ST-FNSF Mission and Performance Dependence on Device Size

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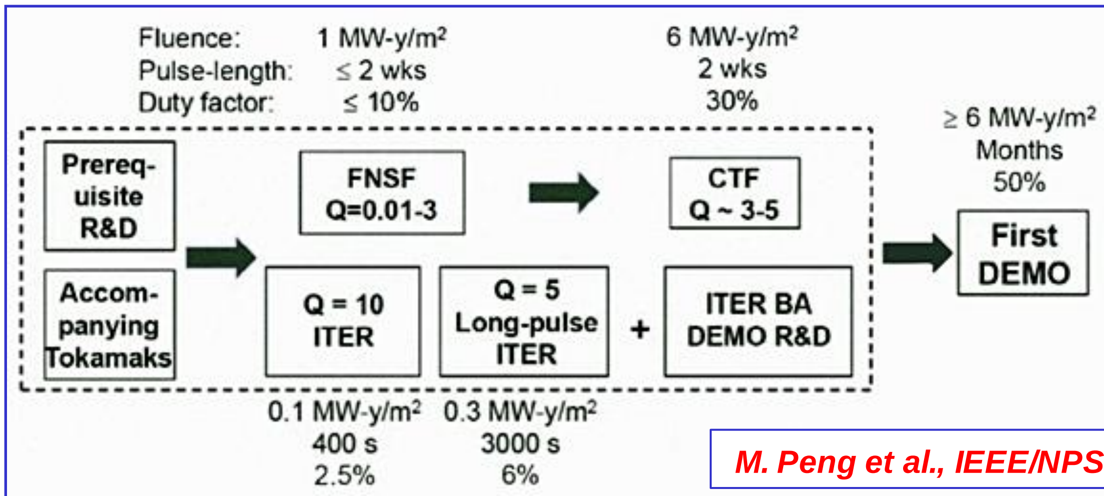
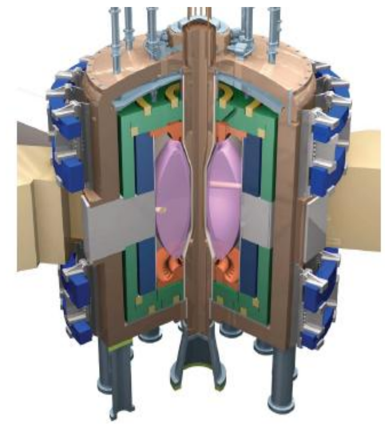
Outline

- Motivation for study
- Physics basis for operating points
- Performance vs. device size
- Tritium breeding ratio calculations
- Divertor poloidal field coil layout and design
- Power exhaust calculations
- Maintenance strategies
- Summary

Successful operation of upgraded STs (NSTX-U/MAST-U) could provide basis for design, operation of ST-based FNSF

- Fusion Nuclear Science Facility (FNSF) mission:
  - Provide continuous fusion neutron source to develop knowledge-base for materials and components, tritium fuel cycle, power extraction
- FNSF → CTF would complement ITER path to DEMO





M. Peng et al., IEEE/NPSS Paper S04A-2 - 24th SOFE Conf. (2011)

- Studying wide range of ST-FNSF configurations to identify advantageous features, incorporate into improved ST design
- Investigating performance vs. device size
  - Require: $W_{neutron} \geq 1$ MW/m^2, test area ≥ 10 m^2, volume ≥ 5 m^3

M. Abdou et al. Fus. Technol. 29 (1996) 1

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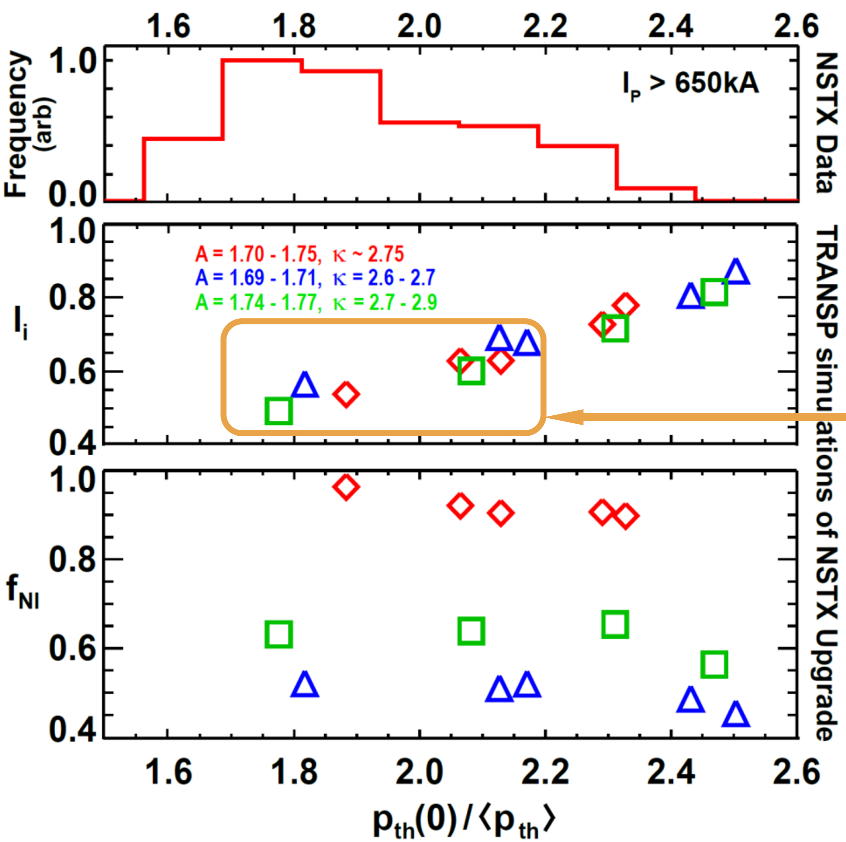
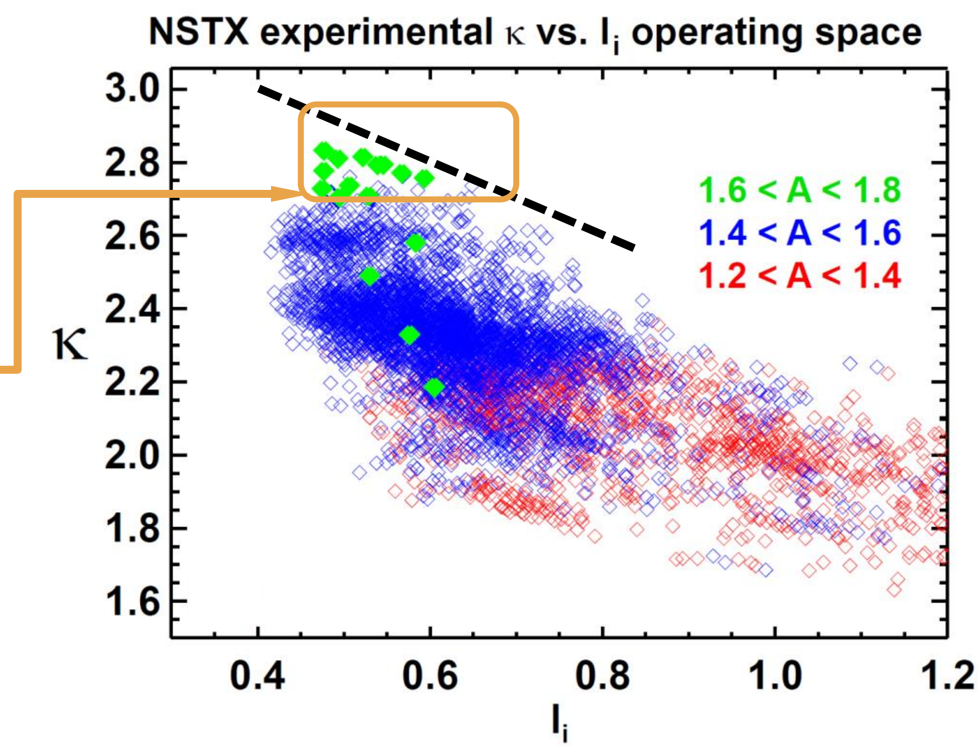
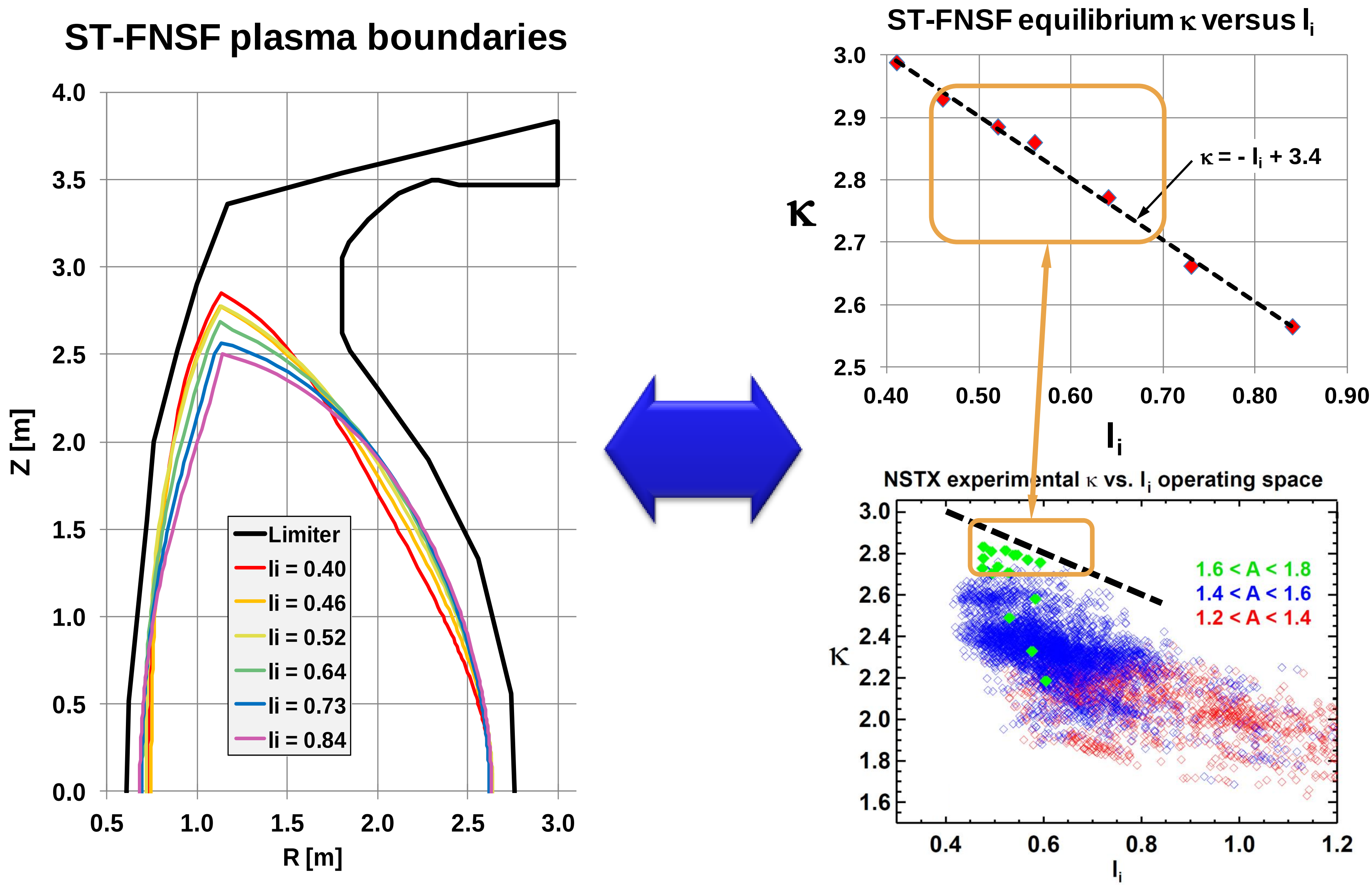
ST-FNSF equilibrium inductance, elongation based on values achieved/anticipated in NSTX/NSTX-U

