The Fusion Nuclear Science Facility (FNSF) is a Critical Step Before Proceeding to Larger and Electricity Producing Fusion Plants

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The Fusion Nuclear Science Facility (FNSF) is examined in a recent study [1-19], as the first significant fusion nuclear step beyond ITER. In the U.S. this device would precede the DEMO, which provides routine electricity production and operations. This two-step approach to power plant realization recognizes the present landscape in the U.S., and internationally, to establish the basis for fusion energy power plants in as few confinement facilities as possible. The motivation for a smaller first step, like the FNSF, has been based on concern over the material and component behaviors in the actual fusion service environment, which cannot be tested in advance of the FNSF. The pre-requisite R&D to qualify fusion core components of the FNSF would mainly consist of fusion relevant neutron exposures of all individual materials, and highly integrated non-nuclear testing of fusion core components. However, this does not adequately represent the complete fusion environment, and the FNSF serves as the platform to establish a new database of material and component behavior. It is strongly anticipated that this actual environment is different and will give rise to behaviors not seen before. This demands an intermediate step before pursuing electricity production at larger scale. The material development experience in fission has many such examples of these issues. Table 1 shows a few parameters characterizing the large leap taken by the FNSF from ITER as one enters the fusion nuclear regime, as well as the requirements for a demonstration power plant (DEMO) and a fusion power plant.

Table 1. Parameters describing the evolution into the fusion nuclear regime, for ITER, the FNSF, the DEMO and a Power Plant.

	ITER	FNSF	DEMO	Power Plant
Max neutron damage, dpa	3	40-80	100-150	150
Tritium breeding ratio		>1	>1.05	1.05
Plasma pulse length / on-time % in a year	3000s / 5%	15 days / 35%	15-365 days / >50%	365 days / 85%
$T^{max}_{blanket}$ / $T^{max}_{coolant}$	285 ^oC / 150 ^oC	550 ^oC / 650 ^oC		
Materials	SS, CuCrZr, Be, H_2O, W	Reduced Activation Ferritic Martinsitic (RAFM) steel, Bainitic steel, W, He, SiC-SiC composite, PbLi		

Figure 1 shows a notional time-line placing the ITER operating phases, with the FNSF beginning ~ 3-4 years after the first ITER DT operations, but only entering its DT phases unitil after another ~ 3 years. The short and long pulse DD tokamak experiments provide plasma configuration demonstrations relevant to the FNSF, in particular 100% non-inductive plasmas, higher beta plasmas, integrated core-edge plasma solutions, and some edge plasma-material evolution. The pre-FNSF R&D occurs over ~ 19 years prior to the FNSF start, with various design activities stretched over ~ 14 years, and a 7 year construction is assumed. A similar design and construction is shown for the DEMO. Important to note is that some R&D areas are expected to continue in parallel with the FNSF, in particular, materials development and irradiation qualification, integrated component demonstrations at more aggressive operating regimes, and an intense study of the material/component observations from the FNSF itself. In addition, pre-DEMO R&D and qualification is performed to prepare for this facility, in particular, in areas such as further materials development to the highest neutron exposures, systems optimizations to enhance plant power balance, prototype the balance of plant systems, and enhance designs based on the FNSF experience.

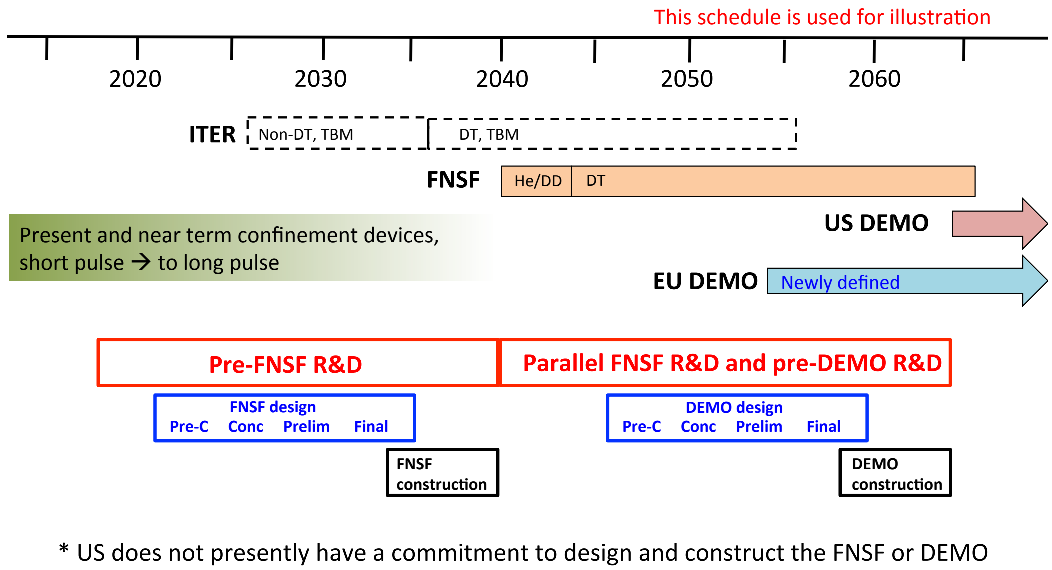


Figure 1. A postulated schedule of FNSF along with ITER, the EU DEMO, and programs supporting the FNSF and US DEMO. This reflects the recent ITER schedule.

The small number of steps to the power plant regime also implies that power plant relevant choices must be made in both the FNSF and DEMO to provide the necessary development time and experience gained from both facilities (e.g. no water in the fusion core, superconducting magnets). Subsequent with this is a strategy to pursue the most power plant attractive blanket, as we understand it today, consistent with materials development and engineering analysis. The Dual Coolant Lead-Lithium (DCLL, RAFM steel structure, He coolant, PbLi breeder/coolant) blanket was justified based on the highest potential thermal conversion efficiencies of blankets utilizing RAFM steel structures, and having in-situ Li-6 and tritium breeding ratio (TBR) control. Alternative blankets were chosen using the same structural material and main coolant (He), and which addressed the most vulnerable aspect of the DCLL, the liquid metal breeder. These were chosen to be the Helium Cooled Lead-Lithium (HCLL) and Helium Cooled Ceramic Breeder (HCCB/PB). These alternate blankets follow the strategy of power plant relevance, and are carried along with test blanket modules in the FNSF.

A set of ten high level missions have been established that encompass all the required aspects of a fusion power plant, and therefore become the mission elements for the FNSF. The FNSF will advance each mission to some degree, and this is measured by a series of metrics. The metrics provide a way to determine progress toward the power plant regime. These missions pertain to the fusion core, the ex-core and overall plant functions, and ultimately include all plant subsystems needed in a fusion power plant. A FNSF can be envisioned to advance a few of these missions by a small degree, or conversely all the missions by a large degree, see Fig. 2. The moderate FNSF has been studied here in detail [6-18], and a possible minimal and maximal FNSF have been examined in systems analysis. This moderate FNSF has been shown to advance all mission elements significantly, with the notable exception of net electricity production. In addition, since electricity is not targeted, pursuing the highest efficiencies in subsystems (e.g. heating and current drive) may also be deferred to the DEMO. Condensed versions of the missions are listed below, see ref [6] for full descriptions and metrics.

1. Strongly advance the fusion neutron exposure of all fusion core (and near and ex-core) components towards the power plant level.
2. Utilize and advance power plant relevant materials
3. Operate in power plant relevant fusion core environmental conditions
4. Produce tritium in quantities that closely approaches or exceeds the consumption in fusion reactions, plant losses and decay.
5. Extract, process, inject and exhaust significant quantities of tritium in a manner that meets all safety criteria
6. Routinely operate plasmas for very long durations
7. Advance and demonstrate enabling technologies that support the very long duration plasma operations with sufficient performance and reliability
8. Demonstrate safe and environmentally friendly plant operations
9. Develop power plant relevant subsystems for robust and high efficiency operation
10. Advance toward high availability

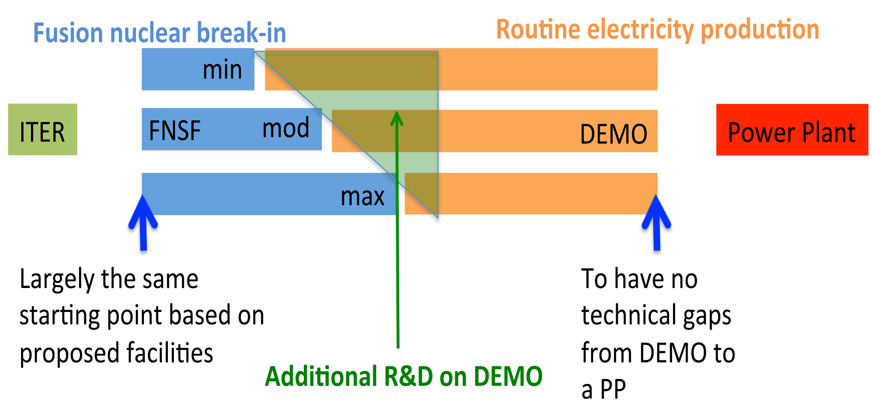


Figure 2. A schematic describing the two facility development path with a FNSF and a DEMO, where the degree that the technical basis is advanced in the FNSF, determines what additional R&D is picked up in the DEMO. A minimal FNSF likely leaves sufficient R&D undone to require an additional fusion nuclear device before DEMO, becoming a three-facility development path.

