ANL/TD/CP--89406
CONF-950961--14

MATERIALS FOR BREEDING BLANKETS*

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September 1995

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* Work supported by the U.S. Department of Energy, Office of Fusion Energy, under Contract W-31-109-Eng-38.

Presented at the Seventh International Conference on Fusion Reactor Materials (ICFRM-7), September 25-29, 1995, Obninsk, Russia

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Abstract

There are several candidate concepts for tritium breeding blankets that make use of a number of special materials. These materials can be classified as Primary Blanket Materials, which have the greatest influence in determining the overall design and performance, and Secondary Blanket Materials, which have key functions in the operation of the blanket but are less important in establishing the overall design and performance. The issues associated with the blanket materials are specified and several examples of materials performance are given. Critical data needs are identified.

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1. Introduction

The breeding blanket represents a key component in the development of fusion power. The blanket must provide shielding to protect personnel and sensitive components, provide power conversion of the fusion power to useful heat, and provide tritium fuel production. The blanket should also operate with a high level of system reliability and safety. The job of the designer is to find the optimum combination of materials that best fulfills the goals of the blanket system. The desirable features to be incorporated into any blanket system are specified in Table 1, and the designer generally arrives at a compromise in design based on the trade-offs between different areas as well as the relative importance given to achieving each feature. There are many possible approaches to meeting these goals, and several recent blanket design are illustrated in Figures 1 to 4.

Figure 1 represents the breeding blanket for ITER [1]. The blanket is composed of modules each of which is approximately 1 m2 x 40 cm deep in size and mechanically attached to a 10 cm thick backplate. The structural material is solution annealed Type 316 LN stainless steel, which was selected primarily because of its ease of fabrication and extensive manufacturing base. The breeding material is solid lithium zirconate, and beryllium is used as the neutron multiplier. The assembly is cooled by low temperature water with an inlet temperature of 150 C. The requirements of the system are a net breeding ratio of > 0.8 (with the rest of the tritium fuel supplied by outside sources) and adequate shielding to protect the magnets and allow for rewelding of coolant manifolds. There is no need to provide useful heat removal, and hence the coolant temperature can remain low.

Figure 2 [2] represents an advanced solid breeder blanket which uses ferritic/martensitic steel as the structure, lithium silicate as the breeding material, and beryllium as the neutron multiplier. Both the breeder and multiplier are in the form of pebbles. The coolant is high temperature helium. This blanket, as well as other advanced blankets, is designed to have a tritium breeding ratio > 1 and also to provide useful heat for conversion to electricity. The ferritic/martensitic steel has advantages over the solution annealed Type 316 LN stainless steel with higher strength (to about 550 C), higher capability for surface heat loads, and low radiation swelling.

Figure 3 [3] represents an advanced liquid metal breeding blanket which uses PbLi as the breeding/multiplier material, ferritic/martensitic steel as the structure, and high temperature water (~300 C) as the coolant. The PbLi is slowly circulated to allow for removal of tritium outside of the reactor, in contrast to solid breeder designs where the tritium is removed in-situ by a separate helium purge stream. A key feature of the design is the need for a tritium permeation barrier between the PbLi and water to prevent excessive permeation into the coolant.

Figure 4 [4] represents an advanced self-cooled liquid metal breeding blanket with liquid lithium as the coolant/breeder and V-4Cr-4Ti as the structure. The vanadium alloy has greater capability to accommodate high heat loads than the ferritic/martensitic alloys and can potentially be used to higher temperatures. It also exhibits good resistance to radiation damage with low swelling and high ductility. A key feature of this design is the need for a thin insulator coating on the surface of the vanadium alloy to reduce the MHD pressure drop of the liquid metal flow.

2. Classification of Breeding Blanket Materials and Conditions

The materials used in the blankets shown in Figures 1 to 4 as well as for other concepts can be divided into two general classifications - Primary Blanket Materials and Secondary Blanket Materials. Primary blanket materials are those materials that have the greatest influence in determining the overall design and performance of the blanket system, and they include structural materials, breeder/multiplier materials, special shielding materials, and coolants. These materials will in large part determine the thickness of the blanket, the allowable operating temperatures, the power conversion efficiency, and the system lifetime. The specific materials that are included as primary blanket materials are listed in Table 2.

Secondary blanket materials include those materials that have key functions in the operation of the blanket but are less important in establishing the overall design and performance. These materials include plasma facing materials, joints and welds, coatings, insulators, and mechanical attachments. The limitations of these materials could set operational limits of the entire blanket system even though they are viewed as having secondary roles. The specific materials that are included as secondary blanket materials are also listed in Table 2.

The operating conditions in the fusion power system represent the primary design drivers which, when coupled with the material choices, help establish the operating window for the blanket. The most important operating conditions are the surface particle and heat flux, the neutron flux/fluence/spectrum, and disruptions. The surface fluxes and neutrons determine response during normal operation, and disruption parameters determine response during expected off-normal events. The surface fluxes directly affect the surface erosion lifetime, the material temperature and stress levels, the cooling requirements, and the reactor power density limits. The neutron flux and fluence directly affects the materials radiation damage lifetime, the shielding thickness, the cooling requirements, the tritium breeding capability, the reactor power density limits, and the material activation/afterheat levels. Disruptions affect the surface erosion lifetime and the structural material requirements for electromagnetic forces. Disruption effects are expected to be much more important in near term devices such as ITER since the understanding and controllability of disruptions should be improved in the long term.

The blanket plays a large role in determining the overall attractiveness of fusion power. There are several desirable features that should be incorporated into the blanket which depend upon the choices of materials. A primary feature is safety, which is enhanced by using materials with reduced activation and afterheat and which have inherently low chemical reactivity and toxicity. The lifetime of the materials should be long enough so that the plant availability is not adversely affected. Generally, this translates into a minimum desirable lifetime of 10-20 MW-a/m^2. The lifetime of blanket materials is affected by surface erosion, neutron radiation damage, and mechanical property considerations that include fatigue and crack growth. The short term performance is enhanced by selecting materials that can operate at high temperatures to increase thermal conversion efficiency, that can operate at high power densities, and that minimize the space required for achieving adequate tritium breeding and radiation shielding. It is also desirable that the materials be easily fabricable and have a modest cost.

3. The Influence of Material Properties on Blanket Design

3.1 Radiation Effects

Radiation swelling has been studied extensively in a large number of alloys and other materials. Although it is desirable to keep the amount of swelling to a low level, there are no well formulated rules governing the degree of allowable swelling in the blanket. From the design viewpoint, swelling is not a primary design driver unless the level is excessive. In the BCSS study [5], the combined amount of swelling and radiation creep allowed was 5%. The actual level of dimensional change allowed can be higher or lower than this depending upon the specific design. The main point is that some level of radiation swelling is acceptable. Swelling can influence other areas, however, and therefore must be taken into account. For example, locally high swelling can have the effect of reducing the effective thermal conductivity, which would result in higher operating temperatures and thermal stresses, which in turn could result in degraded mechanical properties and a reduced lifetime.

