ST-FNSF Mission and Performance Dependence on Size*

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Abstract

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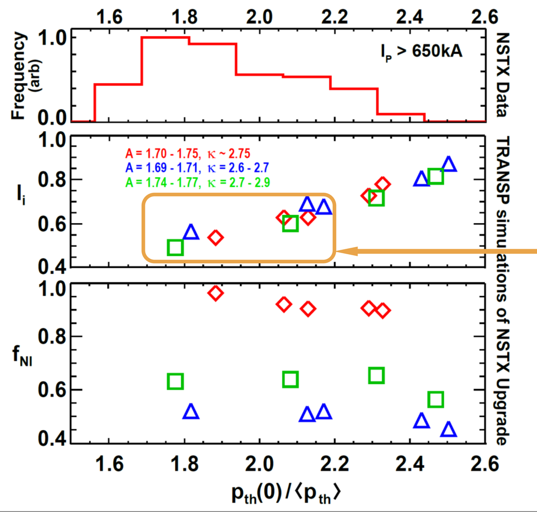
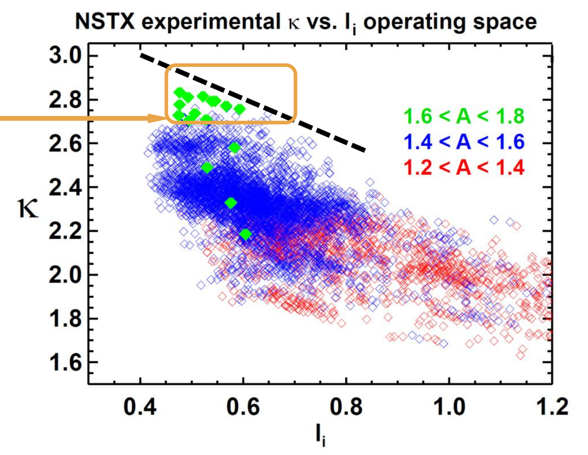
A Fusion Nuclear Science Facility (FNSF) could play an important role in the development of fusion energy by providing the high neutron flux and fluence environment needed to develop fusion materials and components. The spherical tokamak (ST) is a leading candidate for a FNSF due to its potential for generating high neutron wall loading in a small major radius device. Previous studies have identified key research needs and design issues for ST-based FNSF devices and have motivated additional studies of the impact of device size on neutron wall loading, tritium breeding, and electricity production. For example, for an ST-FNSF with A=1.7, κ=3, B_T=3T, 500keV NNBI for heating and current drive, H_{98}=1.2, $f_{Greenwald}$ =0.8, and constrained to have average neutron wall loading of 1MW/m^2, as the plasma major radius R is increased from 1m to 2.2m, the impact is stabilizing, since β_T = 19 → 14%, β_N = 4.5 → 3.8, and q* = 3.5→4.2. However, the overall fusion power = 60MW → 300MW, the tritium consumption also therefore increases by a factor of 5, and the electric power consumed increases from 350MW → 500MW. With respect to higher performance operation targeting net electricity production with fixed B_T=2.6T, H_{98}=1.5, β_N = 6, β_T = 35%, and q* = 2.5, as R is increased from 1m to 2.2m the smallest possible ST device that can achieve electricity break-even (Q_{eng}=1) has R=1.6m assuming very high blanket thermal conversion efficiency η_{th} = 0.59. For η_{th} = 0.45, the device size increases to R=1.9-2m, and still larger devices are required for lower η_{th}. A key issue under study is the impact of device size on tritium breeding ratio (TBR) where smaller devices will likely have more difficulty achieving TBR > 1 since a higher fraction of in-vessel surface area must be dedicated to auxiliary heating ports and blanket test modules. Initial calculations for a R_0 = 1.6m ST-FNSF with Dual Cooled Lithium Lead (DCLL) blankets and wall penetrations for NBI heating indicate TBR near 1 is achievable. The divertor region is also a critical and challenging area. For the ST-FNSF configurations considered here, the divertor Cu PF coils are placed in the ends of the center-stack to enable high-triangularity plasma shapes compatible with demountable TF legs and a removable center-stack. Conventional and high-flux-expansion "snowflake" divertor configurations designed for mitigating high heat fluxes have also been generated, and neutron shielding calculations for the PF coils indicate that ceramic insulators (such as MgO) would be required.

