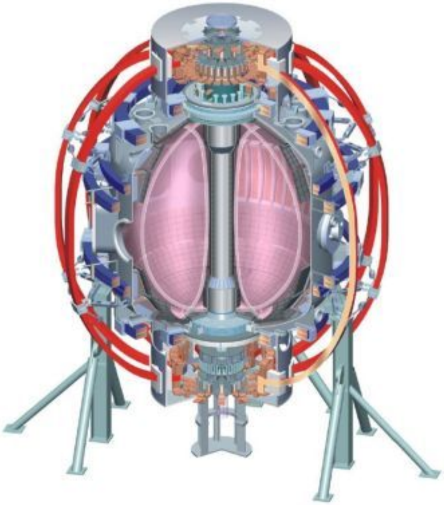
Overview of the NSTX Research Program for 2009-2013

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

J.E. Menard, PPPL

NSTX 5 Year Plan Review for 2009-13
Conference Room LSB-B318, PPPL
July 28-30, 2008



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
JAEA
Hebrew U
Ioffe Inst
RRC Kurchatov Inst
TRINITI
KBSI
KAIST
POSTECH
ASIPP
ENEA, Frascati
CEA, Cadarache
IPP, Jülich
IPP, Garching
ASCR, Czech Rep
U Quebec

Outline:

- NSTX Mission
- Unique Parameter Regimes Accessed by NSTX
  - Macroscopic Stability
  - Transport and Turbulence
  - Waves and Energetic Particles
  - Boundary Physics
  - Plasma Formation and Sustainment
- Next-step ST Missions
- Gaps Between Present and Next-step STs
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NSTX Mission Elements for 2009-2013

(Prioritized)

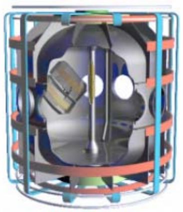
1. Establish attractive ST operating scenarios & configurations
   - **Long-term goal:** Understand and utilize advantages of the ST configuration for addressing key gaps between ITER performance and the expected performance of DEMO (including an ST-DEMO)

2. Complement tokamak physics and support ITER
   - Exploit unique ST features to improve tokamak understanding
   - Contribute to ITER final design activities and research preparation
   - Participate strongly in ITPA and U.S. BPO, benefit from tokamak R&D

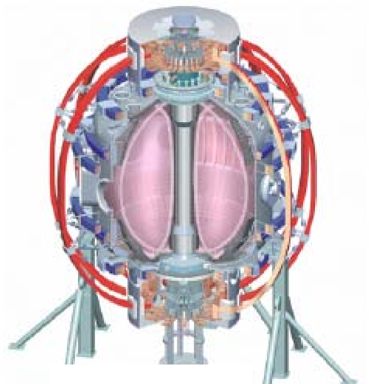
3. Understand unique physics properties of the ST
   - Understand impact of low A, very high β, high v_{fast} / v_A, …
   - **ST understanding underpins missions 1 and 2 above**

Present and future spherical tori complement ITER and accelerate the development paths of all DEMO concepts

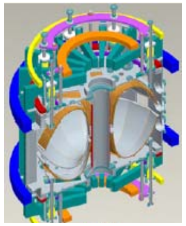
STs



PEGASUS



NSTX

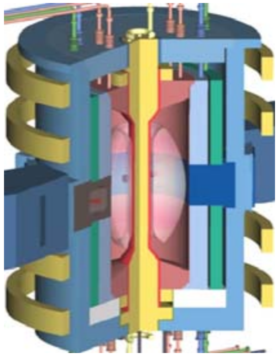


LTX

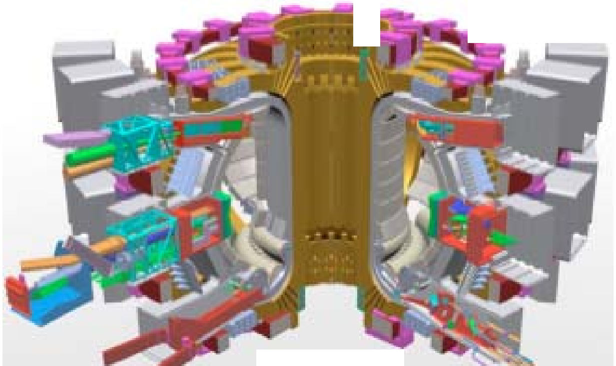
NHTX

Plasma-Material Interface R&D + Advanced Physics

Nuclear Component Testing



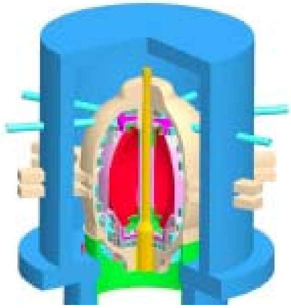
ST-CTF



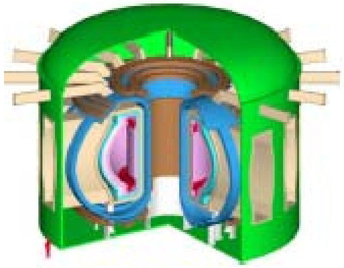
ITER

Burning Plasma Physics

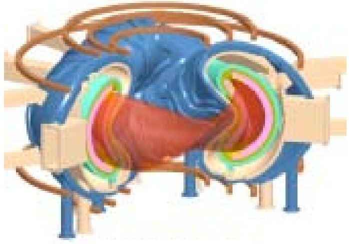
DEMO



ARIES-ST



ARIES-AT



ARIES-CS

The ST can contribute to all FESAC Priority Panel “Themes”

ST expands knowledge-base for all aspects of Theme A

A. Creating predictable high-performance steady-state plasmas
  - Measurement
  - Integration of high-performance, steady-state, burning plasmas
  - Validated predictive modeling
  - Control
  - Off-normal plasma events
  - Plasma modification by auxiliary systems
  - Magnets

ST offers simplified, maintainable, affordable magnets for DEMO

B. Taming the plasma material Interface (PMI)
  - Plasma wall interactions
  - Plasma facing components
  - RF antennas, launching structures, and other internal structures

ST offers high heat flux at small size and cost for PMI R&D

C. Harnessing fusion power
  - Fusion fuel cycle
  - Power extraction
  - Materials science in the fusion environment
  - Safety

ST offers high neutron flux at small size and cost for testing fusion nuclear components

NSTX will make world-leading contributions to ST development and contribute strongly to ITER and fundamental toroidal science

- NSTX Mission
- Unique Parameter Regimes Accessed by NSTX
  - Macroscopic Stability
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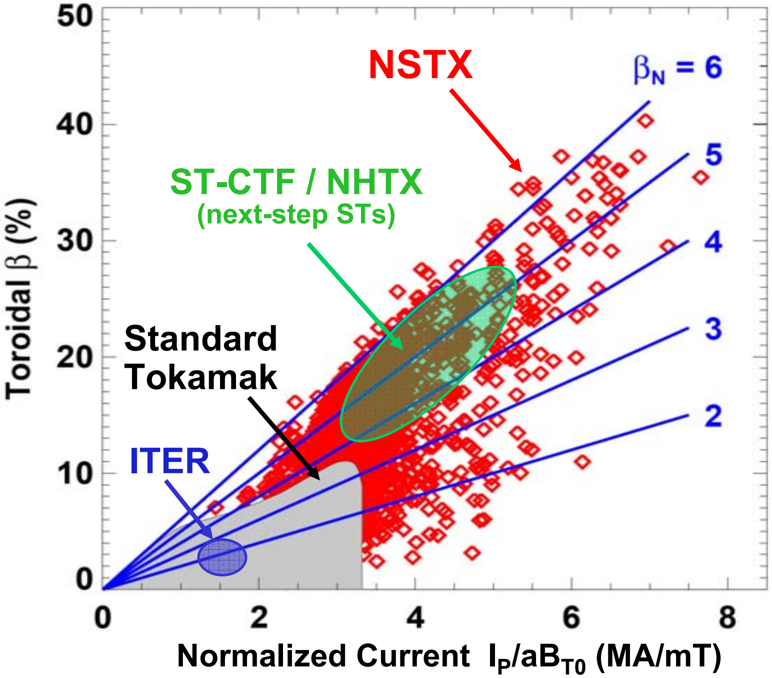
NSTX creates stable, well diagnosed plasmas at high β enabling a wide range of toroidal physics studies

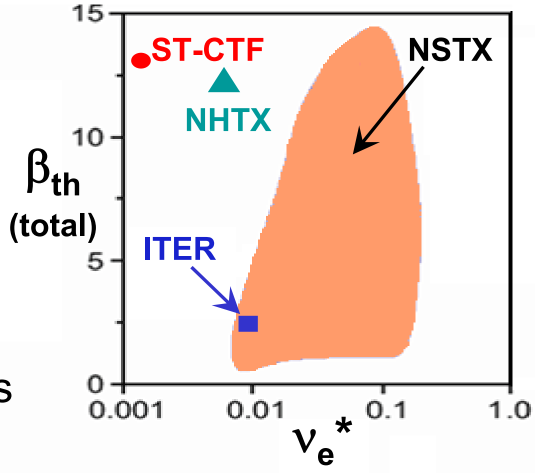


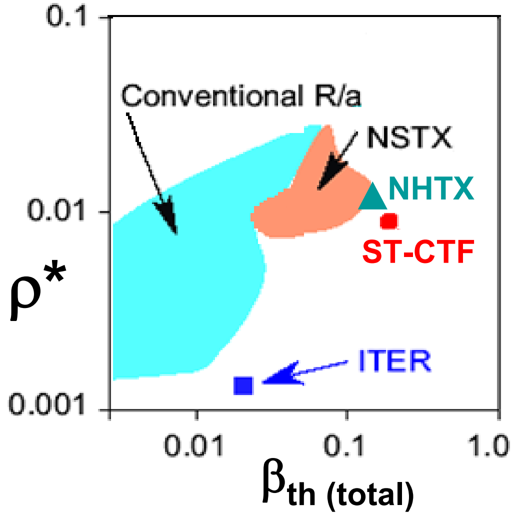
- ST accesses higher normalized current & higher normalized β

➡ higher $\beta_{Toroidal}$

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

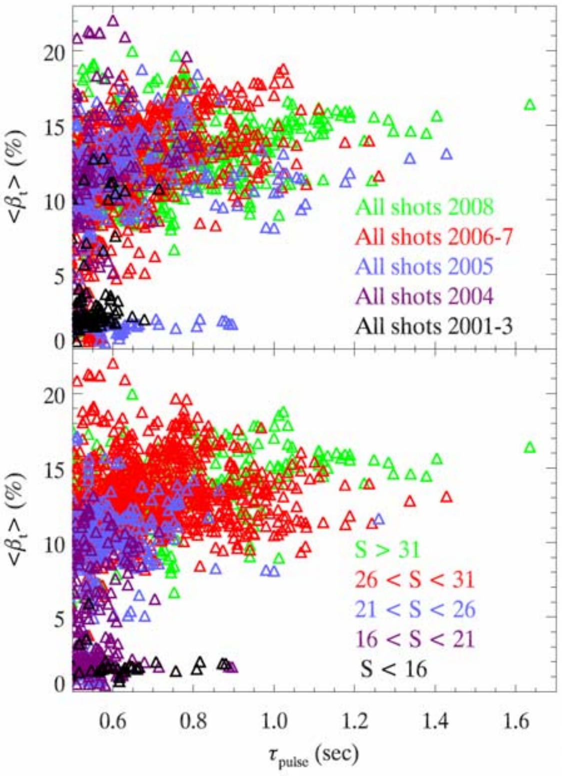
- Access ITER-level ν^*, extending confinement understanding to high β
- Next-step STs expected to operate at significantly lower ν^* than present STs
- ST operates at higher ρ^* than tokamaks / ITER - impacts thermal and fast-ion transport, MHD
- Extrapolation in ρ^* from present STs to next-step STs is small



