U.S. SOLID BREEDER BLANKET DESIGN FOR ITER

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ABSTRACT

The U.S. blanket design activity has focused on the developments and the analyses of a solid breeder blanket concept for ITER. The main function of this blanket is to produce the necessary tritium required for the ITER operation and the test program. Safety, power reactor relevance, low tritium inventory, and design flexibility are the main reasons for the blanket selection. The blanket is designed to operate satisfactorily in the physics and the technology phases of ITER without the need for hardware changes. Mechanical simplicity, predictability, performance, minimum cost, and minimum R&D requirements are the other criteria used to guide the design process. The design aspects of the blanket are summarized in this paper.

I. INTRODUCTION

A solid-breeder water-cooled blanket concept has been developed for ITER based on a multilayer configuration[1]. Two versions of this blanket have been studied. The difference among the two versions is in the fabricated forms of the breeder material. The breeder material form is sintered products (blocks) or packed bed of small pebbles. Both versions have beryllium for neutron multiplication and solid-breeder temperature control. Beryllium has a sintered product form (blocks). The blanket does not use helium gaps or insulator material to control the solid breeder temperature. The beryllium zones provide the desired temperature gradient between the low temperature coolant and the breeder. Lithium oxide (Li_2O) and lithium zirconate (Li_2ZrO_3) are the primary and the backup breeder materials, respectively. The lithium-6 enrichment is 95%. The use of high lithium-6 enrichment reduces the solid breeder volume required in the blanket and consequently the total tritium inventory in the solid breeder material. Also, it increases the blanket capability to accommodate power variation. The multilayer blanket configuration can operate at 150% the nominal power without violating the different design guidelines.

* Work supported by the U.S. Department of Energy, Office of Fusion Energy, under Contract No. W-31-109-Eng-38.

The blanket is designed to produce the necessary tritium required for the ITER operation and to operate at power reactor conditions as much as possible. Also, the reliability and the safety aspects of the blanket are enhanced by using low-pressure coolant and the separation of the tritium purge flow from the coolant system by several barriers. The other criteria used to guide the design process are mechanical simplicity, predictability, performance, cost, and minimum R&D requirements.

The inboard blanket has a single breeder zone embedded in a beryllium zone. The poloidal coolant of the first wall and the shield behind the blanket were used to cool the inboard blanket by conducting the nuclear heating to these coolant zones. This results in a simple design. The outboard blanket has two (or three) breeder zones with toroidal coolant which improves the performance and the mechanical design of the blanket. An additional coolant panel is used in the beryllium zone between the two breeder zones to get the appropriate temperature profile for the blanket materials.

The net tritium breeding ratio based on three-dimensional analysis is in the range of 0.81 to 0.92 depending on whether two or three breeder zones are used in the outboard blanket. These values do not account for any tritium generated from the test sections. The analysis uses detailed models for the different reactor components including the divertor zones, the sector side walls, the assembly gaps, the copper stabilizer, and the spatial source distribution.

The blanket box is designed to accommodate the plasma disruption conditions without exceeding the stress limits for the Type 316 austenitic steel. The accommodation of the electromagnetic pressure on the blanket box insures a maximum first-wall deformation of ss than 100 μm during normal operation from e helium purge gas and the surface heat ux. This results in a satisfactory thermal anket performance.

Each breeder zone is purged by He with 2% H_2 for continuous tritium recovery. The otium is added to the purge gas to reduce lubility and adsorption, important at higher mperatures; and to enhance desorption kinecs, important at lower temperatures. The rerage H/T ratio in the blanket is 30. The flow rate is chosen to be high enough to eep the total moisture (H_2O + HTO + T_2O) essure to <10 Pa throughout the whole blanet. The tritium inventory is calculated to e <14 g in the Li_2O breeders for both phases f pulsed operation. Assuming no tritium ecovery from the beryllium material, the toal inventory is ~1.4 kg at the end-of-life of .8 full power year. However, the blanket is esigned with separate helium purge loop for he beryllium mulitplier which will reduce his inventory.