- 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

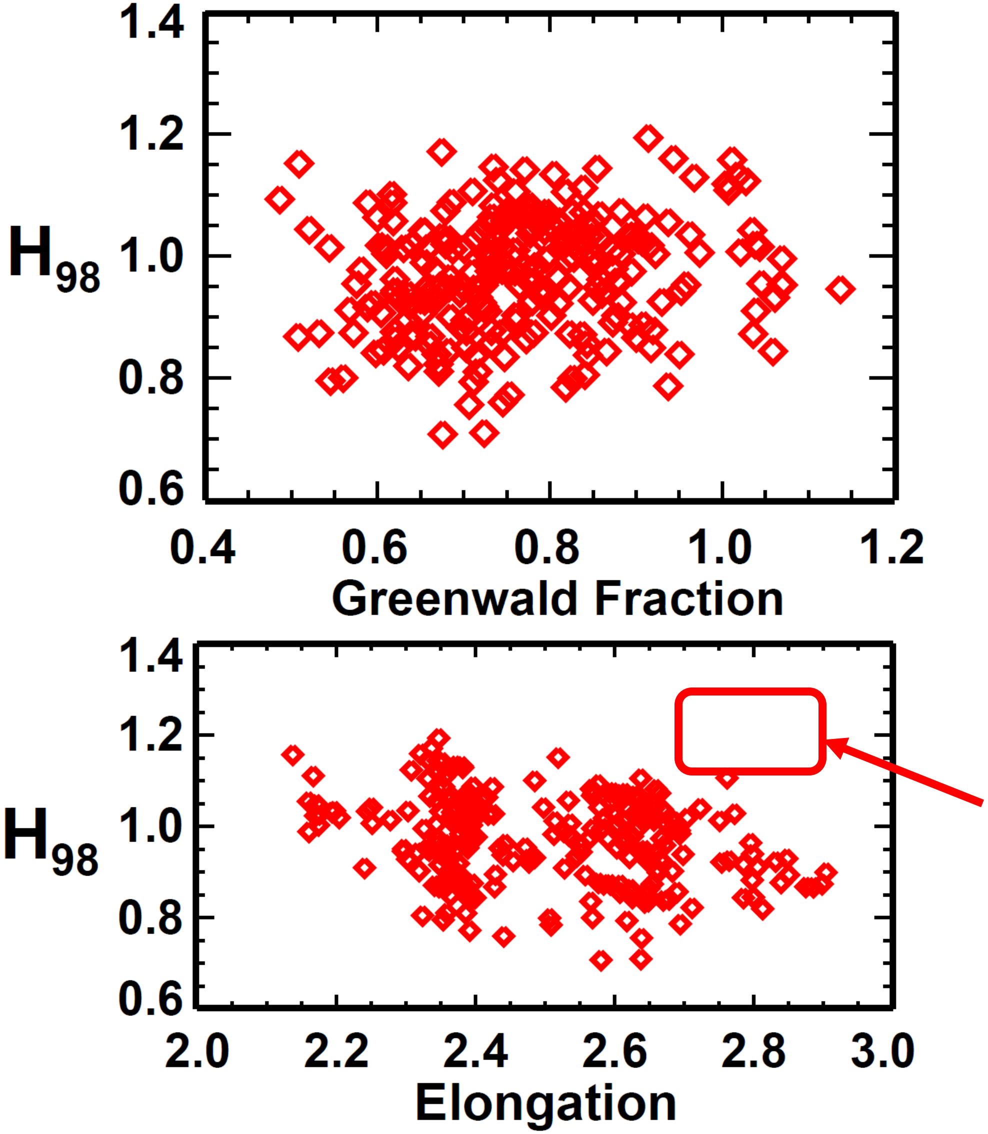




ST-FNSF free-boundary elongation is reduced with increasing l_i to match NSTX/NSTX-U trends



