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

Dr. Jonathan Menard

on behalf of:

M. Boyer[1], T. Brown[1], J. Canik[2], B. Covele[3], C. D'Angelo[4], A. Davis[4], L. El-Guebaly[4], S. Gerhardt[1], S. Kaye[1], C. Kessel[1], M. Kotschenreuther[3], S. Mahajan[3], R. Maingi[1], E. Marriott[4], L. Mynsberge[4], C. Neumeyer[1], M. Ono[1], R. Raman[5], S. Sabbagh[6], V. Soukhanovskii[7], P. Valanju[3], R. Woolley[1], A. Zolfaghari[1]

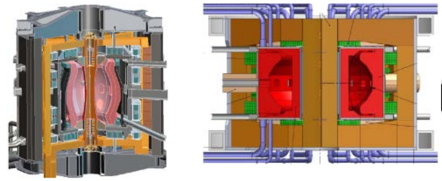
[1]Princeton Plasma Physics Laboratory, Princeton, NJ 08543
[2]Oak Ridge National Laboratory, Oak Ridge, TN, USA
[3]University of Texas, Austin, TX, USA
[4]University of Wisconsin, Madison, WI, USA
[5]University of Washington, Seattle, WA, USA
[6]Columbia University, New York, NY, USA
[7]Lawrence Livermore National Laboratory, Livermore, CA, USA

American Nuclear Society
21st Topical Meeting on the Technology of Fusion Energy (TOFE)
9-13 November 2014, Anaheim, CA

This work supported by the US DOE Contract No. DE-AC02-09CH11466

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

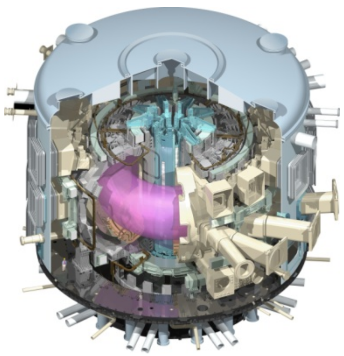
FNSF = Fusion Nuclear Science Facility
CTF = Component Test Facility



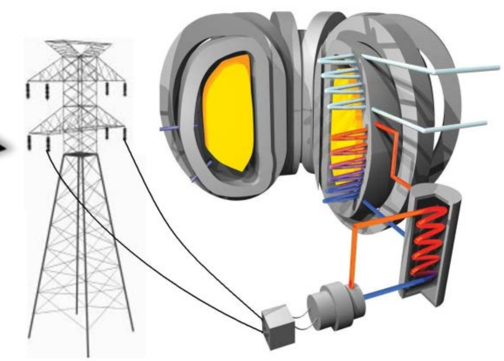
DEMO

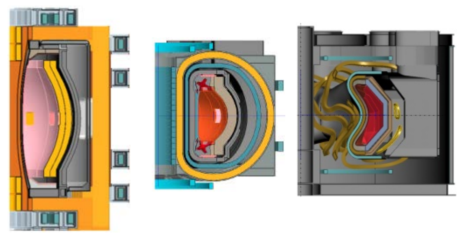
FNSF/CTF
Blanket R&D, T self-sufficiency

ITER



First of a kind Power Plant



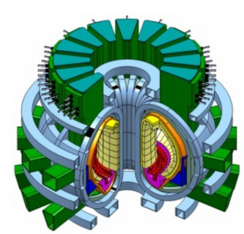


Pilot Plant
FNSF/CTF with power-plant like maintenance, $Q_{eng} \geq 1$

Supporting Physics and Technology

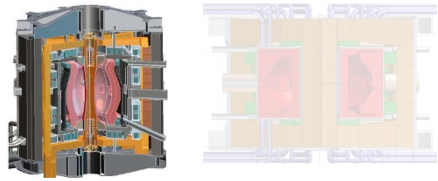
- Core Physics
- Materials R&D
- Plasma Material Interface

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



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

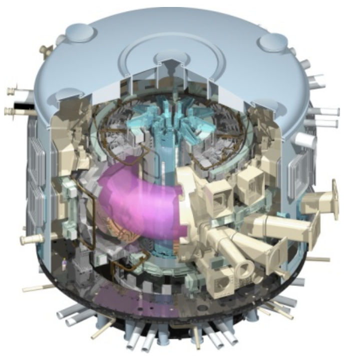
FNSF = Fusion Nuclear Science Facility
CTF = Component Test Facility



FNSF/CTF
Blanket R&D, T self-sufficiency

DEMO

ITER



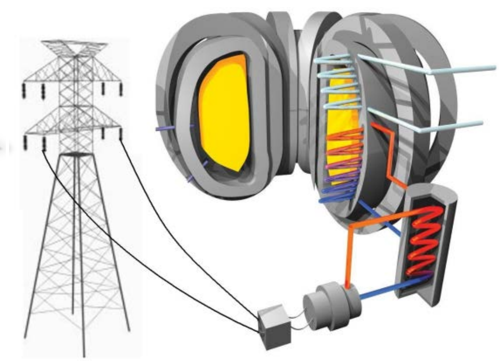
Supporting Physics and Technology

- **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



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)

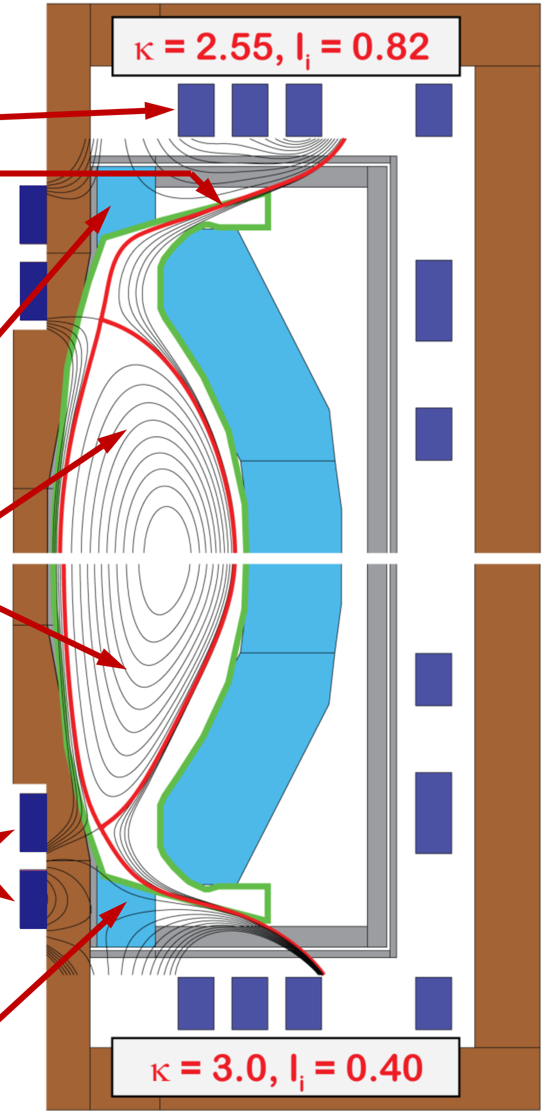
Outline

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

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

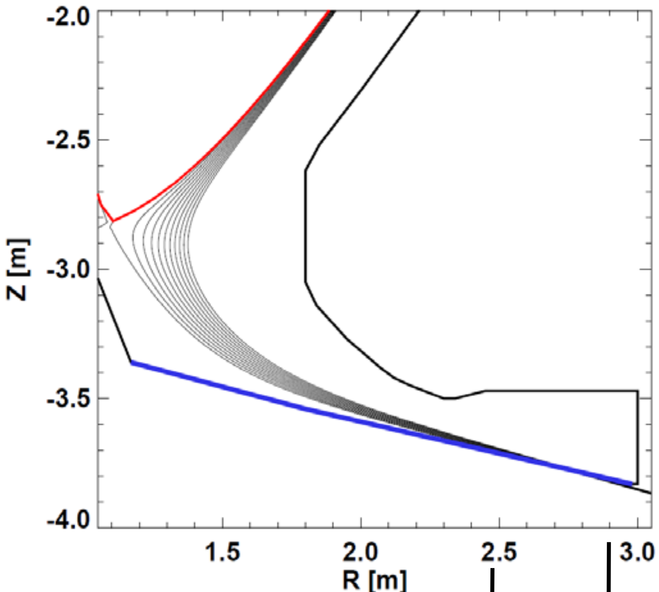
Components: TF coil | PF coil | Vessel | Blanket

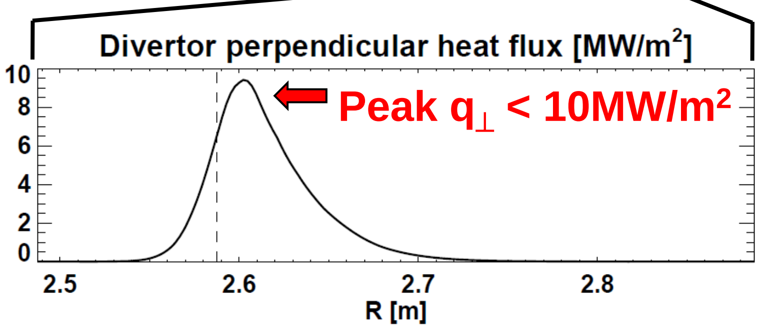
- All equilibrium PF coils outside vacuum vessel
- Increased strike-point radius reduces B, $q_{||}$
  Strike-point PFCs also shielded by blankets
- 2nd X-point/snowflake increases SOL line-length
- PF coil set supports wide range of l_i: 0.4 – 0.8
  - Elongation and squareness change with l_i variation
  - 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



