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Abstract

The Fusion Nuclear Science Facility is examined here as part of a two step program from ITER to commercial power plants. This first step is considered mandatory to establish the materials and component database in the real fusion in-service environment before proceeding to larger electricity producing facilities. The FNSF can be shown to make tremendous advances beyond ITER, toward a power plant, particularly in plasma duration and fusion nuclear environment. A moderate FNSF is studied in detail, which does not generate net electricity, but does reach the power plant blanket operating temperatures. The full poloidal DCLL blanket is chosen, with alternates being the HCLL and HCCB/PB. Several power plant relevant choices are made in order to follow the philosophy of targeted technologies. Any fusion core component must be qualified by fusion relevant neutron testing and highly integrated non-nuclear testing before it can be installed on the FNSF in order to avoid the high probability of constant failures in a plasma-vacuum system. A range of missions for the FNSF, or any fusion nuclear facility on the path toward fusion power plants, are established and characterized by several metrics. A conservative physics strategy is pursued to accommodate the transition to ultra-long plasma pulses, and parameters are chosen to represent the power plant regime to the extent possible. An operating space is identified, and from this one point is chosen for further detailed analysis, with $R = 4.8$ m, $a = 1.2$ m, $I_P = 7.9$ MA, $B_T = 7.5$ T, $\beta_N < 2.7$, $n/n_{Gr} = 0.9$, $f_{BS} = 0.52$, $q_{95} = 6.0$, $H_{98} \sim 1.0$, and $Q = 4.0$. The operating space is shown to be robust to parameter variations. A program is established for the FNSF to show how the missions for the facility are met, with a He/H, a DD and 5 DT phases. The facility requires ~ 25 years to complete its DT operation, including 7.8 years of neutron production, and the remaining spent on inspections and maintenance. The DD phase is critical to establish the ultra-long plasma pulse lengths. The blanket testing strategy is examined, and shows that many sectors have penetrations for H/CD, diagnostics, or TBMs. The hot cell is a critical facility element in order for the FNSF to perform its function of developing the in-service material and component database. The pre-FNSF R&D is laid out in terms of priority topics, with the FNSF phases driving the time-lines for R&D completion. A series of detailed technical assessments of the FNSF operating point are reported in this issue, showing the credibility of such a step, and more detailed emphasis on R&D items to pursue. These include nuclear analysis, thermo-mechsnics and thermal-hydraulics, liquid metal thermohydraulics, transient thermo-mechanics, tritium analysis, maintenance assessment, magnet specification and analysis, materials assessments, core and SOL/divertor plasma examinations.

I. Introduction

For fusion research to take the step beyond ITER it will have to embrace the fusion nuclear science along with fusion plasma science. The hardware that surrounds and supports the plasma will become part of the challenge for research and development since fusion power plants will rely on these structures to recover the power emitted, breed the tritium fuel, provide neutron and gamma shielding, and provide the magnetic fields and the vacuum environment the plasma requires. The Fusion Nuclear Science Facility (FNSF) is a fusion nuclear device that is considered as the first step in a two-step pathway from ITER to commercial power plants in the U.S [1]. The project reported here is exploring this facility to better understand its characteristics and how it moves the demonstration of sustained fusion energy production toward our present vision of power plants.

In order to address this facility several technical strategies and choices had to be established, including the need for a fusion break-in step, the importance of power plant relevance, the practicality of a single primary blanket approach, the need for a fusion core component qualifications, the need for a plasma strategy, and a series of technical decisions that stem from these. A set of missions that must be accomplished to reach an electricity producing power plant are described, and several metrics are proposed for measuring their progress. A program is postulated for the FNSF to expose the steps required to advance these missions, and force the consideration of allocating time to plasma operations, inspections, and maintenance. Although these steps are dominated by the neutron fluence they reach, and blanket operating temperatures, they can also include other incremental technical steps. The blanket testing part of the program is developed by considering plasma support systems (e.g. heating and current drive, fueling, diagnostics), inspection needs, maintenance, and the hot cell. Similar testing would be performed for the divertor, and possibly the other special plasma facing components (e.g. RF launchers and diagnostics).

Systems analysis is used to identify a conventional aspect ratio operating point and its surrounding operating space, with focus on the plasma and engineering constraints, and the need for robustness to account for the considerable uncertainty in reaching the desired parameters. The operating point (its geometry) is used in detailed analysis of the plasma core, scrape-off layer and divertor, nuclear analysis, steady and transient thermo-mechanics, thermal hydraulics, liquid metal MHD breeder analysis, magnets, maintenance, radio-frequency structures and apparatus, tritium behavior and inventory, and materials considerations. These calculations are being used to establish the credibility of such a facility at its smaller size, identify the benefits/penalties of specific technical decisions, uncover vulnerabilities and approaches to provide margin, and help in establishing targeted R&D for the FNSF. The accompanying papers in this issue provide the detailed assessments [2-13], and will only be summarized here.

This paper is organized as follows. In Section II the background for the FNSF is outlined by briefly describing the present fusion landscape, and how the FNSF appears in and impacts the fusion development pathway. Its importance is motivated by the need for a fusion nuclear step that provides an actual fusion environment for the first time, and technical strategies are described. Section III describes the facility mission, and metrics for measuring progress. Section IV describes the physics assumptions and supporting experimental observations. Section V describes the systems analysis and results in deriving the operating space for the device. A program is described in Section VI. Summaries of the detailed technical analysis are given in Section VII, and pre-FNSF R&D is described in Section VIII. A summary and conclusions are presented in Section IX.

II. Background

The FNSF is examined as part of the development path toward commercial fusion energy-based electricity production in the U.S. The FNSF can take on many possible missions, and this is demonstrated by several different forms previously reported [14-17] ranging from a volumetric fusion neutron source to an electricity producing pilot plant. The present study is focused on an FNSF that will contribute to the development path in a definable way, as opposed to the focus on nuclear effects studies or plasma configurations that dominate earlier studies. The landscape in which fusion energy research finds itself now has evolved over the last 40 years, and plays an important role in what is conceivable as a development path. Early roadmaps [19] (1976) for the U.S. fusion program often identified multiple engineering steps before a commercial fusion power plant. These included TFTR (which was built and operated), an engineering research facility or engineering test reactor, a prototype experimental power reactor or ignition test reactor, an experimental power reactor, and finally a demonstration reactor. The list also includes several plasma physics facilities. By the mid 1980's [20] this view had changed significantly, with discussion of a burning plasma facility, international cooperation on an engineering test reactor (referred to as ETR or ITER), and several plasma physics experiments and non-confinement facility engineering test stands (including a materials test facility). International collaboration took a much stronger position at this point due to significant budget reductions in the 1980's. Finally in the mid 1990's [21] a restructuring of the U.S. Fusion Energy Sciences Program took place, moving the emphasis of the program to advancing the plasma science, fusion science and fusion technology, with ITER as the only new fusion facility, directly associated with fusion energy, on the landscape. A much richer description of the history of the U.S. fusion program, and many program studies produced, can be accessed on the FIRE website [22]. In the U.S., and globally, the appetite for several fusion engineering facilities to advance toward a power plant has diminished and there exists now increased pressure to advance any fusion nuclear facilities in as few steps as possible. By 2010 and later, with the international commitments to the ITER project and construction in place, several countries turned to examining what might follow, or proceed in staggered-parallel, with ITER to move toward fusion energy-based electricity production [23-26]. As part of this, this project is targeting a better understanding of 1) what such a next step facility must accomplish, 2) how is this accomplishment measured, 3) how does the facility accomplish its mission, 4) what is the pre-requisite R&D for this facility, and 5) how does this facility fit into a pathway to commercial fusion power.

The development pathway for fusion energy in the U.S. is assumed to have two facilities before commercial fusion power plants are pursued [1], to strike a balance of minimizing technical risk and advancing in as few steps as possible. These are the fusion nuclear science facility (FNSF) and a demonstration power plant (DEMO). The demonstration power plant is intended to demonstrate routine electricity production and plant operations, most likely at reduced availability and somewhat less competitive economics compared to a power plant. At the end of the demonstration power plant facility operation there should be no technical gaps to a commercial power plant, that is, no additional R&D is required, only technical scaling of DEMO systems to power plant parameters remains. This also applies to the economic assessment, such that projections to profitability can be made. The design of a DEMO facility would require projecting all aspects of the facility to a commercial power plant in order to develop a convincing argument that this was achieved. Although the DEMO must end its operation with no technical gaps to power plants, it does not preclude the need for some development in its early phases.

