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UWFDM-824

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ABSTRACT

Aqueous lithium salt blankets have some very attractive features which make them likely candidates for use in the next generation of fusion devices such as ITER and beyond. The main attraction derives from the fact that such a blanket can go from a non-tritium-breeding state to a tritium breeding state by the simple dissolution of a lithium salt into the cooling water. Such a blanket is also relatively simple to design and analyze, and is amenable to accommodating penetrations. On the other hand, the salt solution raises concern with respect to radiolysis, electrolysis and corrosion, as well as safety issues related to them. This paper describes the design of the U.S. entry of an aqueous salt blanket for ITER.

1. INTRODUCTION

The use of an aqueous lithium salt solution as a coolant in a fusion reactor blanket was first proposed by Steiner, et al. in 1985 [1]. Since then it has attracted considerable attention and has been used in at least one reactor conceptual design study [2]. Corrosion studies utilizing aqueous lithium salts have become part of the U.S. R&D effort and similar studies are taking place in Europe and Canada. The extensive experience of the Canadians in the area of tritium recovery from water is directly applicable.

Although the use of aqueous lithium salt blankets (ALSB) has not been excluded for commercial fusion reactors, their use has greater relevancy in near term devices such as ITER and NET [3], which have extensive physics phases followed by technology phases. During the physics phase when there is no need for tritium breeding, the blanket is cooled with water. As the project moves into the technology phase and tritium breeding is needed, the appropriate salt is dissolved in the water so that tritium can be bred. A tritium recovery system can be provided sometime prior to initiation of the technology phase. It is easy to see how such a scheme can be effective in saving both time and money. This is particularly true if the same plasma configuration can be used in both the physics and technology phases, in which case no blanket or first wall changeout will be required. As in any high tritium breeding ratio blanket, the ALS blanket requires large neutron multiplication which can be achieved with the addition of Be or Pb metals.

The use of Be is preferred because of its higher neutron multiplication and the avoidance of the generation of radioactive polonium in Pb. The neutron multiplier can be placed in the blanket during initial assembly and used effectively during the technology phase.

There are several other advantages of the ALS blankets, among the most prominent being its inherent simplicity. In the original ITER design, rectangular box containers were filled with Be and steel balls and the aqueous salt coolant at low temperature and pressure simply flowed through the pebble bed performing both cooling and breeding. Since then, the problem of radiolysis has emerged indicating that a higher pressure will be needed if the products of radiolysis are to be kept in solution. The design evolved into a modified version, in which the aqueous salt solution is confined to cylindrically shaped containers capable of high pressure operation and the rest of the blanket is cooled with low pressure water. Beryllium rods are used instead of pebbles to maximize the Be fraction while keeping the pressure drop at a reasonable value. This design underwent further modifications in an attempt to further reduce the salt operating pressure, the surface area in contact with the salt and the overall salt inventory which directly determines the tritium inventory. The latest design is described in this paper.

Two lithium salts have been considered for use, $LiNO_3$ and LiOH. Both salts can provide adequate breeding at concentrations below the solubility limits in cold water. At temperatures below 100°C, corrosion does not appear to be a problem for either salt although at higher temperatures, the LiOH is predicted to be more corrosive than $LiNO_3$. A perceived problem for the $LiNO_3$ is the generation of ^{14}C from neutron interaction with ^{14}N. If the ^{14}C is not removed from the solution, its accumulation over time will become excessive. Further, whereas a H_2 and O_2 overpressure can keep radiolysis products in solution in LiOH, it is not clear that the same will work for $LiNO_3$. Since for ITER conditions, the coolant will always be below 100°C, we have decided to choose LiOH as the breeding salt for the reference blanket. In the next several sections we describe the reference design for the outboard and the inboard blankets for ITER. Issues of corrosion, electrolysis and safety will not be addressed in this paper.

2. GENERAL DESCRIPTION

Figure 1 is a cross section of ITER in the technology phase. The outboard (OB) blanket is divided into 48 modules, three of which fit between the centerlines of adjacent TF coils. The center module of the three has a penetration at the mid-plane extending 110 cm in the toroidal direction. For this reason, the central module is split into two halves, an upper and a lower. The space in between will be occupied by a test module or some other necessary equipment. The two side modules will be only 85 cm wide and will extend the full height of the blanket. The depth of both central and side modules is 85 cm. Since, on the OB side, the toroidal extent of the blanket varies from top to bottom, we will use the average width for each, namely 95.7 cm for the central module and 81.4 cm for the side modules. The parameters in the tables are normalized to these widths. The inboard (IB) blanket is divided into 16 modules 137 cm wide and 71.5 cm deep. Only the front 23.5 cm depth constitutes the FW and blanket and the remainder is shield and manifolding.

The blanket is cooled with two coolants, water and salt solution. The water is used to cool the first wall and the rear shield zones and the salt is used to cool the beryllium neutron multiplication zones. The first wall design is the same for the IB and OB blankets with the exception that the IB side will have a 2 cm layer of graphite tiles for disruption protection. Modification of the IB first wall will be needed to accommodate tile attachment.

The first wall consists of three layers of stainless steel separated by two water channels. Dimensions and radial build for both IB and OB blankets are given in Figure 2a and 2b. The blanket zone which follows immediately behind the first wall has layers of solid Be blocks divided by flattened tubes of aqueous salt running in the poloidal direction. Toroidally, the blanket zone is separated by radial structural plates which are welded to the first wall and to the back plates. The back plates are bolted to the rear structure of the shield, thus providing support for the first wall to withstand the pressure of the salt channels as well as the disruption loads. It is important to point out that the Be plates are not counted on for load transmission. That is to say, the first wall can withstand the pressure and disruption forces even if the Be blocks are fragmented to pieces.

