NSTX Upgrade Scientific Motivation and Project Requirements

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NSTX Upgrade Project

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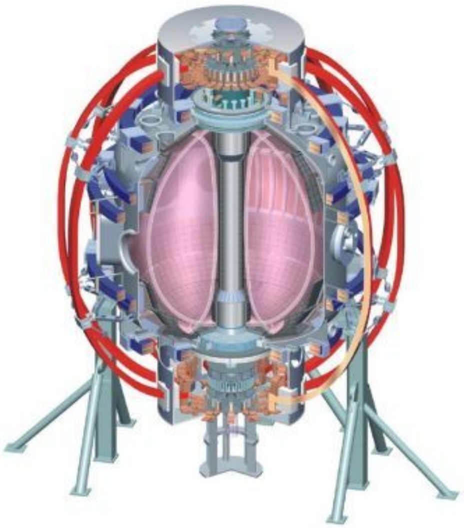

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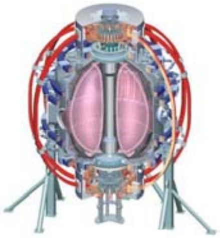
Outline

- Motivation
- Design Point Selection Process
- General Requirements Document, Design Point Info
- Discussion of Impact of Operational Scenarios

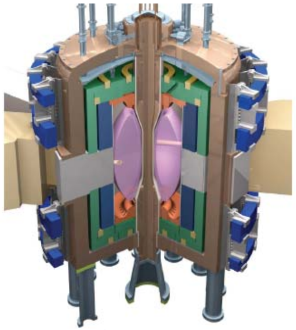
NSTX Upgrade will contribute strongly to toroidal plasma science and preparation for a fusion nuclear science (FNS) program

- 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 $W_{neutron} \sim 0.2\text{-}0.4 \rightarrow 1\text{-}2 MW/m^2$, $\tau_{pulse} = 10^3 \rightarrow 10^6 s$
  - 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 $P_{heat}/S \sim 1 MW/m^2$, high T_{wall}, $\tau_{pulse} \sim 10^3 s$

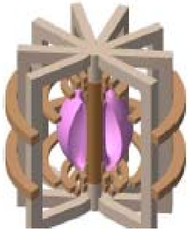


NSTX

NSTX-U

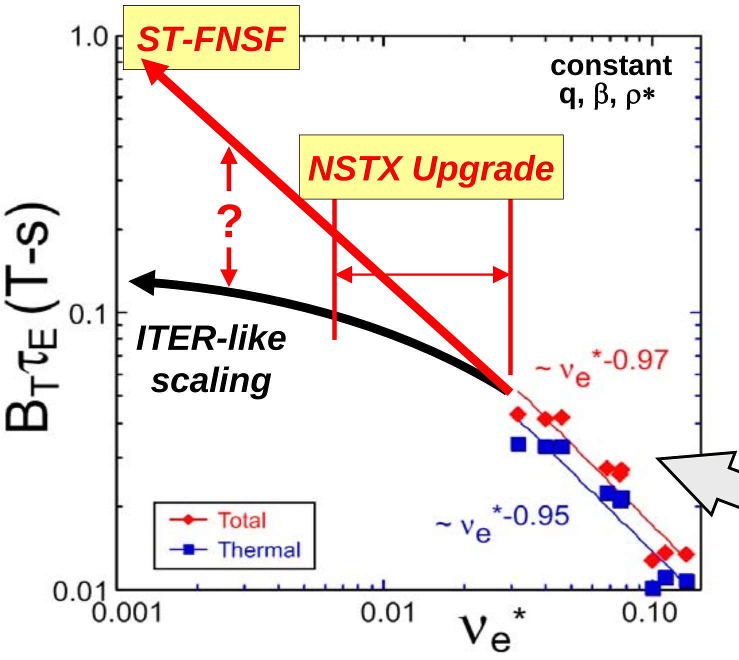


FNSF (ST-CTF)



PMIF (NHTX)

Access to reduced collisionality is needed to understand underlying causes of ST transport, scaling to next-steps



Normalized electron collisionality

$\nu_e^* \propto n_e / T_e^2$

- Future ST's are projected to operate at 10-100× lower normalized collisionality ν^*
- Conventional tokamaks observe weak inverse dependence of confinement on ν^*

ITER $B\tau_E$ (e-static g-Bohm) $\propto \rho_*^{-3}\, \beta^0\, \boxed{\nu_*^{-0.14}}\, q^{-1.7}$

Petty et al., PoP, Vol. 11 (2004)

- NSTX observes much stronger scaling vs. ν^*
  - Does favorable scaling extend to lower ν^* ?
  - What modes dominate e-transport in ST ?
    - Electrostatic or electromagnetic?

- Higher toroidal field & plasma current → higher confinement, temperature
- Higher temperature reduces collisionality, but increases equilibration time

- **Upgrade:** **Double field and current for ~3-6× decrease in collisionality**

  → require ~3-5× increase in pulse duration for profile equilibration

Increased auxiliary heating and current drive are needed to address ST start-up, sustainment, and boundary issues

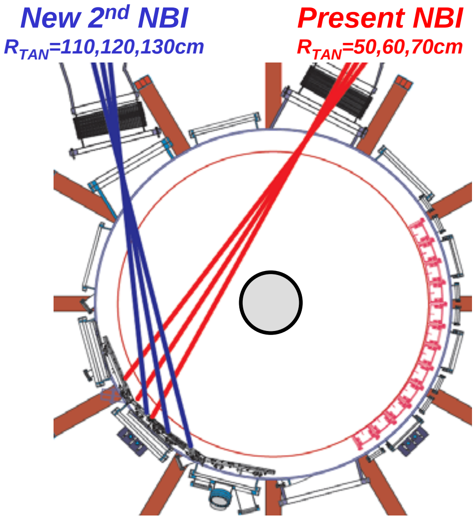
- Need additional heating power to access high temperature and β at low ν^*

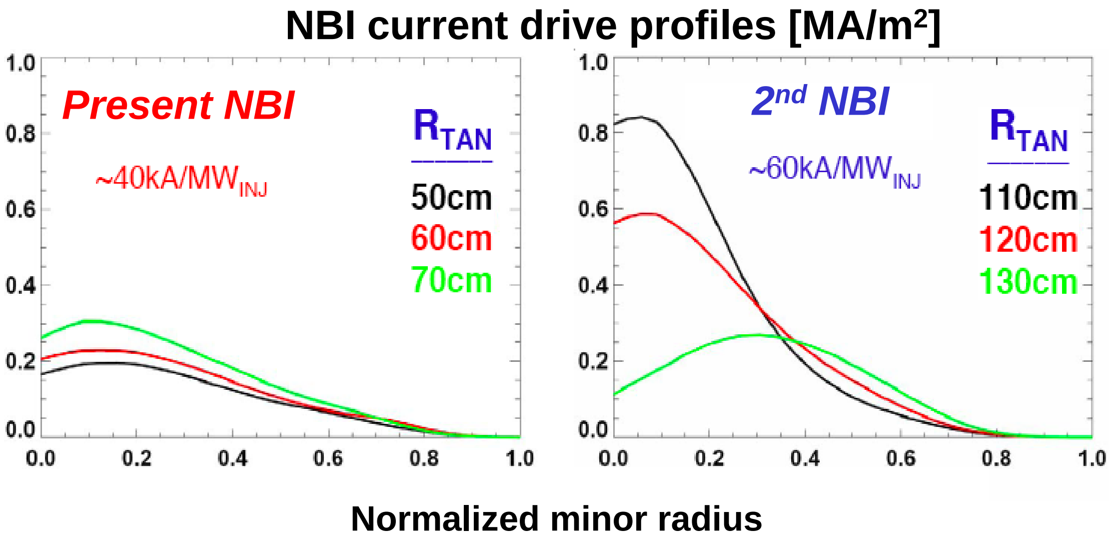
  → 4-10MW, depending on confinement scaling

- Need increased current drive to access and study 100% non-inductive

  → 0.25-0.5MA current drive compatible with ramp-up, sustainment plasmas

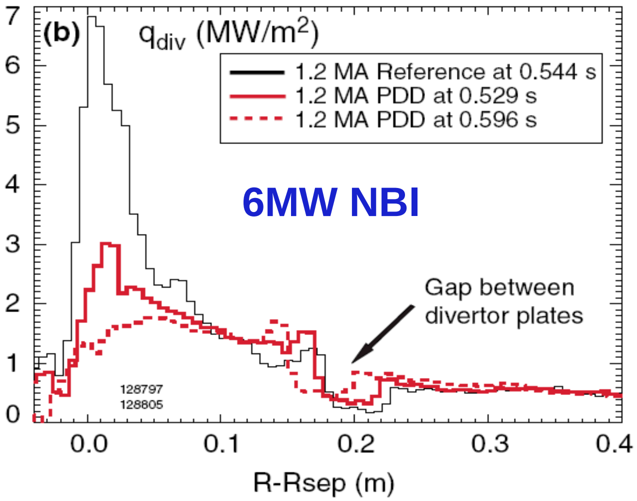
- Neutral beam injection is the only fully developed method for simultaneous heating and bulk non-inductive current drive in over-dense ST plasmas
- **Upgrade: Double neutral beam power + more tangential injection**
  - More tangential injection → up to 2 times higher efficiency, current profile control