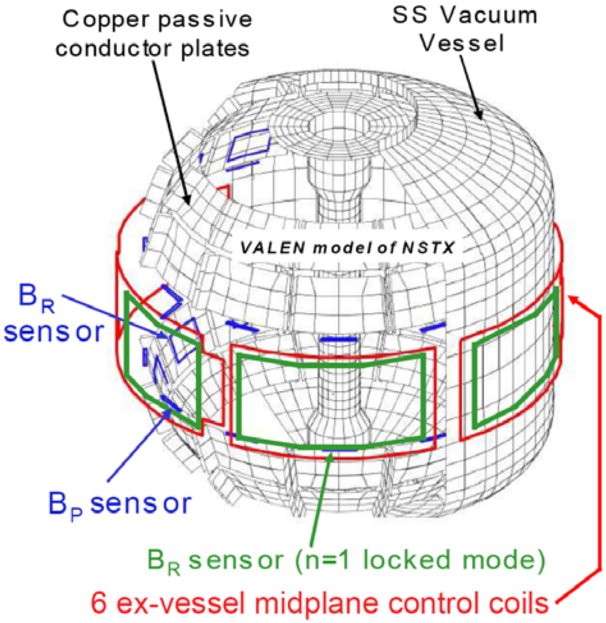


Improved control of plasma instabilities has significantly increased the duration of sustained high β in NSTX

Increased plasma shaping from improved n=0 control for high κ and δ operation **+ $n \geq 1$ EF/RWM control =** **Duration of $\beta_T > 15\%$ increased factor of 4 from 2002 to 2008**



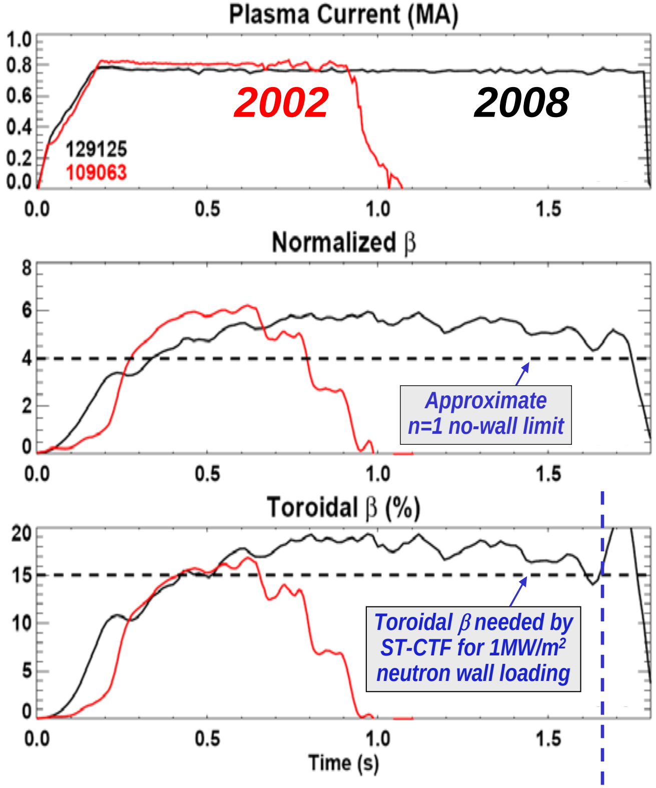
$S \equiv q_{95} I_P/aB_T$ [MA/mT]



Control coils also used to study:

- Locked mode thresholds
- Resonant field amplification
- Rotation damping from NTV
- Anomalous momentum transport
- Pedestal transport and stability

NSTX has sustained β_T needed for ST-CTF for 4 current redistribution times



TF ramp-down due to coil heating limit

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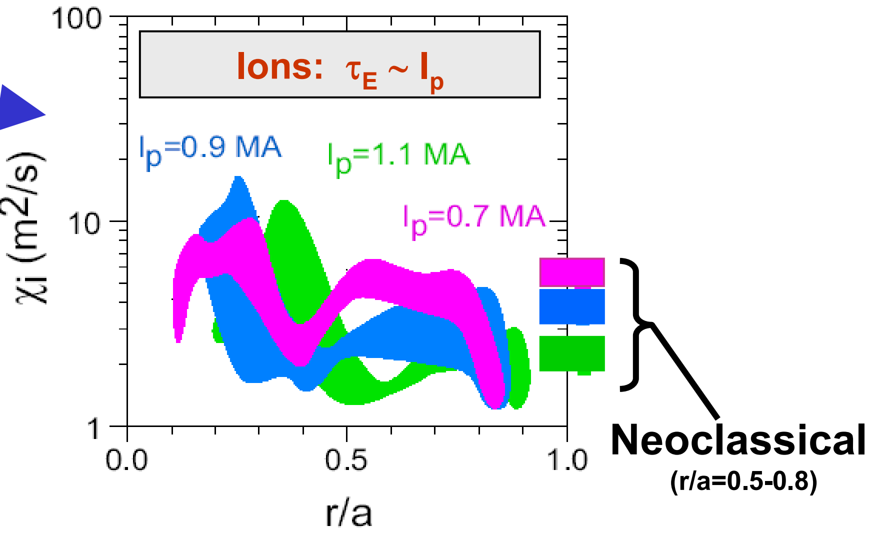
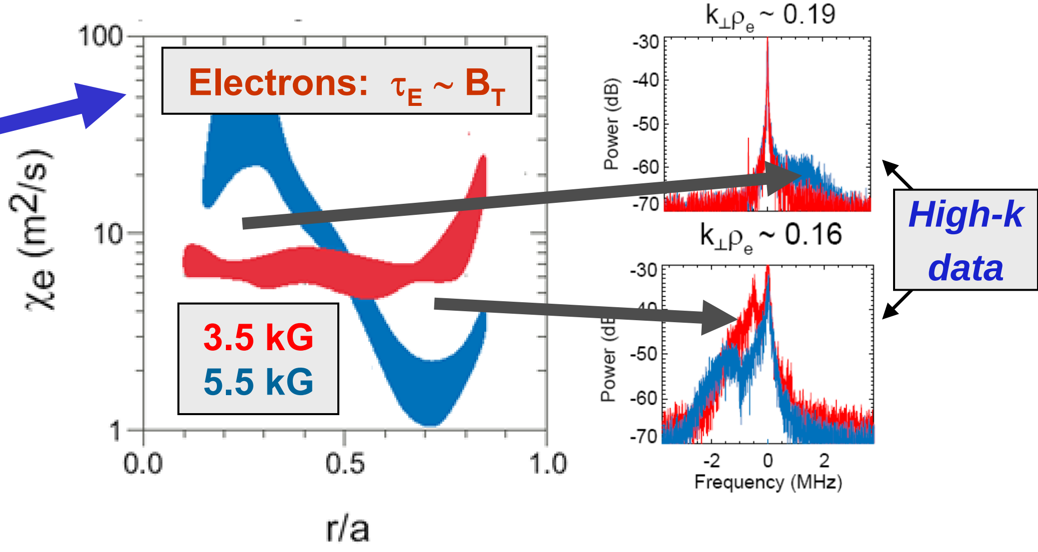
NSTX is developing a deeper understanding of ion and electron energy transport for STs and tokamaks

- Ion $\tau_E \sim I_P$, consistent with neoclassical ion transport
  - Implies ion turb. suppressed by high E×B shear → possibility of isolating causes of e-transport

- Electron & ion τ_E scale differently, and different than at higher A:
  - Ion $\tau_E \sim I_P$, electron $\tau_E \sim B_T$

- High-k scattering data indicates χ_e correlated w/ high-k density fluctuations
  - Correlation holds both spatially and versus B_T
  - Consistent with ETG at large r/a (i.e. in T_e gradient region)

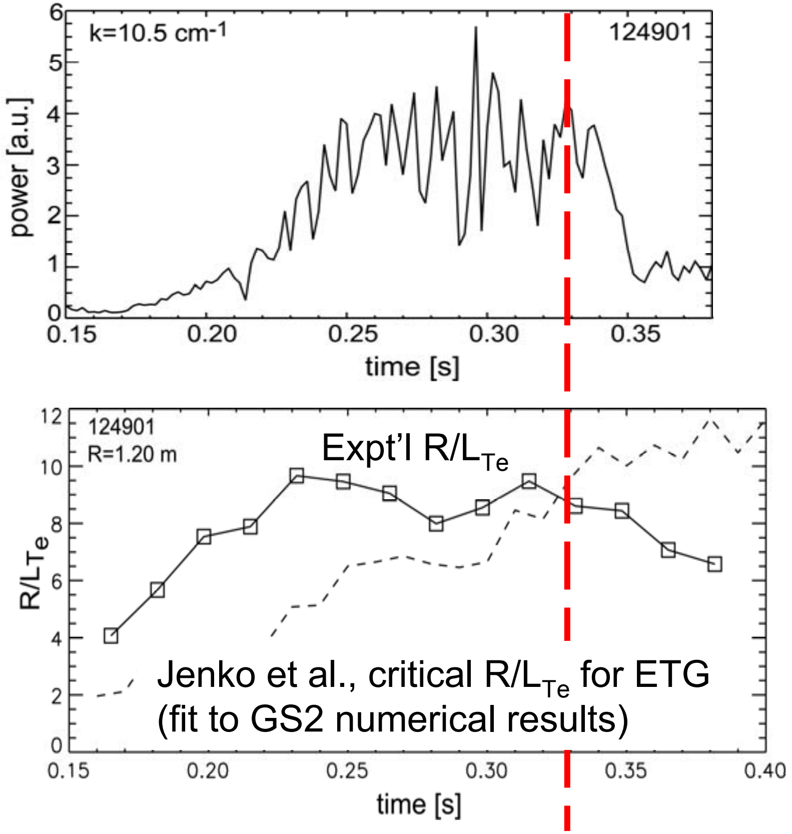


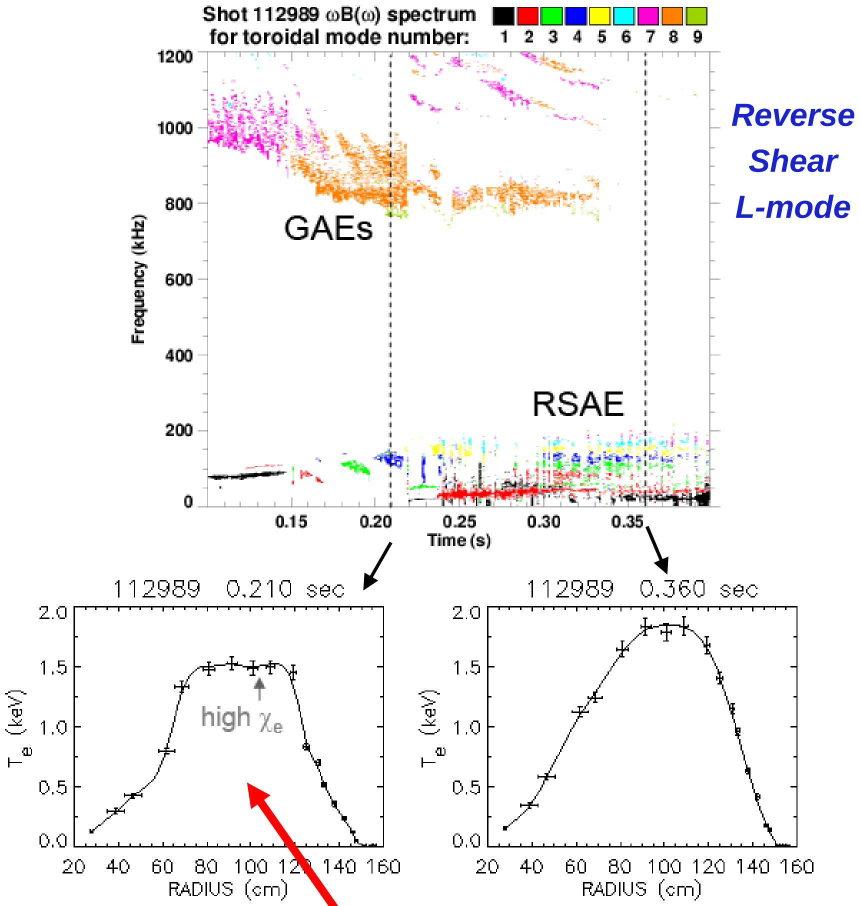


Unique diagnostics and plasma regimes of NSTX indicate multiple modes may influence e-turbulence and transport