I. MATERIAL SELECTION

Li_2O was chosen as the reference breeder aterial because of its excellent thermal and ritium transport properties. Li_2ZrO_3, which as good tritium properties, good stability, and low thermal and in-reactor swelling, but oor thermal conductivity, was chosen as the ackup material. $LiAlO_2$ has good stability, low thermal and in-reactor swelling, good thermal conductivity, but poor tritium release characteristics below 450°C. Thus, it was not considered seriously for ITER application. Finally, Li_4SiO_4, which has very low thermal conductivity and is not superior in any category, was eliminated for ITER application. Figure 1 is a comparison plot of the tritium residency times for the breeders based on correlation fits to experimental data.

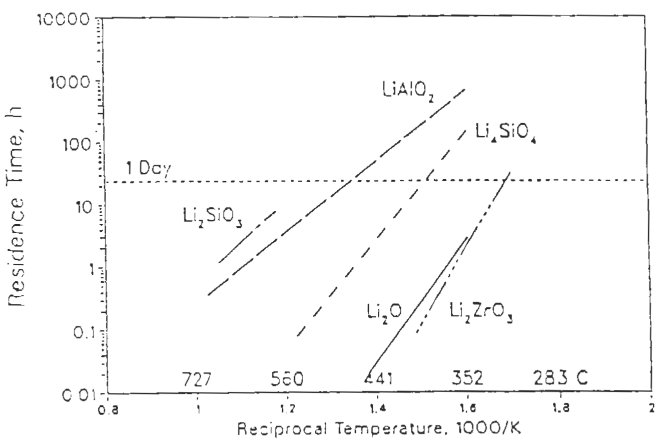


Figure 1. Comparison of tritium residency times for solid breeders based on extrapolations from tritium release experiments.

Austenitic steel (Type 316 solution annealed) was selected as the reference structural material on the basis of an extensive database and ease of fabrication. Water coolant with low temperature (60-100°C), and low pressure is specified for safety considerations.

The desire to achieve a net tritium breeding ratio close to unity with limited breeding volume because of inboard shielding requirements, plasma systems, and provisions for nuclear testing dictates the use of beryllium as a neutron multiplier.

III. DESIGN GUIDELINES

Design guidelines and limits were established for blanket components to satisfy the ITER mission of 3 MWa/m^2 fluence and insure satisfactory performance. The maximum load structural temperature during normal operation for annealed Type 316 Stainless Steel is 400°C based on swelling consideration. While the corresponding temperature during off-normal conditions is 800°C based on deformation consideration to avoid damage to the neighbor segments. The allowable stress intensity, S_m, for annealed Type 316 SS at 400°C is 110 MPa.

Temperature limits are defined for the non-structural components of the blanket: solid-breeders, Be multiplier, and stainless steel cladding separating Be/breeder zones. The temperature limits are based on tritium retention/recovery, materials stability, mass transfer, and compatibility. It should be emphasized that the design guidelines and limits give general target ranges based on materials performance. Within these target ranges, detailed thermal, mechanical, tritium and mass-transfer analyses were done to answer design-dependent issues and insure satisfactory performance for a particular design configuration and set of operating parameters. Table 1 summarizes the temperature limits for the blanket materials based on experimental data.

While there are no constraints on T_{min} at the material interfaces, Be/steel and breeder/steel interaction rates determine T_{max} at the interfaces. A limit of 0.1 mm is chosen somewhat arbitrarily assuming a total steel cladding thickness of 1 mm. The corresponding temperatures are 480°C for Be, and 500°C for Li_2O. Very little interaction was observed experimentally between Li_2ZrO_3 and steel, thus the 750°C limit is arbitrary.

IV. MECHANICAL DESIGN

The outboard (OB) blanket is divided into 48 poloidal segments of equal toroidal extent, three segments for each toroidal field (TF) coil sector. The three segments consist of a central segment and two side segments. The side segments extend the full height of the

Table 1
Summary of Long-Time, Steady-State Temperature Limits for Non-Structural Blanket Components

Material	T_{min}, °C	Basis	T_{max}, °C	Basis
Solid Breeders				
Li_2O	320	tritium transport	1000	mass transfer/sintering
Li_2ZrO_3	320	tritium transport	1000	sintering
Be multiplier	none	---	600	swelling
Be/steel	none	---	480	≤0.1 mm wastage
Li_2O/steel	none	---	500	≤0.1 mm wastage
Li_2ZrO_3/steel	none	---	<750	<0.1 mm wastage

reactor. The central segment is located between TF coils and is divided into an upper and lower segment. A penetration is situated at midplane between the upper and lower central segments. Side segments and upper central segments have service connections at the top, while the lower central segment has service connections at the bottom.