Figure 3 is a cross section of an IB blanket module. The radial build shows a 2 cm layer of graphite followed by the first wall which has an overall thickness of 1.6 cm of which 0.6 cm is water and

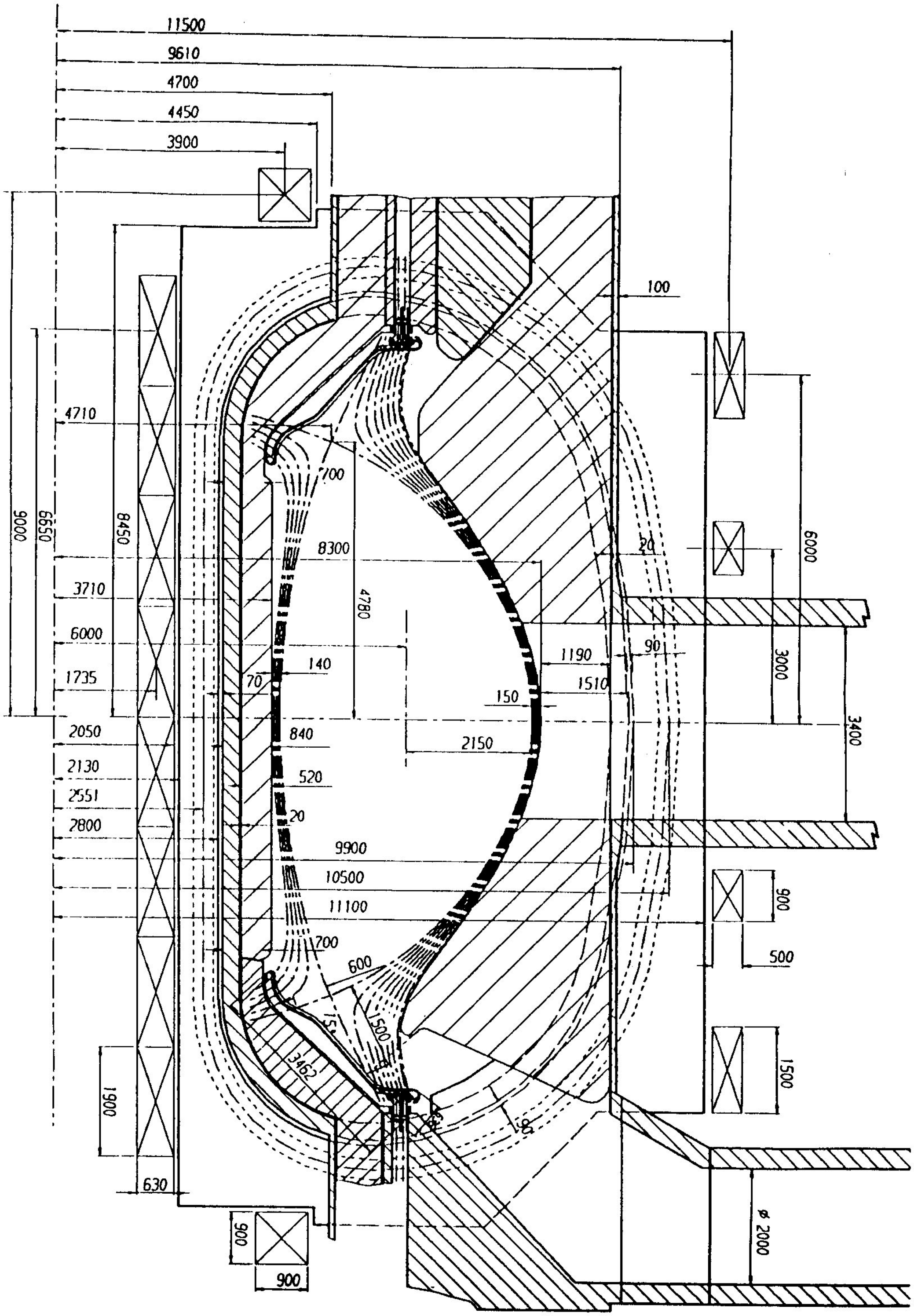
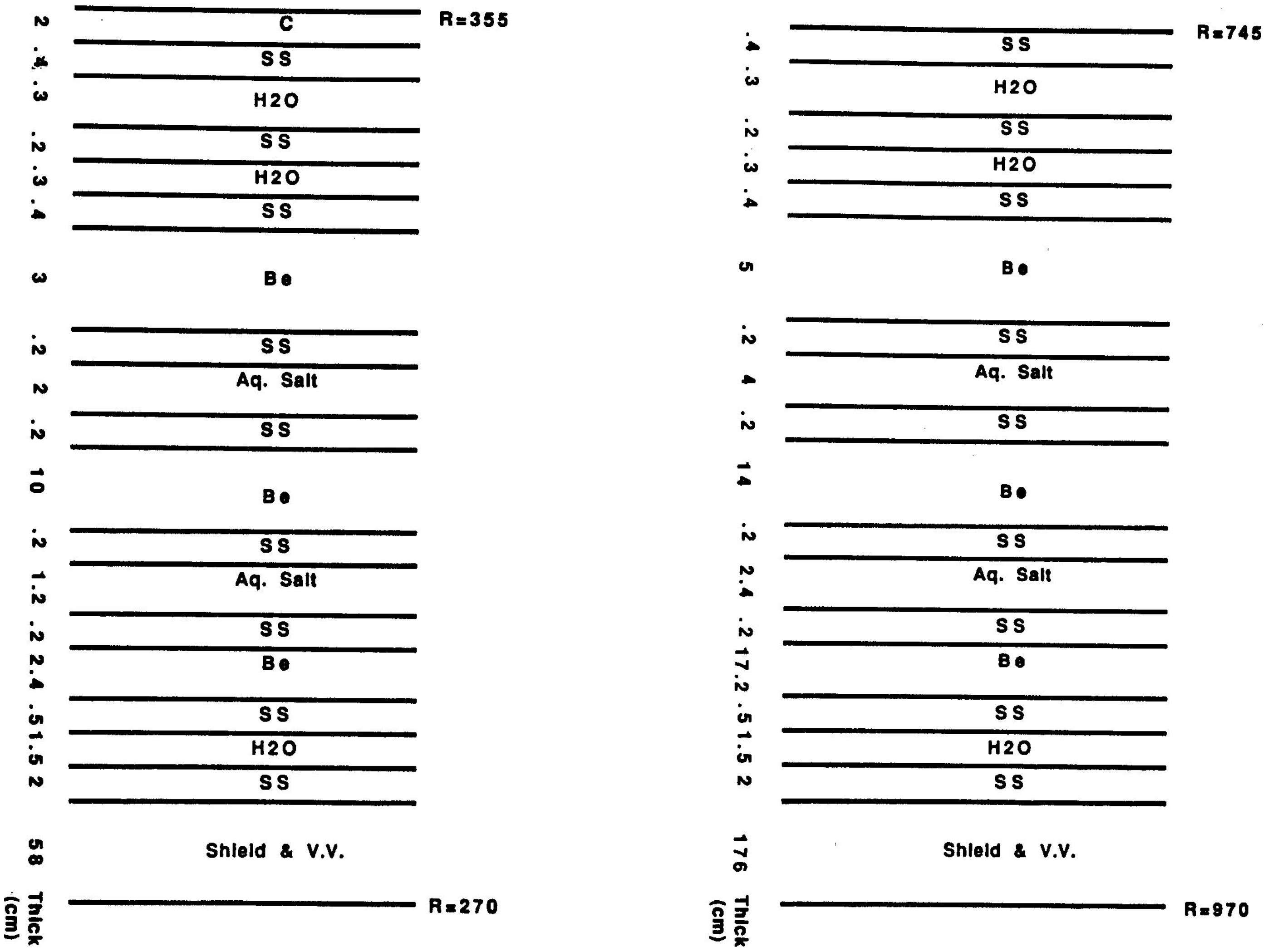


