NSTX

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Scientific Goals, Mission, and Objectives

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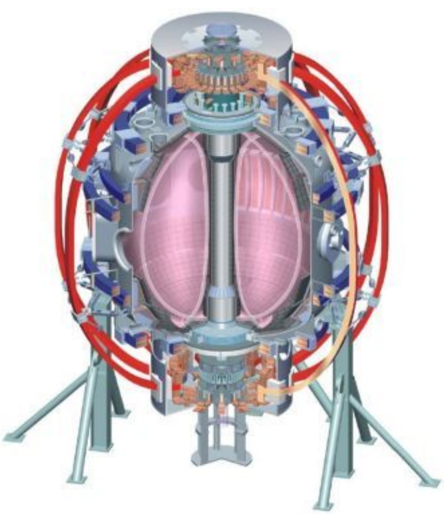
J.E. Menard, PPPL

For the NSTX Research Team

NSTX Facility Review
Director's Conference Room, PPPL
July 30-31, 2008



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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
- Upgrades and Understanding to Narrow Gaps
- Contributions to ITER and Tokamak Research
- Summary

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**

2007 FESAC Priorities Panel prioritized issues for getting from ITER to DEMO

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

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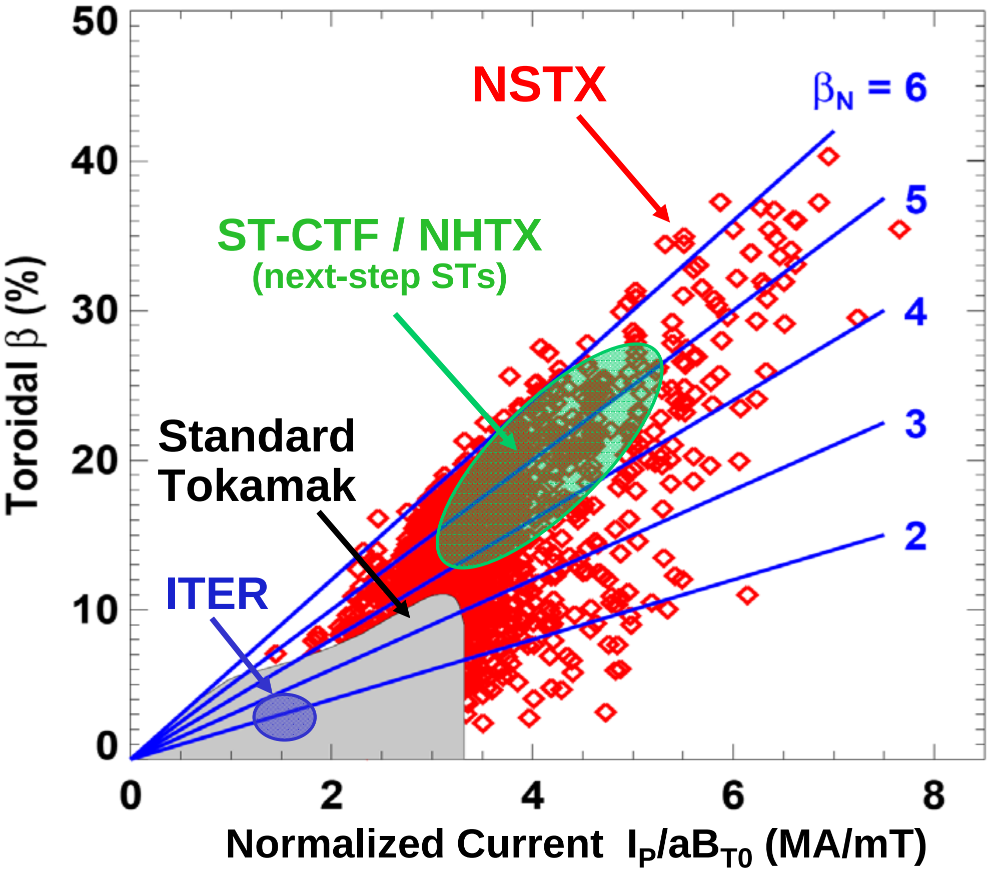
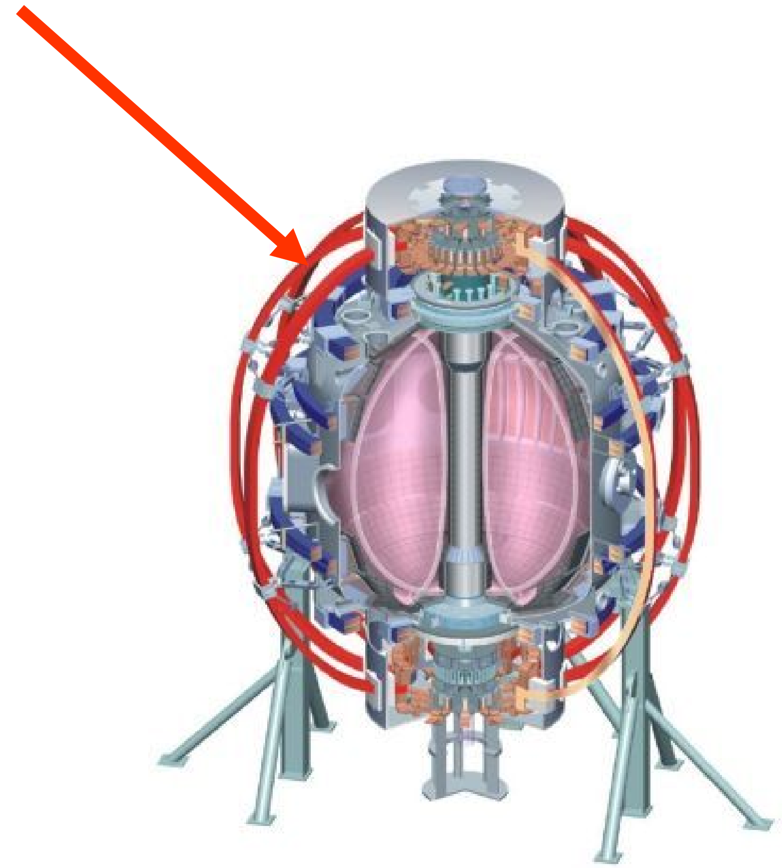
NSTX creates high β plasmas, and is assessing if the ST can be used as a compact high-performance fusion reactor

- ST accesses higher normalized current and higher normalized β
  - → higher $\beta_{Toroidal}$ = plasma pressure / toroidal magnetic field pressure
  - → high plasma pressure with smaller magnets

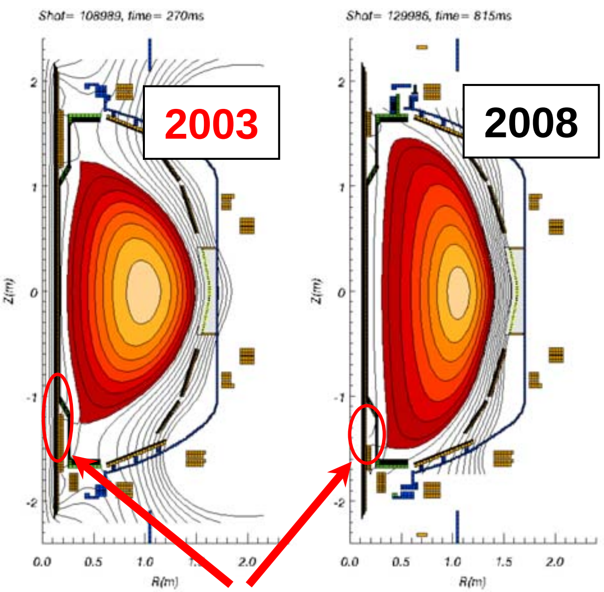




NSTX is improving control of plasma instabilities to increase the duration of sustained high β

