NSTX

Supported by U.S. DEPARTMENT OF ENERGY | Office of Science

Research Results and Opportunities on the National Spherical Torus Experiment (NSTX)

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NSTX Program Director

for the NSTX Research Team

AST-558 Graduate Plasma Seminar
Theory Conference Room
September 20, 2010

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Columbia U
CompX
General Atomics
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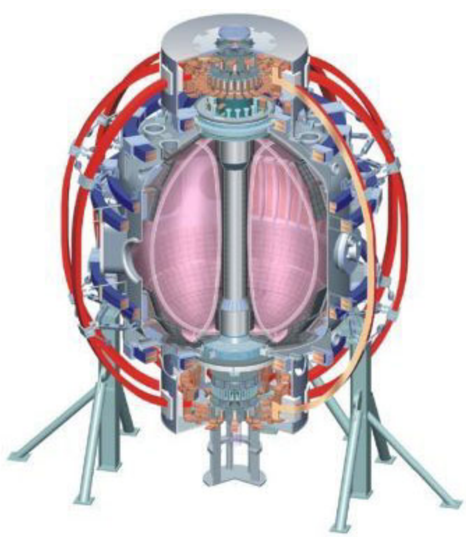

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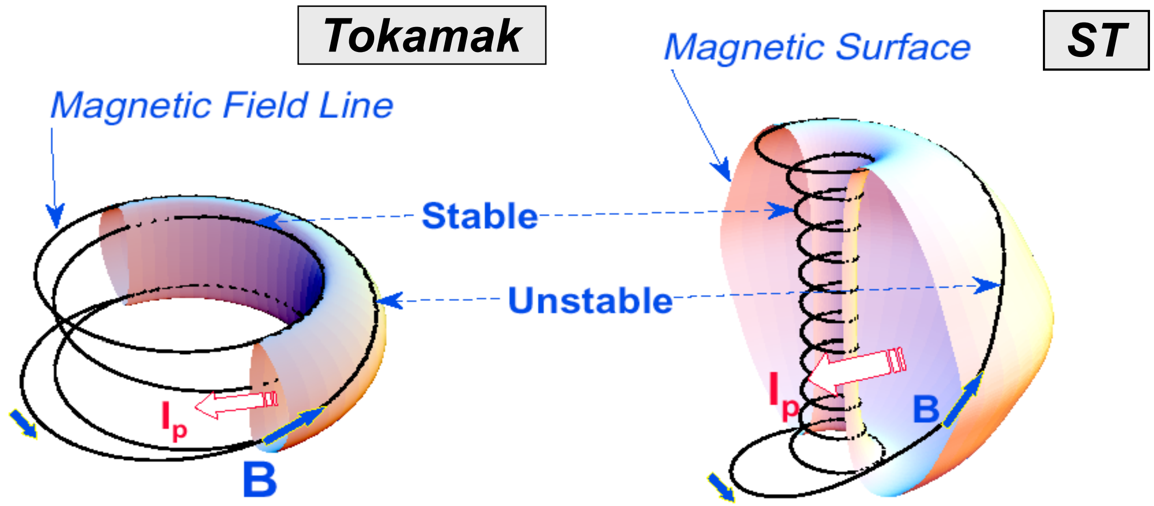
ASIPP
ENEA, Frascati
CEA, Cadarache
IPP, Jülich
IPP, Garching
ASCR, Czech Rep
U Quebec

A Spherical Torus (ST) is a low aspect ratio, high-β Tokamak

Aspect Ratio A = R /a
(plasma average major radius / plasma half-width)

$\beta_T = \langle p \rangle / (B_{T0}^2 / 2\mu_0)$
(ratio of kinetic pressure to magnetic pressure)



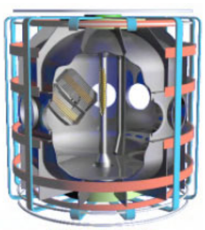
R/a~3-4, κ=1.5-2, q_a=3-4, β_T=3-10%   R/a~1.5, κ=2-3, q_a=8-12, β_T=10-40%

- **New physics regimes are accessed at low aspect ratio, enhancing the understanding of toroidal confinement physics:**
  - Lower A → increased toroidicity → higher β
  - Higher β → enhanced electromagnetic effects in turbulence
  - Higher fraction of trapped particles, increased normalized orbit size, plasma flow, and flow shear → broad range of effects on transport and stability
  - Increased normalized fast-ion speed → simulate fast-ion transport/losses of ITER
  - Compact geometry → high power/particle/neutron flux relevant to ITER, reactors

The ST offers attractive near-term applications for fusion development complementary to ITER

ST characteristics:

High normalized pressure
Compact geometry
Simplified magnets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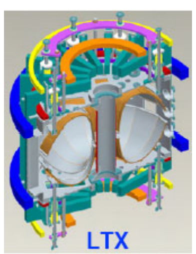
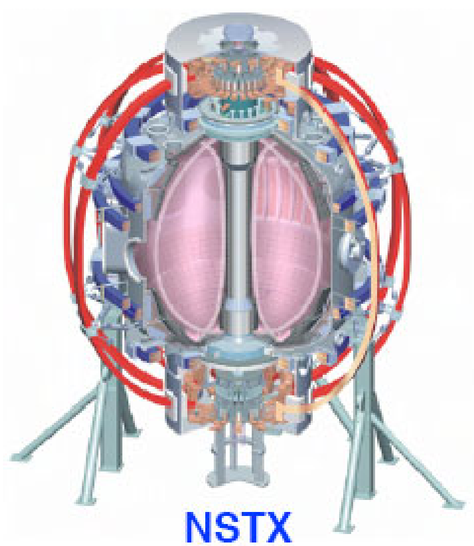




Implications:

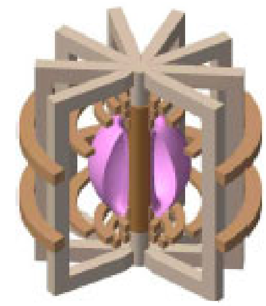
High heat flux at small size and reduced cost

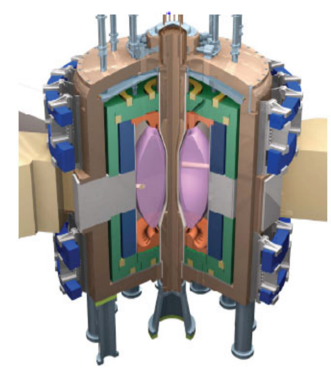
Simplified construction, access, and maintenance

High neutron flux at small size and reduced cost, reduced tritium consumption

Near-term ST Applications:

Plasma-Material Interface R&D + Advanced Physics



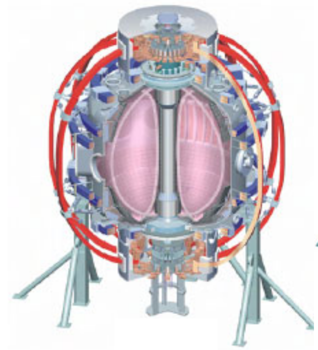


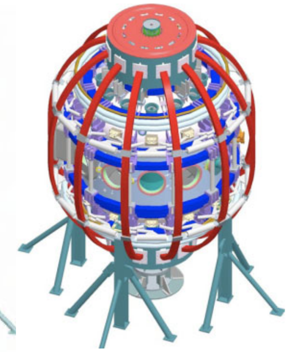
Fusion Nuclear Science, Component Testing

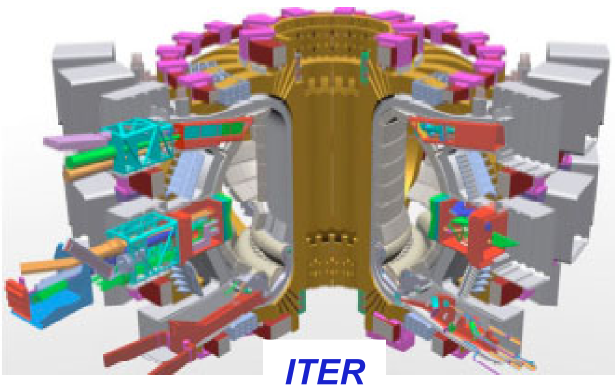
Longer term: ST Power Plant offers simplest magnets, easiest maintenance

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, ITER physics understanding**
  - 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
  - Advance ST as fusion power source


NSTX


NSTX-U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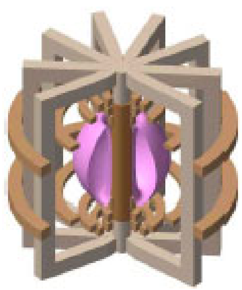
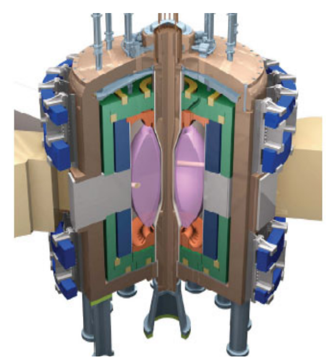
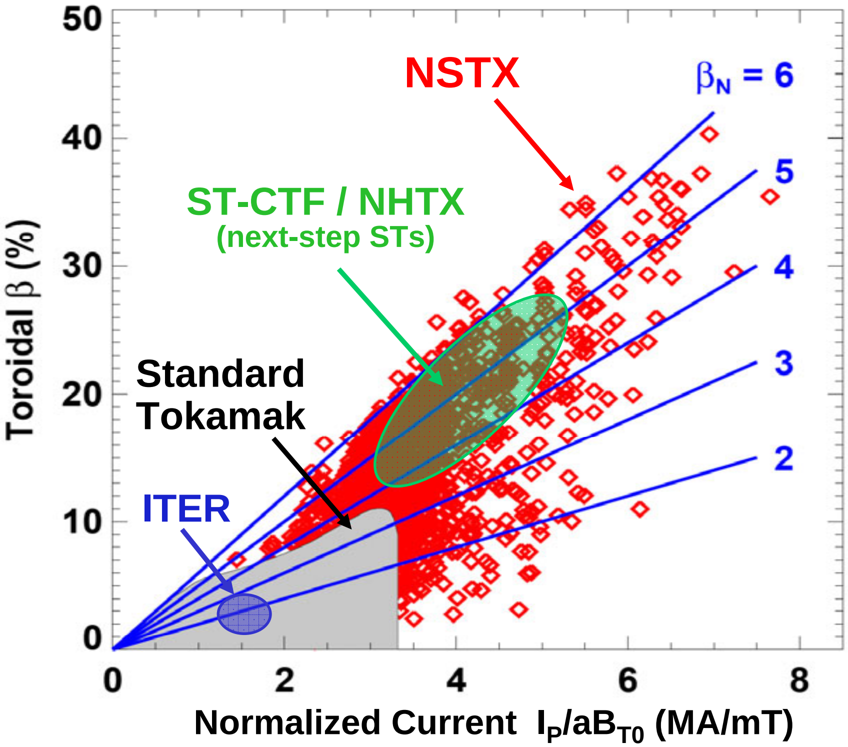

ST-based Plasma Material Interface (PMI) Science Facility


ST-based Fusion Nuclear Science (FNS) Facility

NSTX Research Areas

- Macroscopic Stability
- Transport and Turbulence
- Waves and Energetic Particles
- Lithium Research, Boundary Physics
- Plasma Formation and Sustainment

NSTX creates stable, well diagnosed plasmas at high β enabling a wide range of toroidal physics studies



- ST accesses higher normalized current & higher normalized β

➡ higher $\beta_{Toroidal}$

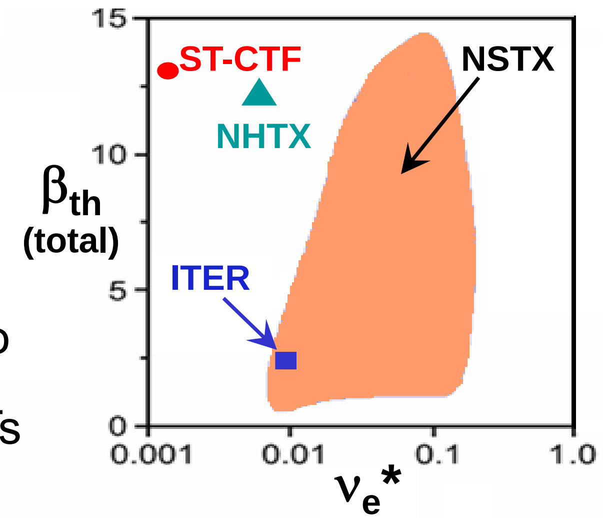
(High β_N results in part from rotational stabilization of resistive wall mode)

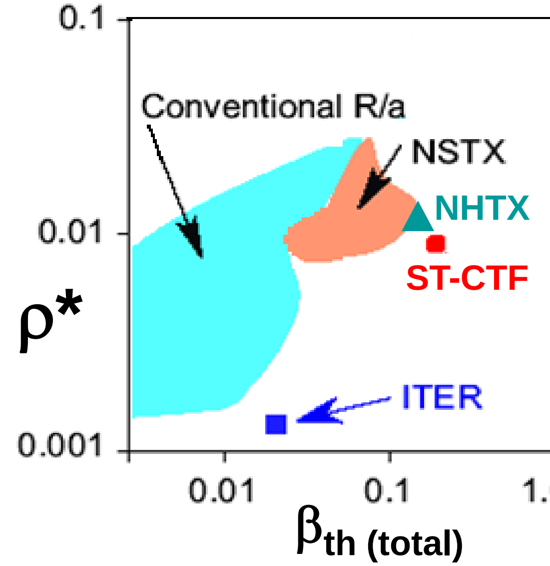
Access ITER-level ν^*, extending confinement understanding to high β

Next-step STs expected to operate at significantly lower ν^* than present STs

ST operates at higher ρ^* than tokamaks / ITER - impacts thermal and fast-ion transport, MHD

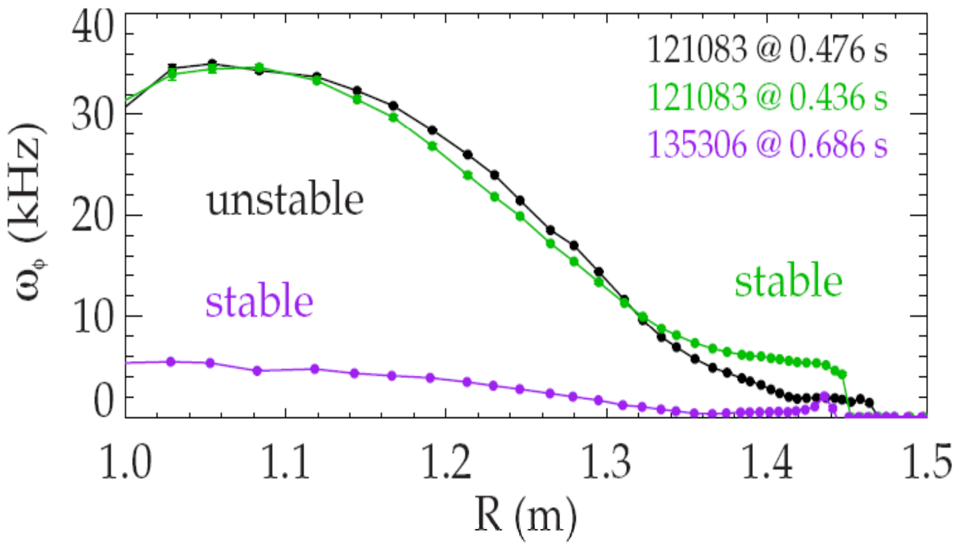
Extrapolation in ρ^* from present STs to next-step STs is small





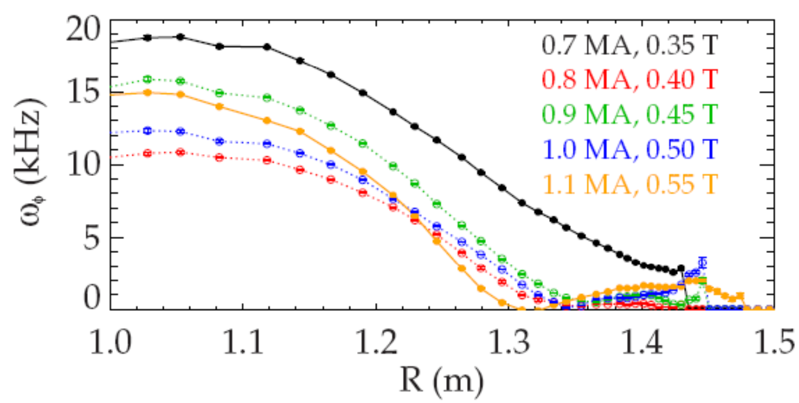
NSTX observes complex n=1 RWM stability behavior

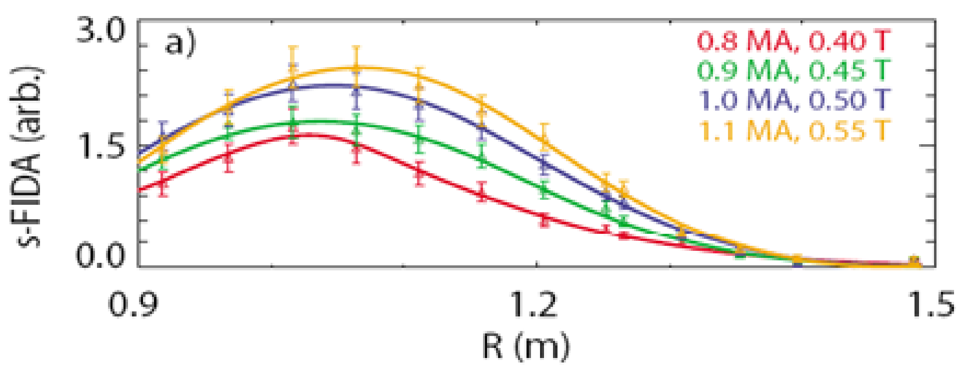
- n=1 RWM marginal stability is non-monotonic function of toroidal rotation



- Simple fluid, semi-kinetic models cannot explain results → need kinetic theory of RWM stability

- n=1 RWM marginal rotation reduced with increased NBI fast ion content (scan at fixed q)





Berkery *et al.* Phys. Plasmas **17**, 082504 (2010)

Modification to Ideal Stability by Kinetic effects (MISK) code:

- **Perturbative approach used to calculate $\delta W_{Kinetic}$** *Betti*
  - **Assumes that kinetic effects do not change eigenfunction**

$$(\gamma - i\omega_r)\tau_w = -\frac{\delta W_\infty + \delta W_K}{\delta W_b + \delta W_K} \qquad \delta W_K = -\frac{1}{2}\int \xi_\perp^* \cdot (\nabla \cdot \tilde{\mathbb{P}}_K) dV$$

- **Thermal particles:**

$$\delta W_K = \frac{\sqrt{2}\pi^2}{m_j^{3/2}} \sum_{l=-\infty}^{\infty} \int d\varepsilon \int d\chi \int \frac{d\Psi}{B_0} \hat{\tau} \left(-2|\chi| \frac{B_0}{B} \right) \times \frac{(\omega_r + i\gamma - \omega_E)\frac{\partial f_j}{\partial \varepsilon} - \frac{1}{eZ_j}\frac{\partial f_j}{\partial \Psi}}{\langle \omega_D^j \rangle + l\omega_b^j - i\nu_{\rm eff}^j + \omega_E - \omega_r - i\gamma} \varepsilon^{5/2} |\langle H/\hat{\varepsilon}\rangle|^2$$

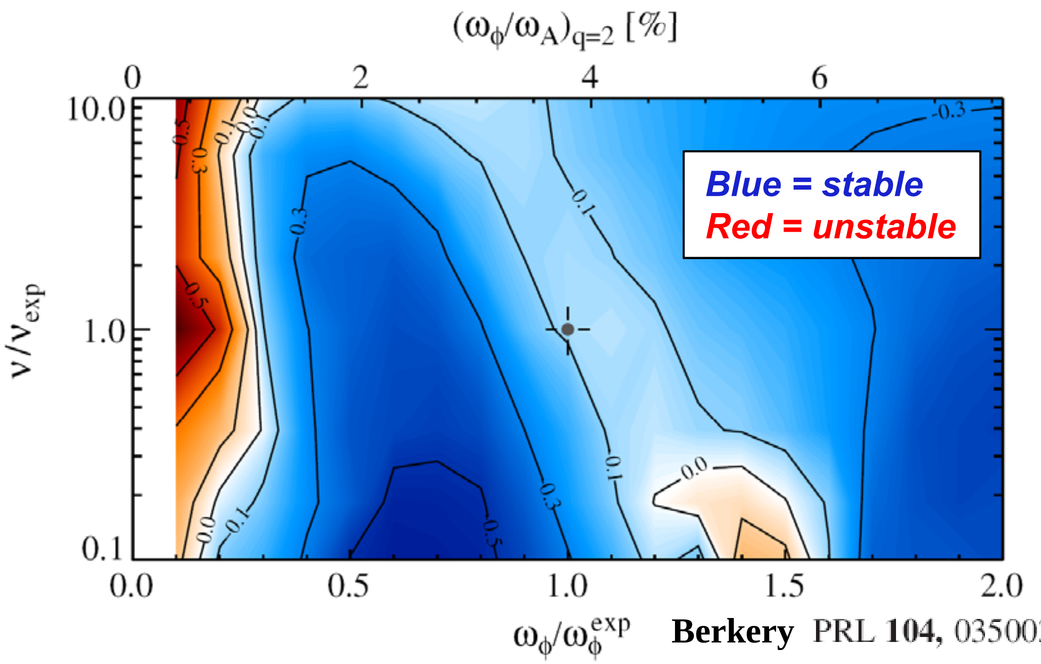
- **Fast-ions (slowing-down distribution):**

$$\delta W_K^a = \frac{\pi}{4} \sum_{l=-\infty}^{\infty} \int d\hat{\varepsilon} \int d\chi \int \frac{d\Psi}{B_0} n_a \varepsilon_a \hat{\tau}^a \left(2|\chi| \frac{B_0}{B} \right) \left(\int \frac{\hat{\varepsilon}^{1/2}}{\hat{\varepsilon}^{3/2} + \hat{\varepsilon}_c^{3/2}} d\hat{\varepsilon} \right)^{-1} \frac{\hat{\varepsilon}^{5/2}}{\hat{\varepsilon}^{3/2} + \hat{\varepsilon}_c^{3/2}} |\langle H/\hat{\varepsilon}\rangle|^2$$

$$\times \left[\frac{\frac{3}{2}\hat{\varepsilon}^{1/2}}{\hat{\varepsilon}^{3/2} + \hat{\varepsilon}_c^{3/2}} (\omega_E - \omega_r - i\gamma) + \frac{\varepsilon_a}{eZ_a} \left(\frac{1}{\hat{\varepsilon}^{3/2} + \hat{\varepsilon}_c^{3/2}} \frac{d\hat{\varepsilon}_c^{3/2}}{d\Psi} - \frac{1}{n_a}\frac{\partial n_a}{\partial \Psi} - \omega_f^a \right) \right] (\langle \omega_D^a \rangle + l\omega_b^a - i\nu_{\rm eff}^a + \omega_E - \omega_r - i\gamma)^{-1}$$

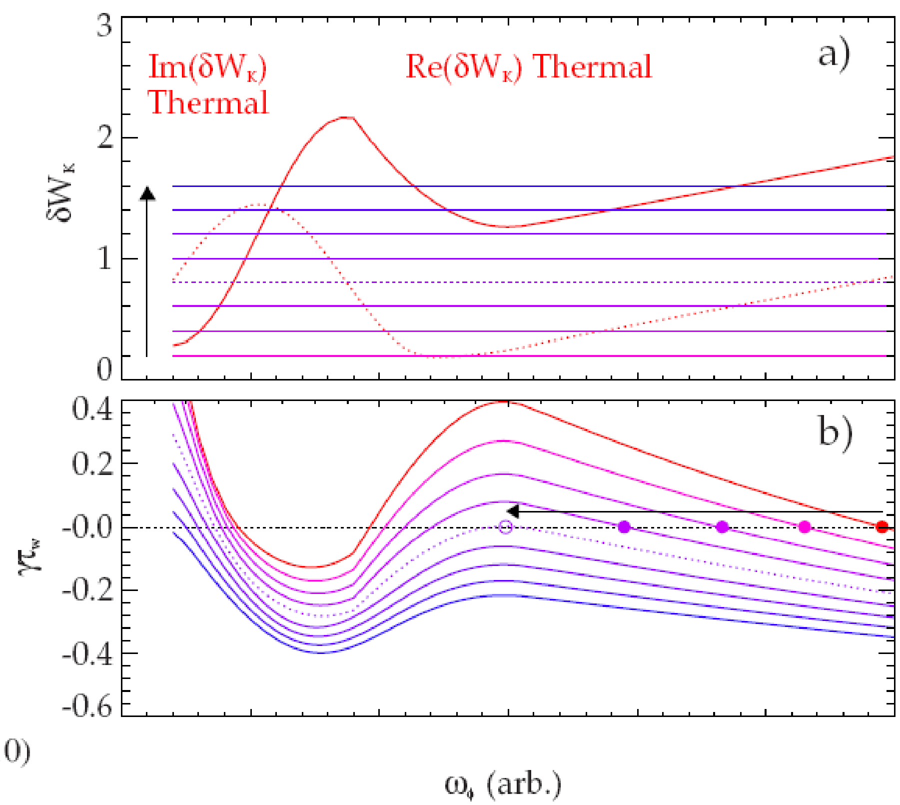
MISK results in good qualitative and near quantitative agreement with NSTX (and DIII-D) experimental observation

- Predict band of marginal stability at experimental rotation and collisionality for plasmas with rotation intermediate between $\omega_{precession}$ and ω_{bounce}



Berkery PRL **104,** 035003 (2010)

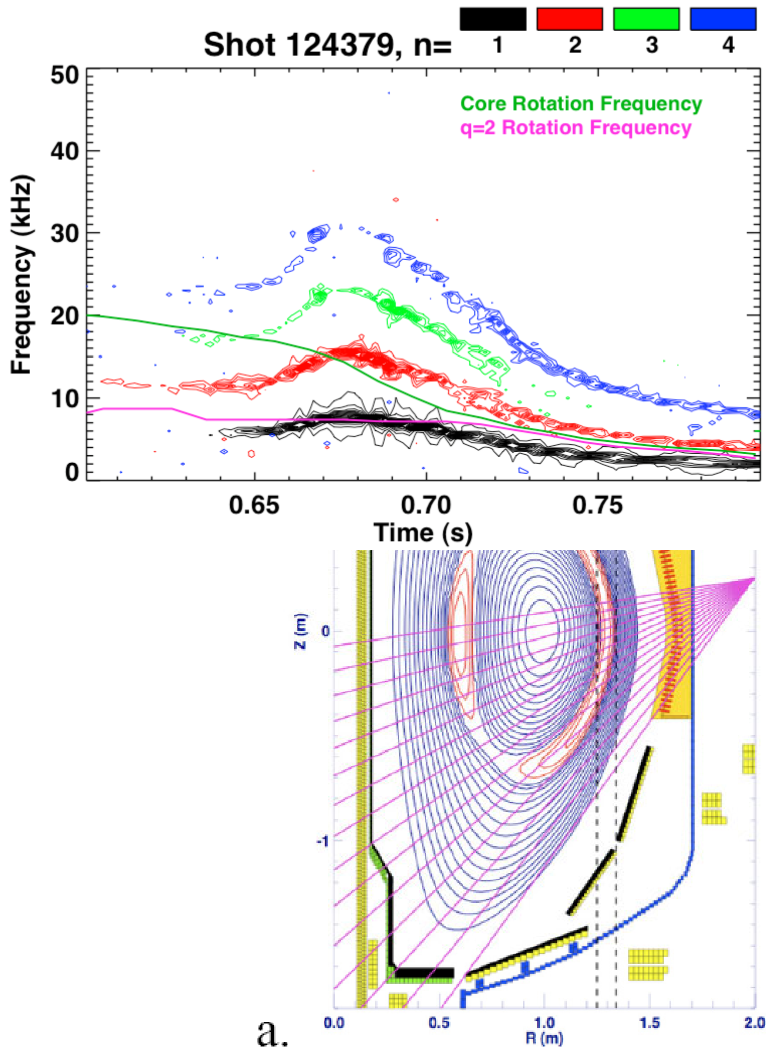
- Increased fast ion content predicted to reduced growth rate



- Now investigating effects of more accurate fast-ion distribution functions
- Finite (large) orbit width of fast-ions on NSTX could also modify stability

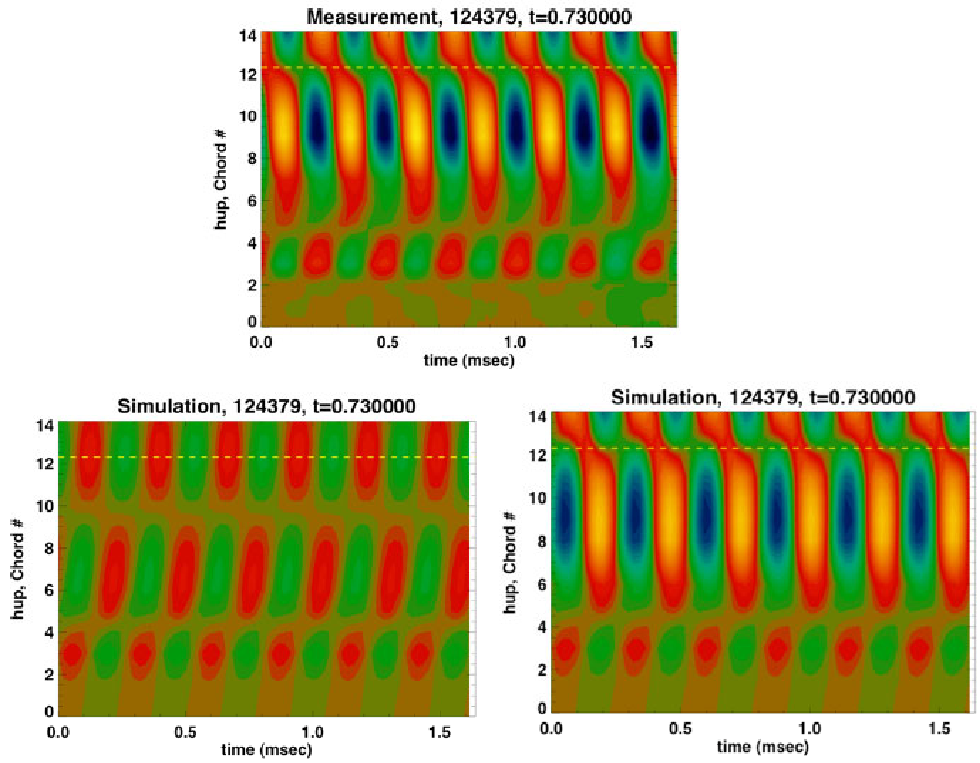
NSTX sometimes observes spontaneous n=1 mode onset with large 2/1+1/1 components

- Precursor-less n=1 mode can lead to rotation decay, disruption



a.

