INTOR FIRST WALL/BLANKET/SHIELD ACTIVITY

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I. Introduction

The main emphasis of the INTOR first wall/blanket/shield (FWBS) during this period has been upon the tritium breeding issues. The objective is to develop a FWBS concept which produces the tritium requirement for INTOR operation and uses a small fraction of the first wall surface area. The FWBS is constrained by the dimensions of the reference design and the protection criteria required for different reactor components. The blanket extrapolation to commercial power reactor conditions and the proper temperature for power extraction have been sacrificed to achieve the highest possible local tritium breeding ratio (TBR). In addition, several other factors that have been considered in the blanket survey study include safety, reliability, lifetime fluence, number of burn cycles, simplicity, cost, and development issues. The implications of different tritium supply scenarios were discussed from the cost and availability for INTOR conditions.

A wide variety of blanket options was explored in a preliminary way to determine feasibility and to see if they can satisfy the INTOR conditions. This survey and related issues are summarized in this report. Also discussed are material design requirements, thermal hydraulic considerations, structure analyses, tritium permeation through the first wall into the coolant, and tritium inventory.

II. Tritium Breeding Issues

The INTOR reference design limits the tritium breeding to the outboard and top sections of the first wall. These sections amount to less than 60% of the total first wall surface area. This constraint requires a local TBR greater than 1.7 to eliminate the need for an external tritium source.

Over the years, lithium and lithium compounds have been considered for tritium generation in fusion reactors. The liquid lithium has an upper theoretical limit of ~ 1.8 tritons per fusion neutron assuming an infinite

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medium. However, several engineering considerations dictate the use of a finite blanket thickness with a significant amount of structure materials to contain the breeder and coolant. In all blanket concepts,[1-3] the calculated tritium breeding ratios are much lower than the upper theoretical limit. For tokamak reactors, the highest reported TBR is less than 1.6 for liquid lithium blankets with steel structure. From this discussion, it appears that the liquid lithium does require external tritium source for tritium fueling unless the inboard section of the reactor is employed to increase the net TBR.

Neutron multipliers can be used to enhance the tritium breeding potential of the liquid lithium blankets. Lithium compounds or combination of materials (excluding fissionable materials) can produce up to 2.7 tritons per fusion neutron. Examination of the nonfissionable elements with significant (n,2n) and (n,3n) cross sections and low neutron absorption indicates that Pb, Bi, Be and Zr have the highest potential for neutron multiplication.[1,4] Table 1 gives a list of the potential candidates along with some relevant parameters. The performance parameters of the different neutron multipliers were compared through the use of different design concepts.[1] The results show that beryllium produces ~ 2.7 neutrons per fusion neutron (n/DTn). For lead is only ~ 1.7 n/DTn. This indicates that blankets with lead neutron multiplier or lithium-lead breeder have about the same tritium breeding potential. Clearly, Be is the only neutron multiplier which has the potential to satisfy INTOR conditions. Table 2 shows the different possible material combinations which have the potential to produce more than 1.7 tritons per fusion neutron.

The main concerns for beryllium are the irradiation swelling caused by helium generation and the tritium inventory due to Be(n,t) reactions. Swelling needs to be accommodated such that the induced stresses in the structural material are not excessive. The recommended approach for INTOR is to use a beryllium with ~ 65 to 85% theoretical density at low temperature $\leq$400°C and interconnected porosity to accommodate swelling. The degradation of thermal conductivity of such a porous beryllium is desirable for solid breeder materials to achieve the minimum temperature required for satisfactory tritium inventory.

The design of FW/B/S without constraint on the coolant outlet temperature provides a significant degree of freedom which eliminates some of the critical feasibility issues addressed in the Blanket Selection and Comparison Study.[4]

Design issues and main characteristics for FW/B/S are discussed for the different blanket categories listed in Table 2.

