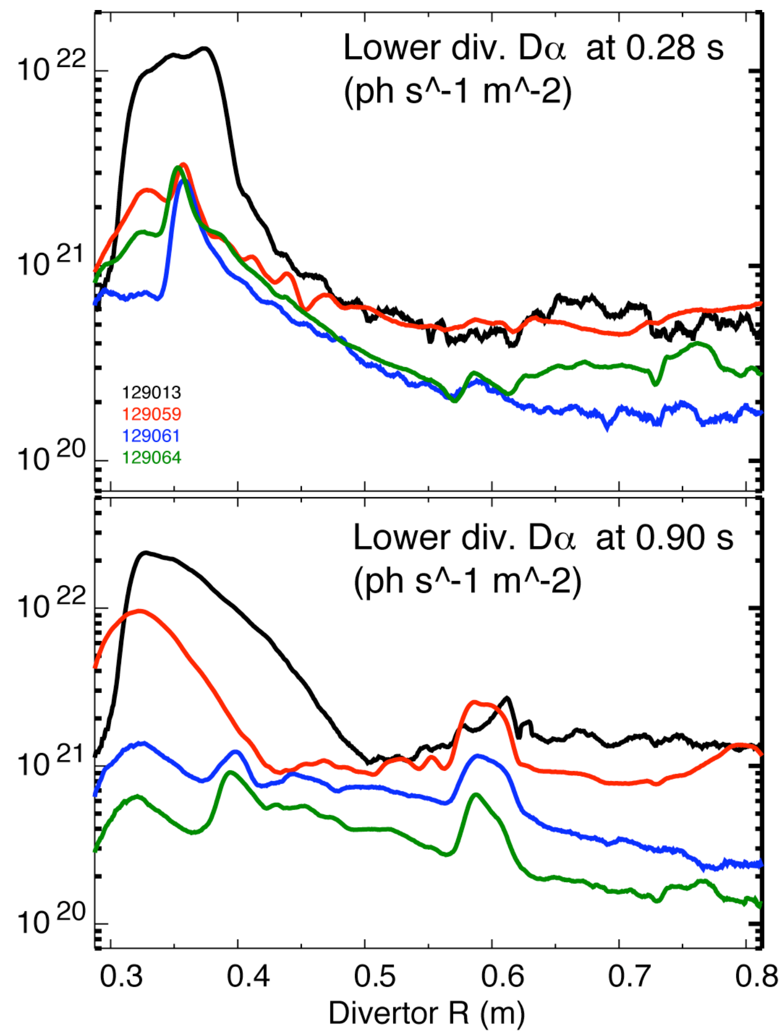
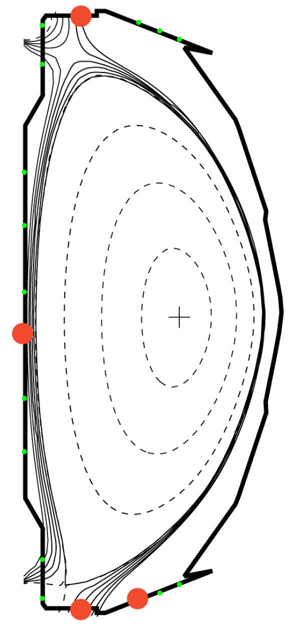
190 mg lithium (129061)
600 mg lithium (129064)

Edge neutral pressure and recycling on PFCs reduced, most strongly in lower divertor



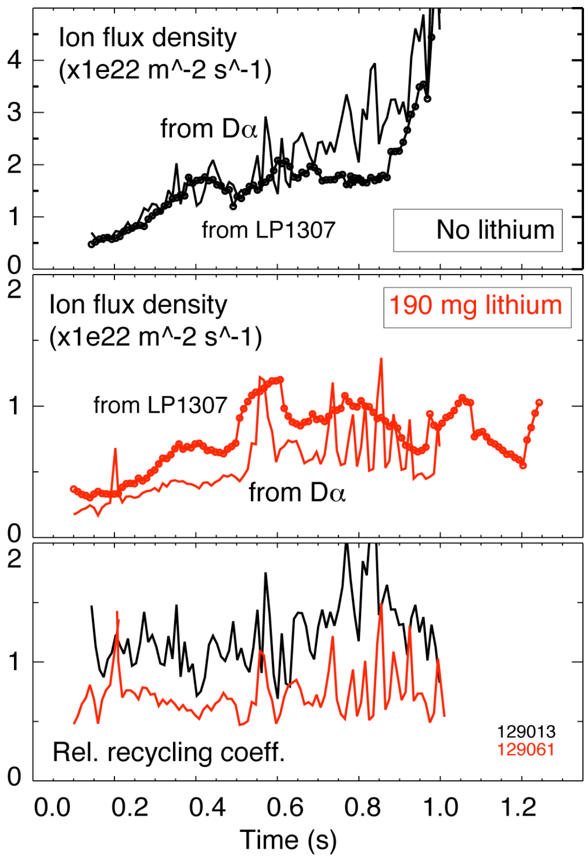


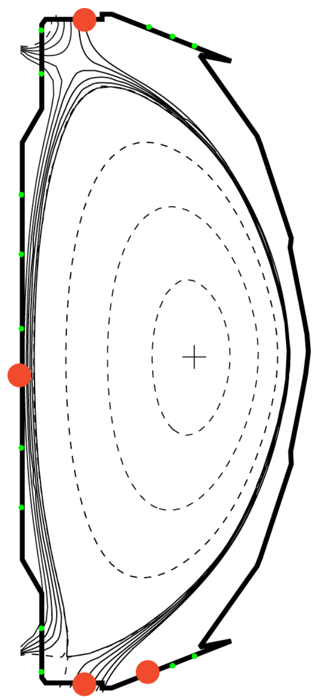


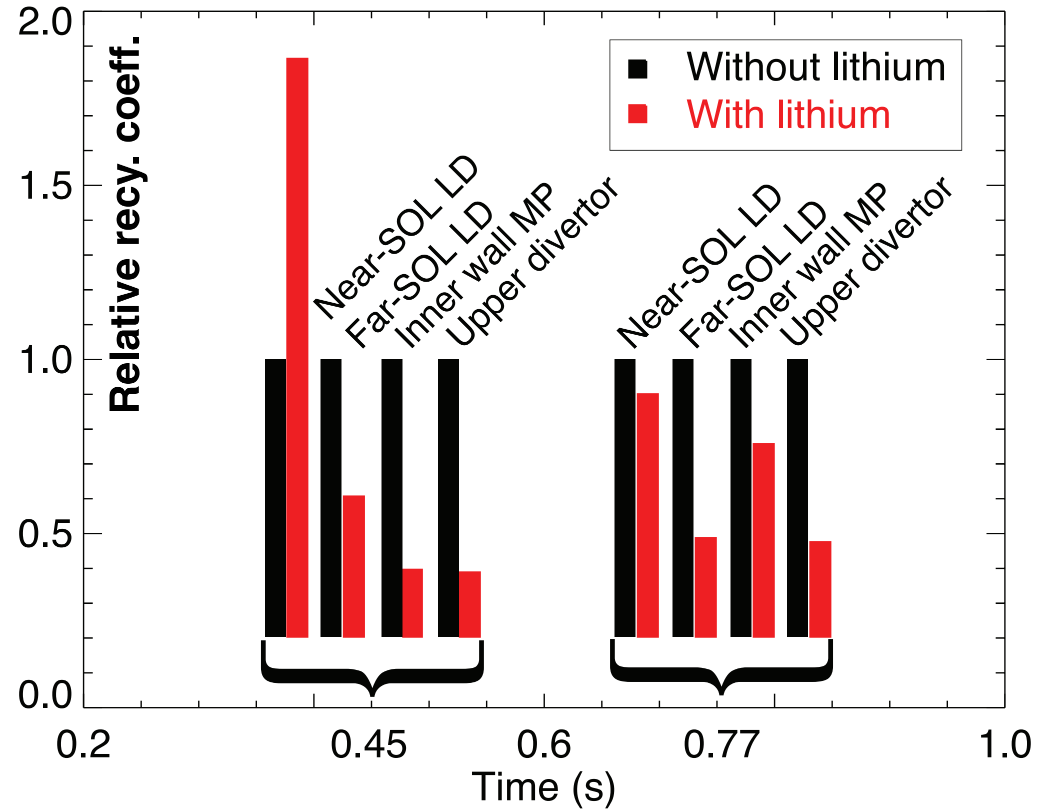
No lithium (129013)
No lithium (129059)
190 mg lithium (129061)
600 mg lithium (129064)

- Divertor ionization source reduced by up to 50 %

Local relative recycling coefficients reduced on all PFCs but in the near-SOL / strike point region



