Aiming at Fusion Power — Tokamak

Design Limits of a Helium-cooled Large Area First Wall Module

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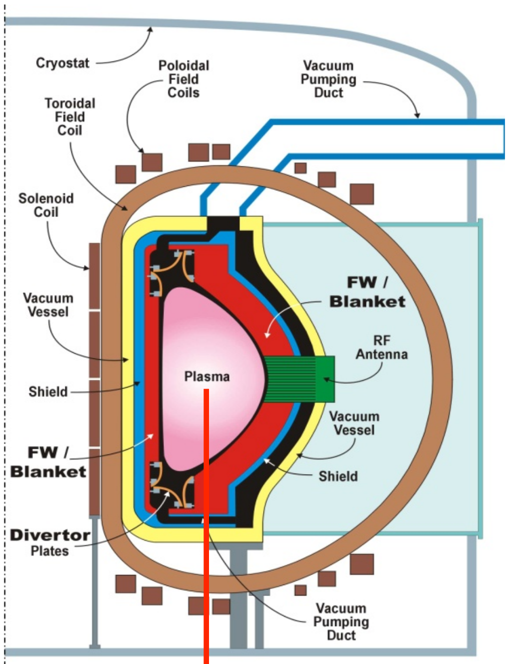
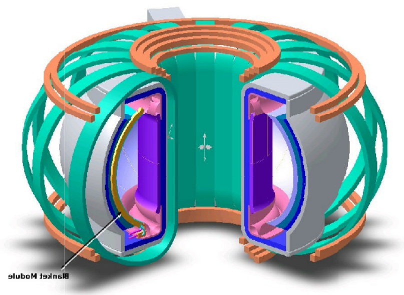
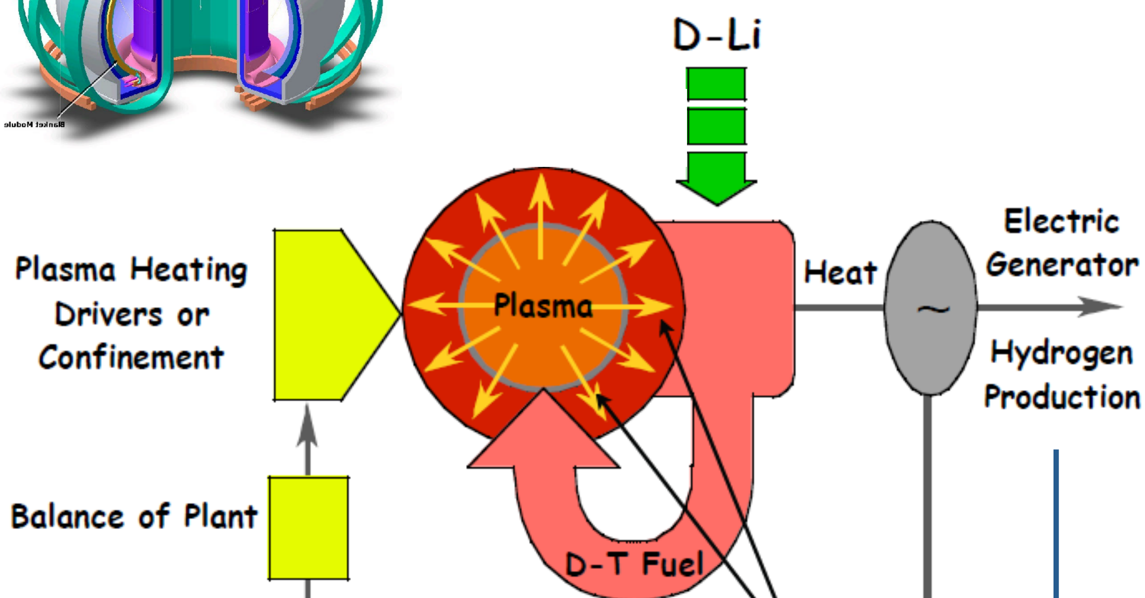
General Atomics

International Workshop on MFE Roadmapping in the ITER Era
Princeton University, Princeton, NJ, U.S.A., 7 -10 September 2011

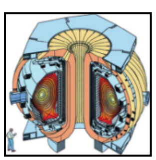
Looking from the Inside Out







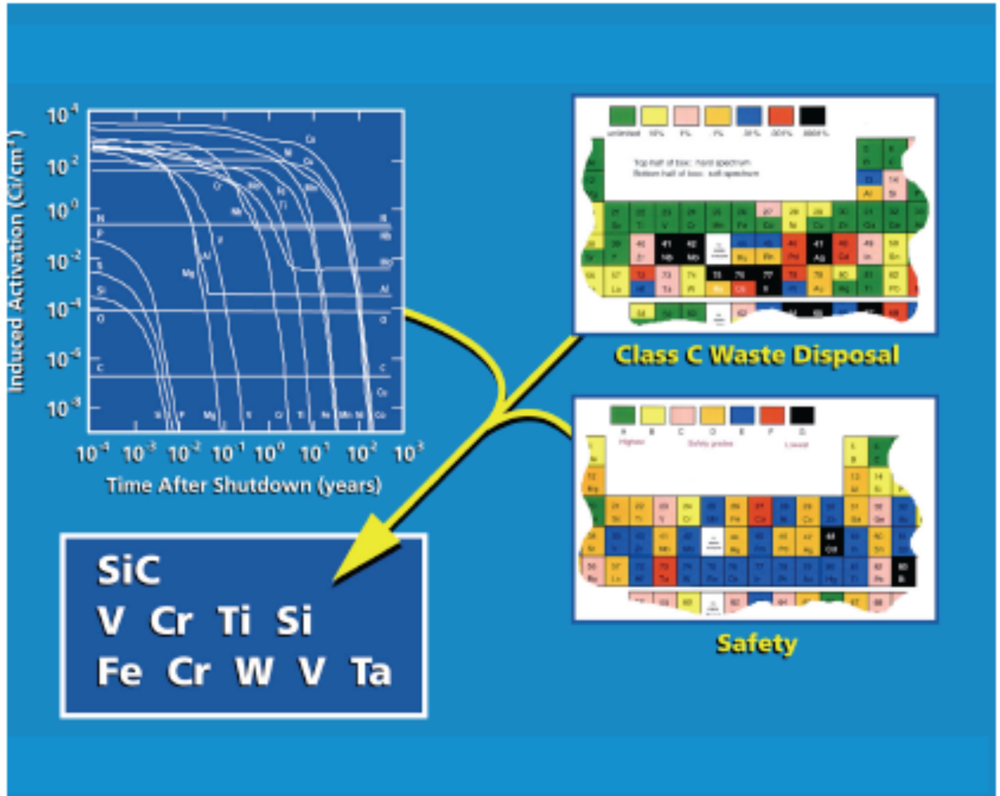
$$\left.\begin{aligned} n+{}^{6}Li &= {}^{4}He + T \\ n+{}^{7}Li &= {}^{4}He + T + n \end{aligned}\right\}\ \text{Blanket}$$



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

Low-Activation Structural Materials for Fusion

- Structural materials most strongly impact economic and environmental attractiveness of fusion power.
- Key issues: thermal stress, compatibility, safety, waste disposal, radiation damage, safe lifetime limits.
- Ti alloys, Ni base superalloys, and most refractory alloys are unacceptable for various technical reasons.
- Based on safety, waste disposal and performance considerations, the 3 leading candidates:
  - **Ferritic/martensitic steels**
  - **Vanadium alloys**
  - **SiC composites**



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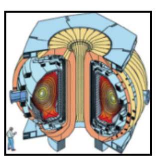
R.J. Kurtz PNNL



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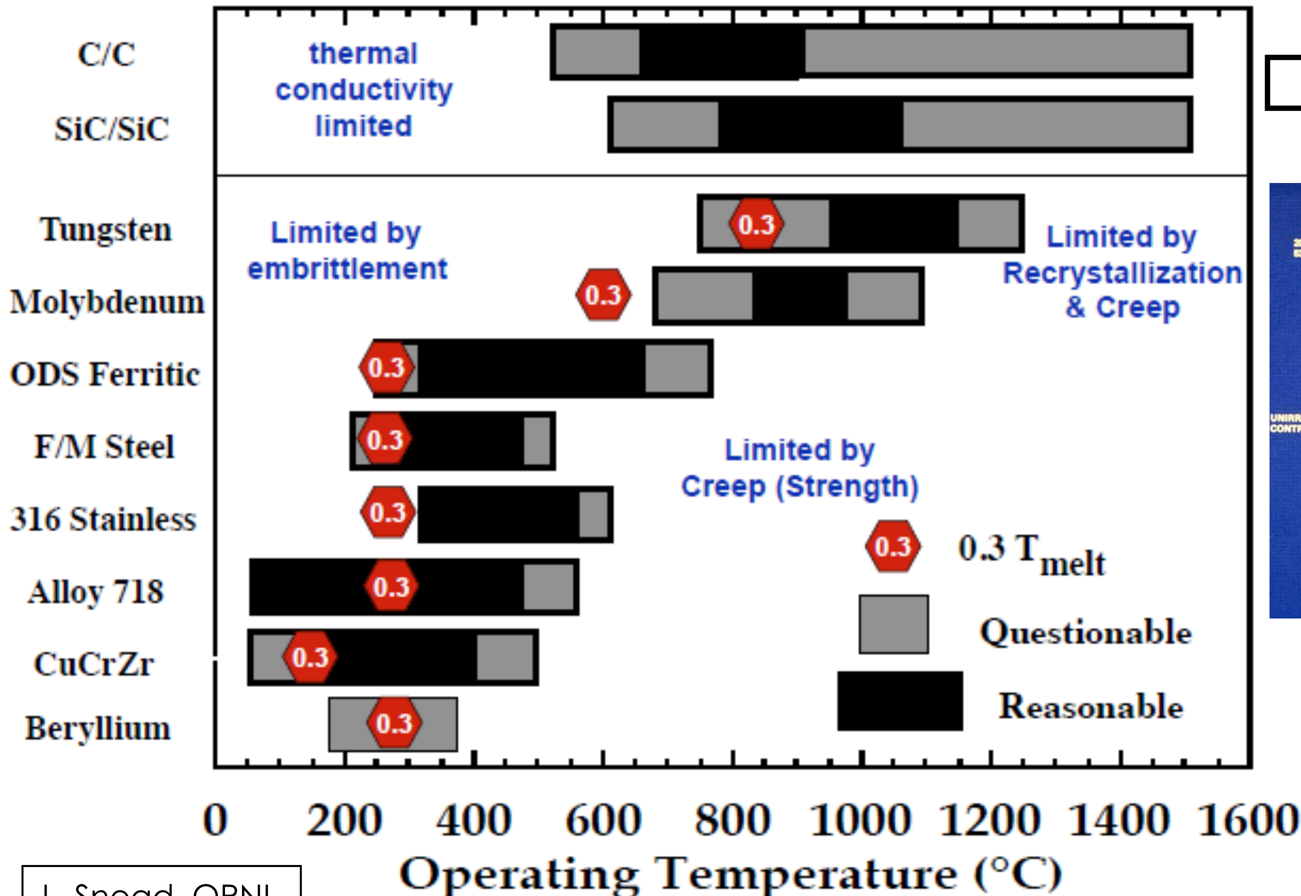
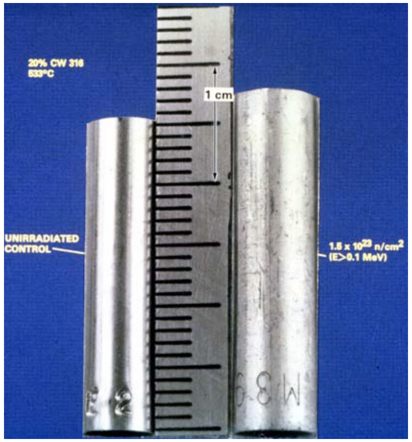
Operating Range, Irradiated Structural Materials

