ST pilot plant studies

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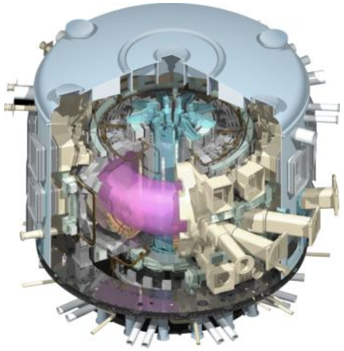
Exploring "Pilot Plant" as a possible pathway from ITER to commercial fusion power plant

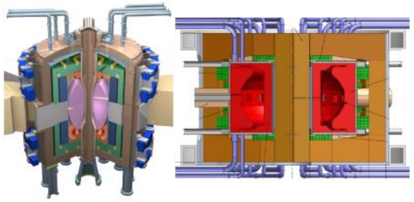
FNSF = Fusion Nuclear Science Facility
CTF = Component Test Facility

ITER



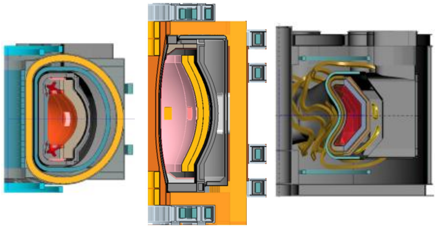
Supporting Physics and Technology

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



FNSF/CTF

- Blanket R&D, T self-sufficiency



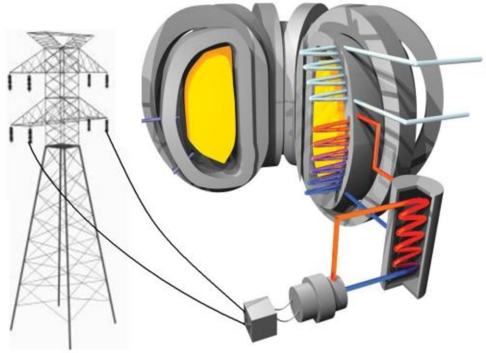
Pilot Plant

- Power-plant-like maintenance, $Q_{eng} = 1$

$Q_{eng} = 3\text{-}5$

- $P_{electric}$ near power plant levels

First of a kind Power Plant (= U.S. Demo)



Pilot plant goals, capabilities

- Pilot Plant goal:

  Integrate key science and technology capabilities of a fusion power plant in a next-step R&D facility.

- Targeted ultimate capabilities:
  - Fusion nuclear S&T development, component testing
    - Steady-state operating scenarios
    - Neutron wall loading ≥ 1MW/m^2
    - Tritium self-sufficiency
  - Maintenance scheme applicable to power plant
    - Demonstrate methods for fast replacement of in-vessel components
  - Net electricity production
    - Bridge gap between ITER/CTF and power plant (~1-1.5 GWe)

Q_{eng} ~ 1 requires improved technology and physics

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

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

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} = $P_{pump} + P_{sub} + P_{coils} + P_{control}$

Blanket and auxiliary heating and current-drive efficiency + fusion gain largely determine electrical efficiency Q_{eng}

Pumping, sub-systems power assumed to be proportional to $P_{thermal}$ – needs further research

0D (XL spreadsheet) model of operating points developed

(Similar to C. Neumeyer version developed for ST-CTF/NHTX, but simpler)

Special attention given to NBI-CD and fast-ion (NBI + alpha) pressure contribution

- **NBI CD efficiency estimated including all trapping and slowing-down effects**
  - D.F.H. Start et al., Plasma physics, Vol. 22, pp. 303 to 316

Normalized current drive efficiency

$$\boxed{\begin{aligned}\frac{I_T}{P} &= \frac{\tau_s v_0 K_1 Z_b}{2\pi R(1+\alpha^3)\varepsilon_0}(1 - Z_b/Z_{\mathrm{eff}} + 1.46\varepsilon^{1/2} A Z_b/Z_{\mathrm{eff}}) \\ &\times[1+(3-2\alpha^3\beta_1)(1+\gamma)\delta/(1+\alpha^3)^2] \\ &\times \int_0^1 x^{3+\beta_1}\left(\frac{1+\alpha^3}{x^3+\alpha^3}\right)^{1+\beta_1/3} \mathrm{d}x\end{aligned}}$$

$$\alpha^3 = 0.75\pi^{1/2} m_e v_e^3 \left(\sum n_i Z_i^2/n_e m_i\right) \Big/ v_0^3$$

$$\beta_n = m_i Z_{\mathrm{eff}} n(n+1)/2m_b,\ \delta = T_e/2\varepsilon_0,\ \rho = n_b/n_e$$

$$\tau_s = 3 m_e v_e^3 m_b/(16\pi^{1/2} e^4 Z_b^2 n_e \ln\Lambda)$$

NBI and alpha pressure derived from energy moment of slowing-down f(E)

- T.H. Stix, Plasma Physics, Vol. 14, pp. 367 to 384

Energy loss rate i-collisions e-collisions

$$\boxed{\left\langle \frac{\mathrm{d}W}{\mathrm{d}x} \right\rangle = -\frac{\alpha}{W} - \beta W^{1/2},}$$

$$\alpha \equiv 1{\cdot}30 \times 10^{-13} A Z^2 \ln\Lambda \sum_j \frac{n_j Z_j^{\,2}}{A_j}$$

$$\beta \equiv 2{\cdot}28 \times 10^{-15} \frac{Z^2}{A^{1/2}} \frac{n_e \ln\Lambda}{(kT_e)^{3/2}},$$

$$\left\langle \frac{\mathrm{d}W}{\mathrm{d}t} \right\rangle = \left\langle \frac{v\,\mathrm{d}W}{\mathrm{d}x} \right\rangle = 1{\cdot}39 \times 10^6 \left(\frac{W}{A}\right)^{1/2} \left\langle \frac{\mathrm{d}W}{\mathrm{d}x} \right\rangle$$

$$\tau = -\int_0^W \frac{\mathrm{d}W}{\mathrm{d}W/\mathrm{d}_t} = \frac{t_s}{3} \ln\left[1 + \left(\frac{W}{W_{\mathrm{crit}}}\right)^{3/2}\right]$$

Assumptions and constraints

- Surface-average neutron wall loading: $\langle W_n \rangle \geq 1$ MW/m^2
  - – ST neutron wall load peaking factor (peak/avg): 1.56

- Blanket thermal conversion:
  - – η_{th} = 0.3, 0.45 – this range incorporates leading concepts: He cooled pebble-bed (HCPB), dual-coolant lead-lithium (DCLL)
    - **M_n = 1.1, blanket coolant pumping power $P_{pump} = 0.03\times P_{th}$, $P_{sub} + P_{control} = 0.04\times P_{th}$**

- Steady-state operating scenarios:
  - – Fully non-inductive CD (BS+NBI)
    - **$\eta_{aux} = 0.4$, $\eta_{CD} = I_{CD}R_0n_e/P_{CD} = 0.3 \times 10^{20}$A/Wm^2**
  - – Superconducting (SC) PF coils

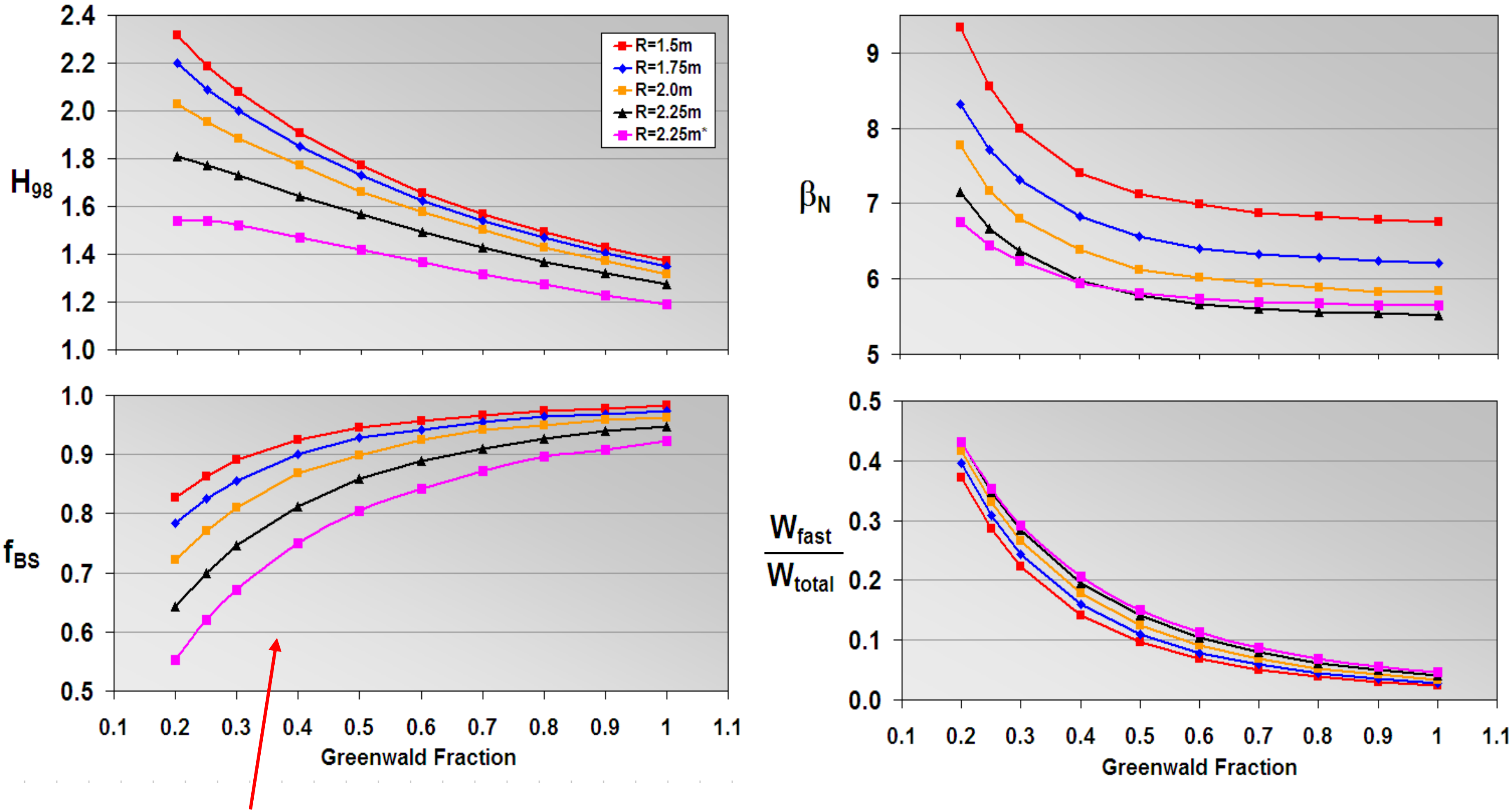
- Confinement and stability:
  - – $\tau_E \propto$ ITER H-mode IPB98(y,2), β near/above no-wall limit
    - **$\beta_N \leq$ present experimental values, density at or below Greenwald limit**

ST pilot plant parameters and scans

Aspect ratio	1.7
Plasma elongation	3.1
Plasma triangularity	0.6
Toroidal field at R_0	2.4T
E_{NBI}	0.5MeV
Non-inductive fraction	100% (BS+NBI)

- Scan major radius and density (Greenwald fraction)
  - Typically choose P_{fusion}, P_{NBI} , Q_{DT} to be independent of n_e
  - Vary I_P and H_{98} to achieve Q_{ENG}=1, f_{NI}=1
- Offset cost of increased R_0 by reducing physics risk in Q_{DT}
  - $q^* > 2$ limits maximum I_P at low n_e
- Solutions become more conservative as R_0 is increased
  - Reduced Q_{DT}, β_N, f_{BS}, H_{98}, and neutron wall loading
- Thermal conversion η=0.45, 0.3, $\Delta_{IB\text{-}shield}$=15cm, SC PF coils

Increased n_e / n_G reduces H_{98}, β_N, I_P, fast ion fraction
Increased R_0 reduces H_{98}, β_N, bootstrap fraction