High-k scattering diagnostic (Δr=±3 cm)

k-range of fluctuations in ETG/high-k TEM range





- **High fluctuation level when R/L_{Te} is greater than critical value for ETG**

- **Core T_e flattening correlated with Global Alfvén Eigenmode (GAE) activity**

Low-k micro-tearing also important - see Transport and Turbulence presentation

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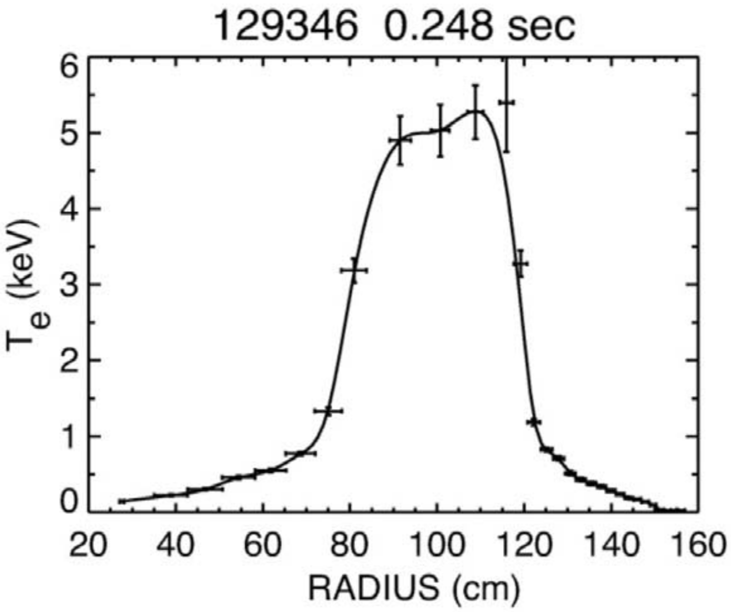
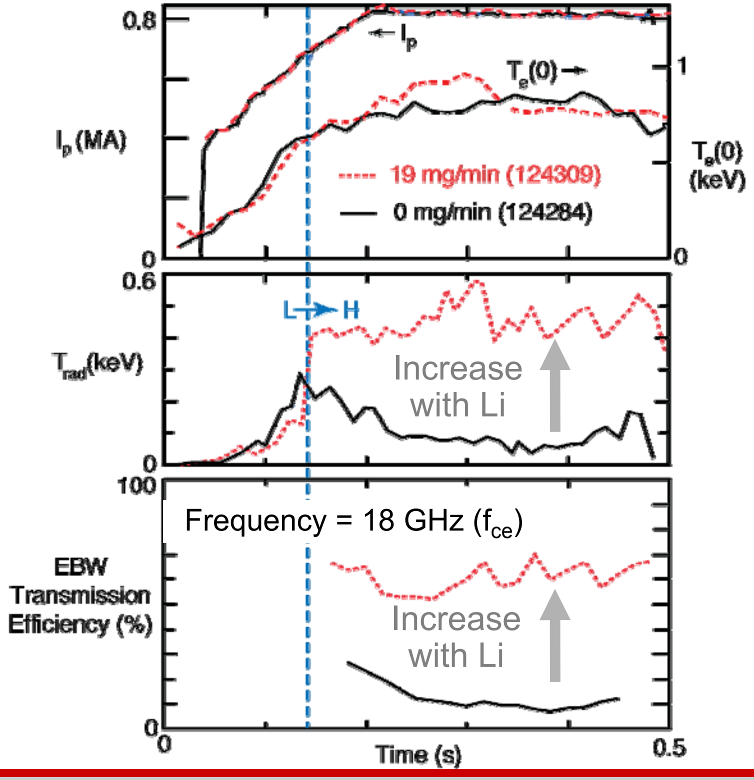
NSTX has improved the understanding and performance of wave heating & CD techniques in over-dense plasmas

- High-harmonic fast-wave (HHFW)
  - Discovered that surface waves reduce heating efficiency if density near antenna is too high
  - Control of edge density improves heating → record T_e = 5keV in NSTX achieved with HHFW
- Electron Bernstein Wave (EBW)
  - Discovered that collisional damping at mode conversion layer reduces coupling
  - Higher T_e at MC layer via Li-conditioning increases EBW transmission efficiency from 10% to 50-60% in H-mode→ Improved prospects for EBW as H&CD tool



$n_e(0) = 1.5 \times 10^{19}$ m^{-3}



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

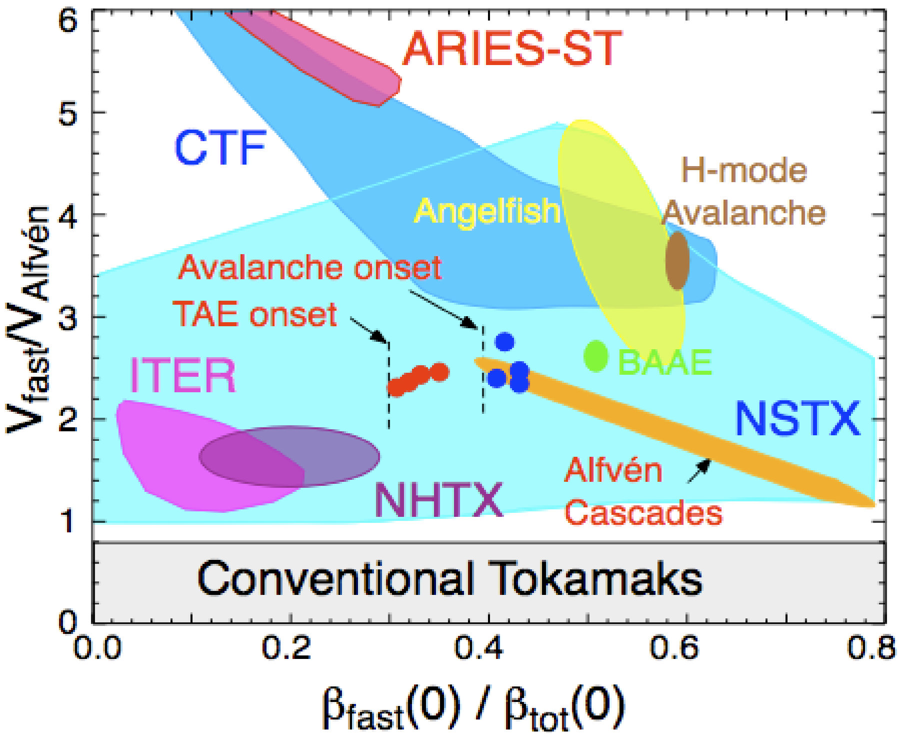
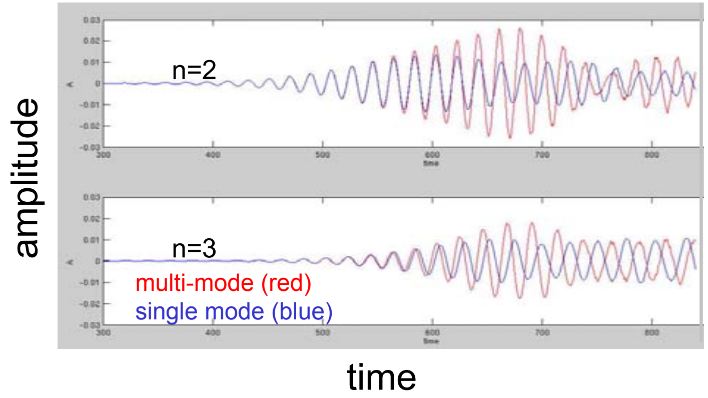
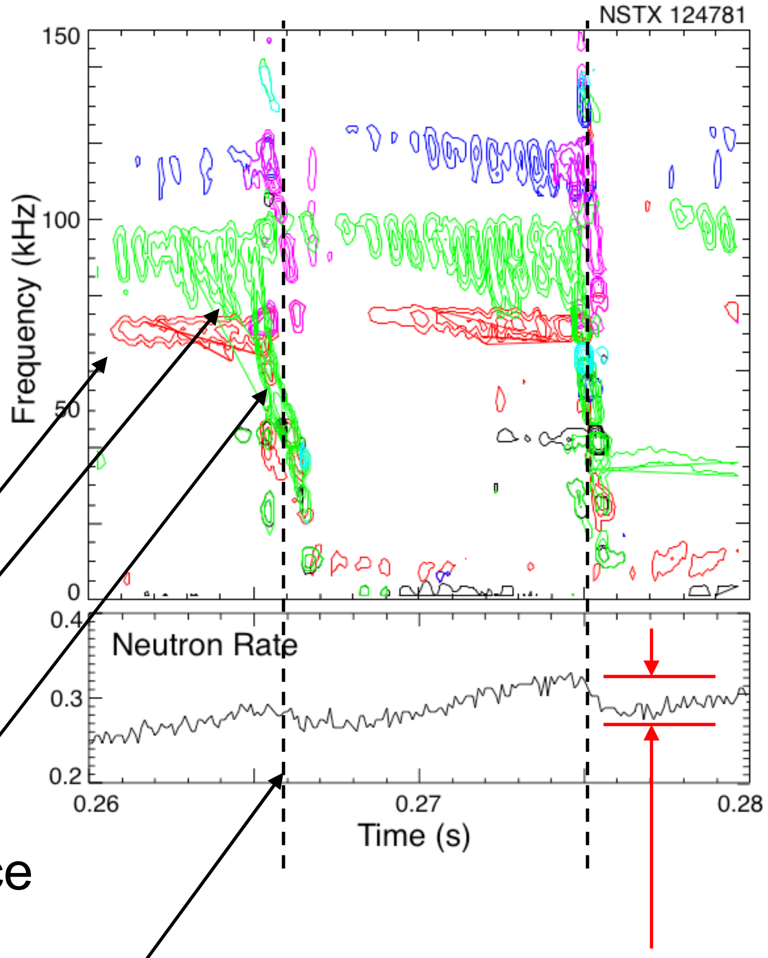


Figure above is simplified picture - there are other dependences, such as *q* profile, ρ^*

NSTX finds AE avalanches can induce fast-ion redistribution and/or loss - potentially important for ITER and ST-CTF

M3D-K simulations: overlapping resonances + multiple modes cause larger mode amplitude → larger fast-ion f(v) perturbation





15% drop in neutron rate

Experiment:

- As power is raised, first see AE
  - then chirping AE
  - then avalanches, multi-mode transport
- Avalanches are strong bursts of multiple AE modes (2 ≤ n ≤ 6) overlapping in space and frequency
- Avalanches correlate with neutron drops indicating fast ion redistribution and/or loss