III. Self-Cooled, Liquid Metal Blankets

The use of the same liquid metal as both tritium breeder and coolant greatly simplifies both materials and design considerations. Coolant-breeder compatibility/reactivity is not a factor, and structure compatibility considerations are less restrictive. Heat transfer requirements are also reduced because a large fraction of the nuclear heating is deposited directly in the breeder-coolant. Lithium or lithium-lead with beryllium provides a high tritium breeding capability, and tritium recovery with relatively low tritium inventory is feasible. However, there are certain design issues related to the use of liquid metals in a fusion reactor. Table 3 gives a summary of the design issues and main characteristics of the four blanket categories considered for INTOR. Compatibility between the coolant and structural material will limit the coolant/structure interface temperature below a certain value which can be accommodated in the INTOR design. Pressure drop of liquid-metal flow through a transverse magnetic field is much higher than that in the absence of a magnetic field. Also, the heat transfer characteristics are degraded because magnetic field suppresses turbulence and natural convection in liquid metal systems.

Two liquid metals; lithium and lithium lead (17Li-83Pb), and two structural materials; austenitic stainless steel (PCA) and ferritic steel (HT-9) are included in this preliminary evaluation. It is assumed that the liquid metal is not used in the inboard section of the reactor. The parameters relevant to the evaluation are listed in Table 4. To determine what parameter range in self-cooled, liquid-metal blankets can operate for the INTOR conditions, preliminary analyses were performed for a simple design, once-through poloidal-flow blanket. It can be observed from Table 4 that the lithium blanket is likely to operate in the temperature range of 400-500°C and pressure range of 1-2 MPa.

From the engineering design point of view, lithium-lead is not a good coolant relative to lithium. This is mainly because lithium lead has relatively (1) poor thermal conductivity, (2) higher melting point, and (3) poor compatibility with structural materials. Thus, the permissible operating temperature window is narrower for lithium-lead. To compensate these effects, lithium-lead velocity has to be higher than that of lithium which will result in a higher pressure drop, even though a lithium-lead blanket may still be feasible, particularly 17Li-83Pb with ferritic steel. In addition, the liquid-metal corrosion of the structural material is a key issue to define the operating temperature for liquid metal. A limited amount of data is available for steel and liquid Li or 17Li-83Pb in forced and thermal convection loops. However, essentially no data exists for liquid-metal corrosion under conditions of forced convection with thermal gradients and magnetic fields. Also, the data for 17Li-83Pb are very limited even under simple test conditions. However, INTOR blankets can be designed with a very conservative temperature limits from the corrosion point of view, by reducing the structure temperature to ~400°C. The other design issues related to liquid metal blankets are highlighted in Table 3.

IV. Helium-Cooled, Liquid Metal Blankets

The design issues for self-cooled liquid metal blankets make a helium-cooled design an attractive option. Helium is one of the better heat transfer media amoung various gasses. It is chemically inert and does not effect the tritium breeding potential of lithium or lithium-lead. Also, using helium as the coolant tends to alleviate the MHD pressure drop problem associated with the self-cooled, liquid metal blankets. Since the liquid metal serves only as the breeder is circulated at relatively low velocities for tritium recovery while helium is pumped at relatively high velocities to remove the heat. The pressure drop associated with the liquid metal flow can be rather small compared to self-cooled, liquid metal blankets. However, the geometrical configuration of helium-cooled, liquid metal blankets will be more complex relative to self-cooled, liquid metal blankets because it requires piping and manifolding systems for two different fluid medias.

A common problem associated with helium-cooled blankets for a power reactor is the large pumping power. This is the result of relatively low specific heat and thermal conductivity of helium. In order to keep the pumping power down to acceptable levels, high gas pressures (50-100 atms) are usually employed to increase helium density in power reactors. Since INTOR is not a power reactor, the pumping power requirement can be relaxed somewhat. Furthermore, since the heat loads (surface and volumetric) of INTOR is more moderate than that of commercial tokamak power reactors, the velocity (or the pressure) of helium can be reduced. Thus, the INTOR blanket can be designed at pressures lower than 50 atmospheres.

The INTOR blanket can also be designed at lower temperatures than that of commerical power reactors. Since INTOR does not have to produce power, the pinch point limit associated with power generation no longer applies. Thus, helium inlet temperatures can be kept at relatively low values (50-100°C) compared to that of commercial power reactors (250-300°C). The modest heat loads of INTOR mentioned previously can also be utilized to reduce the blanket temperature if the designer chooses to do so (instead of using it to reduce the system pressure).

