Fusion Energy Systems Studies-FNSF and the Fusion Nuclear Materials Program

C. E. Kessel and the FESS Team

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Fusion Materials Workshop, July 25-29, 2016, UTK & ORNL

Examination of Integrated Fusion Energy Facilities, Fusion Nuclear Science R&D, Critical Fusion Topics

The study team attempts to covers as many areas as it can (present **Fusion Nuclear Science Facility study** includes), and is multi-institutional

Core plasma physics (PPPL)
Edge Plasma physics (LLNL)
Nuclear Analysis (Univ Wis)
Thermo-mechanics (UCLA, Univ Wis, consultants)
Tritium and Safety analysis (INL, LLNL)
Liquid Metal (breeder) analysis, MHD, thermal, and mass transport (UCLA)
Materials assessments and developments (ORNL, consultant)
Magnets (PPPL)
RF Launchers physics and design (MIT)
Diagnostics (PPPL)
CAD (Univ Wis)
Maintenance, facility, schedule (PPPL, consultant, team)
Systems analysis (PPPL, consultant)
Thermal-hydraulics (consultant)

Fus Mat Community Continues to be a Good Partner for Systems Studies

Rowcliffe participates with our team

FESS had a project meeting with the Fus Mat folks at ORNL, winter of 2015 (the snowy, stormy, icy winter)

- Pre-FNSF and Parallel FNSF Material Testing - Stoller
- FCI/SiC - Katoh
- Bainitic Steel – Yamamoto
- Cast Nanostructured Alloy (CNA) – Tan
- Material Limitations for Diagnostics – Zinkle
- Structural Alloys for the FNSF – Rowcliffe

Recently….2 presentations on Tungsten Developments – Garrison

Rapp has joined our meetings and conf calls

FESS provided input on materials issues, request from Garrison

What Does the FNSF Need to Accomplish?

Missions Identified: (shown as ITER – FNSF – DEMO – Power Plant)

- Fusion neutron exposure (fluence and dpa)
- Materials (structural, functional, coolants, breeders, shield…)
- Operating temperature/other environmental variables
- Tritium breeding, fuel cycle sustainability
- Tritium behavior, control, inventories, accounting
- Long plasma durations at required performance
- Plasma enabling technologies
- Demonstration of safe and environmentally friendly plant operations
- Power plant relevant subsystems at high efficiency
- Availability, maintenance, inspectability, reliability advances toward DEMO and power plants

Each mission contains a table with quantifiable metrics

Expect to use ARIES-ACT2 (DCLL blanket) as power plant example

Defining the Fusion Core Out to, and Including the Magnets

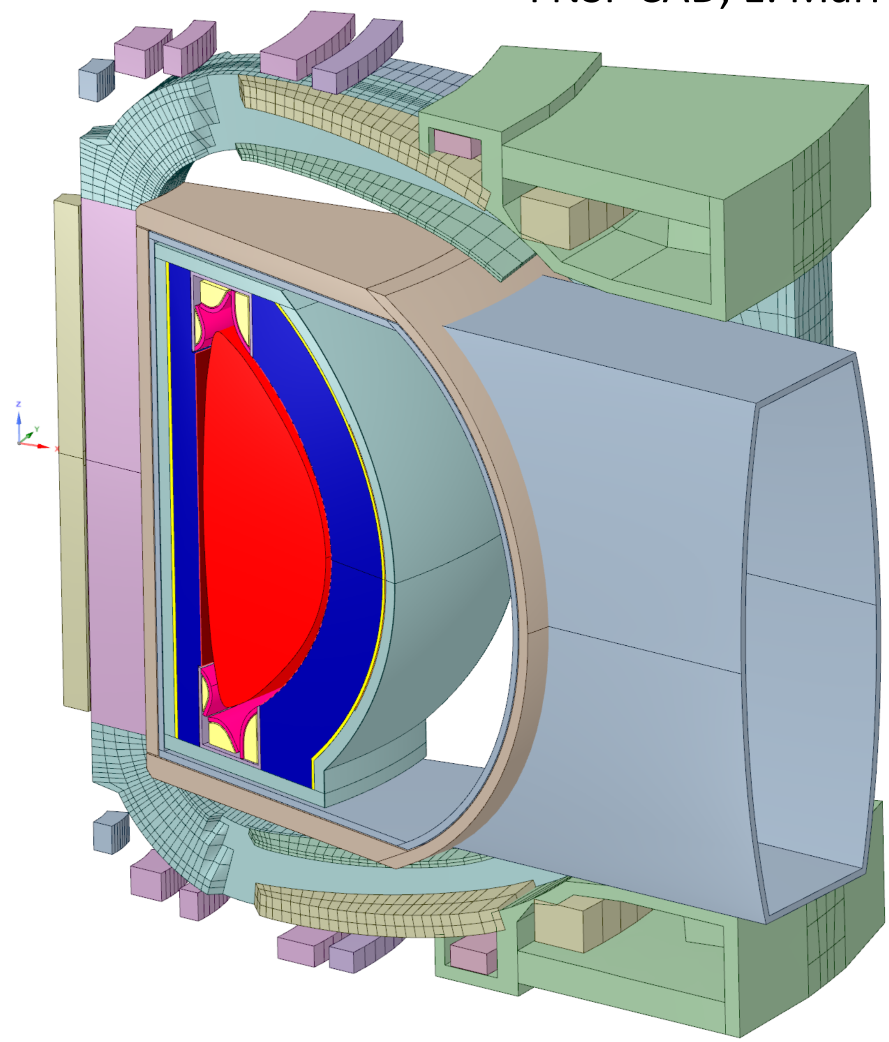
Primary components:

- FW/blanket
- Divertor
- Launchers/TBMs/diagnostics
- Structural Ring
- Vacuum Vessel
- LT Shield
- Magnets

Service parameters:

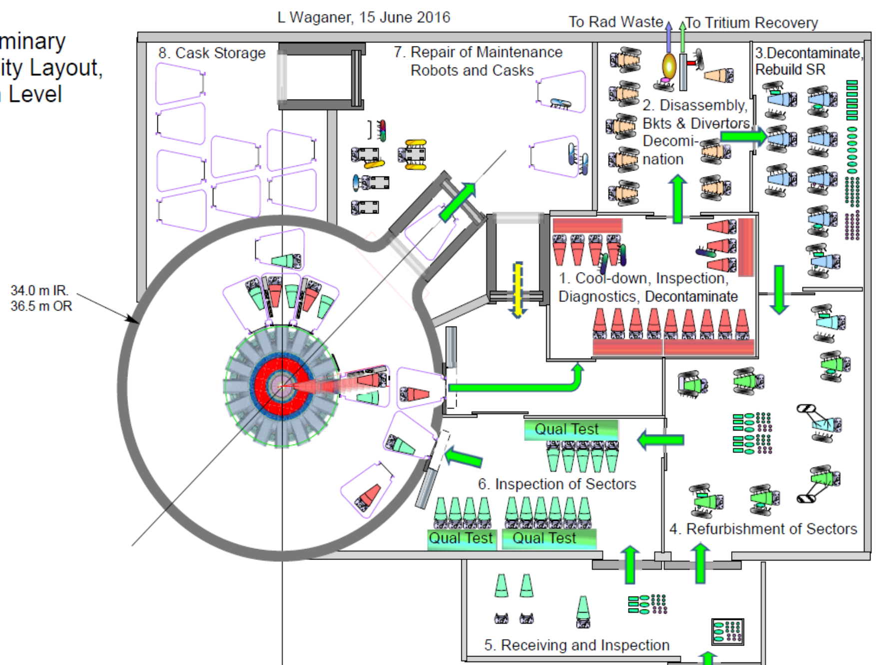
- Plasma op point, loads, NWL
- B-fields
- Temperatures
- Pressures/stresses
- Flow rates
- Volumetric heating
- Dpa/He/H
- Surface heating/particles
- Tritium

FNSF CAD, E. Marriott

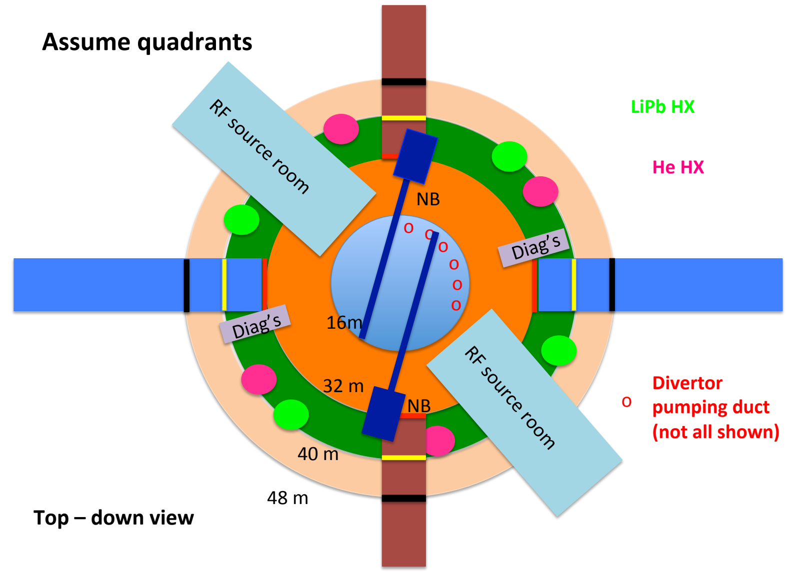


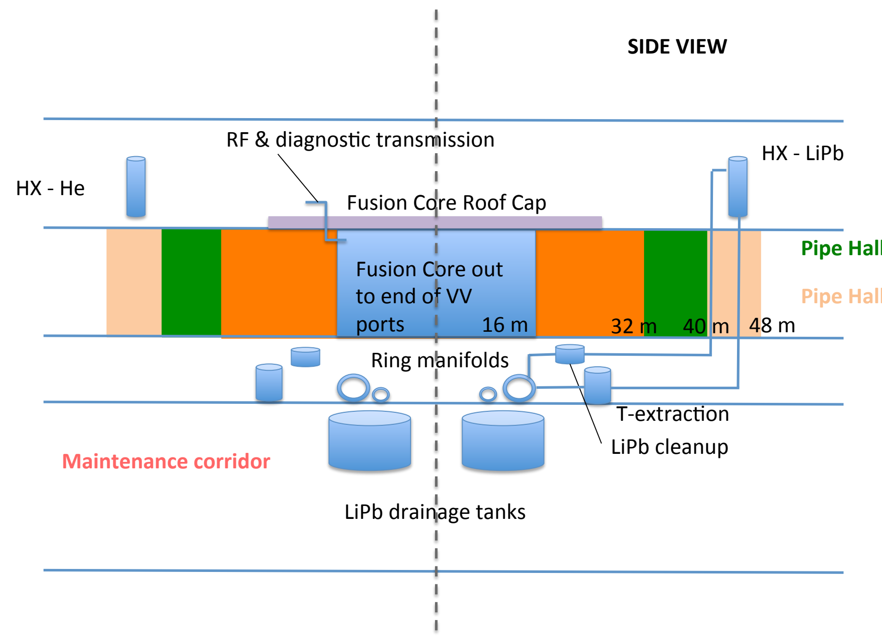
FNSF Hot Cell and Near Core Layout are Used to Assess Operations and Provide Data for Analysis

Preliminary Facility Layout, Main Level



Hot Cell Conceptualization, Waganer





FNSF Power Core Geometry, 6/15/2016

Original CAD PC radius to ends of ports is 13 m

The FNSF must fill the tremendous gap between ITER and DEMO by providing the break-in to the fusion nuclear regime

First strongly burning plasma

Demonstrate routine Power Plant Ops

No technical gaps

	ITER	FNSF	DEMO	Power Plant
Max damage	3 dpa	37-74 dpa	100-150 dpa	150+ dpa
Max plasma pulse	500-3000s	1-15 days	15-365 days	365+ days
TBR	~ 0	~ 1.0	1.05+	1.05
$T_{blanket}$, $T_{cool,exit}$	285C, 150C	550C, 650C	550C, 650C	550C, 650C
Materials	316SS, CuCrZr, Be, W, H_2O, SS304, SS430		RAFM, PbLi, He, SiC-c, Borated-RAFM, W, bainitic steel	

A nearer term facility like the FNSF requires a number of technical philosophies/approaches to be defined/explored

Physics strategy – how do we choose plasma parameters, what's their impact

Long term power plant relevance – design choices are made to keep the scientific/technology development on track, avoid diversions that do not contribute to the power plant vision

Minimal, Moderate, and Maximal FNSF

Qualification requirements to install a component/material in the FNSF – fusion neutron exposure to the dpa level, highly integrated non-nuclear testing…..plasma-vacuum systems are not consistent with "cook and look" approach to FNS

FNSF program plan – phases, material/temperature/dpa evolution, operation and maintenance

Blanket (divertor/launchers) choices and testing strategy – provide the process by which we test and advance fusion core components, and backups

Hot Cell – how do we access and process the information from the FNSF operation

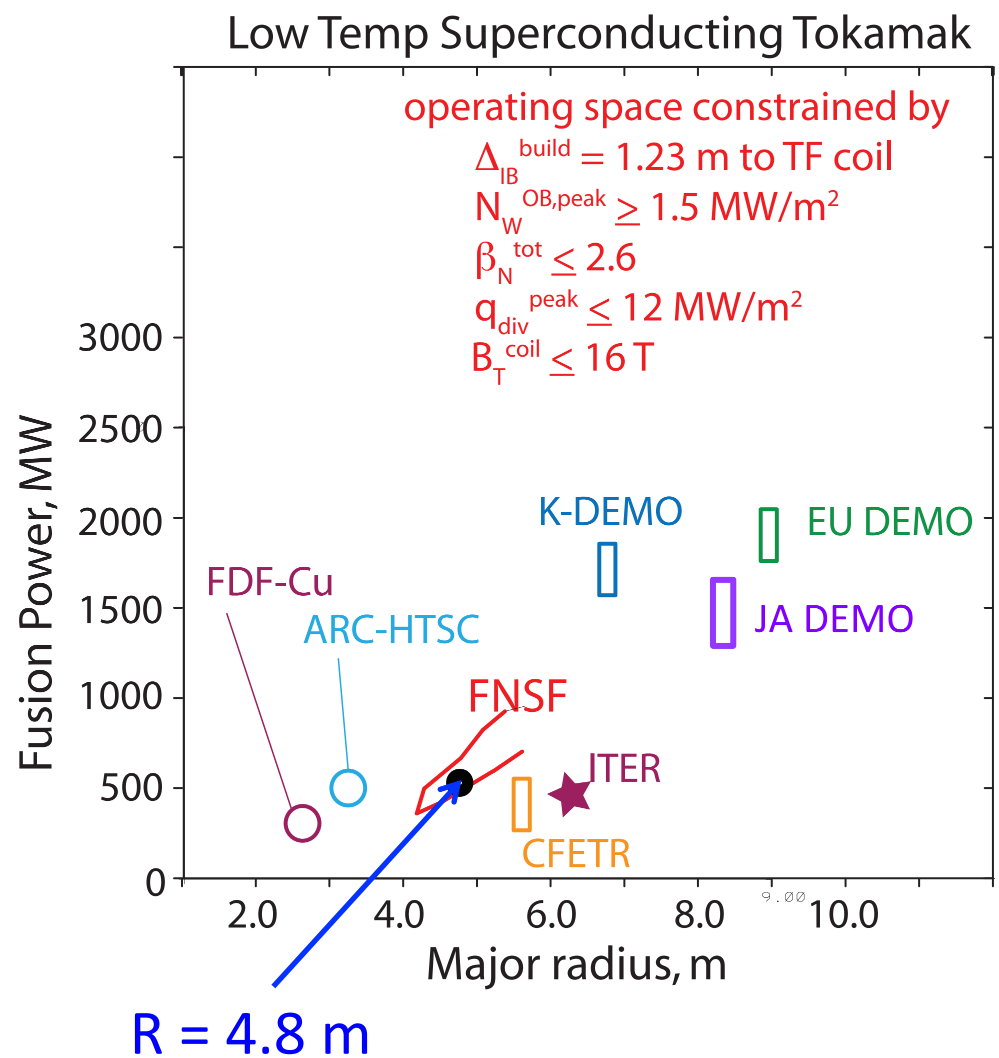
Pre-FNSF R&D – how do we see the R&D evolution to prepare us for the FNSF, design and operation

The FNSF Would Be Smaller Than a DEMO Plant, to Reduce Cost and Facilitate a Break-in Program

Configuration for the FNSF study:

- Conventional aspect ratio (= 4)
- Conservative tokamak physics basis with extensions to higher performance (β_N < 2.6)
- 100% non-inductive plasma current
- Low temperature superconducting coils, advanced Nb_3Sn
- Helium cooling in blanket, shield, divertor, and vacuum vessel
- Focus on DCLL blanket concept with backup concepts (HCLL, HCCB/PB)
- Net electricity is NOT a facility target, but electricity generation can be demonstrated

These devices do not all use the same level of assumptions/goals as the FNSF



The Program on the FNSF Defines It, Not Its Operating Point

	He/H	DD	DT	DT	DT	DT	DT	Power Plant
Yrs	1.5	2-3	2.5	4.2	4.2	5.9	5.9	40 FPY
Neutron wall load, MW/m^2			1.78	1.78	1.78	1.78	1.78	2.25
Plasma on-time, % /year	10-25	10-50	15	25	35	35	35	85
Plasma pulse length, days		Up to 10	1	2	5	10	10	310
Plasma duty cycle, %		33-95	33	67	91	95	95	100
Neutron damage, dpa			7	19	26	37	37 or 74	100-150
blanket	RAFM 400C	RAFM 400C	RAFM 400C	RAFM ODS 450C	RAFM ODS (NS) 500C	RAFM ODS (NS) 5500C	RAFM ODS (NS) 5500C	

Plasma pulse extension
1 hr to 10 days

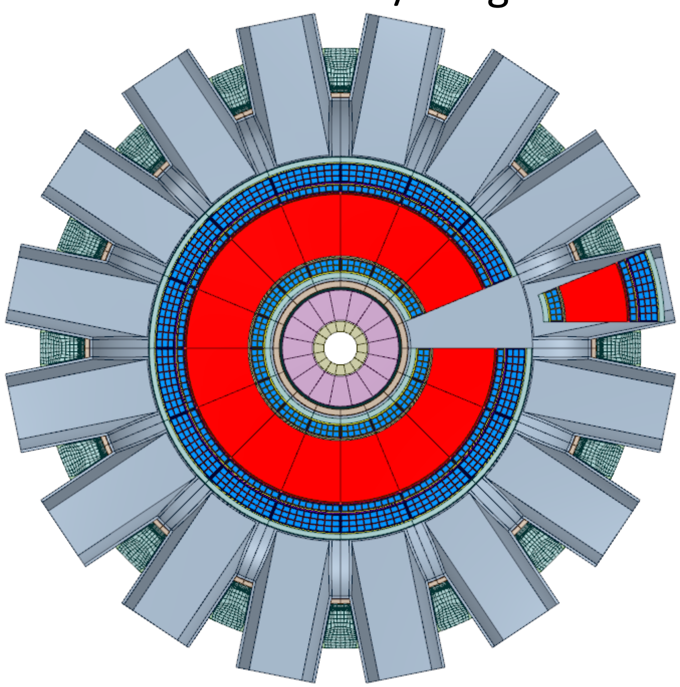
23 years of DT operations, 8.4 years of neutron exposure

Higher N_W, faster plasma pulse development, and efficient maintenance/plasma operation distribution can reduce years

Blanket Testing

Blanket:

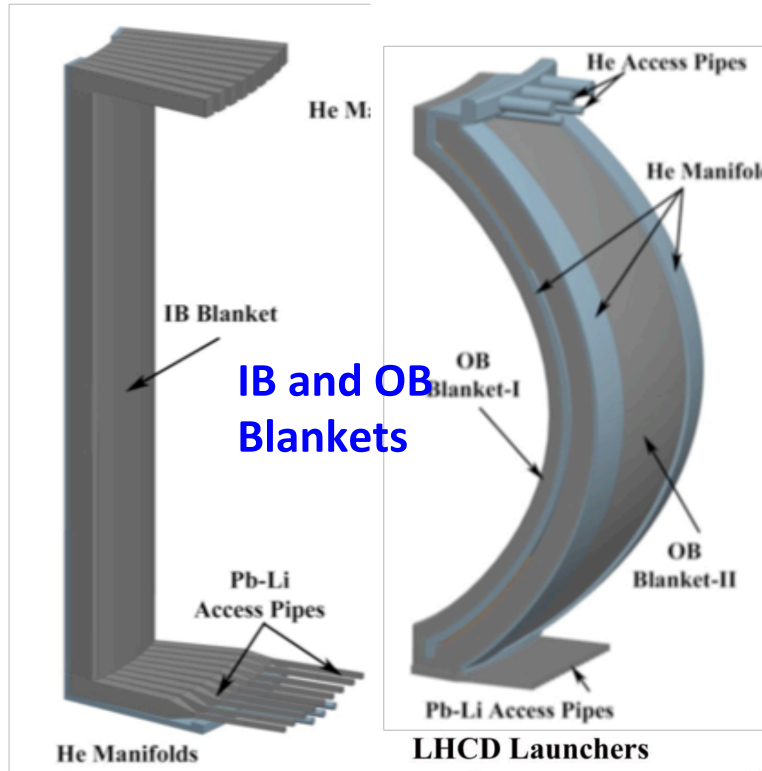
- **DCLL 400C RAFM** (some are taken for autopsy)
- DCLL 400C RAFM/ H&CD
- DCLL 450C RAFM (higher T)
- DCLL 450C RAFM GII (next phase T and FS)
- DCLL 400C RAFM/ MTM
- DCLL 400C RAFM/ TBM-HCLL
- DCLL 400C RAFM/ TBM-HCCB(PB)
- DCLL 400C RAFM/ Diagnostic

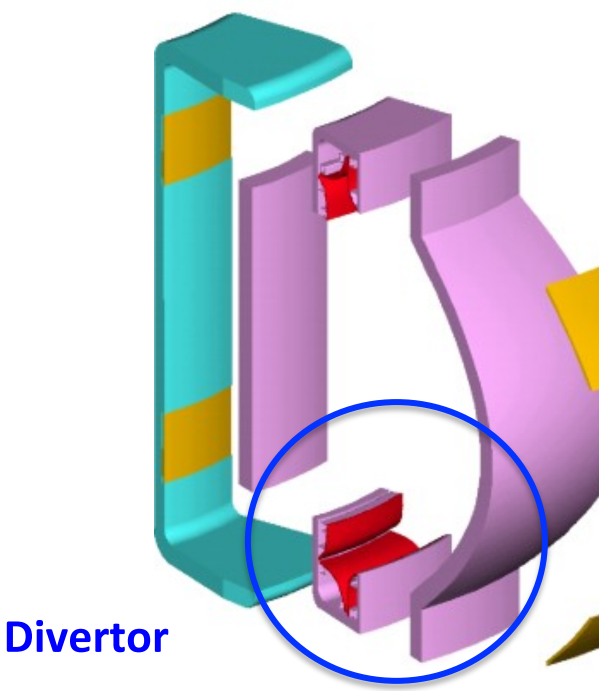


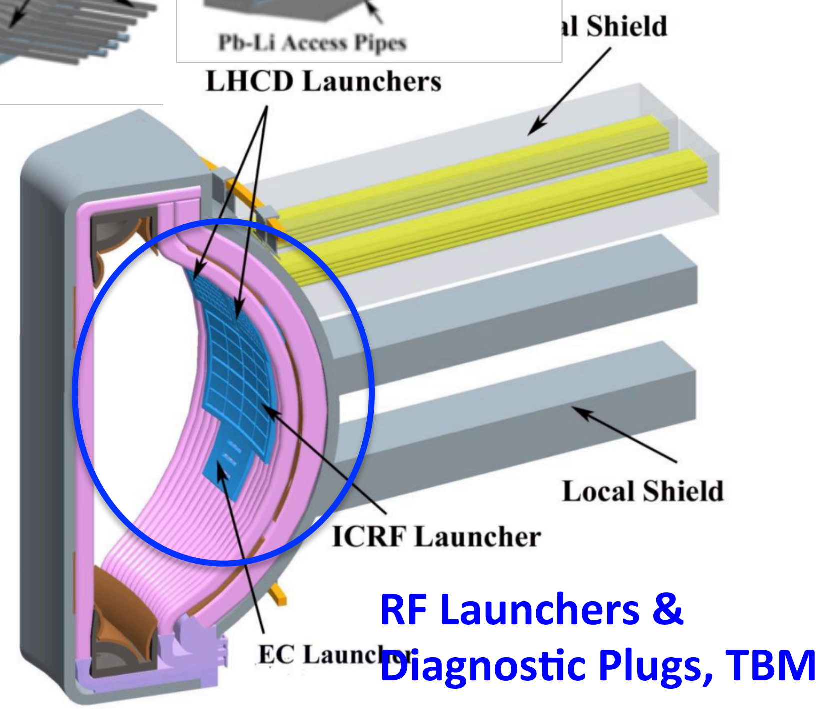
	Phase X – part 1	Phase X – part 2	Phase X – part 3	….
S-1	**DCLL 400C RAFM**	**DCLL 400C RAFM – R1**	**DCLL 400C RAFM – R1**	
S-2	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM – R2**	
S-3	**DCLL 400C RAFM – LH/IC/EC**	**DCLL 400C RAFM – LH/IC/EC**	**DCLL 400C RAFM – LH/IC/EC**	
S-4-MTM	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM**	
S-5	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM**	
S-6	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM – R2**	
S-7	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM**	
S-8	**DCLL 400C RAFM**	**DCLL 400C RAFM – R1**	**DCLL 400C RAFM – R1**	
S-9	**DCLL 450C RAFM GenII**	**DCLL 450C RAFM GenII**	**DCLL 450C RAFM GenII**	
S-10	**DCLL 400C RAFM – NB**	**DCLL 400C RAFM – NB**	**DCLL 400C RAFM – NB**	
S-11	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM**	
S-12	**DCLL 450C RAFM**	**DCLL 450C RAFM**	**DCLL 450C RAFM**	
S-13	**DCLL 400C RAFM - NB**	**DCLL 400C RAFM - NB**	**DCLL 400C RAFM – NB**	
S-14	**DCLL 450C RAFM GenII**	**DCLL 450C RAFM GenII**	**DCLL 450C RAFM GenII**	
S-15-TBM/HCCB	**DCLL 400C RAFM / HCCB**	**DCLL 400C RAFM / HCCB**	**DCLL 400C RAFM / HCCB**	
S-16-TBM/HCLL	**DCLL 400C RAFM / HCLL**	**DCLL 400C RAFM / HCLL**	**HCLL 400C RAFM / HCLL**	

Components in fusion core

We have concentrated on the blankets, but there are others that may have a testing sequence…..materials, temperatures, design, etc.





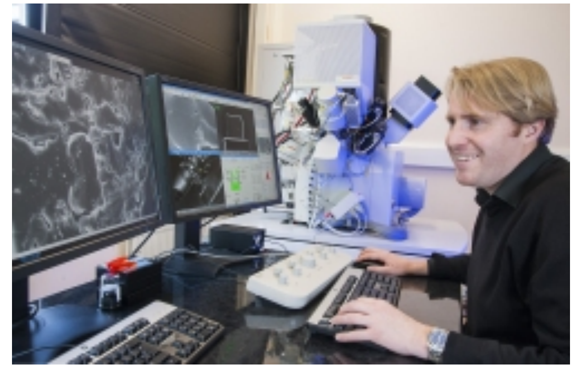


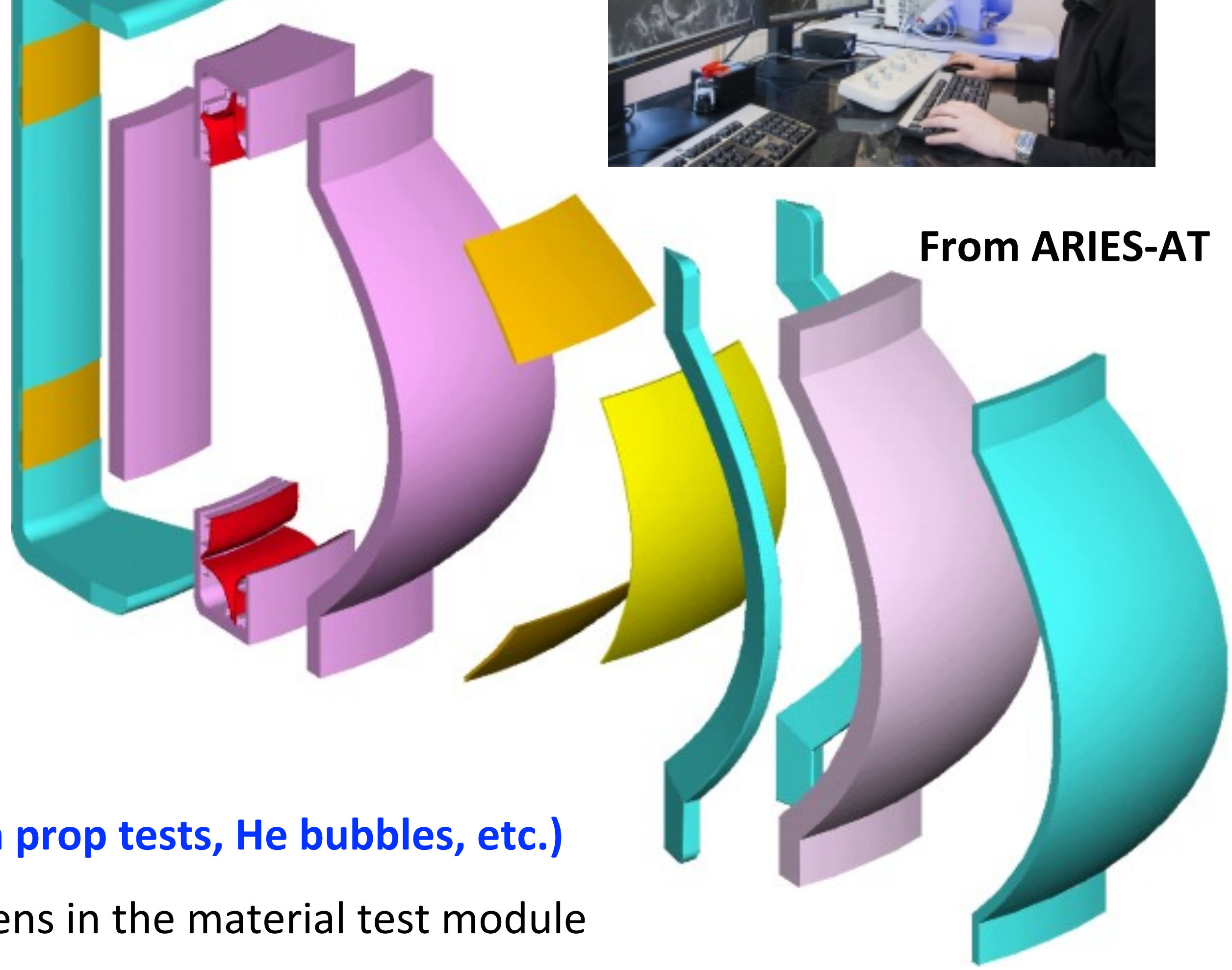
What do we do with the Sectors, Divertors, Launchers in the Hot Cells?

Inspect
Decontaminate (clean off)
Inspect
Dismantle
Inspect
Examine untreated surfaces
Examine mounts/connectors
Cut samples
- FW
- Side wall
- Grid plates
- Mounting hardware
- SR
- Div armor
- Div structure
- FCI
- W stabilizer
- ……

Material examinations (mech prop tests, He bubbles, etc.)

Also examine the test specimens in the material test module





From ARIES-AT

The Hot Cell – a critical mission of the FNSF

The performance of materials in the components of the fusion core is not accessible prior to the FNSF

What is accessible: 1) DONES/IFMIF or similar fusion relevant neutron exposure, 2) fission neutron, ion and doping approaches testing, 3) non-nuclear highly integrated component testing.....would we proceed if #1 was missing?

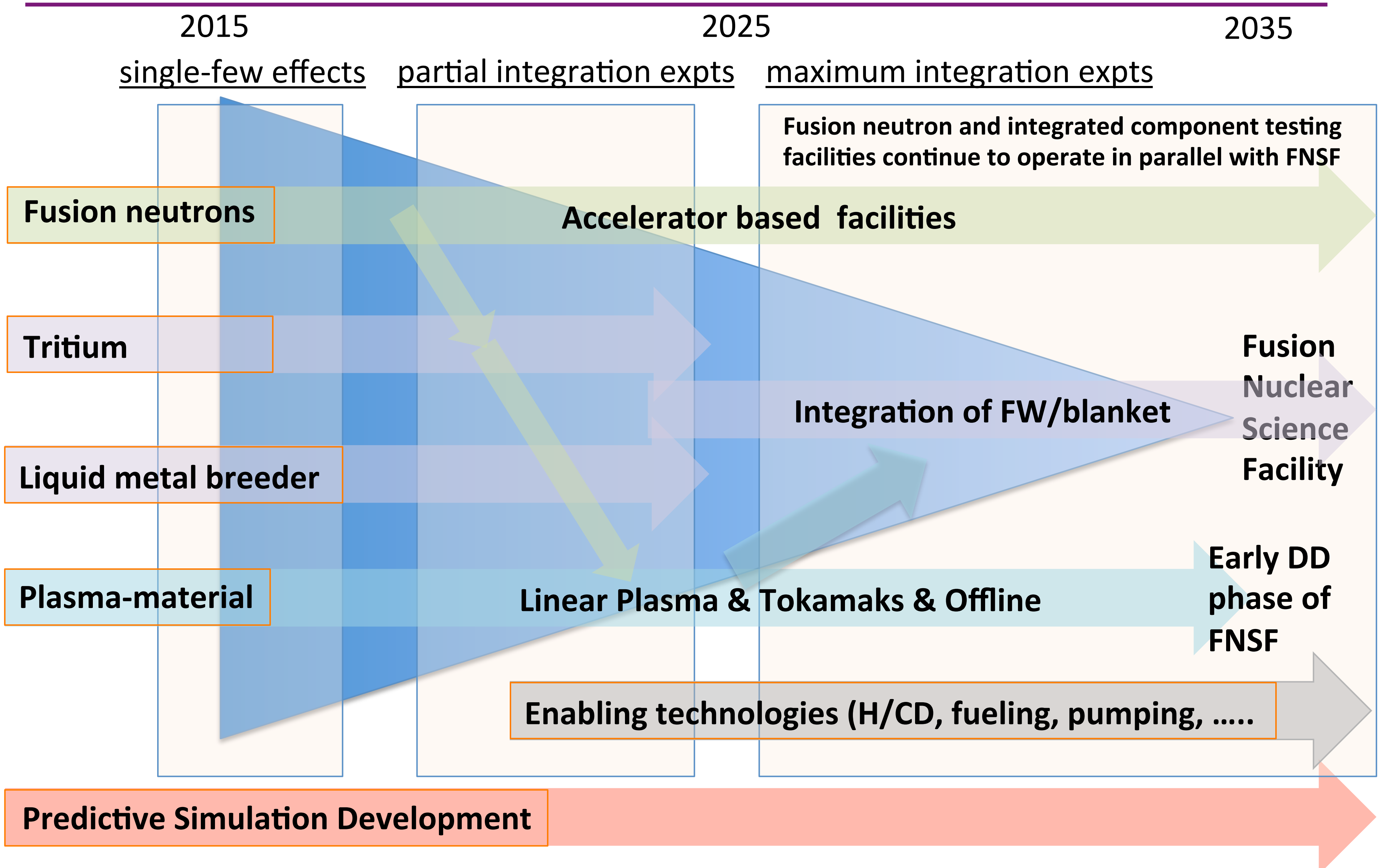
The in-service conditions include more than neutrons and temperature (like IFMIF)

Materials are under stress (pressure)
Materials have hydrogen in their matrix, tritium and transmutation hydrogen
Materials have contact with liquid metal in B-field (MHD, other flow properties, chemical reactions)
Materials see strong neutron damage, and He production gradients into material
Materials see heating, temperature and stress gradients

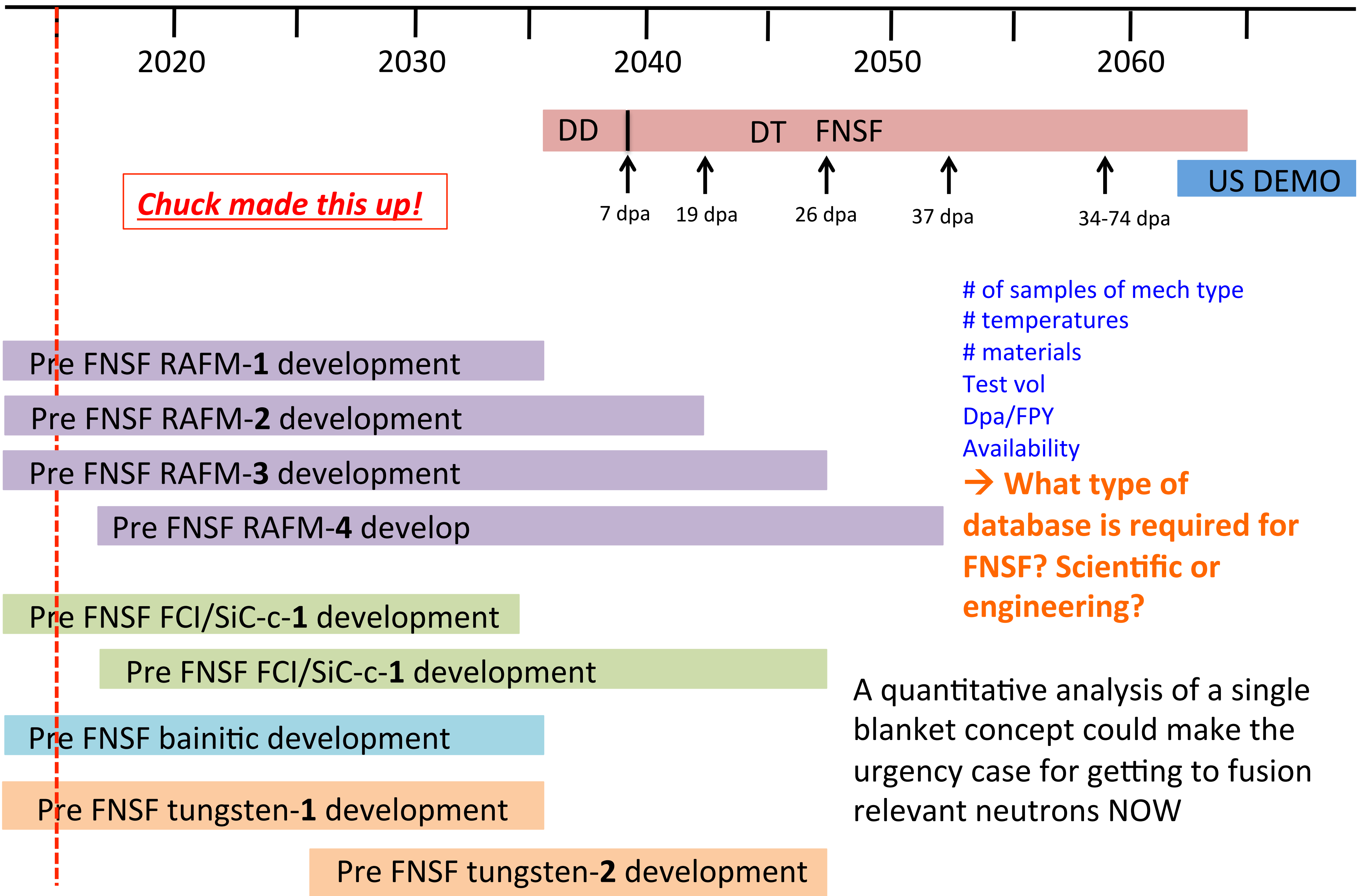
The sectors of the fusion core will be removed, inspected, dismantled, decontaminated, inspected again, cut into samples......and will be examined to determine the property changes, surface changes, microstructure changes.....*this is done in the Hot Cell, <u>and will produce the database upon which DEMO can be based</u>*

we need to make decisions about the materials behavior and next phases based on this information in the FNSF itself.....turnaround must be fast, materials will be HOT

Pre-FNSF R&D Major Topics and Evolution Toward FNSF



Fusion Materials Science assumed timeline



# of samples of mech type
temperatures
materials
Test vol
Dpa/FPY
Availability

→ What type of database is required for FNSF? Scientific or engineering?

