Neutronics Analysis of HTS-ST

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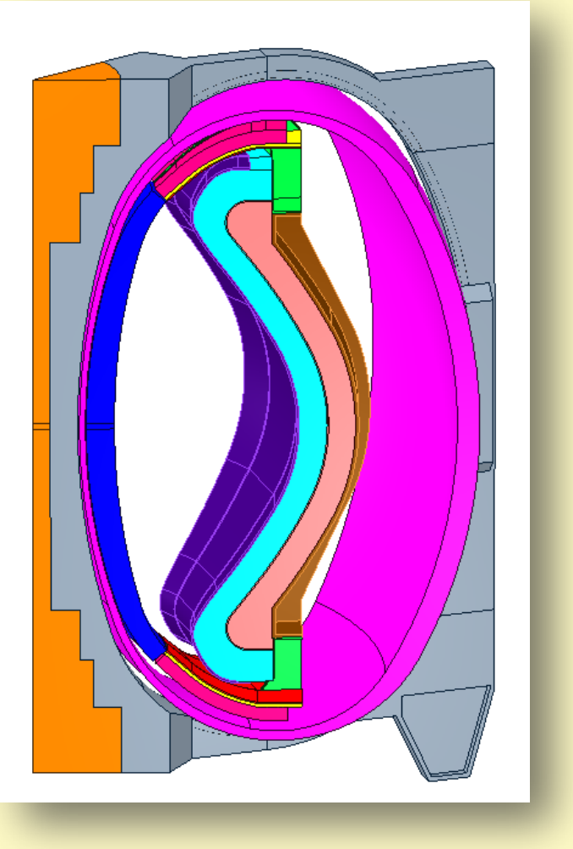
Neutronics Analysis of High Temperature Superconducting Spherical Tokamaks

High Temperature Superconducting Spherical tokamaks (HTS-ST's):

- **compact reactor** – potentially a more cost-effective approach to fusion power,
- smaller physical size of tokamak **limits space** available for neutron **shielding** of the centre column,
- limited inboard space also restricts the **tritium breeding blankets** to the outboard.

Neutronics analysis has been performed on a **HTS-ST model** drawing on results from previous analysis, using simplified parameterised ST models.



Key neutronic results:

Average neutron wall loading (NWL)	~1.29 MW/m^2
Peak centre column toroidal field magnet heating (CC-TF)	5.6 kW/m^3
Peak centre column central solenoid magnet heating (CC-CS)	0.4 kW/m^3
Total tritium breeding ratio (TBR)	1.12

