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OVERVIEW OF FUSION REACTOR BLANKET DESIGNS AND THEIR ASSOCIATED TECHNOLOGICAL ISSUES

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ABSTRACT

Among the reactor components classified as Fusion Nuclear Technology, the critical path to the development of a DEMO reactor is determined by the blanket. A large variety of blanket options have been advanced over the years, and currently there are several designs advocated for testing in ITER by different members of the world fusion community. In this paper, the definition of Fusion Nuclear Technology and the functions of fusion reactor blankets are reviewed. Then different blanket design concepts are summarized and the issues associated with their development, and eventual ITER testing, are explored.

I. INTRODUCTION

A complete fusion reactor is comprised of many different systems, utilizing manifold technologies. For the deuterium/tritium fuel cycle, on which first generation fusion reactors will operate, the technology to accommodate and fully utilize the high energy (14 MeV) neutrons is referred to as *fusion nuclear technology* (FNT). FNT is defined, then, as the technology necessary to simultaneously accomplish the following functions:

1) convert the fusion energy into heat, and to efficiently extract this heat and convert it to a useful product,
2) produce, extract and recycle tritium so as to close the fuel cycle,
3) provide the vacuum boundary for the plasma-containing chamber,
4) provide radiation protection to components, personnel, and public.

The reactor systems that are included under FNT are outlined in Table 1. Of these different FNT systems, this paper provides an overview of the blanket component.

In a tokamak, the most heavily researched reactor type in magnetic confinement fusion, the blanket is physically located directly behind the toroidal first wall (FW), which is often integrally attached. The function of the blanket is to

Table 1. Fusion Nuclear Technology Components and Other Components Affected by the Nuclear Environment

1. Blanket/ First Wall* Components
2. Plasma Interactive and High Heat Flux Components -Divertor, limiter -rf antennas, launchers, wave guides
3. Shield Components
4. Tritium Processing Systems
5. Instrumentation and Control Systems
6. Remote Maintenance Components
7. Heat Transport and Power Conversion Systems

*The blanket determines the critical path to fusion nuclear technology development

Table 2. Worldwide Blanket Options for DEMO

Breeder	Coolant	Structural Material
A. Solid Breeders Li_2O, Li_4SiO_4, Li_2ZrO_3, Li_2TiO_3	He or H_2O	FS†, V alloy, SiC Composites
B. Self Cooled Liquid Metal Breeders Li, LiPb	Li, LiPb	FS, V alloy with Electric Insulator, SiC Composites with LiPb only
C. Separately Cooled Liquid Metal Breeders Li LiPb	He He or H_2O	FS, V alloy FS, V alloy, SiC Composites

* almost all concepts use beryllium as neutron multiplier
† FS = Ferritic Steel

capture the fusion neutrons which, due to their electric neutrality, are not confined magnetically in the plasma. The neutrons give up their energy to the blanket materials in a manner suitable for conversion to electricity through some standard power cycle. But additionally, the blanket must utilize the captured neutrons to breed tritium by a nuclear reaction with lithium. The blanket must breed tritium *in excess* of that consumed in the plasma and that lost through decay or trapped in reactor materials, and the tritium must be extractable during the operation of the reactor. Also, it is desirable to minimize the amount of residual radioactivity so that the blanket components are inherently safe and disposal is not problematic.

The goal of the current world fusion program is the development of a Demonstration Reactor (DEMO), which has all the technologies required of a functioning electric power plant. It is envisioned that the International Thermonuclear Experimental Reactor (ITER), a collaborative, multinational tokamak that will reach plasma ignition, will provide a test bed on which different DEMO blanket concepts can be tested. This paper is intended for professionals who are not well acquainted with the fusion field. Section II presents several of the existing design concepts for blankets, and Section III discusses the main issues involved with developing these conceptual blanket designs into functioning fusion components.

