Nuclear Aspects and Blanket Testing/Development Strategy for ST-FNSF*

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***Abstract*— One of the main technology missions of a Fusion Nuclear Science Facility (FNSF) is to validate the performance of an integrated set of in-vessel components in prototypical fusion operating conditions prior to inclusion in demonstration and/or first-of-a-kind power plant. The FNSF developed by Princeton Plasma Physics Laboratory will enable such integral testing of fusion technologies. The blanket testing and development strategy requires access for a number of test blanket modules (TBM) and a base blanket installed in the available space surrounding the TBMs and heating/CD ports. A unique feature of the proposed strategy is that the TBMs play a key role and serve as "forerunners" for a more advanced version of the base blanket. The maximum achievable tritium breeding ratio (TBR), the shielding of all magnets, and the radial build definition are among numerous design issues investigated in detail. Potential means to increase the TBR were also investigated.**

Keywords*— *fusion nuclear science facility; blanket testing strategy; tritium breeding ratio; radial build; neutron wall loading distribution

I. INTRODUCTION

International fusion roadmaps, from ITER to the first power plant, take different approaches and levels of risk, depending on the degree of extrapolation beyond ITER and the wide range of near-term and advanced power plants. Figure 1 displays three potential pathways to fusion energy. In the US, the Fusion Nuclear Science Facility (FNSF) is viewed as an essential element of the fusion roadmap. It provides a validated technical basis for a fusion demonstration (DEMO) plant and could be constructed in parallel or follow the ITER experiment.

Prior to DEMO and power plant construction, fusion power components and subsystems must be tested in a dedicated facility with a fusion-relevant environment. The FNSF [1-4] will enable such integral testing and development of fusion technologies under prototypical fusion power conditions. A leading candidate for such a facility is the ST-based FNSF due to its compactness and modular configuration with demountable toridal field (TF) Cu coils [1]. Figure 2 shows an isometric view of the recently developed ST-FNSF by the Princeton Plasma Physics Laboratory (PPPL). The main components are the OB blanket (pink), TF Cu coil legs (orange), center-stack (dark orange) with Bitter plates at top/bottom, and vacuum vessel (VV; gray) located inside the TF coils [5].

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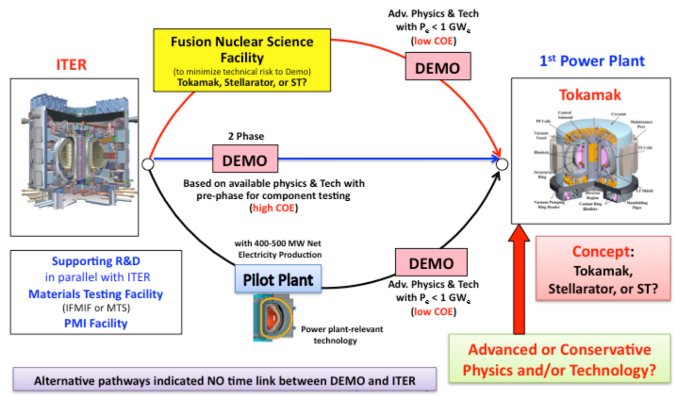


Fig. 1. Potential pathways to fusion energy.

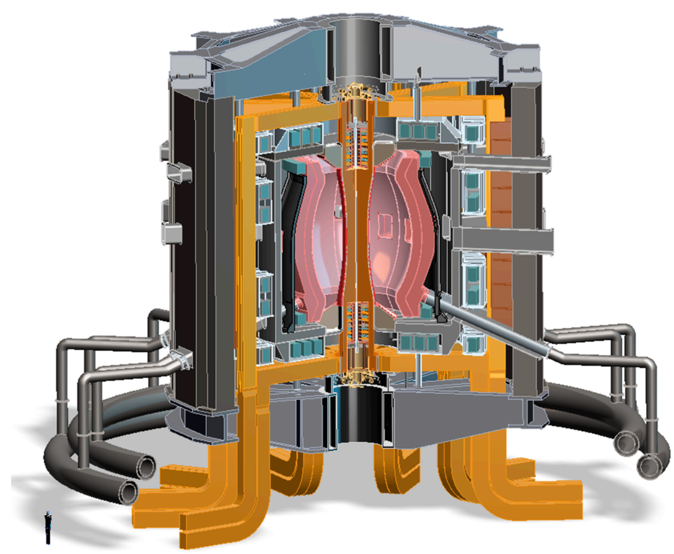


Fig. 2. Isometric view of the ST-FNSF [5].

