Princeton Plasma Physics Laboratory

PPPL-5123

PPPL ST-FNSF Engineering Design Details

T. Brown, J. Menard, L. El Gueblay, A. Davis

July 2015

Prepared for the U.S. Department of Energy under Contract DE-AC02-09CH11466.

PPPL ST-FNSF ENGINEERING DESIGN DETAILS

T. Brown[1], J. Menard[1], L. El Gueblay[2], A. Davis[2]

[a] *Princeton Plasma Physics Laboratory, Princeton, NJ, USA*
[b] *University of Wisconsin, Madison, WI, USA*
Author email: tbrown@pppl.gov

One of the goals of the PPPL Spherical Tokamak (ST) Fusion Nuclear Science Facility (FNSF) study was to generate a self-consistent conceptual design of an ST-FNSF device with sufficient physics and engineering details to evaluate the advantages and disadvantages of different designs and to assess various ST-FNSF missions. This included striving to achieve tritium self-sufficiency; the ability to provide shielding protection of vital components and to develop maintenance strategies that could be used to maintain the in-vessel components (divertors, breeding blankets, shield modules and services) and characterize design upgrade potentials to expanded mission evolutions.

With the conceptual design of a 2.2 m ST pilot plant design already completed emphasis was placed on evaluating a range of ST machine sizes looking at a major radius of 1m and a mid-range device size between 1 m and 2.2 m.

This paper will present an engineering summary of the design details developed from this study, expanding on earlier progress reports presented at earlier conferences that focused on a mid-size 1.7 m device. Further development has been made by physics in defining a Super-X divertor arrangement that provides an expanded divertor surface area and places all PF coils outside the TF coil inner bore, in regions that improve the device maintenance characteristics. Physics, engineering design and neutronics analysis for both the 1.7 m and 1 m device have been enhanced. The engineering results of the PPPL ST-FNSF study will be presented along with comments on possible future directions.

I. INTRODUCTION

A number of roadmaps have been prescribed that lead to a fusion power plant from ITER. Some Countries suggest to move from ITER by constructing a prototypical demonstration device (DEMO) that precedes a power plant; other institutions define a smaller scale "Pilot Plant" that generates net electricity $Q_{eng} \geq 1$ as quickly as possible before building DEMO and some suggest that prior to building a DEMO device, it would be best to first operate a smaller Fusion Nuclear Science Facility (FNSF) to develop the blanket technology used for thermal power conversion and tritium breeding along with developing the myriad of H&CD and auxiliary systems. The primary objective of the FNSF is to provide a fusion-relevant neutron wall loading (1MW/m^2) and neutron fluence of 6MW-yr/m^2 to develop and test fusion blankets. Broader mission requirements for FNSF will impact the selection process and design options, but the goal of this study was to obtain a better understanding of the copper ST option in sizing a device to achieve a tritium breeding ratio TBR $\geq$ 1 and to understand the opportunities offered by a smaller (TBR < 1) device. This paper reviews the engineering details in developing the Spherical Tokamak (ST) approach for FNSF by balancing physics requirements and engineering constraints within a developed configuration arrangement that is amenable to in-vessel component maintenance. It will finalize the conceptual design activities at PPPL looking at copper ST-FNSF devices reported on in earlier papers.[1, 2]

II. PHYSICS DESIGN REQUIREMENTS

As reported in earlier PPPL ST-FNSF studies, physics requirements set engineering constraints for the device with respect to plasma parameters, plasma shaping (elongation and triangularity), divertor geometry and PF arrangement. Soft constraints were set that allowed design revisions to be made as needed to meet engineering requirements governed by stress limitations, component sizing and a planned in-vessel maintenance approach. The basic arrangement shown in past studies are retained with PF coils located outside the vacuum vessel and inside a copper jointed TF coil system, with the exception of a pair of PF coils (top & bottom) that are located within the confines of a single turn, hour-glass shaped TF center post. Figure 1 shows the arrangement details with TF coils, outboard PF coils, vacuum vessel and breeding blankets.