In summary, the helium-cooled, liquid-metal blanket for INTOR can be designed to operate at lower temperatures and pressures than commercial power reactors. The actual temperature and pressure of the blanket depends on specific design configurations. However, other design issues require further analyses as highlighted in Table 3.

V. Water-Cooled, Solid Breeder Blankets

Blankets with pressurized water coolants have been examined for power producing reactors. Water has good materials compatibility data, excellent heat transfer characteristics, and is very low in cost. For INTOR conditions, a water-cooled blanket can be designed with a low temperature (60-80°C) and pressure which will simplify the design and improve the reliability. Also, the water coolant is isolated from the breeder material by several zones to avoid tritium removal from the water which is costly.

Another concern with this design is the inherent incompatibility of Li_2O and H_2O. Thermochemical experiments[3] with Li_2O have demonstrated that Li_2O is very hygroscopic with the by-product being the low-melting (744 K), highly corrosive LiOH. For example, at a moisture partial pressure of only 34 Pa, LiOH will form and precipitate. This is considered a design issue rather than a feasibility issue because, with the non-power INTOR blanket, the H_2O coolant can be physically isolated from the Li_2O breeder.

Lithium oxide is compatible with structural materials such as ferritic steels (e.g., HT-9) and stainless steels (e.g., PCA) provided that its moisture content is very low. Problems are encountered if LiOH is present. For a given moisture content, the corrosion is less severe the lower the temperature. As a general rule, the temperature of the Li_2O/structure interface should be less than 744K.

In this design, the main design issues are the lower temperature limits for solid breeders based on tritium inventory, bulk diffusion, solubility and surface adsorption. Of the three candidates, Li_2O has the highest diffusivity and $LiAlO_2$ has the lowest. However, for Li_2O, the trapping of tritium in the form of LiOT is a major concern at the low temperatures. The temperature limit(3) of >683K and moisture partial pressure limit of <34 Pa is based on LiOT precipitation. In the presence of a reducing environment such as Be and H, it is possible a lower temperature limit can be found.

Because of its extremely low tritium diffusivity, $LiAlO_2$ has problems at low operating temperatures unless the grain size is kept very small (~ 0.2 μm diameter). Such material has been fabricated and found to be quite stable in thermal and neutron environments.(5) Depending on the upper temperature of the breeder and the temperature distribution, local temperatures as low as 620 K can be tolerated. However, surface adsorption and solubility can become dominant even if the diffusive inventory is controlled. Again, the presence of Be and H as a reducing agent should act to mitigate concerns over these other inventory components.

Preliminary data on Li_4SO_4 indicate that it might be the best compromise as a breeding material for INTOR. Its low melting point should present no concern for the non-power INTOR blanket. However, these data need to be examined with the same degree of thoroughness as the Li_2O and $LiAlO_2$ data before a valid comparison can be made.

VI. Helium-Cooled, Solid Breeder Blankets

For the INTOR design, the substitution of the helium coolant for the water coolant with solid breeders produces several design issues requiring further analyses. Table 3 provides a list of these issues. However, the helium coolant permits solid breeders to operate at their proper temperature windows from the tritium inventory point of view by varying the inlet temperature.

VII. Structural Materials

Two structural materials, cold-worked austenitic 316 stainless steel (CW 316 SS/PCA, reference material) and ferritic-martensitic steel (HT-9), are considered as structural materials for the U.S. INTOR blanket designs.

The data base of CW 316 SS and HT-9 has been extensively reviewed in the past,(6) mainly for high-temperature (up to 550°C), long-life (up to 10 MW-yr/m^2) blankets in steady-state fusion reactors. The INTOR blankets, on the other hand, are being designed to operate at relatively low temperature (60-400°C), low fluence (3.25 MW-yr/m^2), and in a pulse mode (~ 10^5 pulses/life) environment. While several materials concerns (e.g., swelling, creep, helium embrittlement, liquid-metal corrosion, etc.) associated with the high-performance, power-generating blankets may be considerably lessened for the INTOR blankets, there are also other issues, and these are addressed in the following.

