NSTX

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Tools For Developing Advanced Spherical Tokamak Plasmas in NSTX

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Innovative Confinement Concepts 2010
LSB Auditorium, PPPL
Feb. 19th, 2010

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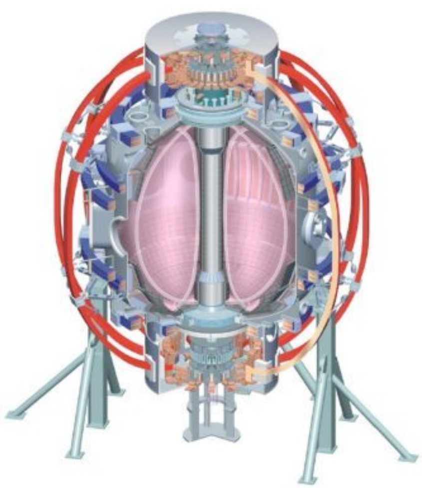

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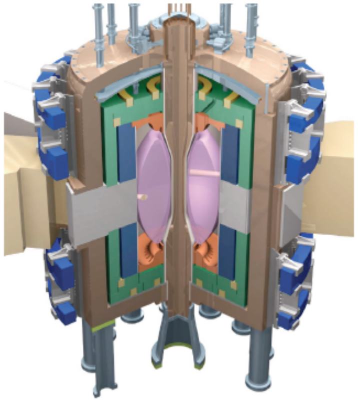
U Quebec

Scenario Development Research in NSTX Focused on Needs of Next-Step Devices

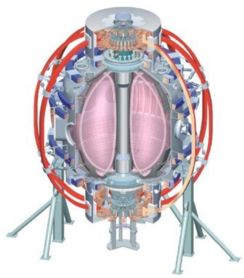
- Next-step STs designed to provide important engineering and physics knowledge for fusion energy:
  - ST Fusion Nuclear Science Facility
    - Develop fusion nuclear science.
    - Test nuclear components for Demo
    - Sustain $W_{neutron}$ ~ 0.2-0.4 → 1-2MW/m^2, τ_{pulse} = 10^3 →10^{6}s
  - ST Plasma Material Interface Facility
    - Develop long-pulse PMI solutions for FNSF / Demo.
    - High P_{heat}/S ~1MW/m^2, high T_{wall}, τ_{pulse} ~ 10^{3}s
- NSTX scenario develop research
  - Maximize the non-inductive current fraction.
  - Study the stability, transport, and overall performance, of plasmas with largely non-inductive current drive.
  - Develop an understanding of the control tools needed to achieve these configurations.

***FNSF** (ST-CTF)*

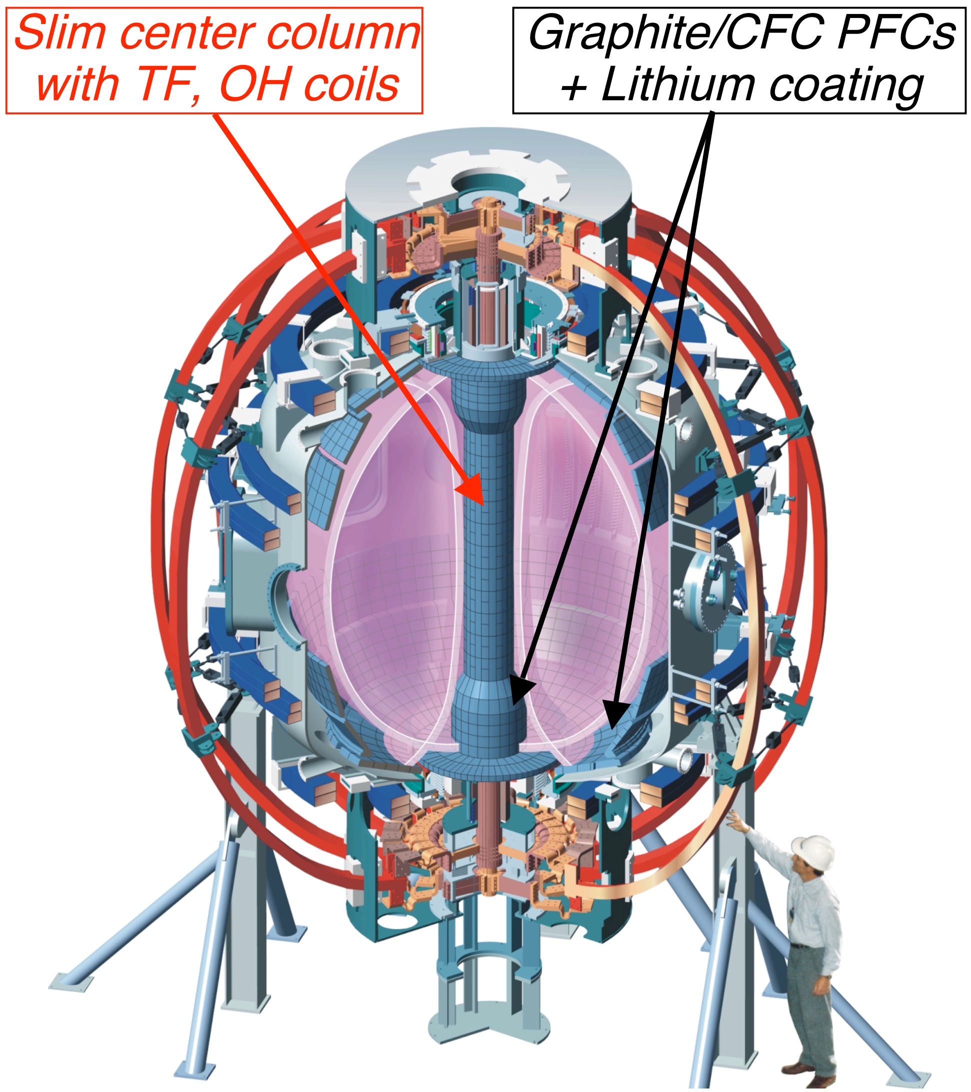


***PMIF** (NHTX)*

NSTX



NSTX Designed to Study High-Temperature Toroidal Plasmas at Low Aspect-Ratio



Aspect ratio A	1.27 – 1.6
Elongation κ	1.8 – 3.0
Triangularity δ	0.2 – 0.8
Toroidal Field B_{T0}	0.4 – 0.55 T
Plasma Current I_p	1.5MA
Auxiliary heating:	
NBI (100kV)	7 MW
RF (30MHz)	6 MW
Central temperature	1 – 5 keV
Central density	$\leq 1.2\times10^{20} m^{-3}$

Outline

- Next-step STs and NSTX.
- Example of optimized discharges in NSTX.
- Important operational and analysis tools.
- Means to increase the non-inductive fraction:
  - Near-term: Liquid Lithium Divertor
  - Long-term: NSTX-Upgrade

Best NSTX Discharges Achieve CTF-level β_N, with Good Confinement and High Non-Inductive Fraction

133994

Optimized for largest non-inductive fraction

B_T=0.48 T

135445

Optimized for longest pulse

B_T=0.375 T

Tools for Integrated Performance

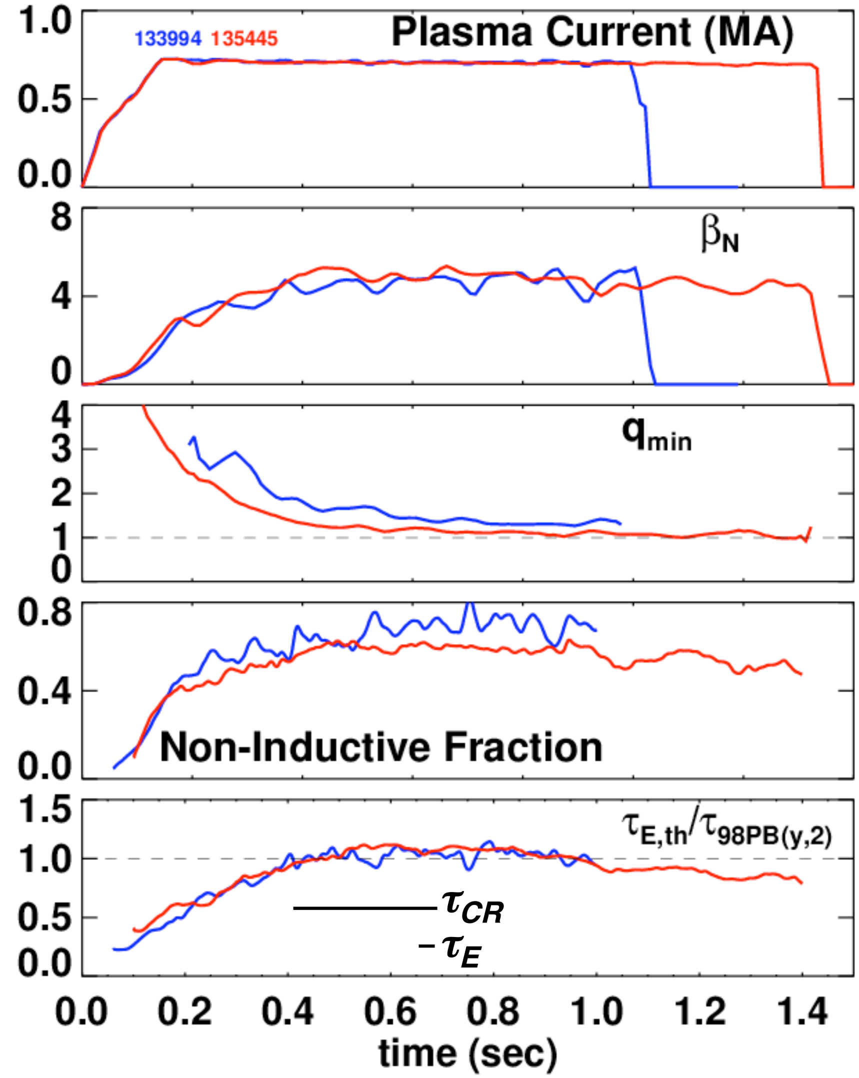
Plasma Shaping

n=1 Mode Control

Lithium Conditioning

Current Profile Analysis

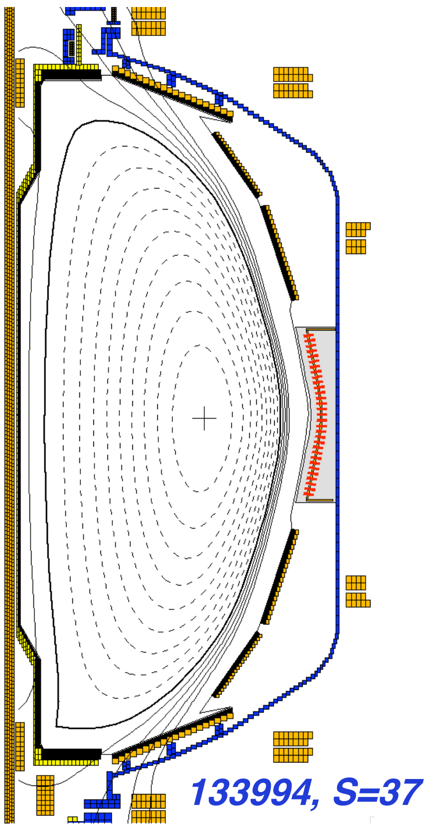
Many more…



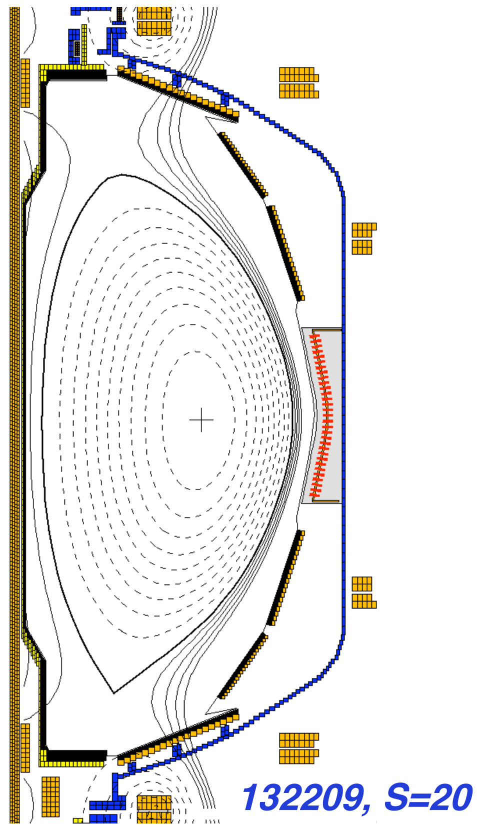
Shaping is Described by the "Shape Parameter" S

- Freedom in plasma boundary shaping[1]
  - Elongation: Increasing elongation increases q, improving the kink stability & increasing bootstrap currents.
  - Triangularity: Increasing the triangularity causes fields lines to spend more time in the good-curvature region.
- "Shape Parameter" S encapsulates both effects.

