GA–A24754

ANCILLARY SYSTEMS FOR DUAL COOLANT LIQUID BREEDER TEST BLANKET MODULES

INTERIM REPORT TO THE ITER TEST BLANKET WORKING GROUP (TBWG)

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JULY 1, 2004

GENERAL ATOMICS

DISCLAIMER

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GENERAL ATOMICS

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SUMMARY

All ITER parties have to provide for the TBWG the required information regarding the interfaces between their TBMs and the external loop systems for heat extraction, coolant circulation, tritium extraction and coolant purification, as well as requirements for all ancillary systems. In the US, we have been interested in the development of dual coolant (DC) liquid breeder/coolant first wall and blanket concepts. Ferritic steel is selected as the structural material and helium is selected as the first wall and blanket structure coolant; and with either Pb-17Li or a low melting point molten salt (MS) as liquid breeder/coolant to be tested in one half of a designated test port. This approach of TBM development leads to the crucial point that the defined ancillary equipment spatial envelope and supporting requirements will have to be able to cover the cases for DC Pb-17Li and DC $LiBeF_3$ concepts, where in our assessment $LiBeF_3$ is selected as an example of the low melting point MS options. Equipment size is designed to the maximum power handling of ITER heat flux of 0.5 MW/m^2 and a neutron wall loading of 0.78 MW/m^2 for half of an ITER test port. Based on the requirements of handling the first wall heat flux and maximizing the outlet temperature of the breeder coolant up to 650°C, we have to evaluate the ancillary equipment required for two coolant loops. The first one is the first wall and structure coolant loop, which we have designed to carry 40% of the total blanket energy at the maximum ITER operation level. Dedicated helium piping is then designed between the TBM and the helium/water heat exchanger at the TCWS vault. The second one is the liquid breeder loop, which we designed to carry 100% of the blanket energy at the maximum ITER operating level. This second loop would also allow the testing of a single breeder/coolant self-cooled breeder concepts if our blanket development evolves to such direction in the future. For this breeder loop we also designed a helium intermediate heat removal loop between the breeder and the TCWS water cooling system. Corresponding helium piping was also designed.

For this task we have also performed a preliminary safety assessment on the designs, which provided guidance to our ancillary equipment design in areas of minimizing the vulnerable breeder volume and the potential loss of tritium through permeation. For the liquid breeder loop design, the requirement of minimizing the potential tritium loss from the breeder to the vicinity, led us to the use of the helium intermediate heat transport loop. This intermediate loop also helps to minimize the required pressure drop when the high viscosity fluid $LiBeF_3$ is utilized by keeping the distance between the TBM and the liquid breeder/helium heat exchanger to a minimum. We also recommended concentric pipes to be used to connect the liquid breeder between the TBM and the breeder/helium heat exchanger. For the FW-coolant loop, to minimize tritium permeation aluminum tubes are recommended for the He/water heat exchanger and permeation barrier like alumina coating or Al outside sleeve are recommended to be applied to the helium-coolant inlet and outlet piping. Results from the preliminary safety assessment for the

ancillary equipment for the two FW/blanket concepts show that ITER safety criterion can be met provided that we take care of controlling the amount of breeder used in the system and the reduction of tritium permeation loss from the FW coolant loop and from the liquid breeder loop.

Details of key ancillary equipments including heat exchangers, circulators, electrical heater, and helium storage and dump tanks for the two loops have been estimated. Results show that all our liquid breeder test equipment can be located within half of the test module transporter envelope. For the heat transport equipment in the TCWS vault, the required foot print is estimated to be a total of 20 m^2 and 5 m high for both helium to water loops. Piping size for the two loops connecting the TBM to the TCWS vault has also been estimated including the necessary inclusion of 10 to 15 cm of thermal insulation. Detailed designs of the PbLi/He and $LiBeF_3$/He heat exchangers are different because of the difference in thermal physical properties between the two breeders. But both heat exchanges will be able to handle the low coolant outlet condition of 520°C and 440°C, respectively, and the high performance condition of 650°C. For the high performance case the heat exchangers will have to be designed with the flexibility of reducing the heat removal surface by about 40%, and tube plugging is an example of achieving such an effect.

For this phase of our ancillary equipment assessment, the main focus was on the space and power requirements of key ancillary equipments to support the development of our DC liquid breeder blanket concepts. More detailed components design, fluid, tritium and safety handling design and analysis will be needed. The scenario of testing FW/blanket concepts in difference phases of ITER operation will have to be developed and the corresponding sequence of installing necessary testing components will have to be coordinated. But the requested envelope for the two testing loops proposed in this report will be able to provide the flexibility of testing the selected DC liquid breeder FW/blanket concepts.

1. INTRODUCTION

This report covers the ancillary systems description of the US dual coolant (DC) liquid breeder test blanket module designs. The goal is to provide TBWG the required information regarding the interfaces between the US TBM and the external loop systems for heat extraction, coolant circulation, tritium extraction and coolant purification, as well as the space requirements for all external systems. We have also included an assessment on the corresponding safety impacts. Based on the selection of ferritic steel as the first wall and blanket structural material and in order to handle the first wall heat flux and maximize the outlet temperature of the blanket coolant, we focused on the DC, which means helium-cooled first-wall/steel structure and self-cooled liquid breeder, design concepts. We will design the test module for both Pb-17Li and a low melting point molten salt (MS) as liquid breeder to be tested in one half of a designated port. For the low melting point MS option, we are interested in the use of $LiBeF_3$ and FLiNaBe breeders. Relatively, FLiNaBe has a lower melting point and viscosity, but with incomplete thermal and physical properties at this time. In our assessment, we have been focusing on the ancillary systems requirements for the Pb-17Li and $LiBeF_3$ options, and these results should be able to cover the case for FLiNaBe. This would allow us to switch to this liquid breeder if its thermal and physical properties continue to be proven favorable.

2. US TEST MODULE PROGRAM STRATEGY

The initial conclusion of the US community, based on the results of the technical assessment to date, is to select two blanket concepts for the US ITER-TBM with the following emphases:

Select a helium-cooled solid breeder concept with ferritic steel structure and neutron multiplier, but without a fully independent TBM. Rather, plan on unit cell and submodule test articles that focus on particular technical issues of interest to all parties.

Focus on testing dual-coolant liquid breeder blanket concepts with ultimate potential for self-cooling. Develop and design TBM with flexibility to test two options:

1. A helium-cooled ferritic structure with self-cooled LiPb breeder zone that uses a flow channel insert, e.g., SiC as MHD and thermal insulator (insulator requirements in dual-coolant concepts are less demanding than those for self-cooled concepts).
2. A helium-cooled ferritic structure with low melting-point molten salt with Be as the neutron multiplier.

The choice of the specific lithium-containing molten salt between $LiBeF_3$ and FLiNaBe will be made based on near-term R&D experiments and modeling. Because of the low thermal and electrical conductivity of molten salts, no insulators are needed.

3. DUAL COOLANT OPTIONS AND INPUT PARAMETERS

3.1. DUAL COOLANT OPTION CONFIGURATION

The basic dual coolant FLiBe+Be blanket design option is shown in Fig. 3–1. A similar configuration can be envisioned with Pb-17Li breeder design option, but with the necessary flow channel insert to reduce the MHD drop and increase the thermal insulation in the poloidal breeder channels. This flow channel insert could be fabricated from SiC_f/SiC composite material. Furthermore, the Pb-17Li options will not need the use of Be to aid tritium breeding.

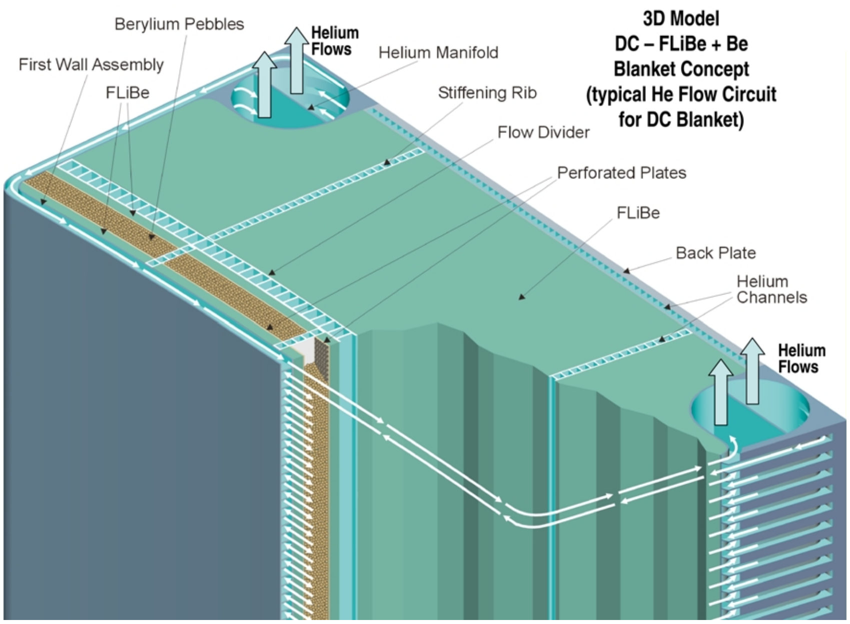

Fig. 3–1. DC FLiBe+Be blanket concept.

3.2. DC TEST MODULE DESIGN PARAMETERS

3.2.1. ITER Parameters

- ITER scenario: Fusion power-500 MW, burn time-400 s.
- Design heat flux: 0.5 MW/m^2 at steady state (earlier guide line was 0.25 MW/m^2 at steady state and 0.5 MW/m^2 with MARFEs for 10 s).
- Design neutron wall loading: 0.78 MW/m^2
  (Under high power operation, the outboard mid-plane could see 1.09 MW/m^2)
- Disruption heat load: 0.55 MJ/m^2
- Disruption heat load during current quench: 0.72 MJ/m^2
- Pulse length: 400 s/ <2000 s.

- Duty factor: <0.3.
- Half module frontal dimensions: 55.4 × 176.0 cm (0.975 m^2)

3.2.2. Test Module Design Input Parameters

	He/FW[(a)]	**$LiBeF_3$**	**FLiNaBe**	**Pb-17Li**
Average neutron wall loading, MW/m^2	0.78	0.78	0.78	0.78
Average surface heat flux, MW/m^2	0.5	0.5	0.5	0.5
Blanket M	1.23	1.2	1.23	1.15
1/2 module power, MW	0.57	1.4	1.423	1.362
Fraction of blanket power, %	40	100	100	100
T_{in}/T_{out}, °C	380/440	420/520	360/520	340/440
Coolant pressure, MPa	8	2[(b)]	2[(b)]	2[(b)]
Mass flow rate, kg/s	1.827	5.88	3.6	72
Volume flow rate, m^3/s	0.343	2.94×10^{-3}	1.9×10^{-3}	7.75×10^{-3}
Input max. flow speed, m/s	100	4	5	2
Material Properties:				
Melting point, °C	Gas	380	~305+15	235
Density, kg/m^3	5.33 @440°C	2000 @540°C	1900 @500°C	9300 @400°C
Specific heat, J/kg-K	5193	2380	2470 @700°C	189 @400°C
Fusion power onto the module, MW	0.951	0.951	0.951	0.951
Tritium breeding ratio[(c)]	1.25	1.25	1.25	1.25
Tritium generation rate,[(d)] gm/s	2.1×10^{-6}	2.1×10^{-6}	2.1×10^{-6}	2.1×10^{-6}
Tritium generation rate, #/s	4.2×10^{17}	4.2×10^{17}	4.2×10^{17}	4.2×10^{17}

[(a)]This column defines the parameters for the helium-cooled first-wall and structure cooling loop

[(b)]A conservative value from TBM coolant pressure drop of 0.6-0.8 MPa

[(c)]Local TBR round off number

[(d)]Normalized to 55.8 kg/year for 1000 MW-fusion (1.6x10-6 gm/s for 1 MW-fusion)

4. POWER MANAGEMENT AND INTERMEDIATE LOOP

For the DC concept, there are two coolant systems. The first one is the helium-cooled system removing the surface and nuclear power from the first wall and blanket structure. The second is the self-cooled liquid breeder system removing the nuclear power form the blanket.

4.1. FIRST WALL HELIUM LOOP

The first wall and structure helium loop will be a self-contained loop including heat transport, tritium extraction, helium purification, and heat exchanger to the TCWS plant cooling water. This system is designed to extract about 40% of the total power generated in the one-half module.

4.2. INTERMEDIATE LOOP BETWEEN LIQUID BREEDER AND WATER SYSTEM

To avoid long liquid breeder pipes running from the TBMs to the HX in the TCWS, we decided to utilize a helium coolant intermediate loop. A liquid breeder to helium heat exchanger is located close to the test module and will handle the liquid breeder with a temperature up to 650°C. This is to minimize the amount of liquid breeder and the corresponding loss of tritium to the surroundings. This approach is applied to both Pb-17Li and MS design options. The liquid breeder transport loop is designed to extract 100% of the total power generated in the one-half module. This allows the possibility for the testing of a complete self-cooled liquid breeder design option.

4.3. CONCENTRIC PIPES FOR THE LIQUID BREEDER ACCESS TUBES

A special issue for the DC coolant liquid breeder blankets is the design of the coolant access tubes. The goal is to achieve liquid breeder exit temperatures of ~650°C in order to enable the use of Brayton cycle power conversion systems. Such a high breeder exit temperature implies, in principle, the following problems:

1. Which structural material can be used for the coolant exit pipes?
2. Are the tritium permeation losses from the liquid breeder through the tube walls to the building atmosphere tolerable?
3. Which material can be used for the heat exchangers to the secondary helium?

The use of concentric tubes with the “hot” exit flow in the inner tube and the “cold” inlet flow in the annulus facilitates the points (1) and (2). The inner tube will always assume a temperature between the hot and cold breeder temperature. This temperature can be influenced by the ratio between the heat transfer coefficients in these two lines and, if required, by providing some thermal insulation in the inner tube. (e.g., gap filled with stagnant fluid). By this means, a maximum temperature of the tube wall below 550°C can be achieved even for a blanket exit temperature of 650°C.

Tritium permeating from the hot liquid breeder has no access to the environment but flows back into the blanket with the “cold” breeder. The temperature of the outer tube can be made much lower, reducing tritium permeation losses into the building atmosphere. Only the tube temperatures in the heat exchange of a power plant will be close to the maximum breeder temperature, but this temperature can significantly lowered in case of ITER TBMs.

An important point in designing concentric coolant access pipes is the possibility of using sliding seals for the inner tube. Only the outer tube has to be cut/rewelded for an exchange, since small leaks in the connection of the inner tube lead to a small bypass flow exclusively from the cold to the hot side. Such sliding seals also facilitate the compensation of differential thermal expansions of the two tubes and, as a result, the high temperature tube need only withstand a small pressure difference.

For these reasons we will apply concentric pipes wherever possible. This becomes another demonstration of the test module technology development directly applicable to the DEMO design.

In the following sections, most of the illustrated liquid breeder piping is in concentric pipes. However, detailed thermal analysis of the concentric piping design will need to be performed.

5. ANCILLARY SYSTEM

This chapter covers the description of ancillary equipment of the liquid breeder options and the first wall helium cooling loop. Corresponding tritium extraction systems are also presented.

5.1. Pb-17Li SYSTEMS

5.1.1. Overview of Flow Circuit Systems

See Fig. 5–1 for a working schematic of the Pb-17Li flow systems and Table 5–1 for a tabulation of assumed and calculated characteristics of the flow loop. Some points to note are listed below.

- The TBM is connected by concentric pipes that contain SiC flow channel inserts in the high magnetic field region beginning roughly 2 m from the back of the TBM.
- A fairly low velocity in the magnetic field region is required to avoid fault conditions with large pressure drops. This velocity was picked at 2 m/s in a rather arbitrary fashion. This sets the outer pipe diameter to be 0.16 m.
- Since the concentric tubes are attached at the bottom of the TBM, a bubble/pressure relief line is required at the top of the TBM to allow filling/draining and venting of bubbles. This line is arbitrarily chosen to have outer diameter 0.025 m.
- Both the main concentric lines and the bubble/pressure relief line penetrate the VV and bioshield plugs back to the transporter region where the rest of the Pb-17Li systems are housed.

Details of the Pb-17Li systems in the transporter and immediate vicinity are given below.

5.1.2. Pumping System

A similar pumping system as discussed in Ref. 5–1 (pp. 66, 76, 110, 114) is adopted with an expansion vessel feeding into a free surface, single stage, centrifugal pump with a long, vertical shaft. The shaft exits the pump housing above the level of the alloy Pb-17Li so seal compatibility with the alloy is not crucial. Such pumps are commonly used for pumping high temperature, heavy metal alloys and are both reliable and efficient. The pump electrical power is approximately 20 kW assuming 80% efficiency.

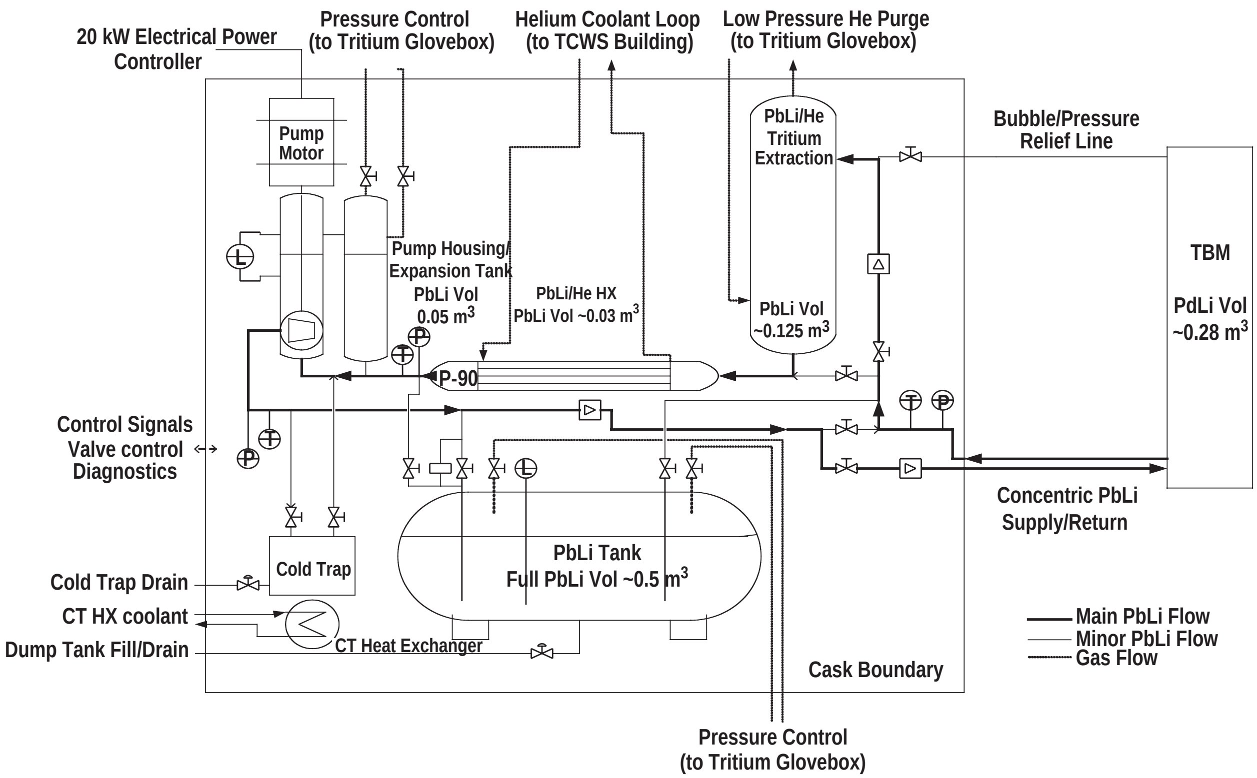

Fig. 5–1. Pb-17Li system layout.

Table 5–1
Pb-17Li Flow System Specifications
(Shaded indicates assumed/supplied value, others are calculated values)

TBM Base Case	Pb-17Li
Average neutron wall loading, MW/m^2	0.78
Average surface heat flux, MW/m^2	0.5
Blanket M	1.15
One-half port power, MW	1.36
TBM width, m	0.554
TBM height, m	1.76
TBM depth, m	0.4
TBM breeder volume fraction, %	0.70
Pb-17Li flow area in TBM (2 pass)	0.079
FW plasma facing area, m^2	0.98

Table 5–1 (Cont.)

Pb-17Li Volume Summary:	
TBM volume, m^3	0.28
Supply pipe volume, m^3	0.062
Pump/exp tank volume, m^3	0.050
Heat exchanger volume, m^3	0.027
Cold trap volume, m^3	0.020
Tritium extraction column, m^3	0.125
Pb-17Li Thermal Hydraulics:	
Pb-17Li T_{in}, °C	340
Pb-17Li T_{out}, °C	440
Pb-17Li pressure drop, MPa	2
Mass flow rate, kg/s	72.12
Volume flow rate, m^3/s	$7.74{\times}10^{-3}$
Pump efficiency, %	0.80
Pump electrical power, kW	19.35
Ave. Pb-17Li velocity in TBM, m/s	0.098
TBM Pb-17Li supply pipes	Concentric
Flow speed, m/s	2
Connection to TBM	Bottom
Pipe length, m	8
Flow area (each way), m^2	0.004
Inner pipe o.d., m (incl. 5 mm FCI)	0.090
Inner pipe wall thickness, m	0.005
Outer pipe outer wall thickness, m	0.010
Outer pipe o.d., m (incl. 2 5 mm FCIs)	0.161

The expansion vessel helps control the liquid metal level in the pump housing. The total Pb-17Li volume is roughly assumed to be 0.5 m^3 total in expansion tank and pump housing. The size of the expansion tank is estimated as a vertically oriented cylinder 1 m in height and 0.25 m in diameter. The size of the pump housing is identical, with an additional vertical height for the motor estimated at 0.5 m with diameter 0.4 m. Gas above the free surface level is controlled by a pressure control system (details not yet available) and is circulated to remove tritium and volatile heavy metal impurities (like Hg, see below).

The pump is located in the cold leg downstream of the heat exchanger.

5.1.3. Detritiation Unit

Current plans are to incorporate a detritiation unit for the Pb-17Li itself similar in size and capacity to that planned by the HCLL team [5–2]. The detritiation unit is a vertical bubble column with a helium purge that is connected to lines that lead to the glovebox in the tritium plant. The size of the cylindrical unit is 0.4 m diam and 1.6 m high. It is located in the hot leg, directly fed by the returning Pb-17Li from the TBM. It has a valved bypass system so not all returning Pb-17Li needs to be fed through the detritiation unit. The bubble/pressure relief line from the top of the TBM also feeds directly into this unit so that helium/tritium bubbles can be separated in the same way. For the tritium extraction from the helium coolant, with the use of a Pd/Ag permeator located in the helium circulation loop, an extraction unit that is $2 \times 1 \times 1$ m is assumed. In addition, a pumping package of similar dimension will be needed.