The data base on austenitic 316 stainless steels is the most extensive among all fusion structural materials. Fabricability and weldability of this class of materials is good, as is their well known water corrosion resistance. Irradiation-induced void swelling is not anticipated to be a problem because (1) the lifetime fluence (30-50 dpa) of the INTOR blanket is considerably less than the swelling incubation fluence (~ 100 dpa) of this material, and (2) the INTOR structural temperature (up to 400°C) is also considerably lower than the peak swelling temperature (~ 525°C) of CW 316 SS. With regard to other irradiation effects, there is an uncertainty related to helium. Although high-temperature (~ 700°C) helium embrittlement of grain boundaries is not expected, preliminary data seem to suggest some helium effects in enhancing swelling at moderate irradiation temperatures.

Below 450°C, irradiation hardening generally raises both yield strength and ultimate tensile strength of CW 316 SS. Also, at this temperature, thermal creep should be relatively insignificant due to its Arhennius temperature dependence. Radiation-enhanced creep may contribute to structural deformation; however, the level of which can be kept low if excessive stress is avoided by design.

The two potential issues regarding the use of CW 316 SS as an INTOR blanket structural material are: (1) low uniform elongation (<1%) between 250-300°C, and (2) fatigue strength. The low uniform elongation is a characteristic of the CW austenitic stainless steels, not affected much by irradiation. The phenomenon could be related to dynamic strain aging which occurs during deformation. If so, the temperature interval within which this phenomenon occurs would also depend on strain rate. In any event, this limited deformation capability of CW 316 SS at low temperatures should be considered in the design. Fatigue strength needs to be considered due to the cyclic nature of the INTOR pulse mode operation. Data on strain-controlled as well as stress-controlled fatigues in CW 316 SS at temperatures below 450°C needs to be accumulated for assessment.

The data base on HT-9 is significant, but moderate in comparison to CW 316 SS. Compared to CW 316 SS, HT-9 has a higher heat flux capability and a better liquid-metal corrosion resistance. However, its fabrication (especially welding) requires special handling such as preheat and post weld heat treatment (PWHT). Irradiation-induced void swelling is less of a concern than for CW 316 SS because HT-9 has a higher swelling incubation fluence (>100 dpa). Just as for the CW 316 SS, thermal creep and high-temperature helium embrittlement of grain boundaries are not expected to cause problems for the INTOR blankets; possible helium effects at low temperatures remain to be investigated. With regard to radiation-enhanced creep, the concern is similar to that of CW 316 SS and can be handled by appropriate design.

The potential issues regarding the use of HT-9 as an INTOR blanket structural material are: (1) shift in the ductile-to-brittle transition temperature (DBTT), and (2) fatigue strength. The shift in DBTT could cause problems in maintenance; fatigue data of HT-9 at temperatures below 450°C also needs to be accumulated for assessment.

Both CW 316 SS/PCA (reference material) and HT-9 can be used as structural materials for the U.S. INTOR blanket designs. Because of the operating characteristics (i.e., relatively low temperature, low fluence, and pulse mode) of the INTOR blankets, swelling, creep, high-temperature helium embrittlement are not expected to cause serious problems, although some uncertainties remain regarding helium effects at low irradiation temperatures. Potential issues for these two materials are limited deformation capability (low uniform elongation) between 250-300°C for CW 316 SS and the DBTT shift for HT-9, and the fatigue strengths of both at low temperatures.

VIII. Tritium Permeation and Inventory in INTOR First Wall

Tritium permeation into the coolant and inventory in the INTOR first wall is considered an important design and economic issue during the operation of the reactor. There are still large uncertainties in key parameters and processes that control tritium permeation. One of these parameters is the sticking (recombination) coefficient, α, for both wall surfaces i.e., plasma side (α_F) and coolant side (α_b). This sticking coefficient is a measure of the degree of cleanliness of the surface, and it ranges from approximately α = 0.5 (clean surface) to $\alpha = 5 \times 10^{-5}$ (dirty surface). For INTOR first wall with water coolant, it is assumed that the water side will always be covered with an oxide layer, thus $\alpha_b = 5 \times 10^{-5}$ is assumed. Because of the continuous erosion of the plasma side wall during operation, the sticking coefficient is assumed to be $\alpha_F = 0.5$ (clean surface). Another value of $\alpha_F = 0.05$ (less clean surface) is assumed to determine the sensitivity of these calculations to the condition of the front wall.

