Studies of ST-FNSF mission and performance dependence on device size

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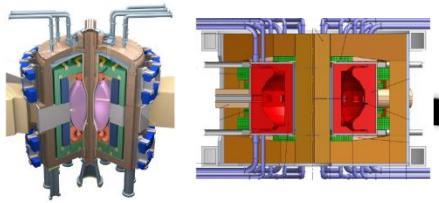
There are many possible pathways from ITER to commercial fusion power plant

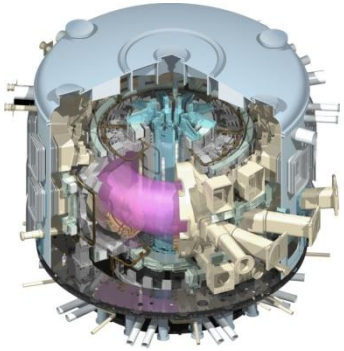
FNSF = Fusion Nuclear Science Facility
CTF = Component Test Facility



FNSF/CTF

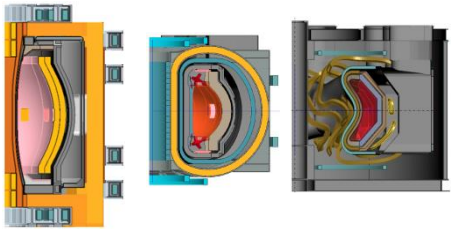
Blanket R&D, T self-sufficiency

ITER



Supporting Physics and Technology

- Core Physics
- Materials R&D
- Plasma Material Interface

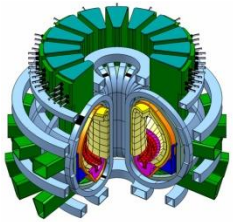


Pilot Plant

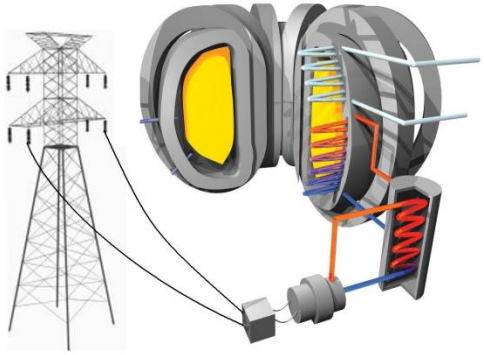
FNSF/CTF with power-plant like maintenance, $Q_{eng} \geq 1$

Q_{eng} = 3-5

e.g. EU Demo

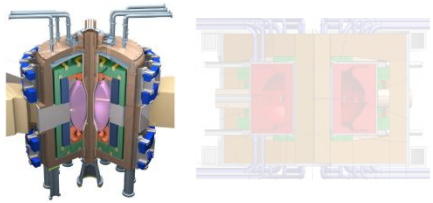


First of a kind Power Plant



This talk focuses on possible spherical tokamak (ST) contributions ranging from FNSF to Pilot Plant

FNSF = Fusion Nuclear Science Facility
CTF = Component Test Facility



FNSF/CTF
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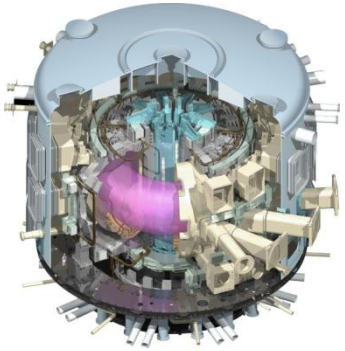

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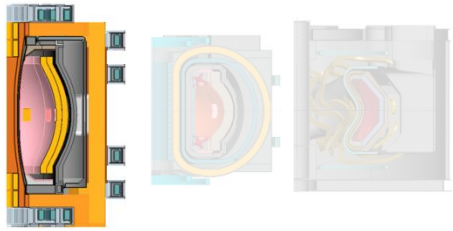
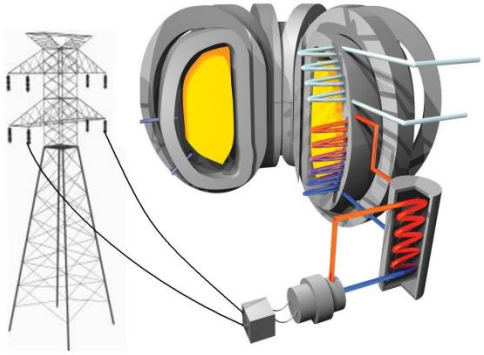


Pilot Plant
FNSF/CTF with power-plant like maintenance, $Q_{eng} \geq 1$

First of a kind Power Plant



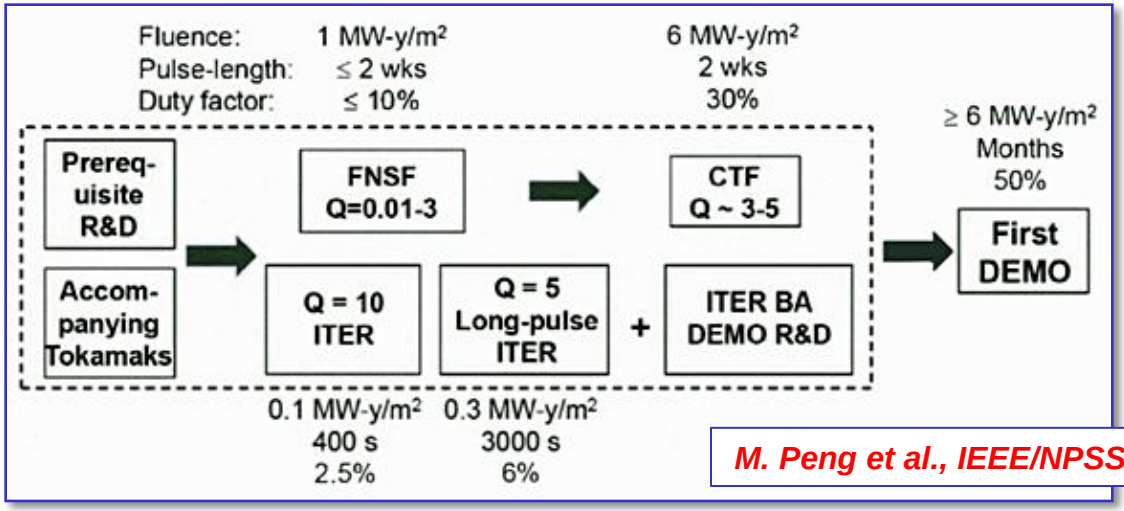
Q_{eng} = 3-5
e.g. EU Demo

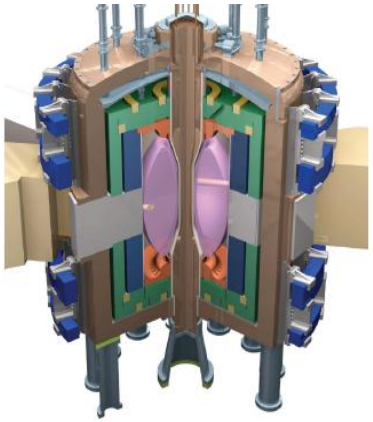
Outline

- 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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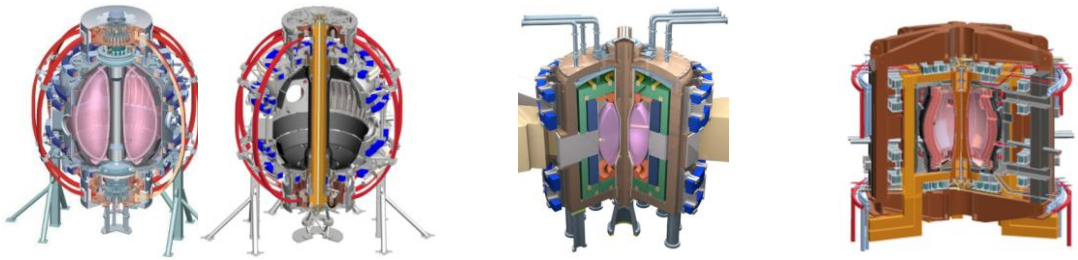
NSTX-U will access next factor of 2 increase in performance to reduce extrapolation to ST-based FNSF/CTF/Pilot



Parameter	NSTX	NSTX Upgrade	Fusion Nuclear Science Facility	Pilot Plant
Major Radius R_0 [m]	0.86	**0.94**	1.3	1.6 – 2.2
Aspect Ratio R_0 / a	≥ 1.3	**≥ 1.5**	≥ 1.5	≥ 1.7
Plasma Current [MA]	1	**2**	4 – 10	11 – 18
Toroidal Field [T]	0.5	**1**	2 – 3	2.4 – 3
Auxiliary Power [MW]	≤ 8	**≤ 19***	22 – 45	50 – 85
P/R [MW/m]	10	**20**	30 – 60	70 – 90
P/S [MW/m^2]	0.2	**0.4**	0.6 – 1.2	0.7 – 0.9
Fusion Gain Q			1 – 2	2 – 10

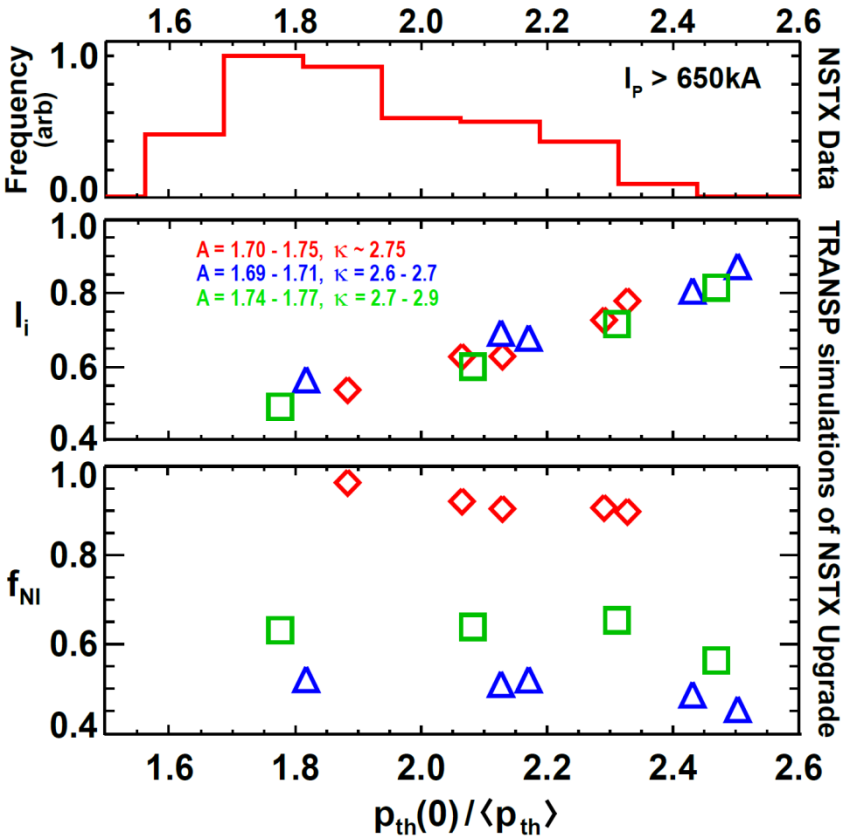
**** Includes 4MW of high-harmonic fast-wave (HHFW) heating power***