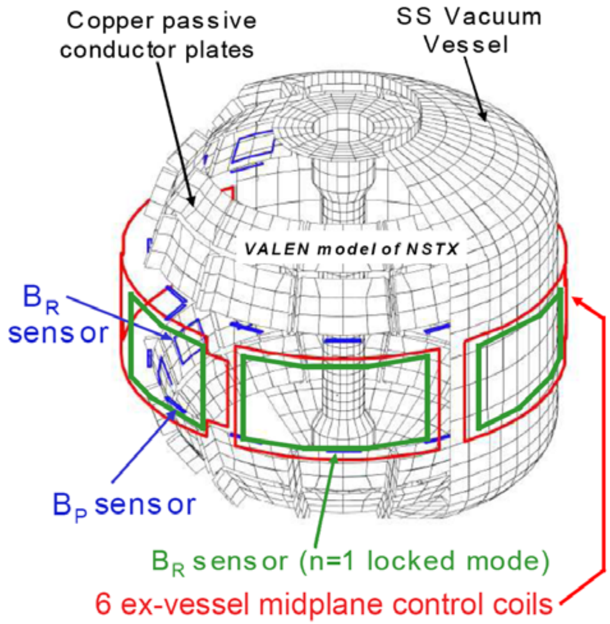
Increased plasma shaping from improved n=0 control for high κ and δ operation **+ $n \geq 1$ EF/RWM control =** **Duration of β_T > 15% increased factor of 4 from 2002 to 2008**

κ = 2.0, δ = 0.8, S ~ 23 κ = 3, δ= 0.8, S ~ 41

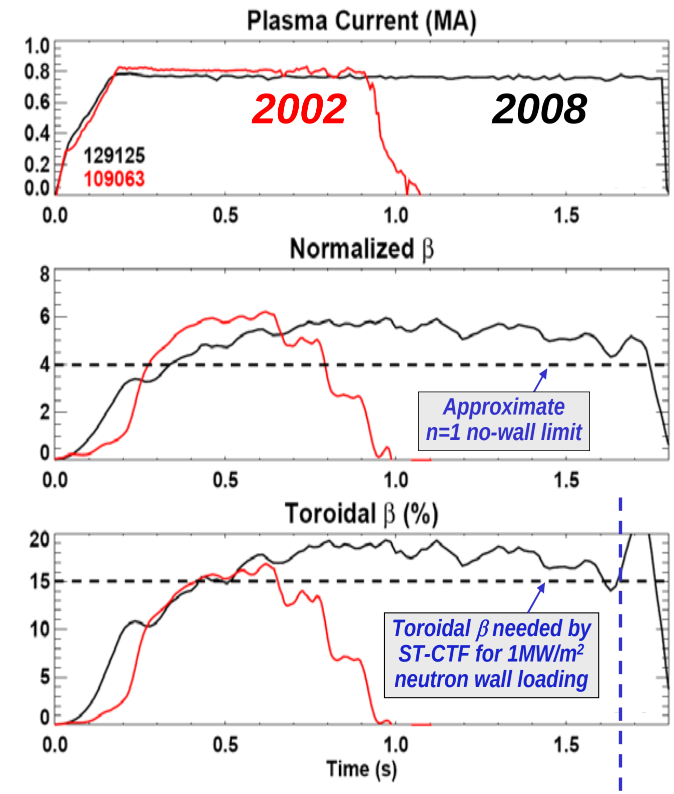


PF1A coil upgraded

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



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



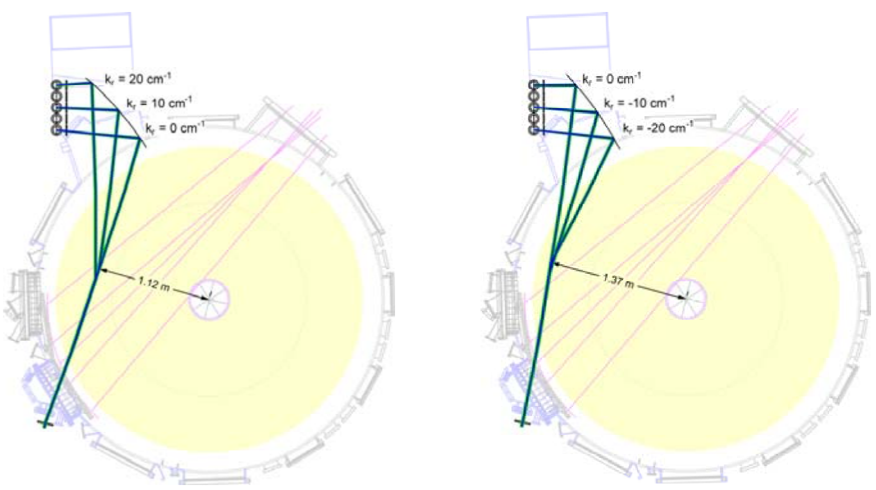
TF ramp-down due to coil heating limit

NSTX is utilizing unique diagnostics and plasma regimes to determine the modes responsible for electron transport

- **Electron transport dominates ST energy losses, is important for ITER burning plasma performance, but is poorly understood.**

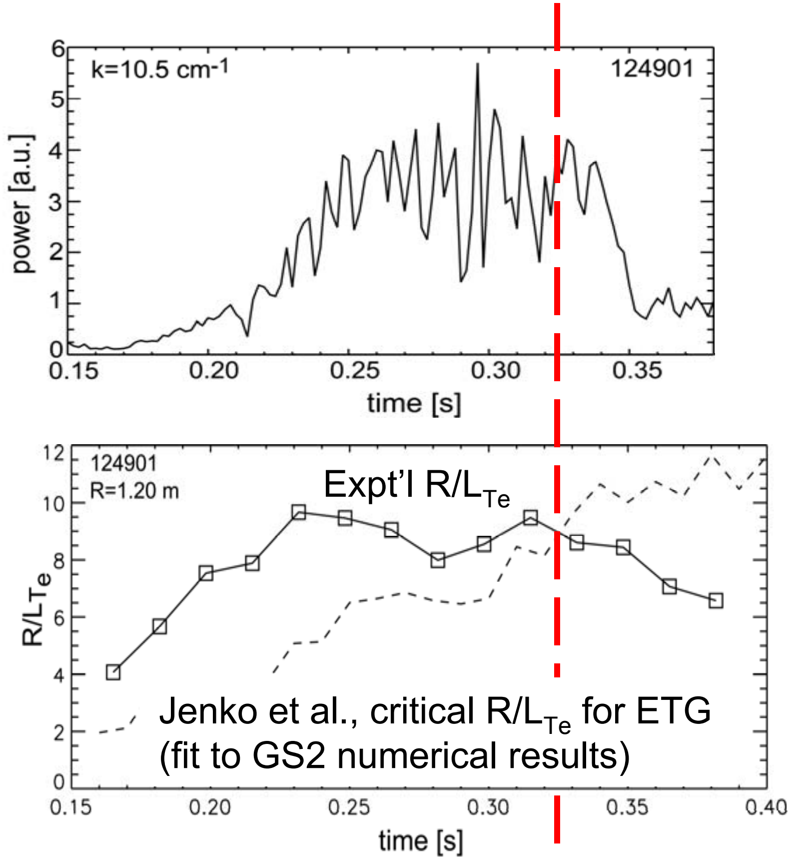
High-k scattering diagnostic ($\Delta r = \pm 3$ cm)

