Supported by

Office of Science

NSTX Physics Results from the FY2005 Run

Jonathan Menard

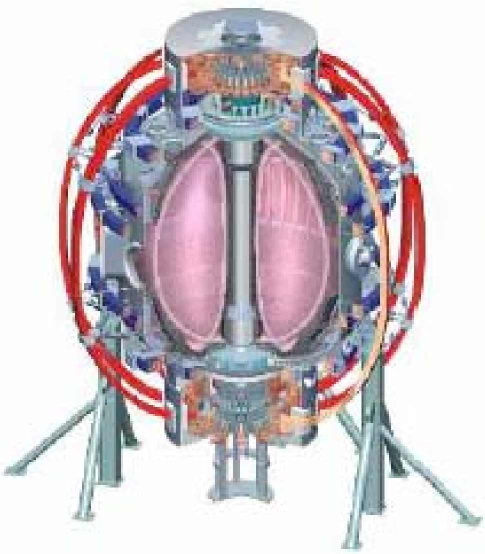
For the NSTX Team

NSTX PAC 19

PPPL

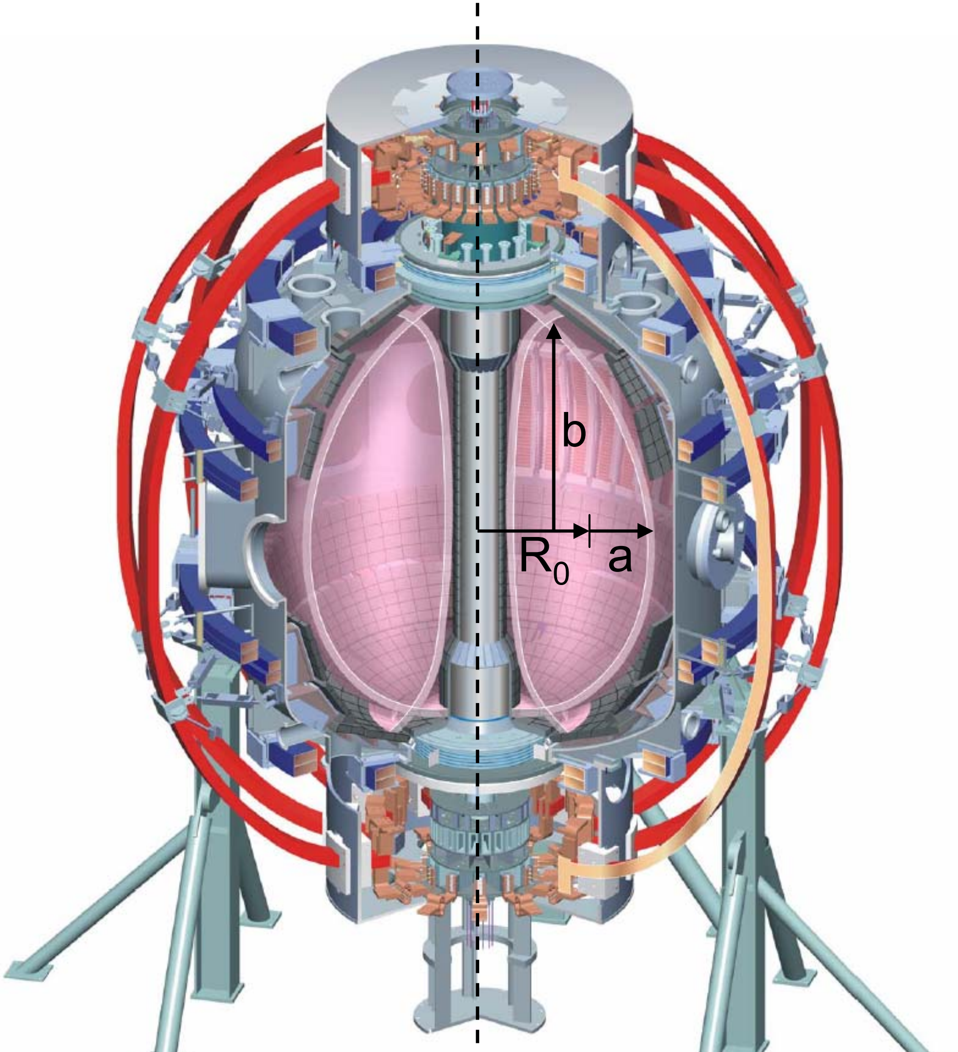
February 22, 2006

College W&M
Colorado Sch Mines
Columbia U
Comp-X
General Atomics
INEL
Johns Hopkins U
LANL
LLNL
Lodestar
MIT
Nova Photonics
New York U
Old Dominion U
ORNL
PPPL
PSI
Princeton U
SNL
Think Tank, Inc.
UC Davis
UC Irvine
UCLA
UCSD
U Colorado
U Maryland
U Rochester
U Washington
U Wisconsin



Culham Sci Ctr
U St. Andrews
York U
Chubu U
Fukui U
Hiroshima U
Hyogo U
Kyoto U
Kyushu U
Kyushu Tokai U
NIFS
Niigata U
U Tokyo
JAERI
Hebrew U
Ioffe Inst
RRC Kurchatov Inst
TRINITI
KBSI
KAIST
ENEA, Frascati
CEA, Cadarache
IPP, Jülich
IPP, Garching
ASCR, Czech Rep
U Quebec

NSTX investigates low collisionality axisymmetric toroidal plasmas at low aspect ratio and high-β



Achieved Parameters

Aspect ratio $A \equiv R_0/a = 1.27$

$\beta \equiv p / (B^2/2\mu_0) \approx 1$, $\langle\beta\rangle = 0.2\text{-}0.4$

Elongation $\kappa = b/a$	2.7
Triangularity δ	0.8
Major radius R_0	0.85m
Plasma Current I_p	1.5MA
Toroidal Field B_{T0}	0.6T
Poloidal flux	1Wb
Pulse Length	1.5s
Te, Ti	**1-4keV**
$\nu^*_{e,i}$	**0.1, 1**

Auxiliary heating & current drive:

RF (30MHz)	6 MW
CHI	0.4MA
NBI (100kV)	**7 MW**

NSTX contributes to fundamental toroidal confinement science in support of ITER and future ST's

- NSTX accesses the plasma β of tokamaks, and extends far beyond this…
  - Improved understanding of roles of toroidicity and β on macro and micro-stability
- Only major US facility studying Li for particle pumping and power handling
  - Power dissipation in divertor very challenging for ITER and devices beyond
- Only ST in world with advanced mode stabilization tools and diagnostics
  - With DIII-D, can validate RWM control methods for ITER
- Unique opportunity for understanding electron gyro-scale turbulence
  - Understanding crucial for all α-particle-heated burning plasmas
- Uniquely able to mimic ITER fast-ion instability drive with full diagnostics
  - Measurement of current profile at high v_{fast} / v_{Alfven} and β_{fast}
- Wave physics in over-dense plasmas ($\omega_{pe} > \Omega_{ce}$)
  - Developing EBW and HHFW heating and CD tools for ST and high-β AT
- Compact geometry + high β for attractive fusion applications:
  - Component Test Facility (CTF) for nuclear testing of reactor components
  - Successful CTF could lead to attractive fusion reactor
  - Both applications require non-inductive current formation techniques (CHI)

NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

The ST configuration provides access to high β plasmas

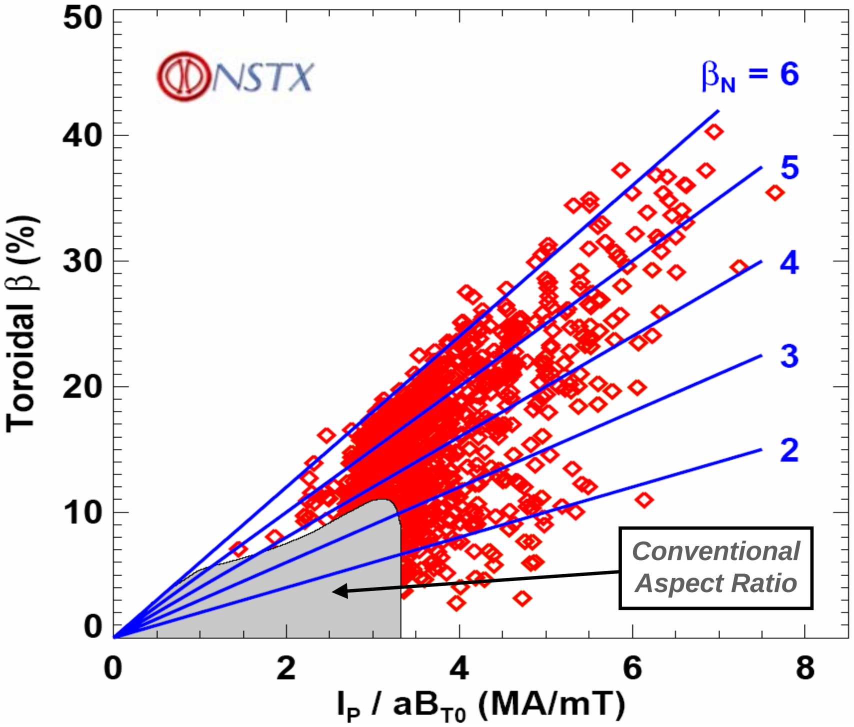
- **Ideal MHD energy principle → Max(β_T) ~ $\beta_N \times I_P / aB_T$**
- **Low A → higher I_P / aB_T stable to dangerous long λ modes**
- **Low A → higher β_N limit w.r.t. short & long λ instabilities**

⇒ Paramagnetic plasma: local β → 50%, diamagnetic: local β → 100%

- **Steady-state ⇒ *need* most of current to be *self-generated* by “bootstrap” effect**
  - Fraction of “bootstrap” current $f_{BS} = I_{BS} / I_P$
- **Efficient reactor ⇒ maximize f_{BS} & β_T**

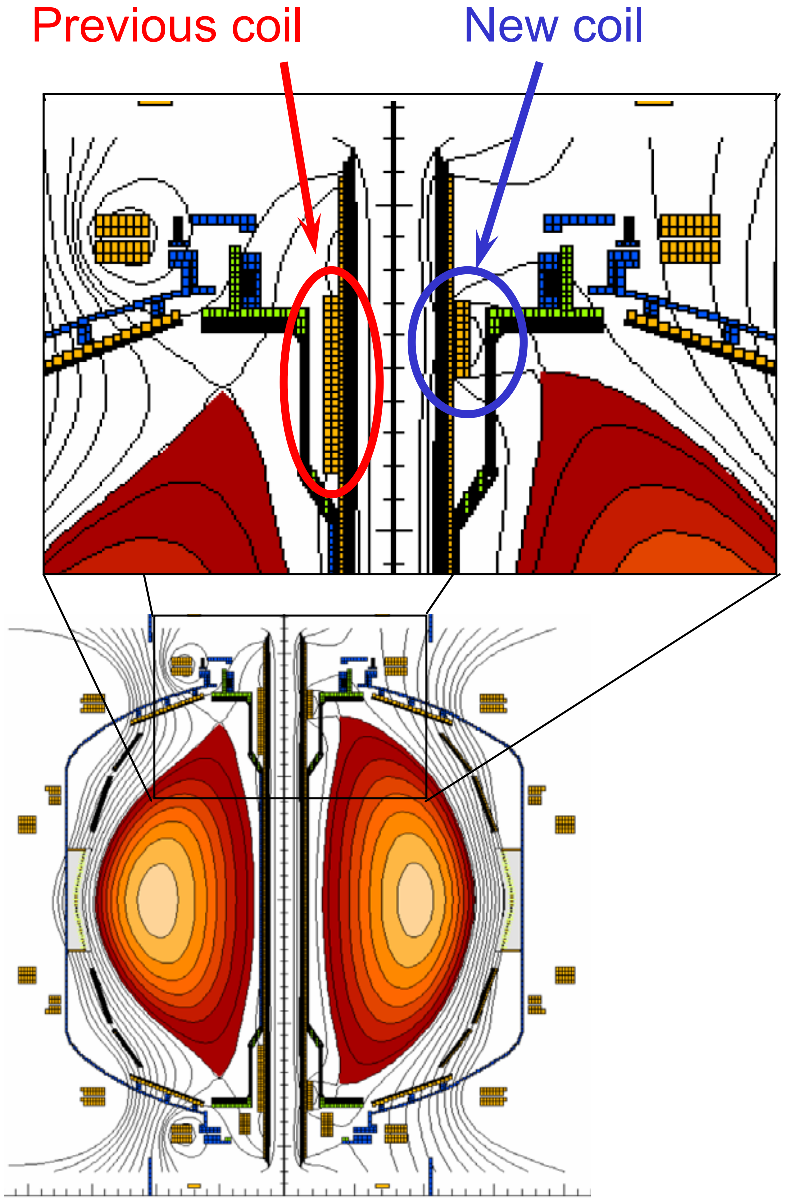
$$f_{BS}\,\beta_T \sim C_{BS}\sqrt{\varepsilon}\left(\frac{1+\kappa^2}{2}\right)\left(\frac{\beta_N}{2}\right)^2$$

- **Motivates elongating plasma (high-κ)**
- **Motivates high normalized β (high β_N)**
  - High β_N at high κ requires high triangularity δ
  - Plasma “spin” and/or active feedback control required to stabilize “Resistive Wall Mode” (RWM) above ideal no-wall stability limit



ST plasmas operate at significantly higher β_N and I_P / aB_T than is possible (i.e. stable) at conventional aspect ratio

