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A Fusion Nuclear Science Facility for a fast-track path to DEMO

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HIGHLIGHTS

- A FNSF is needed to reduce the knowledge gaps to a fusion DEMO and accelerate progress toward fusion energy.
- FNSF will test and qualify first-wall/blanket components and materials in a DEMO-relevant fusion environment.
- The Advanced Tokamak approach enables reduced size and risks, and is on a direct path to an attractive target power plant.
- Near term research focus on specific tasks can enable starting FNSF construction within the next ten years.

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ABSTRACT

An accelerated fusion energy development program, a “fast-track” approach, requires proceeding with a nuclear and materials testing program in parallel with research on burning plasmas, ITER. A Fusion Nuclear Science Facility (FNSF) would address many of the key issues that need to be addressed prior to DEMO, including breeding tritium and completing the fuel cycle, qualifying nuclear materials for high fluence, developing suitable materials for the plasma-boundary interface, and demonstrating power extraction. The Advanced Tokamak (AT) is a strong candidate for an FNSF as a consequence of its mature physics base, capability to address the key issues, and the direct relevance to an attractive target power plant. The standard aspect ratio provides space for a solenoid, assuring robust plasma current initiation, and for an inboard blanket, assuring robust tritium breeding ratio (TBR) > 1 for FNSF tritium self-sufficiency and building of inventory needed to start up DEMO. An example design point gives a moderate sized Cu-coil device with $R/a = 2.7\,\text{m}/0.77\,\text{m}$, $\kappa = 2.3$, $B_T = 5.4\,\text{T}$, $I_P = 6.6$ MA, $\beta_N = 2.75$, $P_{fus} = 127\,\text{MW}$. The modest bootstrap fraction of $f_{BS} = 0.55$ provides an opportunity to develop steady state with sufficient current drive for adequate control. Proceeding with a FNSF in parallel with ITER provides a strong basis to begin construction of DEMO upon the achievement of $Q \sim 10$ in ITER.

1. Introduction

An accelerated fusion energy development program, a “fast-track” approach, requires developing an understanding of fusion nuclear science (FNS) in parallel with research on ITER to study burning plasmas. A Fusion Nuclear Science Facility (FNSF) in parallel with ITER provides the capability to resolve FNS feasibility issues related to power extraction, tritium fuel sustainability, and reliability, and to begin construction of DEMO upon the achievement of $Q \sim 10$ in ITER.

Fusion nuclear components, including the first wall (FW)/blanket, divertor, heating/fueling systems, etc. are complex *systems* with many inter-related functions and different materials, fluids, and physical interfaces. These in-vessel nuclear components must operate continuously and reliably with: (a) *Plasma exposure*, surface particle & radiation loads, (b) *High energy neutron fluxes* and their interactions in materials (e.g. peaked volumetric heating with steep gradients, tritium production, activation, atomic displacements, gas production, etc.), (c) *Strong magnetic fields* with temporal and spatial

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variations (electromagnetic coupling to the plasma including off-normal events like disruptions), and (d) a *High temperature*, high vacuum, chemically active environment. While many of these conditions and effects are being studied with separate and multiple effect experimental test stands and modeling, fusion nuclear conditions cannot be completely simulated outside the fusion environment. This means there are many new multi-physics, multi-scale phenomena and synergistic effects yet to be discovered and accounted for in the understanding, design and operation of fusion as a self-sustaining, energy producing system, and significant experimentation and operational experience in a true fusion environment is an essential requirement.

In the following sections we discuss the FNSF objectives, describe the facility requirements and a facility concept and operation approach that can accomplish those objectives, and assess the readiness to construct with respect to several key FNSF issues: materials, steady-state operation, disruptions, power exhaust, and breeding blanket. Finally we present our conclusions.