A strategy for the plasma physics is established in order to provide the required plasma performance and duration that supports the fusion nuclear exposure mission and all subsequent missions (e.g. materials, tritium breeding, etc.). The need to extend the plasma durations to 10-14 days weighed heavily on the choice for the plasma operating space, see Fig. 3. Conservative beta, energy confinement, bootstrap fraction, and current drive efficiency were assumed. Strong plasma shaping was assumed based on the known physical basis for designing passive stabilizer elements and feedback control coils outside the blanket and shield, but inside the vacuum vessel. The density ratio to Greenwald density was taken to be close to 1.0 to better represent the power plant regime, even though lower values are accessible. Mitigated disruptions were assumed, and the double-null divertor configuration and highly radiating divertor solutions were pursued. The plasma is steady state in the flattop, and three of the major H/CD systems are assessed for driving the plasma current (EC is scaled from previous analysis and is used for q-profile feedback control). Plasma discharges with high non-inductive current fraction and significant beta, in combination with the highest multiples of the current relaxation times from experiments, dominated by JT-60U and DIII-D, are examined for guidance. Recent discharges on ASDEX-U show important high density operations with metallic walls and controlled radiation in the core and divertor plasmas. In addition, EAST and KSTAR are rapidly extending their pulse lengths in H-mode plasmas.

Table 2. Parameters for experimental discharges achieved in JT-60U and DIII-D, and the FNSF operating point.

	JT-60U	JT-60U	DIII-D	DIII-D	DIII-D	DIII-D	FNSF
β_N	2.4	1.7	3.5	$\geq$ 3.5	2.0	3.1-3.4	$\leq$ 2.6
τ_{flat} / τ_{CR}	2.8	2.7	2.0	~ 1.5	>2.0	~0.4-1.0	~18000
q_{95}	4.5	~8	6.7	5.5-6.5	4.7	5.0-5.5	6.0
f_{BS}	45%	80%	40-50%	50-60%		60%	52%
f_{NI}	90%	100%	75%	100%		80-100%	100%
H_{98}	1.0	1.7	1.0	1.6	1.3	$\geq$ 1.2-1.3	1.0
q_{min}	1.5		1.5	~ 1.0		1.4	~1.0-1.2
	Steady state	Steady state	Steady state, off-axis NB	Steady state hybrid, hi rot	QH-mode, no ELMs	Steady state	

Systems analysis was used to search the conventional aspect ratio range (3-5) for an operating point that had a small major radius and could satisfy the physics and engineering constraints established for the facility. Although the FNSF can take on various forms, the moderate FNSF was chosen for detailed study, while a minimal and a maximal FNSF were identified with separately. The operating point is chosen to provide a significant operating space, and not lie at the boundary of the space. Lower aspect ratio led to high plasma currents, and otherwise showed no advantage in any other technical comparison. The aspect ratio of 4.0 was chosen, in spite of attractive high aspect ratio points, mainly because these higher values lacked any experimental database. The reference operating point had a major radius of R = 4.8 m, B_T = 7.5 T, and I_P = 7.9 MA, and Q = 4.0, see Table 1. The operating space is explored, in order to understand the robustness of the operating point, by examining parameter variations around the reference operating point (e.g. n/n_{Gr}), and examining the operating space as assumptions and filters were changed (e.g. maximum B-field at the magnet). A minimal FNSF option was examined with the systems code, utilizing copper TF and CS/PF coils, water cooling in the shield and vacuum vessel, and lower lifetime fluence. This led to an operating point at R = 3.5 m major radius, B_T = 5.8 T, I_P = 6.4 MA, and Q = 2.0. A maximal FNSF option was examined with $Q_{engr} \geq 1$, leading to an operating point with R = 5.8 m, B_T = 8.0 T, I_P = 8.0 MA and Q = 7.0.

Table 3. Parameters for the Reference FNSF Operating Point

A = 4			
R, m	4.8	Z_{eff}	2.4
κ_x, δ_x	2.2, 0.63	P_{alpha}, MW	104
Ip, MA	7.87	P_{CD}, P_{total}, MW	121, 129
I_{CD}, MA	3.78	P_{brem}, MW	24.6
B_T, B_T^{coil}, T	7.5, 15.9	P_{cycl}, MW	1.9
$l_i(1)$	0.7	P_{line}, MW	29.3
q_{95}	6.0	$P_{rad,core}$, MW	56
n/n_{Gr}	0.9	$P_{rad,div}$, MW	177
W_{th}, MJ	170	$P_{L\text{-}H\ threshold}$, MW	90
n(0), x10^{20} /m^3	1.93	P_{net} / $P_{L\text{-}H}$	1.96
$\langle n\rangle_v$, x10^{20} /m^3	1.38	n_{He}/n_e	0.0245
n(0)/<n>	1.4	n_{DT}/n_e	0.87
β_N^{th}, β_N^{fast}	2.2, 0.23	n_{Ar}/n_e	0.0045
f_{BS}	0.52	Q, Q_{engr}	4.0, 0.86
τ_E, s	0.86	η_{CD}, A/m^2-W	0.2 (assumed)
$H_{98(y,2)}$	0.99	$\langle N_w\rangle_{plas}$, $N_W^{peak, FW}$, MW/m^2	1.18, 1.78
$T_{e,i}(0)$, keV	23.7	q_{div}^{peak} (OB,IB), MW/m^2	10.0, 3.9
T(0)/<T>	2.6		

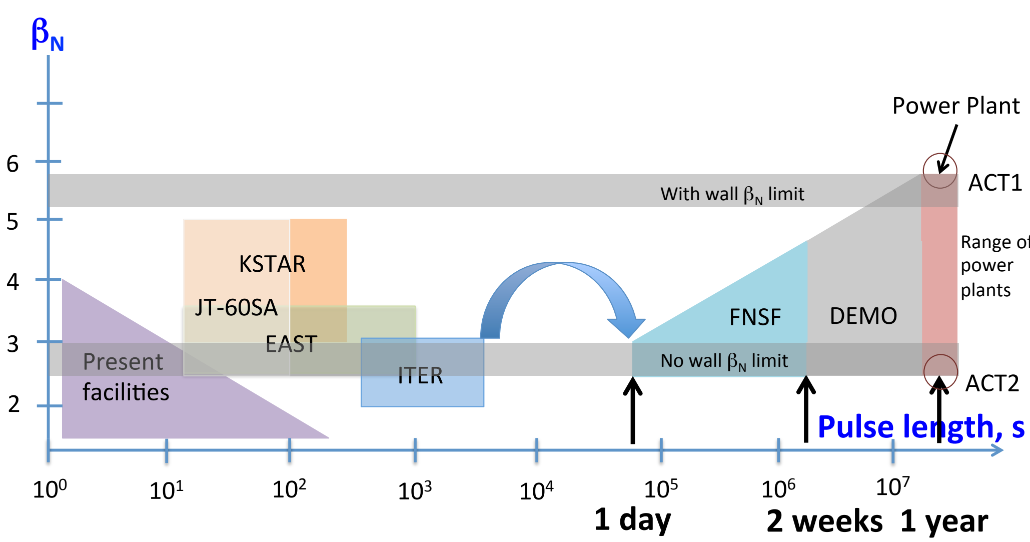


Figure 3. Illustration of the plasma normalized beta versus the plasma pulse duration, showing approximately the present tokamak achievements, the target β_N-duration space for JT-60SA, KSTAR, EAST, and ITER. The FNSF, DEMO and power plant regime are also shown, demonstrating a significant gap between planned facilities and the fusion nuclear next steps. Whether the operating space would remain around the no wall beta limit or approach the with wall beta limit still needs to be established.

A program is established for the FNSF, Table 2, to demonstrate how it accomplishes the advancing of the missions. The moderate FNSF program shows a He/H startup phase for the facility, a DD phase whose primary mission is to establish the longest plasma pulse lengths, and 5 DT phases that progressively advance the neutron fluence on fusion core components through ~ 0.7, 2.0, 3.0, 4.0, and 8.0 MW-yr/m^2 (approximate damage (dpa) for Fe given by 10x these numbers). In addition, the blanket operating temperatures increases, and the structural material advances to higher temperature and higher radiation resistant variants of the RAFM steel. The plasma pulse lengths, duty cycle, and total plasma on-time also increase across these phases. There will be material and operating conditions evolution in the divertor and RF launcher components, but these are too immature to describe at present. The program results in maximum (approximately) fusion core material neutron damage levels of 80 dpa and power plant prototypical blanket operating temperatures. The program timeframes are estimated to provide the required neutron exposure time, and maintenance times. There are 25.3 years of DT operation, 7.8 years of plasma on-time (making neutrons), leaving 17.5 years for the wide range of maintenance activities from inspections to removal of all sectors in the fusion core at the end of each phase. The He/H and DD phases were assigned 1-2 and 2-3 years, respectively based on examining discharge sequences and plasma physics and engineering tasks. Variations in the program plan were considered, such as advancing the plasma duration and duty cycle more rapidly in Phase 3, and operating at reduced fusion power to examine lower blanket temperatures, but in all cases the approximate number of years for the total DT operation remained similar.

