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I. Introduction

The Fusion Nuclear Science Facility (FNSF) is the first strongly nuclear fusion facility in the U.S. pathway to fusion energy development. This facility provides the critical database on ultra-long pulse plasma operation, materials, integrated components, the integrated fusion environment, and operating behavior that is needed to pursue a demonstration fusion power plant (DEMO), and ultimately a commercial fusion power plant. The FNSF bridges the technical parameters from ITER to the DEMO by advancing several missions, such as fusion neutron fluence on the blanket. The pathway from ITER to the first commercial power plant, including the FNSF, is considered a two facility development path, with the FNSF as the first step providing a substantial database to qualify all subsystems for the DEMO, and the DEMO providing both the routine electricity demonstration, and any remaining technical advances not provided by the FNSF. The optimal time frame for the FNSF to begin operation is during the ITER operation, likely after the first DT plasma demonstration, however, there are no commitments to this time frame at present in the U.S. Fig. 1 shows this development pathway schematically. The FNSF will advance several critical parameters toward a fusion power plant, but all of these may not reach the necessary levels, requiring that the DEMO provide the platform for further advancements. At the end of the DEMO program, there can be no further technical gaps remaining, so that a commercial power plant can be pursued by utilities with high technical confidence. The FNSF requires a substantial R&D program as pre-requisite to its design, construction and operation. The arrows in Fig. 1 on the FNSF and DEMO refer to the range of possible mission scopes and the intimate relationship between these devices. The mission scope chosen for the FNSF can be minimal or far-reaching, but whatever the FNSF does not advance, the DEMO must accommodate in its program, before moving to routine power plant operations.

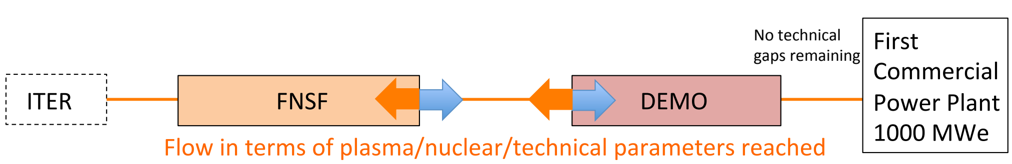


Fig. 1. A schematic view of the fusion energy development pathway, with the two facility assumption of a FNSF followed by a DEMO. The flow from left to right is not time, but rather characterization of plasma, nuclear, and enabling parameters that represent progress toward a commercial power plant.

Shown in Fig. 2 is a timeline with the FNSF identified relative to present or imminent tokamak experiments, ITER, and several FNSF-like or DEMO proposals from various world fusion programs. The arrows on the ends of the proposed facilities indicate that these facility programs have not been defined and so the ends of those programs are unknown. The critical period to establish the plasma physics basis is highlighted between the present tokamak experiments, ITER non-DT and early DT operations, and the DD phase of the FNSF itself. Also identified is the pre-requisite R&D activities prior to operation of the FNSF, some of which will continue in parallel with the FNSF program, and at least partially into the DEMO facility program. The development of predictive simulation capability for all physics and engineering systems in the fusion plant is consistently making progress, as it is the final product of the fusion energy development program and includes simulation codes, databases, design criteria, and a range of physics models.

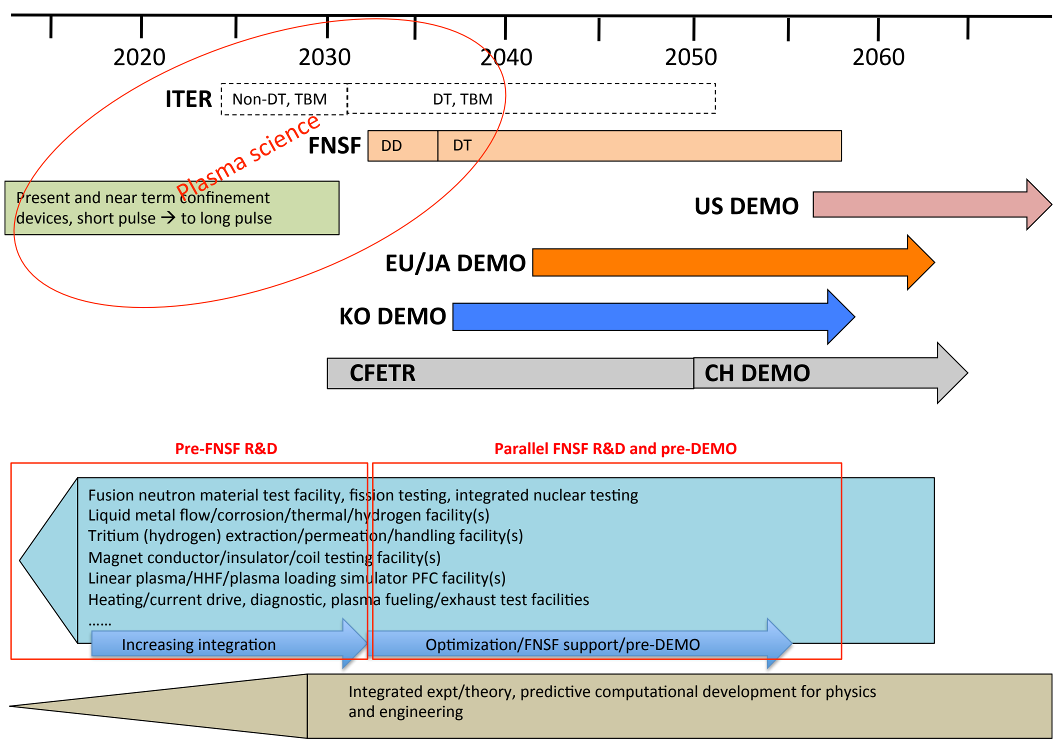


Fig. 2. This diagram provides some perspective on the duration of the FNSF program and how it fits into the landscape of the major components of the international tokamak fusion program. Arrows on the various DEMO proposals indicates that the programs and their durations are not defined. *The timeframe shown for the FNSF is not endorsed or supported in the U.S. at present.* Also shown are ITER, the present tokamak facilities (JET, AUG, DIII-D, C-Mod, NSTX-U,

MAST, KSTAR, EAST, JT-60SA, etc.), the FNSF, U. S. DEMO, and non-confinement R&D facilities are shown.

I.A. Distinguishing the FNSF Mission from the ITER Mission

ITER[1] is taking on several significant technical challenges associated with burning plasma physics and the physics/engineering interfaces associated with that mission. In addition, ITER is at a size that is prototypical of a power plant, utilizes superconducting PF and TF coils, will exercise plasma heating systems approaching the 100 MW level, will experience high divertor heat and particle fluxes, and will take large steps toward a power plant capabilities in tritium handling, cryogenic plant and distribution system, magnet and subsystem power distribution, and activated material handling. The FNSF is intended to pursue the fusion nuclear aspects, and therefore requires sufficiently high performance plasma with very long durations. The neutron fluence at the outboard (OB) first wall over the FNSF's lifetime would reach levels of 15-25 times that reached in ITER, the materials used in the fusion core, including the vacuum vessel, would be different (e.g. reduced activation ferritic steel vs. stainless steel). This is to accommodate the higher neutron exposure and 3-4 times higher operating temperatures for the fusion core components to target conditions for electricity production. The FNSF would breed its tritium for sustainment of the fuel cycle while it is externally supplied to ITER. The plasma pulse durations will need to be 30-1000 times longer than ITER, and the total plasma on-time in a calendar year would need to be 7 times higher. The maintenance approaches for the FNSF will move toward fewer large pieces to facilitate the fusion core component (blanket, divertor) maintenance demonstrations that will achieve rapid replacement, inspection, higher reliability, and overall availability. Table 1 highlights these differences with particular parameter values. The complementarity of the FNSF to ITER cannot be emphasized enough, in spite of the fact that both devices rely on a burning plasma to accomplish their missions.

Table 1. Parameters for ITER, FNSF and a power plant that reflect the significant differences between ITER with burning plasma physics emphasis and a FNSF with ultra long plasma sustainment and fusion nuclear science emphasis.

	ITER	FNSF	Power Plant
Neutron exposure, life of plant OB peak FW, MW-yr/m^2 (dpa)	0.3 (3.0)	8.5-12.5 (85-125)	60-98 (600-980)
Materials	316 SS, 304 SS, 430 SS CuCrZr Be W H_2O	Reduced activation ferritic martensitic (RAFM) steel SiC-composite Borated RAFM Bainitic steel W He	Reduced activation ferritic martensitic (RAFM) steel SiC-composite Borated RAFM Bainitic steel W He
Blanket operating temperature, ^oC	100-150	400-650	600-700
Tritium breeding ratio	~ 0.003	~ 1.0	1.05
Plasma on-time in a year, %	5	35	85
Plasma pulse duration, s	500-3000	$9x10^4$ (1 day) – 10^6 (1 week)	$3x10^7$ (10.5 months)

I.B. The FNSF Will Be Smaller Than the DEMO or Power Plant

The FNSF is intended to be physically smaller than a DEMO, and even than ITER. The primary reason for this additional facility is to reduce overall program cost and risk by allowing a gradual program to break in to the complex fusion nuclear regime at the minimum scale allowed, depending on technology choices. This rationale is balanced on one side by the higher risk of transitioning directly from ITER to a DEMO plant and then to the first commercial power plant. While on the other hand, the maximum of two facilities to the first commercial power plant, provides considerable resistance to very small and low mission scope (non reactor relevant) facilities, since if the FNSF mission scope is too limited it could require an additional FNSF before the DEMO. Fig. 3 shows the fusion power as a function of the plasma major radius for several conventional aspect ratio tokamak DEMO, engineering test reactor, and FNSF proposals. The operating space where solutions are being examined for this study lies inside the red curves, with the lowest radius solutions identified by the large oval. The EU DEMO[2], Korean K-DEMO[3], and Chinese CFETR[4] are shown, as well as the ARC[5] high temperature superconductor design, and the copper coil Fusion Development Facility (FDF).[6] The FNSF operating space displayed is for low temperature superconducting magnets, with the constraints on β_N, q_{div}^{peak}, and $N_w^{OB,peak}$ noted in the figure. The choices for magnet technology and aggressiveness in plasma or other technology features could result in smaller or larger devices. The assumptions used in this study will be described later.

The uncertainty and complexity associated with operating integrated components in the multi-factor fusion environment requires a gradual program that progressively increases the neutron fluence, the temperature, coolant and breeder flows, and pressures. Prior to the FNSF operation, the qualification database for the materials and engineering subsystems inside and including the vacuum vessel will consist of three main components, 1) fusion relevant neutron irradiation of single materials, 2) fission neutron irradiation of individual materials and small partial assemblies (e.g. structure/breeder), and 3) non-nuclear and fully integrated component testing (blanket, divertor, launcher or other). None of these experimental platforms can provide the entire environment, and tend to be missing significant factors in each case.

The fission pressurized water reactor (PWR) and breeder program provides relevant experience on the impacts of a new environment in a nuclear system. Concentrating on material "surprises" as described in Ref. 7, a similar set of unexpected outcomes is likely in the more aggressive fusion neutron environment in combination with high temperature, stresses, hydrogen, flows and corrosion, and associated gradients. In the PWR development experience, it was found that irradiation induced swelling in steels was not observed for temperatures below 302C, but it was observed when above 307C, demonstrating severe sensitivity to operating parameters. Increased neutron irradiation dose rates led to increased steel hardening, while a lower dose rate resulted in earlier onset of swelling, demonstrating that different phenomena respond differently and are not all "co-linear". Small constituents in steels (e.g. 0.5 wt %) could lead to drastically lower ductile crack propagation energy with a shift to higher DBTT and onset of intergranular stress corrosion cracking (IGSCC). In other cases these small differences in metal composition led to an increase in the dpa level (from 2 to 50) before the onset of void formation. This demonstrates both positive and negative features observed for small changes in material constituents. Surface conditions, welds, and metallurgical variability (different heats) in components introduced a constant source of variable behavior in the neutron environment. Incubation periods before the onset of material phenomena were common, and the presence of multiple gradients (e.g. neutron fluence, temperature, and stress) severely complicated the material responses. These examples only involve material effects, while thermo-mechanics, thermal hydraulics, mass transfer, and fluid MHD can provide additional complications. In addition, the number of blanket sectors, divertor sectors, and other components (H/CD apparatus, test blanket modules, In light of these observations it is prudent to pursue a more gradual introduction to the fusion nuclear regime with a smaller size facility. The goal for the FNSF is to establish a database on materials and components in the facility up to relevant parameters (e.g. fluence reaching 40-75 dpa, blanket temperatures reaching 500-600C) before proceeding to larger size in the DEMO.

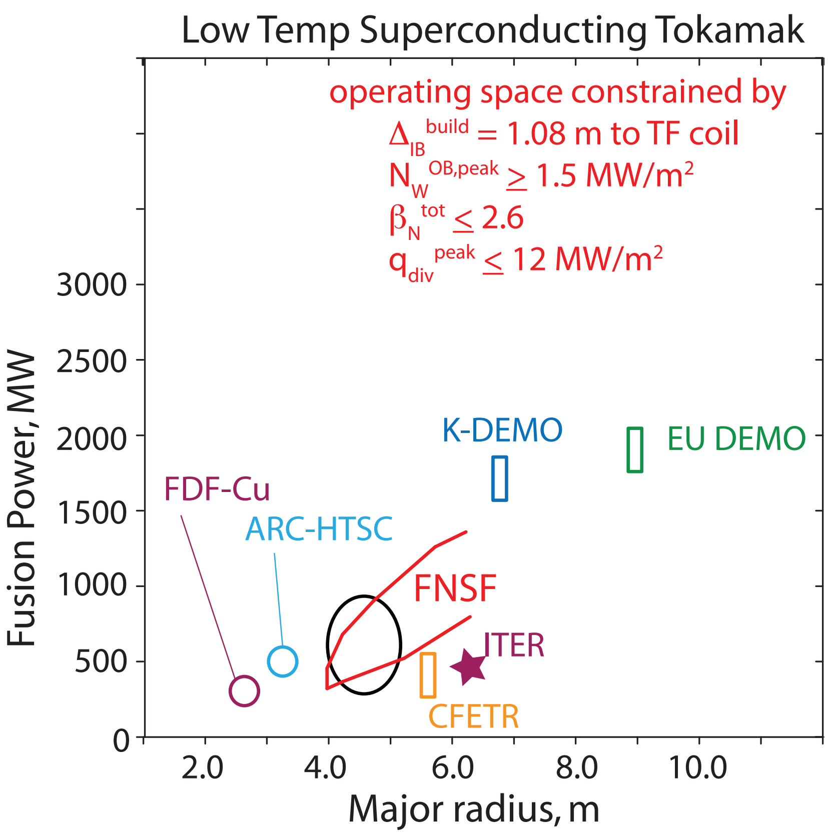


Fig. 3. The fusion power versus the facility major radius for a range of proposed DEMO and FNSF-type facilities. The operating space identified for the moderate FNSF to be studied in this activity is enclosed inside the red contours, and the black oval that identifies the approximate operating point region. The FNSF major radius would be smaller than the EU-DEMO, K-DEMO, ITER, and CFETR.

II. Missions and Metrics for the FNSF

As noted previously, the purpose of the FNSF is to advance the integrated systems in the integrated environment by moving toward power plant relevant parameters. These “advances” can be characterized by missions, and given detail by providing metrics that measure the mission progress. Appendix 1 contains a table of the missions and several metrics being considered for each mission. The missions identified for the FNSF are,

1. Strongly advance the fusion neutron exposure of all fusion core (and ex-core) components towards the power plant level
2. Utilize and advance power plant relevant materials in terms of radiation resistance, low activation, operating temperature range, chemical compatibility and plasma material damage resistance
3. Operate in power plant relevant fusion core environmental conditions including temperatures, coolant/breeder flow rates, pressures/stresses, hydrogen (tritium), B-field, and neutrons, and with gradients in all quantities.
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, requiring a high level of inventory prediction, control, and accountancy.
6. Routinely operate very long plasma durations, much longer than core plasma time constants and long enough for nuclear, chemical, and PMI processes to be accessible, at sufficient plasma performance to advance the fusion nuclear mission, generally considered to be days to weeks.
7. Advance and demonstrate enabling technologies that support the very long duration plasma operations with sufficient performance and reliability to project to DEMO and a power plant, including heating and current drive, fueling/pumping, particle control, PFC lifetime, disruption avoidance and mitigation, plasma transient mitigation, feedback control, diagnostics, etc.
8. Demonstrate safe and environmentally friendly plant operations, in particular with respect to tritium leakage, hot cell operation, onsite radioactive material processing and storage, no need for evacuation plan and other regulatory aspects.
9. Develop power plant relevant subsystems for robust and high efficiency operation, including heating and current drive, pumps, heat exchanger, fluid purity control, cryo-plant, etc.
10. Advance toward high availability, including gains in subsystem and component reliability, progress in capabilities and efficiency of remote maintenance operations, accumulation of reliability and failure rate data that can be used to project and design future systems.

Each of these missions is characterized quantitatively by several metrics for determining how far they advance toward a power plant. The FNSF would be expected to advance most or all missions significantly, however, any remaining shortfalls would have to be accommodated in the DEMO facility before it could move to routine power plant operations.

There are a number of metrics used to quantify the advance in any of the missions and these values are identified for ITER, the FNSF, DEMO and a power plant. Here we list only a few examples for each mission described above while complete (and developing) lists can be found in Appendix 1,

1. Peak first wall fluence before replacing the blanket (MW-yr/m^2, dpa)
   Peak first wall neutron wall load (MW/m^2)

2. First wall (FW) / blanket structural material
   Vacuum vessel (VV) coolant

3. Temperatures, flow velocities, pressures; $T_{str,blnkt}$, $T_{LiPb,blnkt}$, $T_{He,blnkt}$, $T_{str,VV}$, $T_{He,VV}$, $T_{W,div}$, $T_{He,}$div, v_{LiPb}, $P_{He,blnkt}$, $P_{He,div}$, etc.

4. Tritium breeding ratio (TBR) global
   Fraction of Li6 in breeder
   Fraction of FW area unavailable for breeding

5. Tritium extraction efficiency
   Tritium inventory in breeder

6. Plasma on-time per year
   Plasma pulse duration

7. Total heating and current drive (H/CD) power
   Heating and current drive (H/CD) source operating lifetime

8. Total plant tritium leakage, Ci/yr
   LOCA $T_{FW,max}$

9. Plasma fusion gain
   Engineering gain
   H/CD wall-plug efficiency

10. Single sector replacement time
    Yearly plant availability

How far any of these metrics are advanced toward power plant values in the FNSF depends on the mission scope, i.e. the total sum of missions taken on in the FNSF. We can roughly characterize the mission scopes as minimal, moderate and maximal here to show the range of possible configurations.

A minimal FNSF could be characterized as the smallest facility that will produce sufficient neutrons for blanket testing, and would largely bypass power plant relevance wherever possible. For example, copper magnets would be used for the TF and PF/CS magnets. The tritium breeding ratio would not reach 1.0, and tritium would need to be purchased for operations. The peak dpa (or MW-yr/m^2) at the OB first wall over the plant life could be limited to < 30 dpa, and conventional materials such as stainless steel might be used for the vacuum vessel. Maintenance schemes with limited potential for a power plant would be considered in order to meet the immediate facility needs. This facility may operate for ~ 15 years to accomplish its mission scope. This mission scope may be incompatible with a two facility step to commercial fusion power.

At the other end of the spectrum a maximal FNSF would have power plant relevance appear in virtually all aspects of the facility, including some net electricity demonstration. Superconducting TF and PF/CS coils would be used. Tritium breeding exceeding 1.0 would be required, and perhaps with sufficient margin to provide tritium for the next facility. The peak dpa at the OB first wall over the plant life could be set to match the power plant level of say 100 dpa. The facility would be the larger of possible FNSF configurations, with engineering gains ($P_{elec,gross}/P_{recir}$) of 1.0 or larger. This requires a full balance of plant including turbines for electricity generation, and high efficiency subsystems. The maintenance scheme would be power plant relevant, horizontal or vertical large sector replacement. Such an FNSF would likely leave no required technical demonstrations for the DEMO, so that the DEMO could pursue a strictly routine power plant program to provide electricity to the grid. This facility could operate for ~ 35 years to accomplish all of its mission scope.