2. Objectives of a FNSF

The research aims of FNSF will be to provide the first opportunity to explore the behavior of nuclear components and materials in the fusion nuclear environment and discover new multiple effects/multiple interactions phenomena, and to develop the definitive database essential for both the validation of fusion nuclear science modeling, design and safety codes, and ultimately the demonstration of DEMO-ready in-vessel component operation and reliability. To accomplish these goals, FNSF will need to pursue the following objectives:

- Demonstrate that fusion can *make its own fuel*, and the capability to produce a tritium supply to start up subsequent machines (e.g. DEMO).
- Develop fusion FW/blanket, and divertor systems that operate effectively in the fusion environment and *extract high grade heat* over an extended lifetime while remaining compatible with plasma operation and tritium production.
- Obtain the essential database on *reliability*, *availability*, *maintainability* and *inspectability* (RAMI) of fusion nuclear components beyond beginning of life.
- Build the knowledge base for *integrated fusion nuclear science phenomena* including but not limited to: mixed material formation and evolution of the near surface; neutron irradiation damage and effects on implanted D, T, and He behavior; evolution of material thermomechanical properties, liquid metal coolant MHD, stability and transport; tritium solubility, diffusivity and permeation; electromagnetic compatibility, chemical compatibilities.

3. Facility description

The technical requirements of a fusion test facility to study and resolve fusion nuclear science questions has been previously specified by international experts [1] as a high-volume, plasma based neutron source for well diagnosed, integrated tests of materials and full-size components under prototypical conditions. In order to accomplish these nuclear science objectives, the FNSF should operate steady-state for periods of up to two weeks, with a significant duty cycle (e.g. 30%) and significant fusion power for a neutron wall loading of $\sim$1–2 MW/m^2 and a neutron fluence of 3–6 MW yr/m^2 on large sample volume over twenty years (10–20 times the fluence in ITER).

An FNSF designed to achieve the objectives above will be complementary to ITER. With its largesize and plasma current, ITER will explore high fusion gain $Q\sim 10$ and superconducting technology and reactor scale maintenance schemes. FNSF is conceived as a smaller, low fusion power, significantly higher neutron flux, externally driven device (low $Q\sim 3$) with copper coils enabling the flexibility and maintainability required to advance our knowledge of in-vessel component and material performance in an integrated fusion environment, and to demonstrate the harnessing of fusion power. The facility would also have the capability to develop advanced steady-state operating modes toward an attractive power plant. Such a FNSF together with ITER and neutron irradiation facilities would fulfill the critical development gaps from present machines to a tokamak DEMO, identified in recent U.S. MFE community initiatives [2]. There are two main candidate machine types proposed recently in the US aimed at the FNSF mission: FNSF-AT (Fusion Nuclear Science Facility-Advanced Tokamak) [3] and FNSF-ST (Fusion Nuclear Science Facility-Spherical Torus) [4], though other variants could still be envisioned. The FNSF-AT is a strong candidate for an FNSF as a consequence of its mature physics base, capability to address the key issues and to demonstrate plasma scenarios directly relevant to an attractive target power plant. The standard aspect ratio provides space for a solenoid, assuring robust plasma current initiation, and for an inboard blanket, assuring robust tritium breeding ratio (TBR) >1 for FNSF tritium self-sufficiency and building of inventory needed to start up DEMO. A 0-D system optimizer model [3] benchmarked against first principle 2-D transport simulations [5] gives a copper coil device with R/a = 2.7 m/0.77 m, κ = 2.3, B_T = 5.4 T, I_P = 6.6 MA (where R and a are the tokamak major and minor radii, k is the plasma elongation, B_T is the toroidal magnetic field evaluated at R, and I_P is the total plasma current). With these engineering parameters, FNSF-AT could achieve the minimum required neutron wall loading of $\sim$1 MW/m^2 with AT physics parameters of normalized performance, β_N = 2.75, β_T = 4.3%, f_{BS} = 0.55, which have already been largely exceeded in DIII-D experiments of moderate pulse length [6] [here $\beta_N = \beta_T/(I_P/aB_T)$, $\beta = 2\mu_0\langle p\rangle/B_0^2$ is the ratio of volume-averaged plasma pressure (including fast ions) to toroidal magnetic field pressure, and f_{BS} is the ratio of self-generated bootstrap current to total plasma current]. A more detailed parameter list for a range of steady state operating scenarios with varying levels of Advanced Tokamak physics is shown in Table 1. The column labeled 2 MW/m^2 is the baseline design target, aimed at achieving the upper range of the neutron wall loading required to meet the nuclear science and technology mission. This is the mode of operation that requires the largest amount of auxiliary power. The column labeled 1 MW/m^2 shows that even with significantly reduce plasma physics performance, the machine can meet the minimum neutron wall loading requirements. The column labeled 3 MW/m^2 looks at increasing the plasma performance toward levels of an advanced DEMO. Achievement of this mode of operation is an open-ended goal foe FNSF; the machine hardware will be capable of such mode (no additional auxiliary power is required) if the physics allows it. The higher performance is achieved by increasing the density: this leads to higher fusion power, higher confinement, higher β_N, and higher bootstrap fraction, so that no additional auxiliary power is required with respect to the baseline scenario. Clearly, higher performance operation, if achieved, would require shorter plasma on-time, in order to stay within acceptable neutron damage lifetime limits.