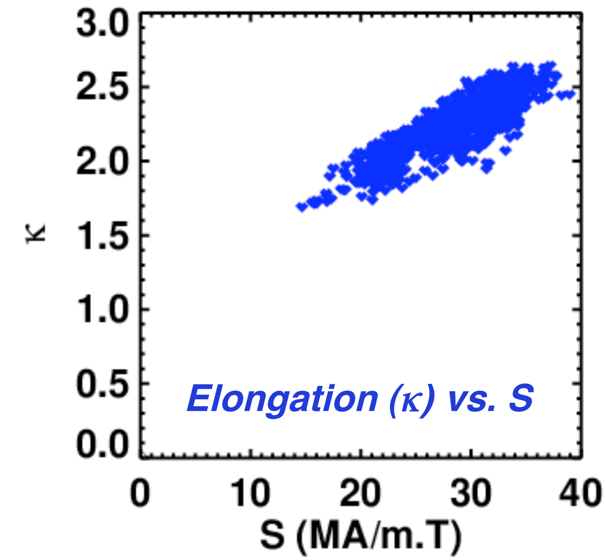
$$S = \frac{q_{95} I_P}{a B_T}$$

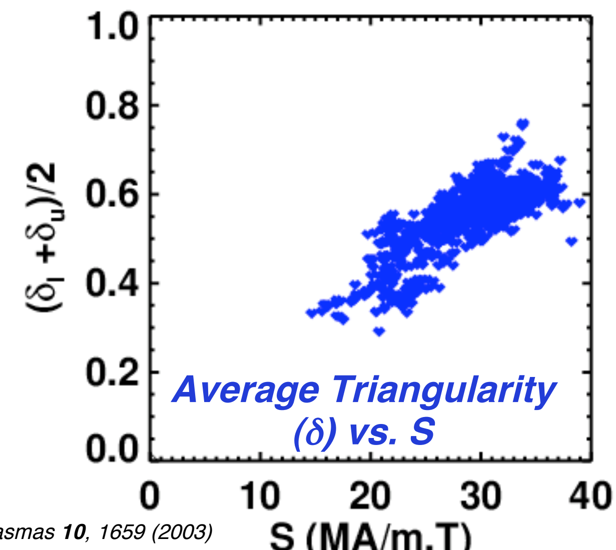


133994, S=37



132209, S=20





[1] D. Gates, et al, Phys. Plasmas **10**, 1659 (2003)
[2] D. Gates, et al, Phys. Plasmas **10**, 1659 (2003)

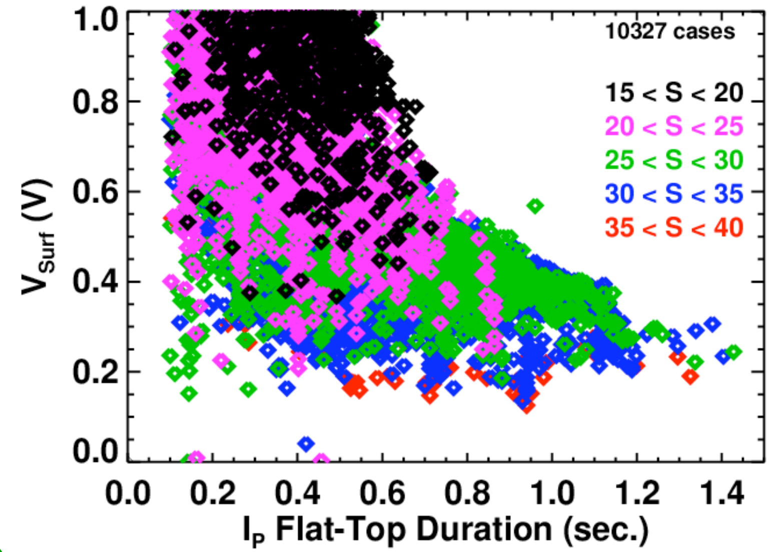
Strong Plasma Shaping is Important For Sustained High-β

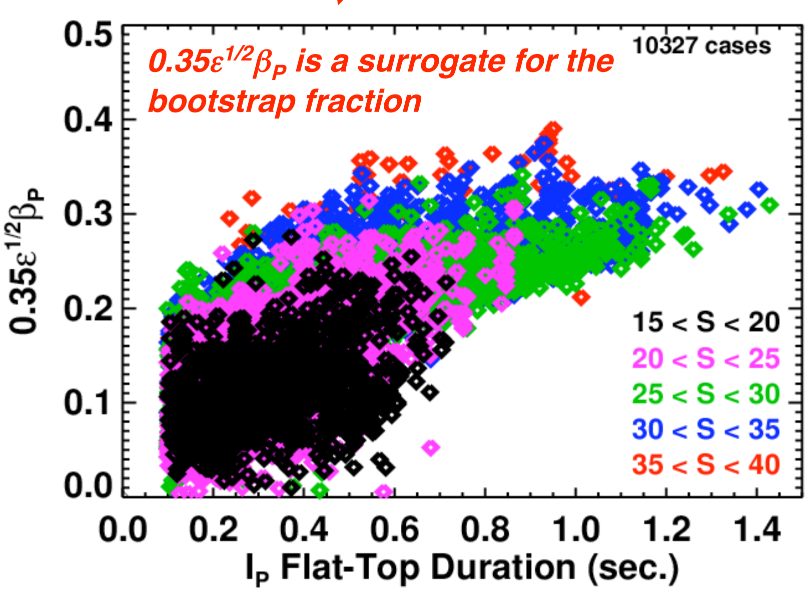
Large values of shaping help achieve:

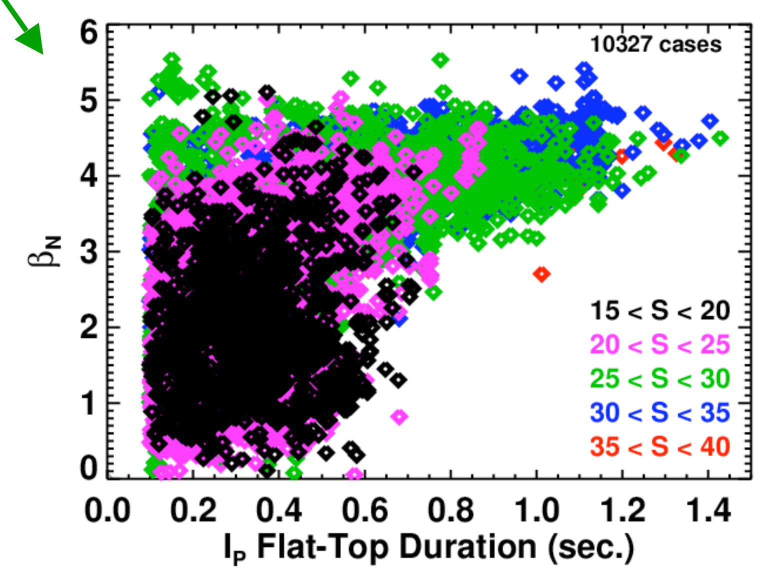
Low surface voltage

Long-pulse at high-β_N

High bootstrap fraction

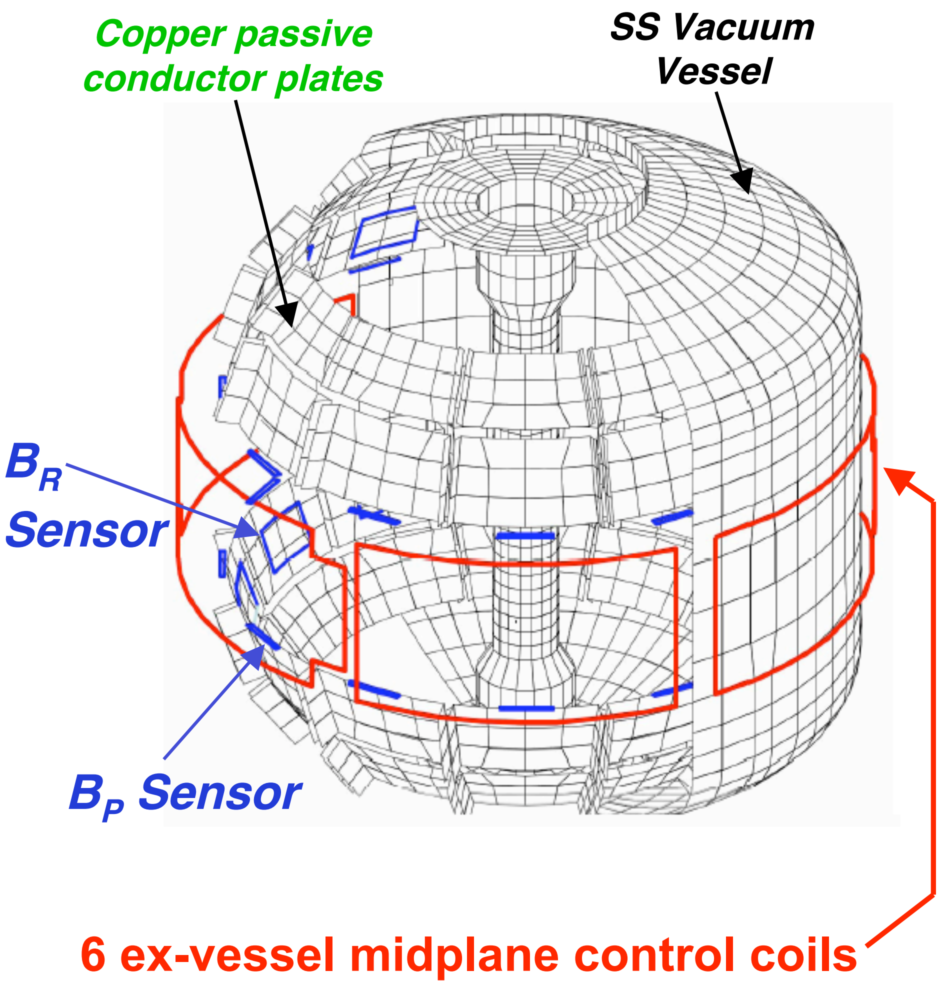






*[1] D. Gates, et al, Phys. Plasmas **10**, 1659 (2003), [2] D. Gates, et al, Phys. Plasmas **10**, 1659 (2003)*

n=1 Mode Control Provided with Internal Sensors and External Midplane Coils



- Copper stabilizing plates to enable high-β operation.
- 6 ex-vessel midplane coils.
- 48 Internal sensors for non-axisymmetric fields.
  - 24 B_R for perturbations.
  - 24 B_P for perturbations.
- Use internal sensors to reconstruct an n=1 amplitude (B_1) and phase (θ_1) at each time.

$$B_{RWM}(\phi) = B_1 \cos(\phi - \theta_1)$$

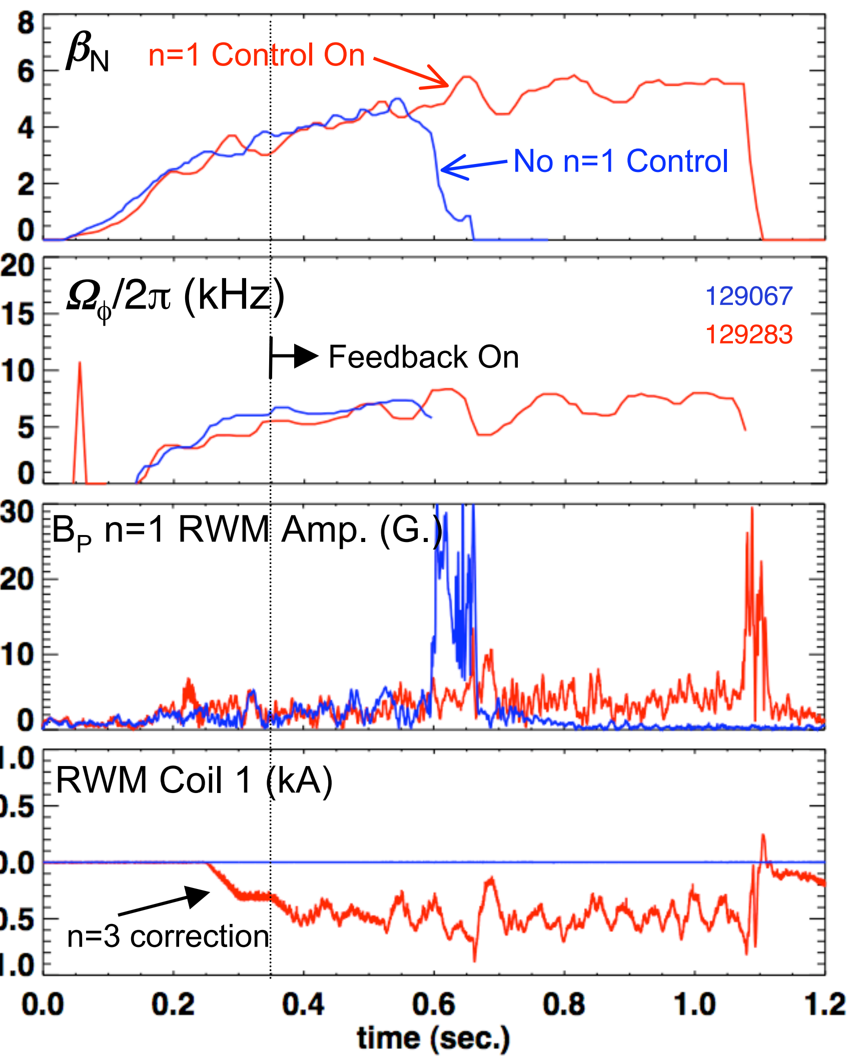
- Apply a phase shifted n=1 field.
  - Feedback Gain G
  - Feedback Phase δ

$$B_{F.B.}(\phi) = GB_1 \cos(\phi - \theta_1 - \delta)$$

Resistive Wall Mode (RWM) Feedback: Rapidly varying applied field.
Dynamic Error Field Correction (DEFC): Slowly varying applied field.

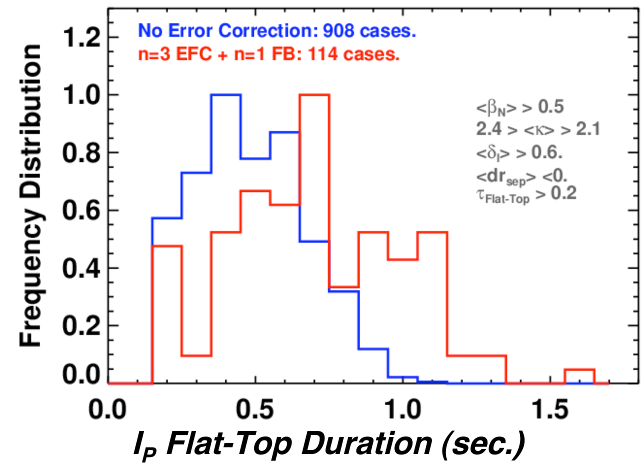
n=1 Mode Control Enables Reliable Access to Higher β

Comparison with & without n=1 control

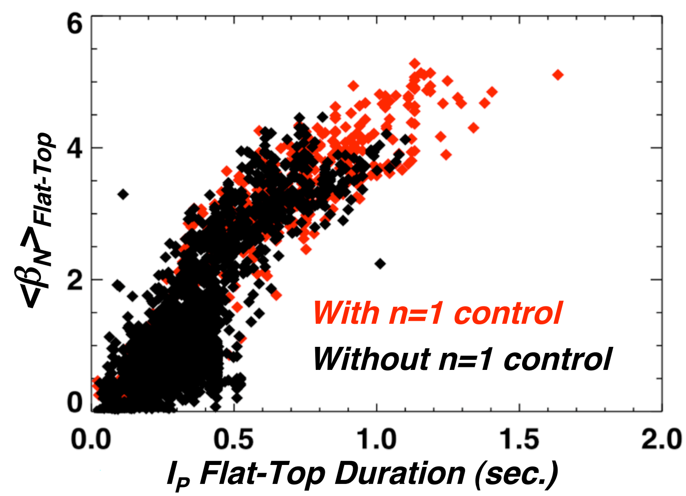


[1] S. Sabbagh, et al, Phys. Rev. Lett **97**, 045004 (2006)   [2] S. Sabbagh et al, Nucl. Fusion **50**, 025020 (2010)

