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Development of a 3-m HTS FNSF Device and the Qualifying Design and Engineering R&D needed to meet the Low AR Design Point *

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Abstract. A Fusion Nuclear Science Facility (FNSF) study based on the Spherical Tokamak (ST) confinement option has progressed through a number of stages of development to understand the requirements to establish a self-consistent conceptual design of an ST-FNSF device. The study objective was to establish sufficient physics and engineering details needed to meet mission objectives centered on achieving tritium self-sufficiency and magnet shield protection within a configuration arrangement develop with a viable maintenance strategy that fosters high availability in the maintenance of the in-vessel components.

The ST physics is centered on lower aspect ratio designs that offer higher confinement times, improved stability and higher beta operation when compared with the conventional high aspect ratio tokamak. One disadvantage of the small major radius ST device is the machine geometry offers limited space on the plasma inboard side for shielding to protect the toroidal field (TF) coils from neutron heating and radiation damage and space to locate an inboard tritium breeding blanket. This is especially the case when working to define a small size FNSF device; greater inboard space is expected when an ST design is scaled to a larger DEMO device. The earlier copper ST-FNSF designs incorporated a copper center stack of wedged TF plates with joints to the outer return legs and a maintenance approach that involved dismantling horizontal legs of the TF coils to gain access to plasma components and replacing the TF coil center stack after a few years of operation, due to radiation damage. In defining a superconducting ST-FNSF device, sufficient inboard shielding is needed to protect the magnet against radiation for 3.1 FPY of operation and a thin inboard breeding blanket is needed to augment the outboard blanket system. To accomplish these requirements, two design features were pursued: incorporating high-temperature superconducting (HTS) TF coils with a winding designed for high current density (reducing the dimensional build of the TF inboard leg) and reducing the size of the plasma by moving to a device with a slightly higher aspect ratio. This paper provides the design details of the 3-m HTS ST-FNSF device - defining engineering R&D qualifying requirements, structural analysis results and any engineering defined limitations that may be imposed within a low aspect ratio tokamak environment.

1. Introduction

The ST-FNSF multi-year study has moved from an early copper based design of a 2.2 m ST Pilot Plant [1,2] $Q_{engr} \geq 1$ mission to the evaluation of two mid-size devices; a 1.7 m FNSF design with TBR = 1 and a 1.0 m copper design that would operate at a reduced level of tritium breeding [3,4]. The ST design further evolved from the vertically maintained copper designs to a vertically maintained configuration incorporating high temperature superconductor (HTS) TF and PF coils, a thin inboard blanket and a liquid metal divertor; a device designed to provide an FNSF tritium breading ratio (TBR) of 1 mission but with a focus on prototyping a 5-m size range device of a fusion demonstration (DEMO) device. Figure 1 highlights the transition of the overall device configurations through this evolution.

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The lower aspect ratio spherical tokamak designs provide improved physics (higher confinement times, improved stability and higher beta operation) compared with the conventional high aspect ratio tokamak but at a smaller major radius the ST places space limitations for shielding and breeding blankets on the plasma inboard side and offers reduced divertor surface area. An effort was made to define a 3m machine design incorporating superconducting TF and PF coils that would be prototypical of a DEMO device - a device configured to meet physics requirements within an arrangement that also provided a credible maintenance scheme. One early difficulty to overcome was the effective placement of external superconducting PF coils needed to shape a Super-X divertor, not wanting to consider internal PF coils or incorporating jointed superconducting TF coils. A solution was found by locating a small vacuum cryostat within the upper dome at the top to contain a pair of superconducting PF coils used to help form the Super-X plasma divertor geometry. Traditional ST designs include internal PF coils within the bore of the TF coil to form the divertor shape and joints in the copper TF coils to provide the access to in-vessel plasma components. Inclusion of a secondary cryostat in the 3-m HTS ST design allows the warm up of two upper PF coils while retaining the cryogenic environment of the remaining superconducting PF and TF coils. The small upper cryostat allows the vertical maintenance approach defined for the 3-m HTS ST device to follow designs being developed within other tokamak DEMO studies, using superconducting magnets to eliminate power supply and circulating power issues brought about with copper coils.

