Divertor Heat Flux Mitigation in High-Performance H-mode Discharges in NSTX

V. A. Soukhanovskii[1], R. Maingi[2], D. A. Gates[3], J. E. Menard[3], S. F. Paul[3], R. Raman[4], A. L. Roquemore[3], R. E. Bell[3], C. E. Bush[2], R. Kaita[3], H. W. Kugel[3], B. P. LeBlanc[3], D. Mueller[3], and the NSTX Research Team

[1] Lawrence Livermore National Laboratory, Livermore, CA, USA
[2] Oak Ridge National Laboratory, Oak Ridge, TN, USA
[3] Princeton Plasma Physics Laboratory, Princeton, NJ, USA
[4] University of Washington, Seattle, WA, USA

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Abstract and Acknowledgments

Abstract. Experiments conducted in high-performance 1.0-1.2 MA 6 MW NBI-heated H-mode plasmas with a high flux expansion radiative divertor in NSTX demonstrate that significant divertor peak heat flux reduction and access to detachment may be facilitated naturally in a highly-shaped spherical torus (ST) configuration. Improved plasma performance with high $\beta_p = 15 - 25$ %, a high bootstrap current fraction $f_{BS} = 45 - 50$ %, longer plasma pulses, and an H-mode regime with smaller ELMs has been achieved in the lower single null configuration with higher-end elongation 2.2-2.4 and triangularity 0.6-0.8. Divertor peak heat fluxes were reduced from 6-12 MW/m^2 to 0.5-2 MW/m^2 in ELMy H-mode discharges using high magnetic flux expansion and partial detachment of the outer strike point at several D_2 injection rates, while good core confinement and pedestal characteristics were maintained. The partially detached divertor regime was characterized by a 30-60 % increase in divertor plasma radiation, a peak heat flux reduction by up to 70 %, measured in a 10 cm radial zone, a five-fold increase in divertor neutral pressure, and a significant volume recombination rate increase.

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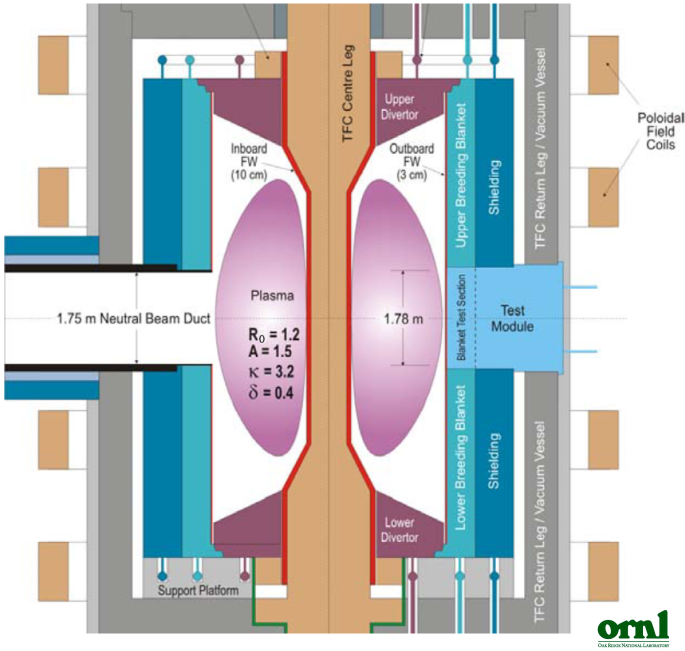
Conclusions

- Significant divertor **peak heat flux reduction** has been demonstrated in highly shaped high-performance H-mode plasmas in NSTX using **divertor magnetic flux expansion and radiative divertor** simultaneously with **high core plasma performance**
  - Good synergy of high performance H-mode regime with partially detached divertor

- Detachment characteristics in NSTX
  - Steady-state PDD regime achieved only with additional gas injection into a high flux expansion divertor
  - High divertor radiated power, neutral pressure, volume recombination rate measured
  - PDD properties appear to be similar to those observed in tokamak
  - SOL geometry limits radiated power and momentum losses to the separatrix region

Divertor heat flux mitigation is key for present and future fusion plasma devices

- **Radiative divertor** is envisioned for present and future devices (e.g. ITER) as the **steady-state** heat flux mitigation solution
  - Divertor $q_{peak} < 10$ MW/m^2
  - Large radiated power fractions ($f_{rad} = 0.50 - 0.80$)
  - Integration with pedestal and core
  - Partially detached divertor (**PDD**) is the most promising regime

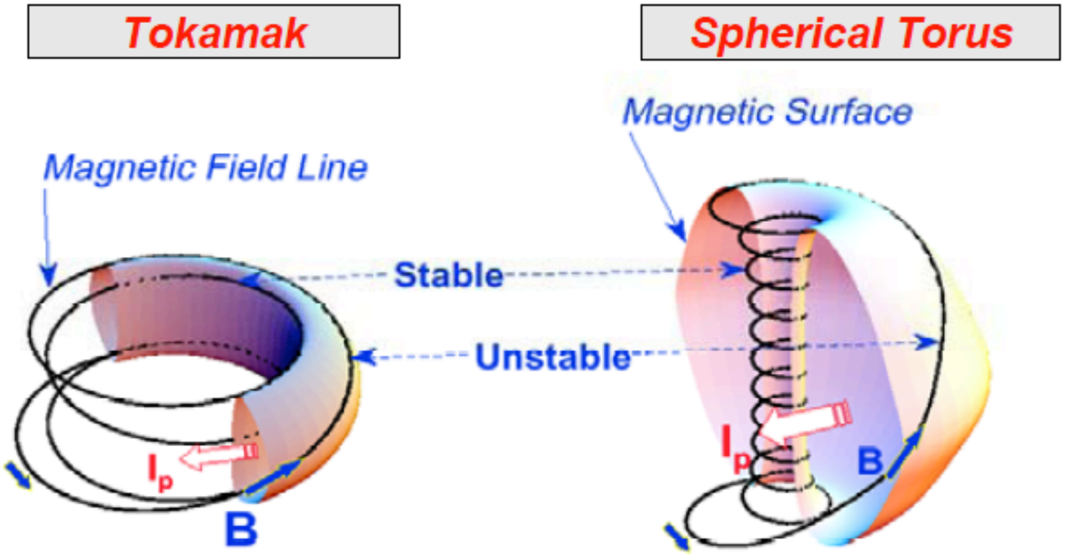


Peng et al, PPCF 47, B263 (2005)

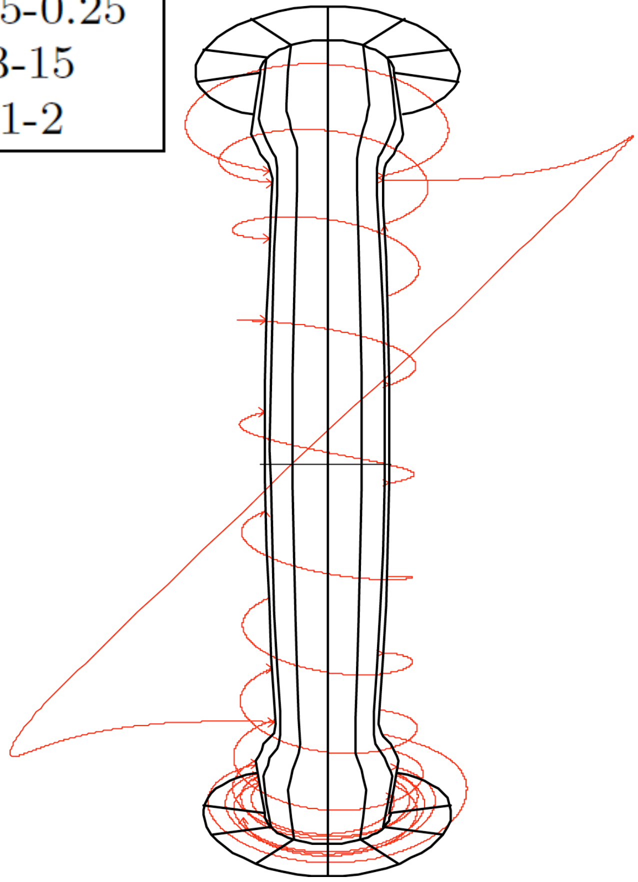
- Radiative divertor in NSTX
  - Does radiative divertor work in a spherical torus (ST) with a compact high $q_{||}$ divertor? What are the limitations?
  - Experimental basis for radiative divertor optimization and projections to ST-CTF

