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## I. Introduction

One of the main missions of the Pilot Plant (PP) is to validate the performance of an integrated set of in-vessel components in prototypical fusion operating conditions prior to inclusion in Demo and/or a first-of-a-kind power plant. These technology validations will reduce the program risk to Demo and future fusion facilities and thus the technology validation testing is an essential mission element for the PP. The primary focus is on testing and validation of blanket and other components in relevant fusion environment, addressing multiple synergistic effects, along with full integration of plasma-facing components (first wall and divertor), blankets, shielding components, and vacuum vessel (VV). All these components must reliably operate with parameters close to conditions expected in Demo and advanced power plants: average neutron wall loading 2-3 $MW/m^2$ and fluence ~10 $MWy/m^2$. Equally important, the PP should test near-term and advanced blanket and divertor concepts, while including provisions for materials irradiation capability and demonstrating that the PP can meet the goal of tritium self-sufficiency, reliable operation, high availability, and efficient remote maintenance.

Candidate fusion concepts for the PP include steady-state advanced tokamak (AT), spherical torus (ST), and compact stellarator (CS). These devices will test all fusion components and validate all the technologies for Demo in a relevant fusion environment, addressing multiple, synergistic effects (neutrons, charged particles, temperature, magnetic field, etc.). Prior to building the PP, laboratory tests should establish the separate, single or some multiple effect database (e.g., temperature effect, magnetic field effect, breeding capacity, tritium recovery, heat removal, etc.).

In summary, the technology validation testing in the PP could demonstrate:

- Integration of PFC, blanket, shield, and VV
- Testing of more advanced blanket concepts
- Materials irradiation testing capability
- Tritium self-sufficiency
- Reliable operation with high availability
- Rapid and efficient remote maintenance.

This document outlines the basic design requirements for the in-vessel components and formulates a blanket development strategy for the PP.

## II. First Wall and Blanket Design

The primary function of the blanket is to breed tritium, capture most of the energetic neutrons and gamma rays, and efficiently convert their energies into useful thermal energy that is transferred to the thermal/electrical conversion system. High-temperature operation is a mandatory requirement to be demonstrated for power plants to achieve high thermal conversion efficiency. The blanket is designed to achieve the necessary tritium breeding ratio (TBR) and efficiently remove the bred tritium to the processing system.

The first wall (FW) and the blanket are designed to be a very synergistic subsystem in that the FW protects the blanket from the high radiant heat and particle fluxes from the plasma. The FW may use the same or a different coolant depending on the plasma conditions. It is likely the FW is attached to and supported by the blanket. The FW has no breeding capability unless a lithium-based liquid metal accomplishes the breeding.

The FW and blanket are subject to high-energy neutrons that continually degrade their materials' strength and properties. To achieve the required sustained level of performance, it is desirable for this subsystem to demonstrate an operational lifetime of several full power years (FPY). High reliability components and provisions for rapid remote maintenance are required to achieve high availability on the order of 50% or more during the PP operational phases.

### *II.1. Tritium Breeding*

The annual tritium (T) consumption in any D-T fuelled fusion device is quite high (5.56 kg per full-power-year per 100 MW of fusion power). Since external sources for such a large quantity of T are predicted to be insufficient, impractical, and/or inaccessible, the PP, as well as future power plants, will produce their own tritium over their lifetime (i.e., an overall TBR > 1) in a blanket surrounding the plasma. Present and larger experimental devices, such as ITER, can acquire the necessary T from external sources. The breeding margin (TBR – 1.0) depends on breeder and multiplier materials, accuracy of the 3- D TBR analysis, and the potential for the PP to supply the startup T inventory for the next device [1]. The Canadians have been recovering T from the CANDU fission reactors at a rate of ~1.7 kg/y. However, ITER will consume most of the T available from CANDU reactors. Therefore, tritium self-sufficiency is a strong requirement for PP and must be considered in the determination of the design point. The AT and CS concepts have unique abilities to breed on the inboard side, offering a larger breeding volume. The ST cannot install a breeding blanket on the central inboard region. Due to neutrons lost to the inboard areas, the ST must rely entirely on the outboard area for T self-sufficiency. The calculated 3-D TBR must be fairly accurate because an uncertainty as small as 1% translates into 0.3 kg of T/FPY per 500 MW of fusion power. This has a significant impact relating to both shortages and surpluses of the overall tritium supply.

