Configuration Studies for Next-Step Spherical Tokamaks*

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18th International Spherical Torus Workshop
Princeton University
Princeton, NJ USA
3-6 November 2015

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

Possible missions for next-step STs

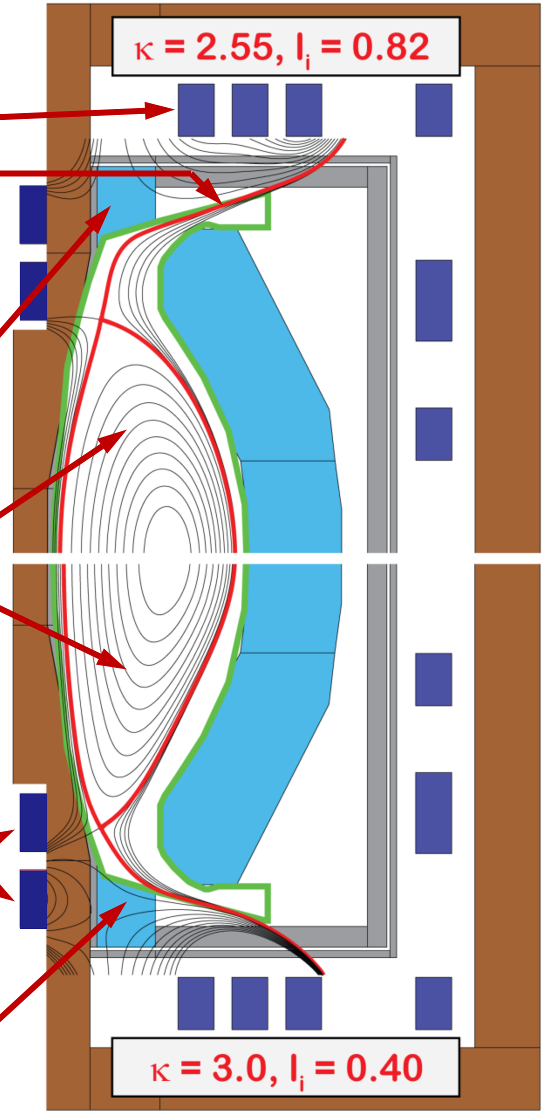
1. Integrate high-performance, steady-state, exhaust
   - Divertor test-tokamak - DTT ← ***Past (& future) PPPL Studies***
2. Fusion-relevant neutron wall loading
   - $\Gamma_n \sim$ 1-2MW/m^2, fluence: $\geq$ 6MW-yr/m^2
3. Tritium self-sufficiency
   - Tritium breeding ratio TBR $\geq$ 1
4. Electrical self-sufficiency
   - $Q_{eng} = P_{electric} / P_{consumed} \sim 1$
5. Large net electricity generation
   - $Q_{eng} >> 1$, $P_{electric}$ = 0.5-1 GWe

Recent / Present PPPL-led Studies

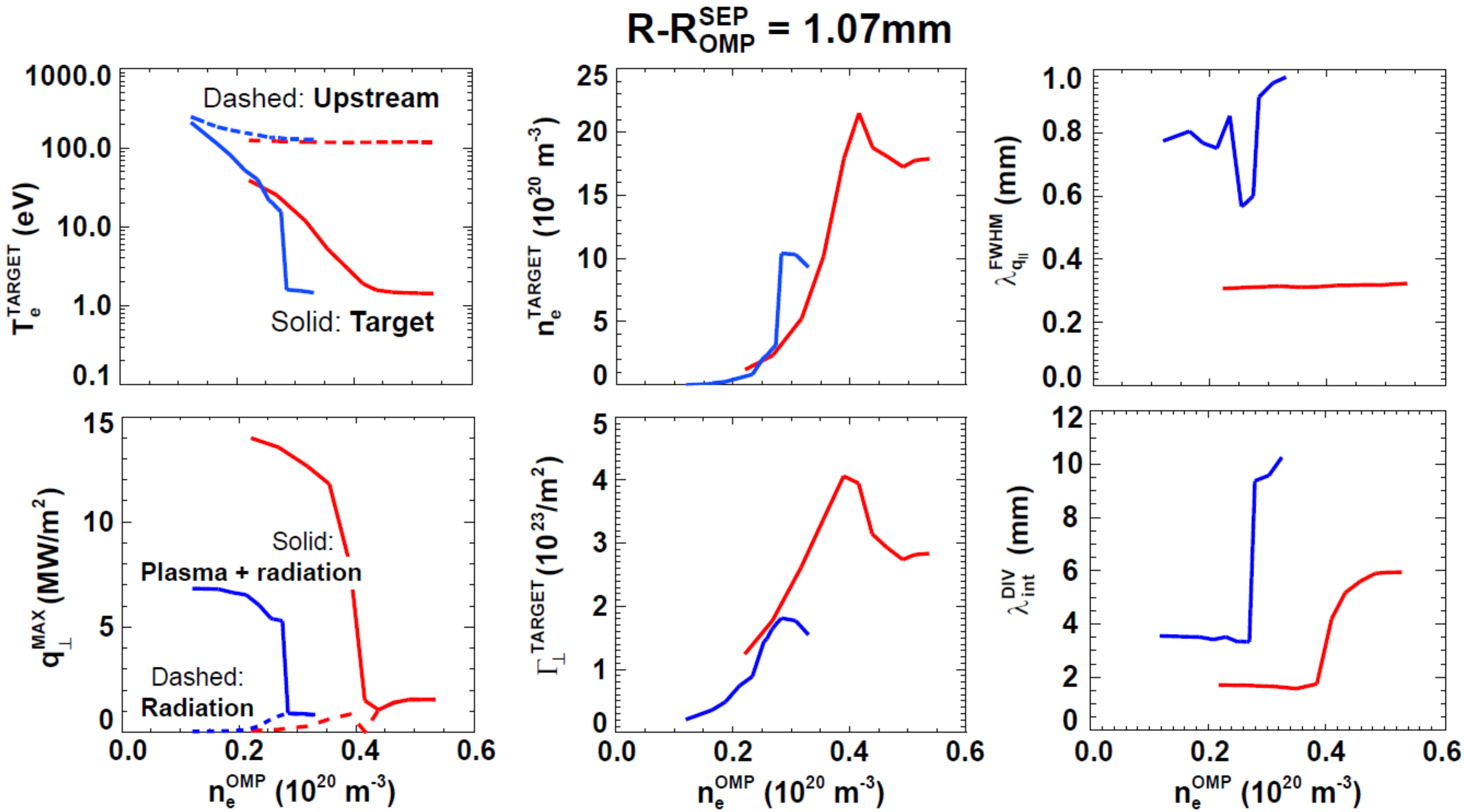
Identified TF + PF coil set that supports long-leg / Super-X divertor for range of equilibria

Components: TF coil | PF coil | Vessel | Blanket

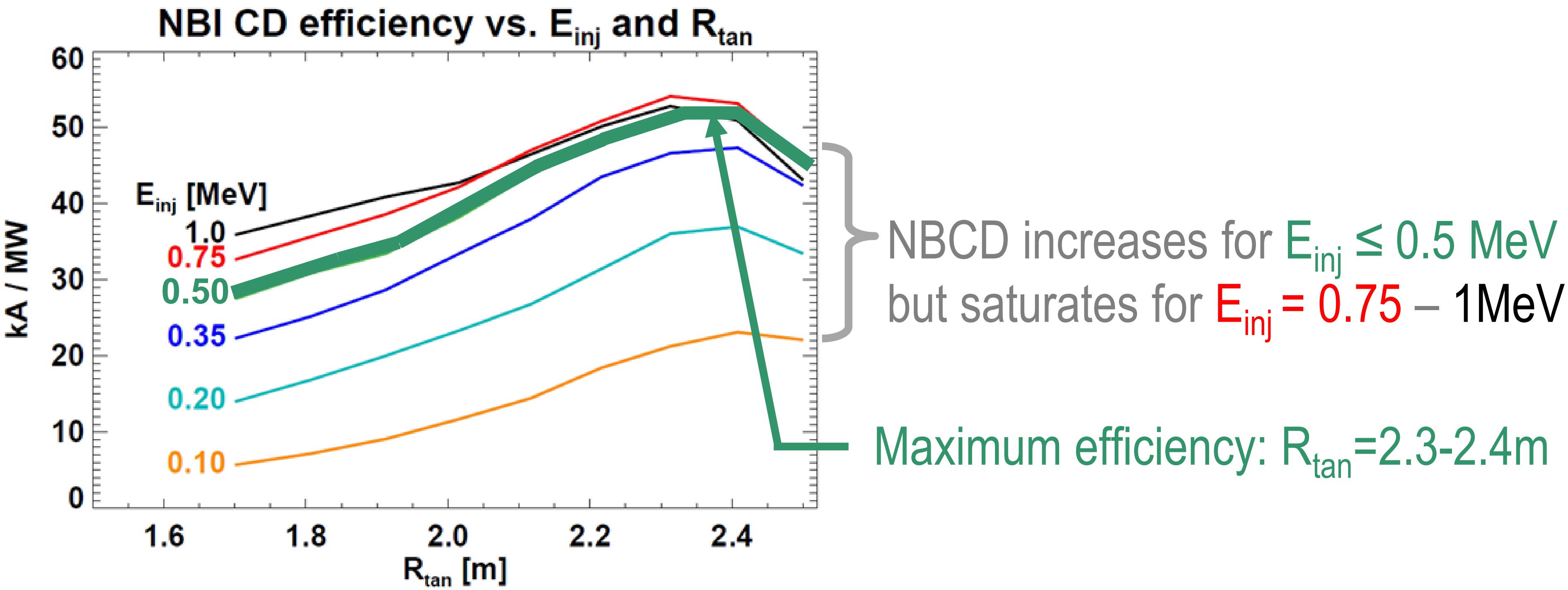
- All equilibrium PF coils outside vacuum vessel
- Increased strike-point radius reduces B, $q_{||}$
  Strike-point PFCs also shielded by blankets
- 2nd X-point/snowflake increases SOL line-length
- PF coil set supports wide range of l_i: 0.4 – 0.8
  - Elongation and squareness change with l_i variation
  - 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 long-leg divertor → $q_{\perp\text{-divertor}}$ < 1-2 MW/m^2 under detached conditions (SOLPS / ORNL)



Negative NBI (0.5 MeV) with large R_{TAN} favorable for heating and current drive (CD) for R=1.7m ST-FNSF



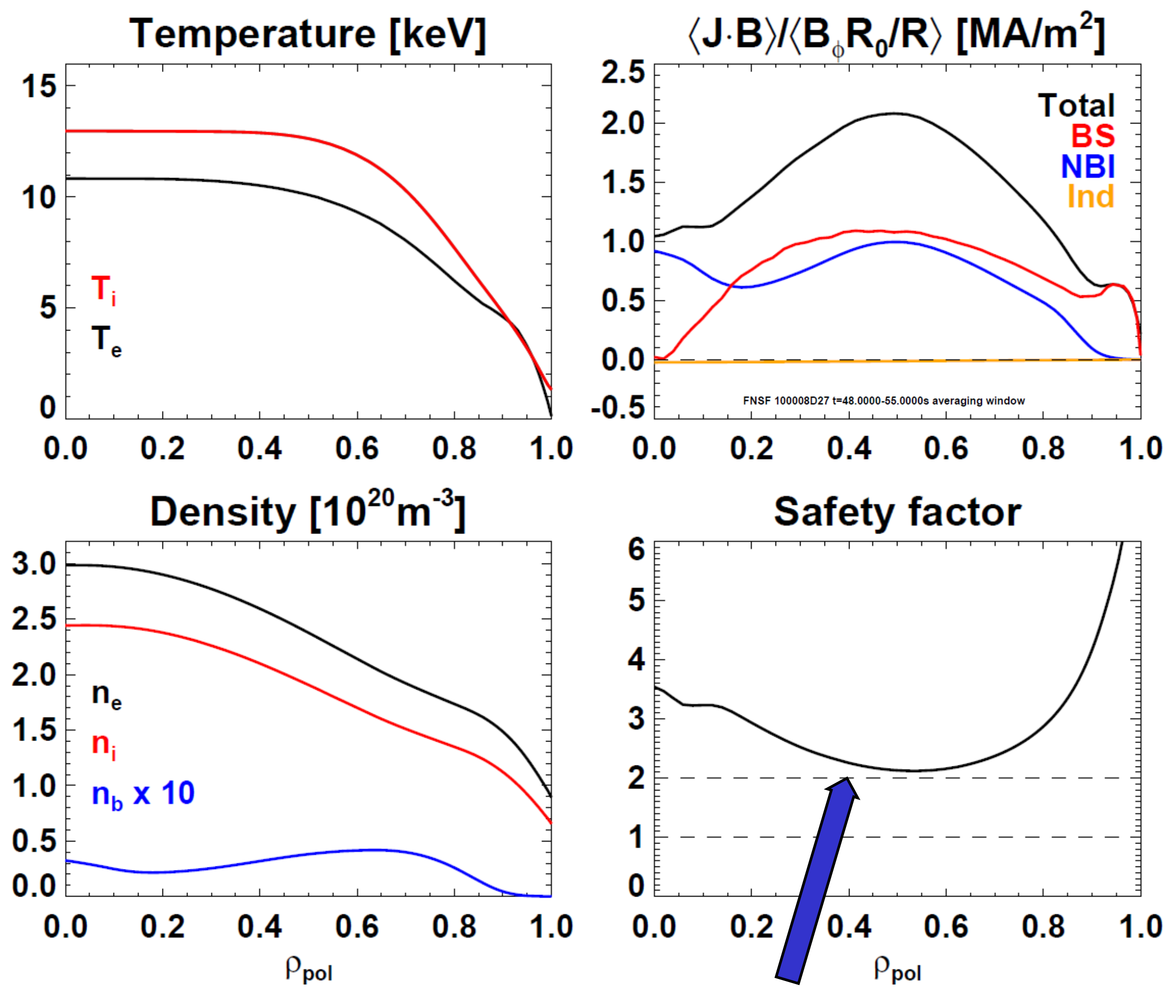
- **Fixed target parameters in DD:**
  - I_P = 7.5MA, β_N = 4.5, l_i = 0.5
  - $n_e / n_{Greenwald}$ = 0.75, $H_{98y,2}$ = 1.5
  - A=1.75, R=1.7m, B_T = 3T, κ = 2.8
  - $\langle T_e \rangle$=5.8keV, $\langle T_i \rangle$=7.4keV

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

Control q(0), q_{min}

Shine-thru limit

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

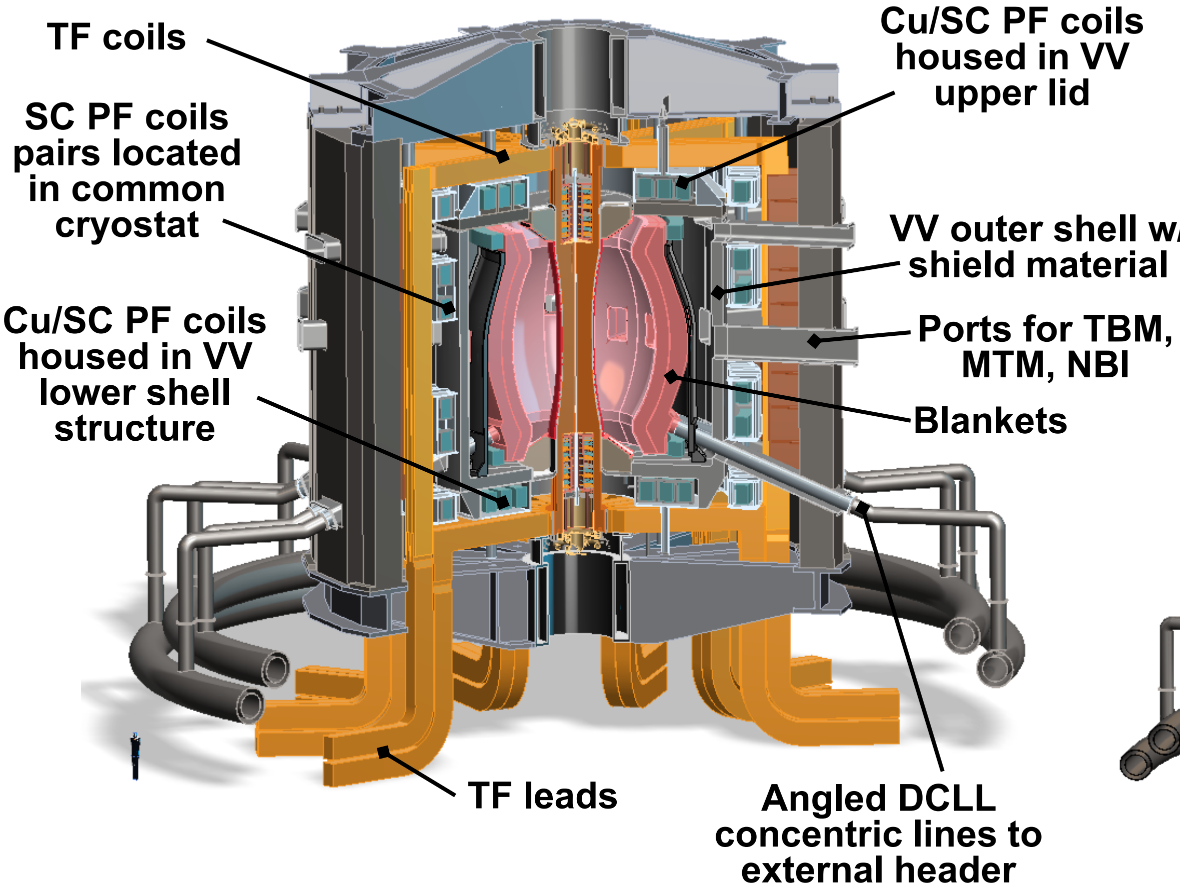


- Neoclassical χ_{ion}
- n_e / $n_{Greenwald}$ = 0.7
- $H_{98,y2}$ = 1.4
- I_P = 8.9MA, B_T = 2.9T
- f_{NICD} = 100%, f_{BS} = 65%
- P_{NNBI} = 80MW (0.5MeV)
- P_{fus} = 200MW (50-50 DT)
  – 2.6% alpha bad orbit loss
- Q_{DT} = 2.5
- β_N = 5.5, W_{tot} = 58MJ
  – W_{fast} / W_{tot} = 14%

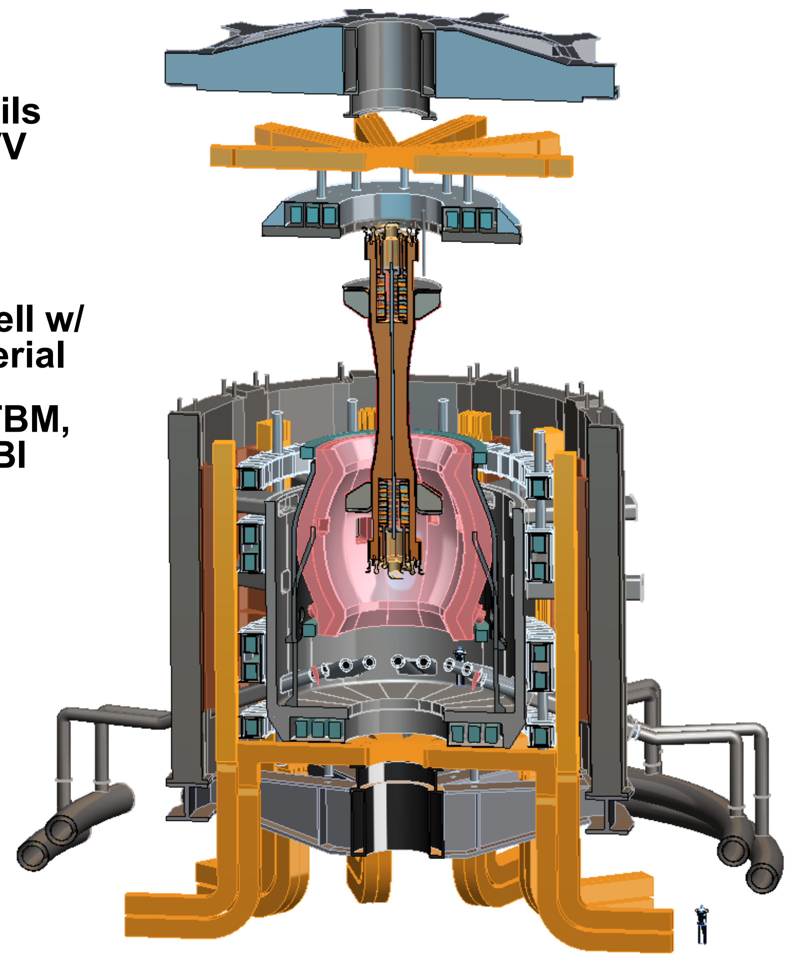
- Maintain q_{min} > 2
- q(0) / q_{min} controllable via R_{tan} and density