5.1.4. Pb-17Li to Helium Heat Exchanger

A heat exchanger has been designed to allow rejection of the full TBM thermal power from the Pb-17Li stream to a secondary helium cooling system that leads to the TCWS building. This heat exchanger is roughly 0.25 m in diameter and 1.4 m in length with details of its design and analysis given in Table 5–2.

The heat exchanger has the capacity to operate between two extreme modes. The base operating mode has Pb-17Li flowrate and temperature rise as indicated in the base case summarized in Table 5–1. An alternate operating extreme is to have the Pb-17Li peak temperature much higher, at around 650°C, with correspondingly lower Pb-17Li mass flowrate. This is to test the true dual coolant feature of high outlet temperature. In both cases, the helium outlet of the HX should not be higher than 400°C to keep permeation from helium pipes to the TCWS building manageable.

5.1.5. Liquid Breeder Purification System

Liquid metal purification systems are, in general, required to control the oxygen content of the system, remove corrosion products, replenish depleted lithium, and remove heavy metal isotopes [5–3]. Diffusion-type cold traps are effective in removing many of these compounds including Polonium. Such a system should be used in the Pb-17Li ancillary loop as a bypass on the cold leg. It is expected that activated mercury vapor may accumulate in all cover gas areas [5–3]. A removal system for mercury has not yet been designed but will be needed on the various cover gas systems controlling the dump tank, pump/expansion vessel housing and detritiation unit. Close coordination on this issue with the HCLL group will be pursued. For now, cold trap size should be estimated based on HCLL [5–1] and will require a separate low-temperature coolant for the local heat exchanger.

Table 5–2
Pb-17Li/He Heat Exchanger

ITER Calculation By: D. P. Carosella	General Atomics **Calculation Sheet** System ITER Test Loop Title: Pb_17Li to Helium Heat Exchanger Design Max Size	Page No.: 36 Calculation No.: 1

IV. RESULTS:

Summary of Design Data for the Helium to Pb-17Li Heat Exchanger:

	Metric		English	
Heat Duty:	Q_{hex}	1362 kW	Q_{hex}	4.647 10^{6} BTU hr^{1}
Effectiveness/NTU :	eff	0.462 / NTU	0.801	
Design Uncertainty	UN_{ht}	15 %		
Pb-17Li Data :				
Flow Rate:	W_{PbLi}	72.08 kg sec^{1}	W_{PbLi}	158.91 lb sec^{-1}
Inlet Temperature:	$TCPbLi_{in}$	440 C	$TFPbLi_{in}$	824 F
Outlet Temperature:	$TCPbLi_{out}$	340 C	$TFPbLi_{out}$	644 F
Pressure Drop:	P_{PbLi}	0.0529 MPa	P_{PbLi}	7.669 psi
Percent Pressure Drop:	$\frac{P_{PbLi}}{P_{PbLimax}}$	5.29 %		
Pumping Power	PP_{PbLi}	408.33 watt	PP_{PbLi}	0.548 hp
Helium Data:				
Flow Rate:	W_{He}	2.2 kg sec^{1}	W_{He}	4.8 lb sec^{-1}
Inlet Temperature:	$TCHe_{in}$	180 C	$TFHe_{in}$	356 F
Outlet Temperature:	$TCHe_{out}$	300 C	$TFHe_{out}$	572 F
Pressure Drop:	P_{tot}	0.075 MPa	P_{tot}	10.9 psi
Percent Pressure Drop	$\frac{P_{tot}}{P_{He}}$	0.94 %		
Pumping Power:	PP_{He}	21.87 kW	PP_{He}	29.33 hp
Maximum He Velocity in Tubes	$VelHe_{out}$	45.8 m sec^{1}	$VelHe_{out}$	150.3 ft sec^{-1}
Velocity In Pipes:	$VelHe_{max}$	68.1 m sec^{1}	$VelHe_{max}$	223.5 ft sec^{-1}
Velocity In The Inlet Orifice	Vel_{ori}	100.8 m sec^{1}	Vel_{ori}	330.6 ft sec^{-1}

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Table 5–2 (Cont.)

ITER	General Atomics **Calculation Sheet**	Page No.: 37
Calculation By: D. P. Carosella	System ITER Test Loop Title: Pb_17Li to Helium Heat Exchanger Design Max Size	Calculation No.: 1

Summary of Design Data for Pb-17Li to Helium Heat Exchanger:

	Metric	English
Geometric Data		
Overall Dimensions		
Total Length	TL_{tot} 1.33 m	TL_{tot} 4.363 ft
Shell Outside Diameter	SOD 0.2527 m	SOD 9.951 in
Heat Transfer Surface Area	TubA 4.22 m^2	TubA 45.45 ft^2
Total Tube Length (Including Tube Sheets)	TL 0.93 m	TL 3.06 ft
Heat Transfer Height:	H 0.876 m	H 2.87 ft
Inside Shell Diameter:	Hexdia 0.2411 m	Hexdia 9.494 in
Head Height:	b 0.06 m	b 2.49 in
Gas Supply & Return Pipe Dia:	dia_{He} 77.9272 mm	dia_{He} 3.07 in
Pb-17Li Supply & Return Pipe Dia:	dia_{PbLi} 62.7126 mm	dia_{PbLi} 2.47 in
Hot Tube Sheet Thickness	TST_{hot} 0.056 m	TST_{hot} 2.22 in
Baffle Thickness:	baffle 6.35 mm	baffle 0.25 in
Total Dry Weight:	Wt_{tot} 95.8 kg	Wt_{tot} 211.3 lb
Tube Parameters:		
Number of Tubes	Ntube 56	
Triangular Pitch	Pitch 17.94 mm	Pitch 0.7064 in
Tube Inside Diameter:	$Tube_{ID}$ 12.7 mm	$Tube_{ID}$ 0.5 in
Tube Outside Diameter:	$Tube_{OD}$ 13.7 mm	$Tube_{OD}$ 0.539 in
Tube Wall Thickness:	wall 0.5 mm	wall 0.02 in
He Inlet Orifice Diameter:	$diaHe_{ori}$ 7.6 mm	$diaHe_{ori}$ 0.3 in
Auxiliary Data		
Total Shell Side Volume	Vol_{ss} 0.028 m^3	Vol_{ss} 0.991 ft^3
Tube Side (He) Nusselt No.	Nu_{He} 251.7905	
Tube Side (He) Heat Transfer Coef.	h_{He} 4414 $\frac{watt}{m^2\ K}$	h_{He} 777 $\frac{BTU}{hr\ ft^2\ R}$
Shell Side (Pb-17Li) Nusselt No.	Nu_{PbLi} 10.6839	
Shell Side (Pb-17Li) HT Coef.	h_{PbLi} 8718 $\frac{watt}{m^2\ K}$	h_{PbLi} 1535 $\frac{BTU}{ft^2\ R\ hr}$

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5.2. MOLTEN SALT SYSTEMS

5.2.1. Overview of Flow Circuit Systems

Figure 5–2 illustrates a schematic of the $LiBeF_3$ flow systems and Table 5–3 lists a tabulation of assumed and calculated characteristics of the flow loop. Some points to note are listed below:

- The TBM is connected by concentric pipes behind the test module to the $LiBeF_3$/helium heat exchanger. At an assumed inlet and outlet flow velocity of 4 m/s, the annular tube has an inner diameter of 0.03 m and external diameter of 0.08 m. The selected tube wall thickness is 0.01 m.
- The $LiBeF_3$ annular pipes are attached to the bottom of the TBM, and a bubble/pressure relief line is located at the top of the TBM to allow fluid filling, draining and venting of the gaseous product.
- The $LiBeF_3$ flow is pumped by the sump-type high temperature centrifugal pump and connected to an expansion tank of similar dimensions.
- The $LiBeF_3$ dump tank and the corresponding flush tank are connected at the bottom of the LBM, and helium gas pressure control can be used to load the $LiBeF_3$ into the TBM.
- All the $LiBeF_3$ containers and piping are electrically heated to ensure that the fluid can be circulated effectively.

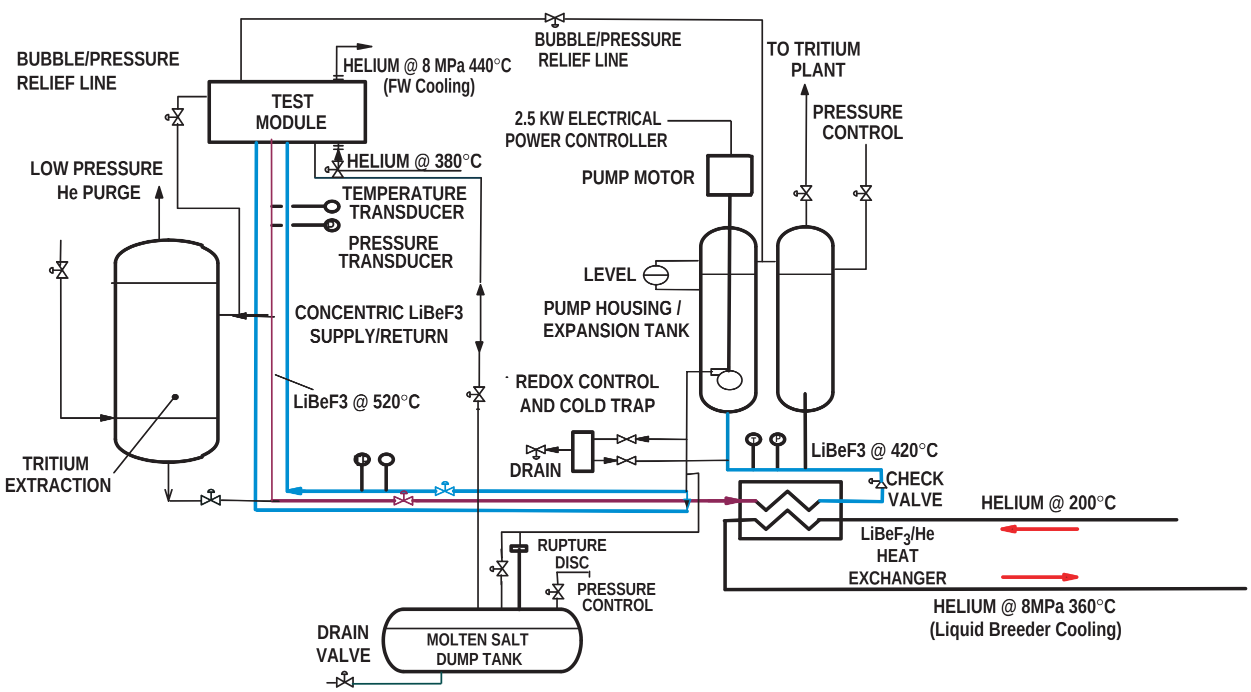

Fig. 5–2. DC $LiBeF_3$ test module subsystem flow diagram.

Table 5–3
$LiBeF_3$ Test Module Parameters

	$LiBeF_3$
Average neutron wall loading, MW/m^2	0.78
Average surface heat flux, MW/m^2	0.5
Blanket M	1.2
1/2 module power, MW	1.4
T_{in}/T_{out}, °C	420/520
TBM pressure drop, MPa	0.6–0.8
Mass flow rate, kg/s	5.88
Volume flow rate, m^3/s	2.94×10^{-3}
Pump electrical power, kW	2.5
Average MS velocity in TBM, m/s	0.1
Breeder volume fraction	0.66
Breeder colume TBM, m^3	0.26
TBM and piping $LiBeF_3$ Volume, m^3	0.5
Dump tank volume, m^3	0.75
TBM width, m	0.554
TBM height, m	1.76
TBM depth, m	0.4
FW plasma facing area, m^2	0.98
Breeder pipe style	Concentric
Supply pipe connection to TBM	Bottom
Average MS flow velocity, m/s	4
Inlet pipe ID, m	0.031
Inner pipe wall thickness, m	0.01
Outer pipe wall thickness, m	0.01
Concentric tube outside D, m	0.08
$LiBeF_3$ material:	
Melting point, °C	380
Density @ 540°C, kg/m^3	2000
Specific heat, J/kg-K	~2380
Tritium breeding ratio	1.25
Tritium generation rate, #/s	4.2×10^{17}
Tritium generation rate, gm/s	2.1×10^{-6}

- What is not shown in Fig. 5–2 are all the other necessary gaseous control and purification systems connected to all the $LiBeF_3$ containers.

Details of the $LiBeF_3$ systems in the transporter and surrounding regions are given in Table 5–3.

5.2.2. Pumping System

The $LiBeF_3$ breeder pump is located in the cold leg downstream of the heat exchanger. The fluid is pumped by a sump-type, single stage centrifugal pump with a long vertical shaft feeding into a free surface as shown in Fig. 5–2. The shaft exits the pump housing above the level of alloy — consequently seal compatibility with the alloy is not crucial. Such pumps are commonly used for pumping molten salt at high temperature and are both reliable and efficient. The required pump electrical power is approximately 2.5 kW assuming an 80% efficiency. The pump housing is connected to an expansion vessel, which helps to control the molten salt level in the pump housing. The total $LiBeF_3$ volume is roughly assumed to be 0.05 m^3 total in expansion tank and pump housing. The size of the expansion tank is estimated as vertically oriented cylinder 1 m in height and 0.25 m in diameter. The size of the pump housing is identical, with an additional vertical height for the motor estimated at 0.5 m with diameter 0.3 m. Gas above the free surface level is controlled by a pressure control system (details not yet available) and is circulated to remove tritium and volatile impurities.

5.2.3. Detritiation Unit

It is currently planned to have a detritiation unit for the $LiBeF_3$ itself similar in size and capacity to that planned by the HCLL team [5–4]. It is a vertical bubble column with helium purge that is connected to lines that go to the glove box in the tritium plant. The size of the cylindrical unit is 0.4 m diam and 1.6 m high. It is located in the hot leg, directly fed by the returning $LiBeF_3$ from the TBM. It has a valved bypass system so not all returning $LiBeF_3$ needs to be fed through the detritiation unit. The bubble/pressure relief line from the top of the TBM also feeds directly into this unit so that helium/tritium bubbles can be separated in the same way.

For the tritium extraction from the helium coolant, with the use of a Pd/Ag permeator located in the He circulation loop, an extraction unit that is 2 m long and 1 m diam is assumed. In addition, a pumping package of similar dimension will be needed.

5.2.4. $LiBeF_3$ to Helium Heat Exchanger

A heat exchanger has been designed to allow rejection of the full TBM thermal power from the $LiBeF_3$ stream to a secondary helium cooling system that goes to the TCWS building. This heat exchanger is roughly 0.37 m in diameter and 2.4 m in length with details of its design and analysis given in Table 5–4.

The heat exchanger has the capacity to operate between two extreme modes. The base operating mode has $LiBeF_3$ flowrate and temperature rise as indicated in the base case summarized in Table 5–3. An alternate operating extreme is to have the $LiBeF_3$ peak at an outlet temperature of 650°C, with correspondingly lower $LiBeF_3$ mass flowrate. This is to test the dual coolant feature of high outlet temperature. In both cases, the helium outlet of the heat exchanger should not be higher than 400°C to keep tritium permeation from helium pipes to the surrounding buildings manageable.

Table 5–4
$LiBeF_3$/He Heat Exchanger Details
Summary of Design Data for the $LiBeF_3$ to Helium Heat Exchanger

	Metric	English
Geometric Data		
Overall Dimensions		
Total Length	$TL_{tot} = 2.403\,m$	$TL_{tot} = 7.884\,ft$
Shell Outside Diameter	$SOD = 0.3697\,m$	$SOD = 14.553\,in$
HT Surface Area (Includes Fins)	$A_{ht} = 37.07\,m^2$	$A_{ht} = 399.04\,ft^2$
Tube Area	$TubA = 15.3\,m^2$	$TubA = 165.1393\,ft^2$
Total Tube Length (Including Tube Sheets)	$TL = 1.85\,m$	$TL = 6.08\,ft$
Inside Shell Diameter:	$Hexdia = 0.3527\,m$	$Hexdia = 13.885\,in$
Head Height:	$b = 0.09\,m$	$b = 3.64\,in$
Gas Supply & Return Pipe Dia:	$dia_{He} = 77.9272\,mm$	$dia_{He} = 3.07\,in$
LiBeF Supply & Return Pipe Dia:	$dia_{LiBeF} = 62.7126\,mm$	$dia_{LiBeF} = 2.47\,in$
Tube Sheet Thickness	$TST_{hot} = 0.071\,m$	$TST_{hot} = 2.78\,in$
Total Dry Weight:	$Wt_{tot} = 378.1\,kg$	$Wt_{tot} = 833.6\,lb$
Tube Parameters:		
Number of Tubes	$Ntube = 100$	
Triangular Pitch	$Pitch = 20.05\,mm$	$Pitch = 0.7894\,in$
Tube Inside Diameter:	$Tube_{ID} = 12.7\,mm$	$Tube_{ID} = 0.5\,in$
Tube Outside Diameter:	$Tube_{OD} = 13.7\,mm$	$Tube_{OD} = 0.539\,in$
Tube Wall Thickness:	$\Delta_{wall} = 0.5\,mm$	$\Delta_{wall} = 0.02\,in$
Fin Thickness:	$Fin\Delta = 0.5\,mm$	$Fin\Delta = 0.0197\,in$
Fin Depth:	$FinH = 3.048\,mm$	$FinH = 0.12\,in$
Number of Fins/Tube	$NFin = 10$	
Orifice Diameter:	$diaHe_{ori} = 6.65\,mm$	$diaHe_{ori} = 0.26\,in$
Heat Duty:	$Q_{hex} = 1400\,kW$	$Q_{hex} = 4.777 \times 10^6\,BTU \cdot hr^{-1}$
Effectiveness/NTU:	$eff = 0.5$ / $NTU = 0.849$	
Design Uncertainty	$UN_{ht} = 15\,\%$	
LiBeF Data:		
Flow Rate:	$W_{LiBeF} = 5.88\,kg \cdot sec^{-1}$	$W_{LiBeF} = 12.97\,lb\,sec^{-1}$
Inlet Temperature:	$TCLiBeF_{in} = 520\,C$	$TFLiBeF_{in} = 968\,F$
Outlet Temperature:	$TCLiBeF_{out} = 420\,C$	$TFLiBeF_{out} = 788\,F$
Pressure Drop:	$\Delta P_{LiBeF} = 0.027\,MPa$	$\Delta P_{LiBeF} = 3.919\,psi$
Percent Pressure Drop:	$\frac{\Delta P_{LiBeF}}{P_{LiBeFmax}} = 2.7\,\%$	
Pumping Power	$PP_{LiBeF} = 78.15\,watt$	$PP_{LiBeF} = 0.105\,hp$
Helium Data:		
Flow Rate:	$W_{He} = 1.7\,kg \cdot sec^{-1}$	$W_{He} = 3.7\,lb\,sec^{-1}$
Inlet Temperature:	$TCHe_{in} = 200\,C$	$TFHe_{in} = 392\,F$
Outlet Temperature:	$TCHe_{out} = 360\,C$	$TFHe_{out} = 680\,F$
Pressure Drop:	$\Delta P_{tot} = 0.033\,MPa$	$\Delta P_{tot} = 4.81\,psi$
Percent Pressure Drop	$\frac{\Delta P_{tot}}{P_{He}} = 0.41\,\%$	
Pumping Power:	$PP_{He} = 8.03\,kW$	$PP_{He} = 10.76\,hp$
Maximum He Velocity in Tubes	$VelHe_{out} = 21.8\,m \cdot sec^{-1}$	$VelHe_{out} = 71.7\,ft\,sec^{-1}$
Velocity In Pipes:	$VelHe_{max} = 58\,m \cdot sec^{-1}$	$VelHe_{max} = 190.4\,ft\,sec^{-1}$
Velocity in the Orifices	$Vel_{ori} = 59.5\,m \cdot sec^{-1}$	$Vel_{ori} = 195.3\,ft\,sec^{-1}$

5.2.5. TF Control and Purification System

In a nuclear application, Be will react with neutrons in a (n,2n) reaction to generate additional neutrons and He, while the Li in $LiBeF_3$ will react with neutrons to produce tritium. With these reactions, either free fluorine or TF will be formed. HF (or TF) will react with many structural materials, including Fe and Cr. Therefore, it is important to control TF activities in the molten salt. This process is called REDuction-OXidation (REDOX) process.

The reduction process is to control all the TF into T_2 form. The most effective reducing agents are Li and Be. Therefore, we can use the following reaction to control the TF activities:

$$Be + 2TF \longrightarrow BeF_2 + T_2 \quad .$$

Based on the free energy of formation, this reaction is very favorable from thermodynamic considerations. However, it is not certain if kinetically this reaction is fast enough. The $LiBeF_3$ REDOX experimental program at INEEL is to confirm the kinetics of this reaction. At the same time, the corrosion product BeF_2 will need to be removed. A REDOX and corresponding corrosion products control system for the $LiBeF_3$ design has not been performed. However, the expected equipment system dimension will be approximately TBD m^3 in volume for the $LiBeF_3$ test module.

5.3. HELIUM COOLING SUBSYSTEMS

5.3.1. Subsystems Description

The helium cooling subsystems include two helium loops: the primary first wall to helium heat transport loop (FW loop), and the liquid breeder to helium heat transport loop (LB loop), which connects to the secondary helium to water loop. Liquid breeders considered are the Pb-17Li and $LiBeF_3$ options. The helium cooling subsystem which interfaces with the secondary water loop as part of the ITER tokamak cooling water system (TCWS). This system has been designed to supply cold water with a temperature of 35°C and accept hot water with a temperature of 75°C. The pressure is moderate — lower than 1 MPa.

The helium cooling subsystems are to be housed in the TCWS vault approximately 70 m away from the TBM (Figs. 5–3 and 5–4). The pipe routing is to be described in Section 7. The piping must be placed 18 m horizontally and 14 m vertically within the shaft and again 60 m horizontally plus 10 m between components. U-bend expansion loops are to be included, as required, to mitigate thermal stresses due to the high temperature operating conditions. This results in a total length for the hot leg and cold leg of about 100 m and 95 m, respectively, or similar length when concentric pipes are used.

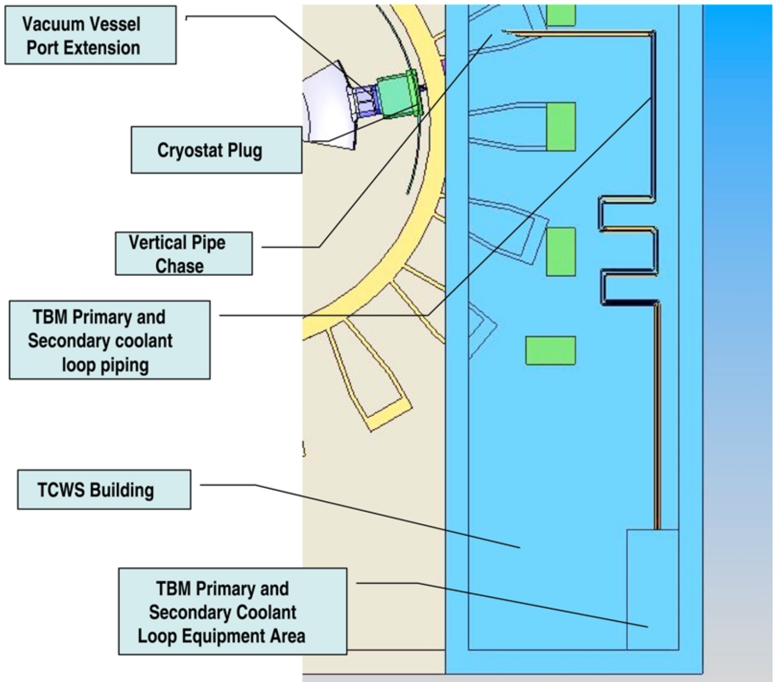

Fig. 5–3. Piping from TBM to TCWS building.

