CONF-881031--39

To be presented at the Eighth Topical Meeting on Technology of Fusion Energy, Salt Lake City, UT, October 9-13, 1988.

CONF-881031--39

DE89 003593

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by

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August 1988

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

WATER-COOLED SOLID-BREEDER BLANKET CONCEPT FOR ITER[a]

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ABSTRACT

A water-cooled solid-breeder blanket concept was developed for ITER. The main function of this blanket is to produce the necessary tritium for the ITER operation. Several design features are incorporated in this blanket concept to increase its attractiveness. The main features are the following: a) a multilayer concept which reduces fabrication cost; b) a simple blanket configuration which results in reliability advantages; c) a very small breeder volume is employed to reduce the tritium inventory and the blanket cost; d) a high tritium breeding ratio eliminates the need for an outside tritium supply; e) a low-pressure system decreases the required steel fraction for structural purposes; f) a low-temperature operation reduces the swelling concerns for beryllium; and g) the small fractions of structure and breeder materials used in the blanket reduce the decay heat source. It is assumed that the blanket operation at commercial power reactor conditions can be sacrificed to achieve a high tritium breeding ratio with minimum additional research and development, and minimal impact on reactor design and operation.

Operating temperature limits are enforced for each material to insure a satisfactory blanket performance. In fact, the design was iterated to maximize the tritium breeding ratio and satisfy these temperature limits. The other design constraint is to permit a large increase in the neutron wall loading without exceeding the temperature limits for the different blanket materials. The blanket concept contains 1.8 cm of Li_2O and 22.5 cm of beryllium both with a 0.8 density factor. The water coolant is isolated from the breeder material by several zones which reduces the tritium buildup in the water by permeation, reduces the chance for water-breeder interaction, and permits the breeder to operate at high temperature with a low temperature coolant. This improves the safety and environmental aspects of the blanket and eliminates the costly process of the tritium recovery from the water. The key features and design analyses of this blanket are summarized in this paper.

[a] Work supported by the U.S. Department of Energy, Office of Fusion Energy.

INTRODUCTION

A water-cooled solid-breeder blanket concept has been developed for ITER using a low-pressure coolant system. The blanket is designed to produce the necessary tritium required for the ITER operation and to operate at commercial power reactor conditions as much as possible. The other consideration is to minimize the impact on the design and operation of ITER.

Because of the ITER test plan and the desire to reduce the reactor size, a large fraction of the fusion neutrons (less than 0.5) is not available for tritium breeding. This mandates the use of beryllium to multiply the fusion neutrons for tritium production. Beryllium is the only non-fissionable material with adequate neutron multiplication for ITER conditions.[1-3] In order to utilize these neutrons, the solid breeder has to contain a large concentration of the lithium-6 isotope. A 90% lithium-6 enrichment is used. The use of high lithium-6 enrichment has several advantages. It reduces the solid breeder volume required in the blanket. This reduction results in a lower tritium inventory and low blanket cost. Also, the high lithium-6 enrichment insures the same breeding ratio at the beginning and end of life for the blanket (3 $MW \cdot y/m^2$ average fluence).

For satisfactory tritium inventory, the solid breeder temperature has to operate within a specific temperature range. The minimum temperature of this range for any of the solid breeders under consideration (Li_2O, $LiAlO_2$, or $LiSiO_4$) is about 350 to 400°C which does not match the desired operating temperature range for a low-pressure water coolant system. The adopted design approach is to locate the beryllium multiplier between the solid breeder and the water coolant to raise the breeder temperature to the required temperature range without relying on an adjustable gap conductance between the solid breeder and the steel clad. This design approach increases the blanket reliability and simplifies the design configuration. This arrangement is also desirable from the neutronics viewpoint to maximize the tritium breeding ratio. The use of the beryllium zone between the water coolant and the solid tritium breeder increases the number of tritium barriers which reduces the tritium permeation from the solid breeder to the water coolant. Carbon is used at the back of the blanket because of its low absorption cross section as a reflector and moderator. In addition, it operates as a thermal insulator between the cold steel shield and the solid breeder.

The above discussion implies that a multilayer configuration is well suited for this blanket concept. Also, it reduces fabrication costs and simplifies the design. Two thin layers of solid breeder (1.8 cm total thickness) with beryllium layers (22.5 cm total thickness) in between are used. In fact, the solid breeder layers are cladded and inserted in the beryllium material. The use of thin layers of solid breeders results in ~125°C temperature variation in the breeder material which is a small fraction of the solid breeder temperature range required for satisfactory performance. This permits an increase in the neutron wall loading up to 40% from the nominal value without exceeding the maximum allowable temperature of any material. A limit of 800°C is used for Li_2O in this design.