Fig. 1. Cross section of ITER in the technology phase.



a)

b)

Fig. 2. a) Inboard blanket radial build. b) Outboard blanket radial build.

the rest steel. There are nine toroidal compartments which contain Be blocks separated by aqueous salt channels. The rear of the blanket contains steel shield material and water manifolds. Salt coolant is admitted at the bottom of the module, is split into two parts feeding the front and rear salt channels finally recombining together and exiting at the top of the module. The extreme rear manifold in the shield feeds and collects water to and from the front first wall channel. This water also enters at the bottom of the module, flows poloidally in the manifold, then flows toroidally through the first wall and, after collecting in the exit manifold finally exits at the top of the module. An intermediate manifold in the same way feeds water to the second coolant channel in the first wall, and is then routed through the rear shield zones picking up the heat generated there before also exiting at the top of the module.

Figure 4 is a mid-plane cross section of an OB side module. The design is very similar to the IB blanket. The central OB modules will have a different salt and water routing because the upper half will have both supply and return lines at the top and the lower half at the bottom. Thus, the central modules have the same width front and rear salt channels with the salt traveling first in the rear channel then turning around and traveling back through the front channel. Water supply and return manifolds are not affected since they can be accessed from either direction with relative ease.

Aqueous salt flow through the IB blanket modules and through the OB side and central modules has been optimized with respect to the mass flow rate in order to minimize operating pressure while maintaining a channel width which places electrolysis potentials within acceptable limits on the potentiodynamic polarization curve. Operating pressure is determined by the pressure drop in the blanket and by the pressure needed to suppress radiolytic decay products from being released. The higher the mass flow rate the lower the pressure needed to suppress release of radiolysis products, but at the expense of a higher pressure drop, larger supply pipes and a higher overall salt inventory. A careful optimization of blanket parameters makes it possible to operate the aqueous salt at a maximum pressure of 1.15 MPa for both IB and OB blankets. The highest velocity is in the IB first salt channel and is equal to 4.7 m/s. The total aqueous salt inventory in the system is 104 m^3, of which 41.5 m^3 is in the reactor including the ring headers, and the rest outside the reactor. A total of 2400 m^2 of surface area is in contact with the salt in the reactor.

