# U.S. Aqueous Salt Blanket for ITER

I.N. Sviatoslavsky, M. Sawan, G.L. Kulcinski, J.
Blanchard, L. El-Guebaly, L. Wittenberg

February 1989

UWFDM-936

Presented at the U.S./Canada Workshop on Lithium Salt Blankets, Toronto, Canada, 7–9 February 1989.

FUSION TECHNOLOGY INSTITUTE

UNIVERSITY OF WISCONSIN

MADISON WISCONSIN

## DISCLAIMER

This report was prepared as an account of work sponsored by an agency of the United States Government. Neither the United States Government, nor any agency thereof, nor any of their employees, makes any warranty, express or implied, or assumes any legal liability or responsibility for the accuracy, completeness, or usefulness of any information, apparatus, product, or process disclosed, or represents that its use would not infringe privately owned rights. Reference herein to any specific commercial product, process, or service by trade name, trademark, manufacturer, or otherwise, does not necessarily constitute or imply its endorsement, recommendation, or favoring by the United States Government or any agency thereof. The views and opinions of authors expressed herein do not necessarily state or reflect those of the United States Government or any agency thereof.

# U.S. Aqueous Salt Blanket for ITER

I.N. Sviatoslavsky, M. Sawan, G.L. Kulcinski, J.
Blanchard, L. El-Guebaly, L. Wittenberg

Fusion Technology Institute
University of Wisconsin
1500 Engineering Drive
Madison, WI 53706

http://fti.neep.wisc.edu

February 1989

UWFDM-936

Presented at the U.S./Canada Workshop on Lithium Salt Blankets, Toronto, Canada, 7–9 February 1989.

# U.S. AQUEOUS SALT BLANKET FOR ITER

I.N. Sviatoslavsky, M. Sawan, G.L. Kulcinski
J. Blanchard, L. El-Guebaly, L. Wittenberg

Fusion Technology Institute
Nuclear Engineering and Engineering Physics Department
University of Wisconsin
Madison, WI  53706-1687

U.S./Canada Workshop on Lithium Salt Blankets

Toronto, Canada

February 7-9, 1989

# ABSTRACT

Aqueous lithium salt (ALS) 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-breeder to a breeder of tritium by the simple dissolution of a lithium salt into the water. Such a blanket is also amenable to simplicity of design and accommodation of 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.

# INTRODUCTION

The use of an aqueous lithium salt solution 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 lithium salts have become part of the U.S. R&D effort and similar studies are taking place in Canada and Europe.

Although the use of ALSB is not excluded for commercial fusion reactors, their use has great relevancy for the near term devices such as ITER which have extensive physics phases followed by technology phases. During the physics phase when tritium is not required, the blanket is cooled with plain water. As the project goes into the technology phase and tritium is needed, the appropriate salt is dissolved in the water and 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 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 the generation of 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 is 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 cylindrical shapes 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. The final design is more complex than the original one and utilizes two coolants, but is still more simple than other blankets of comparable performance.

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 $C^{14}$ 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 ALS blankets for ITER. Issues of corrosion, electrolysis and safety will be covered separately at this workshop and therefore will not be addressed in this paper.

## GENERAL DESCRIPTION

Figure 1 is a cross section of ITER in the technology phase. The outboard (OB) blanket in ITER is divided into 48 modules 82 cm deep with three modules fitting between the centerlines of adjacent TF coils. In the technology phase the toroidal extent of an OB blanket module at midplane is 97.5 cm and at the extremities 77.5 cm. The parameters quoted in this paper will be for the average value of 87.5 cm. Figure 2 is a cross section of a blanket module at midplane. For reference we will divide the module into two zones. The zone containing the large cylinders will be called Zone I and the rear zone containing the steel pebbles, Zone II. Zone I has a front and a rear half and in this zone flow is always from the bottom to the top. Zone II is used to duct salt and water to Zone I and thus the flow in it is always from top to bottom. All connections to the modules are made at the top.