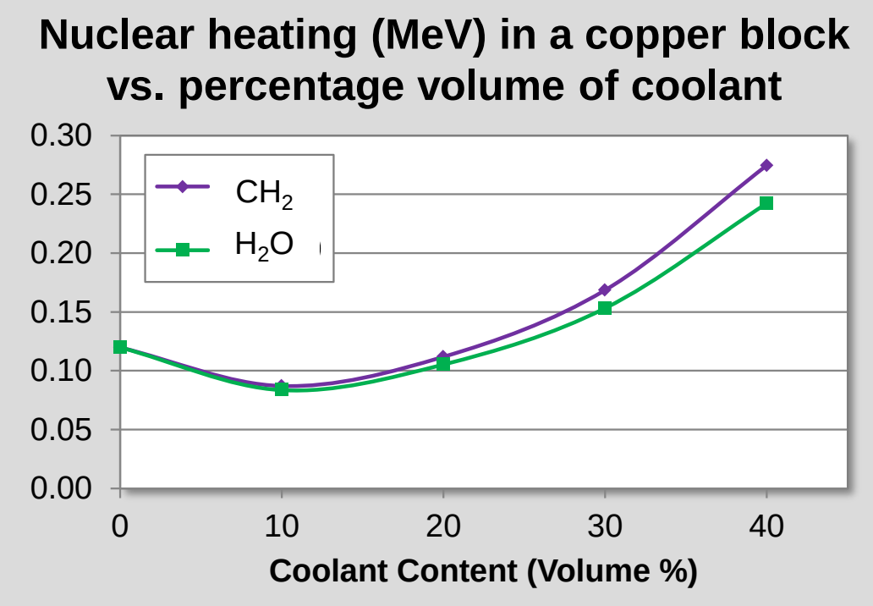
Nuclear Heating

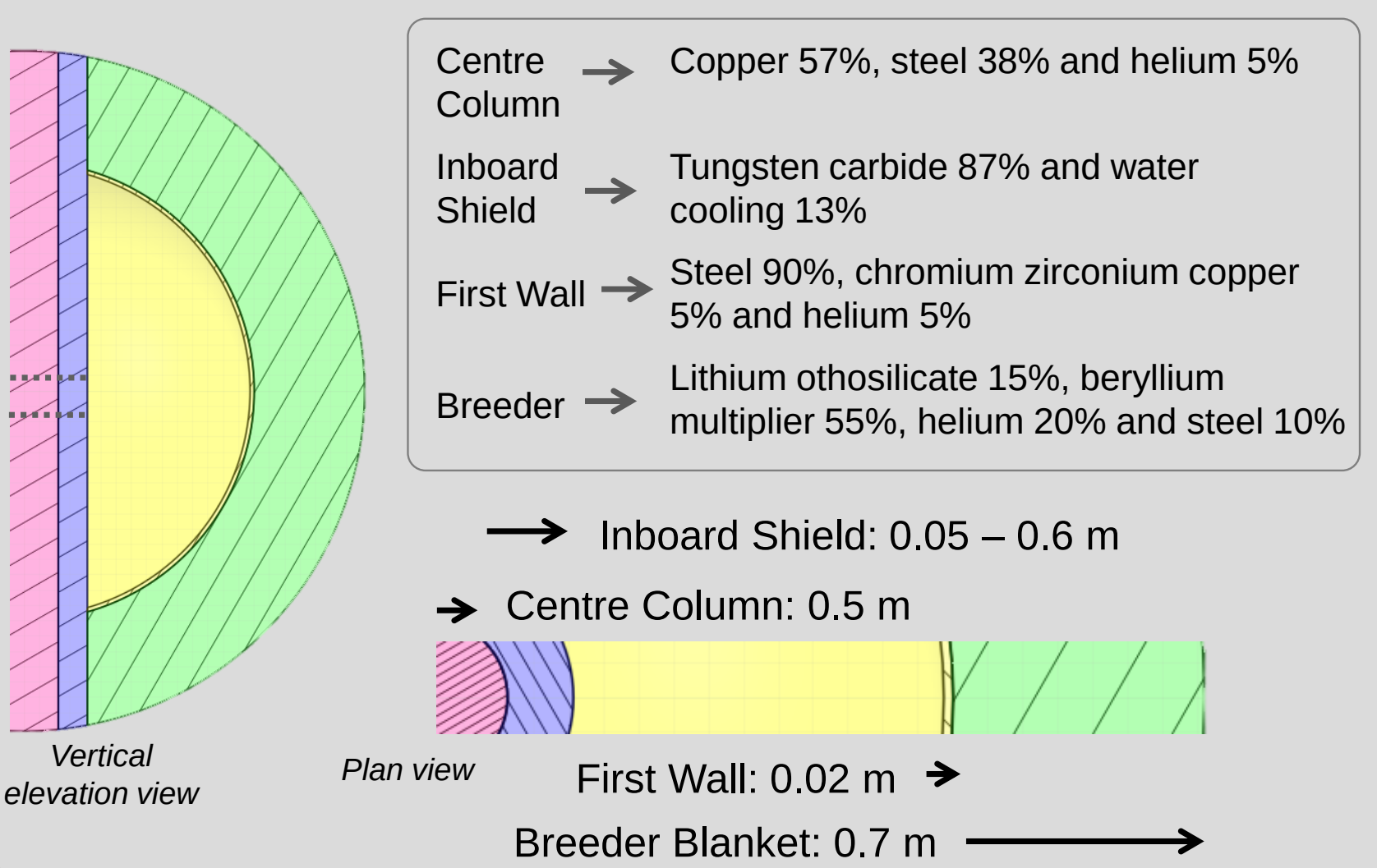
To mitigate heating & damage to the HTS magnets, radiation must be prevented from reaching the coils or attenuated to an acceptable level. In a conventional fusion tokamak, the shielding is provided by the neutron absorbing thick breeding blanket. **Spherical tokamaks** must have an explicit radiation **shield** on the **inboard**.