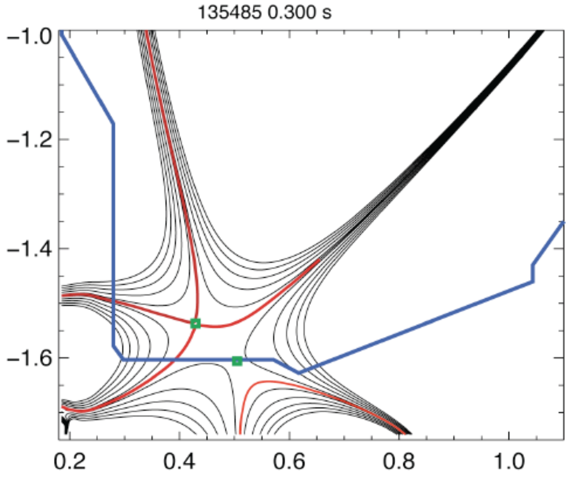


A combination of advanced PMI solutions will likely be required to manage the power exhaust of NSTX Upgrade

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



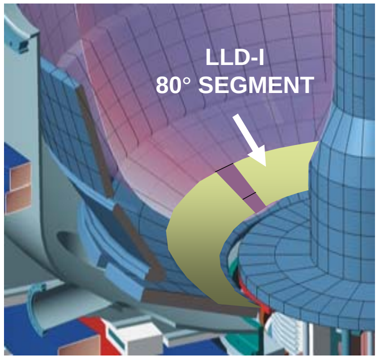
- NSTX has demonstrated formation of high flux-expansion “snow-flake” divertor



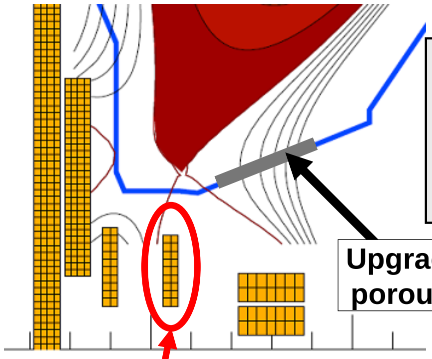
- The PDD operating regime and other PMI solutions will be challenged in NSTX-U:
  - 2-3× higher input power
  - 1.5-2× lower Greenwald density fraction
  - 3-5× longer pulse duration, leading to substantial increase in $T_{divertor}$

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

- **NSTX LLD**



- **NSTX-U “snow-flake”**



Flux expansion factor (strike/mid)

f_{exp} **= 32**

Upgraded LLD, capillary porous system, other…

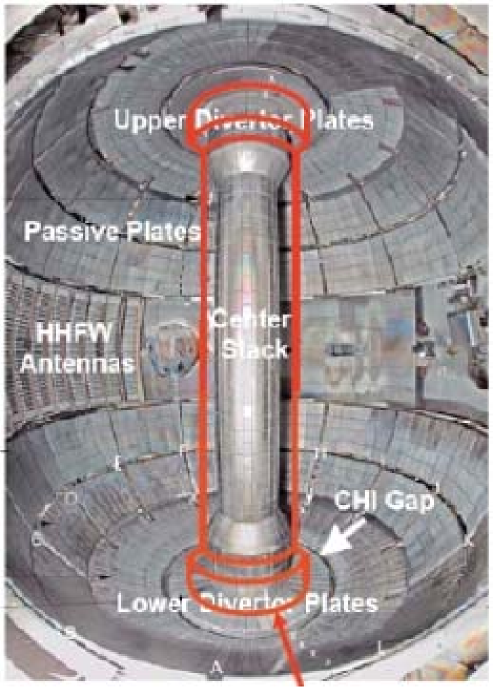
Additional divertor coils added to NSTX-U to simultaneously control strike point location and flux expansion factor

Upgrades provide major step along ST development path

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

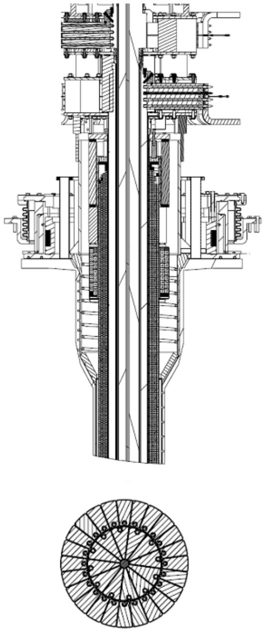
	NSTX	NSTX Upgrade	Plasma-Material Interface Facility	Fusion Nuclear Science Facility
Aspect Ratio = R_0 / a	≥ 1.3	≥ 1.5	≥ 1.7	≥ 1.5
Plasma Current (MA)	1	2	3.5	10
Toroidal Field (T)	0.5	1	2	2.5
P/R, P/S (MW/m,m^2)	10, 0.2*	20, 0.4*	40, 0.7	40-60, 0.8-1.2

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



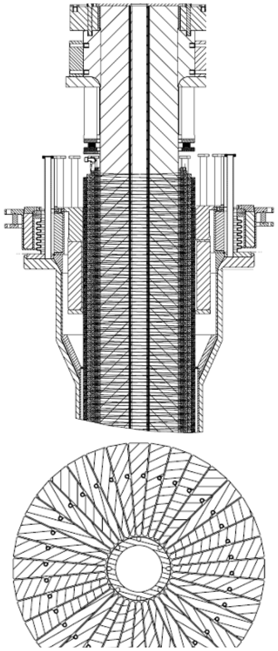
Outline of new center-stack (CS)

Present CS



TF OD = 20cm

New CS



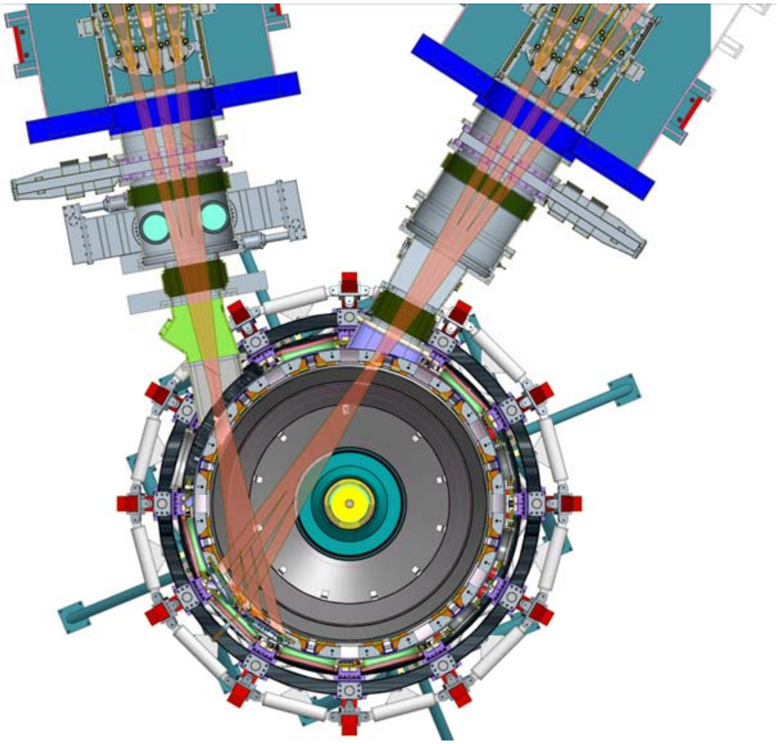
TF OD = 40cm

New 2nd NBI

(R_{TAN}=110, 120, 130cm)

Present NBI

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



Design Point Selection Process (1)

- Design point spreadsheet studies were initiated in April ‘08
- Guiding assumptions:
  1. Completely replace center stack
  2. New TF same dZ as average turn of original
  3. New OH same dZ as old
  4. Retain existing TF outer legs
  5. TF at flat top for full duration of Ip
  6. Provide OH flux sufficient for I_P ramp in 1st swing
     - Conservative $dI_P / dt = 2MA/s$ – assumes 2× higher T_e with upgrade
  7. Use OH 2nd swing as thermal/stress permits
  8. Retain existing PF outer coils
  9. Coil temperature range* 12-100C, adiabatic, allow for L/R decay
  10. Simple formulae for TF von Mise stress**, OH hoop stress*** (peak)
      - VM allowable stress 133 MPA, peak allowable stress 200MPA
  11. 1kV TF, 8kV/24kA OH, 1 MG
  12. Two TFTR NBI systems imposing MG loads

* TF adiabatic allowable slightly above 100C to account for entrained water benefit
** neglects tension due to force from outer leg
*** neglects interaction with PF coils and plasma

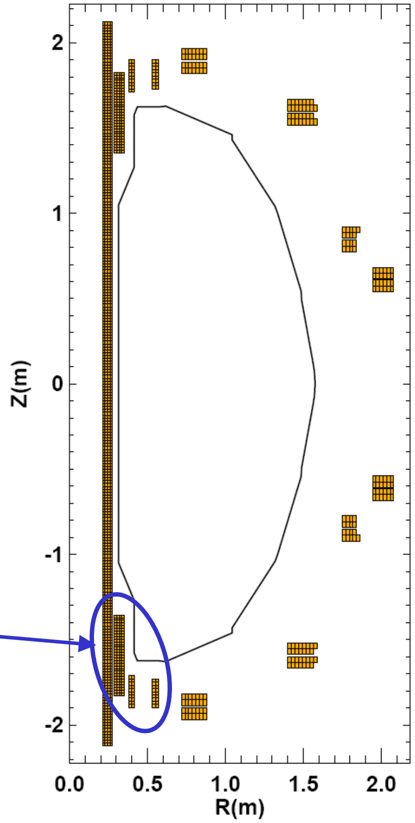
Design Point Selection Process (2)