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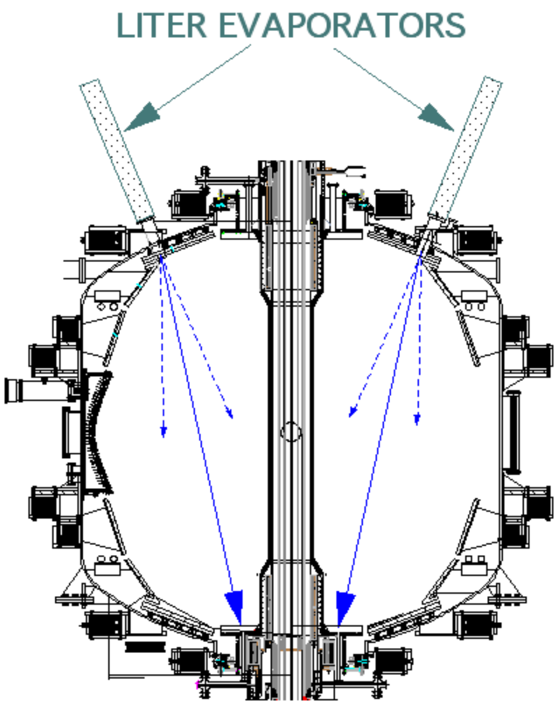
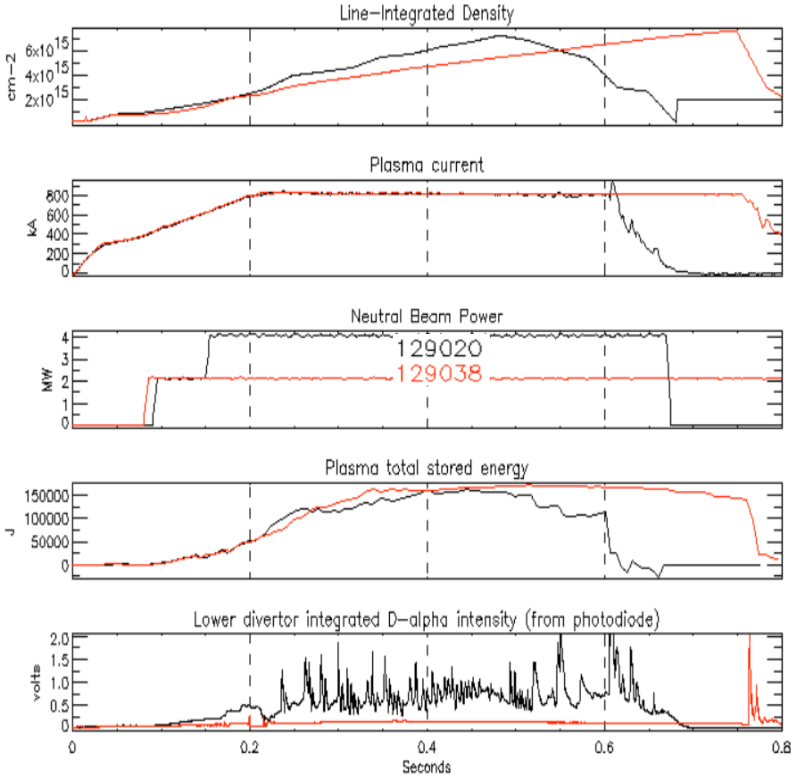
NSTX is unique in the world program in exploring lithium in a diverted H-mode plasma

- Dual Lithium evaporators (LITERs) provide complete toroidal coverage of lower divertor
  - Improved performance vs. 1 LITER
  - 2008: High-performance operation with **NO** between-shot He glow → increased shot-rate



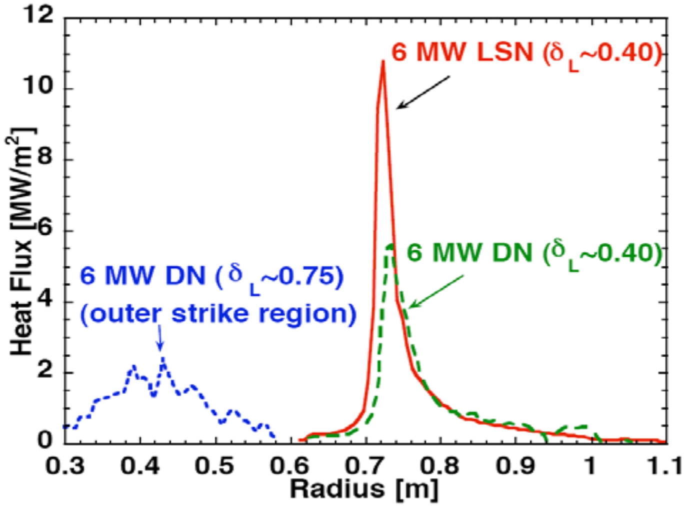
- Reproducible ELM elimination from Li
  - Plasma density reduced
  - Pulse-length extended
  - At 800kA, power must be reduced to avoid β limit
  - Confinement time doubled (up to 80ms)
  - Large reduction in divertor D_{α} → reduced recycling

No Li (black) **With Li (red)**

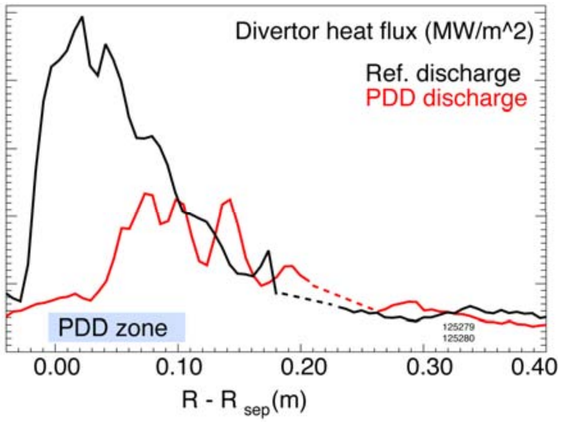


NSTX accesses ITER-level divertor heat fluxes and is exploring mitigation of steady-state and transient heat fluxes

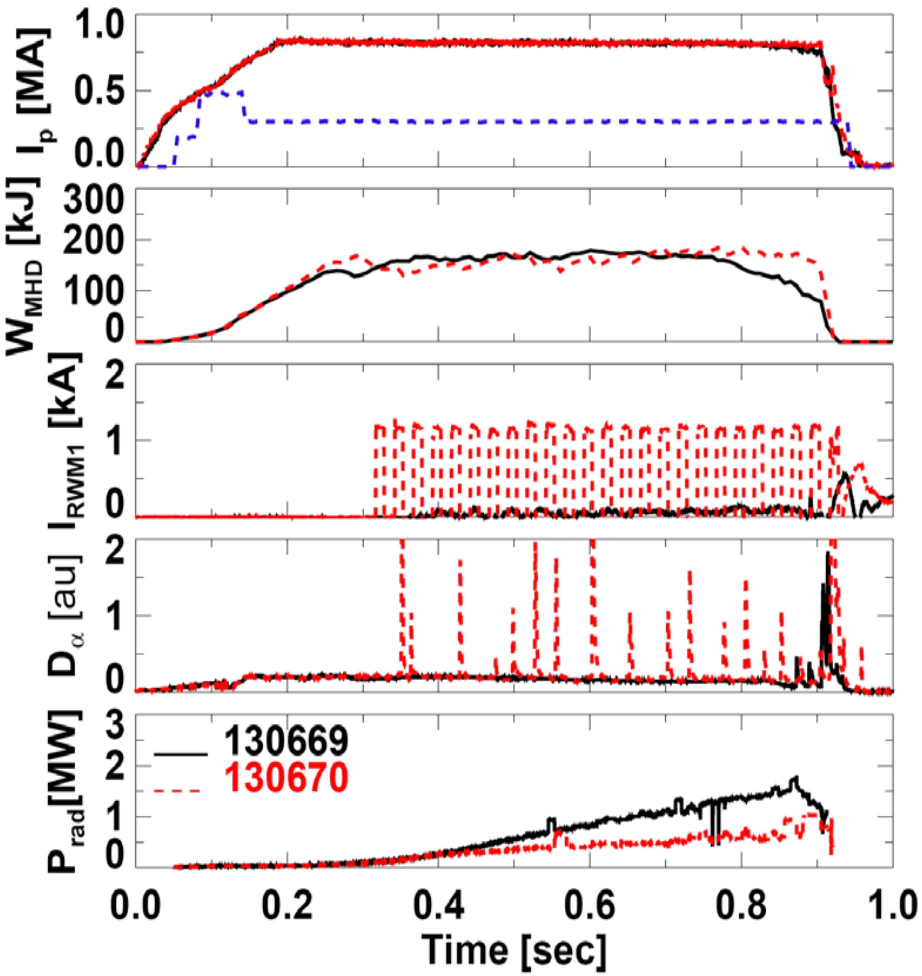
- **Magnetic geometry strongly influences peak heat flux**



- **Partial detachment reduces peak heat flux**



- **Lithium conditioning can eliminate ELMs**
- **RMPs can controllablly trigger ELMs and expel impurities from Li-ELM-free plasmas**



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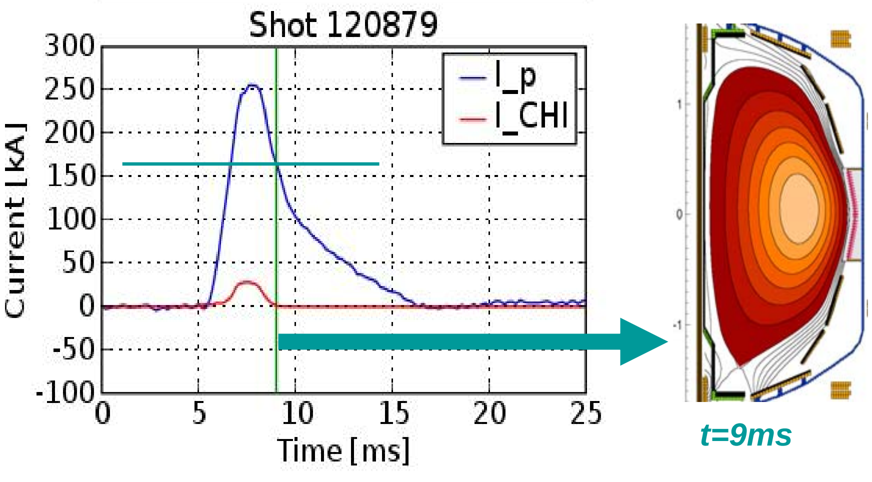
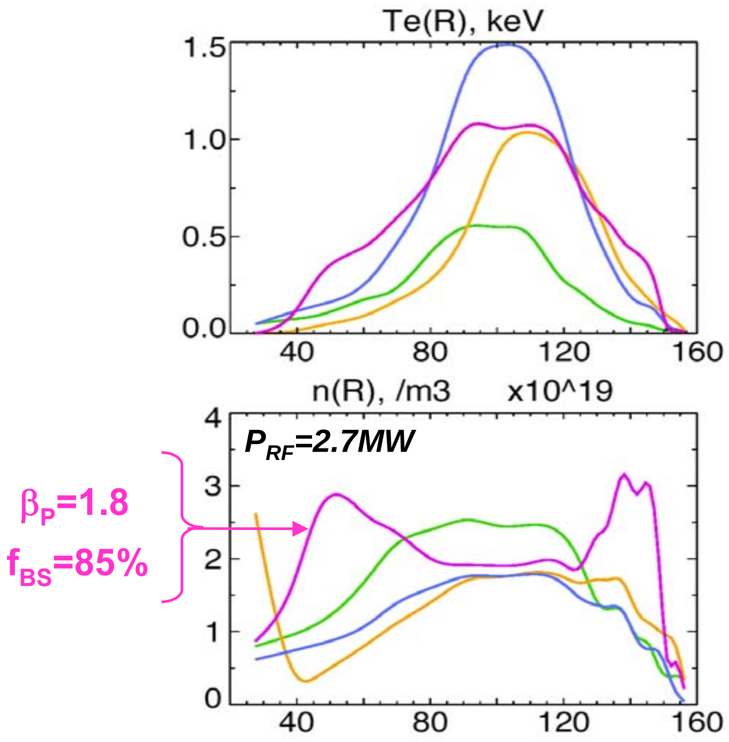
NSTX is testing unique methods of non-solenoidal plasma current start-up and ramp-up for STs