II. BLANKET DESIGNS

Among FNT components, blankets determine the critical path to DEMO. Many different designs for fusion reactor blankets have been advanced[1,2] and advocated through the years. The primary blanket options presently being considered worldwide as candidates for a DEMO reactor are summarized in Table 2. These can be classified into: a) solid breeders, b) self-cooled liquid metal breeders, and c) separately cooled liquid metal breeders. Both helium and pressurized water are considered as coolants for solid breeders. Two types of liquid metals are being considered, lithium and lithium-lead. Only three classes of structural materials are presently considered as candidates for DEMO and commercial reactors: martensitic or ferritic steels, V alloys, and SiC composites. The different ITER partners have each developed designs which they favor for testing in ITER, but these concepts by no means cover all the different designs that are possible. However, a look at these designs will provide an understanding of the current focus of the world fusion community, as well as provide insight into the issues and problems associated with both solid breeder and liquid breeder blankets.

A. Self-Cooled Lithium/Vanadium (Li/V) Blanket

The Li/V blanket, currently advocated by the United States[3] and the Russian Federation, uses liquid lithium, flowing in a vanadium structure, as both a breeder and coolant (see Figure 1). The Li flows down in the channels nearest the FW, where the most severe neutron deposition and plasma surface heating occurs, then turns at the bottom of the segment and flows back up in the rear channels to collection headers at the top of the module. The Li is heated in the process to a temperature suitable for energy conversion. The Li then exits the reactor where the heat is transferred to a secondary loop, and the bred tritium is removed and eventually recycled to the plasma. The concept relies on the use of a self-

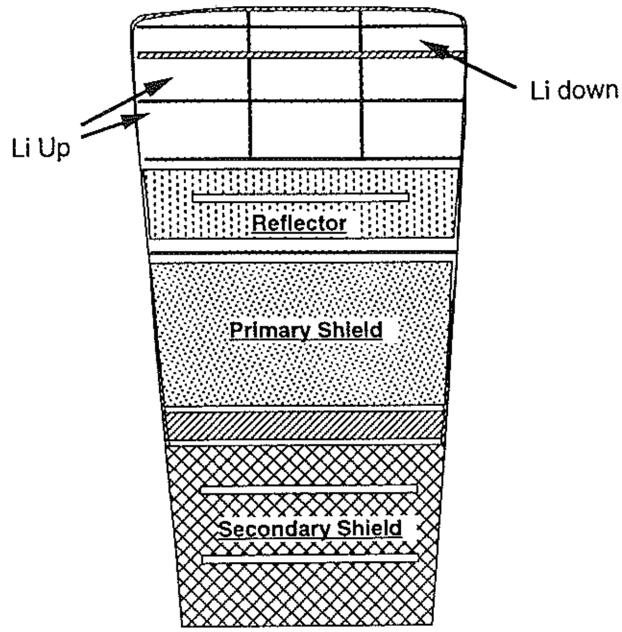

Figure 1: Cross-section of a Li/V breeder blanket module

healing electrical insulator coating on the inside surface of the vanadium channels. This coating electrically decouples the moving, electrically conducting, liquid metal from the vanadium structure, and so reduces the magnetohydrodynamic (MHD) pressure drop to a manageable level (~1 MPa).

Li is the preferred choice as the breeder/coolant because of its good tritium breeding capability, good thermal conductivity and low mass density. Vanadium is chosen as the structural material because of its superior thermal and mechanical properties at high temperature, good compatibility with Li and its low activation in a neutron environment.

B. Water-Cooled Pb-17Li Blanket

This concept is one of the two developed by the European Union (EU)[4], and utilizes a nearly stagnant pool of the liquid eutectic $Pb_{83}Li_{17}$ (abbreviated Pb-17Li, MP = 235°C) as the breeder, cooled by high pressure water flowing poloidally in double-walled U-tubes. The tubes carries water down the front of the pool, turn at the bottom of the module, and return to the top in the back of the module (see Figure 2). The liquid Pb-17Li is confined by a steel box divided into channels by radial and toroidal stiffener plates, with circulation of the alloy counter to the direction of the water flow in the double-walled tubes. The FW is cooled by water in an independent coolant loop with collection headers at the back of the module.

The Pb-17Li is flowed at a rate fast enough to allow efficient tritium removal from the liquid breeder, without excessive tritium permeation into the water coolant, but slow enough to avoid excessive MHD effects. The Pb-17Li breeder has a low tritium solubility so the need for a tritium permeation barrier between the breeder and the water coolant is necessary for this concept. This breeder alloy also contains lead which functions as a neutron multiplier, and has a significantly reduced reactivity with water as compared to pure Li.