The OB solid breeder blanket is of layered configuration consisting of Be blocks interleaved with solid breeder zones and coolant panels, all contained within a stainless steel box. Figure 2 shows an isometric view of the outboard blanket internals. Figure 3 shows a poloidal cross section of the upper central segment. It is noted that the multiplier zone thickness increases from midplane to the upper extremity. This is done to accommodate the change in the neutron wall loading. There are two solid breeder zones and two blanket coolant panels extending the full height of the module. The present design can use three solid breeder zones and three coolant panels to enhance the tritium breeding capability. It should be noted that the solid breeder zones and coolant panels are of constant thickness regardless of their poloidal location.

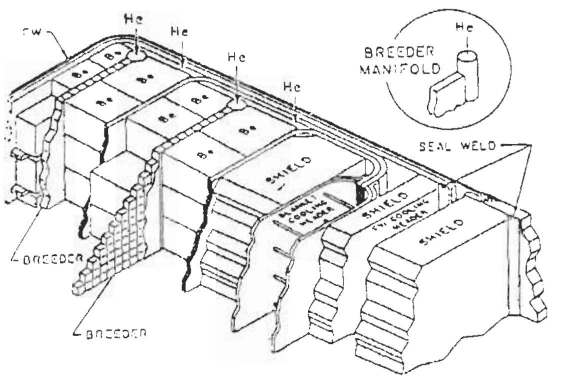
The Be zones are designed with the option of purging with He gas for tritium recovery. This is accomplished by providing spaces at the interfaces between the Be plates and the side walls as shown in Fig. 2. These spaces act like manifolds for distributing He gas poloidally. This gas then flows toroidally across the Be plates and comes out on the other side of the module.

The solid breeder consists of Li_2O panels, 0.8 cm thick, clad in 0.1 cm thick SS sheets. The panels are continuous from top to bottom and have built in manifolds on the sides running in the poloidal direction. Purge gas flows poloidally through the manifold, then toroidally across the panel through semi-circular cylindrical grooves at the breeder cladding interface and finally back out through the return manifold. This purge gas carries with it the produced tritium from the solid breeder.

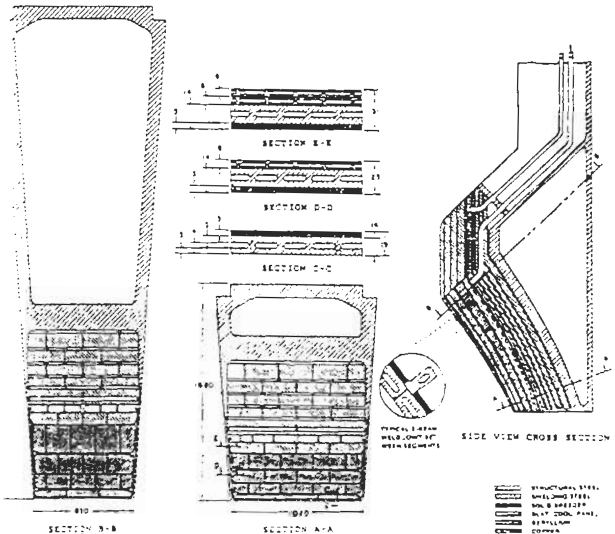
Three neutral beam ports are integrated with the OB blanket, which comes in tangent to the circumferential centerline of the plasma and thus sweeps across two side modules and one central module. Figure 4 shows a modified blanket segment with a missing front corner.

The inboard (IB) blanket is divided into 32 toroidally equal segments, or two segments per TF coil. Figure 5 has a side view of an IB module with cross sections at midplane and at the top extremity (Z= ±3.4 m). As in the OB blanket, the radial build is smaller at midplane than at the extremities. Water and purge gas connections are all at the top.