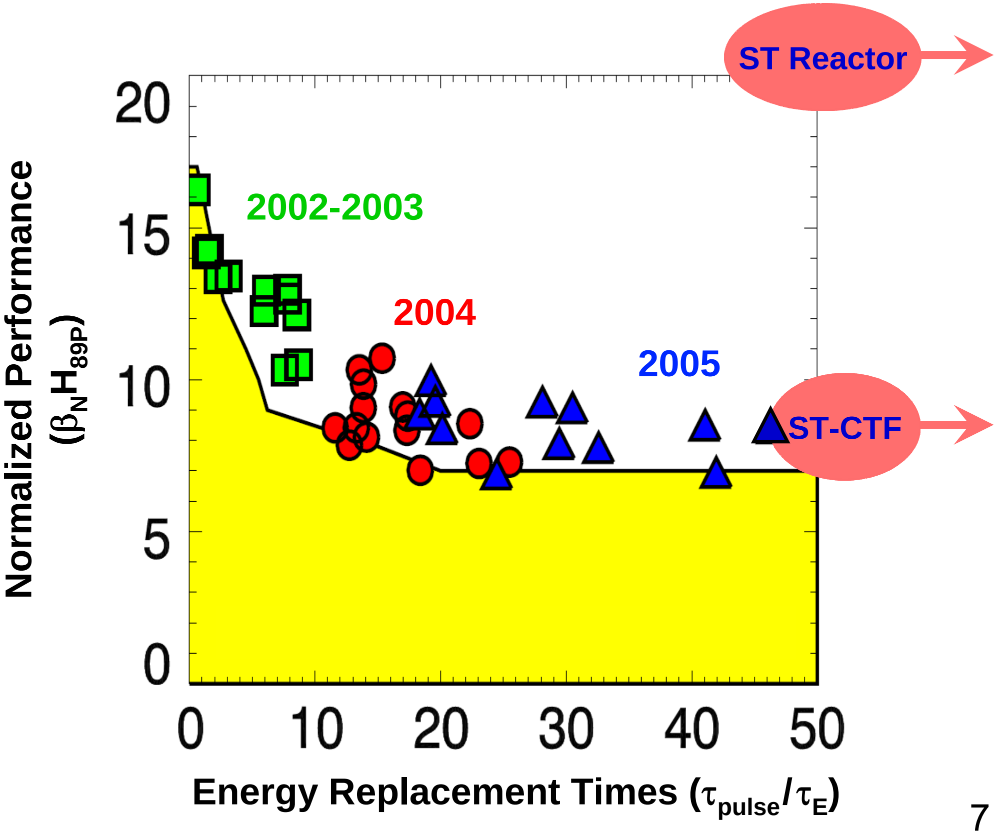
New divertor poloidal field coils provide enhanced ELM stability and plasma shaping capabilities



High stable β_N at high κ requires plasma to be "D" shaped (high triangularity δ)

$\Rightarrow$ **new NSTX coils provide this shape**

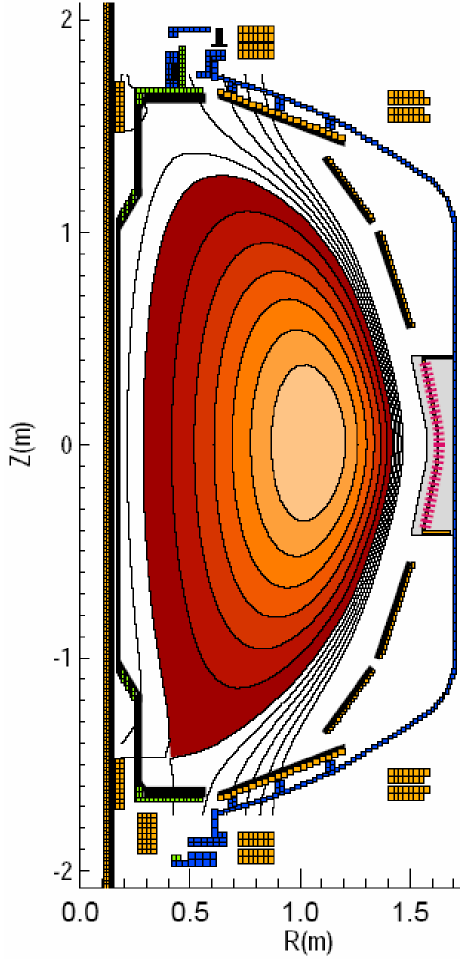
$\Rightarrow$ **NSTX record pulse lengths at f_{BS}=50% and normalized performance of CTF**

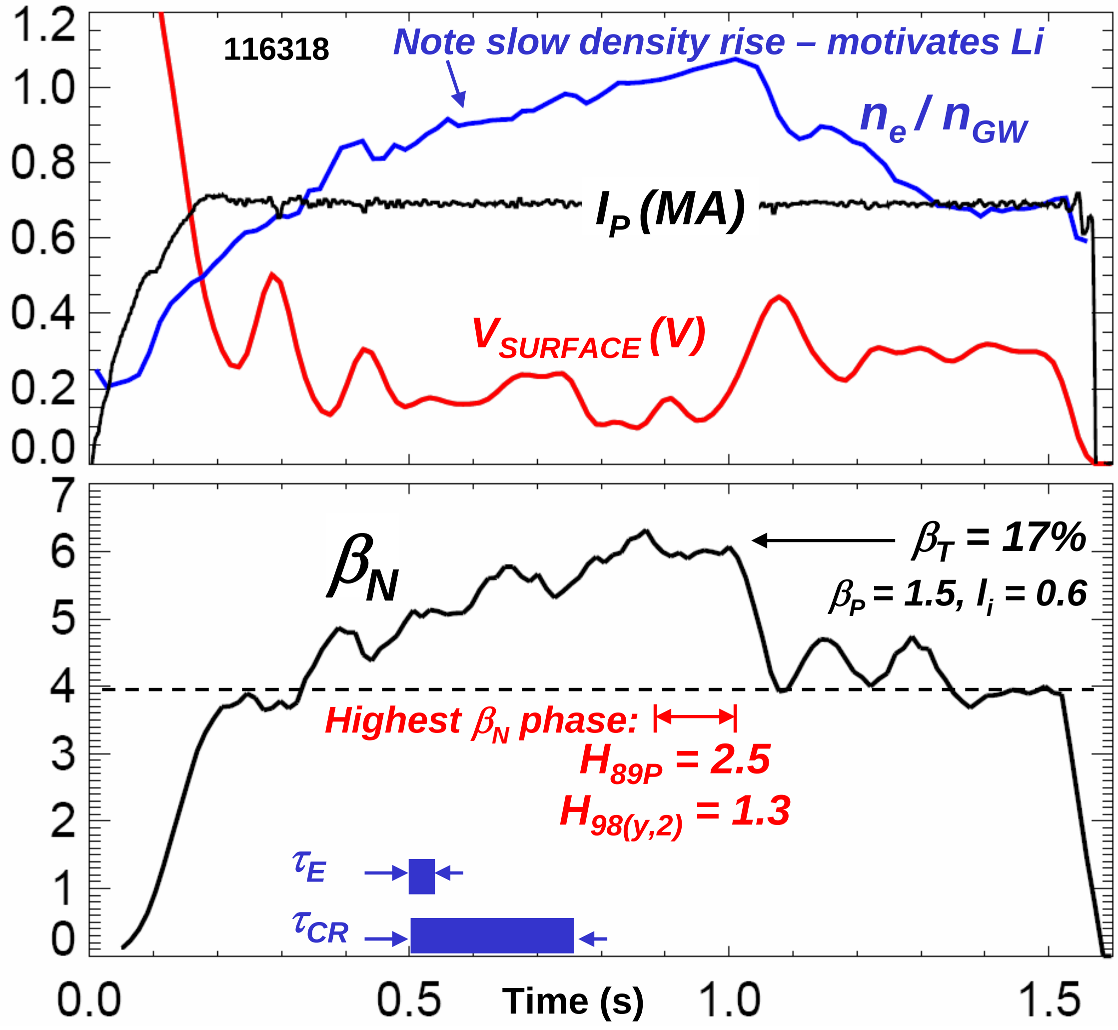


Record discharge pulse-lengths have been achieved by operating with sustained H-mode and high β_N

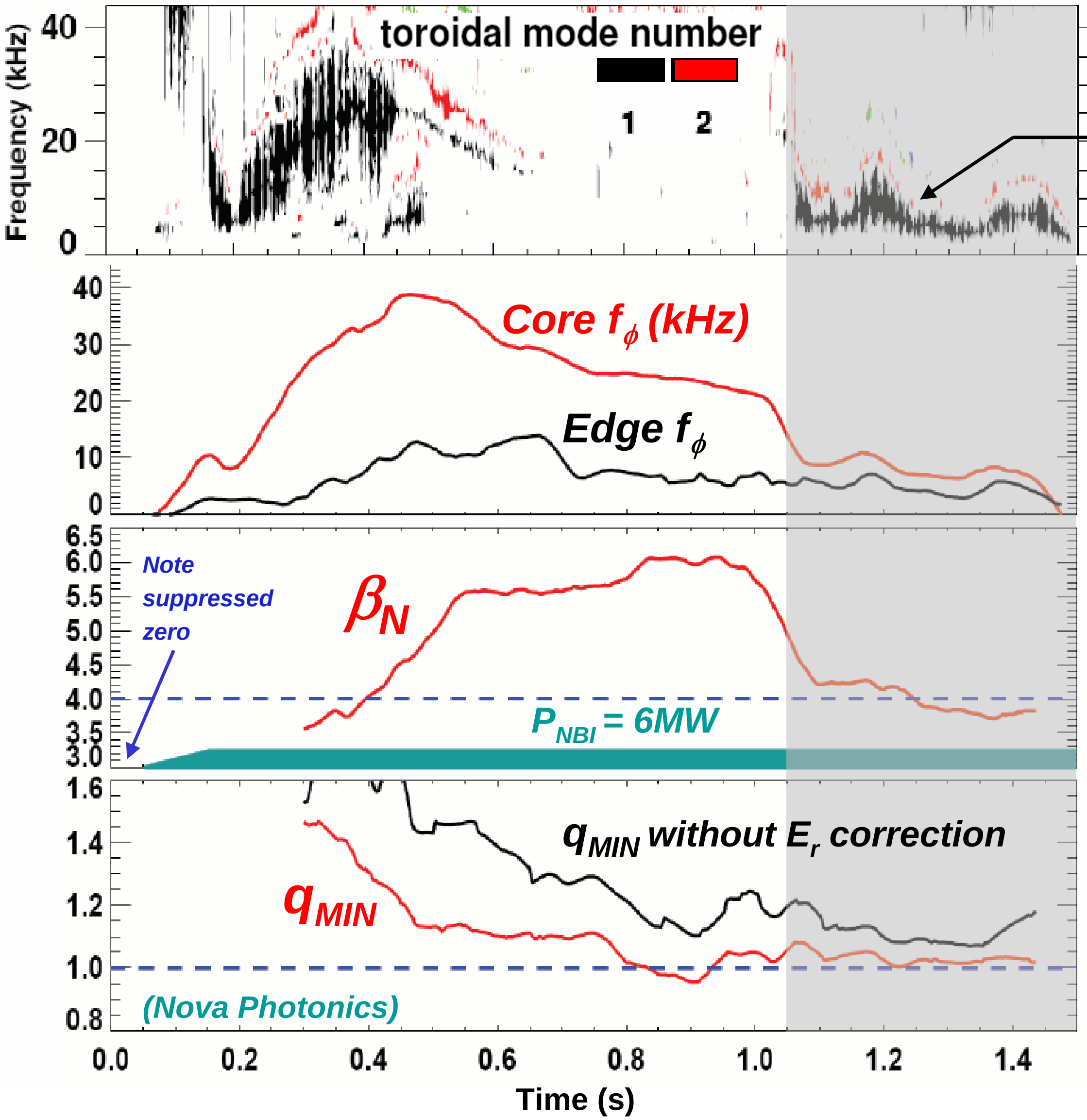
- H-mode with small ELMS ⇒ reduced flux consumption, slow density rise
- $\beta_N > 4$ for $\Delta t > 1$s at high $\beta_P > 1$ increases bootstrap fraction, lowers V_{LOOP}

$\kappa = 2.4$, $\delta_L = 0.77$, $\delta R_{SEP} = -1$cm





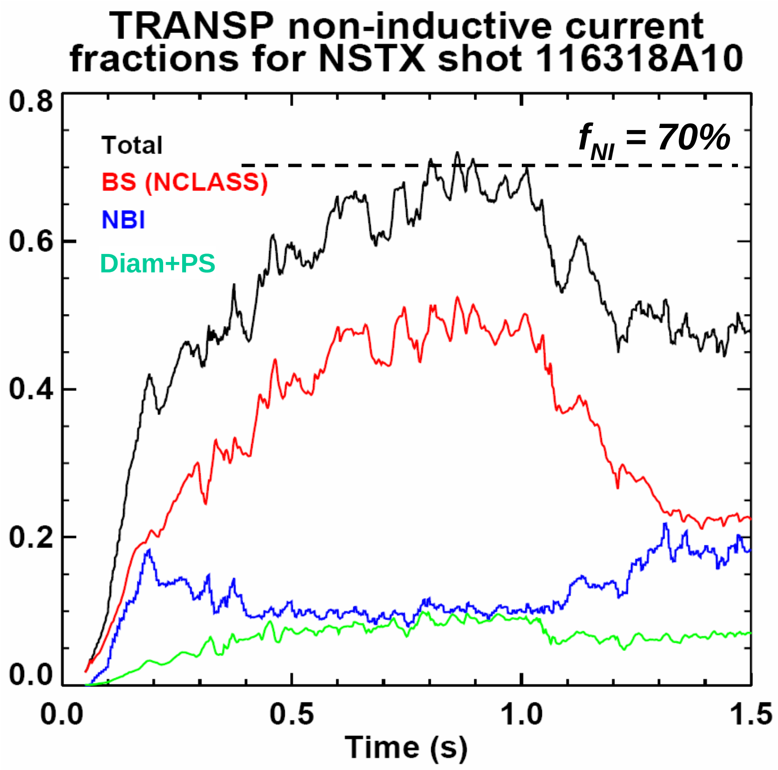
Onset of saturated n=1 mode degrades confinement, but results in a long-pulse "hybrid" state with q_{MIN} sustained near 1