Probability of a Long-Pulse Discharge Increases with n=1 Control

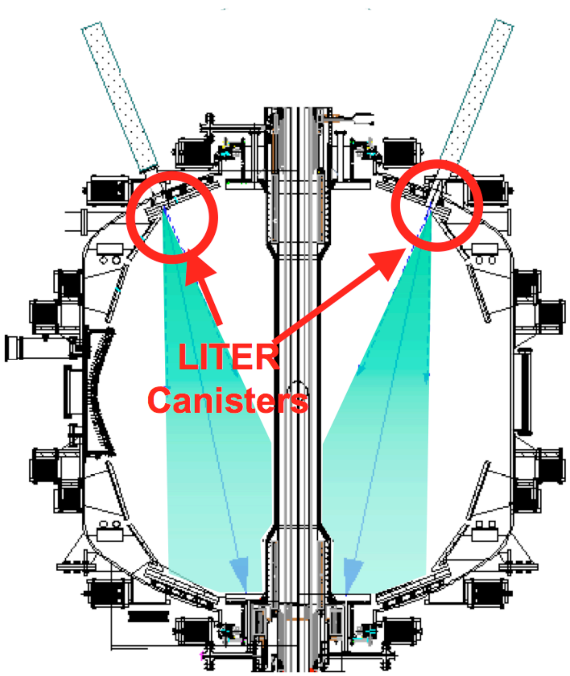


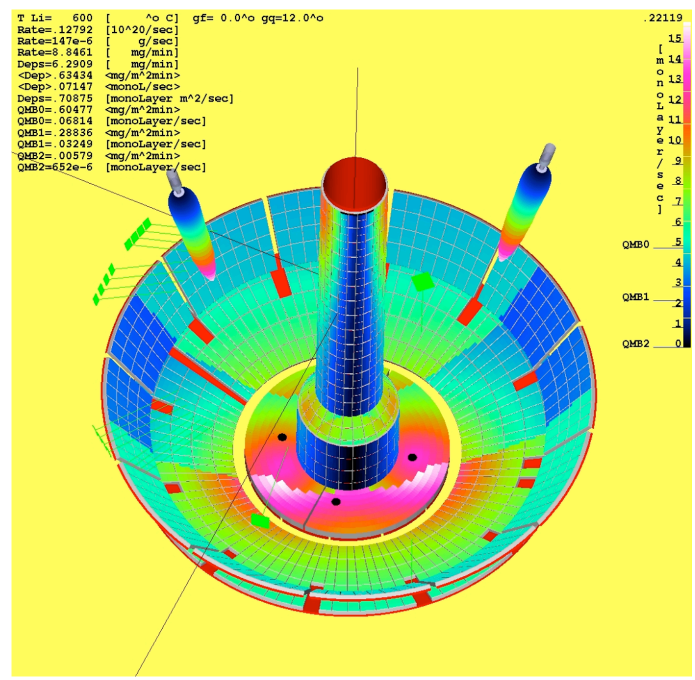
Long-Pulse High-β_N Facilitated by n=1 Control



Lithium Conditioning Provided by Dual Lithium Evaporators

- Two evaporators, separated by ~150º toroidal, deposit solid lithium on graphite PFCs
  - LITER=LIThium EvaporatoR
- Typically deposit 50-300 mg of lithium between discharges.
  - In-situ QMB data implies deposited lithium thickness is 5 – 160 nm on inner divertor plate.
- Need 40-60% more gas with Li conditioning to match density evolution.
- Eliminated the need for both helium GDC between discharges and bi-weekly boronization.
  - Also increases shot-to-shot reproducibility and reliability.

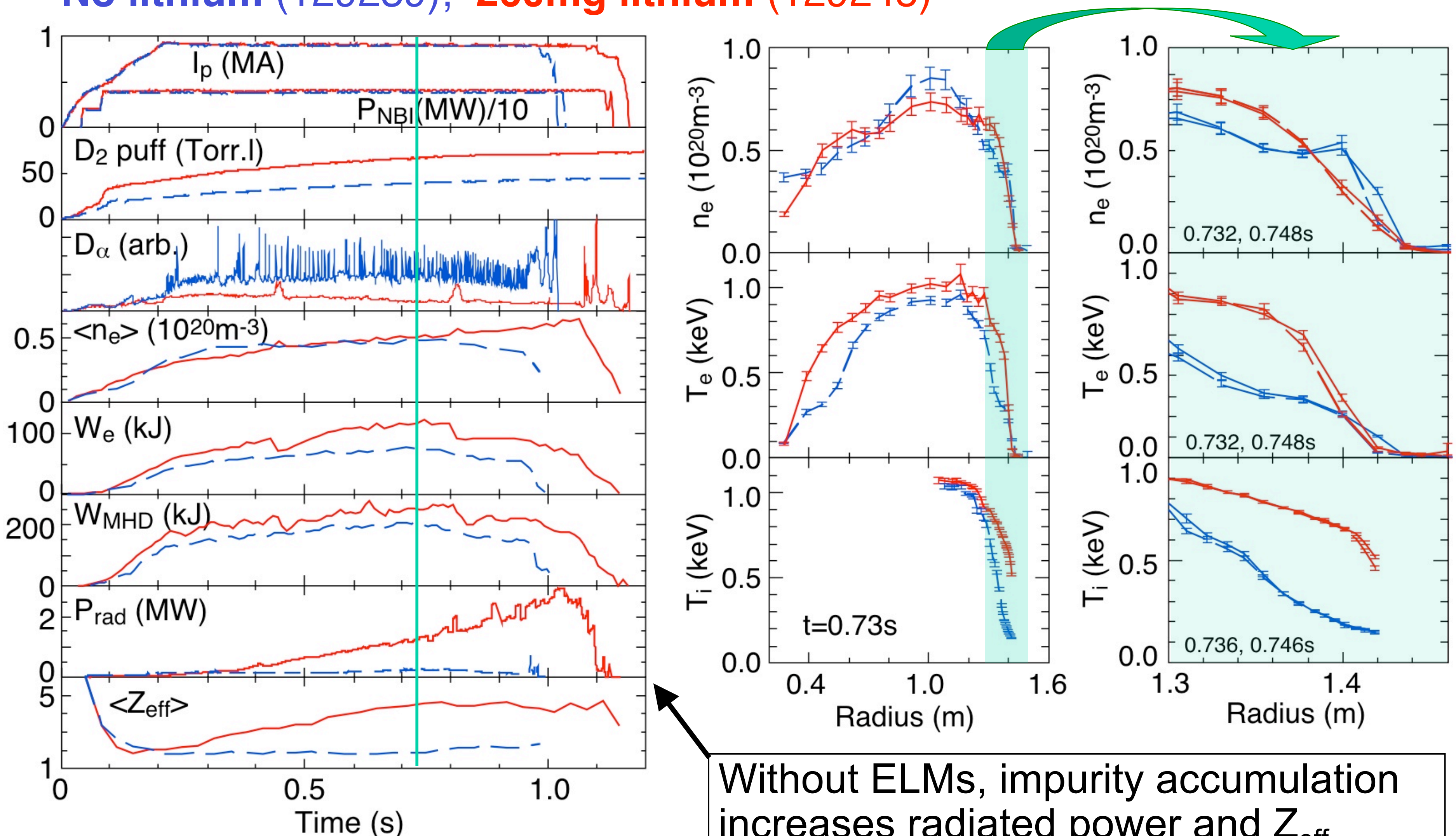




[1] H. Kugel Phys. Plasmas ***15****, 056118 (2008)*   *[2] M. Bell et al, Plasma Phys Control Fusion* ***51****, 124054 (2009)*

Lithium Coating Reduces Deuterium Recycling, Suppresses ELMs, Improves Confinement

No lithium (129239); **260mg lithium** (129245)

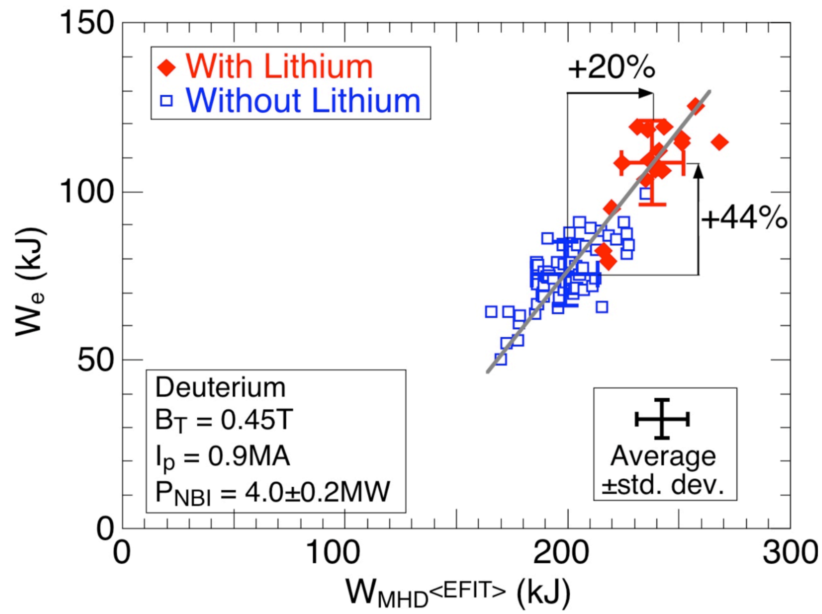


Without ELMs, impurity accumulation increases radiated power and Z_{eff}

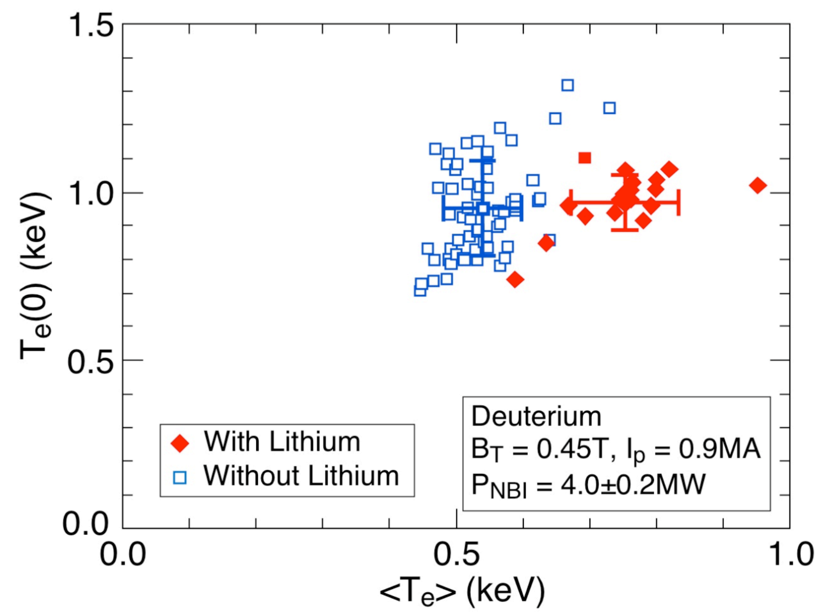
[1] M. Bell et al, Plasma Phys Control Fusion **51**, 124054 (2009) [2] R. Maingi, et al, Phys. Rev. Lett. **103**, 075001 (2009)

Confinement Improves, and Temperature Profiles Broaden, With Li Conditioning of the PFCs

Both Total and Electron Stored Energy Increase with Lithium Conditioning[1]



Electron Temperature Profile Broadens with Lithium[1]

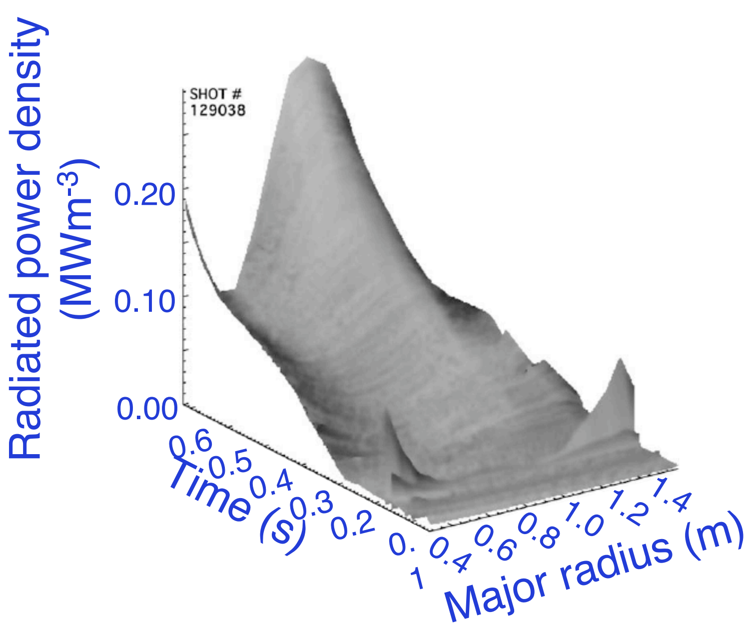


- TRANSP analysis confirms electron thermal transport in outer region progressively reduced by lithium.[2]
- Root cause of confinement improvement with lithium is not understood.
  - Ions remain approximately neoclassical.
  - Electron transport in NB-heated H-mode ST plasmas is not understood.

*[1] M. Bell et al, Plasma Phys Control Fusion **51**, 124054 (2009), [2] S. Ding, Plasma Phys. Control Fusion **52**, 015001 (2010)*

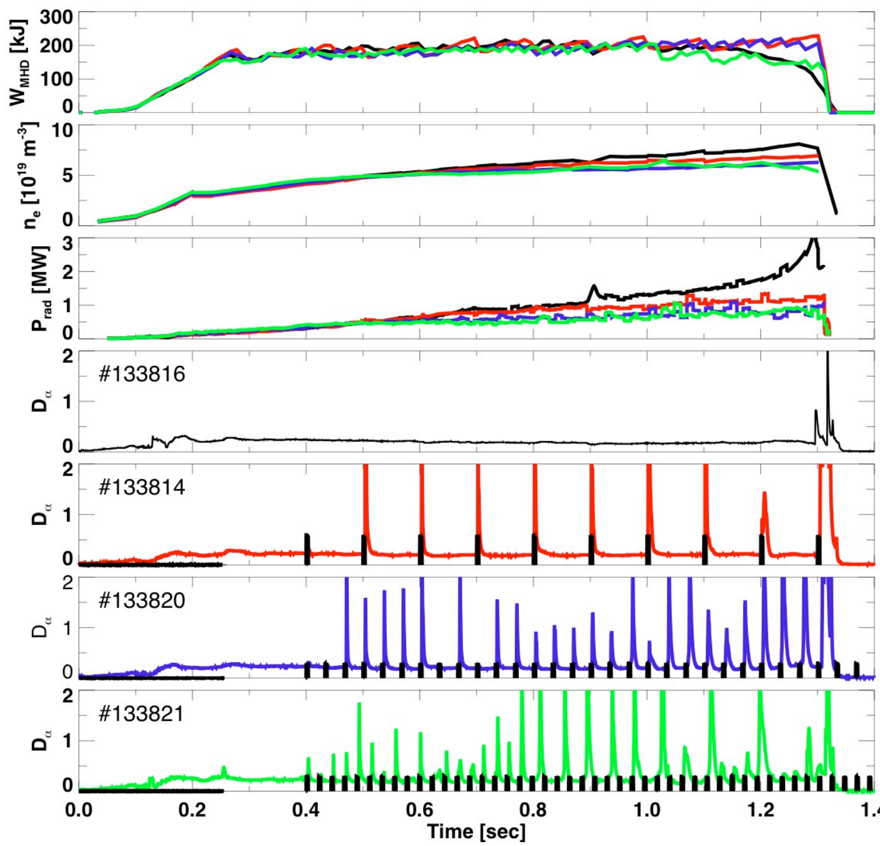
Impurity Accumulation in ELM-Free H-Mode Can Be Arrested With Triggered ELMs.

