Mobile Tiles for Inertial Fusion First Wall/Blanket

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Abstract— **A conceptually simple first wall (FW) and blanket design for an inertial fusion system based on utilizing mobile FW tiles is presented. Using these tiles that are periodically removed, annealed, and reinstalled, tritium retention and surface erosion concerns for inertial fusion FW could be mitigated. A conceptual configuration has been developed with consideration for laser beam port accommodation and a simple tile insertion and removal scheme. Tritium self-sufficiency can be achieved with a variety of options. The current preferred design option utilizes liquid Li breeder that serves also as coolant for both the FW tiles and blanket with ferritic steel structural material. Thermal analysis for the carbon fiber composite FW tiles in the HAPL nuclear environment indicates that the maximum temperature will be ~1300°C.**

Keywords-inertial fusion; first wall; tiles; tritium breeding; nuclear heating

I. Introduction

Inertial fusion power systems have both added challenges, and the potential for simplified design, when compared to magnetic fusion energy systems. Currently, the approach to design of the FW and blanket for inertial fusion systems closely follows the technological path taken by magnetic fusion systems. The High Average Power Laser (HAPL) program [1] aims at developing laser inertial fusion energy based on direct drive targets and a dry wall chamber. Power plant designs are assessed with 350 MJ yield targets driven by forty KrF laser beams. In HAPL, a tungsten armored ferritic steel (FS) FW is considered along with a "conventional" blanket. However, the extreme flux of intermediate energy ions poses a significant, potentially catastrophic, issue of extremely high-pulsed temperatures and erosion/ablation of FW [1]. These conditions limit material choice and lifetime of the FW materials. For this reason alternative designs are being considered.

In this paper, we present a conceptually simple FW and blanket design for an inertial fusion system. Both FW and blanket segments are composed of monolithic ceramic blocks that slowly, and intermittently, traverse the chamber. These blocks would serve the dual purposes of handling the heat and ion loading from the chamber and breeding the tritium required for reactor operation. At the end of their transit through the chamber they are removed and thermally processed to remove tritium. The thermal annealing may also restore the block thermophysical properties such that it can be reinstalled, otherwise the material would be deconsolidated and recycled into a fresh block. This concept is conceptually similar to that of the pebble bed modular fission reactors.

A HAPL chamber design configuration that utilizes the mobile tiles was developed with consideration for laser beam port accommodation and tile insertion and removal. The FW tiles contain carbon fiber composites and possibly beryllium carbide and are cooled by conduction to a metallic guide rail. Neutronics analysis was performed to assess the tritium breeding potential for different blanket design options. In addition, nuclear heating results were used in thermal analysis to determine the expected temperature distribution in the tiles.

II. Concept Description

The FW tiles are inserted at the top of the reactor. While they are not moving, they transit the core at some schedule determined by erosion and radiation damage. As these tiles are irradiated in the reactor chamber, mechanical properties such as strength, elastic modulus, fracture toughness, and thermal conductivity are affected. Following their removal from the chamber, the tiles are annealed to remove tritium and reverse these irradiation-degraded property changes. Fig. 1 shows degradation and recovery in thermal conductivity of 3-D carbon fiber composite FMI-222 following neutron irradiation and annealing [2]. The overall amount of degradation in thermal properties would depend on the amount of residence time (hence integrated dose) in reactor. Fig. 1 exhibits both the temperature and dose dependence of the thermal conductivity. It is worth noting that a higher fraction of recovery in thermal conductivity will occur for less highly irradiated graphite [3]. Fig. 2 gives tritium release from highly irradiated graphite composite following annealing outside the reactor. Most of the tritium is released at ~1400°C [4]. Following this thermal processing, the tiles are inspected and then either reprocessed or reinserted into the fusion chamber. This concept allows for mitigating the tritium retention and surface erosion concerns.