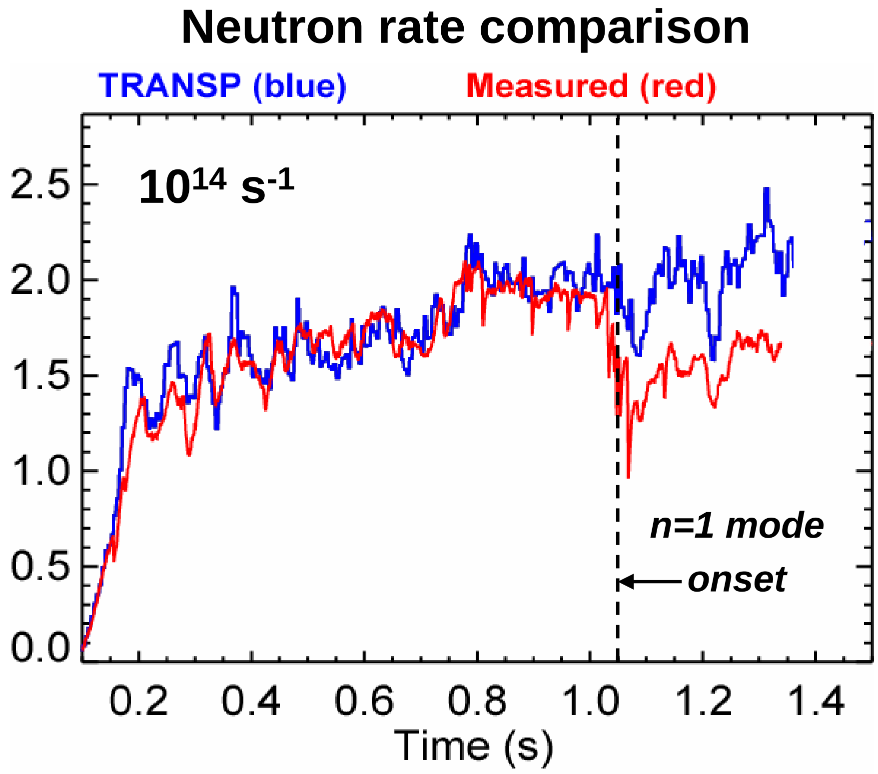


- Saturated n=1 mode persists for 0.5s late in discharge evolution
- Central rotation drops by factor of 3 at mode onset
  - Edge f_ϕ maintained
  - T_i / T_e → 1 (not shown)
- β_N = 6 decreases to 4
  - β_N = 6 above no-wall limit
  - β_N = 4 near no-wall limit
    - No RWM observed…
- q_{MIN} sustained near 1
  - No sawteeth observed
  - Discharge runs out of OH flux and TF flat-top
  - **Possible "hybrid" mode, physics relevant to ITER**

Longest duration discharges approach 70% non-inductive current fraction prior to onset of n=1 mode

- **85% of non-inductive current is ∇p-driven = BS + Diamagnetic + PS**





- Investigating role of MHD in diffusion/loss of fast-ions and CD in elevating q(0)
  - TRANSP agrees with measured neutron rate to within ± 10% during high-β phase
  - TRANSP over-predicts neutron rate during late n=1 activity (assumes no anomalous diffusion)

Hybrid scenario is high priority research area in ITPA SSO group

See presentation by D. Gates for more on long-pulse operation

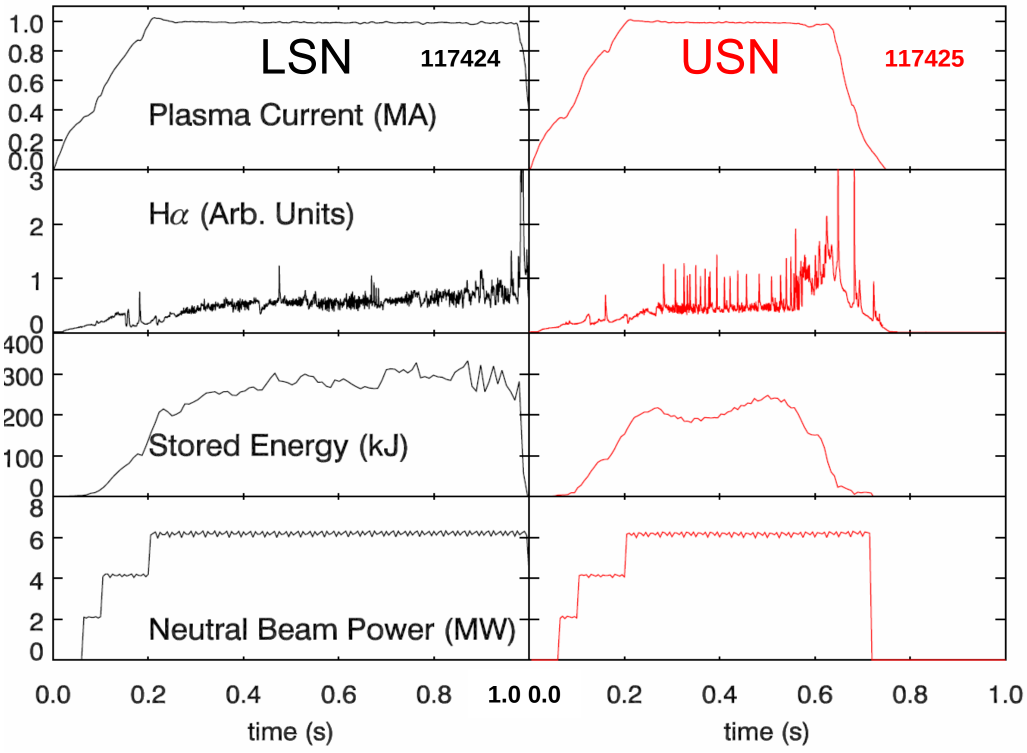
NSTX Research Highlights from 2005 Run

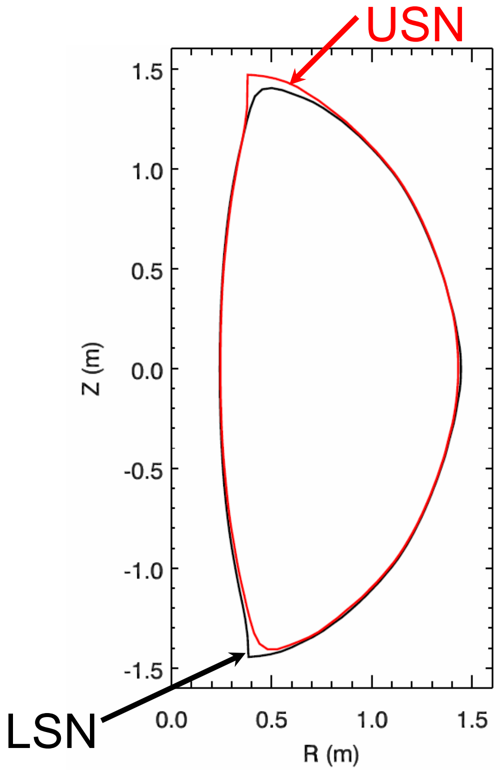
- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

Precise control of X-point balance important to performance

Made possible by using rt-EFIT for shape control – collaboration with GA

- Very small changes in the plasma boundary reproducibly lead to large differences in edge stability – “edge-localized-mode” (ELM) instabilities
- ELMs have a major impact on performance - controlling them is crucial





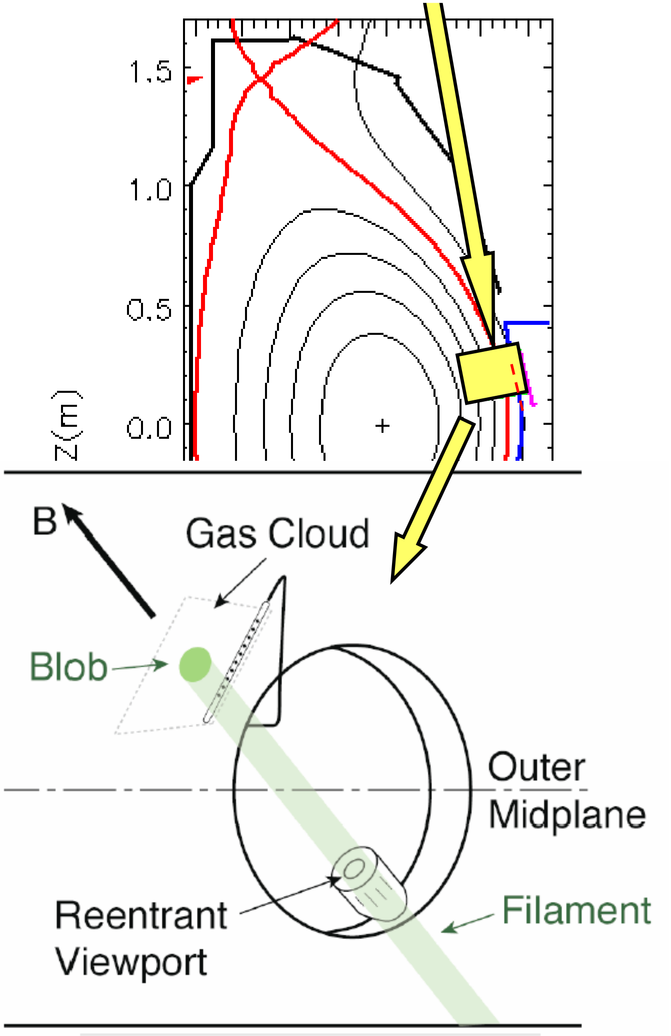
Large heat pulses from ELMs can erode divertor material – important for ITER and CTF

Small ELM scenarios high priority in ITPA PEP group

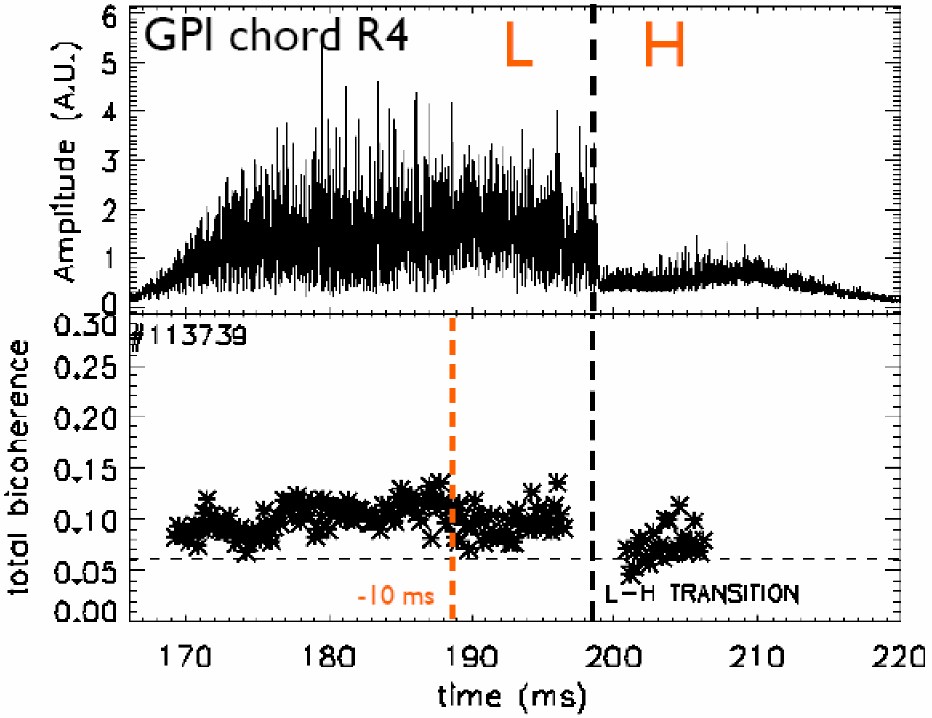
See presentation by R. Maingi for more on pedestal physics

Gas-puff imaging (GPI) diagnostic provides high-time-resolution diagnosis of near-edge transport phenomena and ELMs

Viewing area just above midplane on outer edge

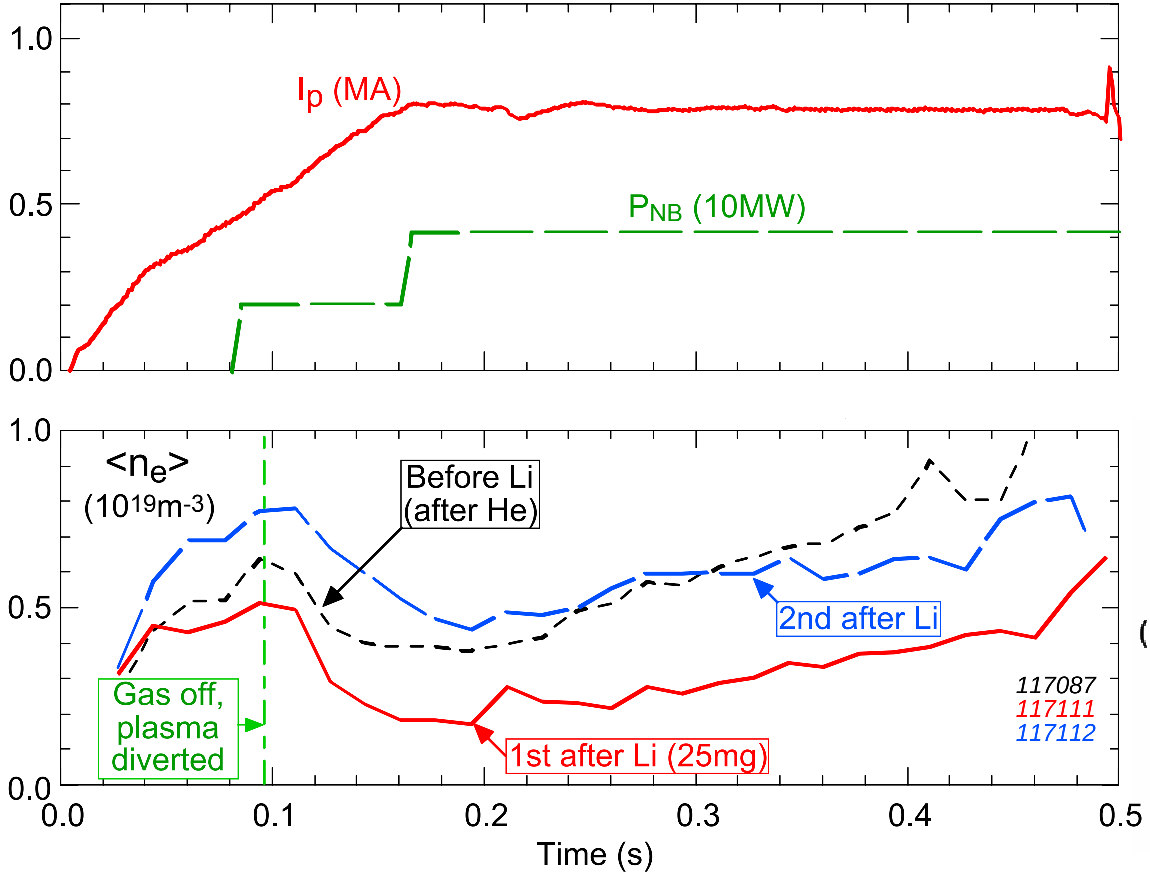


- Example: turbulence and L-H transition
  - Testing hypothesis that Reynold’s stress induced zonal/shear flow suppresses turbulence ⇒
    - Expect increase in pre-transition low/high-f bicoherence
  - Most shots exhibit *no* increase in bicoherence before L-H transition – different from DIII-D result