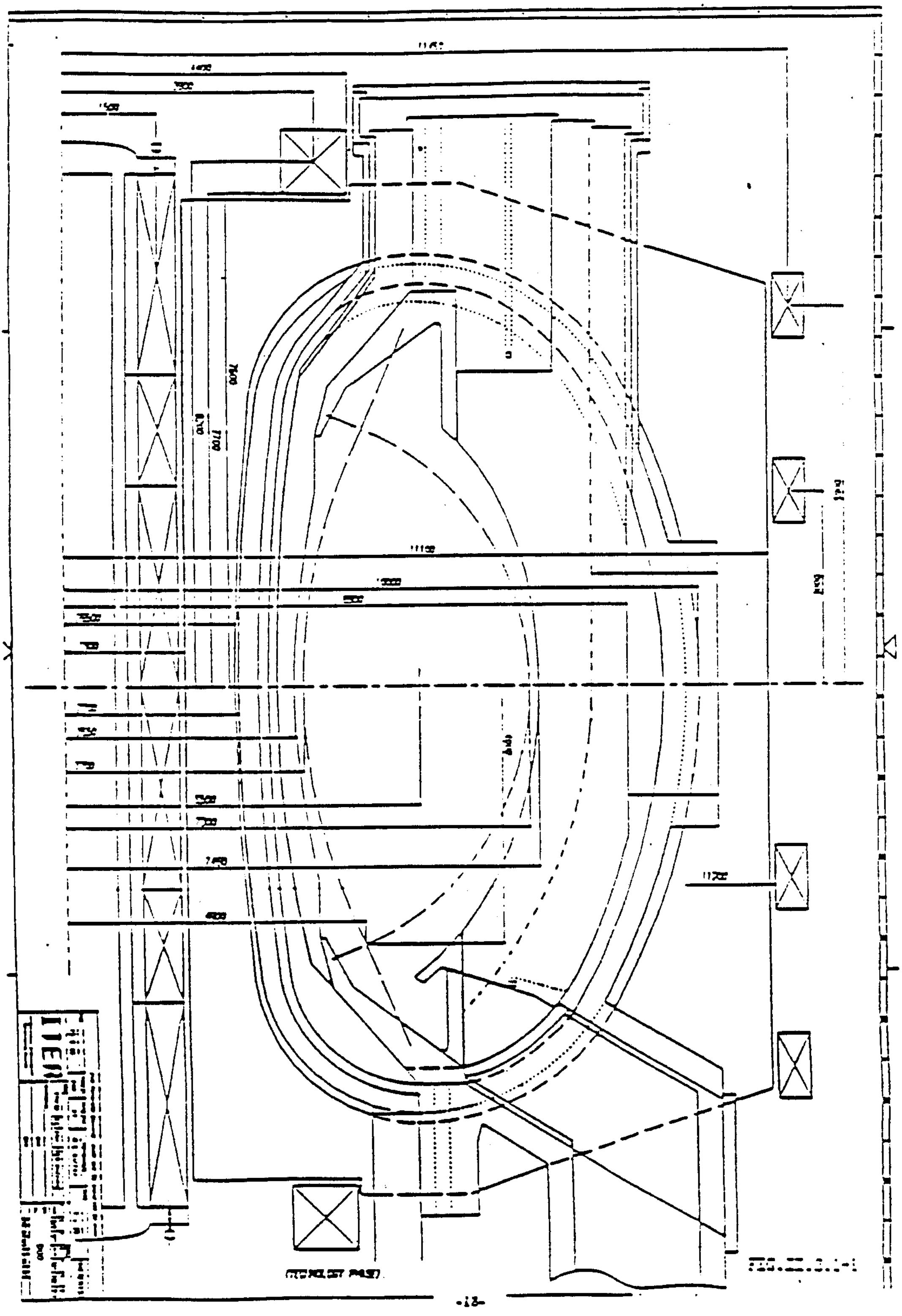


Figure 1
Cross section of ITER in the Technology Phase

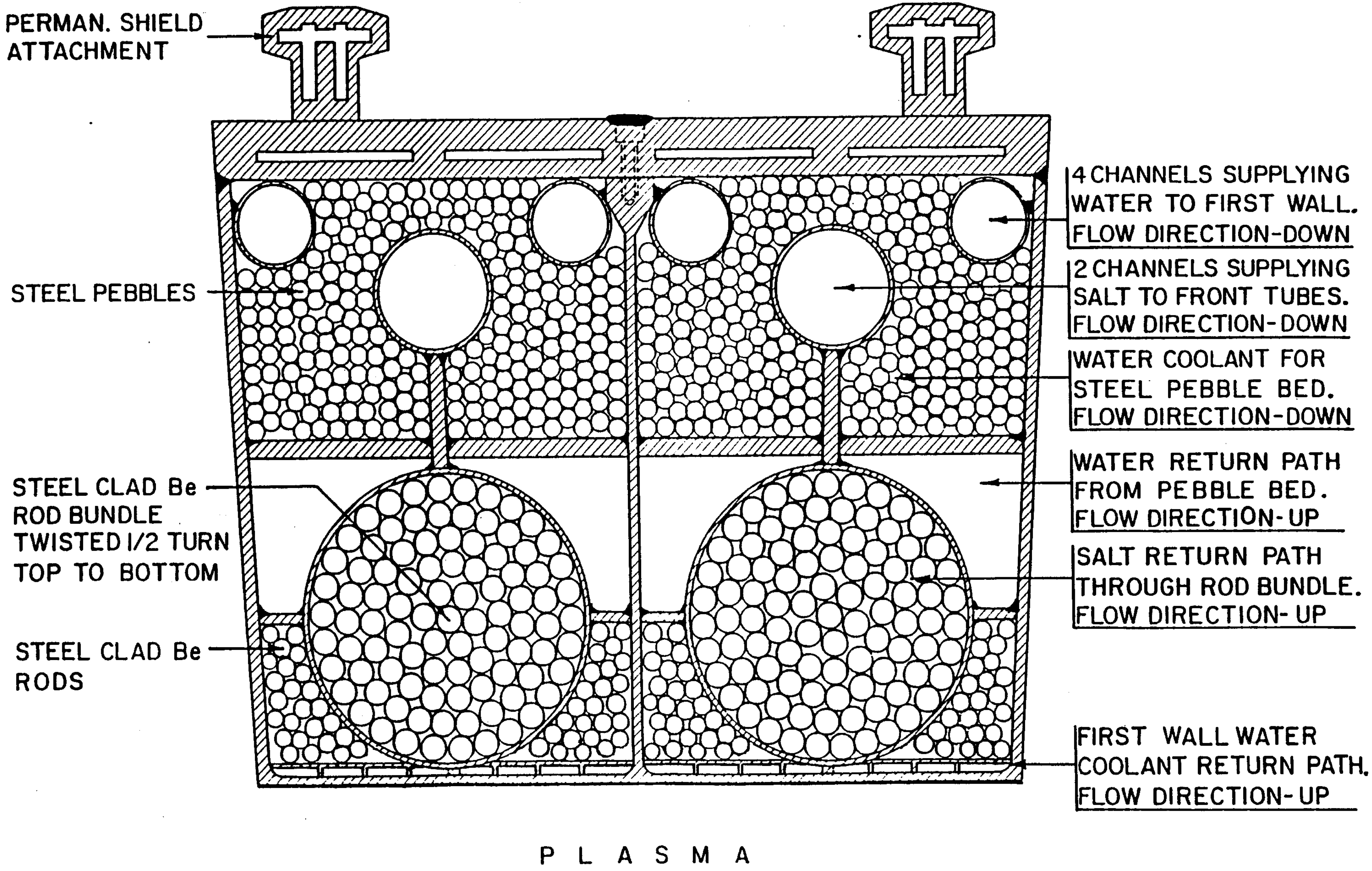


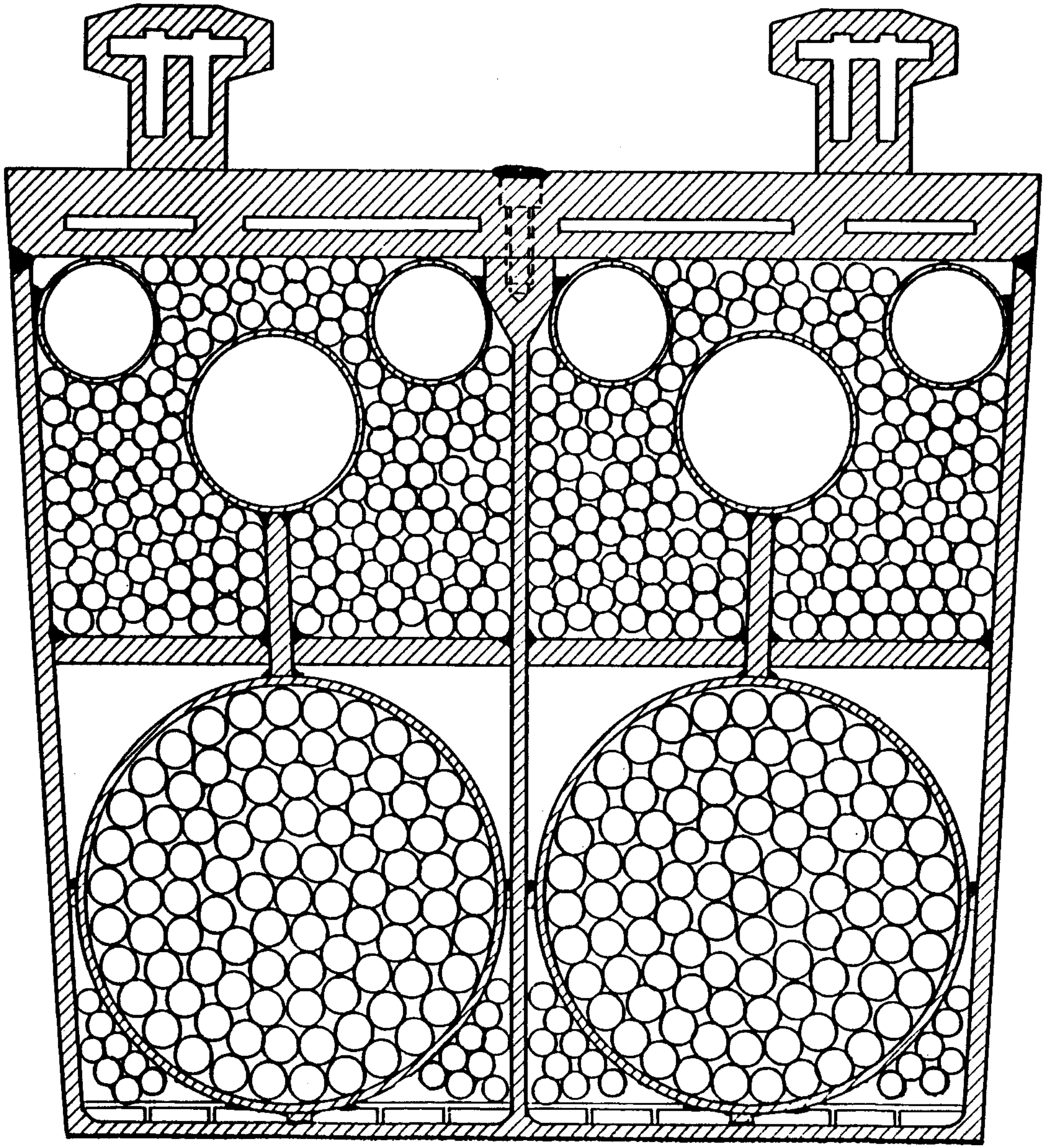
Figure 2

University of Wisconsin-Madison

Zone I is the primary breeding zone. The two large cylinders per module are 36.75 cm OD and 35.75 cm ID. The cylinders extend the full length of the modules following the contour of the first wall. Each cylinder contains 94 steel clad Be rods 3.25 cm OD where the cladding thickness is 0.1 mm. Since all the rods are the same diameter, the orientation is random not providing any particular fit between the rods. The rod bundle is twisted one half turn top to bottom. This allows a flow distribution in the coolant and equalization of temperature and radiation dose as the coolant travels from the bottom to the top. In Zone I the salt solution is confined to the cylinder while the rest of the zone is water cooled. The front half of Zone I also has Be rods only 1.0 cm in diameter. Figure 3 which is a cross section at the extremities shows that not all the rods go the full length of the module. There are more rods at the midplane to fill the additional toroidal extent.