The FNSF on the other hand represents a critical and necessary break-in for fusion, where the fusion nuclear regime is experienced at a significant level and duration for the first time. It is considered here to be the facility that would follow some level of ITER DT controlled burning plasma demonstrations, but precede the DEMO facility. It provides the fully integrated fusion nuclear environment in combination with fully integrated fusion core components (blanket, divertor, heating/current drive, and diagnostics), near-core components (vacuum vessel, cryostat, TF coils, PF/CS coils, maintenance and inspection equipment, hot cell, feed pipes and transmission lines), and finally ex-core components (tritium extraction, heat exchanger, fluid cleanup, plasma heating/current drive sources, vacuum pumps, etc.). This facility also provides the ultra-long plasma pulse operation required to move toward power plants, in conjunction with the plasma facing material/components, at performance levels that provide the strong neutron and plasma loading environment. These aspects of this device require remote-handling and a long-term relevant maintenance strategy untypical of present tokamaks and ITER facilities. Fig. 1 is an illustration of this basic incremental philosophy, in terms of large fusion confinement facilities, ITER, FNSF, DEMO and commercial power plants.

In Figure 1 appears a short list of descriptive metrics to quantify the jumps taken from facility to facility, most notable is the large increase in nuclear damage from ITER to the FNSF, some 10-20x, with an ultimate increase of 30-50x for a DEMO and commercial power plant. The plasma pulse length would be increased by > 400x, along with a tremendous reduction in the dwell time between plasma pulses, and a large increase in the plasma on-time in a calendar year. The overall availability of the plasma must be advanced by ~ 3000 times compared to present tokamak operations in the US, or ~ 20x compared to ITER. This has significant implications for all systems that support and interface the plasma, and such targets cannot be achieved without significant improvements in these system's reliability, and maintenance procedures and equipment. The FNSF will make the first inroads to breeding the tritium it requires to fuel its fusion reactions self-sufficiently. The materials used in an FNSF must move beyond conventional structural and functional materials commonly used in present fusion facilities, and used in ITER, by developing and applying fusion irradiation resistant and low activation (waste) materials. In addition, the environment anticipated in the fusion core of a DEMO and commercial power plant are the same targets for the FNSF, to firmly establish the basis for the larger electricity producing next steps.

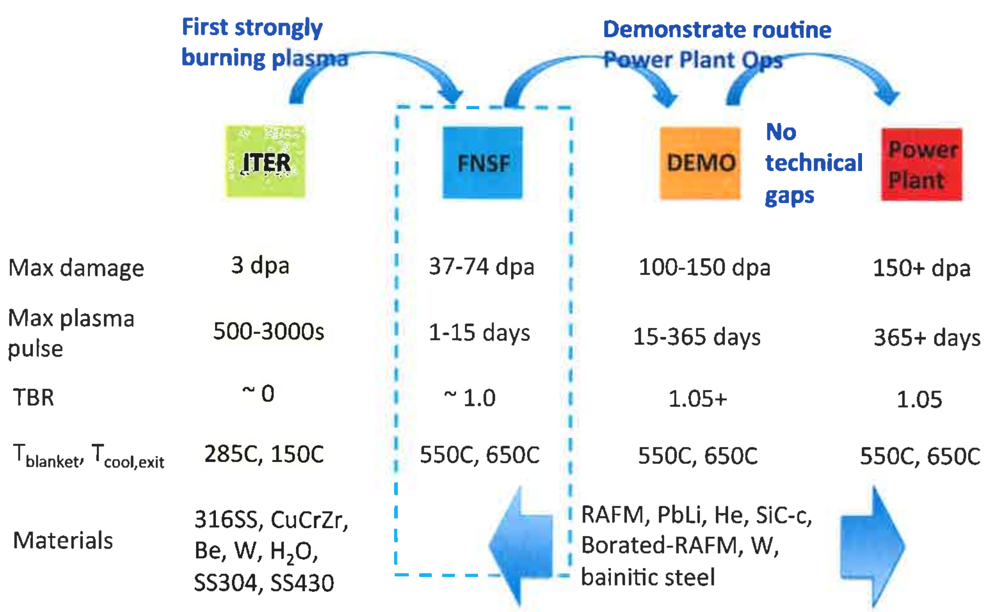


Figure 1. The large confinement facility pathway to commercial fusion power illustrated along with several metrics describing how each facility incrementally approaches power plant parameters, as well as the significant distinction between ITER, with a strong plasma mission, and the FNSF which pushes strongly into the fusion nuclear regime.

Focusing on the two facilities between ITER and a commercial power plant, the pathway includes a pre-FNSF R&D program to establish the scientific basis for the many systems that come together in the facility, the FNSF facility itself as a research tool through its program, parallel-FNSF R&D occurring simultaneously with the FNSF program (e.g. radiation resistant materials), a pre-DEMO R&D program aimed at the science and technologies required for DEMO (not established in the FNSF), and the DEMO facility itself via its program. The connection between the FNSF and DEMO can be viewed as in Fig. 2 and Fig. 3, where the progress made in the FNSF toward a commercial power plant defines what research is left to do in the early phase of the DEMO, if any. This study of the FNSF will focus in detail on the moderate FNSF, and compare minimal, moderate and maximal configurations only at the systems level.

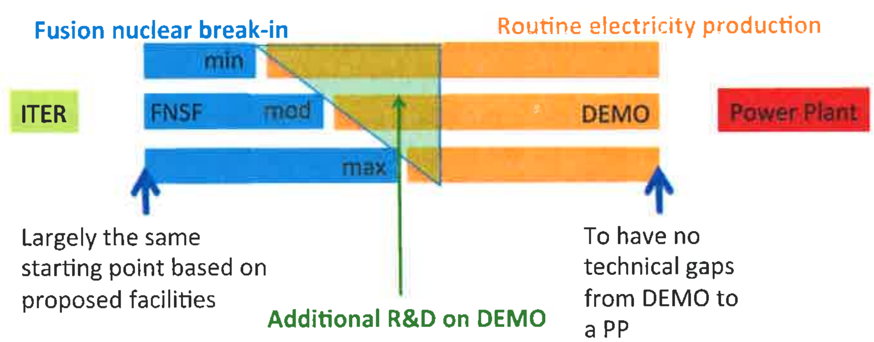
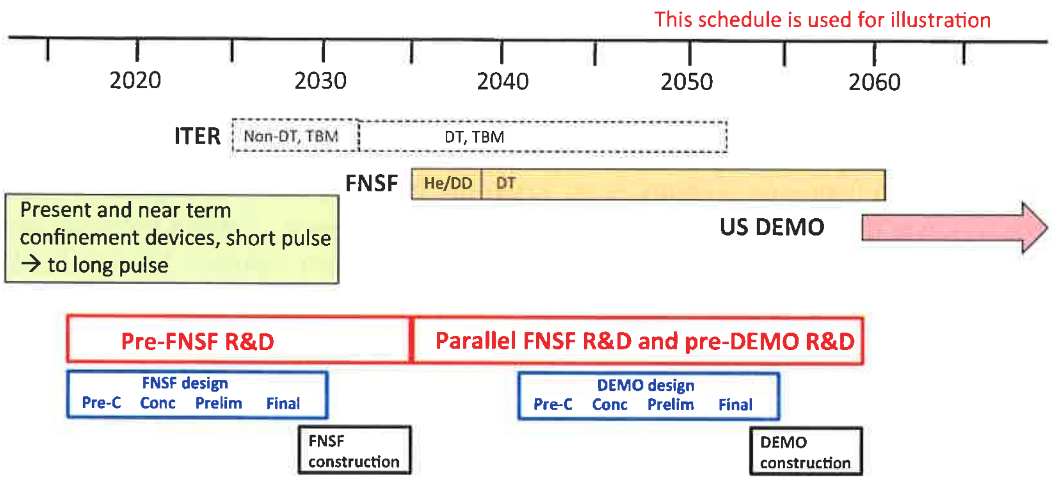


Figure 2. Illustration of the U.S. major confinement facilities in the fusion development path to commercial power production, noting the impact of the various possible FNSF mission scopes on the required R&D in DEMO. Here min, mod and max refer to minimal, moderate and maximal FNSF scopes.