The moderate FNSF falls somewhere in between these two limiting cases and is the mission scope being considered in this study. The TF and PF/CS coils would be considered either superconducting or Cu. The TBR is targeted to be 1.0, although trade-offs are expected in first wall hole area, and design of the blanket, which could lead to small shortfalls or slight over-breeding. Materials are taken to be power plant relevant out to and including the vacuum vessel. It is desired to reach maximum dpa levels of > 40 on the first wall. The use of water inside the vacuum vessel will likely be rejected in this mission scope to operate at power plant relevant temperatures, even though generation of electricity may not be a primary goal, and net electricity may not be possible. This facility would operate for ~ 25 years to accomplish its mission scope. Table 2 provides some approximate parameters for these three mission scopes to provide some perspective on the facility capabilities. These mission scopes will be examined in the systems analysis in the next year of the FNSF activity.

Table 2. Selected parameters to describe 3 possible FNSF mission scopes, with the advance toward power plant parameters lower for the minimal mission scope and high for the maximal mission scope.

	Minimal	Moderate	Maximal	Power plant
Blanket type, temperature (T_{LiPb}^{outlet})	DCLL 400C	DCLL 400-600C	DCLL 400-600C	DCLL 650C
Structural Material	RAFM	RAFM→RAFM-nano	RAFM→RAFM-nano	RAFM-ODS/nano
$N_W^{OB,peak}$, MW/m^2	1.0	1.5	2.25	
Plant life peak dpa at OB FW	32	88	202	840
Max dpa on OB FW (dpa to replacement)	5-20	10-40	10-70	150-200*
Q_{engr}	<<1	$\leq$ 1	> 1	4
TBR	$\leq$ 1	~ 1	> 1	1.05
VV material	SS	Bainitic	Bainitic	Bainitic

R, m	small	medium	large	9.75
Divertor materials	W / CuCrZr	W / W-alloy	W / W-alloy	W/W-alloy
Divertor coolant	H_2O	He	He	He
Plant lifetime	~15 yr	~25 yr	~35 yr	47 yr (40 FPY)
TF/PF conductor	Cu	LTSC or Cu	LTSC or HTSC	LTSC or HTSC
Plasma on-time per year	10-35%	10-35%	10-45%	85%
Plasma duty cycle	0.33-0.95	0.33-0.95	0.33-0.95	1.0
Maintenance	Full toroidal vertical	Horiz or vert	Horiz or vert	Horiz**

*this is being revisited, and likely to drop to 100-150 dpa
**ARIES designs use horizontal maintenance

III. The FNSF Program, What is Done on the Facility

In order to better understand what the FNSF must accomplish, a program has been identified with a series of phases and estimated timeframes. The moderate FNSF mission scope is assumed here. Shown in Table 3 is the program with a He/H phase for startup and shakedown of various plant systems, followed by a DD phase with the primary mission of ultra-long plasma pulse length demonstrations. This is followed in the nominal program by four DT phases, with increasing plasma pulse length and duty cycle, resulting in an increasing neutron fluence (peak at OB FW reaching 7, 19, 26, and 37 dpa), and FW/blanket/shld, divertor, and special PFC (launchers) evolution to higher performance parameters. The accumulated neutron fluence results in a plant lifetime peak fluence of about 88 dpa. The peak neutron wall load is taken to be 1.5 MW/m^2, which is found appropriate from systems analysis. The primary blanket concept assumed is the Dual Coolant Lead Lithium (DCLL) design due to its projected favorable power plant performance and perceived near term development. This design has an reduced activation ferritic martensitic (RAFM) steel structural material, since there are no viable alternatives at present. Higher performance blanket upgrades include advancing the RAFM steel (e.g. EUROFER, F82H) to an oxide dispersion strengthened (ODS) RAFM and to a nano-structured RAFM. Simultaneously, the operating blanket temperatures are increased from a low LiPb outlet temperature of 400C up to 650C, which demonstrates the level needed for high thermal conversion efficiency. Significant maintenance time has been allocated to each session, which includes activities during plasma operations, at the end of each session when sectors are pulled out for autopsy, and finally when the phase ends and all sectors are removed and replaced for the next phase. Maintenance time will also include contingency for unscheduled failures.

The blanket testing strategy is prescribed by sectors, and an example is shown in Table 4 for the Phase 3 of the program. Each sector is defined by 1) blanket type, 2) structural material and operating temperature (T_{LiPb}^{outlet}), 3) whether it will be pulled for autopsy during the phase or left in for the entire phase, 4) whether it has a plasma heating and current drive (or other) penetration, 5) whether it has a test blanket module (TBM) penetration and what is being tested, and 6) whether there is a material test module(s) in the sector. In general the TBM is testing the blanket upgrade for the next phase, conducting engineering scaling studies, or a backup blanket concept. An entire sector can also be testing a backup blanket concept. The backup blanket concepts, helium cooled lead lithium[8] (HCLL) and helium cooled ceramic breeder or pebble bed[9] (HCCB or HCPB), were chosen based on common features and anticipated weaknesses in the

DCLL concept, namely liquid metal MHD and liquid metal interaction issues. Since the RAFM family of steels is the only qualified structural material at present, there are no alternatives, and this is the same in the backups. In addition, for power plant relevance, safety, thermal conversion efficiency, and material compatibility, water has been disallowed for use in the fusion core (inside and including the vacuum vessel). Since few other coolants are considered viable alternatives, the backup blankets also use helium like the main DCLL blanket concept. This is also motivated by the realization that testing many blanket concepts with varying structural materials, breeders, coolant and functional materials is not an effective use of resources when considering the time and R&D investment just to advance a single blanket concept toward power plant parameters.

An additional 7-year DT phase that reaches a peak damage at the OB first wall of 37 dpa is being accommodated in the FNSF program in the event of successful or unsuccessful operations. If the program is executed successfully, sectors from the phase 6 could be left in for the phase 7 operation, advancing the peak neutron fluence and damage levels further to a maximum of 74 dpa. On the other hand, if the blankets (or other components) are performing poorly, either a backup or a re-designed blanket can be tested to the full 37 dpa in phase 7. The incremental increase in shielding required to maintain lifetime components under their limits is a few centimeters.

The years of operation are nominally 31.5 including the extra phase. Some additional time between phases might be required when all sectors are typically replaced. This includes ~8.4 years of DT plasma on-time (neutron producing) operations. The maintenance time requirements in later phases may be reduced as the procedures become more optimized. The organization of the maintenance time within a phase must be optimized to provide the needed time during plasma operations and required time for various scheduled and unscheduled maintenance activities. The neutron wall loading may be increased by operating at a higher plasma beta, as discussed in the Sec. IV.A, and this can accelerate the neutron exposure and shorten facility operational times. Although not discussed here, activities associated with divertor and special PFCs (launchers) optimization will also be taking place on the facility. A tentative DEMO program has also been defined in order to demonstrate the connection between these two facilities. In particular, the advance of fusion neutron exposure (damage) in the moderate FNSF up to ~ 40 dpa, requires that the DEMO spend some period in its early stages to increase the exposure up to anticipated power plant levels before initiating power plant operations. A program for the DEMO is given in Appendix 2. Better definition of the DEMO requirements and its program is needed and will be determined with systems analysis in the next year.

Table 3. The tentative program on the FNSF, identifying phases, their durations, and progress in parameter achievements.

	He/H		DD		DT		DT		DT		DT	DT
	Plasma physics				Low Fluence Fusion Nuclear Break-in				High Fluence Fusion Nuclear Operation			
Phase	1		2		3		4		5		6	7
Phase time, years	1.5	1	2-3	1	3	1	5	1	5	1	7	7

Cumulative operation time, years	1.5	3.5-4.5	6.5-7.5	11.5-12.5	16.5-17.5	23.5-24.5	30.5-31.5
N_w^{peak}, MW/m^2		~0.009	1.5	1.5	1.5	1.5	1.5
Plasma on-time per year (days)	10-25% (37-91)	10-50% (37-183)	10-15% (37-55)	25% (91)	35% (128)	35% (128)	35% (128)
Plasma duty cycle (days on/days off)		0.33-0.95 1/2 – 10/0.5	0.33 1/2	0.67 2/1	0.91 5/0.5	0.95 10/0.5	0.95 10/0.5
Operation / Maintenance per year (days)			111-165/254-200	137 / 228	141 / 224	135 / 230	135/230
End of Phase Peak Fluence ($MW\text{-}yr/m^2$)			0.45-0.68	1.88	2.63	3.68	3.68
Cumulative peak fluence, $MW\text{-}yr/m^2$			0.45-0.68	2.33-2.56	4.96-5.19	8.64-8.87	12.3-12.6
End of Phase Peak damage (dpa)			4.5-6.8	18.8	26.3	36.8	36.8
Cumulative Peak damage (dpa)			4.5-6.8	23.3-25.6	49.6-51.9	86.4-88.7	123-126
Total # plasma cycles			111-165	230	130	91	91
DCLL	RAFM	RAFM	RAFM	RAFM-ODS	RAFM-nano	RAFM-nano	
DCLL T_{LiPb}^{out}, ^oC	400C	400C	400C	500C	600C	650C	
DCLL TBM port	-	-	RAFM-ODS	RAFM-nano	RAFM-nano	-	

					500C		600C		650C			

Table 4. Sample of the detailed blanket testing process in Phase 3 of the FNSF program, identifying by sector, the blanket type and its testing timeline and functionality.

	Phase 3-A (year 1)	Phase 3-B (year 2)	Phase 3-C (year 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	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	DCLL 400C RAFM / HCCB	DCLL 400C RAFM / HCCB	DCLL 400C RAFM / HCCB
S-16	HCLL 400C RAFM	HCLL 400C RAFM	HCLL 400C RAFM

R1 = sector removed after first year for autopsy (blue)
R2 = sector removed after second year for autopsy (blue)
LH = lower hybrid launcher in sector (green)

NB = neutral beam in sector (green)
IC = Ion cyclotron in sector (green)
TBM = test blanket module in sector
Backup blankets are HCLL and HCCB (purple)

IV. R&D Activities in Preparation for the FNSF

A significant research and development program must precede the FNSF. The philosophy for this study is to prevent failures on the facility to the maximum extent possible. This implies a testing program that obtains thorough qualification of all facility components, fusion core in particular, are carried out in advance of installation on the facility, to the extent possible. It is not credible to operate a plasma-vacuum device under frequent failure conditions, since removing components, repairing, cleaning, and reassembling the fusion core is extremely time consuming, and will significantly compromise the fusion nuclear science mission, regardless of the maintenance approach. This is the primary reason that the FNSF program is defined as gradually as it is. The R&D activities can be described broadly in 5 major categories, 1) fusion neutrons, 2) tritium science, 3) liquid metal science (for the DCLL and other LM concepts), 4) plasma material interactions and plasma facing components, and 5) enabling technologies (heating and current drive, magnets, fueling, vacuum pumping systems, diagnostics, feedback control, remote maintenance, and balance of plant). This is shown schematically in Fig. 5, indicating a progression from single to few effects, partial integration, and finally maximum integration experiments that will be required.

Each of the topical areas can be subdivided into more specific activities. For example the fusion neutron area would include fusion relevant neutron source exposure of the many single materials in a fusion core at varying temperatures (SNS[10], FAFNIR[11], IFMIF[12] or other). It would include non-nuclear characterization of the materials, and fission neutron exposure data as well. It may be possible to integrate two materials or put samples under stress (or other conditions) depending on the available volume, which is more difficult in fusion and more likely in fission spectrum facilities. This area cannot be integrated with others and largely provides a database on individual materials under specific conditions.

The tritium science area, shown in Fig. 5, breaks into plasma tritium implantation/permeation/retention, behavior in materials (and multi-materials), extraction from LiPb, and breeder/structure tritium extraction in a fission integrated experiment. Specific experimental facilities and activities are identified to examine these issues. This would likely include deuterium as a surrogate where possible, or tritium where necessary. This area merges into an integrated blanket testing experiment in later years, which would likely be with deuterium surrogate.

The liquid metal science area breaks into primary topics of liquid metal MHD and heat transfer, MHD flow effects on corrosion and re-deposition (mass transfer), flow channels inserts and their interactions, tritium in the liquid metal and constituency control. Specific experimental facilities (including present, upgrades and new) and activities can be identified to examine these issues.

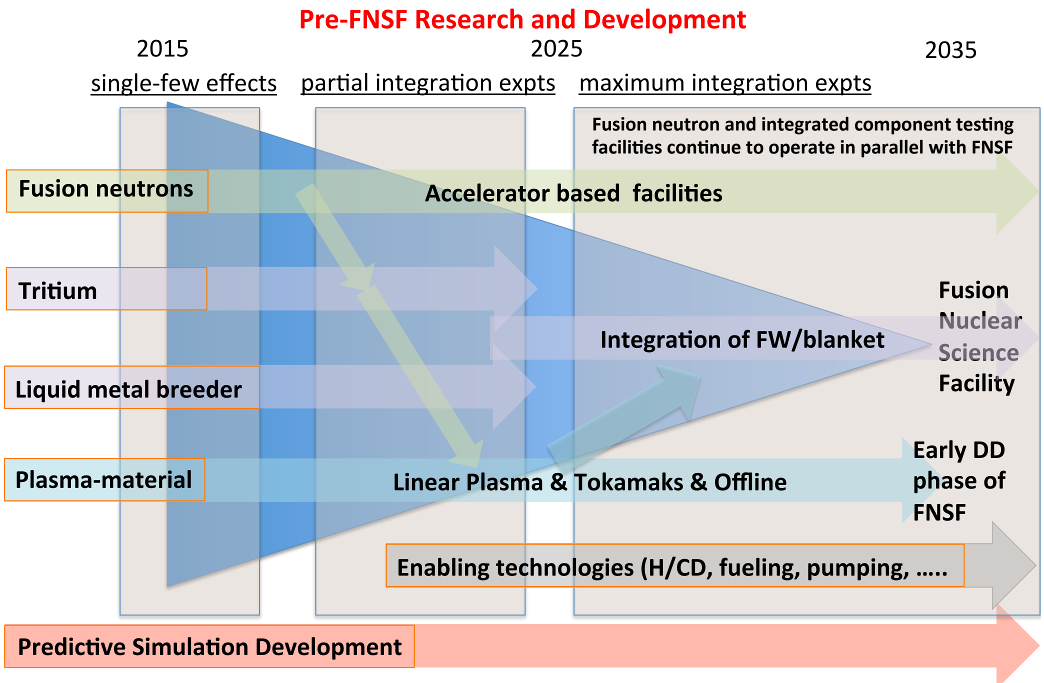
The PMI and PFC area requires an interactive program between tokamak experiments, scrape-off layer plasma/atomic physics simulation, linear plasma simulators, high heat flux facilities, and design integration. This also implies a critical multi-discipline cooperation among physicists, material scientists, and engineers, in order to address the divertor, first wall, and special PFCs. Primary near term thrusts[13] should include 1) significant initiative on expanding SOL and PFC measurements in tokamaks, 2) aggressive programs to eliminate or ameliorate ELMs and disruptions, 3) examination of advanced magnetic configurations, and 4) develop theory and computational tools for SOL physics, divertor physics, PMI, neutral transport and atomic/molecular processes. Linear plasma devices should be upgraded to provide platforms for FNSF loading conditions, establishment of tungsten materials properties and development of tungsten materials for the fusion plasma and nuclear environment. Tungsten divertor and tungsten/RAFM concepts should be tested for high heat flux capability based on relevant design approaches. Finally, the development of RF launchers and viable diagnostics for the FNSF environment is needed.

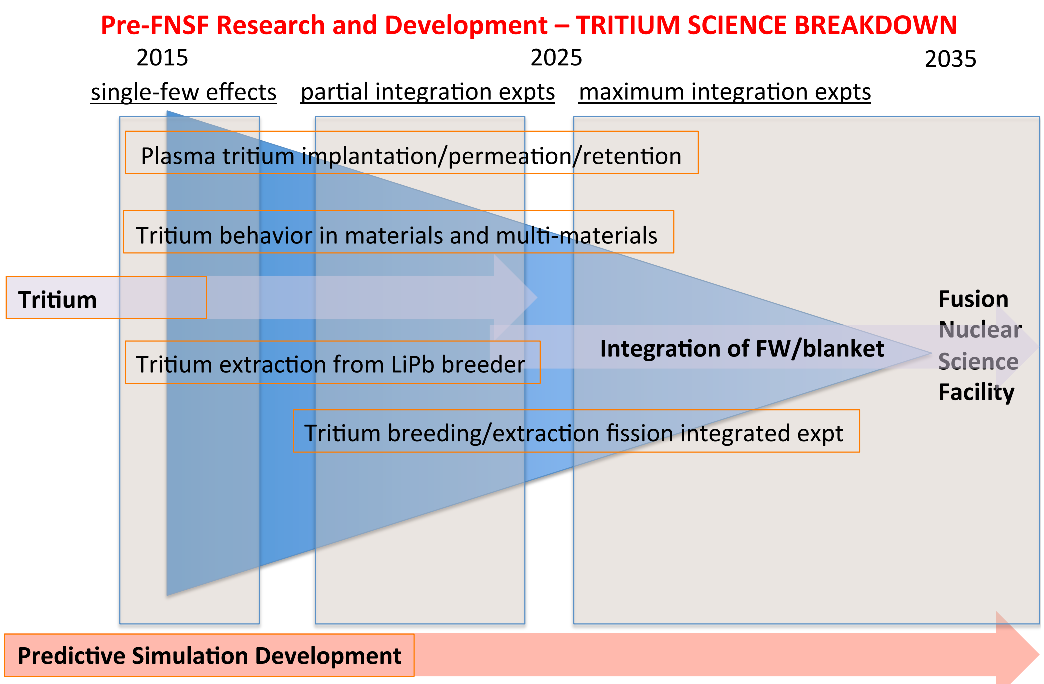
Enabling technologies is a broad category including heating and current drive, fueling and pumping, magnets, diagnostics, maintenance, and balance of plant components (e.g. heat exchanger, tritium extraction, turbines). All these subsystems in the fusion core must be advanced to use fusion relevant materials, extremely long plasma duration, high efficiency and reliability, and long lifetime in the neutron and plasma environments. These issues are described in Ref (14).

Shown in Fig. 5, is the pre-FNSF R&D program as part of the larger pathway, indicating how these thrusts persist into the FNSF program, and in some cases continue into the DEMO program, such as enabling technology and fusion neutrons. For example, the fusion relevant neutron irradiation of materials and fully integrated non-nuclear blanket testing is expected to continue in parallel with the FNSF to provide the qualification of components before installation during the various DT phases where the blanket materials and operating parameters are advanced. The enabling technologies area is expected to continue into the DEMO operation because of the importance of high efficiency and high reliability of components at this stage, as well as the need for balance of plant components that may not be fully developed in the FNSF.

IV.A. Plasma Requirements and the Plasma Strategy for the FNSF

The approach to the physics operating point(s) in the FNSF is to pursue conservative parameters, while allowing higher performance with clearly defined hardware or operation that can support it, should it be possible. The plasma current is targeted to be 100% non-inductive ($f_{NI} = 1.0$) to provide very long uninterrupted plasma operation, with acombination of bootstrap and externally driven currents. It may be possible to support very high non-inductive current fraction plasmas ($f_{NI} > 0.85$) for long durations (several hours) with a relatively small central solenoid if there is some robustness to be gained in the operating space.