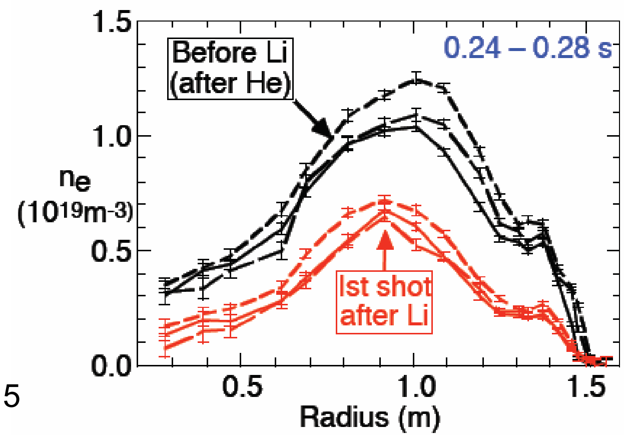


Particle pumping was achieved using Li deposition on lower divertor - exhibited factor ~2 decrease in density

…but required injecting Li pellets for many discharges



Lower single-null divertor discharges,
0.45T,
D_2 gas fueled 3.5mg



- 25mg of Li pumping of edge density saturated after 3 discharges
  - ***Li evaporator in 2006 ⇒ deposit more Li, improved control***

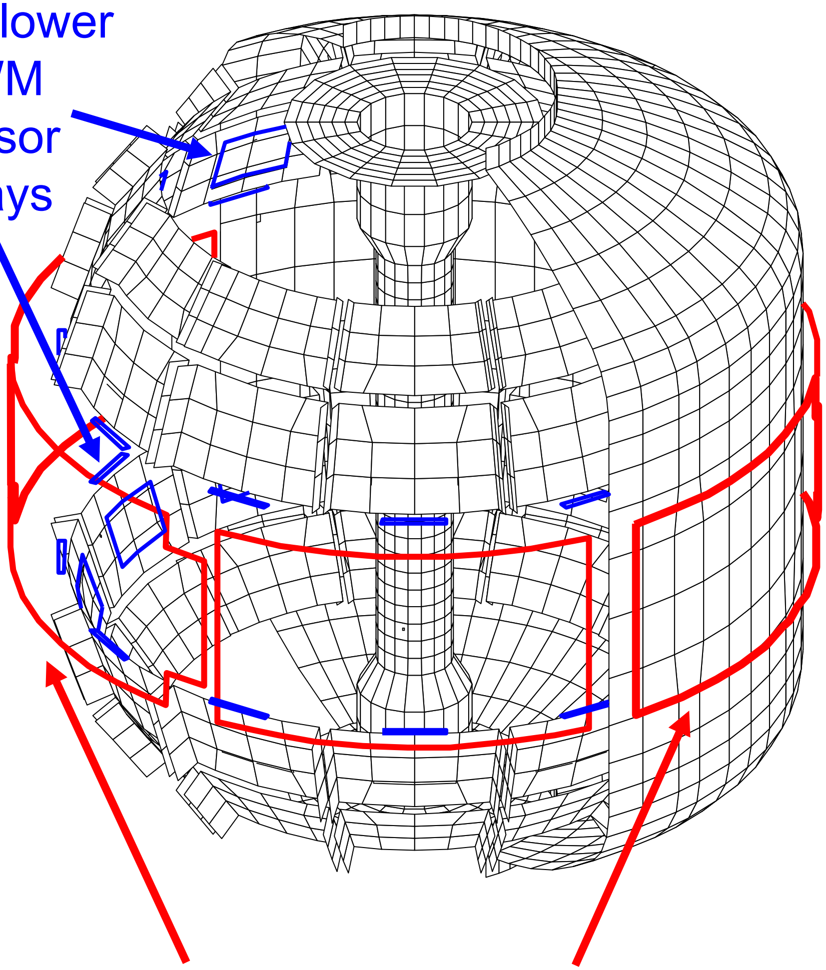
See presentation by H. Kugel for more on Li and recycling control

NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

Non-axisymmetric coils for advanced MHD mode stability and control experiments

- Six coils spanning midplane
  - ❑ *Toroidal mode # n* = 1 - 3
  - ❑ DC or AC standing waves
  - ❑ Toroidally propagating fields
- Recent experiments addressed:
  - ❑ Error field physics/correction/control
    - Static and ramped n=1 fields
    - Island formation
    - Rotation damping
  - ❑ Physics of marginally stable RWM
    - MHD Spectroscopy
      - → Toroidally propagating n = 1 field
    - Resonant field amplification
  - ❑ Rotation control
    - Static “non-resonant” n=3 fields
- 2006 experiments will emphasize <u>feedback</u> control of EF and RWM

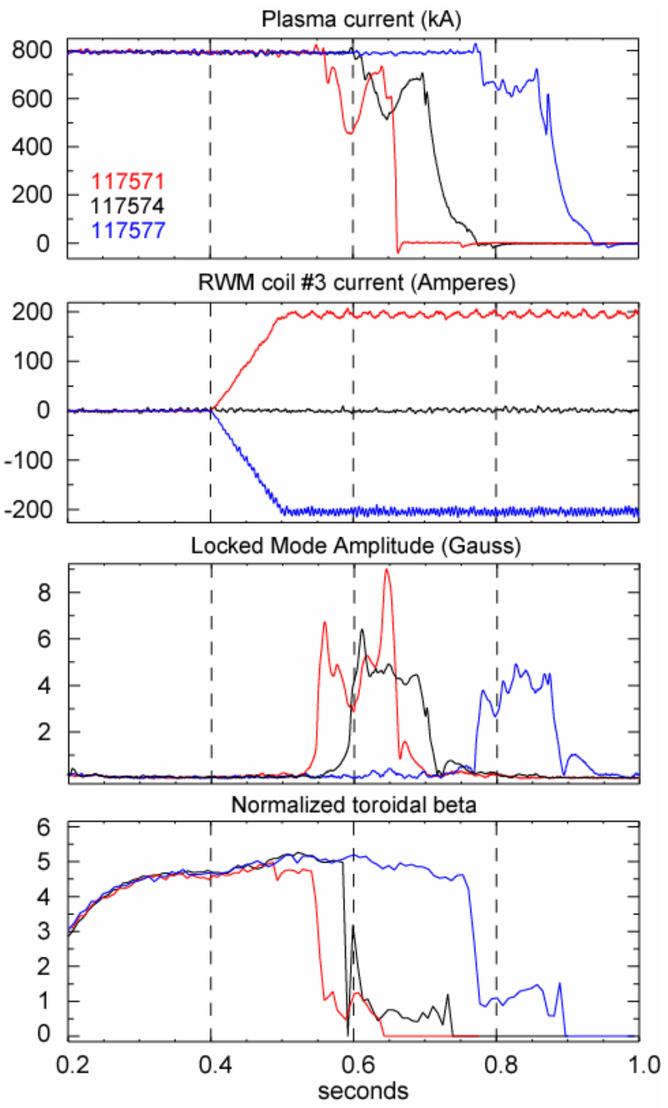


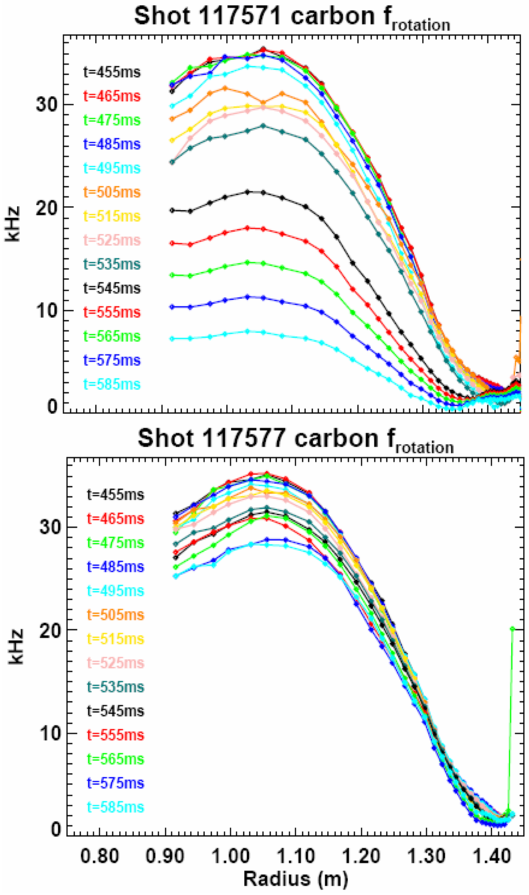
Upper/lower RWM Sensor Arrays

Non-axisymmetric ex-vessel coils

Plasma/conductor/coil geometry similar to US proposal for ITER

Pulse-lengths have been extended at high β_N by reducing intrinsic n=1 error-fields (EF) using the non-axisymmetric coils





- Applying field in “non-correcting” direction damps rotation, leading to earlier locked mode (LM) and/or RWM formation

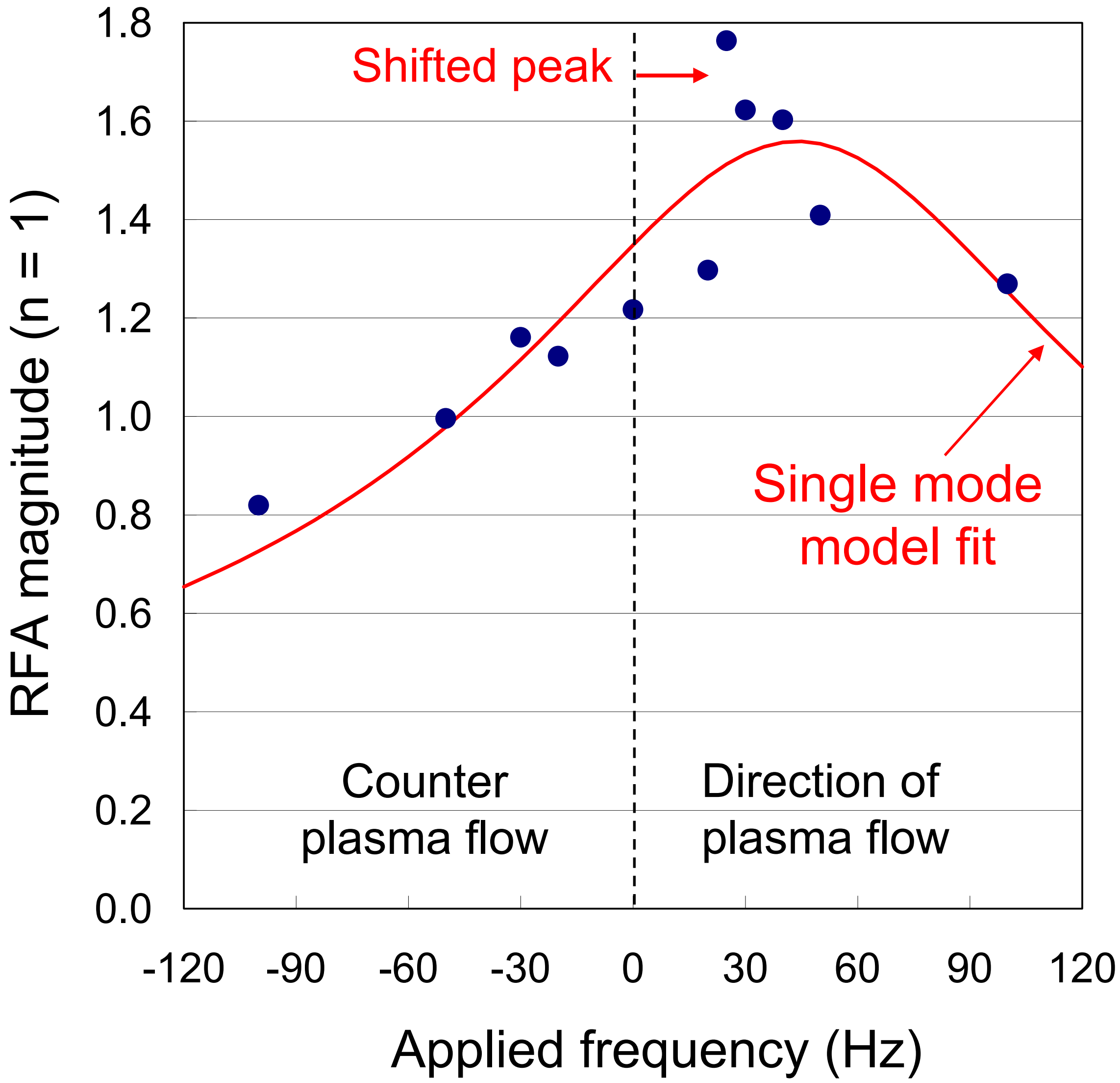
- Applying field in “correcting” direction, allows sustained central rotation. Near-edge rotation locking is avoided - extending pulse length at high-β

2006 experiments → “Dynamic” EF correction

- ***Measure plasma response to EF in real-time***
- ***Slowly feedback control EF to small value***

EF & LM physics high-priority research area in ITPA MHD group

MHD spectroscopy - measured n=1 resonant field amplification (RFA) of RWM stabilized by plasma rotation