- Spreadsheet modeling features:
  - TF and OH conductor sizing
    - Adiabatic conductor heating models (G-function)
    - Allowance for “fill factor” due to conductor cooling hole, corner radii, electrical insulation
    - Presently optimizing conductor and cooling hole size to minimize cool-down time
  - Simple formulae for TF inner leg Von Mises stress and OH peak hoop stress
    - Not included are TF inner leg torsion, TF outer leg, VV, etc.
    - Full OH Von Mises/Tresca stress calculation w/axial stress is not included
  - Full OH waveform including plasma loop voltage and flux requirement
    - Accounts for flux consumption during plasma initiation
    - Computes ramp and flat top flux using Hirshman-Neilson formulation
  - Simplified linear models for AC/DC converter behavior
  - TF and OH L-R circuit models $V = L \bullet dI/dt + I \bullet R$ w/ temperature dependent R’s
  - MG power and energy models

- XL Solver (non-linear optimizer) is used to compute design point…
  - Finds radius of TF necessary to meet B_T and pulse length requirement
  - Designs OH coil to meet flux requirement of 1st swing, maximizes 2nd swing within thermal constraints

Design Point Selection Process (3)

- Initial approach was aggressive (e.g. 2kV TF, 10kV OH, 2 MG) at A ~ 1.5-1.6 to understand possible operating envelope
  - Found that maximum usage of center stack area (based on spreadsheet analysis) would allow I_P=3 MA with 5 sec flat top at B_T=1.4T
  - But this ignored TF joint design, PF forces on TF, PF interaction forces, …
- Decided to limit to 1kV TF, 8kV OH, 1 MG → I_P=2MA, 5 flat-top, B_T = 1T and investigate design concepts in detail
  - First design point proposed in November 2008
  - Physics analysis performed to confirm assumptions
  - First "official" design point for engineering study issued 2/10/09
- Recent iterations in summer/fall 2009:
  - TF conductor details based on manufacturing considerations
  - OH coil wound directly on TF - eliminates "tension tube", gap
  - Refinements in insulation thicknesses (more conservative)
  - Refined design of inner PF coils (PF1A/B/C coils)
  - Added short pulse double swing scenario
  - Inclusion of force influence matrices and force calculations



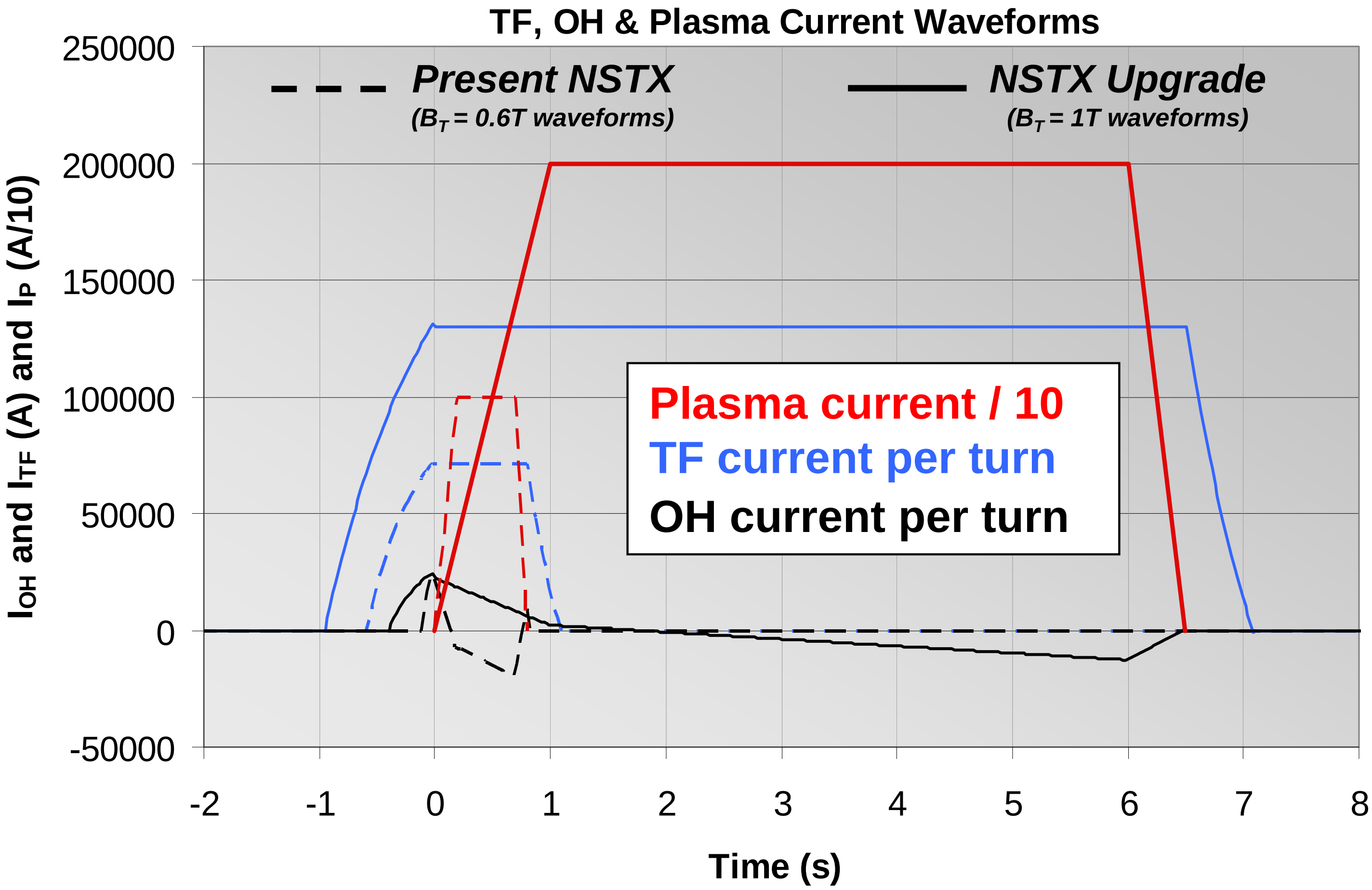
Comparison of NSTX and NSTX Upgrade parameters and waveforms:

Relative performance of Upgraded NSTX vs. Base:

- ***Center-stack radius increased 13cm → A=1.3 → 1.5***
- ***Available OH flux increased 3.5-4×, 3-5× longer flat-top***
- ***I_P increased 2×, B_T increased 2× at same major radius***
- ***But, inter-shot cool-down period increased 2 to 4-fold***

	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	2400*
$R_{center\_stack}=R_0-a$ [m]	0.185	0.315
$R_{antenna}=R_0+a$ [m]	1.574	1.574
Total OH flux [Wb]	0.75	2.8

****$T_{repetition}$ upgradable to 1200s, and OH conductor optimization could further reduce to 600-900s***



General Requirements Document (GRD) and Web-based Design Point Information for Centerstack Upgrade

- GRD was signed and issued on March 30, 2009
- Contains top level mission performance requirements
  - Includes appropriate level of specificity for mission performance
  - Refers to web-based design point data as vehicle for tracking/coordinating details subject to iteration
- Organized according to original NSTX WBS structure
  - Changes required to each WBS element are described
  - Ensures that no work scope is overlooked

- Comprehensive design point data is also maintained on web site to ensure coordination of all design activities
  - NSTX CSU design team is notified when new data is posted
  - Changes indicated in “blue, records of prior revisions maintained
- Web data contains both base NSTX with CS upgrade data
  - Useful for comparing “old” vs. “new” - MKS and English units

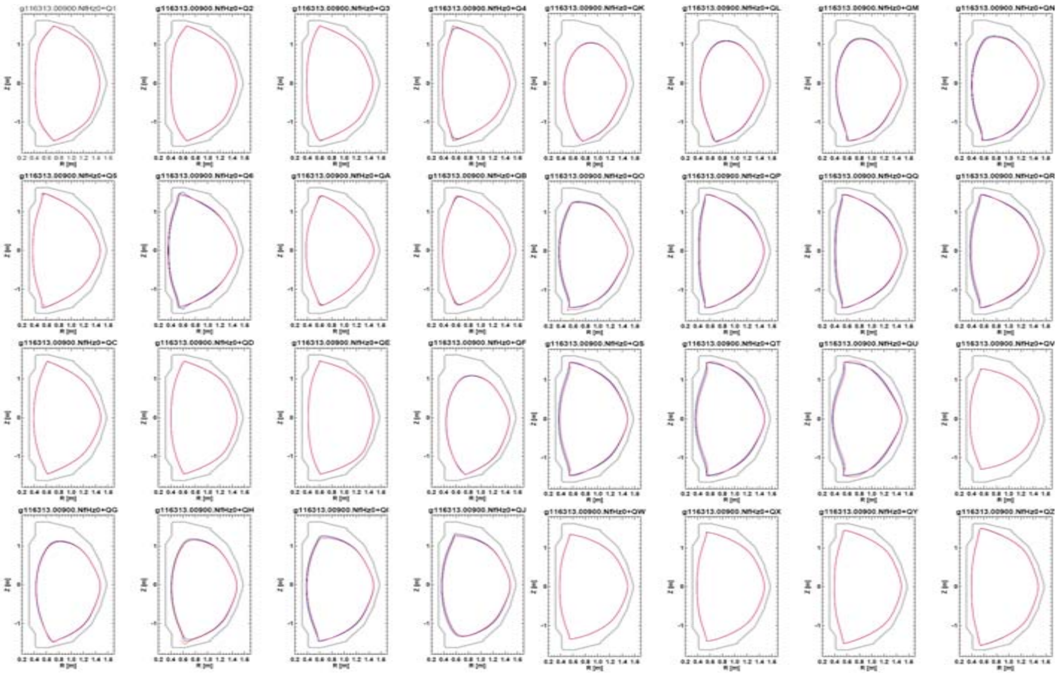
    http://www.pppl.gov/~neumeyer/NSTX_CSU/Design_Point.html