A parameterised model

Initial studies on a parameterised spherical model show that a **shielding thickness** of **at least 0.4 m** is required to reduce centre column heating to 2 kw/m^3.

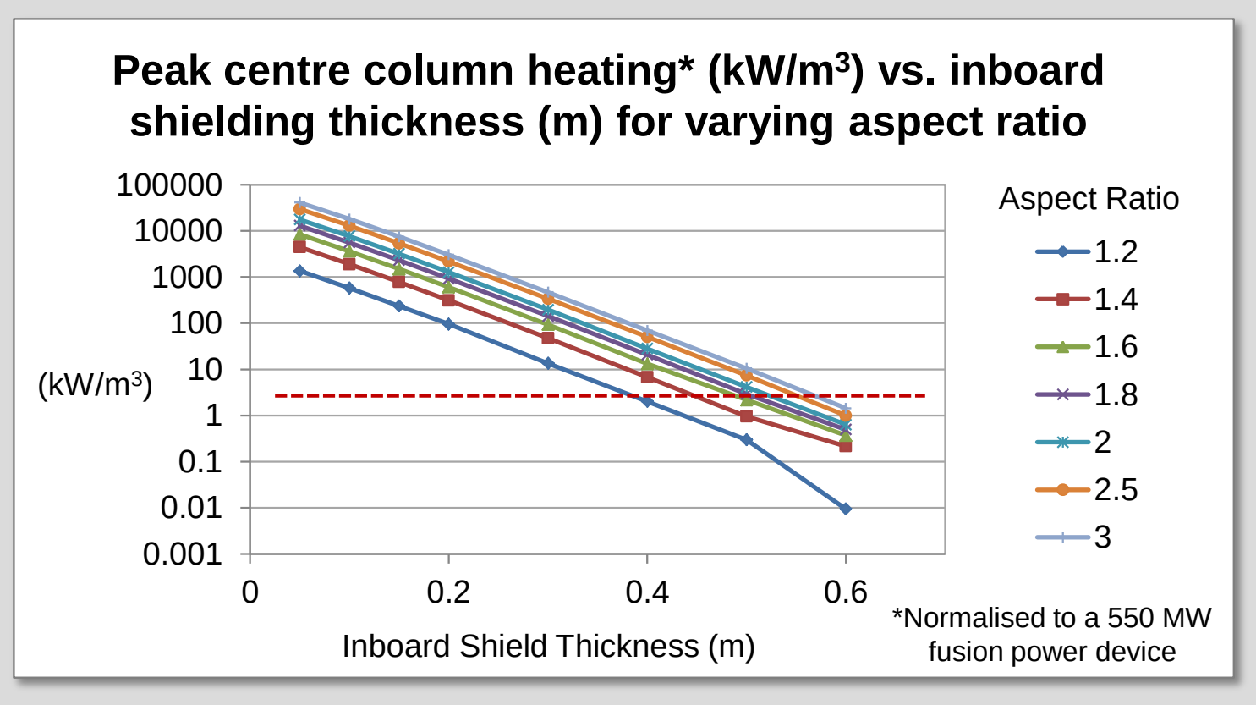
A study of the shielding capabilities of a range of elements verified the use of tungsten carbide, with 13% water coolant.