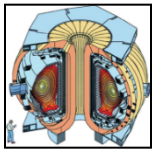


L. Snead, ORNL

Non-negligible Neutron radiation damage



SS 316 can
Have significant
Swelling at low
neutron fluence



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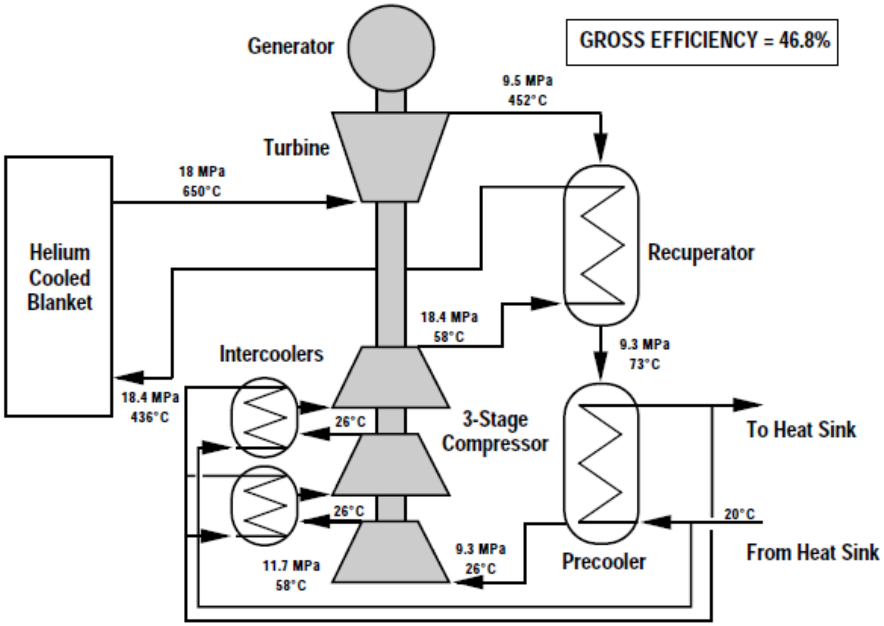
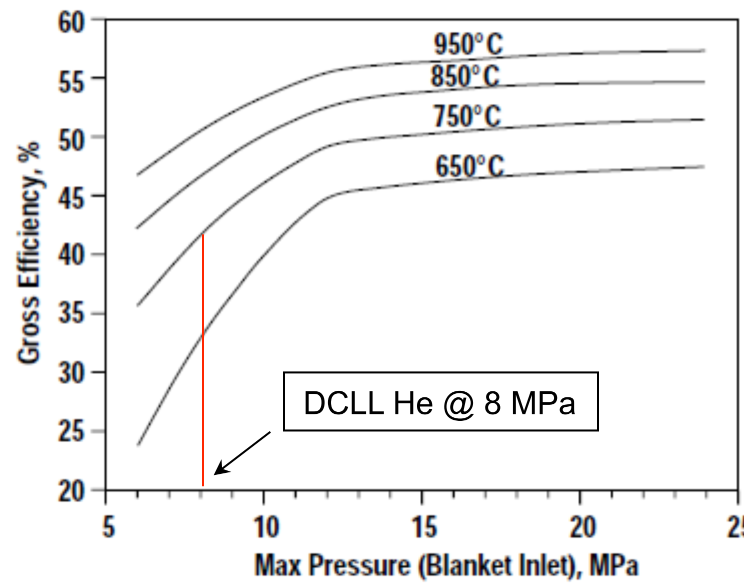
He Brayton Cycle for Fusion Application

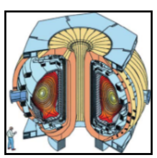
For comparison: Fort St. Vrain HTGR (1981-1989),
He pressure @ 4.8 MPa, Tin/Tout: 405°C/775°C, Steam Rankine cycle



Simplified Brayton Cycle (CCGT) at
He pressure of 18 MPa



Gross η_{th} versus He pressure and Tout



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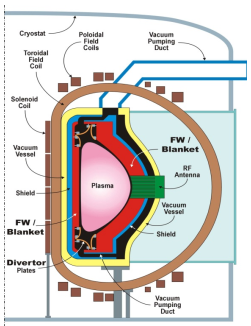
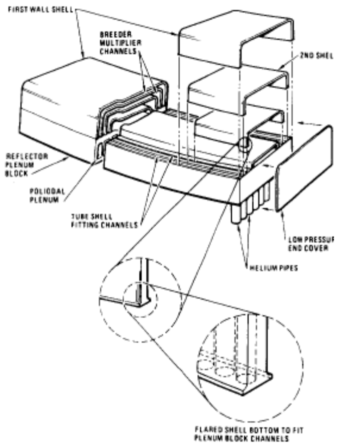
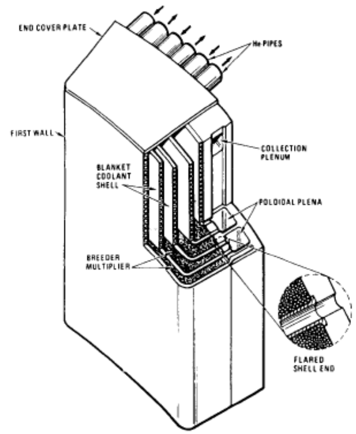
Advanced Materials and Blanket Designs

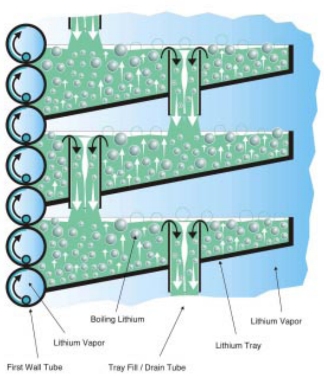


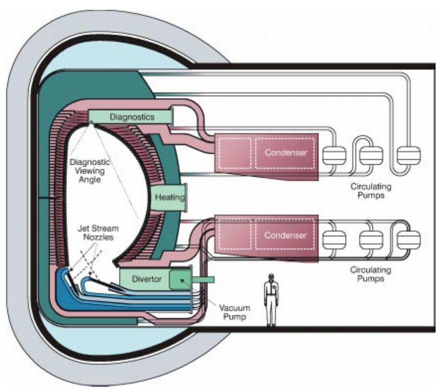




- SiC/SiC composite
- Low activation blanket design
- K_{th} not high enough to take high surface heat flux
- Radiation damage from 14 MeV neutrons can be severe

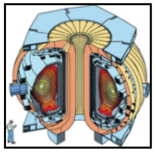
1994





- Li-W-alloy design
- Low pressure @0.05 MPa
- Neutron wall loading @ 10 MW/m^{2}, surface loading @ 2 MW/m^{2}
- 2-D TBR=1.37, Li Tout @ 1200 °C, liquid can be converted to He @ 700-800 °C
- Li vapor can be converted to He @ ~1000 °C, resulting in η_{th}-gross ~57.7%
- Don't know how to fabricate with W-alloy

2002



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DIII-D NATIONAL FUSION FACILITY

