Design of the ITER First Wall and Blanket

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ITER Organization
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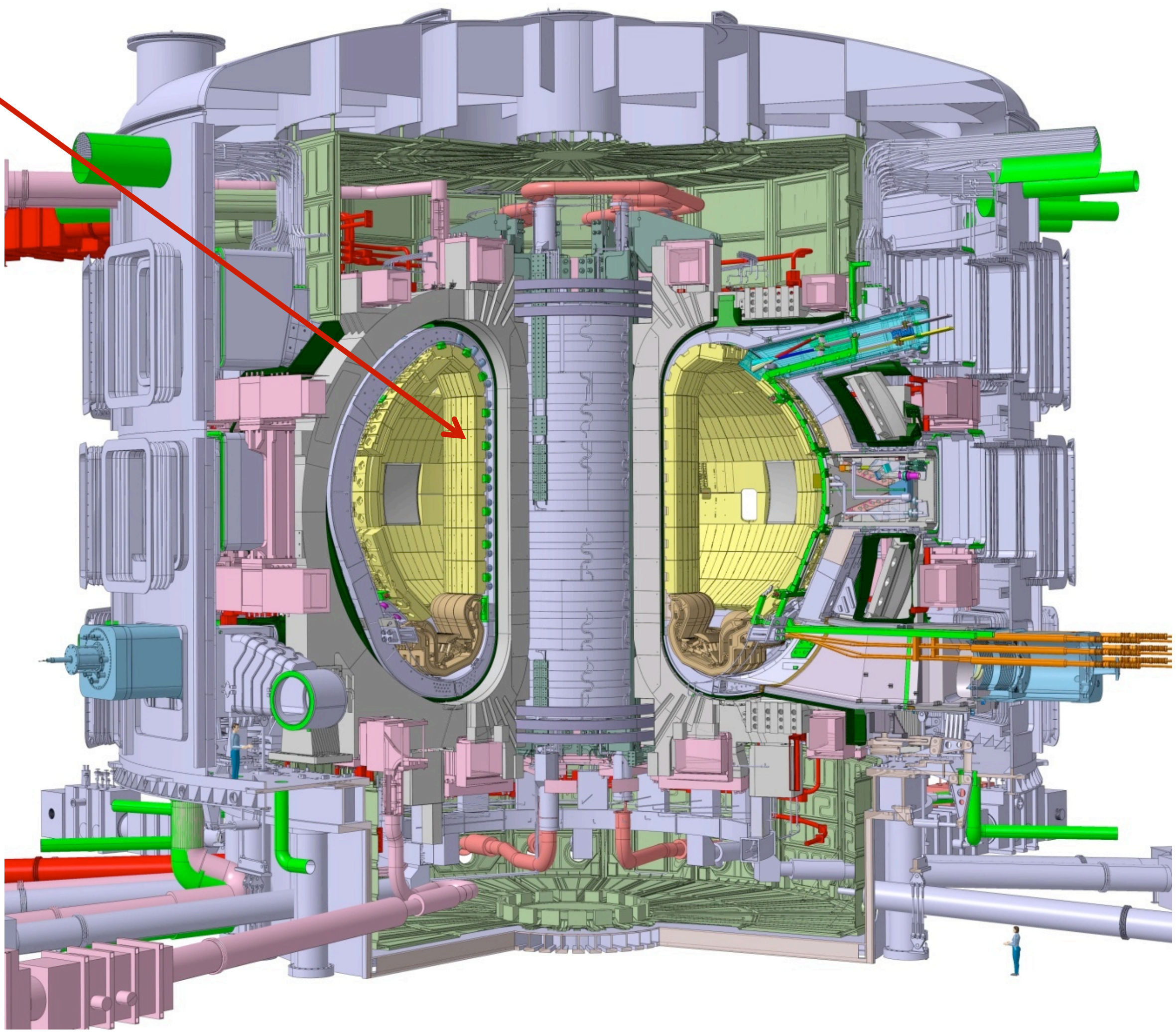
Blanket Effort Conducted within BIPT

- Include resources from DA's to help in major design and analysis effort
- Direct involvement of procuring DA's in design
  - Sense of design ownership
  - Would facilitate procurement

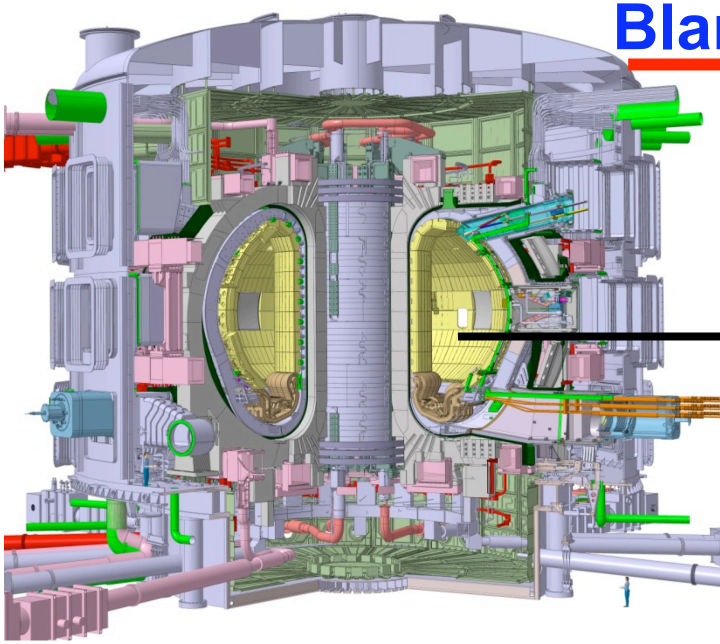
Blanket System Functions

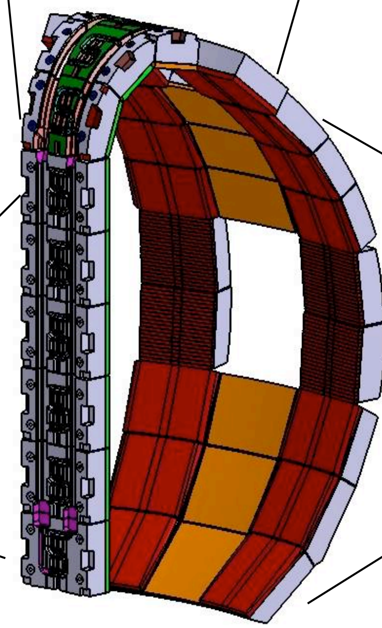
Main functions of ITER Blanket System:

- Exhaust the majority of the plasma power
- Contribute in providing neutron shielding to superconducting coils
- Provide limiting surfaces that define the plasma boundary during startup and shutdown.

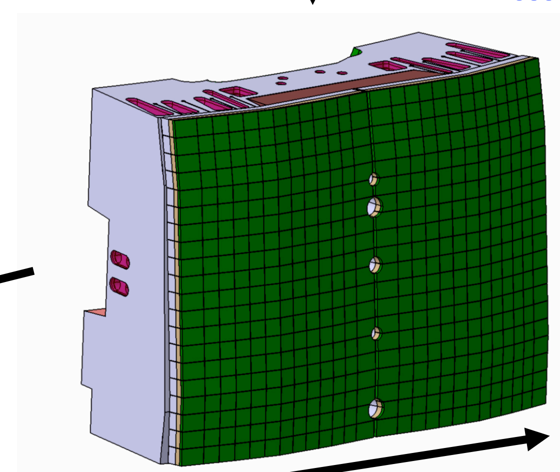


Blanket System



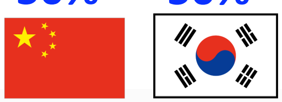


Blanket Module



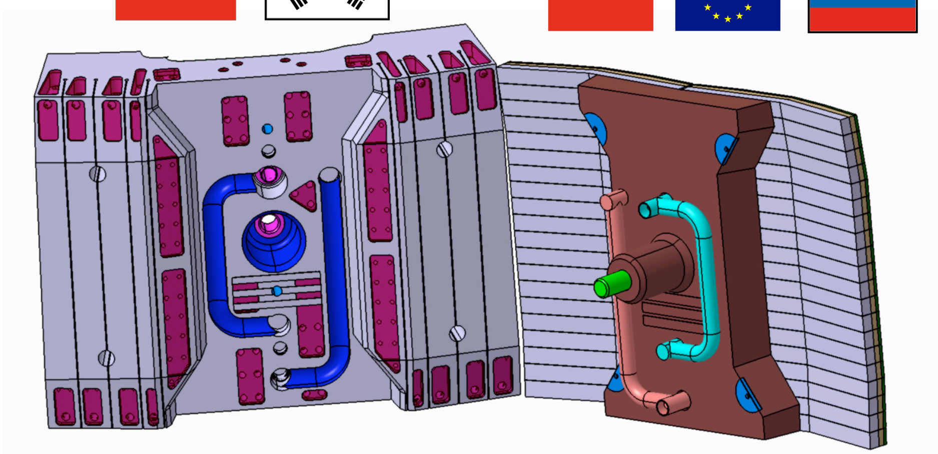
Shield Block (semi-permanent)

50% 50%



FW Panel (separable)