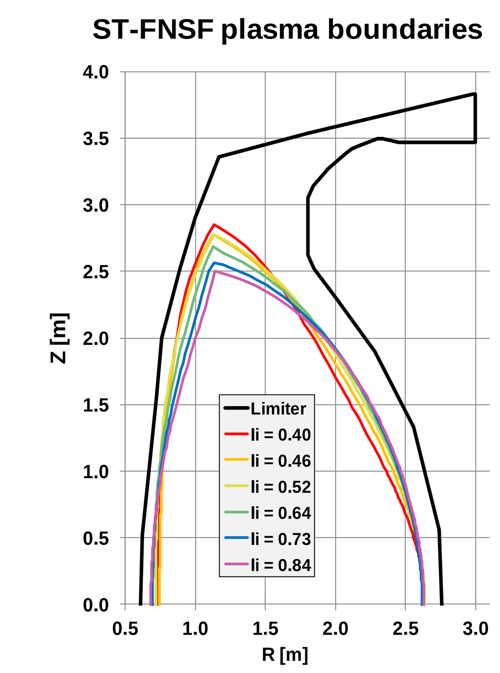
ST-FNSF equilibrium inductance, elongation based on values achieved/anticipated in NSTX/NSTX-U

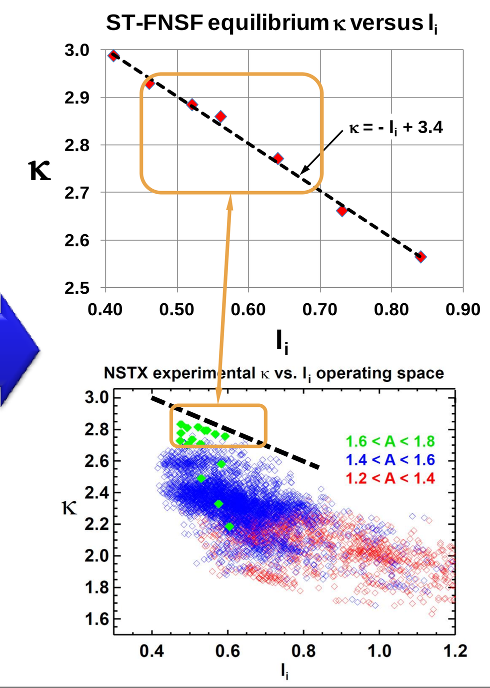
- Most probable NSTX thermal pressure peaking ~ 1.7 – 2.2
  - If similar in NSTX-U/FNSF → full non-inductive l_i ~ 0.45 – 0.7 (BS + NBI)
- NSTX A=1.7, l_i = 0.45 – 0.7 plasmas can operate stably at κ ~ 2.7 – 2.9
  - Expect to improve n=0 control in NSTX-U
  - Anticipate κ → 3 possible in NSTX-U/FNSF