109-11/CW/rs

Test Blanket Systems Testing in ITER Era

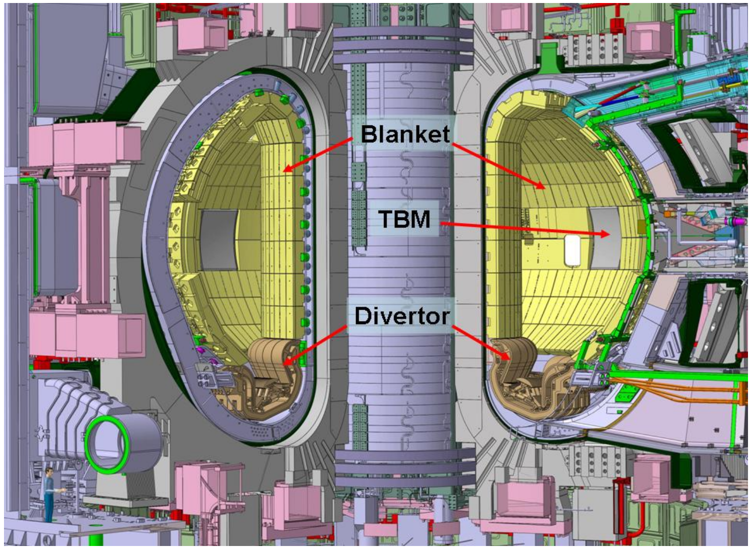
- **Tritium Breeding Blankets are complex components, subjected to very severe working conditions, needed in DEMO but not present in ITER ➔ ITER is a unique opportunity to test breeder blanket mockups in DEMO-relevant conditions : Test Blanket Modules (TBMs)**

- **It is one of the ITER missions : *"ITER should test tritium breeding module concepts that would lead in a future reactor to tritium self-sufficiency, the extraction of high grade heat and electricity production."***

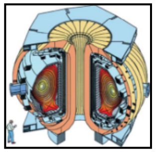
- **All the activities related to this mission form the "TBM Program". All ITER Members participate in the TBM Program**

Definition:

➔ Each TBM and the associated ancillary systems (cooling system, tritium extraction system, measurement systems, etc..) are defined as Test Blanket System (TBS)



L. Giancarli et al.,TBSs/ITER Interfaces, ISFNT-9, Dalian, Oct. 14, 2009



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Test Blanket Systems Testing in ITER

The 6 TBSs to be installed in ITER during H/He phase are the following:
(PM : Port Master, TL : TBM Leader)

Port No. and PM	TBM Concept	TBM Concept
16 (PM : EU)	HCLL (TL : EU)	HCPB (TL : EU)
18 (PM : JA)	WCCB (TL : JA)	DCLL (InCo : US (KO))*
2 (PM : CN)	HCCB (TL : CN)	LLCB (TL : IN)

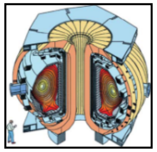
*InCo=Interfaces Coordinator (=TBM delivery not committed)

HCLL : Helium-cooled Lithium Lead, **HCPB** : He-cooled Pebble Beds (Ceramic/Beryllium)
WCCB : Water-cooled Ceramic Breeder (+Be), DCLL : Dual Coolant (He & LiPb) Lithium Lead
HCCB : He-cooled Ceramic Breeder (+Be), **LLCB** : Lithium-Lead Ceramic Breeder (DC type, He & LiPb)

In order to be representative of the corresponding DEMO breeding blanket, all Test Blanket Modules are required to use **Ferritic/Martensitic steels**

➔ **These types of steels are Ferromagnetic** and therefore their presence induces a magnetic field that interferes with the main ITER magnetic field needed for confining the plasma

L. Giancarli et al.,TBSs/ITER Interfaces, ISFNT-9, Dalian, Oct. 14, 2009



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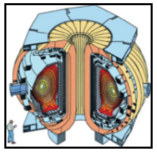
Solid Breeder Blankets for DEMO from ITER Parties

FW surface heat flux ~0.5-0.7 MW/m^2 and large modules

	EU	JA	RF	CHN	KOR	US	JA
Label	HCPB-2003	DEMO 2001-He	CHC	HCCB-DEMO			DEMO2001 WCCB
Breeder	Li4SiO4 (Li2TiO3)	Li_2TiO_3, other Li ceramics	Li_4SiO_4	Li_4SiO_4 (Li_2TiO_3)	Li4SiO4	Li_4SiO_4 (Li_2TiO_3)	Li_2TiO_3, other Li ceramics
FW heat flux, MW/m^2	**0.5 peak**	**0.5, peak 1**	**0.4 0, peak 0.7**	**0.4, peak 0.7**	**< 1**	**0.5**	**0.5, peak 1**
Neutron wall loading, MW/m^2	2.4 peak	3.5, peak 5	2.7, peak 4.4	2.64, peak 4-5	>2	2-3	3.5, peak 5
TBR	1.14	>1.05	Self Sufficient @1.06	1.05-1.11	~1.05	>1.0	> 1.05
Structural material	EUROFER	F82H	FS 9CrMoVNb	CLAM	EUROFER	RAF	F82H
Coolant , pressure, MPa	He @ 8	He @ 10	He @10	He @ 8	He @ 8	He @ 8	Water @ 15, 25
Operating temperature. °C	300-500	220-500	300-500	300-500	300-500	300-500	280-325 280-510
Characteristic Dimension, m	**2 x 2**	**1 x 2**	**1-1,5 toroidal 6.4 poloidal**	**1 x 2.98**	**TBD**	**1 to 2 x 1 to 2**	**1 x 2**

Ref: TBWG during ITER ITA phase, Working Subgroup 1, "Assessment report on Solid Breeder Blankets", Compiled by: L.V. Boccaccini, March 2006



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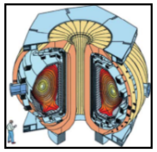
Liquid Breeder DEMO Blankets from ITER Parties

FW surface heat flux ~0.5-0.7 MW/m^2 and large modules

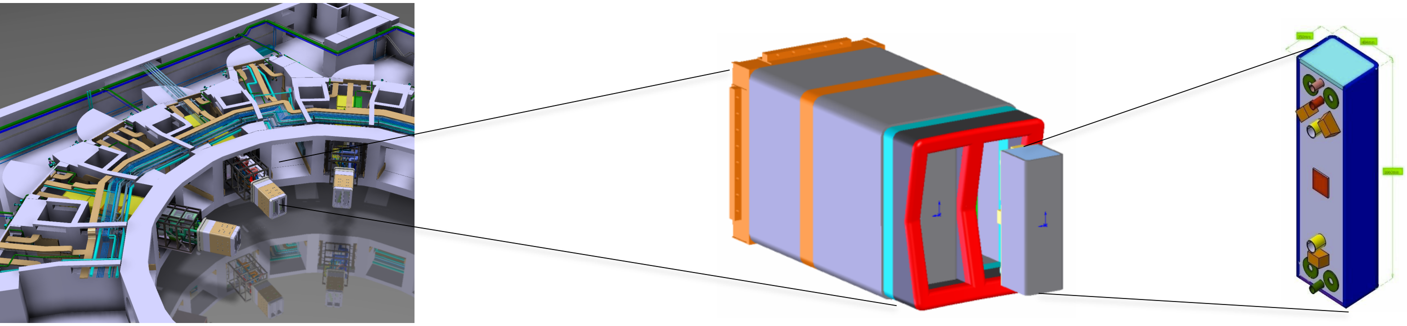
	EU	RF	CHN	KOR	US	India
Label	HCLL	Li/V	DFLL	HCML	DCLL	LLCB
Breeder	PbLi	PbLi	PbLi	Li	PbLi	PbLi and Li_2TiO_3,
FW heat flux, MW/m^2	**0.5 peak**	**0.7 peak**	**0.7 peak**	**< 1**	**0.5 peak**	**0.5, peak**
Neutron wall loading, MW/m^2	2.4 peak	3.4, peak	3.54 peak	>2	3 peak	1.7 average
3-D TBR	1.15	1.05-1.09	1.2	~1.05-1.1	1.17	> 1.2
Structural material	EUROPER	V-Cr-Ti	CLAM	ODS FFS	F82H	IN-LAFMS
Coolant , pressure, MPa	He @ 8	Li @ 1	He @ 8	He @ 8	He @ 8	He @ 8
Operating temperature. °C	300-500 (He)	350-600 (Li)	300-450 (He) 480-700 (PbLi)	250-350/550(Li)	350-450 (He) 460-700 (PbLi)	350-480 (He) 370-480 (PbLi)
Characteristic Dimension, m	**2 x 1.8**	**1.25-1.45 x 2.9-7.3**	**2x2.2**	**TBD**	**1 to 2 x 1 to 2**	**~2x2**

Ref: C. Wong et al., "Overview of Liquid Metal TBM concepts and programs", Fus. Engr & Design 83 (2008) 850-857



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

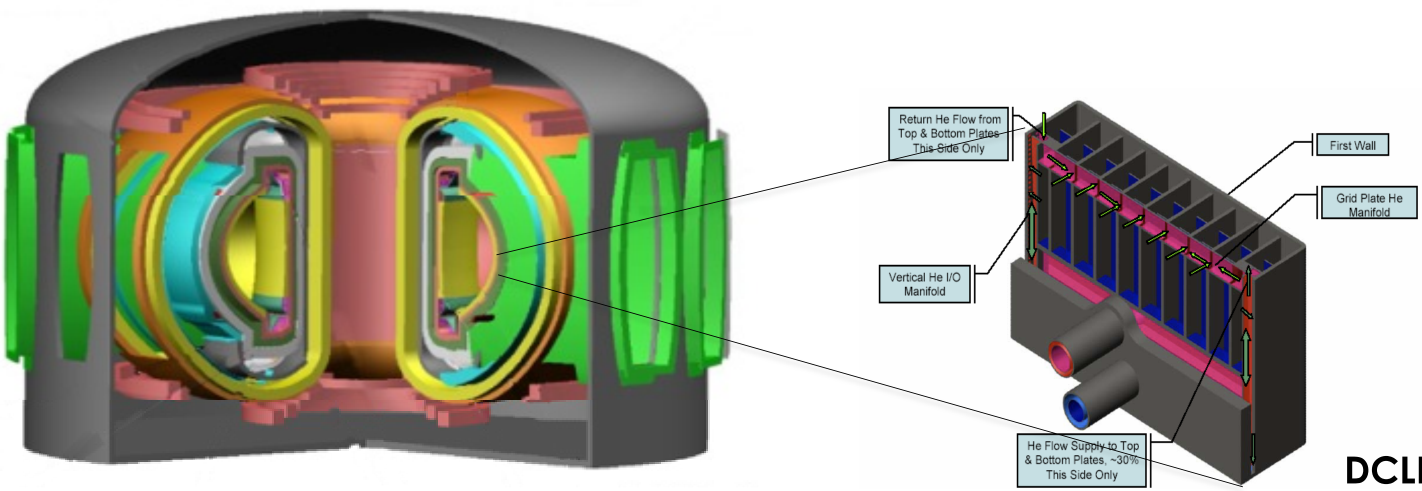
0.5 m width for ITER TBM and m size width for DEMO/Reactor Blanket



