Nuclear Analysis at UW in Support ITER Design

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USA

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US Neutronics Support for ITER

- DCLL TBM
- FW/Shield modules
- In-vessel coils (ELM, VS)
- Diagnostics (PPPL, UCLA)
- Code development (DAG-MCNP)
- Data development (FENDL-3 validation)

Background

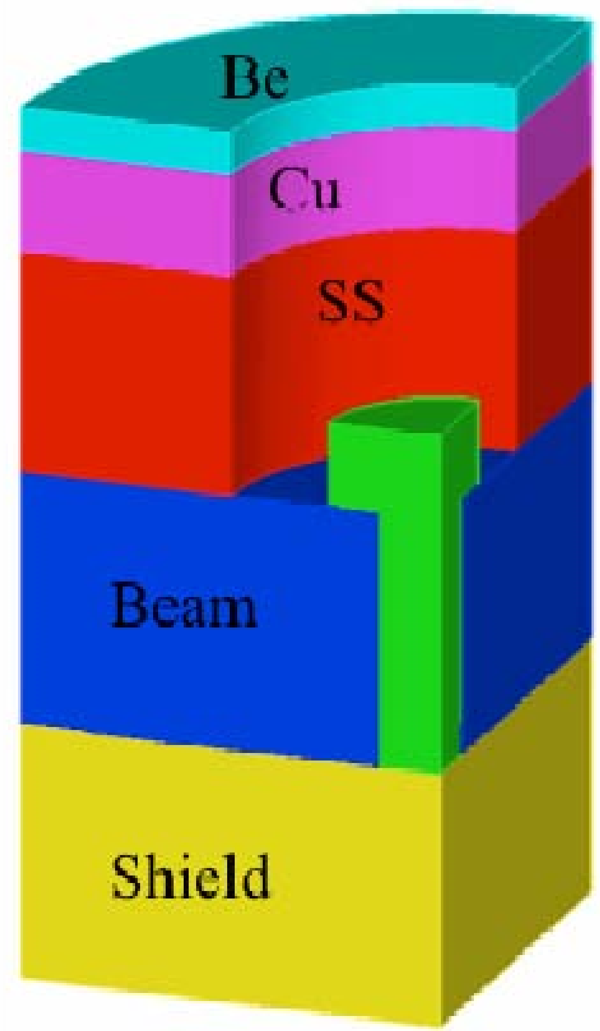
- Design of FW/shield modules of ITER has been going through several changes with different design options considered
- Nuclear analysis performed to guide the design evolution to a new baseline
- We summarize here analysis of various options for shield modules completed over past 18-24 months
- Design variations in shield and VV were assessed aiming at reducing total TFC nuclear heating to goal value of 14 kW
- More details on analyses performed are available

Outline

- Heating and damage in bolts used to secure the FW to the shield module
- Heating of the FW support beam due to a slot in FW
- Heating in the FW attachment mechanism with hinge, knuckle, yoke, pin, bolt
- Optimization of water content in shield and VV
- Shielding impact of water distribution and channel configuration in shield module
- Impact of Flexible Joints on Inboard TF Magnet Shielding
- Effect of FW shaping and added IB shield thickness on magnet heating
- Assessment of adding tungsten in IB shield modules
- Changing material in IB VV to reduce magnet heating

Nuclear Analysis for FW/Shield Attachment Options

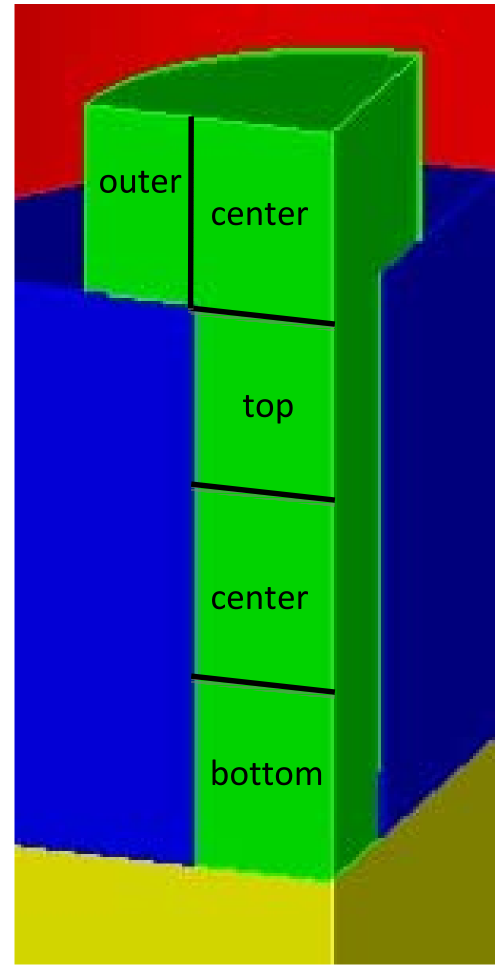
- Performed neutronics calculations to determine nuclear heating and radiation damage in bolts used to attach FW to shield module
- Three diameters considered for opening around bolt: 60, 80, 100 mm
- Used PARTISN discrete ordinates code in R-Z with S_{12} and ray tracing first collision source to mitigate ray effects
- Results normalized to 0.567 MW/m^2 at module 4 with contribution from OB modules included
- Bolt is made of SS316LN-IG
- The option of using Inconel-718 bolts was assessed



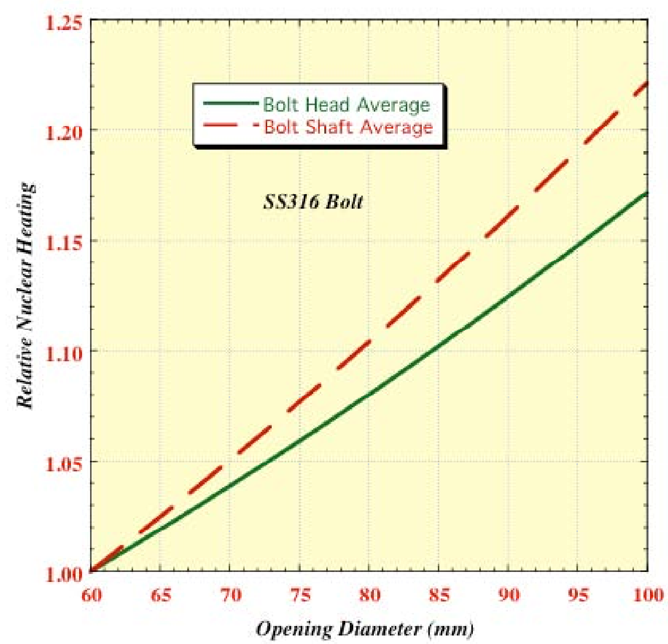
Nuclear Heating in SS316 FW Bolts

		Nuclear Heating (W/cm³)		
		60 mm	80 mm	100 mm
Bolt Head	Center	4.24	4.57	4.97
	Outer	4.26	4.60	4.99
	Average	4.25	4.59	4.98
Bolt Shaft	Top	3.47	3.84	4.26
	Center	2.85	3.15	3.47
	Bottom	2.35	2.60	2.88
	Average	2.89	3.19	3.53

- Increasing opening diameter from 60 to 100 mm enhances heating in bolt head by 17%



- Impact of increasing opening size is larger on bolt shaft heating (22%) due to enhanced contribution from streaming neutrons incident on exposed area of beam steel plate around bolt



Nuclear Parameters in Inconel-718 vs. SS FW Bolts

		Nuclear Heating (W/cm³)		
		60 mm	80 mm	100 mm
Bolt Head	Center	4.53	4.91	5.37
	Outer	4.58	4.97	5.44
	Average	4.56	4.95	5.41
Bolt Shaft	Top	3.65	4.05	4.52
	Center	3.01	3.33	3.70
	Bottom	2.47	2.73	3.04
	Average	3.04	3.37	3.75

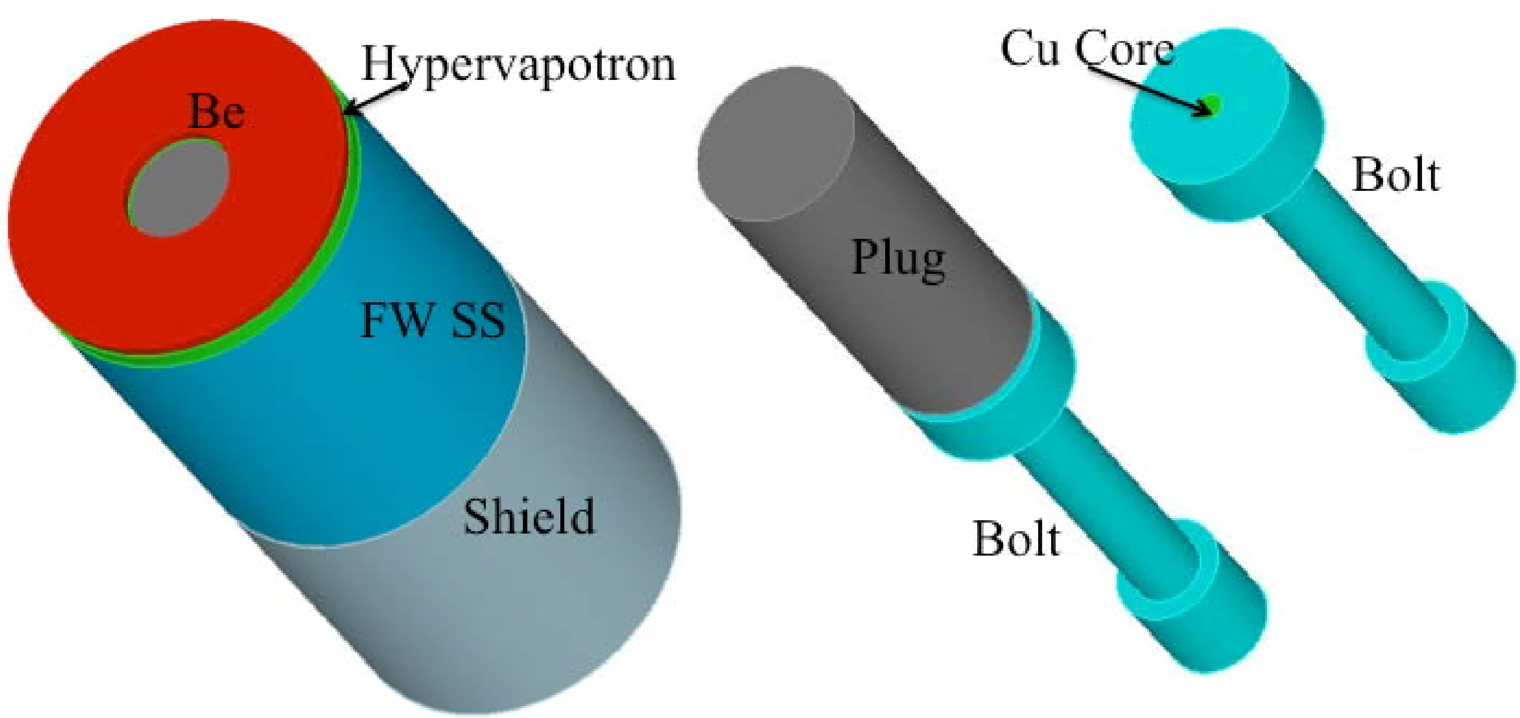
- Nuclear heating in Inconel is 5-10% higher than in SS
- Increasing opening diameter from 60 to 100 mm enhances heating in bolt head by ~19% and in bolt shaft by ~23%

