Configuration Studies for Next-Step Spherical Tokamaks*

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Possible missions for next-step STs

1. Integrate high-performance, steady-state, exhaust
   - Divertor test-tokamak - DTT

2. Fusion-relevant neutron wall loading
   - $\Gamma_n \sim 1-2\text{MW/m}^2$, fluence: $\geq 6\text{MW-yr/m}^2$

3. Tritium self-sufficiency
   - Tritium breeding ratio TBR $\geq 1$

4. Electrical self-sufficiency
   - $Q_{\text{eng}} = \frac{P_{\text{electric}}}{P_{\text{consumed}}} \sim 1$

5. Large net electricity generation
   - $Q_{\text{eng}} \gg 1$, $P_{\text{electric}} = 0.5-1\text{ GWe}$

Recent / Present PPPL-led Studies

Past (& future) PPPL Studies
Identified TF + PF coil set that supports long-leg / Super-X divertor for range of equilibria

- All equilibrium PF coils outside vacuum vessel
- Increased strike-point radius reduces B, q$_\parallel$
  Strike-point PFCs also shielded by blankets
- 2$^{\text{nd}}$ 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 $\delta$
- Breeding in CS ends important for maximizing TBR
Up/down-symmetric long-leg divertor $\Rightarrow q_{\perp}$-divertor $< 1-2 \text{ MW/m}^2$ under detached conditions (SOLPS / ORNL)
Negative NBI (0.5 MeV) with large $R_{\text{TAN}}$ favorable for heating and current drive (CD) for $R=1.7\text{m}$ ST-FNSF.

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

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

- **Optimal tangency radii:**
  - $1.7\text{m} \leq R_{\text{tan}} \leq 2.4\text{m}$

- Control $q(0)$, $q_{\text{min}}$
- Shine-thru limit
Free-boundary TRANSP/NUBEAM used to compute profiles for 100% non-inductive plasmas with $Q_{DT} \sim 2$

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

- Maintain $q_{min} > 2$
- $q(0) / q_{min}$ controllable via $R_{tan}$ and density
**R=1.7m configuration with Super-X divertor**

**Design features**
- TF coils
- SC PF coils pairs located in common cryostat
- Cu/SC PF coils housed in VV lower shell structure
- Cu/SC PF coils housed in VV upper lid
- VV outer shell with shield material
- Ports for TBM, MTM, NBI
- Blankets
- TF leads
- Angled DCLL concentric lines to external header

**Vertical maintenance**
Breeding at CS ends important: \( \Delta TBR = +0.07 \)

Inner Blanket Segment = 0.81

Outer Blanket Segment = 0.15

Total TBR ~ 1.03 with no penetrations or ports (heterogenous outboard blanket)
Summary of TBR vs. device size for A=1.7 Cu-TF ST-FNSF

R=1.7m:  \( \text{TBR} \approx 1 \)

R=1.0m:  \( \text{TBR} < 1 \) (≈ 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 $\leq R_0 = 3m$ 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: $\kappa(\varepsilon), \beta_N(\varepsilon)$

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

\[
\beta_T \sim A^{-1/2} (1 + \kappa^2) \beta_N^2 / f_{BS}
\]
\[
P_f \propto \epsilon (\kappa \beta_N B_T)^4
\]
\[ Q_{\text{eng}} \text{ maximized between } A = 1.7-2.3 \text{ at fixed } R_0, \]

Optimal \( A \) depends on inboard WC shield thickness.

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

\[
Q_{\text{engineering}} = \frac{P_{\text{electric}}}{P_{\text{consumed}}}
\]

Want \( \Delta_{\text{shield}} / R_0 < \sim 20\% \) for \( Q_{\text{eng}} > 1 \).
Configuration Studies for Next-Step STs (J. Menard)

Selection of device HTS-ST performance goals

- Attempt to satisfy FNSF (fluence) and Pilot (net electric) goals
  - 6MWy/m² neutron wall loading (peak) at outboard midplane
  - $Q_{\text{eng}} \sim 1$ – similar to previous PPPL Pilot Plant Study