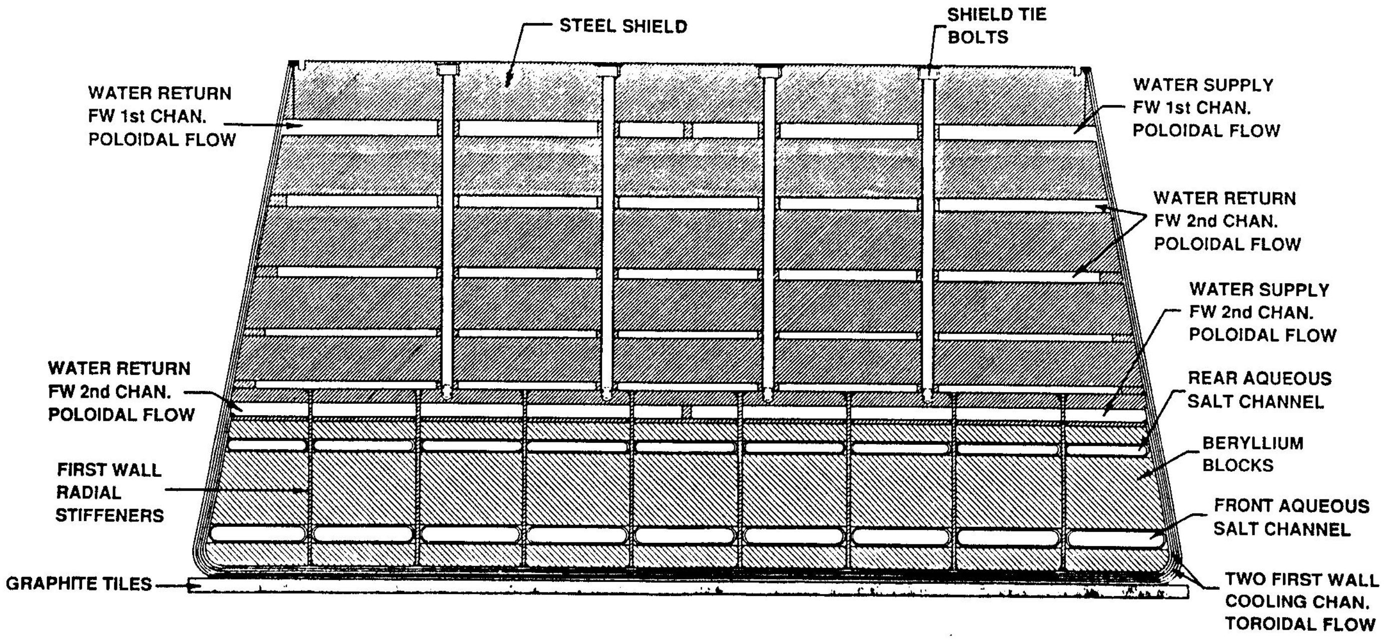


Figure 3. Midplane Cross-section of Inboard Blanket Module

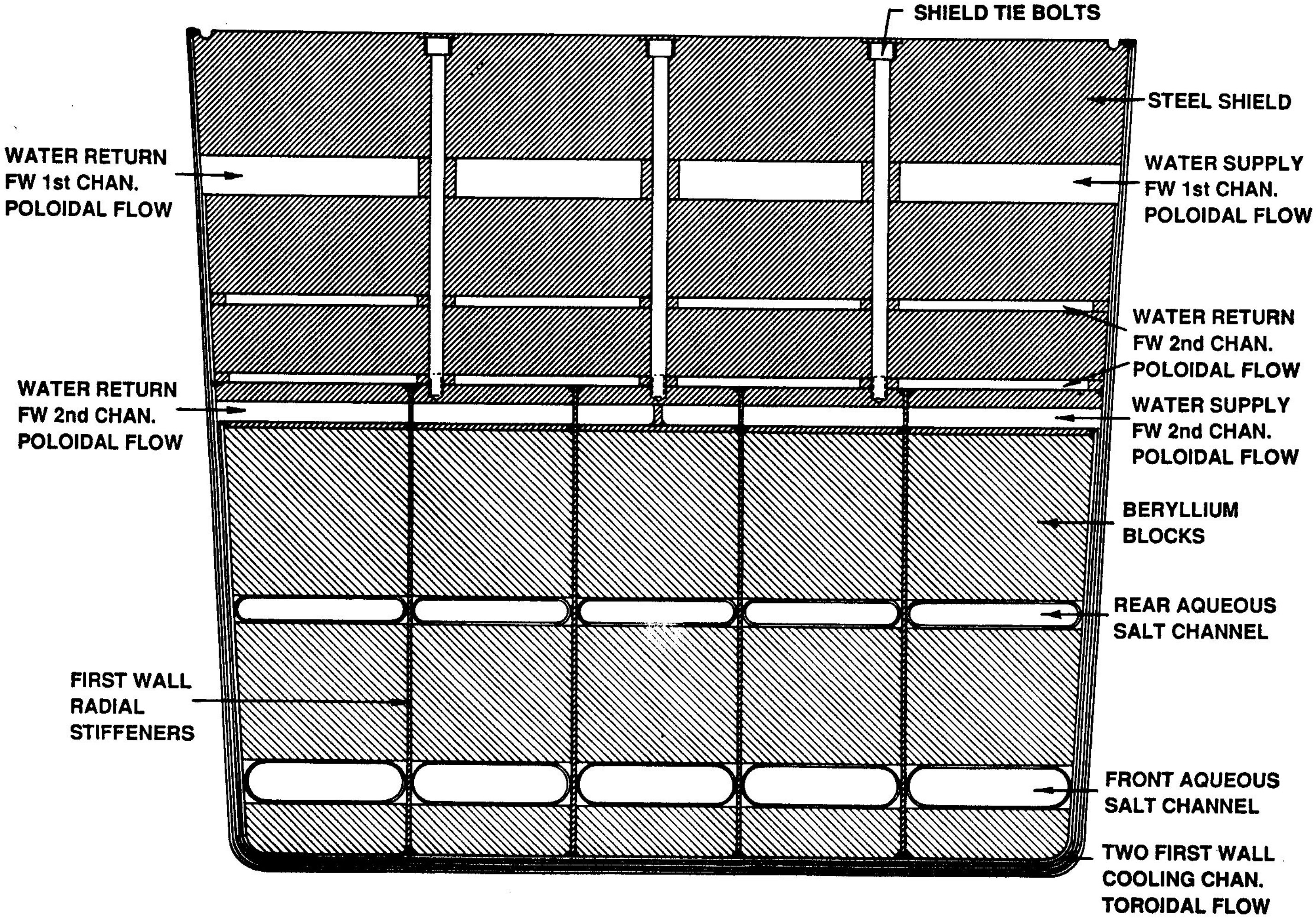


Figure 4. Midplane Cross-section of Outboard Blanket Module

Similarly the water velocity in the first wall channel has been optimized to maintain a low system pressure while keeping component temperatures at reasonable values. Radiolysis effects in the water coolant also have to be taken into account and have dictated that flow in the first wall should be in the toroidal direction to minimize the time spent in the high radiation zone. The maximum water velocity occurs in the IB first channel and is equal to 2.40 m/s, and the highest system pressure required is in the OB side module, equal to 0.3 MPa. Accounting for an additional pressure drop of 0.1 MPa in the outer loop and a safety factor against boiling of another 0.15 MPa, we have designed the water coolant system to operate at 0.55 MPa.

3. NEUTRONICS

The IB and OB blanket configurations have been optimized to maximize the tritium breeding ratio (TBR). Figures 2a and 2b give the optimum radial build for the IB and OB blankets, respectively. These configurations result in local TBR values of 1.03 and 1.44 for the IB and OB blankets, respectively. A one-dimensional toroidal cylindrical model, in which the IB and OB regions are modeled simultaneously, has been used to determine the nuclear heat deposition in the different zones. The peak neutron wall loading is 1.13 MW/m^2 for the IB blanket and 1.59 MW/m^2 for the OB blanket. The average values are 0.72 and 1.06 MW/m^2 for the IB and OB blankets, respectively. Tables 1 and 2 give the nuclear heating in the IB and OB blankets, respectively. The results are given for the peak conditions at the reactor midplane. The total thermal power has been determined to be 1180 MW. Detailed three-dimensional neutronics calculations indicated that the achievable overall TBR is 0.88 with breeding in the test modules excluded. Table 3 gives the tritium breeding calculated in the different regions of the ALSB. Numbers in parentheses indicate the statistical uncertainty in the Monte Carlo calculations. If tritium bred in the test modules is accounted for, an overall TBR of 0.96 is achieved. The calculational model includes assembly gaps, copper side structure, structural stiffeners, divertor plates, and vacuum pumping ducts in addition to 16 radial ports for plasma heating, current drive, and test modules.

3.1 Pressure Needed to Keep Gaseous Radiolysis Products in Solution

The amount of radiolysis products generated in the aqueous salt depends on the rate of energy deposition by the different types of ionizing radiation. The primary yields (G-values for the production of radiolytic products) are different for the different types of ionizing radiation depending on their linear energy transfer (LET) values [4]. The dose rate contributions from gammas, recoil protons, tritons, and alpha particles have been determined in the neutronics calculations. The dose for the aqueous solution in the front and back channels of the OB and IB blankets is given in Table 4 per DT neutron.