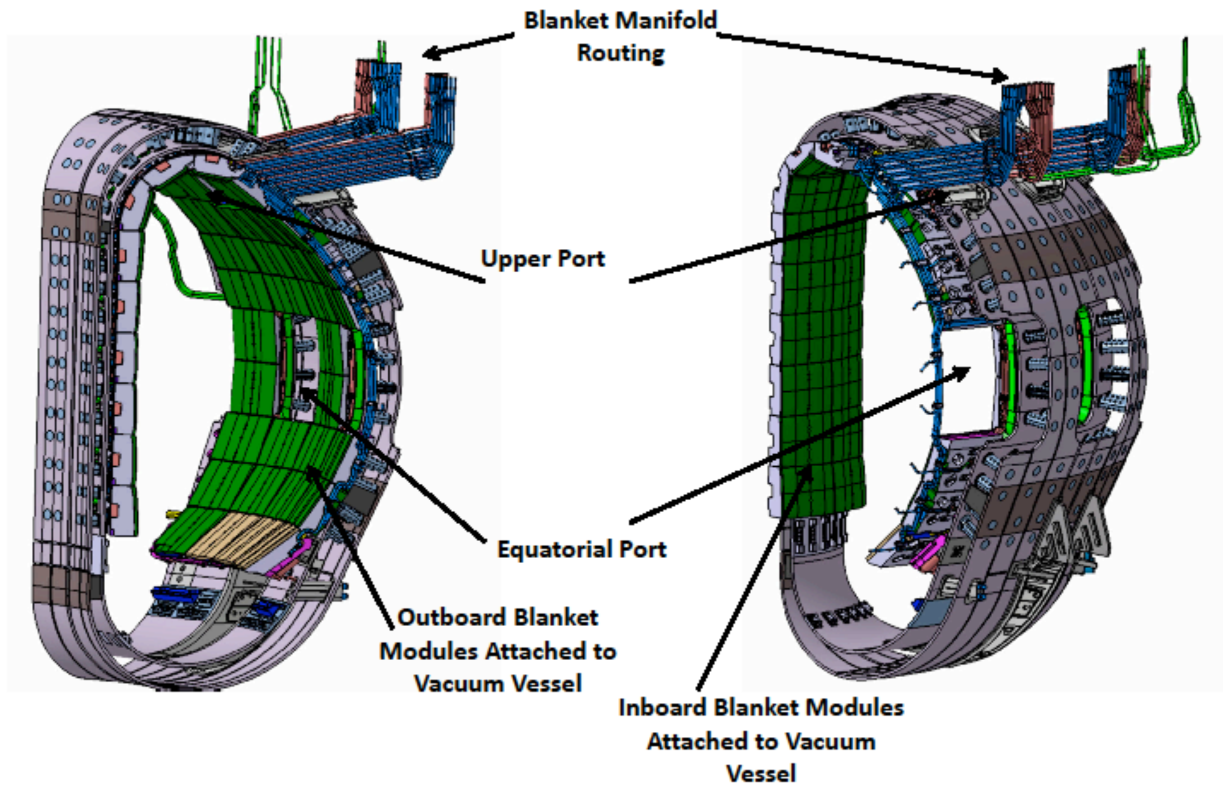
10% 50% 40%



Blanket System Layout

- The Blanket System consists of Blanket Modules (BM) comprising two major components: a plasma facing First Wall (FW) panel and a Shield Block (SB).

- It covers ~600 m^2



- Cooling water (3 MPa and 70°C) is supplied to the BM by manifolds supported off the vacuum vessel behind or to the side of the SB.

Blanket Design

- Major evolution since the ITER design review of 2007
  - Need to account for large plasma heat fluxes to the first wall
    - Replacement of port limiter by first wall poloidal limiters
    - Shaped first wall
  - Need for efficient maintenance of first wall components.
    - Full replacement of FW at least once over ITER lifetime
    - Remote Handling Class 1

- Design change presented at the Conceptual Design Review (CDR) in February 2010 and accepted in the ITER baseline in May 2010.

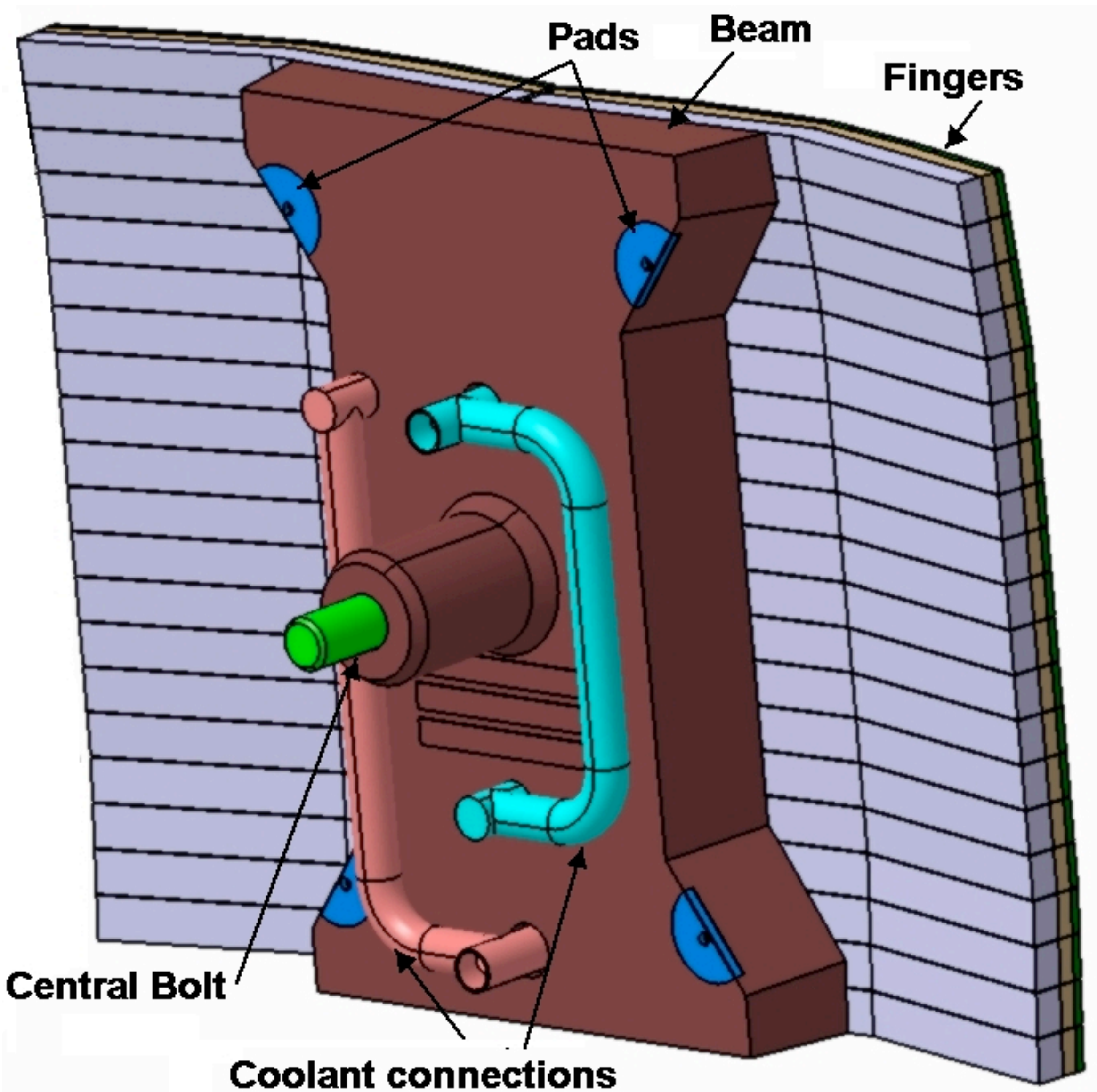
- Post-CDR effort focused on resolving key issues from CDR, particularly on improving the design of the first wall and shield block attachments to better accommodate the anticipated electromagnetic (EM) loads.

Blanket System in Numbers

Number of Blanket Modules:	440
Max allowable mass per module:	4.5 tons
Total Mass:	1530 tons
First Wall Coverage:	~600 m^2
Materials:	
- Armor:	Beryllium
- Heat Sink:	CuCrZr
- Steel Structure:	316L(N)-IG
n-damage (Be / heat sink / steel):	1.6 / 5.3 / 3.4 (FW) 2.3 (SB) dpa
Max total thermal load:	736 MW

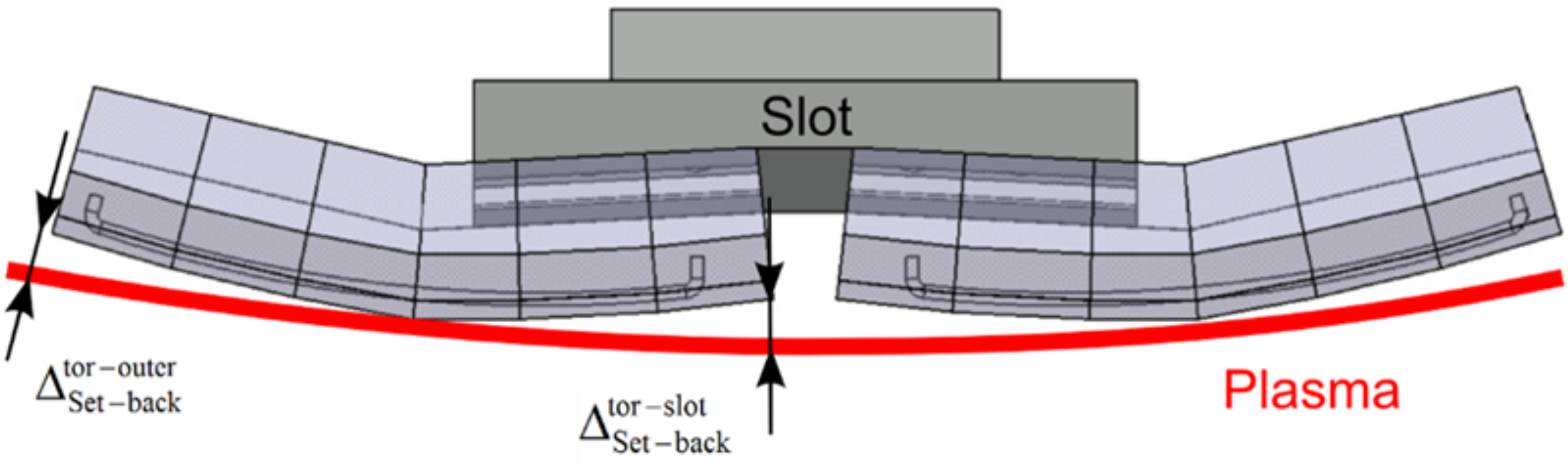
Design of First Wall Panel

I-shaped beam to accommodate poloidal torque

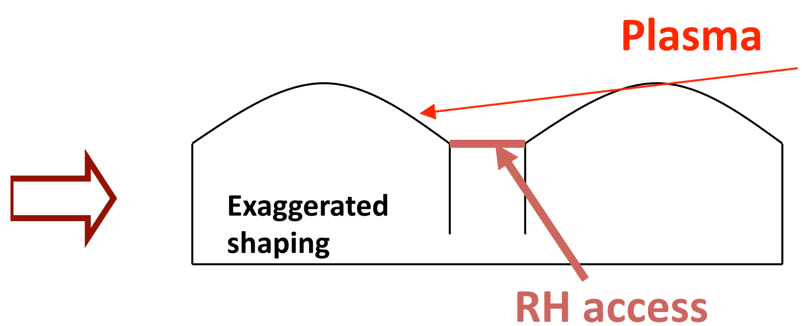


Basis for Shaping of First Wall Panel

- Heat load associated to charged particles is a major component of heat flux to first wall
- The heat flux is oriented along the field line
- Thus, the incident heat flux strongly design-dependent (incidence angle of the field line on the component surface)
- Shaping of FW to shadow leading edges and penetrations