A quantitative analysis of a single blanket concept could make the urgency case for getting to fusion relevant neutrons NOW

Rowcliffe presented at April 2016 FNSF meeting

Early IFMIF:

what is the high-flux payload?

Type of Specimen	Geometry (mm)	No of Specimens*	Volume of packets (cm^3)
Microstructure	TEM disk (3 diam. × 0.25)	800	4
Tensile	Sheet tensile specimens (25 × 4.8 × 0.76)	156	45
Fatigue	Cylindrical specimens (25 × 4.8 × 1.52)	96	56
Fracture toughness	Disk compact tension (11.5 x 11.5 × 4.6)	66	89
Crack growth	Disk compact tension (11.5 x 11.5 × 2.3)	40	32
Dyn. fracture toughness	Notched bar (3.3 × 3.3 × 25)	120	62
Creep	Pressurized tube (25 × 2.5 diam.)	104	37
	Total	1382	325

Size achievable?!

?!

How many different IRRADIATION temperatures are needed
How many TEST temperatures each property
How many data points for a single property
How many specimen/single data point (eg to determine DBBT > 10 samples) >
4 different Tirr and three "materials" ... (base + 2 joints)

IFMIF high-flux payload example shown above (≥325 cc).
ENS and FAFNIR proposals have high flux volume ~ 20-25 cc.
→marginal to measure more than Dynamic Fracture Toughness (highest priority).
Fission irradiation 'pods' would suggest ~ 70-100 cc
Evaluation is needed by the Materials Community – could early IFMIFs optimise?

IFMIF/EVEDA TEST MATRIX

Rowcliffe presented at April 2016 FNSF meeting

Specimen	Type of Test Specimen	Temp. in PIE Test	Base or Weld Metal	Test Number for Each Condition	Irradiation Temp.	Total Numbers	Total Volume* ~cm3 !
Tensile	SS-3 ~Plate type!	Two conditions: RT, Irradiation temp.	Two conditions: Base Metal and Weld Metal	Three times	Three conditions: 3008C 4008C 5008C	36	3.492 ~0.097 X 36!
Creep	SS-3 ~Plate type!	One: Irradiation temp	Two conditions: Base and Weld Metals	Six: 1 000, 2 000, 5 000, 10 000, 20 000, 50 000 hr	Three conditions: 3008C 4008C 5008C	36	3.492 ~0.097 X 36!
Fatigue	SF-1	Two conditions: RT, Irradiation temp.	Two conditions: Base and Weld Metals	Six conditions and two times: 400, 1 000, 3 000, 10 000, 30 000, 100 000 cycle	Three conditions: 3008C 4008C 5008C	144	27.504 ~0.191 X 144!
Fracture toughness	0.2-CT	Three conditions	Two conditions: Base and Weld Metals	Five times	Three conditions: 3008C 4008C 5008C	90	54.72 ~0.608 X 90!
Creep tube	25.4mm X f 2.5mm	—	Two conditions: Base and Weld Metals	Six conditions and two times	Three conditions: 3008C 4008C 5008C	144	18.00 ~0.125 X 144!
Crack growth rate	0.2-CT 102t	Two conditions: RT, Irradiation temp.	Two conditions: Base and Weld Metals	Three conditions and three times	Three conditions: 3008C 4008C 5008C	108	32.832 ~0.304 X 108!
Microstructure0swelling	25.4 mm X 4.95 mm X 0.15 mm	RT	Two conditions: Base and Weld Metals	One	Three conditions: 3008C 4008C 5008C	6	0.1134 ~0.0189 X 6!
Total	—	—	—	—	—	564	140.153

IFMIF/EVEDA TEST MATRIX

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	0.2 CT				Three conditions:		54.72

Table 1

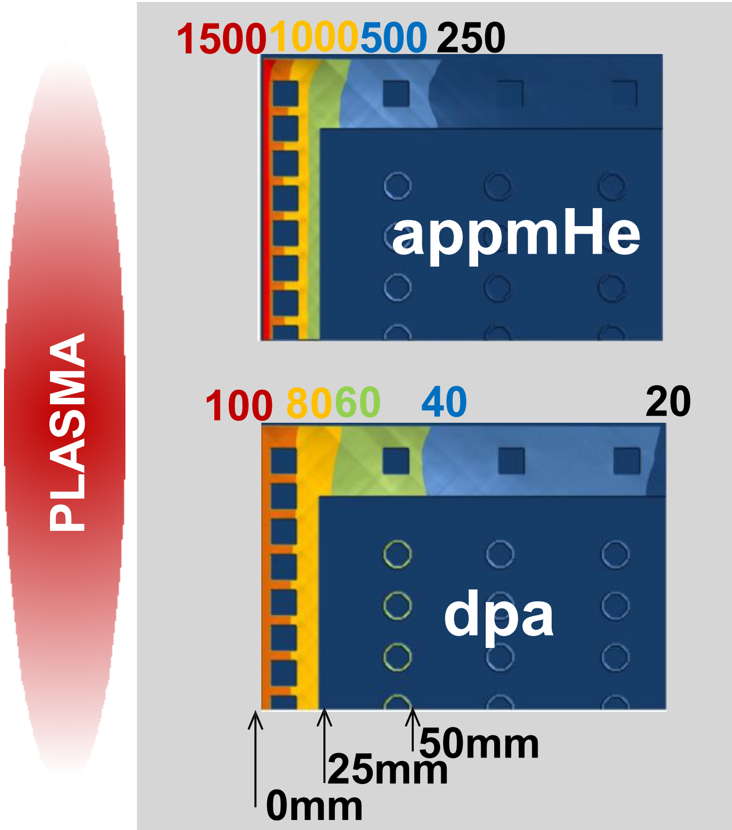
Summary of ferritic/martensitic steel irradiation parameters including damage rate per full power year (fpy) for several current and proposed neutron irradiation facilities.

Facility		Displacement damage rate (dpa/fpy)	He (appm/dpa)	H (appm/dpa)	Ca (appm/dpa)	Cl (appm/dpa)	Capsule individual/total volume (l)	
DEMO 1st wall, 3.5 MW/m^2 [84,85		30	11	41	<0.001	<0.001		
IFMIF high flux test module [84,85	IFMIF	20–55	10–12	35–54	<0.001	<0.001	~0.035/0.5	0.5 liter
HFR fission reactor, position F8 [84,		2.5	0.3	0.8			2.2/37	
HFIR fission reactor, RB* [86,87]		9	0.2	–			0.75/3	
HFIR fission reactor, target [86,88]		24	0.35	5			0.10/3.7	
BOR60 fast reactor, position D23 [84,89]		20	0.29	0.7			0.4/5	
ESS spallation source, reflector [84]		5–10	5–6	33–36				
ESS spallation source, target hull [83]		20–33	25–30	250–300				
SNS spallation source FMITS, 5 cm [9	US	5	20	100			0.02/0.04	0.04 liter
SNS spallation source FMITS, 3 cm [9		10	75	310			0.02/0.04	
SINQ spallation source, center rod 1 [91,92]		≤10	≤70	≤470			~0.006/3	
MTS spallation, fuel positions, 15 cm [44	US	17.5	29	–			~0.001/0.0	0.04 liter
MTS spallation, fuel positions, 5 cm [44,		32	16	–	1	0.1	~0.001/0.0	

Zinkle, Moeslang, FED2013, 472

Displacement damage and He production in Blankets

Helium production (appm) for 100 dpa at plasma facing side



H. Tanigawa, E.Wakai 2012

Separating our materials by their environment and formulating the testing strategy → look to power plants

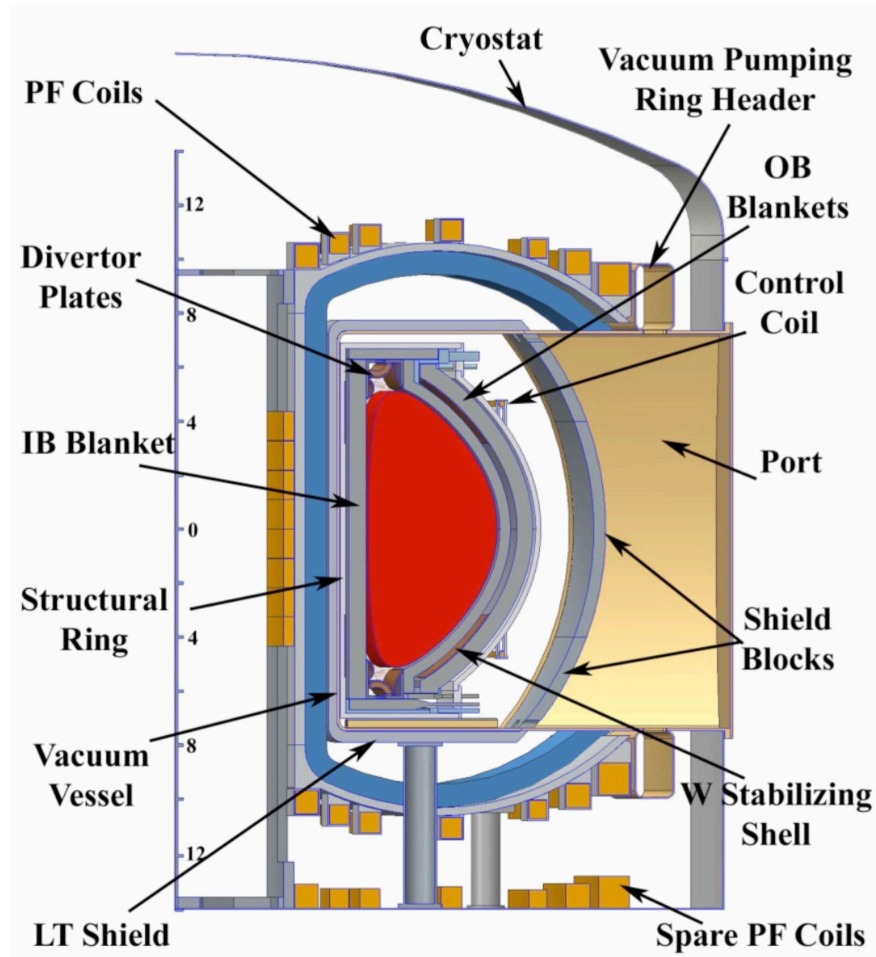
- “Only” the first centimeters have high He/dpa and H/dpa ratios
- In addition this part of the blanket carries the highest thermo-mechanical loads
- Therefore,
  - fission reactor irradiations are still meaningful for a significant fraction of in-vessel components
  - Nevertheless, a dedicated fusion neutron source is indispensable, but has to focus on plasma-near materials and loading conditions

Shown by A. Rowcliffe

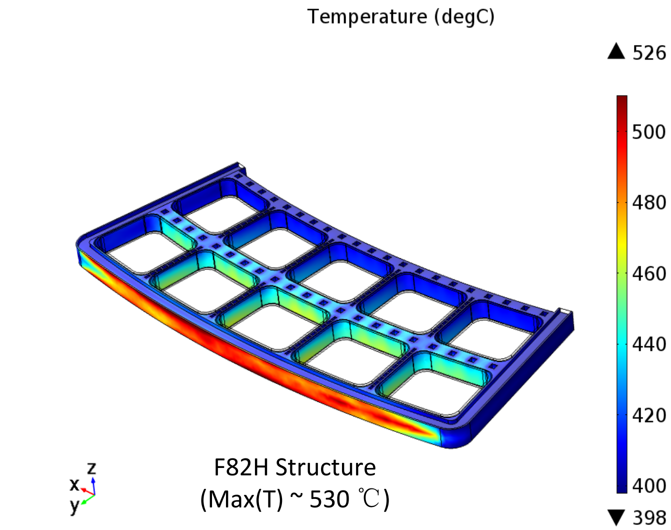
Information from the systems studies design analysis

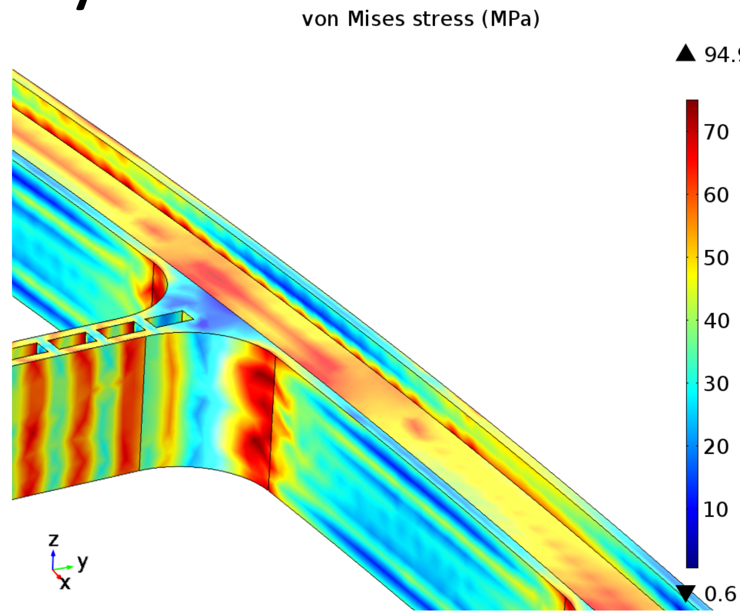
	ARIES-ACT2 DCLL
	9.75 m
FW/Blankets	He-cooled FW and Pb-17Li cooled RAFS(F82H) structure Pb-17Li inlet pressure>~1.65 MPa T_{min}(F82H)>~350 ˚C T_{max}(F82H)<~550 ˚C Helium T_{inlet}/T_{outlet}= ~385/470 ˚C Pb-17Li T_{inlet}/T_{outlet}=~460/647 ˚C η_{th}=45%
Structural ring	Helium-cooled F82H steel structure T_{min}(F82H)>~350 ˚C T_{max}(F82H)<~550 ˚C Helium T_{inlet}/T_{outlet}=380/385 ˚C
Upper/Lower divertor sectors	Helium-cooled W-based divertor and ODS steel or Ta cartridge T_{min}(w-structure)> 800 ˚C T_{max}(w-structure)<1300 ˚C q_{peak}<~10 MW/m^2 Helium T_{inlet}/T_{outlet}=676/720 ˚C
Vacuum vessel	He-cooled ribbed bainitic FS (3Cr-3WV) structure, operating temperature at ~500 ˚C
Water-cooled LT shield	Water-cooled bainitic FS (3Cr-3WV) and WC, operating at room temperature

1) Temperature distribution
2) Stress distribution
3) Nuclear heat, dpa, He distribution
4) Material/coolant
5) Flow rates of coolant/breeder

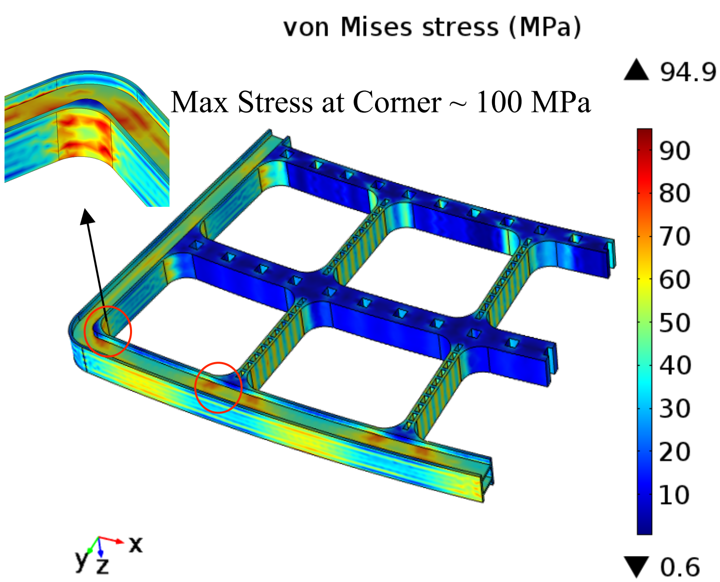


Thermo-mechanics analysis for the FNSF

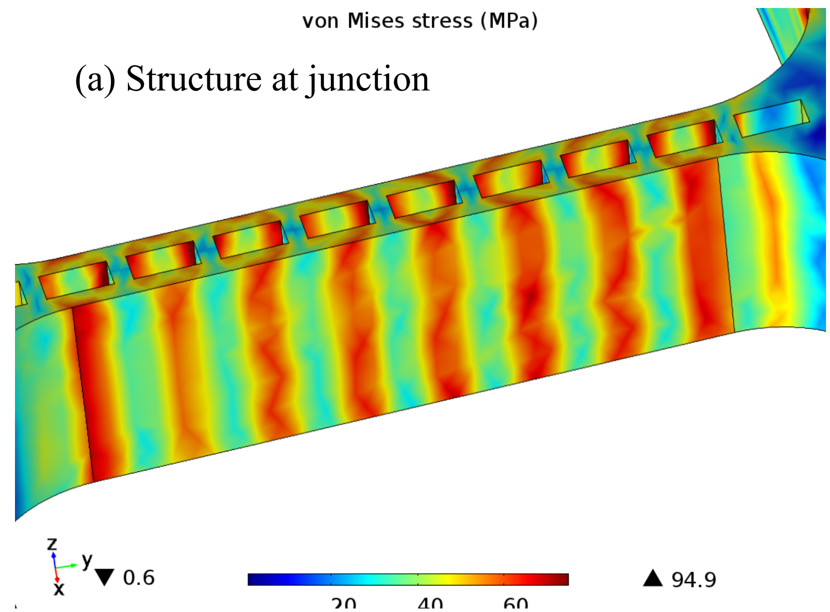




(a) Structure at junction



Global primary stress distribution
– Undeformed configuration

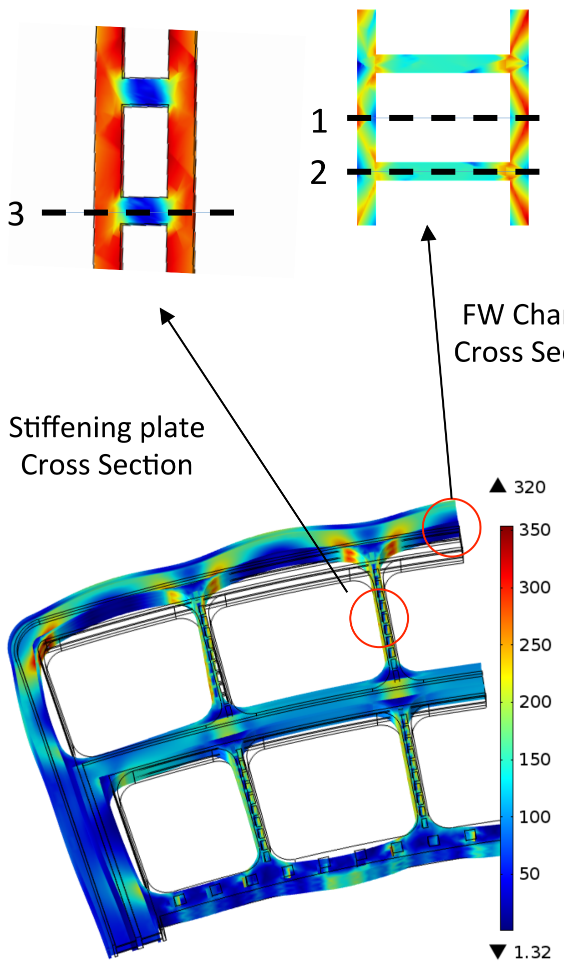


(b) Flow channels

Yue Huang, Nasr Ghoniem, UCLA

Safety Factors

Yue Huang, Nasr Ghoniem, UCLA



(off-normal condition)

$$\overline{P_m} \leq S_m(T_m, \varphi t_m)$$

Path	S_m [MPa]	[MPa]		Safety Factor	
		off-normal	normal	off-normal	normal
1	147.3	86.6	21.5	1.7	6.9
2	151.2	84.1	28.8	1.8	5.2
3	148.9	175.2	56.0	0.85	2.7

$$\overline{P_L + Q_L} \leq S_e(T_m, \varphi t_m)$$

Path	S_e [MPa]	[MPa]		Safety Factor	
		off-normal	normal	off-normal	normal
1	207.9	94.5	37.9	2.2	5.5
2	213.3	125.5	44.5	1.7	4.8
3	210.2	225.9	57.4	0.93	3.7

$$\overline{P_L + P_B + Q} \leq S_d(T, \varphi t, r_3)$$

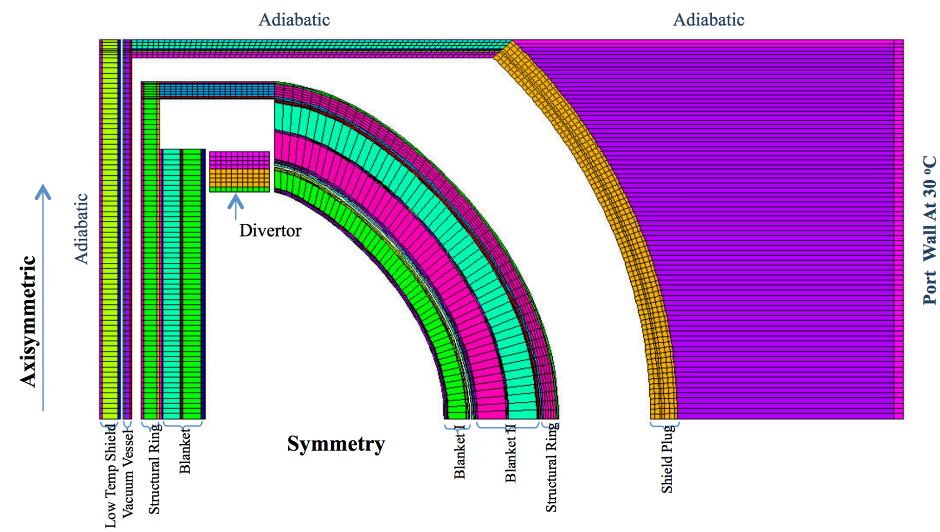
Path	S_d [MPa]	[MPa]		Safety Factor	
		off-normal	normal	off-normal	normal
1	415.8	259.9	194.7	1.6	2.1
2	426.6	177.8	65.2	2.4	6.5
3	420.3	217.1	46.8	1.9	9.0

