FUSION NUCLEAR SCIENCE FACILITY (FNSF) BEFORE UPGRADE TO COMPONENT TEST FACILITY (CTF)

Y.K.M. Peng 1), J.M. Canik 1), S.J. Diem 1), S.L. Milora 1), J.M. Park 1), A.C. Sontag 1), P.J. Fogarty 2), A. Lumsdaine 1), M. Murakami 1), T.W. Burgess 1), M.J. Cole 1), Y. Katoh 1), K. Korsah1), B.D. Patton 1), J.C. Wagner 1), G.L. Yoder 1), R. Stambaugh 3), G. Staebler 3), M. Kotschenreuther 4), P. Valanju 4), S. Mahajan 4), M. Sawan 5)

1) Oak Ridge National Laboratory, Oak Ridge, TN, USA, pengym@ornl.gov
2) Innovative Design Inc., Oak Ridge, TN, USA
3) General Atomics, La Jolla, CA, USA
4) University of Texas, Austin, TX, USA
5) University of Wisconsin, Madison, WI, USA

The compact (R_0~1.2-1.3m) Fusion Nuclear Science Facility (FNSF) is aimed at providing a fully integrated, continuously driven fusion nuclear environment of copious fusion neutrons. This facility would be used to test, discover, and understand the complex challenges of fusion plasma material interactions, nuclear material interactions, tritium fuel management, and power extraction. Such a facility properly designed would provide, initially at the JET-level plasma pressure (~30%T^2) and conditions (e.g., Hot-Ion H-Mode, Q<1)), an outboard fusion neutron flux of 0.25 MW/m^2 while requiring a fusion power of ~19 MW. If and when this research is successful, its performance can be extended to 1 MW/m^2 and ~76 MW by reaching for twice the JET plasma pressure and Q. High-safety factor q and moderate-β plasmas are used to minimize or eliminate plasma-induced disruptions, to deliver reliably a neutron fluence of 1 MW-yr/m^2 and a duty factor of 10% presently anticipated for the FNS research. Success of this research will depend on achieving time-efficient installation and replacement of all internal components using remote handling (RH). This in turn requires modular designs for the internal components, including the single-turn toroidal field coil center-post. These device goals would further dictate placement of support structures and vacuum weld seals behind the internal and shielding components. If these goals could be achieved, the FNSF would further provide a ready upgrade path to the Component Test Facility (CTF), which would aim to test, for ≤6 MW-yr/m^2 and 30% duty cycle, the demanding fusion nuclear engineering and technologies for DEMO. This FNSF-CTF would thereby complement the ITER Program, and support and help mitigate the risks of an aggressive world fusion DEMO R&D Program. The key physics and technology research needed in the next decade to manage the potential risks of this FNSF are identified.

I. INTRODUCTION

A U.S. fusion community-based workshop on Research Needs for Magnetic Fusion Energy Sciences (ReNeW)[1] was conducted recently to assess the key scientific and technical foci of magnetic fusion research aimed at bridging the remaining knowledge gaps to practical fusion energy. Complementary to the research topical areas of the burning plasma and plasma boundary championed by the ITER Program and the advanced performance plasmas needed for DEMO, are the areas of *i*) plasma material interactions (PMI); *ii*) plasma facing components (PFC); *iii*) fusion power extraction; *iv*) tritium sustainability; *v*) radiation effects on materials; and *vi*) integrated design and modeling accounting for safety and environment, and reliability, availability, maintainability, inspectability (RAMI).

The needed research categories identified for these latter areas combined are extensive, encompassing the following:

i) Microstructure properties of material and material combinations under simulated plasma and neutron irradiations;
ii) Macroscopic properties of mock-ups and test loops of increased complexity involving multiple time and size scales;
iii) Interactive properties of partially integrated components using test stands of increased approximation toward the fusion nuclear environment; and
iv) Multiscale synergistic properties in the internal test components to be discovered and investigated in the full fusion nuclear environment.

The recently introduced[2,3] compact (R_0=1.2-1.3m) Fusion Nuclear Science Facility (FNSF) concept with

A=1.5-1.7, Q=0.5-3.0, $P_{DT}\leq75$MW, and $W_L\leq1$MW/m^2, was envisioned to provide such a full fusion nuclear environment. This FNSF was introduced as Stage-1 for "*fusion break-in & scientific exploration*" in a "*High-Volume Plasma-Based Neutron Source*", which was introduced earlier[4] for fusion blanket development. This volume neutron source (VNS) was subsequently renamed Component Test Facility (CTF).[5,6] The mission of CTF, relative to FNSF, therefore remains to provide the more demanding fusion environment of increased neutron fluence and duty factor required to address the Stage 2 "*engineering feasibility and performance verification*" and Stage 3 "*component engineering development and reliability growth*" research assigned to the VNS (Ref. 4).