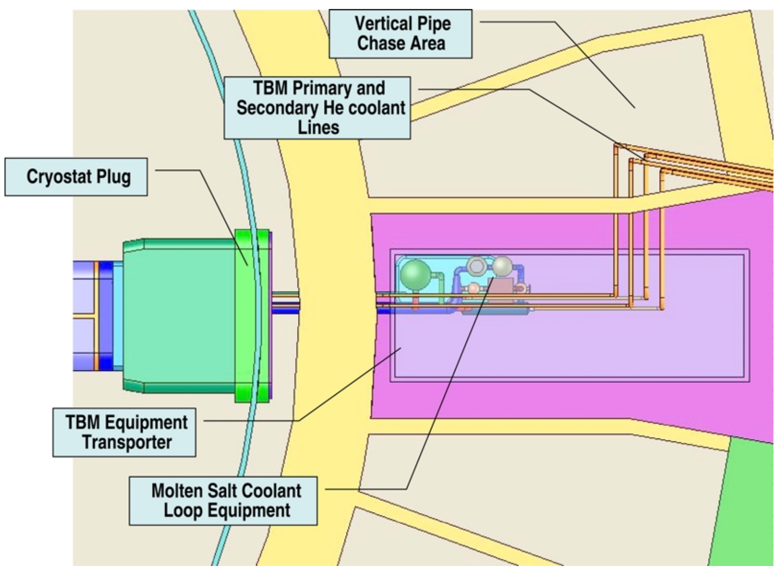

Fig. 5–4. TBM and transfer cask.

The envisaged space division among parties is as illustrated in Fig. 5–5). It is assumed that there will be no crane available for vertical component handling. A total foot print size of 20 m^2 will be needed for the FW loop and LB loop helium cooling subsystems and approximately 5 m height in the TCWS vault.

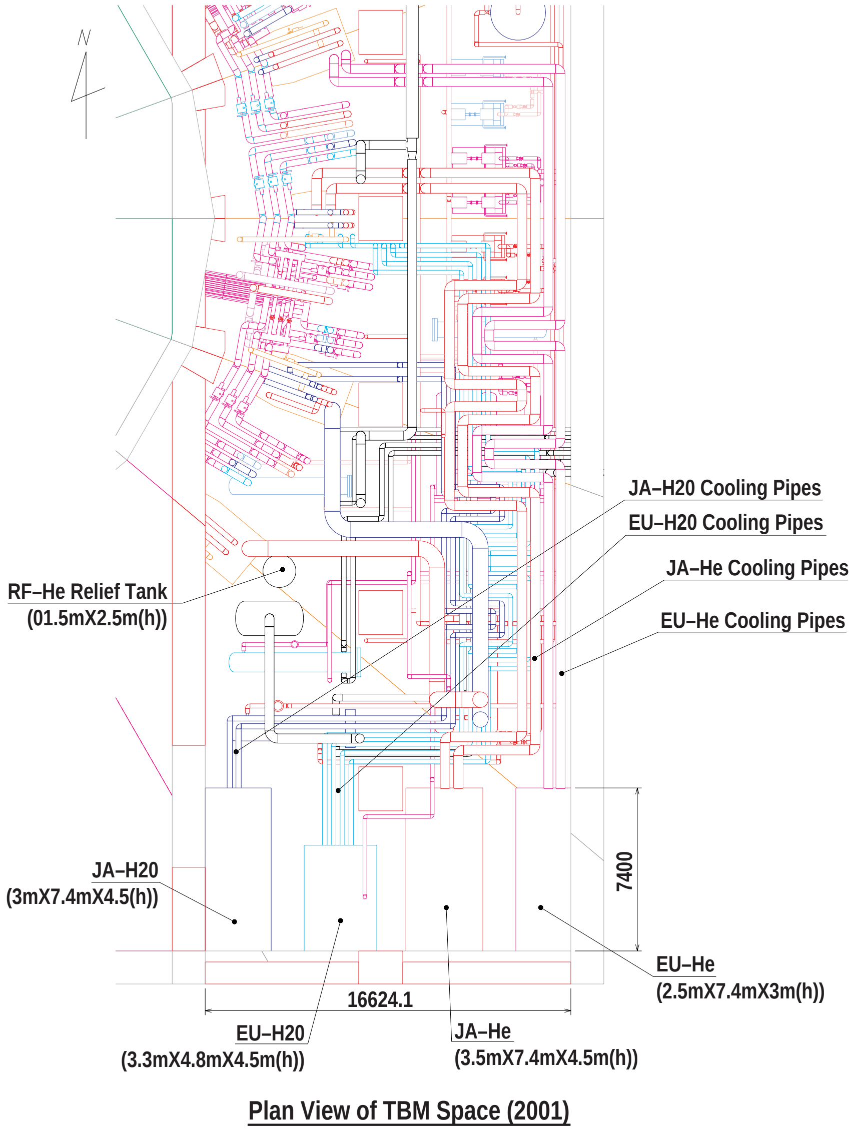

Fig. 5–5. Arrangement of piping and TBM ancillary equipment areas.

Figure 5–6 shows a flow diagram of the FW loop and interfaces to ancillary equipment. Figures 5–7 and 5–8 show the flow diagrams of the LB loops for Pb-17Li and $LiBeF_3$ liquid breeder options, respectively. Related main components for the FW loop and one of the LB loops and their arrangement in the allocated space are illustrated in Section 7. Besides the main components, several sets of tanks, a rack for pressure control equipment and a cubicle for local electrical equipment have to be accommodated for the two loops. A more detailed component description is given in Section 5.3.2.

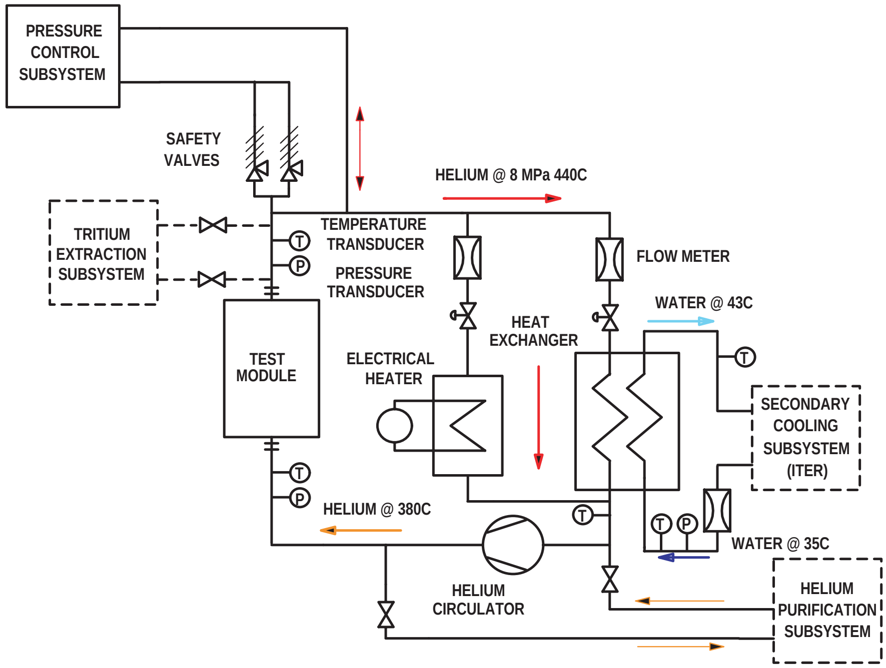

Fig. 5–6. Helium cooling subsystem flow diagram.

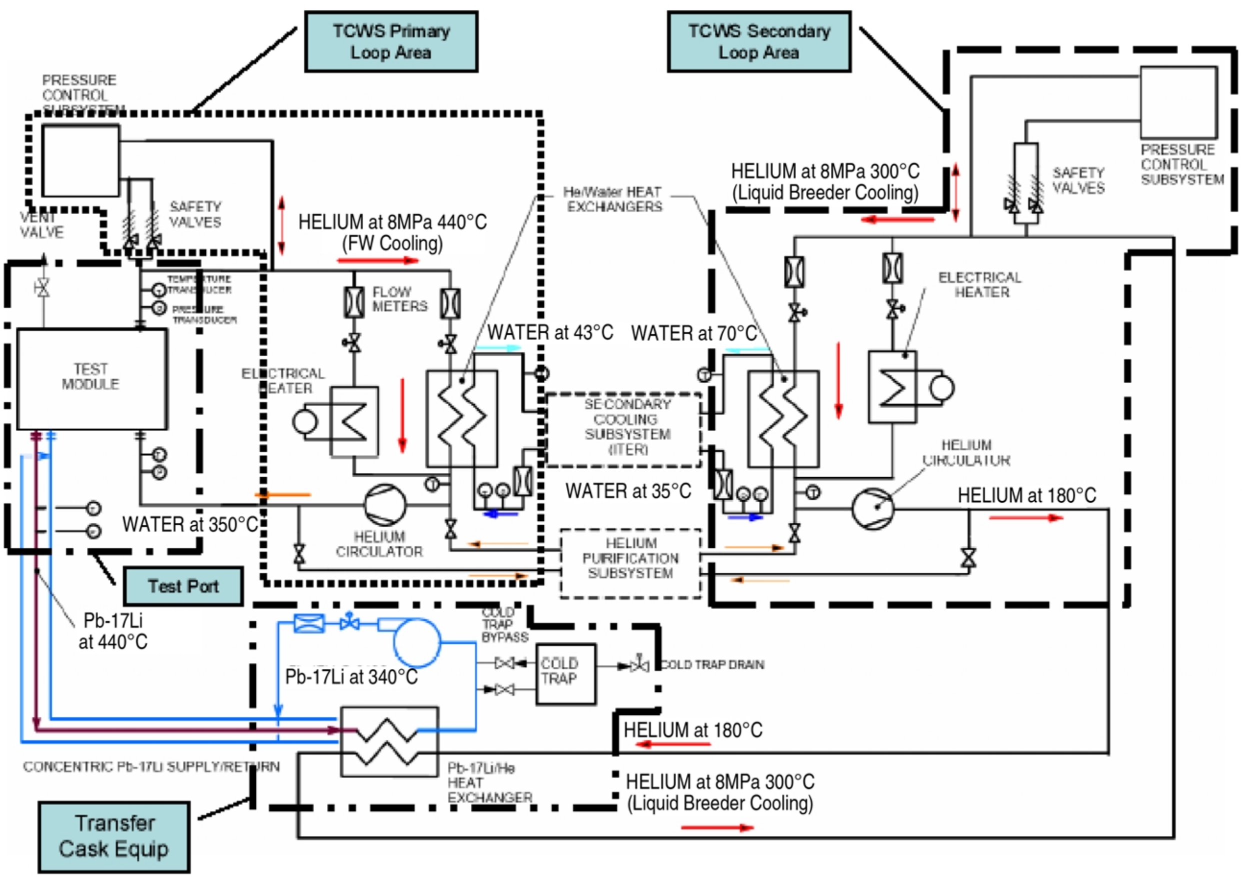

Fig. 5–7. The Pb-17Li loop.

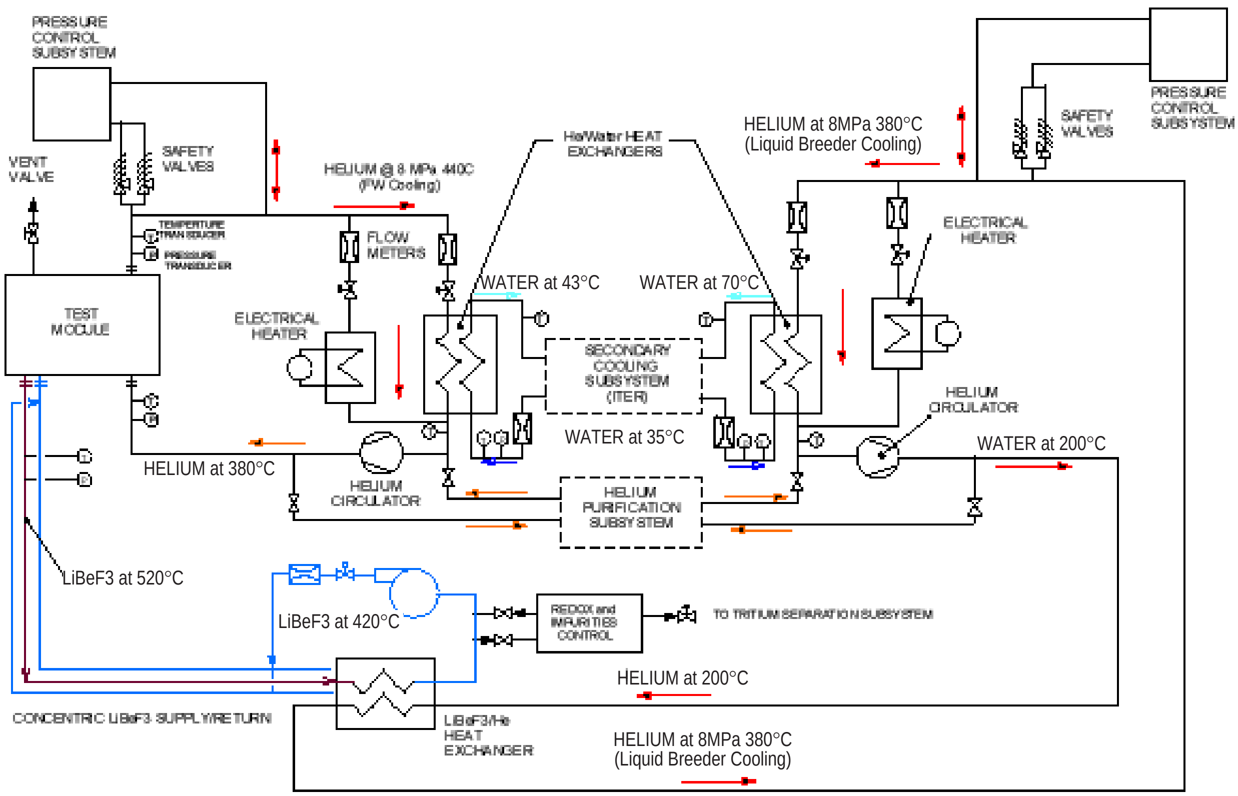

Fig. 5–8. The $LiBeF_3$ loop.

The control panel for operation/monitoring of the cooling subsystem will be installed in the main control room requiring a space of approximately 1 to 2 m^2 for each one of the two loops. Small electrical control equipment, like signal transducers, are planned to be placed inside the transfer cask, where also some equipment of the tritium monitoring and extraction subsystem is housed.

5.3.1.1. Thermal Design Requirements. The helium cooling subsystem must be designed for the operating conditions given in Table 5–5.

5.3.1.2. Layout of Heat Removal Loops

First Wall Loop. Main components of the FW helium cooling loop are (besides the test module) a heat exchanger, circulator, electrical heater, control valves, and pipework. A flow diagram is shown in Fig. 5–6. The primary loop is directly connected to the helium purification subsystem via small pipes taking a small bypass flow. The primary function of the purification system is to remove tritium that diffuses into the helium coolant. No other helium cleanup provisions are required, such as a dust filter, since there are no sources of dust or other contaminants. Further interfaces are shown in the flow diagram to the pressure control unit, which is needed for system evacuation, helium supply, and protection against overpressure. Also shown is the minimum required instrumentation for process control.

Table 5–5
Design Parameters of FW and LB Cooling Subsystems

	First Wall Helium/Water	Liquid Breeder, Pb-17Li/Helium/Water	Liquid Breeder, $LiBeF_3$/Helium/Water
Inner port dimensions (width x height), m × m	1.31 × 0.78	1.31 × 0.78	1.31 × 0.78
Projected area of module facing the plasma, m × m	1.27 × 0.74	1.27 × 0.74	1.27 × 0.74
Surface heat flux, MW/m^2	0.5	0.5	0.5
Neutron wall loading, MW/m^2	0.78	0.78	0.78
Total heat to be removed, MW	0.57	1.36	1.4
Temperature at module in/out, °C	380/440	340/440	420/520
Pressure, MPa	8	2	2
Number of circuits	1	2	2
Mass flow rate, kg/s	1.82	72	5.88
Secondary coolant	Water	Helium:water	Helium:water
Temperature at heat exchanger in/out, °C	35/43	180/300:35/70	200/360:35/70
Pressure, MPa	<1	8:<1	8:<1
Number of circuits	1	1	1
Mass flow rate, kg/s	17 (water)	2.2 (helium)	1.7 (helium)

An overview of thermal-hydraulic data such as pressure loss, helium volume, and helium mass inventory in the different components is displayed in Table 5–6. The overall pressure loss is 0.65 MPa at the extreme load conditions. During operation the total helium mass inventory in the loop including the buffer tank and 10% margin amounts to 18.26 kg.

The primary loop components are described in Sections 5.3.2 with a summary of the main dimensions and masses involved, as far as thermal inertia is concerned, being displayed in Table 5–7. Heat losses are assessed in Section 5.3.1.9. An overview of component size is given in Table 5–7.

Pb-17Li Breeder Loop. As an option for the liquid breeder test module design, Fig. 5–7 shows the layout of the Pb-17Li breeder loop. It includes the intermediate breeder to helium loop. Details of the Pb-17Li breeder circulation system are given in Section 5.1. Main components of the Pb-17Li breeder helium cooling loop are identical to the FW helium loop as described above.

$LiBeF_3$ Breeder Loop. As the second option for the liquid breeder test module design, Fig. 5–8 shows the layout of the $LiBeF_3$ breeder loop. It consists of the intermediate breeder to helium loop. Details of the $LiBeF_3$ breeder circulation system is given in Section 5.2. Main components of the $LiBeF_3$ helium cooling loop are identical to the FW helium loop as described above.

Table 5–6
First Wall Helium Cooling Loop Pressure Loss and Helium Inventory Under Extreme Operating Conditions

Component	Pressure Loss: MPa	Helium Volume (m^3)	Helium Mass (kg)
Hot leg pipework	0.078	0.82	4.433
Cold leg pipework	0.068	0.78	4.599
Main pipe elbows	0.172	Incl. in pipes	Incl. in pipes
Bypass to heat exchanger	0.029	0.0148	0.0873
Valves	0.166	0.0011	0.004
Heat exchanger	0.044	0.010	0.040
Circulator	—	0.3	1.78
Electrical heater	Bypassed	0.043	0.184
Buffer tank	—	0.236	4.92
Test module	0.094	0.09	0.51
Totals	0.65	2.295	16.52[a]

[a]10% margin not shown.

Table 5–7
Enveloping Dimensions and Weights of the First Wall Helium Cooling Loop Components (Dimensions Not Including Thermal Insulation)

Component	Number per loop	Diameter (m)	Length (m)	Weight (kg)
Pipework (including bypass)	1	0.1023	180	2959
Heat exchanger	1	0.223 (o.d.)	0.63	45.7
Circulator	1	1.54	0.30	1979
Circulator motor	1	0.88	1.46	1670
Electrical heater	1	0.35	1.65	287
Helium storage tanks	9	0.4	2.6	4808
Helium dump tanks	4	0.4	2.6	2137
Buffer tank	1	0.4	2.6	534
Test module	1			2000
Total weight				16420

FLiNaBe Breeder Loop. As mentioned in Section 2, a third liquid breeder option is the molten salt FLiNaBe, which has a lower melting point than $LiBeF_3$ and a projected lower viscosity. Basic research is ongoing to quantify the thermal-physical properties of this molten salt. We expect that the thermal performance of FLiNaBe would be higher than $LiBeF_3$, and the general heat removal and tritium control approach should be similar to both liquid molten salt breeders. We will evaluate this breeder option further when more is known about its thermal and physical properties.

As stated in Section 2, the US intends to finalize the selection of the reference liquid breeder option in about two years.

5.3.1.3. TCWS Secondary Heat Removal Loop. As noted above, both FW and LB loops will be connected to the secondary water cooled system. The secondary heat removal system is considered to be part of the ITER cooling system. For the layout of the helium cooling subsystem, the secondary heat removal system was assumed to provide a steady water flow with nominal inlet temperature of 35°C at the secondary side of the heat exchanger (see Section 5.3.2.1). For the FW cooling loop, Al piping is used to reduce the permeation of tritium into the water cooling system. At a water flow rate of about 17 kg/s, the outlet temperature of water is at 43°C. This reduced temperature is selected to maintain the Al piping at a maximum temperature of <160°C. Other means of reducing tritium permeation will also be evaluated in the future. Some of the alternatives are Al sleeve/coating and Alumina coating on SS tubes. These alternatives could allow higher water outlet temperature operation.

For the LB loops, the intermediate loop helium is connected to the secondary water cooled system. The outlet temperature will then vary according to the burn and dwell cycles between 70°C and 35°C. Flow, pressure, and temperature monitoring are needed.

Different from that indicated in Figs. 5–7 and 5–8, the recommended piping from the liquid-breeder/helium heat exchange to the helium/water exchanger is connected also by concentric pipes to reduce thermal tritium losses. Al sleeve and/or Alumina coating applying to the external SS pipes will also be assessed to further control the migration of tritium.

5.3.1.4. Maintenance/Remote Handling. Activation of cooling subsystem components installed in the TCWS vault is expected to be generally low allowing controlled personnel access after plant shutdown. In-service inspection such as examination of selected welds by different methods (visual, eddy current, ultrasonic), inspection of circulator internals, functional tests of valves, leak tightness of heat exchangers, etc., occur during test module change-out or during planned or unplanned machine shutdown periods.

Remote handling is envisaged for connection and disconnection of the TBM. The procedure will be standardized for all the test modules. For this purpose all the FW loop the attachment to the frame, pipe size, elbow radii close to the TBM, tools, etc., are assumed to be standardized details developed by ITER. At present, pipe sections of about 10.5 m length of the hot leg and cold leg each next to the TBM with enlarged inner/outer diameters of 102.3/114.3 mm are foreseen. In the case of any defects in heat exchangers or electrical heaters, replacement of the whole component may be more appropriate than an in-situ repair.

TBM dismantling and transfer to the hot cells after testing will be performed with the aid of the transfer cask, beginning after a shutdown period of at least one day. The decay heat of the whole module is expected to be less than 1 kW at that time which would result in an adiabatic heating up of the isolated TBM at a rate of about 2 K/h. It is assumed that during the dismantling procedure, the vacuum vessel (VV) is vented and purged with dry air at 50°C. Under these conditions, it is expected that the decay heat of the TBM can be dissipated to the surrounding without active cooling. Flooding the transfer cask with dry air during the transfer period would likewise stabilize the TBM temperature at a tolerable level. Thus, no active cooling of the TBM is needed during dismantling and transfer.