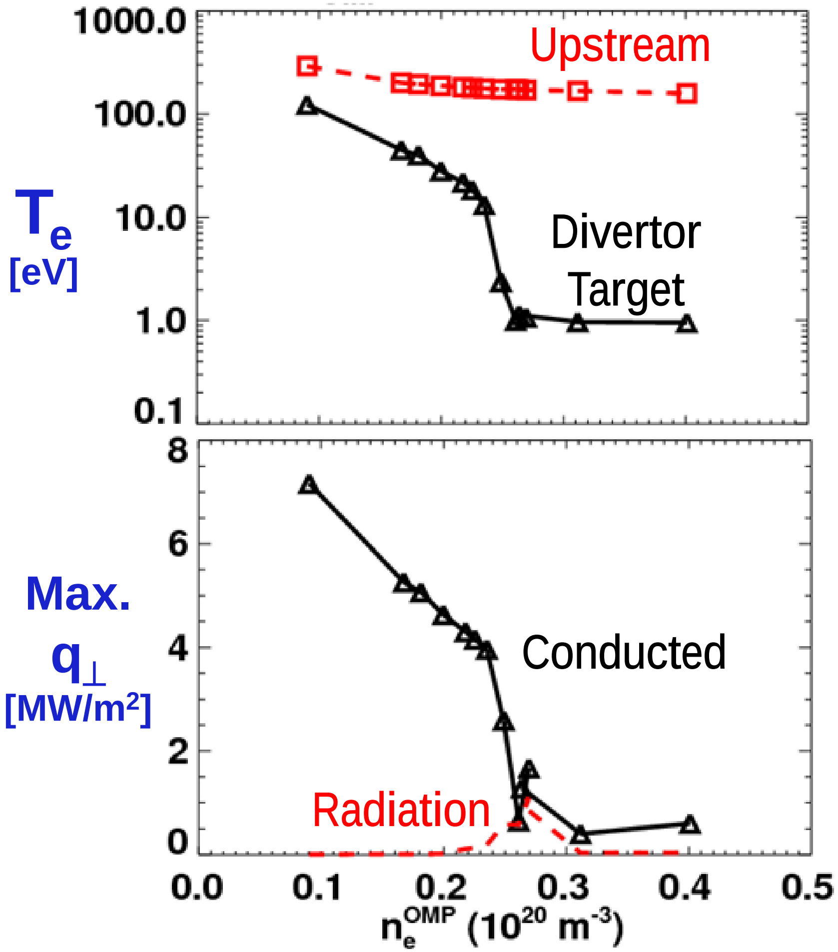
Up/down-symmetric Super-X/snowflake projected to maintain peak divertor heat flux below material limits

λ_q = 0.8mm, assume S ≈ λ_q (closed divertor)
(T. Eich NF 2013)



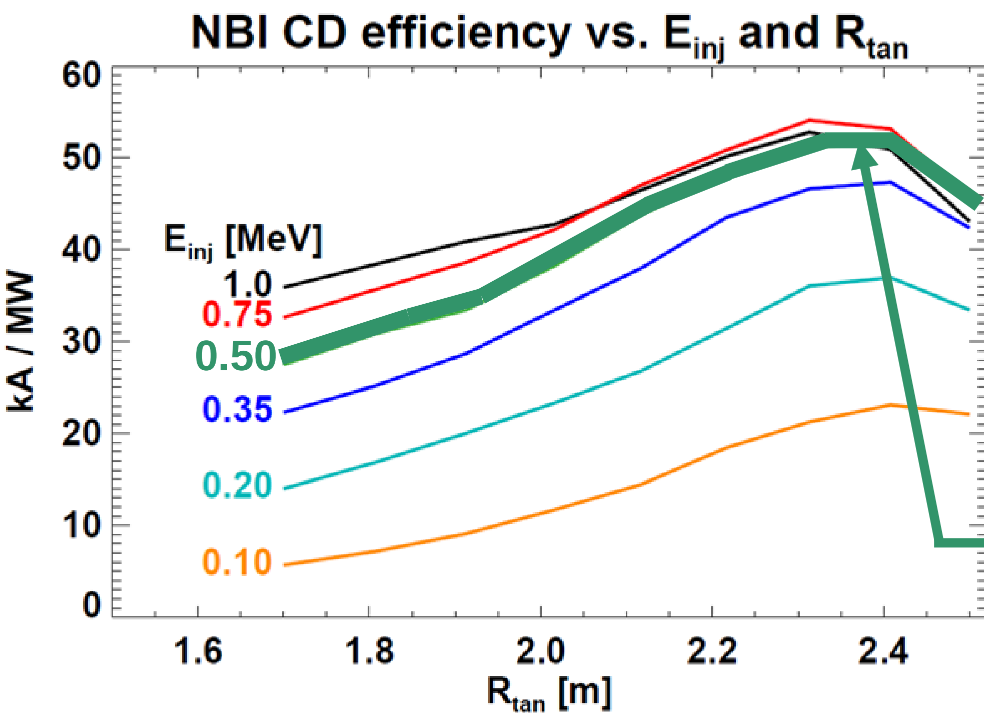


λ_q ≈ 1.2mm (J. Canik IEEE 2013)



(Partial) detachment projected to reduce peak $q_\perp$ to < 2MW/m²

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



NBCD increases for $E_{inj} \leq 0.5$ MeV but saturates for $E_{inj} = 0.75$ – 1MeV

Maximum efficiency: R_{tan}=2.3-2.4m

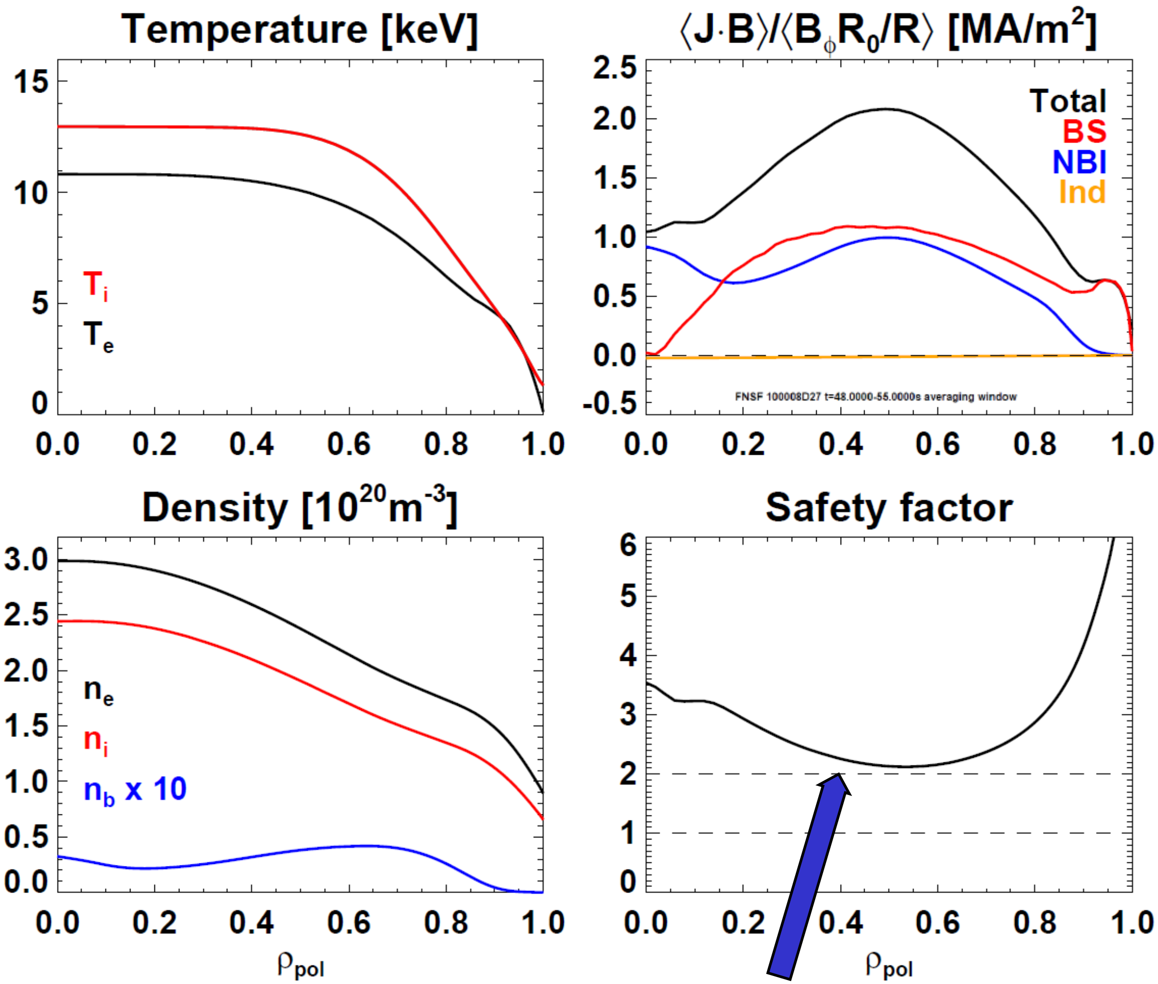
- Fixed target parameters in DD:
  - I_P = 7.5MA, β_N = 4.5, l_i = 0.5
  - $n_e / n_{Greenwald}$ = 0.75, $H_{98y,2}$ = 1.5
  - A=1.75, R=1.7m, B_T = 3T, κ = 2.8
  - $\langle T_e \rangle$=5.8keV, $\langle T_i \rangle$=7.4keV

