DESIGN AND ANALYSIS OF THE PROMETHEUS WETTED WALL IFE REACTOR CAVITY *

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The design and engineering of a wetted wall cavity is in progress at UCLA as a part of an Inertial Fusion Energy (IFE) reactor study led by McDonnell Douglas. In this paper, we outline the design methodology, analyze the thermal, mechanical, nuclear and energy conversion attributes of the cavity, and present possible solutions to outstanding problems.

Introduction

A multi-institutional team led by McDonnell Douglas has undertaken a study which includes the design of both a laser-driven (KrF) and heavy-ion-driven (HI) commercial fusion power reactor. A more general paper describing the overall reactor designs can be found in this proceedings [1]. The cavity, which is the subject of this paper, involves the components surrounding the pellet reaction chamber, including the first wall, blanket, coolant manifolding and shield.

The choice of a wetted wall cavity was made following an assessment of previous IFE cavity designs and evaluation of a number of key design goals, including:

- Safety and environmental attractiveness
- Reliability
- Ease of maintenance
- High thermal cycle efficiency
- Long component lifetime

Wetted walls have many potential engineering advantages, including good beam line accommodation, relaxed repetition rate limitations (as compared with thick films), flexible engineering features, and low inventory and flow rate of the liquid film. The choice of materials was strongly influenced by the desire to maximize safety and environmental attractiveness. Engineering choices were made to provide the most robust design possible using only moderate extrapolation of existing technologies.

The cavity design features the use of a thin Pb film to withstand the high transient heat loads from the microexplosions. Pb has desirable temperature limits and, being a neutron multiplier, is very useful in attaining the required breeding ratio with little or no Be in the blanket. The film is fed through a thin, porous SiC composite structure designed with high flexural stiffness. Pb coolant removes the heat to the first wall, which constitutes approximately 40% of the total reactor power. A separate blanket is provided behind the wall protection system for tritium breeding and energy conversion for the remainder of the reactor power. The blanket uses Li_2O breeder with SiC structure, He coolant and a separate He purge.

Design of the present wetted wall concept considers several important issues, including: (1) limits on cavity clearing time due to the requirement to conduct heat out radially, (2) film flow uniformity, wetting, and drainage, and (3) mechanical response of the first wall system. The remainder of this paper gives an overview of the design, describes the key issues, and concludes with an evaluation of the concept and its R&D requirements.

Description of the Wetted Wall Cavity Design Concept

Selection of Materials

Selection of materials in the cavity determines to a large extent the engineering performance attributes as well as the safety and environmental characteristics of the reactor. Structures in the first wall system and blanket are all made from SiC/SiC composite. These materials are currently commercially available, have low long-term activation, high temperature capability, and are resistant to radiation damage.

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Pb is used for the first wall (FW) protectant and coolant. The FW coolant must have acceptable neutronic properties (either breed, multiply neutrons, or be transparent), such that the choices are limited to Li-bearing materials and neutron multipliers. Pb was selected for a number of reasons. It has a safety advantage over Li, good neutron multiplication, and chemical compatibility with SiC. It's thermophysical properties provide good operating temperature ranges. Its relatively high saturation temperature leads to good conduction heat transfer into the coolant, its boiling point is not too high for materials temperature limits and compatibility, and the relatively high bulk coolant temperature gives good thermal conversion efficiency. Bi and BiPb were considered as alternate multipliers, but they have much higher radioactivity. Major disadvantages of Pb include high density and activation.

Li_2O breeder was chosen due to its low chemical reactivity (as compared with Li), low activation, good temperature window, low tritium inventory, and good existing data base from MFE. Of the solid breeder candidates, it offers the best TBR. With 5 cm of Pb in the first wall system, Be is not required in the blanket, eliminating its associated concerns.

Overall Configuration

Figure 1 shows an overall perspective view of the cavity. The cylindrical shape allows easier maintenance through vertical paths, with independent removal of FW and blanket modules. The cylindrical shape also leads to better control of film flow and is more consistent with conventional plant layouts than a spherical shape. The main disadvantage is the nonuniform power distribution, which leads to higher peak loadings. Upper and lower hemispherical end caps are needed to close the cavity. Since they are located farther from the blast, the environmental conditions are less severe. In addition to their contribution to tritium breeding, these regions are used for film injection and removal and for maintenance access (top).