ITER

0.484 m (W)
1.66 m (H)
0.35 m thick

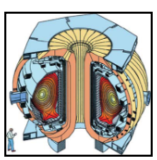
DCLL TBM



ARIES-RS

~1.5 m (W)
~1 m (H)
~0.8 m thick

DCLL DEMO module



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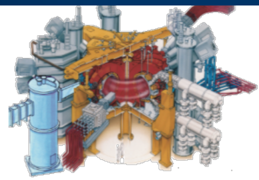
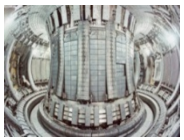
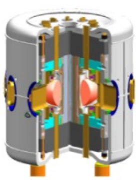
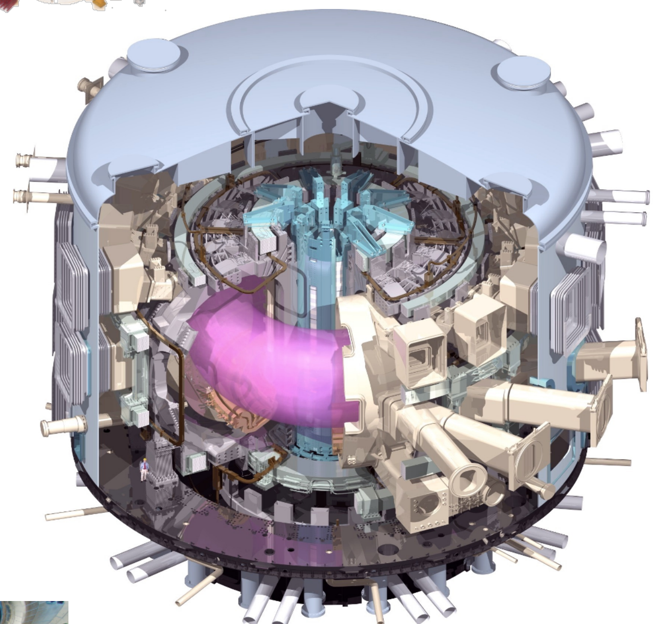
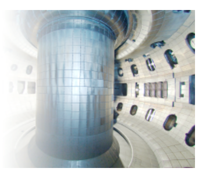
ITER has Helped to Focus World Fusion Resources it has Also Identified Critical Issues when Extended to DEMO





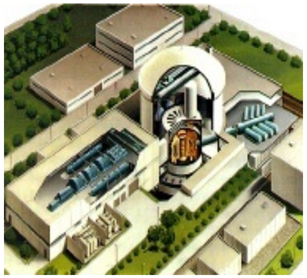




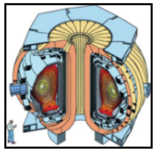


? ?

FNSF → **DEMO** →



Fusion Power Reactor



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Distribution of ITER FW Panel Design Heat Load

Steady state:

$q_{||} \sim 8$ MWm^{-2}, $\lambda_{q||} > 4.0$ cm
$q_{||} \sim 24$ MWm^{-2}, $\lambda_{q||} > 2.5$ cm
(ELMs)

Disruptions

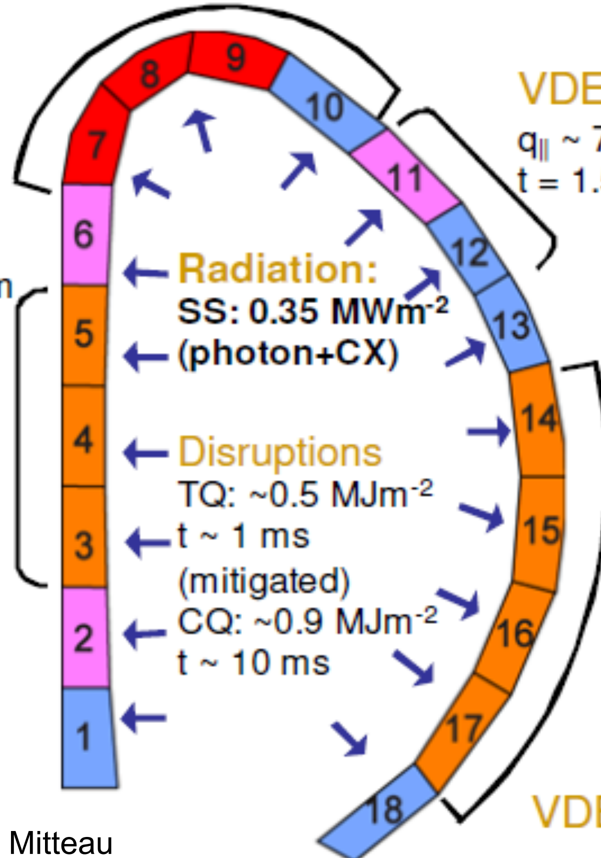
$q_{||} \sim 45\text{-}120$ MJm^{-2}, $\lambda_{q||} > 20$ cm
t = 3.0-6.0 ms

Start-up:

$q_{||} \sim 25$ MWm^{-2}, $\lambda_{q||} \sim 5.0$ cm
Several seconds

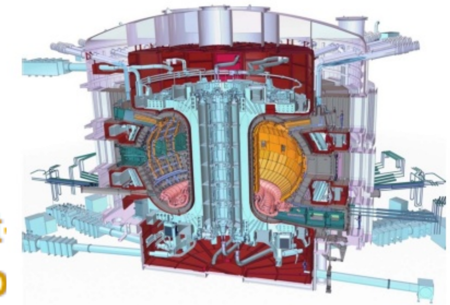
Confinement transients

$q_{||} \sim 250$ MWm^{-2}, ~2-3 secs



VDE (up):

$q_{||} \sim 70\text{-}270$ MJm^{-2}, $\lambda_{q||} > 3.0$ cm
t = 1.5-3.0 ms



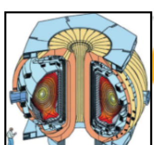
Start ramp

$q_{||} \sim 40$ MWm^{-2},
$\lambda_{q||} > 1.2$ cm
Several seconds

VDE (down):

$q_{||} \sim 90\text{-}300$ MJm^{-2}, $\lambda_{q||} > 3.0$ cm

PFMC-13-9-13 May 2011-Rosenheim-R. Mitteau



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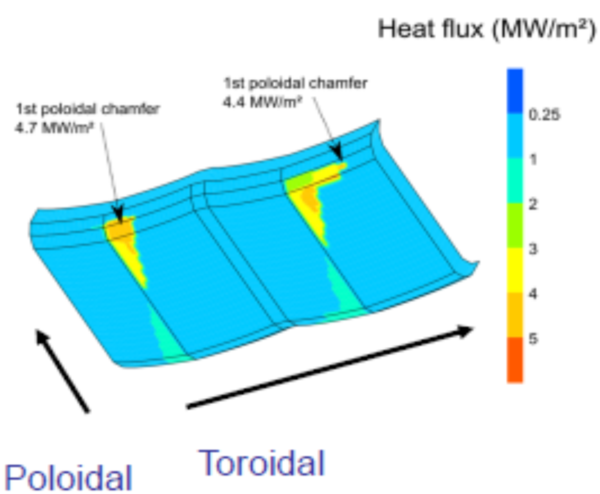
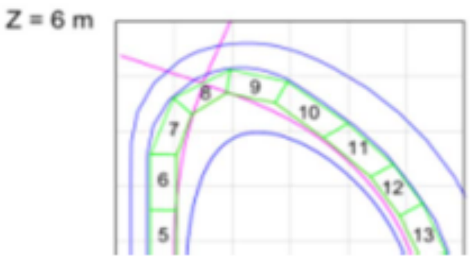
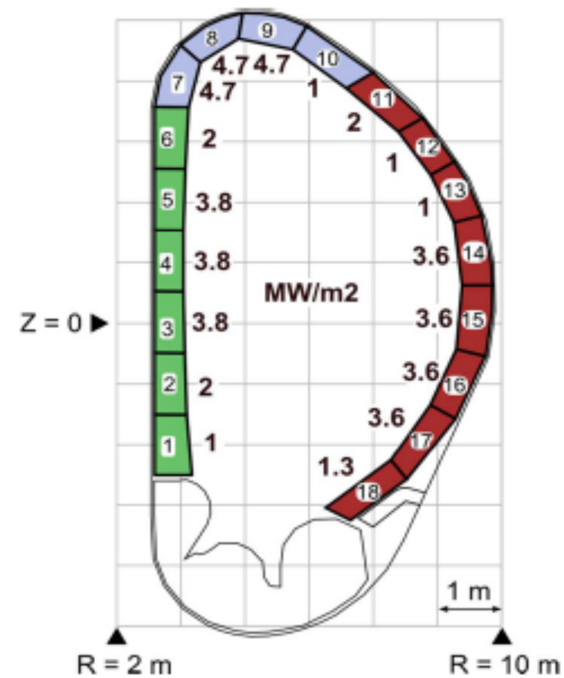
Design Heat Load on ITER Blanket/Shield

Outboard flat top heat load (550 MW)

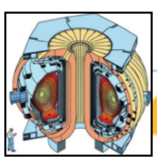






- Group 1 : 1 – 2 MW/m²
  Normal heat flux panels
- Group 2 : 3.5 – 5 MW/m²
  Enhanced heat flux panels