Optimal tangency radii:
1.7m ≤ R_{tan} ≤ 2.4m

Control q(0), q_{min}

Shine-thru limit

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%**

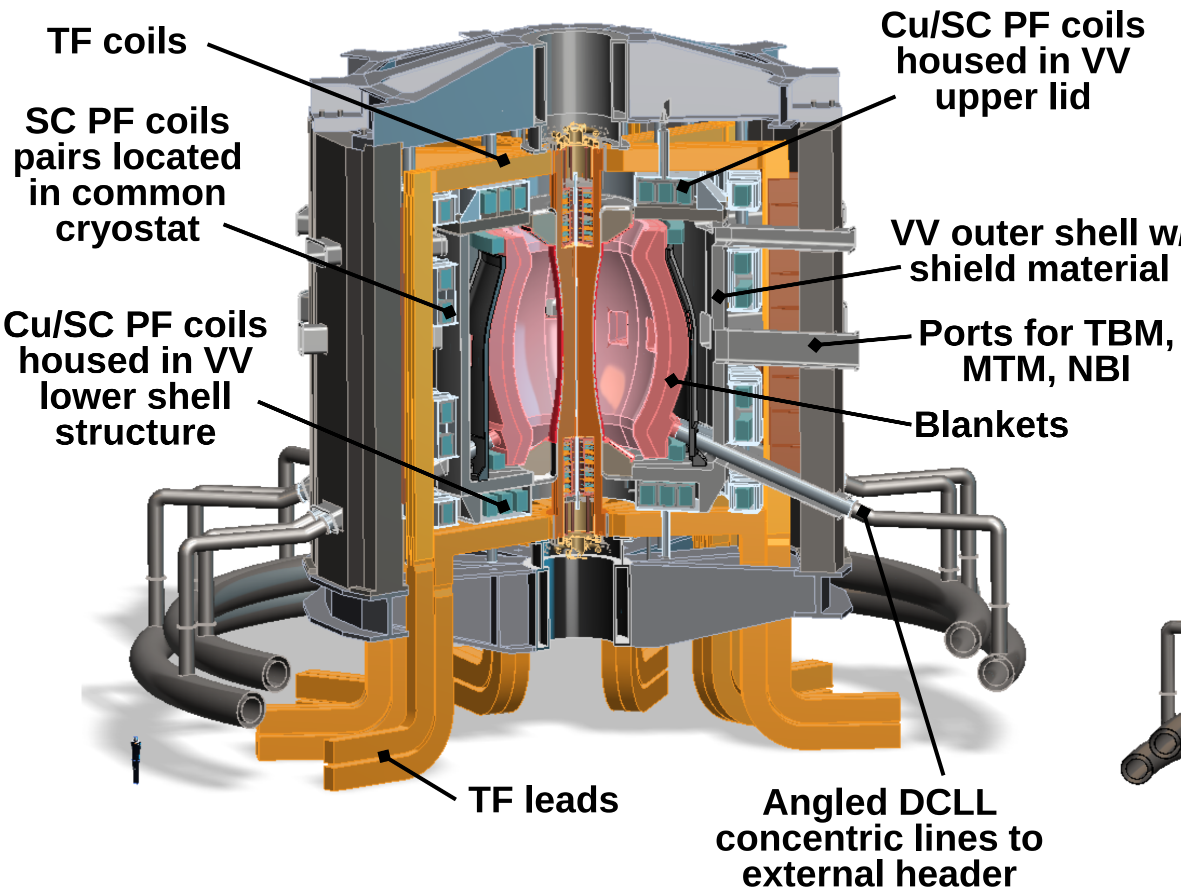
- **Maintain q_{min} > 2**
- **q(0) / q_{min} controllable via R_{tan} and density**

Outline

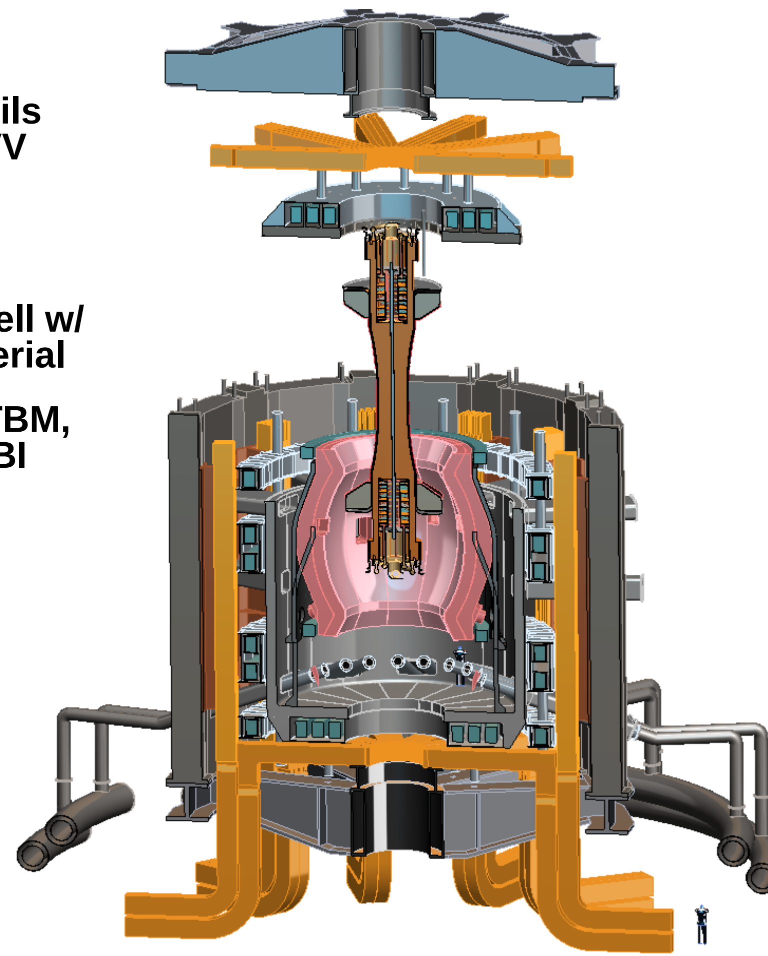
- Physics design
- Configuration, shielding, tritium breeding
- Conclusions

R=1.7m configuration with Super-X divertor

Design features

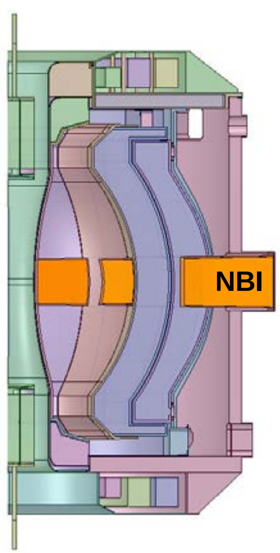


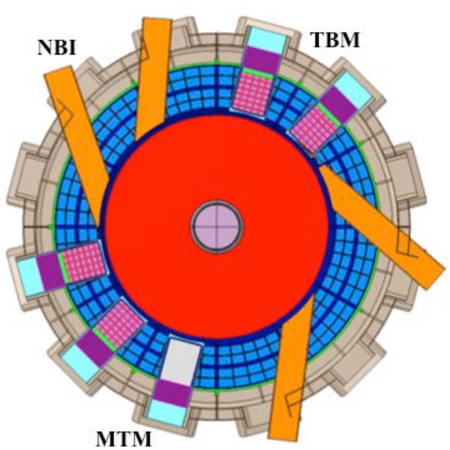
Vertical maintenance

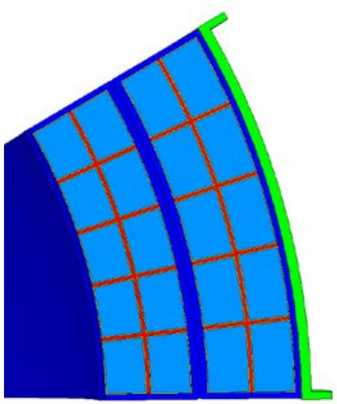


ST-FNSF shielding and TBR analyzed with sophisticated 3-D neutronics codes

- CAD coupled with MCNP using UW DAGMC code
- Fully accurate representation of entire torus
- 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







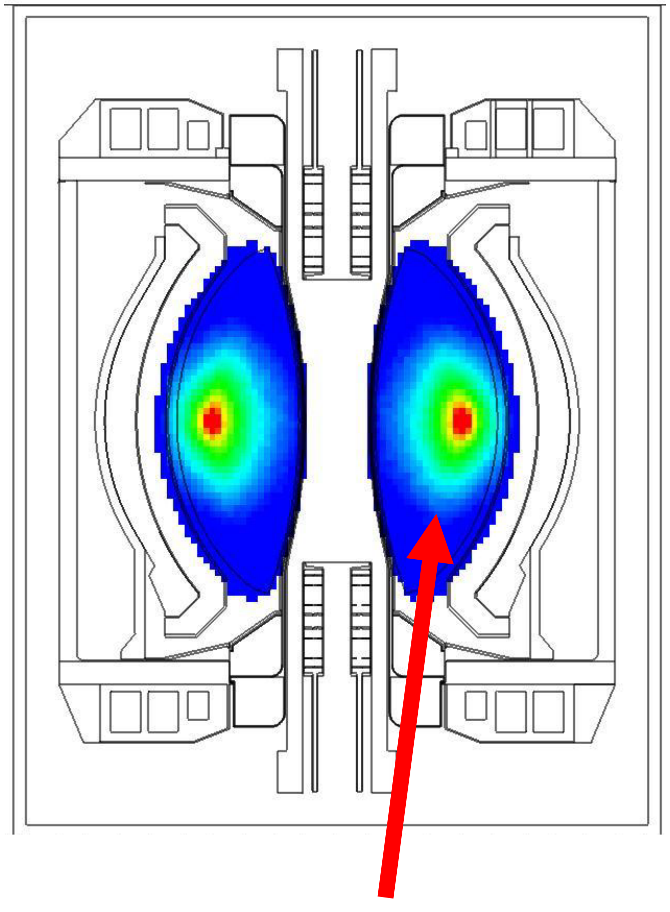
Heterogeneous OB Blanket Model, including FW, side/back/top/bottom walls, cooling channels, and SiC FCI

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)	**$1MW/m^2$**	**$1MW/m^2$**
TF coils	**12**	**10**
TBM ports	**4**	**4**
MTM ports	**1**	**1**
NBI ports	**4**	**3**
Plant Lifetime	~20 years	