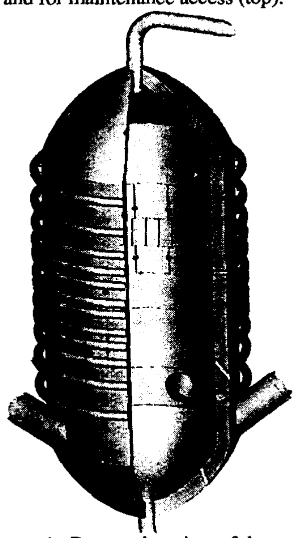

Figure 1. Perspective view of the cavity

The blanket consists of 6 rings through the cylindrical section and separate hemispheres at the top and bottom. Blanket modules are preassembled into the rings, which stack vertically on top of one another. The first wall system consists of individual plates which are locked into a support system attached to the blanket. The ability to provide removable panels which lock into the blanket is essential to allow more frequent maintenance of the first wall panels and still mitigate the mechanical effects of the blast by absorbing the loads into the blanket and support system.

Cavity components are maintained by removing the upper end cap. The first wall panels can then be removed separately, or the entire blanket rings can be lifted. Preliminary analysis suggests that the first wall service life is of the order of 2–3 years, whereas the blanket might last considerably longer.

Figure 2 shows the radial build from the first wall through the shield. Relatively large manifolding is needed behind the blanket to keep the He coolant pressure drop low. The manifolding is made from SiC composite up to the vacuum vessel and shield, where a transition is made to more conventional steel.

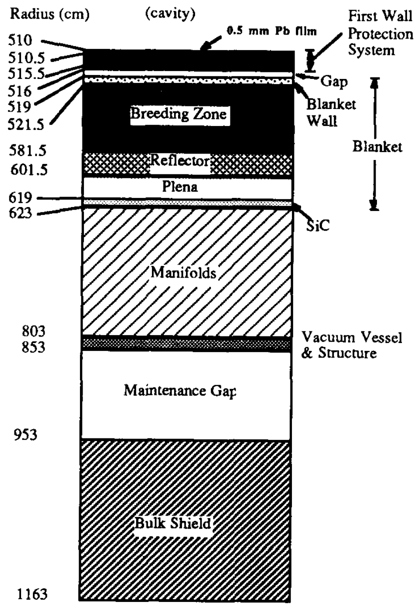

Figure 2. Radial build of the cavity

First wall protection

Configuration. The first wall system consists of SiC/SiC composite made from woven SiC fabric and CVD-deposited bulk SiC. The porosity of the bulk SiC is tailored to allow Pb from the cooling channels to slowly seep onto the surface facing the pellet explosions.

Figure 3 shows a cross sectional view of the first wall system. The film is nominally 0.5 mm in thickness. The cooling channels are shaped to provide high Pb volume fraction. The Pb temperature rise is maximized to reduce the flow rate. The Pb must be kept well above the melting point, but low enough to avoid compatibility limits with the steel heat transport loop; this led to inlet and outlet temperatures of 375 and 525°C. The resulting inlet/outlet pressures and pumping power are 2.0/1.5 MPa and 12 MW, respectively.

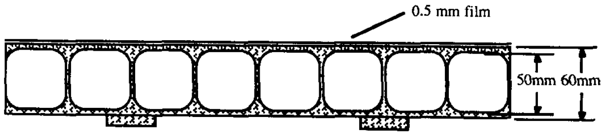

Figure 3. First wall system, cross sectional view

An aspect of this design which differentiates it from MFE designs is the large amount of energy deposited in the first wall system. This results from the conversion of neutron energy to x-ray and debris energy in the pellet, and the large amount of neutron power deposited in the thick Pb coolant. Table 1 summarizes the power balance.