* US does not presently have a commitment to design and construct the FNSF or DEMO

Figure 3. An illustrative time-frame showing ITER first plasma and DT operations, with the FNSF following ITER plasma burn demonstrations, the pre-FNSF R&D and FNSF design activities (pre-conceptual, conceptual, preliminary, and final) and construction. This is similarly shown for the DEMO. During the FNSF, R&D continues in several areas to support the FNSF and prepare for the electricity producing DEMO.

Table 1. Some parameters to distinguish a minimal, moderate and maximal FNSF, identifying large, some, and small departures from power plant characteristics.

Large departure from PP

Some departure

Small departure

	minimal	**moderate**	maximal	Power plant
Plant DT operations	~ 15 yr	~ 25 yr	~ 35 yr	47 yr (40 FPY)
Peak neutron wall load, MW/m^2	1.0	1.5	2.25	2.25
Plasma on-time per year	10-35%	10-35%	10-45%	85%
Max dpa on first wall (or max dpa to replace)	5 -18,36	7 - 37,74	10 - 70,140	150-200
Q_{engr}	<< 1	< 1	> 1	4
Tritium breeding ratio	< 1	~ 1	> 1	1.05
Plant life, peak dpa	50	126	274	765
TF/PF magnet	Cu	LTSC or HTSC	LTSC or HTSC	LTSC or HTSC
Vacuum vessel material	SS	Baintic steel	Bainitic steel	Bainitic steel
Divertor	W/CuCrZr/H_2O	W/W/He	W/W/He	W/W/He

The FNSF can take on multiple forms, characterized by how far it pushes toward a power plant. Three rough characterizations have been used to examine this, minimal, moderate, and maximal. Systems analysis solutions for these will be discussed in more detail in Section V. The variations among these possible FNSF's can be generally described at least roughly by 1) years of operation, 2) plant lifetime neutron fluence and maximum fluence seen by a fusion core component, 3) tritium breeding ratio, 4) engineering gain or electrical power production, and 5) materials and coolants used. Table I provides some estimates for this, highlighting the departure from power plant values. The minimal FNSF would not advance these attributes toward a power as much as the moderate or maximal FNSF's, and the maximal FNSF would attempt to advance most technical aspects toward the power plant. In general, these devices range from smaller to larger as they go from minimal to maximal. However, the minimal FNSF would leave a significant undeveloped scope that must be developed on the DEMO in its early phases, while the maximal FNSF would leave little scope unaddressed. The connection between the FNSF and DEMO is most clearly represented by the fact that the possible FNSF's largely begin with a similar technical basis, and the DEMO must complete its operation delivering the same technical basis to power plants, and so the scope of technical advancement in the FNSF and DEMO are intimately connected in between.

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

II.1. Motivation for a careful break-in step

The significant cost and complexity of a nuclear fusion facility, the time-scales required for the development of various technologies, and the need to provide the many subsystems that support a fusion confinement facility (from plasma to remote maintenance) generally precludes the approach of having several fusion nuclear devices along the development pathway. On the other hand, assuming only one facility is required is similarly difficult, primarily because the complex environment seen by the components in the fusion core (e.g. blanket, divertor, launchers) and near-core (e.g. structure/shields, vacuum vessel, magnets) cannot be re-created in offline testing facilities before the FNSF. The environment experienced by these components prior to the FNSF includes both nuclear and non-nuclear features, however these can only be created separately. The nuclear environment would be sampled with a fusion relevant neutron source (typically an accelerator with a target that provides neutrons in the energy range of fusion). The test volumes available in these facilities are highly limited, only providing for small material coupons at controlled temperatures. In addition, these facilities only provide an approximation to the actual fusion neutron energy spectrum. Facilities like the International Fusion Materials Irradiation Facility (IFMIF) [27] do have low neutron flux zones with larger volumes. It is possible to take advantage of the larger test volumes in fission reactors for testing small assemblies, perhaps a RAFM shell with LiPb flowing through it, but even these volumes are limited, and the neutron spectrum is significantly different from fusion. The non-nuclear environment can be sampled with highly integrated facility(s) testing complete blanket components, for example. This facility would be the culmination of a number of smaller testing facilities that addressed more specific technical issues separately (e.g. liquid metal flow, tritium permeation, high heat flux). In such an integrated facility, heating, fluid flow, hydrogen permeation, operating temperatures and pressures, magnetic fields, can all be created and tested simultaneously, but without neutrons. Only the FNSF can provide the actual combination of fusion nuclear and non-nuclear aspects, however this is not simply a sum of non-interacting pieces. It is well known that stress and temperature play a tremendous role in the behavior of irradiation damage and transmutation gas evolution in irradiated materials, for example. It is also prudent to realize that gradients in irradiation damage, stress, temperature, and hydrogen concentration will undoubtedly produce new behaviors we have not seen in fusion relevant neutron source testing prior to the FNSF. In a broader sense, the combination of the separate nuclear materials testing and non-nuclear integrated component testing is sufficient to design and operate a FNSF, however in order to design and operate an electricity producing DEMO and power plant, an entirely new in-service fusion nuclear database is required to describe and project material and component behavior, because we expect it will be sufficiently different. The FNSF facility is a critical bridge in this development path because of its unique role in establishing this in-service database.

It is worth examining the development of materials for the fission fast breeder program [28,29] in the U.S. as an example of the complications that arise from in-service conditions. A quote from ref [28] indicates the potential impact of materials behavior in fusion, "*Indeed, unexpected material behavior can cause major disruptions to a development program. For example, the first open literature report of void formation during neutron irradiation raised concerns about swelling which dramatically slowed the development of the liquid metal fast breeder reactor program (LMFBR). It required nearly a decade to fully understand the phenomena of swelling and another decade to develop materials with satisfactory performance.*" This reference cites a number of material phenomena relevant to fusion's irradiation at high temperature regime, including radiation-induced segregation (RIS, redistribution of elements in material under irradiation), radiation-induced precipitation, radiation-modified/enhanced/retarded thermal precipitation, helium embrittlement and coupled helium-RIS effects. An unusual result is presented where tensile tests conducted on un-irradiated, post-irradiated, and in-reactor samples showed that the post-irradiated sample had significantly reduced creep rupture strength relative to the un-irradiated sample, while the in-reactor sample showed better creep rupture strength. This unexpected improvement was attributed to short-lived defects generated during irradiation that ultimately annealed out during elevated temperature post-irradiation testing. This points to the importance of complex interactions experienced by materials between irradiation, loading, and the environment. Reference [28] presents a number of examples in fission materials in service that showed unexpected and significant behaviors based on temperature, dose rate, composition/microstructure, coupling among features, welds, surface conditions, gradients, incubation (delays in the emergence of a phenomena), metallurgical variability, and the prototypical environment actually experienced. It is also pointed out that for fusion power plants to be successful the development of failure prediction models and materials management is needed, and these require the true fusion in-service environment to be characterized, which the FNSF can perform.

There are examples from the FNSF study where we can see the variations in service parameters that may lead to important effects under irradiation. Shown in Fig. 4 are the variations in the displacements per atom (dpa) and atomic parts per million (appm) of He resulting from the fusion neutron irradiation near the first wall, for the Dual Coolant Lead Lithium (DCLL) blanket design. The dpa varies from 13 to 7 dpa in 10 cm and the He varies from 120 to 30 appm over the same distance. Also shown are thermo-mechanics results for a slice of the inboard blanket of the FNSF, where the temperatures show a peak value at the first wall facing the plasma, with various transitions to lower temperatures as one moves through to the breeding blanket, with variations of up to ~ 130 °C. The von Mises stress in the breeder flow channel walls inside the breeding blanket show variations of about 30-60 MPa over and over as a grid plate is traversed, due to high pressure helium coolant channels. The combination of the hydrogen produced from transmutations and the tritium bred in the blanket will introduce varying levels of hydrogen into the solid structure matrices. The anticipated trapping of some of this hydrogen can contribute to other irradiation materials behavior. These can be contrasted with a material test coupon used in a fusion relevant neutron source at a specific temperature that we are using to develop the basic information we require to pursue a FNSF.