In this paper we refer in some detail to a recent advance in understanding how neutron radiation damages in materials[7] can couple strongly with nonnuclear plasma material interactions in otherwise disparate physical phenomena that are collocated in space, coincidental in time, and overlap in activation energies (Section II). A working description of the FNSF research mission is presented that is consistent with such synergistic multiscale interacting phenomena[8,9] in a fusion nuclear environment (Section III). Achieving this research mission during the ITER era, including verifying readiness to upgrade to the CTF, would support and help mitigate the risks of an aggressive DEMO R&D program as defined in the ITER Broader Approach[10]. Examples of the FNSF goals in performance, configuration, and remote handling capabilities can be derived from the mission and will be described at the conceptual level (Section IV). These in turn help identify the R&D required to achieve these goals and manage the risks of FNSF (Section V), accounting for the strong commonality of the scientific and technical basis of the normal aspect ratio tokamak and the spherical tokamak. A wider range of FNSF research goals of relevance to the realization of fusion energy will be discussed in Section VI. There it will be clarified that the integrated FNSF R&D capabilities in small and normal aspect ratio, jointly with the supporting R&D programs, will likely pace the progress of FNS research toward the first fusion DEMO.

II. CONDITIONS FOR SYNERGISTIC COUPLING PHENOMENA INVOLVING PMI AND NEUTRON-MATERIAL INTERACTIONS (NMI)

It is nearly impossible to predict with confidence new physics phenomena that would result from hitherto new coincidence and juxtaposition of otherwise disparate physics mechanisms of similar activation energies. However, experience tells us that new phenomena have been encountered in such situations, which in turn strongly shaped the subsequent R&D. Examples include the discovery of H-mode plasmas when the poloidal divertor configuration was introduced to the tokamak, and the development of radiation resistant ferritic steel following the discovery of severe nuclear damages of carbon steel used in early fission critical testing facilities. It is nevertheless appropriate to search for clues for potential synergies involving such as NMI and PMI in a fusion nuclear environment, and use such possibilities to shape the FNSF mission.

It has been broadly anticipated that fusion neutron dose levels up to 10 dpa, introducing in ferritic steel ~100 appm He, can be reached in appropriately designed internal components before substantial deleterious modifications to the material properties would occur.[9] Such radiation-induced changes to the material microstructure, for temperatures above the ductile to brittle transition temperatures (DBTT), include precipitation and solute segregation, permanent deformation under applied stress, volumetric swelling, and high-temperature He-induced embrittlement, etc. However, as can be seen in the following case, it is likely that PMI and NMI can have substantial synergistic interactions, even for damages substantially below 1 dpa.

That the conditions are present for strong synergistic coupling is suggested by several recent advances in NMI in such as the bcc iron regarding the formation and dynamics of microstructure damages by neutrons, particularly in the case of clusters of defects:

1) A fusion neutron damages material through displacement cascades involving >10^3 atoms over a region of ~10^2 nm in size following a collision of the "primary knock-on atom";[9]
2) In addition to nearly isolated point defects, both interstitial and vacancy clusters of 20 or more point defects, with stored energy in the range of ~eV, are formed during the displacement cascades;[11]
3) Such clusters can form planar dislocation loops of sizes up to 20 nm, and undergo 1D motion with fairly low activation barriers (~0.1eV); they were observed via TEM to move at an unexpectedly high rate of up to 5 nm/s at ~300 °C temperature, likely through interactions with interstitial impurity atoms.[7]

It is therefore of interest to determine whether such clusters of defects, when and if born within ~10^2 nm of the plasma facing surface, would migrate to within ~20 nm of the surface and affect those PMI properties that have activation energies of ~eV. A fusion nuclear environment is therefore required to test and understand the properties of this nuclear-nonnuclear coupling effect, and determine whether a similar process occurs in such materials as tungsten at temperatures of ~10^3 °C and the reduced activation ferritic steels at ~500 °C. Both of these materials are of interest to DEMO.

Additional examples of such coupling in the internal components include: degradation of thermal conductivity and changes in surface morphology and physical and chemical erosion rates in graphite at as low as ~0.1 dpa; (Ref. 12) changes in tritium retention and permeation as neutron induced defects accumulate through <1 dpa; (Ref. 13) and property changes of tritium permeation barrier in nuclear environment.[14] In the latter example, substantial reductions of the permeation reduction factor (PRF) were measured at low irradiation doses in a research fission reactor. Tested in this case was the ceramic coating of Cr_2O_3–SiO_2 including $CrPO_4$ on the inner surface of a ferritic steel F82H container of liquid lithium lead eutectics, $Pb_{17}Li$.

Solutions for internal components therefore need to be tested in a fusion environment to investigate synergistic coupling phenomena before confidence can be established for more rigorous engineering and technology testing in a CTF and for use in DEMO.