Radiation creep results in localized dimensional changes, but the deformations are generally considered to be non-damaging, in contrast to thermal creep where the creep deformation can lead to material rupture and component failure. At the operating temperatures of interest, radiation creep occurs at much higher rates than thermal creep. From a design viewpoint, the total acceptable level of radiation creep deformation over the desired lifetime of the component can set the allowable stress level in the structure. In many cases, the allowable stress for radiation creep is lower than the allowable stress established from ASME Design Code criteria. Radiation creep will over time also change the secondary stress distributions. The stresses induced by thermal gradients, material mismatch in thermal expansion coefficients, and differential swelling will tend to relax during operation as a result of radiation creep. This means that the lowest secondary stress will eventually occur during the plasma burn, and the highest secondary stresses will eventually occur during the dwell period.

Radiation will also degrade the mechanical properties. Loss of ductility is probably the greatest concern of the changes in tensile properties. In some alloys, such as 316 Stainless Steel, the uniform elongation can be reduced to essentially zero after irradiation of 5-10 dpa. Even though the total elongation is still adequate, the loss of uniform ductility will result in severe strain localization during plastic deformation leading to component failure. Loss of ductility does not mean that a material has reached the end of its lifetime, but the design rules for the blanket must then be reformulated to insure that the structure remains in the elastic regime at all times during operation. This could result in a reduced level of allowable stress compared with unirradiated material. Radiation damage also degrades the fracture toughness. A high fracture toughness is desirable particularly when the structure experiences its highest impact loads due to the large electromagnetic forces that occur during disruptions.

Fatigue and crack growth represent the most likely failure mode for the blanket. Fatigue and crack growth are influenced by the degree of local ductility and the local environment at the crack tip. Irradiation will tend to decrease the ductility, as mentioned above, and can accelerate stress corrosion cracking (SCC) in a water environment, both of which increase the crack growth rate. Specific areas of the blanket will tend to be particularly vulnerable. These areas include welds with their heat affected zones, coolant side cracks where SCC can occur, joints between different materials where stresses may be high, and notched areas where stresses are intensified.

3.2 Power Density Limits

An attractive fusion device should be capable of operating at high power densities in order to reduce the size of the fusion core which should translate into a reduced cost of electricity. The ability to accommodate high power densities depends upon the thermophysical properties of the blanket materials. In particular, the choice of the structural material will limit the capability of the first wall to accommodate high surface heat fluxes. The power density capability can be quantified by using the thermal stress allowable for surface heat flux:

$$PD\ [(MW/m^2)\bullet mm] = UTS^*k^*(1-\gamma)/(E^*\alpha)$$

where

UTS is the ultimate tensile strength

k is the thermal conductivity

γ is Poisson's ratio

E is Young's modulus

α is the mean coefficient of thermal expansion

A comparison of allowable heat flux is illustrated in Figure 5. The V-4Cr-4Ti alloy exhibits the highest heat load capability up to 800 C compared with SiC, the 9Cr-1Mo ferritic alloy, and 316 LN. The ferritic alloy exhibits relatively high heat load capability at lower temperatures, but it drops off significantly past 500 C as the mechanical strength decreases. Solution annealed 316 LN has the lowest heat load capability primarily due to its low thermal conductivity and high thermal expansion coefficient. SiC has a heat load capability somewhat above 316 LN but below the other materials. The advantage of SiC is at high temperatures, above 800 C where the strength of the other materials is greatly reduced.

3.3 Activation and Afterheat

Activation and afterheat considerations play an important role in the safety of the blanket. Short term values are important for plant accident considerations, and long term activation is important for waste disposal and recycling considerations. Short term radioactivity and afterheat for 316 LN, SiC, 9Cr-1Mo, and V-4Cr-4Ti are shown in Figures 6 and 7 respectively [6]. The radioactivity of SiC and V-4Cr-4Ti are significantly lower in the short term than the iron based alloys. The afterheat of V-4Cr-4Ti is the lowest up to about 300 s after shutdown, after which the SiC exhibits the lowest value. The long term activation is shown in Figure 8 which indicates that SiC has the lowest value up to about 10^9 seconds (about 30 y) after which V-4Cr-4Ti exhibits the lowest value.

4. Solid Breeder and Beryllium Performance Issues

The primary functions of the solid breeder and beryllium are heat deposition and transport, tritium breeding and transport, and nuclear shielding. Nuclear power is deposited in the solid breeder and beryllium, and it is then transported to the cooling system. The properties affecting power deposition and transport are thermal conductivity and neutron energy multiplication. The solid breeder must also have a net tritium breeding ratio greater than one and must easily release the tritium at typical blanket operating temperatures. There are a number of chemical properties that affect tritium release including tritium bulk diffusivity, surface recombination and release, and chemical interaction with the purge gas. Both the solid breeder and beryllium also serve as shielding materials, so that neutronic cross sections and reactions are important. Safety considerations dictate that the level of retained tritium should be low and that the materials should exhibit low activation and afterheat. The materials should also be chemically compatible with the coolant and structural materials. In order to achieve a long lifetime, the solid breeder and beryllium should exhibit good thermal and mechanical stability. In particular, swelling and/or densification as well as cracking and mass transfer should be minimized. The relative ranking of candidate solid breeding materials and beryllium is summarized in Table 3. It is clear that no single breeding material is superior in all categories, and therefore the selection of breeding materials is still in progress.

An important property related to power transport is thermal conductivity. A comparison of calculated thermal conductivities for different materials and densities is illustrated in Figure 9. The thermal conductivity of all solid breeder materials is quite low, and it is reduced even further when the material is used in porous form. The design implication is that the size of breeding zones between coolant channels in the blanket will be quite narrow in order to maintain the temperatures of the breeder within acceptable limits. One of those temperature limits is the minimum allowable temperature which usually set by tritium inventory requirements. Figure 10 illustrates the calculated retained tritium inventory for Li_2O and Li_2ZrO_3. The minimum temperature of Li_2O is set by the point where LiOH forms with the moisture in the purge stream, which in this case is ~370 C. Li_2ZrO_3 can achieve a lower operating temperature since there is no LiOH formation, and for a tritium limit of one day generation, the minimum temperature is reduced to ~230 C.