Table 2. The baseline program on the FNSF showing the various phases.

Phase	**1**	**2**	**3**	**4**	**5**	**6**	**7**	
	He/H	DD	DT	DT	DT	DT	DT	PP
years	1-2	2-3	2.75	4.5	5.0	6.5	6.5	40 FPY
N_w^{peak}, MW/m^2			1.75	1.75	1.75	1.75	1.75	2.25
Plasma on-time, %/year		15-50	15 55 d/yr	25 91 d/yr	35 128 d/yr	35 128 d/yr	35 128 d/yr	85 310 d/yr
Plasma duty cycle, % (pulse/ dwell)			33 (1d/2d)	67 (2d/1d)	91 (5d/.5d)	95 (10d/.5d)	95 (10d/.5d)	100%
Total maintenance time, days			550 d 200 d/yr	1131 d 229 d/yr	1120 d 224 d/yr	1495 d 230 d/yr	1495 d 230 d/yr	2585 d 55 d/yr
Peak* dpa Peak appm He			7.2 73	19.7 200	30.6 310	39.8 403	79.6 806	150-200 1500-

Peak appm H			327	894	1388	1806	3612	2000 6800-9100
Max blanket structure op temp, ^{o}C	<550	<550	350-550	350-550	400-600	450-650	450-650	600
Blanket Structure material	Gen I - RAFM	Gen I - RAFM	Gen I - RAFM	Gen I - RAFM	CNA	CNA/ ODS	ODS	

*peak refers to outboard midplane at the first wall, and refers to the Fe approximation for dpa/MW-yr/m^2
**CNA – cast nano-structured alloy, ODS – oxide dispersion strengthened

As part of the program, a distribution of sectors (16 total), comprising the fusion core, is assigned with penetrations for heating and current drive, diagnostics port plugs, 2-4 test blanket modules for alternative blanket concepts and look-forward of the next phase blanket, and one material test module are postulated. The phases are segmented to allow inspections and maintenance. The full distribution of the penetrations in the sectors was examined with 3D nuclear analysis, showing a TBR of 1.07 for the worst case, see Fig. 4. The basic maintenance actions and hot cell functions are described [6]. The Hot Cell is found to be a critical aspect of the facility to provide the material and component in-service characterization, including post-irradiation examination. This is how the FNSF will provide the working material/component database for future fusion facilities.

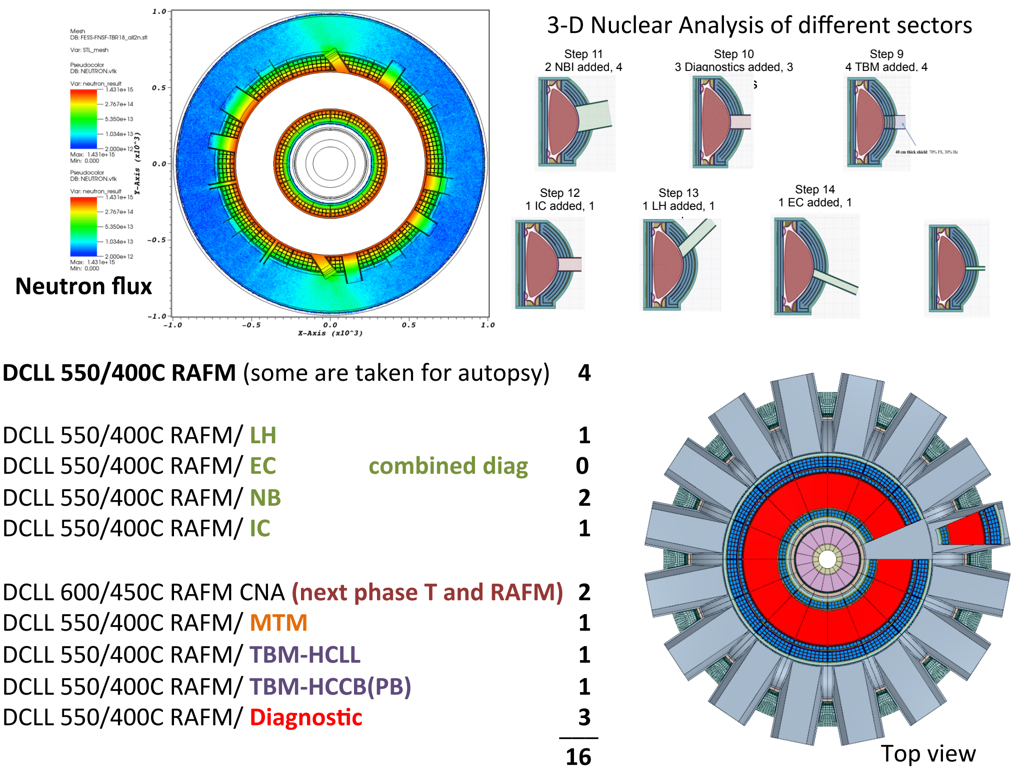
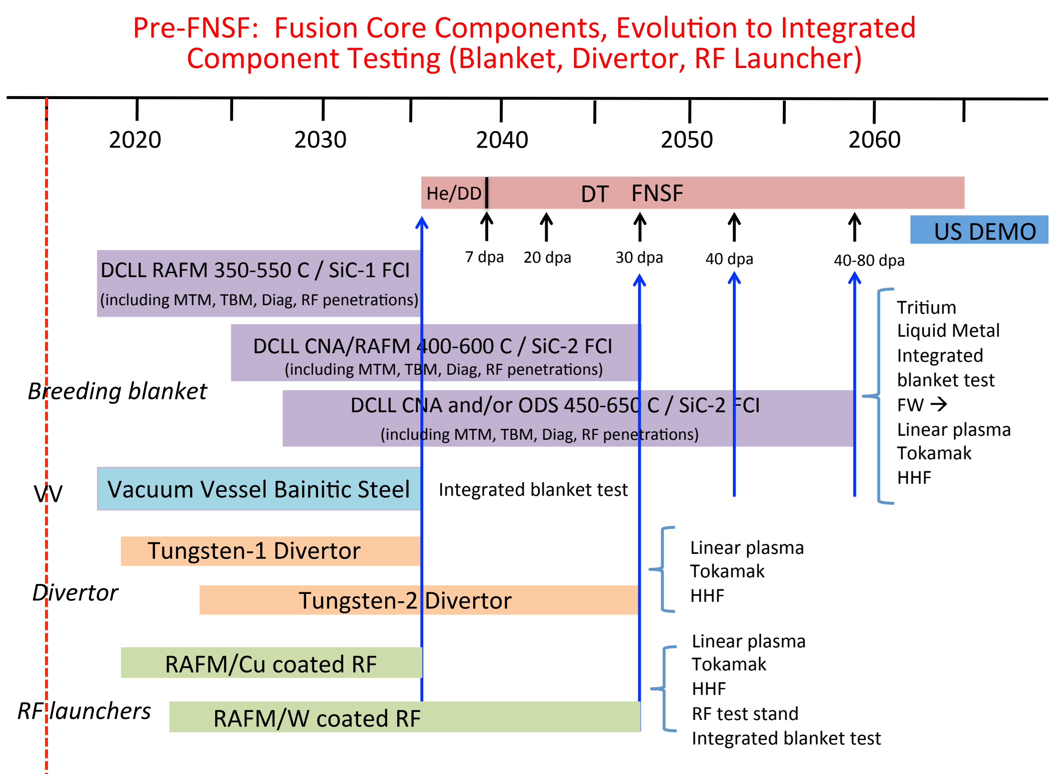


Figure 4. The various sectors are shown from a top view, described by their blanket concept and penetration for H/CD, diagnostics, MTM or TBM. Nuclear analysis showed a TBR of 1.07 with all penetrations included. Sectors are removed after their exposure in each phase and taken to a hot cell for examination. The neutron flux is shown in a color contour map with the wide range of sector penetrations.

A wide range of detailed assessments were performed in physics and engineering to better quantify the parameters characterizing the moderate FNSF configuration identified in the systems analysis. These are reported in companion papers [6-18], and briefly described in Appendix 1. Overall, they confirm the ability of this configuration to meet the fusion nuclear and plasma requirements.