The one-dimensional tritium breeding ratio is 1.7 for this two breeder layers configuration. The energy multiplication factor is about 1.6. The tritium purge system design shows a total tritium inventory of <1.4 g in the Li_2O breeder and the beryllium multiplier assuming no chemical or defect trapping in Be. The maximum partial tritium pressures are the same in the breeder and the multiplier to eliminate the tritium permeation between these materials. The total pumping power required for the helium purge system is 3.5 kW. The radioactivity analysis shows the blanket materials satisfies class C for waste disposal.

CONCEPT DESIGN

Several blanket arrangements were analyzed to determine the tritium breeding potential based on one-dimensional calculations. The one-dimensional discrete ordinates code ONEDANT[4] was used to perform the transport calculations with a P_5 approximation for the scattering cross sections and an S_8 angular quadrature set. A 67-coupled group nuclear data library (46-neutron and 21-gamma) based on ENDF/B-IV was employed for these calculations. Vitamin-C[5] and MACKLIB-IV[6] libraries were used to obtain this library.

Design configurations with three breeder layers and high tritium breeding ratios (greater than 2.0) were developed.[7] The total thickness of the three breeder layers is 6 cm. The thermal hydraulics analysis of this configuration suggested that different gap conductance values be used at the breeder interfaces with the steel clad to maintain the breeder material in the required temperature range.

A simpler configuration has been developed based on two breeder layers (1.8 cm total thickness) as shown in Table 1. This configuration has a 1.7 tritium breeding ratio with 22.5-cm beryllium which results in a net tritium breeding ratio greater than unity for ITER. The first wall/blanket/shield design and optimization system (BSDOS)[8] was employed to carry out the neutronics and thermal hydraulics analyses in an integrated manner where the beryllium zone thicknesses were adjusted for satisfactory breeder temperature limits without using gap conductances. The analysis was performed for a 1.0 MW/m^2 neutron wall loading. The breeder temperatures were adjusted at the bottom of the breeder temperature range for satisfactory tritium recovery. This configuration can accommodate a 40% increase in the neutron wall without exceeding the maximum allowable temperature of any material as shown in Table 1. If an extra margin in this configuration with respect to the changes in the neutron wall loading is required, the reduction of the breeder zone thickness by 1 mm increases the margin by ~10%.

The mechanical design of the blanket is shown in Figs. 1 and 2. A whole sector is shown in Fig. 1 where the reactor is divided into 16 sectors. On the outboard, the sector consists of two sections. Each section has its coolant and tritium purge lines, and it can be removed separately from the reactor. The water coolant flows in the toroidal direction inside the sector as shown in Fig. 2. Two poloidal manifolds are integrated in the shield at the back of the sector where the water manifolds are utilized for shielding purposes. In the inboard section, the water flows only in the poloidal direction as shown in Fig. 1. Coolant panels or water tubes can be used to cool the blanket. The outboard shield is integrated with the blanket, and it is designed to achieve a 2.5 mrem/h dose one day after shutdown in the reactor hall.

Table 1. Blanket parameters for two breeder layers configuration

Zone	Material	Thickness, cm	Min.-Max. Temp., °C 1 MW/m^2	1.4 MW/m^2
First Wall	Type 316 SS	0.5	76-194	79-203
Coolant	H_2O	0.3		
Back Wall	Type 316 SS	0.2	91-163	111-208
Multiplier	Be (0.8 DF)	5.0	157-388	201-558
Clad	Type 316 SS	0.1	386-400	555-572
Breeder	Li_2O^a (0.8 DF)	0.8	397-521	569-782
Clad	Type 316 SS	0.1	426-437	627-636
Multiplier	Be (0.8 DF)	7.5	133-427	168-625
Coolant Channel	Type 316 SS	0.2	80-139	95-177
Coolant	H_2O	0.2		
Coolant Channel	Type 316 SS	0.2	72-121	84-151
Multiplier	Be (0.8 DF)	10.0	115-390	143-565
Clad	Type 316 SS	0.1	389-398	564-574
Breeder	Li_2O^a (0.8 DF)	1.0	397-518	573-795
Clad	Type 316 SS	0.1	526-528	792-795
Reflector	C	30.0	74-525	87-792
Shield	Steel Shield	88.7		

[a] A 90% lithium-6 enrichment.