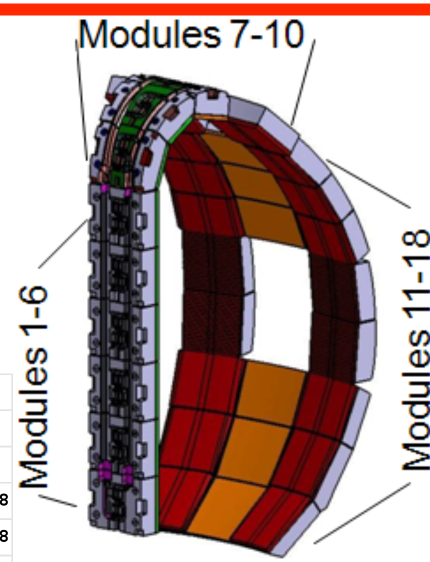


- Allow good access for RH
- Shadow leading edges



First Wall Panels: Design Heat Flux

0	176	<= 1.3 MW/m²	40%
1.4 MW/m² <=	42	<= 2.0 MW/m²	10%
3.0 MW/m² <=	162	<= 3.9 MW/m²	37%
4.0 MW/m² <=	60		14%
Total	440		100%



FWs distribution with design heat load

Poloidal Rows																																					
1		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0		1.0	18
2		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0	18
3		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8	18
4		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8	18
5		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8		3.8	18
6		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0		2.0	18
7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7	18
8		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7	18
9		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7		4.7	18
10	1.0	Diag	1.0	Diag	1.0	EC	1.0	EC	1.0	Diag	1.0	EC	1.0	EC	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	1.0	Diag	18
11	1.0	1.0	1.0	1.0	2.0	1.0	2.0	1.0	2.0	1.0	2.0	1.0	2.0	1.0	2.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	36
12	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	36
13	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	1.0	36
14	3.6	TBM	3.6	port	3.6	TBM	3.6	IC	3.6	EC	3.6	IC	3.6	Diag	3.6	Diag	3.6	Diag	3.6	Diag	3.6	port	3.6	0.5	3.6	0.5	3.6	0.5	3.6	0.5 DNB	3.6	Diag	3.6	TBM	3.6	Diag	22
15	4.0		4.0		4.0		3.6		3.6		3.6		3.6		3.6		3.6		3.6		3.6		3.6	0.5	3.6	3.6	0.5	3.6	0.5	3.6	0.5		3.6		3.6		22
16	4.0	3.6	4.0	3.6	4.0	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	36
17	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	3.6	36
18	1.3	1.3	1.3	3.6	1.3	1.3	1.3	1.3	3.6	1.3	1.3	1.3	1.3	1.3	1.3	3.6	1.3	1.3	1.3	1.3	3.6	1.3	1.3	1.3	1.3	1.3	1.3	3.6	1.3	1.3	1.3	1.3	3.6	1.3	1.3	1.3	36
	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	S	C	440
	18		17		16		15		14		13		12		11		10		9		8		7		6		5		4		3		2		1		

Toroidal Index

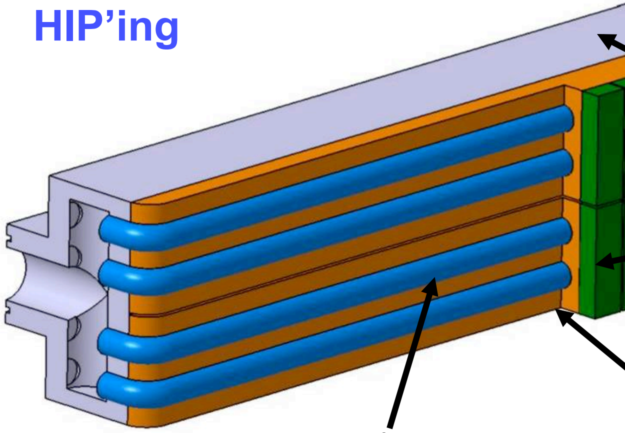
First Wall Finger Design

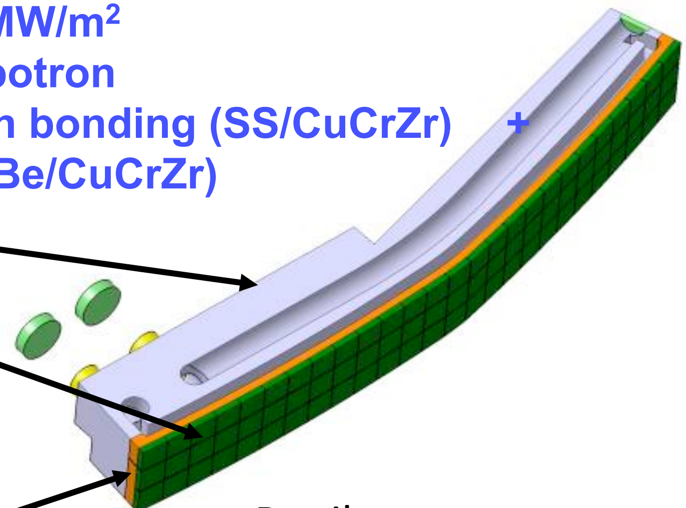
Normal Heat Flux Finger:

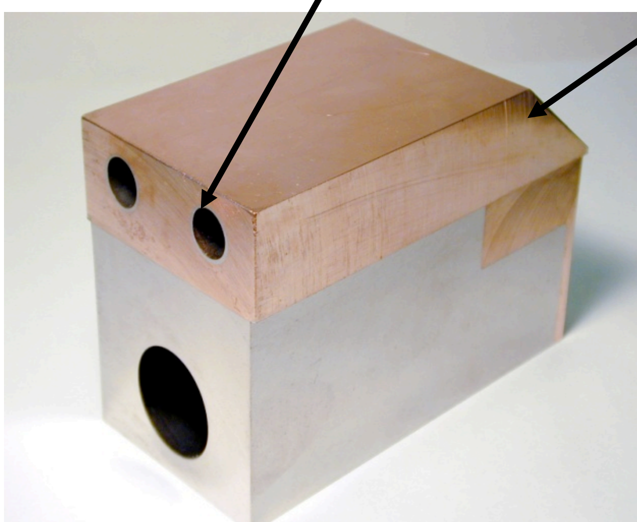
- q" = ~ 1-2 MW/m^2
- Steel Cooling Pipes
- HIP'ing

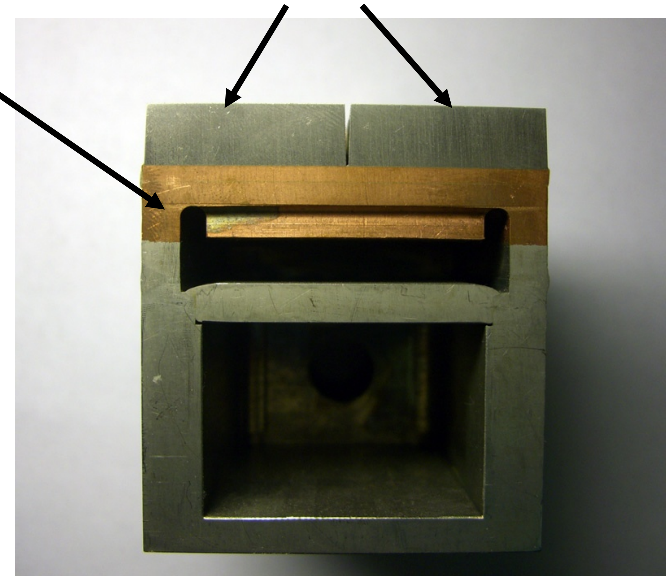
Enhanced Heat Flux Finger:

- q" < ~ 5 MW/m^2
- Hypervapotron
- Explosion bonding (SS/CuCrZr) + brazing (Be/CuCrZr)