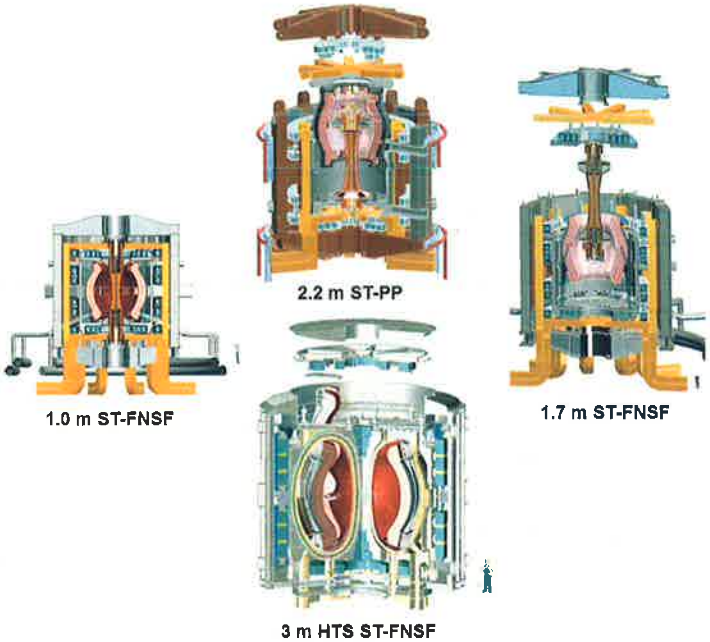

Figure1. Concept evolutions in moving from the copper designs to the more advanced HTS ST-FNSF configuration

In defining a superconducting ST-FNSF device sufficient inboard shielding is needed to protect against neutron heating and radiation damage under the 3.1 FPY operating mission. In addition, a thin inboard breeding blanket was also needed [5] to augment the outboard dual-cooled PbLi (DCLL) blanket system in order to operate with a TBR > 1. To provide the space for shielding and a thin inboard blanket within the condition of a low aspect ratio ST design, the winding of the superconducting TF coils required the operation at high current density (Cd) to minimize its physical size. To add space to the inboard side, the plasma minor radius was reduced, moving to a slightly higher aspect ratio and the TF was designed using a high current density HTS conductor, reducing the dimensional build of the TF inboard leg.

2. 3m HTS ST-FNSF design

The general arrangement of the 3m HTS ST-FNSF design is shown in the exploded view of Figure 2a showing a local upper cryostat containment system that houses two divertor shaping coils with their support structure raised. The major design parameters that the device were an aspect ratio (AR) of 2, 4.0T field on axis and a peak TF field of ~16T. The low aspect ratio ST plasma operates at lower plasma edge ripple while operating with fewer number of TF coils than required for a conventional high AR tokamak design. This allows the 3m device to be designed with ten TF coils when the TF back legs are placed at a location that provides space for the outboard blankets and piping services. Figure 2b shows a close up of the upper cryostat region highlighting the local cryostat insert and the upper PF coil support structure with coils supported beneath it. One of the ten VV access covers is lifted off and an individual blanket sector is shown partially removed.