R. Mitteau @ PFMC-13



T Hirai, PFMC-13 Rosenheim Germany May 2011

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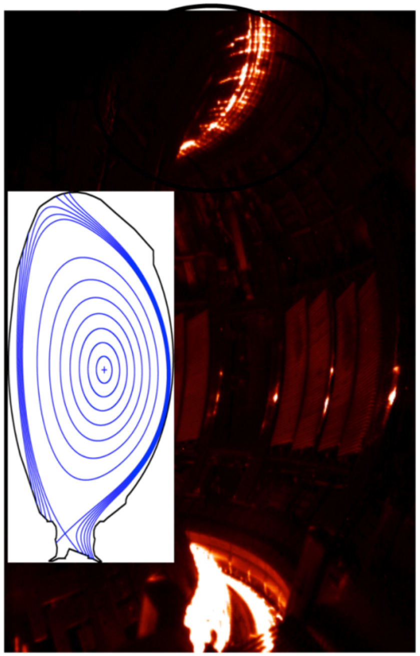
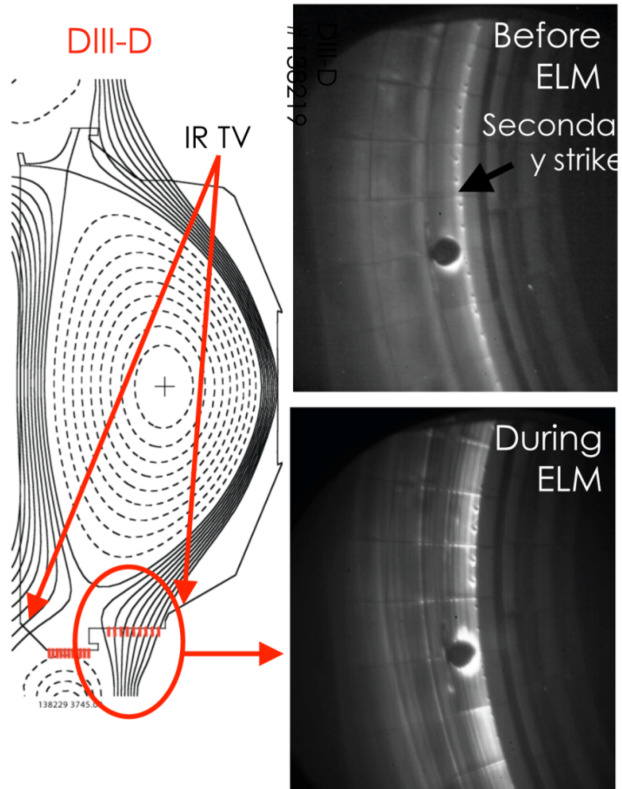
PSI Main Chamber Edge-localized Mode (ELM) Loads

- **Clearly present in higher triangularity configurations**

JET



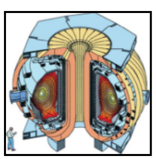
DIII-D



ELM free H-mode and QH mode: are also being studied

R.A. Pitts, 19th PSI conference May 2010, San Diego

J. G. Watkins, 19th PSI conference Poster P2-66



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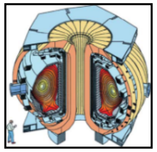
Basic Questions on the DEMO Chamber Wall Heat Flux

- Based on the physics and the projection of plasma edge control, what would be the corresponding projected steady state heat flux distribution for FNSF and DEMO?

  (The projection has to be extended from a 500 MW_{fusion} ITER device to a ~3000 MW_{fusion} DEMO device)

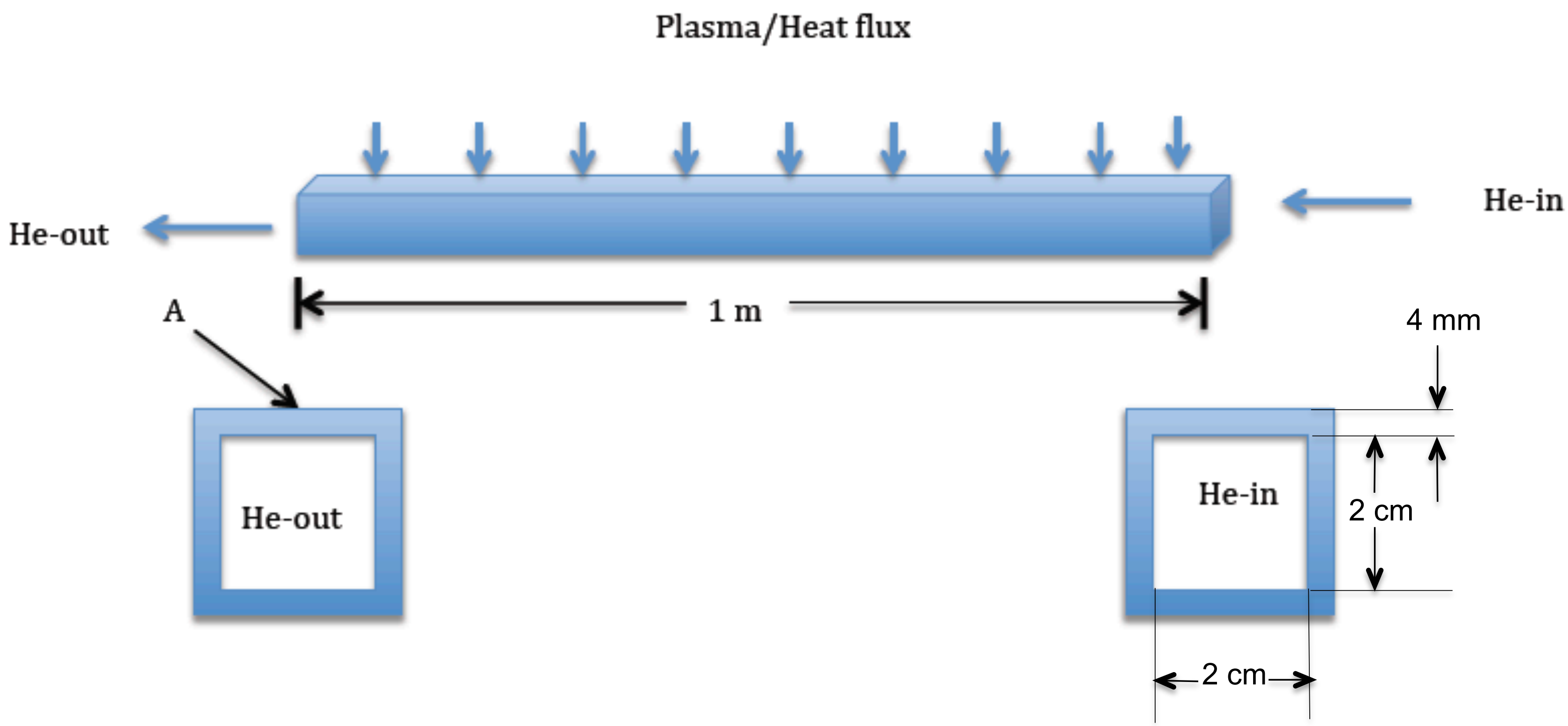
- What would be the temporal pulse and local heat flux distributions that the FW will have to be designed for, including startups/shutdowns and transient events?

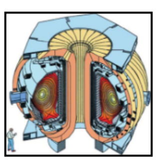
- What are the heat removal design limits of a conventional large area helium-cooled FW/blanket design?



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

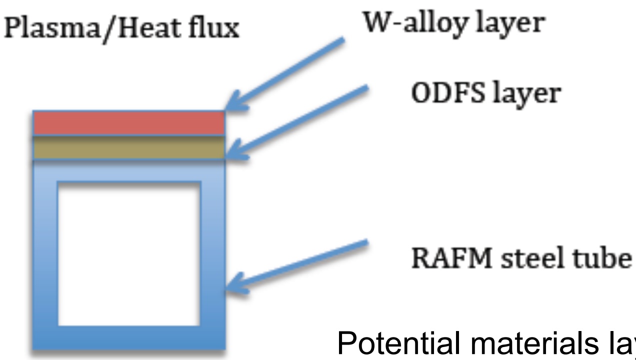
Simplest Input Coolant Channel Geometry for DEMO FW





Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

Input Chamber Wall Materials K_{th} and Temperature Limits



	RAFM	ODFS	W-alloy
K_{th}, W/m.k	20	20	50[1]
T_{min}[2] °C	350	350	700
T_{max}[3] °C	550	700	1300

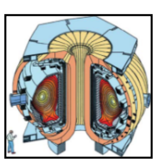
Potential materials layer, thickness TBD
Interface materials undefined and assumes perfect contact

[1] This is a conservative value reduced from a value of ~100 W/m.K at 1000° C

[2] Lower temperature limit based on radiation hardening/fracture toughness embrittlement [4]

[3] Upper temperature limit based on 150 MPa creep strength (1% in 1000h) [4]

Ref [4] S. J. Zinkle and N.M. Ghoniem, Fus. Eng. Des. 51-52 (2000) 55.