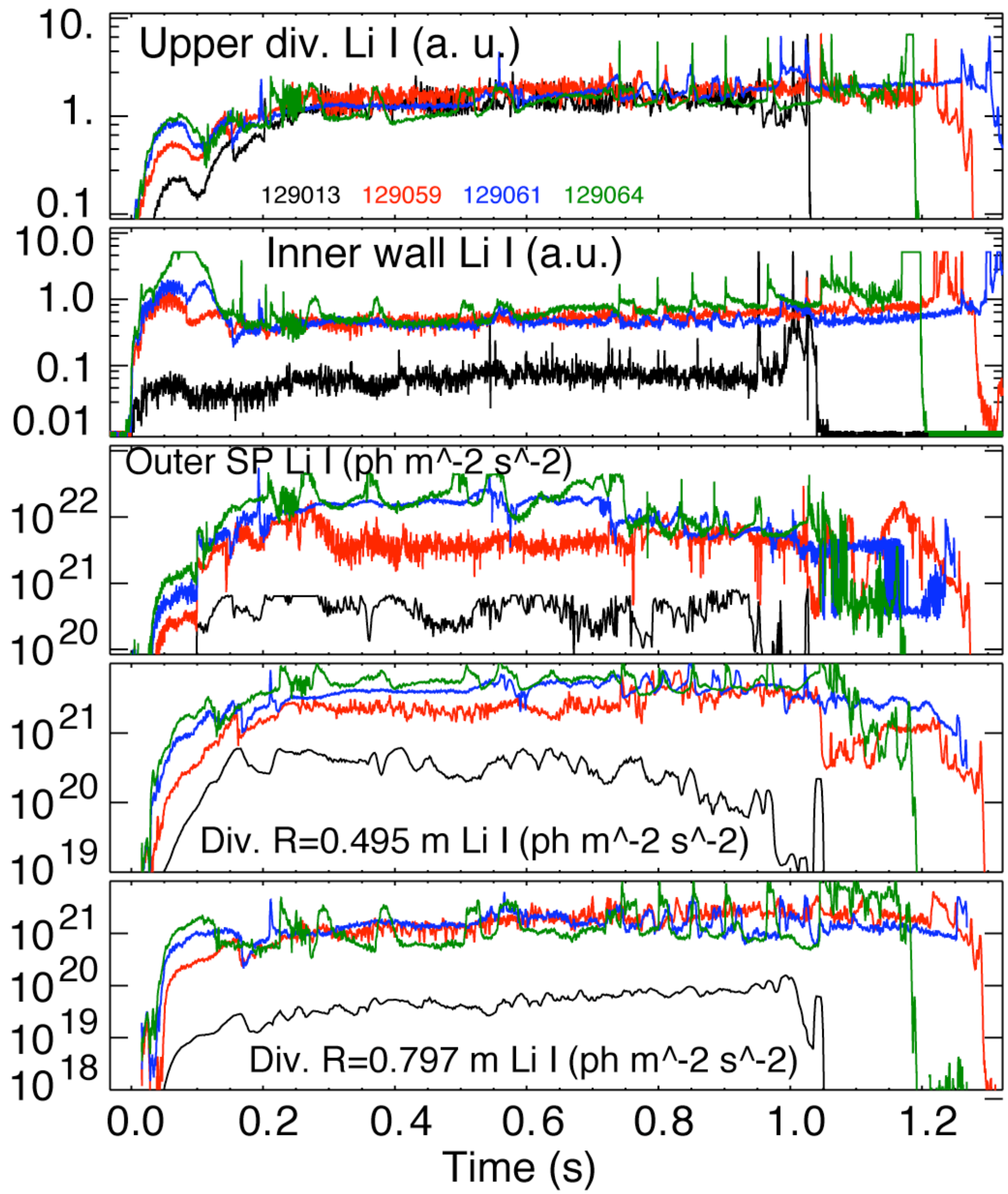




- Local recycling coefficient $R_{local}=\Gamma_i^{out} / \Gamma_i^{in}$
  - Ion flux into surface Γ_i^{in} is measured by Langmuir Probes (LPs)
  - Ion outflux Γ_i^{out} estimated from measured Dα intensity and S/XB (ionizations/photon) coefficient from ADAS

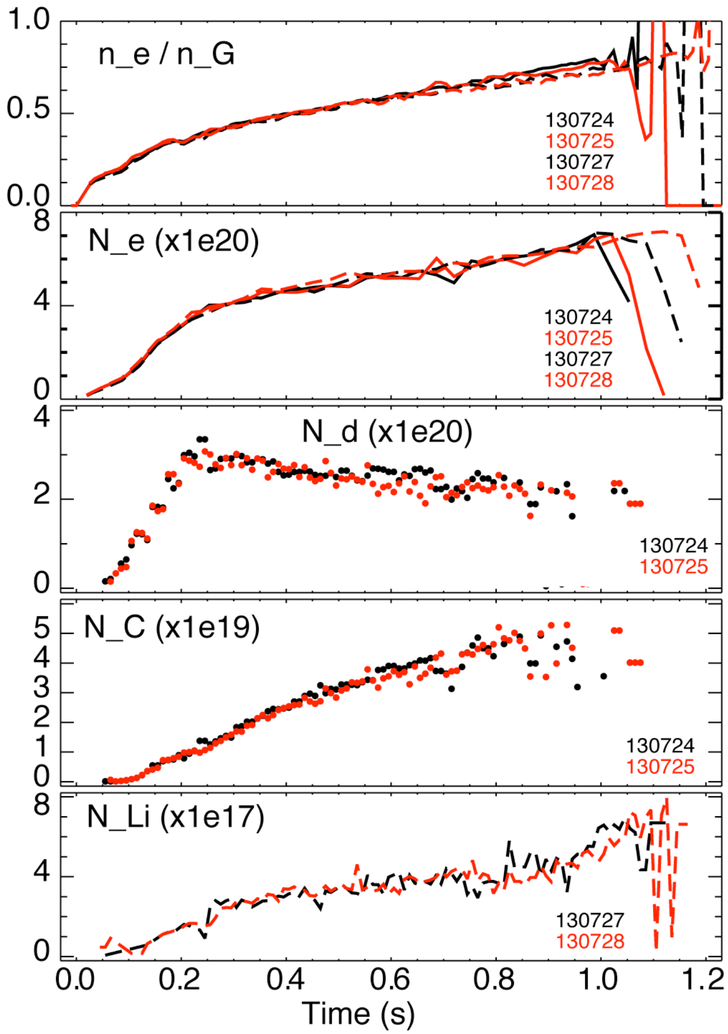
Lithium flux measurements suggest lithium source is in lower divertor, degrades in one discharge

- Strong scaling of lithium fluxes with evaporated amount in early phase on all PFCs
  - In later phase, no scaling in upper divertor, inner wall and far SOL
- In near SOL and strike point, strong scaling until end of discharge (cumulative effect)
- Large difference between “no lithium” reference discharges



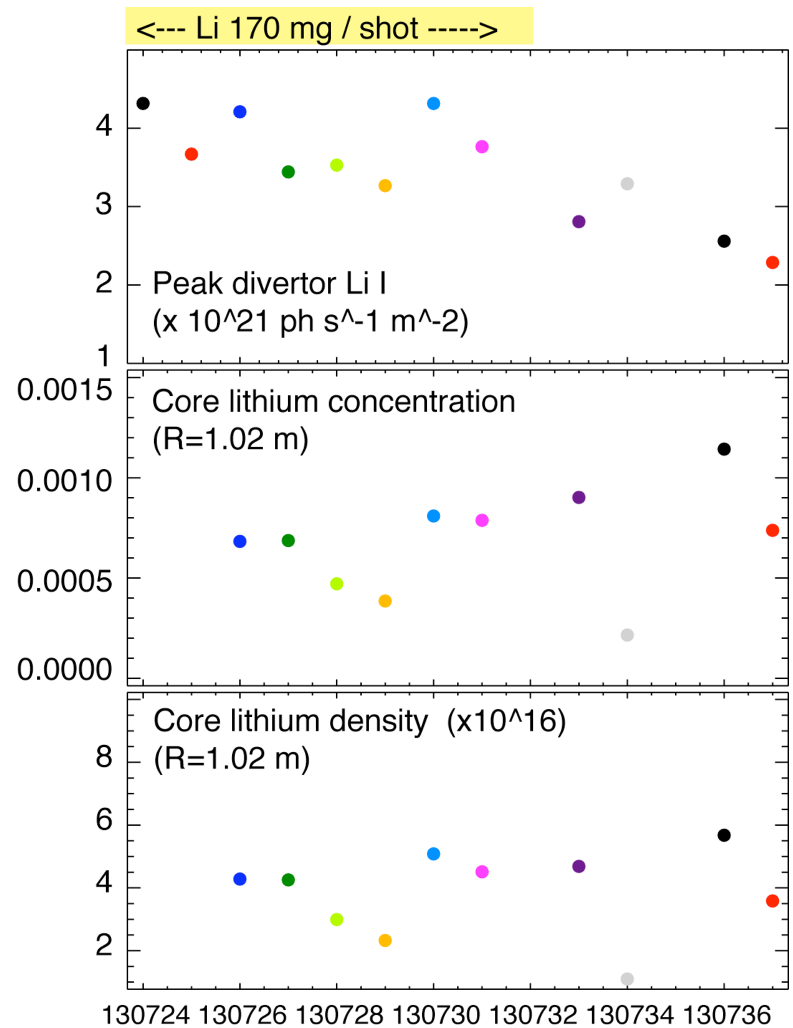
No lithium (129013)
No lithium (129059)
190 mg lithium (129061)
600 mg lithium (129064)

Core lithium density low, does not scale with divertor source, lithium weakly accumulates in core