R=1.7m configuration with Super-X divertor

Design features



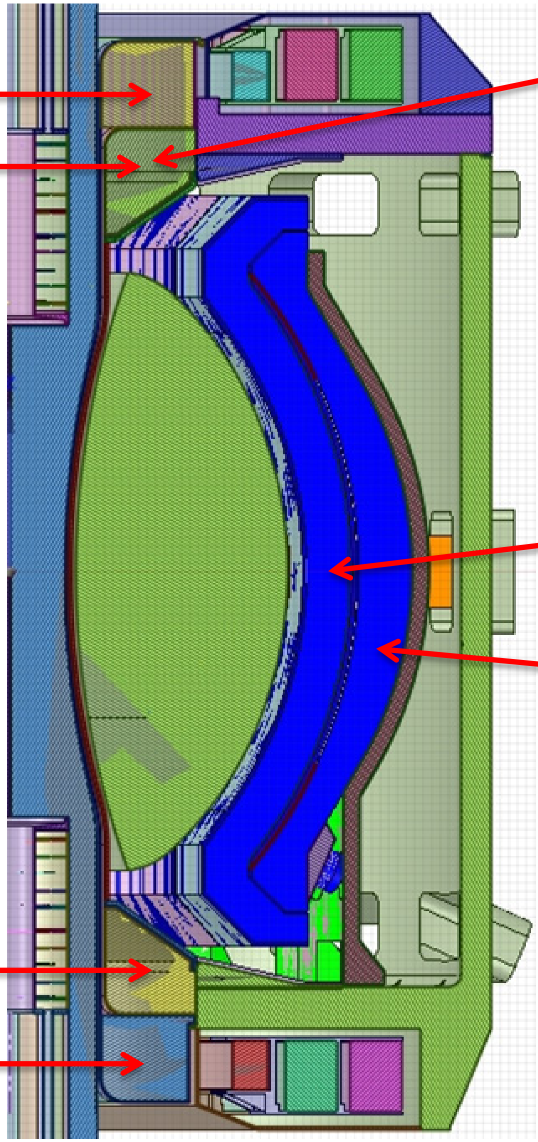
Vertical maintenance



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

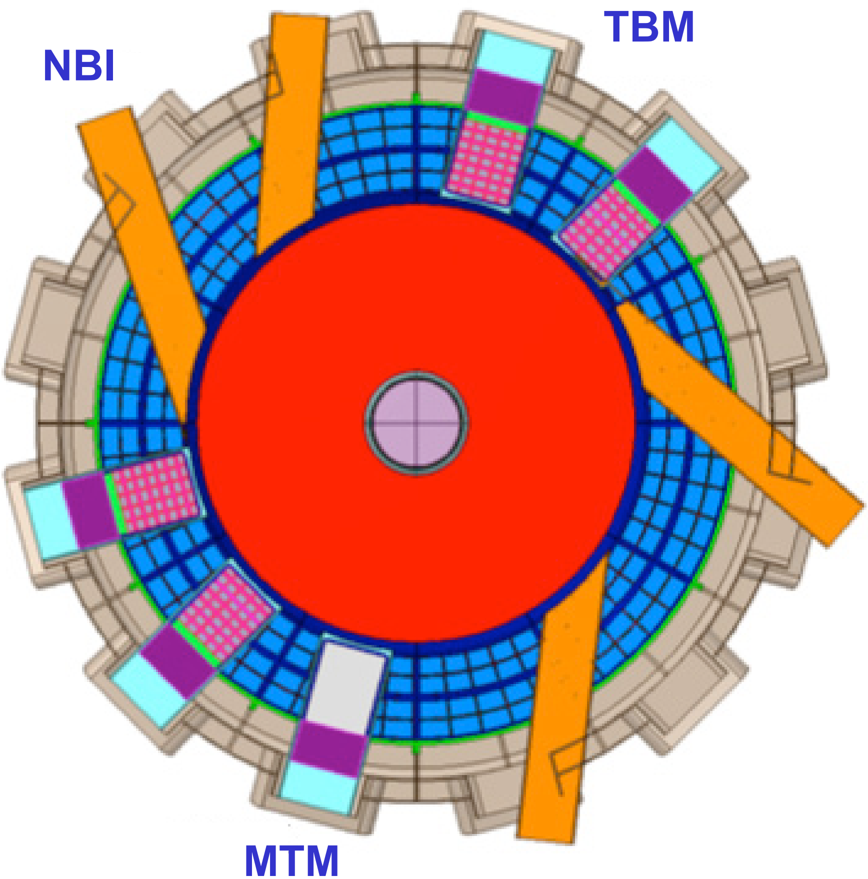
Outer Blanket Segment = 0.15

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

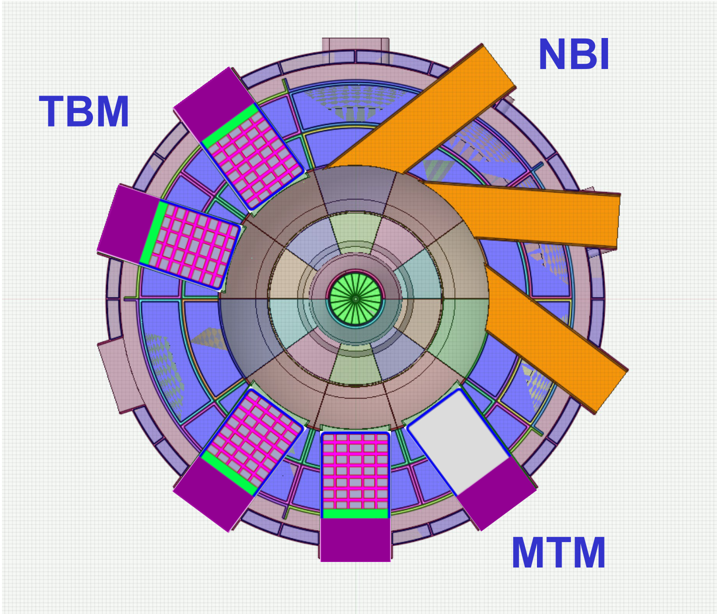
R=1.7m configuration

Summary of TBR vs. device size for A=1.7 Cu-TF ST-FNSF

R=1.7m: **TBR ≈ 1**



R=1.0m: **TBR < 1 (≈ 0.9)**



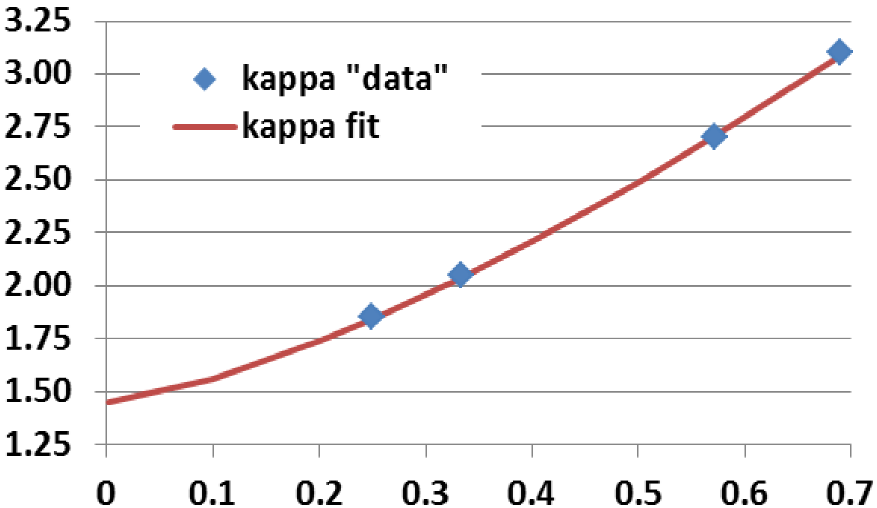
- **1m device cannot achieve TBR > 1 even with design changes**
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

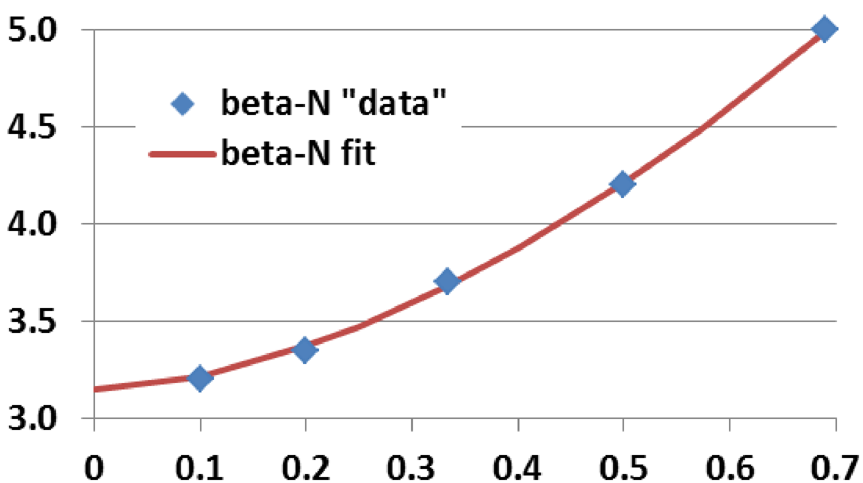
What is optimal A for HTS ST FNSF / Pilot?

Approach:

- Fix plasma major radius and heating power
  - Choose compact device ≤ R_0 = 3m to have any hope of achieving Pilot mission with AT/ST at ~few $B level
- Apply magnet and core plasma constraints (see subsequent slides)
- **Vary aspect ratio from A = 1.6 to 4**
- **Vary HFS WC shield thickness: 30-70cm**
- Calculate achievable Q_{DT} , Q_{eng} , required H_{98}
- Assess various trade-offs

Aspect ratio dependence of limits: $\kappa(\varepsilon)$, $\beta_N(\varepsilon)$





$\varepsilon = A^{-1}$

- **NSTX data at low-A (+ NSTX-U/ST-FNSF modelling)**
- **DIII-D, EAST for higher-A**
  - $\kappa \rightarrow 1.4$ for $A \rightarrow \infty$
- **Profile-optimized no-wall stability limit at $f_{BS} \approx 50\%$**
  - **Menard PoP 2004**
- $\beta_N \rightarrow 3.1$ **for** $A \rightarrow \infty$

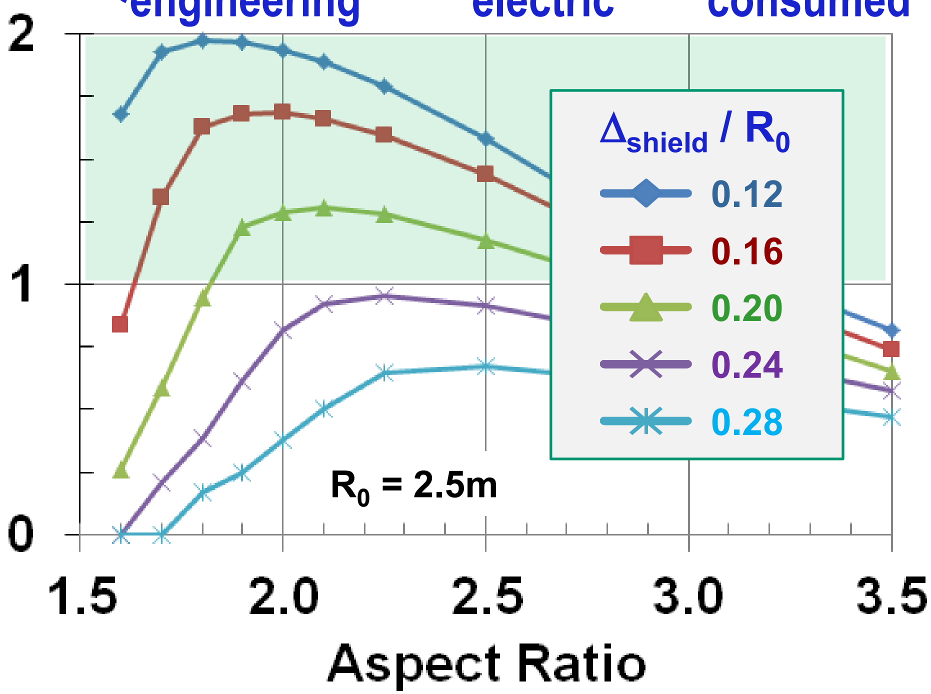
$$\beta_T \sim A^{-1/2}(1+\kappa^2)\beta_N^2/f_{BS}$$

$$\Rightarrow P_f \propto \epsilon(\kappa\beta_N B_T)^4$$

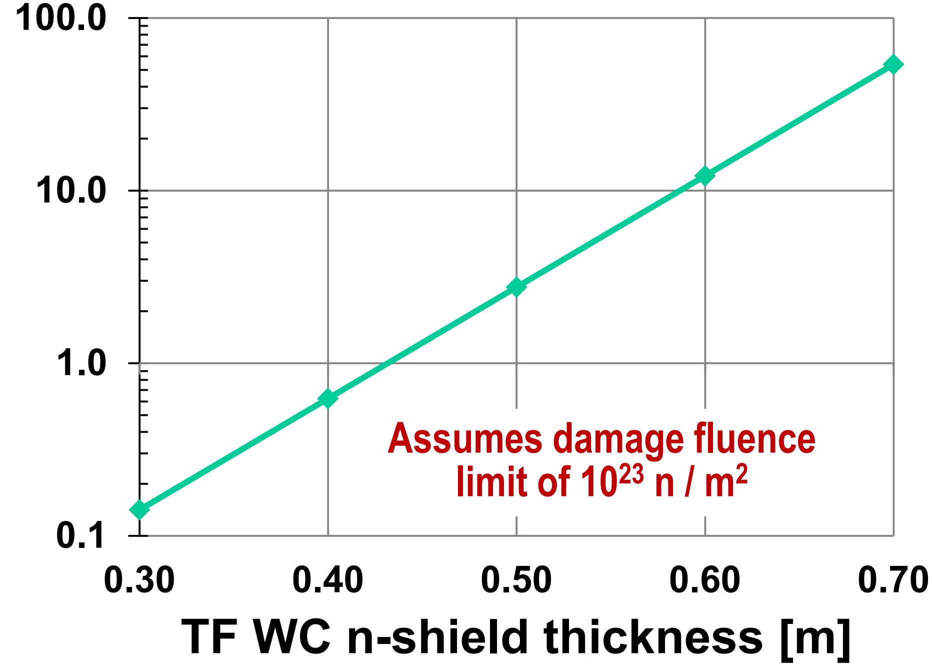
Q_{eng} maximized between A = 1.7-2.3 at fixed R_0, Optimal A depends on inboard WC shield thickness

Key assumption: can minimize or eliminate inboard T breeding, central solenoid

$Q_{engineering} = P_{electric} / P_{consumed}$



Avg Neutron Fluence [MWy/m^2]



Want Δ_{shield} / R_0 < ~20% for Q_{eng} > 1

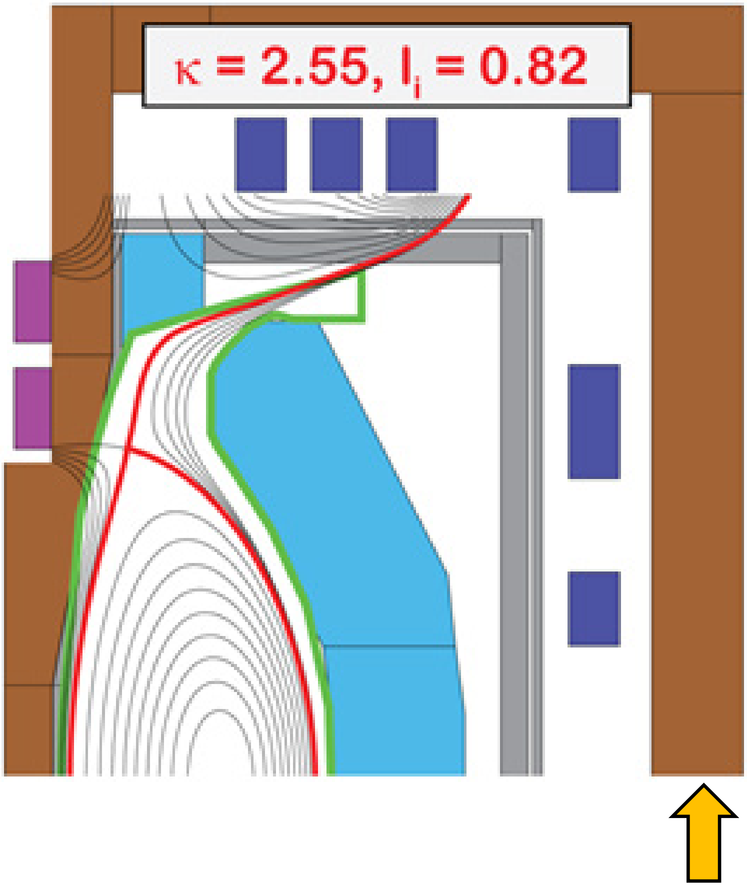
Selection of device HTS-ST performance goals

- Attempt to satisfy FNSF (fluence) and Pilot (net electric) goals
  - $6 MWy/m^2$ neutron wall loading (peak) at outboard midplane
  - $Q_{eng} \sim 1$ – similar to previous PPPL Pilot Plant Study
- Assume n-radiation damage limit of $3\text{-}5 \times 10^{22}/m^2$
  - HTS already tested to this damage fluence range (see next slide)

  → WC shield thickness ~ 60cm, $\Delta/R = 0.2$ → $R_0 = 3m$
- With small / no inboard breeding, optimal A ~ 2.1-2.4
- But, for TBR ~ 1 probably need A ≤ 2 → chose / try A=2
- **Chosen design point (so far):**
  - $R=3m$, $B_T = 4T$, $A=2$, $\kappa=2.5$, $\beta_N = 4.2$ (~no-wall limit)
  - $H_{98y2} \sim 1.7$, $H_{Petty} \sim 1.2\text{-}1.3$, $H_{ST} \sim 0.7$, $P_{fusion} \sim 500\text{-}600MW$
  - 80% Greenwald fraction, 50MW of 0.5-0.7 MeV NNBI
  - $I_P = 12MA$, double-swing of small OH provides ~ 2-3MA

PF coil layout, long-leg divertor, vertical maintenance similar between Cu and HTS FNSFs

