Fusion
Energy
Systems
Studies

Tokamak Fusion Nuclear Science Facility

C. E. Kessel
Princeton Plasma Physics Laboratory

Fusion Energy Systems Studies Team
J. Blanchard, A. Davis, L. El-Geubaly, L. Garrison, N. Ghoniem, P. Humrickhouse, Y. Katoh, A. Khodak, S. Malang, N. Morley, M. Rensink, T. Rognlien, A. Rowcliffe, S. Smolentsev, P. Snyder, M. Tillack, P. Titus, L. Waganer, A. Ying, K. Young, and Y. Zhai

**Fusion Power Associate's Annual Meeting, Wash. DC,
December, 16-17, 2015**

The Fusion Energy Systems Studies Team is Examining the Fusion Nuclear Science Facility

What does an FNSF have to accomplish?

How do we measure the FNSF progress for fusion development?

How does the FNSF accomplish its mission?

What is the pre-requisite R&D needed for an FNSF? What does the FNSF require from our program to succeed?

How does an FNSF fit in the larger fusion development program?

What critical insights about this facility can be uncovered, impacts of assumptions, technical choices and philosophies,…?

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		

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
- 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

ARIES-ACT2 (DCLL blanket) as power plant example

The pre-FNSF Component Development and Phased Operation on the FNSF are Essential for Success

We will have some failures, but the presence of constant failures are incompatible with the plasma-vacuum systems and the need for radioactive materials remote handling

We will use a high level of pre-qualification of materials and components (*NO cook and look!*)

We will test all materials in the fusion core up to the anticipated dpa level before operating to that dpa level, with fission and fusion relevant neutron exposures

We will test the most integrated prototype possible of blanket and divertor components before installation, in a non-nuclear fully integrated facility

On the FNSF, the phases rampup the operating parameters slowly to provide monitoring

The plasma durations, duty cycles, dpa's, and operating temperatures are advanced through the 1 DD, and 5 DT program phases

Inspections and autopsy of components is used to monitor evolution of materials, requiring highly efficient hot cell turn-around, during any given phase and at the end of a phase

Test blanket modules will be used for a "look forward", engineering testing, and backup blanket concepts, ad even material sample testing

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 500C		RAFM ODS (NS) 600C		RAFM ODS (NS) 600C			

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

The DEMO Program has been laid out to provide the rampup in dpa and demonstrate routine electricity production

R&D still required in early DEMO

	He/H		DD		DT		DT		DT		DT	PP
Yrs	1	1	2	1	6	1	6	1	8	1	8	40 FPY
Neutron wall load, MW/m^2					2.5		2.5		2.5		2.5	2.0-3.3
Plasma on-time, % /year			35-75		35		50		67		75	85
Plasma duty cycle, %			95		95		98		98		99	100
Neutron damage, dpa					52.5		75		134		150	100-150
blanket					RAFM nano 600C		RAFM nano 600C		RAFM nano 600C		RAFM nano 600C	

The FNSF will provide 37 or up to 74 dpa

What Types of FNSF’s Can We Envision

Minimal mission:

- Largely ignore reactor relevance
- Provide a neutron source
- TBR < 1

Moderate mission:

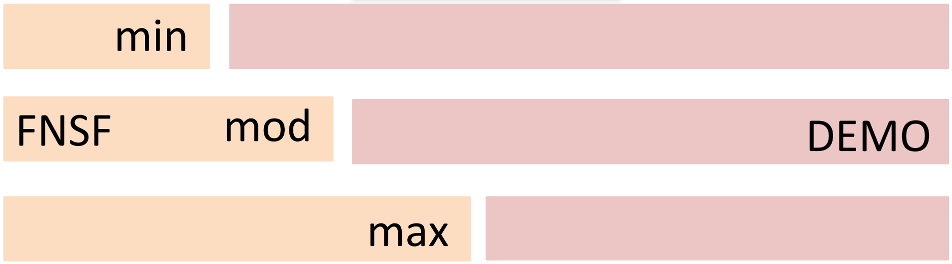
- Significant advance toward DEMO in most aspects
- Possibly provide electricity
- TBR ~ 1

Maximal mission:

- Provide net electricity
- Reach most or all DEMO parameters
- TBR > 1

Long term relevance weighs heavily on technical decisions, since there are too few devices for developing and demonstrating

	minimal	moderate	maximal	power plant
Plant lifetime	~15 yr	~25 yr	~35 yr	47 yr (40 FPY)
$N_W^{OB,peak}$, MW/m^2	1.0	1.5	2.25	
Plasma on-time per year	10-35%	10-35%	10-45%	85%
Max dpa on OB FW (dpa to replace)	5-20	10-40	10-70	150-200*
Q_{engr}	<<1	< 1	> 1	4
TBR	≤ 1	~ 1	> 1	1.05
Plant life peak dpa at OB FW	32	88	202	840
Plant lifetime	~15 yr	~25 yr	~35 yr	47 yr (40 FPY)
TF/PF	Cu	LTSC or Cu	LTSC or HTSC	LTSC or HTSC
VV	SS	Bainitic	Bainitic	Bainitic



Blanket Testing, Each Sector Specified

Begin with lower performance DCLL blanket

- T_{LiPb}^{exit} = 450 C, RAFM steel structure

Backup blankets are HCLL and HCCB

Full sector

- Partial phase life (autopsy)
- Full phase life

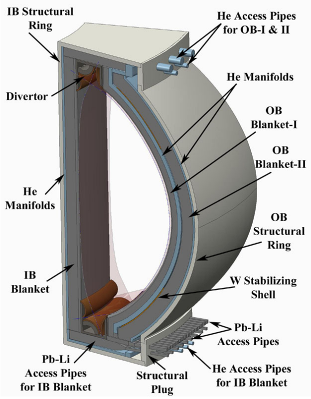
H/CD sector

- Tailored for specific penetration

TBM sector

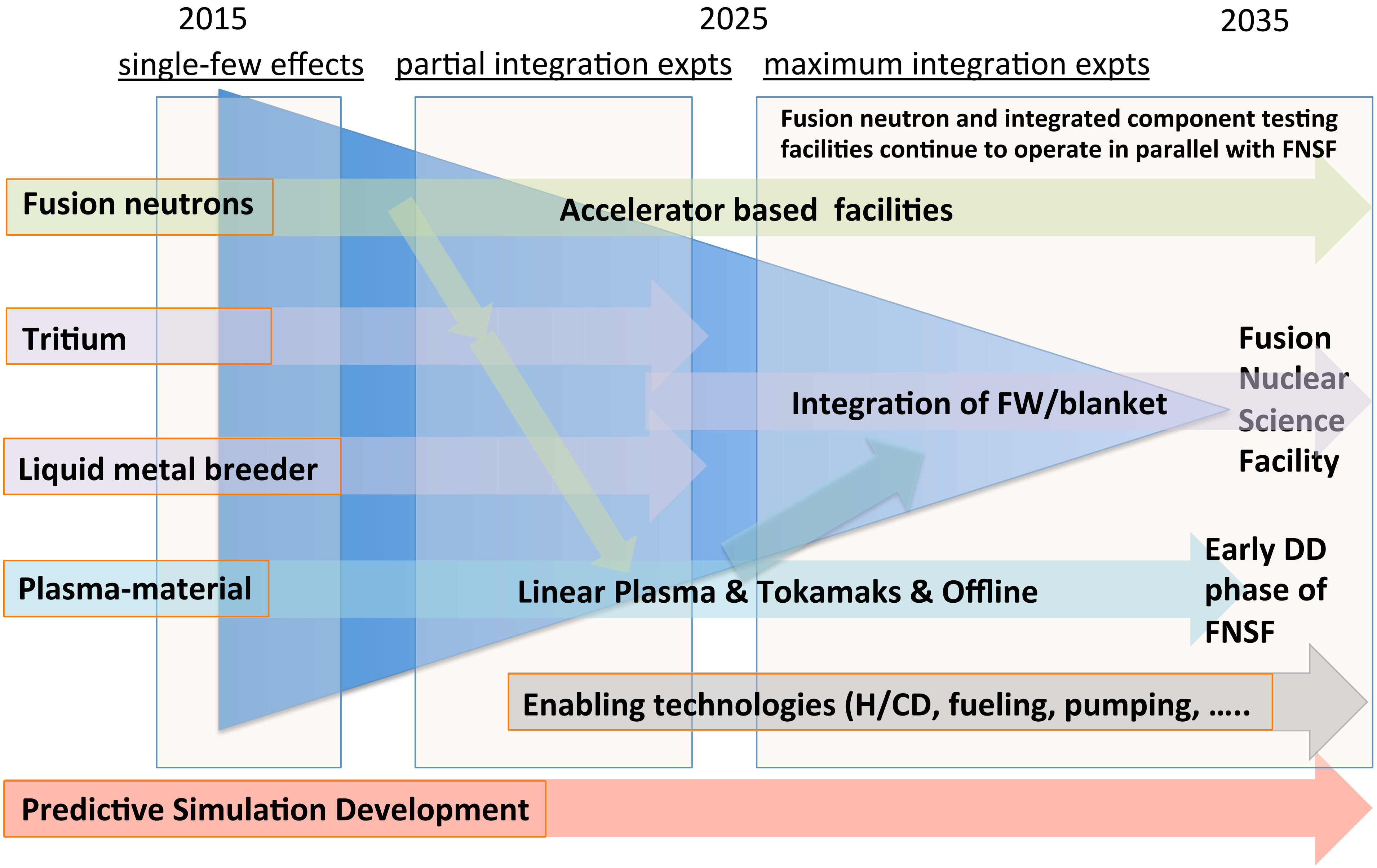
- Examine next phase blanket
- Can also be pulled for autopsy
- Use for backup blankets (HCCB, HCLL)