The G-values for the production of hydrogen are 0.45, 1.1, 1.0, and 1.7 molecules per 100 eV absorbed energy from γ, p, t, and α, respectively.[4,5]. Using the results of Table 4 indicates that the effective primary yields for H_2 production in the front and back channels of the OB blanket are 1.2 and 1.24 molecules of H_2 per 100 eV of absorbed energy, respectively. The corresponding values for the IB blanket are 1.21 and 1.23. It will be assumed that one O_2 molecule is produced for every two H_2 molecules produced, implying that the G-value for O_2 will be taken as half that for H_2. It should be noted that using these effective primary yields leads to a conservative estimate of the pressure required to suppress the release of H_2 and O_2 since no back reactions between these molecular products and the radical intermediates produced by the low LET radiation are considered.

For a fusion power of 880 MW, the total absorbed dose rates in the front and back channels of the side OB blanket module are 4.76×10^{24} and 1.37×10^{24} eV/s, respectively. The dose rates in the aqueous salt channel of the top and bottom segments of the central OB blanket module are 1.85×10^{24} and 1.64×10^{24} eV/s, respectively. The dose rates in the front and back channels of the IB blanket are 2.41×10^{24} and 7.87×10^{23} eV/s, respectively.

Using Henry's law constants and considering the solubility of H_2 and O_2 in the LiOH solution to be 60% of that in pure water, the pressure required to keep the H_2 and O_2 produced by radiolysis in solution was determined for the different aqueous salt channels. This pressure is inversely proportional to the flow velocity, V (m/s). The required pressure in MPa is given by 4.66/V and 2.3/V for the front and back channels of the side OB module, respectively, and 1.73/V and 1.53/V for the aqueous salt in the top and bottom segments of the central OB module, respectively. The pressure values for the front and back channels of the IB blanket are 4.92/V and 2.67/V, respectively. These relationships have been used to determine the optimum flow velocity in each aqueous salt channel that minimizes the system pressure which includes, in addition to the above mentioned pressure, the vapor pressure of water and the friction pressure drop.

4. THERMAL HYDRAULICS

The thermal hydraulics of the ALSB is relatively straightforward. Nuclear heating determined by neutronic analysis is given in Table 1 and 2. Both aqueous salt and water mass flow rates have been optimized to give the lowest operating system pressure. Once the velocities are fixed for the various blanket zones, heat transfer coefficients can be determined and temperature profiles calculated.

The water velocity in the first channel of the first wall has been selected to be 2.4 m/s and the velocity in the second channel, 1.4 m/s for all the different blanket modules. Table 5 gives the thermal hydraulic parameters for the aqueous salt blanket. The water inlet temperature is 40°C and the highest average outlet temperature is in the IB second cooling channel, and is equal to 66°C. The maximum stainless steel/water interface temperature occurs at the IB first wall and is equal to 96°C.

The highest pressure needed to overcome friction and suppress the release of radiolysis products is 0.31 MPa. Accounting for the pressure drop outside the reactor and an additional factor for safety against boiling at hot spots, we have designed the system pressure at 0.55 MPa. The total water mass flow rate in the reactor is equal to 6500 kg/s.

The aqueous salt inlet temperature is also 40°C and the highest outlet temperature is in the OB upper central module, equal to 60.5°C. It can be seen from Table 5 that the pressure drop within the blanket is trivial because of the well defined channels. The largest contribution to the aqueous salt system pressure comes from that needed to keep the products of radiolysis in solution, making the total system pressure equal to 1.15 MPa. Both IB and OB blankets have identical cooling systems.

Table 1

Nuclear Heating in Inboard Blanket at Midplane

(Γ = 1.13 MW/m^2)

Zone	Thickness (cm)	MW/cm	W/cm^3
Graphite Tiles	2	0.0330	7.40
First Wall			
SS	0.4	0.0146	16.46
H_2O	0.3	0.0078	11.82
SS	0.2	0.0070	15.72
H_2O	0.3	0.0073	11.03
SS	0.4	0.0129	14.54
Blanket			
Be	3	0.0370	5.61
SS	0.2	0.0049	11.10
Aq. salt	2	0.1015	23.26
SS	0.2	0.0042	9.70
Be	10	0.0554	2.59
SS	0.2	0.0025	6.00
Aq. salt	1.2	0.0328	12.97
SS	0.2	0.0023	5.49
Be	2.4	0.0070	1.39
Coolant Manifold			
SS	0.5	0.0052	5.01
H_2O	1.5	0.0048	1.54
SS	2	0.0165	3.98
Shield & V.V.	58	0.0400	0.37

Table 2

Nuclear Heating in Outboard Blanket at Midplane

($\Gamma = 1.59$ MW/m^2)

Zone	Thickness (cm)	MW/cm	W/cm^3
First Wall			
SS	0.4	0.0401	21.46
H_2O	0.3	0.0246	17.41
SS	0.2	0.0188	20.09
H_2O	0.3	0.0231	16.37
SS	0.4	0.0344	18.29
Blanket			
Be	5	0.1775	7.54
SS	0.2	0.0110	11.59
Aq. salt	4	0.3839	20.26
SS	0.2	0.0087	9.13
Be	14	0.1541	2.30
SS	0.2	0.0042	4.38
Aq. salt	2.4	0.1107	9.52
SS	0.2	0.0037	3.77
Be	17.2	0.0527	0.63
Coolant Manifold			
SS	0.5	0.0056	2.27
H_2O	1.5	0.0031	0.41
SS	2	0.0171	1.72
Shield & V.V.	176	0.0290	0.03

Table 3

Tritium Breeding (Tritons/DT Fusion) in the Different Regions of ALSB

Inboard Blanket	
Front Channel	0.153 (±1.4%)
Back Channel	0.059 (± 2.1%)
Total	0.212 (± 1.2%)
Outboard Side Modules	
Front Channel	0.361 (±0.9%)
Back Channel	0.145 (± 1.5%)
Total	0.506 (± 0.8%)
Outboard Top Central Modules	
Front Channel	0.077 (± 2.1%)
Back Channel	0.029 (± 3.5%)
Total	0.106 (± 1.9%)
Outboard Bottom Central Modules	
Front Channel	0.039 (±2.9%)
Back Channel	0.015 (± 4.8%)
Total	0.054 (± 2.6%)
Total in IB and OB Modules	0.878 (± 0.6%)