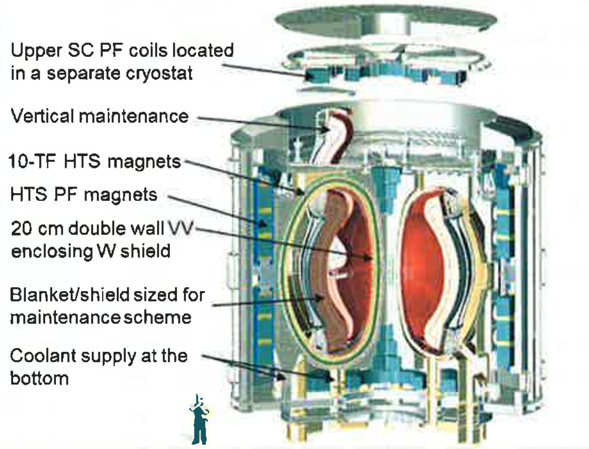

Figure 2a. 3m HTS ST-FNSF device

Figure 2b. Local details of upper cryostat

2.1. TF and PF arrangement

Thermally isolated structural supports connect the small cryostat enclosed PF coils to the TF coil structure, providing a load path for the PF vertical loads back to the TF coil structure. Ten local vacuum panels within the secondary cryostat provide access to remove twenty segmented blanket/shield sectors from within the vacuum vessel plasma chamber. After the warm up and removal of the two upper divertor PF coils, blanket modules can be removed while retaining the cryogenic conditions of the TF coils and all remaining PF coils. ST confinement options provide precious little space on the inner bore for the TF winding and structure, PF coils that help shape the divertor flux lines and inboard shielding. Space conditions improve somewhat as an ST is sized for a larger DEMO design but for the proposed 3m FNSF configuration, the only feasible path forward was to incorporate high current density HTS windings within the TF and PF coils; the current density of advanced low temperature superconductors (LTS) are too low. Figure 3 shows the initial scoping design where the allocated space for the TF inboard leg winding area was set at an overall current density of 36 MA/m^2 to allow sufficient space for TF structure and inboard shielding. To accommodate the higher Cd a number of HTS cables were investigated with an MIT cable arrangement selected as an initial starting point. A 12 sub-cable YBCO twisted stacked tape (TSTC) design shown in Figure 4 provided a higher winding Cd than the winding pack area initially allocated, as identified by the solid area. To

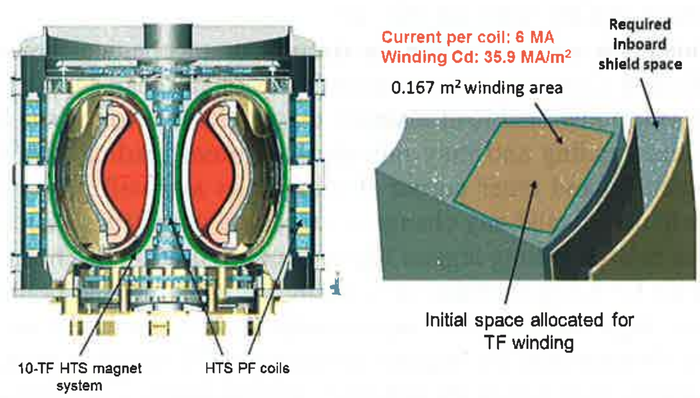

Figure 3. Machine elevation view and TF winding space

provide a small OH flux a small HTS solenoid sized with 70 MA/m^2 Cd resides at the inner bore of the TF coil.

2.2. In-vessel arrangement

The general arrangement of in-vessel components is illustrated in Figure 5 showing a sectioned isometric view with one of the twenty outboard blankets partially removed and a mid-plane plan view highlighting the allocation of port space around the device. As stated earlier a 3m ST device has low ripple even with ten TF coils which provide ample toroidal space on the outside to establish neutral beam orientations that are favorable for current drive. The plan view shows beam plasma tangency radius ranging from 3m to 3.9m interfacing with the 3m major radius, 1.5m minor radius plasma. The NB rotational position on the outside of the device will be optimized to minimize blanket fragmentation which can complicate the blanket design.