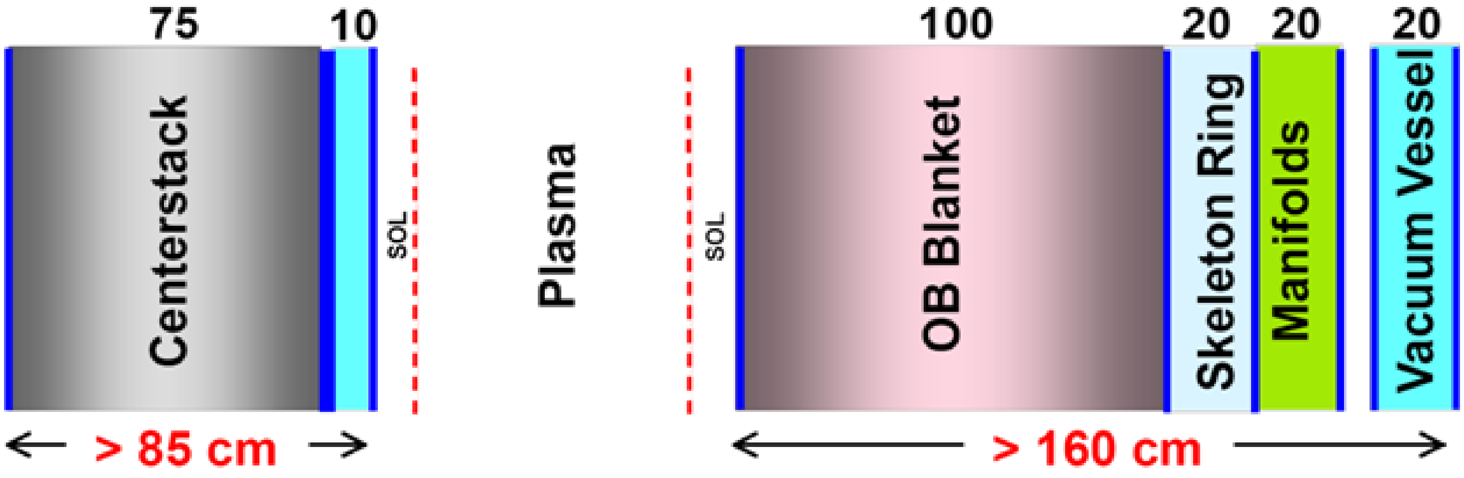


But one disadvantage of increased density is increase in required f_{BS}

NOTE: R=2.25m* case is same as R=2.25m case but with P_{NBI} = 40 → 60MW

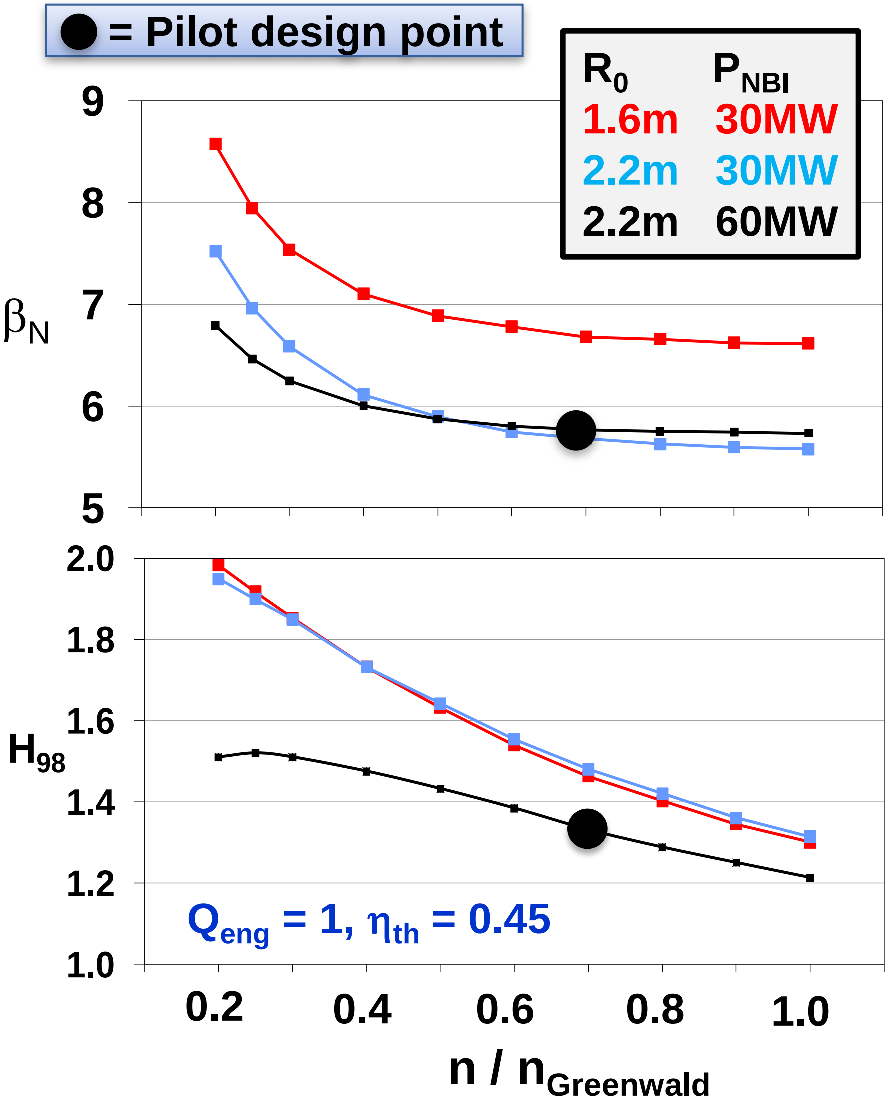
1D neutronics calculations used to develop preliminary ST pilot plant radial builds

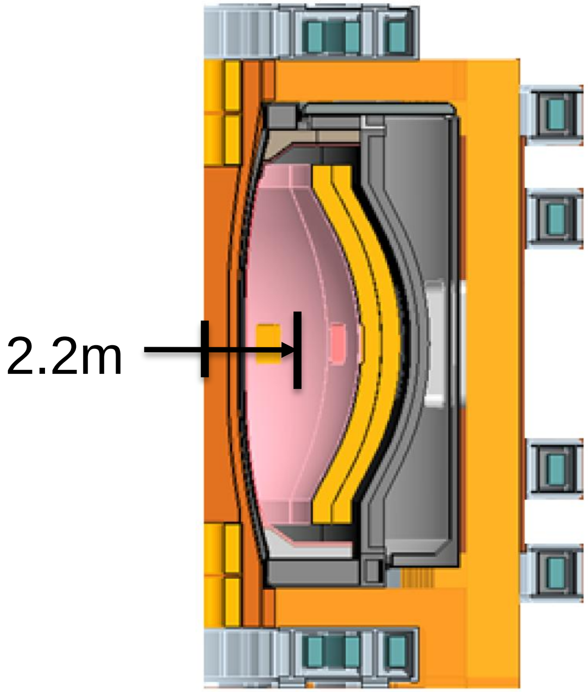
- 20 year plant lifetime, 6 full power years (FPY), 30% average availability,
- ST blanket replacement: 1.8/1.4 FPY inboard/outboard
- Skeleton-ring, vessel, SC coils are lifetime components, vessel re-weldable



- **Use DCLL blankets, TBR ~1.1 for 1.0 net** **(assuming full blanket coverage)**
- Damage to FS ≤ 80 dpa, Re-weldability: ≤ 1 He appm
- SC magnets operated at 4K
  - Peak fast neutron fluence to Nb_3Sn (E_n > 0.1 MeV) ≤ 10^{19} n/cm^2, Peak nuclear heating ≤ 2mW/cm^3,
  - Peak dpa to Cu stabilizer ≤ 6×10^{-3} dpa, Peak dose to electric insul. ≤ 10^{10} rads

Size of ST pilot depends primarily on achievable β_N



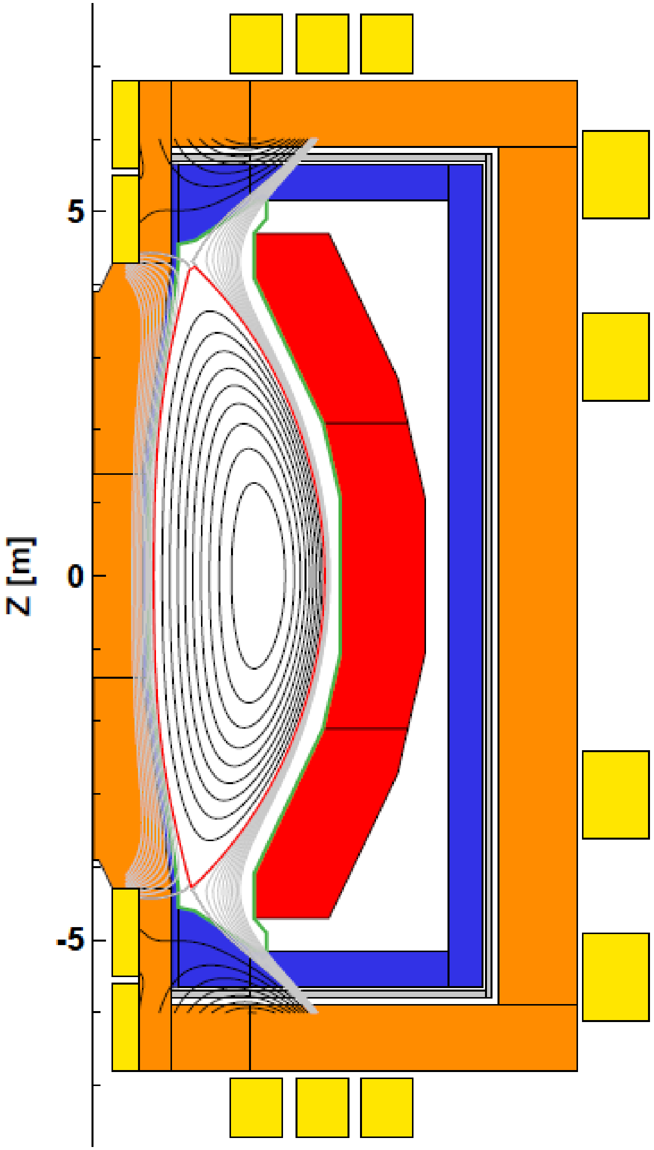


- A = 1.7 = 2.2m / 1.3m
- B_T = 2.4T, I_P = 18-20MA
- Avg. W_n = 1.9-2.9 MW/m^2
- Peak W_n = 3-4.5 MW/m^2

Higher density favorable for reducing β_N and H_{98} (also fast ion fraction)

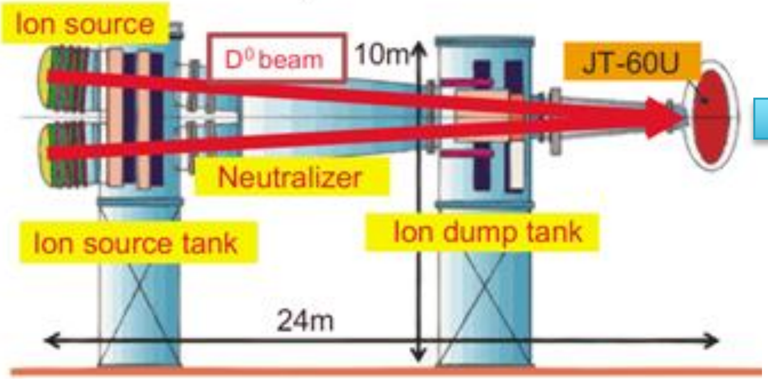
Some ST pilot design features

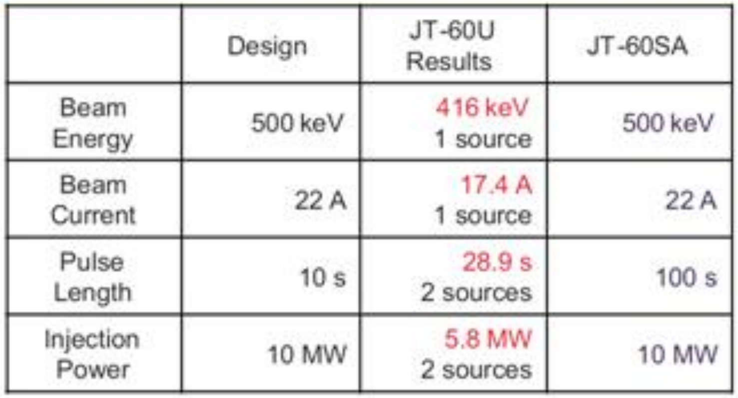
- Flared TF rod to reduce power: 150-200MW
- Strong shaping for stability, bootstrap current
  - Elongation ~3 and triangularity ~ 0.6
- DN divertor for power handling
  - Avg. heat flux over wetted area = 7MW/m^2
  - Peak heat flux could be much higher
  - May need snowflake, flowing Li, Super-X, radiation…
- PF coils in ends of TF rod to produce diverted high δ plasma, protect PF coils
  - All other PF coils superconducting
- Vacuum vessel independent of TF legs
  - 10 TF outer legs, ripple < 0.25% at plasma
- Conformal blankets to maximize TBR
  - Entire blanket structure removable vertically
- Shielding for vessel, TF outer legs, PF coils outside center-stack → lifetime components
- Center-stack shielded for 1-2 FPY