The PF coil set was developed to form a double null Super-X/snowflake divertor which can reduce the divertor heat load by a factor of 3 relative to a conventional divertor arrangement.[2] This Super-X/snowflake PF arrangement was used in developing both the 1.7m and 1m device configurations. Equilibrium calculations were made with a single turn TF center post defined with an hour-glass shape to allow PF coils to be placed within the interior of the coil in the upper and lower region. The

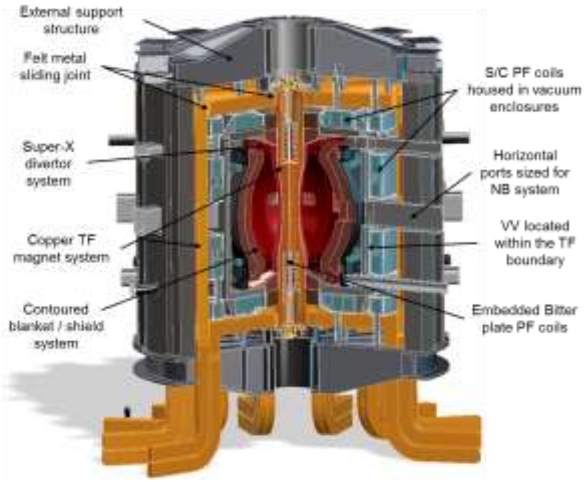

Fig.1. 1.7m ST-FNSF general arrangement

copper TF coil supplied shielding for the copper PF coils. PF coils were sized for engineering current density limits and positioned to provide spacing for interfacing systems, services and structural supports.

III. DESIGN OVERVIEW

The configuration design developed for both size options retain the same vertically-maintained, jointed copper TF magnet system design features as illustrated in the isometric view of Figure 1. The ST-FNSF designs allows the removal of the TF center post and individual sectors or the full blanket system independent of each other once the upper beam structure, TF horizontal legs and vacuum vessel lid is removed (see Figure 2).

The configuration developed for both options use TF coil legs (12 on the 1.7m device, 10 legs for the 1m device) that connect to a single turn hour-glass shaped TF center post with MIT Alcator C-mod style Felt metal sliding joints. To minimize resistive losses in the water cooled TF system, the return legs were expanded in cross-section and superconducting PF coils were positioned where sufficient shielding is present. In keeping with the ground rule of using near term manufacturing processes, a plate assembled TF center post design was incorporated. The plates of the center post incorporate vertical cooling holes that run the full length – an approach proposed in other ST neutron source concepts and analyzed in an earlier study showing acceptable thermal stress conditions.[1] A radial coolant option was analyzed as an alternate approach and was found to be more attractive relative to space and operational conditions than the straight through coolant design used in the baseline design [1], however piping and manifold design details for this approach have not been developed yet.

One challenging area was defining a workable design concept to meet the high current densities needed to

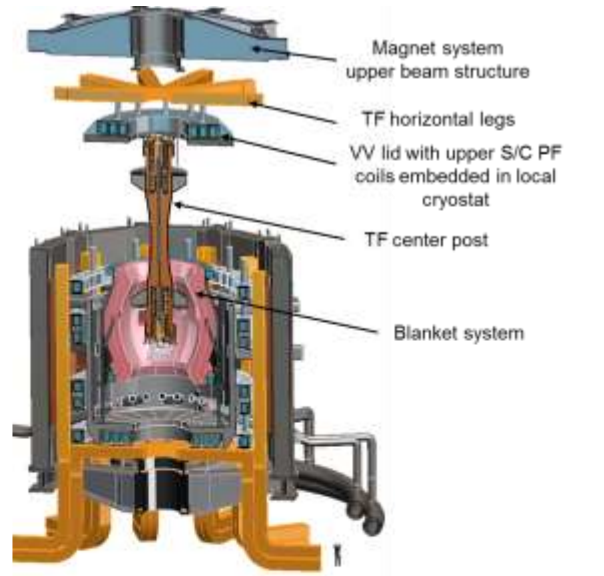

Fig.2. Vertical maintenance approach adopted for TF center post and blanket maintenance

support the results of the physics equilibrium calculations. Equilibrium results indicated that a pair of coils was needed within the TF center post having current densities in the range of 30-40 MA/m^2. This capability was beyond the limit of a normal water-cooled copper coil, but may be feasible for a Bitter plate design.