- Assume n-radiation damage limit of $3-5 \times 10^{22}/m^2$
  - HTS already tested to this damage fluence range (see next slide)
  - WC shield thickness $\sim 60\text{cm}$, $\Delta/R = 0.2 \rightarrow R_0 = 3\text{m}$

- With small / no inboard breeding, optimal $A \sim 2.1-2.4$

- But, for TBR $\sim 1$ probably need $A \leq 2 \rightarrow$ chose / try $A=2$

- Chosen design point (so far):
  - $R=3\text{m}$, $B_T = 4\text{T}$, $A=2$, $\kappa=2.5$, $\beta_N = 4.2$ (~no-wall limit)
  - $H_{98\text{y}2} \sim 1.7$, $H_{\text{Petty}} \sim 1.2-1.3$, $H_{\text{ST}} \sim 0.7$, $P_{\text{fusion}} \sim 500-600\text{MW}$
  - 80% Greenwald fraction, 50MW of 0.5-0.7 MeV NNBI
  - $I_p = 12\text{MA}$, double-swing of small OH provides $\sim 2-3\text{MA}$
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 $\rightarrow$ 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 \( (E_n > 0.1 \text{ MeV}) \)  \( 4.3 \times 10^{18} \ \text{n/cm}^2 \)
  - Peak nuclear heating @ WP  \( 1.7 \ \text{mW/cm}^3 \)
  - Peak dose to electrical insulator  \( 4 \times 10^9 \ \text{rads} \)
  - Total nuclear heating in IB magnet  \( 8.7 \ \text{kW} \)
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. \( \text{Li}_{17}\text{Pb}_{83} \) 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

Expect final TBR ≈ 0.95-1. Options to increase:
- Thin inboard breeding region (assessing now)
- Reduce aspect ratio (reduces \( Q_{\text{eng}} \), no CS)
Summary

- Developed self-consistent $A=1.7$ Cu TF ST configurations w/ high TBR for fluence mission
  - $R_0 = 1 \text{ m} \rightarrow \text{TBR} \sim 0.9$
  - $R_0 = 1.7\text{m} \rightarrow \text{TBR} \sim 1.0$
- Optimal $A$ for fusion performance with HTS TF and small/no inboard breeding or CS is $A \approx 2$
  - High confinement may be required to exploit higher toroidal field potentially achievable with HTS
- $A \approx 2 \ R_0 = 3\text{m}$ HTS FNSF / Pilot with fluence $6\text{MWy/m}^2$ (peak) and $Q_{\text{eng}} \sim 1$ 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 ($\text{Nb}_3\text{Sn}$):
  - Fast neutron fluence: $10^{23} \text{ n/m}^2 \ (E_n > 0.1\text{MeV})$
    - PF3: $1.15 \times 10^{23} \text{ n/m}^2$
  - Peak Dose to Insulator: $2 \times 10^{10} \text{ rads}$
    - PF3: $1.08 \times 10^{10} \text{ rads}$
  - Peak Nuclear Heating: $2 \text{ kW/m}^3$
    - PF3: $2.2 \text{ kW/m}^3$
Up/down-symmetric long-leg divertor $\Rightarrow$ $q_\perp$-divertor $< 10$MW/m$^2$

even under attached conditions (if integral heat-flux width $\lambda_{q\text{-int}} > 2$mm)

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

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

Eich NF 2013: $\lambda_{q\text{-int}} = \lambda_q + 1.64 \times S$, $\lambda_q = 0.78$mm, $S \approx \lambda_q$ (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 $S = q_{95}l_p/aB_T > 25$ 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.