Pre-FNSF Research and Development
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
Fusion Nuclear Science Facility
Integration of FW/blanket
Liquid metal breeder
Early DD phase of FNSF
Plasma-material
Linear Plasma & Tokamaks & Offline
Enabling technologies (H/CD, fueling, pumping, ...
Predictive Simulation Development


Pre-FNSF Research and Development – TRITIUM SCIENCE BREAKDOWN
2015
2025
2035
single-few effects
partial integration expts
maximum integration expts
Plasma tritium implantation/permeation/retention
Tritium behavior in materials and multi-materials
Tritium
Fusion Nuclear Science Facility
Integration of FW/blanket
Tritium extraction from LiPb breeder
Tritium breeding/extraction fission integrated expt
Predictive Simulation Development

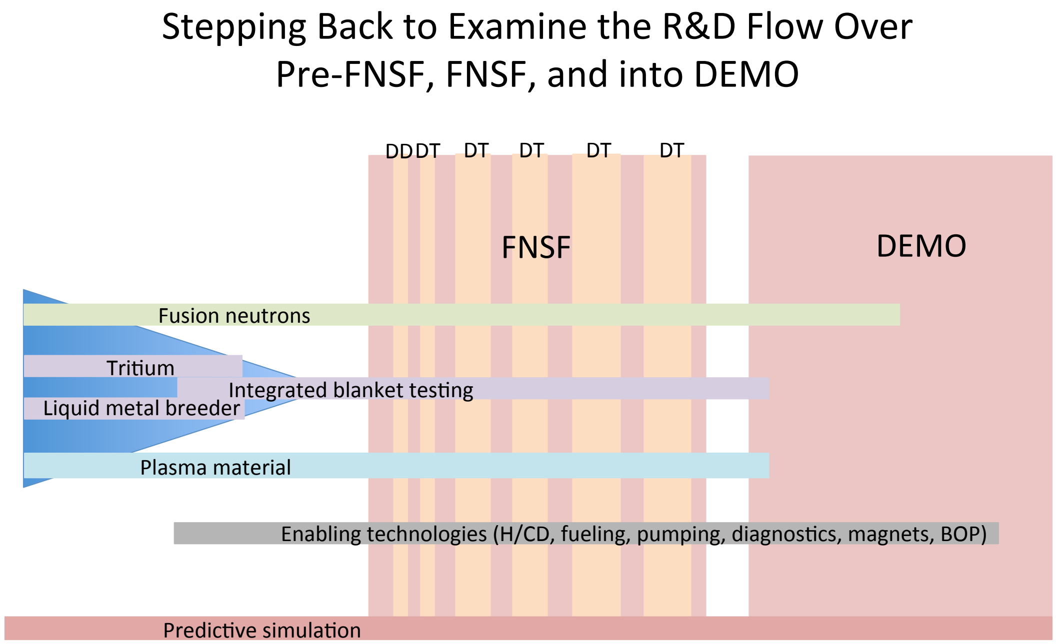


Fig. 5. Diagram showing the primary thrust areas for fusion nuclear material science over time, progressing from single to few effects, to partial integration and finally to full integration experiments prior to the FNSF. This diagram can be expanded in detail for each thrust area, and the R&D activities can be viewed across the time frame of the FNSF and DEMO.

The β_N^{total} ($\beta_N^{th}+\beta_N^{fast}$) is at or below the no wall beta limit, here defined to be ~ 2.5. This is based on ideal MHD analysis for the ARIES-ACT2 study[15], where a range of current profiles from bootstrap, lower hybrid, neutral beam, and ICRF fast wave were examined. Shown in Fig. 6 is the stable β_N versus $l_i(1)$ (internal self-inductance, high values correspond to peaked current profiles, and low values to broad current profiles in the plasma) without and with wall stabilization (which requires feedback, rotation and/or kinetic stabilization). Without wall stabilization the maximum β_N is 2.5, and decreases with decreasing l_i, while with wall stabilization at b/a = 0.55 (b = distance to wall measured from OB plasma boundary, a = minor radius) allows β_N to rise to 2.8-3.3 as l_i varies from 0.85-0.65. Therefore, an increased value of β_N up to 3.25 will be examined for improved performance and hardware requirements for this stabilization identified, while the baseline design will be made with the assumption of $\beta_N \leq 2.5$. The requirements for resistive wall mode (RWM) feedback (and error correction) coils located outside the shield on the OB side will be examined, along with plasma rotation or kinetic stabilization requirements.

The plasma density relative to the Greenwald density limit ($n_{Gr} = I_P/\pi a^2$) is often found to approach or exceed 1.0 when pursuing burning plasma or power plant configurations. Tokamak experiments[16-18] have demonstrated ratios exceeding 1.0 while maintaining reasonable energy confinement in the plasma ($H_{98} \leq 1$). These regimes are facilitated by pellet injection fueling, strong plasma shaping, and careful control of gas injection, recycling locations, and pumping. In general plasma solutions are sought with the lowest density ratio, however this tends to make the global energy confinement requirement higher (higher H_{98}).

The plasma shaping is strong with an elongation of $\kappa_x = 2.2$, and triangularity of $\delta_x \sim 0.6$. The double null (DN) configuration is used to enhance the beta limits (no wall and with wall), to accommodate the close-by x-point that comes with strong shaping, and provide some reduction of the power to the divertor. The stabilizing conductor[19] for the elongated plasma is made of tungsten and located at b/a = 0.33, with poloidal extent from about 45-90^o on the OB side, measured from the plasma major radius. This puts the conductor in the middle of the breeding blanket. An elongation of 2.0 would allow the conductor to move to about b/a = 0.4, which still would be located in the breeder zone. Conductor shells are also located on the IB side. Feedback control coils are made of inorganic insulated Cu and located behind the shield/structural ring on the OB side, but inside the vacuum vessel.

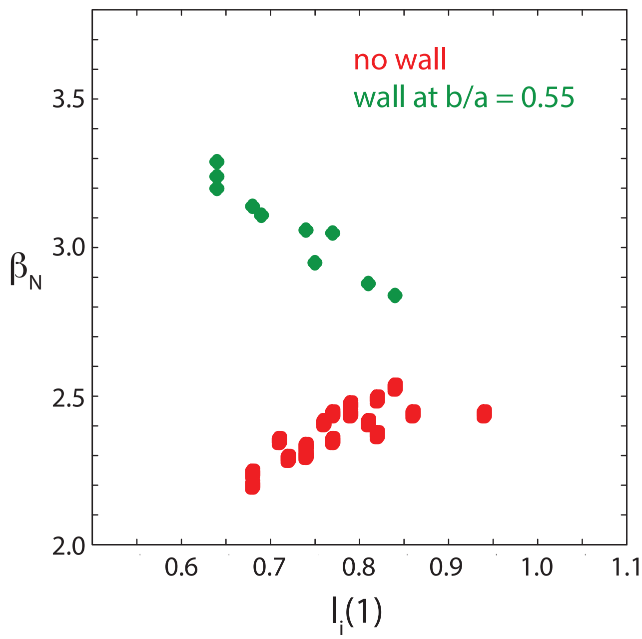


Fig. 6. The maximum stable β_N as a function of the current profile peaking (internal self-inductance), with red points indicating without wall stabilization, and green showing with wall stabilization b/a = 0.55, from the ARIES-ACT2 study. The highest without wall β_N is ~ 2.5, which can be increased to 3.0-3.25 with stabilization.

Although the divertor heat flux is a plasma-engineering interface parameter, it provides a significant constraint on the allowed plasma configurations. Here a heat flux is calculated by using a formulation for the power scrape-off width from Fundamenski.[20] The ratio of scrape-off layer power to the major radius is also calculated. The maximum value for the heat flux is set to be ≤ 10 MW/m^2, since He cooled designs[21] have been identified as being capable of peak heat fluxes ≤ 15 MW/m^2 with acceptable pumping powers. There is considerable uncertainty in the prediction of the power scrape-off width, however, a formula is used to provide some actual constraint on both plasma and engineering operating space. The target is to operate in a partial or full detachment regime[22] with an ITER-like or slot type divertor, and in the systems analysis 90% of the power entering the divertor is assumed to be radiated. Advanced divertor configurations, such as the X-divertor[23] or snowflake[24], will be examined to quantify their potential benefits. The divertor material is taken to be tungsten armor on a tungsten structure, with the tungsten structural material requiring better definition.

The heating and current drive systems demonstrated on tokamaks will be examined, including NB, LH, EC, ICRF, and high frequency ICRF (helicon). For initial systems studies, the current drive efficiency will be taken to be $\eta_{CD(20)}$ (n_{20}RI/P) = 0.2 A/W-m^2. Compared to recent ARIES-ACT2 studies[13] this is conservative, $\eta_{CD(20)}$ = 0.26 (ICRF/FW), 0.35 (NINB), 0.25 (LH), 0.16 (EC). The wall plug efficiency used to calculate the electricity required is taken to be 0.4 for all sources. The ITER projections[25] for wall plug efficiencies are 0.35-0.44 for EC, 0.48 for ICRF ignoring coupling losses, and 0.32 for NB or up to 0.53 including advances beyond ITER. For LH the wall-plug efficiency is estimated to be 0.5 ignoring coupling to the plasma[26].

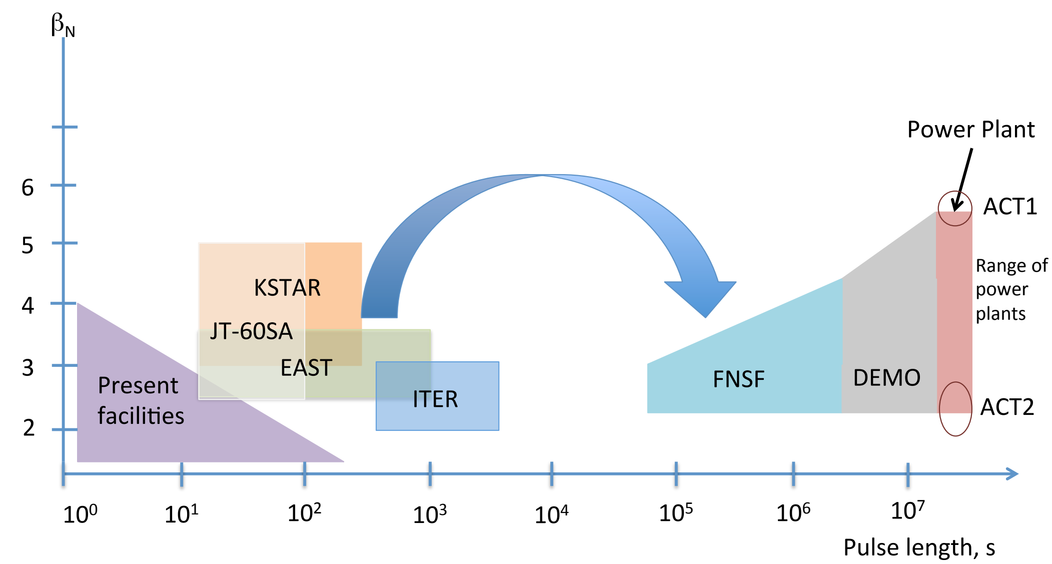


Fig. 7. A diagram showing approximately the plasma performance in terms of β_N versus the plasma pulse duration. Those for present facilities are achieved, while those for KSTAR, EAST, JT-60SA and ITER are targeted. The significant gap left by these facilities before the FNSF shows the critical need for the combination of linear plasma simulators, tokamak experiments, high heat flux facilities, predictive simulation, and the DD phase of the FNSF.

The plasma duration presents a significant challenge, since the target is days to weeks for a plasma pulse, while tokamaks have demonstrated a maximum of ~30 s for high performance plasmas. The tremendous increase in plasma duration required for the FNSF is demonstrated in Fig. 7, with present and anticipated tokamaks, and the significant gap to the FNSF with 1 day and 2 weeks pulse lengths.

The best demonstrations of long duration and high plasma performance, with high non-inductive current fraction are from DIII-D and JT-60U. The longest time scale for the core plasma is the current diffusion time, $\tau_{CR} = \mu_o a^2\kappa/12<\eta_{neo}>$, where $<\eta_{neo}>$ is the volume average neoclassical resistivity, and the longest tokamak discharges relative to this are ~ 15 τ_{CR} in JT-60U.[27-29] However, these longest pulses are not in plasmas with 100% non-inductive current, or the high q_{95} values expected, or the high densities relative to Greenwald, however, they do achieve sufficient β_N ~ 2.6, H_{98} ~ 1.0, n/n_{Gr} ~ 0.55, and f_{BS} ~ 0.43. These discharges avoided neo-classical tearing modes (NTMs) by operating at low q_{95} ~ 3.2, where the potentially unstable rational magnetic surfaces (3,2) and (2,1) were separated from the dominant pressure gradient. Utilizing the vacuum vessel and plasma rotation the β_N was increased above the no-wall beta limit to 3.0 and sustained for 3 τ_{CR}, with f_{BS}, f_{NI} rising to 0.5 and 0.85, respectively. RWMs were observed in these discharges. Plasmas with β_N ~ 2.4, H_{98} ~ 1.0, f_{BS} ~ 0.45, f_{NI} > 90%, and minimum safety factor q_{min} ~ 1.5 were maintained for 2.8 τ_{CR}. Using reversed shear plasmas, f_{NI} reached 1.0, with f_{BS} ~ 0.8, H_{98} = 1.7, q_{95} ~ 8 and β_N ~ 1.7, and was sustained for 2.7 τ_{CR}. Neither of these high f_{NI} plasmas experienced NTMs, presumably due to high safety factors and sufficiently low beta. JT-60U also demonstrated operation at high densities, with n/n_{Gr} ranging from 0.7-1.1, H_{98} values from 0.85-1.1, in reverse shear and high poloidal beta discharges. These utilized high field side pellet injection and impurity seeding, obtaining up to β_N ~ 2.1.

DIII-D has obtained β_N ~ 3.1-3.4, H_{98} > 1.2-1.3, q_{95} = 5.0-5.5, f_{BS} ~ 0.6, f_{NI} ~ 0.8-1.0 and sustained them for $\leq$ 1 τ_{CR}.[30] More recently[31,32] with off-axis neutral beam injection plasmas have reached β_N ~ 3.5, H_{98} > 1.0, q_{95} = 6.7, f_{BS} ~ 0.4-0.5, f_{NI} ~ 0.75 for 2 τ_{CR}. These later discharges with off-axis NBs were not terminated by NTMs while earlier steady state plasmas often were. Notably DIII-D has created plasmas with β_N ~ 2.0, H_{98} = 1.3, q_{95} = 4.6 in the QH-mode with no ELMs, for 2 τ_{CR}. DIII-D routinely takes advantage of error field correction, and some plasma rotation to operate above the no wall beta limit. They have determined that low plasma rotations are acceptable with wall stabilization due to kinetic stabilization mechanisms. DIII-D has also demonstrated stationary hybrid scenarios with f_{BS} ~ 0.4, that were sustained for 6 τ_{CR}, however these discharges have a significant inductive current fraction. It is of interest to explore very high non-inductive (or fully non-inductive) fraction hybrid discharges for their viability for FNSF.

V. Preliminary Systems Analysis for FNSF Operating Point Identification

Systems analysis is used to identify interesting operating plasma points that satisfy engineering constraints. This type of analysis uses 0D plasma power and particle balance, and a series of simple engineering models for heat flux, power balance components, TF coil, bucking cylinder and PF/CS coils. The inboard build is provided by using the neutronic radial build derived for the Pilot Plant studies[33] properly scaled for the FNSF inboard fluence. The inboard radial build is 0.88 m of first wall, blanket, shield, and vacuum vessel, with an additional 0.2 m added for gaps. The inboard SOL thickness is 0.1 m. The TF and PF/CS coils have an overall (SC, insulator, helium, Cu, conduit and structure) current density of 15 MA/m^2, and the peak field at the TF coil is restricted to be $\leq$ 15.5 T. This maximum field with Nb_3Sn is being pursued by K-DEMO[3] and for the next step large hadron collider accelerator. The peak heat flux in the divertor is determined using a formulation for the SOL power width from Ref (20). It is also assumed that 90% of the SOL power is radiated in the divertor in a partially or fully detached regime.[21]

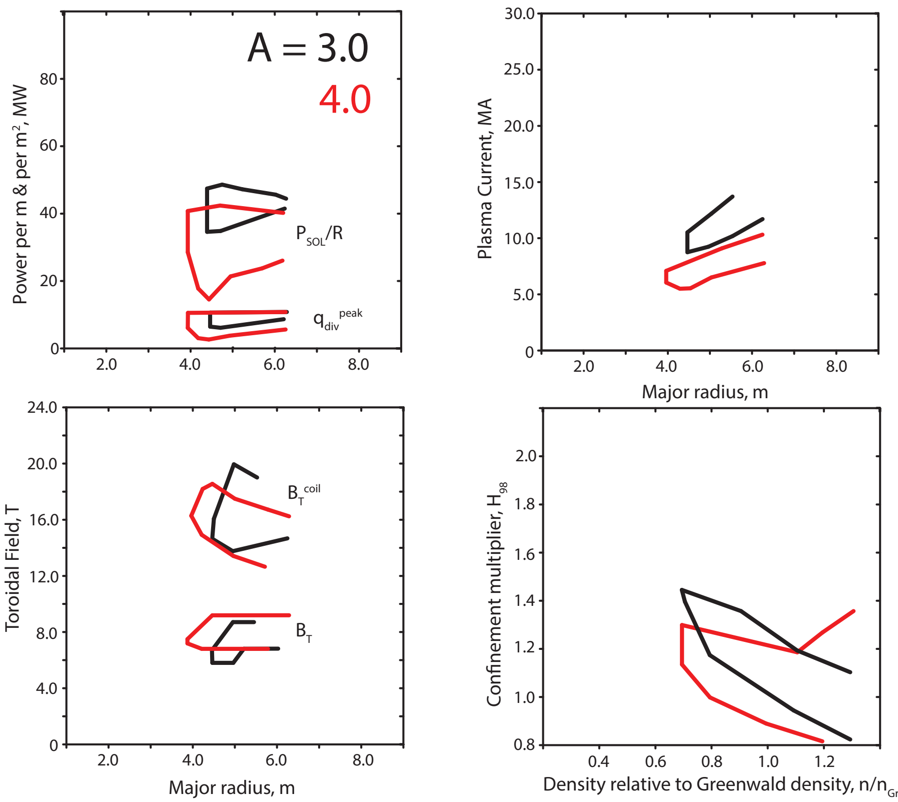


Fig. 8. Systems analysis scans for plasma configurations that satisfy the β_N, neutron wall, and peak heat flux in the divertor, for aspect ratios 3.0 and 4.0, showing that A = 4.0 configurations can access lower plasma current, smaller major radii, lower divertor heating, and lower required energy confinement.

The scanned variables were the major radius from 1.5-6.25 m, toroidal field at the plasma from 4.5-9.0 T, plasma β_N from 0.0175-0.0375, edge safety factor q_{95} from 4.5-8.75, density relative to Greenwald density from 0.7-1.3, fusion gain from 2.0-10.0, argon impurity fraction from 0.15-0.45% (Z_{eff} = 1.5-2.65), and plasma elongation at 1.9 and 2.1 (corresponding to 2.0 and 2.2 at the separatrix). The fixed variables are plasma aspect ratio A = 4.0, triangularity of 0.58, density profile n(0)/<n> = 1.4, temperature profile T(0)/<T> = 2.6, global particle confinement time τ_P^*/τ_E = 5.0, and current drive efficiency at 0.2 A/W-m^2. The filters used to isolate solutions of interest were peak outboard neutron wall load $N_w^{peak} \geq 1.5$ MW/m^2, $\beta_N^{tot} \leq 0.025$, and peak divertor heat flux $q_{div}^{peak} \leq 10$ MW/m^2.

The first systems scans were done at aspect ratios (A = R/a) of 3.0 and 4.0 for comparison. It was found that the A = 4.0 could access lower major radii at lower fusion power, lower P_{SOL}/R and peak divertor heat flux, lower plasma current, and lower required energy confinement multiplier. The lower plasma current is desirable for weakening the effects of a disruption and reducing the current that must be driven by external sources. The lower energy confinement at a given n/n_{Gr} is also desirable for conservative configurations. The A = 3.0 solutions could not access major radius below 4.5 m, while the A= 4.0 could reach 4.0 m. However, this lowest R leads to a peak field at the TF coil that exceeds 16 T, which falls outside our contraints. Fig. 8 shows the plasma current, energy confinement, toroidal fields and divertor power. The aspect ratio of 4.0 was chosen for further FNSF scans. The aspect ratios of the most recent eleven tokamak experiments ranges from 2.48-5.5 with an average aspect ratio of 3.48. Those closest to a value of 4.0 are EAST at 4.07, KSTAR at 3.60, SST-1 at 5.50, and TCV at 3.52.

Systems scans to identify viable operating points for the FNSF were done using the database method where large numbers of physics operating points are identified, which are then processed through an engineering module, and ultimately filtered by constraints that isolate points with the desired parameters. Shown in Table 5 are a reference point along with a number of variants used to examine the trade-offs in assumptions. The second column examines the impact of lower value for the maximum toroidal field at the TF coil, B_T^{coil} = 12.3 T. ITER values for the peak fields are 11.5 T for the TF and 13 T for the CS, with overall coil current densities of 12 and 14 MA/m^2, respectively. An increase in β_N to 3.2 recovers the major radius, lowers the CD power, and increases the H_{98} requirement slightly. If we do not allow the β_N to increase then the major radius increases to 5.0 m, the CD power increases because the plasma current increases, and the peak heat flux ends up above 10 MW/m^2. If we lower the n/n_{Gr} from 1.0 to 0.8, a slight increase in β_N from 2.5 to 2.6 can almost recover a similar operating point, although the peak divertor heat flux is 11.0 MW/m^2. Allowing higher β_N from 2.5 to 2.8, the major radius can shrink to 4.0 m, with most parameters preserved. Finally enforcing a net electricity with Q_{engr} = 1 ($P_{elec,gross}/P_{recirc}$), the β_N rises to 2.9, the major radius is still at 4.5 m, the plasma current drops raising q_{95}, and the CD power drops from 150 to 97 MW.