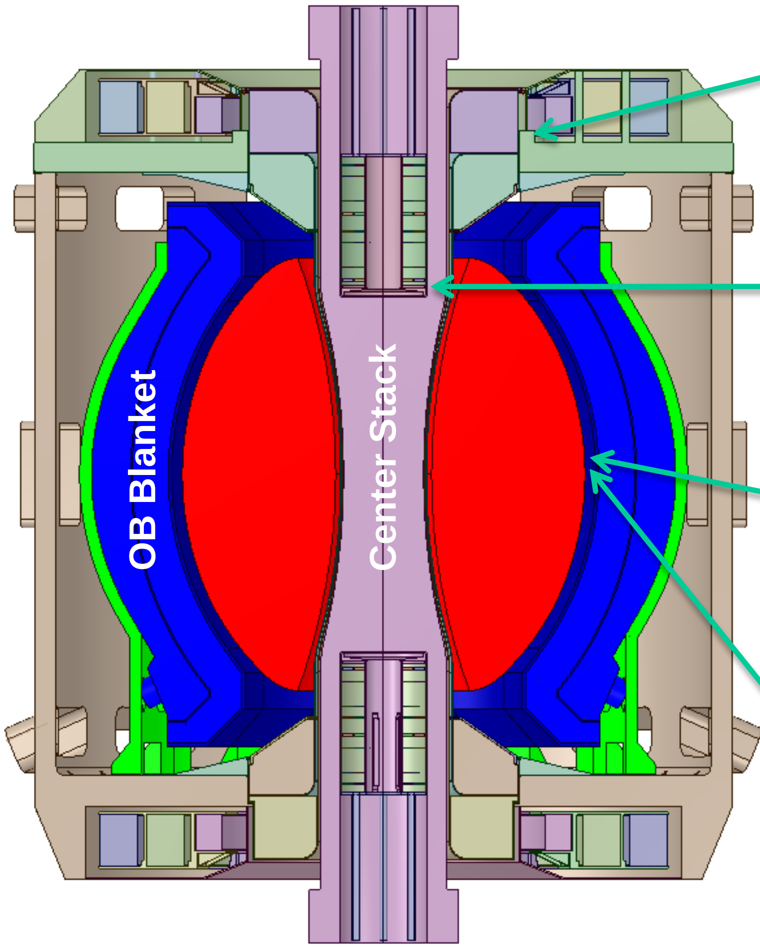
Availability 10-50% / 30% avg } **6 Full Power Years (FPY)**



Neutron source distribution

Peak Damage at OB FW and Insulator of Cu Magnets

R=1.7m configuration



3-D Neutronics Model of Entire Torus

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

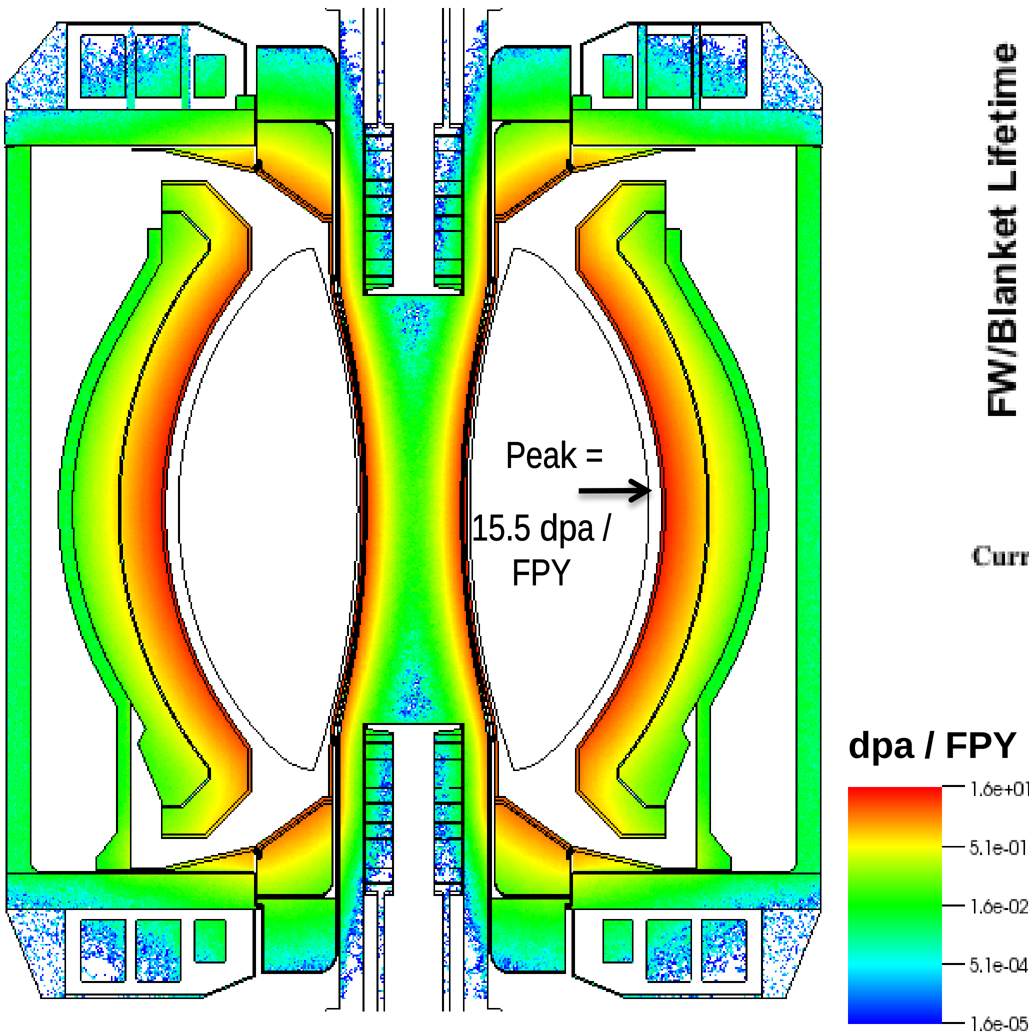
Peak dpa at OB midplane = 15.5 dpa / FPY

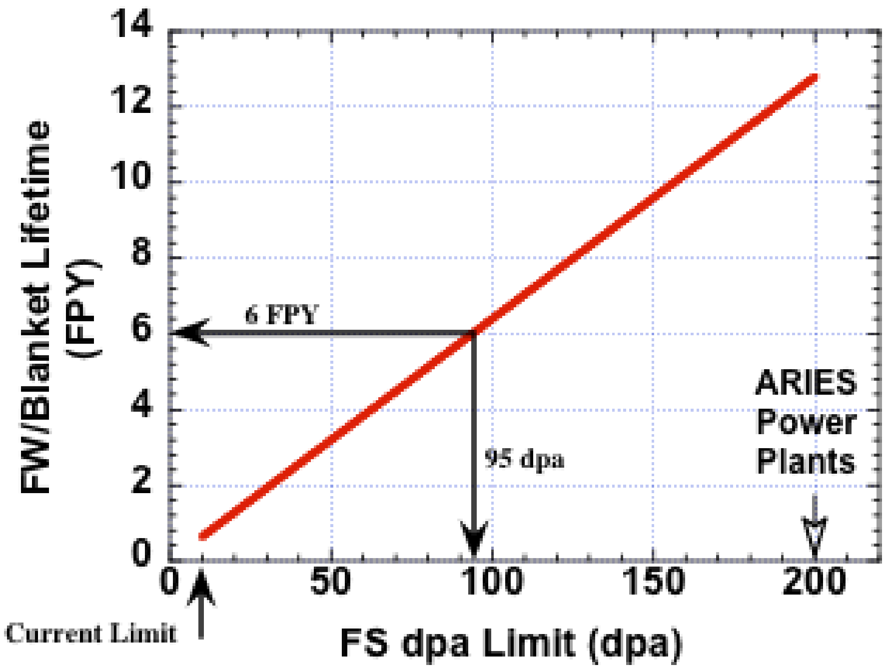
Peak He production at OB midplane = 174 appm/FPY