	Peak End-of-life Radiation Damage (dpa @ 0.3 MWa/m²)		
	60 mm	80 mm	100 mm
SS316	1.20	1.41	1.65
Inconel-718	1.27	1.47	1.73

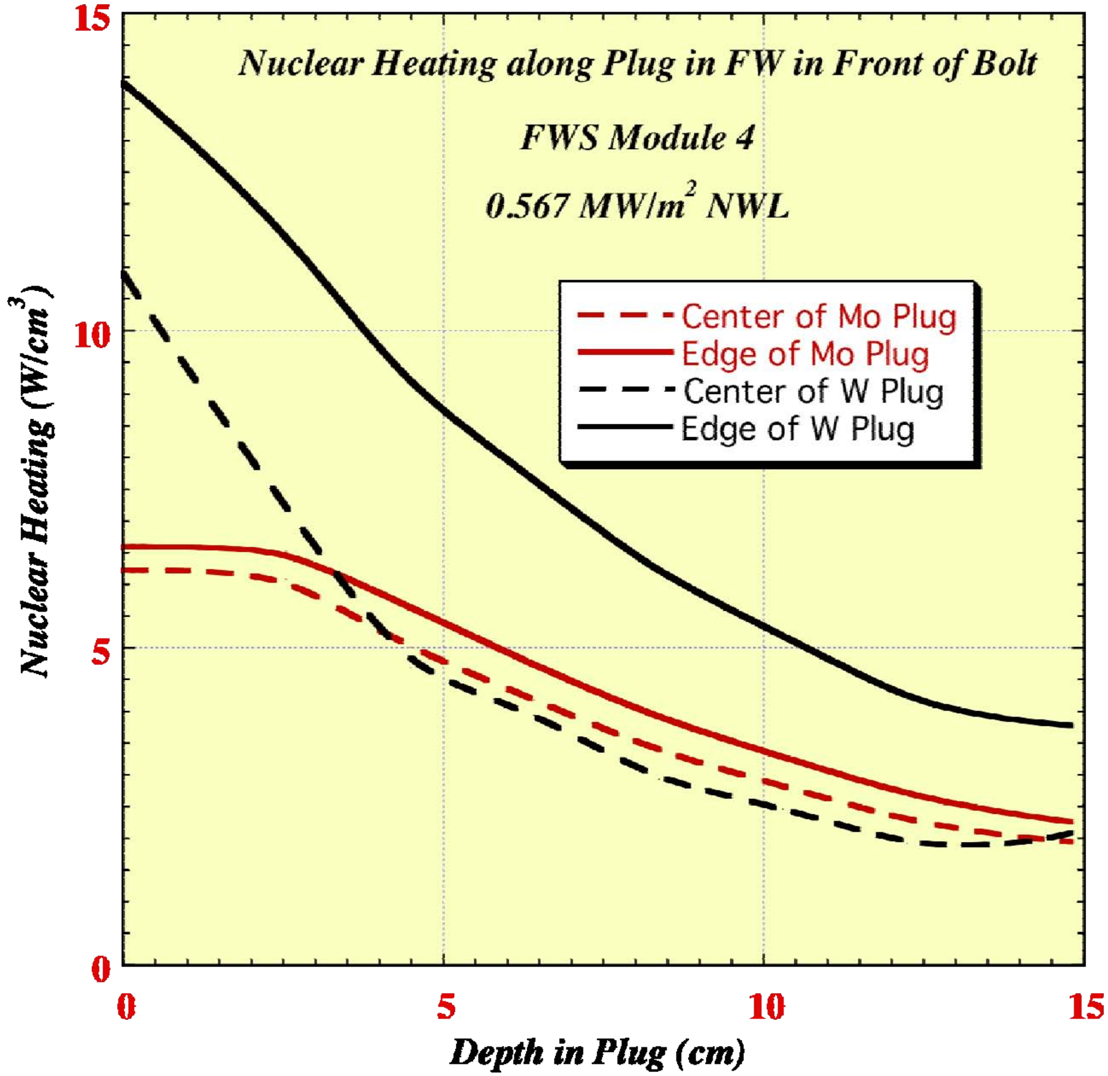
- dpa in Inconel is ~5% higher than in SS
- Increasing opening diameter from 60 to 100 mm enhances peak dpa in bolt head by ~37%

Using Shielding Plugs for Protection of FW Attachment Bolts

- An option for shielding the FW attachment bolts is to use shielding plugs in the FW opening in front of the bolt head
- Two materials were considered for the plug: Mo, W
- Plug is 62 mm in diameter and 148 mm in height
- Bolt is made of Inconel-718 with a 10 mm diameter inner Cu core

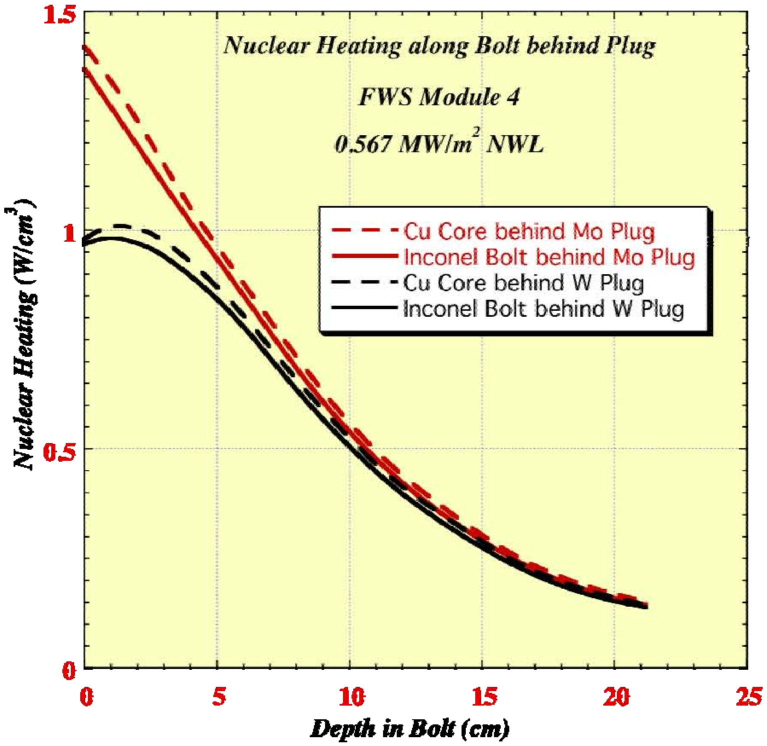


Nuclear Heating in Shielding Plug



- Peak heating in W plug is much higher than that in Mo plug
- Due to the larger attenuation in W, gradient in nuclear heating is more pronounced
- Nuclear heating at the outer edge of plug is higher than that at the center of the plug due to effect of water in surrounding shield

Nuclear Parameters in Shielded Bolt



- Peak heating in bolt is higher behind Mo plug that has weaker attenuation
- Nuclear heating in the Cu core is slightly higher than that in the Inconel bolt
- *Using plugs reduces bolt heating by a factor of ~4*

	Peak End-of-life Radiation Damage (dpa @ 0.3 MWa/m^2)
Mo plug	0.25
W plug	0.22

W is preferred if large heating in plug can be handled:

- Smaller heating in bolt and surrounding shield
- Lower bolt radiation damage
- Less activation concern

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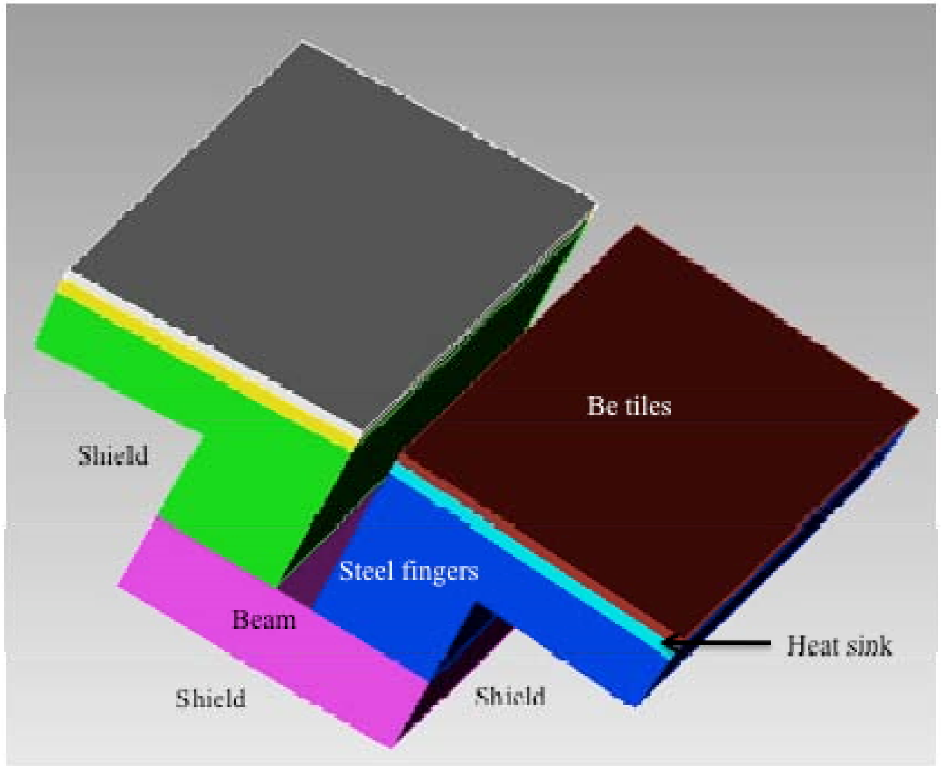
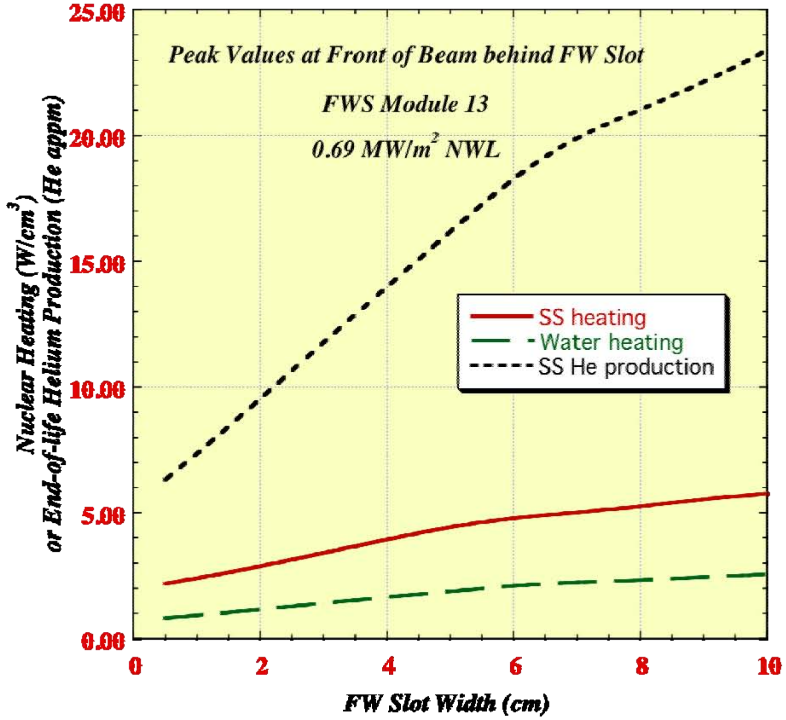
Effect of Poloidal Slot in FW on Nuclear Parameters in Joining Beam