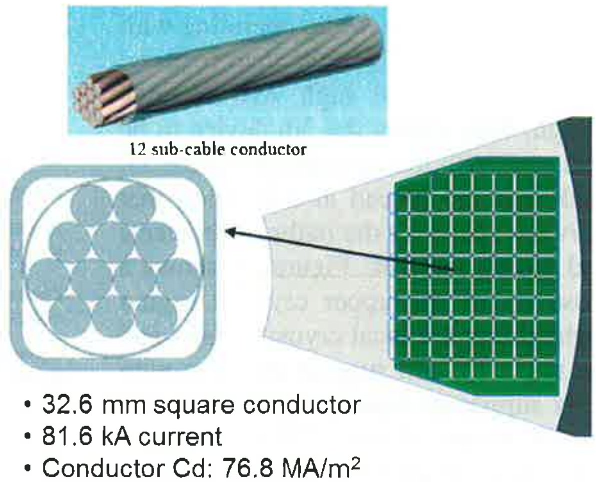

Figure 4. MIT 12 sub-cable YBCO twisted stacked-tape cable (TSTC) conductor

3. TF analysis

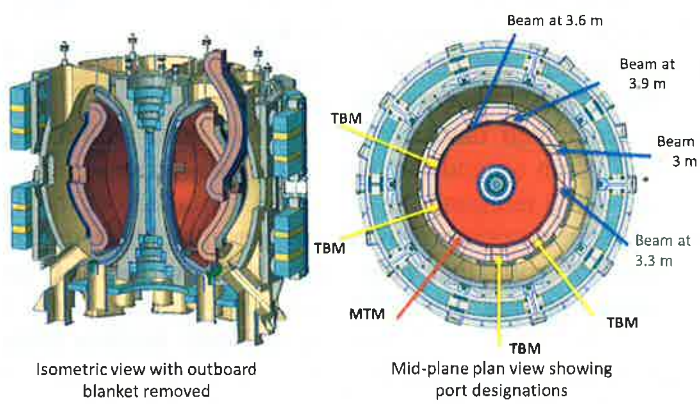

Figure 5. In-vessel arrangement and port allocation

Spherical tokomaks typically have lower fields which reduce the wedging pressures in the TF inner leg. The smaller radial build of the central column also reduces the wedging stress. If a conventional multi-coil case arrangement is chosen rather than the large single central conductor, then the out-of-plane (OOP) load on the TF inner leg must be taken by friction or mechanical keys. With the lower wedge pressure, friction can provide a marginal torsional shear support mechanism. PF coils defined to develop low heat load divertor shapes (X-divertor or snowflake) will impose different out of plane loading and may introduce different loading conditions. ST's offer little space for a solenoid and inner corner shaping coils and will pose new PF coil support challenges. To help understand any change in structural conditions structural analysis of the 3m design with the reference long legged Super-X configuration has been made. The TF coils are cased coils with HTS superconductor winding packs; structural contributions from the tape structure of the high temperature superconductor are included in the FEA analysis. The analysis performed used the original prescribed HTS winding pack area with a Cd that was somewhat smaller than found though later investigations – implying that more nose section structural steel can be used.

A large outer shell structure is used to resist the out-of-plane moments and provide support for the outer PF coils. Openings between the upper-outer spans of the TF coils are needed to allow the vertical servicing logic proposed for the device design. TF case bending must be acceptable around the servicing ports. The inner PF coils are carried on the TF structure and add to the TF in-plane loading including bending of the horizontal legs and the stress state of the TF inner leg equatorial plane section. Torque on the inner legs is taken by a combination of frictional shear and corner pins – like those used in ITER. The results of the analysis illustrated in Figure 6 show TF coil stress levels ~900 Mpa are predominantly located at the TF inboard leg and in very local areas of the vertical port region. This value is above the 666 Mpa allowable and adjustments will be needed to the winding current densities and inner leg cross sections – and/or improved yield case materials will be needed. As stated earlier the winding current density used in this analysis was smaller than finally defined when a more detailed definition of the winding was made which means that there will be more structural steel area available in the nose section to lower the stress level. The OOP Loads for the currently specified PF currents result in stress levels in the vertical port region that are too high but the structural sections defined in the area can be increased in size to bring stress levels down to acceptable levels. The area of major concern is the TF inboard leg where both high TF field and OOP PF forces are applied. Aside from altering the machine physics parameters with reduced TF field and/or incorporating improved structural material; an option to be considered is to reduce the PF currents and OOP forces on the TF by incorporating a liquid metal (LM) divertor – an option to be discussed in Section 5.