The R&D required for the FNSF is described in 6 major categories 1) fusion nuclear material science, 2) PMI and PFC science, 3) tritium science, 4) liquid metal breeder science, 5) enabling technologies, and 6) plasma development. The first four are considered very high priority in order to demonstrate the feasibility of fusion as an energy source. The first two are strongly driven by the access to facilities that provide the exposures to fusion relevant neutrons and plasma, respectively. The tritium and liquid metal breeder areas begin as separate thrusts and later transition into the integrated blanket component testing. The enabling technologies area holds many important support systems that must be advanced in order to provide power plant relevant systems with very high reliability (e.g. magnets, tritium processing, heating and current drive). For the FNSF plasma development is focused on integrated plasma operating scenarios targeting power plant relevant features for ultra-long pulses, with some ability to access higher beta and fusion gain. Fig. 3 shows how the blanket, vacuum vessel, divertor and an RF launcher would be developed through non-nuclear materials, fission and fusion nuclear materials testing, industrial/manufacturing, and progressive integrated testing to arrive at the FNSF for a specific phase corresponding to dpa, operating temperature, and so forth. The blanket component is further broken down to show the R&D elements involved in establishing its technical basis.

This study has shown how important a moderate FNSF can be for breaking into the fusion nuclear regime and significantly advancing fusion development down the pathway to commercial fusion electricity production. The facility is a combination of discovery science and technical demonstration, and is required due to the severe complexity of the integrated fusion system. This study has recognized that the FNSF is an integrated fusion facility, requiring all its science and technologies to be advanced, and the important connections the device has to R&D programs and facilities preceding and following it.



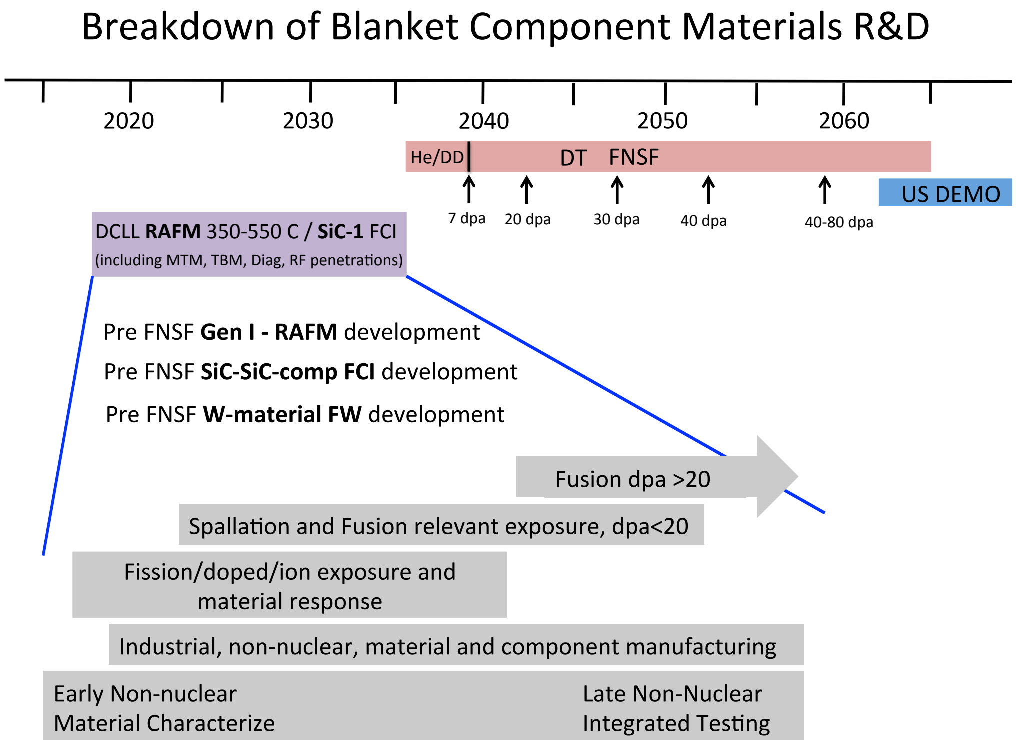


Figure 5. The fusion core components development and delivery to the FNSF, along with subsequent more advanced components at later phases of the FNSF where higher neutron fluences and operating parameters are reached.

Appendix 1: Summary of Detailed Technical Analysis

A series of technical analyses were performed on the reference operating point for the FNSF in order to obtain more detailed information about its performance and features. Both core plasma physics and SOL/divertor simulations were used to examine the operating point and find the requirements for its viability. Engineering analyses included nuclear analysis, thermo-mechanics and thermal hydraulics, transient thermo-mechanics, liquid metal MHD thermofluids, tritium behavior, TF and CS/PF magnets, ion cyclotron and lower hybrid radio-frequency assessments, and materials issues. Figure 6 shows a cross-section of the FNSF operating point. Brief summaries are given here and companion papers with more detail are noted. Some highlight pictures are supplied from each of the technical areas below.

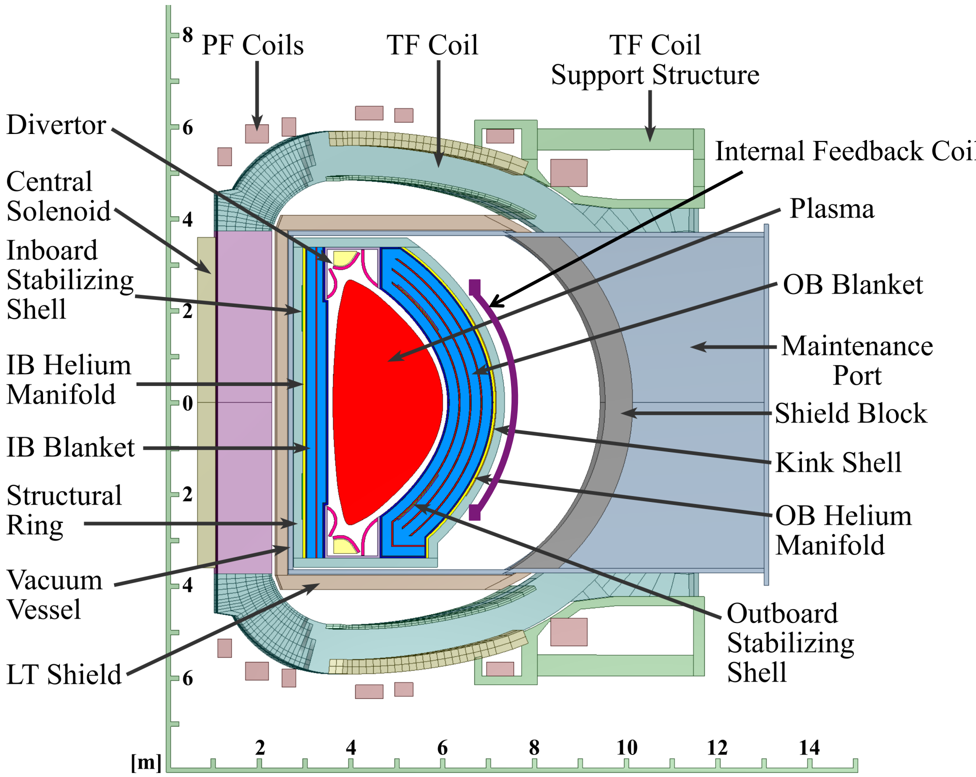


Figure 6. The cross-section of the FNSF, identifying the various components in and near the fusion core.

Core plasma physics [10] established the CS/PF coil currents, pedestal height with the EPED model, confirmation of ideal MHD no wall beta limit to n=1 kink modes, heating and current drive assessments for NINB (ACCOME), LH (GENRAY-CQL3D) and ICRF (TORIC5) with scaling from previous work for the EC (GENRAY), and time-dependent transport evolution into the flattop phase. These showed for NB, LH, and ICRF, that the current drive efficiencies were 0.28, 0.22-0.26, and 0.2 A/W-m^2, respectively. These imply certain penetrations in the FW that are used in the nuclear analysis for the tritium breeding. Time dependent free-boundary plasma evolution scenarios were produced with the Tokamak Simulation Code (TSC) from the 0.5 MA to 7.9 MA, and into flattop until the plasma was fully relaxed. The Coppi-Tang semi-empirical and GLF23 theoretical models were examined for energy confinement, with the former showing that $H_{98} \sim 1.1$ was required to reach 100% non-inductive plasma current and 80-100 MW of alpha power. GLF23 predicted lower confinement $H_{98} \sim 0.8$, and therefore lower alpha power and non-inductive current fractions. Both argon and tungsten were included as impurities, and the EPED pedestal correlation $p_{ped} = f(n_{ped}, Z_{eff,ped})$ was enforced in the simulation. The plasma fueling and exhaust rates are determined to maintain a steady state He concentration in the core plasma ($f_{He} = n_{He}/n_e$) with a corresponding He enrichment in the divertor ($\eta_{He} = n_{He}{}^{o}n_e/2n_{DT}{}^{o}n_{He}$), giving 256, 250 Pa-m^3/s, and 1.3% for $\eta_{He} = 0.2$, and 56, 50, and 6.4% for $\eta_{He} = 1.0$, for the fueling rate, exhaust rate, and tritium burnup, respectively. This does not include any additional gas demands.