A Bitter coil design incorporating MgO insulated Glidcop^{TM} plates are enclosed in a copper alloy structure that incorporates matched drilled coolant holes at the base that interface with holes emanating from the plates of the TF center post. A separate coolant supply is provided to each PF center post assembly with a common return system. The TF center post is Glidcop^{TM}; a dispersion strengthened copper-alloy. A leak free system would be created by furnace brazing the entire center post-PF containment structure. The center post details are shown in Figure 3.

The design intent was to incorporate superconducting

Fig. 3. TF center post details

PF coils at locations outside the TF center post region to minimize power losses. The low and high field materials were chosen depending on PF current density levels and shielding conditions. Shielding calculations show the dose peaks at $6x10^9$ Gy at 6 FPY in the lower region of PF1 and $2x10^8$ Gy at the lower surface of PF3.[3,4] Identified locations can be found in Figure 4, showing PF coil and key in-vessel component details. The allowed dose on the divertor PF coils (PF1 and PF2) Cu Bitter plate MgO

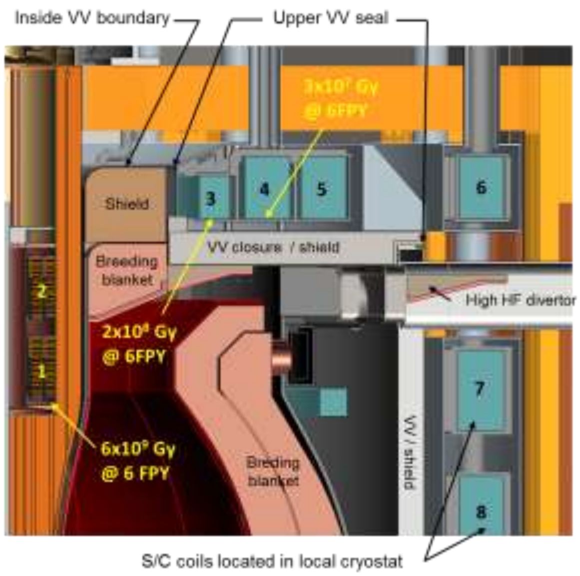

Fig. 4. View of in-vessel components

insulation located in the TF center post exceeds the applied dose value by close to two orders of magnitude (10^{11} Gy) based on best estimates.[4] Dose levels at the base of PF3 are near the limit of Cyanate ester blend insulation with some margin existing when using a pure Cyanate insulation (factor of 2). From a coil service standpoint it's best to keep PF coils (3 thru 5) as S/C coils. To further reduce the fluence on PF3 the coil can be raised vertically and the insulation thickness beneath it increased. The peak operating coil current densities for PF4 thru PF8 is 25 MA/m^2 which may exceed the limit the LTS Nb_3Sn but will easily meet HTS cable current density alowables. The possibility of increasing the coil area to reduce the current density and the ability to operate with reduced cycles may also allow the use of LTS - further investigation is needed for the final recommendation.

Figure 4 also shows general arrangement details of other in-vessel components. The upper vacuum vessel closure is developed as a double wall structure that contains tungsten carbide (WC) balls and borated water (for shielding) with an external super-structure that forms a cryogenic vacuum environment to house S/C PF coils. A vertical cylinder with internal cold-to-warm transition structure provides a structural tie to the external support structure (shown in Figure 1). A vacuum connection between the closure lid and inner vacuum member is made at the top and bottom at the seal interface, which is shown in Figure 4. A double-wall vacuum shield structure is located on the outside of a plasma-contoured blanket. Local ports shown in the upper region can be used for divertor pumping, diagnostics and maintenance access to the high heat flux divertor segments.

A key feature of the PPPL ST-FNSF design is the development of the Super-X/snowflake divertor configuration that moves the divertor strike-point region to a larger radius away from the relatively high neutron flux at the top and bottom ends of the center post. This divertor arrangement substantially reduces the projected peak divertor heat-flux by a factor of 3 relative to a conventional divertor, which is typically $\leq$ 10MW/m^2 for nominally-attached conditions for a surface-averaged neutron wall loading $\langle Wn \rangle$ = 1MW/m^2 (Ref. 2). In this proposed design, the divertor strike-points at large major radius leaves space for breeding in the center post ends, which is important for maximizing the tritium breeding ratio (TBR) in the ST configuration. Tritium breeding calculations were performed on both the 1.7 m and 1 m devices with TBR results of 0.97 and 0.88 respectively.[3, 4]

The calculated TBR of 0.88 on the smaller device substantially reduces the tritium consumption when compared to not breeding any tritium. Port openings were a contributing factor in lowering the TBR. Scaled side-by-side plan and elevation views compare the 1.7 m and 1 m ST-FNSF in Figure 5 with expected TBR values. The test blanket module, material test module and neutral beam ports are also noted on Figure 5.