Key issues to resolve for FNSF

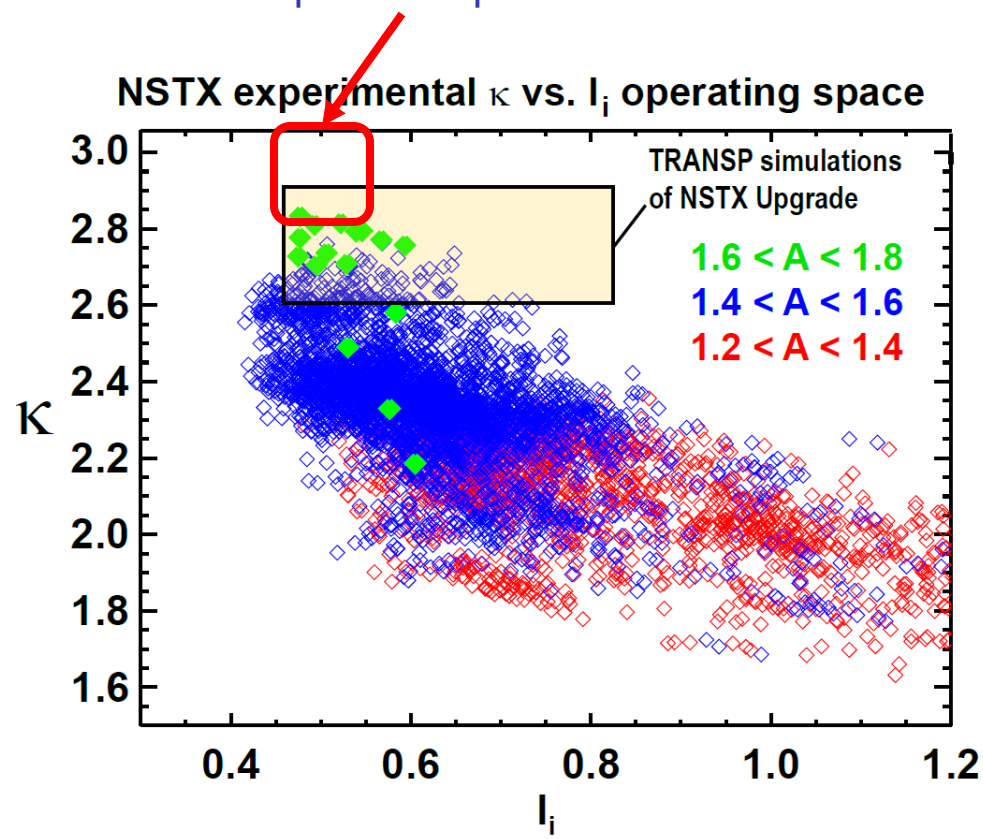
- **Achievable confinement**
- **Non-inductive ramp-up and sustainment**
- **Divertor solutions for mitigating high heat flux**
- **Radiation-tolerant magnets**

ST-FNSF operating point of A=1.7, l_i = 0.5, and κ=3 chosen to be at/near values anticipated for NSTX-U

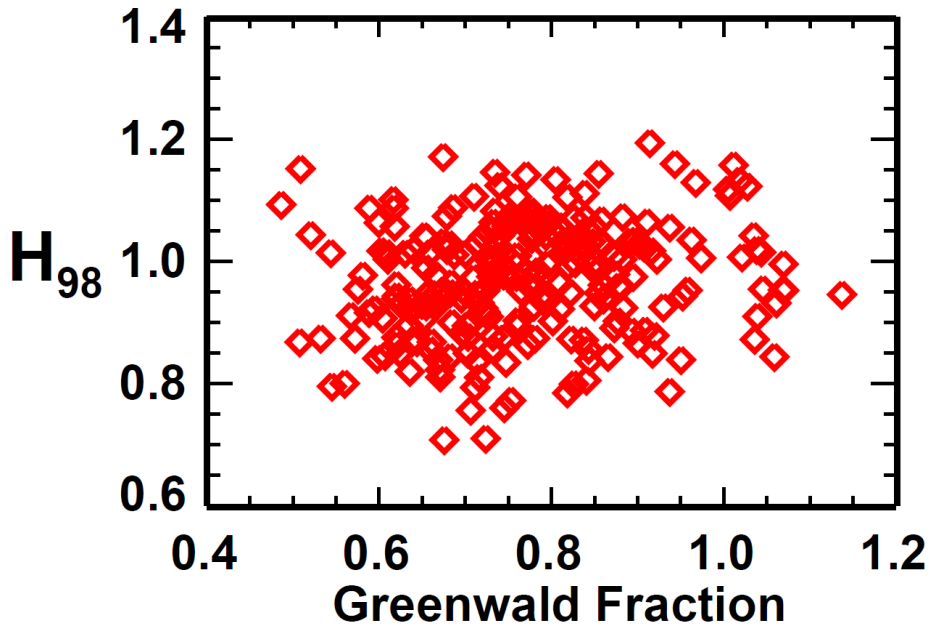
- Most probable NSTX thermal pressure peaking factor ~1.8
  - If similar in NSTX-U/FNSF → full non-inductive l_i ~ 0.5 (BS + NBI)

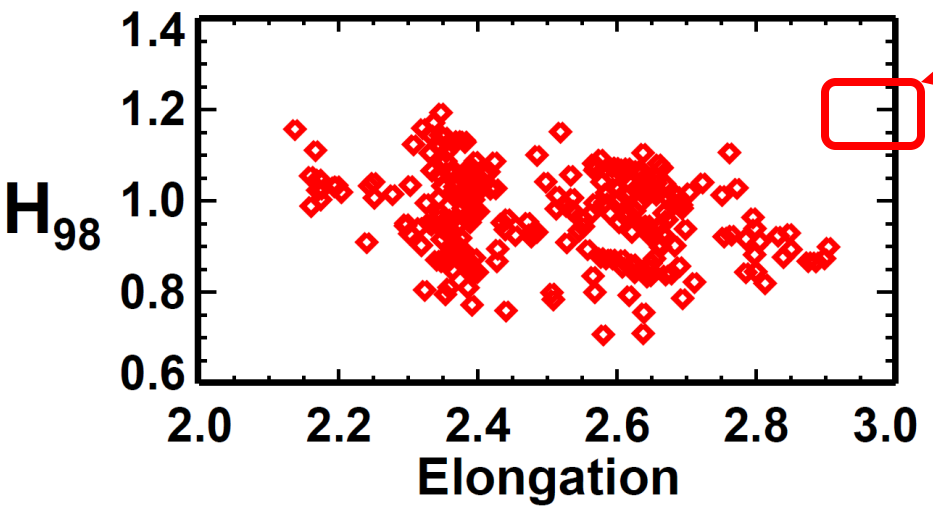


- NSTX A=1.7, l_i = 0.5 plasmas can operate stably at κ ~ 2.8
  - Expect to improve n=0 control in NSTX-U → anticipate κ~3 possible in NSTX-U/FNSF



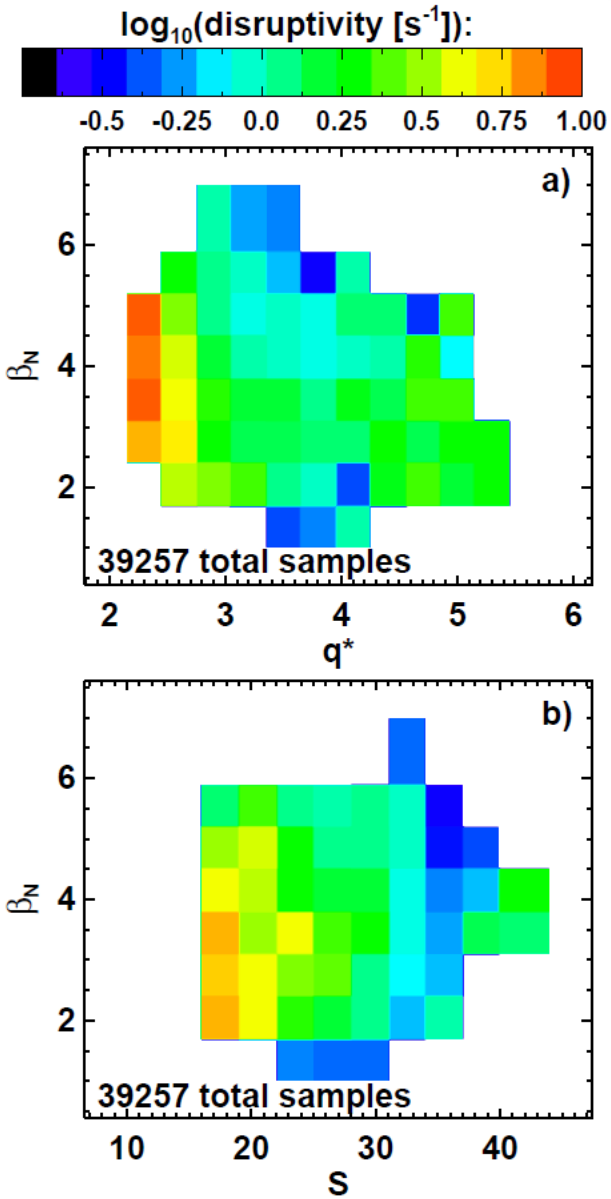
ST-FNSF operating point of $f_{Greenwald}$ = 0.8, $H_{98y,2}$=1.2 chosen to be at/near values anticipated for NSTX-U