Table 1. Power Balance

	Power (MW)
Total Fusion Power	2819
Surface Heating	798
Neutron Power	2020
Net Energy Multiplication	1.14
Nuclear Heating	
First Wall	486
Blanket	1775
Shield & Vacuum Vessel	41
Total Thermal Power	3100

Film Flow. The film coating the surface is created by coolant seeping through the porous, fibrous wall. The film supply is mainly driven by the pressure difference between the main Pb coolant and the cavity. To obtain proper flow conditions, the reference design has a mean fiber size of 25 microns and porosity of 7.5%. In the upper hemisphere, the protective film is initialized by an injector and maintained at an uniform thickness by continuous flow seeping through the porous wall.

The governing equations for the film flow were simplified by assuming that the velocity normal to the wall is much smaller than the velocity component parallel to the wall for a thin film, and then solved numerically for the film height distribution and film velocity. The film is driven by gravity, which is then balanced by the drag along the wall and the flow acceleration. For the film thickness of interest (0.5 mm), the flow is found to be turbulent. The resulting film thickness is found to vary between 0.4~0.6 mm, which is considered acceptable. More details on the film calculations are given in [2].

Mechanical Response. The first wall panels are locked into the blanket, such that the surface of the first wall acts as a stiffened plate subjected to ablation and pressure loading from the pellet explosion. Calculations were performed to estimate the stresses and deformation of a clamped plate representing the first wall. The mechanical loading on the plate results from two mechanisms: (1) surface ablation momentum as a result of the early deposition of x-rays and (2) pressure loading caused by evaporation of liquid lead. Ablation momentum is introduced into the mechanical model as an initial velocity (~0.03 m/s) for the plate's surface, and the pressure loading is computed from a separate one-dimensional hydrodynamic model for vapor flow in the cavity. The governing equations for compressible flow (i.e., time-dependent momentum, continuity and energy) are numerically solved using an equation of state relating the pressure to the temperature and density. The solution is explained in more detail in [2].

The fundamental frequency of the FW/B surface is of the order of 850 Hz. This high frequency is a direct result of the relatively small dimensions of the clamped plate. It is generally desirable to obtain such high frequencies so that resonance coupling with the hydrodynamic loading does not occur. Maximum displacements of the FW/B surface are of the order of 0.1 mm, and maximum stresses of about 20 MPa occur at the corners of the plate. Since the ultimate tensile strength of SEP SiC/SiC fiber-reinforced composite is about 200 MPa, these results may be interpreted as giving adequate fatigue life. The endurance limit of SEP SiC/SiC is not known, but may be of the order of 0.1–0.2 times the ultimate strength. More accurate fatigue crack growth analysis must be performed before final conclusions can be made.

Cavity Clearing. Energy released from target microexplosions results in evaporation of substantial mass from the wall protection scheme. This mass must recondense or be evacuated in order to provide a sufficiently low pressure in the cavity gas to allow penetration of the laser or HI beams and the targets. For the laser driver, a guideline of ~1 mTorr for the Pb vapor is suggested. The main issue for cavity clearing is whether we can ensure that this low pressure indeed will be obtained. In addition to the Pb vapor, a few grams of noncondensible gas remain in the cavity from He and unburned D and T. This mass must be evacuated through the vacuum pumping system.

To help understand cavity clearing following the explosions, a heat and mass transfer computer model was developed. The model computes the energy deposition profile due to the blast x-rays and debris, and solves the time-dependent heat and mass transfer rates in the cavity and film [2].

For this target, the x-ray yield fraction in is small (~6%) and the spectrum is soft. This causes most of the blast energy to be absorbed in the 1 mTorr cavity gas. Approximately 3 kg of Pb are evaporated by direct energy deposition. The initial (averaged) cavity vapor pressure and temperature are estimated as 49 kPa and 550,000°K respectively. A much larger amount of Pb is evaporated due to rapid radiation cooling of the cavity vapor. Before the recondensation phase begins, about 80 kg of Pb (10 microns) is evaporated.