- Start-up: Coaxial Helicity Injection
  - Generated record closed-flux I_P=160kA
  - Demonstrated coupling to induction and compatibility with high performance H-mode
  - Higher I_P limited by lack of auxiliary heating, possibly impurities/divertor conditions

- Ramp-up: High Harmonic Fast Wave
  - HHFW heats 250kA plasma to T_e=1keV
  - Produces f_{BS}=85% H-mode plasma
  - Limited by antenna voltage stand-off, ELMs

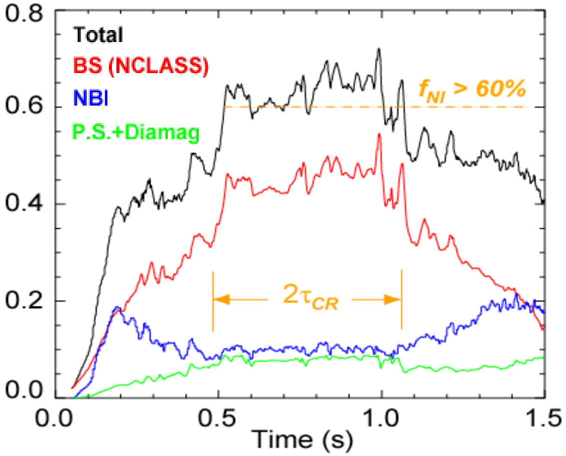




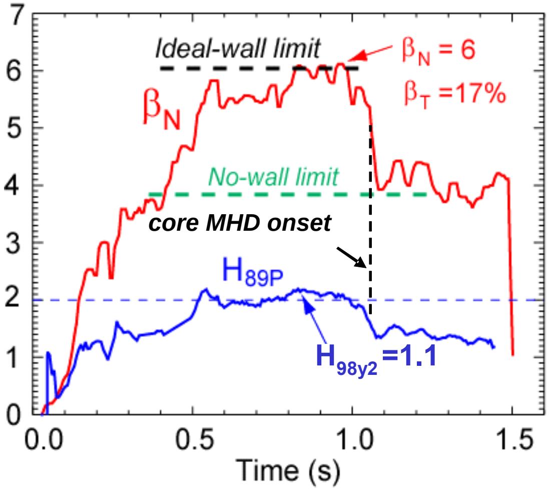
NSTX has developed and sustained scenarios with high non-inductive fraction and high normalized β

- f_{NICD} = 65%
- $f_{\nabla p}$ = 55%

TRANSP non-inductive current fractions for NSTX shot 116313G12

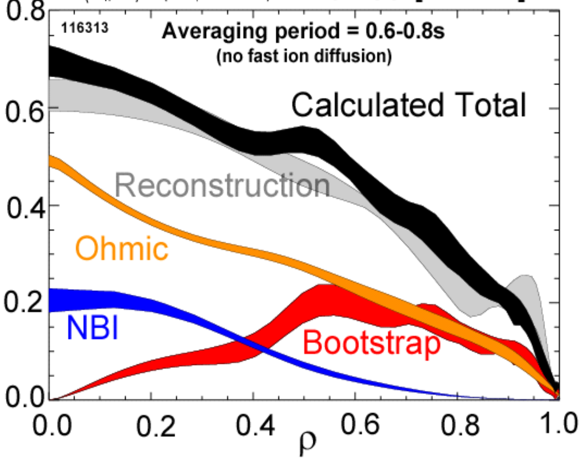


- β_N = 5.5-6
- H_{98} = 1-1.1
- $f_{Greenwald}$ → 1



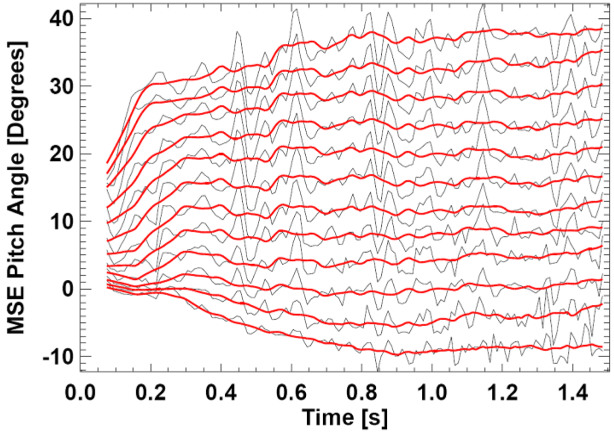
Predicted and reconstructed J profiles are in agreement when MHD activity is weak

$\langle J_{\parallel} B \rangle / \langle B_{\phi} R_0 / R \rangle$ Profiles [MA/m²]



- Recent long-pulse discharges which avoid core rotating MHD activity exhibit J-profile equilibration
  - Spikes in MSE pitch angle are low-f MHD (early) and large ELMs (late)

NSTX Shot 129125



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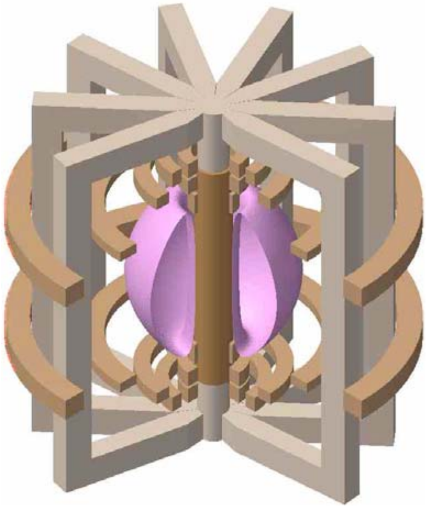
ST is attractive configuration for "Taming the plasma-material interface"

- FESAC-PP identified PMI issue as highest priority: "…solutions needed for DEMO not in hand, …require major extrapolation and substantial development"

Scientific mission of **N**ational **H**igh-power advanced **T**orus e**X**periment (NHTX):
"Integration of a fusion-relevant plasma-material interface with stable sustained high-performance plasma operation"

- **PMI research and integration goals:**
  - Create/study DEMO-relevant heat-fluxes
  - Perform rapid testing of new PMI concepts
    - Liquid metals, X-divertor, Super-X divertor
  - PMI research at DEMO-relevant T_{wall} ~ 600°C
  - Plasma-wall equilibration: τ_{pulse} = 200-1000s
  - Develop methods to avoid T retention
  - Demonstrate compatibility of PMI solutions with high plasma performance:
    - High confinement without ELMs
    - High beta without disruptions
    - Steady-state, fully non-inductive
  - Study high β_N, f_{BS} for ST-DEMO and ST-CTF
  - Test start-up/ramp-up for ST-CTF and ST-DEMO



National **H**igh-power advanced **T**orus e**X**periment (**NHTX**)

Baseline operating scenario:

P_{heat}	50MW
R_0	1m
A	1.8-2
κ	$\leq$ 3
B_T	2T
I_P	3-3.5MA
β_N	4.5
β_T	14%
n_e/n_{GW}	0.4-0.5
f_{BS}	$\approx$ 70%
f_{NICD}	100%
$H_{98Y,2}$	$\leq$ 1.3
E_{NB}	110keV
P/R	50MW/m
Solenoid	½ swing to full I_P

ST-based Component Test Facility (ST-CTF) is attractive concept for "Harnessing Fusion Power"

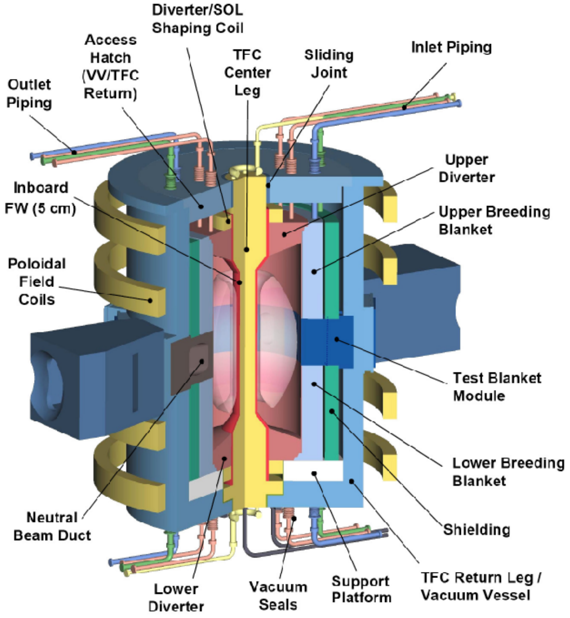
- **ST-CTF Required Conditions:**

From M. Peng APS-2007, based on NCT presentation to FESAC 8/7/2007

Performance metrics	ITER	Required Conditions	Demo Goals
Continuous operation	~hour	weeks	~months
14-MeV neutron flux on module (MW/m^2)	~0.8	1.0-2.0	~3
Total neutron fluence goal ($MW\text{-}yr/m^2$)	~0.3	6	~6-15
Duty factor goal	~1%	30%	~80%
Tritium self-sufficiency goal (%)	~0	~100	≥100

W_L [MW/m^2]	0.1	1.0	2.0
R0 [m]	1.20		
A	1.50		
kappa	3.07		
qcyl	4.6	3.7	3.0
Bt [T]	1.13	2.18	
Ip [MA]	3.4	8.2	10.1
Beta_N	3.8		5.9
Beta_T	0.14	0.18	0.28
n_e [$10^{20}/m^3$]	0.43	1.05	1.28
f_{BS}	0.58	0.49	0.50
T_{avgi} [keV]	5.4	10.3	13.3
T_{avge} [keV]	3.1	6.8	8.1
HH98	1.5		
Q	0.50	2.5	3.5
$P_{aux\text{-}CD}$ [MW]	15	31	43
E_{NB} [keV]	100	239	294
P_{Fusion} [MW]	7.5	75	150
T M height [m]	1.64		
T M area [m^2]	14		
Blanket A [m^2]	66		
$F_{n\text{-}capture}$	0.76		
P/R [MW/m]	**14**	**38**	**61**
Solenoid	Iron core or MIC solenoid for startup		

- **ST advantages for CTF:**
  - Compact device, high β
    - Reduced device cost
    - Reduced operating cost ($P_{electric}$)
    - Reduced T consumption
  - Simplified vessel and magnets
    - Fully modularized core components
    - Fully remote assembly/disassembly



ST-based Component Test Facility (**ST-CTF**)

NSTX will make world-leading contributions to ST development and contribute strongly to ITER and fundamental toroidal science

- NSTX Mission
- Unique Parameter Regimes Accessed by NSTX
  - Macroscopic Stability
  - Transport and Turbulence
  - Waves and Energetic Particles
  - Boundary Physics
  - Plasma Formation and Sustainment
- Next-step ST Missions
- Gaps Between Present and Next-step STs
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Performance gaps between present and next-step STs

For NHTX, ST-CTF scenarios: reduce n_e, increase NBI-CD, confinement, start-up/ramp-up

For ST-DEMO scenarios: increase elongation, β_N, f_{BS}, confinement, start-up/ramp-up

Present high β_N & f_{NICD}	NSTX	NSTX-U	NHTX	ST-CTF	ST-DEMO
A	1.53	1.65	1.8	1.5	1.6
κ	2.6-2.7	2.6-2.8	2.8	3.1	3.7
β_T [%]	14	10-16	12-16	18-28	50
β_N [%-mT/MA]	5.7	5.1-6.2	4.5-5	4-6	7.5
f_{NICD}	0.65	1.0	1.0	1.0	1.0
$f_{BS+PS+Diam}$	0.54	0.6-0.8	0.65-0.75	0.45-0.5	0.99
$f_{NBI\text{-}CD}$	0.11	0.2-0.4	0.25-0.35	0.5-0.55	0.01
$f_{Greenwald}$	0.8-1.0	0.6-0.8	0.4-0.5	0.25-0.3	0.8
H_{98y2}	1.1	1.15-1.25	1.3	1.5	1.3