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



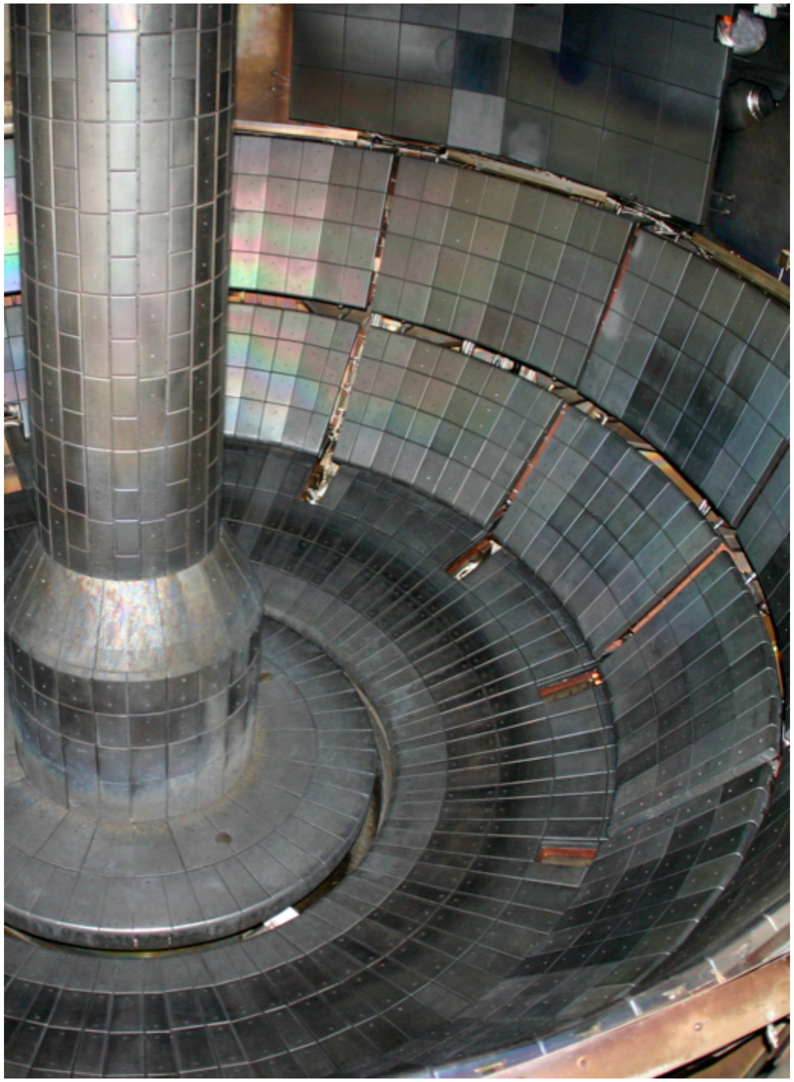
$R/a=2.5\text{-}4$, $\kappa=1.5\text{-}2$, $q_e=3\text{-}5$ $R/a=1.1\text{-}1.6$, $\kappa=2\text{-}3$, $q_e=6\text{-}15$



Open geometry NSTX divertor enables flexibility in plasma shaping

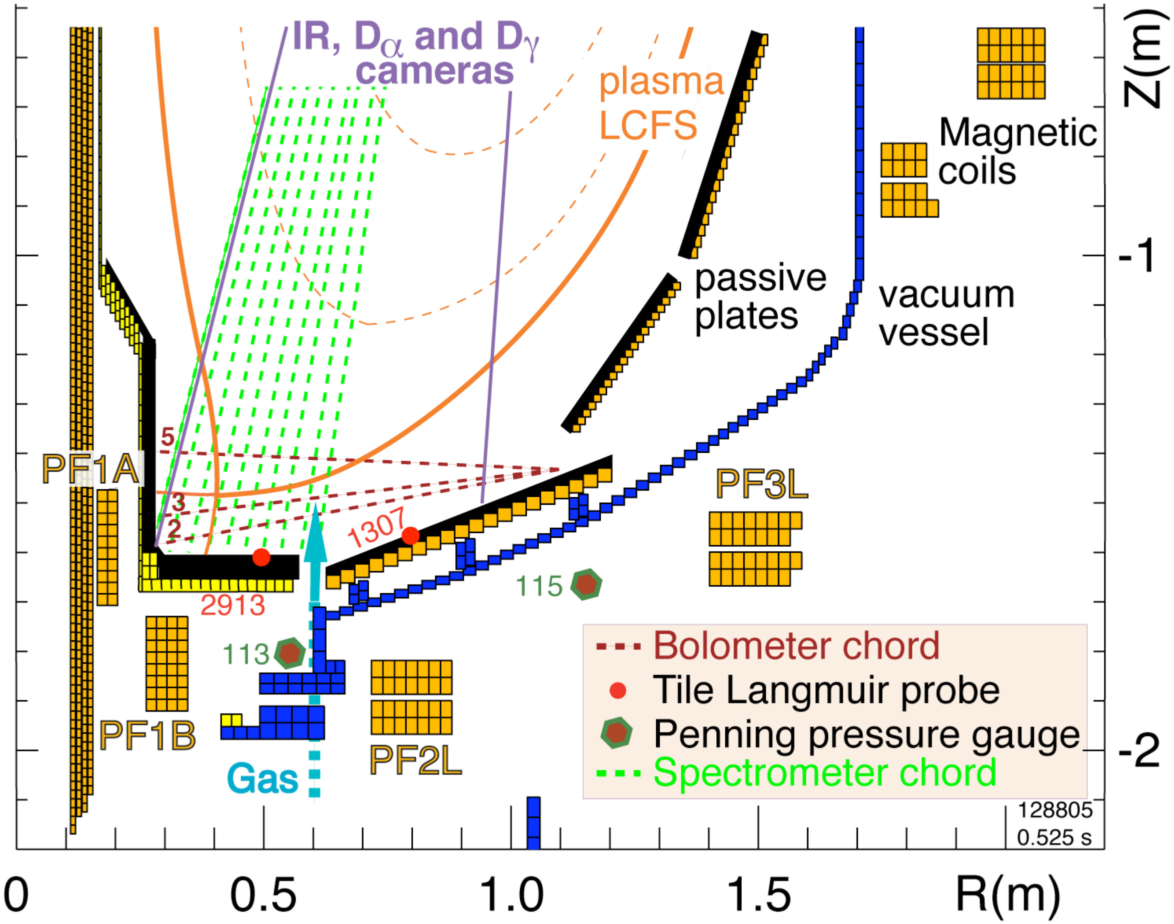
- Plasma facing components
  - ATJ and CFC tiles
  - Carbon - erosion, sputtering
  - Max P_{rad} fraction limited by carbon radiation efficiency
  - Typical divertor tile temperature in 1 s pulses $T < 500$ C ($q_{peak} \leq 10$ MW/m^2)

- No active divertor pumping
  - Experiments with lithium coatings for reduced recycling (see Kaita et al., EX/P4-9)

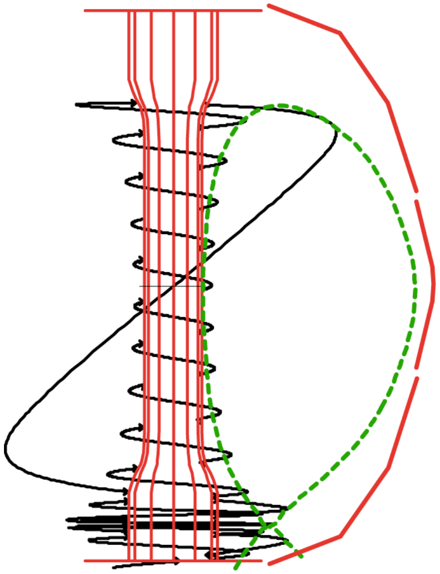


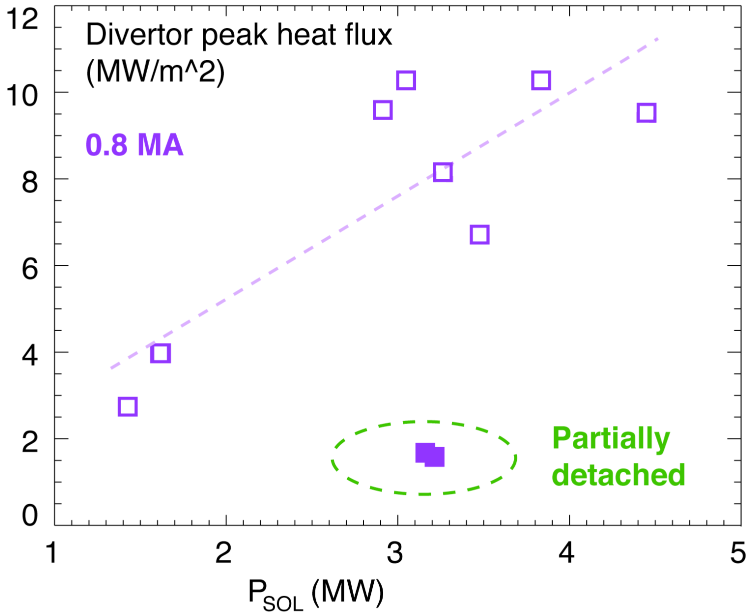
Multiple diagnostic measurements are analyzed to elucidate on radiative divertor physics in NSTX