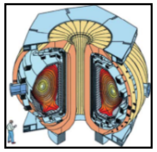


Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

He Coolant Power Input Parameters

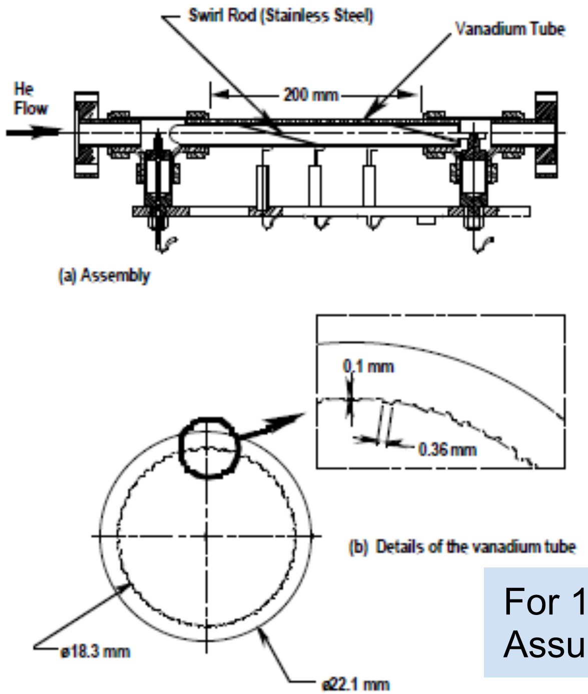
Helium: Pressure @ 8 MPa
- **He Tin@ 350 °C**
- **He velocity @ 100 m/s**
- **He thermal/physical properties as a function of local Temperature**

Power input:
- **Heat flux**
- **Volumetric power from a Γ_n of 3 MW/m^2**



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

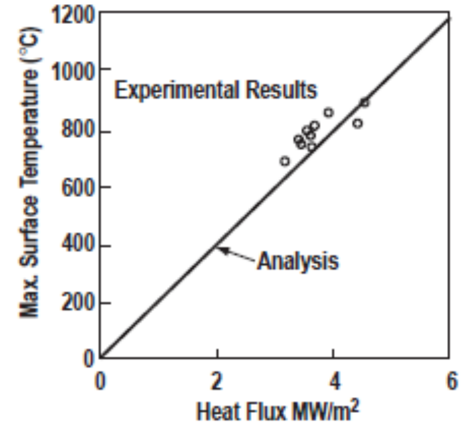
He Coolant Heat Transfer Enhancement and Assumptions



(a) Assembly
(b) Details of the vanadium tube

Heat Transfer Enhancement Methods

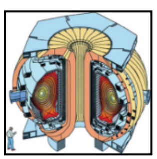
		hx	fx	Ref.
1	Microfins	8	30	2
2	Porous media	5	20	3
3	Jet impingement	3	7	4
4	Particulate Addition	10	30	5
5	Swirl Tape	2	4	6
6	Two D roughness	1.8	3.8	7
7	Three D roughness	3.	7	7
8	Swirl Rod Insert	2.5	5	8
8A	Swirl Rod Insert with 2 D roughness	3.5	7	8



[2] C.B. Baxi, "Design, Fabrication and Testing of a Helium Cooled Module," Fusion Engineering and Design, **28** (1995), 22–26.
[3] J.H. Rosenfeld, "Test results from a Pumped Single Phase Porous Metal Heat Exchanger," SPIE Vol. 1997, HHF Engineering II, 1993.
[4] R. Gordon and J. Cobonque, "Heat Transfer Between a Flat Plate and Jets of Air Impinging On it," International Developments in Heat Transfer, pp. 454.
[5] A. Shimuzu, "Gas Solid Suspension Cooled Fusion Power Reactor Concept," Presented at US-Japan Workshop on Fusion High Power Density Devices, February 17–21, 1997, San Diego, California.
[6] J. Schlosser, *et al.*, "Development of High Heat Flux Components for Continuous Operation in Tokamaks ," 4th Symposium on Fusion Engineering, San Diego, California.
[7] S. Kakac, *et al.*, "Handbook of Single Phase Heat Transfer," J. Wiley & Sons, 1987.

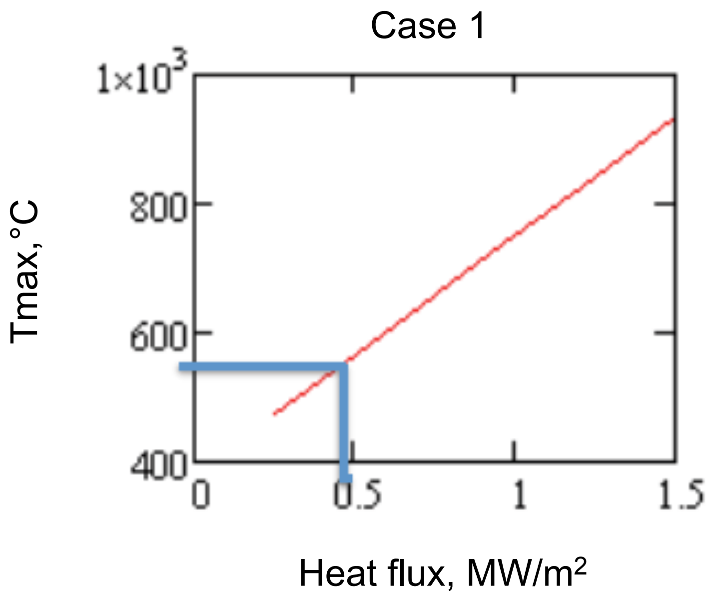
For 1-sided 2-D roughening
Assume: h=2*h, f=1.5*f

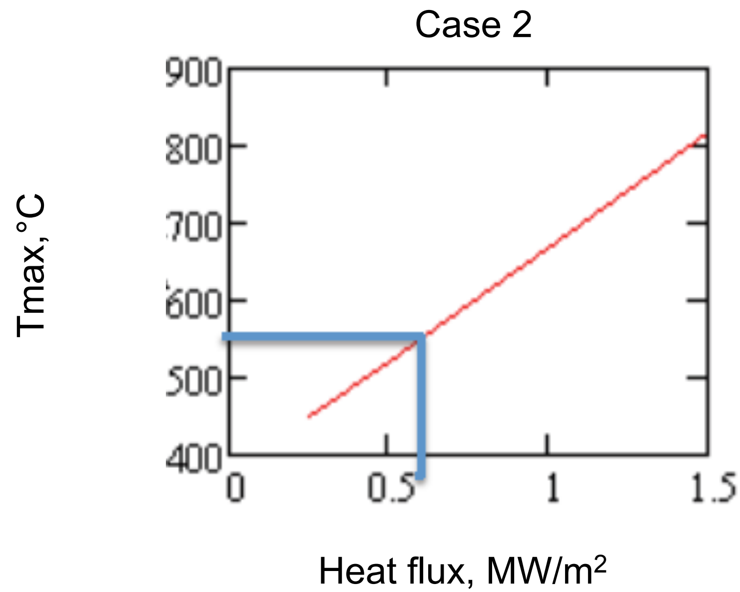
h: heat transfer coefficient
f: friction factor



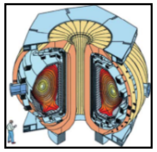
Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

Case 1: 4 mm FS without h enhancement, L=1m, φ_{max}~ 0.49 MW/m^2
Case 2: 4 mm FS with h enhancement, L=1m, φ_{max}~ 0.62 MW/m^2





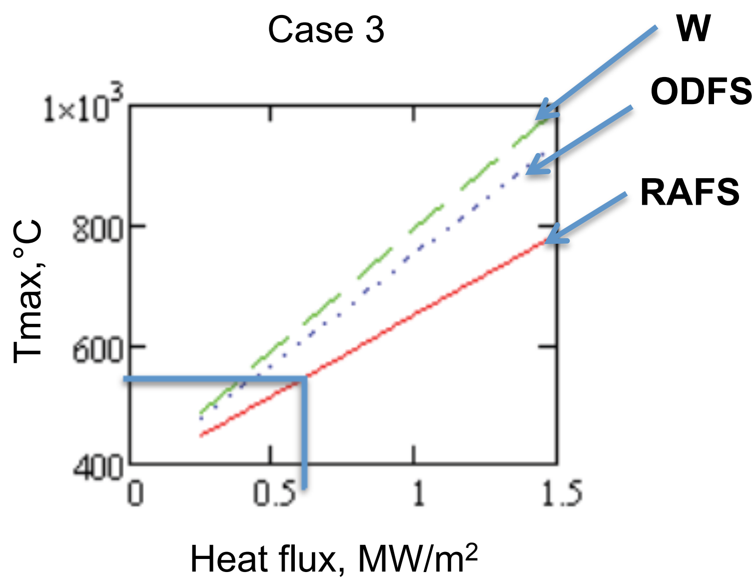
FW heat flux, MW/m^2	0.25	0.5	0.75	1.0	1.25	1.5
Case 1, 4mm wall without h enhancement, L=1 m	476	567	658	750	842	934
Case 2, 4 mm wall with h enhancement, L=1 m	450	522	596	669	742	816

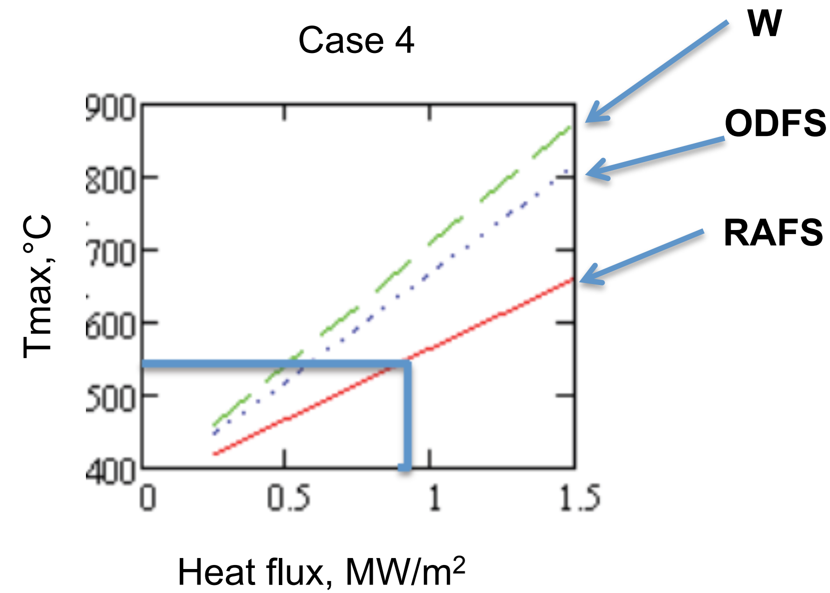


Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

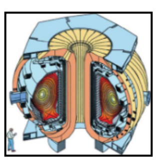
Case 3: 2 mm each FS, ODFS, W without h enhancement, L=1m, φ_{max}~ 0.63 MW/m^2