- ELMs are eliminated with lithium conditioning.
  - Modifications to the edge profiles results in modifications to the peeling-ballooning stability boundary.[2]
- Core radiation grows to unacceptable levels.[1]
- Magnitude of apparent metals concentration depends on plasma current
  - Consistent with sputtering from lost fast-ions being an important impurity source.



- Use 3-D field (n=3) pulses to reintroduce ELMs and reduce radiated power.
  - Short duration (~3 msec)
  - Large amplitude (~2.5kA)
  - Reliable ELM triggering from 10-70 Hz

Double-null, κ=2.4, δ=0.8, 0.8MA, 0.45T, NBI 4 MW



[1] M. Bell et al, Plasma Phys Control Fusion **51**, 124054 (2009), [2] R. Maingi, et al, Phys. Rev. Lett. **103**, 075001 (2009), [3] J.M. Canik Phys. Rev. Lett **104**, 045001 (2010)

Current Profile Analysis Shows That Present Configurations Are Limited to f_{NI}<70%

- ***Separate calculations of each current profile constituent in NSTX***
  - **Inductive Currents:** Electric field from time derivates of equilibria, and neoclassical resistivity[1]

$$\langle J_{OH}\cdot B\rangle = \sigma_{Neo}\langle E\cdot B\rangle$$

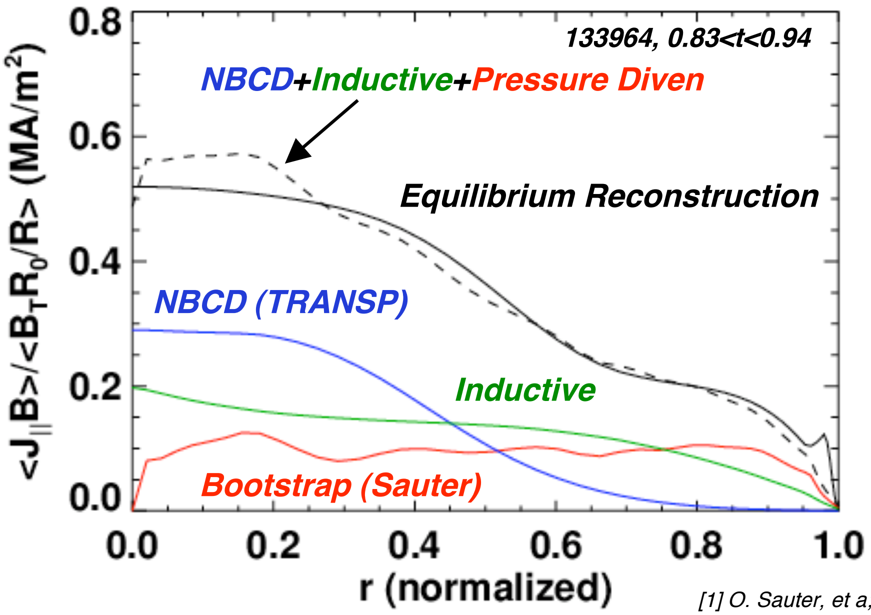
  - **Bootstrap Currents:** Calculate using either the NCLASS model in TRANSP, or the Sauter model.[1]

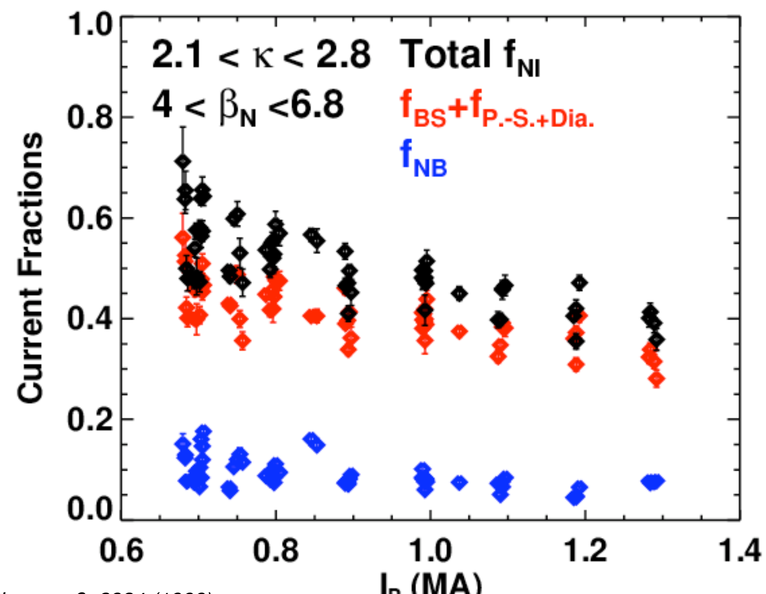
$$\langle J_{BS}\cdot B\rangle = (RB_\phi)p_e\left[L_{31}\frac{1}{p_e}\left(\frac{\partial p_e}{\partial\psi}+\frac{\partial p_i}{\partial\psi}\right)+L_{32}\frac{1}{T_e}\frac{\partial T_e}{\partial\psi}+L_{34}\alpha\frac{1-R_{pe}}{R_{pe}}\frac{1}{T_i}\frac{\partial T_i}{\partial\psi}\right]$$

  - **Neutral Beam Currents:** Calculate using the NUBEAM module in TRANSP.

$$J_{NB} = J_F\left[1-\frac{Z_F}{Z_{eff}}(1-G)\right] \quad J_F \textit{ is the current density of circulating ions}$$

- ***Process repeated for ~80 high-β discharges over a range of parameters.***



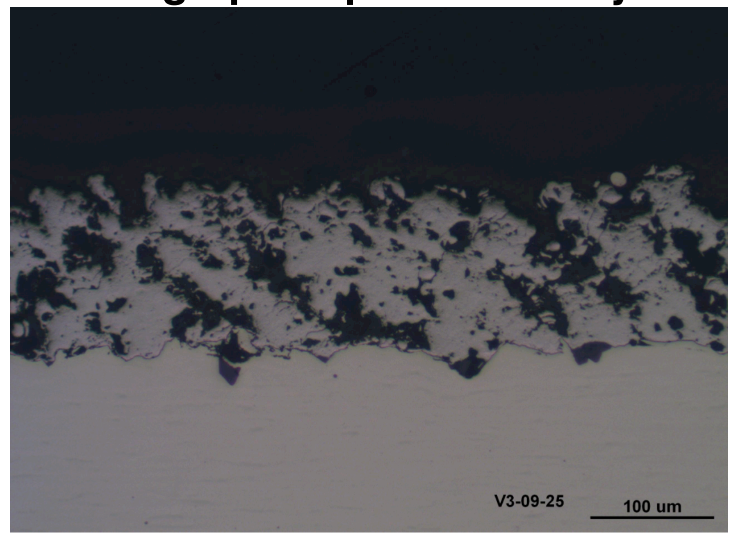


[1] O. Sauter, et a;., Phys. Plasmas **6**, 2834 (1999)

Liquid Lithium Divertor Plates Have Been Installed in NSTX For Improved Pumping of Hydrogenic Species

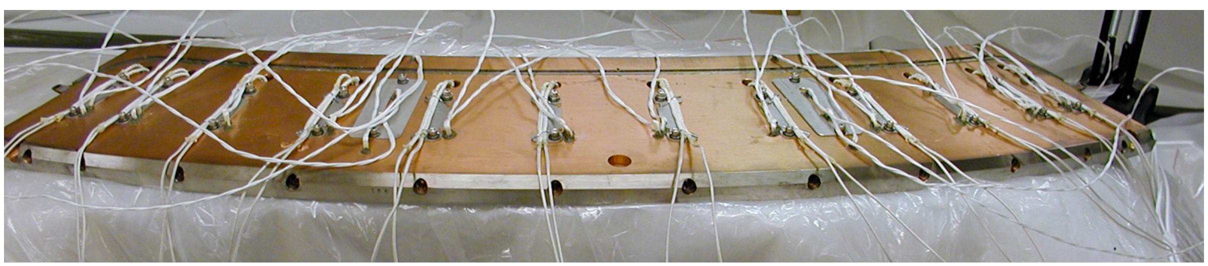
[1] H. Kugel et al., Fusion Engineering and Design **84**, 1125 (2009)

Micrograph of porous Mo layer



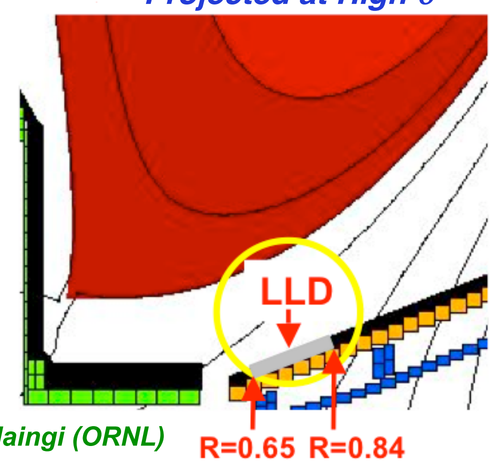
H. Kugel, R. Nygren (SNL), S. O'Dell (PPI), E. Starkman

Back side of plate with heaters and thermocouples installed



LLD to be filled with lithium from the dual LITER evaporator system

25-30% Density Reduction Projected at High-δ



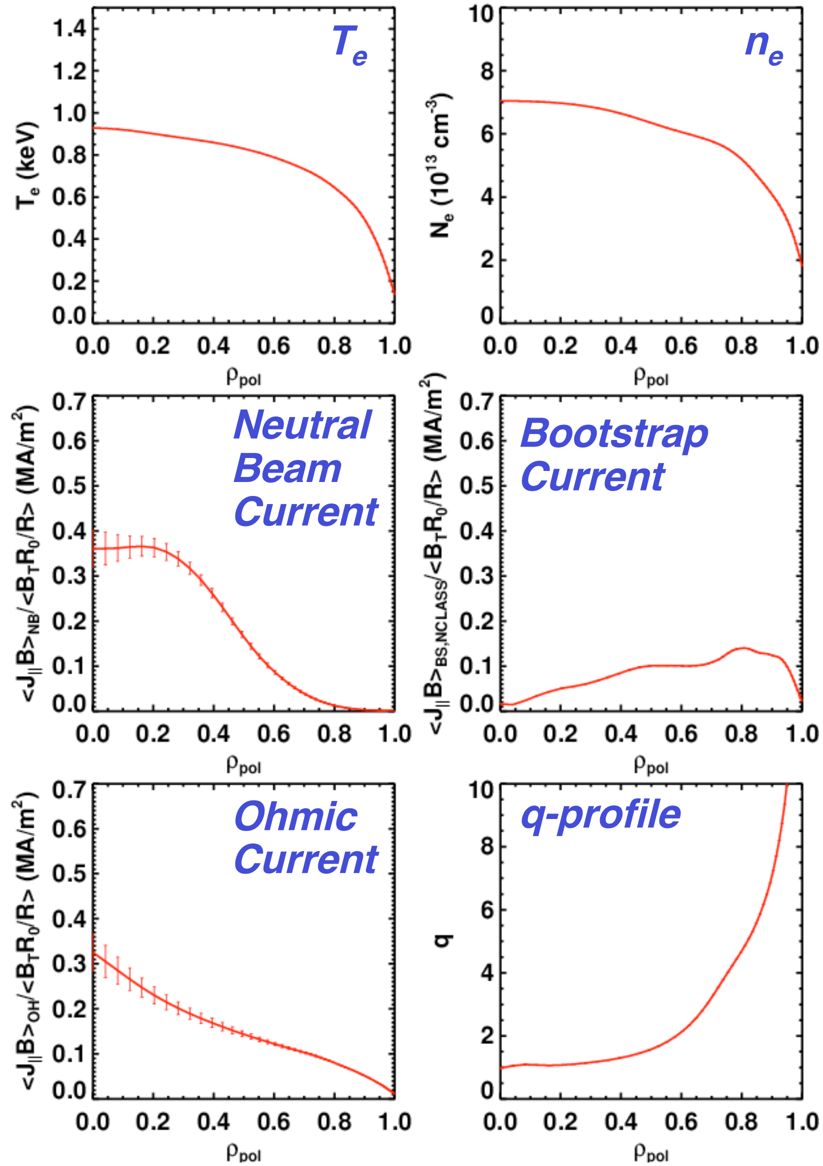
R. Maingi (ORNL)

Will LLD Pumping Increase the Non-Inductive Fraction? Depends on Impurity Accumulation and Confinement

- Fix plasma boundary and profile shapes from high-β_P discharge 133964.
- Modify the TRANSP input data to predict fully evolved current profiles.

