Configuration Studies for an ST-Based Fusion Nuclear Science Facility (FNSF)

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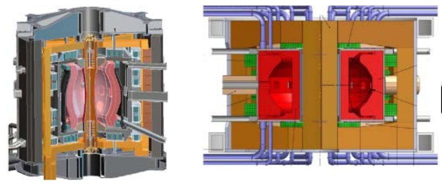
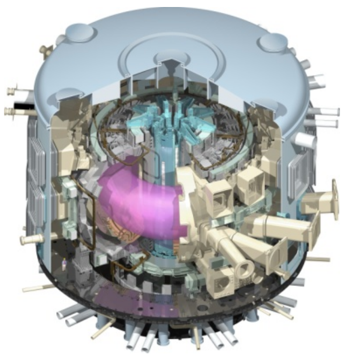
There are several possible pathways from ITER to a commercial fusion power plant

FNSF = Fusion Nuclear Science Facility
CTF = Component Test Facility



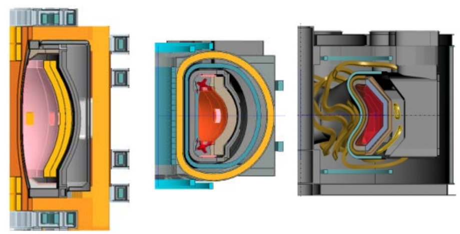
DEMO

ITER



FNSF/CTF

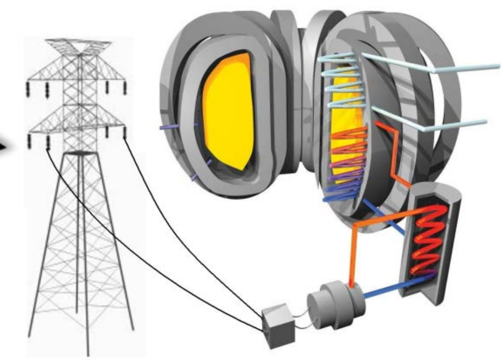
Blanket R&D, T self-sufficiency



Pilot Plant

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

First of a kind Power Plant

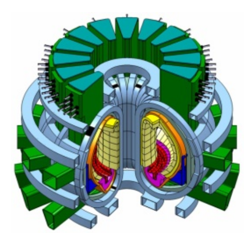


Supporting Physics and Technology

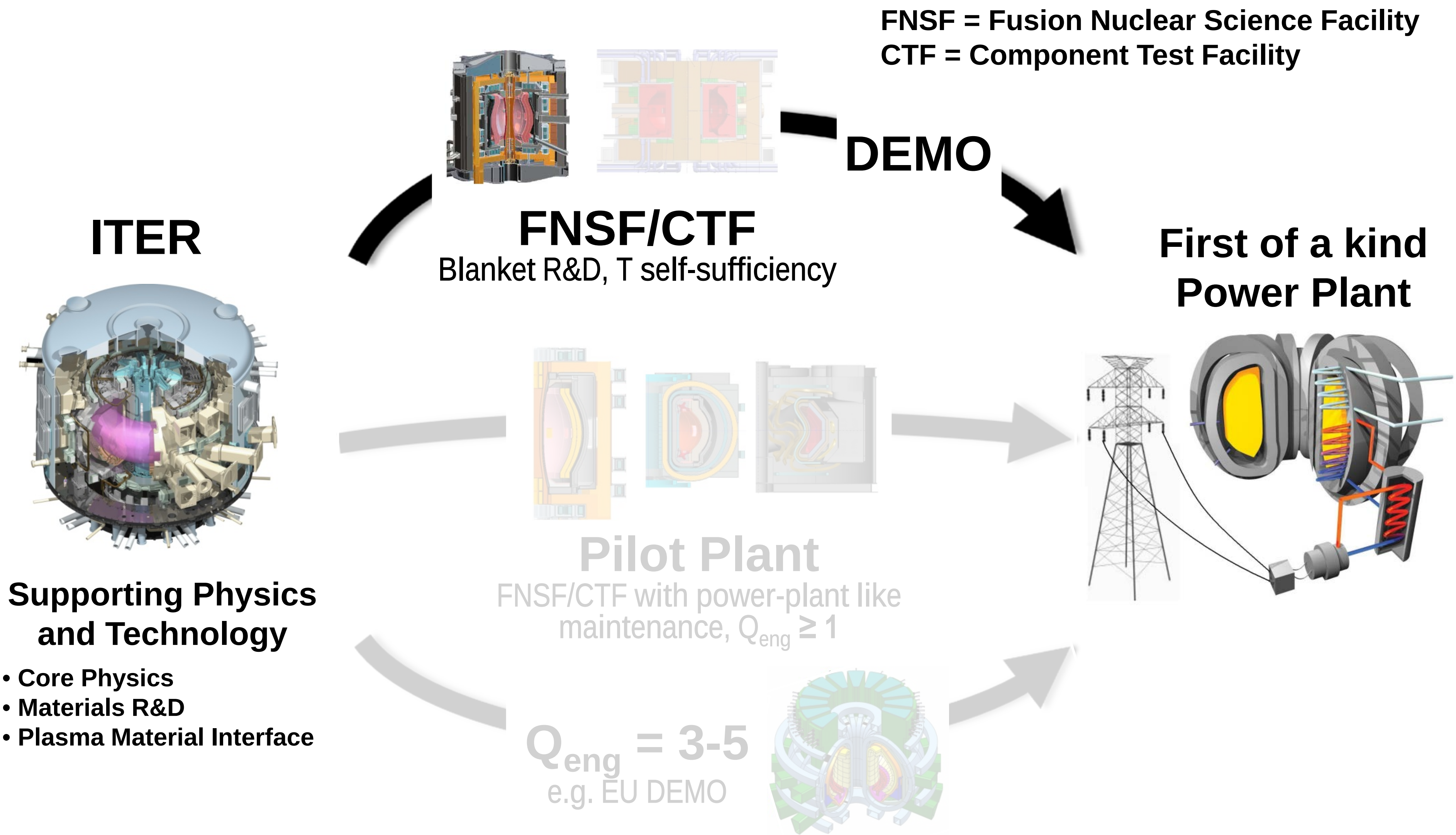
- **Core Physics**
- **Materials R&D**
- **Plasma Material Interface**

Q_{eng} = 3-5

e.g. EU DEMO



This talk considers possible spherical tokamak (ST) Fusion Nuclear Science Facility (FNSF) options



Overview

- Recent U.S. studies for ST-FNSF have focused on assessing achievable missions versus device size
- Possible missions:
  - Electricity break-even
    - Motivated 2010-12 analysis of R=2.2m ST Pilot Plant
  - Tritium self-sufficiency (tritium breeding ratio TBR ≥ 1)
    - Motivates present (2013-14) analysis of R=1m, 1.7m ST FNSF devices to address key questions:
      - How large must ST device be to achieve TBR ≥ 1?
      - How much externally supplied T would be needed for smaller ST?
      - What are device and component lifetimes?
  - Fusion-relevant neutron wall loading and fluence
    - STs studied here access 1MW/m^2, 6MW-yr/m^2 (surface-avg. values)