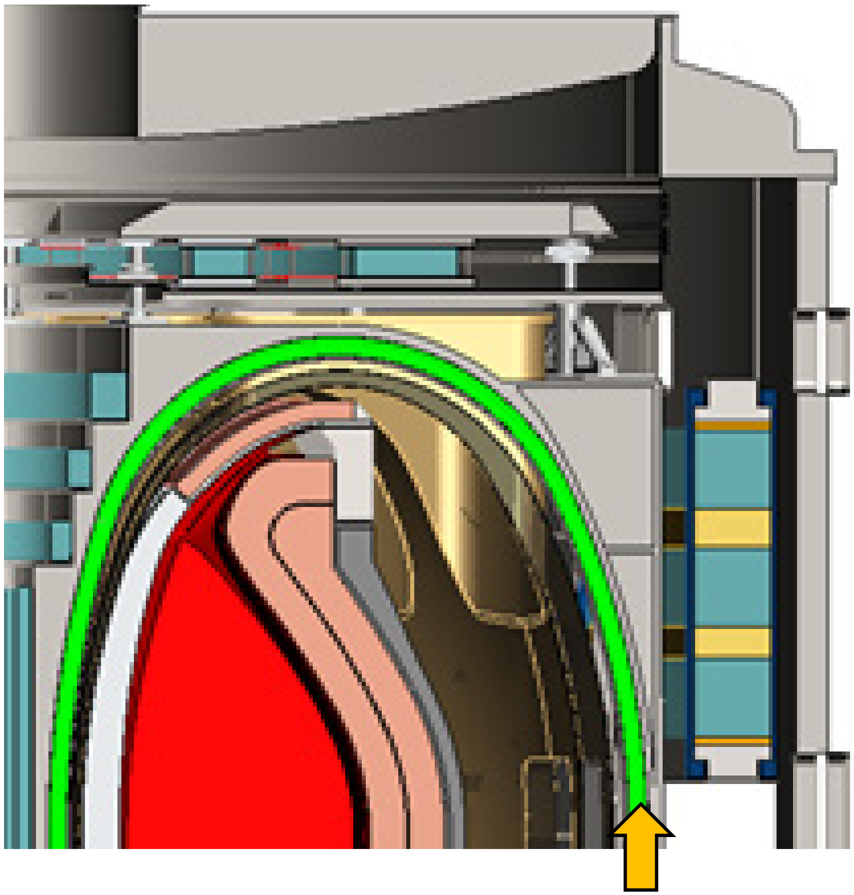
A=1.7 Copper TF FNSF



Outboard PF coils enclosed by TF coil

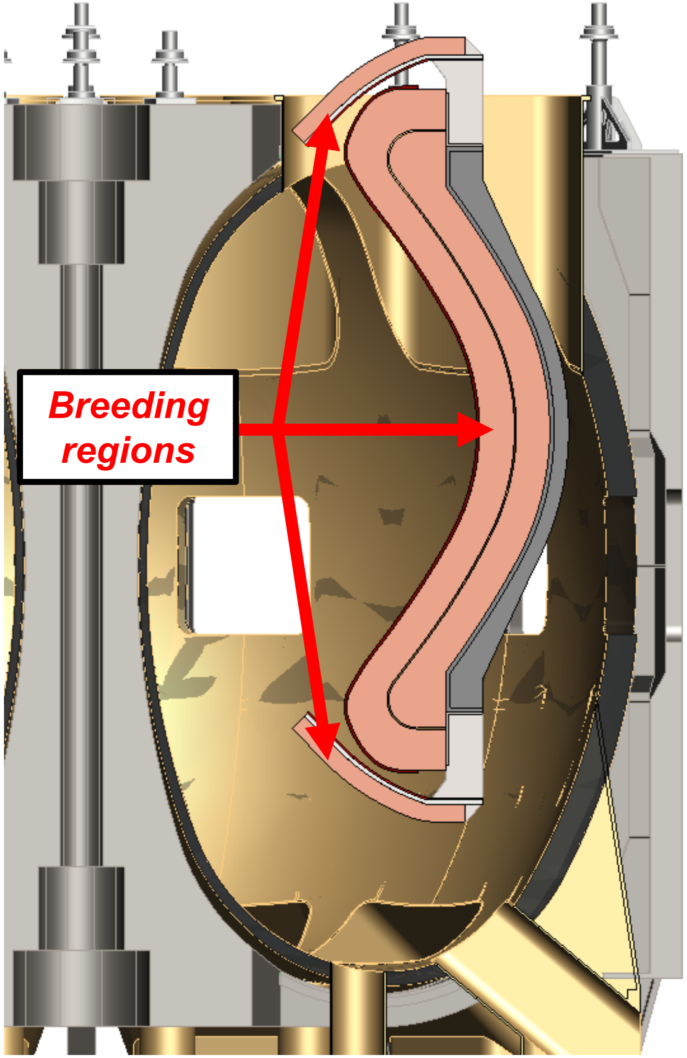
A=2 HTS TF FNSF/Pilot

VECTOR-like A, but with small CS



All PF coils outside TF coil

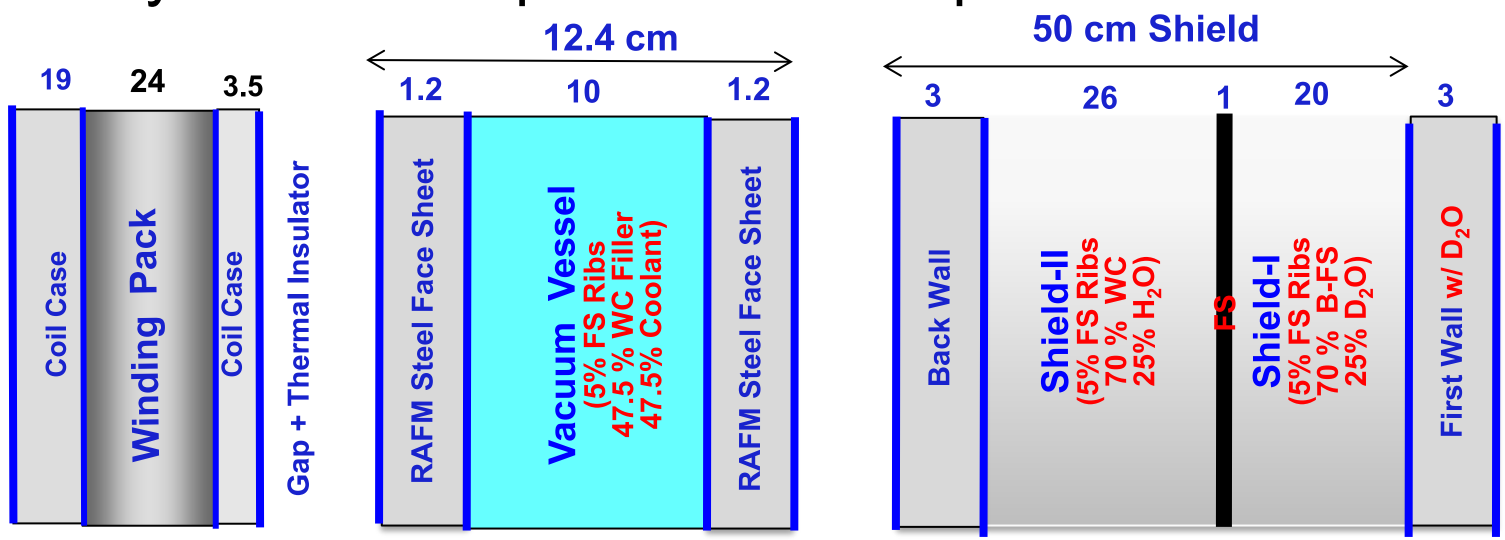
Vertical port maintenance used for OB blanket and divertor modules via separate cryostat for upper PFs



- Potential advantages of this low-A configuration:
  - Reduced part count + no / small inboard breeding → simplified maintenance (?)
- Need to include some breeding at top + bottom
  - Similar to Cu ST-FNSF
- 2016 - will study LM/Li wall and divertor compatibility with this HTS configuration

A=2 HTS ST Shielding Assessment

- **Focus on inboard (IB) shield** - main functions are:
  - Protect IB magnet for machine lifetime (3.1 FPY)
  - Enhance OB breeding by reflecting neutrons to OB
  - Generate low decay heat to control temperature response during accident → avoid using WC filler near FW.
- Two-layer IB shield presents best option:

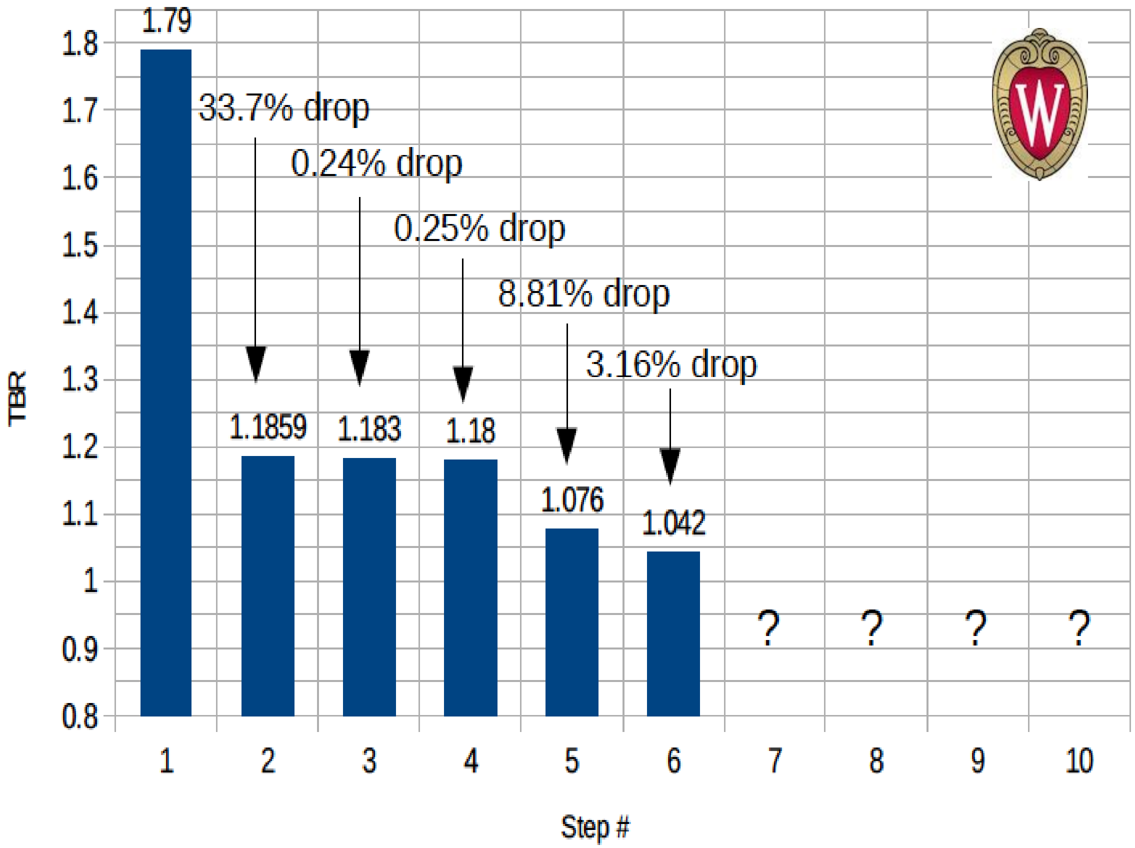


- 3-D analysis confirms radiation damage at IB magnet is near/below limits:
  - Peak fast n fluence to HTS (E_n > 0.1 MeV) 4.3×10^{18} n / cm^2
  - Peak nuclear heating @ WP 1.7 mW / cm^3
  - Peak dose to electrical insulator 4×10^9 rads
  - Total nuclear heating in IB magnet 8.7 kW

Detailed analysis of impact of blanket internals on TBR is being evaluated step-by-step

Steps:

1. 1-D infinite Cylinder: 100% LiPb breeder with 90% enriched Li
2. $Li_{17}Pb_{83}$ confined to OB blanket region and blanket behind divertor
3. 2 cm assembly gap between blanket modules
4. FS structure and FCI added to homogeneous mixture of blanket at top/bottom ends and behind divertor only
5. Materials assigned to 4 cm thick OB FW
6. Materials assigned to side, bottom/top, and back walls of blanket



To be added:

7. IB and OB cooling channels
8. SiC FCI
9. W Stabilizing shell
10. Penetrations.

Expect final TBR ≈ 0.95-1. Options to increase:

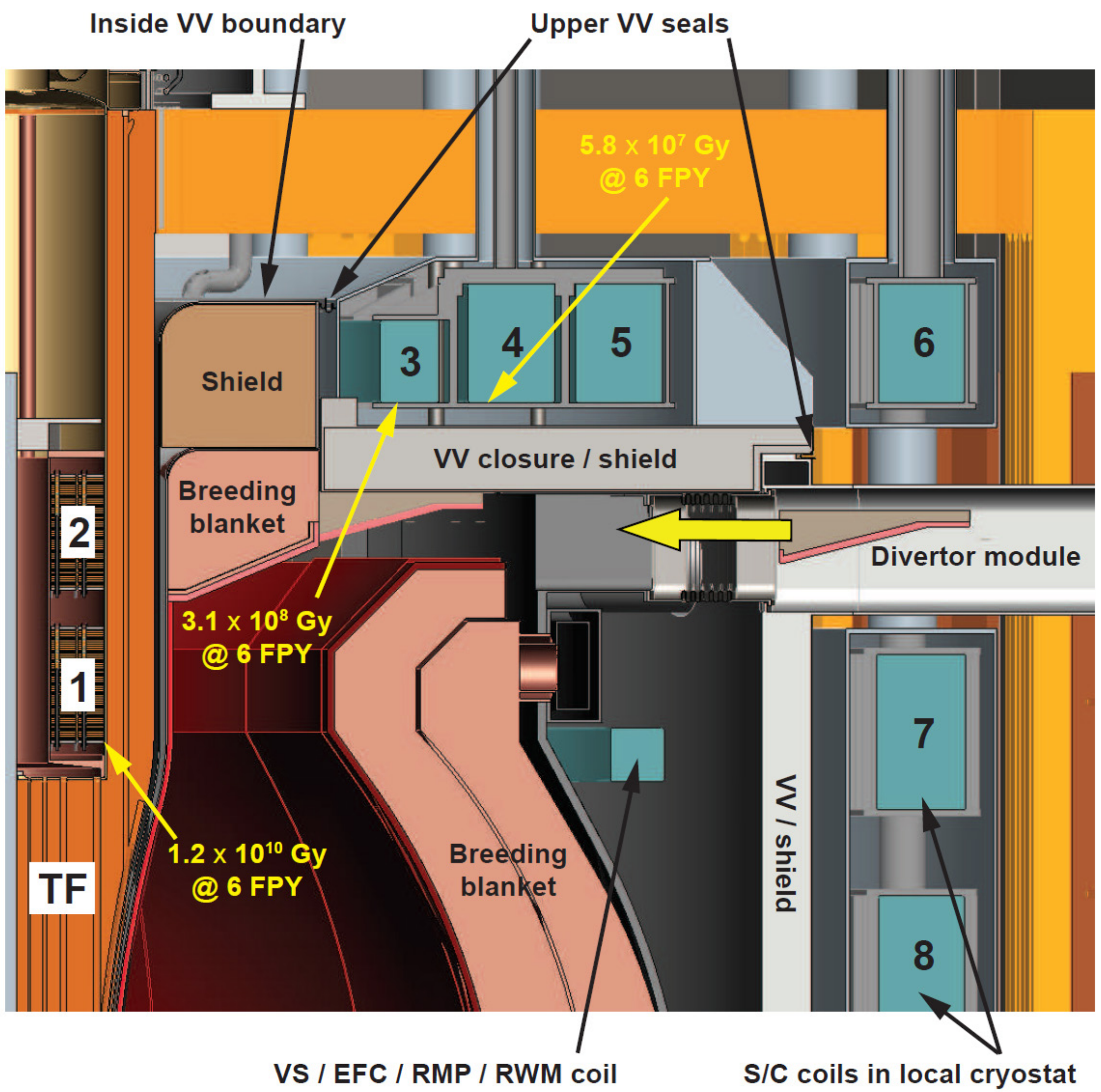
- **Thin inboard breeding region (assessing now)**
- **Reduce aspect ratio (reduces Q_{eng}, no CS)**

Summary

- Developed self-consistent A=1.7 Cu TF ST configurations w/ high TBR for fluence mission
  - R_0 = 1 m → TBR ~ 0.9
  - R_0 = 1.7m → TBR ~ 1.0
- Optimal A for fusion performance with HTS TF and small/no inboard breeding or CS is A ≈ 2
  - High confinement may be required to exploit higher toroidal field potentially achievable with HTS
- A ≈ 2 R_0 = 3m HTS FNSF / Pilot with fluence 6MWy/m^2 (peak) and Q_{eng} ~ 1 may be feasible

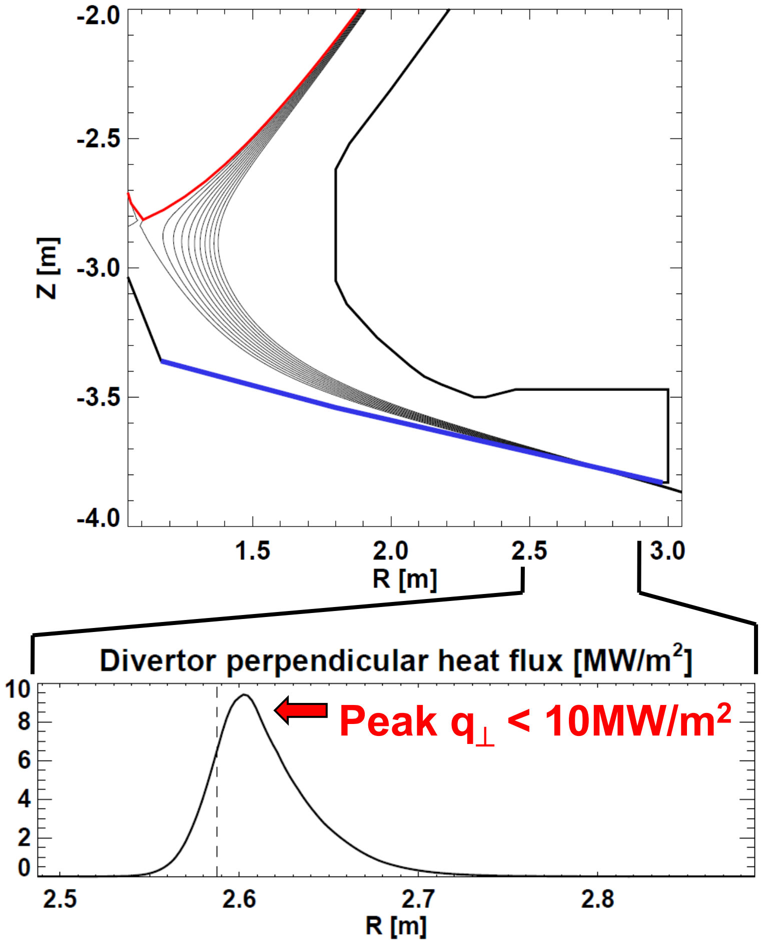
Backup – A=1.7 Cu TF FNSF

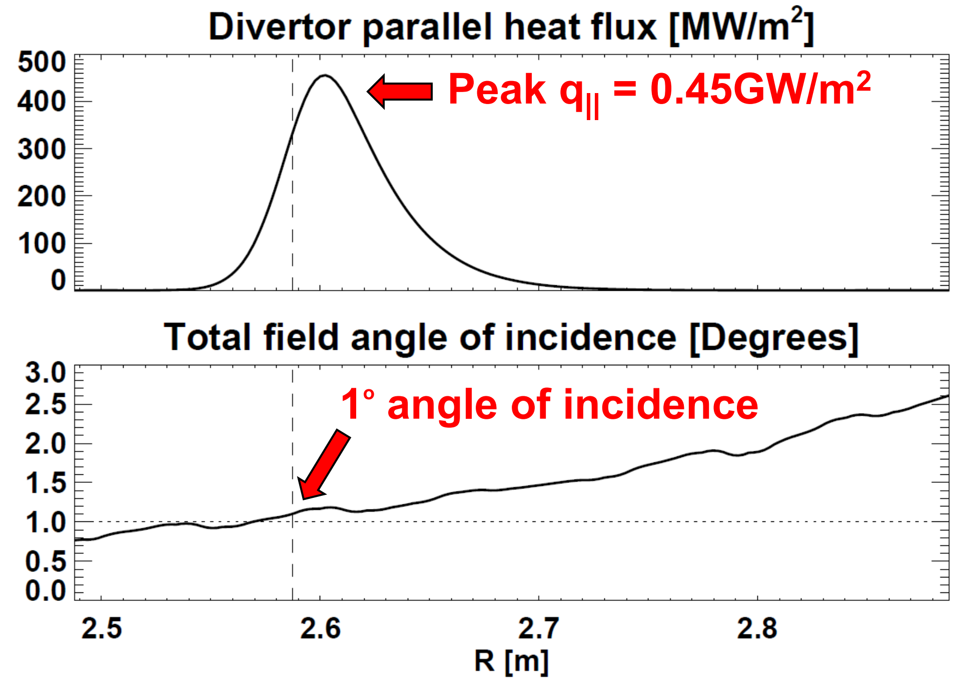
Peak radiation damage at PF coils are within allowable limits for different coil types (IB: Cu + MgO, OB: LTS)