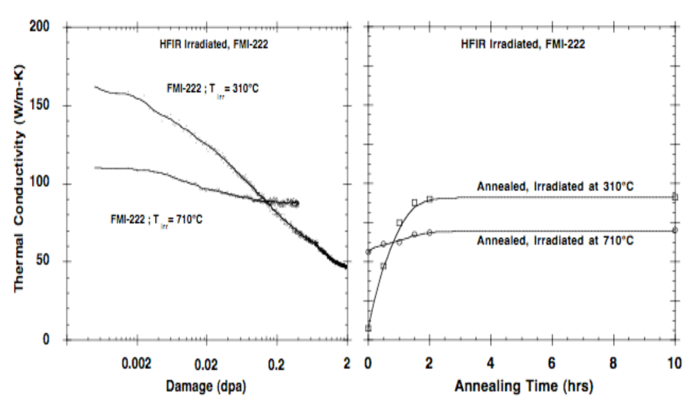

Figure 1. Thermal conductivity recovery with annealing.

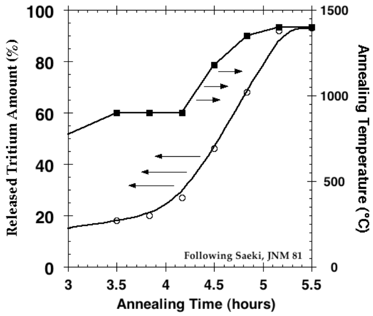

Figure 2. Tritium release with annealing.

III. Tile Configuration in Chamber

We developed a HAPL chamber design configuration that utilizes the mobile tiles. This configuration accommodates the forty laser beam ports. A scheme is outlined for tile insertion and removal. A cylindrical reactor chamber is considered with 10 m radius and 20 m height. The tiles will traverse the cylindrical chamber walls over a period of time. Once removed, they are annealed, inspected, reprocessed, and reinserted along the chamber walls. The top and bottom tiles are stationary and will be removed and reinserted as needed.

Special tiles are utilized at sections of the cylindrical chamber wall with laser beam ports. These tiles traverse the chamber along a coolant rod as shown in Fig. 3. There are 16 locations where these special tiles are used. At the location of the laser port, the tiles will rotate around the coolant rod by following a guiding rail on the coolant rod to avoid interference with the laser beam. For other sections of chamber wall, larger tiles are used and traverse vertically through the chamber without the need to twist. These tiles along with their coolant plates are shown in Fig. 4. The top and bottom tiles will be stationary. Four tiles on the top and bottom each will have an opening for the laser beams. The tiles are installed by sliding them into place on the coolant plates as shown in Fig. 5.

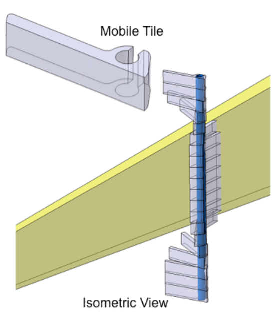

Figure 3. Tile configuration at laser beam ports.

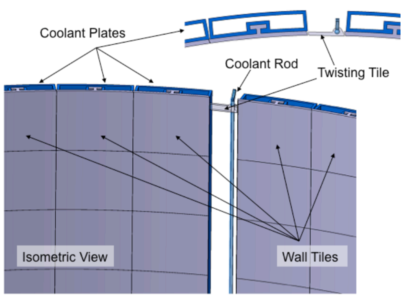

Figure 4. Overall configuration of tiles on chamber side wall.

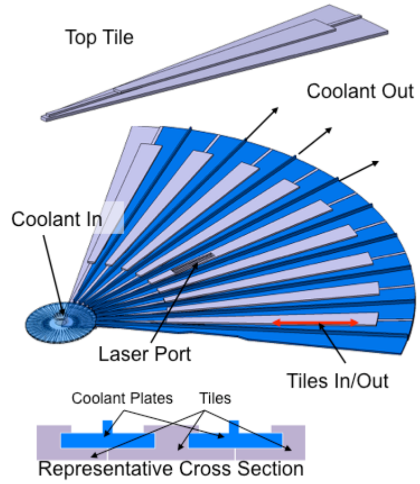

Figure 5. Configuration of top tiles.