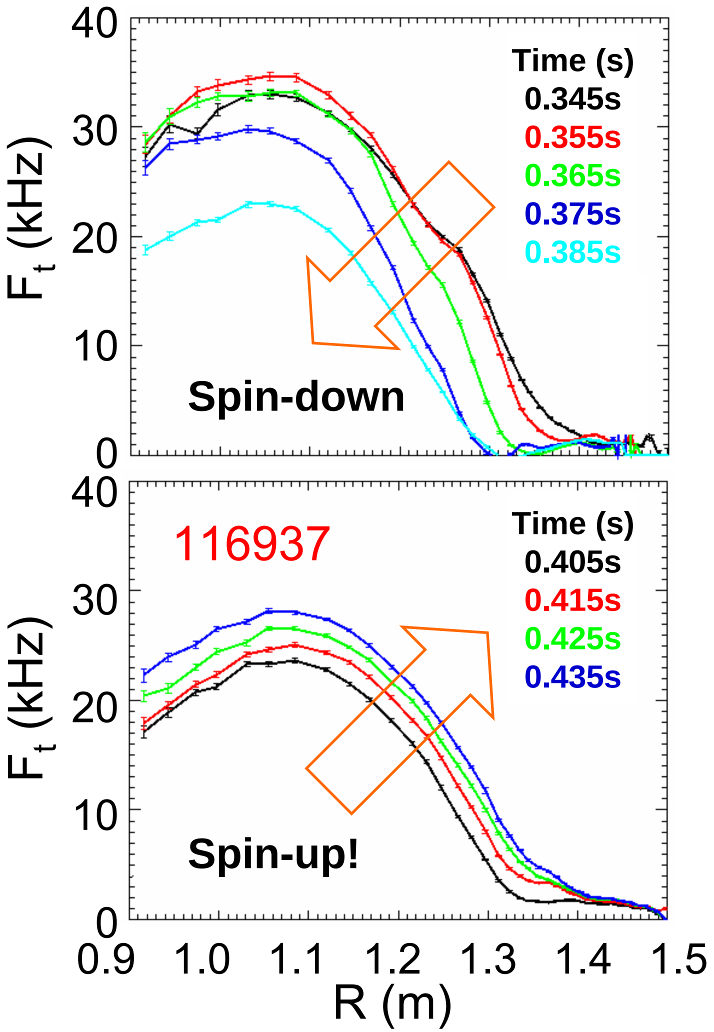
$$\text{RFA} = \frac{B_{\text{plasma}}}{B_{\text{applied}}}$$

- Applied field phased to create traveling wave in toroidal direction
- Peak in RFA(ω) shifted in direction of plasma flow
  - ❑ Expected by RWM theory / experiment
- Confirms resonance with weakly damped, slowly propagating mode = RWM

RWM physics high-priority research area in ITPA MHD group

Rotation control: n=3 non-resonant externally applied fields decrease the plasma rotation initially near edge, later in core

Rotation decreases during n=3,
Increases when n=3 removed



- Coils have allowed detailed tests of 3D neoclassical toroidal viscosity (NTV) models
  - Damping stronger & more controllable on NSTX
- Study impact of flow on thermal transport
- Very useful for MHD stability studies:

2006-7 experiments →

Simulate ITER RWMs

1. ***Slow plasma rotation to destabilize RWM***
2. ***Feedback stabilize RWM***

Complements DIII-D RWM control research

RWM control system design for ITER

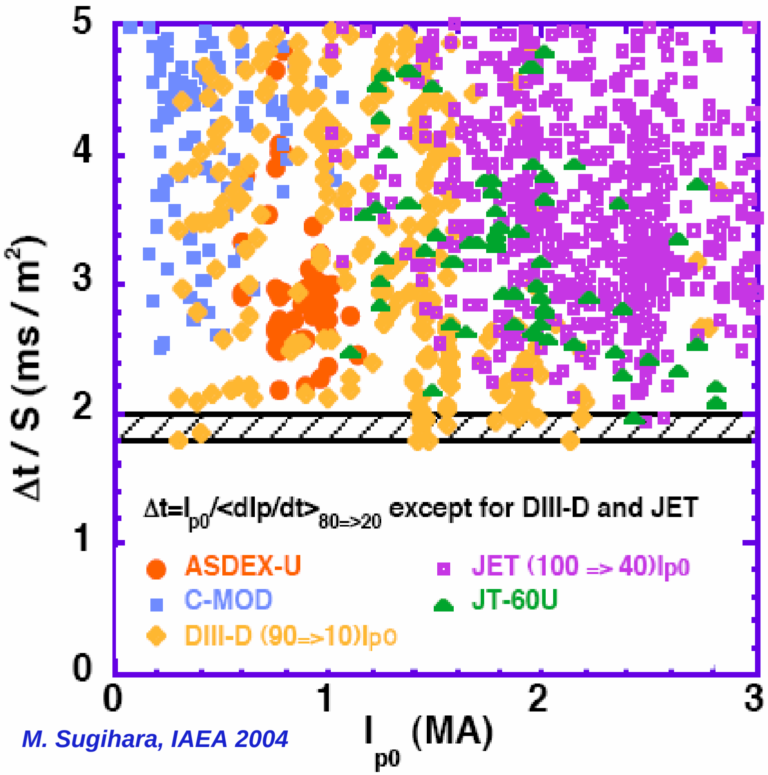
high-priority research area in ITPA MHD group

NSTX is making unique low-A contributions to disruption physics understanding needed for ITER and CTF

Contributing to new ITPA disruption database to determine current-quench-time scaling for ITER blanket module stress calculations

L/R decay time τ proportional to plasma cross-sectional area S $\frac{\tau}{S} = \frac{L_p^{eff}/2\pi R_0}{\eta_p}$

***Tokamak data:* τ/S ≥ 1.6ms/m²**

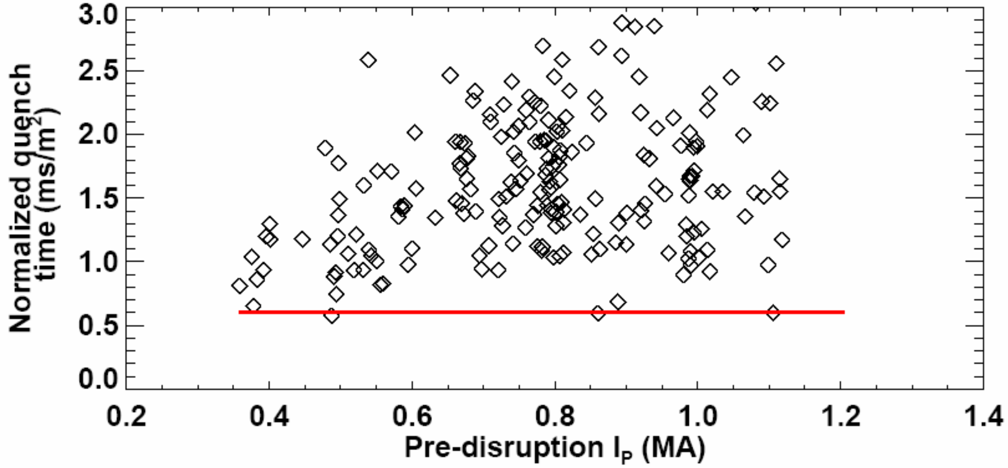


M. Sugihara, IAEA 2004

Inductance also proportional to <u>aspect ratio</u>

- **NSTX has 2-3 × lower τ/S**
  - **consistent with expected scaling**
- **Supports basic model of T_e= 5-10eV with impurity line radiation balancing P_{OHMIC} during current quench phase**

***NSTX data:* τ/S ≥ 0.6-1ms/m²**



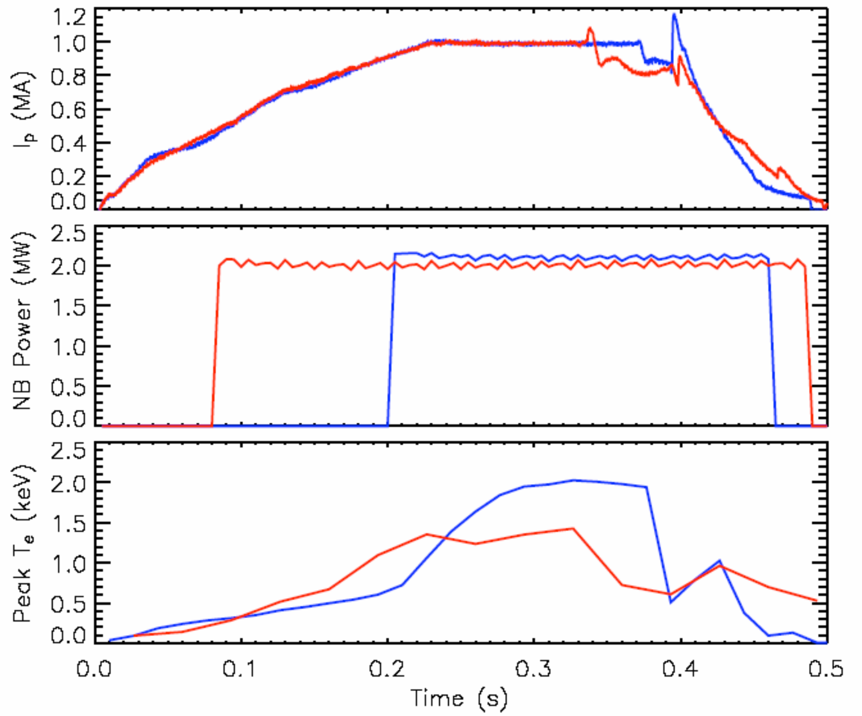
See presentation by A. Sontag for more on MHD

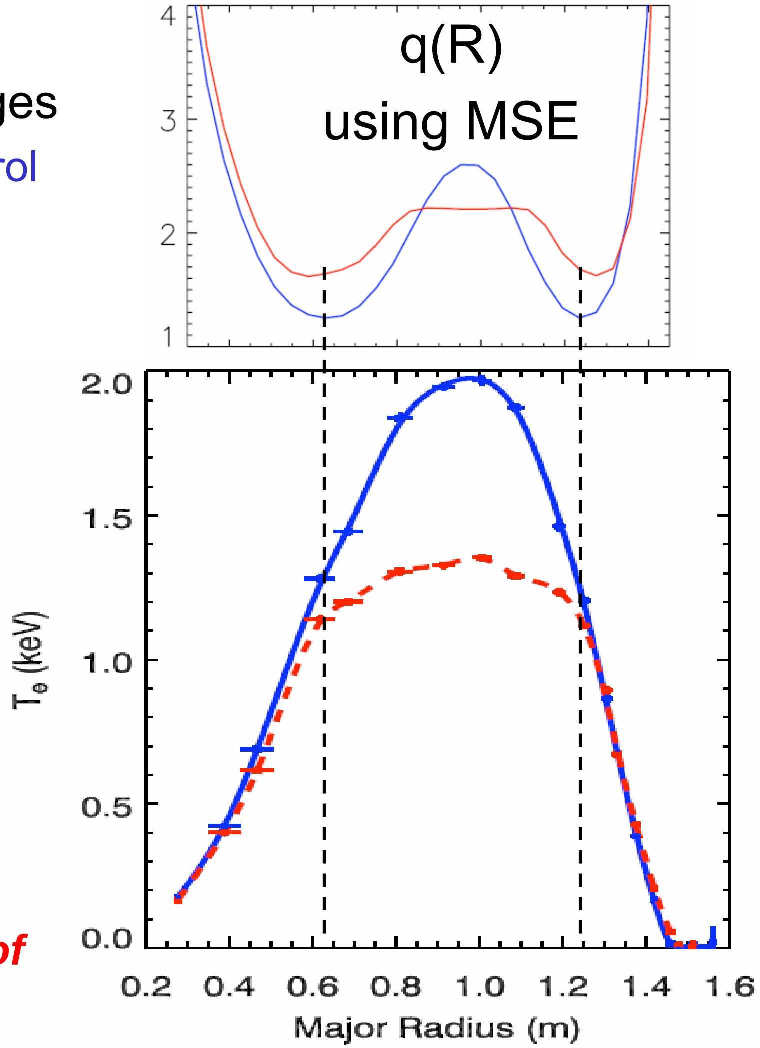
NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

In low-density L-mode, improved electron energy confinement correlates with degree of magnetic shear reversal

- τ_E = 60-80ms – better than standard "H-mode"
  - up to 100ms with late H-mode transition
- Inboard limited 1MA, 4.5kG L-mode discharges
  - nearly DN diverted → few mm inner gap control

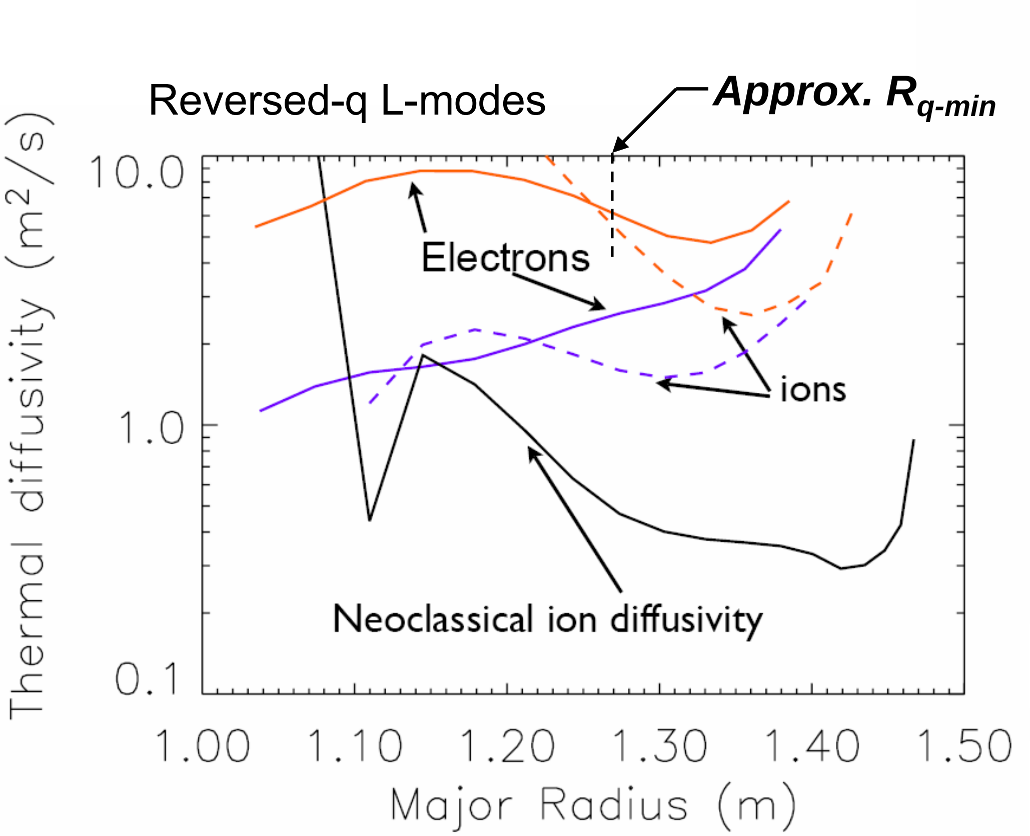




Control of transport may be possible via control of J profile → RF wave current drive (HHFW & EBW)