Zone II contains tubes that supply coolant to the first wall and the high heat flux front half of Zone I. The large tube in Zone II is the aqueous salt supply tube which is connected to the front cylinder at the bottom of the module. Such tubes are used to minimize the pressure drop of the coolants for the high heat flux region. The water which flows down through the pebble bed returns through the rear half of Zone I. The first wall has a separate cooling channel 1.0 cm thick and is independently supplied by tubes located in Zone II.

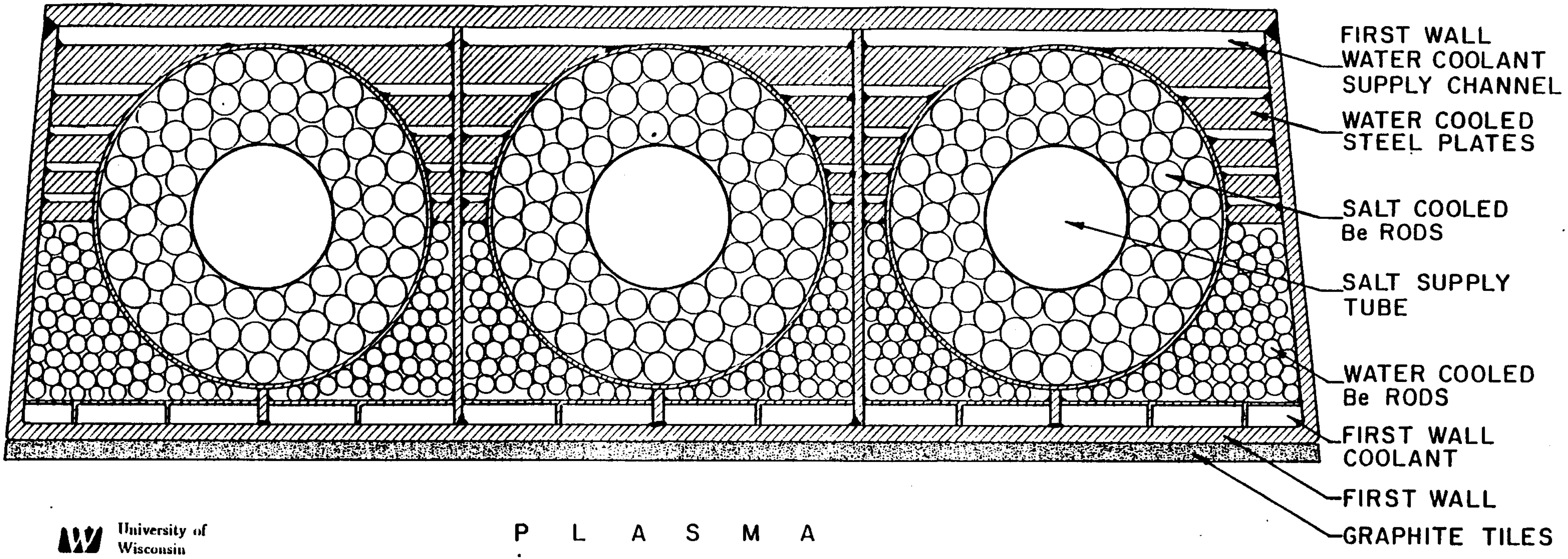
The inboard (IB) blanket is divided into 32 modules and is only 23 cm deep. Figure 4 is a cross section of an IB blanket module. Here again we divide the module into a front Zone I and a rear Zone II, while the aqueous salt cylinders straddle both zones. As seen from the figure there are three cylinders per module each 18.3 cm OD and 17.7 cm ID, and each has a supply



