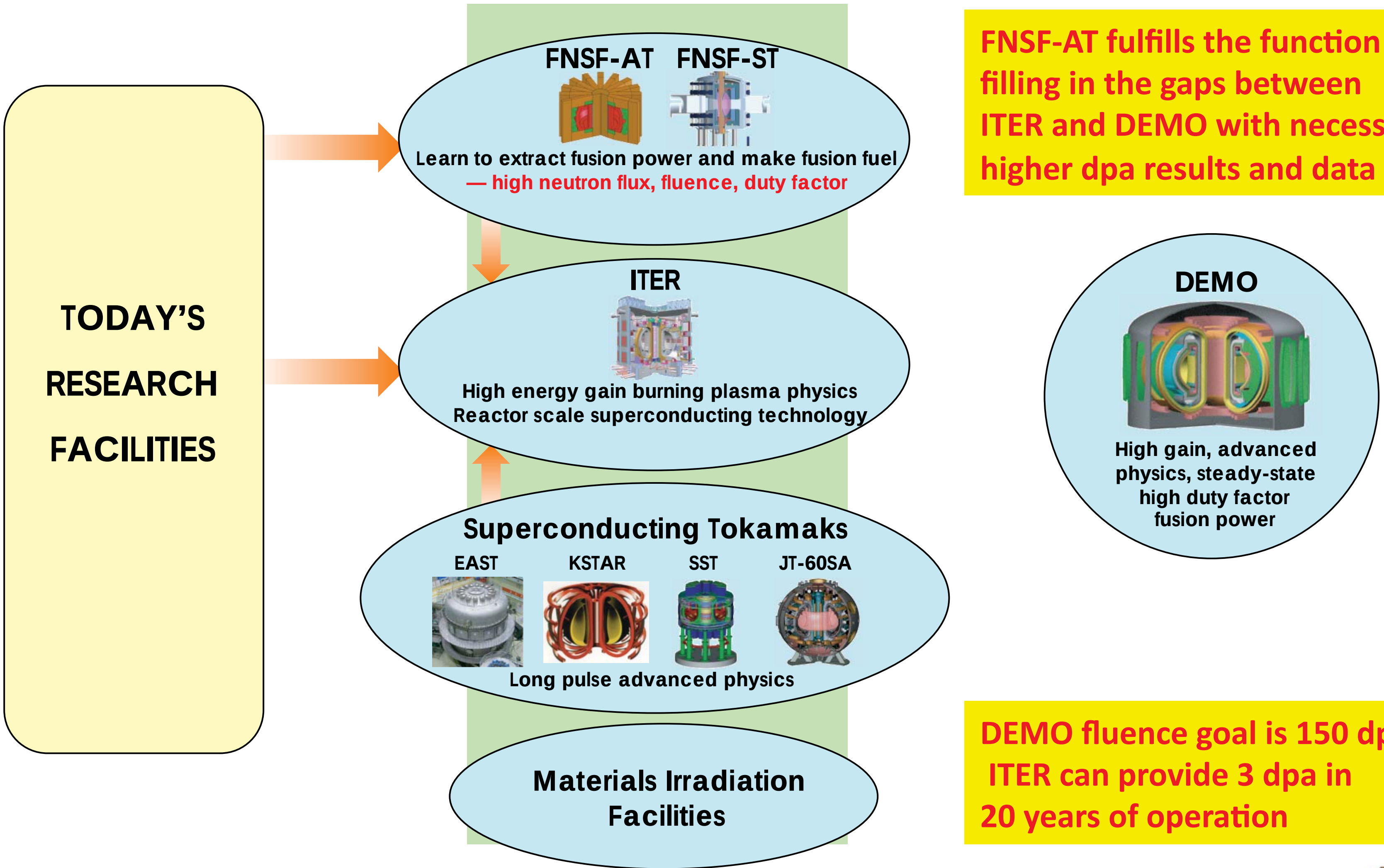
Neutronics Analysis in Support of the Fusion Development Facility Design Evolution

M. Sawan, A. Ibrahim, P. Wilson, E. Marriott
University of Wisconsin-Madison

R. Stambaugh, C. Wong
General Atomics

19th Topical Meeting on the Technology of Fusion Energy (TOFE)
November 7-11, 2010, Las Vegas, Nevada

A Fusion Nuclear Science Facility Provides the U.S. With a World Leading Research Facility Running in Parallel With ITER