Table 1: Comparison of radiation resistant materials

<table>
<thead>
<tr>
<th>Insulation</th>
<th>Organic</th>
<th>Inorganic</th>
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</thead>
<tbody>
<tr>
<td></td>
<td>Epoxy</td>
<td>MgO</td>
</tr>
<tr>
<td></td>
<td>&gt;10^7 Gy</td>
<td>&gt;10^11 Gy</td>
</tr>
<tr>
<td></td>
<td>Polyimide</td>
<td>&gt;10^9 Gy</td>
</tr>
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</table>
**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 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
### Parameter:

<table>
<thead>
<tr>
<th>Parameter</th>
<th>R=1.7m</th>
<th>R=1.0m</th>
</tr>
</thead>
<tbody>
<tr>
<td>Major Radius</td>
<td>1.68m</td>
<td>1.0m</td>
</tr>
<tr>
<td>Minor Radius</td>
<td>0.95m</td>
<td>0.6m</td>
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<tr>
<td>Fusion Power</td>
<td>162MW</td>
<td>62MW</td>
</tr>
<tr>
<td>Wall loading (avg)</td>
<td>1MW/m²</td>
<td>1MW/m²</td>
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<tr>
<td>TF coils</td>
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<td>10</td>
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<td>TBM ports</td>
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<td>4</td>
</tr>
<tr>
<td>MTM ports</td>
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<td>1</td>
</tr>
<tr>
<td>NBI ports</td>
<td>4</td>
<td>3</td>
</tr>
</tbody>
</table>

- **Plant Lifetime**: ~20 years
- **Availability**: 10-50%

### Neutron source distribution

- Neutron source distribution image

---

**Two sizes (R=1.7m, 1m) assessed for shielding, TBR**
Mapping of dpa and FW/blanket lifetime (R=1.7 m Device)

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

Peak = 15.5 dpa / FPY

Peak EOL Fluence = 11 MWy/m²
Peak radiation damage at PF coils are within allowable limits for different coil types (Cu with MgO, LTS/HTS)

- Peak dpa at OB midplane = 15.5 dpa / FPY
- Dose to MgO insulator = 3.1x10^8 Gy @ 6 FPY < 10^{11} Gy limit
- Dose to MgO insulator = 1.2x10^{10} Gy @ 6 FPY < 10^{11} Gy limit
- Peak He production at OB midplane = 174 appm/FPY
  ⇒ He/dpa ratio = 11.2
0.5MeV NNBI well confined down to $\sim$2MA
Options to increase TBR > 1

- Add to PF coil shield a thin breeding blanket ($\Delta TBR \sim +3\%$)
- 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
$R_0 = 1\text{m ST-FNSF achieves TBR} = 0.88$

- **1m device cannot achieve TBR > 1 even with design changes**
- **Solution**: purchase $\sim 0.4-0.55\text{kg of T/FPY}$ from outside sources at $\$30-100/\text{kg of T}$, costing $\$12-55\text{M/FPY}$
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 ($\Delta$TBR = -0.01)

Approx. $\Delta$TBR 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 $Q_{\text{eng}} > 1$
• Inboard plasma/FW gap = 4cm
• Use $\varepsilon$ dependent $\kappa(\varepsilon), \beta_N(\varepsilon)$ (see next slide)
• Greenwald fraction = 0.8
• $q^*$ not constrained
  – $q^*$ is better $\varepsilon$-invariant than $q_{95}$ for current limit
  – Want to operate with $q^* > 3$ to reduce disruptivity
• 0.5MeV NNBI for heating/CD – fixed $P_{\text{NBI}} = 50\text{MW}$
• $H_{98y2}$ 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/m²
  – OH at small R → higher OH solenoid flux swing for higher A

• Shielding / blankets
  – Assume HTS fluence limit of 3-5x10²²
  – 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

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 $2.3 \cdot 10^{22} \text{ m}^{-2}$.

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.

Fig. 3 The change of positron mean lifetime (MLT) after irradiation and annealing (a); Critical temperatures measured in SQUID magnetometer (b).
### Configuration Studies for Next-Step STs (J. Menard)

**3.00 m R0**
CCFE HTS ST radial build

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<tr>
<th>10 TF coils</th>
<th>2.00</th>
<th>AR</th>
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<td><strong>TOTAL</strong></td>
<td><strong>TOTAL</strong></td>
</tr>
<tr>
<td><strong>(in)</strong></td>
<td><strong>(mm)</strong></td>
<td><strong>(in)</strong></td>
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<tr>
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<tr>
<td></td>
<td>Plasma minor radii</td>
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</table>