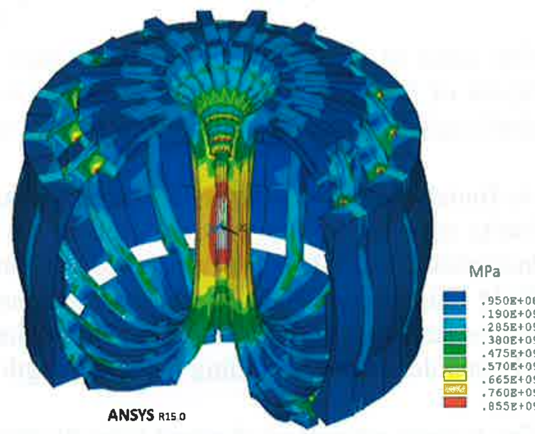

Figure 6. Preliminary FEA analysis

4. Overview of HTS investigation

While an initial HTS winding was established to develop a starting point for a 3m HTS ST-FNSF design, further investigation of HTS options, technology issues and their implementation within a fusion magnet system was carried out. High current cables (a few tens of kA's - ITER 68 kA) are needed for coil protection of fusion magnets during fast discharge (limit coil terminal voltage). At present the most promising HTS material for next-step fusion magnets is YBCO tape or Bi-2212. State-of-the-art technology of YBCO tape fabrication has achieved ~1 km in length of stable conductor performance. The cost at present is a factor of 5 -10 higher than Nb_3Sn wires. Mechanical property of YBCO tape (up to 600 MPa axial stress without performance degradation) is far better and more stable than the strain-sensitive Nb_3Sn. Conductors on round core (CORC®) cables wound from RE-$Ba_2Cu_3O_{7-\delta}$ (REBCO) coated conductors are currently being developed for next generation magnets because of their high flexibility and potential for high engineering current densities. Instead of the circular shaped cross section CORC® cable, a square shape cable can provide higher winding pack current density. REBCO HTS conductors are also more radiation resistant, offering an advantage in the low aspect ratio ST device with limited inboard shielding.

One issue of the YBCO tape is anisotropic in-plane reversible strain effect when wrapping layers of the tapes onto the cable former to make a high current cable. This is particularly challenging when wrapping layers of tapes onto a square shape conductor former.

As found within the analysis of the preliminary reference design an outstanding issue is stress limits imposed by the material used in the coil case and support structure. Higher strength materials are needed. Work has been initiated over a decade ago at National High Magnet Field Laboratory (NHMFL) during the construction of the Series-connected hybrid magnets; Hynes steel was found to be promising when compared to properties of the commonly used 316 stainless steel, providing higher strength.

The biggest advantage of round wire Bi-2212 is that its shape and dimension are very similar to the Nb3Sn wire, so all ITER conductor fabrication technology and facility can be directly used to make Bi-2212 HTS cables. However, the biggest drawback in using Bi-2212 is that it is mechanically very weak (not much mechanical strength) and it incorporates a large amount of silver in the composite wire which would not be acceptable with respect to low activation requirements of a fusion reactor.

