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THE FUSION NUCLEAR SCIENCE FACILITY, THE CRITICAL STEP IN THE PATHWAY TO FUSION ENERGY

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The proposed Fusion Nuclear Science Facility (FNSF) represents the first facility to enter the complex fusion nuclear regime, and its technical mission and attributes are being developed. The FNSF represents one part of the fusion energy development pathway to the first commercial power plant with other major components being the pre-FNSF research and development, research in parallel with the FNSF, pre-DEMO research and development, and the demonstration power plant (DEMO). The Fusion Energy Systems Studies group is developing the technical basis for the FNSF in order to provide a better understanding of the demands on the fusion plasma and fusion nuclear science programs.

I. INTRODUCTION

The proposed Fusion Nuclear Science Facility (FNSF) is the first strongly fusion nuclear facility in the US 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, fusion neutron fluence on the blanket, for example. The pathway from ITER to the first commercial power plant, including the FNSF, is considered a 2-facility development path, with the DEMO providing both the routine electricity demonstration, and any remaining technical advances not provided by the FNSF. The approximate time frame for the FNSF to begin operation is during the ITER operation, likely after the first DT plasma demonstrations, however, there are no commitments to this time frame at present. Figure 1 shows this development pathway schematically. The

FNSF will advance several parameters toward a fusion power plant, but all may not reach these levels, requiring that the DEMO provide the platform for further advance. Meanwhile, at the end of the DEMO program, there can be no further technical gaps remaining, so that a commercial power plant can be pursued with high technical confidence by utilities. The FNSF requires a substantial R&D program¹⁻³ as pre-requisite to its design, construction and operation.

Based on fission experience, the Shippingport nuclear power station, considered the DEMO for fission, which started in 1958, operated for ~ 30 years before being decommissioned, generating about 60 MW electric, and utilizing three separate core configurations.⁴ Over the course of the following decade 12 other power stations were built⁵ and operated with electric powers ranging from 12-210 MW. Only 2 years elapsed after Shippingport began before two other stations were constructed, indicating that the successful demonstration of routine power plant operations was compelling to other utilities or utility consortiums. Finally after a decade of fission electricity production, the unit electric power output was scaled up to > 500 MW. There were many research fission reactors in support of the DEMO step at various laboratories prior to 1958. With only 2 facilities in the pathway to a large fusion commercial power station, the critical need for the FNSF and the DEMO in combination, and the intimate connection of their programs cannot be over emphasized.

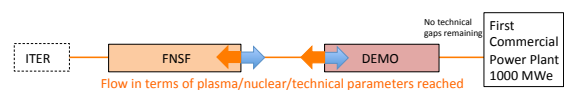


Fig. 1. Schematic fusion energy development pathway in terms of parameters reached toward a power plant, and not chronological.

II. DISTINGUISHING THE FNSF MISSION FROM THE ITER MISSION

ITER (Ref. 6) 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 scale 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 in tritium handling, cryogenic plant systems, magnet and subsystem power distribution, and activated material handling. 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 reach their missions. The FNSF is intended to pursue the fusion nuclear aspects, and therefore requires a sufficiently high performance plasma with very long durations. The neutron fluence at the outboard (OB) first wall over the device's lifetime would reach levels of 15-25x 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) to accommodate the higher neutron exposure; the operating temperature for the fusion core structure and coolant exit would be 1.8x and 3-6x higher, respectively, to target conditions for electricity production; tritium is bred for sustainment of the fuel cycle while it is externally supplied to ITER; the plasma pulse durations will need to be 30-1000x longer than ITER, and the total plasma on-time in a calendar year would need to be 7x higher. The maintenance approaches for the FNSF will move toward few larger pieces to facilitate the fusion core component (blanket, divertor) demonstrations with rapid replacement, inspection, and overall availability.

