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SOLID BREEDER BLANKET OPTION FOR THE ITER CONCEPTUAL DESIGN*

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Abstract

A solid-breeder water-cooled blanket option was developed for ITER based on a multilayer configuration. The blanket uses beryllium for neutron multiplication and lithium oxide for tritium breeding. The material forms are sintered products for both material with 0.8 density factor. The lithium-6 enrichment is 90%. This blanket has the capability to accommodate a factor of two change in the neutron wall loading without violating the different design guidelines.

The design philosophy adopted for the blanket is to produce the necessary tritium required for the ITER operation and to operate at power reactor conditions as much as possible. At the same time, the reliability and the safety aspects of the blanket are enhanced by the use of a low-pressure coolant and the separation of the tritium purge lines from the coolant system. The blanket modules are made by hot vacuum forming and diffusion bonding a double wall structure with integral cooling channels. The different aspects of the blanket design including tritium breeding, nuclear heat deposition, activation analyses, thermal-hydraulics, tritium inventory, structural analyses, and water coolant conditions are summarized in this paper.

Introduction

A solid-breeder water-cooled blanket option was developed for ITER based on a multilayer configuration. The blanket uses beryllium for neutron multiplication and lithium oxide for tritium breeding. Li_2O is recommended as the solid breeder material because of its superior performance relative to the other candidates (Li_4SiO_4, Li_2ZrO_3, and $LiAlO_2$). The recommended backup material is Li_4SiO_4 subject to future review based on the expected new results from the current research programs. The material forms are sintered products for both materials with 0.8 density factor. The lithium-6 enrichment is 90%. The use of high lithium-6 enrichment reduces the solid breeder volume required in the blanket and the total tritium inventory in the solid breeder material. Also, it increases the blanket capability to accommodate power variation. The multilayer blanket was configured to accommodate a factor of two change in the neutron wall loading without violating the different design guidelines. The key parameters of the blanket are given in Table 1 and the blanket configurations are shown in Figs. 1 and 2.

The design philosophy adopted for the blanket is to produce the necessary tritium required for the ITER operation and to operate at power reactor conditions as much as possible. At the same time, the reliability and the safety aspects of the blanket are enhanced by the use of a low-pressure coolant and the separation of the tritium purge lines from the coolant system.

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Table 1
Key Parameters of the Solid-Breeder Water-Cooled Blanket Option

Parameters	Outboard	Inboard
Total lithium oxide thickness (0.8 DF), cm	1.6	0.9
Total beryllium thickness (0.8 DF), cm	34.0	11.5
Total blanket thickness, cm	43.0	23.0
Local tritium breeding ratio	1.6	0.93
Energy deposited per fusion neutron, MeV/DTn	21.9	18.1
Inlet-outlet coolant temperature, C	40-60	40-60
Water mass flow rate, Kg/s	279	94
Water velocity in the blanket, m/s	4	4
Tritium inventory in lithium oxide, g	9.1	47.9
Helium pumping power for the lithium oxide zones, KW	2.1	11.8
Helium pumping power for the beryllium zones, KW	9.1×10^{-2}	3.1×10^{-2}

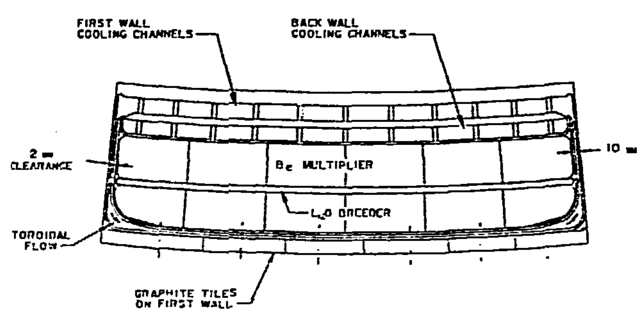

Figure 1. Inboard solid breeder blanket section at midplane.

The blanket uses one and two thin solid breeder layers in the inboard and the outboard sections, respectively. Each breeder layer has a steel clad and it is inserted inside a beryllium zone. The water coolant removes the heat from the beryllium zone surfaces which are parallel to the first wall. The blanket has a poloidal manifold for each segment to supply the water coolant for each blanket module in the segment. The water flows in the radial direction in the side walls,

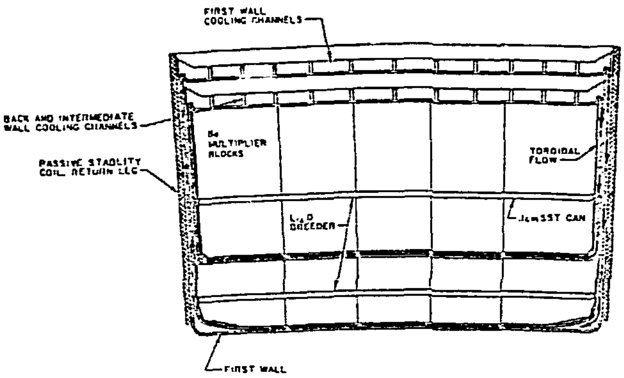

Figure 2. Outboard solid breeder blanket section at modplane.

in the toroidal direction to cool the blanket module, and in the radial direction to exit from the other side walls to the poloidal manifold.