5.0 Liquid Metal Performance Issues

The primary issues related to the liquid metals are related to tritium transport and containment and MHD effects of flowing liquid metals in high magnetic fields. In both cases, the use of insulator coatings are viewed the means for dealing with these concerns. In the case of $Pb_{83}Li_{17}$, the tritium generated during operation has almost no solubility which results is a high partial pressure of tritium in the system and potentially high permeation through the structure. A thin coating of Al_2O_3 applied to the structure surface can act as a permeation barrier and provide for tritium containment to safe levels. In the case of a self-cooled blanket using pure Li, insulator coatings are to be used to reduce the MHD pressure drop through the system. The leading candidates for insulator coatings are CaO and AlN. There are a number requirements that are common to the insulator coatings. The coatings should be able to be applied to large, complex shapes typical of the blanket, they should be chemically compatible with the blanket environment, and they should if possible be self-healing if cracks should develop. They should also be relatively thin (1-10 μm) and have a similar thermal expansion coefficient as the substrate in order to minimize thermal stresses, and they should be resistant to radiation damage. The tritium permeation barriers should provide a reduction of tritium permeation by about a factor of 100 to meet the containment requirements. The main requirement for the MHD insulator coatings is to meet the electrical resistivity values to prevent currents from flowing in walls of the cooling channels. For blankets in tokamaks this requirement is:

$$\rho^*t > 10^{-2} - 10^{-1}\,\Omega m^2$$

This requirement is quite modest for most insulating materials. For example, for a 10 μm thick coating, unirradiated candidate materials exceed the above requirement by 10^3 to 10^6. Irradiation damage will reduce the resistivity, but it appears that sufficient resistivity can be retained to meet the coating requirements.

6.0 Materials R&D Needs

From a design point of view, there are two types of information that are needed; materials properties under nominal operating conditions, and material behavior at the limits of performance. The former information allows the designer to calculate the blanket behavior during normal operation, and the latter allows the designer to establish an operating window for the materials. It should be noted, that it is desirable that a complete and consistent set of data be established for a specific composition and thermo-mechanical treatment and that the properties be representative of those of as-fabricated structures. It is also desirable that the tensile property data include the actual stress-strain curves, particularly for irradiated material where loss of ductility is an issue. There is a wide range of information that is needed for design, and it is not possible here to provide a comprehensive list of data needs. Instead, critical data needs will be highlighted.

Table 4 outlines the critical data needs. The emphasis in this table is on the secondary blanket materials that can set operating limits for the blanket. Less emphasis is placed on the structural materials, since their data needs are covered more completely in other papers in this conference. There are a few items that are particularly important. The first area is coatings. Both Li and PbLi blankets are dependent on the success of coatings for attractive performance, and this area is at an early stage of development. The second area is joint development. Joints are required in all types of blankets, and they are likely to have properties that are no better and perhaps inferior to the base structural materials. Therefore, the response of the joints to the fusion environment is likely to set the limits of performance. Again, the work here is at an early stage of development. The last area is beryllium. Beryllium is used in all solid breeder designs as a neutron multiplier and it is used in some blankets as a plasma facing material. There is a limited understanding of the effects of radiation in beryllium at typical operating temperatures, particularly for beryllium that is less than 100% theoretical density that could be used as a neutron multiplier and for plasma sprayed beryllium that could be used as a plasma facing material.

7. Conclusions

The blanket represents a system of materials that are chosen to best fulfill the design goals for performance, lifetime, and safety. The job of the designer is to find the optimum combination of materials that best fulfills these goals for the blanket system. Blanket materials can be divided into two general categories, primary blanket materials that set the overall design geometry and performance and secondary blanket materials that have important specific functions that can limit performance.

The materials issues and data needs for fusion blankets has been reviewed. Comparisons of materials and examples of key materials issues have been presented. In general, development of primary materials is relatively more advanced than for secondary materials. R&D need have been identified with special emphasis on secondary materials.

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[5] D. Smith, et al., "Blanket Comparison and Selection Study, Final Report," ANL/FPP-84-1, Argonne National Laboratory (1984).

[6] H. Attaya and D. Smith, "The Effects of Impurities on the Activation of SiC, Vanadium, and Ferritic Alloys," Fusion Materials, Semi-Annual Progress Report for Period Ending March 31, 1995, DOE/ER-0313/18 (1995).

Figure Captions

Figure 10. Predicted Tritium Inventory in an ITER/EPP Driver Blanket Composed of Li_2O or Li_2ZrO_3 with 184 g/day Tritium Generation Rate and T_{max}-T_{min}) = 100°C across the Breeder Layer.