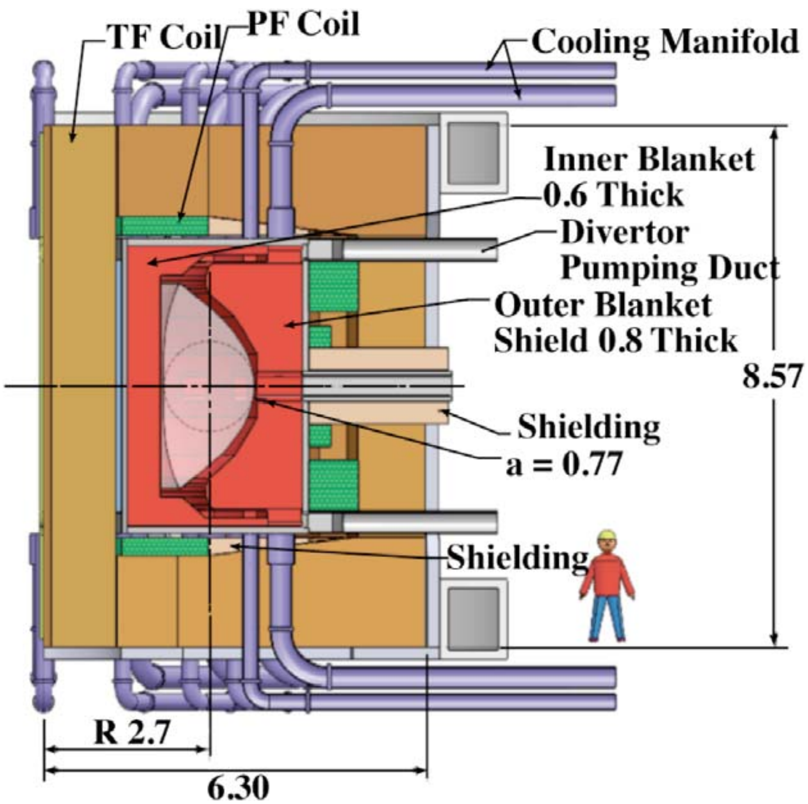
FNSF-AT fulfills the function of filling in the gaps between ITER and DEMO with necessary higher dpa results and data

DEMO fluence goal is 150 dpa, ITER can provide 3 dpa in 20 years of operation

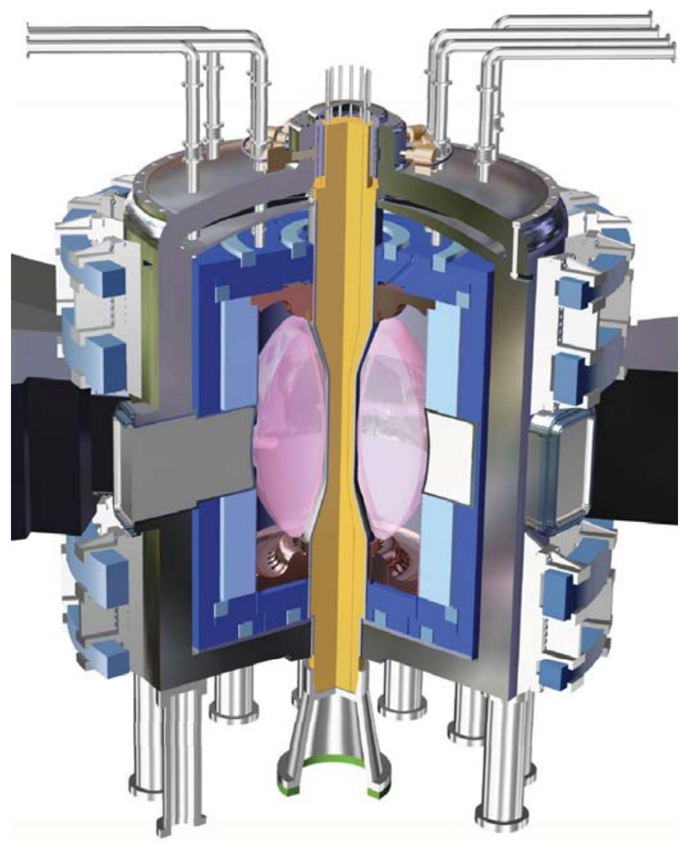
FNSF-AT Mission: Carry Forward Advanced Tokamak Physics and Enable Development of Fusion's Energy Applications

- **Demonstrate advanced physics operation of a tokamak in steady-state with Burn**
  - **Utilize conservative expressions of all elements of Advanced Tokamak physics to produce 100-250 MW fusion power with modest energy gain (Q<5) in a modest sized device**
  - **Utilize full non-inductive, high bootstrap operation to achieve continuous operation for > 2 weeks**
  - **Further develop all elements of Advanced Tokamak physics, qualifying them for an advanced performance DEMO**
- **Develop fusion nuclear technology**
  - **Test materials and components to high neutron fluence (3-8 MW-yr/m^2) with duty factor of 0.3 per year with maintainability**
  - **Demonstrate tritium self-sufficiency**
  - **Develop fusion blankets that produce both tritium and electricity at 1-3 MW/m^2 neutron fluxes**
  - **Develop advanced fusion blankets with high temperature process heat**
- ***With ITER and Material Irradiation Facility, provide the basis for a fusion DEMO Power Plant***

Two FNSF Candidates



FDF
AT Physics
(FNSF-AT)