**CALCULATIONS for OH at inner bore**

\[
B_{ph} = \mu_0 J_s (R_0 - R_i) \text{ where } \mu_0 = 4\pi \times 10^{-7}
\]

Assume \( J_s \) (MA/m²) = 70

\[ B (T) = 19.8 \]

**OH stress = \( B R_i J_s I_z \)**

\[ \text{stress} = 47.11 \text{ Mpa} \]

**OH flux = \( B \times \text{mean area of xolenoid} \)**

\[ \text{flux} = 2.03 \text{ volt seconds (weber)} \]

**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:**

\[ 2 \text{Vs with 70 Jc solenoid and 19.8 T} \]
TF and OH magnet parameters vs. aspect ratio using models from Brown and Zhai.

<table>
<thead>
<tr>
<th>WC thk (cm)</th>
<th>R0 (m)</th>
<th>AR</th>
<th>ε = 1/AR</th>
<th>a (m)</th>
<th>B0 (T)</th>
<th>WP thk case thk</th>
<th>O-Cd (MA/m²)</th>
<th>W-Cd (MA/m²)</th>
<th>Bmax (Mpa)</th>
<th>Ave stress Tresca (ksi)</th>
<th>Ave stress Tresca (ksi)</th>
<th>OH IR (cm)</th>
<th>OH thk (cm)</th>
<th>OH B (T)</th>
<th>OH V-s</th>
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Assessing long-leg / deep-V divertor

- PF coils outside TF
- Increase strike-point radius $\sim 2\times$ 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

\( \lambda_q \sim 1\text{mm}, \text{assume } S \approx \lambda_q \) (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)

\[ \lambda_q \sim 1\text{mm}, \text{ assume } S \approx \lambda_q \text{ (closed divertor)} \]

(T. Eich NF 2013)

(Partial) detachment likely reduces peak \( q_\perp \) by further factor of 2-4
Latest HTS-ST: R=3m, A=2, $P_{\text{fusion}} \sim 500\text{MW}$, $Q_{\text{eng}} \sim 1-1.5$
## R=3 m Configuration and Key Parameters

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<th>Parameter</th>
<th>Value</th>
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<td>Minor Radius</td>
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<td>Fusion Power</td>
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<td>Plant Lifetime</td>
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<td>Availability</td>
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10 Sectors

3.1 Full Power Years (FPY)
Neutron Wall Loading Distribution

Evaluated with 3 methods:

- Average NWL: 1.13 MW/m²
- Peak IB NWL: 1.57 MW/m²
- Peak OB NWL: 2.15 MW/m²
- Peak Div NWL: 0.42 MW/m²

![Graphs showing IB and OB neutron wall loading distributions with different sources.

Fusion power density

Red: Plasma Boundary
Black: Limiter

![Graph showing fusion power density with contour lines.

Configuration Studies for Next-Step STs (J. Menard)
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).
Radiation Limits

Overall TBR
(for T self-sufficiency)

Damage to RAFM steel structure
20 dpa – GEN-I
< 50 dpa – GEN-II
> 65 dpa – ODS (NS)

Helium Production
(for reweldability of FS)

HTS Magnet (@ 20-40 K):
Peak fast n fluence to HT superconductor (E_n > 0.1 MeV) 5 x 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 5-10 x 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:**
• Fast neutron fluence to HTS drives IB shield design.
• Combination of WC and H$_2$O represents superior shielding option as it helps reduce both fluence and magnet heating.
• Avoid:
  – Using B-H$_2$O and hydrides (having less shielding performance compared to WC/H$_2$O)
  – 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) $4.3 \times 10^{18}$ n/cm$^2$
  - Peak nuclear heating @ WP $1.7$ mW/cm$^3$
  - Peak dose to electrical insulator $4 \times 10^9$ 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 (FCI).
  - 2 cm thick W vertical stabilizing shell between OB blanket segments.
  - Ports (4 TBMs, 1 MTM, NNBIs).
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

- B-FS/D$_2$O Shield-I helps enhance OB breeding and control IB decay heat
- VV composition optimizes at 60% WC and 35% H$_2$O
- Magnet heating limit is well met
- Need 1-2 cm additional shield to meet fluence limit.