Based on this analysis, the cavity pressure drops below 1 mTorr before the next shot (at 0.25 s). The SiC temperature is always well below the compatibility limit with Pb (~1000°C). The surface temperature drops very quickly immediately following the blast and asymptotically falls towards the bulk coolant temperature. The ultimate cavity vapor pressure is very sensitive to this slow conduction-dominated phase of the process because the vapor pressure is strongly temperature-dependent. By maintaining a thin first wall and low coolant temperature, adequate recondensation can be expected (within the scope of this simplified model). More details of the analysis are given in [2].

Blanket

The approach for developing the PROMETHEUS blanket design was strongly based on considerations of safety and reliability. Safety considerations led to the choice of a helium-cooled solid breeder blanket with low activation materials: SiC structure and neutron reflector, and Li_2O breeder. Use of Li_2O in conjunction with the first wall Pb coolant provided the potential for adequate tritium breeding without the need for Be as a multiplier.

The blanket configuration consists of 6 rings vertically stacked around the cavity chamber. Each ring contains a number of modules placed circumferentially next to one another. At each laser beam penetration hole, the corresponding module length is shortened to allow for penetration space. A schematic of a module is shown in Figure 4. It consists of a simple layered configuration. The module is made of SiC and contains a number of U-bend woven SiC tube sheets inside which the pressurized He coolant flows. The Li_2O is placed in packed bed form between the tube sheets and is purged by He flowing along the axis of the module.

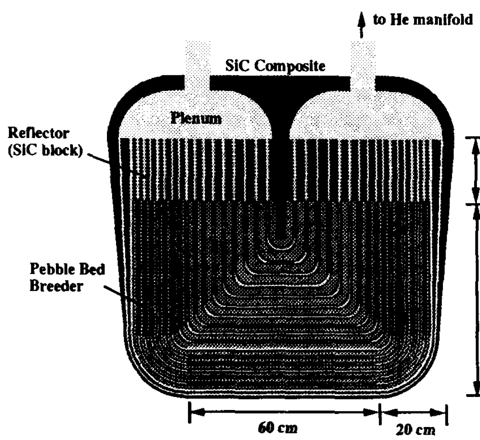

Figure 4. Schematic of a blanket module

Neutronics. A series of parametric neutron/gamma ray transport calculations were performed in a 1-D spherical model to optimize the design of the first wall and blanket systems for larger nuclear power multiplication (M) and tritium breeding ratio (TBR). Factors varied in this parametric study were the ^6Li enrichment, diameter of the Pb cooling channels, and the thickness of the breeding zone. The optimal value of ^6Li enrichment was shown to be ~25%.

The power multiplication is less sensitive to design variations as compared to the TBR, which is most sensitive to the thickness of the Pb zone. When the Pb thickness was varied from 1 to 7 cm, the TBR increases by ~30%, M increases by 5%, and the fraction of nuclear power deposited in the FWS increases from ~0.1 to ~0.25.

The breeding zone thickness was varied from 40 to 100 cm while keeping the Pb cooling channel diameter fixed at 5 cm. In this case, M decreases by ~2%, while the TBR increases by ~9% (from 1.13 to 1.23). The fraction of nuclear power deposited in the FWS (~21%) is insensitive to the breeder thickness, whereas the power deposited in the blanket increases by ~7%. In the reference design, with a Pb zone thickness of 5 cm and breeding zone thickness of 60 cm, M~1.14 and TBR~1.2.

A study was made to determine the advantages of including Be in the blanket. It was found that replacing Li_2O in the first 8 cm of the breeding zone with beryllium (same radial build) increases the TBR by ~7% while M increases by ~5%.

Thermal-Hydraulics. The first wall protection system absorbs virtually all the energy from the x-rays and debris resulting from the micro-explosion as well as a significant fraction of the neutron energy. This results in less than 60% of the total thermal power being deposited in the blanket, which opens the possibility of flowing the He at moderate pressure while keeping the pressure drop and velocity at reasonable levels.