- SXR data indicates coupled 2/1 and 1/1 components during saturated phase

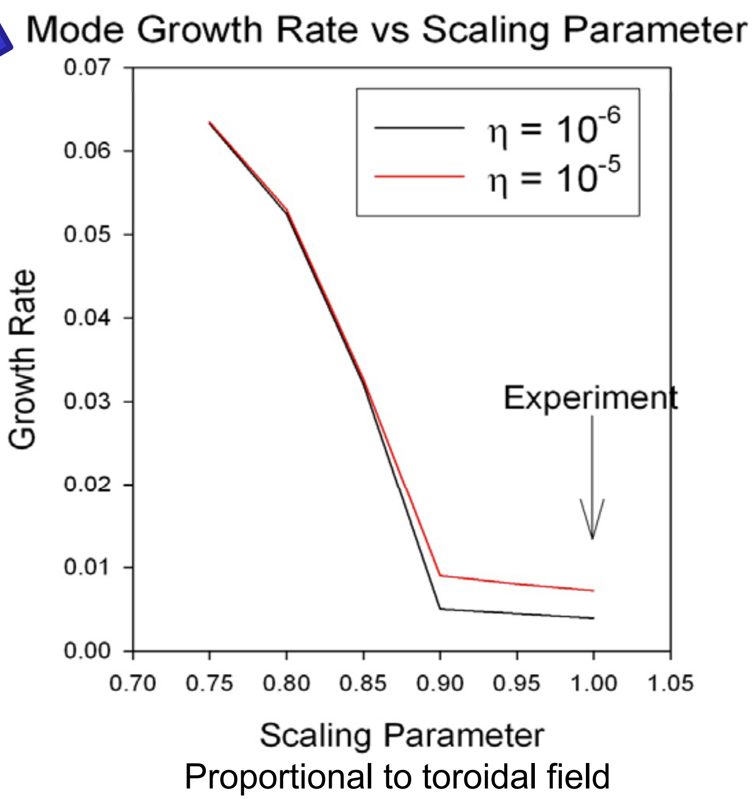


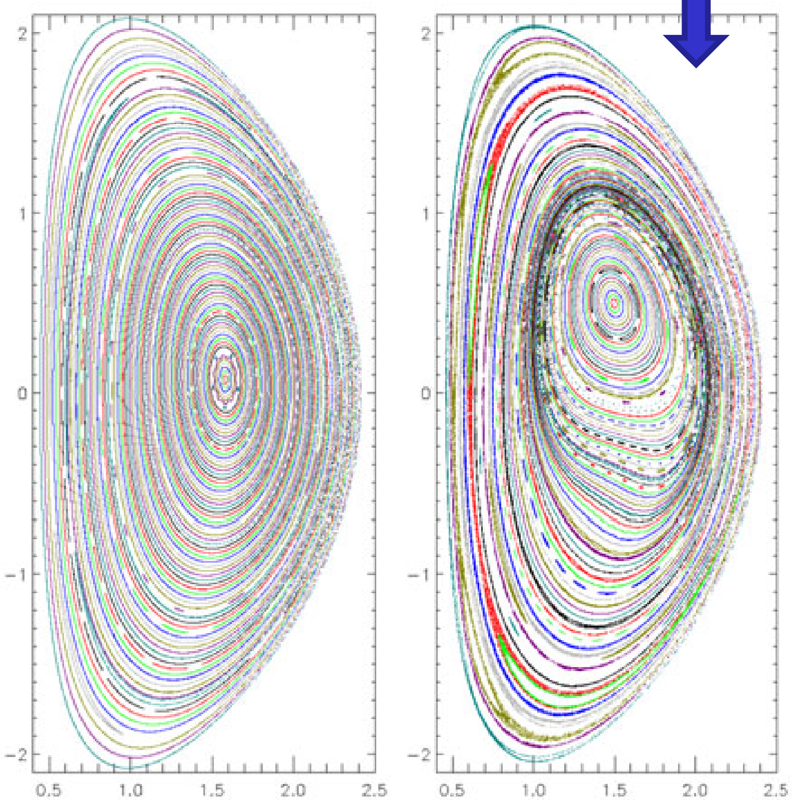
b. 2,1 only 2,1 + 1,1 pert

M3D simulations showing promise for understanding core mode onset, coupling in NSTX

Breslau

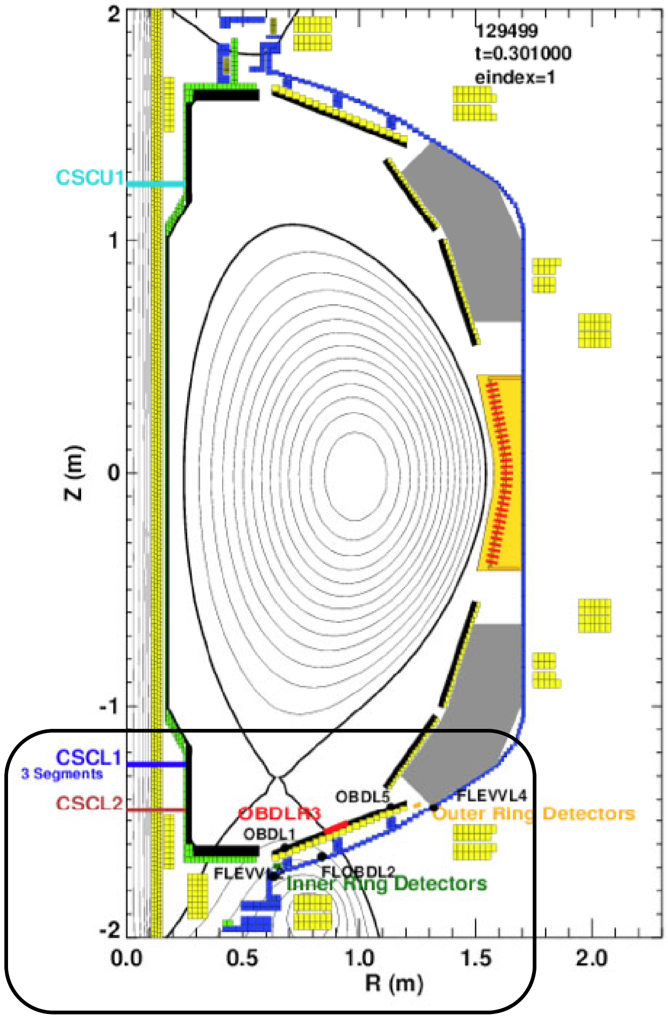
- Equilibrium has very flat central q profile with q_0 slightly above one.
- Linear numerical analysis with the M3D and M3D-C1 codes indicate marginal stability to an ideal n=1 MHD mode with both m=1 and m=2 components
- Saturated n=1 mode can set develop when q_0 slightly > 1.
- Can helically distort core flux surfaces and also drive m=2 islands.





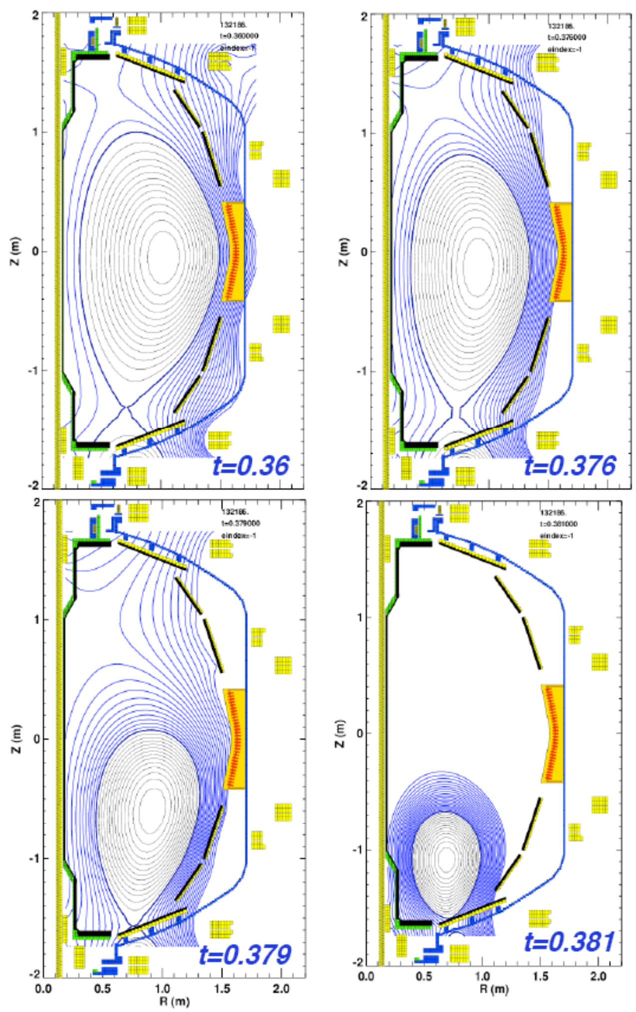
IAEA 2010

NSTX has implemented extensive halo current detection diagnostics for ITER support, preparation for NSTX Upgrade

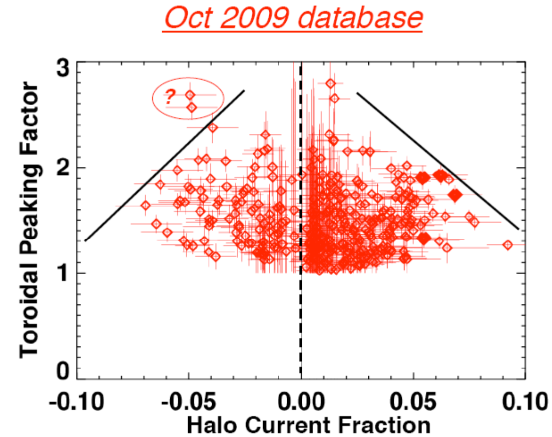


Halo Diagnostics

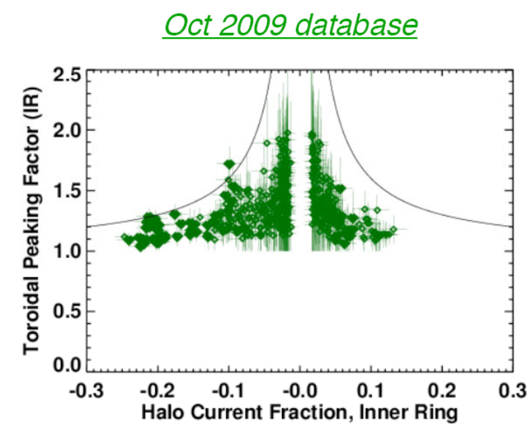
S.P. Gerhardt and J.E. Menard Nucl. Fusion 49 (2009) 025005



Typical downward VDE



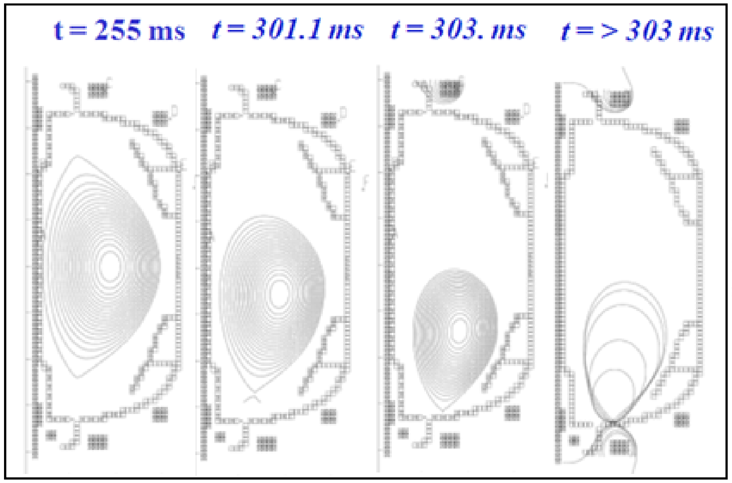
Halo Currents Through Outboard Divertor Row #3 Tiles



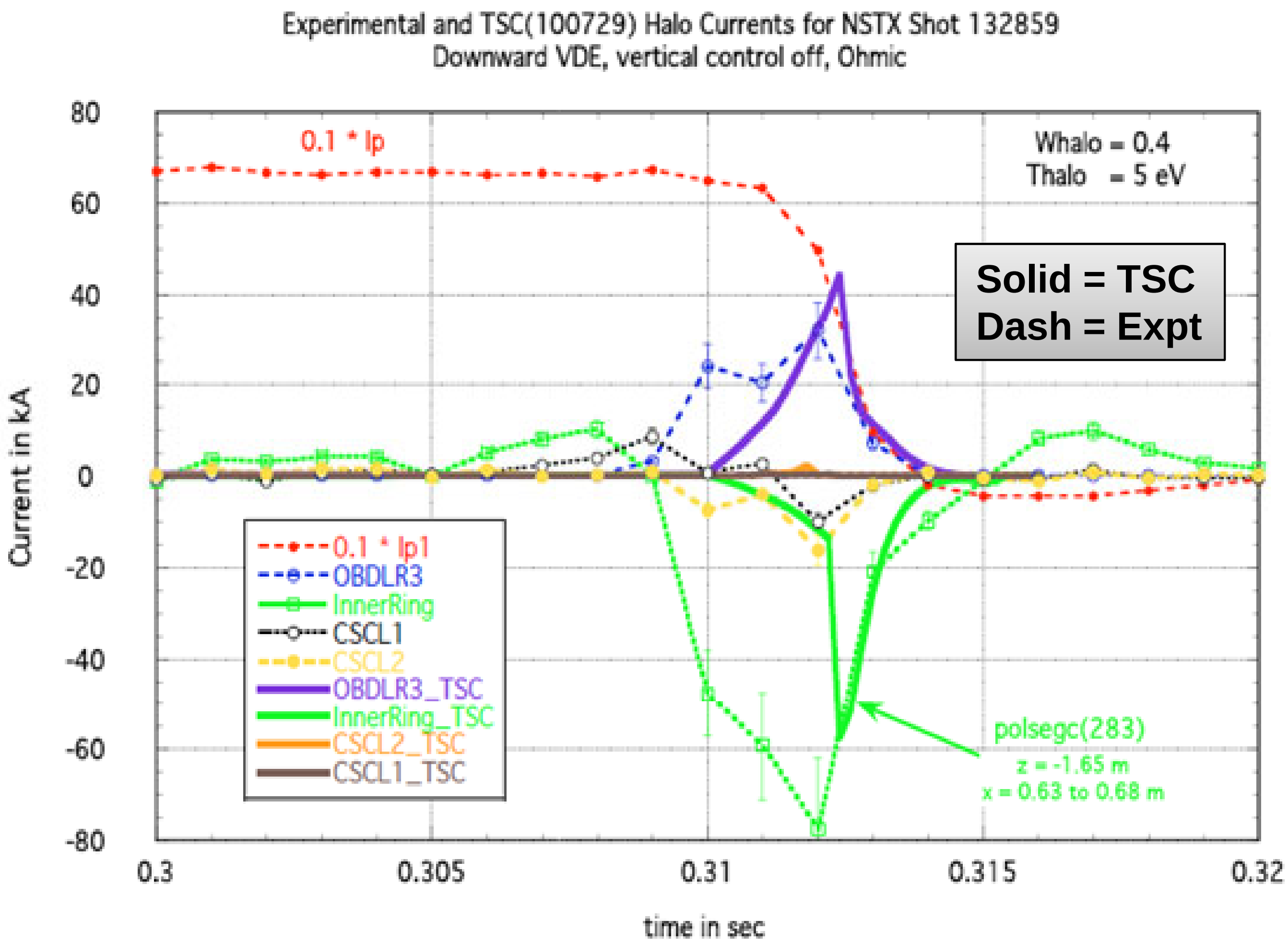
Halo Currents at Vessel Bottom

Preliminary TSC simulations show approximate agreement in halo current magnitude

Jardin



TSC Simulation



- But here are differences between TSC and experiment for halo time-evolution
- Now being investigated, and will be extended to 3D studies with M3D
- Results sensitive to initial conditions – may need to treat statistically

MHD research opportunities

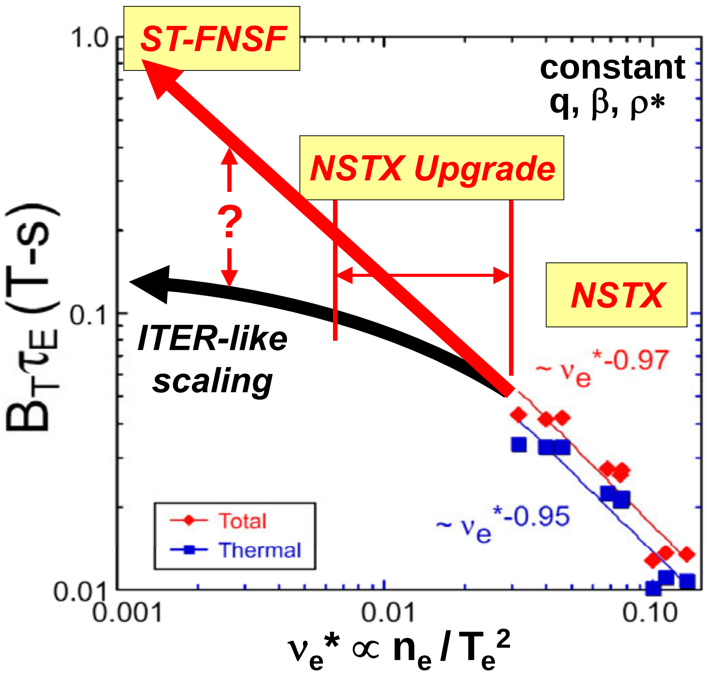
- Kinetic stability of Resistive Wall Mode
- Kink and tearing mode triggering, saturation, coupling to other modes
- Disruption dynamics and modeling

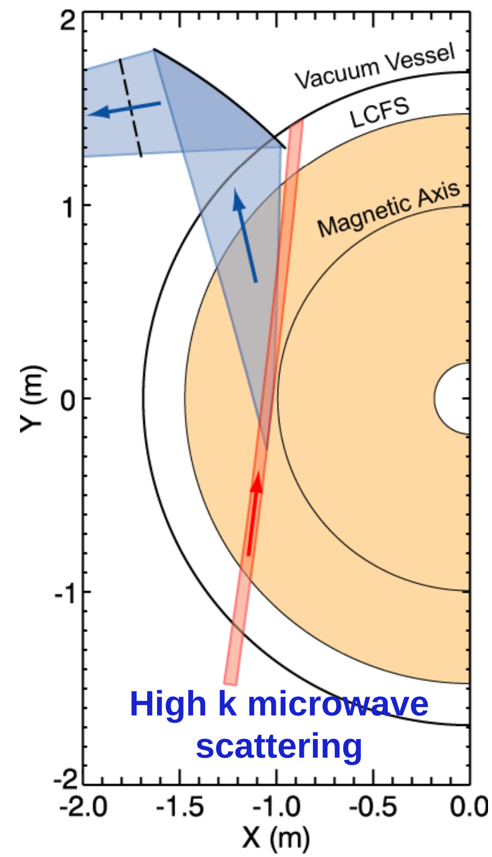
NSTX Research Areas

- Macroscopic Stability
- Transport and Turbulence
- Waves and Energetic Particles
- Lithium Research, Boundary Physics
- Plasma Formation and Sustainment

Understanding electron thermal transport is a top priority for both ST and tokamak programs

- ST has unique physics regime which can provide new insights into transport physics
  - high β, strong E×B flow shear, large ρ_e
  - high-k scattering diagnostic for electron gyro-scale fluctuations
- Transport mechanism understanding for tokamaks, especially for electrons, is limited
- NSTX can achieve neoclassical ion transport due to strong rotational flow shear, but electron transport is anomalous and high

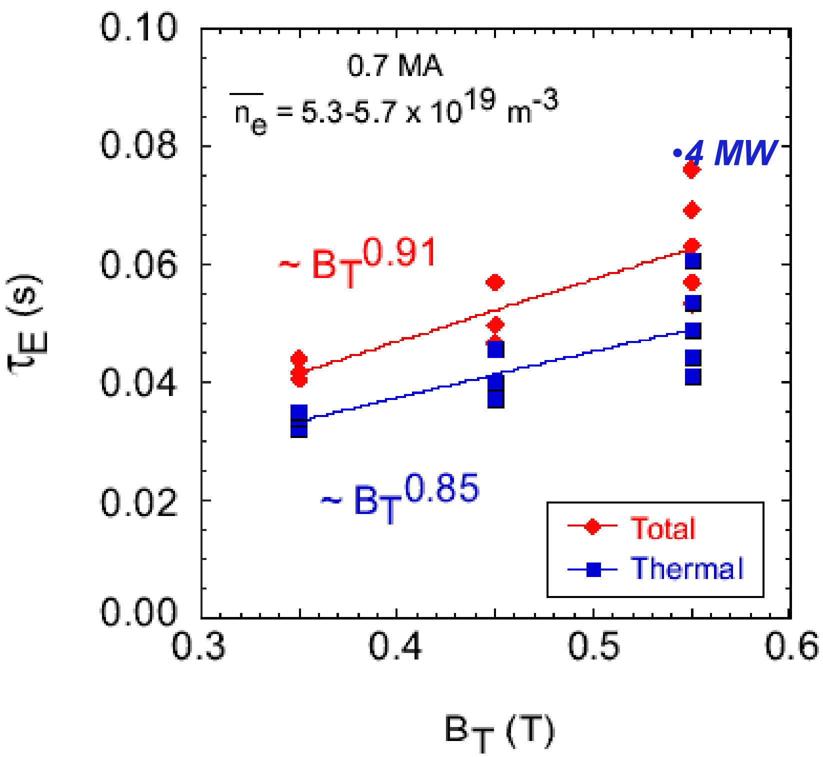




Additional understanding of ST transport at low collisionality is required for future devices

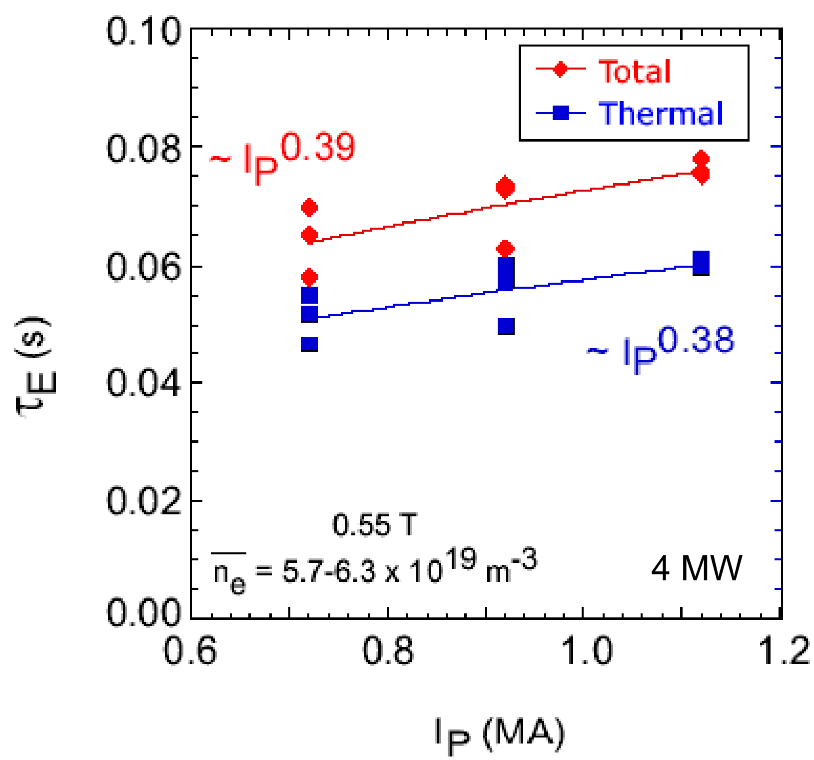
Experiments in NSTX Have Revealed Confinement Scalings Different From Those at Higher Aspect Ratio

- *Strong dependence of τ_E on B_T*



$\tau_{E,98y,2} \sim B_T^{0.15}$

- *Weaker dependence on I_p*

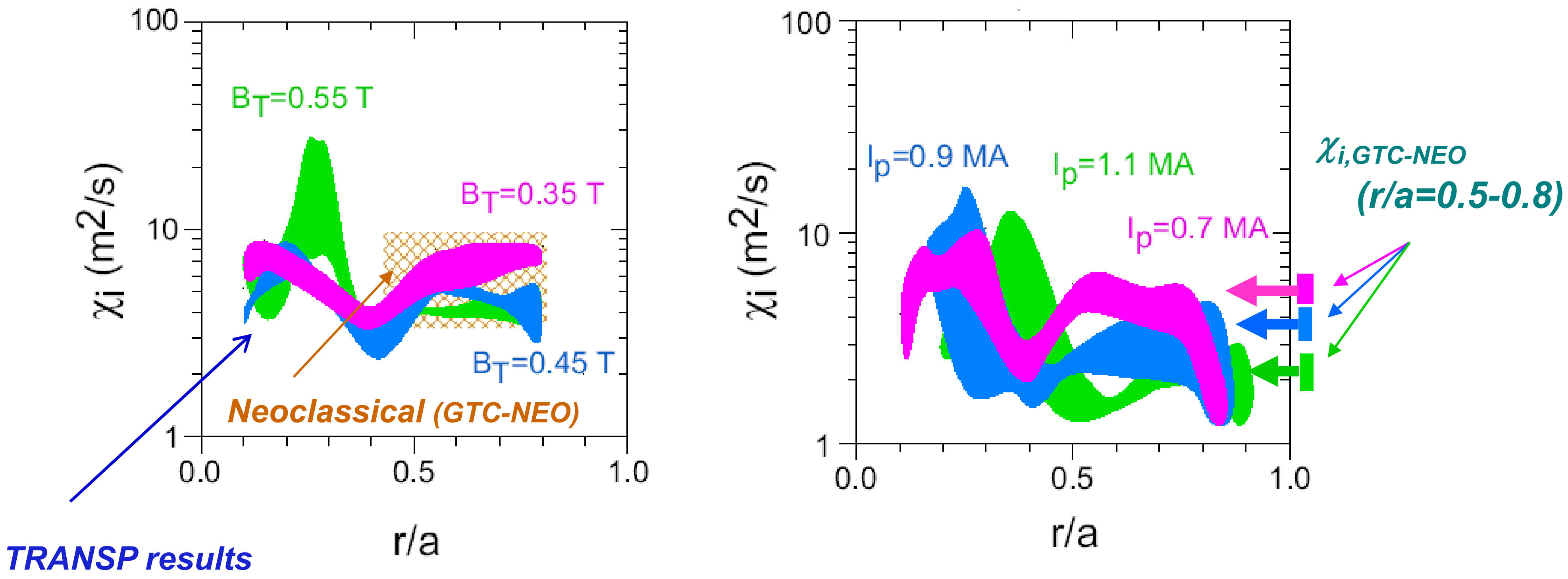


$\tau_{E,98y,2} \sim I_p^{0.93}$

GTC-NEO Calculations Have Shown That the Ion Thermal Transport is Neoclassical

Compare TRANSP results to GTC-NEO

- GTC-NEO includes finite banana width effects (non-local)



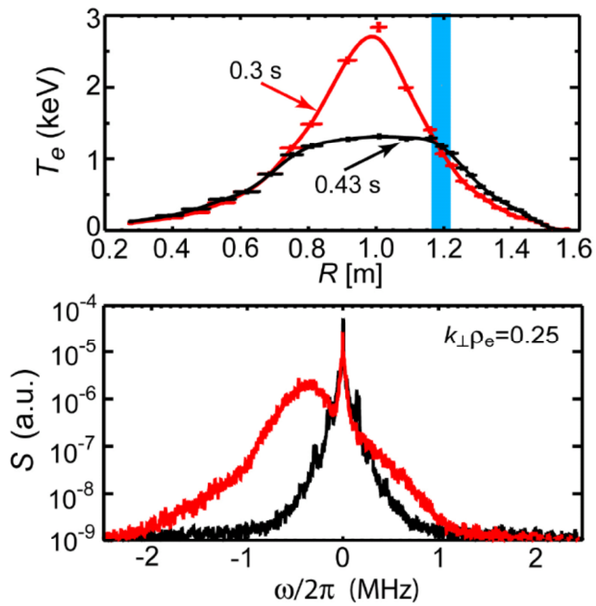
Ion neoclassical determines I_p scaling

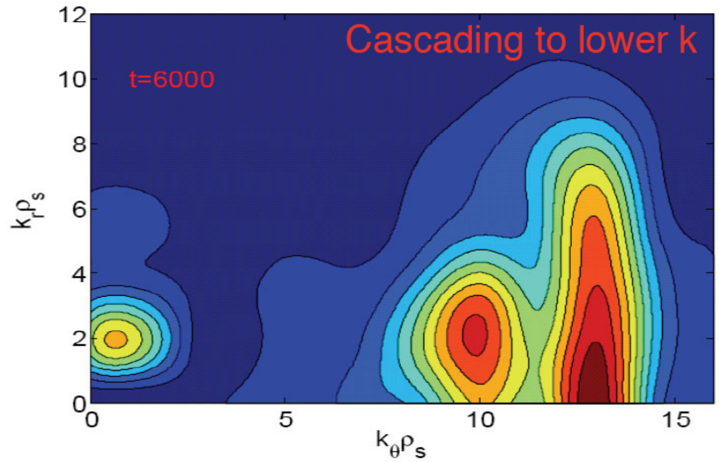
(Rewoldt, Wang)

First global, Nonlinear ETG Simulations Carried Out for Direct Validation Against Experimental Measurements in NSTX

- GTS calculations have indicated that ETG modes can contribute significantly to anomalous electron transport

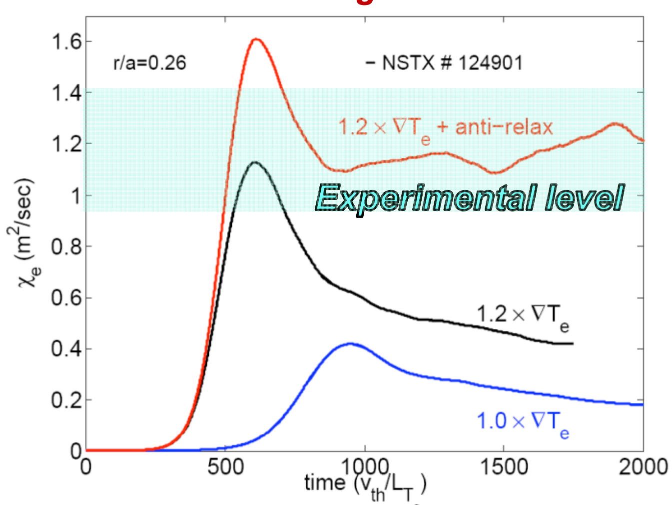
(Wang, Ethier)





Non-linear spectrum dynamics (energy flow to lower-k electron GAM and zonal flow)

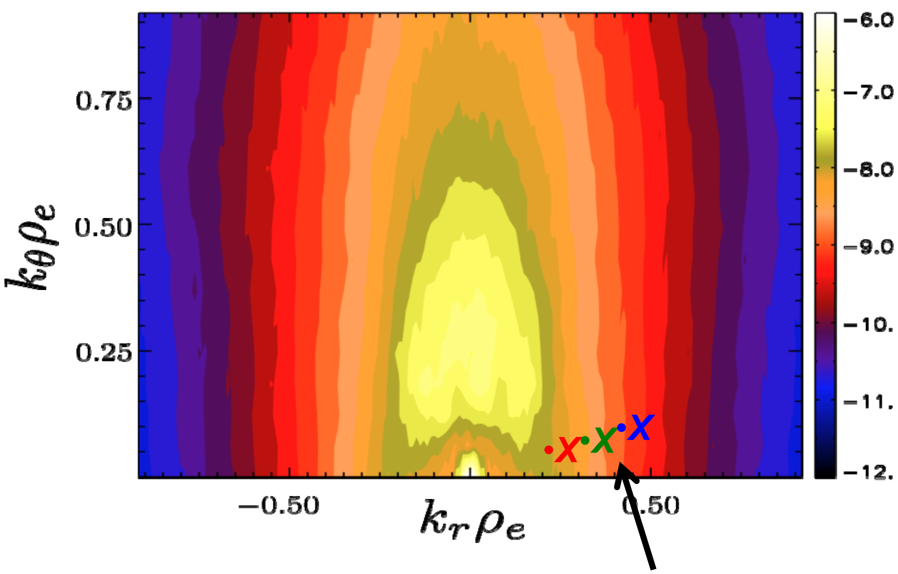
"Anti-relax" - maintain ∇T_e until convergence



- Transport levels/turbulence spectra are comparable to experimental levels ($k_r^{-(2.5\ to\ 5.3)}$ GTS, $k_r^{-4.5}$ in expt)

GYRO Calculations Have Been Done Also to Explore the Source of the Electron Thermal Transport

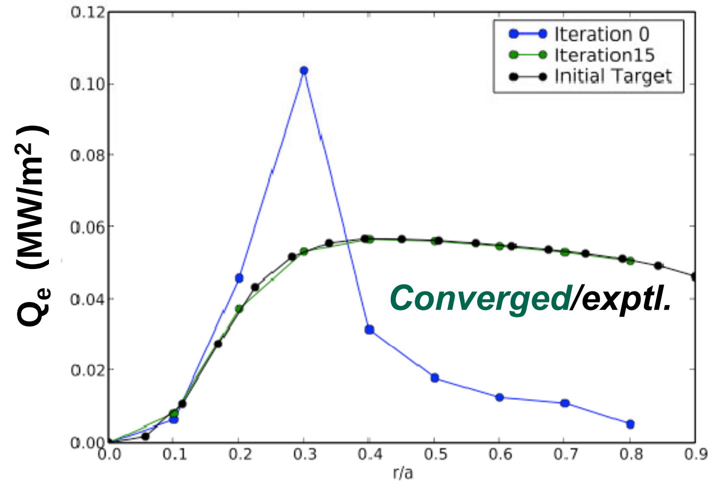
- GYRO indicates high-k diagnostic may not see peak of ETG