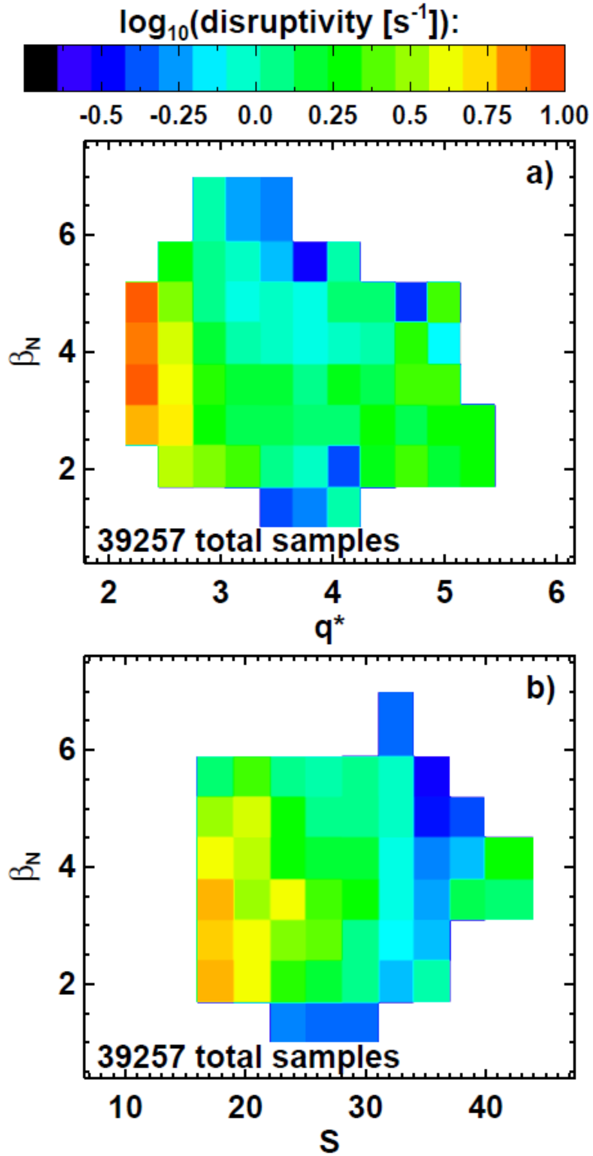
ST-FNSF operating point of $f_{Greenwald}$ = 0.8, $H_{98y,2}$=1.2 chosen to be at/near values anticipated for NSTX-U



- **$H_{98y,2}$ → 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
  - $S \equiv q_{95} I_P / aB_T$
  - S > 30 provides strongest stabilization
  - S > 22-25 good stability
  - S < 22 unfavorable

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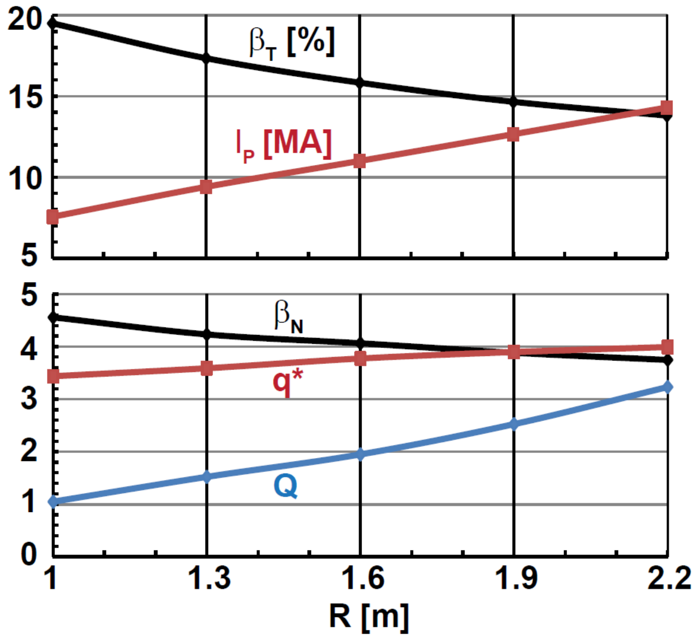
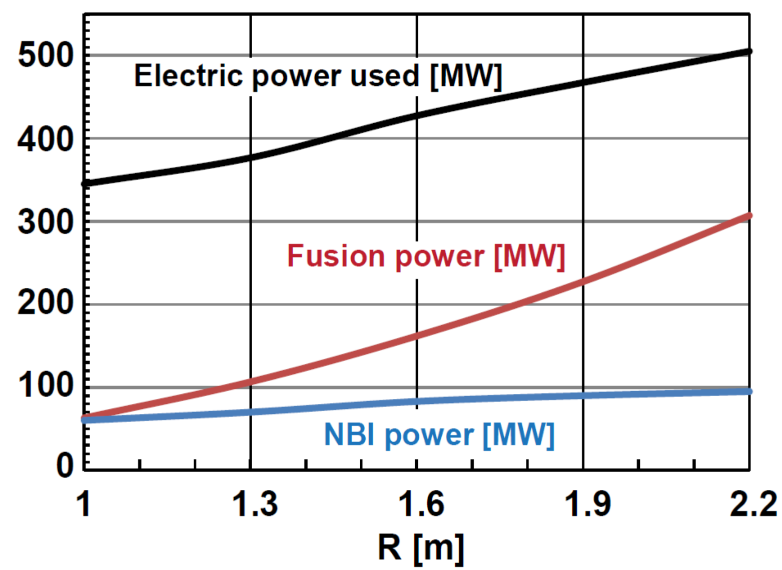
Increased device size provides modest increase in stability, but significantly increases T consumption

- Scan R = 1m → 2.2m (smallest FNSF → pilot plant with Q_{eng} ~ 1)
- Fixed average neutron wall loading = $1 MW/m^2$
- 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}

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

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

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):

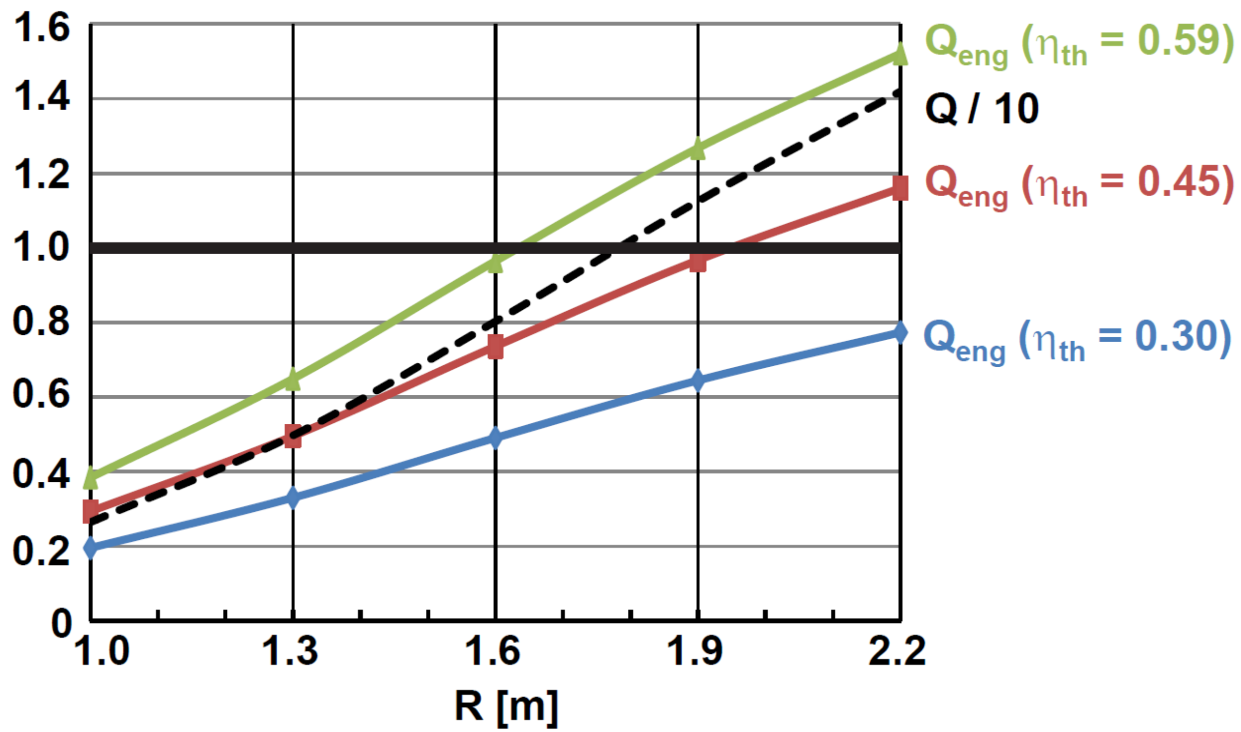
- $M_n = 1.1$
- $P_{pump} = 0.03 \times P_{th}$
- $P_{sub} + P_{control} = 0.04 \times P_{th}$
- $\eta_{aux} = 0.4$ (presently unrealistically high)
- $\eta_{CD} = I_{CD} R_0 n_e / P_{CD} = 0.3 \times 10^{20} A/Wm^2$

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

$$Q_{eng} \equiv \frac{P_{electric}\ \text{produced}}{P_{electric}\ \text{consumed}}$$