- MgO limits: 10^{11} Gy
- LTS limits (Nb_3Sn):
  - Fast neutron fluence: 10^{23} n/m^2 (E_n > 0.1MeV)
    - PF3: 1.15×10^{23} n/m^2
  - Peak Dose to Insulator: 2×10^{10} rads
    - PF3: 1.08×10^{10} rads
  - Peak Nuclear Heating: 2 kW/m^3
    - PF3: 2.2 kW/m^3

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





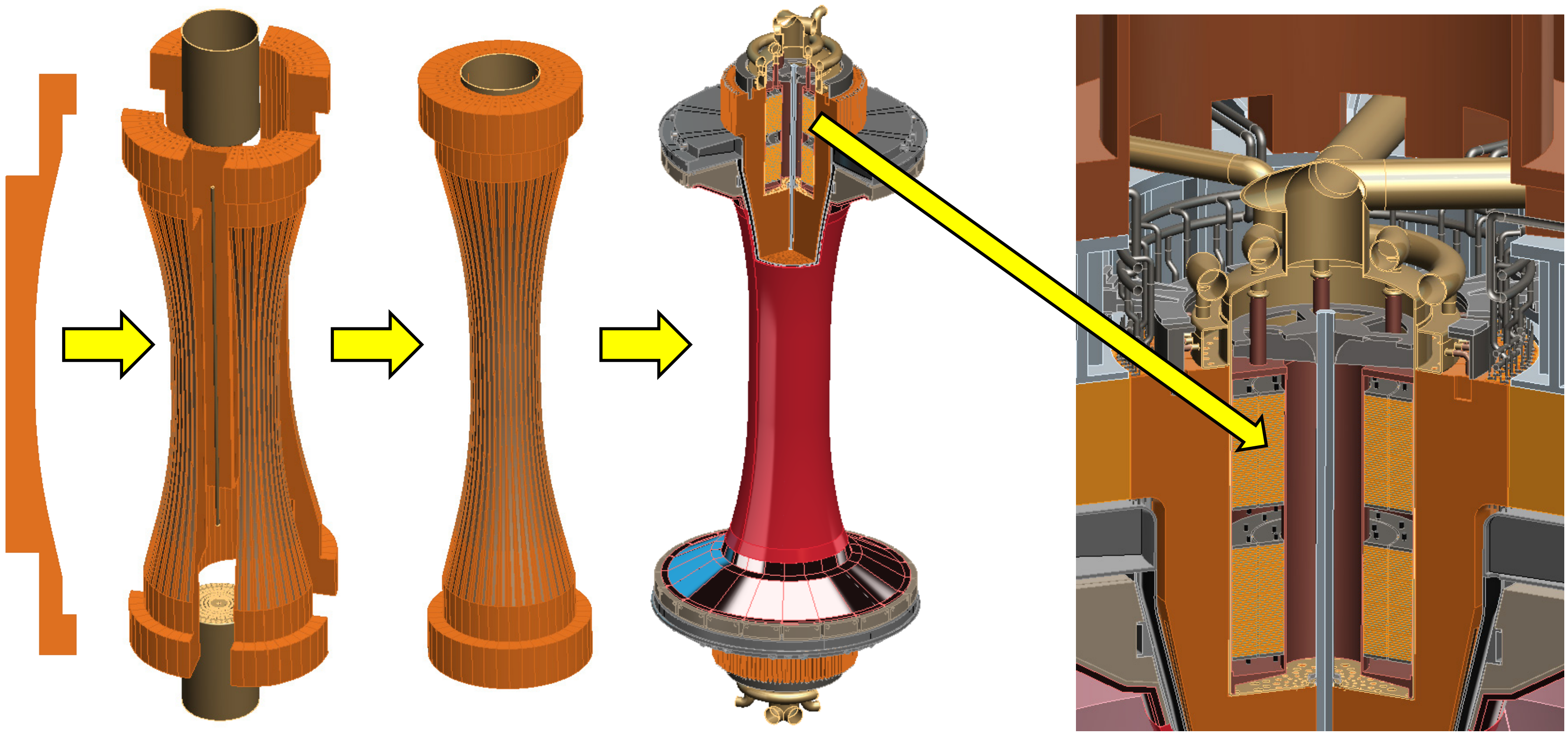
Partial detachment expected to further reduce peak $q_\perp$ factor of 2-5×

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

- 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

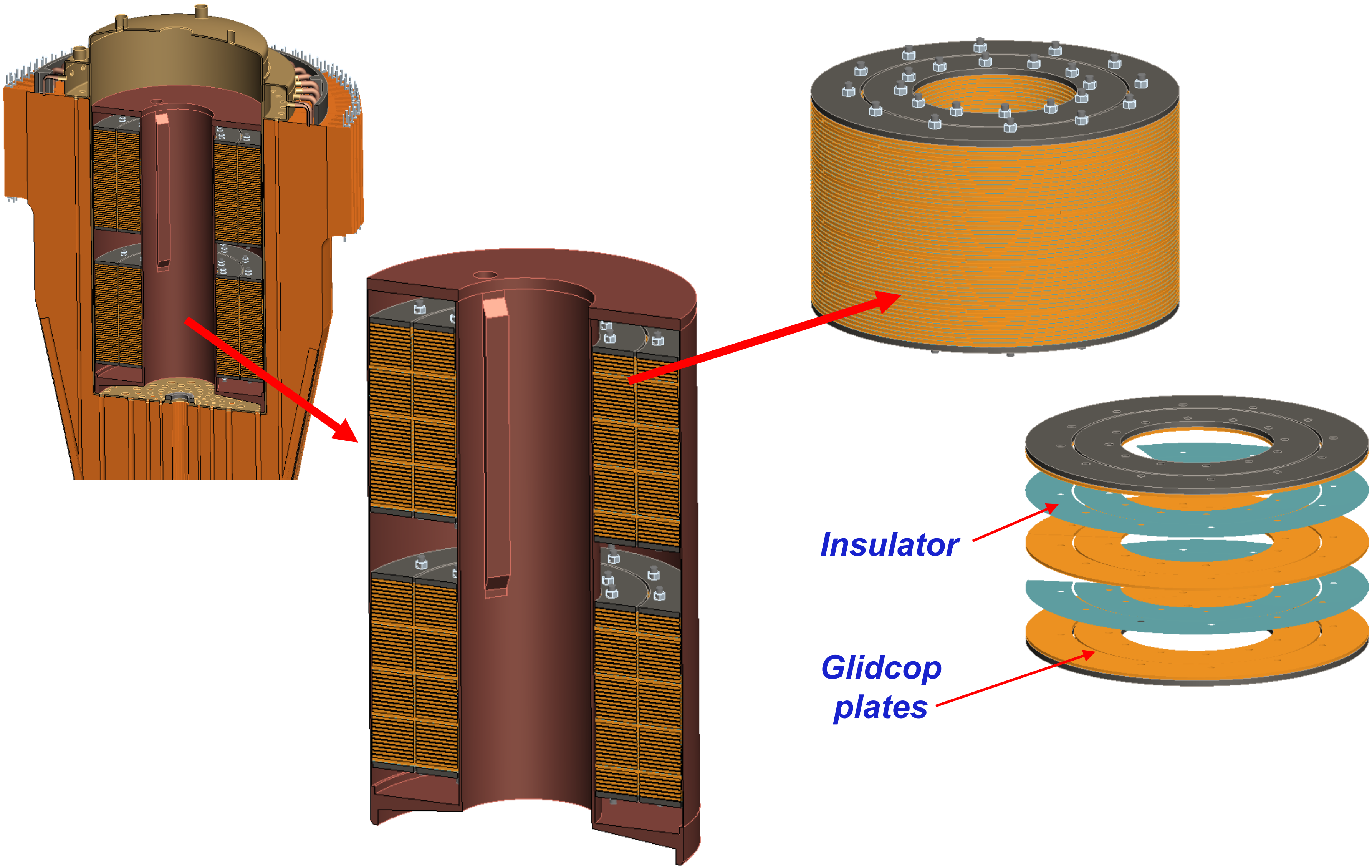
Eich NF 2013: $\lambda_{q\text{-int}} = \lambda_q + 1.64 \times S$, λ_q = 0.78mm, $S \approx \lambda_q$ (closed divertor)

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

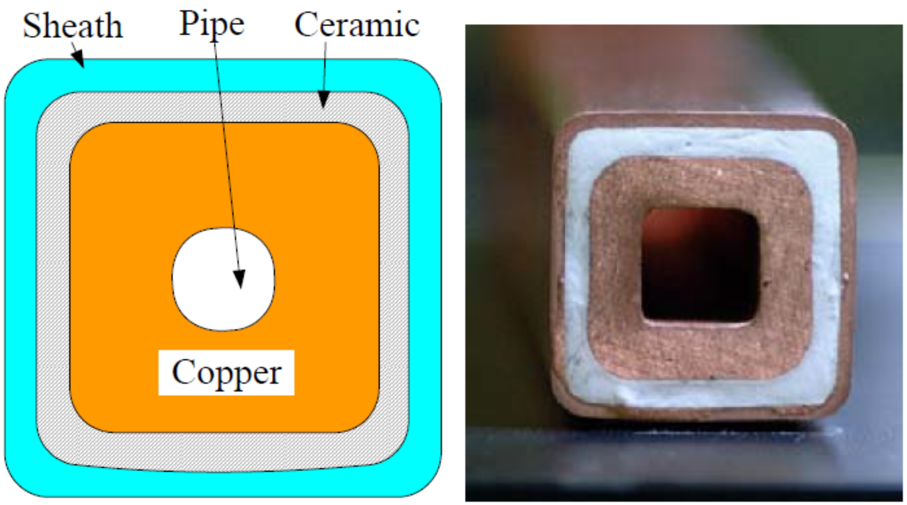


Fig. 3 Cross section of MIC

Table 1: Comparison of radiation resistant

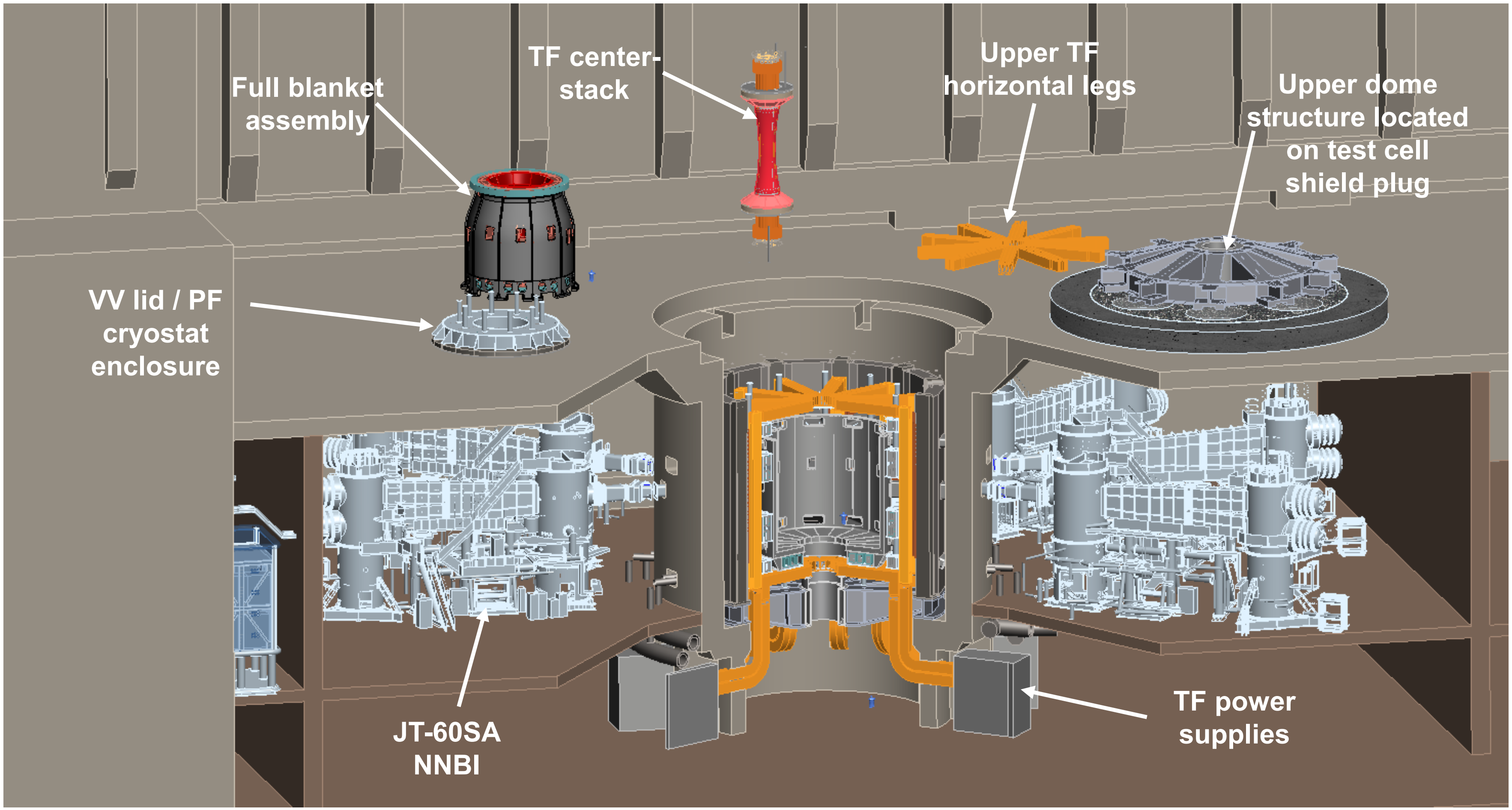
	Organic		Inorganic
Insulation	Epoxy	Polyimide	MgO
Resistant	$>10^7$ Gy	$>10^9$ Gy	$>10^{11}$ Gy

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)

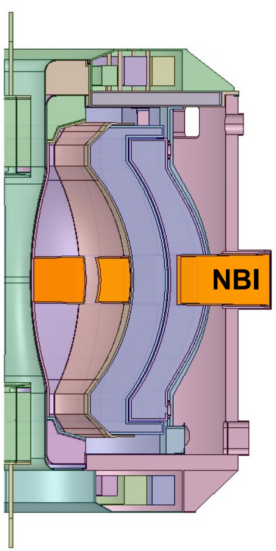
Kuanjun Fan [1,A)], Hiroshi Matsumoto [A)], Koji Ishii [A)], Noriyuki Matsumoto [B)]
A) High Energy Accelerator Research Organization (KEK)
1-1 OHO, Tsukuba, Ibaraki, 305-0801, Japan
B) 2NEC/Token

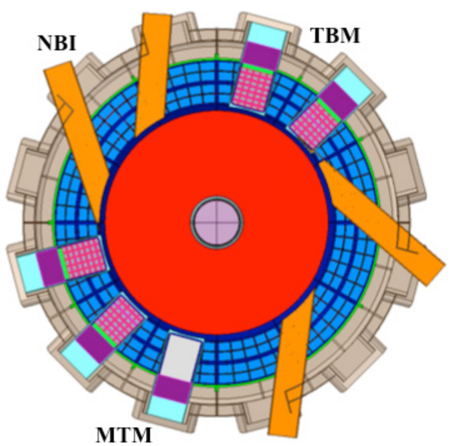
R=1.7m ST-FNS facility layout
using an extended ITER building

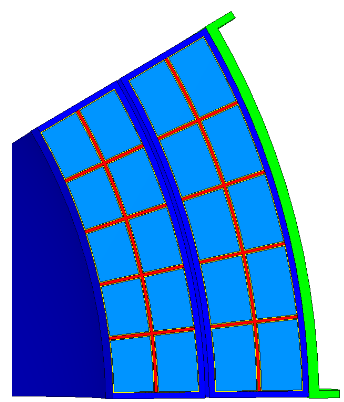


ST-FNSF shielding and TBR analyzed with sophisticated 3-D neutronics codes

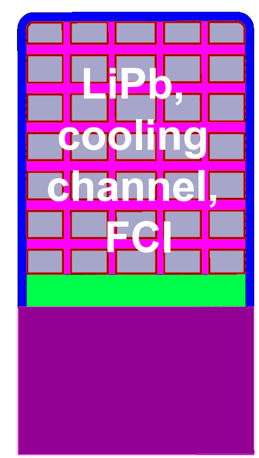
- CAD coupled with MCNP using UW DAGMC code
- Fully accurate representation of <u>entire torus</u>
- No approximation/simplification involved at any step:
  - Internals of two OB DCLL blanket segments modeled in great detail, including:
    - FW, side, top/bottom, and back walls, cooling channels, SiC 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