Water coolant conditions are defined to have low conductivity, i.e., 3 μmho/cm, be neutral, i.e., pH ~ 7, and contain no additives if possible. A 0.06 MPa pressure is required to keep the hydrogen from radiolysis in solution at < 80°C at the first wall. Maintenance of this water quality during reactor operation, i.e., over ten years of operation, will require continuous use of a purification system containing filters and ion-exchange resins and careful attention to the conductivity. Specific ions which enhance corrosion, chloride, etc., will have strict limits, ~ 15 ppb.

The blanket is divided into 32 inboard and 64 outboard segments to accommodate the plasma disruption forces. The blanket modules are made by hot vacuum forming and diffusion bonding a double wall structure with integral cooling channels.

Li_2O is recommended as the ITER solid breeder ceramic. The recommended backup material is Li_4SiO_4. Li_2O is superior to the other candidates (Li_4SiO_4, Li_2ZrO_3, and $LiAlO_2$) in terms of lithium atom density, thermal conductivity, thermal response time, and tritium lattice diffusivity. In terms of overall tritium transport and tritium inventory, it ranks at least as high, if not higher, than Li_4SiO_4 and Li_2ZrO_3, and it is clearly superior to $LiAlO_2$. It also ranks high in terms of activation and afterheat considerations. Based on data currently available, it appears that possible problems associated with Li_2O (e.g., lithium mass transport, chemical interaction with metals, tritium trapping in the form of solid-phase LiOT, thermal expansion, and swelling) can be accommodated in the design and/or minimized by proper choice of purge flow rates and purge chemistry within the recommended temperature window.

With regard to the backup material, $LiAlO_2$ is attractive with regard to its chemical and radiation stability, but is has very poor tritium release characteristics below 500°C. Li_2ZrO_3 is attractive in terms of its chemical and radiation stability and its low-temperature tritium release characteristics, but it has the low thermal conductivity and possible problems associated with activation and afterheat. The primary concern with Li_4SiO_4 which resulted in its being relegated to a backup role to Li_2O is the speculation that its tritium release characteristics will deteriorate with burnup. Otherwise, it has reasonably good thermal, tritium, chemical, and radiation properties.

Nuclear and Thermal-Hydraulic Analyses

The design philosophy of this blanket option is to produce the necessary tritium required for the ITER operation and to operate at power reactor conditions as much as possible while ensuring the reliability and safety of the blanket which has been accomplished by the use of a low-pressure coolant and the separation of the tritium purge lines from the coolant system.

The desire to operate at power reactor conditions requires continuous tritium recovery from the blanket. In order to operate in such mode, the solid breeder temperature has to be within a specific temperature range (temperature window). The minimum temperature of this range is 350 to 400° which does not match the desired operating temperature range for a low-pressure water coolant system. The design approach is to utilize the beryllium multiplier between the solid breeder and the water coolant to get the breeder temperature distribution within the required temperature window. This approach does not make use of an adjustable gap conductance or helium gaps to adjust the solid breeder temperature which increases the robustness of the design. A multilayer configuration [1-3] is employed for the design. The first wall/blanket/shield design and optimization system (BSDOS) [4] was used to carry out the neutronics and thermal hydraulics analyses in an integrated manner.

The first step in the analysis was to define a blanket configuration which maximizes the tritium breeding ratio and satisfies the temperature limits for the different materials. Table 2 gives the outboard blanket configuration. The BSDOS employed the one-dimensional discrete ordinates code ONEDANT [5] to perform the transport calculations with a P_3 approximation for the scattering cross sections and an S_8 angular quadrature set. A 67-coupled group nuclear data library [6] (46 neutron and 21 gamma) based on ENDF/B-V was employed for these calculations. VITAMIN-E [7] and KAOS/LIB-V [8] libraries were used to obtain this working library. For the heat transfer and the thermal hydraulics analyses, BSDOS used the three-dimensional mesh generator for modeling nonlinear systems INGRID [9] to model one blanket module at the midplane and the three-dimensional *finite element heat transfer code TOPAZ3D [10] to* perform the heat transfer part of the analyses. The physical properties of the different materials were evaluated at each node as a function of the temperature and the density factor. Also, a detailed model is used to calculate the heat transfer coefficients between the different layers as a function of the temperature and the pressure.