(MTM) material test modules that expose samples in the blanket region



	Phase 3-A	Phase 3-B	Phase 3-C
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**	**DCLL 400C RAFM - LH**	**DCLL 400C RAFM – LH**
S-4-TBM	**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-TBM	**DCLL 400C RAFM / ODS**	**DCLL 400C RAFM / ODS**	**DCLL 400C RAFM / ODS**
S-10	**DCLL 400C RAFM – IC**	**DCLL 400C RAFM – IC**	**DCLL 400C RAFM – IC**
S-11	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM – R2**
S-12	**DCLL 400C RAFM**	**DCLL 400C RAFM**	**DCLL 400C RAFM**
S-13	**DCLL 400C RAFM - NB**	**DCLL 400C RAFM - NB**	**DCLL 400C RAFM – NB**
S-14-TBM	**DCLL 400C RAFM / ODS**	**DCLL 400C RAFM / ODS**	**DCLL 400C RAFM / ODS**
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**
	1st year	2nd year	3rd year

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



2015

2025

2035

single-few effects

partial integration expts

maximum integration expts

Fusion neutron and integrated component testing facilities continue to operate in parallel with FNSF

Fusion neutrons

Accelerator based facilities

Tritium

Liquid metal breeder

Integration of FW/blanket

Fusion Nuclear Science Facility

Plasma-material

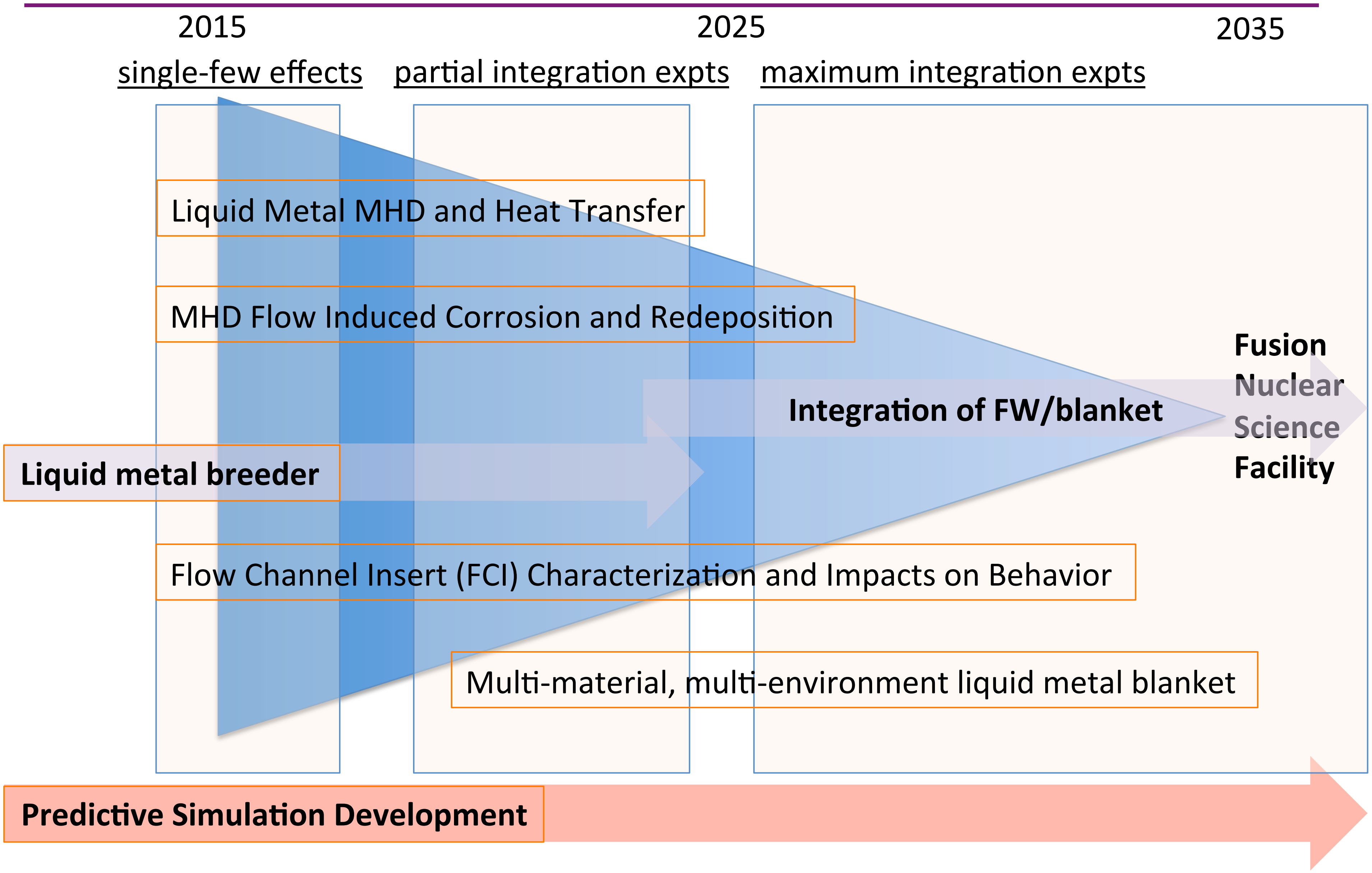
Linear Plasma & Tokamaks & Offline

Early DD phase of FNSF

Enabling technologies (H/CD, fueling, pumping, …..