III. THE FNSF WILL BE SMALLER THAN THE DEMO OR POWER PLANT

The proposed FNSF is intended to be smaller than a DEMO, and even than ITER. The primary reason for this is to reduce cost and allow a gradual program to break in to the complex fusion nuclear regime at the minimum scale allowed, depending on technology choices. This is balanced against maintaining a two facility step to the first commercial power plant, so that there is some resistance to very small and low mission scope facilities. Figure 2 shows the fusion power as a function of the plasma major radius for several DEMO, engineering test reactor, and FNSF proposals, all for conventional aspect ratio tokamaks. The operating space where solutions are being examined for this study lies inside the curves, with the lowest radius solutions identified by the large oval. The EU DEMO (Ref. 7),

Korean K-DEMO (Ref. 8), and Chinese CFETR (Ref. 9) are shown, as well as the ARC (Ref. 10) high temperature superconductor design, and the copper coil Fusion Development Facility, FDF (Ref. 11). The FNSF operating space displayed is defined for low temperature superconducting magnets, with the constraints on β_N , q_{div}^{peak} , and $N_w^{OB,peak}$ noted.

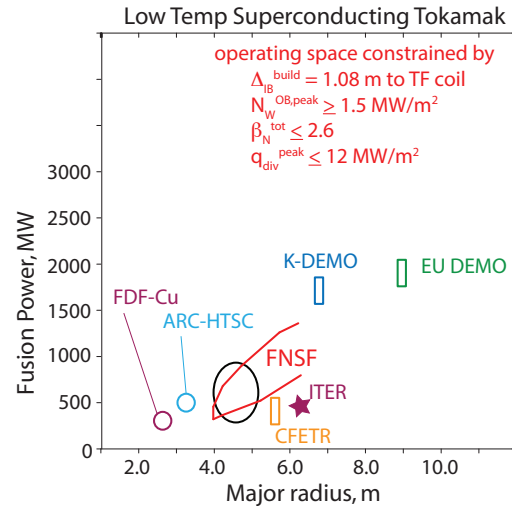


Fig. 2. Fusion power versus plasma major radius for various DEMO and fusion nuclear testing facility proposals, and the operating space of solutions determined for this study for the FNSF.

The choices for magnet technology and aggressiveness in plasma or other technology features can 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 endured by the fusion core components (which are replaced in each phase, Table I), 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 fully integrated component testing (blanket, divertor, launcher or other). None of these can provide the entire environment, and tend to be missing significant factors in each case.

The impact of a new environment in a nuclear system can be understood from the fission PWR and breeder program experience. Concentrating on material “surprises” described in Ref. 12, 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. It was found that swelling from void formation in an austenitic steel was not observed for temperatures below 302C, but it was observed when above 307C, over the irradiation dose range of 3-57 dpa, indicating a very strong dependence of swelling on temperature. For the same neutron dose, a lower neutron dose rate could lead to increased hardening in pressure vessel steels, while a lower dose rate in 304 stainless steel resulted in earlier onset of swelling from 20 dpa to 5 dpa. Small constituents in steels (e.g. 0.5 wt %) such as Cu in a reactor pressure vessel steel, could lead to drastically lower ductile crack propagation energy with a shift to higher DBTT, while Si content at ~ 1 wt % in an austenitic stainless steel led to severe intergranular stress corrosion cracking (IGSCC) and increased crack growth rates. In other cases the addition of 1 wt % hafnium to 316 SS led to an increase in the dpa (from 2 to 50) before the onset of void formation in a Fe-Cr-Ni alloy. Surface conditions and 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 precise issues are not likely to be those that arise in the fusion environment, however, they give some perspective on the impact of the actual environment which may not be testable in advance in individual material testing in a fusion or a fission neutron facility. The additional effects of transmutation He and H produced in the fusion neutron spectrum, as well as the resulting transmutant, at high operating temperatures, on a wide range of fusion blanket and divertor materials will undoubtedly provide a series of new challenges when exposed to the fully integrated environment.¹³ The sensitivity of various phenomena to temperature, neutron dose, neutron dose rate, He/dpa ratio, stress, and many other factors can not be over-emphasized. As noted in Ref 13, several critical material behaviors led to “major disruptions in the development program” for the liquid metal fast breeder program. These only involve material effects, while thermo-mechanics, thermal hydraulics, mass transfer, and fluid MHD can provide additional complications. In light of these observations it is prudent to pursue the break-in to the fusion nuclear regime with a smaller size facility. The goal for the FNSF is to establish a database on all components in the facility up to relevant parameters (e.g. fluence reaching 40-60 dpa, blanket temperatures reaching 500-600C) before proceeding to larger size in the DEMO.