A parametric study was done to explore this possibility. The He pressure drop in the blanket was computed as a function of the fractional flow area for different inlet pressures and fixed inlet and outlet temperatures of 400 and 650°C respectively. For a flow area of 8% (which corresponds to 16% of the total blanket area, since space has to be provided for both inlet and outlet flows) and He velocity of 60 m/s, the blanket pressure drop is only 33 kPa for an inlet pressure of 1.5 MPa. This is a key feature of this blanket since previous designs made use of He at high pressure (5 MPa or more). A moderate pressure of 1.5 MPa would result in improved safety and reliability. The total loop pressure drop is estimated at ~85 kPa.

Thermo-Mechanics. As an added safety feature, the module dimensions are chosen so that each module can independently withstand the coolant pressure of 1.5 MPa in the unlikely event of a SiC tube sheet leaking or rupturing. This would in effect provide double coolant pressure boundaries. The side wall and front wall of the module were approximated as a rectangular plate of length 60 cm with all edges built-in. The plate can be considered as a 2.5-cm SiC tube sheet, each tube being 0.5 cm in diameter. The thickness of the front side of the plate is 1 cm. The corresponding temperature rise was estimated at about 35°C, resulting in a thermal stress of about 18 MPa.

The maximum bending stress in the plate was estimated to be ~117 MPa for a coolant pressure of 1.5 MPa and plate dimensions of 60 cm x 25 cm x 2 cm. This results in a total stress of 135 MPa, which is acceptable. Development of SiC composite with higher tensile strength would relax the dimensional constraints.

From this analysis, inclusion of a U-shape strengthening structure at about every 25 cm along the length of the module would result in the module withstanding the 1.5 MPa coolant pressure in case of leak or failure.

Tritium Analysis. The tritium inventory in the solid breeder was estimated using the MISTRAL code [3]. The breeder temperature is kept in the range from 425–800°C, the purge pressure is 0.2 MPa, and 0.1% H_2 is added to assist in release. Under these conditions, the inventory in the blanket is estimated as less than 10 g. Tritium permeation through the SiC tube sheet wall from the purge to the coolant was estimated based on a diffusion coefficient of 3 x 10^{-19} m^2/s at 600°C for β-SiC [4]. The tritium permeation flux was found to be so small as to be negligible.

Shielding

The radiation shield for PROMETHEUS has been designed to protect components and personnel. It consists of 1. a bulk shield circumscribing the blanket, 2. penetration shield around the laser beam lines and vacuum ducts, and 3. biological shield, which also serves as the reactor building wall. The system has been designed to achieve the following goals: (a) the biological dose rate outside the reactor building during operation is below 2.5 mrem/hr, (b) neutron-induced activation in all components outside the blanket but inside the reactor building (e.g., heat transport system and steam generators) is minimized, and (c) the biological dose rate inside the reactor building but outside the blanket decays to <2.5 mrem/hr within 48 hours after shutdown in order to permit personnel access, if needed, although the reactor system is designed for fully-remote maintenance operations.

Tradeoff studies were carried out to select shield materials and thicknesses. The minimum total cost occurs when the larger part of the shield is placed "close-in" surrounding the blanket and penetrations, rather than at the reactor building. Therefore, imposing personnel access criterion (c) does not increase the capital cost. The increase in the cost of the bulk and penetration shields is compensated by a large reduction in the building cost. Pending further analysis, the shield specifications are as follows:

- The thickness of the bulk shield is ~2.1 m with a material composition of 30% water, 20% B_4C, 20% Pb, and 30% SiC (or TiH_2).
- The structural material in the shield is an aluminum alloy.
- The shield is cooled by low-temperature water.
- The thickness of the laser beamline shield varies with location of the beam line (the penetration opening is location-dependent), but generally is ~1.5 times the diameter of the opening at the first wall.

Safety and Environmental Assessment

Safety and environmental attractiveness are currently recognized as crucial attributes for fusion in order to play a competitive role in the future energy mix. In this study, these issues have played a major role in all of the design decisions which were made. The primary goal associated with cavity safety is to design a cavity which is close to inherently safe. The approaches include: (1) to select low-activation materials; (2) to minimize corrosion and chemical interaction; (3) to promote tritium release; and (4) to minimize and control tritium permeation. The safety concerns include source term characterization (of activation products), accident tolerance, normal radioactive effluents, occupational exposure and waste management.