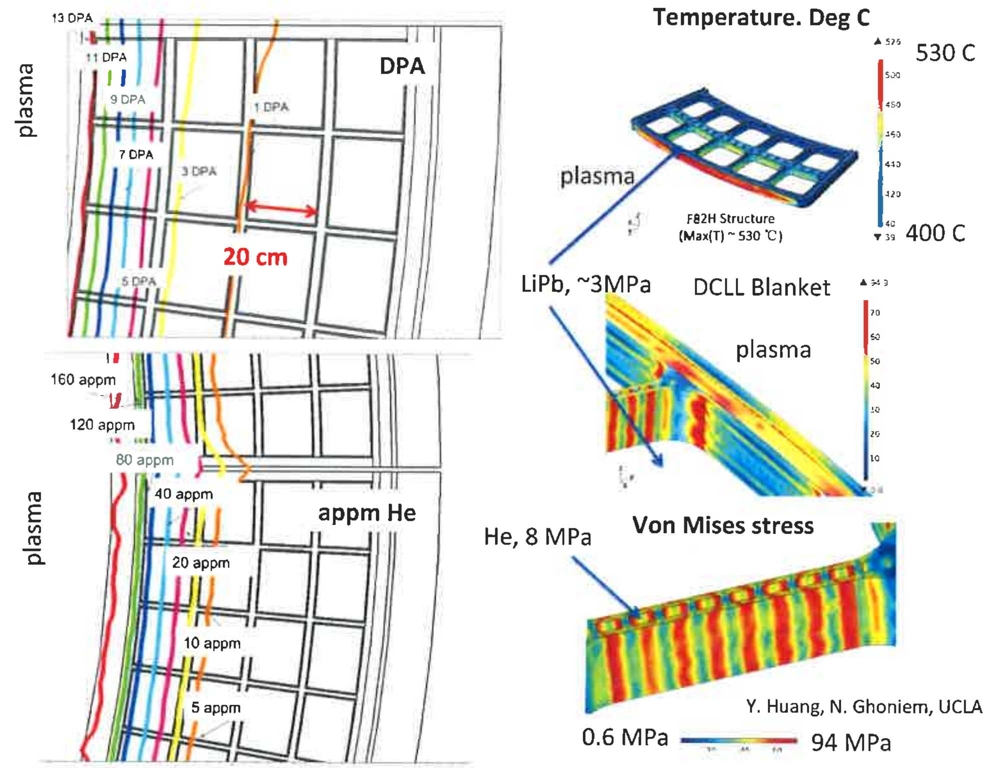


Figure 4. The variation in the displacement per atom (dpa) and atomic parts per million (appm) He from fusion neutron irradiation in the vicinity of the first wall, for the FNSF outboard blanket, left. Thermo-mechanics analysis of a slice through the inboard blanket, shows the temperature variations between the first wall, inside the large LiPb conduits and the back of the blanket. In addition, zooming into the side wall of the LiPb conduit the Von Mises stress can be seen to vary back and forth between approximately 30 and 60 MPa.

II.2. Power Plant Relevance

In view of the limited number of fusion confinement facilities in the pathway from the ITER-era to power plants, the need for consistent advancing of all the required science and technologies can be understood. The FNSF and DEMO become the platforms for significant preceding R&D, focused on the specific facility's needs. The development of advanced superconducting magnets, high heat flux components, materials science for plasma and nuclear loading, breeding blanket materials/design integration, tritium extraction, and plasma heating, fueling and exhaust are (only a few) examples of systems that must be advanced to prepare for and meet the conditions on a new next step fusion facility. The few steps to reach power plants indicate that there is little to no room for non-power-plant relevant development. Taking long time scales (1-2 decades) to develop a system that will never be relevant to a power plant is undesirable, even with the uncertainty in projecting precisely what a power plant will be. Although the discussion here focuses on many hardware elements, major focus on power plant relevant plasma physics is also required by focusing on the long term plasma facing materials, and core and divertor operating regimes. Based on this reality for fusion, it is chosen to target only power plant relevant subsystems, at least based on our power plant studies to date. This is balanced against the FNSF being a first of a kind, and one of a kind, facility in the fusion nuclear regime. An example of this decision process considering long term relevance and near term risk aversion would be the choice to use copper TF and PF coils, when it is understood that superconducting coils are the only credible solution for a power plant. Copper coils are anticipated to cost less than superconducting coils and provide some ability to reduce the device size due to higher tolerance for irradiation and/or nuclear heating. It is clear that they would suffer from high power consumption during the long pulse plasma operations in the FNSF. Then on the other hand, Nb_3Sn low temperature superconductors (LTSC) have an established database extending from the Large Coil Test program [30] in the 1980's and the ITER coil design [31] up to present day, while high temperature superconductor (HTSC) have made important progress, but no fusion scale HTSC TF magnet has ever been built or tested. The FNSF technical decision was to pursue an advanced low temperature superconducting design based on higher performing Nb_3Sn superconductor [32] and enhanced design to reach 16 T at the coil and ~15 MA/m^2 overall coil current density, based on the intentions of the Korean fusion program [33] and the High Energy Physics program [34.35] to pursue these parameters. Along with this direction, a "watch" is applied to the HTSC developments, anticipating progress while understanding that development can take several years, especially to yield magnets that produce the suggested peak fields of ~ 23 T and high current density simultaneously [36].

Another example of this philosophy was the FNSF choice to eliminate water from the fusion core, more specifically, from inside the vacuum vessel (VV) or within the VV itself. The VV is the primary radionuclide and pressure barrier, and represents also the boundary isolating the fusion core components from other ancillary equipment. This choice was reported in ref [37], and justified based on water interactions with LiPb, higher operating pressures (16 MPa) to avoid boiling, low maximum water temperatures (300-350 ^{o}C) being inconsistent with RAFM structural materials and leading to low thermal conversion efficiencies, potential for hydrogen explosions in high temperature accident scenarios, and enhancing the tritiated water load for processing. The primary penalty for this choice was thicker radial builds to accomplish the required shielding for the TF magnet. Concomitant with this choice is the choice to use helium as the primary coolant, which stems from the numerous power plant studies [38-41] that show that it is power plant relevant due to its ability to achieve high temperatures for high thermal conversion efficiencies, chemical inertness, retaining of tritium in gaseous form, and low neutron cross-section. In addition, there is an extensive helium coolant database, both experimental and computational, on the viability of He cooling strategies that address its shortcomings relative to more dense fluids like water. These have addressed heat transfer enhancements from roughening, impingement, and turbulence inducers, as well as pumping power requirements [42-45]. The gas cooled fission experience [46,47], both older and more recent, provides a practical basis for projecting He cooling to fusion. In addition, the highly successful CO_2 gas cooled fission experience, although not identical, provides further basis for the credibility of gas cooling. Fusion power plant studies have pursued both analysis and experiments, and a number of DOE sponsored SBIR and

CRADA studies [42,48-52] have established a sound basis for He cooling technology. The choice to pursue He cooling for the FNSF was to initiate and advance the technology that was clearly superior in the power plant regime, rather than spend many years developing workarounds for the many disadvantages of water cooling.

Several other technical decisions for the FNSF were developed with similar power plant relevance in mind. These are listed in Appendix A, along with a short description of justifications, benefits, and/or penalties, with topics listed below.

Dual Coolant Lead Lithium blanket concept
Helium Cooled Lead Lithium and Helium Cooled Ceramic Breeder as alternatives
Tungsten Carbide as shield filler on the inboard
Irradiation limits on the TF coil
Horizontal maintenance
A thin tungsten coating on the FW RAFM steel
Concentric hot/cold leg coolant and breeder piping
No electricity production requirement for the moderate FNSF
Tungsten structure/armor divertor

II.3. Primary Blanket Strategy and Required Qualification for Fusion Core Components