Table 5. Select parameters for a reference FNSF operating point, and several nearby operating points with different assumptions on plasma physics or technology limits.

	REF	**Lower B_T^{coil}**	**Lower B_T^{coil}**	**Lower n/n_{Gr}**	**Higher β_N**	**Q_{engr}**
Ip, MA	7.51	7.08	8.52	7.51	6.82	6.88
B_T, T (B_T^{coil})	7.0 (14.4)	6.0 (12.3)	6.5 (12.6)	7.0 (14.4)	6.50 (14.3)	7.0 (14.4)
R, m	4.5	4.5	5.0	4.5	4.0	4.5
β_N^{tot}	2.5	3.2	2.48	2.59	2.82	2.92
H_{98}	0.9	1.1	0.8	1.0	1.0	1.1
n/n_{Gr}	1.0	0.9	1.1	0.8	0.9	0.9
q_{95}	5.5	5.0	5.0	5.5	5.0	6.0
$P_{H/CD}$, MW	150	113	196	114	120	97
f_{BS}	0.50	0.57	0.46	0.50	0.51	0.62
q_{div}^{peak}, MW/m^2	9.88	9.19	13.3	11.0	8.8	9.5
N_w^{peak}, MW/m^2	1.54	1.54	1.62	1.55	1.55	1.64
P_{fusion}, MW	450	452	588	456	360	485
Q_{engr} (η_{th}=0.4)	0.7	0.86	0.7	0.85	0.7	1.0

The systems analysis will continue next year under different primary assumptions involving the 1) magnet type, 2) blanket concept (composition), 3) power balance and efficiencies, and 4) physics strategy. Since the FNSF is not pursuing economic electricity production in these studies, the primary assumptions associated with power balance and efficiencies is not considered critical, but will be monitored to see how much electricity could potentially be generated from thermal power.

VI. Design Choices and State of Understanding Documentation

A series of design decisions will be made in assessing the FNSF, as well as documenting the state of understanding in many areas associated with the facility. These are initially promoted as white papers, which may remain as project documents or may be submitted for publication. It is necessary to critically review all technical aspects of a FNSF and the short summaries below provide that documentation on several topics examined in 2014.

VI.A. The use of water in a fusion power core

M. S. Tillack, P. W. Humrickhouse, S. Malang and A. F. Rowcliffe

In the U.S. power plant studies, water has not been chosen as a fusion power plant core coolant internal to the vacuum vessel for decades. At the same time, researchers in other countries continue to adopt water in their future DEMO or power plant designs, in some cases as the leading or sole candidate. In order to assist in the establishment and validation of reference design concepts for FNSF, we documented the technical challenges resulting from the choice of water coolant and the differences in approach and assumptions that lead to different design decisions amongst researchers in this field. The documentation includes a review of past power plant design studies in the U.S., Asia and Europe. Concerns with water are described in the areas of chemical reactivity, tritium safety (inventory, control, and extraction), performance limitations in the blanket and divertor, materials compatibility and neutronic aspects.

The choice of water as a fusion reactor coolant is based to a large extent on the commercial availability of large thermal conversion components and a vast industrial experience base, although the relevance of this experience base to the unique conditions in a fusion reactor is questionable.

The choice of water leads to several negative consequences, including:

- Operation in the liquid phase requires high pressure and low temperature, leading to low thermal conversion efficiency, of the order of 33% or less, leading to higher cost of electricity
- Performance limits in the divertor, restricting the heat flux to 8-10 MW/m^2
- Low ductility of structural materials under irradiation at the lower water coolant temperatures, which would restrict fluence lifetime (to perhaps one year of operation for the divertor) and/or require the development of new materials
- For the water-cooled PbLi blanket requires the use of more complex double-walled tubes to avoid energetic interaction of water with PbLi

- More difficult tritium management and higher inventories, leading to higher occupational and accidental doses
- The risk of hydrogen explosion during accident scenarios
- Additional challenges on tritium breeding as a result of the higher required structure fraction of steel (due to high coolant pressure) and the moderating effect of water on neutrons.
- Coolant activation from the $O^{16}(n, p)N^{16}$ reaction and corrosion products.

Due to the large number of negative consequences, continued effort to identify and develop more attractive coolants is prudent. In the US, the reference blanket and divertor coolants are helium and PbLi (in a dual-cooled blanket configuration). The issues surrounding the use of He and PbLi are described elsewhere in this report.

M. S. Tillack, P. W. Humrickhouse, S. Malang and A. F. Rowcliffe, "The use of water in a fusion power core," University of California, San Diego report number UCSD-CER-14-01, October 2014.

VI.B. Technology readiness of helium as a fusion power core coolant

M. S. Tillack, P. W. Humrickhouse, S. Malang

Helium is an attractive coolant for fusion power plant applications due to its chemical inertness (resulting in safety and performance advantages), compatibility with other reactor materials, low neutron cross section, and high temperature capability that enables high thermodynamic efficiencies. Worldwide, a large number of fusion power plant studies have proposed using helium as a coolant in the blanket, divertor, and recently even in the vacuum vessel. At present, the mainline candidates for both the blanket and divertor of an FNSF use helium coolant. The technical readiness and remaining challenges related to helium as a coolant are documented in order to support design decisions within the project. A summary and findings are provided below.

The technologies needed for large-scale high-temperature helium-cooled systems already has been developed and implemented in the fission industry. Prismatic and pebble bed reactors have operated in the US, Europe and Asia, with electric power generation demonstrated up to levels of 300 MW or more. Notwithstanding the advantages and past experience base, concerns have been expressed over the use of helium as a coolant in fusion power plants, including limitations in cooling capability, pumping power, impact on power core size (due to the transparency of He to neutrons) and coolant manifold size, a limited industrial supply chain, and even limitations in helium resources.

The existing design concepts and operating parameters for fusion power plant components were summarized that use helium as a coolant, described issues with the use of helium, and explained the rationale for concluding that the issues either can be avoided by design or solved through further R&D. The emphasis was on in-vessel components, including the blanket and divertor, and their required ancillary systems (manifolds, piping and heat exchangers). Power cycles using helium were excluded from our consideration.

Recent conceptual studies of fusion power plant blankets and divertors were summarized, together with the main design parameters and justification of design choices. A more detailed examination of the performance issues and existing database were included, as was a review of the experience base and lessons learned from the fission industry, which has operated helium and CO_2 gas-cooled reactors since the 1960's.

In summary, helium has been used as a coolant in several fission reactors around the world with coolant temperature, pressure and flow rate similar to those of fusion blanket and divertor designs. This indicates that an industrial basis exists. Helium offers unique challenges as compared with water, and those challenges have been addressed in conceptual design studies. Experiments have been performed on small divertor mockups to validate heat transfer models and demonstrate performance under high heat flux. Larger validation tests have been performed for the several ITER test blanket modules that use helium as coolant. Heat removal capabilities are not worse than water in a properly designed system. The main penalty is increased pumping power, but this can be managed to levels that are small compared with the recirculating power needed for plasma sustainment.

The use of helium allows several important advantages to the designer, including chemical compatibility with pressure vessel materials, tritium safety and the ability to operate at high temperature without constraints from the coolant itself. Helium provides a pathway to improvements, allowing the introduction of fusion power core technology at modest temperature levels without limitations enabled by future materials development. Due to its advantages in the near term, as well as its long-term prospects, we consider that the choice of helium as a fusion power core coolant is well justified.

M. S. Tillack, P. W. Humrickhouse, S. Malang, "Technology readiness of helium as a fusion power core coolant," University of California, San Diego report number UCSD-CER-14-03, December 2014.

VI.C. Single Null versus Double Null Divertor Configuratons

M.E. Rensink and T.D. Rognlien

The design-study decision for FNSF tokamak or ST devices includes selecting either a single-null (SN) or a double-null (DN) divertor magnetic configuration. While the decision clearly impacts a portion of the first wall and blankets, it also impacts the heat-flux distribution to plasma facing components and operating characteristics of the plasma discharge. Note that even for a device with exact magnetic symmetry about an equatorial midplane, the plasma does not respond symmetrically owing to magnetic and electric particle drifts, and these differences are most apparent in the behavior of the boundary plasma. Also, the ITER project selected a single-null configuration. A study of many experimental and theory/simulation papers has been performed to elucidate the issues and provide guidance for the SN/DN decision. Here a short summary of the key results is given, with more detail and a list of references available in the white-paper report.

Four main areas are considered:

1. Cost of divertor components
2. Divertor effectiveness: influence of cross-field drifts, power distribution between and during ELMs, pumping, impurity distribution
3. Impact on pedestal/core physics: ELM characteristics including impurity flushing, core confinement, density limit, H-mode power threshold, disruptions, plasma shaping
4. Other issues: magnetic control; flaking of material surface on top falling through plasma discharge

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Divertor cost: An obvious argument for a SN configuration is that it requires only one set of divertor components whereas a double-null configuration requires two, with a corresponding increase in the cost of construction, vertical size, and possibly maintenance.

Divertor effectiveness: An argument for DN is that it gives twice the area for heat-flux removal, but the edge plasma dynamics significantly reduces the useful area. The power split between inner and outer and upper and lower divertor targets depends on a number of effects, such as plasma density and radiation losses (detachment conditions) at each divertor target, direction of the vertical plasma magnetic drift (into or out of the divertor from the x-point), and the degree of magnetic balance between primary and secondary x-points. Of particular importance is that experimentally, the inner two divertor plates for a balanced DN receive very little steady-state heat-flux compared to the inner plate of a SN divertor, or the outer plates. This effect is attributed to low turbulence on the (stable) inner separatrix, no magnetic connection between inner and outer scrape-off layers (SOLs), and larger plasma radial gradients across the outer separatrix.

For ELM heat loads, the inner divertor of a SN can receive ~1/2 of the power, though owing to it being at a smaller major radius, the area may be smaller unless increased by poloidal magnetic flux expansion. For DN, the two inner plates again see little power. Finally, up/down symmetry of the outer plate heat fluxes in DN require that a slight magnetic imbalance be maintained because the particle magnetic drifts drive an asymmetry themselves.

Pedestal and core plasma performance: DN operation can also affect the pedestal and core plasma characteristics as observed in a number of present-day tokamak. For SN, large Type-I ELMs produce the best energy confinement, while the smaller, more frequent Type-III ELMs lower confinement time significantly and can lead to return to the L-mode. DN generally have Type-II ELMs, intermediate to Types I and III, where the decrease in confinement time is generally modest. Experimental papers on the behavior of ELMs for DN discharges come from DIII-D, NSTX, JET, ASDEX-U, and MAST. Control of core impurity accumulation is associated with ELM "flushing," and it appears that Type-II ELMs provide this positive role.

L-H transition power can be ~20% lower in DN, but this requires being very close to up/down magnetic balance, and it isn't clear if the up/down heat flux is then balanced. Vertical-displacement disruptions are better controlled in DN. No information was found on the comparative density limits for SN versus DN.

For plasma shaping, the DN allows magnetic equilibria with high triangularity in both the upper and lower regions, which may be favorable for high-beta operation. The SN can have similar shapes, but then a secondary x-point is typically located just outside the first wall, as in the ITER design. Such a nearby x-point can enhance particle and heat fluxes to PFCs in that region. The behavior of impurity intrusion to the core for radiative divertor discharges differs some for SN and DN, and depends on details of the ion gradient-B drift in SN and the degree of magnetic balance in DN.

Other SN/DN issues: Optimum divertor heat loading in DN, both steady-state and from ELMs, probably requires operating with a slightly unbalanced DN owing to classical cross-field magnetic particle drifts that cause asymmetries in the plasma properties. Inadequate control of the magnetic balance, or operational errors could result in unanticipated large power fluxes to either divertor. Other effects such as ELM behavior, plasma shaping, impurity ejection by ELMs, and disruptions may prefer a different level of magnetic balance.

Dust flakes are created on divertor surfaces and some may be large. In a DN, such macro-particles can fall through the main discharge, perhaps causing a strong perturbation to the core plasma, whereas in the SN, the dust stays on the bottom unless mobilized by plasma charging. For tritium breeding, the DN reduces blanket module coverage at the top because the upper divertor structure must be accommodated.

Overview of divertor structure in each leg region: In addition to the general SN/DN issue just summarized, there are options for the detailed geometrical configuration of both the divertor plates and the magnetic flux surfaces along the dominant path that the plasma exhaust power flows. For our study of the ACT-1 divertor (see January issue of Fusion Science and Technology), we compared a tilted plate divertor similar to that designed for ITER with a flat-plate divertor (no tilt with respect to the flux surfaces). The tilting produces two desirable effects: (1) it reduces the peak heat load by increasing the "wetted" area exposed to the heat flux and (2) the tilting directs recycled neutral particles toward the high heat-flux region, further reducing the peak heat flux by producing a partially detached plasma. The flat-plate divertor (no tilt with respect to the flux surfaces) with a sufficient width to allow recycled neutrals to interact with the high heat flux exhaust from two sides results in full plasma detachment, resulting in a lower peak power than the tilted-plate design. Such detached plasmas have been observed in DIII-D and other devices, and are presently being vigorously investigated. While the lower peak heat flux from full detachment is attractive, there are concerns that the flat-plate divertor is less flexible for plasma start-up and shut-down, and that ELM damage may be more severe; these issues need further analysis and data.

The magnetic configuration can also help reduce the peak heat load in the divertor region and may improve other edge-related features such as ELM characteristics and ion orbit loss. These include a variety of designs that feature secondary magnetic X-points in the divertor region(s). Examples are the snowflake divertor with two magnetic X-points in close proximity, the X-divertor with a second X-point just behind the divertor plate or a related design with the second X-point in front of the divertor plate, and finally, the Super-X divertor with the second X-point at a substantially larger major radius location, R, to gain the increased divertor plate surface area ($\propto$ R). A number of these designs are or will be investigated experimentally. The results of these studies will greatly aid in focusing on which options could/should be included in future devices.

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VI.D. The Materials-Design Interface for Fusion Power Core Components

M. S. Tillack, N. M. Ghoniem, J. P. Blanchard and R. E. Nygren

A fusion nuclear science facility is expected to encounter conditions far more severe than those in ITER, with higher operating temperatures, much higher neutron fluences, and higher demands on reliability. The desire to construct a fusion nuclear facility in the US within the coming decades requires examination of the readiness to license such a facility and operate it for long periods of time with the requisite reliability. At present, neither functional materials, nor the requisite computational tools, nor the underlying knowledge base currently exist for reliable integrity and lifetime assessments of fusion in-vessel structures. The lack of a significant program of research on component-level materials issues places FNSF at serious risk. We examined the area we call the “materials-design interface” and described the necessary near-term R&D needed to begin to address the challenges. A summary of our white paper follows.

What is the “materials design interface” and why is it important?

Research on individual material properties, informed by conceptual design studies, is not sufficient to resolve the fundamental issues of survivability and performance of in-vessel components, which is absolutely required in order for fusion to be useful as an energy source. The mechanical behavior of components in the fusion environment is highly complex and design-dependent, requiring research into the critical design-dependent phenomena that might lead to failure. The research area that we describe as "the materials-design interface" requires strongly coupled investigations of the mechanical behavior of materials within a design context.

This topic is critical for the success of fusion as an energy source.

In-vessel components must survive a challenging, unique and unexplored environment involving extreme conditions of heat flux, plasma particle flux, radiation fields (high-energy neutrons and gamma rays), strong magnetic fields and the ubiquitous presence of hydrogen. They must satisfy a set of requirements to fulfill their own functions as well as overall plant requirements. Because failures can have catastrophic consequences on plant operations, and overall plant availability must be high, high confidence in the reliability of components is needed. Given our current understanding of how to produce and sustain burning plasmas, the primary remaining challenge is how to extract the energy in a way that is commercially and environmentally acceptable. Without structural materials that can function reliably in real components, and not only as small test specimens, fusion energy will not be realized as a viable power source.

This area has been neglected in the past, leading to a very low level of maturity.

The amount of past research in this area has been small within the U.S., and much of the work is not relevant to next-step nuclear devices or Demo. Large gaps in knowledge remain. Related efforts on the mechanical behavior of components have been performed within the ITER project, which has advanced the state-of-the-art in methods for fusion component "design by analysis", design rules and component validation. However, the requirements, designs and materials for ITER are all very different from those of a fusion power plant. ITER has no breeding requirement (which impacts design choices and design details), operates at low temperature, and will experience very low neutron dose. The materials chosen for ITER could not be used in a power plant. Furthermore, our involvement in ITER has declined: for example, the U.S. has chosen not to participate in the fabrication of first wall modules or the divertor, and our connection with the first wall design ended in 2013.

Research must expand immediately for FNSF and Demo to succeed in this century.

The time required to build and operate experiments, generate data, develop design rules, and prepare for qualification of nuclear components can be measured in decades rather than years. Starting from the current state of neglect, a rapid increase in funding in this area of R&D will be needed to meet the timelines under discussion for FNSF and Demo, as noted in the recently completed "FESAC Report on Strategic Planning: Priorities Assessment And Budget Scenarios". In addition, being such a crucial aspect of in-vessel component behavior, results from this program should be used in overall fusion program planning and design selection. Without strong input from the materials-design interface, the basis for decision-making will be incomplete.

Needed research includes modeling, design rules, fabrication techniques and experiments.

At present, neither functional materials, nor the requisite computational tools, nor the underlying knowledge base currently exist for reliable integrity and lifetime assessments of fusion in-vessel structures. New design and in-service performance computational tools must be developed to replace simplistic high temperature design and operational rules. These tools must ultimately be incorporated in design codes and regulatory requirements.

The greatest challenge is a lack of understanding with respect to material behavior. A few examples of this limited understanding include failure mechanisms in tungsten alloys, radiation damage effects on mechanical properties in the presence of fusion-relevant helium concentrations, surface morphology of plasma-facing structures and their effects, effects of synergistic radiation and thermomechanical damage on first wall and blanket components, and models of ferromagnetic materials, especially in the presence of transient magnetic fields.

In addition to these deficiencies, there is only limited understanding of macroscopic failure mechanisms, especially in the harsh environment experienced by a fusion component. For example, the damage due to the interaction of creep and fatigue is difficult to model under normal conditions, but adding radiation damage, helium, *etc*. increases the uncertainty dramatically. Similarly, the dual nature of brittleness and ductility in tungsten is not well understood, especially in plasma and neutron environments, because it has not typically been used as a structural material. It is possible to make some progress on enhanced understanding of these phenomena using coupon tests, but it is impossible to properly address failure mechanisms without comprehensive structural models, which include coolant pressure, coolant chemistry, static thermal gradients, thermal transients, and radiation damage. Hence, a multi-disciplinary, multi-scale effort is needed to comprehensively address the materials-design interface and permit substantial progress towards the design of high performance, optimized components.

This program can begin to make progress at a modest funding level.

The resources needed to fully develop, test and qualify fusion in-vessel components will be large. However, significant progress can be made to establish the scientific foundations for this field and provide a credible path forward, to FNSF and beyond, with levels of funding that could be obtained within the current OFES budget. A coordinated program involving participation from universities, laboratories and industry, as well as international collaborations, provides the most effective path forward. We estimate that funding of the order of $2M per year for several years would be sufficient to lay the groundwork.

In order to develop expertise required to advance this area, active outreach (beyond the current set of contributors to OFES-sponsored programs), planning workshops and programs to support student training should be an integral part of the near-term program. All of these efforts will require financial resources. At the end of the first phase of this activity, we would be in a better position to evaluate the required tools and experiments to establish the feasibility and lifetime of in-vessel components and to determine the funding and research requirements for the next phase of research toward qualification of FNSF components.

M. S. Tillack, N. M. Ghoniem, J. P. Blanchard and R. E. Nygren, "The Materials-Design Interface for Fusion Power Core Components," University of California, San Diego report number UCSD-CER-14-02, October 2014.