University of
Wisconsin-Madison

Figure 3

BLANKET CROSS SECTION AT BOTTOM



University of Wisconsin

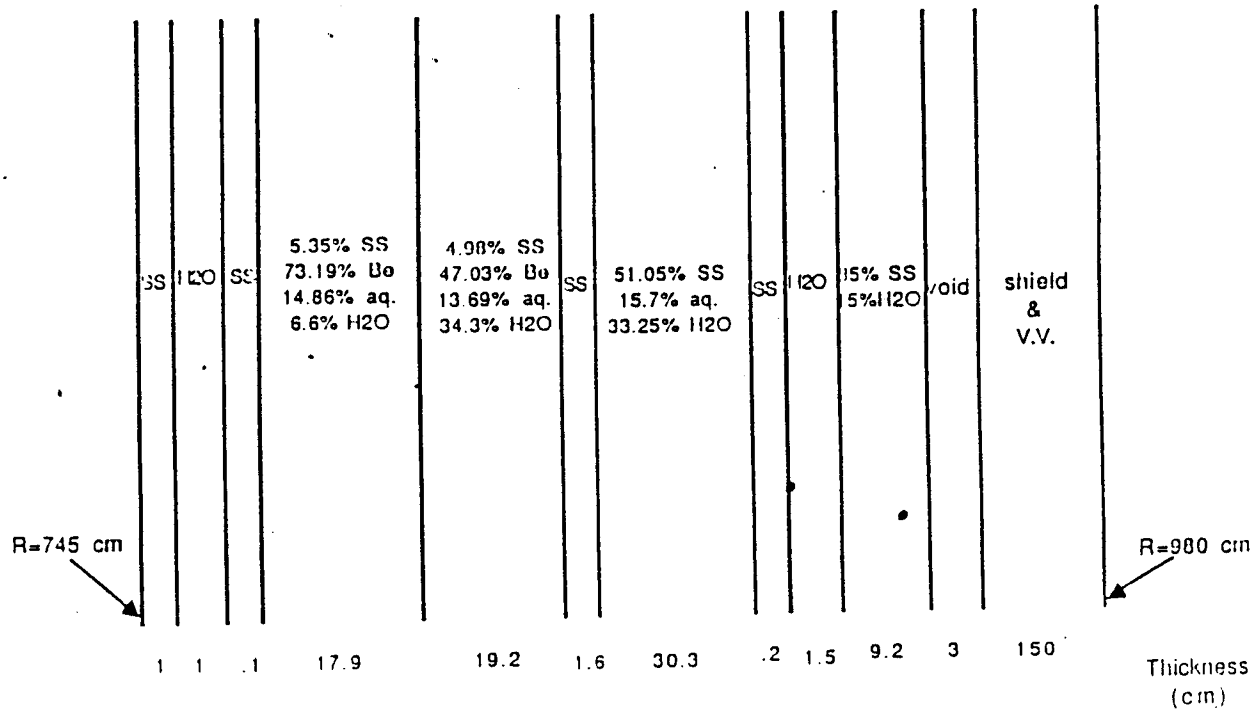
CROSS SECTION OF I/B BLANKET MODULE

Figure 4

tube coming down the center. The outer annulus of the cylinder contains 73 Be rods 1.6 cm in diameter also clad with 0.1 mm thick steel. In this configuration the aqueous salt flows down the center tube then emerges at the bottom and flows up in the annular region cooling the Be rods. The rest of Zone I is filled with 1.0 on diameter Be rods cooled with low pressure water. Zone II is devoted to ducting water to Zone I including the first wall. Instead of steel pebbles, we are using steel plates. the IB blanket has a straight forward geometry with no penetrations and is more amenable to plates. Further, the water pressure drop through plates is lower than through pebbles, especially since this water is used to cool a high heat flux region of the blanket. The first wall has a separate cooling channel of 0.7 cm thickness and is supplied by the rear channel in Zone II.

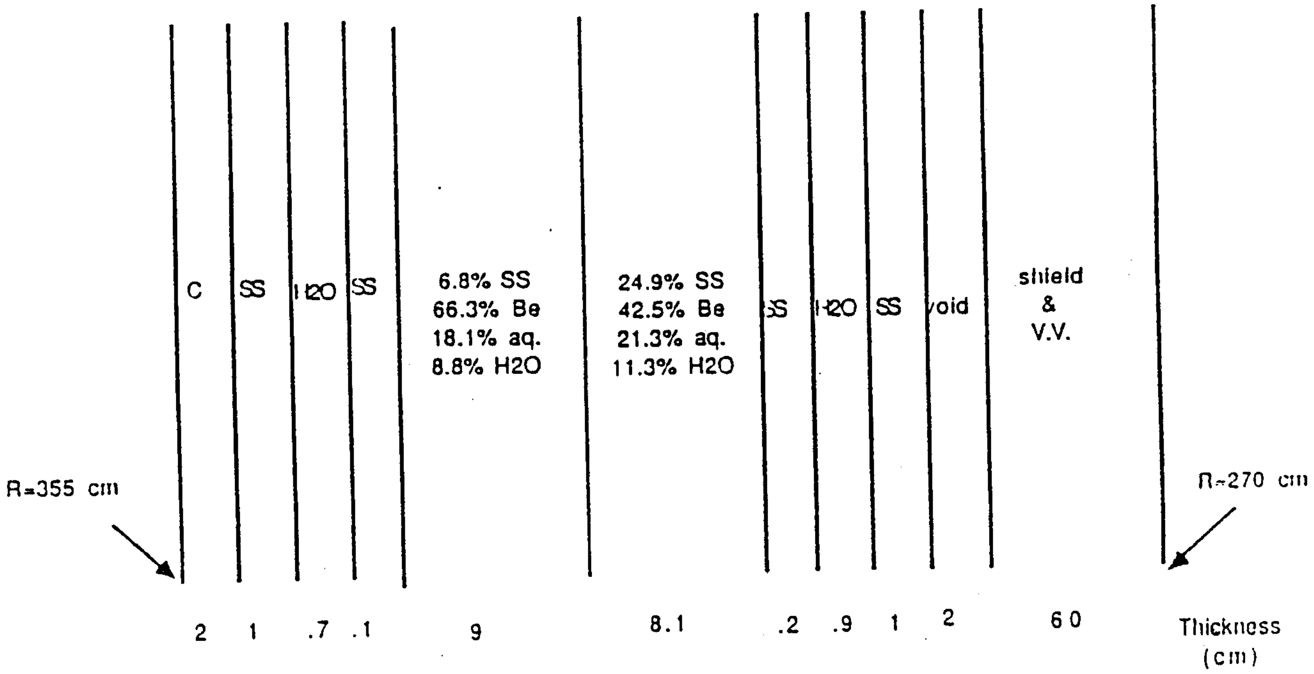
## NEUTRONICS

Neutronics analysis has been performed for the OB and IB ALS blankets to determine the tritium breeding ratio and the nuclear heat deposition in the different zones. One-dimensional toroidal cylindrical model, in which the IB and OB regions are modeled simultaneously, has been used. Figures 5a and 5b give the radial build used in the calculations for the OB and IB blankets, respectively. The breeding Zone I of the OB blanket is divided into front and rear zones to closely model the actual configuration of the blanket. The composition used represents the poloidal average conditions. The peak neutron wall loading is 1.5 MW/m$^2$ for the OB blanket and 1.1 MW/m$^2$ for the IB blanket. The average values are 1.1 and 0.67 MW/m$^2$ for the OB and IB blankets, respectively. Tables I and II give the nuclear heating in the OB and IB blankets, respectively. The results are given for the peak conditions



RADIAL BUILD OF OUTBOARD BLANKET AND SHIELD

Figure 5a



RADIAL BUILD OF INBOARD BLANKET AND SHIELD

Figure 5b

## Table I

## Nuclear Heating in Outboard Blanket and Shield

(at Reactor Midplane, $\Gamma = 1.5$ MW/m$^2$)

| Zone | Thickness (cm) | MW/cm | W/cm$^3$ |
|---|---|---|---|
| First wall | 1 | 0.086 | 18.47 |
| FW coolant | 1 | 0.062 | 13.26 |
| Baffle | 0.1 | 0.008 | 17.42 |
| Breeding Zone I (front) | 17.9 | 0.629 | 7.4 |
| Breeding Zone I (rear) | 19.2 | 0.160 | 1.72 |
| Strut | 1.6 | 0.010 | 1.31 |
| Breeding Zone II | 30.3 | 0.031 | 0.19 |
| Baffle | 0.2 | $2.5 \times 10^{-5}$ | 0.024 |
| Coolant | 1.5 | $5.4 \times 10^{-5}$ | 0.007 |
| Back wall | 9.2 | $4.7 \times 10^{-4}$ | 0.01 |
| Semipermanent shield and V.V. | 150 | $1.7 \times 10^{-4}$ | $1.9 \times 10^{-4}$ |

## Table II

## Nuclear Heating in Inboard Blanket and Shield

(at Reactor Midplane, $\Gamma = 1.1$ MW/m$^2$)

| Zone | Thickness (cm) | MW/cm | W/cm$^3$ |
|---|---|---|---|
| C tile | 2 | 0.036 | 7.87 |
| First wall | 1 | 0.032 | 14.28 |
| FW coolant | 0.7 | 0.014 | 8.97 |
| Baffle | 0.1 | 0.003 | 12.51 |
| Breeding Zone I | 9 | 0.147 | 7.51 |
| Breeding Zone II | 8.1 | 0.071 | 4.13 |
| Baffle | 0.2 | 0.002 | 4.15 |
| Coolant | 0.9 | 0.003 | 1.47 |
| Black wall | 1 | 0.007 | 3.3 |
| Semipermanent shield and V.V. | 60 | 0.042 | 0.37 |

at the reactor midplane. The total nuclear heating in the OB blanket is 700 MW while that in the IB blanket is 195 MW. An estimate for the achievable overall tritium breeding ratio (TBR) has been made by coupling the results of the one-dimensional toroidal geometry calculations with the coverage fractions of the IB and OB blankets excluding areas devoted for test modules, penetrations and blanket module side walls. The overall TBR is estimated to be 1.05 with the OB and IB blankets contributing 0.83 and 0.22, respectively.