### *II.2. Structural Materials*

The currently recommended structural material for the PP FW, blanket, and shield is F82H – low-activation ferritic steel (FS) [2]. This material could operate at high temperatures with remarkable strength in the intense neutron environment and offers safety and environmental advantages compared to commercially available steels. Irradiated bulk material property data from a dedicated high-intensity neutron source will be needed to qualify F82H for use in PP, Demo, and power plants for acceptable operating lifetimes (3-4 FPY). With this structural material, the temperature range is limited to > 350$^{o}$C (to avoid embrittlement) and < 550$^{o}$C (for creep strength). Higher performance low-activation structural materials that offer higher temperature operation (such as ODS-FS (oxide-dispersion strengthened ferritic steel)) may also be employed in the PP and validated in dedicated test module(s). ODS-FS offers remarkable advantages in terms of the operating temperature range and the potential for higher allowable helium concentration. For example, plating the FW with ODS-FS allows higher surface temperature (700-800$^{o}$C). The U.S. materials program is currently examining a range of alloys to build an understanding of the ability of ODS materials to sequester helium produced simultaneously with neutron damage [3].

One of the failure mechanisms for the FS structure is related to the atomic displacements per atom (dpa), limiting its service lifetime at ~100 dpa for conventional FS and ~200 dpa for advanced FS, with the latter being a more attractive limit for power plants. Fusion materials R&D efforts to date have demonstrated FS with radiation resistance to doses of only ~30 dpa in a typical fusion neutron spectrum [4]. Significant developmental and testing efforts are still needed to qualify advanced FS structural materials for future power plants that may operate at high dpa levels up to 200 dpa for acceptable FW lifetimes. Therefore, a few outboard test modules in the PP should be devoted for materials testing with a cumulative neutron fluence approaching 10 MWy/m$^2$.

### *II.3. First Wall*

The FW is an integral and important part of the blanket. The radiant heat flux, energetic particle flux, and neutron wall loading requirements at the FW dictate the structural material, surface armor, and type of coolant. The FW structural material should probably be the same material as the blanket for compatibility reasons. The structural material will have similar thermal, radiation, strength, and lifetime requirements as the blanket. It is likely the structural material is the F82H – a low-activation FS. The FW will have a requirement to withstand a severe environment of intense radiation heat flux and energetic particle flux, but must have a service liftetime of 3-4 FPY. This suggests an armor material that will have minimal sputtering and erosion and exhibit no cracking or spalling. Transient plasma events or disruptions will determine transient heat, particle and electromagnetic loads on the FW, thus strengthening the need for an armor material. The current candidate material will likely be made out of tungsten (W).

The candidate coolants are water, liquid metal (also serves as a breeder) and helium. The coolant/structure compatibility and creep strength of structural materials set the upper limit on the operating temperature whereas the exit temperature from the blanket establishes the upper limit on the power conversion efficiency. Water requires high pressure to remain liquid, has the potential for chemical reactions with the structural materials, liquid metal breeders and/or beryllium, and is limited in its ability to achieve sufficiently high thermal conversion efficiencies. Liquid metals or metal eutectics offer higher operating temperatures at lower pressures, but must use electrically insulating coatings or sleeves to minimize the MHD pumping losses. Corrosion and erosion with high velocity liquid metals is problematic. Liquid lithium alloys or eutectics as FW coolants offer enhanced tritium breeding. The use of high-pressure helium for FW cooling is possible and compatible with the recently developed ARIES He-cooled divertors. Moreover, the use of He in the FW instead of liquid breeder avoids the need for the electric insulator required to mitigate the MHD pumping losses. For these reasons, helium will probably be the favored FW coolant. The maximum operating temperature is dictated by the structure (550$^{o}$C for F82H and 700-800$^{o}$C for ODS-FS). The lifetime could be limited by radiation damage to structure (> 60 dpa; 200 dpa goal for power plants).

The PP FW should resemble and perform like the Demo-type FW. Transient plasma events or disruptions can produce significant thermal and electromagnetic loads on the FW, thus reducing the FW lifetime or possibly causing surface or structural failures. Such off-normal events may require W armor on the FW and/or a sizable structural upgrade within the FW and/or blanket – both solutions may degrade the tritium breeding and/or cooling, depending on the final design and W and FS contents. Sophisticated plasma confinement control and diagnostic systems are necessary to mitigate or avoid the detrimental transient off-normal events in magnetic fusion machines, especially tokamaks. In the meantime and as a fail-safe strategy, the PP FW/blanket should be designed to withstand a few major transient events (disruption, VDE, ELM, etc.) with minimal or no damage. The degradation in tritium breeding due to the added structure to the FW/blanket must be addressed.