$\Rightarrow$ He/dpa ratio = 11.2

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

R=1.7m configuration

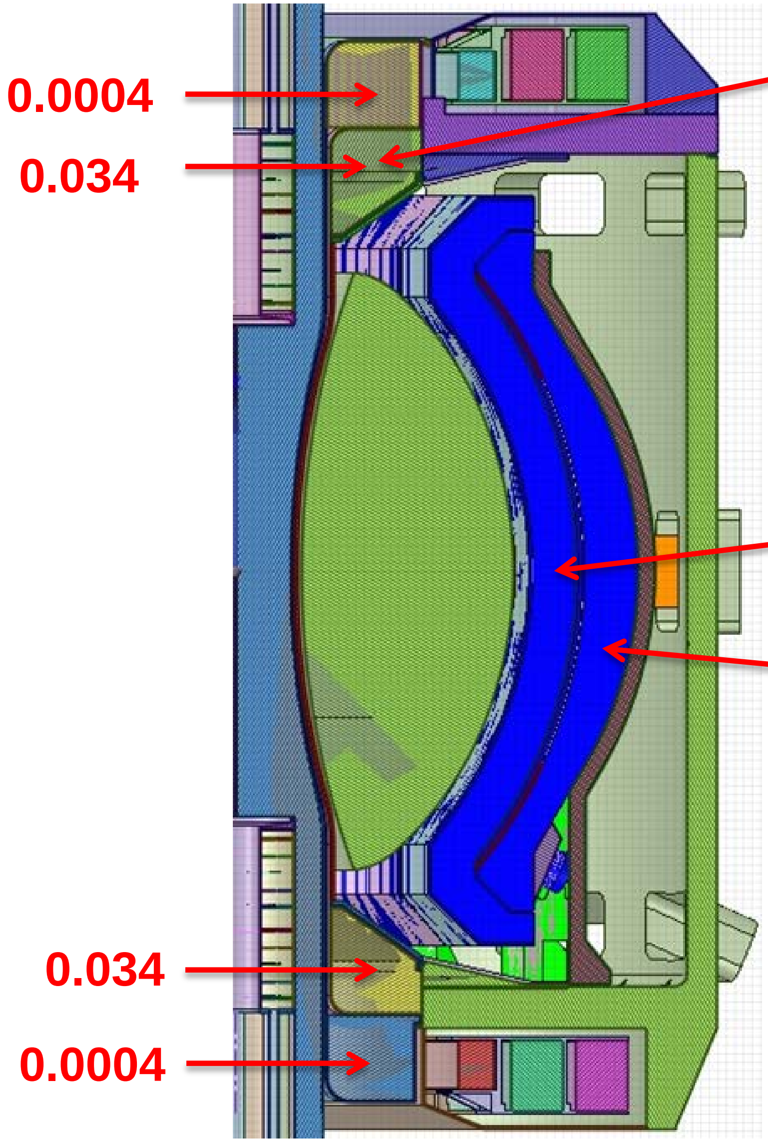




FW/blanket could operate for 6 FPY if allowable damage limit is 95 dpa

→ Peak EOL Fluence = 11 MWy/m^2

TBR contributions by blanket region



Breeding at CS ends important:
ΔTBR = +0.07

Inner Blanket Segment = 0.81

Outer Blanket Segment = 0.15

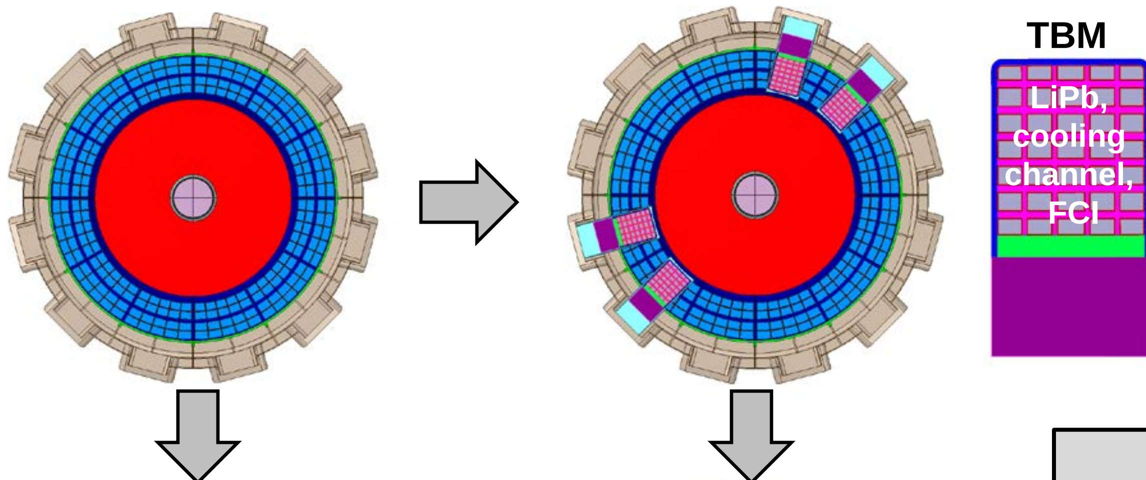
Total TBR ~ 1.03 with no penetrations or ports
(heterogenous outboard blanket)

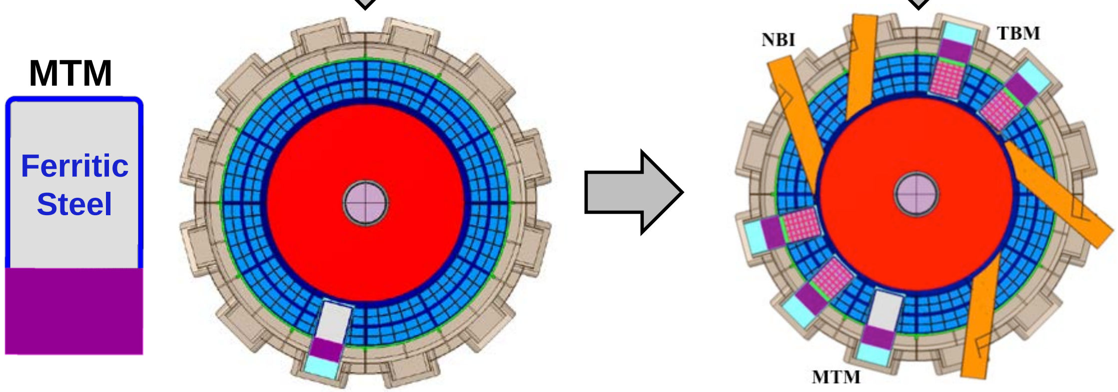
R=1.7m configuration

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)





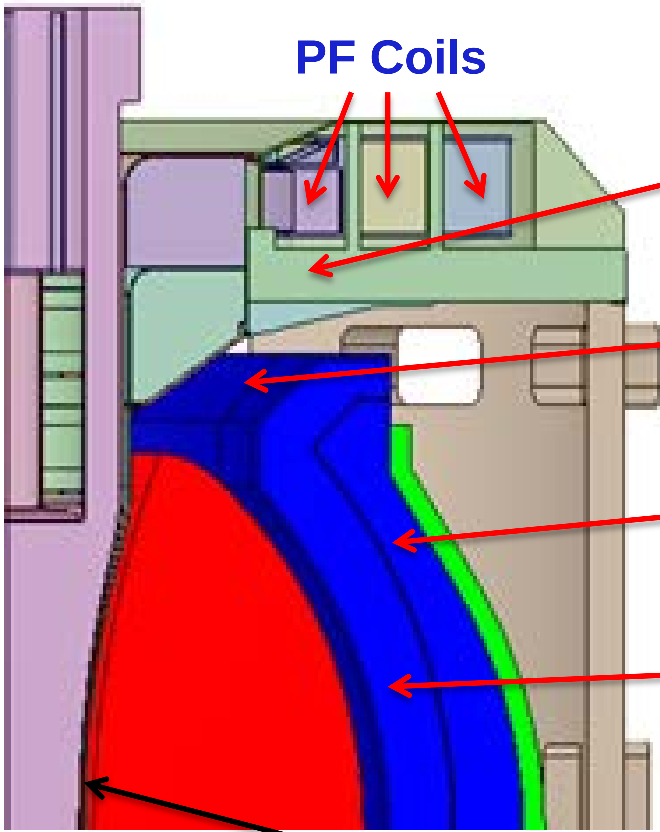
1 Materials Test Module (MTM)
TBR = 1.01 (ΔTBR = -0.02)