PRESSURE NEEDED TO SUPPRESS RADIOLYSIS

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.(3) The dose rate contributions from gammas, recoil protons, triton, and alpha particles have been determined in the neutronics calculations. The average dose rates for the aqueous solution in front breeder cylinders of the OB and IB blankets are given in Table III. Average values are used since the Be rod bundles are twisted resulting in the aqueous salt being exposed to the average radiation dose as it travels from the bottom to the top. The aqueous salt supply tube at the back of the OB blanket receives less than 1% of the dose in the front cylinder and hence is neglected.

The G-values for the production of hydrogen are 0.45, 1.1, 1, and 1.7 molecules per 100 eV absorbed energy from $\gamma$, p,t, and $\alpha$, respectively.(3) Using the results of Table III indicates that the effective primary yields for $H_2$ production in the OB and IB blankets are 1.227 and 1.214, respectively. It

Table III

Absorbed Dose in Aqueous Solution

| Radiation | Average Dose Rate (eV/cm$^3$s) OB | Average Dose Rate (eV/cm$^3$s) IB | % of Total Dose OB | % of Total Dose IB |
|---|---|---|---|---|
| $\gamma$ | $2.64 \times 10^{18}$ | $2.99 \times 10^{18}$ | 3.8 | 5 |
| p | $1.39 \times 10^{19}$ | $1.32 \times 10^{19}$ | 20.1 | 22 |
| t | $2.99 \times 10^{19}$ | $2.49 \times 10^{19}$ | 43.5 | 41.7 |
| $\alpha$ | $2.25 \times 10^{19}$ | $1.88 \times 10^{19}$ | 32.5 | 31.3 |
| Total | $6.89 \times 10^{19}$ | $5.99 \times 10^{19}$ | 100 | 100 |

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.

The total absorbed dose rate in the aqueous solution per cylinder is $1.82 \times 10^{25}$ eV/s in the OB blanket and $4.41 \times 10^{24}$ eV/s in the IB blanket. Using the mass flow rates of 72 and 23 kg/s in the OB and IB cylinders, respectively, yields $H_2$ productions of $3.1 \times 10^{21}$ and $2.3 \times 10^{21}$ molecules per kg of solution in the OB and IB cylinders, 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 results in $H_2$ pressures of 1.1 and 0.82 MPa for the OB and IB blankets, respectively. The corresponding values for the $O_2$ pressure are 0.55 and 0.41 MPa. Adding the vapor pressure of water at 80°C (0.05 MPa) results in the total pressure required to keep the radiolytic gas products in solution and prevent boiling to be 1.7 MPa for the OB blanket and 1.28 MPa for the IB blanket.

## THERMAL HYDRAULICS

The thermal hydraulics of the ALS blanket is relatively straight forward. The nuclear heating determined by neutronic analysis is given in Tables I and II. Mass flow rate in the water cooled zones is adjusted for a temperature rise of 30°C. The velocity is then determined as well as the heat transfer coefficients. This is used to determine the temperature at the coolant interface. The first wall surface heat load is taken to be 20% of the incident neutron wall current.

The pressure drop due to flow between rods is determined by means of an equivalent hydraulic diameter. Because of random tube orientation the flow passages are not uniform. The hydraulic diameter is determined from a passage created when four rods are touching on a square matrix. This gives an average pressure drop for uniform sized rods randomly packed into a cylinder. Thus the hydraulic diameter for the OB salt cylinder is 0.89 cm, the IB salt cylinder 0.44 cm and for the water cooled Be rods it is 0.27 cm.

The aqueous salt inlet temperature is 60°C and the outlet is 75°C for the OB and 72°C for the IB blankets, respectively. The corresponding velocities are 3.1 m/s and 4.3 m/s and the pressure drops are 0.116 MPa and 0.548 MPa. An additional 0.1 Mpa is needed for the pressure drop outside the reactor. Further, a $H_2$ and $O_2$ overpressure of 1.7 MPa is needed for the OB and 1.28 MPa for the IB blankets to suppress evolution of radiolysis products. Thus the total pressure needed in the aqueous salt coolant system is 1.93 MPa for both IB and OB blankets. At this pressure the hoop stress in the cylinders is on the order of 60-70 Mpa. The maximum salt coolant interface temperature occurs at an OB blanket front Be rod and is equal to 86°C.

The plain water inlet temperature is 40°C and the outlet 70°C. The highest pressure drop of 0.32 MPa occurs in the IB blanket first wall coolant where the velocity is 4.1 m/s. An additional 0.1 Mpa is provided for the pressure drop outside the reactor and another 0.1 Mpa as protection against boiling, giving a total of 0.42 MPa. The value is rounded off to 0.45 Mpa as the water operating pressure.

Both IB and OB blankets have the same coolant systems operating at the same temperature and pressure conditions. Tables IV and V give the pertinent thermal hydraulic parameters.

## 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 fraction as compared to 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

## Table IV

### Preliminary OB Blanket Parameters

| Salt Cooled Cylinders | Outboard |
|---|---|
| OD/ID of cylinder (cm) | 37.5/36.5 |
| No. of cylinders/module | 2 |
| ID of supply tube (cm) | 15.0 |
| No. of modules in reactor | 48 |
| Vertical extent of cylinder (m) | 9.7 |
| OD of Be rods (cm) | 3.25 |
| Equivalent hydraulic diameter (cm) | 0.89 |
| Inlet salt temperature (°C) | 60 |
| Outlet salt temperature (°C) | 75 |
| Salt velocity in cylinders (m/s) | 3.1 |
| Pressure drop in blanket (MPa) | 0.13 |
| Pressure drop outside (MPa) | 0.10 |
| Pressure needed to suppress radiolysis products (MPa) | 1.70 |
| Absolute pressure (MPa) | 1.93 |
| Mass flow rate/cylinder (kg/s) | 71.9 |
| Salt/cylinder interface $T_{max}$ (°C) | 78.5 |
| Salt/Be rod interface $T_{max}$ (°C) | 80 |
| **Water Cooled Blanket Zones** | |
| OD of Be rods (cm) | 1.0 |
| Equivalent hydraulic diameter (cm) | 0.27 |
| Velocity in Be rods (m/s) | 1.34 |
| Maximum $\Delta p$ in blanket (MPa) | 0.17 |
| $\Delta p$ outside blanket (MPa) | 0.10 |
| Absolute water pressure (MPa) | 0.45 |
| Inlet temperature (°C) | 40 |
| Mass flow rate/module (kg/s) | 28.1 |
| Outlet temperature (°C) | 70 |
| Water/Be interface $T_{max}$ (°C) | 78 |
| **FW Cooling** | |
| FW graphite thickness (cm) | 0 |
| FW thickness | 1.0 |
| FW coolant channel thickness (cm) | 1.0 |
| Coolant velocity (m/s) | 3.75 |
| Inlet temperature (°C) | 40 |
| Outlet temperature (°C) | 70 |
| Pressure drop (MPa) | 0.11 |
| Absolute pressure (MPa) | 0.45 |
| Mass flow rate/module (kg/s) | 32.8 |
| FW/coolant interface $T_{max}$ | 90 |
| **General** | |
| Total water mass flow/module (kg/s) | 60.9 |
| Total salt mass flow/module (kg/s) | 143.8 |
| Total thermal power/module ($MW_{th}$) | 16.46 |