Given the compact nature of the low aspect ratio ST design, basic blanket details and piping service arrangements were developed to assure that self-

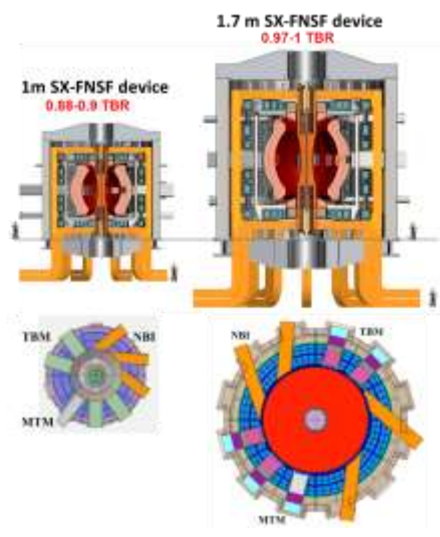

Fig. 5. Comparison of 1.7m and 1m devices

consistency could be developed in meeting general requirement and space constraints of configuration driving systems (magnets, in-vessel systems, services and support structure). Solenoid free plasma startup also forms one of the key features of a low aspect ratio ST design by eliminating the ohmic heating (OH) component from the device build. Planned experiments on Coaxial Helicity Injection (CHI) on NSTX and electron Bernstein waves (EBW) on MAST will be used to demonstrate the viability of assisted start-up methods .[5]

Within the PPPL ST-FNSF study, design features were added to a DCLL blanket segment to support the requirements of a CHI start-up scenario. Figure 6 highlights details for an assumed DCLL blanket type along with the features to meet isolation methods for CHI start-up. Developing viable component piping service arrangements will continue to present challenges in all fusion designs. The concentric pipe plenum feed located at the top and bottom to supply PbLi coolant to the inner and outer blanket segments (shown in Figure 6) was changed to a single pass system with one concentric pipe system at the bottom supplying the inner blanket and

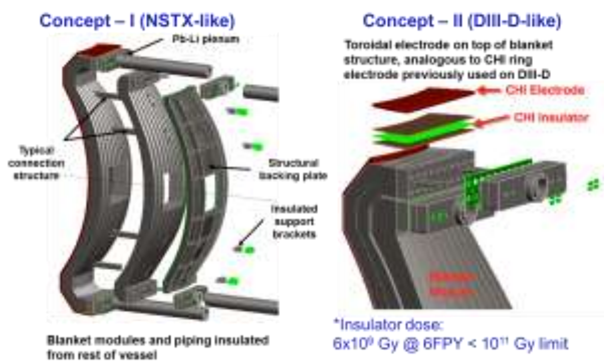

Fig. 6. DCLL blanket supporting CHI plasma start-up

second concentric pipe system at the top to supply the outer blanket. The blanket helium piping details have not been incorporated at this time although access is available, given the spacing of the outer PF coils. Piping services for the inboard blankets, FW and VV have also not been detailed yet. Space is available above the upper VV lid that can be expanded in areas between the TF coil horizontal legs. Greater complications will be found in supplying piping services to interior VV, blanket and shield components at the bottom of the device where disconnect access is required to remove the center post. Further design details would be required to resolve potential space issues.