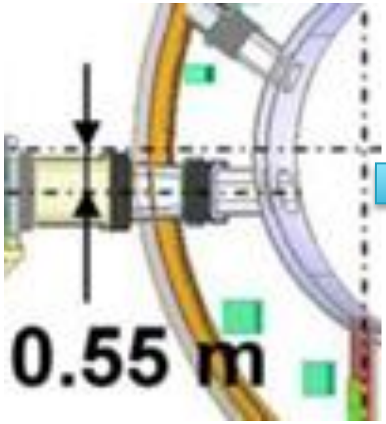
JT60-SA NNBI well matched to ST Pilot

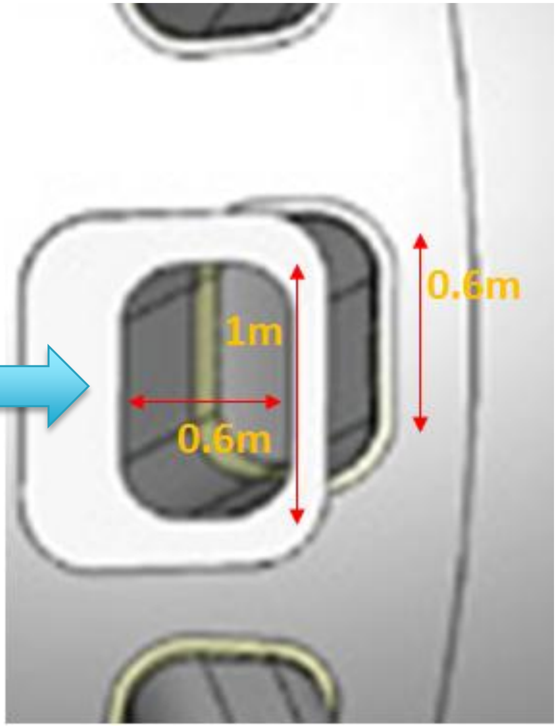
Kojima *et al.*





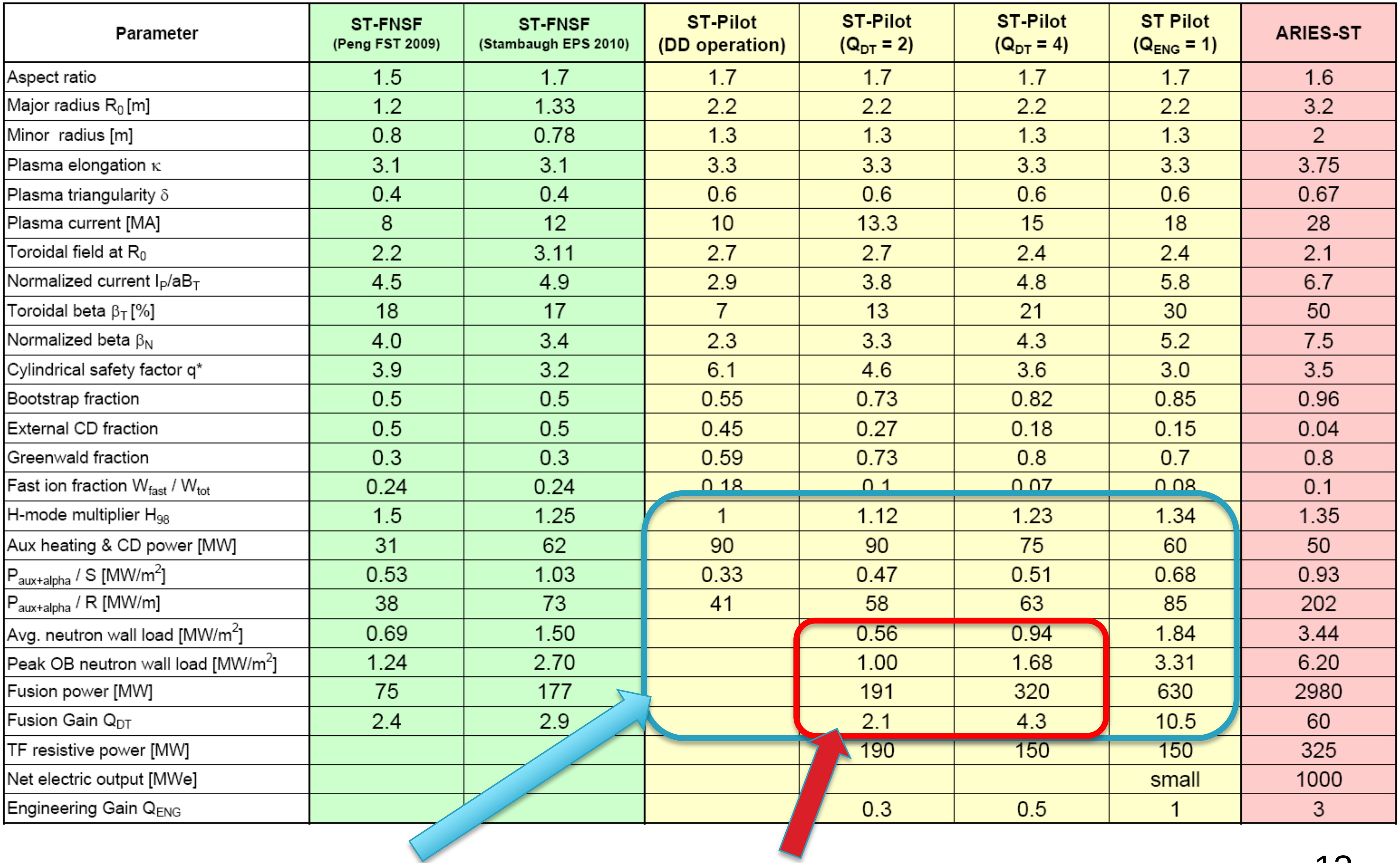
	Design	JT-60U Results	JT-60SA
Beam Energy	500 keV	416 keV 1 source	500 keV
Beam Current	22 A	17.4 A 1 source	22 A
Pulse Length	10 s	28.9 s 2 sources	100 s
Injection Power	10 MW	5.8 MW 2 sources	10 MW





- Estimated aperture at front of blanket would be ~0.4m^2
- 60MW → 6 ports → 2.4m^2 → 1.5% of ST pilot blanket area

ST pilot performance intermediate between ST-FNSF and ARIES-ST in size, β, fusion performance



Parameter	ST-FNSF (Peng FST 2009)	ST-FNSF (Stambaugh EPS 2010)	ST-Pilot (DD operation)	ST-Pilot (Q_{DT} = 2)	ST-Pilot (Q_{DT} = 4)	ST Pilot (Q_{ENG} = 1)	ARIES-ST
Aspect ratio	1.5	1.7	1.7	1.7	1.7	1.7	1.6
Major radius R_0 [m]	1.2	1.33	2.2	2.2	2.2	2.2	3.2
Minor radius [m]	0.8	0.78	1.3	1.3	1.3	1.3	2
Plasma elongation κ	3.1	3.1	3.3	3.3	3.3	3.3	3.75
Plasma triangularity δ	0.4	0.4	0.6	0.6	0.6	0.6	0.67
Plasma current [MA]	8	12	10	13.3	15	18	28
Toroidal field at R_0	2.2	3.11	2.7	2.7	2.4	2.4	2.1
Normalized current I_P/aB_T	4.5	4.9	2.9	3.8	4.8	5.8	6.7
Toroidal beta β_T [%]	18	17	7	13	21	30	50
Normalized beta β_N	4.0	3.4	2.3	3.3	4.3	5.2	7.5
Cylindrical safety factor q*	3.9	3.2	6.1	4.6	3.6	3.0	3.5
Bootstrap fraction	0.5	0.5	0.55	0.73	0.82	0.85	0.96
External CD fraction	0.5	0.5	0.45	0.27	0.18	0.15	0.04
Greenwald fraction	0.3	0.3	0.59	0.73	0.8	0.7	0.8
Fast ion fraction W_{fast} / W_{tot}	0.24	0.24	0.18	0.1	0.07	0.08	0.1
H-mode multiplier H_{98}	1.5	1.25	1	1.12	1.23	1.34	1.35
Aux heating & CD power [MW]	31	62	90	90	75	60	50
$P_{aux+alpha}$ / S [MW/m^2]	0.53	1.03	0.33	0.47	0.51	0.68	0.93
$P_{aux+alpha}$ / R [MW/m]	38	73	41	58	63	85	202
Avg. neutron wall load [MW/m^2]	0.69	1.50		0.56	0.94	1.84	3.44
Peak OB neutron wall load [MW/m^2]	1.24	2.70		1.00	1.68	3.31	6.20
Fusion power [MW]	75	177		191	320	630	2980
Fusion Gain Q_{DT}	2.4	2.9		2.1	4.3	10.5	60
TF resistive power [MW]				190	150	150	325
Net electric output [MWe]						small	1000
Engineering Gain Q_{ENG}				0.3	0.5	1	3

Possible ST progression: DD, PMI validate, FNS, component test, Q_{ENG} → 1

Pilot Plant can perform blanket testing

- Blanket development requirements:

– Local $W_{neutron} \geq 1$ MW/m^2, test area ≥ 10 m^2, volume ≥ 5 m^3

– Three phases:

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

I. Fusion break-in ~ 0.3 MWy/m^2
II. Engineering feasibility ~ 1−3 MWy/m^2
III. Engineering development, reliability growth, ≥ 4-6 MWy/m^2 accumulated

- $Q_{eng} \geq 1$ → P_{fus}=0.3-1 GWth → 17-56kg of T per FPY

– World T supply (CANDU) peaks at ~25-30 kg by 2025-2030

– ITER + T decay projected to consume most of this amount

- All three pilots have sufficient testing area, volume

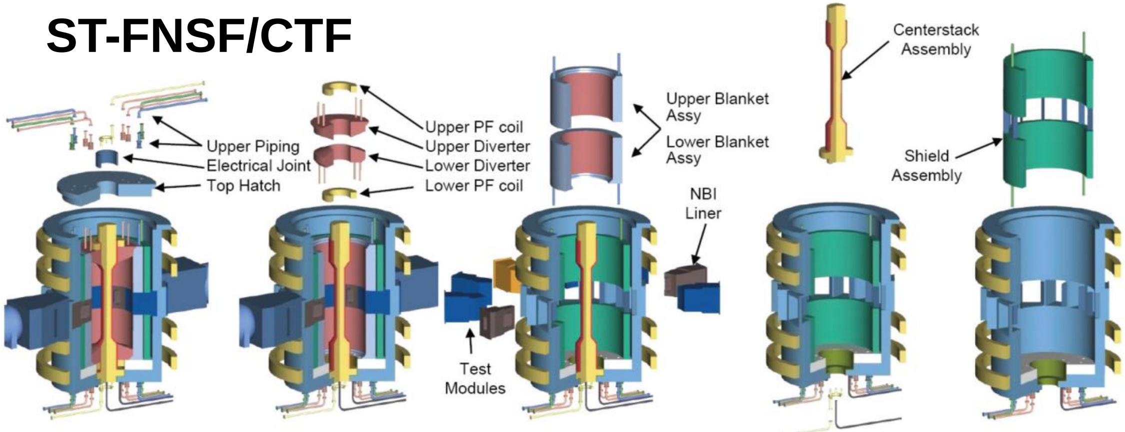
- To achieve Phase III 6MWy/m^2 (peak) → 45-72 kg T

➔ Need TBR ≈ 1 (Example: need TBR ≥ 0.9 for 5-7 kg available T)

ST/Cu magnets offer attractive vertical maintenance

Present ST Pilot design uses combination of ST-FCTF, FDF, ARIES-ST features

ST-FNSF/CTF



Disconnect upper piping
Remove sliding electrical joint
Remove top hatch

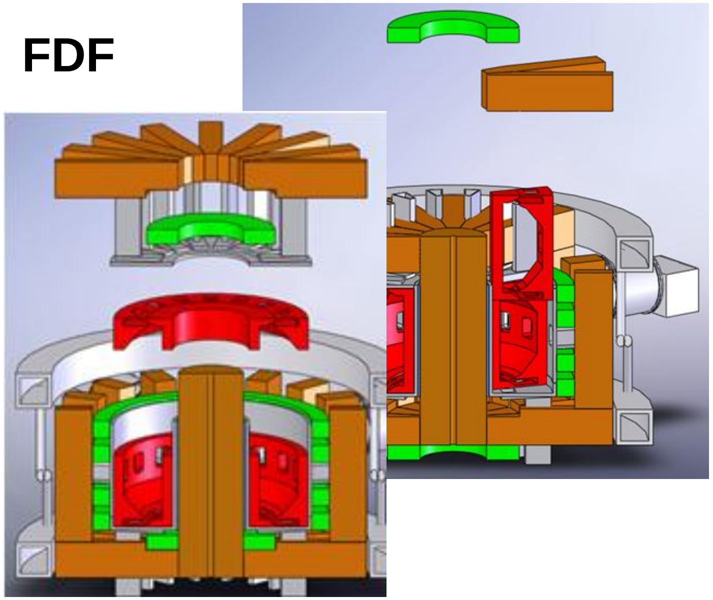
Remove upper PF coil
Remove upper diverter
Remove lower diverter
Remove lower PF coil

Extract NBI liner
Extract test modules
Remove upper blanket assembly
Remove lower blanket assembly