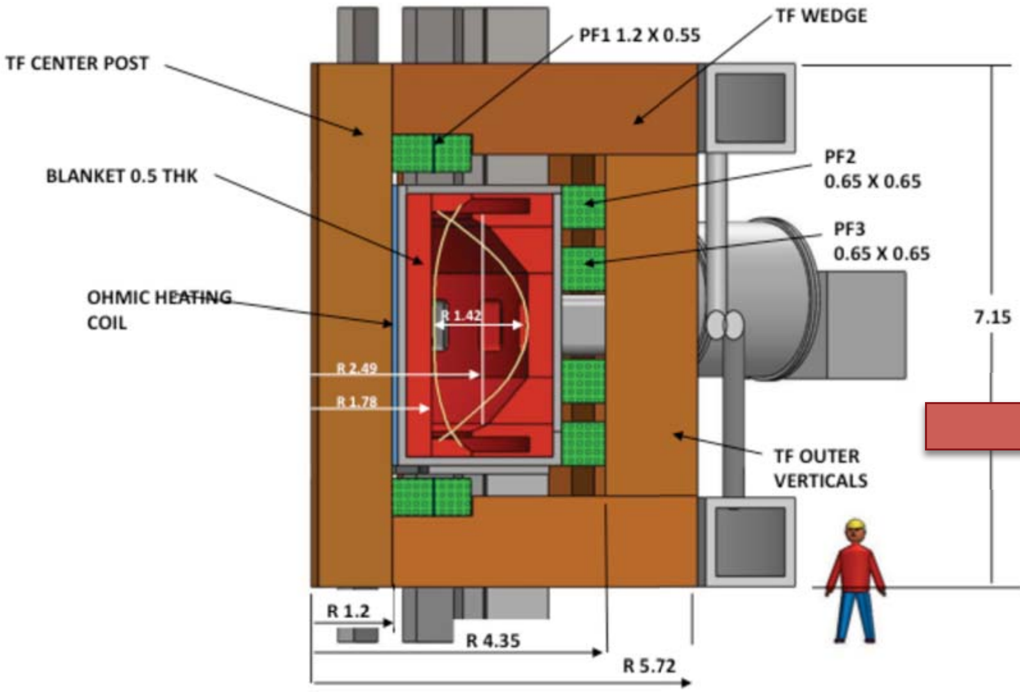
ST-CTF
ST Physics
(FNSF-ST)

THE UNIVERSITY of WISCONSIN MADISON

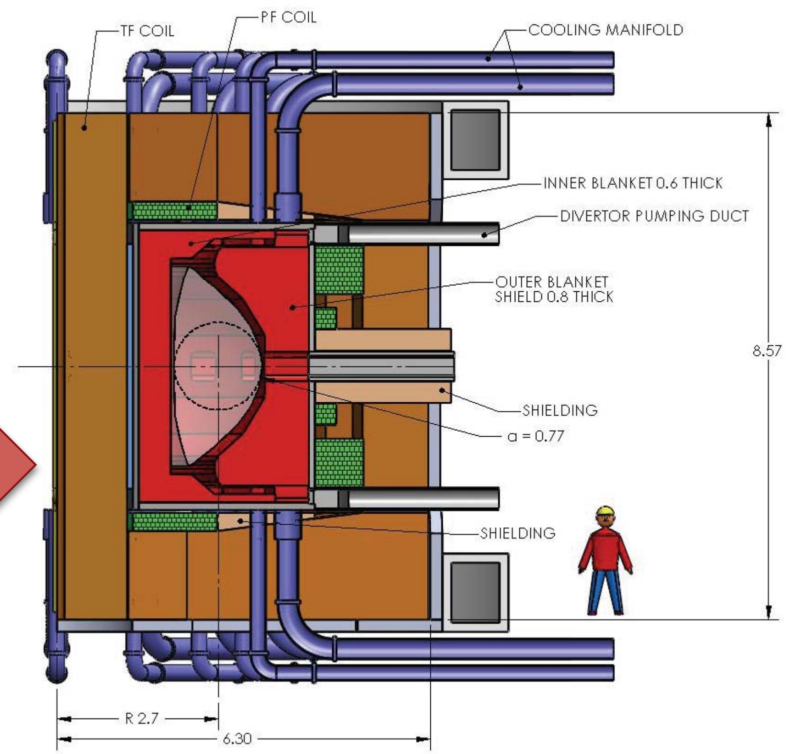
Scoping Analysis for Initial FDF Baseline Design

- Several options were considered for material choice in IB zone
- Preferred option divides 50 cm IB blanket/shield space between blanket and water-cooled FS shield with WC filler
- Two blanket concepts were considered; DCLL and HCCB
- With preferred option of organic insulators, we determined that IB blanket/shield thickness should be increased to 60 cm and the OB blanket/shield thickness should be 80 cm
- Re-baselined FDF design that incorporates increased blanket/shield thickness, with realistic divertor geometry and plasma/wall gaps, was developed with R=2.7 m, a= 0.77 m