Range of I_P = 2MA free-boundary equilibria generated to enable design of TF and PF coil support structures

Free boundary equilibrium parameters:

- Aspect ratio A: 1.6 – 1.9
- Internal inductance l_i: 0.4 – 1.1
- Elongation κ: 2.1 – 2.9
- Triangularity δ: 0.2 – 0.7
- Squareness ζ: -0.15 – 0.12
- Magnetic balance: -1.5 – 0cm
- I_{OH}: zero and +/- supply limit
  - For computing PF needed for cancellation of OH leakage flux
- Pressure variation: β_N = 1, 5, 8

32 free boundary equilibria × 3 OH conditions = 96 cases



- NOTE: Negative "squareness" boundary shape cases are included:
  - More shaping flexibility/capability than in present NSTX (requires PF4 usage)
  - Expect could be important for controlling edge stability (NSTX will test in FY2010)
  - For conservative (worst-case) power-supply fault conditions, would require substantial inter-coil PF support structure, which could be challenging
  - With coil/machine protection system and nominal operating currents, preliminary analysis indicates simplified support structure is feasible

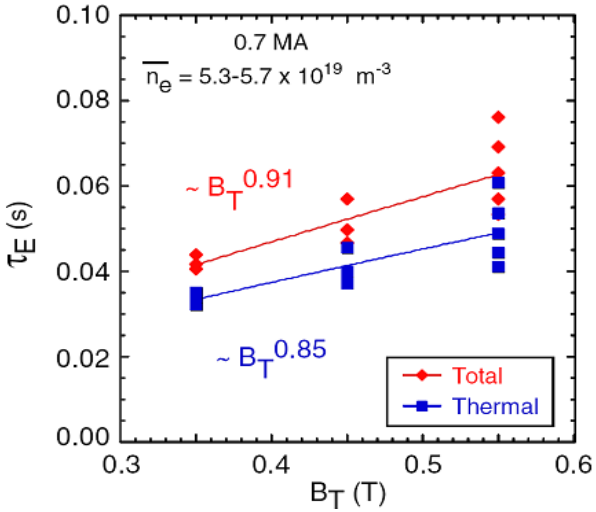
Summary: NSTX Upgrades will greatly expand the research capabilities of NSTX and narrow key gaps to future STs/tokamaks

- Design doubles B_T, I_P, P_{NBI}, and extends pulse 3-5× while increasing divertor flexibility for power exhaust
  - Access and understand impact of reduced collisionality
  - Access fully non-inductive ramp-up and sustainment
  - Assess plasma-material interface solutions for FNSF/Demo

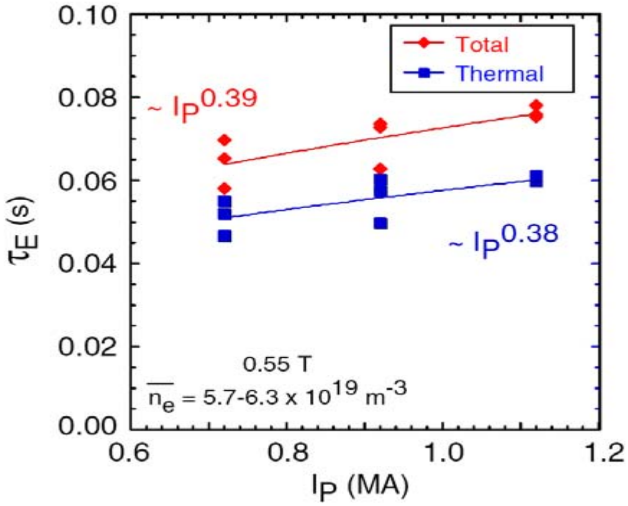
- Design point is feasible from engineering standpoint
  - Increased loads on vessel and coils from increased fields (and shaping flexibility) can be accommodated with enhanced support structures and coil protection system
  - See subsequent presentations

CSU / Confinement Backup Slides

Access to higher field and current is needed to understand scaling of ST confinement, implications for next-steps



NSTX Data



- NSTX (and MAST) energy confinement time τ_E scales much more strongly with magnetic field and more weakly with current than ITER scaling

ST H-mode: $\tau_E \propto B_T^{1.2} I_P^{0.6} n^{0.2} P^{-0.6}$ **ITER H-mode: $\tau_E \propto B_T^{0.15} I_P^{0.9} n^{0.4} P^{-0.7}$**

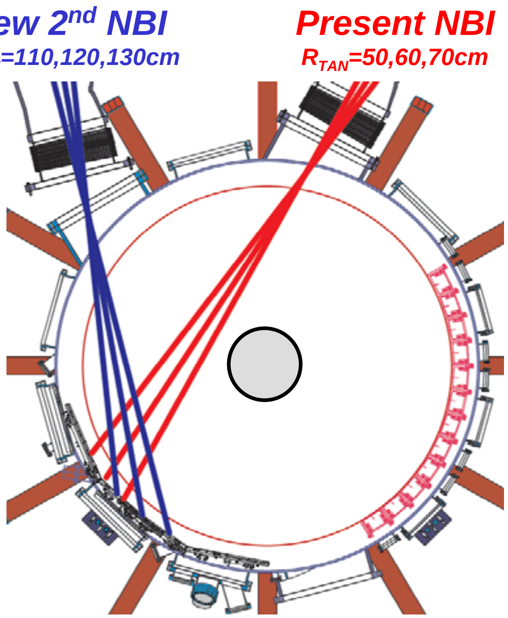
- For scaling from NSTX to NSTX-U assume:
  - $n / n_{\text{Greenwald}}$ decreases 30% (~1 → ~0.7) via planned density control
  - Toroidal, normalized beta held ~constant: increase -20% (ITER) to +10% (ST)

- **To achieve: 3-6× reduction in collisionality →**
  - Field and current must double, heating power P = 6MW increases to 10-16MW
  - Also require 3-5× increase in pulse duration for profile equilibration

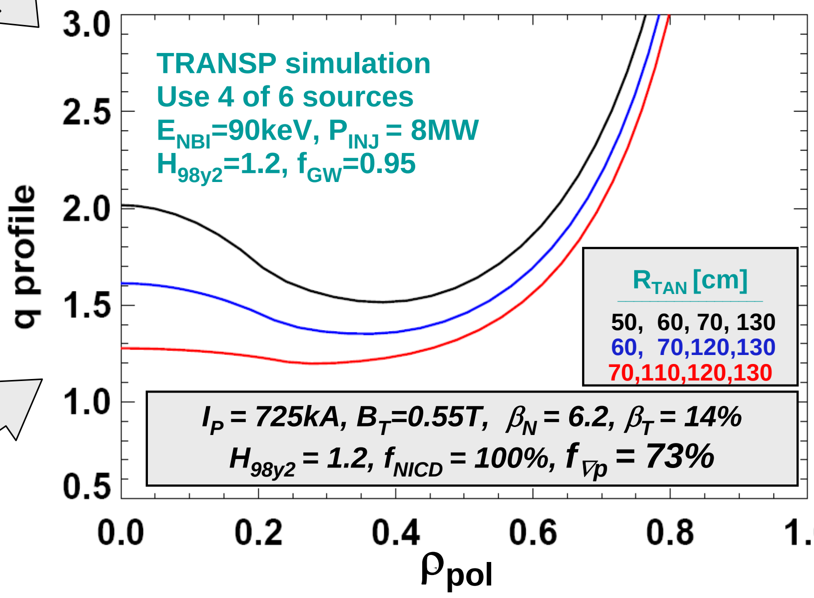
NBI Backup Slides

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

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



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

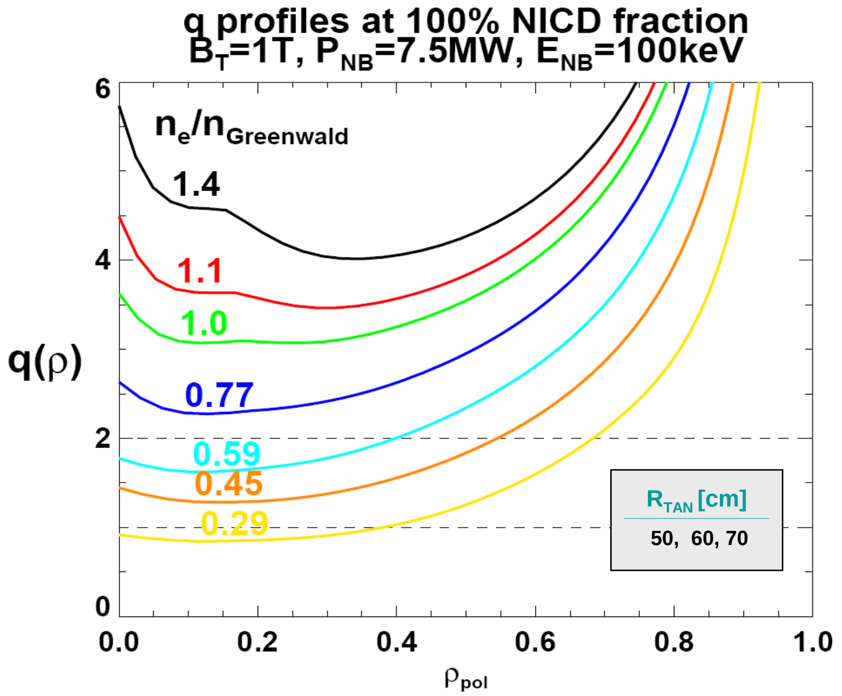


- q(r) profile variation and control very important for global stability, electron transport, Alfvénic instability behavior