Case 4: 2 mm each FS, ODFS, W with h enhancement, L=1m, φ_{max}~ 0.92 MW/m^2





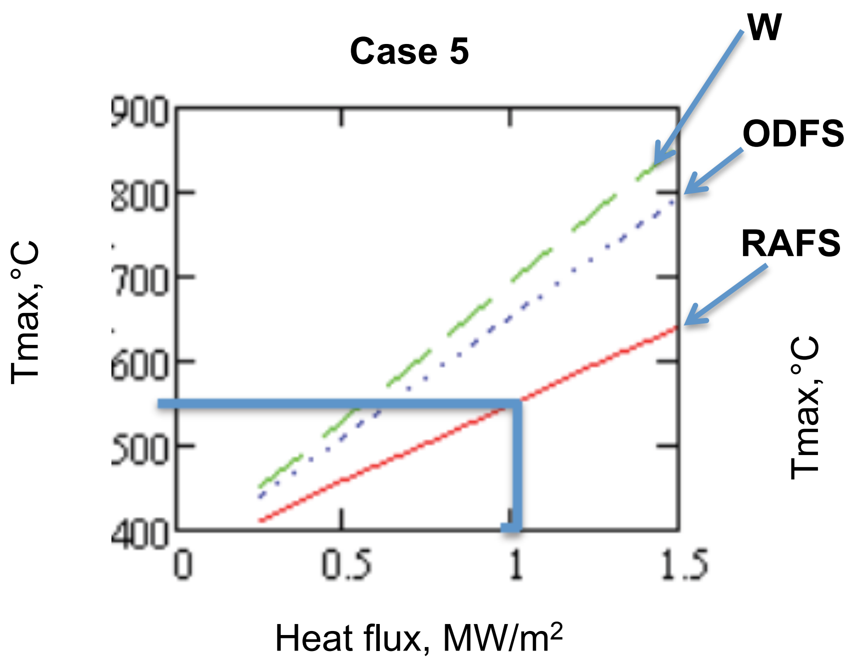
FW heat flux, MW/m^2	0.25	0.5	0.75	1.0	1.25	1.5
Case 3, 2 mm FS, ODFS & W without h enhancement, L=1m	450	516	582	649	716	783
Case 4, 2 mm FS, ODFS & W with h enhancement, L=1 m	420	468	516	565	613	662

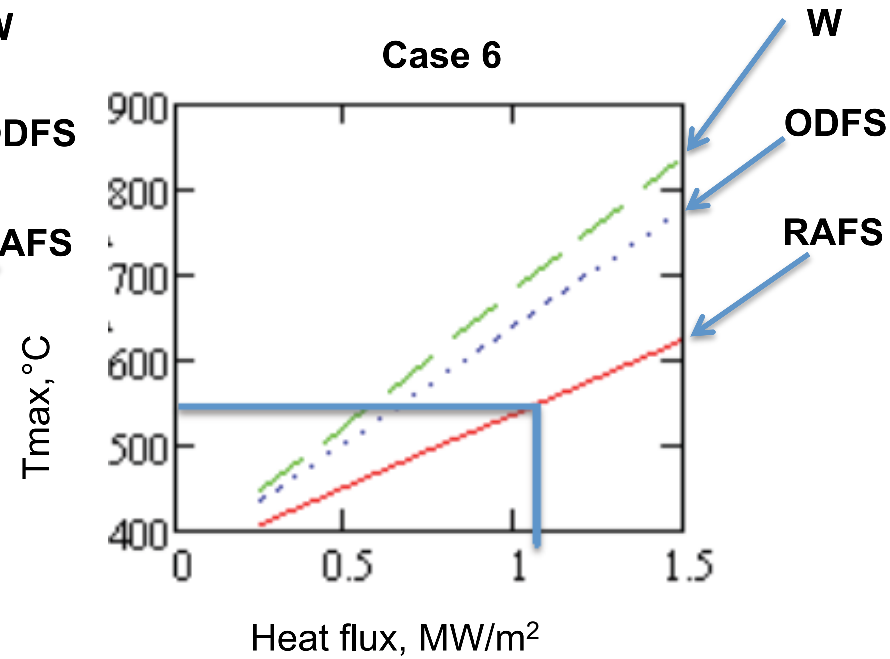


Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

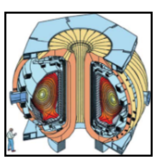
Case 5: 2 mm each FS, ODFS, W with h enhancement, L=0.5 m, $\varphi_{max} \sim 1.02$ MW/m^2

Case 6: 2 mm each FS, ODFS, W with h enhancement, L=0.1 m, $\varphi_{max} \sim 1.1$ MW/m^2



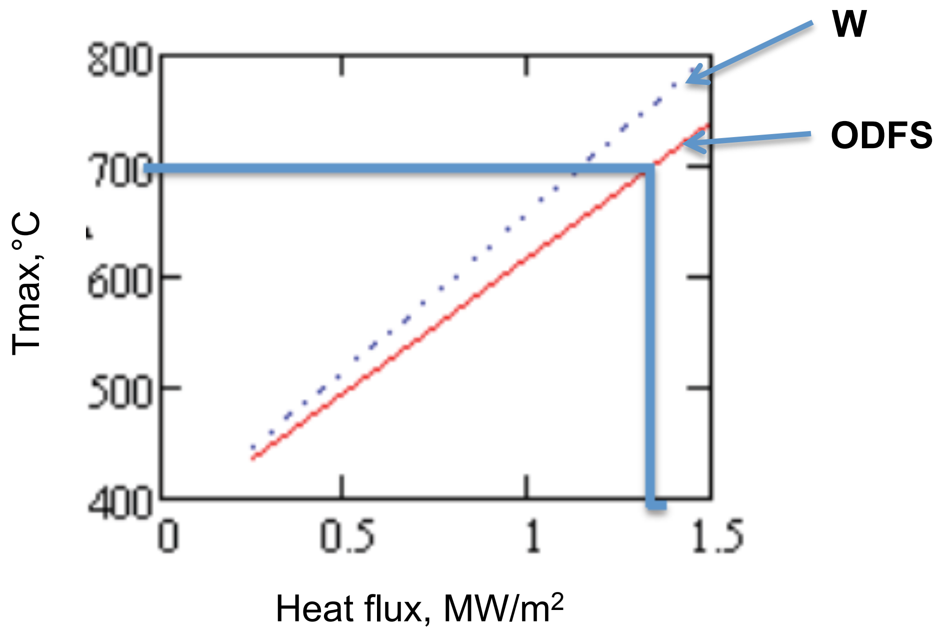


FW heat flux, MW/m^2	0.25	0.5	0.75	1.0	1.25	1.5
Case 5, same as case 4, L=0.5 m	414	459	504	550	595	640
Case 6, same as case 4, L=0.1 m	408	452	495	538	581	624



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils>Power conversion->General

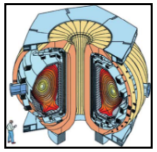
Case 7: 3 mm ODFS + 2 mm W, Enhanced h and l=1 m



With the use of ODFS as the structural material the φ_{max} ~1.35 MW/m²

Example:
@ Location A and φ=1.25 MW/m²
h=13690 W/m².K
He Tin=350 °C
He Tout =382.6 °C
ΔT-film= 100.9 °C
ΔT-ODFS=194.2 °C
Tmax-ODFS=677.7 °C

FW heat flux, MW/m²	0.25	0.5	0.75	1.0	1.25	1.5
Case 7, 3 mm ODFS and 2 mm W wall with heat transfer enhancement of 2, L=1 m	435	495	556	616	677	739



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

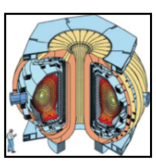
Observations from Heat Transfer Results

- **For the conventional/typical helium cooled first wall channel design, with RAFM steel as the structural material, and with heat transfer enhancement, it is projected that the design can handle surface heat flux of ~0.62 MW/m^2**

- **For multilayer designs with the use of RAFM steel, ODFS and W-alloy layers, the heat flux handling capability can be extended to ~0.92 MW/m^2**

- **The reduction of flow path length could improve the heat flux removal capability to ~1.1 MW/m^2, but the increased pressure drops and pumping power and the corresponding increase in the number of blanket modules make this an unattractive approach**

- **With the successful development of ODFS as the structural material and W-alloy is used as the coating material, the heat removal capability could be extended to ~1.35 MW/m^2**



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

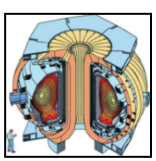
Observations from Heat Transfer Results

- **Based on the ITER design specific surface heat flux design of 1-5 MW/m^2 with a 500 MW-fusion machine, there is a disconnect when compared to the conventional/typical projection of ~0.5 MW/m^2 for DEMO, which could be a 3000 MW-fusion machine**

- **Limitation on surface heat removal capability is due to the minimum temperature limit of 350 °C for RAFS and ODFS, and the corresponding maximum allowed temperatures limits of 550 °C and 700 °C, respectively**

- **The use of ODFS and W-alloy will have higher heat removal capabilities but their use as structural material will have to be demonstrated**