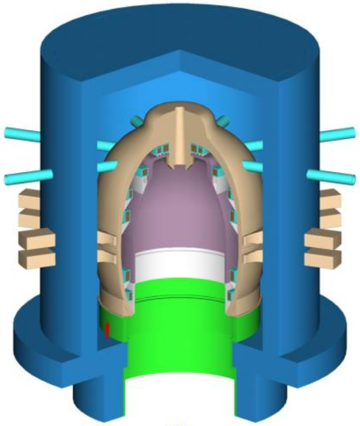
Remove centerstack assembly

Remove shield assembly

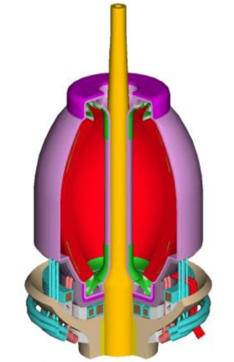
FDF



ARIES-ST

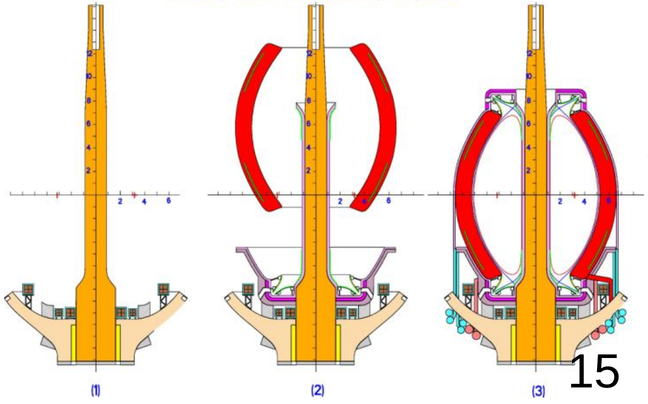


Permanent Components



Replaceable Components

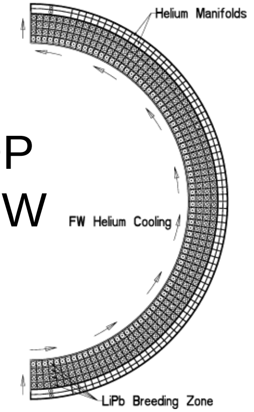
ARIES-ST Power Core Assembly Sequence



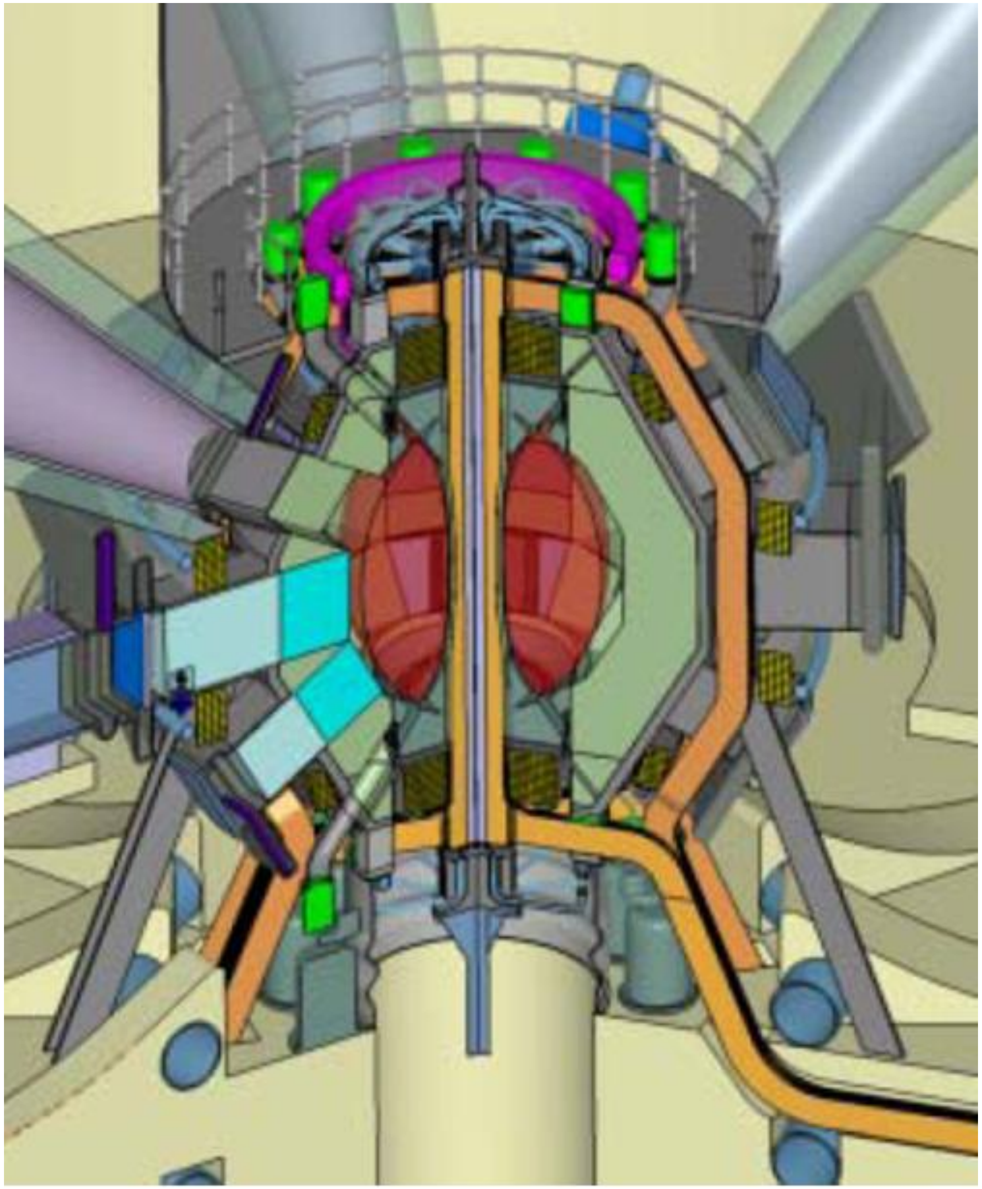
TOP VIEW

Cross Section of ARIES-ST Outboard Blanket

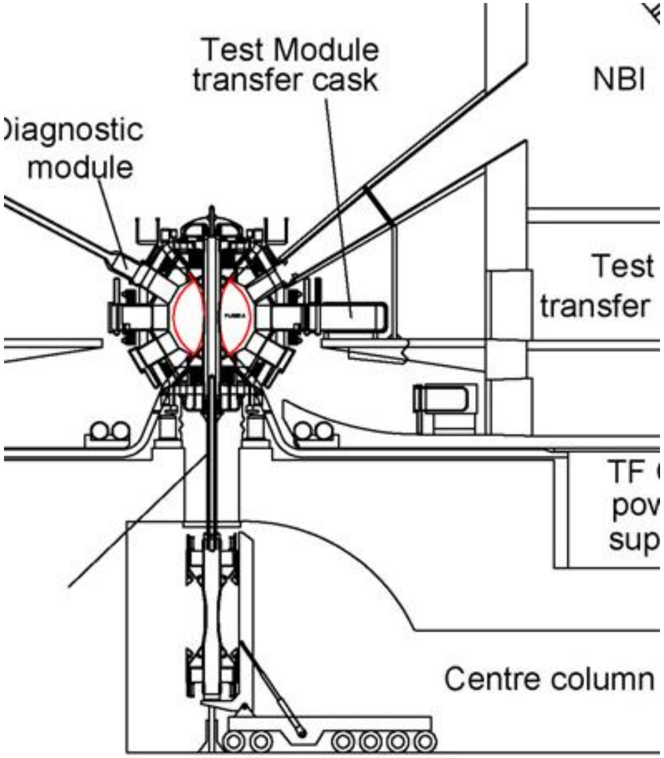
(One of Two Sectors)



Culham CTF maintenance features



- Test modules installed horizontally
- CS removed vertically
- NBI off-midplane

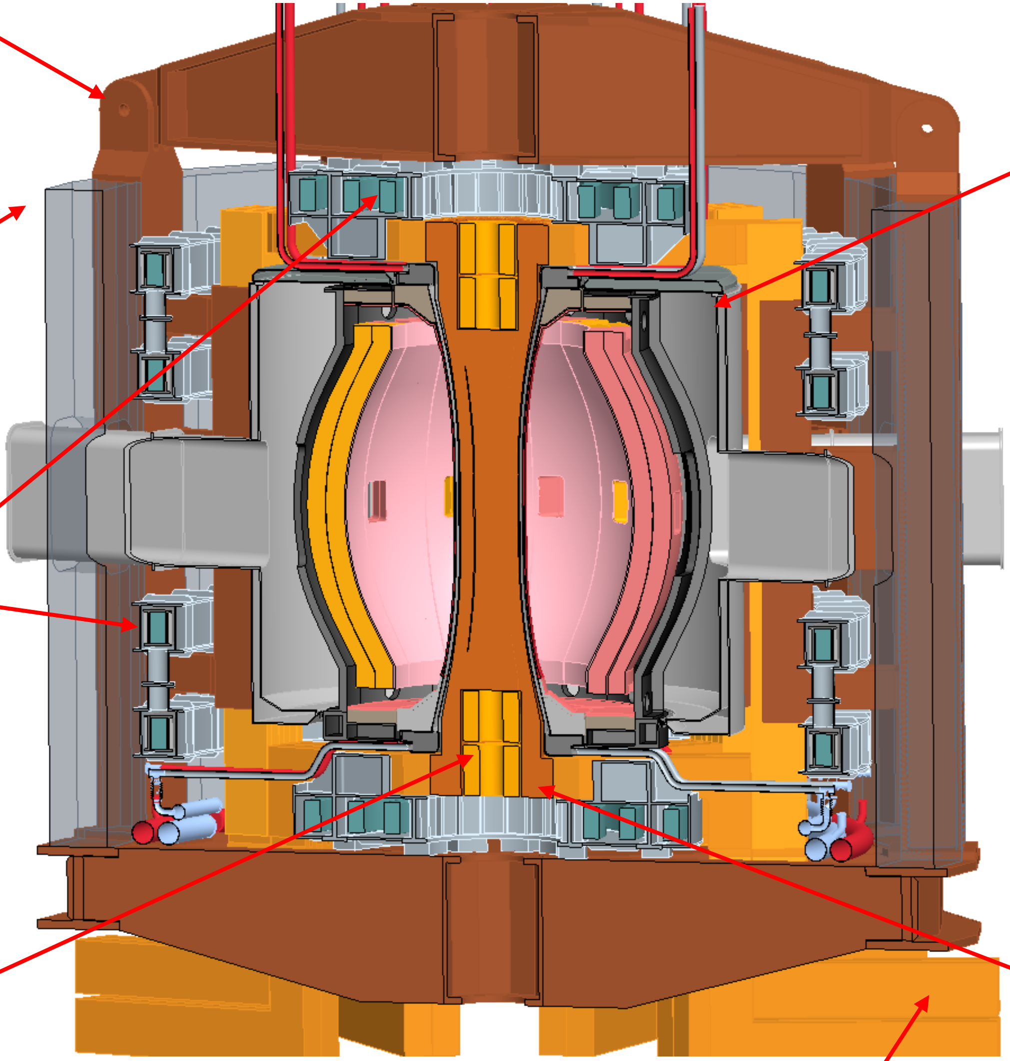


Would be beneficial for Culham/PPPL develop common understanding, vision, design of ST-based FNSF/CTF/Pilot

- What is acceptable risk for next step?
- What is optimum mission scope?
  - Limited to test modules with small total surface area?
  - Try for TBR = 1?
  - Aim for net electricity production?
- What are wall loading requirements, assumptions?
  - How does this drive assumed physics scenarios?
  - How does this impact ongoing research on NSTX, MAST?
- What are best design, maintenance approaches?
  - Sharing of engineering, design expertise most valuable
- Upgrade outages are excellent opportunity for joint physics and engineering work on CTF – thoughts?

Resources needed: at least ~1-2 FTE design, engineering + much more physics input

Engineering design features of ST Pilot



External structure supporting a sliding joint TF system

External structure also supports PF coils, torsional loads

S/C external PF coils housed in a vacuum enclosure

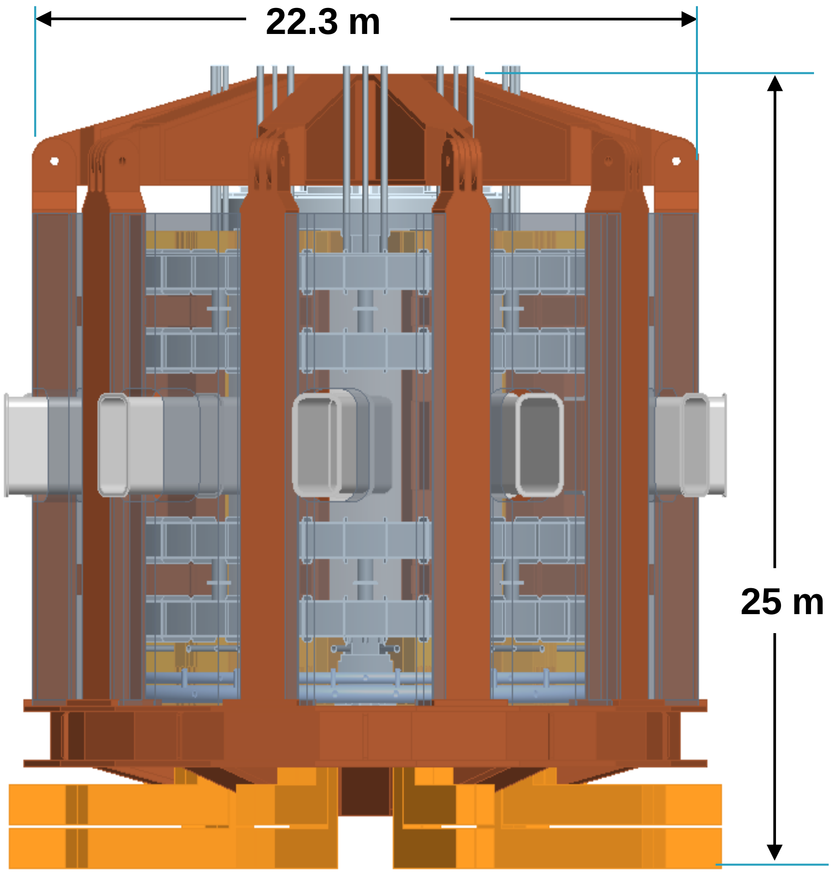
Copper divertor shaping coils embedded within the TF centerstack