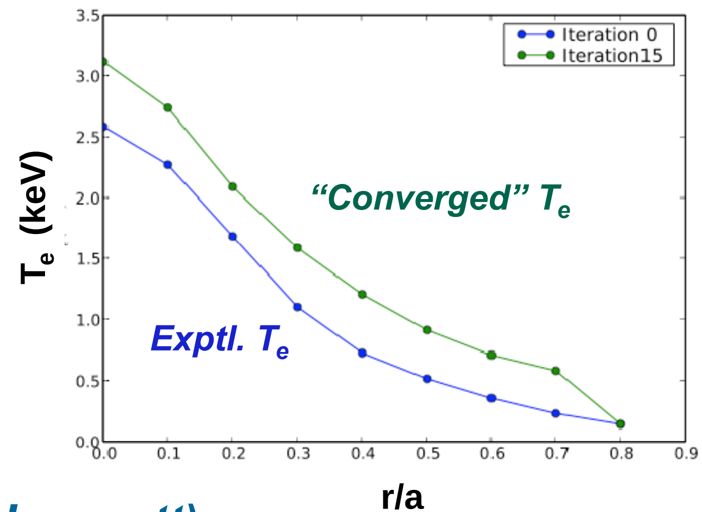


•Approx. high k locations:
•Ch. 3 Ch. 4 Ch. 5

- Synthetic diagnostic under development (Poli et al., PoP in press)

- TGYRO/TGLF predicts steady state NSTX heat flux profiles (ETG & TEM contribute to transport)

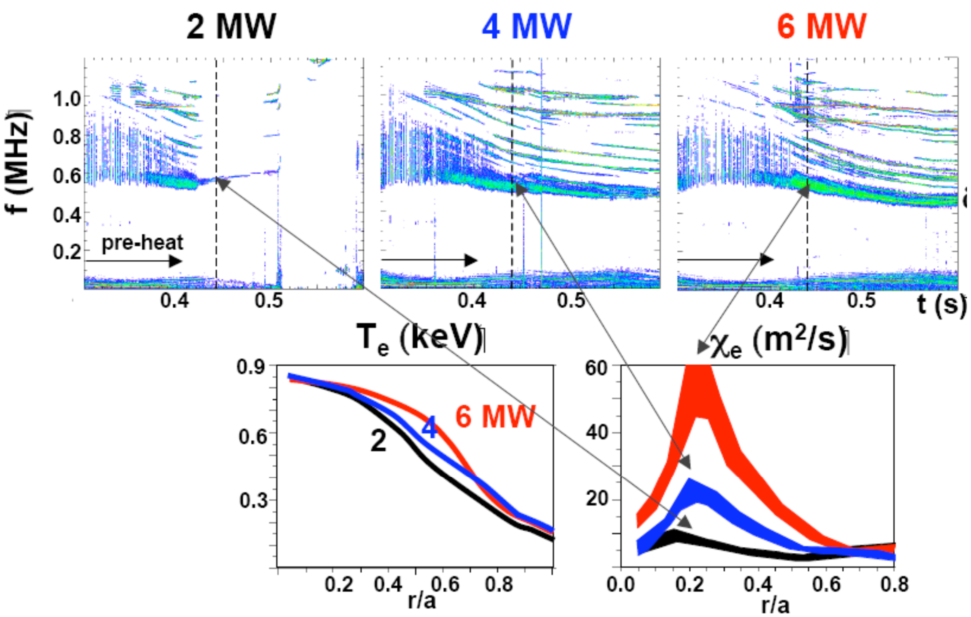


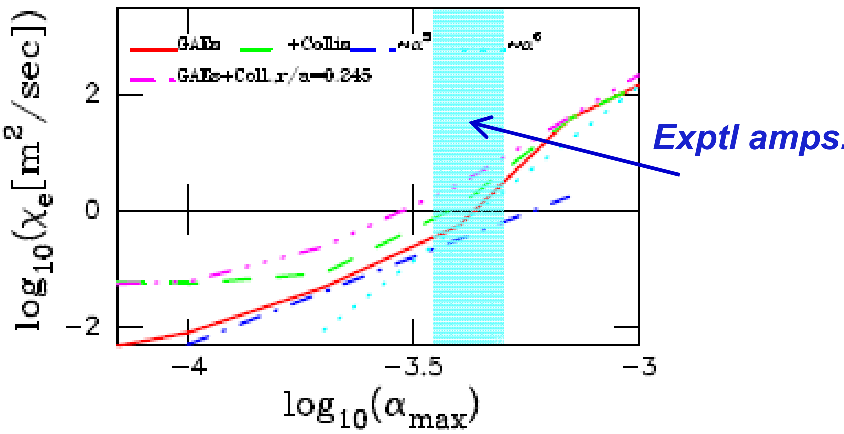


(Peterson [grad student], Hammett)

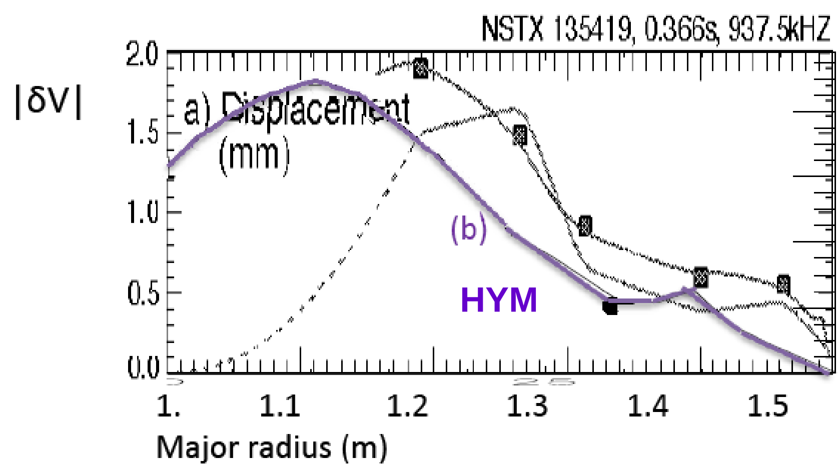
At High Power, Global Alfvén Eigenmodes Can Drive Electron Transport in the Core Region

- Absence of ∇T_e, ∇n_e driving terms for ETG, TEM, μtearing, etc.
- Theory indicates multiple GAE modes can produce significant electron transport from this core region





(Theory+ORBIT: Gorelenkov, White, NF 2010)



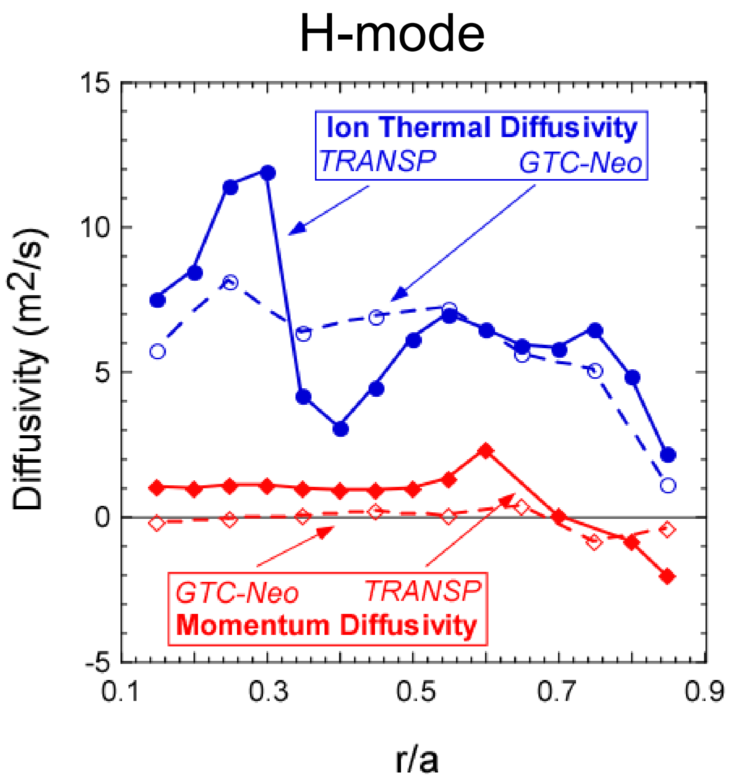
Non-linear HYM calculations yield GAE mode structure, saturated amplitudes similar to that observed experimentally (Belova, APS Invited 2010)

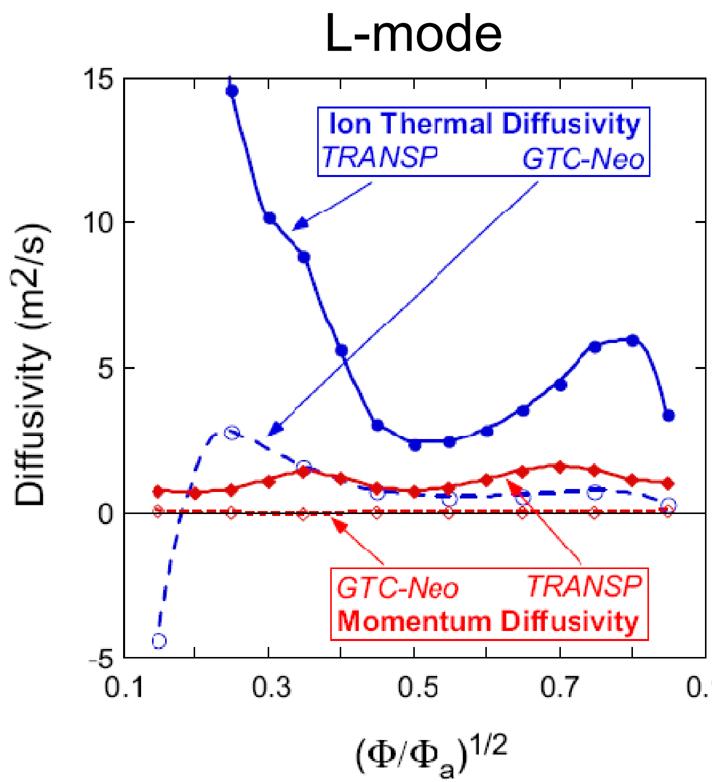
- ***Plan to use as basis for studying effect of GAE on electron transport***

Momentum Transport Studies in NSTX Have Benefited from Both Analytical and Computational Efforts

- $\chi_\phi >> \chi_{\phi,neo}$ in both H- and L-mode plasmas, irrespective of $\chi_i/\chi_{i,neo}$

(Rewoldt, Wang)



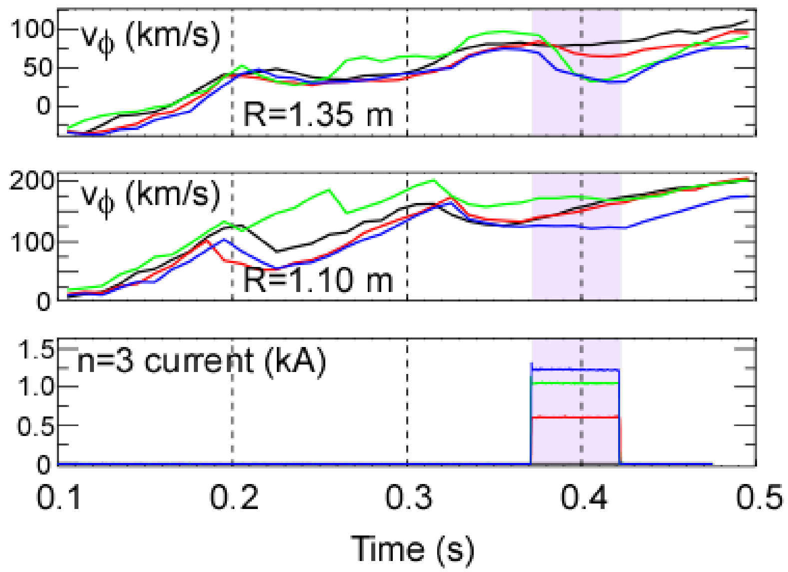


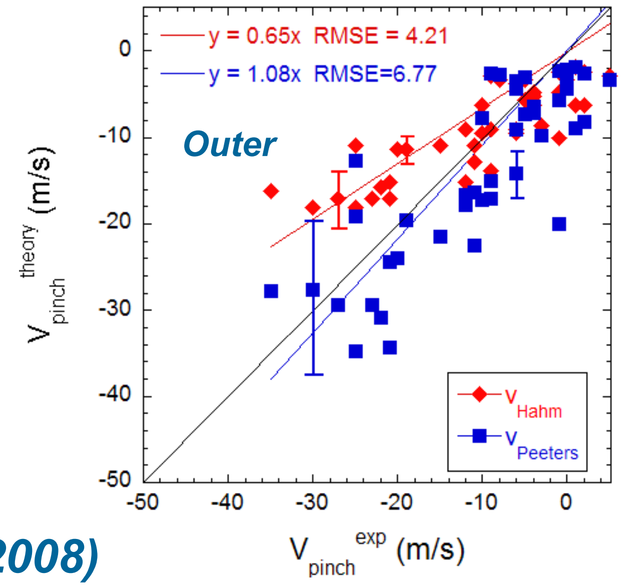
Is χ_ϕ controlled by low-k turbulence?

Motivated perturbative experiments to determine

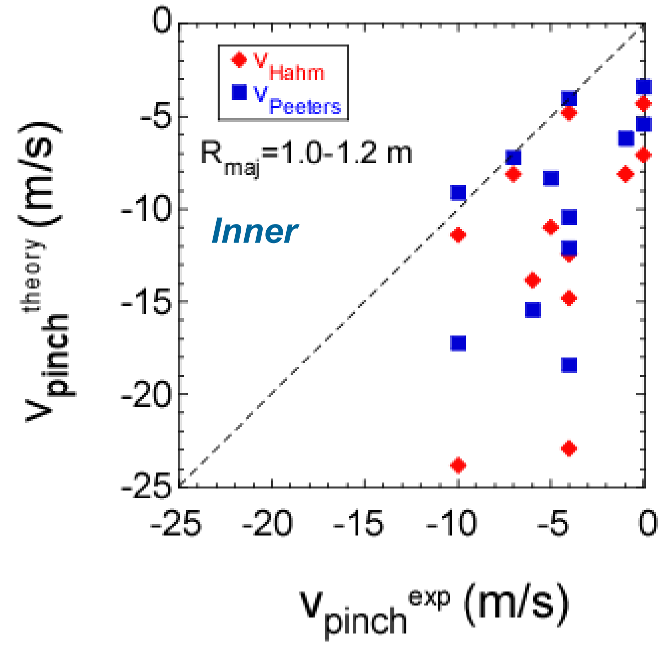
Inferred Momentum Pinch is Consistent With Predictions Based on Low-k Turbulence Theory

n=3 braking pulses to determine χ_ϕ, v_{pinch} in outer region, NBI pulses for inner region





(Hahm, PoP 2008)



Why is there a difference between theories at high v_{pinch}? ***L_n dependence***

Why does theory match in outer region better than in core? ***ITG/TEM stable in core***

Core turbulence research opportunities

- Electron and ion thermal transport
  - Predict experimental/NSTX profiles?
    - Neoclassical, ITG, TEM, μ-tearing, GAE, ETG
- Momentum transport
- Particle transport – main ion, impurities

NSTX Research Areas

- Macroscopic Stability
- Transport and Turbulence
- Waves and Energetic Particles
- Lithium Research, Boundary Physics
- Plasma Formation and Sustainment

NSTX accesses broad range of fast ion parameters, and a broad range of fast particle modes

- Figure at right illustrates NSTX operational space, as well as projected operational regimes for: ITER (α's only), ST-CTF (α+NBI), ARIES-ST (α's)
- Also shown are parameters where typical fast particle modes (FPMs) have been studied.
- Conventional beam heated tokamaks typically operate with $V_{fast}/V_{Alfven} < 1$.
- CTF in avalanche regime motivates studies of fast ion redistribution
  - ITER with NBI also unstable to AE
- Higher ρ^* of NSTX compensated by higher beam beta

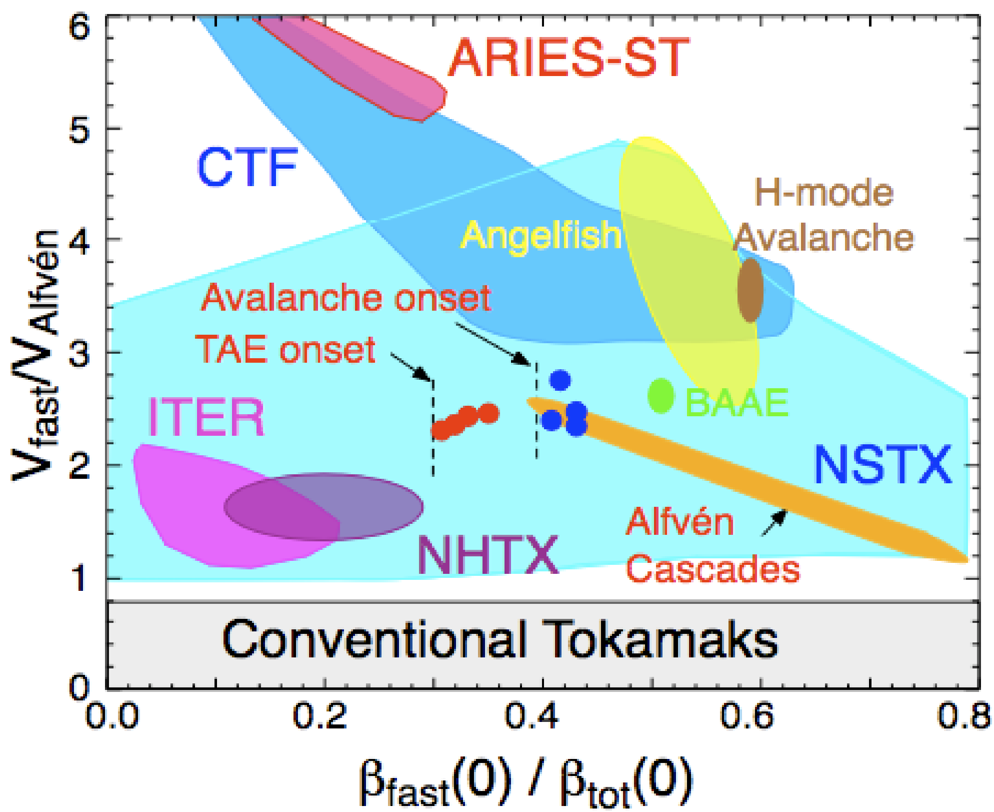
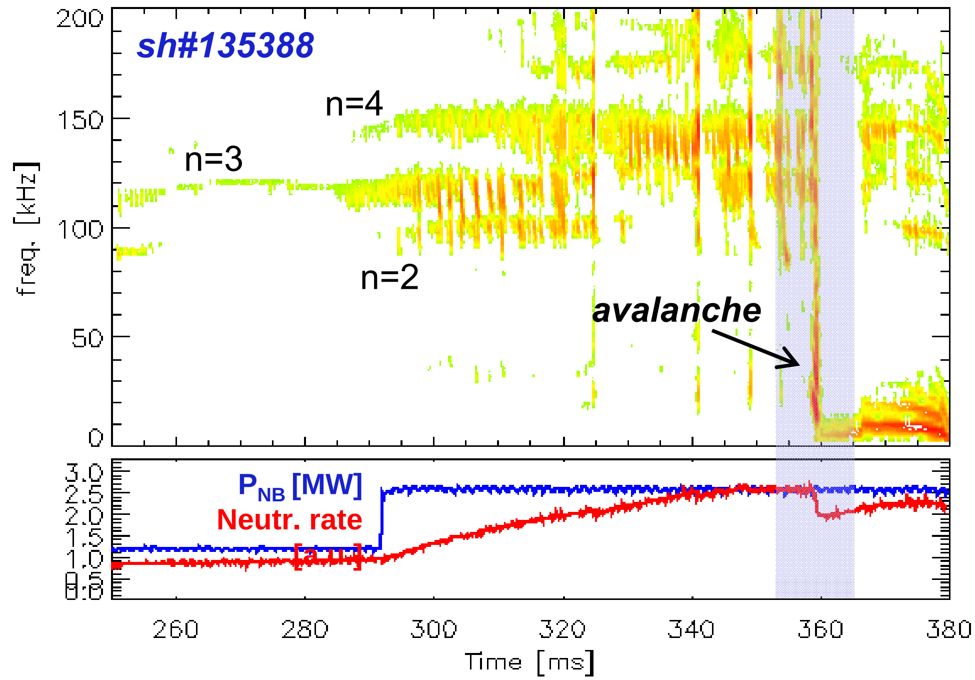
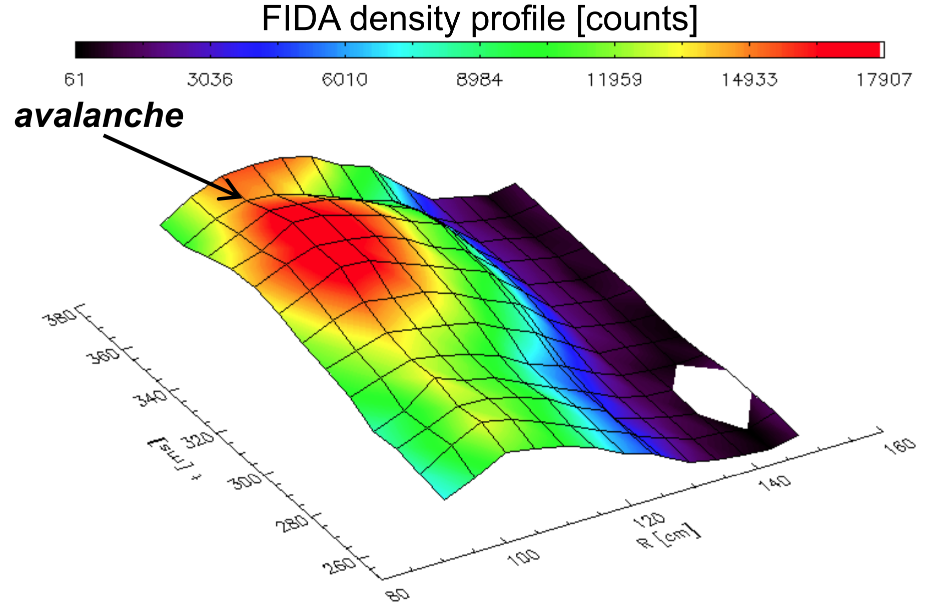


Figure above is simplified picture - there are other dependences, such as *q* profile, ρ^*

Non-linear evolution of multiple TAE modes into *avalanches* causes large fast ion losses

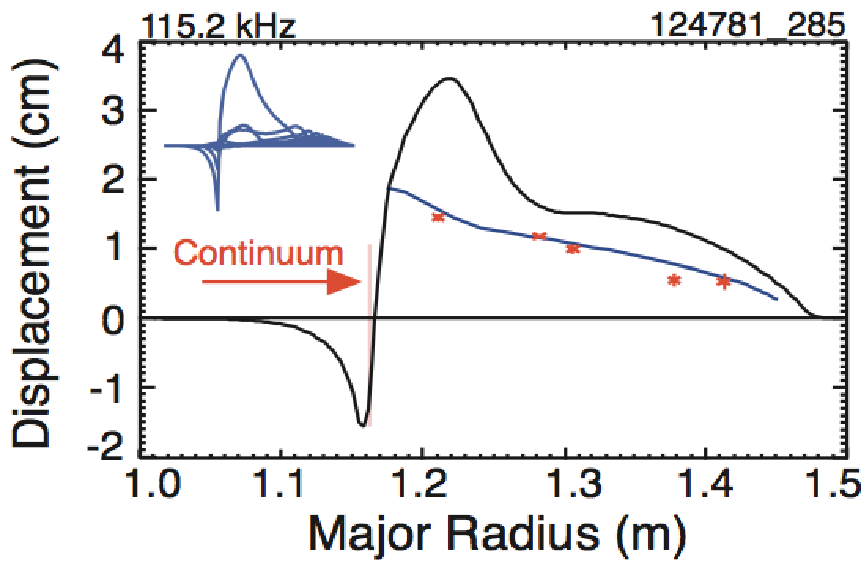
- Multiple TAEs are simultaneously destabilized by NB injection
  - No (detectable) losses caused by weakly chirping modes, but…
  - Eventually, modes undergo strong frequency sweep with increasing amplitude: ***avalanche***
    - Neutron rate drops up to 30-35%
    - Expected loss mechanism in ITER
- TAE studies will be extended to H-mode plasmas in 2010-2011
  - Mode structure from BES, interferometer
  - Use BES/high-K to search for kinetic effects, e.g. continuum damping through kinetic Alfvén waves (KAWs)

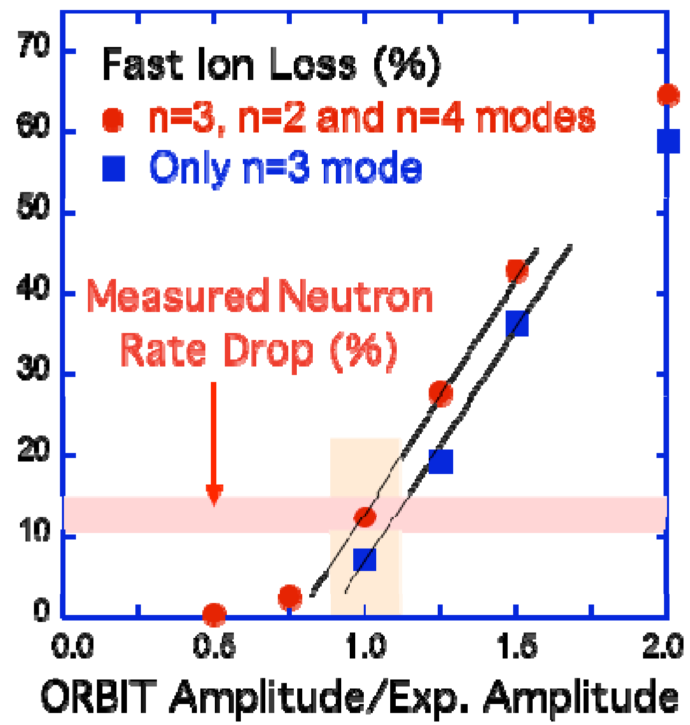




Modeling TAE avalanches with linear codes (NOVA-K, ORBIT) can reproduce measured level of fast ion losses

- Procedure:
  - Calculate linear eigenmodes with NOVA-K
  - Match and select "observed" modes based on measured mode structure (reflectometer)
  - Rescale amplitude according to measured displacement
    - 2x mode amplitude is needed when compressibility is included (2009)
  - Calculate fast ion loss with ORBIT
    - Including potential fluctuations enhances losses (2009)
- Simulated losses agree with experiment
  - Comparable mode amplitude
    - All dominant modes must be retained
    - No simple (linear) dependence of losses upon mode amplitude
- Main limitations:
  - *Linear*; *not self-consistent*; mode amplitude/frequency in ORBIT adjusted to mimic data



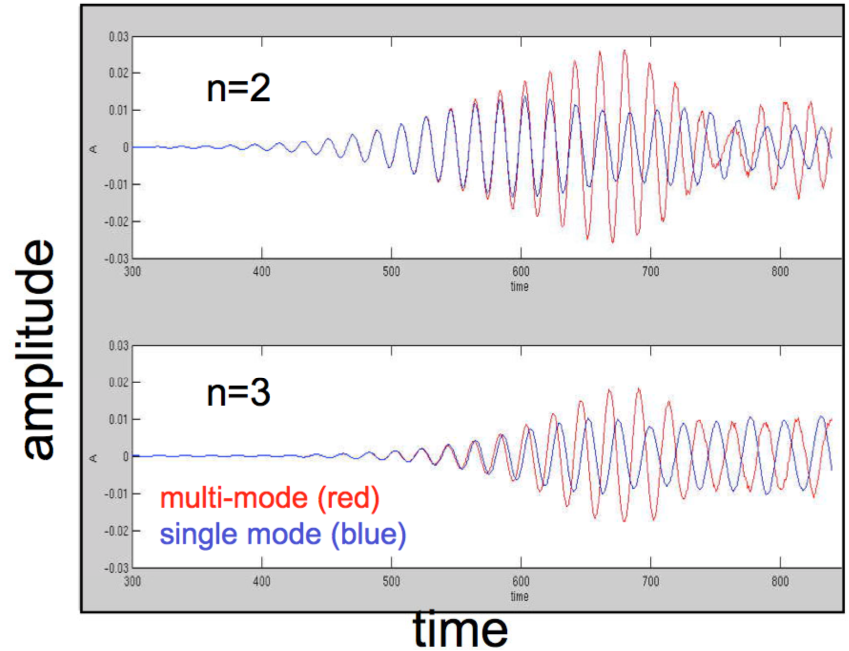


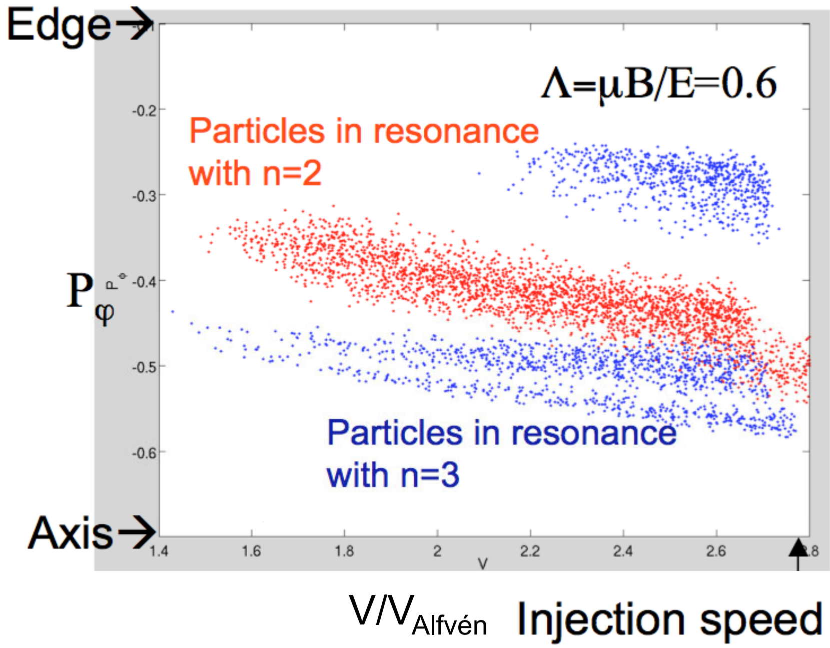
Phys. Plasmas **16**, 122505 (2009)

Benchmarking non-linear, self-consistent code M3D-K is planned to improve predictive capability

- Past results highlight importance of multi-mode dynamics
  - Enhanced saturation level is larger in multi-mode simulation
  - Broad, overlapping resonance regions in phase space

- 2010-2011: dedicated experiments planned for validating the M3D-K code
  - Optimize measurements of mode structure(BES, reflectometer)
  - Initial focus on L-mode plasmas with weakly turbulent TAE activity
  - Use stability predictions (growth, damping rates) from linear analysis
    - Self-consistent stability calculation would require additional code development
  - Compare predicted mode structure and multi-mode (non-linear) dynamics of TAEs with experiment