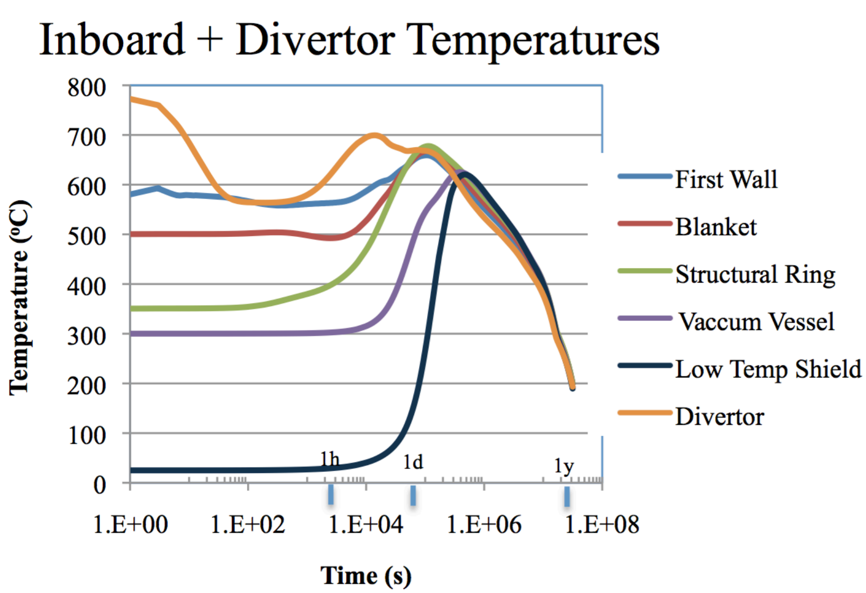
Loss of Coolant Accident Analysis – these could be an issue for materials like bainitic steel in the VV

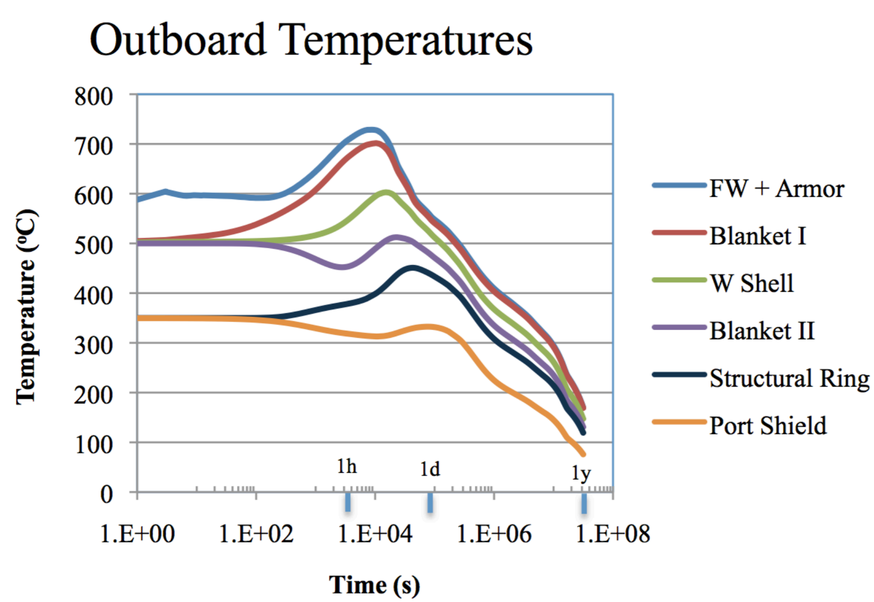
Simulation indicates that the RAFM remains below 730C

Bainitic VV reaches 625C….OK??

Univ. Wisconsin







The story of the 470C LiPb/RAFM Corrosion Limit

Based on experience with Na and NaK-loops, compatibility criteria for the allowable temperature at the steel/PbLi interface in the breeding blanket had been discussed already during US BCSS (1983):

1) Hands-on maintenance of external loop components must be possible.
2) Plugging of valves, cold traps and other components must be avoided.
3) Thinning of structural walls by corrosion must not endanger the integrity of components.

BCSS adopted criterion #2 since plugging had been observed in Na-loops if the maximum corrosion rate in the hottest section was > 5 mm/year.

For the different conditions in a PbLi blanket loop, a maximum allowable corrosion depth of 2.0 mm was selected. It was estimated at that time that this corresponds for ferritic steels to a maximum interface temperature of 470 C.

This very rough criterion is still used worldwide today for the layout of breeding blankets based on PbLi as breeder/coolant and ferritic steel as structural material.

Pursuing LiPb corrosion of RAFM or RAFM-variants; Limit should be 10% of materials thickness

Li Puma, 2012

As a design constraint, the mass loss per year is an more effective parameter (Siegfried Malang has suggested this)

This is parameterized as f(v, T)

Experimental data (Eurofer)

552C, 0.1 m/s, 0.25 mm/year

477C, ~0.2 m/s, 0.10 mm/year

552C, ~**0.2 m/s**, 0.40 mm/year

**typical flow speed is 0.05 m/s

What are complicating features to corrosion

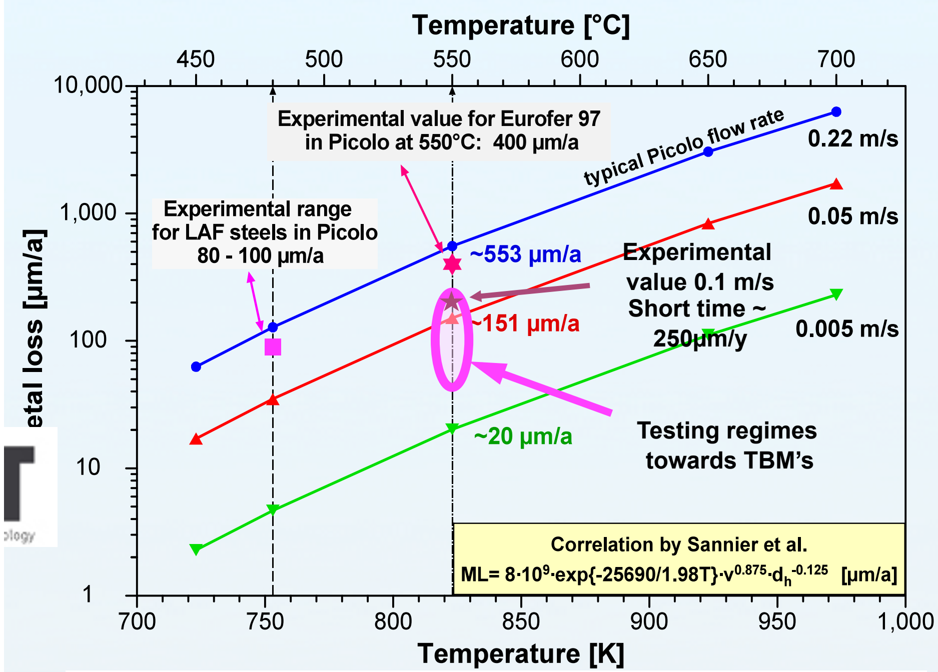
LM MHD turbulence (B-field)

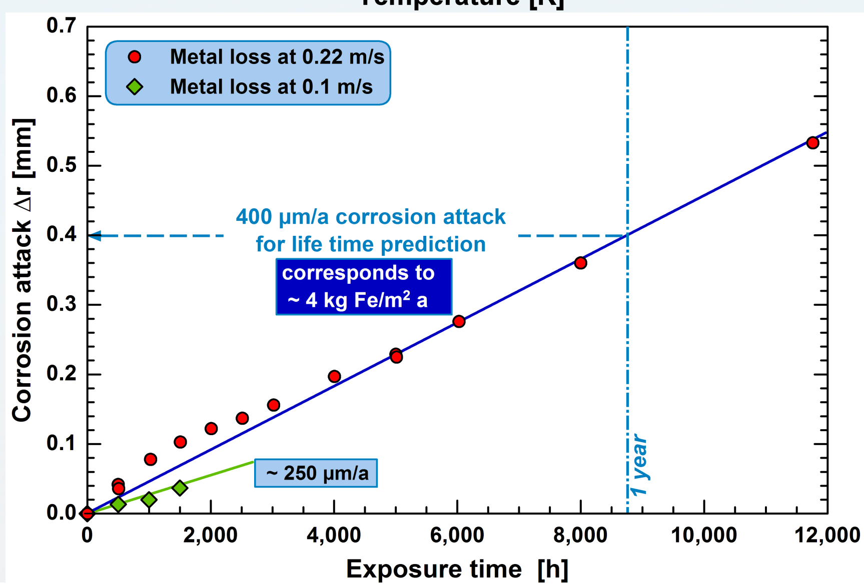
LiPb constituents, intermetallics?

Irradiation

SCC

In order to take advantage of corrosion resistance (Al), the steel must have hi T creep resistance, hi T irradiation resistance, etc





Examining the Extraction of Tritium from the LiPb breeder – This Component is Like a Large Heat Exchanger

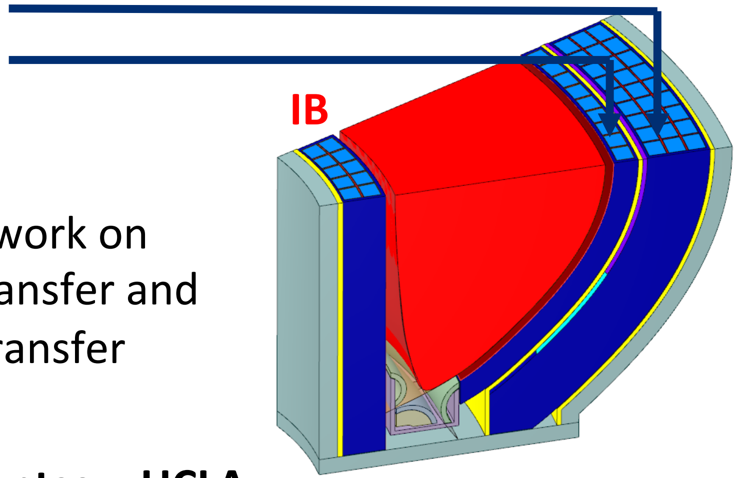
RAFM permeator comparison

Humrickhouse, INL

	B&W PWR steam generator	RAFM 470 ˚C	RAFM 470 ˚C	RAFM 470 ˚C	Vanadium 400 ˚C	Vanadium 500 ˚C	Vanadium 600 ˚C	Vanadium 700 ˚C
η (low solubility)		0.7	0.7	0.7	0.7	0.7	0.7	0.7
Tubes (#)	15,000	343,521	68,704	19,432	13,347	10,136	8,274	7,095
Tube length (m)	20.7	8.54	16.61	37.3	18.25	11.15	7.65	5.7
v (m/s)		0.1	0.5	1.77	2.55	3.4	4.22	4.98
Total volume (m^3)	61.8	278.7	108.42	69.0	23.15	10.74	6.01	3.84
ζ		4.85	1.27	0.45	1681	425	148	65
η (high solubility)		0.10	0.04	0.03	0.47	0.36	0.29	0.23

Vanadium has high tritium permeability
Vanadium is very sensitive to volatiles like oxygen
Although commercial hydrogen purifiers exist that use palladium to control this, the T is limited
Active research to utilize ceramics to allow higher temperatures

LiPb Pressure Drop Analysis shows the FCI and flow speed/op temperature where Δp is acceptable



Addt'l work on heat transfer and mass transfer

S. Smolentsev, UCLA

CASE IB	DELTA P, MPa				
	Poloidal duct	Inlet/outlet manifold	Inlet/outlet pipe	Pipe through fringing B-field	TOTAL
Sandwich FCI Tin=350C Tout=470C	2.67	2x0.98=1.96	2x0.1=0.2	2x0.57=1.14	**5.97**
SiC FCI Tin=400C Tout=500C	0.194	2x1.18=2.36	2x0.04=0.08	2x0.67=1.34	**3.97**
SiC FCI Tin=400C Tout=600C	0.097	2x0.59=1.18	2x0.02=0.04	2x0.35=0.70	**2.02**
SiC FCI Tin=400C Tout=650C	0.078	2x0.44=0.88	2x0.01=0.02	2x0.28=0.56	**1.54**

OB I	DELTA P, MPa				
	Poloidal flow	Inlet/outlet manifold	Access pipe	Access pipe in fringing B-field	TOTAL
LT DCLL with sandwich FCI Tin=350C Tout=470C	0.49	2x0.36=0.72	2x1.31=2.62	2x1.30=2.60	**6.43**
HT DCLL with SiC FCI Tin=400C Tout=600C	0.003	2x0.26=0.52	2x0.003=0.006	2x0.44=0.88	**1.41**
HT DCLL with SiC FCI Tin=400C Tout=650C	0.002	2x0.21=0.42	2x0.002=0.004	2x0.39=0.78	**1.21**

OB II	DELTA P, MPa				
	Poloidal flow	Inlet/outlet manifold	Access pipe	Access pipe in fringing B-field	TOTAL
LT DCLL with sandwich FCI Tin=350C Tout=470C	0.21	2x0.352=0.704	2x0.764=1.53	2x0.48=0.96	**3.404**
HT DCLL with SiC FCI Tin=400C Tout=600C	0.002	2x0.591=1.182	2x0.002=0.004	2x0.40=0.80	**1.99**
HT DCLL with SiC FCI Tin=400C Tout=650C	0.001	2x0.474=0.948	2x0.003=0.006 28	2x0.32=0.64	**1.60**

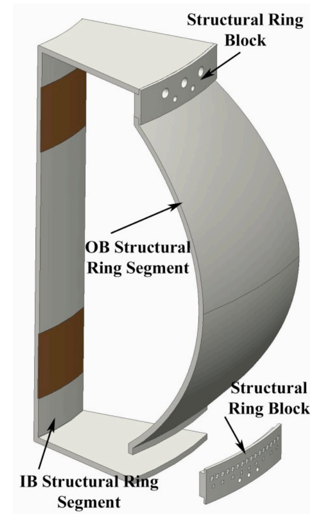
Other stuff that is actually important

The structural ring is the primary stiff structural element that the blanket, divertor and launchers (TBMs, MTMs, diagnostic plugs) are all mounted to

We assume this is also an RAFM steel since it generally receives a sufficient fluence/dpa, it operates at ~ 450C to be compatible with the blanket structure

It is filled with a shield filler material, WC or B-Fe steel

Structural Ring

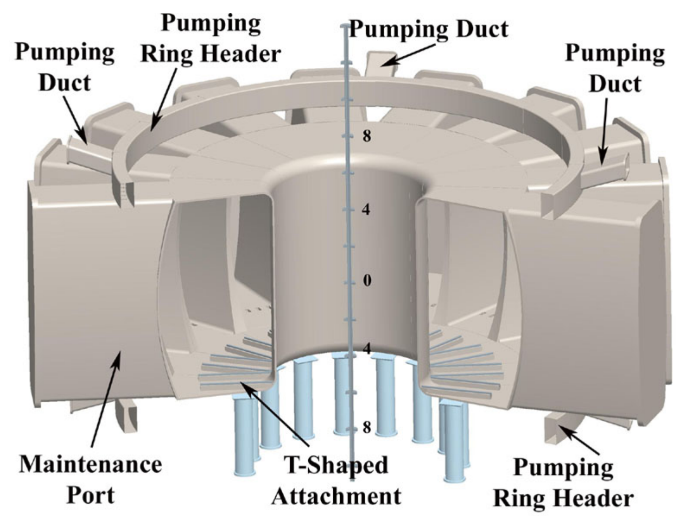


Vacuum vessel is the primary radio-nuclide barrier and pressure barrier

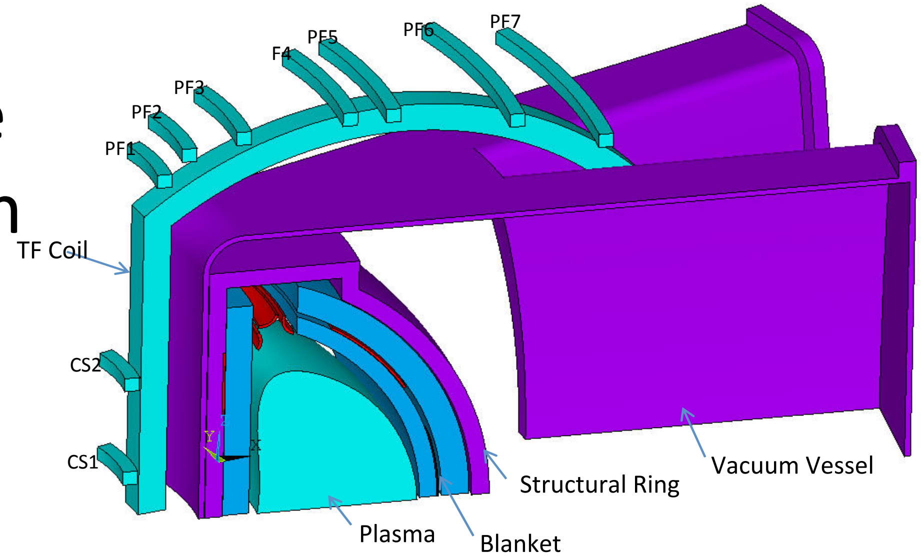
We are considering bainitic steel to avoid PWHT if/when the VV requires any re-welding, but the dpa and He must be low

In the FNSF we chose to maintain an inert gas (no oxygen) atmosphere or vacuum inside the VV, we do not open to air or other reactive gases….this allows the fastest recovery and does not contaminate our materials

Vacuum Vessel

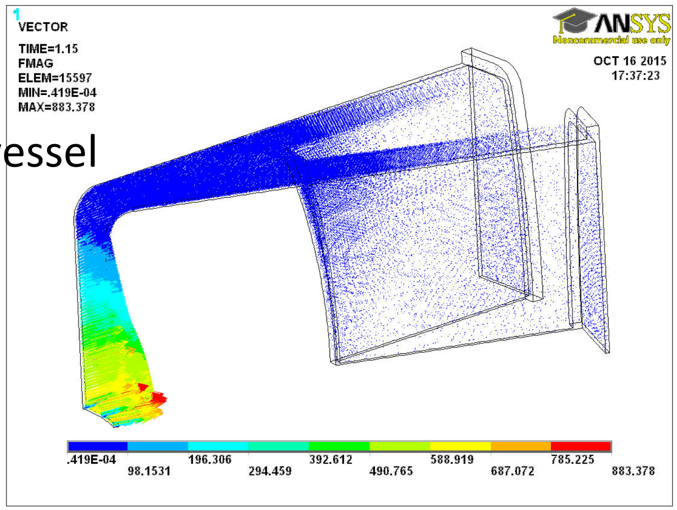


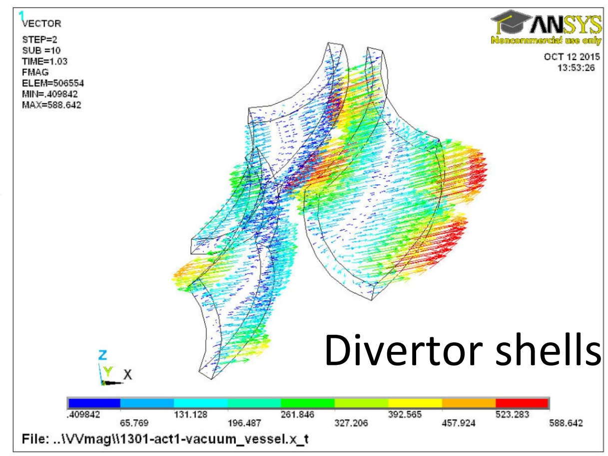
Disruption Modeling is exploring the impact of the thermal and current quench