## Table V

## Preliminary IB Blanket Parameters

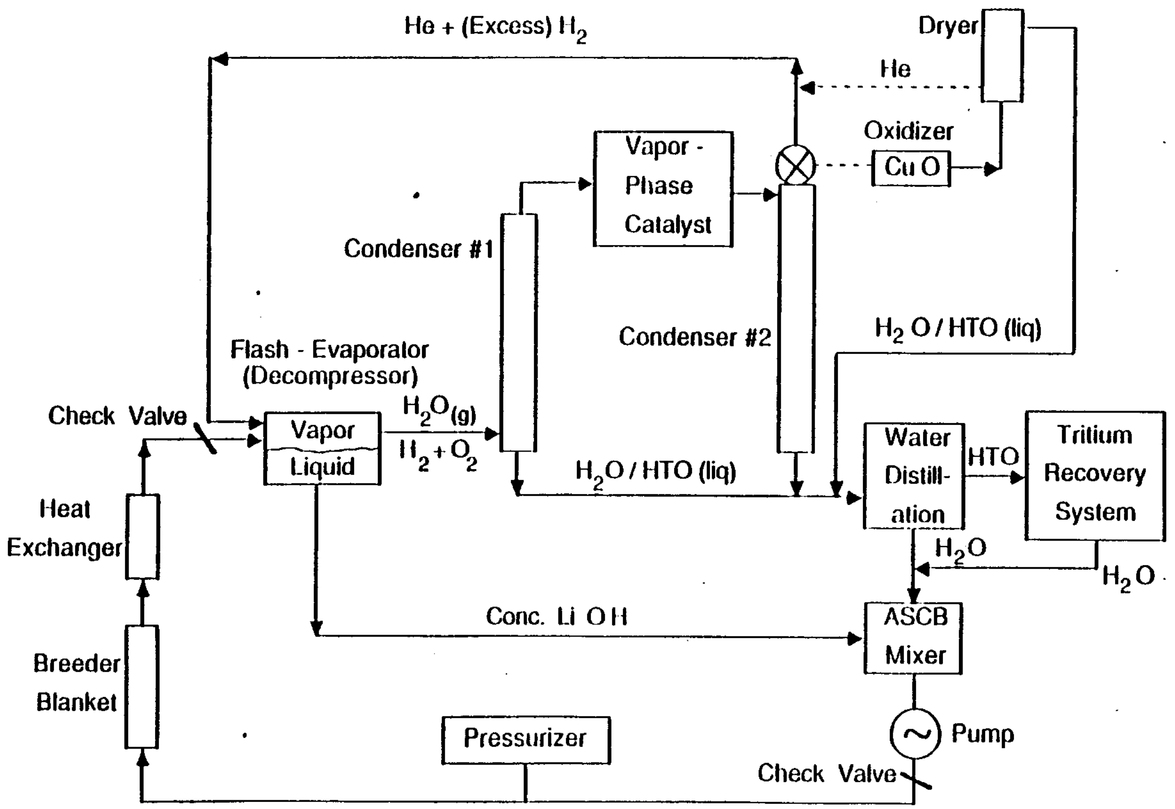
| Salt Cooled Cylinders | Inboard |
|---|---|
| OD/ID of cylinder (cm) | 18.3/17.7 |
| No. of cylinders/module | 3 |
| ID of supply tube (cm) | 6.8 |
| No. of modules in reactor | 32 |
| Vertical extent of cylinder (m) | 10.2 |
| OD of Be rods (cm) | 1.6 |
| Equivalent hydraulic diameter (cm) | 0.44 |
| Inlet salt temperature (°C) | 60 |
| Outlet salt temperature (°C) | 72 |
| Salt velocity in cylinders (m/s) | 4.3 |
| Pressure drop in blanket (MPa) | 0.55 |
| Pressure drop outside (MPa) | 0.10 |
| Pressure needed to suppress radiolysis products (MPa) | 1.28 |
| Absolute pressure (MPa) | 1.93 |
| Mass flow rate/cylinder (kg/s) | 23.2 |
| Salt/cylinder interface $T_{max}$ (°C) | 73 |
| Salt/Be rod interface $T_{max}$ (°C) | 73.8 |
| **Water Cooled Blanket Zones** | |
| OD of Be rods (cm) | 1.0 |
| Equivalent hydraulic diameter (cm) | 0.27 |
| Velocity in Be rods (m/s) | 1.69 |
| Maximum $\Delta p$ in blanket (MPa) | 0.23 |
| $\Delta p$ outside blanket (MPa) | 0.10 |
| Absolute water pressure (MPa) | 0.45 |
| Inlet temperature (°C) | 40 |
| Mass flow rate/module (kg/s) | 9.5 |
| Outlet temperature (°C) | 70 |
| Water/Be interface $T_{max}$ (°C) | 76 |
| **FW Cooling** | |
| FW graphite thickness (cm) | 1.0 |
| FW thickness | 1.0 |
| FW coolant channel thickness (cm) | 0.7 |
| Coolant velocity (m/s) | 4.1 |
| Inlet temperature (°C) | 40 |
| Outlet temperature (°C) | 70 |
| Pressure drop (MPa) | 0.21 |
| Absolute pressure (MPa) | 0.45 |
| Mass flow rate/module (kg/s) | 20.5 |
| FW/coolant interface $T_{max}$ | 86 |
| **General** | |
| Total water mass flow/module (kg/s) | 30 |
| Total salt mass flow/module (kg/s) | 69.6 |
| Total thermal power/module ($MW_{th}$) | 6.79 |

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.

Vapor Phase Catalytic Recombiner

The flow-diagram for this scheme, Fig. 6, indicates that the ASCB 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 the 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 during any leakage the large volume of tritiated water would not vent to the atmosphere, but the external atmosphere would be drawn inwardly. The combined flow to the Flash Evaporator from the OB and IB blankets is $9.14 \times 10^3$ kg/s of aqueous solution containing 10 Ci/kg of tritiated water (HTO) but, also, $5 \times 10^{-3}$ moles of $H_2$, $2.5 \times 10^{-3}$ moles of $O_2$ and $10^{-3}$ Ci of HT per kg. The total tritium production rate in the solution is 15.8 Ci/s which should 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 non-condensible gaseous, $H_2 + O_2 +$ HT, will be released and some portion of the $H_2O$ (HTO) from the aqueous solution will evaporate. Because these combined gases constitute a potentially explosive mixture, these 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

Figure 6



ASCB FLOW DIAGRAM USING ($H_2 + O_2$) VAPOR RECOMBINER the Water Distillation (WD) System. The non-condensible $H_2$ and $O_2$ pass through a Recombiner catalyst to form $H_2O$ and HTO. Some excess $H_2$ may be required to completely remove all the $O_2$. The condensible $H_2O$ vapors are removed by Condenser #2 which has product flow of 0.83 kg/s containing 9.1 Ci/s of HTO which is sent, also, to the WD section of the Tritium Removal system.