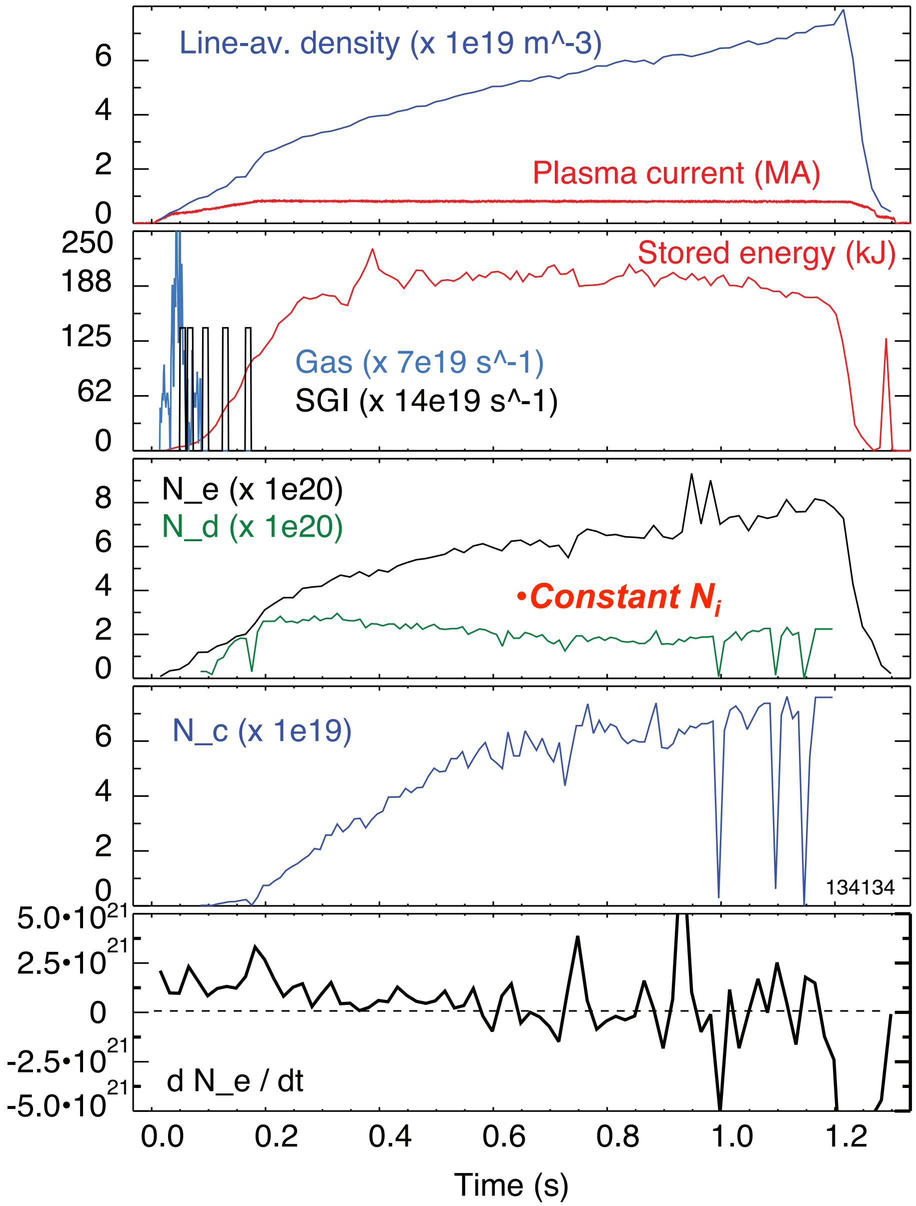
Impurity density profiles from CHERS

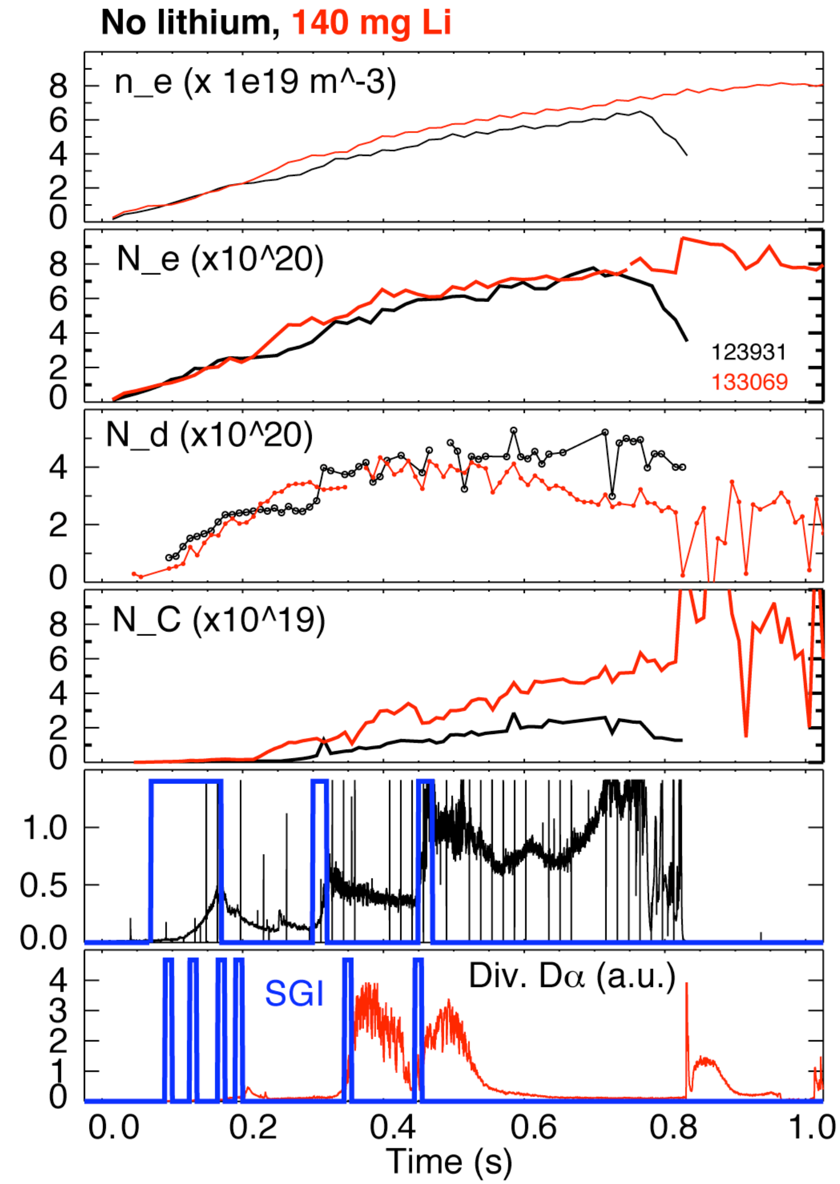
– C VI, n = 8-7, 529.1 nm
– Li III, n = 7-5, 516.7 nm



- Lithium screening efficiency high, penetration factor N_{Li} / Γ_{li} ~ 0.0001

A long pulse H-mode discharge scenario with SGI fueling and lithium-controlled N_i





Divertor with lithium coatings provides pumping – but what about impurity and heat flux handling ?

- On-going study of impurity sources and impurity parallel and radial transport in SOL and pedestal
- Carbon sources: wall and divertor, physical and chemical sputtering
- Reduce physical sputtering yield by lowering divertor temperature
  - $E_i = 2kT_i + 3\,Z_i\,kT_e$
  - $E_i \sim 50 – 300$ eV -> $Y_C \sim 0.01$
  - Need to obtain $E_i \leq 20\text{-}40$ eV ($T_e \leq 5$ eV)
- Low T_e divertor operation established in NSTX
  - Divertor gas puffing
  - Snowflake divertor