IV. THE MISSIONS AND METRICS FOR A FNSF

The missions identified for the proposed 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 approach or exceed 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 by several metrics for determining how far they are advanced

relative to a power plant. The FNSF would be expected to advance most or all missions significantly, however, any shortfalls would have to be accommodated in the DEMO facility, before it could move to routine power plant operations. Underlying all of these missions are significant scientific activities involving complex measurements in a harsh environment, component inspections, autopsy, and identification, integrated simulation validation, and multi-physics behavior never observed in advance. The FNSF will serve as a significant materials test platform, adding critical information to materials and design databases and criteria. The development of multi-discipline integrated simulations, e.g. combining neutronics, thermo-mechanics, CFD, liquid metal MHD, and tritium diffusion/convection, is important as the FNS program approaches the FNSF operation, and the operation of the FNSF itself provides the penultimate validation platform for such sophisticated modeling. The plasma pulse lengths will be the longest ever obtained in a tokamak confinement facility, and provide an unprecedented view of the simultaneous core, scrape-off layer, and plasma material interface coupling at representative fusion power levels (several 100 MWs). The DD plasma phase would provide a strongly diagnosed platform for integrated plasma simulation validation from plasma core to the plasma facing wall. In the DT phase this diagnostic coverage would be reduced and the simulation capability would be relied on to augment the resulting lower resolution.

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,

1. Peak first wall fluence before replacing the blanket (MW-yr/m², dpa)
Peak first wall neutron wall load (MW/m²)
2. FW/blanket structural material
VV coolant
3. $T_{str,blnkt}$, $T_{LiPb,blnkt}$, $T_{He,blnkt}$, $T_{str,VV}$, $T_{He,VV}$, $T_{W,div}$, $T_{He,div}$, v_{LiPb} , $P_{He,blnkt}$, $P_{He,div}$, etc.
4. 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 H/CD power
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, 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. As shown in Fig. 1, what the FNSF does not accomplish in its program (e.g. reaches a peak dpa at the FW of 37 relative to a target of ~ 100-150 dpa required in a power plant) the DEMO facility must absorb into its program before moving to routine power plant operations, and ultimately ending with no technical gaps to the commercial fusion power production. The separation of possible FNSF into three categories is artificial, but useful to identify the most critical differences among them. These differences tend to be most significant in 1) maximum neutron fluence or dpa reached, 2) magnet technology, 3) electricity production, 4) maintenance approaches, 5) tritium breeding ratio, 6) divertor technology, and 7) years of operation. Meanwhile the access to long plasma durations, and high duty cycles is considered accessible to all configurations. The minimal FNSF configuration would likely cost the least and provide an early view of fusion neutron exposure of blanket components, while the maximal configuration would be pursued if the primary objective is to demonstrate net electricity, and likely cost the most. This study will concentrate on the moderate configuration with systems and detailed physics and engineering analysis, while the minimal and maximal configurations will be examined in systems analysis only.

For the moderate FNSF 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, with the possibility to reach 75 dpa. The use of water inside and including the VV 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 could operate for ~ 25-30 years to accomplish its mission scope. The power plant relevant maintenance options of vertical sub-sector or horizontal sector will be examined.

V. THE PROGRAM ON THE FNSF

In order to better understand what the proposed FNSF must accomplish, a program has been identified

with a series of phases and estimated time-frames. The moderate FNSF mission scope is assumed here. Shown in Table I 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 demonstration. This is followed in the nominal program by 4 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/shield, divertor, and special PFC (launchers) evolution to higher performance parameters. The neutron fluence buildup 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 is assumed to be the Dual Coolant Lead Lithium (DCLL) design due to its favorable power plant performance and perceived near term development. This has an 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 RAFM and to a nano-structured RAFM. Simultaneously the blanket temperature is 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.