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



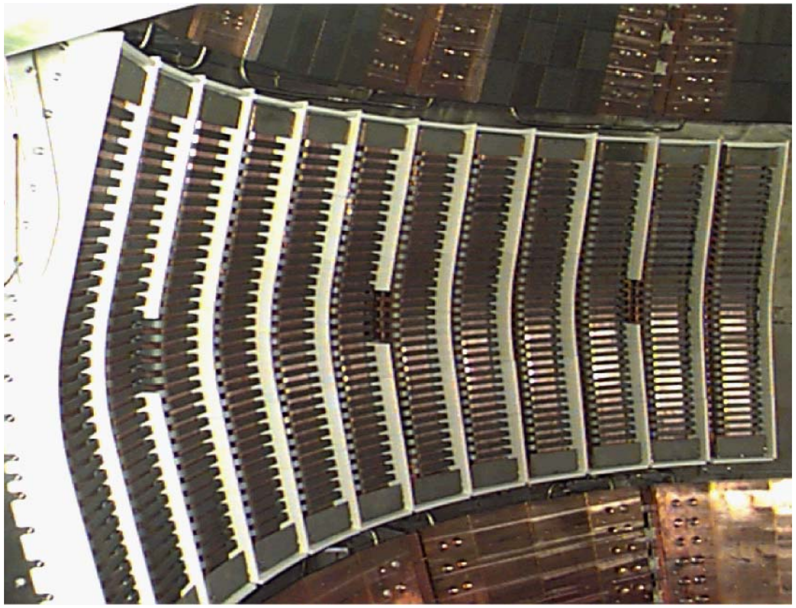
- k_r=2 (upper ITG/TEM) to ~24 (ETG) cm^{-1} (ρ_e~0.01 cm)
- Δr ~ 6 cm, Δk ~ 1 cm^{-1}
- Can vary scattering volume location (axis to near edge)

- **High fluctuation level when R/L_{Te} > critical value for electron temperature gradient mode (ETG)**



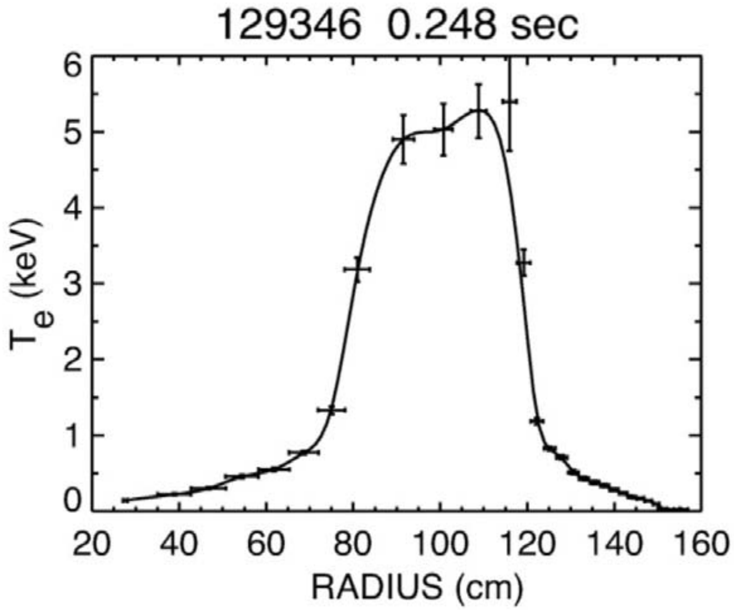
NSTX is improving the understanding and performance of wave heating techniques for high-β (over-dense) plasmas

HHFW Antenna Array



- **Twelve antennas**
- **Six 1 MW transmitters**
- **Real time phasing**
- **Top-fed**
- **Wave reflectometers**
- **Edge RF probes**

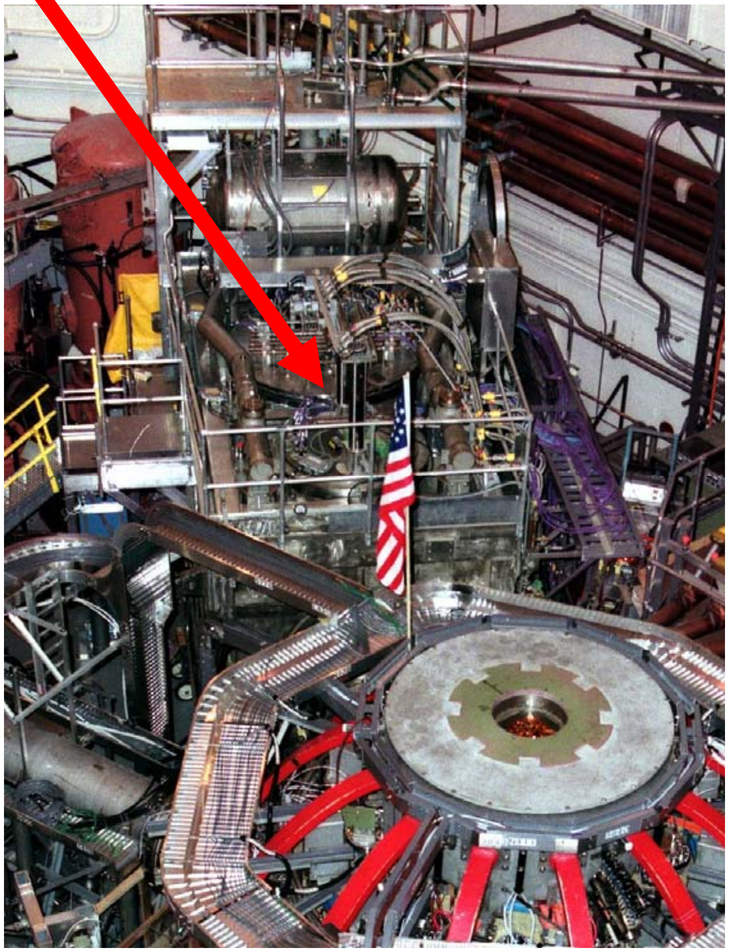
- High-harmonic fast-wave (HHFW)
  - Discovered that surface waves reduce heating efficiency if density near antenna is too high
  - Control of edge density improves heating → record T_e = 5keV in NSTX achieved with HHFW