The combined flow from Condensers #1 and #2 is ~ 1.8 kg/s (6500 kg/hrs) containing ~ 19 Ci (HTO)/s, ~ 10 Ci/kg. The WD section of the TRS is designed to remove ~ 5.7 Ci/kg from water containing 10 Ci/kg at the rate of ~ 2700 kg/hr; therefore, one WD system would remove HTO at ~ 33% of its generation rate, which is compatible with the planned 30% availability rate for ITER. If greater availability of ITER becomes likely a second WD system could be added. The enriched HTO from the WD system is then conveyed to the remainder of the TRS processes(4) (as discussed elsewhere) 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 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, Fig. 6, is then reconstituted in the MIXER apparatus and is probably filtered and corrosion products removed. The solution must, subsequently, be repressurized by use of the pump and pressurizer before entering the breeder blanket. Some excess $H_2$ may need 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 Fig. 6 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 sample 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 becomes very large, 1.8 m dia.

## Liquid-Phase Recombiner

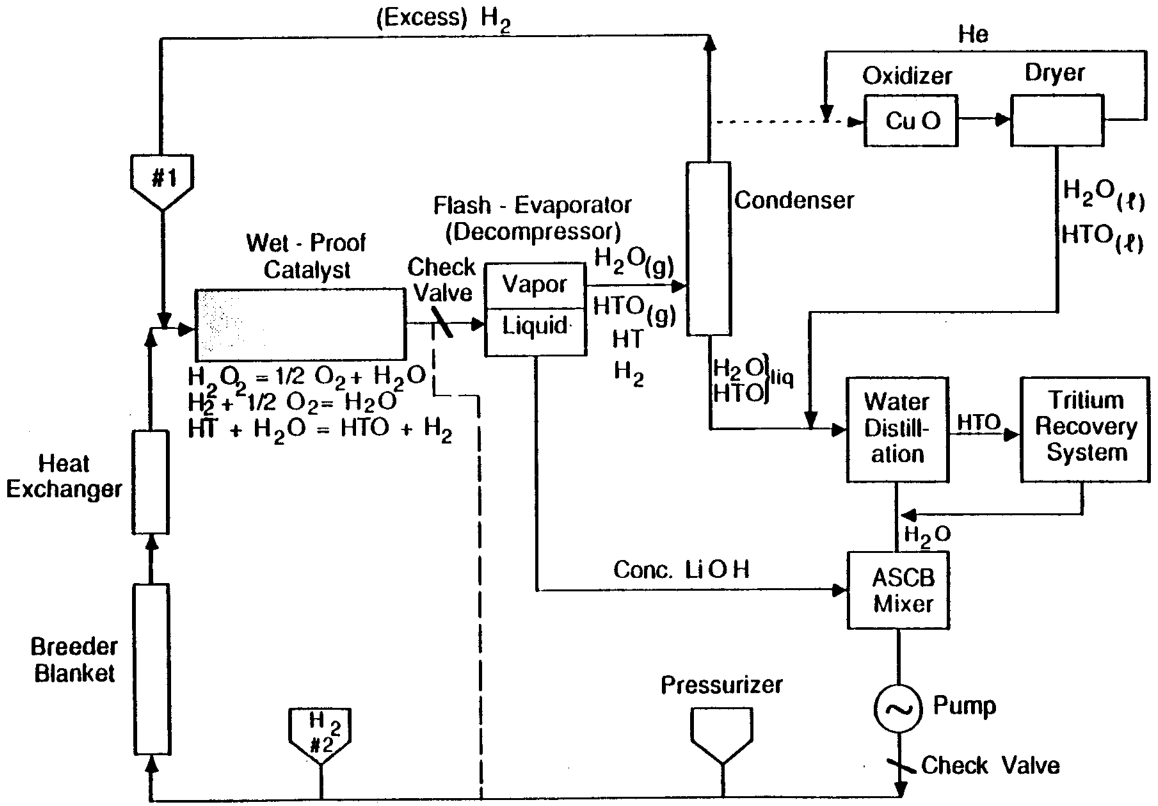
An advanced ASCB flow diagram, Fig. 7, utilizes an $H_2 + O_2$ liquid recombiner which employs a precious metal catalyst with a porous, hydrophobic coating, hence, the name "wet-proof catalyst." Such a catalyst has been used successfully (1) to reduce the oxygen content of boiler-feed water when excess $H_2$ is added,(5) and (2) to catalyze the hydrogen isotopic equilibrium between water and the molecular species,(6) such as

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

This catalyst offers many advantages for the ASCB 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

Figure 7



ASCB FLOW DIAGRAM USING ($H_2$+ $O_2$) LIQUID RECOMBINER radiolytically produced $H_2O_2$ begins to decompose and the recombination of $H_2 + O_2$ is initiated. Excess hydrogen ~ 10 to 20% is added to the flow to insure that all of the $O_2$ is consumed. 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 ratio is the same for HTO then the dissolved molecular HT is $10^{-3}$ Ci/kg; however, ~ 80% is oxidized with the $H_2$ on the catalyst. Of the remaining $HT_{(g)}$ it will be probably, 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 at $10^{-3}$ Ci/kg of solution 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 honey-comb 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, now, 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 ASCB is evaporated in the Flash Evaporator, only 10 kg/s needs to be introduced. The remaining 9130 kg/s of 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 non-condensible gas $H_2$(HT) which exits from the condenser 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 WD section of the TRS system.

Radiolytic experiments of the candidate ASCB salt solutions in a radiation environment similar to that expected in ITER may indicate that the production of radiolytic gases is decreased when excess $H_2$ pressure is maintained. Demonstrations in operational Boiling Water Reactors indicate(7) that for a ~ 10% increase in $H_2$ pressure above that expected from radiolysis, the further production of radiolytic gases ceases. The BWR operates, however, at a much higher temperature than the proposed ITER blanket. Some back-reactions may be possible in the ITER blanket at its lower temperature and would help to destroy some of the highly oxidizing species which may form at the high pH of the LiOH solutions. If experimental evidence indicates such benefits, then, the excess $H_2$ would be injected at position #2 ahead of the breeder blanket.

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) the need to dilute the radiolytic gases with helium below the explosive limit of $H_2$ in $O_2$ does not exist; consequently, the pipe dimensions for these gases can be much smaller.

The total tritium inventory existing as HTO in the proposed ASCB flow-diagrams is approximately 262 g of tritium, as shown in Table VI.