SOL/divertor simulations [15] have verified that the two high radiating divertor solutions sought are accessible in the FNSF, the partially detached ITER-like (tilted plate) and the fully detached orthogonal plate configurations. In fact, it is confirmed that a continuum of solutions exists transitioning from one to the other. The ITER-like divertor had peak outboard divertor heat flux values > 6 MW/m^2, with ~ 75% of the power entering the divertor radiated. The orthogonal plate solution obtained < 3 MW/m^2, and radiated > 95% of the power entering the divertor. For the perpendicular transport assumptions, the heat flux width at the outboard midplane lies between 1.5 and 2.0 mm over the first ~ 4 mm, and becomes progressively larger across the SOL. The impact of molecules versus atom-only treatments indicate the strongest effects are in the private flux region, but otherwise are minor. The tungsten sputtering and transport is examined to begin developing a description of the erosion/re-deposition/migration phenomena. Estimates for the particle fluxes and energies to the wall surfaces are determined, and tungsten is introduced into the SOL and followed to steady state. These activities will continue to refine the description of PMI for plasma facing component design.

Nuclear analysis for the FNSF [3,14] began with 1D models to provide early input on the inboard radial build, shielding and other nuclear impacts. This was followed by 3D CAD-based modeling. The plasma neutron source was examined to improve its treatment, showing slightly higher dpa/FPY. The peak neutron wall load on the outboard midplane is 1.75 MW/m^2, while the peak on the inboard is 1.31 MW/m^2, and the maximum in the divertor is 0.79 MW/m^2. First of a kind tritium breeding ratio calculations were done examining > 10 different sector geometries that accommodate diagnostics, 4 different heating and current drive sources, test blanket modules and a material test module. The full sector TBR was 1.12, and the final value with the most pessimistic penetration assumptions gave 1.07, with a Li-6 enrichment of 90%. Analysis also determined the radiation damage and gas production, heating, and shutdown decay heating and dose over the fusion core and near core components, as well as the neutron flux and dose at the TF magnet.

Thermal-hydraulics, thermo-mechanics and computational fluid dynamics [9] were used to examine the DCLL blanket. A multi-physics approach is used for the inboard blanket assessment with the COMSOL suite. Normal and off-normal loading are examined and tested against ITER SDC-IC design criteria including nuclear effects. The RAFM structure could be maintained within temperature limits for both 5 and 8 MPa He coolant pressures. The performance for off-normal conditions at 8 MPa were found to be marginal, and parametric analysis on the first wall design found approaches to improve this. High heat flux capable first wall designs for power plant regimes are lacking, and efforts to explore possible solutions were performed [13], optimizing structural materials and geometry, found solutions reaching up to ~ 4.5 MW/m^2 for tungsten/RAFM.

The liquid metal breeder/coolant LiPb was analyzed with MHD thermohydraulics [11] in the full poloidal DCLL blanket. This blanket design requires an electrical and thermal insulating flow channel insert to separate the majority of the liquid metal from the RAFM steel walls, and SiC and a RAFM/alumina/RAFM sandwich were examined. Pressure drops were determined for the blanket flow, blanket inlet/outlet, access pipe flow, and at flow direction changes in the blanket. The SiC FCI was found to yield low pressure drops, while the sandwich exceeded the allowable. In addition, the pressure drops associated with complex 3D MHD flows were explored in detail computationally and new correlations were derived to represent them. Required R&D for the LiPb breeder is described, and the FNSF is shown to have similar dimensionless numbers (Ha, Re, Gr, N) to power plants, indicating it is a viable platform to establish the LM flow characteristics.

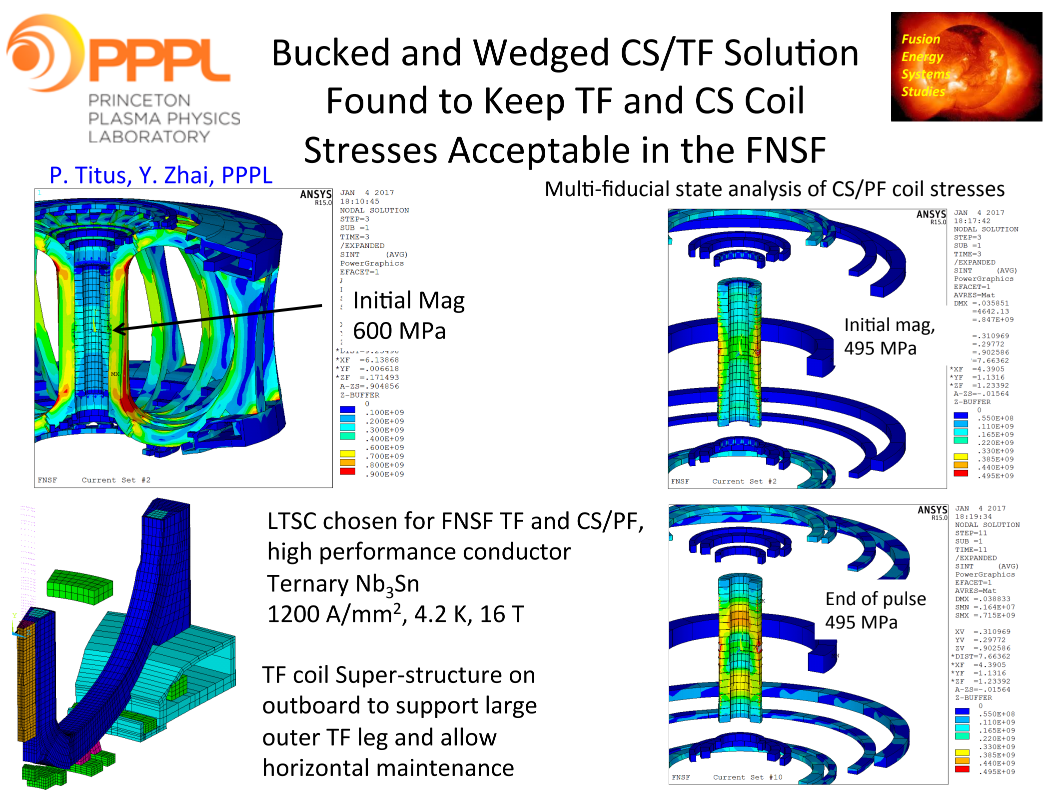
Transient thermo-mechanics in the FNSF is examined, exploring ELMs and mitigated disruptions [17]. The divertor is assumed to be the plate-type all tungsten design that employs jet impingement to enhance heat transfer. The first wall is RAFM steel with a thin (~0.2-0.5 mm) tungsten coating. ELMs resulted in the divertor tungsten armor almost reaching melting at the surface, but only a 100 ^{o}C change at 1 mm depth. Cyclic plastic strain is only seen to 1 mm depth as well. The ELM affects the FW by raising its temperature to ~ 740 ^{o}C, well below the melting for tungsten, with an RAFM/tungsten interface temperature reaching about 540 ^{o}C. The stresses remained within allowables for the materials. For the double-null plasma in the FNSF, the midplane disruption is the most likely, since the plasma sits at or near its neutral point. Disruption thermal quench and current quench are also examined. The induced currents and electromagnetic body forces are determined for the major components. The largest currents occur in the various tungsten parts, including the FW coating and the stabilizer plates.

The tritium breeding circuit is analyzed with the TMAP code [16], and indicates that < 6 g/year for all LiPb and He loops can be achieved, in the worst case, without any additional enclosures. This makes the target of < 1 g/year losses to the environment a realistic outcome, particularly since all these tritium bearing circuits would have enclosures. This has been facilitated by high LiPb flow rates, SiC flow channel inserts, efficient tritium extraction (vacuum permeator), and co-axial piping (hot leg inside annular cold leg). The uncertainty in tritium transport parameters was also examined, showing that the total leakage could be as low as 0.05 g/year. The tritium extraction has been studied in detail, and group 5 metals are the primary permeation window candidates, however modifications are needed for the fusion specific environment. Other tritium loss channels are examined for their sensitivities to assumed operating conditions. The tritium inventories are also reported, for the fusion core components, and breeder and coolant loops through heat exchangers to the environment. These indicate very low values of < 1.5 grams, dominated by structures and the PbLi. The tritium fueling and exhaust loop is not modeled in this study. With the extensive development of this tritium loop, and processing on ITER, this is not considered the most critical development area for tritium on the FNSF. It would be necessary to model this in order to determine the total plant inventories, exhaust-to-fuel times, storage requirements, and transient responses like startup. The safety and licensing strategy for the FNSF is not developed in this study, primarily because the pre-conceptual design is insufficient to allow detailed analysis, and the governing/licensing agency is not determined (e.g. Department of Energy (DOE) or Nuclear Regulatory Agency). The safety culture in the US fusion program is reflected in the DOE Standard, Safety of Magnetic Fusion Facilities: Requirements. In fact, virtually all the technical decisions made in the FNSF study have strong safety aspects imbedded in them. The ITER experience will also be an important element in the evolution to a US fusion licensing framework.