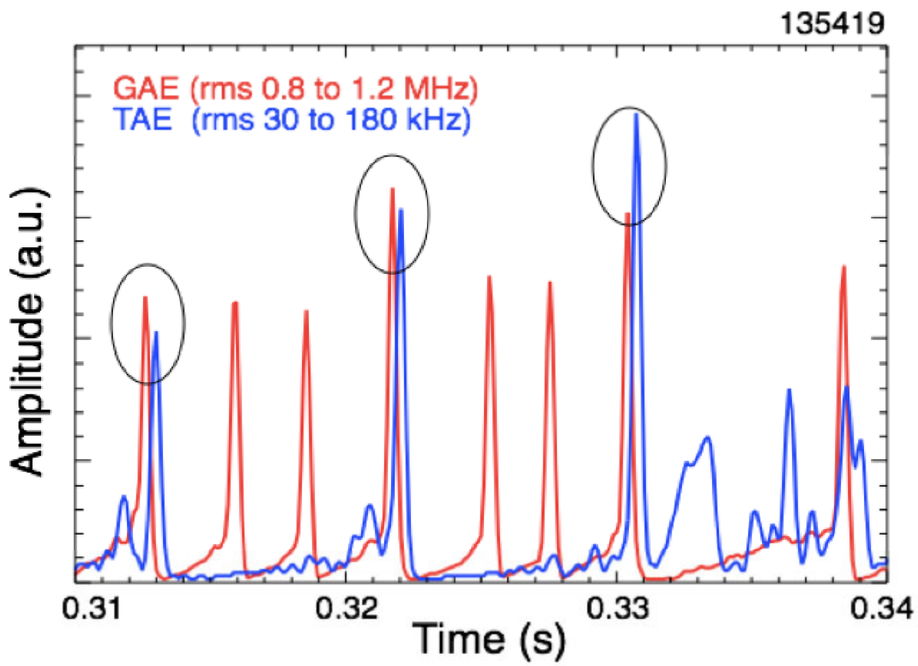


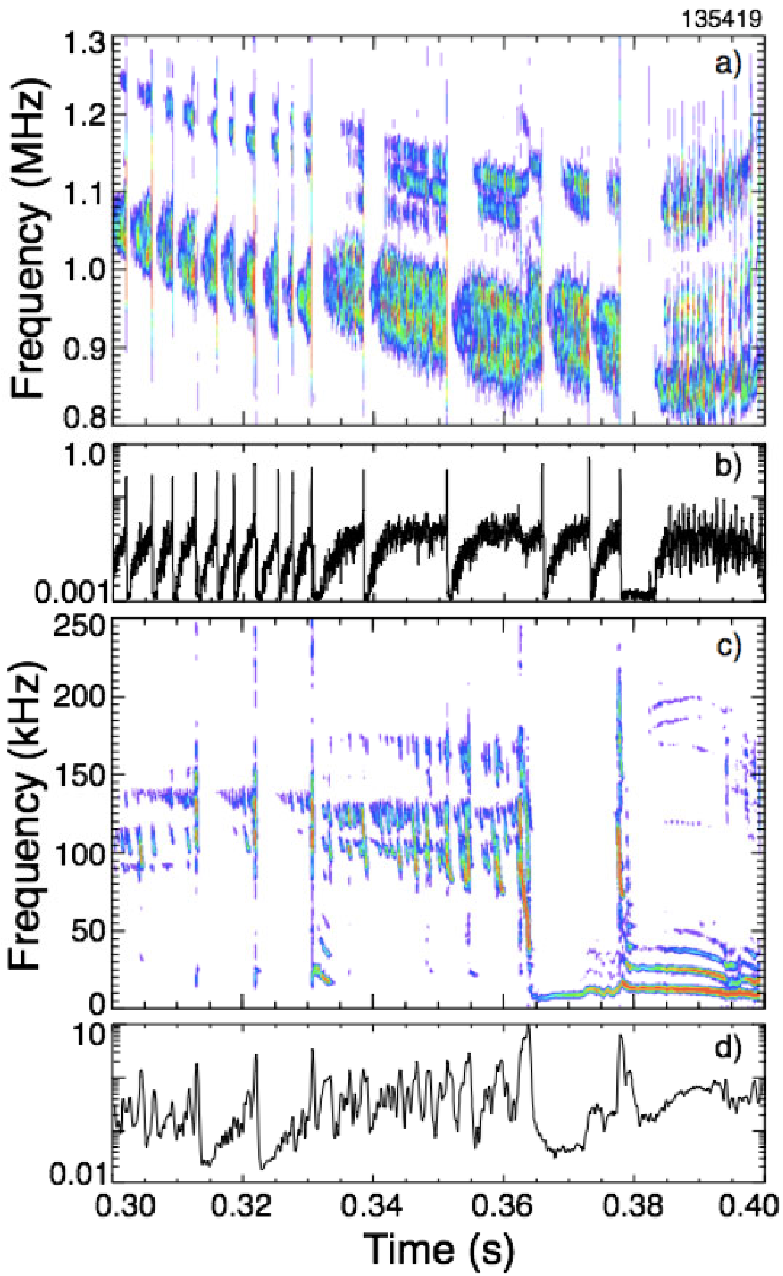


GAEs/CAEs show avalanching behavior that correlates with TAE avalanches at lower frequency

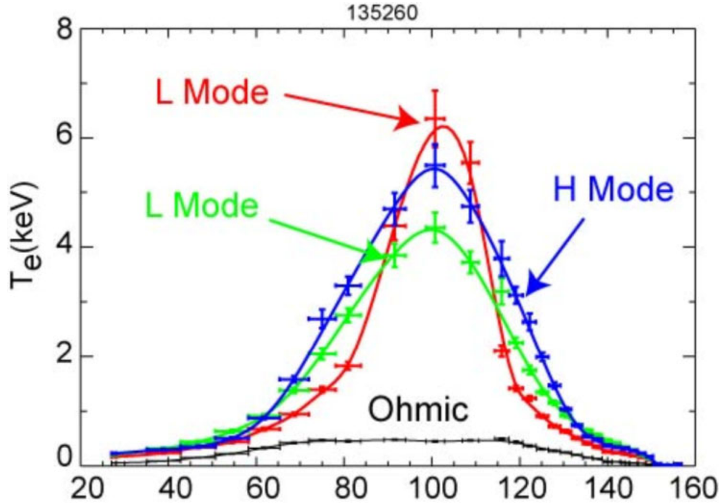
Bursts can trigger TAE/EPM avalanches

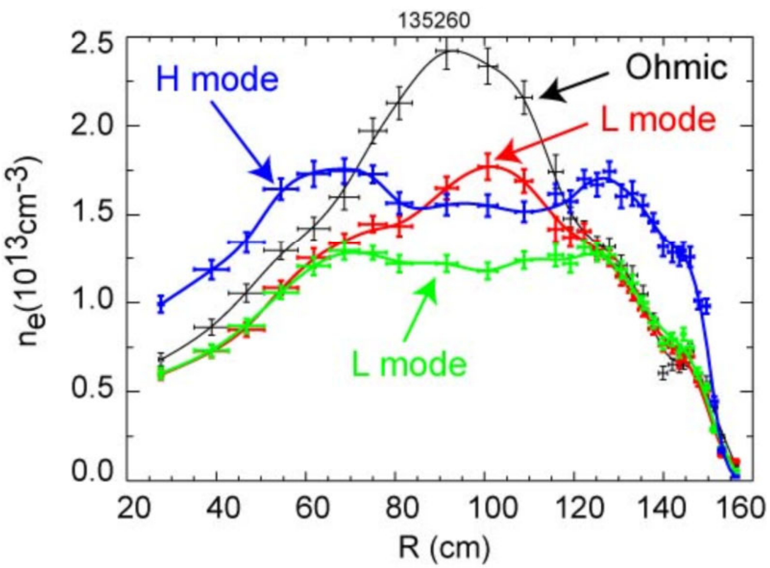
- Implies significant fast ion redistribution
  - Effects on fast ion confinement might be masked by dominant TAEs/EPMs
- Extend study to H-mode in 2010-2011
- Codes (HYM, M3D-K) may reveal underlying non-linear physics of mode-mode coupling (2011-2012)



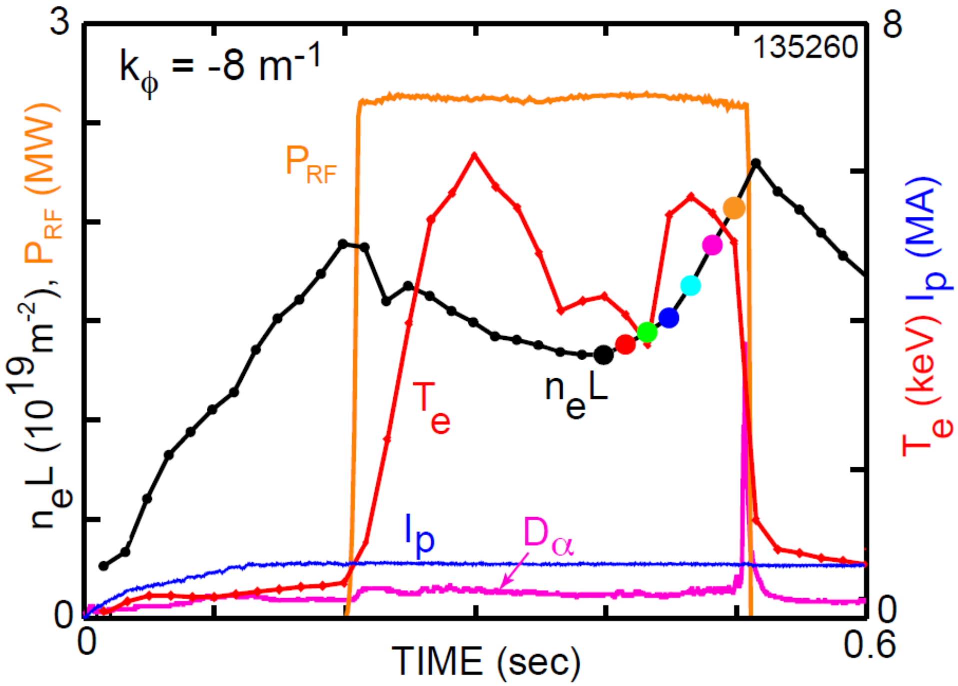


Improved Coupling Efficiency and Achieved Record T_e = 6.2keV Using Upgraded HHFW Antenna





Maintained coupling through L-H transition in presence of ELMs

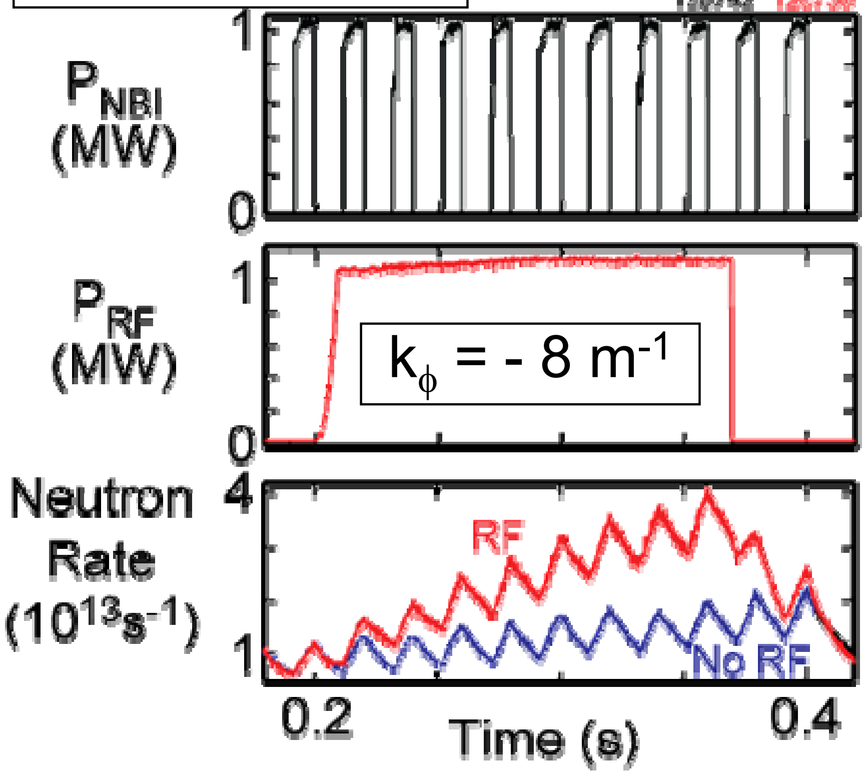


- Moved antenna grounding point to center of strap to reduce voltage per current
  - System quickly commissioned to previous power levels (2-3 MW)
  - Additional conditioning, combined with improved ELM discrimination should allow P_{RF} > 5MW

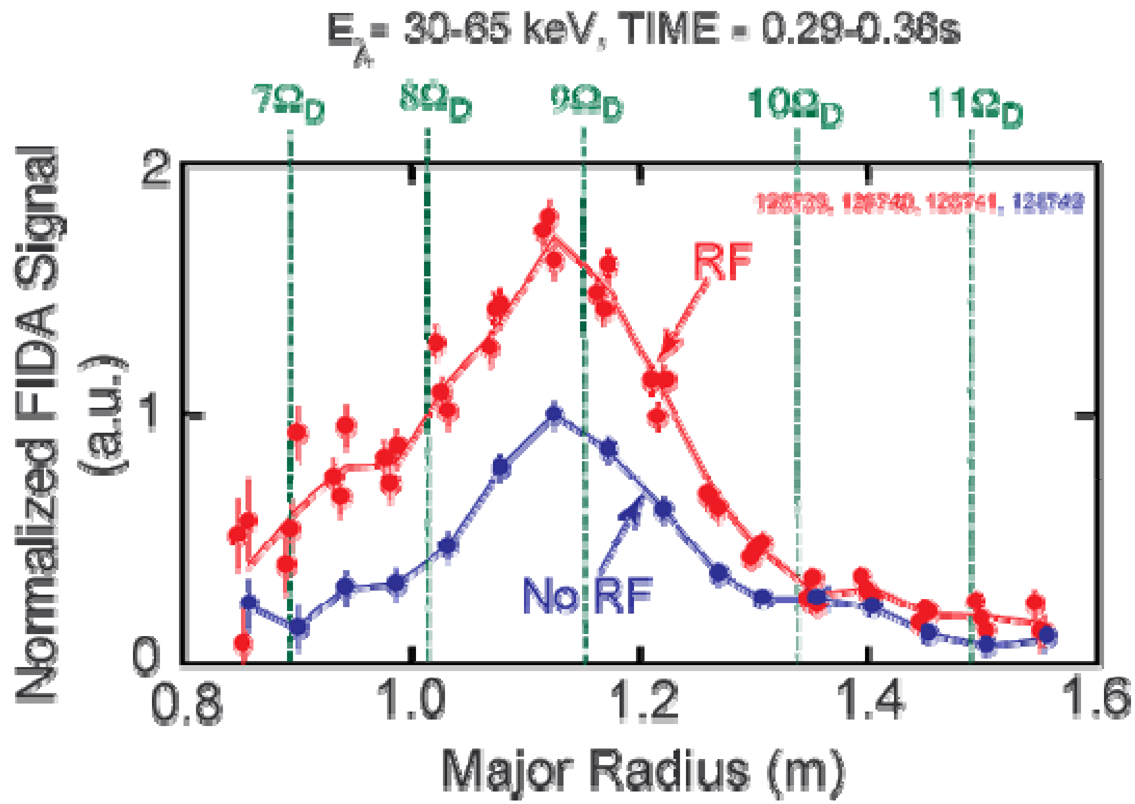
J. Hosea, P. Ryan, G. Taylor

Significant Interaction Between HHFW & NBI Fast-Ions Over Multiple Cyclotron Harmonics

$B_T(0) = 5.5$ kG



Fast-Ion D_α (FIDA) Measurements

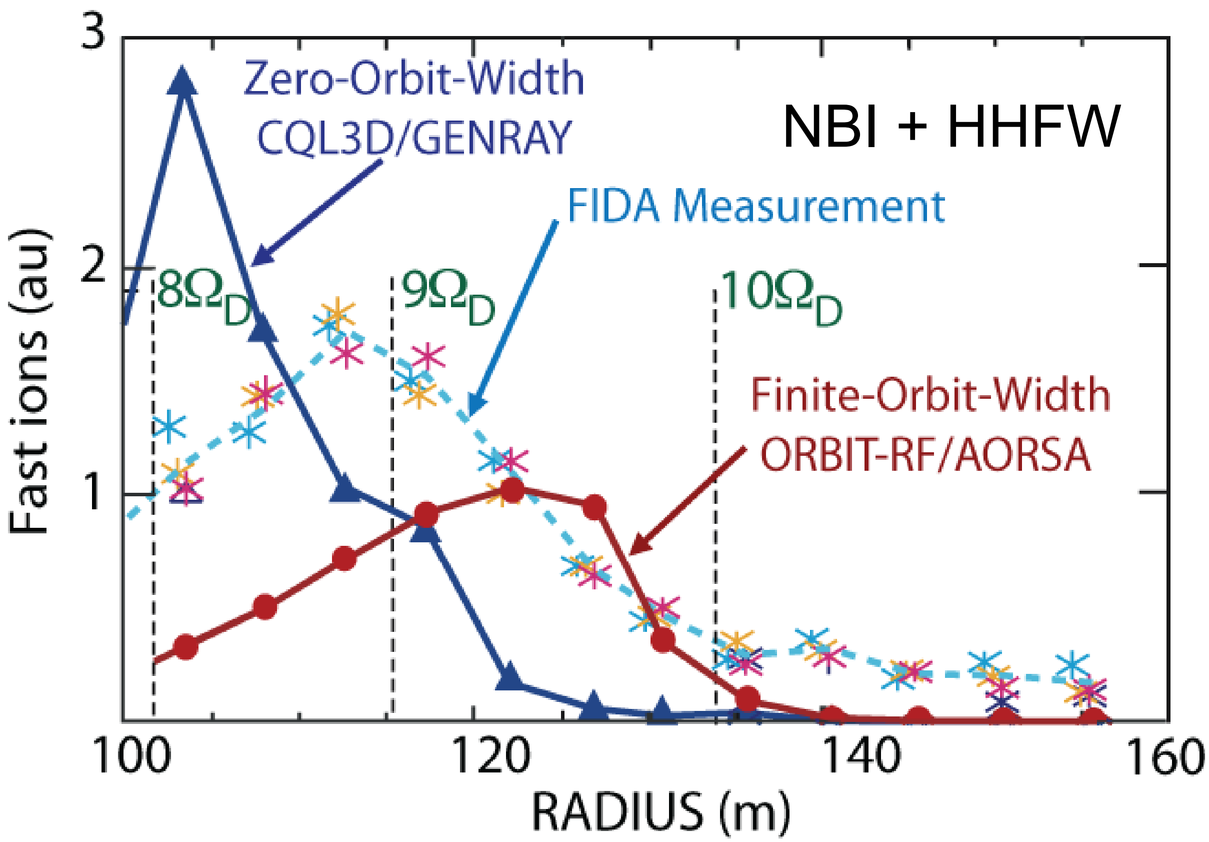


Measured acceleration of NBI fast-ions and large increase in neutron rate during HHFW + NBI plasmas

- As predicted originally by CQL3D/GENRAY

- Measured significant enhancement & broadening of fast-ion profile when HHFW power is applied

Finite Lamor Radius & Banana-Width Effects Broaden Fast-Ion Profile in NSTX



- Zero-orbit-width Fokker-Planck CQL3D/GENRAY ray tracing model predicts fast-ion profile peaked on axis
- Finite-orbit-width Monte- Carlo ORBIT-RF/AORSA 2D full wave model predicts broader outwardly shifted fast-ion profile

- Differences between the ORBIT-RF/AORSA simulation and the FIDA data are being investigated

- CQL3D modeling with first order orbit-width correction in progress this year
- A full-finite orbit width version of CQL3D is planned for 2011

Energetic particle research opportunities

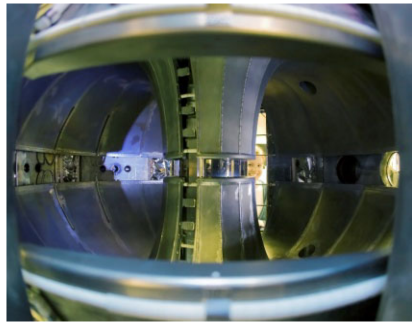
- Measurement and non-linear simulation of Alfvén Eigenmode (AE) avalanche events
- Develop of reduced models for fast-ion transport for use in predictive modeling
- Interaction between fast-ions and HHFW
- HHFW beat-wave AE excitation/suppression

NSTX Research Areas

- Macroscopic Stability
- Transport and Turbulence
- Waves and Energetic Particles
- Lithium Research, Boundary Physics
- Plasma Formation and Sustainment

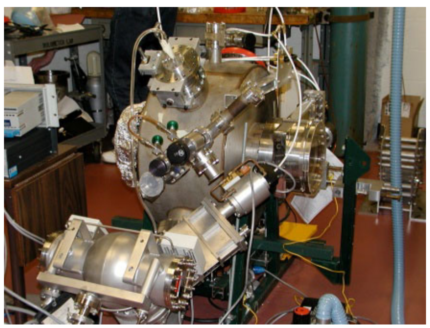
NSTX lithium research is an integral part of a program to develop lithium as a PFC concept for magnetic fusion

LTX lithium handling facility

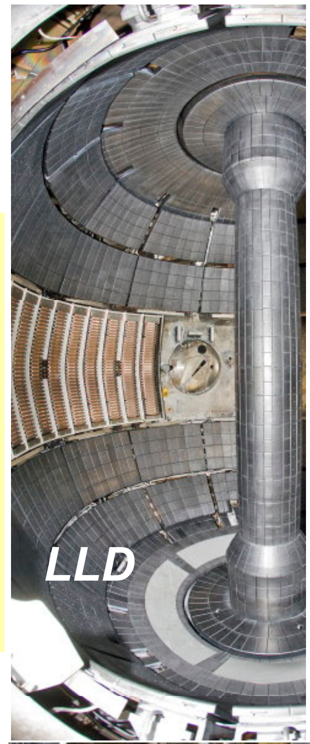


LTX operations commence 2010
- ***Fully-nonrecycling liquid lithium PFC's***
- ***Profile control with core fueling***
- ***No-carbon comparison to NSTX***

LTX PFC test facility



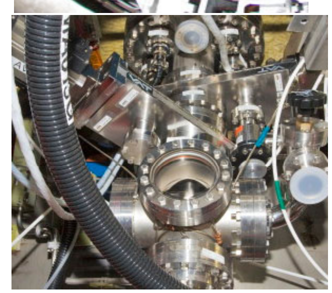
NSTX w/LLD
- ***Only diverted, NBI-heated tokamak studying Li.***
- ***LLD to extend density control for NB CD***
- ***LLD compatible with high flux expansion div. solutions***



NSTX upgrade, Fusion next-steps

Purdue surface analysis facilities

NSTX materials analysis probe

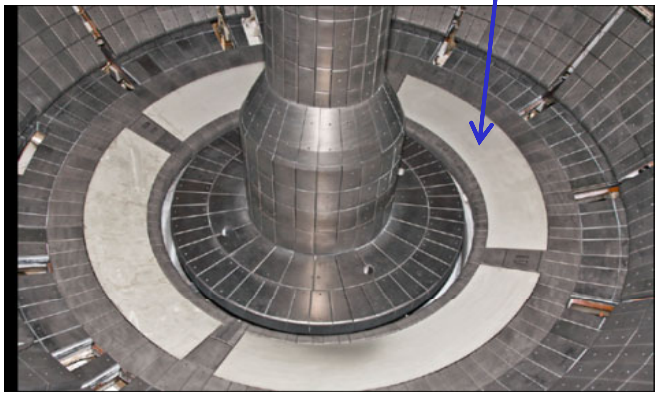


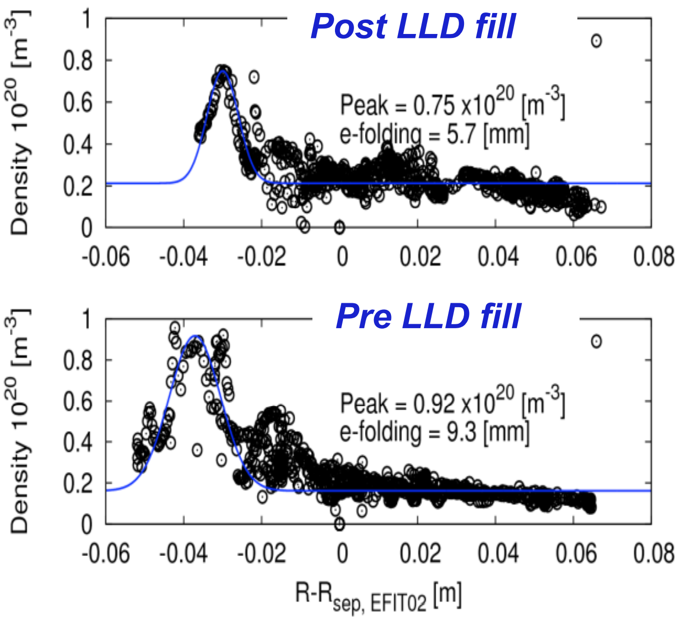
Understand surface physics and chemistry

New Liquid Lithium Divertor (LLD) Capability

- Liquid Lithium Divertor (LLD)
  - LLD commissioned this run to test pumping capability of liquid Lithium
  - Unique capability in world program
    - Only diverted H-mode expt testing liquid Li
  - First signs of density reduction and profile variation at divertor strike-point from liquid Li
    - Peak plasma density reduced by ~20%
    - Profile width reduced by ~40%
  - Challenges
    - Competition with pumping from Li evaporated onto surrounding C surfaces
    - Problems with heaters
    - Impurities possibly interfering with Li pumping

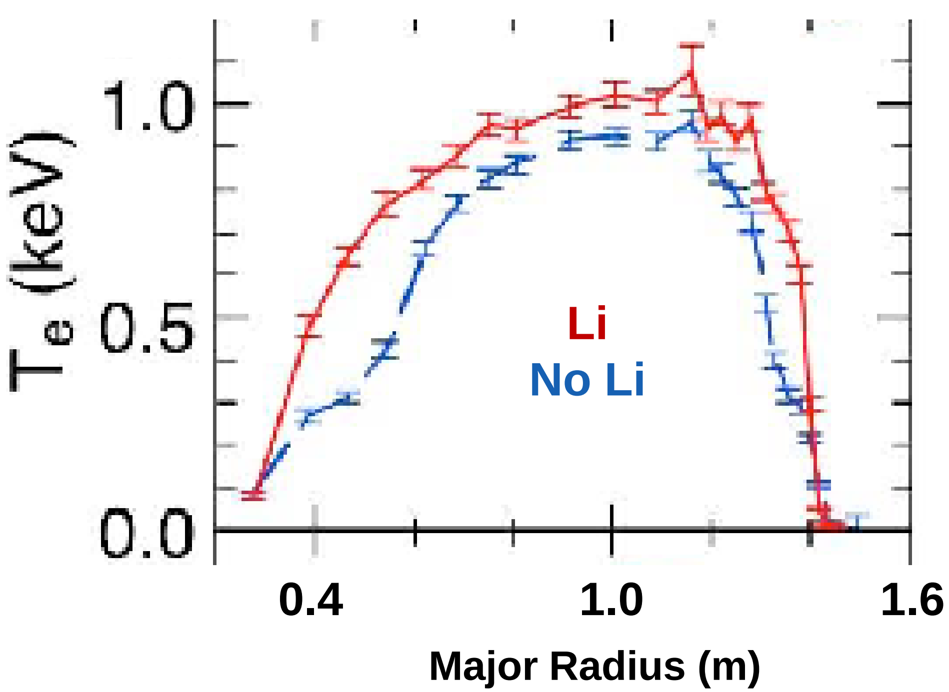
Heatable Mo-surface LLD plate

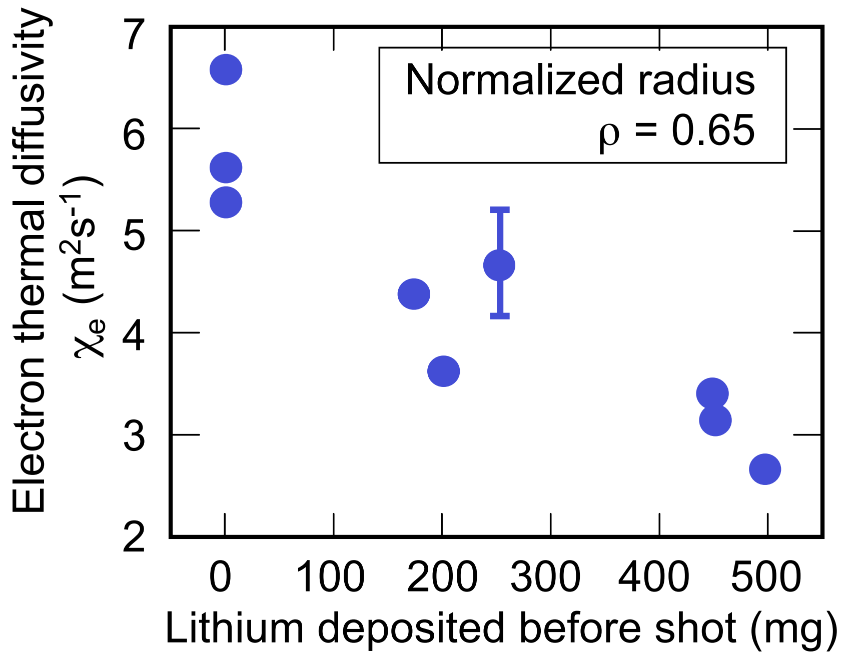




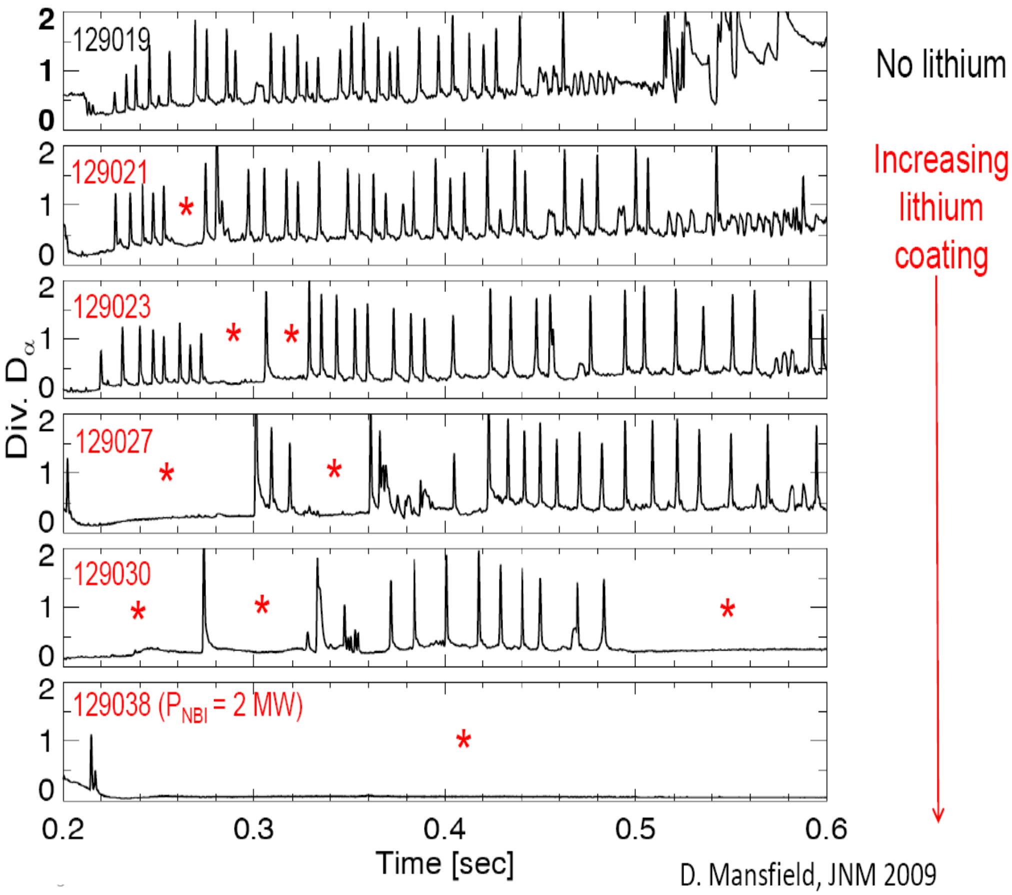
Studying effects of Li-coated PFCs on turbulence and transport

- T_e broadening effect from LITER in 2009 shown to be improvement in local electron confinement leading to broadening of T_e profile
- Fluctuation measurements to be made inside improved confinement region
  - k spectrum to be covered using BES, high-k, and reflectometry
- Test confinement sensitivity to ν* reduction expected with LLD