VV located inside the TF

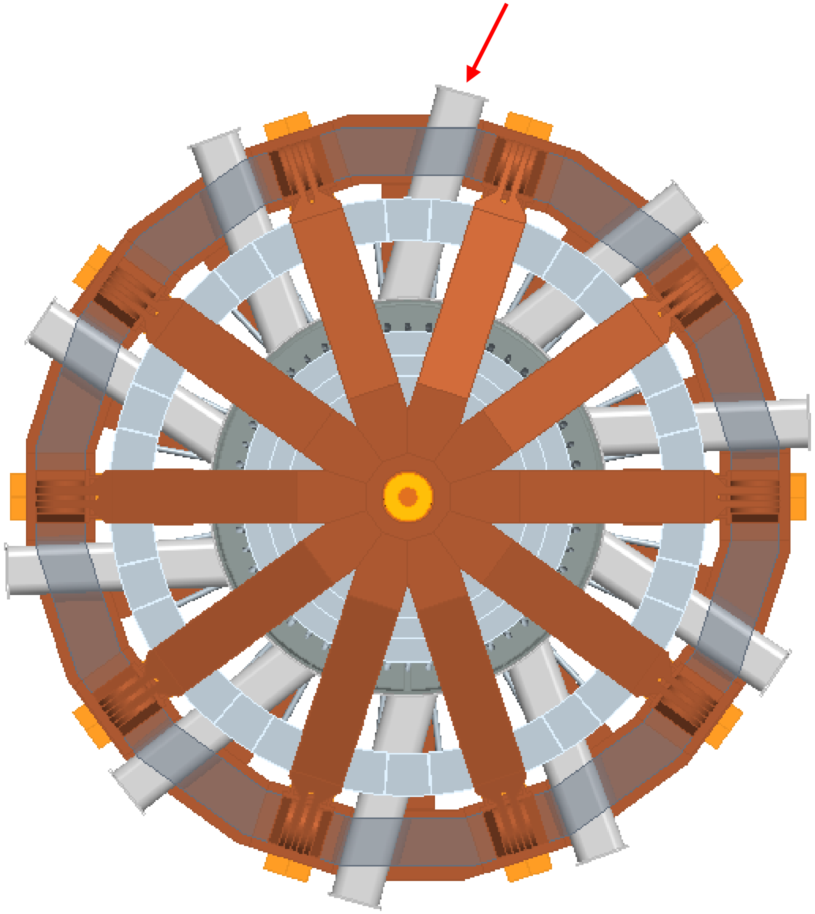
Felt metal sliding joint

Leads for TF power supplies

Side and top views, overall device size

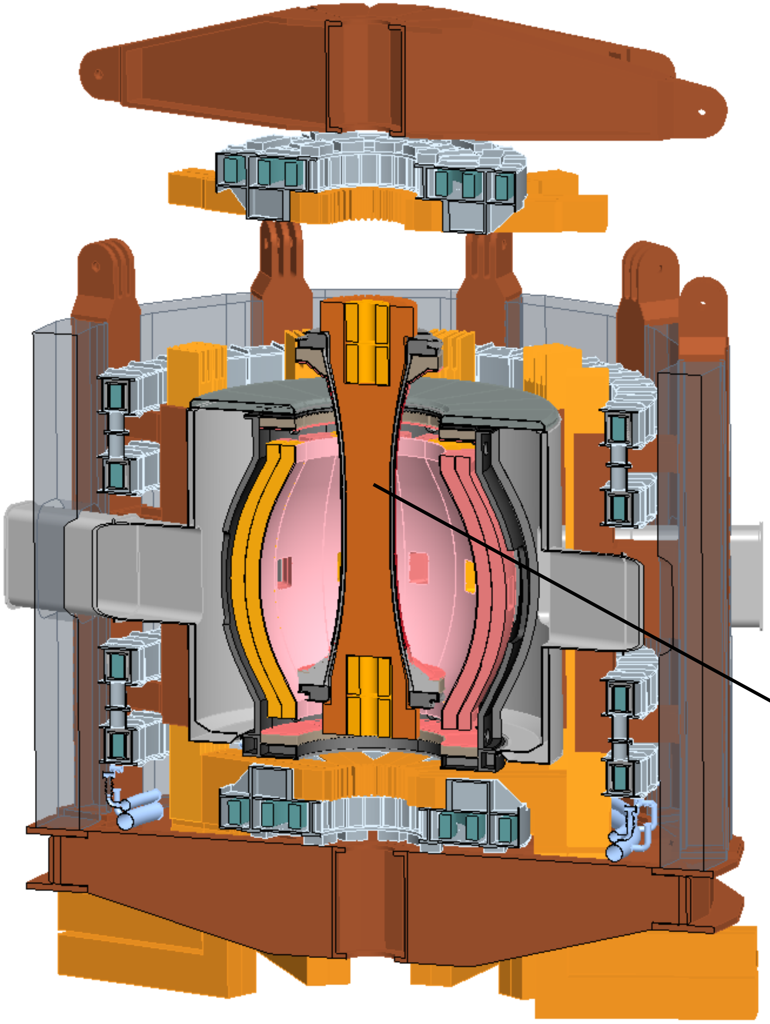


Front View

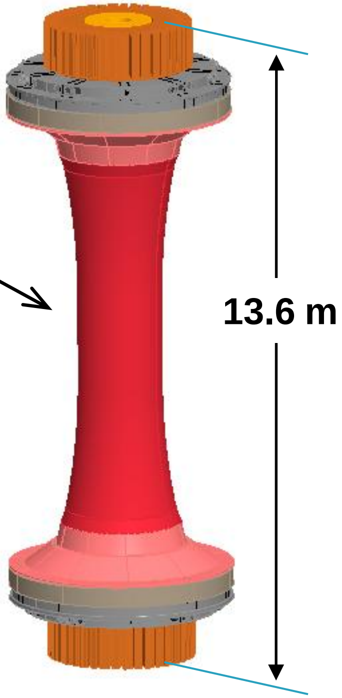

Tangential injection port for NBI

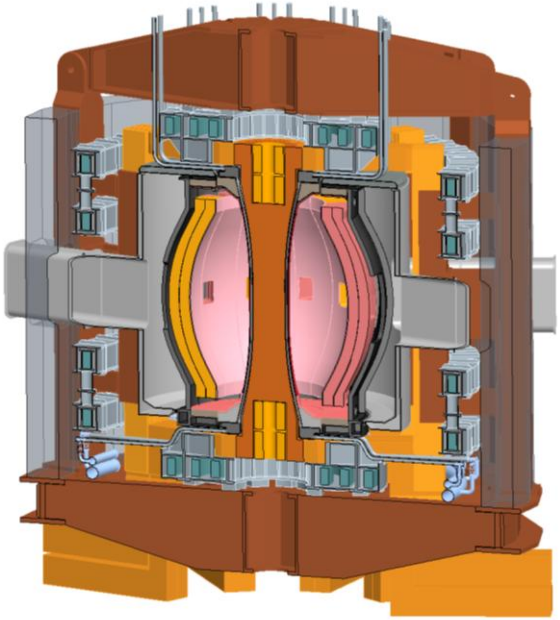
Plan View

Center-stack removal scheme



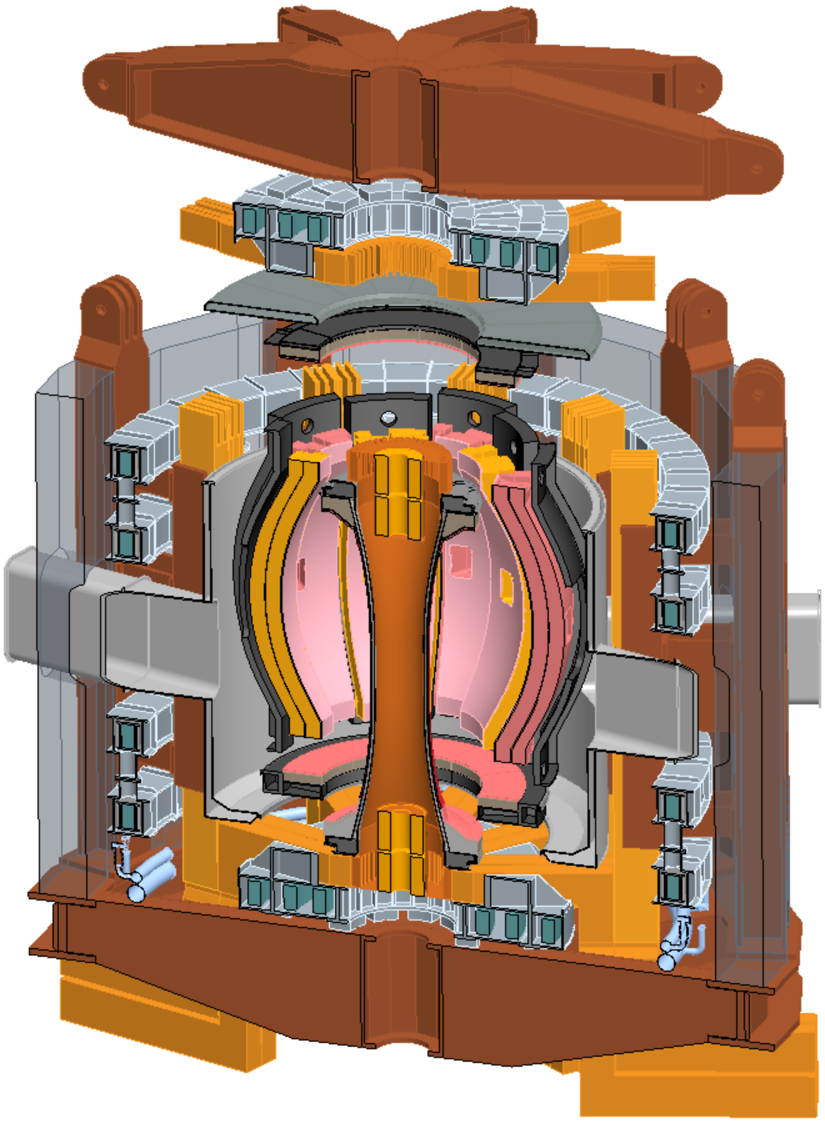
Centerstack removal independent of outer blanket assembly

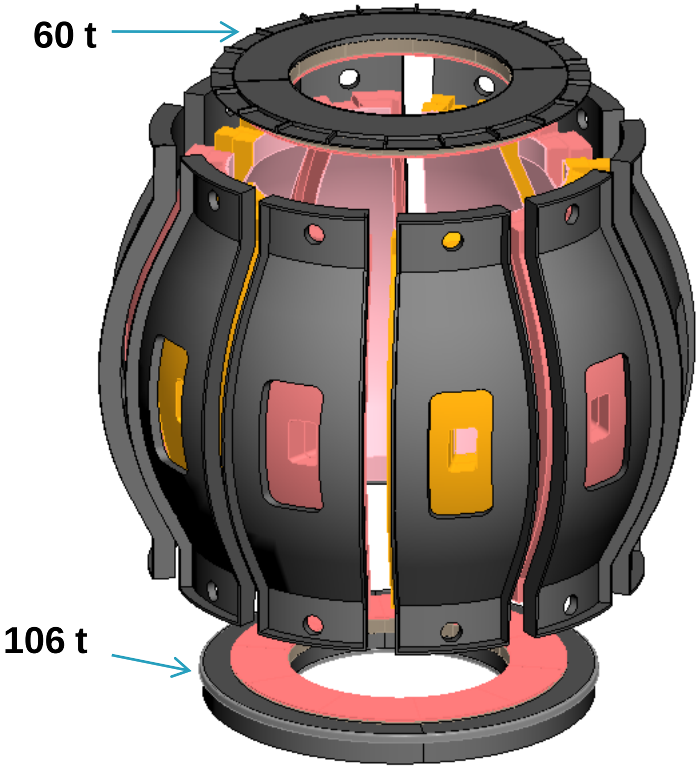




Centerstack weight 525 tonnes

Blanket / shield removal scheme



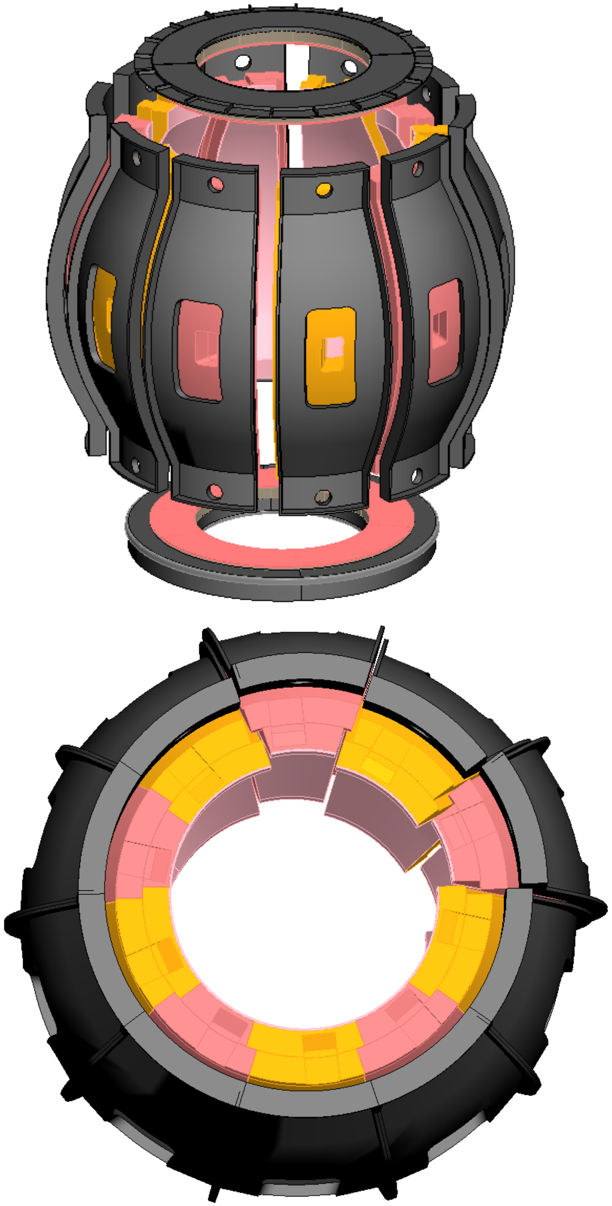


Complete blanket / shield assembly is 1501 tonnes (assumes 40% void fraction when drained)