The blanket testing strategy is prescribed by sectors. Each sector is defined by 1) blanket type, 2) structural material and operating temperature ($T_{\text{LiPb}}^{\text{outlet}}$), 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 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 lithium¹⁴ (HCLL) and helium cooled ceramic breeder or pebble bed¹⁵ (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 rejected for use in the fusion core¹⁶ (inside and including the vacuum vessel). With few other coolants

being considered viable, the backup blankets also use helium like the main DCLL blanket concept. The commonality of the primary and backup blanket concepts, in terms of structural material and coolant, lend credibility to being able to support these concepts within the DCLL design of the service manifolding and maintenance of the facility. These backup blanket concepts are modular in nature, while the DCLL concept can be modular or fully poloidal within a sector. The detailed assessment for the primary and backup blanket sectors has not been performed yet.

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 difficult 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 blanket (or other component) is 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 number of years of operation are 31.5 including the extra phase, and might require additional time between phases when all sectors are typically replaced. This includes ~8.4 years of DT plasma on-time. The maintenance time requirements in later phases may be reduced as the procedures become more routine. The organization of the maintenance time within a phase must be optimized to provide the needed time during plasma operations and when not in plasma operations. It may be possible to accelerate the evolution to the longest plasma on-time in DT operation (~ 10 days), say establishing them rapidly in Phase 3, and obtaining 35% on-time and 95% duty cycle for Phases 4-7. The neutron wall loading may be increased by operating at a higher plasma beta, as discussed in the Sec. VII, and this can accelerate the neutron exposure and shorten facility time frames. Although not discussed here, activities associated with divertor and special PFCs (launchers) optimization will also be taking place on the facility. The present program plan for the proposed FNSF will be revisited often as part of the study to better establish timeframes and activities during all phases of the plan.

VI. RESEARCH AND DEVELOPMENT BEFORE THE FNSF

A significant research and development program must precede the proposed FNSF, and the philosophy for this study is to prevent failures on the facility to the maximum extent possible. This implies that thorough qualification of the all facility components, fusion core in particular, are carried out in advance of installation on the

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, 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. A range of these issues are described in Ref (1). Shown in Fig. 4, 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 persist into the DEMO, such as enabling technology and fusion neutrons. The synchronization of the R&D programs with the major fusion facilities is determined in order to establish the deliverables for the R&D, and guarantee the progress toward a power plant.

VII. PLASMA PHYSICS STRATEGY FOR THE FNSF

The approach to the physics operating point(s) in the proposed 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, so that it is the combination 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.

The β_N^{total} ($\beta_N^{\text{th}} + \beta_N^{\text{fast}}$) 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,²³ where a range of current profiles from bootstrap, lower hybrid, neutral beam, and ICRF fast wave were examined. Shown in Fig. 5 is the stable β_N versus $I_i(1)$ 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 I_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 I_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 experiments²⁴⁻²⁶ have demonstrated ratios exceeding 1.0 while maintaining reasonable energy confinement in the plasma ($H_{98} \leq 1$). This is achieved 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 conductor²⁷ for the elongated plasma is made of tungsten and located at $b/a = 0.33$, with poloidal extent from about 45-90° 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 coils²⁷ are made of inorganic insulated Cu and located behind the shield/structural ring on the OB side, but inside the vacuum vessel.

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.²⁸ 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/m², since He cooled designs²⁹ have been identified as being capable of peak heat fluxes ≤ 15 MW/m² with acceptable pumping powers. There is considerable uncertainty in the power scrape-off width prediction, 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 regime³⁰ 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-divertor³¹ or snowflake³², 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.