Table 2. Steel inboard shield for the water-cooled solid-breeder blanket

Zone	Thickness cm	Composition Fraction by Volume
Shield Arrangement		
FW	1	0.7 Type 316 SS, 0.3 H_2O
Steel	34	0.9 Type 316 SS, 0.1 H_2O
TiH_2	17	0.05 Type 316 SS, 0.95 TiH_2
Pb	3	Pb
Total	55	
Shield Performance[a]		
Maximum dose in the insulator material, rads		1.0×10^{10}
Maximum fast fluence in the superconductor, n/cm^2		9.8×10^{18}
Total heat load in the TF coils, kw		51.9
Maximum dpa in the copper stabilizer, dpa		7.9×10^{-3}

[a] Values are calculated at the end-of-life and no credit is taken for the 2-cm carbon tile.

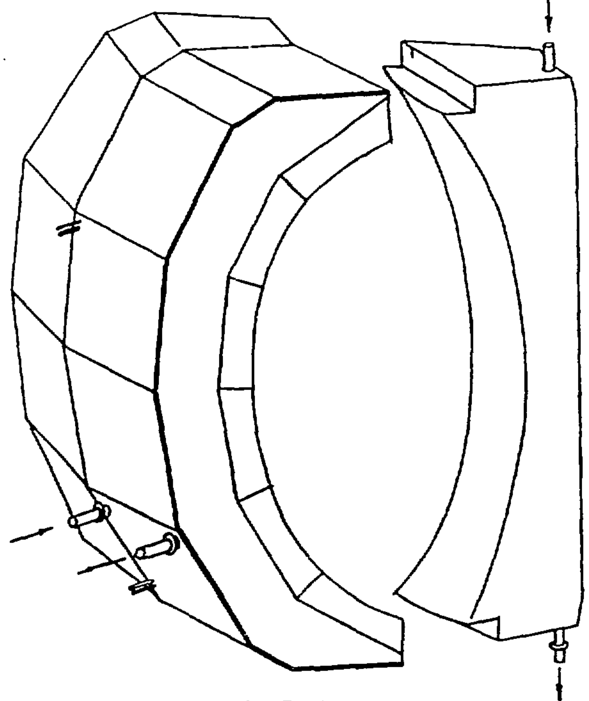

Fig. 1. Water-cooled solid breeder blanket sector

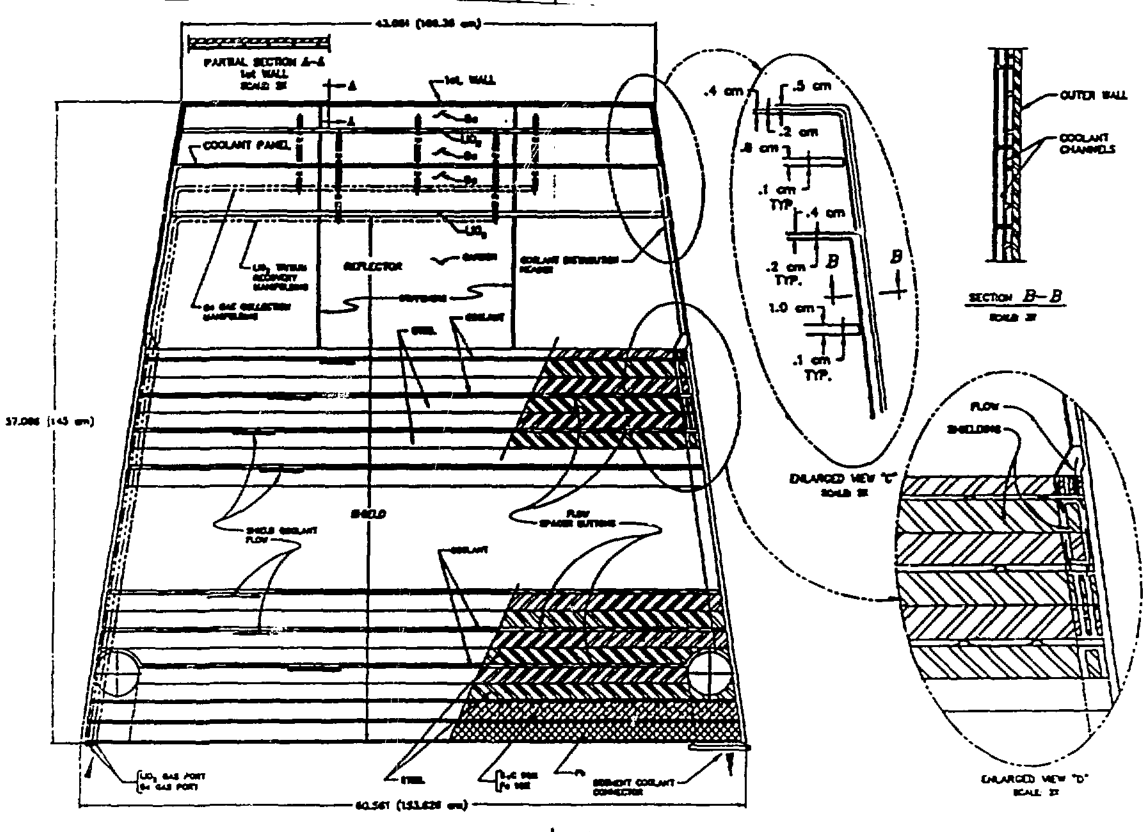

Fig. 2. Water-cooled solid-breeder blanket

The radioactivity analysis for this blanket shows that it is qualified for a class C waste at the end of life based on the Nuclear Regulatory Commission regulations and 3 $MW \cdot y/m^2$ average fluence. The total carbon-14 production is 0.69 and 0.99 g in the Li_2O breeder and the water coolant, respectively from $^{17}O(n,He)^{14}C$ reaction. The corresponding ^{14}C concentration in these materials are 0.79 and 0.12 curies/m^3 which qualify the blanket for class A waste based on ^{14}C only.

The inboard shield configuration used for the water-cooled solid-breeder blanket has type 316 SS, H_2O, TiH_2, and lead shielding materials. The total shield thickness is limited to 55 cm. The zone thicknesses in the shield were optimized to satisfy the design guidelines.[9] The compositions are not optimized yet. The shield performance is also shown in Table 2.

The tritium inventory formalism presented in the ITER nuclear design guidelines[9] was used to calculate tritium inventory in the Li_2O breeder. Input to these models and correlations was: tritium generation rate ($\dot{G}$) = 57.7 (g/day)/(MW/m^2), grain size (r_g) = 10 μm, temperature profile (T) = $T_{max} - 4 (T_{max} - T_{min}) x^2/L^2$, k, where $-0.5 L<x<0.5 L$, maximum H_2O and HTO partial pressures of P_{H_2O} = 29 Pa and P_{HTO} = 5 Pa, and the total breeder mass (M_b) = 5.1 x 10^6 g. The assumptions and results are shown in Table 3. Under conditions of protium swamping, the major contribution to the inventory is due to solubility. The overall uncertainty in these calculations is taken to be a factor of 2. While the solubility is known to much better accuracy than this, the overall estimate includes uncertainties in the purge flow analysis which determines the partial pressures of H_2O and HTO. With regard to diffusion and surface desorption, even a two-order-of magnitude uncertainty in these would only increase the calculated inventory by 1 g in the Li_2O.