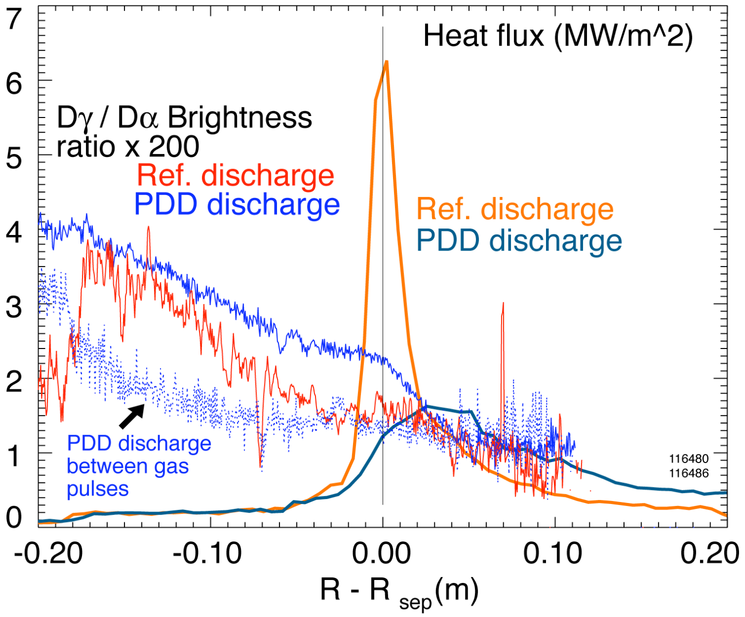
- Diagnostic set for divertor studies:
  - IR cameras
  - Bolometers
  - Neutral pressure gauges
  - Tile Langmuir probes
  - D_α, D_γ filtered CCD arrays
  - UV-VIS spectrometer (10 divertor chords)
- Midplane Thomson scattering and CHERS systems
- Divertor gas injector Γ_{gas} = 20-200 Torr l / s



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



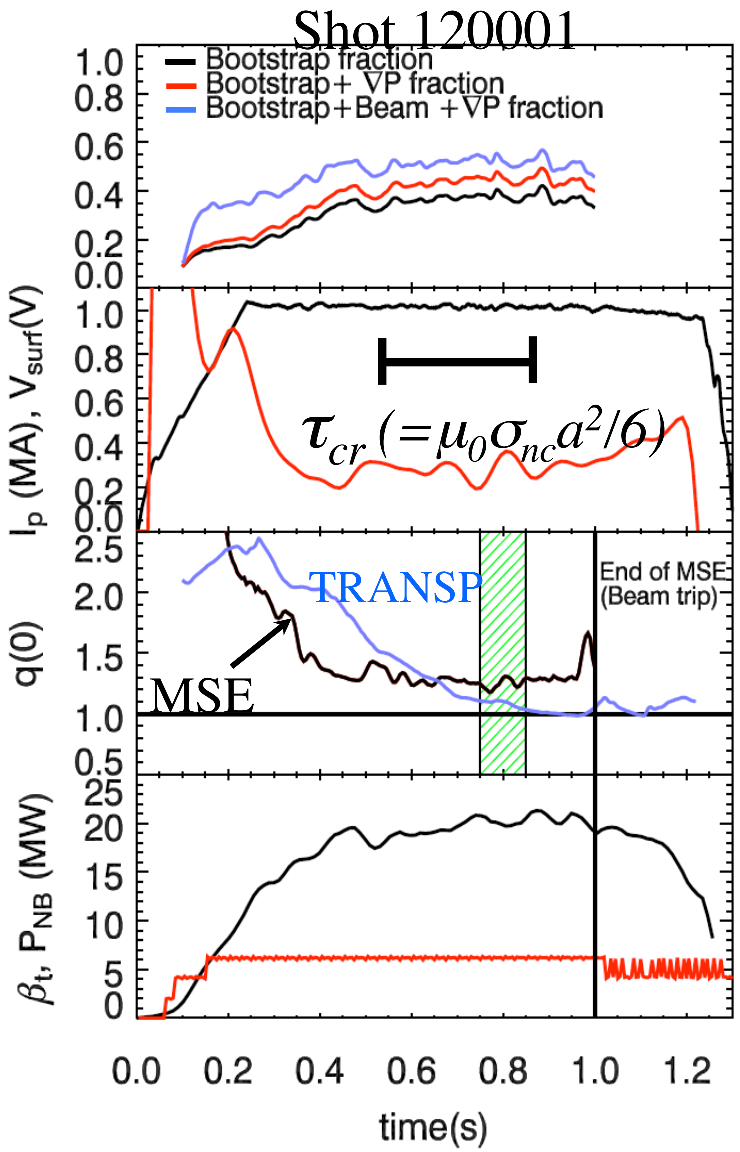




- Peak heat flux in outer divertor (**Maingi JNM 363-365, 196 (2007)**):
  - ITER-level q_{out}< 10 MW/m^2
  - Scaling of q_{peak}: linear with P_{sol} (P_{NBI}), linear-monotonic with I_p
  - Large q_{peak} asymmetry - 2-10; inner divertor always detached
- Experiments using D$_2$ injection (**Soukhanovskii IAEA 2006**):
  - q_{peak} reduced by up to 60 % in transient PDD regime
  - X-point MARFE degraded confinement within 2-3 x τ_E

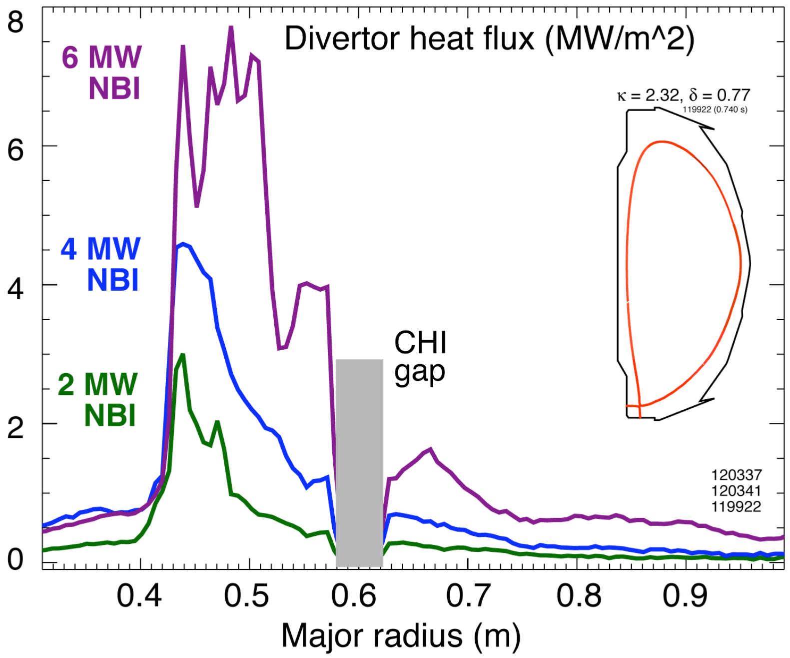
High plasma performance and reduced q_{pk} are attained in highly shaped plasmas

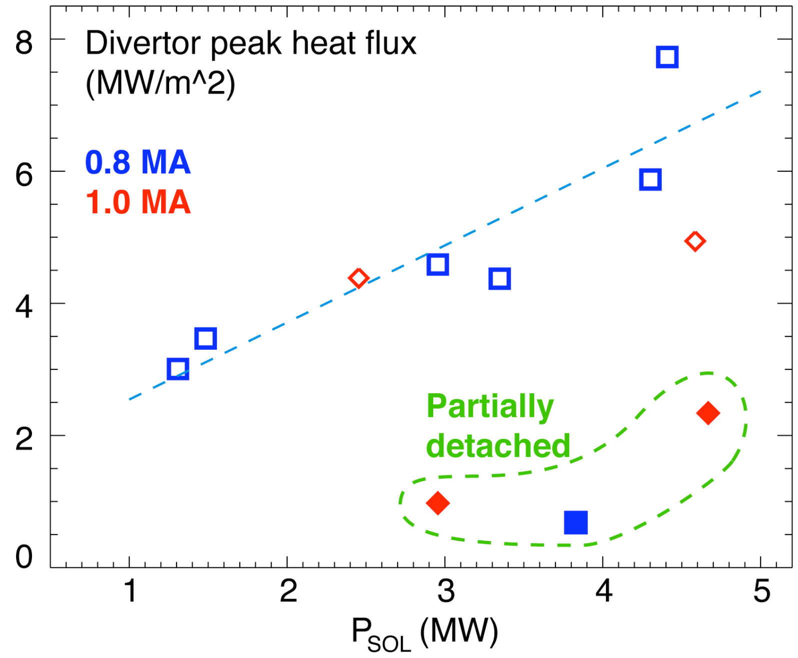
- High performance H-mode (**Gates APS 2005, Maingi APS 2005, Menard IAEA 2006**)
  - κ = 2.2-2.3, δ = 0.65-0.75, *drsep* ~ 5-10 mm
  - H89P ~ 2.0
  - β_t = 15 - 25 %
  - f_{bs} = 45 - 50 %
  - longer pulses ~50 x τ_E
  - smaller ELMs



- Divertor in highly shaped plasmas
  - High flux expansion, area expansion (q_{peak} ↓)
  - Higher isothermal SOL volume (P_{rad} ↑)
  - Lower L_p (neutral penetration ↑)
  - Neutrals recycle toward separatrix