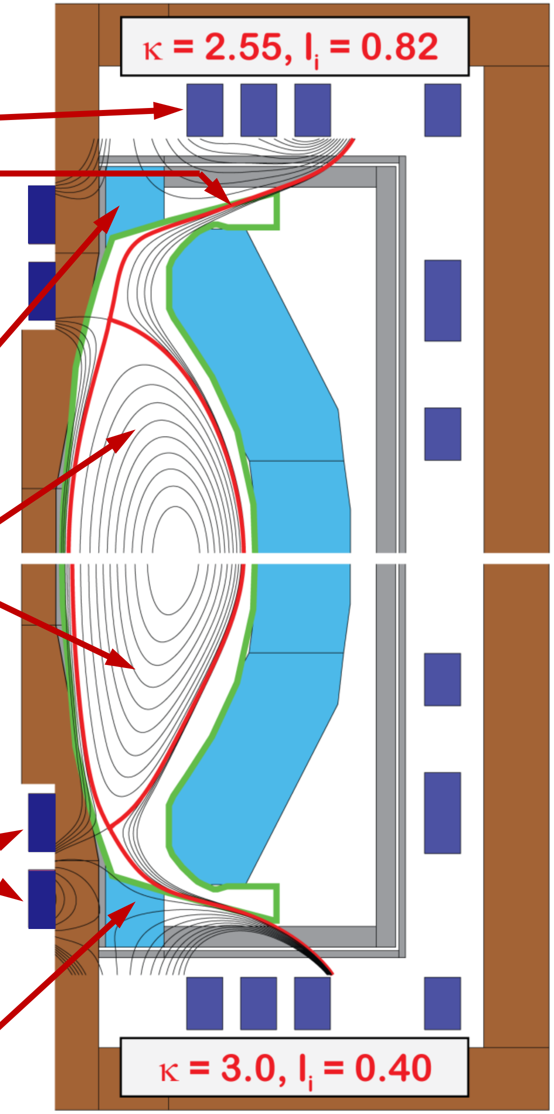
Outline

- Physics design
- Configuration, shielding, tritium breeding
- Conclusions

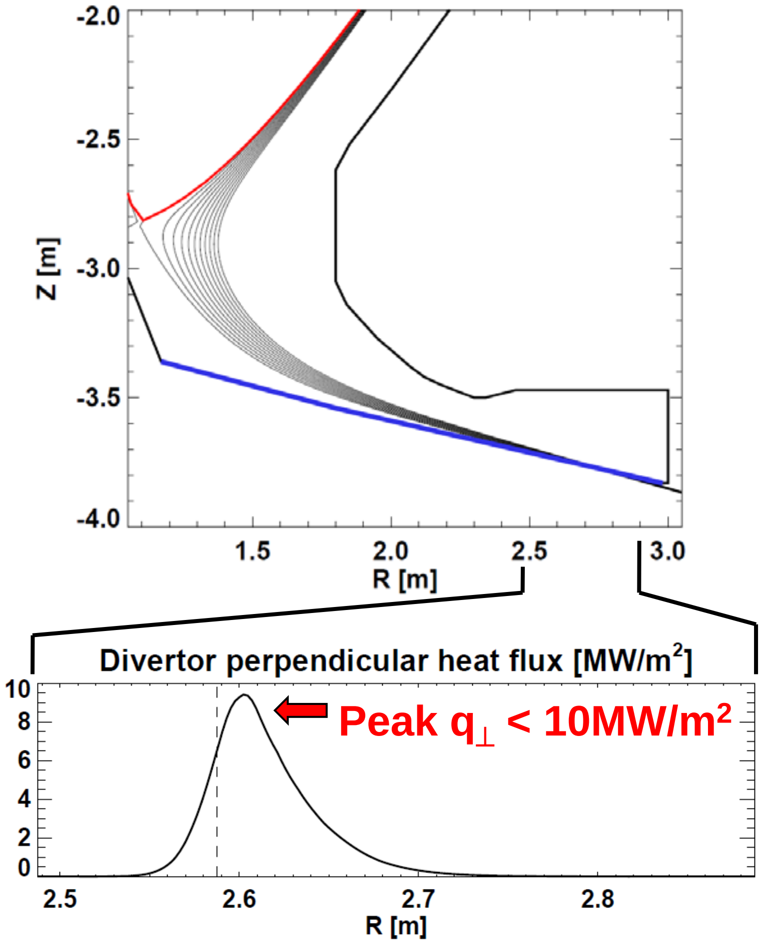
PF coil set identified that supports combined Super-X + snowflake divertor for range of equilibria

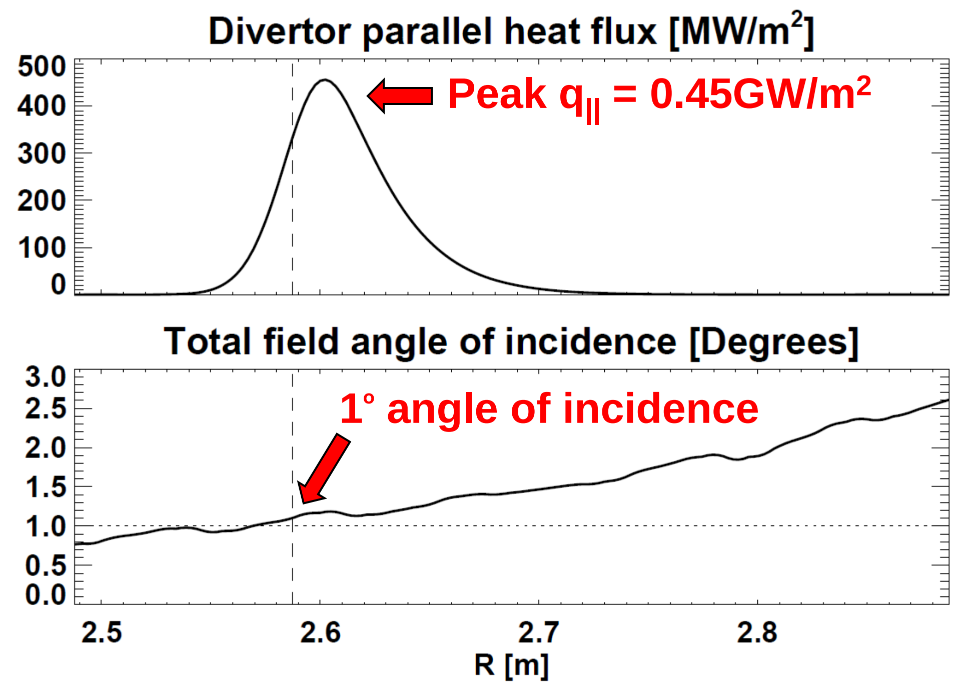
Components: TF coil | PF coil | Vessel | Blanket

- All equilibrium PF coils outside vacuum vessel
- Increased strike-point radius reduces B, $q_{||}$
  Strike-point PFCs also shielded by blankets
- 2nd X-point/snowflake increases SOL line-length
- PF coil set supports wide range of l_i: 0.4 – 0.8
  - Elongation and squareness change with l_i variation
  - Fixed strike-point R, controllable B-field angle of incidence (0.5-5˚)
- Divertor coils in TF coil ends for equilibrium, high δ
- Breeding in CS ends important for maximizing TBR



Up/down-symmetric Super-X/snowflake → $q_{\perp\text{-divertor}} < 10\,\text{MW/m}^2$ even under attached conditions (if integral heat-flux width $\lambda_{q\text{-int}} > 2$mm)





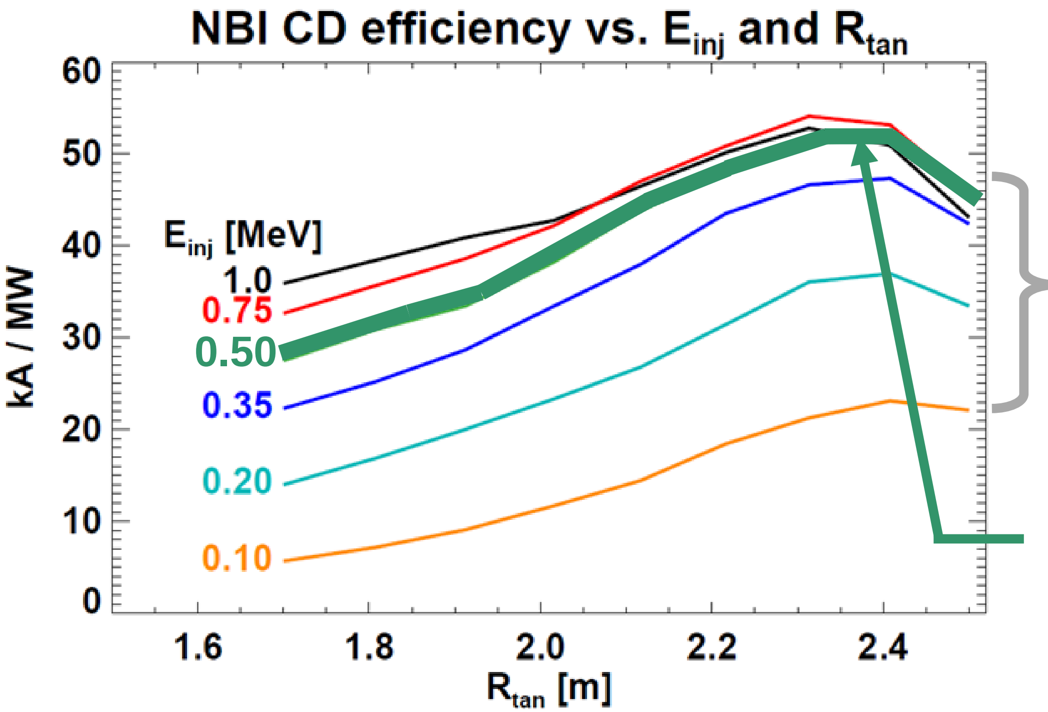
Partial detachment expected to further reduce peak $q_\perp$ factor of 2-5×

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

- P_{heat} = 115MW, f_{rad}=0.8, f_{obd}=0.8, θ_{pol} = 2.1°
- R_{strike} = 2.6m, f_{exp} = 1.4, $\lambda_{q\text{-int}}$ =2.05mm, N_{div} = 2

Eich NF 2013: $\lambda_{q\text{-int}} = \lambda_q + 1.64 \times S$, **$\lambda_q$ = 0.78mm, S ≈ λ_q** (closed divertor)

0.5 MeV NNBI favorable for heating and current drive (CD) for R=1.7m ST-FNSF



- Fixed target parameters in DD:
  - I_P = 7.5MA, β_N = 4.5, l_i = 0.5
  - $n_e / n_{Greenwald}$ = 0.75, $H_{98y,2}$ = 1.5
  - A=1.75, R=1.7m, B_T = 3T, κ = 2.8
  - $\langle T_e \rangle$=5.8keV, $\langle T_i \rangle$=7.4keV

Optimal tangency radii:

1.7m ≤ R_{tan} ≤ 2.4m

Control $q(0)$, q_{min}

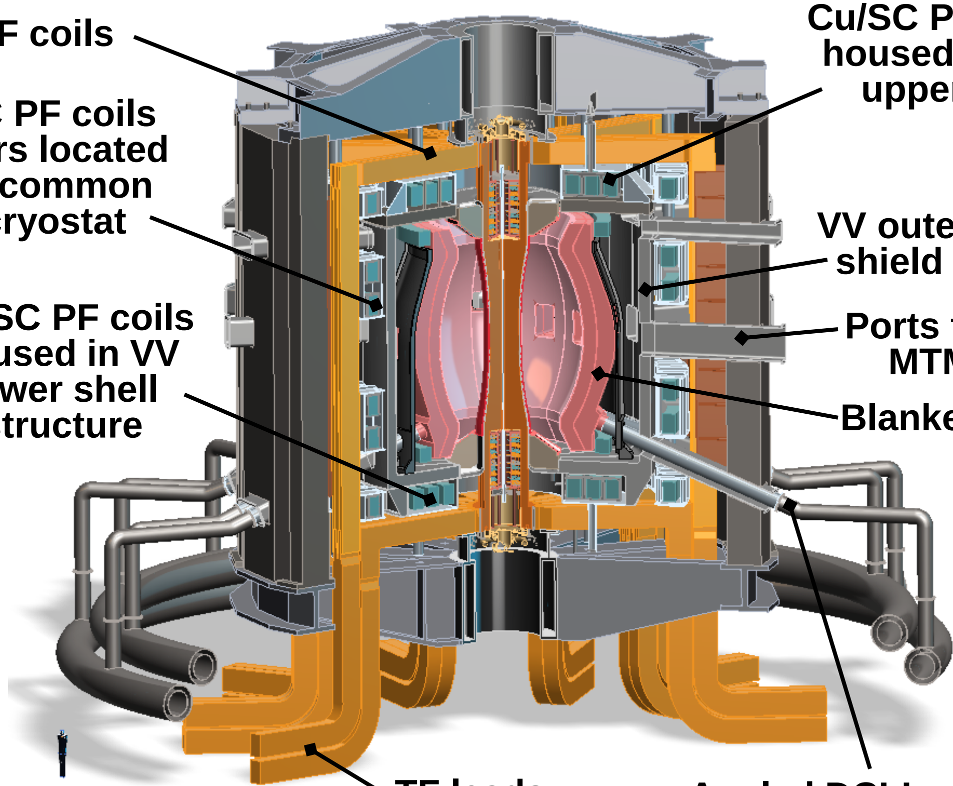
Shine-thru limit

Outline

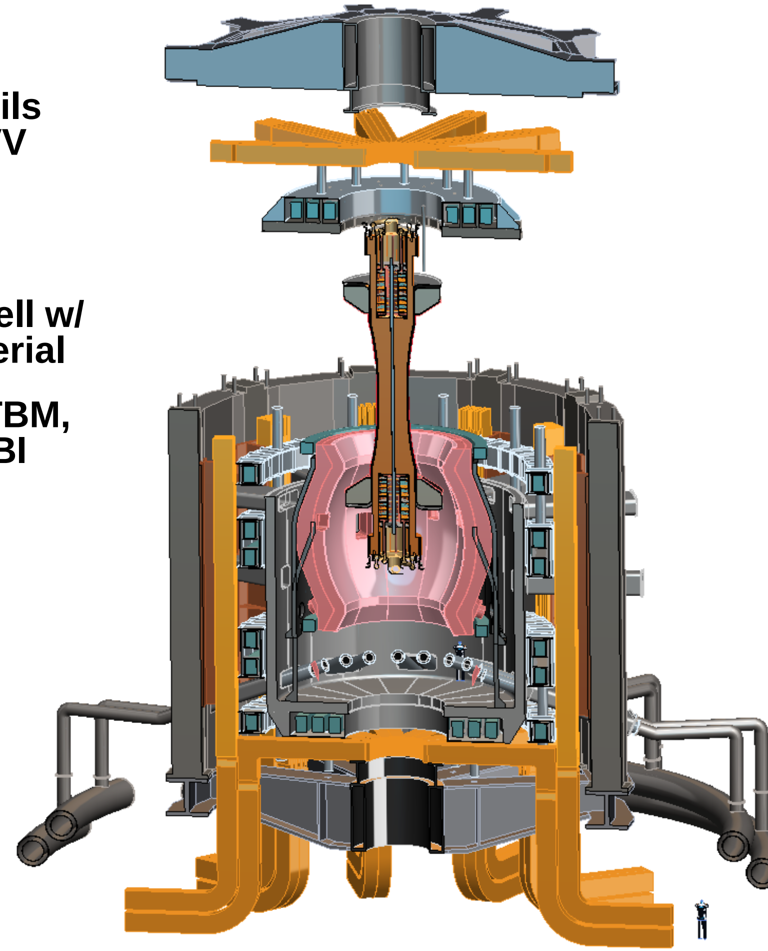
- Physics design
- Configuration, shielding, tritium breeding
- Conclusions

R=1.7m configuration with Super-X divertor

Design features

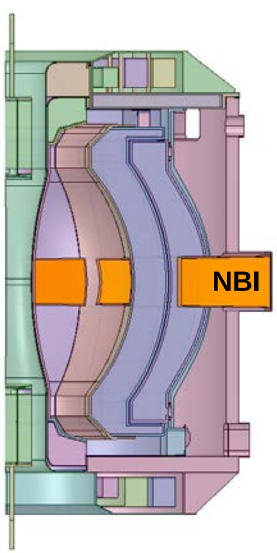


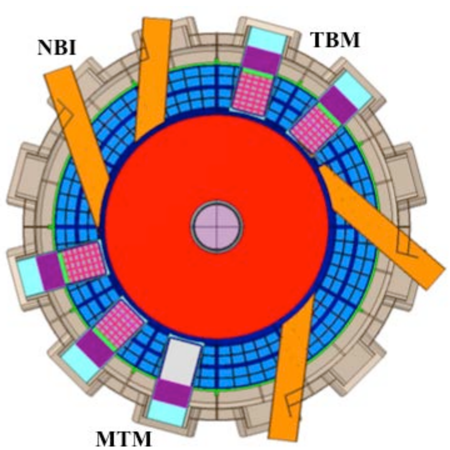
Vertical maintenance

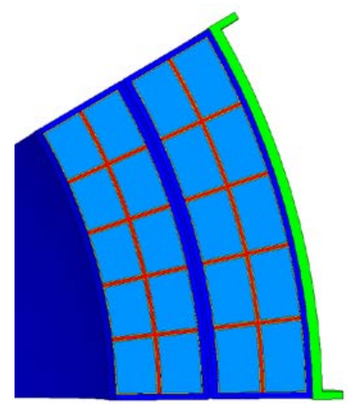


ST-FNSF shielding and TBR analyzed with sophisticated 3-D neutronics codes

- CAD coupled with MCNP using UW DAGMC code
- Fully accurate representation of <u>entire torus</u>
- No approximation/simplification involved at any step:
  - Internals of two OB DCLL blanket segments modeled in great detail, including:
    - FW, side, top/bottom, and back walls, cooling channels, SiC FCI
  - 2 cm wide assembly gaps between toroidal sectors
  - 2 cm thick W vertical stabilizing shell between OB blanket segments
  - Ports and FS walls for test blanket / materials test modules (TBM/MTM) and NNBI







Heterogeneous OB Blanket Model, including FW, side/back/top/bottom walls, cooling channels, and SiC FCI

TBM

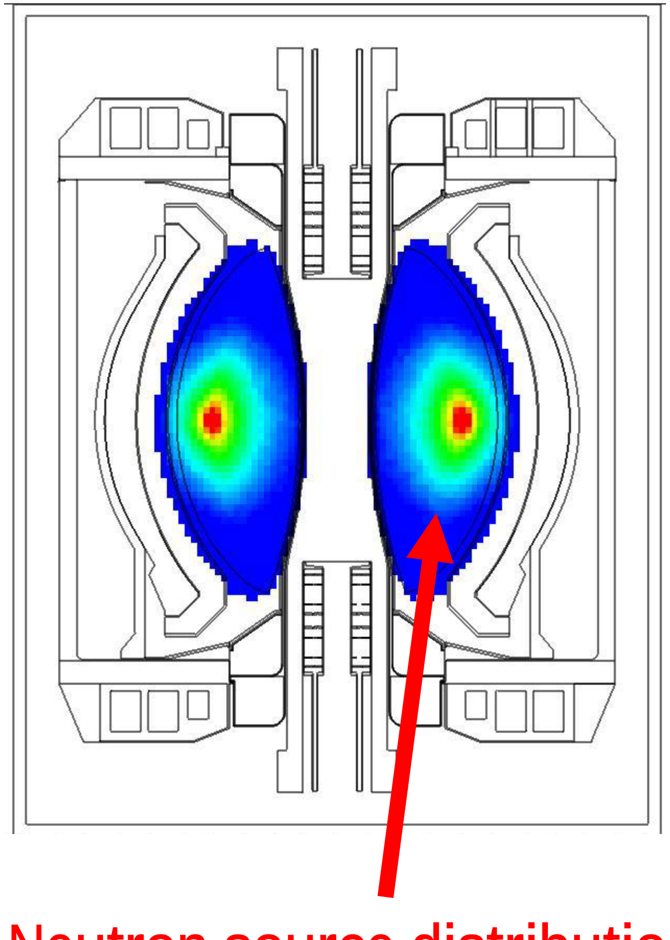
Two sizes (R=1.7m, 1m) assessed for shielding, TBR