List of Tables

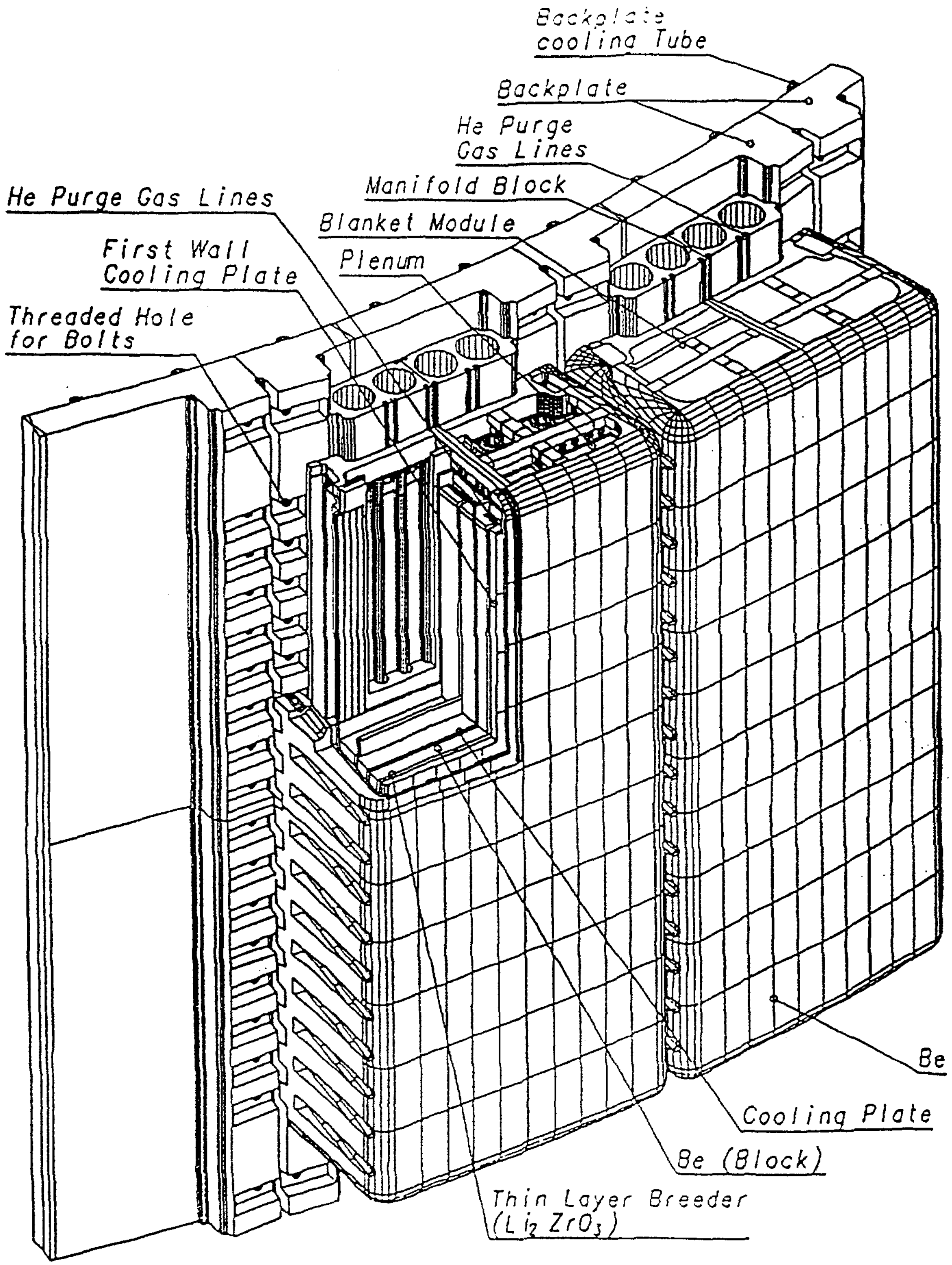

Figure 1 - ITER Inboard Solid Breeder Blanket

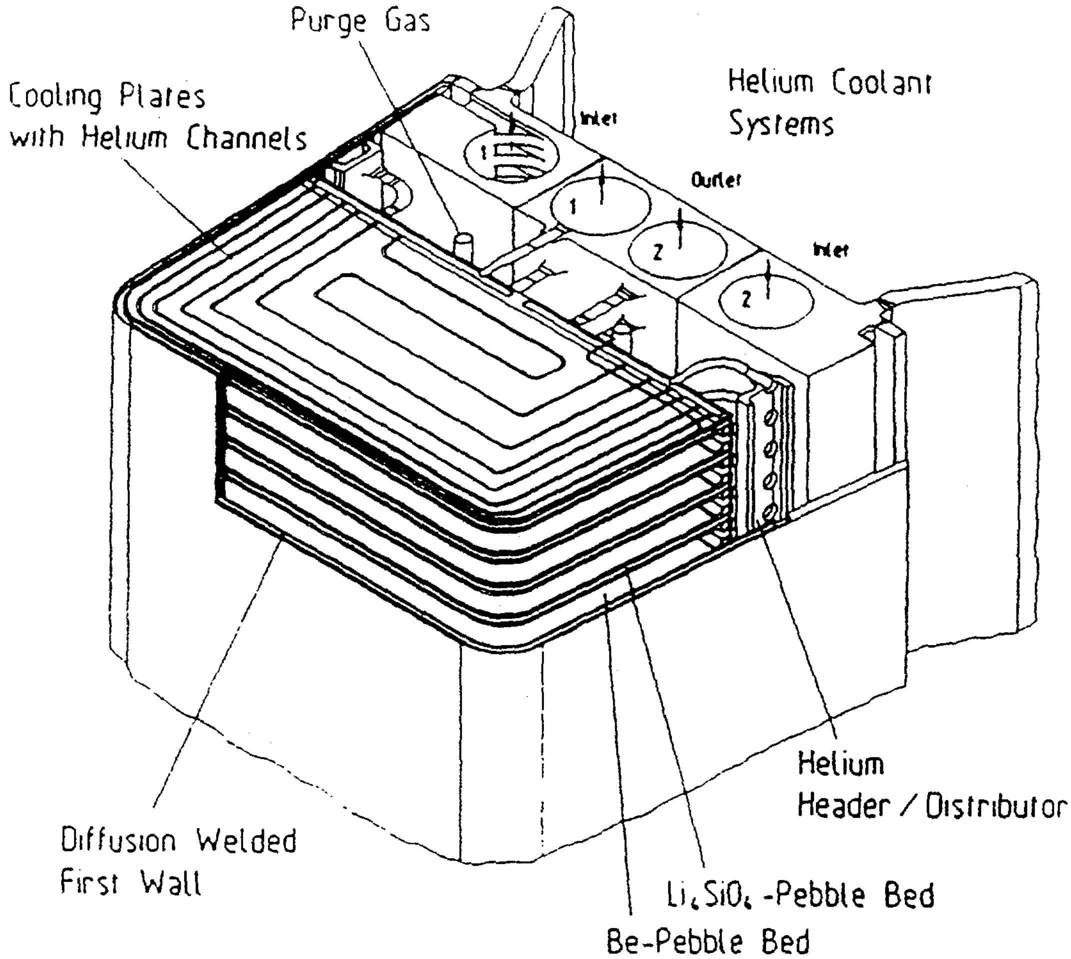

Figure 2 - EU He/Solid Breeder BOT Blanket