- $H_{98y,2}$ ~ 1.2 has been accessed for a range of Greenwald fractions
  - However, much more research needs to be carried out in NSTX-U to determine if HH=1.2 can be achieved reliably
- Also need to assess feasibility of routine access to $H_{98y,2}$ ~ 1.2 at κ~3 in NSTX-U
- **Note: there is sufficient shaping margin to reduce κ to 2.7-2.9 for FNS mission, but $H_{98y,2}$ ~ 1 would require much higher P_{aux} (~1.8×)**

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

Gerhardt IAEA FEC EX/9-3

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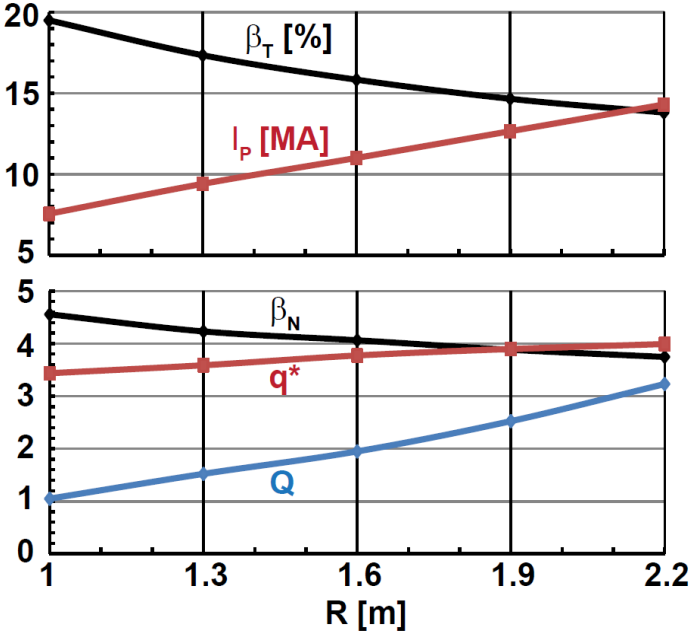
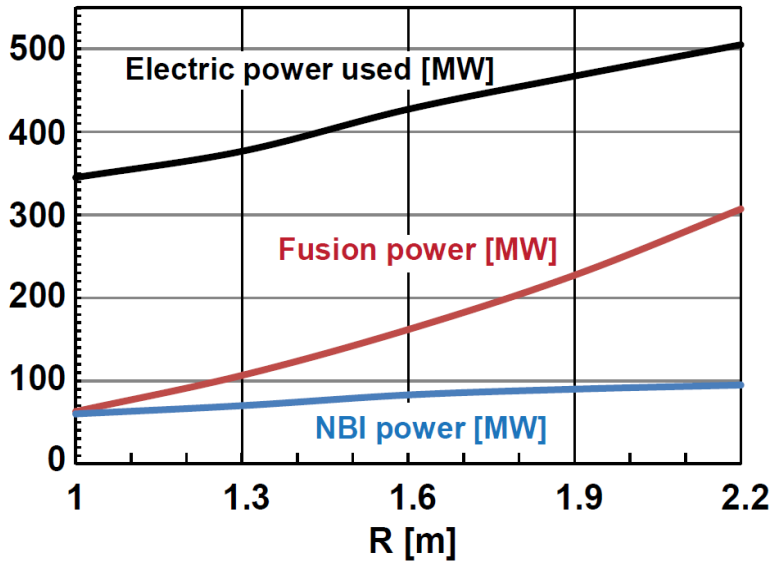
Increased device size provides modest increase in stability, but significantly increases tritium consumption

- 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

In addition to neutron wall loading and tritium breeding, FNSF study also tracking overall 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):

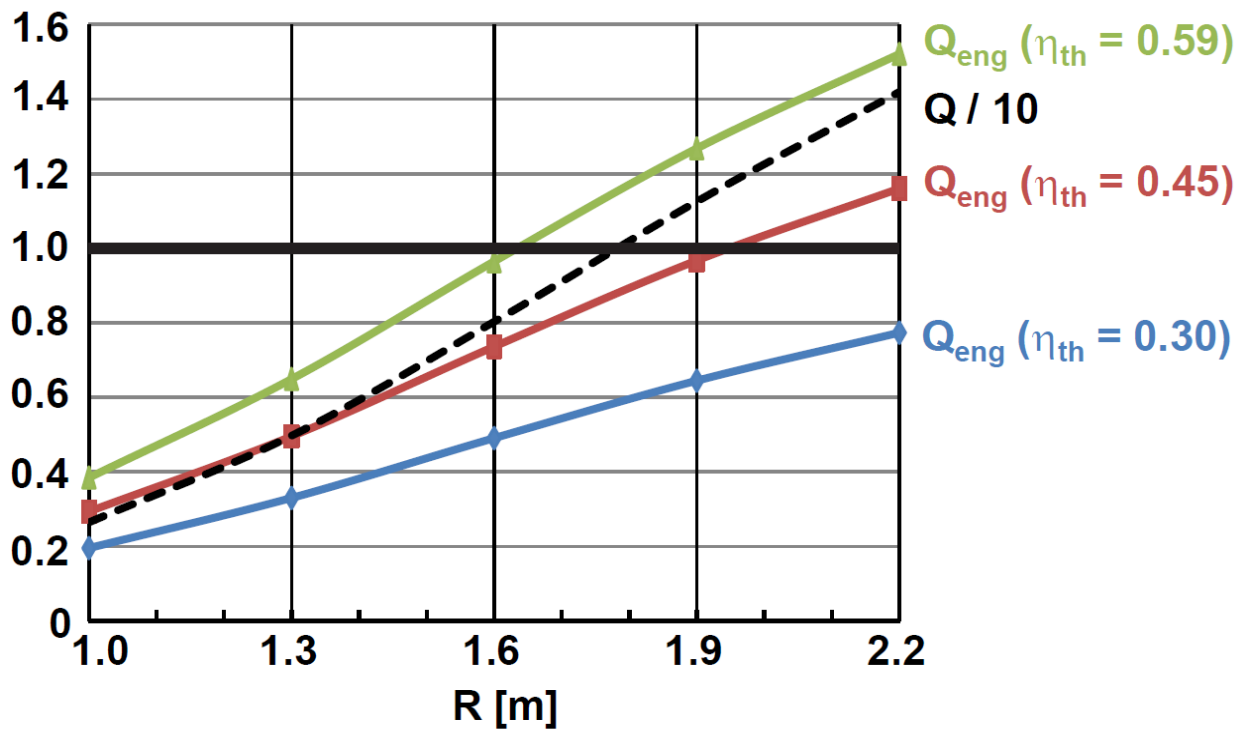
- $M_n = 1.1$
- $P_{pump} = 0.03 \times P_{th}$
- $P_{sub} + P_{control} = 0.04 \times P_{th}$
- $\eta_{aux} = 0.4$ (presently unrealistically high)
- $\eta_{CD} = I_{CD} R_0 n_e / P_{CD} = 0.3 \times 10^{20} A/Wm^2$

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

High performance scenarios can access increased neutron wall loading and engineering gain > 1 for sufficiently 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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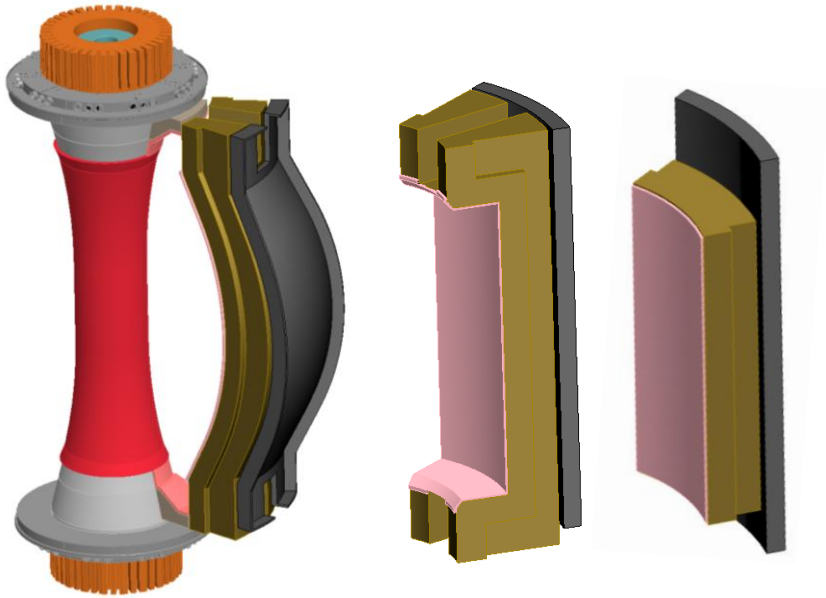
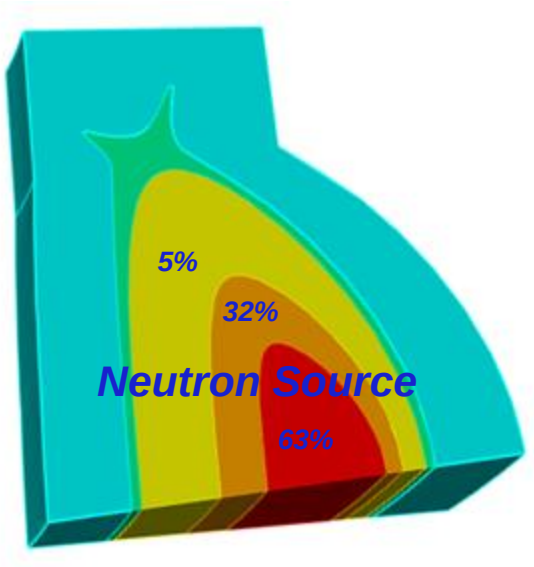
Cost of tritium and need to demonstrate T self-sufficiency motivate analysis of tritium breeding ratio (TBR)