Experimental Reference

Z_{eff}=3
Exp. Density
Exp.Temperature
f_{BS}=45%, f_{NBCD}=17%



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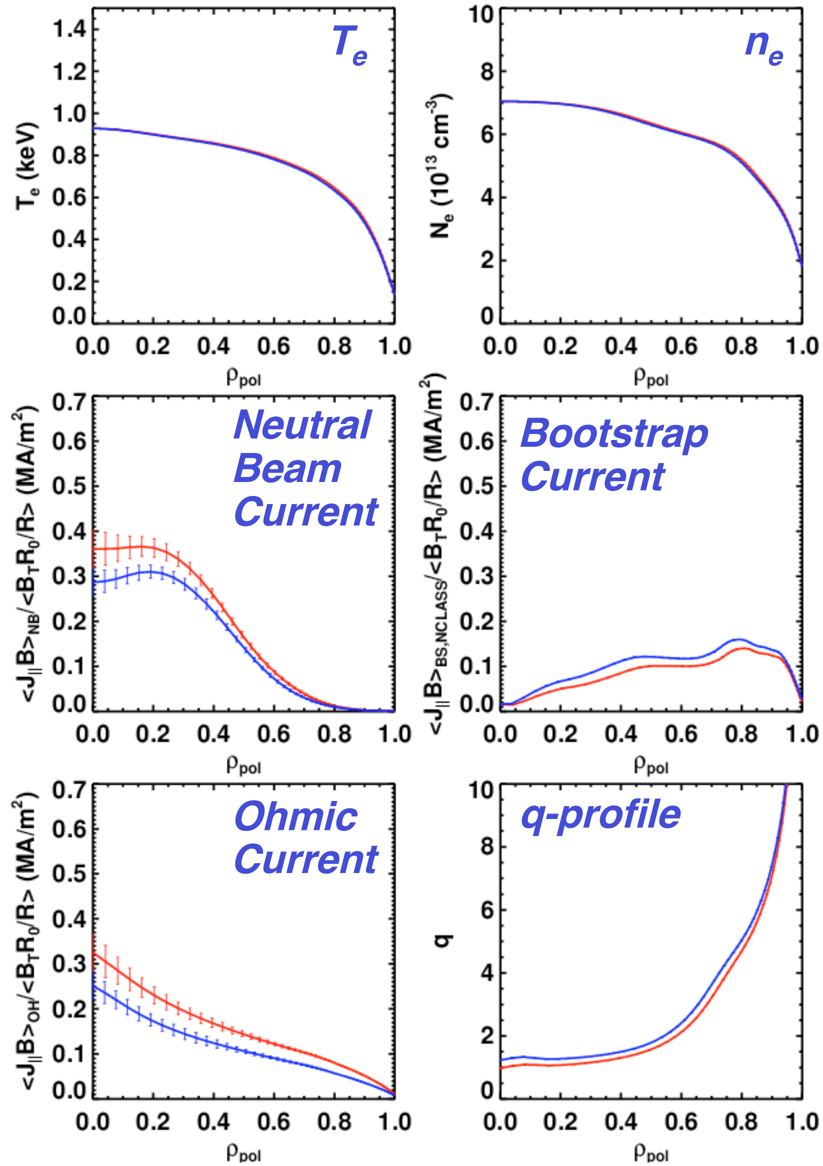
Reduce Z_{eff}

Z_{eff}=2
Exp. Density
Exp.Temperature
f_{BS}=54%, f_{NBCD}=15%



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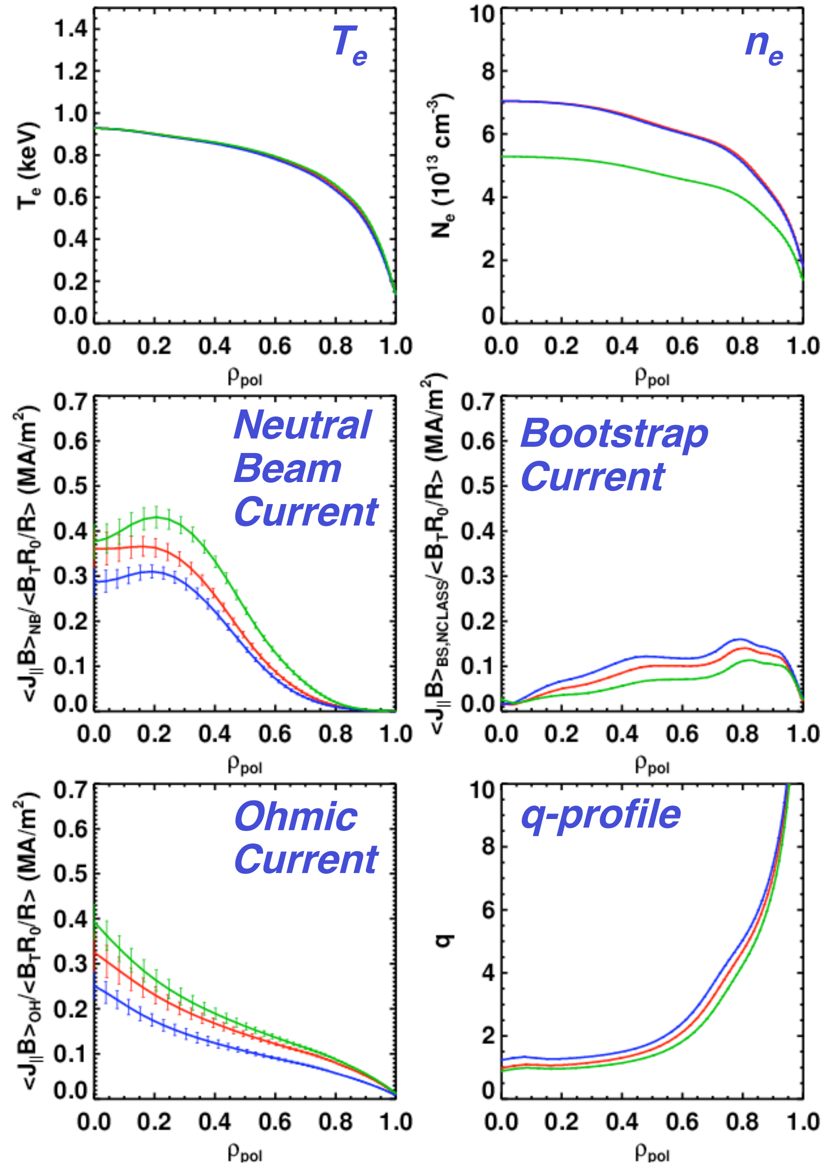
Exp.Temperature
f_{BS}=45%, f_{NBCD}=17%

Reduce Z_{eff}

Z_{eff}=2
Exp. Density
Exp.Temperature
f_{BS}=54%, f_{NBCD}=15%

Reduce the Density

Z_{eff}=2
Reduce Density 25%
Exp. Temperature
f_{BS}=38%, f_{NBCD}=22%



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- Fix plasma boundary and profile shapes from high-β_P discharge 133964.
- Modify the TRANSP input data to predict fully evolved current profiles.

Experimental Reference

Z_{eff}=3
Exp. Density
Exp. Temperature
f_{BS}=45%, f_{NBCD}=17%

Reduce Z_{eff}

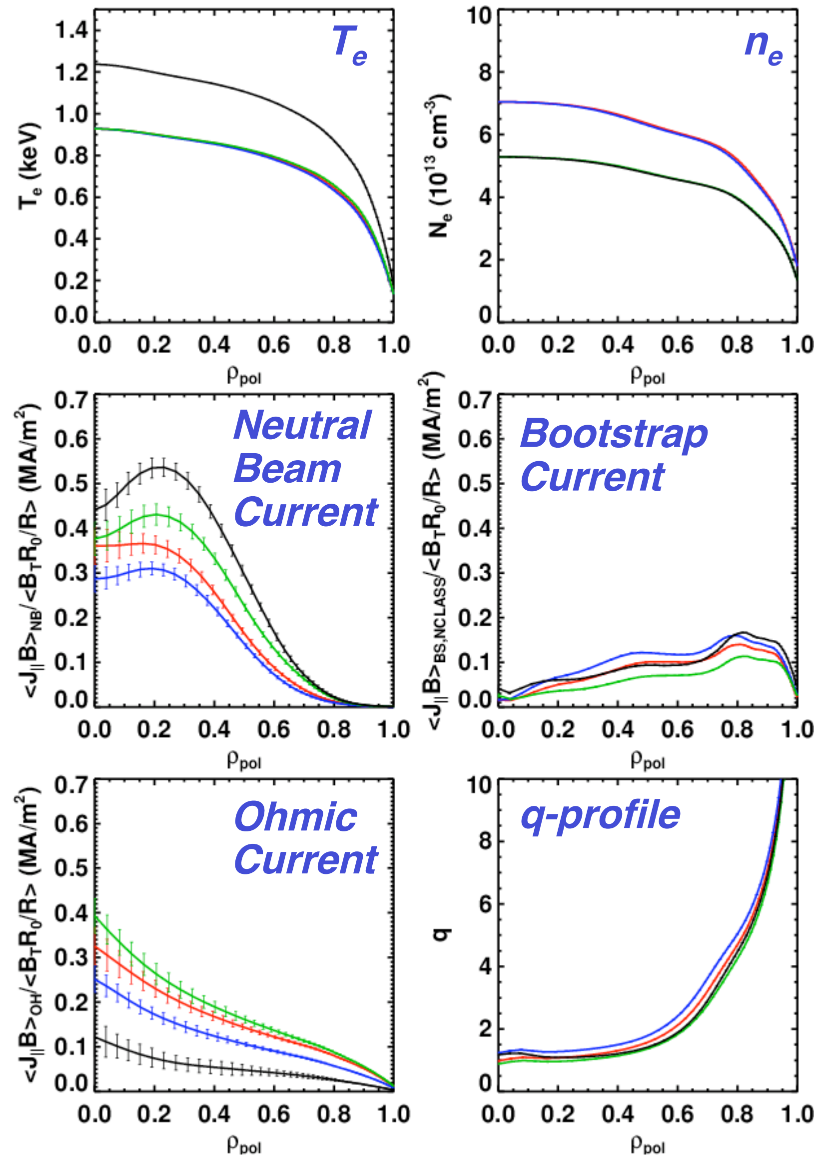
Z_{eff}=2
Exp. Density
Exp. Temperature
f_{BS}=54%, f_{NBCD}=15%

Reduce the Density

Z_{eff}=2
Reduce Density 25%
Exp. Temperature
f_{BS}=38%, f_{NBCD}=22%

Increase the Temperature

Z_{eff}=2
Reduce Density 25%
Increase Temperature 33%
f_{BS}=54%, f_{NBCD}=27%

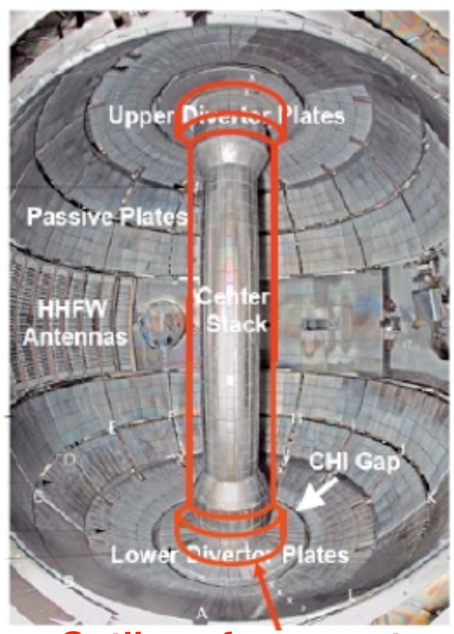


NSTX Upgrade Would Be A Major Step Along ST Development Path

(next factor of 2 increase in current, field, and power density)

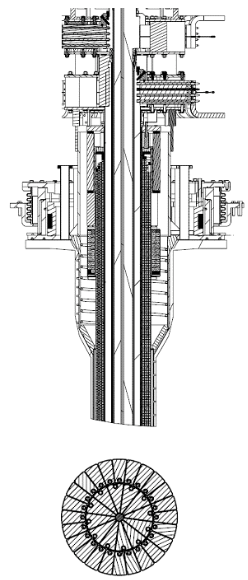
	NSTX	NSTX Upgrade	Plasma-Material Interface Facility	Fusion Nuclear Science Facility
Aspect Ratio = R_0 / a	≥ 1.3	≥ 1.5	≥ 1.7	≥ 1.5
Plasma Current (MA)	1	2	3.5	10
Toroidal Field (T)	0.5	1	2	2.5
P/R, P/S (MW/m,m^2)	10, 0.2*	20, 0.4*	40, 0.7	40-60, 0.8-1.2

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



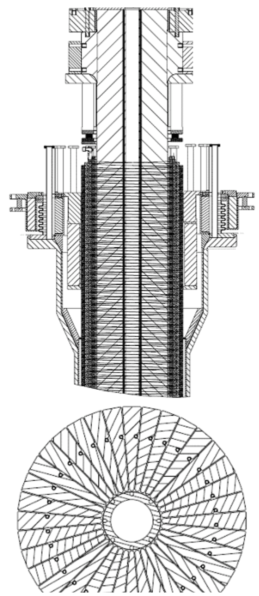
Outline of new center-stack (CS)

Present CS



TF OD = 20cm

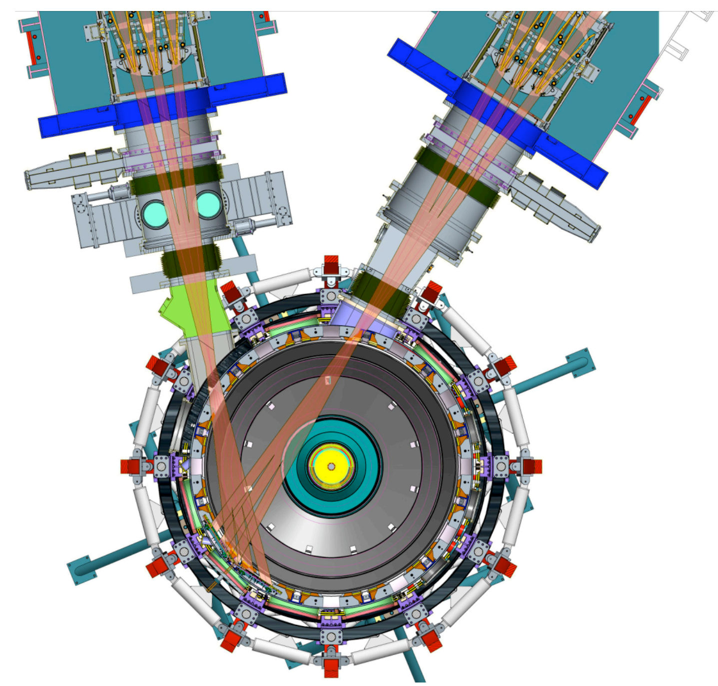
New CS



TF OD = 40cm

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

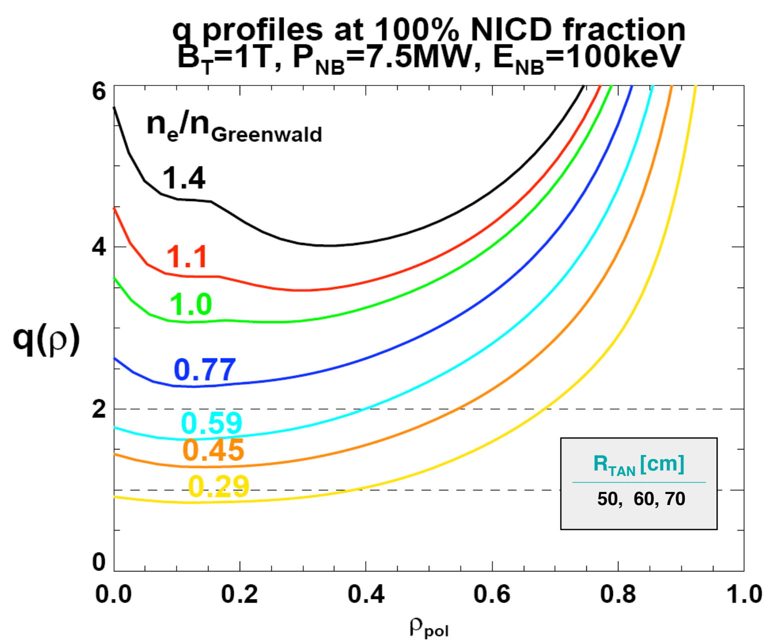
Present NBI
(R_{TAN}= 50, 60, 70cm)