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

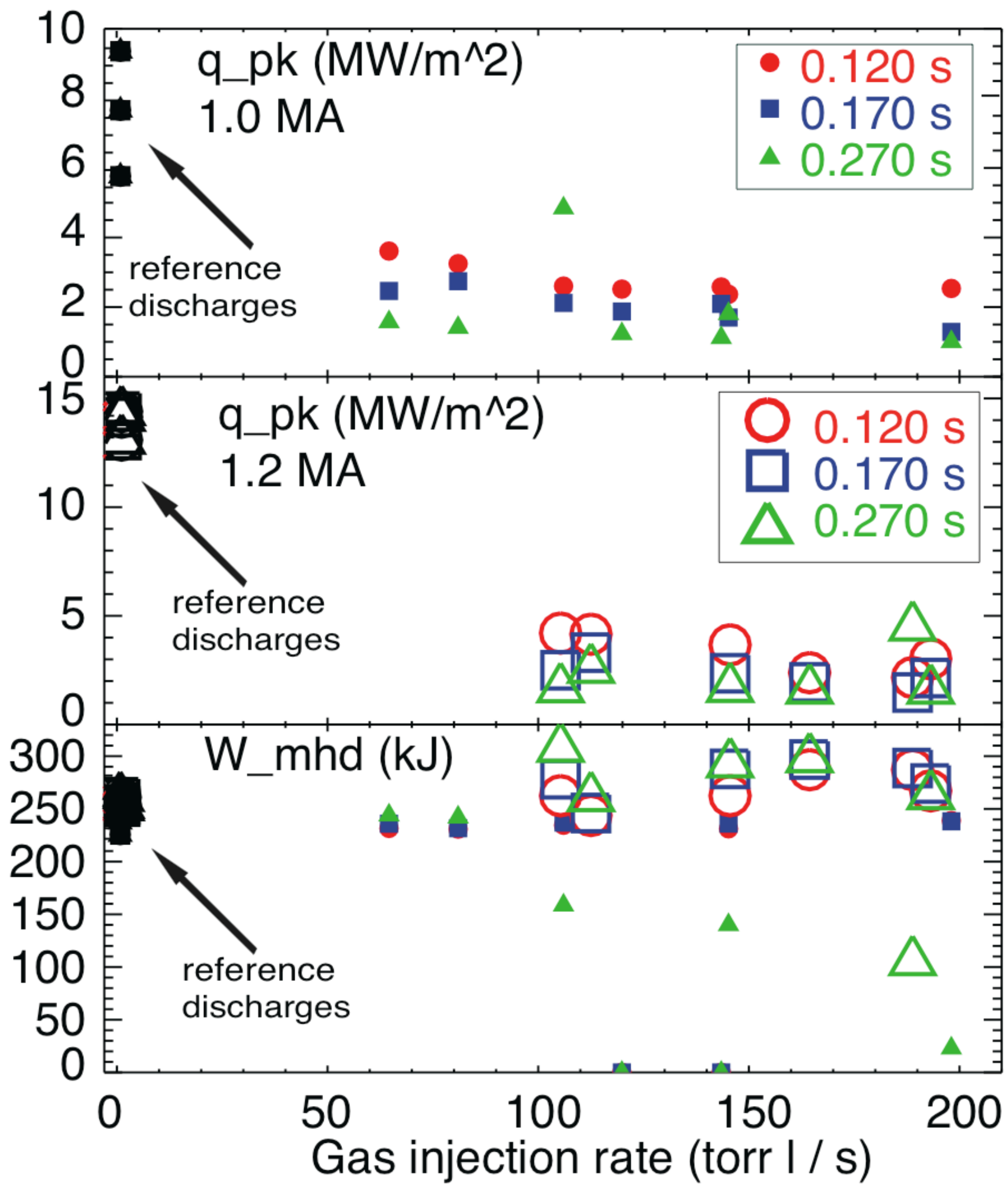




- Experiments conducted in 0.8-1.0 MA 4-6 MW NBI discharges with κ=2.2-2.3, δ=0.6-0.75 (**Soukhanovskii APS 2007)**
- Obtained partially detached divertor (PDD) outer strike point using divertor D_2 injection, however, P_{rad} due to intrinsic carbon and helium
- q_{peak} reduced by 60 - 80 % in PDD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

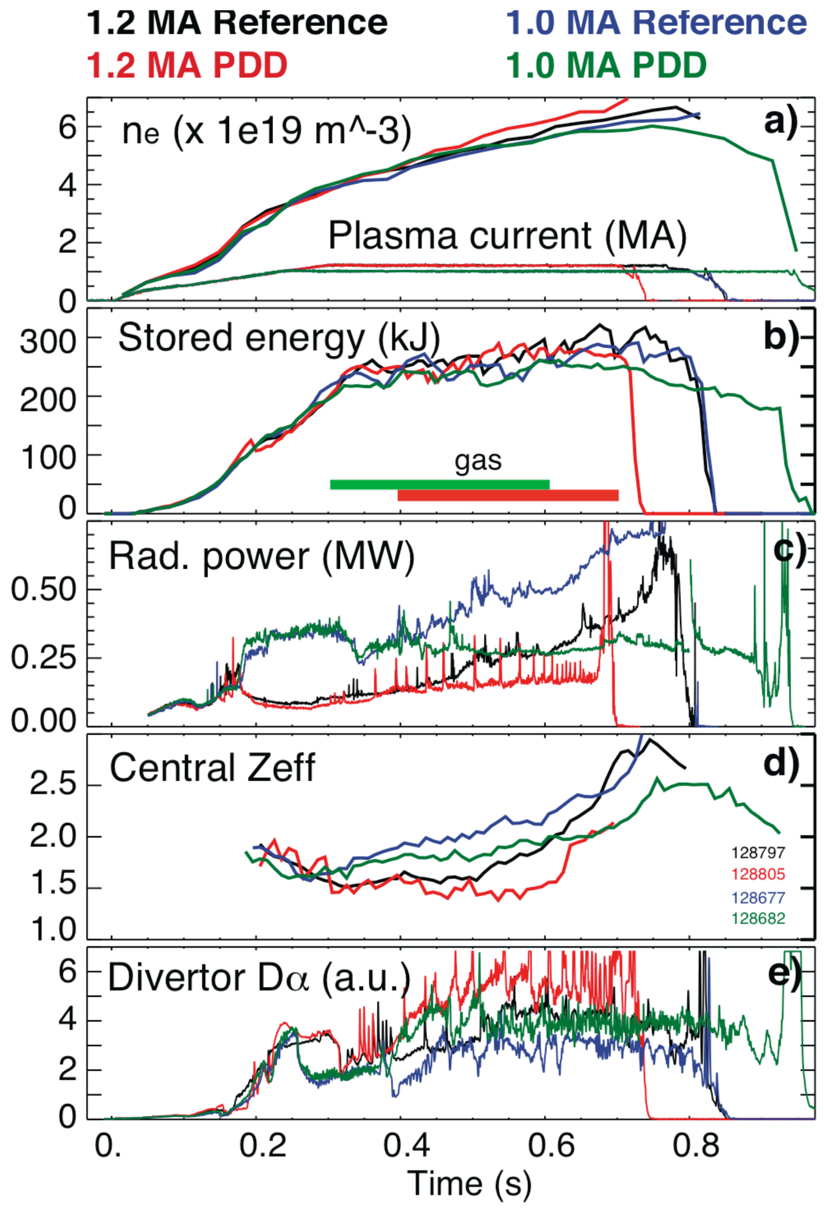
Radiative divertor conditions were optimized in 1.0 MA and 1.2 MA 6 MW H-mode discharges

- Criteria of optimization - find gas injection rate to obtain PDD with minimal confinement degradation
- $q_{||}$ was higher in 1.2 MA discharges thus more gas was needed to reduce q_{pk}
- After 0.250-0.270 ms peak heat flux reached low steady-state level
- Optimal gas injection found (used 300 ms pulses)
    - 50-100 Torr l /s for 1.0 MA discharges
    - 110-160 Torr l /s for 1.2 MA discharges