Key ST-FNSF parameters are fusion power of 162 MW, major radius of 1.7 m, machine average neutron wall loading of 1 MW/m^2, and 6 full power years (FPY) of operation with up to 9 MWy/m^2 end-of-life fluence at the outboard midplane. For such a high power density ST device, the divertor region is a critical and challenging area [1]. The snowflake (or Super-X) divertor enables flux expansions to mitigate the high heat flux problem. A unique ST-FNSF configuration has been identified for such advanced divertors in which Bitter Plate Cu coils are placed at the ends of the center-stack to enable high triangularity plasma shapes with flux expansion compatible with demountable TF coils and removable center-stack [1]. Other resistive coils are added near the divertor (just outside the VV) to help shape the snowflake divertor arrangement. The design allows vertical access from the top to replace the blanket on a regular basis and remotely maintain all internal components [5].

A staged blanket testing strategy was developed for ST-FNSF to be able to test, learn, and enhance the blanket performance during operation. Three generations of LiPb-based blanket could be tested as the operating and heat recovery systems are incrementally improved. This blanket development strategy requires access to a number of test blanket modules (TBM) arranged on the outboard (OB) midplane of the ST-FNSF where the neutron flux peaks. A low-technology, but robust and highly reliable base blanket capable of breeding adequate tritium is installed from the beginning of operation in the available space surrounding the test modules (and other penetrations) to supply the tritium needed for plasma operation. As discussed in Section V, the combined results from the upgraded base blanket and TBMs are essential to build a sufficiently high confidence level for a successful operation of an advanced blanket in DEMO from the outset of its operation.

The ability of the base blanket to achieve tritium self-sufficiency, the neutron wall loading profile, the dose to the insulator of the Cu Bitter Plate and divertor coils, and the radiation protection of the OB superconducting poloidal field (PF) magnets are among several nuclear issues investigated in detail for the ST-FNSF design. High fidelity in nuclear predictions mandates performing state-of-the-art nuclear 3-D analyses. This has recently been achieved through coupling the CAD system directly with the 3-D MCNP code [6], using DAGMC [7] – a newly developed code by the University of Wisconsin. Such a coupling preserves the essential design elements for the individual calculations and speeds up feedback and iterations to quickly converge on an acceptable configuration that satisfies the ST-FNSF goal, mission, and requirements. Currently, the ST-FNSF design considers a range of machine sizes with 1 - 2.2 m major radius. Understanding the impact of the various device sizes on the TBR is another important ongoing research activity to determine the threshold in device size to achieve T self-sufficiency.

II. TRITIUM BREEDING RATIO

The calculated TBR represents a metric for T self-sufficiency. Achieving a TBR of 1 (or more) is a strong requirement for ST-FNSF to avoid purchasing a large quantity of T from external sources with uncertain pricing. Current estimates range from \$30,000 to \$100,000 for a gram of T. A 100 MW fusion power machine consumes 5.56 kg of T annually. For the R=1.7 m device with 162 MW fusion power, a deficiency in TBR as small as 1% is equivalent to 90 g of T/FPY, costing \$2.7-9M annually. The implication is that a TBR of 0.8 results in an annual T cost of \$54-180M, as shown in Fig. 3, representing a significant contribution to the operational cost. Clearly, the cost of T is expensive enough to enter the evaluation process in terms of defining the FNSF mission (e.g., T self-sufficiency) and possibly constraining the machine size and fusion power.

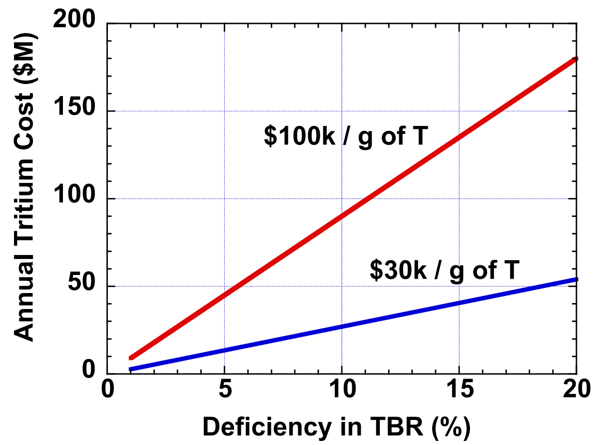


Fig. 3. Cost of puchasing T from exteranl sources versus deficiency in TBR.

Several design elements could easily degrade the TBR [8]. Efforts should be made from the outset to maximize the TBR via increasing the blanket coverage, reducing the structural content within the first wall (FW) and blanket, and minimizing the OB penetrations. We performed a preliminary 3-D TBR analysis to highlight the importance of the blanket coverage to the achievable TBR for the R=1.7 m device. The concept of choice for the base blanket is the dual-cooled LiPb (DCLL) with ferritic steel (FS) structure, LiPb breeder/coolant, and helium coolant [9]. A typical DCLL blanket design is shown in Fig. 4.