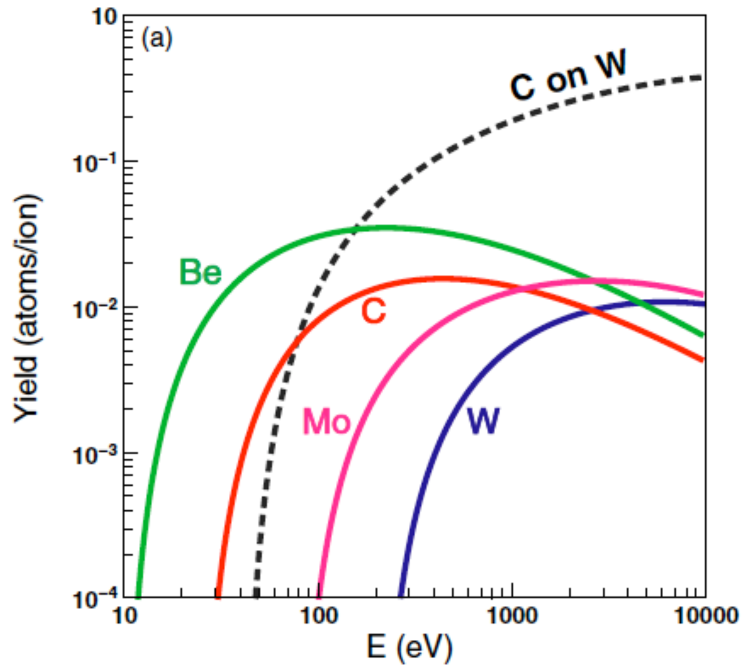
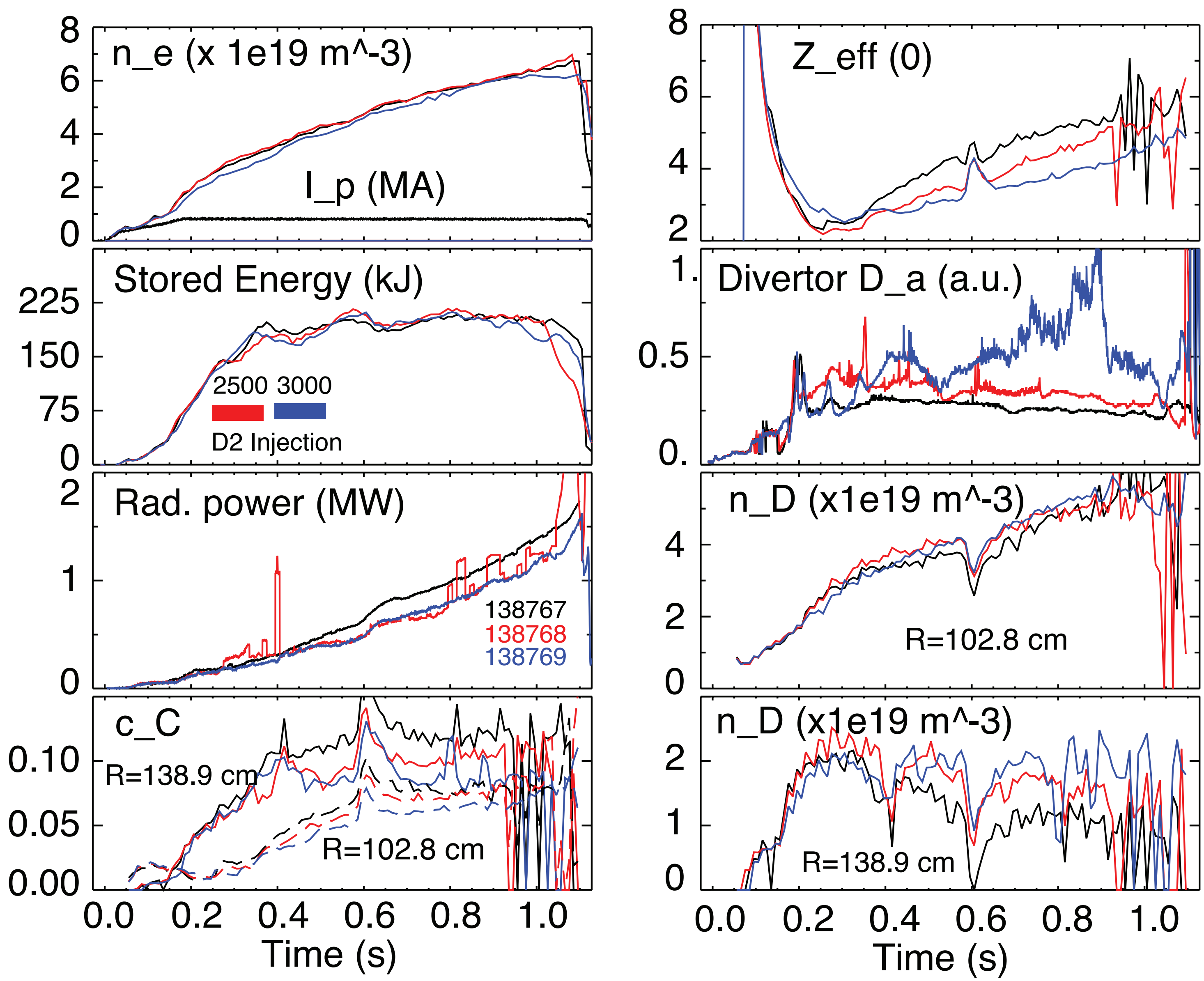


Figure from R. A. Pitts et. al, PPCF (2005) B303

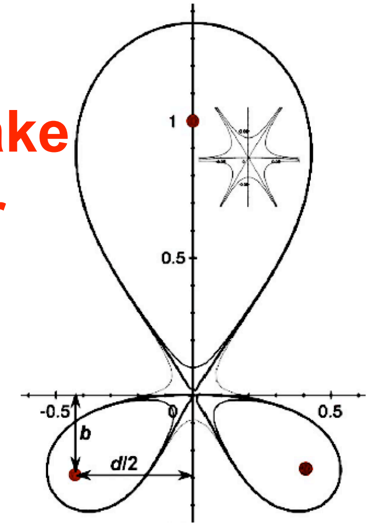
Divertor D_2 puffing used to reduce divertor carbon source and core carbon accumulation

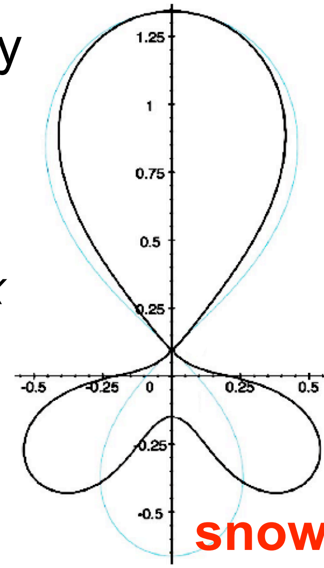


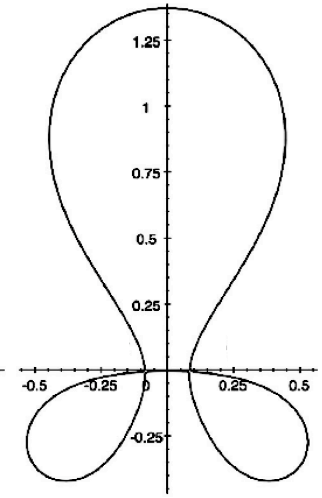
Snowflake divertor geometry attractive for heat flux mitigation

- Snowflake divertor
  - Second-order null
    - $B_p \sim 0$ and *grad* $B_p \sim 0$ (Cf. first-order null: $B_p \sim 0$)
  - Obtained with existing divertor coils (min. 2)
  - Exact snowflake topologically unstable
- Predicted geometry properties (cf. standard divertor)
  - Larger region with low B_p around X-point: ped. stability
  - Larger plasma wetted-area A_{wet} : reduce q_{div}
  - Larger X-point connection length L_x : reduce q_{II}
  - Larger effective divertor volume V_{div} : incr. P_{rad} , P_{CX}
- Experiments
  - TCV (F. Piras *et. al*, PRL 105, 155003 (2010))
  - NSTX

Exact snowflake divertor







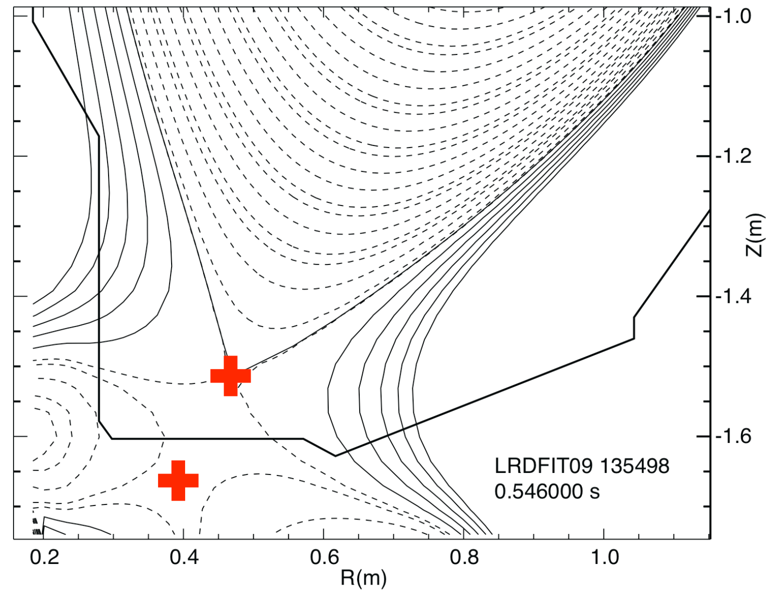
snowflake-minus

snowflake-plus

D. D. Ryutov, PoP 14, 064502 2007

Snowflake configuration as the laboratory for pedestal MHD stability and parallel SOL transport

- Obtained with three divertor coils PF1A, PF2L and rev. pol. PF1B
- Properties
  - Overall, same shaping and core plasma parameters as with standard divertor
  - Outer SP flux expansion can be same as in standard divertor !
  - Pedestal magnetic shear and SOL connection length higher than in standard divertor
- Test parallel transport with 3D fields
  - Role of increased line length
  - Role of radial heat diffusion (to common and private flux regions)
- Test pedestal stability models
  - A knob to for peeling-balooning stability



$$q_{peak} \simeq \frac{P_{SOL}(1-f_{rad})f_{geo}\sin\alpha}{2\pi R_{SP} f_{exp} \lambda_{q_{\|}}}$$

$$A_{wet} = 2\pi R\ f_{exp}\ \lambda_{q_{\|}}$$

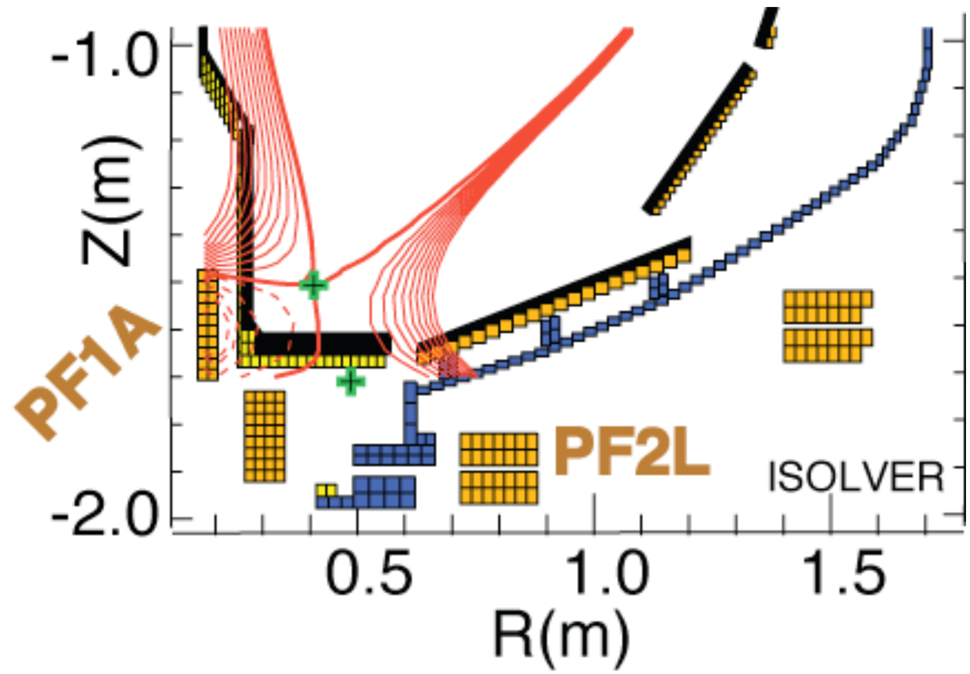
$$f_{exp} = \frac{(B_p/B_{tot})_{MP}}{(B_p/B_{tot})_{OSP}}$$