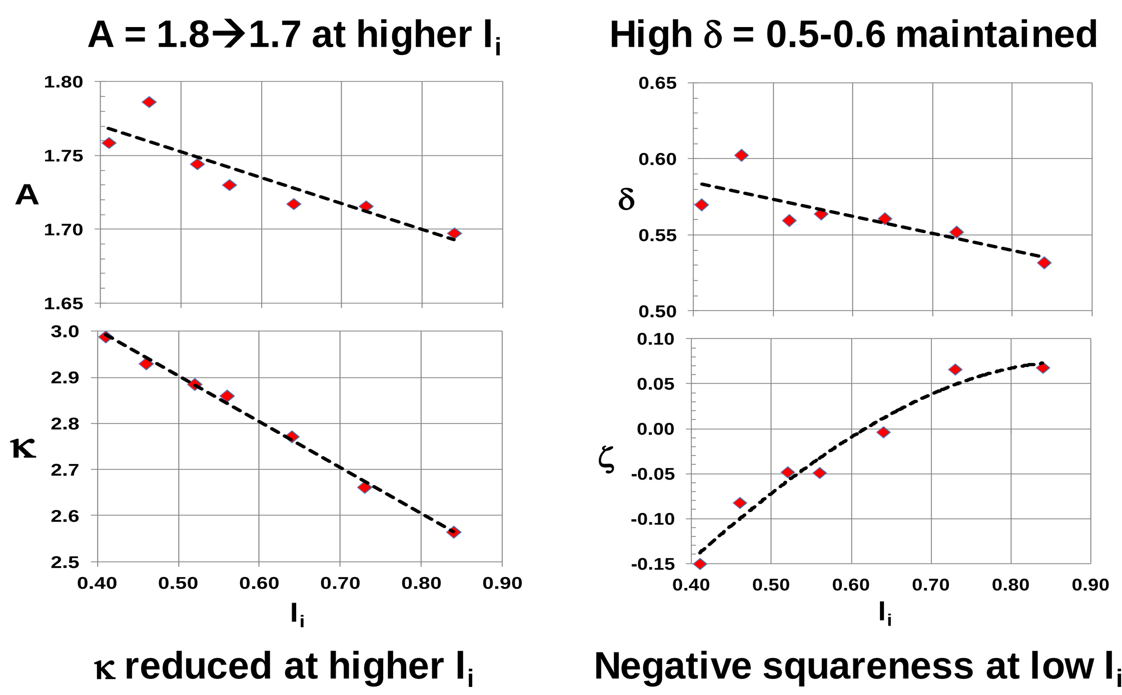


ST-FNSF free-boundary elongation is reduced with increasing l_i to match NSTX/NSTX-U trends

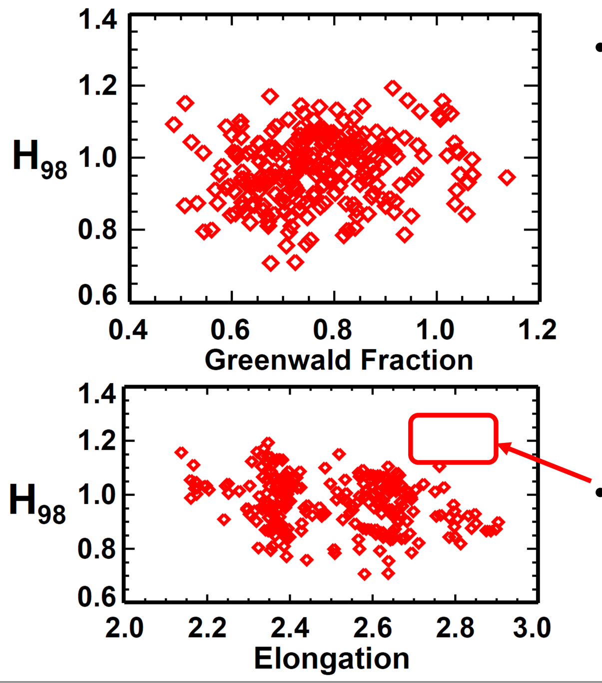




Boundary shape parameters vs. internal inductance

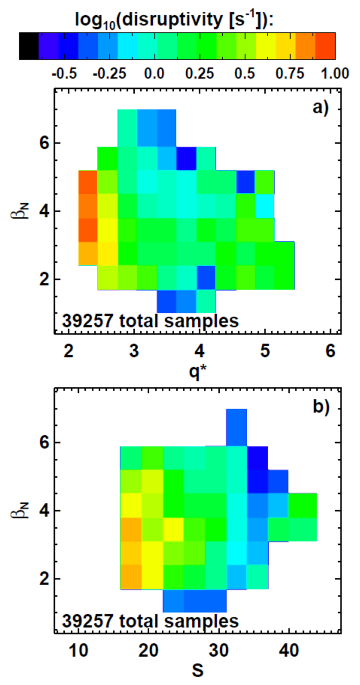


ST-FNSF operating point of $f_{Greenwald}$ = 0.8, $H_{98y,2}$=1.2 chosen to be at/near values anticipated for NSTX-U



- $H_{98y,2}$ → 1.2 accessed for a range of Greenwald fractions in NSTX
  - However, much more research needs to be carried out in NSTX-U to determine if H = 1.2 can be achieved reliably
  - Note: $H_{98y,2}$ ~ 1 would require much higher P_{aux} (~1.8×)
- Need to assess feasibility of access to $H_{98y,2}$ ~ 1.2 at κ ~ 2.7-2.9 in NSTX-U