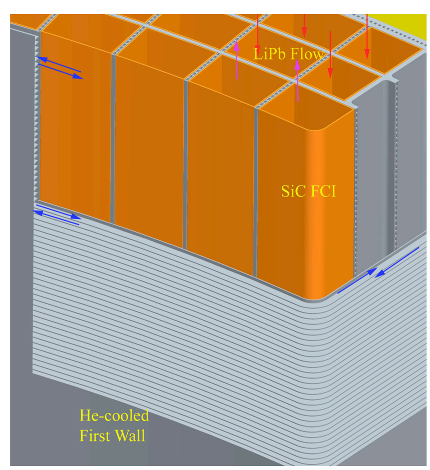


Fig. 4. Typical DCLL blanket design [Courtsey of X. Wang, University of California, San Diego].

The initial 3-D neutronics model, shown in Fig. 5, represents 1/20th of the entire device. There are 10 blanket modules that span 36^o each, thus the 36^o wedge in the 3-D model represents the upper half of a complete blanket module. Reflecting surfaces were placed on both sides of the 36^o wedge as well as at the midplane. The neutron source distribution within the plasma region was approximated using three-nested source regions with varying intensities (63%, 32%, and 5%).

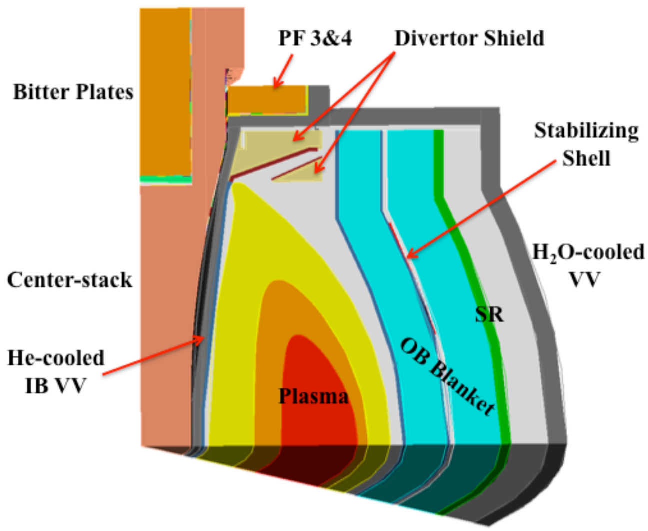


Fig. 5. Initial 3-D TBR and shielding model.

At this early stage of the design, homogenization was considered for the internal LiPb flow channels and He/FS cooling channels. However, several design elements that degrade the breeding significantly [8] were included in this model. For instance, for the inner blanket only, the 3.8 cm thick He-cooled FW and 3 cm thick back wall are modeled separately. Also, the 2 cm thick W stabilizing shell is placed between the two blanket segments (50 cm thick each). The divertor shield, structural ring (SR), vacuum vessel (VV) and magnets are all included in the 3-D model of Fig. 4. The He and LiPb manifolds and blanket feeding pipes are located at the bottom of the machine [5]. A few other components were omitted since their impact on the TBR is insignificant. The ^{6}Li enrichment is 90% for the LiPb eutectic that contains 15.7 atom % Li and 84.3 atom % Pb [10]. Figure 6 displays the T production within the LiPb breeder. It peaks around the midplane and fades out as one moves upward/downward and outward. This emphasizes the importance of the midplane for breeding and suggests keeping the OB midplane free of penetrations to maximize the TBR. The figure also illustrates the reduction in breeding around the stabilizing shell due to the strong neutron absorption by W.

The computed 3-D TBR for several blanket configurations is given in Fig. 7. The initial configuration has a TBR of 0.9. It is less than unity due to the less conformal blanket configuration resulting in high neutron losses in the large divertor slot – a typical feature of Snowflake (Super-X) divertors. We investigated means to enhance the breeding through several consecutive changes to the blanket configuration and coverage. In each step, we evaluated the impact on TBR as illustrated in Fig. 7:

- Extending the upper/lower ends of the blanket inward with more conformal FW increased the TBR from 0.9 to 1.04
- Replacing the upper divertor shield by blanket increased the TBR further to 1.08
- Adding LiPb in the 8 cm thick inboard (IB) VV increased the TBR to it maximum value of 1.15.