Evolution to New Baseline



Initial Baseline

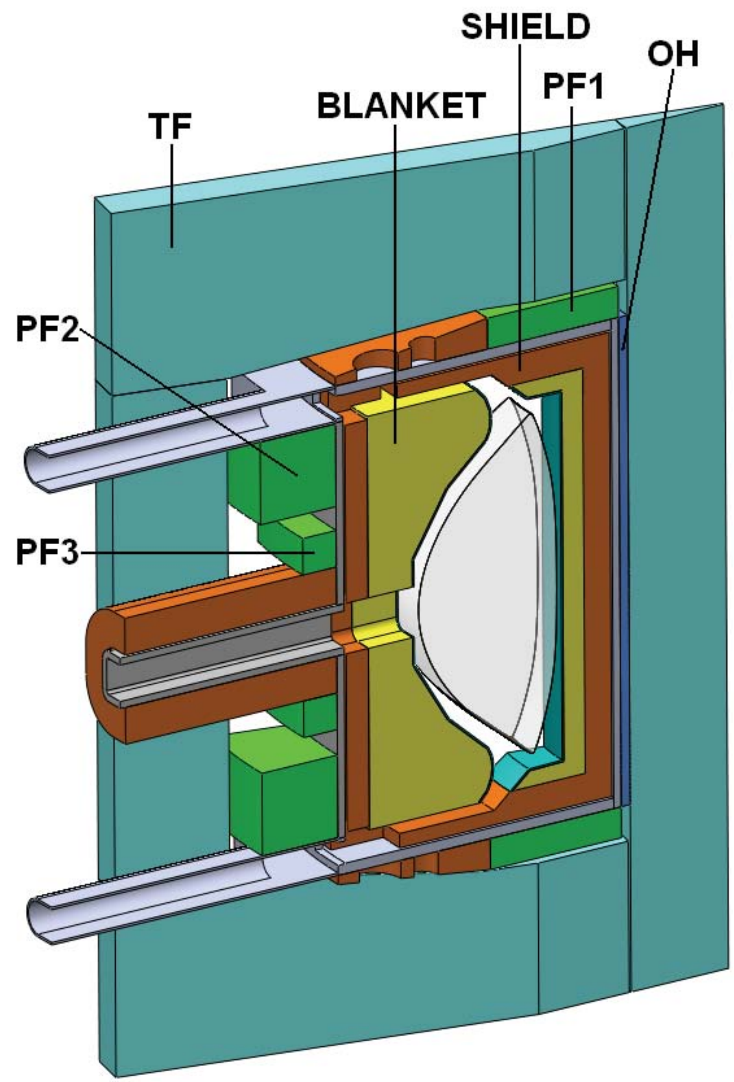


Present Baseline

THE UNIVERSITY of WISCONSIN MADISON

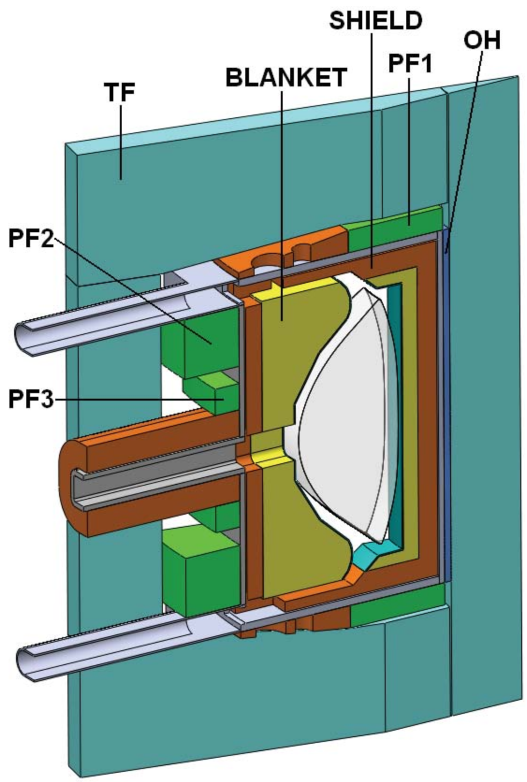
3-D Neutronics for New FDF Baseline

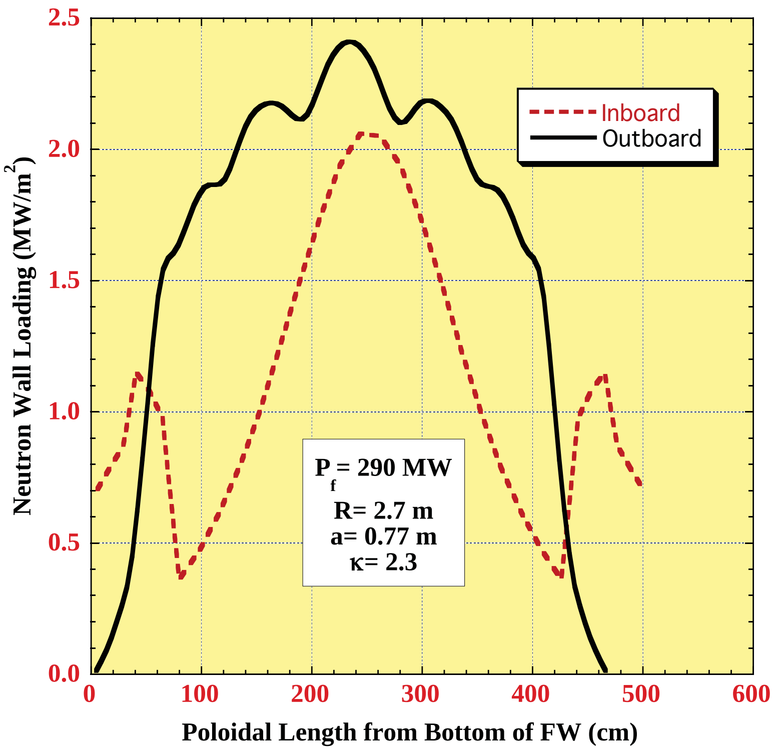
- Utilized the FDF CAD model to perform 3-D neutronics using the DAG-MCNP code
- 1/32 toroidal sector used with reflecting boundaries
- Source sampled from a peaked distribution in the plasma zone
- Calculated:
  - Neutron wall loading distribution
  - TBR for both DCLL and HCCB blankets
  - Nuclear parameters in VV, OH, TF, and PF coils