IV. Neutronics Assessment

Neutronics calculations were performed to assess the breeding potential for different design options and determine the amount of nuclear heating. The PARTISN 4.0 discrete ordinates particle transport code system [5] was used to perform the neutronics calculations utilizing the FENDL-2.1 nuclear data library [6]. The chamber was modeled in spherical geometry with a point source at the center emitting neutrons with a softened energy spectrum resulting from interactions between fusion neutrons and the dense target materials. 74.7% of the target yield is carried by neutrons with an average energy of 12.3 MeV [7]. 175 neutron and 42 gamma energy groups were used. The breeder options considered include lithium silicate ceramic breeder (Li_4SiO_4), the molten salt Flibe, liquid Li, and the LiPb eutectic ($Li_{17}Pb_{83}$). Coolant options assessed are liquid Na and the liquid breeder. Three structural materials were considered. These are FS, the vanadium alloy V-4Cr-4Ti, and SiC_f/SiC composite. In addition, we considered adding Be_2C in the FW tiles and blanket to improve the tritium breeding ratio (TBR). 7 and 10 cm average FW tile thicknesses were considered followed by a meter thick blanket. A zone consisting of 85% FS and 15% He was used behind the blanket to represent reflection from the shield and vacuum vessel (VV). To ensure tritium self-sufficiency, the calculated TBR should exceed unity by a margin that allows for tritium decay, startup inventory, reserve inventory, and uncertainties in data and modeling. In this study, the TBR is required to be at least 1.1.

A. *Tritium Breeding Potential of Design Options*

We analyzed several design options that utilize Li_4SiO_4. The TBR maximizes with 30% Li-6 enrichment. FS structure was used in both the FW tiles and blanket with liquid Na coolant. The highest TBR was obtained for FW tiles made of 75% C, 10% FS, and 15% Na, followed by a blanket consisting of 79% Li_4SiO_4, 10% C, 10% FS, and 10% Na. The TBR is only 0.911 with 7 cm FW tiles and reduces to 0.868 with 10 cm tiles. Adding Be_2C resulted in improving the TBR. Adding 30% Be_2C in the FW tiles and blanket, the TBR is 1.127 with 7 cm FW tiles and 1.056 with 10 cm tiles. This implies that, if a ceramic breeder is used, at least 30% Be_2C should be added in tiles and blanket and the average FW tile thickness should not exceed 7 cm to ensure tritium self-sufficiency. Notice that tritium breeding for other ceramic breeders is nearly the same and this conclusion applies to any ceramic breeder material.

Calculations of the TBR were performed for the three liquid breeder options (Li, Flibe, LiPb) with three structural materials (FS, V, SiC). Natural Li is used except for LiPb where 90% Li-6 enrichment was also considered. The FW tiles consist of 75% C, 10% structure, and 15% Na. The blanket consists of 90% liquid breeder and 10% structure. Table I gives the results with 10 and 7 cm FW tiles. The results indicate that natural Li and enriched LiPb yield adequate TBR with any structural material for 7 cm or less tiles. The V structure provides the best neutron economy with FS giving the least TBR. Using Flibe does not allow tritium self-sufficiency with any structural material.

TABLE I. TBR WITH SODIUM COOLANT IN FW TILES

	Flibe	Li	LiPb (nat)	LiPb (90% Li-6)
10 cm FWTiles				
FS	0.865	1.045	0.690	1.075
V	0.933	1.119	0.817	1.130
SiC	0.959	1.080	1.042	1.149
7 cm FWTiles				
FS	0.949	1.150	0.812	1.213
V	1.014	1.223	0.954	1.258
SiC	1.012	1.159	1.144	1.248

To avoid using two coolants we considered the option of cooling the FW tiles with the same liquid breeder used in the blanket. The FW tiles consist of 75% C, 10% structure, and 15% liquid breeder. The TBR results are given in Table II. Tritium breeding increased by ~2-5% when the liquid breeder is used instead of Na to cool the FW tiles. However, the conclusions regarding adequacy of TBR are similar to those obtained with Na FW cooling.