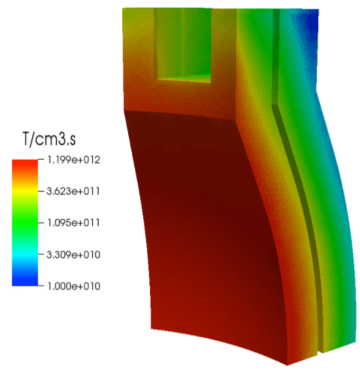


Fig. 6. Tritium production distribution in OB LiPb breeder. The slot at the end of the blanket is for divertor replacement.

The first two options were judged practical and will be implemented in future designs. However, placing a blanket behind the divertor could best be coupled with an IB blanket system. The reported TBR is based on a partially homogenized base blanket covering the OB side completely. We expect modeling the details of the blanket internals, adding the NBI/heating/CD ports, and inclusion of the TBMs on the OB midplane to degrade the 1.08 TBR by 7-8%. The uncertainty in LiPb nuclear data [8] could further reduce the TBR by 3%. However, a few design changes could enhance the TBR and bring it closer to the threshold for T self-sufficiency. These include only five (not 10) but deeper divertor access slots at the end of the OB blanket, lower water content in the center-stack (< 35%), optimization of the W stabilizing shell, and off midplane ports.

III. Magnet Radiation Protection

For STs, achieving high triangularity requires sets of PF coils to be placed both inboard and close to the divertor X-points [1,2]. This could be challenging since these Cu coils will be subject to a high radiation environment. Radiation damage to the insulation of these coils could substantially limit their service lifetime. Figure 8 shows a potential arrangement in which Bitter plate coils are installed at both ends of the center-stack and PF coils are placed behind the divertor plates.

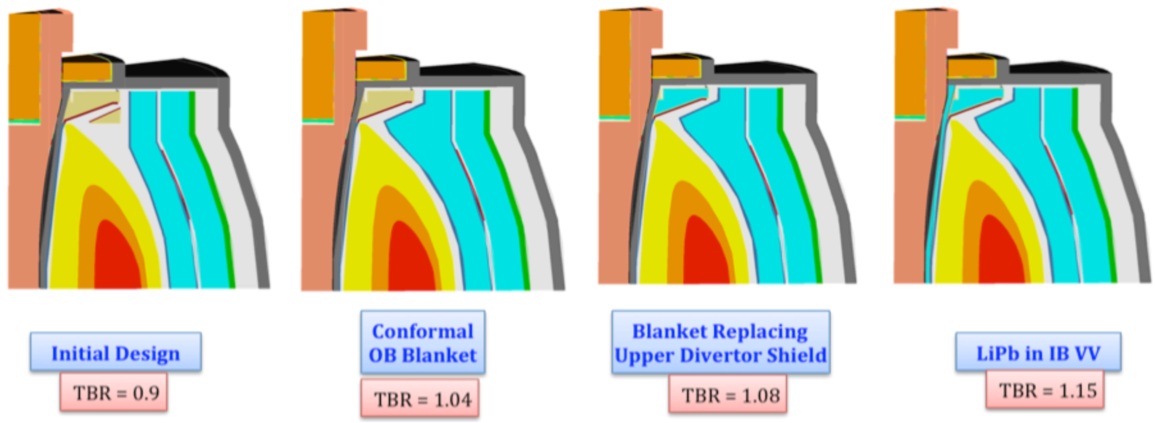


Fig. 7. Sensitivity of TBR to blanket coverage, replacing the divertor shield by blanket, and filling the IB VV by LiPb.

Preliminary shielding analysis indicated the dose reaches ~2 x 10^{10} Gy at 6 FPY. This dose is two orders of magnitude above the allowable limit for the cyanate ester/epoxy organic insulator. However, the MgO ceramic insulator is more radiation-resistant and can stand 10^{11} Gy (10^{13} rad) [11]. We evaluated the dose to the MgO insulator of the Bitter plates and PF 3&4 coils along the red lines shown in Fig. 8. Borated-FS filler was considered for the He-cooled divertor shield and the 20 cm thick top/bottom water-cooled VV. Figures 9 and 10 display the 3-D results showing doses < 10^{11} Gy (10^{13} rad) limit for MgO. Peakings in the dose occur at the lower edge of the Bitter plates (closest to the plasma) and at the outermost edge of PF 4 coil. The latter peaking is mainly due to neutron streaming through the vertical gap between the OB blanket and divertor shield. Closing this gap would reduce the dose significantly. We also evaluated the dose to the PF 3&4 coil for the proposed design changes to enhance the TBR. Replacing the divertor shield by blanket, as shown in Fig. 7, will increase the dose to PF 3&4 coils by a factor of two – still below the MgO dose limit.