With the foundation of high-performance ST physics being developed involve NBI heating, the device configuration and conceptual layout of the test cell was developed with particular attention to beam heating. Four angled beams were placed in the 1.7 m device (three for the 1m) with tangency values ranging from R0, R0+a/2 to R0+.75a (as seen in Figure 5). A number of iterations were made to balance port details with respect to physics beam angle, port allocations and device neutron shielding requirements. The JT60SA neutral beam design also was changed from a two source to a three source beam arrangement to minimize the number of beams on the test cell floor. Although not updated to the latest three-source NBI beam design, Figure 7 highlights the general layout of a 1.7 m ST-FNSF test cell showing an arrangement of JT-60SA NNBI's, extracted components resting on the upper test cell floor and the remaining core components that are left in the test cell pit. The ITER building was used in sizing the test cell for the 1.7 m case, resulting in

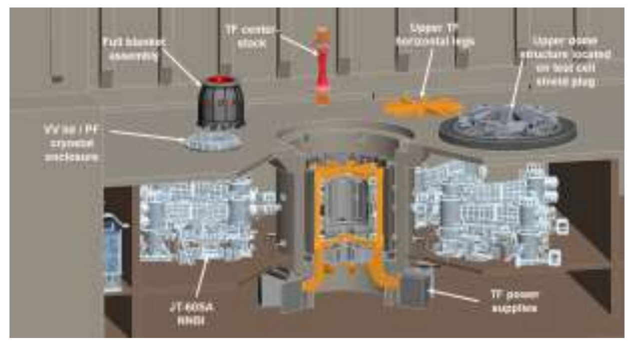

Fig. 7. ST-FNSF facility layout

a building of similar length but somewhat reduced width and height. Cask tritium containment systems will be needed for the extracted components with size and handling details left to a later study.

One final area of review was to examine the basic requirements to supply power to the single turn TF coils. A 86 m wide by 162 m long single floor building was needed to locate an arrangement of twenty-four 1 MA units each comprising four groups of ABB 250 KA power supplies. Figure 8 shows a building layout housing the power supplies. The single turn copper TF coil system carried in an ST design will present issues with regard to operating power balance and costs associated with the power supply equipment – a design feature that will

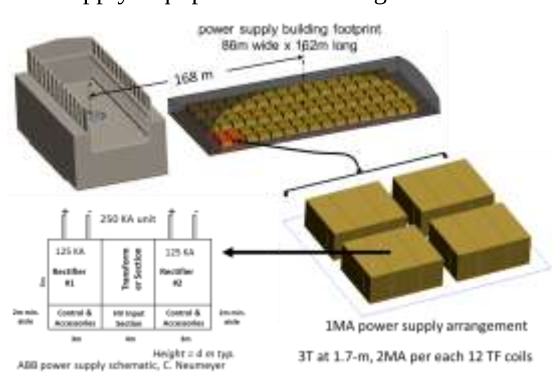

Fig. 8. TF power supplies for a 1.7m device

restrict a copper ST device from being a viable power plant option.

III. B, HTS ST-FNSF Design

Figure 9 shows a High Temperature Superconductor (HTS) ST Pilot Plant design that was developed under a contract with Tokamak Energy (TE) (UK). Design details include: 1.8 aspect ratio, 1.4m R_0, 3.2T B_0, P_{fusion} ~

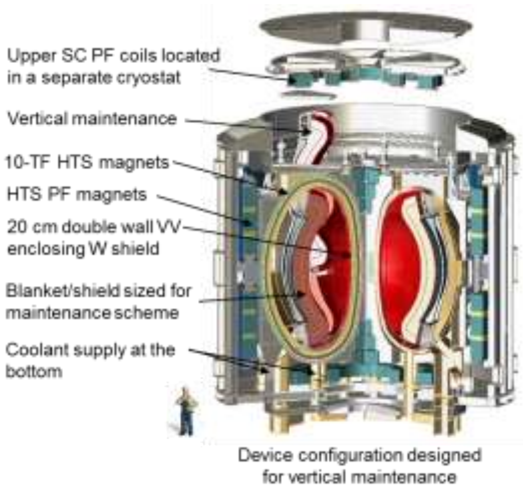

Fig. 9. TE 1.4m HTS ST Design

100MW and Q_{DT} ~ 10. PF coils are configured for a Super-X/snowflake divertor and utilize negative neutral beam injection for heating, current drive, and momentum input. The design incorporates a novel vertical maintenance scheme that allows access to in-vessel components once upper S/C PF coils located in a separate cryostat are removed. With the upper coils and VV access panel removed, in-vessel blanket/shield components can be accessed with vertical maintenance – keeping the eight wedged HTS TF coils and remaining S/C PF coils in their cryogenic state. The TE mission was to develop an ST design that allowed operation for a period of time with electricity break-even conditions. This would require sufficient shielding to allow the TF coils to operate for a week or month at 1MW/m^2 average neutron wall loading before the TF magnet neutron irradiation lifetime limit is met. The design that evolved provides a good starting point to develop a 1000 MW HTS-ST, TBR>1 design to be followed with the evaluation a comparative devices to meet an FNSF mission.