Table 3 also summarizes the results of the Be calculations using the methodology established for the ITER nuclear design guidelines.[9] The data of Jones and Gibson[10] (based on unirradiated arc-cast Be) in the range of 300-900°C indicate that the diffusivity of tritium in Be is quite high and the solubility is quite low, consistent with the very low tritium inventory shown in Table 3. However, for decreasing test temperatures, increasing fractions of tritium remained in their samples even after 50-100 hours of annealing. This suggests the possibility of chemical trapping due to impurities. Also, it has been observed[11] that Be specimens (containing 1-3 wt.% BeO), which have been bombarded by deuterons, release tritium very slowly from room temperature up to ~350°C. Using an upper-bound envelop of these two data sets for tritium which may be trapped in Be at impurities and radiation-induced defects results in the estimates in the footnote of Table 3 of 735 g (1 MW/m^2) and 868 g (1.4 MW/m^2) at the end of life. Clearly more work needs to be done in this area to characterize tritium retention in Be as a function of impurities and defects.

From tritium inventory considerations, it is desirable to introduce protium in the purge stream in quantities large enough to swamp the amount of tritium generated. This protium swamping effectively minimizes the amount of tritium adsorbed on the surfaces and dissolved in the bulk of the breeder (or multiplier). A ratio of protium flow rate to tritium generation rate of ~30 has been chosen to minimize tritium inventory in the breeder. The combined effect of this "protium swamping" and the stainless steel cladding means that the tritium in the gas phase is mostly in the HT molecular form. With tritium in the HT form, there is a greater potential for tritium permeation through the breeder cladding. The Be purge design is such that the HT partial pressures in the breeder purge and the Be purge are about the same; there is very little driving force for tritium leakage out of the breeder plates. Also, at the Be/H_2O boundary, the stainless steel cladding temperature is quite low, yielding a very small leakage rate from the Be plates to the coolant.