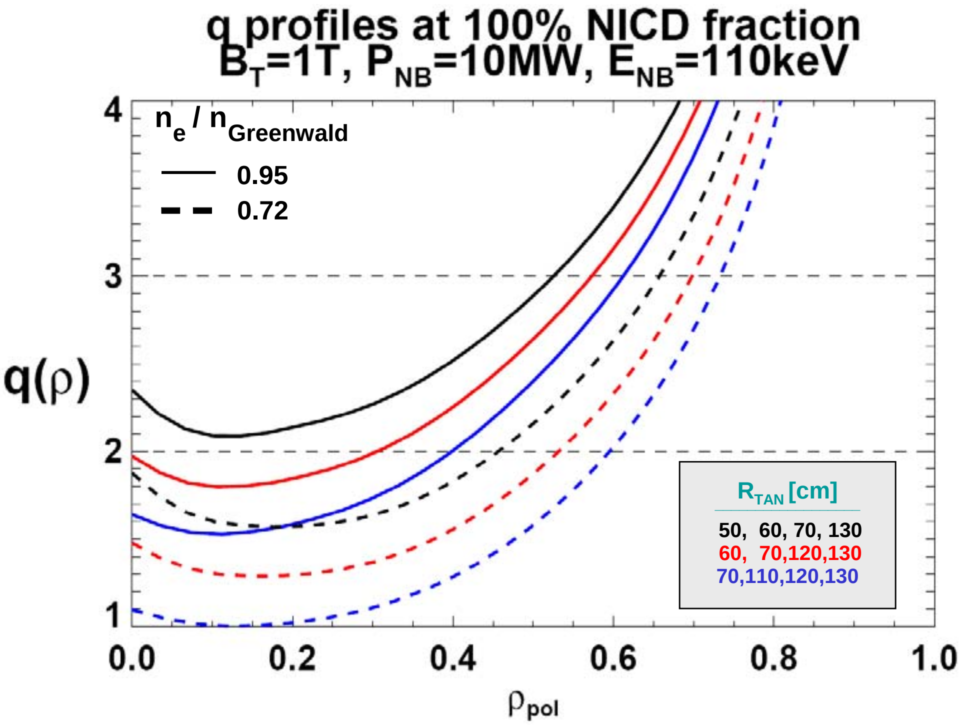
Higher field B_T=1T from new CS + 2nd NBI would enable access to wide range of 100% non-inductive scenarios

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

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


q profiles at 100% NICD fraction
B_T=1T, P_{NB}=7.5MW, E_{NB}=100keV

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


q profiles at 100% NICD fraction
B_T=1T, P_{NB}=10MW, E_{NB}=110keV

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

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, …

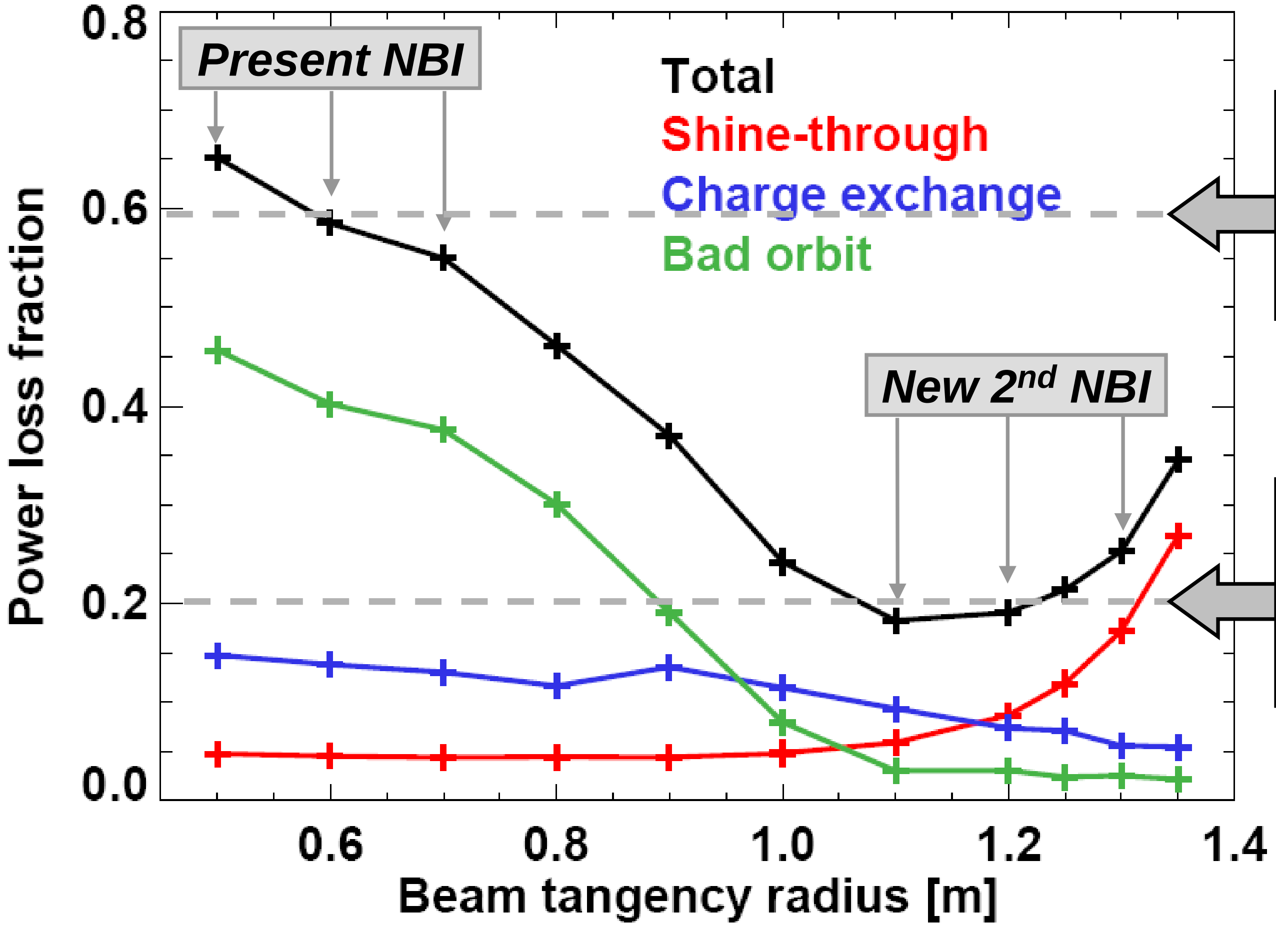
- I_P =1.6-2MA and B_T = 1T partially-inductively driven scenarios identified:
  - f_{NICD} = 50-65% with **q_{min} > 1**, β_N = 4-5, NBI profile computed with TRANSP
    - Similar to present high NI-fraction discharges, but with 2× field and current
  - These scenarios also require $\geq$ 8MW of NBI heating power for $H_{98} \leq 1.2$

- New solenoid can support 2MA plasmas for 5s (flat-top $\Delta\Phi_{OH} \sim$ 1Vs)

For NBI I_P ramp-up, more tangential 2nd NBI has 3× lower power loss than present NBI at low I_P = 400kA

E_{NBI}=80keV, I_P=0.40MA, f_{GW}=0.62

$\overline{n_e} = 2.5\text{x}10^{19}\text{m}^{-3}$, $\overline{T_e} = 0.83\text{keV}$