To reduce the plasma disruption effects, each segment is subdivided into three parts electrically insulated from each other. The ılated zone extends 27 cm at midplane and cm at the extremities. The solid breeder the Be zone are purged with He gas. The blanket has poloidal coolant flow and one id breeder zone.



gure 2. Isometric view of the outboard anket internals.



igure 3. Poloidal cross section of the upper entral module.

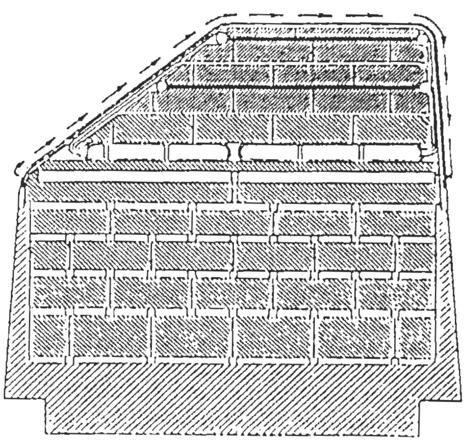


Figure 4. Modified blanket segment to accomodate neutral beam port.

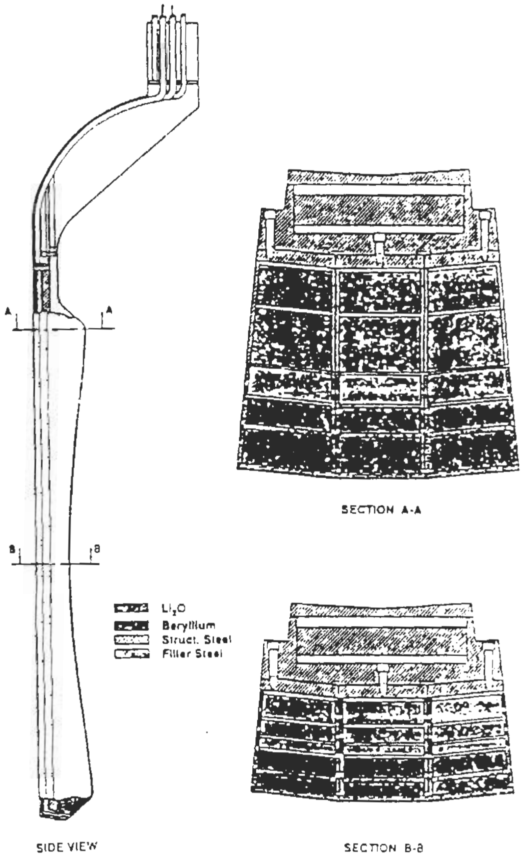


Figure 5. Inboard module with cross sections at midplane and the top extremity.

V. NEUTRONICS ANALYSIS

The first wall/blanket/shield design and optimization system² (BSDOS) was used to carry out the neutronics and thermal-hydraulics analysis in an integrated manner. The analysis is performed to define a blanket configuration that maximizes the tritium breeding ratio and satisfies the temperatures limits for the different materials. BSDOS uses the one/two-dimensional discrete ordinates code ONE/TWODANT to carry out the transport calculations.

The radial build of the blanket is defined at six locations to insure the accommodation of the poloidal distribution of the neutron wall loading. The radial build is defined at the midplane and the end of the blanket in the poloidal direction for the inboard and the outboard sections. The other two locations are at the starting point of the copper stabilizer in the poloidal direction. At each location, the calculated neutron wall loading is used to determine the radial build of the blanket.

The total inboard blanket and shield thickness is defined based on the global system studies. This thickness does not permit the use of a full blanket module and an adequate shield thickness to protect the inboard section of the toroidal field coils. Therefore, the inboard blanket thickness is minimized to provide an adequate radial space for the shield.

In the inboard section, the beryllium material is used with a 0.65 density factor to reduce the thermal conductivity of the sintered block. This reduces the beryllium thickness required to get the temperature distribution of the solid breeder material within the temperature window. On the contrary, the beryllium material of the first outboard breeder zone has a density factor of 0.85 to get high thermal conductivity. This permits the use of a thick beryllium zone in the front section of the blanket where it is needed from the neutronics point of view. The beryllium material of the second breeder zone has a 0.65 density factor to reduce the required beryllium thickness similar to the inboard section. The range of the beryllium density factor of 0.65 to 0.85 is defined based on material considerations including swelling and mechanical properties.