Possible NSTX snowflake divertor configurations were modeled with ISOLVER code

- ISOLVER - predictive free-boundary axisymmetric Grad-Shafranov equilibrium solver
  - Input: normalized profiles (P, I_p), boundary shape
  - Match a specified I_p and β
  - Output: magnetic coil currents

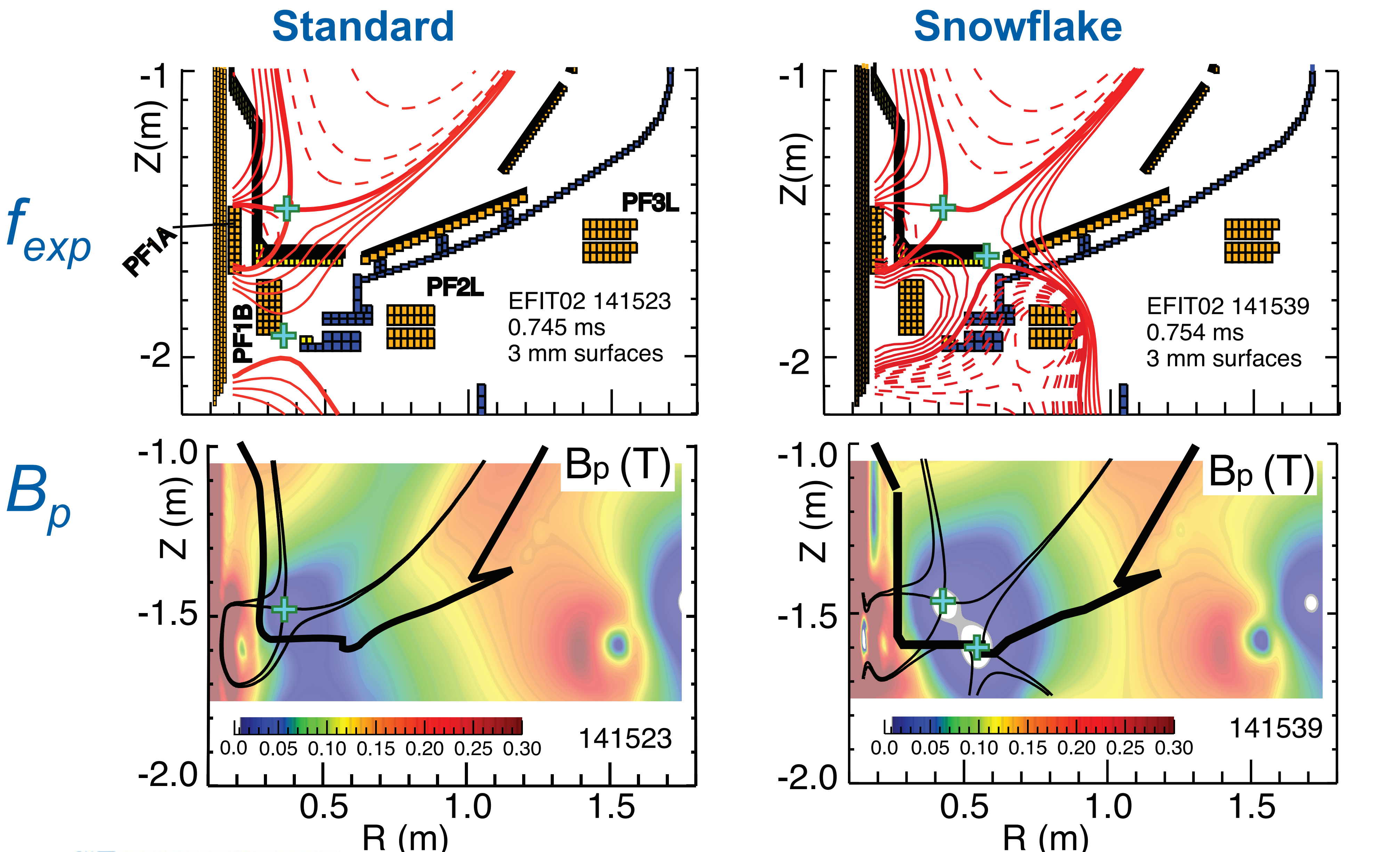
✓ Standard divertor discharge below:

B_t=0.4 T, I_p=0.8 MA, δ_{bot}~0.6, κ~2.1



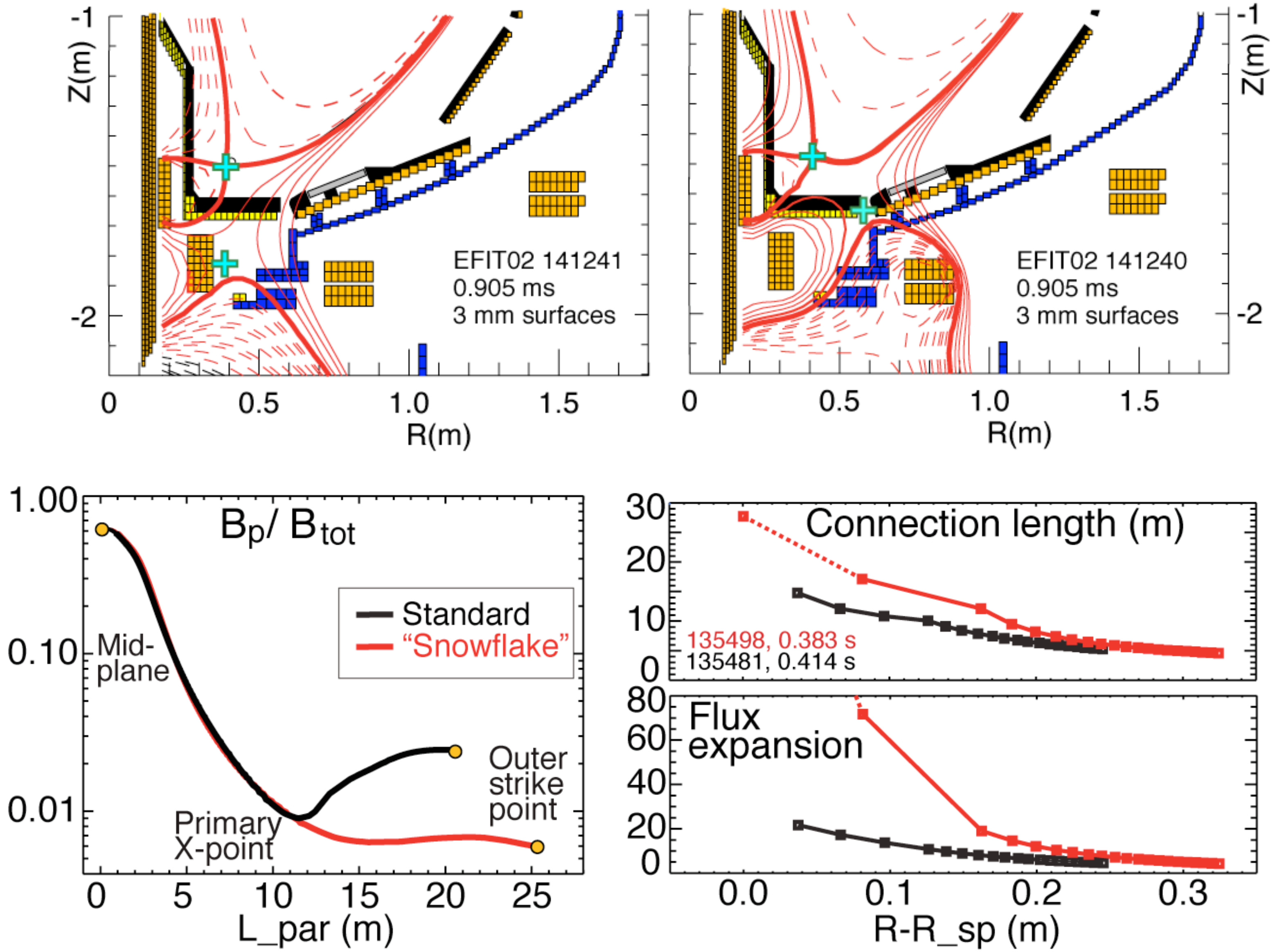
Quantity	Standard divertor	Simulated snowflake
X-point to target parallel length L_x (m)	**5-10**	**10**
Poloidal magnetic flux expansion f_{exp} at outer SP	**10-24**	**30-60**
Magnetic field angle at outer SP (deg.)	**1.5-5**	**~1-2**
Plasma-wetted area A_{wet} (m^2)	**≤ 0.4**	**0.95**

Snowflake divertor configurations obtained in NSTX confirm analytic theory and modeling