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

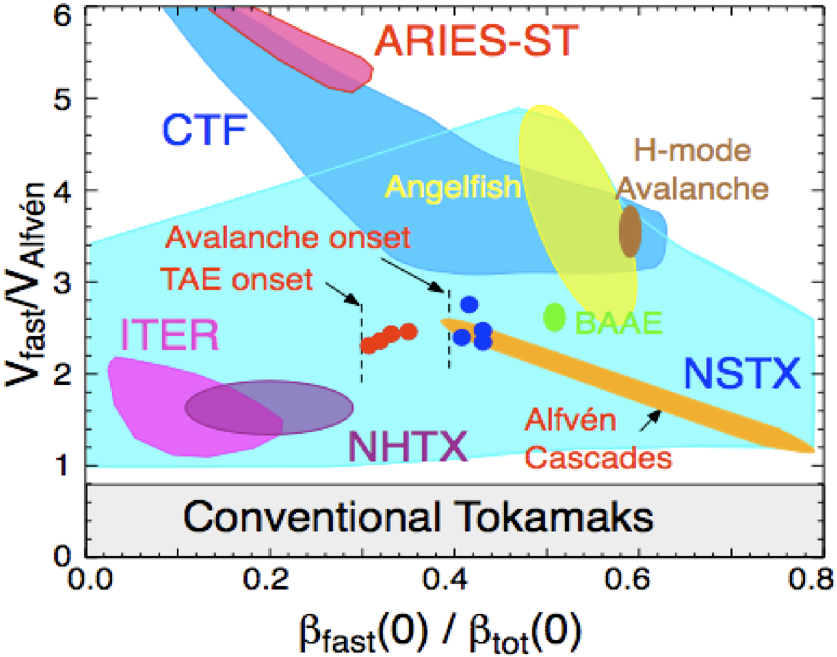
Fast-ions from NBI are used to simulate α-particles in ITER, and enable studies of fast-ion physics for next-step STs.

- NSTX neutral beam injection (NBI) used for heating, current drive, driving plasma rotation, and fast-ion physics studies



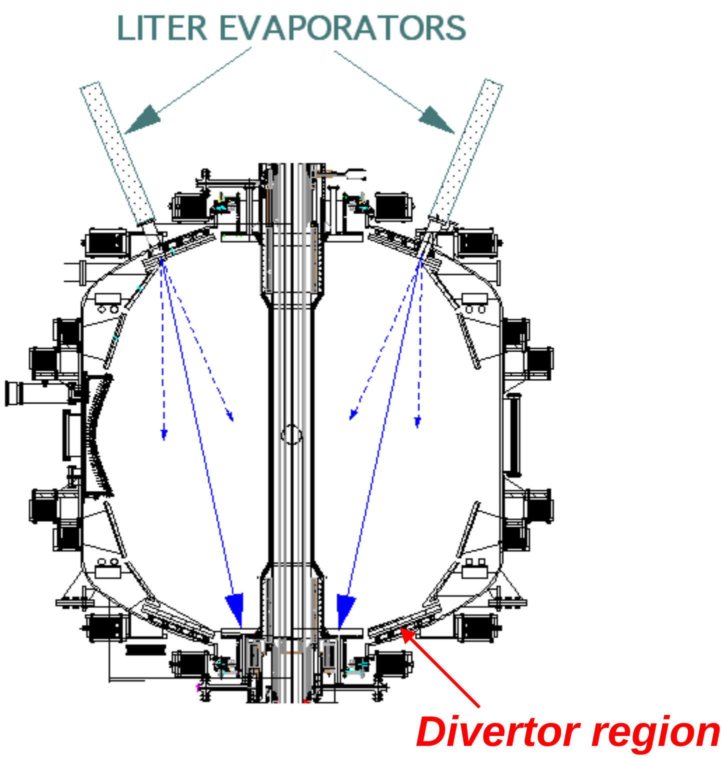
- NSTX studies the range of instabilities excited by the fast-ions, and the effects of the instabilities on the fast-ions

Fast ion instability space of NSTX, ITER, and future ST devices

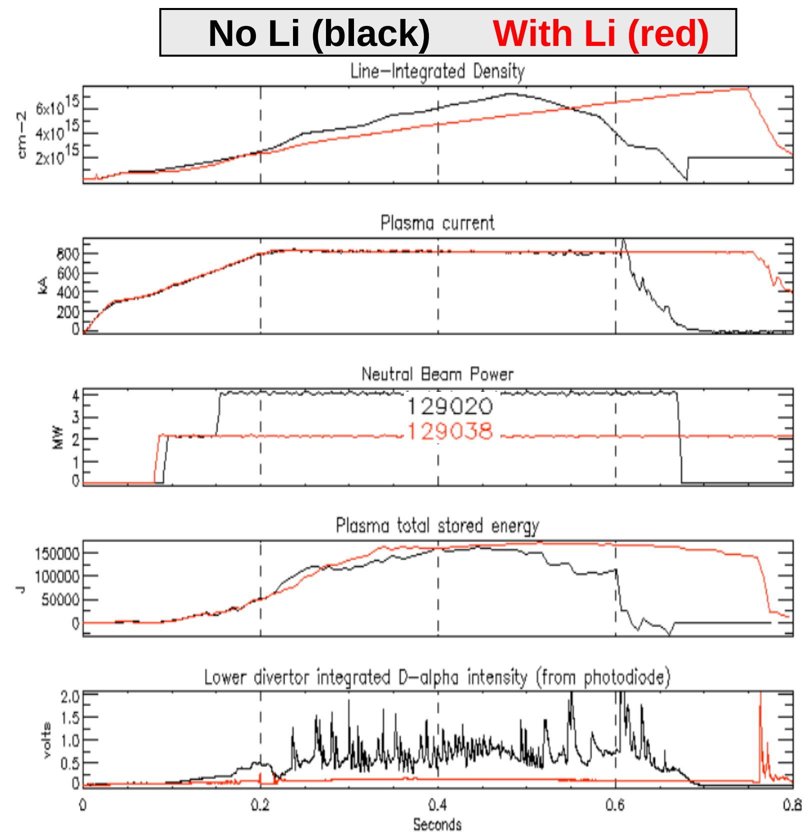


NSTX is unique in the world program in exploring lithium in a diverted H-mode plasma