The calculated radial build of the outboard and inboard blankets at the different poloidal locations are given in Table 2 for the blanket with breeder blocks. The change in the beryllium material thickness in the poloidal direction is similar to the poloidal change of the neutron wall loading on the first wall. The local tritium breeding ratio for this concept varies from 1.375 at $Z = 0.0$ to 1.461 at $Z = 2.7$ where the copper stabilizer starts. At the end of the blanket, the local tritium breeding ratio is 1.310. The blanket thickness varies from 26.5 cm at the midplane to 58.5 cm at the end. The local tritium breeding ratio of the inboard blanket changes from 0.755 at $Z = 0$ to 0.895 at the end of the blanket. The corresponding blanket thicknesses are 10.7 and 18.3 cm, respectively.

The outboard blanket is reconfigured with three breeder zones and the same minimum breeder temperature. The local poloidal tritium breeding ratio at the midplane for the new configuration is 1.634 compared to 1.375 for the blanket with two breeder zones, which is about 19% increase. However the blanket thickness is increased from 26.5 to 45.1 cm.

Three-dimensional neutronics calculations have been performed for the solid-breeder water-cooled blanket design with sintered product materials to determine the net tritium breeding ratio as well as tritium breeding and nuclear heating in the different components of the blanket. The continuous energy coupled neutron-gamma Monte Carlo Code MCNP, version 3B, has been used with cross section data based on the ENDF/B-V evaluation.

The blanket segments were modeled in detail with the poloidally varying radial builds required for breeder temperature control. The sidewalls, the 2-cm-thick assembly gaps between blanket segments and the detailed layered configuration of the FW and blanket are included in the model. The copper stabilizer loops used in the outboard region were modeled. The FW configuration in the inboard, outboard, and divertor regions was modeled in detail. The divertor plates and vacuum pumping ducts in the lower divertor region were included in the model. Sixteen standard 1.07 m × 3.4 m radial ports were used at the middle of the outboard region. These ports are utilized for testing, plasma heating, startup and maintenance. A typical Li/V blanket was used in the ports to represent a blanket test module.

The results indicate that the net TBR is 0.81 with 15% of it contributed by the inboard blanket. It is interesting to note that coupling the 1-D toroidal geometry results with coverage fractions of the different breeding zones, the net TBR was estimated to be 0.84 which is only 3.7% different from the value obtained form the detailed 3-D calculation. In addition, the 1-D analysis for the blanket design with small breeder pebbles resulted in 3.2% lower net TBR compared to the design with breeder blocks implying that the 3-D calculation for the packed bed design is expected to yield 0.78 for net TBR. Furthermore, the 1-D analysis indicated that the overall TBR increases by 9.6% if three breeder plates are utilized in the outboard blanket in the zone $-2.7 \text{ m} < z < 2.7 \text{ m}$. Therefore, the net TBR from the 3-D calculations is expected to increase to 0.89 and 0.86 with three breeder plates for the block and packed bed designs, respectively.

VI. THERMAL ANALYSES

The thermal analysis is always performed for each blanket configuration after the neutronics analysis to insure the appropriate temperature profiles of the different materials. In all the blanket configurations, the zone dimensions and material density factors are defined to get 450°C as a minimum temperature for the solid breeder material for the technology phase. This choice results in a satisfactory tritium inventory for the physics and the technology phases. Also, it limits the maximum temperature at the clad beryllium interface to <430°C. Which results in a steel reaction layer of <0.1 mm at the end-of-life.