Table 3 summarizes the design and operating conditions for the helium purge system for both Li_2O breeder and Be multiplier. While the breeder requires semi-circular notches at the breeder/clad interface because of its high tritium generation rate, the Be multiplier with its low generation rate does not require such holes. For each module, a single inlet tube is required for the breeder layers and the multiplier layers.

KEY TECHNICAL ISSUES

The water-cooled, solid-breeder blanket developed for ITER makes use of current technology and data bases. However, several key technical issues require further investigations to provide extra confidence in the design and to insure a satisfactory blanket performance. The issues for this class of blanket are the following:

- Lithium-6 burnup and irradiation effects impact on the properties of the solid breeder materials are not completely known. During operation, the middle section of the solid breeder plate will have a compressive stress and a high temperature which may result in changes in the microstructure of the solid breeder materials. The main concerns are the changes in the tritium release characteristics and the thermal conductivity of the solid breeders.

Table 3. Main tritium parameters

Parameter	Li_2O 1 MW/m^2	Li_2O 1.4 MW/m^2	Be 1 MW/m^2	Be 1.4 MW/m^2
Mass, 10^6 g	5.2		63	
Density, % TD	80		80	
Grain Size, μm	20		20	
Pore Diameter, μm	4.8		4.8	
Purge Holes Diameter, mm	0.36		-	
Purge Holes #/cm^2	1		-	
Tritium Generation Rate, g/day	57.7	80.7	0.90	1.26
T_{min}, °C	397	569	115	143
T_{max}, °C	521	795	427	625
Average Purge Temperature, °C	397	571	311	444
Permeability, 10^{-12} m^2	4.2	4.2	4.0×10^{-2}	5.0×10^{-2}
Helium Flow Rate, moles/s	0.500	0.700	3.8×10^{-2}	5.3×10^{-2}
Helium Pressure, MPa				
Inlet	0.157	0.199	0.133	0.148
Outlet	0.101	0.101	0.101	0.101
Total Pumping Power, kw	1.23	3.33	0.05	0.12
Hydrogen Flow Rate, moles/s	3.32×10^{-3}	4.65×10^{-3}	1.53×10^{-4}	2.14×10^{-4}
Hydrogen Pressure, Pa				
Inlet	1040	1320	534	593
Outlet	631	631	394	394
Maximum Tritium Pressures, Pa				
HT	61.9	62.9	62	62
T_2	-	-	2.9	2.9
HTO	5.04	4.15	-	-
T_2O	0.22	0.14	-	-
Tritium Inventory, g				
Diffusion	6.62×10^{-3}	2.51×10^{-3}	6.28×10^{-6}	8.05×10^{-6}
Desorption	2.03×10^{-2}	7.38×10^{-3}	-	-
Solubility	1.07×10^{-2}	9.36×10^{-1}	2.80×10^{-1}	2.80×10^{-1}
Gas Phase	1.01×10^{-2}	8.09×10^{-3}	1.30×10^{-1}	1.06×10^{-1}
Trapping	-	-	(a)	(a)
Total	4.48×10^{-2}	9.54×10^{-1}	4.10×10^{-1}	3.86×10^{-1}

[a] These calculations assume relatively pure Be and no radiation-defect trapping of tritium in Be. Based on upper-bound estimates of chemical and radiation-defect trapping from two sets of data,[10,11] as much as 735 g of tritium may be trapped in the 1 MW/m^2 case and 868 g for the 1.4 MW/m^2 case.

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Also, the change in the gap conductance at the breeder-to-structure interface is another important factor for the blanket performance.

- The data base for tritium retention in Be is inadequate, leading to a large uncertainty in the upper-bound estimate of tritium inventory. More experimental work is needed to characterize tritium retention as a function of impurities, defects, and operating conditions.

- Irradiation data base for austenitic steel at a low temperature (50 to 350°C) at an appropriate He to dpa ratio needs to be determined. This issue applies to all blanket and shield options for ITER.

- Stress corrosion cracking of austenitic steel in water blanket options needs to be assessed. Especially, the design solutions developed to mitigate stress corrosion in light water reactor materials which involve: a) stringent water chemistry control, b) modification of residual stress distributions in weldments, and c) more resistant materials, e.g., type 316NG, need to be assessed for fusion conditions.

- The ability of the blanket to accommodate a large change in the neutron wall loading greater than 50% of the nominal value needs to be addressed if ITER design requires this feature.. The main concern is the maximum temperature of the breeder materials will exceed the design temperature range during the transient period.

CONCLUSIONS

The main conclusion from this study is that a low pressure high tritium breeding blanket based on the water-cooled solid breeder concept is a very promising concept for ITER. This concept requires a low first wall coverage and provides adequate tritium for ITER operation without an external tritium source.

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