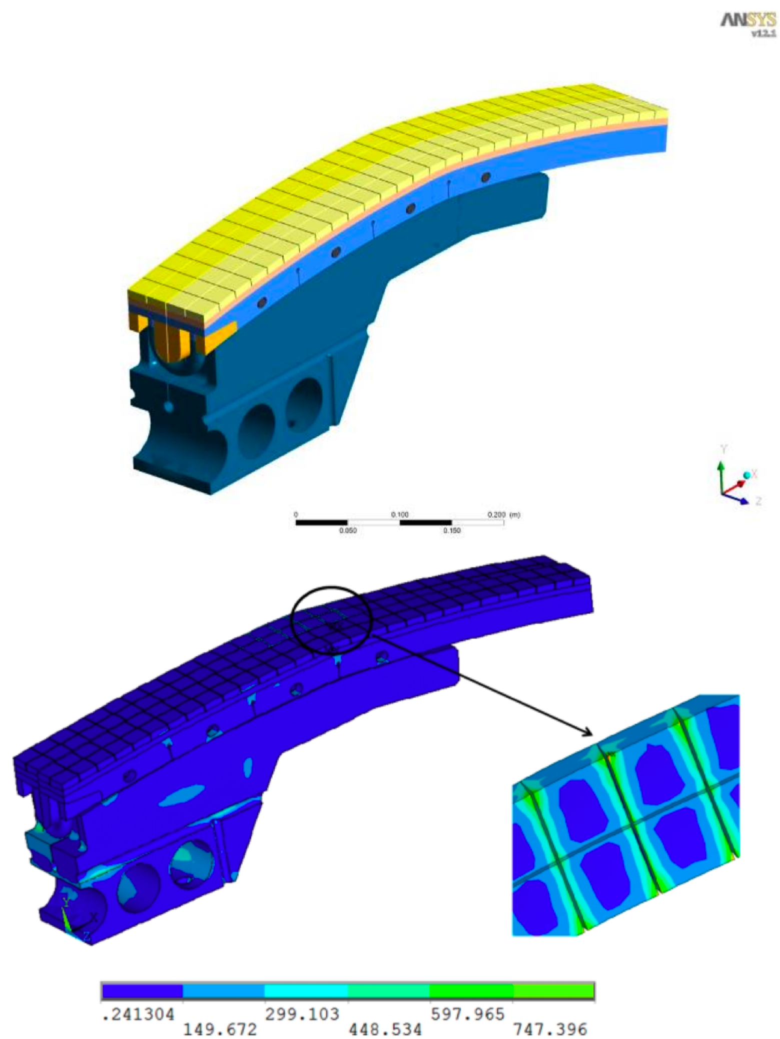




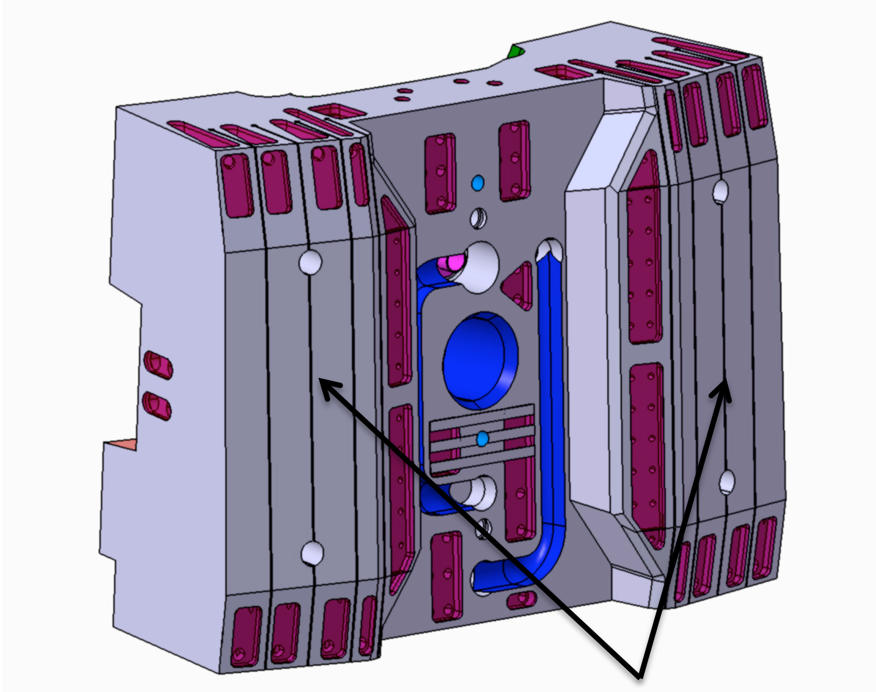
First Wall Analysis

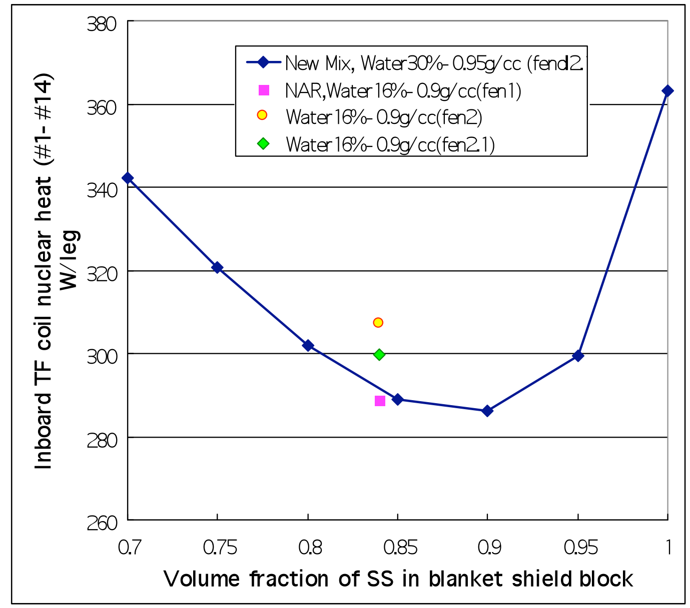
- Detailed blanket design activities on-going in parallel with supporting analyses in preparation for PDR.
- They address EM, thermo-hydraulic, thermal and structural aspects based on ITER Load Specifications and requirements.
- Design cases are categorized according to their probability of occurrence and allowable stress or temperature levels depend on the event category.
- The capability of the design to withstand the design number of cycles for each of the events must be demonstrated.

Schematic of EHF FW finger and Example Thermal Stress Results



Shield Block Design

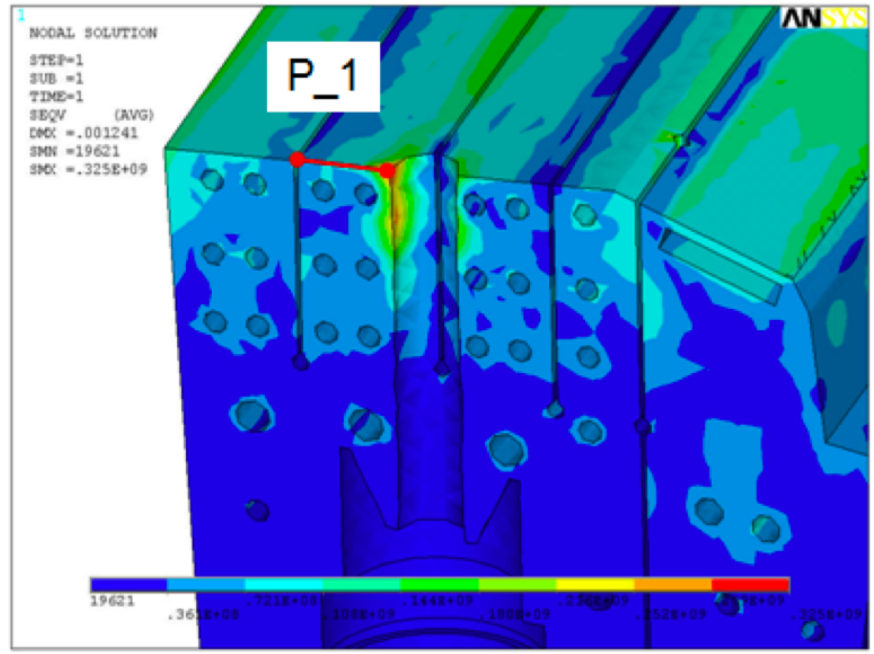




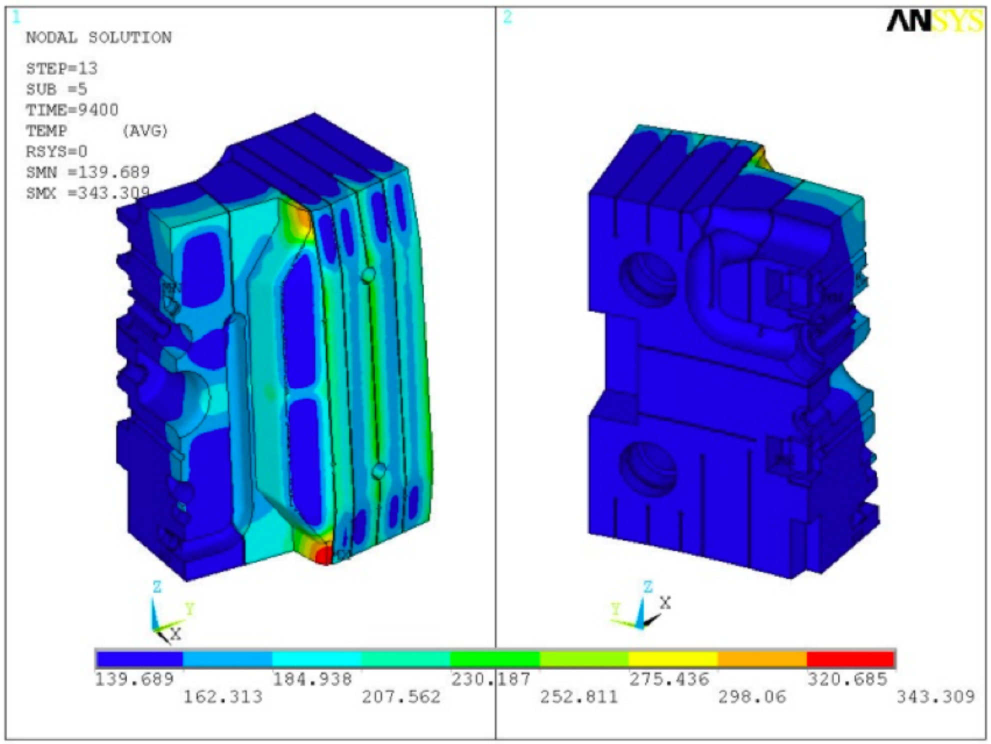
- **Slits to reduce EM loads and minimize thermal expansion and bowing.**
- **Poloidal coolant arrangement.**
- **Cooling holes are optimized for Water/SS ratio (Improving nuclear shielding performance).**
- **Cut-outs at the back to accommodate many interfaces (Manifod, Attachment, In-Vessel Coils).**
- **Basic fabrication method from either a single or multiple-forged steel blocks and includes drilling of holes, welding of cover plates of water headers, and final machining of the interfaces.**