The obtained blanket configuration has a 1.6 local tritium breeding ratio and the temperature distributions satisfy the design guidelines for the different materials. The same configuration was subjected to two abnormal fault conditions. The first abnormal condition is to increase the heat transfer resistance at the first breeder surface by a factor of ten between the clad and the Li_2O breeder (between Zone 6 and Zone 7 of Table 2). The second abnormal condition is similar except it is located at the last breeder surface (between Zone 16 and Zone 17 of Table 2). The extreme temperatures of each material in the blanket are given in Table 3. The maximum lithium oxide material is 565.5 C instead of 487.3 C for the normal design which is much lower than the 1000 C design limit. Also, all other temperatures are within the design guidelines.

The second step in the analyses was to study the impact of the lithium-6 enrichment on the blanket performance parameters. The lithium-6 enrichment was varied from 20 to 95%. The tritium breeding ratio changes from 1.47 to 1.60 with ~ 60% of the tritium production in the first breeder layer. Therefore, the use of high

Table 2
Outboard Blanket Configuration

Zone Number	Zone Thickness, cm	Zone Material
2	0.5	First wall steel[a]
3	0.3	Coolant-water
4	0.2	Back wall steel
5	4.5	Multiplier-Be (0.8 DF)
6	0.1	Clad steel
7	0.8	Breeder-Li_2O (0.8 DF)
8	0.1	Clad steel
9	5.5	Multiplier-Be (0.8 DF)
10	0.2	Coolant channel steel
11	0.2	Coolant-water
12	0.2	Coolant channel steel
13	9.0	Multiplier-Be (0.8 DF)
14	0.1	Clad steel
15	0.8	Breeder-Li_2O (0.8 DF)
16	0.1	Clad steel
17	15.0	Multiplier-Be (0.8 DF)
18	0.2	Coolant steel
19	0.2	Coolant-water
20	5	Utility zone steel
	43	Total thickness

[a] Type 316 stainless steel.

lithium-6 enrichment is recommended to achieve the highest possible tritium breeding ratio. For the ITER design at the 3 MW·y/m^2 average DT neutron fluence, the maximum burnup of lithium-6 in the first lithium oxide layer is about 3% which indicates that the tritium breeding ratio is almost constant during the technology phase of ITER. In addition, the results show that the maximum temperature in the solid breeder decreases with the burnup of lithium-6. Such low burnup of lithium-6 allows the blanket to use a thinner lithium oxide layer. Therefore, the impact of changing the thickness of the solid breeder layers was studied.

The thickness of both breeder layers were changed simultaneously from 0.2 to 1.4 cm. In this blanket configuration the tritium breeding has a saturation value of 1.62 at a thickness of 1 cm for both lithium oxide layers. However, the tritium production in the first lithium oxide layer increases as the thickness increases. In the second layer, the tritium production peaks at about 0.6 cm. Also, the energy deposited in each lithium oxide layer has a similar behavior. The minimum temperature of the first lithium oxide layer increases with the thickness because of the increase of the energy deposited. In the second lithium oxide layer the minimum temperature decreases as the lithium oxide thickness increases in both layers. While the maximum temperature in both layers of lithium oxide increases because of the increase in the energy deposited in the lithium oxide layers. In summary, the blanket does not

Table 3
Extreme Temperatures of Each Material in the Outboard Blanket Under Normal and Abnormal Conditions for 1 MW/m^2 Neutron Wall Loading