Parameter:

Major Radius	**1.68m**	**1.0m**
Minor Radius	**0.95m**	**0.6m**
Fusion Power	**162MW**	**62MW**
Wall loading (avg)	**1MW/m²**	**1MW/m²**
TF coils	**12**	**10**
TBM ports	**4**	**4**
MTM ports	**1**	**1**
NBI ports	**4**	**3**

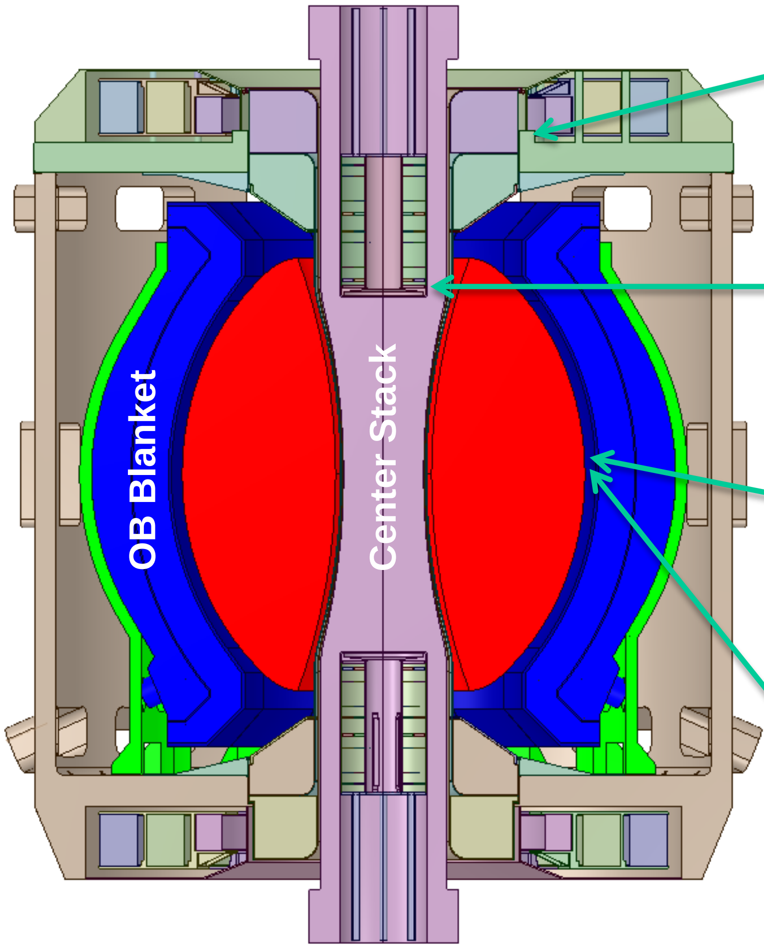
Plant Lifetime ~20 years

Availability 10-50%, 30% avg } **6 Full Power Years (FPY)**



Peak Damage at OB FW and Insulator of Cu Magnets

R=1.7m configuration



Dose to MgO insulator = 2×10^{8} Gy @ 6 FPY $< 10^{11}$ Gy limit

Dose to MgO insulator = 6×10^{9} Gy @ 6 FPY $< 10^{11}$ Gy limit

Peak dpa at OB midplane = 15.5 dpa / FPY

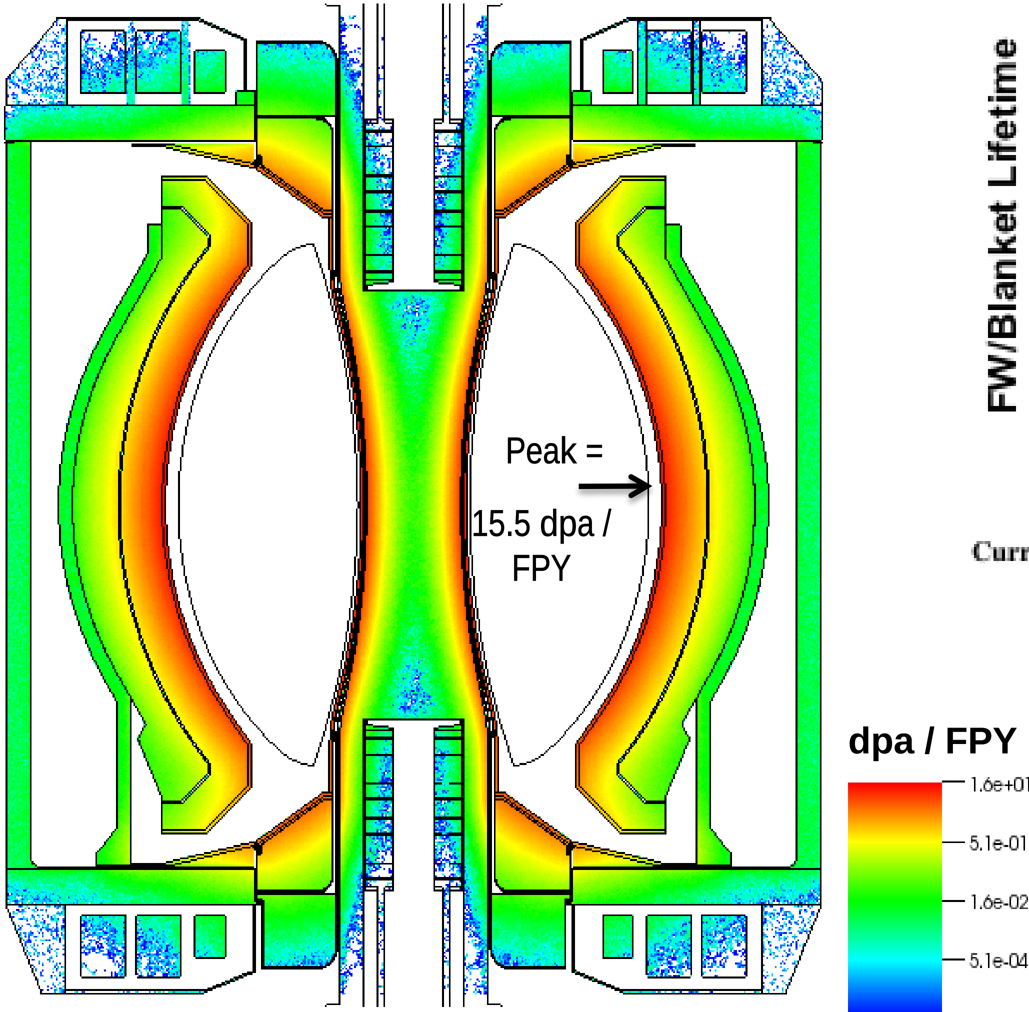
Peak He production at OB midplane = 174 appm/FPY

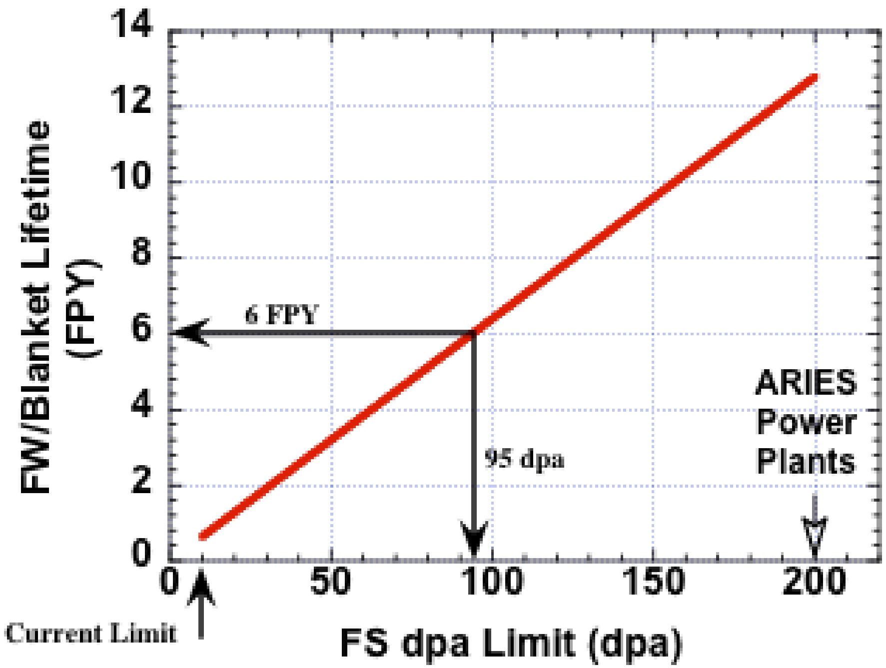
$\Rightarrow$ **He/dpa ratio = 11.2**

3-D Neutronics Model of Entire Torus

Mapping of dpa and FW/blanket lifetime
(R=1.7 m Device)