III. FNSF MISSION AND MISSION COMPONENTS

The FNSF mission is therefore *to provide an integrated, continuously fusion nuclear environment of copious neutrons that can be used to test, discover, and understand the multiscale synergistic coupling phenomena involving fusion plasma material interactions, tritium fuel management, and power extraction, accounting for the nuclear effects on materials*. The interactions range in scale from picoseconds to years, from nm to meters, and involve up to four states of matter. Improvements to the internal components based on the new understanding so obtained would be developed and further tested, until adequate scientific and technical basis for DEMO-capable components are established.

The FNSF mission complements the ITER mission,[15] which is to demonstrate the scientific and technological feasibility of fusion energy for peaceful purposes. Its inductive operation is expected to produce significant fusion power (~500MW) through the D–T reaction with high fusion gain Q ~ 10 for 300–500 s, and Hybrid operation to produce ~400 MW with Q~5 for 3000 s. Fig. 1 indicates this complementarity, indicating that FNSF could utilize low Q (~1) conditions to achieve its mission for a fluence of 1 MW-yr/m^2, pulse lengths of up to 2 weeks (~10^6 s), and a testing duty factor up to 10%. Also indicated is the upgrade to a CTF requiring increased fluence and duty factor.[4]

Accomplishing the FNSF mission will therefore reduce the risks for DEMO R&D and accelerate it.

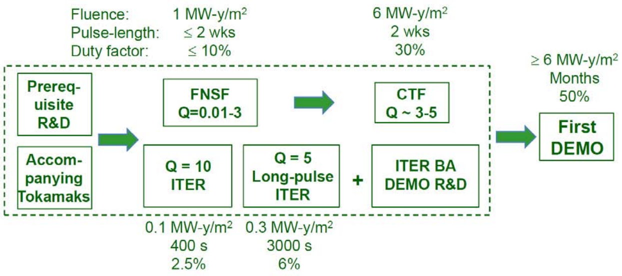


FIG. 1. Mission complementarity between FNSF-CTF and ITER & ITER-BA to develop basis for DEMO.

Efficient upgradability to the CTF will further require that the following conditions and capabilities be provided:

1) For an adequate fusion environment – fusion neutron flux $W_L \leq 1$ MW/m^2 at the outboard mid-plane.
2) To address long time constant interactive phenomena – plasma duration ~10^6 s.
3) For reliable plasma operation to limit or eliminate plasma induced disruptions and transients – large margins to MHD stability limits assuming such as $\beta_N \leq 0.75\, \beta_N^{\text{no-wall}}$; $q_{cyl} \geq 4$.
4) For continuous plasma operation – I_p maintained non-inductively, likely using co-NBI and RF heating and current drive in the presence of substantial bootstrap current.
5) To maintain plasma T and n profiles and purity for ~10^6 s – divertors capable of handling peak heat flux ≤ 10 MW/m^2 and control the plasma density continuously using fueling, pumping, heating and current drive.
6) To enable time-efficient cycles in which to test, discover, understand and innovate solutions for internal components – modular internal components in configurations that allow RH, and extensive RH capabilities to install and replace these components.
7) To prepare, maintain, and repair activated internal components – extensive hot-cell capabilities based on RH.
8) To carry out post-mortem research and investigations of nearly failed components to discover and understand new phenomena – extensive diagnostic, analysis, and component manipulation capabilities in a hot-cell laboratory to examine conditions of critical areas in the components.

To deliver these FNSF operational capabilities, accompanying R&D programs will be required to develop the basis for the plasma operation, design the appropriate test modules for all the internal components, and manage the risks inherent in such a new fusion research facility. These will include R&D to develop, including optional and innovative approaches when appropriate:

1) Adequate database, predictability, actuators, and modular RH capabilities to control the plasma dynamics, including measurements, heating, fueling, current drive, and stability control.
2) Divertor modules, mid-plane test blanket modules, and off-mid-plane tritium breeding modules, including optional and innovative designs for these modules.
3) RH systems and tools, hot cell systems, tools, and post-mortem research capabilities.

These research capabilities should apply equally to the low aspect ratio and the normal aspect ratio configurations and are compatible with component designs to be recommended by the accompanying R&D programs. In the case of the low aspect ratio spherical torus,[2,3] which requires the use of a single-turn normal conducting, water-cooled toroidal field coil center post with limited nuclear shielding and no central solenoid. This center post is an internal component to be determined by an accompanying R&D program. Solenoid-less startup and ramp up of the plasma current will also be required in this case. These may further influence the choices of the internal component designs. The aspect ratio choices for FNSF will require comparative assessments of performance, cost and risks.