Zone Material	Min.-Max. Temp. °C Normal Design	Abnormal Condition 1	Abnormal Condition 2
First wall-steel	55.7-155.0	55.7-154.8	55.7-155.0
Coolant-H_2O	40.0- 45.9	40.0- 45.7	40.0- 45.9
Back wall-steel[a]	62.0-123.9	60.4-118.4	62.0-123.9
Neutron multiplier-Be	177.9-351.9	168.3-321.5	177.9-351.9
Clad steel	373.1-386.9	338.8-351.1	373.1-386.9
Solid breeder Li_2O	398.3-487.3	441.1-565.5	398.4-487.3
Clad steel	370.4-385.1	408.9-425.0	370.4-385.1
Neutron multiplier-Be	154.1-349.8	163.1-384.5	156.1-349.9
Coolant channel steel	56.7-111.2	58.0-116.4	56.7-111.3
Coolant-H_2O	40.0- 48.0	40.0- 48.4	40.0- 48.2
Coolant channel steel	53.9-100.6	53.9-100.9	54.4-102.8
Neutron multiplier-Be	135.1-363.7	135.1-363.9	138.8-383.5
Clad steel	376.6-387.1	376.6-387.1	397.8-409.0
Solid breeder-Li_2O	395.4-460.8	395.5-461.0	418.2-496.4
Clad steel	406.9-414.8	407.0-414.9	377.3-384.5
Neutron multiplier-Be	104.5-396.8	104.5-396.9	101.2-368.5
Coolant channel steel	49.3- 80.1	49.3- 80.1	48.8- 78.2
Coolant-H_2O	40.0- 43.9	40.0- 43.9	40.0- 43.7
Utility zone-steel	44.0-166.6	44.0-166.6	44.0-166.5

[a] Type 316 stainless steel.

benefit from increasing the breeder layer thickness above 1 cm for this configuration. Also, the lithium oxide layer thickness can be reduced up to 0.3 cm.

A blanket configuration with 0.4 cm thick lithium oxide layers was examined for neutron wall loading range of 1 to 2 MW/m^2 to define the blanket performance. At 2 MW/m^2, the maximum temperature in the lithium oxide is ~ 900 C which is lower than 1000 C design limit. This shows the capability of the blanket to accommodate a range of neutron wall loading of about a factor of two.

The backup breeder material (Li_4SiO_4) was employed in the blanket configuration given in Table 2 to compare the blanket performance parameters. The results show that both breeders have very similar results except for the maximum temperature in the breeder material. The temperature drop across the solid breeder is doubled in the Li_4SiO_4 case which is related to the difference in the thermal conductivity of both materials.

The inboard blanket configuration has one layer of lithium oxide as shown in Fig. 1. The one-dimensional tritium breeding ratio from this configuration is 0.925. The total thickness of this blanket is about 16 cm where the last steel zone behind the blanket is employed for mechanical and structural purposes.

Thermal-Mechanical Analyses of the Be Multiplier Zones

In the blanket region there are Be multiplier zones with thicknesses (h) ranging from 4.5-15.0 cm in the radial direction. Using modules of ~ 1 m^2 surface area facing the plasma, gives the toroidal and poloidal dimensions of each Be region as ~ 1 m × 1 m. For the sintered product (80% dense), which has been selected for the design, temperature increases (ΔT) across each of the four outboard zones were calculated to be 186, 208, 242 and 308 K.

Based on the temperature distributions within each Be region, stress and deformation analyses were performed to address the impact of possible thermal-stress cracking and Be deformation on the thermal performance of sintered Be. A detailed finite element, thermal-mechanical analysis was performed for the Be region. Although the ΔT values across the Be regions are large, the calculated thermal stresses are small (< 1/4 the yield strength and < 1/6 the ultimate strength) for unrestrained Be. This is due to the fact that the temperature distribution is essentially linear across the Be. The displacement analyses showed that the Be curvature due to the temperature gradient was acceptably small from a heat transfer perspective if the Be were fabricated in the form of sintered cubes (h × h × h, where h is 4.5, 5.5, 9.0, and 15.0 cm for the four respective outboard zones). Upper-bound estimates of temperature uncertainties, due to materials properties, Be/stainless-steel interfaces and Be curvature were also made. The results show that the design guidelines are satisfied. In addition, block-to-block interactions were analyzed. Several design concepts are considered for minimizing the effects of block-to-block interactions.

Tritium Inventory Analysis

In the current design, the inlet hydrogen (protium) molar flow rate is 25 times the tritium generation rate, with a minimum local value of 20. The He flow rate is 26,000 times the tritium generation rate. Under these conditions, the total tritium inventory is 57 g in the Li_2O breeder due to diffusion, desorption, solubility, and gas phase inventories for the 1 MW/m^2 operating condition with T_{min} = 400°C. No separate phase LiOT or LiOH is predicted to occur. If T_{min} is decreased by ~ 40°C while the temperature gradient across each breeder zone is maintained, the inventory due to these four mechanisms increases to 200 g. However, solid phase precipitation of LiOT and LiOH begins to occur at ~ 360°C. As T_{min} is decreased below this temperature, the potential exists for trapping large inventories of tritium until the temperature is increased. Thus, for pulsed operation, it is important that for a significant portion of the burn time, the minimum breeder temperature is above 360°C. For burn/dwell cycles for which this is not true, the purge flow rate needs to be increased to inhibit LiOT formation for T < 360°C. By increasing the He flow rate by a factor of 10, the critical temperature for LiOT precipitation is reduced to 310°C.