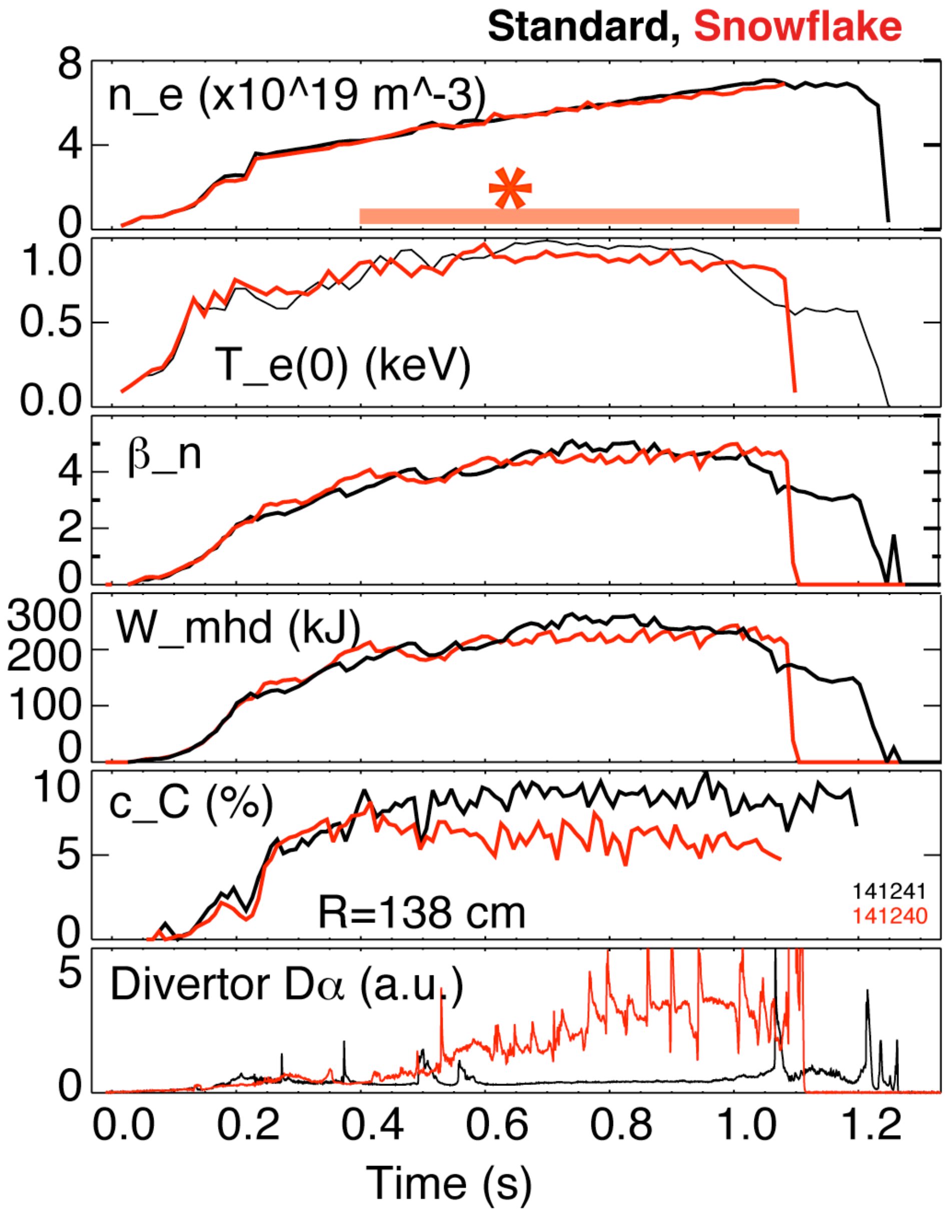


Plasma-wetted area and connection length are increased by 50-90 % in snowflake divertor

- These properties observed in first 30-50 % of SOL width
- B_{tot} angles in the strike point region: 1-2°, sometimes < 1°

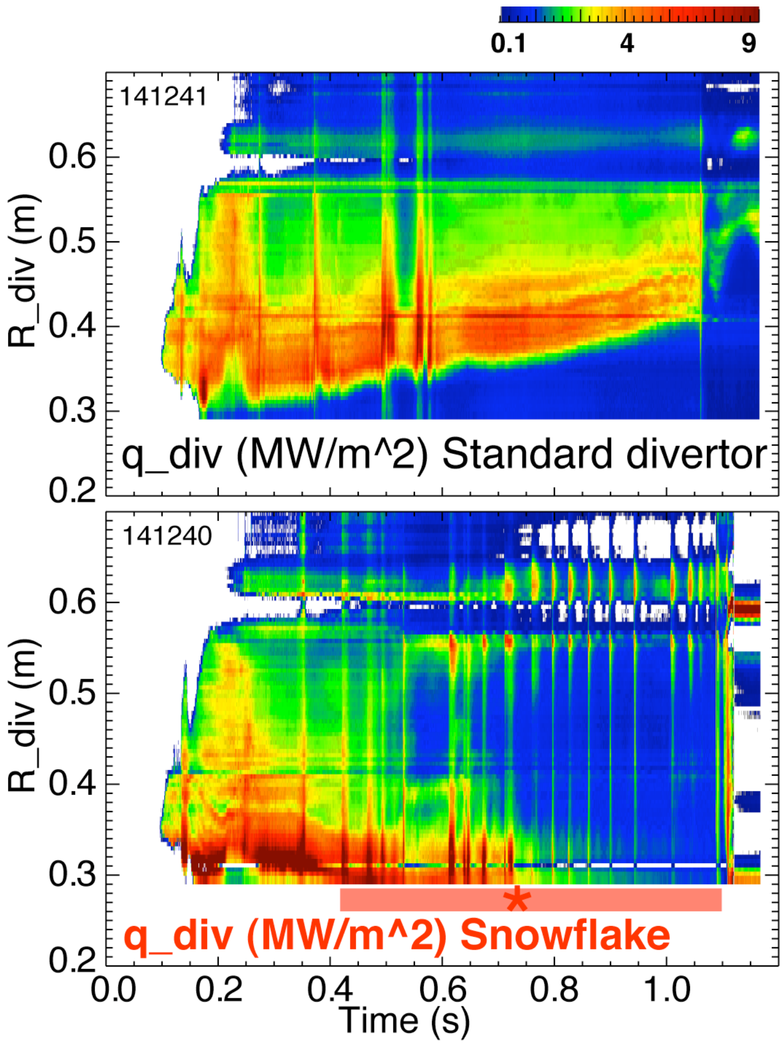


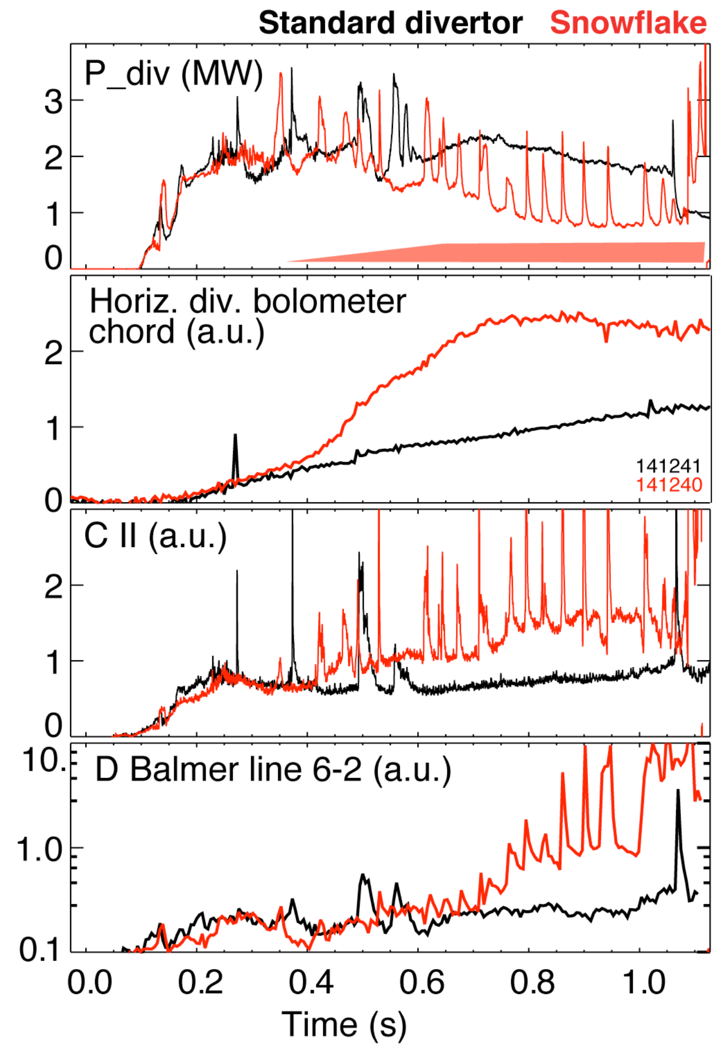
Good H-mode confinement properties and core impurity reduction obtained with snowflake divertor



- 0.8 MA, 4 MW H-mode
- κ=2.1, δ=0.8
- Core T_e ~ 0.8-1 keV, T_i ~ 1 keV
- β_N ~ 4-5
- Plasma stored energy ~ 250 kJ
- H98(y,2) ~ 1 (from TRANSP)
- Core carbon reduction due to
  - Type I ELMs
  - Edge source reduction
    - Divertor sputtering rates reduced due to partial detachment