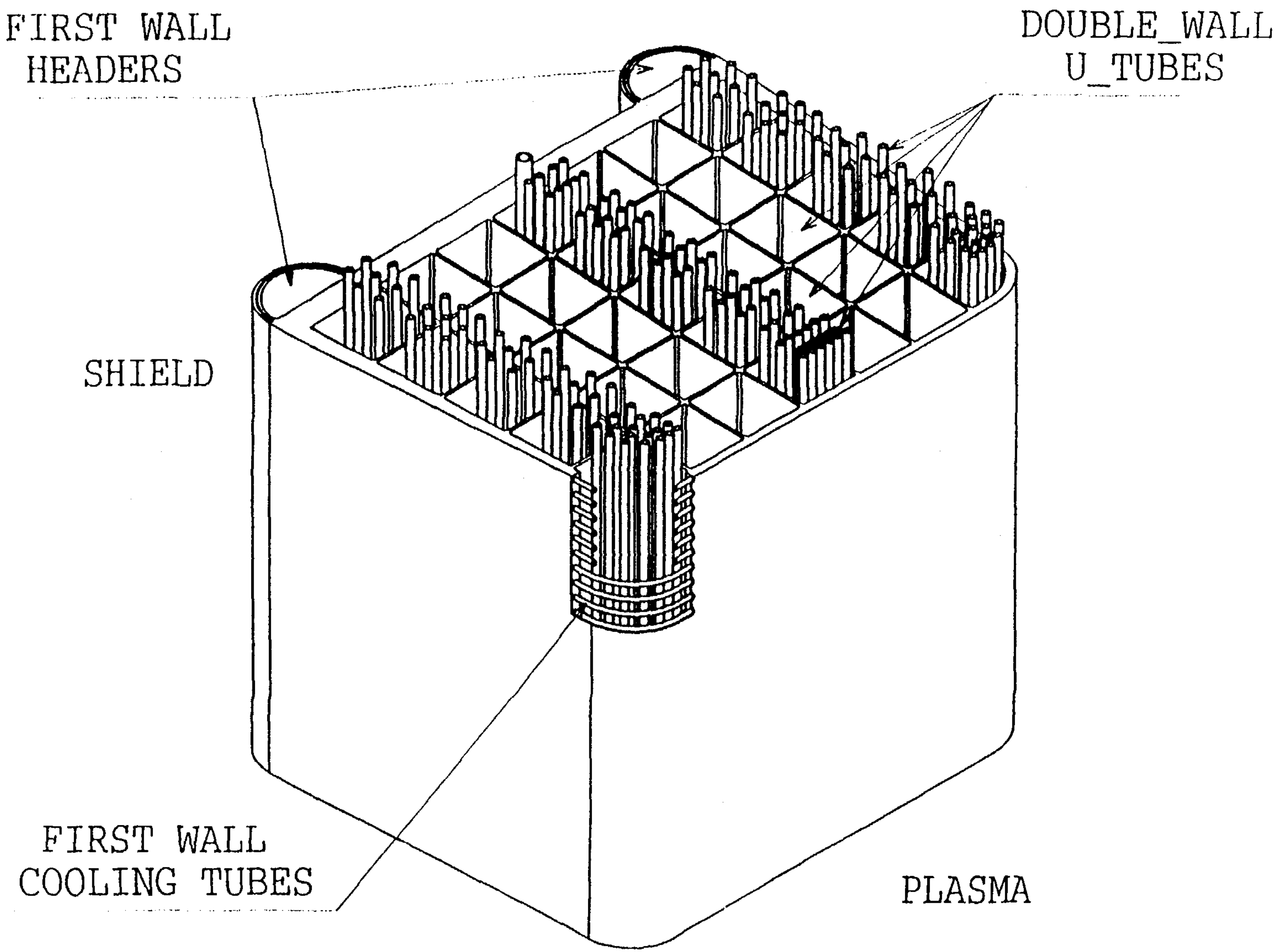

Figure 3 - EU Water/PbLi Blanket

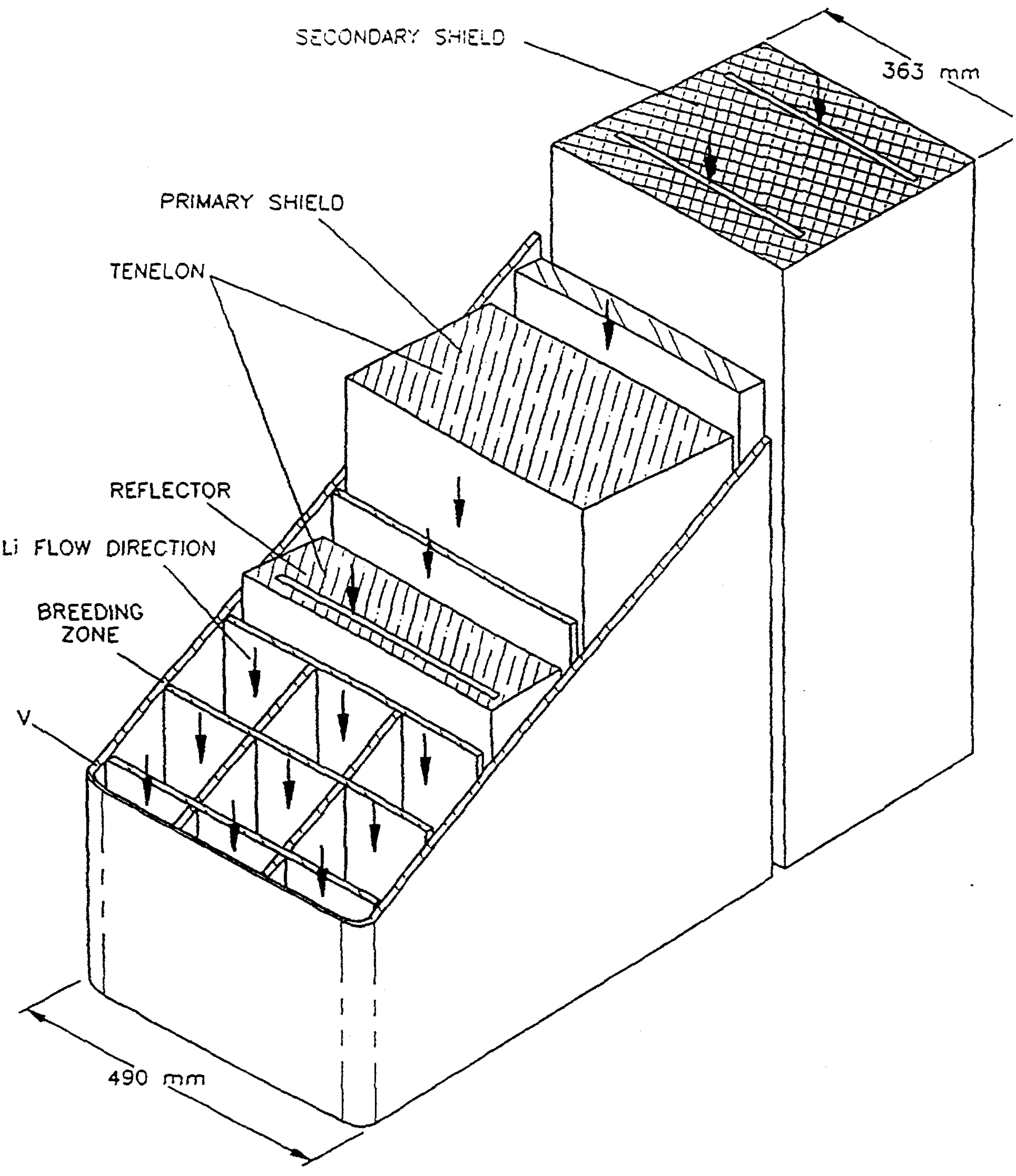

Figure 4 - Cutaway View of Vanadium/Lithium Demo Inboard Blanket/Shield Segment.