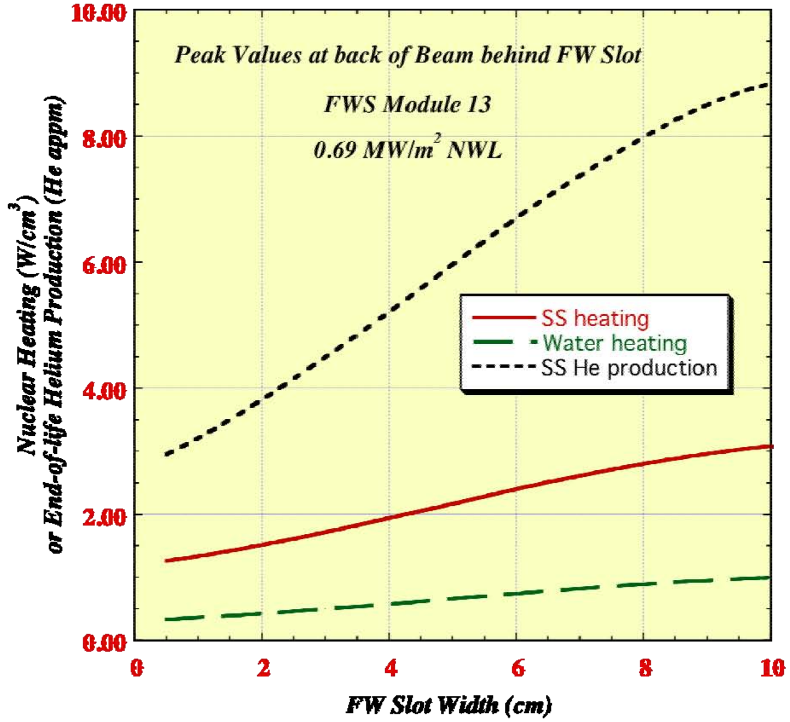
- Performed neutronics calculations to determine nuclear heating and helium production in the joining beam behind a proposed poloidal slot (canyon) in the first wall
- 2-D neutronics analysis performed using PARTISN in X-Y with S_{12} and ray tracing first collision source



- Attachment beam is 7.5 cm thick with 29.4 cm toroidal width
- Slot is 17.5 cm deep
- Calculations performed for slot widths ranging from 0.5 cm to 10 cm
- Results normalized to the neutron wall loading of 0.69 MW/m^2 at module 13 with contribution from facing modules included

Peak Parameters in Beam





- Increasing slot width from 0.5 cm to 10 cm results in a factor of ~3 increase in peak heating and He production in beam
- Attenuation factor in 7.5 cm thick beam is ~2.5 for He and ~1.8 for heating
- Peak He production in beam back at module 13 location exceeds 1 appm (rewelding limit for thick welds) even with 0.5 cm slot width
- Rewelding will not be possible at the front of the beam for any slot width

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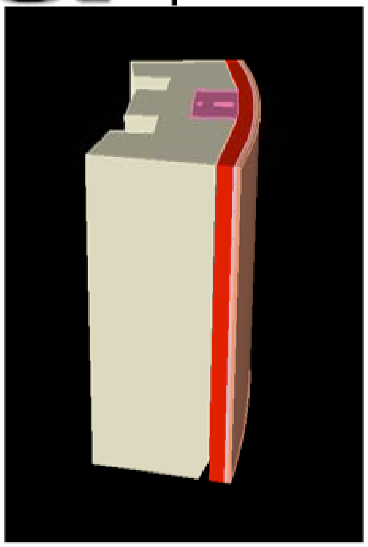
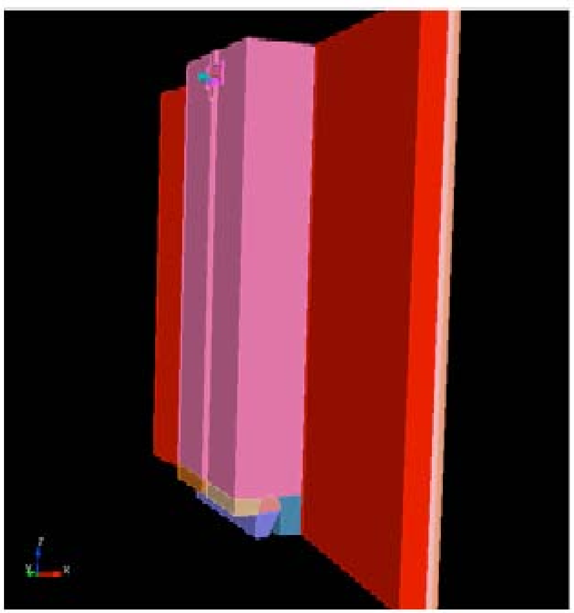
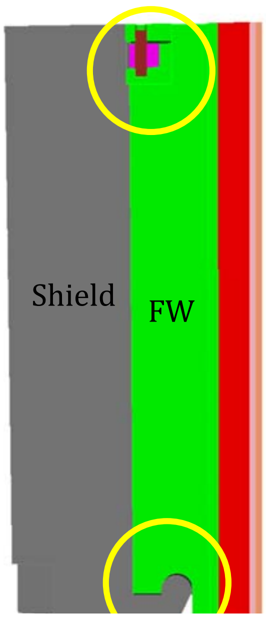
Nuclear Heating Analysis of New FW Attachment Scheme

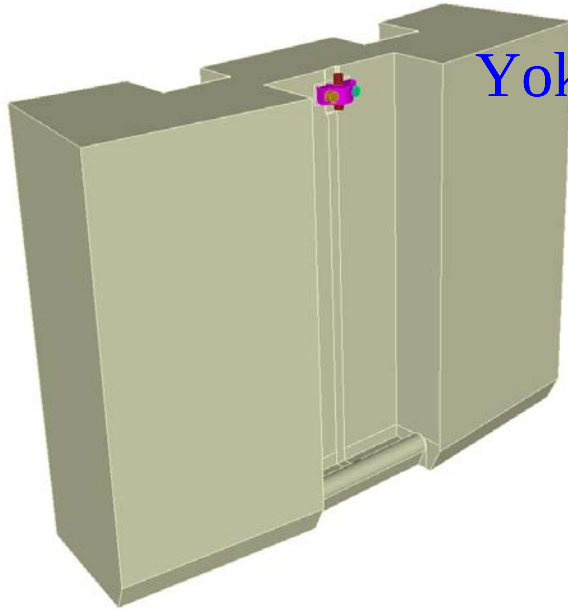


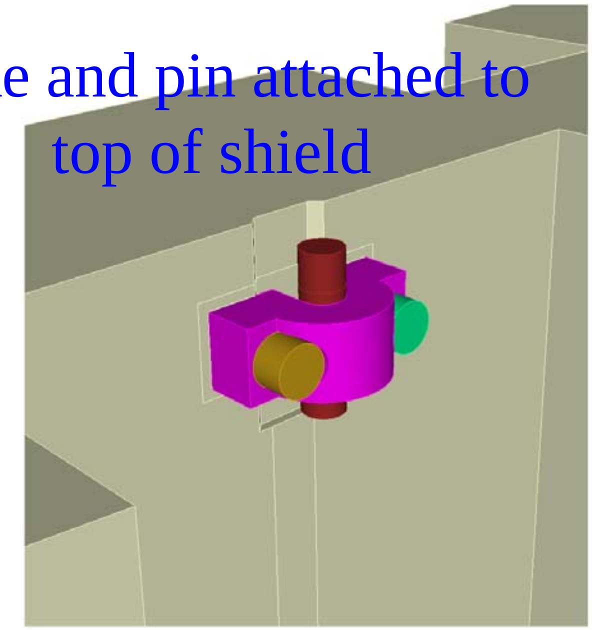


- A beam on back of FW fits over a "knuckle" that is attached to base of shield
- The top of FW attached to shield with a Cu yoke around an Inconel pin with two Inconel bolts
- There are two holes through all the first wall layers to provide access to the bolts that attach the yoke to the shield module
- There are concerns about nuclear heating and the ability to cool both the hinge components and the yoke/pin components
- Performed 3-D neutronics analysis of this system using DAG-MCNP5
- Results scaled to Module 4 NWL of 0.567 MW/m^2