IV. FNSF PERFORMANCE, CONFIGURATION, AND REMOTE HANDLING DESIGN CONCEPTS

The systems code of Refs. 2 and 3 was applied to update working examples of the FNSF goals in performance, configuration and operational capabilities, as driven by the FNSF mission and mission components described in Section III. The results are summarized in Figs. 2a & 2b. The internal component modules similar to earlier discussions[2,3] include, from device center, the toroidal field coil center post with He-cooled tungsten first wall; the top and bottom toroidally complete divertor modules with extended channels with strong pumping, and normal conducting divertor coils; out board mid-plane full function test blanket modules; outboard off mid-plane tritium breeding modules, and outboard poloidal field coils with large access that could be superconducting. The NBI access, RF, and diagnostic modules at the mid-plane would replace the mid-plane test blanket module as required, using identical access ports (12 in the present conceptions). NBI brings in substantial beam-plasma fusion, up to ~40% in fusion power for Q~1. All modules are enclosed by additional shielding modules. The vacuum boundary, the toroidal field coil current return path, and the support structures for the internal modules are integrated into a single system with removable lids to enable RH access to all the internal modules, while the mid-plane modules have radial access.

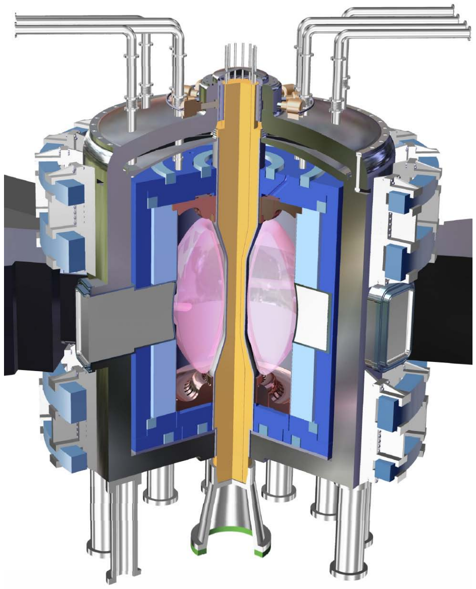


FIG. 2a. Updated FNSF configuration with A=1.7, R_0 = 1.3 m, H-mode H factor ≤ 1.25, and $J_{TF\text{-}avg} \leq 4$ kA/cm^2.

Stage-Fuel	I-DD	II-DT	III-DT	IV-DT
Current, I_p (MA)	4.2	4.2	6.7	8.4
Plasma pressure (MPa)	0.16	0.16	0.43	0.70
W_L (MW/m^2)	0.005	0.25	1.0	2.0
Fusion gain Q	0.01	0.86	1.7	2.5
Fusion power (MW)	0.2	19	76	152
Tritium burn rate (g/yr)	0	≤105	≤420	≤840
Field, B_T (T)	2.7	2.7	2.9	3.6
Safety factor, q_{cyl}	6.0	6.0	4.1	4.1
Toroidal beta, β_T (%)	4.4	4.4	10.1	10.8
Normal beta, β_N	2.1	2.1	3.3	3.5
Avg density, n_e (10^{20}/m^3)	0.54	0.54	1.1	1.5
Avg ion T_i (keV)	7.7	7.6	10.2	11.8
Avg electron T_e (keV)	4.2	4.3	5.7	7.2
BS current fraction	0.45	0.47	0.50	0.53
NBI H&CD power (MW)	26	22	44	61
NBI energy to core (kV)	120	120	235	330

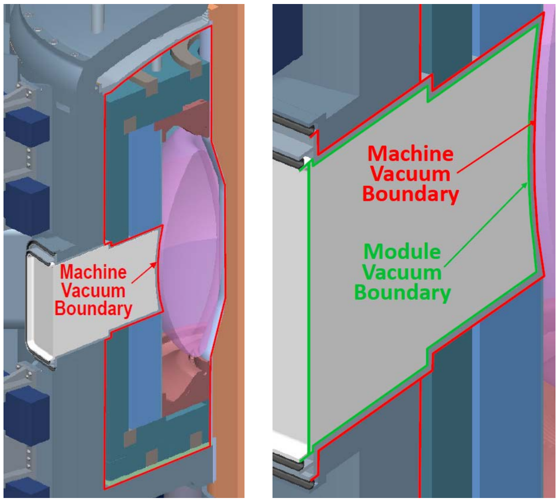
FIG. 2b. Example FNSF parameters for Stage-I) D-only operation at I_p = 4.2 MA, II) D-T operation at the same current (W_L = 0.25 MW/m^2), III) D-T operation at 6.7 MA (1 MW/m^2), and IV) at 8.4 MA (2 MW/m^2).

Note further that, relative to Refs. 2 and 3, a reduced average current density $J_{TF\text{-}avg}$ up to 4 kA/cm^2 over the narrow part of the center post, a decreased $\beta_N / \beta_N^{\text{no-wall}}$ to 0.75, an increased q_{cyl} to ≥4, and a reduced H-mode confinement-factor to ≤1.25 (in Hot Ion H-Mode[16]) have increased A and R_0 to 1.6 and 1.3m, respectively.