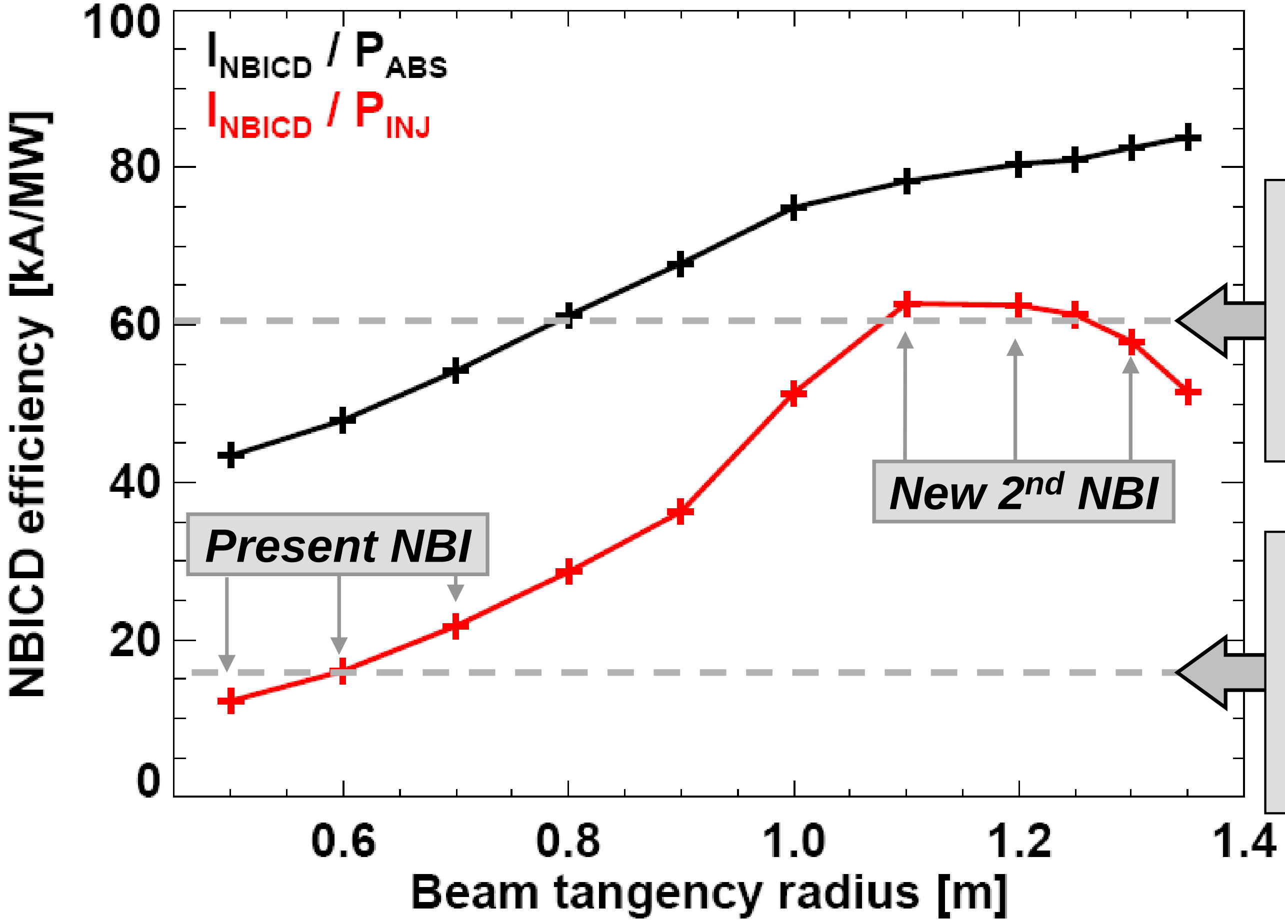


- 60% of power of present NBI lost
  - Absorb 2 of 5 MW (80kV)

- 20% of power of new 2nd NBI lost
  - Absorb 4 of 5 MW (80kV)

→ *2nd NBI can efficiently heat 400kA HHFW-driven ramp-up plasma*

For NBI I_P ramp-up, more tangential 2nd NBI has 4× higher NBI-CD than present NBI at low I_P = 400kA



E_{NBI}=100keV, I_P=0.40MA, f_{GW}=0.62

$\overline{n}_e = 2.5\times10^{19} m^{-3}$, $\overline{T}_e$ = 0.83keV

- 2nd NBI → 60kA/MW current drive efficiency
  - 450kA CD for 7.5MW injected at E=100keV

- Present → 15kA/MW current drive efficiency
  - 110kA CD for 7.5MW injected at E=100keV

→ 2nd NBI can provide sufficient current for ramp-up to ~800kA

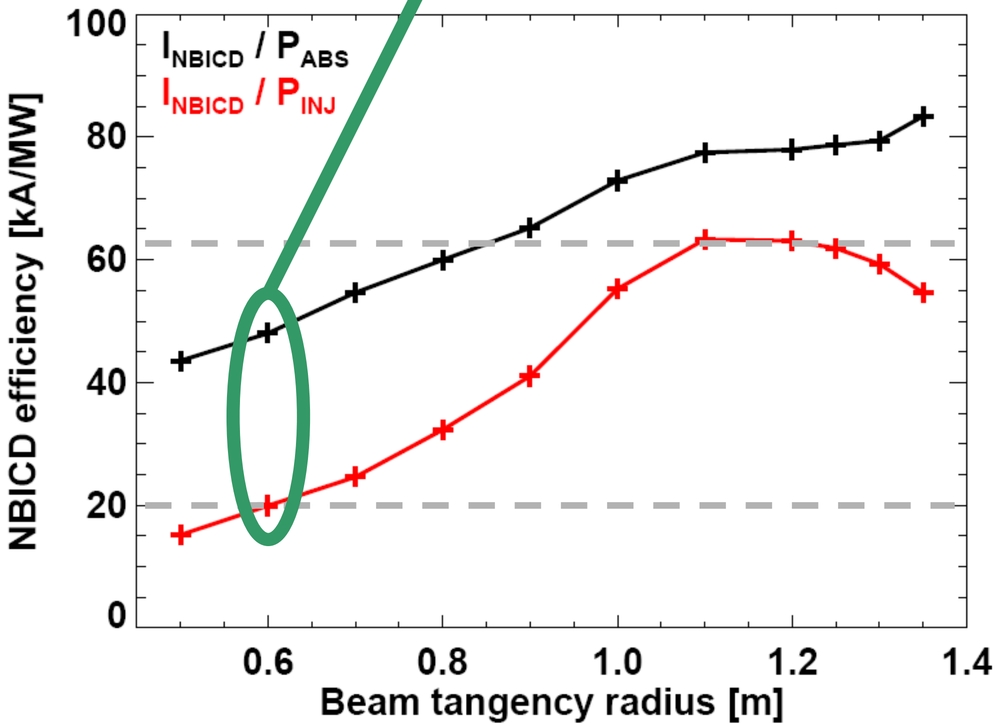
For NBI I_P ramp-up, absorbed fraction and CD of *present* NBI increases by factor of 1.7 for plasma current = 400kA→600kA

- I_P = 400kA, present NBI:
  - 60% loss, 20kA/MW

- I_P = 600kA, present NBI:
  - 32% loss, 33kA/MW

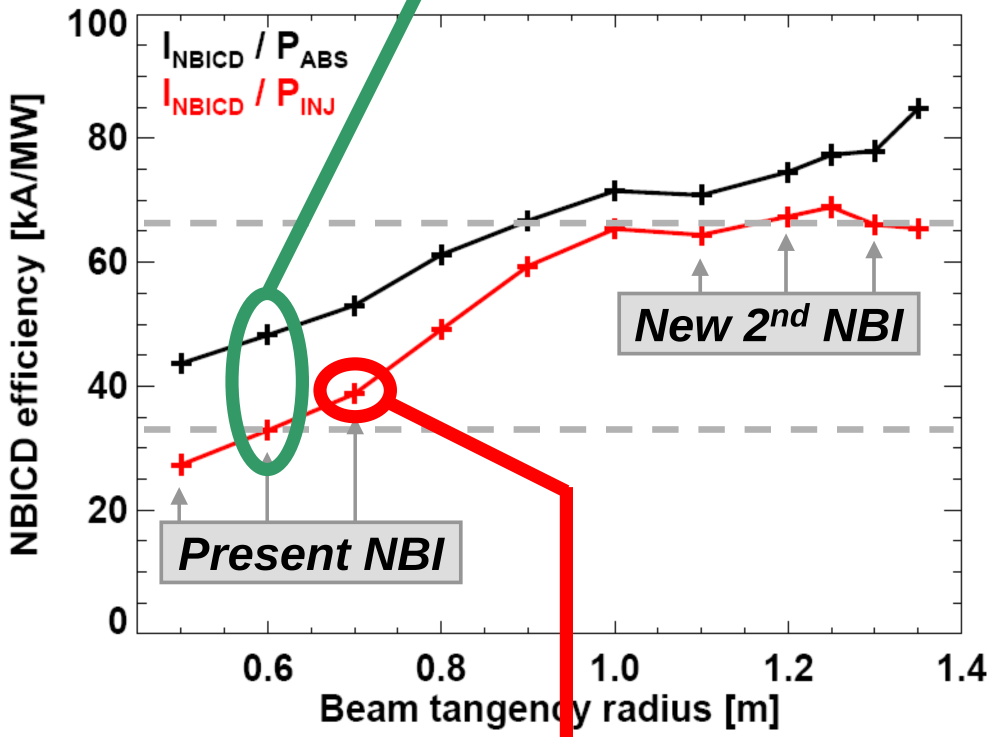
E_{NBI}=80keV, I_P=0.40MA, f_{GW}=0.62

$\overline{n_e}$ = 2.5x$10^{19}$$m^{-3}$, $\overline{T_e}$ = 0.83keV



E_{NBI}=80keV, I_P=0.60MA, f_{GW}=0.62

$\overline{n_e}$ = 3.6x$10^{19}$$m^{-3}$, $\overline{T_e}$ = 1.2keV



Most tangential of present sources has > 70% absorption for $I_P \geq$ 600kA and would be the most effective of the present sources for ramp-up