In addition to tritium generation, release and retention in the breeder region, there is a low level of tritium generated in the Be multiplier regions. At the end of life, ~ 1 kg of tritium (minus decay) would be in the Be if no tritium were released. Based on crude estimates of irradiation (1 data set) and chemical (1 data set) trapping of hydrogen isotopes in Be, it appears that 80-90% of the tritium generated in the Be will be retained. An experimental program is underway to provide better data on the behavior of hydrogen isotopes in Be.

Activation Analyses and Decay Heat Calculations

The total and the specific values of the activation responses (the radioactivity, the decay heat, and the biological hazard potential) have been calculated for the inboard and outboard blankets and the shields during and after the 3 MWa/m^2 average fluence of the technology phase. The calculations are made using the radioactivity code RACC [11] and the 46 neutron group flux calculated by the neutron transport code ONEDANT.

In the neutron transport calculations, both inboard and outboard blankets and shields have been modeled using one-dimensional cylindrical toroidal geometry. The results are normalized to the average neutron wall loading on each blanket section. The average inboard neutron wall loading values are .641 and 1.024 MW/m^2 for the inboard and outboard sections, respectively.

After the 3 MWa/m^2 fluence, the outboard blanket activity reaches about 1.4 MCi/cm and dominates the activity of the whole machine. About half of this activity is due to ^6He which decays in a few seconds after which the outboard blanket activity (per cm of height) decreases to .4 MCi, .15 MCi, 14 kCi, 460 Ci, and 13 Ci after 1 day, 1 year, 10 years, 100 years, and 1000 years, respectively.

The inboard blanket activity is less than half the outboard blanket activity. However, the radioactivity is more concentrated in the inboard blanket than the outboard blanket. The inboard specific activities are: 7.7, 4.3, 2.8, 1.1, .1, 3×10^{-3} and 2×10^{-4} Ci/cm^3 at shutdown, after 1 hour, 1 day, 1 year, 10 years, 100 years, and 1000 years, respectively. The outboard specific activities at the same respective times are 6.6, 2.5, 1.9, .74, .07, 2.2×10^{-3} and 6.7×10^{-5} Ci/cm^3.

The decay heat calculations show that there is total of 15 kW/cm decay heat in the machine at shutdown, half of which is generated by the short lived ^6He isotope. The total decay heat is dominated by the outboard blanket. However, as in the radioactivity, the specific decay heat of the inboard blanket is larger than the outboard blanket decay heat.

The waste disposal ratings of the inboard/outboard blanket have been calculated according to the U.S. waste disposal regulations [12]. The outboard blanket have been found to satisfy the low level waste Class C limits, while the inboard blanket exceeds the Class C limits.

Mechanical Design

The blanket is divided into 32-inboard and 48-outboard segments. The inboard segments are essentially identical except for piping details, while the outer blanket is split between center and side segments. The side segments lie in the bore of the toroidal field coils and are continuous from top to bottom. The center segments are interrupted at the radial ports. All the segments are installed through the large top ports, but some of the segment piping must be routed through the bottom ports.

The structural attachment design philosophy to minimize the number of interfaces and simplify the types of interface that must be made remotely. Both the inboard and outboard blanket segments are attached to removable sections of the shield, and the blanket-shield assembly is attached at discreet support points to the vacuum vessel. In this way, the shield forms a structural backbone for the blanket. The semi-continuous blanket support can be carefully assembled and controlled outside the reactor, while the remote connections between the shield and vacuum vessel are very simple with statically determinate support reactions. These attachment concepts also allow for thermal growth and swelling of the components. Both sides of the remote attachment are adjustable for initial installation and the shield side is electrically isolated.

The blanket fabrication concept must provide a thin, cooled, first wall and intermediate cooling consistent with the blanket internal heating and temperature requirements. It must provide a stable, dimensionally accurate structure that will withstand the internal pressure and the electromagnetic disruption forces but be compliant enough to allow thermal expansion of the internal components. The proposed concept is a double-wall structure. The "autovac" process would be used to simultaneously form the face sheets and diffusion bond them to the transverse ribs. In this process, the sheets are heated to 1100°C and vacuum formed in a ceramic mold. The result is a fully annealed structure with integral cooling passages. One assembly would form the front and side walls of the box and a second assembly would form the back. These assemblies are welded together. Cooling water would be fed toroidally through the side walls and across the front from the poloidal manifold chambers at the back wall. Detailed sections through inboard and outboard segments are shown in Figs. 1 and 2.

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