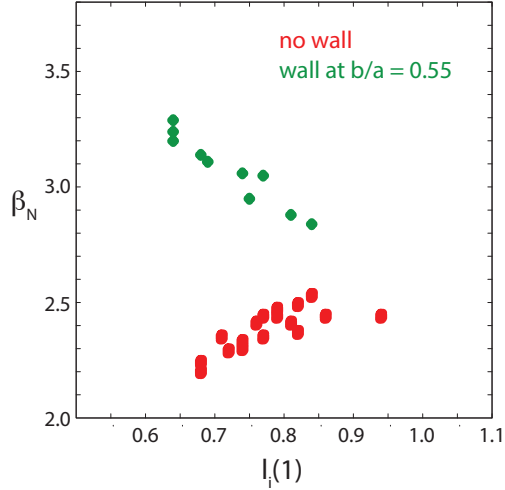


Fig. 5. The maximum β_N limit as a function of current profile peaking (I_i), with (upper) and without (lower) wall stabilization and feedback of RWMs, from ARIES-ACT2 study.

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 $\eta_{CD(20)} (n_{20}RI/P) = 0.2 \text{ A/W}\cdot\text{m}^2$. Compared to recent ARIES-ACT2 studies²³ 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 projections³³ 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 plasma³⁴.

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 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, $\tau_{CR} = \mu_0 a^2 \kappa / 12 \langle \eta_{neo} \rangle$, where $\langle \eta_{neo} \rangle$ is the volume average neoclassical resistivity, and the longest tokamak discharges relative to this are $\sim 15 \tau_{CR}$ in JT-60U.³⁵⁻³⁷ 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 $\beta_N \sim 2.6$, $H_{98} \sim 1.0$, $n/n_{Gr} \sim 0.55$, and $f_{BS} \sim 0.43$. These discharges avoided neo-classical tearing modes (NTMs) by operating at low $q_{95} \sim 3.2$, 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. Resistive wall modes (RWMs) were observed in these discharges. Plasmas with $\beta_N \sim 2.4$, $H_{98} \sim 1.0$, $f_{BS} \sim 0.45$, $f_{NI} > 90\%$, and minimum safety factor $q_{min} \sim 1.5$ were maintained for 2.8 τ_{CR} . Using reversed shear plasmas, f_{NI} reached 1.0, with $f_{BS} \sim 0.8$, $H_{98} = 1.7$, $q_{95} \sim 8$ and $\beta_N \sim 1.7$, 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 $\beta_N \sim 2.1$.

DIII-D has obtained $\beta_N \sim 3.1-3.4$, $H_{98} > 1.2-1.3$, $q_{95} = 5.0-5.5$, $f_{BS} \sim 0.6$, $f_{NI} \sim 0.8-1.0$ and sustained them for $\leq 1 \tau_{CR}$.³⁸ More recently³⁹⁻⁴⁰ with off-axis neutral beam injection plasmas have reached $\beta_N \sim 3.5$, $H_{98} > 1.0$, $q_{95} = 6.7$, $f_{BS} \sim 0.4-0.5$, $f_{NI} \sim 0.75$ 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 $\beta_N \sim 2.0$, $H_{98} = 1.3$, $q_{95} = 4.6$ 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 $f_{BS} \sim 0.4$, 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.

VIII. PRELIMINARY SYSTEMS ANALYSIS FOR A LOW TEMPERATURE SC MAGNET FNSF

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,⁴¹ properly scaled for the FNSF inboard fluence. Here 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/m², and the peak field at the TF coil is restricted to be $\leq 15.5 \text{ T}$. This maximum field with

Nb₃Sn is being pursued by K-DEMO (Ref. 8) 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 (18). It is also assumed that 90% of the SOL power is radiated in the divertor in a partially or fully detached regime.³⁰

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% ($Z_{\text{eff}} = 1.5-2.65$), 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 $n(0)/\langle n \rangle = 1.4$, temperature profile $T(0)/\langle T \rangle = 2.6$, global particle confinement time $\tau_p^*/\tau_E = 5.0$, and current drive efficiency at 0.2 A/W-m². The filters used to isolate solutions of interest were peak outboard neutron wall load $N_w^{\text{peak}} \geq 1.5 \text{ MW/m}^2$, $\beta_N^{\text{tot}} \leq 0.025$, and peak divertor heat flux $q_{\text{div}}^{\text{peak}} \leq 10 \text{ MW/m}^2$.