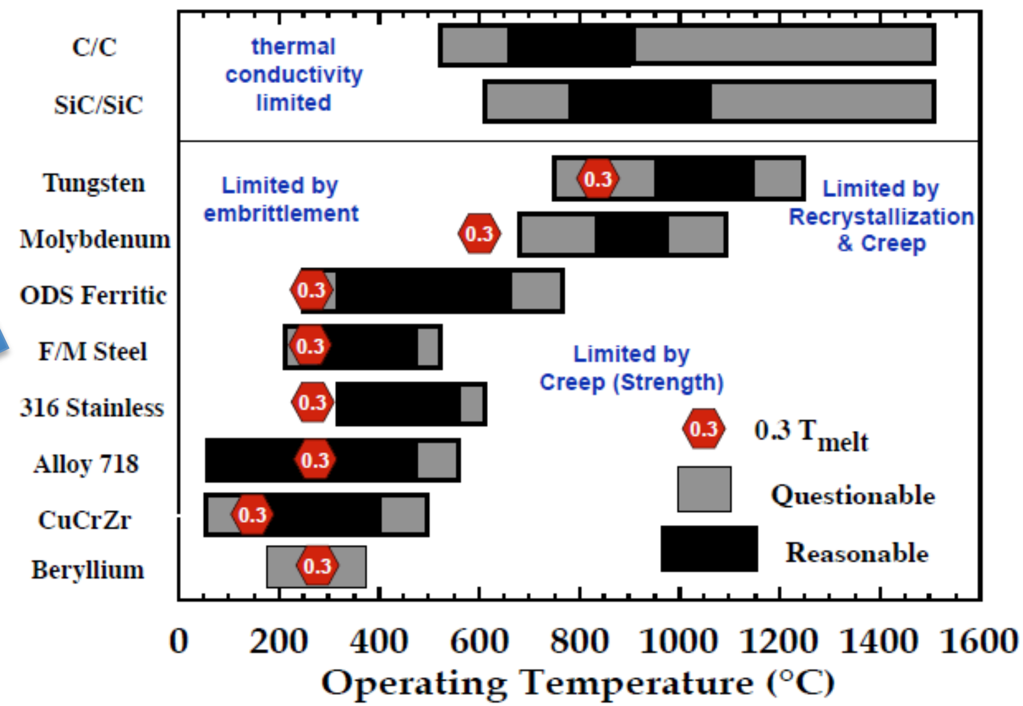
- **This presentation is mainly focused on the steady state operation of the FW design; the corresponding assessment on startup, shutdown and different transient effects will have to be performed**

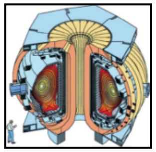


Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

Key Observation

The heat flux removal capability of Helium cooled chamber wall design is limited by the present sole acceptable structural material of Reduced Activation Ferritic Martinsitic (RAFM) Steel





Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

Recommendations

- A concerted effort is needed to quantify the chamber wall heat flux distributions from present tokamaks as a function of different modes of plasma operation and changes in SOL thickness. The aim is to identify a uniform radiation design and corresponding regime of high performance.

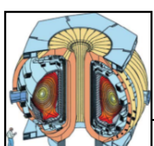
- A joint core/boundary physics and FW/blanket design study should be initiated to look into design details, including: projection of optimum radiation uniformity, impacts to front face surface topology, heat removal module design and support, including impacts to tritium breeding for FNSF and DEMO.

- In concert with the above, a detailed review should be performed on the capability of helium-cooled first wall designs for FNSF and DEMO. Innovative high heat flux removal designs should be considered.

- For structural material development, ODFS structure and W coating should be emphasized in coordination with the developmental schedule of ITER-TBM, FNSF and DEMO.

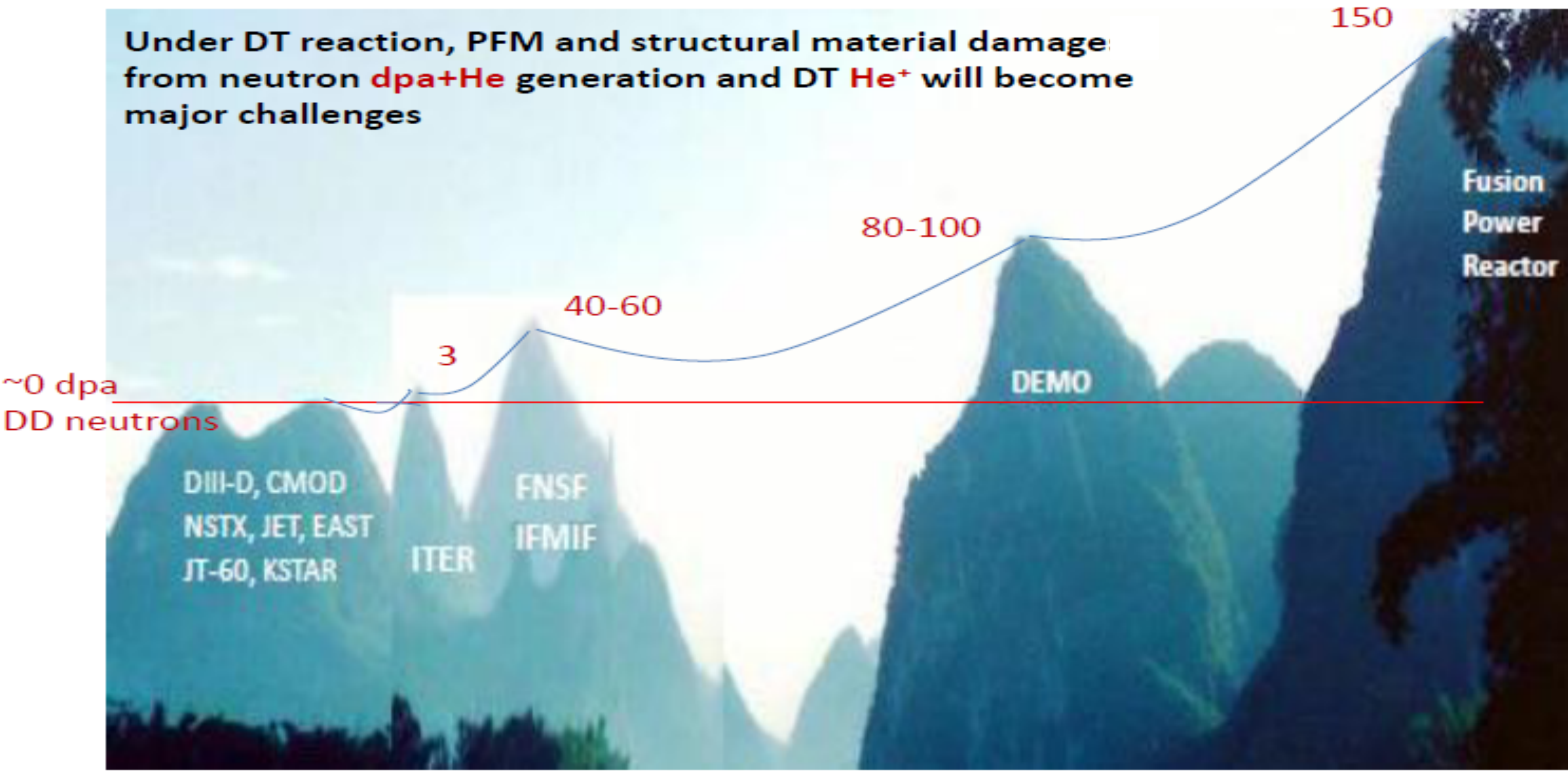
- If the chamber wall surface heat flux is projected to be higher than 1 MW/m^2, water should be re-considered, perhaps for even partial coverage, as the first wall and divertor coolant for the FNSF and DEMO designs* (*It should be noted that 1 mm thickness increase of metallic FW is ~0.5% reduction in TBR[1])

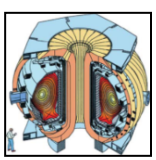
[1] M. E. Sawan, Fu. Engr. & Design 81 (2006) 1131-1144



Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

We Need to Provide Technical Connections Between Present and Future Tokamaks High Neutron Fluence will Limit Material Performance

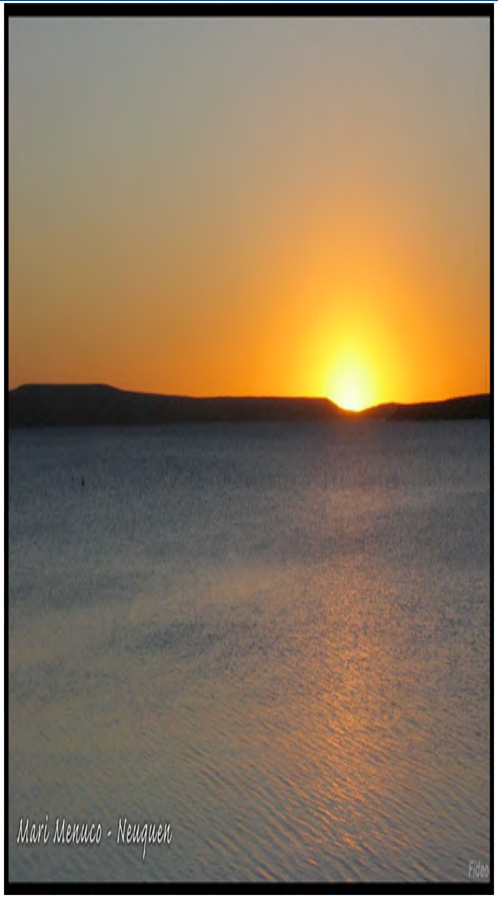


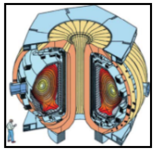


Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General

Aiming at Fusion Power – Tokamak
Addressing disconnects

- **Significant progress has been made in fusion power generation development in the last 40 years**
- **Fusion power generation is entering a very challenging period of development**
- **No doubt: a physics, material, technology, safety, testing and licensing integrated approach is necessary, taking ITER as an example**
- **In parallel with the effort on ITER, the community should continue to identify potential physics and technology disconnects, and to address them in a coordinated manner**
- **The chamber heat flux distribution and the helium-cooled heat removal capability at the chamber wall is identified as a relatively new disconnect**
- **We need to continue identifying and resolving other significant disconnects in order to pave the way to a credible fusion power development including FNSF and DEMO**





Plasma -> Boundary->PMI ->PFC ->Materials ->FW H/T->Blanket->Coils->Power conversion->General