The essential design feature of FNSF that allows it to be an effective research machine, enabling frequent planned changeouts and maintenance of the in-vessel components, is its jointed copper TF coil, as in DIII-D, Alcator C-Mod, and NSTX. The choice of copper coils minimizes shielding requirements, significantly reducing the size and cost of the facility. On the other hand, the choice of copper coils leads to large electrical powered dissipated in the coils, and large overall power required to run the machine, estimated

Table 1
FNSF-AT parameters for three steady-state operating modes with varying levels of Advanced Tokamak physics, from conservative to aggressive, all satisfying the requirements of the nuclear technology mission.

	Nominal neutron wall loading	1 MW/m^2	2 MW/m^2	3 MW/m^2
		ECCD + LHCD	Only ECCD	Only ECCD
A	Aspect ratio	3.52	3.52	3.52
a	Plasma minor radius (m)	0.77	0.77	0.77
R_o	Plasma major radius (m)	2.70	2.70	2.70
k	Plasma elongation	2.31	2.31	2.31
P_{fus}	Fusion power (MW)	127	228	386
Q_{plasma}	P_{fus}/P_{cd}	1.73	2.66	5.58
Φ_n	Peak neutron flux at wall (MW/m^2)	1.05	1.89	3.19
β_T	Toroidal beta (%)	4.3	5.8	7.6
β_N	Normalized beta (% mT/MA)	2.75	3.69	4.50
f_{BS}	Bootstrap fraction	0.55	0.74	0.85
P_{cd}	Current drive power (MW)	73.2	85.8	69.2
I_P	Plasma current (MA)	6.60	6.60	7.03
B_o	Field on axis (T)	5.46	5.46	5.46
q	Safety factor	4.93	4.93	4.63
$T_i(0)$	Ion temperature (keV)	20	20	20
$n(0)$	Electron density (10^{20}/m^3)	2.18	2.92	3.8
$\langle n\rangle/n_{GR}$	Ratio to Greenwald limit	0.47	0.63	0.77
Z_{eff}	Effective atomic number	1.8	1.8	1.8
$H_{98(y,2)}$	H factor over ELMy H	1.13	1.23	1.33
P_{heat}/A_{wall}	Surface average power density (MW/m^2)	0.62	0.82	0.91