TABLE II. Parameters for a Preliminary Reference FNSF and Variations about this Point.

	REF	Lower B_T^{coil}	Lower B_T^{coil}	Lower n/n_{Gr}	Higher β_N	Q_{engr}
I_p , 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
$P_{\text{H/CD}}$, MW	150	113	196	114	120	97
f_{BS}	0.50	0.57	0.46	0.50	0.51	0.62
$q_{\text{div}}^{\text{peak}}$, 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} ($\eta_{\text{th}}=0.4$)	0.7	0.86	0.7	0.85	0.7	1.0

The first scans were done at aspect ratios of 3.0 and 4.0 for comparison. It was found that the $A = 4.0$ could access lower major radius at lower fusion power, lower P_{SOL}/R and peak divertor heat flux, lower plasma current, and lower energy confinement multiplier. The lower plasma current is desirable for weakening the effects of disruption and on the absolute MA of 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, and is not acceptable. Figure 6 shows the plasma current, energy confinement, toroidal fields and divertor power. The aspect ratio of 4.0 was chosen for the FNSF scans. The aspect ratios of the most recent 11 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 passed through an engineering module, and ultimately filtered by constraints that isolate points with the desired parameters. Shown in Table II are a reference point and a number of variants 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, $B_T^{\text{coil}} = 12.3 \text{ T}$. 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/m², respectively. An increase in β_N to 3.2 recovers the major radius, lowers the CD power, and increases the H_{98} slightly. If we do not allow the β_N to increase then the major radius increases to 5.0 m, the CD power increases, since the plasma current increases, and the peak heat flux ends up above 10 MW/m². 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/m². 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 $Q_{\text{engr}} = 1$ ($P_{\text{elec, gross}}/P_{\text{recirc}}$), the β_N rises to 2.9, the major radius is still 4.5 m, the plasma current drops raising q_{95} , and the CD power drops from 150 to 97 MW.

The systems analysis will continue under different primary assumptions of 1) magnet type, 2) blanket concept (composition) and neutronic build, 3) power balance and efficiencies, and 3) 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 be generated from thermal power.

IX. CONCLUSIONS

The proposed 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 fission. 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 penetration. 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 to move to 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

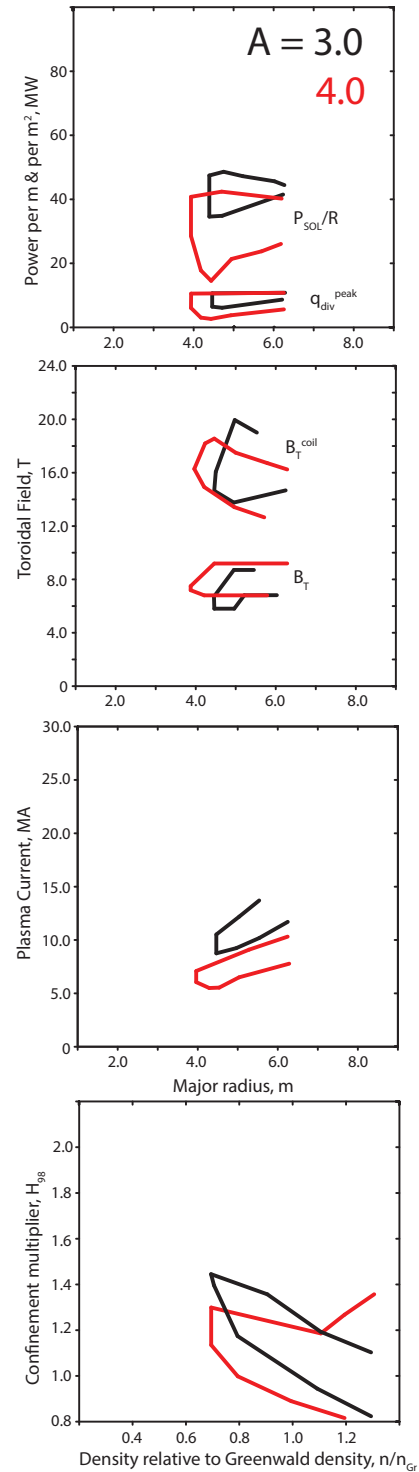


Fig. 6. Systems scans for aspect ratios 3.0 and 4.0, indicating that 4.0 is preferable for lower plasma current, lower energy confinement requirement, lower divertor heating, and lower radius albeit at higher B_T^{coil} .