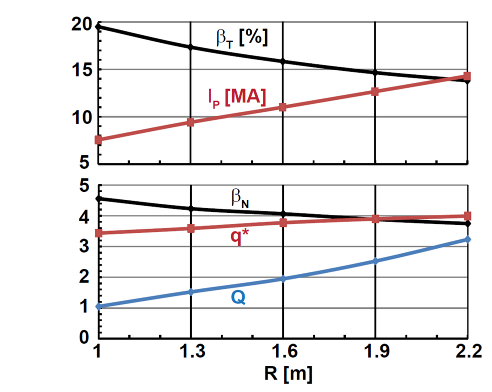
NSTX disruptivity data informs FNSF operating point with respect to global stability

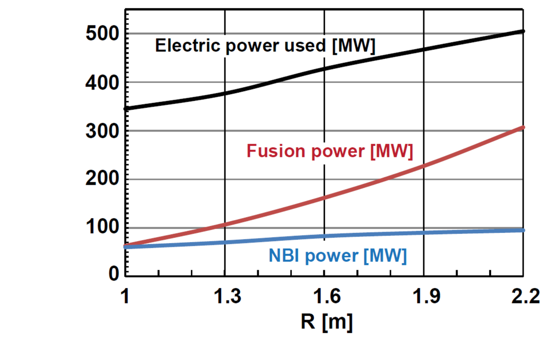


- Increased disruptivity for q* < 2.7
  - Significantly increased for q* < 2.5
- Lower disruptivity for β_N = 4-6 compared to lower β_N
  - Higher β_N increases f_{BS}, broadens J profile, elevates q_{min}
  - Operation above no-wall limit aided by:
    - NBI co-rotation
    - Close-fitting conducting wall
    - Active error-field and RWM control
- Strong shaping also important
  - S ≡ q_{95} I_P/aB_T
  - S > 30 provides strongest stabilization
  - S > 22-25 good stability
  - S < 22 unfavorable

Increased device size provides modest increase in stability, but significantly increases T consumption

- Scan R = 1m → 2.2m (smallest FNSF → pilot plant with Q_{eng} ~ 1)
- Fixed average neutron wall loading = 1MW/m^2
- B_T = 3T, A=1.7, κ=3, H_{98} = 1.2, $f_{Greenwald}$ = 0.8
- 100% non-inductive: f_{BS} = 75-85% + NNBI-CD (E_{NBI}=0.5MeV JT60-SA design)





- Larger R lowers β_T & β_N, increases q*
- Comparable/higher β_T and β_N values already sustained in NSTX

- Q = 1 → 3, P_{fusion} = 60MW → 300MW
  → 5× increase in T consumption

Beyond neutron wall loading and T breeding, FNSF study is also tracking electrical efficiency Q_{eng}

$$Q_{eng} = \frac{\text{Electricity produced}}{\text{Electricity consumed}} = \frac{\eta_{th}(M_n P_n + P_\alpha + P_{aux} + P_{pump})}{\frac{P_{aux}}{\eta_{aux}} + P_{pump} + P_{sub} + P_{coils} + P_{control}}$$

$$Q_{eng} = \frac{\eta_{th}\eta_{aux}Q(4M_n + 1 + 5/Q + 5P_{pump}/P_{fus})}{5(1+\eta_{aux}QP_{extra}/P_{fus})}$$

η_{th}	=	thermal conversion efficiency
η_{aux}	=	injected power wall plug efficiency
Q	=	fusion power / auxiliary power
M_n	=	neutron energy multiplier
P_n	=	neutron power from fusion
P_α	=	alpha power from fusion
P_{aux}	=	injected power (heat + CD + control)
P_{pump}	=	coolant pumping power
P_{sub}	=	subsystems power
P_{coils}	=	power lost in coils (Cu)
$P_{control}$	=	power used in plasma or plant control that is not included in P_{inj}
P_{extra}	=	P_{pump} + P_{sub} + P_{coils} + $P_{control}$