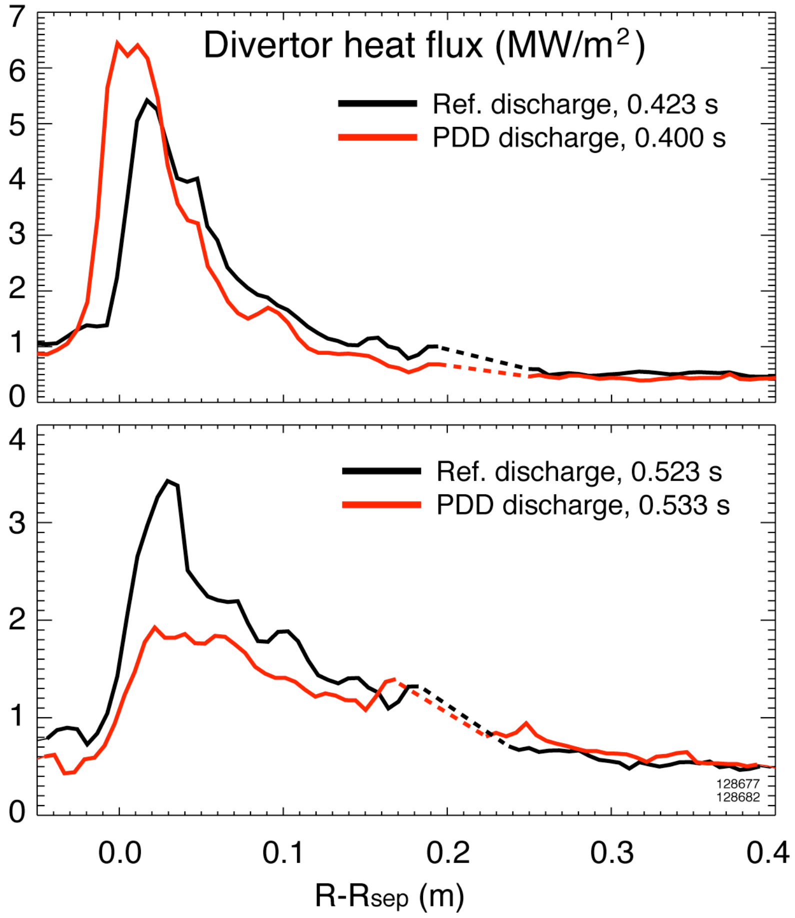


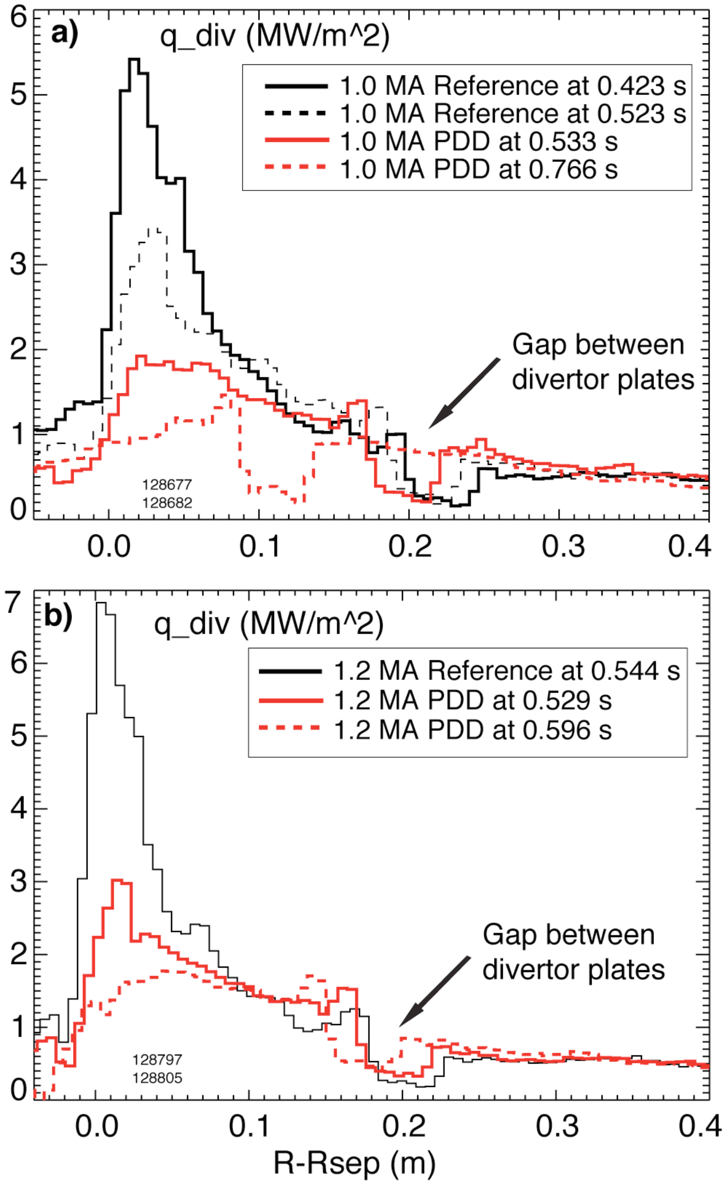
High core and pedestal plasma performance during PDD is achieved in high κ, δ configuation

- These experiments
    - **I_p = 1.0-1.2 MA**
    - P_{NBI} = 6 MW
    - n_e=(0.7-0.8) x n_G
    - D_2 injection in divertor
    - $q_{||}=P_{SOL}/(4\pi R(B_p/B_{tot})\lambda_q)$ = 50-80 MW/m^2
    - **Carbon is main impurity**
- High core plasma performance during PDD phase
    - Minimal effect on W_{MHD} or pedestal
    - Core P_{rad} and n_c decreased
    - Small ELMs ($\Delta W_{MHD}/W_{MHD} \leq 1\%$) and mixed ELMs



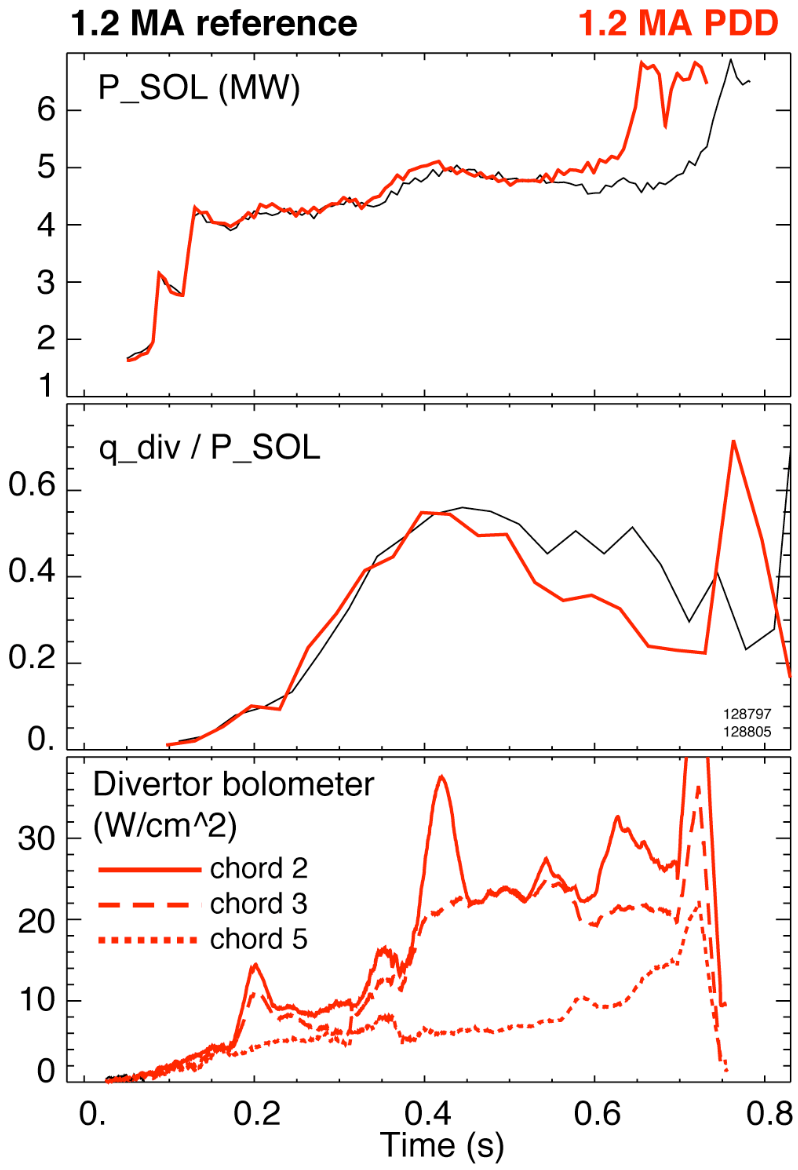
Peak heat flux reduced by up to 60 % as a result of outer strike point partial detachment





- λ_q changed from 5-10 cm to 10-15 cm
- PDD zone 10-15 cm
- No q reduction outside of PDD zone

Divertor heat flux reduction was attributed in part to divertor radiated power loss



- Total P_{SOL} = 4.5 - 5 MW
- $Q_{out.div.}$= 2-3 MW (reference discharge)
- $Q_{out.div.}$= 1-2 MW (PDD discharge)