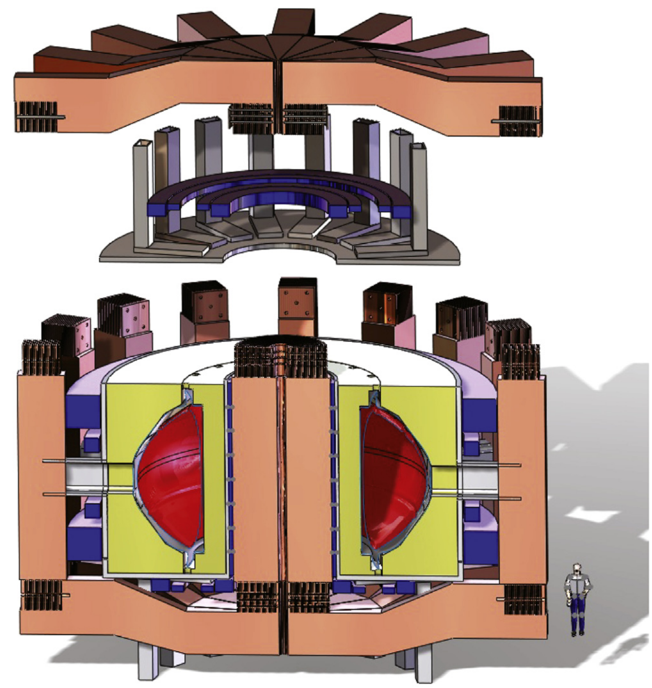

Fig. 1. Baseline maintenance scheme for FNSF-AT is based on vertical crane lift of toroidally continuous ring structures.

to be ~600 MW, that is between the steady-state and peak power demands expected for ITER (200–800 MW) [7].

The TF coil joint allows a vertical maintenance scheme in which the divertor and FW/blanket structures inside the TF coil can be built as axisymmetric ring structures and maintained and changed out as large units. The construction as axisymmetric rings also enables a baseline option, based on precision toroidal alignment of plasma-facing surfaces, for handling of the plasma exhaust. This vertical maintenance approach is shown in Fig. 1.

4. Staged testing approach

A staged research and development schedule is a defining characteristic of FNSF, essential in order to be able to learn and improve over time the operating scenarios, diagnostics, nuclear components, and structural and plasma facing materials. This enables an early start for FNSF that maximizes synergy with the parallel ITER burning plasma program. A schematic time-table of the FNSF development schedule referenced to major ITER and DEMO milestones is shown in Fig. 2. FNSF construction should begin in the early 2020s in order to begin operation by the year 2030. An initial commissioning period of 3–4 years is envisioned in which the working fuel will progress from H to D to D–T. The basic operating modes of the machine can be developed in this phase with sufficient auxiliary power without dependence on fusion power. This first stage would also allow the study of plasma surface interactions, enabling improved understanding of materials under simultaneous exposure to high plasma heat and ion fluxes, and testing FW, divertor, and MHD liquid-metal flow in the blanket components together at sufficient engineering size and prototypic integrated environment with all conditions except neutrons. In the following D–T operation period, three upgrades of full-coverage ("main") blanket systems could be tested, while the plasma performance, the blanket system design and operation, the closed loop tritium system design and operation, are gradually improved. Until the first main blanket starts to produce net tritium, the facility will be a net tritium consumer. This strongly motivates a reduced facility size, hence the choice of copper coils (less shielding necessary compared to superconductive coils) and AT physics (higher power density compared to standard tokamak physics). Planned shutdowns will enable each main blanket change-out. In parallel to main blanket research, several port-based, test blanket modules (TBMs) could be studied in specially designed FNSF test ports. Thus, the third main blanket design could take advantage of input from the TBM tests, using more advanced materials if available, to operate to higher fluence. The price paid for this necessary flexibility in the FNSF testing strategy is the high power consumption in the copper coils and significant downtime for in-vessel component change-outs.

5. Readiness to construct – materials

An aggressive fusion nuclear science research program is urgent and essential to advance understanding and simulation of FW/blanket, divertor and tritium system materials.

5.1. Structural materials

The current generation reduced activation ferritic steel (e.g. F82H, EUROFER) has a sufficient irradiation database to serve as the main structural material for first phases of FNSF, able to tolerate up to ~30 dpa at 500–520 °C [8]. Key uncertainties include ductile-brittle transition temperature (DBTT) increase due to fusion H, He effects and dose limits in fusion neutron environment. Risk mitigation options include oxide dispersion strengthened (ODS) steels and new ferritic/martensitic steels with a very high precipitate density designed with computational thermodynamics tools. These more advanced neutron-resistant materials can be tested during the early phases of FNSF for possible adoption in the later, higher fluence phases of operation. Required functional materials (insulators, tritium barriers, armors, breeders, etc.) are less developed, especially in the US, and need more focus in the near term research program.