Table 4

Absorbed Dose in Aqueous Solution per DT Neutron

Radiation	Absorbed Dose (eV/DT neutron)			
	OB		IB	
	Front	Back	Front	Back
γ	3.69×10^5	9.23×10^4	2.60×10^5	8.65×10^4
p	2.09×10^6	2.70×10^5	1.29×10^6	2.49×10^5
t	2.75×10^6	9.88×10^5	2.32×10^6	8.46×10^5
α	2.06×10^6	7.42×10^5	1.74×10^6	6.36×10^5
Total	7.27×10^6	2.09×10^6	5.61×10^6	1.82×10^6

Table 5

Thermal Hydraulic Parameters

	INBOARD	OUTBOARD CENTRAL UPPER LOWER	OUTBOARD CENTRAL SIDE	OUTBOARD
General Parameters				
No. of modules in reactor	16	16	16	32
Width at midplane (cm)	137	N/A	N/A	85
Vertical extent (cm)	970	390	240	970
Toroidal segments/module	9	6	6	5
Max. neutron wall loading (MW/m^2)	1.13	1.59	1.59	1.59
Avg. neutron wall loading (MW/m^2)	0.72	1.06	1.06	1.06
Aqueous Salt Parameters				
No of channels/module	18	12	12	10
Width of channel at midplane (cm)	14.2	12	12	16.3
Depth of front channel (cm)	2.0	2.0	2.0	4.0
Depth of rear channel (cm)	1.2	2.0	2.0	2.4
Inlet salt temp.(°C)	40	40	40	40
Velocity in front channel (m/s)	4.7	1.58	1.4	4.35
Velocity in rear channel (m/s)	2.48	1.58	1.4	2.12
Avg. outlet temp. front channel (°C)	45	60.5	60.4	50.8
Avg. outlet temp. rear channel (°C)	49	N/A	N/A	51.2
Press. drop front channel (MPa)	0.046	0.006	0.003	0.019
Press. drop rear channel (MPa)	0.026	0.006	0.003	0.009
System pressure (MPa)	1.15	1.15	1.15	1.15
T_{max} salt/SS interface (°C)	60.1	94.4	88.4	67.7
Mass flow rate/module (kg/s)	167	28.5	25.3	165
Mass flow rate in reactor (kg/s)	2672	456	405	5293
Area in contact with/salt (m^2)	818	306	188	1096
Water Coolant Parameters				
FW First Channel				
Velocity (m/s)	2.4	2.4	2.4	2.4
Inlet temp. (°C)	40	40	40	40
Avg. outlet temp. (°C)	59.5	55.7	50	51.7
Pressure drop (MPa)	0.3	0.24	0.15	0.31
Mass flow rate/module (kg/s)	67.6	30.7	18.9	69.8
T_{max} steel outside surface (°C)	212	191	191	191
T_{max} graphite outside surface (°C)	570	N/A	N/A	N/A
FW Second Channel				
Velocity (m/s)	1.4	1.4	1.4	1.4
Inlet temp.(°C)	40	40	40	40
Avg. outlet temp. (°C)	66.1	61.4	53.2	58.4
Pressure drop (MPa)	0.31	0.31	0.19	0.31
Mass flow rate/module (kg/s)	39.5	17.9	11.0	40.7
Both Channels				
System pressure (MPa)	0.55	0.55	0.55	0.55
Mass flow rate/module (kg/s)	107.4	48.6	29.9	110.5
Mass flow rate in reactor (kg/s)	1718	777	478	3536

5. TRITIUM RECOVERY

The tritium produced by neutronic reactions in the aqueous salt breeder during its passage through the reactor breeder/shield is bound, chiefly in the thermodynamically stable molecule HTO. Some tritium can potentially reside in the gaseous molecule HT due to radiolysis but its relative concentration will be only 10^{-4} mole HT per mole of liquid HTO. The gaseous HT will be oxidized, subsequently, and the HTO produced will be combined with the aqueous phase. The gaseous tritium recovery will, therefore, begin with the isotopic separation of HTO/H_2O by the use of a water distillation column operating below atmospheric pressure.

Before the water distillation process can begin, the pressure of the aqueous breeder as it exits the reactor and heat exchanger must be reduced. Two techniques are presented to accomplish this depressurization. The first technique utilizes a vapor phase catalyst while the second technique uses a liquid phase, wet-proof, catalyst in order to oxidize the H_2 (HT) to H_2O (HTO), as described in the following summaries. The choice between these two techniques requires further analysis and experimental data.

5.1 Vapor Phase Catalytic Recombiner

The flow-diagram for this scheme, Figure 5, indicates that the ALSB solution is pumped via a pressurized circuit through the breeder blanket and its heat exchanger. The required pressure, as previously calculated, is sufficient so that no gaseous phase exists in these components. At the exit from the heat exchanger this dissolved H_2 and O_2 must be vented and the tritium produced in the blanket must be recovered. Both of these objectives are accomplished by use of the Flash-Evaporator. This component will probably be operated below atmospheric pressure so that in the event of any leakage, the large volume of tritiated water would not vent to the atmosphere, but the external atmosphere would be drawn inward. The combined flow to the Flash Evaporator from the OB and IB blankets is 8.92×10^3 kg/s of aqueous solution containing 10 Ci/kg of tritiated water (HTO) but, also, 3.4×10^{-3} moles of H_2, 1.7×10^{-3} moles of O_2 and 6×10^{-4}Ci of HT per kg. The total tritium production rate in the breeder solution is 15.8 Ci/s which must be removed at this rate so that the concentration of tritiated water does not exceed 10 Ci/kg. In the Flash Evaporator all of the noncondensable gaseous $H_2 + O_2$ + HT will be released and some