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

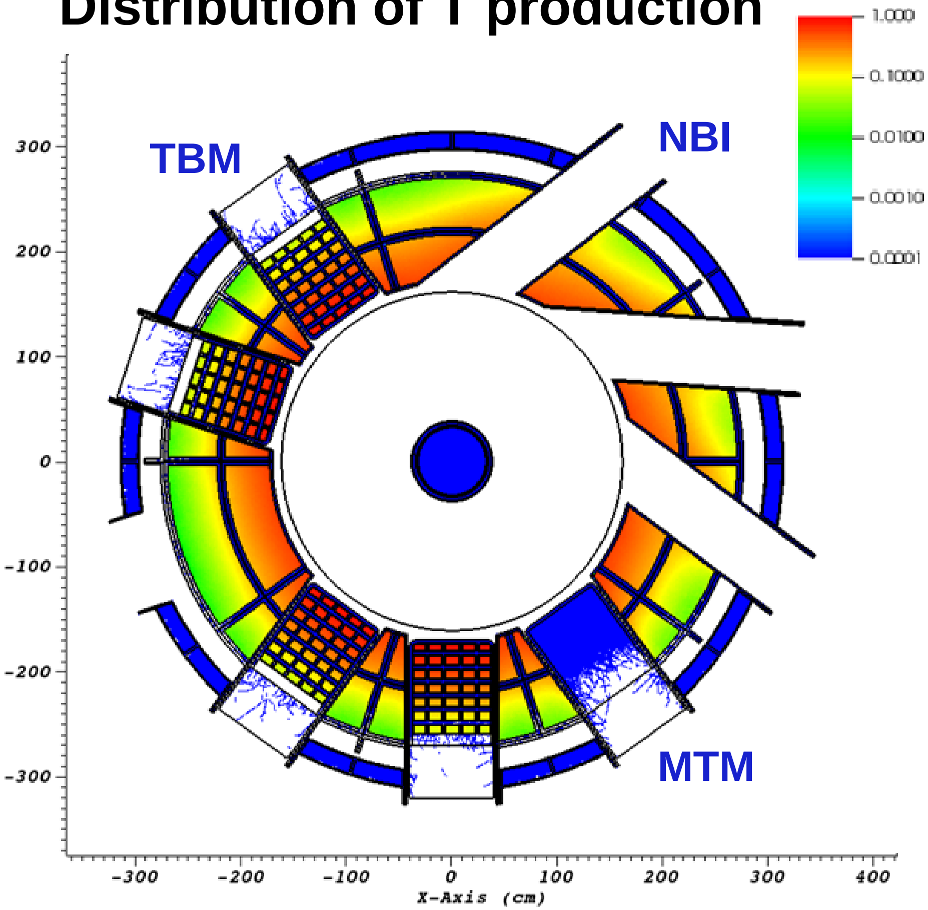


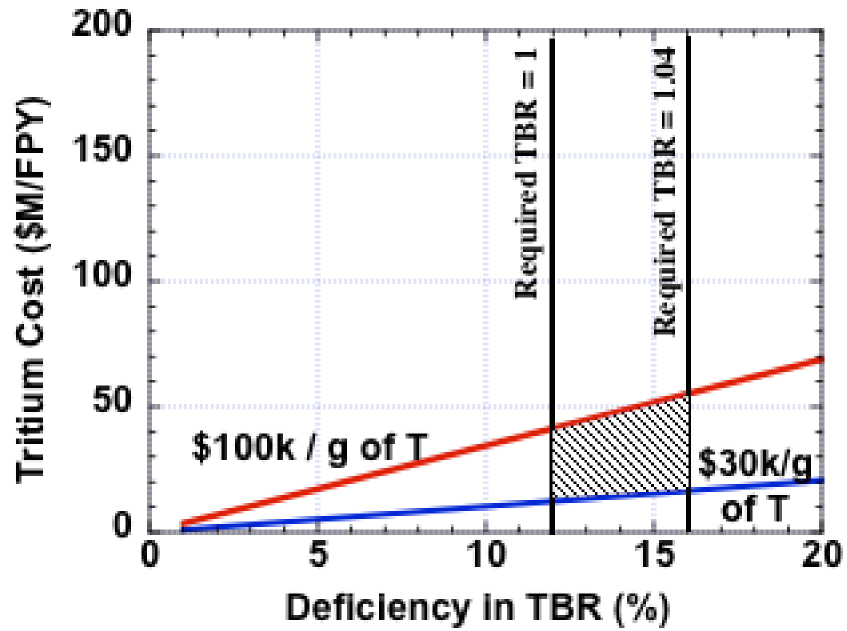
- Add to PF coil shield a thin breeding blanket (ΔTBR ~ +3%)
- Smaller opening to divertor to reduce neutron leakage
- Uniform OB blanket (1m thick everywhere; no thinning)
- Reduce cooling channels and FCIs within blanket (need thermal analysis to confirm)
- Thicker IB VV with breeding

Potential for TBR > 1 at R=1.7m

R_0 = 1m ST-FNSF achieves TBR = 0.88

Distribution of T production





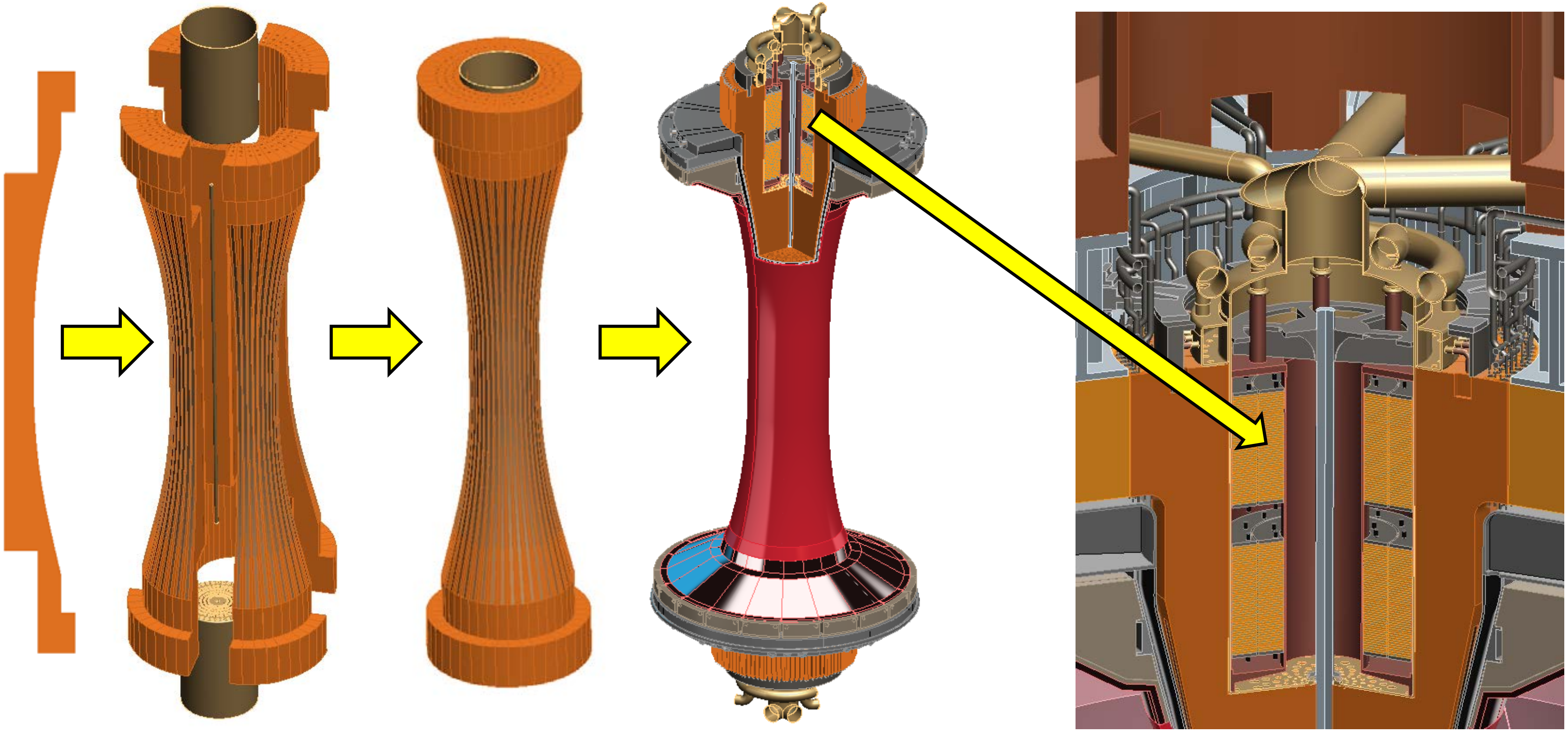
- 1m device cannot achieve TBR > 1 even with design changes
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

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 $R_0 \geq$ 1.7m**
- **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