The high fusion power and high neutron fluence goal of the FNSF necessitate an effective tritium breeding blanket in order to assure availability of fuel for operations. An extension of the power plant relevance approach was the choice of a primary breeding blanket, and this was the Dual Coolant Lead Lithium (DCLL) concept, due to its high potential for thermal performance and thermal conversion to electricity in a power plant [53]. This blanket concept uses RAFM (or higher temperature and/or radiation resistant variants) steel as structural material, He as primary coolant, and LiPb liquid metal breeder/coolant. A SiC-composite (flow channel insert) electrical and thermal insulator is required between the LiPb and the RAFM steel in the blanket to minimize MHD pressure drops for the liquid metal, and the LiPb flows at ~ 5-25 cm/s. The full poloidal extent of the channels in this blanket [54] was also pursued, based on a significant simplification of the coolant and breeder flows and piping, and reduction of structure leading to improved tritium breeding. It was recognized that accommodating multiple blanket concepts was impractical from the engineering and layout points of view, based on providing the different/independent coolant feeds to each sector, the large number of potential sector penetrations (TBMs, MTM, RF launchers, NB, and diagnostics), and the demand for only qualified blanket concepts on the FNSF (which requires an R&D program for that concept). The ability to test any blanket concept, which has been advocated previously, was rejected because of the incompatibilities with safety (e.g. water cooled concepts at high pressure), insufficient database, development and/or qualification (e.g. using a SiC-c or Vanadium structural material), and the fact that it was not considered credible that multiple primary blankets would be developed in the U.S. fusion program. Alternative blankets were identified by examining blanket concepts with similar basic features, while also considering difficulties that could arise in the coupled fusion nuclear and non-nuclear environment in the FNSF. Most non-nuclear setbacks for a blanket concept should have been identified and corrected by the pre-FNSF R&D program. The major weakness of virtually all fusion blanket concepts is the breeder and its requirements. The alternative blankets focused on this element, being the Helium Cooled Lead Lithium (HCLL) and the Helium Cooled Ceramic Breeder / Pebble Bed (HCCB/PB). The HCLL concept reduces the LiPb flow speed to mm's/s, thereby removing the need for the SiC-c flow channel insert since the low fluid speed strongly reduces the liquid metal MHD phenomena. The HCCB/PB removes the liquid metal altogether, and replaces it with a solid ceramic breeder material. The alternative concepts are closely related to the DCLL, but generally have degraded performance in a power plant, and may suffer from their own shortfalls, such as high tritium partial pressure in the HCLL since the fluid moves so slowly, and thermo-mechanics issues with solid breeders at high temperature and under irradiation. Their commonality made it conceivable to provide the required coolant feeds, although these designs were not pursued any further in this study. They are the frontrunner concepts for the EU [55,56] in modular blanket concepts.

The FNSF requires a long pulse plasma operating at high performance to provide the nuclear environment essential for testing and developing a basis for the fusion facilities that follow. The plasma-vacuum environment required for plasma operations is very demanding and is incompatible with a high frequency of fusion core component failures. It has been proposed that an FNSF can simply test fusion core components until they fail, as the program for the facility [14,16]. Based on present tokamak operation, failures of hardware in vacuum are severely compromising to plasma operations, requiring shutdown, up to atmosphere and subsequent inspection, repair and cleaning, before re-entering pump-down to high vacuum conditions. The time scales for this can be long, weeks to months depending on the severity of the failure. In some cases plasma operations have been compromised by failures that were not "visible", but were clearly present due to lower plasma performance, and subsequently identified after the run campaign. In a nuclear device like the FNSF, all inspection and maintenance processes are accomplished with remote handling equipment, either human or computer controlled. Maintenance will require considerable time to accomplish actions typically performed by humans on present tokamaks. Since a device like the FNSF can only succeed if it is running and providing the needed nuclear environment, the "cook and look" approach is not considered viable. In light of this, all fusion core components are required to have two primary qualifications before being installed in the device; fusion relevant neutron testing of individual materials to the fluence (or dpa) level reached in a given phase, and highly integrated non-nuclear component testing to the operating parameters expected in the phase (heating, temperature, flow, pressures and stresses, B-field, and hydrogen) to the extent possible.

During operation on the FNSF, in-situ measurements, regular inspections, surveillance material samples, and full sector autopsies are taken to monitor the evolution of the fusion core components during any phase in its program. At the end of a phase in the program all sectors are removed for examination, and broken down into smaller samples for more detailed post irradiation examination.

III. Mission and Metrics for Assessing Progress

The primary missions associated with the FNSF have been identified and are listed below. These encompass the main technical areas that require advancing from the ITER era status to a power plant. In the FNSF these may or may not be advanced, and each mission may be advanced to a greater or lesser degree than another. Technical areas that are advanced, and the degree to which they are advanced, comprises the mission scope for the FNSF. Also listed are some metrics under each mission that can help to characterize this advance. In general a few to several of these metrics per mission is all that is necessary to demonstrate progress, and they represent critical parameters

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

    a. Life of plant outboard peak neutron fluence, MW-yr/m^2 (or displacements per atom)
    b. Outboard peak neutron fluence reached before replacing first wall & blanket, MW-yr/m^2 (or displacements per atom)
    c. Peak outboard neutron wall load, MW/m^2

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.

    a. First wall armor
    b. First wall structure
    c. Blanket structure
    d. Breeder (coolant)
    e. Blanket coolant
    f. Structural ring/shield
    g. .....

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.

    a. FW armor temperature
    b. FW structure temperature, FW coolant I/O temperature, pressure, flow rate
    c. Blanket structure temperature
    d. Breeder temperature I/O, pressure, flow rate
    e. Blanket coolant temperature I/O, pressure, flow rate
    f. .....

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

a. Tritium breeding ratio
b. Lithium-6 enrichment required, %
c. Outboard first wall hole/loss area, %
d. Tritium lost to decay, kg/year
e. Tritium lost to the environment, g/year

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.

    a. Tritium extraction efficiency
    b. Tritium leakage through HX, cleanup apparatus, pipe runs
    c. Tritium inventory in blanket materials, coolants and breeder
    d. Tritium inventory in divertor materials, coolant
    e. Tritium fueling rate to and exhaust rate from plasma chamber
    f. Tritium burnup

6. Routinely operate plasmas for very long 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, and ultimately ~ 1 year in a power plant.

    a. Plasma on-time per year
    b. Plasma pulse duration
    c. Plasma pulse duty cycle
    d. $\beta_N H_{98} / q_{95}$
    e. Q (fusion gain, P_{fus} / P_{aux})
    f. f_{BS} (bootstrap current fraction)
    g. $P_{core,rad} / (P_{alpha} + P_{aux})$
    h. $P_{div,rad} / P_{SOL}$
    i. Disruption frequency
    j. ELM energy release and frequency

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.

    a. H/CD sources, maximum power, maximum duration, source lifetime, CD efficiency, wall plug efficiency
    b. Fueling sources, fueling rates, fueling efficiencies
    c. Pumping type, exhaust efficiency, He enrichment, regeneration rates

   d. Disruption avoidance/mitigation, number per year, unmitigated disruptions per year
   e. ELM mitigation
   f. TF coil type, plasma TF, coil TF (max), $\langle j_{TF} \rangle_{winding}$, $\langle j_{TF} \rangle_{total}$
   g. PF/CS coil type, max B, max I, $\langle j_{CS} \rangle_{winding}$, $\langle j_{CS} \rangle_{total}$, $\langle j_{PF} \rangle$
   h. Divertor type, peak heat flux, peak transient heat flux, max erosion rate, lifetime to replace

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 in the worst accident scenario, and meet or exceed all other regulatory aspects.

   a. Radioactive waste classification
   b. Plant maximum tritium leakage per year
   c. Peak decay heating, W/m^3
   d. Peak specific activity, Ci/kg
   e. Peak absorbed dose to human, to equipment, and/or dose rate

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

   a. Plasma gain, Q (P_{fus} / P_{aux})
   b. Engineering gain, Q_{engr} ($P_{elec,gross}$ / $P_{recirculate}$)
   c. Net electricity production, $P_{elec,net}$
   d. Thermal conversion efficiency
   e. Heating and current drive source to plasma wall plug efficiency
   f. Coolant/breeder pumping power efficiency (P_{pump}^{He} / P_{th}^{He})
   g. Total plant subsystems electric requirement, MW

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.

    a. Plasma on-time per year, %
    b. Plasma pulse duty cycle, %
    c. In-core inspection frequency, time to perform inspection
    d. In-core minor and major maintenance frequency, time to perform maintenance
    e. Ex-core inspection frequency, time to perform inspection
    f. Ex-core maintenance frequency, time to perform maintenance

The breadth (how many missions are being addressed) and the depth (how far toward a power plant) that the mission scope provides is a critical measure of effectiveness for any next step facility. A sampling of these metrics for the moderate FNSF studied here are given along with the ITER, and power plant values in Table II. These are reported in longer form in Appendix B.

From the first section in the Table on advancing fusion neutron exposure, the life of plant fluence reached in the FNSF, which is relevant to lifetime components like the VV, low temperature (LT) shield, and magnets, is 45 x higher than ITER, and > 6 x smaller than a power plant. These components can only be assessed after the facility is decommissioned, but material surveillance samples can be placed over a wide range of locations to monitor behavior during its operation. The DEMO must provide a significant advance in this parameter to approach power plant levels. In the FNSF the peak FW neutron fluence seen by the blanket before replacement ratchets up over five DT phases from 2 x to 25 x the ITER value, but the highest obtained would still be about 2-2.5 x lower than a power plant. Fusion core components are changed multiple times throughout the FNSF program. The peak neutron wall load, which is a measure of the irradiation rate, is over 2 x higher than ITER, and is just 78% of the power plant value. This parameter is rarely discussed in fusion in the damage of materials, in spite of phenomena related to the rate of damage observed in fission materials research. Power plant studies have reported peak neutron wall load values ranging from ~ 2-6 MW/m^2, with the lower end becoming more common recently as divertor heat load limitations are addressed and these plants tend to larger sizes.