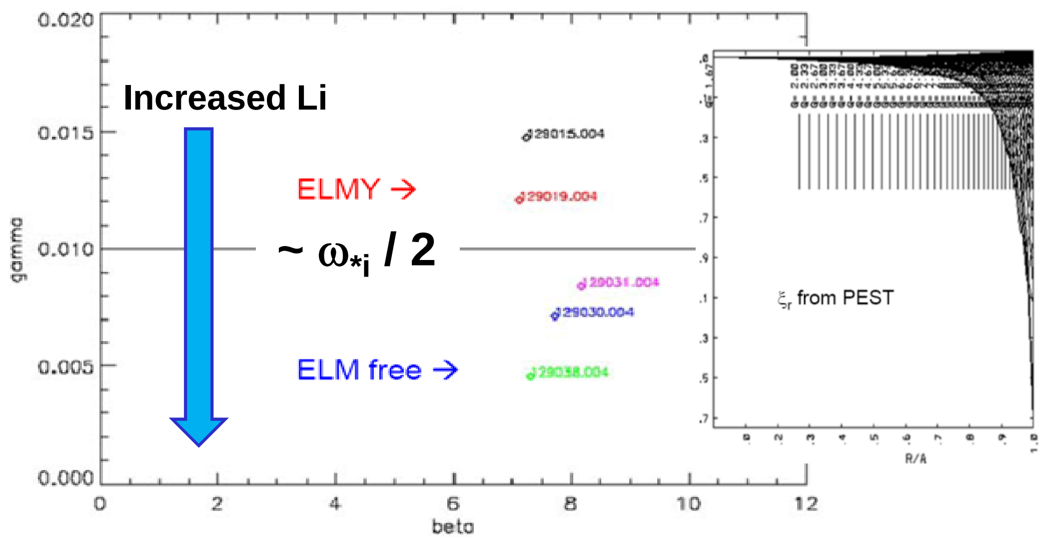




Increasing Li coating observed to suppress ELMs in NSTX



- Measured precursors are low-n: n=2-5 (often 3-4)
- PEST n=3 growth rate reduced with increased Li

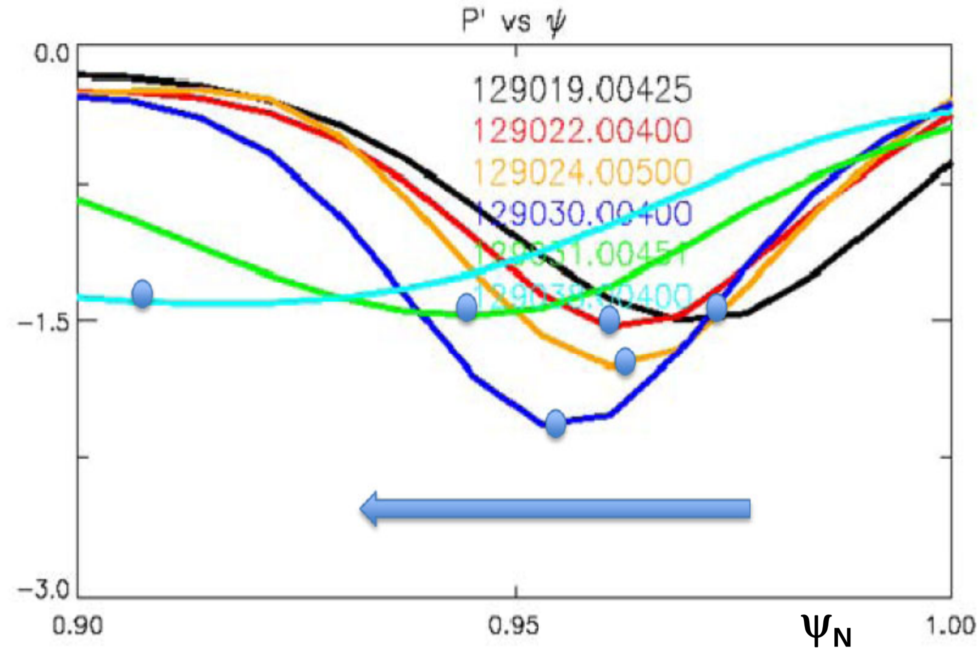


Manickam

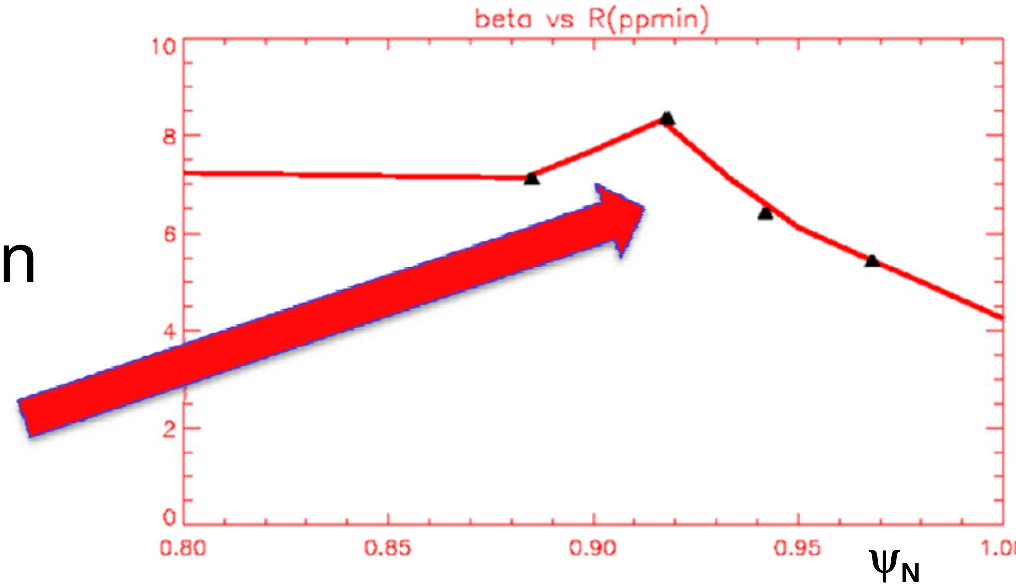
- Low-n of ELM precursor motivates use of ideal kink stability analysis

PEST used to assess stability of low-n kink-peeling mode

- Li moves peak p′ in pedestal to smaller minor radius

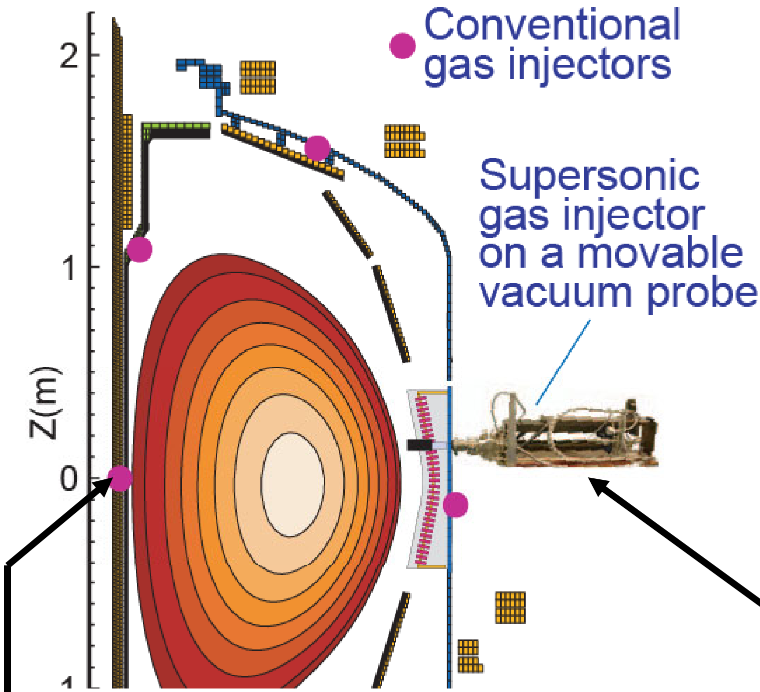


- PEST analysis predicts n=3 β limit maximized for location of peak p′ near ψ_N = 0.9
  - Dependence similar to expt



Supersonic Gas Injection (SGI) Enables Control of D^+ Content in LITER ELM-free Discharges, but C^{6+} Dominates N_e

PAC25-7



Replaced high-field-side (HFS) injection with SGI-only fueling

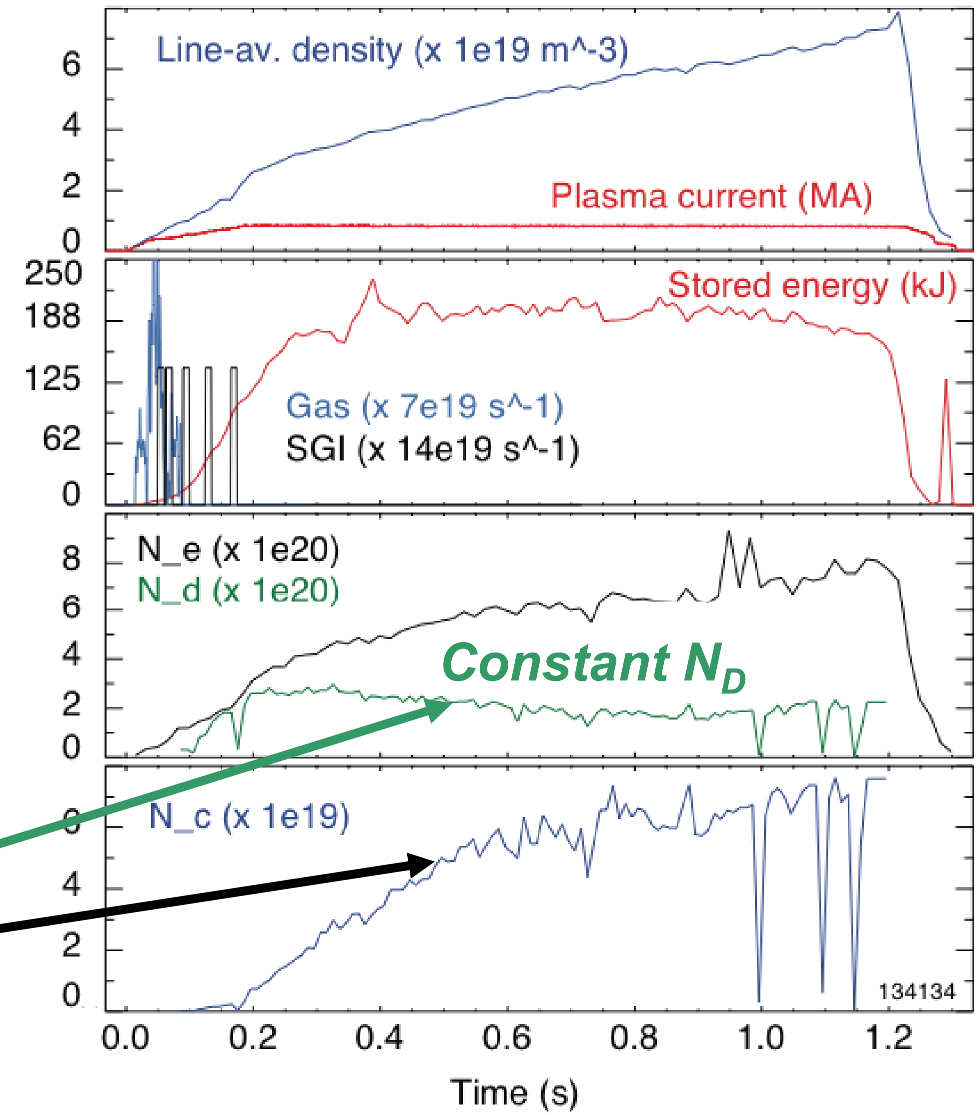
- HFS used for H-mode access
- HFS long tube → slow response

SGI only → D ion density control

***N_e* is rising due to carbon**

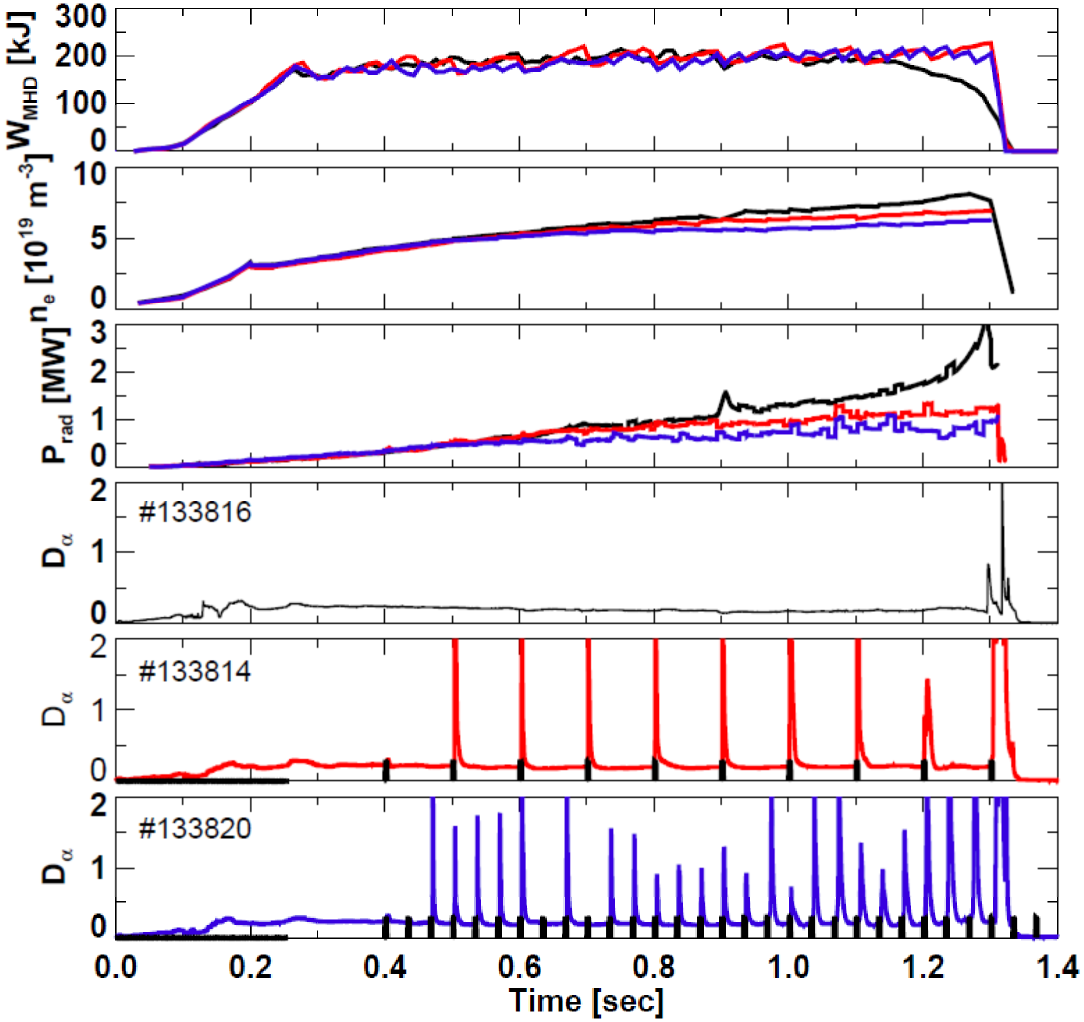
- **C confinement too good…**

LITER at 9 mg/min, ELM-free



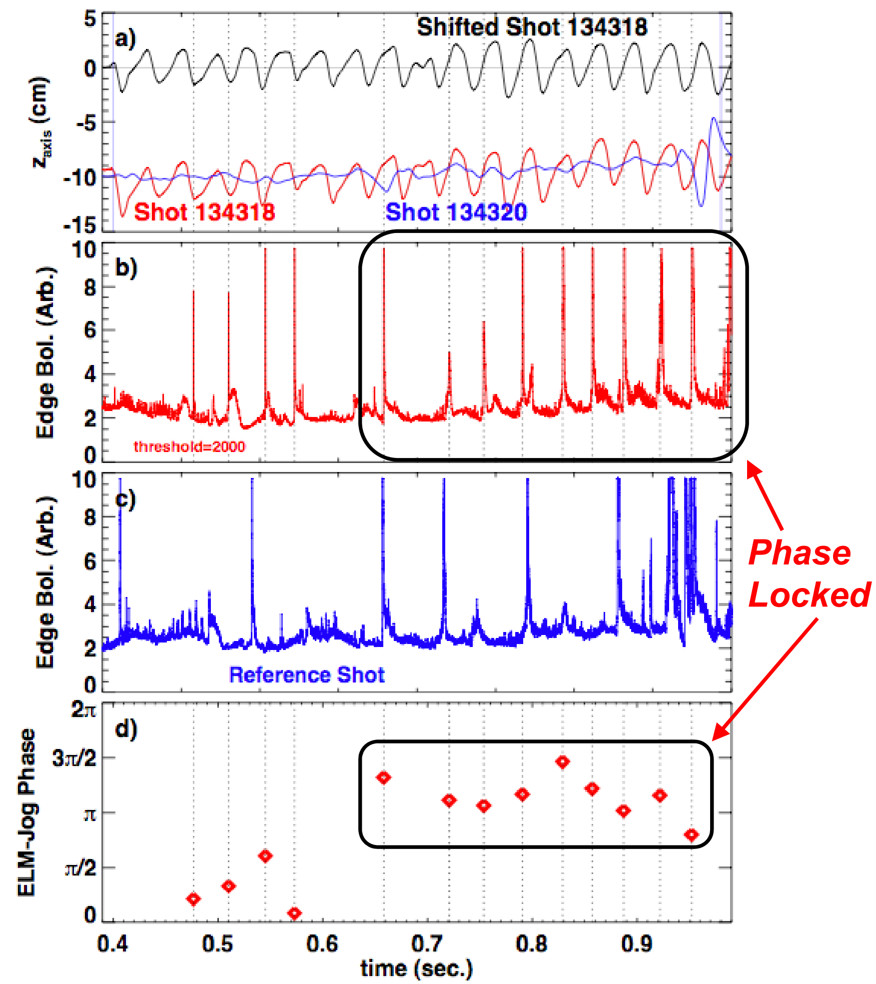
ELM Pacing Developed With Pulsed Non-Resonant Fields and Vertical Jogs

Rapid, Reliable Triggering with Pulsed 3-D Fields



- Reduction in radiated power
- Rapid ELMs lead to smaller per-ELM energy loss [see BP and/or ASC talks for more information]

ELM Pacing Via Vertical Jogs



- Vertical jogging successful despite thick continuous vacuum vessel.
- ELMs become phase locked to upward motion

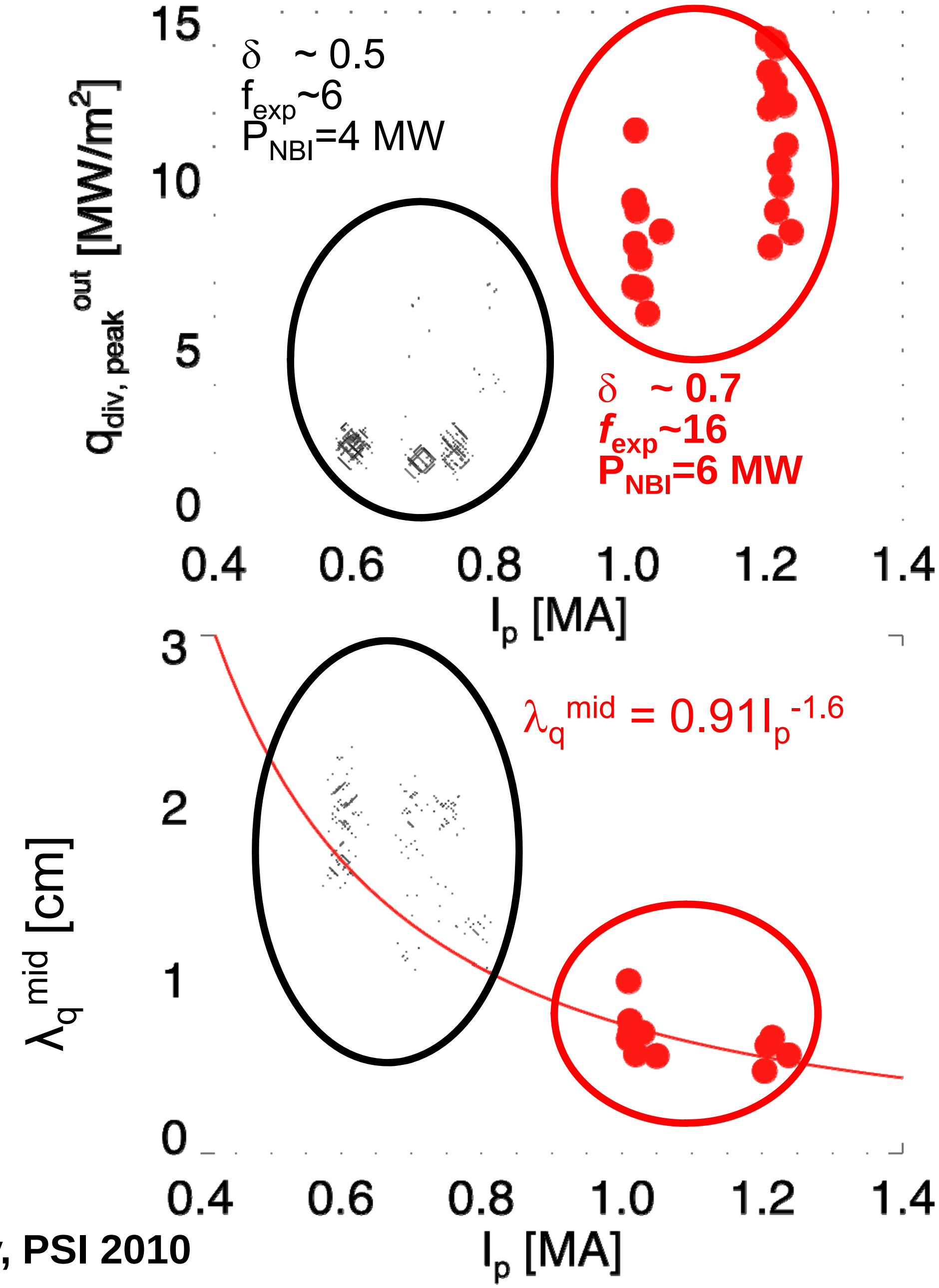
Heat flux width, λ_q^{mid} contracts with I_p

- Combined data from dedicated I_p scans in low δ and high δ discharges
  - Different P_{NBI} and f_{exp}, but previous slides shows no P_{loss} or f_{exp} effect on λ_q^{mid}
  - I_p dependence also in DIII-D, JET
  - q_{95}, $\ell_{||}$ different

- λ_q^{mid} found to scale accordingly:

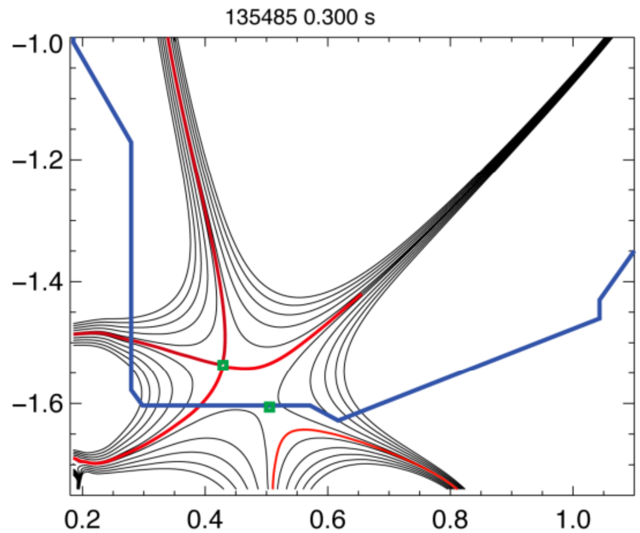
$$\lambda_q^{mid} = 0.91 I_p^{-1.6}$$

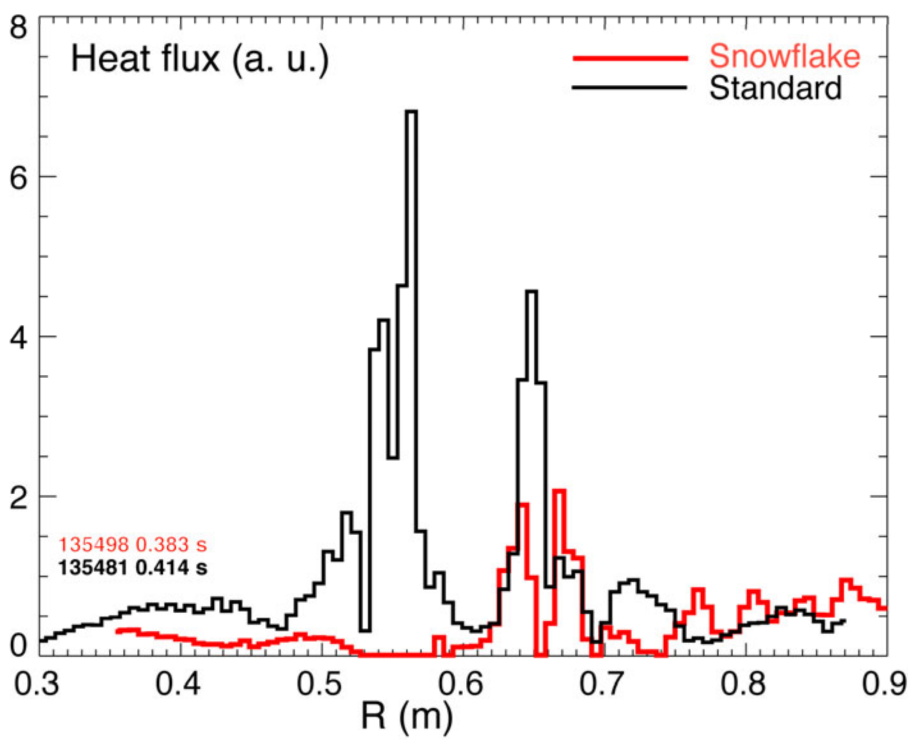
  – Suggests that for NSTX-U, with I_p = 2 MA, λ_q^{mid} = 3 ± 0.5 mm



Gray, PSI 2010

Strike Point Control Development for LLD was Used to Enable "Snow-flake" Divertor Research





Maintained "snowflake"-like configuration for 100s ms
Obtained with lithium
Maintained H-mode confinement with core carbon reduction by 50 %

OSP partial detachment, reduction in divertor peak heat flux

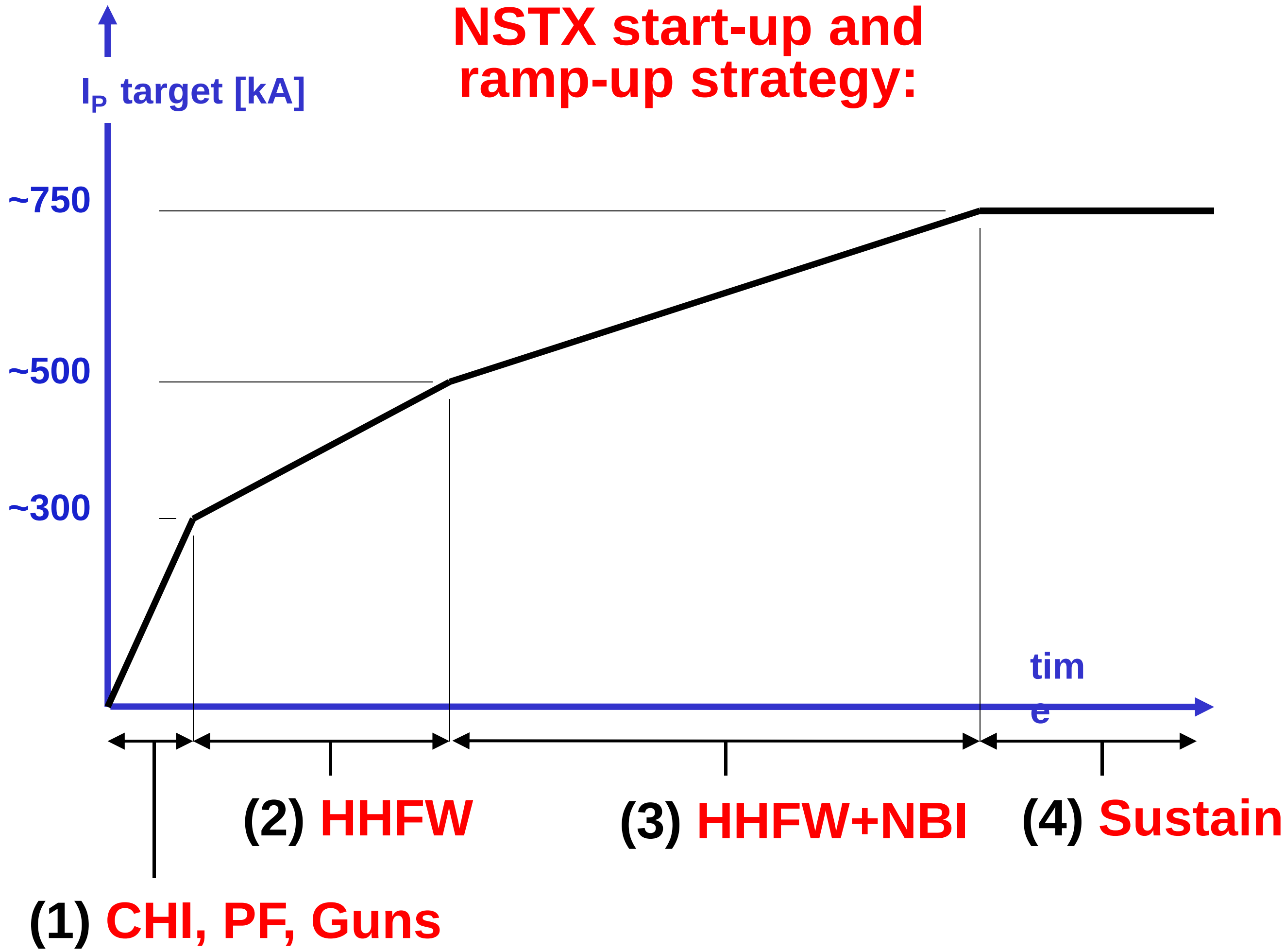
Li & boundary physics research opportunities

- Physics and engineering of Li PFCs
  - How does Li modify transport profiles, stability?
- High-confinement mode (H-mode) onset
- H-mode pedestal structure, ELMs, control
- Power exhaust width, control, novel divertors

NSTX Research Areas

- Macroscopic Stability
- Transport and Turbulence
- Waves and Energetic Particles
- Lithium Research, Boundary Physics
- Plasma Formation and Sustainment

Strategy for Current Start-up and Ramp-up in NSTX



NSTX start-up and ramp-up strategy:

Start-up/ramp-up requirements:

(1→2) I_P, T_e, RF coupling must be sufficiently high for HHFW to be absorbed

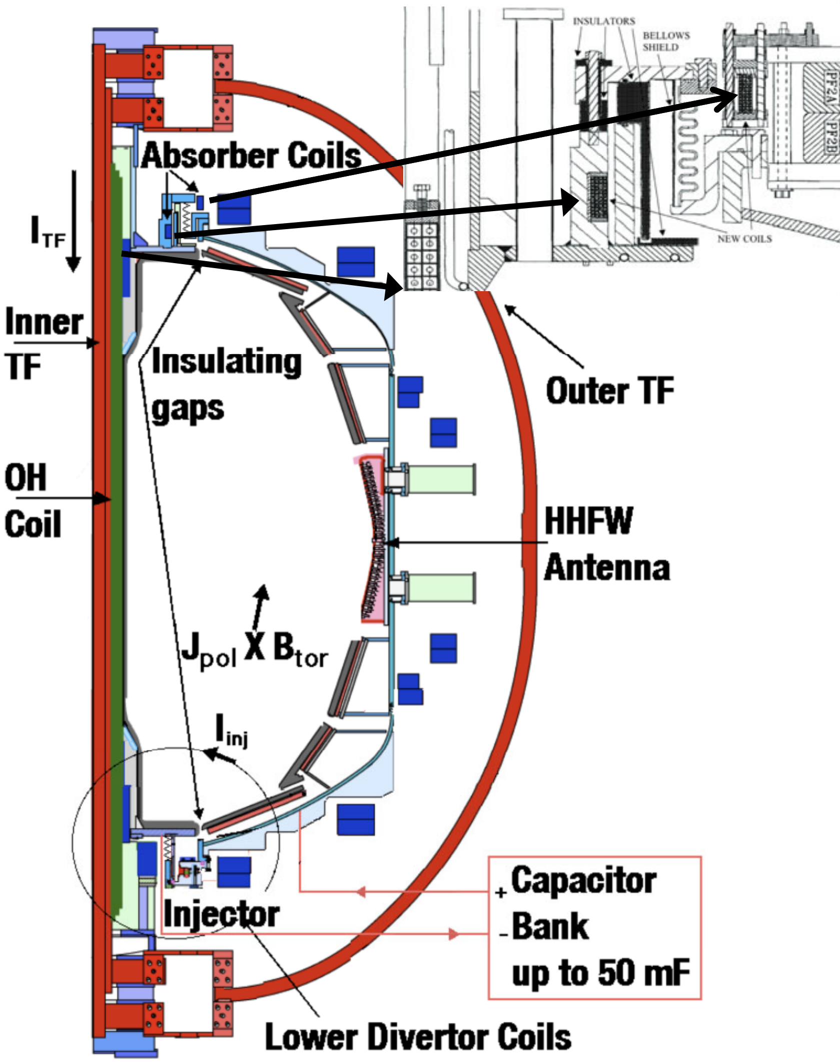
(2) Sufficiently high P_{RF}, τ_E must be achieved for I_P overdrive using BS and HHFW current drive