THE UNIVERSITY of WISCONSIN MADISON

Neutron Wall Loading Distribution

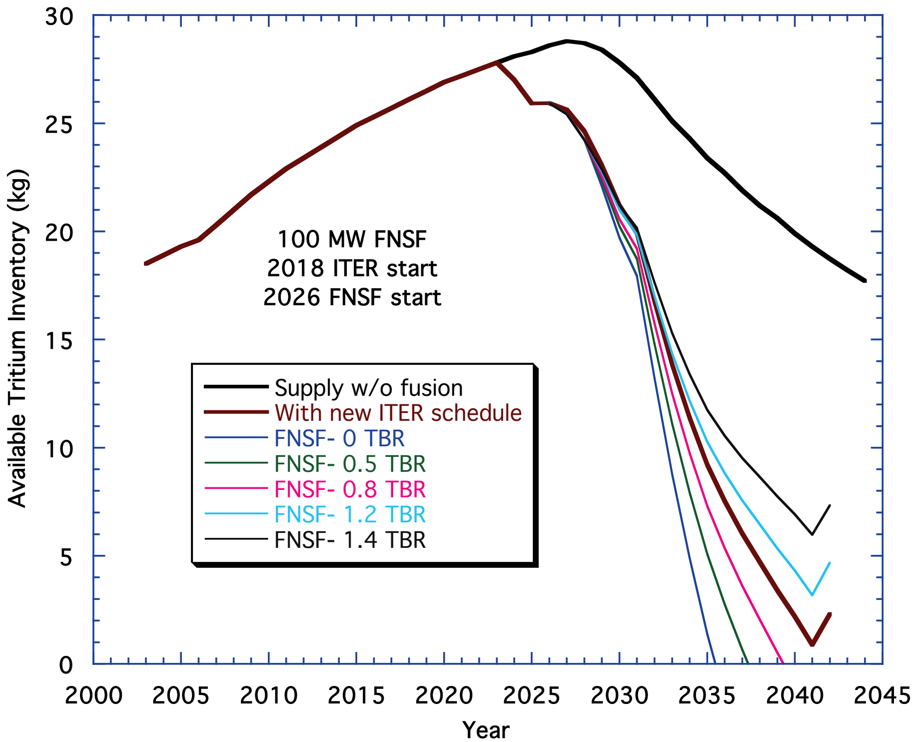


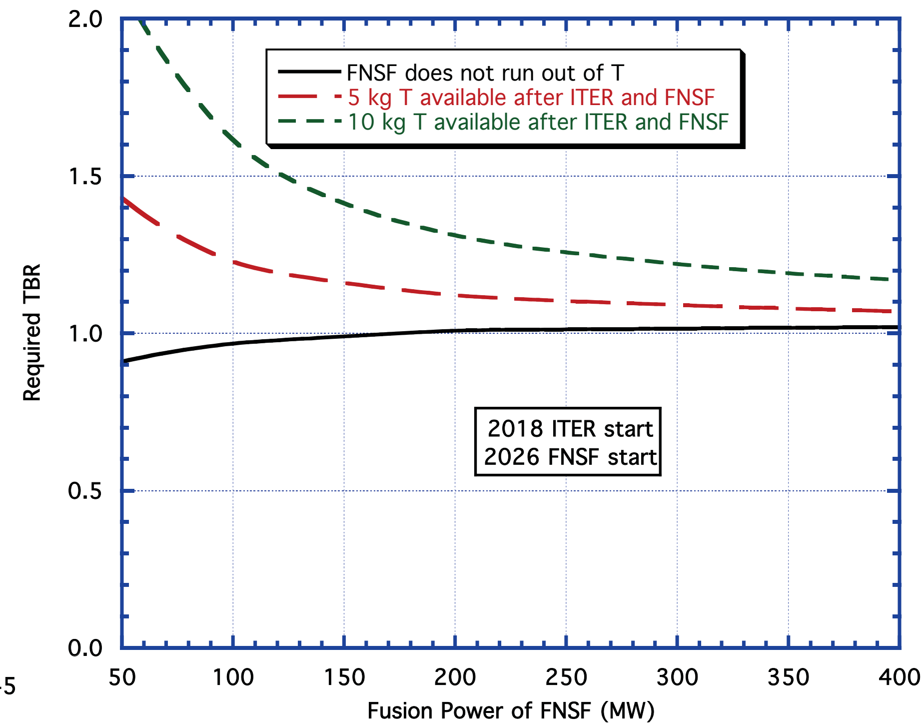


	FW Area (m^2)	Peak Neutron Wall Loading (MW/m^2)	Average Neutron Wall Loading (MW/m^2)
Inboard	64.0	2.06	1.09
Outboard	92.4	2.41	1.69
IB+OB	156.4	2.41	1.44