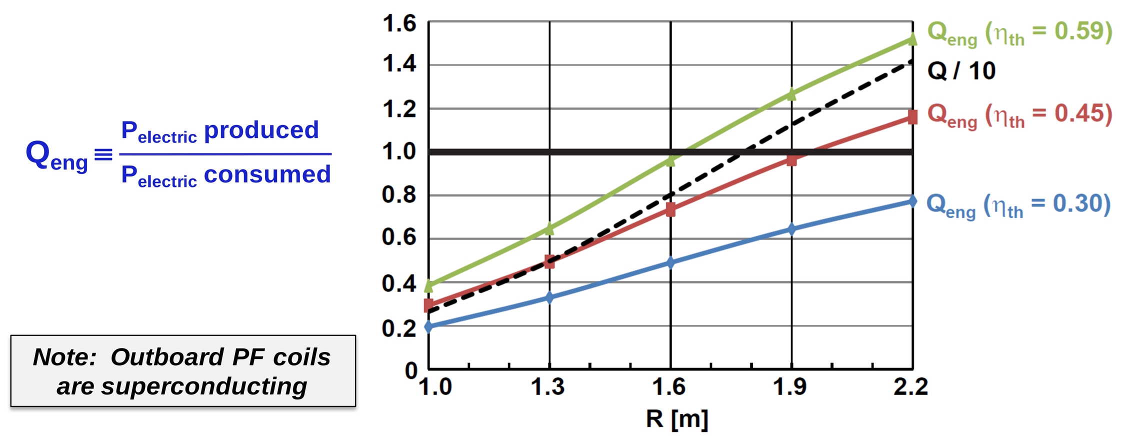
Note: blanket and auxiliary heating and current-drive efficiency + fusion gain largely determine Q_{eng}

FNSF assumptions (from Pilot study):
- M_n = 1.1
- P_{pump} = 0.03×P_{th}
- P_{sub} + $P_{control}$ = 0.04×P_{th}
- η_{aux} = 0.4 (presently unrealistically high)
- η_{CD} = $I_{CD}R_0n_e/P_{CD}$ = 0.3 × 10^{20}A/Wm^2

For more details see J. Menard, et al., Nucl. Fusion 51 (2011) 103014

High performance scenarios can access increased neutron wall loading and Q_{eng} > 1 at large R

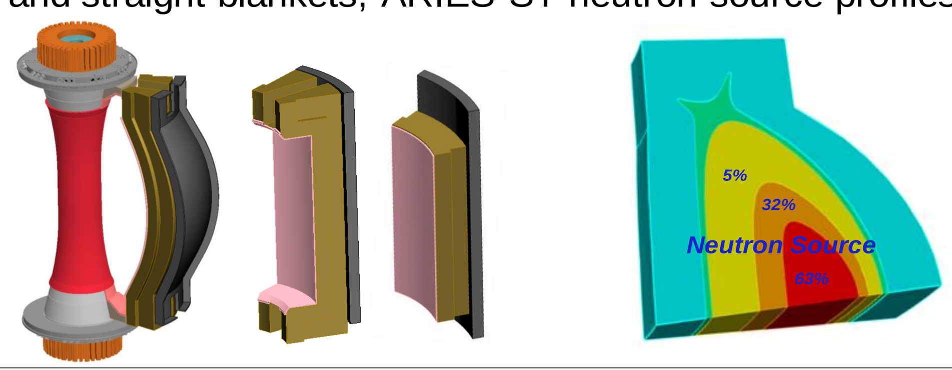
- Decrease B_T = 3T → 2.6T, increase H_{98} = 1.2 → 1.5
- Fix β_N = 6, β_T = 35%, q* = 2.5, $f_{Greenwald}$ varies: 0.66 to 0.47