VI.E. Tritium Breeding Ratio

L. El-Guebaly, University of Wisconsin

The tritium breeding capability of the blanket is of particular interest since it defines a critical element of the FNSF mission (e.g., tritium self-sufficiency). Once the FNSF enters its technological Phase 3, the FW and blanket must have sufficient neutronics-related characteristics (blanket coverage area, thickness, and materials) to generate the required tritium (T) for sustained plasma operation. However, it is challenging to achieve tritium self-sufficiency in FNSF, especially for compact devices where a higher fraction of the outboard (OB) surface area will be devoted to test modules and ports/penetrations. The tritium breeding ratio (TBR) is a key metric for tritium self-sufficiency. A calculated net TBR slightly above unity implies the machine breeds tritium at a level exceeding the combination of consumption, uncertainties in the calculated TBR, and T holdups, losses, decay, etc. The pertinent questions are: How high the TBR should be? Could breeding-related R&D programs reduce the uncertainties in TBR prediction? Does a shortage of T present an economic burden or place the FNSF operation at risk? Do external sources for large quantities of tritium exist in the U.S. or abroad? How to control the overproduction of tritium in case of an over-breeding blanket?

Do external sources for large quantities of tritium exist in the U.S. or abroad? There is worldwide interest in building several D-T fueled FNSFs, DEMOs, and other experimental devices between 2030 and 2040. Such devices will have to either generate their own T or compete for the very limited external sources of tritium. Tritium has been produced and collected for commercial use as a byproduct from non-military sources, such as the Canadian heavy water-cooled CANDU reactors [1]. There is a strong indication that ITER will consume almost all T recovered from CANDU reactors (~1.7 kg/y). Other sources of T exist in other countries, but they are limited in supply, classified, and/or inaccessible for general use. For these reasons of limited or no T-supply, fusion devices generating hundreds MW of fusion power (such as FNSF) must breed their own T to mitigate the risk of relying on external supplies to provide/control the essential fuel of the machine.

Does a shortage of T present an economic burden or place the FNSF operation at risk? The TBR should be estimated with high fidelity. A FNSF with 500 MW fusion power consumes 27.8 kg of T per full power year (FPY) of operation. A small 1% deficiency in the TBR is equivalent to ~0.28 kg of T/FPY. At a unit cost for tritium ranging from ~$30k to ~$118k per gram of T [2], this 1% deficiency implies a FPY operational cost of $8-33M to purchase T from external sources. Larger deficiencies in TBR represent a significant burden on the FNSF operational cost, as shown in Fig. 1.

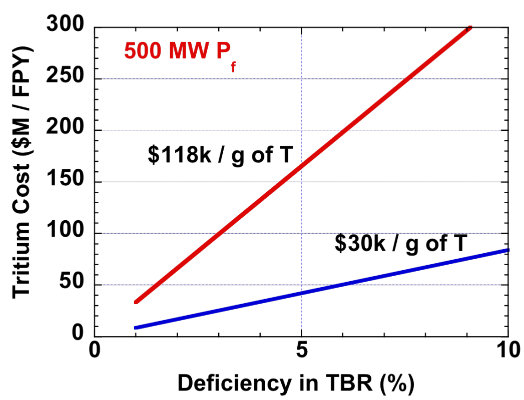

Fig. 1. FPY cost of purchasing T from external sources versus deficiency in TBR of FNSF with 500 MW fusion power.

How high the TBR should be? The required TBR is design and breeder dependent [3]. In the contemplated FNSF design with a DCLL blanket, a calculated TBR of 1.04 should achieve T self-sufficiency. This is a computational target that may be raised or lowered, if needed, during operation with online adjustment of ^{6}Li enrichment of the PbLi breeder/coolant. The 0.04 breeding margin provides for currently known deficiencies in the nuclear data, approximations in the 3-D modeling of DCLL blanket, and T holdups, losses, and decay [4]. In the future, these unknowns should be reduced. The current 0.04 breeding margin is relatively low for these reasons:

- Recent integral experiments at ENEA in Italy [5] predicted < 3% uncertainties in the nuclear data of two European blanket designs: He-cooled PbLi and He-cooled pebble bed. No data available for the DCLL blanket concept
- Recent advances in University of Wisconsin computational techniques [6] enabled computing the TBR with high fidelity for the complex configuration developed by blanket designers. This sophisticated computational tool (that couples the CAD with 3-D neutronics code) also enables good modeling of experimental facilities (first point above) to help obtain good agreement between calculated T production and measured values

- Unlike power plants, FNSF will not generate excess T nor provide a start-up inventory for future devices
- Aggressive assumptions were made related to minimal T inventory/holdups in all subsystems, highly efficient T extraction system with redundant components, short times for T reprocessing, efficient detritiation systems, negligible T losses to the environment, and an efficient T accountancy system.
-

Several design elements significantly affect the TBR of FNSF: the FW thickness, design of side/top/bottom/back walls, choice of coolant, cooling channel geometry, flow channel insert (FCI) material and thickness, W stabilizing shells, assembly gaps, support structure/frame of TBMs, and size and orientation of H/CD penetrations. Another approach suggests designing all in-vessel components around the TBR requirement by maximizing the blanket coverage, reducing the structural content within the FW and blanket, minimizing the H/CD aperture size and maximizing its tangential radius. Past findings provide useful insight into the understanding of how the individual design elements influence the TBR of the DCLL blanket and what conditions or changes are more damaging/enhancing to the breeding [4,7].

Could breeding-related R&D programs reduce the uncertainties in TBR prediction? Well-planned R&D program is needed to reduce the unknowns involving the T production, storage, processing, etc. A large gap exists between near-term fusion experiments (such as ITER that generates ~4 g of T/y) and FNSF (that could produce ~30 kg of T/FPY) – which would be the largest T production facility in the U.S. fusion history. Prior to operating FNSF, a dedicated R&D program, involving both analytical and lab-based studies, could close this gap, reduce the breeding margin to ~1%, validate the computed T production rates, demonstrate the T recovery and storage processes, and determine the T inventory, holdups, and efficiency of T processing and detritiation systems.

How to control the overproduction of tritium in case of an over-breeding blanket? Because some uncertainties in the operating system govern the achievable breeding level, the net TBR will not be verified until after operating the FNSF with fully integrated blanket, T extraction system, and T processing system. Therefore, it is necessary for the blanket to have a flexible approach such as a feasible scheme to adjust the ^{6}Li enrichment online during operation [8,9]. A net TBR ≈ 1 can easily be achieved by adjusting the ^{6}Li enrichment ratio online, thus avoiding the problem of storing/disposing of excess tritium or purchasing tritium if there is a deficiency.

In summary, tritium self-sufficiency is appealing and necessary from an economic perspective. The FNSF ability to breed all the tritium required to sustain the plasma operation requires production of 27.8 kg of T per FPY for 500 MW of fusion power. This is a technically challenging requirement for the blanket and suggests designing all in-vessel components around the TBR requirement. FNSF will provide power plant-relevant operational information for attaining tritium self-sufficiency (such as tritium inventory, holdups, losses, and reliability and efficiency of tritium processing and detritiation systems).

VI.F. Fusion Radioactive Waste Management

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Fusion has long been envisioned as possessing an inherent advantage for benign environmental impact, mainly due to the absence of high-level waste (HLW) generation. However, fusion tends to generate a sizable amount of mildly radioactive materials. Such a potential problem has been overlooked in early fusion studies and/or relegated to the back-end as only a disposal issue in low-level waste (LLW) repositories, adopting the preferred radioactive waste (radwaste) management approach of the 1960s. The large volume of fusion LLW generated during operation and after decommissioning will fill existing US LLW repositories rapidly. To put matters into perspective, we compared in Fig. 1 the power core volumes of the ITER experimental device, the advanced ARIES power plants (ARIES-ACT-1&2) and the European Power Plant Conceptual Study (PPCS) to ESBWR (Economic Simplified Boiling Water Reactor) – a Gen-III^{+} advanced fission reactor.

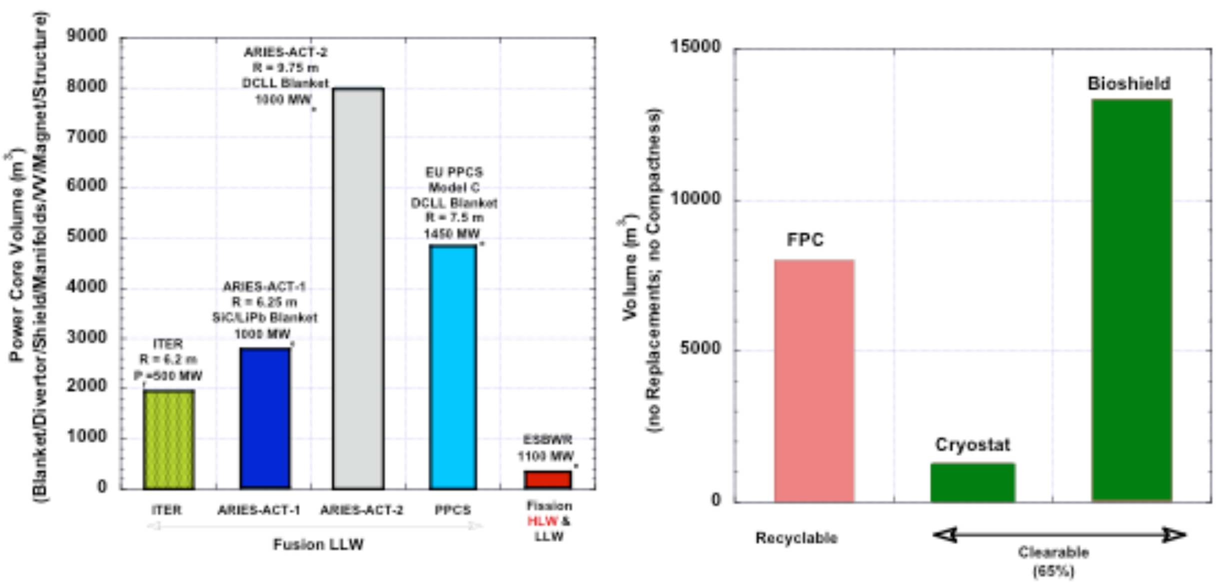

Fig. 1. Comparison of radioactive waste from power core of fusion and fission designs (actual volumes of components; not compacted; no replacement).

The shallow land burial is not a viable option for fusion and it is essential to think of a more environmentally attractive framework to keep the fusion LLW volume to a minimum. This is important to the future of fusion energy. Concerns about the environment, radwaste burden for future generations, lack of geological repositories, and high disposal cost directed our attention to recycling of the radioactive materials (for reuse within the nuclear industry) and clearance (the unconditional release to the commercial market if materials contain traces of radioactivity).

In recent years, the recycling and clearance approaches became more technically feasible with the development of advanced radiation-hardened remote handling (RH) tools that can recycle highly irradiated materials [1] along with the introduction of the clearance category for slightly radioactive materials by national and international nuclear agencies. If integrated properly at an early stage of the design process, fusion will eventually reach the ultimate goal of radwaste-free energy source. At present, the US experience with recycling/clearance is limited, but will be augmented significantly by advances in fission reactor dismantling, spent fuel reprocessing, and bioshield clearance before fusion is committed to commercialization in the second half of the 21st century.

Beginning in 2000, numerous fusion studies indicated the recycling/clearance approach is relatively easy to envision and apply from a science perspective. To support this argument, we applied all three scenarios (disposal, recycling, and clearance) to ARIES designs [2,3,4]. The technical feasibility of recycling could be based on the dose rate to the remote handling (RH) equipment. Essentially, the dose determines the RH needs (hands-on, conventional, or advanced tools to handle the radioactive components) and the interim storage period necessary to meet the dose limit. Advanced RH equipment has been used in the nuclear industry, in hot cells and reprocessing plants, and in spent fuel facilities. While the fission processes may have no direct relevance to fusion, their success gives confidence that advanced RH techniques could be developed to handle high doses (> 10,000 Sv/h) for the recycling of fusion materials. Beside the recycling dose, other important criteria include the decay heat level during reprocessing, recycling of T-containing materials, physical properties of recycled products, and economics of fabricating complex shapes remotely. All fusion materials, even the highly irradiated first wall, can potentially be recycled in less than a year after shutdown with advanced RH equipment that can handle 10,000 Sv/h or more. ^{54}Mn (from Fe) is the main contributor to the dose of RAFM-based components at early cooling periods (<10 y), while impurities have no contribution to the recycling dose during such a period.

A material qualifies for clearance if its clearance index (CI) drops below one at any time during the 100-y storage period following replacement or decommissioning. In ARIES-ACT-2, the CIs for all internal components (blanket, SR, VV, shield) exceed unity by a wide margin. The 2 m thick external concrete building (bioshield) that surrounds the tokamak represents the largest single component of the decommissioned materials, (refer to Fig. 1). Fortunately, if adequately protected, the bioshield along with the cryostat qualifies for clearance, representing ~65% of the total volume of ARIES-ACT-2 radioactive materials.

In summary, the amount of fusion LLW is large, so efforts to avoid the geological disposal, and promote the recycling and clearance of all components are essential. This new approach for managing fusion radioactive materials will relax/eliminate the stringent requirement of generating only LLW imposed on alloying elements and impurities of fusion materials.

VI.G. The Dual Coolant Lead Lithium (DCLL) Blanket Concept as the Primary Candidate for FNSF

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A blanket concept is characterized by the breeder material, the structural material, and the coolant used to extract the heat. There are concepts where the FW facing the plasma is separated from the breeding blanket. For simplicity reasons, however, the followings will be concentrated on blanket concepts with integrated FW, built of the same material as the blanket structure.

Candidate breeder materials in a large number of blanket studies as well as power plant studies utilize either ceramic breeders or liquid metal breeders i.e. lithium or the eutectic lithium lead alloy Pb-17 Li. Both classes of breeders have their advantages, and disadvantages.

a) Ceramic breeder

In most of the ceramic breeder blankets the breeder, as well as the Beryllium required as neutron multiplier, are arranged as pebble beds filled into the space between cooling plates. These plates as well as the FW is cooled with He of ~ 8 MPa pressure in most of the present blanket concepts.

Unfortunately the behavior of the pebble beds under fusion irradiation conditions will not be sufficiently known prior to the operation of a FNSF. A further disadvantage of ceramic breeder blankets is the relatively small temperature window in which the ceramic breeder (high enough to release tritium while low enough to avoid sintering or other material degradation) as well as the beryllium multiplier can be operated. This limits the allowable neutron wall load as well as the achievable coolant outlet temperature, resulting in a thermal efficiency of the power conversion system to values < 38 %.

A crucial issue of ceramic breeder blankets in fusion power plants is the missing possibility to adjust TBR without replacing blanket modules. The most sophisticated neutronics model have still an uncertainty of > 3 % in predicting the effective TBR of a power plant. However, it is clear that a real TBR-value < 1 would require huge costs for buying the missing tritium (if possible at all). If the net TBR would be > 1.02, the storage of the excess T could become a real safety issue.

b) Liquid metal breeder

In principle, liquid metal breeders have the following inherent features making them to attractive candidate materials:

- Immunity to radiation damage,

- Potential for tritium self-sufficiency without requiring an additional neutron multiplier,

- Tritium extraction can be performed outside the blanket,

- On-line adjustment of TBR possible,

- Relatively low operational pressure required,

- High thermal conductivity of the breeder (compared to ceramic breeder)

There is a growing agreement in the international community that blanket concepts based on liquid metal breeders lead in the long run to more attractive power plants than ceramic breeder blankets.

Main candidate liquid metal breeder is the eutectic PbLi alloy because its potential for chemical reaction with water or air is much lower than for of Li, and the achievable TBR is in general higher than with Li due to its content of Pb as neutron multiplier.

There are blanket concepts where the liquid metal is quasi stagnant as a pool, circulated slowly to the outside for tritium extraction. This pool is cooled by He similar to ceramic breeder blankets, but the higher thermal conductivity allows larger distances between these plates and fewer of them.

The simplest design can be obtained when the liquid metal serves not only as breeder material but as coolant too, circulated between the blanket and an external heat exchanger for heat extraction.

Critical issue of such self-cooled blanket concepts is the impact of the strong magnetic field on flowing liquid metal. Without an electrical insulator between the liquid metal and metallic duct walls the resulting MHD pressure drops would be prohibitive high and unfavorable flow velocity profiles as well as the limited compatibility of the liquid breeder with the structural materials are still feasibility issues.

A promising method to provide the required electrical insulation in advanced blanket concepts is to use a SiC-composite as structural material, but the performance of this material under high fluence neutron irradiation remains an open question. Fission neutron exposure has shown that SiC-composites are a qualified material for fission power core use. However, it remains to be seen if this material meets the requirements of low electrical conductivity as well as the mechanical integrity for the use as a structural material in the fusion typical high neutron energy of a fusion power plant.

Fortunately, the investigation of the MHD issues does not require irradiation tests and can be performed to a large degree in available MHD and corrosion test facilities.

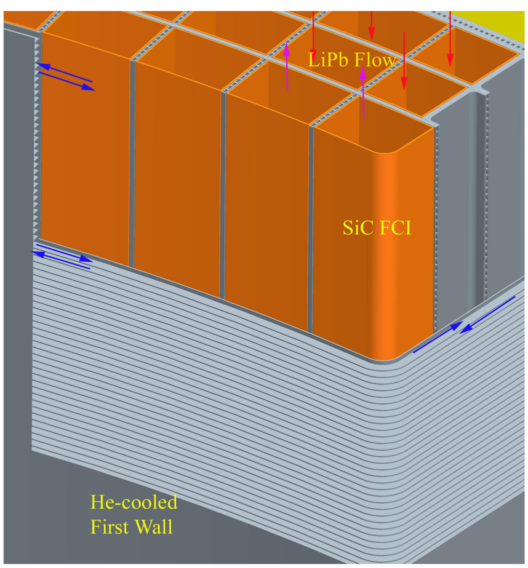
Dual Coolant blanket concept

The really difficult cooling of the FW with the LM coolant in self-cooled blankets was the incentive to develop a blanket concept with a self-cooled PbLi breeding zone but with a He-cooled FW and blanket structure. The liquid metal breeder is flowing with low velocity (a few cm/s) in large poloidal ducts and can achieve a considerably higher outlet temperature than the temperature of the steel structure.

At the beginning of the design development for the DCLL concept, an electrically insulating coating at the surface of the duct walls was suggested. Alternatively the use of flow channel inserts (FCI) made as a sandwich steel-alumina-steel was considered as electrical insulator in such Dual Coolant Lead Lithium (DCLL) blanket concepts.

A breakthrough was achieved in the ARIES-ST study with the idea to replace the sandwich FCI by an insert made of the electrically and thermally insulating SiC-composite.

The figure below shows an isometric view of a typical DCLL blanket.



Here, the FW and the entire blanket structure is cooled with helium at an inlet pressure of ~ 8 MPa. The large poloidal ducts for the PbLi flow have typical cross sections of 200 mm by 200 mm, and the SiC-FCI's a thickness of ~ 5 mm. In most tokamak designs such blanket segments cover the entire height of the power core with the PbLi flowing upwards in the first row of ducts and down in the back rows with lower velocity.

With such a concept it is feasible to obtain in a blanket made of ferritc-martensitic-steel a LM exit temperature of ~ 700 C, maintaining the maximum steel temperature < 550 C and the interface temperature steel-PbLi < 500 C. About 50 % of the total heat is retracted with the LM, the remaining 50 % with He in the temperature range between 350 C and 480 C. These temperatures allow the use of a Brayton cycle power conversion system with a thermal efficiency of up to 45 %.

Advantages of such a DCLL blanket system compared to He-cooled concepts (such as the HCLL) are:

- No large internal heat exchange surfaces inside the blanket required (reliability issue!)

- Steel content in the breeding zone considerably lower (higher TBR achievable) than in He-cooled PbLi blankets,

- High liquid breeder circulation rate allows to minimize tritium partial pressure in the PbLi to values < 1 Pa (typical a few hundred Pa in He-cooled PbLi blankets)

- Thermal efficiency in the power conversion system up to 45 % achievable (limited in He-cooled blankets to <38 %)

- Possibility to extract the after heat by natural convection of the PbLi in case of LOCA, LOFT, or loss of power accidents. Evaluations have shown that this is feasible even with the magnets energized.

A crucial issue for such a blanket concept is the impact of the strong magnetic field on the liquid metal and especially on the flow distribution into parallel channels. A key issue is here the flow distribution from one coolant access pipe/module into ~8 parallel poloidal ducts inside the module. With the presently available tools it is not possible describe this distribution and to ensure equal flow rates in the parallel ducts.