Low temperature superconductor was chosen for the toroidal field coils [7], taking advantage of advanced Nb_3Sn conductor, with higher current density and field capability, and advanced winding pack design. These allow the field at the coil to reach 16 T, and 7.5 T at the plasma center. Recent irradiation data was used along with ITER coil data to arrive at the detailed winding pack description of materials, their fractions, and distributions. A “watch” is placed on high temperature superconductor research to keep track of this development. Mechanical analysis of the TF magnet (and CS/PF) has identified a structural solution for the large outer TF leg associated with the horizontal maintenance scheme. The CS coil can be a LTSC with a bucked and wedged TF/CS structural solution, or it can pursue HTSC solution and possibly remain in the pure wedged configuration.

Radio-frequency heating and current drive for ion cyclotron and lower hybrid frequencies is examined for the FNSF [8]. GENRAY-CQL3D analysis provided the basis for the current drive utilized in time-dependent simulations, and TORIC5 analysis was used to identify the optimal frequency for the ICRF system to provide heating and fast wave current drive. The high field side (HFS) was studied as a less harsh launching environment compared to the low field side, and for LH it provides ~ 20-30% higher CD efficiency with somewhat deeper deposition. The waveguide design for the HFS LH launching could also be used on the LFS, resulting in considerably lower loss of breeding blanket volume. The cooling, materials and system efficiencies are discussed for the FNSF environment.

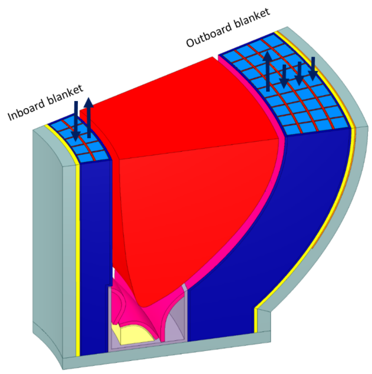
Materials science permeates all of the fusion nuclear science, and coordinating integrated studies with material assessments and developments is essential [12]. The RAFM Generation I material (e.g. EUROFER, F82H, CLAM) provides the basic structural material for the fusion core in the FNSF early phases, while advances to cast nanostructured alloy RAFM and oxide-dispersion strengthened RAFM are being developed to increase strength, operating temperature and irradiation resistance. In addition, LiPb corrosion resistance is being examined through the introduction of aluminum to a high Cr ferritic steel. Tungsten, tungsten composites, tungsten fiber/matrix, and tungsten alloys are all being considered/examined to understand their behavior when subject to plasma and nuclear loading. SiC composite have been qualified for use in fission, and can provide the basis for a SiC flow channel insert in the DCLL blanket. A bainitic steel is being explored as the structural material for the vacuum vessel in order to avoid post-weld heat treatment. Characterizing the fusion core and near core environment in terms of dpa, He/dpa, H/dpa, transmutations, temperature, stress, tritium/hydrogen concentration, B-field, fluid flows, and gradients in these quantities is critical for the materials science community to address and optimize material development and coordinate with engineering design.



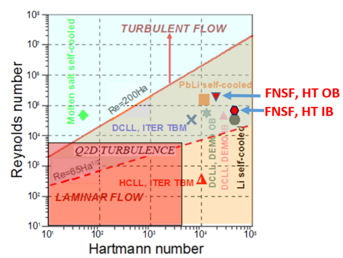
MHD Flow Model with 2D and 3D MHD Solvers are Used to Determine the Pressure Drops in PbLi flows in the FNSF DCLL Blanket

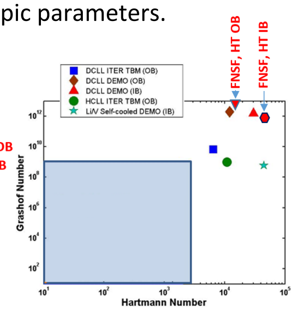
S. Smolentsev, UCLA

For 518 MW of fusion power and a peak neutron wall load of 1.75 MW/m2, a SiC Flow Channel Insert (FCI) was found to have acceptable pressure drops in the blanket, while the sandwich FCI did not unless the fusion power could be reduced.



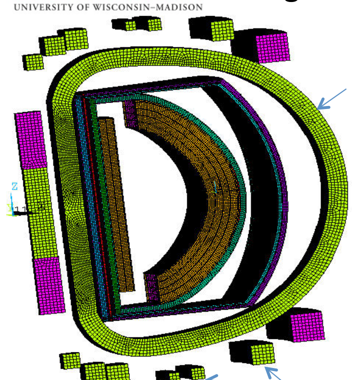
The dimensionless numbers, *Re*, *Ha*, *Gr* can be used to identify flow regimes, and to serve as metrics to measure R&D progress on the pathway from the FNSF to DEMO. The FNSF provides DEMO-prototypic parameters.





Disruption Induced Stresses Can Cause Surface Cracks or Damage from ELMs to Progress Deeper into the Armor Than They Would on Their Own

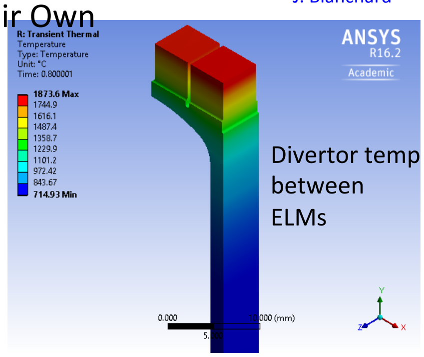
J. Blanchard



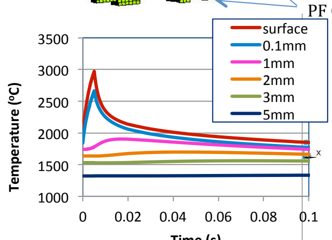
Examining:

ELM→ divertor
ELM→ FW
TQ→divertor
TQ→ FW

CQ

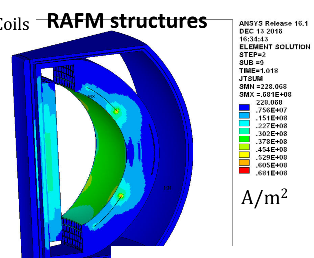


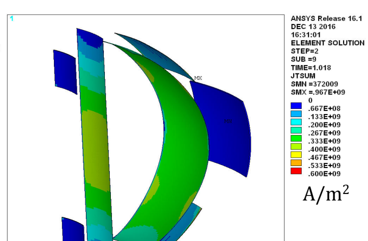
Divertor temp between ELMs



Divertor during TQ disruption

RAFM structures





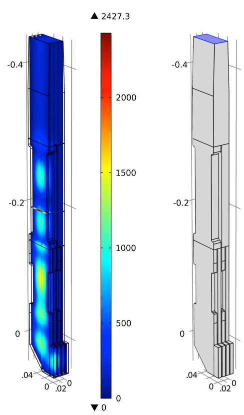
W structures

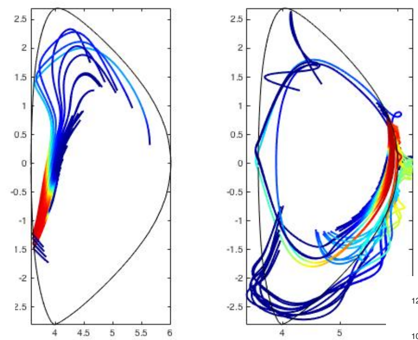
Induced currents disruption

Lower Hybrid Current Drive Analysis for the FNSF Shows High CD Efficiency and Effective Launch Locations on the HFS and LFS

G. Wallace, MIT

HFS antenna takes advantage of quiescent plasma and high B-field to obtain deeper wave penetration to core plasma ➔

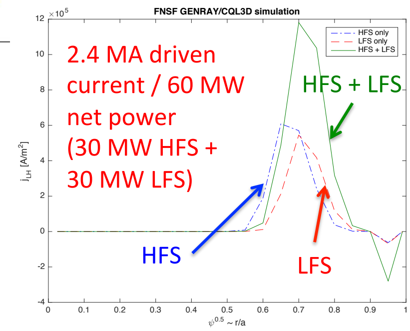




Ray-tracing results from GENRAY-CQL3D, power is very low after red rays

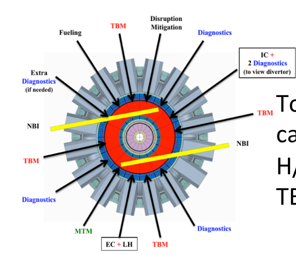
🡐 LFS antenna launches from above the midplane to increase its η_{CD}, but avoiding interference with passive stabilizer plates