Dimensional/Device Parameters:

Solenoid Capability	Ramp+flat-top	Ramp+flat-top	Ramp to full I_P	No/partial	No
I_P [MA]	0.72	1.0	3-3.5	8-10	28
B_T [T]	0.52	0.75-1.0	2.0	2.5	2.1
R_0 [m]	0.86	0.92	1.0	1.2	3.2
a [m]	0.56	0.56	0.55	0.8	2.0
I_P / aB_{T0} [MA/mT]	2.5	1.8-2.4	2.7-3.2	4-5	6.7

Near-term highest priority is to assess proposed ST-CTF operating scenarios

Gaps between present and future STs motivate NSTX scientific goals and associated upgrades

1. Increase and understand beam-driven current at lower n_e, ν^*
   - Next-step STs ***require*** full NICD to achieve missions, and NBI-CD is largest gap
   - But, lower n_e, ν^* also impacts AE avalanches, transport, MHD, pedestal, ELMs

   → ***Test increased NBI-CD with density reduction, higher T_e, higher NBI power***

2. Increase and understand H-mode confinement at low ν^*
   - ST energy confinement – in particular electron energy confinement - not sufficiently well understood to make extrapolation to next-steps with high confidence

   → ***Determine modes responsible for transport, determine scaling vs. B_T, I_P, P_{HEAT}***

3. Demonstrate and understand non-inductive start-up and ramp-up
   - Non-inductive ramp-up essential to ST-CTF and ST-DEMO
   - Increased non-inductive start-up current must also be demonstrated

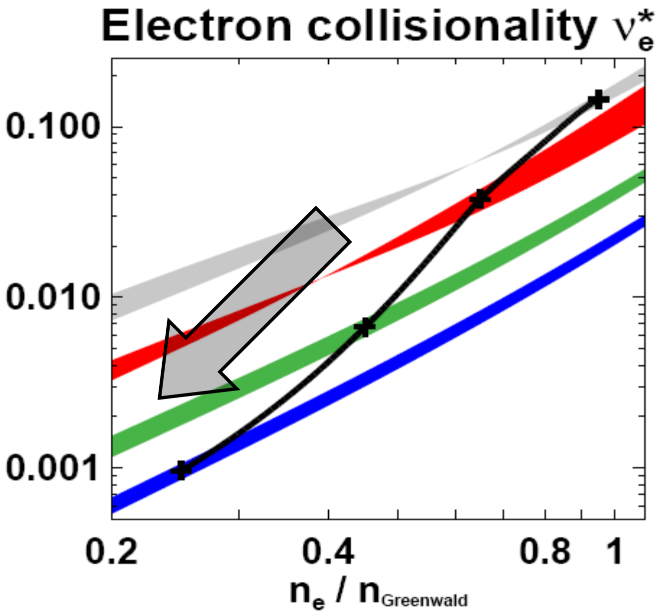
   → ***Increase ramp-up heating power & current drive to test I_P ramp-up techniques***

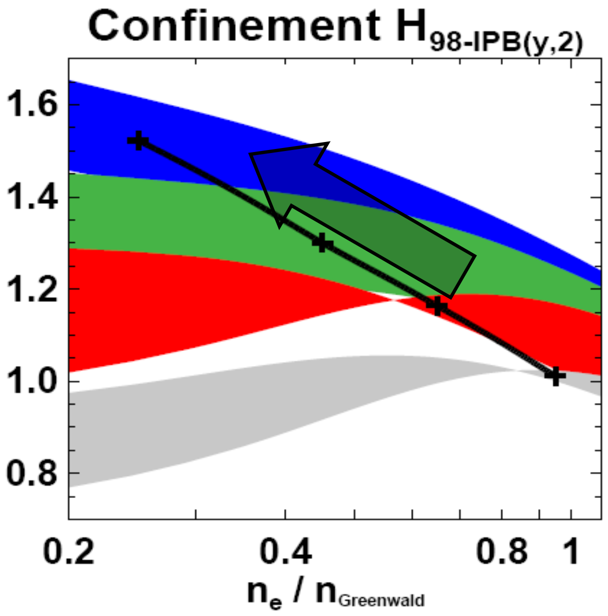
4. Sustain β_N and understand MHD near and above no-wall limit
   - Operation at no-wall limit assumed as baseline for NHTX and ST-CTF designs
   - Increased β_N, κ increases f_{BS}, β_T - would enhance ST-CTF, needed for ST-DEMO

   → ***Improve control of β, RWM/EF, rotation and q profiles to optimize stability***

Extrapolation from NSTX to ST-CTF is 2 orders of magnitude in ν_e*, factor of 1.4 in H_{98}, factor of 1-2 in ρ*

- **Collisionality dependence of ST confinement not yet understood**
- **H_{98} = 1.5 → 1 implies factor of 3 increase in required heating power**


Electron collisionality ν_e^*


Confinement $H_{98\text{-}IPB(y,2)}$

Upgraded NSTX could access ≥ factor of 4 lower ν* by increasing pumping, B_T, I_P, P_{HEAT}

Device	R_0/a	R_0	B_{T0}	β_N	P_{HEAT}	P_{NBI}	f_{NICD}
NSTX	1.5	0.86m	0.45T	5.8	6 MW	6 MW	50-70%
NSTX-U	1.6	0.92m	1.0T	5.0	14 MW	10 MW	50-100%
NHTX	1.8	1.00m	2.0T	4.5-5	50 MW	30 MW	100%
ST-CTF	1.5	1.20m	2.5T	3.5-4	65 MW	30 MW	100%

Decreased collisionality and density could impact physics and plasma performance across all topical science areas

- Macroscopic Stability
  - RWM critical rotation and neoclassical viscous torques may increase at lower ν_i
- Transport & Turbulence
  - Underlying instabilities (micro-tearing, CTEM, and ETG) scale differently versus ν^*
  - If $T_e(r)$ is set by a critical ∇T_e, H-mode confinement may be reduced at reduced n_e
- Boundary Physics
  - ELM ΔW increases at lower ν_e * - could impact confinement, plasma purity, divertor
  - ELM stability may improve at lower ν_e* - possible second-stability access
  - Detachment for heat flux reduction will be more challenging at reduced SOL density
- Wave-Particle Interactions
  - AE avalanches may be more easily triggered at reduced n_e due to increased fast-ion pressure fraction resulting in possible fast-ion redistribution and/or loss
- Plasma Start-up, Ramp-up, Sustainment
  - NBI-CD and RF-CD efficiency for ramp-up are increased at reduced n_e, increased T_e
  - ST-CTF scenarios rely on reduced n_e and increased T_e to increase NBI current drive efficiency to achieve 100% non-inductive current fraction.

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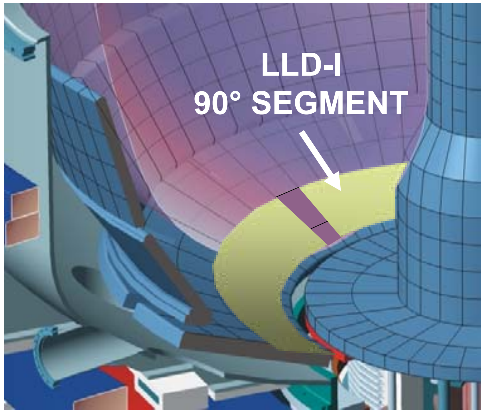
2009-10 upgrades will enable unique and exciting research in support of 3 highest priority research goals

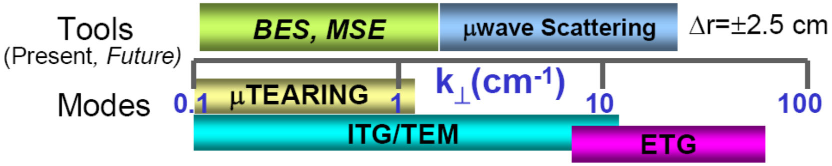
1. Reduce electron density using ***liquid*** lithium, improve understanding of how Li improves confinement and reduces/eliminates ELMs

   →Implement liquid lithium divertor (LLD)

   LLD-I = porous Mo surface bonded to heated Cu plate



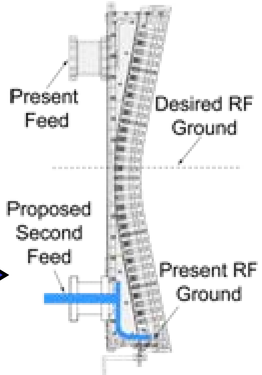
2. Measure full wave-number spectrum of turbulence to determine modes responsible for anomalous transport



   → Implement BES to complement existing high-k scattering diagnostic

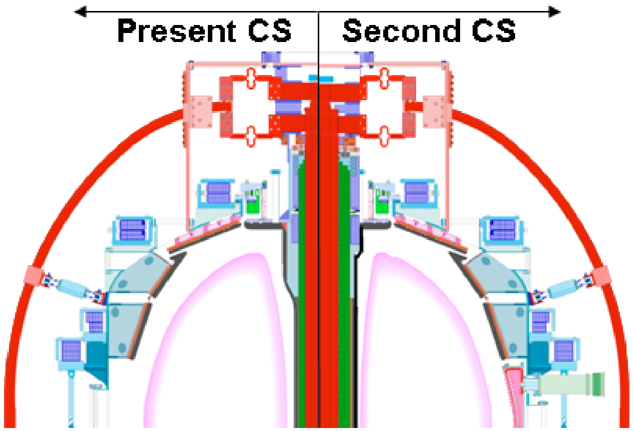
3. Asses if higher power HHFW can ramp-up I_P in H-mode (BS+RF overdrive) and heat high-β_N NBI H-mode scenarios

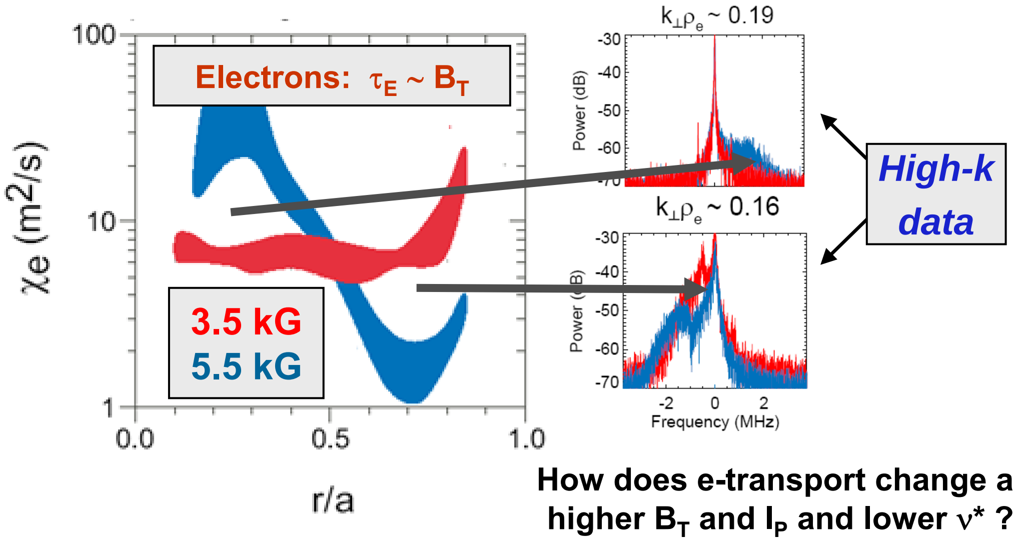
   → Upgrade HHFW system for higher P_{RF} + ELM resilience