### *II.4. Breeding Blanket Concepts*

Numerous Li-based liquid and ceramic breeder blanket concepts have been proposed over the past 40-50 years to help breed T and provide efficient neutron-thermal conversion. Examples include Li, LiPb, Flibe, and LiSn for liquid breeders and $Li_4SiO_4$, $Li_2TiO_3$, and $Li_2ZrO_3$ for ceramic breeders. Beryllium is an essential neutron multiplier to boost the T production for Flibe and ceramic breeders while Li enrichment (20-90%) is necessary for LiPb and ceramic breeders.

Recent ARIES power plant designs [5,6,7] employed the LiPb breeder in self-cooled blankets based on:

- A moderately aggressive blanket concept: dual-cooled LiPb (DCLL) configuration with He-cooled FW/blanket FS structure and self-cooled breeding zones with SiC flow channel inserts to control the MHD pumping losses [5,6,8]
- An advanced blanket concept: SiC/SiC composite structure with LiPb coolant/breeder in the FW and blanket to achieve very high LiPb temperature (~1100$^{o}$C) and thermal conversion efficiencies (~60%) [7].

The U.S. technology program strongly supports the DCLL concept. It is now being considered as a U.S. test blanket module candidate for ITER. However, the ITER contribution to the overall blanket technology program is very limited as the ITER test blanket modules will be subjected to very low neutron fluence (~0.3 MWy/m$^2$; ~3 dpa), breed only a few grams of T per year, and start providing data by 2030 or beyond. Therefore, the DCLL blanket concept should be tested and validated in the PP. For meaningful testing, the PP should provide a neutron wall loading of ~3 MW/m$^2$ at the outboard test modules to shorten the testing period and reach the fluence goal of 10 MWy/m$^2$.

## III. Blanket Development Strategy

The proposed blanket development strategy requires access for a number of test blanket modules (TBM) arranged on the outboard midplane of the PP where the neutron flux peaks. This means the majority of the outboard midplane should be dedicated for blanket/materials testing and validation. Ports for blanket/materials testing should be easily accessible from outside the VV. A low-technology, but robust and highly reliable base-blanket capable of breeding adequate tritium should be installed at the beginning of PP operation in the available space surrounding the test modules and other penetrations to supply all the tritium needed for plasma operation. As discussed below, the combined results from the base-blanket and TBMs are essential to build a sufficiently high confidence level for a successful operation of the blanket in Demo from the outset of its operation.

### *III.1. Base-Blanket*

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 all the tritium needed for plasma operation (for T self-sufficiency) with partial but significant electricity production during the first phase of PP operation (~2 years). To maximize the breeding, the base-blanket should cover the entire space surrounding the test modules and penetrations for plasma control and diagnostics. Since the PP is intended to recover the heat at moderate thermal conversion efficiency, very high coolant temperatures (> 550$^{o}$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 program for LiPb breeder and flow channel inserts (FCI) before use in the PP.

A favorite candidate for such a base-blanket is the DCLL concept. It can operate with rather low coolant temperature (e.g., LiPb and He inlet/outlet temperatures of 350/450$^{o}$C). As mentioned earlier, this concept requires FCIs to serve as thermal and/or electric insulators. If the more advanced SiC-based FCIs cannot be developed and qualified within the PP timeframe, low-technology sandwich-like inserts made of a steel/alumina/steel multilayer could be employed for the base-blanket. Since its operating temperature is not too high, the steel/alumina/steel 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 (F82H) operating at 400-500$^{o}$C
- Helium-cooled FW and blanket structure
- Temperature in FW and blanket structure as uniform as possible (to minimize thermal stresses)
- FCI made of SiC, if available, or sandwich-like steel/alumina/steel
- Be multiplier to enhance breeding, if needed.

As will be discussed shortly, based on the PP TBM results, more advanced operating conditions can be demonstrated for the base-blanket at later stages of PP operation with higher thermal conversion efficiency and thus larger electricity production (e.g., up to 700$^o$C LiPb exit temperature – as required for Demo).