- 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

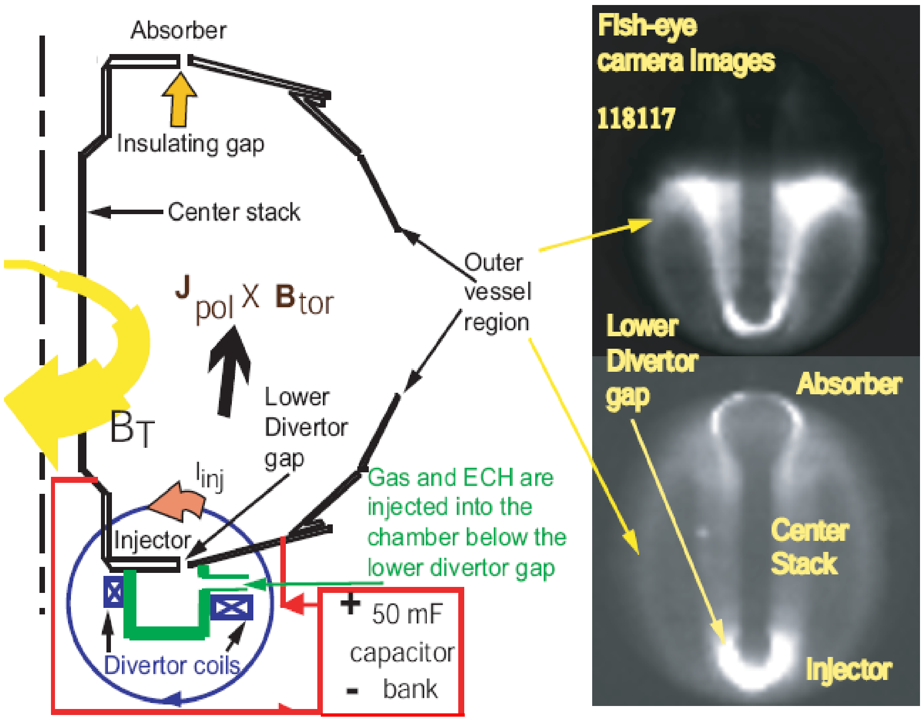


- Reproducible ELM elimination from Li
  - Large reduction in divertor D_α → reduced recycling
  - Plasma density reduced
  - Pulse-length extended
  - At 800kA, power must be reduced to avoid β limit
  - Confinement time doubled (up to 80ms)

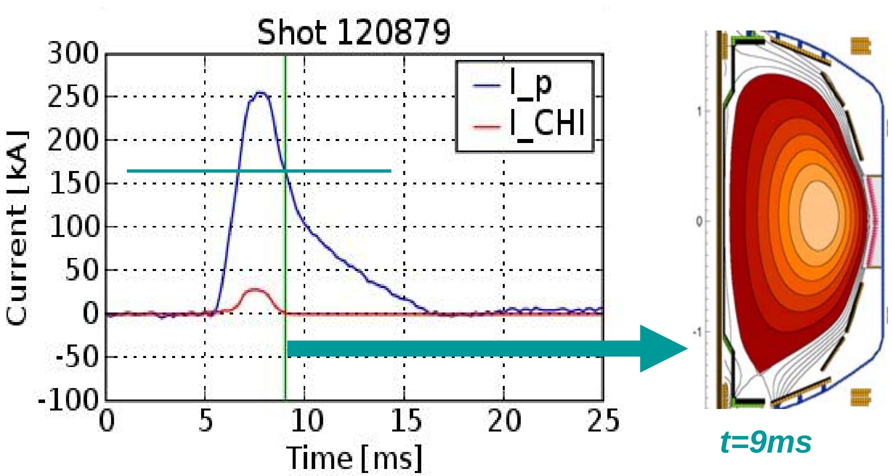


NSTX is testing unique methods of non-solenoidal plasma current start-up and ramp-up for STs

- Coaxial Helicity Injection (CHI)
  - Apply voltage between inner and outer vacuum vessel - up to 1.7kV thus far
  - J×B force pushes plasma into vessel
  - Current reconnects, forms tokamak

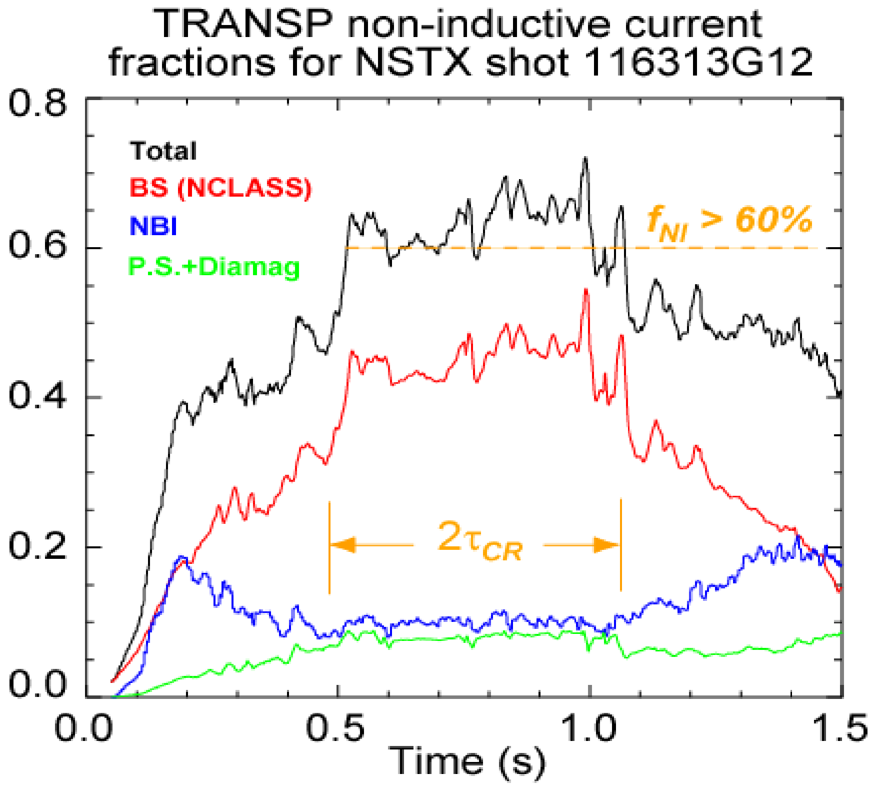


- Coaxial Helicity Injection Performance:
  - Generated record closed-flux I_P=160kA
  - Demonstrated coupling to induction and compatibility with high performance H-mode
  - Higher I_P limited by lack of auxiliary heating, possibly impurities/divertor conditions
    - Will upgrade "magnetic insulation" at absorber
    - Will modify outboard divertor material (C → Mo)

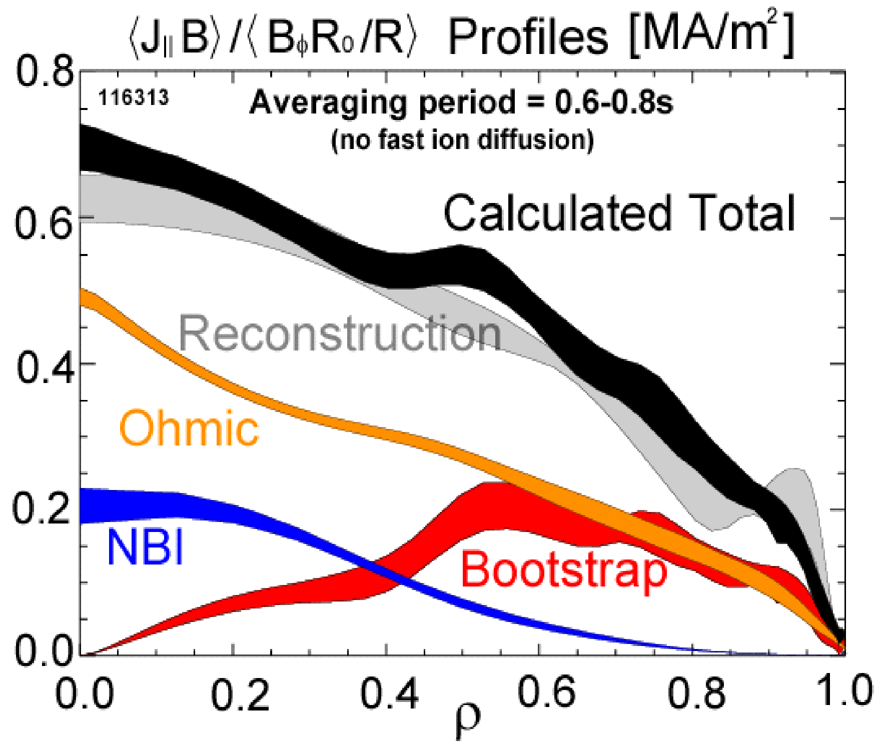


NSTX is developing sustained scenarios with a majority of the current driven non-inductively (i.e. w/o central solenoid)