## Table VI

## ASCB Tritium Inventory as Tritiated Water

| Location | Volume, $m^3$ | Concentration, Ci/kg | Tritium, g |
|---|---|---|---|
| Blanket | 47 | 10 | 47 |
| Heat Exchanger | 25 | 10 | 25 |
| Evaporator | 10 | 10 | 10 |
| Piping and Pumps | 100 | 10 | 100 |
| Water Distillation | - | $10-10^3$ | 80 |
| TOTAL | | | 262 |

## Table VII

## Number and Type of Penetration in ITER

| | Number Penetration | Shape | Orientation | Dimensions W x H (m) | Modules Effected |
|---|---|---|---|---|---|
| 1) Test Modules | 8 | Rectangular | Radial | 1.0x2.0 | 8 |
| 2) Neutral Beams | 3 | Rectangular | at 45° | (0.8-1.0) x (2.4-3.4) | 9 |
| 3) Lower Hybrid | 2 | Rectangular | Radial | (1.1x1.3) x (2.5-3.5) | 6 |

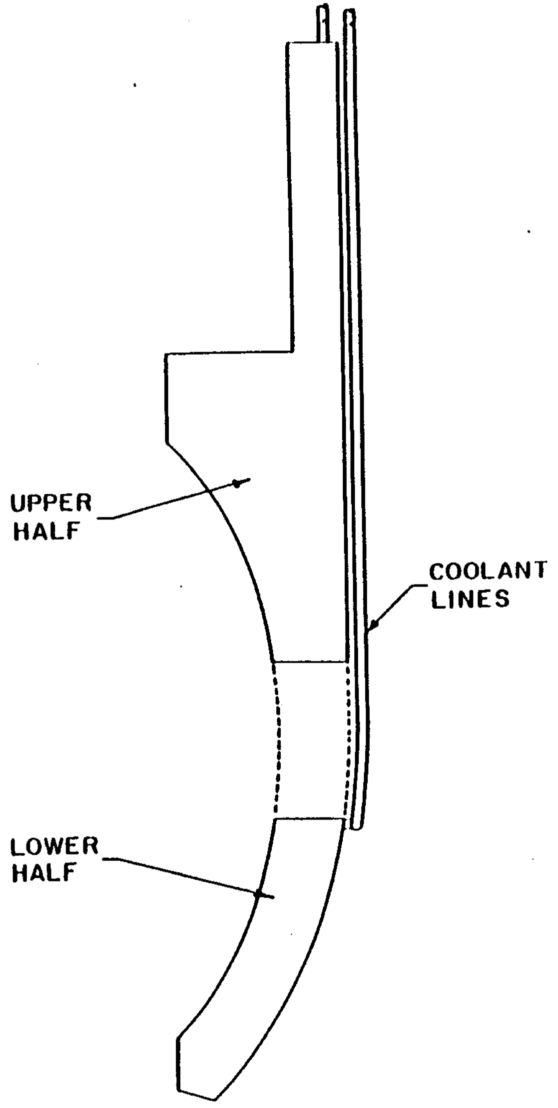
## PENETRATION ACCOMMODATIONS

The ALS blanket is amenable to accommodating penetrations on the OB side. Table VII gives the information on penetrations. The test modules and the lower hybrid heating modules are rectangular and have a radial orientation. These penetrations effectively divide a blanket module into an upper and a lower half as shown in Fig. 8a. Moreover, since the coolant lines must come from the top, this means that the lower half will need its own set of coolant lines which have to be recessed into the permanent shield of the adjacent modules as shown in Fig. 8b. This is done to leave an unobstructed path for the test modules and the lower hybrid heating modules.

Neutral beam penetrations have a 45° orientation and each affect three modules to varying degrees. However, we find that sufficient area is left in each of the effected modules to allow coolant ducting to and from the lower half. This means that the cylinders are truncated at the penetration edge and the coolants ducted through the remaining connecting blanket module area.

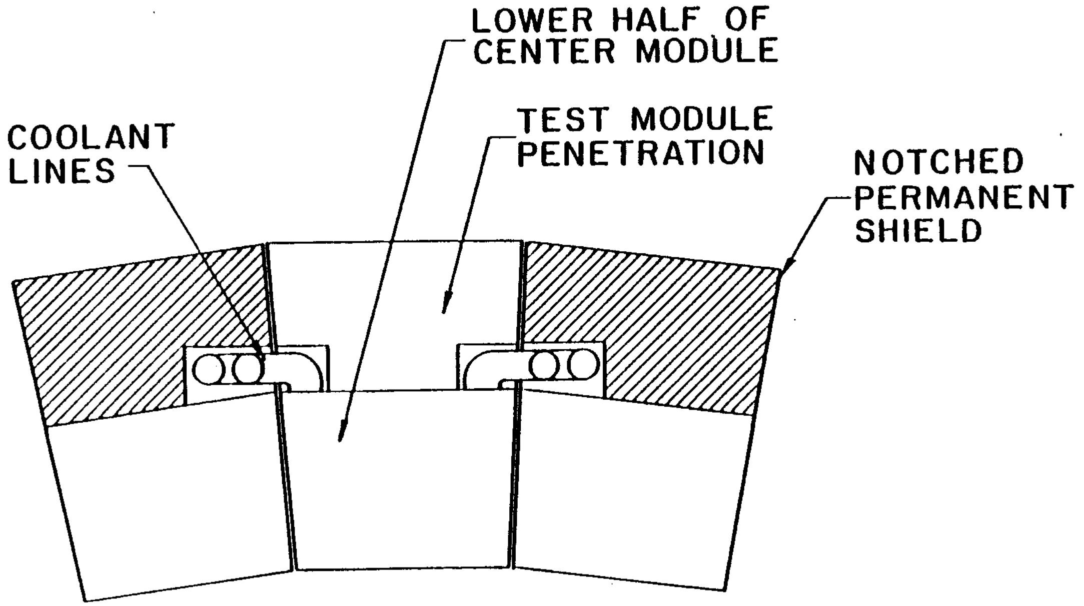
## SUMMARY

An aqueous salt blanket for ITER has been designed which utilized predominantly current technology, is relatively easy to fabricate and has a tritium breeding ratio of 1.05. It employs a dual cooling system, an intermediate pressure LiOH aqueous salt and a low pressure water system. The maximum aqueous salt pressure is 1.93 MPa and the maximum structural temperature is 90°C. A credible tritium extraction system and recombiner for radiolysis products is presented. The blanket design is amenable for accommodating the penetrations planned for ITER.



TEST MODULE PENETRATION

Figure 8a



TOP VIEW OF A SECTOR
OF O/B BLANKET MODULES

Figure 8b

## Acknowledgement

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

## REFERENCES

1. D. Steiner, et al., "A Self-Cooled Heavy Water Breeding Blanket," 11th Symposium on Engineering Problems of Fusion Research, Houston, Texas, Nov. 1985.

2. F. Najmabadi, R.W. Conn, N.M. Ghoniem, S.P. Grotz, J. Blanchard, et al., "The TITAN Reversed-Field Pinch Fusion Reactor Study; The Final Report," UCLA-PPG-1200, Joint Report of University of California - Los Angeles, GA Technologies, Inc., Los Alamos National Laboratory, and Rensselaer Polytechnic Institute, January 1988.

3. A.O. Allen, "The Radiation Chemistry of Water and Aqueous Solutions," D. Van Nostrand Co., 1961.

4. M.R. Galley, et al., "Recovery of Tritium from an Aqueous Lithium Salt Blanket on the NET Fusion Device," Fusion Technology, 14, 822 (1988).

5. D.A. Spagnolo, private communication (1988).

6. D.A. Spagnolo, A.E. Everatt, P.W.K. Seto, and K.T. Chuang, "Enrichment and Volume Reduction of Tritiated Water Using Combined Electrolysis Catalytic Exchange," Fusion Technology, 14, 501 (1988).

7. R.J. Law, M.E. Indig, C.C. Lin, and R.L. Cowan, "Suppression of Radiolytic Oxygen Produced in a BWR by Feed Water Hydrogen Addition," in Water Chemistry 3, British Nuclear Energy Society, London, pp. 23-29 (1984).