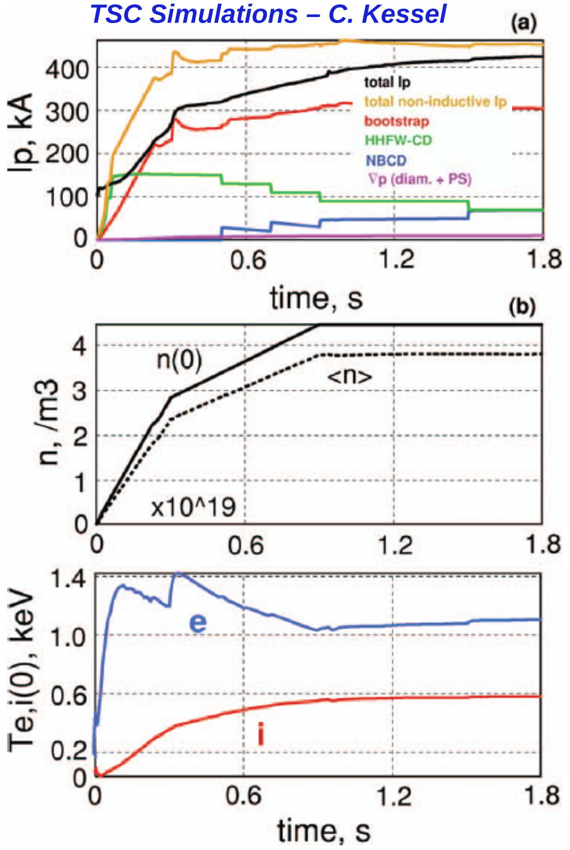
Non-inductive ramp-up to ~0.4MA possible with RF + new CS, ramp-up to ~1MA possible with new CS + more tangential 2nd NBI

Ramp to ~0.4MA with fast wave heating:

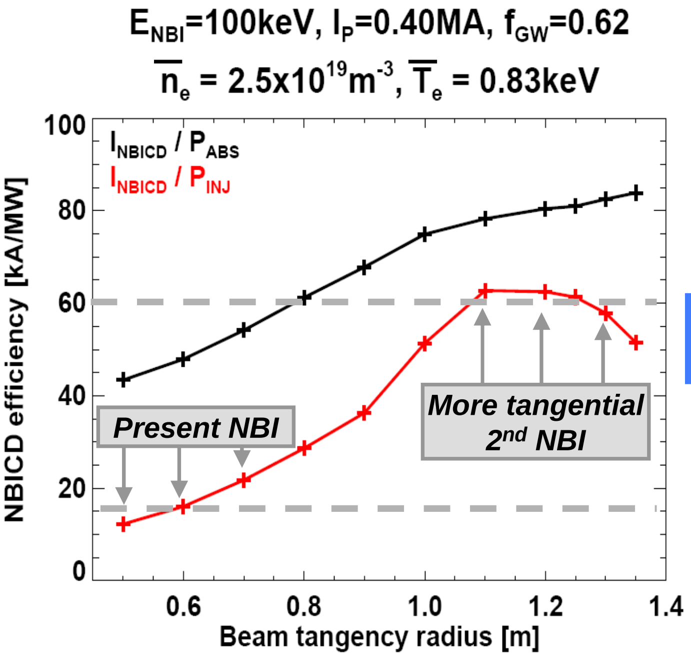
- High field ≥ 0.5T needed for efficient RF heating
- ~2s duration needed for ramp-up equilibration
- Higher field 0.5→1T projected to increase electron temperature and bootstrap current fraction

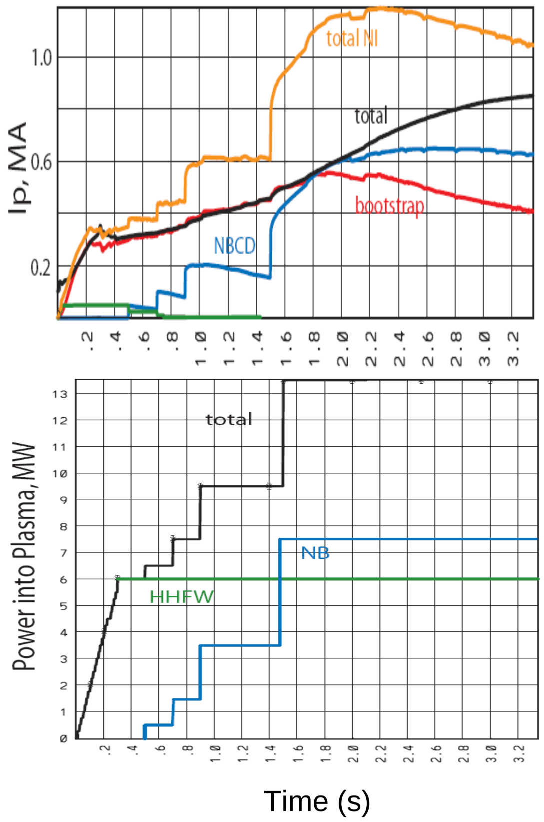
Extend ramp to 0.8-1MA with 2nd NBI:

- 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



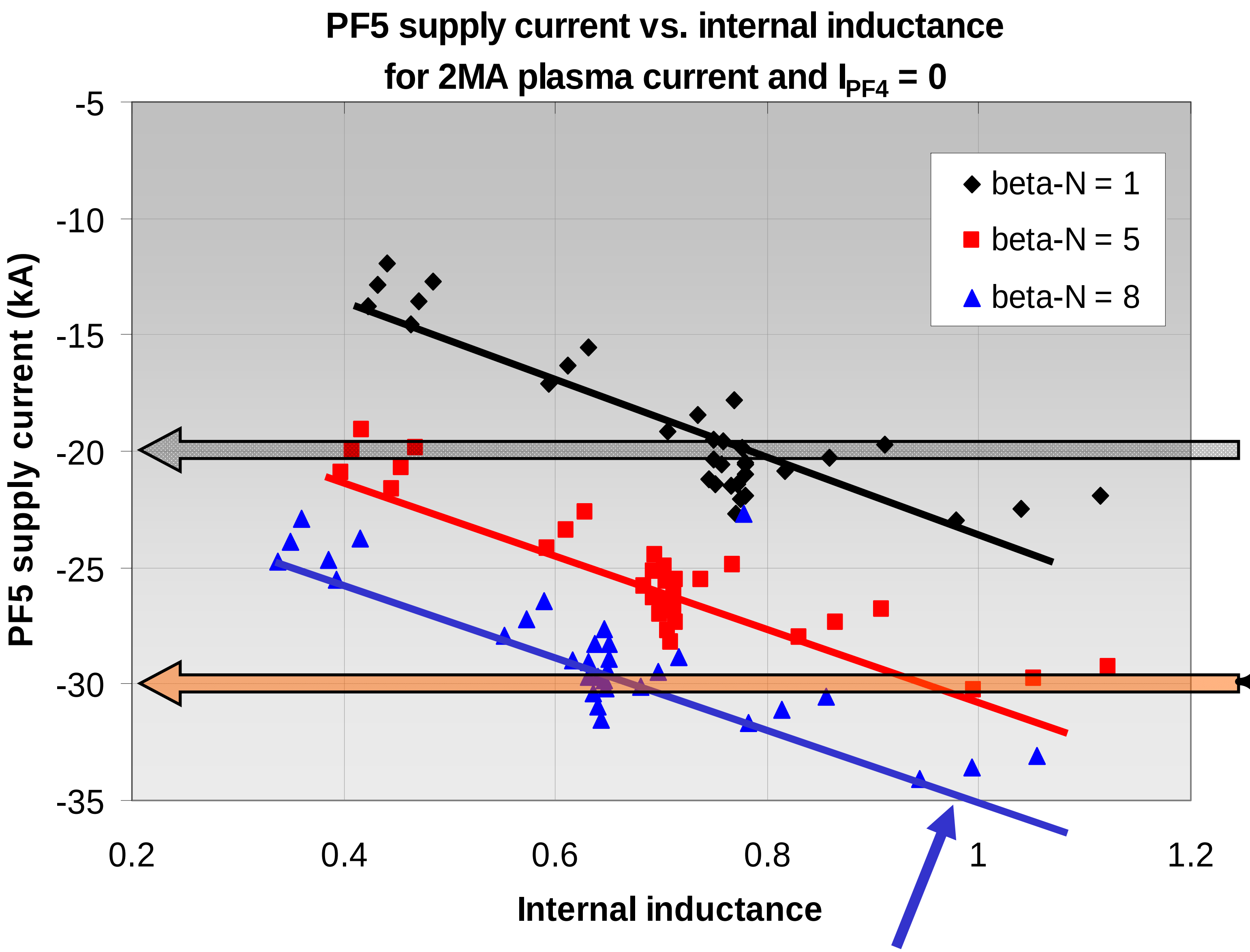
TSC Simulations – C. Kessel





PF Coil Design Backup Slides

Vertical field upgrade being designed to support $\beta_N = 5, l_i \le 1$ and $\beta_N = 8, l_i \le 0.6$ at I_P = 2MA