Further issues requiring more R&D work is the development of the FCI's made of SiC, the compatibility between PbLi, ferritic steel, and the qualification of SiC used for FCI's. Fortunately, most of these issues can be investigated without irradiation tests in fusion typical environment. It is possible to test small assemblies of these materials in a fission reactor or accelerator based environment before the FNSF.

FCI's made of SiC can only be tested for fusion neutron exposure as an individual material, but cannot be qualified for the integrated irradiation conditions in a fusion power plant prior to the start of FNSF operation. A more conservative low temperature DCLL blanket based on sandwich FCI's can be used at the beginning.

VI.H. Comparison of Horizontal and Vertical Maintenance Approaches, Robotic Remote Handling for Maintenance, and Special Maintenance for the Test Blanket Modules

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Description of Maintenance Approaches for High Plant Availability - At the present time of 2014 there two schools of thought on how to maintain and repair the tokamak fusion power core. It is widely recognized the ITER scheme of using manipulators to disengage smaller modules inside the power core would result in too much contamination and is too time consuming for any future facility to achieve high plant availability. Instead, the thinking is to significantly increase the replacement article size up to a complete sector (representing fraction of the power core divided by the number of TF coils) extracted horizontally or a subdivision of the sector into smaller segments to be removed vertically in upper and/or lower ports. The horizontal and vertical maintenance approaches will be analyzed and compared in this document.

Full Sector, Horizontal Maintenance - Most of the U.S. and Japanese conceptual tokamak power plant designs employed a complete sector maintenance approach with horizontal removal. These studies assumed the first wall, blanket, divertor and H/CD launcher all had similar lifetimes that enabled simultaneous replacement of a complete sector. However with the present state of inadequate component lifetime knowledge, the first-order assumption is that the divertor may have a shorter lifetime. This assumption either requires a very expeditious sector removal/replacement action or a separate dedicated divertor maintenance approach. The full sector horizontal removal approach has been analyzed in substantial detail in ARIES-RS and ARIES-AT studies.

Most authorities on fusion maintenance agree that the horizontal sector maintenance approach will yield the shortest maintenance time, the easiest and fewest hydraulic connections, the least in-vessel contamination and the most reliable replacement of any proposed maintenance approach. The detriments for the horizontal approach are that the TF coils must be increased in size to accommodate the sectors passing between the TF outer legs, the power core building must increase in size to accommodate the maintenance corridor and the ability to easily move and align the large sector must be validated.

A slight variation of the horizontal sector maintenance approach is investigated by the Japanese for the JAEA DEMO. They are proposing using only four horizontal ports to access and maintain the sectors. This approach allows fewer large ports, a stronger anti-torque structure, however it requires 2/3s of the sectors to be moved toroidally inside the vacuum vessel to align with one of the maintenance ports. These sectors must be first disconnected (structurally and hydraulically) inside the vessel, toroidally moved to the ports via a rail system or inserted mobile platforms and then removed from the power core like the normal horizontal approach. This four-port approach requires longer maintenance duration and more likelihood of damage and contamination within the vessel. Another downside characteristic is that when a sector requires unscheduled maintenance, there is a 66% chance that at least two sectors will require extraction for the repair. The power core building, maintenance corridor and hot cell would be identical to the full sector horizontal maintenance approach. Due to the longer maintenance period, this may allow fewer transporters and casks.

Vertical Segment Maintenance – The another candidate option for maintaining the fusion power core is the vertical segment approach that uses fewer, smaller access ports at the top of the vacuum vessel between the TF coils. There the available space for a maintenance port is more limited, thus requiring a smaller subdivision of first wall, blanket, shield and hot support structure elements. In the context of the vertical maintenance approach, the use of the word "segment" denotes a subset of the sector. Depending on each team's design approach, the subdivision of the sector is different.

The EU DEMO approach assumes each sector is divided into inboard (IB) segments and outboard (OB) segments called Multi-Module Segment (MMS). The combined weight for each IB and OB FWBS and structure segment is around 100 tonnes for the EU-DEMO. The lower divertor modules accessed through separate lower maintenance ports (number not available). The PPPL-AT Pilot Plant (4.0 m) and K-Demo (6.8 m) studies adopted the vertical approach with a minor modification. Both machines have 16 TF coils. They are using a semi-permanent inboard shield to be used for alignment, shielding and handling the disruption loads. They have segmented the blanket and use a vertical removal approach with 8 maintenance ports and 8 coolant manifolds. For each TF coil, there are two outboard modules and one inboard module. For the 16 coil case, there are 48 blanket segments to be removed plus four divertor segments associated with each TF coil, resulting in 64 divertor segments. This results in at least 112 segments to be removed to replace the entire power core. These designs are still evolving. One version of the Japanese Demo recommends vertical maintenance as opposed to the baseline horizontal maintenance approach. Their DEMO has 12 TF coils. They subdivided the Power Core into (36 segments in total) that are narrow enough to pass through the upper vertical maintenance ports and weigh about 130 tonnes each. Each segment has a complete poloidal structural ring upon which the blankets and shields are attached. The divertors are also included in the segment, so all the FWBS, structure and divertors are serviced at the same time.

Metrics to Compare and Assess Alternate Maintenance Schemes – There are several high-level key metrics that might be used to compare and evaluate the proposed fusion plant maintenance schemes. The intent in this assessment is to adopt addressable physical evaluation criteria that would equally apply to FNSF, DEMO and future power plants. It would unwise to adopt one maintenance approach on the early developmental facilities and then transition to a different approach for future machines, hence the common set of criteria. It is felt that the comparison criteria should not contain design issues, rather it should be related to physical attributes and how these attributes translate into positive or negative characteristics.
Thus the following addressable criteria are proposed.

- Number of replaceable PC units including divertors (fewer are better)
- The number of hydraulic connections (with cutting and rewelding) inside the VV should be either none or a very limited number (addresses time, reliability and contamination issues)
- The number of RH equipment needed inside the VV should be either none or a very limited number (addresses time, activation and contamination issues)
- Complexity of maneuvers and in-vessel equipment (addresses damage)
- Compatibility with H/CD systems, diagnostic systems and test blanket modules
- Easy and quick access to any sector or segment to replace a prematurely failed internal PC element (addresses unscheduled failures)
- Flexibility to alter downtimes (addresses achieving availability goals)
-

The two horizontal maintenance approaches and the three vertical maintenance approaches were compared and assessed using the addressable criteria.

Conclusions – The horizontal access maintenance approach with one full sector and port per TF offers the easiest, most expedient, most reliable, most flexible approach to achieve the required plant availability. The other approaches do offer the desired maintenance of the power core, but their solutions entail a higher level of technical risk, longer maintenance period durations and do not offer flexibility to accommodate access for special systems and an approach for higher levels of availability. Based on this assessment, it is believed that the horizontal maintenance option with one port per TF coil offers the most favorable power plant maintenance approach.

Use of Automated Remote Handling for Maintenance of the Fusion Core – The FNSF is envisioned to have a high flux fusion neutron power core that that will be well-shielded for long-term operation. As a result of this intense environment, these high energy neutrons will activate the power core materials that are harmful to humans. Therefore, the environment within the power core and extending out to the bio-shield cannot be accessed by humans either during operation or between operations for years. Thus the maintenance of all subsystems inside the bioshield cannot be done hands-on and must be remotely maintained and disassembled at the time of decommissioning. This approach applies to all large fusion devices including the FNSF, any DT DEMO plant and all following DT fusion power plants.
The remote assembly, maintenance and disassembly of the power core and its subsystems inside the bioshield can be accomplished with several means, including remote manipulators executed by skilled operators, special purpose robots guided by a combination of humans and computers or by autonomous robots with human oversight (command and control). However these remote operations require high precision to assure extremely reliable operations, very repeatable execution and very expedient maintenance actions to achieve high plant availability. These requirements suggest the exclusion of human actions would be advisable and recommended, starting with the construction of the FNSF project.

The technology of advanced robots to execute highly detailed, precise and dangerous operations has advanced tremendously over the past few decades. This trend will continue and accelerate, thus providing the necessary robotic database needed for fusion power core assembly, maintenance and disassembly. It has the advantages of lower labor cost and predictable and precise geometry control with repeatable movements. All of this can be accomplished without any human being exposed to dangerous environments inside the bioshield or hot cell. Obviously, the maintenance of a highly complex fusion experimental plant cannot begin operations with a completely autonomous robotic approach. Initially, there would likely be human oversight, but in the latter phases of FNSF, the maintenance actions would transition to be completely autonomous. These maintenance actions would include removal of the sector from the power core and transport of the sectors from the power core through the access corridors to the Hot Cell. Inside the Hot Cell, there would be other specialized automated robots to disassemble, inspect and reassemble the refurbished sector, segment or module. After the sector is completely quality verified, it can be returned and reassembled in the power core by the autonomous robots.

Conclusions – The need to safeguard the health of the plant workers mandates the use of robotic assembly, maintenance and disassembly of all fusion power core elements inside the bioshield. The use of highly automated, autonomous robotic assembly and maintenance operations will be commonplace when the FNSF (or the next large developmental fusion facility) is designed and built. This capability will enable faster, higher precision and more reliable assembly and maintenance of the power core that will result in a project cost savings and ensure the highest plant availability possible. These qualities are essential to the project success not only of FNSF, but also to all following facilities leading to the first fusion power plant. The use of autonomous robotic operations must be a keystone element of the FNSF design.

Assessing Design and Maintenance of the Test Blankets in FNSF – The need for a test blanket module (or a material test module) in the FNSF is a universally accepted idea. However, there are no conceptual design approaches on how to attach and maintain the FNSF Test Blanket Modules (TBMs) in the FNSF power core. Moreover it is thought that the TBM should be easy to access and quick to maintain (in much less time than removing an entire sector).

This qualitative analysis of how the TBM might be accessed and maintained discovered the process and time to remove a TBM is very complicated and lengthy due to the layers of protection inherent in FNSF or any high power fusion device. These barriers for the vacuum and shielding inhibit quick and easy access for either the sector or the test modules.
Although there is no design information about the TBM definition or how the TBM could be maintained, there are two general approaches on how the TBM might be maintained.

1) Removal of Test Blanket Module Through a TBM Port – The TBMs and the MTM would be inserted through the basic blanket structural ring and attached to the back surface of that structural ring. Port doors for the TBMs would be installed in all the vacuum vessel maintenance port barriers and the dedicated TBM lines would run through those barrier doors. This would require disconnecting the lines at each barrier so that each barrier door could be removed to access the next barrier door. This concept would be much more time-consuming with more disconnections and connections than removing an entire sector.

   A slightly different option of the distinct TBM port concept is to have all the port barrier doors and connections combined in a TBM port access module that would have only one set of line disconnects at the outside of the bioshield. Although this approach seems attractive, the multiple interior physical connections to the base hot structure, shielding door, vacuum vessel door and the bioshield was judged to be not feasible. This approach also complicates the sector removal.

2) Removal of Test Blanket Modules Without a TBM Port – In this case, there are no TBM port doors in the sector maintenance port barriers. Instead, the TBM lines are routed to the sector line disconnect area. The intent is to remove the maintenance port barriers per the normal sector removal process. Then the TBM lines could be disconnected and the TBM removed from the sector. This method would require a maintenance time very similar to that of a sector removal. The difficulty is that the TBM first wall alignment with the sector first wall might be very complicated, time consuming and result in poorer first wall alignment. So the TBM replacement time might be somewhat longer than the

sector replacement and the alignment of the TBM front surfaces would be difficult and time-intensive.

3) A more appealing option would be to keep the TBM attached to the sector and remove the entire sector to service the TBM. All the sector and TBM lines would be disconnected at the same time and location. The complete sector would be moved to the hot cell and a new refurbished sector with the new TBM would be returned and reinstalled in the same time as a sector. The TBM would be pre-aligned in the Hot Cell.

After analyzing all the options for TBM maintenance, it was concluded the most efficient and shortest TBM maintenance time is to replace the entire sector with the TBM. This approach is especially attractive as the sector replacement will be optimized, fewer maintenance equipment sets will be required and the fewest maintenance procedures will required. It also suggests the highest reliability because it has the fewest connections. It also assures the first wall alignment is correct as it is done in the Hot Cell. Thus it is recommended the TBM remain affixed to the sector for maintenance and no TBM ports are needed.

VI.I Critical Issues in Tritium Science, Tritium Research Needs in Support of FNSF

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Significant advances in tritium science are required for safe and successful operation of an FNSF. FNSF will need to burn 1-10 times the amount of tritium that ITER does (~100 times for DEMO) and will need to breed ~1000 times more tritium than the ITER TBMs to achieve TBR~1. This must be achieved while meeting the same stringent limits for radiation exposure and environmental release. Given the extremely mobile nature of tritium, this scale-up is a very significant technical challenge that will require a substantial investment in pre-FNSF R&D. We outline some of the critical issues in this are in this section.

Tritium retention

Tritium that is not burned must be extracted for reuse, and safety additionally demands that mobilizable tritium inventories remain low. It has been noted in the past that fuel retention in carbon PFCs (e.g. in TFTR and JET) was rather high (10-50%). This has prompted a shift away from carbon PFCs; ITER now plans to operate with a full-tungsten divertor and beryllium first wall. This configuration is now being studied in JET as a part of the ITER-like wall campaign, which has recently demonstrated reduced fuel retention by an order of magnitude in a D-D campaign.

Synergistic effects of plasma exposure and neutron irradiation remain to be addressed. Tritium retention is understood to be due in part to trapping of T atoms at defect sites within a structure. Neutron irradiation creates more such defects, and thus can increase retention by creating more traps. These effects are being studied in neutron-irradiated tungsten as a part of the TITAN collaboration. Results thus far suggest that trap densities continue to increase with fluence, that the traps are not saturated following plasma exposure, and that the trap energies are difficult to characterize. Significantly increased tritium retention was observed even as high as 500 °C plasma exposure. It should be noted that these observations were for specimens irradiated only at 0.025 and 0.3 dpa (and without thermal neutron shielding), so any functional relationship between trap density and dpa, and its applicability to fusion neutrons, is difficult to discern, and clearly there is a need for longer irradiations and continued investigations in this area.

From a safety perspective, the largest inventories are in the exhaust control, or fuel reprocessing plant. The throughput of the plasma exhaust stream for this tritium plant is large even for ITER. This throughput is driven by very low tritium burnup fraction in the plasma, <3% for ITER. Techniques to improve the burnup fraction must be developed. Increasing the burnup fraction will reduce the tritium plant throughput and inventory.

Tritium recovery from the blanket

For reasons outlined elsewhere in this report, the DCLL blanket concept is preferred for FNSF. The DCLL places some unique requirements on tritium management systems relative to ceramic breeders and even the HCLL, some of which are outlined below.

Tritium extraction from PbLi

The higher flow rates of PbLi in the DCLL (relative to the HCLL) in principle allow for lower circulating tritium concentrations in the PbLi. This, however, requires a reasonably efficient tritium extraction system. The vacuum permeator is the preferred concept for the DCLL. The vacuum permeator concept seeks to exploit tritium permeation through solids as a means of extraction. By maintaining a high vacuum outside the pipes containing tritium-laden PbLi, a concentration gradient across the pipe wall is established that drives permeation across it.

This is a very attractive concept, as it removes only tritium and in principle can be relatively compact. In order to be compact, however, materials with a very high tritium permeability must be used; it has been shown (as a part of this project) that more conventional materials (e.g. RAFM at its corrosion limit of 470 °C) require a permeator of enormous size. Group 5 metals such as vanadium, niobium, and tantalum offer the highest tritium permeabilities, but some technical challenges must be resolved by pre-FNSF R&D prior to their adoption. These include in particular a strategy to prevent oxidation, either by application of protective coatings or maintenance of oxygen partial pressures below 10^{-8} Pa. The effectiveness of this strategy, and the performance of the device overall, need to be verified beginning with small scale experiments and subsequently by larger ones.

Permeation and permeation barriers

High temperatures, and the low solubility of PbLi, will lead to significant permeation losses, to the helium coolant and elsewhere (e.g. secondary coolants, the building, and the environment). In addition to efficient extraction systems, a long-pursued strategy to mitigate this unwanted permeation is through the application of tritium permeation barriers. These typically consist of a low-permeability oxide layer coating applied to the structural material in uses (e.g. RAFM steel). While these have achieved dramatic permeation reduction factors (1,000-10,000) in small scale laboratory experiments, they have not performed as advertised in radiation environments, and a fundamental understanding of this is still lacking. If permeation barriers must be relied upon, further investigation in this area (including in radiation environments) must be undertaken.

VI.J. Neutron Dose Limits (dpa) for Blanket Structural Materials, and How This Limitation Impacts Power Plant Economics and Availability

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Introduction - There are several key requirements for the Fusion Nuclear Science Facility (FNSF). First, FNSF is expected to create a high power (high flux), 14-MeV neutron fusion plasma to test and validate in-core and out of core subsystems and systems. Second, it is mandatory for this plasma to be sustained for long periods of time to accumulate high neutron fluence levels on the power core and test modules to develop, understand and qualify both materials and subsystems for future use in the FNSF and DEMO. A third requirement is to deploy advanced structural materials that will not only perform and survive the intense nuclear environment but will perform safely in terms of accident response and exhibit favorable decay heat and waste disposal characteristics. Ultimately the structural (and other functional) materials will reach a neutron fluence at which their properties will degrade to a point that they must be replaced. Since fusion-relevant neutron sources do not exist, our knowledge of this exposure/damage limit is poorly predicted. Here we will review fission experience for guidance, structural material development to address the anticipated fusion neutron damage, and assess the impact of damage limits (200, 100, and 50 dpa) on the economics and availability of a power plant.

Brief Review of Fission Experience for Maximum dpa Levels – The only basis we have for making an assessment of the maximum neutron exposure or damage is the experience with fast reactor in-core components that was developed in the U.S., EU and Japan in the 1970-1995 timeframe. Over a period of 20-25 years of materials R&D, engineering design, fabrication technology development and component testing, these programs demonstrated that component lifetimes of 100-150 dpa could be achieved using austenitic stainless steels and Ni-base alloys for fuel pin cladding operating up to ~ 650°C and also for ferritic-martensitic steels as wrapper/duct material operating up to ~500°C. The problem with extrapolating this fission experience to fusion components is that the fission neutron energy spectrum is significantly different from the fusion neutron energy spectrum, leading to much lower gas production (helium and hydrogen) and transmutations than in fusion. In particular, the generation of helium, with the higher energy fusion neutrons, impacts swelling behavior, grain boundary cohesion and creep rupture life, fracture toughness, etc. to an extent that we cannot predict without adequate experience in a 14 MeV neutron environment. For comparison, in a fast fission reactor environment the maximum helium production is ~ 0.1-0.5 He appm/dpa, while in the fusion environment this value is ~ 10-15 He appm/dpa. The highest achieved dpa levels for the ferritic martensitic steel HT-9 in the fast reactor fission environment at 400-500°C, were ~ 200 dpa without failure, and incurring only 2% swelling.

In fusion power plant studies, the maximum damage level before replacing the first wall and blanket has been assumed to be at 200 dpa. This maximum damage occurs at the outboard midplane where the neutron flux is the highest. In spite of the fact that the structural material at other locations has not reached this limit, they all must be replaced. There is no current fusion relevant neutron source data to support this 200 dpa assumption, and, more importantly, the He generation from fusion neutrons is expected to aggravate the impact of damage significantly, thus adopting the fission experience maximum damage level is not a credible solution.

The present class of reduced activation ferritic martensitic steels (Generation I RAFM or RAFS), such as EUROFER(EU), F82H(JA), and CLAM(CH) are expected only to tolerate a maximum damage level of ~ 20-30 dpa, based on limited fusion neutron simulated experiments and computer simulations of damage and transmutations. Advances in the RAFM steels to oxide-dispersion strengthened (ODS) alloys and nano-scale (NS) particle strengthened alloys show promise in allowing higher temperature operation without the loss of strength, and providing some radiation resistance by trapping the He produced and thereby not allowing it to combine and form voids.

Based on the estimated damage capability of presently well-developed RAFM alloys of ~ 20-30 dpa, and the anticipation of increased radiation resistance modifications, the nuclear materials community would regard a ~100 dpa goal in a fusion environment as a credible challenge. The present RAFM alloys have been exposed to fission neutron spectrum damage levels of ~ 100 dpa so far. The incentive to pursue even higher levels is discussed later.