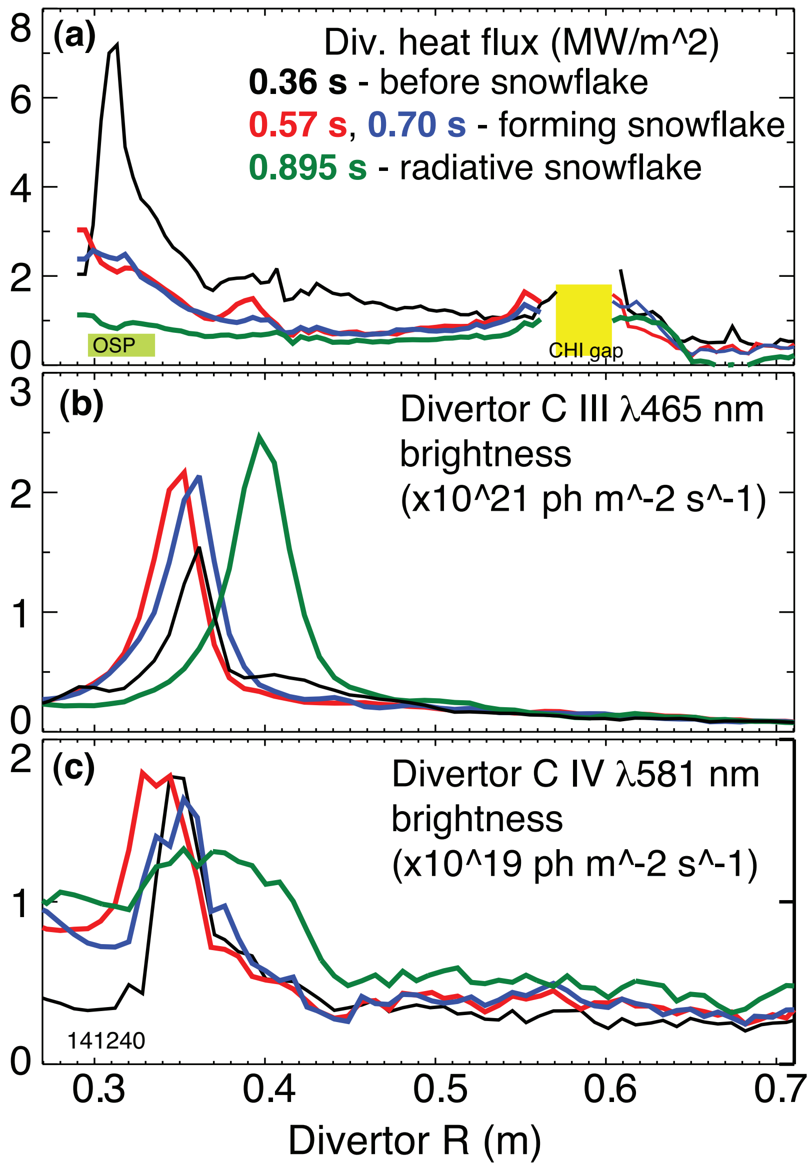
Significant reduction of steady-state divertor heat flux observed in snowflake divertor (at P_{SOL}~ 3 MW)



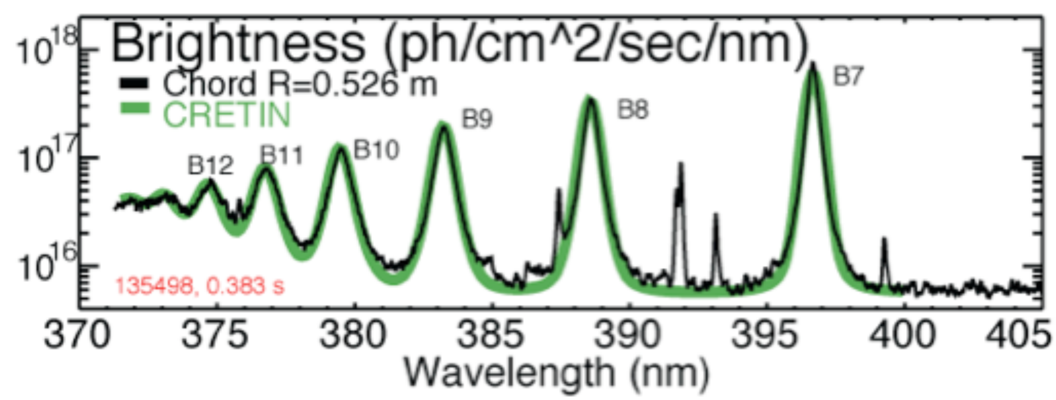


- Partial detachment at or after snowflake formation time
  - Heat and ion fluxes in the outer strike point region decreased
  - Divertor recombination rate and radiated power are increased

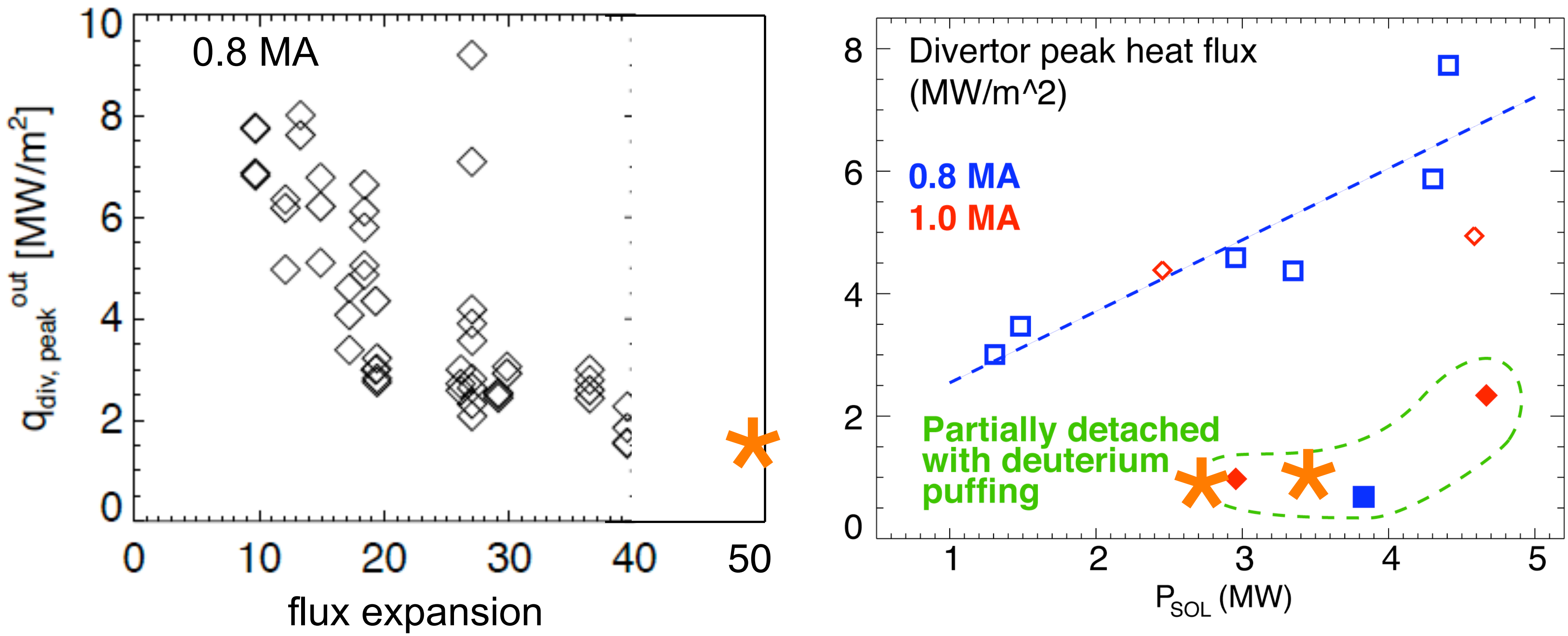
Divertor profiles show low heat flux, broadened C III and C IV radiation zones in the snowflake divertor phase



- Heat flux profiles reduced to nearly flat low levels, characteristic of radiative heating
- Divertor C III and C IV brightness profiles broaden
- High-*n* Balmer line spectroscopy and CRETIN code modeling confirm outer SP detachment with $T_e \leq 1.5$ eV, $n_e \leq 5 \times 10^{20}$ m^{-3}
  - Also suggests a reduction of carbon physical and chemical sputtering rates



Snowflake divertor heat flux consistent with NSTX divertor heat flux scalings



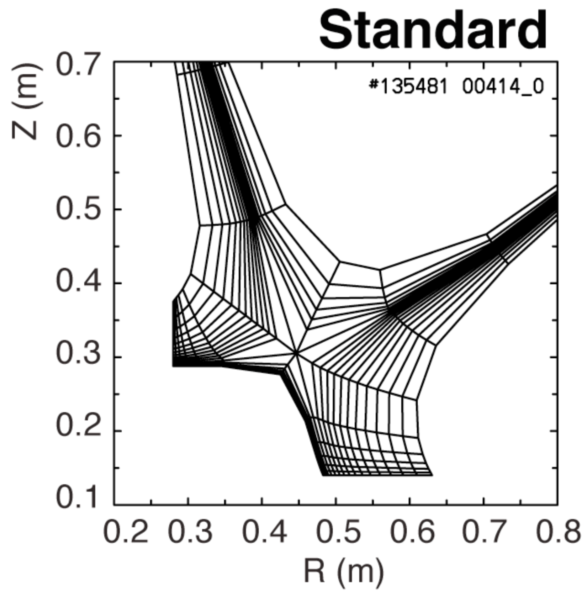
- Snowflake divertor (*): P_{SOL}~3-4 MW, f_{exp}~40-60, q_{peak}~0.5-1.5 MW/m^2