Shield Block Analysis

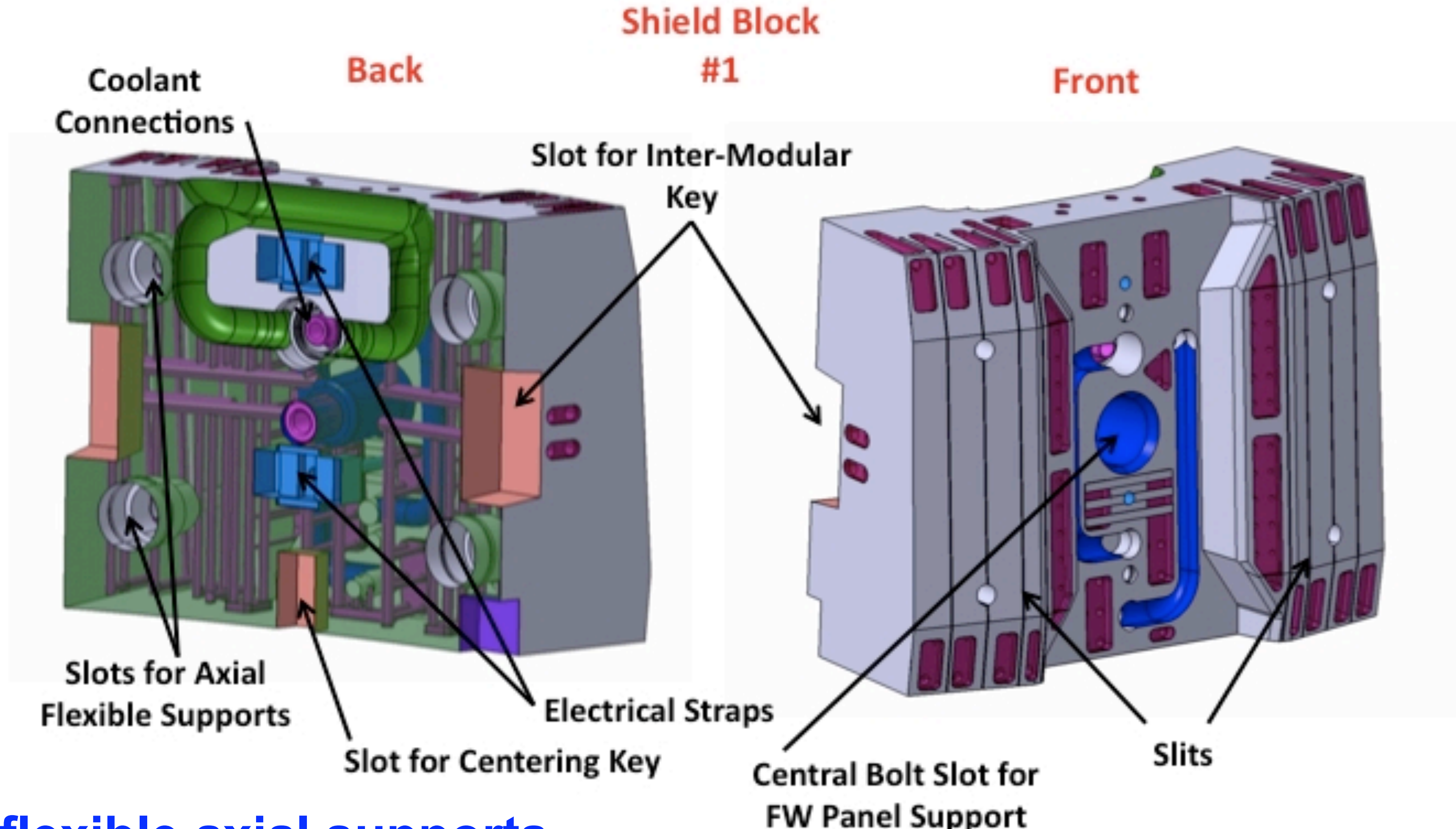
- **Thermo-mechanical analysis indicates that the stress levels are acceptable and that the temperature level <~350°C , as illustrated by the example results for SB 1 shown here.**



Path Location	Stress Category	Equivalent Stress (MPa)	Allowable Stress* (MPa)
P_1	P+Q	71	327
	P+Q+Bending	201	327
	P+Q+F	325	-

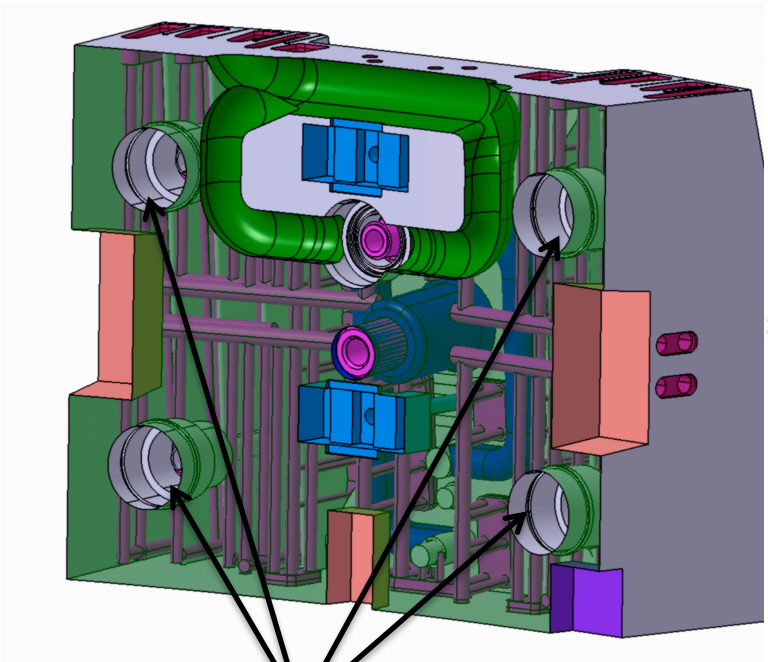


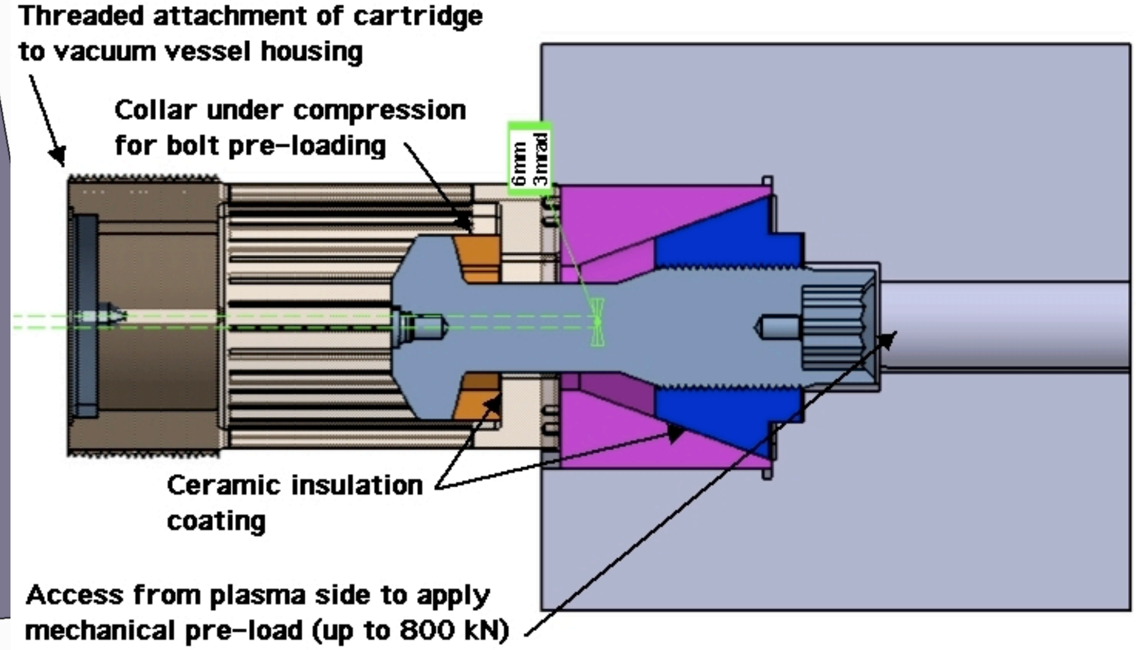
Shield Block Attachment



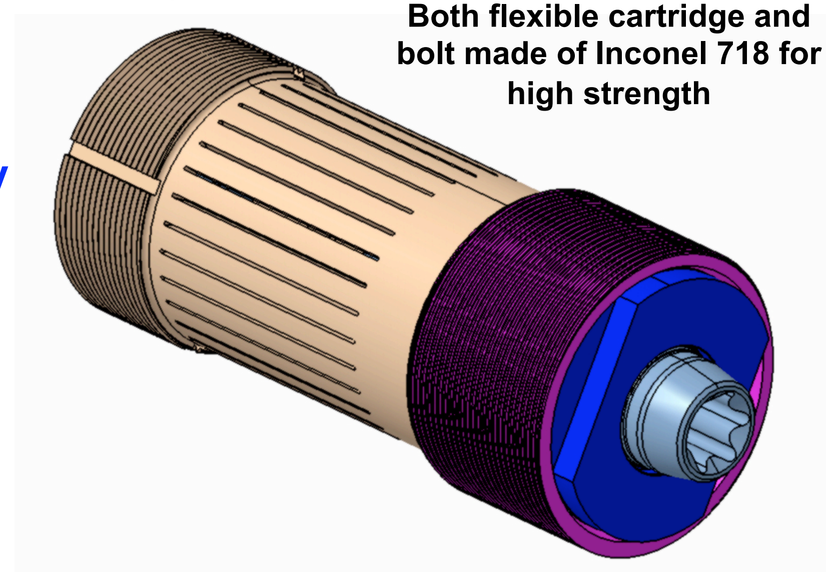
- 4 flexible axial supports
- Keys to take moments and forces
- Electrical straps to conduct current to vacuum vessel
- Coolant connections

Flexible Axial Support

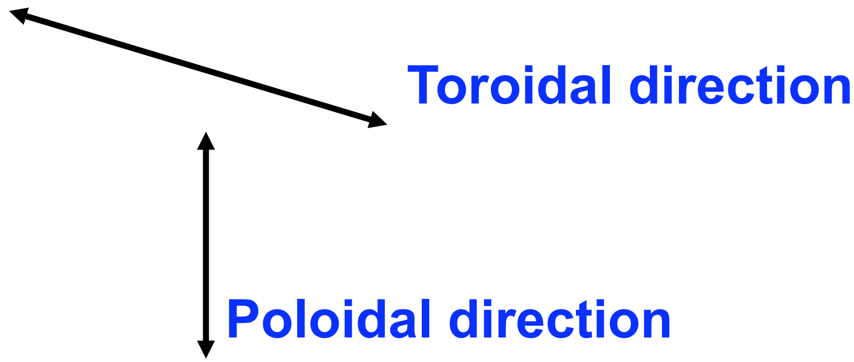


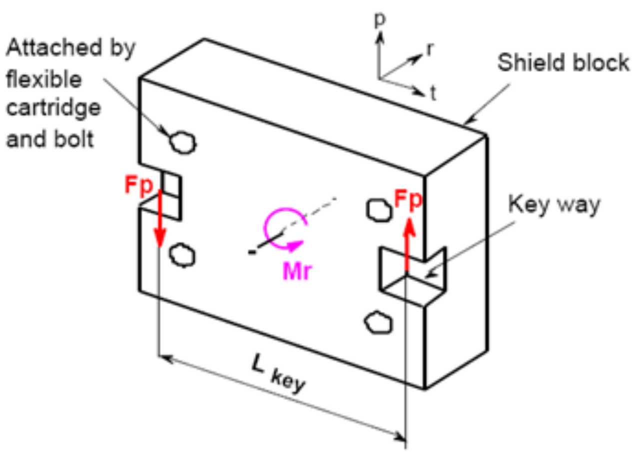


- **4 flexible axial supports located at the rear of SB, where nuclear irradiation is lower.**
- **Compensate radial positioning of SB on VV wall by means of custom machining.**
- **Adjustment of up to ±10 mm in the axial direction and ±5 mm transversely (on key pads) built into design of the supports for custom-machining process.**
- **Designed for 800 kN preload and to accommodate up to 600 kN Category III radial load.**