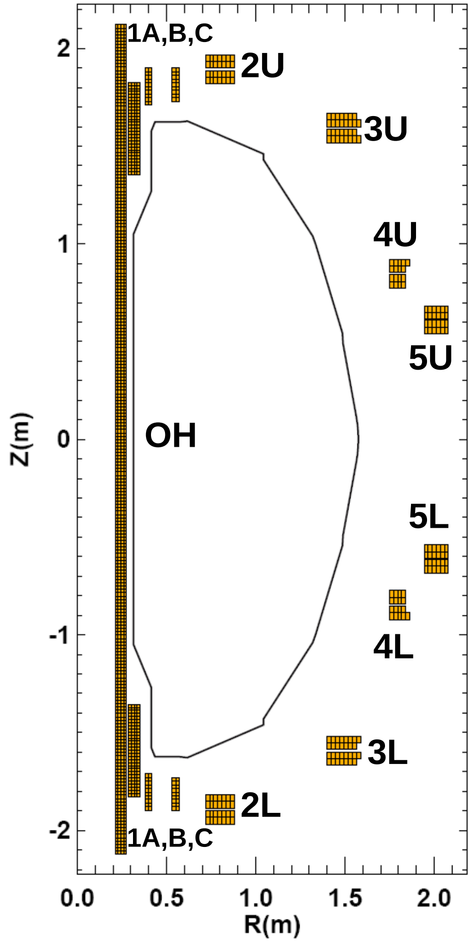
Present PF5 limit

Upgrade design targets $I_{PF5} \le$ 30kA with 10% current margin for radial position control

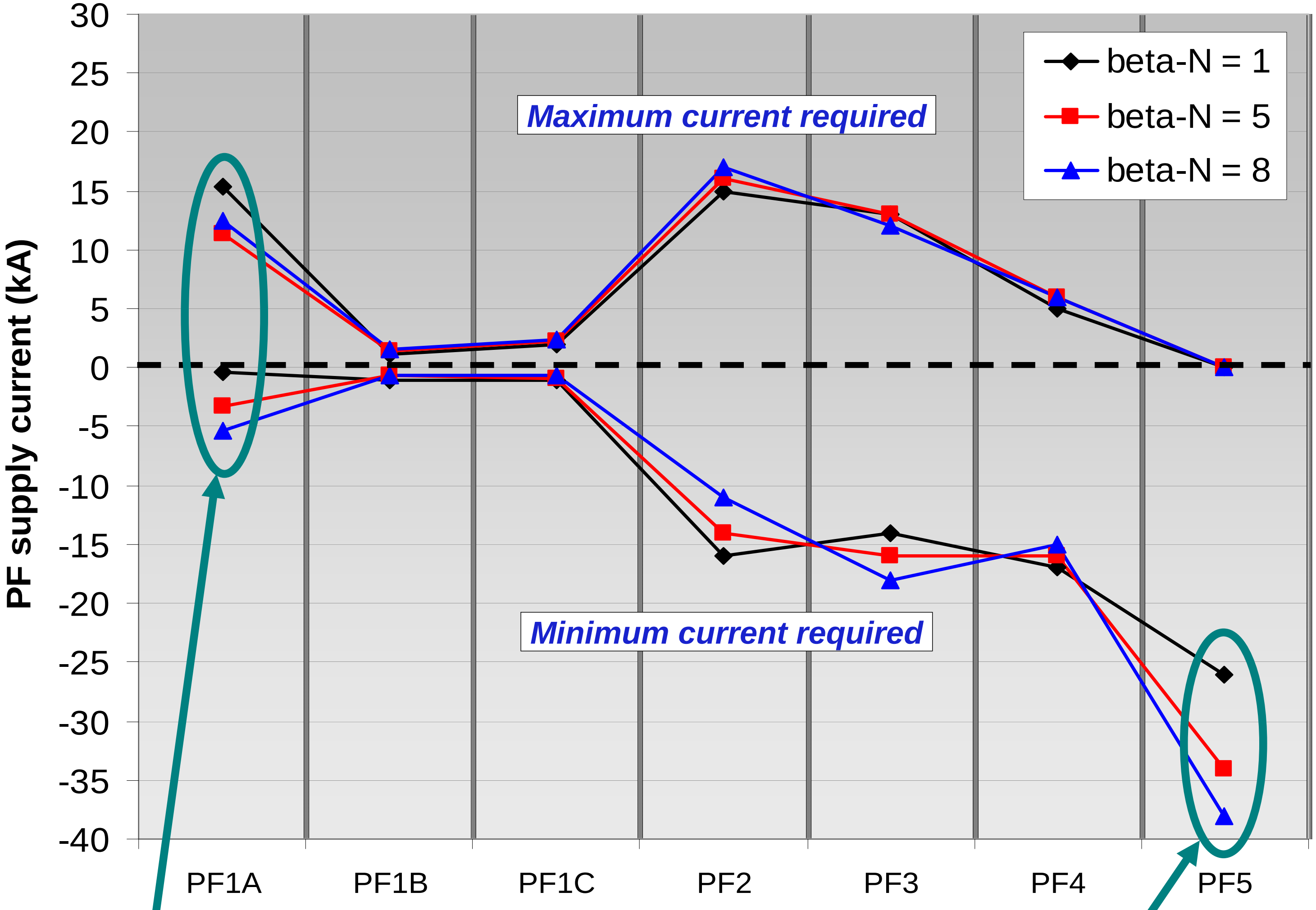
High l_i, high-β_N scenarios determine maximum PF5 current required

High β_N increases vertical field requirement, and shifts primary divertor coil (PF1A) current requirement to bipolar

Note: all current limits are 10% above current required for actual equilibrium



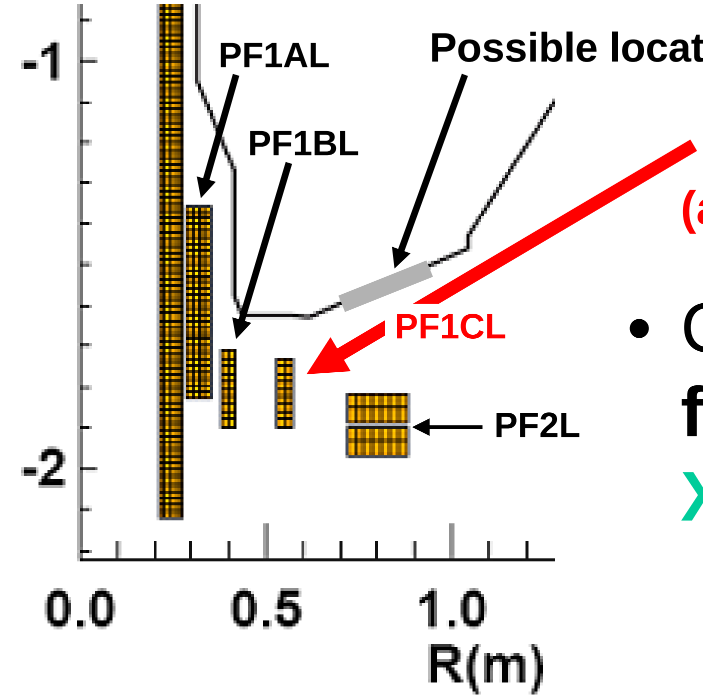
PF supply limits - 2MA, expected OH operation, full PF



Primary divertor field (PF1A) requires -5kA reduction at high β_N

Vertical field (PF5) required increases ~50% from low to high β_N

The divertor PF coil system for NSTX Upgrade includes an additional coil to enhance control of power exhaust

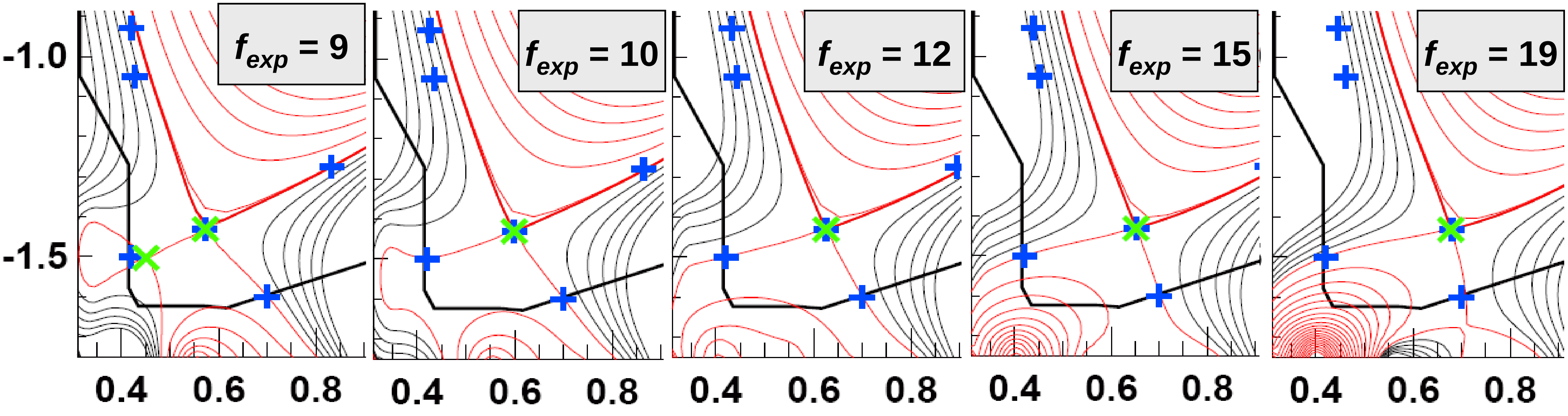


Additional coil added = PF1CL

(and added 2 upper coils PF1BU, PF1CU for U/L symmetry)

- Combination of PF1A,B,C + PF2 enables **flux expansion variation** with fixed X-point height and strike-point location:

Outboard poloidal flux expansion factor $f_{exp} \equiv |\nabla\psi|_{\text{mid-plane}} / |\nabla\psi|_{\text{strike-point}}$



PMI Backup Slides

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