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

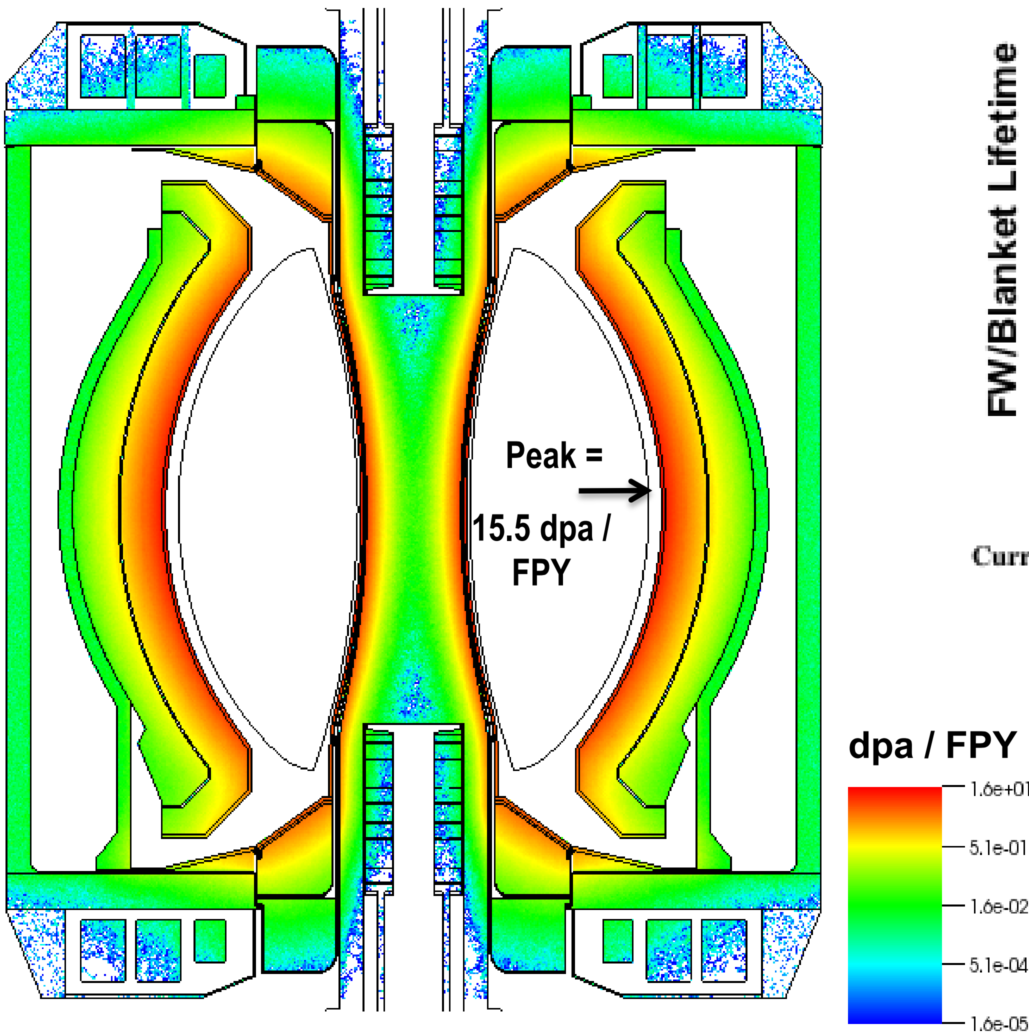
Availability 10-50%
30% avg

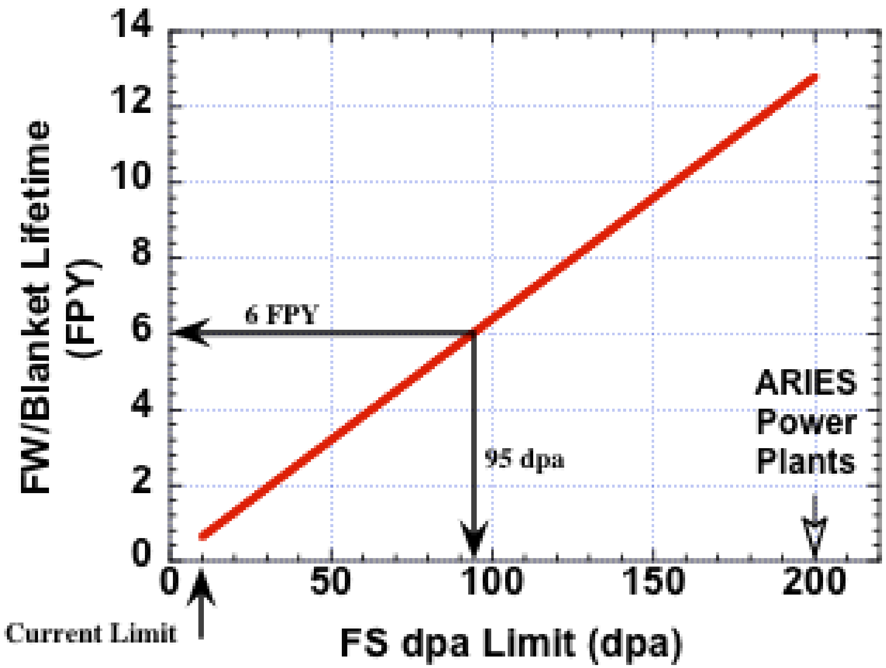
} 6 Full Power Years (FPY)

Neutron source distribution

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

R=1.7m configuration



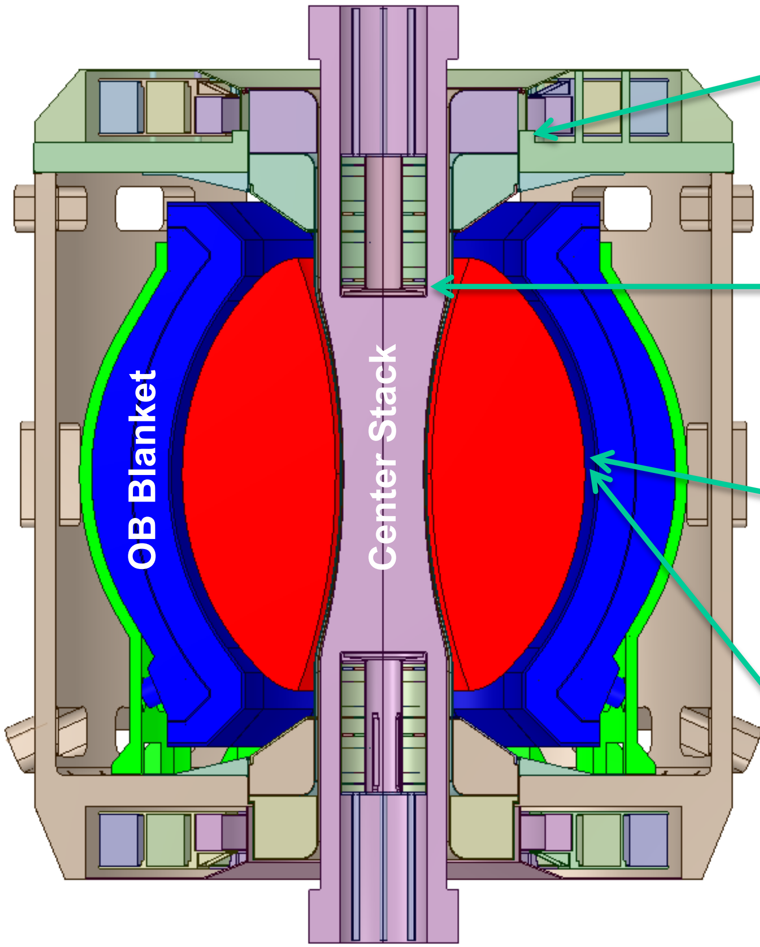


FW/blanket could operate for 6 FPY if allowable damage limit is 95 dpa

→ **Peak EOL Fluence = 11 MWy/m^2**

Peak radiation damage at PF coils are within allowable limits for different coil types (Cu with MgO, LTS/HTS)

R=1.7m configuration



3-D Neutronics Model of Entire Torus

Dose to MgO insulator = 3.1×10^{8} Gy @ 6 FPY < 10^{11} Gy limit

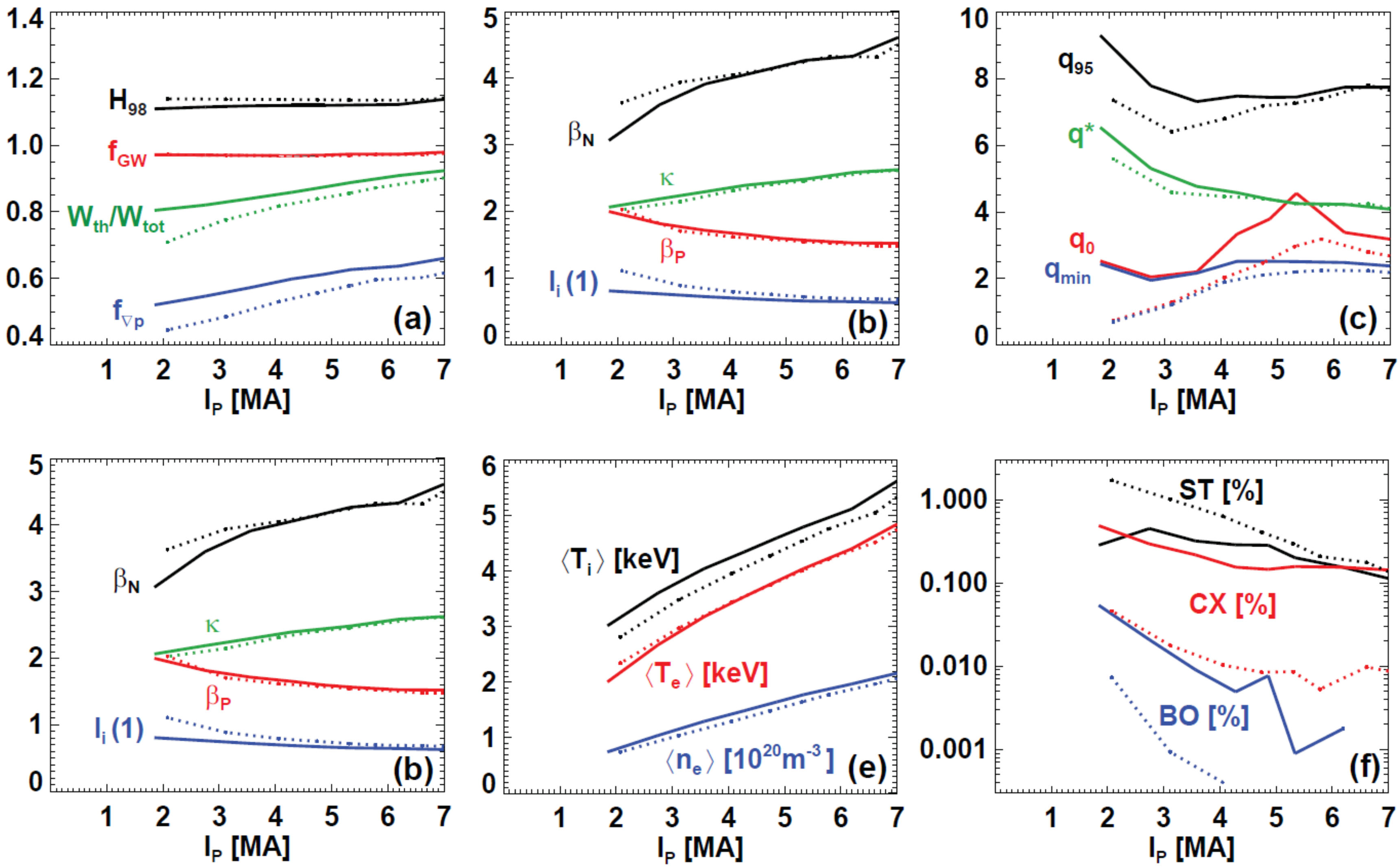
Dose to MgO insulator = 1.2×10^{10} Gy @ 6 FPY < 10^{11} Gy limit

Peak dpa at OB midplane = 15.5 dpa / FPY

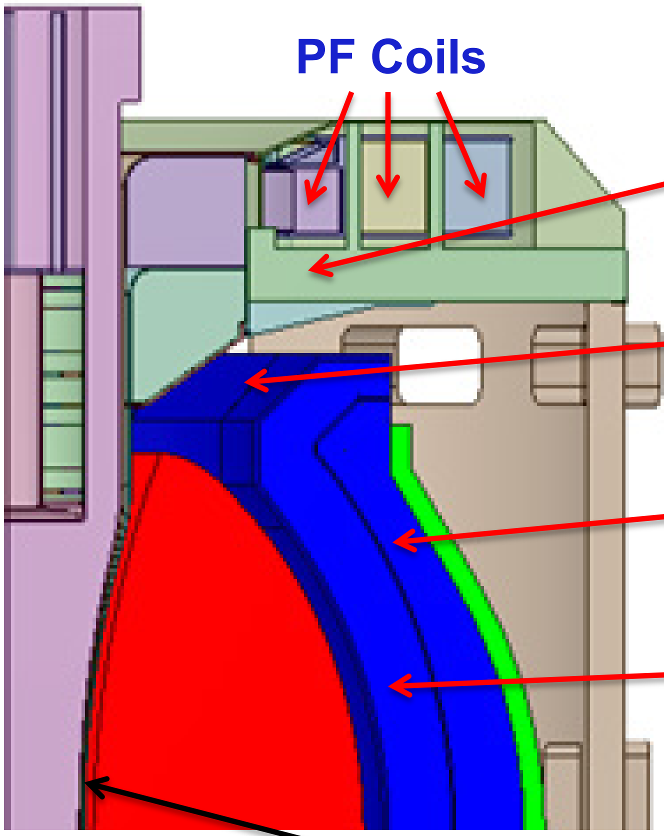
Peak He production at OB midplane = 174 appm/FPY

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

0.5MeV NNBI well confined down to ~2MA



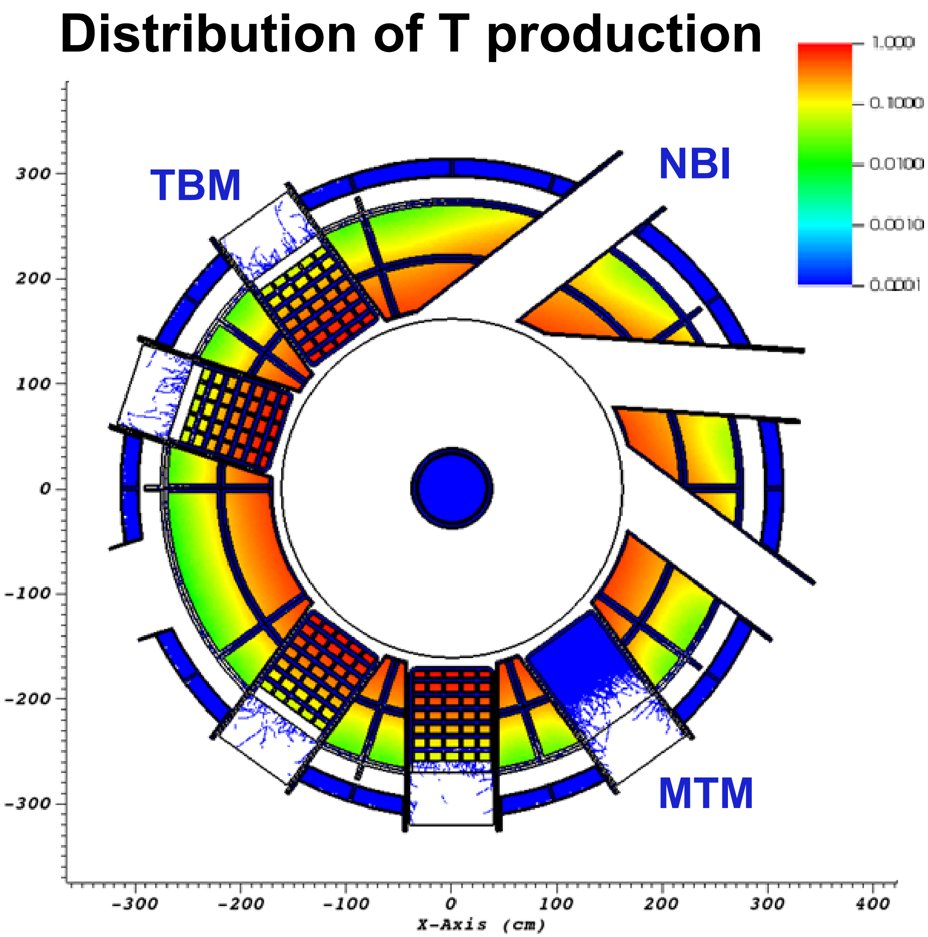
Options to increase TBR > 1

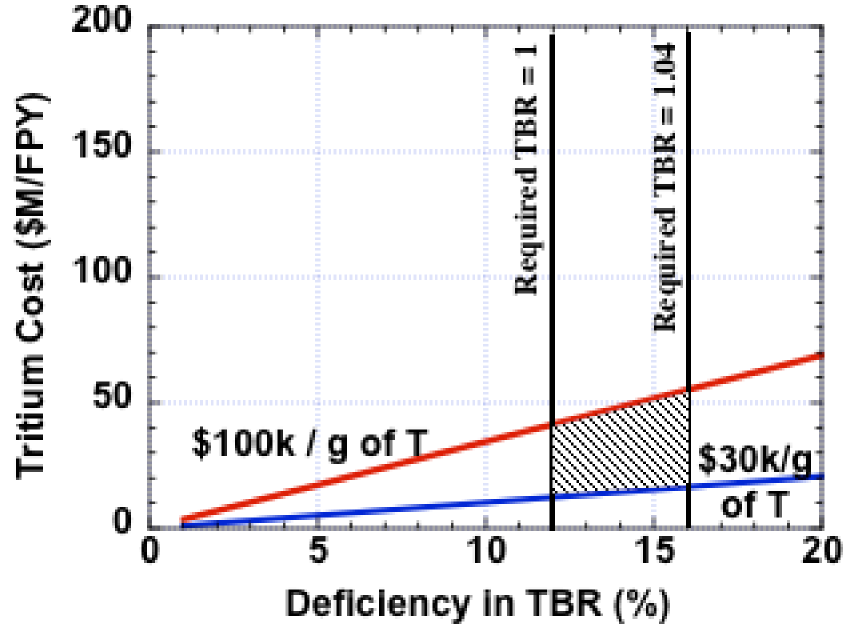


- Add to PF coil shield a thin breeding blanket (ΔTBR ~ +3%)
- Smaller opening to divertor to reduce neutron leakage
- Uniform OB blanket (1m thick everywhere; no thinning)
- Reduce cooling channels and FCIs within blanket (need thermal analysis to confirm)
- Thicker IB VV with breeding

Potential for TBR > 1 at R=1.7m

R_0 = 1m ST-FNSF achieves TBR = 0.88



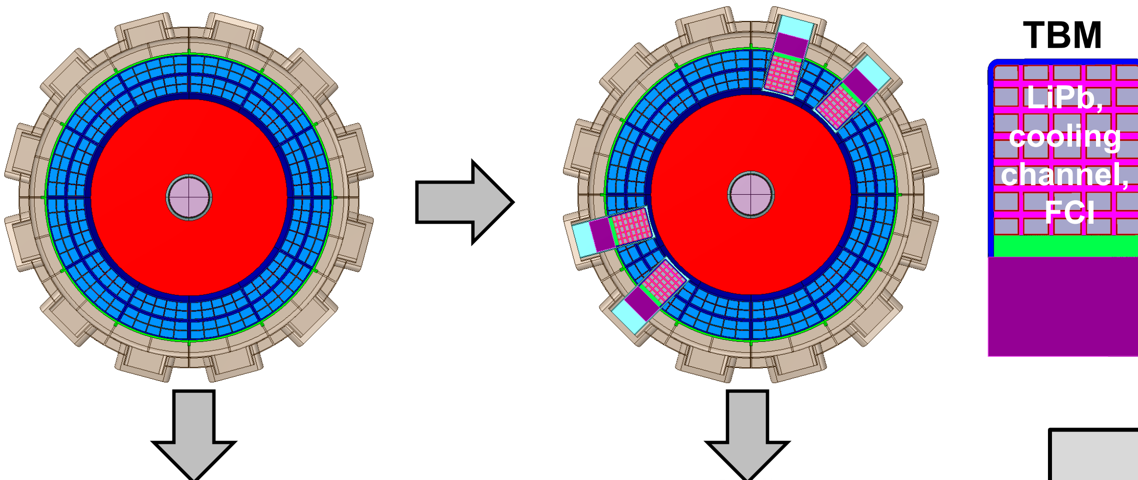


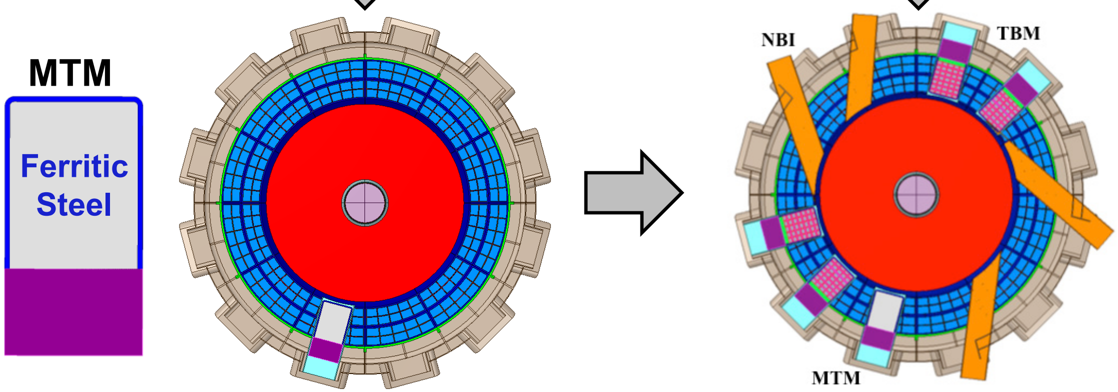
- **1m device cannot achieve TBR > 1 even with design changes**
- **Solution**: purchase ~0.4-0.55kg of T/FPY from outside sources at $30-100k/g of T, costing $12-55M/FPY