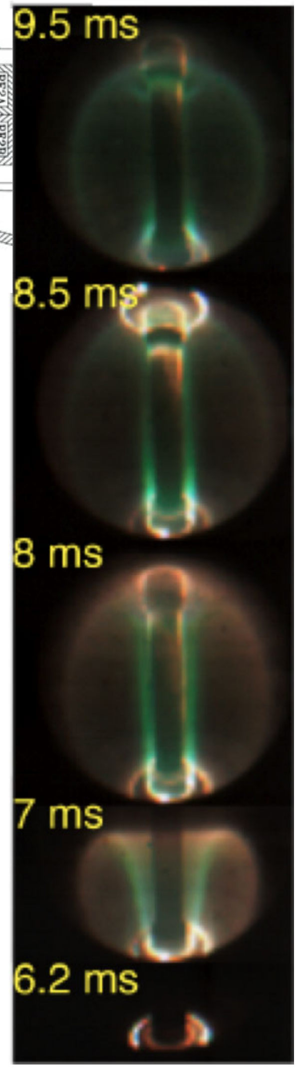
(2→3) Sufficiently high I_P needed to absorb NBI, high P_{HEAT}, τ_E, β_P needed for current overdrive

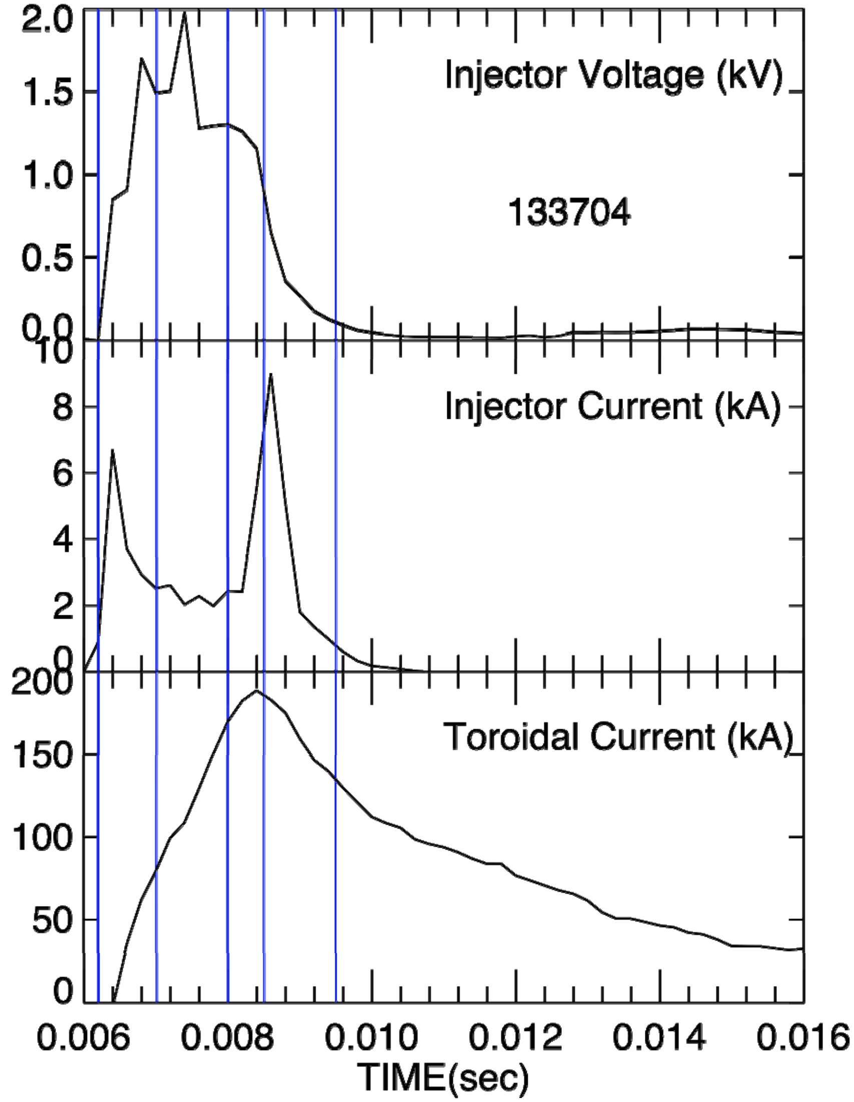
(3→4) Ramp-up plasma must be consistent with sustained high-f_{NI} scenario

NSTX FY2009-13 – Progressively reduce use of central solenoid

Transient CHI: Axisymmetric reconnection leads to formation of closed flux surfaces



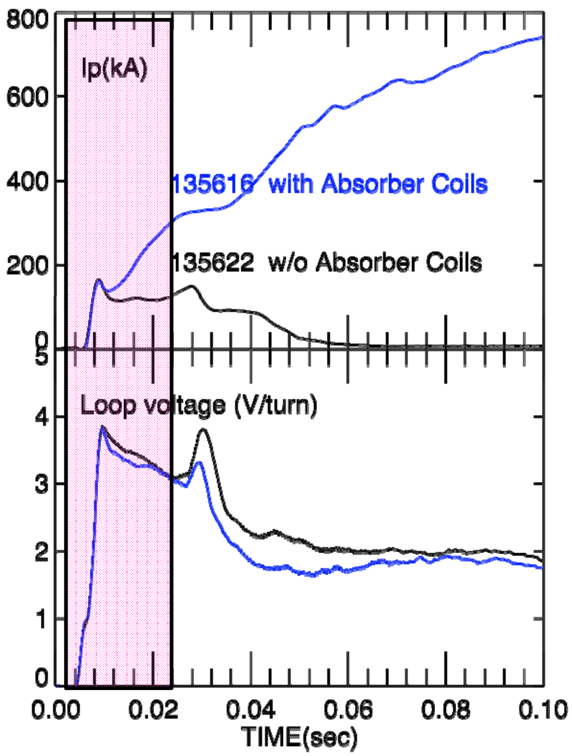


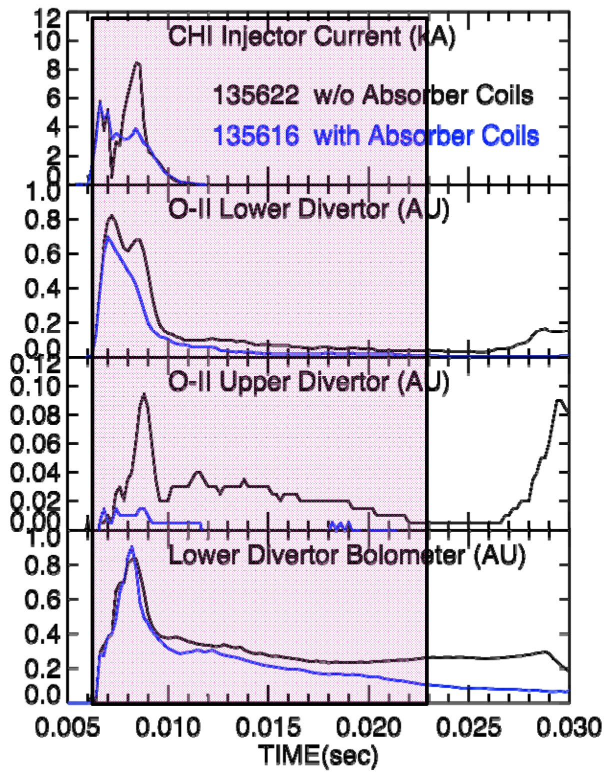


Demonstration of coupling to induction and NBI H-mode (2008)
Improved coupling at higher injection current (2009)

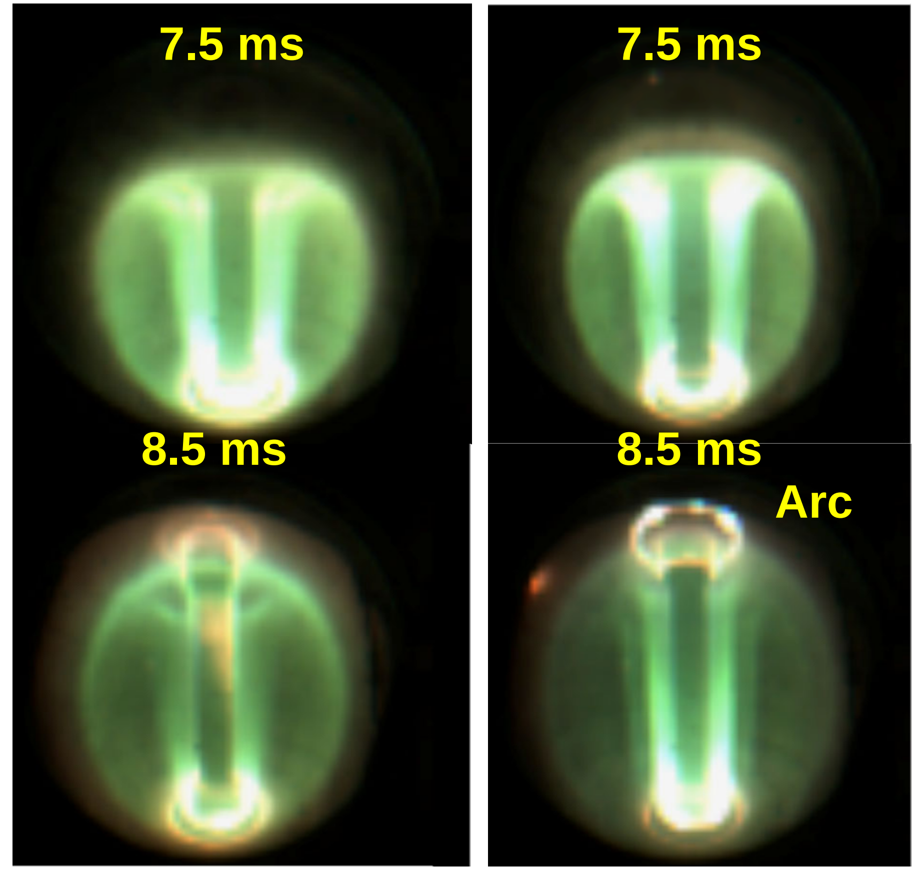
Radial field from absorber coils can prevent plasma from reaching absorber gap and arcing during CHI

15 mF at 1.7 kV



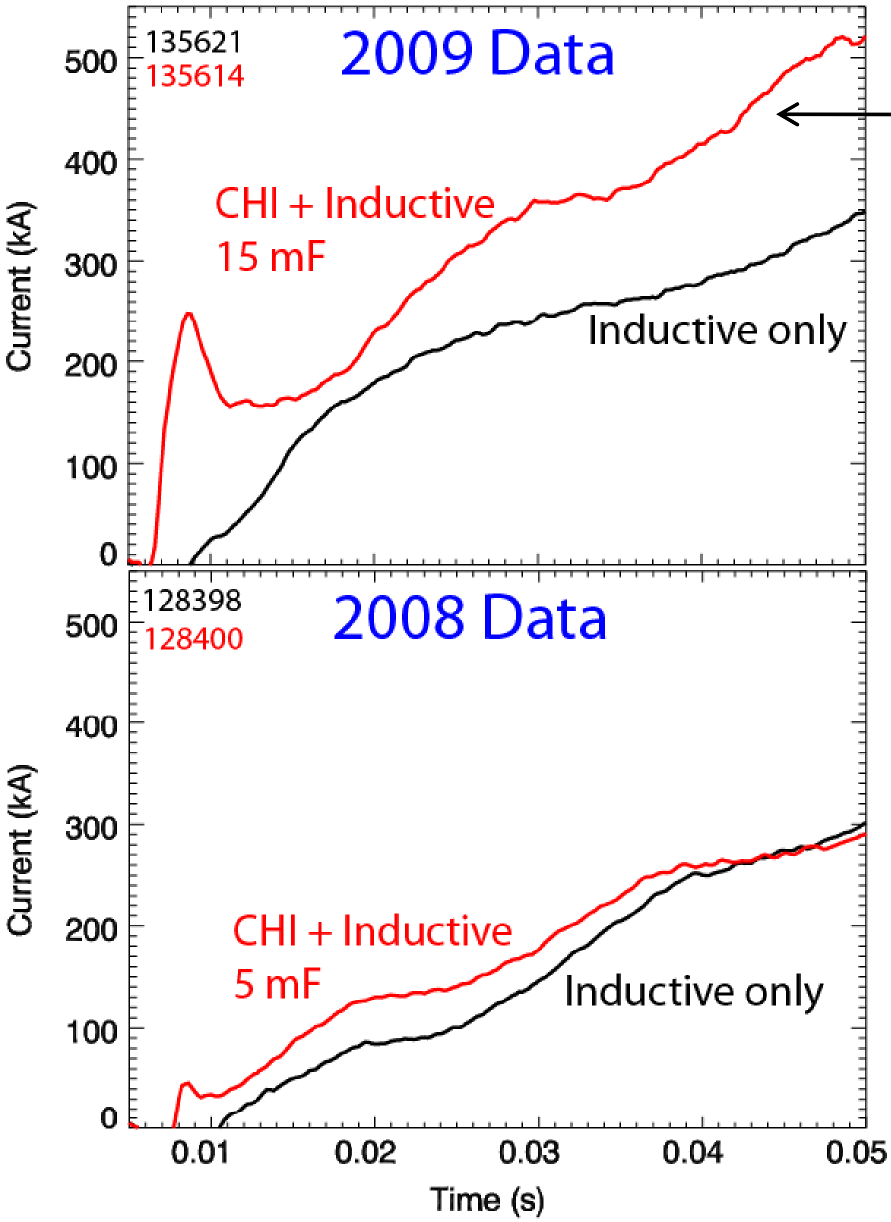


With Absorber Coils **Without**



- Only the discharge without the absorber arc couples to inductive ramp-up
- Even without an arc, low Z impurities limit the ability to couple to ramp-up
- It is important to condition the lower divertors

300kA of CHI Discharge Generated and Coupled to Induction at an Efficiency of 10A/Joule



Discharges with 3-capacitors (20kJ) reaches 525kA

- **-200kA higher than induction-only discharge**
- **-Applied loop voltage is same for all cases**

Methods used to reduce Low-Z impurities:

Long-pulse (400ms) CHI conditioning

Deuterium GDC to reduce oxygen

Buffer field in Absorber to reduce oxygen

Lithium evaporation

Higher cap bank energy leads to arc – will improve in FY10

Study High Elongation Discharge Scenarios
(high κ favorable for increasing β, bootstrap fraction)

High-β_P, q_{95}:

Maximize non-inductive fraction

Limited by I^2t on TF coil

Long pulse moderate q_{95}:

Fully equilibrated profiles

Match TF I^2t and solenoid current limit

High-β_N at low q_{95}:

*Toward reactor I_N, β_T and q**

Limited by solenoid current or MHD.

<u>All configurations:</u>

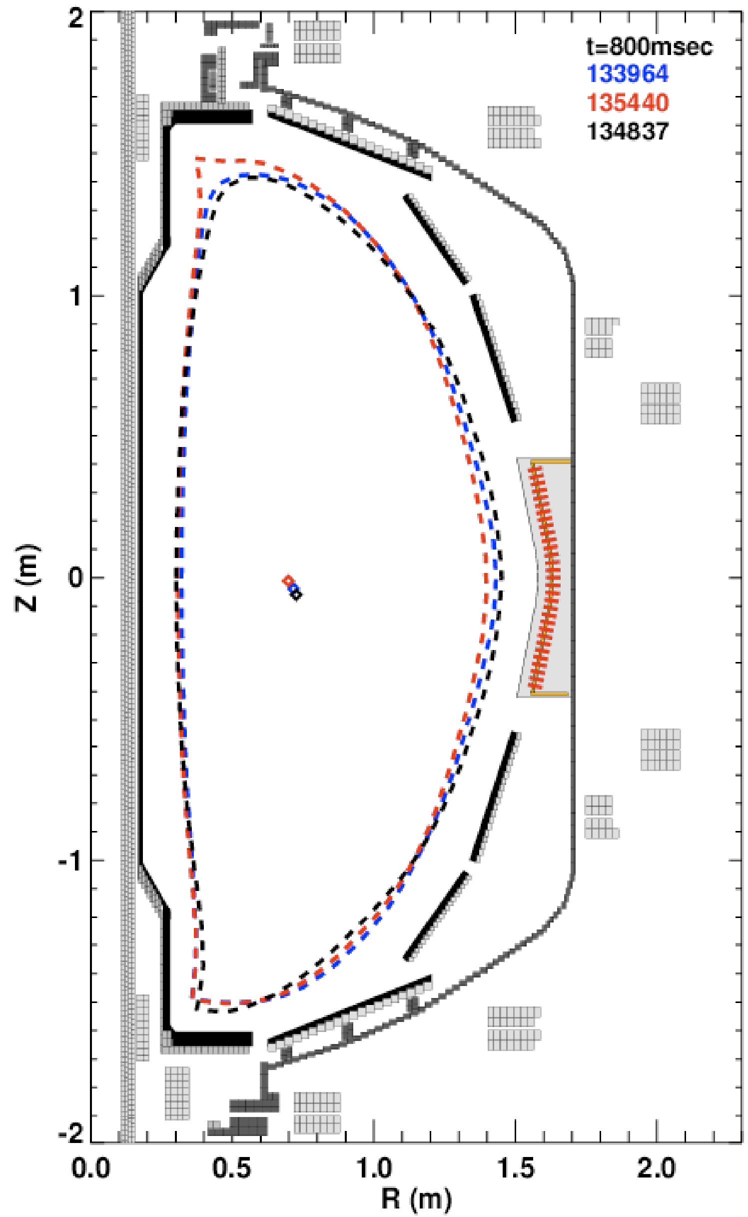
High-κ and δ (κ~2.7 & δ~0.8)

Near double-null (|dr_{sep}|<3mm)

(Shaping and improved power handling)

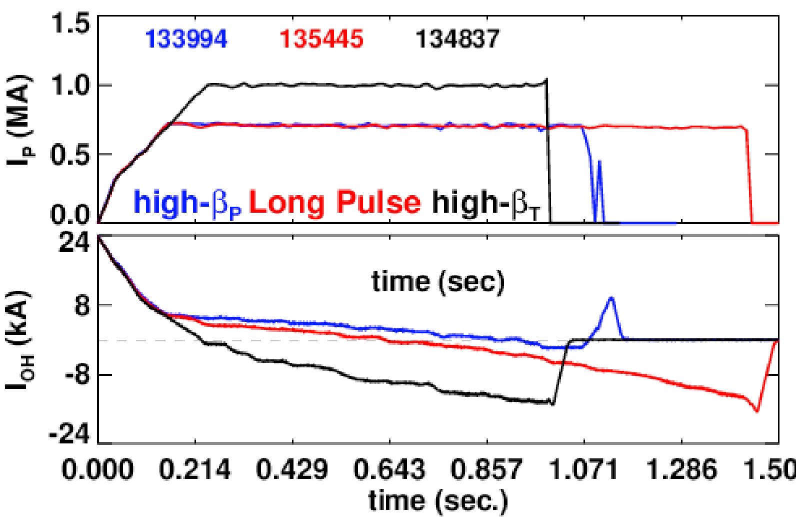
Lithium Conditioning

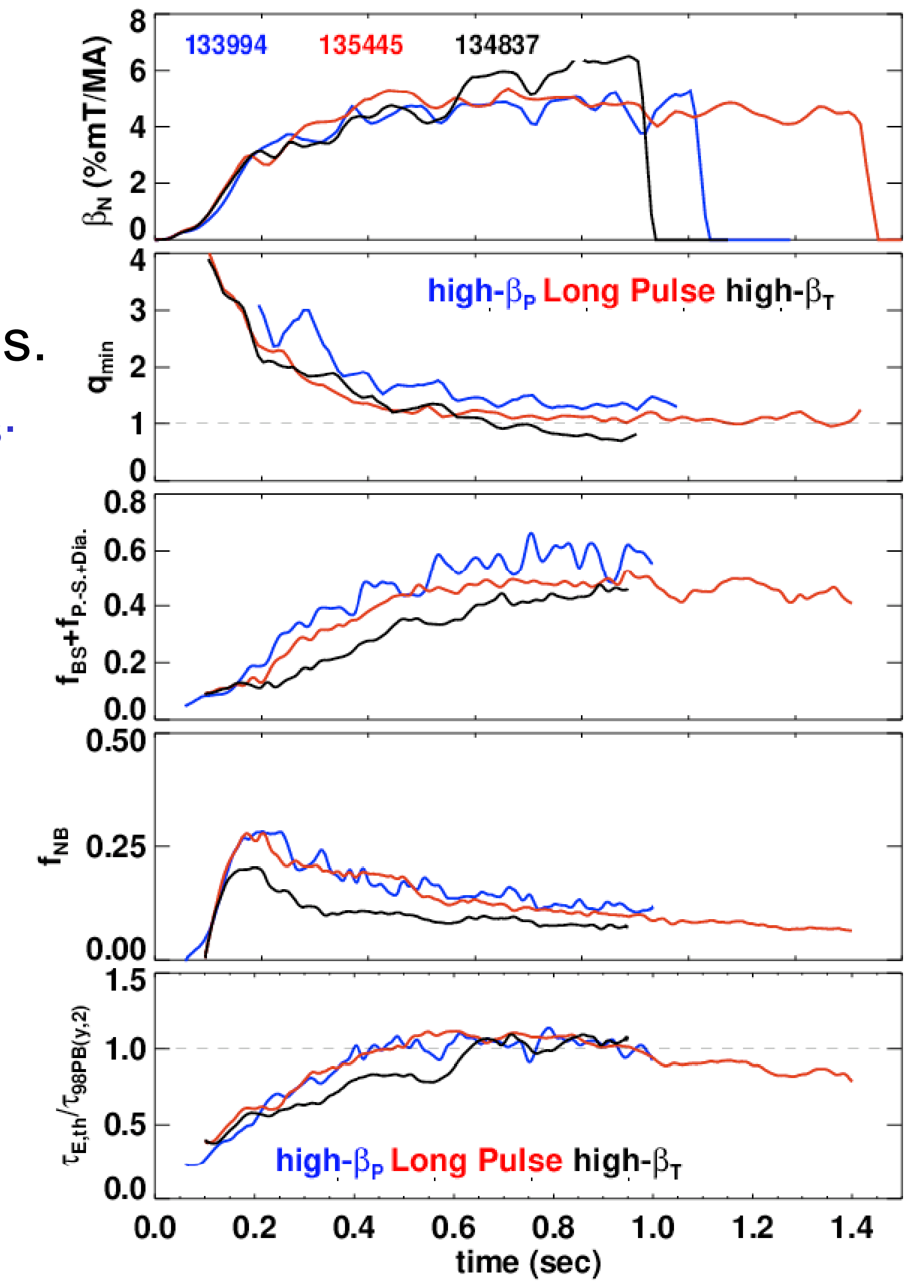
Dynamic Error Field Correction+RWM Control



Large Non-Inductive Fraction and Good Confinement Achieved Over a Range of q at High-κ

- $\beta_N \geq 4.5$ for all scenarios.
  - Matches ST-CTF design point.
- f_{BS} approaching 50%.
  - Matches ST-CTF design point.
- Early f_{NB}>25%, decreases as density rises.
  - Loss in f_{NBCD} partially made up for with f_{BS}.
- H_{98}~1 in all cases.
  - Further confinement improvements are desirable.





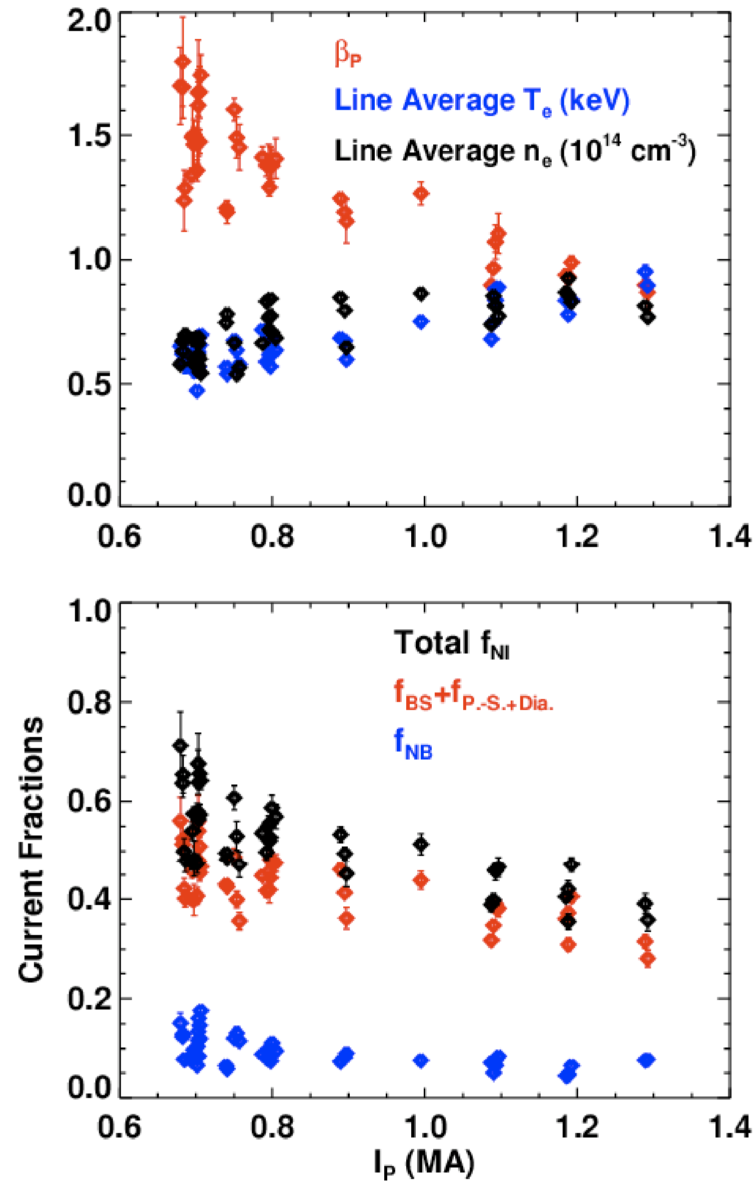
Present Configurations Are Limited to f_{NI}<70%

Loss of NB heating efficiency prevents operating at lower plasma current.

Near-term options for increasing f_{NI} in high-power NBI scenarios:

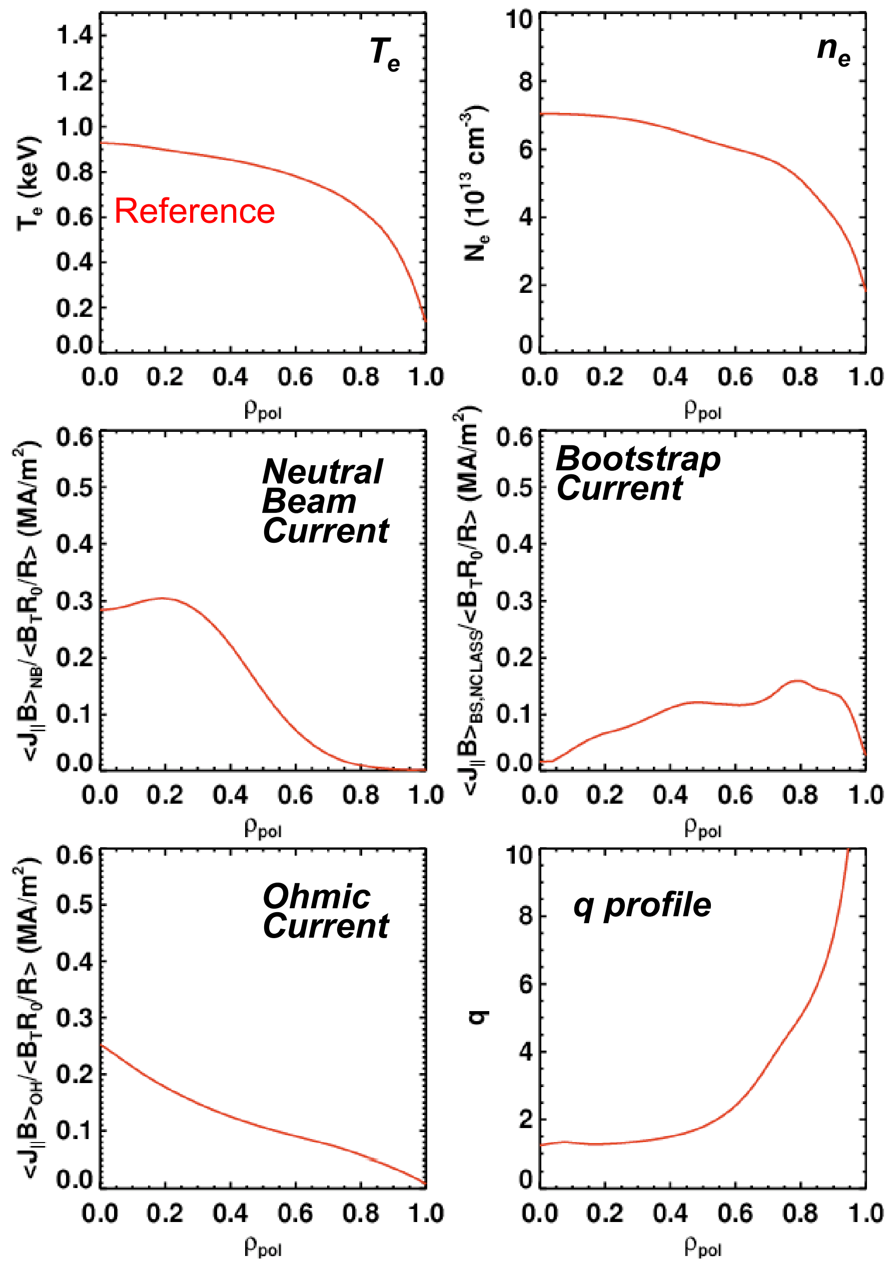
- Reduce density for increased NBCD.
  - –Pumping with LLD.
- Increase the temperature for higher NBCD and bootstrap current.
  - –Confinement improvements with LLD and/or HHFW heating.