Ready upgradability of all internal components would enable effective staging of the FNSF plasma and neutron flux performance goals, with plasma durations up to 2 weeks per pulse.

1) Stage-I (DD): to commission PFC-divertor capabilities, continuous control of plasma dynamics in steady state, neutron transport, shielding and safety integrity, RH operation, etc., at low levels of activation that is equivalent to JET DD plasma operation.
2) Stage-II (DT): to test and verify predictability of tritium breeding, power extraction, full RH operations, etc. Note that, at a duty factor of 10% with pulse lengths up to 2 weeks, the tritium burn rate would be ~100 g/yr during this stage and ease tritium supply while the basis for tritium breeding and recovery are tested toward a TBR goal of ~100%. This aims to surpass the ITER TBM testing goals, which are limited to burning plasma pulses up to 3000 s and a duty factor of less than 3%.
3) Stage-III (DT): to test and achieve the full fusion nuclear science research capabilities, following upgrade of NBI to negative ion source systems, etc.
4) Stage-IV (DT): to stretch the FNSF goals to $W_L = 2$ MW/m^2, if Stage-III turns out to be successful.

The goals of ready upgradability to enable effective staging, and a testing duty factor of 10%, will also require effective approaches to place vacuum weld seals behind the internal components and shielding to allow repeated cutting and welding. Design concepts of such vacuum seal configurations and the associated vacuum boundaries for the mid-plane test modules are developed (Figs. 3). Remote access to these weld seals becomes feasible, after plasma shutdown and radiation cool down in the FNSF experimental hall to allow hands-on draining and disconnection of the service lines to the test module.



(i) (ii)

FIG. 3. Vacuum boundary and weld seals concepts for (i) FNSF chamber, and (ii) mid-plane test module.

Using such weld seals, the assembly and disassembly procedures for the mid-plain test modules via RH can be described. A maintenance cask[12] with radiation shielding, vacuum capability to isolate dusts and gases, tools to cut and re-weld seals and to move the test modules can be designed to execute the following:

1) Mate cask to the test module larger seal and evacuate,
2) Cut the smaller vacuum seal of the test module while maintaining vacuum of the torus vessel,
3) Withdraw the test module into the cask,
4) Move a prepared vacuum shield plug in place of the test module and reseal the smaller weld,
5) Move a prepared vacuum door in place and establish independent vacuum of cask, now containing an activated and / or contaminated test module,
6) Transport the test module into a RH laboratory for maintenance, repair, diagnosis and analysis, and

If desired, install a replacement test module using another transport cask in the reversed order. Hands-on maintenance can then be applied to reconnect services to the installed test module before preparation for the experimental testing research operation.

The RH of the toroidal field coil (TFC) center post will be more demanding. As shown in Fig. 4, a new sliding joint concept is developed to allow simultaneous smooth motion in the vertical and the toroidal directions while maintaining an adequate electrical contact. It is seen that the total area of contact, and hence the contact current density, can be adjusted by the height of this "bi-directional" sliding joint. The pressure of the contact can be adjusted by the stiffness of the curved conductors. The flexibility and strength of the contact copper can be ensured by placing the sliding joints behind adequate shielding to prevent neutron irradiation. A strong top dome containing the TFC current path can be designed to use actuators to ensure adequate concentricity of the center leg during operation. Mechanical bolts can be used for the removable joint at the outer rim shown in Fig. 4.

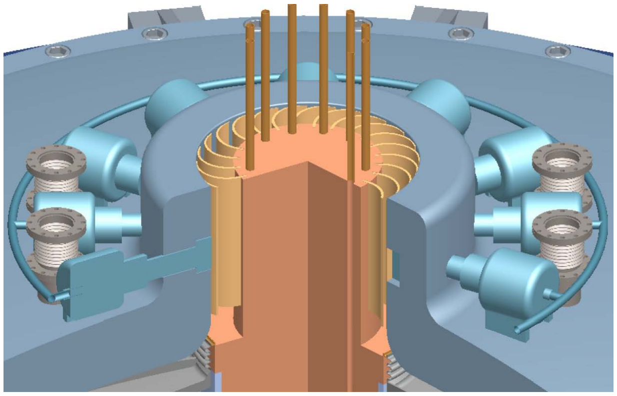


FIG. 4. Concepts for bi-directional (vertical and toroidal) sliding joint at the top of the TF coil center post.

A new design concept for a top vacuum dome lid with shielded vacuum weld seals is also required for this purpose and is developed (Fig. 5). There it is shown after the TFC dome has been removed with crane lift, which in turn follows hands-on operation to remove the TFC bolts (see, Fig. 4) and to compress the sliding joint fins to a smaller radius. This in turn allows access to the vacuum lid and its two weld seals for manual cutting and removal.