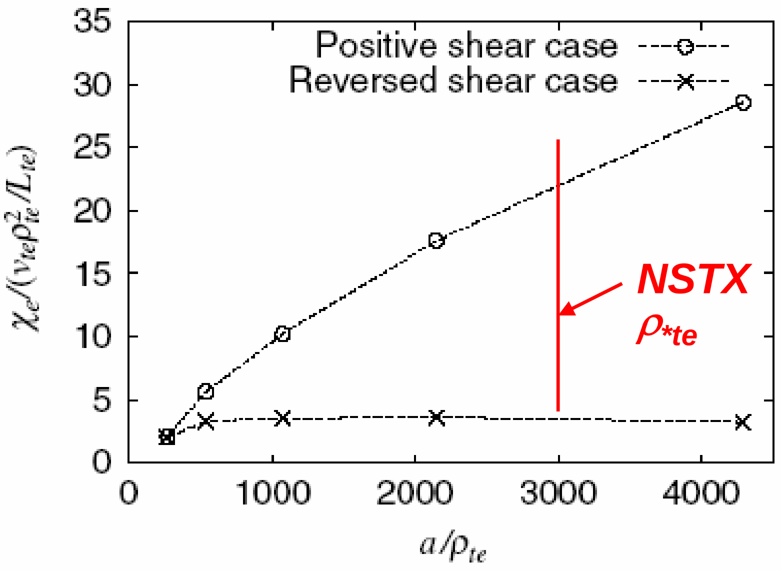
Measurements of electron thermal diffusivity vs. magnetic shear will test models of anomalous electron thermal transport

Blue curves, with stronger shear reversal, have lower electron and ion thermal diffusivities



- **EXAMPLE:**

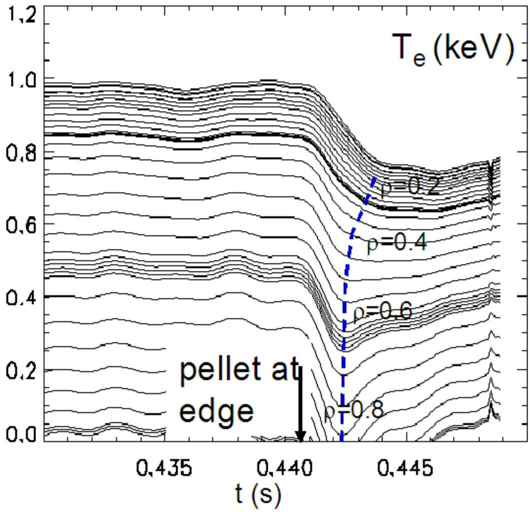
  Electron Temperature Gradient (ETG) micro-instability-driven turbulence is predicted to be sensitive to magnetic-shear

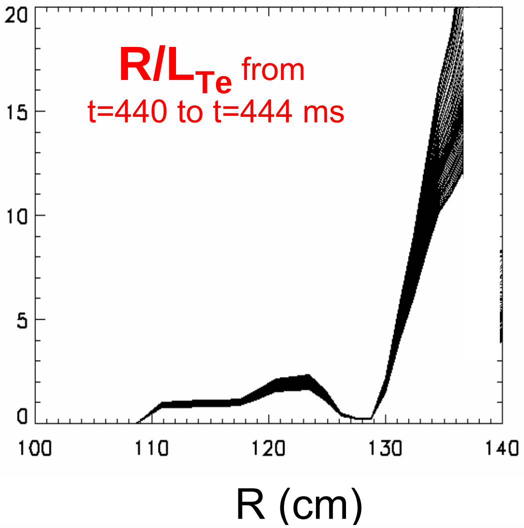


Y. Idomura, S. Tokuda and Y. Kishimoto
Nucl. Fusion 45 (2005) 1571–1581

Lithium pellet injection into the edge probes electron thermal transport by inducing edge T_e perturbation

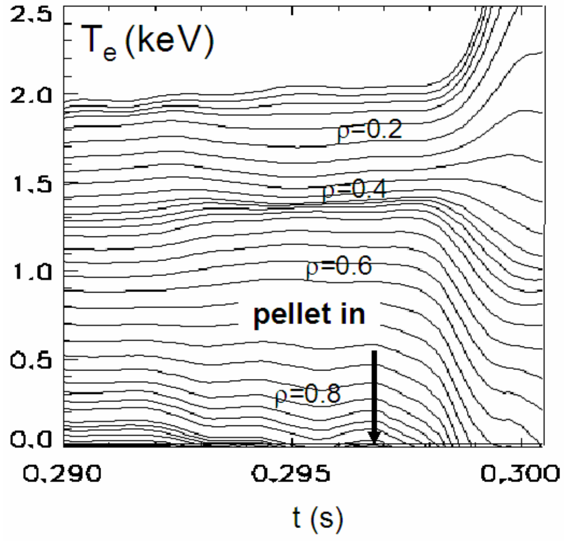
- **“Two-color” SXR measures T_e profile evolution with high time resolution**

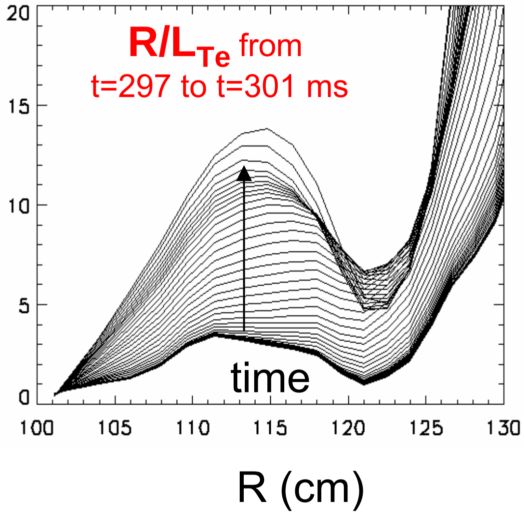




6MW monotonic-q H-mode

- Core T_e profile exhibits clear critical gradient behavior – ETG?





2MW reversed-q L-mode

- Core SXR T_e actually ***increases*** ⇒ very different transport properties

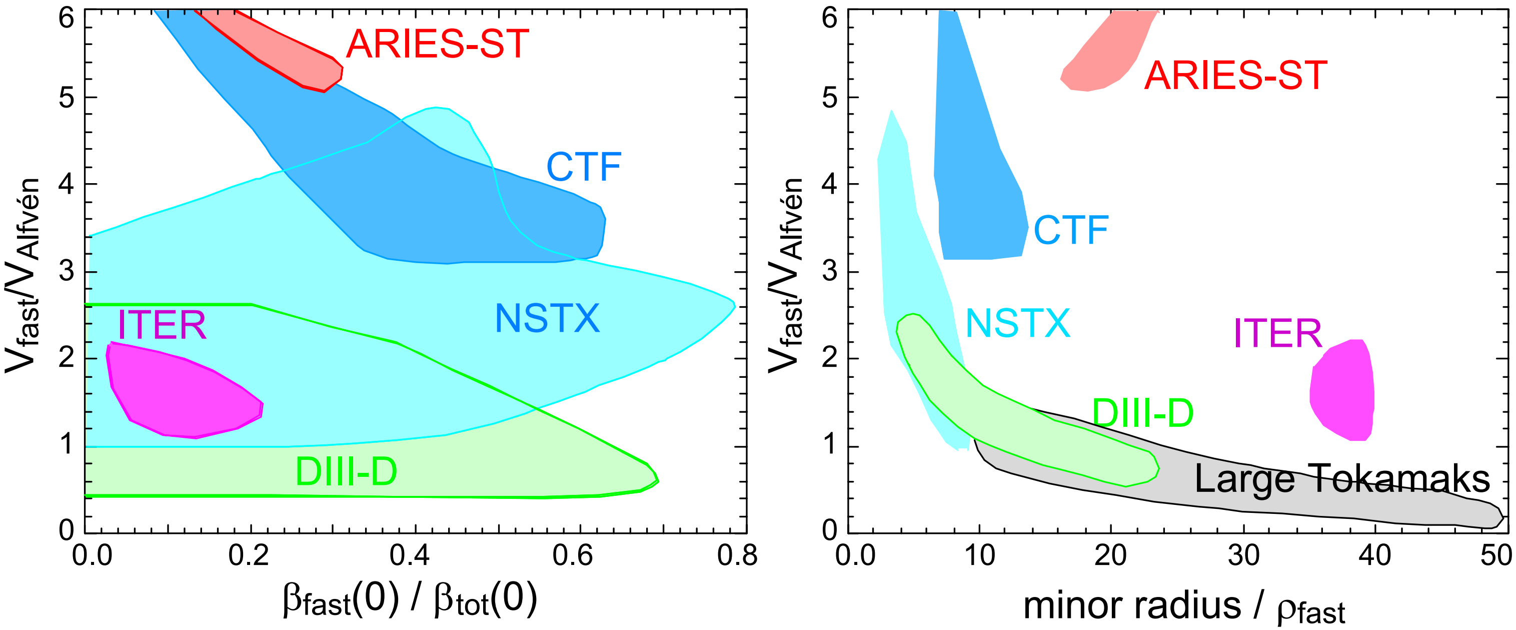
See presentation by M. Bell for more on transport and turbulence

NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

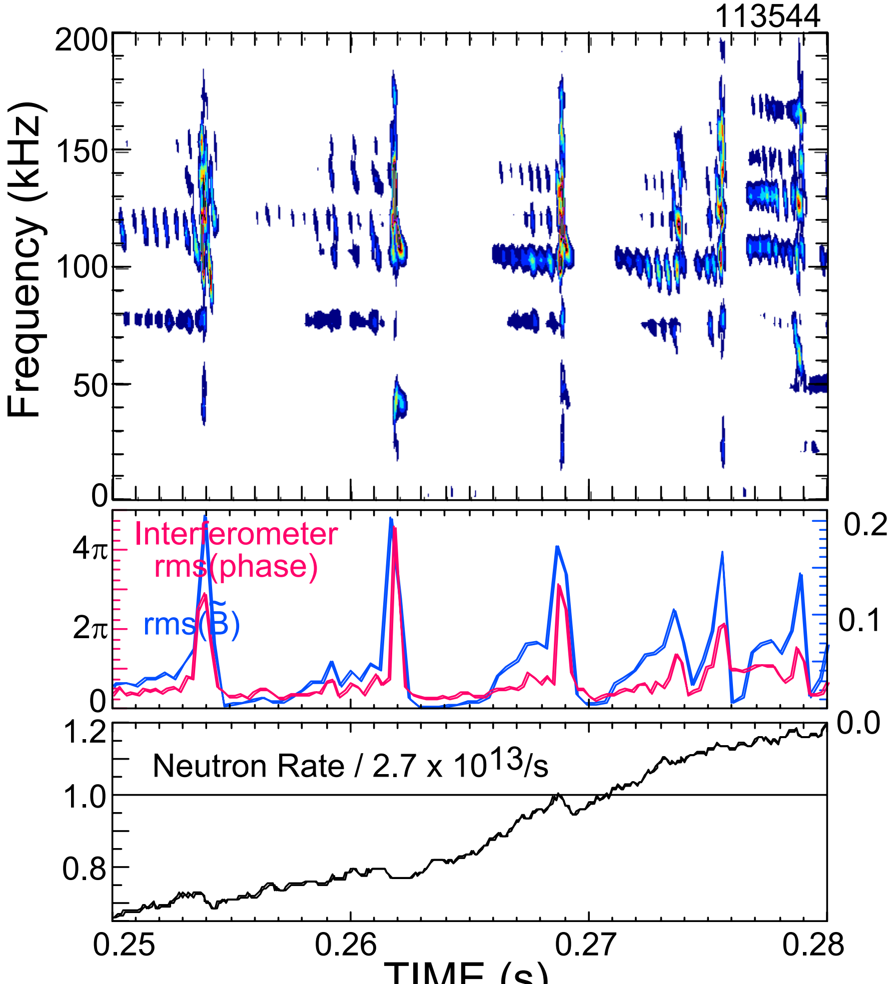
ITER will operate with a large, super-Alfvénic, fast ion population

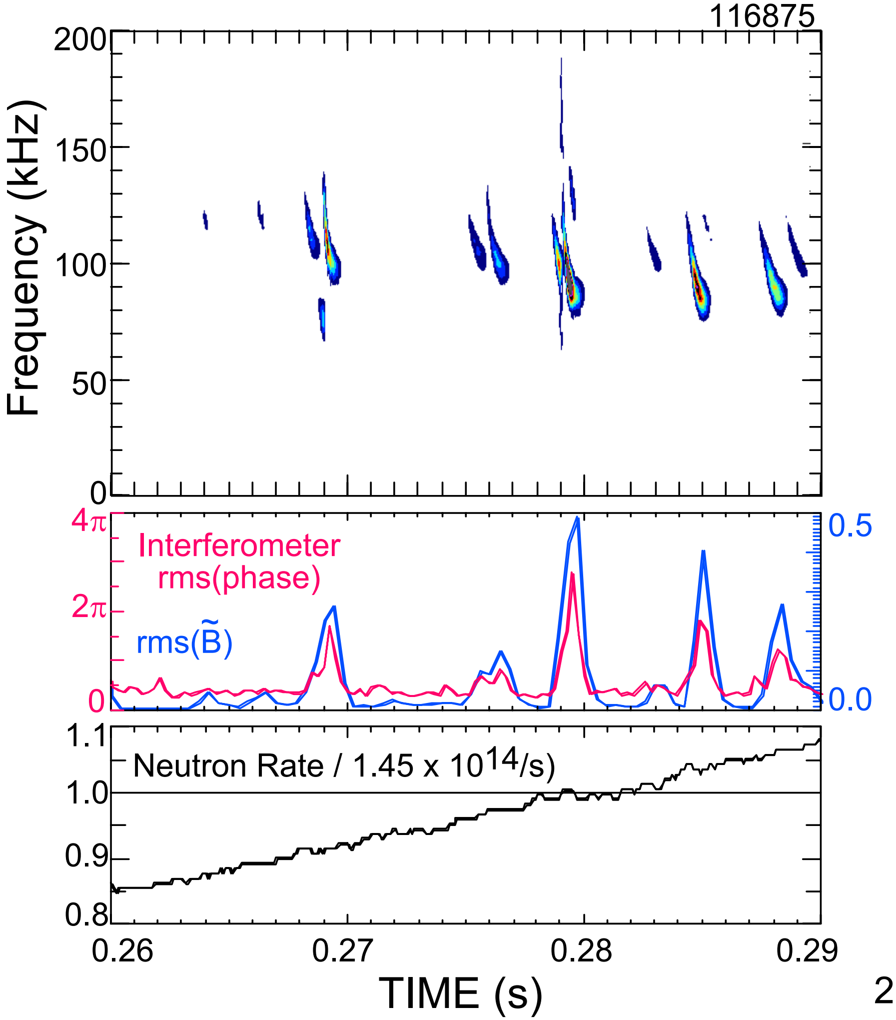
- ITER in new, small ρ^* regime for fast ion transport
  - $k_\perp \rho \approx 1$, "short" wavelength Alfvén modes
  - fast ion transport from interaction of many modes
- NSTX also routinely operates with super-Alfvénic fast ions;
  - Although ρ^* is large, can study multi-mode transport
  - Only machine capable of measuring q profile at large v_{fast} / v_{Alfven}