5.3.1.5. Assembly. All components of the cooling subsystems such as heat exchanger, circulator, electrical heater, tanks, and valves will be pre-assembled at the factory and delivered to the site as functional units. Connection of the components will be performed on site by conventional means. They are mounted on the floor of the TCWS vault and require space for horizontal translation by means of a lift truck with a load capacity of about 1000 kg. Field-welded joints will be subjected to surface and/or volumetric inspection, followed by pressure and leak tests. Thermal insulation will be installed after leak testing of the loop.

An exception is the installation of the test module. It will be brought into place by the aid of a special transfer cask that is aligned with the test port. It is to be equipped with all tools needed for positioning, aligning, locking, and connecting the module in the test port.

5.3.1.6. Subsystem Startup, Control, and Shutdown. For the first startup or startup after a major repair, the cooling subsystem is assumed to be clean and proof tested, components are at room temperature and filled with air. Correspondingly, the ITER machine is supposed to be simultaneously conditioned for startup. The following steps will then be taken with the cooling subsystem:

- Subsystem evacuation to <102 Pa within about 24 h
- Subsystem flooding with helium and pressurization to approx. 4.5 MPa at 25°C
- Heating to approximately 380°C within a few hours by a combination of the electrical heater and circulator at full or reduced speed (see Section 5.3.2.3) with the HX bypassed
- Establishing secondary cooling water flow in the HX
- Establishing temperature control at desired baking temperature, (about 375°C at circulator outlet) by controlling the flow through HX with bypass heater power still on
- Keeping subsystem stable for baking period
- Driving circulator to nominal speed
- Establishing temperature and pressure control at stand-by level: 380°C, 8±0.3 MPa at circulator outlet, heater power off. Subsystem is then ready for operation.

During operation, the typical ITER load cycle is envisaged, i.e., pulse duration of 400 s and repetition time of 1800 s with specified power ramp-up and ramp-down. The total power removed by the FW and LB cooling loops thus varies between the maximum of about 1.4 MW (depending on the actual load conditions at the TBM plus circulator power) and the minimum of the order of 0.14 MW, the latter coming from circulator at reduced flow and from decay heat. This is a ratio of about 10:1.

Because of the given large mean temperature difference in the HX between the primary and secondary side of the FW loop, the heat removed in the HX can most effectively be influenced by primary helium flow control. Hence, the following preliminary subsystem control scheme is proposed for pulsed operation, based on an initial thermodynamic analysis with consideration of the thermal inertia of all components:

- The principal objective is to keep the test module FW inlet temperature at 380°C.
- The secondary cooling water inlet temperature is kept at 35°C.
- The circulator is operated at rated speed.
- The electrical heater is turned off.
- Flow partition through the HX and heater bypass is controlled as to maintain the inlet temperature to the TBM as close as possible to 380°C.

If for some reason much longer dwell times or shutdown periods have to be bridged, decay heat removal at reduced circulator speed, or even by natural convection, is envisaged.

5.3.1.7. Materials. All of the piping and components in the primary cooling subsystem have been designed. Austenitic stainless steel (AISI 316L) in contact with a liquid breeder cannot be used for temperatures ≳420°C to 440°C due to corrosion. This means that for concentric pipes arrangement, we can only use AISI 316L for the outer tube which is the only one to be cut/welded during the exchange of a TBM. This would meet the ITER requirement for remote handling. The inner tube will be designed with ferritic steel sliding seals and must be made of feritic steel, which is designed for operation up to ~550°C, allowing a maximum liquid breeder exit temperature of 650°C. This interface temperature between ferritic steel and liquid breeder will have to be verified. As noted above, to reduce tritium permeation to the water system, Al tubes are proposed for the FW loop helium/water heat exchanger. All of the piping and components will be equipped with 10 to 15 cm of a mineral thermal insulation.

5.3.1.8. Safety. The main safety concerns with the cooling subsystem are the loss of coolant accident with regard to release of tritium and/or activation products, and the loss of flow or loss of heat sink accident with respect to decay heat removal. Various scenarios are to be analyzed in the safety assessment. Details are given in Section 6. Among them are large in-vessel coolant leaks, large ex-vessel coolant leaks, and small and large in-TBM leaks. The issues addressed are VV pressurization, vault pressurization, purge gas system pressurization, temperature evolution in the TBM, decay heat removal capability, tritium and activation products release from the TBM system, hydrogen and heat production from Be/steam reaction, and Be/air reaction exothermic heat production. The expected results will show that all of the effects are inherently small (like pressurization, heat production, radioactive inventory) and will not add significant safety hazards to the basic ITER machine.

5.3.1.9. Interfaces to Other Equipment. Using the FW loop as an example, Table 5–8 summarizes the interfaces of helium cooling subsystem and gives estimates on space requirements for installation and operation of the FW loop subsystem, based on the assumptions and layout described in the foregoing. Also estimates on heat losses from the cooling circuit are added (Table 5–9).

Table 5–8
Space Requirements and Supplies to HCPB Helium Cooling Subsystem

Space Requirements

- Foot total print area in TCWS for main components, ancillary equipment and all tanks is about 20 m^2, 5 m high (Fig. 5–8 and Section 5.3.1.2)
- Space in transfer cask (parking cask) for measuring equipment rack, $0.5 \times 0.5 \times 2$ m^3 (Section 5.3.1)
- Space in main control room for subsystem control panel and operator, 12 m^2 (Section 5.3.1)
- Space for piping from TBM to main components, about 195 m, up to 0.4 m diam with thermal insulation, routing according to ITER drawings of TCWS vault (schematic shown in Figs. 5–3 and 5–4).

Supplies and Services

- Infrequent evacuation (to ~l00 Pa) of the main loop after maintenance, repair (Sections 5.3.1.6).
- Secondary cooling water (up to 17 kg/s, 1 MPa, 35°C/43°C) to heat exchanger in TCWS, about 63 mm diam piping.
- Electrical power for circulator up to about 300 kW (Section 5.3.2.2) and heater up to 170 kW (Section 5.3.2.3).
- Helium supply for main loop (approx. 2.52 m^3/ 18.26 kg at operating conditions)
- TBM installation, connection/disconnection, and transfer to the hot cells by the transfer cask, i.e., cutting and welding of two main pipes 89 mm o.d. × 6.0 mm wall, two purge gas pipes of approximately 25 × 2 mm
- Instrumentation and electrical cabling

The heat lost through the thermal insulation from the cooling circuit (piping and components, but excluding the TBM proper) has been calculated. In the calculations, air at 30°C is assumed as ambient atmosphere, stainless steel as structural material, and mineral wool as insulating material. Insulation thickness is assumed to be 15 cm for all the piping and components. Table 5–9 shows main results for the operating conditions specified in Table 5–5.

Table 5–9
Heat Losses from the First Wall Cooling Subsystem for Different Operational Conditions

	Nominal
Helium temp. inlet/outlet (°C)	380/440
Circuit heat losses other than the circulator motor (kW)	15.29
Peak insulation surface temp. (°C)	68

The heat losses amount to 15.29 kW. The insulation surface temperature runs up to 68°C. About 99.6% of the heat losses result from the piping, with more than 57.5% coming from the hot leg. In addition, there is a total of 51.4 kW lost from the circulator motor. The circulator motor is not insulated.

5.3.1.10. Tritum Extraction System. For the FW loop, in addition to the use of Al piping in the heat exchanger for the reduction of the tritium into the water coolant system, tritium extraction from the circulation helium is also envisioned. Description of this tritium extraction system is presented in Section 5.4.

5.3.2. Component Description

The helium cooling subsystem is described in Section 5.3.1. This section provides design descriptions of the subsystem components.

5.3.2.1. Heat Exchanger. A first layout has been performed for the FW loop HX assuming a conventional tube and shell U-tube heat exchanger configuration with high pressure helium flowing inside the tubes and low pressure water flowing outside. The tube bundle data are listed in the upper half of Table 5–10. The helium volume in the HX will be 0.0096 m^3 (0.0031 m^3 in the tubes and 0.0065 m^3 in the inlet and outlet plenums). The thermal-hydraulic data like flow velocity, pressure drops and heat transfer coefficients have been taken from detailed MATHCAD models.

Table 5–10
Heat Exchanger Layout Data

Parameter	Unit	Extreme
Type of HX		U-tube bundle
Number of HX		1
Tube size (outer/inner diam)	mm	13.7/12.7
“U” Tube length (incl. plate)	m	0.59
Number of “U” tubes per HX		41
Tube bundle diam × length	m	0.21 × 0.53
Overall HX diam × length	m	0.22 × 0.63
Heat to be removed per HX	MW	0.57
Primary coolant		Helium inside
Pressure	MPa	8
Temperature in/out	°C	380/440
Mass flow rate per HX	kg/s	1.82
Flow velocity (maximum)	m/s	65.1
Pressure drop (approx.)	MPa	0.044
Heat transfer coeff.	$W/(m^2K)$	3974
Secondary coolant		Water outside tubes
Pressure	MPa	1
Temperature in/out	°C	35/43
Mass flow rate per HX	kg/s	16.99
Flow velocity (maximum)	m/s	1.73
Heat transfer coeff.	$W/(m^2K)$	1.21×10^4

5.3.2.2. Circulator. Variable speed helium circulators will be installed in the cold leg of the helium loops. An encapsulated type circulator with vertical shaft is envisaged where the type of bearings (gas lubricated or magnetic) still has to be decided upon. As an example, the design specification for the FW coolant loop circulator is as follows:

- Design temperature 440°C (maximum temperature __TBD__ °C short term).
- Rated pressure 8 MPa (plus approximately 10% margin for overpressure control).
- Mass flow rate 1.82 kg/s at a pumping head of 0.65 MPa at rated speed and at 380°C at the outlet nozzle.
- Rated speed about TBD rpm, maximum speed TBD rpm, speed variation max/min of at least TBD.

Under these conditions, the input electrical power of the drive motor would be about 295 kW when assuming a combined circulator and motor efficiency of 0.68. The helium volume contained in the circulator is estimated as 0.3 m^3 and the overall dimensions of the circulator and drive unit are expected to be 1.54 m diam by 1.76 m in height.

5.3.2.3. Electrical Heater. The electrical heater is needed for baking the test module first wall at 375°C and for heating the whole cooling subsystem, including the test module, to operating temperatures after maintenance or repair periods. The heater will be positioned in a bypass to the HX, assuming that the HX is isolated during heating periods and the circulator is operating at rated or reduced speed. It has been estimated that an electrical power of 170 kW, together with a circulator power of 80 kW, would enable the whole TBM system (including the circuit components involved and heat losses, see Section 5.3.1.9) to be heated at a rate of about 200°C per hour in the case of ideal uniform heating. The main dimensions of the heater are 0.21 m diam by 1.25 m height, approximately 18% of which is occupied by the heating rods. This yields a helium volume of 0.031 m^3. The estimated pressure loss is small, ~500 Pa. The overall dimensions are assumed to be 0.35 m diam (at flanges) by 1.65 m height (including the end dome foreseen for electrical terminals).

5.3.2.4. Pipework. For the main pipework of the FW loop, i.e., hot leg, cold leg and elbows, an outer diameter of 114.3 mm and a wall thickness of 6.0 mm have been chosen. This results in flow velocities of between 37.6 and 41.1 m/s for the cold and hot extreme operating conditions, respectively. The pipe length is determined on the basis of the component arrangement in the TCWS vault shown schematically in Figs. 5–3 and 5–4, considering two U-bends for thermal expansion in the long horizontal pipe sections. Altogether, it yields a total length of the main pipework of 195 m (95 m for the cold leg and 100 m for the hot leg). The total number of elbows amounts to 44. Overall, the pipework contributes about 75% to the pressure losses in the loop.

The heater bypass line is supposed to be almost closed during burn times and open during dwell times (see Section 5.3.1.6), in which the flow rate could be reduced compared to the rated mass flow rate by slowing down the circulator. Also, the baking and heating procedure can be performed at a reduced flow rate. Thus, the bypass to the HX would allow smaller dimensions than the main pipework. Nevertheless, for simplicity, the same pipe size has been chosen for the FW loop design, i.e., outer/inner diameter of 114.3 mm. Its length is only about 4 m.

5.3.2.5. Valves. The number of valves in the FW loop has been minimized to avoid inadvertent closure, which would mean loss of heat sink, and to avoid additional pressure loss. Hence, only one valve is installed before the HX in the main loop and another one in the bypass line before the electrical heater. These two valves are needed for temperature control in normal cyclic operation and must be position-controlled. Valve size corresponds to pipe dimensions (see Section 5.3.2.4). A similar approach is required for the LB helium loop design.

5.3.2.6. Pressure Control Unit. This is a combination of equipment needed for evacuation of the cooling subsystem, helium supply, pressure control, and overpressure protection. The components are conventional and of relatively small size, except for the battery of tanks (see below). The pressure control unit is essentially isolated from the main cooling loop during normal operation. However, in case of a pressure drop caused by a small leak or by a loss of coolant accident, the buffer tank will discharge into the main loop.

The evacuation unit is needed for the first startup as well as after repair of the main cooling loop. It is assumed that a vacuum pipe line is provided for the ITER vacuum system. The pipe line has to be reliably isolated from the loop after evacuation to avoid inadvertent interconnection of the loop with other subsystems or pressurization of the vacuum system.

The helium supply and storage unit consists of a storage tank, buffer tank, compressor, and pressure regulators. Except for the fresh helium supply and the decommissioning of the used helium, which are supposed to be provided by ITER, the supply and storage unit is designed to be self-sufficient during the different TBM testing campaigns.

For the FW loop design, the storage tanks are therefore sized to handle the entire helium inventory of the loop with 10% margin (excluding the inventory in the buffer tank). The total storage tank inventory must be 2.269 m^3. This can be achieved with nine tanks of 0.4 m diam and 2.6 m long. With storage conditions of 14 MPa and 50°C, this results in a total mass storage of 47.3 kg. During normal operation, the helium inventory including 10% margin and one buffer tank, is 18.26 kg. A multi-stage compressor and cooler will be needed to load the storage tank for emptying the main loop. Similarly, the following parameters apply to the LB helium loop. The maximum helium inventory (based on the LB case) is 55.9 kg. Hence, a maximum mass of 61.5 kg must be stored at about 50°C, 14 MPa, resulting in a tank volume of 2.95 m^3. This can be achieved by, e.g., twelve tanks of 0.4 m diam, 2.6 m long.

Pressure control in the helium loop during nominal operation is achieved in the following way. The storage tank is kept at low pressure (~l.5 MPa) so that the main loop can discharge to the storage tank via the pressure regulator if the set point “pressure high” is reached. The buffer tank, on the other hand, has to compensate for the loop pressure if the set point “pressure low” is reached. As it discharges to the loop, a compressor from the storage tank will recharge it. A buffer tank volume of 10% of the FW loop volume is chosen, about 0.24 m^3, at a maximum operating pressure of 14 MPa. If one tank were to be used, its dimensions could be, e.g., 0.4/0.36 m outer/inner diameter by 2.6 m long for the FW loop design. Because the helium operating temperature is lower for the LB loops, one buffer tank will also be adequate for the secondary helium loop.

The overpressure protection of the cooling loop consists of two redundant safety valves or a combination of one safety valve plus a burst disc. The safety valves discharge into a group of dump tanks which are kept at controlled low pressure (near atmospheric) during normal operation. This avoids releasing contaminated helium into the building and completely depressurizing the main loop in case the valve would fail to close. The dump tanks are created in anticipation of the primary loop inadvertently pressurizing to the nominal pressure (8 MPa) at room temperature causing the whole subsystem to subsequently heat up to 380°C, the nominal operating inlet temperature. If, in this case, the pressure regulator should fail to open, the safety valve would respond and balance the pressure between the loop and the dump tanks. In order to limit the pressure to 8 MPa, a dump volume of ~50% of the loop volume (excluding the buffer tank volume), i.e., 1.03 m^3, would be required This can be achieved by, e.g., four tanks of 0.4 m diam by 2.6 m long. The lower helium temperature and the higher helium inventory in the secondary loop result in the need for five dump tanks for the secondary helium loop.

5.3.3. Procurement Packaging

To be determined.

5.4. TRITIUM EXTRACTION

5.4.1. Introduction

The blanket will breed tritium according to the reactions:

$$n^1 + Li^6 = T^3 + He^4 \quad ,$$

and

$$n^1 + Li^7 = T^3 + He^4 + n' \quad .$$

Thus, the reaction product will be equal amounts of tritium and helium. Tritium will be born in the liquid phase and will ultimately be recovered as a gas. The subject of this section is the recovery of this tritium.

5.4.2. Specifications

Specifications important for tritium recovery are summarized in Table 5–11.

The first tritium generation rate in Table 5–11 is the rate generated during a plasma pulse. When the plasma is off, of course, no tritium is generated. The operational scenario considered was: plasma on for 450 s, plasma off for 1800 s and overall 25% availability. This gives the time averaged tritium generation rate shown.

Table 5–11
Tritium Recovery Specifications

	He	$LiBeF_3$	FLiNaBe	Pb-17Li
T_{in}/T_{out} (°C)	380/440	420/520	360/520	340/440
Coolant pressure (MPa/atm)	8/7.9	1/9.9	1/.9.9	1/9.9
Mass flow rate (kg/s)	1.514	4.86	2.98	59.2
Density (kg/m^3)	5.33	2000	1900	9300
	@440°C	@540°C	@500°C	@400°C
Tritium generation rate during pulse (gm/s)	2.09×10^{-6}	2.09×10^{-6}	2.09×10^{-6}	2.09×10^{-6}
Average tritium generation rate (gm/s)	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}	1.05×10^{-7}
MW (gm/mole)		72.95	57.94	17315.58
Molar flowrate (mole/s)		66.62	51.43	3.419
Tritium molar flowrate (mole/s)		3.48×10^{-7}	3.48×10^{-7}	3.48×10^{-7}
Loop volume (m^3)		0.5	0.5	0.5
Heat exchanger area (m^2)				
Breeder to He		37		4.22
He to water	0.536	4.7		4.97
Heat exchanger tube thickness (mm)	0.5	0.5		0.5
Solubility (at. fr./Pa$^{0.5}$)				1.00×10^{-8}

5.4.3. Physical Properties

The permeability K_p of deuterium through austenitic stainless steel is given by Louthan and Derrick [5–5] as:

$$K_p = 6 \times 10^{-3} \exp(-14{,}300/RT) \text{ cc (STP)/(cm-atm}^{0.5}\text{-s)} \quad .$$

[Note: R = 1.987 cal/(K-mole)]

For the purposes of this analysis, the tritium permeation is taken to be the same as the deuterium permeation rate.

The rate of permeation of tritium through a metal such as stainless steel is given by:

$$F = \frac{A\,K_p}{l}\left(\sqrt{p_a} - \sqrt{p_b}\right) \quad .$$

Where F is flowrate, A is permeation area, l is the permeation length, p_a is the tritium partial pressure on the "upstream" side of the metal, and p_b is the tritium partial pressure on the "downstream" side of the metal.

The solubility of tritium in Pb-17Li is given by Caorlin and Gervasini [5–6] as approximately 1×10^{-8} atom fraction/Pa$^{0.5}$. This number can be considered to be independent of temperature. This can be converted to:

$$s = 5.5 \times 10^{\pm 8} \frac{\text{kgT}}{\text{kg Pb-17Li}\sqrt{\text{atm}}} \quad .$$

The equilibrium partial pressure of tritium over the Pb-17Li, p*, is related to the mass fraction of tritium in the Pb-17Li, x, by Sievert's law:

$$s\sqrt{p^*} = x .$$

5.4.4. Model

5.4.4.1. Model Description. A flow diagram for the tritium generation and extraction system is given in Fig. 5–9:

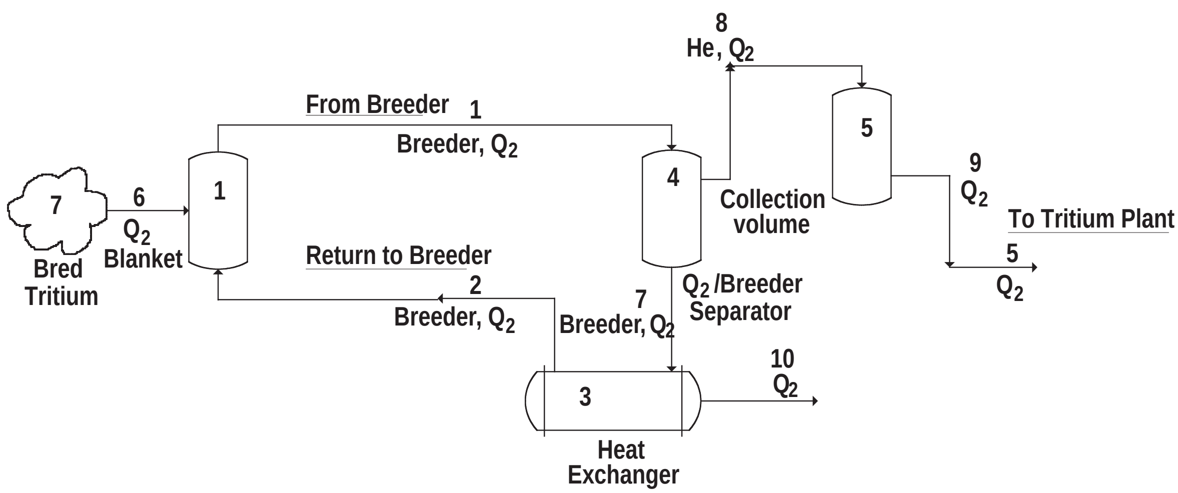

Fig. 5–9. Tritium extraction loop from MS.

Vessel 1 is the blanket. Breeder material leaves the blanket in stream 1 and returns in stream 2. Tritium is bred in the blanket and this tritium is shown entering the blanket as stream 6. The change in mass fraction of tritium and Pb-17Li in the blanket over time is given by:

$$\frac{dx}{dt} = \frac{F_2 x_3 - F_1 x_1 + F_6 x_7}{m_1} .$$

Where:

- x_i is the mass fraction of a given component (tritium or Pb-17Li) in vessel i
- F_j is the total mass flowrate of stream j
- m_i is the total mass of material in vessel i

- t is time

Vessel 4 is a Q_2/breeder separator. This is a tank which allows tritium and helium to accumulate in the head space. The mass fraction of tritium in the Pb-17Li is given by:

$$\frac{d x_4}{d t} = \frac{F_1 x_1 - F_7 x_4 - F_8}{m_4} .$$

Where F_8 if the flowrate of tritium from the vessel 4 headspace to a collection volume (vessel 5). The mole fraction of Pb-17Li in vessel 4 is given by the same equation, but F_8 is zero.