With FDF goal of 2 MW/m² peak OB NWL fusion power can be reduced to 240 MW

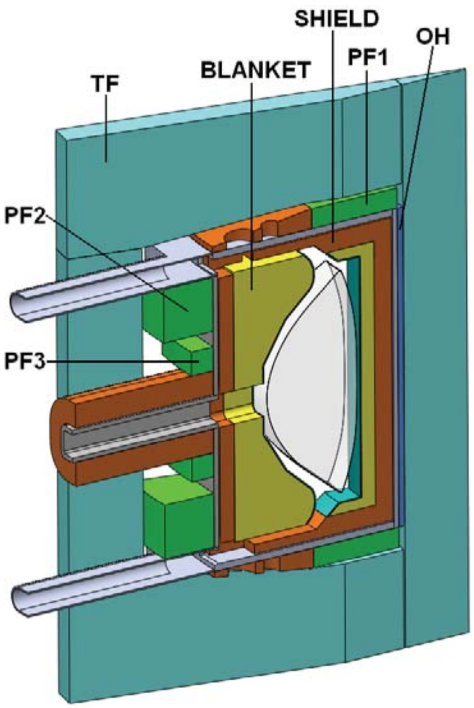
Required TBR in FNSF





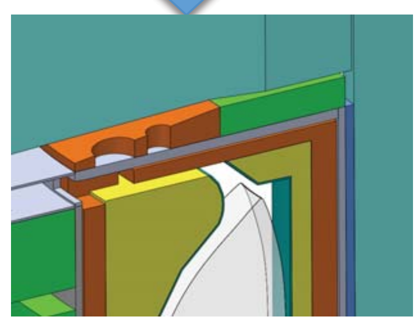
- *Almost all tritium supply will be used by ITER and FNSF has to be self-sufficient in tritium in addition to providing initial startup inventory for DEMO*

Tritium Breeding



	DCLL Blanket	HCCB Blanket
Inboard	0.21	0.26
Outboard	0.69	0.75
IB+OB	0.90	1.01

- *TBR is smaller by ~15% than the full coverage case (1-D) due to lost coverage by divertor and 16 mid-plane ports*
- *TBR enhancement possible with more blanket utilization in divertor regions*