5. 3m HTS ST-FNSF developed with a LM divertor

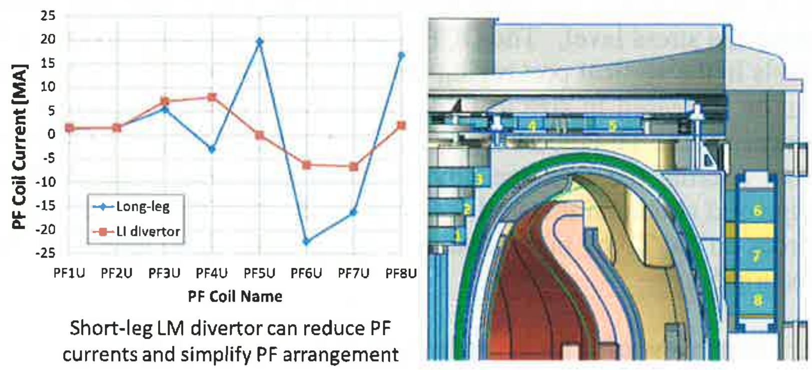

Figure 7. Comparison of Super-X and LM divertor PF currents

Developing the high elongation plasma with expanded divertor flux lines within a Super-X divertor require an increased number of higher current PF coils than needed for the design of a conventional divertor. The issue with the conventional divertor is that the heat load would be unacceptable for a low aspect ratio ST design. To place an ST design in its most advantage arrangement would be to incorporate a liquid metal divertor system. The graph of Figure **7** shows a comparison of PF current requirements for the reference PF coil arrangement supporting a Super-X divertor (blue line) with respect to PF currents needed for a short leg LM divertor (red line) [6]. PF coil designations that are defined at the bottom of the graph are numbered on the adjacent figure. Of great interest in simplifying the ST design is that PF5 is no longer needed which will allow a more conventional vertical maintenance design to be developed with the in-vessel access ports extending through the cryostat, similar to the DEMO tokamak design. Likewise the LM divertor PF arrangement will reduce currents in the PF coils which will reduce TF coil loading conditions. With anticipation that there exist design advantages by incorporating a LM divertor a small effort was made to transition the 3m HTS ST-FNSF design to a LM incorporated option.

Basic design features include:

- A double null divertor with Li flowing over the upper divertor, down the inboard blanket and existing just beyond the lower inboard divertor surface
- A thin metal sheet encompasses the thin inboard DCLL blanket that forms an electrically continuous girdle over which the Li flows
- With no electrical breaks in any inboard Li components any circulating currents occurring during plasma disruptions will result in compression forces which are expected to be manageable
- The lower divertor has a separate Li surface coolant system to keep the Li temperatures to acceptable levels
- Both the upper and lower divertor systems can be installed prior to the installation of the inboard blanket/shield and the outboard blanket systems

Table 1. 3m LM device build

	COMP BUILD, Z=0 (in)	(mm)	(mm)	TOTAL (mm)
	Machine Center			0
	TF center bore	68		68.0
	OH coil	225.0	225	293.0
	OH - TF gap	10		303.0
TF inbd leg	Ext structure	190.0		493.0
	Clearance	2.00		
	ground wrap	4.00		499.0
	Winding pack thk	240		739.0
	ground wrap	4.00		743.0
	Clearance	2.00		745.0
	Ext structure	35.0	477	780.0
	TF-OH TPT	2.0		
	VV TPT	5.0		
	wedge coil asmbly fit up	1.0		
Thermal Insul	Thermal Shield	8.0		
	Min TF/VV Gap	5.0	21	801
inbd VV	VV shell thk	13		
	borated water W shield	100		
	VV shell thk	13	126	927
Shield	WC inboard shield	400		1327
	PbLi blanket	100		1427
	WC / bkt TPT	5.0		1432
Lq Li	Li shell	18		1450
	Liquid Li	10		
	Plasma SO	40		1500
	Plasma minor radii	1500		
Plasma R0				3000

- The inboard and outboard blankets can be maintained independent of the divertor systems

In making changes to the reference 3m design developed with the Super-X divertor some additions were added to the machine build. The spreadsheet listing of Table 1 defines the components make-up and spacing being carried up to the plasma major radius. To achieve a TBR>1 a thin inboard blanket was added to the reference design and the new LM machine configuration adds a thin shell to the outside blanket surface to carry the flow of the LM. Four to five metal sheets are welded together to form a continuous toroidal surface (or girdle) on the inboard side, collecting the Li flow off the upper divertor and carrying it down to a lower channel circulating just inboard of the lower divertor. Figure 8 presents some of the concept details defined for the LM ST-FNSF design. The complete removal of the LM surface is required prior to any replacement of the inboard blankets.