In the section containing tritium production, the TBR obtained in the FNSF is larger than the power plant value. The power plant value is an optimized parameter to avoid producing too much tritium or too little tritium, and is actively adjusted during operation through the Li-6 fraction. In addition, it anticipates significant improvements in nuclear data, analysis, and modeling, and minimized plant inventories and processing times. The FNSF is found to provide sufficient tritium even with a pessimistic distribution of penetrations [2], with a Li-6 fraction of 80-90%. The FNSF value for the tritium produced per year is lower than a power plant due to the lower plasma on-time in a year and the lower fusion power compared to a power plant. The higher Li-6 enrichment in the FNSF is due to the larger fraction of the OB FW lost to penetrations, although it was found that 80% would provide a sufficiently high TBR of 1.04. For Li-6 enrichments below this the FNSF would not have the required margin. The margin in the TBR is determined by nuclear cross-section uncertainties, nuclear modeling approaches, blanket materials and their distribution, penetrations, and a range of tritium processing features (plasma fueling and exhaust, burn-up, doubling time, processing time, and reserve inventories [57]).

For the section describing the plasma performance, the plasma on-time per year combined with the duty cycle show that the FNSF is advancing strongly beyond ITER. Although the plasma pulse length is 400 x the longest ITER pulse, it is still about 20 x short of the power plant. The fusion gain is far from the power plant value, but can be increased by operating at higher normalized beta operation and lower density, which is one of the reasons to install the required systems to exceed the no-wall beta limit. This would be an important parameter left to the DEMO to advance significantly, and could prove to be challenging since the nonlinearity of the plasma behavior would increase as the ratio P_{alpha}/P_{aux} increases. This is further enhanced by a rising bootstrap current fraction, a current driven by the plasma's own density and temperature gradients. Based on existing and proposed facilities, it is not clear how the plasma gain can be explored, while high bootstrap fractions can be obtained in the long pulse Asian tokamaks and ITER. Roughly speaking a minimum fusion gain for a power plant is ~ 20.

The DEMO column is blank because this facility has not been sufficiently designed to accomplish the missions and technical basis we know it must deliver for commercial power plants to be pursued.

Table 2. A sampling of metrics for ITER, the moderate FNSF, the DEMO, and a power plant.

	ITER	Moderate FNSF	DEMO	Power Plant ARIES-ACT2
1. Strongly advance the fusion neutron exposure.....				
Life of plant peak FW fluence, MW-yr/m^2 (life of plant)	0.3	**13.7** **(7.8 FPY)**		88 (40 FPY)
Peak FW fluence to replace blanket, MW-yr/m^2 (dpa) (replacements)	0.3 (3) (0)	**0.7, 1.9, 3.1, 4.0, 8.0** **(7, 19, 31, 40, 80)** **(5)**		15-20 (150-200) (4-6)
Peak FW neutron wall load, MW/m^2 (average at plasma)	0.76 (0.56)	**1.77** **(1.18)**		2.2 (1.46)
4. Produce tritium in quantities that.....				
TBR - total		**1.07***		1.05
Tritium produced per year, kg	0.004	**10.7**		101-146
Li-6 enrichment		**90%**		40%
OB FW hole/loss fraction		**12-15 m^2 / 208 m^2**		36 m^2 / 1021 m^2
6. Routinely operate very long plasma durations....				
Plasma on-time per year (ave)	5%	**35%**		85%
Plasma pulse duration, s	500-3000	**$1.2x10^6$**		$2.7x10^7$
Plasma duty cycle	25%	**95%**		100%
$\beta_N H_{98} / q_{95}$	0.6	**0.4**		0.4 (2.1)**
Q	5-10	**4-6**		25 (48)
f_{BS}	0.25-0.5	**0.52**		0.77 (0.91)
$P_{core,rad} / (P_{alpha} + P_{aux})$	0.27	**0.24**		0.28 (0.46)
$P_{div,rad} / P_{SOL}$	0.7	**0.75-1.0**		0.75-1.0

*depending on assumed sector penetrations; TBR = 1.04 when Li-6 enrichment is 80%

**values in parentheses indicate those for an aggressive physics tokamak, ARIES-ACT1

IV. Plasma Physics Strategy

For the conventional aspect ratio tokamak FNSF studied here, the strategy for choosing plasma parameters to target for the FNSF was heavily influenced by the move to ultra-long plasma pulses, which is critical to establishing a basis for a power plant, with an anticipated plasma pulse lasting approximately a year. The longest plasma pulses prior to a FNSF would be those from JT-60SA at ~ 100 s [58], KSTAR at ~ 300 s [59], EAST at ~ 1000 s [60], and those from ITER at ~ 500-3000 s [61]. This is illustrated in Fig. 5, showing the plasma normalized beta versus pulse length for existing and planned tokamaks, and the FNSF, DEMO and power plants. The pulse lengths for 1 day, 2 weeks and 1 year are highlighted, showing the large gap between the longest ITER pulse lengths and one-day pulses on an FNSF, indicating the FNSF must fill this gap in its DD operating phase. Whether plasma operation above the no wall beta limit can be established and maintained is an important goal for shorter and longer pulse tokamaks, in order for such an operating regime to be projected to power plants. For power plants the target pulse length is ~ 3×10^7 s, a factor of > 10000 beyond ITER. In addition to this, the plasma pulse duty cycle (plasma on-time / (plasma on-time + dwell time)) must reach 100% in a power plant, while it is typically ~ 0.2% on present tokamaks, and anticipated to reach 25% in ITER. And finally the actual plasma on-time achieved per year must reach $\geq$ 85%, compared to ~ 0.05% in present tokamaks, and an anticipated 5% in ITER. This required push to much longer sustained plasma durations has tremendous impacts on all systems that support the plasma (e.g. fueling, heating), the plasma facing materials and components, and the need to diagnose and monitor the plasma chamber.

For the core plasma properties a conservative approach is taken on the plasma beta and energy confinement, while pursuing 100% non-inductive plasma current, high density relative to the Greenwald density, strong plasma shaping, and high radiative power fraction divertor concepts. In addition, the double-null (DN) divertor, high magnetic field in the plasma, and ELMs and mitigated disruptions are assumed in the design studies.

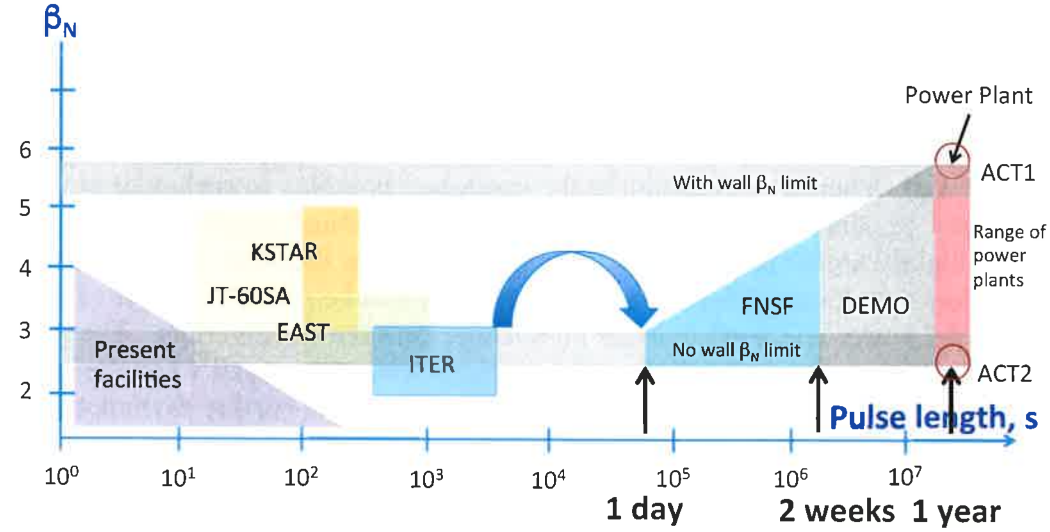


Figure 5. Illustration of the plasma normalized beta versus the plasma pulse duration, showing approximately the present tokamak achievements, the target β_N-duration space for JT-60SA, KSTAR, EAST, and ITER. The FNSF, DEMO and power plant regime are also shown, demonstrating a significant gap between planned facilities and the fusion nuclear next steps. Whether the operating space would remain around the no wall beta limit or approach the with wall beta limit still needs to be established.