- f_{NICD} = 65% (total non-inductive fraction)
- $f_{\nabla p}$ = 55% (fraction driven by plasma pressure)



Predicted and reconstructed current profiles are in agreement
(for plasmas free of core instability activity)



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

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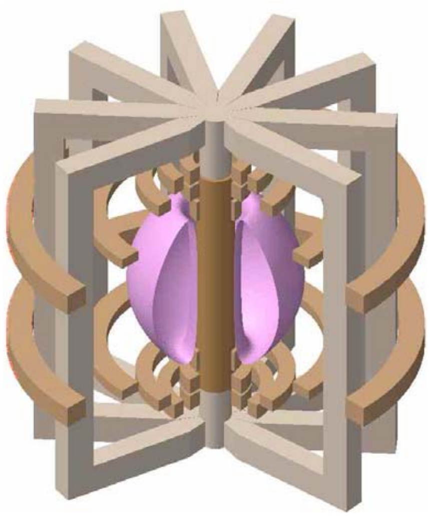
ST is attractive configuration for "Taming the plasma-material interface"

- 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"

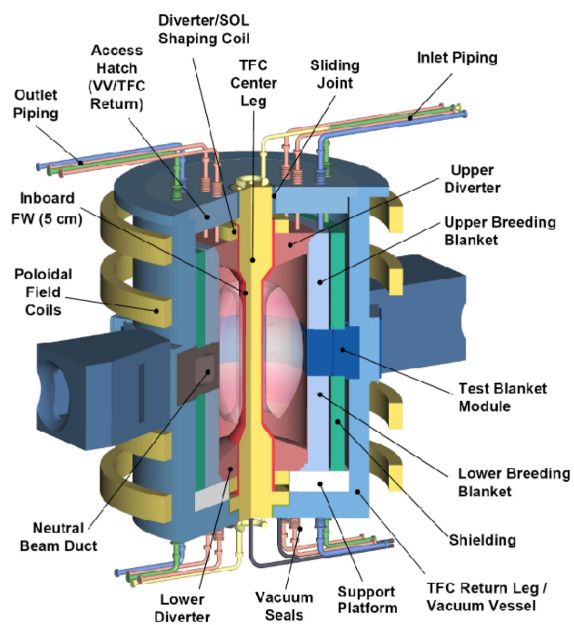
- **ST-CTF Required Conditions:**

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-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 **C**omponent **T**est **F**acility (**ST-CTF**)

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

1. Increase and understand beam-driven current lower n_e, ν^*

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

2. Increase and understand H-mode confinement at low ν^*

   → ***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

   → ***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

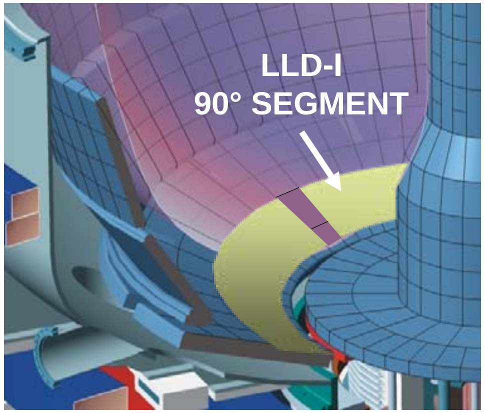
   → ***Improve control of β, RWM/EF, rotation and q profiles to optimize stability***

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

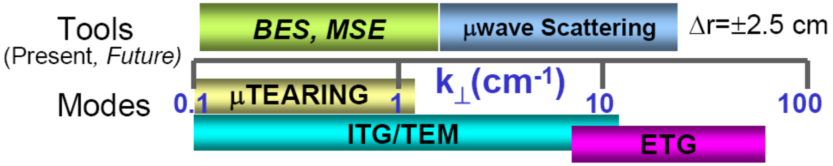
1. Reduce electron density using ***liquid*** lithium, improve understanding of how Li improves confinement and reduces/eliminates ELMs

   →Implement liquid lithium divertor (LLD)

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



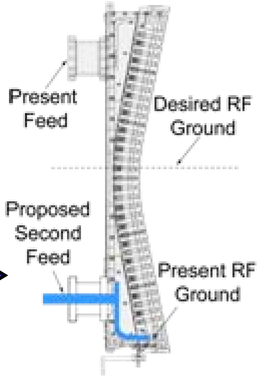
2. Measure full wave-number spectrum of turbulence to determine modes responsible for anomalous transport



   → Implement BES to complement existing high-k scattering diagnostic

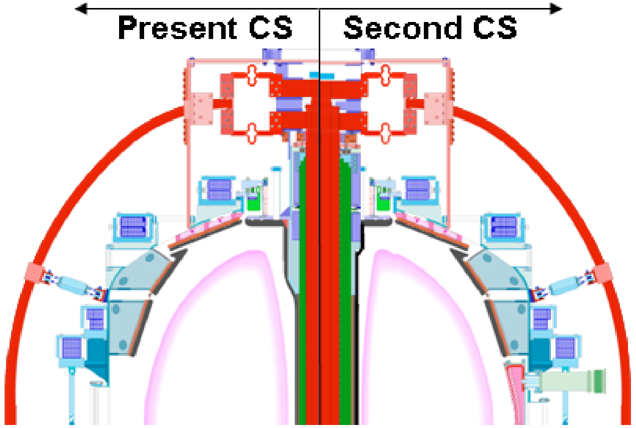
3. Asses if higher power HHFW can ramp-up I_P in H-mode (BS+RF overdrive) and heat high-β_N NBI H-mode scenarios

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

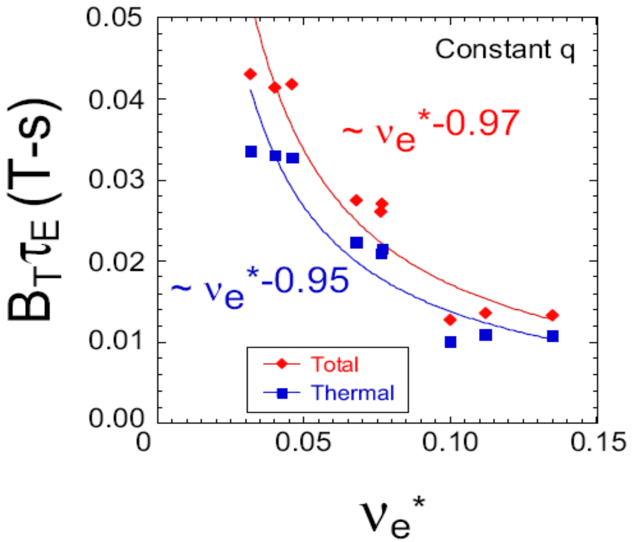


Upgrade for FY12 (FY11) incremental : New center stack for 1T, 2MA, 5s will expand understanding and performance of ST plasmas