Non-Inductive Current Fractions vs. I_P, $4<\beta_N<6.5$, $2.3<\kappa<2.8$



Successful LLD Could Have Major Impact on Non-Inductive Currents

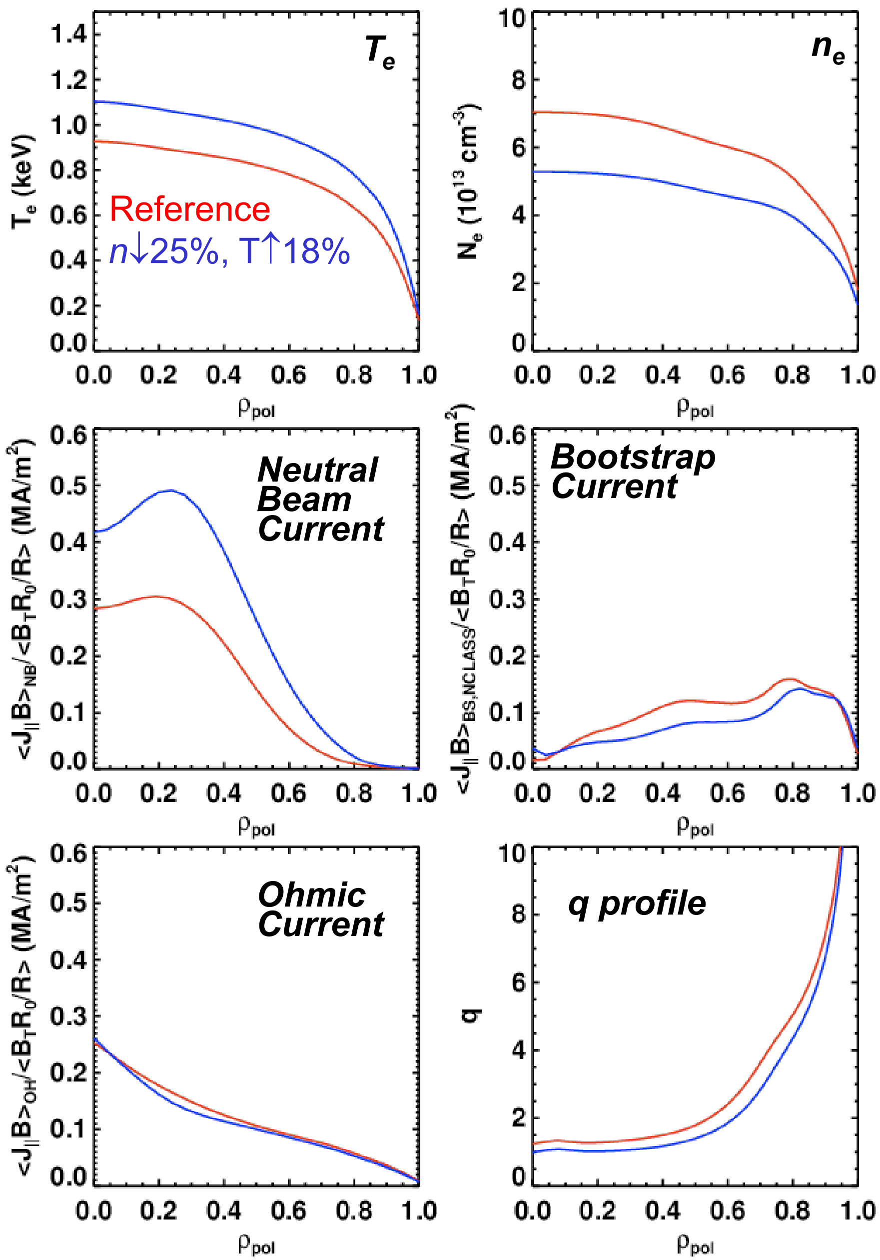
- Utilize profiles from high-κ, high-β_P shot.
  - Fix plasma boundary and Z_{eff}=2.
- Scales profiles to examine effect of f_{NI}.
  - Reference
    - f_{NBCD}=15% , f_{NI}=75%, H_{98}=1.1



PAC25-32

Successful LLD Could Have Major Impact on Non-Inductive Currents

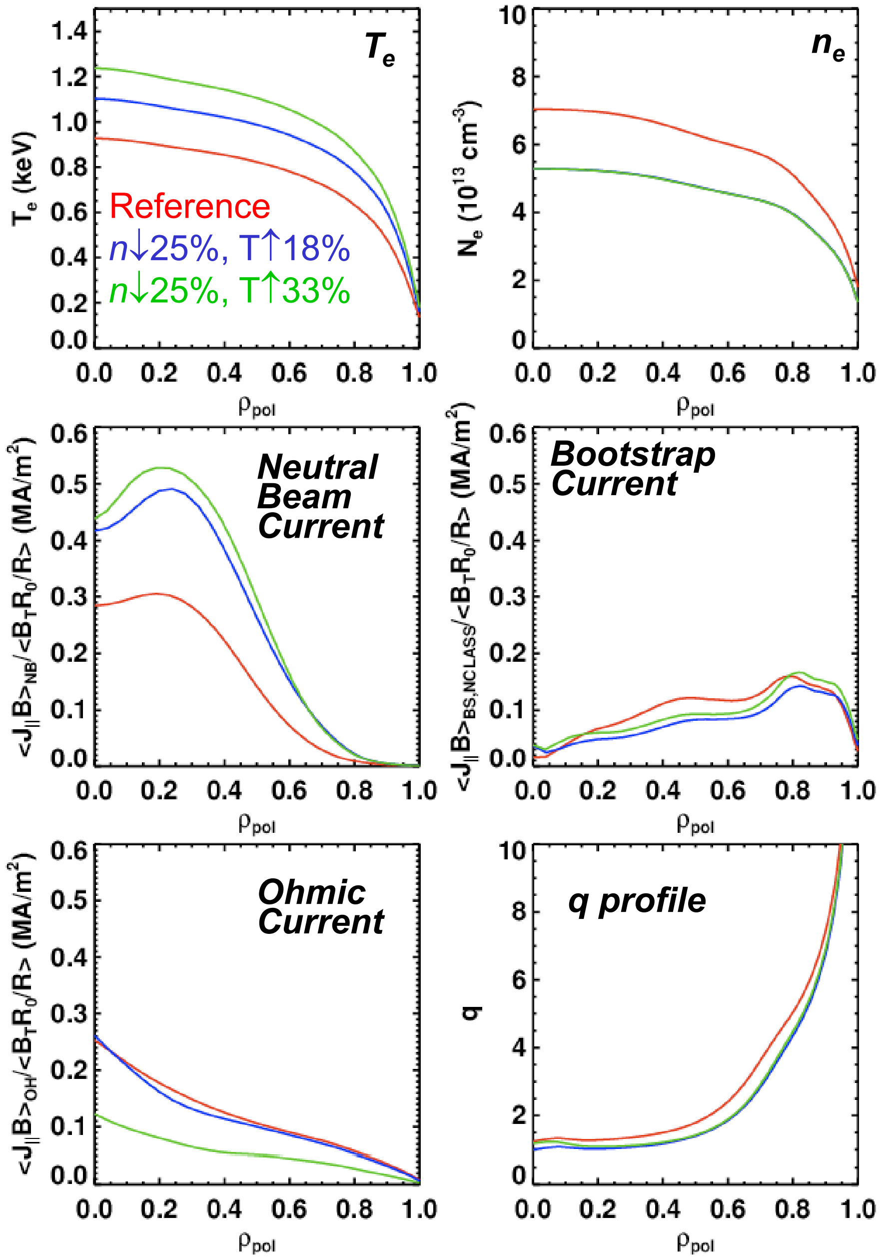
- Utilize profiles from high-κ, high-β_P shot.
  - Fix plasma boundary and Z_{eff}=2.
- Scales profiles to examine effect of f_{NI}.
  - Reference
    - f_{NBCD}=15% , f_{NI}=75%, H_{98}=1.1
  - Density ↓ 25%, Temperature ↑ 18%
    - f_{NBCD}=26% , f_{NI}=80%, H_{98}=1.1



PAC25-32

Successful LLD Could Have Major Impact on Non-Inductive Currents

- Utilize profiles from high-κ, high-β_P shot.
  - Fix plasma boundary and Z_{eff}=2.
- Scales profiles to examine effect of f_{NI}.
  - Reference
    - f_{NBCD}=15% , f_{NI}=75%, H_{98}=1.1
  - Density↓ 25%, Temperature↑ 18%
    - f_{NBCD}=26% , f_{NI}=80%, H_{98}=1.1
  - Density↓ 25%, Temperature↑ 33%
    - f_{NBCD}=27% , f_{NI}=90%, H_{98}=1.3
- Increasing T_e and T_i by 25% in Z_{eff}=2 reference case yields fully non-inductive operation.
  - Z_{eff}=3 requires 40% increases in the temperatures.



PAC25-32

Successful LLD Could Have Major Impact on Non-Inductive Currents

- Utilize profiles from high-κ, high-β_P shot.
  - Fix plasma boundary and Z_{eff}=2.
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- Increasing T_e and T_i by 25% in Z_{eff}=2 reference case yields fully non-inductive operation.
  - Z_{eff}=3 requires 40% increases in the temperatures.

Simulations demonstrate the importance of the thermal transport response to LLD.
- New post-doc for transport modeling starting in March.

Recently revisiting TSC models for NSTX:
- Realistic vessel model for time dependent simulations (R. Sayer, ORNL)
- Calibrated against flux loop data for single coil vacuum shots.

Beginning to model discharge evolution with TSC+NUBEAM.
- Make discharge evolution modeling more routine.

Plans for increased f_{NI} operation

Study the effect of LLD on NBI high-κ scenarios.
- Reduced density for increased NBCD.
- Confinement improvements at fixed density?

Develop scenarios with HHFW+NBI for core electron heating.

Assist with HHFW-only experiments at reduced-I_p.

PAC25-32

Start-up, sustainment research opportunities

- Time-dependent modeling of helicity injection start-up, non-inductive current ramp-up, projection to next-step devices

- Real-time control development and simulation
  - Rotation profile control (NSTX, NSTX Upgrade)
  - Current profile control (NSTX Upgrade)

Long-term future of NSTX

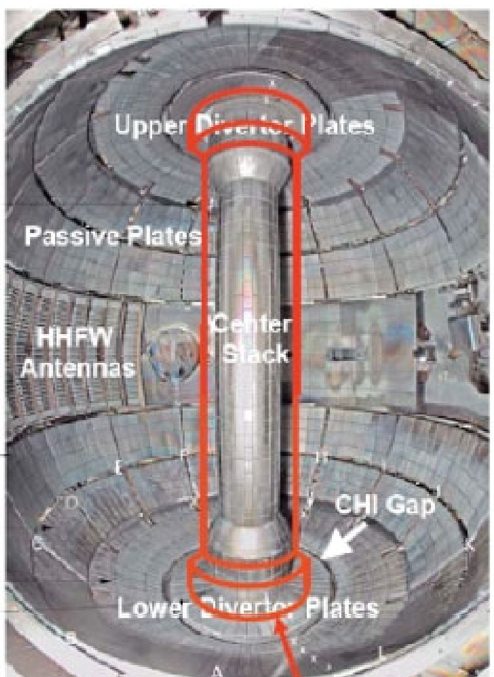
If all goes according to plan, NSTX will not be operating in 2013-14 in order to undergo a major Upgrade

Upgrades provide major step along ST development path

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

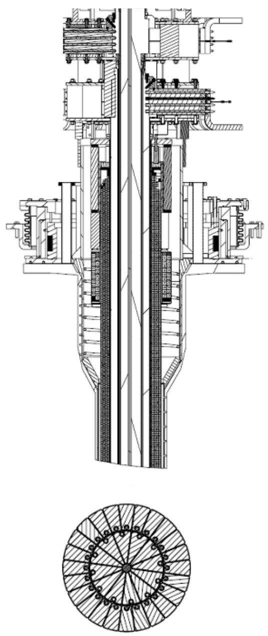
	NSTX	NSTX Upgrade	Plasma-Material Interface Facility	Fusion Nuclear Science Facility
Aspect Ratio = R_0 / a	≥ 1.3	≥ 1.5	≥ 1.7	≥ 1.5
Plasma Current (MA)	1	2	3.5	10
Toroidal Field (T)	0.5	1	2	2.5
P/R, P/S (MW/m,m^2)	10, 0.2*	20, 0.4*	40, 0.7	40-60, 0.8-1.2

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



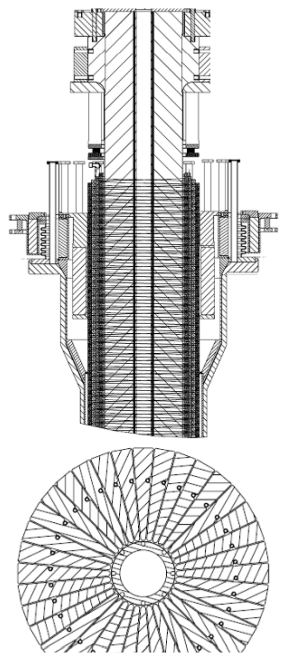
Outline of new center-stack (CS)

Present CS



TF OD = 20cm

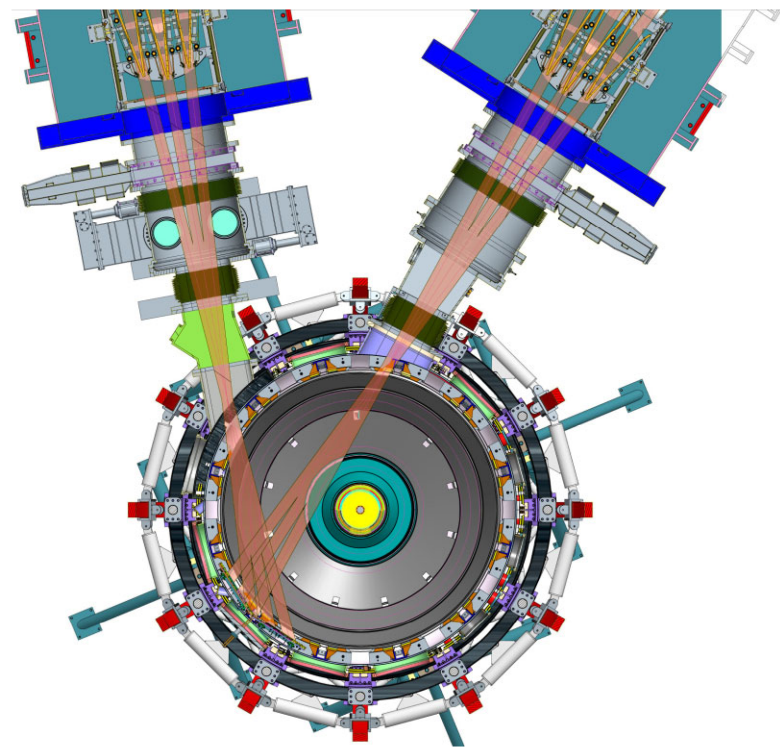
New CS



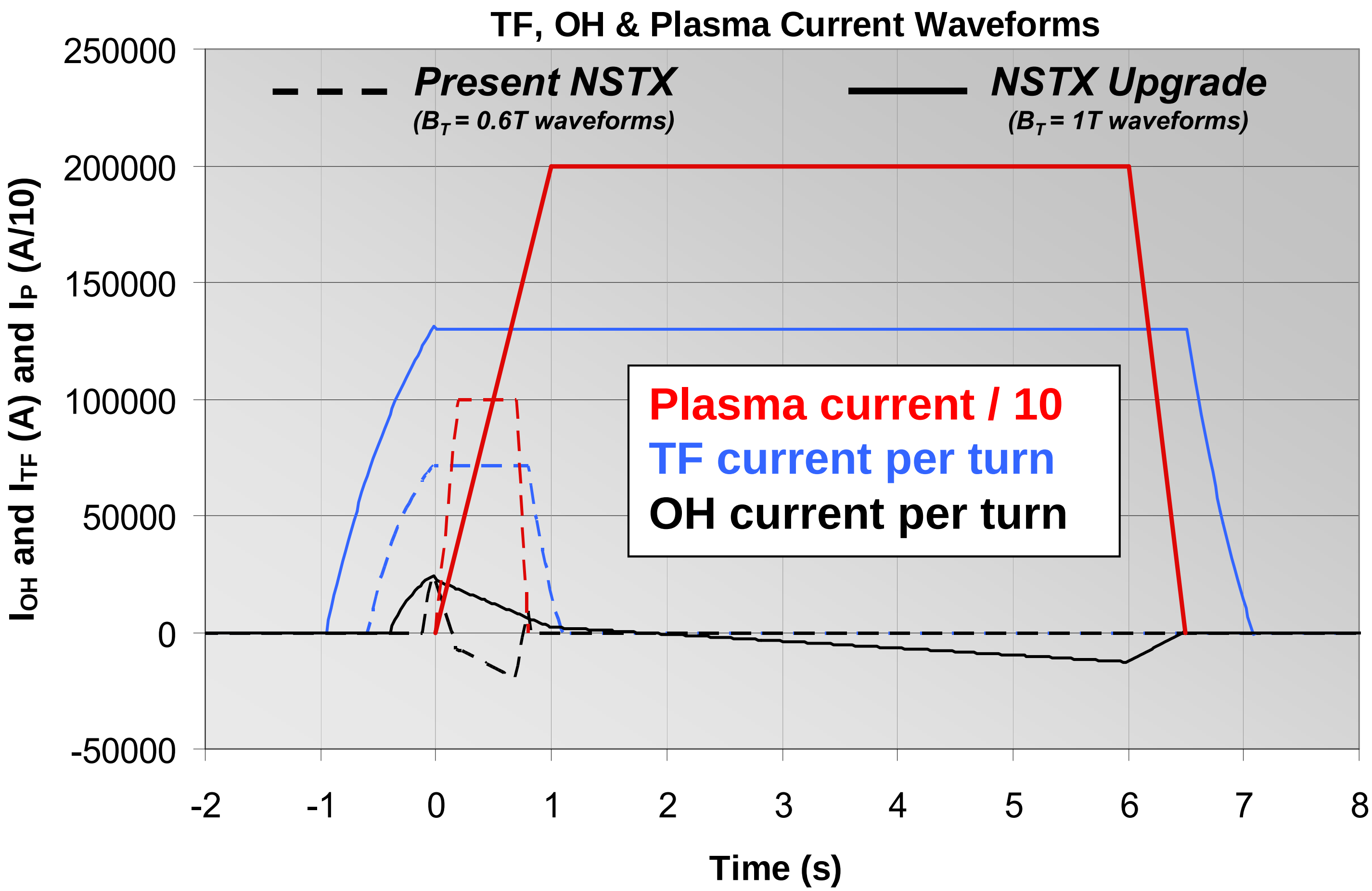
TF OD = 40cm

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

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



Upgrade provides substantial increase in device performance



	Base NSTX	NSTX Upgrade
R_0 [m]	0.854	0.934
Min. aspect ratio	1.28	1.5
I_p [MA]	1	2
B_T [T]	0.55	1
T_{pulse} [s]	1	5
$T_{repetition}$ [s]	600	1000
$R_{center\_stack}=R_0-a$ [m]	0.185	0.315
$R_{antenna}=R_0+a$ [m]	1.574	1.574
Total OH flux [Wb]	0.75	2.1

Relative performance of Upgraded NSTX vs. Base:

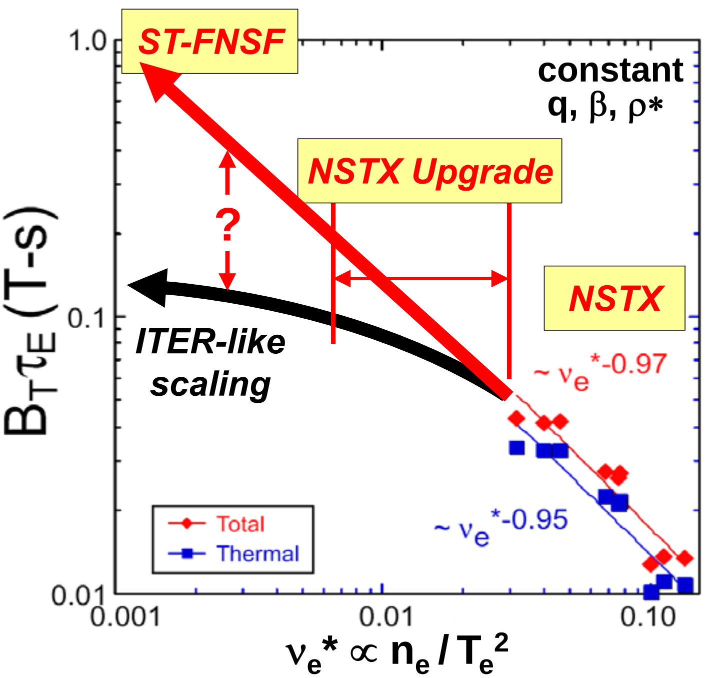
Center-stack radius increased 13cm → A=1.3 → 1.5
Available OH flux increased 3×, 3-5× longer flat-top
I_P increased 2×, B_T increased 2× at same major radius
Plasma stored energy increased up to 4× (0.25→ 1MJ)

NSTX Upgrade will address many important questions for fusion

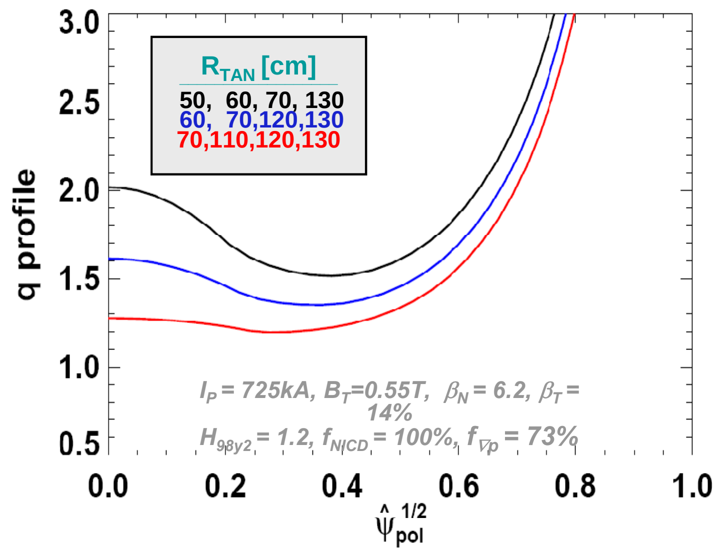
How does confinement vary with normalized temperature, pressure?

Can we create, sustain, and control high β, low l_i ST plasmas without induction?

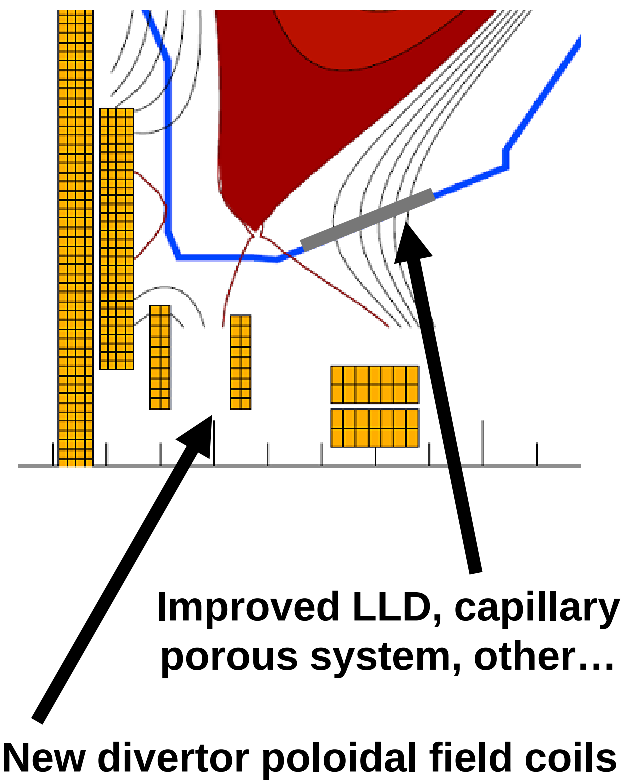
Can we manage the power & particle exhaust of high-performance plasmas?



Normalized electron collisionality reduction from higher temperature from higher field, current, heating



q profile control in 100% non-inductive plasma using mix of existing and additional NBI sources



Non-inductive ramp-up to ~0.4MA possible with RF + new CS, ramp-up to ~1MA possible with new CS + more tangential 2nd NBI

Ramp to ~0.4MA with fast wave heating:

- High field ≥ 0.5T needed for efficient RF heating
- ~2s duration needed for ramp-up equilibration
- Higher field 0.5→1T projected to increase electron temperature and bootstrap current fraction

Extend ramp to 0.8-1MA with 2nd NBI:

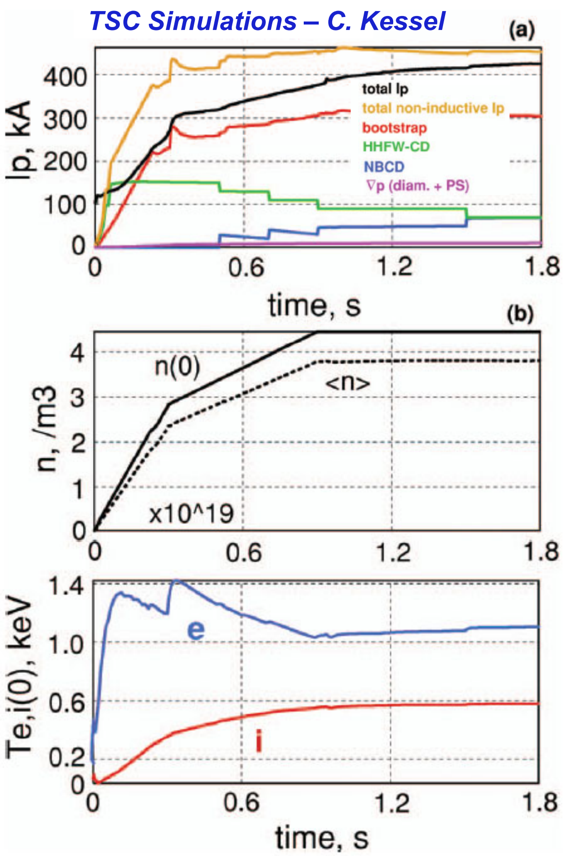
Benefits of more tangential injection:

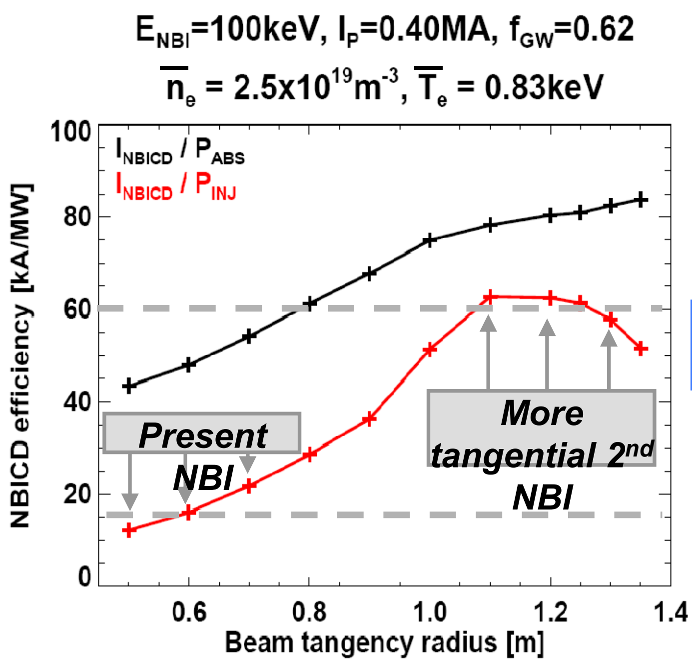
Increased NBI absorption = 40→80% at low I_P

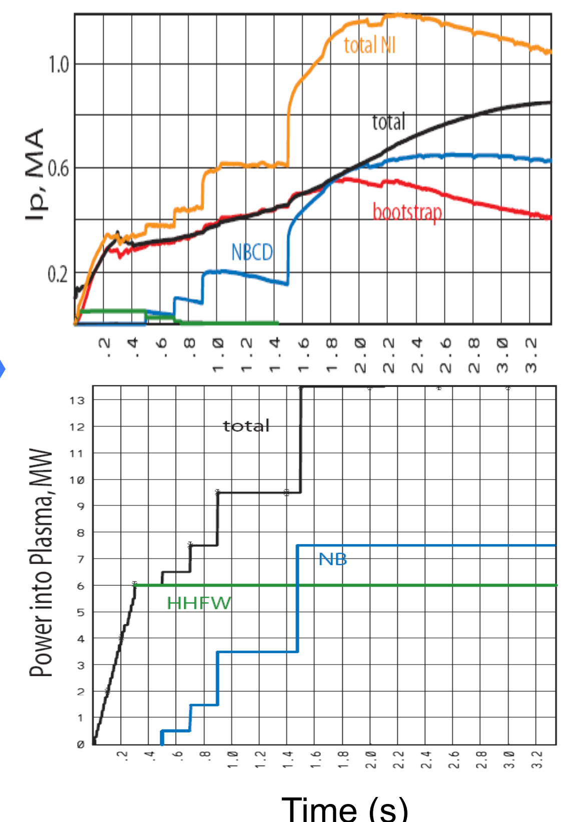
Current drive efficiency increases: ×1.5-2

New CS needed for ~3-5s for ramp-up equilibration

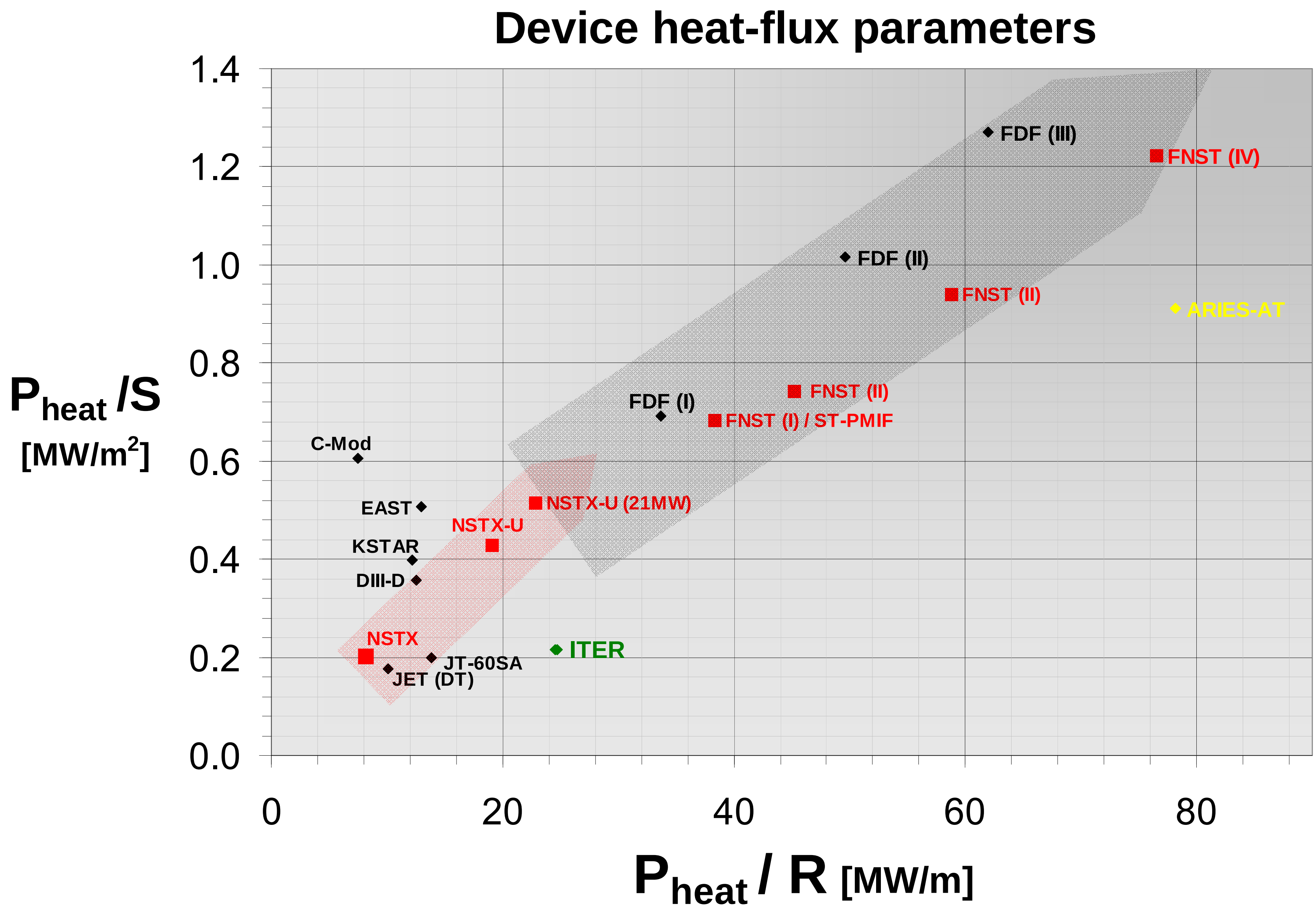
Higher field 0.5→1T also projected to increase electron temperature and NBI-CD efficiency