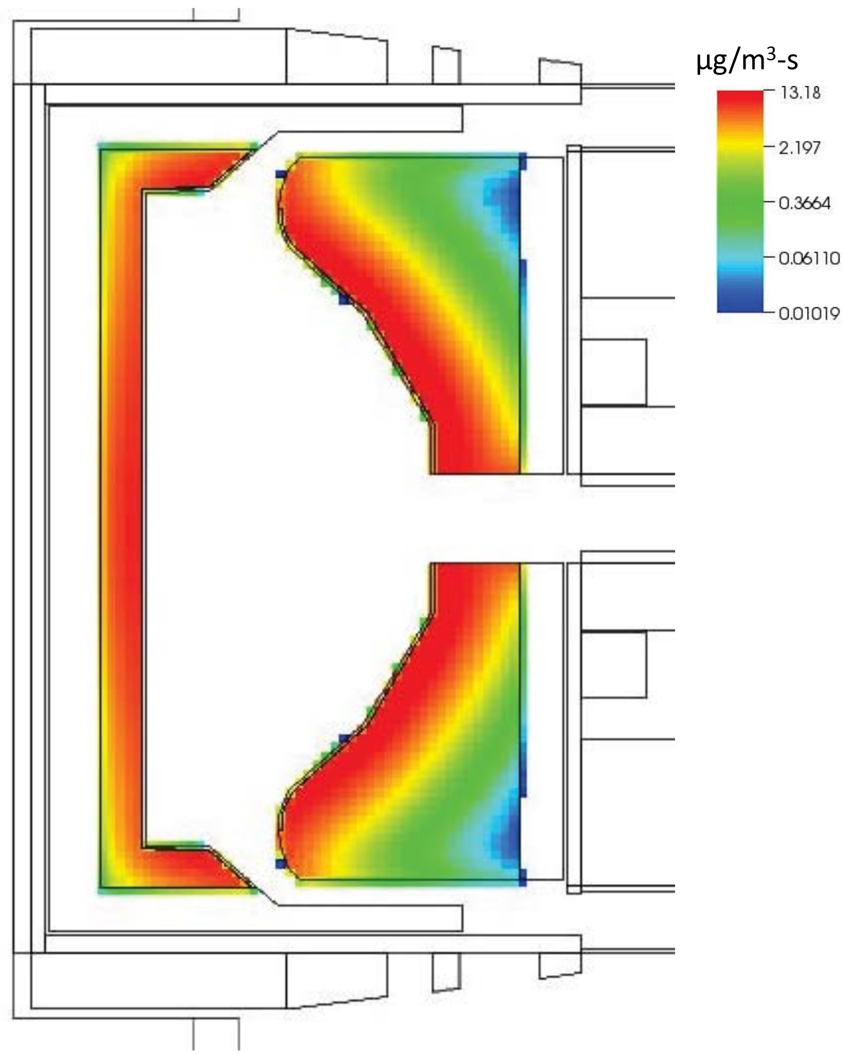


	DCLL Blanket	HCCB Blanket
Inboard	0.26	0.31
Outboard	0.74	0.78
IB+OB	1.00	1.09

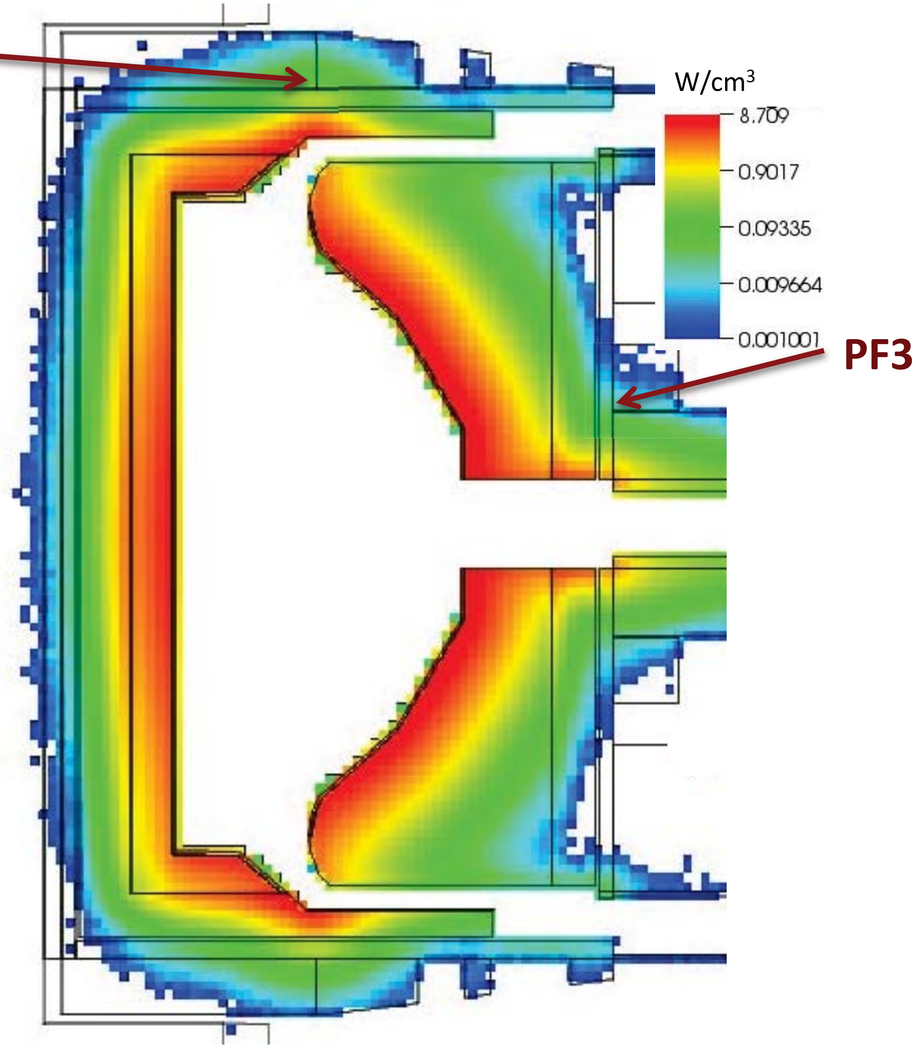
- Lost TBR due to 16 ports is 6%
- Some of this can be recovered by breeding in test blanket modules

Tritium Production in HCCB Blanket

Tritium production drops rapidly as one moves deeper in blanket with negligible contribution from back of blanket above/ below OB mid-plane



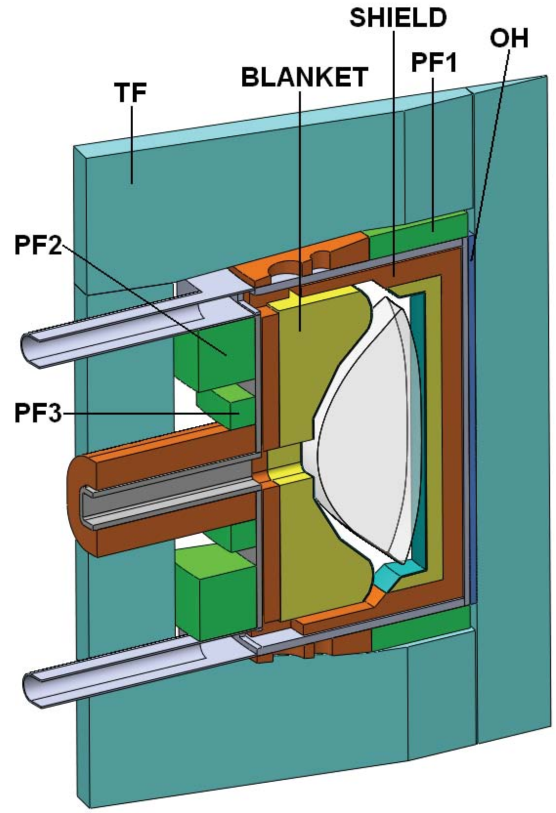
Nuclear Heating with HCCB Blanket



Effect of streaming in mid-plane ports and divertor clearly demonstrated with hot spots in PF1 and PF3