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).

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REFERENCES

1. C. E. KESSEL, et al, "Fusion Nuclear Science Pathways Assessment", Princeton Plasma Physics Laboratory report, PPPL-4735, 2012; http://bp.pppl.gov/pub_report//2012/PPPL-4736-abs.html
2. S. ZINKLE, et al, FESAC Report, "Materials Science and Technology Research Now and in the ITER Era", <http://science.energy.gov/~media/fes/pdf/workshop-reports/20120309/FESAC-Materials-Science-final-report.pdf>
3. M. KOEPKE, et al, FESAC Report, "Strategic Planning: Priorities Assessment and Budget Scenarios", 2014; <http://science.energy.gov/~media/fes/fesac/pdf/2014/October/FESAC Strategic Panel Report Final Draft Oct 10 2014.pdf>
4. http://en.wikipedia.org/wiki/Shippingport_Atomic_Power_Station
5. http://en.wikipedia.org/wiki/List_of_nuclear_reactors#United_States_of_America
6. ITER Physics Basis, *Nucl. Fusion*, **39**, (1999); K. Ikeda, Progress in ITER Physics Basis, *Nucl. Fusion*, **47**, (2007).
7. G. FEDERICI, et al, *Fusion Eng. Des.*, **89**, 882, (2014).
8. K. KIM, et al, "A Preliminary Conceptual Design Study for Korean Fusion DEMO Reactor Magnet", 25th Symp. On Fusion Engineering, San Francisco, June 10-14, 2013.
9. Y. T. SONG, et al, "Concept Design of CFETR Tokamak Machine", 25th Symp. On Fusion Engineering, San Francisco, June 10-14, 2013.
10. B. N. SORBOM, et al, "ARC: A compact high field fusion nuclear science facility and demonstration power plant with demountable magnets", submitted to *Fusion Eng. Des.*; <http://arxiv.org/pdf/1409.3540v1.pdf>
11. V. S. CHAN, et al, *Nucl. Fus.*, **51**, 083019, (2011).
12. G. S. WAS, *J. Nucl. Mater.*, **367-370**, 11, (2007).
13. E. E. BLOOM, et al, *J. Nucl. Mater.*, **367-370**, 1, (2007).
14. A. LI PUMA, et al, *Fusion Eng. Des.*, **81**, 469, (2006).
15. S. HERMSMEYER, et al, *Fusion Eng. Des.*, **75-79**, 779, (2005).
16. M. S. TILLACK, et al, *Fusion Eng. Des.*, **91**, 52, (2015).
17. M. WENDEL, et al, "Fusion Material Irradiation Test Facility (FMITS) at SNS", IAEA Fusion Energy Conference 2012; http://www-naweb.iaea.org/napc/physics/FEC/FEC2012/papers/6_FTPP127.pdf
18. E. SURREY, et al, *Fusion Eng. Des.*, **89**, 2108, (2014).
19. R. HEIDINGER, et al, *Fusion Eng. Des.*, **88**, 631, (2013).
20. S. ZINKLE AND A. MOESLANG, *Fusion Eng. Des.*, **88**, 472, (2013).
21. P. W. HUMRICKHOUSE, et al, "Tritium research needs in support of long pulse burning plasmas: new initiatives", White Paper Submission, FESAC Strategic Planning Sub-Panel, July 2014; <https://www.burningplasma.org/resources/ref/fspp/wHITEPAPERS/Tritium2.pdf>
22. S. SMOLENTSEV, et al, this TOFE2014 conference.
23. C. E. KESSEL AND F. M. POLI, *Fusion Sci. Tech.*, **67**, 220, (2015).
24. J. ONGENA, et al, *Phys. Plasmas*, **8**, 2188, (2001).
25. M. A. MAHDAVI, et al, *Nucl. Fusion*, **42**, 52, (2002).
26. G. MANK, et al, *Phys. Rev. Lett.*, **85**, 2312, (2000).
27. C. E. KESSEL, et al, *Fusion Eng. Des.*, **80**, 63, (2006).
28. W. FUNDAMENSKI, et al, *Nucl. Fusion*, **45**, 950, (2005).
29. M. S. TILLACK, et al, *Fusion Eng. Des.*, **86**, 71, (2011).
30. T. D. ROGNLIEN AND M. E. RENSINK, *Fusion Sci. Tech.*, **67**, 125, (2015).
31. B. COVELE, et al, *Nucl. Fusion*, **54**, 072006, (2014).
32. D. D. RYUTOV, *Phys Plasmas*, **14**, 064502, (2007).
33. P. Thomas, "Heating and Current Drive Systems, Their Impact on Scenario/Economics (Lessons Learned from ITER)", 2nd IAEA DEMO Workshop, Vienna, Austria, Dec. 2013; <http://www-naweb.iaea.org/napc/physics/meetings/TM45256/talks/Thomas.pdf>
34. D. STORK, "Technical Challenges on the Path to DEMO", MFE Roadmapping in the ITER Era, Sept. 2011; <http://advprojects.pppl.gov/ROADMAPPING/presentations.asp>.
35. S. IDE, et al, *Nucl. Fusion*, **45**, S48, (2004).
36. N. OYAMA, et al, *Nucl. Fusion*, **49**, 104007, (2009).
37. N. ASAKURA, et al, *Nucl. Fusion*, **49**, 115010, (2009).
38. M. MURAKAMI, et al, *Phys. Plasmas*, **13**, 056106, (2006).