Table 5 shows the calculated tritium permeation and inventory in the 316 SS first wall. These results, for the conditions shown, are for a continuous operation of 2-1/2 years (10 years of operation at 25% availability). The tritium permeation into the water coolant is negligible for both cases with and without traps induced by neutron damage. The inventory of tritium in the wall increases for the case of traps and for a less clean front surface. In all cases shown, the inventory is less than 1 kg over the proposed lifetime of the reactor. The steady state of tritium permeation and inventory in the first wall occurs at fluence greater than the 3.25 MW•Y/m^2 fluence assumed for INTOR. The continuous erosion of the wall during operation and the reduction of the wall thickness and the front surface temperature will still have an effect on the tritium permeation to the coolant.

IX. Benefits of Different Tritium Breeding Blankets in INTOR

This section examines the relative advantages and disadvantages of different tritium breeding ratios in INTOR in the range from 0.0 to 1.08. These options consider cases of no breeding and also breeding sufficient to handle both the burn requirement and the decay and processing losses expected in INTOR. The issues investigated for each option include the net present value of all costs, the probability of accidents during tritium transport, reliability and safety.

Costs: The net present value of a given blanket alternative is the sum of all costs in year zero plus the sum of the costs in all following years, each of these latter costs being discounted at the U.S. Federal Treasury rate (7%).

The types of costs assessed for each blanket case were: (1) differential capital costs between a $Be/LiAlO_2$/steel/H_2O breeder and a steel/H_2O non-breeder blanket; (2) tritium supply costs at $1/Curie; (3) tritium transport costs; and (4) differential operating costs which include maintenance and pumping power. The total cost was the sum of all four of these costs.

The capital costs were developed using data in reference (1) and appropriate scaling factors. Severe permeation losses does not appear to be a problem in INTOR, water cleaning systems were deleted from the data. To calculate the differential operating costs, the following assumptions were made. Maintenance costs occured over 14 years are ~ 1.5% of the direct capital cost each year.[6] Additional pumping power needs with a breeding blanket are 1.5 MWe for water at a cost of electricity of 200 mill/kWh.

The tritium supply costs all assume an initial startup inventory of 3 kg which is valued at ~ $29M. Tritium supply need at a tritium breeding ratio of 0.0 is 6.8 Hg/y. Needs are reduced proportionately as the breeding ratio is increased. At a tritium breeding ratio of 1.08, no tritium is supplied after startup.

To assess the cost of transporting tritium to the site, several assumptions were made. They were: (1) qualified containers holding 100 g of tritium would be available; (2) each truckload would consist of 100 g; (3) the distance traveled would be < 3000 miles; and (4) the cost of transporting is \$6000/1000 miles.[7]

The net present value of the total cost and each contributing cost for each of the net breeding ratios considered is shown in Table 6. For all options considered, the lowest cost is associated with on-site breeding. When more detailed designs are available, this observation should still be true.

Accidents/Terrorism: The probability per year of an accident is a product of the vehicle miles per year times the probability per mile of having an accident. Information on probabilities per mile is compiled in WASH-1236.[8] The total probabilities for minor, moderate, and severe accidents are 1.3×10^{-6}/mi, 3×10^{-7}/mi, and 8×10^{-9}/mi, respectively.

The distances and size of each shipment are < 3000 miles and 100 g/shipment load. The risk of an accident increases as the amount of tritium transported increases. The probability of terrorist attacks is assumed to be equivalent to that for severe transportation accidents. This increases as the breeding ratio decreases. Table 7 contains a summary of these probabilities for a 1000 mile distance per shipment.

Reliability: The reliability of the reactor system is dependent on a stable tritium supply. When the reactor is self sufficient or has at least a net breeding ratio of 0.6, then there is a level of flexibility built into reactor operations. With a reserve storage of ~1.5 kg (~ 16 d burn), a breakdown in the tritium recovery system can be accommodated without a perceptible interruption in the operating schedule.