*III.2. Test Blanket Modules*

The flexibility of the PP TBM configuration offers the opportunity to test a wide spectrum of blanket concepts in an environment representative of the demo or power plant. This would include conventional GEN-I blanket technologies (ceramic breeders and liquid breeders with FS structure operating at 400-500$^o$C), moderately aggressive concepts (GEN-II blanket such as DCLL with LiPb exit temperature of 700-800$^o$C), and advanced blanket concepts (GEN-III blanket with SiC/SiC composite structure operating at ~1000$^o$C). They could all be tested in the 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 test modules 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.

S. Malang suggests a stepwise upgrade for the base-blanket [9] 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 be run for ~2 years and then replaced with a new set of sectors containing the GEN-II blanket. During the initial 2-y phase of operation while the GEN-I base-blanket is primarily utilized for tritium breeding and partial energy recovery, the TBMs could develop a GEN-II blanket (e.g., a moderately aggressive DCLL concept with SiC FCI and 700-800$^o$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$^o$C exit LiPb temperature).

The more advanced GEN-II blanket (that offers substantially more electricity production) could replace the GEN-I base-blanket in the second phase of PP operation to test and validate such an advanced blanket concept on a larger scale before utilized for Demo. In other words, the TBMs could serve as "forerunners" for a more advanced version of the base-blanket, allowing the PP to start with a "low-tech" highly reliable base-blanket, followed by a stepwise upgrade of the base-blanket using results obtained from TBMs to ultimately validate the characteristics and features of more advanced GEN-II and -III blankets for Demo and advanced power plants.

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 FCL)
- 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 PP 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, 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 this case, the PbLi 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. As mentioned earlier, the low exit temperature at the start of the PP operation helps minimize the thermal stresses in the blanket structure and improve its reliability. However, a challenging issue with this scenario is the primary LiPb loop and the power conversion system. Operation at full fusion power level and with low LiPb exit temperature would require a considerable increase in the LiPb flow rate and would make the layout of the power conversion system very difficult. A compromise would be to operate the PP for a short initial phase at partial load, then raise both fusion power and LiPb exit temperature simultaneously.

*III.3. Pertinent Questions and Answers*

– *How long does it take to replace GEN-I DCLL base-blanket by GEN-II DCLL blanket?*

  It is likely to take on the order of 6-9 months to replace the first wall and blanket to meet the desired availability constraints. No problem for replacing GEN-I by GEN-II blankets if the ancillary systems (heat transfer and recovery system, and tritium extraction system) are designed to handle it from the beginning. If the blankets are not physically similar, it might take a few months longer. If the ancillary system is entirely new and different, this might need a few more additional months.

– *How many blanket concepts could be employed for GEN-II base-blanket?*

  Switching the base-blanket concept (e.g., from liquid breeder to ceramic breeder) is technically feasible, but requires approximately a year of shutdown and presents an additional cost as different blankets require different coolants and ancillary systems.

– *How many blanket concepts could be tested simultaneously in TBMs? Is it realistic to test both liquid and ceramic breeders in TBMs and, at a later stage, install two GEN-II base-blanket concepts?*

Testing a small number (2 or 3) of blanket concepts is feasible if all are cooled with the same helium coolant for the DCLL base-blanket. If a water-cooled blanket concept is tested in TBMs, it is necessary to build a separate high temperature/pressure water-cooling ancillary system with water detritiation capability as the VV low temperature/pressure water system is inadequate for TBM cooling.

Note that splitting the base-blanket modules between two concepts (e.g., DCLL concept and ceramic breeder concept) doubles the R&D program and could statistically reduce the confidence level for successful operation of Demo from the outset.

– *Could GEN-III LiPb/SiC blanket eventually replace GEN-II DCLL base-blanket?*

No. The GEN-III blanket should only be tested in the TBM form, and not in the complete power core. The reason is that the GEN-I and -II balance of plant (BOP) presently, as designed, cannot handle the much higher LiPb outlet temperature (1100$^{o}$C) of the SiC blanket. However, the PP could continue operating after Demo construction while the GEN-III blanket is tested in the TBMs with limited upgrade to the BOP to handle the higher temperature. Nevertheless, achieving high reliability through full blanket deployment is still needed before employing the GEN-III blanket for Demo.

*III.4. Summary*

The following observations and conclusions can be drawn:

1. The PP asset is too valuable and powerful tool to be used for screening, selecting, and testing of large number of blanket concepts. Instead, this should be done with an extensive R&D program, including dedicated modeling, out-of-pile tests, and irradiation tests in fission reactors. ITER can contribute to this effort as all seven parties have a large number of TBM concepts.