- Outer leg radiated power estimate:
  - V=0.1 m^3
  - Total P_{rad}=0.5 MW

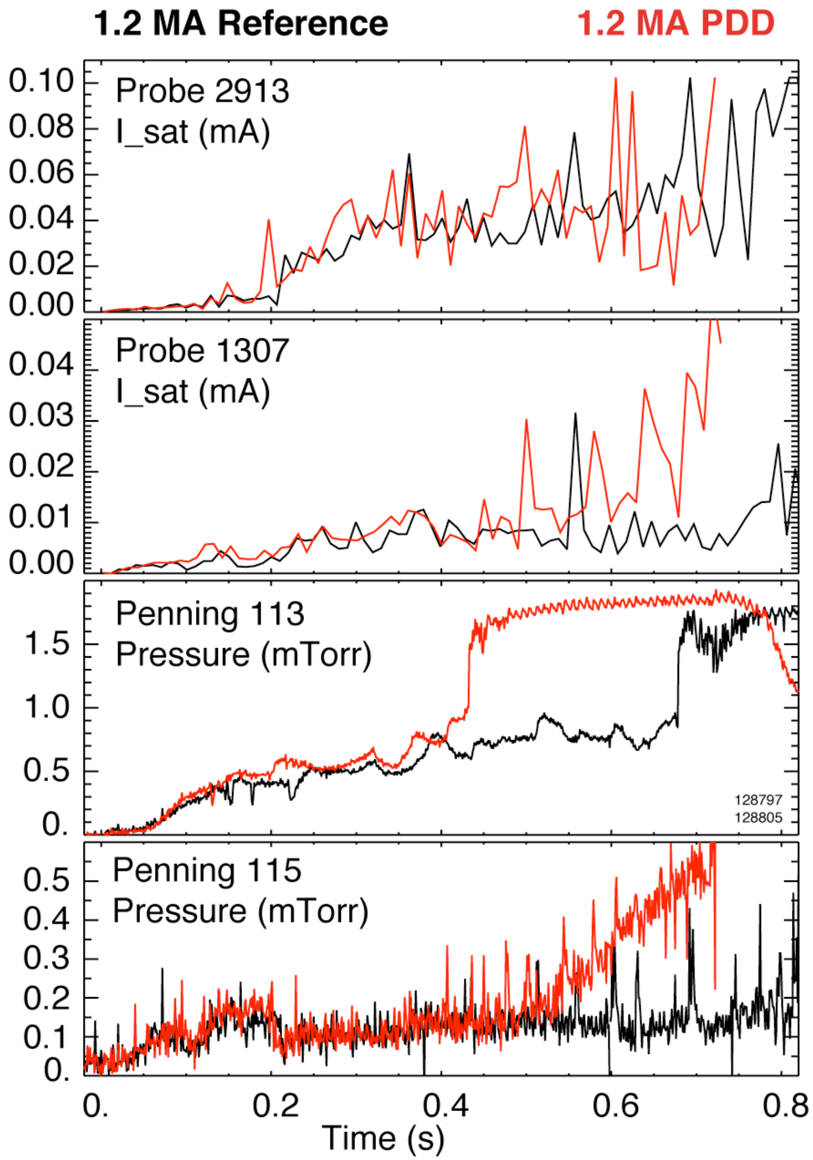
$$\frac{\partial(nv)}{\partial x} = n(n_n\langle\sigma v\rangle_i - n\langle\sigma v\rangle_{rec}) + S_\perp$$

$$\frac{\partial(mnv^2 + 2nT)}{\partial x} = -mnv(n_n\langle\sigma v\rangle_{cx+el} + n\langle\sigma v\rangle_{rec})$$

$$\frac{\partial}{\partial x}\left(-\kappa_0 T^{5/2} + \frac{1}{2}mnv^3 + 5nTv\right) =$$

$$-n^2 f_Z L_Z - \frac{3}{2}Tnn_n\langle\sigma v\rangle_{cx+el} - nE_{ion}\langle\sigma v\rangle_i + Q_\perp$$

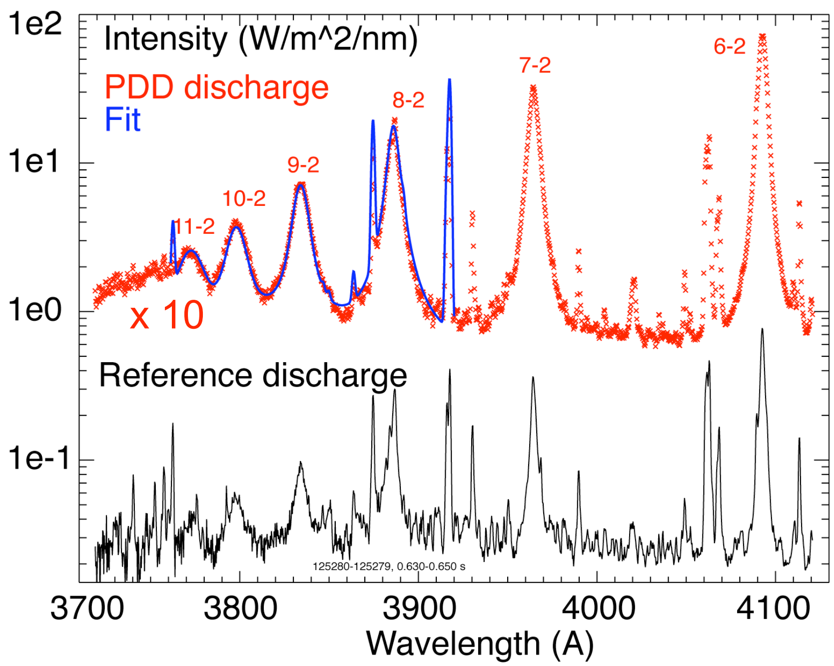
Momentum loss was evidenced by divertor neutral pressure increase and particle flux decrease

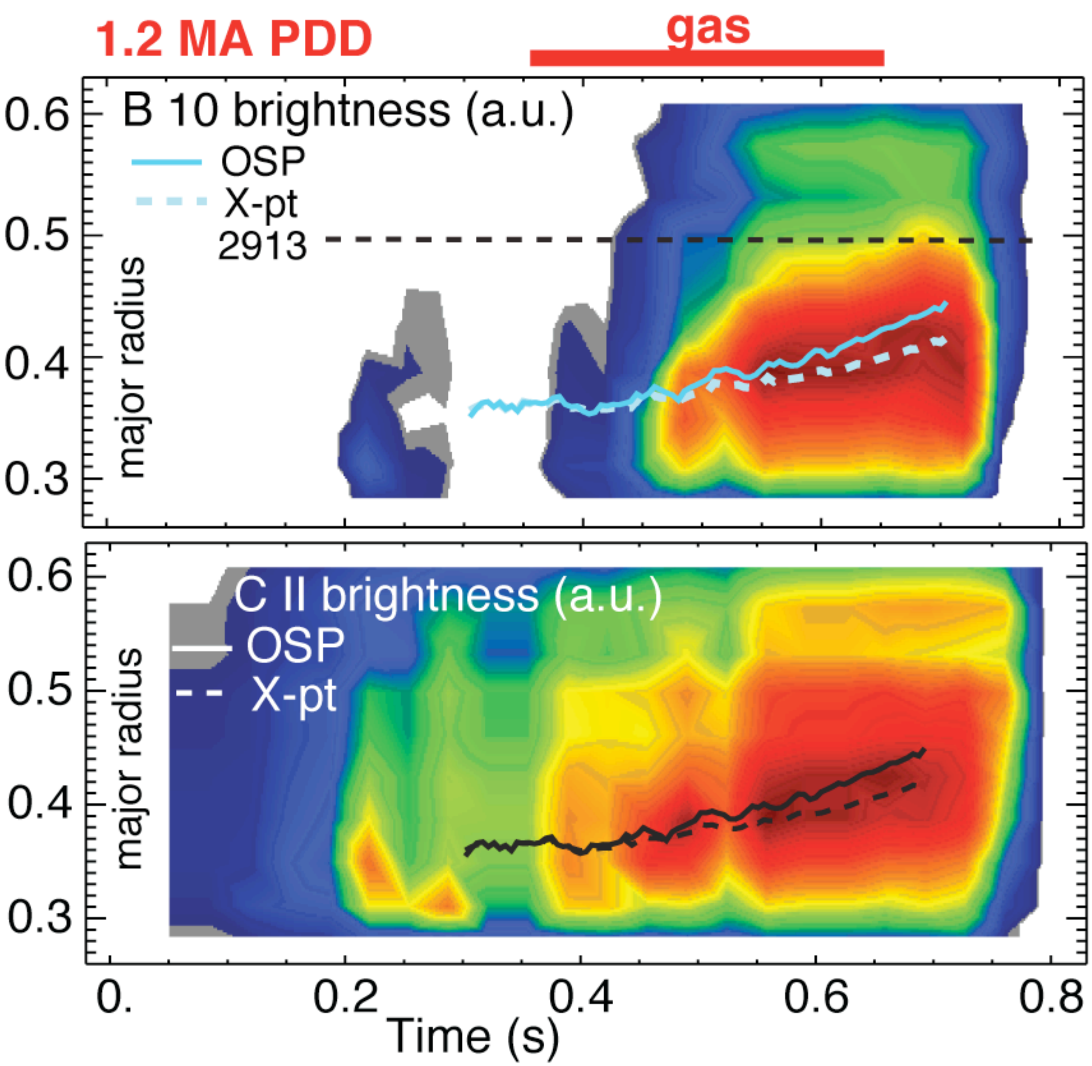


- Langmuir probe in PDD zone showed particle flux decrease during gas puff
- Langmuir probe outside of PDD zone showed particle flux increase during gas puff – as expected in high-recy. regime
- Neutral pressure increased in outer div. region from 0.5 to 2-3 mTorr
- Neutral pressure of 2-3 mTorr is required to explain plasma pressure drop of dp/dx = 9-10 Pa/m