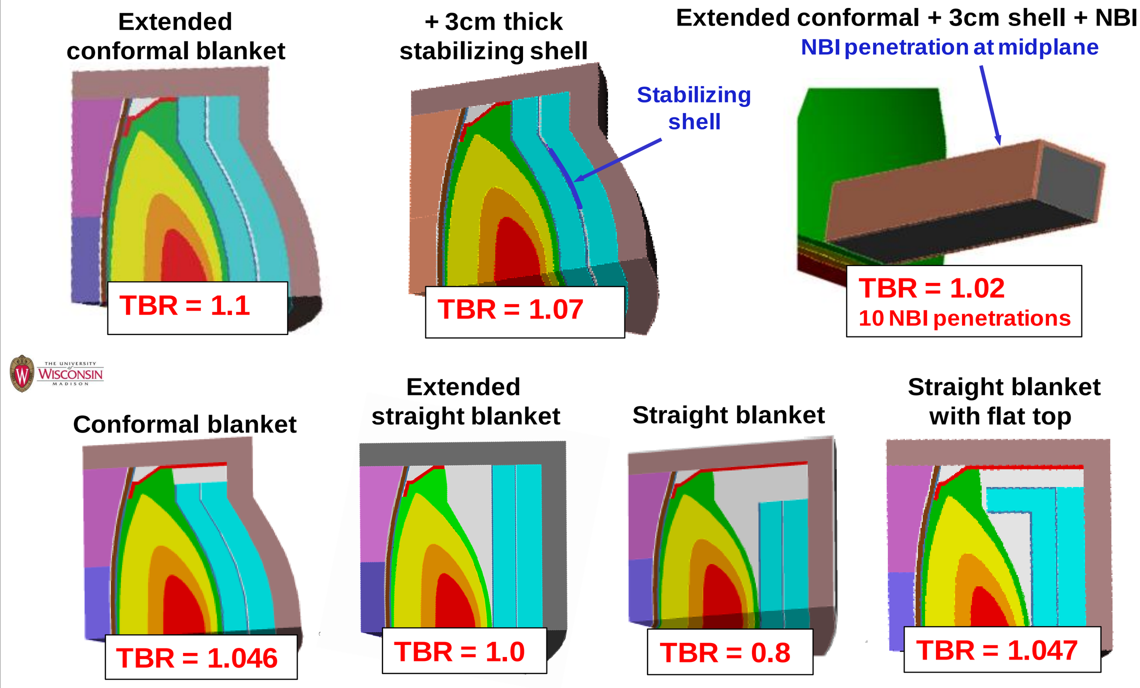
- Size scan: Q increases from 3 (R=1m) to 14 (R=2.2m)
- Average neutron wall loading increases from 1.8 to 3 MW/m^2 (not shown)
- Smallest ST for Q_{eng} ~ 1 is R=1.6m → requires very efficient blankets

Cost of T and need to demonstrate self-sufficiency motivate analysis of tritium breeding ratio (TBR)

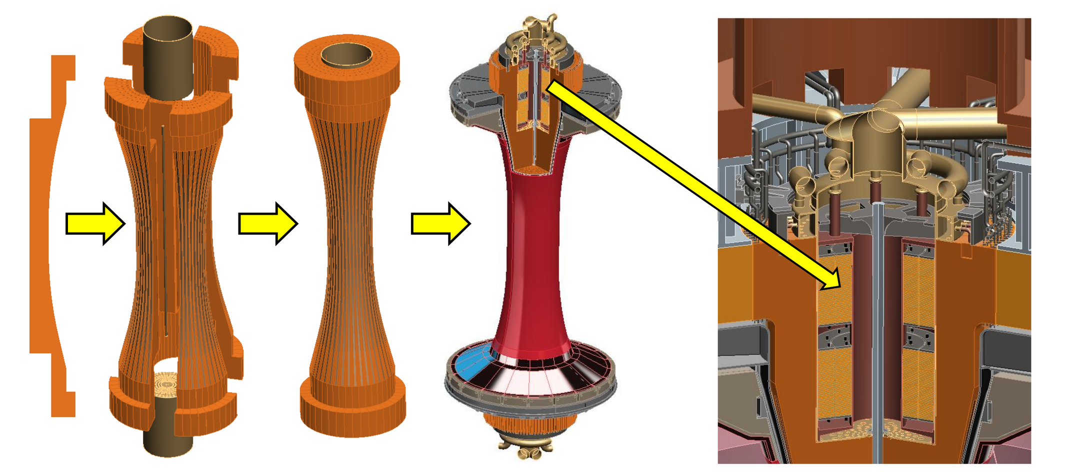
- Example costs of T w/o breeding at \$0.1B/kg for R=1 → 1.6m
  - FNS mission: 1MWy/m^2 — \$0.33B → \$0.9B
  - Component testing: 6MWy/m^2 — \$2B → \$5.4B
- Implications:
  - TBR << 1 likely affordable for FNS mission with R ~ 1m
  - Component testing arguably requires TBR approaching 1 for all R
- Performed initial analysis of R=1.6m FNSF using conformal and straight blankets, ARIES-ST neutron source profiles:



R=1.6m TBR calculations highlight importance of shells, penetrations, and top/bottom blankets



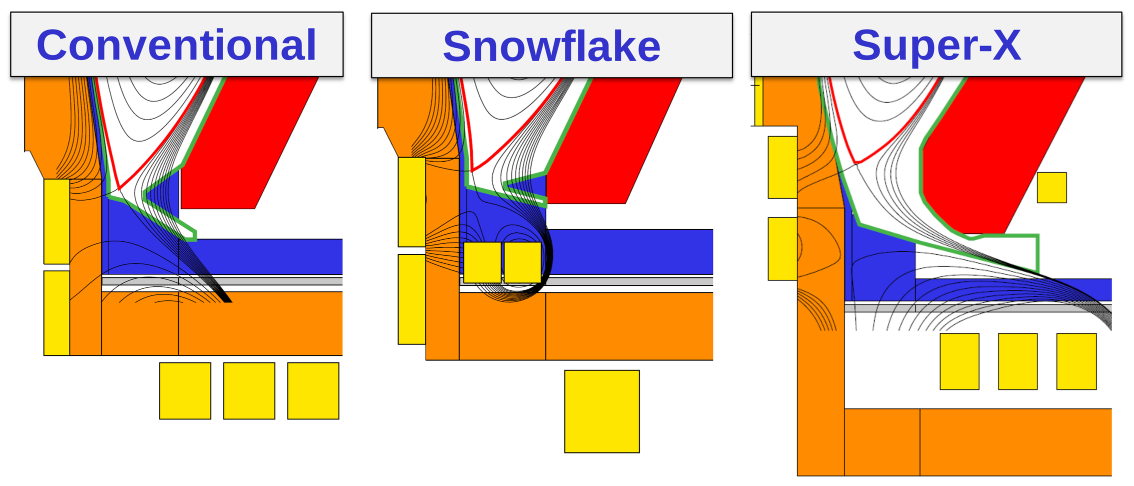
FNSF center-stack can build upon NSTX-U design, incorporate NSTX stability results



- Like NSTX-U, use TF wedge segments (but brazed/pressed-fit together)
  - Coolant paths: gun-drilled holes or NSTX-U-like grooves in wedge + welded tube
- Bitter-plate divertor PF magnets in ends of TF enable high triangularity
  - NSTX data: High δ > 0.55 and shaping S ≡ $q_{95}I_P/aB_T$ > 25 minimizes disruptivity
  - Neutronics: MgO insulation can withstand lifetime (6 FPY) radiation dose

Divertor PF coil configurations identified to achieve high δ, maintain peak divertor heat flux ≤ 10MW/m^2

Field-line angle of incidence at strike-point = 1°



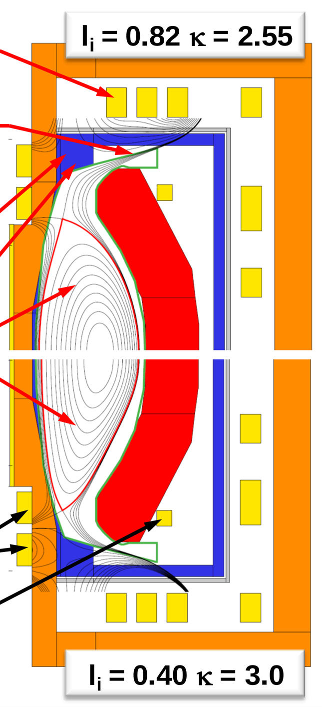
Conventional	Snowflake	Super-X
q_{peak} ~ 10MW/m^2	q_{peak} ~ 10MW/m^2	q_{peak} ~ 3MW/m^2
Flux expansion = 15-25	Flux expansion = 40-60	Flux expansion = 2
1/sin(θ_{plate}) = 2-3	1/sin(θ_{plate}) = 1-1.5	1/sin(θ_{plate}) = 15
R_{strike} = 1.15m, δ_x ~ 0.55	R_{strike} = 1.05m, δ_x ~ 0.62	R_{strike} = 2.6m, δ_x ~ 0.56