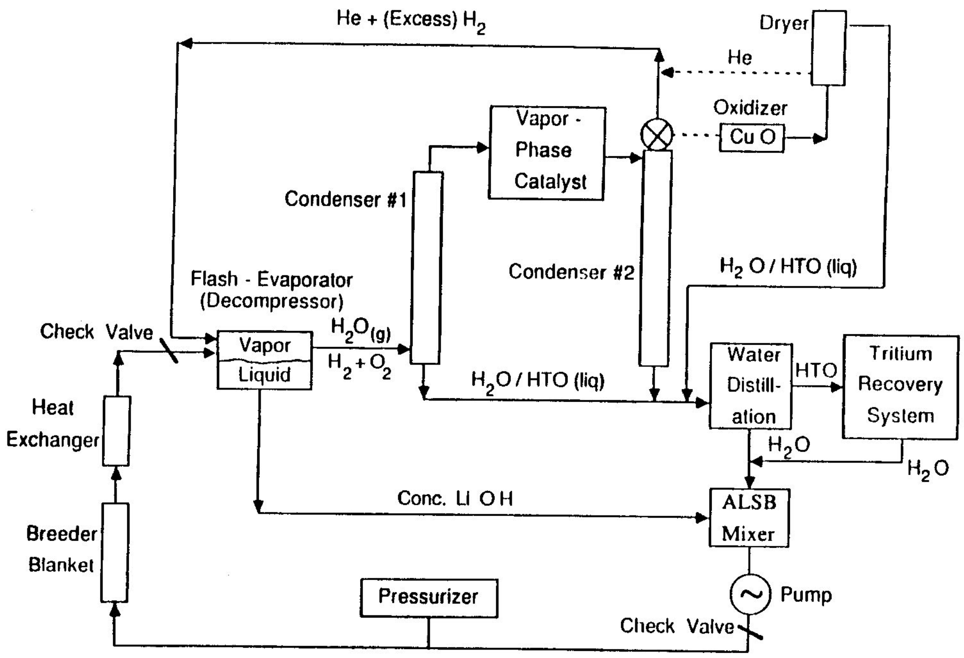


Figure 5. Flow Diagram Using ($H_2 + O_2$) Vapor Recombiner

portion of the H_2O (HTO) from the aqueous solution will evaporate. Because these combined gases constitute a potentially explosive mixture, the vapors must be diluted with He so that the concentration of H_2 is $< 4\%$. This He flow will transport the gases to Condenser #1 where the H_2O (HTO) will be condensed to liquid at ~ 10 Ci (HTO)/kg. One kg/s of this water is sent to the Water Distillation (WD) System. The noncondensable H_2, HT and O_2 pass through a Recombiner, gas phase catalyst, to form H_2O and HTO. Some excess H_2 may be required to completely remove all the O_2. The condensable H_2O (HTO) vapors are removed by Condenser #2 which has a product flow of 0.5 kg/s containing 5.3 Ci/s of HTO which is also sent to the WD section of the Tritium Removal System (TRS). The WD system processes this tritiated water at the rate of 3400 kg/hr. The combined flow from Condensers #1 and #2 is ~ 1.5 kg/s containing ~ 15 Ci (HTO)/s, ~ 10 Ci/kg.

The WD section of the TRS consists of two water distillation columns 2.6 m dia x 20 m high and are operated below atmospheric pressure in order to prevent the leakage of tritiated water. Each column is designed for an efficiency of 56% for the separation of HTO/H_2O at a flow rate of 1700 kg/hr; therefore, each column will remove ~ 5.7 Ci/kg from water containing 10 Ci/kg. At a flow rate of 1700 kg/hr, one WD system would remove 2.65 Ci/s of HTO. The two columns will provide tritiated water separation at 40% of its breeding rate, which is compatible with the planned 30% availability rate for ITER. In this mode of operation each column will have a tritium inventory of ~ 20 g. The enriched HTO from the WD system is then conveyed to the remainder of the TRS processes [6,7] in which refined T_2 and HT are produced.

Following Condenser #2, the He is recirculated to the Flash Evaporator. The H_2 and HT concentrations in the He are controlled when a side-stream is shunted to an oxidizing circuit so that these gases can be removed as H_2O and HTO.

The aqueous salt solution, Figure 5, is then reconstituted in the MIXER apparatus. The aqueous salt solution must be filtered and treated for corrosion products removal. The solution must, subsequently, be repressurized by use of the pump and pressurizer before entering the breeder blanket. Some excess H_2 may have to be introduced at this point in order to reduce some highly oxidizing species which may form as a result of radiolysis.

The system described in Figure 5 is based upon known technologies; however, the ability of the overpressure to eliminate completely the formation of bubbles of H_2 or O_2 in the breeder blanket or the heat exchanger must be demonstrated in radiolysis experiments. The engineering draw-backs to this flow-diagram are: (1) all of the salt solution exiting from the reactor must be depressurized and subsequently repressurized before returning to the blanket which requires a large pumping power, ~ 18 MW, and (2) because H_2 and O_2 exist in the Flash Evaporator, they must be diluted with He; consequently, the conduits for these gases become very large, ~ 1.0 m dia.

5.2 Liquid-Phase Recombiner

An advanced ALSB flow diagram, Figure 6, utilizes an $H_2 + O_2$ liquid recombiner which employs a precious metal catalyst with a porous, hydrophobic coating, hence, the name "wet-proof catalyst." Such a catalyst has been used successfully (1) to reduce the oxygen content of boiler-feedwater when excess H_2 is added [8], and (2) to catalyze the hydrogen isotopic equilibrium between water and the molecular hydrogen gas species [9], such as

$$HT_{(g)} + H_2O_{(liq)} = HTO_{(liq)} + H_{2(g)} \, . \tag{1}$$

This catalyst offers many advantages for the ALSB system, but it has not been demonstrated in the combined temperature and pressure ranges needed. Its temperature limit is ~ 260°C, the softening temperature of the coating at atmospheric pressure. It has been used in the gas phase up to 6 MPa and is being tested up to 15 MPa in the coolant from PWR's.