Nuclear heating (MeV) in a copper block vs. percentage volume of coolant



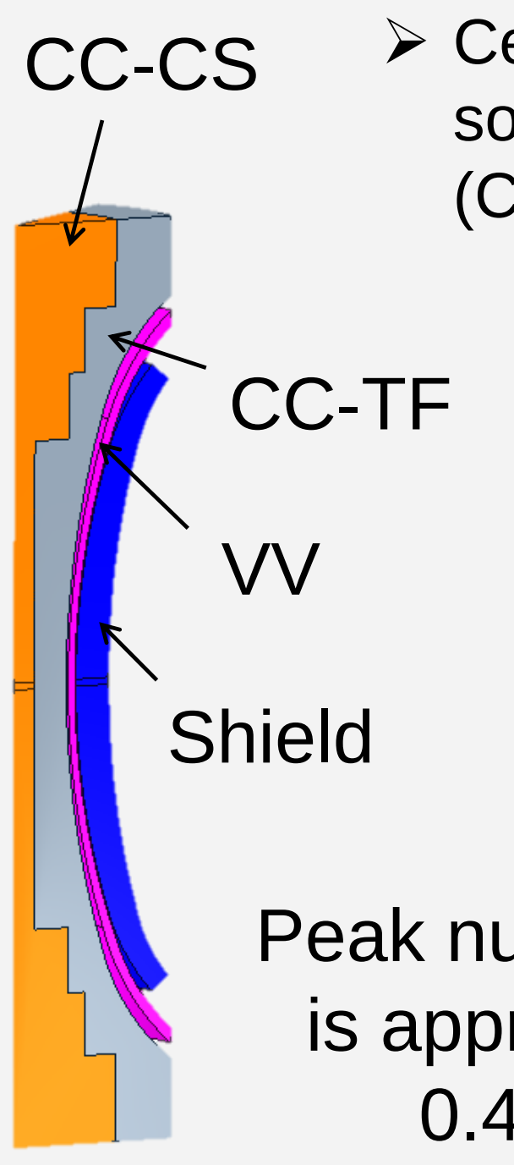
Centre Column →	Copper 57%, steel 38% and helium 5%
Inboard Shield →	Tungsten carbide 87% and water cooling 13%
First Wall →	Steel 90%, chromium zirconium copper 5% and helium 5%