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%**

Backup – A=2 HTS FNSF / Pilot

Plasma constraints

- Fix plasma major radius at R_0 = 2.5-3m
  - Chosen to be large enough to allow space for HTS neutron shield and access $Q_{eng} > 1$
- Inboard plasma/FW gap = 4cm
- Use ε dependent $\kappa(\varepsilon)$, β_N (ε) (see next slide)
- Greenwald fraction = 0.8
- q* not constrained
  - q* is better ε-invariant than q_{95} for current limit
  - Want to operate with q* > 3 to reduce disruptivity
- 0.5MeV NNBI for heating/CD – fixed P_{NBI} = 50MW
- H_{98y2} adjusted to achieve full non-inductive CD

Engineering constraints

- Magnet constraints (T. Brown ST-HTS Pilot, K-DEMO)
  - Maximum stress at TF magnet = 0.67-0.9GPa
  - Maximum effective TF current density = 65MA/m^2
  - OH at small R → higher OH solenoid flux swing for higher A
- Shielding / blankets
  - Assume HTS fluence limit of $3\text{-}5\times10^{22}$
  - No/thin inboard blanket, ~1+ m thick outboard blanket
    - 10x n-shielding factor per 15-16cm WC for HTS TF
    - Also 8cm inboard thermal shield + other standard radial builds
- Electrical system efficiency assumptions:
  - 30% wall plug efficiency for H&CD - typical of NNBI
  - 45% thermal conversion efficiency - typical of DCLL
    - Also include pumping, controls, other sub-systems
    - See Pilot Plant NF 2011 paper for more details

HTS performance vs. field and fast neutron fluence

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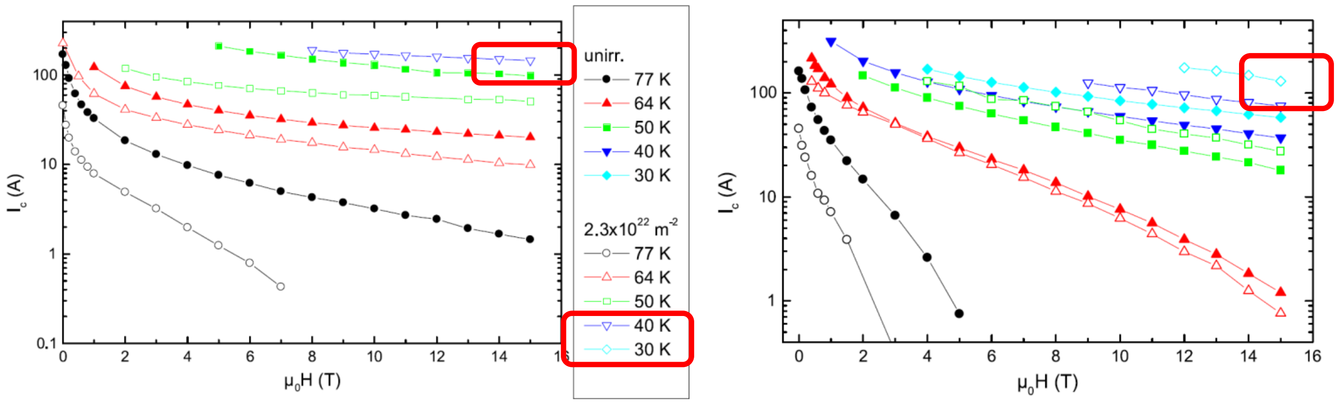

Figure 6. Critical currents (ASC-40) in magnetic fields applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) before and after irradiation to a fast neutron fluence of $2.3 \cdot 10^{22}$ m^{-2}.

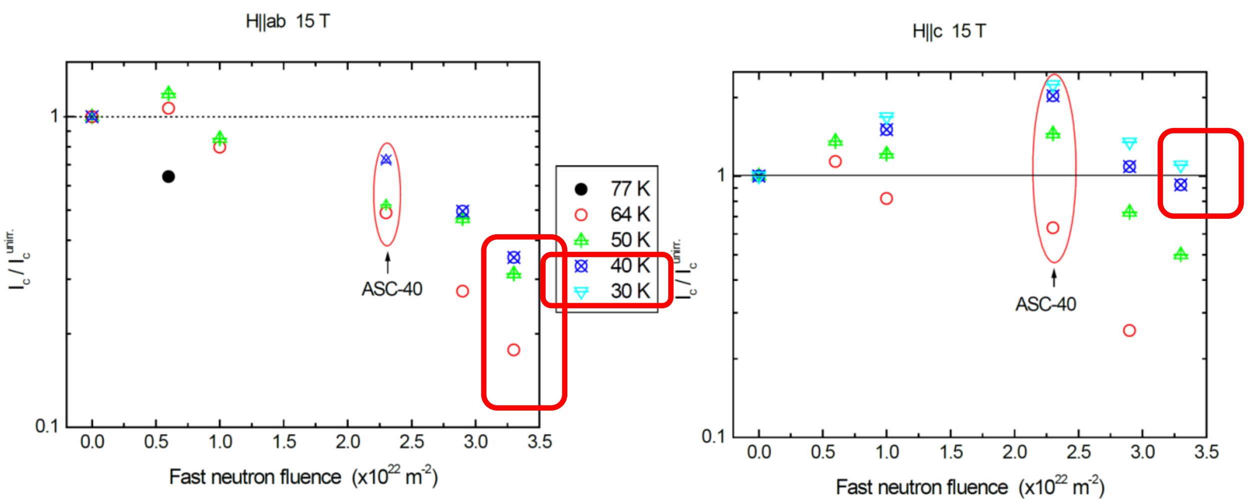

Figure 8. Normalized critical currents in a magnetic field of 15 T applied parallel to the ab-plane (left) and parallel to the *c*-axis (right) as a function of neutron fluence.

Very preliminary work suggests annealing may be able to reverse some effects of radiation damage in HTS conductors

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Physics Procedia

Positron Studies of Defects 2011

Investigation of Radiation Affected High Temperature Superconductors - YBCO

J. Veterníková[a,*], M. Chudý[b], V. Slugeň[a], S. Sojak[a], J. Degmová[a], J. Snopek[a]

[a] Institute of Nuclear and Physical Engineering, Faculty of Electrical Engineering and Information Technology, Slovak University of Technology, Ilkovičova 3, 812 19 Bratislava, Slovakia

[b] Vienna University of Technology – Atominstitut, Stadionallee 2, 1020 Vienna, Austria

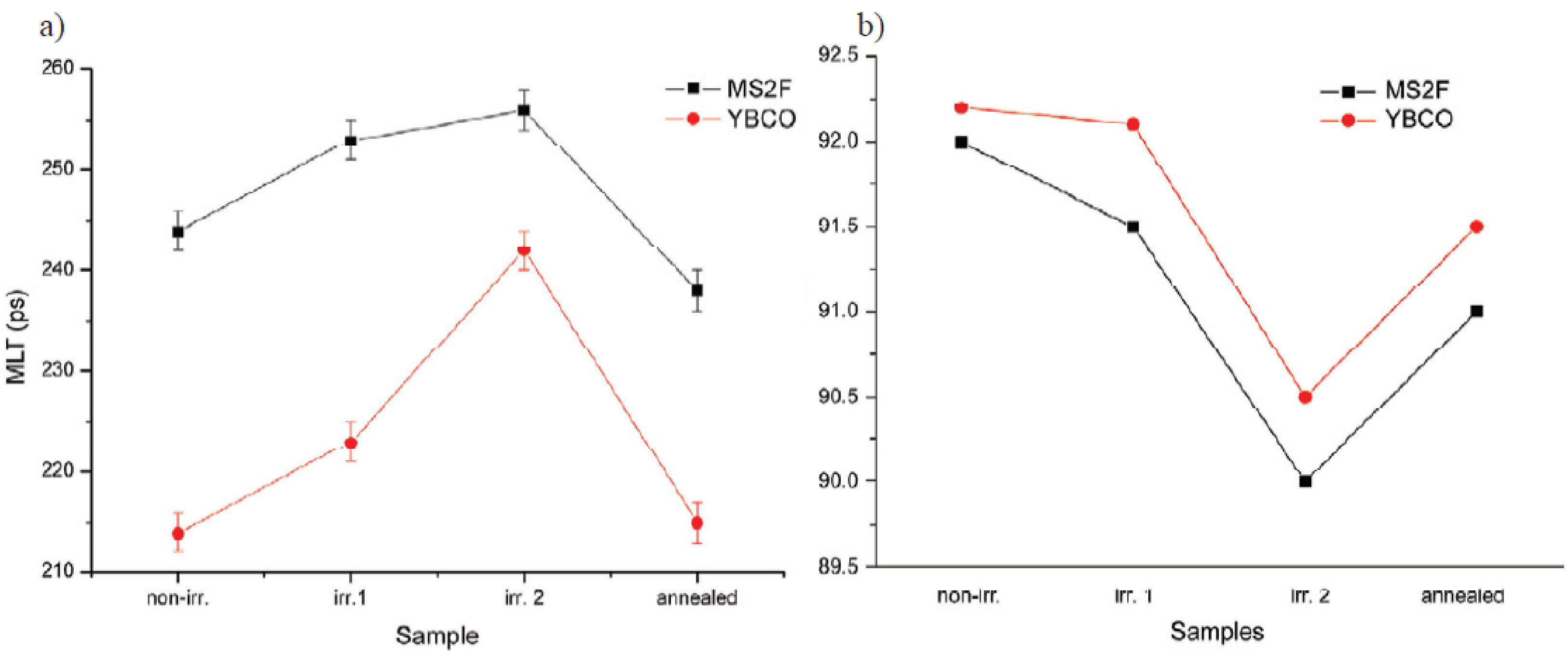


Fig. 3 The change of positron mean lifetime (MLT) after irradiation and annealing (a); Critical temperatures measured in SQUID magnetometer (b).

Radial build (includes small OH) - T. Brown

	3.00 m R0 CCFE HTS ST radial build						
	10 TF coils	2.00	AR				
	COMP BUILD, Z=0				TOTAL	TOTAL	
	(in)	(mm)	(in)	(mm	(mm)	(in)	
	Machine Center				0		180
	TF center bore	68	2.677		68.0		
	OH coil	225.0	8.858	225	293.0	11.535	
	OH - TF gap	10	0.394		303.0		
TF inbd leg	Ext structure	190.0	7.480		493.0		0 needed to add to nose stu
	Clearance	2.00	0.079			230.000	
	ground wrap	4.00	0.157		499.0	19.646	
	Winding pack thk	240	9.449		729.0	28.701	
	ground wrap	4.00	0.157		733.0	28.858	
	Clearance	2.00	0.079				
	Ext structure	35.0	1.378	477	780.0	30.709	
	TF-OH TPT	2.0	0.079				
	VV TPT	5.0	0.197				
	wedge coil asmbly fit up	1.0	0.039				
Thermal Insu	Thermal Shield	8.0	0.315				
	Min TF/VV Gap	5.0	0.197	21	801	31.535	
inbd VV	VV shell thk	12	0.472				
	borated water/W shield	100	3.937				
	VV shell thk	12	0.472	124	925	36.417	
	WC inboard shield	500	19.685		1425	56.102	
	VV TPT	5.0	0.197				
FW	FW	30	1.181		1460	57.480	
	Plasma SO	40	1.575		1500	59.055	
	Plasma minor radii	1500	59.055				
Plasma R0					3000		

4.00	B0
16.5	Bmax
854	Ave Tresca stress (Mpa)

CALCULATIONS for OH at inner bore

$B = \mu_0 J_0 (Ro - Ri)$ where $\mu_0 = 4\pi \times 10^{-7}$

Assume J_0 (MA/m^2) 70

B (T) = 19.8

OH stress = $B_i R_i J_0 / 2$

stress = 47.11 Mpa

OH flux = B x mean area of xolenoid

flux = 2.03 volt seconds (weber)

Added space between OH and TF

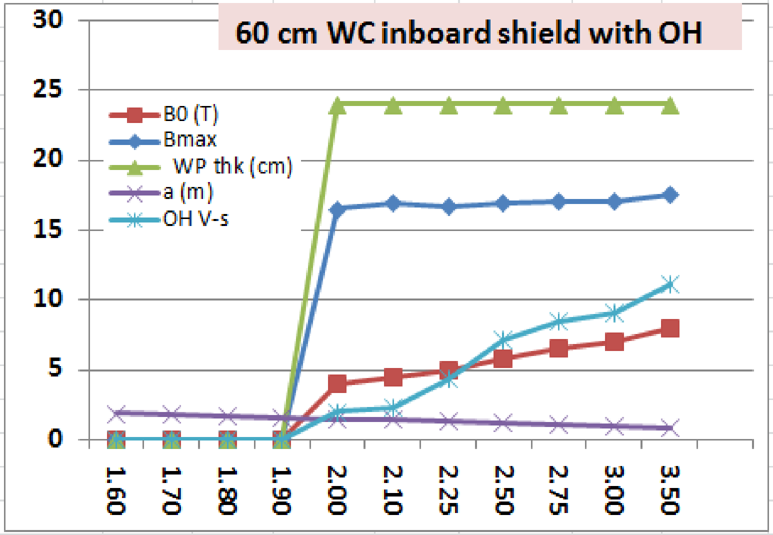
60 cm WC uses VV shield and shield in front of it.

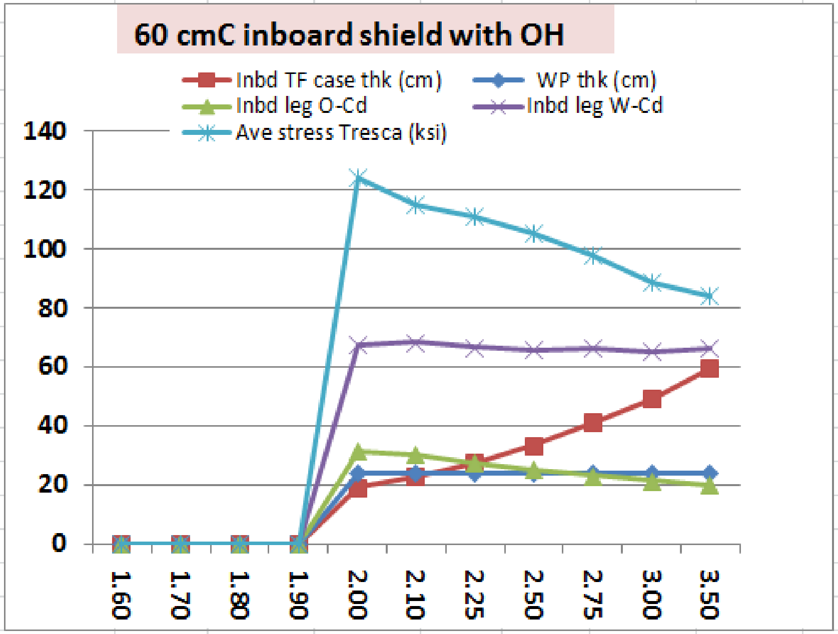
For AR 2 device:

OH located inside of TF bore: *854 Mpa Tresca stress with 4.0 B0 and 176.5 TF Bmax*

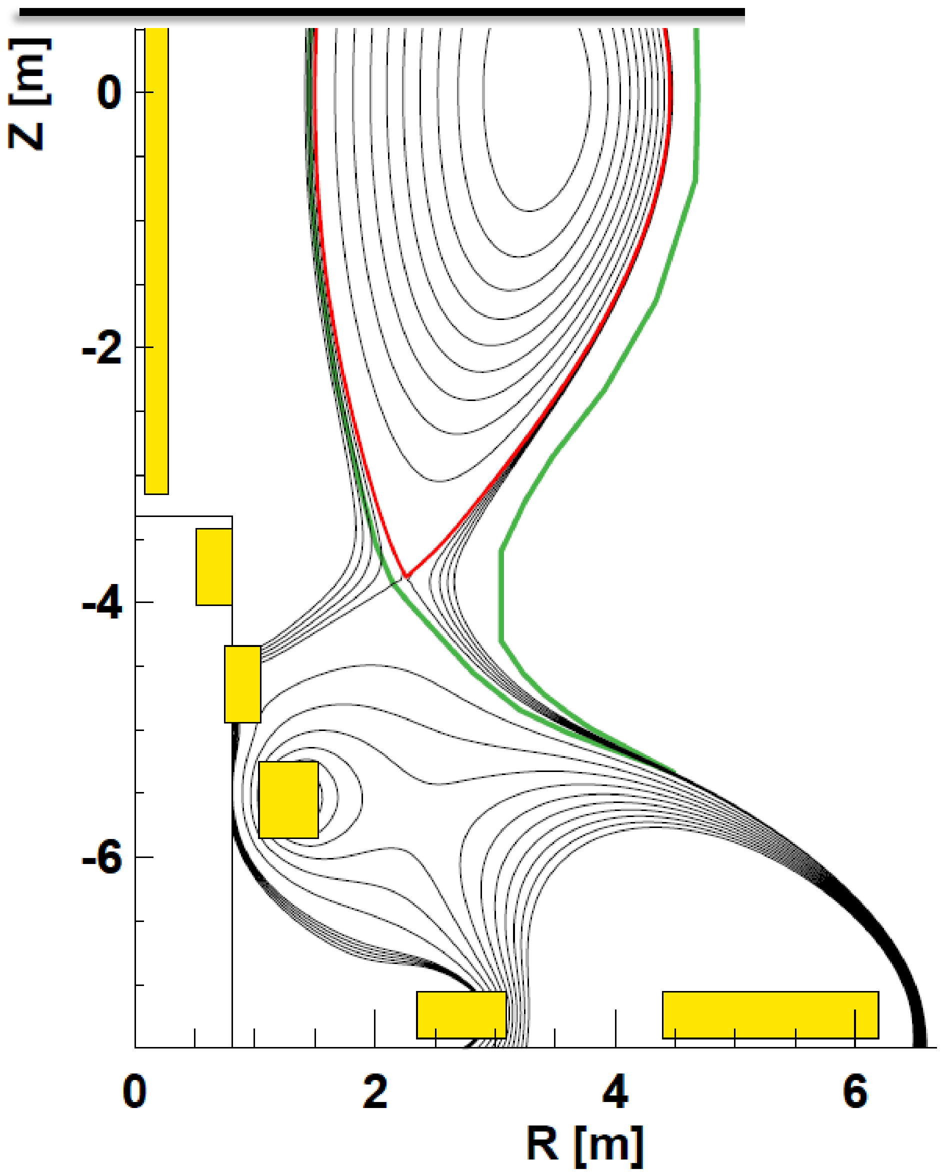
OH flux: 2 Vs with 70 Jc solenoid and 19.8 T