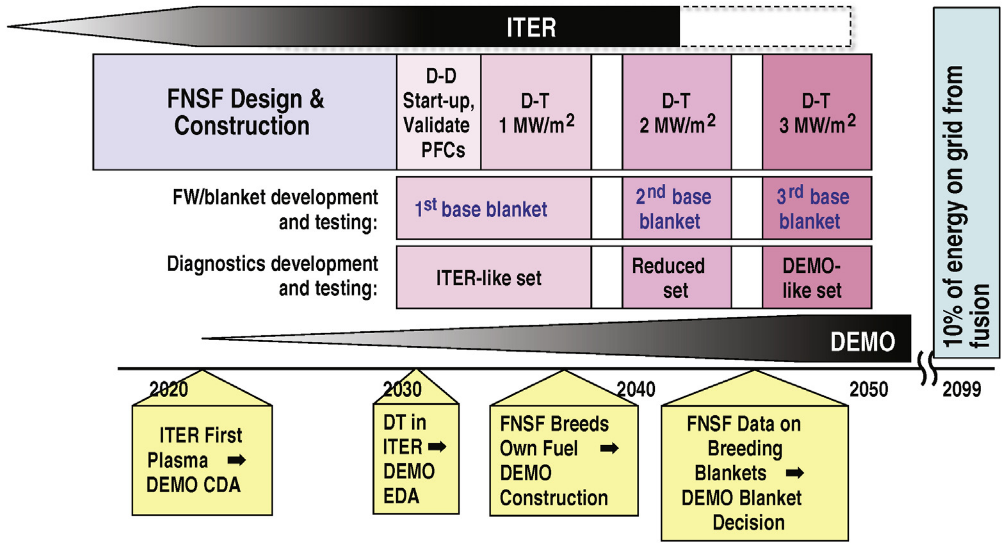

Fig. 2. A staged development to learning and improving nuclear components, diagnostics, and operating scenarios enables an early start for FNSF that maximizes synergy with the parallel ITER program: the "fast-track" approach.

5.2. Plasma-material interaction

The requirement to avoid the cyclic heat loads from ELMs has to be addressed for ITER, so no additional research is required for FNSF. The main challenges are erosion of plasma facing surfaces, the transport and redeposition of eroded material, and the trapping of tritium in redeposited layers and tokamak dust. With 10–20 times greater plasma fluence onto surfaces than ITER, FNSF-AT will make a major contribution toward testing DEMO relevant solutions. Key issues to be resolved for FNSF design include use of tungsten armor at high wall temperatures, and use of innovative divertor approaches (e.g., Snowflake, Super-X, or liquid walls). The existing database shows tungsten armor can work if the electron temperature is <10 eV at the divertor strike points (detached or semi-detached operation), and the ion impact energy is <200 eV at the chamber wall. Melting should be minimized to avoid killer contamination (avoid exposed edges). Long-term material migration may be a concern. Measurements in existing tokamaks (total exposure $\sim 10^4$ s/yr in DIII-D, 10^5 s/yr in JET) can determine rates, using surface analysis. Recent results from JET [9] showing the mitigation of erosion of the tungsten divertor by local nitrogen seeding are very promising. New improved divertor geometries, flux expansion and physical isolation concepts can be tested in existing or upgraded facilities, which have the comprehensive set of diagnostics needed to interpret results and move beyond empiricism toward validated models.

Because of the major goal of demonstrating TBR ≥ 1, FNSF-AT must develop solutions to the tritium retention and permeation issues. Operation with hot walls will enable research in the reactor-relevant regime that builds on a growing database from supporting materials science facilities.