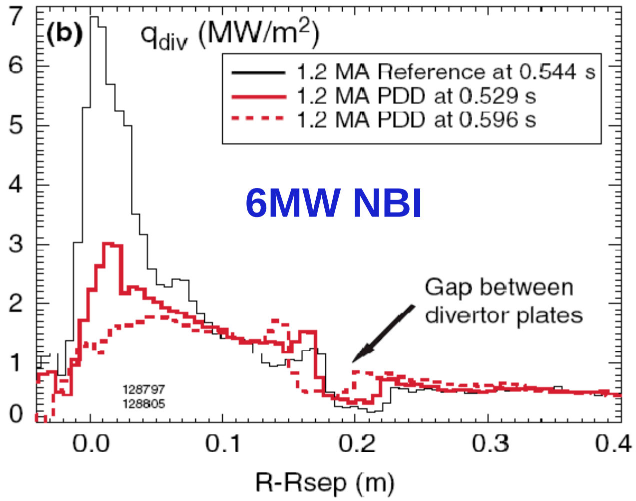


NSTX Upgrade will extend normalized divertor and first-wall heat-loads much closer to FNS and Demo regimes

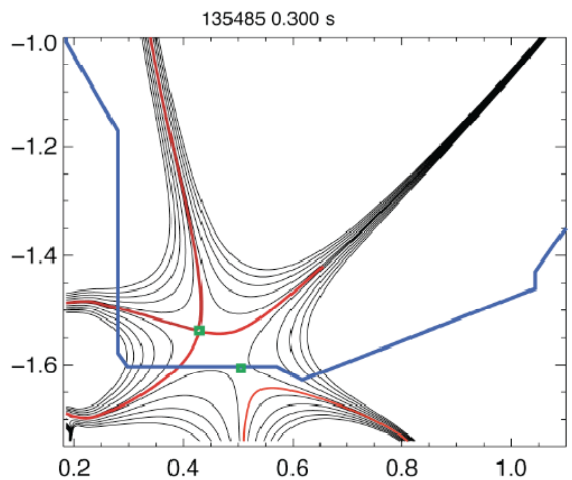


A combination of advanced PMI solutions will likely be required to manage the power exhaust of NSTX Upgrade

High divertor heat flux can be reduced in NSTX with partially detached divertor (PDD)



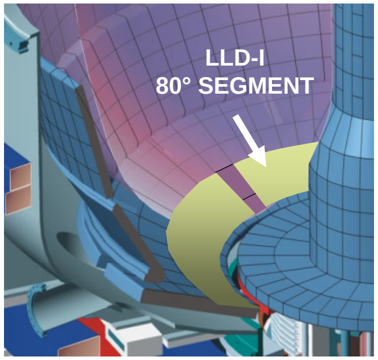
NSTX has demonstrated the formation of high flux-expansion “snowflake” divertor



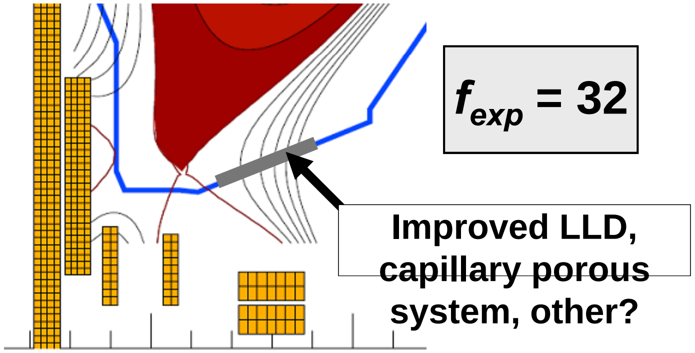
- **The PDD operating regime and other PMI solutions will be challenged in NSTX-U due to:**
  - 2-3× higher input power
  - 30-50% reduction in Greenwald fraction
  - 3-5× longer pulse duration, leading to substantial increase in $T_{divertor}$

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

NSTX LLD



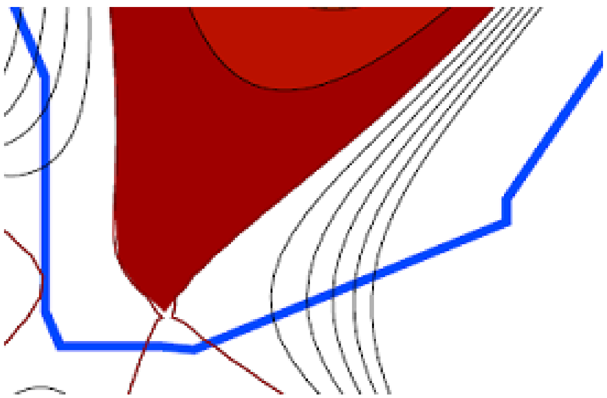
NSTX-U high flux expansion:



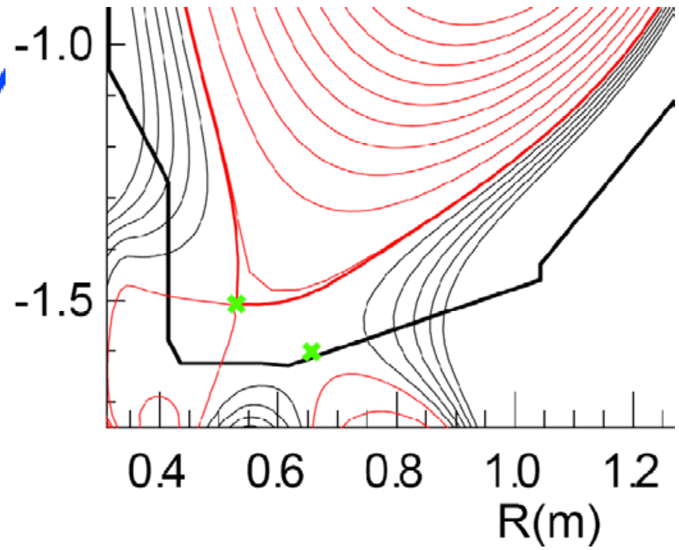
Center-stack Upgrade divertor coil set supports conventional, snowflake, and X/Super-X divertor options

- **Implication:** CS divertor coil location and configuration now finalized

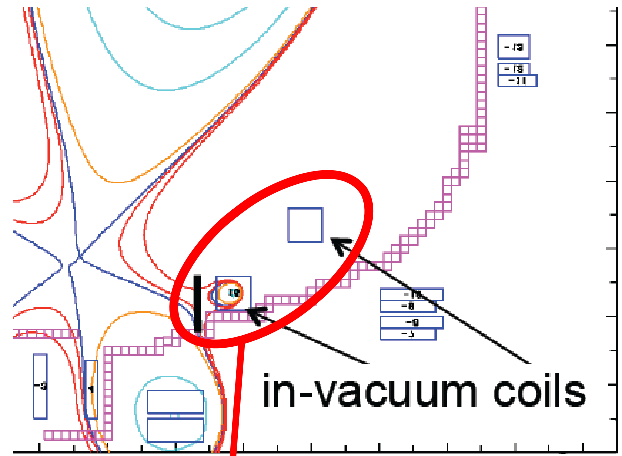
Conventional



Snowflake



X/Super-X



Possible location for cryo-pumps?

X/Super-X requires in-vessel PF coils which are **NOT** part of Upgrade project
Design/analysis of Upgrade divertor is collaborative effort (ORNL, LLNL, UT, PPPL)

NSTX-U divertor design will be strongly influenced by NSTX LLD results
– To be prepared for possible favorable results from LLD, NSTX is initiating a conceptual design study of heated inboard Mo divertor tiles to support test of high-δ LLD-pumped plasma

Interested in any of these topics?

- Talk to me, or better yet, talk to the experts:

Topical Science Group	Leader	Deputy	Theory and Modeling
Advanced Scenarios and Control	Stefan Gerhardt	Michael Bell	Egemen Kolemen
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	609-243-2823	609-243-3282	609-243-3731
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	vlad@pppl.gov	rmaingi@pppl.gov	dstotler@pppl.gov
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Lithium Research	Charles Skinner	Bob Kaita	Daren Stotler
	cskinner@pppl.gov	rkaita@pppl.gov	dstotler@pppl.gov
	609-243-2214	609-243-3275	609-243-2063
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	sabbagh@pppl.gov	jmenard@pppl.gov	jpark@pppl.gov
	609-243-2645	609-243-2037	609-243-3513
Solenoid-free Start-up and Ramp-up	Roger Raman	Dennis Mueller	Steve Jardin
	rraman@pppl.gov	mueller@pppl.gov	sjardin@pppl.gov
	609-243-2855	609-243-3239	609-243-2635
Transport and Turbulence	Howard Yuh	Stan Kaye	Taik-Soo Hahm
	hyuh@pppl.gov	skaye@pppl.gov	thahm@pppl.gov
	609-243-2710	609-243-3162	609-243-2611
Wave-Particle Interactions	Gary Taylor	Mario Podesta	Nikolai Gorelenkov
	gtaylor@pppl.gov	mpodesta@pppl.gov	ngorelen@pppl.gov
	609-243-2573	609-243-3526	609-243-2552

NSTX Scientific Organization for the FY2010 Run

Backup Slides

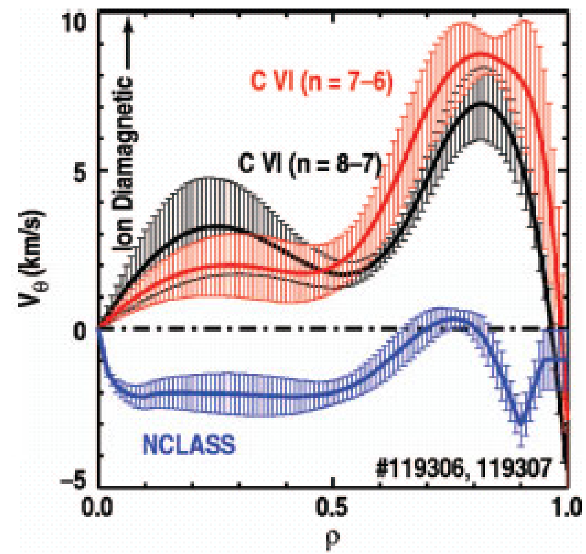
Recent Poloidal Rotation Velocity Measurements Consistent With Neoclassical Estimates

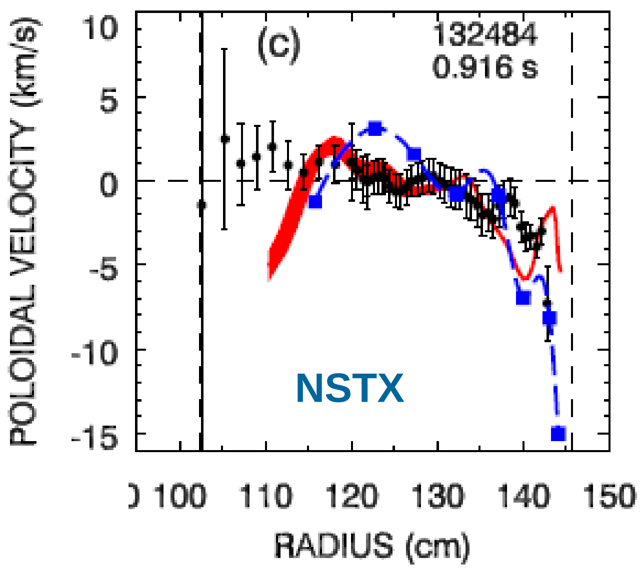
- Implementation of impurities in GTC-NEO enables direct comparison with carbon poloidal rotation measurements (Kolesnikov, PoP 2010)
  - v_θ much different than neoclassical at high aspect ratio

- *v_θ close to neoclassical in NSTX*
  - Difference between NSTX, DIII-D results motivated joint expt)
    - Lower B_T in DIII-D brings measured v_θ closer to neoclassical
  - GTS to explore impact of turbulence on poloidal rotation

(Kolesnikov, Rewoldt, Wang)

DIII-D (from Solomon et al., PoP 2005)





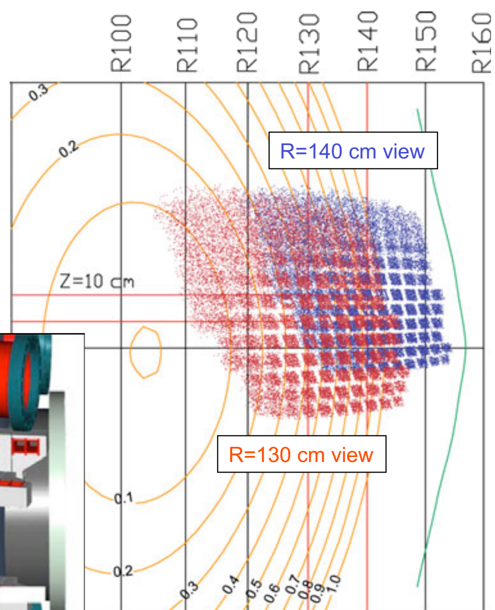
In 2010, new NSTX diagnostics extend wavenumber and spatial coverage of turbulence and transport measurements

- BES (**B**eam **E**mission **S**pectroscopy)
  - 2 viewing optic sets necessitated by steep NSTX pitch angles
  - First light, commissioning, calibration, in 2010
  - Low-k turbulence measurements continues in 2011

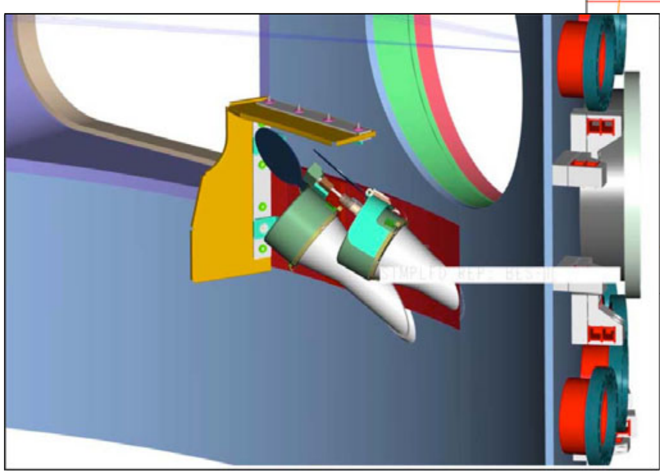
ME-SXR (**M**ulti-**E**nergy **S**oft **X**-**R**ay array)

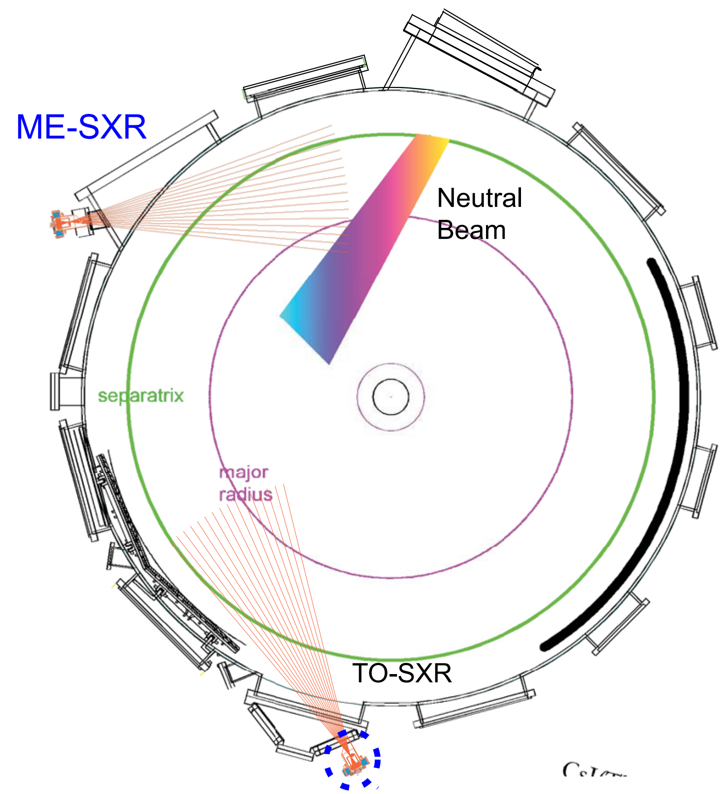
- Prototyped in 2009, full system 2010
- Fast measurements of n_e, T_e, n_{imp} edge profiles using 5 energies

BES sightlines cover most of outboard plasma

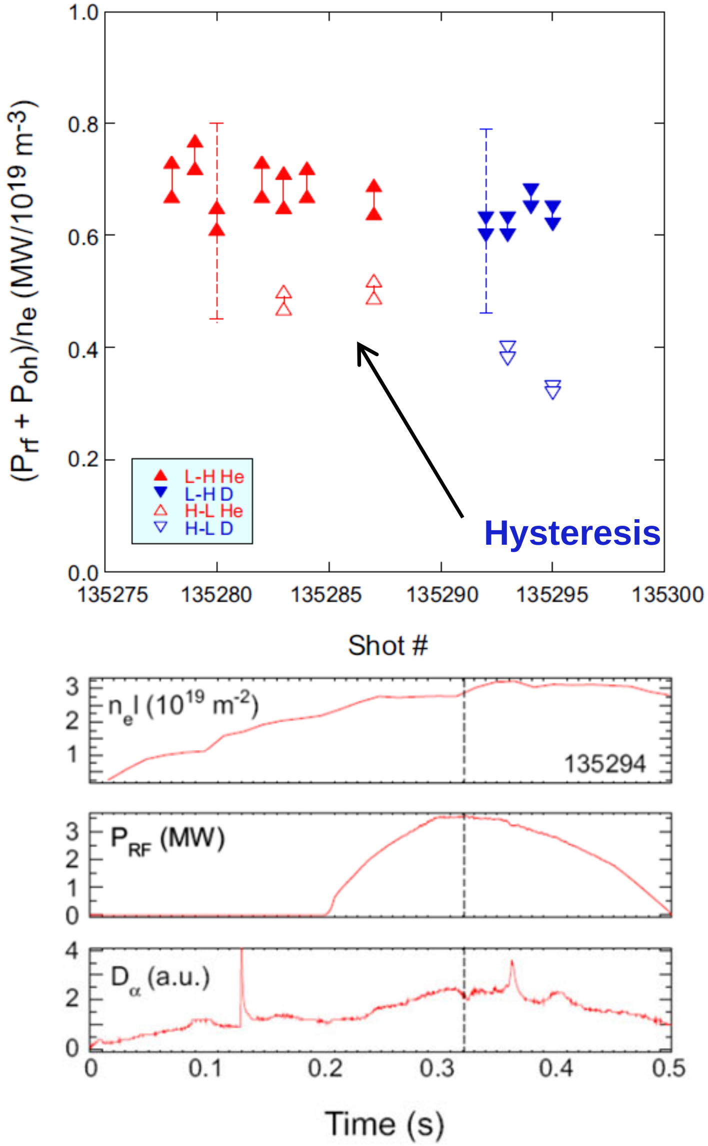


CAD rendering of dual BES vacuum side optics





Normalized L-H Threshold Power P_{LH}/n_e Similar for He and D Plasmas Heated by HHFW



Continuous ramp in P_{RF} allowed fine resolution

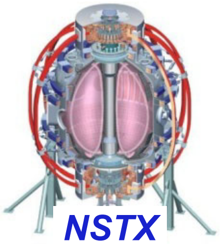
L-H threshold power vs. species is ITER high priority task

Other scalings observed in NSTX:

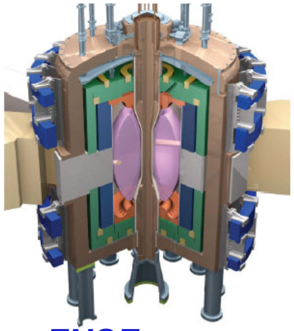
- Plasma current
  - P_{LH} / n_e increased ~2 x for I_P = 0.7MA → 1MA
- Lithium coatings
  - P_{LH} / n_e decreased ~35% with Li evaporation
- 3D field strength
  - P_{LH} / n_e increased ~65% with 3-4 higher n=3 field
- τ_E weakly dependent on B_T

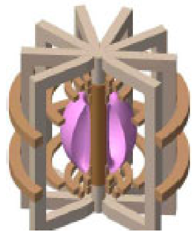
NSTX Upgrade will contribute strongly to toroidal plasma science and preparation for a fusion nuclear science (FNS) program

- 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

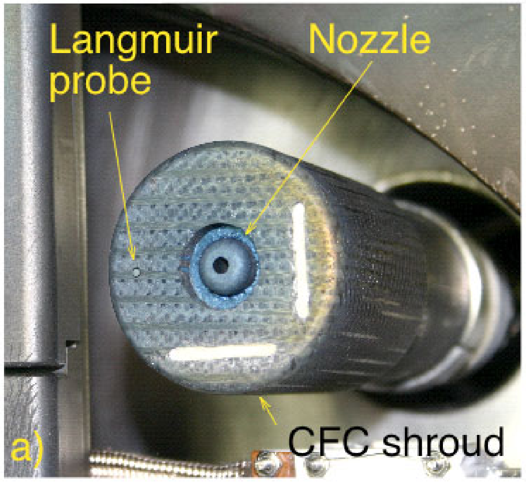

NSTX


NSTX-U


FNSF (ST-CTF)


PMIF (NHTX)

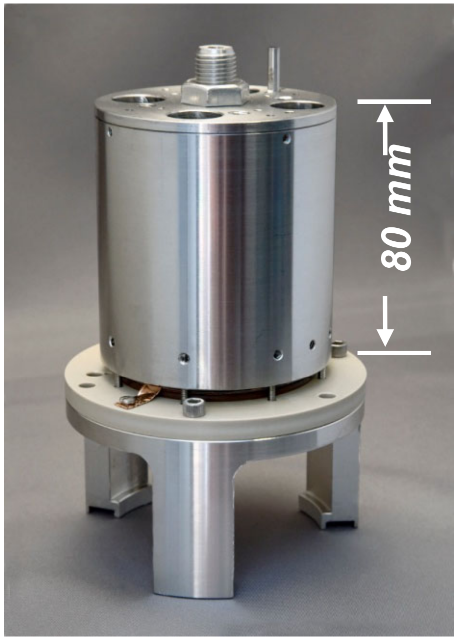
Tools for lithium program



Supersonic Gas Injector 10 ms pulses

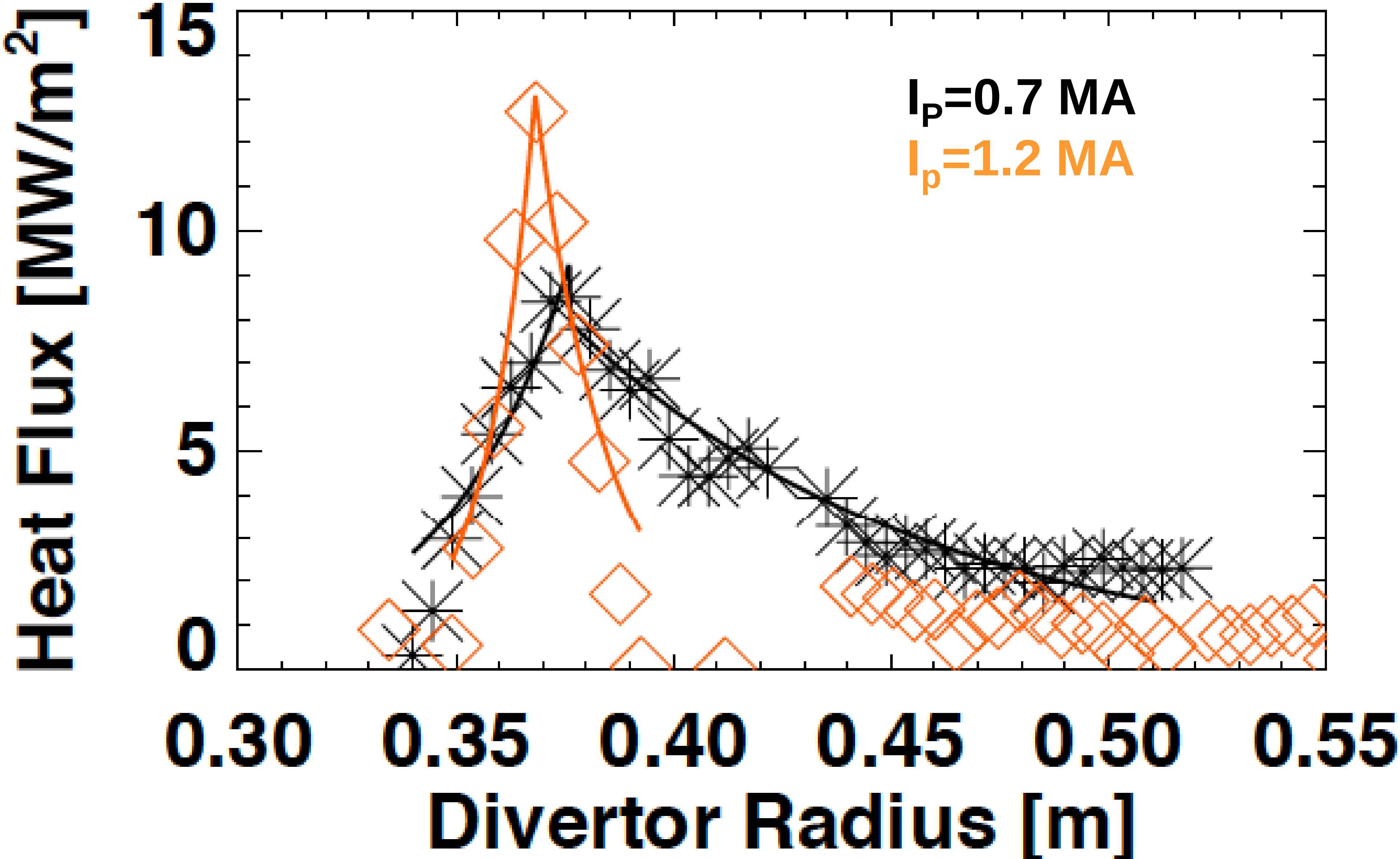
PMI probe. prompt ex-vessel surface analysis

Dual LiTER evaporators ≤ 40 mg Li / min each



Dual Li Droppers ≤ 7 g / min each

Heat Flux Profile Width Narrows with I_p in Lithiated Discharges (New 2-color Camera Data)



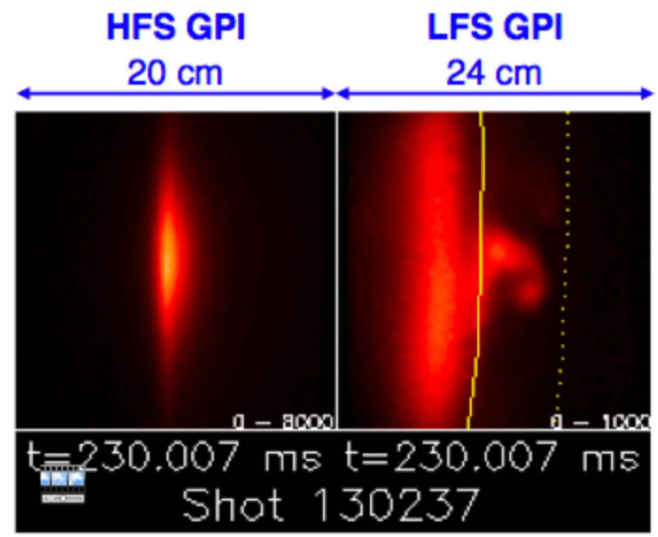
Preliminary

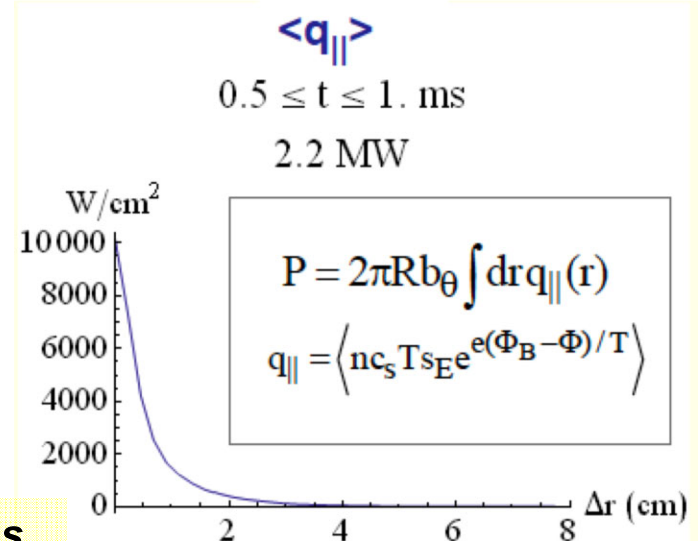
Investigating SOL turbulence and blob properties, relation to divertor heat flux scalings and SOL width

- FY 2009 highlights
  - Statistical analysis of intermittency
    - Divertor, inboard and outboard SOL
  - Local SOL control with asymmetric midplane biasing
  - Modeling of measured SOL profiles and turbulence with SOLT code

- Plans for FY 2010 – FY2012
  - SOL and divertor turbulence scaling with midplane SOL width and major parameters (JRT FY2010)
  - Comparison of edge turbulence properties to theory
    - Poloidal and 2-point correlation in L- & H-mode
    - SOL filament length wrt to magnetic balance
  - Convective cell generation with divertor biasing

In-out turbulence asymmetry measured with gas puff imaging

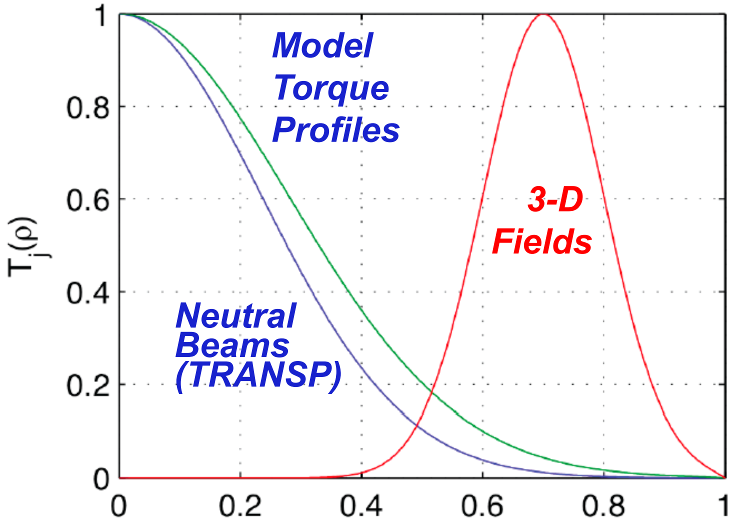


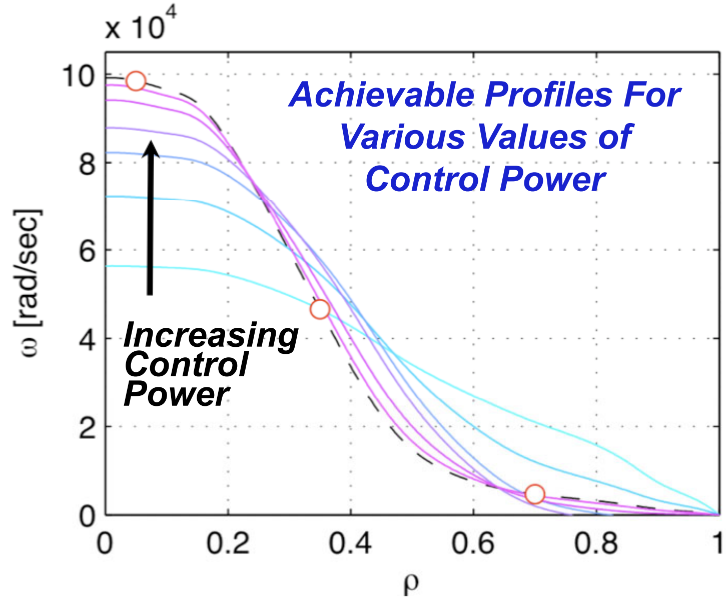


***TOOLS*: Gas Puff Imaging (GPI), fast cameras, new Langmuir probes, biasing electrodes, edge turbulence codes SOLT, BOUT, XGC0**

Development of Real-Time NB Control Enables β_N and Rotation Control

- ☐ β_N control demonstrated in 2009.
- Long-term plan to control the rotation profile.
  - RWM & EF physics as a function of β and rotation.
  - Transport dynamics vs. rotation shear.
  - Pedestal stability vs. edge rotation.
  - ***What is the optimal rotation profile for integrated plasma performance?***
- Use a state-space controller based on a momentum balance model.
  - Neutral beams provide torque.
  - 3-D fields provide braking.
  - Different toroidal mode numbers provide different magnetic braking profiles.
    - Use 2nd Switching Power Amplifier (SPA) for simultaneous n=1,2 &3 fields.
- FY-12 milestone on the *physics*, *measurement*, and *control* aspects of rotation control.
  - Progress in off-line algorithm development.
  - Developing rt-V_ϕ diagnostic for the FY-11.





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