Predictive Simulation Development

Zoom-In: *Liquid Metal Breeder* Science



2015 2025 2035

single-few effects

partial integration expts

maximum integration expts

Liquid Metal MHD and Heat Transfer

MHD Flow Induced Corrosion and Redeposition

Liquid metal breeder

Integration of FW/blanket

Fusion Nuclear Science Facility

Flow Channel Insert (FCI) Characterization and Impacts on Behavior

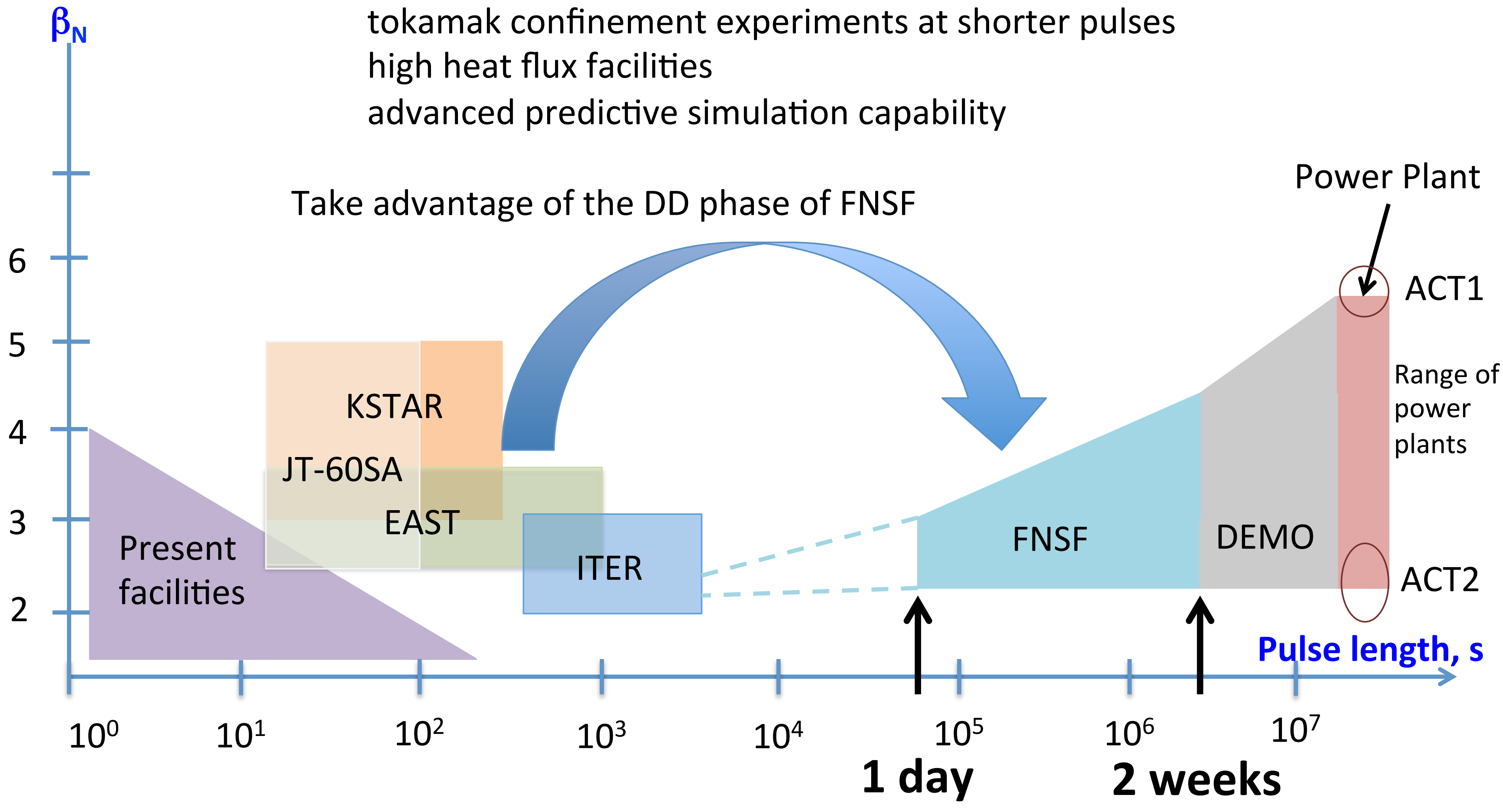
Multi-material, multi-environment liquid metal blanket

Predictive Simulation Development

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

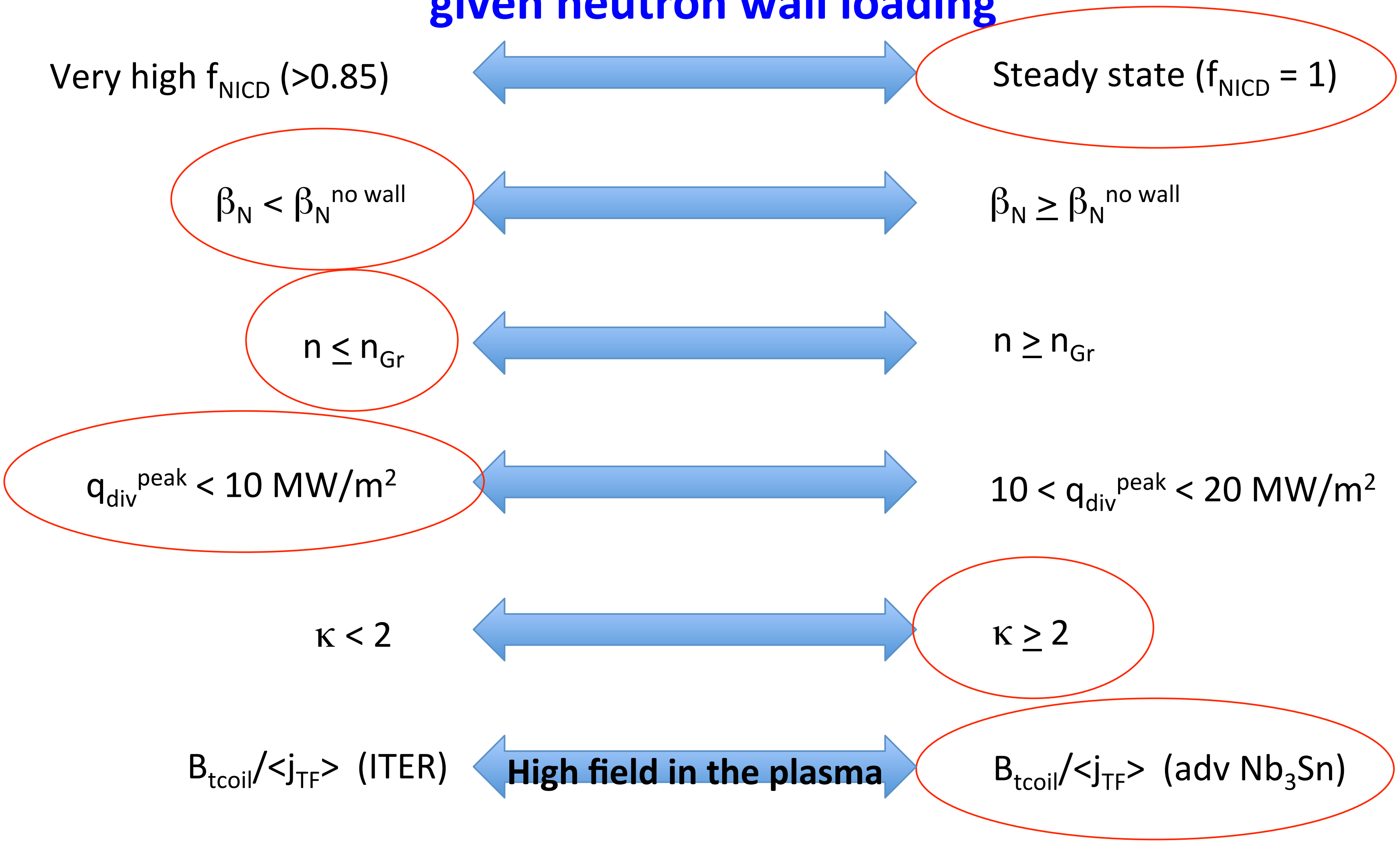
Before the FNSF, must combine
- ultra-long pulse linear plasma facilities
- tokamak confinement experiments at shorter pulses
- high heat flux facilities
- advanced predictive simulation capability

Take advantage of the DD phase of FNSF



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 ($f_{NICD} = 1$)
$\beta_N < \beta_N^{no\ wall}$	↔	$\beta_N \geq \beta_N^{no\ wall}$
$n \leq n_{Gr}$	↔	$n \geq n_{Gr}$
$q_{div}^{peak} < 10\ MW/m^2$	↔	$10 < q_{div}^{peak} < 20\ MW/m^2$
$\kappa < 2$	↔	$\kappa \geq 2$
$B_{tcoil}/<j_{TF}>$ (ITER)	**High field in the plasma**	$B_{tcoil}/<j_{TF}>$ (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, $\beta_N \sim 1.15 \times \beta_N^{\text{no wall}}$

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

Why Pursue a Smaller First Step, like the FNSF?