Note: Outboard PF coils are superconducting

- **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**

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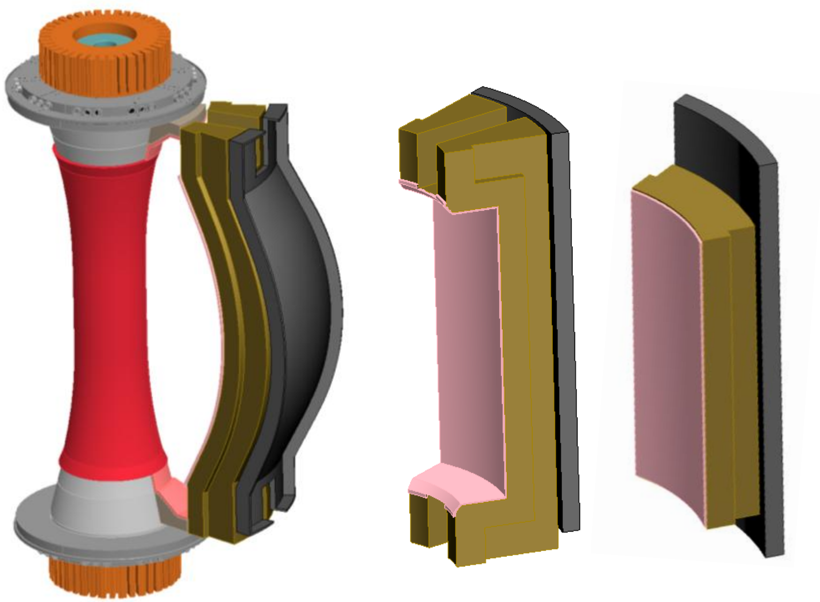
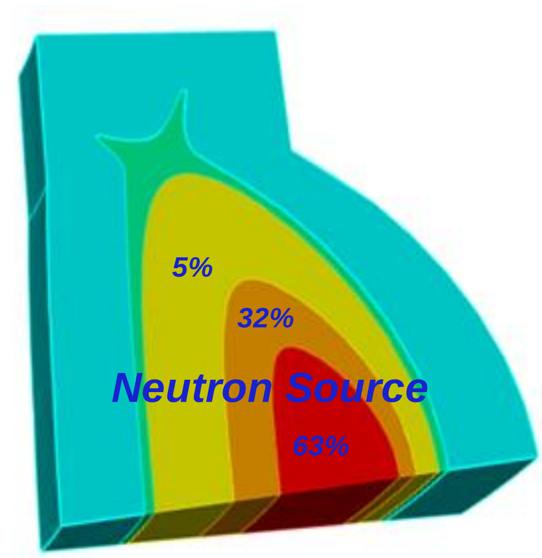
Cost of T and need to demonstrate self-sufficiency motivate analysis of tritium breeding ratio (TBR)

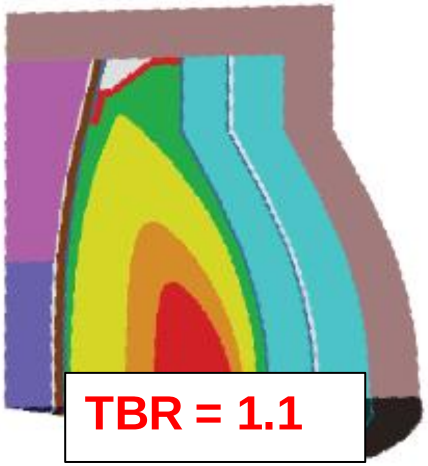
- Example costs of T w/o breeding at \$0.1B/kg for R=1 → 1.6m
  - FNS mission: 1MWy/m^2 \$0.33B → \$0.9B
  - Component testing: 6MWy/m^2 \$2B → \$5.4B
- Implications:
  - TBR << 1 likely affordable for FNS mission with R ~ 1m
  - Component testing arguably requires TBR approaching 1 for all R
- Performed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



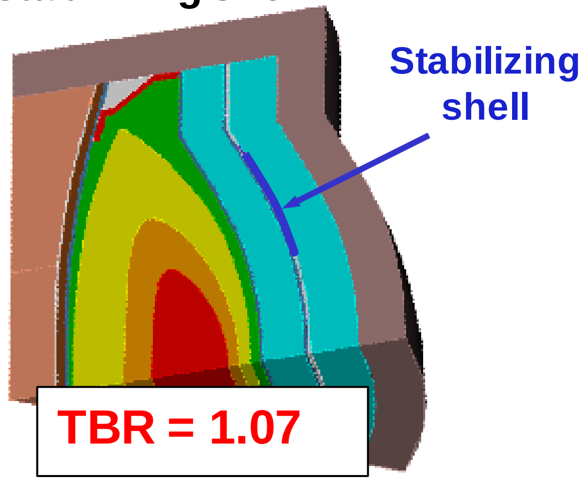


R=1.6m TBR calculations highlight importance of shells, penetrations, and top/bottom blankets