TF and OH magnet parameters vs. aspect ratio using models from Brown and Zhai





WC thk (cm)	R0 (m)	AR	Ɛ= 1/AR	a (m)	B0 (T)	WP thk (cm)	Inbd TF case thk (cm)	Inbd leg O-Cd (MA/m^2)	Inbd leg W-Cd (MA/m^2	Bmax	Ave stress Tresca (Mpa)	Ave stress Tresca (ksi)	OH IR (cm)	OH thk (cm)	OH B (T)	OH V-s
60.00	3.0	1.60	0.63	1.875												
		1.70	0.59	1.765												
		1.80	0.56	1.667												
		1.90	0.53	1.579												
		2.00	0.50	1.500	4.00	24	19.0	31.5	67.6	16.50	854	124	6.8	22.5	19.8	2.0
		2.10	0.48	1.429	4.50	24	22.5	29.9	68.2	16.90	791	115	7.9	22.5	19.8	2.3
		2.25	0.44	1.333	5.00	24	27.5	27.3	66.5	16.70	764	111	15.0	22.5	19.8	4.3
		2.50	0.40	1.200	5.80	24	33.3	24.8	65.9	16.90	725	105	22.5	22.5	19.8	7.1
		2.75	0.36	1.091	6.50	24	41.0	23	66.0	17.10	672	97	25.7	22.5	19.8	8.5
		3.00	0.33	1.000	7.00	24	48.8	21.3	65.2	17.10	609	88	27.0	22.5	19.8	9.1
		3.50	0.29	0.857	8.00	24	59.1	19.8	66.1	17.50	580	84	31.0	22.5	19.8	11.1

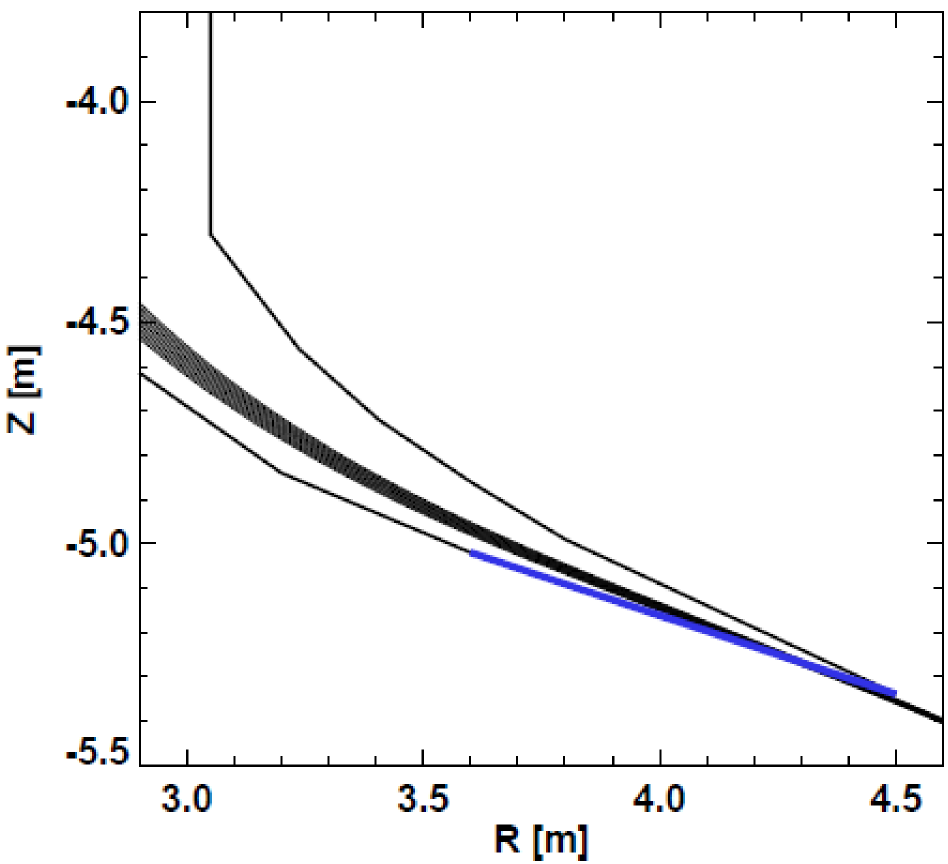


Assessing long-leg / deep-V divertor

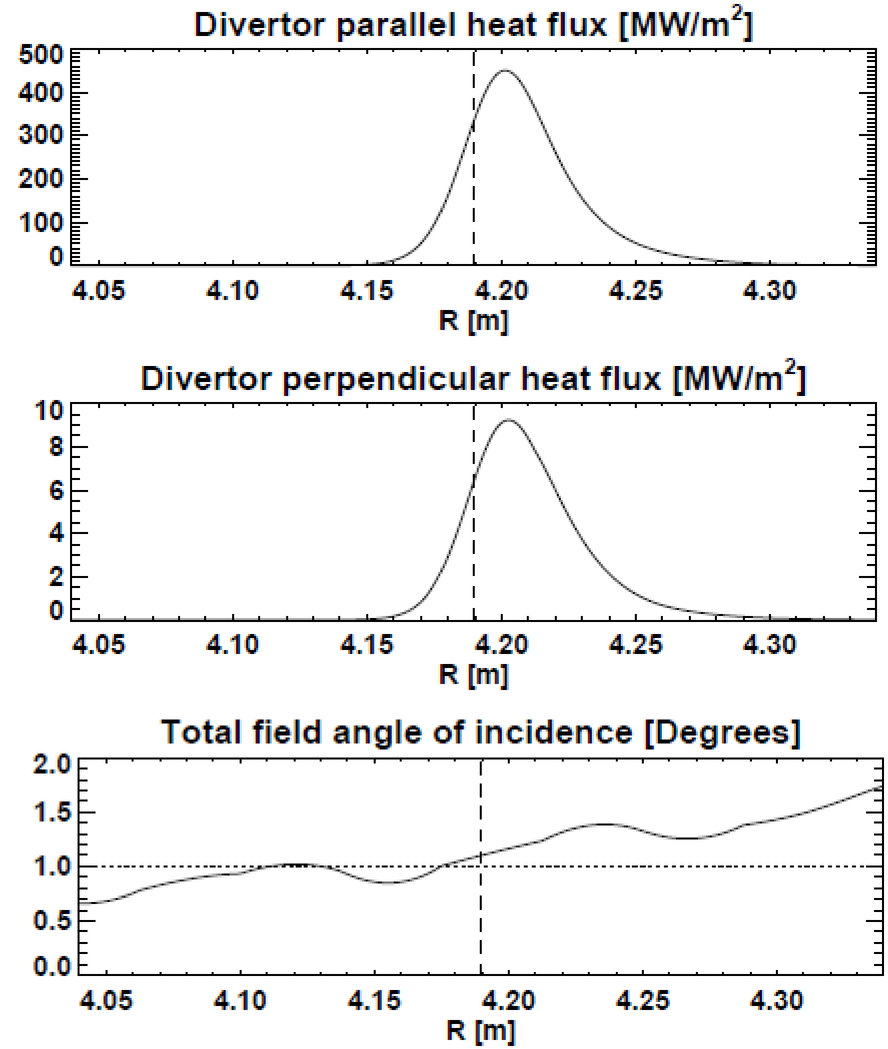
- PF coils outside TF
- Increase strike-point radius ~2× to reduce $q_{||}$ and peak heat flux
- Divertor PFCs in region of reduced neutron flux
- Narrow divertor aperture for increased TBR
- More space for breeding at top/bottom of device

Long-leg divertor aids heat flux reduction

A=2 HTS TF FNSF/Pilot

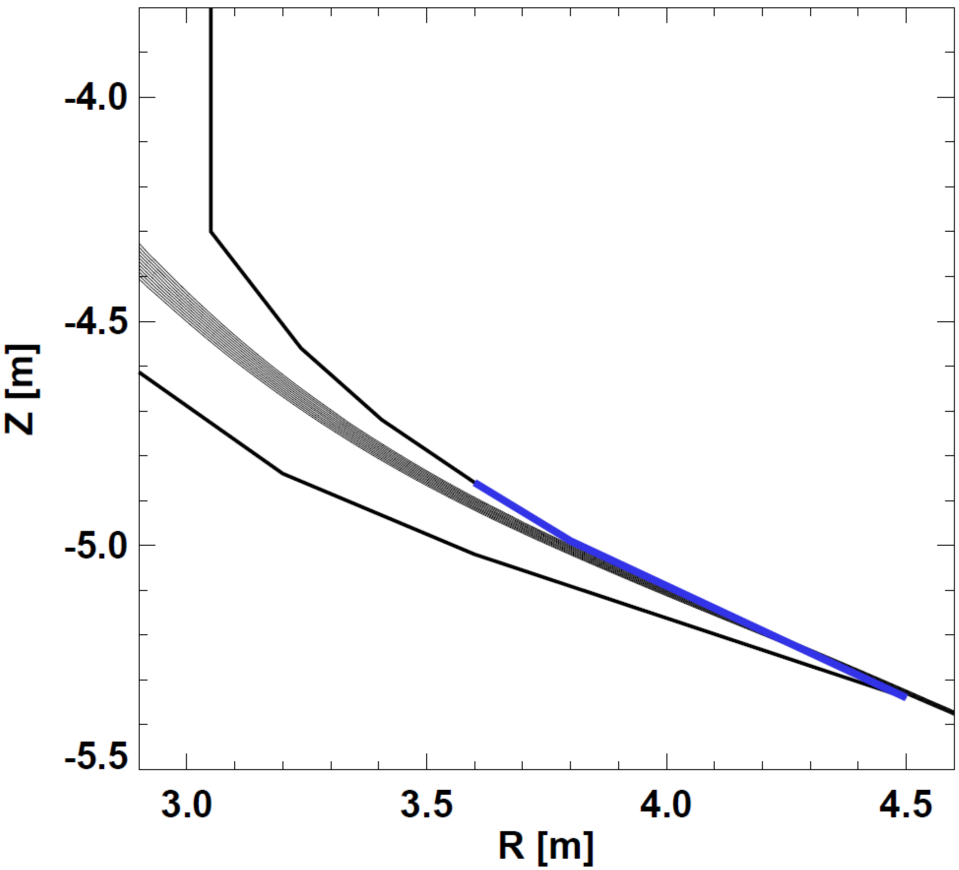


λ_q ~ 1mm, assume $S \approx \lambda_q$ (closed divertor)
(T. Eich NF 2013)

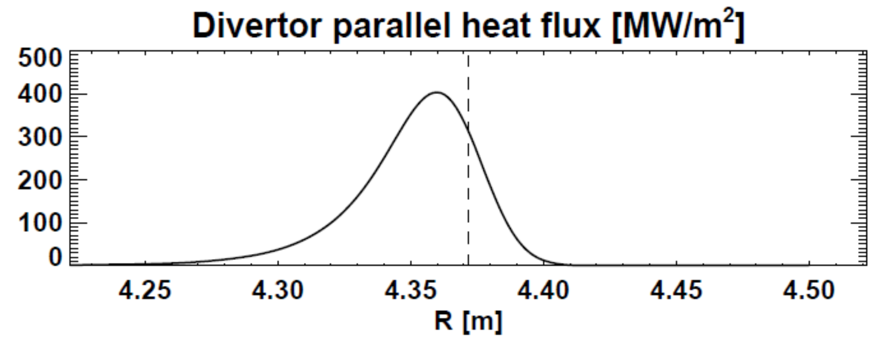


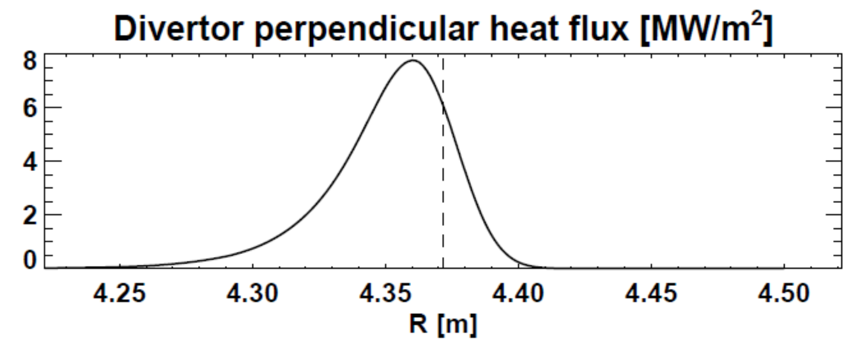
(Partial) detachment likely reduces peak $q_{\perp}$ by further factor of 2-4

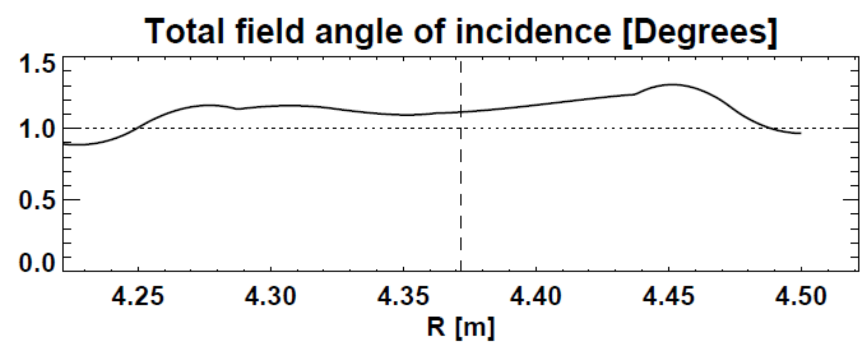
Can also exhaust onto back of OB blanket (like vertical target in conventional divertor)



$\lambda_q \sim$ 1mm, assume $S \approx \lambda_q$ (closed divertor)
(T. Eich NF 2013)

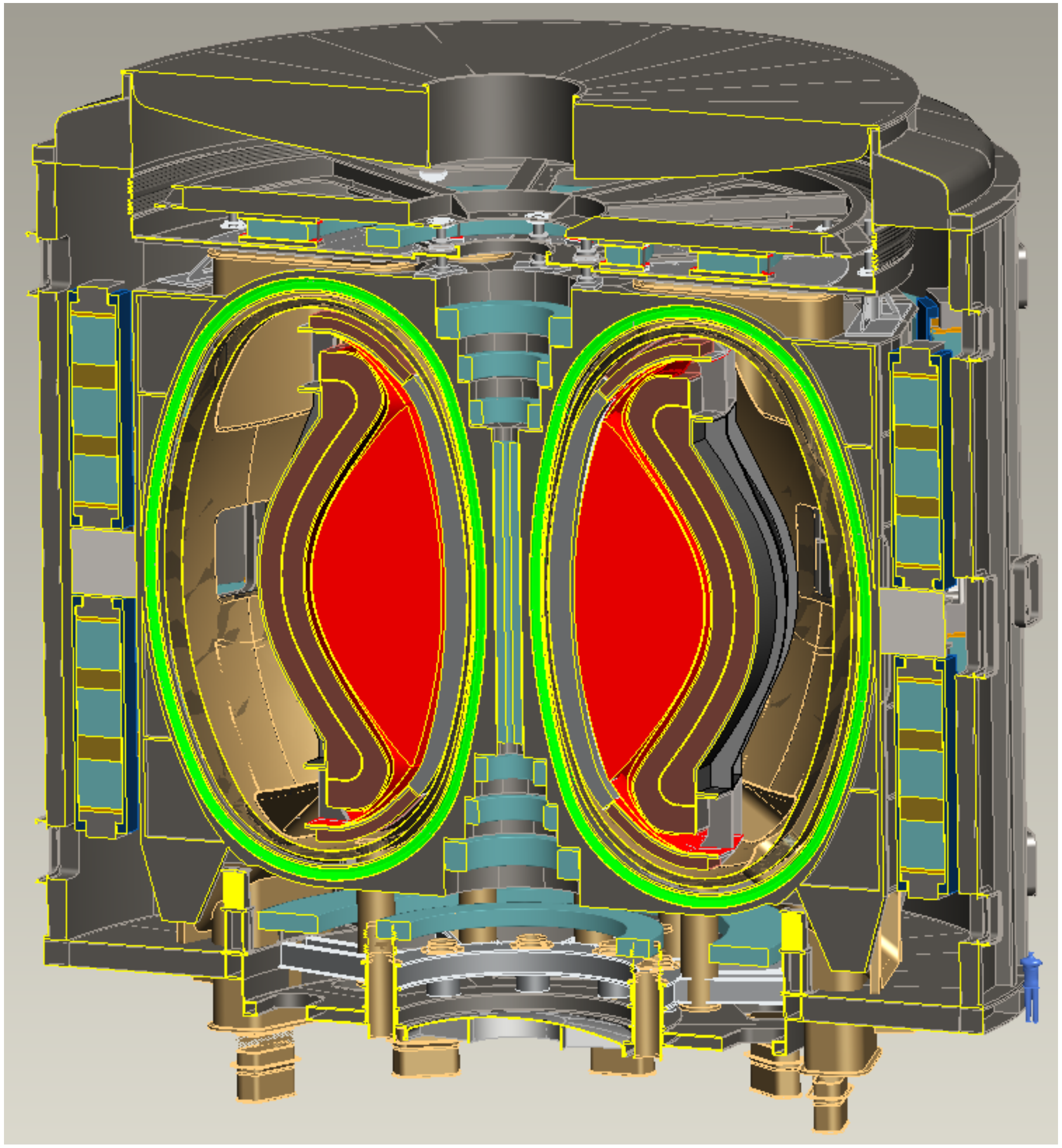






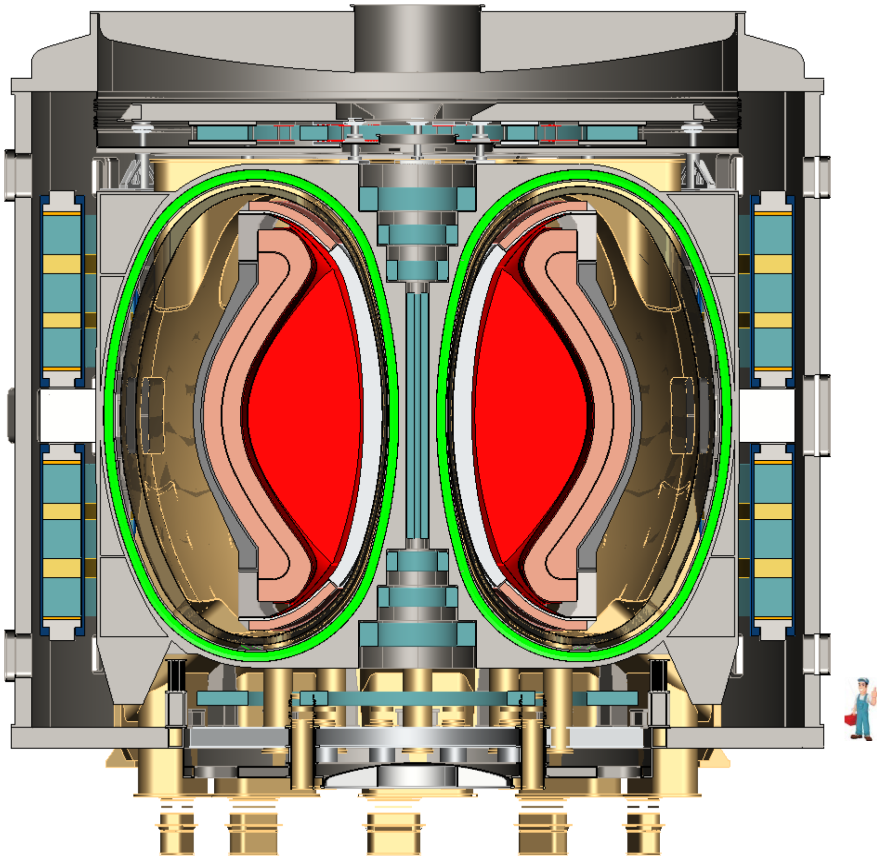
(Partial) detachment likely reduces peak $q_{\perp}$ by further factor of 2-4