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

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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)

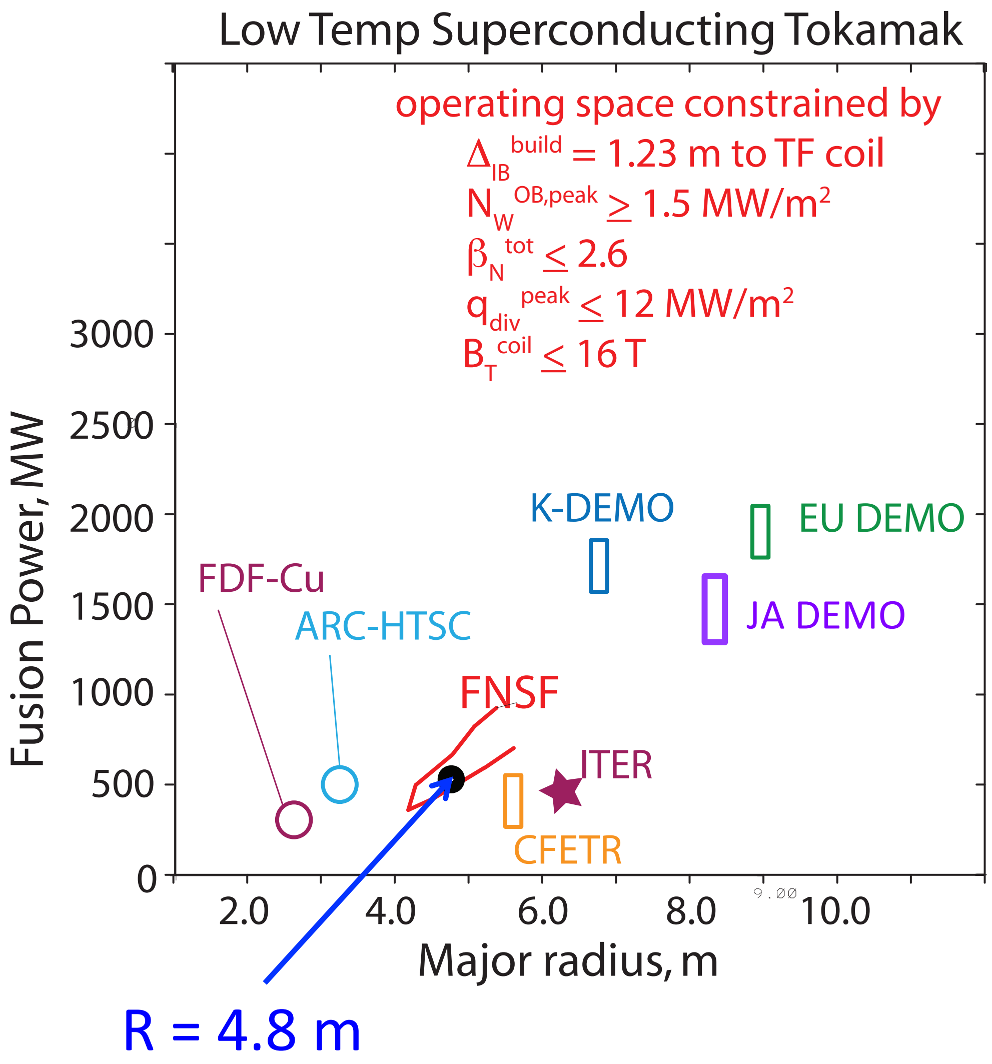
Goal is to establish the database on all components in the fusion neutron environment and in the overall environment before moving to larger size and routine electricity production

The 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 ($\beta_N < 2.6$)
- 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)
- 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 FNSF is a One of 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 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

Systems Code Identification

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

$\langle j_{TF} \rangle = 15\ MA/m^2$
$f_{div,rad} = 90\%\ (\lambda_{pow}^{Fundamenski})$

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
$\Delta_{SR} = 20$ cm
$\Delta_{VV} = 10$ cm
$\Delta_{LT\ shield} = 23$ cm
$\Delta_{gaps} = 20$ 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
$\langle j_{TF} \rangle$, 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
$n(0)/\langle n \rangle$, $T(0)/\langle T \rangle$	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)
$\langle N_w \rangle$, N_w^{peak}, MW/m^2	1.18, 1.77
q_{div}^{peak} (OB, IB), MW/m^2	10.7, 3.9

Why Pursue a Smaller First Step, like the FNSF?

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

Before

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Fission

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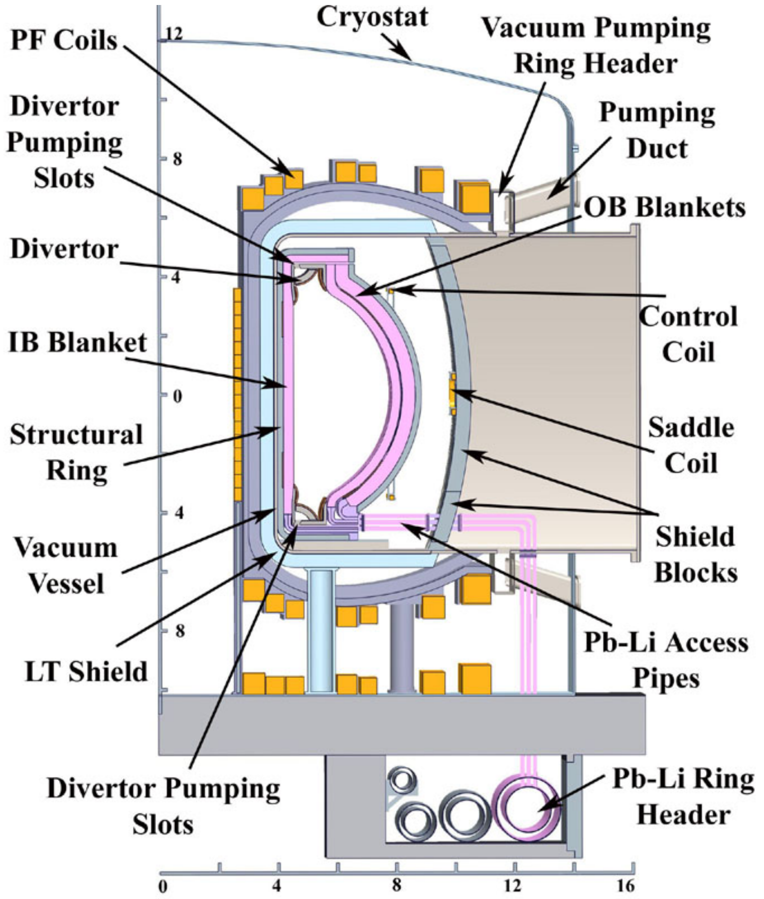
→ Seve

liquid r

Environmental variables in the fusion core

- B-field
- Temperature
- Pressure/stress
- Flow
- Hydrogen
- Dpa
- q'', q'''
- chemical
- gradients in r,θ



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

How Do We Measure Progress in These Missions - **Metrics**

Tritium Breeding				
	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
TBR - total		**~ 1.0**		**1.05**
Tritium produced/ year	**4 g (TBMs)**	**4.3-10.0 kg**		**101-146 kg**
Li-6 enrichment		**90%**		**40%**
OB FW hole/loss fraction		**10%**		**4%**
Tritium lost to decay, kg/year				**0.3**
Tritium lost to environment, kg/ year				**0.004**

How Do We Measure Progress in These Missions - **Metrics**

Long Plasma Durations at Required Performance				
	ITER	FNSF	DEMO	Power Plant ACT1/ACT2
Plasma on-time per year	**5%**	**15-35%**		**85%**
Plasma pulse duration, s	**500-3000**	**$0.09\text{-}1.2\text{x}10^6$**		**$2.7\text{x}10^7$**
Plasma duty cycle	**25%**	**33-95%**		**100%**
$\beta_N H_{98} / q_{95}$	**0.6**	**0.4**		**0.4-2.1**
Q	**5-10**	**4**		**25-48**
f_{BS}	**0.25-0.5**	**0.5**		**0.77-0.91**
$P_{core,rad}/(P_{alpha} + P_{aux})$	**0.27**	**0.26**		**0.28-0.46**
$P_{div,rad}/P_{SOL}$	**0.7**	**0.9**		**0.9**