The selected structural material is DIN 1.4914 martensitic steel (MANET), which has the advantage of acceptable strength at high neutron fluence and high temperature.

C. Helium-Cooled Pebble Bed Blanket

This solid breeder blanket, also under development in the EU[5] (the US, Russians and Japanese have similar designs), utilizes Li_4SiO_4 ceramic in pebble bed form as the breeder material with Be pebbles as a neutron multiplier. The pebble beds are layered vertically with coolant channel plates separating each layer, as seen in Figure 3. The structural box and the coolant plates are again made from MANET. The coolant plates also serve as stiffeners to support the box structure in

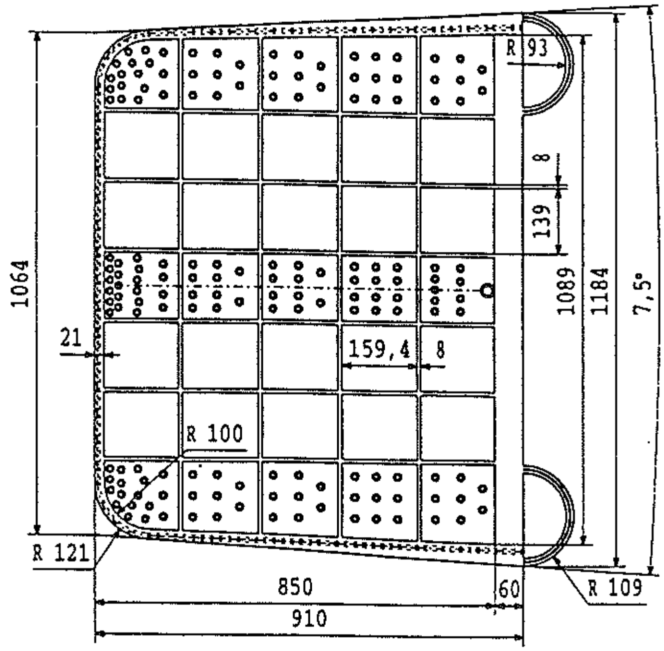

Figure 2. Equatorial cross-section of a water-cooled Pb-17Li blanket segment

the event of box pressurization due to a He leak or large electromagnetic loads due to a plasma current disruption. The box exterior, including the FW, is cooled by an independent He circuit, with all collection/distribution headers located at the rear of the module.

The solid breeder designs require the presence of a separate neutron multiplier in order to achieve acceptable tritium breeding characteristics. Tritium bred in the Li_4SiO_4 is removed from the blanket *in situ* by a separate He purge stream at low pressure, which circulates through the pebble bed and carries out diffusing tritium to a processing unit.

D. Water-Cooled Pebble Bed Blanket

This concept, developed for ITER testing by the Japanese[6], uses Li_2O ceramic breeder and Be multiplier again in the pebble bed form (see Figure 4). In this design, though, the pebble beds are layered radially and are cooled by pressurized