R_0/a = 1.25-1.3 → 1.5-1.6



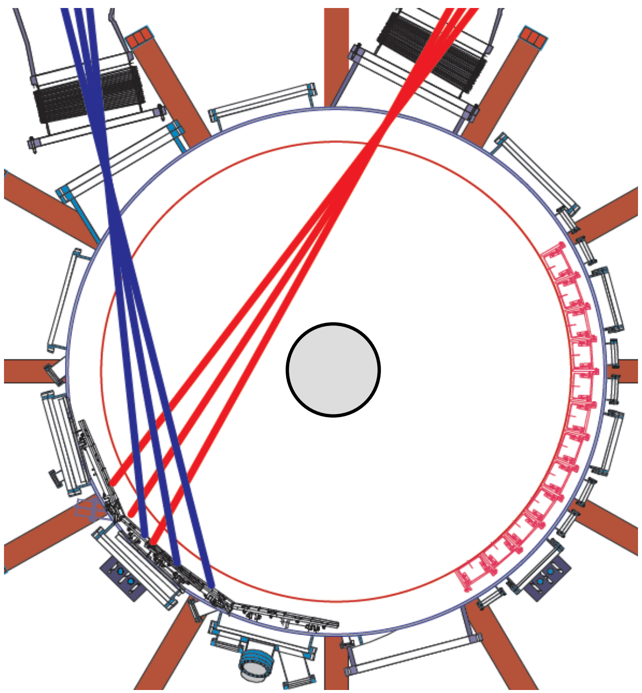
NSTX confinement exhibits strong dependence on collisionality ν^*



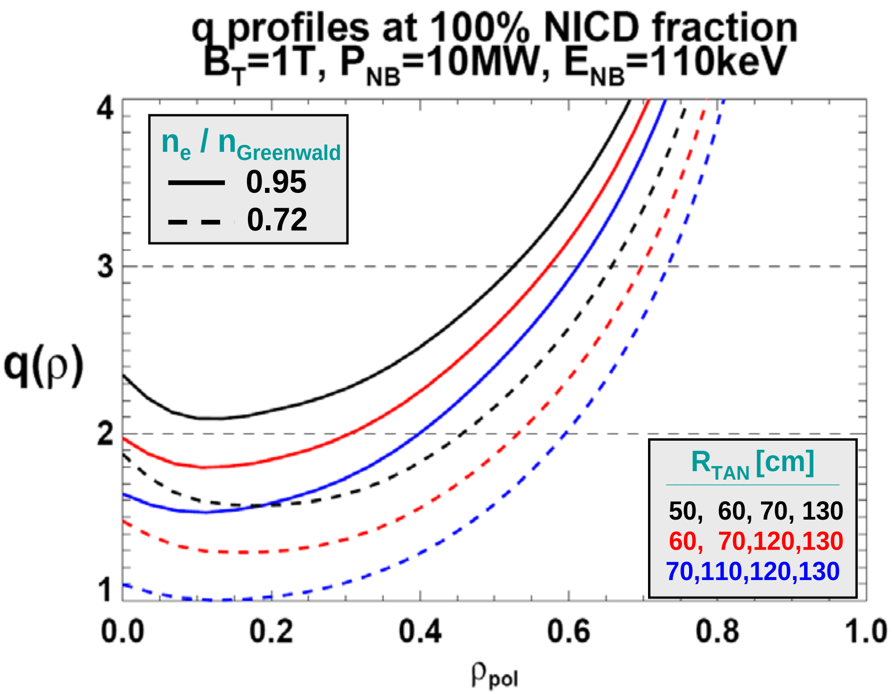
- Increase B_T and I_P to access higher temperature, lower collisionality plasma
- Improve understanding of transport and turbulence:
  - Assess if electron $\tau_E \sim B_T$ is result of low B_T, high β, suppressed ion transport, other
  - Assess ion turbulence scaling as field increases, neoclassical 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)

2^{nd} NBI in FY14 incremental (FY13) will support long-pulse (5s) fully non-inductive scenarios at high power at full TF (B_T = 1T)

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



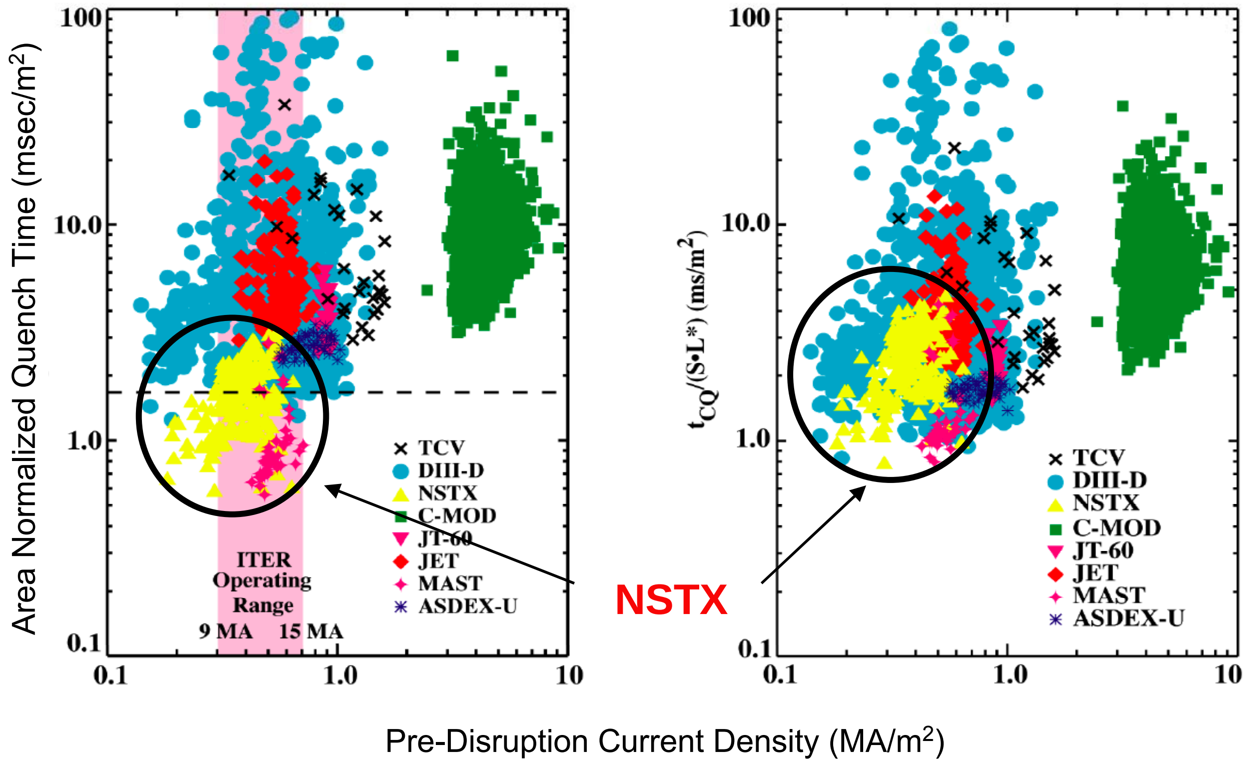
- **2^{nd} NBI + 1T CS → fully non-inductive**
- Source variation + density variation enable control of the core q value
  - Important for plasma stability, transport
  - Prototypical of next-step STs