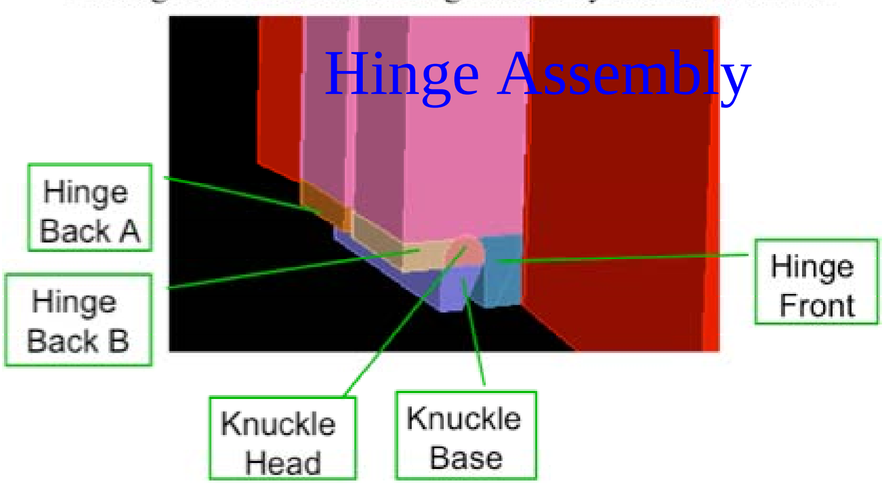
Attachment Configuration details

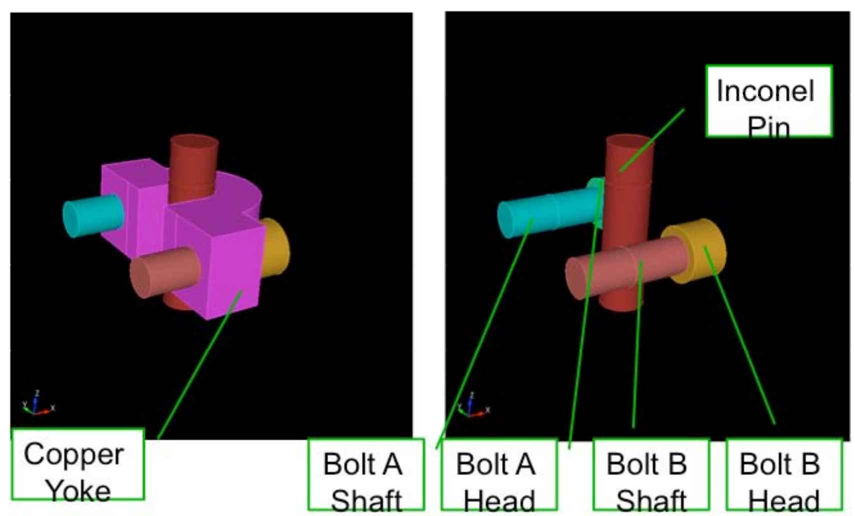






All regions of interest in hinge assembly are 100% SS316



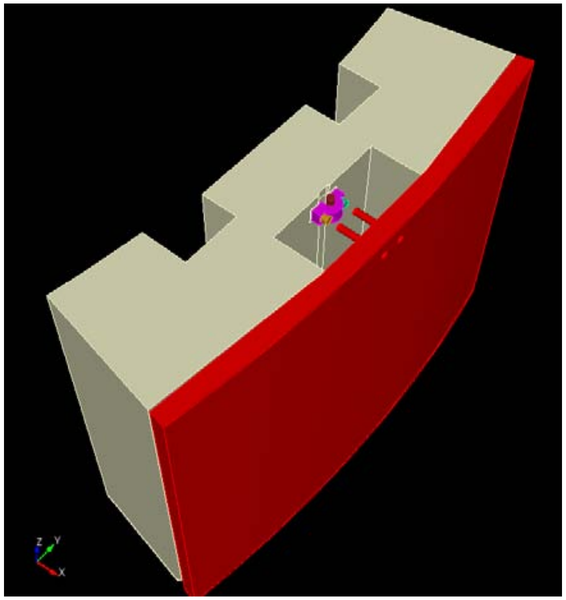


Nuclear Heating Results

Part	Total Nuclear Heating (W/cc)
Knuckle Head	2.83 (±0.45%)
Knuckle Base	3.22 (±0.40%)
Hinge Front	5.07 (±0.28%)
Hinge Back A	1.87 (±0.77%)
Hinge Back B	1.86 (±0.76%)
Yoke	2.44 (±1.05%)
Pin	2.44 (±1.63%)

Bolt Variations

- Bolts have direct line of sight through access holes in FW
- Bolt material choices
  - Inconel 718
  - Moly
- Open vs. plugged (with SS) holes



Total Nuclear Heating in W/cc

Part	Inconel-718 Bolts		Mo Bolts	
	Open Holes	Plugged Holes	Open Holes	Plugged Holes
Bolt A Head	2.91 (±2.36%)	2.68 (±2.79%)	3.48 (±2.22%)	3.24 (±3.05%)
Bolt B Head	2.85 (±2.40%)	2.84 (±2.82%)	3.42 (±2.25%)	3.46 (±3.08%)
Bolt A Shaft	2.10 (±2.61%)	2.00 (±5.98%)	2.48 (±2.31%)	2.40 (±3.22%)
Bolt B Shaft	2.07 (±2.53%)	2.08 (±2.95%)	2.48 (±2.34%)	2.51 (±3.19%)
Plug A/B	N/A	5.99/5.83	N/A	6.02/5.85

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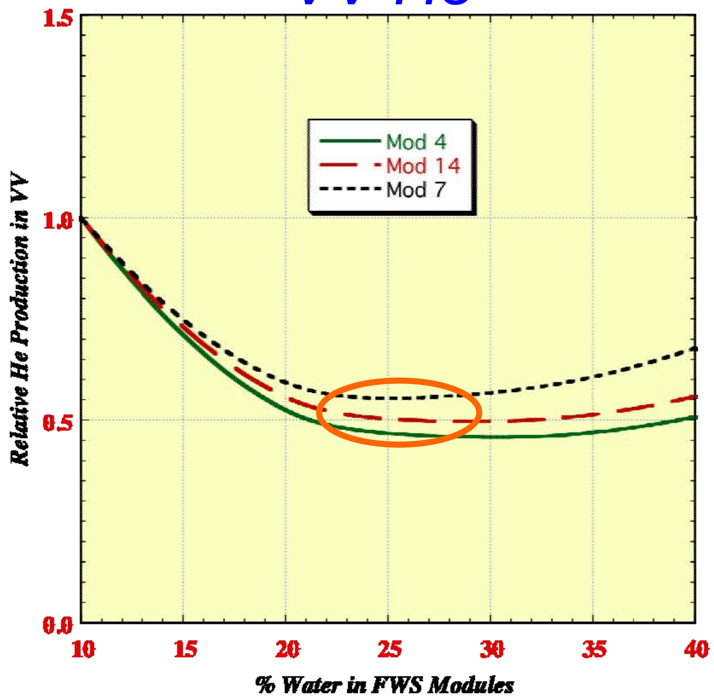
Impact of Water Content on VV and Magnet Shielding

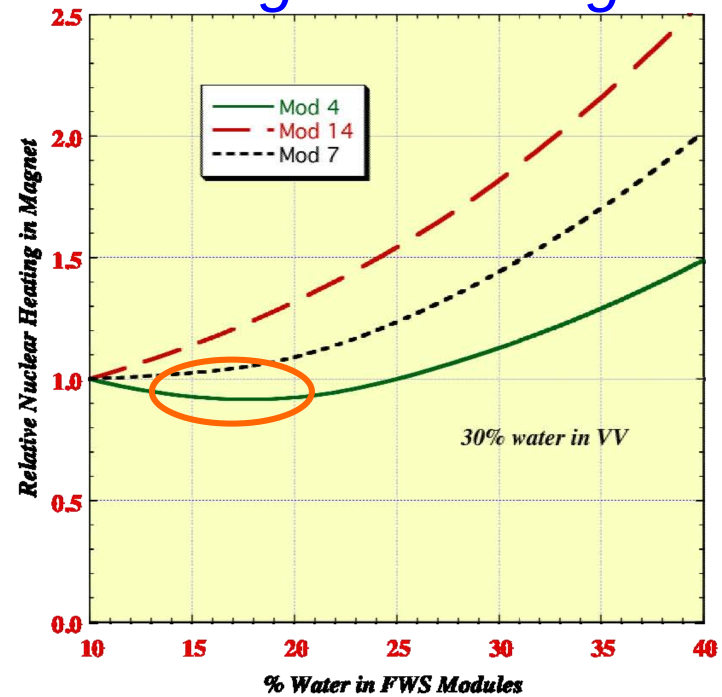
- Performed simple parametric calculations to determine optimum water content in FW/shield module (FWS) and VV
- Effect of water content varies depending on the component to be shielded (VV, magnet), radiation parameter to be minimized (heating, He production, etc) and poloidal location
- VV helium production peaks at OB mid-plane
- Magnet heating peaks at IB mid-plane
- 1-D calculations using the ITER radial build at mid-plane provided by the ITER IO (Eduard Polunovskiy)
- FWS module is ~45 cm thick in both IB and OB regions. Average FWS Mod 7 thickness is ~50 cm
- VV is ~34 cm thick in IB and increases to ~75 cm thick in OB
- For optimization of water content in FWS, VV assumed to have 40% water in shielding zone between shells

Optimization of Water Content in FWS

VV He



Magnet Heating



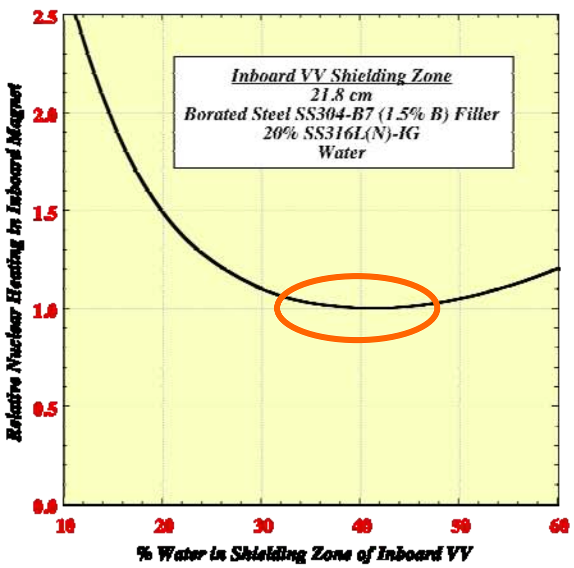
- Less water fraction needed in FWS for magnet than VV shielding
- Less water needed in FWS for magnet protection as we move from IB to OB because of increased VV thickness with its large water content
- Minima are quite shallow and small changes will have small impact
- Use 10-20% water in IB FWS (where magnet heating is driver) that could increase to 20-30% as one moves poloidally to OB where VV He is driver

VV Water Content Optimization

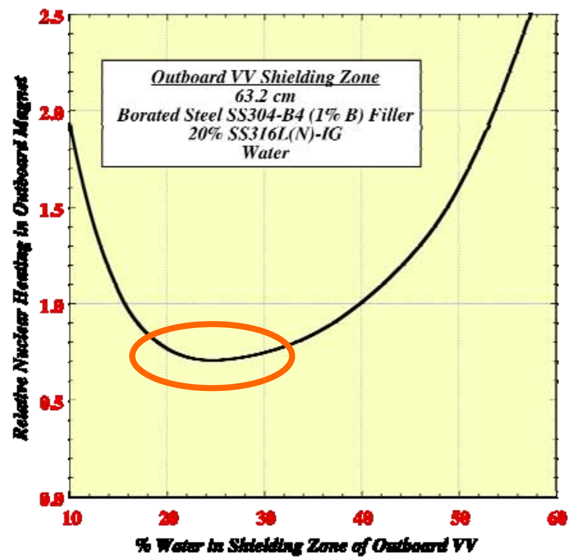
- Assessed the effect of water content in the shielding zone between VV shells on magnet heating
- Inner and outer shells of VV are 6 cm thick and made of SS316L(N)-IG
- VV shielding zone is 21.8 cm IB and 63.2 cm OB
- VV shielding zone has 20% SS316L(N)-IG for ribs and brackets with the rest including water and shielding filler
- Filler is SS304-B7 (1.5% B) in IB and SS304-B4 (1% B) in OB
- In locations near the equatorial port and at the center of the TF coil ferromagnetic steel SS430 plates are used in VV for ripple control
- Current reference uses 40% water in shielding zone of VV
- We determine optimum water content for the IB VV with SS304-B7 filler and for OB VV with SS304-B4 and SS430 fillers

Optimum VV Water Content

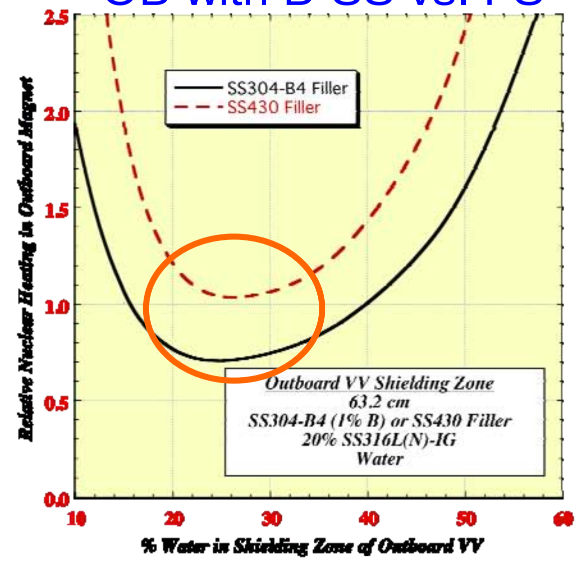
IB with B-SS



OB with B-SS



OB with B-SS vs. FS



- Results confirm that the dominant inboard magnet heating minimizes at the baseline 40% water in shielding zone of IB VV
- While OB magnet heating minimizes at lower (~25%) water in the thicker VV shielding zone, a higher value up to 40% is acceptable due to small contribution of OB magnet heating and larger filler cost with small water content (<0.4 kW reduction in total TF heating when water content reduced from 40% to 25%)
- Using SS430 instead of SS304-B4 increases magnet heating by 42%
- Results show a trend of larger optimum water content for thinner VV implying that using more water (~40%) at upper ports will help with magnet heating there