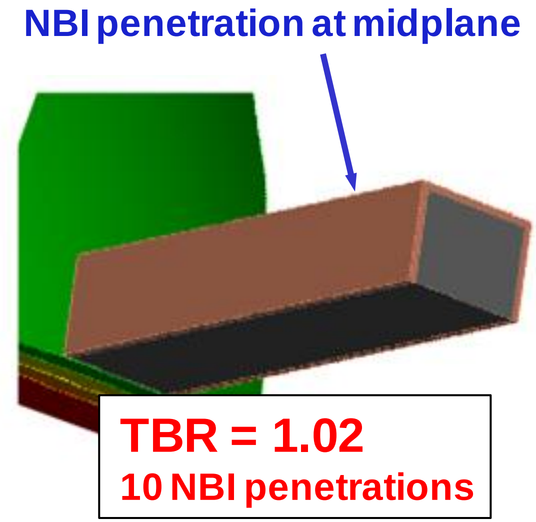
Extended conformal blanket



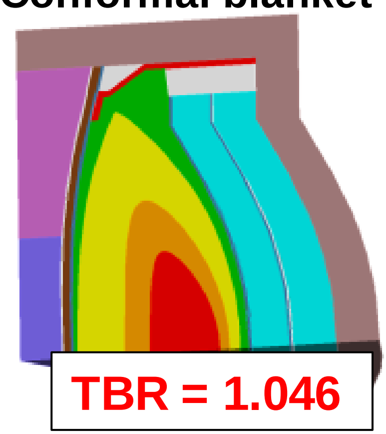
+ 3cm thick stabilizing shell



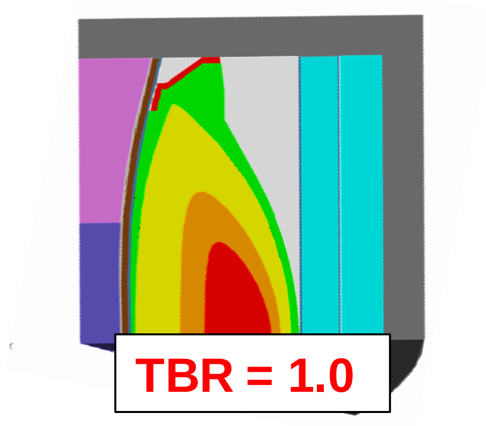
Extended conformal + 3cm shell + NBI



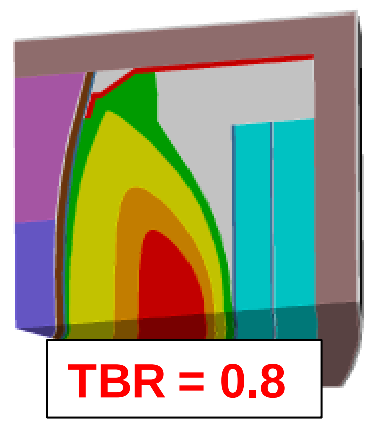
Conformal blanket



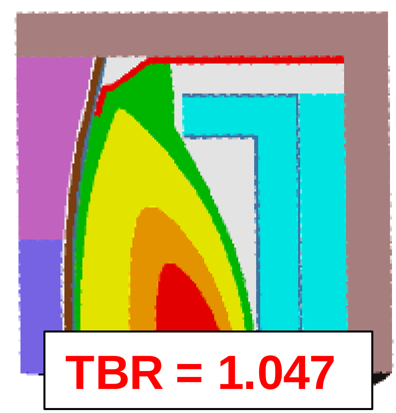
Extended straight blanket



Straight blanket



Straight blanket with flat top



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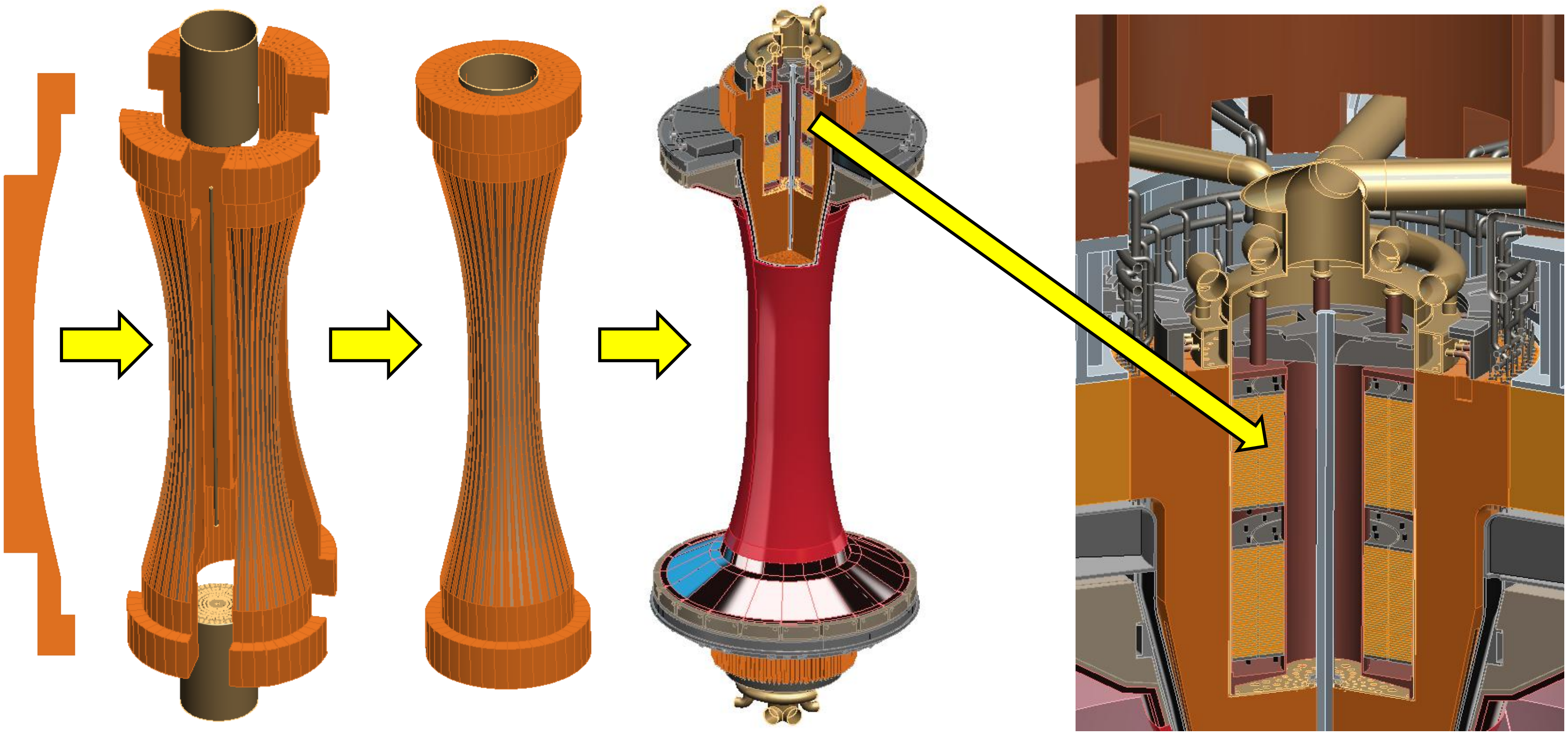
FNSF center-stack can build upon NSTX-U design, incorporate NSTX stability results



- **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 $\delta > 0.55$ and shaping $S \equiv q_{95}I_P/aB_T > 25$ minimizes disruptivity
  - Neutronics: MgO insulation can withstand lifetime (6 FPY) radiation dose

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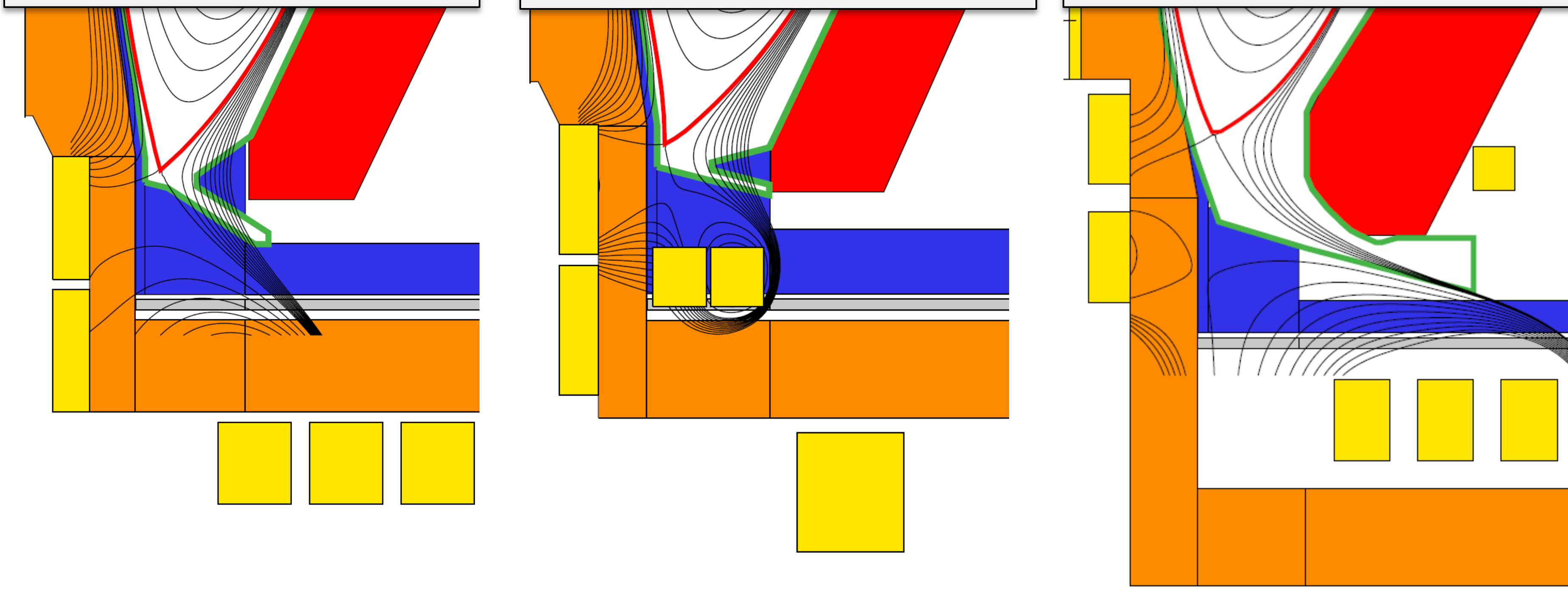
Divertor PF coil configurations identified to achieve high δ, maintain peak divertor heat flux ≤ 10MW/m^2

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

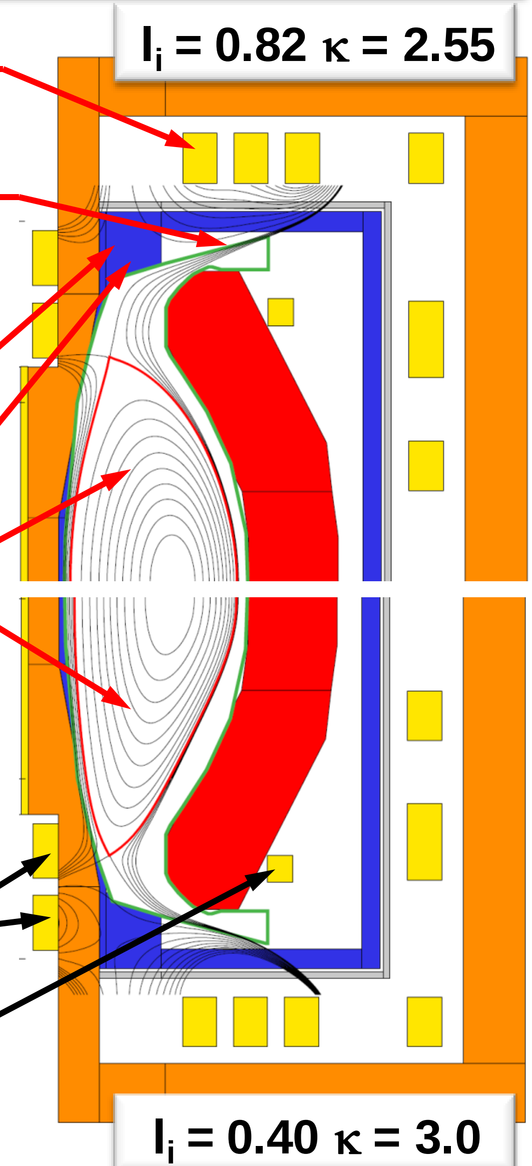
Conventional	Snowflake	Super-X



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
• $1/\sin(\theta_{plate})$ = 2-3	• $1/\sin(\theta_{plate})$ = 1-1.5	• $1/\sin(\theta_{plate})$ = 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