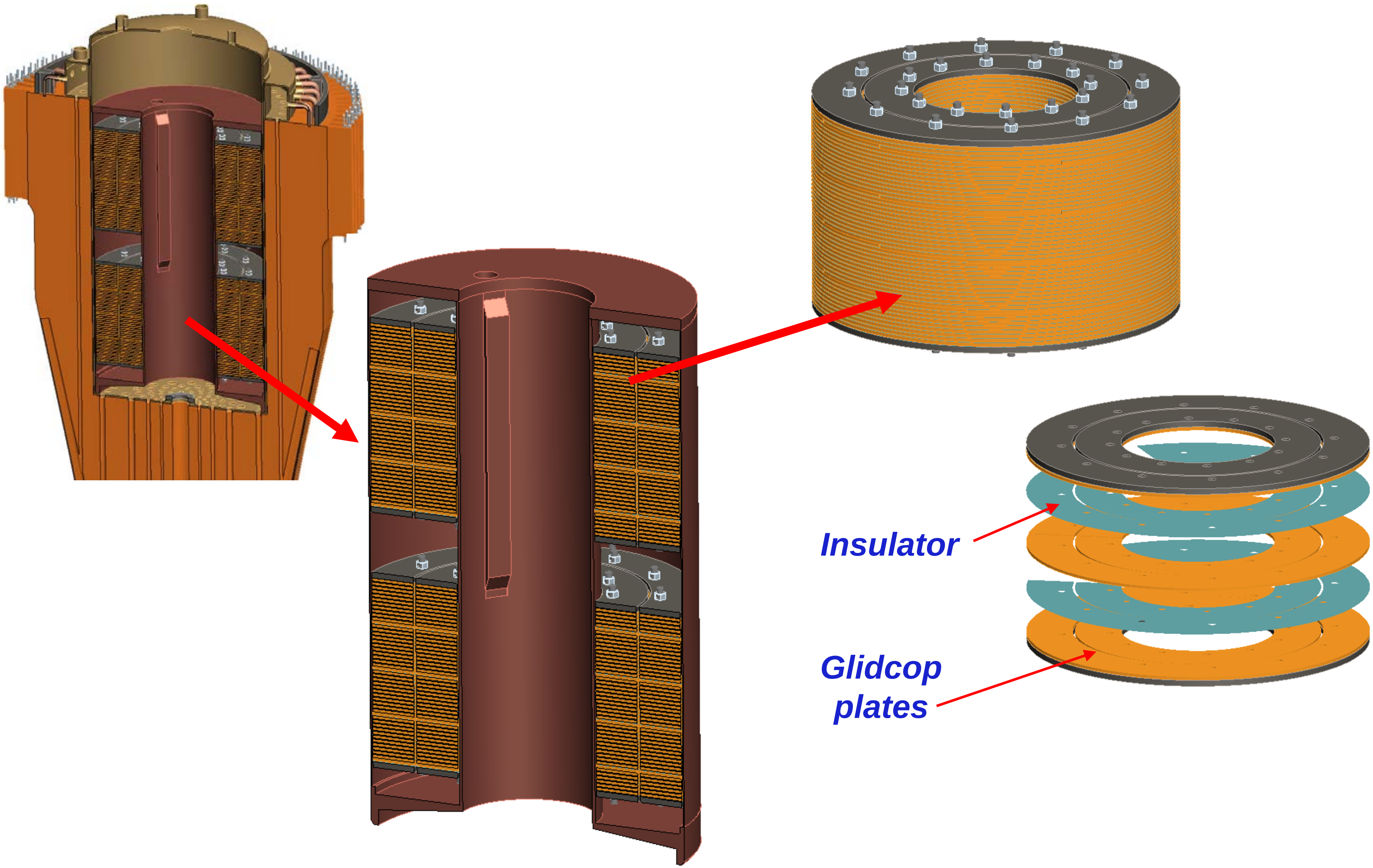
Backup slides

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

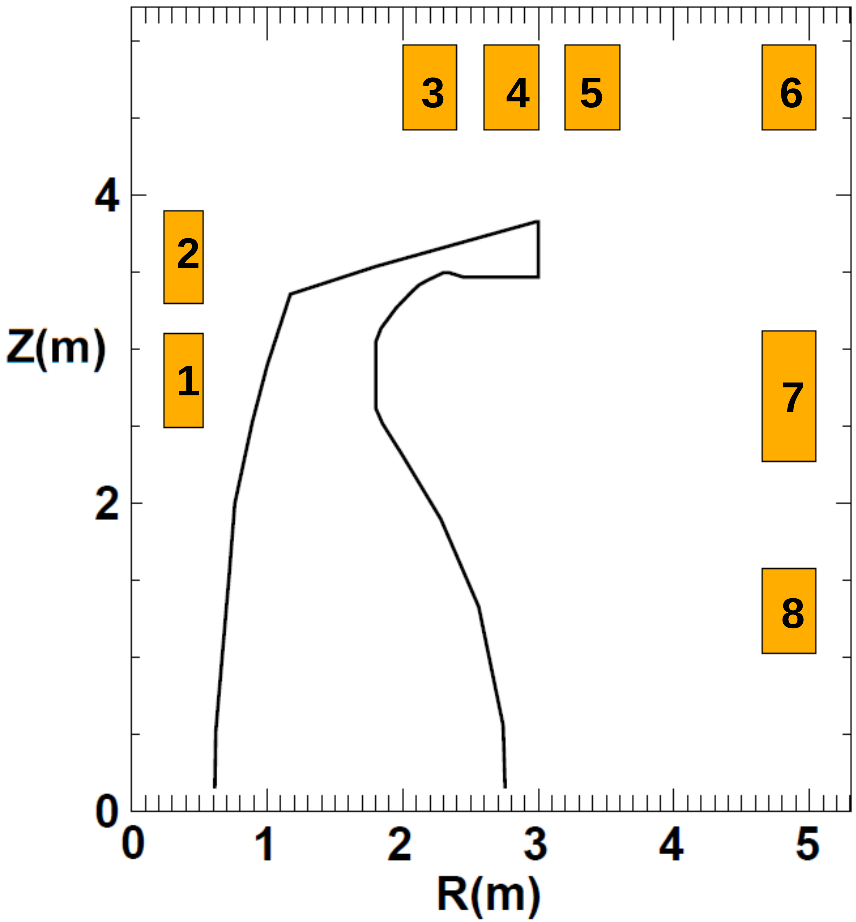
Bitter coil insert for divertor coils in ends of TF

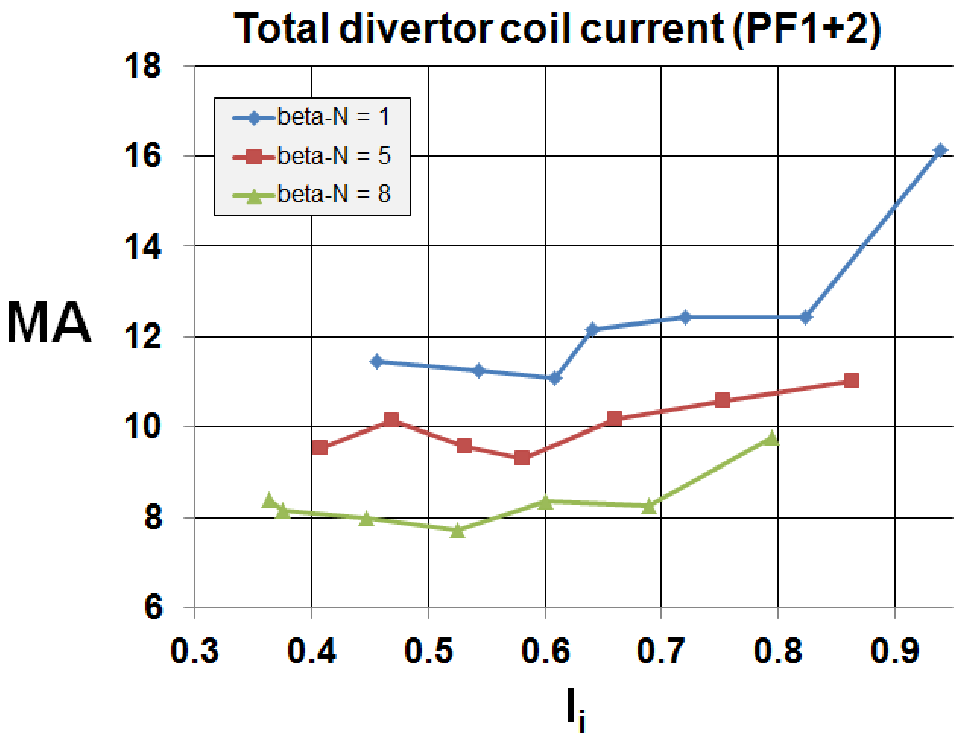


PF coil currents and current densities

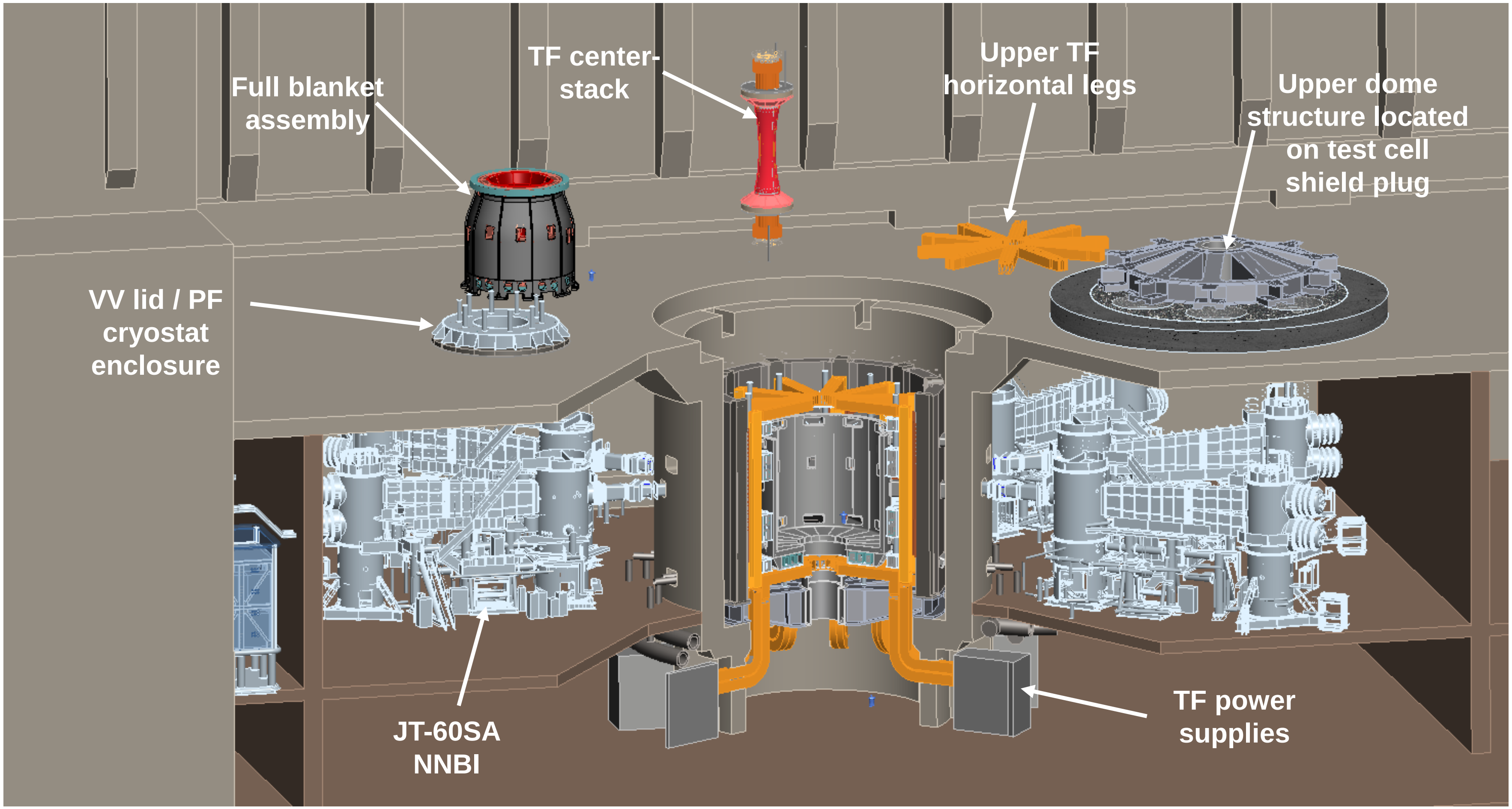
$\beta_N = 5$
$l_i = 0.58$

Coil	PF1U	PF2U	PF3U	PF4U	PF5U	PF6U	PF7U	PF8U
MA turns	2.3	7.0	1.3	0.0	6.2	-0.6	-9.7	0.4
MA/m^2	12.8	39.6	5.9	0.0	28.2	-2.7	-28.5	1.8