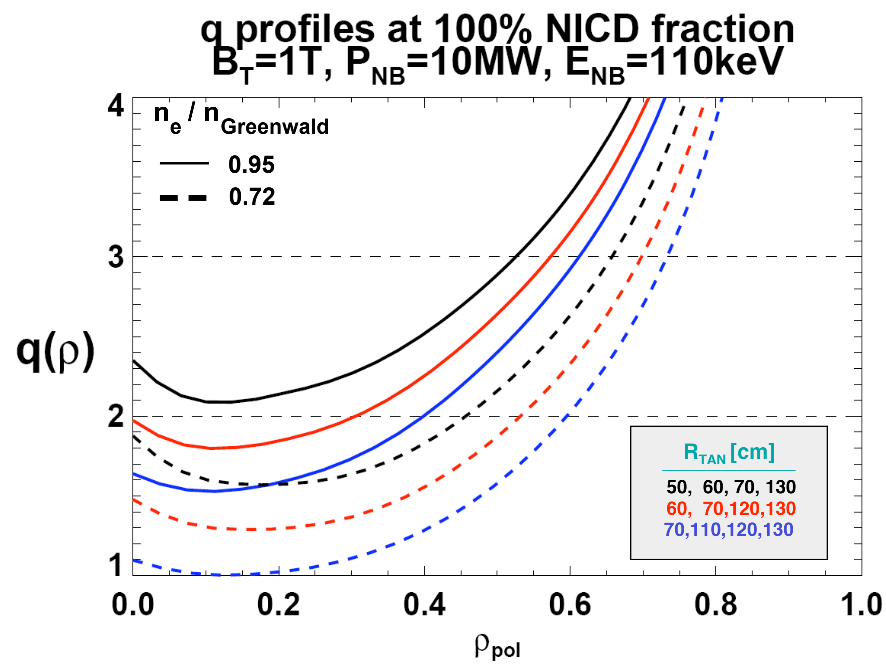
Higher Field B_T=1T from new CS + 2nd NBI Would Enable Access to Wide Range of 100% Non-Inductive Scenarios

- **New CS + present NBI-CD + fast wave:**
  - Study confinement scaling vs. I_P and B_T
    - Limited range of auxiliary power levels
  - 100% non-inductive for 1-1.5s (~1 τ_{CR})
    - NBI duration limited to 2s at 7.5MW
    - Vary q_{min} with density (CD efficiency $\propto T_e/n_e$)

- **Addition of 2nd NBI would enable:**
  - Study confinement scaling vs. I_P and B_T with:
    - Full range of auxiliary power available
    - Assured access to high-β at reduced ν^*
  - 100% non-inductive for 3-4 τ_{CR} → relaxed J(r)
    - 10MW NBI available for 5s
    - Control q_{min} & q-shear w/ NBI source, n_e, & B_T
    - Study long-pulse NTM stability with q > 2
  - Study compatibility of high-β w/ PMI solutions



I_P = 0.8-1.2MA, H_{98y2} = 1.2-1.4, β_N = 4.5-5, β_T = 10-12%, 4MW RF



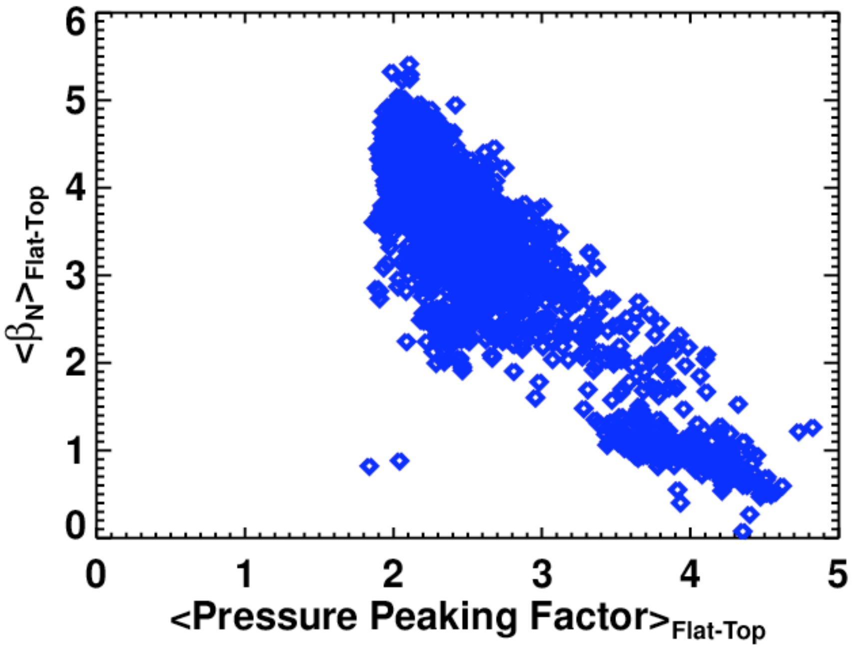
I_P = 0.95MA, H_{98y2} = 1.2, β_N = 5, β_T = 10%, 4MW RF

Summary

- NSTX discharges have achieved CTF levels of bootstrap current and β_N.
  - Up to 70% of the current has been driven non-inductively in long-pulse quiescent discharges.
- A large number of analysis and operational tools facilitate this:
  - Plasma shaping
  - n=1 mode control
  - Lithium conditioning
  - Current profile analysis
  - Many others…
- Near and long- term upgrades will enhance integrated ST research.
  - Enhanced pumping with LLD.
  - Higher field & current + off-axis beam current drive in the upgraded NSTX.

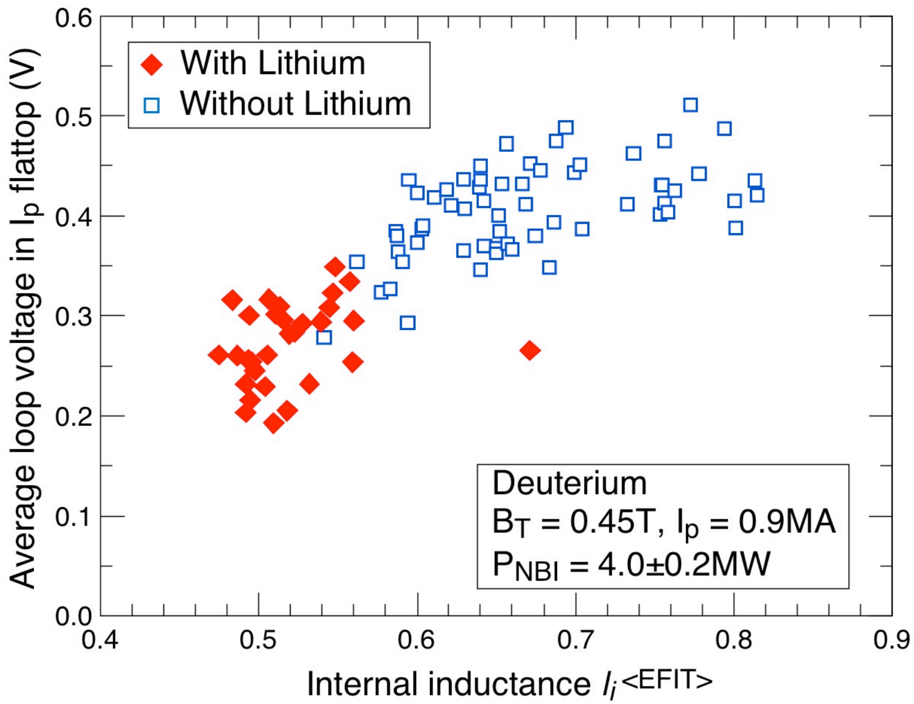
Backup

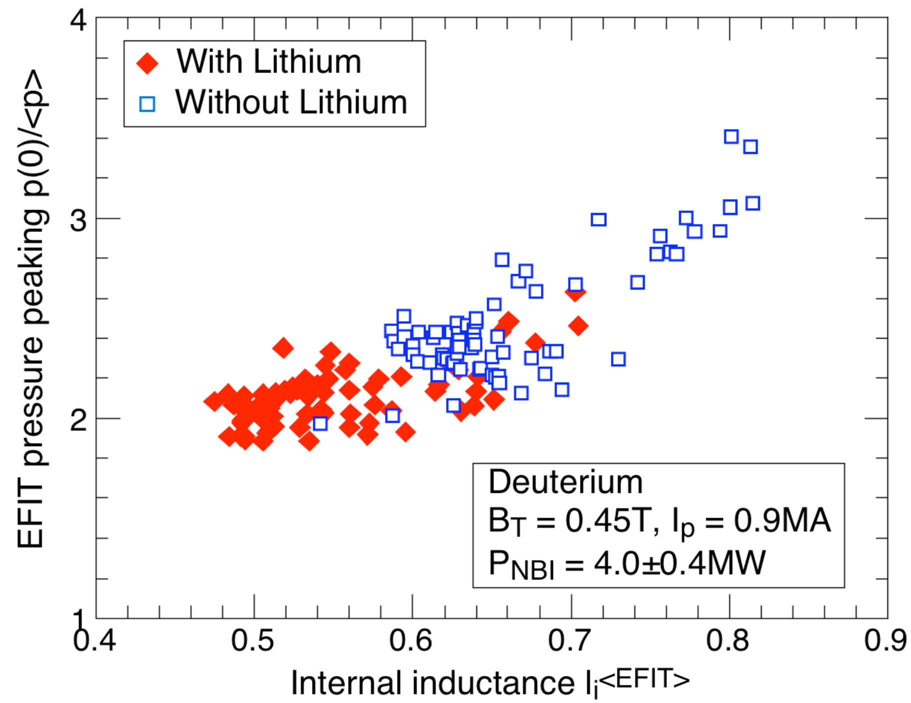
H-Mode Access Provides Improved Stability and Broad Current Profile



Broader T_e Profile with Lithium Coating Reduces Both Inductive and Resistive Flux Consumption

- Critical issue for development of low-aspect ratio tokamaks
  - Little space for conventional central solenoid providing inductive current drive



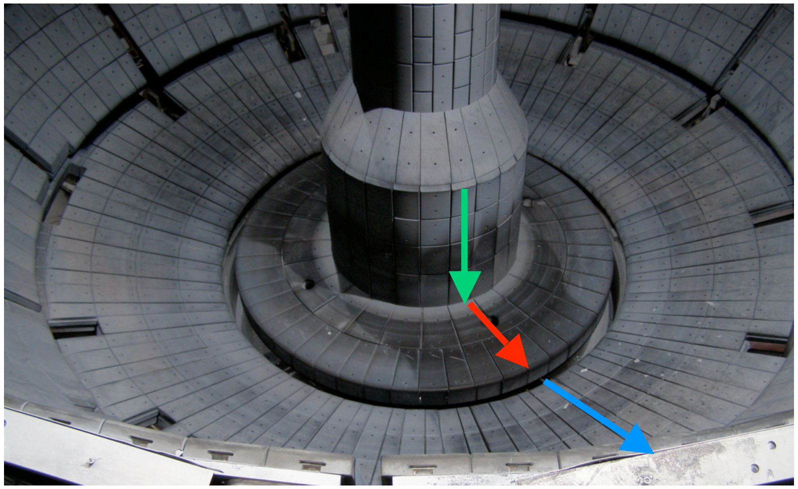


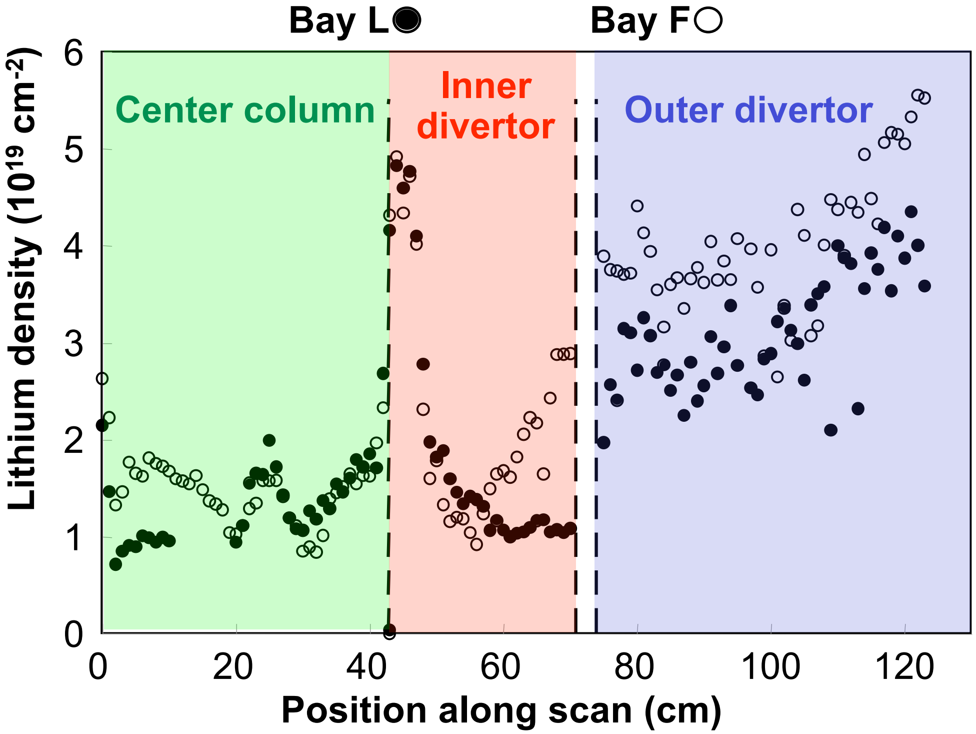
- Reduction occurs despite increase in $\langle Z_{eff} \rangle$ in ELM-free H-modes after lithium coating

Analysis of Carbon Tile Surfaces Confirms Migration of Lithium Under Plasma Fluxes