Focus for 2015 is Detailed Analysis in Engineering and Physics of the FNSF – Access Critical R&D Issues

Engineering:

Neutronics, 1D this year to develop builds and heating, 3D next year for more accuracy, streaming and other issues (El-guebaly, UW)

Liquid metal MHD analysis by Smolentsev (UCLA) on IB and OB LiPb flow

Thermo-mechanics of blanket, FW and divertor by Y. Huang/N. Ghoniem (UCLA), J. Blanchard at UW, S. Malang (retired), M. Tillack (UCSD)

TF coil (and PF) coils stress analysis and winding pack design by Y. Zhai, P. Titus (PPPL)

Tritium inventory, extraction, implantation analysis (and accident) by P. Humrickhouse (INL)

Materials science development and assessments by FusMat group at ORNL (A. Rowcliffe, L. Garrison, and Y. Katoh)

CAD, establishing layouts for FNSF from systems code and design activities (E. Marriott)

Physics:

Core plasma equilibrium, ideal stability, time-dependent transport evolution, H/CD (Kessel, PPPL)

SOL/divertor analysis by Rognlien and Rensink (LLNL)

Other Critical Activities in 2015

Identification of accident scenarios and their categorization, rare to less-rare

- → Identify design features that ameliorate or minimize accident consequences

More detailed examination of maintenance and inspections

- → Hot cell requirements and turnaround (likely well beyond present capability)
- → In-vessel inspections/no vacuum break, minor maintenance/no vacuum break, maintenance/with vacuum break (or gas), etc.
- → Examination of the maintenance approach for TBMs and H/CD systems

Plasma and fusion core diagnostics

- → Different sets in He/H, DD and DT

Plasma physics strategy and DIII-D experiments/interactions, extensions to EAST/KSTAR, and also interaction with broader community

- → Plasma operating point candidates, projections with burning plasma

Challenging the program plan on the FNSF

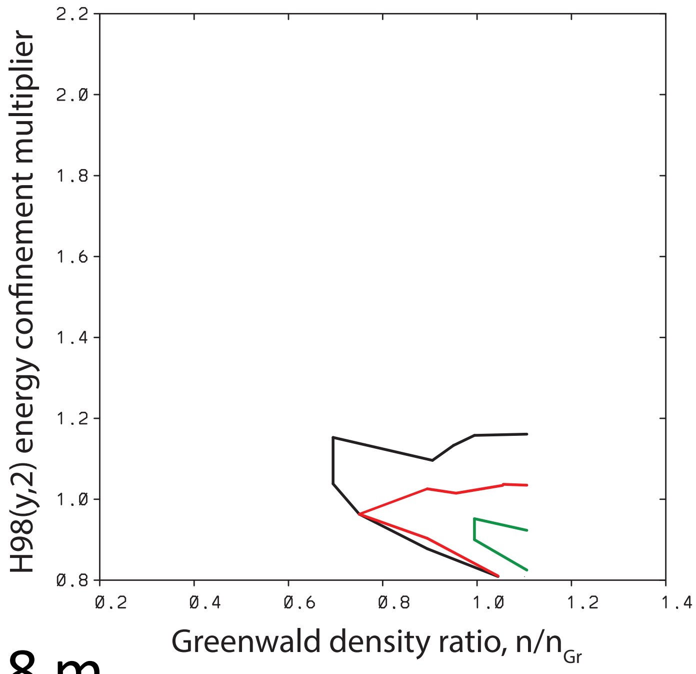
- → Accelerate the plasma pulse durations
- → Re-organize the distribution of plasma ops and maintenance
- → Identify more specifically hot plasma on, warm plasma off, and cold plasma off states

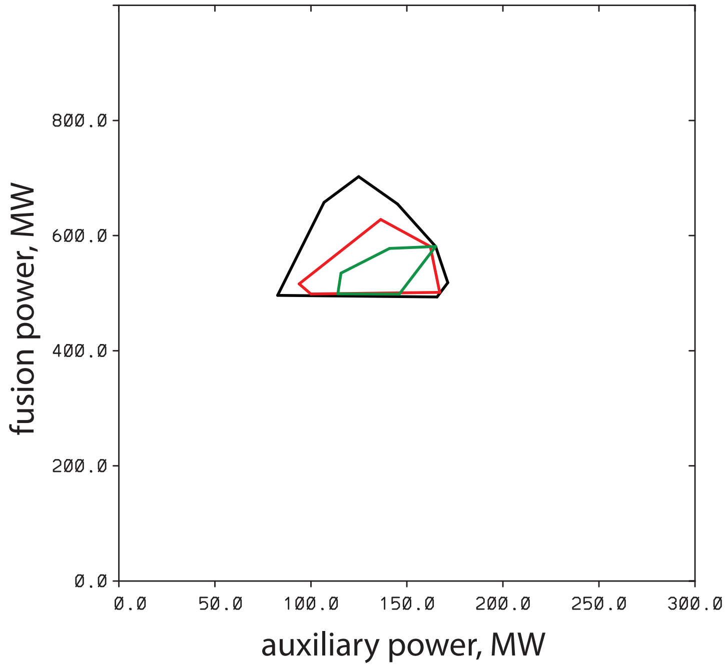
What if the maximum B-field at the TF coil does not reach 16 T?

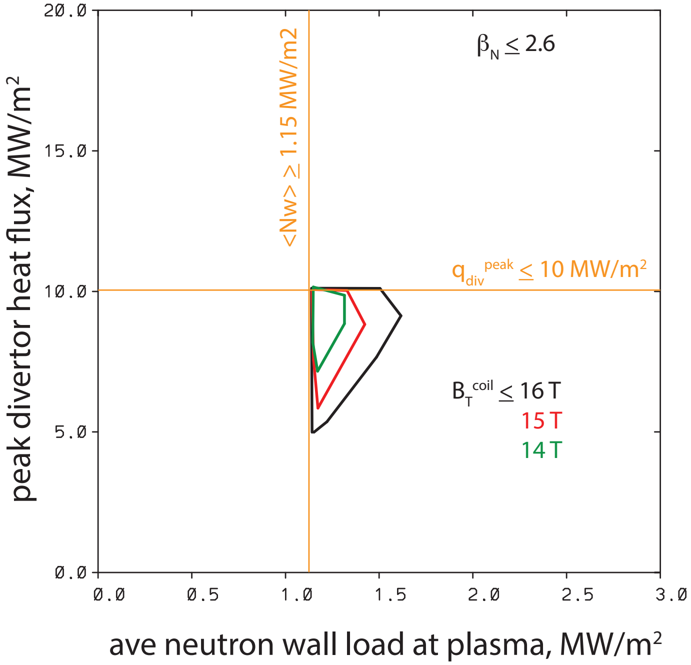
We are assuming that the Nb_3Sn technology will improve with R&D, and exceed ITER performance (*based on K-DEMO and HEP targets*)