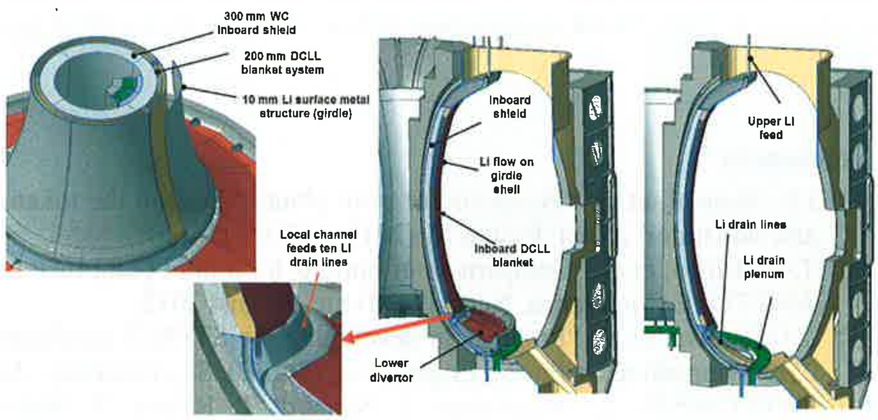

Figure 8. LM local details

6. Conclusions

A 3m HTS ST-FNSF design has been established that promotes in-vessel component maintenance while incorporating continuous TF coils and external PF coils. Physics benefits can be realized by improvements in the device design brought on by incorporating a liquid metal divertor system. A LM divertor provides a higher heat capacity within a smaller footprint than required by the expanded divertor flux line system of a Super-X PF coil configuration. PF coil currents will be substantially reduced with some coils eliminated, allowing design simplification of the cryogenic containment system, improvements in access and maintenance of the in-vessel components and lower TF loading conditions. If confinement time improves with the introduction of a LM as initial experiments suggest, further optimization of the machine size with lower TF fields may be achieved. Based on the PPPL ST-FNSF concept studies, high current densities provided by HTS is needed to meet the physics requirements for a small radius device (~ 3m) within the machine constraints associated with inboard shielding, inclusion of a thin inboard breeding blanket and the structural requirements needed to support the TF coil inner leg. Although not completed, the initial sizing of a 5m ST DEMO device also shows favorability with the inclusion of HTS TF windings.

The objective of this study was to establish sufficient physics and engineering details needed to meet the mission objectives of achieving tritium self-sufficiency and shield protection of a superconducting magnet system within a configuration develop with a viable maintenance strategy that fostered high availability in the maintenance of the in-vessel components. A concept design was developed based on specified physics parameters and an iterated set of PF coils that defined a Super-X divertor geometry. Follow on structural and neutronics analysis benchmarked engineering performances with results indicating that through any follow-on activities, re-optimized components and/or physics performance conditions would allow the mission objectives to be met. Sufficient work also has been done to understand the positive characteristics associated with moving to more advanced technologies (HTS and LM divertors) to place the ST confinement option in its best competitive position.

References

[1] J.E. Menard, et al. "Prospects for pilot plants based on the tokamak, spherical tokamak and stellarator", Nucl. Fusion 51 (2011) 103014 (13pp)

[2] T.G. Brown, et al. "Comparison of options for a pilot plant fusion nuclear mission", 20th ANS TOFE Conference, Nashville, Tenn., August 2012.

[3] T.G. Brown, et al. "Progress in developing the STFNSF configuration", Proceedings of 25th Symposium on Fusion Engineering (SOFE), San Francisco, June 10-14, 2013.

[4] L. El-Guebaly, L. Mynsberge, J. Menard, T. Brown, S. Malang, and L. Waganer, "Nuclear Aspects and Blanket Testing/Development Strategy for ST-FNSF," IEEE Transactions on Plasma Science, 42, No. 5, 1457-1463 (May 2014).

[5] L. El-Guebaly, M. Harb, A. Davis, J. Menard, T. Brown, "ST-Based Fusion Nuclear Science Facility: Breeding Issues and Challenges of Protecting HTS Magnets," to be published in Fusion Science and Technology.

[6] J.E. Menard, "Low-A Tokamak FNSF / Pilot Plant Studies", PPPL Advisory Committee, May 2016

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