The rate of transfer of material from vessel 4 to vessel 5 is presumed to be described by a mass transfer coefficient according to:

$$F_8 = k_{m1} \left(p_2^* - p_5 \right) .$$

And the rate of removal of material from vessel 5 to the tritium plant is also assumed to follow a mass transfer-like equation as:

$$F_9 = k_{m2} \left(p_5 - p_{vacuum} \right) .$$

The amount of material in vessel 5, then, will follow:

$$\frac{d m_5}{d t} = F_8 - F_9 .$$

Vessel 3 is the heat exchanger. The mass fraction of tritium in this vessel's Pb-17Li is described by:

$$\frac{d x_3}{d t} = \frac{F_7 x_4 - F_2 x_3 - F_{10}}{m_3} .$$

Where F_{10} is the rate of tritium permeation through the stainless steel heat exchanger tubes. The mass fraction of Pb-17Li in vessel 3 is also given by this equation, but in this case F_{10} is zero.

5.4.4.2. Model Input. Besides the values given in Table 5–11, the following values were also set:

- m1 = 2000 kg (for all t)
- m4 = 1000 kg (for all t)
- m3 = 1650 kg (for all t)
- m5 = 0 kg (at t=0)
- V5 = 100 liters (volume of vessel 5)

Initially vessels 1, 4 and 3 are filled with tritium-free Pb-17Li.

The model was run to time = 100 days.

5.4.4.3. Model Results

No Extraction/Permeation. It is instructive to run the model with no removal of tritium from the Pb-17Li in vessel 4 (i.e., the mass transfer coefficient is set to zero) and no permeation occurs in the heat exchanger. While not practical, this is an instructive case to explore operational limits. The tritium partial pressure which develops is shown on Fig. 5–10.

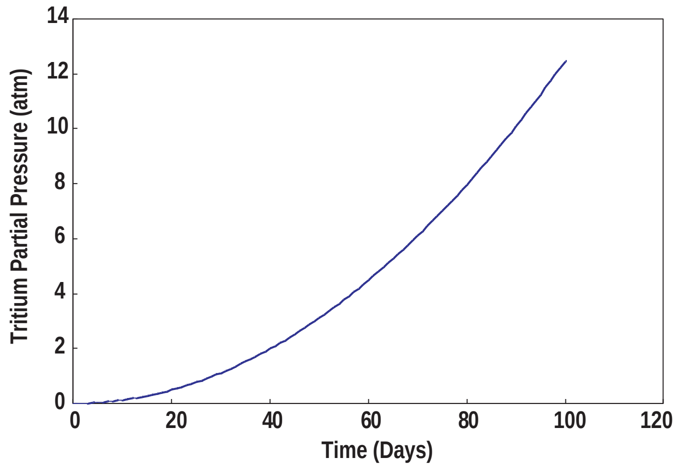

Fig. 5–10. No extraction/permeation.

While the tritium generation rate is linear, the pressure builds up according to t^2 due to the square root in the solubility equation. For this case, at t = 100 days, the tritium partial pressure is 12.5 atm, the mass fraction of tritium in the Pb-17Li is 1.9×10^{-7} and the total tritium in the Pb-17Li is 0.9 gm.

While the amount of tritium is rather small, this generates a substantial partial pressure due to its low solubility.

Tritium Extraction by Collection in Tank. For the next case, a finite mass transfer coefficient is used for transfer of tritium from vessel 4 to vessel 5. However, the coefficient for transfer out of vessel 5 remains at zero. Thus, all tritium that can be removed by mass transfer is collected in vessel 5. At present, there has not been sufficient time to research and estimate the mass transfer coefficient. Rather, what is believed to be an optimistic value for the coefficient was selected as 1×10^{-7} kg/s/atm. Using this value, the equilibrium partial pressure of tritium and the partial pressure of tritium in vessel 5 develop as shown in Fig. 5–11.

As shown, the partial pressure of tritium in vessel 5 (100 l) and the equilibrium partial pressure of tritium exerted by the Pb-17Li are nearly identical. If a high mass transfer coefficient were selected, these two values would be identical. If a lower value were selected, then a significant mass transfer resistance would be observed. This would seem to be a reasonable estimate.

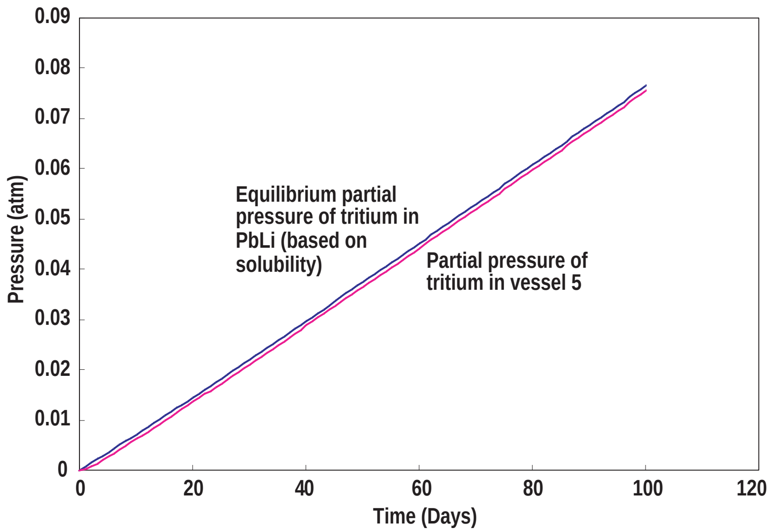

Fig. 5–11. Tritium extraction by collection in tank.

For this case, after 100 days, the tritium mass fraction in the Pb-17Li rose to 1.5×10^{-8}. This results in 0.71 gm tritium in the Pb-17Li, and 0.83 gm tritium in the amount in the tank. The total is the same as in the previous case since no tritium has been sent to the tritium plant. Compared to the previous case, the reduced amount of tritium in the Pb-17Li results in a much reduced pressure exerted by the tritium in the Pb-17Li (i.e., 0.076 versus 12.5 atm). It is observed that providing an expansion volume (100 l) greatly reduces the pressure exerted by the tritium in the Pb-17Li.

Tritium Extraction by Collection in Tank with Removal to Tritium Plant. For the next case, the second “mass transfer coefficient” (k_{m2}) is given a finite value which is a factor of 10 larger than the other mass transfer coefficient. This is justified since there should be less resistance to moving tritium through a pipe compared with removing tritium gas from solution. A vacuum pressure of 10^{-5} torr is maintained on the outlet of vessel 5. The resulting pressure history is shown in Fig. 5–12.

As shown, the pressures increase for the first three days, but then reaches a steady state value as tritium is being removed to the tritium plant as fast as it is being generated. The partial pressure of tritium from the Pb-17Li is 0.0011 atm (down from 0.076 atm in the previous case). The mass fraction of tritium in the Pb-17Li is 1.9×10^{-9}. The amount of tritium in the Pb-17Li is 0.0087 gm and in the tank is 0.0012 gm. Thus, the total tritium in the blanket module system is 0.0099. Most of the tritium has been sent to the tritium plant.

Tritium Extraction by Collection in Tank with Removal to Tritium Plant and by Permeation through Heat Exchanger Tubes. Now to bring all phenomena into the model, permeation through the heat exchanger tubes is allowed. All the values used for the previous case are also maintained.

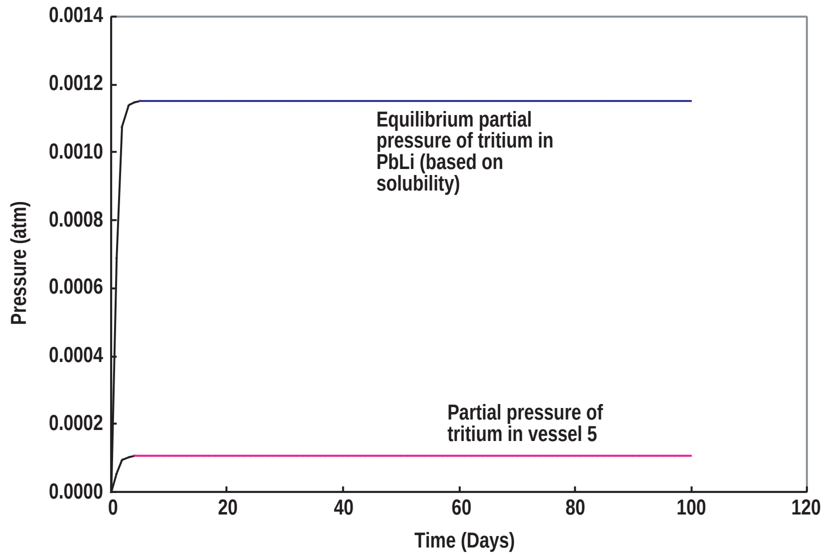

Fig. 5–12. Tritium extraction by collection in tank with removal to tritium plant.

The permeability for tritium through stainless steel was developed using gas with metal. For the test module, we are considering tritium migration from the liquid Pb-17Li directly to the metal. It is unclear what effect this will have on permeation. For the gas-to-solid case, hydrogen isotopes must dissociate to enter the metal, while in the Pb-17Li, the hydrogen isotopes are already dissociated. This would cause the permeation for Pb-17Li to be enhanced. However, gas phase diffusion is much faster than liquid phase diffusion, so this would tend to retard permeation for Pb-17Li. For this study, it was assumed that the permeation for Pb-17Li was the same as the gas-solid case. The “upstream” tritium partial pressure was assumed to be the equilibrium partial pressure of tritium for Pb-17Li (i.e., the equation given above). For the “downstream” pressure, it was assumed that the tritium partial pressure in the helium coolant was 10^{-4} torr. The resulting pressures are given in Fig. 5–13.

Qualitatively, the pressure history is similar to the previous case, but the pressures are much lower.

For this present case, tritium is removed from the test blanket module system both by vacuum pumping out of the collection tank and by permeation. Figure 5–14 was prepared to compare these two pathways:

As shown, much more tritium leaves the system through heat exchanger tube permeation than by mass transfer from the collection tank.

The mass fraction of tritium in the Pb-17Li is 2.9×10^{-10}. The amount of tritium in the test blanket module system is 0.0014 gm in the liquid and 0.000030 gm in the tank for a total of 0.0014 gm.

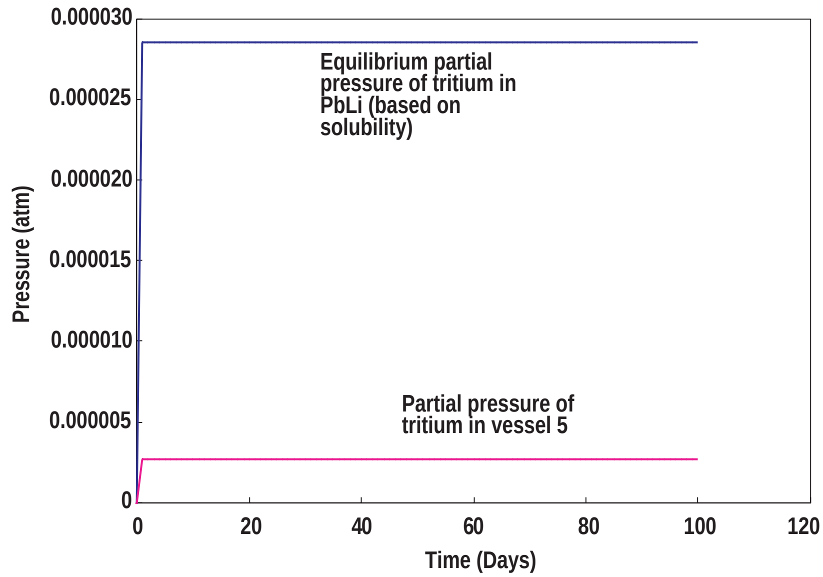

Fig. 5–13. Tritium extraction with all three phenomena.

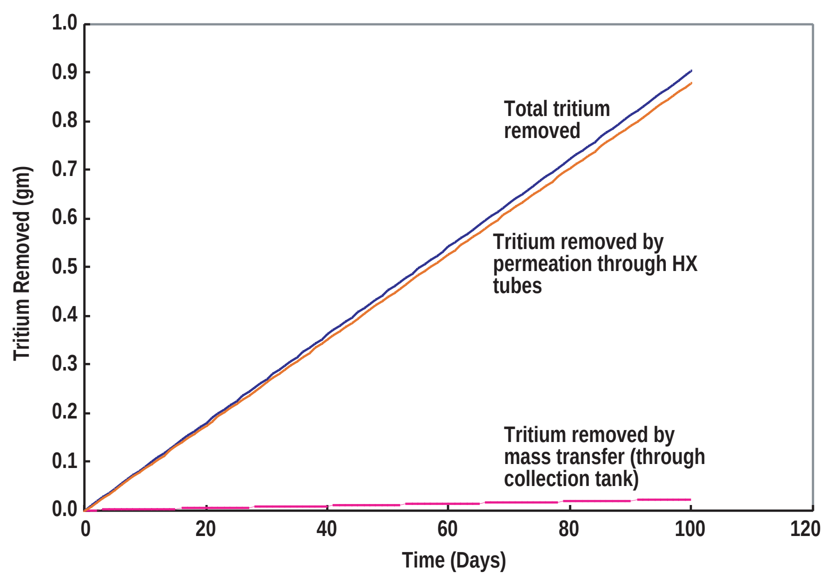

Fig. 5–14. Comparison of results.

Using mass transfer coefficients which are 100x's those listed above, the model shows that most of the tritium leaves the system via the collection tank, but still a substantial amount (36%) leaves via heat exchanger permeation.

5.4.5. Bubbles Nucleation/Outgassing

There has not been sufficient time to determine whether or not tritium (and helium) degassing will be by mass transfer/permeation alone, or whether bubbles will nucleate and rise to the surface. For bubbles to form, the amount of dissolved gas must rise to a threshold level.

From the low mass fractions listed here, it is likely that bubbles will not nucleate, but further study is needed to determine this.

Whether or not tritium/helium bubbles nucleate, these gases will have a tendency to evolve from any liquid surface. This gas will migrate to the highest point in the system. Thus, the entire liquid Pb-17Li loop should be constructed in a manner so that gas will collect at planned for points. Pressure vents should be provided at these points.

5.4.6. Active Versus Passive Tritium Removal

The analysis performed here did not specify whether tritium was actively or passively removed from the extraction tank (i.e., an arbitrary mass transfer coefficient was used).

The active way of removing tritium from the tank would be sparge helium into the bottom of the tank and, as these bubble rise through the tank, tritium will diffuse into the bubbles which will accumulate in the headspace. The passive method does not employ sparging; it just allows tritium to naturally accumulate in the headspace. The active method should have significantly higher mass transfer, but it is presently unclear how effective this will be and whether or not it will be desirable.

It is expected that this will be part of the practical testing of the tritium extraction system. It is proposed that helium sparging be included as a system capability so that operations can be performed with and without helium bubbling.

5.4.7. Other Products

This analysis has not considered other gases which might evolve from the breeder due to transmutation and radiolysis.

5.4.8. Recovery of Tritium from Helium Coolant

Much and very possibly most of the bred tritium will accumulate in the helium coolant. This tritium can be recovered by a “low-pressure permeator” as described by Willms, et al. [5–7] and Willms [5–8]. Such a system will recover tritium in a usable form (i.e., T_2) and will not require the addition of oxygen to the helium coolant loop. The system places Pd/Ag alloy tubes in the helium flow stream. Tritium permeates through the tubes until the tritium partial pressure is the same on both sides of the tubes.

Alternatively, an oxidation/adsorption system could be used, but in this case the tritium will be converted to the T_2O form.

In the safety section of this report, without choosing the technology, this separation was assumed to be performed on a 10% side-stream with a 30% efficiency. As proposed in this, the “Tritium Extraction” section, it is being proposed that a low pressure permeator will be incorporated into the full helium flow loop. However, the effect will be similar to that described in the “Safety” section, so the analysis performed there is valid.

5.4.9. Equipment Size

5.4.9.1. Tritium Extraction from Pb-17Li. Exact sizing of this equipment is not possible at this point, but it is expected that a tank which holds 100 l of active volume should be sufficient (50 l of headspace and 50 l for liquid). Insulation and the like would add another 50% to the volume, so the tank would occupy 150 l of space. The tank would likely be mounted vertically. This would be the preferred configuration for bubble column operation. If, however, no bubbles were used, horizontal mounting would be preferred.

5.4.9.2. Tritium Extraction from Helium Coolant. A low-pressure permeator system consists of a permeator and a high vacuum pumping system. The latter includes a turbo pump and a backing pump. The permeator will be approximately 2 m long × 1 m in diameter (cylinder). The pumping package will occupy a space of about 2 × 1 m × 1 m (rectangle).

5.4.9.3. Tritium Extraction from First Wall Helium. This can be performed with a system identical to the Helium Coolant system (immediately above).

5.4.10. Molten Salt

It is assumed that the same equipment will be needed for the molten salt coolant. The tritium solubility in the molten salt is lower than for Pb-17Li, so the driving force for mass transfer to the gas phase or through the heat exchanger tubes will be higher. The basic tritium pathways, however, will be quite similar, and the same equipment will be appropriate. Further refinement can be performed later.

5.4.11. Higher Temperatures

There may be reasons to operate blankets at temperatures higher than those used for this analysis. In this case, the tritium permeability in the heat exchanger will be increased while the solubility in Pb-17Li will not change appreciably. This will result in a larger fraction of tritium permeating through the heat exchanger relative to the amount of tritium removed in the tritium extraction tank. Nonetheless, the same basic equipment and pathways will be needed.

5.4.12. Conclusions

1. Both tritium and helium will be produced in the breeder.
2. Tritium is only sparingly soluble in Pb-17Li. Thus, it will have a tendency to leave the Pb-17Li by mass transfer.

3. Tritium is also substantial permeation through stainless steel at the blanket temperatures. Approximately half or more of the tritium will permeate through the heat exchanger tubes into the helium coolant.

4. Helium that is bred will not permeate through the heat exchanger. Much of it should accumulate in the collection tank together with some of the bred tritium.

5. A collection tank is needed to accumulate tritium and helium which will be evolved by mass transfer and possibly by bubble accumulation.

6. In this analysis, a 100 l (of gas) collection tank was used. This is probably larger than needed, but would be a good size if space is available.

7. It would be valuable to include the capability to bubble helium through the Pb-17Li in the collection tank to enhance the transfer of tritium from the liquid.

8. Any high point in the Pb-17Li circulation loop should include a gas vent line.

9. A system will be needed to recover tritium from the helium coolant system. This can be a low-pressure permeator or an oxidation/adsorption system.

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6. SAFETY

The US Home Team is developing an ITER TBM based on a dual-cooled (DC) DEMO blanket concept. This blanket concept employs high-pressure helium to cool the blanket first wall (FW) and side wall structures (SW), and employs a liquid breeder to self-cool the interior of the blanket module, including a beryllium multiplier pebble bed. The structural material for the TBM is F82H low activation martensitic steel. There are three liquid breeder material options presently under consideration: $LiBeF_3$, FLiNaBe, and Pb-17Li.

As stated in Appendix A of the ITER-FEAT Generic Site Safety Report [6–1], the safety assessment to date of TBMs has addressed a number of concerns or issues that are directly caused by TBM system failures. The concerns stem from three basic accident event sequences, which are: (1) in-vessel TBM coolant leaks, (2) in-TBM breeder box coolant leaks, and (3) ex-vessel TBM ancillary coolant leaks. These events were selected to address, where applicable, the following ITER-FEAT reactor safety concerns: (1) VV pressurization, (2) vault pressure build-up, (3) purge gas system pressurization, (4) temperature evolution in the TBM, (5) decay heat removal capability, (6) tritium and activation products release from the TBM system, and (7) hydrogen and heat production from chemical reactions.

A detailed safety analysis of a DC TBM can be conducted when the design of this TBM and ancillary systems matures. However, the impact on ITER safety from this proposed TBM concept can be inferred from results already reported for other TBM concepts [6–1], and the DC DEMO blanket concept. The following sections present a general description of this TBM and associate ancillary equipment, safety related source terms of these components, and a preliminary assessment of the components during the event sequences mentioned above.

6.1. SYSTEM DESCRIPTION AND SOURCE TERMS INVOLVED

6.1.1. System Description

The DC TBM is approximately a box that is 55.4 cm in width, 176 cm in height, and 40 cm in depth, which is half of an ITER TBM port. The TBM walls (FW and SW) are cooled by 8 MPa helium, which enters the module at 380°C and exits the module at 440°C. This helium is delivered to the TBM by concentric pipes with the outlet stream contained by the inner pipe. Once outside of the ITER VV, this concentric pipe runs approximately 100 m to a heat exchanger located in the TCWS vault. Here, the helium rejects the TBM wall heating to water through an aluminum tube heat exchanger. The primary safety reason for adopting concentric pipes and an aluminum tube heat exchanger is to reduce tritium permeation into the ITER confinement building and into the ancillary cooling water. This particular cooling system will contain approximately 18 kg of helium (see Section 5.3.2.6).

The self-cooled breeder zone of the TBM is cooled by a liquid breeder material, which enters the blanket at a temperature between 340°C and 420°C and exits the TBM at a temperature between 440°C to 520°C, depending on the selected breeder material. Compared to the helium cooling system, this system is a relatively low-pressure cooling system, and the adopted breeding materials contained within this loop are very low-vapor-pressure fluids.

The pipes that supply the breeding material to the TBM are also concentric pipes that run from the TBM into the TBM cell, where it enters the shell side of a helium-cooled heat exchanger. The volume of breeder material in the TBM will be limited for safety reasons to 0.28 m^3 [6–1]. The total volume for the TBM plus ancillary equipment is presently estimated to be 0.5 m^3. There will be a beryllium multiplier pebble bed in the TBM just behind the TBM FW for the molten salt (MS) breeder options. This zone will contain ~54 kg of beryllium.

The helium in the tube side of the breeder heat exchanger is supplied by concentric pipes that run ~90 m from the breeder heat exchanger to a water-cooled helium heat exchanger that resides in the TCWS vault. This 8 MPa helium enters the breeder heat exchanger at 200°C and exits the heat exchanger at 360°C. The total helium inventory for this intermediate helium loop is ~17 kg.

A heated drain tank will be part of the ancillary equipment to contain the liquid breeding material prior to charging of the breeder-cooling loop and to keep the breeder from freezing during reactor down times. This tank will be pressurized to charge the loop prior to operation. Once the loop has been charged, the tank will be isolated from the loop by valves and the tank pressure reduced. There will be two safety components associated with this drain tank. The first is a rupture disk that will open a line into the drain tank at a set pressure below the maximum allowable design pressure for the TBM breeder zone. The second will be an isolation valve located downstream from the loop pump that will open a line into the drain tank based on a low-pressure signal at the pump inlet.

6.1.2. Tritium Inventory

Tritium will be bred in this DC TBM in the liquid breeding material and in the case of the MS breeder options in the beryllium multiplier. The tritium production rate for the liquid breeders is estimated to be 2.103×10^{-6} gm/s. The quantity of tritium bred in the beryllium, based on 30,000 full power ITER-FEAT pulses and production rates calculated by a reactor blanket concept described in Ref. 6–2, is preliminarily estimated to be 30 mg. This tritium will be trapped in the beryllium, and at the anticipated neutron fluence for this TBM will likely not be released unless the beryllium temperatures exceed about 800°C [6–3].