Previous analysis [62] showed that the no wall beta limit associated with self-consistent 1.5D transport profiles and current drive profiles from lower hybrid (LH), ion cyclotron fast wave, (ICRF-FW), and negative ion neutral beams (NB) analysis yielded no wall normalized beta limits around 2.2 at l_i = 0.65 and 2.5 for l_i = 0.8-0.95. A conducting wall at 0.55a (a = minor radius) measured from the plasma surface would increase these to 2.8 at l_i = 0.85 and 3.25 at l_i = 0.65. These calculations also show that particular choices for pressure and current profiles can lead to slightly higher no wall beta values, up to 2.65 for example. For the FNSF design point conducting shells located at 1.0a (behind the OB breeding blanket) led to approximately no improvement to no wall limits, and 0.33a (behind the first OB breeding zone) showed > 20-40% increases in the beta limit over no wall limits. It was chosen to put the normalized beta limit at 2.7 for the systems analysis search for an operating point, guaranteeing that the fusion nuclear mission could be met without an aggressive plasma requirement. On the other hand, the FNSF would pursue the hardware (feedback coils, sensors) required to access above the no wall beta limit operation that has been achieved on DIII-D [63]. If the higher beta could be reached and sustained for the long plasma durations, the neutron wall load could be higher, the operating space could be more forgiving for other plasma parameters, and the program on the FNSF could be accomplished in less time.

Strong plasma shaping is chosen for the FNSF, with κ_X = 2.2, and δ_X = 0.58, due to its significant improvements to the beta limits and accessible operating space. This is also consistent with the choice for a double-null (DN) divertor since the X-points can not be separated significantly from the plasma boundary when the plasma is strongly shaped. The control of the vertical position is determined by the location of conducting structures and feedback coils [64], and this plasma elongation can be supported within the FNSF design, with tungsten plates located in the breeding blanket and feedback control coils located behind the blanket structure inside the vacuum vessel. A DN divertor is chosen to provide a dedicated power handling component to both upper and lower X-points, to avoid the difficulties with the in-active X-point in SN and the associated flux geometry on the first wall. There is a reduction in the maximum heat flux to each divertor, but this is not exactly ½, since it is expected that the plasma vertical position will be drifting, leading to slightly higher power and particle loads to one or the other of the divertors. In fact, intentional position shifting on a slow time-scale will be actively controlled and used to balance the power and particle loads on average between the divertors. The power to the upper and lower divertors is known from experiments to be equal when the plasma is not up-down symmetric [65-67], and the particle loads are also weakly asymmetric.

The plasma density is chosen to lie near the Greenwald density ($n/n_{Gr} = 0.9$), in spite of operating points at lower values being accessible. This is because power plant studies have shown how difficult it is to operate below the Greenwald density [68], due to the need to drive 100% of the plasma current non-inductively, driving I_P down, and the increasing size of the device, driving minor radius up ($n_{Gr} = I_P/\pi a^2$). A discussion can be found in [68] of tokamaks exceeding the Greenwald density. More recent results from ASDEX-U, with tungsten plasma facing material, shows [69,70] operation of plasmas in partially, pronounced, and fully detached divertor regimes, all with > 80% of the Greenwald density limit. These experiments provide important integration demonstrations for sustained high-density plasmas, with high radiation power fractions, reasonable energy confinement, and metal plasma facing components.

The divertor operating regimes considered for the FNSF in this study are two highly radiating conventional slot configurations. One is an ITER-like tilted plate concept which reaches $P_{div,rad} / P_{SOL} \sim 75\%$, and the other is an orthogonal plate wide slot concept that leads to $P_{div,rad} / P_{SOL} \sim 100\%$, shown in Fig. 6. These divertors are examined with 2D SOL/divertor simulations [8]. Radiating regime divertors for power handling are considered the only solutions for power plants, even with advanced divertor configurations or liquid metals. In fact, ITER already must operate with a radiating divertor to handle its power into the scrape-off layer, and their solution is a partial detachment regime. The ITER divertor cannot operate for more than ~ 1 s in an attached regime [71]. The two concepts chosen for FNSF have a minimum geometric footprint within the fusion core, and combines well with the other requirements of the blanket, structure/shield, and vacuum vessel such as tritium breeding and shielding. In systems analysis to identify operating points in this radiative regime, the radiated power fraction in the divertor of $P_{div,rad} / P_{SOL} = 90\%$.

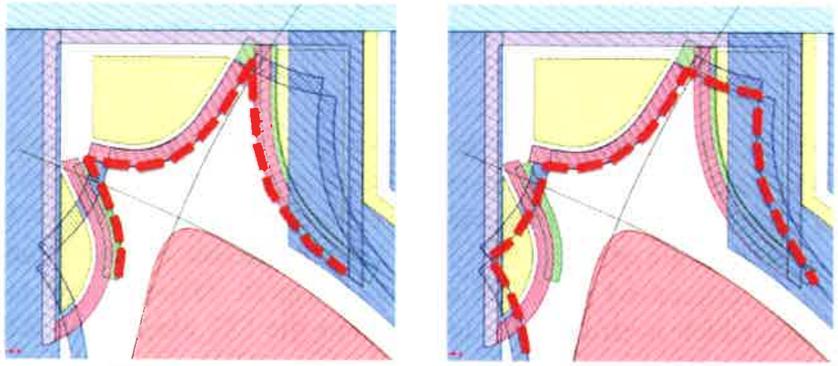


Figure 6. Two divertor configurations examined in the FNSF, can each be accommodated within the blanket, structure/shield and vacuum vessel and their requirements. The ITER-like tilted plate divertor obtains 75% and the orthogonal plate divertor obtains 100% radiated power fractions. The fully detached divertor requires a slightly larger envelope, but this is acceptable based on 3D nuclear analysis.

The divertor heat flux is targeted at a maximum of 10 MW/m^2, considered reachable with tungsten armor and structure, and He cooling based on thermo-mechanics, computational fluid dynamics (CFD), and experiments [42-49]. Although higher heat fluxes have been obtained, this value represents a compromise between achievable heat flux, pumping power compared to thermal power removed, and design complexity. The tungsten plate design [72] is considered the basis for the FNSF. All loading conditions relevant to the divertor have not been considered in this design, and this effort to characterize these is continuing. The systems analysis uses this target value when searching for viable operating points, while the independent 2D SOL/divertor simulations provide solutions that are checked for this condition as well.

The plasma current is taken to be 100% non-inductive in the flattop phase, while the central solenoid (CS) coils and poloidal field (PF) coils provide inductive assistance in the rampup phase and in feedback control during the flattop (this will depend on the ultimate control algorithm used). The plasma current is composed of the bootstrap current, and external contributions from NB, LH, ICRF-FW, and possibly EC. Different combinations are examined due to the uncertainty in which sources will become most viable, and capable of being extrapolated to the DEMO and power plants. The installed power to support these various sources to drive current in the plasma must be sufficient to provide a robust solution in spite of possible shortfalls in plasma parameters. This has made the FNSF choices more conservative in terms of current drive efficiency and operating density, generally avoiding highly optimistic projections based on low density operating points with high beta. These sources have a negative impact on the tritium breeding and neutron shielding that is significant and has been examined in detail in nuclear analysis. Power densities through the FW for each of the sources are based on experimental and ITER values where available [73-74]. The systems analysis uses a generic low-end current drive efficiency of 0.2 A/W-m^2 in order to guarantee sufficient installed power, and is combined with operating points at higher density relative to

Greenwald. Detailed simulations of each of the heating and current drive schemes for flattop conditions provides accurate current and current profiles, which are reported in [13].