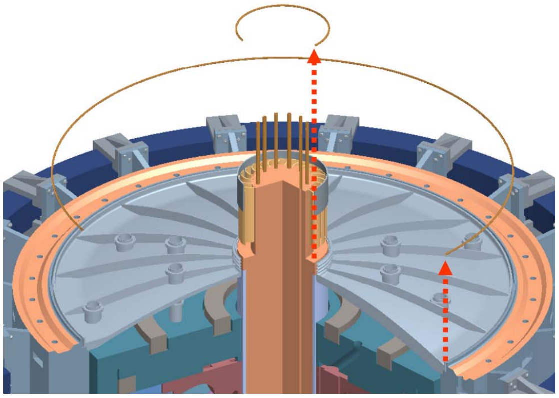
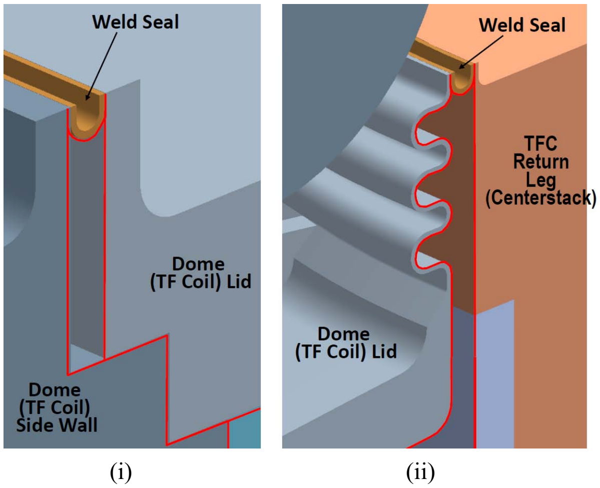


FIG. 5. Concepts for removable vacuum weld seals before vertical removal of the top vacuum dome and then the TF coil center post.

The design concept for the vacuum boundary and weld seals are also developed and shown in Fig. 6. Near the center post, a vacuum bellow is needed to allow relative motion between the center stack and the vacuum lid. The vacuum boundary extends the length of the TFC center stack. A large radius weld seal is used to connect the vacuum boundary defined by the TFC current return path. Automatic welding and cutting tools will be used for this purpose.



(i) (ii)

FIG. 6. Vacuum boundary and weld seals concepts for the dome lid joining (i) the outer TFC current path, and (ii) the TFC center post.

The foregoing design features will most likely be needed to realize full remote handling in FNSF. In Fig.7 are shown the steps for full disassembly and disassembly of FNSF. Only linear motion is allowed for all internal modules and the auxiliary systems (NBI, RF, fueling and diagnostics). The service lines to these components are also linear in the radial or vertical direction, where hands-on connection and disconnection can be permitted outside of the shielding shown in Fig. 2a, after plasma shutdown and experimental hall cool down. Remote handling casks and remote handling hot cell laboratories[3] will also be needed to manage the removed components.