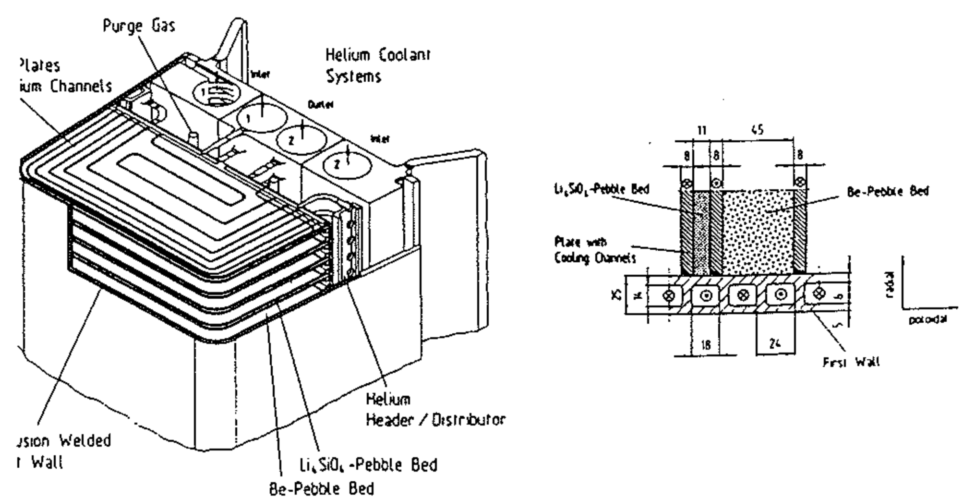

Figure 3: Perspective view of EU Helium-cooled pebble bed DEMO blanket design

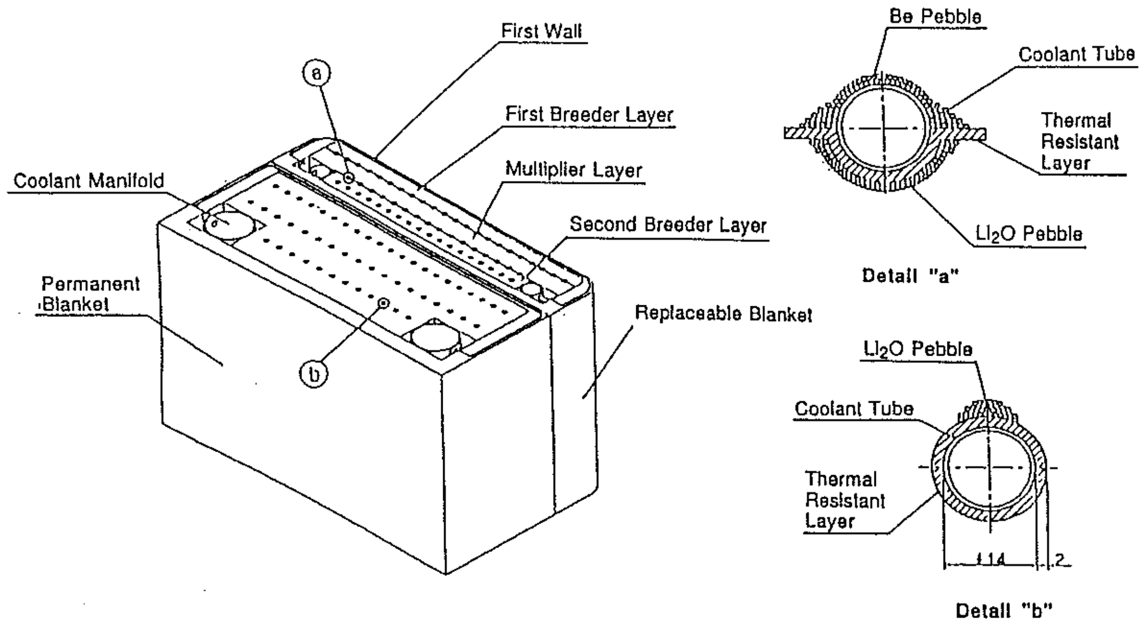

Figure 4: Water-cooled Li_2O DEMO blanket test article

water flowing in coolant tubes through the pebble beds. The front section of the blanket is made replaceable due to the higher Li burn-up in this region. The rear permanent blanket is composed of Li_2O only, with no Be multiplier. The maximum temperature limits of water require the placement of a thermal resistant layer between the breeder and water tubes, to insure that the breeder will operate at a temperature sufficient for tritium release into the separate He purge stream flowing through the pebble bed.

The structural material for the confining box and coolant tubes is Japanese F-82H ferritic steel. Due to use of Li_2O as the breeder, much less Be is required to achieve an acceptable tritium breeding.

III. ISSUES ASSOCIATED WITH BLANKET DEVELOPMENT

All of the different designs cited above have technical issues that must be resolved before the concepts can be utilized successfully in a DEMO reactor. The blanket test program in ITER is designed to help address some of these issues, and to allow the blanket concepts to be evaluated based on real performance in a fusion device.

Fusion nuclear technology testing include feasibility issues and attractiveness issues. Feasibility issues are those whose negative resolution will have the following impact:

a. may close the design window,
b. may result in unacceptable safety risk,
c. may result in unacceptable reliability, availability or lifetime.