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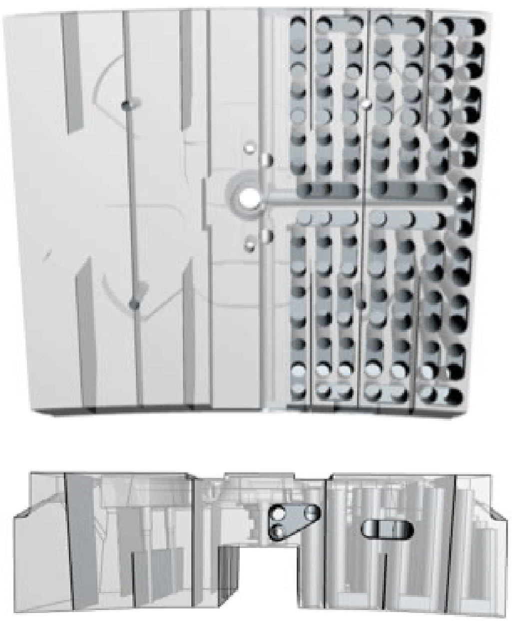
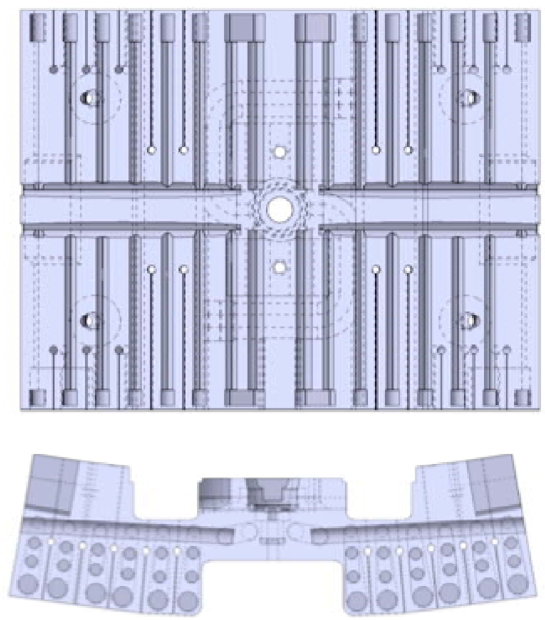
Shielding Impact of Water Distribution and Channel Configuration in Shield Module

Option 1 – Radial Cooling



Option 2 – Poloidal Cooling



- Radial water channels are equivalent to streaming paths but not as bad as void and they do not penetrate all the way through the module in a realistic configuration
- To quantify these effects we performed simple generic calculations

- Base case has homogeneous 75% SS, 25% water uniformly distributed in FWS
- Two variations of this shield were considered:
    - 35% water in font half and 15% water in back half (resembling poloidal channels configuration)
    - 15% water in front half and 35% water in back half (preferred neutronically)
- An option with separate 2 cm radius radial water channel considered to assess impact of radial channels

Results for impact of water distribution and configuration in shield

Shielding performance improved if more water is used in back of shield

	Uniform 25% water distribution	35% water in front, 15% water in back	15% water in front, 35% water in back
VV He	1	1.23	0.89
VV heating	1	1.11	0.94
Magnet heating	1	1.11	0.96
Magnet fast neutrons	1	1.03	0.96

Peaking factors due to radial water channels are smaller in magnet

	Radial water channel penetrating through whole FWS
VV He	1.30
VV heating	1.10
Magnet heating	1.06
Magnet fast neutrons	1.03

Average nuclear parameters in VV and magnet are higher with radial water channels

	Uniform 25% water distribution	35% water in front, 15% water in back (poloidal conf.)	Radial water channel penetrating through whole FWS
VV He	1	1.23	1.41
VV heating	1	1.11	1.38
Magnet heating	1	1.11	1.37
Magnet fast ns	1	1.03	1.27

- Lower effect expected for realistic configuration with radial channels not fully penetrating FWS and addition of poloidal FW channels and thick back plate
- For realistic configuration poloidal and radial channels in shield are expected to give comparable radiation parameters

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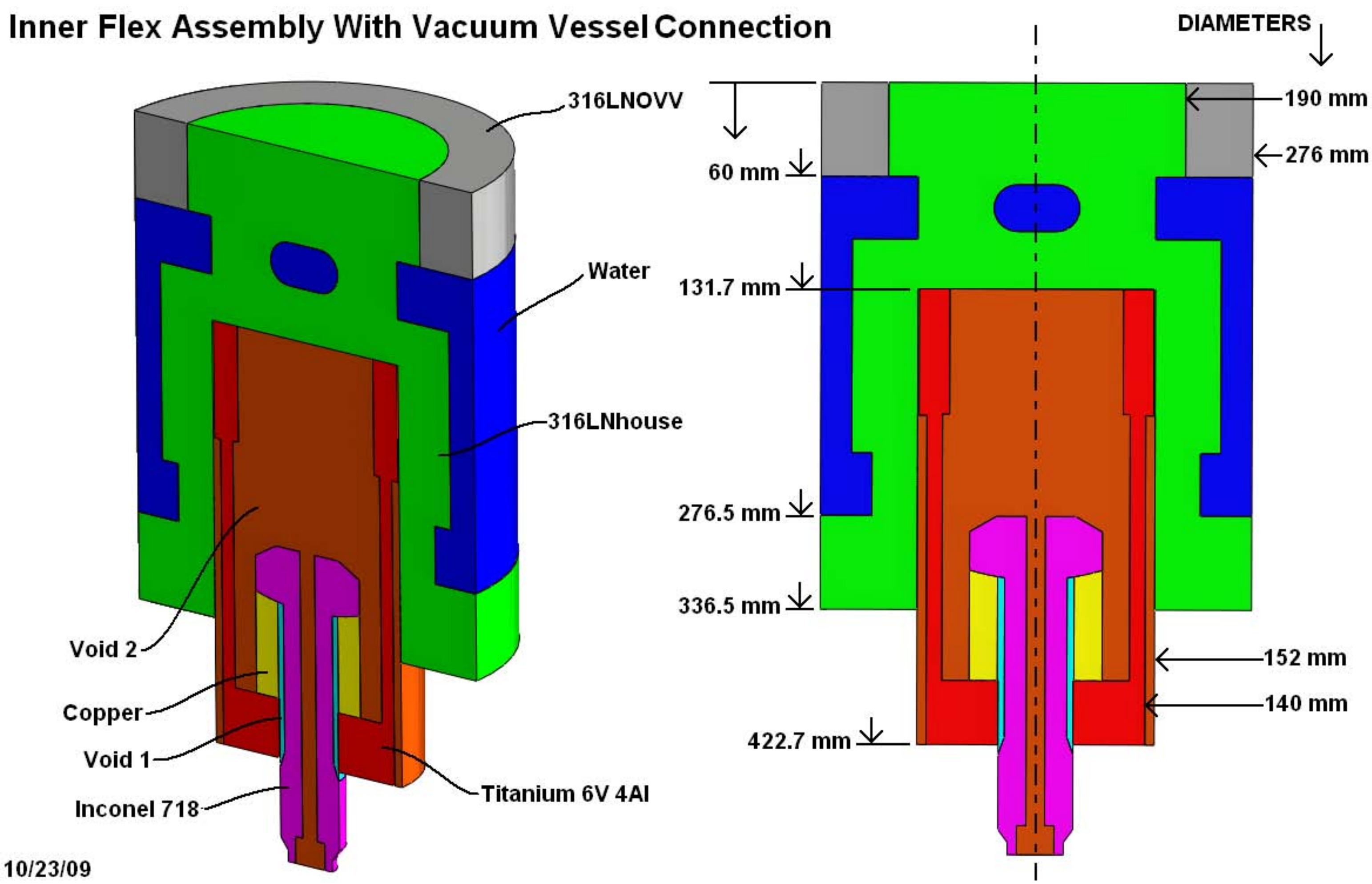
Impact of Flexible Joints on Inboard TF Magnet Shielding

- Recent 3-D analysis by IO yield ~19.3 kW total nuclear heating in TF coils
  - ITER_D_2V34SH v 1.0, 12 October 2009
  - ITER_D_2LF7NM v 1.0, 21 October 2009
- Homogenized composition (41% water, 15.9% SS316, 36.9% borated SS304, 3.4% void, 1.9% Ti-6Al-4V, 0.5% Inconel-718, 0.4% Cu) in shielding zone between the two VV shells accounts for flexible joints
- We investigate here the impact of the flexible joints on peaked parameters and integrated heating in IB TF coils
- Used PARTISN discrete ordinates code in R-Z (around axis of flexible joint) with S_{12} and ray tracing first collision source to mitigate ray effects
- Used FENDL-2.1 nuclear data in 175n-42g groups
- Used radial build at mid-plane with both IB and OB modules
- VV radial build used

6 cm inner shell	SS316 100%
21.65 cm shielding zone	SS316 20%, B-SS304 40%, water 40%
6 cm outer shell	SS316 100%

- Calculations performed with and without the flexible joints
- Since 4 flexible mounts and housings are used per module, an outer reflecting boundary at radius of 31.5 cm, corresponding to ¼ the back area of shield module 4, was used