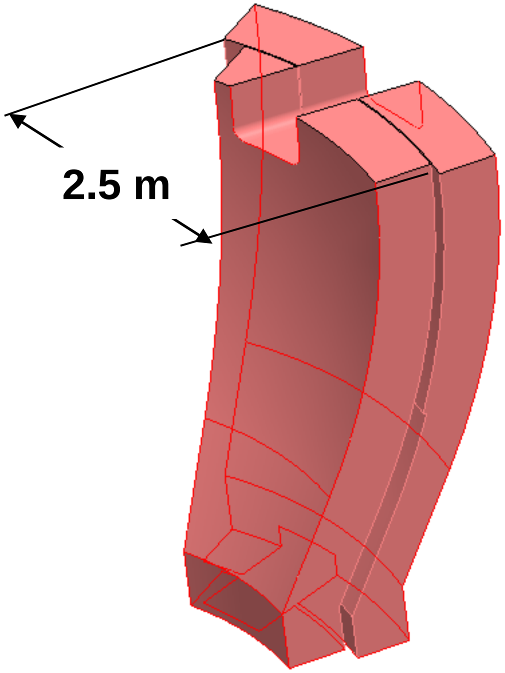
Goal is to limit crane to 1500 tonnes

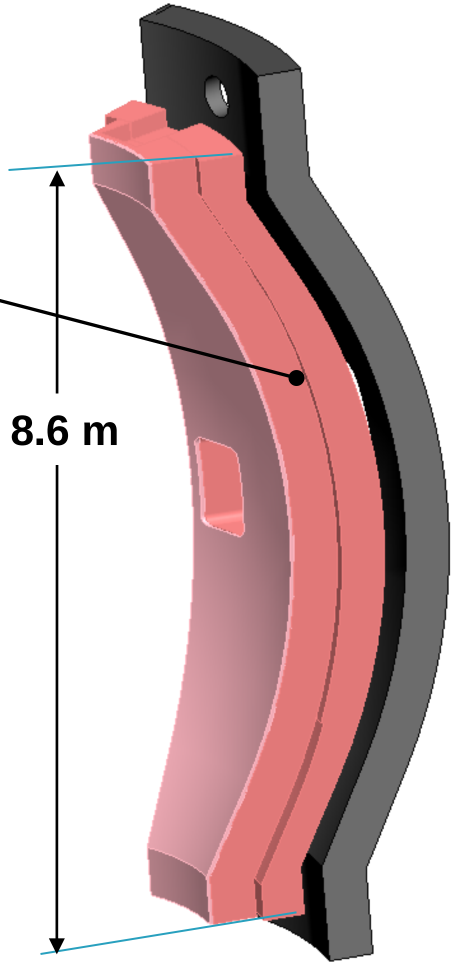
Lift as unit or as sub-assemblies

Blanket support structure, neutron labyrinth

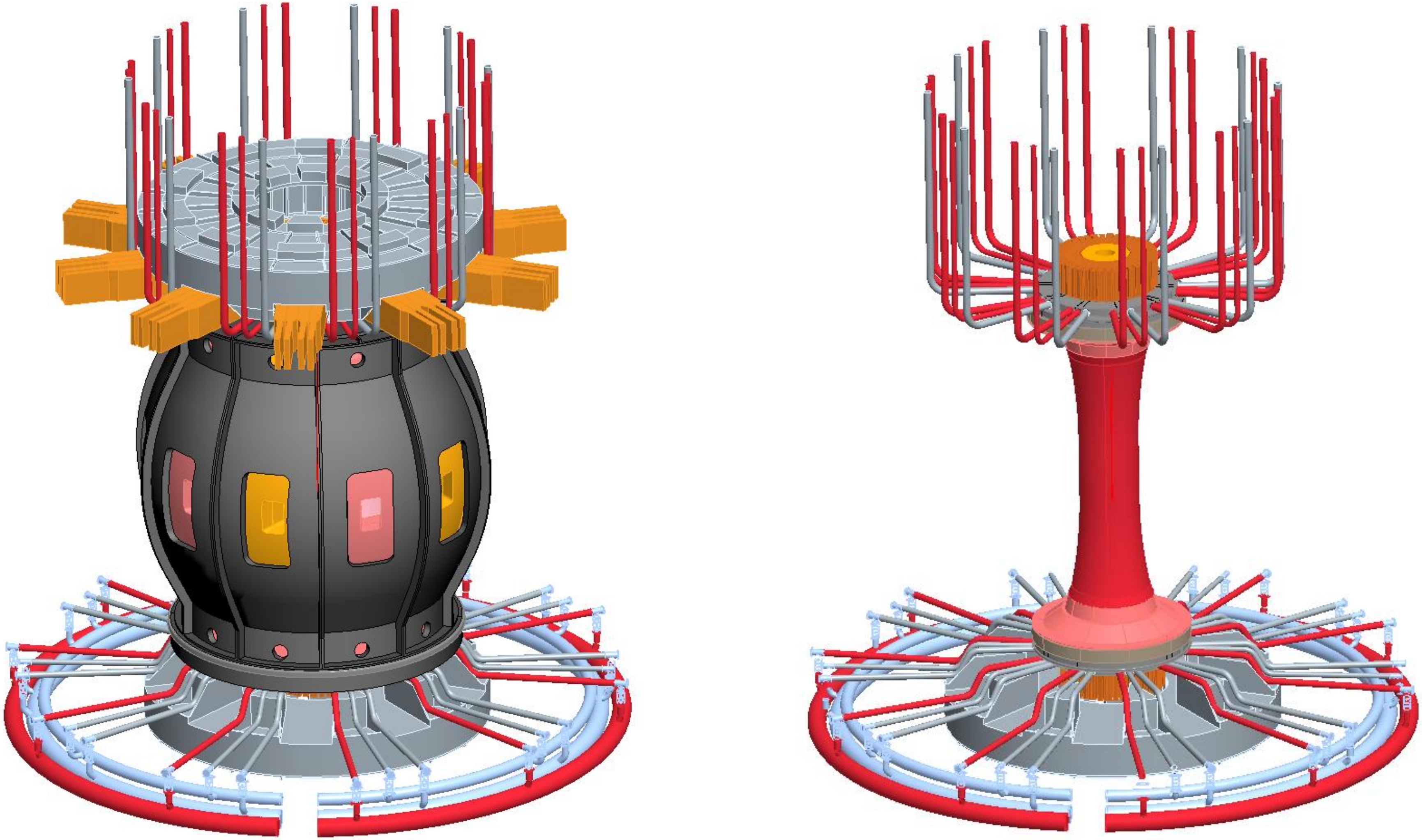


The local blanket assembly is 78.5 tonnes (assuming a 40% void fraction)



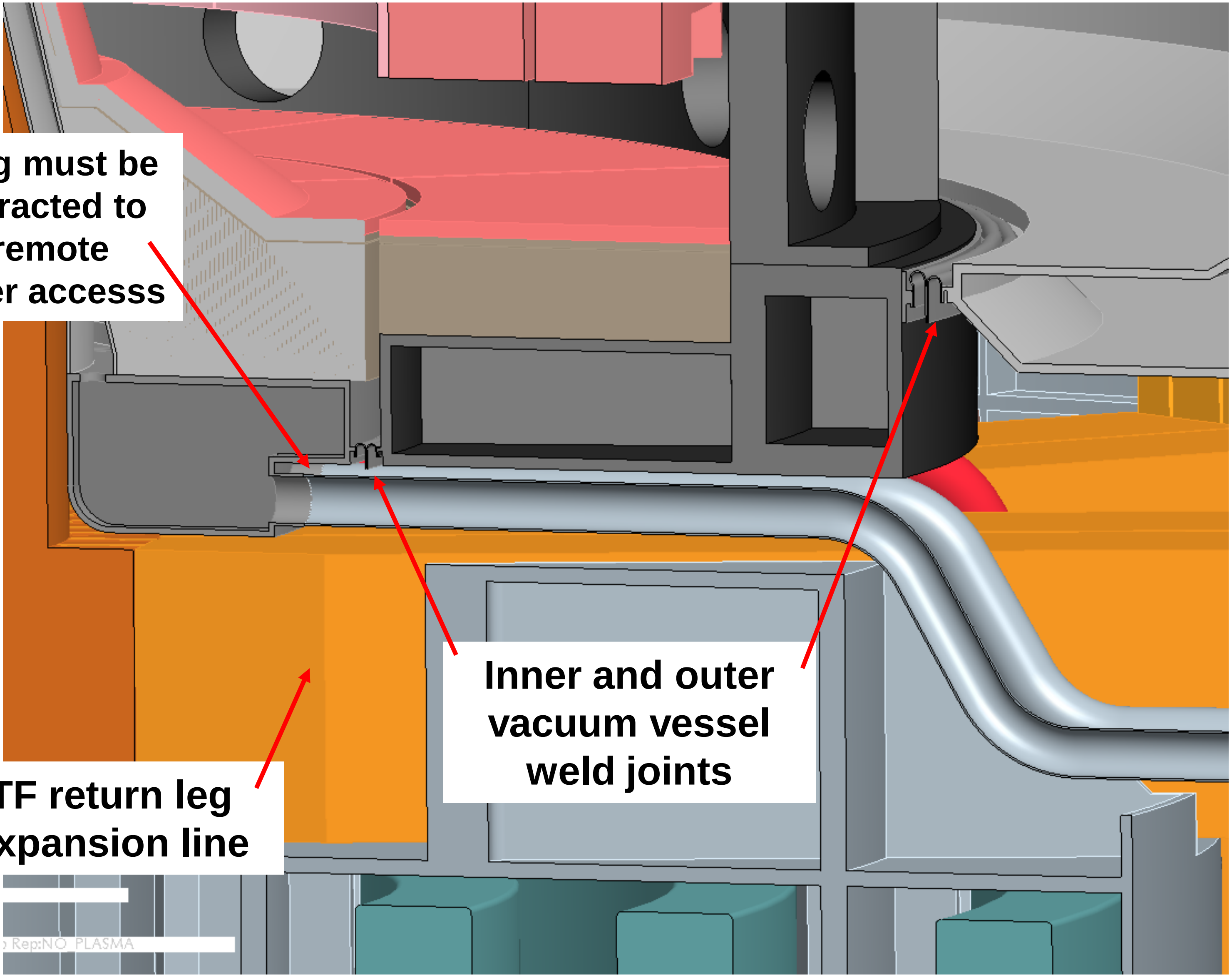


CS coolant channels around TF, blankets



Future work: blanket manifolds, improved divertor definition

Local interface details – joints, welds



Local piping must be cut and retracted to provide remote welder/cutter accesss

Inner and outer vacuum vessel weld joints

TF return leg expansion line

Substantial R&D needed for FNSFs, pilots

- Improved magnet technology:
  - ST: Large single-turn radiation-tolerant Cu TF magnets
- High-efficiency non-inductive current drive
- Advanced physics:
  - 100% non-inductive sustainment, high κ and β, low disruptivity
  - Non-inductive current ramp-up
- Plasma-material interface capabilities beyond ITER:
  - Long-pulses (~10^6s), high duty-factor (10-50% availability goal)
  - High power loading (P/S_{wall}~1MW/m^2, P/R~30-60MW/m, W/S~0.5-1MJ/m^2)
  - High-temperature first-wall (T_{wall} ~ 350-550C, possibly up to 700C)

Preliminary summary of pilot studies

- Identified Pilot Plant configurations sized between FNSF/CTF and a conventional Demo incorporating:
  - Radial builds compatible with shielding requirements, TBR~1
  - Neutron wall loading ≥ 1MW/m^2 for blanket development
    - Average W_n up to 2-3 MW/m^2 → accelerated blanket development
  - Maintenance schemes applicable to power plants
  - Small net electricity to bridge gap to GWe power plant

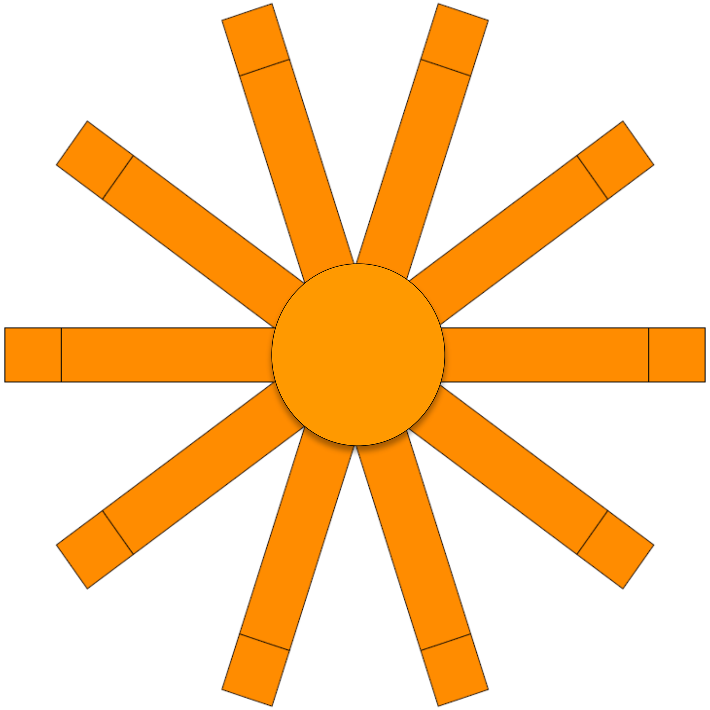
Appears feasible to integrate R&D capabilities needed for fusion commercialization in modest size device