R=1.7m configuration

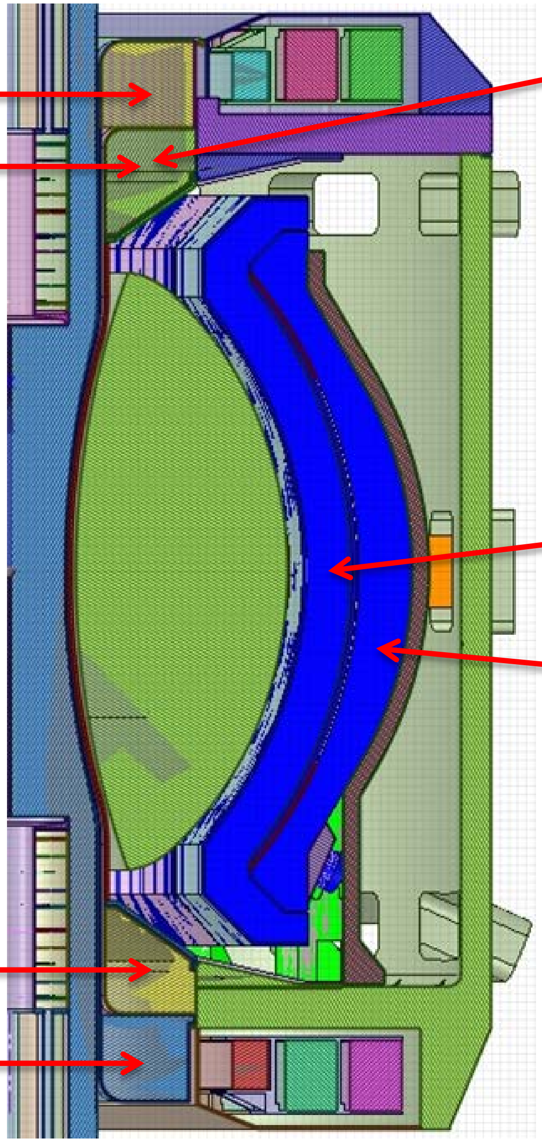




FW/blanket could operate for 6 FPY if allowable damage limit is 95 dpa

→ Peak EOL Fluence = 11 MWy/m^2

TBR contributions by blanket region



0.0004

0.034

0.034

0.0004

Breeding at CS ends important:
ΔTBR = +0.07

Inner Blanket Segment = 0.81

Outer Blanket Segment = 0.15

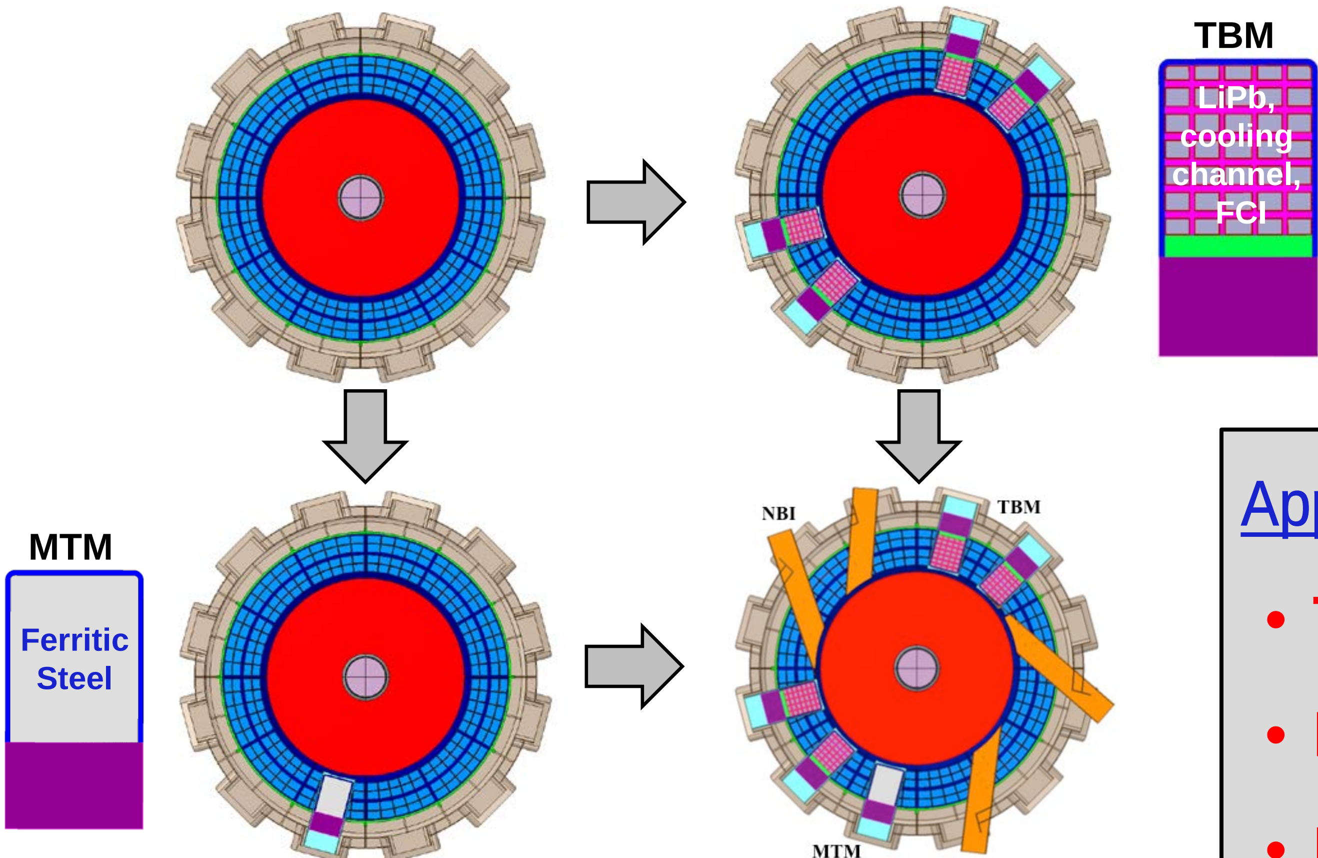
Total TBR ~ 1.03 with no penetrations or ports
(heterogenous outboard blanket)

R=1.7m configuration

Impact of TBM, MTM, NBI ports on TBR

No ports or penetrations, homogeneous breeding zones:
TBR = 1.03

Add 4 Test Blanket Modules (TBMs)
TBR = 1.02 (ΔTBR = -0.01)



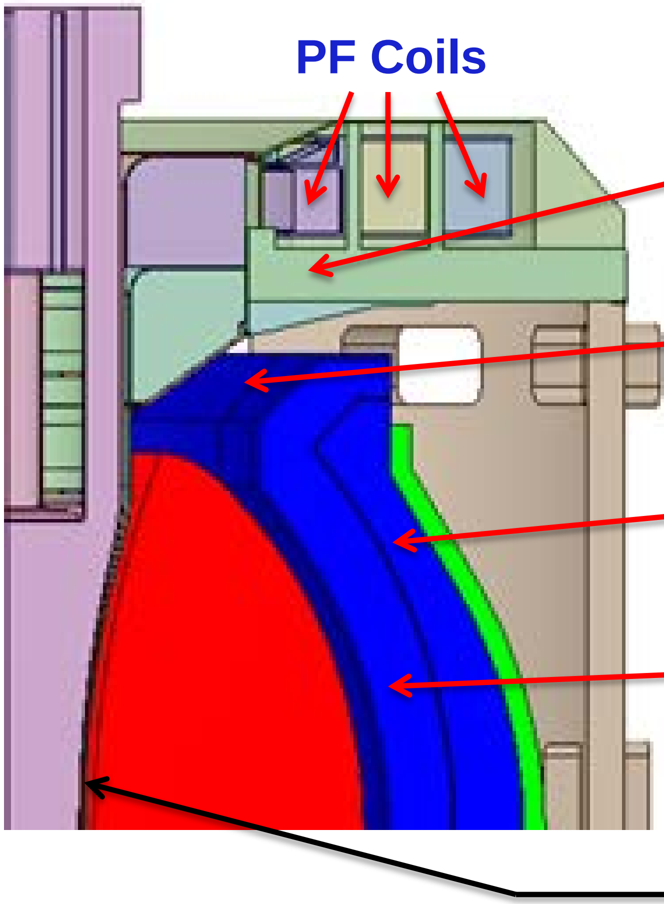
1 Materials Test Module (MTM)
TBR = 1.01 (ΔTBR = -0.02)

4 TBM + 1 MTM + 4 NBI
TBR = 0.97

Approx. ΔTBR per port:

- **TBM: -0.25%**
- **MTM: -2.0%**
- **NBI: -0.75%**

Options to increase TBR > 1

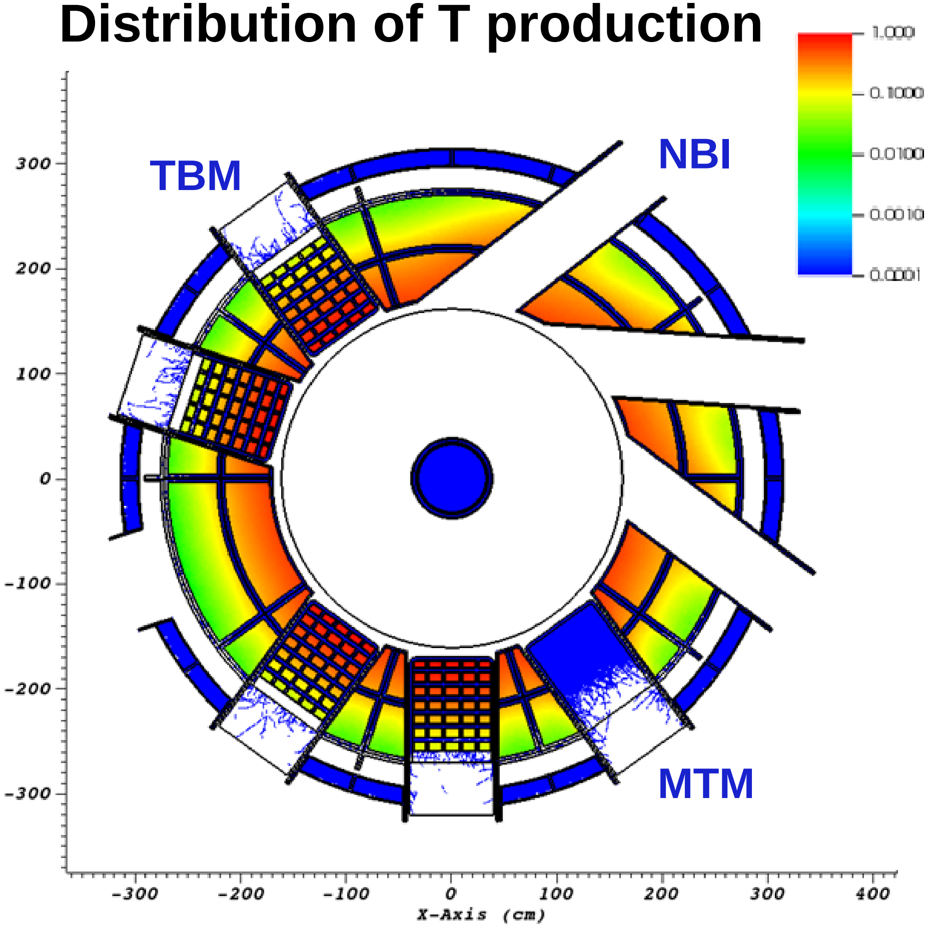


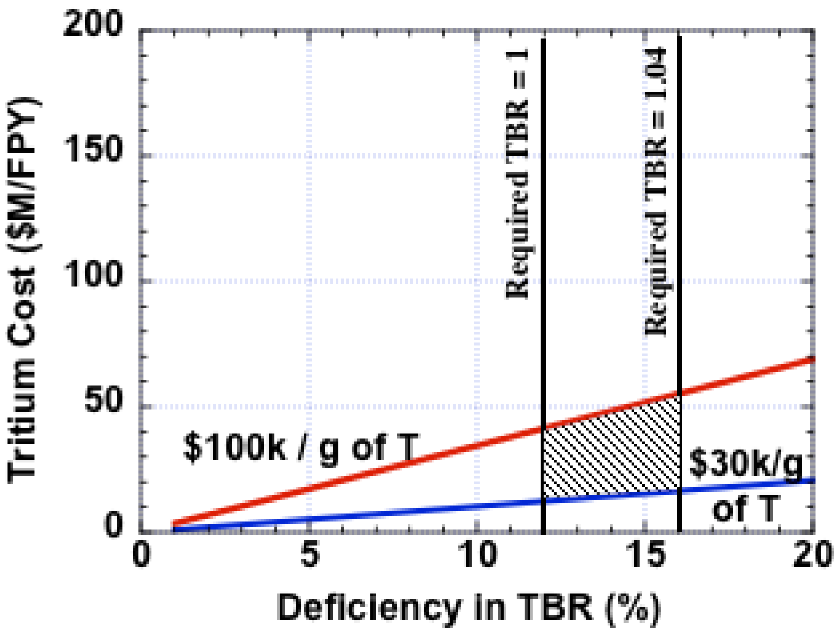
- Add to PF coil shield a thin breeding blanket (ΔTBR ~ +3%)
- Smaller opening to divertor to reduce neutron leakage
- Uniform OB blanket (1m thick everywhere; no thinning)
- Reduce cooling channels and FCIs within blanket (need thermal analysis to confirm)
- Thicker IB VV with breeding

Potential for TBR > 1 at R=1.7m

R_0 = 1m ST-FNSF achieves TBR = 0.88

Distribution of T production





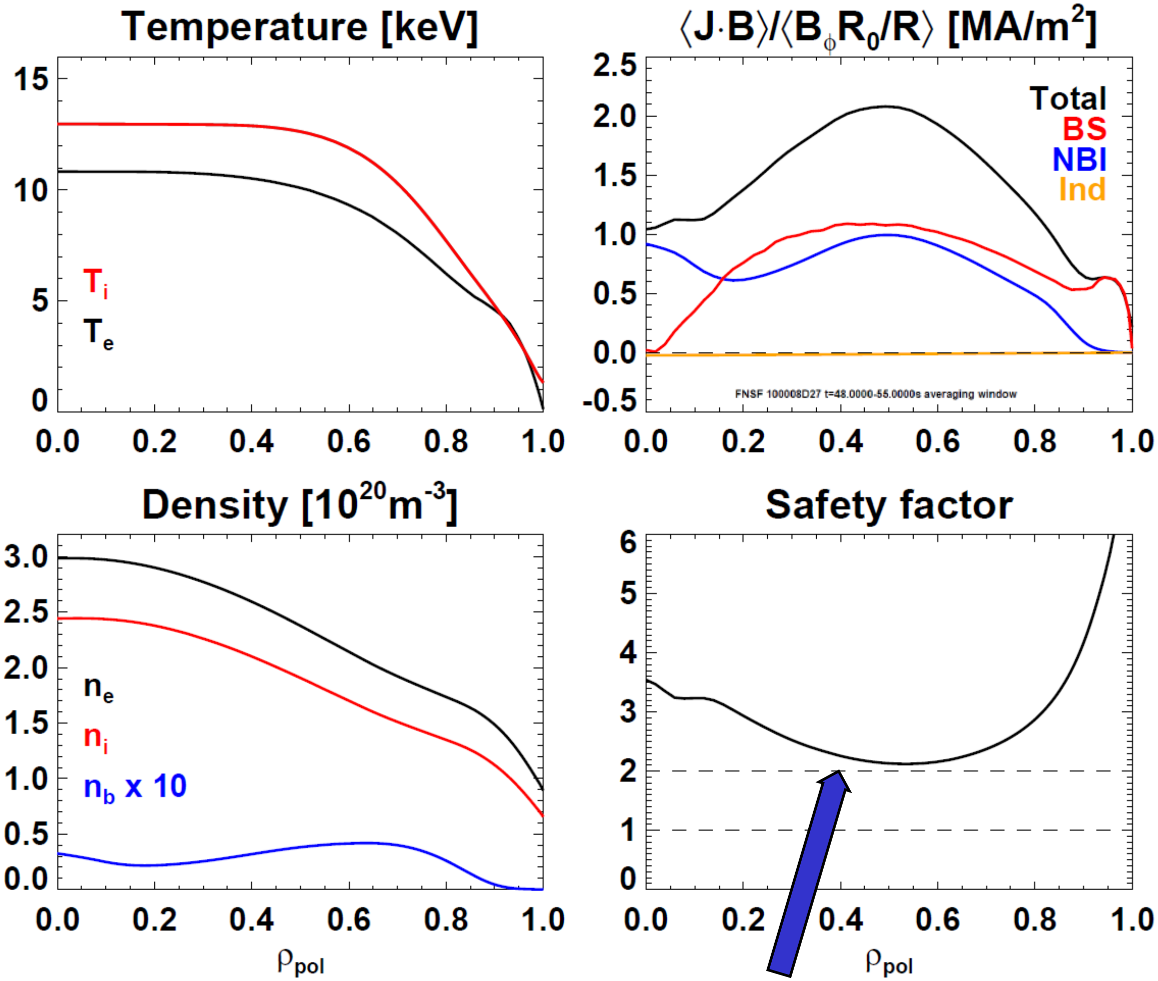
- 1m device cannot achieve TBR > 1 even with design changes
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

Summary: R = 1m and 1.7m STs with Γ_n = 1MW/m^2 and Q_{DT} = 1-2 assessed for FNS mission

- Ex-vessel PF coil set identified to support range of equilibria and Super-X/snowflake divertor to mitigate high heat flux
- 0.5MeV NNBI optimal for heating & current drive for R=1.7m
- Vertical maintenance approach, NBI & test-cell layouts identified
- Shielding adequate for MgO insulated inboard Cu PF coils
  - Outboard PF coils (behind outboard blankets) can be superconducting
- Calculated full 3D TBR; TBR reduction from TBM, MTM, NBI
- **Threshold major radius for TBR ~ 1 is $R_0 \geq 1.7$m**
- **R=1m TBR = 0.88 → 0.4-0.55kg of T/FPY → $12-55M/FPY**
- R=1m device will have lower electricity and capital cost → future work could assess size/cost trade-offs in more detail