IV. CONCLUSIONS

Significant progress was made within the ST-FNSF study these past few years to develop physics, engineering and neutronics details to enhance the selection process of an FNSF program. The two ST-FNSF designs developed support ex-vessel PF coils to form a Super-X/snowflake divertor that operate with low heat loads, a credible vertical maintenance scheme and an internal arrangement of blanket modules that provide proper port cut-outs to support NNBI yet leave sufficient blanket material to generate high TBR values. The study found that for a copper TF device, 1.7 m was the threshold major radius to operate with a TBR~1 and that a device sized at 1m could provide sufficiently high tritium breeding with lower capital and operating cost – meriting further detailed considerations. Broader mission requirements for FNSF will impact the selection process in choosing between competing fusion options (Cu-ST, conventional Cu tokamak and S/C AT designs). Although the 1.7 m ST device reaches the TBR goal within a realistic engineering design (though still requiring some areas of R&D) there are areas which will weaken its ability to compete with alternate FNSF options. The size of device core is somewhat larger than the 4m PPPL pilot plant advanced tokamak (AT) S/C design which conceptually provides a TBR~1 with $Q_{eng} \geq 1$. In addition the single turn copper TF system dissipates substantial power which reduces Q_{eng} or requires significantly higher fusion power to achieve electricity breakeven. It is also noted that the power supplies needed for 1.7m or larger devices would contribute a high cost penalty unless more compact low-voltage / high-current power supply technology could be developed such as a homopolar generator. For a copper ST device the smaller R ~1m design is likely more attractive for lowering the device cost for achieving an FNSF mission.

Within the time frame of this study the concept details were integrated to form an HTS ST design that may help to conceptualize a feasible ST power plant – this should be pursued to see if it also fits with the expectations of an FNSF mission.

As physics and engineering concept details of all options mature (copper ST, conventional Cu tokamak, HTS-ST, advanced tokamak and stellarators), it would be prudent to evaluate the roadmap to fusion with a substantive overview of all options considering timing, risk, physics and engineering performance, device availability and perceived cost to obtain a better understanding in how best to move from ITER to a commercial power plant. This should include the evaluation of an FNSF mission within the context of competing fusion concept options.

REFERENCES

1. T. BROWN et.al, Progress in Developing a High-Availability Advanced Tokamak Pilot Plant, 24th Fusion Energy conference, San Diego, Calif., 8 October 2012.
2. J. MENARD et.al, "Configuration Studies for an ST-based Fusion Nuclear Science Facility," Presented at

25th IAEA Fusion Energy Conference, Oct. 13-18, 2014, St. Petersburg, Russia.

3. L. El-GUEBALY, et.al, "TBR and shielding analysis in support of ST-FNSF study," CD-ROM of 2013 IEEE 25th Symposium on Fusion Engineering (SOFE), San Francisco, June 10-14, 2013. ISBN: 978-1-4799-0169-2, IEEE Cat. Number: CFP13SOF-CDR (2013).
4. L. El-GUEBALY, et.al, "Nuclear Aspects and Blanket Testing/Development Strategy for ST-FNSF," *IEEE Transactions on Plasma Science*, 42, **No. 5**, 1457-1463 (2014).
5. R. RAMAN, et.al. "Design Description for a Coaxial Helicity Injection Plasma Start-up System for a ST-FNSF", 2014 ANS Winter Meeting and Nuclear Technology Expo, Anaheim, CA.

Princeton Plasma Physics Laboratory
Office of Reports and Publications

Managed by
Princeton University

under contract with the
U.S. Department of Energy
(DE-AC02-09CH11466)

P.O. Box 451, Princeton, NJ 08543
Phone: 609-243-2245
Fax: 609-243-2751

E-mail: publications@pppl.gov

Website: http://www.pppl.gov