When the net breeding ratio is zero, one is subject to three constraints. These are: (1) the availability of tritium from the source(s) that one has chosen; (2) price fluctuation in the cost of tritium from the source(s); and (3) accidents and/or terrorist attacks along the supply line. Since it seems that no one material source would provide the total tritium supply, one would also add the cost and time for the documentation and coordination necessary to ensure a continuous supply from multiple sources. It appears that a breeding blanket on-site provides the greater reliability for the INTOR design.

Safety: Safety considerations enter into both the breeding and the non-breeding options. For the breeding blanket, the net increase in safety related questions are those associated with possible chemical reactions of the breeder material. With a solid breeder, these are minimal. With liquid lithium, the engineering design would have to be such as to minimize any potentially destructive interactions to the reactor or to its operation.

The non-breeding option has its own safety ramifications. These are: (1) the accident probability previously developed; (2) the need for high temperature operation to decompose the solid hydride, and (3) the potential for a requirement for a higher reserve inventory on-site.

Conclusions: The advantages and disadvantages of different breeding options have been assessed. These options were breeding blankets with net breeding ratios of 0.0 and 0.6, 0.8, 1.0, and 1.08. On balance, the optimum choice is a self-sufficient breeding blanket.

Prime drivers for this choice are the lower net present value for the cost of the breeding options and the lower accident probabilities associated with these options. Reliability and safety are harder to evaluate, but these also are more favorable for a self-sufficient breeder blanket.

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Table 1. Properties of Candidate Neutron Multiplier Materials

Material	Be	BeO	Pb	PbO	Bi	Zr	Zr_5Pb_3	PbBi
Density, g/cm^3	1.85	2.96	11.34	9.53	9.8	7.6	8.93	10.46
Atoms or molecules/ cm^3, x 10^{-24}	0.124	0.0713	0.0335	0.0257	0.0282	0.0429	0.00468	0.0152
$\sigma(n,2n)$ at 14 MeV, barns	0.5	0.5	2.2	2.2	2.2	0.6	9.6	4.4
$\Sigma(n,2n)$ at 14 MeV, cm^{-1}	0.0618	0.0256	0.0737	0.0565	0.0621	0.0257	0.0449	0.0670
Threshold energy for (n,2n) cross section, MeV	1.868	1.868	6.765	6.765	7.442	7.274	6.765	6.765
$\sigma(n,\gamma)$ at 0.0253 eV, barns	0.0095	0.0095	0.17	0.17	0.034	0.18	1.41	0.204
$\Sigma(n,\gamma)$ at 0.0253 eV, cm^{-1}	0.00117	0.000671	0.00569	0.00437	0.000960	0.00772	0.00660	0.0031
Radioactivity isotopes	^{10}Be	^{10}Be	^{205}Pb	^{205}Pb	^{210}Po	^{93}Zr	^{93}Zr, ^{205}Pb	^{205}Pb, ^{210}Po
Decay types	B^-	B^-	Ec	Ec	α	B^-	B^-, Ec	Ec, α
Half lives	1.6×10^6 y	1.6×10^6 y	1.5×10^7 y	1.5×10^7 y	138.4 d	1.5×10^6 y	1.5×10^6 y 1.5×10^7 y	1.5×10^7 y 138.4 d
Melting point, °C	1278	2520	327.5	888	271.3	1852	1400	125
Thermal conductivity[a] at 25°C, W/m-°K	201	216[b]	35.3	2.8	7.92[c]	22.7	----	2.3[d]

[a] At 25°C.
[b] Pure beryllium oxide, hot pressed.
[c] Polycrystalline.
[d] At 200°C.

Table 2. Material Combinations with TBRs > 1.7

Neutron multiplier/breeder/reflector/structure/coolant
A - Be/Li (17Li83Pb)/C/steel/Li(17Li83Pb)
B - Be/Li (17Li83Pb, Li_7Pb_2)/C/steel/He
C - Be/Li_2O ($LiAlO_2$, Li_4SiO_4)/C (H_2O)/steel/H_2O
D - Be/Li_2O ($LiAlO_2$, Li_4SiO_4)/C/steel/He

Table 3. Design Issues to be Addressed During the Design Analysis Phase and Main Characteristics of Four Blanket Categories

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I Common Design Issues

- Beryllium performance problems
  - Beryllium swelling
    (low operating temperature and powder with low density accommodating swelling)
  - Tritium recovery
    (Tritium generated from Be(n,t) reactions)