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

Carbon radiation and ion recombination rates increased in divertor detachment phase





- Increase in recombination rate
- D I Balmer spectra (8…11 - 2) indicate
  - T_e < 0.7-1.2 eV (from line intensity ratio according to Saha-Boltzman formula)
  - $n_e \sim$ 2-6 x 10^{20} m^{-3} (from Stark broadening and MMM calculations)

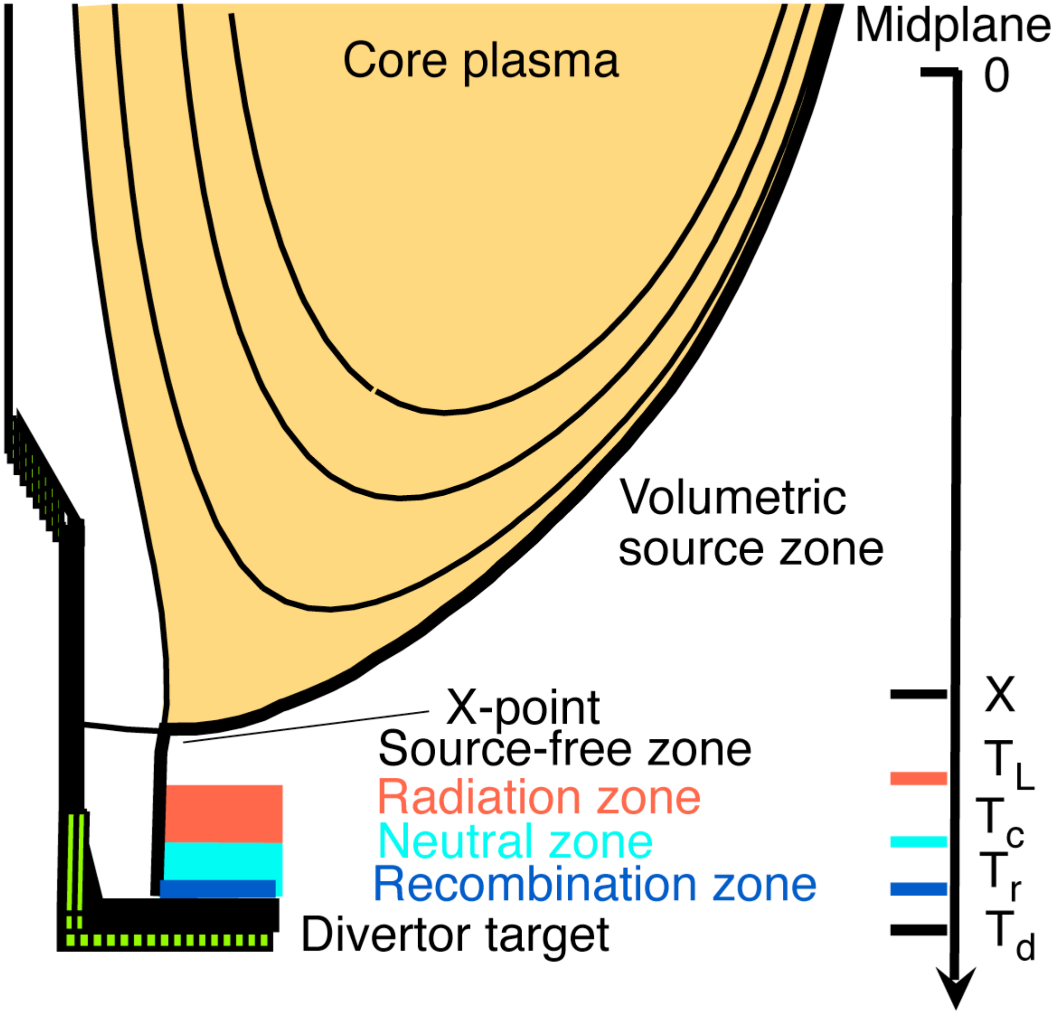
Six-zone 1D analytic SOL / divertor model captures essential features of detachment

- **Goswami PoP 8, 857 (2001)**
- Zone locations defined by temperature of process
- Sources and sinks $Q_\perp$, $S_\perp$, $\Gamma_{i\text{-}div}$, f_{rad}, R_{rec}, $\nu_{i\text{-}n}$ as input

$$\text{Continuity:} \quad \frac{d}{dx}(nv) = S_n$$

$$\text{Momentum:} \quad \frac{dp}{dx} = S_p$$

$$\text{Energy:} \quad \frac{d}{dx}\left(\kappa_0 \, T_e^{5/2} \, \frac{dT_e}{dx}\right) = S_q$$

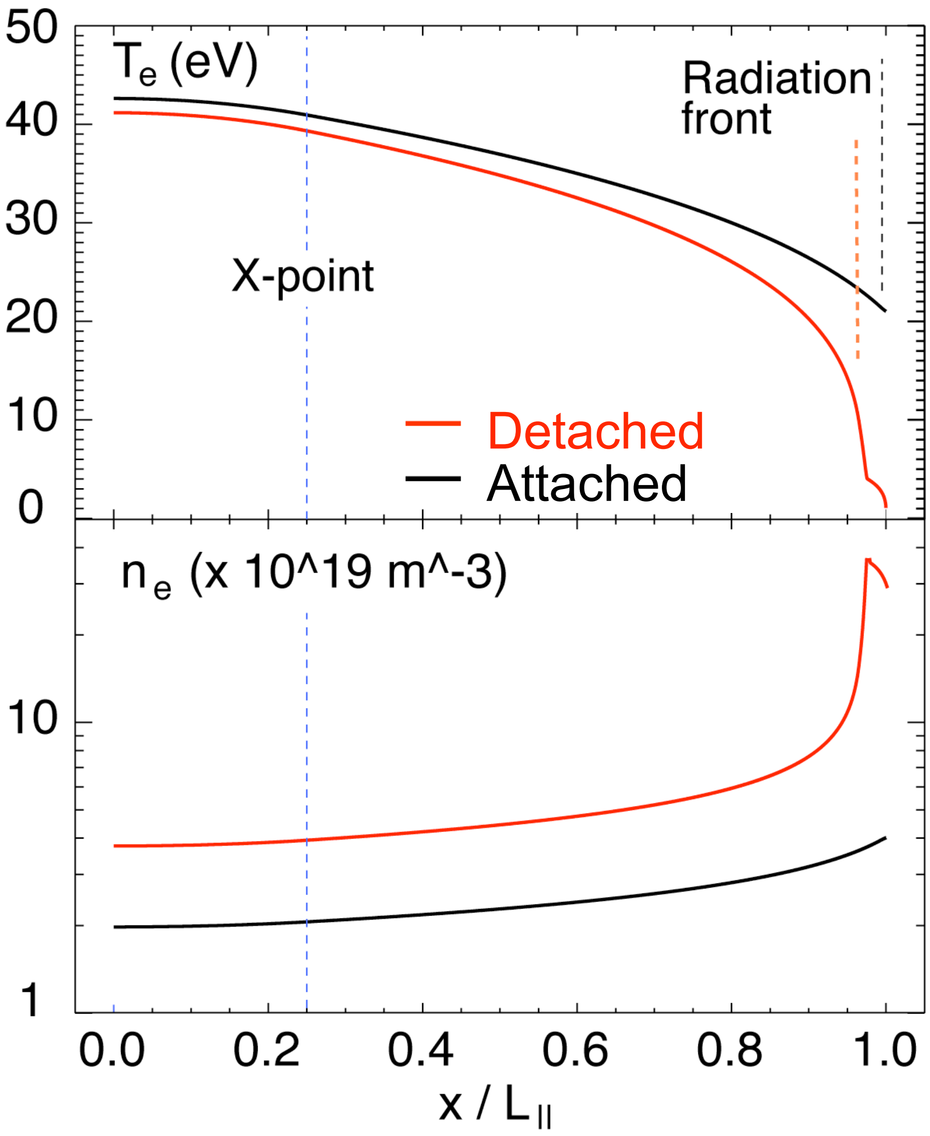


Model predictions consistent with experiment within NSTX range of SOL parameters

- NSTX SOL / divertor parameters
  - $Q_{\perp}$ = 0.5 - 20 MW m^{-3} (high)
  - $S_{\perp}$ = 0.01-3 $\times 10^{23}$ s^{-1} m^{-3}
  - L_x = 5-10 m (low)
  - R_{rec} = 10^{23} s^{-1} m^{-3}