Breeder →	Lithium othosilicate 15%, beryllium multiplier 55%, helium 20% and steel 10%

→ Inboard Shield: 0.05 – 0.6 m
→ Centre Column: 0.5 m
First Wall: 0.02 m →
Breeder Blanket: 0.7 m →


Peak centre column heating* (kW/m^3) vs. inboard shielding thickness (m) for varying aspect ratio

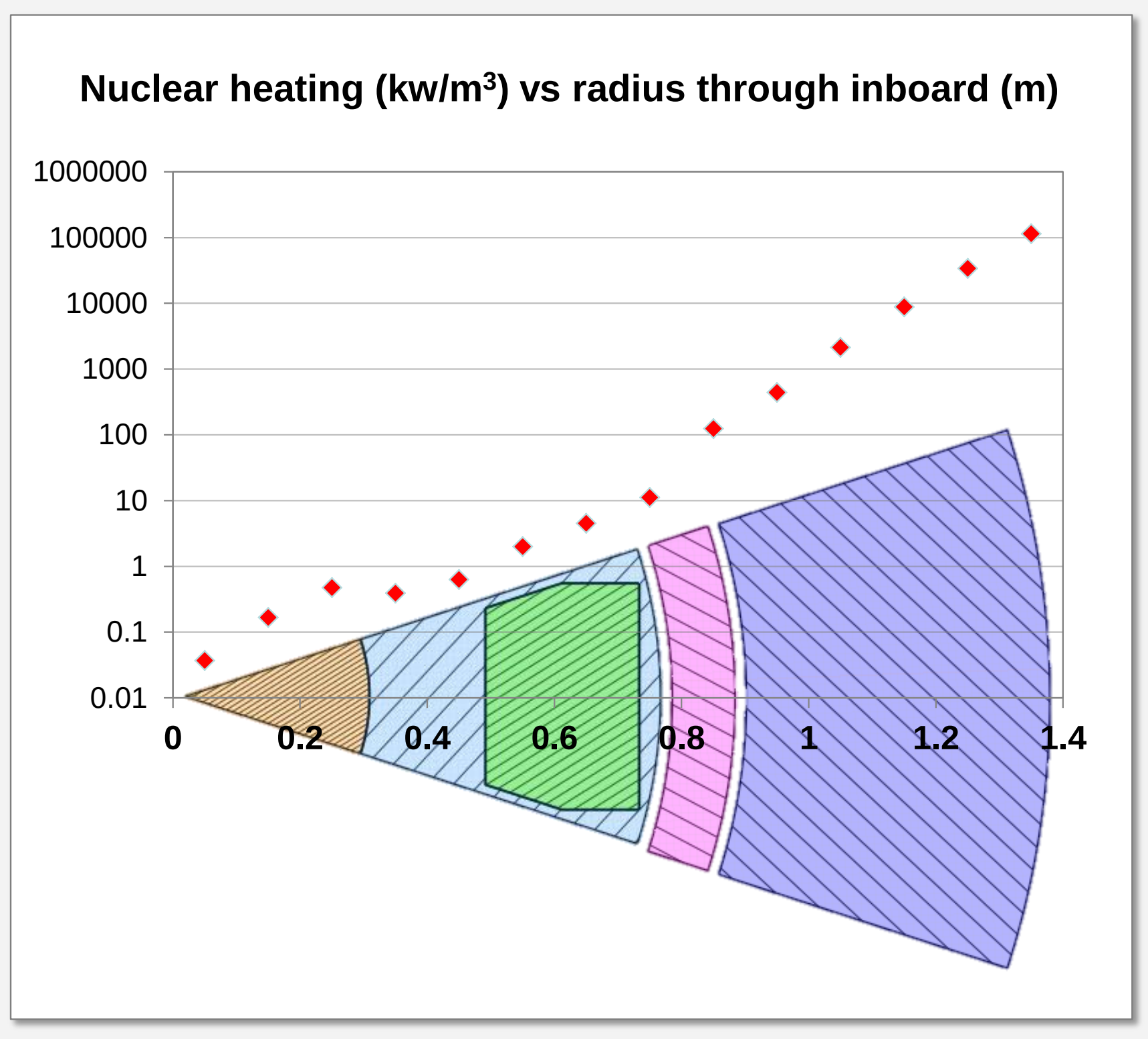
*Normalised to a 550 MW fusion power device