In order to estimate the tritium inventory and permeation rates for the DC TBM cooling systems based on the production rate of 2.103×10^{-6} gm/s, a TMAP code [6–4] model of the entire TBM system was developed. This model includes the structural material associated with the most of the major TBM and ancillary loop components (e.g., pipes, heat exchangers, and tritium extract systems). Material properties of tritium diffusion coefficients and solubilities used in this model are those adopted by Ref. 6–2 for the structural materials and MS of these components. The same parameters for the Pb-17Li breeder were taken from Ref. 6–5. Bulk transport of tritium within the liquid breeder material was also included as described in Ref. 6–2. Only tritium extraction from the helium cooling loops was considered in this model, with a slip steam of 10% of the helium flow taken per loop and the tritium extracted from this slip stream with an efficiency of 30%. This would be a conservative estimate when additional tritium is extracted form the liquid breeder.

Based on the assumption of 3000 consecutive ITER-FEAT pulses per year, the tritium inventories in DC TBM and TBM ancillary equipment reach equilibrium after 40 pulses. The structural material of this system is predicted to contain ~8 mg of tritium, and the breeder material contains ~5 mg in the case of Pb-17Li and ~0.3 mg in the case of MS. When this tritium is combined with the tritium in the multiplier beryllium the total is less that 40 mg. This inventory is considerable less than the tritium inventory contained within the VV due to normal operation of ITER.

Tritium permeation from the helium cooling system pipes is predicted to be ~45 mg after 3000 pulses, and since this represents one year of operation, the tritium release into the TCWS vault would be 45 mg/a. This is 4.5% of the release guidelines to the environment for ITER tritium plant. Because of the higher temperature of the breeder cooling loops, the permeation is estimated to be much larger for the breeder cooling system at about 555 mg/a. An Alumina coating is being considered for all ancillary system pipes to reduce tritium permeation by more than a factor of 1000. Because the helium to water heat exchanger tubes are made of aluminum, the predicted permeation rate into the cooling water is only 0.05 mg/a of tritium or 0.35 mg/a of HTO. This permeation rate is very small and is less than 0.7% of the anticipated ITER HTO tritium plant environmental release per year.

6.1.3. Breeder Material Radioactive Inventory

Based on activation calculations performed by Ref. 6–2, the long-term Flinabe, which is one form of molten salt being considered, activity is dominated by Na-22, Na-24, and F-18 at intermediate times. For Flibe, the dominant isotope is F-18. The specific dose for the FLiNaBe, based on radionuclide inventories scaled to ITER according to FW neutron flux, average weather conditions, and a stacked release, is 0.013 mSv/kg,[1] with 91% of the dose from F-18, 7% of the dose from Na-22, and 2% of the dose from Na-24. The specific dose for Flibe is 0.007 mSv/kg, with 99% of the dose due to F-18. If the entire TBM Flinabe or Flibe inventory (~1170 kg) were released to the environment by way of the stack, the dose at the site boundary would not exceed the ITER site boundary limit value of 25 mSv.

[1]Average weather conditions are Pasquill-Gifford stability class D atmospheric conditions for a wind velocity of 4 m/s.

Based on activation calculations performed for the self-cooled Pb-17Li power reactor blanket concept, the activity is primarily due to Pb-203 and Pb-209 [6–6]. The specific dose for Pb-17Li will likely be dominated by Po-210 and Hg-203 because of the volatility of these isotopes. The estimated Po-210 production rate will be 6.4×10^{-10} Ci/kg-s based on an extrapolation from that power reactor blanket with a neutron wall loading of 10 MW/m^2 to the DC TBM with a neutron wall loading of 0.78 MW/m^2. At this production rate, the total inventory for 0.5 m^2 of Pb-17Li in the DC TBM system after operating for 30,000 pulses of 400 s is 37 Ci, which should be a conservative estimate because the decay of Po-210 has not been included in this estimate. If this entire inventory of Po-210 were to be released to the environment as a ground level release during average weather conditions, the dose at the site boundary would be 14 mSv. A similar calculation for Hg-203 results in a dose of 0.05 mSv. As was the case for a MS DC TBM, the anticipated dose associated with Pb-17Li even under very conservative assumptions will be less than the ITER site limit value of 25 mSv.

In addition, the above dose estimates will be conservative because these breeder materials are low vapor pressure fluids that will quickly solidify if spilled making a complete evaporation of the breeder volume highly unlikely, and because the specific dose estimates used are based on steady-state TBM operation. Therefore, the real difference between anticipated release and allowable release is much larger than indicated.

6.2. DECAY HEAT REMOVAL

The decay heat removal capability must be demonstrated for the DC TBM in order to meet ITER safety requirements. A CHEMCON [6–7] 1D heat transfer model has been developed to analyze the anticipated temperature response of this module during a loss-of-flow event under decay heating. Because DC TBM design details are not available, the DC DEMO blanket design was modified to give a breeder zone depth that is consistent with the required in-vessel Pb-17Li breeder volume limit of 0.28 m^3. A 20.5 cm steel shield was assumed to reside behind the TBM, with decay heating of this shield included to introduce additional conservatism. The radial build for this CHEMCON model appears in Table 6–1. The low-temperature Flibe option was selected for this study because the low thermal conductivity of this salt represents the largest resistance to radial heat flow. The decay heat input for this model was scaled from DC DEMO blanket activation calculations by the ratio of the neutron flux for ITER TBMs (0.78 MW/m^2) to that for the DC DEMO [6–8] (3.0 MW/m^2). The resulting decay heat at shutdown for the TBM is estimated to be 23.2 kW. This decay heat decreases to 0.5 kW after one day. Only radiation heat transfer is assumed between the back of the TBM and the shield and between the shield and the ITER VV. For this study, the VV temperature was held at 120°C.

Table 6–1
Radial Build of CHEMCON Thermal Model of DC TBM

TBM Region	Thickness (cm)	Steel Fraction	Beryllium Fraction	Flibe Fraction	Helium Fraction
First wall	0.3	1			
Cooling channel	2.2	0.13			0.87
Second wall	0.3	1			
Flibe front channel	2	0.03		0.93	0.04
Multiplier front wall	0.3	0.88		0.1	0.02
Be pebble bed	5	0.03	0.6	0.35	0.02
Multiplier back wall	0.3	0.88		0.1	0.02
Flibe back channel	2	0.03		0.93	0.04
Breeder zone front wall	3.8	0.32			0.68
Breeder zone	28.8	0.03		0.93	0.04
Module back wall	3.4	0.66			0.34
Shield	20.5	0.7			

Figure 6–1 shows TBM FW temperature evolution for this decay heat removal analysis for three parametric cases. Case 1 is the base case model as described above. Case 2 assumes that the shield is an integral part of TBM; that is, no gap exists between the TBM and the shield. Case 3 allows for natural convection of the salt within the breeding zone to enhance the radial heat transport through this zone of the TBM. In all cases, the FW starts at 530°C (the initial conditions used here are those from the DEMO analysis) and rapidly transfers the heat towards the cooler back of the FW in the first few seconds. Later on, the FW temperature reaches a peak at 555°C for Case 1 after about 5 h and then decreases due to heat transfer to the cooler structures. This peak temperature drops to 550°C and 530°C for Cases 2 and 3, demonstrating the impact of design decisions and modeling details on predicted TBM decay heat removal. For comparison, the predicted temperature history for the ITER FW due to decay heating has been included in Fig. 6–1. The ITER curve heats to a peak of ~465°C and then drops as the decay heat is removed by the ITER VV by natural convection.

These are positive results for the DC TBM in the sense that it shows that the TBM temperature will remain very close to the operational values during the first few hours and will passively decrease at later times. It is also worth mentioning that these results are conservative, because approximate decay heating rates based on a steady state DEMO design were extrapolated to the DC TBM, and that under more realistic decay heat predictions the peak temperature is expected to be closer to that predicted for the ITER FW during this event.

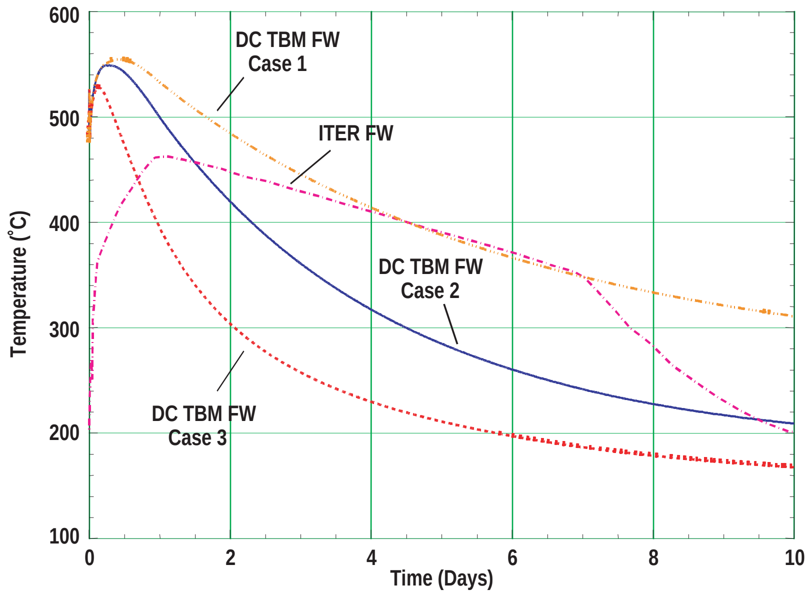

Fig. 6–1. DC TBM FW temperature evolution due to decay heat during loss-of-flow event.

6.3. IN-VESSEL LOSS OF COOLANT

Detailed calculations for this event can be performed when the designs of the DC TBM and the TBM ancillary system become more mature. However, the impact on ITER safety from in-vessel coolant leaks from this proposed TBM concept can be inferred from results already reported for other TBM concepts [6–1]. There are two independent coolant loops for the DC TBM concept that must be considered for this event. The FW cooling system contains high-pressure helium, and the ancillary equipment for this cooling system is similar to that for the EU helium cooled pebble bed (HCPB) TBM. The leak of helium into the plasma chamber is assumed to immediately induce a plasma disruption. This disruption not only fractures the ITER FW, releasing water into the ITER VV, but forces created by the disruption on the DC TBM fractures the breeding region of the TBM, releasing breeder material into the VV. Pressurization of the VV will occur, primarily driven by the ITER FW/shield cooling water, until the VV pressure suppression system opens. Because the amount of helium in the DC TBM FW cooling system is less than 40 kg, the VV pressure suppression system should retain its suppression function and return the VV to sub-atmospheric pressures. The release of helium from the TBM should only raise the VV pressure by ~8.7 kPa according to EU HCPB predictions.

For the Pb-17Li breeder concept, there is a safety issue surrounding the leakage of the Pb-17Li breeder into the ITER VV from the DC TBM, which is the chemical reaction of this material with water to produce hydrogen. The stoichiometric equations for this chemical reaction are the following:

$$\mathrm{Li} + \mathrm{H_2O} \Rightarrow \mathrm{LiOH} + \tfrac{1}{2}\mathrm{H_2} + 204\ \mathrm{kJ}; \quad T < 450^{\circ}\mathrm{C} \quad ,$$

$$2\,\mathrm{Li} + \mathrm{H_2O} \Rightarrow \mathrm{Li_2O} + \mathrm{H_2} + 186\ \mathrm{kJ}; \quad T > 450^{\circ}\mathrm{C} \quad .$$

Experiments [6–9] have shown that the fraction of lithium reacted in the Pb-17Li liquid metal (LM) strongly depends on the temperature of the LM and the contact mode between the LM and the water. In literature, these contact modes have been defined as:

- Injection — pressurized injection of water into liquid metal (LM).
- Pouring — pouring of LM into water.
- Layered — pouring of water onto LM.
- Pool — steam environment over LM pool.
- Spray — steam environment present during LM spray.

For the layered, pool, and spray contact modes the above reactions have been found to be self-limiting by the formation of a solid LiOH or Li_2O layer that shields the LM from water or steam at the LM/water interface [6–9]. Because the accident under consideration does not inject water into the LM, the only remaining contact mode of concern is that of pouring.

The quantity of hydrogen generated depends on the mass of LM poured into the water pool that forms in the VV as a result of the ITER FW break and reaction rates associated with the contact mode of pouring. The reaction rates are faster for LM at temperatures above the melting temperature of LiOH; that is, for conditions when Li_2O formation dominates the reaction. Reference 6–10 presents data for pouring Pb-17Li into excess water, which is the anticipated condition for this event. For this test, 20 g of LM at 600°C was poured into 4000 g of water at 95°C. The quantity of hydrogen generated was measured to be 2.5×10^{-4} mol-H_2/g LM. The volume of Pb-17Li in the DC TBM and ancillary system must be limited to 0.5 m^3. Given this reaction measurement, the mass of hydrogen generated from 0.5 m^3 of Pb-17Li is 2.4 kg, which is just below the VV limit of 2.5 kg. This hydrogen generation estimate should be conservative for two reasons. First, the volume of Pb-17Li spilt into the VV will be limited by an isolation valve in the ancillary system downstream of the LM pump that will actuate during this event from a low pump inlet pressure signal, pumping LM into the drain tank. Second, the adopted reaction rate will be in excess of that expected for the LM during this accident because the LM temperature in the TBM will be less than the tested LM temperature of 600°C.

Hydrogen generation regarding the molten salts being considered for the DC TBM should not be a safety concern. Hydrogen generation will be limited by the amount of free fluorine in the salt, which was demonstrated in Section 5.2 to be insignificant. These salts should readily dissolve in the water. There is ~54 kg of beryllium in the multiplier pebble bed of the DC TBM that could pour into the VV water pool, producing hydrogen. The temperature of these 5 mm diam pebbles will not exceed 500°C during operation. At this temperature, the oxidation rate for beryllium in water based on the rate formula in Ref. 6–11 SADL is 5.8×10^{-9} kg/m^2-s. The total surface of these pebbles is 35.3 m^2, which means that at this oxidation rate it would take 1.7 years to generate 2.5 kg of hydrogen. In addition, the chemical heat liberated at this temperature is less than 1 W/m^2-s. Because this heating is four orders in magnitude less than the thermal radiation heat transfer between the beryllium pebbles and the ITER FW cooling water, a thermal excursion should not occur. These pebbles will quickly cool to the temperature of the water (125°C).

6.4. LOSS OF COOLANT INSIDE BREEDER ZONE

A TBM FW coolant leak into the breeder zone of the TBM would create a pressure rise in this region of the DC TBM. However, the ancillary system for the TBM has been designed such that a pressure rise in the breeder-cooling loop would burst a rupture disk that leads to the drain tank. The purpose of this safety measure is to drain the breeder from the cooling loop prior to a structural failure occurring in the breeder box of this TBM. Even if this safety system fails to drain the breeder from the breeder-cooling loop, the consequence of this event should not be much different than those discussed for in-vessel coolant leaks (Section 6.3).

6.5. EX-VESSEL LOSS OF COOLANT

A DC TBM has the safety advantage that if correctly designed, the loss of one of the cooling systems should not lead to damage of the TBM. This is particularly true for the breeder cooling loop, because an ex-vessel coolant leak would not only remove the cooling from the breeder zone of the TBM, but would also remove the heat source from this zone (e.g., nuclear heat in the breeding material). To determine if a loss-of-coolant event in the FW cooling loop could be tolerated without causing FW melting, the CHEMCON 1D heat conduction model described in Section 6.2 was used for this analysis. The nuclear heating for the module was scaled from the heating rates calculated for the DEMO blanket [6–12] based on neutron flux. A parametric study was performed with this model by applying surface heat fluxes of 0.1 MW/m^2, 0.25 MW/m^2, and 0.5 MW/m^2. The results of FW temperature with time for these three cases appear in Fig. 6–2. If we use a melting temperature of 1500°C for F82H steel, at a 0.1 MW/m^2 heat flux the FW will remain below melting. For the case of 0.25 MW/m^2, melting will occur during the first pulse, at ~200 s. Finally, at 0.5 MW/m^2, the FW would melt after only 70 s of operation.

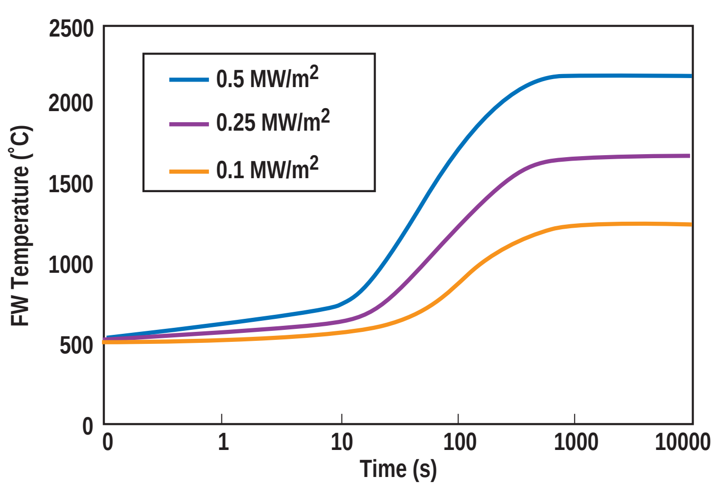

Fig. 6–2. DC TBM FW temperature evolution for various surface heat flux values during FW coolant LOCA at power.

This result seems to be in conflict with that for the ex-vessel loss-of-coolant analysis of the EU HCPB TBM [6–1]. The analysis assumed that a uniform surface heat flux of 0.1 to 0.25 MW/m^2 was present at the TBM prior to the accident. It was found that the beryllium FW surface temperature would come close to the melting point of beryllium at the end of the first pulse, but would reach the shutdown condition (1290°C) only in the middle of the next pulse. As more accurate nuclear heating and TBM design detail becomes available, we hope to demonstrate that the FW of the DC TBM will also not melt during a single pulse. Given the dwell time between pulses, the FW temperature of the DC TBM should return to normal operating values between pulses as a result of continued cooling by the breeder-cooling loop. However, if the FW does melt during a single pulse, a plasma disruption will ensue and the consequences internal to the ITER VV will be very similar to those for an in-vessel coolant leak discussed in Section 6.3.

The ex-vessel pressurization consequences for a loss-of-coolant event in the FW helium-cooling loop will be very similar to those predicted for the EU HCPB TBM [6–1]. The pressure rise in the TCWS vault will be less than 280 Pa for this event. As design detail become available, we will perform a similar analysis for the DC TBM to demonstrate this point. An ex-vessel coolant leak for the breeder-cooling loop will not cause pressurization of the TBM cell because the liquid breeding materials under consideration are low-vapor pressure fluids that will quickly solidify after contact with the floor of the cell.

6.6. SUMMARY

A preliminary assessment of the safety impact on ITER of a DC TBM concept shows that the anticipated radiological inventories are small in comparison to those produced in the ITER VV due to normal operation of ITER. Possible hydrogen sources were examined in this assessment and the conclusion was drawn that the maximum quantities produced during accident conditions should be less than the ITER limit of 2.5 kg. Pressurization of the ITER VV and TCWS vault by the helium coolant from the TBM ancillary system does not pose a serious threat to these confinement structures. Tritium permeation from the ancillary system has been analyzed and found to be below ITER release guidelines. However, it cannot be stressed strongly enough at this time that these are preliminary conclusions that are based on assumptions that will require more rigorous analyses once more design details become available for the DC TBM concept.

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7. SYSTEMS DESIGN AND LAYOUT

7.1. GENERAL SPACE ALLOCATIONS AND DESCRIPTION FOR TBM

The equatorial ports allocated for the test blanket modules have limited space and limited access through the port plugs. Space is also limited out side the port area to accommodate all the ancillary equipment needed to service and run the TBM. This space must be shared between the parties occupying the port. Furthermore, ITER safety, remote handling and remote operations requirements must be met when designing the TBM module and all of its support equipment. Figure 7–1 shows the standard test port arrangement where the TBM is connected to the transfer cask situated outside the Bio-Shield port opening. The TBM module is mounted inside the shielding frame, which is water-cooled, providing the support of the TBM and proper shielding behind the TBM as required by ITER. The TBM module along with the shielding frame and the VV port plug forms one complete assembly called the VV port plug. The port plug will be assembled and tested in the hot cell building prior to installation in the VV port extension. A special transporter will be utilized to carry and install the port assembly into the VV plug. Once the installation process is completed, the transporter will be moved out making room for the transfer cask, which houses the supporting equipment for the TBM as shown in Figs. 7–2 and 7–3.

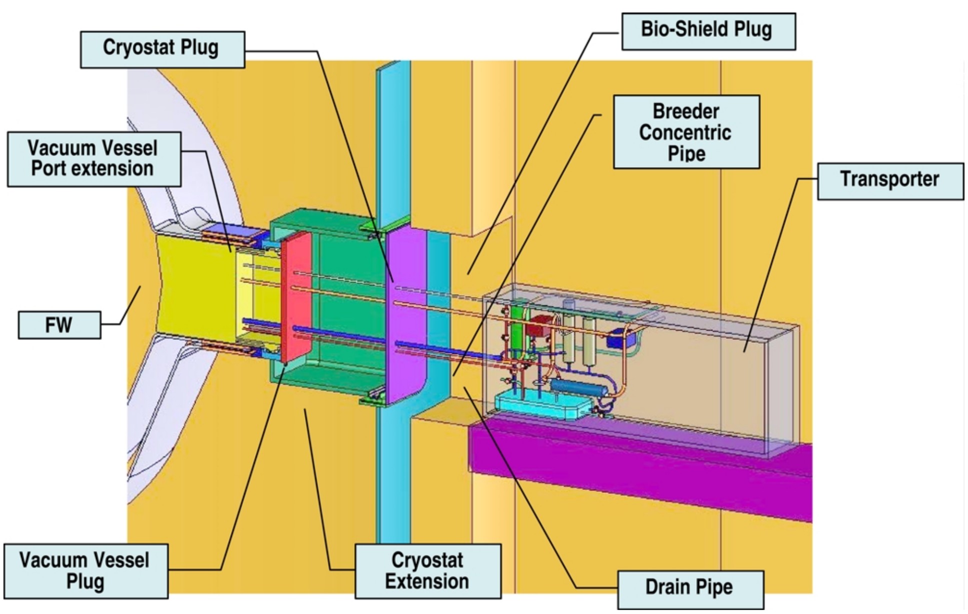

Fig. 7–1. Test port general arrangement.