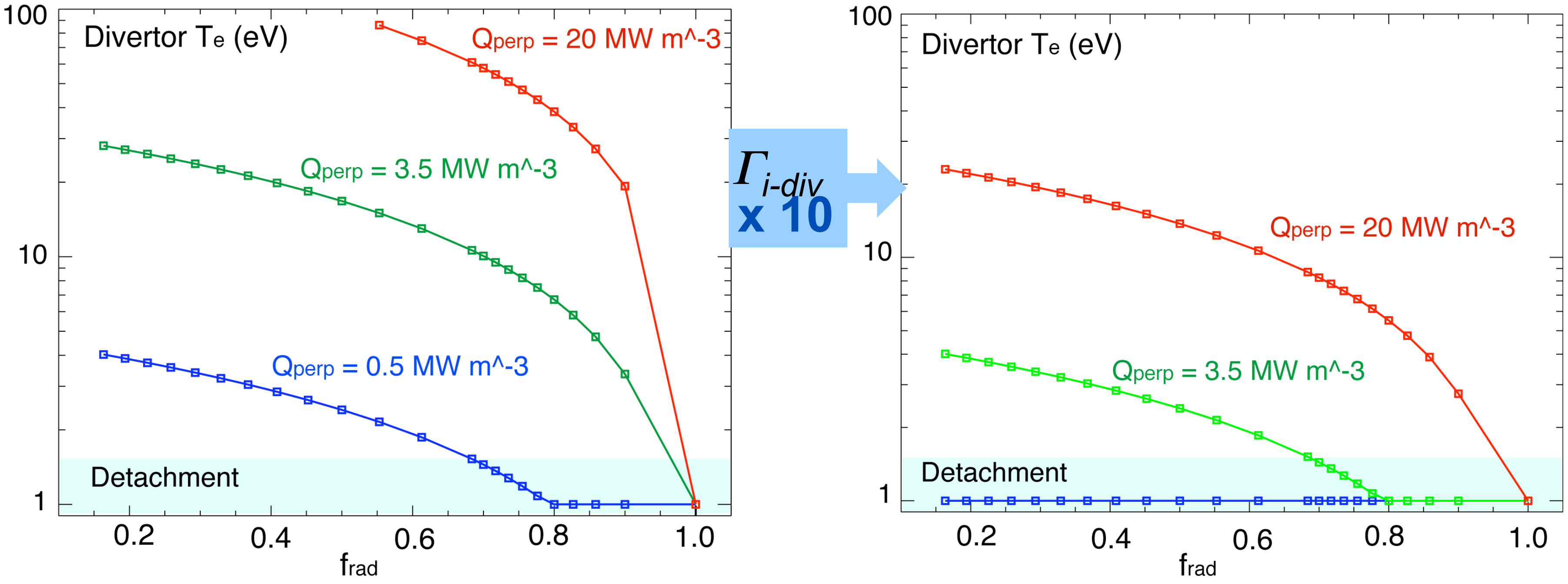
- Example calculation
  - $Q_{\perp}$ = 10 MW m^{-3}
  - $S_{\perp}$ = 6 x 10^{22} s^{-1} m^{-3}
  - f_{rad} = 0.3 (attached)
  - f_{rad} = 0.9 (detached)

- Recombination onset at $T_e < 1.5$ eV
- Detachment at $T_e < 1.0$ eV



All routes to detachment predicted by model involve high f_{rad}

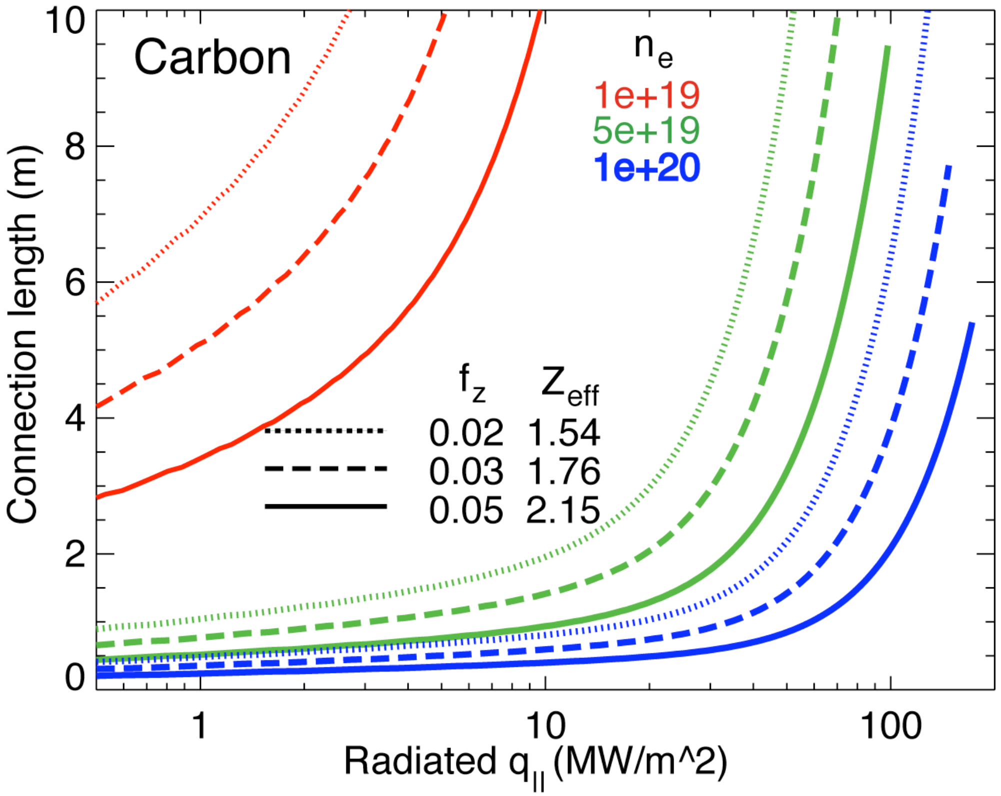
- Detachment at NSTX-range of $Q_{\perp}$, $S_{\perp}$ can be achieved in model by
  - increasing f_{rad} (shown)
  - increasing $\Gamma_{i\text{-}div}$ (gas puff)
  - increasing $S_{\perp}$ (not shown)



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



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

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

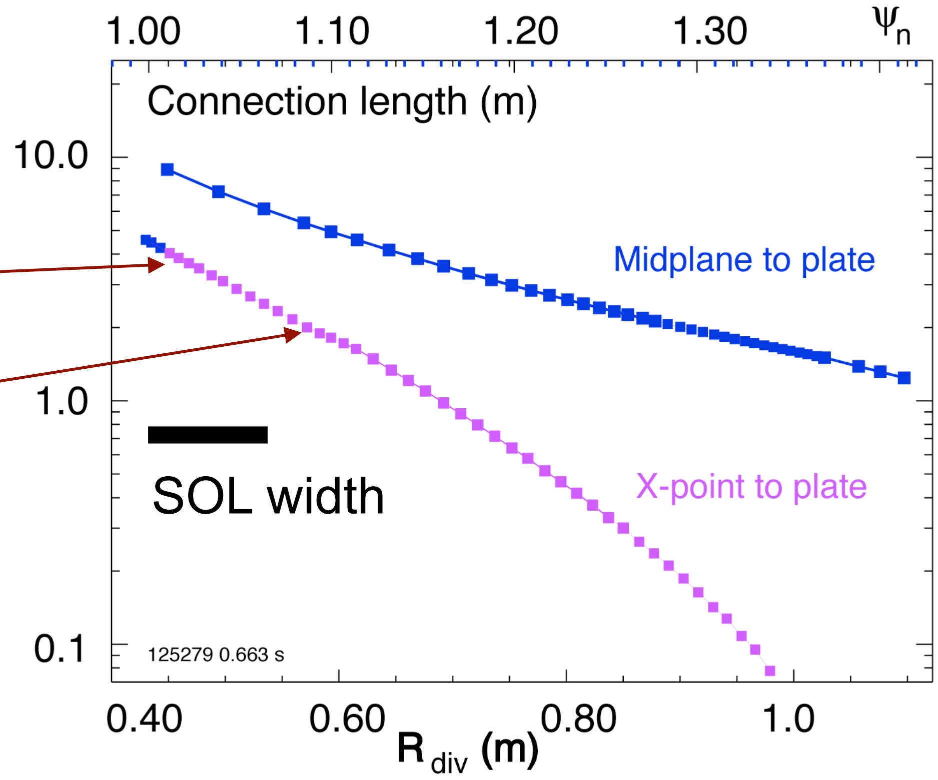
Volumetric power and momentum losses are limited by L_x (R) at high magnetic field shear

- Fraction of $q_{||}$ to be radiated is a function of L_x for given impurity
  - high f_{rad} only where L_x longest

- 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

possible to detach
L_x ~ *5-6 m*

difficult (impossible)
to detach L_x ~ *1-2 m*



Discussion

- PDD regime with reduced q_{pk} and good core confinement demonstrated in open geometry un-pumped divertor in a high power spherical torus

- In an ST, modest $q_{||}$ can yield high divertor q_{pk}
  - in NSTX, $q_{||}$= 50-80 MW/m^2 and q_{pk}=6-12 MW/m^2
  - Large radiated power and momentum losses are needed to reduce $q_{||}$

- In NSTX density ramp discharges do not necessarily lead to PDD
  - n_{sep} weakly coupled with n-bar

- In NSTX, PDD regime is accessible only
  - in highly-shaped plasma configuration with high flux expansion divertor (high plasma plugging efficiency, reduced $q_{||})$
  - modest divertor D_2 injection still needed

- ST SOL geometric effects appear to play dominant role in the above