Damage Parameters

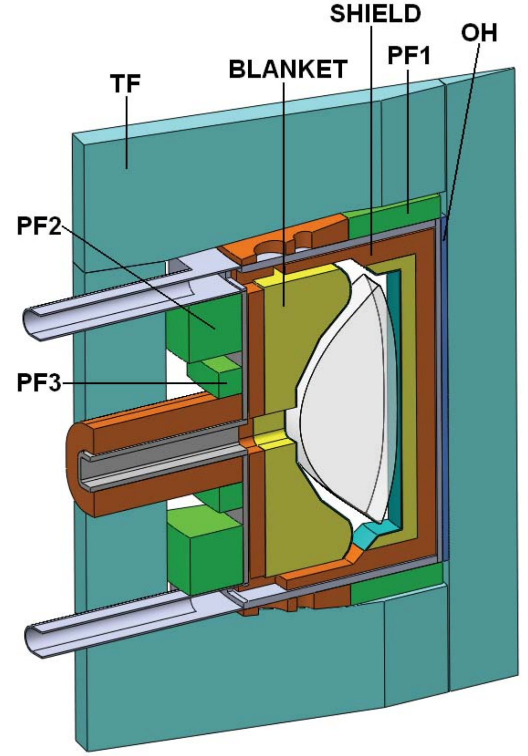
	DCLL	HCCB
Peak He appm in VV	0.084	0.096
Peak power density in OH coil (mW/cm^3)	27.7	27.9
Peak fast neutron fluence in OH coil (n/cm^2)	7.2×10^{18}	8.3×10^{18}
Peak organic insulator dose in OH coil (Rads)	1.5×10^{10}	1.6×10^{10}
Peak Cu dpa in OH coil (dpa)	5.2×10^{-3}	6.3×10^{-3}
Peak Cu resistivity increase in OH (nΩm)	0.49 (3.1%)	0.56 (3.5%)
Peak power density in TF coil (mW/cm^3)	7.1	7.2
Peak fast neutron fluence in TF coil (n/cm^2)	6.2×10^{18}	7.2×10^{18}
Peak organic insulator dose in TF coil (Rads)	1.1×10^{10}	1.2×10^{10}
Peak Cu dpa in TF coil (dpa)	4.3×10^{-3}	5.2×10^{-3}
Peak Cu resistivity increase in TF (nΩm)	0.42 (2.6%)	0.49 (3.1%)



- *Detailed 3-D calculations for the new FDF baseline confirm that VV is reweldable with modest peak nuclear heating and Cu resistivity increase in magnets*
- *Peak insulator dose levels slightly exceed 10^{10} Rads. Advanced cyanate ester resins showed no degradation in properties following irradiation to several 10^{10} Rads*
- *Shielding of insulator layer by coil case was not accounted for*
- *Damage parameters for VV, TF, and OH are comparable for both blanket options*

Peak Damage Parameters for PF Coil

	DCLL	HCCB
Peak power density in PF1 coil (mW/cm³)	214	193
Peak fast neutron fluence in PF1 coil (n/cm²)	2.2×10^{20}	2.1×10^{20}
Peak organic insulator dose in PF1 coil (Rads)	3.8×10^{11}	3.6×10^{11}
Peak Cu dpa in PF1 coil (dpa)	1.7×10^{-1}	1.7×10^{-1}
Peak Cu resistivity increase in PF1 (nΩm)	1.25 (7.8%)	1.25 (7.8%)
Peak power density in PF3 coil (mW/cm³)	19.3	13.7
Peak fast neutron fluence in PF3 coil (n/cm²)	1.3×10^{19}	1.2×10^{19}
Peak organic insulator dose in PF3 coil (Rads)	3.0×10^{10}	2.4×10^{10}
Peak Cu dpa in PF3 coil (dpa)	7.9×10^{-3}	8.7×10^{-3}
Peak Cu resistivity increase in PF3 (nΩm)	0.74 (4.7%)	0.70 (4.4%)



- *Peak insulator dose levels exceed 10^{10} Rads locally at PF coils adjacent to mid-plane port and in divertor region. No credit was taken for added shielding by material in the port. Shielding of insulator layer by coil case was not accounted for*
- *Damage parameters for PF are higher for DCLL blanket*
- *Excessive hot spot dose at corner of PF1 with only 20 cm shield*
- *Recommend moving PF1 upward by ~15 cm to add more shield*

THE UNIVERSITY of WISCONSIN MADISON

Summary and Conclusions

- 3-D neutronics analysis performed using FDF CAD
- Two blanket concepts considered; DCLL and HCCB
- Based on NWL results, fusion power reduced to 240 MW yielding a peak OB NWL of 2 MW/m^2 and a fluence of 6 $MW\text{-}yr/m^2$
- TBR adequate for both blankets with minor blanket configuration change in divertor region
- VV is re-weldable during FDF lifetime
- Modest nuclear heating, atomic displacements and conductor resistivity increase occur in coils
- Cumulative end-of-life organic insulator dose levels in TF and OH coils are acceptable
- Excessive hot spot dose at corner of PF1 with only 20 cm shield
- Recommend moving PF1 upward by ~15 cm to add more shield