The BSDOS system gets the zone dimensions and the radial distribution of the nuclear heating over a fine mesh from the neutronics analysis. BSDOS transfers these data to the

Table 2. Radial Build of the Blanket

ZONE	MATERIAL (DF)	THICKNESS (cm) Outboard 1.2 MW/m^2 Z = 0	0.958 MW/m^2 Z = ±2.7 m		0.6 MW/m^2 Z = ± 4.3 m	Inboard 0.884 MW/m^2 Z = 0	0.325 MW/m^2 Z > ±3.4 m
First Wall Layers							
Tile[a]	C	2.0	2.0	2.0	2.0	2.0	2.0
First wall	steel	0.5	0.5	0.5	0.5	0.5	0.5
Coolant	H_2O	0.4	0.4	0.4	0.4	0.4	0.4
Back wall	steel	0.5	0.5	0.5	0.5	0.5	0.5
Stabilizer	Cu			0.5	0.5		
Blanket							
Multiplier	Be	3.4[a]	4.8[a]	4.3[a]	7.2[a]	3.3[b]	9.0[b]
Clad	steel	0.1	0.1	0.1	0.1	0.1	0.1
Breeder	Li_2O (0.80)	0.8	0.8	0.8	0.8	1.0	1.0
Clad	steel	0.1	0.1	0.1	0.1	0.1	0.1
Multiplier	Be	5.9[a]	7.9[a]	7.9[a]	12.7[a]	4.6[b]	6.5[b]
Coolant channel	steel	0.2	0.2	0.2	0.2	0.2	2.8[c]
Coolant	H_2O	0.2	0.2	0.2	0.2	0.2	0.2
Coolant channel	steel	0.2	0.2	0.2	0.2		
Multiplier	Be (0.65)	5.7	8.4	8.4	19.0		
Clad	steel	0.1	0.1	0.1	0.1		
Breeder	Li_2O (0.80)	0.8	0.8	0.8	0.8		
Clad	steel	0.1	0.1	0.1	0.1		
Multiplier	Be (0.65)	7.1	11.6	11.6	15.2		
Coolant channel	steel	0.2	0.2	0.2	7.2[d]		
Coolant	H_2O	0.2	0.2	0.2	0.2		
Total first wall/blanket thickness		26.5	37.1	37.1	58.5	10.9	18.5
Local tritium breeding ratio		1.375	1.461	1.356	1.310	0.755	0.895

a- A 0.85 density factor.
b- A 0.65 density factor.
c- 2.6 cm of the steel in this zone is part of the bulk shield and it is not included in the total first wall/blanket thickness.
d- 7.0 cm of the steel in this zone is a part of the bulk shield and it is not included in the total first wall/blanket thickness.

hree-dimensional mesh generator for modeling onlinear systems, INGRID, to model the blanet segment. Geometrical data, coolant condiions, and gap conductance model are included n the calculations. INGRID generates a hree-dimensional finite element model for the lanket segment. The three-dimensional finite lement heat transfer code TOPAZ3D uses this odel to calculate the temperature distribuion profiles and the change in the coolant onditions. The physical properties of the ifferent materials are evaluated at each node s a function of the temperature and the mateial density factor. The value of the surface eat flux is taken 0.25 the neutron wall loadng. The coolant inlet temperature is 60°C ind the water pressure is 10 atm. The coolant low direction is included in the model.

The neutronics and the thermal calculaions are iterated. The purpose of this teration is to get the minimum temperature of he solid breeder at 450°C for the technology hase. The results from these analyses give the radial build and the temperature distribuion of the blanket. The extreme temperatures are given in Table 3. The solid breeder material in this concept uses only about 90°C of the 600°C temperature window (400 to 1000°C) for the lithium oxide.

In the physics phase, the reactor fusion power is 1100 instead of 860 MW for the technology phase, and carbon tiles are used for first wall protection. Also, the flat DT burn time is relatively short. It is 400 s instead of 2290 s for the technology phase. The neutronics and the thermal analyses were performed for the physics phase. Figure 6 gives the temperature history of the first solid breeder zone at the midplane of the outboard blanket. At the midplane section, the solid breeder temperatures reach very close to saturation during the second pulse, while the blanket extremity requires 3 to 10 pulses to reach saturation values.