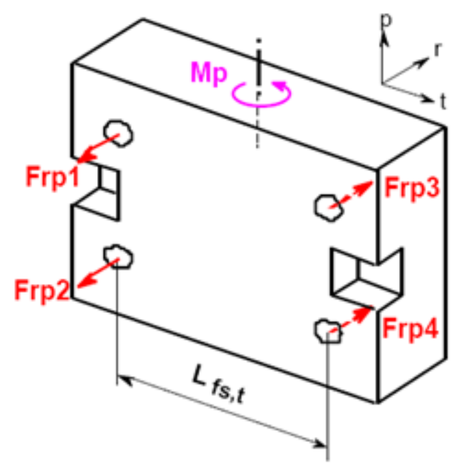


Shear Keys Used to Accommodate Moments from EM Loads

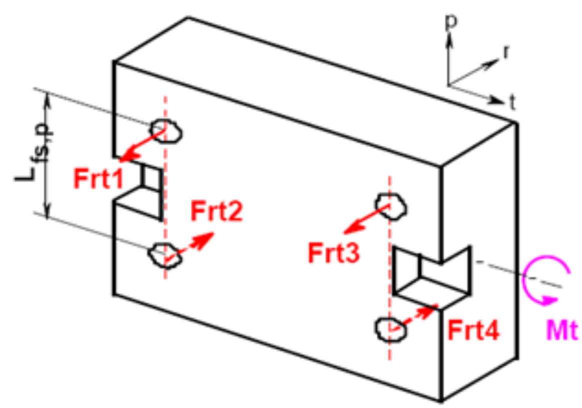




Reaction scheme to Radial moments



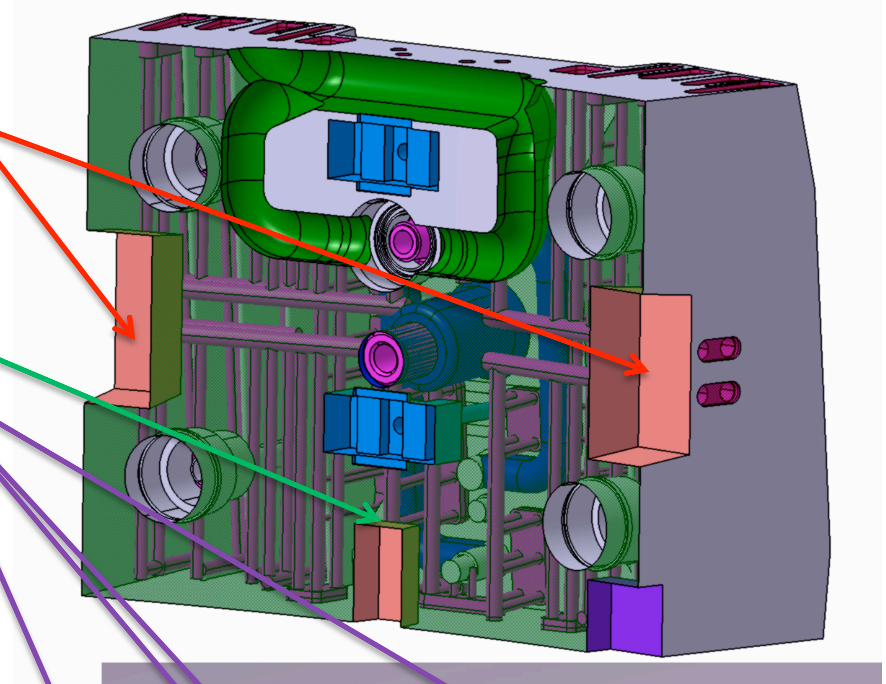
Reaction scheme to Poloidal moments

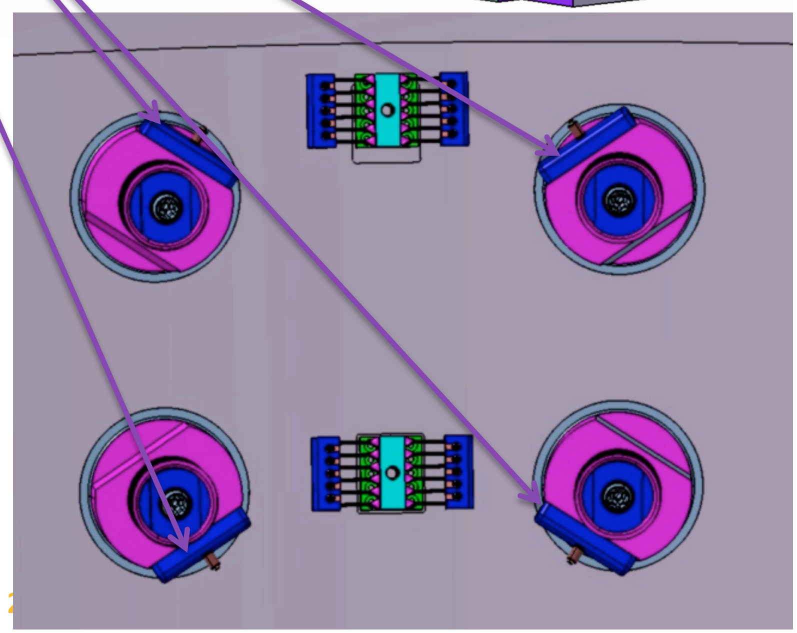


Reaction scheme to Toroidal moments

Keys in Inboard and Outboard Modules

- Each inboard SB has two inter-modular keys and a centering key to react the toroidal forces.
- Each outboard SB has four stub keys concentric with the flexible supports
- Bronze pads are attached to the SB and allow sliding of the module interfaces during relative thermal expansion.
- Key pads are custom-machined to recover manufacturing tolerances of the VV and SB.
- Electrical isolation of the pads through insulating ceramic coating on their internal surfaces (must withstand normal and lateral loads and any lateral displacement – R&D required and planned).