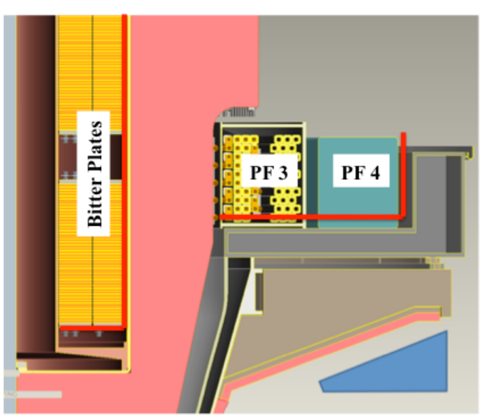
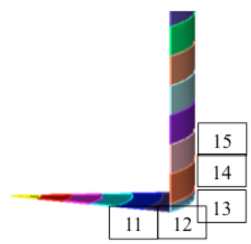
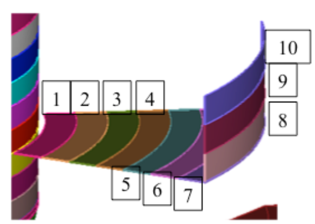


Fig. 8. Dose to MgO evaluated at PF coil surfaces.



Surface #	Heating [W/cm^3]	Dose @ 6 FPY [Rads]	Statistical Error [%]
11	7.90E-03	4.16902E+10	10.94%
12	1.13E-02	5.98327E+10	6.74%
13	1.14E-02	6.00234E+10	7.80%
14	7.33E-03	3.87231E+10	5.35%
15	3.60E-03	1.89989E+10	8.19%

Fig. 9. Dose to MgO at outermost surface of Bitter plates.



Surface #	Heating [W/cm^3]	Dose @ 6 FPY [Rads]	Statistical Error [%]
1	9.49E-03	5.00976E+10	3.72%
2	1.31E-02	6.89907E+10	3.15%
3	1.34E-02	7.08298E+10	2.94%
4	1.43E-02	7.57211E+10	2.71%
5	1.54E-02	8.10700E+10	2.55%
6	1.77E-02	9.34978E+10	2.49%
7	2.03E-02	1.06908E+11	2.25%
8	1.78E-02	9.39528E+10	2.53%
9	9.63E-03	5.08576E+10	3.45%
10	3.58E-03	1.89079E+10	5.04%

Fig. 10. Dose to MgO at outermost surface of PF 3&4 coils.

Other interesting data needed for the thermal hydraulic analysis is the vertical distribution of the nuclear heating along the innermost surface of the center-stack. This is shown in Fig. 11 indicating a peak of 4.6 W/cm^3 at the midplane with considerable reduction at the top/bottom due to the additional protection provided by the divertor shield.

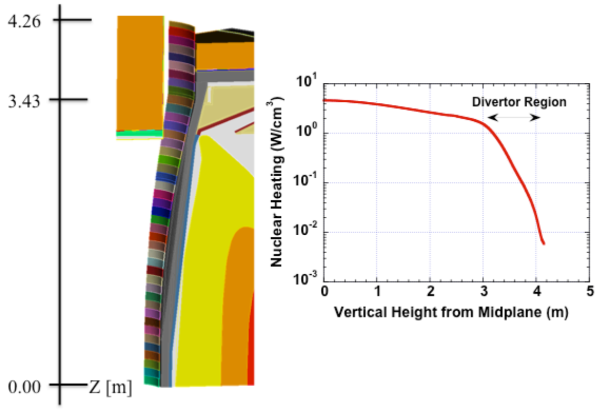


Fig. 11. Vertical distribution of nuclear heating at the outermost surface of center-stack.

IV. Radial Build Definition

After determining the OB blanket dimensions (that breed sufficient T for plasma operation), the OB shielding design proceeded with close interaction between the effectiveness of the preferred shielding materials, their activation characteristics, and safety impact. The design requirements of Table I determined the combined dimensions of the OB blanket and structural ring (SR) needed to protect the OB VV for 6 FPY (life of plant). A common goal for all specialized components (blanket, SR, and VV) is to provide a shielding function to collectively satisfy the radiation protection requirements for the OB superconducting (S/C) PF magnets. This helps define the most compact operational space of the device with minimum radial standoff to ultimately free ex-vessel space for structural connections, cooling pipes, coil leads, etc.