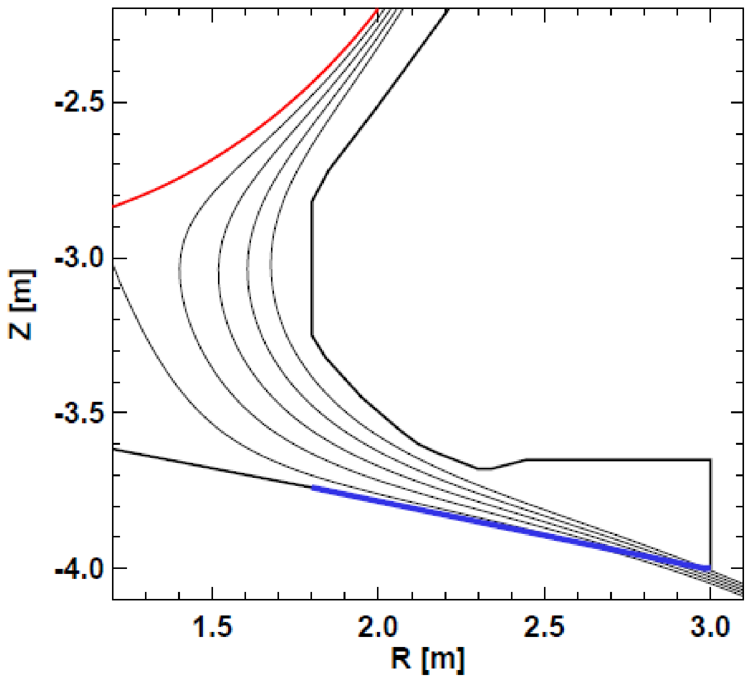
- Outboard PF coils shielded by blankets → can be SC
- Increased strike-point radius to reduce B and $q_{||}$, further increase line-length
- Strike-point PFCs shielded by blankets
- 2^{nd} X-point/snowflake lowers B_P, increases line-length
- Possible location for T breeding to increase TBR
- PF coil 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°)

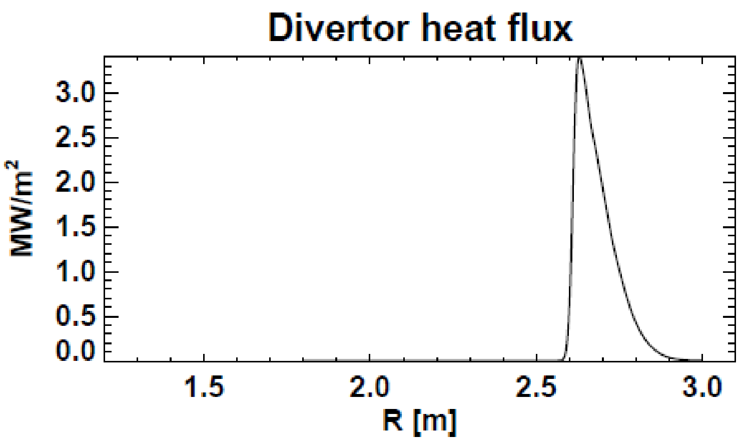


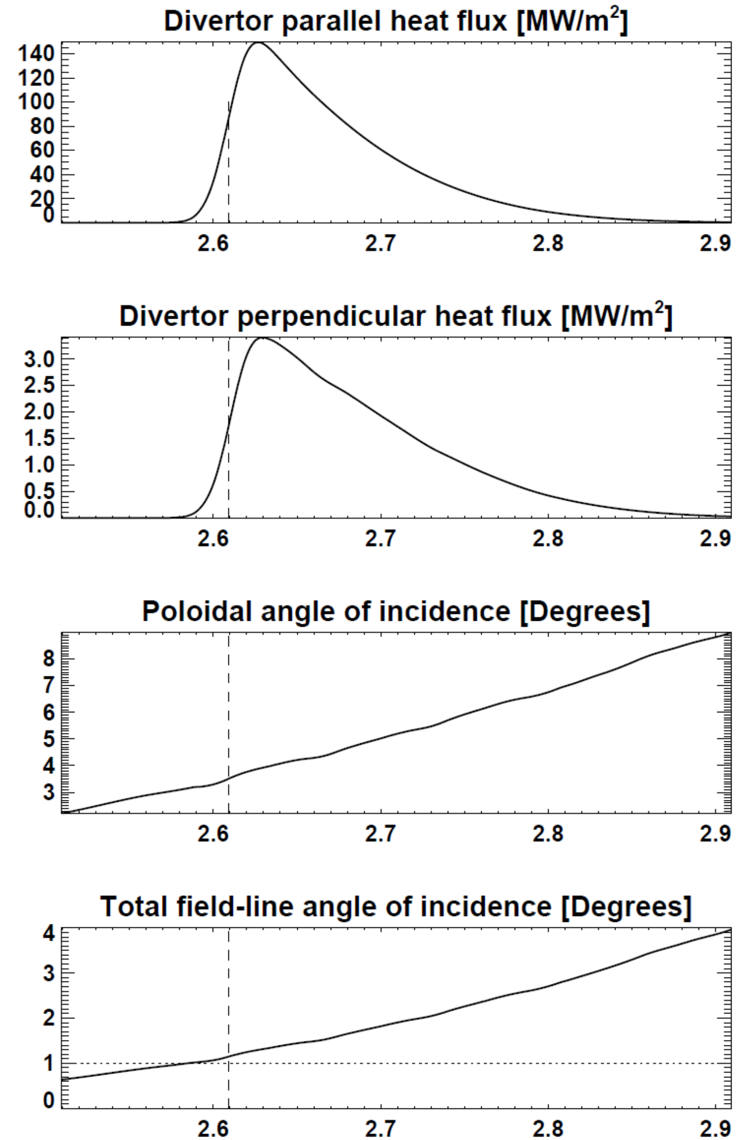
Normally conducting PF coil features:

- 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)

Super-X → ~3× reduction in q_{peak}: 10 → 3MW/m² for fixed radiation fraction and angle of incidence