Current Status of RAFM Structural Materials – The most credible approach to establishing a structural material for fusion next step facilities is to build upon the promising radiation damage resistance that is a characteristic of the reduced activation ferritic steels (RAFMS), and to pursue efforts to widen their operating temperature range and to develop strategies for mitigating the potentially damaging effects of high helium concentrations. In parallel, the development of nano- structured oxide dispersion strengthened (ODS) steels produced by mechanical alloying offers the prospect of a revolutionary approach to mitigating radiation damage effects at high helium generation rates. In addition there is a potential for further alloying (using Al and/or Zr) to address the needs for improved corrosion resistance in liquid Pb-Li. The current set of potential FW/blanket alloys may be conveniently classified as follows.

Gen 1 RAFM: These are exemplified by the 8-9% Cr alloys, such as EUROFER, F82H and CLAM. The effects of radiation damage on the properties of these materials have been explored using fission reactors, heavy ion irradiation facilities and spallation neutron sources. While none of these facilities are capable of fully simulating the principal damage characteristics of the fusion neutron environment, a high level of scientific understanding has been established regarding the fundamental mechanisms of property degradation and a credible approach to the long term development of improved materials has evolved. The existing radiation effects database for RAFMS suggests that significant impacts on mechanical behavior and void swelling could occur in the fusion neutron environment when helium concentrations approach ~300 appm. For purposes of defining an R&D program for the development of increasingly robust FW/blanket components for FNSF (Fig.1) it is proposed to adopt a neutron damage limit of ~20 dpa (or 200 appm helium) for these alloys . Gen 1 RAFMS would be used for the FNSF Gen1 blanket which would be designed to survive through the 4th phase of FNSF operations.

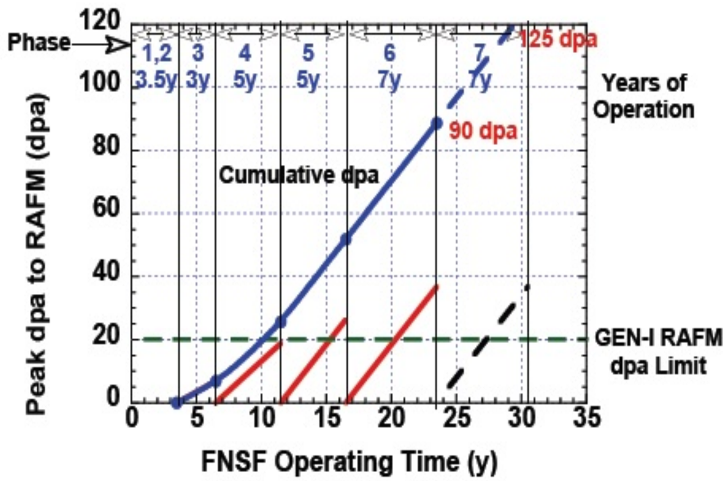

Fig.1 The peak fusion neutron damage level experienced by structural materials in the first wall and blanket at OB mid-plane of FNSF, in each phase, with Phase 3-7 being the DT fusion phases.

Gen II RAFMS: These alloys currently are at an early developmental stage in the U.S. and EU. Based on modifications to the concentrations of N, C, W and Ta currently used in the Gen1 RAFMs combined with controlled thermo-mechanical processing, the objective is to create a nano-scale sub-grain structure combined with a high-number density of nano-scale particles to provide improved high-temperature creep resistance up to 650°C combined with enhanced trapping of helium to inhibit swelling and migration of helium to interfaces. In this approach, the required nano-scale microstructure and properties are achieved via well-established conventional processing rather than via mechanical alloying. On the basis of the advanced nano-scale microstructure produced in these materials, it is proposed to adopt a projected neutron damage limit of ~50dpa/500appm helium for these materials. These advanced alloys would be deployed for Phase 5 of FNSF operations.

ODS (NS): These alloys are produced via mechanical alloying (powder preparation-ball milling-consolidation by extrusion or HIPping) and characterized by nano-scale (NS) microstructures with superior high-temperature strength and helium trapping capability. For both alloy variants, it is assumed that damage limits in excess of 65 dpa/650 appm helium are achievable and that high performance materials could be developed capable of surviving Phase 6 of FNSF operations and beyond.

There are basically two types of alloys in this group namely,

a) ODS alloys containing 9-12 % Cr which are transformable, i.e., they are ferritic–martensitic steels which are further strengthened by a dispersion of Y_2O_3 particles introduced via mechanical alloying, and EUROFER ODS is a leading example. Examples

of improved 9%Cr ODS alloys at an early stage of development are the 9Cr1WVTa (T2), (L. Tan, et.al, ORNL) and the 9YWTV-PM2 alloy (T. S. Byunm et. al., ORNL)
b) ODS steels with >12% Cr such as 14YWT which are non-transformable (i.e., they maintain an entirely ferritic structure at all temperatures). The oxide particle dispersion materials have remarkable thermal stability and these materials have the potential for operating at temperatures in excess of 700°C with superior resistance to off-normal temperature excursions.

Impact of dpa Restrictions on FNSF and Power Plant Performance -- It is recognized, in a broad sense, that the useful life of the power core structural elements will have a profound impact on the performance of any proposed high-power fusion facility. The current interest is focused on how the FNSF (or any near term large experimental facility) will perform its intended mission, while recognizing that the ultimate materials lifetime in the fusion neutron environment will also have a significant impact on how the commercial fusion power plants will operate in a competitive environment.

To address this issue, the ARIES series of 10th of a kind power plant designs and data were used as the baseline facility to assess the impact of various structural material lifetimes, characterized by their end of life dpa (damage) levels. In lieu of any quantified material data in the fusion neutron spectrum, the ARIES team historically had adopted a structural material lifetime of 200 dpa, which given a peak neutron wall load, would translate into a certain number of Full Power Years (FPY) before the component was replaced. For the ARIES-AT example used here, a 200 dpa material damage limit corresponds to 4 FPYs, and a lower limit of 100 dpa would correspond to 2 FPYs. Moreover, the present or very near term dpa limit might be as low as 50 dpa. So how do these limits impact the fusion plant performance?

Table 1, Cost Impacts for Decreasing First Wall & Blanket Structure Lifetime

ARIES-ACT2 uses RAFM power core structure	ARIES-ACT2	ARIES-ACT2	ARIES-ACT2
Structural Material lifetime dpa limit	200	100	50
Lifetime of replaceable FWB subsystem, FPY	4	2	1
Plant Lifetime, FPY	40	40	40
Number of FWB Replacements (Lifetime/FWB life-1)	9	19	39
Rplsbl FWB (no shield or divertor) costs, 2014M$	$98.44	$98.44	$98.44
Plant Lifetime FWB replacement cost, 2014M$	$885.99	$1,870.41	$3,839.27
Plant Total Capital Cost, escal (w/ Indir),2014$	$4,355.50	$4,355.50	$4,355.50
Plant Lifetime FWB Repl % of Total Cap Cost, %	20.3%	42.9%	88.1%
COE for Replaceable Components, 2014$. Mills/kWh	5.38	10.75	21.50
Total COE, 2014$, Mills/kWh	72.79	78.17	88.92
Replaceable COE/Total COE, %	7.38%	13.75%	24.18%

The dpa lifetime issues do not affect the Total Capital Cost of the plant. Rather, the reduced lifetime impacts the frequency of the scheduled maintenance intervals, thus increasing the operational costs and decreasing the plant availability, both of which have a negative impact on the Cost of Electricity (COE). For the baseline ARIES-AT case with its published database, the horizontal sector replacement scenario is adopted with two casks and two transporters utilized to remove and replace the sectors. Note that the ARIES-AT power core employs SiC-composite first wall and blanket (FWB) structural elements. The costs are higher, but the lifetime effect should be similar if this were a metallic structural material.

Table 1 illustrates the impact of reducing the power core lifetime in one half and one quarter. The baseline lifetime of 4 FPY is reduced to 2 FPY and then 1 FPY, before the FWB are replaced. The FWB replacement cost over the plant lifetime divided by the plant total capital cost starts out at 20% and then increases to 43% and then to 88%. It should be noted that these values are higher for SiC-composite structure used in ARIES-AT, and use of RAFM structural material would be lower, but still proportional. The values for the COE for the replaceable components (viewed as fuel costs) are 7%, 14% and 24%, respectively. However, the total COE varies from the reference case of 72.8 mills/kW-hr to 78.2 and 88.9.

The other impact of decreasing FWB lifetime relates to the Plant Availability. The shorter FWB lifetime demands more frequent replacement of these items, thus decreasing the plant availability. This effect is shown in Tables 2, 3 and 4. The baseline availability in ARIES-AT was computed to be 87.5%. Reducing the FWB life in half would reduce the availability to 86.5% and correspondingly, a quarter of the original lifetime would further reduce the availability to 84.6%. Although this does not sound like much of a difference, the 100 dpa case equates to 3.6 days per year of lost revenue and the 50 dpa represents about 10 days per year of lost revenue.

Table 2. ARIES-AT w/ 1/2 core replaced and 2 casks and transporters: FWB = Life 4 FPY

System Group Maintenance	Maintenance Days/FPY	System Availability
Power Core, Major, Scheduled	4.23	0.9885
Power Core, Minor, Scheduled	6.05	0.9837
Power Core, Unscheduled	20.56	0.9467
Reactor Plant Equipment, S+US	9.37	0.9750
BOP Equipment, S+US	9.37	0.9750
Plant Availability		0.8750

Table 3. ARIES-AT w/ 1/2 core replaced and 2 casks and transporters ; FWB Life = 2 FPY

System Group Maintenance	Maintenance Days/FPY	System Availability
Power Core, Major, Scheduled	8.47	0.9773
Power Core, Minor, Scheduled	6.05	0.9837
Power Core, Unscheduled	20.56	0.9467
Reactor Plant Equipment, S+US	9.37	0.9750
BOP Equipment, S+US	9.37	0.9750
Plant Availability		0.8651

Table 4. ARIES-AT w/ 1/2 core replaced and 2 casks and transporters; FWB Life - 1 FPY

System Group Maintenance	Maintenance Days/FPY	System Availability
Power Core, Major, Scheduled	16.93	0.9557
Power Core, Minor, Scheduled	6.05	0.9837
Power Core, Unscheduled	20.56	0.9467
Reactor Plant Equipment, S+US	9.37	0.9750
BOP Equipment, S+US	9.37	0.9750
Plant Availability		0.8459

Conclusions – The FNSF is intended to progressively provide higher neutron exposures over its lifetime, while simultaneously offering advances in the blanket and divertor operational environment (temperatures, pressures, flow rates). The structural materials will advance from the baseline RAFM to the most radiation-resistant alloys that are available. The purpose is to establish a database of the fully integrated set of components and subsystems in high power relevant environment, to prepare for the construction and operation of the DEMO and ultimately the first commercial power plant.

Due to the uncertainly in the expected operation lifetime (until material properties are degraded and the components are replaced) of structural materials in the fusion neutron environment, the reduction of expected structural material lifetime from 200 dpa (typical power plant study level) to 100 dpa, and to 50 dpa was examined. The present database from fission development, simulated fusion effects experiments, and computer simulations indicate that the fusion neutron damage and transmutation effects will induce significantly more material degradation than that from fission neutrons due to the more intense neutron energy spectra. It is expected that in the fusion neutron environment, a target value for maximum damage of ~ 100 dpa before replacement is a credible goal. On the other hand, the impacts of shorter first wall and blanket lifetimes on the economics and availability of power plants can be significant. The most negative impacts on the power plant were 1) increase in the operational cost of blanket replacements of 2x and 4x (resulting in a ratio of replacement costs to total capital costs to rise from 20% to 43 % to 88%) , 2) increase of the total cost of electricity of ~ 7% and 22%, 3) increase in the first wall and blanket nuclear waste of 2x and 4x. There is clearly an incentive to increase the maximum damage level by developing and qualifying fusion neutron radiation-resistant materials.

VI.K. Choices for Magnets in the FNSF

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1. Introduction

The Fusion Nuclear Science Facility (FNSF) is the first nuclear fusion device to provide an integrated fusion environment with fully integrated components to bridge the technical gaps of fusion plasma and fusion nuclear science between ITER and the demonstration power plant (DEMO). Both resistive copper magnet and low-temperature superconducting (LTS) magnet systems have been proposed in the past as a favourable choice for FNSF. In recent years, the high-temperature superconducting (HTS) magnets are gaining worldwide attention with considerable research and development activities. Better understanding of irradiation damage to conductors and insulation materials is required for both LTS and HTS magnet options. In addition, research and development programs to advance LTS and HTS superconducting technology while reducing system cost are essential for the successful development of magnets for FNSF, as well as for DEMO and power plants. We will discuss all magnet options as well as the design considerations in the radiation environment for FNSF magnets. We will also address the status of magnet radiation limits as they influence the size of in-vessel components.

The main advantage of the resistive magnet option is that less space is needed for radiation shielding. On the other hand, the resistive magnet system is not a power plant-relevant technology and will not give a long-term solution due to its large power consumption to drive the current and overcome the Joule heating loss. The LTS superconducting magnet system, using advanced Nb_3Sn wires, is the present day state-of-the-art magnet technology option following the ITER coil design and fabrication experience. The intrinsic upper critical field of Nb_3Sn and lack of high field pinning capacity (main reason of Nb_3Sn not reaching 80% of its intrinsic limit) limit the LTS magnet technology to a practical field of 16~18 T for both fusion and High Energy Physics applications. In addition, the availability of liquid helium for future supply may provide a limitation and could be a cost constraint for the LTS magnet option for FNSF and DEMO.

With recent success in developing and testing the high field solenoid HTS magnets using YBCO tapes at the National High Magnetic Field Laboratory (NHMFL) and the development of the Facility for Rare Isotope Beams (FRIB) accelerator magnets at BNL, the HTS magnet system can be a practical, long-term option for fusion. Under various constraints, however, significant progress in understanding the critical magnet design requirements toward the FNSF missions is necessary. Advanced high current cable design, practical joint development and the selection of irradiation tolerant structural and insulation materials with sufficient fracture toughness will be among the most challenging issues for HTS magnets.

2. Radiation Limits

Compared to ITER, the FNSF has higher field requirement and much higher neutron fluence to the magnet conductor and structures. The plasma from FNSF is "on" making neutrons for 7 times longer per year, and plasma pulses are 1000 times longer, therefore, much higher radiation dose rate is expected to the insulation materials. Radiation limits of all relevant materials are summarized in this section.

Figure 1 presents the radiation limits of binary and ternary Nb_3Sn LTS conductors. Sensitivity of binary and ternary Nb_3Sn to neutron radiation is quite different and the selection of ternary Nb_3Sn wires that can provide higher J_c and higher field for FNSF is not straightforward. To select advanced ternary Nb_3Sn for high field FNSF LTS magnets, we need to assess the impact on the machine size of its lower radiation limit (by an order of magnitude) compared to the binary pure Nb_3Sn with a relatively lower current density. Figure 1 also showed that J_c initially increases at low radiation limits for both binary and ternary Nb_3Sn but decreases significantly at high fast neutron fluence (10^{22} n/m^2 for ternary Nb_3Sn and 10^{23} n/m^2 for binary Nb_3Sn).

According to ITER heat and nuclear load specification, fast neutron fluence to the TF coils must not exceed $5x10^{21}$ n/m^2 in areas subjected to large stress fields and fast neutron fluence limit is 10^{22} n/m^2 where material is not stressed. The peak value occurs at inboard mid-plane where the total integrated fast fluence is $3x10^{21}$ n/m^2.

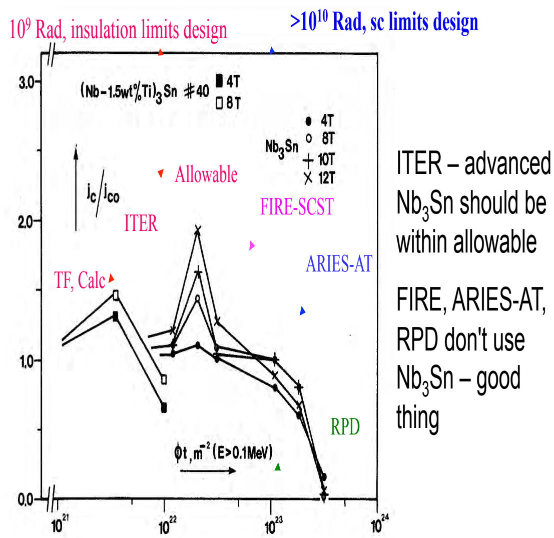
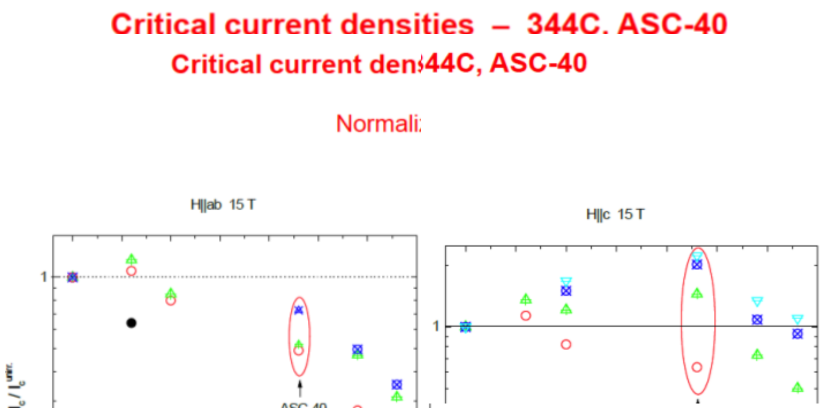


Fig. 1 Radiation limits of binary and ternary Nb_3Sn.

The HTS irradiation data is very limited. At present, Weber and Eisterer (Atominstitut, Vienna University of Technology, Austria) are leading the worldwide effort on irradiation test of HTS materials. The main conclusions include 1) Degradation of the critical current I_c of HTS is strongly magnetic field and temperature dependent and I_c degradation of YBCO tapes is significant at high field (>10 T) and high temperature (>60 K). 2) Reduction of critical temperature T_c, on the other hand, is insignificant compared to the I_c degradation. 3) The radiation limit of YBCO tapes will be driven by the reduction of I_c (not T_c) since most HTS magnets will be sub-cooled and operated at relative low temperature (20-40 K), far from the critical temperature (>90 K).

Figure 2 presents the radiation limits of American Superconductor (ASC) YBCO tape HTS conductors. Similar to the LTS Nb_3Sn conductor, Critical current initially increases at low radiation level for both parallel and perpendicular directions, but J_c starts to drop at 10^{22} n/m^2 fast neutron level for the in-plane parallel direction and $2x10^{22}$ fast neutron level for the perpendicular direction. Results also show higher than 50% I_c degradation for 60 K operation at the $2x10^{22}$ n/m^2 fast neutron radiation. This is probably unacceptable for FNSF application. Impact of radiation is better at lower temperature. The 30% Ic degradation for 40 K operation and $3x10^{22}$ n/m^2 may be acceptable for FNSF HTS magnets at below 40 K operation.



Weber, ASC-2014

Fig. 2 Radiation limits of ASC YBCO tape conductor.

Table I presents a summary of radiation limits for various LTS and HTS magnet materials. Fluence up to $5x10^{22}$ n/m^2 do not cause any change in the metallic material properties except for copper. Nuclear radiation, however, will cause martensitic transition in some austenitic SS resulting in significant embrittlement of the material. Table II presents the radiation limit specification established for fusion magnet design.

TABLE I. Summary of Radiation Limits of LTS and HTS Conductors, Copper Stabilizers and Insulations

Materials		Fluence	limit	
LTS	Binary Nb_3Sn	fast neutron fluence	10^{23}	n/m^2
	Ternary Nb_3Sn		10^{22}	n/m^2
HTS	YBCO	fast neutron fluence	$3x10^{22}$	
	Gd-123		$3x10^{22}$	n/m^2
Stabilizer	Copper	Fluence accumulation	$2x10^{21}$	n/m^2
		Warm-up requirement	10^{-4}	dpa
Organic	Epoxy	Radiation dose	10^6	Gy
	Polyimide (Kapton)		10^7	Gy
	CE/epoxy		$2x10^8$	Gy
	Hybrid		$5x10^8$	Gy
Inorganic	MgO		10^{11}	Gy
Structural		fluence	$5x10^{22}$	n/m^2

TABLE II. Radiation Limit Specification for Fusion Magnet Design

	Lifetime fluence (n/m^2)	Fast neutron fluence (n/m^2)	Insulation dose (MGy)	Fast neutron in winding pack (n/m^2)
ITER	$1x10^{22}$	10^{22}	10	10^{22}
ARIES-AT	$5x10^{22}$	10^{23}	<1000	10^{23}

FNSF		10^{23}	500	10^{23}

3. Magnet Options

3.1 Resistive Magnets

The resistive magnet option for FNSF requires much lower initial construction cost than the superconducting magnet options. However, a large amount of power is needed to compensate the resistive dissipation in the TF coil. This is particularly true for any fusion magnets with long plasma pulse operations. The fusion power of resistive magnet options can always be scaled up but at a larger coil size than ITER. The concept of jointed TF coils, which allows the simple vertical maintenance, enables a rapid change of the entire blanket and divertor, but there is limited space for radiation shielding to protect the center stack of spherical tokamaks, and possibly also PF coils in the tokamak.