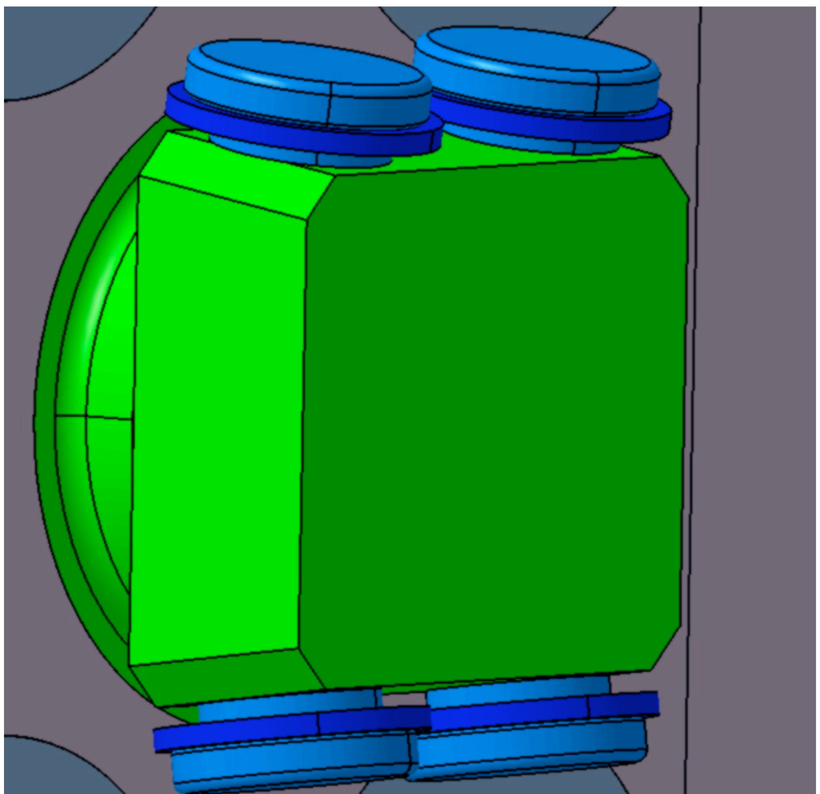


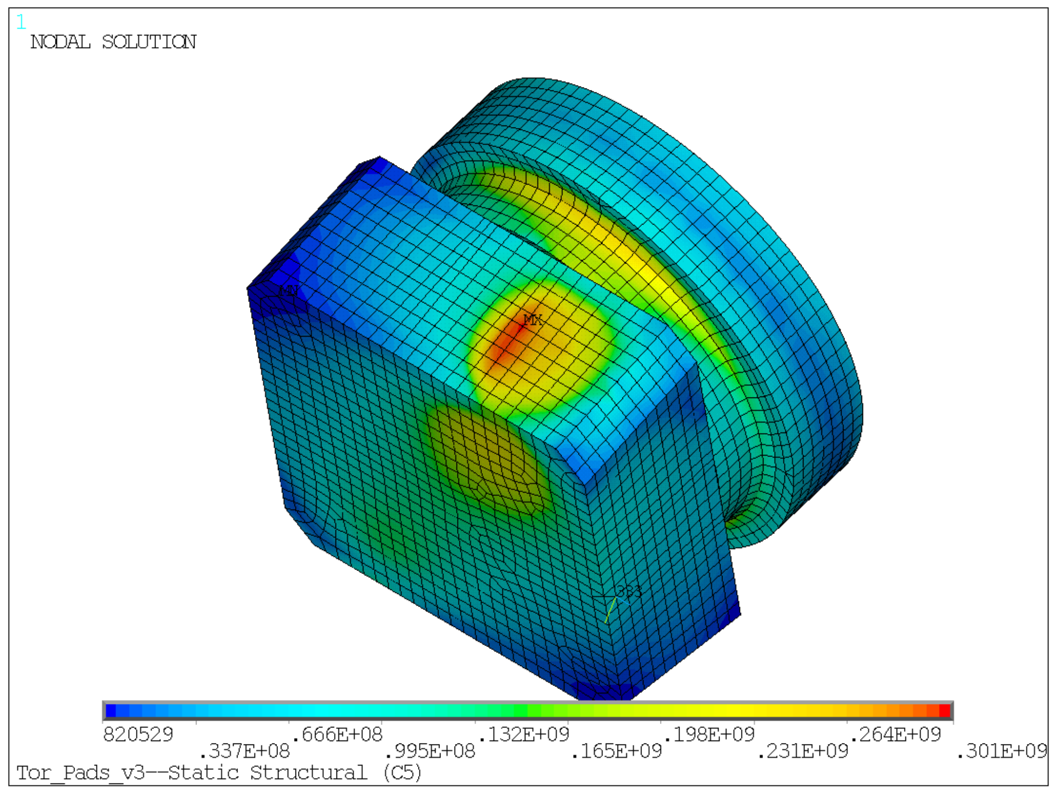


Example Analysis of Inter-Modular Key

- Analysis of the inter-modular keys indicate stresses above yield (~172 MPa at 100°C) in the case of Category III load.

- Limit analysis then performed to check margin.

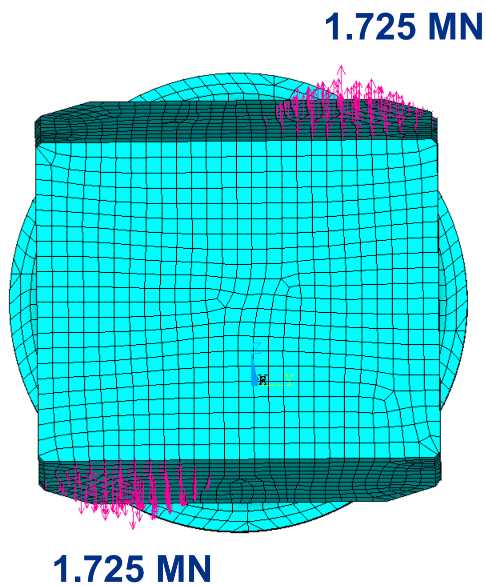


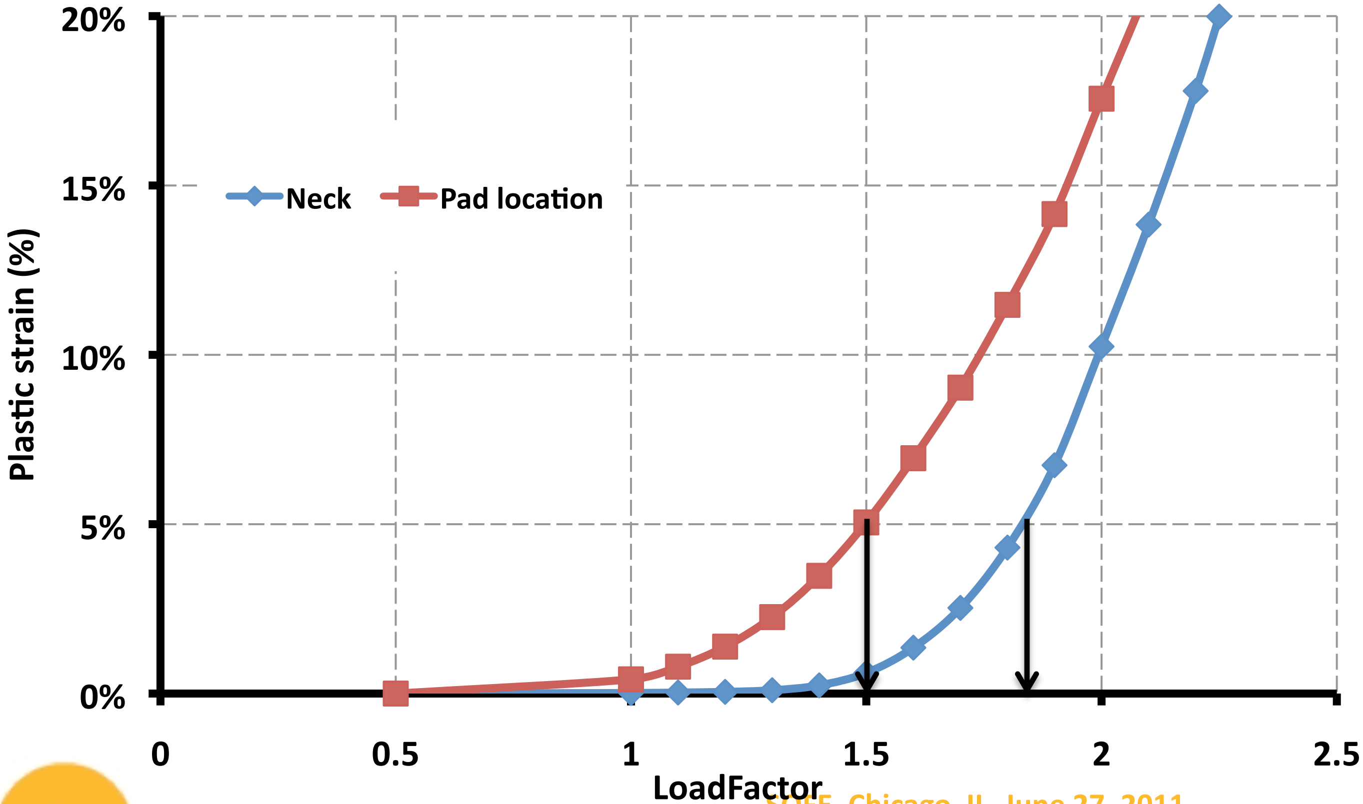


Limit Analysis of Inter-Modular Key

- **Reasonable load factors of 1.5 for the pads and 1.9 for the neck of the key are obtained based on limit analysis under Category III load with 5% plastic strain.**

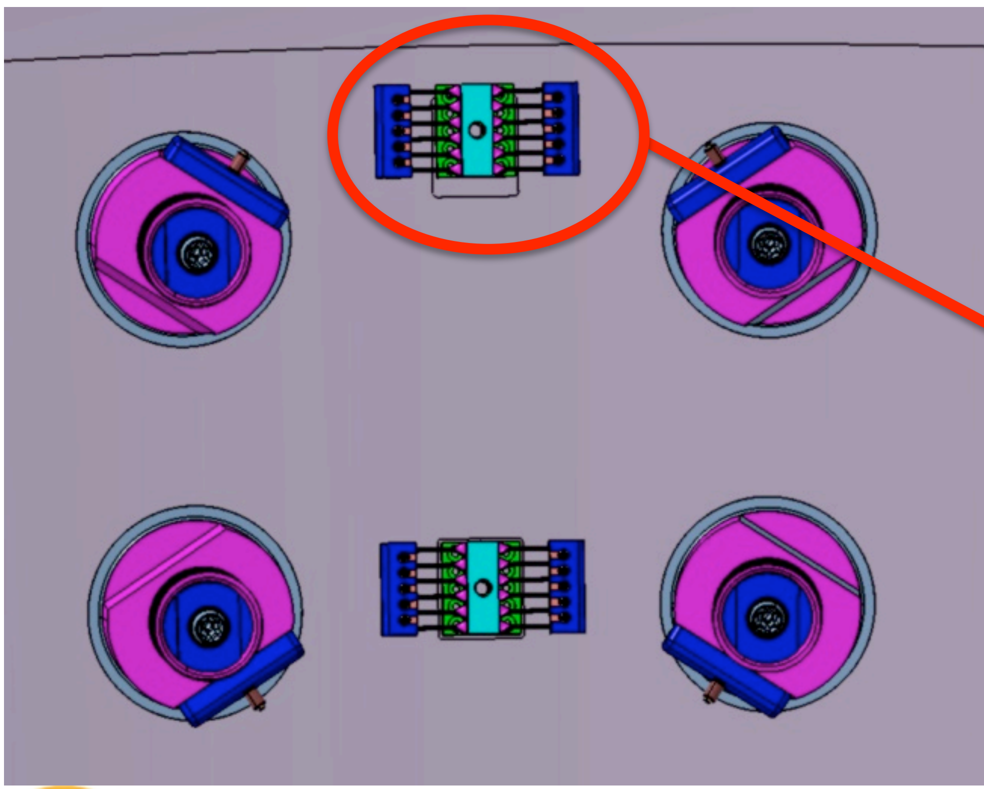
Eddy Forces Applied

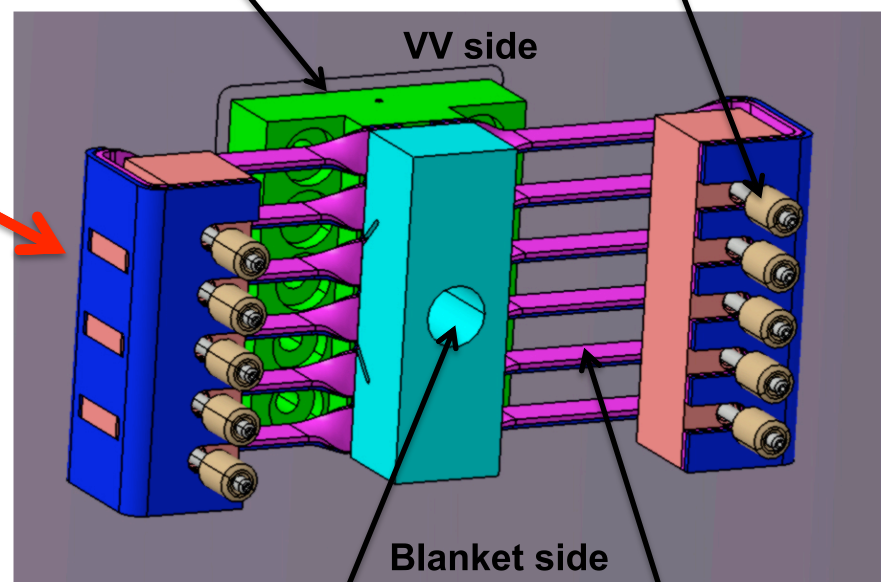




Electrical Straps

- Each SB is electrically joined to the VV by two electrical straps, formed and louvered from two sheets of CuCrZr alloy to achieve flexibility.
- Each strap is bolted on to the rear of a SB using M8 bolts. The socket is welded to the VV.
- An M20 bolt inserted through the front face of the SB connects the straps to the vessel socket via a compression block.
- One electrical connection can handle up to 180 kA of electrical current.
- Cu plating on VV and blanket side for lower contact resistance (to reduce local joule heating and resulting high temperature).