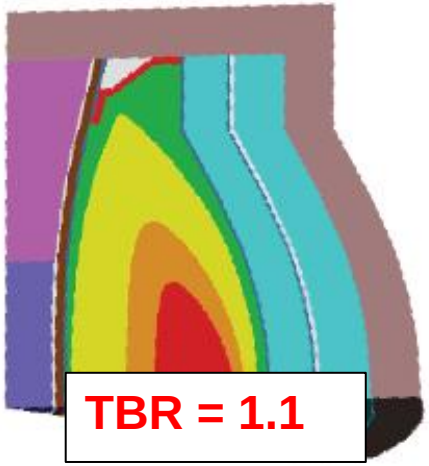
- Example costs of T w/o breeding at $0.1B/kg for R=1 → 1.6m
  - FNS mission: 1MWy/m^2 $0.33B → $0.9B
  - Component testing: 6MWy/m^2 $2B → $5.4B
- Implications:
  - TBR << 1 likely affordable for FNS mission with R ~ 1m
  - Component testing arguably requires TBR approaching 1 for all R
- Performing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



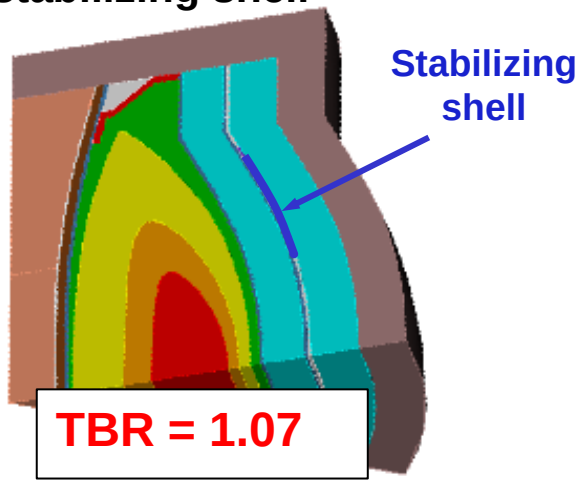


R=1.6m TBR calculations highlight importance of shells, penetrations, and top/bottom blankets

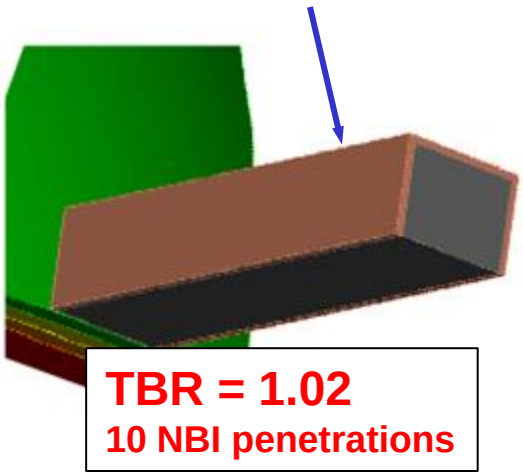
Extended conformal blanket



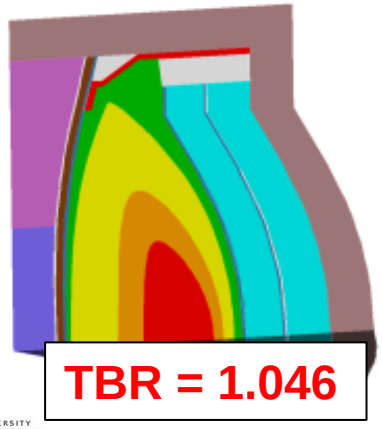
+ 3cm thick stabilizing shell



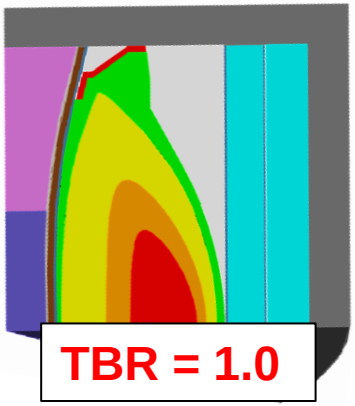
Extended conformal + 3cm shell + NBI



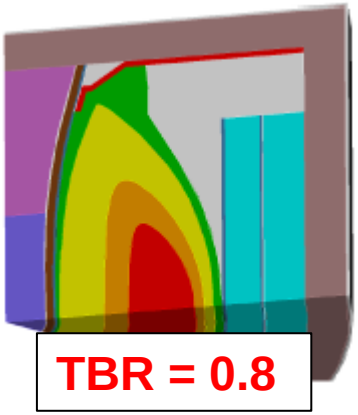
Conformal blanket



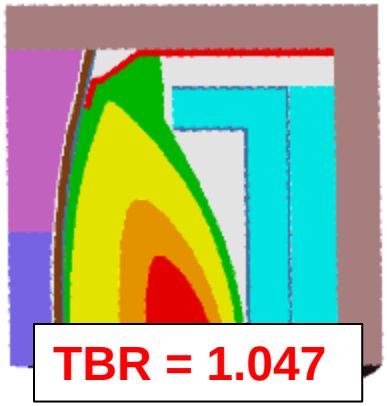
Extended straight blanket



Straight blanket



Straight blanket with flat top



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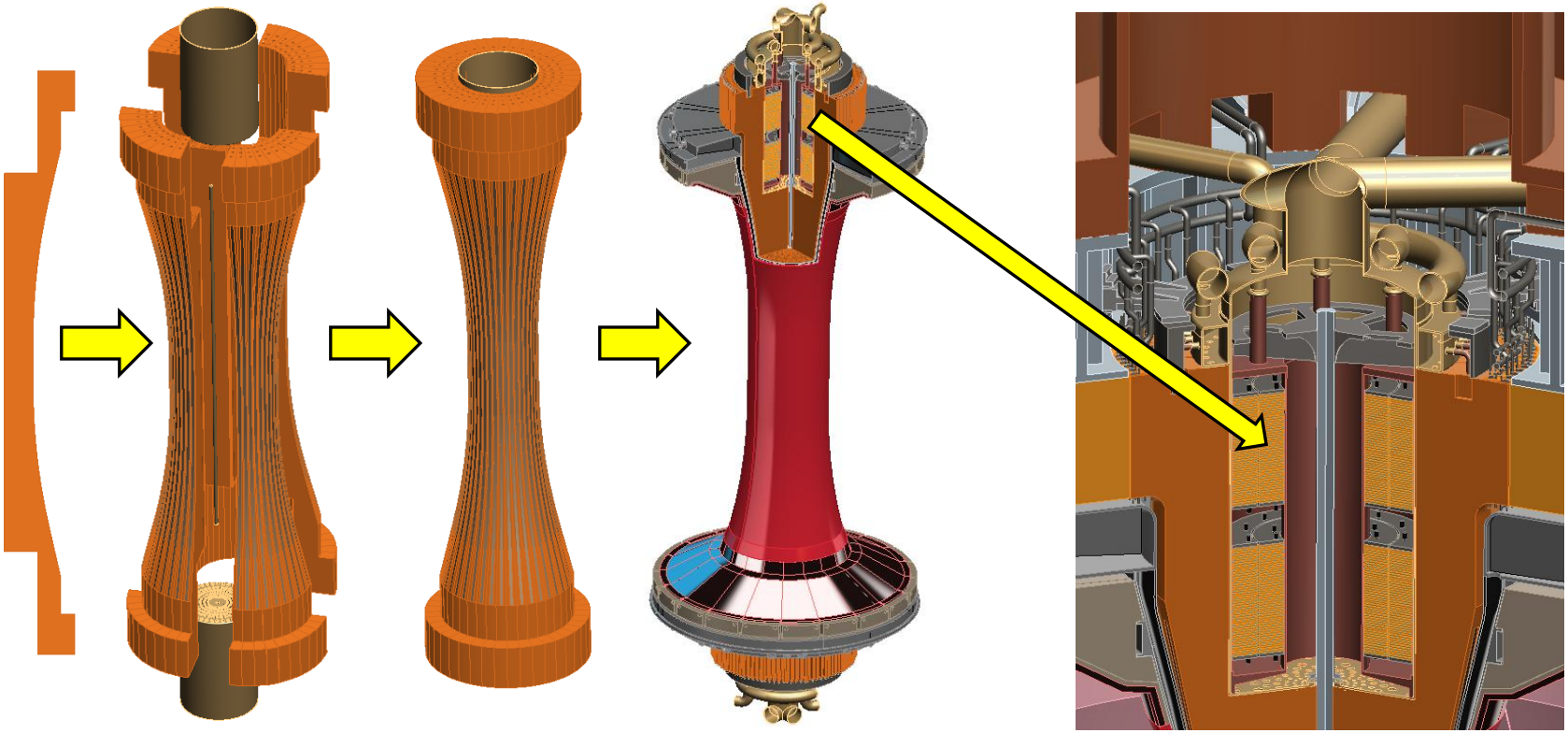
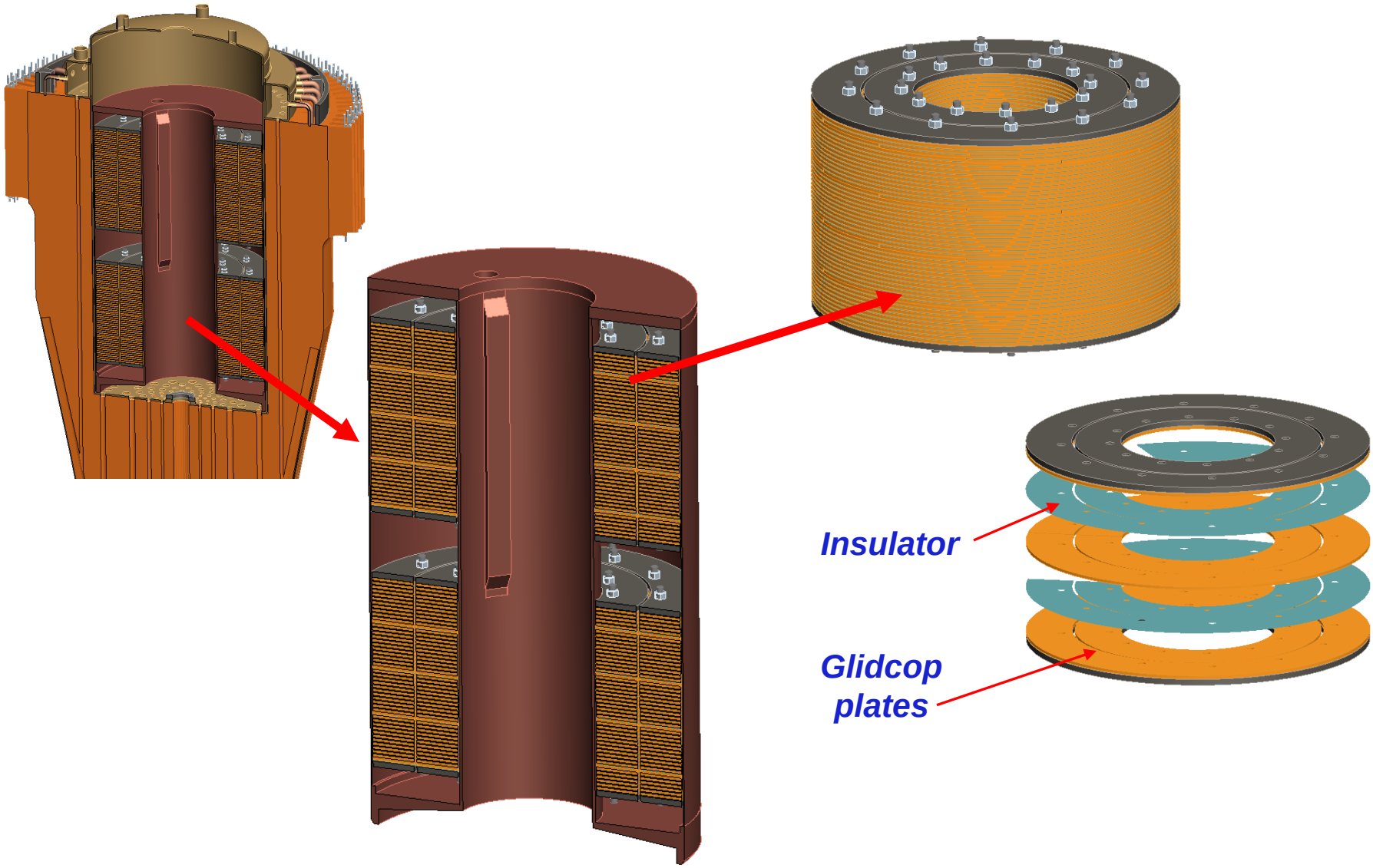
FNSF center-stack can build upon NSTX-U design and is incorporating NSTX stability results