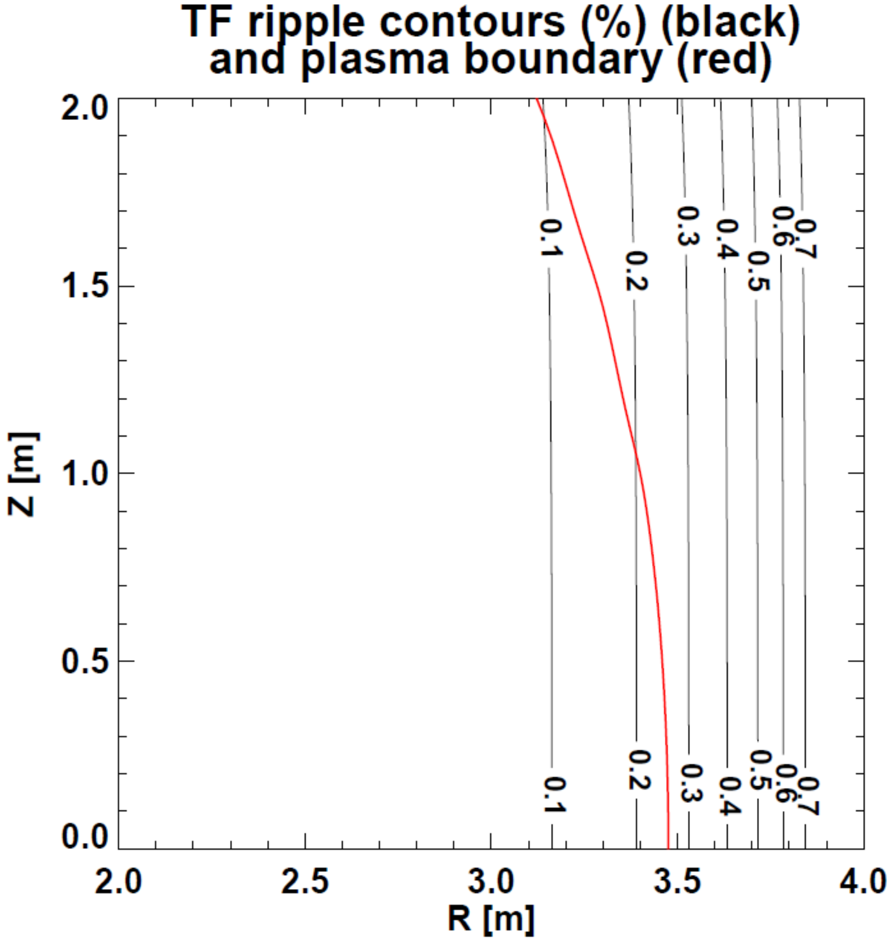
Pilot Plant could be last step before first-generation commercial fusion system