6. Readiness to construct – steady-state operation

FNSF-AT/ST design concepts feature fully non-inductive current drive, strong plasma cross section shaping, internal profiles consistent with high bootstrap fraction, and operation at moderate to high beta. The FNSF-AT vision is largely based on the advanced physics performance and key features demonstrated on DIII-D, JT-60U, JET, and ASDEX-U, extended to very long pulse in a nuclear environment. At $\kappa = 2.3$, $\beta_N = 3.7$, and $f_{BS} \sim 0.75$ the baseline AT scenario achieves peak neutron flux at the outboard wall of ~ 2 MW/m^2, sufficient for nuclear science objectives with ample margin over the minimum requirement (~ 1 MW/m^2). This provides risk mitigation against shortfalls in physics and/or engineering performance. For this baseline scenario, physics based transport simulations presented in Ref. [5] have found steady-state equilibria with good stability and controllability properties. A high level of $n = 0$ mode controllability even at elongation $\kappa = 2.3$ follows mainly from two features of the FNSF-AT concept: construction in toroidally continuous rings, which increases the stabilizing effect of the wall, and low internal inductance of the plasma, which increases the plasma-wall coupling.

Experiments on various tokamaks have shown that fully non-inductive scenarios with the minimum levels of normalized fusion performance required by FNSF-AT (1 MW/m^2 column in Table 1) are obtained and sustained for many energy confinement times [10–12]. Key remaining challenges are to sustain these AT scenarios for several current redistribution times, and develop high heat and particle fluence boundary solutions consistent with high plasma performance. Experiments in superconducting tokamaks such as EAST and KSTAR will address these issues in the next few years.

7. Readiness to construct – disruptions

Disruptivity in FNSF should be as low as reasonably achievable. Disruptions that do occur will be mitigated. FNSF would require methods of limiting unmitigated disruptions to about one per year. Disruption avoidance includes several layers of plasma control. The plasma shape, pressure profile, and current density profile will be controlled to avoid stability limits; for greatest accuracy the stability limits should be calculated in real time with measured plasma parameters. Some instabilities (e.g., neoclassical tearing modes) are amenable to active suppression. Conditions potentially leading to a disruption may still result from unanticipated events such as a sudden influx of impurities or failure of a control actuator. If a disruption threatens, the control system will attempt a controlled shutdown, or "soft landing". As a last resort, when a disruption is inevitable, the mitigation system will create a rapid shutdown by gas injection or other means, now under development for ITER.

Algorithms for disruption avoidance and actuators for disruption mitigation are problems that must be solved before ITER begins high power operation. Present facilities have already demonstrated advanced disruption warning systems that can predict oncoming

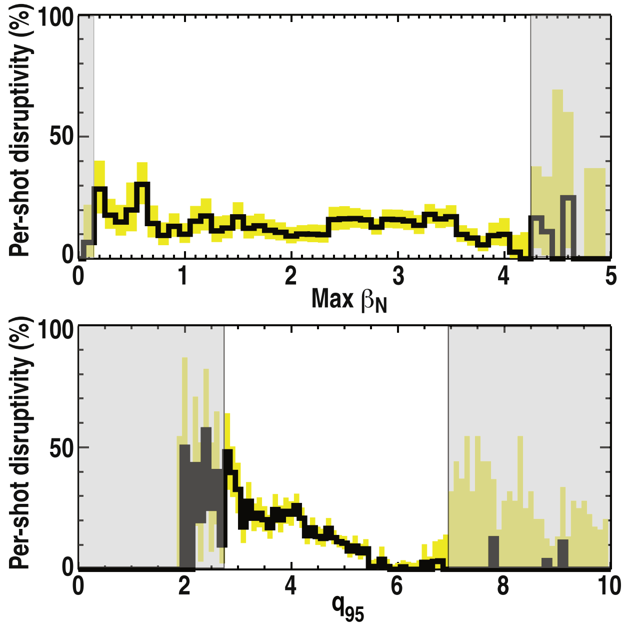

Fig. 3. Analysis of the flattop disruptivity database from DIII-D. Only disruptions not caused by operator error or power supply failures are counted as disruptions in this database of about 6000 cases. Plots show the per-shot disruptivity as a function of (a) the maximum β_N reached during the flat-top before the disruption, and (b) q_{95} measured at the time of maximum β_N. The yellow band around the solid lines is the 90% confidence interval. The gray shaded areas cover parameter ranges where the discharge population drops below 20 per bin. (For interpretation of the references to color in this figure legend, the reader is referred to the web version of this article.)