Backup slides

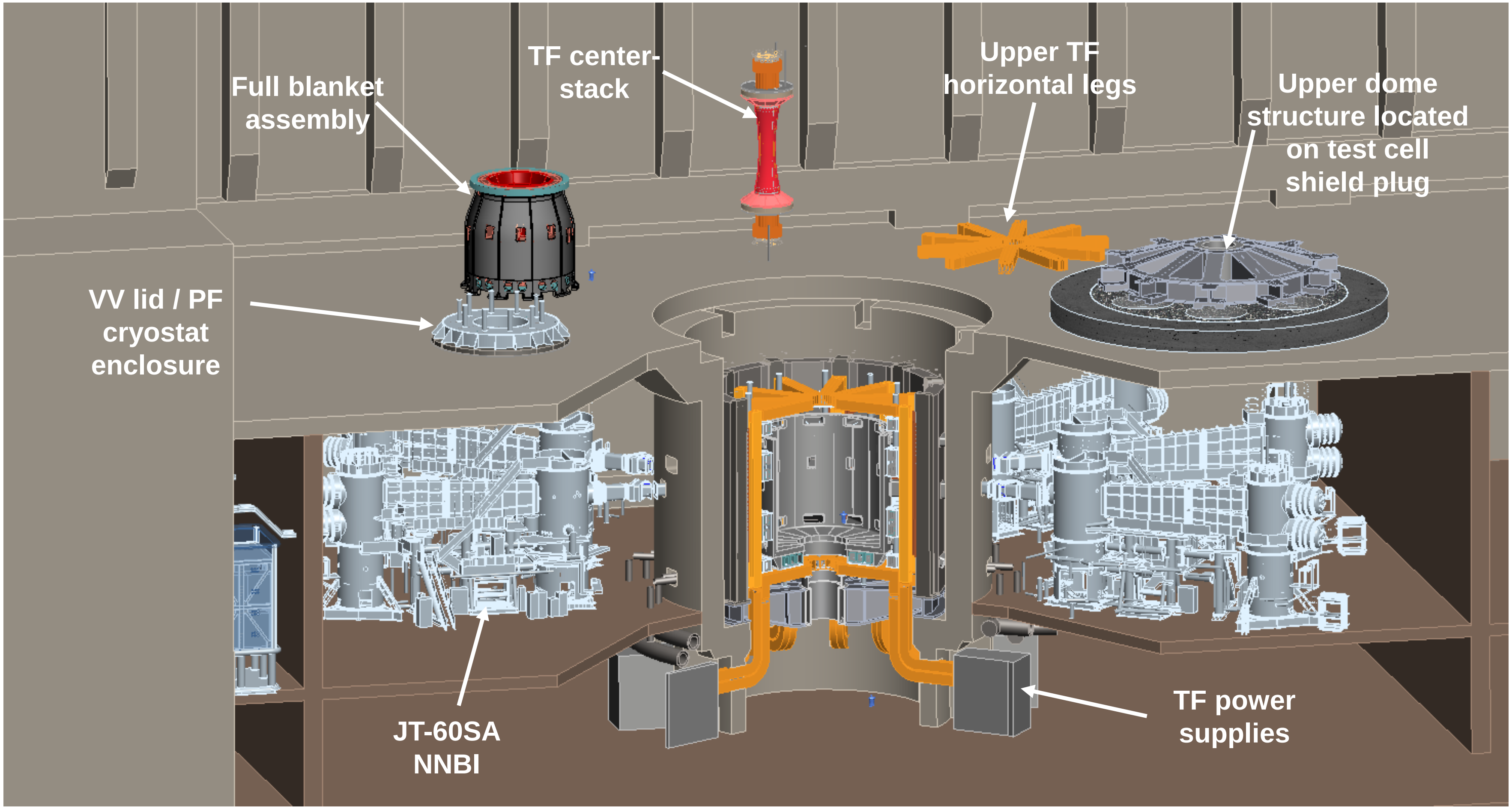
Free-boundary TRANSP/NUBEAM used to compute profiles for 100% non-inductive plasmas with Q_{DT} ~2



- **Neoclassical χ_{ion}**
- **n_e / $n_{Greenwald}$ = 0.7**
- **$H_{98,y2}$ = 1.4**
- **I_P = 8.9MA, B_T = 2.9T**
- **f_{NICD} = 100%, f_{BS} = 65%**
- **P_{NNBI} = 80MW (0.5MeV)**
- **P_{fus} = 200MW** (50-50 DT)
  - **2.6% alpha bad orbit loss**
- **Q_{DT} = 2.5**
- **β_N = 5.5, W_{tot} = 58MJ**
  - **W_{fast} / W_{tot} = 14%**

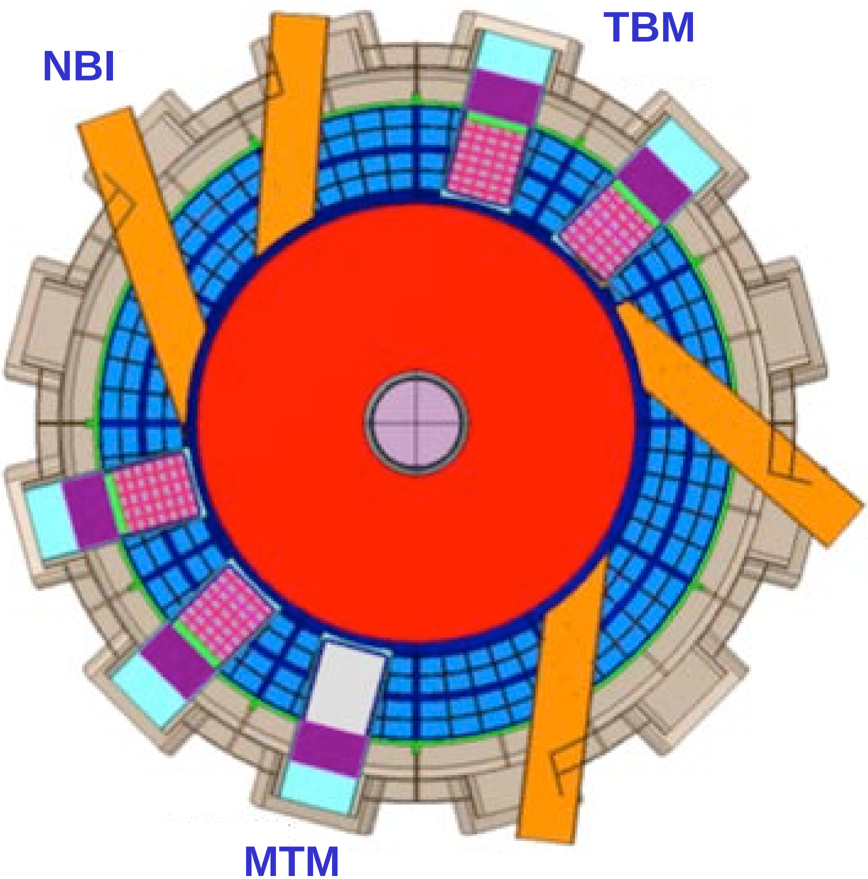
- **Maintain q_{min} > 2**
- **q(0) / q_{min} controllable via R_{tan} and density**