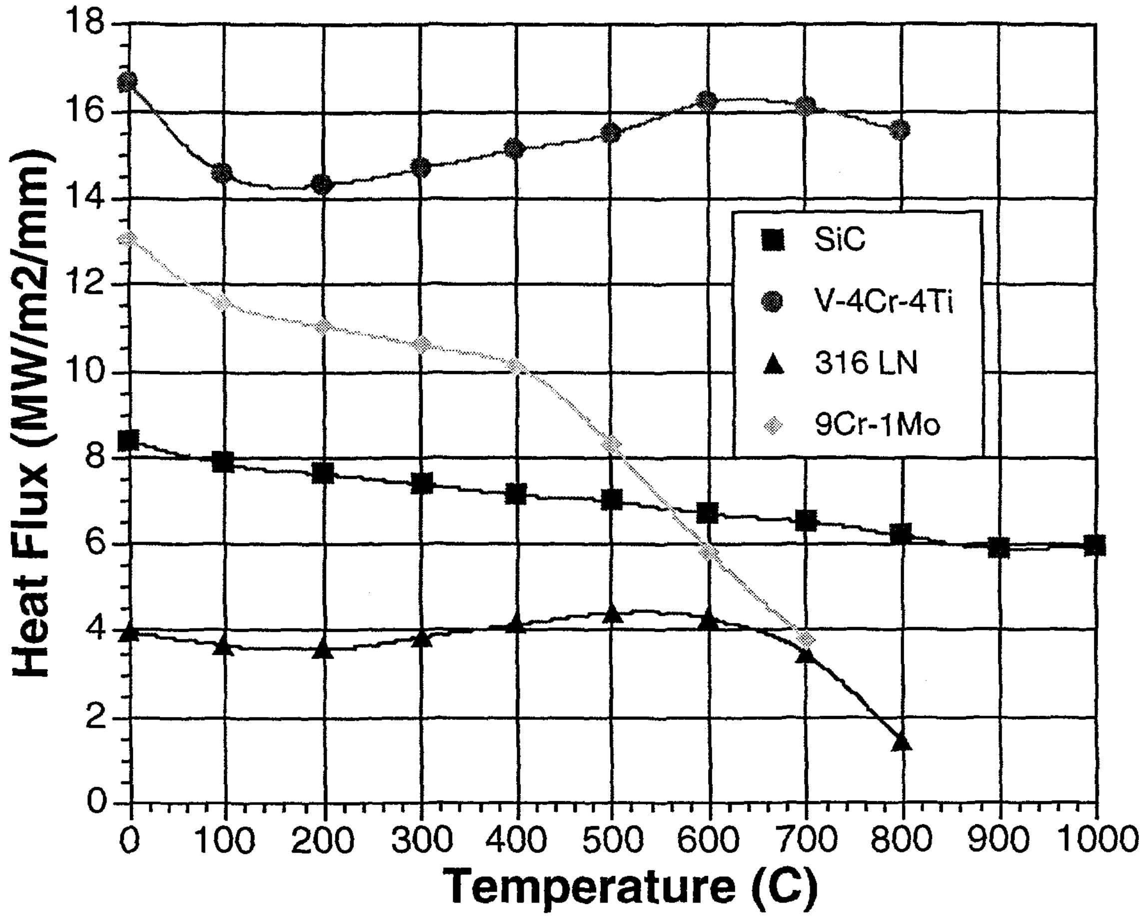

Figure 5 - Structure Material Heat Load Comparison

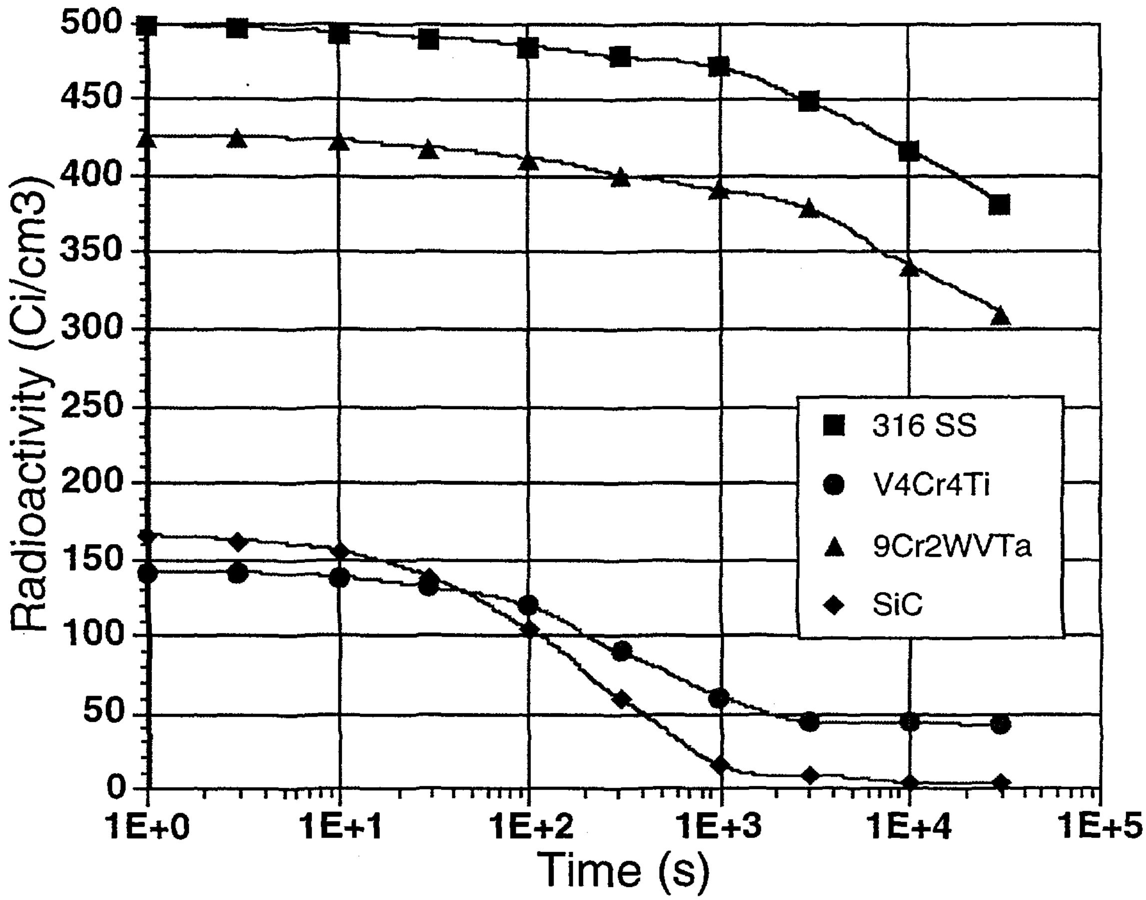

Figure 6. - Short Term Radioactive Inventories

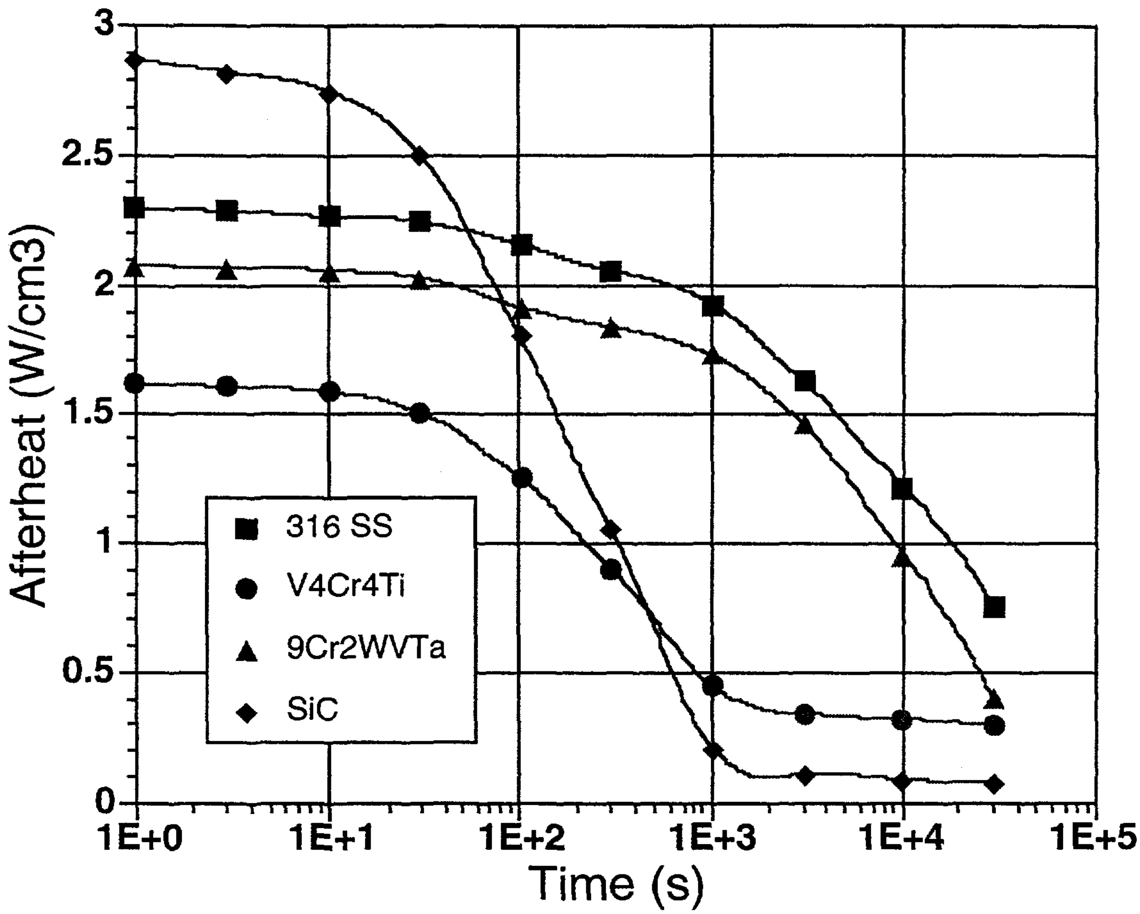

Figure 7 - Short Term Afterheat