39. D. N. HILL, et al, *Nucl. Fusion*, **53**, 104001, (2013).
 40. C. T. HOLCOMB, et al, *Nucl. Fusion*, **54**, 093009, (2014).

41. J. E. MENARD, et al, *Nucl. Fus.*, **51**, 103014, (2011).

TABLE I. FNSF Program Table for a Moderate Mission Scope.

	He/H	DD	DT	DT	DT	DT	DT
Phase	1	2	3	4	5	6	7 (additional)
Phase time, yr	1.5	2-3	3	5	5	7	7
$N_W^{OB,peak}$ MW/m ²			1.5*	1.5*	1.5*	1.5*	1.5*
Plasma on-time/yr, % (days)	10-25	10-50	10-15 (37-55)	25 (91)	35 (128)	35 (128)	35 (128)
Plasma duty cycle			0.33	0.67	0.91	0.95	0.95
Total maintenance time**, days			762-600	1140	1120	1610	1610
End of phase peak fluence, MW-yr/m ² (dpa)			0.45-0.68 (4.5-6.8)	1.88 (18.8)	2.63 (26.3)	3.68 (36.8)	3.68 (36.8)
Cumulative peak damage (dpa)			4.5-6.8	23.3-25.6	49.6-51.9	86.4-88.7	123.2-125.5
DCLL blanket	RAFM T _{LiPb} = 400C	RAFM T _{LiPb} = 400C	RAFM T _{LiPb} = 400C	RAFM-ODS T _{LiPb} = 500C	RAFM-ODS(NS) T _{LiPb} = 550C	RAFM-ODS(NS) T _{LiPb} = 600C	RAFM-ODS(NS) T _{LiPb} = 600C
Blanket TBM			RAFM-ODS T _{LiPb} = 500C	RAFM-ODS(NS) T _{LiPb} = 550C	RAFM-ODS(NS) T _{LiPb} = 600C	RAFM-ODS(NS) T _{LiPb} = 650C	

*higher β_N operation can allow higher neutron wall load

**1 year has been allocated at the end of each phase in addition to the specified maintenance time

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