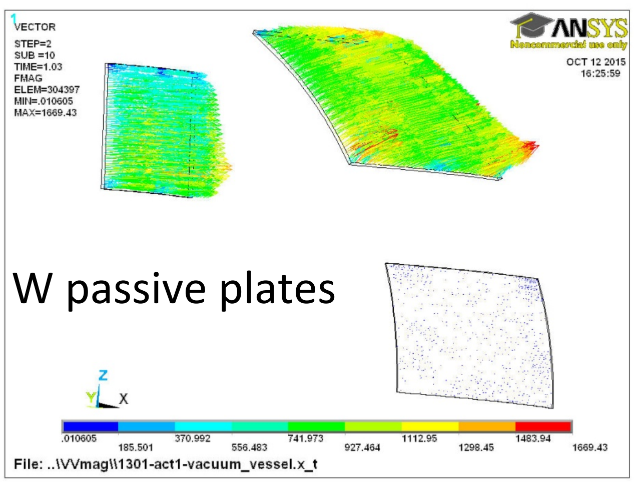
Nodal Forces for current quench

Vacuum vessel

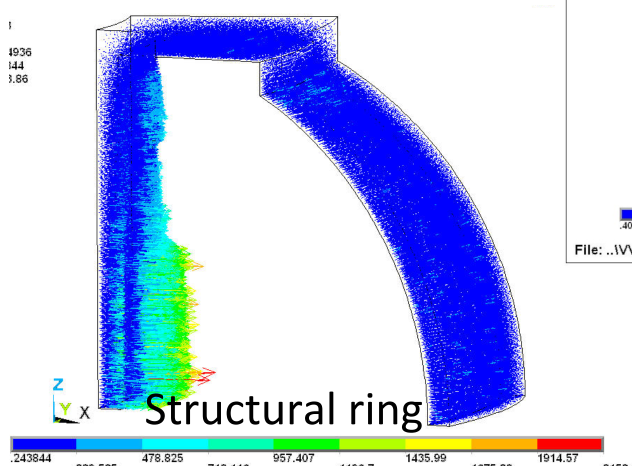




Divertor shells



W passive plates



Structural ring

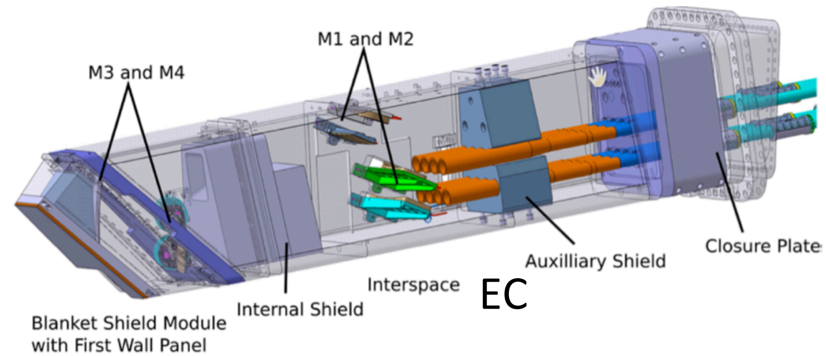
Blanchard, UW

RF launchers

What are the replacements for presently used materials in RF launching structures (Cu, SS) and how are windows, insulators and transmission lines affected

These systems must operate at high temperatures like the blanket, experience fusion **neutron exposure** and **plasma exposure**

G. Wallace and S. Wukitch MIT to assess RF launchers

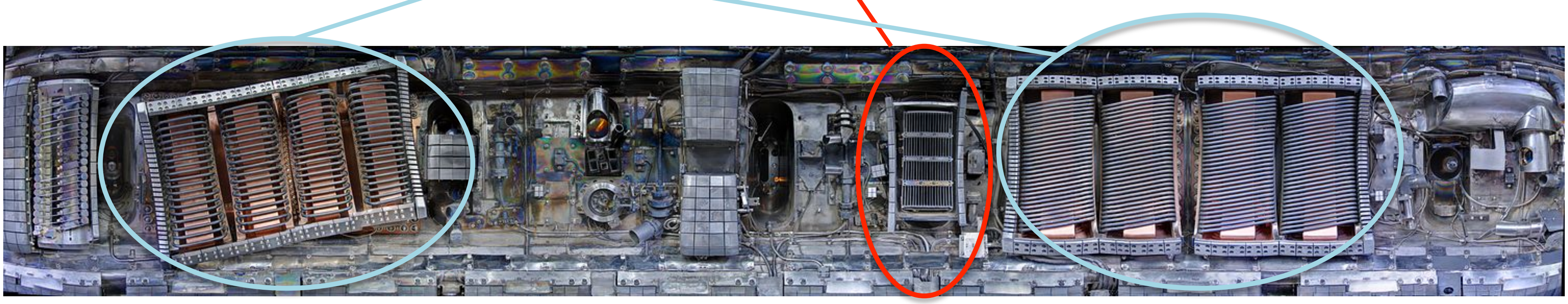


ICRF

Lower Hybrid Waveguide Launcher

Ion Cyclotron Launchers

Alcator C-Mod Outboard wall unfolded



First Wall - Loading

Steady state heat loading: <0.5 MW/m^2 (radiation); ITER considers some other sources but caps the heat load at <0.5 MW/m^2

→ We have considered up to 2 MW/m^2 for seconds in ACT, as fusion power excursion or plasma motion during operation phases, exploring maximum FW heat flux designs

→ ITER assumes 5 MW/m^2 maximum heating during startup, VDE disruptions, and in-active x-pt region

Blobs: particles and heat emerging from plasma into SOL resulting from turbulence, *we do not have a prescription*, and are relying on distance to dissipate these (SOL min thickness is 10 cm)

Steady state particle flux: thermal and some higher energy, LLNL to supply estimate for these (includes charge exchange)

ELMs: use heat flux prescription from experiments as done for ITER, particle flux is hard to find

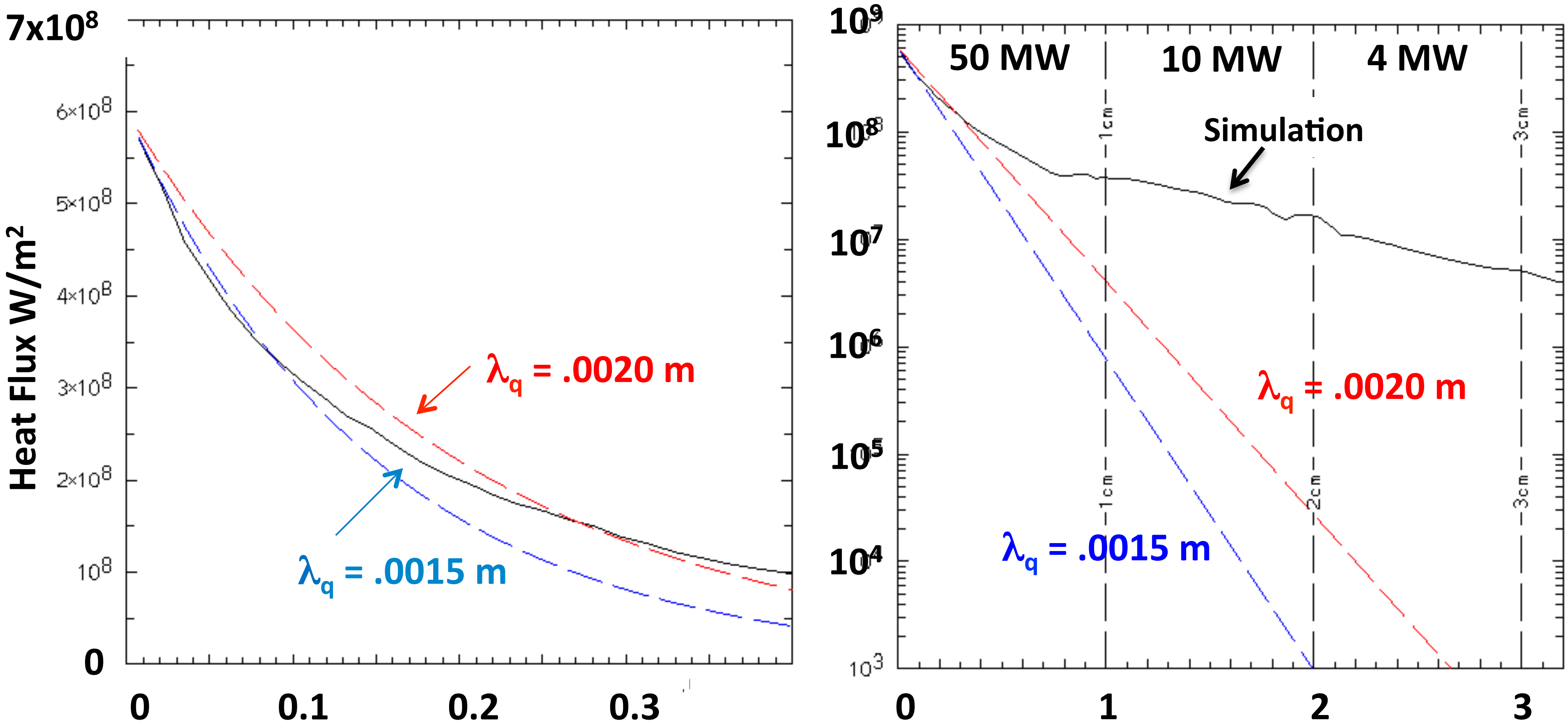
Disruption: use prescription from experiments as done for ITER, assume mitigated, radiative heat flux (midplane disruption)

Erosion/re-deposition/migration: LLNL to estimate erosion, prompt re-deposition, and migration

What is reconstituted surface thickness after what exposure, like erosion? We would consider the material no longer functional

2D SOL Modeling Integrating the Plasma Edge into Device

Heat flux into divertor – mapped to midplane



LLNL, Rognlien & Rensink

Radial Distance (cm) from separatrix

First Wall, cont’d

Design for FW (IB and OB)

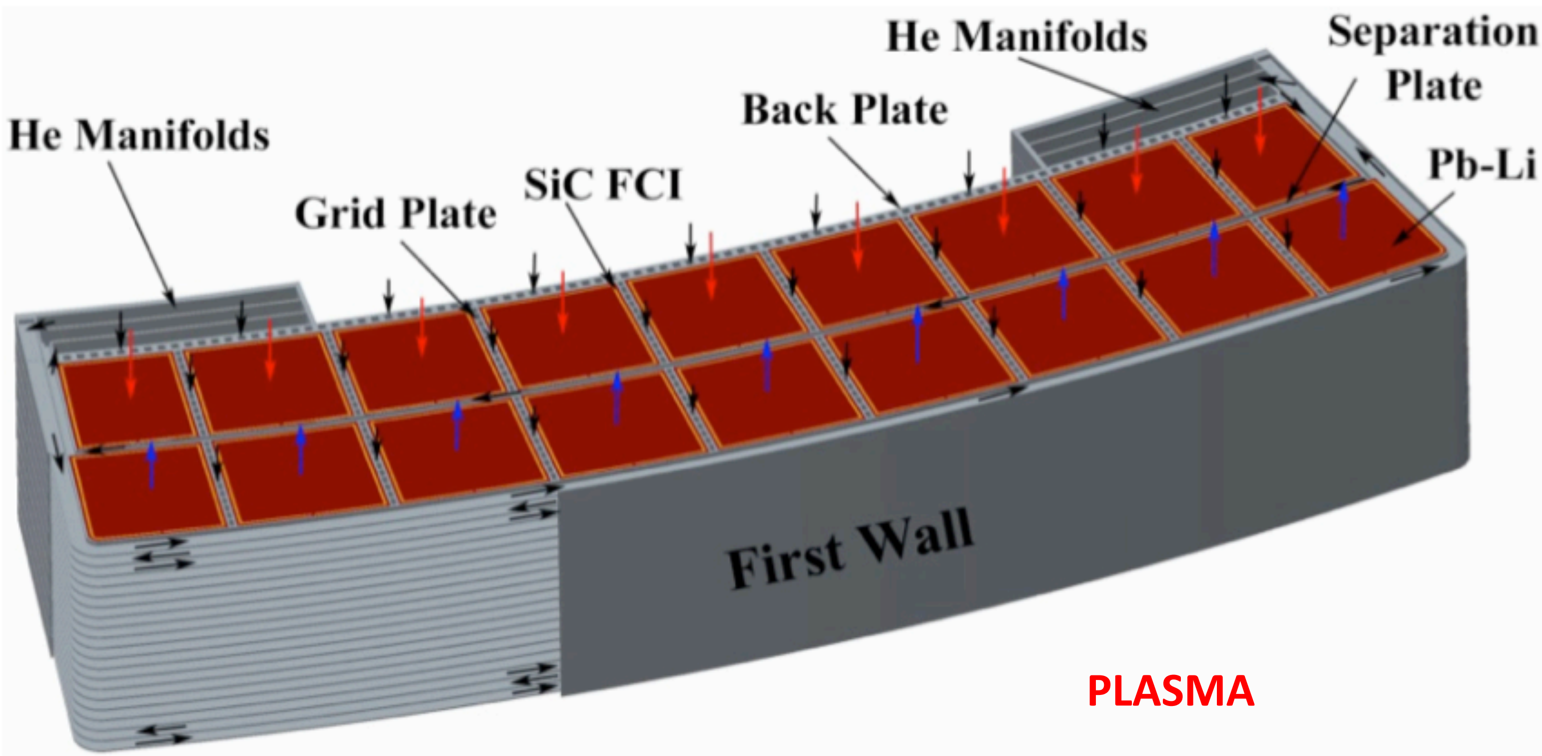
- RAFM or variant (blanket structural material)
- W coating 200 microns (sprayed)

0.02 cm W

0.4 cm RAFM

3.0 cm He channel

0.4 cm RAFM



Divertor - Loading

Steady State heat loading: using $q_{div}^{peak} = P_{SOL} f_{psi} f_{tilt} f_{vert} \{ f_{rad}/A_{div,rad} + (1-f_{rad})/A_{div,cond}\}$ in systems analysis, and 2D SOL solutions from UEDGE analysis

Use Fundamenski formula for λ_{int}, gives ~ 5 mm for FNSF (in systems analysis)

Steady state particle flux: estimate from LLNL analysis (expect ~100x FW value)

ELMs: use heat flux prescription from experiments as done for ITER, conducted heat load with some expansion, and recalculate the inter-ELM (or steady) heat flux

Examining melting threshold as function of q_{ELM} and $q_{inter\text{-}ELM}$

Disruption: : use prescription from experiments as done for ITER, assume mitigated, conducted heat load with large expansion factor in divertor

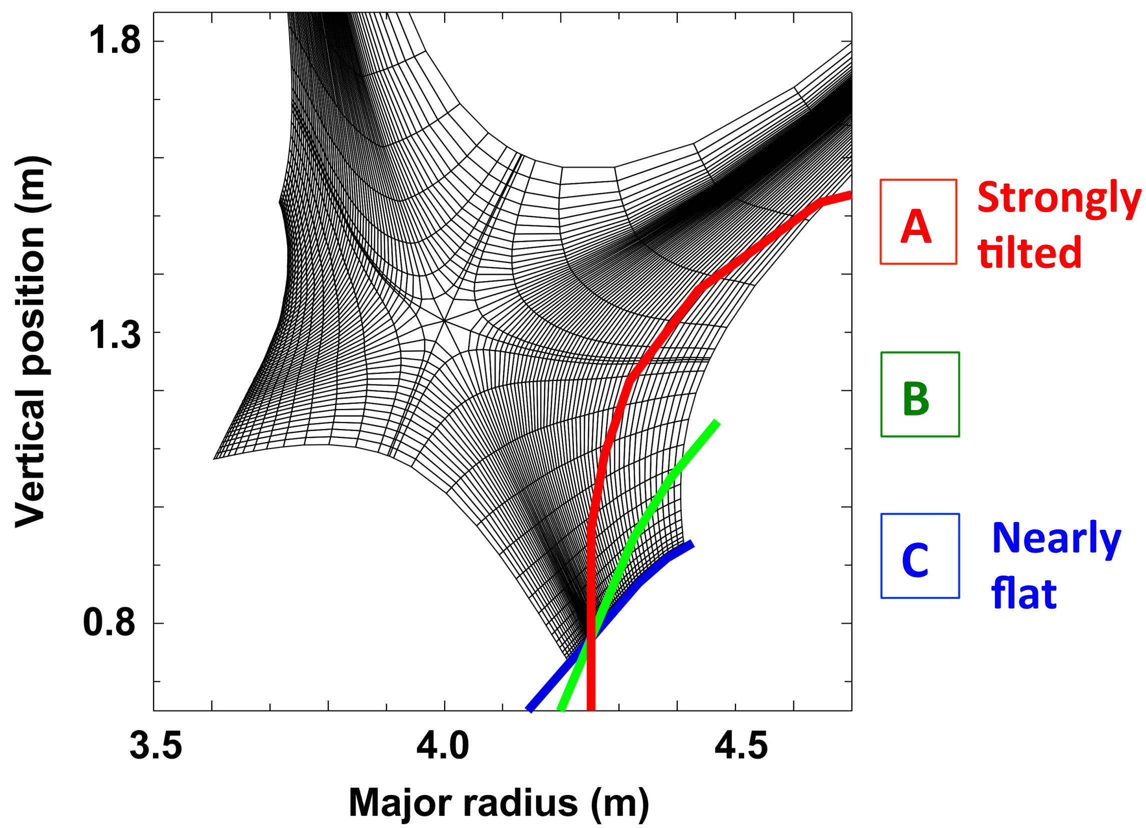
Erosion/re-deposition/migration: LLNL to estimate erosion, prompt re-deposition, and migration

What is reconstituted surface thickness after what exposure, like erosion? We would consider the material no longer functional

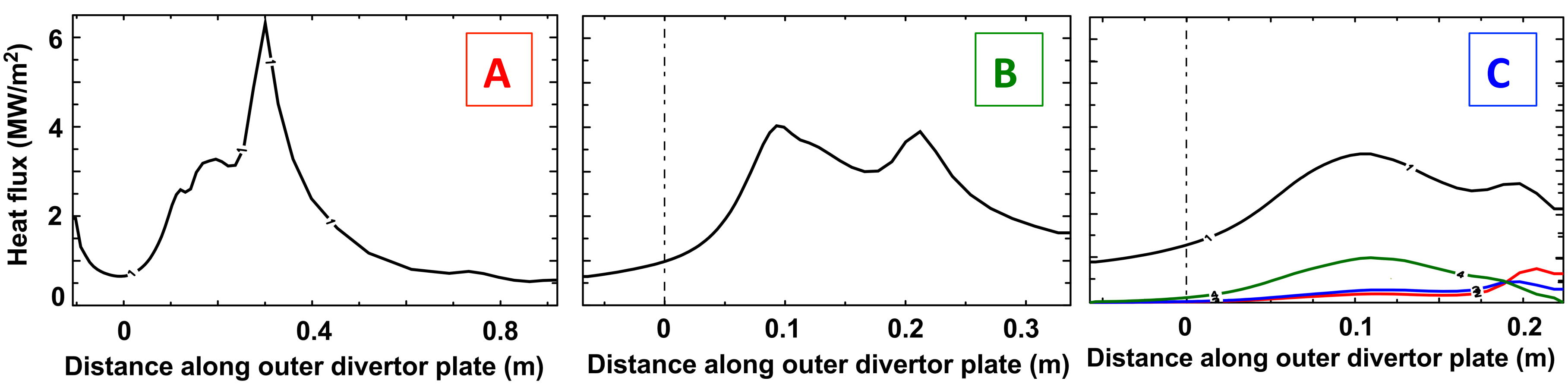
Divertor configurations being examined:

Tilted plate, ITER-like → partial detach

Wide slot → full detach



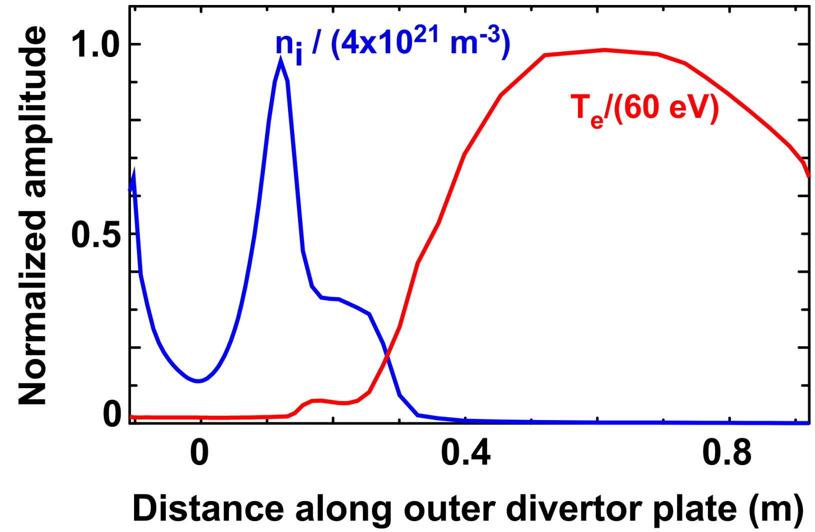
LLNL, Rognlien & Rensink

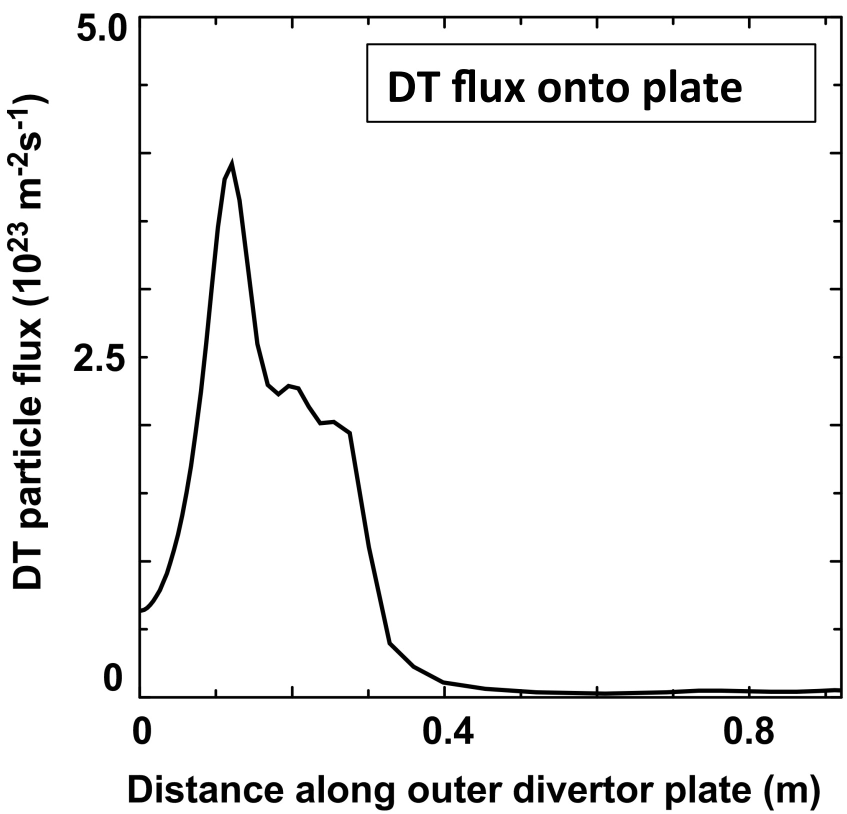


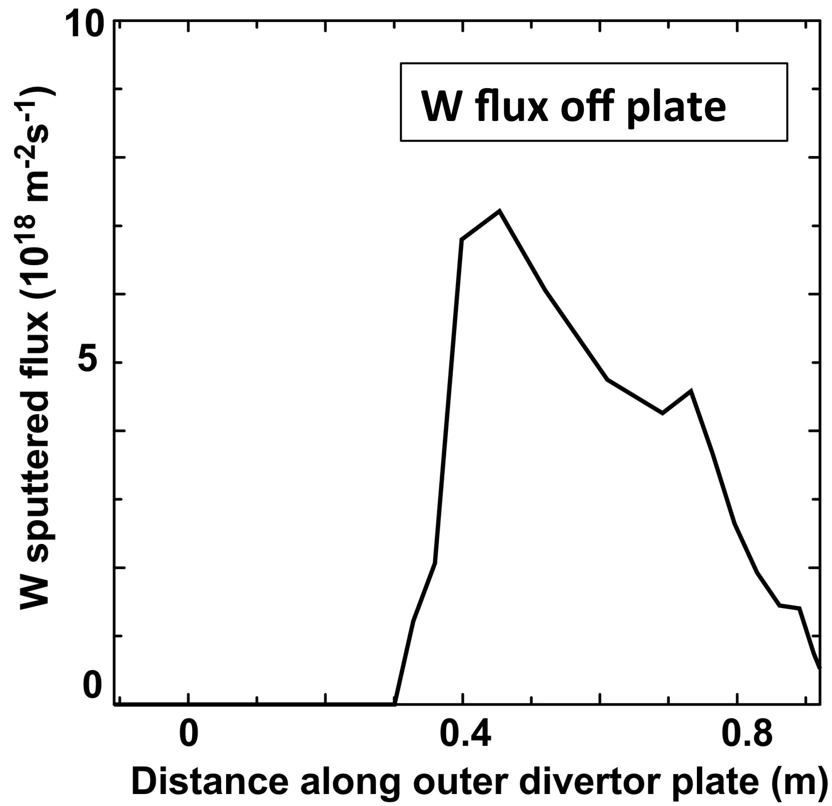
Using 2D SOL analysis to access particle loading parameters – Tilted Plate, ITER-like

LLNL, Rognlien & Rensink

Tilted ITER-like divertor configuration

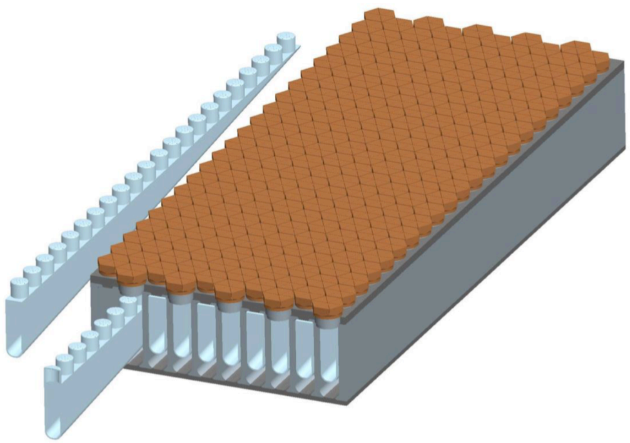


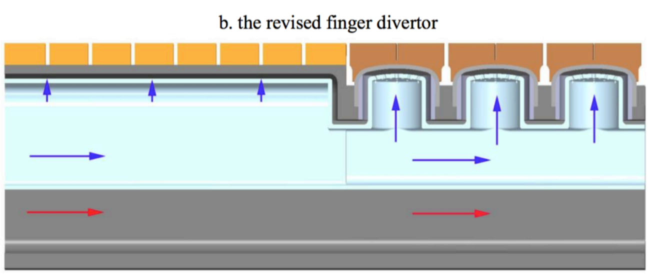




Divertor, cont'd

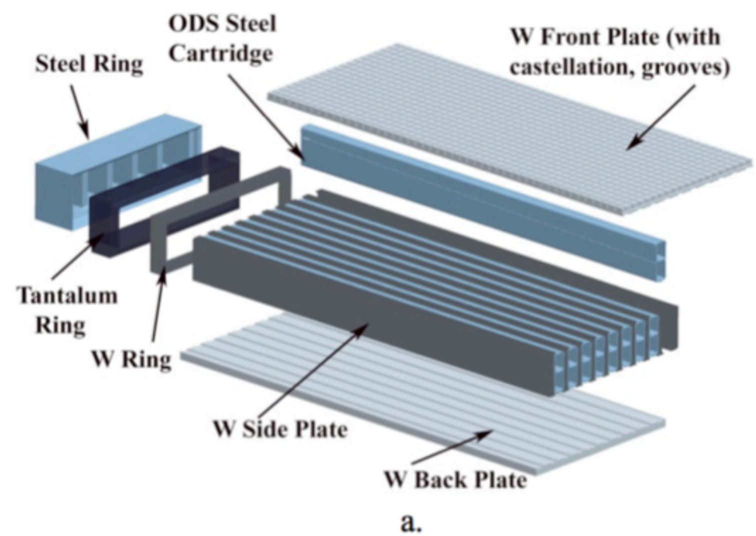
Finger divertor, > 10 MW/m²

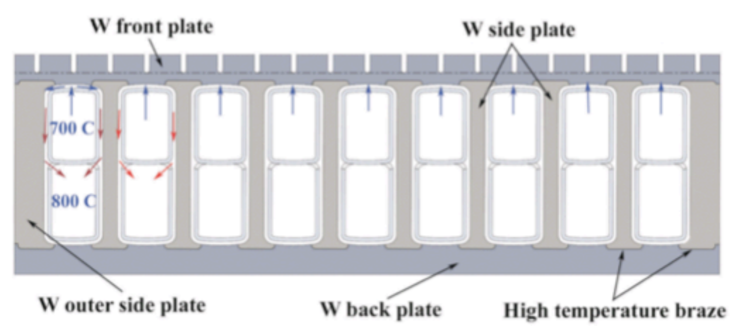




Combination finger and plate to accommodate spatially varying heat flux

Plate divertor, < 10 MW/m²





The FNSF is a One of a Kind Facility that Must Bridge the Tremendous Gap from ITER to DEMO and Power Plants

The FNSF takes a significant fusion nuclear and fusion plasma step beyond ITER and present operating tokamaks

The deliberate caution in taking this step is driven by the complexity of the the simultaneous fusion neutron and multi-factor non-nuclear environmental parameters seen by the materials/components

Separate materials qualification with fusion neutrons and non-nuclear integrated testing should provide a sufficient basis for the FNSF, but ultimately the FNSF will provide the in-service materials basis to move to power production with the DEMO and commercial PPs

This activity is trying to identify what the FNSF must demonstrate, identify the R&D program to prepare for the FNSF operation, and establish its connection to the demonstration and commercial power plants

BACKUP SLIDES

Large departure from PP

Some departure

Small departure

	minimal	**moderate**	maximal	Power plant
Plant DT operations	~ 15 yr	~ 25 yr	~ 35 yr	47 yr (40 FPY)
Peak neutron wall load, MW/m^2	1.0	1.5	2.25	2.25
Plasma on-time per year	10-35%	10-35%	10-45%	85%
Max dpa on first wall (or max dpa to replace)	5 -18,36	7 - 37,74	10 - 70,140	150-200
Q_{engr}	<< 1	< 1	> 1	4
Tritium breeding ratio	< 1	~ 1	> 1	1.05
Plant life, peak dpa	32, 50	88, 126	202, 274	765
TF/PF magnet	Cu	LTSC or HTSC	LTSC or HTSC	LTSC or HTSC
Vacuum vessel material	SS	Baintic steel	Bainitic steel	Bainitic steel
Divertor	W/CuCrZr/H_2O	W/W/He	W/W/He	W/W/He

Min, Mod, Max Programs

	He/H	DD	DT	DT	DT	DT	DT	DT	PP
Minimal									
Yrs (minimal)	1.5	2.5	3	4	5	5			
$<N_w>_{peak}$			**1.0**	**1.0**	**1.0**	**1.0**			
Plasma on-time %			15	25	35	35			
Neutron damage (dpa)			**4.5**	**10**	**17.5**	**17.5 or 35**			
Blanket									
Moderate									
Yrs (moderate)	1.5	2.5	3	5	5	7	7		40 FPY
$<N_w>_{peak}$			**1.5**	**1.5**	**1.5**	**1.5**	**1.5**		2.0-3.3
Plasma on-time %			15	25	35	35	35		85
Neutron damage (dpa)			**7**	**19**	**26**	**37**	**37 or 74**		100-150
Blanket			RAFM 400C	RAFM ODS 450-500C	RAFM NS 500C	RAFM NS 550-600C	RAFM NS 550-600C		
Maximal									
Yrs (maximal)	1.5	2.5	3	5	5	7	7	7	
$<N_w>_{peak}$			**2.25**	**2.25**	**2.25**	**2.25**	**2.25**	**2.25**	
Plasma on-time %			15	25	35	35	45	45	
Neutron damage (dpa)			**10**	**28**	**39**	**55**	**71**	**71 or 142**	
Blanket									

Why Pursue a Smaller First Step, like the FNSF?

Untested regime of fusion neutrons on multi-materials under multi-factor environment

Before FNSF we would have in hand:

- Fusion relevant neutron exposure of individual materials
- Fission exposure of small subassemblies (breeder and structural material)
- Non-nuclear fully integrated "as much as possible" FW/blanket, divertor, other PFC testing

Fission experience with materials (learned from PWR and breeder development programs)

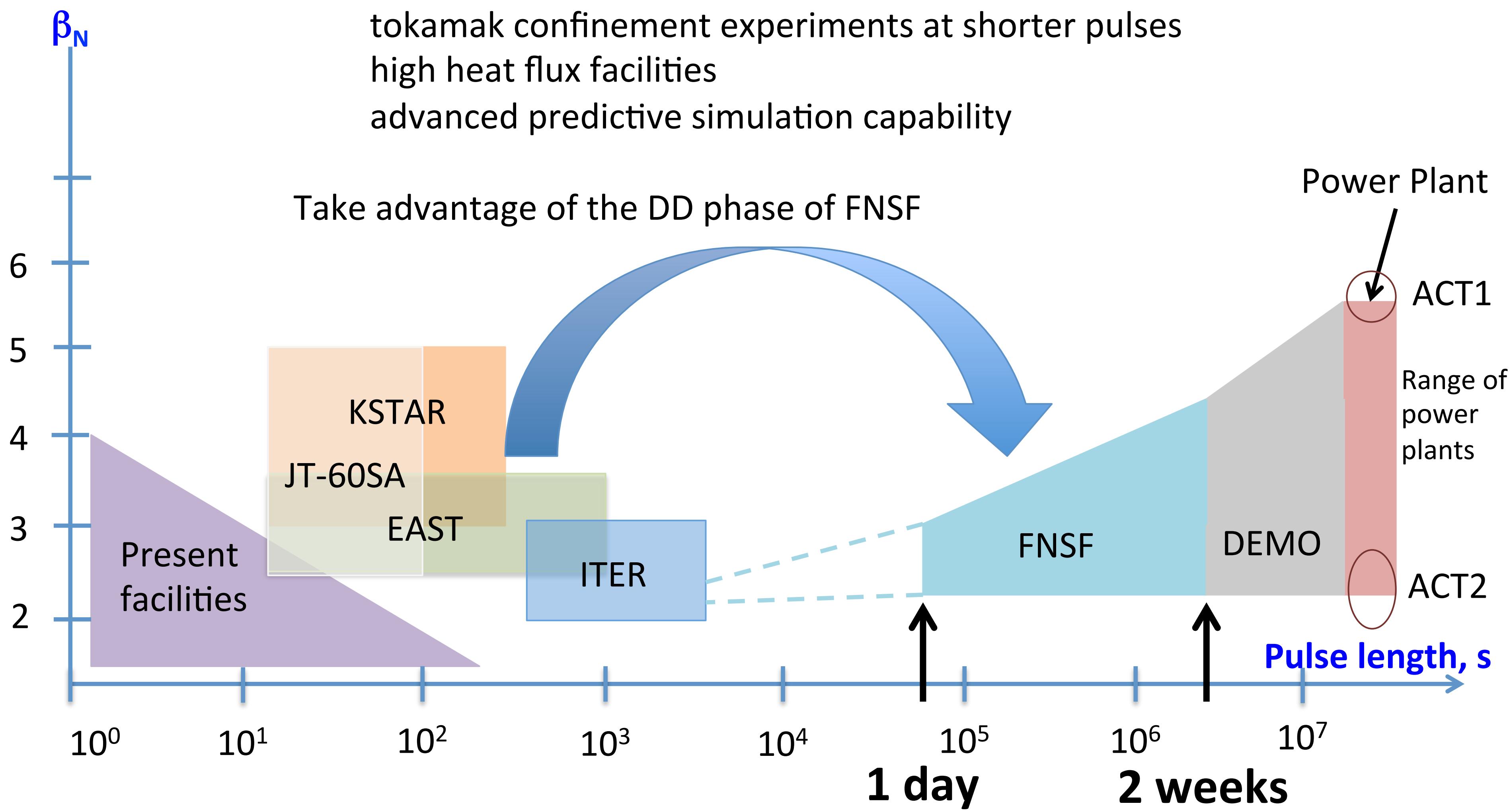
- Extreme sensitivity of swelling with temperature
- Impacts of irradiation dose rate increased hardening and threshold for swelling
- Impacts of smaller constituents ~ 0.5 wt% can lead to positive and negative effects
- Surface conditions, welds, and metallurgic variability provided wide variations in irradiation behavior
- Incubation periods that delay the emergence of a phenomena
- Simultaneous multiple variable gradients (neutron fluence, temperature, stress) on crack behavior

→ Several critical materials behaviors led to major disturbances in the development program for the liquid metal fast breeder program (Bloom et al, JNM 2007 & Was, JNM 2007)

Goal is to establish the database on all components in the fusion neutron environment and in the overall environment before moving to larger size and routine electricity production

The Plasma Durations Required in the FNSF is a Large Leap Compared to Present/Planned Tokamaks

Before the FNSF, must combine
- ultra-long pulse linear plasma facilities
- tokamak confinement experiments at shorter pulses
- high heat flux facilities
- advanced predictive simulation capability



Plasma Strategy – Finding Plasma Solutions That Can Provide a Robust Basis for the FNSF

Access very long plasma on-time, very high duty cycle → provide a given neutron wall loading

Very high f_{NICD} (>0.85)	⟷	Steady state (f_{NICD} = 1)
$\beta_N < \beta_N^{no\ wall}$	⟷	$\beta_N \geq \beta_N^{no\ wall}$
$n \leq n_{Gr}$	⟷	$n \geq n_{Gr}$
$q_{div}^{peak} < 10\ MW/m^2$	⟷	$10 < q_{div}^{peak} < 20\ MW/m^2$
$\kappa < 2$	⟷	$\kappa \geq 2$
$B_{tcoil}/\langle j_{TF}\rangle$ (ITER)	**High field in the plasma**	$B_{tcoil}/\langle j_{TF}\rangle$ (adv Nb_3Sn)

Plasma *Performance and Duration* in DIII-D and JT-60U Looking at Experiments for Guidance

	JT-60U	JT-60U	JT-60U	DIII-D	DIII-D	DIII-D
β_N	2.3	2.4	1.7	3.5*	2.0	3.1-3.4*
$\tau_{flattop}/\tau_{CR}$	13.1	2.8	2.7	2.0	> 2	~ 0.4-1.0
q_{95}	3.2	4.5	~ 8	6.7	4.7	5.0-5.5
f_{BS}	35-40%	45%	80%	40-50%		~60%
f_{NI}		90%	100%	75%		80-100%
H_{98}		1.0	1.7	1.0	1.3	$\geq$ 1.2-1.3
q_{min}	~ 1	~ 1.5		1.5		1.4
	hybrid	~ steady state	steady state	→ steady state, off-axis NB	QH-mode, no ELMs	steady state

EAST and KSTAR will soon contribute

*utilize active error field correction, plasma rotation, $\beta_N \sim 1.15 \times \beta_N^{no\ wall}$

Additional experiments on JT-60U and DIII-D have 1) approached and exceeded density limit, 2) high radiated power in the plasma and divertor, 3) avoiding or actively suppressed NTMs, 4) low plasma rotation, and 5) PFC materials

Systems Code Identification

Large scans over R, B_T, q_{95}, β_N, Q, Z_{eff}, n/n_{Gr}

$<j_{TF}>$ = 15 MA/m^2

$f_{div,rad}$ = 90% ($\lambda_{pow}^{Fundamenski}$)