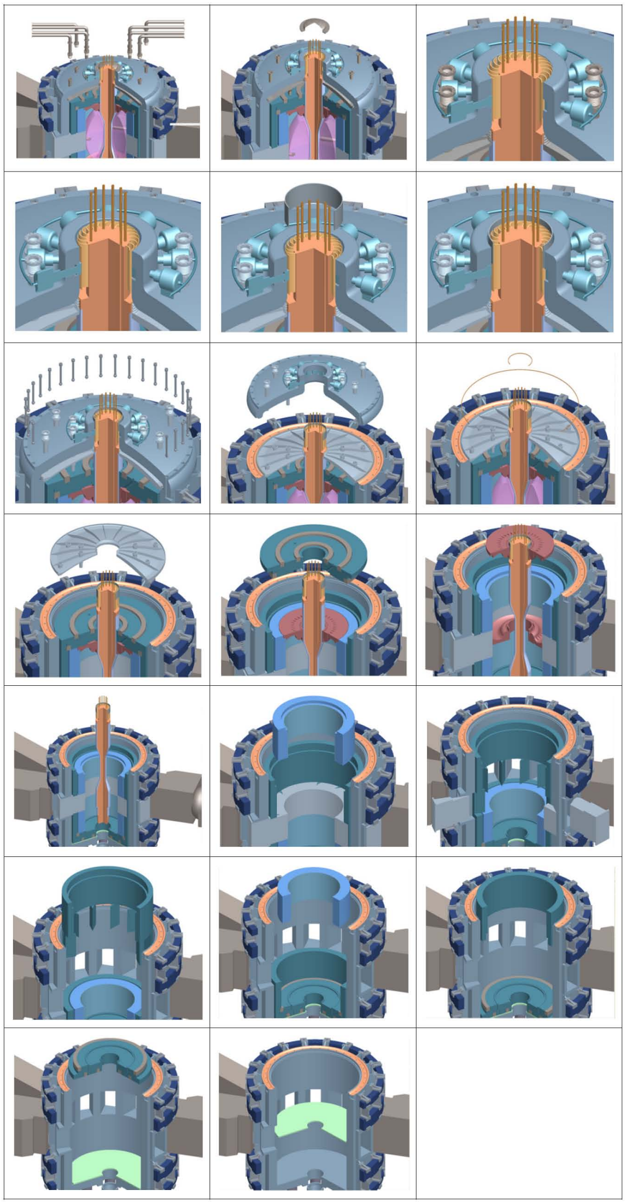


FIG. 7. Steps (left to right starting from top) in disassembly of the internal activated test modules for the FNSF based on linear motion of the modules.

V. PREREQUISITE R&D

The FNSF mission and its goals in performance, configuration, and remote handling design in the preceding two sections also serve to identify the prerequisite R&D to establish the basis for FNSF design, and reduce the risks in achieving its goals. It is assumed that an accompanying fusion nuclear science R&D programs (FNSP) would be in place to determine the designs and options of the internal component test modules, including their material choices. The following R&D topics are provided as guide to further development of the FNSF concept and the FNSP:

1) Plasma gun Helicity Injection[17] and Electron Bernstein Wave start up[18] research to establish the basis and predictive modeling capabilities.
2) Predictive modeling capability, using such as the SWIM (Ref. 19) and the GLF23 (Ref. 20) codes, to estimate steady-state operation conditions, assuming the Hot-Ion H-Mode[16] operational scenario based on dominating NBI heating and current drive.
3) Predictive modeling capability, using such as the SOLPS-EIRENE codes,[21] to estimate appropriate divertor designs and operational scenarios, using such options as the extended divertor channels, to limit the peak divertor heat fluxes to below 10 MW/m^2 for neutron W_L up to 1 MW/m^2.
4) Assess the engineering requirements for the RH systems capabilities to reduce the mean-time to replace (MTTR) all internal component modules adequately to achieve a duty factor of 10%.
5) Assess the engineering requirements[22] for the TF coil center post to achieve reliable operation, and determine the R&D needed for its fabrication.
6) Assess the engineering requirements for the low voltage, high current dc power supply with relatively stiff control of current, and determine the R&D needed for its fabrication.
7) Assess the hot-cell and RH capabilities required to support the fusion nuclear science research onsite in concert with the accompanying FNSP offsite, jointly to manage the risks of the overall program.

These prerequisite R&D can be defined when conceptual designs of sufficient detail for the FNSF become available.

VI. FNSF-AT, FNST-STRETCH, AND PACING

The FNSF mission, performance, configuration, and remote handling goals can be implemented similarly for the case of normal A. In this case, an FNSF-AT concept has been identified[23,24] with increased H-factor, bootstrap current fraction, and β_N (to just beyond the no-wall limit), providing W_L =1-2 MW/m^2 at moderate Q. These cases are therefore equivalent to the Stages III-DT and IV-DT cases shown in Fig. 2b. These R&D stages on FNSF, with either aspect ratio, are placed in Fig. 8 in the space of fusion gain Q and fusion neutron flux. DD superconducting tokamaks, JET DT (Ref. 16), ITER (Ref. 15), and DEMO are included for comparison. It is seen that substantially smaller Q values than ITER characterize the FNSF mission space with the potential capability to reach W_L values closer to DEMO.

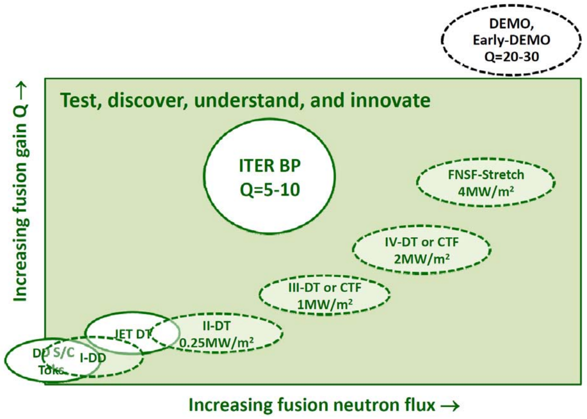


FIG. 8. FNSF performance goals from Fig. 2 in comparison with superconducting tokamaks, JET-DT, ITER, and DEMO in the space of neutron flux and fusion gain.

Large margins to plasma and engineering limits are included in these FNSF device concepts (see, fig. 2b). However, their hardware capabilities could be more fully utilized to support research at even more advanced physics performance and higher neutron fluxes[25], such as $Q > 7$ and $W_L \sim 4$ MW/m^2. This case is included in Fig. 8 as "FNSF-Stretch", which can be realized only if the preceding stages of research have been successful.