R=1.7m ST-FNS facility layout
using an extended ITER building

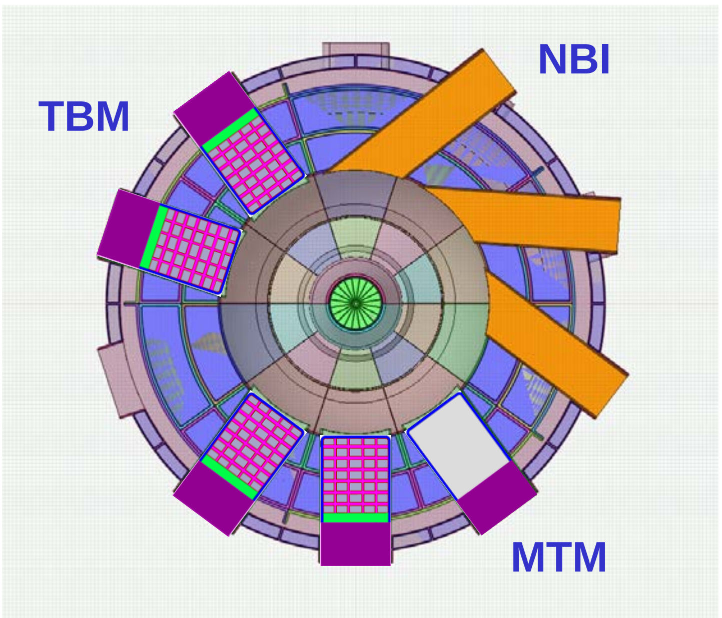


Summary of ST-FNSF TBR vs. device size

R=1.7m: **TBR ≥ 1**

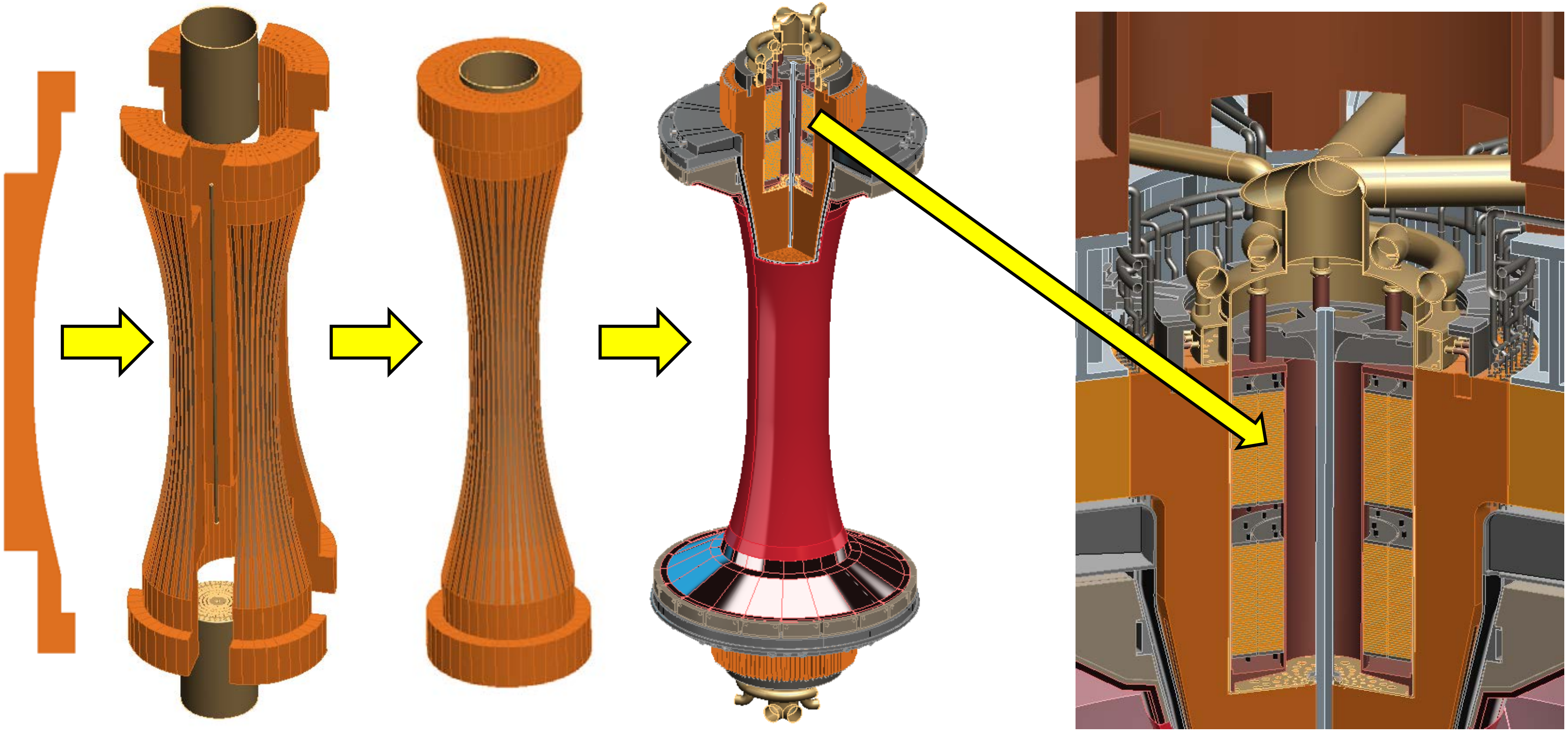


R=1.0m: **TBR < 1 (≈ 0.9)**



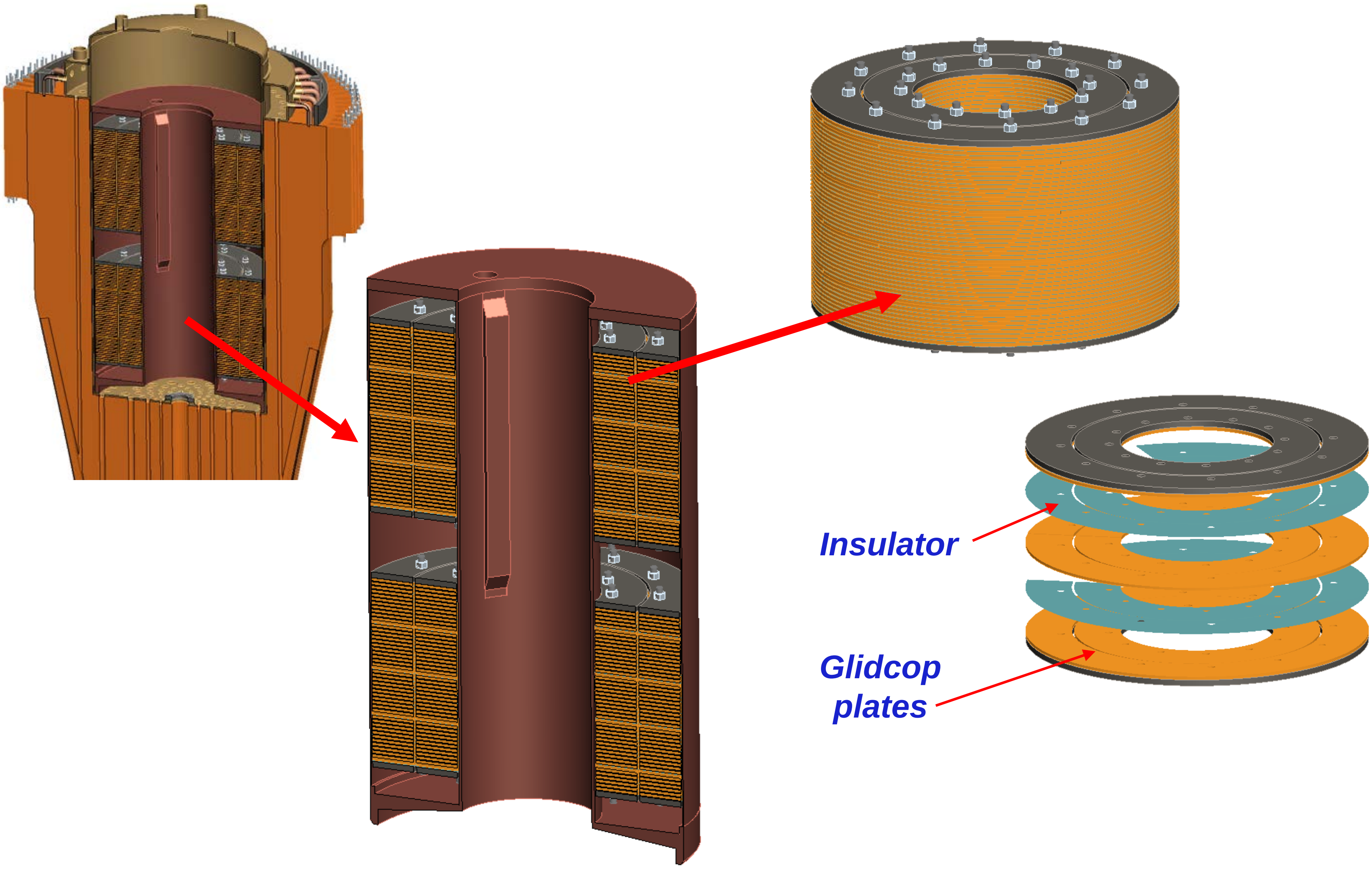
- **1m device cannot achieve TBR > 1 even with design changes**
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

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

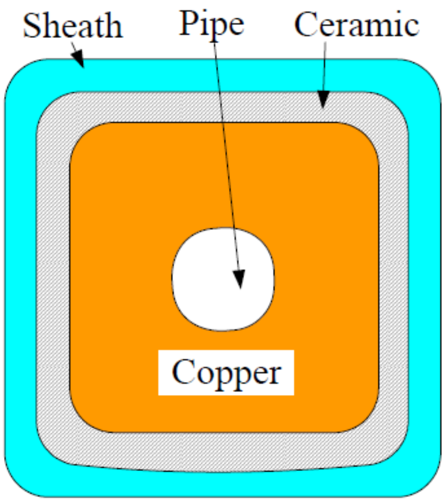


- **Like NSTX-U, use TF wedge segments (but brazed/pressed-fit together)**
  - Coolant paths: gun-drilled holes or grooves in side of wedges + welded tube
- **Bitter-plate divertor PF magnets in ends of TF achieve high triangularity**
  - **NSTX data:** High $\delta > 0.55$ and shaping $S \equiv q_{95}I_P/aB_T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

Bitter coil insert for divertor coils in ends of TF



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



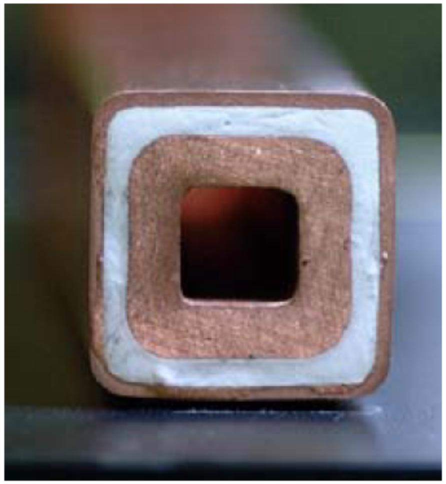


Fig. 3 Cross section of MIC

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

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)

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