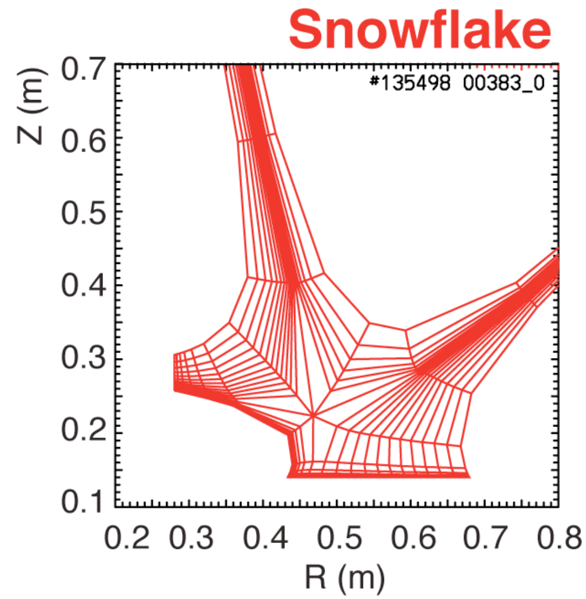
T. K. Gray et. al, EX/D P3-13, IAEA FEC 2010

V. A. Soukhanovskii et. al, PoP 16, 022501 (2009)

Modeling shows a trend toward reduced temperature, heat and particle fluxes in the snowflake divertor

- 2D multi-fluid code UEDGE
  - Fluid (Braginskii) model for ions and electrons
  - Fluid for neutrals
  - Classical parallel transport, anomalous radial transport
    - $D = 0.25\ m^2/s$
    - $\chi_{e,i} = 0.5\ m^2/s$



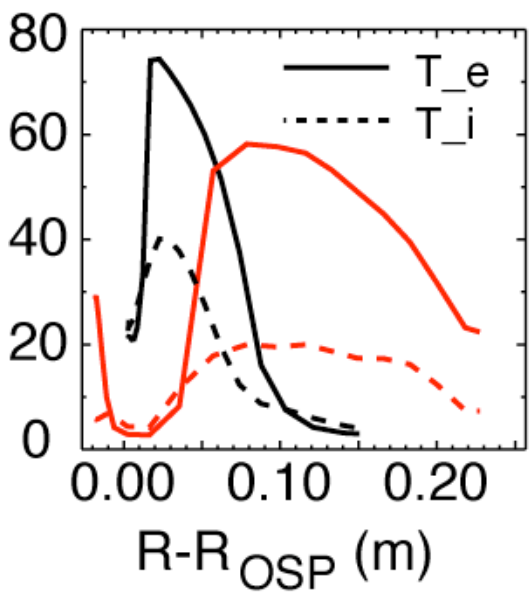


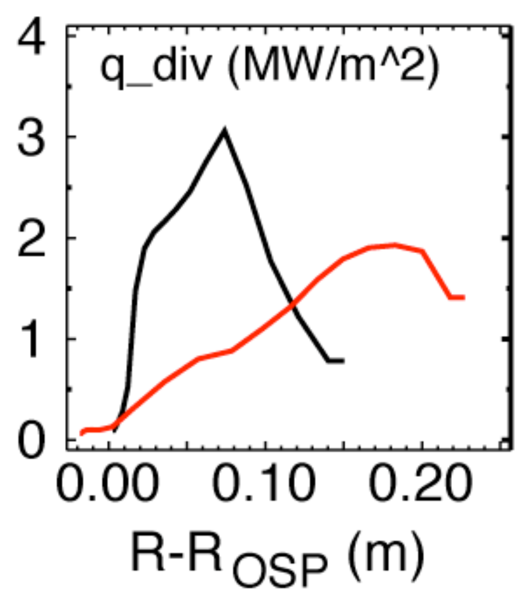
Core interface:

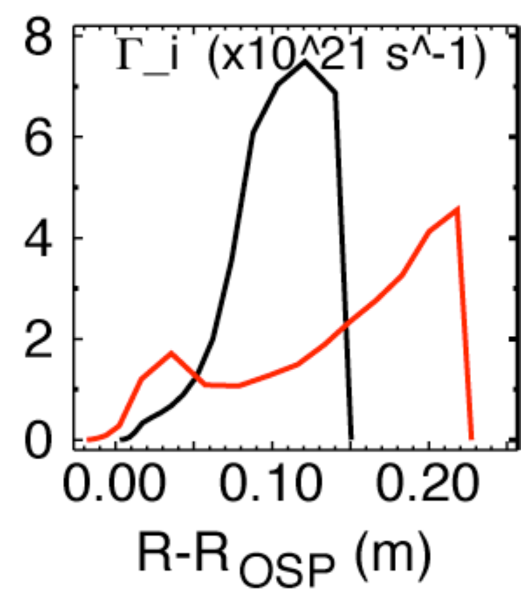
- $T_{e,i}$ = 120 eV
- $n_e = 4.5 \times 10^{19}$

R_{recy} = 0.95

Carbon 3 %

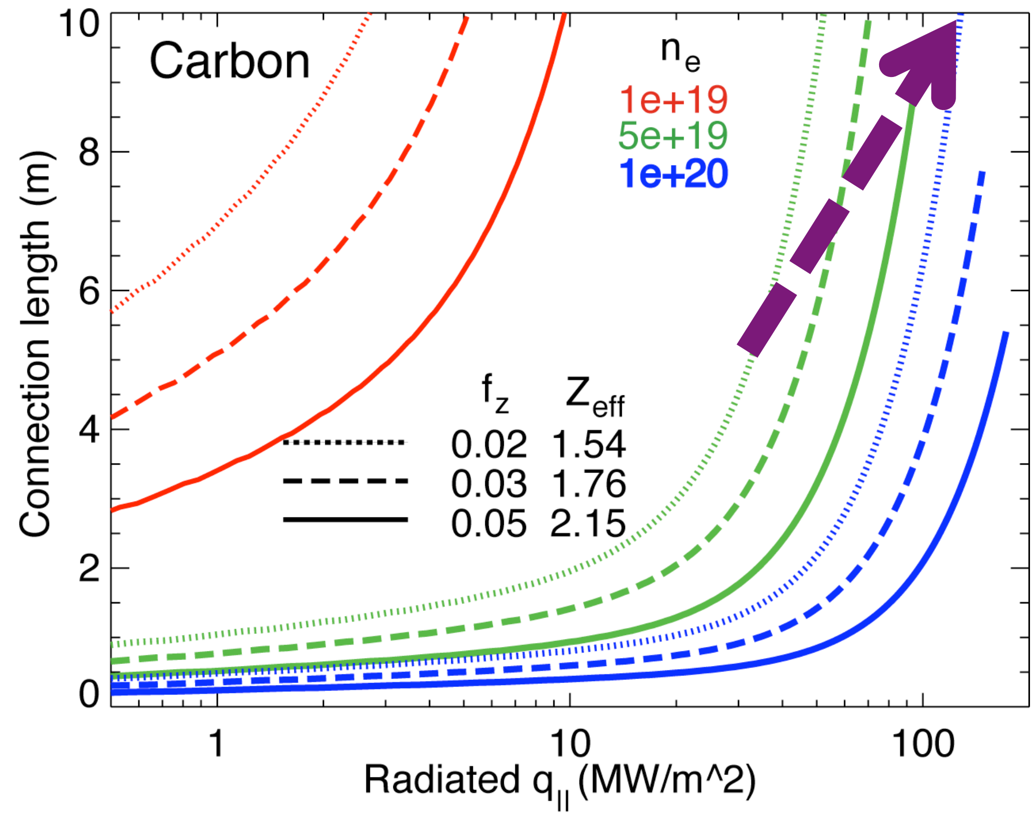






1D estimates indicate power and momentum losses are increased in snowflake divertor

- 1D divertor detachment model by Post
  - Electron conduction with non-coronal carbon radiation
  - Max $q_{||}$ that can be radiated as function of connection length for range of f_z *and* n_e
- Three-body 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: $\tau_{ion} \leq$ 3-6 ms in standard divertor, x 2 in snowflake

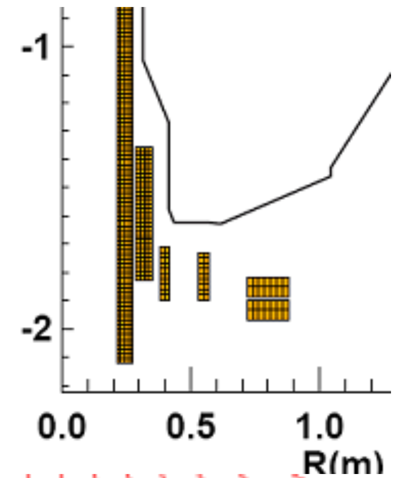


$$q_{||} = -\kappa_0 T_e^{5/2} \frac{\partial T_e}{\partial x}$$

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

Outlook for NSTX-Upgrade

- Development of divertor solutions to address
  - 2-3x higher input power
    - $P_{NBI} < 12$ MW, $I_p < 2$ MA
  - 3-5 x longer pulse duration (t< 5 s)
  - Up to 30 % reduction in Greenwald fraction
  - Projected peak divertor heat fluxes up to 25 MW/m^2



- Heat-flux handling solutions
  - Lower snowflake configuration
  - Upper and lower snowflake divertor
  - Snowflake + impurity radiation
- Density control
  - Lithium coatings or liquid lithium
  - Conceptual cryo-pump design
- Divertor control
  - Additional divertor coil PF1C
  - Developing snowflake control
  - Developing plans for radiative feedback control