The FNSF will be designed to withstand disruptions, however, it is assumed that the science of disruption mitigation will be established, moving the vast majority (~90%) of plasma stored energy released in the thermal quench to radiation to the first wall [75], and the runaway electrons are eliminated. The current quench is largely unchanged and scaled to meet the criteria set for ITER maximum dIp/dt. Halo currents would be with the inboard wall, which is structurally robust. For the DN plasma the most dangerous and most likely disruption is the midplane disruption (MD), where the plasma releases 100% of its flattop stored energy in the thermal quench, and the plasma remains at or close to the midplane. The up-down symmetry of this configuration means that the plasma sits at its neutral point [76]. Motion away from the neutral point is slow and easily detectable, allowing a mitigation pellet injection to terminate the plasma before any significant vertical excursion.

An examination of FNSF relevant plasma discharges from experiments shows some important trends. Although very long pulse duration do not exist, the long discharge extension in JT-60U from ~ 5 s to 20-30 s discharges, with extensive hardware upgrades in the early 2000's, showed that plasma configurations identified at shorter pulse could in fact be recovered and sustained in much longer duration [77]. All of the high performance scenarios, high beta-poloidal, high bootstrap fraction, and high normalized beta were re-created.

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, $\tau_{CR} = \mu_o a^2 \kappa / 12 \langle \eta_{neo} \rangle$, where $\langle \eta_{neo} \rangle$ is the volume average neoclassical resistivity, and the longest tokamak discharges relative to this are ~ 15 τ_{CR} in JT-60U [77-79]. 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. 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 $\leq 1\ \tau_{CR}$ [80]. More recently [81,82] 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 [83-85]. 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, and recent experiments have begun to explore this regime [86], obtaining 100% non-inductive plasmas with $\beta_N \sim 3.5$, $H_{98} \sim 1.6$, $f_{BS} \sim$ 50-60%, $q_{95} \sim$ 5.5-6.5, lasting for about 1.5 τ_{CR}.

Table 3. Parameters for experimental discharges achieved in JT-60U and DIII-D, and the FNSF operating point.

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

The interplay between resistive wall mode stabilization, error field correction, and the behavior in the vicinity of the no wall beta limit is an important research area to provide the means to project to a FNSF. Plasma performance in DIII-D can be lower when the plasma rotation is reduced. Ongoing research is focused on creating high performance plasmas with long term relevant characteristics including low rotation, dominant electron heating and $T_e \sim T_i$, and integration with radiating divertors.

Recent experiments in ASDEX-U [69,70] have demonstrated operation simultaneously with high plasma density ($n/n_{Gr} > 0.75$-1.0), with tungsten plasma facing material, and controlled radiation from the core and divertor plasmas. Both partial and full detachment regimes have been accessed and sustained. These are important integration demonstrations for the FNSF operating regime and address other features than those in Table III.

KSTAR has recently reported their longest sustained H-mode discharges, extending a ~ 15 s discharge to 70 s, with parameters $\beta_N \sim 2$, $f_{NI} \sim 1$, $f_{BS} \sim 50\%$ [87]. The EAST experiment [88] has obtained 100% non-inductive plasma current for ~ 6 s, which was 15 τ_{CR}, with $H_{98} \sim 1.1$ and $q_{95} \sim 6.3$, with all RF heating. This has been extended to 60 s. EAST has established an ELM suppressed regime and sustained it for over 26 s. These facilities will contribute significantly in the future to the basis for an ultra-long pulse needed in the FNSF.

V. Systems Analysis to Establish an Operating Point and Operating Space

Systems analysis is used to search for operating points that obey global physics, engineering, and imposed constraints. This mainly establishes the plasma geometry around which the device can be built, and allows exploration over a wide range of parameters, that is not practical with more detailed simulations. The systems analysis includes a zero-dimensional plasma description for power and particle balance, along with plasma radiation, external heating and current drive, bootstrap current, using parabolic profiles with finite edge profiles for density and temperature, and up to 3 impurities. This analysis includes a series of simple engineering models for heat flux, power balance components, TF coil, bucking cylinder and PF/CS coils. The inboard radial build is provided by preliminary nuclear analysis. The systems code uses a database approach where large numbers of viable operating points are identified, and filtered by constraints to produce a range of desirable configurations for a viable "operating space". One particular operating point is selected for specifying the device for use with detailed physics and engineering analyses.

V.1. Establishing a Reference Operating Point Geometry

Preliminary 1D nuclear analysis [2] was generated in order to construct the inboard radial build of first wall, breeding blanket, structural/shield ring, vacuum vessel, and low temperature shield. The definition of the inboard radial build is critical to the device's size. The neutron damage to the superconducting TF coils, dose to the insulator in the TF coils, heating in the winding pack and coil case, and damage to the copper stabilizer were examined to set the requirements for overall shielding (reduction in neutron flux and high energy component of the neutron energy spectrum). Since there is a desire to keep the device smaller at the break-in step of an FNSF, tungsten carbide (WC) was used for the shielding filler in the structural ring, vacuum vessel and low temperature shield, due its very high shielding efficiency Only helium cooling is allowed inside and in the vacuum vessel, and water-cooling is only allowed outside the vacuum vessel. In the DCLL blanket concept, the lead-lithium breeder also cools the blanket. These cooling choices have strongly impacted the nuclear aspects of the device. Shown in Fig. 7, the resulting inboard build is composed of 50 cm of first wall and breeding blanket, 20 cm for the structural ring, 10 cm for the vacuum vessel, and 23 cm for the low temperature shield, giving a total material build to the TF coil of 103 cm. The systems code has assumed 20 additional cm's of undefined gaps, and a 10 cm plasma scrape-off width on the inboard, leading to the inboard build distance from the plasma to the TF coil of 133 cm.

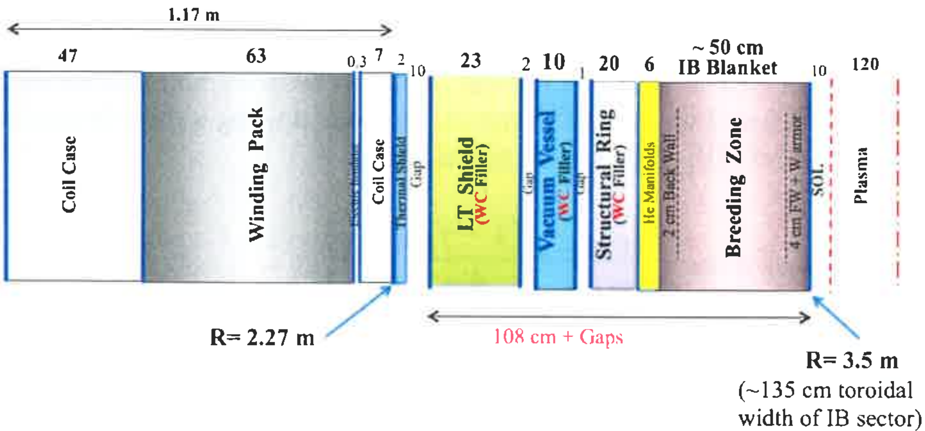


Figure 7. Midplane inboard radial build for the FNSF determined by 1D nuclear analysis and confirmed by 3D nuclear analysis to meet breeding and shielding requirements. This illustrates the build components, whose detailed geometry and composition are reported elsewhere [2]. The total radial build from the IB first wall to the TF coil is 1.23 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 $\leq$ 16 T. The overall current density in the TF coil can be related directly to the stress allowable with the following global equation,

$$< j_{TF} > = \frac{f_{str}\,\sigma_{all} - B^2_{TF,coil}/2\mu_o}{\sigma_{all}\left(\frac{1}{j_{SC}} + \frac{1}{j_{stab}}\right) + \left(\frac{B_{TF}R}{4}\right)\ln\left(\frac{R_{outer}}{R_{inner}}\right) - \frac{\sigma_{stab}}{j_{stab}}}$$

Here σ_{all} is the allowable stress in the steel structure, f_{str} is the fraction of structure in the coil, σ_{stab} is the allowable stress in the stabilizer, $B_{TF,coil}$ is the toroidal field at the coil, B_{TF} is the toroidal field at the plasma center (R), and R_{outer} and R_{inner} are the midplane outboard and inboard TF coil radii. A bucking cylinder can be included in the build and is sized by TF coil pressure. The CS coil is sized by the required flux swing to rampup the plasma current, and along with the PF coils, is also sized to provide equilibrium coil currents scaled from previous equilibria. This provides an envelope for these coils, although detailed equilibria are required to size the final coils.

The plant power balance is represented by

$$P_{elec,net} = \eta_{th}\left(M_n P_n + P_{plas}\right)\left(1 - f_{subs}\right) - \frac{P_{aux}}{\eta_{aux}} - \frac{P_{pump}}{\eta_{pump}}$$

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