Attractiveness issues are those whose negative resolution will have the following impact:

a. reduced system performance,
b. reduced component lifetime,
c. increased system cost,
d. less desirable safety or environmental implications.

A summary of the testing issues for the blanket/first wall system is shown in Table 3. Many issues are common to all types of blankets. Examples are tritium self-sufficiency, allowable operating temperatures, reliability and failure modes, effects and rates. However, the specific details of all the issues are different. A very brief summary of key issues for different types of blankets is given below. A summary of the critical issues of fusion nuclear technology, which stresses the key functional aspects of the fusion reactor that must be resolved through testing, is given in Table 4.

Table 3. List of Blanket/First Wall Testing Issues

A. Structure
  1. Changes in properties and behavior of materials
  2. Deformation and/or breach of components
     a. Effect of first-wall heat flux and cycling on fatigue or crack growth-related failure
     b. Magnetic forces within the structure (including disruptions)
     c. Premature failure at welds and discontinuities
     d. Failures due to hot spots
     e. Interaction of primary and secondary stresses and deformation
     f. Effect of swelling, creep, and thermal gradients on stress concentrations (e.g., in grooved surfaces)
     g. Failure due to shutdown residual stress
     h. Interaction between surface effects and first-wall failures
     i. Self-welding of similar and dissimilar metals
  3. Tritium permeation through the structure
     a. Effectiveness of tritium permeation barriers
     b. Effect of radiation on tritium permeation
  4. Structural activation product inventory and volatility
  5. Hermiticity of SiC

B. Coolant
  1. MHD pressure drop and pressure stresses
  2. MHD and geometric effects on flow distribution
  3. MHD insulating coating fabrication, integrity, and in-situ self-healing
  4. Stability/kinetics of tritium oxidation in the coolant
  5. Helium bubble formation leading to hot spots
  6. Coolant/purge stream containment and leakage
  7. Activation products in Pb-Li
  8. Liquid metal purification

C. Breeder and Purge
  1. Tritium recovery and inventory in solid breeder materials
  2. Liquid breeder tritium extraction
  3. Temperature limits and variability in solid breeder materials
     a. Temperature limits
     b. Thermal conductivity changes under irradiation
     c. Effect of cracking
     d. Effect of LiOT mass transfer
  4. Breeder behavior at high burn-up/high dpa

D. Coolant/structure interactions
  1. Mechanical and materials interactions
     a. Corrosion
     b. Mechanical wear and fatigue from flow-induced vibrations
     c. Failure of coolant wall due to stress corrosion cracking
     d. Failure of coolant wall due to liquid metal embrittlement
  2. Thermal Interactions
     a. MHD effects on first-wall cooling and hot spots
     b. Response to cooling system transients
     c. Flow sensitivity to dimensional changes
  3. Coolant/Coatings/Structure interactions

E. Solid breeder/multiplier/structure interactions
  1. Solid breeder mechanical and materials interactions
     a. Clad corrosion from breeder burnup products
     b. Strain accommodation by creep and plastic flow
     c. Swelling driving force
     d. Stress concentrations at cracks and discontinuities
     e. Thermal expansion driving force
  2. Neutron multiplier mechanical interactions
     a. Beryllium swelling (Swelling driving force in Be)
     b. Strain accommodation by creep in beryllium
     c. Mechanical integrity of unclad beryllium
  3. Thermal interactions
     a. Breeder/structure and multiplier/structure interface heat transfer (gap conductance)

F. General blanket
  1. D-T fuel self-sufficiency
     a. Uncertainties in achievable breeding ratio
     b. Uncertainties in required breeding ratio
  2. Tritium permeation
     a. Permeation from breeder to blanket coolant
     b. Permeation from beryllium to coolant
     c. Permeation characteristics at low pressure
  3. Chemical reactions
  4. Tritium inventory
  5. Failure modes and frequencies
  6. Nuclear heating rate predictions
  7. Time constant for magnetic field penetration for plasma control
  8. Blanket response to near blanket failures
  9. Assembly and fabrication of blankets
  10. Recycling of irradiated lithium and beryllium
  11. Prediction and control of normal effluents associated with fluid radioactivity
  12. Liquid-metal blanket insulator fabrication, effectiveness, and lifetime
  13. Tritium trapping in beryllium