TABLE I. ST-FNSF DESIGN REQUIREMENTS AND RADIATION LIMITS

Calculated overall TBR (for T self-sufficiency)	~1 with 90% Li-6 enrichment
Damage to FS structure (for structural integrity)	60 dpa
Reweldability of FS	1 He appm
S/C PF coils (@ 4 K):	
Peak fast n fluence to Nb_3Sn (E_n > 0.1 MeV)	$3x10^{18}$ n/cm^2
Peak nuclear heating	2 mW/cm^3
Peak dpa to Cu stabilizer	$6x10^{-3}$ dpa
Peak dose to cyanate ester/epoxy insulator	$2x10^{10}$ rads ($2x10^8$ Gy)
Plant lifetime	20 y (= 6 FPY)
Plant availability	10-50% (30% average)
Operational dose rate to workers and public	< 2.5 mrem/h

A fundamental constraint for all nuclear testing facilities is to achieve an average NWL of at least 1 MW/m^2 to accelerate the testing process. For ST-FNSF, an average NWL of 1 MW/m^2 is established [1]. Figure 12 displays the poloidal distribution along the IB and OB first walls. The NWL peaks at ~1.5 MW/m^2 at the OB midplane – the preferred location for test blanket modules.

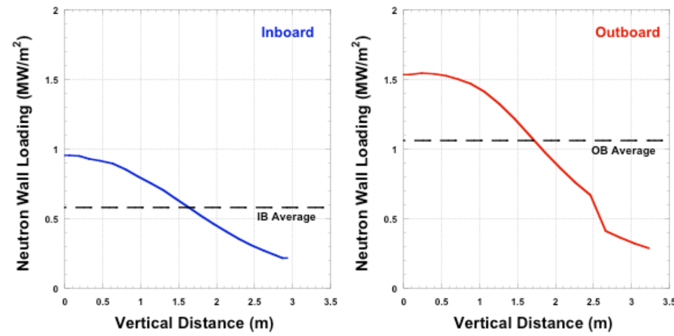


Fig. 12. IB and OB neutron wall loading distributions.

The SR, VV, and all coils are assumed to be lifetime components. The center-stack and divertor may require frequent replacement during operation, but their life-limiting criteria are unknown. The peak damage at the midplane of the OB FW is 13 dpa/FPY. Considering a conservative 60 dpa limit for FS structure, the inner blanket segment should be replaced every 4-5 FPY. The SR and VV are reweldable away from assembly gaps and penetrations. A tradeoff analysis of water and B-FS filler defined the optimal composition of the OB VV: 33% FS structure, 20% B-FS, and 47% H_2O, by volume. Figure 13 shows the optimum radial build based on these specifications.

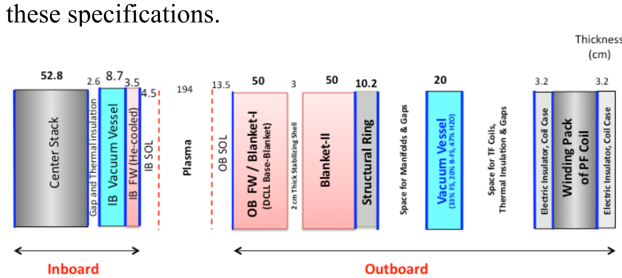


Fig. 13. Optimum IB and OB radial builds of ST-FNSF.

On the IB side, there is 3.5 cm thick He-cooled FW and 8.7 cm thick VV to reduce the nuclear heating and radiation damage to the center-stack. Our effort focused on selecting the optimal VV composition that maximizes the OB breeding – a strong design requirement. Candidate fillers and coolants for the 6.3 cm space between the double walls of the VV include FS, WC, He, H_2O, B-H_2O, and D_2O. As shown below, no single material satisfies all requirments, calling for a compromise. Figure 14 a,b illustrates the sensitivity of the TBR of the OB DCLL blanket to the IB shielding materials. In general, the OB TBR decreases with higher coolant content in IB VV. FS/He IB VV maximizes the OB breeding. Water slows down and absorbs the neutrons resulting in less reflection to the OB, degrading the breeding. Heavy water (D_2O) has less impact on the TBR because of the lower neutron absorption cross section compared to water.

The dependence of the damage to the Cu of the center-stack on the He content in the VV is displayed in Fig. 14 c,d. Both nuclear heating and fast neutron fluence (indicative of the transmutation in Cu) increase with He content. It is recommended to keep the He in the VV to a minimum of 10%, by volume. Even though other materials could provide better shielding for the center-stack, as shown in Fig. 14 e-h, they were excluded for degrading the OB TBR. Therefore, the He-cooled FS is the preferred option for the IB VV as it offers a superior advantage in terms of enhancing the OB breeding.