Inboard heating of the HTS-ST neutronics model



- ➢ Centre column consists of central solenoid (CC-CS), toroidal field (CC-TF) coil and steel casing .
- ➢ The centre column is separated from the plasma by the inboard shield and vacuum vessel structure (VV).

Peak nuclear heating in the CC-TF is approximately 5.6 kW/m^3 and 0.4 kW/m^3 in the CC-CS.


Nuclear heating (kw/m^3) vs radius through inboard (m)

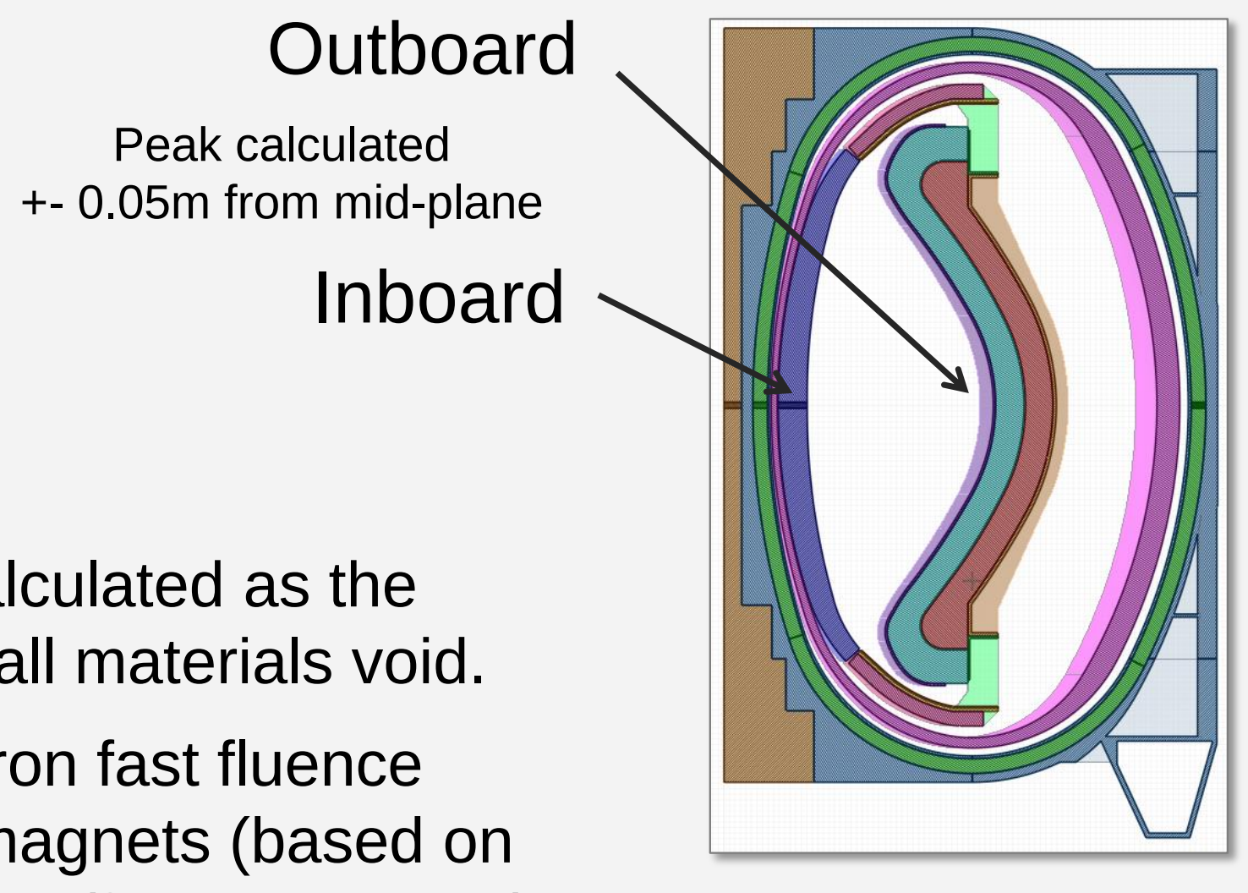
Model, Materials and Assumptions

- Neutronics model is based on **PPPL ST-FNSF radial build** and preliminary CAD model; modified using ANSYS SpaceClaim and converted to an MCNP input using MCAM [1].
  - ➢ **Major radius of 3 metres, aspect ratio of 2, fusion power 550 MW**
- Some **homogenisation of materials** to create a simple neutronics model for preliminary parameter studies.
  - ➢ **Blankets-** homogeneous mix of lithium orthosilicate (Li_4SiO_4) breeder and beryllium neutron multiplier pebbles with a packing fraction of 0.7, reduced activation ferritic martensitic (RAFM) steel, and helium coolant. No gaps or holes for ports are considered which could reduce the TBR considerably.
  - ➢ **Shield-** homogeneous mix of tungsten carbide, RAFM steel and water coolant.
  - ➢ **HTS magnets-** homogeneous mix of 57% REBCO, 38% hastelloy steel 38% and 5% helium.

Calculations have been performed using the radiation transport code MCNP (Monte Carlo N-Particle) [2] with the FENDL-2.1 nuclear data library.

Neutron Wall Loading (NWL) & Fast Flux (FF)

Peak inboard NWL	1.45 MW/m^2
Peak outboard NWL	1.77 MW/m^2
Machine average NWL	~1.29 MW/m^2
Peak CC-TF FF	3.8 x 10^{14} neutrons/s/m^2
Peak CC-CS FF	3.6 x 10^{13} neutrons/s/m^2