- Tritium inventory
  (first wall, clad, Be, and breeder)

II Specific Design Issues and Characteristics

A. Be/Li (17Li83Pb)/C/steel/Li (17Li83Pb)

- Thermal hydraulics
  - Startup/shutdown procedure
  - External heating between pulses
  - Manifolding

- Structure stresses
  - MHD pressure stresses
  - First wall thermal stresses
  - Weight load (lithium-lead case)

- Liquid metal
  - Coolant choice for other reactor components (shield, divertor, etc.)
  - Operational procedure to accommodate the liquid metal

B. Be/Li (17Li83Pb, Li_7Pb_2)/C/steel/He

- Thermal hydraulics
  - Startup/shutdown procedure
  - External heating between pulses

- Structure stresses
  - First wall/thermal stresses
  - Weight load (lithium-lead case)

- Shielding
  - Neutron streaming from coolant lines

- Leak detection
  - He coolant

Table 3. Design Issues to be Addressed During the Design Analysis Phase and Main Characteristics of Four Blanket Categories (Continued)

C. Be/Li_2O ($LiAlO_2$, Li_4SiO_4)/C (H_2O)/steel/H_2O

- Tritium recovery
  - Purge gas from solid breeder

- Unique advantages
  - low pressure
  - low structure temperature
  - low structure stresses
  - breeder separated from coolant by several zones

D. Be/Li_2O ($LiAlO_2$, Li_4SiO_4)/C/steel/He

- Tritium recovery
  - Purge gas from solid breeder

- Shielding
  - Neutron streaming from coolant lines

- Leak detection
  - (He coolant)

- Unique advantages
  - Medium pressure
  - Low structure temperature
  - Medium structure stress
  - Breeder separated from coolant several zones

Table 4. Parameters Relevant to Self-Cooled Lithium Blanket

Neutron wall loading, MW/m^2		1.30
Surface heat flux, MW/m^2		0.14
Magnetic flux density at the outboard blanket, T		4.70
Heated length in the poloidal direction, M		5.00
Availability, %		25.0
First wall thickness, mm (BOL)		9.0
(EOL)		4.0
First wall coolant channel		
radial span, mm		30.0
toroidal span, mm		80.0
side wall thickness, mm		4.0
STRUCTURE MATERIAL	PCA	HT-9
Coolant average velocity, M/s	0.25	0.15
Maximum interface temperature, °C	365	420
Maximum structure temperature, °C	470	485
Total pressure drop, MPa	2.2	1.5

Table 5. Tritium Permeation and Inventory in INTOR First Wall After 2-1/2 Years of Continuous Operation

Particle flux = $2.5 \times 10^{16}\ m^{2}.s^{1}$

Wall thickness = 9 mm

Material = 316 SS

Wall area = 400 m^2

$\alpha_b = 5 \times 10^{-5}$

α_F	Tritium Permeation (g/d) No Traps	Tritium Permeation (g/d) With Traps	Tritium Inventory (g) No Traps	Tritium Inventory (g) With Traps
0.5	2.3×10^{-9}	~ 0	125	365
0.05	1.56×10^{-8}	~ 0	325	890

Table 6. Net Present Value of the Cost of Each $Be/LiAlO_2/Steel/H_2O$ Blanket System

Net TBR	Differential Capital	Differential Operating	Tritium Supply	Tritium Transport	Total
0	-	-	425	8	433
0.6	47	9	229	4.3	289
0.8	80	9	144	2.7	236
1.0	113	19	60	1.1	193
1.08	127	19	29	0.5	175

Table 7. Minimum Accident Probability

NET-BR	Minor	Moderate	Severe
0	0.95	0.21	5.9×10^{-3}
0.6	0.44	0.10	2.8×10^{-3}
0.8	0.28	0.06	1.7×10^{-3}
1.0	0.10	0.02	6.3×10^{-4}
1.08	0.04	0.01	2.4×10^{-4}

U.S. CONTRIBUTION

to

INTOR WORKSHOP

Phase II-A, Part 3

March 10-21, 1986

GROUP F

BLANKET AND FIRST WALL

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