Radioactive materials existing in the cavity include: (1) tritium; (2) activated debris from burned pellets; (3) activated structural and shielding material in first wall, blanket, reflector and radiation shield; and (4) activated coolant. The tritium inventory within the blanket is estimated as less than 10 grams, which is small. The permeation of tritium from the coolant through the walls of the steam generator represents a difficult problem. It is expected that the inclusion of coolant tritium processing system and use of a double-walled steam generator would reduce tritium permeation rate.

The induced activation of the first wall and blanket systems were computed after 2 years of full power operation using the DKRICF code. In the first wall system, at 1 hour or more following shutdown, the use of Pb results in a considerable inventory compared to the blanket system. The highest activity of 5×10^8 Ci at shutdown is due to ^{203}Pb having a half-life of 52 hours. Care should be given to bismuth impurity control, removal and safe storage of ^{210}Po from the Pb coolant. In the blanket, the highest activity of 1.2×10^9 Ci at shutdown is due to ^{28}Al having a half-life of 2.24 min.

The decay heat level in the first wall system at 1 hr or later after shutdown comes mainly from lead radionuclides. If an adiabatic condition is assumed, the temperature of the FW system reaches 1000 °C at ~9 hours after shutdown. This calculation indicates that an active cooling system is required for any containment of lead radionuclides.

Conclusions

By adopting a set of design goals from the start of the design process, the cavity design has many attractive features, as summarized below:

1. Maximum safety and minimum environmental impact. Low activation materials were selected whenever possible, including SiC structures, Li_2O breeder, and He coolant. Some structures outside the cavity (e.g., vacuum vessel and final mirrors) employ low-activation ferritic steel. The use of Li_2O leads to low tritium inventory in the cavity. Li an LiPb were rejected due to chemical reactivity. While Pb has known concerns with radioactivity, the inventory is low, and means to minimize this problem (e.g. using recycling) are addressed. The use of Pb also eliminates the need for Be multiplier.

2. High level of reliability. By separating the functions of the first wall and blanket, the blanket can be made simple, using relatively conventional approaches. The blanket He coolant can be maintained at very low pressure (1.5 MPa) due to the relatively low power density in the blanket, allowing reduced pressure stresses and leakage concerns. The main tradeoff is with pumping power and plant efficiency. The simple overall configuration and use of flow control at the first wall also lend confidence to the "robustness" of the design.

3. Easy maintenance. The cylindrical cavity configuration was selected to allow simple vertical maintenance paths and compatibility with the remaining engineering systems. Maintenance is accomplished by removing the upper end cap and then removing the first wall panels or the blanket rings independently.

4. High thermal cycle efficiency. Choice of He blanket coolant, Pb FW coolant and SiC structure allows high temperature operation, resulting in high thermal cycle efficiency. The pumping power for Pb is fairly low; the pumping power for He is based on a tradeoff between lower pressure for higher reliability and higher pressure for higher net efficiency.

5. Long lifetime. The use of radiation damage resistant structures allows us the possibility of blankets which are lifetime or near-lifetime components. The first wall system is likely to require changeout every 1–2 years, but the maintenance scheme adopted allow rapid changeout with relatively small impact on availability.

As with other IFE cavity designs, several critical issues exist due to lack of data or understanding of the phenomena. A complete list is not provided here; some of the more critical issues and the uncertainties include:

1. Cavity clearing
   - Accuracy of radiation model and effect on recondensation
   - Nucleate recondensation and aerosol transport
   - Droplet generation and behavior
   - 3-D effects
2. Film flow
   - Wetting characteristics of Pb on SiC composite
   - Flow on inverted surfaces
   - Uniform coverage of engineering structures
   - Stability of the film subjected to pressure impulses
   - Integrity of SiC composite in the pulsed fusion environment
   - Consequences of dry spots
3. Mechanical response

The issues have been identified and addressed, but much further R&D will be required to provide confidence in this design.

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