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

Bitter coil insert for divertor coils in ends of TF

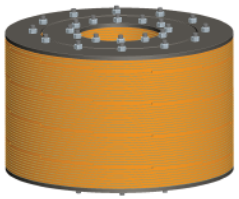


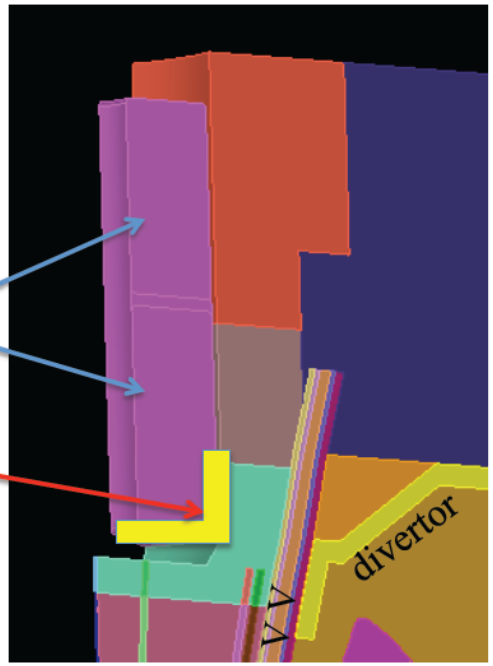
Neutronics analysis indicates organic insulator for divertor PF coils unacceptable

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





3-D Model

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

Peak dose to insulator = 3×10^{11} rad @ 1 FPY
= 1.8×10^{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

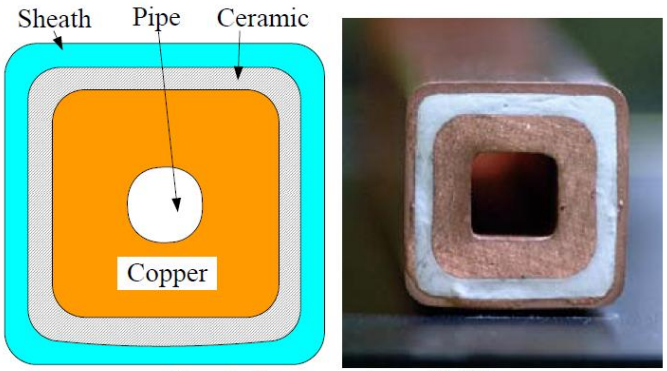


Fig. 3 Cross section of MIC

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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[B)] 2NEC/Token

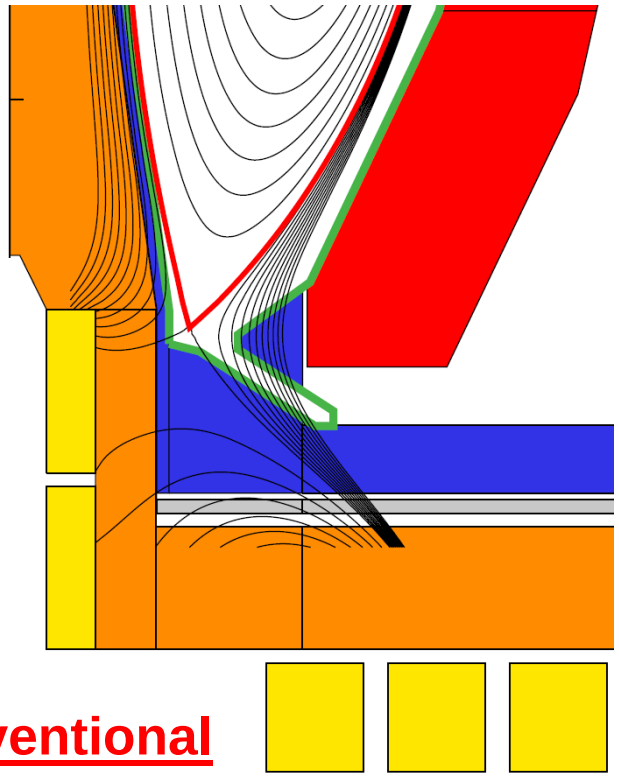
Proceedings of the 5th Annual Meeting of Particle Accelerator Society of Japan and the 33rd Linear Accelerator Meeting in Japan (August 6-8, 2008, Higashihiroshima, Japan)

- 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

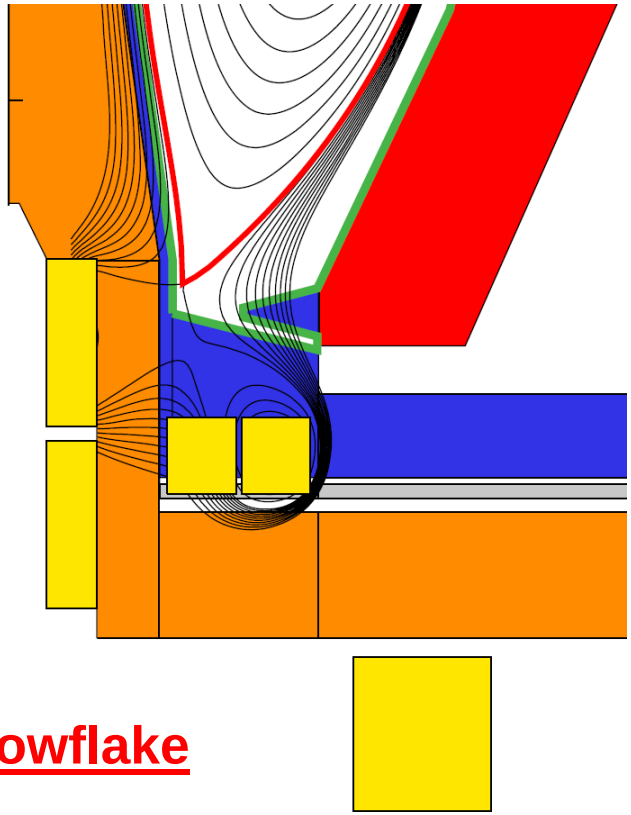
Outline

- 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

Divertor PF coil configurations identified to achieve high δ while maintaining peak divertor heat flux < 10MW/m^2



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



Conventional

- Flux expansion = 15-25, δ_x ~ 0.55
- $1/\sin(\theta_{plate})$ = 2-3
- Detachment, pumping questionable
  - Future: assess long-leg, V-shape divertor (JA)

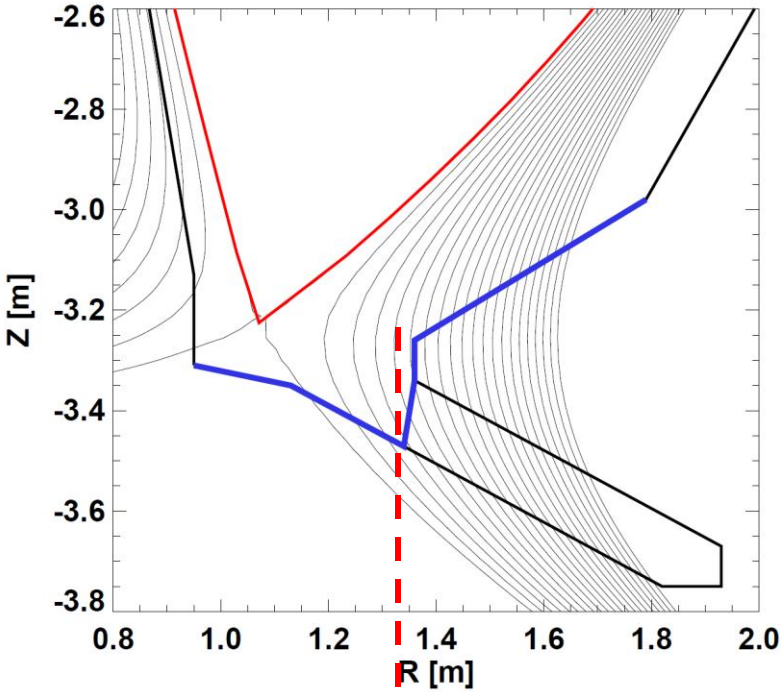
Snowflake

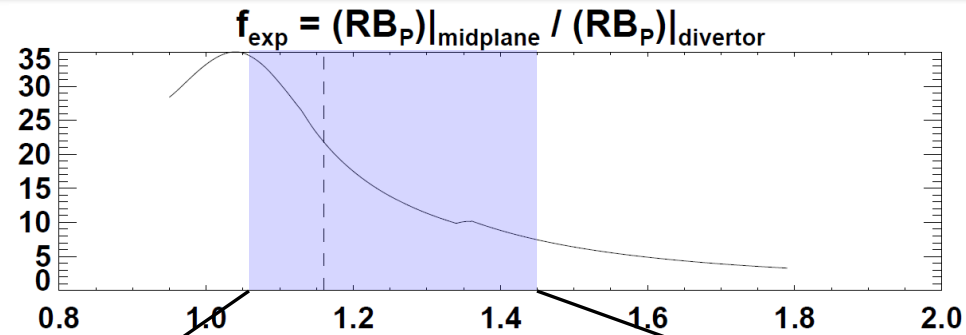
- Flux expansion = 40-60, δ_x ~ 0.62
- $1/\sin(\theta_{plate})$ = 1-1.5
- Good detachment (NSTX data) and cryo-pumping (NSTX-U modeling)