Backup slides

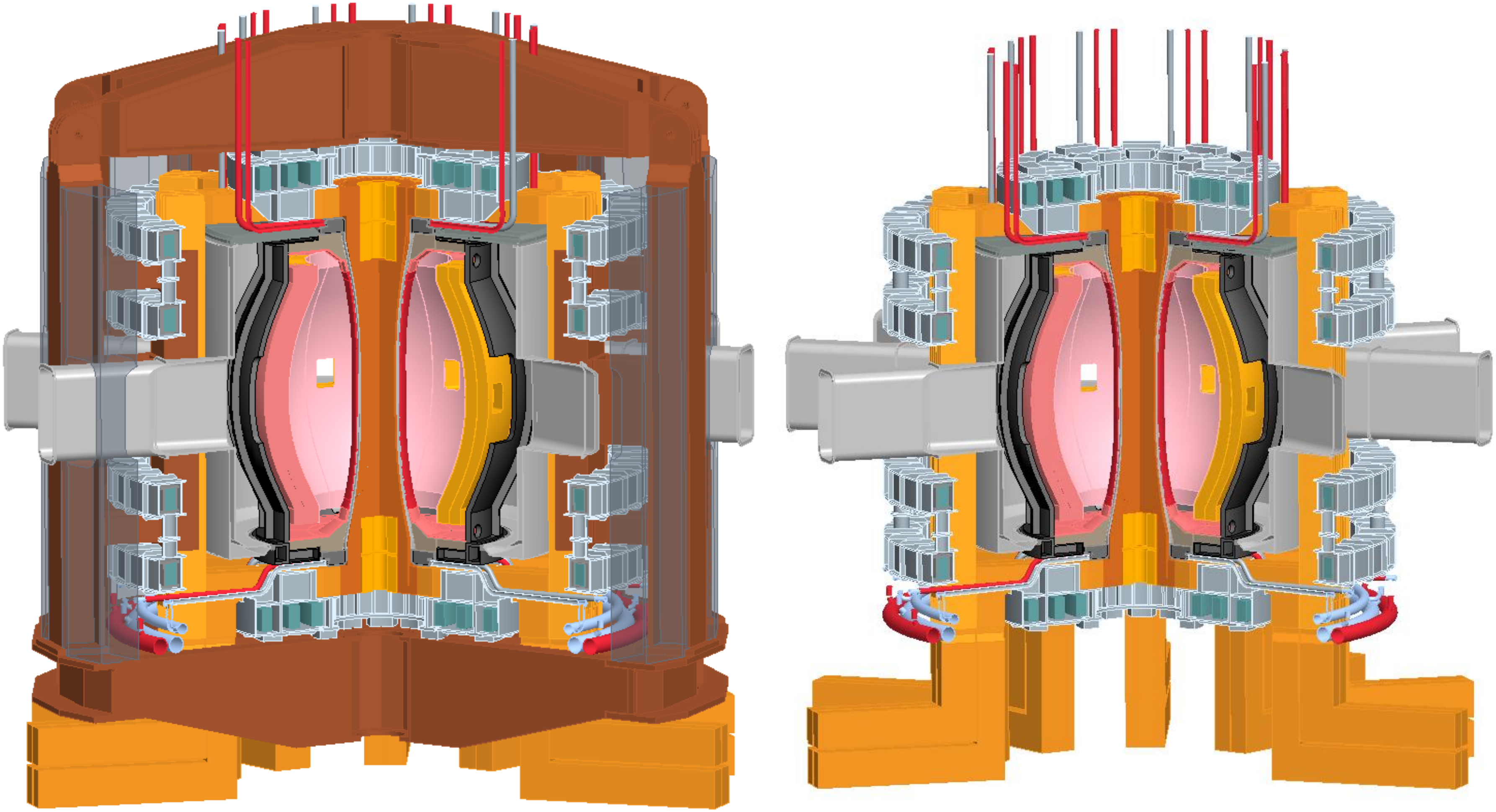
TF outer leg design

- 10 TF outer legs, ripple < 0.25% at plasma

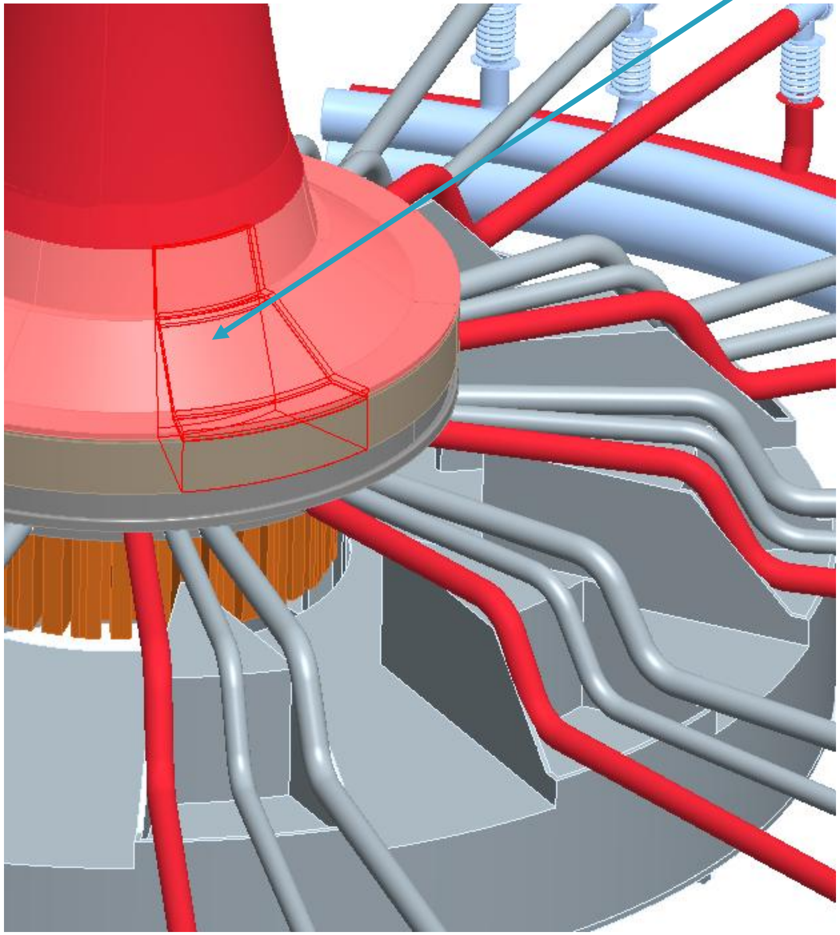




Section views shown with/without external structure



Divertor, TF joint, coolant line details



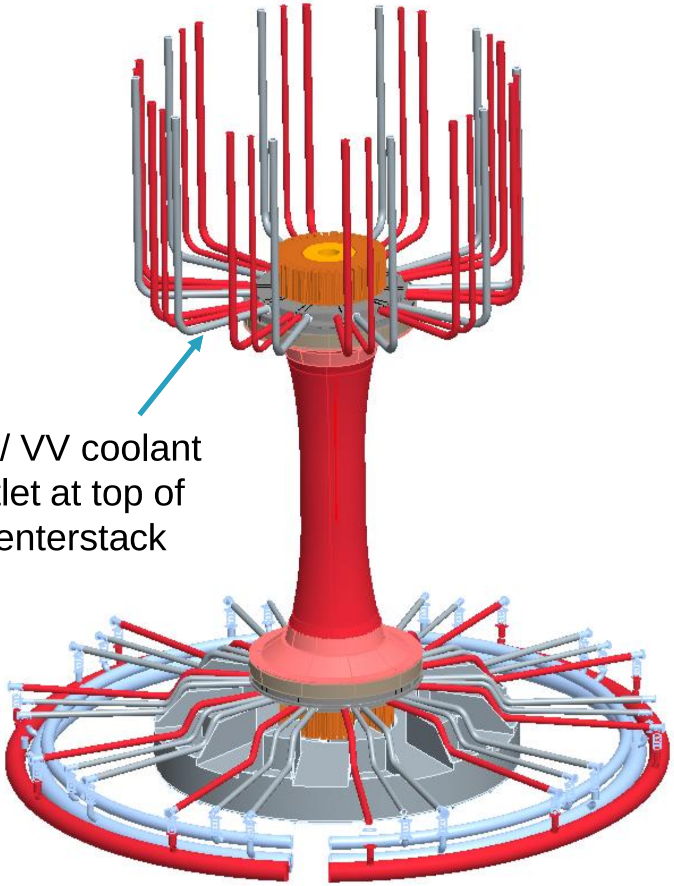
Lower Div Assembly

FW / VV coolant inlet

Lower Div coolant outlet

FW / VV coolant inlet

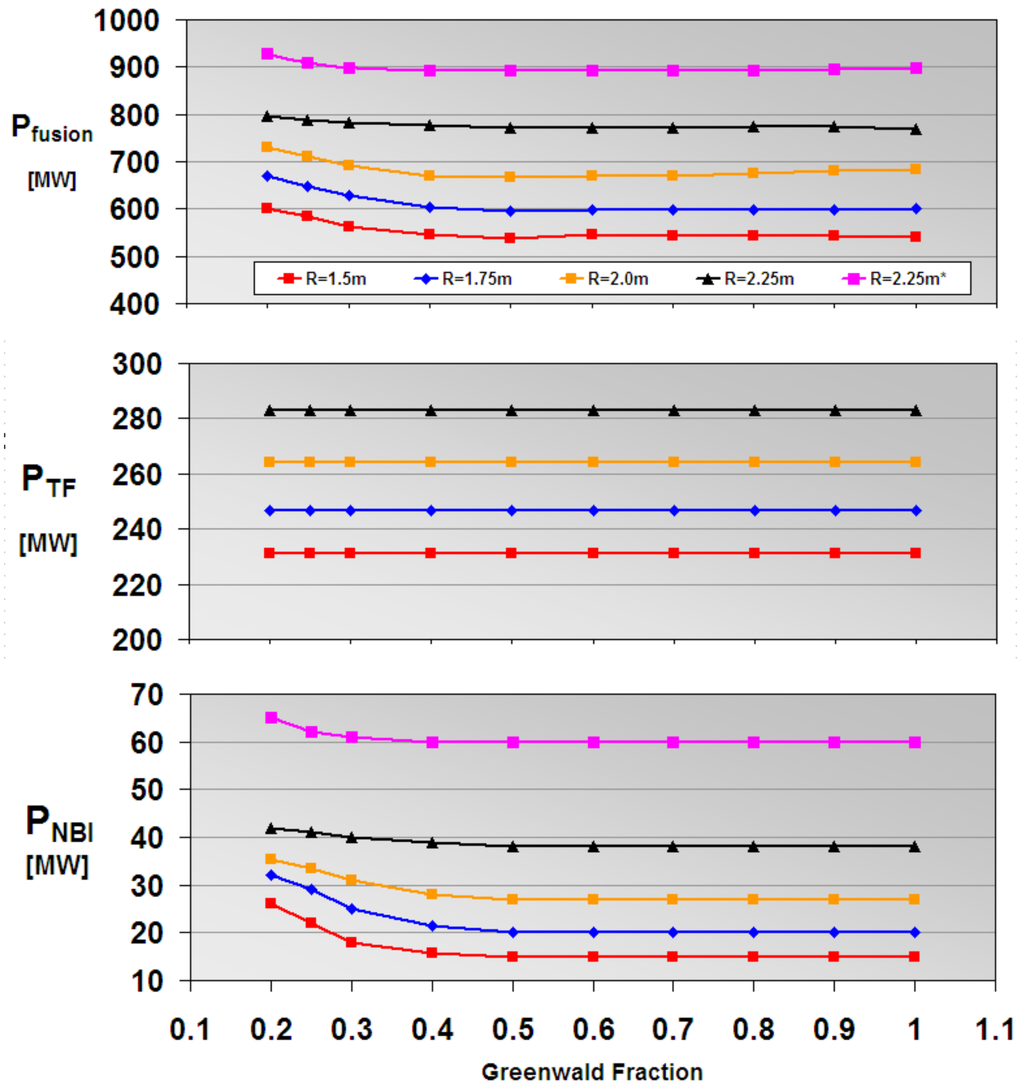
Lower Div coolant inlet



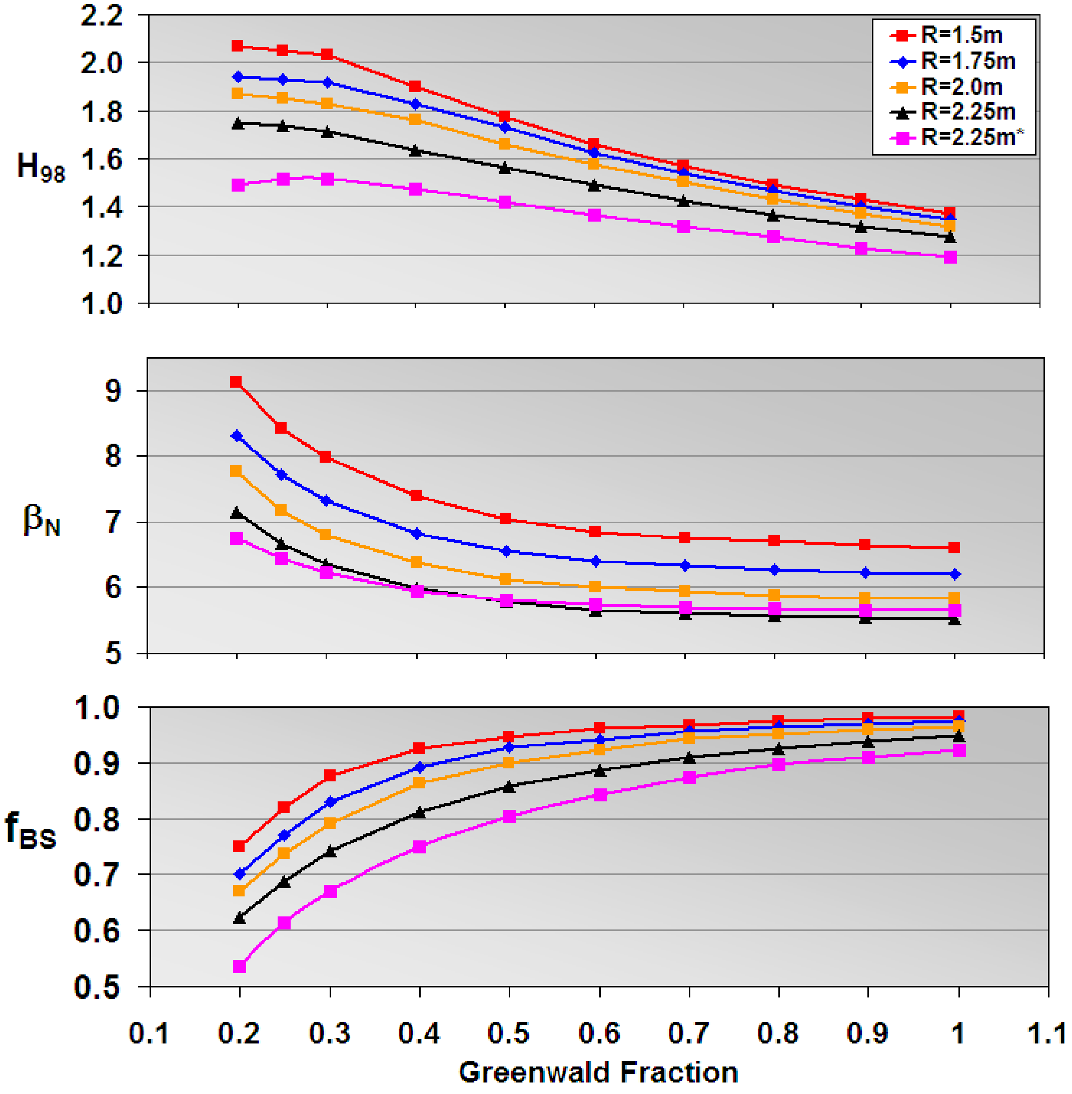
FW / VV coolant outlet at top of Centerstack

Lower inlet / outlet coolant headers

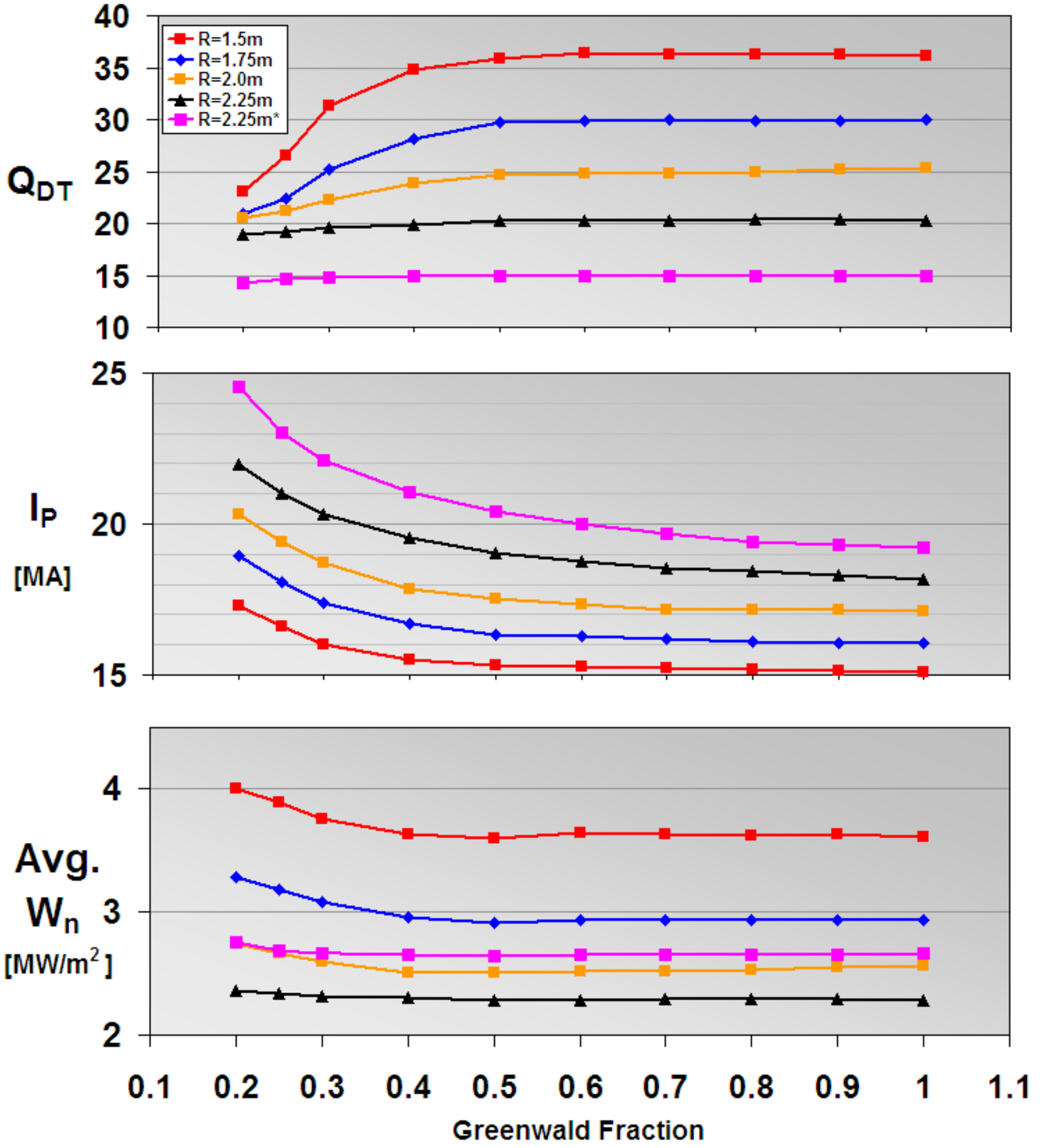
Fusion power, toroidal field dissipation, and NBI power versus major radius at Q_{ENG} ~ 1



H_{98}, β_N, and f_{BS} versus major radius at Q_{ENG} ~ 1

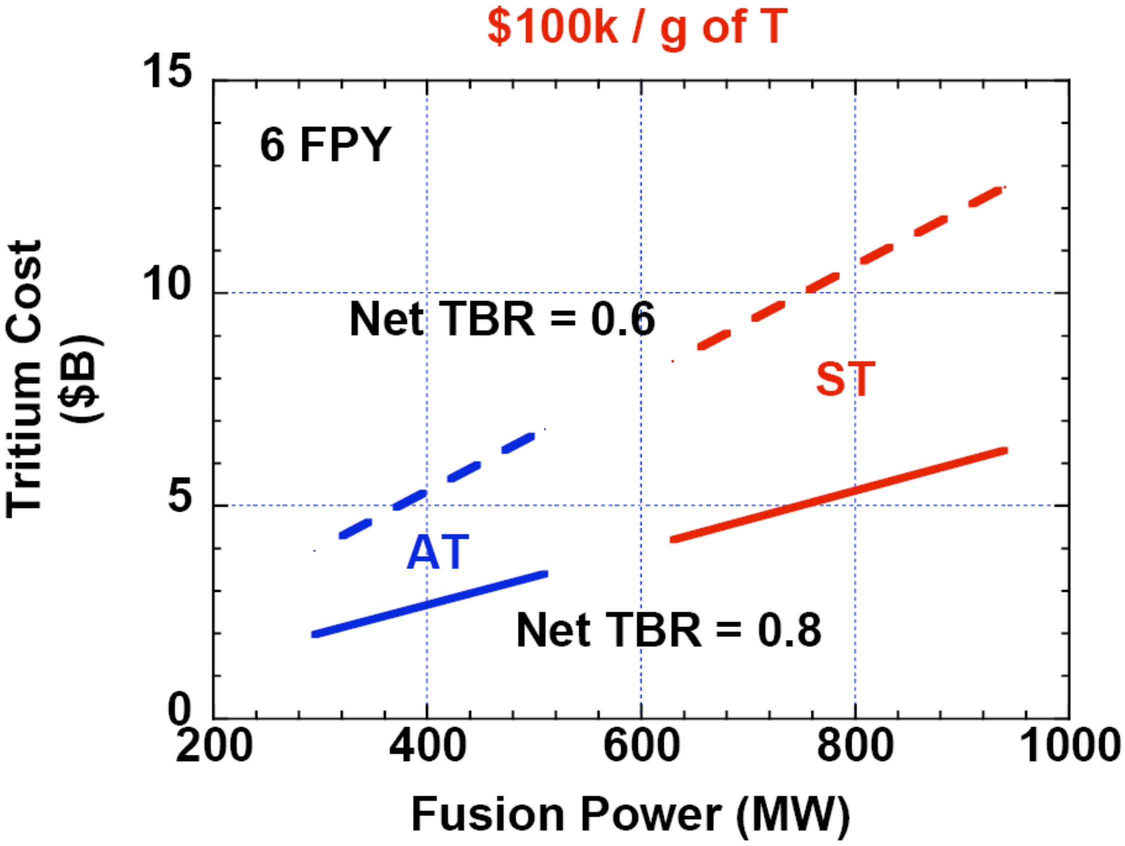


Q_{DT}, I_P, and average neutron wall loading versus major radius at Q_{ENG} = 1



TBR << 1 would increase operating cost of pilots

- Estimated T cost for 6 FPY, TBR = 0.6-0.8 would be $1-10B
- Higher fusion power of ST increases T consumption, cost
- Strongly motivates achieving TBR ≥ 1 early in operation



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July 30, 2010