🡐 Compact multijunction antenna concept for use with HFS launch of LH waves, and same approach can reduce LFS LH footprint in breeding blanket



3D Nuclear Analysis Used to Analyze Wide Range of Sectors to Determine the Tritium Breeding Ratio for FNSF

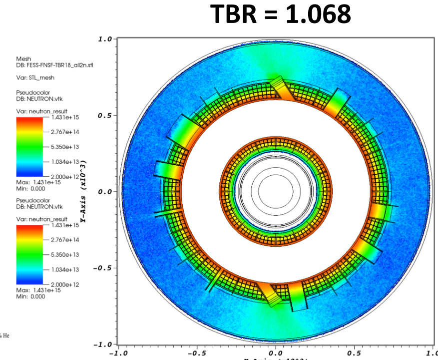
A. Davis, L. El-Guebaly, M. Harb, UW



Top view with worst case penetrations; H/CD, diagnostics, TBM, MTM

Neutron flux, all sectors together

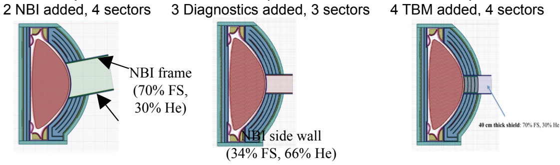
TBR = 1.068



Step 11
2 NBI added, 4 sectors

Step 10
3 Diagnostics added, 3 sectors

Step 9
4 TBM added, 4 sectors



Step 8
1 MTM added, 1 sector

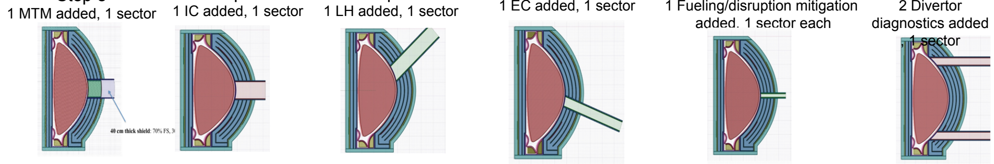
Step 12
1 IC added, 1 sector

Step 13
1 LH added, 1 sector

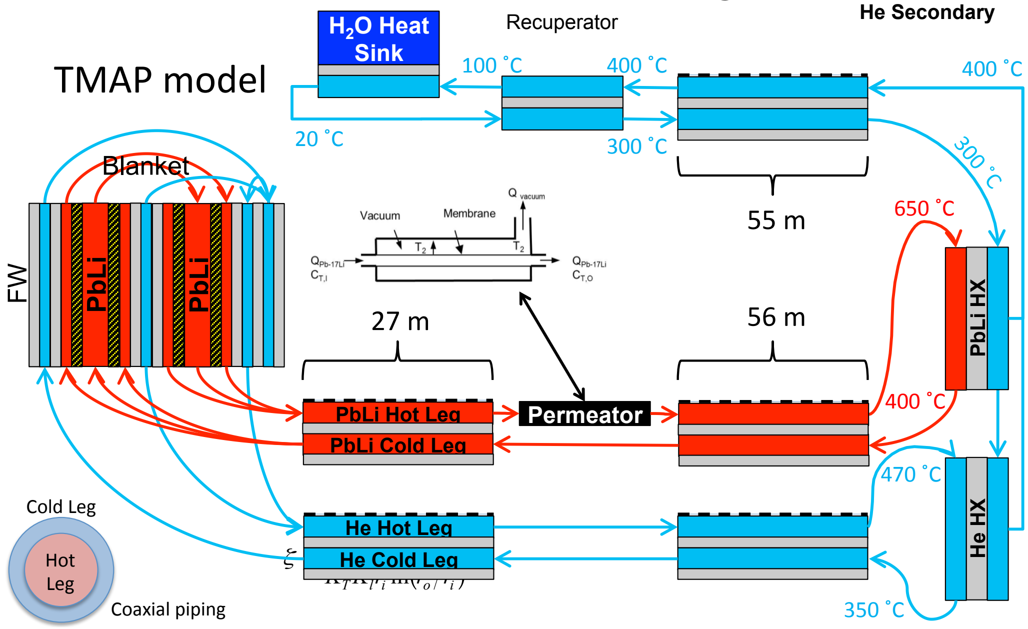
Step 14
1 EC added, 1 sector

Step 15/16
1 Fueling/disruption mitigation added, 1 sector each

Step 17
2 Divertor diagnostics added, 1 sector



Tritium Modeling Shows Low Leakage, < 3 gm/yr for Entire Coolant Loops, with No Enclosures…Checking VV

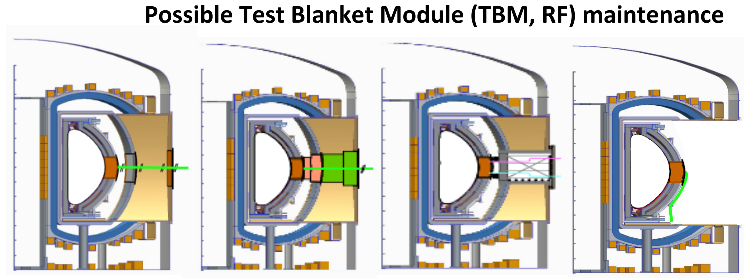


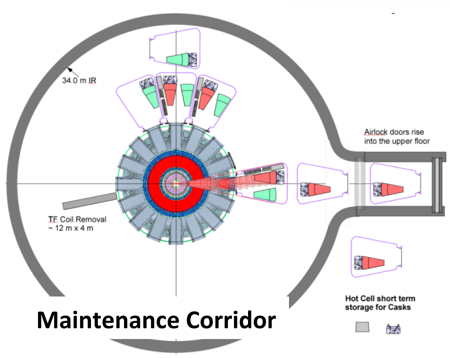
Full Sector Maintenance is Pursued to Provide Fast, Flexible and Reliable Approach

Possible Test Blanket Module (TBM, RF) maintenance

Inspection
Minor Maintenance
Major Maintenance

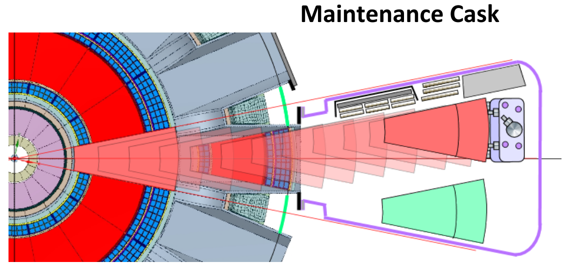
How much time does it take?



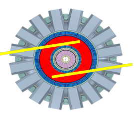


Maintenance Corridor

Maintenance Cask

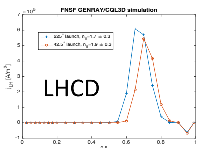


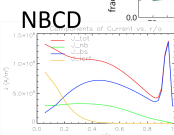
Free-Boundary Plasma Evolution, H/CD analysis, Stability, and CS/PF equilibria Show Viable Core Plasma

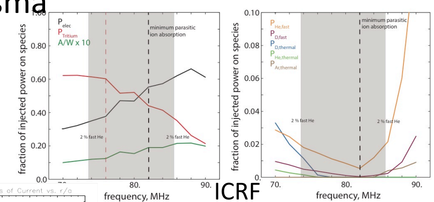


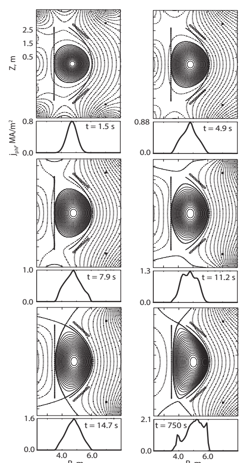
NBs on FNSF

- CS/PF coil
- EPED & n=1 stability
- Free-boundary time-dependent
- H/CD
- Energy Transport