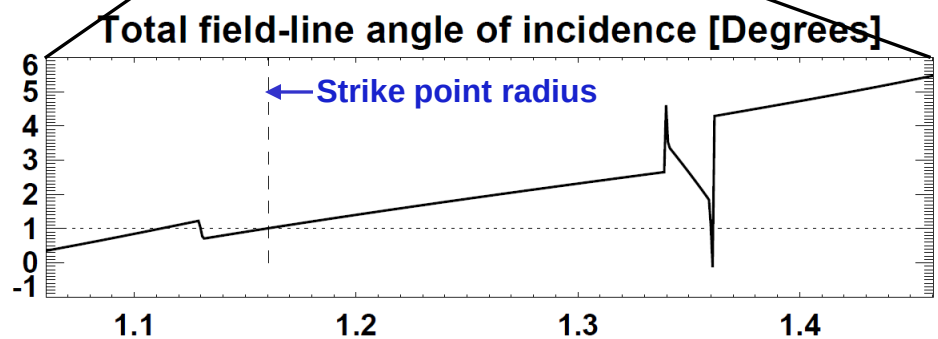
- **Will also test liquid metal PFCs in NSTX-U for power-handling, surface replenishment**

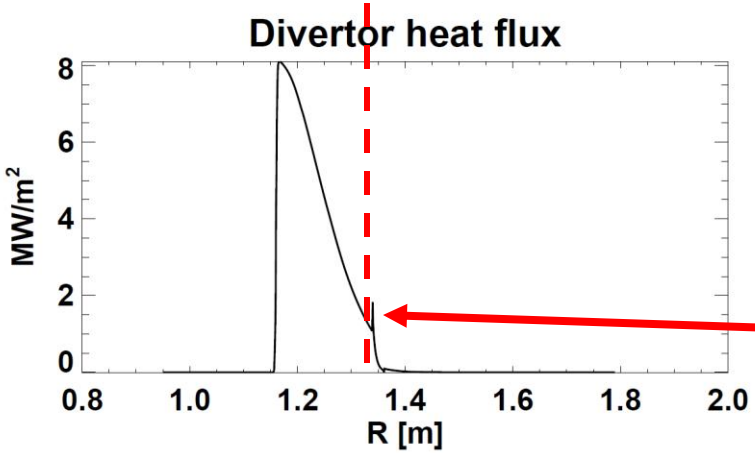
Jaworski - IAEA FEC EX/P5-31

Parameters and profiles for conventional divertor
(using simple exponential heat-flux profile in R=1.6m ST-FNSF)







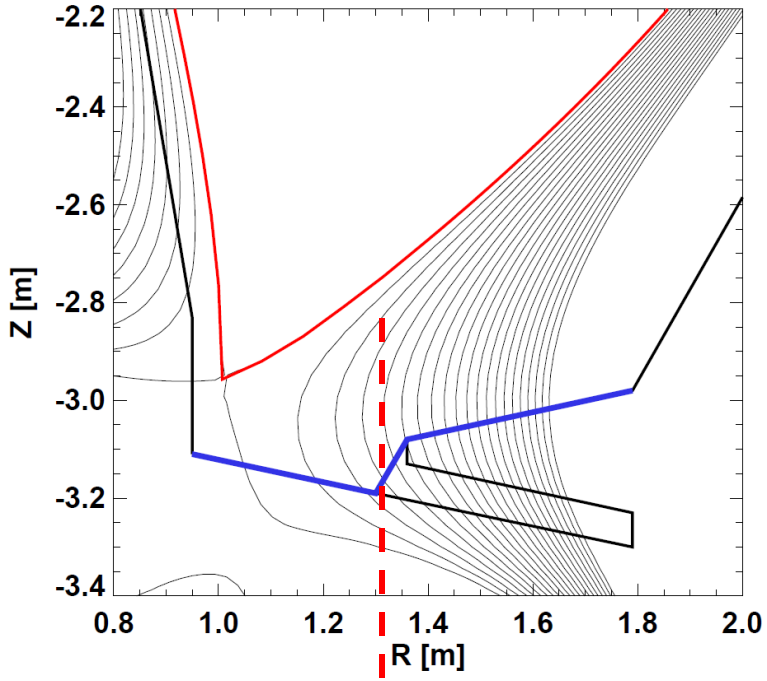


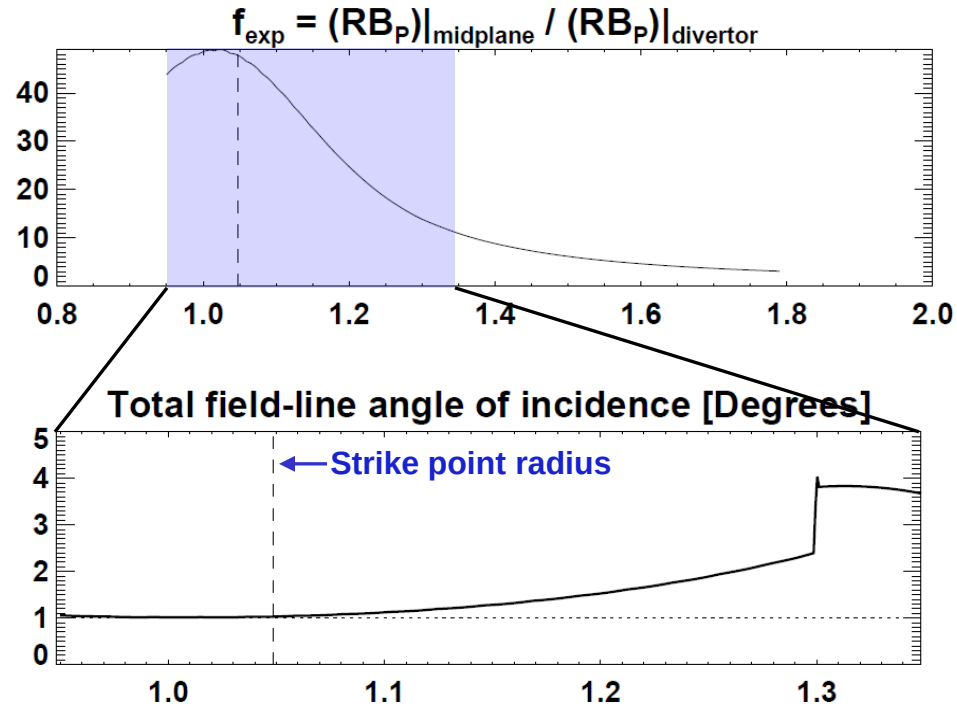
$$\text{Peak } q_{\perp\text{-div}} \approx \frac{P_{heat}\,(1-f_{rad})\,f_{obd}\,\sin(\theta_{pol})}{2\pi R_{strike}\,f_{exp}\,\lambda_{q\text{-mid}}\,N_{div}}$$

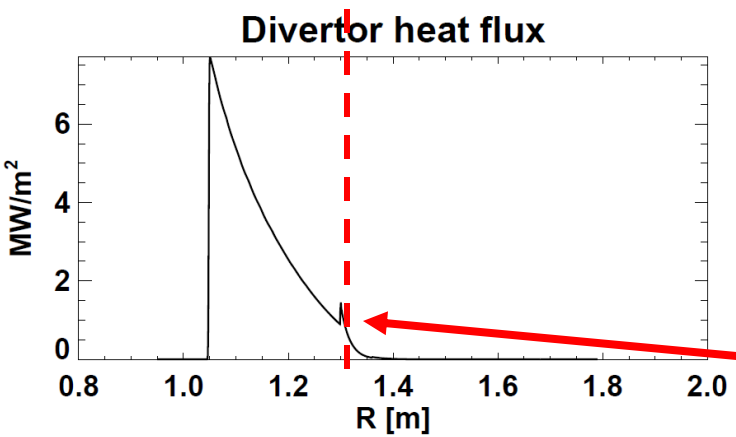
- P_{heat} = 115MW, f_{rad}=0.8, f_{obd}=0.8, $\sin(\theta_{pol})$ = 0.39
- R_{strike} = 1.16m, f_{exp} = 22, $\lambda_{q\text{-mid}}$ =2.7mm, N_{div} = 2

NSTX-U simulations find $q_\perp$ at pump entrance should be ≥ 1-2MW/m² for efficient pumping → guesstimate $R_{entrance}$ ~ 1.3m for R=1.6m ST-FNSF

Parameters and profiles for snowflake divertor
(using simple exponential heat-flux profile in R=1.6m ST-FNSF)







$$\text{Peak } q_{\perp\text{-div}} \approx \frac{P_{heat}\,(1-f_{rad})\,f_{obd}\,\sin(\theta_{pol})}{2\pi R_{strike}\,f_{exp}\,\lambda_{q\text{-mid}}\,N_{div}}$$

- P_{heat} = 115MW, f_{rad}=0.8, f_{obd}=0.8, $\sin(\theta_{pol})$ = 0.87
- R_{strike} = 1.05m, f_{exp} = 50, $\lambda_{q\text{-mid}}$ =2.7mm, N_{div} = 2

Snowflake would also want divertor cryo-plenum entrance radius $R_{entrance}$ ~ 1.3m for R=1.6m ST-FNSF