Combined super-X + snowflake divertor configuration has many attractive features

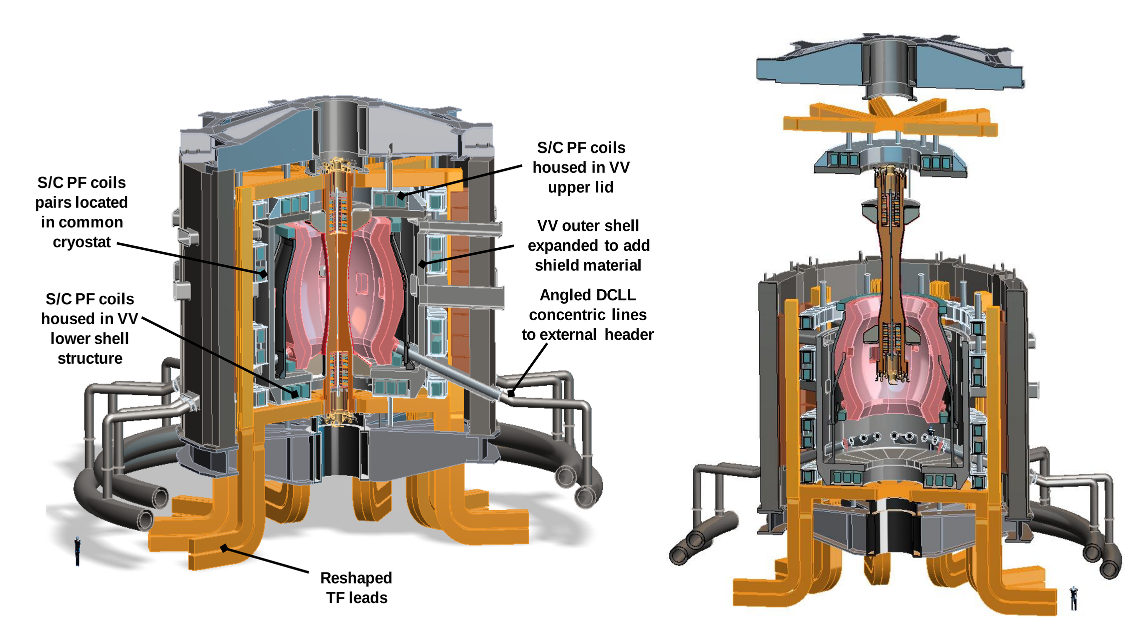
- Outboard PF coils shielded by blankets → can be SC
- Increased strike-point radius to reduce B and $q_{||}$, further increase line-length
- Strike-point PFCs shielded by blankets
- 2nd X-point/snowflake lowers B_P, increases line-length
- Possible location for T breeding to increase TBR
- PF coil design supports wide range of l_i values (0.4 – 0.8) with fixed strike-point location/region and controllable B-field angle of incidence (0.5-5°)

Normally conducting PF coil features:

- Divertor coils in TF coil ends for equilibrium, high δ
- In-vessel coils not-required for shaping – will be used for vertical control (to be studied in future)



R=1.6 device configuration with Super-X



Summary

- Present STs (NSTX/MAST) providing preliminary physics basis for ST-FNSF performance studies
  - Upgraded devices will provide more extensive and definitive basis
- Neutron wall loading of 1MW/m^2 feasible for range of major radii for β and H_{98} values already achieved
  - High wall loading and/or pilot-level performance require β_N ~ 6 and H_{98} ~ 1.5 which are at/near maximum values attained in present STs
- TBR near 1 possible if top/bottom neutron losses minimized
  - TBR ≥ 1 may only be possible for R ≥ 1.6m – under active investigation
- Divertor PF coils in ends of TF bundle enable high δ, shaping
- Conventional, snowflake, super-X divertors investigated, PF coils incorporated to reduce peak heat flux << 10MW/m^2
- Vertical maintenance strategies for either full and/or toroidally segmented blankets being investigated