Multi-mode bursts can induce significant fast-ion losses, while weaker losses occur with single-mode bursts

- Opportunity to examine multi-mode fast ion transport

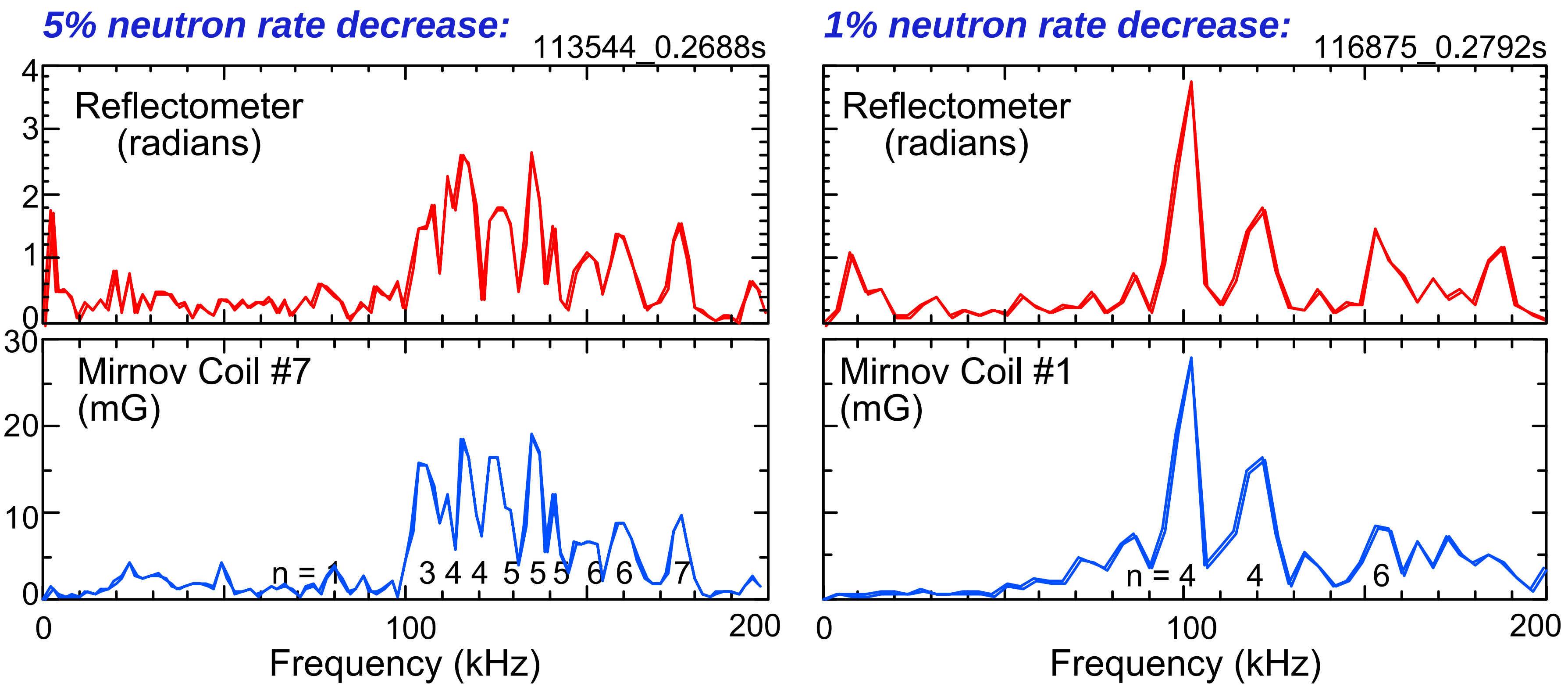




Reflectometer and Mirnov data indicates “sea of TAE” with n=1-7 vs. a few scattered modes…

- Neutron rate decrease correlates with number of modes, not just amplitude

(Mode spectra below are from TAE burst events of previous viewgraph)



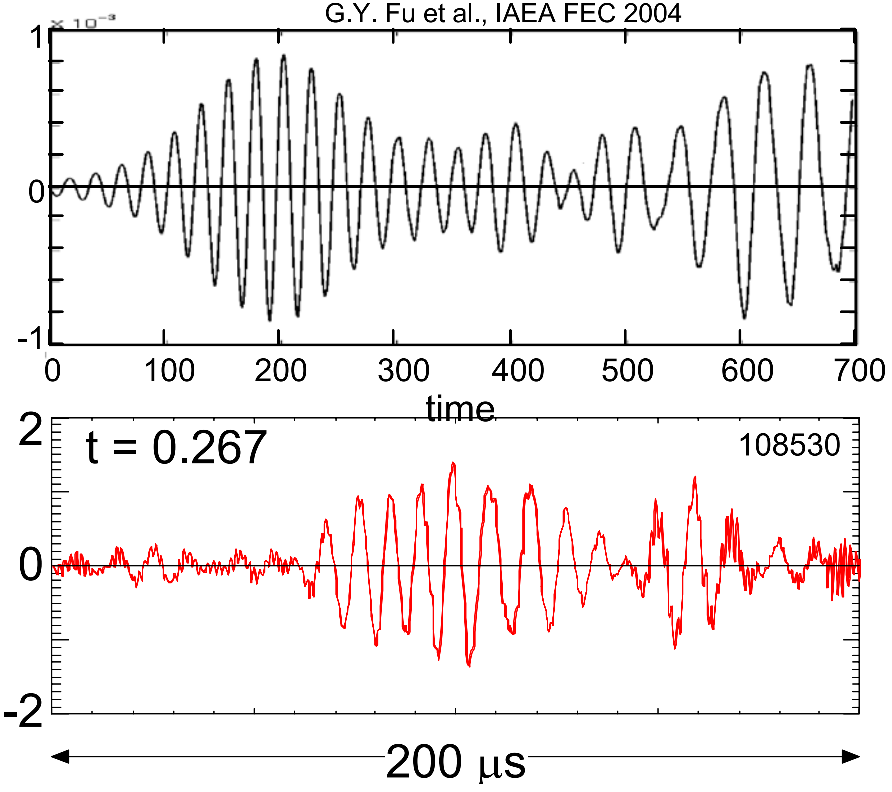
5% neutron rate decrease:

1% neutron rate decrease:

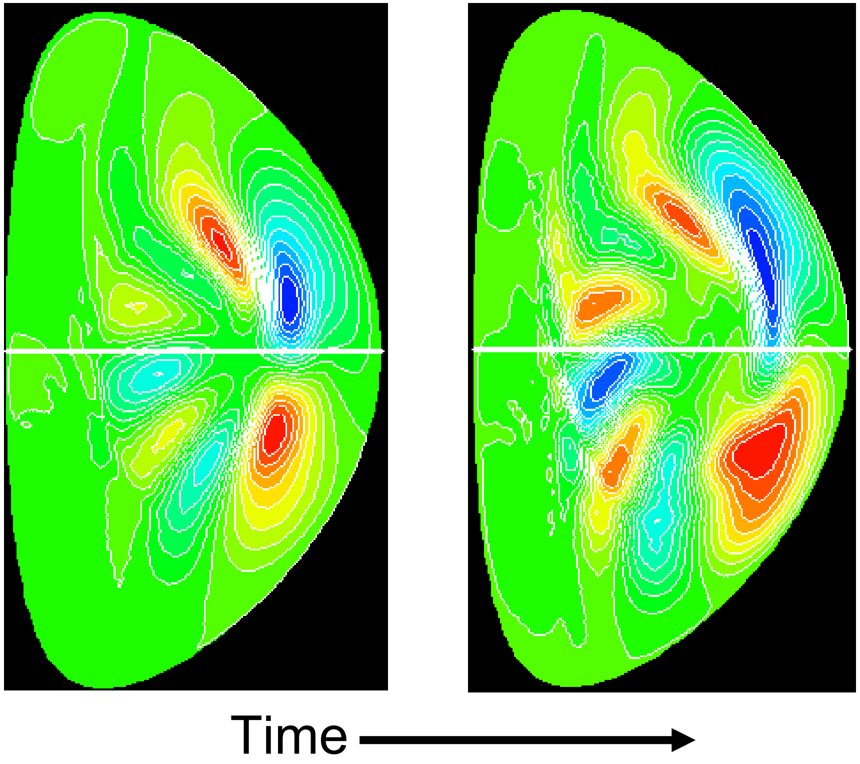
- High n's imply large $k_\perp\rho_{fastest}$ = 2 - 5, but for ions with $V_{beam} \approx V_{Alfvén}$, $k_\perp\rho_{beam}$ = 0.5 - 1.5 ($\rho^* \approx 0.06$)

Non-linear simulations of single-n TAEs are promising – now in early stages of simulating multiple simultaneous modes

M3D nonlinear hybrid simulation of beam-driven modes in NSTX shows a bursting n=2 TAE as the mode moves out radially:



G.Y. Fu et al., IAEA FEC 2004



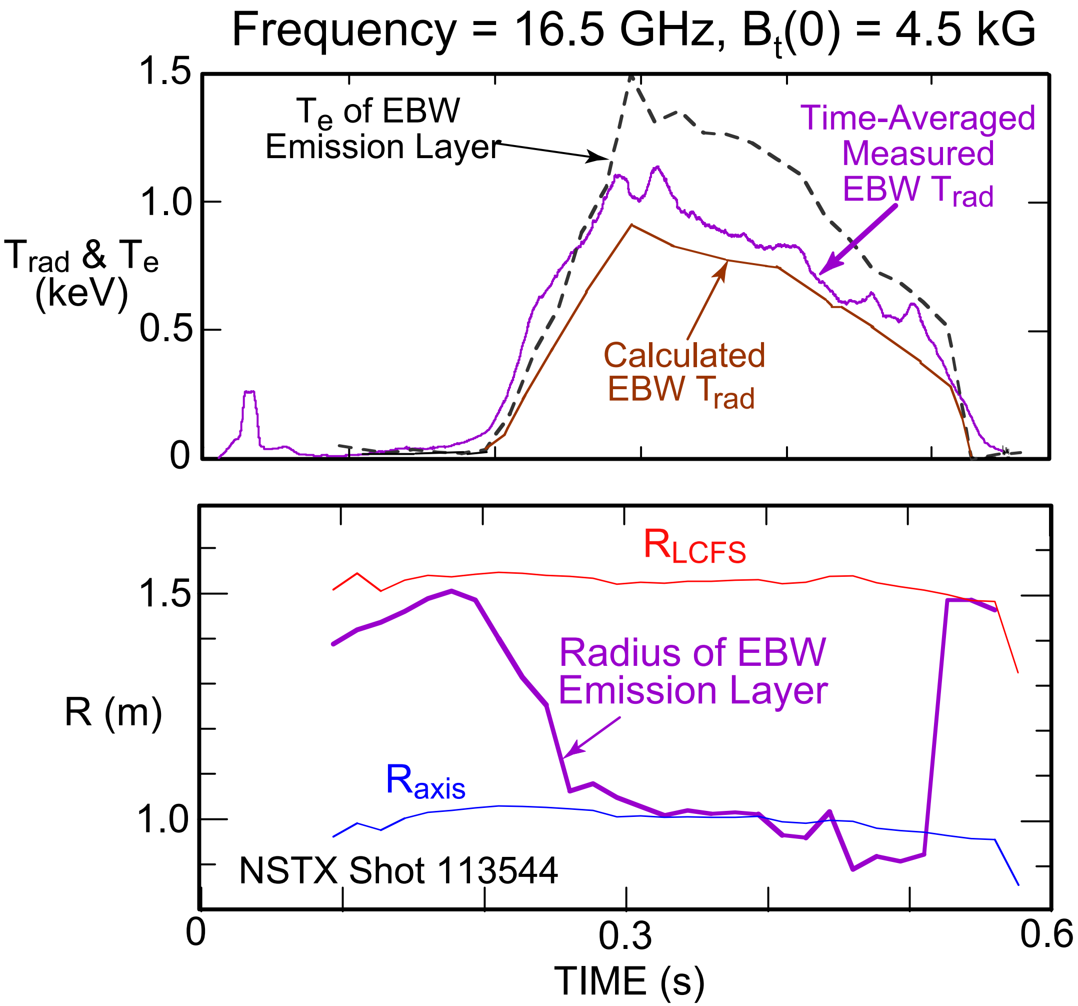
Fast-ion redistribution from TAEs high-priority in ITPA MHD group