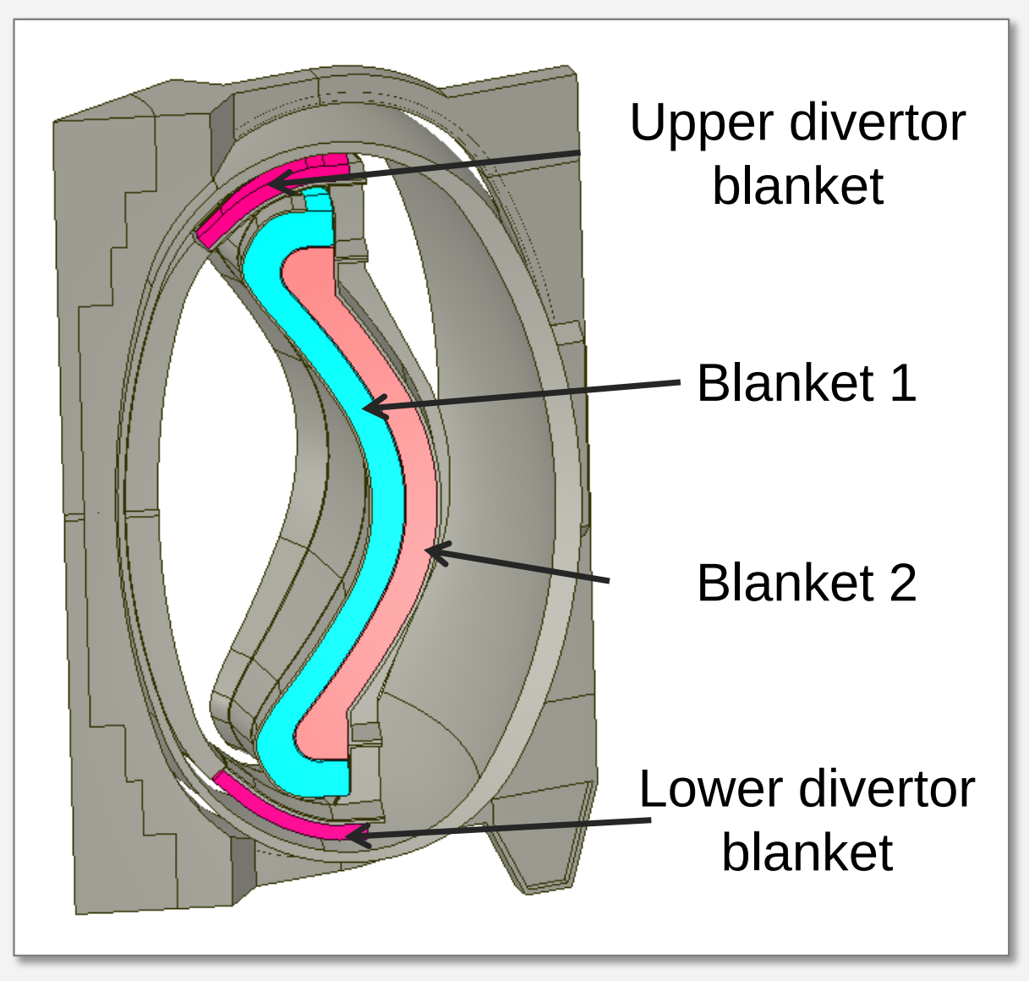
Peak calculated +- 0.05m from mid-plane

The HTS-ST provides a high NWL. This has been calculated as the uncollided neutrons using the F1 tally in MCNP with all materials void.

If the superconducting magnets are limited to a neutron fast fluence (E_n > 0.1MeV) of 3 x 10^{22} neutrons/m^2 then the TF magnets (based on CC-TF values) have a lifetime of approximately 2.5 fpy (full power year).

Tritium breeding ratio (TBR)

- Due to the geometry of ST's the breeder blanket is limited to the outboard (light blue and salmon pink regions).
- In the PPPL design, divertor blankets have also been considered (dark pink region).
- **TBR = 1.12** using the solid ceramic breeder Li_4SiO_4 , enriched with 40% lithium-6.