Power consumption of resistive magnets is the limiting factor for any large-scale fusion devices with long plasma pulse operations. This is particularly true for the FNSF and DEMO that aim at long pulse or steady state operation. The cost of running NHMFL 20 MW resistive magnets is ~$1,500/hour. Table VI presents the study of electric power required for TF magnets if copper magnet technology is used for these fusion devices.

TABLE III. Summary of Electric Power Required for TF Magnets if Copper Magnet Technology is Used

	Field on Axis (T)	Plasma Duration (s)	Fusion Power (MW)	Electric Power (MW)
Tor supra	4.5	1,000	0	150
Jet upgrade	4	10	~20	~500
ITER	5.3	500	~500	~800-900

For FDF steady state plasma pulse duration of 2 weeks, the cost to run its resistive magnets of 300-400 MW could easily be ~10 million dollars per pulse.

3.2 LTS Magnets

No additional power consumption except small amount of power for cryogenics is needed for the LTS magnets. Higher current density and higher field can be achieved for the same size of coils compared to resistive magnets. LTS magnets are better for long time plasma operation as required by FNSF. The construction cost of the LTS magnets, however, is ~30% of the total machine cost. In addition, there is may be a limited availability of helium for cooling, or at least a higher cost, of the LTS magnets. The cost of helium has risen drastically in the past decade likely driving users to capture and recycle rather than release as has been the common practice. Moreover, more space is needed for thermal and radiation shielding of LTS coils. Better ways of integrating advanced insulations are also needed.

SULTAN test of ITER CS Nb_3Sn CICC showed significant degradation with both thermal and magnetic load cycling that is unacceptable for ITER. The cyclic load degradation problem was solved by adjusting the 1st stage cabling pattern to a short twist pitch. It is not fully understood about the mechanism of this performance degradation other than that it is related to filament fracture. SULTAN test of ITER TF CICC conductors also showed degradation after 1000 load cycling. This may be related to the strand properties. Unlike the CS conductor tests, this result of TF CICC performance degradation is directly relevant to the FNSF LTS magnet option since a few thousand load cycles are expected for the FNSF. Recent studies are focused on the correlation of the strand irreversible limits with the wire initial voids/defects induced stress concentration.

Due to the lack of high field pinning capability, the intrinsic limitation of Nb_3Sn magnets is at about 16 T for 4.2 K operation. Even if high field pinning capability in wires could be slightly improved, Nb_3Sn would still be intrinsically limited to below 20 T at 1.9 K. In practice, for the FNSF magnets with a peak field on the TF coils higher than that for ITER, design of LTS magnets using high J_c Nb_3Sn wires may be challenging as it will push Nb_3Sn close to its intrinsic field limit. In addition, availability of liquid helium for cooling of the LTS magnets may become limited and thus increase significantly operating cost for the FNSF LTS magnet option of long pulse plasma operation.

Recent studies also indicate that additional heating to ITER TF coils may be an issue. The peak nuclear heating to the ITER TF coils must not exceed 14 kW but the most recent studies showing that this is challenging to meet at the ITER neutron fluence level. In addition, the dose limit to insulation is 10 MGy. The maximum occurs in the front insulation of the coil at inboard mid-plane where the total integrated dose during ITER lifetime is ~3 MGy. These issues on ITER are due to a lack of proper neutronic design, which will be accommodated in the FNSF with its much stronger emphasis on the fusion nuclear environment.

3.2 HTS Magnets

It is reasonable to consider the HTS options with higher field limit and much higher temperature margin for FNSF. High energy margin is obtained for the high operating temperatures. The ideal operation temperature is in the 20-40 K range considering radiation limit, total heating to the coils, availability of helium, etc. With HTS magnets, the recent MIT proposal of options to build joints into winding that can be disconnected or reconnected on site, the disassemble and reassemble to allow for maintenance and change of internal components, and the demountable joints and coils becomes possible.

Compared to Nb_3Sn, the YBCO superconductor has a relatively low stress and strain limit but YBCO has a relatively low strain sensitivity, low temperature and field dependence to critical current density. HTS such as Bi2212 and YBCO are still a factor of 10-20 times the cost of Nb_3Sn. It is expected the cost gap can be reduced in the next decades for the FNSF. A robust cable design suitable for FNSF magnets is needed based on the TF coil winding structural configuration. Compared to Nb_3Sn, the YBCO has low stress and strain limits, low strain sensitivity and low temperature and field dependence on critical current J_c.

4. Conclusions

For any large-scale fusion magnet designed for long pulse plasma or steady state operation after ITER, copper magnets cannot be a long-term option due to the costly power consumption for long plasma operation. LTS magnet is the present-day state-of-the-art technology option. Initial construction cost can be reduced by conductor grading. The magnet materials with high radiation limits should be selected and tested. The ITER experience of CICC degradation over load cycling is not a critical issue for the steady state plasma operation. The intrinsic field limitation and lack of high field pinning capacity, as well as availability of liquid helium for future supply, will be the constraint for the FNSF LTS magnet option.

HTS magnet is costly according to present day price quote, but offers a potentially better long term option. Research and development needs for the FNSF magnets include advanced high Jc wire, cable design, joint for demountable coils and better structural materials. The YBCO irradiation resistance is better than the high J_c ternary Nb_3Sn but less tolerant than the binary Nb_3Sn superconductor.

VII. Conclusions

The FNSF is the critical break-in step for fusion energy development, offering a smaller facility to obtain the significant database over a broad range of integrated subsystems operating in the fully integrated fusion nuclear environment. It is very different from ITER although both devices require a burning plasma. The considerable complexity of the fusion nuclear regime can be understood by examining the many technical “surprises” found in the fission experience. Prior to operation of the FNSF the available data will not include the full integration provided by the FNSF, in particular the fusion neutron influence on all other phenomena (e.g. corrosion, gradients, material composition). This is the primary reason the smaller first step is chosen.

The FNSF study is beginning with the identification of the advances that the facility must provide, and quantifiable parameters to measure this progress against anticipated power plant parameters. An initial program on the FNSF has been established to clarify the steps and timeframes for progressing toward these mission goals. A deeper analysis of the blanket testing strategy has begun, assigning each sector a task in terms of its functionality (full phase life or partial), and whether it contains a TBM, H/CD, or material testing penetrations. In addition, backup blanket concepts to the primary DCLL have been determined to be the HCLL and HCCB/HCPB. The focus on helium cooled fusion cores has been established, avoiding water until outside the vacuum vessel.

The pre-FNSF R&D activities have been identified in terms of the topical science areas of 1) fusion neutrons, 2) tritium, 3) liquid metal breeder, 4) PMI/PFC, and 5) enabling technologies. Always in parallel with these activities is the predictive computational development. Each of these areas has been defined by high priority experiments required in preparation for FNSF. The evolution of this R&D leads to fusion neutron material testing facility(s), an integrated blanket testing facility, and an aggregate of facilities for testing the divertor and first wall/PFC components. The later ultimately converges on the DD phase of the FNSF itself where ultra-long plasma operation is developed before entering the DT phases. The R&D activities continue in parallel with FNSF to support its evolution through neutron exposure of structural alloys, operating temperatures, and design optimizations. Fusion neutron testing can continue into the DEMO phase to reach the high exposures at power plant levels. The enabling technologies also continue in support of the DEMO requirement for higher efficiency and reliability of all subsystems including the balance of plant.

A physics strategy is being developed in order to provide a basis for plasma parameter choices. In general, conservative choices are preferred in order to allow for very long pulse lengths without interruption for up to weeks in duration. A range of experimental tokamak accomplishments in duration, β_N, energy confinement, non-inductive current fraction, q_{95}, high density, elimination of ELMs, consistency with divertor, NTMs and low plasma rotation are being reviewed to understand the main trends to project to FNSF. Systems analysis is used to scan large areas of parameter space to identify attractive operating points for the FNSF. In addition, nearby operating points with higher or lower parameters are examined to see how the FNSF might be impacted (beneficial or not).

As part of describing the FNSF and what it must accomplish, several technical issues are being critically reviewed. These correspond to decisions that must be made, or at a minimum understood, for the FNSF definition. These can include what technologies to assume, what power plant parameters are credible from our present viewpoint, or what are the critical factors in a particular FNSF mission. These are referred to as white papers initially, and can become either project documents or publications. A number of these have been developed that include,

1) the use of water in the fusion core
2) technical readiness of helium as a fusion power core coolant
3) single-null versus double-null divertor configurations
4) material design interface for fusion power core components
4) the tritium breeding ratio
5) fusion nuclear waste management
6) the DCLL blanket concept as the primary blanket candidate for FNSF
7) maintenance approach, remote robotic handling and TBM maintenance
9) tritium research needs in support of the FNSF
9) fusion neutron dose limits and impact on power plant economics
10) choices for magnets on the FNSF

List of papers presented at the Technology of Fusion Energy (TOFE) conference (11/10-14, 2014), and submitted for publication in Fusion Sci. and Tech.

C. E. Kessel, et al, “The Fusion Nuclear Science Facility, the Critical Step in the Pathway to Fusion Energy”

P. Humrickhouse and B. Merrill, “Vacuum Permeator Analysis for Extraction of Tritium from DCLL Blankets”

J. P. Blanchard, et al, “Modeling the Thermo-mechanical Behavior of Plasma Facing Components”

S. Smolentsev, et al, “R&D Needs and Approach to Measure Progress for Liquid Metal Blankets and Systems on the Pathway from Present Experimental Facilities to FNSF”

Y. Zhai, et al, “Magnet Options for Fusion Nuclear Science Facility”

L. El-Guebaly, et al, “Breeding Potential and Blanket/Materials Testing Strategy for FNSF”

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Appendix 1: Missions and Metrics Table

The FNSF missions and metrics are described by a series of tables, one for each mission, where several parameters are used.

1. Strongly advance the fusion neutron exposure of all fusion core (and ex-core) components towards the power plant level

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
Life of plant peak FW fluence, MW-yr/m^2 (life of plant)	0.3	10 (6 FPY)	41 (16+ FPY)	88 (40 FPY)
Peak FW fluence to replace blanket, MW-yr/m^2 (dpa) (replacements)	0.3 (3) (0)	0.7, 1.9, 2.6, 3.7 (7, 19, 27, 37) (4-5)	3.7-15 (50-150) (4-5)	15-20 (150-200) (4-6)
Peak FW neutron wall load, MW/m^2 (average)	0.76 (0.56)	1.5 (1.0)	2.5 (1.67)	2.2 (1.46)
Peak Structural Ring damage, dpa (appm He)				
Peak VV damage, dpa (appm He)				
TF Magnet E>0.1 MeV fluence (n/cm^2), heating (mW/cm^3), insulator dose (rads), Cu damage (dpa)				10^{19} 2.0 10^{10} $6x10^{-3}$
Divertor peak fluence, MW-yr/m^2				
Divertor damage, dpa (appm He)				
Cryostat damage, dpa (appm He)				

Bio-shield dose, mrem/hr				0.25

2. Utilize and advance power plant relevant materials in terms of radiation resistance, low activation, operating temperature range, chemical compatibility and plasma material damage resistance

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
First wall	Be			ODS Fe-steel*
Blanket structural				Fe-steel*
Breeder				LiPb
Blanket Coolant				He/LiPb
Shield/Str ring structural material	316 SS / CuCrZr			Fe-steel*
Shield/Str ring coolant	H_2O			He
Shield/Str ring shielding/filler				Borated Fe-steel*
Vacuum vessel structural material	316 SS			Bainitic steel
Vacuum vessel coolant	H_2O			He
Vacuum vessel filler	SS304-borated SS430 ferritic			
Ex-VV shield				Fe-steel*, H_2O, borated Fe-steel
Divertor structural	316SS/CuCrZr			W-alloy/ODS Fe-steel*
Divertor armor	W			W
Divertor coolant	H_2O			He

3. Operate in power plant relevant fusion core environmental conditions including temperatures, coolant/breeder flow rates, pressures/stresses, hydrogen (tritium), B-field, and neutrons, and with gradients in all quantities.

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
First wall structure	100-150 ^{o}C			550 ^{o}C
First wall coolant				
Blanket structural				350-550 ^{o}C
Breeder				460-647 ^{o}C
Blanket Coolant				385-470 ^{o}C
Shield/Str ring structural material				350-550 ^{o}C
Shield/Str ring				380-385 ^{o}C

coolant				
Shield/Str ring shielding/filler				
Vacuum vessel structural material	120 ^{o}C			~500 ^{o}C
Vacuum vessel coolant	100 ^{o}C			
Vacuum vessel filler				
Ex-VV shield				RT
Divertor structural	150-400 ^{o}C			800-1300 ^{o}C
Divertor armor	700-1100 ^{o}C			~1500 ^{o}C
Divertor coolant	100-150 ^{o}C			676-720 ^{o}C

4. Produce tritium in quantities that closely approaches or exceeds the consumption in fusion reactions, plant losses and decay.

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
TBR - total				1.05
Tritium produced/year	4 g			101-146 kg
Li-6 enrichment				40%
OB FW hole/loss fraction				4%
Tritium lost to decay, kg/year				0.3
Tritium lost to environment, kg/year				0.004

5. Extract, process, inject and exhaust significant quantities of tritium in a manner that meets all safety criteria, requiring a high level of inventory prediction, control, and accountancy.

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
Tritium extraction efficiency				
Tritium leakage rate thru HX, kg/yr				
Tritium inventory in FW/B/S structures				
Tritium inventory in Breeder				
Tritium inventory in main coolant (He)				
Tritium inventory in VV				
Tritium inventory in processing				

Tritium inventory in storage				
Tritium fuel rate into plasma chamber				
Tritium exhaust rate from plasma chamber				
Tritium burnup				

6. Routinely operate very long plasma durations, much longer than core plasma time constants and long enough for nuclear, chemical, and PMI processes to be accessible, at sufficient plasma performance to advance the fusion nuclear mission, generally considered to be days to weeks.

	ITER	FNSF	DEMO	Power Plant ACT1/ACT2
Plasma on-time per year	5%			85%
Plasma pulse duration, s	500-3000			2.7×10^7
Plasma duty cycle	25%			100%
$\beta_N H_{98} / q_{95}$	0.6			0.4-2.1
Q	5-10			25-48
f_{BS}	0.25-0.5			0.77-0.91
$P_{core,rad}/(P_{alpha} + P_{aux})$	0.27			0.28-0.46
$P_{div,rad}/P_{SOL}$	0.7			0.9

7. Advance and demonstrate enabling technologies that support the very long duration plasma operations with sufficient performance and reliability to project to DEMO and a power plant, including heating and current drive, fueling/pumping, particle control, PFC lifetime, disruption avoidance and mitigation, plasma transient mitigation, feedback control, diagnostics, etc.

		ITER	FNSF	DEMO	Power Plant ARIES-ACT2
H/CD source	NB				
Max $P_{H/CD}^{total}$, MW		33			65-80
H/CD max injection duration, s		500-3000			2.7×10^7
Source operating lifetime, years					
Source availability					
η_{CD} ($n_{20}RI/P$)					0.35
$\eta_{wall\text{-}plug}$					0.4
$\eta_{coupling}$		1.0			1.0
H/CD source	EC				
Max $P_{H/CD}^{total}$, MW		20			20
H/CD max injection duration,		500-3000			2.7×10^7

s					
Source operating lifetime, years					
Source availability					
η_{CD} (n_{20}RI/P)					0.18
$\eta_{wall\text{-}plug}$					0.4
$\eta_{coupling}$					1.0
H/CD source	ICRF				
Max $P_{H/CD}^{total}$, MW		20			30
H/CD max injection duration, s		500-3000			2.7×10^7
Source operating lifetime, years					
Source availability					
η_{CD} (n_{20}RI/P)					0.25
$\eta_{wall\text{-}plug}$					0.4
$\eta_{coupling}$					
H/CD source	LH				
Max $P_{H/CD}^{total}$, MW		0			30
H/CD max injection duration, s					2.7×10^7
Source operating lifetime, years					
Source availability					
η_{CD} (n_{20}RI/P)					0.26
$\eta_{wall\text{-}plug}$					0.4
$\eta_{coupling}$					
Fueling source	HFSpellet				
Total DT fuel particle rate, /s					
η_{fuel}					
	LFSpellet				
Total DT fuel particle rate, /s					
η_{fuel}					
Pumping	Divertor/cryo				
Total exhaust particle rate /s					
η_{He}					
Cryo regeneration frequency					
Disruption mitigation	Pellet/frac pellet (Ar/D)				
Unmitigated disruptions/year					
Mitigated disruptions/year					

TF Coil LTSC/Nb_3Sn					
B_T at R_o, T	5.3				8.75
B_T^{max} at TF, T	11.5				14.4
$\langle j_{TF}\rangle_{winding,}$ MA/m^2					
$\langle j_{TF}\rangle_{total}$, MA/m^2	11.5				13.0
CS Coil LTSC/Nb_3Sn					
Max B	13.0				
Max $\langle j_{CS}\rangle$	14.0				
PFC divertor	W armor / CuCrZr str				
$q_{peak,}$ MW/m^2					
$q_{trans}/\Delta t_{trans}$, MW/m^2-s					
Max erosion rate, mm/yr					
Lifetime to replace, years					

8. Demonstrate safe and environmentally friendly plant operations, in particular with respect to tritium leakage, hot cell operation, onsite radioactive material processing and storage, no need for evacuation plan and other regulatory aspects.

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
Peak FW specific activity, Ci/m^3				
Radioactive waste classification				
Peak decay heating, MW/m^3				

9. Develop power plant relevant subsystems for robust and high efficiency operation, including heating and current drive, pumps, heat exchanger, fluid purity control, cryo-plant, etc.

	ITER	FNSF	DEMO	Power Plant ARIES-ACT2
Q (P_{fus}/P_{aux})	5-10			25
Q_{engr} ($P_{elec,g}/P_{recir}$)	0			3.0
η_{th}	0			45
$\eta_{H/CD,wall\text{-}plug}$	~0.4			0.4
η_{pump}^{He}				0.9
η_{pump}^{LiPb}				

10. Advance toward high availability, including gains in subsystem and component reliability, progress in capabilities and efficiency of remote maintenance operations,

accumulation of reliability and failure rate data that can be used to project and design future systems.

Appendix 2: DEMO Program Table

The demonstration power plant in the U.S. definition requires that no technical gaps remain at end of its operation with respect to all plant systems. Ths does not mean that the DEMO must be a full size power plant, but it does mean that plasma physics and technologies are all established, and that scale-ups of systems are only acceptable if they are with high technical confidence. At present a detailed definition of the DEMO does not exist. Power plant studies provide the targets for all parameters. The program provided below is intended to provide the connection to the FNSF and the demonstration required by industry and utilities that fusion power is reliable, safe, and economically viable.

	He/H		DD		DT		DT		DT		DT	Power Plant
Phase	1		2		3		4		5		6	
Phase time, yr	1	?	3	?	6	?	6	?	8	?	8	35-47 years/ 30-40 FPY
N_w^{peak}, MW/m^2					2.5		2.5		2.5		2.5	2.0-3.25
Plasma on-time per year (days)			35-75% (128-274)		35% (128)		50% (183)		67% (245)		75% (274)	85% (308)
Plasma duty cycle (days on/days off)			0.95 20-90/1		0.95 20/1		0.98 40/1		0.98 60/1		0.99 90/1	1.0
Operation / Maintenance per year (days)					135/230		188/177		249 / 116		277 / 88	308/56
End of phase peak fluence (MW-yr/m^2)					5.25		7.5		13.4		15.0	15.0 to replace
Cumulative peak fluence, MW-yr/m^2					5.25		12.75		26.15		41.15	60-130
End of phase peak damage (dpa)					52.5		75		134		150	150 to replace

Cumulative peak damage (dpa)					52.5		127.5		261.5		411.5	600-1300
Total # plasma cycles					38		27		32		24	

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