VII. TRITIUM DESIGN ANALYSIS AND PERFORMANCE

The solid breeder zones of the blanket consist of blocks of 80% dense Li_2O with radial thicknesses of 8 mm and 10 mm for outboard and inboard zones, respectively. The breeder is purged by He + 0.2% H_2 gas which flows through semi-circular cylindrical grooves at the breeder cladding interface. The purge flow rates (2.71 moles/s outboard and 0.53 moles/s inboard) are chosen based on limiting the maximum local moisture (H_2O + HTO + T_2O) to be <12 Pa to avoid precipitation of separate phased LiOH(T). The minimum long-time

Table 3. Steady State Extreme Temperatures of the Different Blanket Materials for the Technology Phase

ZONE	MATERIAL	OUTBOARD TEMPERATURES, °C				INBOARD TEMPERATURES, °C	
		1.2 MW/m^2	0.958 MW/m^2		0.6 MW/m^2	0.884 MW/m^2	0.325 MW/m^2
		Z = 0	Z = ±2.7 m		Z = ± 4.3 m	Z = 0	Z = ±3.4 m
First Wall Layers[a]							
First wall	steel	77-191	74-168	74-168	69-132	92-174	65-98
Back wall	steel	82-225	81-222	82-228	79-203	94-187	69-131
Stabilizer	Cu			230-237	205-210		
Blanket							
Multiplier	Be	280-401	277-440	285-412	252-418	224-413	155-427
Clad	steel	426-438	462-472	432-442	432-439	433-440	435-438
Breeder	Li_2O	453-537	484-559	455-522	448-492	452-526	442-472
Clad	steel	424-437	459-470	433-443	432-440	431-438	435-437
Multiplier	Be	192-401	180-437	174-413	143-418	162-415	288-427
Coolant channel	steel	77-138	75-131	75-127	71-109	91-126	67-275
Coolant channel	steel	70-109	68-101	68-99	65-84		
Multiplier	Be	138-421	124-454	121-431	97-440		
Clad	steel	435-446	465-472	441-449	444-447		
Breeder	Li_2O	451-505	476-514	452-487	449-460		
Clad	steel	434-443	465-471	442-447	446-447		
Multiplier	Be	123-422	106-457	104-434	309-444		
Coolant channel	steel	68-99	66-88	66-86	63-303		
Total first wall/blanket thickness		26.5	37.1	37.1	58.5	10.9	18.5
Local tritium breeding ratio		1.375	1.468	1.356	1.310	0.755	0.895

a- The surface heat flux is 25% the DT neutron wall load value.

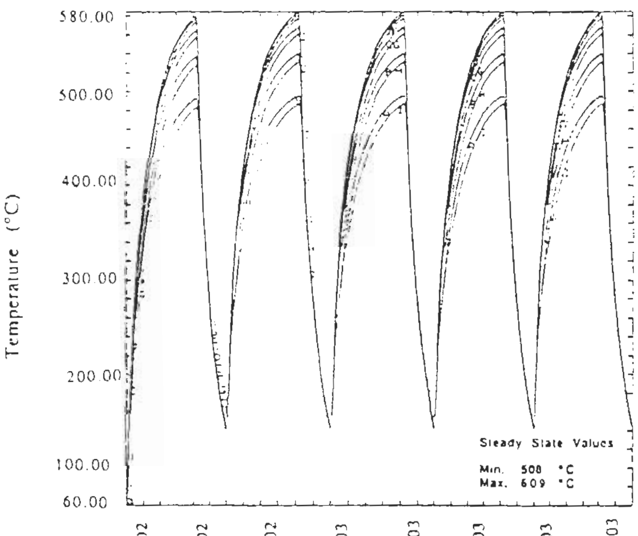


Figure 6. Lowest temperature history of the first breeder zone during five physics pulses at the outboard midplane section.

Detailed steady-state and transient analyses of tritium inventory and release rate were performed for the Li_2O breeder in the layered design. Poloidal variations in tritium generation rate and temperature prc file, were included in the steady-state analy sis, as well as uncertainties in model parame ters. The calculated steady-state inventor is only 14 grams using nominal model paramet ers and technology phase operating conditions With uncertainties included, the inventor ranges from a lower bound of 6 grams to a upper bound of 45 g. Using the same mode parameters for the pulsed mode of technolog phase operation the calculated transier inventory is only 18% higher than the steady state operating mode due to the thermal- an tritium- response lags. Figure 7a shows th inventory build-up at the outboard core midplane as a function of the number o technology pulses (2290-s flat burn, 20- ramps, 2490-s total cycle time).