Improved SC, conduit and winding pack optimization, and structural approaches

R = 4.8 m

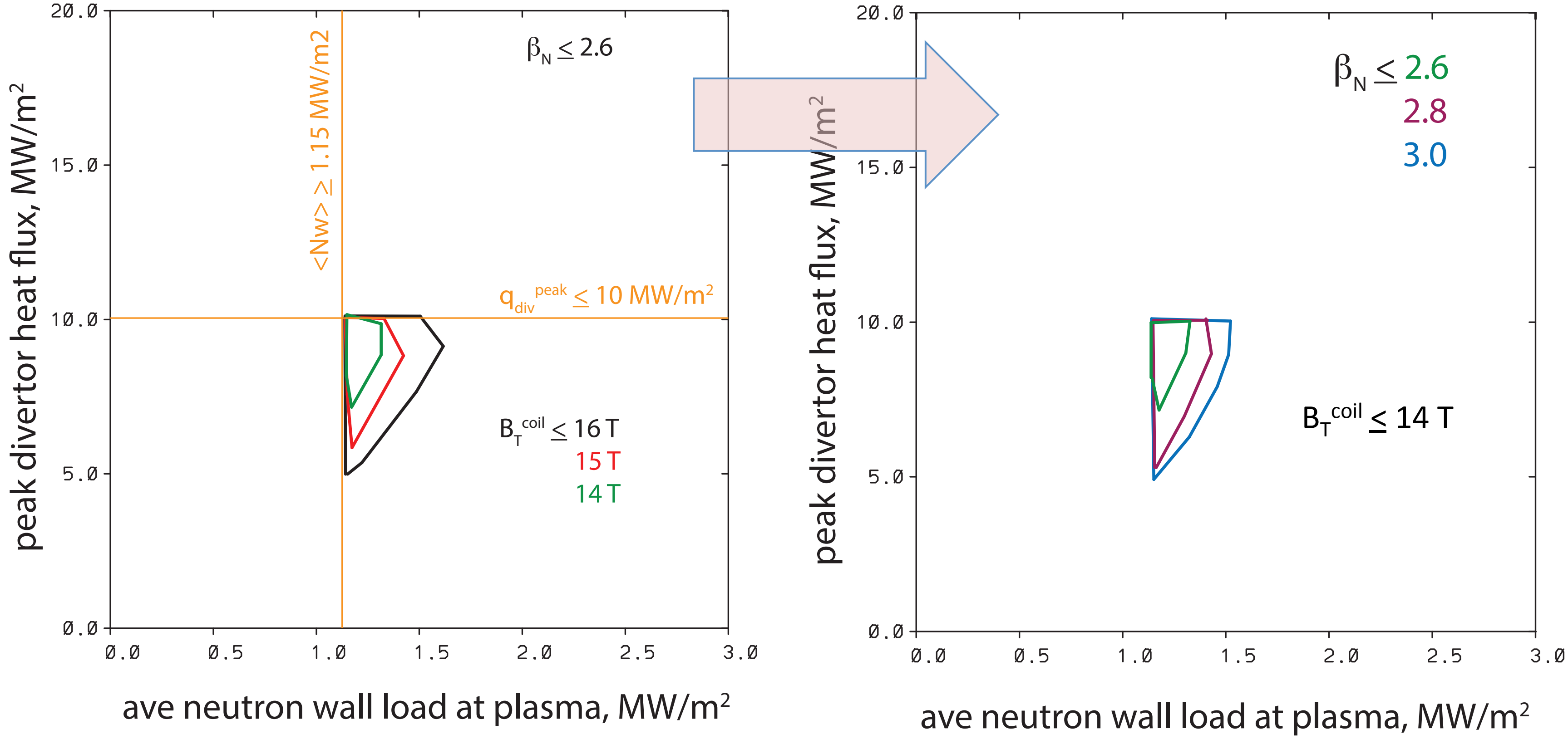






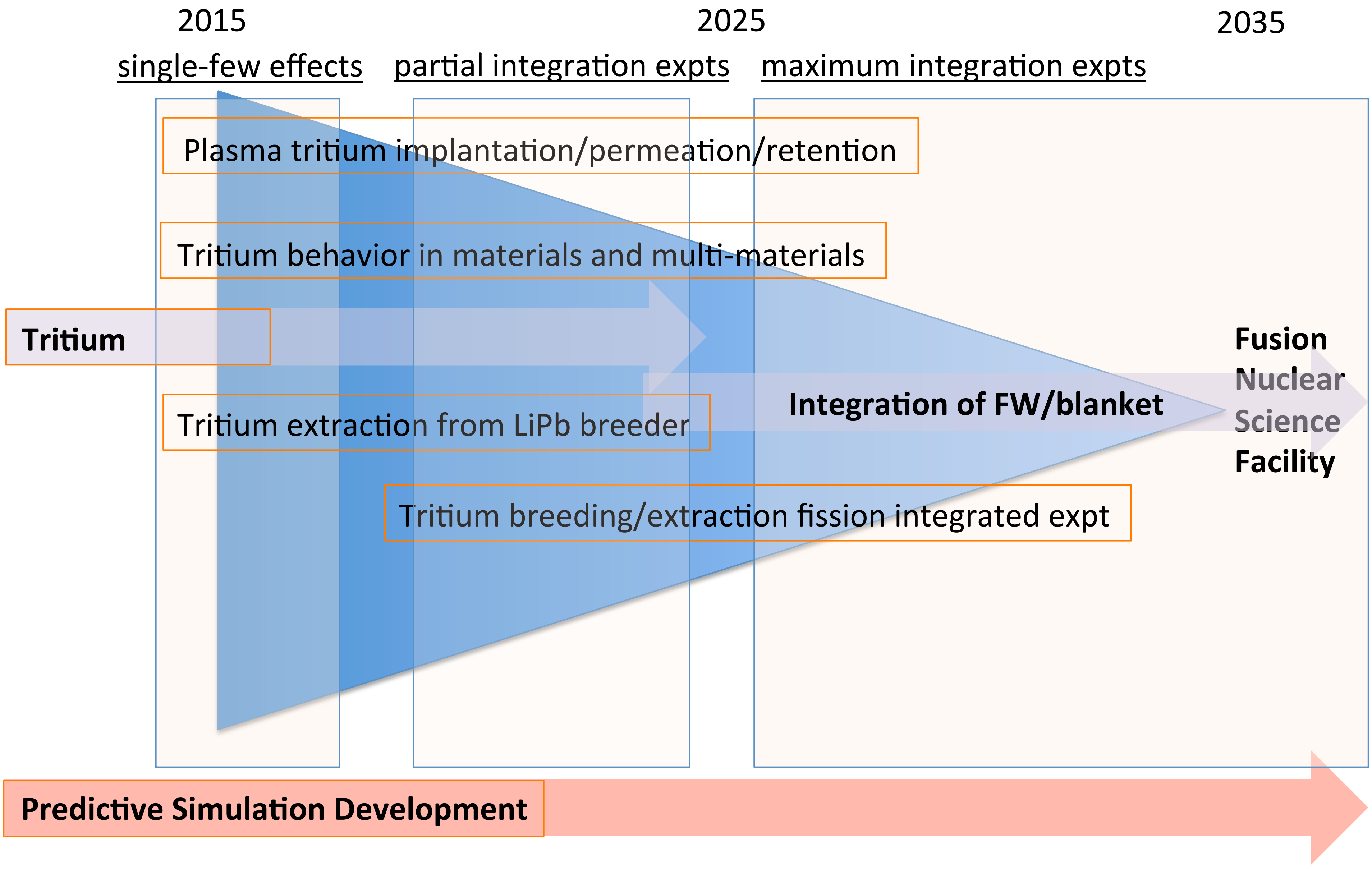
Raising the β_N can compensate the reduction in B_T, and still produce the N_w

The increase in β_N above the no-wall β limit requires 1) kinetic stabilization, 2) plasma rotation, and/or 3) feedback control