disruptions with high reliability: more than 96% disruptions are recognized in advance in NSTX [13], and more than 98% in JET [14]. Indeed, the requirements for disruption avoidance in FNSF are intermediate between those of ITER and DEMO. However, reduced disruptivity is observed in high β operation in regimes with high q_{95}, high $q_{\min}$, and broad current profiles (low l_i), such as the regimes envisioned for FNSF and DEMO. This can be seen clearly in the results of recent analysis of the DIII-D disruptivity database [15], shown in Fig. 3: in a database of about 6000 cases, disruptivity is constant or decreasing with higher β_N, and decreases strongly with increasing q_{95}. Thus the Advanced Tokamak approach, which pursues high β_T by increasing β_N instead of the plasma current, strongly reduces the disruption risk. Key challenges for FNSF will be to provide the diagnostic measurements in a high neutron fluence environment along with adequate actuators.

8. Readiness to construct – power exhaust

Peak power deposition in the FNSF-AT divertor is calculated to be similar or lower than that in ITER, even though FNSF-AT has a higher surface-average power density (~0.8 shown in Table 1 vs. ~0.2 in ITER) that is comparable to that in DEMO (~1.0). The key assumptions that lead to this conclusion are: (1) achieving ~50% core radiation fraction, and (2) achieving excellent alignment of the divertor target plates.

With ~50% core radiation fraction, the ratio $P_{\rm sol}/R_{\rm div}$ ($P_{\rm sol}$ = heating power into the scrape off layer, $R_{\rm div}$ = major radius at divertor strike point) of ~25 in FNSF-AT is only about 40% higher than the value for ITER ~18. In addition, the FNSF-AT construction scheme (demountable TF coils and toroidally continuous ring structures that can be crane-lifted vertically) allows achieving and maintaining excellent alignment of the top surfaces of the divertor tiles (<0.1 mm, as achieved for example in DIII-D [16]), thus enabling full use of strong divertor plate tilting and flux expansion: the angle between magnetic field and plate can be as low as ~1° in FNSF-AT versus ~2.5° in ITER. This increases by more than a factor of two the wetted area of the divertor. Therefore, for similar transport coefficients, it is reasonable to expect lower heat fluxes in FNSF-AT than in ITER. More detailed divertor modeling fully confirms these simple estimates: 2-D analysis [5] of the baseline scenario (2 MW/m^2) predicts peak that heat flux <10 MW/m^2 at the outer divertor targets and electron temperature at the target below 10 eV can be achieved using low to moderate deuterium gas puffing in the divertor, and assuming ITER heat and particle diffusion coefficients in the SOL (which yield a heat flux width at the X-point of about 1.4 mm). Note that the poloidal field is lower in FNSF-AT then in ITER, thus yielding a larger heat flux width according to recent scaling studies [17].

In support of the assumption (1), we note that feedback techniques developed on ASDEX-Upgrade for optimization of power exhaust with a standard vertical target divertor have achieved and maintained $P_{\rm rad}/P_{\rm NBI}\sim 87\%$ with $Z_{\rm eff}\sim 2.0$, $\beta_N = 3$, $H_{98(y,2)} = 1$ [18]. However, the core radiation impact on performance at FNSF-AT conditions is so far not known. First principle transport simulations including the radiation losses are required to investigate the effects of different impurity species, such as neon, argon, and krypton. With regards to assumption (2), we note again that an essential feature of the FNSF concept is the demountable copper coil, which enables a vertical maintenance scheme and the construction of divertor components as toroidally continuous ring structures, where precision placement of divertor plates can be readily achieved.