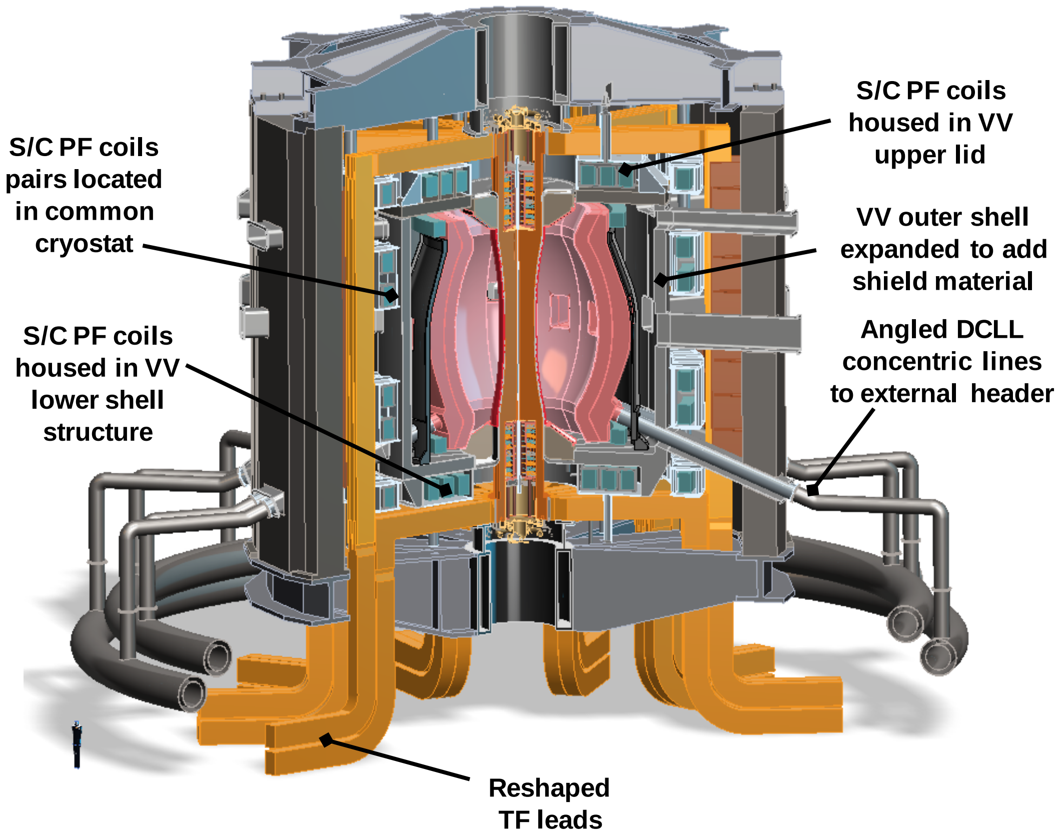




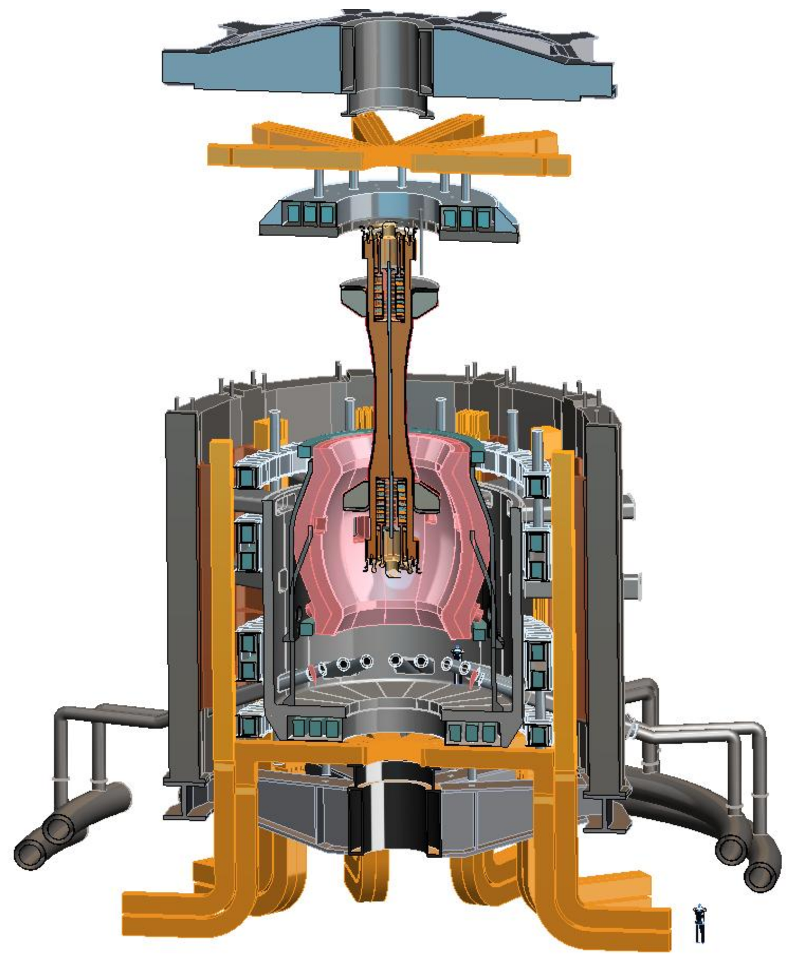


R=1.6 device configuration with Super-X

Design features



Vertical maintenance approach



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 at/near 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

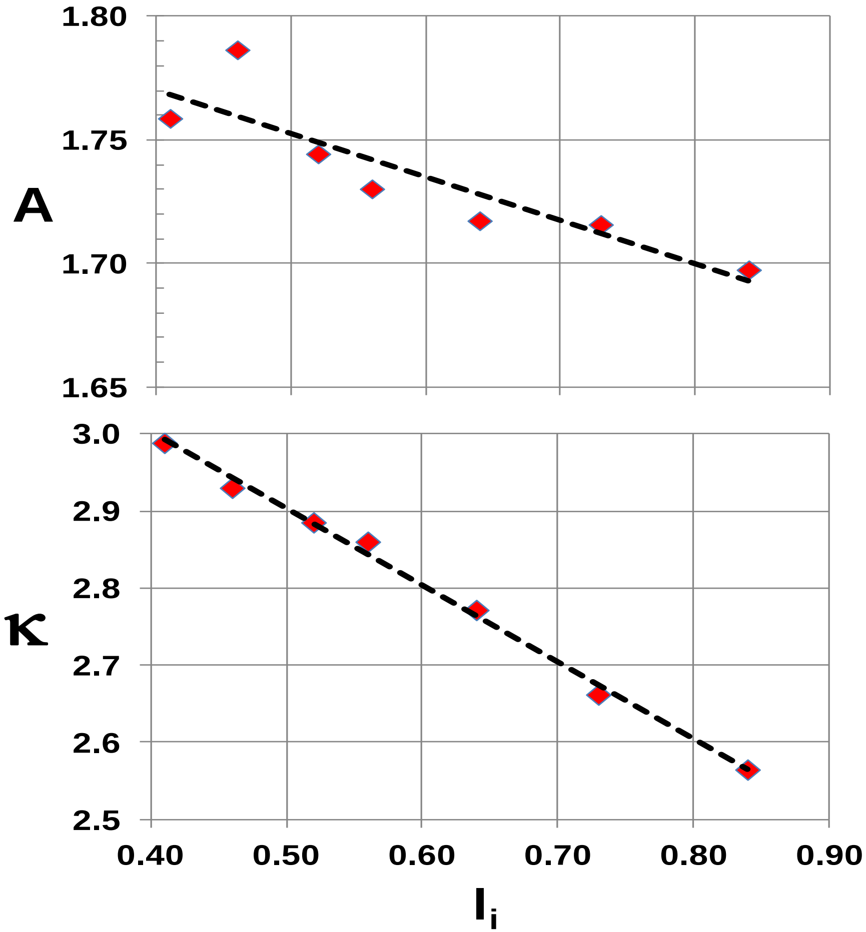
Future work

- Physics basis for operating points
  - Perform sensitivity study of achievable performance vs. baseline configuration assumptions: A, κ, $H_{98y,2}$, ST vs. tokamak τ_E scaling
  - TRANSP calculations of NBI heating, current drive, neutron production
- Performance vs. device size
  - Could/should overall machine configuration change at smaller R?
    - Example questions: could/should vessel take more load?
  - Is there sufficient shielding for divertor PF coils at smaller R?
- Tritium breeding ratio calculations
  - Extend calculations to smaller R
  - Include 3D effects and final machine layout
- Maintenance strategies
  - Assess space/lift requirements above machine for vertical maintenance

Backup

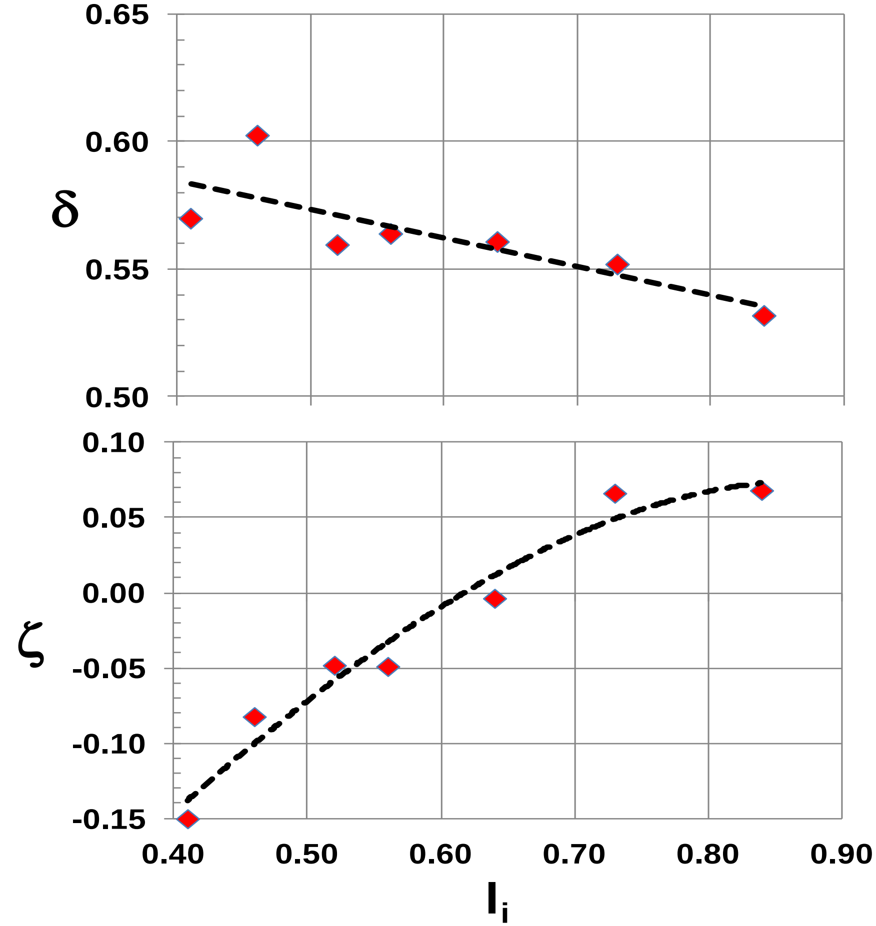
Boundary shape parameters vs. internal inductance