Filters for solutions

$\beta_N \leq 2.6^*$

$q_{div}^{peak} \leq 10$ MW/m^2

$N_w^{peak} \geq 1.5$ MW/m^2

$B_T^{coil} \leq 16$ T (LTSC)

IB Radial build from neutronics:

$\Delta_{FW/blkt}$ = 50 cm

Δ_{SR} = 20 cm

Δ_{VV} = 10 cm

$\Delta_{LT\ shield}$ = 23 cm

Δ_{gaps} = 20 cm

*examining benefits of RWM feedback to raise this toward 3.0-3.2

A = 4	
R, m	4.80
κ_X, δ_X	2.2, 0.63
I_P, MA	7.87
B_T, B_T^{coil}, T	7.5, 15.85
$<j_{TF}>$, MA/m^2	15 MA/m^2
β_N^{th}, β_N^{fast}	2.2, 0.23
q_{95}	6.0
H_{98}	0.99
f_{BS}	0.52
Z_{eff}	2.43
n/n_{Gr}	0.90
n(0)/<n>, T(0)/<T>	1.4, 2.6
P_{fusion}, $P_{rad,core}$, $P_{rad,div}$, P_{aux}, MW	517, 60, 160, 130
Q, Q_{engr}	4.0, 0.86
η_{CD}, A-m^2/W	0.2 (assumed)
$<N_w>$, N_w^{peak}, MW/m^2	1.18, 1.77
q_{div}^{peak} (OB, IB), MW/m^2	10.7, 3.9

Design Rules: Allowables

- *Allowable primary membrane stress intensity (S_m)*: it is a temperature (T) and fluence (ϕ_t) dependent allowable stress intensity defined as the least of the quantities:

$$S_m = min(\frac{1}{3}S_{u,min}(RT,0), \frac{1}{3}S_{u,min}(T,0), \frac{1}{3}S_{u,min}(T,\phi_t), \\ \frac{2}{3}S_{y,min}(RT,0), \frac{1}{3}S_{y,min}(T,0), \frac{1}{3}S_{y,min}(T,\phi_t))$$

- *Allowable total stress intensity (S_d)*: it is a temperature (T), fluence (ϕ_t), and r-factor dependent allowable stress intensity for total primary plus secondary stress in radiation embrittled materials, defined as:

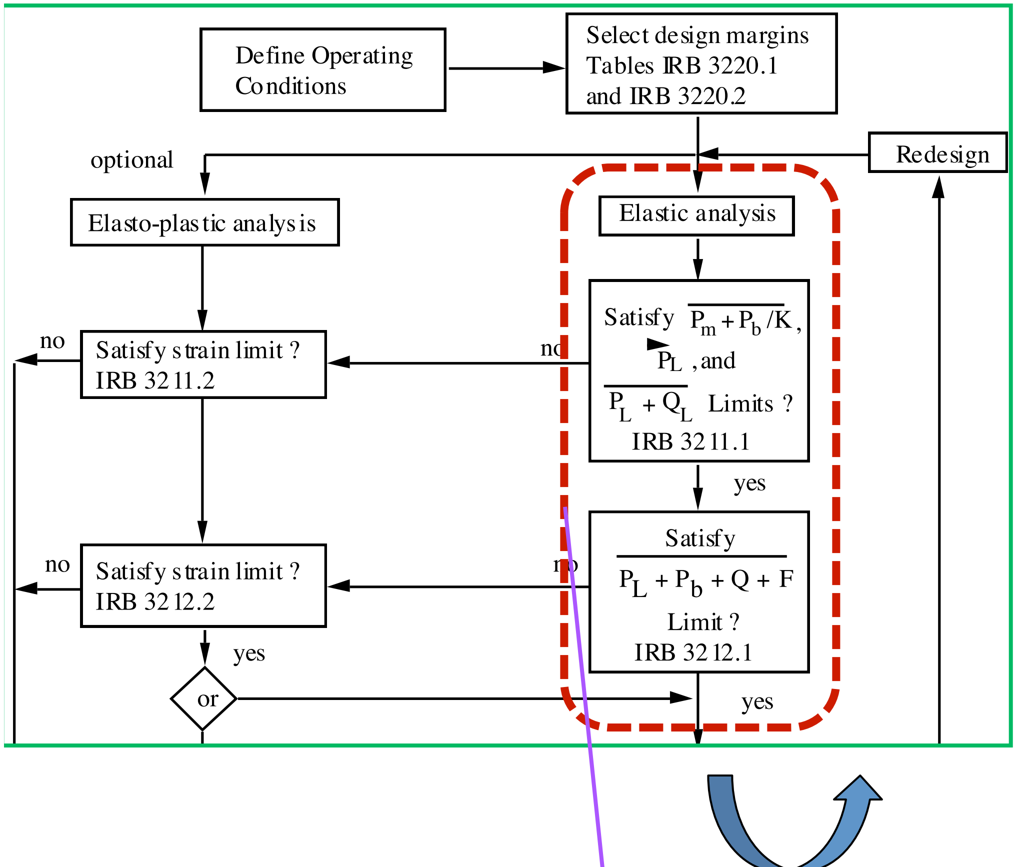
$$S_d = \frac{2}{3}\left(S_{u,min}(T,\phi_t) + \frac{E\epsilon_{tr}(T,\phi_t)}{r \times TF}\right)$$

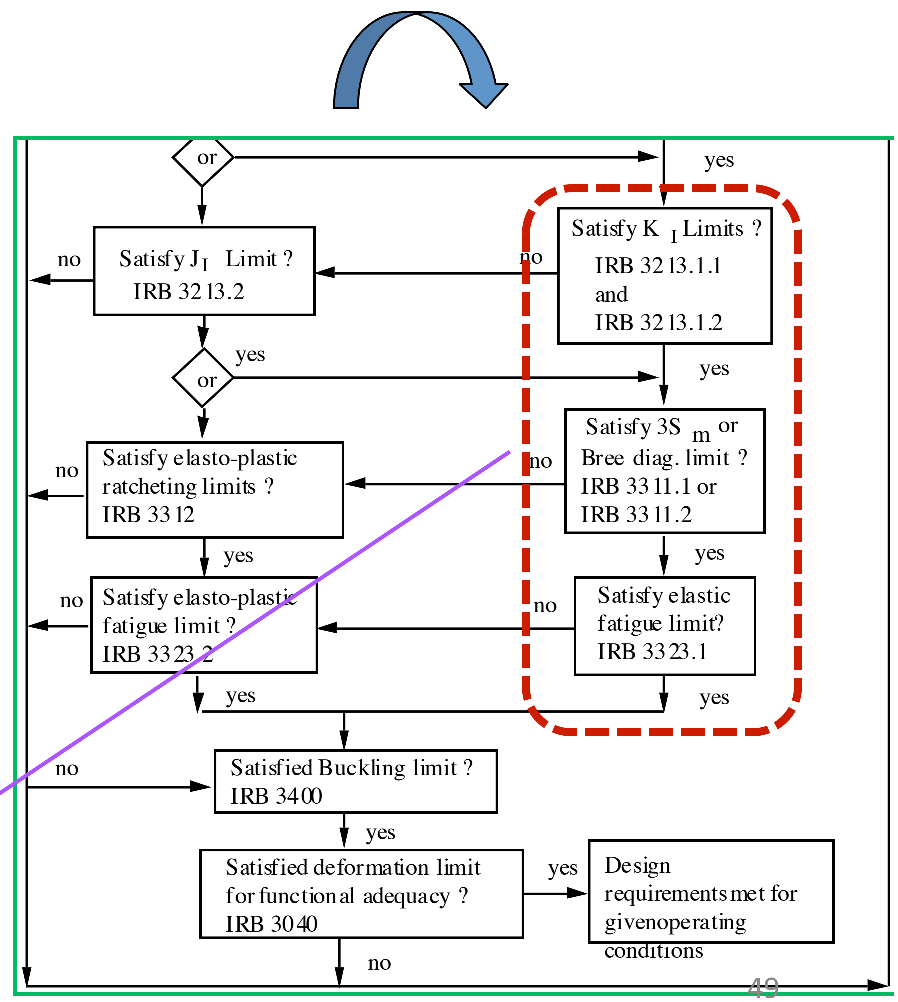
- *Allowable primary plus secondary membrane stress intensity (S_e)*: it is a temperature (T) and fluence (ϕ_t) dependent allowable stress intensity for a material with severe loss of uniform elongation due to irradiation and is defined as follows:

$$S_e = \begin{cases} \frac{1}{3}S_{u,min}(T,\phi_t) + \frac{E\alpha_1}{r_1}[\epsilon_u(T,\phi_t) - 0.02] & if \quad \epsilon_u > 2\% \\ \frac{1}{3}S_{u,min}(T,\phi_t) & if \quad \epsilon_u < 2\% \end{cases}$$

Ghoniem, Nasr M., Giacomo Po, and Shahram Sharafat. "Deformation mechanisms in ferritic/martensitic steels and the impact on mechanical design." *Journal of Nuclear Materials* 441.1 (2013): 704-712.

Analysis Flow Chart for Low-Temperature SDC-IC Design Rules for Given Operating Conditions





DCLL TBM Screening Analysis Performed for Normal Operating Conditions

Definition of Design SDC-IC Criteria

LOW TEMPERATURE DESIGN RULES:	
Necking and Plastic Instability Limit - Primary membrane stress (Immediate plastic collapse and plastic instability)	$\overline{P_m} \le S_m(T_m, \phi t_m)$
Necking and Plastic Instability Limit - Primary membrane and bending stress	$\overline{P_L + P_b} \le K_{eff}\, S_m(T_m, \phi t_m)$
Local primary membrane stress – (Immediate plastic collapse and plastic instability)	$\overline{P_L} \le \min\left[1.5\, S_m(T_m, \phi t_m),\, S_{y,min}(T_m, \phi t_m)\right]$
Local primary membrane stress – (Immediate plastic collapse and plastic instability)	$\overline{P_L} \le 1.1\, S_m(T_m, \phi t_m)$
Plastic Flow Localization Limit - Primary plus secondary membrane stress (Immediate plastic flow localization)	$\overline{P_L + Q_L} \le S_e(T_m, \phi t_m)$
Ductility Exhaustion Limit - (Local fracture, exhaustion of ductility)	$\overline{P_L + P_b + Q + F} \le S_d(T, \phi t, r_2)$
Ductility Exhaustion Limit – Without peak stress (Local fracture, exhaustion of ductility)	$\overline{P_L + P_b + Q} \le S_d(T, \phi t, r_3)$
HIGH TEMPERATURE DESIGN RULES:	
Creep Damage Limit	$P_L + {}^{P_b}/_{K_t} \le S_t$
Ratcheting Limit - Progressive deformation or ratcheting	$X \cdot Y \le 1 \quad for \quad 0 \le X \le 0.5$
Ratcheting Limit - Progressive deformation or ratcheting	$\frac{Y}{4(1-X)} \le 1 \quad for\ 0.5 \le X \le 1.0$
Ratcheting Limit - Progressive deformation or ratcheting: TEST No. A.2	$X + Y \le 1$

Some F82H Properties

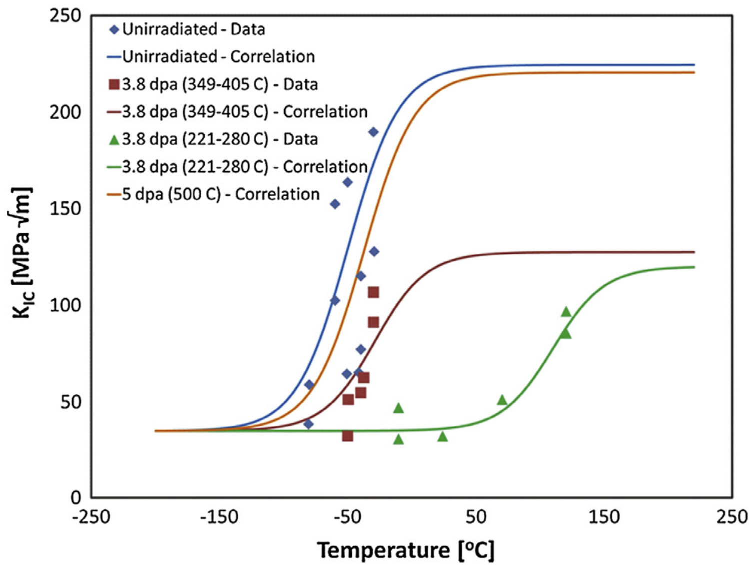


Fig. 10. $K_{IC(irr)}$ correlation (Eq. (26)) fitted to experimental conditions [22].

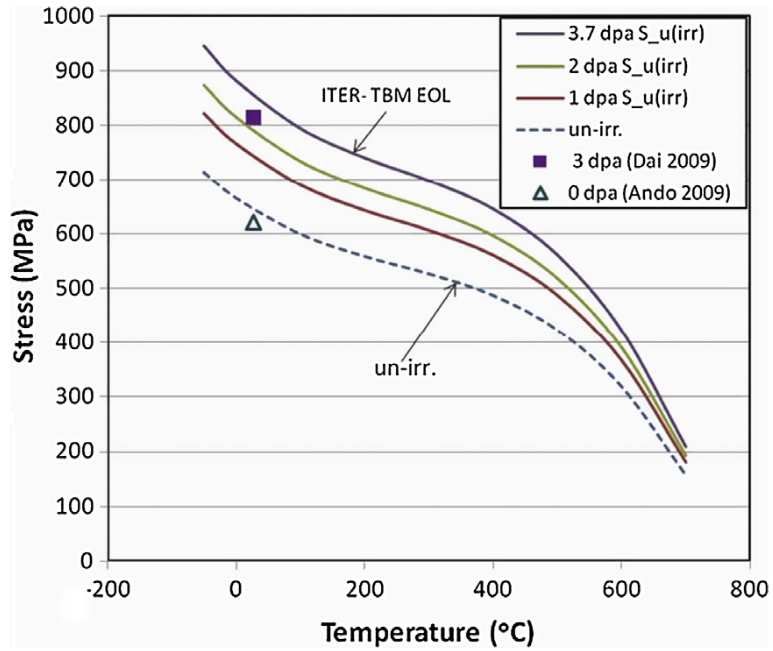


Fig. 5. Irradiated ultimate tensile ($S_{u(irr)}$) strength of F82H (IAE Heats 9741, 9753) as a function of temperature based on correlations given by Eq. (25) (3 dpa at T_{irr} = 85–100 °C and 180 He-appm[19]).

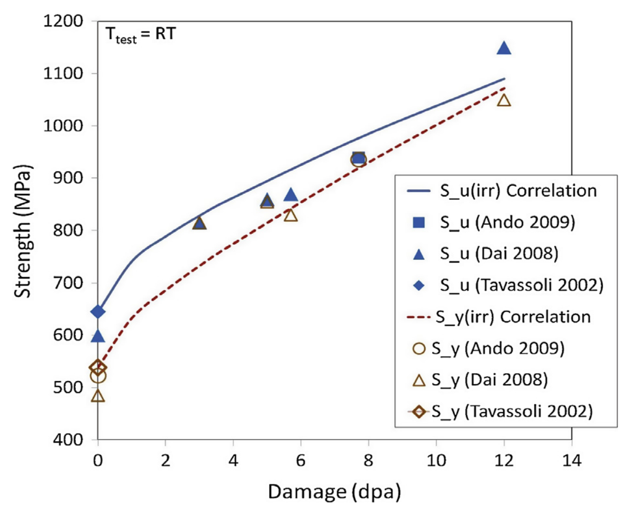


Fig. 4. Tensile yield and ultimate strengths for F82H as a function of displacement damage dose (HFIR: Tirr = 300 °C; STIP-I: Tirr = 90 °C to 375 °C – HFIR IEA Heat 9741, 9753; STIP-I: IEA Heat 974 [17]).

Sharafat, Shahram, Aaron T. Aoyama, and Nasr Ghoniem. "Assessment of the DCLL TBM Thermostructural Response Based on ITER Design Criteria."*Fusion Science and Technology* 60.1 (2011): 264-271.

What is the cumulative time required to perform inspections, minor maintenance, single sector removal, 16 sector removal

We want to figure out how much time is required for these anticipated activities, then we can add contingency for unanticipated activities

→ We can replace the "maintenance time" that we've tentatively allocated in the program, with a better estimate?...*hopefully to shorten the overall program time*

Diagnostics relied upon for continuous monitoring

Inspections ex-VVessel (1 week) assume 10 or these
Inspections in-VVessel (1 week) assume 10 of these

Minor maintenance ex-VVessel (1 week, includes testing) assume 5 or these
Minor maintenance in-VVessel (2 weeks, includes testing) assume 5 of these

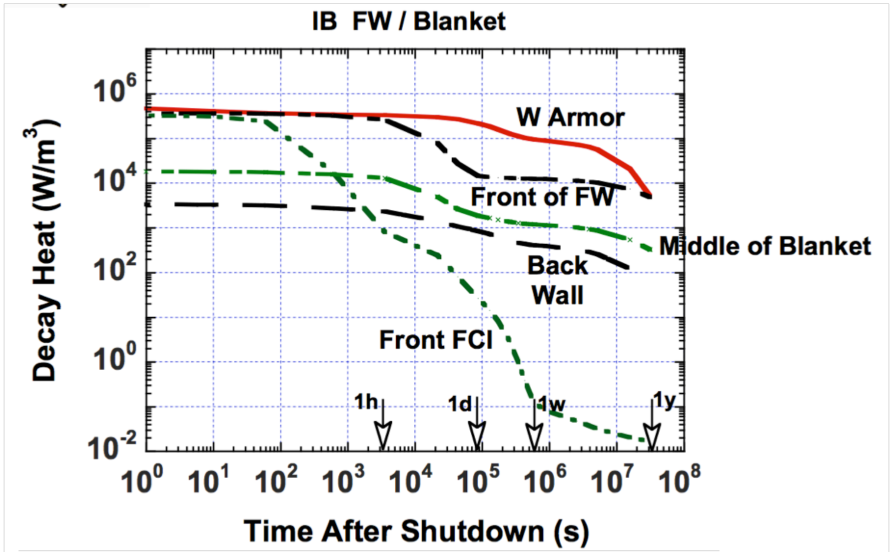
Major in-vessel, sector removal and replace (includes PFC, divertor, blanket, RF,TBM or other) (77 days (Les, 1 cask, shutdown, remove, replace, startup), 30 days assessment) assume 2 of these

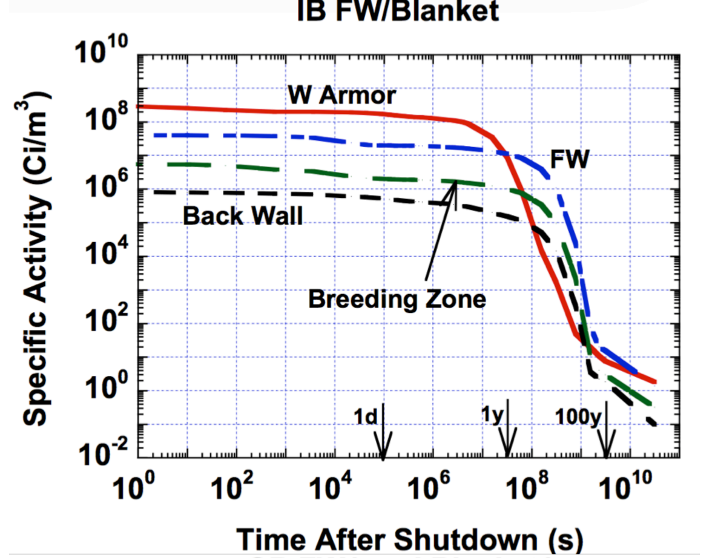
Major in-vessel, end of phase 16 sector removal, clean, replace .....(167 days, Les, 4 casks) assume 1 of these

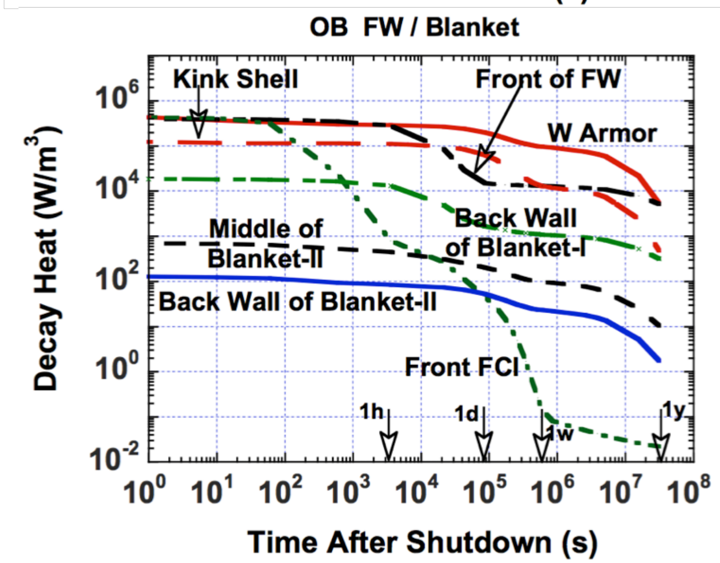
→ For Phase 3, (70+70+35+70+154+167 days) = 566 days
To this we must add the time to assess material and component behavior, make decisions on proceeding with phase, making changes to next phase, etc.