Example of NSTX contribution to ITER physics basis:

- **MHD:** ST has faster current quench rate during a current disruption
  - Reduced normalized external inductance of ST explains difference in I_P quench-rate
  - Implies tokamaks & STs have similar T_e during I_P quench phase
    - Consistent with impurity radiation dominating dissipation of plasma inductive energy

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



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

BACKUP

Performance gaps between present and next-step STs

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

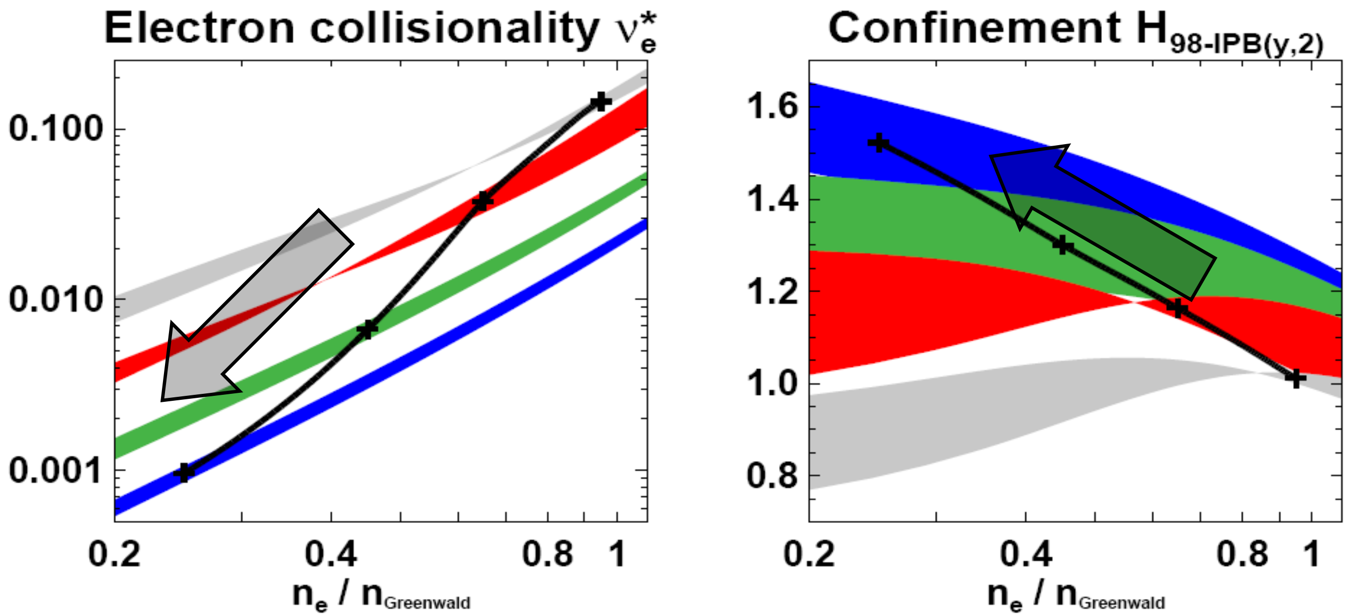
For ST-DEMO scenarios: increase elongation, β_N, f_{BS}, confinement, start-up/ramp-up

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-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

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**



Upgraded NSTX will 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%

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

NSTX actively involved in 17 joint experiments
ITPA experiments receive increased run priority

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

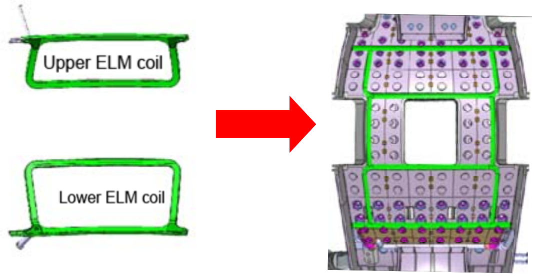
Wave Particle Interactions
- MDC-11 Fast ion losses and redistribution from localized Alfvén Eigenmodes

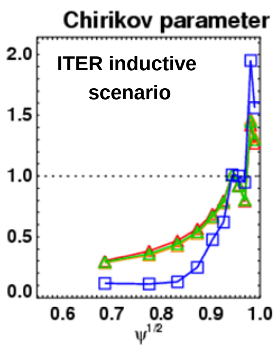
Boundary Physics
- PEP-6 Pedestal structure and ELM stability in DN
- PEP-9 NSTX/MAST/DIII-D pedestal similarity
- PEP-16 C-MOD/NSTX/MAST small ELM regime comparison
- DSOL-15 Inter-machine comparison of blob characteristics
- DSOL-17 Cross-machine comparison of pulse-by-pulse deposition

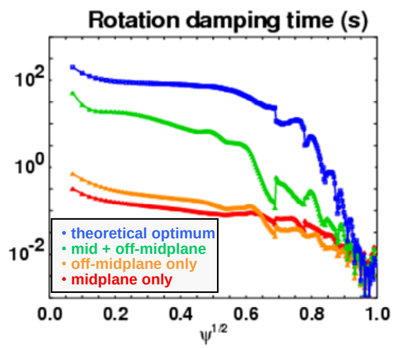
Advanced Scenarios and Control
- SSO-2.2 MHD in hybrid scenarios and effects on q-profile
- MDC-14: Vertical Stability Physics and Performance Limits in Tokamaks with Highly Elongated Plasmas

NSTX is actively engaged in ITER design activities

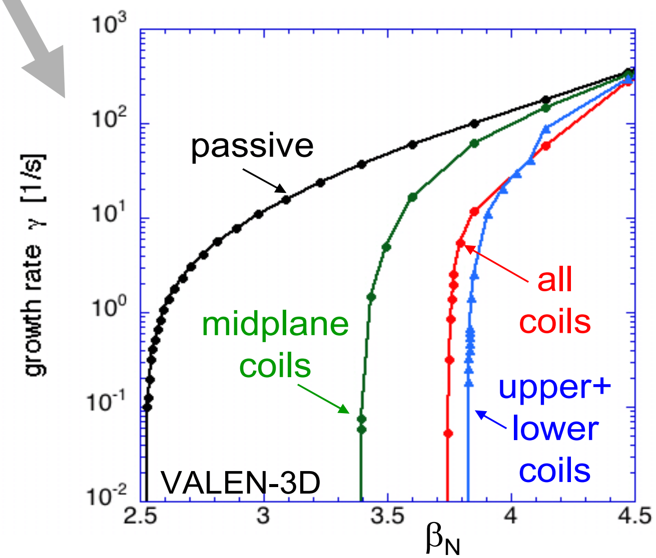
- Ideal Perturbed Equilibrium Code (IPEC) for ELM suppression
  - Physics analysis of design of internal coils proposed for ITER







- VALEN code for RWM control
  - ITER internal coils can stabilize n=1 RWM in ITER Q~5 steady-state scenario 4



- 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