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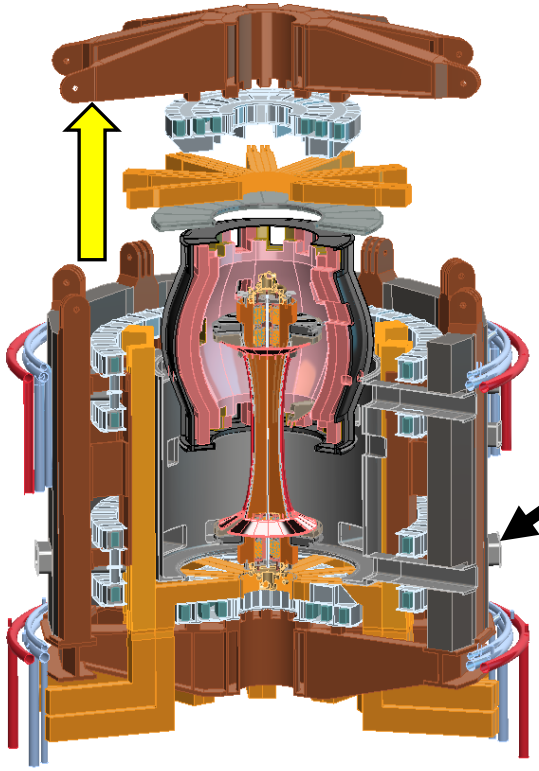
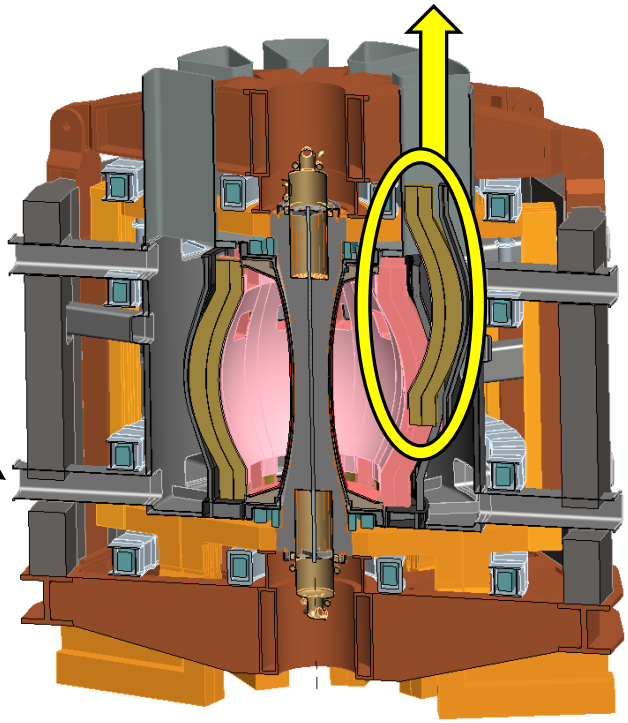
Flexible and efficient in-vessel access important for testing, replacing, improving components, maximizing availability

Several maintenance approaches under consideration:

- Vertically remove entire blanket and/or center-stack
  - **Better for full blanket replacement?**
- Translate blanket segments radially then vertically
  - **Better for more frequent blanket module replacement and/or repair?**