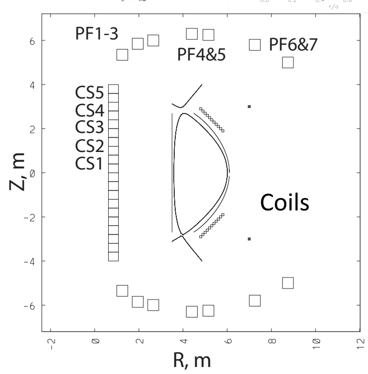


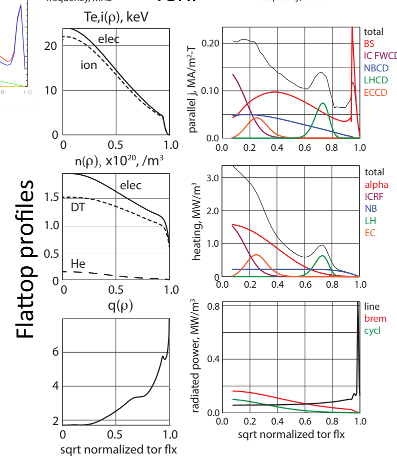




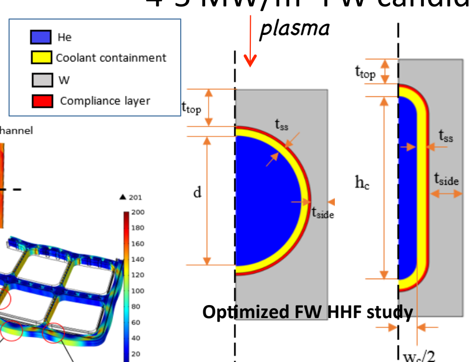


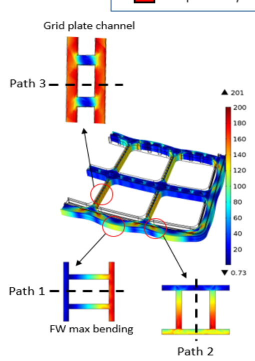
rampup

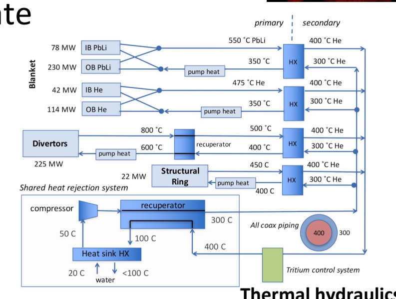




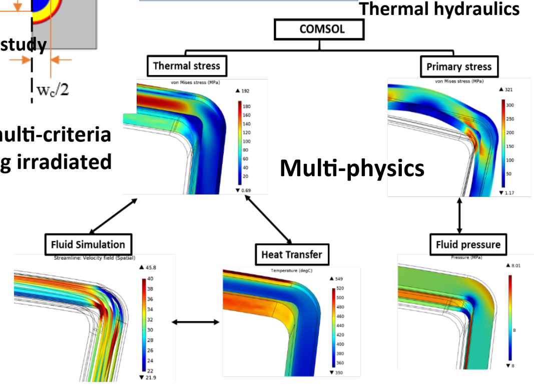
Blanket and FW Thermo-Mechanics Show Robustness to Over-pressure Events and ~ 4-5 MW/m^2 FW candidate







Multi-load path & multi-criteria assessment including irradiated properties



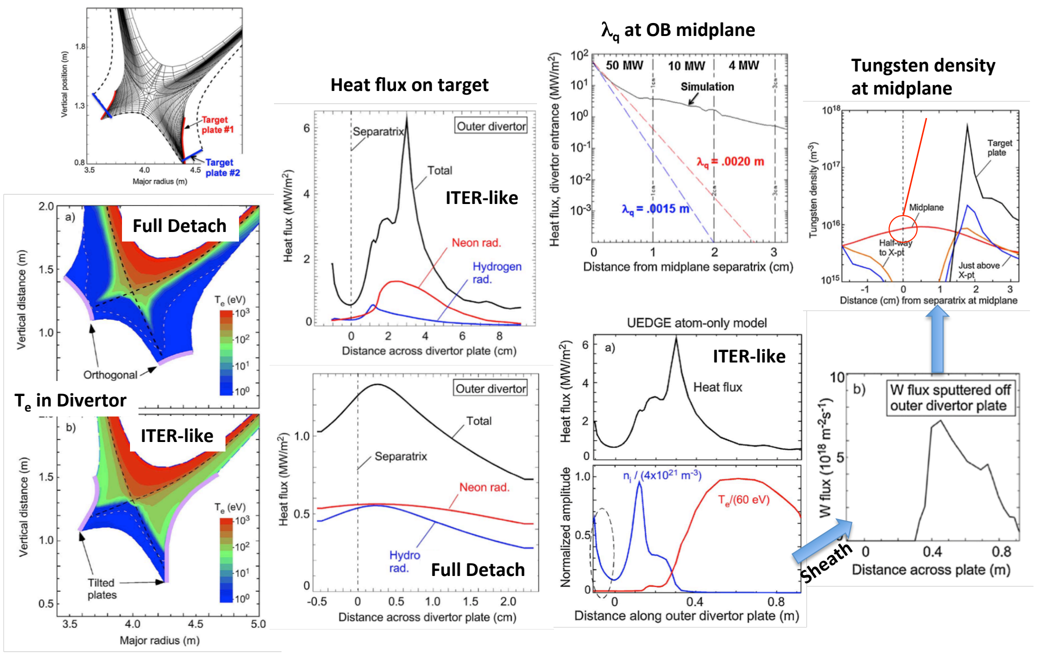
Allowable primary membrane stress intensity (S_m):
Allowable primary plus secondary membrane stress intensity (S_e):
Allowable total stress intensity (S_d):

Necking and plastic instability limit:
$3S_m$ Criteria for primary (P) and secondary (Q):

Plastic flow localization limit:

2D SOL/Divertor Modeling Shows Radiative Solutions and Impurity Generation/Transport

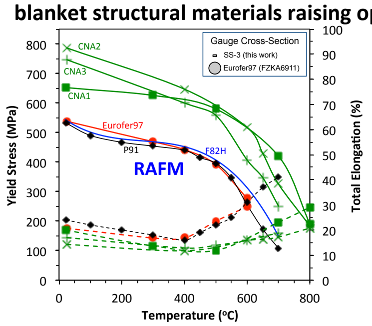
Fusion Energy Systems Studies



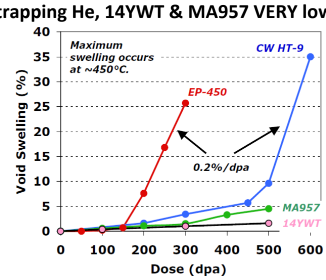
Structural Steels for Fusion, Moving from RAFM to Advanced Alloys for Radiation Resistance, High Strength at High Temp

Fusion Energy Systems Studies

Cast Nano-structured Alloys (CNA) for blanket structural materials raising op temp



Ion Irradiation Tests of FM steel with precipitates for trapping He, 14YWT & MA957 VERY low swelling



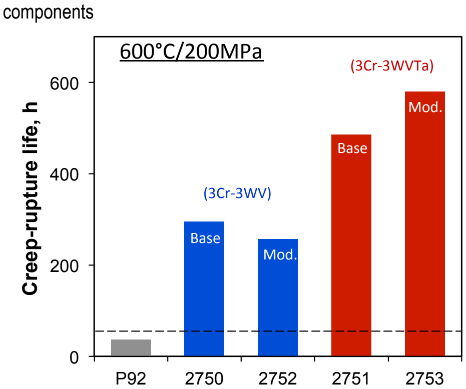
High dpa

Bainitic Steels: Fusion structural applications in next-step fusion devices (FNSF or DEMO);

- Vacuum vessel (>400-500°C, relatively low dose)
- Structural ring, magnet shields

3Cr-3WV(Ta) steels:

- Inexpensive low alloy steel
- Improved creep properties due to formation of carbide-free acicular bainite ferrite (lower bainitic microstructure)
- potentially no requirement of PWHT, suitable for large volume components

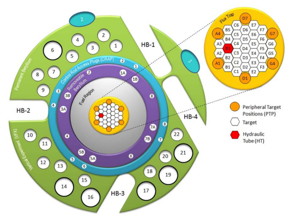


A. Rowcliffe, L. Tan, Y. Yamamoto

We need to understand the basic science of how tungsten behaves under neutron irradiation so we can design better tungsten-based materials for future reactors

L. Garrison, and ORNL FusMat

HFIR tests of W, TITAN and PHENIX projects



Identified impurity contents for commercial “high purity” tungsten

Keeping FESS up to date on irradiation results from HFIR

Explaining International experiments and results

VM-tungsten (K addition and grain distortion) was reviewed, but found not to help with recrystallization or irradiation

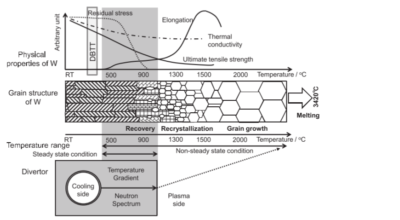
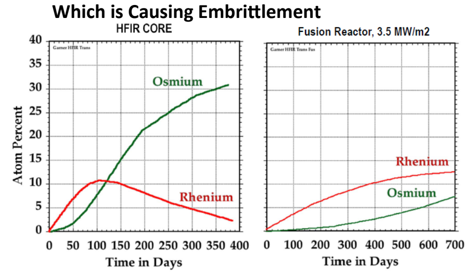


Fig. 1. Schematic of the temperature dependence of W properties, the grain structure of worked W, and the temperature distribution of a mono block divertor of fusion reactor.

Fusion Neutron Spectra Create 10x Lower levels of Re and Os Than Fission Spectra, Which is Causing Embrittlement



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[2] P. W. Humrickhouse and B. J. Merrill, *Vacuum Permeator Analysis for Extraction of Tritium from DCLL Blankets*, Fusion Sci. Technol., **68**, (2015), 295.

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