TABLE II. TBR WITH LIQUID BREEDER COOLANT IN FW TILES

	Flibe	Li	LiPb (nat)	LiPb (90% Li-6)
10 cm FWTiles				
FS	0.934	1.107	0.808	1.185
V	1.001	1.177	0.948	1.229
SiC	0.992	1.116	1.128	1.210
7 cm FWTiles				
FS	0.983	1.182	0.876	1.267
V	1.043	1.251	1.022	1.303
SiC	1.030	1.182	1.191	1.286

Using Flibe as breeder does not provide adequate tritium breeding with any of the candidate structural materials. We assessed the effect on TBR of adding Be_2C to the FW tiles. The tiles have 10% structure and 15% Na with the remaining 75% being split between C and Be_2C. The blanket consists of 90% Flibe and 10% structure. The TBR results with different amounts of the Be_2C added in the 7 cm FW tiles are given in Table III. Tritium self-sufficiency with a Flibe blanket can be achieved only with at least 30% Be_2C added in the FW tiles and either SiC or V structure being used.

TABLE III. TBR FOR FLIBE WITH F Be_2C ADDED IN FW TILES

	0% Be_2C	20% Be_2C	30% Be_2C	40% Be_2C
FS	0.983	1.007	1.034	1.061
V	1.043	1.075	1.104	1.131
SiC	1.030	1.094	1.134	1.175

B. *Preferred Design Option*

The high surface heat flux (~0.4 MW/m^2) and volumetric heating (~4 W/cm^3) in the FW tiles require a coolant with good heat removal capability. Among the coolants considered here, liquid Na is the best with liquid Li a close second and Flibe being the worst. With its low melting point and light weight, liquid Na is the preferred option for cooling the FW tiles but adds the complication of having two coolants in the power cycle. To avoid the complexity of having two coolants in the power cycle, it is preferred to cool the FW tiles with the same liquid breeder used in the blanket. While both Li and LiPb can provide adequate TBR, Li is the preferred option due to its better heat removal capability, light weight leading to less pumping power, and no need for enrichment. The main issue is the safety concern that can be mitigated by using He cooling in the shield and VV. The choice of structural material depends on compatibility with Li. While V and SiC yield better TBR and can operate at higher temperatures than FS, they are more expensive, require more R&D and compatibility with Li could limit their operating temperature. The current preferred design option utilizes liquid Li breeder that serves also as coolant for both the FW tiles and blanket. FS is used as structural material.

V. TEMPERATURE DISTRIBUTION IN TILES

Nuclear heating and the surface heat flux were calculated for use in thermal analysis. The reference HAPL target yield is 367.1 MJ. For a repetition rate of 5 Hz, this corresponds to a total fusion power of 1836 MW. The surface heat load and neutron wall loading distributions were determined at the inner surface of the cylindrical chamber. These distributions peak at the mid-plane and centers of the chamber's top and bottom. The peak surface heat flux is 0.37 MW/m^2 and the peak neutron wall loading is 1.09 MW/m^2. The average surface heat flux values are 0.26 MW/m^2 on the side and 0.22 MW/m^2 at top/bottom. The average neutron wall loading is 0.77 MW/m^2 on the side and 0.64 MW/m^2 on the top/bottom. Fig. 6 gives the radial variation of nuclear heating at mid-plane in the different constituent materials for the preferred design option. Nuclear heating results scale linearly with the neutron wall loading. The nuclear heating results along with the surface heat flux and neutron wall loading distributions were used to determine the temperature distribution in the FW tiles at mid-plane (Fig. 7) and for average conditions (Fig. 8). The maximum temperature in the 3-D carbon fiber composite is ~1300°C.