Flexible Mount and Housing



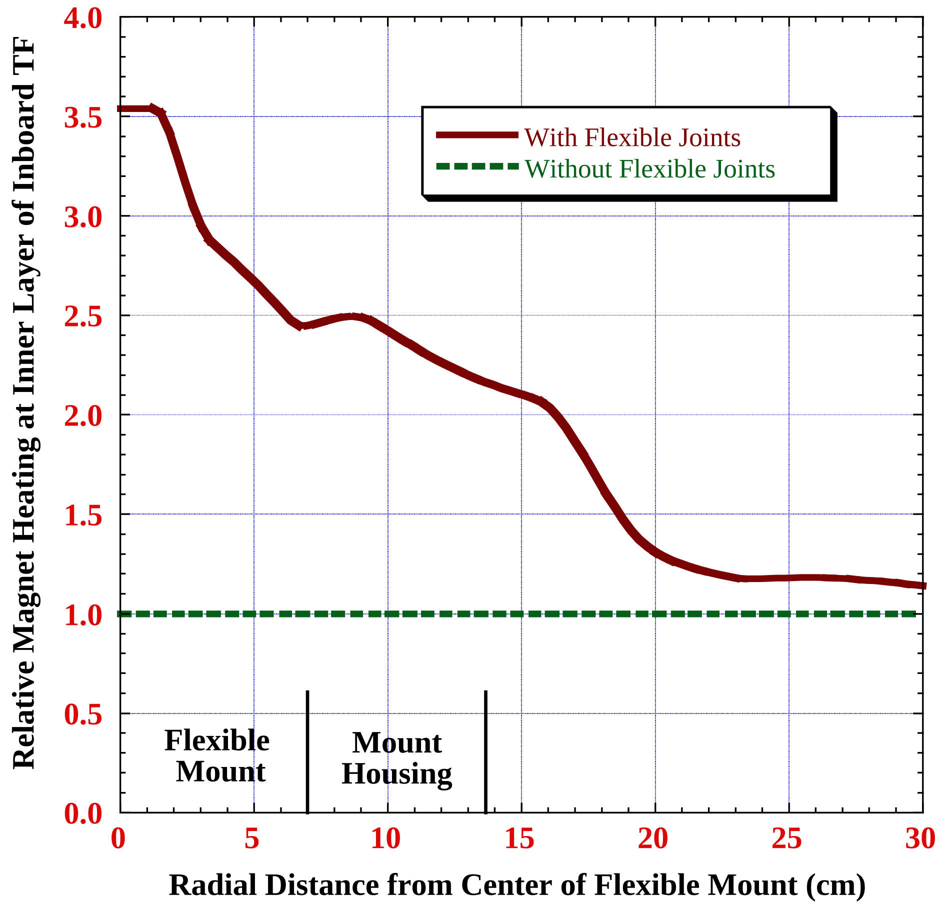
The housing fits within the full thickness of VV with the extended part of bolt and Ti casing penetrating into the back of the shield module

Impact of Flexibles on TFC Parameters

Parameter	Relative Value
Peak Fast Neutron Fluence	2.91
Average Fast Neutron Fluence at Inner Surface of Coil	1.40
Peak Insulator Dose	3.35
Average Insulator Dose at Inner Surface of Coil	1.38
Peak Power Density	3.54
Average Power Density at Inner Surface of Coil	1.41

Hot spots with up to a factor of 3.5 peaking occur behind the flexible joint

Total IB magnet heating increases by 41% due to the flexible joints



Effect of Detailed Geometry of Flexible Joint

- The IO analysis uses the Alite model with homogenized composition (41% water, 15.9% SS316, 36.9% B-SS304, 3.4% void, 1.9% Ti-6Al-4V, 0.5% Inconel-718, 0.4% Cu) in shielding zone between the two VV shells to approximately account for flexible joints without detailed configuration
- Total magnet heating was 19.3 kW including a correction factor of 1.1 for effect of flexibles
- We performed additional calculation with the the homogenized composition (Alite composition) to assess the effect of heterogeneity and configuration of flexibles

Inboard (seg. 1-14), kW	15.60	8.76	14.14	13.88	13.48
Difference with the previous case, kW		6.84	-5.38	0.26	0.40
Upper port segments (seg. 22-29), kW	0.30	2.42	2.48	2.48	2.48
Difference with the previous case, kW		-2.12	-0.06	0.00	0.01
Lower port segments (seg. 35-45), kW	2.03	1.52	1.57	1.57	1.58
Difference with the previous case, kW		0.51	-0.05	0.00	0.00
total, kW	19.99	14.37	19.93	19.67	19.27
Difference with the previous case, kW		5.62	-5.57	0.26	0.40

Yellow cells denote the inboard straight leg of the TF coil

The totals were computed using the following formulae:

$$H_{inboard} = F_1 \times F_2 \times F_3 \times F_4 \times 9 \times \sum_{i=1}^{14} H_i$$

$$H_{outboard} = 9 \times F_4 \times \sum_{i=15}^{90} H_i$$

where

H_i – the heating (W) in the two TFC segments at position i

F_1=1.2 – correction factor for vacuum vessel heterogeneity

F_2=1.1 – correction factor for the effect of the flexible joint housings

F_3=1.06 – correction factor for the blanket heterogeneity

F_4=1.3 – confidence factor

Case	Relative Value
No Flexible Joints	1
Flexible Joints Homogenized in VV	1.14
Detailed Configuration of Flexibles	1.41

Total IB magnet heating increases by a factor of 1.24 as a result of detailed configuration of flexible joints

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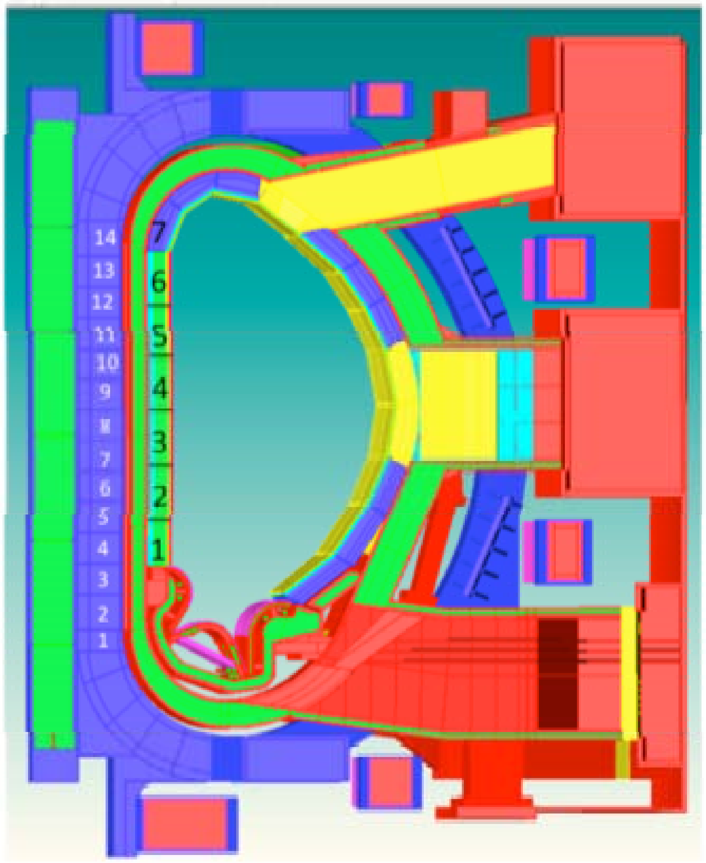
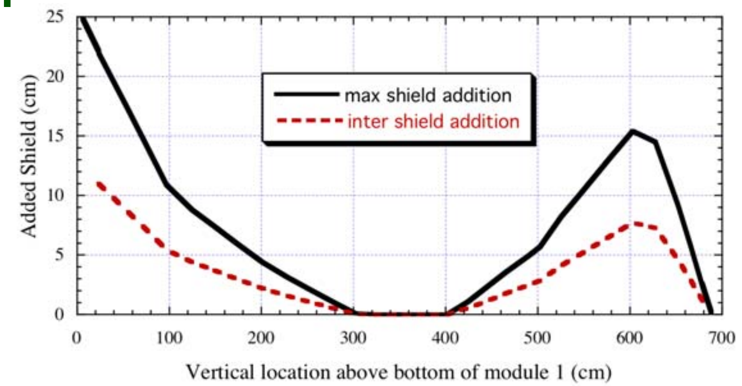
- Assessment of adding tungsten in IB shield modules
- Changing material in IB VV to reduce magnet heating

Impact of FW Shaping and Added IB Shield Thickness on Total IB Magnet Heating

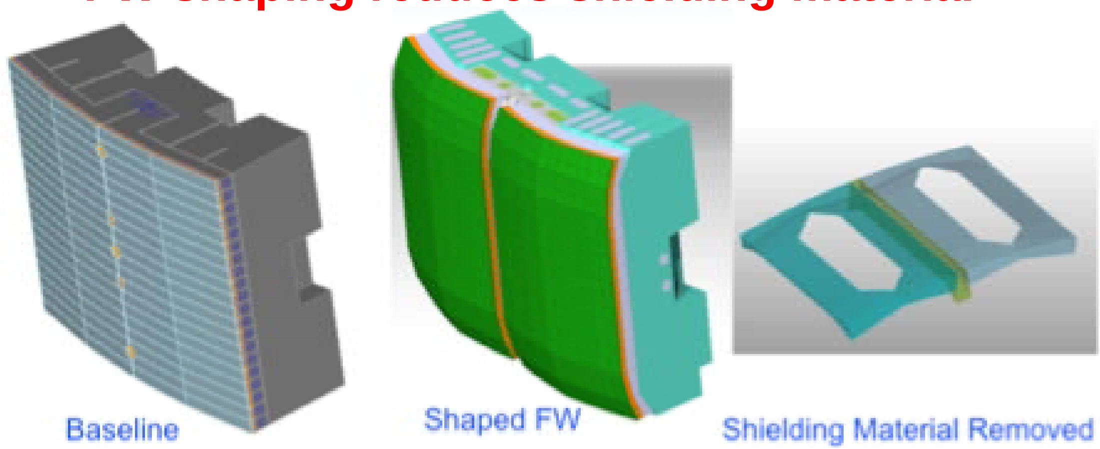
ITER model with location of FWS modules 1-7 relative to IB magnet segments



Proposed added thickness for FWS modules 1-7



FW shaping reduces shielding material



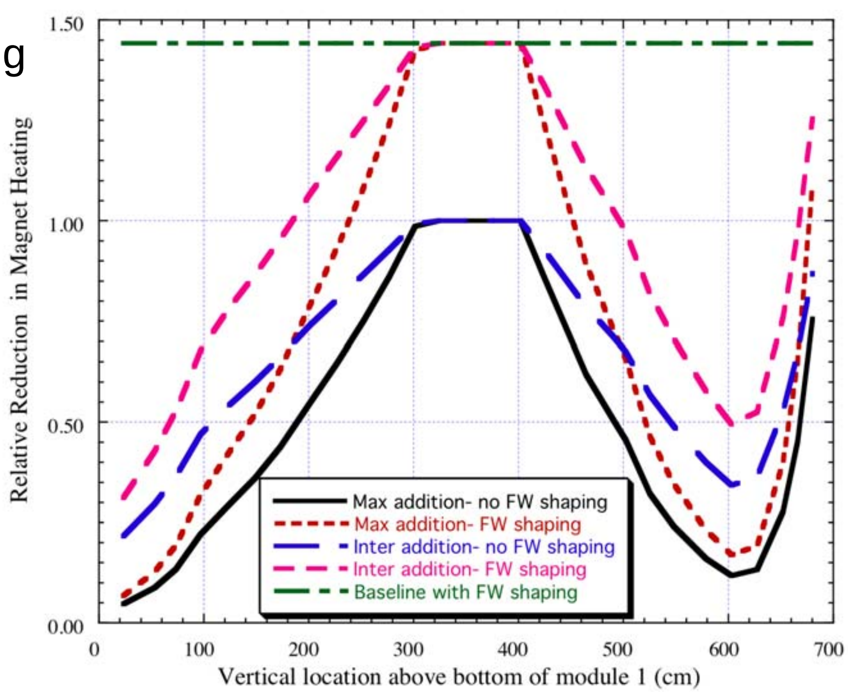
- 6 % of volume of baseline FWS removed by FW shaping
- Effective FWS thickness reduces from 440 to 414 mm