Each of the operating conditions included in Fig. 8 will require simultaneous success of all internal components to handle the fusion neutron and plasma fluxes, and to control the plasma dynamics to deliver the required fusion neutrons. This would imply that the AT-level fusion plasma dynamics and control, and fusion nuclear sciences properties should continue to be pursued hand-in-hand in FNSF. Nevertheless, the fusion nuclear science studies can begin at the JET level conditions, which do not require advances in tokamak or spherical tokamak physics beyond the no-wall beta limit (Fig. 2b).

These considerations suggest that the FNSF capabilities in handling continuous plasma and neutron fluxes, controlling the plasma dynamics to produce these fluxes, and recovering operational capabilities with a minimized down time, jointly with the associated FNSP, will likely pace the progress of research in fusion nuclear science and engineering toward the realization of DEMO.

The FNSF research mission and performance capabilities described here depict a first-of-kind research users' facility for fusion nuclear science and engineering. In this facility will be enabled the R&D on multi-scale nuclear-nonnuclear coupling, anticipated in a fusion nuclear environment. The research will address this coupling effect in timescales from picoseconds to years and spatial scales from nm to meters, involving up to four states of matter (plasma, gas, liquid, solid).

That such a research facility will most likely be necessary can be surmised from the extensive experience in R&D leading up to the first commercial demonstration in 1958 of fission electrical power of ~60MWe at Shippingport, PA, U.S.A. (Ref. 26) Much of the fission nuclear science, material, and engineering data for nuclear fission power were obtained at the Idaho Reactor Testing Station, where 49 fission test facilities were built and operated before 1960 (Ref. 27). Several times more fission test facilities were built and operated during the same period within the U.S. National Defense Programs.[28]

Nuclear-nonnuclear coupling interactions in the nuclear science and engineering of a full fusion nuclear environment is anticipated to be very broad in scientific scope. It is therefore appropriate to suggest that the first FNSF, if built and operated, will only begin to discover new scientific opportunities to pursue further.

ACKNOWLEDGMENTS

This work is supported by U.S. DEPARTMENT OF ENERGY under Contract No. DE-AC05-84OR21400.

REFERENCES

1. http://burningplasma.org/web/ReNeW/ReNeW.report.press1.pdf.
2. Y.K.M. PENG, et al., paper FT/P3-14, IAEA Fusion Energy Conference, 2008.
3. Y.K.M. PENG, Fusion Technol. **56** (2009) 957.
4. M. ABDOU, Fusion Technol. **29** (1996) 1.
5. G.M. VOSS, et al., Fusion Engineering Design **83** (2008) 1648.
6. Y.K.M. PENG, et al., Plasma Phys. Control. Fusion **47** (2005) B263.
7. K. ARAKAWA, et al., Science **318** (2007) 956.
8. B.D. WIRTH, et al., J. Nuclear Mat. **329-333** (2004) 103.
9. S.J. ZINKLE, Phys. Plasmas **12** (2005) 058101.
10. T. TSUNEMATSU, Fusion Engineering Design **84** (2009) 122.
11. D.J. BACON, et al., J. Nuclear Mat. **323** (2003) 152.
12. B.I. KHIRPUNOV, et al., J. Nuclear Mat. **390-391** (2009) 921.
13. D.G. WHYTE, J. Nuclear Mat. **390-391** (2009) 911.
14. M. NAKAMICHI, et al., Fusion Engineering Design **82** (2007) 2246.
15. M. SHIMADA, et al., Nucl. Fusion **47** (2007) S1.
16. JET TEAM (by M. KEILHACKER), Plasma Phys. Controlled Fusion **39** (1997) B1.
17. D.J. BATTAGLIA, et al., Phys. Rev. Lett. **102** (2009) 225003.
18. V. SHEVCHENKO, et al*., RF-17, 7 -9 May 2007, Clearwater, Florida, USA.*
19. D.B. BATCHELOR, et al., J. Phys. **180** (2009) 012054.
20. L.E. KINSEY, et al., Phys. Plasmas **9** (2002) 1676.
21. R. SCHNEIDER, et al., Contrib. Plasma Phys. **46** (2006) 3.
22. A. LUMSDAINE, et al., P4-127, 26th Symp. Fusion Technology, 2010.
23. V.S. CHAN, et al., Fusion Science Technology **57** (2010) 66.
24. A.M. GAROFALO, et al., IEEE Transaction Plasma Science **38** (2010) 461.
25. R.D. STAMBAUGH, et al., P2-110, 37th EPS Conf. Plasma Phys. Controlled Fusion.
26. Operating History of U.S. Nuclear Power Reactors, WASH-1203, US AEC, 1972.
27. https://inlportal.inl.gov/portal/server.pt?open=514&objID=4182&mode=2&featurestory=DA_143293.
28. H. FORSEN, *private communications, October 2010.*