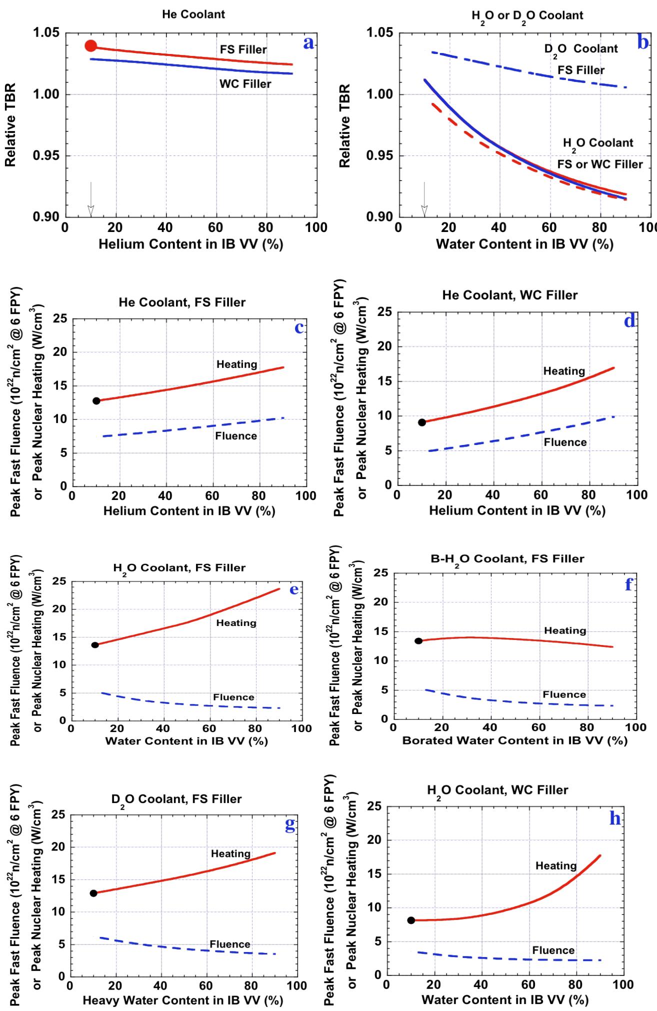


Fig. 14. Sensitivity of OB TBR and center-stack damage to IB VV materials.

V. BLANKET TESTING STRATEGY

The majority of the OB midplane is dedicated for blanket/materials testing and validation. To maximize the breeding, the base blanket should cover the entire space surrounding the TBMs and ports for plasma control. The main features of the base blanket include low technology (to reduce risk), robustness, high reliablility (for lower failure rate and reasonable availability), and, along with the TBMs, capability of breeding most, if not all, the tritium needed for plasma operation (for T self-sufficiency). High coolant temperatures (> 550°C) are not required for this base blanket. However, its manufacturing should be possible with minimum extrapolation from the present technology database. To assure high reliability, sufficiently large margins from the absolute limits (maximum structure temperatures, inter-phase temperatures to the coolant, and mechanical stresses) should all be considered in designing the base blanket coupled with an extensive R&D and validation programs before use in the ST-FNSF.

As mentioned earlier, a favorite candidate for such a base blanket is the DCLL concept. It can operate during an initial stage with rather low coolant temperature (e.g., LiPb and He inlet/outlet temperatures of 350/450°C). This concept requires flow channel inserts (FCI) to serve as thermal and/or electric insulators [9]. If the more advanced SiC-based FCIs (that allow high LiPb exit temperature of 700-800°C) cannot be developed and qualified within the FNSF timeframe, low-technology sandwich-like inserts made of a FS/alumina/FS multilayer could be employed for the base blanket. Since its operating temperature is not too high, the FS/alumina/FS multilayer inserts do not actually serve as a thermal insulator, rather they act only as an electric insulation to control the MHD pressure drop for LiPb. Other features of this first-generation (GEN-I) base blanket include:

- Low-activation FS structure operating at 400-500°C
- Helium-cooled FW and blanket structure
- Temperature in FW and blanket structure as uniform as possible (to minimize the thermal stresses)
- FCI made of SiC, if available, or sandwich-like FS/alumina/FS
- Be multiplier to enhance the breeding, if needed.

Later, we will introduce a new concept of changing out the initial base blanket with a higher technology version required for DEMO and advanced power plants.

The TBM configuration could offer the opportunity to test a wide spectrum of blanket concepts in an environment representative of DEMO or power plant. This would include conventional GEN-I blanket technologies (ceramic breeders and liquid breeders with FS structure operating at 400-500°C), moderately aggressive concepts (GEN-II blanket such as DCLL with LiPb exit temperature of 700-800°C), and advanced blanket concepts (GEN-III blanket with SiC/SiC composite structure operating at ~1000°C). They could all be tested in 4-6 TBM ports. For liquid breeder blankets, the footprint at the FW could range from 1.5 to 2 m poloidally and 0.5 to 1 m toroidally. Two or more ports could be assigned for each blanket concept to enable a reasonable database for "reliability growth testing." A high degree of symmetry for the neutron flux at the TBMs is desirable in order to compare the blanket performances under the same operating conditions. A number of special ports arranged around the OB midplane can be designed to exchange the TBMs without large openings in the VV or without breaking the vacuum.