9. Readiness to construct – breeding blanket

The approach taken will be to engineer the first full blanket as breeding blanket of the same type as options proposed for DEMO, but operated in conservative temperature range to reduce risk. Candidate blankets with broader operating parameters and less conservative temperature range will be tested in port blanket modules. Experience from the first stage of operation will be utilized to engineer the second-generation full blankets. FNSF-AT will be designed to facilitate changeout of the full first wall/blanket structures and will do so at least twice in the life of the project. Dual Coolant Lead Lithium (DCLL) is considered the leading system in the US. Features and R&D issues are typical of a family of PbLi and/or helium cooled FW/blankets. Backup options include the helium-cooled ceramic breeder (HCCB) design, which is widely used in the ITER TBM program, and water-cooled blanket designs, which offer the advantage of higher heat removal capability at the cost of reduced potential for high thermal efficiency. 3-D neutronics calculations for the FNSF-AT baseline configuration have been carried out for the DCLL and HCCB options, yielding TBR > 1 when the loss of blanket coverage from midplane ports required by NBI is avoided or minimized [19]. Note that the TBR values from these calculations might be slightly optimistic because heterogeneous lattices of materials making up the blanket are replaced with homogenized mixtures in the model. More accurate calculations using heterogeneous models are in progress.

FNSF will be the facility for testing FW/blanket structures and improving reliability from initial designs. To achieve DEMO readiness, the MTBF (mean time between failures) of these components will have to reach up to ~11 years, and the MTTR (mean time to repair) down to ~2 weeks. These will be the most serious challenges for FNSF. Performance, design margin, and failure modes/rates should now be the R&D focus of the FNS program.

Blanket readiness for FNSF will require the utilization, in the ~15 years between now and the envisioned start of FNSF operation, of several multiple effect facilities to simulate to the degree possible all conditions. These include a Blanket Thermomechanics Thermofluid Test Facility (with simulated surface and volume heating, reactor like magnetic fields, for testing of mockups and ancillary systems of prototypical size, scale, materials), and a Tritium Breeding and Extraction Facility (with unit cell mockups exposed to fission neutrons for testing coolant loops coupled to ex situ tritium processing and chemistry systems).

10. Enabling technology

In addition to the specific topics discussed above, FSNF enabling technology issues must also be addressed, such as cooling of the TF-coil joints, fusion power core components maintenance, continuous operation of heating and current drive systems, and diagnostics for the harsh neutron environment.

11. Conclusion

An accelerated fusion energy development program, a "fast-track" approach, requires proceeding with a nuclear and materials testing program in parallel with research on burning plasmas, ITER. A FNSF would address many of the key issues that need to be addressed prior to DEMO, including breeding tritium and completing the fuel cycle, qualifying nuclear materials for high fluence, developing suitable materials for the plasma-boundary interface, and demonstrating power extraction. The Advanced Tokamak is a strong candidate for an FNSF as a consequence of its mature physics base, capability to address the key issues, and the direct relevance to an attractive target power plant.

Near term research focus on specific tasks can enable starting FNSF engineering design within the next five years. Extension of long pulse high performance operation and core-boundary integration should be addressed by the superconducting tokamaks in Asia in the next few years. Most significant risk reduction for the divertor/FW is provided by better models: for understanding, optimization, and extrapolation of experimental results, e.g. on detached divertor operation and mitigation of divertor erosion by local gas injection. Additionally, an aggressive research program on fusion materials and systems with single purpose facilities is needed to improve the scientific understanding of multiple effects in FW/blanket and tritium systems.

FNSF would be the facility for addressing critical issues for fusion in realistic fusion conditions, and would enable confident design of DEMO following the achievement of $Q \sim 10$ in ITER. With the level of readiness described in this paper and a focused effort on research described above, construction of FNSF could begin within the next 10 years.

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