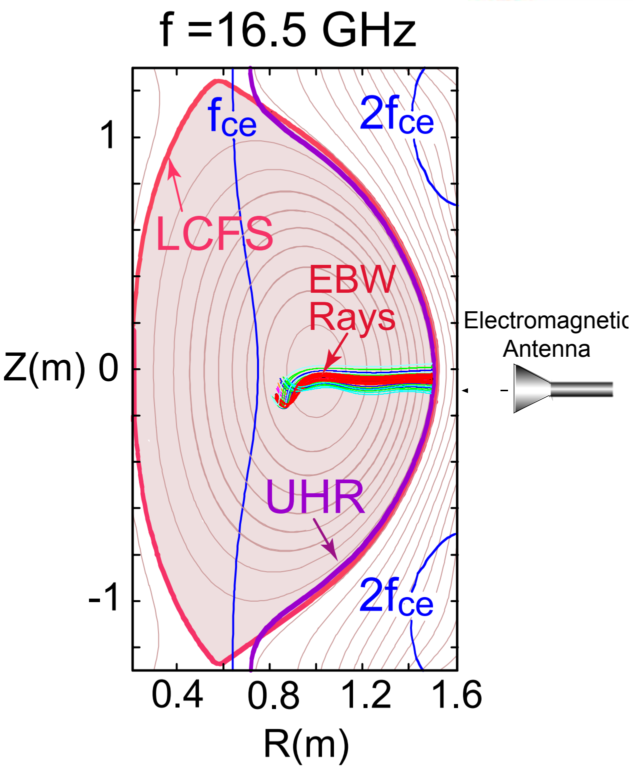
See presentation by E. Fredrickson for more details

NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

2004: Measured 80% B-X-O coupling of intrinsic thermal EBW - consistent with modeling

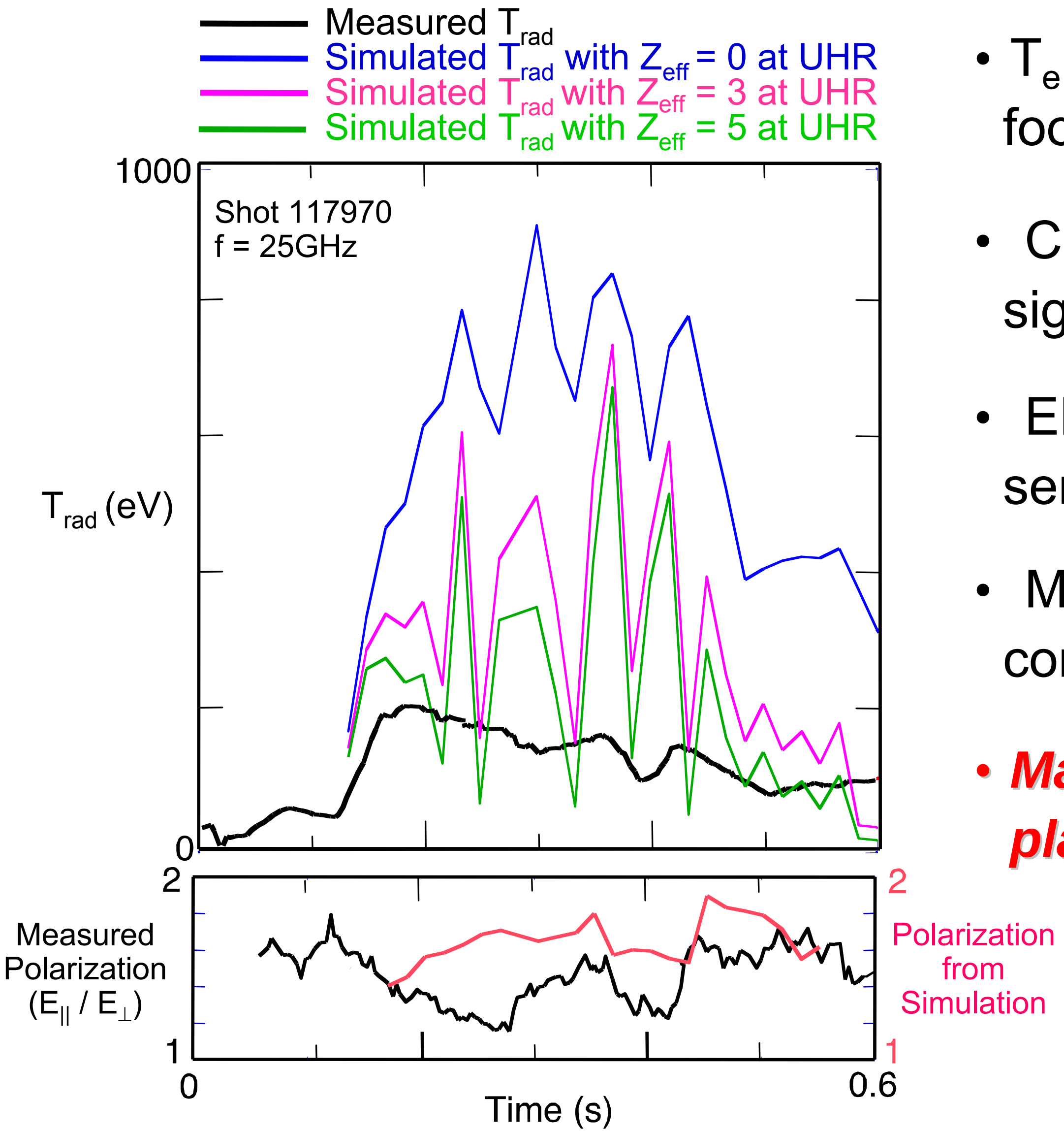




- 3-D ray tracing & full wave EBW mode conversion model using EFIT magnetic equilibrium & Thomson scattering T_e & n_e

Taylor et al., PoP 2005
J. Preinhaelter et al., AIP Proc. ***787****, 349 (2005)*

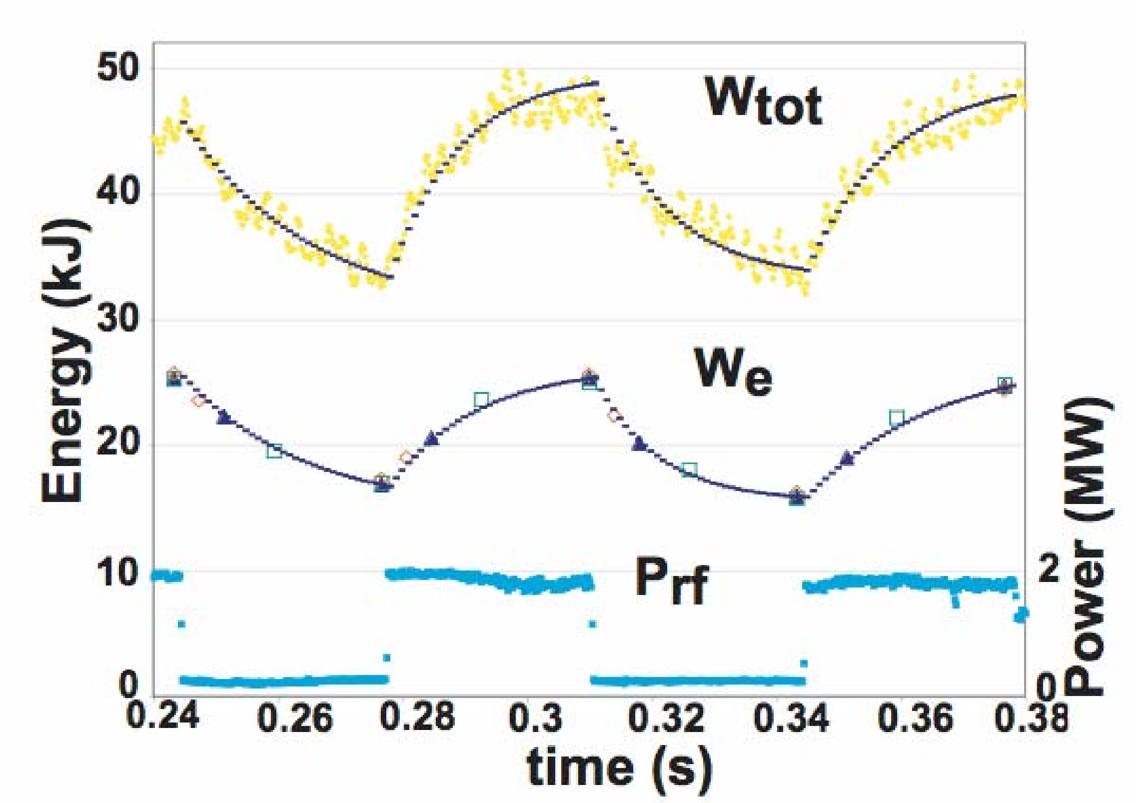
2005: Much lower B-X-O coupling ≈ 20% in some H-Modes may be due to EBW damping at Upper Hybrid Resonance (UHR)



- T_e ~ 10 - 30 eV near UHR, near foot of H-mode pedestal
- Collisional losses can be significant for $T_e < 30$ eV
- EBW conversion efficiency sensitive to Z_{eff} at low T_e
- Measured emission polarization consistent with simulation
- ***May need improved control of plasma parameters near UHR***

HHFW power modulation experiments measured reduced power absorption fraction with current-drive phasings

k_T (m^{-1})	% Power absorbed
14	80
+7	70
-7	55
+3	< 20

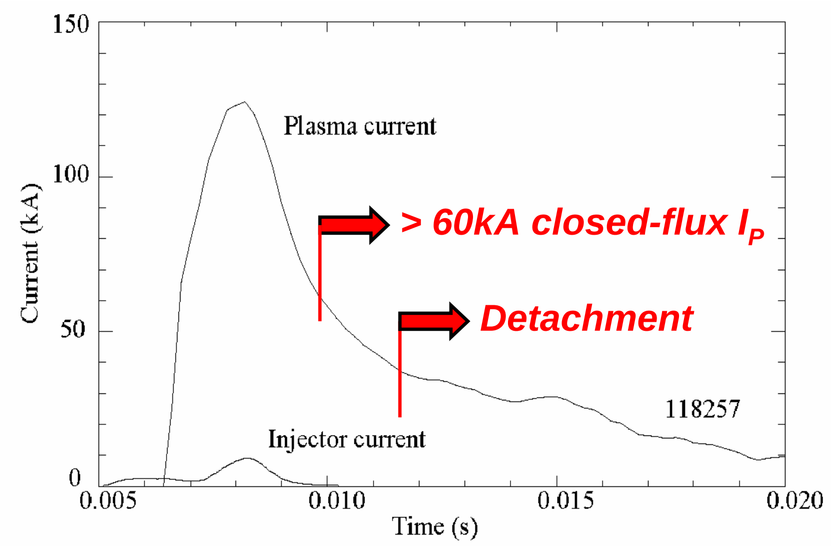


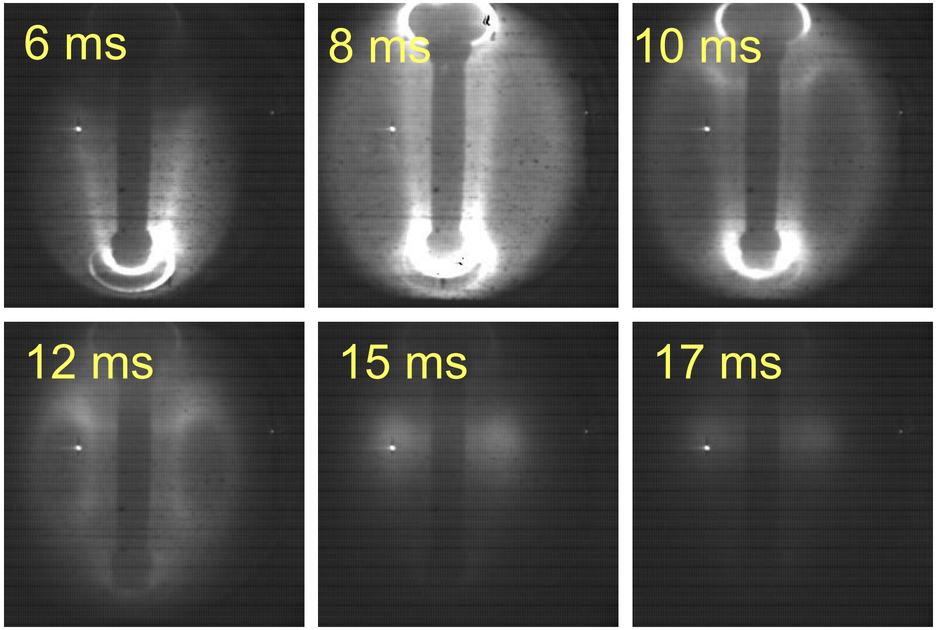
- Parametric decay into surface waves may explain some of the power absorption dependence on k_T
- Field pitch angle may explain differences between co & counter

NSTX Research Highlights from 2005 Run

- Long-pulse operation - enhanced plasma shaping & control
- Boundary physics
- Macroscopic stability
- Electron thermal transport
- Fast-ion-instabilities and confinement
- Wave heating and current drive
- Plasma formation with Coaxial Helicity Injection

For 1st time, closed-flux current has been generated in large ST using Transient CHI *(Previously studied on smaller HIT-II – UW)*





- Plasma current amplified many times relative to injected current
- > 60kA of closed flux current generated using transient CHI
- Camera images from 12 to 17ms show clear detachment of plasma from injector region
- **GOAL for 2006 and beyond:**
  - Increase I_P to 0.5MA without using central transformer action
- **METHOD:**
  - Make sufficient CHI plasma current to allow auxiliary heating and current drive tools to heat & further ramp I_P
  - **The tools needed for this are only available on NSTX**

See CHI presentation by D. Mueller

Summary of 2005 NSTX Research Highlights

- Achieved record NSTX pulse-length discharges in a favorable ELM regime obtained with strong shaping and enhanced shape control
- Demonstrated particle control with Lithium coating
- Dramatically improved physics understanding of error fields, resistive wall modes, plasma rotation damping, and disruptions
- Correlated improved electron confinement with measured reversed q-shear
- Correlated significant fast particle loss with multi-mode “sea-of-TAE” bursts
- Improved understanding of EBW and HHFW coupling efficiency
- Demonstrated 60kA closed-flux plasma formation in NSTX using CHI

NSTX makes important contributions to plasma science, ITER, and next-step STs

- Access to β of order unity → new physics in transport and MHD
- Understand role of plasma geometry on stability (edge and core)
- Only major US facility investigating Li for pumping and power handling
- Understand error fields and resistive wall modes - complements AT
- Understanding electron transport highly relevant to burning plasmas
- Observe mode-induced fast ion loss - important to burning plasmas
- Non-inductive plasma formation (CHI) and current drive (EBW, HHFW) essential for ST, useful for AT
- Developing knowledge for extrapolating ST to CTF and reactor