We suggest a stepwise upgrade for the base blanket in an effort to reach beyond the traditional TBM testing through piloting advanced blankets for DEMO and advanced power plants. In principle, the GEN-I base blanket could operate for ~2 years and then replaced with a new set of sectors containing the GEN-II blanket to test and validate such an advanced blanket concept on a larger scale before utilized for DEMO. During the initial 2-y phase of operation while the GEN-I base blanket is primarily utilized for tritium breeding, the TBMs could develop a GEN-II blanket (e.g., a moderately aggressive DCLL concept with SiC FCI and 700-800°C exit LiPb temperature). Later, the TBMs could be used to develop a GEN-III blanket (e.g., an advanced LiPb concept with SiC/SiC composite structure and 1100°C exit LiPb temperature).

In summary, there are two scenarios determined by the availability of the SiC FCI and advanced FS:

- Scenario-I suggests three generations of blankets (as discussed above) if the SiC FCI and advanced FS are not available for the base blanket at the beginning of PP operation:
  - GEN-I (low-tech base blanket, DCLL, with an exit temperature of 450ºC and FS/Alumina/FS FCI)
  - GEN-II (moderately aggressive DCLL blanket with LiPb exit temperature of 700-800ºC and SiC FCI)
  - GEN-III (aggressive SiC/LiPb blanket with LiPb exit temperature of 1100ºC).
- Scenario-II suggests only two generations of blankets if the SiC FCI and advanced FS are available for use in the base blanket at the beginning of FNSF operation:
  - De-rated GEN-I (base blanket, DCLL, with LiPb exit temperature of 450ºC and SiC FCI)
  - GEN-I (moderately aggressive DCLL blanket with LiPb exit temperature of 700-800ºC and SiC FCI)
  - GEN-II (aggressive SiC/LiPb blanket with LiPb exit temperature of 1100ºC).

In this latter case, the base blanket (GEN-I) could be designed at the outset to be capable of operation at higher temperatures (LiPb exit temperature of 700-800^{o}C, helium exit temperature of ~500^{o}C, and Brayton cycle power conversion system with ~45% efficiency). However, this blanket would initially operate in a de-rated mode to validate its intrinsic subsystem reliability and availability. Then, in the second operational phase, the operating temperature would be increased to the full capability of the DCLL blanket. In other words, the LiPb exit temperature can be gradually increased from a conservatively low value of ~450^{o}C to the higher design value without an exchange of the base blanket.

A unique feature of this staged blanket testing strategy is that the TBMs play a key role and in fact, serve as "forerunners" for a more advanced version of the base blanket, allowing the ST-FNSF to start with a "low-tech" highly reliable base blanket, followed by a stepwise upgrade of the base blanket using results obtained from the TBMs to ultimately validate the characteristics and features of more advanced GEN-II and GEN-III blankets for DEMO and advanced power plants.

VI. Conclusions

In the ST-FNSF with R=1.7 m, a tritium self-sufficiency is achievable with a few modifications to the baseline design, the end-of-life fluence reaches 9 MWy/m^2 at the OB TBMs, and adequate shielding is provided for the divertor PF coils with MgO insulator. Future study will determine the threshold in device size for achieving the breeding goal (TBR ~ 1.04). It could be challenging for smaller devices (R < 1.7 m) to meet this goal since a higher fraction of the OB surface area will likely be devoted to several TBMs and heating ports.

We developed a staged blanket testing strategy for to test and enhance the blanket performance during ST-FNSF operation. Such a strategy requires access to a number of TBMs. Three generations of LiPb-based blanket could be tested. The main emphasis in designing the GEN-I base blanket (that surrounds the TBMs and heating ports) should be on high reliability. The second version of the base blanket (GEN-II) should demonstrate performance and reliability of an advanced blanket concept with sufficiently high temperature for the startup of DEMO. The TBMs could be used to develop a more advanced GEN-III blanket for future power plants (e.g., a LiPb concept with SiC/SiC composite structure). Noteworthy is that, according to this strategy, the TBMs play a pivotal role and serve as "forerunners" for more advanced versions of the base blanket to validate the characteristics and features of more advanced blankets for DEMO and advanced power plants.

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