- Analysis performed on surface of carbon tiles as removed from vessel
- Used ion-beam nuclear-reaction analysis for lithium and deuterium areal density in surface layer

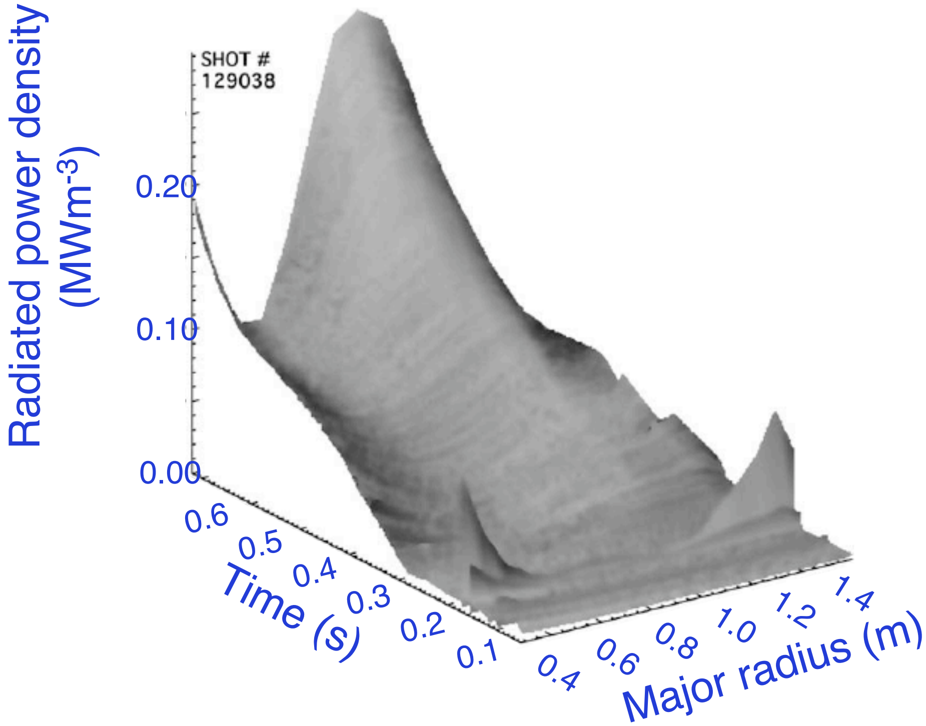
Scan across lower divertor

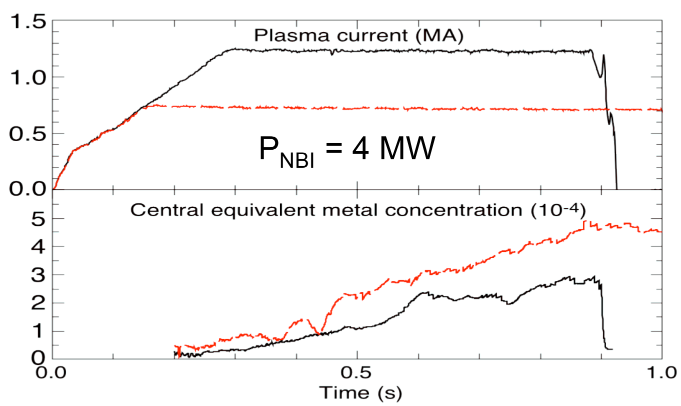




- Peak lithium density remaining on inner divertor ~0.6 mg·cm^{-2}
- Total deposition there estimated at ~8 mg·cm^{-2}

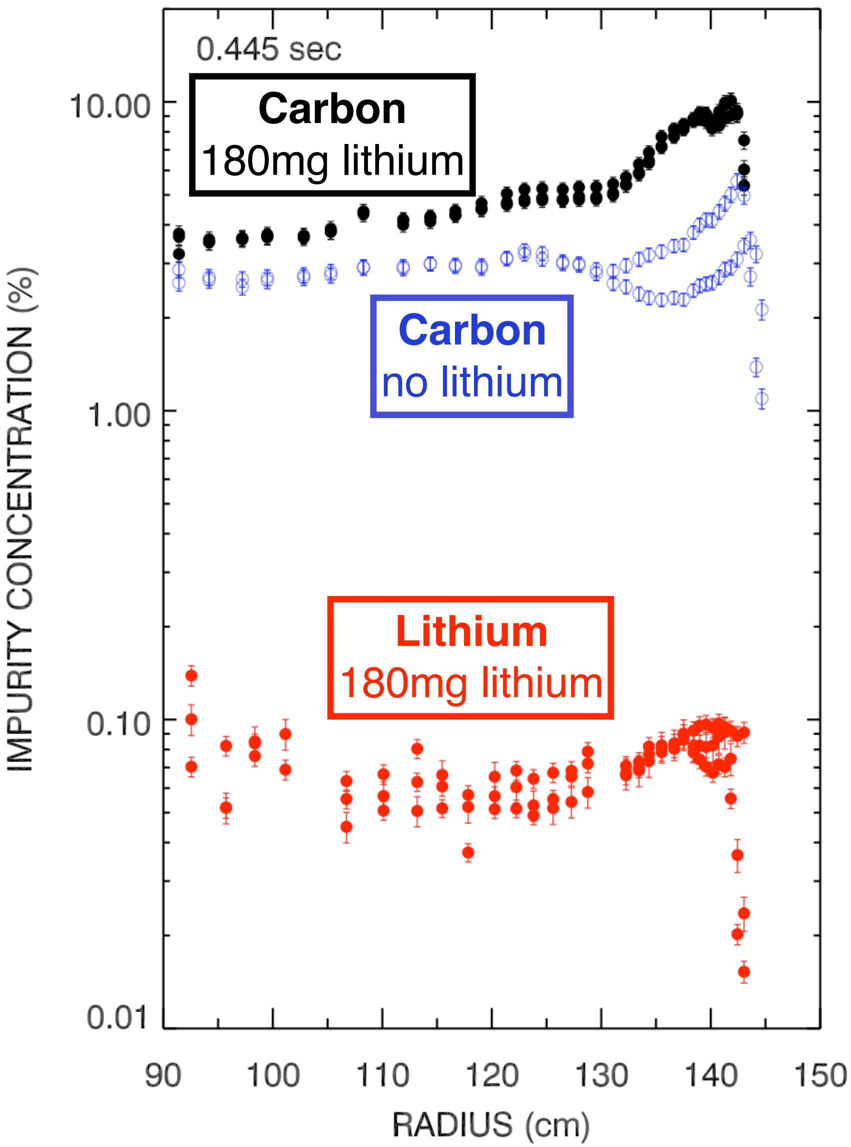
Metals Responsible for Most of the Increase in Radiation When ELMs Suppressed by Lithium



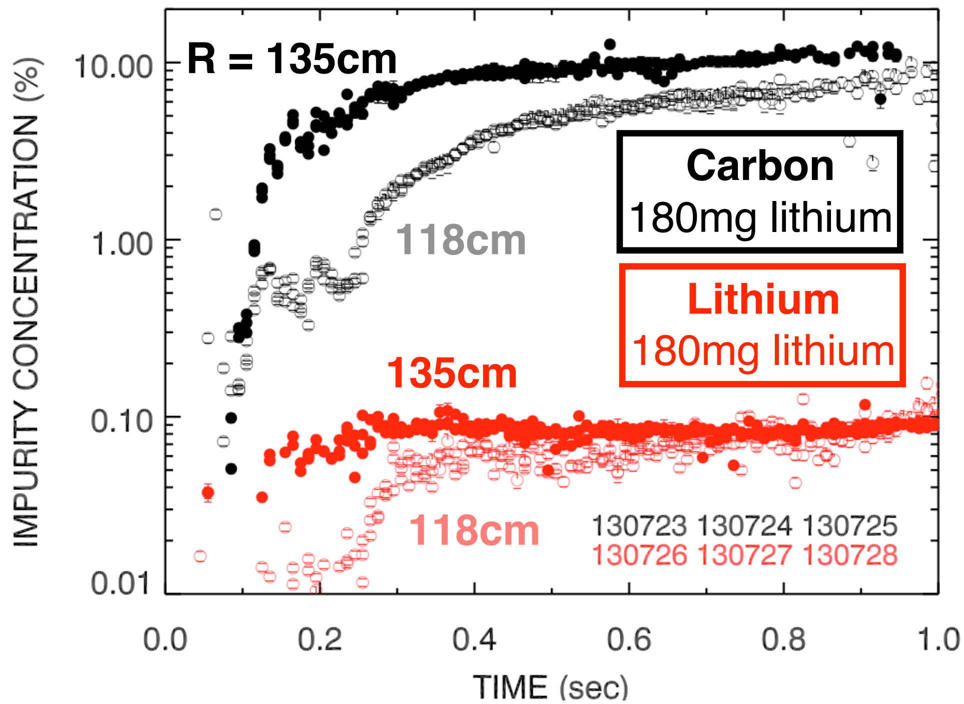


- Radiated power centrally peaked in ELM-free discharges
- VUV and SXR spectra show iron lines (Fe X – XVIII) increasing during ELM-free periods
- Radiated power profile remains hollow when ELMs are present
  - Metals still present early but do not accumulate
- If increase in radiation is ascribed to iron-like metals:
  - $n_{\text{"Fe"}}/n_e \sim 0.1\%$
  - $\Delta Z_{eff}(\text{"Fe"}) \sim 0.3$
- Dependence of rate of rise of radiation on I_p suggests sputtering by unconfined NB ions is source

Lithium Concentration in Plasmas Remains Low but Carbon Concentration Rises with Lithium Coating

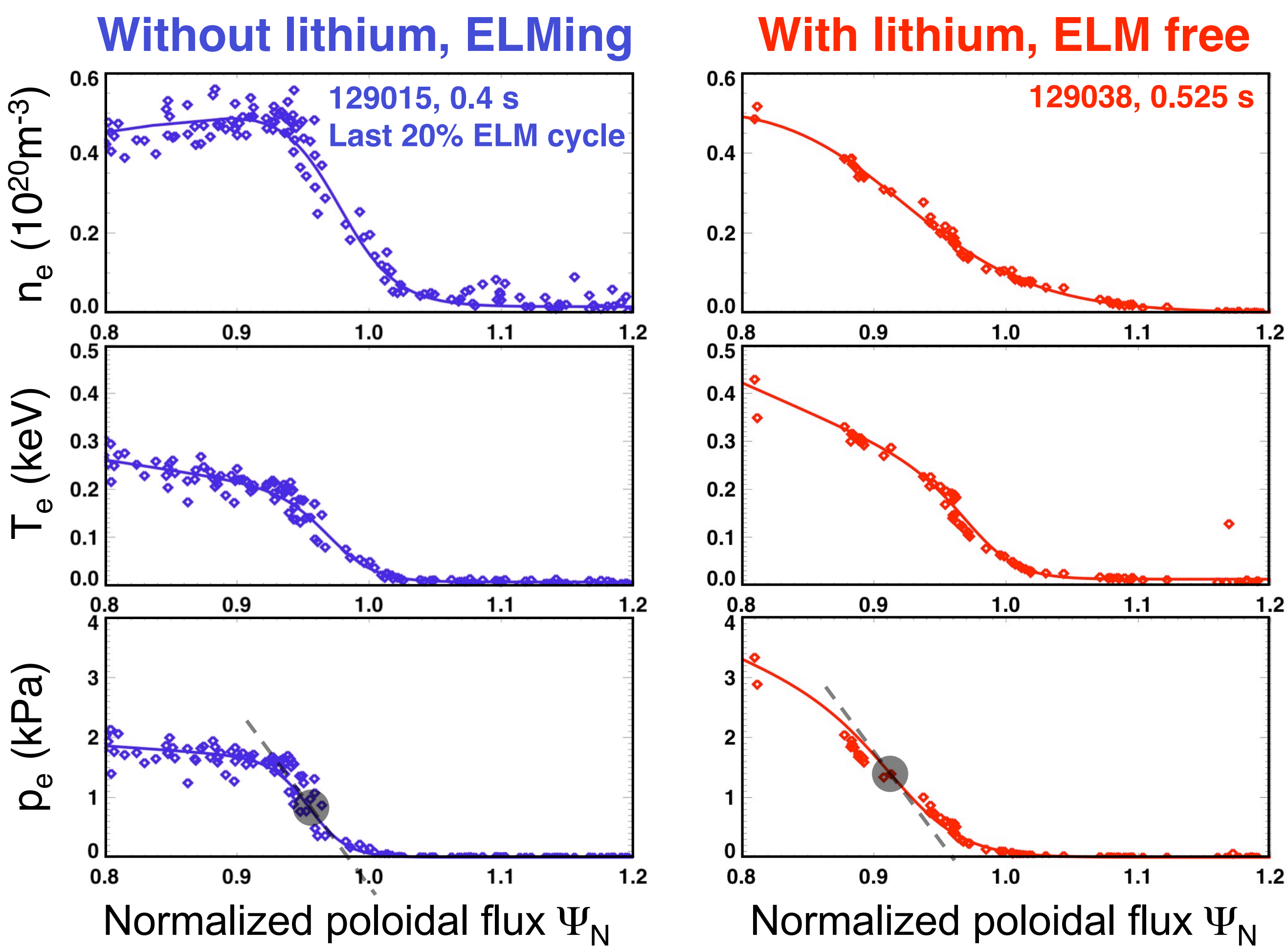


- Quantitative measurements of C^{6+}, Li^{3+} with charge-exchange recombination spectroscopy
- **n_C/n_{Li} = 30 – 100**
- Hollow profiles early for both C and Li fill in as time progresses