Table 4. Summary of Critical R&D Issues for Fusion Nuclear Technology

1.	D-T fuel cycle **self sufficiency**
2.	**Thermomechanical** loading and response of blanket components under normal and off-normal operation
3.	Materials **compatibility**
4.	Identification and characterization of **failure modes, effects, and rates**
5.	Effect of imperfections in electric (MHD) **insulators** in self cooled liquid metal blanket under thermal/mechanical/electrical/nuclear loading
6.	**Tritium inventory** and recovery in the solid breeder under actual operating conditions
7.	**Tritium permeation** and inventory in the structure
8.	Radiation Shielding: accuracy of prediction and quantification of radiation production requirements
9.	Plasma-facing component thermomechanical response and lifetime
10.	**Lifetime** of first wall and blanket components
11.	Remote maintenance with acceptable machine shutdown time.

A. Solid Breeder Blanket Issues

For solid breeder blankets, the major classes of issues include:

- Tritium self-sufficiency
- Breeder/multiplier/structure interactive effects under nuclear heating and irradiation
- Tritium inventory, recovery and control; development of tritium permeation barriers
- Thermal control
- Allowable operating temperature window for breeder
- Failure modes, effects, and rates
- Mass transfer
- Temperature limits for structural materials and coolants
- Mechanical loads caused by major plasma disruption
- Response to off-normal conditions

Test modules in ITER will be specifically designed to evaluate and correlate neutronics in a fusion environment, demonstrate effective heat removal and sufficient resistance to thermal and electromagnetic loads, and confirm tritium breeding calculations and in-situ tritium recovery.

B. Liquid Breeder Blanket Issues

For self-cooled liquid metal blankets, the main feasibility issue is the electrical insulation between the flowing liquid metal and the load carrying duct walls. The most attractive solution is the use of insulating coatings on the interior duct surface. Such coating of the structural material is also required, this time as a tritium permeation barrier, in the separately-cooled Pb-17Li blanket concept. Tritium removal from water is costly and difficult, as so must be minimized. For both electrical insulation and tritium permeation barriers, the common issues for coatings are fabrication technology, stability, and long term performance under irradiation in the presence of temperature, stress and electric potential gradients.

The tritium control issue is different for lithium and Pb-17Li. Tritium extraction is a key issue for lithium while tritium permeation is a primary issue for Pb-17Li. Activation of Pb-17Li under neutron irradiation is also a concern, especially in the case of a liquid metal spill, because of the production of the α-emitter ^{210}Po. This may require an on-line bismuth-removal technique. Corrosion and mass transfer is an issue for both Li and Pb-17Li. Because of this temperature limits for the structural material and coolant are important. For the lithium/vanadium the heat transport system outside the blanket must be constructed of a different structural material because vanadium is not economical to use outside the blanket. Interstitial impurity transfer in such bi-metallic loops is a key concern.

The large stored chemical reactivity of Li is a serious issue if water cannot be excluded from the system (possibly needed for divertor or rf antennae cooling). Water-liquid metal interaction is an issue also for water-cooled Pb-17Li blankets, but not to the same extent as with pure Li.

Transient electromagnetics is an issue for liquid metal blankets particularly in the case of plasma disruption. The large electrical currents, which can be induced in a electrically conducting liquid metal, can interact with the magnetic field and lead to large forces and stresses in the blanket.

IV. FINAL REMARKS

In closing it should be stressed that the DEMO blanket concepts and their ITER test article equivalents are still undergoing redefinition and redesign as the R&D progresses and the ITER design and DEMO requirements[7] solidify. A good case in point is the possible prohibition against the use of the current European martensitic steel due to radiation-induced decrease in ductile-to-brittle transition temperature, found after irradiation experiments in fission reactors. It is highly likely that other such changes are in store before the initiation of ITER testing.

I would like to thank the various institutions involved in DEMO blanket development from whom I have borrowed Figures 1-4, and also Drs. N. Morley and A. Ying for their assistance with this document.

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