Upgrade for FY12 incremental (FY11) : New center stack for 1T, 2MA, 5s to expand understanding and performance of ST plasmas

R_0 / a = 1.25-1.3 → 1.5-1.6

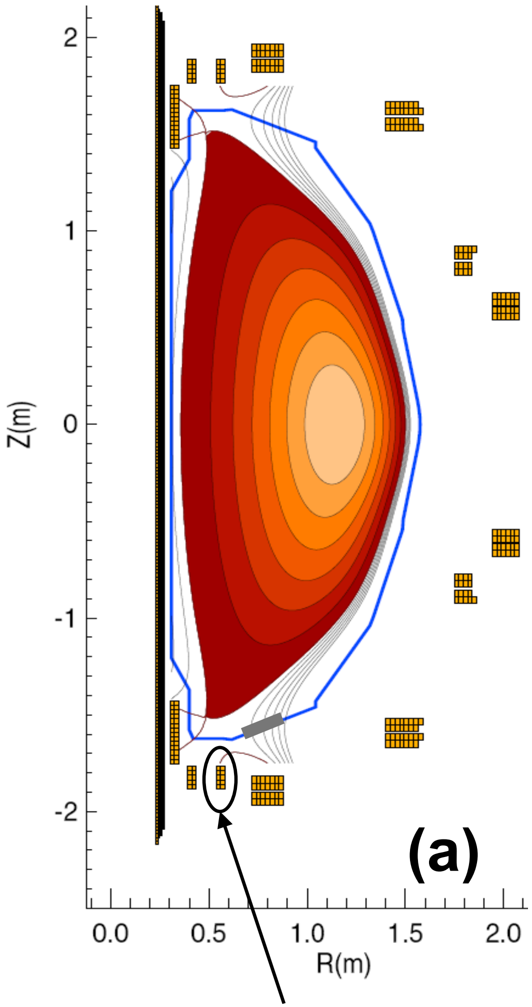




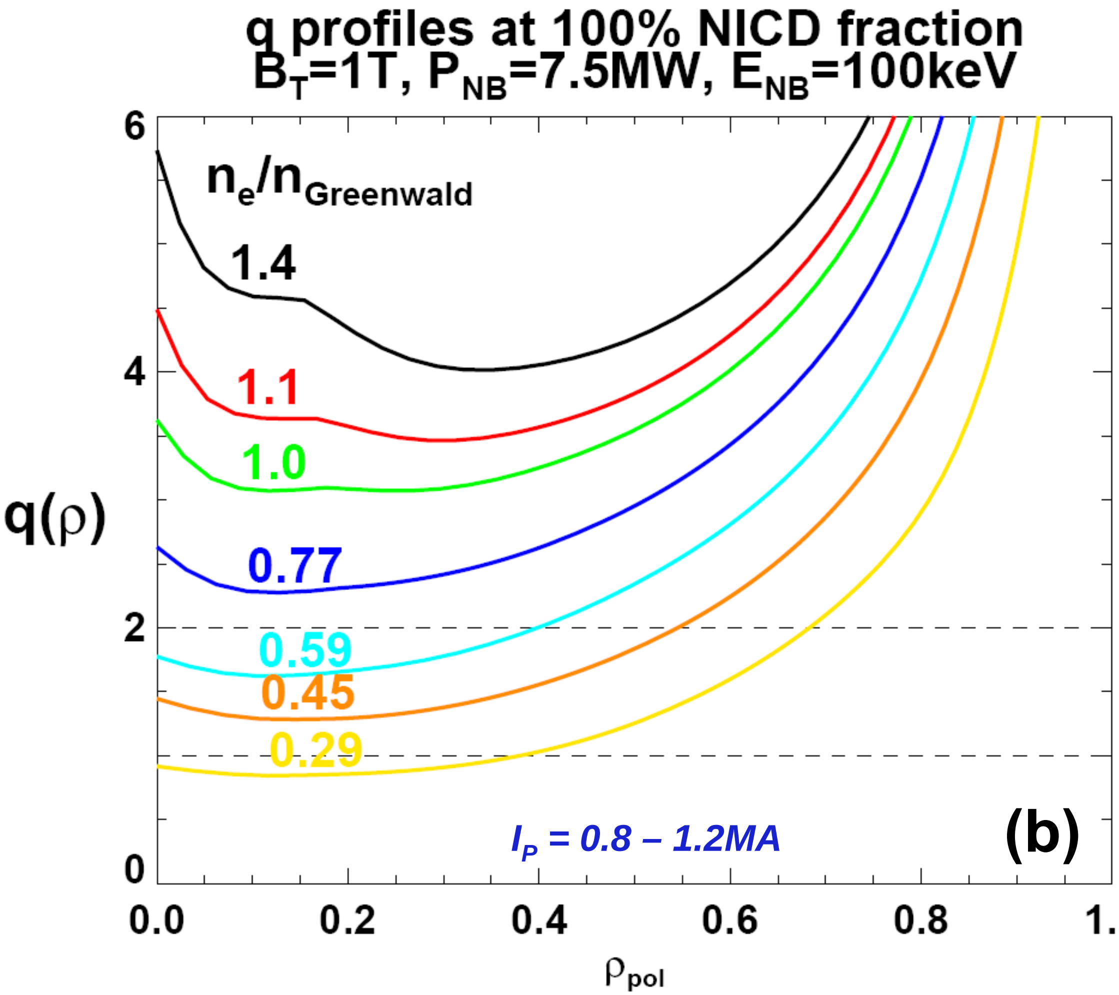
How does e-transport change at higher B_T and I_P and lower ν^* ?

- Access higher temperature, lower collisionality plasma
  - Understand impact of reduced ν^* for all topical science areas
- Improve understanding of transport and turbulence:
  - Assess if electron $\tau_E \sim B_T$ is result of low B_T, high β, suppressed ion transport, other
  - Assess ion turbulence scaling as field increases, neoclassical transport decrease
- Assess heating, start-up, ramp-up closer to parameters of next-step STs:
  - NBI $v_{fast} / v_{Alfvén}$ lower → fast-ion instability drive modified/reduced
  - HHFW surface waves reduced → improved power coupling
  - Higher B_T, T_e aids plasma start-up (Coaxial Helicity Injection, plasma guns, PF)

New CS will include additional PF coils for “X-divertor”, and higher B_T enables q_{min} control using n_e to control NBI-CD



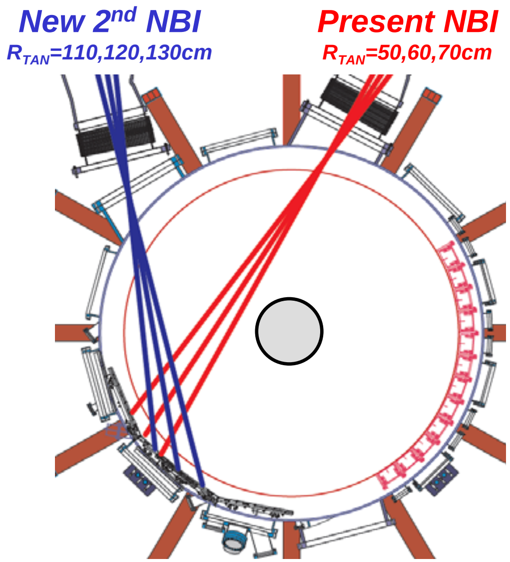
Additional divertor coils for very high flux expansion (up to 60) exhaust onto LLD in outboard “X-divertor” configuration



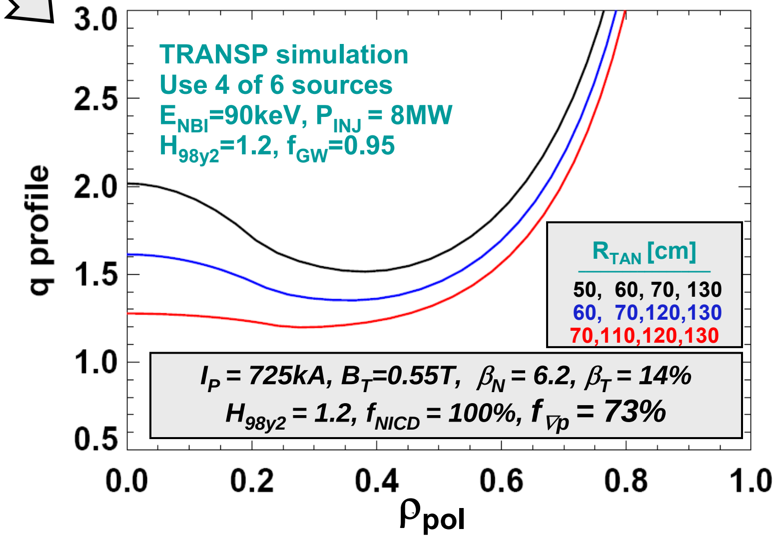
q_{min} > 1 achievable using existing 3 NBI sources (100keV, 7.5MW) + additional 4MW of HHFW for H_{98} = 1.2-1.4, β_N = 4.5-5, β_T = 10-12%

incremental

Upgrade for FY14 (FY13): 2nd NBI injecting at larger $R_{tangency}$ to expand performance and understanding of ST plasmas



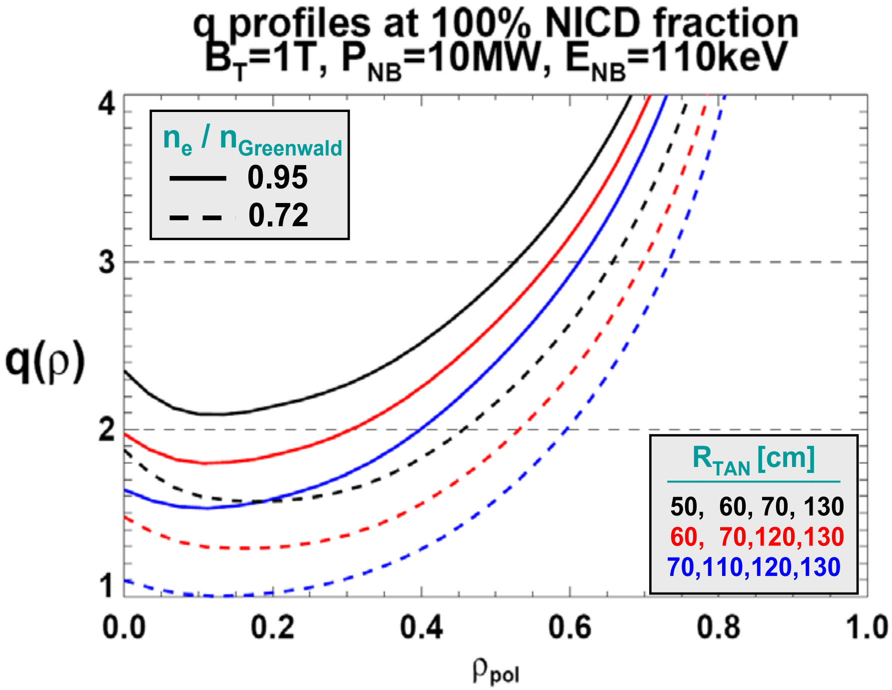
- Improved NBI-CD and plasma performance
  - Higher CD efficiency from large R_{TAN}
  - Higher NBI current drive from higher P_{NBI}
  - Higher β_P, f_{BS} at present $H_{98y2} \leq 1.2$ from higher P_{HEAT}
  - Large R_{TAN} → off-axis CD for maintaining $q_{min} > 1$
  - Achieve 100% non-inductive 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



2^{nd} NBI needed to support long-pulse (5s) fully non-inductive scenarios at high power at full TF ($B_T = 1T$)

- NBI duration 5s for 80kV → 5MW total per NBI, ~2s limit for ~7MW
  - 2^{nd} NBI can double maximum power or double duration at fixed power