Vertical variation of magnet heating change from baseline

- Increased FWS module thickness results in enhancement in neutron wall loading by <3% with maximum addition and <1.7% with intermediate addition
- The e-fold thickness for radial build of FWS module 4 with 13% water in shield was determined to be 7.1 cm
- Calculated the net reduction factor in IB magnet heating for five options

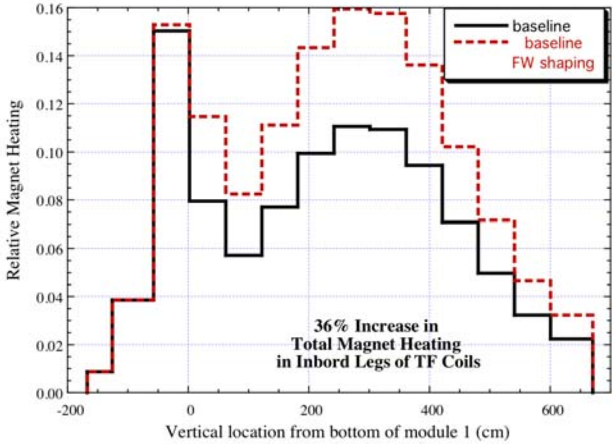
1. Baseline with shaped FW
2. Intermediate addition without FW shaping
3. Intermediate addition with FW shaping
4. Maximum addition without FW shaping
5. Maximum addition with FW shaping

Results of vertical variation of magnet heating relative to baseline were combined with calculated relative nuclear heating profile in the 14 segments of IB leg (as reported in "Nuclear Heat of TF Inboard Legs with Fine Structures of Inboard Blanket", INAR-001, Rev. 1, 9/11/07) to determine expected change in IB magnet heating

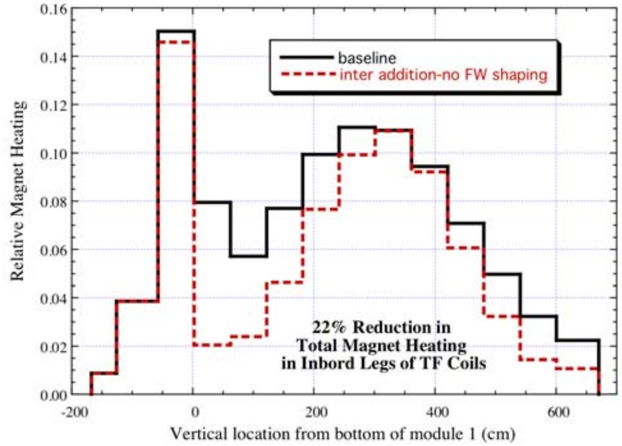


Expected change in IB magnet heating due to FW shaping and Added FWS Thickness

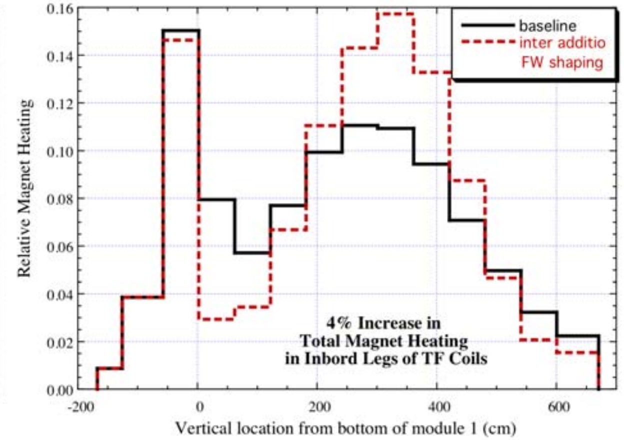
1) Baseline+Shaping



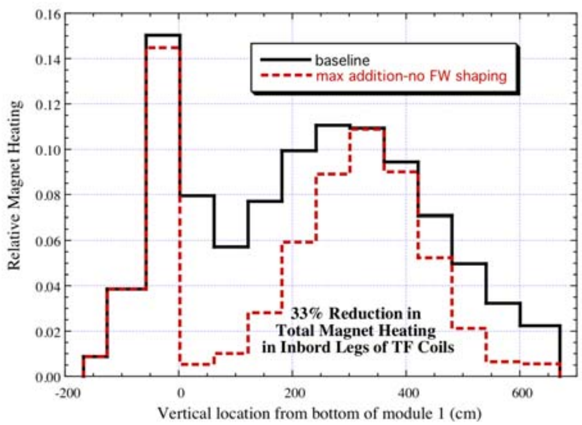
2) Inter. Add.- No Shaping



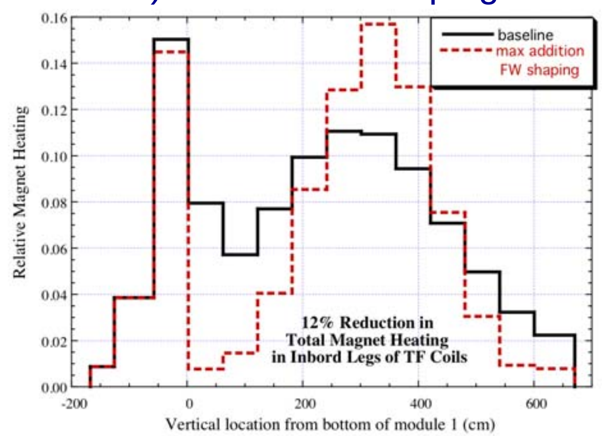
3) Inter. Add.+Shaping



4) Max. Add.- No Shaping



5) Max. Add.+Shaping



Design Option	Relative IB Magnet Heating	% Change from Baseline
Baseline	1	0
1	1.36	+36%
2	0.78	-22%
3	1.04	+4%
4	0.67	-33%
5	0.88	-12%

- We expect a smaller impact on total magnet heating since IB region contributes ~80-85%
- Shaping FW counteracts benefits of adding shield thickness with impact being only ~12% reduction in IB magnet heating with the proposed maximum shield thickness addition

Recent Estimate of TF Nuclear Heating

ITER_D_2LF7NM v 1.0, 21 October 2009

MCNP best estimate of the TFC nuclear heating, W per 2 TFC

Segment	Case 1	Case 2	Case 3	Case 4	Case 5
1	8.88E+00	6.71E+00	6.83E+00	7.03E+00	7.06E+00
2	2.04E+01	1.05E+01	1.12E+01	1.13E+01	1.11E+01
3	6.53E+01	1.63E+01	1.87E+01	1.79E+01	1.67E+01
4	5.35E+01	1.36E+01	1.98E+01	1.92E+01	1.71E+01
5	6.06E+01	2.08E+01	3.28E+01	3.31E+01	2.93E+01
6	8.27E+01	3.45E+01	6.15E+01	6.16E+01	5.66E+01
7	1.12E+02	6.32E+01	1.06E+02	1.02E+02	1.02E+02
8	1.21E+02	8.72E+01	1.43E+02	1.39E+02	1.37E+02
9	1.22E+02	9.46E+01	1.53E+02	1.51E+02	1.50E+02
10	1.09E+02	8.41E+01	1.38E+02	1.36E+02	1.37E+02
11	7.65E+01	5.47E+01	8.79E+01	8.75E+01	8.55E+01
12	5.39E+01	2.91E+01	4.88E+01	4.79E+01	4.30E+01
13	3.68E+01	1.22E+01	2.36E+01	2.18E+01	1.92E+01
14	3.04E+01	7.54E+00	1.25E+01	1.24E+01	1.20E+01
15	8.95E+00	3.00E+00	3.97E+00	3.96E+00	4.06E+00
16	1.64E+01	9.04E+00	1.08E+01	1.08E+01	1.08E+01
17	1.49E+01	1.12E+01	1.17E+01	1.16E+01	1.15E+01
18	1.51E+01	1.16E+01	1.23E+01	1.23E+01	1.22E+01
19	1.82E+01	1.52E+01	1.54E+01	1.55E+01	1.54E+01
20	1.01E+01	9.54E+00	1.02E+01	1.02E+01	1.02E+01
21	5.69E+00	7.97E+00	8.04E+00	8.05E+00	7.91E+00
22	3.52E+00	6.29E+00	6.54E+00	6.54E+00	6.56E+00
23	3.82E+00	2.94E+01	3.02E+01	3.02E+01	3.04E+01
24	2.75E+00	6.82E+01	7.05E+01	7.05E+01	7.01E+01
25	2.56E+00	5.88E+01	5.88E+01	5.88E+01	5.89E+01
26	2.64E+00	2.53E+01	2.68E+01	2.68E+01	2.65E+01
27	2.89E+00	8.27E+00	8.23E+00	8.23E+00	8.22E+00
28	3.27E+00	5.15E+00	5.76E+00	5.76E+00	5.64E+00
29	3.80E+00	5.39E+00	5.42E+00	5.42E+00	5.43E+00
30	8.19E+00	8.85E+00	9.06E+00	9.06E+00	9.23E+00
31	6.14E+00	5.19E+00	5.12E+00	5.12E+00	5.16E+00
32	4.39E+00	3.03E+00	3.25E+00	3.25E+00	3.25E+00
33	4.92E+00	2.79E+00	3.00E+00	3.00E+00	3.04E+00
34	4.76E+00	3.20E+00	3.30E+00	3.30E+00	3.27E+00
35	5.16E+00	2.95E+00	3.26E+00	3.26E+00	3.27E+00
36	5.97E+00	4.24E+00	3.74E+00	3.74E+00	3.81E+00
37	1.07E+01	7.92E+00	7.68E+00	7.68E+00	7.60E+00
38	3.05E+01	1.78E+01	1.81E+01	1.81E+01	1.83E+01
39	3.22E+01	2.16E+01	2.24E+01	2.24E+01	2.25E+01
40	2.61E+01	2.11E+01	2.30E+01	2.30E+01	2.30E+01
41	2.02E+01	1.73E+01	1.70E+01	1.70E+01	1.71E+01
42	1.38E+01	1.25E+01	1.28E+01	1.28E+01	1.29E+01
43	9.28E+00	7.74E+00	8.49E+00	8.49E+00	8.49E+00
44	1.00E+01	9.37E+00	9.86E+00	9.86E+00	9.77E+00
45	9.76E+00	7.76E+00	8.14E+00	8.14E+00	8.12E+00
46	1.36E+01	1.15E+01	1.23E+01	1.22E+01	1.22E+01
47	1.25E+01	1.25E+01	1.31E+01	1.31E+01	1.31E+01
48	1.09E+01	9.49E+00	9.05E+00	8.90E+00	8.91E+00
49	1.39E+01	1.18E+01	1.18E+01	1.19E+01	1.18E+01
50	7.39E+00	6.29E+00	6.24E+00	6.30E+00	6.27E+00
Inboard (seg. 1-14), kW	*15.60*	*8.76*	*14.14*	*13.88*	*13.48*
Difference with the previous case, kW		*6.84*	*-5.38*	*0.26*	*0.40*
Upper port segments (seg. 22-29), kW	*0.30*	*2.42*	*2.48*	*2.48*	*2.48*
Difference with the previous case, kW		*-2.12*	*-0.06*	*0.00*	*0.01*
Lower port segments (seg. 35-45), kW	*2.03*	*1.52*	*1.57*	*1.57*	*1.58*
Difference with the previous case, kW		*0.51*	*-0.05*	*0.00*	*0.00*
total, kW	*19.99*	*14.37*	*19.93*	*19.67*	*19.27*
Difference with the previous case, kW		*5.62*	*-5.57*	*0.26*	*0.40*