2. In designing the GEN-I base-blanket to be installed from the beginning of the PP operation, the main emphasis should be on high reliability. Efficient conversion to electricity can be demonstrated later with a more efficient version(s) of the base-blanket. This approach allows first designing a version of the base-blanket with reduced LiPb exit temperature, leading to more uniform temperature fields across the blanket and therefore reduced thermal stresses.

3. The second version of the base-blanket (GEN-II) should demonstrate performance and reliability of a more advanced blanket concept with sufficiently high temperature for the startup of Demo.

4. The transition from such a startup base-blanket (GEN-I) to a more advanced GEN-II version (typical for Demo) has to be accommodated from the beginning of PP construction.

5. Depending on progress and results of worldwide blanket R&D program, it remains to be determined if two blanket concepts (e.g., DCLL and ceramic breeder blankets) should be employed as base-blankets in the PP. A few options could be considered:

    a. Prepare the PP design for later transition of base-blanket from Concept-A to Concept-B. A rough estimate is a 6-9 month shutdown and more than double the R&D cost of blanket development.

    b. Install two base-blankets for Concept-A and Concept-B from the beginning of PP operation. The R&D cost would not double if both concepts were developed in parallel instead of in series. However, this option reduces the confidence level for successful operation in the Demo plant from startup and halves the reliability database for each blanket, unless each blanket operates for a longer time. The time for changing from Concept-A to Concept-B may be eliminated, but the life limitations may prevail.

6. Regardless of the number of concepts to be used for the base-blanket in the PP, additional advanced blanket concepts can be tested in the TBM ports. However, high priority should be given to developing the main concept selected for the base-blanket and especially to the reliability growth testing.

## IV. Shielding Criteria and Design

All internal power core components will provide a shielding function. The FW/blanket sufficiently protects the bulk shield so that it can achieve a lifetime equal to the PP machine lifetime (6 FPY; 20 years operating at 30% average availability). In turn, the shield along with the FW/blanket protects the VV and assures its reweldability at any time during operation and also allows the VV to achieve a 6-FPY lifetime. The vacuum vessel, including all of its maintenance, diagnostic and heating/current drive ports, also provides a shielding function to protect all external components. All three components (FW/blanket, shield, and VV) protect the magnets, which are more sensitive to neutron damage. The hefty components (shield, VV, and magnets) are sufficiently protected so they can be considered to be permanent subsystems, requiring no replacement due to radiation damage during the machine lifetime.

The shielding criteria for these components are sufficient to stay below the damage limits of the best presently available data for all power core components and to permit limited human access, if allowed, around the power core after a reasonable cooling period (~1 week). This access is for unusual circumstances as the planned maintenance actions are fully remote operations with no human access inside the bioshield.

There are several options for the shielding materials that vary in effectiveness and cost. Understanding the cost/performance/safety implication of the various options is needed before making a proper choice. Water coolant and bulk shielding fillers (such as borated FS and WC) can be used to enhance the shielding effectiveness of the VV.

For superconducting magnets, the peak fast neutron fluence at the $Nb_3Sn$ superconductor, nuclear heating, dose to the electrical insulator, and radiation-induced resistivity for the Cu stabilizer will all be below the radiation limits specified by the magnet designers ($10^{19}$ n/cm$^2$, 2 mW/cm$^3$, $10^{10}$ rads, and $6x10^{-3}$ dpa, respectively). Well-optimized shield and VV configurations provide lifetime protection for the magnet with the least amount of shielding possible. For ST normal conducting coils with multiple turns, the radiation dose to the insulator is the most limiting factor. Activation of Cu and the restrictive goal of generating only low-level waste may limit the lifetime of the ST Cu-based center stack, thus requiring regular replacement of the center stack.

The Demo and power plant will likely require a hot shield, cooled with the same coolant as the blanket. The shield may also serve as a Skeleton Support or Ring for structural and maintenance purposes. This design concept needs to be demonstrated and validated in the PP.