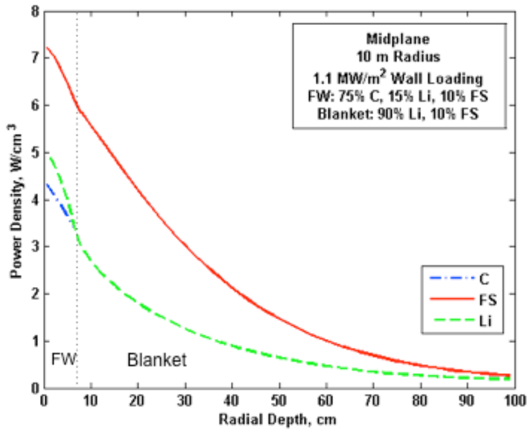

Figure 6. Nuclear heating distribution at mid-plane for the preferred design.

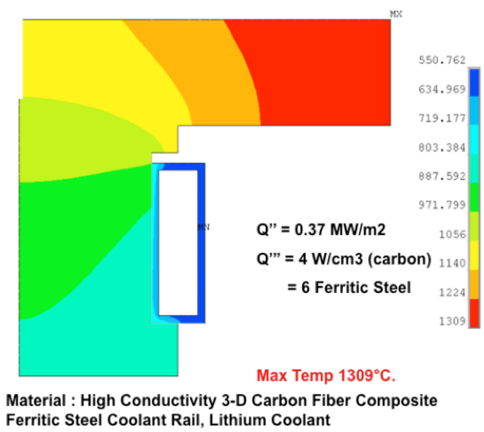

Figure 7. Temperature distribution in FW tile at mid-plane.

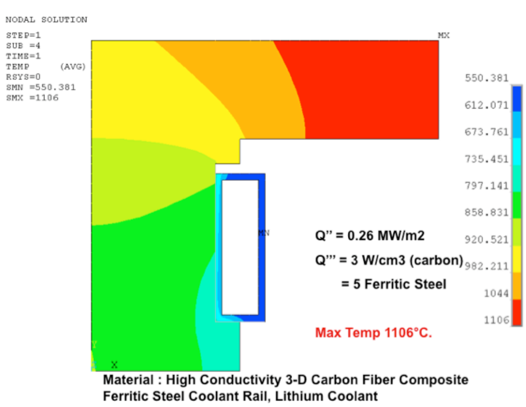

Figure 8. Temperature distribution at average FW tile conditions.

VI. Conclusions

Using mobile FW tiles that are periodically removed, annealed, and reinstalled, tritium retention and surface erosion concerns for inertial fusion first walls could be mitigated. A conceptual configuration has been developed with consideration for laser beam port accommodation and a simple tile insertion and removal scheme. Tritium self-sufficiency can be achieved with a variety of options employing FW mobile tiles. Using ceramic breeders or Flibe requires at least 30% Be_2C added in the FW tiles and blanket to ensure tritium self-sufficiency. This added complication makes these options unattractive. On the other hand, using liquid Li or LiPb as breeder allows achieving tritium self-sufficiency. While liquid Na has the best heat removal capability for FW tiles, it adds the complexity of having two coolants in the power cycle. For this reason, it is preferred that either liquid Li or LiPb be used also to cool the FW tiles. Li is the preferred breeder/coolant due to better heat removal capability, lighter weight, and no need for enrichment. The choice of structural material depends primarily on compatibility with Li. The current preferred design option utilizes liquid Li breeder that serves also as coolant for both the FW tiles and blanket with ferritic steel structural material. Thermal analysis for the carbon fiber composite FW tiles in the HAPL nuclear environment indicates that the maximum temperature will be ~1300°C.

Acknowledgment

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