A = 1.8→1.7 at higher l_i



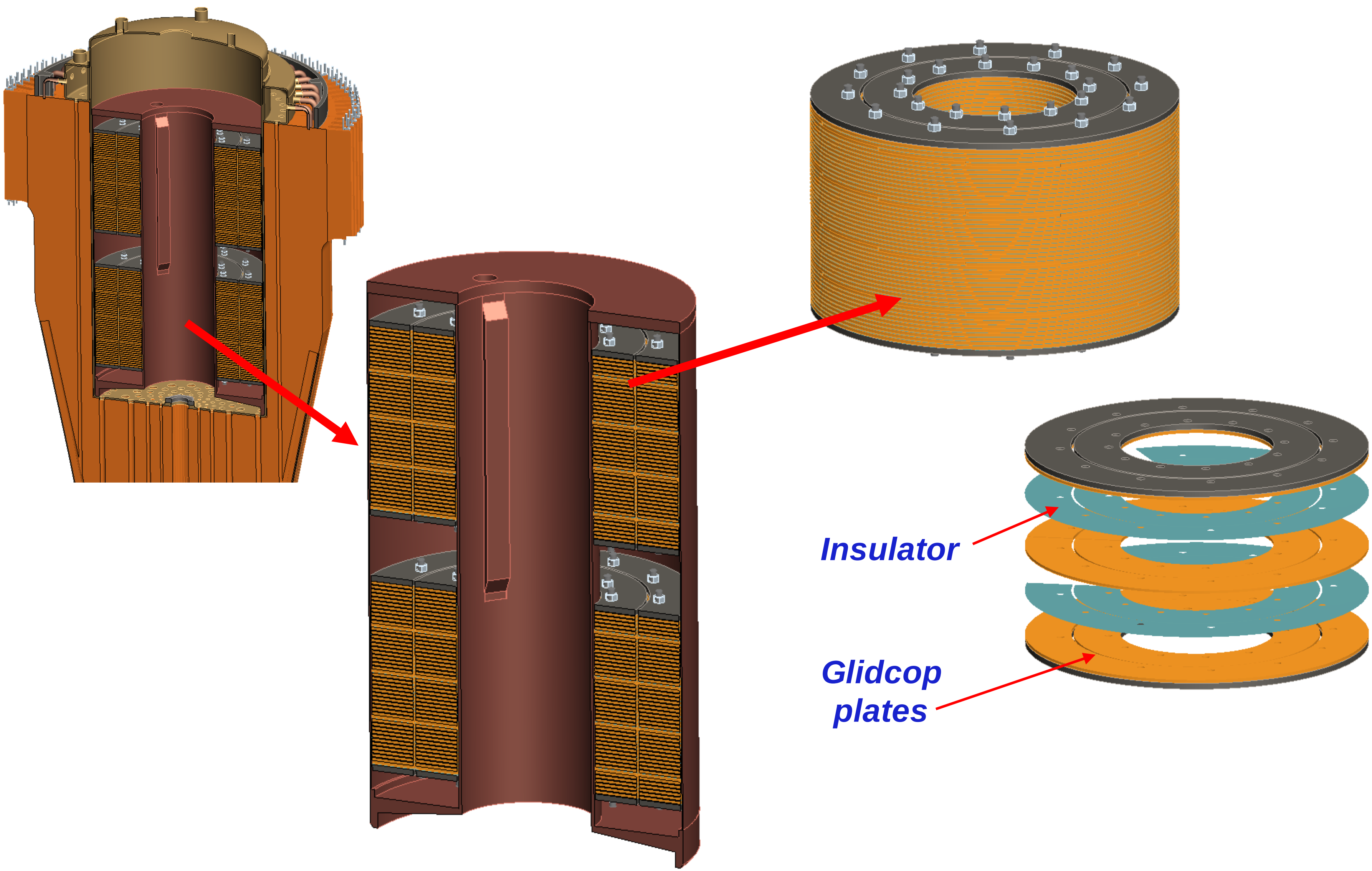
κ reduced at higher l_i

High δ = 0.5-0.6 maintained



Negative squareness at low l_i

Bitter coil insert for divertor coils in ends of TF

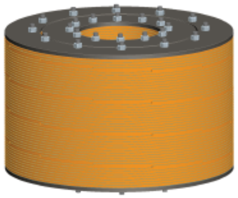


Neutronics analysis indicates organic insulator for divertor PF coils unacceptable

Dose Results

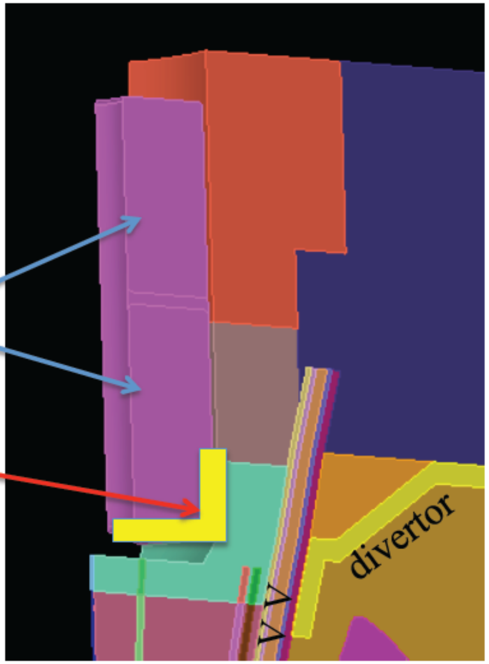
PF Coil **Composition**, per T. Brown

	Volume (in^3)	%
water	5017	20
copper	17059	68
insulator	1254	5
hardware	1756	7
Total	25087	100



PF Coils

Insulator (cyanate ester / epoxy blend)
placed at corner of PF coil to calculate peak dose



3-D Model

Peak dose to insulator $= 3 \times 10^{11}$ **rad @ 1 FPY**
$= 1.8 \times 10^{12}$ **rad @ 6 FPY**
>> 2×10^{10} rad limit

Replace inner PF coil every 24 days (not practical).

Is ceramic insulator more radiation resistant?

MgO insulation appears to have good radiation resistance for divertor PF coils

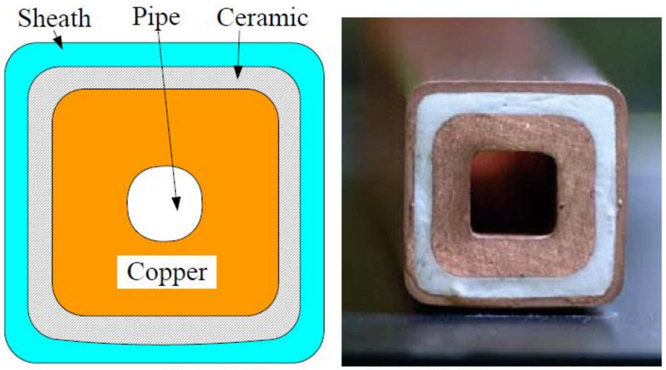


Fig. 3 Cross section of MIC

Table 1: Comparison of radiation resistant

	Organic		Inorganic
Insulation	Epoxy	Polyimide	MgO
Resistant	$>10^7$ Gy	$>10^9$ Gy	$>10^{11}$ Gy

R&D of a Septum Magnet Using MIC coil

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Proceedings of the 5th Annual Meeting of Particle Accelerator Society of Japan and the 33rd Linear Accelerator Meeting in Japan (August 6-8, 2008, Higashihiroshima, Japan)

- UW analysis of divertor PFs
  - 1.8×10^{12} rad = 1.8×10^{10} Gy at 6FPY for P_{fus} = 160MW
- Pilot mission for R=1.6m:
  - P_{fus} = 420MW vs. 160MW → 2.6x higher → 4.7×10^{10} Gy
  - Even for Pilot mission, dose is < limit of 10^{11} Gy
- Limiting factor may be Cu
- Need to analyze CS lifetime
- Revisit option for multi-turn TF and small OH solenoid