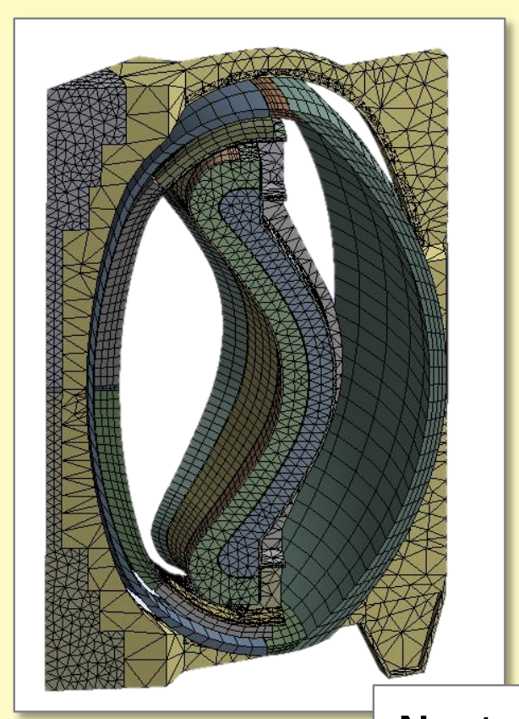


- For a self sustaining tritium supply **a tritium breeding ratio** of **>1.1** is required. The use of homogenised blankets has been shown [3] to over estimate TBR values by 2%.
- The **majority** (~97%) of the **TBR** contribution is from **blanket 1**, with blanket 2 contributing ~2% and ~1% from the divertor blankets. Of the total TBR, ~97% is from the **Li-6 reaction**.
- Further **optimisation** of the **enrichment** and quantity of neutron **multiplying material** (in this case beryllium) could be used to increase the TBR.

Ongoing Studies & Future Developments

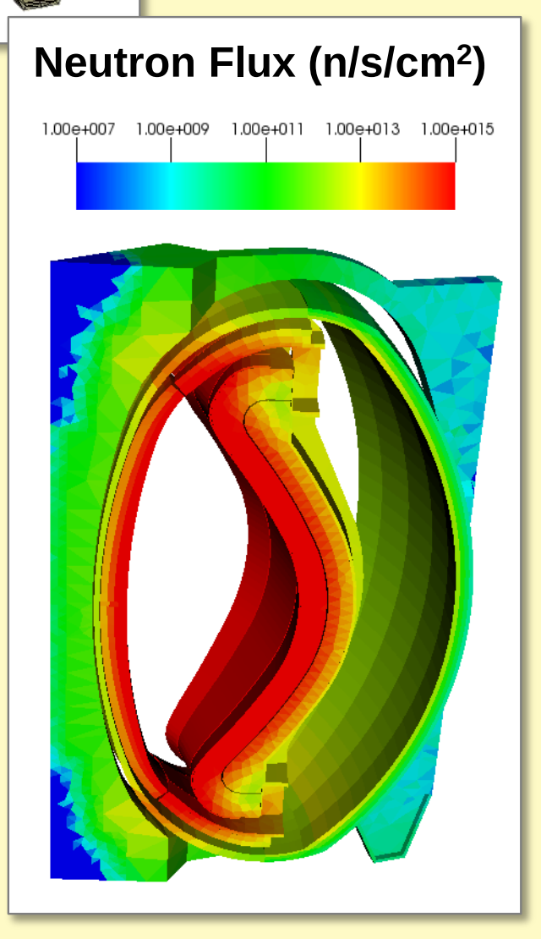
Ongoing simulations will investigate damage to the centre column and further tritium breeding optimisation.

Using neutron flux spectra (in 175 energy groups) the materials will be irradiated using the inventory code FISPACT-II to calculate **activation**, **decay heat** and **waste inventory** of the device.


The MCNP geometry is represented using an unstructured mesh created in ANSYS.

The use of **unstructured mesh** (a new feature of MCNP6) is also being investigated.

- ➢ Preliminary results show reasonably good agreement in the centre column heating cell tallies; peak nuclear heating in the CC-TF is approximately 5.7 kW/m^3 and 0.7 kW/m^3 in the CC-CS.
- ➢ Due to difficulties in using a weight window (WW) with the unstructured mesh, global variance reduction has not been used, requiring higher particle histories and resulting in an increase in statistical uncertainty; even higher particle histories are required and the use of WW with unstructured mesh will be investigated.


Neutron Flux (n/s/cm^2)

Acknowledgements & References

The author wishes to thank Princeton Plasma Physics Laboratory (PPPL) for the radial build and preliminary CAD of the PPPL-FNSF design.

[1] Y. Wu, FDS Team, CAD-based interface program for fusion neutron transport simulation, Fusion Engineering and Design 84 (2009) 1987-1992
[2] X-5 Monte Carlo Team, MCNP—A General Monte Carlo N-particle Transport.
[3] C.W. Lee, et al Sensitivity of the homogenized model in the neutronics analysis for the Korea Helium Cooled Solid Breeder Test Blanket Module, Fusion Eng. Des., 87 (2012), pp. 575–579

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