R = 4.8 m

Zoom–In: *Tritium Science* Breakdown



2015

2025

2035

single-few effects

partial integration expts

maximum integration expts

Tritium

Plasma tritium implantation/permeation/retention

Tritium behavior in materials and multi-materials

Tritium extraction from LiPb breeder

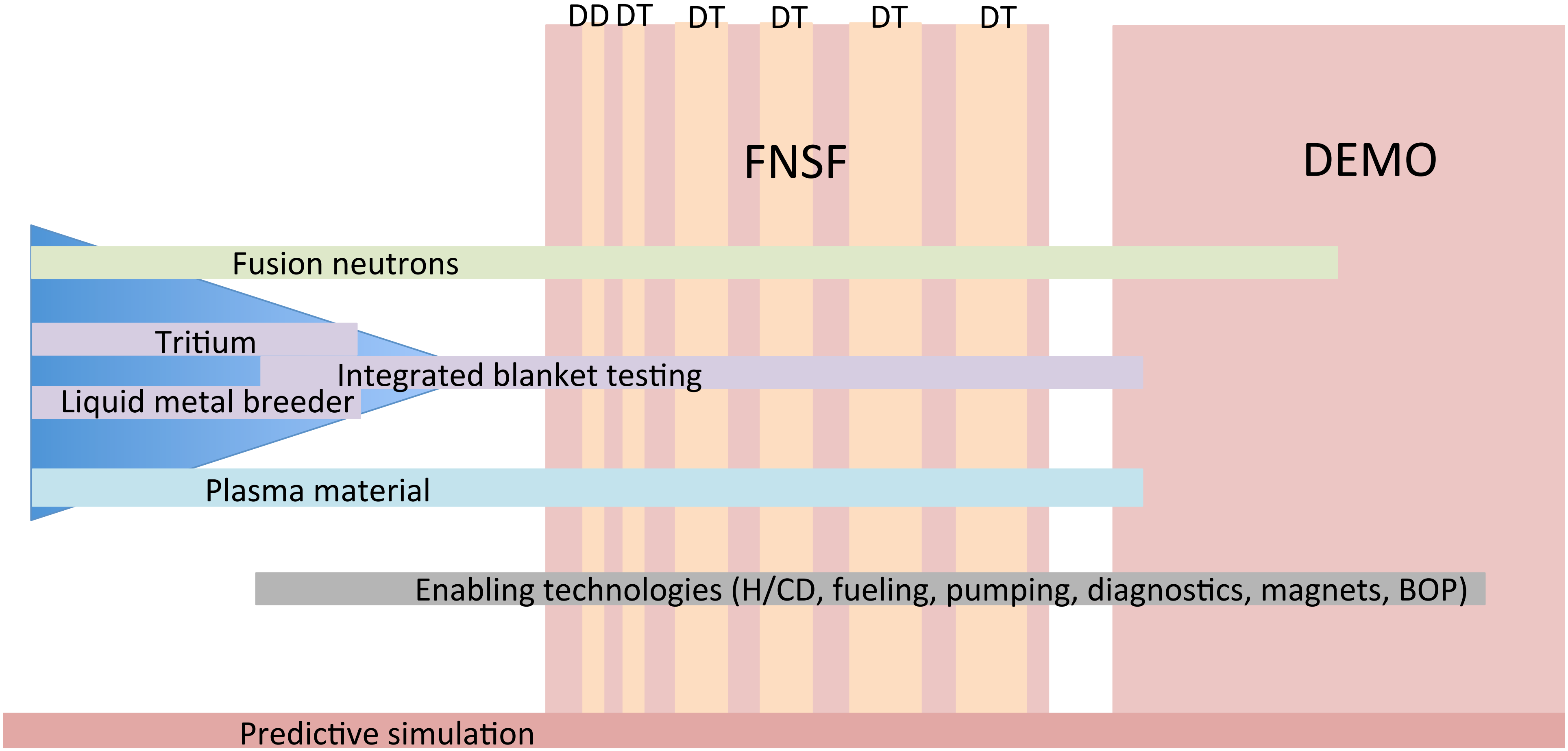
Tritium breeding/extraction fission integrated expt

Integration of FW/blanket

Fusion Nuclear Science Facility

Predictive Simulation Development

Zoom-Out: Examine the R&D Flow Over Pre-FNSF, FNSF, and into DEMO



White Papers and Design/Strategy/Philosophy Decisions

1) Use of water in the fusion core → no water inside the vacuum vessel
2) Helium cooling in the fusion core → He cooling is a viable approach with a technical basis and significant advantages
3) Single null versus Double null → Undecided, we are pursuing DN
4) Tritium breeding in the FNSF → this will be challenging, TBR ~ 1, may need to purchase (fission plant generation or int'l)
5) DCLL blanket concept → provides significant power plant advantages
6) Fusion core maintenance approach → qualitatively horizontal maintenance is baseline
7) Maximum dpa before replacement → reduced max dpa from 200 to 100 dpa, economic impacts can be significant for lower dpa's
8) TF/PF magnet options → pursue advanced LTSC, watch HTSC, Cu is not power plant relevant

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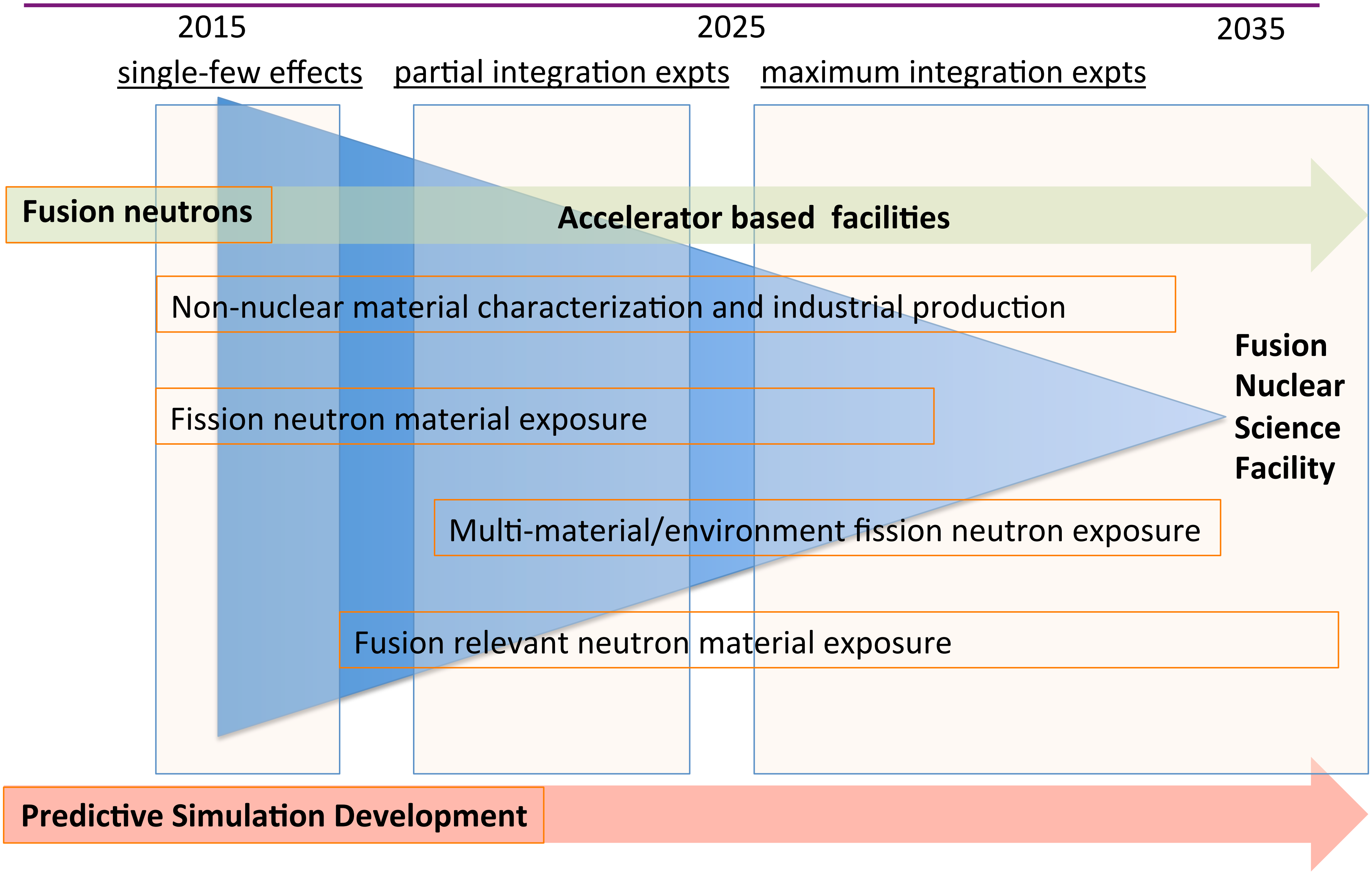
Zoom-In: *Fusion Neutron Science* (preliminary)



2015 2025 2035

single-few effects

partial integration expts

maximum integration expts

Fusion neutrons

Accelerator based facilities

Non-nuclear material characterization and industrial production

Fission neutron material exposure

Multi-material/environment fission neutron exposure

Fusion relevant neutron material exposure

Fusion Nuclear Science Facility

Predictive Simulation Development

Materials Assumptions for the FNSF, DCLL Blanket

Basing fusion structural components (blanket, structural ring, shield filler, manifolding) on the "family" of RAFM steel