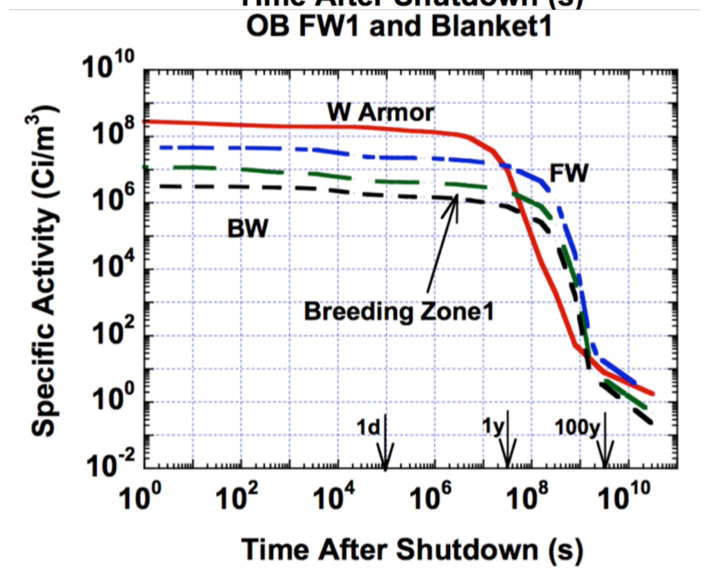
Decay heat and specific activity vs time for FNSF

L. El-Guebaly

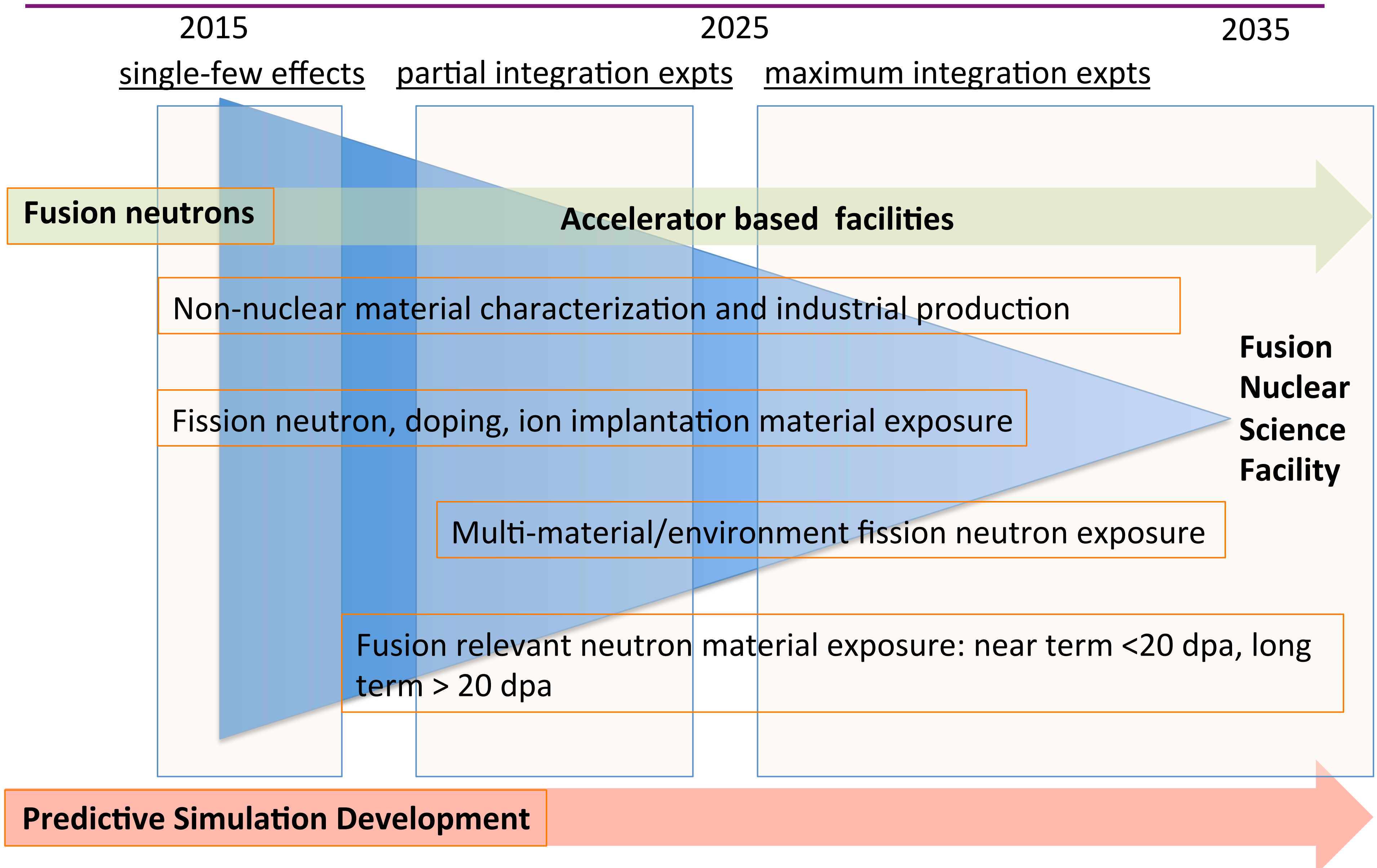




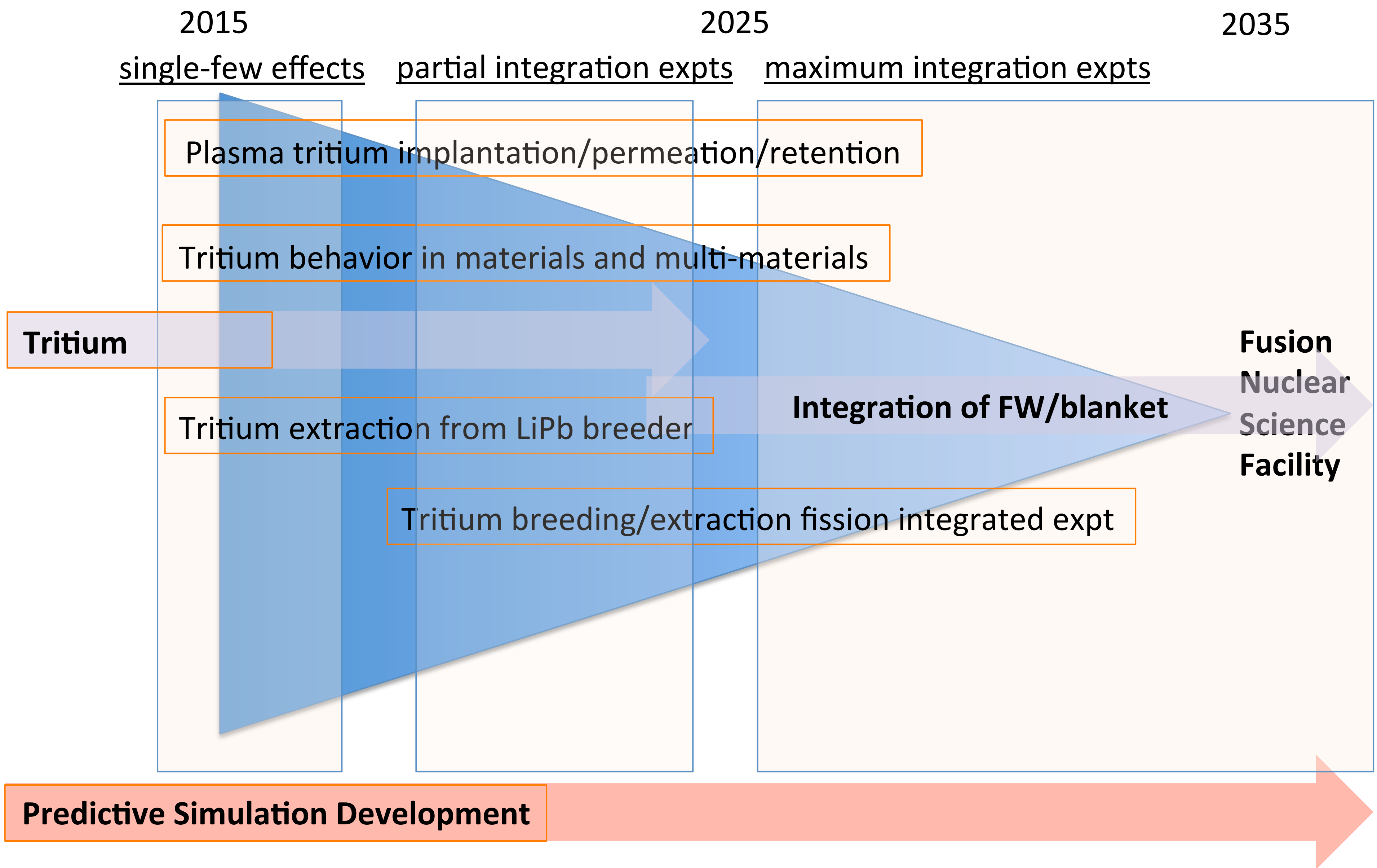




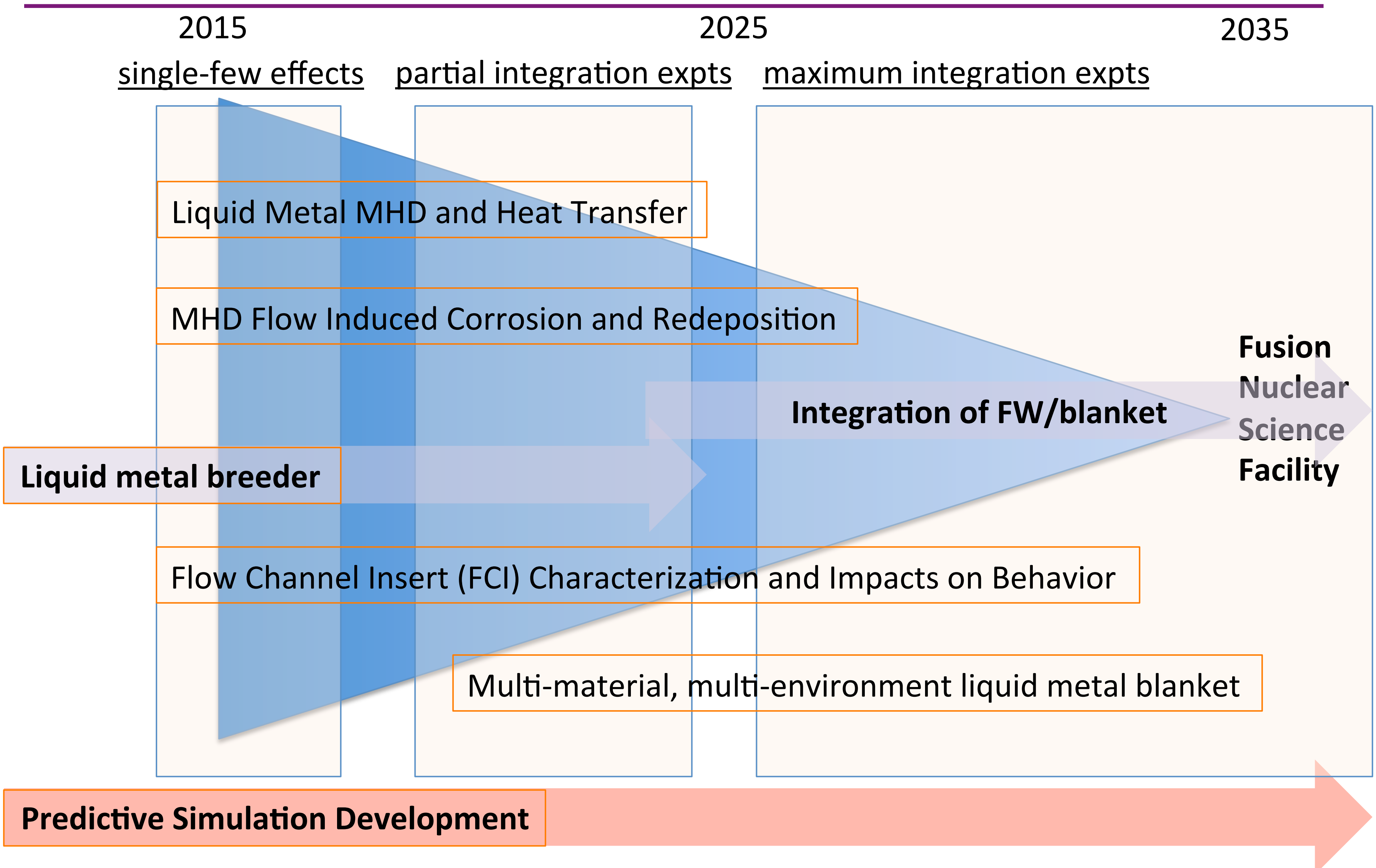
Zoom-In: *Fusion Nuclear Materials Science* (preliminary)



Zoom–In: *Tritium Science* Breakdown



Zoom-In: *Liquid Metal Breeder* Science



Flow Channel Insert – Fundamental to the DCLL Blanket Concept

The Flow Channel Insert (FCI) is what makes the LiPb concept reach a potentially attractive operating regime (high temperature and low LM pressure drop)

SiC-c is considered the material of choice, providing both low thermal and electrical conductivity, which allows sufficiently low pressure drops in the LiPb loop

Sandwich materials (Fe/Al_2O_3/Fe) have been proposed, but are significantly more limited (lower temperature)

We know very little about this application of SiC-c

Can the SiC-c avoid LiPb entering its matrix? Over long exposure times? At high temperature?

What is the SiC-c's behavior in the integrated environment (neutron damage, transmutations, temperature, vibrations, temp gradient,…..)

So far, the FNSF study has shown that the sandwich leads to excessive pressure drops (Smolentsev, UCLA), but may be due to the full banana blanket configuration rather than a modular blanket

LiPb liquid metal

What does this liquid look like when operated for long periods of time in contact with RAFM/SiC-c, at high temperature

Ultimately the LiPb cycle is much larger than the blanket, thru long pipe runs, tritium extraction, heat exchanger, and cleanup apparatus

What is in the otherwise stoichiometric $Li_{15.7}Pb_{84.3}$ liquid metal, inter-metallics, impurities, gases (H, He)

How do these affect the interaction of LiPb with the FCI and RAFM steels

We do not know the tritium properties of LiPb with sufficient accuracy to predict tritium behavior with the required accuracy

The electrical resistivity of LiPb has a 2 orders of magnitude spread

The tritium extraction permeator window material requires development, as part of the overall extraction system

The RAFM family of alloys – this is our only blanket structural material option

From the integrated systems studies perspective, the RAFM is not one alloy, but rather a broad range of alloys

Gen I – F82H, EUROFER

Gen II –

ODS(NS) –

Low, med and high Cr –

Significant database already exists and is being further developed

High fission dpa exposures
Industrial large heats

Has the basis on which future alloys can be developed in the timeframe for an FNSF

….goals arising from integrated design are to increase

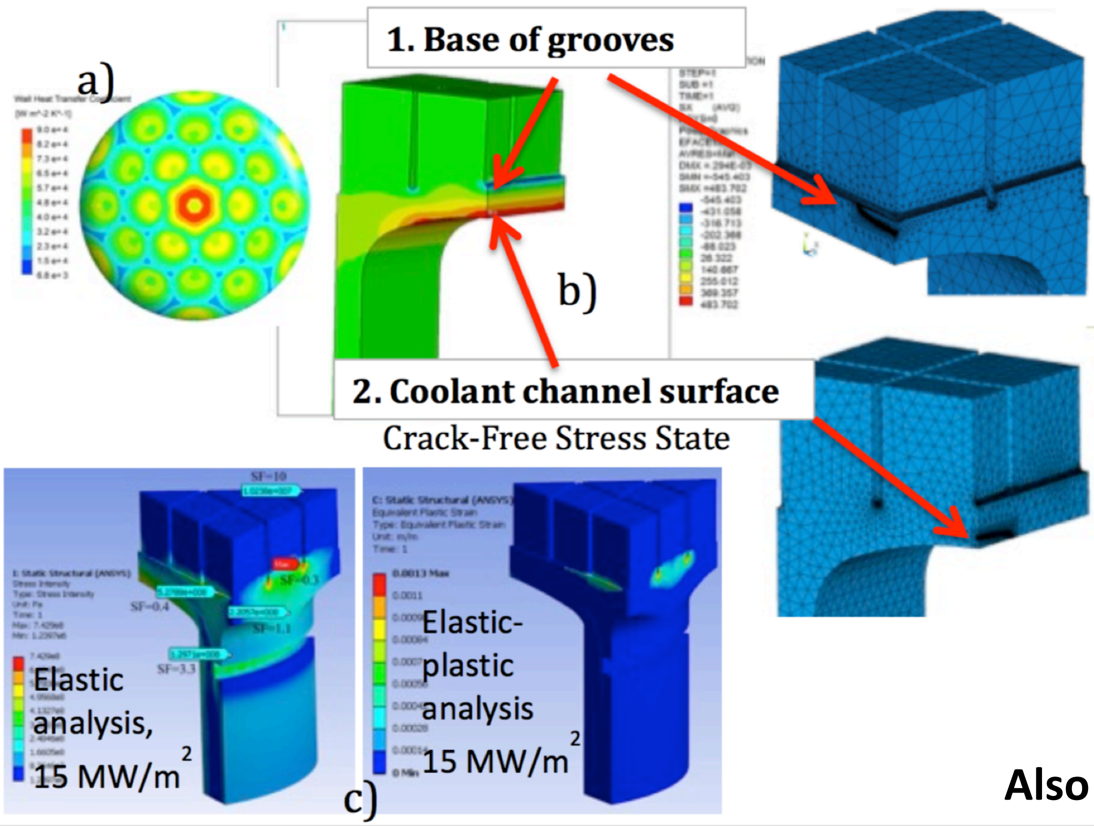
1) resistance to irradiation degradation (holding off He aggregation),
2) higher temperature operation, high creep rupture strength,
3) compatibility with LiPb,
4) optimizing RAFM variants for dpa, He, temperature environment

http://ceramics.org/wp-content/uploads/2011/08/applications-ceramic-apps-auto-hoffmann.pdf
http://www.forbes.com/2008/03/06/solutions-green-car-ceramics-oped-cx_atg_0307ceramics.html

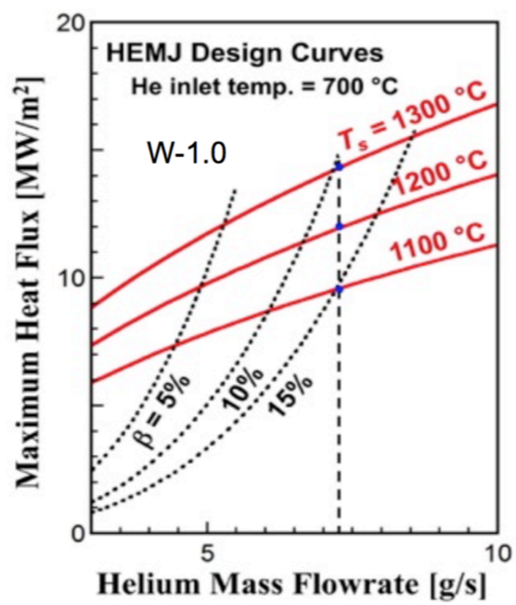
Analysis and Expts support the divertor design in the absence of neutrons & plasma

Materials data (even unirradiated) is not plentiful

CFD, Thermo-mechanics, fracture mechanics



GIT expts



Also expts at KIT with finger design