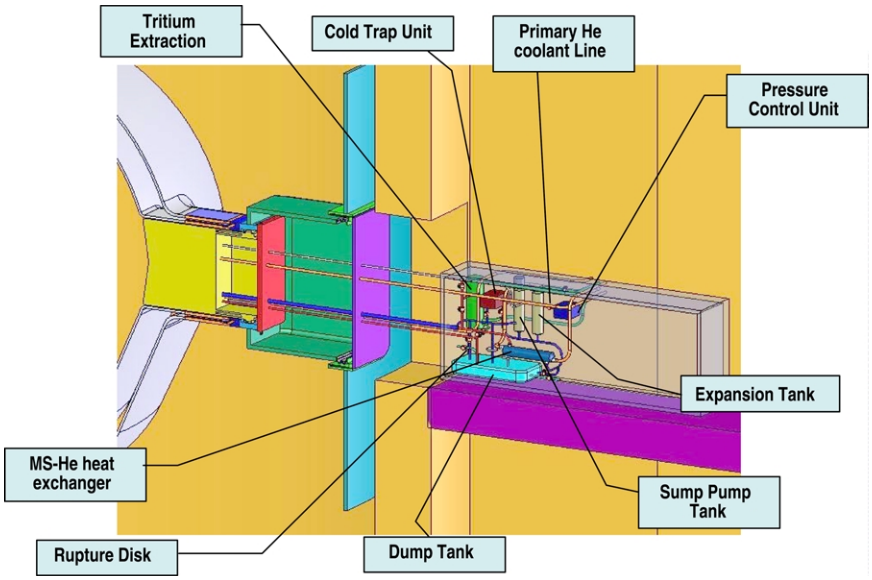

Fig. 7–2. Breeder coolant loop.

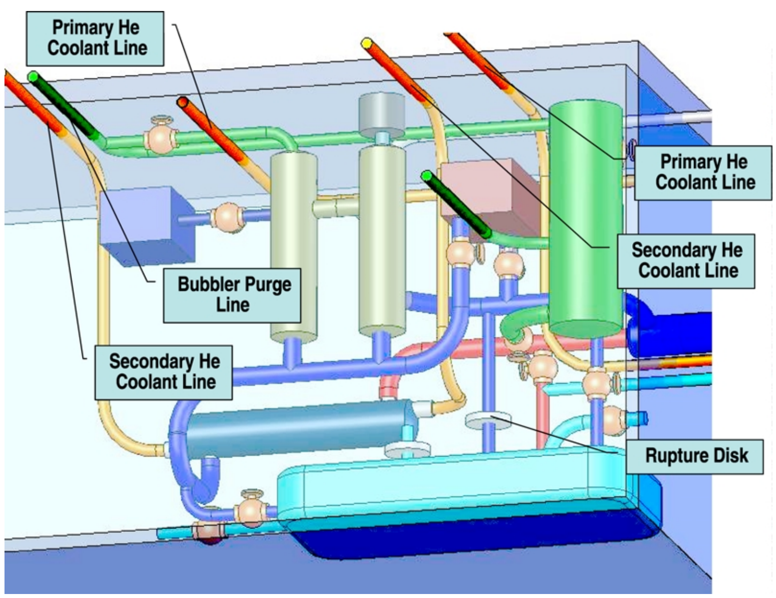

Fig. 7–3. Coolant loop details.

The TBM is connected to the transporter through series of pipes providing all the needed service mainly for cooling and diagnostics. These pipes must penetrate three barriers as they are routed between the transporter and the TBM. The US TBM design relies on the two coolant loops and purge lines to provide all the operational services needed. There are six pipes running between the transporter and the TBM. One concentric pipe carrying the Breeder MS or Pb-17Li, with the hot liquid running in the inner pipe, two He coolant lines for cooling the first-wall/structure and the breeder, a drain pipe for draining the MS liquid from the TBM and the VV plug assembly prior to removal or in case of emergency, and finally a gaseous product purge line from the top of the TBM module.

7.2. EQUATORIAL TEST PORT

TBM Module will be tested in predesignated test ports on the equatorial level. These test ports are similar in design to the other equatorial ports and could be used interchangeably. A water cooled shield plug is designed to fill the port in case it was not used for testing. However a similarly designed shield plug with provisions for the TBM will be used to test the different blanket modules. The shield plug extends to form one assembly with the VV port extension plug. This assembly is supported from the VV plug and is independent of the shielding blanket.

7.2.1. Test Module Assembly

The TBM is mounted inside a specially designed shield plug capable of holding two test modules with two different design concepts. As a result this assembly must accommodate the two different modules and allow for services to support both modules. The US TBM module assembly shown in Fig. 7–4 includes the test module, the shield plug and the VV port plug. The three major components will form on the complete assembly that will be mounted inside the equatorial port extension. The shield plug is composed of two major segments. They are the shielding frame which houses the TBM and the shielding module which is designed to fit directly behind the TBM and will provide the necessary shielding based on ITER requirements. Both the shielding frame and the shield plug are water cooled, with water pipes running from the VV port to the TCWS building.

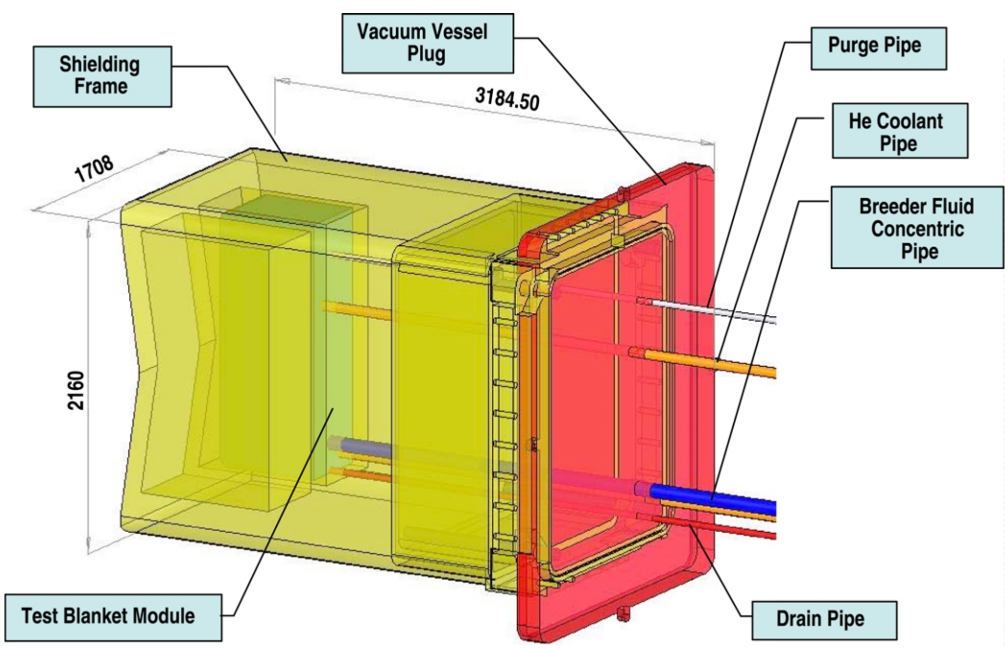

Fig. 7–4. Test module assembly.

7.2.2. Pipe Penetration through VV plug

As shown in Fig. 7–4, the test module assembly consists of three major components, the test blanket module, the shielding frame and the VV plug. The shielding frame is designed to be a housing support frame for the TBM and provide the required shielding for the magnets and surrounding structure. All coolant and service pipes as well as connectors such as power, control and instrumentation cables must run through the shielding in order to connect to the TBM. In order to minimize streaming because of the penetrations through the shielding, piping must be designed to prevent straight through access. This is especially critical for larger size pipes, however further analysis is needed in order to determine the proper piping penetration through the shielding block. Pipes connecting TBM to the VV plug are fixed on both ends because the penetrations through the VV plug are rigid sealed connections. Thus expansion loops will be used between the shielding and the VV plug to allow for thermal expansion.

The VV plug is designed as part of the test module assembly to provide the extended vacuum boundary through the VV port extension. Therefore, all penetration through this plug must be sealed to prevent any vacuum leaks. The TBM penetrations through the VV plug shown in Fig. 7–5 are mainly for the He coolant line, the breeder fluid, and drain and purge lines. Table 7–1 provides a summary of these penetrations. Space allocation on the plug surface must take into account any additional area requirements for special penetration design. Isolation valves will be required on all pipes outside the VV plug to seal and close all piping during transport, maintenance and remote operations. These valves will also require certain clearances around them to avoid interference. More details about the space requirements will be provided once the assembly procedure for the test module assembly is developed and more details become available. This also applies to all cable connectors running through the VV vessel plug (power, control and sensors). Additional area around each plug will be needed to allow for a special sealed plug design.

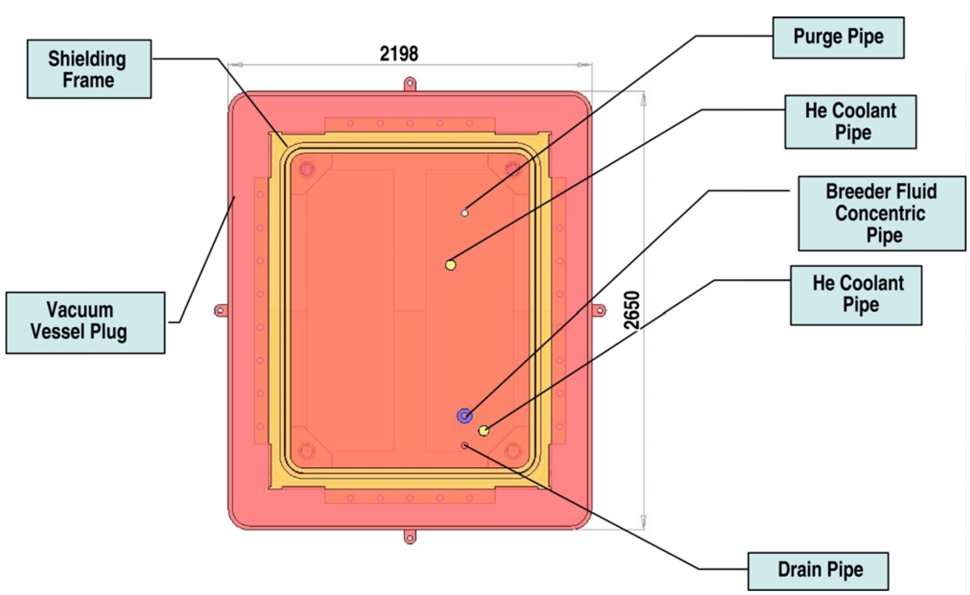

Fig. 7–5. Vacuum vessel plug.

Table 7–1
Vacuum Vessel Plug Penetration Summary

Description	Size (mm)	Number of Pipes	Approx. Penetration Size (mm)	$T_{in/out}$
Concentric Pipe penetration breeder fluid (MS or PbLi) Inner pipe o.d. 90mm	Outer pipe o.d. 160	1	190	
Primary FW coolant lines, He	o.d. 88.9	2	100	
Breeder fluid drain line	o.d. 88.9	1	100	
Bubbler/pressure relief line	o.d. 46	1	65	NA
Power/control cable connection	75	2	110	NA
Instrumentation cable connections	50	2	75	

7.2.3. Remote Maintenance and Handling

The TBM assembly including the shielding plug and the VV plug are preassembled in the hot cell area and transported to the equatorial ports using a special transporter design for equipment handling and insertion into the VV port. The transporter will be positioned in front of the bio-shield port opening for test module assembly installation. Once the cryostat plug is removed and the VV plug with the shielding assembly is taken out then the remote handling equipment inside the transporter will be used to insert and secure the TBM plug assembly into the VV port extension. The next step in this process is to make all the necessary piping connections between the module assembly and the cryostat plug. More details of the step-by-step process will be provided later.

7.3. CRYOSTAT PASS-THROUGH DESIGN

Pipe penetrations through the cryostat plug are designed to facilitate the pipe runs between the VV plug and the transfer cask outside the bio-shield door. Procedures for installation and removal of the test blanket plug assembly are carefully planned to utilize both hands on and remote operations. Hands-on maintenance operations are permitted to the point when the VV plug seal is opened. At this stage, remote operations are required. However, hands-on operation has limitation on the amount of time maintenance personnel are exposed after the removal of the bio-shield plug. Therefore, pipe penetrations through the cryostat plug must facilitate easy assembly and disassembly as well as allow for pipe expansion either through expansion loops in the interstitial area between the VV plug and the cryostat plug or through expansion joints (bellows) at the cryostat plug. Table 7–2 list all piping and other penetration requirements needed for the cryostat plug. This allows for clearances to easily setup cutting and welding equipment and facilitate access to all joints and plugs.

The interstitial space between the VV plug and the cryostat plug shown in Fig. 7–6 will accommodate the pipe runs and other service connections. In order to facilitate the removal of the test blanket assembly, a hands on cutting operation using the internal boring tool will be used to cut the pipe on the inboard side of the cryostat plug. After all connections are removed, the cryostat plug will be removed thus exposing the pipes and the VV plug. The concentric pipe assembly used to transport the breeder fluid will require a special procedure for separating the inner pipe and then cutting the outer the pipe. Once all connections are removed and the cryostat is removed, additional cuts will be performed to remove all piping and connections in preparation for the removal of the test blanket assembly. Detailed procedures will later be developed based on final configuration designs.

Table 7–2
Cryostat Plug and Bio-shield Penetration Summary

Description	Size (mm)	Number of Pipes	Approx. Penetration Size (mm)	$T_{in/out}$
Concentric pipe penetration breeder fluid (MS or PbLi)	Outer pipe o.d. 160 Inner pipe o.d. 90	1	190	
Primary FW coolant lines, He	o.d. 88.9	2	170	
Breeder fluid drain line	o.d. 88.9	1	170	
Bubbler/pressure relief line	o.d. 46	1	100	NA
Power/control cable connection	75	2	150	NA
Instrumentation cable connections	50	2	100	

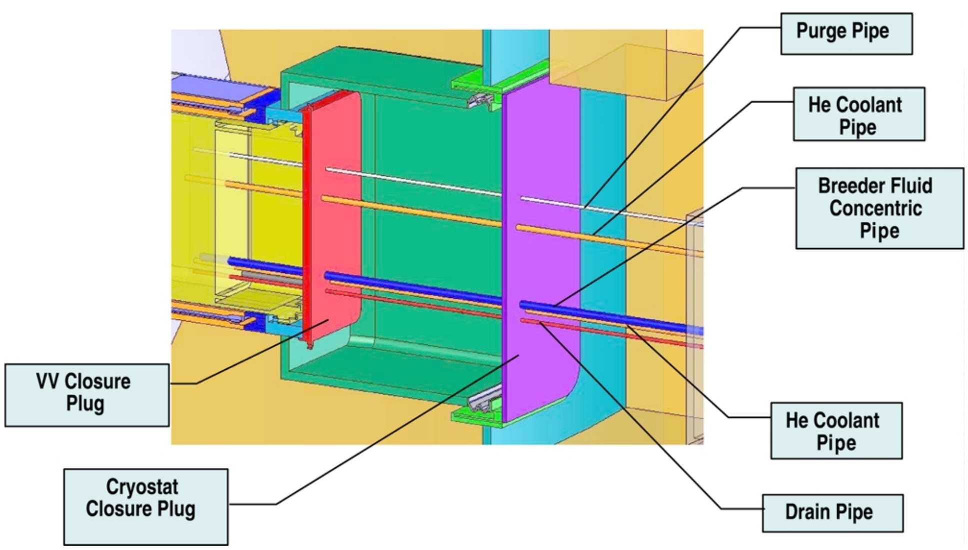

Fig. 7–6. VV-cryostat interstitial space.

Figure 7–7 shows the pipe penetrations through the cryostat plug. Because of the use of liquid metal or molten salts, the use of expansion loops to accommodate the thermal expansion between the VV and the cryostat plug may be limited due to drainage issues. This will require the expansion joints at the cryostat plug such as bellows to allow the pipe to through the VV plug without causing any failures. Final design recommendation on the use of expansion loops or bellows will be considered later, however pipe penetrations through the cryostat plug must be spaced to accommodate these possible design requirements.

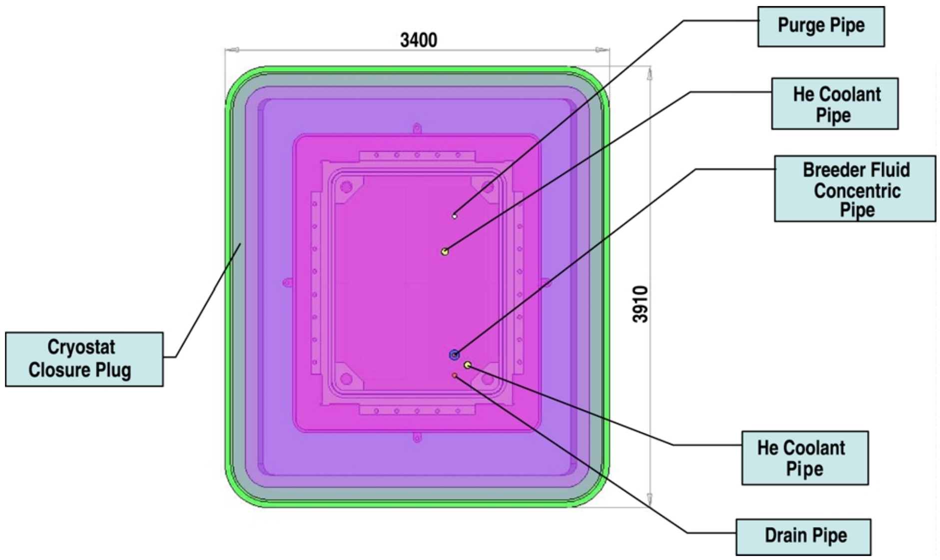

Fig. 7–7. Cryostat closure plug.

7.4. BIO-SHIELD PLUG

The Bio-shield plug is a 2-m thick concrete plug designed to provide shielding of the equipment in the equatorial test port area during operation. The plug is composed of block designed to accommodate the pipe penetrations as they are connected to the transporter just out side the bio-shield. Operations outside the bio-shield are considered hands on, thus facilitating the installation and separation of the transporter from the related services during maintenance operations.

7.5. SPACE ALLOCATION OF EQUATORIAL PORT AREA

The equatorial port area just outside the bio-shield is used as a staging area for a multitude of operations in conjunction with the installation, removal and operation of the TBM. The area is large enough to accommodate a container 7 m long, 2.65 m high and 2.6 m wide. Longer containers could be used up to 7.6 m long in case confinement is not required for the particular operation. A series of special design transporter will be utilized during the course of the test blanket module operation. Transporters will be used to remove and install the bio-shield plug as well as the cryostat plug. Installing and removing the test module assembly will require a special transporter with heavy lift equipment built into it and designed to perform remote operation for the test module insertion. During operation the equipment transporter shown in Fig. 7–8 will be parked in this area and will contain all the primary coolant loops and tritium extraction system as well as all instrumentation and control systems to monitor the test module status and collect the test data.

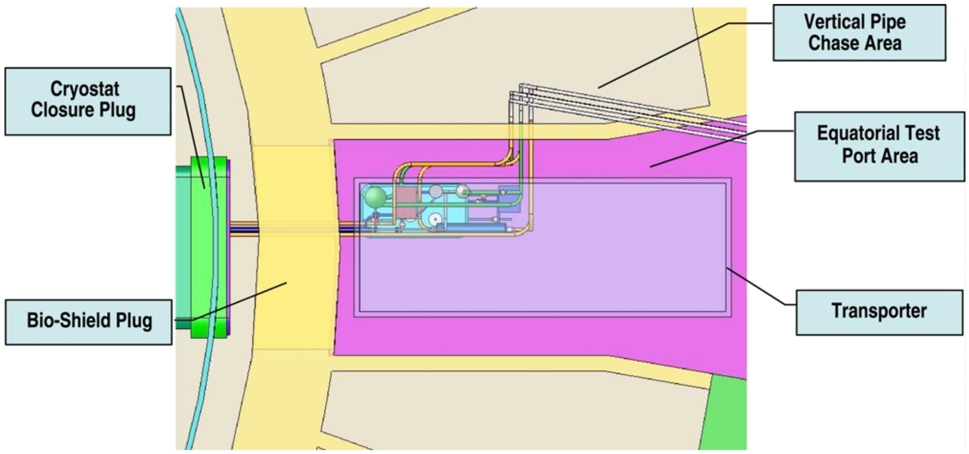

Fig. 7–8. Equatorial test port area with transporter.

Table 7–3 is a summary of the equipment installed in the transporter. The space inside the transporter will be used to house the all the equipment needed for both test modules that are being tested in the corresponding port. This table lists only the equipment needed for the US test module.

Table 7–3
Equatorial Port Transporter Equipment Summary

Description	Size (m)	No. of Pipe Connections	Number of Units Used
He/breeder fluid heat exchanger	0.37 diam 2.4 long	4	1
Sump pump tank	0.25 m diam 1 m long	3	1
Expansion tank	0.25 m diam 1 m long	4	1
Tritium extraction tank	1 m diam 2 m long	4	1
Dump tank	~0.5 m^3	6	1
Redox control and cold trap unit[a]	TBD	3	1
Pressure control unit[a]	TBD	1	1

[a]Unit are composed of various components, details will be provided later.

The transporter will be stationed out side the bio-shield and has various connections to the test modules for the coolant lines, purge and drain lines as well all power control and instrumentation. Table 7–4 lists a penetration summary for the transporter. All lines connecting the transporter to the test module assembly penetrate the transporter through the front side while all lines going between the test port area and the TCWS area penetrate the transporter through the side as shown in Fig. 7–9. Prior to moving the transporter, these connections will be removed. Internal power inside the transporter may have to be supplied to provide the trace heaters with power during transit between the test port area and the hot cell area. This is needed to prevent any freezing of the molten salt or the liquid metal if the system is already filled.

Table 7–4
Transporter Pipe Penetration Summary

Description	Size (mm)	Number of Pipes	Approx. Penetration Size (mm)	$T_{in/out}$
Concentric pipe penetration breeder fluid (MS or PbLi) Inner pipe o.d. 90 mm	Outer Pipe o.d. 160	1	190	
Primary FW coolant lines, He	o.d. 88.9	4	100	
Secondary loop He coolant pipes	o.d. 88.9	2	100	
Breeder fluid drain line	o.d. 88.9	1	100	
Dump tank pressure control pipe	o.d. 46	1	65	
Dump tank drain pipe	o.d. 63.5	1	85	
Bubbler/pressure relief line	o.d. 46	1	65	NA
Power/control cable connection	75	5	110	NA
Instrumentation cable connections	50	5	75	

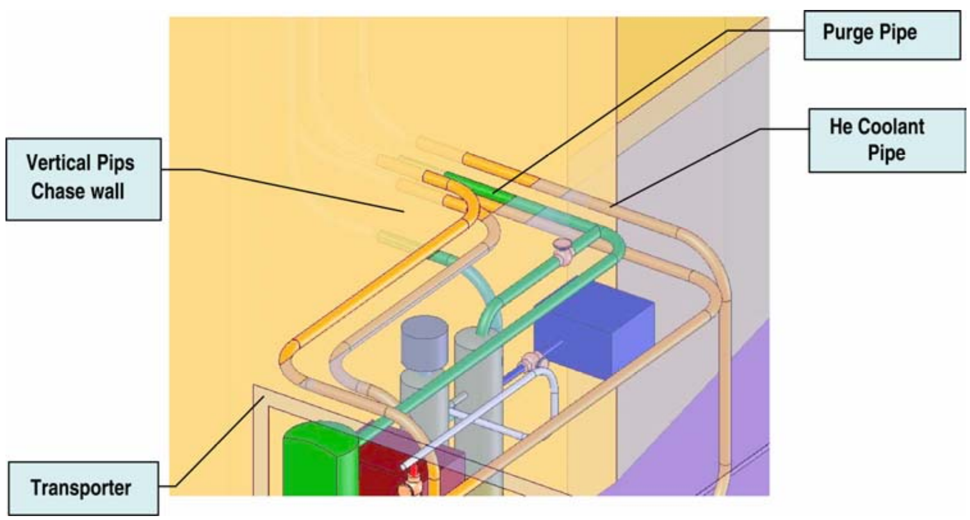

Fig. 7–9. Transporter side wall pipe penetrations to vertical pipe chase area.