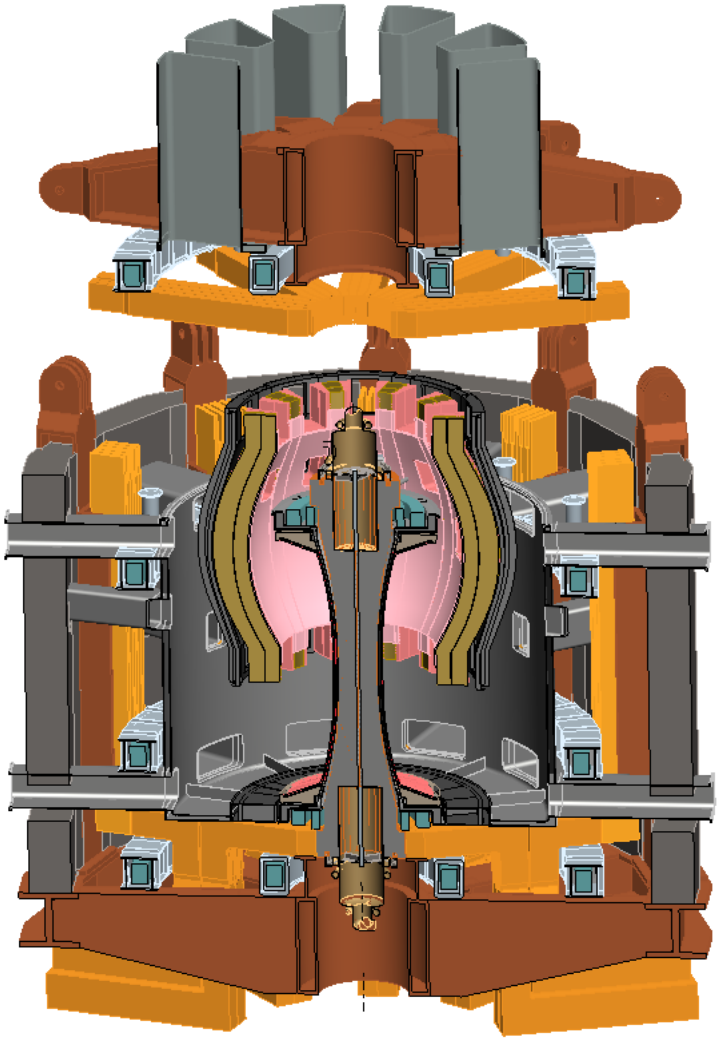


Radial ports for divertor maintenance or pumping

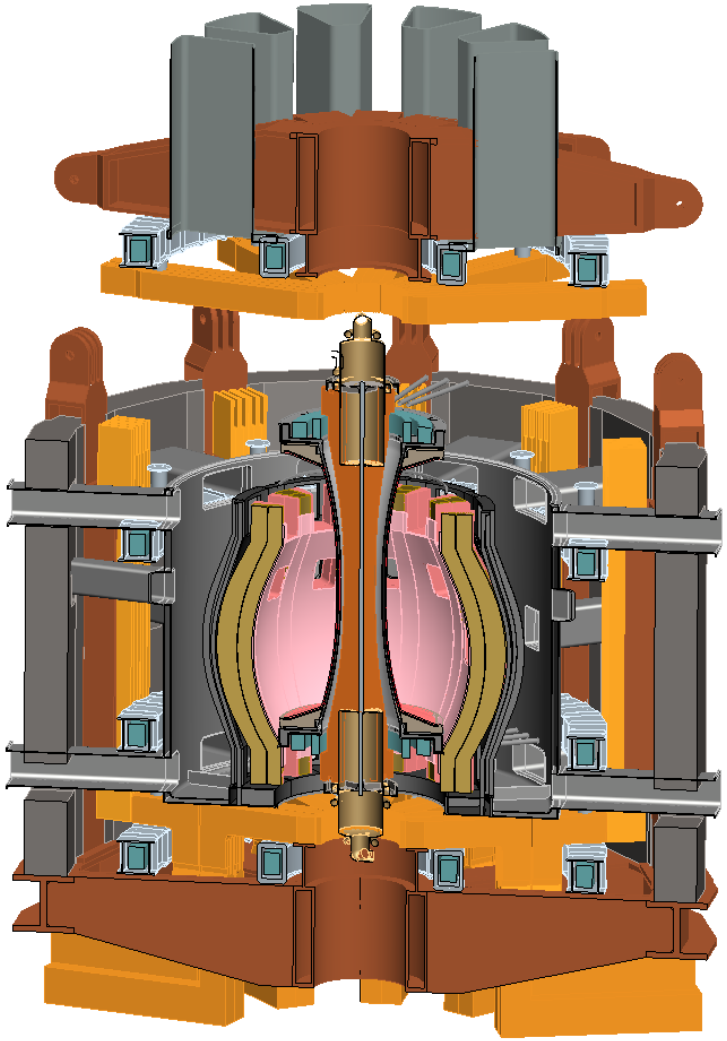


- **May be possible to combine features of both approaches**

Assembly and maintenance schemes with snowflake divertor and vertical ports

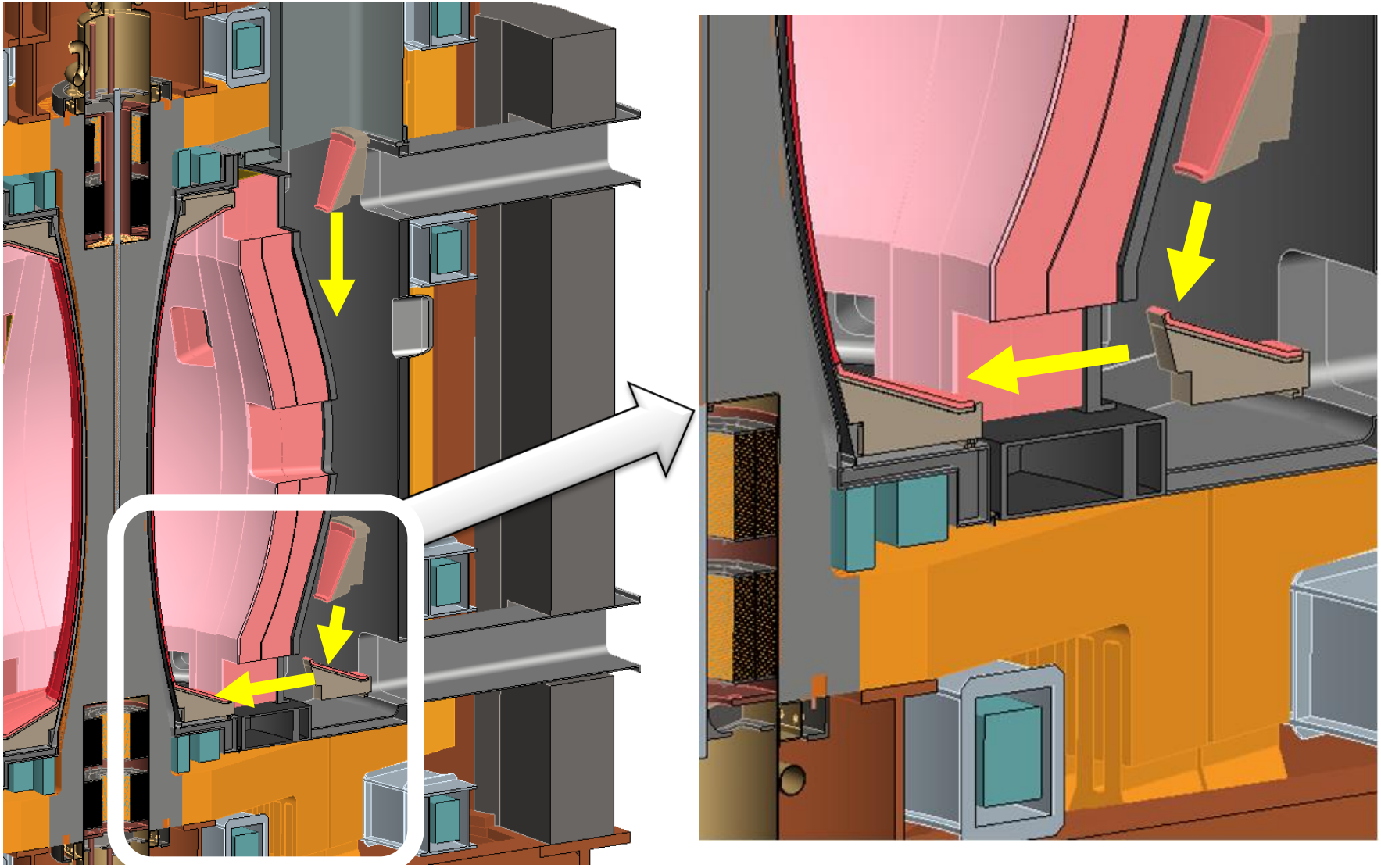


Full Blanket Assembly Removed

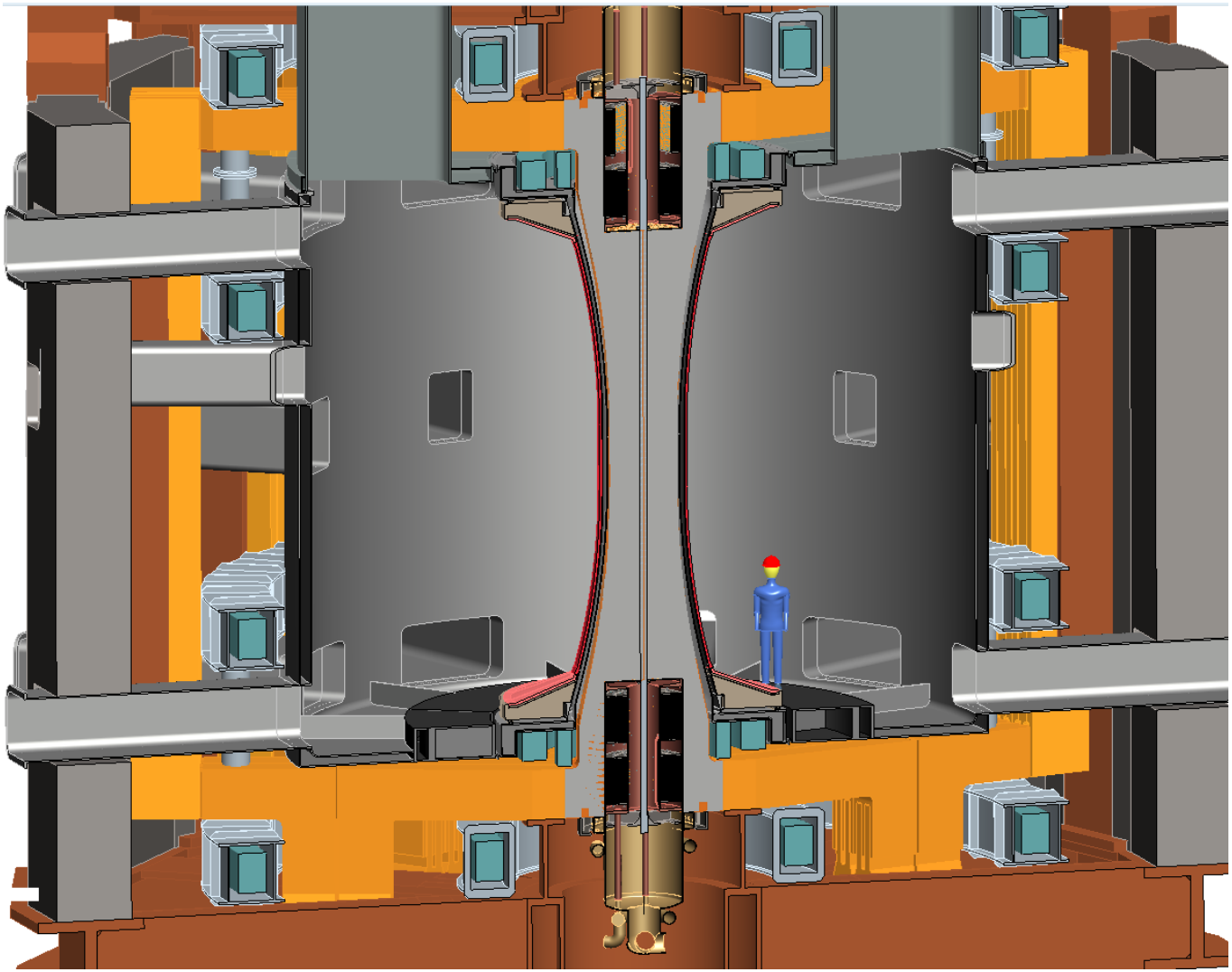


Centerstack Assembly Removed

Possible divertor module maintenance scheme using radial installation and vertical translation through vertical ports

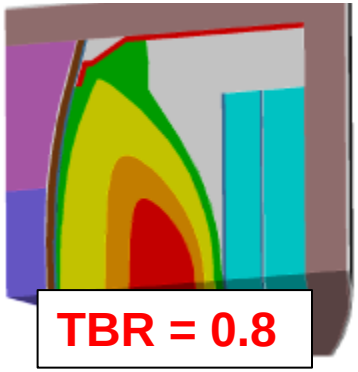


Large cylindrical vessel of R=1.6m FNSF could be used for PMI R&D (hot walls, Super-X?), other blanket configurations

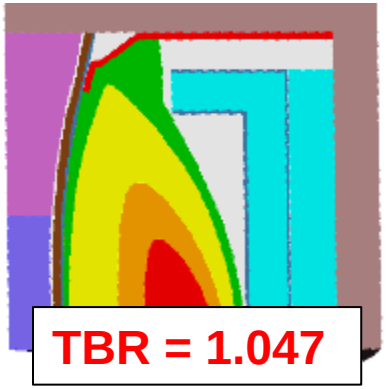


NOTE: TBR values do not include stabilizing shells or penetrations

Straight blanket



Straight blanket with flat top

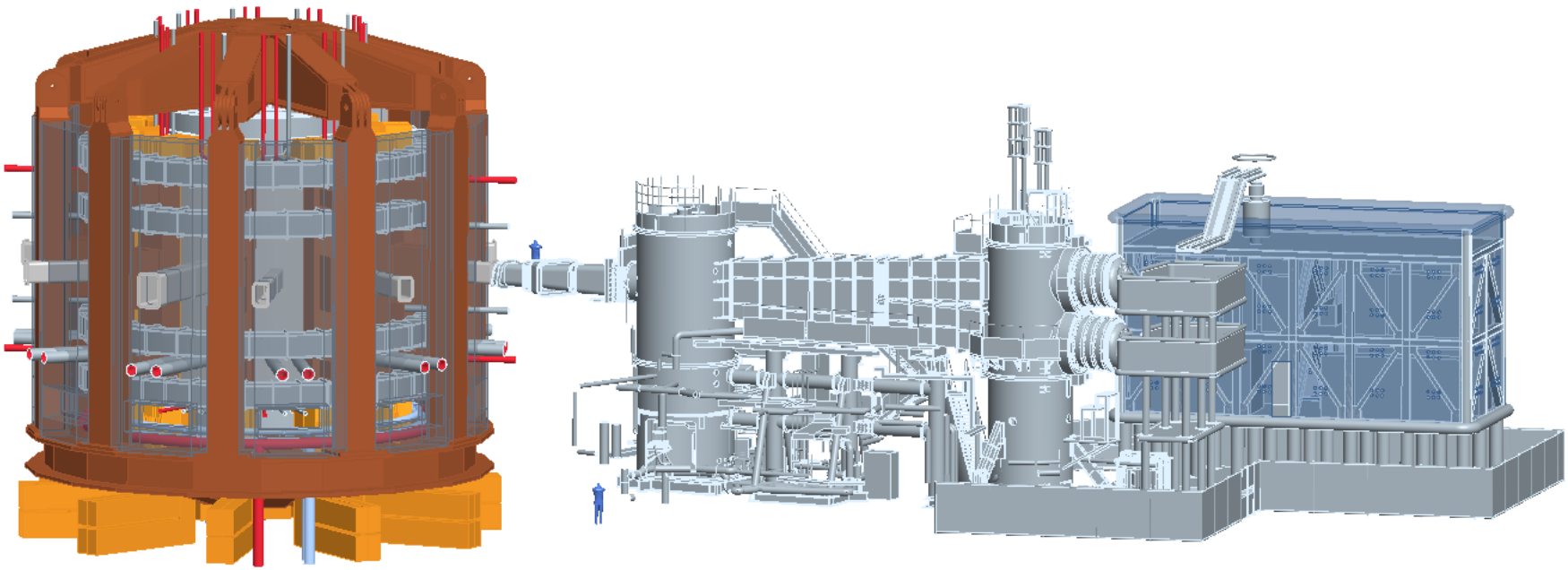


Summary

- 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 ~ 1 possible if top/bottom neutron losses are minimized
- Divertor PF coils in ends of TF bundle enable high δ, shaping
- Conventional and snowflake 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
(a highly incomplete list!)

- 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
- Divertor poloidal field coil layout and design
  - Assess radiation induced conductivity (RIC) in ceramic insulators
- Power exhaust calculations
  - Perform pumping and detachment calculations, include Super-X divertor
- Maintenance strategies
  - Assess space/lifting requirements above machine for vertical maintenance

Backup

R=1.6 m ST FNSF with JT-60SA NNBI system