Latest HTS-ST: R=3m, A=2, P_{fusion}~ 500MW, Q_{eng}~1-1.5



R=3 m Configuration and Key Parameters

Major Radius	**3**	m
Minor Radius	**1.5**	m
Fusion Power	**556**	MW



10 Sectors

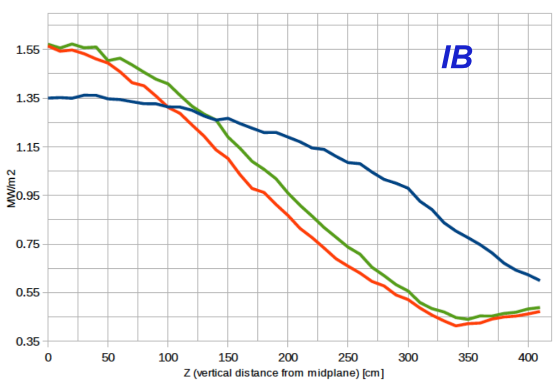
Plant Lifetime	**~10.3**	years
Availability	**~30%**	

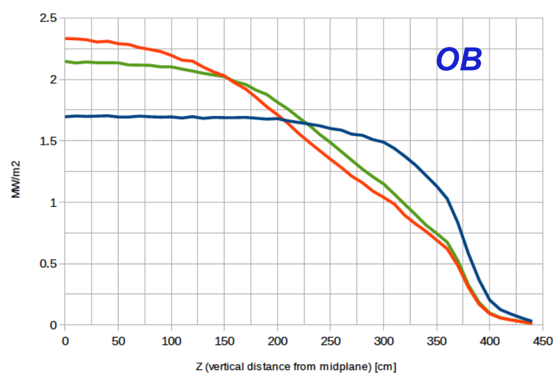
3.1 Full Power Years (FPY)

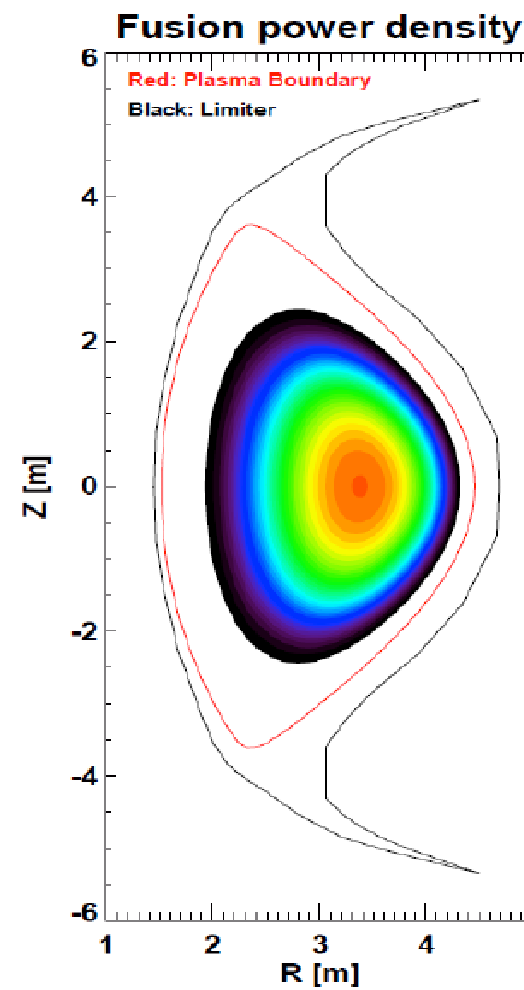
Neutron Wall Loading Distribution

Evaluated with 3 methods:

Average NWL	1.13 MW/m^2
Peak IB NWL	1.57 MW/m^2
Peak OB NWL	2.15 MW/m^2
Peak Div NWL	0.42 MW/m^2







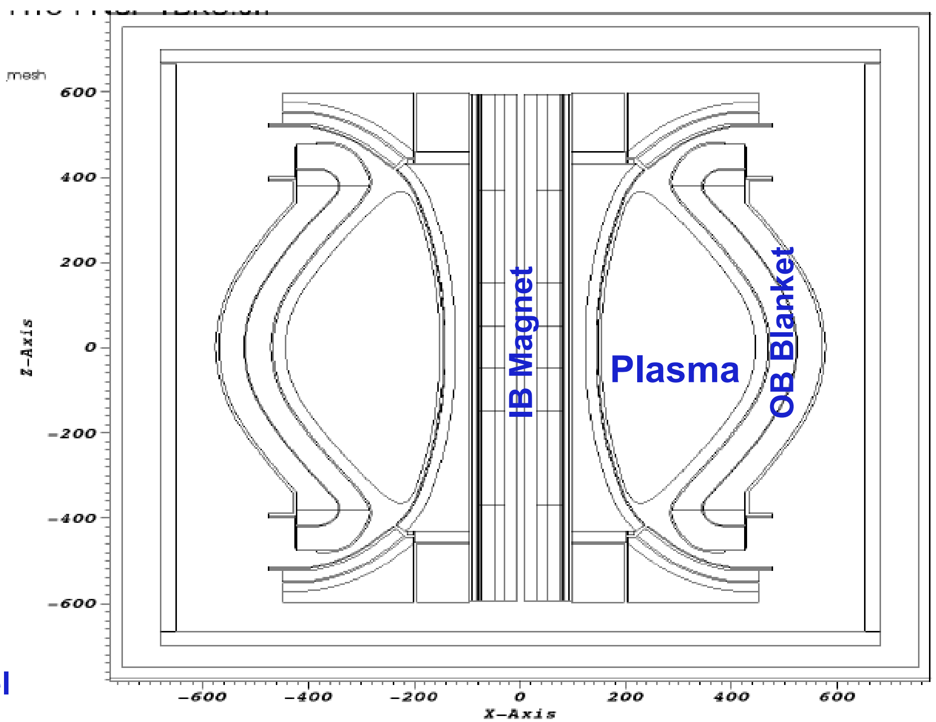
uniform source

3-region source

R-Z source

Nuclear Analysis Performed with Sophisticated 3-D Neutronics Codes

- CAD coupled with MCNP using UW DAGMC code.
- Fully accurate presentation of entire torus.
- **Neutron source model on R-Z grid, presenting fusion power density.**
- No approximation or simplification involved in 3-D model.
- **Evaluated:**
  - Neutron wall loading distribution
  - **Radiation damage at IB magnet**
  - Tritium breeding ratio (ongoing).



3-D Model

Radiation Limits

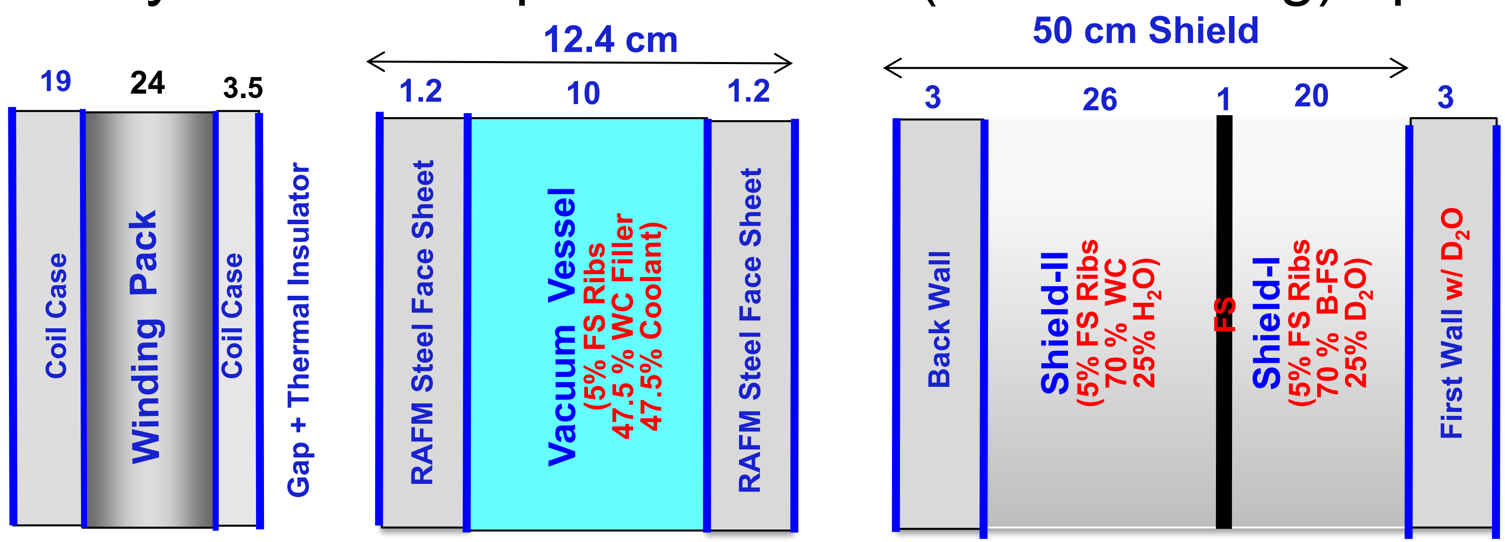
Overall TBR (for T self-sufficiency)	~ 1	
Damage to RAFM steel structure	20	dpa – GEN-I
	< 50	dpa – GEN-II
	> 65	dpa – ODS (NS)
Helium Production (for reweldability of FS)	1 ?	He appm

HTS Magnet (@ 20-40 K):

Peak fast n fluence to HT superconductor (E_n > 0.1 MeV)	**5×10^{18}**	**n/cm^2**
Peak nuclear **heating @ WP**	5	mW/cm^3
Peak nuclear heating @ coil case	**?**	**mW/cm^3**
Peak **dose** to electrical insulator	$5\text{-}10 \times 10^{10}$	rads
Total nuclear heating in 10 TF coils	**?**	**kW**

A=2 HTS ST Shielding Assessment

- **Focus on inboard (IB) shield** - main functions are:
  - Protect IB magnet for machine lifetime (3.1 FPY)
  - Enhance OB breeding by reflecting neutrons to OB
  - Generate low decay heat to control temperature response during accident → avoid using WC filler near FW.
- Assessed impact of candidate IB materials
  - Ferritic steel, tungsten carbide, hydrides, water, borated water, and heavy water) on magnet shielding as well as reflecting neutrons to OB blanket to enhance TBR.
- Two-layer IB shield presents best (non-breeding) option:



HTS-ST Shielding Assessment (Cont.)

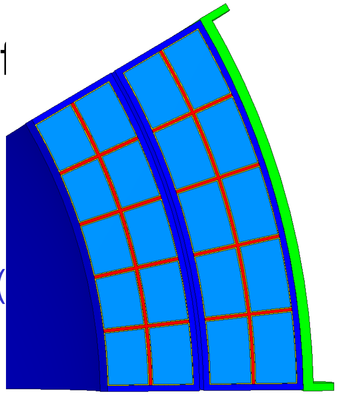
- Fast neutron fluence to HTS drives IB shield design.
- Combination of WC and H_2O represents superior shielding option as it helps reduce both fluence and magnet heating.
- Avoid:
  - Using B-H_2O **and** hydrides (having less shielding performance compared to WC/H_2O)
  - Straight radial assembly gaps.
- **3-D analysis confirmed radiation damage at IB magnet are below limits:**

Peak fast n fluence to HTS (E_n > 0.1 MeV)	**4.3×10^{18}**	**n/cm^2**
Peak nuclear heating @ WP	**1.7**	**mW/cm^3**
Peak dose to electrical insulator	**4×10^9**	**rads**
Total nuclear heating in IB magnet	**8.7**	**kW**

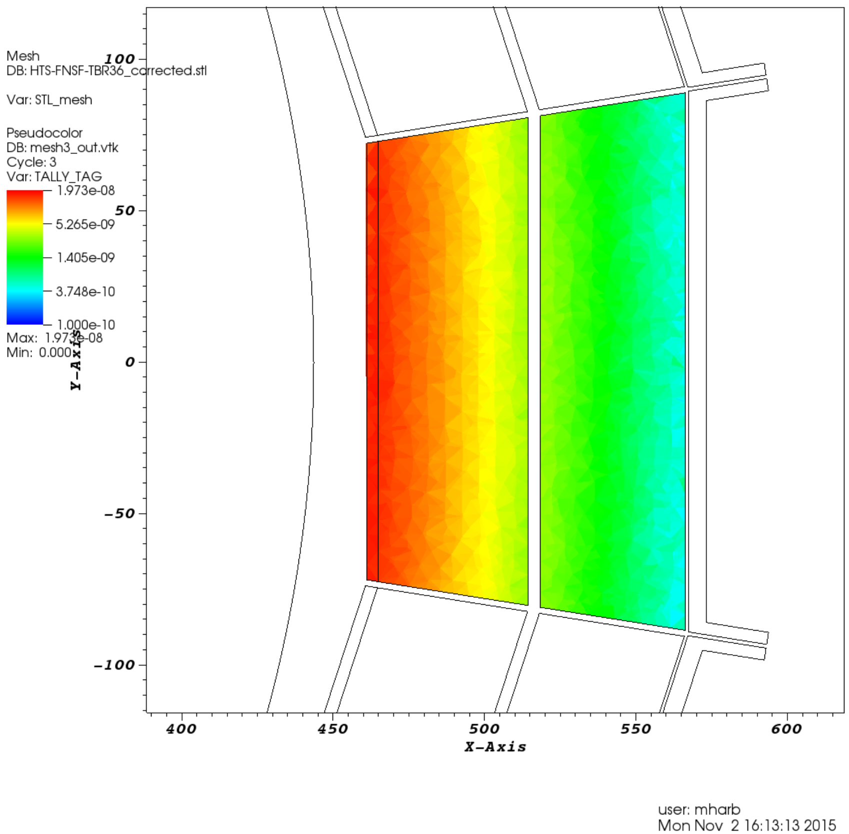
Blanket Design and Breeding Potential

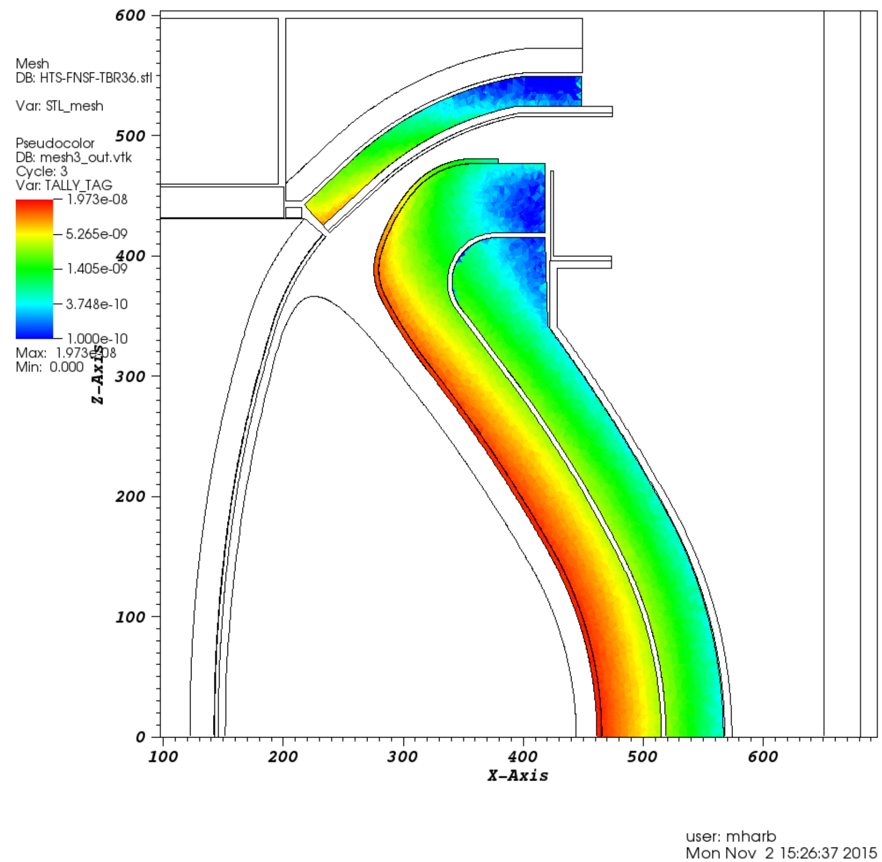
- Dual-cooled LiPb blanket (DCLL) – preferred US blanket concept for DEMO and power plants.
- 1 m thick OB blanket divided into two segments (to accommodate vertical stabilizing shells)
- He-cooled structural ring (SR) supports 20 OB blanket sectors.
- Several ports penetrate VV, SR, and blanket.
- During operation, 4 tritium breeding modules (TBM) and one Materials Testing Module (MTM) develop more advanced blanket/materials technologies for GEN-II, III, and IV DCLL blanket systems.
- To accurately estimate the overall TBR, 3-D model included details of internals and externals:
  - 2 cm wide assembly gaps between toroidal sectors
  - Internals of two OB DCLL blanket segments modeled in great details, including: FW, side, top/bottom, and back walls, cooling channels, SiC Flow Channel Inserts (
  - 2 cm thick W vertical stabilizing shell between OB blanket segments.
  - Ports (4 TBMs, 1 MTM, NNBIs).

3-D Blanket Model



Mapping of Tritium Production





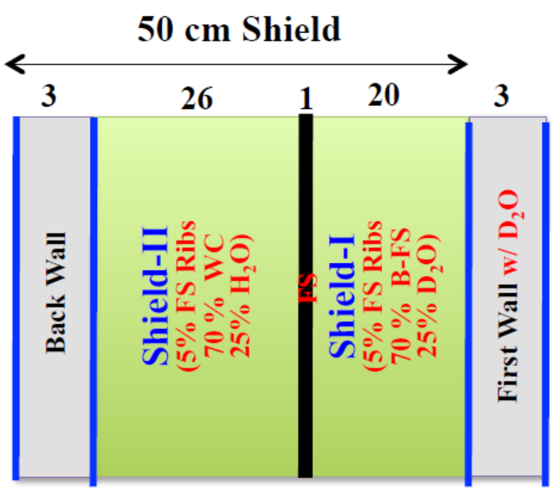
Horizontal Cross Section at OB Midplane

Vertical Cross Section Through Blanket regions

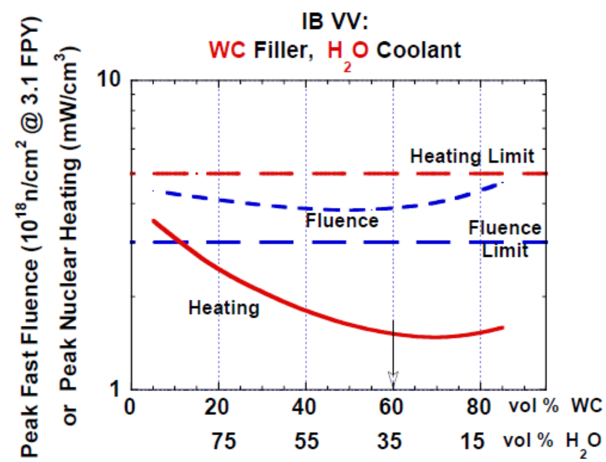
Inner segment of OB blanket provides highest breeding

Shield design and HTS radiation limits are critical issues for device size, lifetime

Two-Layer IB Shield and VV Optimization



Vacuum Vessel Optimization



- B-FS/D_2O Shield-I helps enhance OB breeding and control IB decay heat
- VV composition optimizes at 60% WC and 35% H_2O
- Magnet heating limit is well met
- Need 1-2 cm additional shield to meet fluence limit.