7.6. VERTICAL PIPE CHASE AREA

All pipes and other connectors running between the test port area and the TCWS will be routed through the vertical pipe chase area. Figure 7–10 shows the pipes penetrations at the equatorial port level.

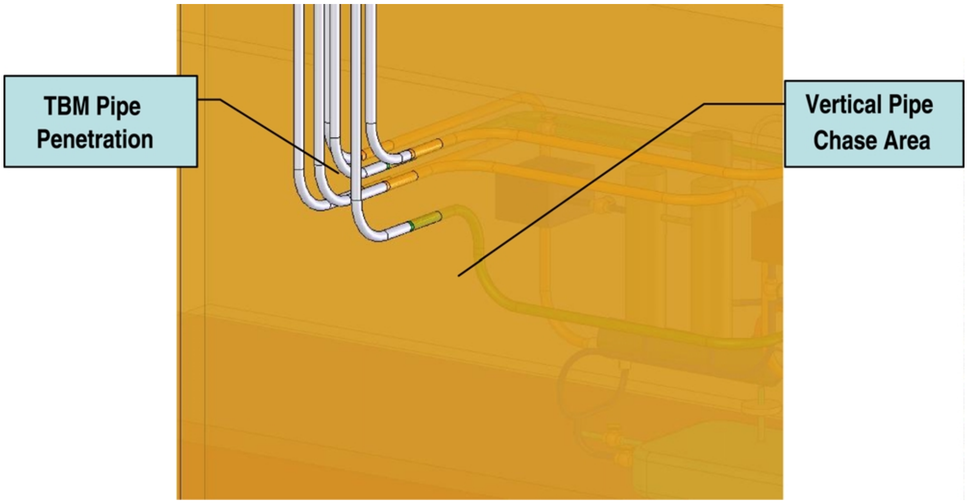

Fig. 7–10. Pipe penetration into vertical pipe chase area.

There are a total of six pipes running from the transporter to the TCWS area through the vertical pipe chase. The vertical run between the equatorial port level and the TCWS level is approximately 15 m long. The pipe bundle will have appropriate vertical supports allowing for thermal expansion. Expansion loops may not be an option through the pipe chase area simply because of space limitations. In addition to all the TBM piping, the pipe chase area will have the blanket coolant lines and other services running through it. As a result thermal expansion of this pipe run will be accommodated in the TCWS area.

Figure 7–11 shows the pipe run through the vertical pipe chase area as it reaches the TCWS area.

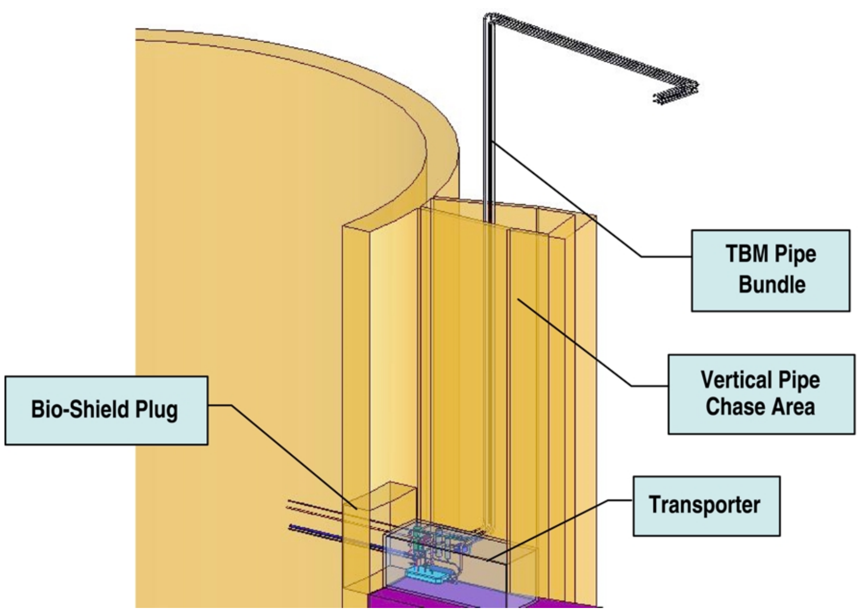

Fig. 7–11. Equatorial port and pipe chase area.

As the pipes reach the upper TCWS area they turn 90 deg toward the west wall in a horizontal run approximately 5 m long. Figure 7–12 shows the pipe run trough the TCWS area. The pipe run is approximately 28 m long and will have incorporate expansion loops to allow for thermal expansion.

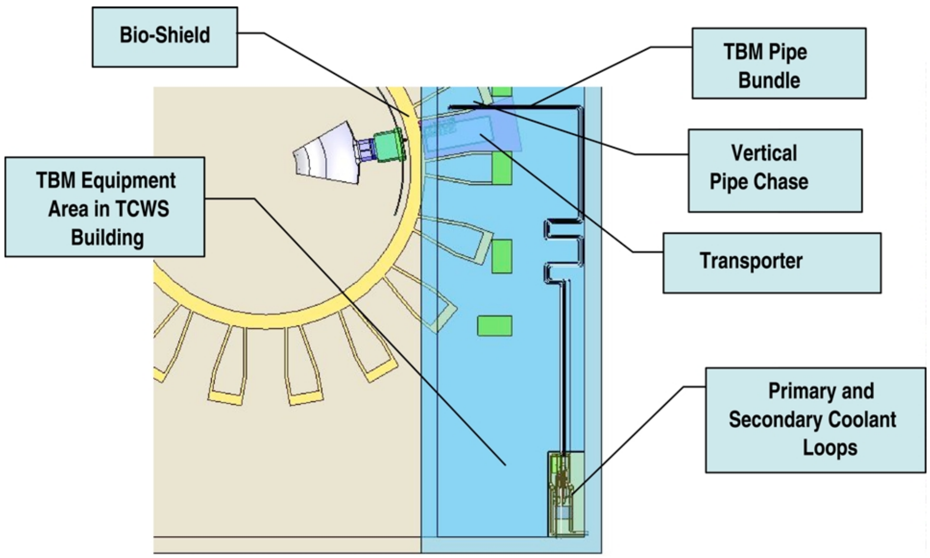

Fig. 7–12. TBM pipe run through TCWS area.

7.7. TCWS BUILDING LAYOUT AND AVAILABLE SPACE

A space of 16.6 × 7.3 m with a clear height of 5 m is assigned in the south end of the TCWS building for all TBM cooling system. This space is shared with all the parties to house the corresponding cooling systems. The US TBM design requires a primary and secondary He coolant loops that are located in this area. Table 7–5 lists a summary of the equipment to be used. The two loops will share some equipment such as the pressure control system and the tritium processing system.

Table 7–5
Primary and Secondary Coolant Loops Equipment Summary

Description	Size (m)	No. of Pipe Connections	Number of Units Used
He/water heat exchanger, primary and secondary loops	0.22 diam 0.63 long	4	2
He heating unit	0.35 diam 1.65 long	2	2
He circulator	1.8 diam 2.4 long	2	2
Pressure control sub system[(a)]	TBD	2	2
Tritium extraction sub-system[(a)]	TBD	2	1
Flow meter	TBD	2	4

[(a)]Subsystems include various components, details will be provided later.

The coolant lines will be connected directly to He-Water heat exchangers. The water pipes from the heat exchanger will be routed to the ITER heat removal system. Interface for this area will be defined in more details at a later time. He circulators located in the TCWS area will circulate the He back into test module assembly. Figure 7–13 shows the general arrangement of the cooling equipment located in the TCWS area.

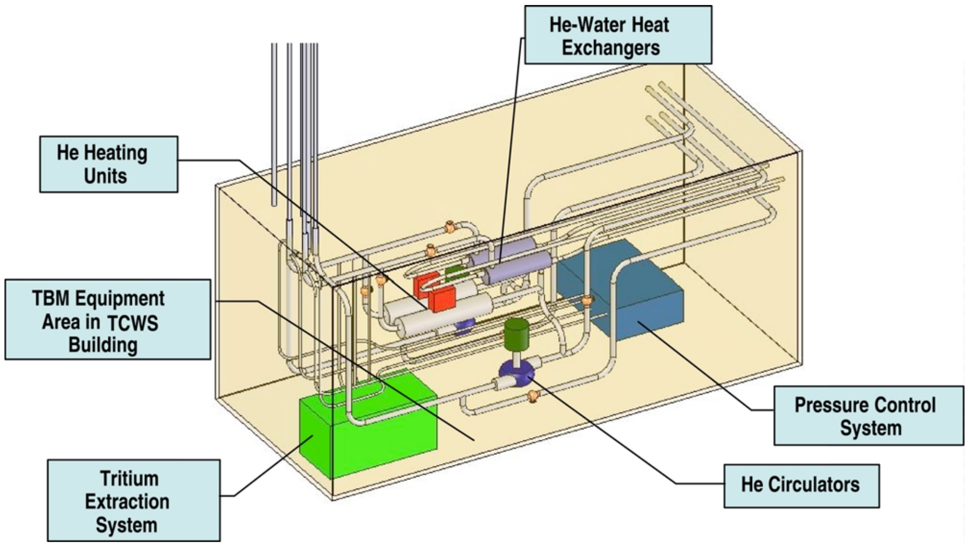

Fig. 7–13. US TBM primary and secondary coolant loops in TCWS.

The tritium processing and extraction system for both primary and secondary coolant loops. are also located in the TCWS area. Based on the current design and the equipment layout the total area requirements for the FW/structure and breeder helium cooling systems in the TCWS building should not exceed 20m^2 for the US TBM design. This requirement may be revised based on additional design details and changes. Figure 7–14 shows a plan view of the equipment layout in the TCWS building. The layout currently occupies an area 6.7 m long by 3 m wide.

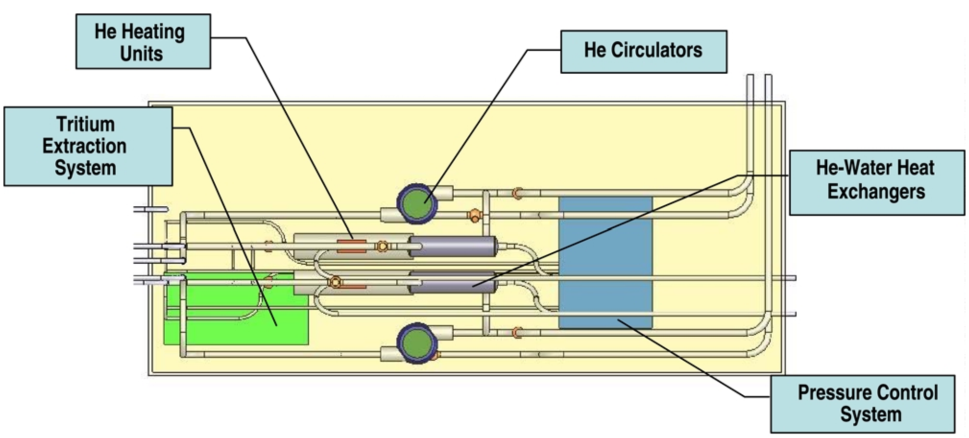

Fig. 7–14. Primary and secondary coolant loops plan view.

Services running from the TCWS area to the general ITER plant include the water coolant for the primary and secondary loops and the He purification bypass lines. The final pipe runs for these lines will be determined later based on more design details.

APPENDIX
ITER CALCULATION SHEETS

Table A–1
Summary of FW-Loop Helium to Water Heat Exchanger Design

ITER	General Atomics **Calculation Sheet**	Page No.: 41
Calculation By: D. P. Carosella	System ITER Test Loop Title: "U" Tube Helium Heat Exchanger Design Cold Aluminum	Calculation No.: 1

IV. RESULTS:

Summary of Design Data for the Helium to Water Heat Exchanger:

	Metric	English
Heat Duty:	Q_{hex} 569.2 kW	Q_{hex} 1.942 10^{6} BTU hr ^{1}
Effectiveness/NTU:	eff 0.148 / NTU 0.162	
Design Uncertainty	UN_{ht} 15 %	
Water Data:		
Flow Rate:	W_{H2O} 16.99 kg sec ^{1}	W_{H2O} 37.47 lb sec^{-1}
Inlet Temperature:	$TCH2O_{in}$ 35 C	$TFH2O_{in}$ 95 F
Outlet Temperature:	$TCH2O_{out}$ 43 C	$TFH2O_{out}$ 109.4 F
Pressure Drop:	P_{H2O} 0.0287 MPa	P_{H2O} 4.162 psi
Percent Pressure Drop:	$\frac{P_{H2O}}{PH2O_{max}}$ 2.87 %	
Pumping Power	PP_{H2O} 491.02 watt	PP_{H2O} 0.658 hp
Maximum Velocity:	$VelH2O_{max}$ 1.73 $\frac{m}{sec}$	$VelH2O_{max}$ 5.68 ft sec^{-1}
Saturation Temperature:	TCSAT 179.9 C	TFSAT 355.8 F
Helium Data:		
Flow Rate:	W_{He} 1.82 kg sec ^{1}	W_{He} 4.02 lb sec^{-1}
Inlet Temperature:	$TCHe_{in}$ 440 C	$TFHe_{in}$ 824 F
Outlet Temperature:	$TCHe_{out}$ 380 C	$TFHe_{out}$ 716 F
Pressure Drop:	P_{tot} 0.044 MPa	P_{tot} 6.43 psi
Percent Pressure Drop	$\frac{P_{tot}}{P_{He}}$ 0.55 %	
Pumping Power:	PP_{He} 14.35 kW	PP_{He} 19.24 hp
Tube Wall Temperature:	$TCwall_{in}$ 157.2 C	$TFwall_{in}$ 315 F
Maximum Velocity:	$VelHe_{max}$ 65.1 m sec ^{1}	$VelHe_{max}$ 213.5 ft sec^{-1}
Velocity in the Orifice	Vel_{ori} 113.9 m sec ^{1}	Vel_{ori} 373.7 ft sec^{-1}

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Table A–1 (Cont.

ITER	General Atomics **Calculation Sheet**	Page No.: 42
Calculation By: D. P. Carosella	System ITER Test Loop Title: "U" Tube Helium Heat Exchanger Design Cold Aluminum	Calculation No.: 1

Summary of Design Data for Helium to Water Heat Exchanger:

	Metric		English	
Geometric Data				
Overall Dimensions				
Total Length	TL_{tot}	0.629 m	TL_{tot}	2.063 ft
Shell Outside Diameter	SOD	0.2231 m	SOD	8.785 in
Heat Transfer Surface Area:	TubA	0.536 m^2	TubA	5.77 ft^2
Total Tube Length ("U" Tube) (Including Tube Sheet)	Len_{tube}	0.59 m	Len_{tube}	1.94 ft
Inside Shell Diameter:	Hexdia	0.206 m	Hexdia	8.096 in
Head Height:	b	0.06 m	b	2.2 in
Gas Supply & Return Pipe Dia:	dia_{He}	102.26 mm	dia_{He}	4.03 in
Water Supply & Return Pipe Dia:	dia_{H2O}	62.71 mm	dia_{H2O}	2.47 in
Hot Tube Sheet Thickness	TST_{hot}	0.063 m	TST_{hot}	2.49 in
Baffle Thickness	$_{baffle}$	6.35 mm	$_{baffle}$	0.25 in
Total Dry Weight	Wt_{tot}	45.7 kg	Wt_{tot}	100.7 lb
Tube Parameters:				
Number of Tubes		Ntube 41		
Inlet Orifice Diameter	$diaHe_{ori}$	9.6 mm	$diaHe_{ori}$	0.3779 in
Rectangular Pitch	Pitch	17.94 mm	Pitch	0.7064 in
Tube Inside Diameter:	$Tube_{ID}$	12.7 mm	$Tube_{ID}$	0.5 in
Tube Outside Diameter:	$Tube_{OD}$	13.7 mm	$Tube_{OD}$	0.539 in
Tube Wall Thickness:	$_{wall}$	0.5 mm	$_{wall}$	0.0197 in

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Table A–1 (Cont.

ITER	General Atomics **Calculation Sheet**	Page No.: 43
Calculation By: D. P. Carosella	System ITER Test Loop Title: "U" Tube Helium Heat Exchanger Design Cold Aluminum	Calculation No.: 1

Auxiliary Data

Total Water Side Volume	$Vol_{ss} = 0.005\,m^3$	$Vol_{ss} = 0.191\,ft^3$
Total Helium Side Volume	$VolHe_{Hex} = 0.0096\,m^3$	$VolHe_{Hex} = 0.339\,ft^3$
Helium Inventory	$WtHe_{Hex} = 0.0395\,kg$	$WtHe_{Hex} = 0.087\,lb$
Tube Side (He) Nusselt No.	$Nu_{He} = 239.9114$	
Tube Side (He) Heat Transfer Coef.	$h_{He} = 5100\,\dfrac{watt}{m^2\ K}$	$h_{He} = 898\,\dfrac{BTU}{hr\ ft^2\ R}$
Shell Side (H2O) Nusselt No.	$Nu_{H2O} = 306.5572$	
Shell Side (H2O) HT Coef.	$h_{H2O} = 1.21 \cdot 10^4\,\dfrac{watt}{m^2\ K}$	$h_{H2O} = 2127\,\dfrac{BTU}{ft^2\ R\ hr}$

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Table A–2
Summary of MS-Loop Helium to Water Heat Exchanger Design

ITER	General Atomics **Calculation Sheet**	Page No.: 42
Calculation By: D. P. Carosella	System LiBeF3 ITER Test Loop Title: "U" Tube Helium to H2O Heat Exchanger	Calculation No.: 1

IV. RESULTS:

Summary of Design Data for the Helium to Water Heat Exchanger:

	Metric	English
Heat Duty:	Q_{hex} 1400 kW	Q_{hex} 4.777 10^{6} BTU hr^{1}
Effectiveness/NTU:	eff 0.492 /	NTU 0.722
Water Data:		
Flow Rate:	W_{H2O} 9.55 kg sec^{1}	W_{H2O} 21.06 lb sec^{-1}
Inlet Temperature:	$TCH2O_{in}$ 35 C	$TFH2O_{in}$ 95 F
Outlet Temperature:	$TCH2O_{out}$ 70 C	$TFH2O_{out}$ 158 F
Pressure Drop:	P_{H2O} 0.0081 MPa	P_{H2O} 1.18 psi
Percent Pressure Drop:	$\frac{P_{H2O}}{PH2O_{max}}$ 0.81 %	
Pumping Power	PP_{H2O} 78.74 watt	PP_{H2O} 0.106 hp
Maximum Velocity:	$VelH2O_{max}$ 0.55 $\frac{m}{sec}$	$VelH2O_{max}$ 1.8 ft sec^{-1}
Helium Data:		
Flow Rate:	W_{He} 1.68 kg sec^{1}	W_{He} 3.71 lb sec^{-1}
Inlet Temperature:	$TCHe_{in}$ 360 C	$TFHe_{in}$ 680 F
Outlet Temperature:	$TCHe_{out}$ 200 C	$TFHe_{out}$ 392 F
Pressure Drop:	P_{tot} 0.029 MPa	P_{tot} 4.27 psi
Percent Pressure Drop	$\frac{P_{tot}}{P_{He}}$ 0.37 %	
Pumping Power:	PP_{He} 7.12 kW	PP_{He} 9.55 hp
Design Uncertainty	UN_{ht} 15 %	
Tube Wall Temperature:	$TCwall_{bar}$ 170.2 C	$TFwall_{bar}$ 338.4 F
Maximum He Velocity in Tubes	$VelHe_{out}$ 21.2 m sec^{1}	$VelHe_{out}$ 69.6 ft sec^{-1}
Velocity In Pipes:	$VelHe_{max}$ 28.4 m sec^{1}	$VelHe_{max}$ 93.1 ft sec^{-1}
Velocity In The Inlet Orifice	Vel_{ori} 68.1 m sec^{1}	Vel_{ori} 223.5 ft sec^{-1}

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Table A–2 (Cont.

ITER Calculation By: D. P. Carosella	General Atomics **Calculation Sheet** System LiBeF3 ITER Test Loop Title: "U" Tube Helium to H2O Heat Exchanger	Page No.: 43 Calculation No.: 1

Summary of Design Data for Helium to Water Heat Exchanger:

	Metric		English	
Geometric Data				
Overall Dimensions				
Total Length	TL_{tot}	1.212 m	TL_{tot}	3.975 ft
Shell Outside Diameter	SOD	0.2901 m	SOD	11.420 in
Heat Exchanger Surface Area	TubA	4.72 m^2	TubA	50.85 ft^2
Total Tube Length ("U" Tube) (Including Tube Sheet)	Len_{tube}	1.55 m	Len_{tube}	5.1 ft
Inside Shell Diameter:	Hexdia	0.277 m	Hexdia	10.896 in
Head Height:	b	0.07 m	b	2.86 in
Gas Supply & Return Pipe Dia:	dia_{He}	77.93 mm	dia_{He}	3.07 in
Water Supply & Return Pipe Dia:	dia_{H2O}	62.71 mm	dia_{H2O}	2.47 in
Hot Tube Sheet Thickness	TST_{hot}	0.065 m	TST_{hot}	2.55 in
Baffle Thickness	baffle	6.35 mm	baffle	0.25 in
Total Dry Weight:	Wt_{tot}	115.7 kg	Wt_{tot}	255.1 lb
Tube Parameters:				
Number of Tubes	Ntube	77		
Inlet Orifice Diameter	$diaHe_{ori}$	8.2 mm	$diaHe_{ori}$	0.32 in
Rectangular Pitch	Pitch	17.94 mm	Pitch	0.7064 in
Tube Inside Diameter:	$Tube_{ID}$	12.7 mm	$Tube_{ID}$	0.5 in
Tube Outside Diameter:	$Tube_{OD}$	13.7 mm	$Tube_{OD}$	0.539 in
Tube Wall Thickness:	wall	0.5 mm	wall	0.0197 in
Auxiliary Data				
Total Shell Side Volume	Vol_{ss}	0.031 m^3	Vol_{ss}	1.102 ft^3
Tube Side (He) Nusselt No.	Nu_{He}	152.2054		
Tube Side (He) Heat Transfer Coef.	h_{He}	$2807\,\frac{watt}{m^2\ K}$	h_{He}	$494\,\frac{BTU}{hr\ ft^2\ R}$
Shell Side (H2O) Nusselt No.	Nu_{H2O}	119.8504		
Shell Side (H2O) HT Coef.	h_{H2O}	$4634\,\frac{watt}{m^2\ K}$	h_{H2O}	$816\,\frac{BTU}{ft^2\ R\ hr}$

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