- Fully non-inductive scenarios require 7-10MW of NBI heating for $H_{98} \leq 1.2$
  - τ_{CR} will increase from 0.35 → 1s if T_e doubles at lower n_e, higher B_T
  - Need 3-4 τ_{CR} times for J(r) relaxation → 5s pulses → need 2^{nd} NBI
  - f_{GW} > 0.7 needed at higher P_{NBI} to reduce core J_{NBICD} to maintain q_{min} > 1



Above: β_N=5, β_T=10%, I_P=0.95MA

β_N=6.1, β_T=16%, q_{min} > 1.3, I_P=1MA at B_T=0.75T also possible

2^{nd} NBI + 1T → study transport, stability (especially NTM) of high q_{min} plasmas for NHTX, ST-CTF

2^{nd} NBI also needed to support long-pulse (5s) high-I_P partial-inductive scenarios at high-power at full TF ($B_T = 1T$)

- Higher current expected to expand range of accessible T and ν^*
  - Accessible ν^* will depend on how confinement scales at higher field and current

- Access to higher current important for variety of physics issues – examples:
  - High-β_T physics at lower ν^* (RWM, NTV) – requires access to high I_P/aB_T
  - Core transport and turbulence at reduced ν^*, reduced $\chi_{i\text{-neoclassical}}$
  - Pedestal transport/stability, SOL width, heat flux scaling vs. current, …

- High I_P =1.6MA and B_T = 1T partially-inductively driven scenarios identified:
  - f_{NICD} = 65% with **q_{min} > 1**, β_N = 5, β_T =14%, NBI profile computed with TRANSP
    - Similar to present high NI-fraction discharges, but with 2× field and current
  - Higher current possible with present PF systems if $l_i < 0.5$
  - These scenarios also require $\geq$ 8MW of NBI heating power for $H_{98} \leq 1.2$

- Solenoid in new CS can support 2MA plasmas for 5s (flat-top $\Delta\Phi_{OH}$=1.6Vs)

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NSTX participation in International Tokamak Physics Activity (ITPA) benefits both ST and tokamak/ITER research

Actively involved in 17 joint experiments – contribute/participate in 24 total

Macroscopic stability
- MDC-2 Joint experiments on resistive wall mode physics
- MDC-3 Joint experiments on neoclassical tearing modes including error field effects
- MDC-12 Non-resonant magnetic braking
- MDC-13: NTM stability at low rotation

Transport and Turbulence
- CDB-2 Confinement scaling in ELMy H-modes: β degradation
- CDB-6 Improving the condition of global ELMy H-mode and pedestal databases: Low A
- CDB-9 Density profiles at low collisionality
- TP-6.3 NBI-driven momentum transport study
- TP-9 H-mode aspect ratio comparison

Wave Particle Interactions
- MDC-11 Fast ion losses and redistribution from localized Alfvén Eigenmodes

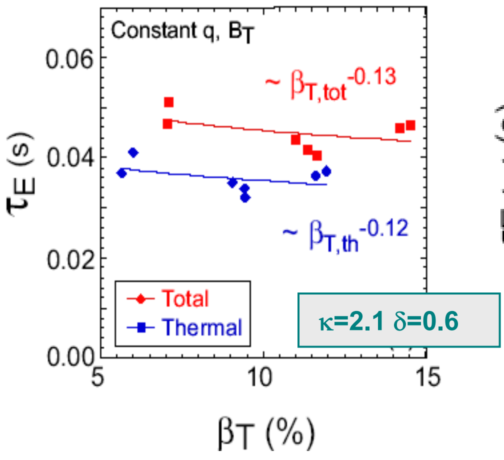
Boundary Physics
- PEP-6 Pedestal structure and ELM stability in DN
- PEP-9 NSTX/MAST/DIII-D pedestal similarity
- PEP-16 C-MOD/NSTX/MAST small ELM regime comparison
- DSOL-15 Inter-machine comparison of blob characteristics
- DSOL-17 Cross-machine comparison of pulse-by-pulse deposition

Advanced Scenarios and Control
- SSO-2.2 MHD in hybrid scenarios and effects on q-profile
- MDC-14: Vertical Stability Physics and Performance Limits in Tokamaks with Highly Elongated Plasmas

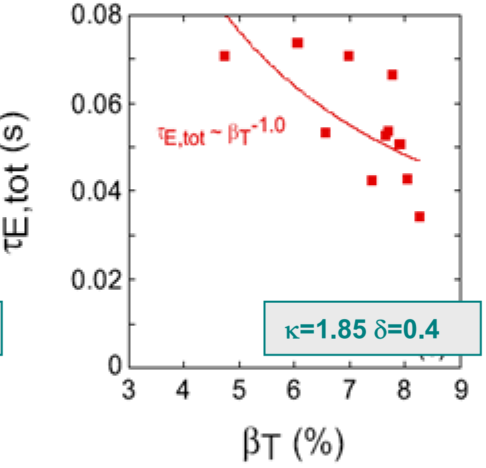
Examples of NSTX contributions to ITPA for ITER:

- **Transport:** β-dependence of H-mode confinement important to ITER advanced scenarios ($B\tau_{98y2} \sim \beta^{-0.9}$)
  - NSTX performed β-scan (factor of 2-2.5) at fixed q, B_T
  - Degradation of τ_E with β weak on NSTX for strongly shaped plasmas, stronger for more weakly shaped plasmas
  - Implies shape and/or ELM-type influences β dependence of H-mode confinement scaling

Small Type V ELMs

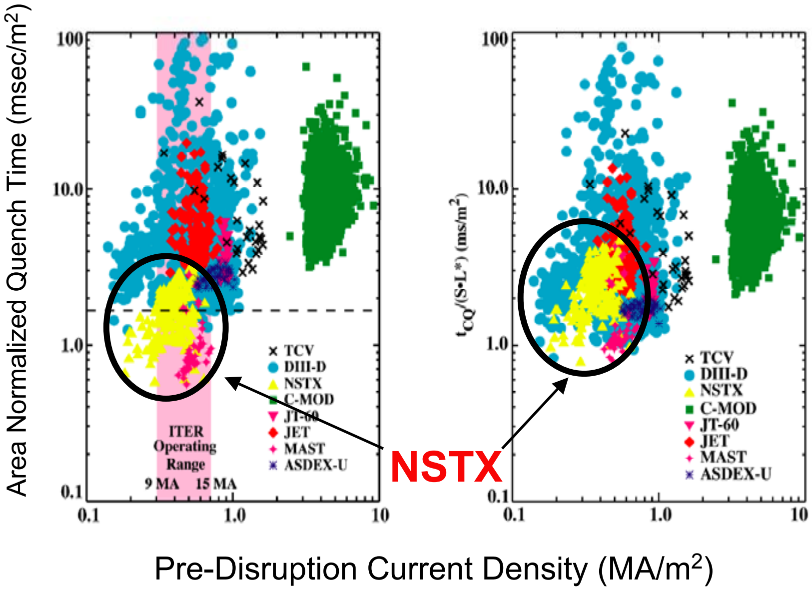


Type III/No → Type I ELMs



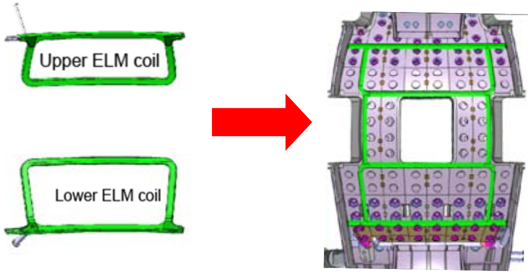
- **MHD:** Reduced normalized external inductance of low-A explains difference in I_P quench-rate
  - Implies tokamaks & STs have similar T_e during I_P quench phase (impurity radiation dominates dissipation of plasma inductive energy)

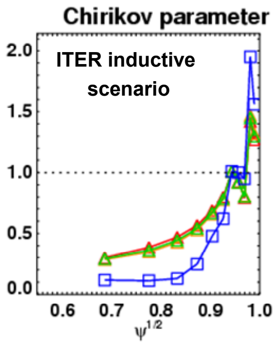
Area-normalized (left), Area and L_{ext}-normalized (right) I_p quench time vs. toroidal J_p (ITER DB)

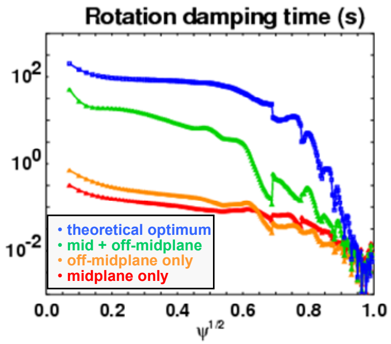


NSTX is actively engaged in ITER design activities

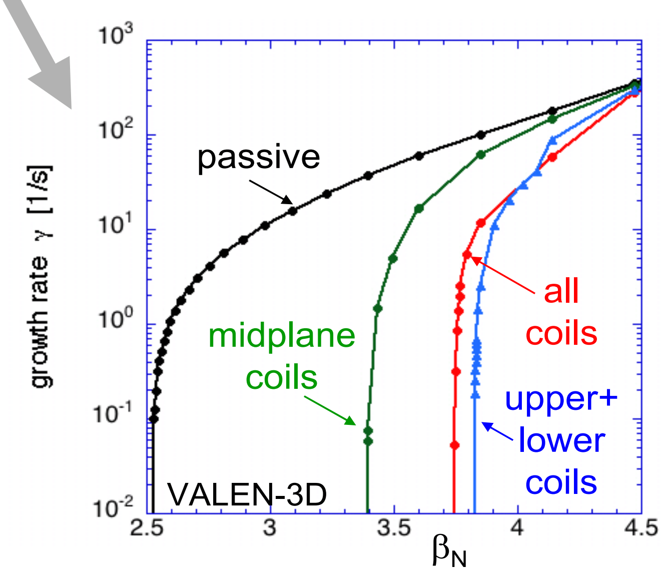
- Ideal Perturbed Equilibrium Code (IPEC)
  - Validated on NSTX for locked modes/error-fields, testing NTV theory
  - Extended to calculate plasma response effects for RMP calculations
    - Calculate upper+mid+lower coils can ergodize edge, minimize core flow damping







- VALEN code
  - Validated against NSTX RWM data
  - ITER RMP coils can stabilize n=1 RWM in ITER Q~5 steady-state scenario 4 ($\beta_N \approx 3$)



- Vertical control experiments
  - Maximum recoverable displacement $\Delta Z_{MAX}/a < 10\%$
  - Consistent with results at higher aspect ratio $A \approx 3$
  - Confirms the potential inadequacy of baseline ITER vertical control

Summary: NSTX will lead the U.S. effort to assess the properties and potential advantages of the ST for fusion

- NSTX will address important questions for ST and fusion science:
  - Can high normalized pressure be sustained with high reliability?
  - What are underlying modes and scalings of anomalous transport?
  - How does large fast-ion content influence Alfvénic MHD & fast-ion loss?
  - Can steady-state & transient edge heat fluxes be understood and controlled?
    - Is liquid Li attractive for taming the plasma-material interface?
  - Are fully non-inductive high-performance scenarios achievable in the ST?
  - Can a next-step ST operate solenoid-free with high confidence?

- Upgrades will greatly expand the scientific capabilities of NSTX to:
  - Access and understand impact of reduced collisionality on ST physics
    - Achievable through density reduction, higher B_T, I_P, power
    - Impacts all topical science areas
  - Access and understand impact of varied NBI deposition profile
    - Achievable through implementation of 2nd NBI
    - Impacts heating, rotation, current profiles, f(v) for fast-ion MHD
    - Access fully non-inductive operation and sustain it

- NSTX research will strongly address key gaps for next-step STs