Supporting R&D

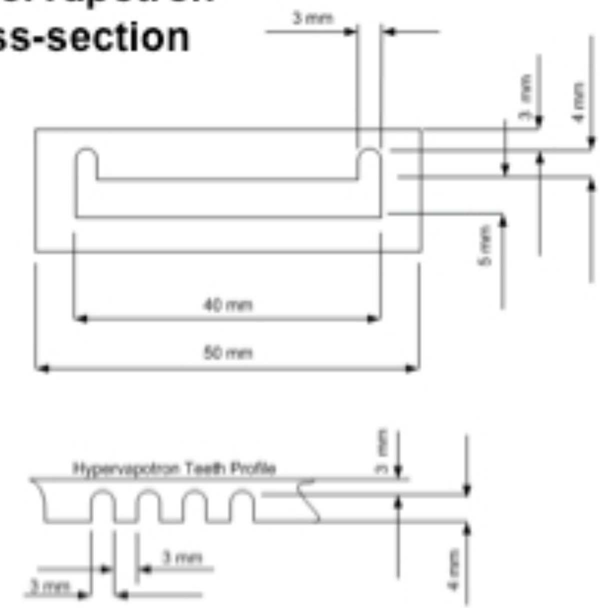
- A detailed R&D program has been planned in support of the design, covering a range of key topics, including:
  - Critical heat flux (CHF) tests on FW mock-ups
  - Experimental determination of the behavior of the attachment and insulating layer under prototypical conditions
  - Material testing under irradiation
  - Demonstration of the different remote handling procedures
- A major goal of the R&D effort is to converge on a qualification program for the SB and FW panels
  - Full-scale SB prototypes (KODA and CNDA)
  - FW semi-prototypes (EUDA for the NHF FW Panels, and RFDA and CNDA for the EHF First Wall Panels).
    - The primary objective of the qualification program is to demonstrate that:
      - Supplying DA can provide FW and SB components of acceptable quality.
      - Components are capable of successfully passing the formal test program including heat flux tests in the case of the FW panel.

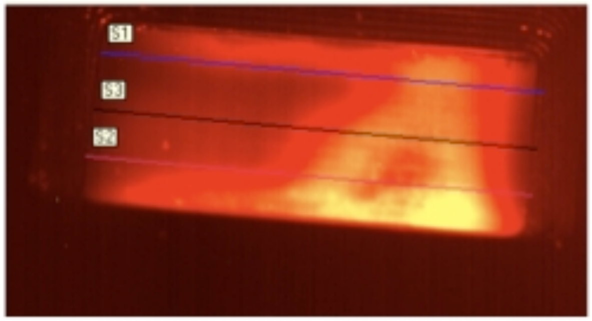
Example R&D for Hypervapotron CHF

- The R&D program in support of the EHF hypervapotron CHF testing was conducted at the Efremov Institute, RF.

- The results confirm the CHF margin of 1.4 for the EHF FW under an incident heat flux of 5 MW/m^2.

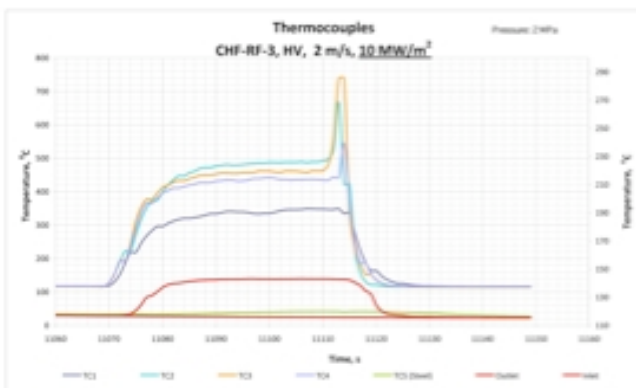
Hypervapotron cross-section



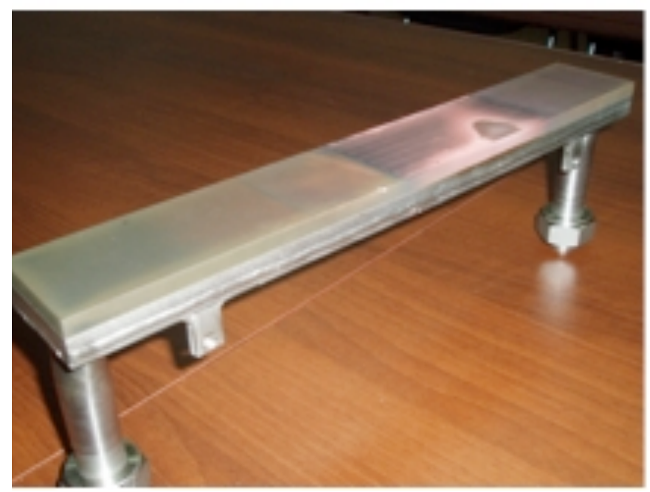


IR image showing a “hot spot”, maximum limit of the heat sink before failure

Heat sink mock up

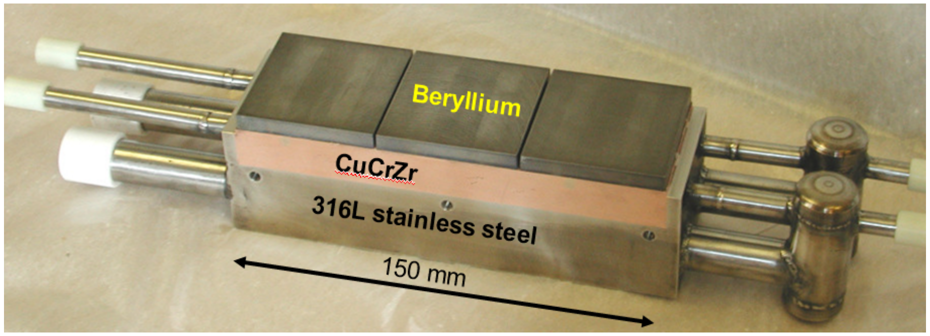


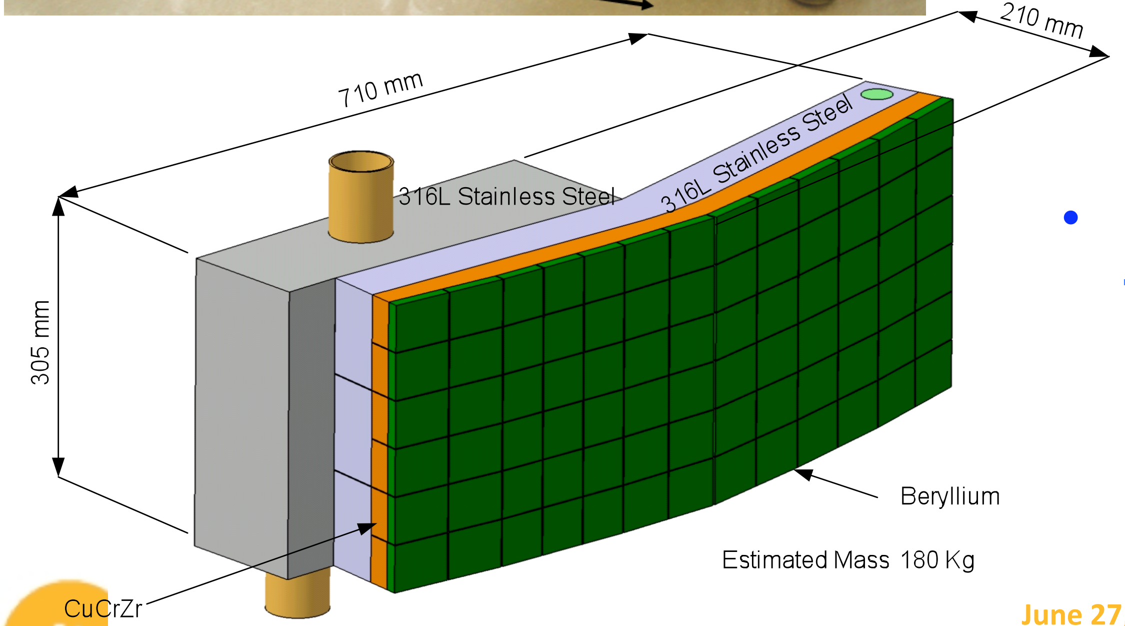
Example results on HV CHF testing in the TSEFEY facility



Example of Qualification Requirements for NHF First Wall

- **DA must demonstrate technical capability prior to start of procurement**
  - **2-phase approach**





- **Pre-qualification small-scale mock-up fabrication and testing**
  - **Demonstration/validation of Be/CuCrZr and SS/CuCrZr joints**

- **Semi-prototype**
  - **Production/validation of large scale components (fabrication and formal test program for acceptance)**

Summary

- The Blanket design is extremely challenging, having to accommodate high heat fluxes from the plasma, large EM loads during off-normal events and demanding interfaces with many key components (in particular the VV and IVC) and the plasma.

- Substantial re-design following the ITER Design Review of 2007. The Blanket CDR in February 2009 has confirmed the correctness of this re-design.

- Effort now focused on finalizing the design work.

- In parallel to that, R&D program and formal qualification process by the manufacturing and testing of full-scale semi-prototypes.

  - Preliminary Design Review in late 2011
  - Final Design Review in late 2012.
  - Procurement to start in early 2013 and should last till 2019.

Disclaimer

The views and opinions expressed herein do not necessarily reflect those of the ITER Organization