  Generation I (Eurofer, F82H, CLAM, etc.) up to 20 dpa
  Generation II (ODS) up to 50 dpa
  Nano-structured (ODS, NS) up to 60+ dpa?
  → activation, waste, accidents

LiPb breeder material for high breeding potential, lower reactivity with oxygen, controllability of tritium breeding in situ

  → MHD impacts on heat transfer, corrosion, pressure drop

SiC-c flow channel insert to provide high electrical/thermal resistance barrier for LiPb

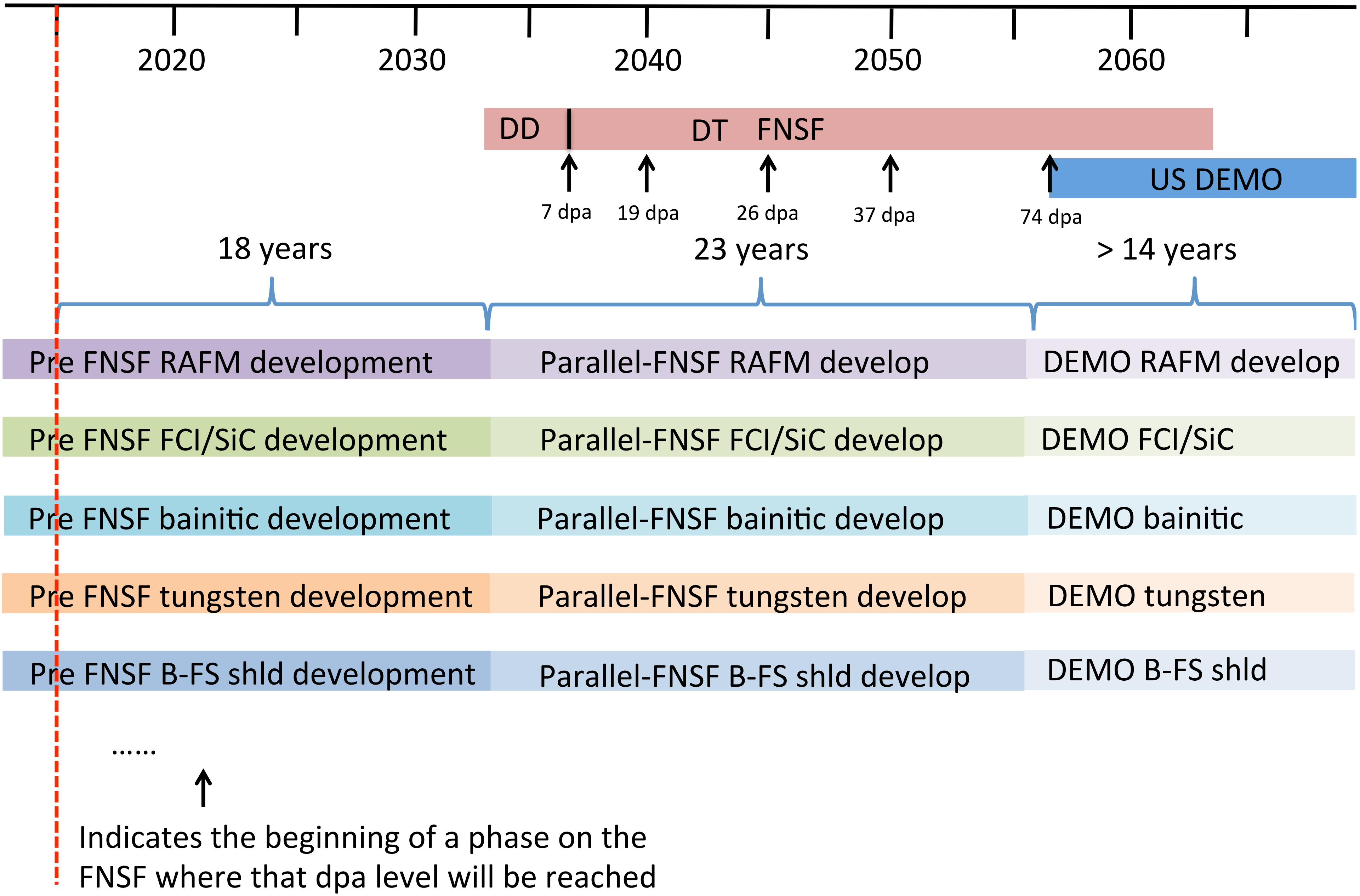
  → Fission testing is successful, ASME code section being written for fission
  → Fusion behavior is still uncertain
  → Interaction with flowing LiPb, high T and neutrons

Using bainitic steel for vacuum vessel due to weaker neutron environment and no need for PWHT (stable micro-structure)

Tungsten (alloy, composite, ODS, ??) in the divertor, and WC as shielding filler in some components

  → Significant uncertainty in tungsten materials as PFC armor and structure, decay heat

Fusion Materials Science assumed timeline



2020 2030 2040 2050 2060

DD | DT FNSF

US DEMO

7 dpa 19 dpa 26 dpa 37 dpa 74 dpa

18 years

23 years

> 14 years

Pre FNSF RAFM development	Parallel-FNSF RAFM develop	DEMO RAFM develop
Pre FNSF FCI/SiC development	Parallel-FNSF FCI/SiC develop	DEMO FCI/SiC
Pre FNSF bainitic development	Parallel-FNSF bainitic develop	DEMO bainitic
Pre FNSF tungsten development	Parallel-FNSF tungsten develop	DEMO tungsten
Pre FNSF B-FS shld development	Parallel-FNSF B-FS shld develop	DEMO B-FS shld

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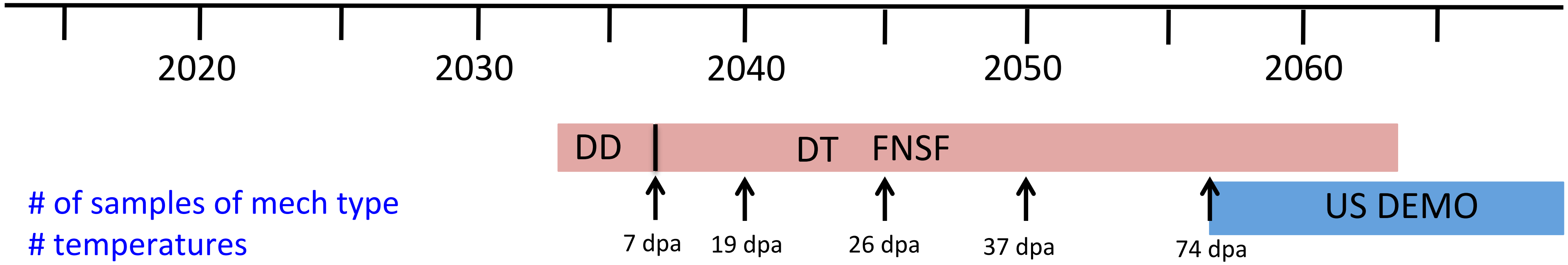
↑

Indicates the beginning of a phase on the FNSF where that dpa level will be reached

Characterizing what neutron irradiation facilities can deliver, how fast, and what volume of samples……


2020
2030
2040
2050
2060

DD

DT FNSF

US DEMO

7 dpa
19 dpa
26 dpa
37 dpa
74 dpa

# of samples of mech type
temperatures
materials

→ **What type of database is required? Scientific or engineering?**

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 [84,86]	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				
US	SNS spallation source FMITS, 5 cm [90]	5	20	100			0.02/0.04	0.04 liter
	SNS spallation source FMITS, 3 cm [90]	10	75	310			0.02/0.04	
	SINQ spallation source, center rod 1 [91,92]	≤10	≤70	≤470			~0.006/3	
US	MTS spallation, fuel positions, 15 cm [44]	17.5	29	–			~0.001/0.04	0.04 liter
	MTS spallation, fuel positions, 5 cm [44,85]	32	16	–	1	0.1	~0.001/0.04	

Zinkle, Moeslang, FED2013, 472