The tritium inventory analysis is base on a conservative (slow release) model to pro vide an upperbound on the inventory. A dif ferent approach was used for calculating th maximum tritium release rate during pulsec operation. For this bounding analysis, the model parameters lending to the fastest release and lowest inventory were used. This approach is important in providing guidance tc the design of the tritium processing units. Figure 7b shows the results of this transient calculation at the outboard midplane locatior during technology pulses. The results apply to the Li_2O at the purge outlet. In this mall region the local tritium release rate an approach ~5 times the generation rate. However, when the toroidal and poloidal variations are considered, the overall increase in release rate is <2 times the generation rate.

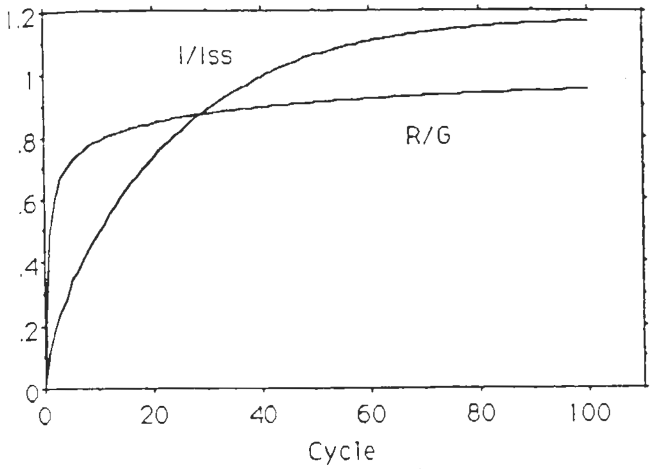


Figure 7a. Inventory (I/Iss) and release (R/G) fractions (technology phase).

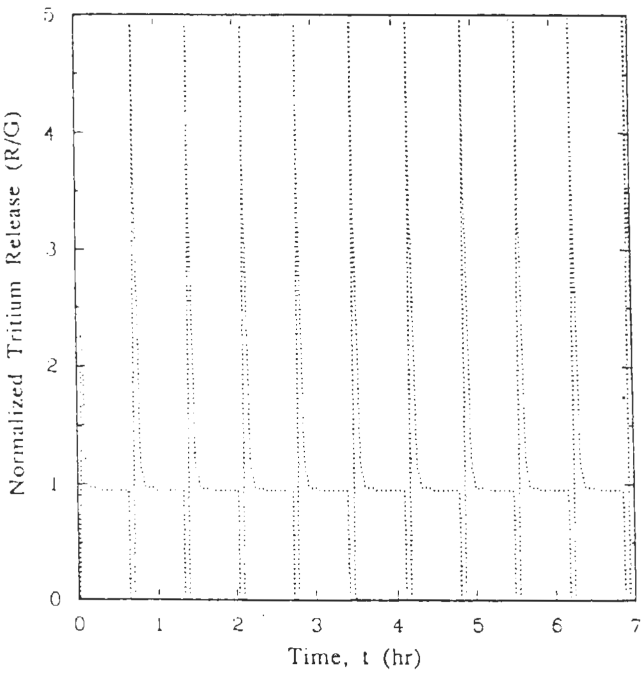


Figure 7b. Tritium release history for the outboard Li_2O region 1 during the technology phase with a 20 Pa tritium partial pressure in the purge.

VIII. BLANKET R&D ISSUES

Several R&D tasks have been identified for the blanket design. It includes the following R&D tasks:

- Characterization of beryllium. There is a need for data on fabrication techniques and irradiation effects such as swelling, tritium retention, and compatibility with other material.

- Characterization of the ceramic breeder. Data on tritium release and irradiation effects on the mechanical properties are required to reduce the design uncertainties.

- Temperature control. The method used to provide a thermal insulation between the structural material and the ceramic breeder require testing under reactor conditions.

- Structural material data base for Type 316 austenitic steel. There is a need for data on irradiation effects on low temperature fracture toughness, cyclic fatigue, and crack growth. Also, data on welds and brazing without and with irradiation are required.

- Fabrication and testing of a blanket module.

The above issues are included in the ITER R&D plan to provide the necessary data during the engineering design phase.

IX. SUMMARY

The U.S. solid breeder blanket design satisfies the design goals of ITER. Also, the R&D requirements have been identified to provide the necessary data during the engineering design phase.

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