The aqueous salt breeder is pressurized as in the previous flow diagram so that no gaseous phase exists as it transits the breeder blanket and the heat exchanger. At this point the solution enters the catalyst bed where the radiolytically produced H_2O_2 begins to decompose and the recombination of $H_2 + O_2$ is initiated. Excess hydrogen of ~ 10 to 20% is added to the flow to insure that the O_2 is consumed. Oxygen removal is necessary in order to reduce the corrosive potential of the aqueous solution. As the solution enters the catalyst bed it contains 10 Ci (HTO)/kg and 10^{-4} mole fraction of H_2 to H_2O. If this

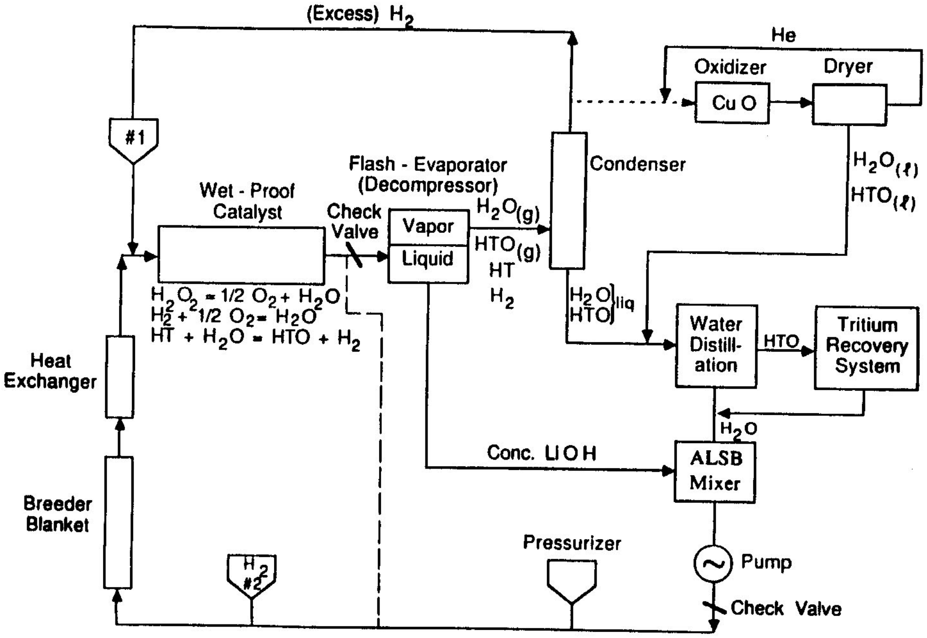


Figure 6. Flow Diagram Using (H_2 + O_2) Liquid Recombiner

ratio is the same for HTO then the dissolved molecular HT is 6 x 10^{-4} Ci/kg; however, ~ 80% is oxidized with the H_2 on the catalyst. Of the remaining $HT_{(g)}$ it will probably be reduced by a factor of 2 due to the catalyzed reaction, Eq. (1), which is thermodynamically favored at 60°C. As a result nearly 90% of the HT is converted to HTO which remains in the aqueous phase along with the residual 10 Ci/kg. If the liquid velocity in the catalytic bed is limited to ~ 2 m/s in order to avoid a large pressure differential, then the bed would need to be ~ 4 m dia. The use of a honeycomb catalysis structure would permit higher fluid flow and could reduce the bed diameter.

A portion of the aqueous solution containing only some excess H_2 is next admitted to the Flash Evaporator. Only 1 kg/s of water containing 10 Ci/kg of HTO is required for the WD process of the TRS. If 10% of the ALSB solution is evaporated in the Flash Evaporator, only 10 kg/s needs to be introduced. The remaining major portion of the aqueous solution can be returned directly to the blanket without the need to repressurize. The 20% excess H_2 in the solution will only increase the total pressure ~ 10% which is probably within the uncertainty of the radiolytic generation rate of the gases.

The noncondensable gas H_2(HT) which exits from the condenser of the Flash Evaporator can be reintroduced at position #1 ahead of the catalyst. As H_2 and HT build-up in this external circuit, their concentrations can be regulated when a portion of this gas is diverted to a side-stream in which the gases are oxidized and adsorbed on a desiccant. As this desiccant is recycled, the liquid H_2O and HTO are sent to the WD section of the TRS system.

A simple, low pressure technique for the suppression of the radiolytic gases may be possible. Computer modelling of the candidate ALSB solutions in a radiation environment similar to that expected in ITER indicates that the production of radiolytic gases is eliminated when excess H_2 at a concentration of 5 x 10^{-4} mol H_2/kg (10^{-2} dm^3 (STP)/kg) is added to the solution ahead of the radiation zone [10]. Demonstrations in operational Boiling Water Reactors indicate [11] that by H_2 addition to the feedwater the further production of radiolytic oxygen decreases and can be eliminated. The BWR coolant is principally exposed to gamma and neutron radiation while the ALSB solution will principally absorb energy from the highly energetic alpha and tritium recoil ions. Because of this difference in types of radiation environment, the ALSB solutions will be tested [12] with neutron irradiation from a CRNL experimental fission reactor, starting in August 1989. If experimental evidence confirms such benefits, then the excess H_2 would be injected at position #2 ahead of the breeder blanket (Figure 6), and the liquid phase recombiner could probably be eliminated.

In summary, the use of the liquid phase recombiner has advantages over the vapor phase recombiner in several features, namely: (1) only a small fraction of the circulating aqueous solution requires decompression and repressurization; consequently, the pumping power is reduced from 18 MW to ~ 18 kW; and (2) there is no need to dilute the radiolytic gases with helium below the explosive limit of H_2 in O_2; consequently the pipe dimensions for these gases can be much smaller.

The total tritium inventory existing as HTO in the proposed ALSB flow-diagrams is approximately 144 g, as shown in Table 6. This includes the tritium inventory in the two distillation columns. Five columns will be needed for tritium to be removed at the same rate as the breeding rate. In this case, the tritium concentration will be maintained at 10 Ci/kg even during extended periods of operation. Drawing on the experience of Ontario Hydro, the cost of the TRS using two distillation columns is ~ 31 M$.

6. SUMMARY

An aqueous salt blanket for ITER has been designed which utilizes predominately current technology and is relatively easy to fabricate. It employs a dual cooling system, an intermediate pressure (1.15 MPa) LiOH aqueous salt and a low pressure (0.55 MPa) water system. The first wall is capable of withstanding a disruption load of 2 MPa. A credible tritium extraction system and recombiner for radiolysis products is presented. In the physics phase the entire blanket can be cooled with plain water and the salt added to the breeder loop when the technology phase is initiated.

Table 6

ALSB Tritium Inventory as Tritiated Water

Location	Volume, m^3	Concentration, Ci/kg	Tritium, g
Blanket			
Inboard	6	10	6.0
Outboard	18.6	10	18.6
Heat Exchanger	12.3	10	12.3
Evaporator	10	10	10
Recombiner	5	10	5
Supply Pipes	35.2	10	35.2
Ring Header	16.9	10	16.9
Water Distillation	---	$10\text{-}10^3$	40
TOTAL			144

ACKNOWLEDGEMENT

This research has been supported by the Office of Fusion Energy of the U.S. Department of Energy.

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