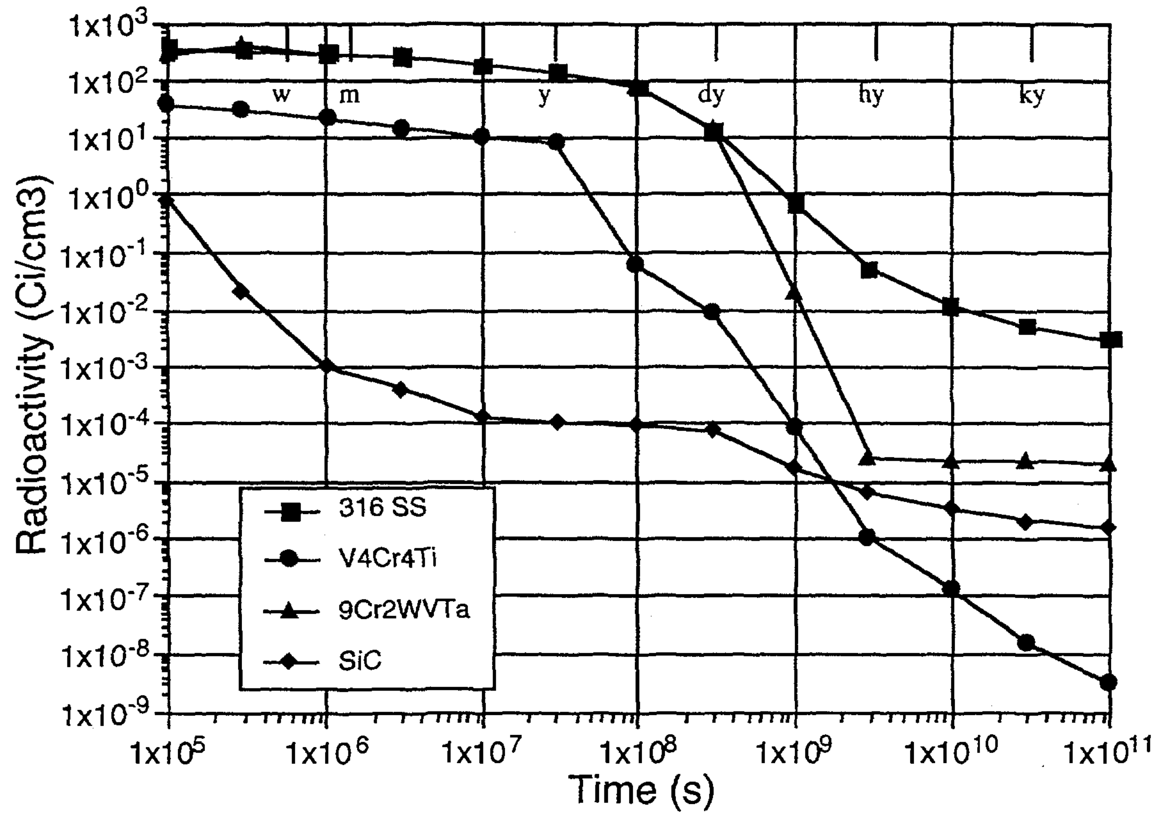

Figure 8 - Long term activation

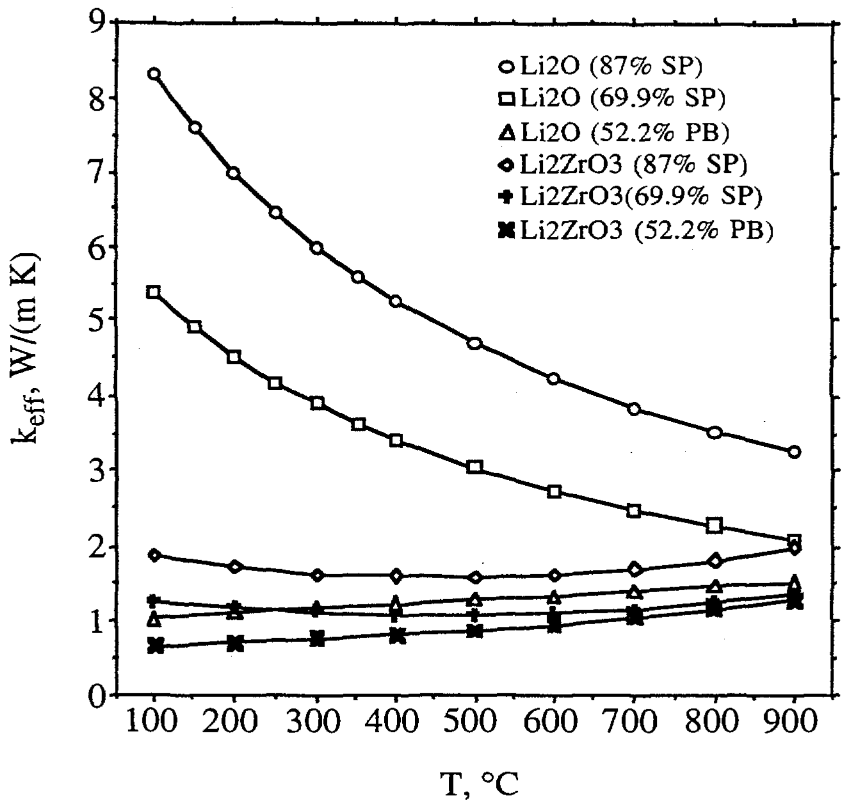

Figure 9. - Thermal conductivity (k_{eff}) of sintered product (SP) and pebble bed (PB) Li_2O and Li_2ZrO_3. Numbers in parenthesis are smear densities.