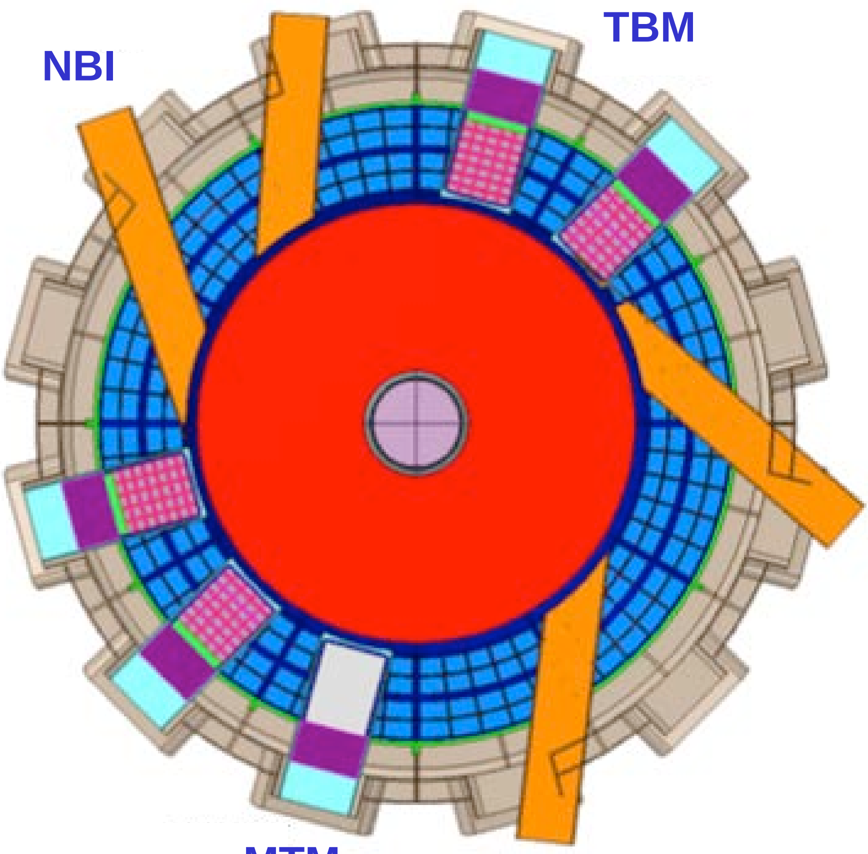


R=1.7m ST-FNS facility layout
using an extended ITER building

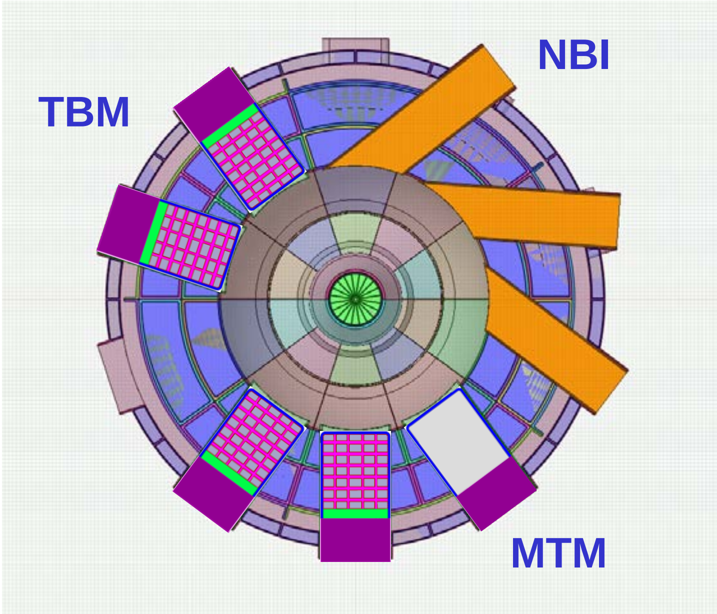


Summary of ST-FNSF TBR vs. device size

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

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

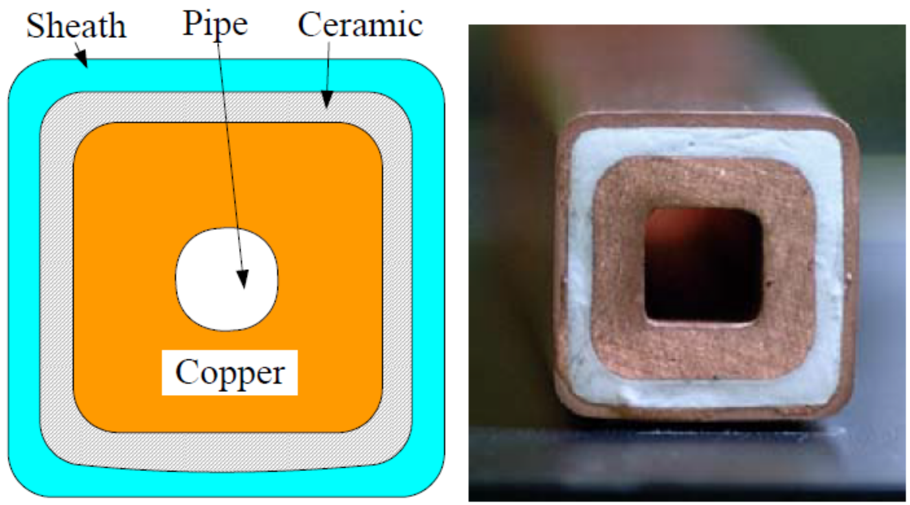


Fig. 3 Cross section of MIC

Table 1: Comparison of radiation resistant

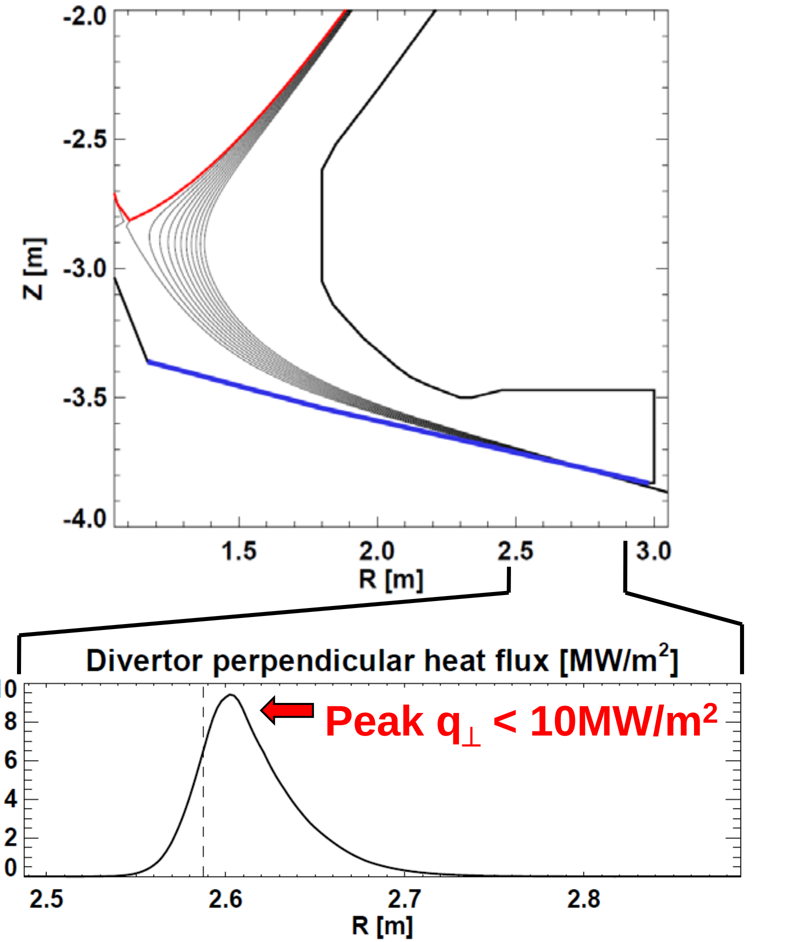
	Organic		Inorganic
Insulation	Epoxy	Polyimide	MgO
Resistant	$>10^7$ Gy	$>10^9$ Gy	$>10^{11}$ Gy

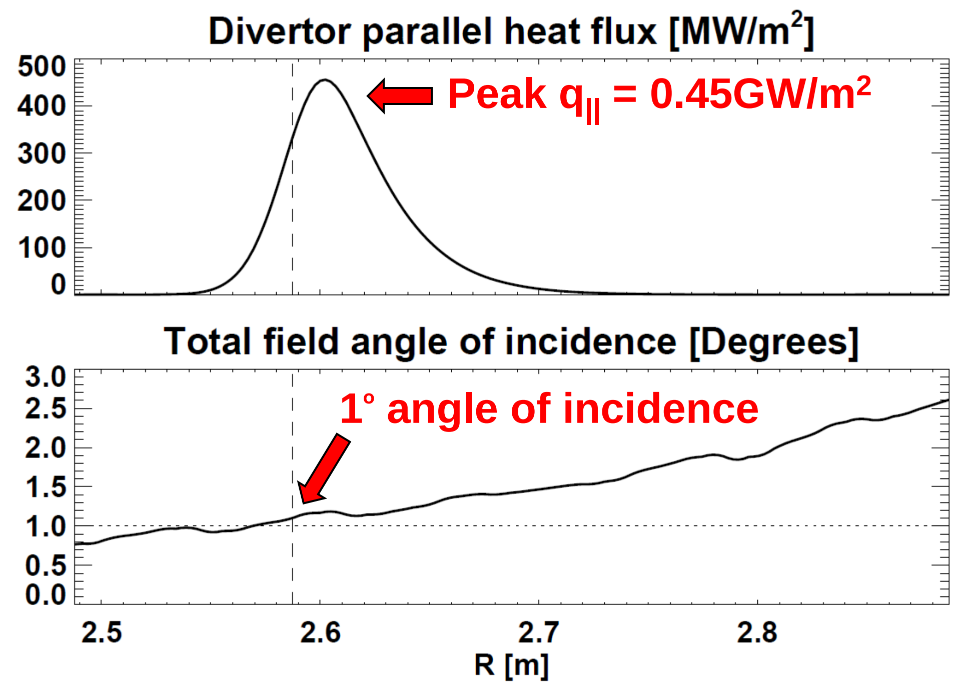
R&D of a Septum Magnet Using MIC coil

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)

Kuanjun Fan [1,A)], Hiroshi Matsumoto [A)], Koji Ishii [A)], Noriyuki Matsumoto [B)]
[A)] High Energy Accelerator Research Organization (KEK)
1-1 OHO, Tsukuba, Ibaraki, 305-0801, Japan
[B)] 2NEC/Token

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





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