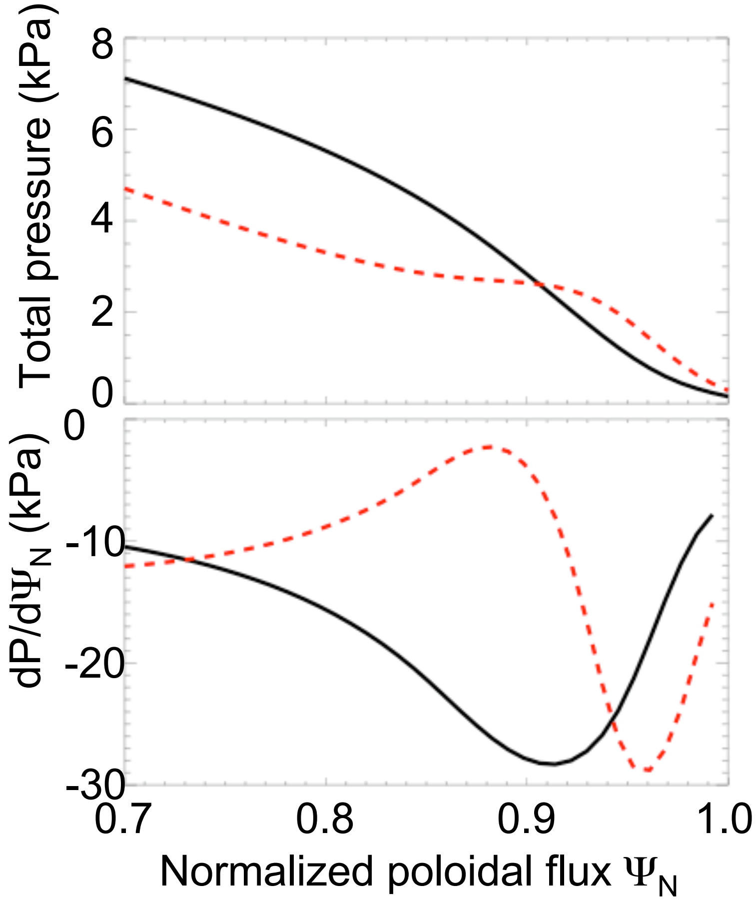
Lithium Affects ELMs Through Changes in Temperature and Pressure Profile at Edge

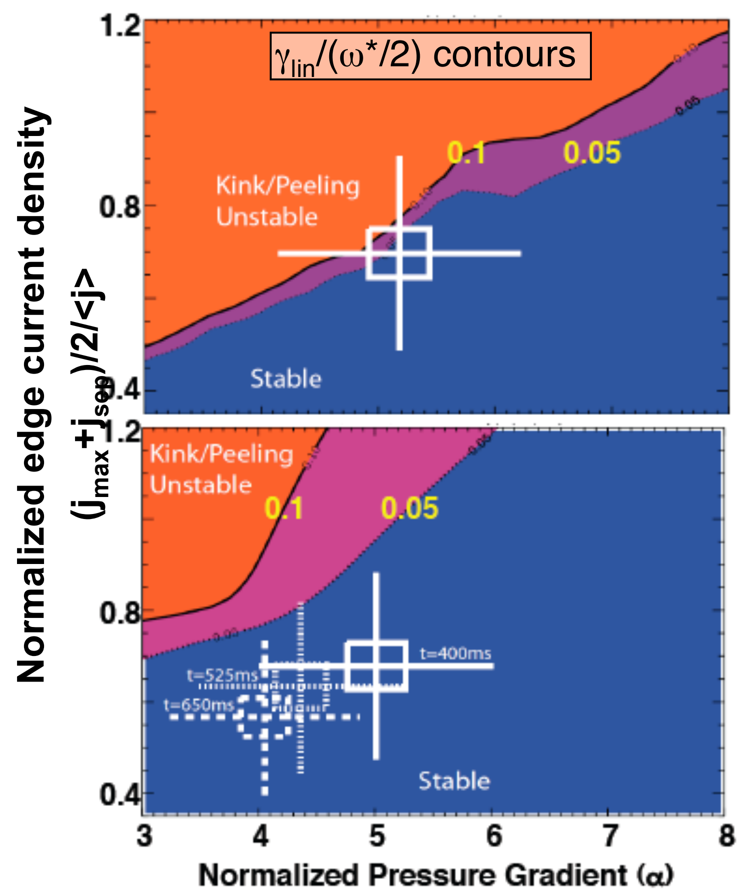
- Multiple timeslices mapped into composite profiles using EFIT equilibrium



Shift of Maximum in ∇p_e to Region of Lower Shear with Lithium Stabilizes Kink/Ballooning

- Analysis with PEST and ELITE codes
- Change in recycling affects edge current
- Precursor activity with n = 1 – 5 observed before ELM onset



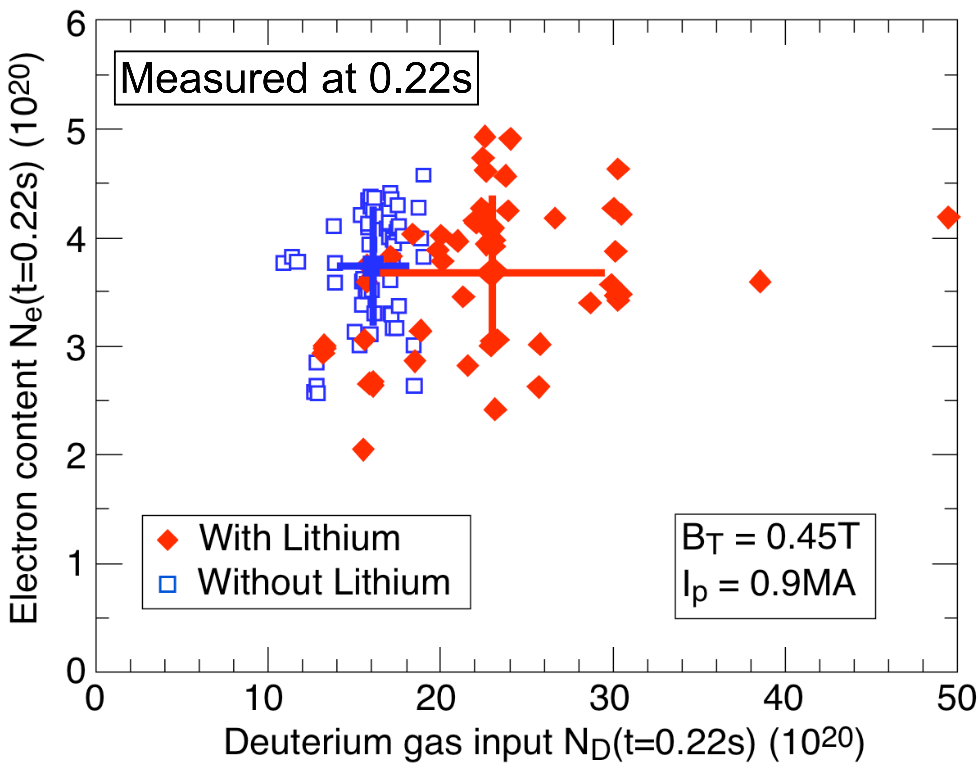


Without lithium (end of ELM cycle)

With lithium

Lithium Reduces Deuterium Recycling but Need to Increase Fueling to Avoid Early Locked Modes

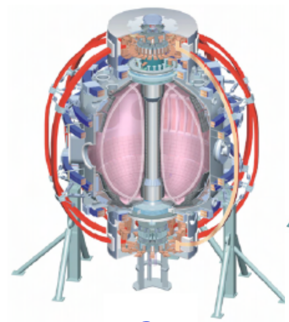
- Lower density achievable early in discharges both with and without lithium but likelihood of deleterious locked modes increases
  - Extensive HeGDC, He ohmic- or HHFW-heated plasmas also effective



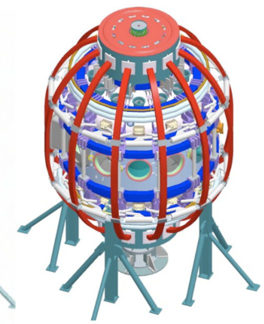
- Tangentially viewing camera for edge D_{α} emission shows greatly reduced neutral D density across outboard midplane with lithium

NSTX Mission Elements

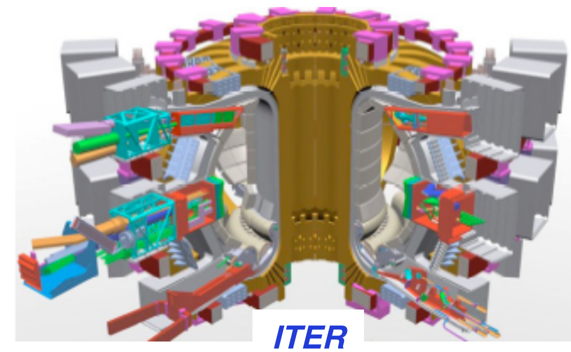
- **Understand unique physics properties of ST**
  - Assess impact of low A, high β, high v_{fast} / v_A on toroidal plasma science + impact of high power density on PMI
  - Longer term NSTX → NSTX Upgrade goals:
    - Study high beta plasmas at reduced collisionality
    - Access full non-inductive start-up, ramp-up, sustainment
    - Prototype solutions for mitigating high heat & particle flux
- **Extend tokamak physics understanding, support ITER**
  - Exploit unique and complementary ST features
  - Benefit from tokamak research and development
- **Establish attractive ST operating conditions**
  - Understand and utilize ST for addressing key gaps between ITER and FNSF / DEMO
    - ReNeW Thrusts 14-15 (FNS), 9-12 (PMI), 8 (self-driven high-Q_{DT})
  - Advance ST as fusion power source



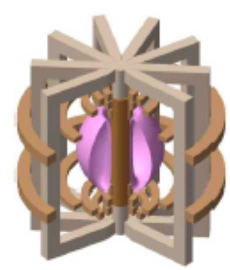
NSTX



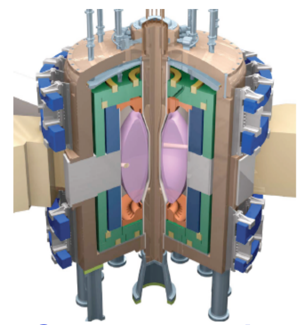
NSTX-U



ITER



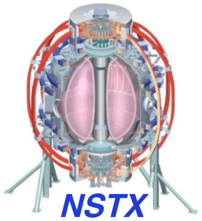
ST-based Plasma Material Interface (PMI) Science Facility

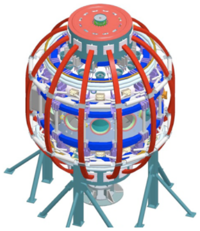


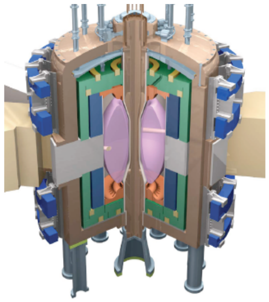
ST-based Fusion Nuclear Science (FNS) Facility

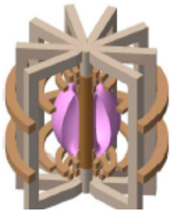
NSTX Upgrade will contribute strongly to toroidal plasma science and preparation for a fusion nuclear science (FNS) program

- NSTX:
  - Providing foundation for understanding ST physics, performance
- NSTX Upgrade:
  - Study high beta plasmas at reduced collisionality
    - Vital for understanding confinement, stability, start-up, sustainment
  - Assess full non-inductive current drive operation
    - Needed for steady-state operating scenarios in ITER and FNS facility
  - Prototype solutions for mitigating high heat, particle exhaust
    - Can access world-leading combination of P/R and P/S
    - Needed for testing integration of high-performance fusion core and edge
- NSTX Upgrade contributes strongly to possible next-step STs:
  - ST Fusion Nuclear Science Facility
    - Develop fusion nuclear science, test nuclear components for Demo
    - Sustain $W_{neutron}$ ~ 0.2-0.4 → 1-2MW/m^2, τ_{pulse} = 10^3 → 10^6s
  - ST Plasma Material Interface Facility
    - Develop long-pulse PMI solutions for FNSF / Demo (low-A and high-A)
    - Further advance start-up, confinement, sustainment for ST
    - High P_{heat}/S ~1MW/m^2, high T_{wall}, τ_{pulse} ~ 10^3s



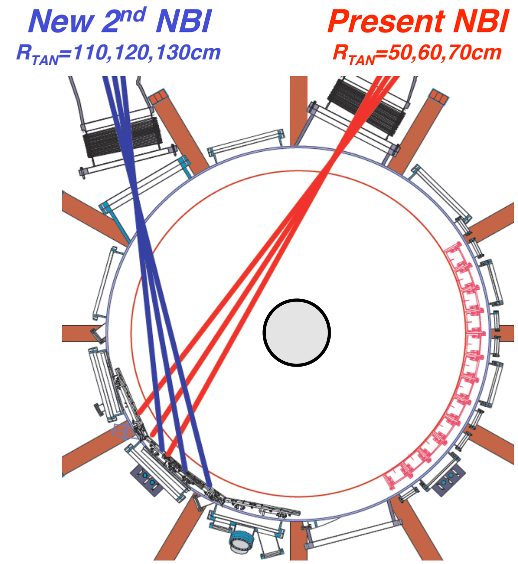




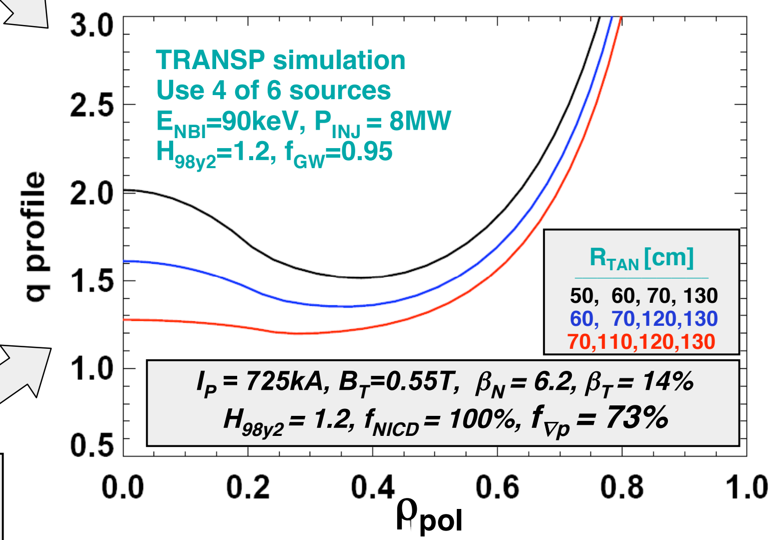


Upgrade 2nd NBI injecting at larger $R_{tangency}$ will greatly expand performance and understanding of ST plasmas

- Improved NBI-CD and plasma performance
  - Higher CD efficiency from large R_{TAN}
  - Higher NBI current drive from higher P_{NBI}
  - Higher β_P, f_{BS} at present $H_{98y2} \leq 1.2$ from higher P_{HEAT}
  - Large R_{TAN} → off-axis CD for maintaining $q_{min} > 1$
  - Achieve 100% non-inductive fraction (presently < 70%)
  - Optimized $q(\rho)$ for integrated high τ_E, β, and f_{NI}



- Expanded research flexibility by varying:
  - *q*-shear for transport, MHD, fast-ion physics
  - Heating, torque, and rotation profiles
  - β, including higher β at higher I_P and B_T
  - Fast-ion $f(v_{||},v_{\perp})$ and *AE instabilities
    - 2nd NBI more tangential – like next-step STs
  - Peak divertor heat flux, SOL width



- q(r) profile variation and control very important for global stability, electron transport, Alfvénic instability behavior