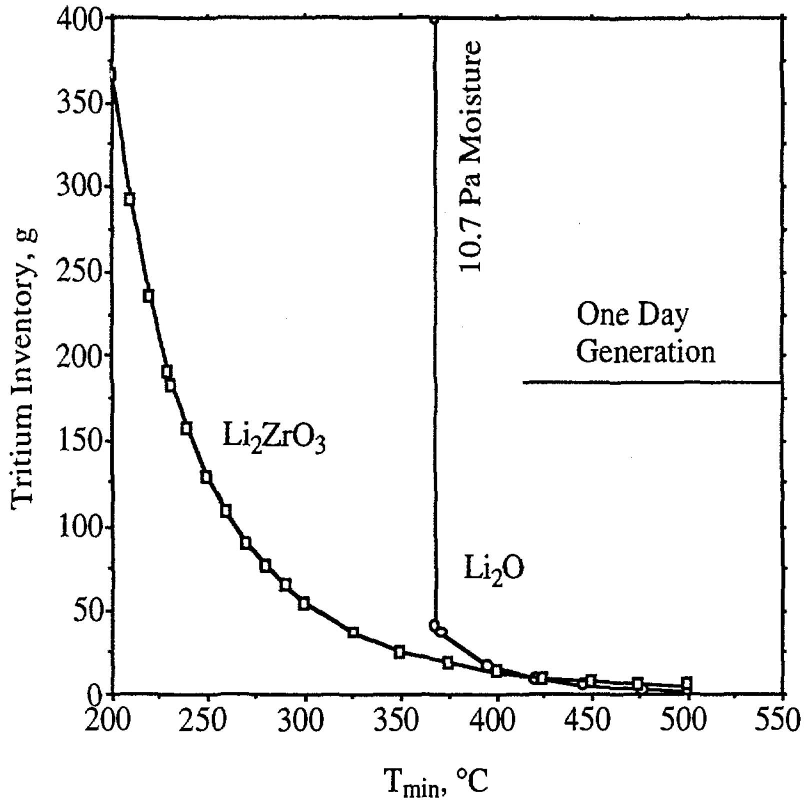

Figure 10 - Predicted tritium inventory in an ITER /EPP driver blanket composed of Li_2O or Li_2ZrO_3 with 184 g/day tritium generation rate and $(T_{max}-T_{min}) = 100°C$ across the breeder layer.

Table 1 - Desirable Features for Blanket materials

- Safety
  - Minimize activation
  - Minimize afterheat
  - Minimize chemical reactivity
  - Minimize chemical toxicity
- Lifetime
  - Maximize surface erosion lifetime
  - Maximize radiation damage lifetime
  - Maximize cyclic fatigue lifetime
  - Maximize reliability
- Performance
  - Maximize thermal conversion efficiency
  - Maximize power density
  - Minimize blanket/shield thickness
- Maintainability
  - Minimize maintenance time
- Fabricability
  - Minimize fabrication difficulty
- Cost
  - Minimize cost

Table 2 - Primary and secondary blanket materials

Primary Material Type	**Candidate Materials**
Structural Materials	Austenitic stainless steel Ferritic/martensitic stainless steel Vanadium alloys Silicon carbide and SiC/SiC composites
Breeder/multiplier	Beryllium Lithium ceramics Liquid Li/Pb-Li
Coolant	Water Helium Liquid Li/Pb-Li
Shield	Structure/coolant/breeder Tungsten Boron Hydrogen bearing compounds

Secondary Material Type	**Candidate Materials**
Plasma Facing Materials	Beryllium Graphite Tungsten
Joints	Welds Brazes Diffusion bonds
Coatings	Tritium barriers MHD insulating coatings Corrosion barriers
Insulators	Disruption current breaks Diagnostics Antennas/waveguide
Mechanical Attachments	Bolts Anti-seizing materials Heat transfer layers

Table 3 - Property Comparison for Solid Breeders and Beryllium

Area	Comparison
Heat Deposition/Transport	
Energy multiplication	$Li_2ZrO_3 > Li_2TiO_3 > Li_2O > LiAlO_2 > Li_4SiO_4 >$ Be
Thermal Conductivity	Be $> Li_2O > LiAlO_2 > Li_2ZrO_3 \sim Li_2TiO_3 > Li_4SiO_4$
Tritium Generation, Release, Retention	
Tritium breeding	$Li_2O > Li_2ZrO_3 > Li_4SiO_4 > Li_2TiO_3 > LiAlO_2 >$ Be
Tritium retention	$Li_2ZrO_3 \sim Li_2TiO_3 < Li_2O < Li_4SiO_4 < LiAlO_2 <<$ Be
	Li_2O is limited by LiOT precipitation (e.g., T > 368°C for 10.7 Pa moisture and T > 328°C for 2.14 Pa moisture) Ternaries: minimum termperature for retention of one-day's generation (287C for $Li_2ZrO_3 \sim Li_2TiO_3$, 390°C for Li_4SiO_4 , 450°C, for $LiAlO_2$)
Shielding, After-Heat, Radioactivity (pure materials)	
Shielding (relatively small impact on shield design)	
After-heat	Be $\sim Li_2O \sim Li_4SiO_4 \sim Li_2TiO_3 < Li_2ZrO_3 \sim LiAlO_2$
Activation	Be $\sim Li_2O \sim Li_4SiO_4 \sim Li_2TiO_3 < Li_2ZrO_3 \sim LiAlO_2$

Table 4 - Critical Material Data Needs

Blanket Material	Property Data Needs
Structural Materials	Radiation creep: V alloys, SiC Helium effects on mechanical properties (high T) Irradiation effects on fatigue/crack growth DBTT in ferritic/martensitic steels Fabrication development: V alloys, SiC
Lithium ceramics	Tritium inventory/release at low temperatures (≤ 300C)
Lithium	Tritium recovery Compatibility
Pb-Li	Tritium permeation Compatibility
Beryllium	Properties in as-fabricated forms (< 100% TD) Helium effects on swelling Tritium retention and release
Water	Radiolosis/recombination in fusion neutron spectrum
Li/Pb-Li as Coolants	Heat transfer characteristics
Plasma Facing Materials	Mechanical properties in as fabricated forms (e.g plasma spray) in irradiated and unirradiated conditions
Joints	Fabrication development - all types Mechanical properties in as fabricated forms Irradiated and unirradiated conditions NDT methods of inspection Rejoining of irradiated structures Helium effects in rewelding
Coatings	Fabrication development in complex geometries Crack formation rates and rehealing kinetics Irradiation effects - RIED -Flux and fluence dependent changes Tritium permeation Compatibility with coolant and structure
Insulators	Irradiation effects on resistivity - RIED - Flux and fluence dependent changes Irradiation effects on mechanical and thermophysical properties
Mechanical Attachments	Irradiation effects on mechanical properties Irradiation creep Stress relaxation Heat transfer capability across interfaces