## V. Vacuum Vessel

The primary functions of the vacuum vessel are to provide the high-level vacuum environment (necessary to achieve and maintain high-quality fusion plasma) and support the internal power core elements. In addition to providing shielding (as noted in the prior section), the VV is a safety-class component that implements safety functions, such as confining the tritium and radioactivity and limiting the public/worker exposure to radiation during accidents. It is a robust water-cooled, double-walled structure with ribs and bulk shielding materials between two 3-5 cm thick structural face sheets. Unlike the blanket and shield, the VV operates at lower temperature (150-200$^o$C) and serves as a heat sink during LOCA/LOFA events. A candidate steel for the VV is the newly developed 3Cr-3WV FS by the Oak Ridge National Laboratory [10,11]. The design requirements for such steels include:

- Operate at low temperature (150-200$^o$C)
- No steel with high strength is required, specially at such a low operating temperature (unlike the high-strength F82H FS for FW/Blanket/Shield)
- No substantial embrittlement at low operating temperature
- Compatible with water cooling
- Generate of only low-level waste
- Generate low decay heat
- Easily rewelded without the need for complex Post Weld Heat Treatment
- Tolerable neutron-induced swelling, particularly behind assembly gaps and near penetrations (10-20 dpa).

## VI. Power Core Penetrations

NBI, RF, or diagnostic ports and divertor pumping ducts have neutron streaming concerns that need special shielding requirements. There will be local ports for the TBMs and maintenance ports to access the base-blanket modules. Local shields surrounding the penetrations help protect the sides of the magnets. Tilting, angling, or bending the ports/ducts help alleviate the streaming problem, but may present access problems for the mounting and maintenance of the TBMs in particular. Locating the ports off the outboard midplane helps achieve tritium self-sufficiency. The internals of both NBI and RF systems survive longer if located at the upper/lower outboard regions, which are subjected to lower neutron fluxes. The NBI footprint at the FW is the largest among all penetrations. The PP could lose up to 5-7% of the tritium generation capability if NBI is used.

## VII. Safety and Environment

The PP, Demo, and fusion power plants must be safe for workers and the general public as well as being environmentally attractive and acceptable. Achieving this goal depends primarily on the activation performance of the FW/blanket and divertor that operate in a harsh neutron environment. Highly irradiated components will have a limited lifetime and require frequent replacement with radiation-resistant remote handling equipment. The alloying elements and impurities in the power core materials along with the operating conditions determine the activation level and dictate the waste management approach: recycling, clearance, or disposal. Continued development and usage of low-activation materials for fusion is required to assure that all activated materials are recyclable and/or clearable. The geological disposal option should be avoided because of concerns about the environment and limited capacity of existing low-level waste repositories [12].

## VIII. Design Integration and Maintenance

The PP must demonstrate that all the power core elements can be designed to effectively integrate all the functions and requirements necessary for the Demo and/or the first power plant. The PP requirements become much more severe if the Demo is not built. In this case, the PP will have to embrace all the Demo requirements.

It might appear each of the power core subsystems could be developed and validated separately, but they cannot be fully validated until they are completely integrated with all the other power core elements in the requisite operating environment in a large-scale fusion facility. This is especially evident when trying to incorporate successful TBM concepts into the entire power core.

One of the most crucial elements of integration and eventual risk mitigation efforts is that of incorporating a highly efficient and fully remote power core inspection and maintenance approach. In the case of scheduled maintenance actions, the maintenance scheme has to be able to quickly remove and replace large power core sectors or segments with high reliability. In the case of unscheduled maintenance actions, the inspection system has to diagnose with high confidence the problem and the maintenance subsystem has to quickly remove and repair or replace the failing or failed component. Plant availability is one of the most critical parameters for a commercial power plant and the PP and/or the Demo must convincingly demonstrate scalable plant availability on the order of 40-50% to suitably reduce this risk factor. This suggests the base-blanket must be able to be removed and replaced in 6-9 months.

## IX. Accommodation of Future Developments

The PP is intended to be the bridging facility between ITER and Demo and must be able to fulfill that mission by validating all subsystems that will be used in Demo and/or future power plants. It is typical for any machine design to specify requirements to be able to accommodate future developments. This means the PP design could be flexible and adaptable for future upgrades of different blanket concepts as there may be a completely new blanket concept discovered in the next two decades. As a goal at this early stage of this study, the PP design may be able to incorporate, test, and validate any advanced blanket concept that emerges in the near future. Admittedly, there will be several changes (design, cost, and schedule) required to make such upgrades a reality [13]. For instance, the primary coolant loop and the tritium handling systems may have to be replaced when the blanket is replaced, if not compatible with the incoming blanket concept. So, designers will have to be very careful about the design details from the beginning and should specify all power core interfaces to assure that the most likely new blanket candidates are essentially compatible with the basic power core geometry and BOP facilities. In some cases, the design process and requirements for accommodating future developments may get crowded out by the basic design requirements of the main mission and achieving such a goal may not be practical in the end.

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