Yellow cells denote the inboard straight leg of the TF coil.
The totals were computed using the following formulae:

$$H_{inboard} = F_1 \times F_2 \times F_3 \times F_4 \times 9 \times \sum_{i=1}^{14} H_i,$$

$$H_{outboard} = 9 \times F_4 \times \sum_{j=15}^{50} H_i,$$

where
H_i – the heating (W) in the two TFC segments at position i,
F_1=1.2 – correction factor for vacuum vessel heterogeneity
F_2=1.1 – correction factor for the effect of the flexible joint housings
F_3=1.06 – correction factor for the blanket heterogeneity
F_4=1.3 – confidence factor

- Case 5 included maximum thickening of IB FWS, FW shaping, and added B-SS at thermal shield
- Total TF heating (with cumulative correction factors of 1.82) is 19.27 kW with 13.48 kW in IB legs
- Upper port area contributes 2.48 kW with 1.58 kW at lower port
- The rest is only 1.73 kW from other regions including OB region

Outline

- Heating and damage in bolts used to secure the FW to the shield module
- Heating of the FW support beam due to a slot in FW
- Heating in the FW attachment mechanism with hinge, knuckle, yoke, pin, bolt
- Optimization of water content in shield and VV
- Shielding impact of water distribution and channel configuration in shield module
- Impact of Flexible Joints on Inboard TF Magnet Shielding
- Effect of FW shaping and added IB shield thickness on magnet heating
- Assessment of adding tungsten in IB shield modules
- Changing material in IB VV to reduce magnet heating

Effect of Adding W to IB Shield Modules

- To achieve the goal of 14 kW total TF heating we need 5.3 kW reduction
- This represents ~40% of the 13.5 kW inboard heating
- We investigate here impact of replacing some of the SS316 in IB shield modules by W
- We will determine the amount of W needed to realize this reduction
- Used radial build at mid-plane with both IB and OB modules
- IB module 4 radial build used

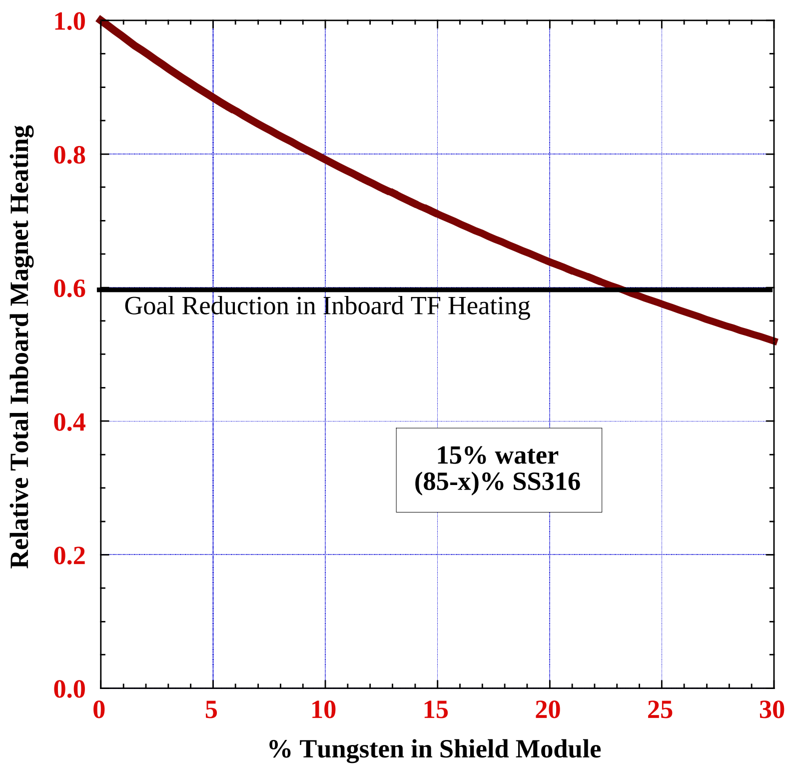
1 cm Be PFC layer	Be 100%
1.2 cm Hypervapotron heat sink	Cu 42%, Steel 4%, Water 54%
4.9 cm SS FW structure	Steel 67%, Water 33%
37.9 cm Shield zone	**Steel (85-x)%, W x%, Water 15%**

- No W added in OB FWS modules
- VV radial build used

6 cm inner shell	SS316 100%
21.5 cm shielding zone	SS316 20%, SS304(2 wt%B) 40%, water 40%
6 cm outer shell	SS316 100%

Effect of Using W in IB Shield Modules

% W in Shield Module	Effective SS thickness replaced	Relative total IB TF heating	% reduction in IB TF heating
0%	0	1	0
5%	2 cm	0.886	11.4%
10%	4 cm	0.794	20.6%
20%	8 cm	0.642	35.8%
30%	12 cm	0.530	47.0%



- Reduction in IB TF nuclear heating increases (non-linearly) as the W content in shield module increases
- About 23% W (~9 cm effective W thickness) is needed to achieve the 14 kW total heating requirement
- More W can be used in modules 3-5 and less in modules 1,2,6,7

Outline

- Heating and damage in bolts used to secure the FW to the shield module
- Heating of the FW support beam due to a slot in FW
- Heating in the FW attachment mechanism with hinge, knuckle, yoke, pin, bolt
- Optimization of water content in shield and VV
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Can We Significantly Reduce IB TF Heating with Material Change in IB VV?

- To achieve the goal of 14 kW total TF heating we need 5.3 kW reduction
- This represents ~40% of the 13.5 kW inboard heating
- Current composition for shielding zone between VV shells is
  SS316 20%, SS304 (1.5 wt%B) 40%, water 40%
- We performed parametric analysis to assess effect of replacing the borated steel by WC and/or replacing the water by borated water
- For borated water we used 5 g boric acid per 100 cc of water

Material in IB VV shielding zone	Relative total IB TF heating	% reduction in IB TF heating	Reduction in TF heating (kW)
20% SS, 40% H2O, 40% B-SS	1	0	0
20% SS, 40% B-H2O, 40% B-SS	0.964	3.6%	0.5
20% SS, 40% H2O, 40% WC	0.589	41.1%	5.5
20% SS, 40% B-H2O, 40% WC	0.567	43.3%	5.8

- Using B-water instead of water in VV has very small effect on TF heating
- Using WC instead of B-SS as filler in shielding zone between shells of IB VV has significant impact that might get us to the goal total TF heating
- While using W instead of WC in VV results in a small improvement, the lower density and cost of WC make it the preferred option

Effect of WC Content in IB VV

- We changed the WC content in the VV shielding zone keeping the SS316 structure content at 20% (water content changed)

Material in IB VV shielding zone	Relative total IB TF heating	% reduction in IB TF heating	Reduction in TF heating (kW)
20% SS, 40% H2O, 40% B-SS	1	0	0
20% SS, 60% H2O, 20% WC	0.902	9.8%	1.3
20% SS, 50% H2O, 30% WC	0.699	30.1%	4.1
20% SS, 40% H2O, 40% WC	0.589	41.1%	5.5
20% SS, 30% H2O, 50% WC	0.535	46.5%	6.3
20% SS, 20% H2O, 60% WC	0.545	45.5%	6.1

- With WC in VV, water content optimizes at lower value than with SS due to moderating effect of C
- Lowest heating obtained with 30% water and 50% WC in shielding zone between VV shells. However, added cost and need for more cooling for the larger nuclear heating in W might render this option unattractive

Can we use intermediate (half) thickening of IB FWS and achieve TF heating goal by using more WC in VV?

NO! Maximum added reduction from increasing WC is only 0.8 kW which is less than the ~2.2 kW increase resulting from using half of thickness increase

Can We Vary WC Content Vertically in IB VV?

- Since largest contribution (10 kW) to heating comes from magnet segments around mid-plane (segments 5-10), we can think of reducing WC content in the top and bottom extremities of IB VV
- We assessed the impact of keeping the water content at 40% and changing the amount of B-SS replaced by WC

Material in IB VV shielding zone	Relative total IB TF heating	% reduction in TF heating
20% SS, 40% H_2O, 40% B-SS	1	0
20% SS, 40% H_2O, 40% WC	0.589	41.1%
20% SS, 40% H_2O, 30% WC, 10% B-SS	0.657	34.3%
20% SS, 40% H_2O, 20% WC, 20% B-SS	0.748	25.2%

Based on TF heating profiles, we estimated net effect on TF heating as follows:

1) Uniform 40% WC 5.5 kW
2) 40% WC in middle and 30% WC at top and bottom 5.3 kW
3) 40% WC in middle and 20% WC at top and bottom 5.0 kW

While option 2 could work, the savings in cost might not be significant to counterbalance added complexity

Publications

- M.E. Sawan, E.P. Marriott, M. Dagher, "Neutronics Performance Parameters for the US Dual Coolant Lead Lithium ITER Test Blanket Module," Proceedings of the 23rd IEEE/NPSS Symposium on Fusion Engineering (SOFE), May 31-June 5, 2009, San Diego, CA, ISBN No. 978-1-4244-2636-2, IEEE Cat. No. CFP09SOF-CDR. Also University of Wisconsin Fusion Technology Institute Report, UWFDM-1360 (May 2009).
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- T.D. Bohm, M.E. Sawan, and P. Wilson, "Radiation Streaming in Gaps Between ITER First Wall/Shield Modules," Fusion Science & Technology, vol. 56, pp 731-735 (August 2009).
- M.E. Sawan, P.P.H. Wilson, T. Tautges, L.A. El-Guebaly, D.L. Henderson, T.D. Bohm, E.P. Marriott, B. Kiedrowski, B. Smith, A. Ibrahim, and R. Slaybaugh, "Application of CAD-Neutronics Coupling to Geometrically Complex Fusion Systems," Proceedings of the 23rd IEEE/NPSS Symposium on Fusion Engineering (SOFE), May 31-June 5, 2009, San Diego, CA, ISBN No. 978-1-4244-2636-2, IEEE Cat. No. CFP09SOF-CDR. Also University of Wisconsin Fusion Technology Institute Report, UWFDM-1363 (May 2009).
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- T.D. Bohm, M.E. Sawan, B. Smith, and P.P.H. Wilson, "Peaking in SS Nuclear Parameters at Water Interface," University of Wisconsin Fusion Technology Institute, UWFDM-1376 (December 2009).
- M. Sawan, "Impact of ENDF/B-VII.0 Release on Fusion Evaluated Nuclear Data Library FENDL-2.1," Fusion Science & Technology, vol. 56, pp 766-770 (August 2009).
- M.E. Sawan, "Benchmarking FENDL-2.1 with Impact of ENDF/B-VII.0 Release on Nuclear Heating and Damage," Proceedings of the 23rd IEEE/NPSS Symposium on Fusion Engineering (SOFE), May 31-June 5, 2009, San Diego, CA, ISBN No. 978-1-4244-2636-2, IEEE Cat. No. CFP09SOF-CDR. Also University of Wisconsin Fusion Technology Institute Report, UWFDM-1361 (May 2009).